CATEGORY 1

REGULATORY INFORMATION DISTRIBUTION SYSTEM (RIDS)

ACCESSION NBR:9703070192 DOC.DATE: 96/12/31 NOTARIZED: NO DOCKET #
FACIL:50-220 Nine Mile Point Nuclear Station, Unit 1, Niagara Powe 05000220
50-410 Nine Mile Point Nuclear Station, Unit 2, Niagara Moha 05000410
AUTH.NAME AUTHOR AFFILIATION
ABBOTT,R.B. Niagara Mohawk Power Corp.
RECIP.NAME RECIPIENT AFFILIATION

SUBJECT: "Annual Occupational Exposure Repts for Nine Mile Point Units 1 & 2 for CY96." W/970227 ltr.

DISTRIBUTION CODE: IE56D COPIES RECEIVED:LTR 1 ENCL 1 SIZE: 5
TITLE: 20.407 Annual Personnel Monitoring Rpt/Annual Rad Exposure Rpt by Fun

NOTES:

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
	PD1-1 PD	1	0	HOOD,D	1	1
INTERNAL:	ACNW	1	1	FILE CENTER	1	1
	NMSS/SSSS T 8F5	1	1	NRR/DRPM/PERB/B	1	1
	RES/RPHEB	1	1	RGN1 01	1	1
EXTERNAL:	LITCO BRYCE,J H	1	1	NOAC	1	1
	NRC PDR	1	1			

C A T E G O R Y 1 D O C U M E N T

NOTE TO ALL "RIDS" RECIPIENTS:
PLEASE HELP US TO REDUCE WASTE! CONTACT THE DOCUMENT CONTROL DESK, ROOM OWFN 5D-5(EXT. 415-2083) TO ELIMINATE YOUR NAME FROM DISTRIBUTION LISTS FOR DOCUMENTS YOU DON'T NEED!

TOTAL NUMBER OF COPIES REQUIRED: LTTR 11 ENCL 10

NIAGARA MOHAWK
GENERATION
BUSINESS GROUP

NINE MILE POINT NUCLEAR STATION/LAKE ROAD, P.O. BOX 63, LYCOMING, NEW YORK 13093/TELEPHONE (315) 349-1812
FAX (315) 349-4417

RICHARD B. ABBOTT
Vice President and
General Manager – Nuclear

February 27, 1997
NMP1L 1186

United States Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

RE:	Nine Mile Point Unit 1	Nine Mile Point Unit 2
	Docket No. 50-220	Docket No. 50-410
	DPR-63	NPF-69

Gentlemen:

In accordance with the Technical Specifications for the Nine Mile Point Nuclear Station Units 1 and 2, Niagara Mohawk Power Corporation is submitting the Annual Occupational Exposure Reports. These reports cover the period January 1, 1996 through December 31, 1996.

Please refer to any questions regarding these reports to Mr. Robert Carlson, Supervisor Dosimetry, (315) 349-2728.

Sincerely,

Richard B. Abbott
Vice President and General Manager - Nuclear

RBA/RC/lmc

IE56

xc: Regional Administrator, Region I
Mr. Barry S. Norris, Senior Resident Inspector
Records Management

Nine Mile Point Nuclear Station Unit # 1 DPR-63 1996 Regulatory Guide 1.16 Report

Work & Job Function	Numbers of Personnel >0 mrem				Exposure Totals			
	Station Employees	Utility Employees	Contract & Others	Total Persons	Station Employees	Utility Employees	Contract & Others	Total milli-rems
Reactor Operations & Surv.								
Maintenance Personnel	92	0	145	237	7,820	0	211	8,031
Operating Personnel	51	0	10	61	12,956	0	0	12,956
Health Physics Personnel	41	0	6	47	4,337	0	8	4,345
Engineering Personnel	24	0	72	96	1,042	0	109	1,151
Supervisory Personnel	37	0	33	70	1,609	0	39	1,648
Total				511				28,131
Routine Maintenance								
Maintenance Personnel	109	0	22	131	8,949	0	1611	10,560
Operating Personnel	51	0	2	53	2,108	0	32	2,140
Health Physics Personnel	42	0	0	42	2,577	0	0	2,577
Engineering Personnel	43	0	37	80	815	0	522	1,337
Supervisory Personnel	19	0	11	30	990	0	517	1,507
Total				336				18,121
In-Service Inspection								
Maintenance Personnel	26	0	9	35	2,430	0	2551	4,981
Operating Personnel	0	0	0	0	0	0	0	0
Health Physics Personnel	6	0	0	6	119	0	0	119
Engineering Personnel	18	0	7	25	1,954	0	1892	3,846
Supervisory Personnel	2	0	2	4	43	0	535	578
Total				70				9,524
Special Maintenance								
Maintenance Personnel	24	0	9	33	985	0	521	1,506
Operating Personnel	2	0	0	2	14	0	0	14
Health Physics Personnel	9	0	0	9	32	0	0	32
Engineering Personnel	16	0	3	19	218		225	443
Supervisory Personnel	4	0	0	4	79	0	0	79
Total								2,074
Waste Processing								
Maintenance Personnel	18	0	3	21	129	0	0	129
Operating Personnel	31	0	5	36	1,878	0	410	2,288
Health Physics Personnel	22	0	0	22	191	0	0	191
Engineering Personnel	4	0	1	5	16	0	0	16
Supervisory Personnel	4	0	1	5	19	0	0	19
Total				89				2,643
Refueling								
Maintenance Personnel	17	0	0	17	0	0	0	0
Operating Personnel	29	0	0	29	30	0	0	30
Health Physics Personnel	9	0	0	9	1	0	0	1
Engineering Personnel	1	0	4	5	0	0	6	6
Supervisory Personnel	5	0	0	5	0	0	0	0
Total				65				37
Summary by Job Function								
Maintenance Personnel	286	0	188	474	20,313	0	4,894	25,207
Operating Personnel	164	0	17	181	16,986	0	442	17,428
Health Physics Personnel	129	0	6	135	7,257	0	8	7,265
Engineering Personnel	106	0	124	230	4,045	0	2,754	6,799
Supervisory Personnel	71	0	47	118	2,740	0	1,091	3,831
Grand Total	756	0	382	1,138	51,341	0	9,189	60,530

1996 Nine Mile Point - Unit 1 Exposure for Special Maintenance Activities

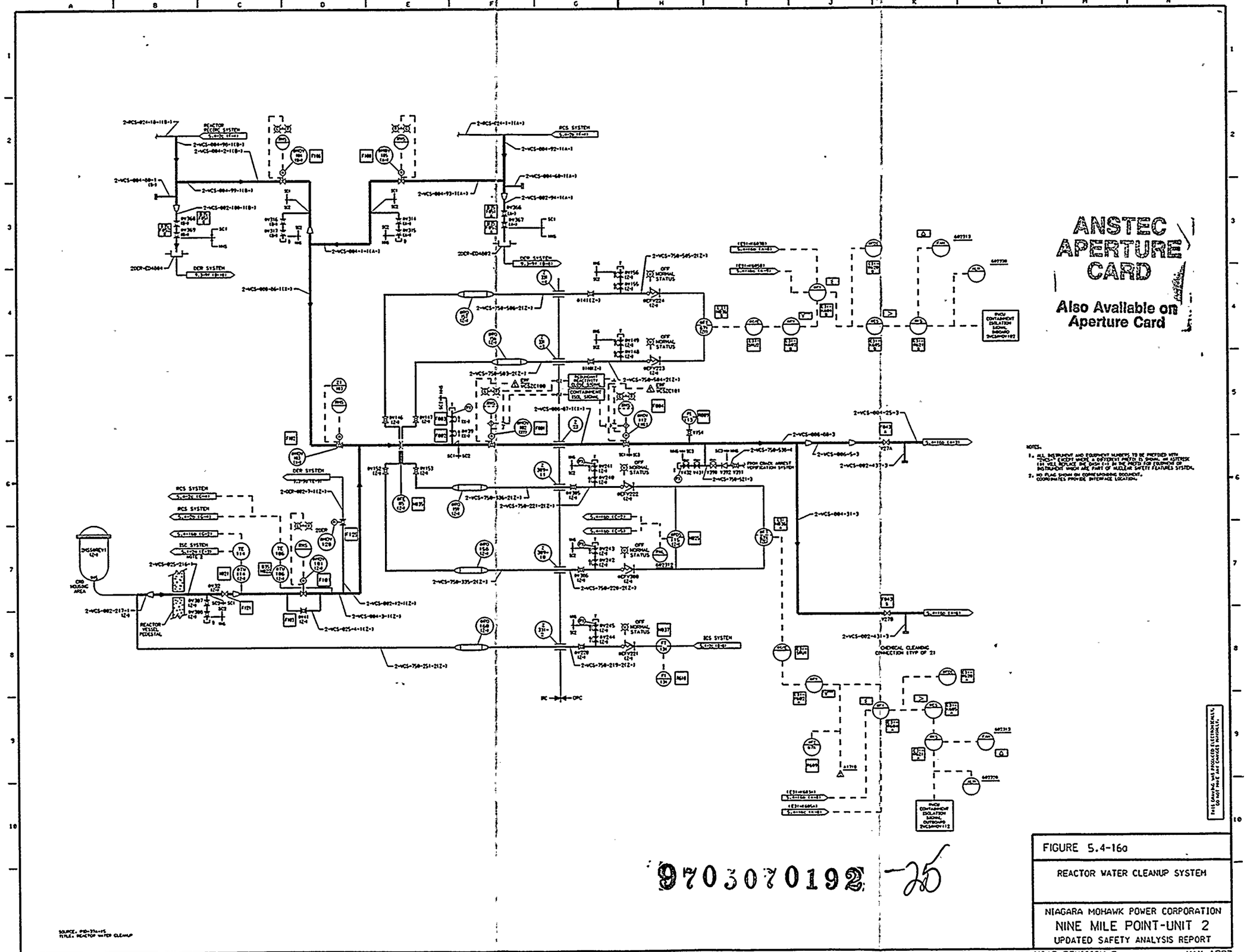
Job Description	MOD / SC	Exposure (mRem)
Feed Water System Modifications	Mod # 1S0028	1,730
RW Remove 11 Sump & Reset Level Controls	SC1-0116-91	127
Modify Spent Fuel Pool Racks	Mod # N1-95-006	126
Miscellaneous	Various	91
	Total Exposure	**2,074**

Nine Mile Point Nuclear Station Unit # 2 NPF-69 1996 Regulatory Guide 1.16 Report

Work & Job Function	Numbers of Personnel > 0 mrem				Exposure Totals			
	Station Employees	Utility Employees	Contract & Others	Total Persons	Station Employees	Utility Employees	Contract & Others	Total milli-rems
Reactor Operations & Surv								
Maintenance Personnel	233	0	430	663	4,658	0	5,820	10,478
Operating Personnel	141	0	12	153	15,221	0	19	15,240
Health Physics Personnel	88	0	42	130	9,049	0	2,070	11,119
Engineering Personnel	42	0	102	144	1,958	0	245	2,203
Supervisory Personnel	78	0	40	118	404	0	77	481
Total				1,208				39,521
Routine Maintenance								
Maintenance Personnel	242	0	304	546	34,705	0	61,165	95,870
Operating Personnel	102	0	9	111	2,420	0	90	2,510
Health Physics Personnel	62	0	42	104	7,591	0	5,599	13,190
Engineering Personnel	181	0	196	377	4,667	0	5,994	10,661
Supervisory Personnel	61	0	48	109	1,185	0	1,956	3,141
Total				1,247				126,338
In-Service Inspection								
Maintenance Personnel	9	0	210	219	185	0	24,386	24,571
Operating Personnel	0	0	1	1	0	0	4	4
Health Physics Personnel	8	0	2	10	101	0	25	126
Engineering Personnel	12	0	35	47	1,021	0	6,960	7,981
Supervisory Personnel	0	0	5	25	0	0	258	258
Total				282				32,940
Special Maintenance								
Maintenance Personnel	62	0	184	246	1,882	0	13,393	15,275
Operating Personnel	5	0	1	6	18	0	0	18
Health Physics Personnel	13	0	6	19	35	0	37	72
Engineering Personnel	26	0	22	48	36	0	78	14
Supervisory Personnel	9	0	11	20	5	0	11	16
Total				339				15,495
Waste Processing								
Maintenance Personnel	4	0	4	8	6	0	0	6
Operating Personnel	26	0	0	26	3,129	0	0	3,129
Health Physics Personnel	16	0	16	32	257	0	1,406	1,663
Engineering Personnel	5	0	7	12	0	0	342	342
Supervisory Personnel	4	0	2	6	23	0	1	24
Total				84				5,164
Refueling								
Maintenance Personnel	86	0	187	273	774	0	11,178	11,952
Operating Personnel	55	0	3	58	1,453	0	35	1,488
Health Physics Personnel	29	0	15	44	1,541	0	581	2,122
Engineering Personnel	27	0	43	70	207	0	3,302	3,509
Supervisory Personnel	11	0	6	17	91	0	12	103
Total				462				19,174
Summary by Job Function								
Maintenance Personnel	636	0	1,319	1,955	42,210	0	115,942	158,152
Operating Personnel	329	0	26	355	22,241	0	148	22,389
Health Physics Personnel	216	0	123	339	18,574	0	9,718	28,292
Engineering Personnel	293	0	405	698	7,889	0	17,887	25,776
Supervisory Personnel	163	0	112	275	1,708	0	2,315	4,023
Grand Total	1,637	0	1,985	3,622	92,622	0	146,010	238,632

1996 Nine Mile Point - Unit 2 Exposure for Special Maintenance Activities

Job Description	MOD / EDC / SC	Exposure (mRem)
Drywell Snubber Mods	2S10971 & 2S10980	10,966
Drywell 300' Permanent Shielding - LPCI A & C	SC2-0144-92	1,447
Turbine 250' HDL Mods	2F01401, 2F01401A & 2F01423	605
Install ADH System	N2-96-004	598
Reactor Building, 175', RHS-MOV- 8A	2M10964,2M10964A,2S10869, DP-437AB,2S10859,2S10859	505
Reactor Building 240', Support MSIV ILRT Re-Test	SC2-0207-91	403
Reactor Building 261', NS Level Tree Shielding	N2-96-019 & 2S10773A	363
Reactor Building, 175', 2SWP-001-65H-3	SC2-0318-91	222
Reactor Building, 215', WCS Seal Water	2F01276	184
Reactor Building, 175', Install Taps and Valves for Drain Cooler	DDC-2S10859	108
Miscellaneous	Various	94
	Total Exposure	**15,495**



ANSTEC APERTURE CARD

Also Available on Aperture Card

NOTES:

1. ALL INSTRUMENT AND EQUIPMENT NUMBERS TO BE PREFIXED WITH "PNCA" EXCEPT WHEN A DIFFERENT PREFIX IS SHOWN, AN ASTERISK (*) WILL REPLACE THE PREFIX (-) IN THE PREFIX FOR EQUIPMENT OR INSTRUMENT WHICH ARE PART OF NUCLEAR SAFETY FEATURES SYSTEM.
2. NO FLAG SHOWN ON CORRESPONDING DOCUMENT. COORDINATES PROVIDE INTERFACE LOCATION.

THIS DRAWING WAS PRODUCED ELECTRONICALLY. DO NOT MAKE ANY CHANGES MANUALLY.

9703070192 -25

FIGURE 5.4-16a

REACTOR WATER CLEANUP SYSTEM

NIAGARA MOHAWK POWER CORPORATION
NINE MILE POINT-UNIT 2
UPDATED SAFETY ANALYSIS REPORT

APPENDIX 4A

INTERGRANULAR STRESS CORROSION CRACKING

APPENDIX 4A

INTERGRANULAR STRESS CORROSION CRACKING

Welded austenitic stainless steel reactor internals are relied upon to permit adequate core cooling for any mode of normal operation or under credible postulated accident conditions. This appendix demonstrates that components which may have carbon content at the maximum limit are not subject to intergranular stress corrosion cracking (IGSCC).

Carbon content of austenitic stainless steels is used as an indication for grain boundary chromium depletion. Chromium depletion, also known as sensitization, can lead to IGSCC in BWR components. IGSCC due to sensitization was the primary cause of cracking in BWR piping components. This appendix addresses the potential for this phenomenon manifesting in BWR internal components. For sensitization-induced IGSCC to occur in reactor internals fabricated from wrought austenitic stainless steel, the following three conditions are all necessary:

1. Sensitized condition.
2. Corrosive environment.
3. Significant tensile stress.

Since carbon content in the wrought austenitic stainless steel is a major factor leading to material sensitization from welding, the level of sensitization can be minimized by keeping the carbon content of the stainless steel below 0.035 percent.

Oxygenated BWR water is considered a corrosive environment when the water temperature exceeds 200°F for sustained periods of time (1,000 hr during the life of the plant). Parts subjected to oxygenated BWR water below this temperature are not susceptible to IGSCC.

Significant tensile stress on a particular component or part is dependent on its residual stress from fabrication and the applied load. The criteria for designating a part's tensile stress as not significant are:

1. The calculated stress levels are judged to be low.
2. No previous IGSCC problems based on long-term GE field experience.

Austenitic stainless steel components associated with the jet pump, low-pressure coolant injection (LPCI), core spray, and feedwater component of the reactor internals permit core cooling for normal and accident conditions.

For jet pump parts, a Type 304 grade (maximum carbon content 0.055 percent) stainless steel was used. Welded parts (riser elbow, riser pipe, and riser support) are subjected to low applied tensile stress, and reactor experience on those parts indicates no IGSCC.

For LPCI parts, a Type 304 grade (maximum carbon content 0.055 percent) stainless steel was used. These welded parts (ring and flange) are subjected to low applied tensile stresses during normal plant operation, and IGSCC is not expected.

For core coolant parts, Type 316L grade and Type 304L grade (maximum carbon content 0.035 percent) stainless steels meet the maximum carbon limits for preventing sensitization by welding. Therefore, IGSCC due to sensitization is not expected.

For feedwater parts, a Type 316L grade (maximum carbon content 0.035 percent) stainless steel meets the maximum carbon limits for preventing sensitization of the material by welding. Therefore, IGSCC is not expected.

In general, IGSCC due to sensitization is not a problem for Unit 2 internal components since applied stresses are low and, in addition, for many parts, low carbon stainless steels that are not expected to experience sensitization were utilized. Continued operation with high-purity coolant will further reduce the potential for IGSCC.

Industry-wide efforts are underway to address the potential for IGSCC in all vessel internal and attachment components.* Ongoing and proposed studies will provide additional information regarding the relative IGSCC susceptibility and expected lifetime of internals and attachment welds over the next several years. These studies will also help quantify benefits of operating with high-purity water under HWC conditions.

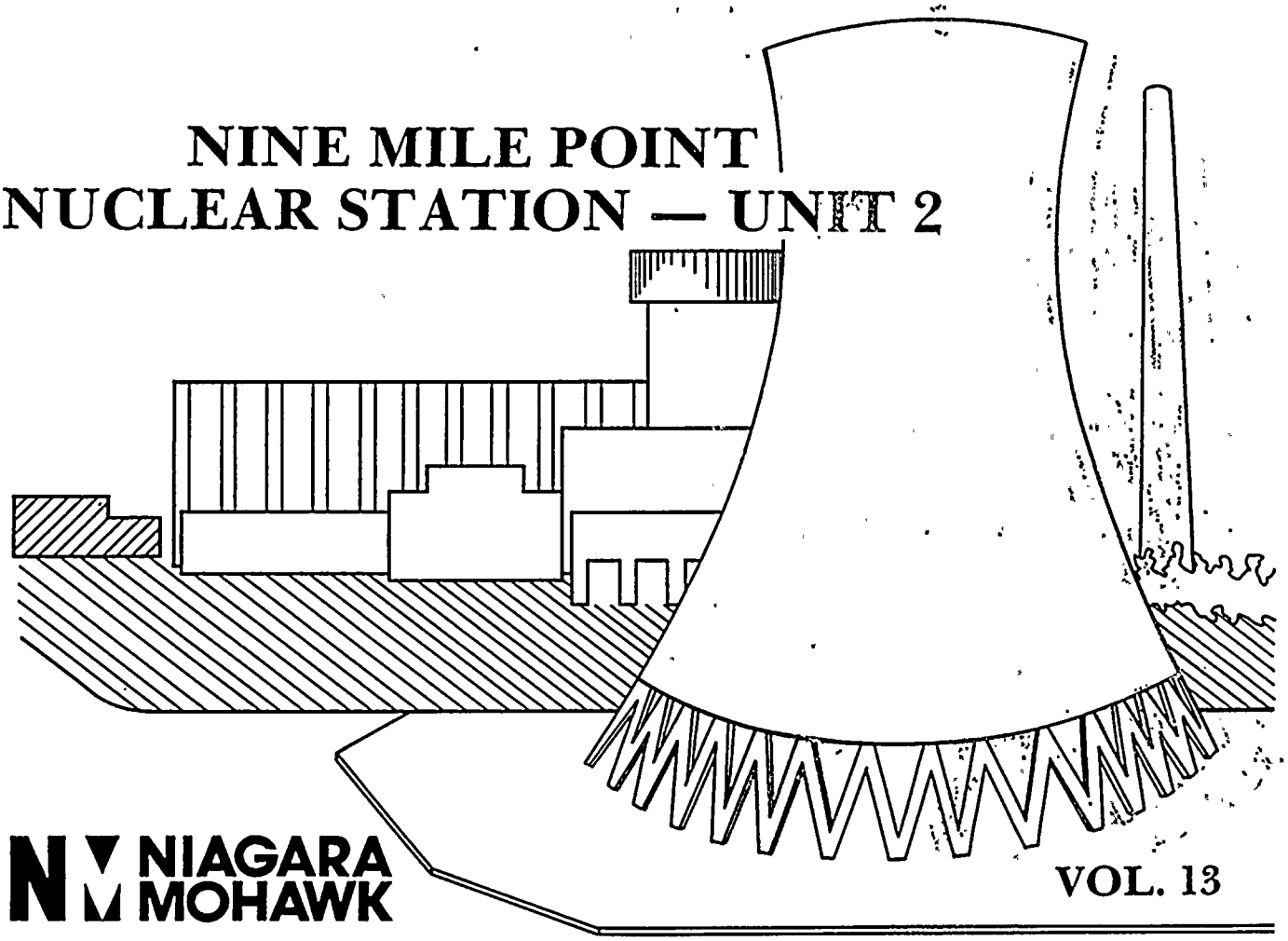
A crack arrest verification (CAV) system is installed in Unit 2 to provide real-time indication of the performance of plant materials in the BWR environment. The CAV system is discussed in Section 5.2.3.2.5.

---

* BWROG/NRC meeting held in Rockville, MD, on March 22, 1989, to review the BWR industry program for RPV and internals inspection.

UPDATED SAFETY ANALYSIS REPORT

NINE MILE POINT NUCLEAR STATION — UNIT 2



NIAGARA MOHAWK

VOL. 13

CHAPTER 5

REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

TABLE OF CONTENTS

CHAPTER 5

TABLE OF CONTENTS (Cont'd.)

CHAPTER 5

TABLE OF CONTENTS (Cont'd.)

CHAPTER 5

TABLE OF CONTENTS (Cont'd.)

CHAPTER 5

TABLE OF CONTENTS (Cont'd.)

CHAPTER 5

TABLE OF CONTENTS (Cont'd.)

CHAPTER 5

LIST OF TABLES

CHAPTER 5

LIST OF FIGURES

CHAPTER 5

LIST OF FIGURES

CHAPTER 5

LIST OF FIGURES (Cont'd.)

TABLE 5.2-5

REACTOR COOLANT PRESSURE BOUNDARY MATERIALS

Component	Form	Material	Specification (ASTM/ASME)	
Reactor Pressure Vessel				
Reactor vessel heads, shells	Rolled plate or forgings	Low alloy steel	SA-533 Gr. B Cl. 1 or SA-508 Cl. 2	
	Welds	Low alloy steel	SFA5.5	
Closure flange	Forged ring	Low alloy steel	SA-508 Cl. 2	
	Welds	Low alloy steel	SFA5.5	
Nozzles	Forged shapes	Low alloy steel	SA-508 Cl. 2	
	Welds	Low alloy steel	SFA5.5	
Nozzle safe ends	Forgings or plates	Stainless steel	SA-182, F304, or F316 SA-336, F8 or F8M SA-240, 304 or 316	
	Welds	Stainless steel	SFA5.9 Tp. 308L or 316L SFA5.4 Tp. 308L or 316L	
Nozzle safe ends	Forgings	Ni-Cr-Fe	SB-166 or SB-167	
	Welds	Ni-Cr-Fe	SFA5.14 Tp. ER NiCr-3 or SFA5.11 Tp. ENi CrFe-3	24
Nozzle safe ends	Forgings	Carbon steel	SA-508 Cl. 1	
	Welds	Carbon steel	SFA5.1, SFA5.18 GPA or SFA5.17 F70	
Cladding	Weld overlay	Austenitic stainless Steel	N/A	
Main Steam Safety Relief Valve				
Body	Cast	Carbon steel	SA-352 LCB	24
Seat	Forging	Carbon steel	SA-350 LF2	

TABLE 5.2-5 (Cont)

Component	Form	Material	Specification (ASTM/ASME)	
Disc	Cast	Stainless steel	SA-351 CF3A	
Main Steam Flow Element				
Instrument nozzle	Forging	Carbon steel	SA-105	
Upstream casting	Cast	Stainless steel	SA-351 Tp. CF8	
Downstream casting	Cast	Stainless steel	SA-216 Gr. WCB	
Main Steam Piping				
Pipe	Seamless	Carbon steel	SA-106 Gr. B, Gr. C	
	Seamless	Chrome-moly	SA-335 P22	
Pipe (penetration)	Seamless	Carbon steel	SA-106 Gr. B	
Contour nozzle (26" x 6" I.D. - 1,500 lb)	Forged	Carbon steel	SA-105	23
Tongue flange (4" - 900 lb)	Forged	Carbon steel	SA-105	
Elbow	Forged fittings	Chrome-moly	SA-182 Gr. F22	23
	Welded or seamless fittings	Carbon steel	SA-234 Gr. WPC or Gr. WPCW SA-234 Gr. WPB	23
Sockolet (1" - 3,000 lb, 2", 3/4" - 6,000 lb)	Forged	Carbon steel	SA-105	23
Head fitting - groove	Forged	Carbon steel	SA-508 Cl. 1	
Concentric reducer	Welded fitting	Carbon steel	SA-234, Gr. WPB	
Flange (8" - 1,500 lb)	Forged	Carbon steel	SA-105	
Sweepolet (26" x 10")	Forged	Carbon steel	SA-105	23

TABLE 5.2-5 (Cont)

Component	Form	Material	Specification (ASTM/ASME)	
Main Steam Isolation Valve				
Valve body	Cast	Carbon steel	SA-216 Gr. WCC	28
Bonnet	Forged	Carbon steel	SA-350 Gr. LF2	
Stem	Forged	Stainless steel	A-182 Gr. F6A Cl. 3 or SA-564, Type 630	
Disk	Forged	Carbon steel	SA-350 Gr. LF2	28
Stem disk	Forged	Carbon steel	SA-350 Gr. LF2	
Bonnet studs	Forged	Alloy Steel	SA-540 Gr. B23 Cl. 4	
Hex nuts	Forged	Alloy steel	SA-194 Gr. 7	
Leakoff plug	Forged	Carbon steel	SA-105	
Recirculation Gate Valve				
Body	Cast	Stainless steel	SA-351 CF8M	
Bonnet	Cast	Stainless steel	SA-351 CF8M	
Disc	Cast	Stainless steel	SA-351 CF3A	
Stem	Rod	Stainless steel	SA-564 Tp. 630 Cond. 1150	
Studs	Bolting	Alloy steel	SA-193 Gr. B7	
Nuts	Bolting	Alloy steel	SA-194 Gr. 7	
Recirculation Flow Control Valve				
Body	Cast	Stainless steel	SA-351 Gr. CF8M, 316	
Housing	Cast	Stainless steel	SA-351 Gr. CF8M, 316	
Bonnet	Cast	Stainless steel	SA-351 Gr. CF8M, 316	
Covers	Cast	Stainless steel	SA-351 CF8M, 316	

TABLE 5.2-5 (Cont)

Component	Form	Material	Specification (ASTM/ASME)
Recirculation Pump			
Pump case assembly	Cast	Stainless steel	SA-351 CF8M
Stud case to stuffing box	Bolting	Alloy steel	SA-540 Gr. B23 Cl. 5
Stud nut (3 1/4-8N)	Bolting	Alloy steel	SA-194 Gr. B7
Stuffing box	Cast	Stainless steel	SA-351 CF8M

TABLE 5.2-5 (Cont)

Component	Form	Material	Specification (ASTM/ASME)
Nozzle - 3/4"	Forging	Stainless steel	SA-182 Tp. F304/F316
Flange nozzle - 1"	Forging	Stainless steel	SA-182 Tp. F304/F316
Thrust ring	Forging	Stainless steel	SA-182 Tp. F316 with flash chrome plated
Pump flange	Forging	Carbon steel	SA-350 Gr. LF2
Seal holder	Forging	Stainless steel	SA-351 Gr. CF8M
Elbow	Plate	Stainless steel	SA-240 Tp. 304/316
Clip - outer	Bar	Stainless steel	A276 Tp. 304/316 Cond A
Clip - inner	Bar	Stainless steel	A276 Tp. 304/316 Cond A
Clip - tube end	Bar	Stainless steel	A276 Tp. 304/316 Cond A
Upper seal gland	Forging	Stainless steel	SA-351 Gr. CF8M
Grayloc clamp	Cast	Stainless steel	SA-351 Gr. CF8M
Pipe - 1" Sch 80	Plate	Stainless steel	SA-312 Gr. Tp. 304/316
Grayloc hub - 1"	Forging	Stainless steel	SA-182 Tp. F316
Tee - 1" pipe 3000#	Forging	Stainless steel	SA-182 Tp. F316
Thermowell	Forging	Stainless steel	SA-182 Tp. F304/F316
Elbow - 1"	Fitting	Stainless steel	SA-403 Tp. WP304/316
Pipe - 1"	Pipe	Stainless steel	SA-312 Gr. Tp. 304/316
Flange - 1"	Forging	Stainless steel	SA-182 Tp. F304/F316
Grayloc hub	Forging	Stainless steel	SA-182 Tp. F316
Tee	Forging	Stainless steel	SA-182 Tp. F316
Pipe plug	Forging	Stainless steel	SA-182 Tp. F304/F316

TABLE 5.2-5 (Cont)

REACTOR COOLANT PRESSURE BOUNDARY MATERIALS

Component	Form	Material	Specification (ASTM/ASME)
Pipe 3/4	Pipe	Stainless steel	SA-312 Gr. Tp. 304/316
Tee	Forging	Stainless steel	SA-182 Tp. F316
Flange 3/4	Forging	Stainless steel	SA-182 Tp. F304/F316
Grayloc hub	Forging	Stainless steel	SA-182 Tp. F316
Valve body	Plate	Stainless steel	SA-240 Tp. 304/316
Valve bonnet	Plate	Stainless steel	SA-240 Tp. 304/316
Coil - inner	Tubing	Stainless Steel	SA-213 Gr. Tp. 316
Pipe cap	Forging	Stainless steel	SA-182 Tp. F316
Recirculation Piping			
Pipe	Rolled and welded	Stainless steel	SA-358 Tp. 316K
Cross, tee, cap, contour nozzle, elbow, concentric reducer	Fittings	Stainless steel	SA-403 Tp. 316K
Control Rod Drive			
Flanges, plugs, head	Forging	Stainless steel	SA-182 F304
Nut	Bar	Stainless steel	SA-193 Gr. B8
Indicator tube	Pipe	Stainless steel	SA-312 Tp. 316
Drive housing	Pipe	Stainless steel	SA-312 Tp. 304
	Welds	Stainless steel	SFA5.9 Tp. 308L SFA5.4 Tp. 308L
	Forging	Stainless steel	SA-182 Tp. F304
In-core housing	Tube	Stainless steel	SA-213 Tp. 304
	Welds	Stainless steel	SFA5.9 Tp. 308L SFA5.4 Tp. 308L
	Forging	Stainless steel	SA-182 Tp. F304

24

TABLE 5.2-5 (Cont)

THE INFORMATION ON THIS PAGE HAS BEEN DELETED.

TABLE 5.2-6

BWR WATER CHEMISTRY

	Concentration (ppb)				Conductivity		
	Iron	Copper	Chloride	Oxygen	µmho/cm (25°C)	pH (25°C)	
Condensate (1)	15-30	3-5	≤20	20-50	0.1	7*	
Condensate treatment effluent (2)	≤10	<2	<2	20-50	<0.1	6.5-7.5	23
Feedwater (3)	5-15	<2	<2	20-200	0.1	6.5-7.5	23
Reactor water (4)							
Normal operation	10-50	<20	<200	100-300	<1	5.6-8.6	23
Shutdown	-	-	500		<1	4-10	
Hot standby	-	-	<20	See Section 5.2.3.2.2	<1	7*	
Depressurized	-	-	500	8,000	<10	5.3-8.6	23
Steam (5)	0	0	0	10,000-30,000	0.1*		
Control rod drive cooling water (6)	50-500	-	<20	≤50	≤0.1	7*	

NOTE: Numerals in parentheses refer to locations delineated on Figure 5.2-6.

*Approximately

TABLE 5.2-9

SUMMARY OF SYSTEM ISOLATION/ALARMS OF SYSTEMS MONITORED AND THE LEAK DETECTION METHODS USED

(SUMMARY OF ISOLATION SIGNALS AND ALARMS SYSTEM ISOLATION VS VARIABLE MONITORED) (3)

SYSTEM ISOLATED (1) \ VARIABLE MONITORED	REACTOR VESSEL WATER LEVEL (2,4)	REACTOR PRESSURE (2)	TURBINE BUILDING LEAK DETECTION (AMBIENT TEMP, HIGH)	MS TUNNEL AMBIENT TEMP, HIGH	MS TUNNEL DIFFERENTIAL TEMP, HIGH	MAIN STEAM LINE FLOW RATE, HIGH	DRYWELL PRESSURE, HIGH (2)	RHR EQUIPMENT AREA AMBIENT TEMP, HIGH	RCIC EQUIPMENT AREA AMBIENT TEMP, HIGH	RCIC EXHAUST DIAPHRAGM PRESSURE, HIGH (2)	RHR/RCIC STEAM SUPPLY DIFFERENTIAL PRESSURE (HIGH FLOW)	RHR/RCIC STEAM SUPPLY DIFFERENTIAL PRESSURE (INSTR LINE BREAK)	RWCU PROCESS PIPING DIFFERENTIAL FLOW, HIGH	RWCU EQUIPMENT AREA AMBIENT TEMP, HIGH	REACTOR BLDG. PIPE CHASE AREA AMBIENT TEMP, HIGH	REACTOR BLDG. GENERAL AREA AMBIENT TEMP, HIGH
MAIN STEAM	1		I	I	I	I										
RECIRC (SAMPLE LINE)	2															
RHR	3	I						I			I	I			I	I
RCIC								I	I	I	I	I			I	I
RWCU	2												I	I	I	
CONTAINMENT ISOLATION	2						I									

I = ISOLATE ALARM, AND INDICATE OR RECORD

(1) SYSTEMS OR SELECTED VALVES WITHIN THE SYSTEM THAT ISOLATE.
(2) THESE LEAK DETECTION SIGNALS ARE PROVIDED BY OTHER SYSTEMS.
(3) AN ALARM IS ASSOCIATED WITH EACH ISOLATION SIGNAL.
(4) NUMERALS IN THIS COLUMN CORRESPOND TO REACTOR WATER LEVELS AS SHOWN ON FIGURE 5.3-2 AND ARE LEVELS AT WHICH ISOLATION VALVES OF THE RELATED SYSTEM ARE CLOSED.

TABLE 5.2-10

SUMMARY OF ISOLATION ALARMS OF SYSTEM MONITORED
AND LEAK DETECTION METHODS USED
(SUMMARY OF VARIABLE TRIP ALARMS LEAKAGE SOURCE VS GENERATED VARIABLES)

SOURCE OF LEAKAGE \ AFFECTED VARIABLE MONITORED	LOCATED INSIDE PRIMARY CONTAINMENT	LOCATED OUTSIDE PRIMARY CONTAINMENT	DRYWELL PRESSURE, HIGH*	REACTOR WATER LEVEL, LOW*	DRYWELL FLOOR DRAIN TANK FILL RATE, HIGH	DRYWELL EQUIPMENT DRAIN TANK FILL RATE, HIGH	FISSION PRODUCT RADIATION, HIGH*	DRYWELL TEMPERATURE, HIGH	SAFETY/RELIEF VALVE DISCHARGE PIPE TEMP, HIGH	MSL TUNNEL RADIATION, HIGH	PUMP SEAL FLOW, HIGH	SEAL PRESSURE, HIGH	FLOW RATE, HIGH	FLOOR DRAIN SUMP, HIGH FILL UP/PUMP OUT (REACTOR BUILDING)	MSL TUNNEL AMBIENT AND DIFFERENTIAL TEMP, HIGH (REAC BLDG)	EQPT AREA AMBIENT TEMP, HIGH (REAC BLDG)	RWCU DIFFERENTIAL FLOW, HIGH	SEAL DRAIN FLOW, HIGH	INTERSYSTEM LEAKAGE (RADIATION), HIGH	ECCS INJECTION LINE LEAKAGE (INTERNAL TO REACTOR VESSEL)	DIFFERENTIAL PRESSURE	REACTOR BLDG. PIPE CHASE AREA AMBIENT TEMP, HIGH	REACTOR BLDG. GENERAL AREA AMBIENT TEMP, HIGH
MAIN STEAM LINE	X		A	A	A		A	A	A				A										
		X		A						A			A		A								
RCIC/RHR STEAM LINE	X		A	A	A		A	A					A										
		X		A									A	A		A						A	A
RCIC STEAM LINE																							
		X		A									A	A		A						A	A
RWCU WATER	X		A	A	A		A	A															
		X		A										A		A	A					A	
HPCS WATER	X				A															A			
		X												A									
LPCS WATER	X				A															A			
		X												A									
RECIRC PUMP SEAL	X		A			A	A	A			A												
FEEDWATER	X		A	A	A		A	A															
		X										A											
RHR WATER	X		A	A	A			A					A							A			
		X		A										A		A			A				A
REACTOR VESSEL HEAD SEAL	X											A											
REFUELING POOL																							
		X												A				A					
MISCELLANEOUS LEAKS	X				A																		
		X												A									
RCIC WATER	X				A																		
		X												A									

KEY:
A = ALARM AND INDICATE OR RECORD ONLY.
X = LOCATION OF LEAKAGE SOURCE.
* THESE LEAK DETECTION SIGNALS ARE PROVIDED BY OTHER SYSTEMS

5.4 COMPONENT AND SUBSYSTEM DESIGN

5.4.1 Reactor Recirculation System

5.4.1.1 Safety Design Bases

The reactor recirculation system has been designed to meet the following safety design bases:

1. An adequate fuel barrier thermal margin will be assured during postulated transients.

2. A failure of piping integrity will not compromise the ability of the reactor vessel internals to provide a refloodable volume.

3. The system will maintain pressure integrity during adverse combinations of loadings and forces occurring during abnormal, accident, and special event conditions.

5.4.1.2 Power Generation Design Bases

The reactor recirculation system meets the following power generation design bases:

1. The system will provide sufficient flow to remove heat from the fuel.

2. System design will minimize maintenance situations that would require fuel removal.

5.4.1.3 Description

The reactor recirculation system consists of the two recirculation pump loops external to the reactor vessel. These loops provide the piping path for the driving flow of water to the reactor vessel jet pumps (Figures 5.4-1 and 5.4-2). Each external loop contains one high-capacity motor-driven recirculation pump, a flow control valve (FCV), and two motor-operated gate valves (for pump maintenance). Each pump suction line contains a flow measuring system. The recirculation loops are part of the RCPB and are located inside the drywell structure. The jet pumps are considered to be reactor vessel internals. Their location and mechanical design are discussed in Section 3.9.5B. However, certain operational characteristics of the jet pumps are discussed in this section. A tabulation of the important design and performance characteristics of the reactor recirculation system is shown in Table 5.4-1. The head, net positive suction head (NPSH), flow, and efficiency curves are shown on Figure 5.4-3. Instrumentation and controls for the recirculation flow control system are described in Section 7.7.1.2.

The recirculated coolant consists of saturated water from the steam separators and dryers that has been subcooled by incoming feedwater. This water passes down the annulus between the reactor vessel wall and the core shroud. A portion of the coolant flows from the vessel, through the two external recirculation loops, and becomes the driving flow for the jet pumps. Each of the two external recirculation loops discharges high pressure flow into an external manifold from which individual recirculation inlet lines are routed to the jet pump risers within the reactor vessel. The remaining portion of the coolant mixture in the annulus becomes the suction flow for the jet pumps. This flow enters the jet pump at suction inlets and is accelerated by the driving flow. The flows, both driving and suction, are mixed in the jet pump throat section and result in partial pressure recovery. The balance of recovery is obtained in the jet pump diffuser (Figure 5.4-4). The adequacy of the total flow to the core is discussed in Section 4.4.

The allowable heatup rate for the recirculation pump casing is 100°F/hr, which is the same as that for the reactor vessel. If one loop is shut down, the idle loop can be kept hot by leaving the loop valves open; with the FCV in the minimum position, this permits the reactor pressure plus the active jet pump head to cause reverse flow in the idle loop.

The objective of the recirculation gate valve trim design is to minimize the need for maintenance of the valve internals. The valves have high-quality backseats that permit renewal or replacement of stem packing while the system is full of water.

The pump/motor operates at two speeds, 25 and 100 percent. When operation is at 25 percent speed, the power comes from the low frequency motor generator (LFMG) set which operates at 15 Hz. Power for 100-percent speed operation comes from a 60-Hz source.

When the pump is operating at 25-percent speed, the head provided by the elevation of the reactor water level above the recirculation pump is sufficient to provide the required NPSH for the recirculation pumps (FCV and jet pumps). When the pump is operating at 100-percent speed, most of the NPSH is supplied by the subcooling provided by the feedwater flow. The subcooling temperature is measured by detectors that are provided in the recirculation lines, and by the steam dome delta pressure that is converted to temperature. The difference between these two readings is a direct measurement of the subcooling. If the subcooling falls below approximately 11°F, the 100-percent speed power supply is tripped to the 25-percent speed power source to prevent cavitation of recirculation pump, jet pumps, and/or the FCV.

The 100-percent speed trip to 25-percent speed is a step as far as the power supplies are concerned but linear as far as motor is concerned. The pump is tripped from 100-percent speed; it coasts

down to 25-percent speed when the LFMG comes on line and the pump is run at 25 percent.

When preparing for hydrostatic tests, the nuclear system temperature must be raised above the vessel NDTT limit.

The vessel is heated by core decay heat and/or by operating the recirculation pumps at 100-percent speed and FCVs at minimum position.

Each recirculation pump is equipped with mechanical shaft seal assemblies. The two seals built into a cartridge can be readily replaced without removing the motor from the pump. Each individual seal in the cartridge is designed for pump operating pressure so that any one seal can adequately limit leakage in the event that the other seal should fail. The pump shaft passes through a breakdown bushing in the pump casing to reduce leakage in the event of a gross failure of both shaft seals. The cavity temperatures and pressures of each individual are continuously monitored.

Each recirculation pump motor is a dual-speed, vertical, solid-shaft, totally-enclosed, air-water cooled, induction motor. The combined rotating inertias of the recirculation pump and motor provide a slow coastdown of flow following loss of power to the drive motors so that the core is adequately cooled during the transient. This inertia requirement is met without a flywheel.

The pump discharge FCV can throttle the discharge flow of the pump proportionally to an instrument signal. The FCV has an equal percentage characteristic. The recirculation loop flow rate can be rapidly changed, within the expected flow range, in response to rapid changes in system demand.

The design objective for the recirculation system equipment is to provide equipment that does not require removal from the system for rework or overhaul. Pump casing and valve bodies are designed for a 40-yr life and are welded to the pipe.

The pump drive motor, impeller, and wear rings and FCV internals are designed for a long operational life. Pump mechanical seal parts and the valve packing are expected to have a life expectancy that affords convenient replacement during the refueling outages.

The recirculation system piping is of all-welded construction. The effective ASME III Code for the recirculation piping system, based on the piping contract award and the requirements of NCA1140, is the 1977 Edition, Summer 1977 Addenda. No one contractor has overall responsibility for the complete scope, including design, procurement, material supply, and installation. Therefore, on the N-5 Data Report (modified), GE certifies that the design, procurement, and material supply activities performed by GE are in accordance with the ASME III Code; and RCI, the

system installer (NA Certificate Holder), certifies that the piping system was installed in accordance with the design specification and ASME III Code. SWEC, as the N Certificate Holder assuming overall responsibility for the balance of the plant, performed the pressure testing activities. The piping system designer (GE) and installer (RCI and its AIA, Hartford Steam Boiler) witnessed the pressure testing of those items for which they were responsible.

The reactor recirculation system pressure boundary equipment is designed as Category I equipment. As such, it is designed to resist sufficiently the response motion for the SSE at the installed location within the supporting structure. The pump is assumed to be filled with water for the analysis. Snubbers located at the top of the motor and at the bottom of the pump casing are designed to resist dynamic reactions.

The recirculation piping, valves, and pumps are supported by hangers to avoid the use of piping expansion loops that would be required if the pumps were anchored. In addition, the recirculation loops have a system of restraints designed so that reaction forces associated with the postulated pipe breaks do not jeopardize drywell integrity. This restraint system provides adequate clearance for normal thermal expansion movement of the loop. The criteria for protection against the dynamic effects associated with a postulated pipe rupture are contained in Section 3.6B. The recirculation system piping, valves, and pump casings are covered with thermal insulation having a total maximum heat transfer rate of 65 Btu/hr-sq ft with the system at rated operating conditions. This heat loss includes losses through joints, laps, pipe supports and restraints, and other openings that may exist in the insulation. The maximum heat transfer is based upon a recirculation system temperature of 550°F and a drywell temperature of 135°F.

The insulation is the metal reflective type of Min-K type. It is prefabricated into components for field installation. Removable insulation is provided at various locations to permit periodic inspection of the equipment.

5.4.1.4 Safety Evaluation

Reactor recirculation system malfunctions that pose threats of damage to the fuel barrier are described and evaluated in Chapter 15; it is shown that none of the malfunctions could result in significant fuel damage. The recirculation system has sufficient flow coastdown characteristics to maintain fuel thermal margins during abnormal operational transients.

The core-flooding capability of a jet pump design plant is discussed in detail in the ECCS document filed with the NRC as a GE topical report[1]. The ability to reflood the BWR core to the top of the jet pumps is shown schematically on Figure 5.4-5 and is discussed in Reference 1.

Piping and pump design pressures for the reactor recirculation system are based on peak steam pressure in the reactor dome, appropriate pump head allowances, and the elevation head above the lowest point in the recirculation loop. Piping and related equipment pressure parts are chosen in accordance with applicable codes shown in Table 3.9B-2. Use of the listed code design criteria assures that a system designed, built, and operated within design limits has an extremely low probability of failure caused by any known failure mechanism.

Purchase specifications require that the recirculation pumps' first critical speed not be less than 130 percent of operating speed. Calculations performed by the vendor to conform to this specification were reviewed and approved by the NSSS vendor. Purchase specifications require that integrity of the pump case be maintained through all transients and that the pump remain operable through all normal and upset transients. The design of the pump and motor bearings is required to be such that dynamic load capability at rated operating conditions is not exceeded during the SSE. Calculations performed by the vendor to substantiate this were reviewed and approved by the NSSS vendor.

Pump overspeed occurs during the course of a LOCA due to blowdown through the pump in the broken loop pump. Design studies determined that the overspeed is not sufficient to cause destruction of the motor.

5.4.1.5 Inspection and Testing

Quality control methods were used during fabrication and assembly of the reactor recirculation system to assure that design specifications are met. Inspection and testing is carried out as described in Chapter 3. The RCS is thoroughly cleaned and flushed before fuel is loaded initially.

Before the preoperational test program, the reactor recirculation system is hydrostatically tested at 125-percent reactor vessel design pressure. Preoperational tests on the reactor recirculation system also include checking operation of the pumps, flow control system, and gate valves and are discussed in Chapter 14.

During the startup test program, horizontal and vertical motions of the reactor recirculation system piping and equipment are observed; supports and restraints are adjusted, as necessary, to assure that components are free to move as designed. Nuclear system responses to RPTs at rated temperatures and pressure are evaluated during the startup tests, and plant power response to recirculation flow control is determined.

5.4.2 Steam Generators (PWR)

Section 5.4.2 is not applicable to this Final Safety Analysis Report (FSAR).

5.4.3 Reactor Coolant Piping

The reactor coolant piping is discussed in Sections 3.9.3.1.4 and 5.4.1. The recirculation loops are shown on Figures 5.4-1 and 5.4-2. The design characteristics are presented in Table 5.4-1. Avoidance of stress corrosion cracking is discussed in Section 5.2.3.4.1.

5.4.4 Main Steam Line Flow Restrictors

5.4.4.1 Safety Design Bases

The MSL flow restrictors are designed:

1. To limit the loss of coolant from the reactor vessel following a steam line rupture outside the containment to the extent that the reactor vessel water level remains high enough to provide cooling within the time required to close the MSIVs.

2. To withstand the maximum pressure difference expected across the restrictor, following complete severance of a MSL.

3. To limit the amount of radiological release outside the drywell prior to MSIV closure.

4. To provide trip signals for MSIV closure.

5. In accordance with ASME Fluid Meters, Sixth Edition, 1971.

5.4.4.2 Description

A main steam flow restrictor (Figure 5.4-6) is provided for each of the four MSLs. The restrictor is a complete assembly welded into the MSL. It is located in the drywell. The restrictor limits the coolant inventory loss and loss rate from the reactor vessel in the event a MSLB occurs outside the containment. The loss is limited to a maximum (choke) flow of 7.08×10^6 lb/hr at 1,015 psig upstream pressure. The restrictor assembly consists of a venturi-type nozzle insert welded (in accordance with applicable code requirements), into the MSL.

The flow restrictor has no moving parts. Its mechanical structure can withstand the velocities and forces associated with a MSLB. The maximum differential pressure is conservatively assumed (by calculation) to be 1,265 psia.

The ratio of venturi throat diameter to steam line inside diameter of approximately 0.551 results in a maximum pressure differential (unrecovered pressure) of about 6.26 psi at 100 percent of rated flow. This design limits the steam flow in a severed line to 190.5 percent of rated flow, yet it results in

negligible increase in steam moisture content during normal operation. The restrictor is also used to measure steam flow to initiate closure of the MSIVs when the steam flow exceeds the preselected operational limits.

5.4.4.3 Safety Evaluation

In the event a MSL should break outside the containment, the critical flow phenomenon would restrict the steam flow rate in the venturi throat to 190.5 percent of the rated value. Prior to isolation valve closure, the total coolant losses from the vessel are not sufficient to cause core uncovering and the core is thus adequately cooled at all times.

Analysis of the steam line rupture accident (Chapter 15) shows that the core remains covered with water and that the amount of radioactive materials released to the environs through the MSLB results in doses to the public that are a fraction of 10CFR100 limits.

Tests on a scale model determined final design and performance characteristics of the flow restrictor. The characteristics include maximum flow rate of the restrictor corresponding to the accident conditions, unrecoverable losses under normal plant operating conditions, and discharge moisture level. The tests showed that flow restriction at critical throat velocities is stable and predictable.

The steam flow restrictor is exposed to steam of about 0.2-percent moisture flowing at velocities of 140 ft/sec (steam piping ID) to 450 ft/sec (steam restrictor throat). ASTM A351 (Type 304 Grade CF8) cast stainless steel was selected for the steam flow restrictor throat material, because it has excellent resistance to erosion and corrosion in a high-velocity steam atmosphere. The excellent performance of stainless steel in high-velocity steam is due to its resistance to erosion and corrosion. A protective surface film forms on the stainless steel, which prevents any surface attack, and this film is not removed by the steam.

Hardness has no significant effect on erosion and corrosion. For example, hardened carbon steel or alloy steel erodes rapidly in applications where soft stainless steel is unaffected.

Surface finish has a minor effect on erosion and corrosion. If very rough surfaces are exposed, the protruding ridges or points erode more rapidly than a smooth surface. Experience shows that a machined or a ground surface is sufficiently smooth and that no detrimental erosion occurs.

5.4.4.4 Inspection and Testing

Because the flow restrictor forms a permanent part of the main steam piping and has no moving components, no testing program is

planned beyond testing the main steam system piping. Only very slow erosion occurs with time, and such a slight enlargement has no safety significance. Stainless steel resistance to erosion has been substantiated by turbine inspections at the Dresden Unit 1 facility which revealed no noticeable effects from erosion on the stainless steel nozzle partitions. The Dresden inlet velocities are about 300 ft/sec and the exit velocities are 600 to 900 ft/sec. However, calculations show that, even if the erosion rates are as high as 0.004 in/yr, after 40 yr of operation the increase in restrictor choked flow rate would be no more than 5 percent. A 5-percent increase in the radiological dose calculated for the postulated MSLB accident is not significant.

5.4.5 Main Steam Isolation System

5.4.5.1 Safety Design Bases

The MSIVs, individually or collectively, will:

1. Isolate the MSLs within the time established by DBA analysis to limit the release of reactor coolant.

2. Isolate the MSLs slowly enough that simultaneous isolation of all steam lines does not induce transients that exceed the nuclear system design limits.

3. Isolate the MSL when required, despite single failure in either valve or in the associated controls, to provide a high level of reliability for the safety function.

4. Use separate energy sources as the motive force to isolate independently the redundant isolation valves in the individual steam lines.

5. Use the local stored energy (compressed air and springs) to close at least one MSIV without relying on the continuity of any variety of electrical power to furnish the motive force to achieve closure.

6. Be able to isolate the MSLs, either during or after seismic and/or hydrodynamic loadings, to assure isolation if the nuclear system is breached.

7. Have capability for testing, during normal operating conditions, to demonstrate that the valves are functional.

5.4.5.2 Description

Two isolation valves are welded in a horizontal run of each of the four MSLs. One valve is as close as possible to the inside

of primary containment and the other is just outside the containment.

Figure 5.4-7 shows a MSIV. Each is a 26-in, Y-pattern, globe valve. Design steam flow rate through each line is 3.750 x 10^6 lb/hr. The main disk or poppet is attached to the lower end of the stem. Normal steam flow tends to close the valve, and higher inlet pressure tends to hold the valve closed.

The bottom end of the valve stem closes a small pressure-balancing hole in the poppet. When the hole is open, it acts as a pilot valve to relieve differential pressure forces on the poppet. Valve stem travel is sufficient to give flow areas past the wide-open poppet approximately equal to the seat port area. The poppet travels approximately 90 percent of the valve stem travel; approximately the last 10 percent of travel closes the pilot hole. The air cylinder can open the poppet with a maximum differential pressure of 200 psi across the isolation valve in a direction that tends to hold the valve closed.

A 45° angle permits the inlet and outlet passages to be streamlined. This minimizes pressure drop during normal steam flow and helps prevent debris blockage. The pressure drop at rated flow is 10 psi maximum. The valve stem penetrates the valve bonnet through a stuffing box that has replaceable packing. To help prevent leakage through the stem packing, the poppet backseats when the valve is fully open.

Attached to the upper end of the stem is an air cylinder that opens and closes the valve and a hydraulic dashpot that controls its speed. The speed is adjusted by a valve in the hydraulic return line bypassing the dashpot piston. Valve closing time is adjustable to between 3 and 10 sec.

The air cylinder is supported on the valve bonnet by actuator support and spring guide shafts. Helical springs around the spring guide shafts close the valve if air pressure is not available. (Spring closure of the valve due to loss of air supply pressure is assisted by air from an air tank accumulator directed to the top of the actuator cylinder). The motion of the spring seat member actuates a scram switch in the 85-percent open valve position and indicator light switches in the 90-percent open and 10-percent open valve positions.

The valve is operated by pneumatic pressure and by the action of compressed springs. The control unit is attached to the air cylinder. This unit contains air pilot valves and solenoid-operated valves (SOVs). The SOVs control opening and closing of the air valves and provide exercising capability at slow speed. Remote manual switches in the main control room enable the Operator to operate the valves.

Operating air is supplied to the outboard valves from the instrument air system, and to the inboard valves from the

instrument nitrogen system (Section 9.3.1). An air accumulator between the control valve and a check valve provides backup operating air. The leak-tightness of the check valve is tested periodically to assure sufficient air pressure in the accumulator to assist in closing the valve on demand.

Each valve is designed to accommodate saturated steam at plant operating conditions. The valves are furnished in conformance with a design pressure and temperature rating in excess of plant operating conditions to accommodate plant accident and transient overpressure conditions.

In the worst case, if the MSL should rupture, steam flow would quickly increase to 190.5 percent of rated flow. Further increase is prevented by the venturi flow restrictor in the MSL inside containment.

During approximately the first 75 percent of closing, the valve has little effect on flow reduction, because the flow is choked by the venturi restrictor. After the valve is approximately 75-percent closed, flow is reduced as a function of the valve area versus travel characteristic.

The design objective for the valve is a minimum of 40-yr service at the specified operating conditions. The estimated operating cycles are estimated to be 50 to 400 cycles/yr (full open to full close and return).

In addition to minimum wall thickness required by applicable codes, a minimum corrosion allowance of 0.120 in is added to provide for 40-yr service.

Design specification ambient conditions inside the primary containment for normal plant operation are specified as 135°F average and 150°F maximum temperature and 90-percent maximum humidity. Design normal gamma plus neutron radiation dose over a 5-yr maintenance period is 7.4×10^6 rads.

The MSIVs (inside and outside the primary containment) are designed to close under accident environmental conditions of 340°F for 1 hr at -5 to +45 psig drywell pressure. In addition, they will not fail open under the following postaccident environmental conditions:

1. 340°F for an additional 2 hr at drywell pressure of -5 to +45 psig.

2. 320°F for an additional 3 hr at -5 to +45 psig.

3. 250°F for an additional 18 hr at -5 to +45 psig.

4. 200°F during the next 99 days at -5 to +45 psig.

5. Relative humidity: 20 to 100 percent.

6. Radiation: 1.5 x 10^8 rads gamma (100-day integrated accident dose).

The MSIV systems are Category I equipment and the valves are designed, fabricated, inspected, and tested in accordance with ASME Section III, Safety Class 1. The valves are designed to be operable when subjected to various combinations of the following loads:

1. Operating base earthquake (OBE) and SSE.

2. A double-ended guillotine pipe break outside containment during plant operation at full power.

3. Worst-case loading imposed by the attached piping.

4. Suppression pool dynamic loads resulting from the discharge of SRVs and LOCA.

Operability, as used above, is defined as the ability of the valve system, when subjected to the described loadings, to close and remain closed.

The valve body and associated internal pressure boundary components are modeled using finite element techniques. Application of the described loadings results in stresses within the limits of ASME Section III, Subarticle NB-3500. Deformations were evaluated for worst-case conditions, for all regions where critical clearances or alignments might conceivably be compromised so as to jeopardize functional capability. Acceptable margins were determined for all such regions. A flaw was identified in the valve-to-pipe weld of MSIV 7A during the baseline examination of the replacement activities for this valve under ASME Section XI. This flaw was evaluated and meets the acceptance criteria of ASME Section XI, Table IWB-3514-1. The area containing this flaw will be reexamined during the next three inspection periods in accordance with IWB-2420(b).

All Class 1E electrical equipment of the MSIV system is qualified in accordance with IEEE-323-1974 and RG 1.89. Assurance of operability of the MSIV actuators and their control logic cabinets is demonstrated by a comprehensive dynamic testing program, in accordance with IEEE-344-1975 and RG 1.100. The ability of the operator to perform its safety function before, during, and after the tests, is demonstrated.

The qualification of all Class 1E electrical equipment of the MSIV system meets or exceeds the requirements for Category II qualification in accordance with NUREG-0588.

5.4.5.3 Safety Evaluation

In a direct-cycle nuclear power plant, the reactor steam goes to the turbine and to other equipment outside the containment.

Radioactive materials in the steam can be released to the environs through process openings (leaks) in the steam system, or escape from pipe breaks. A large break in the steam system can drain the water from the reactor core faster than it is replaced by feedwater. The MSIVs are provided to limit both the release of radioactive material and the drainage of water from the reactor due to a MSLB outside containment.

The analysis of a complete, sudden steam line break outside the containment is described in Chapter 15. The analysis shows that the fuel barrier is protected against loss of cooling if MSIV closure is within specified limits, including instrumentation delay to initiate valve closure after the break. The calculated radiological effects of the radioactive material postulated to be released following a MSLB are presented in Section 15.6.4.

The shortest closing time (approximately 3 sec) of the MSIVs is also shown in Chapter 15 to be satisfactory. The switches on the valves initiate reactor scram when specific conditions (extent of valve closure, number of pipelines included, and reactor power level) are exceeded (Section 7.2.1). The pressure rise in the system from stored and decay heat may cause the nuclear system relief valves to open briefly, but the rise in fuel cladding temperature is insignificant. No fuel damage results.

The ability of this 45-degree, Y-design globe valve to close in a few seconds after a steam line break, under conditions of high pressure differentials and fluid flows with fluid mixtures ranging from mostly steam to mostly water, has been demonstrated in a series of dynamic tests. A full-size, 20-in valve was tested in a range of steam-water blowdown conditions simulating postulated accident conditions.[3]

The following specified hydrostatic, leakage, and stroking tests, as a minimum, are performed by the valve manufacturer in shop tests:

1. To verify its capability to close between 3 and 10 sec, each valve is tested at 1,000 psig line pressure and no flow. The valve is stroked several times, and the closing time is recorded. The valve is closed by spring only and by the combination of air cylinder and springs. The closing time is slightly greater when closure is by springs only.

2. Leakage is measured with the valve seated and backseated. The specified maximum seat leakage, using cold water at design pressure, is 2 cm^3/hr/in of nominal valve size. In addition, an air seat leakage test is conducted using 50 psig pressure upstream. Maximum permissible leakage is 0.1 scfh/in of nominal valve size. There must be no visible leakage from either set of stem packing at hydrostatic test pressure. The valve stem is operated a minimum of

three times from full open to full closed and return to open position, and the packing leakage still must be zero by visual examination.

3. Each valve is hydrostatically tested in accordance with the requirements of the applicable edition and addenda of the ASME Code. During valve fabrication, extensive nondestructive tests and examinations are conducted. Tests include radiographic, liquid penetrant, or magnetic particle examinations of casting, forgings, welds, hardfacings, and bolts.

4. The spring guides, the guiding of the spring seat member on support shafts, and rigid attachment of the seat member assure correct alignment of the actuating components. Binding of the valve poppet in the internal guides is prevented by making the poppet in the form of a cylinder longer than its diameter and by applying stem force near the bottom of the poppet.

After the valves are installed in the MSLs, each valve is tested as discussed in Chapter 14.

Two isolation valves provide redundancy in each steam line so either can perform the isolation function, and either can be tested for leakage. The inside valve, the outside valve, and their respective control systems are separated physically. The design of the isolation valve system has been analyzed for earthquake and suppresssion pool dynamic loading. The stress caused by the specified dynamic loading on the actuator does not exceed material allowables or prevent the valve from closing as required.

Electrical equipment that is associated with the isolation valves and is required to operate in an accident environment is limited to the wiring, terminal blocks, wire lugs, relays, solenoid valves, trip solenoids, and position switches.

The expected pressure and temperature transients following an accident are discussed in Chapter 15. All these components are tested for their specified performance and qualified under the specified environmental conditions.

5.4.5.4 Inspection and Testing

The MSIVs will be functionally tested for operability during plant operation and refueling outages. The test provisions are as follows:

1. During a refueling outage, the MSIVs can be functionally tested, leak tested, and visually inspected.

2. During plant operation the MSIVs can be tested and exercised individually to a partially closed position without a significant effect on plant operation.

3. The MSIVs can also be tested and exercised individually to the fully closed position in accordance with the requirements of the Technical Specifications.

Leakage from the valve stem packings can be detected visually during shutdown. The leak rate through the valve seats can be measured accurately during shutdown.

During prestartup tests following an extensive shutdown, the valves receive the same hydrostatic tests as those imposed on the primary system. Such a test and leakage measurement program ensures that the valves are operating correctly and that a leakage trend is detected.

5.4.6 Reactor Core Isolation Cooling System

5.4.6.1 Design Bases

The RCIC system is a safety system that consists of a turbine, pump, piping, valves, accessories, and instrumentation designed to assure that sufficient reactor water inventory is maintained in the reactor vessel to permit adequate core cooling to take place. This prevents the reactor fuel from overheating in the event that:

1. The vessel is isolated and maintained in the hot standby condition.

2. The vessel is isolated in conjunction with the loss of coolant flow from the reactor feedwater system.

3. A complete plant shutdown occurs under conditions of loss of normal feedwater system before the reactor is depressurized to a level where the shutdown cooling system can be placed into operation.

Following a reactor scram, steam generation continues at a reduced rate due to core fission product decay heat. At this time, the turbine bypass system diverts the steam to the main condenser, and the feedwater system supplies the makeup water required to maintain reactor vessel inventory.

In the event the reactor vessel is isolated and the feedwater supply unavailable, relief valves are provided to automatically (or remote manually) maintain vessel pressure within desirable limits. The water level in the reactor vessel drops due to continued steam generation by decay heat. Upon reaching a predetermined low level, the RCIC system is initiated automatically. The turbine-driven pump supplies demineralized makeup water from the condensate storage tank (CST) to the

reactor vessel. An alternate source of water is available from the suppression pool. The RCIC system allows automatic switchover of pump suction from the CST to the suppression pool if the RCIC pump suction pressure falls to a preset low level. Two pressure transmitters are used to detect low pressure at the RCIC pump suction. If either transmitter senses low pressure (indicating low CST level), pump suction is automatically transferred to the suppression pool. The turbine is driven with a portion of the decay heat steam from the reactor vessel and exhausts to the suppression pool. Suppression pool water is not maintained demineralized and is used only in the event all sources of demineralized water have been exhausted.

If the main feedwater system is not operable, a reactor scram is automatically initiated when reactor water level falls to Level 3. The Operator can then remotely manually initiate the RCIC system from the main control room, or the system is automatically initiated as follows. Reactor water level continues to decrease due to boiloff until Level 2 is reached. At this point, the HPCS and the RCIC systems are automatically initiated to supply makeup water to the RPV. These systems continue automatic injection until the reactor water level reaches Level 8, at which time the HPCS injection valve is closed and the RCIC steam supply valve is closed.

In the nonaccident case, the RCIC system is normally the only makeup system used to furnish subsequent makeup water to the RPV. The Operator remotely manually shuts down the HPCS system from the main control room. When level reaches Level 2 again due to loss of inventory through the main steam relief valves or to the main condenser, the RCIC system automatically restarts as described in Section II.K.3.13. This system then maintains the coolant makeup supply. RPV pressure is regulated by the automatic or remote manual operation of the main steam relief valves which discharge to the suppression pool.

To remove decay heat during a planned isolation event, assuming that the main condenser is not available, the steam-condensing mode of the RHR system can be manually initiated. Residual steam is routed through the RHR heat exchangers where it is condensed and cooled, then returned to the RPV through an interconnection with the RCIC pump. Thus, closed loop cooling is provided by this mode.

If the steam-condensing mode is unavailable for any reason, the SRVs can be used to dump the residual steam to the suppression pool. The suppression pool will then be cooled by remote manual alignment of the RHR system in the suppression pool cooling mode which routes the pool water through the RHR heat exchangers, cools it, and returns it to the suppression pool in a closed cycle. Makeup water to the RPV is still supplied by the RCIC system.

For the accident case with the RPV at high pressure, the HPCS system can also be used to automatically provide the required makeup flow. No manual operations are required. If the HPCS system is postulated to fail at these conditions and the RCIC capacity is insufficient, the ADS will automatically initiate depressurization of the RPV to permit the condensate pumps or the low-pressure ECCS (LPCI and LPCS) to provide makeup coolant.

Whenever the RCIC system is initiated, the large steam turbine generator (LSTG) turbine is tripped to prevent water induction into the turbine, and the control room is alarmed that the RCIC injection valve is open.

Therefore, although manual actions can be taken to mitigate the consequences of a loss of feedwater, there are no short-term manual actions which must be taken. Sufficient systems exist to automatically mitigate these consequences.

During RCIC operation, the suppression pool acts as the heat sink for steam generated by reactor decay heat. This results in a rise in pool water temperature. Heat exchangers in the RHR system are used to maintain the pool water temperature within acceptable limits by cooling the water directly or by condensing generated steam. The condensate discharge from the RHR heat exchangers may be used as RCIC pump suction supply or it may be directed to the suppression pool.

The RCIC system is equipped with a discharge line fill pump that operates to maintain the pump discharge line in a filled condition. Keeping the discharge line filled reduces the lag time between pump startup and attainment of full flow to the RPV. Additionally, its operation eliminates the possibility of RCIC pumps discharging into a dry pipe and minimizes water hammer effects. The fill pump is classified as Category I and Safety Class 2. The pump motor is Class 1E and is powered from a Class 1E source. Indication of pump operating status is provided in the main control room. Low discharge line pressure is also indicated in the main control room.

Pump discharge pipe routing and valve locations inside and outside containment ensure that a maximum amount of piping is maintained full of water.

In addition to the fill pump, the RCIC water discharge line is designed to accommodate water hammer loads due to postulated voids in the piping between the reactor and injection valve (MOV126). This section of piping is normally isolated from the fill pump circuit by the isolation valve. Also, appropriate drain and vent lines are provided in the discharge line.

In addition to the physical design features described above, the RCIC pump discharge piping has been analyzed and conservatively designed for the effects of possible water hammer forces using the methods described in Appendix 3A, Section 3A.24.

5.4.6.1.1 Residual Heat Removal and Isolation

Residual Heat Removal

The RCIC system initiates and discharges, within 30 sec, a 600-gpm constant flow into the reactor vessel over a 165 to 1,215 psia pressure range. The temperature of RCIC water discharged into the reactor vessel varies from 40° to 140°F when using water from the CST. The mixture of the cool RCIC water and the hot steam does the following:

1. Quenches the steam.

2. Removes reactor residual heat.

3. Replenishes reactor vessel inventory.

Redundantly, the HPCS system performs the same function, hence providing single-failure protection. Both systems use separate and independent electrical power sources of high reliability, which permit operation with either onsite or offsite power. Additionally, the RHR system performs a residual heat removal function.

RCIC system design includes interfaces with redundant leak detection devices, namely:

1. High pressure drop across a flow device in the steam supply line equivalent to 300 percent of the steady state steam flow (with a 3 to 5 sec delay for TMI modification) at the reactor high-pressure steam condition.

2. High area temperature, utilizing temperature switches as described in the LDS. High area temperature is alarmed in the main control room.

3. Low reactor pressure of 50 psig minimum.

4. High pressure between the RCIC turbine exhaust rupture diaphragms.

These redundant leak detection devices, activated by the redundant power supplies, automatically isolate the steam supply to the RCIC turbine. Other isolation bases are defined in the following section. The HPCS provides redundancy for the RCIC should the RCIC become isolated, hence providing single-failure protection.

Isolation

Isolation valve arrangements include the following:

1. Two RCIC lines penetrate the RCPB. The first is the RCIC steam line which branches off one of the MSLs between the reactor vessel and the MSIV. This line has two automatic motor-operated isolation valves, one located inside and the other outside the primary containment. An automatic motor-operated inboard RCIC isolation bypass valve is used to equalize the line pressure across the inboard isolation valves and warm up the downstream line. The isolation signals noted earlier close these isolation valves.

2. The second RCIC line that penetrates the RCPB is the RCIC pump discharge line, which has two testable check valves (one inside the primary containment and the other outside). Additionally, an automatic MOV is located outside primary containment.

3. The RCIC turbine exhaust line vacuum breaker system line has two automatic MOVs and two check valves. This line runs between the suppression pool air space and the turbine exhaust line downstream of the exhaust line check valve. Positive isolation is automatic via a combination of low reactor pressure and high drywell pressure. The vacuum breaker valve complex is placed outside the primary containment where there is a more desirable environment. In addition, the valves are readily accessible for maintenance and testing.

4. The RCIC pump suction line, minimum flow pump discharge line, and turbine exhaust line all penetrate the primary containment and are submerged in the suppression pool. The isolation valves for these lines are all outside the primary containment and require remote-manual operation, except for the minimum flow valves that actuate automatically.

5.4.6.1.2 Reliability, Operability, and Manual Operation

Reliability and Operability

The RCIC system (Table 3.2-1) is designed commensurate with the safety importance of the system and its equipment. Each component is individually tested to confirm compliance with system requirements. The system as a whole is tested during both the startup and preoperational phases of the plant to set a baseline for system reliability. To confirm that the system maintains this line, functional and operability testing is performed at predetermined intervals throughout the life of the reactor plant in accordance with Technical Specifications.

A design flow functional test of the RCIC system may be performed during normal plant operation by drawing suction from the CST and discharging through a full flow test return line to the CST. The discharge valve to the head cooling spray nozzle remains closed

during the test, and reactor operation remains undisturbed. All components of the RCIC system are capable of individual functional testing during normal plant operation. System control provides automatic return from test to operating mode if system initiation is required, with the following three exceptions:

1. Auto/manual initiation on the flow controller is required for Operator flexibility during system operation.

2. Closure of either or both of the steam inside/outside isolation valves requires Operator action to properly sequence their opening. An alarm sounds when either of these valves leaves the fully-open position.

3. Other bypassed or otherwise deliberately rendered inoperable parts of the system are automatically indicated in the main control room at the system level.

Manual Operation

In addition to the automatic operational features, provisions are included for remote-manual startup, operation, and shutdown of the RCIC system, provided initiation or shutdown signals do not exist.

After the RHR system is placed in the steam-condensing mode, the Operator selects the condensate discharge from the RHR steam-condensing heat exchangers as the RCIC pump suction supply. The steam-condensing mode of the RHR system is manually placed in operation. Once steam condensing has been established, water level in the RHR heat exchangers is automatically maintained by means of a regulating valve in the condensate discharge line. Initially, the condensate discharge is directed to the suppression pool. After proper water quality is obtained, the condensate discharge may be directed to the RCIC pump suction. The level control for the RHR heat exchangers is independent from the RCIC control system. The Operator selects the flow setpoint of the RCIC system to match the condensate flow rate from the RHR heat exchangers.

5.4.6.1.3 Loss of Offsite Power

The RCIC system power is derived from an emergency auxiliary power distribution system that is normally energized from offsite power sources. Upon loss of offsite power (LOOP), this is automatically energized from standby onsite power sources (diesel generator or battery). All components necessary for initiation of the RCIC system are capable of startup independent of auxiliary ac power, plant service air, and external cooling water systems.

5.4.6.1.4 Physical Damage

The system is designed to the requirements of Table 3.2-1 commensurate with the safety importance of the system and its equipment. The RCIC turbine and pump are located in a different quadrant of the reactor building and utilize different divisional power (and separate electrical routings) than that of its redundant system HPCS (Sections 5.4.6.1.1 and 5.4.6.2.4).

5.4.6.1.5 Environment

The system operates for the time intervals and the environmental conditions specified in Section 3.11.

The RCIC system takes suction from the CSTs during normal modes of operation. The CSTs are located within the condensate storage building which is maintained at a minimum temperature of 65°F, as described in Section 9.4.7.2.5.

All interconnecting piping is located within piping tunnels which are beneath heated structures and below the frost line. To provide a Category I source of cooling water for the RCIC system, automatic transfer circuitry has been provided to transfer suction from the CSTs to the suppression pool, which is inside the reactor building and protected from cold weather. All other RCIC piping is located within the reactor building and is protected from cold weather.

5.4.6.2 System Design

5.4.6.2.1 General

Description

A summary description of the RCIC system is presented in Section 5.4.6.1, which defines in general the system functions and components. The detailed description of the system, its components, and operation is presented in the following sections.

Diagrams

The following diagrams are included for the RCIC systems:

1. A schematic P&ID (Figure 5.4-9) shows all components, piping, points where interface system and subsystems tie together, and instrumentation and controls associated with subsystem and component actuation.

2. A schematic process diagram (Figure 5.4-10) shows temperature, pressures, and flows for RCIC operation and system process data hydraulic requirements.

3. Performance curves showing temperature, pressure, steam flow, brakehorsepower, and shaft speed for the RCIC

turbine manufactured by the Terry Corporation, are shown on Figures 5.4-10a and 5.4-10b.

Interlocks

The following defines the various electrical interlocks:

1. There are three keylocked switches controlling valves F063, F064, F068 (2ICS-MOV128, -MOV121, -MOV122), and two keylocked reset switches that reset the isolation signal seal-in feature.

2. The F031 (21CS-MOV136) limit switch activates when fully open and closes F010-MOV129, F022-MOV108, and F059-MOV124.

3. The F068 (21CS-MOV122) limit switch activates when fully open and clears F045 (-MOV120) permissive so F045 (-MOV120) can open.

4. The F045 (1CS-MOV120) limit switch activates when F045 (-MOV120) is not fully closed and energizes a 15-sec time delay for alarms for low pump discharge flow, low turbine bearing oil pressure, and low gland seal air pressure, closes F019 (MOV143), and initiates startup ramp function. This ramp resets each time F045 (-MOV120) is closed.

5. The F045 (1CS-MOV120) limit switch activates when fully closed; this permits F004 (AOV109), F005 (AOV110), F025 (AOV131), and F026 (AOV130) to open and closes F013 (MOV126).

6. The turbine trip throttle valve limit switch activates when fully closed and closes F013 (MOV126) and F019 (MOV143).

7. The steam line isolation valves F063 (MOV120) and F064 (MOV121) are closed by RCIC Division I and II isolation signals. These divisions are different from the ECCS divisions.

8. High turbine exhaust pressure, low pump suction pressure, or an isolation signal actuate and close the turbine trip throttle valve. When the signal is cleared, the trip throttle valve must be reset from the main control room.

9. Overspeed of 125 percent trips the mechanical trip at the turbine which closes the trip throttle valve. The mechanical trip is reset at the turbine.

10. An isolation signal closes F063 (MOV120), F064 (MOV121), F076 (MOV170), and other valves as noted in Items 6 and 8.

11. An initiation signal opens F095 (MOV159), F010 (MOV129), if closed, and F045 (MOV120), and F013 (MOV126) and closes F059 (MOV124) if open. The initiation signal causes F022 (MOV108) to receive a close signal; however, this is not sealed in. Valve F059 (MOV124) closes and seals in. An initiation signal also trips the LSTG turbine.

12. High and low inlet RCIC steam line drain pot levels, respectively, and open and close F054 (LV132).

13. The combined signal of low flow plus high pump discharge pressure opens and with increased flow closes F019 (MOV143) (Items 5 and 6).

14. The F013 (MOV126) limit switch activates when F013 (MOV126) is not fully closed and energizes a control room relay to alarm the control room that valve F013 (MOV126) is not fully closed.

15. A LOCA signal will electrically isolate the motor control center (MCC) for the turbine trip and throttling valve motor from its safety-related bus.

5.4.6.2.2 Equipment and Component Description

Design Conditions

Operating parameters for the components of the RCIC system, defined as follows, are shown on Figure 5.4-10. The RCIC components are:

1. One 100-percent capacity turbine and accessories.

2. One 100-percent capacity pump assembly and accessories.

3. Piping, valves, and instrumentation for:

   a. Steam supply to the turbine.

   b. Steam supply to RHR steam-condensing heat exchanger.

   c. Turbine exhaust to the suppression pool.

   d. Supply from the CST to the pump suction.

   e. Supply from the suppression pool to the pump suction.

f. Makeup supply from the RHR steam-condensing heat exchangers.

g. Pump discharge to the head cooling spray nozzle, including a test line to the CST, a minimum flow bypass line to the suppression pool, and a coolant water supply to the turbine lube oil cooler.

4. System pressure pump to maintain injection lines full of water to the outside containment isolation valves.

The basis for the design conditions is ASME Section III.

Design Parameters

Design parameters for the RCIC system components are listed as follows (see Figure 5.4-9 for a cross-reference of component numbers):

1. RCIC Pump Operation (C001)

Flow rate	Injection flow - 600 gpm Cooling water flow - 25 gpm Total pump discharge - 625 gpm (includes no margin for pump wear)
Water temperature range	40° to 140°F
NPSH	21 ft minimum
Developed head	3,080 ft @ 1,215 psia reactor pressure 610 ft @ 165 psia reactor pressure
BHP, not to exceed	720 hp @ 3,080 ft developed head 130 hp @ 610 ft developed head
Design pressure	1,525 psig
Design ambient temperature	60° to 122°F

2. RCIC Turbine Operation (C002)

	H.P. Condition (psia)	L.P. Condition (psia)
Reactor pressure (saturated temperature)	1,215	165
Steam inlet pressure	1,190, min	150, min

	H.P. Condition (psia)	L.P. Condition (psia)
Turbine exhaust pressure	65, max	65, max
Turbine design inlet pressure	1,250 psig at saturated temperature	
Turbine exhaust casing design pressure	165 psig at saturated temperature	

3. RCIC Orifice Sizing

Coolant loop orifice (D012)	Sized to maintain 16 to 25 gpm to the lube oil cooler based upon pump suction line pressure varying from 50 psig to minimum NPSH value. (The estimated diameter is 0.343 in.)
Minimum flow orifice (D005)	Sized with piping arrangement to ensure minimum flow of 75 gpm with MO-F019 (MOV143) fully open.
Test return orifice (D006)	Sized with piping arrangement to simulate pump discharge pressure required when the RCIC system is injecting design flow with the reactor vessel pressure at 165 psia.
Leakoff orifices (D008, D010)	Sized for 1/8-in diameter minimum, 3/16-in diameter maximum.

4. Valve Design Requirements

The following are the design differential pressure requirements. However, the actuator sizing/setting is based on maximum operating differential pressure.

Steam supply valve (F045) (MOV120)	Open and/or close against full differential pressure of 1,200 psi within 15 sec.
Pump discharge valve (F013) (MOV126)	Open and/or close against full differential pressure of 1,450 psi within 15 sec.
Pump minimum flow bypass valve (F019) (MOV143)	Open and/or close against full differential pressure of 1,450 psi within 5 sec.
RHR steam supply isolation valves (F063 & F064) (MOV128 and MOV121)	Open and/or close against full differential pressure of 1,200 psi within 30 sec.

Cooling water pressure control valve (F015) (PCV115)	Self-contained downstream sensing control valve capable of maintaining constant downstream pressure of 125 psia.
Pump suction relief valve (F017) (RV114)	150 psig relief setting; 10 gpm at 10% accumulation.
Cooling water relief valve (F018) (RV112)	Sized to prevent overpressurizing piping, valves, and equipment in the coolant loop in event of failure of pressure control valve F015.
Pump test return valve (F022) (FV108)	Capable of throttling against 1,000 psi differential pressure and closure against differential pressures of 1,450 psi.
Pump suction valve, suppression pool (F031) (MOV136)	Located outside as close as practical to the primary containment.
Testable check valves (F065-F066) (AOV156+AOV157)	System test mode bypasses this valve, and its functional capability is demonstrated separately. Therefore, valve test provisions are provided, including limit switches to indicate disc movement. The valve and valve-associated equipment are capable of proper functional operation during maximum ambient conditions.
Warmup line isolation valve (F076) (MOV170)	Opens and/or closes against differential pressure of 1,200 psi with minimum travel of 4 in/min.
Vacuum breaker valves (F080, F086) (MOV148+MOV164)	Opens and/or closes against differential pressure of 160 psi at a minimum rate of 4 in/min.
5. <u>Rupture Disc Assemblies (D001 & D002)</u>	Utilized for turbine casing protection; includes a mated vacuum support to prevent rupture disc reversing under vacuum conditions.
Rupture pressure flow capacity	150 ±10 psig. 60,000 lb/hr at 165 psig.
6. <u>Instrumentation</u>	For instruments and control definition refer to Chapter 7.

7. Condensate Storage Requirements — Total required reserve storage for RCIC and HPCS systems is 135,000 gal.

8. Piping RCIC Water Temperature — The maximum water temperature range for continuous system operation will not exceed 140°F. However, due to potential short-term operation at higher temperatures, piping expansion calculations were based on 170°F.

9. Turbine Exhaust Vertical Reaction Force — The turbine exhaust sparger is capable of withstanding a vertical pressure unbalance of 20 psi. Pressure unbalance is due to turbine steam discharge below the suppression pool water level.

10. Ambient Conditions

	Temperature (°F)	Relative Humidity (%)
Normal plant operation	60-100	95
Isolation conditions (Isolation of the primary system requiring RCIC operation)	150	100

5.4.6.2.3 Applicable Codes and Classifications

The RCIC system components within the drywell up to and including the outer isolation valve are designed in accordance with ASME Section III, Safety Class 1. The RCIC system component classifications and those for the condensate storage system are given in Table 3.2-1.

5.4.6.2.4 System Reliability Considerations

To assure that the RCIC operates when necessary and in time to prevent inadequate core cooling (ICC), the power supply for the system is taken from immediately available energy sources of high reliability. Added assurance is given in the capability for periodic testing during Station operation. Evaluation of reliability of the instrumentation for the RCIC shows that no failure of a single initiating sensor either prevents or falsely starts the system. In order to assure HPCS or RCIC availability for the operational events noted previously, certain design considerations are utilized in design of both systems.

The most limiting operating condition for the RCIC pump occurs when the pump takes suction from the suppression pool and

discharges at its rated flow of 625 gpm. This represents the limiting operating condition because of the minimum static suction head (17.4 ft) and the maximum temperature/vapor pressure (170°F/6.0 psia) of the water that might exist during RCIC system operation. The NPSH margin during this condition is 9.0 ft (NPSH available = 30.0; NPSH required = 21 ft). The RCIC system meets the requirements of RG 1.1, since the calculation of NPSH available takes no credit for increased containment atmospheric pressure accompanied by a LOCA, and is computed using the maximum anticipated water temperature of 170°F.

Physical Independence

The HPCS and RCIC systems are located in separate areas of the secondary containment. Piping runs are separated, and the water delivered from each system enters the reactor vessel via different nozzles.

Prime Mover Diversity and Independence

Prime mover independence is achieved by using a steam turbine to drive the RCIC pump and an electric motor-driven pump for the HPCS system. The HPCS motor is supplied from emergency ac power or a separate diesel generator.

Control Independence

Control independence of HPCS and RCIC is provided by using different battery systems to provide control power to each unit. Separate detection initiation logics are also used for each system.

Portions of HPCS and RCIC within the RCPB are designed to meet Safety Class 1 requirements. Environment in the equipment rooms is maintained by separate auxiliary systems.

Periodic Testing

A design flow functional test of the RCIC can be performed during plant operation (Section 5.4.6.1.2). Periodic inspections and maintenance of the turbine-pump unit are conducted in accordance with manufacturer's instructions. Valve position indication and instrumentation alarms are displayed in the main control room.

5.4.6.2.5 System Operation

Automatic startup of the RCIC system due to an initiation signal from reactor low water level requires no Operator action. The test operation mode and hot standby steam-condensing mode are both manually initiated by the Operator. The Operator actions associated with these modes are defined in the operating procedures.

The most limiting single failure with the RCIC system and its HPCS backup system is the failure of the HPCS. With a HPCS failure, if the capacity of the RCIC system is adequate to maintain reactor water level, the Operator follows specific procedures to facilitate the automatic operation. If, however, the RCIC capacity is inadequate, the same procedures still apply, but the Operator may also initiate the ADS (Section 6.3.2).

Operation of the RCIC system following a Station blackout is addressed in Section 8.3.1.5.

5.4.6.3 Performance Evaluation

The analytical methods and assumptions in evaluating the RCIC system are presented in Chapter 15 and Appendix 15A. The RCIC system provides the flows required from the analysis (Figure 5.4-10) within a 30-sec interval based upon considerations noted in Section 5.4.6.2.4.

5.4.6.4 Preoperational Testing

The preoperational and initial startup test program for the RCIC system is presented in Chapter 14.

5.4.7 Residual Heat Removal System

5.4.7.1 Design Bases

The RHR system is composed of three independent loops, each containing a motor-driven pump, piping, valves, instrumentation, and controls. Each loop has a suction source from the suppression pool and is capable of discharging water to either the reactor vessel via a separate nozzle, or back to the suppression pool via a full-flow test line. The A and B loops have heat exchangers that are cooled by service water. Loops A and B can also take suction from the reactor recirculation system suction and can discharge into the reactor recirculation discharge or to the suppression pool and drywell spray spargers. The A and B loops also have connections to reactor steam via the RCIC steam line and can discharge the resultant condensate to the RCIC pump suction or to the suppression pool. In addition, Loops A and B take suction from the fuel pool and discharge to the fuel pool cooling discharge.

5.4.7.1.1 Functional Design Basis

The RHR system has five subsystems, each of which has its own functional requirements. Each subsystem is discussed separately as follows.

<u>Residual Heat Removal Mode (Shutdown Cooling Mode)</u>

The functional design basis of the shutdown cooling mode is to have the capability to remove decay and sensible heat from the

reactor primary system so that the reactor outlet temperature is reduced to 125°F, in approximately 20 hr after the control rods have been inserted, to permit refueling when the service water temperature is 82°F, the core is "mature," and the RHR heat exchanger tubes are assumed to be completely fouled (see Section 5.4.7.2.2 for exchanger design details). The capacity of the heat exchangers is such that the time to reduce the vessel outlet water temperature to 212°F corresponds to a maximum cooldown rate of 100°F/hr with both loops in service. However, the flushing operation associated with normal initiation of the shutdown cooling mode prevents attaining 212°F coolant temperature at the minimum time.

If flushing is performed in 2 hr, the minimum time required to reduce vessel coolant temperature to 212°F is depicted on Figure 5.4-11.

The design basis for the most limiting single failure for the RHR system (shutdown cooling mode) is that the shutdown line can be made usable by manual action (Section 15.2.9) and the plant is then shut down using the capacity of a single RHR heat exchanger and related service water capability. Figure 5.4-12 shows the time required to reduce vessel coolant temperature to 212°F using one RHR heat exchanger and allowing 2 hr for flushing.

In the event that the RHR shutdown cooling suction line is not available because of single failure, the alternate shutdown cooling method may be used to accomplish the shutdown cooling function as discussed in Section 15.2.9. This alternate shutdown cooling path uses the RHR and ADS/SRV systems. The RHR pump flow is directed to the RPV from the suppression pool through the RHR heat exchanger via the LPCI lines. A sufficient number of SRVs are powered open to establish a liquid flow path back to the suppression pool.

Alternatively, when the Operator is using EOPs to control RPV parameters and shutdown cooling is not available, the Operator is permitted to continue cooldown using the systems previously used for depressurization. These systems include SRVs, MSL drains, RWCU, RCIC and RHR steam condensing.

Further operational description of the alternate shutdown cooling method is discussed in Section 15.2.9. The adequacy of the SRVs for liquid flow in this mode of operation is discussed in Section 1.12.

The RHR pumps have sufficient head to satisfy the requirement of the alternate shutdown cooling mode of operation. The following calculation demonstrates the adequacy of the pump flow/head requirement.

The following design features/criteria were applied to the calculation.

1. Suppression pool level is assumed at minimum drawdown level of 197 ft 8 in.

2. Frictional flow losses are based on a 40°F suppression pool water temperature.

The total developed head (TDH) required to provide adequate cooling water flow rate in this mode of operation is given by the following formula:

$$TDH = Z_2 - Z_1 + \frac{P_2 - P_1}{\rho} + \frac{V_2^2}{2g} - \frac{V_1^2}{2g} + h_L$$

Where:

Z_2	=	Elevation of the RPV outlet nozzle (el 322′-0")
Z_1	=	Elevation of the suppression pool water level (el 197′-8") corresponding to the minimum pool drawdown level
$\frac{P_2-P_1}{\rho}$	=	Difference in pressure between the RPV and the suppression chamber
$\frac{V^2_2}{2g}$	=	Velocity head at the water surface within the RPV, assumed to be negligible
$\frac{V^2_1}{2g}$	=	Velocity head at the water surface, in the suppression pool assumed to be negligible
h_L	=	Frictional head loss from all components in the flow path, including the RHS suction strainers (50-percent clogged); RHS heat exchangers, RPV, and all interconnecting piping, valves, etc., assuming a flow of 7060 gpm

Evaluation of the terms in the foregoing equation at this flow rate is shown as follows:

$Z_2 - Z_1$ = 322 ft - 197 ft-8 in = 124.3 ft

$\frac{P_2-P_1}{\rho} = \frac{35 \text{ psig} - 0 \text{ psig}}{\rho}$ = 80.8 ft

h_L = 3.9 ft (suction strainer, RHR pump suction piping and valves)

+ 157.8 ft (RHR heat exchanger, RHR pump discharge piping and valves)

+ 35.9 ft (RPV)

Total h_L = 197.6 ft

Calculating TDH required using preceding values:

TDH = 124.3 ft + 80.8 ft + 197.6 ft = 402.7 ft

Based upon the RHR pump characteristics shown on Figure 6.3-5B, the RHR pump will deliver a flow rate of approximately 7060 gpm at 402.7 ft. This equates to the 982 lb/sec assumed in the alternate shutdown cooling analysis per Table 15.2-13.

Sufficient head exists to "push" the water back down to the suppression pool via the SRVs. This is demonstrated by the following calculation:

$$\left(\frac{P_2 - P_3}{\rho}\right) + \left(\frac{V_2^2 - V_3^2}{2g}\right) + Z_2 - Z_3 \geq h_L$$

Where:

$\frac{P_2-P_3}{\rho}$	=	Difference in pressure between the RPV and the suppression chamber, assumed to be zero for this part of the calculation (i.e., no credit is taken for any pressure head within the RPV)
Z_2-Z_3	=	124.3 ft, as before
$\frac{V_2^2-V_3^2}{2g}$	=	0, as before
h_L	=	Frictional head loss from all piping and components in the flow path, including the SRV and the T-Quencher = 47.8 ft

The head loss across the SRVs is based on data provided in the Analysis of Generic BWR Safety/Relief Valve Operability Test Results, GE-NEG, NEDE-24988-P, dated October 1981. For this calculation, four SRVs are assumed open in the ADS mode of operation. The flow coefficient Cv for each valve was taken as 415, in accordance with NEDE-24988-P, Table 5.2-1.

Evaluating these terms demonstrates that the head requirement is met:

124.3 ft > 47.8 ft

Therefore, sufficient head exists to ensure the return of the water from the RPV to the suppression pool, even with no credit taken for any pressure head which may exist within the RPV.

Low-Pressure Coolant Injection Mode

The functional design basis for the LPCI mode is to pump a total of 7,450 gpm of water per loop, using the separate pump loops from the suppression pool, into the core region of the vessel when the vessel pressure is 20 psid over drywell pressure. Injection flow commences at 225 psid vessel pressure above drywell pressure.

The initiating signals are: vessel level 1.0 ft above the active core or drywell pressure greater than or equal to 2.0 psig. The pumps attain rated speed in 27 sec and injection valves fully open in 40 sec.

Suppression Pool Cooling Mode

The functional design basis for the suppression pool cooling mode (SPCM) is that it will have the capacity to ensure that the suppression pool temperature immediately after a blowdown does not exceed 170°F.

Containment Spray Cooling Mode

The functional design basis for the containment spray cooling mode is that there should be two redundant means to spray into the drywell and suppression pool vapor space to reduce internal pressure to below design limits.

Reactor Steam-Condensing Mode

The functional design basis for the reactor steam-condensing mode is that the heat exchanger in one loop of the RHR system, in conjunction with the RCIC turbine, is able to condense all the steam generated 1 1/2 hr after a reactor scram.

5.4.7.1.2 Design Basis for Isolation of RHR System from Reactor Coolant System

The low-pressure portions of the RHR system are isolated from full reactor pressure whenever the primary system pressure is above the RHR system design pressure (Section 5.4.7.1.3). In addition, automatic isolation may occur as a result of low reactor water level which is unrelated to line pressure rating (see Section 5.2.5 for an explanation of the LDS and the isolation signals).

The RHR pumps are protected against damage from a closed discharge valve by means of automatic minimum flow valves, which open on low main line flow and close on high main line flow.

Possible Operator errors during plant startup and cooldown when the RHR system is not isolated from the RCS have been minimized through the implementation of the following design bases:

1. The low-pressure suction piping is protected from inadvertent opening of valves F008 (2RHS*MOV113) and F009 (2RHS*MOV112) by pressure interlocks on these valves.

2. The probability of draining some reactor water to the suppression pool is reduced by the existence of an interlock on valve F006 (2RHS*MOV2) which prevents its opening unless valves F004 (2RHS*MOV1), F024 (2RHS*FV38), and F027 (2RHS*MOV33) are closed.

3. The pump cannot be started unless a suction path is open.

5.4.7.1.3 Design Basis for Pressure Relief Capacity

The relief valves in the RHR system are sized on one of three bases:

1. Thermal relief only.

2. Valve bypass leakage only.

3. Control valve failure and the subsequent uncontrolled flow which results.

Transients are treated by bases 1 and 3; basis 2 results from an excessive leak past isolation valves. E12-F055 (2RHS*SV34) and E12-F230 (2RHS*SV62) are sized to maintain upstream piping at 500 psig and 10-percent accumulation with E12-F051 (2RHS*PV21) fully open and a reactor pressure equal to the lowest nuclear boiler SRV spring setpoint. E12-F036 (2RHS*RV108) is sized to limit the pressure downstream of E12-F026A and B (2RHS*MOV32A and B) to 150 psig and 10-percent accumulation with both PCV E12-F065A (2RHS*LV17A) and PCV E12-F065B (2RHS*LV17B) failed open. E12-F005 (2RHS*RV110), F025 (2RHS*RV20), F030 (2RHS*RV139), F088 (2RHS*RV61), F231 (2RHS*RV152), 2RHS*RV56, 2RHS*RV42, and F236 (2RHS*RV117) are set at the design pressure specified in the process data drawing, with allowance for piping elevation.

Redundant interlocks prevent opening valves to the low-pressure suction piping when the reactor pressure is above the shutdown range. These same interlocks initiate valve closure on increasing reactor pressure. A pressure interlock prevents connecting the discharge piping to the primary system whenever the pressure difference across the discharge valve is greater than the design differential. In addition, a high-pressure check valve closes to prevent reverse flow if the reactor pressure should increase. Relief valves in the discharge piping are sized to account for leakage past the check valves and motor-operated injection valves.

The relief and safety valve capacities and settings are shown in Table 5.4-2. All relief and safety valves are purchased,

maintained, and installed to Safety Class 1, 2, or 3 requirements to match the requirements of the piping and equipment to which they are installed. All valves discharge to the suppression pool with the exception of E12-F231 (2RHS*RV152), located on the shutdown cooling suction line; 2RHS*RV42A and B, located on the RHR heat exchanger tubeside; and E12-F236 (2RHS*RV117), located on the RHR pressure pump discharge piping. These valves discharge to the reactor building equipment drain system.

With the exception of the steam SRVs (E12-F055 and E12-F230) discharge piping, all pressure relief valve discharge lines in the RHR system are designed to accommodate the dynamic loading resulting from relief valve actuation. For a discussion of the analyses used to verify the design adequacy of these discharge lines, see Section 3.9.1.5.2A.

5.4.7.1.4 Design Basis for Reliability and Operability

The design basis for the shutdown cooling mode of the RHR system is that this mode is controlled by the Operator from the control room. The only operations performed outside the control room for a normal shutdown are certain manual valve lineups and system flushings.

Two separate shutdown cooling loops are provided. Although both loops are normally used for shutdown, the reactor coolant can be brought to 212°F in less than 20 hr after control rod insertion with only one loop in operation. A single RHR suction line can supply either or both shutdown cooling loops. With the exception of the shutdown cooling suction, shutdown cooling return, and steam supply and condensate discharge lines, the entire RHR system is part of the ECCS and containment spray and suppression pool cooling system. It is designed with the redundancy, flooding protection, piping protection, power separation, and other features required of such systems (see Section 6.3 for an explanation of the design bases for the ECCS). Shutdown cooling suction and discharge valves are provided with both offsite and standby emergency power supplies for purposes of isolation and shutdown. In the event either of the two shutdown cooling supply valves fails to operate, an Operator is sent to open the valve manually. If this is not feasible, the shutdown line is isolated using manual valve E12-F020 and repairs are made to the shutdown cooling valves so that they can be opened to supply shutdown cooling suction to the RHR pumps. While repairs are in process, residual heat is absorbed by the main condenser or by the suppression pool, which is cooled by the RHR system. In the event that the RHR shutdown cooling suction line is not available because of single failure, alternate shutdown cooling methods may be used to accomplish the shutdown cooling function (see Section 5.4.7.1.1). Thus, no single failure in either system design or power source will result in the loss of shutdown cooling capability, because the plant may use the normal shutdown cooling through the recirculation loops or the alternate shutdown cooling using the ADS SRVs and suppression pool cooling.

The RHR system takes suction from either the recirculation piping or the suppression pool. All piping is within the reactor building and is protected from cold weather.

The RHR heat exchangers dissipate their heat to the service water system. All service water piping and components supplying the RHR heat exchangers are either within heated structures or underground piping tunnels located below the frost line. Design provisions which protect water in the service water intake and discharge system from freezing are described in Section 9.2.5.

5.4.7.1.5 Design Basis for Protection from Physical Damage

Pumps A, B, and C as well as heat exchangers A and B are physically separated. Each is housed in a separate cubicle. Pump A and heat exchanger A are in separate cubicles in the reactor building auxiliary bay north. Pumps B and C and heat exchanger B are in separate cubicles in the reactor building auxiliary bay south. The RHR system pressure pump P2, which maintains the loop B and C discharge header full of water and pressurized, is located in the same cubicle as pump C. The same function is provided by LPCS system pressure pump P2 for RHR loop A, located in a separate cubicle with LPCS pump P1.

The design basis for protection from physical damage, such as flooding, internally-generated missiles, pipe break, and seismic effects, is discussed in Sections 3.4, 3.5, 3.6B, and 3.7B, respectively.

5.4.7.2 System Design

5.4.7.2.1 System Diagrams

All components of the RHR system are shown on Figure 5.4-13. A description of the controls and instrumentation is presented in Sections 7.3.1.1.4 and 7.6.1.2.

Figure 5.4-14 is the RHR process diagram and data. All sizing modes of the system are shown in the process data. The functional control diagram (FCD) for the RHR system is provided on Figure 7.3-10.

Interlocks are provided: (1) to prevent draining vessel water to the suppression pool, (2) to prevent opening the RHR shutdown cooling suction valve if vessel pressure is above the RHR suction line design pressure, or the discharge line design pressure with the pump at shutoff head, (3) to prevent inadvertent opening of drywell spray valves while in shutdown, (4) to prevent opening low-pressure steam supply valve F087 when vessel pressure is above line design rating, and (5) to prevent pump start when suction valve(s) are not open.

The RHR system is connected to higher-pressure piping at shutdown suction, shutdown return, LPCI injection, head spray, and heat

exchanger steam supply lines. The vulnerability to overpressurization of each location is discussed in the following paragraphs.

Shutdown suction has two gate valves in series, F008 and F009, that have independent pressure interlocks to prevent opening at high inboard pressure for each valve. The pressure interlock setpoint for RHR shutdown suction valves F008 and F009 is 128 psig. Shutdown suction valves F006A and B are not interlocked because valves F008 and F009 provide the required overpressurization protection. No single-active failure nor Operator error will result in overpressurization of the low-pressure piping.

In the event of leakage past F008 and F009, pressure transmitter N057 provides indication and alarm to the Control Room Operator.

The shutdown return line has a swing check valve, F050, to protect it from overpressurization. Additionally, a globe valve, F053, is located in series and has pressure interlock to prevent opening at high inboard pressures. No single-active failure nor Operator error with the shutdown return line will cause overpressurization of the lower-pressure piping.

LPCI injection valves F042A, B, and C, along with LPCI injection line check valves F041A, B, and C, provide overpressurization protection for the low-pressure RHR discharge line piping. The pressure interlock setpoint of LPCI injection valves F042A, B, and C is 130 psid. The differential pressure transmitters N658A, B, and C connected to LPCI injection valves F042A, B, and C are not an overpressurization protection device. In situations where the injection valves F042A, B, and C are open, i.e., surveillance/operability tests, check valves F041A, B, and C, and relief valves F025A, B, and C prevent overpressurization of the low-pressure piping. The routine verification of low leak rates of check valves F041A, B, and C ensures that valves F041A, B, and C and F025A, B, and C can provide adequate overpressurization protection when LPCI injection valves F042A, B, and C are open. This feature improves the reliability of the LPCI system by reducing the differential pressure which permits valve actuation. This decreases the size of the actuators required for the F042A, B, and C injection valves. The smaller valve actuators have correspondingly smaller actuator/valve-stem/disc loads and hence increased reliability. Valve installations are modified to prevent bonnet pressure locking.

The head spray line has a swing check valve, F019, to protect it from overpressurization. Additionally, a gate valve, F023, is located in series and has pressure interlocks to prevent opening at high inboard pressure. No single-active failure nor Operator error with the head spray line will cause overpressurization of the lower-pressure piping.

The heat exchanger steam supply line has a globe valve, F052, for shutoff. The Operator admits steam through F052 and sets the pressure regulating valve F051 to limit heat exchanger pressure. Also, F087 can be opened when the steam supply pressure is below the pressure interlock to provide additional steam flow to the heat exchangers. Safety valves F055 and F230 are provided downstream of F051 and F087 to protect the low-pressure piping should F051 fail open. No single-active failure nor single Operator error with the heat exchanger steam supply line will cause overpressurization of the low-pressure piping.

5.4.7.2.2 Equipment and Component Description

System Main Pumps

The RHR main system pumps are motor-driven deepwell pumps with mechanical seals and cyclone separators. The motors are air cooled by the reactor building ventilating system. The pumps are sized on the basis of the LPCI mode (Mode A) and the minimum flow bypass mode (Mode G) of process data on Figure 5.4-14. Design pressure for the pump suction structure is 220 psig with a temperature range from 40° to 360°F. Design pressure for the pump discharge structure is 500 psig. The bases for the design temperature and pressure are maximum shutdown cut-in pressures and temperature, minimum ambient temperature, and maximum shutoff head. The pump pressure vessel is carbon steel; the shaft and impellers are stainless steel. A comparison between the available and the required NPSH can be obtained from Section 6.3.2.2 and the pump characteristic curve provided on Figure 5.4-15. Available NPSH is calculated in accordance with RG 1.1 as shown in Section 6.3.2.2.

Pressure Pumps

RHR pressure pump P2 is provided to maintain RHR Loop B and C discharge header full of water and pressurized to avoid water hammer upon system initiation. This function is provided by the LPCS pressure pump P2 for RHR loop A. RHR pump P2 and CSL pump P2 are similar.

The RHR pump P2 is a horizontal-mounted, one-stage centrifugal pump. The pump is rated for 50 gpm with 175 ft TDH.

Heat Exchangers

The RHR heat exchangers are sized on the basis of the duty for shutdown cooling mode (Mode E). All other uses of these exchangers, including steam condensing, require less cooling surface. Flow rates are 7,450 gpm (rated) on the shellside and 7,400 gpm (rated) on the tubeside (service water side). Rated inlet temperature is 125°F shellside and 67°F tubeside. The overall heat transfer coefficient is 273 Btu/hr sq ft-°F. The exchangers contain 4,240 sq ft of effective surface. Design temperature of the shellside is 40° to 480°F. Design temperature

on the tubeside is 32° to 480°F. Design pressure is 500 psig on both sides; fouling factors are 0.0005 shellside and 0.001 tubeside. The construction materials are carbon steel for the pressure vessel with Type 304L stainless steel tubes and stainless steel clad tube sheet.

Valves

All directional valves in the system are conventional gate, globe, butterfly, stop check, and check valves designed for nuclear service. The injection valves, reactor coolant isolation valves, and pump minimum flow valves are high-speed valves, as required for LPCI or vessel isolation. Valve pressure ratings are as necessary to provide the control or isolation function; that is, all vessel isolation valves are rated Safety Class 1 nuclear valves rated at the same pressure as the primary system. All other containment isolation valves are Safety Class 2 nuclear valves rated at the same pressure as the piping in which they are installed. Steam pressure-reducing valves are designed to regulate steam flow into the heat exchangers from full reactor pressure upstream to maintain a maximum downstream pressure at 200 psig.

ECCS Portions of the RHR System

The ECCS portions of the RHR system include those sections described through Mode A-1 and A-2 of Figure 5.4-14. The flow path includes suppression pool suction strainers, suction piping, RHR pumps, discharge piping, injection valves, and drywell piping into the vessel nozzles and core region of the reactor vessel. Steam-condensing components include steam supply piping and valves, heat exchangers, and condensate piping. Pool cooling components include pool suction strainers, suction piping, pumps, heat exchangers and pool return lines. Containment spray components are the same as suppression pool cooling components except that the spray headers replace the suppression pool return lines.

5.4.7.2.3 Controls and Instrumentation

Controls and instrumentation for the RHR system are described in Sections 7.3.1.1.1 and 7.4.1.3.

5.4.7.2.4 Applicable Standards, Codes, and Classifications

Piping, Pumps, and Valves

Process side	Safety Class 1 and 2
Service water side	Safety Class 3

Heat Exchangers

Process side	Safety Class 2 TEMA Class C
Service water side	Safety Class 3 TEMA Class C

Electrical Portions

BOD	NSSS
IEEE-279-1971	IEEE-279-1971
IEEE-308-1974	IEEE-308-1971
IEEE-384-1974	IEEE-323-1971
IEEE-379-1977	IEEE-384-1974
IEEE-323-1974	

5.4.7.2.5 Reliability Considerations

The RHR system design includes the redundancy requirements of Section 5.4.7.1.5. Two completely redundant loops, each powered from a separate emergency bus, remove residual heat. All mechanical and electrical components, except for the common cooling shutdown suction line, are separate. Either loop is capable of shutting down the reactor within a reasonable length of time. The system design features, which assure that the systems connected to the RHR system do not degrade the reliability of the RHR system, are discussed in Section 6.3.2.5.

5.4.7.2.6 Manual Action

Residual Heat Removal (Shutdown Cooling Mode)

In shutdown cooling operation, when vessel pressure is 128 psig or less, the pool suction valve may be closed for the initial shutdown loop or loops. Flushing valves connecting RHR to condensate and makeup system are manually opened, and the stagnant water flushed to radwaste via MOVs E12-F040 and F049 which are operated from the control room. At the end of this flush, the upper portion of the chosen loop may be prewarmed by opening the testable check bypass valve in the shutdown return line permitting vessel water to warm the loop. Effluent is directed to radwaste and a temperature element is used to monitor effluent temperature. The testable check bypass valve is closed and vessel suction valves are opened to allow prewarming of the lower half of the shutdown loop with effluent directed to radwaste as before. The radwaste effluent valves are closed, the heat exchanger bypass valves opened (the exchanger valves were closed after the initial cold water flush), then the RHR pump is started at a regulated flow through return valve E12-F053. After an interval of several minutes to permit loop internal stability to be established, the service water pump is started, the service water valves are opened, the heat exchanger inlet and outlet

valves are opened, and cooldown of the vessel is in progress. Cooldown rate is subsequently controlled via valves E12-F053 (total flow) and E12-F048 (heat exchanger bypass flow).

The manual actions required for the most limiting failure are discussed in Section 15.2.9.

Steam Condensing

In order to initiate or terminate the steam-condensing mode, certain manual actions must be taken by the main control room Operator in conjunction with local Operators. In preparation for steam-condensing operations, the heat exchanger is initially isolated by closing the shellside (RHR) inlet and outlet valves. Prior to steam admission, the Operator will start additional service water (SWP) pumps, if necessary, and establish cooling water flow by opening the tubeside (SWP) inlet and outlet valves. Also prior to steam admission, the water level in the heat exchanger is lowered to a preset value, and the motor-operated shellside vent is opened to allow for the removal of noncondensable gases during steam condensing.

A 1-in bypass line with a MOV is provided around the steam supply block valve (E12-F052). This bypass MOV is opened first to allow for a slow, controlled pressurization and warmup of the piping between E12-F052 and the pressure control and bypass valves (E12-F051 and F087). Pressurization can be monitored by a pressure transmitter located in the section of piping with indication on a local control panel. After this line is warmed up, the steam supply block valve can be fully opened.

When the piping up to the pressure control valve is adequately warmed up and the steam supply block valve has been opened, the control valve is slowly opened to admit steam to the heat exchanger. This valve is normally controlled by an automatic controller which, during this initial warmup, will be manually set at a low pressure and gradually increased to the 200 psig operating pressure.

Drains with traps are provided at all local piping low points to automatically remove any condensate that is formed during the warmup of the steam supply lines. The steam supply piping is, therefore, designed to minimize the possibility of water slugs or high-pressure steam being introduced into the heat exchangers and to minimize the occurrence of water/steam hammer in the steam supply lines.

Once the steam-condensing mode is initiated, the automatic pressure regulator controls steam flow to maintain steam pressure within the heat exchanger. When the condensate quality attains the appropriate level, the Operator switches condensate from the pool to the RCIC pump suction.

The Operator regulates the opening of the shellside vent as required to prevent a buildup of noncondensables in the heat exchanger. The Operator can also regulate the water level within the heat exchanger to preclude the onset of flow-induced vibration when the reactor pressure falls below the 200 psig normal operating pressure of the steam-condensing mode.

When the steam-condensing mode is terminated, the condensate discharge control valve (E12-F065) will be closed. As the remaining steam is condensed, the level within the shellside of the heat exchanger will rise. When the rate of level increase slows down due to low steam flow, the steam supply will be isolated, the system valve lineup will be realigned to connect the heat exchanger to the main RHR pump discharge (i.e., E12-F047 and F003 will be reopened), and the system will be slowly refilled by the operation of the jockey pumps and throttling of valves.

Vent valves are located at all piping high points to ensure that no air pockets will exist when the system is returned to its normal standby configuration. Since the LPCI portion of the system is isolated from the steam-condensing portion during this mode by the closure of E12-F047 and F003, this portion will not be affected by RHR operation in the steam-condensing mode. Level transmitters are provided on the RHR piping at the LPCI injection valves to ensure that the piping is full of water up to this point. The provision of high-point vents and the ability to monitor the water level in the LPCI injection lines ensures that water hammer upon subsequent system operation is not a concern.

5.4.7.3 Performance Evaluation

Thermal performance of the RHR heat exchangers is based on the residual heat generated at 20 hr after rod insertion, a 125°F heat exchanger inlet temperature, and the flow of two loops in operation. Because shutdown is usually a controlled operation, maximum expected service water temperature less 10°F is used as the service water inlet temperature. These are nominal design conditions; if the service water temperature is higher, the exchanger capabilities are reduced and the shutdown time is longer and vice versa.

5.4.7.3.1 Shutdown With All Components Available

No typical curve of vessel cooldown temperatures versus time is shown here due to the infinite variety of such curves that may be due to: (1) clean steam systems that may allow the main condenser to be used as the heat sink when nuclear steam pressure is insufficient to maintain steam air ejector performance, (2) the condition of fouling of the exchangers, (3) Operator use of one or two cooling loops, (4) coolant water temperature, and (5) system flushing time. Since the exchangers are designed for the fouled condition with relatively high service water temperature, the units have excess capability to cool when first cut in at

high vessel temperature. Total flow and mix temperature must be controlled to avoid exceeding 100°F/hr cooldown rate (see Section 5.4.7.1.1 for minimum shutdown time to reach 212°F).

5.4.7.3.2 Shutdown with Most Limiting Failure

Shutdown under conditions of the most limiting failure is discussed in Section 15.2.9. The capability of the heat exchanger for any time period is balanced against reactor residual heat, pump heat, and sensible heat. The excess over residual heat and pump heat is used to reduce the sensible heat.

5.4.7.4 Preoperational Testing

The preoperational test program and startup test program (Chapter 14) are used to generate data to verify the operational capabilities of each piece of equipment in the system: instruments, setpoints, logic elements, pumps, heat exchangers, valves, and limit switches. In addition, these programs verify the capabilities of the system to provide the flows, pressures, condensing rates, cooldown rates, and reaction times required to perform all system functions as specified for the system or component in the system data sheets and process data. Logic elements are tested electrically; valves, pumps, controllers, and relief valves are tested mechanically. Finally, the system is tested for total system performance against the design requirements, as specified above, using both offsite power and standby emergency power. Preliminary heat exchanger performance can be evaluated by operating in the pool cooling mode, but a vessel shutdown is required for the final check due to the small temperature differences available with suppression pool cooling.

5.4.8 Reactor Water Cleanup System

The RWCU system is classified as a primary power generation system (not an engineered safety feature [ESF]), a small part of which is part of the RCPB up to and including the outside isolation valve. The other portions of the system are not part of the RCPB and can be isolated from the reactor. The RWCU system may be operated at any time during planned reactor operations or it may be shut down if water quality is within the Technical Specification limits. The seismic and safety class for this system are provided in Table 3.2-1.

The reactor chemistry limits outlined in RG 1.56 Revision 1, Table 1, will be met. These chemistry limits, as well as corrective action, will be established in the Technical Specifications. The following methods of chemical analyses will be used:

Parameter	Analysis Method*
Chloride	ASTM D512-81C or Ion Chromatographic*
pH	ASTM D1293-84B
Conductivity	ASTM D1125-82B

* The method used will provide adequate sensitivity to ensure that the limits described in RG 1.56 Rev. 1, Table 1, will be met. Alternative methods are acceptable provided adequate analytical sensitivity is ensured.

THIS PAGE INTENTIONALLY BLANK

5.4.8.1 Design Bases

5.4.8.1.1 Safety Design Basis

The RCPB portion and the high-pressure portion beyond the outside containment isolation valves are assessed to meet the requirements of RG 1.26 and 1.29 in order to:

1. Prevent excessive loss of reactor coolant.
2. Prevent the release of radioactive material from the reactor.
3. Isolate the major portion of the RWCU system from the RCPB.

5.4.8.1.2 Power Generation Design Basis

The RWCU system:

1. Removes solid and dissolved impurities from reactor coolant. The RWCU system is assessed to be in compliance with RG 1.56.
2. Discharges excess reactor water during startup, shutdown, and hot standby conditions to the main condenser or liquid radwaste system.
3. Minimizes temperature gradients in the reactor recirculation piping and RPV during periods when the reactor recirculation system pumps are unavailable.

The RWCU system is designed to:

1. Minimize system heat loss.
2. Enable the major portion of the system to be serviced during reactor operation.
3. Prevent the standby liquid reactivity control material from being removed from the reactor water by the cleanup system when required for shutdown.

5.4.8.2 System Description

The system takes its suction from the inlet of each reactor recirculation pump and from the RPV bottom head. The process fluid is circulated by the cleanup pumps through the regenerative and nonregenerative heat exchangers for cooling, through the filter demineralizers for cleanup, and back through the regenerative heat exchanger for reheating. The processed water is returned to the RPV, the main condenser, or liquid radwaste system (Figures 5.4-16a through 5.4-16f and 5.4-17).

The major components of the RWCU system are located outside the drywell. These components include pumps, regenerative and nonregenerative heat exchangers, filter demineralizers, and associated precoat equipment. Flow rate capacities for the major components are presented in Table 5.4-3.

Because the operating temperature of the filter demineralizer units is constrained by the resin operating temperature limit, the reactor coolant must be cooled before being processed in the filter demineralizer units. The regenerative heat exchanger transfers heat from the influent (tubeside) to the effluent (shellside). The effluent returns to the reactor. The nonregenerative heat exchanger cools the process influent further by transferring heat to the RBCLCW system.

The filter demineralizer units (Figures 5.4-16d through 5.4-16f and 5.4-19) are pressure precoat-type filters, using precoat filter aid material and finely ground, nonregenerable, mixed ion-exchange resins. Spent resins are not regenerable and are sluiced from the filter demineralizer unit to a phase separator tank in the radwaste system for processing and disposal (the resin backwash transfer system is described in Section 11.2.2.4). Resins are discarded based on filter demineralizer performance, as indicated by monitoring effluent conductivity, differential pressure across the unit, and sample analysis. Initial total capacity is not measured on resin that is finely ground and mixed since separation into anion and cation components is not practical. To prevent resins from entering the reactor coolant system in the event of failure of a filter demineralizer resin support, a strainer is installed in the effluent line of each filter demineralizer. Each strainer and filter demineralizer vessel has a control room alarm that is energized by high differential pressure. Upon further increase in differential pressure from the alarm point, the filter demineralizer automatically isolates.

The backwash and precoat cycle for each filter demineralizer is semiautomatic; however, permissives and interlocks are installed in the logic program to prevent human operational errors, such as inadvertent opening of valves that would initiate a backwash or contaminate reactor water with precoat material or resins. The filter demineralizer piping configuration is arranged to ensure that transfers are complete and crud traps are eliminated. A bypass line is provided around the filter demineralizer units.

The vent line from each filter demineralizer is routed to the phase separator tanks which are vented directly to the reactor building heating, ventilating, and air conditioning (HVAC) system (Section 9.4.2).

In the event of low flow or loss of flow in the system, flow is maintained through each filter demineralizer by its own holding pump. This ensures that the precoat and resin material are held in place on the septum screens. Sample points are provided in

the common influent header and in each effluent line of the filter demineralizer units for continuous indication and recording of system conductivity. High conductivity is annunciated in the main control room. The control room alarm setpoints of the conductivity meters at the inlet and outlet lines are 1.0 umho/cm and 0.1 umho/cm, respectively. The influent sample point is also used as the normal source of reactor coolant grab samples. Sample analysis also indicates the effectiveness of the filter demineralizer units.

The suction line (RCPB portion) of the RWCU system contains two motor-operated isolation valves, which automatically close in response to signals from the RPV (low water level), the LDS, and actuation of the SLCS. Activation of the SLCS closes the outside isolation valve from Division I logic and the inside isolation valve from Division II logic. Nonregenerative heat exchanger high outlet temperature closes the outside isolation valve only. Section 7.6 describes the LDS requirements which are summarized in Table 5.2-8. This isolation prevents the loss of reactor coolant and release of radioactive material from the reactor. The isolation valves close automatically to prevent removal of liquid boron reactivity control material from the reactor vessel in the event of SLCS activation. In addition, the outside isolation valve closes automatically to prevent damage of the filter demineralizer resins if the outlet temperature of the nonregenerative heat exchanger is high. The RCPB isolation valves may be remote manually operated to isolate the system equipment for maintenance or servicing. The requirements for the RCPB are specified in Section 5.2.

A remote manually-operated globe valve on the return line to the reactor provides long-term leakage control. Instantaneous reverse flow isolation is provided by check valves in the RWCU system piping.

The RWCU return line is split into two branches connecting to the two feedwater loops outside the primary containment. Each branch is equipped with a MOV. Under normal plant operation, both MOVs are open and RWCU water is returned to the RPV via two feedwater loops. During plant startup when the reactor power is below 20 percent, one of the MOVs may be closed directing all RWCU flow into a single feedwater loop. This will be done to minimize thermal stratification in the feedwater line during startup and shutdown and allow sharing the thermal cycles between the two feedwater loops. Both valves may be closed as necessary to support surveillance testing during plant shutdown.

Operation of the RWCU system is controlled from the main control room. Resin-changing operations, which include backwashing and precoating, are controlled from a local control panel.

A FCD is provided on Figure 7.3-7. Controls for valves 2WCS*MOV404A and B are shown on Figure 10.4-11.

5.4.8.3 System Evaluation

The RWCU system, in conjunction with the CND system, maintains reactor water quality during all reactor operating modes (normal, hot standby, startup, shutdown, and refueling). During refueling mode, the spent fuel pool cooling and cleanup system (SFC) also contributes to this function. This type of pressure precoat cleanup system has been used in all operating BWR plants since 1971. Operating plant experience has shown that the RWCU system, as designed, maintains the required BWR water quality. The nonregenerative heat exchanger is sized to maintain the required process temperature for filter demineralization when the cooling capacity of the regenerative heat exchanger is reduced due to diverting a portion of the return flow to the main condenser or radwaste. The control requirements of the RCPB isolation valves are designed to the requirements of Section 7.3.1. The component design data (flow rates, pressure, and temperature) are presented in Table 5.4-2. All components are designed to the requirements of Section 3.2, according to the requirements of Figures 5.4-16a through 5.4-16f and 5.4-17.

5.4.9 Main Steam Line and Feedwater Piping

This section describes the design of main steam piping (Sections 3.6.1A, 3.6.2A and 3.6.2B, and 10.3) and feedwater piping up to and including the piping through the jet impingement wall (Sections 3.6.1A, 3.6.2A, and 3.6.2B).

5.4.9.1 Safety Design Bases

In order to satisfy the safety design bases, the main steam and feedwater lines have been designed:

1. To withstand operational stresses, i.e., internal pressures, thermal expansion, SRV loads, fluid and thermal transients, and SSE loads, without a failure that could lead to the release of radioactivity in excess of the guideline values in published regulations.

2. With suitable access to permit ISI and IST.

3. To withstand without loss of safety function the fluid jets missiles, reaction forces, pressures, and temperatures postulated to result from pipe breaks (Sections 3.6.1A, 3.6.2A, and 3.6.2B).

5.4.9.2 Power Generation Design Bases

In support of reliable power generation, the following design bases have been employed for the main steam and feedwater piping:

1. The MSLs are designed to conduct steam from the reactor vessel throughout the full range of reactor power operation.

2. The feedwater lines are designed to conduct water to the reactor vessel throughout the full range of reactor power operation.

5.4.9.3 Description

The main steam piping is described in Sections 3.6.1A, 3.6.2A, 3.6.2B, and 10.3. The main steam and feedwater piping is shown on Figures 10.1-3 and 10.1-6. The feedwater piping consists of two 24-in outside diameter lines, each of which penetrates the drywell and branch into three 12-in lines that connect to the reactor vessel. Each line includes three containment isolation valves consisting of one check valve inside the drywell, and one motor-operated gate valve, and one spring-loaded piston-actuated check valve outside the drywell. The design pressure and temperature of the feedwater piping between the reactor and long-term isolation valve are 1,300 psig and 575°F. The design pressure and temperature of the feedwater piping between the reactor feed pumps and the sixth-point heaters are 2,200 psig and 375°F, and between the sixth-point heaters and the long-term isolation valves are 2,200 psig and 450°F. The Category I design requirements are invoked on the feedwater piping from the reactor through the outboard isolation valve and jet impingement wall. The materials used in the piping are in accordance with the applicable design code and supplementary requirements (Section 3.2). The general requirements of the feedwater system are described in Sections 7.7.1.3, 7.7.2, and 10.4.7.5.

5.4.9.4 Safety Evaluation

Differential pressure on reactor internals under the assumed accident condition of a ruptured steam line is limited by the use of flow restrictors and by the use of four MSLs. All main steam and feedwater lines of the RCPB are designed in accordance with the requirements defined in Sections 3.2, 3.6.1A, 3.6.2A, and 3.6.2B. Design of the piping in accordance with these requirements ensures that the safety design bases are met.

5.4.9.5 Inspection and Testing

Inspection and testing are carried out in accordance with Sections 3.9.1A, 3.9.1B, 5.2.4 and Chapter 14. ISI is considered in the design of the main steam and feedwater piping. This consideration assures adequate working space and access for the inspection of selected components.

5.4.10 Pressurizer

(Not applicable to BWR)

5.4.11 Pressurizer Relief Discharge System

(Not applicable to BWR)

5.4.12 Valves

5.4.12.1 Safety Design Bases

Line valves are located in the fluid systems to perform the mechanical function of controlling or prohibiting flow in one or more directions. They are components of the system pressure boundary and are designed to operate efficiently to maintain the integrity of that boundary. The valves operate under the internal pressure/temperature loading experienced during the various system transient operating conditions and various plant conditions such as seismic events or main steam system relief valve blowdown. The design, loading, and acceptability criteria are as required for Safety Class 1, 2, and 3 valves (Sections 3.9.3A and 3.9.3B) and ANSI B31.1 for Safety Class 4 valves. Compliances with ASME Codes are discussed in Section 5.2.1. The ADS valves comply with the requirements of NUREG-0737 for reactor coolant system venting, as described in Section 1.10.

5.4.12.2 Description

Line valves are standard manufactured types such as gate, globe, check, ball, and butterfly. They are designed and constructed in accordance with the requirements of Safety Class 1, 2, and 3 valves and ANSI B31.1 for Safety Class 4 valves. All materials, exclusive of seals, packing, and wearing components, are designed to remain functional for a 40-yr life under the environmental and design conditions applicable to the particular system and physical location in the plant when appropriate maintenance is periodically performed.

Valves requiring remote power operation utilize motor, electrohydraulic, hydraulic/mechanical, pneumatic or solenoid-actuated operators, or a combination thereof. The actuators have been sized by the manufacturer to operate successfully under the specified maximum operating conditions at the specified time rate stated in the design specification. Control and operation of these power-operated valves are discussed in the sections covering the systems in which particular valves are located.

5.4.12.3 Safety Evaluation

Line valves have been hydrostatically tested by the manufacturer for their performance as a pressure boundary. Pressure-retaining parts are subject to the testing and examination requirements of Safety Class 1, 2, and 3 valves and ANSI B31.1 for Safety Class 4 valves.

All electrical components of power-actuated valve operators utilized on Safety Class 1, 2, and 3 valves have been designed and qualified as described in Sections 3.10A, 3.10B, and 3.11.

To prevent motor overheating due to frequent cycling of ECCS/RCIC MOVs, operating procedures have been developed to caution plant Operators to be aware of the allowable duty cycles on these valves. The allowable duty cycle of ECCS QA Category I MOVs is five cycles open and closed per hour. The duty rating for RCIC MOVs is also five cycles per hour except for a two-cycle-per-hour rating for the turbine exhaust valve (E51-F068). These allowable cycles per hour envelope the required duty for ECCS and RCIC valves during normal, transient, and accident modes of operation. The frequent cycling of the ECCS/RCIC MOVs is ended after the first hour of LOCA or transient event.

To verify their operability, Class 1E MOVs are qualified in accordance with IEEE-382-1980, IEEE-323-1974, and IEEE-344-1975 as part of the Unit 2 Equipment Qualification Program.

5.4.12.4 Inspection and Testing

Inspection and testing of all valves and valve power actuators have been performed in accordance with Safety Class 1, 2, and 3 valves, ANSI B31.1 for Safety Class 4 valves, and the additional requirements of the design specification as applicable.

Hydrostatic shell tests have been performed in accordance with Article NB-6000 for Safety Class 1 valves, Article NC-6000 for Safety Class 2 valves, and Article ND-6000 for Safety Class 3 valves. Seat-tightness tests for Safety Class 1, 2, and 3 valves, and shell and seat-tightness tests for Safety Class 4 valves were performed in accordance with Manufacturer's Standardization Society Standard MSS-SP61 as a minimum. Certain Class 4 air-actuated valves have had seat-tightness testing performed in accordance with ANSI B16.104 in lieu of MSS-SP61.

Nondestructive testing of Safety Class 1, 2, and 3 valves was performed in accordance with the applicable subsection of ASME Section III and the additional requirements of the design specification when required. Personnel performing nondestructive testing on Safety Class 1, 2, and 3 valves were qualified in accordance with Society of Nondestructive Testing Standard SNT-TC-1A. Nondestructive testing of Safety Class 4 valves was performed in accordance with ANSI B31.1 and the additional requirements of the design specification when applicable.

All power-actuated valve operators have been assembled, factory tested, and adjusted on the valve for proper operation, position and torque switch setting, position transmitter function (where applicable), and speed requirements at the manufacturer's shop. Valve actuator electric motors have been furnished in accordance with applicable sections of NEMA Standard MG-1. Assembled power-actuated valves have been tested to demonstrate adequate

stem thrust (or torque) capability to operate the valve within the specified time at specified differential pressure. Tests verified that no mechanical damage to valve components occurred during full stroking of the valve.

Operational Analysis

Preoperational and operational testing performed on the installed valves consists of total circuit checkout and performance tests.

Valves that function as containment isolation valves will be exercised in accordance with Technical Specifications to assure their operability at the time of an emergency or faulted condition. Other valves, serving as system blocks or throttling valves, will be exercised when appropriate.

5.4.13 Safety and Relief Valves

5.4.13.1 Safety Design Bases

Overpressure protection has been provided at isolatable portions of systems in accordance with the rules set forth in Safety Class 1, 2, and 3 components.

5.4.13.2 Description

Pressure relief valves have been designed and constructed in accordance with the same code class as that of the line valves in the system.

Table 3.2-1 lists the applicable code classes for valves. The design criteria, design loading, and design procedure are described in Section 3.9.3B. Specific data (e.g., capacity, setpoint) are discussed in Section 5.2.2.

5.4.13.3 Safety Evaluation

The use of pressure-relieving devices assures that overpressure does not exceed 10 percent above the design pressure of the system. The number of pressure-relieving devices on a system or portion of a system has been determined on this basis.

5.4.13.4 Inspection and Testing

No provisions are made for in-line testing of pressure relief valves. Certified set pressures and relieving capacities are stamped on the body of the valves by the manufacturer, and further examinations would necessitate removal of the component.

5.4.14 Component Supports

Support elements are provided for those components included in the RCPB and the connected systems.

5.4.14.1 Safety Design Bases

The structural integrity of component supports is such that the supports are capable of safely sustaining the maximum combination of design loads.

Design loading combinations, design procedures, and acceptability criteria are as described in Sections 3.9.3A and 3.9.3B. Flexibility calculations and seismic analysis for Safety Class 1, 2, and 3 components conform to the appropriate requirements of ASME Section III.

Support types and materials used for fabricated support elements conform to Articles NF-2000 and NF-3000 of ASME Section III. Pipe support spacing guidelines of Table NF-3133.1-1 of ASME Section III were followed.

5.4.14.2 Description

The use and location of rigid-type supports, variable or constant spring-type supports, snubbers, and anchors or guides are determined by flexibility and seismic and stress analysis. Component support elements are manufacturer's standard items. Direct weldments to thin-wall pipe are avoided where possible.

5.4.14.3 Safety Evaluation

The flexibility and seismic/dynamic analyses performed for the design of adequate component support systems included all transient loading conditions expected by each component. Provisions have been made to protect spring-type supports from the initial deadweight loading due to hydrostatic testing of steam systems to prevent damage to this type of support.

5.4.14.4 Inspection and Testing

After completion of the installation of a support system, all hanger elements are visually examined to assure that they are in correct adjustment to their cold setting positions. Upon hot startup operations, thermal growth will be observed to confirm that spring-type hangers will function properly between their hot and cold setting positions. Final adjustment capability is provided on all spring hanger-type supports.

5.4.15 References

1. Ianni, P. W. Effectiveness of Core Standby Cooling Systems for General Electric Boiling Water Reactors. APED-5458, March 1968.

2. Analysis of Recirculation Pump Under Accident Conditions, Revision 2, General Electric Company, March 30, 1979.

3. Design and Performance of General Electric Boiling Water Reactor Main Steam Line Isolation Valves, General Electric Company Atomic Power Equipment Department, APED-5750, March 1969.

4. Licensing Topical Report, Power Uprate Licensing Evaluation for Nine Mile Point Nuclear Power Station, Unit 2, NEDC-31994P, Revision 1, May 1993.

CHAPTER 6

ENGINEERED SAFETY FEATURES

TABLE OF CONTENTS

CHAPTER 6

TABLE OF CONTENTS (Cont'd.)

CHAPTER 6

TABLE OF CONTENTS (Cont'd.)

CHAPTER 6

TABLE OF CONTENTS (Cont'd.)

CHAPTER 6

TABLE OF CONTENTS (Cont'd.)

CHAPTER 6

TABLE OF CONTENTS (Cont'd.)

CHAPTER 6

LIST OF TABLES

CHAPTER 6

LIST OF TABLES (Cont'd.)

CHAPTER 6

LIST OF TABLES (Cont'd.)

CHAPTER 6

LIST OF TABLES (Cont'd.)

CHAPTER 6

LIST OF TABLES (Cont'd.)

CHAPTER 6

LIST OF TABLES (Cont'd.)

CHAPTER 6

LIST OF FIGURES

CHAPTER 6

LIST OF FIGURES (Cont'd.)

CHAPTER 6

LIST OF FIGURES (Cont'd.)

CHAPTER 6

LIST OF FIGURES (Cont'd.)

CHAPTER 6

LIST OF FIGURES (Cont'd.)

CHAPTER 6

LIST OF FIGURES (Cont'd.)

CHAPTER 6

LIST OF FIGURES (Cont'd.)

CHAPTER 6

LIST OF FIGURES (Cont'd.)

CHAPTER 6

LIST OF FIGURES (Cont'd.)

CHAPTER 6

LIST OF FIGURES (Cont'd.)

CHAPTER 6

LIST OF FIGURES (Cont'd.)

CHAPTER 6

LIST OF FIGURES (Cont'd.)

CHAPTER 6

LIST OF FIGURES (Cont'd.)

TABLE 6.1-1

PRINCIPAL PRESSURE-RETAINING MATERIAL FOR ESF COMPONENTS
(Non-NSSS Scope of Supply)

Component	Form	Material	Specification (ASME/ASTM)
Containment			
Drywell liner			See Section 3.8.1.6.2 for material specifications
Suppression chamber liner			
Drywell head			
Access openings			
Penetrations			
Suppression vent downcomers	Pipe	Stainless steel	SA-312 Type 304
Vacuum relief valves	Forging	Stainless steel	SA-182 F316L
Reactor pedestal liner	Plate	Stainless steel	SA-240 Type 304L
DBA hydrogen recombiner system			
Piping-external	Pipe	Carbon steel	SA-106 Gr. B
	Pipe	Stainless steel	SA-312 Type 304
Fittings-external	Forging	Carbon steel	SA-105
	Forging	Carbon steel	SA-234 WPB
	Forging	Stainless steel	SA-182 F304
	Forging	Stainless steel	SA-403 WP304 or WP304W
Valves-External	Forging	Carbon steel	SA-105
	Forging	Stainless steel	SA-182 F316
Bolts	Bar	Low alloy steel	SA-193 Gr. B7
	Bar	Stainless steel	SA-193 Gr. B6
Nuts	Forging	Carbon steel	SA-194 Gr. 2H
	Forging	Stainless steel	SA-194 Gr. 6
Strainers			
Body	Casting	Stainless steel	SA-351 Gr. CF8
Flange	Forging	Stainless steel	SA-182 F304
Recombiner unit			
Heat exchanger			
Piping	Pipe	Stainless steel	SA-312 Type 316
Fittings	Forgings	Stainless steel	SA-182 Type 316
Reaction chamber			
Piping	Pipe	Stainless steel	SA-312 Type 304 SA-358 Type 304 SA-376 Type 304
Piping	Pipe	Carbon steel	SA-333 Gr. 6
	Pipe	Carbon steel	SA-106 Gr. B

27

TABLE 6.1-1 (Cont)

Component	Form	Material	Specification (ASME/ASTM)
Valves	Forging	Carbon steel	SA-105
ECCS			
Piping	Pipe	Carbon steel	SA-106 Gr.B
	Pipe	Stainless steel	SA-358 Type 304 Cl.1 or Type 316
	Pipe	Stainless steel	SA-376 Type 304 or SA-312 Type 304
Fittings	Forging	Carbon steel	SA-105
		Carbon steel	SA-234 WPB
		Stainless steel	SA-182 F304
		Stainless steel	SA-403 WP304 or WP304W
Valves	Forging	Carbon steel	SA-105
	Castings	Carbon steel	SA-216 Gr. WCB
	Plates	Carbon steel	SA-515 Gr. 70
	Forgings	Stainless steel	SA-182 F316 or F304
	Castings	Stainless steel	SA-351 Gr. CF8
Standby liquid control system			
Injection line	Pipe	Stainless steel	SA-312 Type 316L
Valves: ASME Safety Class I	Forgings	Stainless steel	SA182-F316L
ASME Safety Class II	Forgings	Stainless Steel	SA182-F316

28

TABLE 6.1-2

PRINCIPAL ESF COMPONENT MATERIALS

(NSSS Scope of Supply)

Component	Form	Material	Specification (ASTM/ASME)
RHR heat exchanger			
Shell, head and channel	Plate	Carbon steel	SA-516 Gr. 70
Tubesheet	Plate	Carbon steel	SA-516 Gr. 70
Tubesheet-cladding on channel side	Cladding	Weld deposit	Type 309L or 309 for first layer, 308L for subsequent layers
Nozzles - shell inlet and outlet	Forgings	Carbon steel	SA-350-LF2
Nozzles-channel inlet and outlet	Forgings	Carbon steel	SA-350-LF2
Flanges - shell side	Forgings	Carbon steel	SA-350-LF2
Flanges - channel side	Forgings	Carbon steel	SA-350-LF2
Tubes	Tubing	Stainless steel	SA-249 T-304L
Studs	Bar	Low alloy steel	SA-193 Gr. B7
Nuts	Bar	Low alloy steel	SA-194 Gr. 7
RHR, HPCS, and LPCS pumps			
Bowl assembly	Casting	Carbon steel	SA-216 Gr. WCB or A-216 Gr. WCB
Discharge head shell	Plate	Carbon steel	SA-516 Gr. 70
Discharge head cover	Forging	Carbon steel	SA-105
Suction barrel shell and dished head	Plate	Carbon steel	SA-516 Gr. 70
Flanges	Forging	Carbon steel	SA-105
Pipe (RHR, LPCS pumps)	Pipe	Carbon steel	SA-106 Gr. B
Pipe (HPCS pump)	Pipe	Carbon steel	SA-106 Gr. B
Shaft	Bar	Stainless steel	A-276 Type 410 Cond. H
Impeller	Casting	Stainless steel	A-351 Gr. CA6NM
Studs	Bar	Low alloy steel	SA-193 Gr. B7
Nuts	Forgings	Low alloy steel	SA-194 Gr. 7
Cyclone separator body and cover	Bar	Stainless steel	SA-479 Type 304
HPCS valves			
Body, bonnet, and disc	Casting	Carbon steel	SA-216 WCB
Stem	Bar	Stainless steel	A-314 Type 410 for valve F004; A-479 Type 410 for all others
Studs	Bar	Alloy steel	SA-193 Gr. B7
Nuts	Forgings	Carbon steel	SA-194 Gr. 2H

TABLE 6.1-2 (Cont)

Component	Form	Material	Specification (ASTM/ASME)
Control rod velocity limiter	Casting	Stainless steel	A351 Gr. CF8
Main steam flow restrictor	Casting	Stainless steel	SA351 Gr. CF8 (upstream insert)
	Casting	Carbon steel	SA216 Gr. WCB (downstream insert)
Main steam safety relief valves			
Body and bonnet	Casting	Carbon steel	SA-352, Gr. LCB
Disc	Casting	Stainless steel	SA-351, Gr. CF3A
Stem	Bar	Stainless steel	A-564 Type 630

TABLE 6.1-3

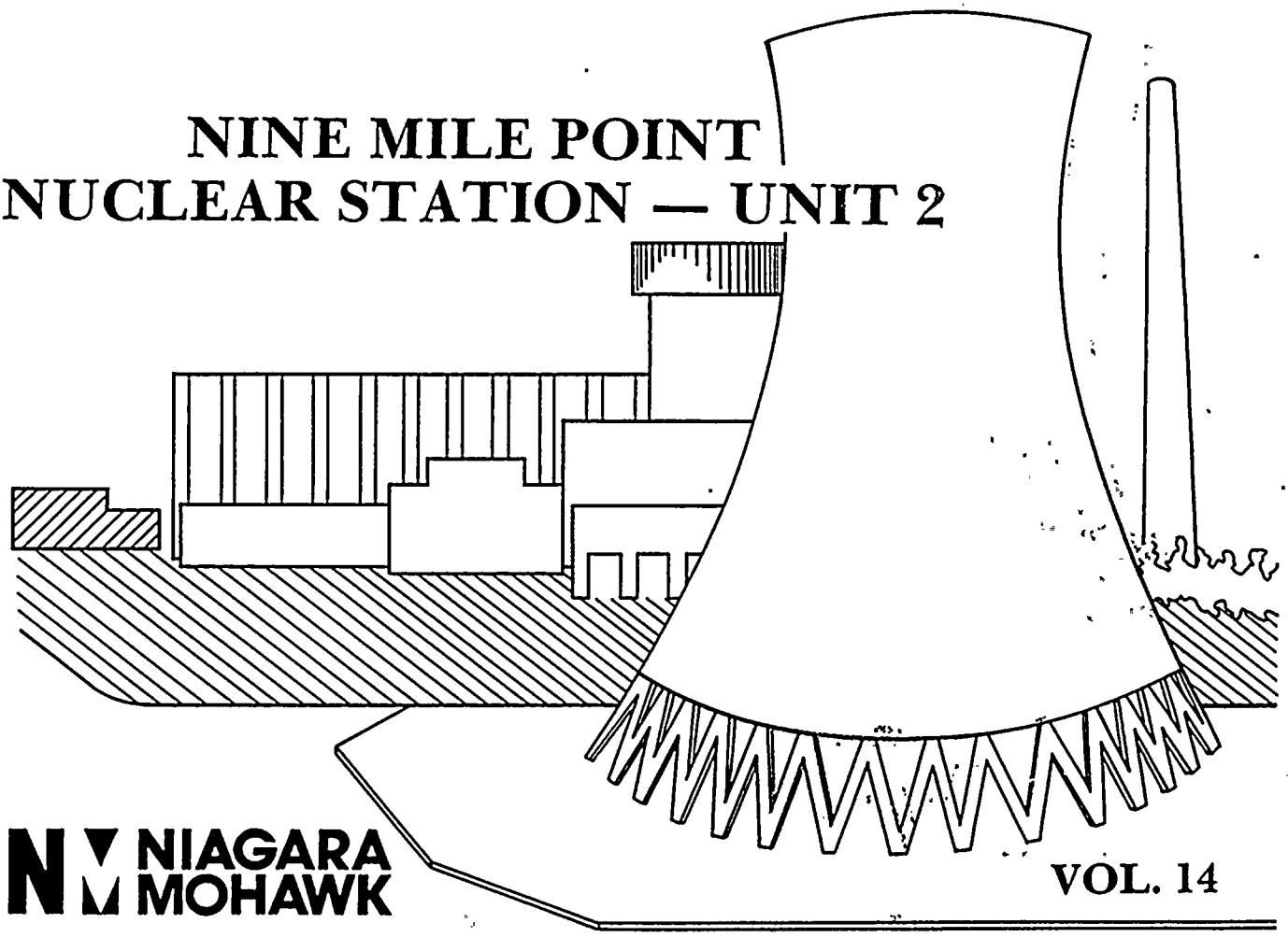
UNQUALIFIED PROTECTIVE COATINGS AND ORGANIC MATERIALS USED INSIDE THE PRIMARY CONTAINMENT

	Material	Quantity
Protective Coatings		
Inside Drywell		
Portions of the liner and supports and also certain misc. equipment	Inorganic zinc	6,700 ft^2 @3.3 to 6.0 mil DFT
	Epoxy based	8,960 ft^2 @8 mil DFT
	Alkyd based	300 ft^2 @3 mil DFT
	Modified phenolic	460 ft^2 @5 mil DFT
Recirculation pumps	Alkyd based	1,100 ft^2 @5 mil DFT
		220 ft^2 @3 mil DFT
Inside Suppression Pool		
Valve actuator enclosures	Alkyd based	50 ft^2 @3 mil DFT
Other Organic Materials		
Cable Insulation	**Covered**	**Uncovered**
Ethylene propylene rubber	1,920 lb @23 ft^3	1,280 lb @16 ft^3
Hypalon	8,890 lb @92 ft^3	2,320 lb @23.3 ft^3
Cross-linked polyethylene	7,225 lb @82.9 ft^3	100 lb @1.1 ft^3
Polypropylene	809 lb @8 ft^3	0
Motor electrical insulation[1]	None	1,390 lb
Shimming material	Devcon plastic steel B (catalyzed epoxy with 80% steel)	300 lb

(1) Approximate weight of recirculation drive motor stator insulation, wedges, and detectors.

UPDATED SAFETY ANALYSIS REPORT

NINE MILE POINT NUCLEAR STATION — UNIT 2



NIAGARA MOHAWK

VOL. 14

6.2 CONTAINMENT SYSTEMS

6.2.1 Containment Functional Design

This section establishes the design basis for the primary containment structure and provides the major design features and evaluation of the primary containment to perform the intended safety functions during all normal and postulated accident conditions throughout the operating life of the plant.

6.2.1.1 Containment Structure

The primary containment structure of Unit 2 consists of the drywell, the pressure suppression chamber which stores a large volume of water, and the drywell floor which separates the drywell and suppression chamber. The drywell is a steel-lined reinforced concrete vessel in the shape of a frustum of a cone, closed by a dome with a torispherical head. The pressure suppression chamber is a cylindrical stainless steel clad steel-lined reinforced concrete vessel located below the drywell. The primary containment structure houses the reactor vessel, the reactor recirculation system, and other branch connections of the RCPB.

6.2.1.1.1 Design Bases

The primary containment structure, including subcompartments (Section 6.2.1.2), meets the following functional design bases.

<u>Containment Vessel Design</u>

The containment vessel design bases are:

1. The primary containment has the capability to maintain its functional integrity during and following the peak transient pressure and temperatures that would occur following any postulated LOCA. The LOCA includes the worst single failure (which leads to the maximum primary containment pressure and temperature) and is further postulated to occur simultaneously with loss of offsite power (LOOP) and a safe shutdown earthquake (SSE).

2. The primary containment structure also withstands the peak environmental transient pressures and temperatures associated with the postulated spectrum of line breaks.

3. The primary containment has the capability to withstand jet forces associated with the flow from the postulated rupture of any pipe within it.

4. The primary containment system is protected from or designed to withstand missiles from internal sources

and excessive motion of pipes that could directly or indirectly endanger the integrity of the containment.

5. The primary containment is designed for the hydrodynamic loads in the Design Assessment Report for Hydrodynamic Loads (DAR, Appendix 6A).

Containment Subcompartment Design

The effects of primary containment subcompartment pressurization (Section 6.2.1.2) on the postulated pipe ruptures have been evaluated.

Drywell Internal Pressure

The DBA postulated for the calculation of the maximum pressure acting on the drywell walls is a double-ended rupture (DER) of a 24-in recirculation suction line. This event is more severe than a DER of the 26-in main steam line. The peak calculated drywell pressure following the recirculation pump suction line DER is shown in Table 6.2-4.

Suppression Chamber Internal Pressure

The DBA for the suppression chamber is a DER of a 24-in recirculation suction line. The peak pressure occurs shortly after the initial reactor blowdown. The maximum calculated suppression chamber pressure is shown in Table 6.2-4.

Drywell Floor Differential Pressure

1. Downward ΔP

   The DBA for the downward floor differential pressure is a DER of a 24-in recirculation suction line. The maximum differential pressure (drywell to suppression chamber) occurs at the time of downcomer vent clearing when steam and air begin to flow to the suppression chamber. The maximum calculated downward differential pressure on the floor is shown in Table 6.2-4. A significant margin remains with respect to the design downward differential pressure of 25 psid. In addition to this, the drywell floor is structurally designed to withstand a pressure load equal to 1.5 times the design pressure (i.e., 25 psid), thereby widening the safety margin.

2. Upward ΔP

   The DBA for the upward floor differential pressure is a small steam line break with rapid drywell depressurization caused by initiation of the drywell sprays considering the worst-case minimum spray water temperature.

In this case all the drywell air is transferred to the suppression chamber prior to spray initiation. It is assumed that the Operator actuates drywell sprays 30 min after the drywell pressure reaches 30 psig and a rapid depressurization of the drywell is initiated.

The drywell depressurization and upward ΔP are limited by the return flow of air to the drywell through the vacuum breakers.

The following major assumptions have been made to maximize the upward floor ΔP.

1. Initial suppression pool temperature is 40°F.
2. The service water temperature is assumed to be 32°F.
3. Containment spray flow to the suppression chamber is ignored.
4. Both loops of drywell sprays are actuated with 100 percent thermal effectiveness.

The maximum upward loading on the drywell floor is 4.70 psid or less (including the effects of power uprate) for the worst-case transient assuming that only three out of the four vacuum breaker flow paths are available. The design upward ΔP is 10 psid which provides a substantial margin relative to the maximum calculated value. In addition to this, the drywell floor is structurally designed to withstand a differential pressure load equal to 1.5 times the design pressure (i.e., 10 psid), thereby widening the safety margin.

Drywell Atmosphere Temperature

The DBA for the drywell atmosphere temperature is a postulated small steam line break. The maximum drywell atmospheric temperature is 340°F and the maximum drywell liner temperature corresponding to the saturation pressure equivalent to the design pressure of 45 psig is 293°F.

Suppression Chamber Temperature

The DBA for the suppression chamber and the suppression pool temperature is the DER of a 24-in recirculation suction line. The design basis temperature of 212°F occurs between 1 and 15 hr after the break depending on the number of residual heat removal (RHR) pumps and heat exchangers used to remove heat from the system.

Mass and Energy Release for the Primary Containment Design

The maximum postulated release of mass and energy to the primary containment is based upon the instantaneous DER of a 24-in

reactor recirculation pump suction line or 26-in main steam line. The effects of metal-water reaction and other chemical reactions following the DBA can be accommodated in the primary containment design. Further description of mass and energy release data is provided in Section 6.2.1.3.

Energy Removal from the Containment

Energy is removed from the primary containment after a LOCA by circulating the suppression pool water through the RHR system heat exchangers where heat is removed by the service water (SWP) system. The primary containment spray mode of the RHR system can also be used to condense steam and reduce the temperature in the primary containment following a LOCA. Section 6.2.2 describes these ESFs in more detail. Energy is also absorbed from the primary containment atmosphere by way of passive heat sinks.

Pressure Suppression Feature

The Unit 2 primary containment conforms to the fundamental principles of a Mark II pressure suppression system. The water stored in the suppression pool is capable of condensing the steam displaced into the pool through the downcomer vents, and the amount of water is sufficient that no Operator action is required for at least 10 min immediately following initiation of a LOCA. In addition, the design allows any significant amounts of water from pipe breaks within the primary containment to drain back to the suppression pool. This closed loop ensures a continuous, adequate supply of water for core cooling.

Primary Containment Leakage Design

The primary containment system, in conjunction with other engineered safeguards, has the capability to limit leakage during the postulated design basis LOCA so that offsite doses do not exceed the design values set forth in 10CFR100.

Hydrostatic Loading Design

The primary containment design permits filling the primary containment with water to the level of the reactor vessel flange.

Primary Containment Leakage Testability

The primary containment system is designed to allow for periodically conducting tests in accordance with 10CFR50 Appendix J to confirm the leak-tight integrity of the primary containment and its penetrations.

Suppression Pool Hydrodynamic Loads

The amplitudes and frequency of the dynamic forcing functions that result from hydrodynamic loads due to steam and air

discharges into the suppression pool are discussed in the Unit 2 DAR for Hydrodynamic Loads (Appendix 6A).

6.2.1.1.2 Design Features

The primary containment design features include the drywell, the pressure suppression chamber, downcomers between the drywell and the suppression chamber, isolation valves, vacuum breakers, and the RHR system for containment heat removal. The primary containment is a steel-lined, reinforced concrete pressure vessel that is closed at the top by the drywell head assembly. The steel drywell head assembly forms a gas-tight enclosure (Section 3.8.1). The analysis of the steel membrane is included in Section 3.8.1. The design, description, and analysis of the reinforced concrete primary containment structure are included in Section 3.8.1. The principal design parameters of the primary containment, suppression pool, and vent downcomers are listed in Table 6.2-3.

Drywell

The drywell is a steel-lined reinforced concrete vessel in the shape of a truncated cone having a base diameter of approximately 91 ft and a top diameter of approximately 34 ft. The floor of the drywell serves both as a pressure barrier between the drywell and the suppression chamber and as the support structure for the reactor pedestal and downcomers.

The drywell houses the reactor and associated equipment. The primary function of the drywell is to contain the radioactivity and withstand pressures and temperatures resulting from a breach of the RCPB, up to and including an instantaneous circumferential break of a single reactor recirculation pump suction pipe, and to provide a holdup time for decay of any radioactive material released. The drywell is designed to resist the forces of an internal design pressure of 45 psig in combination with thermal, seismic, and other forces as outlined in Chapter 3.

Pressure Suppression Chamber

The pressure suppression chamber is a stainless steel clad steel-lined, reinforced concrete vessel in the shape of a cylinder, having an inside diameter of 91 ft. The foundation mat, to which the vessel is anchored, is lined with steel plates within the inside diameter of the cylinder. The steel plates are welded to each other and to steel embedments to maintain the primary containment function of a gas-tight enclosure.

Pressure Suppression Pool

The pressure suppression pool, which is contained in the pressure suppression chamber, stores sufficient water to condense the steam released from blowdown of the reactor coolant system (RCS) after a LOCA or from safety/relief valve (SRV) discharge during

accident or normal operational transients. Steam is transferred to the pressure suppression pool by the downcomers and the discharge piping of the SRVs.

Approximately 150,000 cu ft of water is contained within the suppression chamber. The suppression pool serves both as a heat sink for transients and accidents and as a reservoir of water for the core standby cooling systems. It is the primary source of water for the low-pressure core spray (LPCS) and low-pressure coolant injection (LPCI) systems, provides a safety-related source of water for the reactor core isolation cooling (RCIC) and high-pressure core spray (HPCS) systems. The water level and temperature of the pressure suppression pool are continuously monitored in the control room.

Downcomers

The downcomers consist of 121 pipes open to the drywell and submerged 9.5 ft below the low water level (operating minimum) of the suppression pool, providing a flow path for uncondensed steam into the pool. The internal diameter of each downcomer is 23.25 in. The downcomers project 3 to 6 in above the drywell sloping (or sloped) floor so that small quantities of water leakage flow past the downcomers and are collected in the drywell floor drain system. Each downcomer opening is shielded by a 2 1/4-in thick steel deflector plate to prevent overloading any single vent pipe by direct flow from a pipe break to that particular vent. The deflector plate also minimizes the potential for downcomer blockage by debris.

Vacuum Breakers

Vacuum breakers provide a return flow path from the suppression chamber gas space to the drywell. The vacuum breakers are designed to limit the negative differential pressure between the drywell and the suppression chamber to less than the design value (10 psid). Each vacuum breaker flow path has two relief valves in series to ensure a leak-tight boundary under positive drywell-to-suppression chamber differential pressure conditions. Three flow paths are required for the vacuum breaker design basis. One additional flow path is provided to accommodate the postulated single failure of one vacuum breaker.

The vacuum breakers are located inside the primary containment drywell and do not, therefore, form an extension of the primary containment boundary. These valves are mounted in piping that connects the drywell and suppression chamber. This location removes the vacuum breakers from the direct effects of chugging transients. Each vacuum breaker is designed to withstand a valve disc velocity of 20.8 radians/sec opening and 22.4 radians/sec closing. The opening and closing velocities for pool swell phenomena have been determined and analyzed by Continuum Dynamics, Inc., for Commonwealth Edison Company's Lasalle Units 1 and 2. Testing was also performed. The Unit 2 vacuum breakers

are of the same size and of similar design as the Lasalle valves. The tests were performed on a "modified" and an "original" as-installed valve configuration with the following results which were compared to Unit 2 opening and closing velocities.

	Opening Velocity	Closing Velocity
Lasalle		
Required	18.5 rad/sec	25.5 rad/sec
"Original" valve test	19.8 rad/sec	25.0 rad/sec
"Modified" valve test	20.2 rad/sec	25.8 rad/sec
Unit 2	20.8 rad/sec	22.4 rad/sec

The "original" configuration showed deformations occurred and the pallet did not seat uniformly. A bypass flow area of 10.8 sq in was measured compared to an allowable of 21.8 sq in. No other damage was observed and it was concluded that the valves would be acceptable for the intended function.

The modified valve showed no damage.

The modifications have been included in the Unit 2 valves and consist of:

1. Addition of an impact load distribution device.
2. Material upgrade of the eccentric shaft and hinge plate.
3. Hinge plate thickness increase.
4. Addition of a bumper stop to soften the opening velocity impact.

The closing test velocities exceeded the Unit 2 velocity for both tests and the "modified" valve test was within 3 percent of the Unit 2 opening velocity.

Analysis for the modified vacuum breakers indicated that margins of safety of the disc were 1.3 rad/sec for closing and 2.7 rad/sec for opening velocity. Since this margin exists and the test velocity was within 3 percent of the Unit 2 opening, the Unit 2 valves are shown to be adequate and need not be tested.

The vacuum breakers have the capability for remote manual testing from the local instrument panel. This design provides assurance of limiting the differential pressure between the drywell and suppression chamber and ensures proper valve operation and testing during normal plant operation. A monthly operability test of the drywell-to-suppression pool vacuum breakers is

conducted by cycling each vacuum breaker through at least one complete cycle of full travel. Verification of actual position is determined by the full open position switch and the full shut position limit switches.

No vacuum relief valves are provided between the drywell and the reactor building atmosphere. The primary containment structure can accommodate subatmospheric pressure of approximately 10 psia at maximum operating water level.

6.2.1.1.3 Design Evaluation

The original, pre-uprate large line break analyses described in this section were performed at 105 percent of the original rated steam flow, corresponding to 104.3 percent of the original rated thermal power (3,323 MWt), or 100 percent of the uprated power (3,467 MWt). These analyses provide the basis for identifying the limiting large break and for establishing the containment leak rate test pressure, and also evaluate the sensitivity of the containment response to variations in initial conditions and analysis assumptions.

For power uprate, the limiting DBA LOCA (a double-ended guillotine break of a recirculation suction line) is analyzed to demonstrate that power uprate operation does not result in exceeding the containment design limits. This limiting case analysis is addressed in Section 6.2.1.1.5. For intermediate and small line breaks, the results of analyses performed for power uprate operating conditions vary only slightly from the original, pre-uprate analysis results, which is consistent with the evaluations documented in Reference 8.

The key design parameters and maximum calculated parameters are provided in Tables 6.2-3 through 6.2-5. These design and maximum calculated accident parameters are not determined from a single accident event but from an envelope of accident conditions. As a result, there is no single DBA for this containment system.

A maximum drywell and suppression chamber pressure occurs near the end of the blowdown phase of a LOCA. Approximately the same peak pressure occurs for the break of either a recirculation line or a main steam line. Both accidents are evaluated.

The most severe drywell temperature condition is predicted for a small steam break above the reactor water level. To demonstrate that breaks smaller than the rupture of the largest primary system pipe will not exceed the primary containment design parameters, the primary containment system responses to an intermediate size liquid break and a small size steam break are evaluated. The results show that the primary containment design conditions are not exceeded for these small break sizes.

Table 6.2-3 lists the key design parameters of the Unit 2 primary containment system including the design characteristics of the

drywell, suppression chamber, and the pressure suppression vent system.

Table 6.2-6 lists the performance parameters of the related ESF systems that supplement the design conditions of Table 6.2-3 for primary containment heat removal during postaccident operation. Performance parameters given include those applicable to design capacity operation and those reduced capacities assumed for primary containment analyses.

Additional containment evaluation is provided in Appendix 15B for operation under single recirculation loop conditions.

Accident Response Analysis

The response of the pressure suppression containment system to postulated pipe breaks in the drywell is analyzed with the LOCTVS computer program. This program models the reactor system, drywell, downcomer vent system, suppression chamber, and active and passive heat removal systems.

The primary results of the LOCTVS accident analyses consist of the transient pressure and temperature response of the drywell and suppression chamber, and the transient temperature response of the suppression pool. The analyses assume instantaneous pipe ruptures concurrent with the LOOP. The worst-case single active component failure is the failure of one of the two electrical divisions (Division I or II) resulting in minimum containment heat removal capability. Failure of either Division I or II results in nearly identical peak pressure and temperature.

Break Spectrum

To ensure that the primary containment system design parameters are not exceeded for pipe breaks smaller than the DER of the largest primary system pipe, a spectrum of break sizes is analyzed. The following breaks are considered:

1. An instantaneous guillotine DER of a recirculation pump suction line.

2. An instantaneous guillotine DER of a main steam line.

3. An intermediate size liquid line rupture.

4. A small size steam line rupture.

Energy release from these accidents is reported in Section 6.2.1.3.

Recirculation Line Rupture

The instantaneous guillotine DER rupture of a recirculation suction line results in the discharge of maximum flow rate of

primary system fluid and energy into the drywell. Figure 6.2-1 shows the location of the break. Immediately following the rupture, the flow out of both sides of the break will be limited to the maximum allowed by critical flow considerations. Figure 6.2-1 shows a schematic view of the flow paths to the break. In the side adjacent to the suction nozzle, the flow will correspond to critical flow in the pipe cross section of 2.598 sq ft. Reverse flow in the discharge side of the recirculation pump will correspond to critical flow at the 50 jet pump nozzles (5 nozzles per jet pump) associated with the broken loop, providing an effective break area of 0.461 sq ft. In addition, a 4-in cleanup line crosstie adds 0.088 sq ft to the critical flow area, yielding a total of 3.147 sq ft (Figure 6.2-1).

Assumptions for Reactor Blowdown The following assumptions are made with respect to the blowdown from the RCPB for the original, pre-uprate analysis (the impact of power uprate on the large break analysis is addressed in Section 6.2.1.1.5):

1. The reactor is operating at 105-percent of the original rated steam flow (corresponding to a core thermal power of 3,467 MWt) at the time of the recirculation line break. The initial conditions are selected to maximize the parameter of interest, that is, primary containment pressure.

2. The instantaneous DER of the recirculation suction line is considered. This results in the most rapid coolant loss and depressurization, with coolant being discharged from both ends of the break.

3. The vessel depressurization flow rates are calculated using Moody's critical flow model assuming liquid-only outflow, since this assumption maximizes the energy release to the drywell[1,2]. Liquid-only outflow implies that all vapor formed in the reactor pressure vessel (RPV) by flashing rises to the surface rather than being entrained in the blowdown. In reality, some of the vapor would be entrained in the break flow which would significantly reduce the RPV discharge flow rates.

4. The core decay heat and the sensible heat (after the accident) released in cooling the fuel to 551°F are included in the RPV depressurization calculation. The rate of energy release is calculated using a conservatively high heat transfer coefficient through the depressurization period. By maximizing the assumed energy release rate, the RPV is maintained at nearly rated pressure (within 10 percent) for approximately 16 sec. This high RPV pressure increases the calculated blowdown flow rates, which is again conservative for containment analysis purposes. The stored energy of the fuel at temperatures below 551°F is released to the

vessel fluid along with the stored energy in the vessel and internals, as vessel fluid temperatures decrease below 551°F during the remainder of the transient calculation.

5. The main steam isolation valves (MSIVs) start closing at 0.5 sec after the accident. They are assumed to fully close in the shortest possible time of 3 sec following closure initiation. The closure signal for the MSIVs may occur from low reactor water level, so the valves may not receive a signal to close for more than 4 sec, and the closing time may be as long as 5 sec. By assuming rapid closure of these valves, the RPV is maintained at a high pressure which maximizes the calculated discharge of high-energy water into the drywell.

6. The reactor feedwater flow continues until all the high-energy feedwater (above 200°F) is depleted. This results in slightly lower pressure initially as compared to a case with no feedwater; however, subsequent peak (maximum peak) pressure is higher.

   The recirculation line blowdown mass flow rate, the corresponding enthalpy, and the RCPB pressure with and without the addition of feedwater are summarized in Tables 6.2-7 and 6.2-8, respectively.

Assumptions for Primary Containment Pressurization The pressure response of the primary containment during the blowdown period of the accident is analyzed using the following assumptions:

1. Thermodynamic equilibrium exists in the drywell and suppression chamber. The analysis assumes complete mixing of the effluents in the drywell and suppression chamber.

2. Downcomer flow consists of the homogeneous mixture of the fluid in the drywell. The use of this assumption results in liquid carryover into the drywell downcomer vents.

3. The fluid flow in the drywell-to-suppression chamber downcomer is compressible except for the liquid phase.

Assumptions for Long-Term Cooling Following the blowdown period, the emergency core cooling systems (ECCS) (Section 6.3) provide water for core flooding and long-term decay heat removal. The primary containment pressure and temperature response during this period are analyzed using the following assumptions:

1. Two RHR pumps in LPCI mode are used to flood the core prior to 10 min after the accident. The HPCS pump is assumed to be available for the entire transient.

2. After 10 min, one of the RHR pumps in LPCI mode may be transferred to the containment spray or pool cooling mode. An additional 10 min (total of 20 min) has been provided to the Operator to turn on the sprays. This is a manual operation. Note that Case C, in Item 5 below, addresses the accident wherein no credit is taken for containment sprays.

3. The effect of decay energy, stored energy, and energy from the metal-water reaction are considered.

4. After approximately 20 min, the RHR heat exchanger is activated to remove heat from the primary containment via the suppression pool.

5. The performance of the ECCS equipment during the long-term cooling period is evaluated for each of the following three cases of interest:

Case A	Offsite power available; all ECCS equipment and containment spray operating
Case B	LOOP; minimum diesel power available for ECCS and containment spray mode (only Electrical Division II available)
Case C	Minimum diesel power available for ECCS and pool cooling mode (only Electrical Division II available)

Initial Conditions for Accident Analysis The accident response analyses assume that the drywell, wetwell, and suppression pool are initially at the maximum normal operating conditions. Table 6.2-9 provides the initial RCPB and primary containment conditions used in the accident response evaluations.

Energy Sources All major sources of energy are considered in the calculation of the mass and energy release to the primary containment from each postulated pipe break. The sources of available energy include:

1. Stored energy in the RCPB.

2. Fission product and heavy element decay heat.

3. Fission power coastdown heat.

4. Stored energy in the RCPB metal piping, structures, and core.

5. Metal water reaction energy.

6. ECCS pump heat.

These data are provided in Tables 6.2-10 through 6.2-14.

Short-Term Accident Response - Original Pre-Uprate Analysis The calculated primary containment pressure and temperature responses for the recirculation line break are shown on Figures 6.2-2 through 6.2-5 and 6.2-6 through 6.2-9, respectively. The calculated peak drywell pressure is 39.75 psig, which is below the primary containment design pressure of 45 psig.

The suppression chamber is initially pressurized by the carryover of noncondensables from the drywell. As the transient continues, steam flows through the vent system and the temperature of the suppression pool water increases, causing an increase in the suppression chamber pressure and a corresponding increase in the drywell pressure. The peak drywell pressure occurs near the end of the blowdown and then the drywell pressure stabilizes at a higher pressure than that of the suppression chamber, the difference being equal to the height of water to the downcomer submergence. During the RPV depressurization phase, most of the noncondensable gases initially in the drywell are forced into the suppression chamber. However, following the depressurization the noncondensables will redistribute between the drywell and suppression chamber via the vacuum breaker system. This redistribution takes place as drywell pressure is decreased by the steam condensation process occurring in the drywell.

The ECCS supplies sufficient core cooling water to control core heatup and limit metal-water reaction to less than 1 percent pursuant to 10CFR50.46. The excess flow discharges through the recirculation line break into the drywell. This flow of water (steam flow is negligible) transports the core decay heat out of the RPV, through the broken recirculation line, in the form of hot water which flows into the suppression pool via the suppression chamber downcomer. This flow, in addition to heat losses to the drywell walls, provides a heat sink for the drywell atmosphere. The resulting depressurization of the drywell causes the vacuum breakers to open, allowing the noncondensables in the suppression chamber to redistribute into the drywell. Table 6.2-4 provides the peak pressure, temperature, and time parameters for the recirculation line break as predicted for the conditions of Tables 6.2-3, 6.2-6, and 6.2-9 corresponding with Figures 6.2-4 and 6.2-8. The peak calculated drywell floor differential pressure is 16.89 psid.

During the blowdown period of a LOCA, the pressure suppression downcomer conducts the flow of the steam-water-gas mixture in the drywell to the suppression pool for condensation of the steam. The pressure differential between the drywell and suppression pool controls this flow versus time. Figure 6.2-10 provides the mass flow versus time relationship through the vent system for this accident.

Long-Term Accident Response - Original Pre-Uprate Analysis To assess the adequacy of the primary containment following the

initial blowdown transient, an analysis was made of the long-term temperature and pressure response following the accident. The analysis assumptions are those discussed previously for the three cases of interest.

Case A - All ECCS Equipment Operating (with Containment Spray) This case assumes that offsite ac power is available to operate all ECCS equipment. During the first 10 min following the pipe break, the HPCS, LPCS, and all three RHR pumps in LPCI mode are assumed operating. All flow is injected directly into the reactor vessel. The Operator will turn on the sprays within 20 min following the signal that drywell pressure has reached 30 psig since, in the case of a large break, the spray system is not available for the first 10 min of the transient. Therefore, the Operator will switch two RHR pumps from the LPCI mode to containment spray mode to remove the energy from the primary containment after 10 min and complete this action within the next 10 min. During this mode of operation the flow from two of the RHR pumps is routed through the RHR heat exchangers, where it is cooled before being discharged into the primary containment spray headers.

The primary containment pressure response to this set of conditions is shown on Figure 6.2-5. The corresponding drywell and suppression pool temperature responses are shown on Figures 6.2-6 and 6.2-11. After the subsequent depressurization due to LPCS and LPCI core spray, energy addition due to core decay heat results in a gradual pressure and temperature rise in the primary containment. When the energy removal rate of the RHR exceeds the energy addition rate from the decay heat, the primary containment pressure and temperature reach a peak and decrease gradually.

Case B - Division I Unavailable (with Containment Spray Mode) This case assumes electrical Divisions II and III are available. For the first 10 min following the accident, one HPCS and two RHR pumps in LPCI mode are used to cool the core. Containment spray is operating and injecting into the drywell within 20 min after the spray initiation signal following the accident. During this mode of operation the RHR pump through one RHR heat exchanger is discharged through the primary containment spray nozzles. Under this condition, one HPCS pump and one RHR pump in LPCI mode continue to cool the core after 10 min while the other RHR pump operates in containment spray mode after 20 min. The containment response to this set of conditions is shown on Figure 6.2-3. The corresponding drywell and suppression pool temperature responses are shown on Figures 6.2-7 and 6.2-11.

Case C - Division I Unavailable (Pool Cooling Mode) This case assumes electrical Divisions II and III are available. For the first 10 min following the accident, one HPCS and two RHR pumps in the LPCI mode are used to cool the core. After 10 min the spray may be manually activated to further reduce the containment pressure. This case assumes that the spray is not activated.

After 10 min, one of the RHR pumps in LPCI mode is transferred to pool cooling mode to remove energy from the suppression pool. Under this condition, one HPCS pump and one RHR pump in LPCI mode continue to flood the reactor vessel after 10 min. The primary containment pressure response to this set of conditions is shown on Figures 6.2-2 and 6.2-4. The corresponding drywell and suppression pool temperature responses are shown on Figures 6.2-8, 6.2-9, and 6.2-11.

When comparing the primary containment spray Case B with the pool cooling Case C, the same duty on the RHR heat exchangers is obtained since the suppression pool temperature response is approximately the same as that shown on Figure 6.2-11. Thus, the same amount of energy is removed from the pool whether the exit flow from the RHR heat exchanger is injected back into the suppression pool or into the drywell as spray. Primary containment pressure is identical for the first 10 min; therefore, the peak pressures for Cases B and C are the same and the pressure is less than the primary containment design pressure of 45 psig.

Figure 6.2-12 shows the rate at which the RHR system heat exchanger removes the heat from the suppression pool following a LOCA (Section 6.2.2.3 describes the pool cooling and containment spray mode of the RHR system). The heat removal rate is shown for the three cases of interest. The first assumes that all the ECCS equipment is available, including both RHR heat exchangers and the necessary service water pumps. The second case is for the very degraded minimum cooling condition that would limit the heat removal capability to that of one RHR heat exchanger. For all cases, it was assumed that at the time of the accident the service water temperature was 77°F. This is the maximum that is expected at this site and is unlikely to exist for more than a limited period in the summer.

The long-term post-LOCA primary containment pressure and suppression pool temperature design evaluation also supports an increased service water temperature of 82°F. Figure 6.2-46A delineates the performance of the RHR heat exchanger used for the design evaluation with a K-factor of 199 Btu/sec-°F (overall U = 202 Btu/hr-°F-sq ft) and service water at 77°F. This figure also shows the actual heat exchanger performance with service water at 82°F and a conservative K-factor of 239 Btu/sec-°F. The heat removal rate is greater for the revised performance when suppression pool temperature is greater than 107°F. The expected pool temperature at the time of the containment heat removal system initiation is generally greater than 107°F following a LOCA. The short-term peak containment pressures are not affected by the service water temperature.

Energy Balance During Accident An energy balance for this accident showing the energy distribution at various times is given in Table 6.2-15. For the purpose of performing the energy balance, the system boundary is defined to be the primary

containment. Everything within the system is grouped according to whether or not it is primarily a heat source or a heat sink, although many items behave as both during the course of the transient. Entries opposite these items represent stored internal energy at a particular time. The reference temperature for stored heat is 32°F except for heat sinks where the reference temperature is their initial temperature. For drywell heat sinks, this is 135°F; for wetwell heat sinks, 90°F. The following energy sources and sinks are considered:

1. Blowdown energy release rates.

2. Decay heat rate and fuel relaxation energy.

3. Sensible heat rate.

4. Pump heat rate.

5. Heat removal rate from the suppression pool.

6. Passive heat sinks (modeling of the passive heat sinks is fully described in Tables 6.2-1 and 6.2-2).

For the case of a DER of the recirculation line, the energy balance is performed for the reactor system, the primary containment system, and the containment heat removal systems at time zero, at the time of peak drywell pressure, at the end of reactor blowdown, and at the time of the long-term peak pressure reached in the primary containment.

Chronology of Accident Events The complete description of the primary containment response to the design basis recirculation line break is given above. A chronological sequence of events for this accident from time zero is provided in Table 6.2-16, based on the original, pre-uprate analysis.

Main Steam Line Break - Original Pre-Uprate Analysis

The instantaneous DER of a main steam line between the RPV and the flow restrictor results in the maximum discharge of primary system fluid and energy to the drywell. The flow on both sides of the break will accelerate to the maximum allowed by critical flow considerations. The critical flow rate is determined by using the Moody flow model with the conservative assumption of zero friction. The flow from the reactor vessel side of the break is critical in the 3.05-sq ft area main steam line nozzle. Blowdown through the other end of the break occurs because the main steam lines are interconnected upstream of the turbine by the main steam header. This interconnection allows primary system fluid to flow to the drywell via the broken line. Flow will be limited by the critical flow in the 0.913-sq ft steam line flow restrictor. The total effective flow area is thus 3.963 sq ft, which is the sum of the steam line cross-sectional

area and the flow restrictor area. Section 6.2.1.3 provides information on the mass and energy release rates.

The decrease in steam pressure at the turbine inlet initiates closure of the MSIVs within approximately 200 msec after the break occurs. Also, MSIV closure signals are generated as the differential pressure across the main steam line flow restrictors increases above isolation setpoints. The instruments sensing flow restrictor differential pressure generate isolation signals within approximately 500 msec after the break occurs.

After 4 sec, the MSIVs in the broken line have closed sufficiently so that the MSIV flow area equals the flow restrictor area. At that time, the critical flow location changes from the flow restrictor to the MSIVs. Subsequent closure of the MSIVs in the broken line terminates flow from the flow restrictor side of the break at 5.5 sec after the postulated failure of the main steam line. Figures 6.2-13 and 6.2-14 show the break schematic and total effective break area versus time, respectively. The closing time of the MSIVs is between 3 and 5 sec.

Immediately following the break, the total flow rate of steam leaving the vessel exceeds the steam generation rate. This steam flow to steam generation mismatch causes an initial depressurization of the reactor vessel, and the resultant formation of steam bubbles within the reactor vessel liquid causes a rapid rise in water level. When the froth level reaches the vessel steam nozzles and the top of the steam dryers, flow out of the break changes from steam to a two-phase mixture. The two-phase critical flow rates are determined from the Moody model with the known values of vessel pressure and mixture enthalpy. During the first second of the blowdown, the blowdown flow will consist of saturated reactor steam. This initial period of all steam discharge results in a drywell atmosphere temperature condition of approximately 310°F. Figures 6.2-15 through 6.2-18 and 6.2-19 through 6.2-22 show the pressure and temperature response of the drywell and suppression chamber during the primary system blowdown phase of the accident. Suppression pool temperature response is shown on Figure 6.2-23.

Figure 6.2-21 shows that the drywell atmosphere temperature approaches approximately 310°F after 1 sec of primary system blowdown. At that time, the water level in the vessel reaches the steam line nozzle elevation and the blowdown flow changes to a two-phase mixture. This increased flow causes a more rapid drywell pressure rise. However, the peak differential pressure is 14.90 psi, which occurs shortly after the downcomer vent clearing transient. As the blowdown proceeds, the primary system pressure and fluid inventory decrease, resulting in reduced break flow rates.

Approximately 75 sec after the start of the accident, the primary system pressure has dropped significantly. At this time the

drywell atmosphere contains only steam. The blowdown continues because the RPV is still being pressurized from the hot feedwater addition until approximately 250 sec, at which time the peak drywell pressure occurs. Following this, drywell pressure is equal to the reactor vessel pressure and initial blowdown is over. Passive heat sinks continue to remove the energy and the drywell and suppression chamber pressure gradually reduce.

Table 6.2-4 presents the peak pressures, peak temperatures, and times of this accident with feedwater addition as compared to the recirculation line break.

The drywell and suppression chamber remain in this equilibrium condition until the RPV refloods. During this period, the ECCS pumps inject cooling water from the suppression pool into the reactor. This injection of water eventually floods the reactor vessel to the level of the steam line nozzles, and at this time, the ECCS flow spills into the drywell and thus reduces the drywell pressure. As soon as the drywell pressure drops below the suppression chamber pressure, the drywell vacuum breakers open and noncondensable gases from the suppression chamber flow back into the drywell. This process continues until the pressure in the two regions equalizes.

Intermediate Breaks

This classification covers those breaks for which the blowdown results in reactor depressurization and operation of the ECCS. This section describes the consequences to the primary containment of a 0.1-sq ft liquid break. This break area was chosen as being representative of the intermediate size break area range for both steam and liquid breaks.

Following an intermediate size break, the drywell pressure increases at a sufficiently slow rate that the dynamic effect of downcomer vent clearing is negligible and the downcomer clear when the drywell-to-wetwell differential pressure is equal to the downcomer submergence pressure. For Unit 2 primary containment design, the maximum distance between the pool surface and the bottom of the downcomer is 11 ft; thus the water level in the downcomers will reach this point when the drywell-to-containment pressure differential reaches 4.77 psid.

Figures 6.2-24 and 6.2-25 show the drywell and suppression chamber pressure and temperature response, respectively. The ECCS response is discussed in Section 6.3. Approximately 5 sec after the break occurs, air, steam, and water start to flow from the drywell to the suppression pool; the steam is condensed and the air enters the suppression chamber. The continual purging of drywell air to the suppression chamber results in a gradual pressurization of both the wetwell and drywell to about 29.8 and 34.6 psig, respectively. Some continuing containment pressurization occurs because of the gradual pool heatup.

The ECCS is initiated by a LOCA signal from the 0.1-sq ft break and provides emergency cooling of the core. The operation of these systems is such that the reactor is depressurized in approximately 600 sec. This terminates the blowdown phase of the transient. The drywell and suppression chamber peak pressures are predicted to be approximately 34.6 psig and approximately 29.8 psig, respectively.

In addition, the suppression pool temperature is the same as following the recirculation suction line rupture (DBA) because essentially the same amount of primary system energy is released during the blowdown. After reactor depressurization, the flow through the break changes to suppression pool water that is being injected into the RPV by the ECCS. This flow condenses the drywell steam and eventually causes the drywell and suppression chamber pressures to equalize in the same manner as following a recirculation suction line rupture (DBA). The subsequent long-term suppression pool and primary containment heatup transient that follows is essentially the same as for the recirculation suction line break (DBA). From this analysis, it is concluded that the consequences of an intermediate size break are less severe than those from a recirculation suction line rupture (DBA).

Small Breaks

This section discusses the primary containment transient response associated with small breaks. The RCPB ruptures in this category are those blowdowns that will not result in reactor depressurization either due to loss of reactor coolant or automatic operation of the ECCS equipment.

Reactor System Blowdown Considerations Following the occurrence of a break of this size, it is assumed that the Reactor Operators will initiate an orderly plant shutdown and depressurization of the reactor system. The thermodynamic process associated with the blowdown of primary system reactor coolant is one of constant enthalpy. If the primary system break is below the water level the blowdown flow will consist of reactor water. Blowdown from reactor pressure to the drywell pressure will flash approximately one-third of the water to steam and two-thirds will remain as liquid. Both phases will be at saturation conditions corresponding to the drywell pressure. Thus, if the drywell is at atmospheric pressure, the steam and liquid associated with a liquid blowdown will be at 212°F. Similarly if the primary containment is assumed to be at its design pressure of 45 psig, the reactor coolant will blow down to approximately 293°F steam and water.

If the primary system rupture is located so that the blowdown flow consists of reactor steam only, the resultant steam temperature in the containment is significantly higher than the temperature associated with liquid blowdown. This is because the

enthalpy of high energy saturated steam is nearly twice that of the saturated liquid.

Based upon this thermodynamic process, it is concluded that a small reactor steam leak will impose the most severe temperature conditions on the drywell structures and the safety-related equipment in the drywell. For larger steam line breaks, the temperature is nearly the same as for small breaks, but the duration of the high temperature condition is shorter. This is because the larger steam breaks will depressurize the reactor more rapidly than the orderly reactor shutdown that is assumed to terminate the small break.

Containment Response For drywell design consideration, the following sequence of events is assumed to occur. With the reactor and primary containment operating at the maximum normal conditions, a small break occurs that allows blowdown of reactor steam to the drywell. The resulting pressure increase in the drywell leads to a high drywell pressure signal that scrams the reactor and activates the primary containment isolation system. The drywell pressure continues to increase at a rate dependent on the size of the steam leak. This pressure increase lowers the water level in the vents until the level reaches the bottom of the vents. At this time, air and steam start to enter the suppression pool. The steam is condensed and the air is carried over to the suppression chamber. The air carryover results in a gradual pressurization of the suppression chamber at a rate dependent on the size of the steam leak. Once all the drywell air is carried over to the suppression chamber, the primary containment pressure very slowly increases due to suppression pool heatup. Figures 6.2-26 and 6.2-27 show the pressure and temperature transients of the drywell and suppression chamber, respectively.

Recovery Operation The Reactor Operators are alerted to the incident by the high drywell pressure signal and the reactor scram. It is assumed that their response will be to cool down the reactor in an orderly manner using the RHR heat exchangers or main condenser and limiting the reactor cooldown rate to 100°F/hr.

It is assumed that the Operator initiates the sprays and they become effective for reduction of containment pressure 20 min after the drywell reaches 30 psig. Vacuum breakers open and the pressure between the drywell and wetwell equalizes. When the suppression pool temperature reaches 185°F the Operator transfers the one RHR pump from the containment spray mode to the reactor shutdown cooling mode. An additional 16 min is provided for the Operators to complete this action, during which time it is assumed that no cooling takes place in the RHR heat exchanger.

The reactor primary system is completely depressurized in 6 hr. At this time, the blowdown flow to the drywell ceases and the primary containment pressure and temperature begin to reduce.

Drywell Environmental Design Temperature Considerations For drywell design purposes, it is assumed that there is a blowdown of reactor steam for the 6-hr cooldown period and the passive heat sinks are neglected. The corresponding design temperature is determined by finding the combination of RCS pressure and drywell pressure that produces the maximum calculated atmospheric temperature. The maximum drywell temperature of 325.8°F occurs when the RCS is at approximately 470 psia and the drywell pressure is 50.2 psia. As noted in Reference 8, the changes in reactor vessel conditions for power uprate will increase the expected peak drywell gas temperature during a LOCA by at most a few degrees, and thus will not exceed the design limit of 340°F. For the drywell structural design the maximum calculated temperature is 281.3°F, which is the saturation temperature corresponding to 50.2 psia drywell pressure.

Steam Bypass

In a pressure suppression type containment, steam released from the primary system following a postulated LOCA is condensed by the suppression pool and does not have an opportunity to produce a significant pressurization effect on the primary containment. This is accomplished by directing the steam into the suppression pool through connecting downcomer vents. This arrangement forces any steam released from the RCPB to be condensed in the suppression pool.
In the highly unlikely event of a reactor depressurization to the drywell accompanied by a simultaneous open bypass path between the drywell and suppression chamber, the leaking steam would significantly pressurize the containment. Therefore, the allowable bypass leakage is defined as the amount of steam that could bypass the suppression pool without exceeding the primary containment design pressure of 45 psig. The allowable bypass leakage is a function of the nature of the leakage path, the duration of the pressure differential across the leakage path, and the rate of condensation of leakage steam inside the suppression chamber.

The bypass area is defined in this analysis as the total flow area for leakage between the drywell and suppression chamber. Following a reactor system break, the air-steam mixture within the drywell passes through various leakage paths into the suppression chamber and causes pressurization of the suppression chamber. However, this analysis conservatively assumes that only steam leaks through the bypass area which results in the lowest bound on the allowable bypass area.

Possible leak paths are as follows:

1. Drywell floor seams.

2. Downcomer and SRV piping and penetrations.

3. Vacuum breakers.

4. Instrumentation test vents and air piping.

5. Floor and equipment drain piping and penetrations.

6. Concrete floor.

The allowable drywell bypass leakage capacity is expressed in terms of the parameter $A/\sqrt{K}$.

Where:

A = Flow area of leakage path, sq ft

K = Geometric and friction loss coefficient

This parameter is dependent only on the geometry of drywell leakage paths and is a convenient numerical definition of the overall drywell bypass leakage capacity. It results from a consideration of the flow process in the leakage paths. Assuming the steady-state noncompressible fluid flow theory (Bernoulli equation) to be applicable to the leakage flow, the pressure loss between the drywell and containment can be written:

$$PD - PW = \frac{KV^2}{2g_c v} \; 1/144 \; psid \qquad (6.2\text{-}1)$$

Where:

PD = Drywell pressure, psi

PW = Wetwell pressure, psi

K = Total geometric and friction loss coefficient of the flow between the drywell and wetwell. These losses include entrance, exit, discontinuities, and friction. The latter is somewhat dependent upon the Reynolds number of the fluid flow but for drywell leakage consideration, it is considered constant.

V = Velocity of flow, ft/sec

g_c = Conversion factor, 32.2 lbm-ft/lbf-sec^2

v = Specific volume of the fluid flowing in the leakage path, ft^3/lbm

If the bypass leakage path flow rate is M (lbm/sec) and the flow area is A (sq ft), the above equation can be rewritten to give:

$$\dot{M} = \frac{A}{\sqrt{K}} \sqrt{2g_c(PD - PW)144/v} \; lbm/sec$$

(6.2-2)

Thus, for a given drywell-to-wetwell pressure differential, the leakage flow (capacity) is dependent on $A/\sqrt{K}$ and the specific volume of the fluid flowing in the leakage path (which depends on the drywell internal pressure).

The purpose of the steam bypass analysis is to determine the leakage rate (in terms of bypass leakage capacity, $A/\sqrt{K}$) that would result in drywell pressurization to design pressure for the complete spectrum of line break sizes. The results of this analysis are summarized on Figure 6.2-28. This figure shows that allowable bypass leakage capacity ranges from approximately 0.061 sq ft for a large break to 0.069 sq ft for small steam line breaks.

The size of RCPB break determines the magnitude and duration of the pressure differential across the drywell leakage paths. When a very large break occurs, such as the DER of a main steam line, the high mass and energy flow from the RCPB pressurizes the drywell, generating a high pressure differential across the assumed leakage paths and producing high leakage flow rates. This short duration of reactor blowdown gives a large allowable bypass leakage capacity. When blowdown is over, the pressure differential across the leakage path dissipates and leakage flow and primary containment pressurization cease.

Small and intermediate breaks, on the other hand, result in slow RCPB depressurization. The reactor is scrammed due to the high drywell pressure resulting from the energy and mass released from the RCPB break.

During this period, the blowdown flow from the RCS forces the drywell air into the wetwell. The blowdown steam is condensed in the suppression pool, after the water level in the downcomer vents is depressed from the incoming steam and air. This results in an essentially continuous pressure differential between the drywell and suppression chamber of at least 4.76 psid. The allowable bypass leakage capacity for these conditions is an $A/\sqrt{K}$ of 0.05 sq ft.

The bypass leakage analysis is performed assuming that only steam leaks through the bypass paths. This assumption conservatively minimizes the allowable bypass leakage capacity by maximizing the primary containment pressurization from the assumed leakage. The results shown on Figure 6.2-28 are also based on the following assumptions and parameters shown in Table 6.2-52.

1. The pipe break, LOOP, and failure of Division II diesel generator occur at time zero.

2. Reactor coolant makeup is provided by high- and low-pressure ECCS pumps depending upon the reactor pressure and water level. Feedwater is unavailable due to the LOOP and unit trip.

3. To maximize steam flow from the reactor vessel through the pipe break, it is assumed that the Operator throttles the ECCS flows at 10 min after the accident. It is also assumed that the Operator does not initiate controlled reactor cooldown or actuate the automatic depressurization system (ADS).

4. For small breaks which do not depressurize the reactor, it is assumed that reactor pressure is maintained by automatic operation of the SRVs in the power-actuated relief mode according to the relief setpoints defined in Table 5.2-2.

5. The containment spray mode of the RHR system is assumed to operate at 30 min after the accident. The system flow details are shown in Table 6.2-52.

6. It is assumed that no heat or mass transfer takes place between the pool surface and the suppression chamber atmosphere.

7. Passive heat sinks, summarized in Tables 6.2-1 and 6.2-2, absorb energy from the drywell and suppression chamber. The UCHIDA heat transfer correlation is used and condensate film revaporization is limited to 8 percent.

The Unit 2 analysis is based on the manual initiation of containment sprays instead of automatic sprays as required by the Standard Review Plan (NUREG-0800). The containment spray system is QA Category I and classified as an ESF.

For the worst-case event of steam bypass, it has been determined that the Operator has 30 min following the accident to establish spray flow with one loop of the RHR system. The manual initiation of the containment spray can be accomplished in approximately 2 to 4 min considering the valve stroke times involved.

The containment transient (before and after Operator action) for the worst-case steam bypass event is shown on Figure 6.2-28A.

The rate of heat transfer to the passive heat sinks is dictated by the temperature difference between the containment atmosphere and the sink surface and by the nitrogen-to-steam mass ratio of the atmosphere. The surface heat transfer coefficient is determined for each heat sink as a function of the appropriate nitrogen-to-steam mass ratio from the UCHIDA correlation, which yields coefficients ranging from the minimum value of 2.0

Btu/hr-ft^2 °F to the maximum value of 280 Btu/hr-ft^2 °F. When the surface temperature of the heat sink (Ts) is greater than the saturation temperature of the atmosphere (Tg), the minimum UCHIDA coefficient is used. When Ts is less than Tg, the UCHIDA condensing heat transfer rate is compared to the convective heat transfer rate and the larger of the two is selected. The condensing heat transfer rate is given by:

$$q\ cond = huA\ (Tg - Ts)$$

The convective heat transfer rate is given by:

$$q\ conv = hcA\ (Ta - Ts)$$

Where:

hu = UCHIDA condensing heat transfer coefficient

hc = 2.0 Btu/hr-ft^2 °F for convection

A = Heat sink surface area

Ta = Drywell or suppression chamber atmosphere temperature

Tg = Saturation temperature of the drywell or suppression chamber atmosphere

Ts = Heat sink surface temperature

Under superheated drywell or suppression chamber atmosphere conditions, heat sink condensate is subject to revaporization due to convective heat gain from the superheated atmosphere. Partial revaporization limited to 8 percent of the condensate mass is assumed in this analysis.

The heat transfer coefficients for the primary containment wall (heat sinks 7 and 14 of Table 6.2-1) are shown on Figure 6.2-28B.

The spray drop thermal efficiency of 90 percent has been assumed in the steam bypass analysis. However, a sensitivity study with spray thermal efficiency reduced to 50 percent demonstrates that there is no limitation on bypass capability due to this assumption. Figure 6.2-28C shows that the drywell and suppression chamber peak pressures occur at the time of spray initiation with either 50 percent or 90 percent spray thermal efficiency.

A sensitivity study considering heat transfer from the drywell atmosphere into the suppression chamber atmosphere through the steel downcomer pipes is included in the bypass analysis. However, at the time of spray initiation (1800 sec), the downcomer metal is at 291°F compared to the spray water temperature of 113°F. Therefore, about 7.55 million Btus of sensible heat is stored in the downcomer pipes. In the worst

case, this energy could vaporize the suppression chamber spray water for about 4 min after spray actuation. The sensitivity study including this additional energy source shows that there is no additional limitation on steam bypass capability due to this heat source.

The steam bypass analytical model assumes that nitrogen purged from the drywell due to steam blowdown is forced through the downcomer vent system and enters the suppression chamber atmosphere as dry nitrogen at the suppression pool temperature. For the limiting steam bypass case, the majority of the drywell nitrogen mass is purged to the suppression chamber approximately 300 sec after the pipe break. In this time interval, pool temperature increases from 90°F to 99°F. A sensitivity study considering nitrogen saturated with vapor at pool temperature indicates that the dry nitrogen carryover assumption has negligible effect on the steam bypass capability. With saturated nitrogen carryover, peak containment pressure at 1,800 sec increased by approximately 0.28 psi.

Energy absorbed (integrated energy) by the containment heat sinks and sprays at various times is provided in Table 6.2-27A. The energy and mass balance is shown in Table 6.2-53.

Unit 2 utilizes all-welded construction to prevent bypass leakage between the drywell and the suppression pool air space.

To ensure that drywell bypass leakage conforms to the design basis, a leak test is conducted at the drywell-to-suppression chamber design pressure differential of 25 psid. The acceptance criterion for this test is that the measured leakage must be less than 10 percent of the bypass leakage capacity based on an $A/\sqrt{K}$ of 0.054 sq ft.

At the first refueling outage a leak test was conducted to verify bypass leakage. The schedule for subsequent bypass leakage tests is in accordance with the Technical Specifications.

At least once per 18 mo, a visual inspection shall be conducted of the exposed accessible interior and exterior surfaces of the suppression chamber (including each vacuum relief valve and associated piping).

Negative Primary Containment Differential Pressure

The maximum negative differential pressure for the primary containment results from the assumed inadvertent actuation of the containment spray system during normal plant operation with minimum spray water temperature and minimum air mass inside the containment. The term air mass is used for simplicity and refers to the nitrogen inerted atmosphere.

Containment depressurization following a postulated pipe break in the drywell is less severe than inadvertent spray actuation

because the suppression pool temperature, which dictates the final containment temperature and pressure, is increased due to blowdown steam condensation in the pool. In addition, the air mass inside the Mark II primary containment is constant during normal operation and the drywell floor vacuum breakers allow air to return to the drywell during post-LOCA drywell depressurization to limit the upward pressure differential on the drywell floor.

The following assumptions and analysis describe the worst-case negative containment pressure differential resulting from inadvertent spray actuation.

1. Air in the drywell and suppression chamber is assumed to be an ideal gas.

2. Vacuum breakers between the suppression chamber and drywell start to open at a differential pressure of 0.25 psid resulting in equalization of the pressure in the two compartments.

3. Initial conditions of minimum pressure (14.2 psia), maximum temperature (drywell at 150°F, wetwell at 122°F), and maximum relative humidity (drywell at 100%) are assumed to minimize the containment air mass.

4. The suppression pool is at the minimum temperature of 70°F and suppression chamber dew point of 70°F.

5. The service water which cools the RHR (containment spray) heat exchanger is assumed to be at the minimum temperature of 32°F.

6. The minimum spray temperature is determined as follows:

$$T_{sp} = T_p - \frac{K\left(T_p - T_{sw}\right)}{m_{sp} C_p} \qquad (6.2\text{-}3)$$

Where:

T_{sp} = Spray temperature

T_p = Pool temperature (70°F)

T_{sw} = Service water temperature (32°F)

m_{sp} = Spray flow rate (3.7×10^6 lbm/hr)

C_p = Specific heat of spray (1 Btu/lbm°F)

K = Heat exchanger performance factor (1,373,800 Btu/hr°F) (unfouled)

Therefore:

T_{sp} = 56°F

7. The drywell and suppression chamber pressure after spray actuation is determined assuming final temperature equal to the spray temperature.

8. The suppression chamber pressure after spray actuation is determined by adding the pressure differential across the vacuum breakers to the drywell pressure.

9. The final minimum containment pressure, considering pressure equalization between the drywell and wetwell due to the vacuum breakers, is calculated as follows:

$$P_{DW}^{F} = \frac{\left(\frac{T_{WW}^{F}}{T_{DW}^{I}}\right)\left(\frac{V_{DW}}{V_{WW}}\right) P_{DW,N}^{I} + \left(\frac{T_{WW}^{F}}{T_{DW}^{I}}\right) P_{WW,N}^{I} - \Delta P}{1 + \frac{V_{DW}}{V_{WW}}} + P_{VAPOR}^{F} \qquad (6.2\text{-}8)$$

Where:

N = Noncondensible

F = Final

I = Initial

WW = Suppression chamber

DW = Drywell

T = Temperature

P = Pressure

V_{DW} = Free volume of the drywell

V_{WW} = Free volume of the suppression chamber

ΔP = Vacuum breaker differential pressure between drywell and suppression chamber

and finally:

$P_{DW,MIN}$ = 10.18 psia

$P_{WW,MIN}$ = 10.68 psia

Assuming atmospheric pressure of 14.7 psia in the reactor building (outside containment), the maximum negative containment differential pressure is:

Maximum negative ΔP = 10.18 - 14.7
= -4.5 psid

Thus, an extremely conservative end-point analysis of this event predicts a maximum negative pressure differential of -4.5 psid compared to the design negative pressure differential of -4.7 psid.

Suppression Pool Dynamic Loads

The manner in which suppression pool dynamic loads resulting from postulated LOCAs, transients, and seismic events have been integrated into the Unit 2 design is described in the Unit 2 DAR for Hydrodynamic Loads (Appendix 6A).

Asymmetric Loading Conditions

The manner in which potential asymmetric loads were considered for Unit 2 is fully described in the DAR for Hydrodynamic Loads (Appendix 6A).

Containment Ventilation System

The primary containment ventilation system is discussed in Section 9.4.

Postaccident Monitoring

A description of the postaccident monitoring system is provided in Section 7.5.2.

Analytical Models

LOCA Containment Response Model For the original, pre-uprate analysis, the pressure and temperature response of the primary containment and the suppression pool temperature response following a LOCA in the RCPB are determined as functions of time with the LOCTVS computer program. Analytical models used for analysis of the limiting large line break for power uprate operating conditions are described in Section 6.2.1.1.5. The

THIS PAGE INTENTIONALLY BLANK

LOCTVS program performs numerical integrations principally of the mass and energy conservation equations, and also solves the momentum conservation equation as required to determine flow rates between nodes. LOCTVS simulates behavior of the pressure suppression containment system, the RCS, the primary containment heat sources and sinks, and the containment heat removal systems. SWECO 8101, a topical report, provides a detailed description of the analytical models used and the assumptions employed[1]. The analytical models and assumptions incorporated in the LOCTVS program to predict conservatively the response of the pressure suppression containment system are summarized in the following paragraphs.

The drywell is modeled as a constant volume system, initially containing a homogeneous mixture of air and water vapor. The total energy and mass of air, water vapor, and water inside the drywell are determined at all times by numerical integration of the appropriate flows. The flows included in the LOCTVS drywell model are the reactor blowdown, the vent flow, and the heat transfer to the primary containment structures.

Blowdown from the RCPB causes the drywell pressure to increase and the level of water in the downcomer to be depressed until these are cleared. A mixture of air, steam, and water is then forced through the downcomers into the suppression pool. The water and condensed steam are added to the suppression pool inventory while the air from the drywell is added to the suppression chamber, increasing the suppression chamber pressure and the pressure at the downcomer exit. The drywell pressure is dictated by the downcomer exit pressure, the dynamic pressure losses associated with a given downcomer flow, the drywell free volume, and the blowdown flow rate. For calculation of flow through the downcomers into the suppression pool, a homogeneous flow model is employed. This model, in combination with the frictionless Moody flow model used for blowdown calculations, provides a computed flow rate into the drywell that is large and a flow rate out of the drywell that is small; thus, the rate of storage in the drywell is large and the pressure calculated is conservatively high.

The thermodynamic state of the drywell atmosphere following a recirculation suction line break (DBA) is assumed to be always saturated. The break effluent is homogeneously mixed with the atmosphere and undergoes a pressure flash, with subsequent dropout of the decompressed and unflashed liquid.

In determining the state of the drywell (i.e., pressure and temperature) for a steam line break, the steam mass and enthalpy are added to the drywell atmosphere. Initially, the blowdown consists of saturated steam, but as the reactor vessel pressure drops the water level in the vessel swells. Eventually the froth level reaches the top of the steam dryers, and the blowdown is assumed to change from steam to a two-phase mixture.

At the time the two-phase blowdown starts, the drywell contains superheated steam and air. If instantaneous and complete mixing between the steam, air, and two-phase mixture is assumed, drywell temperature and pressure decrease. This decrease results from heat being extracted from the high temperature steam and air to evaporate the liquid phase of the blowdown. For the sake of conservatism, the assumption of instantaneous homogeneous mixing is modified for the period immediately following the start of two-phase flow until after peak drywell pressure and temperature occur (except for downcomer flow calculations where, again to be conservative, a homogeneous drywell mixture is always assumed).

The modification of the homogeneous mixing assumption is as follows. The steam portion of the blowdown goes directly to the drywell atmosphere. The liquid portion undergoes a pressure flash, where the flashed steam goes directly to the drywell atmosphere and the decompressed liquid falls to the drywell floor. Saturated steam resulting directly from the blowdown and the flashing process mixes uniformly with the existing drywell air and steam.

The mode of heat transfer to the drywell and primary containment heat sinks is determined from the individual heat sink surface temperature and the saturation temperature of the atmosphere adjoining the heat sink surface. The criteria used to determine the heat transfer mode are described in Section 8.3 of the LOCTVS topical report[1]. The condensing heat transfer coefficient is determined as a function of the air-steam mass ratio of the adjoining volume using the UCHIDA correlation. Accordingly, the heat transfer coefficient used by LOCTVS varies from a maximum value of 280 Btu/hr-sq ft-°F to a minimum value of 2 Btu/hr-sq ft-°F, as shown on Figure 6.2-29.

LOCTVS incorporates models for downcomer clearing, downcomer flow, suppression pool swell, and suppression chamber pressurization. These models are similar to those of the GE Mark II analytical model described in NEDM-10320. The Mark II version of LOCTVS has shown excellent agreement with the analytical results presented in NEDM-10320. LOCTVS results have also been favorably compared to the Pressure Suppression Test Facility (PSTF) test results.

In this comparison the PSTF geometry and conditions were simulated using LOCTVS for two of the test cases - one with liquid blowdown and one with steam. The blowdown was released in the drywell, thereby producing high-energy pipe break conditions which cause vent clearing and pool swell in the suppression chamber.

The blowdown was calculated based on Moody flow through a venturi of given diameter. This simulated the test condition which actually controlled blowdown using a critical flow venturi. Other test parameters were duplicated such as vent submergence, vent pipe diameter, and drywell/suppression pool initial

temperature. The LOCTVS model was then executed for one complete cycle of pool swell and fallback. The LOCTVS results conservatively envelope the PSTF test data as shown in Figures 6.2-29A through 6.2-29J.

In general, primary containment peak pressure is sensitive only to those variables that affect the long-term analysis, such as initial suppression pool temperature, decay heat rate, RHR heat exchanger heat transfer coefficient, and passive heat sink area (Table 6.2-2). Drywell floor peak pressure differential is sensitive only to those variables that affect the short-term analysis, such as downcomer area, submergence, air carryover rate, blowdown flow rate, and RPV water level swell time. The sensitivity analysis results are provided in Tables 6.2-17 through 6.2-27.

Models for Analysis of Intermediate and Small Breaks The CONSBA computer program has been utilized for the analysis of intermediate and small breaks. This program calculates the thermodynamic response of BWR primary containment and uses a finite difference technique based on input-specified time steps to solve the transient equations. In each time step, the program determines the mass and energy flow across all control volumes and performs state calculations for the reactor vessel, suppression pool, drywell, suppression chamber atmosphere, and water on the drywell floor assuming equilibrium conditions.

CONSBA has the capability to follow an input-specified reactor cooldown rate by controlling the opening and closing of valves in the SRV system. Five groups of manually-actuated SRVs are included in the model and available to discharge steam from the reactor vessel to the suppression pool, thus allowing the plant Operators to cool down the reactor vessel. At the user's option the program will calculate the integrated mass and energy flow of each compartment and system and print a mass and energy balance.

Ten groups of automatically-actuated SRVs are included in the model and are available to discharge steam from the reactor vessel to the suppression pool, thus relieving the pressure in the reactor vessel. Valves open when reactor vessel pressure reaches a specified setpoint and remain open until the pressure drops below a specified setpoint. The flow through the SRV is assumed to be frictionless Moody flow at all times and the assumed enthalpy is that of the reactor vessel steam.

The reactor coolant is assumed to be in a saturated equilibrium condition. In the case of a pipe break in the drywell, either steam or water will be discharged from the reactor vessel depending on the location of the break and water level in the reactor vessel. Frictionless Moody critical flow tables are interpolated to determine the blowdown mass flux as a function of reactor vessel pressure and for either reactor coolant saturated liquid or steam enthalpy, depending on the type of the break

specified. Blowdown flow becomes zero when reactor vessel pressure drops below the drywell pressure.

The drywell atmosphere is either a homogeneous mixture of air and steam or all steam. The suppression chamber atmosphere is always a homogeneous mixture of air and steam.

Downcomers and vacuum breakers are modeled in the program to relieve pressure buildup in the drywell or suppression chamber. Since this program is written for small or intermediate breaks, the pressure buildup in the drywell is expected to be significantly slower than the large breaks. The dynamic clearing at the downcomer is not analyzed in the program. The hydrostatic pressure at the downcomer discharge is calculated and compared with the pressure differential between the drywell and suppression chamber to determine if there is any vent flow. Flow through the downcomer is a homogeneous mixture of air and steam.

Water in excess of an input-specified maximum limit on the drywell floor is assumed to overflow through the downcomer system and is added to the suppression pool inventory in the next time step. The suppression pool and water accumulated on the drywell floor are assumed to be saturated water. The vacuum breakers will be open if the pressure differential between the suppression chamber and drywell is greater than the input vacuum breaker pressure setpoint. Vacuum breaker flow is assumed to be a homogeneous steam and air mixture.

Heat sinks are modeled in CONSBA to determine the amount of energy absorbed by various structures of the primary containment. Concrete and steel structures (passive heat sinks) may absorb energy from the local environment after an accident due to elevated temperature in the drywell, suppression chamber, and suppression pool.

CONSBA models the ECCS and heat removal through the RHR heat exchangers. Unit 2 has two RHR heat exchanger loops. Both loops can be used for pool cooling; only one can be used for reactor shutdown cooling at a given time. Pool cooling mode is achieved by pumping suppression pool water through the heat exchanger and discharging it back to the pool. Shutdown cooling is achieved by recirculating the reactor vessel coolant through the RHR heat exchanger.

CONSBA also determines the steam or air/steam leakage rate from the drywell to the suppression chamber atmosphere bypassing the suppression pool. The Darcy equation is used to calculate this leakage.

6.2.1.1.4 Sensitivity of Suppression Chamber Air Space Temperature Increase on LOCAs

The large break LOCA analysis described in Section 6.2.1.1.3 is done using 90°F suppression chamber air space temperature. The

maximum allowed temperature is 122°F. This change results in a reduction of initial suppression chamber air mass and a small decrease in the maximum calculated drywell pressure. Thus, use of 90°F is conservative.

6.2.1.1.5 Impact of Power Uprate on Large Break Containment Response Analysis

Section 6.2.1.1.3 provides the results of the original analyses of the Unit 2 containment response to various postulated accidents that constitute the design basis for the containment, based on operation at the original rated core thermal power of 3,323 MWt. Operation with power uprate changes some of the conditions for the large break analyses. For example, the short-term DBA LOCA containment response during the blowdown is governed by the blowdown flow rate. This blowdown flow rate is dependent on the reactor initial thermal-hydraulic conditions, such as vessel dome pressure and the mass and energy of the vessel fluid inventory, which change slightly with power uprate. Also, the long-term heatup of the suppression pool following a LOCA is governed by the ability of the RHR system to remove decay heat. Since the decay heat depends on the initial reactor power level, the long-term containment response is affected by power uprate. The Unit 2 containment pressure and temperature response has been reanalyzed to demonstrate the plant's capability to operate with a rated power increase to 3,467 MWt. These analyses use General Electric Company (GE) codes and models (References 6 and 7) and ANS 5.1-1979 decay heat assumptions. The input assumptions are consistent with those used for the original containment analyses, as described in Section 6.2.1.1.3.

Short-Term Accident Response - DBA LOCA

Short-term containment response analyses are performed for the limiting DBA LOCA (a double-ended guillotine break of a recirculation suction line) to demonstrate that operation with power uprate will not result in exceeding the containment design limits. The short-term analysis covers the blowdown period during which the maximum drywell pressure, wetwell pressure and differential pressure between the drywell and wetwell occur. These analyses are performed at 102 percent of the uprated power level. The results of these short-term analyses are summarized in Table 6.2-4. As shown by these results, the maximum pressure values with power uprate are bounded by the original analysis values and by the design pressures.

Analysis of the short-term containment response was also performed for power uprate operating conditions using the original analysis methods (i.e., LOCTVS computer code) and assumptions described in Section 6.2.1.1.3 to confirm the expected minor impact of power uprate. This analysis confirmed that power uprate results in peak values for drywell pressure, suppression chamber pressure, and drywell floor differential

downward pressure that are slightly higher than the original analysis values but continue to be less than design values.

The drywell design temperature (340°F) has been determined based on a bounding analysis of the superheated gas temperature which can be reached with blowdown of steam to the drywell during a LOCA. The short-term peak drywell temperature is controlled by the initial steam flow rate during a large steam line break. Since the vessel dome pressure assumed for the original containment analysis (1,055 psia) is unchanged by power uprate, the initial break flow rate for this event will be unchanged. Therefore, there is no change to the original analysis short-term peak drywell temperature value of 303°F.

The wetwell gas space peak temperature response is calculated assuming thermal equilibrium between the pool and wetwell gas space. As discussed below, the bulk pool temperature will increase by approximately 2°F with power uprate. Therefore, the wetwell gas space will also increase by approximately 2°F. This increase results in wetwell gas space temperatures which are well below the design temperature of 270°F.

Long-Term Accident Response - DBA LOCA

The long-term bulk pool temperature response with power uprate is evaluated with the GE containment model for the limiting DBA LOCA, identified in Section 6.2.1.1.3 as Case C. The analysis is performed at 102 percent of the uprated power, and uses current values for the RHR heat exchanger coefficient (240.2 Btu/°F-sec) and service water temperature (82°F). Table 6.2-4 summarizes the analysis results. The peak bulk suppression pool temperature calculated with the uprated power is 207.9°F. This temperature is approximately 1°F higher than the originally calculated value but is within the design value of 212°F.

6.2.1.2 Containment Subcompartments

6.2.1.2.1 Design Bases

The drywell subcompartments are designed in accordance with the following criteria:

1. A pressure response analysis is given for each primary containment subcompartment containing high-energy piping in which breaks are postulated. The definition of high-energy piping and the criteria for postulating breaks are outlined in Section 3.6A.

   The break selected for the design evaluation produced, by virtue of its size and location, the greatest release of blowdown mass and energy into the subcompartment, during normal operation and hot standby condition. The breaks used in the design evaluations are listed in Section 6.2.1.2.3.

2. All circumferential breaks are considered to be fully double ended and no credit is taken for limiting blowdown generation due to pipe restraint locations. The effective cross-sectional flow area of the pipe is used in the jet discharge evaluation for breaks.

3. The suppression chamber and suppression pool are assumed not available to relieve pressure from the drywell region.

4. No heat sink credit is taken.

5. The design pressure differentials for all subcompartments are higher than the calculated peak pressure differentials resulting from the postulated pipe breaks.

6. No credit is taken for blowout panels in the subcompartment analyses.

6.2.1.2.2 Design Features

For the most part, the drywell is a large continuous volume interrupted at various locations by piping, grating, ventilation ducting, etc. Two volumes within the drywell classified as subcompartments are:

1. Reactor Pressure Vessel (RPV) - Biological Shield Wall (BSW) Annulus The 1 ft 8 1/2-in thick cylindrical primary shield wall surrounds the RPV. It has an outside radius of 15 ft 9 1/4 in and extends from the reactor pedestal elevation to el 314 ft 1 1/2 in. Breaks in the recirculation water discharge and suction piping, LPCI piping, LPCS piping, and feedwater piping are analyzed. Venting occurs through the top of this annular region to the drywell and also through the flow diverter doors on the recirculation suction lines for the case of a double-ended rupture of a recirculation suction line.

2. Drywell Head The drywell head surrounds the RPV head. The detachable portion connects to the refueling bulkhead (Figure 6.2-30) at el 329 ft 7 1/8 in. The vent area supplied through the refueling bulkhead consists of two ventilation exhaust openings at azimuths 45 and 285 deg and four annular vent areas at azimuths 105, 165, 225, and 345 deg (Figure 6.2-30A). All vent areas are normally open and are closed only during refueling.

   The ductwork that extends up to or penetrates the refueling bulkhead openings is not considered available vent area. However, the open annular areas around the ductwork are considered available as open vent area.

Breaks are postulated in the RCIC head spray line and the recirculation suction piping.

Drawings depicting piping, equipment, and compartment/venting locations are provided in Section 3.6A. The volumes and vent areas are discussed in Section 6.2.1.2.3. The subcompartments described do not incorporate blowout panels. No credit is taken for vent areas that become available after the pipe break occurs.

6.2.1.2.3 Design Evaluation

The evaluations described in this section are based on original rated design conditions. The impact of power uprate is addressed in Section 6.2.1.2.5.

The breaks utilized in the design evaluation of the primary containment subcompartments are listed in Table 6.2-28. The tables and figures that contain the nodal parameters and results for each analysis are also listed in Table 6.2-28.

The primary containment subcompartment design evaluations were performed with the SWEC THREED computer code. THREED considers two-phase, two-component (steam-water-air) flow through the vents and accounts for the fluid inertia effects. A description of the THREED analytical model is provided in Appendix 6B.

The blowdown mass and energy releases for each of the breaks are provided in the tables that are cross-referenced in Table 6.2-28. For all cases, the blowdown data are based upon conservative methodology developed by GE using the Moody steady-slip flow model with subcooling[3].

The assumed initial conditions for the subcompartment volumes are conservatively chosen to maximize transient differential pressure responses. The initial conditions are given in the subcompartment nodal description tables. The minimum relative humidity of 20 percent assumed for several of the subcompartment analyses is based on meteorological data. The review of 2 yr of these data indicates that the relative humidity of the environment was at or above 20 percent for 99.85 percent of the time. However, a sensitivity run, assuming zero percent relative humidity, for the case of a 12-in feedwater line break in the RPV-BSW annulus, showed a pressure increase of only 0.2 psi in the break node, and a maximum pressure increase of 0.85 psi adjacent nodes.

The piping systems assumed to rupture in the subcompartments are identified in Table 6.2-28. Break locations are discussed in Section 3.6A.

The description of and justification for the subsonic and sonic flow models, and the degree of entrainment used in vent flow calculations are given in Appendix 6B.

The subcompartment nodalization schemes are tabulated and provided. The nodalization schemes are selected to maximize pressure differentials across node boundaries. Restrictions resulting from structural components or equipment placement were selected as nodal boundaries for the flow models.

RPV-BSW Annulus Models

RPV-BSW annulus volumes modeled are shown on Figures 6.2-52, 6.2-58, 6.2-61, 6.2-63, and 6.2-65. A 20-node model was used for the recirculation suction line break, the recirculation discharge line break, and the LPCI line break, and a 21-node model was used for the feedwater line break and the LPCS line break. Only half of the annulus is modeled in each case due to symmetry about the location of the rupture. Nodes close to the break are made comparatively small with respect to the other nodes since the pressure gradient is greater close to the break node.

Flow diverters are incorporated in the BSW penetration sleeves for the two recirculation water suction lines. These diverters help to minimize the asymmetric loads resulting from the pressurization of the RPV-BSW annulus. Flow diverters are not used for the recirculation water discharge lines, feedwater lines, LPCI lines, or the LPCS line.

No credit is taken for any penetrations through the BSW, other than flow diverters described earlier, that might allow additional venting out of the annulus. All mass flow from the RPV-BSW annulus is required to vent around the stabilizers on top of the BSW. When the blowdown occurs, pressure differentials develop across the BSW. When the mass flow leaves the top of the annulus and begins to enter the drywell, the pressure differentials decrease to values well below the peaks. Vent paths where choked flow occurs are indicated in the vent path description tables for each analysis. During the blowdown the RPV insulation panels are assumed to displace toward the BSW. The insulation is considered to be incompressible.

Flow around pipes and nozzles is modeled as flow encountering sudden contractions and expansions in the flow area. Pipes obstructing flow are projected onto the next junction.

Drywell Head Analysis

The drywell head analysis considers pipe breaks that produce upward and downward loadings on the refueling bulkhead. Downward pressure on the refueling bulkhead is determined by the RCIC head spray line case. Upward pressure values on the refueling bulkhead are based on a recirculation suction line break. The ductwork is assumed to stay in place for the duration of the

analysis. No flow is considered through the ducts. The volume between the RPV head and the surrounding insulation is considered available during the transient and the insulation is assumed incompressible.

Flow Coefficients

The flow coefficient (C) for a particular geometry is determined as a function of the equivalent head loss coefficient (K_{eff}) for that flow system as follows:

$$C = \frac{1}{\sqrt{K_{eff}}} \qquad (6.2\text{-}11)$$

The value of K_{eff} is simply the sum of the head losses for separate parts of the system. These head losses are defined as follows:

1. Entrance Loss or Contraction Determined as a function of the ratio of the upstream cross-sectional area to the cross-sectional area of the contraction.

2. Exit Loss or Expansion Determined as a function of the ratio of the cross-sectional area upstream of the expansion to the cross-sectional area of the expansion.

3. Resistance of Bends to Flow of Fluid Determined by the angle and length of the bend.

4. Friction Loss Although generally very small, it is calculated as an fl/d term, where f is the nondimensional friction coefficient, l is the junction length, and d is the cross-sectional hydraulic diameter.

5. Form Losses They are due to objects in the flow path, such as grating, and included in the vent path description tables with the friction losses. The formulas in APED-4378[2] are used to calculate the form losses.

The previously-listed losses are defined specifically in APED-4378[2] and NEDO-24548[3]. Values for the respective components are listed in the vent path description table for each break analyzed.

The pressure response graphs of nodes within each subcompartment model are provided with the figure numbers listed in Table 6.2-28.

Sensitivity Studies

Force and moment time-histories are calculated for all five postulated pipe breaks in the RPV-BSW annulus. The feedwater line DER produces the greatest moment on the BSW. The recirculation inlet DER produces the greatest forces on the BSW along the break axis.

A force and moment sensitivity study has been done on the feedwater line DER nodalization model. This nodalization model has been expanded from 21 to 37 nodes to determine the effect additional nodes have on the force and moment values. The 37-node model is shown on Figure 6.2-55. Nodal and vent path descriptions are shown on Tables 6.2-38 and 6.2-39, Sheets 1 through 6. The blowdown mass and energy release for the 21- and 37-node models are identical (Table 6.2-37).

Peak forces and moments in the 37-node model are approximately 9 percent less than those calculated in the 21-node model. This is due to the difference in the configuration of the two models. The 21-node model projects the areas of pipes within a node onto the nodal boundaries. This produces smaller and more conservative junction areas and restricts flow to a greater extent. In the 37-node model, finer meshing reduces the need to project pipe areas.

Force and moment data and results are provided with the figure and table numbers of Table 6.2-46.

The 37-node feedwater line DER model produces a peak nodal pressure differential of 34.83 psid versus 28.62 psid for the 21-node model. Nodal pressure and pressure differential responses are shown on Figures 6.2-56, Sheets 1 through 10, and Figure 6.2-57.

No sensitivity study has been done for the drywell head subcompartment because it is an open hemispherical volume.

6.2.1.2.4 Asymmetric LOCA Loads

The guidance of NUREG-0609 is used to analyze asymmetric LOCA loads. The following is a brief description of the methodology:

1. Pressure-Time Histories

   The pressure-time histories in the annulus region between the RPV and shield wall are generated from feedwater, LPCI, LPCS, and recirculation line breaks. The SWEC THREED code, which uses nodalized mass and energy balance, is applied in this analysis.

2. Concentrated Force-Time Histories

The forcing function of jet impingement on the shield wall is obtained from the break flow transient caused by these line breaks. Forcing functions of jet reaction on the RPV, jet impingement on the RPV, and the pipe whip restraint load on restraint anchors are obtained from these line breaks.

3. Integrated Dynamic Analyses

Beam models are used with the pressure-time and concentrated force-time histories to determine the effects on the shield wall, pedestal, vessel support skirt, core support and internals, and control rod drives (CRDs). These dynamic analyses yield displacements, accelerations, forces, stresses, and moments.

4. Attached Piping Analysis

Acceleration-time history from the integrated dynamic analysis is used to generate response spectra for the stress analysis of the attached piping. This analysis covers ECCS lines, primary coolant piping, and associated piping supports.

5. Load Combination for Vessel and Piping

Asymmetric LOCA loads, combined with SSE by the square root of the sum of the squares (SRSS) methodology, are treated as a faulted condition for evaluation versus the ASME Code. Load combinations are given in Tables 3.9A-2 and 3.9B-2.

6.2.1.2.5 Impact of Power Uprate on Subcompartment Pressurization and Annulus Pressurization Evaluations

Peak subcompartment pressures occur very quickly (during the first 2 sec) for the limiting subcompartment pressurization events identified in Section 6.2.1.2.3. Therefore, the pressurization is controlled by the initial break flow rates, which are governed by the initial reactor thermal-hydraulic conditions such as reactor pressure. The values used for the power uprate evaluation at 102 percent of the uprated power are not significantly changed from the values used in the analyses described in Section 6.2.1.2.3 (at 104.3 percent of the original rated power). Therefore, the subcompartment pressurization loads are not significantly affected by power uprate.

6.2.1.3 Mass and Energy Release for Postulated Loss-of-Coolant Accidents

The original containment analyses use the LOCTVS computer program to calculate the mass and energy released following postulated large break LOCAs, as described in the following subsections. LOCTVS also calculates the containment system response as described in Section 6.2.1.1.3, Short-Term Accident Response. For a description of the mass and energy release models incorporated by LOCTVS, see SWECO 8101[1].

Methods used for power uprate containment response analyses are addressed in Section 6.2.1.1.5.

6.2.1.3.1 Mass and Energy Release Data

Table 6.2-7 provides the mass and enthalpy release data for the recirculation suction line DER. Figures 6.2-34 and 6.2-35 show the blowdown flow rates and enthalpy for the recirculation line break. Table 6.2-50 provides the mass and enthalpy release data for the main steam line break. Figures 6.2-36 and 6.2-37 show the vessel blowdown flow rates and enthalpy for the main steam line break as a function of time after the postulated rupture.

6.2.1.3.2 Energy Sources

The RCPB conditions prior to the DER of recirculation suction line are presented in Table 6.2-9. Reactor blowdown calculations for containment response analyses are based on these conditions during a LOCA.

Following each postulated accident, the stored energy in the RCPB and the energy generated by fission product decay will be released. The rate of release of core decay heat for the evaluation of the primary containment response to a LOCA is provided in Table 6.2-10. Following a LOCA, the sensible energy stored in the RCPB metal will be transferred to the recirculating ECCS water and will thus contribute to suppression pool and primary containment heatup. Table 6.2-12 shows the reactor system metal and core-stored energy release rate to the reactor coolant following a recirculation line break.

6.2.1.3.3 Effects of Metal-Water Reaction

The primary containment systems are designed to accommodate the effects of metal-water reactions and other chemical reactions following a postulated DBA. The amount of metal-water reaction considered in the LOCTVS blowdown data is 0.7 percent which is consistent with the performance objectives of the ECCS assumed to occur in the first 2 min following the pipe break (Table 6.2-13). The above metal-water reaction value is 5 times that calculated in accordance with 10CFR50 Appendix K.

6.2.1.4 Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures Inside Containment (PWR)

This is not applicable.

6.2.1.5 Minimum Containment Pressure Analysis for Performance Capability Studies on Emergency Core Cooling System (PWR)

This is not applicable.

6.2.1.6 Testing and Inspection

Containment testing and inspection programs to verify the structural adequacy of the primary containment are described in Section 3.8.1.7. The tests for verifying that the containment and drywell leakage rates are within allowable limits are described in Section 6.2.6.

6.2.1.7 Instrumentation Requirements

Description

Safety-related instruments and controls are provided for automatic and manual control of the containment atmosphere monitoring system (CMS). The controls and monitors described below are located in the main control room. The control logic is shown on Figure 6.2-38.

Additional requirements for containment atmosphere monitoring will be provided in accordance with Section 1.10, Task II.F.1.

Operation

The CMS monitors and/or samples the following parameters:

1. Primary containment pressure.
2. Primary containment atmosphere and suppression pool water temperatures.
3. Primary containment hydrogen and oxygen.
4. Primary containment particulate radiation.
5. Primary containment gaseous radiation.
6. Primary containment radioactive iodine (sample collection only).
7. Suppression chamber pressure.
8. Suppression pool level.

The CMS can continuously sample the hydrogen/oxygen content of the primary containment by drawing samples from five different areas: three from the drywell and two from the suppression chamber. The sampling point can be automatically switched by a cycle timer for the drywell samples or manually selected for both the drywell and suppression chamber areas. Sampling consists of continuous drawing, analyzing, and returning the sample to the area from which it was drawn.

The hydrogen-oxygen analyzer is controlled manually. An interlock is provided to automatically stop the analyzer, or prevent it from starting, when either of the hydrogen-oxygen analyzer instrument isolation valves is closed or when a LOCA or manual isolation signal is present. The CMS primary containment isolation valves close automatically on receipt of manual isolation or LOCA signal. The valves can also be controlled manually. The LOCA signal can be overridden by a keylock switch. The CMS drywell inboard sampling valves are controlled automatically by the cycle timer and the sample path selector. The valves can also be operated manually.

Monitoring

Indicators are provided for:

1. Primary containment hydrogen-oxygen concentration.
2. Drywell pressure (normal and wide ranges).
3. Suppression chamber pressure (normal and wide ranges).
4. Suppression pool level (normal and wide ranges).
5. Drywell temperature.
6. Suppression chamber temperature.
7. Suppression pool water temperature.
8. Drywell atmosphere train leakage gaseous radiation level.
9. Drywell atmosphere train leakage particulate radiation level.

Recorders are provided for:

1. Primary containment hydrogen-oxygen concentration.
2. Drywell pressure (normal and wide ranges).
3. Suppression chamber pressure.
4. Suppression pool level.

5. Drywell temperature.

6. Suppression chamber temperature.

7. Suppression pool water temperature.

8. Drywell atmosphere train leakage gaseous radiation level.

9. Drywell atmosphere train leakage particulate radiation level.

Alarms are provided for each CMS division:

1. Primary containment hydrogen-oxygen concentration high.

2. Drywell pressure high.

3. Drywell temperature high.

4. CMS primary containment isolation valve inoperable.

5. CMS LOCA override.

6. Suppression pool water temperature high.

7. Process airborne radiation monitor activated.

6.2.2 Containment Heat Removal System

The containment heat removal function is accomplished by either the suppression pool cooling or the containment spray mode of the RHR system. Containment spray consists of two independent spray headers in the drywell and one common spray header in the suppression chamber. Containment spray water is discharged through spray nozzles by the RHR pumps and cooled by the RHR heat exchangers.

No credit is taken for containment spray heat removal or fission product control following a large break accident. However, the sprays are necessary to limit the primary containment pressure following a small or intermediate steam line break considering steam bypass factor of 0.05 sq ft (A/√K).

Systems that control fission products in order to reduce the concentration and quantity of fission products released to the environment are discussed in Section 6.5.

6.2.2.1 Design Bases

Systems utilized for postaccident containment heat removal meet the following safety design bases:

1. Systems are designed to limit the long-term bulk temperature of the suppression pool water to 212°F when considering various sources of energy addition to the containment following a LOCA (Section 6.2.1.1).

2. Systems are designed so that no single failure results in loss of the safety function.

3. Systems are qualified for the environmental conditions imposed by a LOCA (Section 3.11).

4. Systems are designed to safety-related requirements including the capability to perform their functions following a SSE. Systems are designed as Category I and Safety Class 2.

5. Systems are designed to withstand dynamic loads resulting from suppression pool hydrodynamic conditions. These various hydrodynamic loads are fully described in the Unit 2 DAR for Hydrodynamic Loads (Appendix 6A).

6. Each active component of the systems is capable of being periodically inspected and tested during plant operation.

6.2.2.2 System Design

When the RHR system is in the containment spray mode, the RHR pumps draw water from the suppression pool, pass it through the RHR heat exchangers, and inject it into the spray headers. When the RHR system is in the pool cooling mode, the RHR pumps draw water from the suppression pool, pass it through the RHR heat exchangers, and return it to the suppression pool. Cooling water from the service water system is pumped through the heat exchanger tubes to exchange heat with the suppression pool water. Two cooling loops are provided, each mechanically and electrically separate from the other to achieve redundancy. The process diagram, including process data (Figure 5.4-14), covers all operating modes and conditions.

RHR heat exchanger details are listed in Table 6.2-6. Figure 6.2-39 shows the schematic of the containment spray system. Codes and standards used for design purposes are addressed in Section 5.4.7. Environmental qualification of the containment heat removal systems is discussed in Section 3.11.

The RHR pumps are automatically initiated in the LPCI mode from diverse signals as described in Section 7.3.1.1.1.4. The LOCA analysis assumed uninterrupted LPCI flow to the reactor core during the first 10 min of the accident. In accordance with emergency operating procedures (EOPs), a RHR pump may be realigned from the LPCI mode to the containment spray or suppression pool cooling mode of operation independent of the 10

min elapsed time criteria assumed in the LOCA analysis. The approach utilized in the EOPs is acceptable as these procedures contain adequate cautions to deter the Operator from premature flow diversion of RHR in the LPCI mode of operation. The guidance provided in EOPs assures that the peak clad temperature (PCT) identified by the Updated Safety Analysis Report (USAR) LOCA analyses is not increased due to Operator realignment of the RHR system during an accident. Therefore, EOPs may result in diversion of RHR flow from the core cooling function prior to 10 min into the accident; however, such action will not increase the PCT of the fuel. The Operator has 30 min after a LOCA to establish the containment spray cooling mode.

In the event that a single failure occurs and the procedure the Operator is following does not result in system initiation, the Operator places the other totally redundant loop into operation by following the same initiation procedure. The Operator would require about 5 min to change over from the LPCI mode of the RHR system to the containment spray cooling mode of the RHR system. The RHR heat exchangers, the service water pump, and the valves are remote manually operated. The RHR pool cooling suction and discharge arrangement is shown on Figure 6.2-87. All suction and discharge points are located near the pool wall with suction taken approximately 14 ft above the pool floor and discharge at 22 ft above the floor, vertically downward. Loop A suction and discharge points are separated circumferentially by 140 deg of arch length (80-ft chord length) and loop B suction and discharge are also approximately 140 deg apart. This particular arrangement will promote the natural circulation and mixing between the discharged water and the suppression pool water. This arrangement precludes suction and discharge interaction when only one RHR loop is operating in the pool cooling mode. It may be noted that one loop operating is the worst case for maximum suppression pool temperature.

Each RHR pump takes suction directly from the suppression pool through a strainer that prevents foreign objects in the suppression pool from entering the ECCS and spray system flow path. The RHR suction strainers are approximately 8 ft below the minimum drawdown water level of the suppression pool. The ECCS suction strainers, including the RHR suction strainers, are all designed to prevent the passage of any particle larger than 1/4 in long and 3/32 in in diameter (Figure 6.2-39a). With the exception of the orifices located in the seal water inlet, piping to the ECCS pumps' cyclone separators, this particle size is smaller than any portion of the ECCS flow path, as shown in the following dimensions:

ECCS pump seal water orifices	1/8-in diameter
Drywell spray nozzle orifices	1 1/64-in diameter
Suppression pool spray nozzle orifices	23/64-in diameter
Minimum core spray internal diameter	5/8-in
Minimum fuel channel spacing	4/10-in

The orifice assemblies in the seal water piping to the ECCS pump cyclone separators all have a bore diameter of 1/8 in. Although the maximum theoretical particle size that can pass through the suction strainers is larger than this orifice size, blockage of these orifices is not considered to be a credible event.

In the unlikely event that a 1/4-in by 3/32-in cylindrical particle or sliver was generated during a LOCA event, transported to the suppression pool and migrated to one of the suction strainers, the particle would still have to traverse a tortuous path to pass through the strainer (see Figure 6.2-39a). The particle would first have to enter the external chamber through the 1/4-in exterior slot in an exactly perpendicular alignment with respect to the surface of the perforated disc. Then the particle would have to align itself directly over one of the 3/32-in perforations and pass perfectly through the perforation into the internal chamber and then into the interior of the strainer.

Once through the strainer, the particle would pass through the piping to the pump. The pump seal piping, which takes suction from the pump discharge, is 3/4-in piping that is designed to pass only 2 1/2 gpm (typical for all ECCS pumps). Therefore, the particle would have to be entrained within this 2 1/2 gpm out of the total flow handled by the pump to at least reach the orifice assemblies located in the seal water piping.

The orifice assemblies located in the seal water inlet piping to the cyclone separator are actually short segments of piping which are bored to 1/8-in diameter. The entrance and exit of each orifice are taper-bored to provide a smooth transition into and out of the orifice. This tapered entrance design would make it very difficult for a 3/32-in diameter by 1/8- to 1/4-in long particle to align itself such that it would become trapped and block seal water flow.

The sequence of improbable events necessary for a particle to block ECCS pump seal water flow is such that this blockage is not a credible concern. Sufficient redundancy and separation exists in the ECCS design to ensure adequate core cooling even if one pump experiences total seal failure due to the blockage of seal water flow.

Loss of seal water flow due to the blockage of an ECCS pump seal water orifice would cause the pump seal to leak at a maximum flow rate of 23 gpm. This leak rate would be insignificant with respect to the total system flow.

To ensure that the system function is maintained, the strainers are oversized to minimize pressure drop and flow velocities if the strainer should remove suspended debris and become partially clogged. The inlet flow area of the strainers is approximately 5 times the suction line flow area.

The RHR suction strainers are designed to withstand any loads during suppression pool transients, and are designed to withstand a pressure differential of 25 psi. Design mechanical loads consisted of the following:

1. Direct Drag Loads (SRV bubble, LOCA bubble, condensation, oscillation, chugging, and seismic sloshing)

2. Deadweight

3. Operating Pressure

4. Head of Static Fluid

5. Earthquake plus SRV plus LOCA vibratory loading

6. Pipe loads

The equipment has been designed and qualified by analyses to be capable of continued operation under the simultaneous application of all normal, operating, and seismic loads caused by operating basis earthquake (OBE), SSE, SRV, discharge loads, LOCA loads, and SS loads caused by the water motion inside the suppression pool and the loads due to differential pressure (H) across the surfaces of the equipment. Equipment adequacy was demonstrated for each of the load combinations described in Table 3.9A-6.

Net Positive Suction Head

Design data at a pump runout flow of 8,400 gpm allow a pressure drop across the strainer of 0.65 psi with 50 percent of the strainer area clogged. A pressure drop of 0.65 psi is assumed across the strainers in all net positive suction head (NPSH) calculations, ensuring adequate available NPSH to the RHR pumps at all times. The available NPSH was determined in accordance with RG 1.1.

Insulation

Types of insulation used for piping and equipment within the drywell and suppression chamber are discussed in the following paragraphs.

For piping and equipment located within the drywell that require insulation to minimize heat loss, primarily metal-reflective-type insulation is used.

Metal-reflective insulation is an all-metal construction-type insulation that has a stainless steel inside and outside jacket which encapsulates multiple layers of stainless steel insulation material. Metal-reflective insulation is installed in sections with overlapping edges and quick-release latches with keepers.

Two other types of insulation are used inside the drywell for special and limited application: Min-k and Temp-Mat insulation. Min-k is a powder-type insulation used where space is limited and is encapsulated in stainless steel so as to be watertight. Temp-Mat is a borated, spun glass, blanket-type insulation used where it is necessary to lower the neutron flux (i.e., at the primary shield wall penetration), and is also encapsulated in stainless steel (see Table 6.2-64).

No antisweat insulations are used within the primary containment.

The mechanism for transport of any insulation debris from the drywell into the suppression pool following an accident involves a series of unlikely occurrences, as discussed in the following paragraphs.

In the event of a postulated pipe break, some insulation in the immediate vicinity of the break could possibly be removed by direct jet impingement. Since the insulation is fabricated and installed in overlapping sections, only sections in the immediate vicinity of the break would likely be affected. The stainless steel jacket minimizes the possibility of the Min-K and Temp-Mat insulation breaking up and becoming transportable debris. Any metal components removed would be expected to fall on the drywell platforms or the drywell floor and remain there throughout the accident. Even if insulation material did reach the drywell floor as suspended debris, it would tend to accumulate outside the downcomer pipes that protrude 3 to 6 in above the sloping drywell floor. Any suspended debris overflowing through downcomer pipes would enter the suppression pool with a vertically downward initial velocity at an elevation of 190 ft 0 in. The suction strainers are located below this elevation; however, they are located closer to the pool wall (2 ft 7 in maximum from the wall) than the outermost downcomer pipes (6 ft from the pool wall). Additionally, the suction strainers are designed for a maximum flow velocity through the perforations of 1 ft/sec with the strainers 50 percent plugged. Therefore, the fluid velocity approaching the strainers is less than 1 ft/sec and the potential for any insulation to migrate toward the strainers is small. Only Min-K insulation will float, whereas Temp-Mat and metal-reflective insulation will sink in the suppression pool. This reduces the possibility of covering strainers which are several feet below the pool surface.

6.2.2.3 Design Evaluation

The DBA for the containment spray system is failure of a steam line having a break area equal to 0.3 sq ft with suppression pool steam bypass of 0.05 sq ft ($A/\sqrt{K}$). In the long term, this accident is similar to the DER of a recirculation suction or a main steam line. If such an event occurred, the short-term (prior to the actuation of the RHR heat exchangers) energy released from the RCS would be absorbed by the suppression pool, and the suppression pool temperature would increase. In the long

term, fission product decay heat would continue to be absorbed by the pool. Unless this energy is removed from the suppression pool, a high containment pressure/temperature would result.

The containment cooling mode of the RHR system with the containment sprays is used to remove heat from the suppression pool and to limit the long-term, post-LOCA primary containment internal pressure and the suppression pool temperature to less than 45 psig and 212°F, respectively.

To evaluate the adequacy of the containment heat removal system, the following sequence of events is assumed to occur:

1. With the reactor initially at 102 percent of rated thermal power, a steam line failure with a rupture area of 0.3 sq ft occurs. The bypass of steam is assumed, with an area of 0.05 sq ft ($A/\sqrt{K}$ factor) through the drywell floor.

2. LOOP occurs and one standby diesel generator fails to start and remains out of service during the entire transient. This is the most limiting single failure, failure of the Division II electrical system.

3. Only three ECCS pumps are functional following the postulated LOOP and one standby diesel generator failure. Thus, one HPCS pump, one LPCS pump, and one RHR pump are available for emergency core cooling and containment heat removal functions.

4. Within 30 min after the accident, the plant Operator actuates the RHR heat exchanger and the containment spray. This involves transferring the RHR pump from LPCI to the containment spray mode and starting another service water pump.

   Once the containment heat removal function is established, no further Operator action is required.

When calculating the long-term, post-LOCA primary containment pressure and suppression pool temperature, it is assumed that the service water temperature is at a maximum value of 82°F. In addition, the RHR heat exchanger is assumed to be in a fully fouled condition during the transient (tubeside fouling, 0.001 hr-°F-sq ft/Btu and shellside fouling, 0.0005 hr-°F-sq ft/Btu). Both of these assumptions are conservative and maximize the containment pressure and the suppression pool temperature response.

6.2.2.3.1 Containment Sprays

6.2.2.3.1.1 Design Bases

The containment spray system is capable of quickly reducing containment pressure during the postaccident period of a LOCA through condensation of steam in the drywell and through cooling of the noncondensable gases in the free volume above the suppression pool. The design of the containment spray system is in accordance with Category I and Safety Class 2 requirements.

6.2.2.3.1.2 System Design

The containment spray system consists of two subsystems, the drywell spray and the suppression chamber spray. The drywell spray consists of two independent loops and spray headers (Figure 6.2-39). The suppression chamber spray consists of one spray header supplied from two otherwise independent loops. Since the water source for all containment sprays is the suppression pool, the system is a closed loop. There are no chemicals added to the spray water. The spray water is cooled by the RHR heat exchangers. The calculated minimum flows for the drywell and suppression chamber sprays are shown in Table 6.2-52. Containment spray is an operational mode of the RHR system (Section 5.4.7).

The containment spray isolation valves are electrically interlocked to allow actuation of the drywell spray only when: 1) there is a LOCA signal or a system-level LPCI manual initiation signal, and 2) there is a high drywell pressure signal present. A second electrical interlock prevents actuation of either the drywell or suppression chamber spray lines until the corresponding LPCI injection valve is shut.

The containment spray system is safety related and, in case of LOOP, supplied with a redundant onsite standby power source.

The system is designed to operate under the conditions indicated in Table 6.2-6.

The LOCA analysis assumed uninterrupted LPCI flow to the reactor core during the first 10 min of the accident. In accordance with EOPs, a RHR pump may be realigned from the LPCI mode to the containment spray or suppression pool cooling mode of operation independent of the 10 min elapsed time criteria assumed in the LOCA analysis. The approach utilized in the EOPs is acceptable as these procedures contain adequate cautions to deter the Operator from premature flow diversion of RHR in the LPCI mode of operation. The guidance provided in EOPs assures that the PCT identified by the USAR LOCA analyses is not increased due to Operator realignment of the RHR system during an accident. Therefore, EOPs may result in diversion of RHR flow from the core cooling function prior to 10 min into the accident; however, such action will not increase the PCT of the fuel.

Distribution of spray in the air space is made as complete and uniform as practical with minimal direct impingement on wall and component surfaces. The sizes, types, number, and location of the spray nozzles are suitable for delivering the required quantity of water in the proper spray pattern and particle size shown on Figure 6.2-91. The suppression chamber spray header and spray nozzle locations are shown on Figure 6.2-92. The lower and upper drywell spray header locations are shown on Figure 6.2-93. The spray nozzle locations for the two drywell spray headers are shown on Figure 6.2-94.

The expected spray pattern of the spray nozzles is hollow cone.

Figures 6.2-40 through 6.2-42 show expected spray coverage several feet below the spray nozzles. Figure 6.2-43 shows the extent of the volume coverage by the sprays. Spray drops are expected to reach thermal equilibrium with the primary containment atmosphere. Table 6.2-51 shows the containment spray system parameters.

The spray system is designed to provide 94 percent of the system flow to the drywell spray header and 6 percent of the flow to the wetwell header with one RHR loop in the spray mode. Following a LOCA, the drywell and wetwell would be pressurized and the downcomer vents would be hot due to steam flow to the suppression pool. If the sprays are initiated in this condition, the drywell and wetwell pressures would drop, as indicated on Figure 6.2-28A, assuming 90-percent spray thermal effectiveness. Reduced wetwell spray effectiveness resulting from spray impingement on the hot downcomer pipes would tend to increase the wetwell pressure. However, this pressure increase will be limited by the vacuum breakers which will open to permit any uncondensed steam in the wetwell to flow into the drywell and condense. For these reasons, the Unit 2 LOCA steam bypass analysis is not sensitive to reduced wetwell spray effectiveness.

6.2.2.3.1.3 Design Evaluation

Due to the redundancy and separation of the containment spray loops, containment spray is available to rapidly reduce containment pressure during the postaccident period. The long-term containment pressure response is shown on Figures 6.2-3 and 6.2-4 (recirculation pump suction line break) and Figures 6.2-16 and 6.2-17 (main steam line break) for the spray and no-spray cases. Even with minimum ECCS operation and no containment spray, the postaccident containment pressure remains significantly below the containment design value of 45 psig.

6.2.2.3.2 NPSH Availability

The available NPSH for the RHR pumps is calculated based on the regulatory position of RG 1.1 considering:

1. Water level of the suppression pool is at a minimum water level of el 197 ft 8 in.

2. Suppression pool water temperature is 212°F.

3. No credit is taken for any increase in containment pressure (atmospheric pressure is assumed).

4. Strainers are 50-percent clogged.

The available NPSH at 8,200 gpm flow is 15.09 ft, which is greater than the RHR pump required NPSH of 14 ft, at a point 2 ft above the pump mounting flange.

6.2.2.3.3 Heat Removal

Analysis of the containment heat removal capability is performed for large and intermediate breaks with the LOCTVS and CONSBA codes (Section 6.2.1.1). The resultant primary containment pressure and temperature responses are described in Section 6.2.1.1.

Even with the very degraded conditions of heat exchanger and service water temperature previously outlined, the peak primary containment pressure does not exceed the containment design pressure of 45 psig, and the peak suppression pool water temperature does not exceed the design suppression pool water temperature of 212°F.

A system-level/qualitative-type plant failure modes and effects analysis (FMEA) of the RHR system is provided in Appendix 15A, Plant Nuclear Safety Operational Analysis (NSOA). The FMEA of the balance-of-plant (BOP) instrumentation and control components of the RHR system (suppression pool cooling mode and containment spray cooling mode) is contained in the Unit 2 FMEA document.

Figures and tables showing the calculated performance of the key variables as a function of time following the occurrence of a DBA, assuming minimum ESFs available, are summarized as follows:

1. Accident parameters used in the analysis (Table 6.2-52).

2. Primary containment pressure for the steam line break area of 0.3 sq ft (Figure 6.2-28A).

3. Suppression pool water temperature for the steam line break area of 0.3 sq ft (Figure 6.2-45).

4. Integrated energy content of the water containment and the suppression pool (Table 6.2-53).

5. Integrated energy absorbed by the passive heat sinks and removed by the RHR heat exchanger (Table 6.2-53).

6. Heat removal rate of the RHR heat exchanger for the steam line break area of 0.3 sq ft (Figure 6.2-46).

7. Suppression pool water temperature for DER of recirculation suction line (Figure 6.2-11).

The very conservative evaluation procedure previously described clearly demonstrates that the RHR system in the containment cooling mode can meet its design objective of safely terminating the postaccident primary containment temperature transient.

6.2.2.4 Tests and Inspections

Preoperational and operational testing and periodic inspection of containment heat removal system components are described in Sections 5.4.7.4, 6.3.2.7, 14.2, and the Technical Specifications. Logic is provided to prevent normal testing of one drywell spray isolation valve when the other valve in the same loop is open.

6.2.2.5 Instrumentation Requirements

The RHR containment spray cooling mode and the RHR suppression pool cooling mode of the RHR system are manually initiated from the main control room. Details of the instrumentation are provided in Sections 7.3.1.1.3 and 7.3.1.1.4, respectively.

6.2.3 Secondary Containment Functional Design

The secondary containment, consisting of the reactor building and auxiliary bay structures, completely surrounds the primary containment. The secondary containment is maintained at a negative pressure of 0.25 in W.G. with respect to the surrounding outside atmosphere. The secondary containment provides a means of controlling fission product leakage to the environment. This section discusses the reactor building design, the role of the reactor building ventilation system, and the standby gas treatment system (SGTS) which is used to depressurize and clean the reactor building atmosphere. The SGTS is discussed in detail in Section 6.5.1.

6.2.3.1 Design Bases

The reactor building structure completely encloses the reactor and the primary containment.

The reactor building structure, in conjunction with the SGTS and portions of the reactor building ventilation system, provides the means of controlling and minimizing leakage from the primary containment to the outside atmosphere during a LOCA and from the refueling facilities (including the spent fuel pool) during a postulated refueling accident.

The normal reactor building ventilation system (HVRS) is designed to automatically shut down and isolate and to automatically start the SGTS and safety-related unit coolers on receipt of any of the following signals that indicate either a LOCA or a refueling accident:

1. High drywell pressure.

2. Reactor vessel low water level.

3. High radiation level in exhaust ducts above or below the refueling floor.

The secondary containment pressure control function utilizes the HVRS (normal operation) and the SGTS (emergency operation) instrumentation and controls to maintain a negative pressure of 0.25 in W.G. with respect to the atmosphere. This ensures that while the systems are operating, any leakage is into the reactor building. All reactor building air is either exhausted through the exhaust air plenum, where it is constantly monitored, or discharged through the filtration units of the SGTS.

The reactor building isolation signals, isolation dampers for the HVRS, and the SGTS are all designed to Category I and Class 1E requirements. The design basis for the SGTS is given in Section 6.5.1.

The reactor building structure houses the refueling and reactor servicing equipment, the new and spent fuel storage facilities, and other reactor auxiliary or service equipment, including the RCIC system, reactor water cleanup (RWCU) demineralizer system, SLCS system, CRD system equipment, HPCS, and electrical equipment components. The reactor building auxiliary bays house the LPCS system, the RHR system heat exchangers and pumps, the reactor building closed loop cooling water (RBCLCW) system heat exchangers, and electrical equipment components. The reactor building structure protects the equipment from the SSE, design basis tornado (DBT) and tornado-generated missiles, and design basis wind. The reactor building structure is designed to meet the following design bases:

1. The reactor building is designed to meet Category I requirements.

2. The reactor building is designed and constructed in accordance with the structural design criteria presented in Section 3.8.

3. The reactor building design provides for low in-leakage and out-leakage during reactor operation.

4. The reactor building is designed to withstand applied wind pressures resulting from the design basis wind velocity (Section 3.3).

5. The reactor building is designed to withstand pipe whip loads plus jet impingement or jet reaction loads due to high-energy pipe breaks outside primary containment (Section 3.6).

6. The reactor building design allows for periodic inspections and functional tests of the penetrations.

7. The reactor building is designed to withstand tornado-generated missiles (Section 3.5).

8. The reactor building is designed for all probable combinations of the design basis wind, DBT velocities, and associated differences of pressure within the structure and atmospheric pressure outside the structure. The reactor building design incorporates the pressure loading from various conditions (i.e., events) that may be encountered during plant operation. These events include:

   a. Negative internal pressure of 0.25 in of water under which the reactor building normally operates.

   b. Negative internal pressure of 0.332 in of water (0.012 psig) relative to the outside atmosphere, which exceeds the 0.25 in of water in Item a, to account for any uncertainty in pressure measurement and to account for any negative pressures actually developed by the SGTS or by unknown causes.

   c. Positive pressure of 6.92 in of water (0.25 psig) relative to the outside atmosphere, to account for: any positive pressure transient that the reactor building may experience following a postulated pipe break in the reactor building, any outward positive differential pressures created by wind loads, and any uncertainty in pressure measurements.

9. All entrances to the reactor building are through double door airlock systems. Table 6.2-62 provides a listing of these entrances and their locations.

   Use of these entrances is under administrative control to maintain a negative reactor building differential pressure, thus ensuring secondary containment integrity.

   All of these entrances are equipped to provide local position indication.

All entrances to the reactor building which are radiation access control doors on the access control computer system have position indicators to detect unauthorized access. Alarms generated by these doors are received on the dedicated monitor and read-only printer in the control room.

6.2.3.2 System Design

Refer to Figures 1.2-6 through 1.2-12 for general arrangement drawings of the reactor building showing plan and elevation views of the boundary of the structure. Also refer to Figures 3.8-1 and 3.8-2. Refer to Table 6.2-54 for the design and performance data for the secondary containment structure.

The reactor building design criteria are described in Section 3.8.4. Refer to Section 3.8.4 for identification of the codes, standards, and guides applied in the design of the reactor building structure.

6.2.3.2.1 Reactor Building Ventilation System

Normal ventilation for the reactor building is described in Section 9.4.2.

6.2.3.2.2 Postaccident Design Provisions

The major design provisions that prevent postaccident primary containment leakage from bypassing the SGTS (except for those lines identified as potential bypass leakage paths in Tables 6.2-55a, b, c, and d) are the secondary containment pressure control instrumentation of the SGTS, the reactor building ventilation isolation system, the isolation signals listed below, and the standby power system.

A portion of HVRS is required to operate during accident conditions. Part of the system is automatically shut down, the HVRS emergency recirculation dampers are aligned, and the SGTS starts in the event of any of the following isolation signals:

1. Reactor vessel low water level.

2. High drywell pressure.

3. High radiation level in exhaust ducts above or below the refueling floor.

The SGTS can also be started manually.

All ventilation system penetrations of the reactor building (except those of the SGTS) are fitted with two fail-closed, air-operated dampers in series. All dampers automatically close on any one of the aforementioned isolation signals.

Penetrations of the reactor building are designed with leakage characteristics that ensure that the leakage requirements of the entire building are not exceeded.

Railcar entrance to the reactor building railroad airlock is through an interlocking double door airlock system. The railroad airlock is completely within and along the northeast side of the reactor building at el 261 ft. One of the interlocked doors is the exterior railcar door at the north end of the railroad airlock, and the other is the interior railcar door at the south end of the railroad airlock. A smaller door for personnel ingress and egress is incorporated into the design of the interior railcar door. All three doors must be closed before any one of them can be opened.

The reactor building pressure control function automatically maintains a subatmospheric pressure of 0.25 in W.G. by monitoring the differential pressure between the reactor building interior and the external atmospheric pressure. The differential pressure is monitored by a differential pressure transmitter. The signal that indicates the differential pressure also controls the position of the recirculation dampers in the HVRS supply fan units. In the event of reactor building isolation, the reactor building pressure control instrumentation regulates the reactor building pressure by controlling the SGTS recirculation flow.

The reactor building pressure control instrumentation is designed to eliminate fluctuations in reactor building pressure caused by such factors as wind gusts. Reactor building pressure is indicated and recorded and loss of negative pressure is alarmed in the main control room.

6.2.3.2.3 Bypass Leakage Paths

Table 6.2-56 presents a tabulation of all primary containment process piping penetrations including the potential reactor building bypass leakage paths. The potential bypass leakage paths are routed through the reactor building and terminate in the radwaste, standby gas treatment, turbine generator buildings, or yard. No guard pipes are used on penetrations and, therefore, guard pipes cannot constitute a bypass leakage path. All process lines that rely on a closed system within the primary containment as a leakage boundary terminate within the reactor building; therefore, these lines are not considered potential bypass leakage paths.

Bypass leakage is included in the radiological evaluation of design basis events. This is discussed in Section 15.6.5.5. Tables 6.2-55a, b, c, and d show the bypass leakage paths considered. They include four main steam lines, two main steam drain lines, one RWCU line, one feedwater line, four postaccident sampling lines, six primary containment purge lines, four drywell floor and equipment vent and drain lines and six nitrogen/instrumentation lines.

All leakage is conservatively assumed to be across isolation valve seats and to remain within the system piping until released to the environment. Any leakage escaping across outboard isolation valve stem packing would be released to the secondary containment or main steam tunnel. Any leakage into the secondary containment would be processed by the SGTS. Contaminants leaked into the main steam tunnel will be transported to the environment more slowly due to the much larger cross-sectional area of the tunnel and the resulting slower average velocities.

No credit is taken for a reduction in bypass leakage due to water inboard of or trapped between isolation valves. The isolation valves are assumed to leak containment atmosphere instantaneously following the accident. No credit is taken for the time required to initially pressurize the volume between the isolation valves.

Leakage transport time to the environment is based on 1/2 of the available horizontal and vertically downward flow piping located between the isolation valve and the environment.

Further conservatism is added to the analysis by the assumption that all isolation valves in these paths, except the MSIV and feedwater check valves, leak at a rate equal to the maximum permissible, ASME Section XI, Subsection IWV-3426, recommended acceptance level of 7.5 scf/day per inch of nominal valve diameter at functional pressure. The MSIVs are assumed to leak at 24 scfh, nearly ten times the valve design limit, with credit for post-LOCA isolation of the more contaminated of the east and west control room outside air intakes. For an assumed MSIV leak rate of 15 scfh or less, no control room air intake isolation is required. Leakage across check valves, except the feedwater check valves, is assumed to be at twice the recommended rate of 7.5 scf/day per inch of nominal valve diameter, as provided for by ASME Section XI, Subsection IWV-3426. Leakage across the feedwater valves is assumed to be 12 scfh.

Several process lines eliminate bypass leakage by the use of water seals. These are discussed below and include condensate makeup and drawoff (CNS), RCIC, and HPCS. Feedwater system (FWS) is also discussed below, but no credit for water seal is applied for that system. A typical loop seal is shown on Figure 6.2-88.

CNS

While not directly connected to the primary containment, the CNS system is used as the alternate fill source to the RHR, HPCS, LPCS, and RWCU systems. Each condensate fill connection to these systems is isolated by means of a normally closed globe valve. The main supply line into the secondary containment contains a check valve at the low point which, in case of a pipe break outside the containment, is sealed by a 70-ft leg of water. Although the CNS system is not of seismic design, any line break within the reactor building would provide a preferential flow path, for containment atmosphere leakage, into the reactor

building atmosphere. Under this condition gaseous leakage would be collected by the SGTS and thus not be classified as bypass leakage.

RCIC

The RCIC path from the primary containment to the condensate storage building is protected from bypass leakage. When RCIC is taking suction from the CST (2CNS-TK1A), the tank static head pressure maintains a 23-psig water seal at valve 2ICS*V28 and/or 2ICS*MOV136 (Figure 6.2-81). Also, the piping arrangement as shown on Figure 6.2-81 provides a loop seal with a high point at 2ICS*MOV136. Thus, any containment atmosphere leakage through this valve during the period that containment pressures exceed water seal pressure would be trapped at this high point. If a LOCA and a SSE take place simultaneously and a condensate line break occurs, 2ICS*MOV129 on the condensate tank line will shut automatically, creating an additional barrier to bypass leakage.

HPCS

The arrangement of the HPCS suction line from CST 2CNS-TK1B provides enough static head pressure to keep a 75-ft (32-psig) water seal at the line low point (valve 2CSH*MOV101) on Figure 6.2-83. Further, the piping arrangement, as shown on Figure 6.2-83, provides two intermediate loop seals with high points at valves 2CSH*MOV118 and 2CSH*V59, ensuring that any containment atmosphere leakage occurring during the 20 min that containment pressures exceed water seal pressure would be trapped between these high points. If a LOCA and a SSE take place simultaneously and a condensate line break occurs, 2CSH*MOV101 on the CST line will shut automatically, creating an additional barrier against bypass leakage.

FWS

For LOCAs not involving a feedwater line break, sufficient water exists in the vertical feedwater piping between the containment penetration and the reactor vessel to prevent bypass leakage for at least 30 days after the accident. See Figure 6.2-84.

For a break in feedwater piping inside containment, bypass leakage through this piping is included in the analysis of Section 15.6. However, as discussed below, a water seal restored after the break will effectively prevent escape of containment atmosphere to the environment after 10 min.

In considering a break in the feedwater piping within the primary containment, credit can be given to the piping arrangement which provides low stress levels along with pipe whip restraints. Consequently, it can be stated that the containment penetration is a break exclusion area. Assuming a break in the feedwater line at the end of the break exclusion region inside the primary containment (see Section 3.6A and Figure 3.6A-20), sufficient

water will remain in the line, even after flashing due to initial depressurization, to maintain a vertical water seal on the feedwater isolation valves (Figure 6.2-84). Water losses due to long-term containment pressure reduction and the associated water vaporization, and the back-leakage through the two isolation check valves for 30 days, will be replenished by reactor water leaking from the break. Within 10 min after the break, the ECCS injection water will reflood the reactor to above the level of the feedwater sparger. At that point water would flood back into the feedwater piping and then into the intact containment penetration piping (Figure 6.2-84). This would more than make up for any losses due to leakage out the containment isolation valves. Thus a continuous water seal is provided to prevent any bypass leakage through the feedwater lines after the initial 10-min refilling period. Notwithstanding the above, bypass leakage through a ruptured feedwater line is included in the radiological analysis for the entire 30-day period to ensure conservative analysis results.

In addition to the two isolation check valves, each feedwater line has a remote-manual gate valve outboard of the isolation valves that may be shut subsequent to a LOCA anytime the Operators determine that feedwater flow is unnecessary or unavailable. The gate valve provides further back-leakage control. However, this valve is assumed to remain open for the purpose of evaluating bypass leakage.

6.2.3.2.4 Bypass Leakage Rates

Bypass leakage rates as a function of time after the postulated LOCA are predicted for each path by two methods, assuming isothermal flow and isentropic flow. Tables 6.2-55a and 55c list the bypass paths considered and their contributions to the total bypass leakage, assuming isothermal flow determined with the following equation:

$$\dot{m} = \frac{A}{\sqrt{K}} \left\{ \frac{g_c(P_u^2 - P_D^2)}{RT_u} \right\}^{1/2} \qquad (6.2\text{-}12)$$

Where:

P_u = Upstream absolute pressure (post-LOCA pressure/temperature profile per Section 6.2.1)

P_D = Downstream absolute pressure

T_u = Upstream absolute temperature

R = Gas constant

K	=	Resistance coefficient
A	=	Flow area
$\dot{m}$	=	Mass flow rate
g_c	=	Conversion constant

To quantify the sensitivity of the bypass leakage analysis to the flow model assumption, the bypass calculation was repeated considering the leakage flow to be characterized as isentropic flow through an orifice. Tables 6.2-55b and 55d summarize the isentropic flow results determined with the following equation:

$$\dot{m} = A \left\{ 2\, g_c \left(\frac{\gamma}{\gamma - 1}\right) \left(\frac{P^2_u}{RT_u}\right) \left(\frac{P_D}{P_u}\right) \frac{2}{\gamma} \left[1 - \left(\frac{P_D}{P_U}\right) \frac{\gamma_{-1}}{\gamma} \right] \right\}^{1/2}$$

(6.2-13)

Where:

P_u	=	Upstream absolute pressure (post-LOCA pressure/temperature profile per Section 6.2.1)
P_D	=	Downstream absolute pressure
T_u	=	Upstream absolute temperature
R	=	Gas constant
γ	=	Specific heat ratio
g_c	=	Conversion constant
A	=	Orifice flow area (to be determined from the Technical Specification of allowable leak rate)
$\dot{m}$	=	Mass flow rate

The isentropic flow is generally 5 to 35 percent higher than the isothermal flow depending upon the number of valves and time after LOCA.

In each case the fractional flow rate is evaluated using the following equation:

$$f = \frac{\dot{m}}{\rho V}$$

(6.2-14)

Where:

$\dot{m}$ = Mass flow rate

ρ = Density of containment air and steam mixture (P/RT)

V = Containment volume

f = Fractional flow rate

The containment bypass leak rate for various paths is calculated based on two closed valves in series or one closed and one open valve depending upon the direct consequence of the postulated failure of one emergency diesel generator or the failure of one MSIV to close.

Two single failure scenarios are included in the analysis to predict the bypass leakage rates. One scenario is the postulated failure of one emergency diesel generator, combined with LOOP which results in loss of one division of electrical power. This scenario results in all motor-operator containment isolation valves on that division failing as is (see Table 6.2-56), thereby reducing the restrictions to bypass leakage. The containment bypass leakage rates for various paths are calculated based on two closed valves in series, or one closed and one open valve depending upon the direct consequence of the postulated failure of one emergency diesel generator. The other single failure scenario is the failure of one MSIV to close. Both single failure scenarios are included in Tables 6.2-55a, b, c, and d.

6.2.3.2.5 Iodine Plateout Considerations

The radiological consequences arising from bypass leakage are provided in Section 15.6.5. The analyses include credits for elemental iodine deposition on the walls of the piping between the isolation valve and the release point. Details of the iodine deposition analysis can be found in Section 15.6.5.5.3.

6.2.3.2.6 Activity Transport Delay Considerations

The leakage of activity from the primary containment to the environment is through a portion of piping downstream of the outer isolation valve. Because of the very low leakage rates, there is a considerable transport delay time between the outer isolation valve and the release point. Therefore, the analyses include the credit of the delay time in the dose calculations. This is further explained in Section 15.6.5.5.3.

6.2.3.3 Design Evaluation

6.2.3.3.1 LOCA Temperature and Pressure Transient

During normal plant operation the reactor building and auxiliary bays are maintained at a negative pressure of 0.25 in W.G., relative to the outside atmosphere by the HVRS described in Section 9.4. In the event of a LOCA, the HVRS is isolated and the SGTS is initiated upon receipt of any of the three signals listed in Section 6.2.3.2.2. Details of the SGTS are provided in Section 6.5.1.

6.2.3.3.1.1 Summary and Conclusions

The following summarizes the drawdown analysis:

1. For a 2,670 acfm in-leakage at 0.25 in W.G. pressure differential between secondary containment and the environment, secondary containment temperature of 105°F, outside air temperature of -20°F, and a temperature differential between the secondary containment and the service water in the range of 0 to 10°F, the time required to reestablish -0.25 in W.G. in the secondary containment following a LOCA (i.e., drawdown time) is estimated to be less than 1 hr. The radiological dose evaluation uses 1-hr drawdown time for conservatism.

2. The SGTS has adequate capacity to maintain the secondary containment at 0.25 in W.G. vacuum for an extended period of time.

The difference between secondary containment in-leakage and SGTS exhaust flow is the net SGTS exhaust rate, and is referred to as differential flow.

During the winter months, the colder air temperature is expected to increase the air in-leakage resulting in a lower effective differential flow. An increased differential temperature is required to offset the effect of this lower differential flow. Figure 6.2-77 illustrates the relationship between external temperature and the required differential temperature for the in-leakage rate specified in Item 1.

6.2.3.3.1.2 Calculation Approach

The secondary containment drawdown time is dependent upon the following parameters:

1. Secondary containment environmental conditions (pressure, temperature, relative humidity) prior to LOCA.

2. Secondary containment in-leakage.

3. SGTS exhaust capacity and startup delay.

4. Building unit cooler and central cooling system capacity and startup delay.

5. Service water temperature.

6. Outside environmental conditions (pressure, temperature, relative humidity).

7. Heat generation and evaporation.

8. Building size.

Due to seasonal variation, secondary containment temperature, service water temperature, outside air temperature, and secondary containment and outside relative humidities are expected to vary within certain limits. Therefore, the drawdown time may vary depending upon these parameters for a given plant condition. The drawdown time is influenced by the differential temperature between the secondary containment and the service water, as well as the differential flow between the secondary in-leakage rate and the SGTS exhaust. A smaller differential temperature results in low heat removal rate which increases the drawdown time. A smaller differential flow results in a lower effective SGTS exhaust rate which also increases the drawdown time. The secondary containment temperature has minor effect on the drawdown time for a constant differential temperature between the secondary containment air and the service water.

The outside air temperature change has little direct effect on the drawdown time. However, outside air temperature affects the in-leakage rate due to a change of air density relative to elevation. For example, in winter months a large temperature differential between the secondary containment and outside air creates a larger differential pressure at ground level than at the roof. Thus, the secondary containment in-leakage increases. Therefore, the outside air temperature indirectly increases the drawdown time.

Depending upon its value, the relative humidity inside the containment can have a significant impact on the drawdown time. Since, during normal operation, ventilation system causes about two air changes per hour in the building, the relative humidity inside the containment is essentially dependent on the outside relative humidity. The evaporation from the spent fuel pool does not have a significant impact on relative humidity.

The drawdown time estimation is done using THREED computer code. The secondary containment is divided into multiple subvolumes based on localized heat loads and cooling rates. The heat load estimation considers the operation of equipment prior to LOCA, initiation of equipment following LOCA, and residual heat of equipment which trips following LOCA. The heat removal of the

cooling systems is based on the differential temperature between the building and the cooling water. The analysis also considers the worst-case single failure of 600 V Division II bus failure without LOOP and appropriate system startup delays.

6.2.3.3.1.3 Assumptions

Some of the assumptions applied to this analysis were:

1. A LOCA and the loss of the Division II 600-V bus are assumed to occur simultaneously. No LOOP is postulated to occur. This combination results in the highest post-LOCA heat load with the lowest cooling capacity within the secondary containment.

2. Adiabatic boundary conditions are assumed for the surface of the reactor building and auxiliary bay structure exposed to the outside air temperature.

3. The initial secondary containment environmental conditions are 0.25 in W.G. vacuum. ECCS rooms modeled in the calculation are conservatively assumed to have an initial air temperature equal to the service water temperature.

4. The initial relative humidity inside the secondary containment is 75 percent or less. The maximum 1-hourly dew point at the site during October through April is 63°F. During this time frame, the minimum reactor building temperature is 70°F or more. When the 63°F dew point air is heated to the containment temperature of 70°F, it results in a relative humidity of about 75 percent. During summer months, the maximum 1-hourly dew point is 73°F. The reactor building temperature during the summer, with outside temperature greater than 73°F, is expected to be approximately 85°F. The 73°F dew point air, when heated to 85°F, results in relative humidity of about 67 percent.

5. The maximum spent fuel heat load calculated for the plant design basis analysis is used.

6. Evaporation and convection effects from spent fuel pool are included in drawdown time estimation. No other source of latent heat is considered.

7. It is assumed that the Operator does not divert the SGTS fan capacity for decay heat cooling of the other loop for a duration of at least 5 hr after LOCA. The SGTS begins operation 60 sec after the LOCA occurs, at a nominal flow rate of 3,720 cfm.

8. The compressive effect of primary containment expansion is assumed to be insignificant.

9. The primary containment wall heat gain is assumed to increase linearly from the normal heat gain at time zero to the full emergency heat gain at 24 hr to account for the heat storage capacity of the 5.25-ft thick concrete wall.

10. The ECCS piping heat gain is based on the calculated pool temperature transient following a LOCA and a reactor building temperature of 70°F.

11. The sensible heat removal rate of each operating unit cooler is determined based on reactor building temperature and relative humidity. The unit coolers are unavailable during the first 90 sec of the LOCA. A 40-percent degradation is assumed for all drawdown-related unit coolers, with the exception of 2HVR*413A and B which are assumed to be degraded 20 percent.

12. The nonessential lighting heat load is assumed to be available.

13. Secondary containment temperature range is 70°F to 105°F.

14. One spent fuel pool heat exchanger and two component cooling heat exchangers are in operation for cooling the spent fuel pool.

6.2.3.3.2 High-Energy Line Break Evaluation

All high-energy lines within the reactor building and the analysis of line rupture for any of these lines are discussed in Sections 3.6.1 and 3.6.2.

6.2.3.4 Test and Inspection

Tests and inspections of the HVRS and the SGTS will be performed prior to initial fuel load and periodically thereafter in accordance with Technical Specification requirements.

To demonstrate that the SGTS will accomplish its design objectives under DBA conditions, the Technical Specifications require that the system must be tested every 18 months ±25 percent.

Four curves are developed to determine the SGTS performance and the secondary containment leak-tightness (see Figures 6.2-95a, 6.2-95b, 6.2-95c and 6.2-95d).

SGTS testing, Technical Specification surveillance 4.6.5.1.c.1, commonly referred to as the secondary containment drawdown test, is performed to ensure that the SGTS is capable of establishing an acceptable negative pressure within the time constraints

imposed by the surveillance test drawdown analysis. This analysis does not model LOCA heat loads and unit cooler performance since a LOCA is not simulated during the test.

In addition, a SGTS subsystem is operated for 1 hr to ensure that measured building in-leakage is within the requirements stipulated in Technical Specification surveillance 4.6.5.1.c.2.

The performance of the SGTS is verified by measuring the time it takes to achieve a -0.25 in W.G. pressure from initiation of the test signal. This time is the drawdown time. The SGTS is run for 1 hr to ensure building stability and then the flow meter reading is recorded in order to determine if building in-leakage requirements are met. The purpose of the four figures (6.2-95a, 6.2-95b, 6.2-95c and 6.2-95d) is to allow comparison of the test results against the design basis without recalculating the LOCA drawdown analysis using test condition parameter values each time a test is run.

To use these figures, the following are required from plant testing:

1. Q - SGTS flow meter reading (cfm),

2. Tout - outside air temperature at the time of test (F),

3. dp - reactor building pressure at equilibrium from test ("WG),

4. Tsw - service water temperature at the time of test (F) and

5. Trb - reactor building temperature at the time of test (F).

The above 5 parameters are used with the 4 figures in the following manner:

1. From Figure 6.2-95a, look up F1 (correction factor) for the appropriate dp and Tout,

2. From Figure 6.2-95b, look up F2 (correction factor) for the appropriate Tout and Tsw

   Qactual = in-leakage rate (cfm of outside air) into building at -0.25 in W.G. at roof elevation, or

   Qactual = F1 * F2 * Q,

3. From Figure 6.2-95c, look up acceptable drawdown time for the appropriate Qactual

Acceptance Criteria: drawdown time test result is less than required drawdown time from Figure 6.2-95c,

4. From Figure 6.2-95d, look up acceptable in-leakage for the appropriate Tout and Trb

Acceptance Criteria: Qactual calculated from above is less than Qallowed from Figure 6.2-95d.

The F1 correction factor converts the test condition flow rate to equivalent in-leakage rate at -0.25 in W.G. The F2 correction factor converts a raw meter reading to building in-leakage. The F2 correction factor also accounts for actual gas conditions at the flow orifices.

The drawdown time limit for each drawdown surveillance test must be determined based on the in-leakage measured during the test (corrected to account for test pressures and temperatures) to confirm the SGTS is capable of performing its intended function in accordance with the design basis.

Figure 6.2-95d provides a family of curves for various secondary containment temperatures (Trb). To determine the appropriate allowable in-leakage limit for test conditions, the curve closest to but not exceeding the secondary containment temperature is used. The point on that curve corresponding to the outside air temperature is used to determine the allowable in-leakage limit. The in-leakage value determined using F1 and F2 from Figures 6.2-95a and 6.2-95b must be less than or equal to the allowable in-leakage limit determined above.

The SGTS fans are rated for a design flow of 4,000 cfm. Under post-LOCA drawdown operation, a nominal flow of 3,720 cfm is drawn from secondary containment through the on-line filter train (see Figure 9.4-8L). Under postaccident conditions decay heat cooling can be provided by manually activating the decay heat removal valves.

6.2.3.5 Instrumentation Requirements

A reactor building negative air pressure of 0.25 in W.G. is automatically maintained under normal operating conditions by the HVRS. Normally, modulating air dampers automatically recirculate supply air to maintain negative pressure in the reactor building. During accident conditions (LOCA), isolation dampers in the air supply and air exhaust ducts will close automatically; the supply and exhaust air fans will stop, and an emergency recirculation air unit cooler and other safety-related unit coolers will start automatically to recirculate air through the reactor building. The SGTS will be automatically initiated and used to filter and maintain the required negative reactor building differential air pressure.

A detailed description of the reactor building heating, ventilating and air conditioning (HVAC) system is provided in Section 9.4.2. Refer to Section 7.3 for a description of instrumentation for the SGTS. Functional design details and logic are described in Section 7.3.1.1.5. Refer to Section 7.3.1.1.2 for a description of the instrumentation, functional design details, and logic for the primary containment and reactor vessel isolation control system (PCRVICS). Primary containment penetration lines and isolation signals applied to each are provided in Table 6.2-56.

6.2.4 Primary Containment Isolation System

6.2.4.1 Design Bases

6.2.4.1.1 Safety Design Bases

The safety design bases for the primary containment isolation system are:

1. Primary containment isolation valves provide for the necessary isolation of the primary containment in the event of accidents or other conditions to preclude radioactive releases from primary containment.

2. Capability for rapid closure or isolation of all pipes or ducts that penetrate the primary containment is provided so that leakage is maintained within permissible limits.

3. The design of isolation valving for lines penetrating the primary containment follows the requirements of General Design Criteria (GDC) 54 through 57 as noted in Table 6.2-56.

4. Isolation valving for instrument lines that penetrate the primary containment conforms to the requirements of RG 1.11.

5. Isolation valves, actuators, and controls are protected against loss of safety function from missiles.

6. The design of the primary containment isolation valves and associated piping and penetrations meets Category I requirements.

7. Primary containment isolation valves and associated piping and penetrations meet the requirements of the ASME Boiler and Pressure Vessel Code, Section III, Safety Classes 1 or 2, as applicable.

8. Primary containment isolation valve closure speeds limit radiological effects from exceeding guideline values established by 10CFR100.

The primary objective of the primary containment isolation system is to provide protection against release of radioactive materials to the environment from accidents occurring to the RCPB or lines connecting to the RCPB or penetrating primary containment. This is accomplished by automatic isolation valve closure of appropriate lines that penetrate the primary containment system. Actuation of the primary containment isolation system is automatically initiated at specific limits defined for reactor plant operation and, after the isolation function is initiated, it goes to completion.

The primary containment isolation system, in general, closes those fluid lines penetrating containment that support systems not required for emergency operation. Those fluid lines penetrating the primary containment which support ESF systems have remote manual isolation valves that can be closed from the control room, if required.

Redundancy and physical separation are required in the electrical and mechanical design to ensure that no single failure in the primary containment isolation system prevents the system from performing its intended functions.

The seismic and safety classification of equipment, systems, and penetration piping, up to and including the first isolation valve, is shown in Table 3.2-1. All piping is seismic Category I and either Safety Class 1 or 2. Conformance to SRP Section 3.6.2 requirements is discussed in Section 3.6.2.1.5A. Leakage detection capabilities for leakage external to the primary containment are discussed in Section 5.2.5. Actuation of the primary containment isolation system is initiated by various signals as listed in Table 6.2-56.

Primary containment isolation valves are designed to minimize leakage from shaft and/or bonnet seals. Additionally, periodic inspection, testing, and maintenance procedures under normal operating conditions are intended to minimize the potential for leakage under off-normal conditions. Radiological consequences of potential leakage from ESF systems are addressed in Section 15.6.5.5.3 and Table 15.6-13.

Criteria for the design of the PCRVICS are listed in Section 7.3.1.1.2. The bases for assigning certain signals for primary containment isolation are also listed and explained in Section 7.3.1.1.2.

Instrument lines that penetrate the primary containment conform to RG 1.11 and GDC 55 and 56.

6.2.4.2 System Design

The general criteria governing the design of the primary containment isolation system are provided in Sections 3.1.2 and

6.2.4.1. Table 6.2-56 summarizes the primary containment penetrations and contains information as to:

1. Status open or closed under normal operating conditions and accident situations.

2. Primary and secondary modes of actuation for the isolation valves.

3. Parameters sensed to initiate isolation valve closure.

4. Closure time for principal isolation valves to secure primary containment isolation.

5. Applicable general design criteria.

P&IDs which show the primary containment penetrations along with their associated piping, branch connections, piping safety class, and pressure boundary piping for each system are provided in each specific system description section.

Protection is provided for isolation valves, actuators, and controls against damage from missiles. All potential sources of missiles are evaluated. Where possible hazards exist, protection is afforded by separation, missile shields, or location. See Section 3.5 for a discussion of evaluation techniques.

Isolation valves are designed to be operable under adverse environmental conditions (Section 3.11) such as maximum differential pressures, extreme seismic occurrences, high temperature, and high humidity.

Redundancy and physical separation are provided in the electrical and/or mechanical design to ensure that no single failure in the primary containment isolation system prevents the system from performing its intended functions. Where a penetration is part of a redundant train in an ESF system, isolation valves for that train receive power from a single electrical division. This is necessary so that single failure of an electrical division cannot disable both trains of the ESF system.

The MSIVs are globe valves designed to fail closed on loss of power.

The MSIVs shall be Type C tested in accordance with 10CFR50 Appendix J. The test medium shall be air/nitrogen and the test pressure shall be 40.00 psig.

The design and operation of the MSIVs is described in Section 5.4.5.

It should be noted that all motor-operated isolation valves remain in the as-is position upon failure of valve power. On the

other hand, all air-operated valves (AOVs) (not applicable to air-testable check valves) close on loss of air.

The swing-check valves located on the drywell-to-wetwell vacuum breaker lines cycle open and closed based on the difference in atmospheric pressure between the drywell and wetwell. A pneumatic source is not required in order for these valves to perform their safety function; however, a pneumatic supply is available for test purposes.

The design of the isolation valve as well as the associated system includes consideration of the possible adverse effects of sudden isolation valve closure when the plant systems are functioning under normal operation.

Outside isolation valves are located as close as practical to the primary containment. Except as listed below, outside isolation valves are within 10 ft of the containment wall.

Valve Number	Penetration Number	Pipe Length From Outside Containment
2RHS*MOV1B	Z-5B	20'-9"
2RHS*MOV33A	Z-7A	18'-3"
2CSH*MOV111	Z-13	50'-0"
2CSH*MOV105	Z-13	45'-6"
2DFR*MOV139	Z-43	20'-10"
2CPS*SOV119	Z-59	14'-6"
2MSS*MOV208	Z-1A-1D	36'-0"
2RHS*MOV30A	Z-6B	10'-6"
2RHS*MOV30B	Z-6A	19'-3"
2RCS*V59A	Z-38A	33'-0"
2RCS*V59B	Z-38B	31'-0"
2CMS*SOV26C	Z-61B	15'-0"
2CMS*SOV35A	Z-61C	18'-3"
2ICS*MOV148	Z-90	23'-10"
2ICS*MOV164	Z-90	29'-11"
2WCS*MOV200	Z-4A	57'-8"
2WCS*MOV200	Z-4B	65'-8"
2RHS*V192	Z-90	26'-6"

For the above outboard isolation valves, locations have been established as close as practical to the containment while also satisfying pipe supporting and flexibility requirements, clearing other equipment in the area, and providing access for maintenance, testing, and in-service inspection (ISI). As discussed in Sections 3.5, 3.6, and Appendix 3C, the lines between the containment and the isolation valves are analyzed to ensure that their integrity is maintained against the effects of missiles, pipe whip, and jet impingement loads.

The penetrations shown on Table 6.2-63 involve relief valve discharge headers which combine inputs from several sources into one pipe penetrating to the primary containment. In this manner,

they reduce the amount of piping and number of containment penetrations needed to satisfy system process requirements. In Table 6.2-63, piping lengths from outside the containment are separated into two lengths; first, from the containment isolation valve to the common piping header to the containment penetration. In all cases, relief and safety valves which serve as outside containment isolation valves have been located as close as practical to the containment, considering available piping arrangements and the requirement of ASME Code Section III, Subsection NC-7100, that relieving devices be located as close as practical to the major source of overpressure.

The 12-in lines passing through containment penetrations Z-88A and Z-88B originate from RHS*SV34A and B and RHS*SV62A and B. The layout of the 12-in lines is dictated by the required location of SV34 and SV62, which are positioned on the inlet piping to the RHR heat exchangers. These 12-in headers are protected from the effects of rapid depressurization, caused by condensation of steam vented by SV34 and SV62, by vacuum breakers 2RHS*RVV35A and B and 2RHS*RVV36A and B. These 12-in headers vent the 1-in RHR heat exchanger vent lines (RHS*MOV26A, B and 27A, B), the 3/4-in relief valve discharge vacuum breaker lines (RHS*V19, V20, V117, and V118), and the 1-in RHR heat exchanger relief valve discharge lines (RHS*RV56A, B). The connection of these lines to the 12-in headers does not significantly increase the containment volume beyond that represented by the actual 12-in headers themselves, while it does significantly reduce the number of containment penetrations and the amount of piping required for the several flow paths returning to the suppression pool. Thus, this arrangement locates outside isolation valves as close as practical to the primary containment. As discussed above, all containment penetration piping is analyzed to ensure integrity for its entire length against the effects of missiles, pipe whip, and jet impingement loads.

Whenever relief or safety valves are used as containment isolation valves, their set pressure is at least 1.5 times the design pressure of the primary containment. The one exception to this is valve 2CSL*RV123, with a set pressure of 46 psig. For this valve, only the discharge side is part of the primary containment boundary. Thus, containment pressure during an accident is applied to the discharge side of the valve.

6.2.4.3 Design Evaluation

6.2.4.3.1 Introduction

The primary objective of the primary containment isolation system is to provide protection by preventing releases of radioactive materials to the environment. This is accomplished by automatic isolation of system lines penetrating the primary containment. Redundancy is provided in all design aspects to satisfy the requirement that any active failure of a single valve or component does not prevent primary containment isolation.

Mechanical components, such as isolation valve arrangements, are redundant to provide backup in the event of accident conditions. The arrangements with appropriate instrumentation are described in Table 6.2-56. The isolation valves have redundancy in mode initiation. Generally, the primary mode is automatic and the secondary mode is remote manual. A program of testing (Section 6.2.4.4) is maintained to ensure valve operability and leak-tightness.

The design specifications require each isolation valve to be operable under the most severe operating conditions that it might experience. Each isolation valve is afforded protection, by separation and/or adequate barriers, from the consequences of potential missiles.

Electrical redundancy is provided in isolation valve arrangements; this eliminates dependency on one power source to attain isolation. Electrical cables for isolation valves in the same line have been routed separately. Cable selection is based on the specific environment (Section 3.11) to which they may be subjected, such as high radiation, high temperature, and high humidity.

Administrative control provisions ensure that the position of all nonpowered isolation valves is maintained and known. The position for all power-operated valves is indicated in the main control room. Discussion of instrumentation and controls for the isolation valves is included in Chapter 7.

Containment isolation considerations in the event of a Station Blackout (SBO) are addressed in Section 8.3.1.5.

6.2.4.3.2 Evaluation Against General Design Criteria

Evaluation Against Criterion 55

The RCPB, as defined in 10CFR50, Section 50.2(v), consists of RPV, pressure-retaining appurtenances attached to the vessel, and valves and pipes that extend from the RPV up to and including the outermost isolation valve. The lines of the RCPB that penetrate the primary containment include provisions for isolation of the primary containment, thereby precluding any significant release of radioactivity. Similarly, for lines that do not penetrate the primary containment but form a portion of the RCPB, the design ensures that isolation of the RCPB can be achieved.

Influent Lines Influent lines that penetrate the primary containment and connect directly to the RCPB are equipped with at least two isolation valves, one inside the drywell and the other as close to the external side of the primary containment as practical. Protection of the environment is provided by these isolation valves.

Table 6.2-56 contains those influent pipes that compose the RCPB and penetrate the primary containment.

1. Feedwater Lines The feedwater lines are part of the RCPB as they penetrate the drywell to connect with the RPV. The isolation valve inside the drywell is a Y-pattern check valve, located as close as practical to the primary containment wall. Outside the primary containment is a testable Y-pattern check valve located as close as practical to the primary containment wall. This valve fails closed upon the loss or reversal of fluid flow. Away from the primary containment is a motor-operated gate valve. Should a break occur in the feedwater line, the check valves prevent significant loss of reactor coolant inventory and offer immediate isolation. However, in case a LOCA occurs without a seismic event, the design allows the condensate and feedwater pumps to supply feedwater to the vessel. For this reason, the outermost gate valve does not automatically isolate upon signal from the protection system. The gate valve does meet the same environmental and seismic qualifications as the outboard isolation valve. The valve can be remotely closed from the control room to provide long-term leakage protection upon Operator judgment that feedwater makeup is unavailable or unnecessary. No credit is taken for feedwater flow in assessing core and containment response to a LOCA.

   An analysis was conducted to evaluate the ability of feedwater check valves 2FWS*AOV023 and 2FWS*F012 to withstand rapid closure following a postulated pipe break outside the containment. Since long-term leakage protection is provided by MOV21, the acceptance criterion is that gross leak rates do not occur because of disk rupture, serious fracture of the seat/disk interface, or misalignment of the disk with respect to the seat from this faulted event.

   Following the guidelines of Appendix F of the ASME III Code, inelastic systems and component analyses were conducted using the nonlinear transient option of the ANSYS (Appendix 3A.25) program. Seismic and dead loads were not considered because of their insignificant magnitude compared to impact. The nonlinear stress/strain relationship was approximated by a bilinear curve adjusted for temperature and strain rate effects.

   Stresses in the rock shaft, tail link, seat, and disk were below ASME III Class 1 allowables for faulted conditions. It is concluded that both valves will remain intact and that any leakage will be within the makeup capability of the HPCS or the RCIC system,

following rupture of the feedwater piping outside the containment.

2. HPCS Line The HPCS line penetrates the drywell to inject directly into the RPV. Isolation is provided by an air-testable check valve located inside the drywell with position indicated in the main control room, and a remote manually-actuated gate valve located as close as practical to the exterior wall of the primary containment. Long-term leakage control is maintained by this gate valve. If a LOCA occurred, this gate valve would receive an automatic signal to open.

3. LPCI and LPCS Lines Satisfaction of isolation criteria for the LPCI and LPCS systems is accomplished by use of remote manually-operated gate valves and check valves. Both types of valves are normally closed with the gate valves receiving an automatic signal to open at the appropriate time to assure that acceptable fuel design limits are not exceeded in the event of a LOCA. The check valves are located as close as practical to the RPV. The normally closed check valves protect against primary containment overpressurization in the event of pipe rupture between the check valve and primary containment wall by preventing high-energy reactor water from entering the primary containment. Once the system is in operation, the low energy of the influent fluid (220°F maximum) excludes any possibility of containment overpressurization should a break occur.

4. CRD Lines The CRD system insert and withdraw lines penetrate the drywell; however, these lines are not part of the RCPB since they do not directly communicate with the reactor coolant. The classification of these lines is Quality Group B, and they are, therefore, designed in accordance with ASME Section III, Safety Class 2. The basis to which the CRD insert and withdraw lines are designed is commensurate with the safety importance of maintaining pressure integrity of these lines. See Note 17 of Table 6.2-56 for further discussion.

5. RCIC Line The head spray line penetrates the drywell and discharges directly into the RPV. The testable check valve inside the drywell is normally closed and has position indication lights in the main control room to verify its position. The testable check valve is located as close as practical to the RPV. Two types of valves, a check valve and a remote manual block valve, are located outside the containment. The check valve assures immediate isolation of the primary containment in the event of a line break.

6. SLCS Lines The SLCS line penetrates the drywell and connects to the RPV. In addition to a simple check valve inside the drywell, a check valve and explosive actuated valves are located outside the drywell. Since the SLCS line is a normally closed, nonflowing line, the possibility of rupture of this line is extremely remote. The explosive actuated valves function as third isolation valves. These valves provide an absolute seal for long-term leakage control and prevent leakage of sodium pentaborate into the RPV during normal reactor operation.

7. RWCU System The RWCU pumps, heat exchangers, and filter demineralizers are located outside the drywell. The return line from the filter demineralizers connects to the feedwater line outside the primary containment between the block valve and the outboard primary containment feedwater check valve. Isolation of this line is provided by the feedwater system check valves inside and outside the primary containment. A motor-operated globe valve is provided in the RWCU return line as a third isolation valve. The valve can be remotely closed from the control room to provide long-term leakage protection.

   Should a break occur in the RWCU return line, the feedwater system isolation check valves would prevent significant loss of inventory and offer immediate isolation, while the RWCU return line isolation valve would provide long-term leakage control.

8. Recirculation Pump Seal Water Supply Line The recirculation pump seal water line extends from the recirculation pump through the drywell and connects to the CRD supply line outside the primary containment. The seal water line forms a part of the RCPB. The recirculation pump seal water line is 3/4 in in diameter and is Class B from the recirculation pump through the second check valve (located outside and as close as practical to the primary containment). From this valve to the CRD connection the line is Class D. In a postulated failure of this line, the flow rate through the broken line has been calculated to be substantially less than that permitted for a broken instrument line.

Continued recirculation pump seal purge is required whenever reactor coolant temperature is above 200°F and the pump is not isolated. Three check valves in series, two outside the primary containment, are used to provide containment isolation while permitting seal purge, if available. This design will prevent seal damage during containment isolation events. Therefore, automatic isolation valves are not desirable.

The seal purge lines are continually pressurized (and therefore leak tested) above reactor pressure. Thus, any leakage from these lines would be detected either through the floor drain system monitors or by routine surveillance by plant Operators. In addition, the seal purge pressure is continually monitored by pressure transmitters with control room indication. Therefore, the integrity of these lines is continuously verified.

Effluent Lines Effluent lines that form part of the RCPB and penetrate primary containment are equipped with at least two isolation valves, one inside the drywell and the other outside, located as close to the primary containment as practical. Table 6.2-56 also contains those effluent lines that compose the RCPB and penetrate the primary containment.

1. Main Steam, Main Steam Drain Lines, and RHR Shutdown Cooling Lines The main steam lines extend from the RPV to the main turbine and condenser system, and penetrate the primary containment. The main steam drain lines also penetrate the containment. The RHR steam supply line/RCIC turbine steam line connect to the main steam line inside the drywell and penetrate the primary containment. Isolation is provided by automatically-actuated block valves inside the primary containment for the RHR steam supply line/RCIC turbine steam line. The RHR shutdown cooling effluent line has automatically-actuated block valves for isolation.

2. Recirculation System Sample Lines A sample line from the recirculation system penetrates the drywell. The sample line is 3/4 in in diameter and is designed to ASME Section III, Safety Class 2. Two solenoid-operated valves (SOVs) which fail closed are provided, one inside and one outside located as close to the primary containment as practical.

Conclusion on Criterion 55

In order to assure protection against the consequences of accidents involving the release of radioactive material, pipes forming the RCPB have been shown to provide adequate isolation capabilities on a case-by-case basis. In all cases, a minimum of two barriers were shown to protect against the release of radioactive materials.

In addition to meeting the isolation requirements stated in Criterion 55, the pressure-retaining components that compose the RCPB are designed to minimize the probability or consequences of an accidental pipe rupture. The quality requirements for these components ensure that they are designed, fabricated, and tested to the highest quality standards of all reactor plant components. These components are designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Safety Class 1, except for piping and valves smaller than 1 in which are Safety Class 2.

It is concluded that the design of the piping system that composes the RCPB and penetrates containment satisfies Criterion 55. For further discussion, see the following:

1. Quality Group Classification, Table 3.2-1.

2. Containment and Reactor Vessel Isolation Control System, Section 7.3.1.1.2.

Evaluation Against Criterion 56

Criterion 56 requires that lines that penetrate the primary containment and communicate with the primary containment interior must have two isolation valves, one inside the primary containment and one outside, unless it can be demonstrated that the primary containment isolation provisions for a specific class of lines are acceptable on some other basis.

Table 6.2-56 includes those lines that penetrate the primary containment and connect to the drywell and suppression chamber. Although a word-for-word comparison with Criterion 56 is not practical in some cases, adequate isolation provisions are demonstrated on a well-defined basis.

Influent Lines to Suppression Pool

1. LPCS, HPCS, and RHR Test Lines The LPCS, HPCS, and RHR test lines have test isolation capabilities commensurate with the importance to safety of isolating these lines. The HPCS line has a normally closed remote manually-actuated motor-operated valve (MOV) located outside the primary containment. Primary containment isolation requirements are met on the basis that the test lines are normally closed, low-pressure lines constructed to the same quality standards as the primary containment. Furthermore, the consequences of a break in these lines result in no significant safety consideration. The LPCS and RHR test return line has a normally open remote manually-actuated isolation valve.

   The test return lines are also used for suppression chamber return flow during other modes of operation. In this manner the number of penetrations are reduced, minimizing the potential pathways for radioactive material release. Typically, pump minimum flow bypass lines join the respective test return lines upstream of the test return isolation valve. The minimum flow bypass lines are isolated by MOVs.

2. RCIC Turbine Exhaust and RCIC Pump Minimum Flow Bypass These lines, which penetrate the containment and discharge to the suppression pool, are equipped with motor-operated, remote manually-actuated isolation valves located as close to the primary containment as

possible. In addition, there is a simple check valve upstream of the isolation valve that provides positive actuation for immediate isolation in the event of a break upstream of this valve. The gate valve in the RCIC turbine exhaust is normally open and interlocked to prevent opening of the inlet steam valve to the turbine while the turbine exhaust valve is not in a full open position. The RCIC pump minimum flow bypass line is isolated by a normally closed globe valve with a check valve installed upstream.

3. RHR Heat Exchanger Vent Lines The RHR heat exchanger vent lines discharge to the RHR safety relief valve discharge line (SRVDL) which in turn discharges to the suppression pool. The vent lines are isolated from the relief valve discharge line by two remote-controlled motor-operated globe valves. These isolation valves are normally closed and valve positions are indicated in the main control room to provide the Operator with the indication of valve status.

4. RHR Relief Valve Discharge Lines The RHR relief valve discharge to the suppression pool has no valve other than the relief valve. This relief valve will not be opened during normal operation and therefore can be considered as normally closed and adequate under the same criteria as the suppression chamber spray line.

5. RCIC Turbine Exhaust Vacuum Breaker System Lines This line has two automatic MOVs and two check valves. This line runs between the suppression pool air space and the RCIC turbine exhaust line downstream of the exhaust line check valve. Positive isolation is automatic via a combination of low reactor steam pressure and high drywell pressure. The vacuum breaker complex is placed outside the primary containment where there is a more desirable environment. In addition, the valves are readily accessible for maintenance and testing.

Effluent Lines from Suppression Chamber The RHR, RCIC, LPCS, and HPCS suction lines contain motor-operated, remote manually-actuated valves that ensure isolation of these lines in the event of a break. These valves also provide long-term leakage control. In addition, the suction piping from the suppression chamber is considered an extension of containment since it must be available for long-term usage following a design basis LOCA and, as such, is designed to the same quality standards as the primary containment. Thus, the need for isolation is conditional. The ECCS discharge line fill system (ECCS water leg pumps) takes suction from the respective ECCS pump effluent line from the suppression pool downstream of the isolation valve. The ECCS discharge line fill system suction line has a manual valve for operational purposes. This system is isolated from the primary containment by the respective ECCS pump

suction valve from the suppression pool, as listed in Table 6.2-56.

Each ECCS pump room is provided with leak detection capabilities, as discussed in Section 9.3.3.3. If leakage from a seal or gasket is detected in one of the pump rooms during normal plant conditions, the remotely-operated valve installed in the pump suction line would be closed, thereby isolating the leaking component from the suppression pool water. Between the isolation valve and the penetration there is a manual valve in the low-pressure core spray (CSL) and RHR systems. These valves are welded on the suppression pool side and gasketed on the pump side. The suction line from the penetration to the ECCS pump rooms is inside flood troughs which carry all leakage to the pump rooms. Leakage from this valve gasket would be minimal. However, if the leakage became significant, the water would be detected in the ECCS pump room by the leak detection system (LDS). After detection, the manual valve could be closed and the gasket repaired prior to any significant loss of suppression pool water.

The only potential path for leakage of suppression pool water into the ECCS pump rooms is through the pump suction lines, as these are the only lines that penetrate the containment at an elevation below the suppression pool water level.

The need to size the ECCS pump room so that the volume of suppression pool water needed to fill the ECCS pump room would not reduce the suppression pool level below the minimum drawdown line is not required. This is due to the leak detection, isolation, and repair capabilities incorporated into the design. The potential reduction in suppression pool water inventory before detection, isolation, and repair of a leaking gasket in the pump room would be insignificant. Suppression pool makeup during normal plant conditions is from the CST.

The elevations of the ECCS pump suction centerlines and the suppression pool minimum drawdown level are 195'-0" and 197'-8", respectively.

Influent and Effluent Lines from Drywell and Suppression Chamber Free Volume

1. Primary Containment Purge Lines The drywell and suppression chamber purge lines have isolation capabilities commensurate with the importance to safety of isolating these lines. Each line has two normally closed/fail closed valves - one located inside (nitrogen operated) and one located outside (air operated) the primary containment. The inboard end of each 12-in and 14-in valve located inside the primary containment is provided with a QA Category I debris screen to prevent entrainment of foreign matter in the valve seat. The isolation valves are interlocked to

preclude opening of the valves while a primary containment isolation signal exists (Table 6.2-56). The radiological consequences of a LOCA occurring during containment purge system (CPS) operation (isolation valves wide open) with the SGTS in the pressure control mode are discussed in Section 15.6.5.

2. Primary Containment Atmosphere Monitoring System Sampling Lines The primary containment atmosphere monitoring system consists of radiation and hydrogen/oxygen monitoring lines. Each line, suction, and discharge penetrates the primary containment and continuously monitors the radiation level and hydrogen/oxygen concentration during normal operation. These lines are equipped with two solenoid-operated isolation valves, one inside the primary containment and the other outside, located as close as possible to the primary containment. The hydrogen/oxygen monitoring lines are also used to continuously monitor the primary containment air during the post-LOCA period. Each isolation valve receives isolation signals. The isolation valves for hydrogen/oxygen monitoring lines are provided with individual keylock switches to override the isolation signal and initiate system operation during the post-LOCA period.

3. Suppression Chamber Spray Lines The suppression chamber spray lines penetrate the primary containment to remove energy by condensing steam and cooling noncondensable gases in the suppression chamber. The line is equipped with a normally closed MOV located outside and as close as possible to the primary containment. This normally closed valve receives an automatic isolation signal. Primary containment isolation requirements are met on the basis that the spray header injection lines are normally closed, and the lines are constructed to the same quality standards as the primary containment.

4. Drywell-to-Wetwell (DW-WW) Vacuum Relief Lines The four DW-WW vacuum relief penetrations are each equipped with two positive closing swing check valves. The air operator on the swing check valve is used only for testing. Swing check valve operation is initiated by the pressure difference between the drywell and wetwell.

5. Reactor Building Closed Loop Cooling Water The RBCLCW lines penetrate the drywell to provide cooling water to the recirculation pumps and motors and drywell unit coolers. Each line that penetrates the primary containment has an automatic isolation valve inside and outside the containment. These valves also have remote manual operation capability from the control room.

As discussed above, the following penetrations rely on a single isolation valve and are closed system outside the containment.

Penetration No.	Description
Z-5A, B, and C	RHS pump suction from suppression pool
Z-6A and B	RHS test return line to suppression pool
Z-7A and B	RHS containment spray to suppression pool
Z-12	HPCS pump suction from suppression pool
Z-13	HPCS test return and minimum flow bypass to suppression pool
Z-15	LPCS pump suction from suppression pool
Z-17	RCIC suction from the suppression pool
Z-18	RCIC minimum flow to suppression pool
Z-19	RCIC turbine exhaust
Z-73	RHS relief valve discharge to suppression pool
Z-88A and B	RHS safety valve discharge to suppression pool
Z-98A and B	RHS relief valve discharge to suppression pool

System piping and valves outside the containment, which are a part of the closed system boundary, are of Category I, Safety Class 2 design; are protected from missiles; and have design temperature and pressure ratings at least equal to that for the containment. Branch lines from the closed system are valved closed and procedurally controlled. In case of an active failure of the isolation valve, primary containment leakage would be contained within the closed system boundary. System reliability is enhanced by the simplicity of design obtained with the use of a single isolation valve outside the containment by reducing the number of possible active failures. Those lines which connect to the suppression pool do not have an isolation valve located inside the primary containment, as this would necessitate placement of the valve underwater. In effect, this would introduce an unreliable design element into a system requiring high reliability.

Conclusion on Criterion 56

In order to ensure protection against the consequences of accidents involving release of significant amounts of radioactive

materials, pipes that penetrate the primary containment have been demonstrated to provide isolation capabilities on a case-by-case basis in accordance with Criterion 56. In addition to meeting isolation requirements, the pressure-retaining components of these systems are designed to the same quality standards as the primary containment.

Evaluation Against Criterion 57

Lines penetrating the primary containment for which neither Criterion 55 nor Criterion 56 governs compose the closed system isolation valve group. Influent and effluent lines of this group are isolated by automatic or remote manual isolation valves located as close as possible to the primary containment boundary. The reactor recirculation system hydraulic control lines to the flow control valve contain an isolation valve located outside the drywell that closes automatically upon receipt of its isolation signal. The hydraulic lines and their isolation valves are discussed in Note 26 of Table 6.2-56.

Evaluation Against Regulatory Guide 1.11

Instrument lines that penetrate the primary containment from the RCPB are equipped with a restricting orifice located inside the drywell and an excess flow check valve located outside and as close as practical to the primary containment, in accordance with RG 1.11. Those instrument lines that do not connect to the RCPB are equipped with isolation valves whose status is indicated in the control room in accordance with RG 1.11.

Traversing in-core probe (TIP) subsystem guide tubes are classified as instrument lines in accordance with RG 1.11. The justification for this classification is provided in GE NEDC-22253, BWROG Evaluation of Containment Isolation Concerns, October 1982. The TIP guide tubes have an isolation valve that closes remote manually after the TIP cable and fission chamber have been retracted. An additional or backup isolation shear valve is included in series with this isolation valve. Both valves are located outside the drywell. The TIP system and isolation provisions are discussed in Note 19 of Table 6.2-56.

6.2.4.3.3 Failure Modes and Effects Analysis

A single failure can be defined as a failure of some component in any safety system that results in a loss or degradation of the system's capability to perform its safety function. Active components are defined in RG 1.48 as components that must perform a mechanical motion during the course of accomplishing a system safety function. Appendix A to 10CFR50 requires that electrical systems be designed against passive single failures as well as active single failures. Chapter 3 describes the implementation of these standards as well as GDC 17, 21, 35, 41, 44, 54, 55, and 56.

In single-failure analysis of electrical systems, no distinction is made between mechanically active or passive components; all fluid system electrically-operated components such as valves are considered electrically active whether or not mechanical action is required. Electrical systems as well as mechanical systems are designed to meet the single-failure criterion for both mechanically active and passive fluid system components regardless of whether the component is required to perform a safety action or not. Even though a component such as an electrically-operated valve is not designed to receive a signal to change state (open or closed) in a safety scheme, it is assumed as a single failure that the system component changes state or fails. Electrically-operated valves include valves such as solenoid valves and solenoid-actuated AOVs or valves that are directly operated by an electrical motor. In addition, all electrically-operated valves that are automatically actuated can also be manually actuated from the main control room. Therefore, a single failure in any electrical system is analyzed regardless of whether the loss of a safety function is caused by a component failing to perform a requisite mechanical motion or a component performing an unnecessary mechanical motion.

6.2.4.3.4 Operator Actions

A trip of an isolation control system channel is annunciated in the main control room so that the Operator is immediately informed. All motor-operated and air-operated isolation valves have open-close status lights. The following general information is presented to the Operator by the isolation system:

1. Annunciation of each process variable that has reached a trip point.

2. Computer readout of trips on main steam line tunnel temperature or main steam line excess flow.

3. Control power failure annunciation for each channel.

4. Annunciation of steam leaks in each of the systems monitored (main steam, RWCU, and RHR).

The leakage detection system detects possible leakage from lines inside/outside containment and provides the Operator in the main control room with information required to isolate fluid systems equipped with remote manual isolation valves. Parameters used to detect leakage are high radiation, high area temperature, high sump level, and RPV level and pressure as discussed in Sections 5.2.5.1.3, 7.6.1.3, and 12.3.4.1. System parameters such as flow, pressure, and temperature are indicated and/or alarmed in the main control room. These enable the Operator to detect degraded system performance attributable to system leakage and take appropriate action to isolate systems that are potential leakage paths.

This information will enable the Operator to decide if he needs to operate a remote manual valve in the event of a LOCA.

6.2.4.4 Tests and Inspections

The primary containment isolation system is scheduled to undergo periodic testing during reactor operation. The functional capabilities of power-operated isolation valves are tested manually from the main control room. By observing position indicators and changes in the affected system operation, the closing ability of a particular isolation valve is demonstrated. A discussion of testing and inspection pertaining to isolation valves is provided in Section 6.2.6 and Technical Specifications. Table 6.2-56 lists all isolation valves.

Instruments will be periodically tested and inspected. Test and/or calibration points will be supplied with each instrument.

Excess flow check valves will be periodically tested by opening a test drain valve downstream of the excess flow check valve and verifying proper operation. Preoperational testing is discussed in Section 14.2.12.

Containment isolation valve leak testing is discussed in Section 6.2.6.

Leakage testing of the closed ESF systems outside containment is performed in accordance with Section XI of the ASME Code, as discussed in Sections 6.6 and 3.9.6. Any airborne radioactivity resulting from leakage from these ESF systems following a LOCA is processed through the SGTS prior to discharge to the environment. The offsite doses from this source are small. This contribution has been accounted for in the radiological assessment of the site. Section 15.6.5.5.3 and Table 15.6-13 discuss the methodology and assumptions used in determining the radiological consequences of leakage from the primary containment and from ESF systems following a LOCA.

Additional requirements for the PCRVICS will be provided in accordance with Section 1-10, Task II.E.4.2.

6.2.5 Combustible Gas Control in Containment

To assure that the primary containment integrity is not endangered by generation of combustible gases following a postulated LOCA, the primary containment (drywell and suppression chamber) will be inerted with nitrogen (Section 1.10). Systems for controlling the relative concentrations of oxygen and hydrogen are provided within the plant. The system includes subsystems for mixing the primary containment atmosphere, monitoring oxygen and hydrogen concentrations, and reducing oxygen and hydrogen concentrations without relying on primary containment purging to the environment. The primary containment

purge system is available to aid in postaccident cleanup operations.

6.2.5.1 Design Bases

The following design bases were used for the combustible gas control system (CGCS) design:

1. The CGCS is designed to limit the oxygen or hydrogen concentration to 5 volume percent within the primary containment following a LOCA.

2. A recombiner mixes the drywell atmosphere and the suppression chamber atmosphere. Prior to initiation of the recombiner, the drywell and the suppression chamber will be mixed uniformly due to natural convection and molecular diffusion. Mixing will be further promoted by operation of the containment sprays. The Operator actuates the containment sprays within 30 min after the LOCA. The criteria for the operation of containment sprays is specified in Section 6.2.1.1.

3. The recombiners will be started manually by the Operator prior to either the hydrogen concentration reaching 4 volume percent or oxygen concentration reaching 4.5 volume percent in the drywell or the suppression chamber. An alarm is provided to alert the Operator to these conditions.

4. Two identical Category I recombiners are provided to limit oxygen or hydrogen concentration. Operation of either recombiner will limit combustible gas concentration to a safe value.

5. The components of the CGCS are protected from missiles and pipe whip to assure proper operation under accident conditions as required for safety-related systems. The recombiners and monitors are located outside the primary containment.

6. The components of the CGCS are designed as Category I and Safety Class 2.

7. All components that are subjected to primary containment atmosphere will be capable of withstanding the humidity, temperature, pressure, and radiation conditions in the containment following a LOCA.

8. The CGCS can be inspected or tested during normal plant conditions.

9. The recombiners are located in the reactor building.

10. The primary containment purge system is provided to aid in the postaccident cleanup operation. The primary containment atmosphere can be purged through the SGTS to the outside environment. Nitrogen makeup will also be available during the purging operation.

6.2.5.2 System Design

The CGCS provides effective control over hydrogen and oxygen generated following a LOCA. The system consists of the following features:

1. Atmospheric mixing is achieved by natural processes. Mixing could be enhanced by operation of the containment sprays, which are used to depressurize the primary containment. The criteria for the manual operation of the containment sprays are provided in Section 6.2.1.1.

2. One of the two 100-percent capacity hydrogen recombiners is manually initiated after a LOCA to preclude the oxygen and hydrogen concentration from exceeding 5 volume percent.

3. The primary containment nitrogen inerting system establishes and maintains an oxygen-deficient atmosphere ($\leq$4 volume percent) in the primary containment during normal operation.

4. The redundant hydrogen and oxygen analyzer system measures hydrogen and oxygen in the drywell and suppression chamber during normal operation and LOCA conditions. Hydrogen and oxygen concentrations are displayed in the main control room. Piping and instrumentation for this system are shown on Figure 6.2-71. Safety-related display instrumentation for containment monitoring is listed in Table 7.5-1.

5. All controls for operating the CGCS (i.e., hydrogen recombiner system (HCS) and monitoring system) are located in the main control room.

6. A tabulation of the design and performance data for each system component is listed in Table 6.2-57.

7. Environmental qualification information for safety-related equipment is given in Section 3.11.

8. Electrical requirements for equipment associated with this system are in accordance with IEEE Class 1E standard.

The CGCS is considered an extension of the primary containment in post-LOCA conditions and consequently will be included within the

boundary of the Type A test (Section 6.2.6). The DBA HCS system meets the criteria of Standard Review Plan (SRP) 6.2.3 for closed loop systems as follows:

1. Containment atmosphere does not directly communicate with the environment following a LOCA.

2. Designed in accordance with Quality Group B standards.

3. Meets Category I design requirements.

4. Is designed to primary containment pressure and temperature design conditions as applicable.

5. Is designed for protection against pipe whip, missiles, and jet forces.

6. Is tested for leakage.

6.2.5.2.1 Atmospheric Mixing

The function of post-LOCA mixing in the drywell and suppression chamber is performed by the primary containment spray system, recombiner system, and natural processes. At approximately 30 min following the postulated accident, the redundant containment spray systems in the drywell and suppression chamber can be initiated to depressurize the containment. The turbulence induced by the spray ensures a well-mixed primary containment atmosphere. In addition to the spray system, the blowdown of steam and water through the broken pipe creates a large degree of turbulence and promotes mixing of the entrained hydrogen and oxygen with the primary containment atmosphere. The natural convection currents arising from temperature differences between the atmosphere and primary containment walls, and diffusion will promote a well-mixed atmosphere and prevent hydrogen and oxygen stratification. With the above mixing capabilities, there is minimal potential for a nonuniform hydrogen and oxygen concentration within the primary containment.

The atmosphere between the drywell and suppression chamber will be mixed during the depressurization phase of the LOCA. When activated the recombiner units will also serve to effect mixing between these two compartments. The recombiner will draw air from the drywell and/or the suppression chamber and discharge to the suppression chamber. This will in turn cause the atmosphere from the suppression chamber to circulate into the drywell via the vacuum breaker lines.

There are three interior subcompartments where gases may not achieve thorough mixing with the bulk of the primary containment atmosphere. The drywell head area, used for reactor vessel refueling purposes, is one such subcompartment. There are no sources of oxygen in this area; therefore, local oxygen concentration is not expected to be greater than the bulk oxygen

concentration. The other two subcompartments are the CRD area in the drywell and the volume enclosed by the pedestal wall in the suppression chamber. Due to the large open area between these two subcompartments and the bulk atmosphere, significant concentration gradients are unlikely.

6.2.5.2.2 Hydrogen Recombiner System

The long-term control of hydrogen and oxygen is achieved by means of two identical 150-scfm thermal hydrogen recombiners, located in the reactor building and controlled from the main control room. The recombiner system removes gas from the drywell or suppression chamber, recombines the hydrogen with oxygen, and returns the gas mixture along with the condensate to the suppression chamber. Flow from the suppression chamber atmosphere to the drywell through the vacuum breakers prevents the suppression chamber pressure from exceeding the drywell pressure by more than 0.25 psi.

Operation of any one recombiner will provide effective control over combustible gases within primary containment. Figure 6.2-72a and b shows the piping and instrumentation diagram (P&ID) of the recombiner system. The manufacturer of the hydrogen recombiner is the Atomics International Division, Energy Systems Group of Rockwell International.

The recombiner unit is skid mounted and is an integral package. All pressure-containing equipment including piping between components is considered an extension of the containment and, therefore, is designed to ASME Section III, Safety Class 2 requirements. The skid and the equipment mounted on it are designed to meet Category I requirements.

The recombiner unit consists of a blower, electric heater, reaction chamber, and water spray cooler. The reaction chamber is capable of processing 150 scfm of gas containing up to either 2 1/2 volume percent of oxygen and unlimited excess hydrogen or 5 volume percent of hydrogen with excess oxygen. Under these conditions, recombination efficiency is virtually 100 percent. The recombiner is not designed to operate when hydrogen concentration exceeds 5 volume percent with excess oxygen.

The recombination process takes place within the recombiner as a result of high temperature. The resulting water vapor is then cooled along with other gases and returned to the suppression chamber.

The recombiner unit, which requires a 1 1/2-hr warmup period, is initiated manually from the control room prior to either hydrogen concentration reaching 4 volume percent or oxygen concentration reaching 4.5 volume percent. This occurs for the hydrogen concentration, approximately 2 days after the DBA. Once placed in operation, the system continues to operate until it is

manually shut down when an adequate margin below the hydrogen or oxygen concentration design limit is reached.

The operation of the system can be tested from the control room. The test consists of energizing the blower and heaters and observing system operation to see if components are performing properly. Flow and pressure measurement devices are periodically calibrated.

Cooling water required for operation of the system after a LOCA is taken from the SWP system after passing through 100-mesh size, Y-type strainers. Demineralized water from the makeup water system is used for functional testing of the recombiner units. The cooling water is used to cool the water vapor and the residual gases leaving the recombiner prior to returning them to the primary containment.

During normal operation the recombiner system will be maintained in an inerted condition with nitrogen, ready for immediate startup.

6.2.5.2.3 Primary Containment Nitrogen Inerting System

Oxygen control within primary containment during normal plant operation is achieved by means of the nitrogen inerting system. During normal plant operation, oxygen concentration is maintained at or below 4 volume percent using this system.

The system is designed to supply nitrogen to the primary containment for initial inerting and for makeup during normal operation.

6.2.5.2.4 Primary Containment Purge

Primary containment purge capability is provided in accordance with RG 1.7 and as an aid in cleanup following an accident. This function is fulfilled by the combined operation of the CPS and the SGTS.

During normal plant operation, the CPS also functions, in conjunction with the nitrogen inerting system (GSN) and the SGTS, to maintain primary containment pressure at about 0.75 psig and oxygen concentration at or below 4 percent by volume. This is accomplished by injecting the required quantity of nitrogen into the primary containment through the CPS and/or extracting the required volume of gas through the CPS exhaust. The exhaust flow is routed through piping to the SGTS, where it passes through the SGTS filters and a radiation monitor before being released from the plant stack to the environment. All CPS primary containment isolation valves are automatically closed after 15 sec when a high radiation level is detected in the exhaust flow. This time delay of 15 sec prevents automatic closure of CPS primary containment isolation valves due to spurious power transients.

The CPS P&ID (Figure 9.4-8) shows the piping and instrumentation used in this mode of operation.

6.2.5.2.5 Hydrogen and Oxygen Monitoring System

The hydrogen and oxygen concentrations are monitored by the two fully-independent hydrogen/oxygen analyzer trains. The redundant system design ensures that the volumes are sampled in the event of the functional failure of one of the analyzer trains. The location of oxygen and hydrogen sample points within the drywell and the suppression chamber are provided in Table 6.2-59A. These sampling points are distributed vertically and radially throughout the drywell and suppression chamber. Structures and equipment within the region of the sampling points are listed in Table 6.2-59B. Following an accident, this system will be activated manually to monitor combustible gas concentrations. After activation, the system continuously monitors the primary containment hydrogen and oxygen concentration by drawing samples from five different areas: three from the drywell and two from the suppression chamber. For the drywell samples, the sample source and the return points are selected by a sequencing timer that controls the opening and closing of SOVs in the sample and return lines. All the samples drawn are returned to their origins. When the sequencing timer is utilized, each sample valve in the drywell remains open for 20 min.

For the suppression chamber, the sample source and the return point are selected manually. The sample is drawn from both areas simultaneously, combined, and then analyzed by the hydrogen/oxygen analyzer and returned to the suppression chamber.

The accuracy of the hydrogen and oxygen analyzer is ±5 percent of full scale, and the 90-percent response time to sample the concentration is less than 60 sec. Both hydrogen and oxygen analyzers are supplied with two range readouts. The hydrogen analyzer has 0-10 and 0-30 percent ranges, and the oxygen analyzer has 0-10 and 0-25 percent ranges.

6.2.5.3 Design Evaluation

The Unit 2 primary containment atmosphere will be inerted with nitrogen during normal operation of the plant. Oxygen concentration within the primary containment will be maintained at or below 4 volume percent (based on noncondensable gases). Following an accident, oxygen and hydrogen concentrations will be controlled by means of the recombiner system.

In evaluating the CGCS design, it is necessary to consider:

1. Oxygen and hydrogen sources in a postaccident environment.

2. Distribution of oxygen and hydrogen in the drywell and the suppression chamber.

3. Primary containment pressure and temperature during the containment cooldown phase of the accident.

6.2.5.3.1 Sources of Oxygen and Hydrogen

Short-Term Hydrogen Generation

In the period immediately after the LOCA, hydrogen is generated by both radiolysis and metal-water reaction. However, the short-term contribution from radiolysis is insignificant compared to that of the metal-water reaction. The metal-water reaction of steam with the zirconium fuel cladding which produces hydrogen is:

$$Zr + 2H_2O \rightarrow ZrO_2 + 2H_2 \qquad (6.2\text{-}14)$$

Based on LOCA calculational procedures and analysis of ECCS performance in conformance with 10CFR50.46 and Appendix K of 10CFR50, the extent of the chemical reaction is estimated to be 0.1 percent of the fuel cladding material. The metal-water reaction generated hydrogen based on a core-wide penetration of 0.00023 in results in a metal-water reaction of 0.8 percent of the fuel clad material that is greater than five times the calculated value of 0.1 percent (0.5 percent). Therefore, 0.8 percent of the fuel cladding is assumed to react with water to produce hydrogen in accordance with RG 1.7. The duration of this reaction is assumed to be 120 sec with a constant reaction rate. The resulting hydrogen is assumed to be uniformly distributed in the drywell. Figures 6.2-72D and 6.2-72E show hydrogen generation rates and integrated values as a function of time following the accident.

Short-Term Oxygen Source

The only source of air addition to primary containment is the operation of relief valves inside the primary containment. These relief valves are part of the breathing and service air systems, and are normally isolated during reactor operation. Due to high temperature following a LOCA inside primary containment, a portion of these systems (inside primary containment) becomes pressurized and relieves pressure by expelling about 126 std cu ft of air into the primary containment.

The primary containment does not have any provision for storing portable air packs for breathing. The operating procedures would have appropriate controls for the use of portable air packs.

The ADS valves are nitrogen operated; therefore, operation of these valves will not result in addition of oxygen in the primary containment. The short-term oxygen source has not been considered in the oxygen concentration evaluation, as it is very small.

Long-Term Hydrogen/Oxygen Generation

Hydrogen and oxygen are produced by decomposition of water due to absorption of the fission product decay energy immediately after a LOCA. Generation of hydrogen and oxygen due to radiolysis of core cooling water is an important factor in determining the long-term gas mixture composition within the primary containment. A fission product distribution model, as outlined in RG 1.7, is used to calculate hydrogen/oxygen generation rates. The in-core radiolysis (due to core gammas) contributes hydrogen and oxygen to the drywell, and radiolysis due to fission products contributes hydrogen and oxygen directly to the suppression chamber and the drywell atmospheres. The division of hydrogen and oxygen between the suppression chamber and the drywell depends upon the fraction of water holdup on the drywell floor and water in the reactor vessel.

Hydrogen can also be formed by corrosion of metals and decomposition of organic materials in the primary containment. The significant portion of this source is from the corrosion of zinc, which is included in the analysis. The temperature-dependent hydrogen production rate is based on NUREG/CR-2812[4]. The temperature-dependent hydrogen generation rate for demineralized water is shown in Table 6.2-59C. The galvanized steel and zinc primer surface areas exposed to sprays are shown in Table 6.2-59D. The surface area used in the analysis is approximately 15 percent higher than tabulated values. The corrosion of aluminum in demineralized water is very small. The Griess and Creek[5] test data suggest the hydrogen production rate to be between 4.76×10^{-5} to 3.23×10^{-3} std cu ft of H_2 per sq ft/hr. Assuming that the corrosion in the Griess and Creek test is mainly due to 285°F and 212°F water temperature, the average rate is 4×10^{-4} std cu ft of H_2 per sq ft/hr. Considering the aluminum surface area directly exposed to the spray environment and the above H_2 generation rate, a total of 250 scf of hydrogen would be evolved within 20 days following a LOCA. Since this source is small compared to other sources of hydrogen, aluminum corrosion and associated hydrogen production is ignored in the analysis.

Figures 6.2-72D through 6.2-72G show hydrogen and oxygen generation rates and integrated values. The quantity of hydrogen initially contained within the RCS is negligible; hence, it is neglected.

6.2.5.3.2 Accident Description

Following the postulated recirculation suction line DER, the metal-water reaction begins in the core region and produces hydrogen immediately. The reaction is assumed to last 2 min, during which 0.8 percent of the active zircaloy fuel cladding reacts. The radiolysis of coolant in the core region, water on the drywell floor, and suppression pool water begins immediately.

The hydrogen and oxygen thus generated evolve to the drywell and suppression chamber atmospheres.

The combustible gases in the drywell and the suppression chamber would approach the flammability limit, if uncontrolled, after 3.75 days. Prior to this, pressure and temperature within the primary containment are shown by analysis (Section 6.2.1) to have dropped to a level that will permit operation of the recombiner. The recombiner system is manually activated when oxygen or hydrogen concentration reaches the limits described in Section 6.2.5.1. The recombiner system takes suction from the primary containment atmosphere, recombines the hydrogen and oxygen to form water vapor, and returns the exhaust to the suppression chamber. This results in a small pressure buildup in the suppression chamber that causes the opening of the vacuum breaker valves between the drywell and suppression chamber. As a result, the flow of the gas mixture from the suppression chamber to the drywell is established. This arrangement of recombiner suction and discharge promotes mixing of the two volumes in the primary containment.

6.2.5.3.3 Analysis

Based on the preceding hydrogen and oxygen generation sources and the accident description, the oxygen and hydrogen concentration in the drywell and suppression chamber is obtained as a function of time. However, the analysis conservatively assumes that the recombiner system is manually activated prior to either the hydrogen concentration reaching 4 volume percent or the oxygen concentration reaching 4.5 volume percent. To calculate the redistribution of the hydrogen and oxygen between the drywell and suppression chamber, a two-region computer model of the primary containment system is used. This model takes into consideration hydrogen and oxygen generation from the metal-water reaction and radiolysis. The calculation determines the inventory, partial pressure, and mole fraction of each atmospheric constituent in both regions as a function of time.

Tables 6.2-58, 6.2-59, 6.2-59C, and 6.2-59D present the parameters used in the analysis of the oxygen and hydrogen buildup within the primary containment. The minimum recombiner flow necessary to control the formation of combustible gases is 120 scfm. Although the recombiner has a design processing capacity of 150 scfm, the analysis to determine postaccident hydrogen and oxygen concentrations within primary containment uses the 120 scfm flow. The hydrogen and oxygen concentration transient plots are shown on Figure 6.2-72H and 6.2-72I.

Operation of the recombiner at 150 scfm, as opposed to 120 scfm, would reduce at a faster rate the postaccident concentrations of combustible gases in primary containment.

6.2.5.3.4 Failure Modes and Effects Analysis

The FMEA for the CGCS is contained in the Unit 2 FMEA document.

6.2.5.4 Tests and Inspections

Each active component of the CGCS is testable during normal reactor power operation. This system will be tested periodically to assure that it will operate correctly whenever required. Preoperational tests of the CGCS are conducted during the final stages of plant construction prior to initial startup. These tests assure correct functioning of all controls, instrumentation, recombiners, piping, and valves. System reference characteristics such as pressure differentials and flow rates are documented during the preoperational tests and will be used as base points for measurement in subsequent operational tests.

During normal operation, the recombiner system piping, valves, instrumentation, wiring, and other components can be inspected visually at any time, since they are outside the primary containment. Further information may be found in Chapter 14.

6.2.5.5 Instrumentation Requirements

Description

Safety-related instruments and controls are provided for automatic and manual control of the hydrogen recombiners. The controls and monitors described below are located in the main control room. The control logic is shown on Figure 6.2-72K.

Instrumentation requirements for the CPS and the SGTS portions of the CGCS are described in Sections 9.4.2.5 and 6.5.1.5, respectively.

Operation

The hydrogen recombiner inlet and outlet isolation valves close automatically on a LOCA or manual isolation signal and can be opened manually during a LOCA by means of the associated hydrogen recombiner LOCA override keylock switch.

The redundant cooling water block valves located in the water supply lines are manually operated. These valves are interlocked with the recombiner discharge line containment isolation valves so that they cannot be opened unless the isolation valves are already open. They will also automatically close if the isolation valves are closed.

The strainer blowdown drain valves are interlocked with the redundant cooling water block valves. In the automatic mode, the blowdown drain valves close when the associated block valve is

opened and will open when the block valves close. The blowdown valves can also be closed manually and opened manually.

Recombiner cooling water inlet valves close automatically when the associated recombiner unit is turned off. The air inlet valves are manually closed after the recombiner unit is turned off and they will stop when the associated control switch is released.

Recombiner gas heaters and the gas blower are turned on manually, after which the reaction chamber temperature is automatically controlled by the SCR controller. Temperatures are set at manual/automatic stations. Interlocks prevent operation of the recombiner when its cooling water inlet and block valves are less than fully open, when through gas flow is low, when heater gas inlet or outlet temperature is high, or when high temperature or pressure conditions prevail. Gas blowers are turned off under recombiner high temperature or pressure conditions.

Monitoring

Indicators are provided for each recombiner for each of the following parameters:

1. Blower inlet temperature.
2. Heater wall temperature.
3. Reaction chamber shell temperature.
4. Return gas temperature.
5. Reaction chamber temperature.
6. Inlet temperature.
7. Heater inlet gas temperature.
8. Heater outlet gas temperature.
9. Inlet pressure.
10. Through gas flow.

Alarms are provided for each of the following conditions:

1. Hydrogen recombiner system inoperable.
2. Recombiner blower inlet temperature high.
3. Recombiner heater wall temperature high.
4. Recombiner heater wall temperature high/high.

5. Recombiner reaction chamber shell temperature high.

6. Recombiner reaction chamber shell temperature high/high.

7. Recombiner return gas temperature high.

8. Recombiner inlet pressure high.

9. Recombiner reaction chamber temperature high.

10. Recombiner reaction chamber temperature low.

11. Recombiner through gas flow low.

12. Recombiner heater gas inlet temperature high.

13. Recombiner heater gas outlet temperature high.

14. Primary containment isolation valve LOCA override.

15. Recombiner containment isolation valves motor overload.

6.2.6 Containment Leakage Testing

This section presents the proposed testing program for the primary containment, containment penetrations, and containment isolation barriers that comply with the requirements of the general design criteria and Appendix J, Option B, to 10CFR50. Compliance with RG 1.163 (September 1995), entitled "Performance-Based Containment Leak-Test Program," is described in Table 1.8 of the USAR, the Appendix J Testing Program Plan and the Technical Specifications. Each of the tests described in this section will be performed during the startup and test program and as periodic tests.

6.2.6.1 Containment Integrated Leakage Rate Test (ILRT) (Type A Test)

Following the completion of the construction, repair, inspection, and testing of welded joints, penetrations, and mechanical closures, including the satisfactory completion of the structural integrity tests as described in Section 3.8.1, a preoperational primary containment leakage rate test was performed to verify that the actual containment leak rate does not exceed the design limits. To ensure a successful ILRT, local leakage tests (Type B and C tests) were performed on penetrations and isolation valves, and repairs were made, if necessary, to ensure that leakage through the containment isolation barriers did not exceed the limits established by Technical Specifications or the Owner.

A general inspection of the accessible interior and exterior surfaces of the primary containment and components will be performed prior to any Type A test. Any structural

deteriorations requiring repair prior to performing Type A tests and the corrective actions taken will be included in the test report. Leakage rates of equipment to be Type B or C tested shall include "as-found" (pre-repair) and "as-left" (post-repair) condition, with the exception of the test performed prior to the initial Type A test.

An ILRT is performed on the entire primary containment to determine that the total leakage through all primary containment isolation barriers does not exceed the design leakage rate of 1.1 percent/day at the primary containment DBA LOCA pressure (P_a). The pertinent test data, including test pressures and acceptance criteria, are presented in Table 6.2-60.

Systems penetrating containment that may not be vented to the primary containment atmosphere during the ILRT, but whose containment isolation valves are Type C tested, are listed below.

	System	Exception Justification
1.	Reactor building closed loop cooling water (RBCLCW)	1
2.	Low-pressure core injection subsystem of residual heat removal (LPCI)	1, 2
3.	High-pressure core spray (HPCS)	1, 2
4.	Low-pressure core spray (LPCS)	1, 2
5.	Reactor core isolation cooling (RCIC)	2
6.	Feedwater (FWS)	2
7.	Standby liquid control (SLC)	2
8.	Reactor coolant recirculation pump seal injection	1
9.	Reactor water cleanup (RWCU)	1
10.	Control rod drive	1, 2
11.	Main steam	1
12.	Shutdown cooling subsystem of residual heat removal	1, 2
13.	Containment spray subsystem of residual heat removal	2
14.	Suppression pool cooling subsystem of residual heat removal	2

Exception Justification

1. Systems that are required for proper conduct of the Type A test or to maintain the plant in a safe condition during the test shall be operable in their normal mode and need not be vented.

2. Systems that are normally filled with water and operating under postaccident conditions need not be vented.

The Type A ILRT is normally performed at the end of a refueling outage. At this time, the RPV head is installed and tensioned. In order to maintain the minimum vessel flange and head flange temperature required by the Technical Specifications, the reactor vessel water level may be raised to the flange level. Thus, the main steam lines may be flooded during the time the ILRT is performed, and may not be vented to the primary containment. The CRD and hydraulic control for the reactor recirculation flow control valves will not be vented during the ILRT as justified by Notes 17 and 26 of Table 6.2-56, respectively.

During the Type A test, the RPS system will be energized (i.e., SCRAM reset). Scram discharge volume (SDV) vent and drain valves (2RDS*AOV123, 124, 130 and 132) will be Type C tested and leakages will be added to Type A results for overall containment leakage.

The preoperational (initial) Type A test was performed in accordance with 10CFR50 Appendix J, ANS-N45.4/ANSI-56.8-1981. This method employs:

4 hr (min) stabilization period

24 hr (min) ILRT test period (utilizing total time analysis of BN-TOP-1 and mass point analysis technique of ANSI-56.8)

1 to 4 hr (min) verification period

The 24-hr Type A test provided the baseline for postoperational tests.

Subsequent refueling outage Type A tests can be performed using total time analysis which provides the bases for a Type A test duration of 8 hr (min) with 20 data sets (min) (BN-TOP-1 referenced).

With the implementation of Option B, "Performance-Based Requirements of Appendix J to 10CFR50," for Type A, B and C leakage rate testing, the Type A test may be performed using a mass point analysis for a test duration of 8-24 hr with 30 data sets (min) in accordance with ANSI/ANS-56.8-1994, or a total time analysis for a test duration of 8 hr (min) with 20 data sets (min) in accordance with BN-TOP-1.

The test method utilized is the absolute method utilizing the total time and mass point analysis techniques applicable to the test duration. Values of primary containment atmosphere dry-bulb temperature, dew point temperature (vapor pressure), and pressure are used in the leakage rate calculations.

The primary containment leakage monitoring system (LMS) provides means for monitoring the primary containment pressure during ILRT. Two independent pressure-sensing lines, each equipped with a quartz digital-type absolute pressure manometer, are provided in LMS system. The quartz manometers are required for the Type A test only. To protect these devices from damage, the quartz manometers will be isolated, disconnected, and stored when not in use. A third quartz manometer is provided as a spare (Figure 6.2-73). Two independent temporary quartz digital-type absolute pressure gauges may be used in place of the instruments installed in LMS.

Eighteen temperature elements and six humidity analyzers are provided in the CMS system to monitor dry-bulb and dew point temperatures, respectively (Figure 6.2-71). Additional instrumentation may be installed as required to complete the test. Temporary temperature elements and temporary humidity analyzers may be used in place of the instruments installed in LMS.

The test procedure, test equipment and facilities, period of testing, and verification of leak test accuracy follow the recommendations of BN-TOP-1, Rev. 1, 1972, or ANSI/ANS-56.8-1994.

Acceptance criteria and test intervals Type A, B, and C tests will be in conformance with 10CFR50 Appendix J, Option B.

6.2.6.2 Containment Penetration Leakage Rate Tests (Type B Tests)

Containment penetrations whose design incorporates resilient seals, gaskets, or sealant compounds; airlock door seals, equipment and access hatches with resilient seals or gaskets; and other such penetrations received a preoperational leak test in accordance with Appendix J to 10CFR50. Periodic leak tests will be in accordance with 10CFR50 Appendix J, Option B.

The following penetrations will be tested to Type B criteria:

1. Equipment hatch (Figure 3.8-6).
2. Personnel airlock/equipment hatch (Figure 3.8-6).
3. Suppression pool access hatch (Figure 3.8-7).
4. CRD removal hatch (Figure 3.8-7).
5. Drywell head (Figure 3.8-8).

6. Electrical penetrations (Figure 3.8-9).

7. TIP 2NMT*Z31A,B,C,D & E penetration flanges (Figure 6.2-75b).

8. Midspan single O-ring flange in 2NMT*Z31B (Figure 6.2-75a).

9. Escape airlock (Figure 3.8-6).

10. Blind flanges on piping penetrations Z46C and Z74.

11. TIP N_2 purge penetration (Figure 6.2-75).

12. TIP 2NMT*Z31A,B,C,D & E penetration bellows.

The makeup pressure test will be utilized to determine primary containment penetration leak rates. In this test, leakage is measured by pressurization between the double seal and measurement of the makeup flow required to maintain the test pressure.

Penetrations that rely on welds for sealing will not be Type B tested but they will be subject to the ILRT conditions. Leakage from these penetrations will be included in the overall leakage rate measured during the ILRT. Test pressures are given in Table 6.2-60.

The primary containment airlocks shall be tested at P_a in accordance with 10CFR50 Appendix J, Option B. The testing frequency is contained in the Technical Specifications.

The acceptance criteria for the preoperational primary containment penetration leakage rate test are in compliance with the criteria given in Appendix J to 10CFR50. The periodic testing acceptance criteria are established in Technical Specifications.

6.2.6.3 Primary Containment Isolation Valve Leakage Rate Tests (Type C Tests)

Primary containment isolation valve leakage rate tests will be performed by local pressurization in accordance with the requirements of Appendix J to 10CFR50, Option B. The pressure will be applied in the same direction as it would be applied when the valve is required to perform its safety function, unless it can be determined and documented that the results from the test for a pressure applied in the opposite direction will provide equivalent or more conservative results. Table 6.2-65 lists all plant primary containment isolation valves which may be reverse tested, and provides the justification necessary to ensure that the reverse test is as conservative as testing the valve in the same direction as postaccident flow.

Containment isolation valves will be Type C tested in accordance with Appendix J, Option B. Each valve to be tested will be closed by normal operation, without any preliminary exercising or adjustment. Table 6.2-56 lists all primary containment isolation valves on pipelines penetrating the primary containment.

Primary containment isolation valves with bonnet vent pathways installed to prevent pressure locking may be tested in accordance with Appendix J by pressurizing the main line and the bonnet vent pathway simultaneously since both pathways will be open during an accident. This method will pressurize the valve between the seats as well as upstream of the disc.

It should be noted that test, vent, and drain (TVD) connections within the boundaries of containment isolation valves are not Type C tested since they are small lines, usually 3/4 in or smaller; they are normally only open for the Type C testing or ILRT; they are part of a double isolation barrier for redundancy, consisting of either the main process inboard isolation valve and a single TVD valve for those TVD connections outside the main process inboard isolation valve, or double TVD valves for those TVD connections inside the main process inboard isolation valve; and they are designed as Quality Assurance (QA) Category I, Safety Class 2. In addition, their use is controlled by administrative procedures.

The test pressures and acceptance criteria for the primary containment isolation valve leakage rate tests are given in Table 6.2-60.

6.2.6.4 Additional Requirements

The combined leakage rate for all penetrations and valves subject to Type B and C tests will be in accordance with 10CFR50 Appendix J, Option B.

6.2.6.5 Scheduling and Reporting of Periodic Tests

The periodic leakage test schedule is given in Technical Specifications.

6.2.6.6 Special Testing Requirements

The reactor building will be tested as required by Technical Specifications.

Preoperational high- and low-pressure suppression pool bypass leakage tests will be performed once prior to fuel load to determine the bypass leakage from the drywell into the suppression chamber. The high-pressure test will be initiated at the pressure differential of 25 psi +0.5, -0.0 between the drywell and suppression chamber. The low-pressure test will be initiated at 3 psid. In each case, the drywell pressure will be

monitored at regular time intervals and compared with expected pressure decay for $A/\sqrt{K}$ of 0.0054 sq ft.

The test pressures and acceptance criteria for the drywell bypass leakage tests are given in Table 6.2-61.

6.2.7 References

1. Models used in LOCTVS - A Computer Code to Determine Pressure and Temperature Response of Vapor Suppression Containments Following a Loss-of-Coolant Accident, Topical Report SWECO 8101, 1981.

2. Maximum Flow Rate of a Single Component Two-Phase Mixture, APED-4378, October 25, 1963.

3. Sharma, D. F. Technical Description Annulus Pressurization Load Adequacy Evaluation, NEDO-24548, January 1979.

4. U.S. Nuclear Regulatory Commission, NUREG/CR-2812, The Relative Importance of Temperature, pH, and Boric Acid Concentration on Rates of H_2 Production from Galvanized Steel Corrosion, January 1984.

5. BNL-NUREG-24532 (Informal Report), Hydrogen Release Rates from Corrosion of Zinc and Aluminum, May 1978.

6. NEDM-10320, The GE Pressure Suppression Pool Containment Analytical Model, March 1971.

7. NEDO-20533, The General Electric Mark III Pressure Suppression Containment System Analytical Model, June 1974.

8. NEDC-31994P, Licensing Topical Report, Power Uprate Licensing Evaluation for Nine Mile Point Nuclear Power Station Unit 2, Revision 1, May 1993.

9. NEI 94-01, Nuclear Energy Institute, Industry Guideline for Implementing Performance-Based Option of 10 CFR Part 50, Appendix J, Revision 0, July 26, 1995.