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U.S. Nuclear Regulatory Commission
Washington, DC 20555-0001

Subject: **Submittal of ABWR Licensing Topical Report (LTR) NEDO-33372, "Advanced Boiling Water Reactor (ABWR) Containment Analysis"**

Reference: Letter MFN 017-97, J. Quirk to NRC, *ABWR Design Control Document, Revision 4*, dated March 28, 1997, Docket No. 52-001

The purpose of this letter is for GE-Hitachi Nuclear Energy Americas (GEH) to request U.S. Nuclear Regulatory Commission (NRC) generic review and approval of the subject LTR in advance of any future combined operating license application (COLA) submittals. This submittal is in support of the ABWR Design Centered Working Group (DCWG) plans and is one of the ABWR-related LTRs discussed in South Texas Project 3&4 project meetings with the NRC. Appendix A to the LTR provides the justification for the changes under the ABWR design certification rule.

Enclosure 1 contains the GEH LTR that reviews the Design Change Document (DCD) description of the ABWR Containment design and analyses. Engineering analysis performed subsequent to ABWR certification identified improvements in the analysis assumptions regarding feedwater flow, decay heat and containment vent modeling that should be reflected in the certified ABWR containment analysis. The proposed changes are based on the updated ABWR containment analysis and conservative results using GE-14 fuel assumptions. This LTR describes the revised ABWR containment analysis, the results, and the proposed changes.

Enclosure 2 contains design basis information and analytical assumptions used by GEH to update the ABWR containment analysis. This information is provided per NRC request and in support of the staff's independent analysis of GEH's proposed changes to the US ABWR DCD.

The enclosures contain no information that GE considers proprietary although full copyright protection applies.

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NRO

Should you have any questions concerning this matter, please contact me at 910-602-1885 or joe.savage@ge.com.

Sincerely,

Joseph A. Savage
Project Manager, ABWR Licensing

JAS/mkg/jll

Enclosure: NEDO-33372 "Advanced Boiling Water Reactor (ABWR) Containment Analysis", September 2007 – Non-Proprietary

Containment Analysis Input Parameters for ABWR, Non-Proprietary

cc:	JA Savage	GE (Wilmington w/ enclosure)
	BE Brown	GE (Wilmington w/ enclosure)
	GB Stramback	GE (San Jose w/o enclosure)
	GF Wunder	NRC (w/ enclosure)
	ME Tonacci	NRC (w/ enclosure)
	MA McBurnett	STP (w/ enclosure)
	Lona Smith	STP (w/ enclosure)
		eDRF 0000-0064-3948

Containment Analysis Input Parameters for ABWR

Section

Containment Analysis Input Parameters for ABWR

1 - Reactor Initial & Analytical Conditions			
No.	**Parameter**	**Units**	**Value**
a	Initial Power (100% rated)	MWt	3926
b	Analysis Power for DBA	MWt	4005
c	100% Rated Core Flow	Mlbm/hr	115.1
d	Initial Dome Pressure at analysis power	psia	1055.0
e	Turbine steam flow rate at analysis power	Mlbm/hr	17.2
f	Feedwater flow rate at analysis power	Mlbm/hr	17.5
g	Vessel inlet feedwater temperature at analysis power	°F	422.4

Containment Analysis Input Parameters for ABWR

No.	Parameter	Units	Value
	2 - Decay Heat		
a	Short-term Response		
	1 Fuel/ Decay Heat Model		ANS 5 - 1971 20%/10%
b	Long-term Response		
	1 Fuel/ Decay Heat Model		Ansi/ANS-5.1-1994 + 2 sigma
	2 Type of fuel		GE14
	3 Fuel Bundle Average Enrichment	%	3.39
	4 End-of-Cycle Core Average Exposure (Select one)		
	a. By short ton	GWd/STU	27.70
	b. By metric ton	GWd/MTU	24.73
	5 Core Average Time at Power (Irradiation Time)	years	3.36
	6 Cycle Duration	months	18

Containment Analysis Input Parameters for ABWR

No.	Parameter	Units	Value
	3 - Vessel Weights and Volumes		
a	Total vessel free volume w/ attached piping	ft^3	25,083
b	Vessel liquid volume w/ attached piping		
1	Subcooled	ft^3	0
2	Saturated	ft^3	15,754
c	Steam mass in main steam line to the first MSIV		
1	One outer steam line	lbm	678
2	One inner steam line	lbm	478
d	Liquid mass in one recirculation loop	lbm	0
e	Liquid mass in connected piping to the first normally closed valve		
1	LPCI piping	lbm	0
2	RWCU piping	lbm	400
3	RCIC piping	lbm	1,628
4	LPCS/CS piping	lbm	1,454
5	HPCI/HPCS piping	lbm	321
f	Metal mass of RPV internals structure (excluding fuel and fuel assembly)	lbm	857,100
g	Metal mass of RPV, including top head but excluding vessel skirt.	lbm	1,841,000
h	Metal mass of RPV connected piping		
1	Recirculation piping for all loops	lbm	0
2	LPFL/RHR piping	lbm	15,522
3	LPCI piping	lbm	0
4	Main Steam Lines to second isolation valve	lbm	95,539
5	RWCU lines to first normally closed valve	lbm	Not Included
6	HPCI/HPCF lines to first normally closed valve	lbm	3,516
7	RCIC lines to first normally closed valve	lbm	Not Included
i	Number of fuel bundles		872
j	Mass of fuel (UO2)	lbm	396,000
k	Mass of fuel assembly	lbm	176,900

Containment Analysis Input Parameters for ABWR

4 - LOCA			
No.	**Parameter**	**Units**	**Value**
a	LOCA Break elevation (from bottom of vessel)		
1	Feedwater Line	ft	38.1000
2	Steam Line	ft	51.0000
b	Time at which MSIVs start to close for DBA-LOCA		
1	Feedwater Line Break	sec	0.5000
2	Main Steam Line Break	sec	0.5000
c	Time at which MSIVs are completely closed for DBA-LOCA		
1	Feedwater Line break	sec	3.5000
2	Main Steam Line Break	sec	5.0/3.5 (ST/LT)
d	Inside vessel height (from vessel zero to top of steam dome)	ft	69.0800
e	Long-term LOCA break area		
1	Feedwater Line Break	ft^2	0.9120
2	Main Steam Line Break	ft^2	1.0600
g	Main steam line data for short-term containment analysis vessel model		
1	Steam line pipe inside diameter	in	25.5430
2	Nozzle safe end inside diameter or vessel nozzle inside diameter whichever is smaller	in	13.9200
3	Steam Line Flow Limiter diameter	in	25.5430
4	Length of nozzle extending from vessel	in	85.7100

Containment Analysis Input Parameters for ABWR

5 - Containment Operating Conditions			
No.	**Parameter**	**Units**	**Value**
a	Atmospheric pressure	psia	14.7
b	Drywell-to-Wetwell operating pressure	psid	2
c	Drywell pressure		
1	Maximum	psig	1.3
2	Minimum	psig	0
3	Nominal	psig	0.75
4	Scram setpoint	psig	2
d	Drywell temperature		
1	Maximum	°F	135
2	Minimum	°F	50
e	Drywell relative humidity		
1	Nominal	%	20
f	Wetwell pressure		
1	Maximum	psig	1.3
2	Minimum	psig	0
3	Nominal	psig	0.75
g	Wetwell temperature		
1	Maximum	°F	95
2	Minimum	°F	50
h	Wetwell relative humidity		
1	Nominal	%	100
i	Suppression pool temperature		
1	Maximum	°F	95
2	Minimum	°F	50

Containment Analysis Input Parameters for ABWR

6 - Drywell			
No.	**Parameter**	**Units**	**Value**
a	Total drywell free volume (including vent system)	ft^3	259,600.0
b	Drywell holdup volume	ft^3	0.5
c	Drywell pool surface area (in contact with drywell airspace)	ft^2	121.7
d	Design Pressure (analytical limit)	psig	45.0
e	Design Temperature	°F	340.0
f	Pressure difference between wetwell and drywell for vacuum breakers to start to open	psid	0.1
g	Pressure difference between wetwell and drywell for vacuum breakers to be fully open	psid	0.5
h	Number of WW-DW vacuum breaker systems in use		
	1 Single Valve		7
	2 Multiple Valve		N/A
i	Total loss coefficient of each WW-DW vacuum breaker line (per system) including entrance and exit coefficients		
	1 Single Valve		3
	2 Multiple Valve		N/A
j	Total flow area of one WW-DW vacuum breaker line (per system)		
	1 Single Valve	ft^2	2.185
	2 Multiple Valve	ft^2	N/A
k	Acceptable effective DW-to-WW bypass leakage area ($A/\sqrt{K}$)	ft^2	0.1
l	Drywell inside diameter	ft	95.167
m	Trash Rack loss coefficient		0.16
o	DCV loss coefficient		0.92
p	DCV Area	ft^2	12.173

Containment Analysis Input Parameters for ABWR

7 - Vent System			
No.	**Parameter**	**Units**	**Value**
j	Number of downcomer / horiz. vents per row		10
k	Flow area of each downcomer	ft^2	12.2
l	Drywell weir annulus pool volume including horizontal vents		
1	High water level	ft^3	N/A
2	Low water level	ft^3	N/A
m	Inside weir wall diameter	ft	34.75
n	Outside weir wall diameter	ft	38
o	Weir annulus surface area	ft^2	121.73
p	Height of weir wall from suppression pool bottom	ft	38.4
q	Length of each horizontal vent	ft	2.95
r	Centerline elevation of top-row vents from pool bottom	ft	11.483
s	Centerline elevation of mid-row vents from pool bottom	ft	6.988
t	Centerline elevation of bottom-row vents from pool bottom	ft	2.493
u	Total loss coefficient for vent system		1.7
v	Horizontal vent diameter	ft	2.3

Containment Analysis Input Parameters for ABWR

No.	Parameter	Units	Value
	8 - Wetwell / Suppression Pool		
a	Total suppression pool volume		
	1 High water level	ft^3	128,000
	2 Low water level	ft^3	122,000
b	Wetwell free airspace volume		
	1 High water level	ft^3	210,000
	2 Low water level	ft^3	216,000
c	Suppression pool depth		
	1 High water level	ft	23.333
	2 Low water level	ft	22.64
d	Suppression pool surface area (in contact with wetwell airspace)		
	1 Below Tunnels	ft^2	5,460
	2 At Low Water level (6.9m)	ft^2	4,846
e	Design Pressure (analytical limit)	psig	45
f	Design Temperature		
	WW	°F	255
	SP	°F	207

Containment Leakage

cc	Appendix J maximum allowable containment leakage for NPSH	%/day	0.5
dd	Appendix J test pressure	psig	39

Containment Analysis Input Parameters for ABWR

No.	Parameter	Units	Value
	9 - SRV		
a	Minimum number of SRV openings in normal set before SRVs switch to low-low set		100000000
b	Average throat area of SRVs	ft^2	1
c	Suppression pool temperature above which vessel controlled cooldown is initiated (Tech Spec value)	°F	1000
d	Number of ADS valves		8
e	Maximum vessel controlled cooldown rate using SRVs	°F/hr	100
f	Average vertical elevation drop from SRV entrance at main steam line to SRV quenchers	ft	10000
g	SRV quenchers initial submergence at LWL	ft	15.6
h	SRV rated flow at pressure		
	1. Rated flow	lbm/hr	6.98E+06
	2. Pressure	psig	1148.56
i	Number of SRVs available for manual pressure and temperature control		4
j	Normal (relief) set mode (nominal)		
	1 a. Number of valves Group 1		2
	1 b. Opening setpoint Group 1	psig	1160
	2 a. Number of valves Group 2		4
	2 b. Opening setpoint Group 2	psig	1170
	3 a. Number of valves Group 3		4
	3 b. Opening setpoint Group 3	psig	1180
	4 a. Number of valves Group 4		4
	4 b. Opening setpoint Group 4	psig	1190
	5 a. Number of valves Group 5		4
	5 b. Opening setpoint Group 5	psig	1200
	6 Difference between opening setpoint and reset pressure	psi	80

Containment Analysis Input Parameters for ABWR

10 - High Pressure ECCS			
No.	**Parameter**	**Units**	**Value**
a	Vessel water level (above vessel zero, AVZ) below which HPCF is automatically actuated	in	395
b	Vessel water level (AVZ) above which HPCF is automatically shut off	in	561
c	Maximum suppression pool liquid volume above which suppression pool replaces CST (if available) as water source for HPCF	ft^3	10
d	Drywell pressure for actuation of HPCF (LOCA signal)	psig	2
e	Maximum HPCF delay time	sec	47
f	Elevation of top of suction strainer from S/P bottom	ft	5
g	If high drywell pressure and high vessel water level coexist, HPCF will cycle between high and low levels	(Yes or No)	YES
m	HPCF pump heat	hp	2000
n	Runout flow	gpm	3200
o	Shut-off head	psid	1195.13
RCIC parameters			
p	Vessel water level (AVZ) below which RCIC is automatically actuated	in	452.1
q	Vessel water level (AVZ) above which RCIC is automatically shut off	in	561.06
r	Rated flow	gpm	800
s	Maximum time delay	sec	30
t	Maximum vessel pressure for RCIC operation	psig	1220
u	Minimum vessel pressure for RCIC operation	psig	150

Containment Analysis Input Parameters for ABWR

10 - High Pressure ECCS			
No.	**Parameter**	**Units**	**Value**
v	RCIC turbine steam flow rates		
	1 a. RPV Pressure	psig	150
	1 b. RCIC steam flow rate	lbm/hr	16,200
	2 a. RPV Pressure	psig	1220
	2 b. RCIC steam flow rate	lbm/hr	37,800
w	Maximum S/P temperature for RCIC operation	°F	170
CST parameters			
x	Available CST volume for vessel makeup	gal	155333
y	CST water temperature	°F	100.2

11 - Low Pressure ECCS			
No.	Parameter	Units	Value
LPFL parameters			
i	LPFL pump heat (per pump)	hp	800
j	LPFL shutoff head	psid	225
k	Total LPFL time delay	sec	37
l	Drywell pressure above which LPFL will automatically be actuated (LOCA signal)	psig	2
m	Vessel water level (AVZ) below which LPFL will automatically be actuated	in	362.3
n	Vessel water level (AVZ) above which LPFL will be shut off	in	561.06
o	LPFL performance		
1	Number of pumps per LPFL loop		1
2	LPFL runout flow for one pump in one loop.	gpm	4200
3	LPFL runout flow for two pumps in one loop.	gpm	N/A
p	Elevation of top of suction strainer from S/P bottom	ft	6.054

Containment Analysis Input Parameters for ABWR

No.	Parameter	Units	Value
	12 - RHR		
a	Pool cooling mode		
	1 RHR flowrate per loop	gpm	4,200
	2 Service water flowrate per loop	gpm	5,615
	3 RHR heat exchanger K-value per loop	Btu/sec-°F	225
	4 Number of RHR pumps per loop		1
b	Containment spray mode		
	1 RHR flowrate per loop		
	a. Drywell	gpm	3,698
	b. Wetwell	gpm	502
	c. Total	gpm	4,200
	2 Service water flowrate per loop	gpm	5,300
	3 RHR heat exchanger K-value per loop	Btu/sec-°F	225
	4 Number of RHR pump(s) per loop		1
c	LPCI cooling mode		
	1 RHR flowrate per loop	gpm	4,200
	2 Service water flowrate per loop	gpm	5,300
	3 RHR heat exchanger K-value per loop	Btu/sec-°F	225
	4 Number of RHR pump(s) per loop		1
	5 Pressure permissive for cooling	psig	225
d	Normal shutdown cooling mode		
	1 RHR flowrate per loop	gpm	4,200
	2 Service water flowrate per loop	gpm	5,300
	3 RHR heat exchanger K-value per loop	Btu/sec-°F	225
	4 Number of RHR pump(s) per loop		1
	5 Shutdown pressure permissive	psig	225
e	Service water temperature	°F	95
f	Average vertical distance between drywell spray nozzles and bottom of drywell	ft	81.9
g	Average drywell spray droplet diameter	ft	0.0052
h	Average vertical distance between wetwell spray nozzles and suppression pool surface at LWL	ft	39
i	Average wetwell spray droplet diameter	ft	0.0052
j	RHR pump heat (per pump)	hp	800
k	Maximum vessel cooldown rate with RHR	°F/hr	100
l	Drywell pressure above which drywell sprays can operate	psig	0
m	Wetwell pressure above which automatic wetwell spray can operate	psig	N/A

Containment Analysis Input Parameters for ABWR

12 - RHR			
No.	**Parameter**	**Units**	**Value**
n	Wetwell pressure below which wetwell spray will be turned off	psig	0

Containment Analysis Input Parameters for ABWR

No.	Parameter	Units	Value
	13 - Feedwater System Mass and Energy		
a	Reference power (at 102% power)	MWt	4,005
b	Node 1 - Vessel to closest FW heater		
	1 Fluid temperature	°F	421.3
	2 Fluid mass	lbm	87,828.0
	3 Metal mass	lbm	423,853.0
c	Node 2		
	1 Fluid temperature	°F	395.8
	2 Fluid mass	lbm	43,989.0
	3 Metal mass	lbm	169,093.0
d	Node 3		
	1 Fluid temperature	°F	370.3
	2 Fluid mass	lbm	13,106.0
	3 Metal mass	lbm	64,356.0
e	Node 4		
	1 Fluid temperature	°F	338.1
	2 Fluid mass	lbm	51,072.0
	3 Metal mass	lbm	192,797.0
f	Node 5		
	1 Fluid temperature	°F	305.9
	2 Fluid mass	lbm	116,055.0
	3 Metal mass	lbm	525,124.0
g	Node 6		
	1 Fluid temperature	°F	304.2
	2 Fluid mass	lbm	734.0
	3 Metal mass	lbm	24,156.0
h	Node 7		
	1 Fluid temperature	°F	304.1
	2 Fluid mass	lbm	292,484.0
	3 Metal mass	lbm	387,773.0
i	Node 8		
	1 Fluid temperature	°F	288.1
	2 Fluid mass	lbm	59,900.0
	3 Metal mass	lbm	83,568.0
j	Node 9		
	1 Fluid temperature	°F	272.0
	2 Fluid mass	lbm	25,703.0
	3 Metal mass	lbm	40,750.0

13 - Feedwater System Mass and Energy			
No.	**Parameter**	**Units**	**Value**
k	Node 10		
1	Fluid temperature	°F	244.2
2	Fluid mass	lbm	73,750.0
3	Metal mass	lbm	117,271.0
l	Node 11		
1	Fluid temperature	°F	216.4
2	Fluid mass	lbm	26,396.0
3	Metal mass	lbm	40,758.0
m	Node 12		
1	Fluid temperature	°F	194.9
2	Fluid mass	lbm	77,555.0
3	Metal mass	lbm	116,927.0

Containment Analysis Input Parameters for ABWR

14 - Others			
No.	**Parameter**	**Units**	**Value**
a	Control Rod Drive		
1	CRD Flow Rate	lbm/sec	9.79
2	CRD Flow Enthapy	Btu/lbm	77.02
b	Elevation from S/P bottom to top of RCIC suction strainer	ft	4.773

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LICENSING TOPICAL REPORT

Advanced Boiling Water Reactor (ABWR) Containment Analysis

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Appendices

1.0 Introduction

This Licensing Topical Report (LTR) requests US Nuclear Regulatory Commission (NRC) approval of a generic change to the design certification for the US Advanced Boiling Water Reactor (ABWR) design. The proposed changes are based on a completely revised ABWR containment analysis. An exemption to the generic Technical Specifications (TS) is necessary to implement the proposed changes, and associated Tier 2 changes are also proposed. This LTR describes the revised ABWR containment analysis and identifies the proposed changes.

This containment analysis is performed using GE 14 fuel, which provides conservative results when compared to the DCD (Design Control Document) referenced GE 7 fuel bundles (P8x8R) as shown in LTR "*GE14 Compliance with Amendment 22 of NEDE-24011-P-A (GESTAR II)*" (Reference 10.1). GE 7 fuel was used for the ABWR certification.

The nuclear safety evaluation in this LTR identifies the required TS and associated Tier 2 changes and explains the bases for them. There are no Tier 1 changes associated with this analysis.

The ABWR containment analysis described in this LTR is based on information subsequently discovered after certification. The analysis performed for other ABWRs identified improvements in the analysis assumptions regarding feedwater flow, decay heat and containment vent modeling that needed to be reflected in the certified ABWR containment analysis. In addition, a design change is proposed to the ABWR protection system to accommodate the feedwater flow modeling changes. The amendment to the ABWR DCD is required to assure compliance with NRC regulations that were applicable at the time the certification was issued and promotes standardization of ABWR certified design material.

As the regulatory processes for generic amendment of approved and certified reactor designs such as the ABWR (10CFR52 Appendix A) are currently in the state of flux, GE understands that a generic change may not be feasible for the NRC to grant until the planned revision to 10CFR52.63 becomes effective. If the NRC does not make the planned revisions to 10CFR52.63, future Combined Operating License Applications (COLA) applicant(s) would then intend to seek specific departures from the DCD based on the content of this LTR. NRC review of the technical content of this LTR is requested with the understanding that this LTR and subsequent discussions between GE and NRC staff may form the basis for site-specific departures requested in one or more future COLAs.

1.1 Acronyms

ACS – Atmospheric Control System
ANSI/ANS – American National Standards Institute /American Nuclear Society
ABWR – Advanced Boiling Water Reactor
BWR – Boiling Water Reactor
COLA – Combined Operating License Application
DBA – Design Basis Accident
DCD – Design Control Document
DCV – Drywell Connecting Vent

DW-WW – Drywell-Wetwell
EOC – End of Cycle
EOC-RPT – End of Cycle Recirculation Pump Trip
FWLB – Feedwater Line Break
LOCA – Loss of Coolant Accident
LTR – Licensing Topical Report
LWR – Light Water Reactor
MSLB – Main Steam Line Break
NBR – Nuclear Boiler Rated
NRC – Nuclear Regulatory Commission
R/B – Reactor Building
RPV – Reactor Pressure Vessel
SSE – Safe Shutdown Earthquake
TS – Technical Specifications

2.0 Description of the Design

This chapter describes the ABWR containment functional requirements and lists the major changes in the analysis (Section 2.1). In Section 2.2, the certified ABWR design is described for the three areas of interest. The required changes for the revised ABWR containment analysis are described in Section 2.3. There were additional changes included in the revised ABWR analysis, mainly based on BWR operational experience, that are described in Section 2.4.

2.1 Containment Analysis Background

The ABWR primary and secondary containments are shown in Figure 1. The ABWR primary containment is the main focus of this LTR and its functional capabilities are listed below. The analysis summarized in this LTR is associated with the first of the nine items. The functional capability for the other eight items are described in the certified DCD. These items are not included in this LTR because they are not impacted by the revised containment analysis.

The ABWR pressure suppression primary containment system, which is comprised of the drywell and wetwell and supporting systems, is designed to have the following functional capabilities:

1. The containment structure has the capability to maintain its functional integrity during and following the peak transient pressures and temperatures, which would occur following any postulated loss-of-coolant accident (LOCA), including the worst LOCA pipe break (which leads to maximum containment and drywell pressure and/or temperature) simultaneously with loss of offsite power and a safe shutdown earthquake (SSE).

   The containment structure is designed for the full range of loading conditions consistent with normal plant operation and accident conditions, including the LOCA-related design loads in and above the suppression pool.

The containment structure is designed to accommodate the negative pressure difference between the drywell and wetwell and relative to the Reactor Building (R/B) surrounding.

2. The containment structure and isolation system, with concurrent operation of other accident mitigation systems, is designed to limit fission product leakage, during and following the postulated DBA, to values less than leakage rates which would result in offsite doses greater than those set forth in 10 CFR 100.

3. Capability for rapid closure or isolation of all pipes or ducts, which penetrate the containment boundary, is provided to maintain leakage within acceptable limits.

4. The containment structure can withstand coincident fluid jet forces associated with the flow from the postulated rupture of any pipe within the containment.

5. The containment structure is designed to accommodate flooding to a sufficient depth above the active fuel to permit safe removal of the fuel assemblies from the reactor core after the postulated Design Basis Accidents (DBA).

6. The containment structure is protected from or designed to withstand hypothetical missiles from internal sources and uncontrolled motion of broken pipes, which could endanger the integrity of the containment.

7. The containment structure provides a means to channel the flow from postulated pipe ruptures in the drywell to the pressure suppression pool.

8. The containment system is designed to allow for periodic tests at the calculated peak or reduced test pressure to measure the leakage from individual penetrations, isolation valves and the integrated leakage rate from the structure to confirm the leaktight integrity of the containment.

9. The Atmospheric Control System (ACS) establishes and maintains the containment atmosphere to less than 3.5% by volume oxygen during normal operating conditions to maintain an inert atmosphere.

There was containment analysis performed subsequent to the ABWR certification, which identified necessary modeling changes for the certified ABWR containment to ensure the analysis is bounding. There are three major types of modeling changes that need to be included in the revised ABWR containment analysis:

- Feedwater line break (FWLB) flow changes
- Decay heat using 2 sigma uncertainty
- Containment vent model

An evaluation of these items was performed under the GE correction action program to determine the cause, extent of condition, and corrective actions to prevent reoccurrence.

2.2 Containment Analysis in the Certified Design

The following subsections describe the certified ABWR design for the three changes that are the subject to the LTR.

2.2.1 Feedwater Line Break

BWR plants prior to the ABWR did not evaluate for containment the FWLB shown in Figure 2 as part of the analysis of DBAs. This is because, for those plants it was determined, the recirculation piping double guillotine break was the most limiting line break (Loss of Coolant Accident – DBA). For the ABWR, there is no recirculation piping and therefore, the FWLB must be evaluated for the containment analysis.

The feedwater system side of the FWLB was modeled (Figure 3) by use of a time variant feedwater mass flow rate and enthalpy directly to the drywell airspace. The time histories of the mass flow and enthalpy were determined from the operating characteristics of anticipated feedwater system performance.

The maximum possible feedwater flow rate was calculated to be 164% of nuclear boiler rated (NBR) flow, based on the response of the feedwater pumps to an instantaneous loss of discharge pressure. Since the Feedwater Control System would respond to the decreasing reactor pressure vessel (RPV) water level by demanding increased feedwater flow, and there was no FWLB logic/mitigation in the certified ABWR design, this maximum feedwater flow was assumed to continue for 120 seconds. This was based on the following assumptions:

1. All feedwater system flow is assumed to go directly to the drywell.
2. Flashing in the broken feedwater line was ignored.
3. Initial feedwater flow was assumed to be 105% NBR.
4. The feedwater pump discharge flow will coast down as the feedwater system pumps trip due to low suction pressure. During the inventory depletion period, the flow rate is less than 164% because of the highly subcooled blowdown. A feedwater line length of 100 meters was assumed on the feedwater system side.

Further analysis for other ABWRs after certification revealed that the sequence of events, operation of systems and maximum flow time interval assumed in the certified ABWR DCD containment analysis were non-conservative.

2.2.2 Decay Heat

The certified ABWR DCD long-term containment analysis was performed with decay heat curves based on GE 7 fuel and ANSI/ANS–5.1 (1979) (Reference 10.2). Additional uncertainty was not applied to the decay heat curves used in the containment analysis based on NRC approval of this methodology (Reference 10.3). For certain safety analyses, (e.g. Emergency Core Cooling Systems LOCA), there was a 2 sigma uncertainty adder that was

applied to the decay heat curves to ensure conservatism, but this was not done or required for the long-term containment analysis.

Background and Description of ANS 5.1 Decay Heat Model

The ANSI/ANS-5.1 (1979) standard at the time of the containment analysis for the certified ABWR DCD had significant technical advantages over previous standards in that it dealt in great detail with the physics involved and was based on a significant body of empirical data. Based on this standard, GE developed generic decay heat curves to provide a more accurate assessment of decay heat during DBA. This methodology was described in Reference 10.4, which was reviewed and approved by the NRC. This decay power curve not only includes fission product decay heat but also includes other major contributors to post-LOCA heat generation. The other contributors include decay of actinides and fission heat due to delayed neutrons.

Decay Heat from Fission Products

The fission product values provided in ANSI/ANS-5.1 are based on the fissioning of the major fissionable nuclides present in Light Water Reactors (LWRs), i.e., U^{235}, Pu^{239} thermal and U^{238} fast. A method is also prescribed for evaluating the total fission product decay heat power from the data given for these specific nuclides. There are fissions produced from other nuclides, however, ANSI/ANS-5.1 (1979) assumed that the nuclides other than Pu^{239} and U^{238} have the same characteristics as U^{235}.

Decay of Actinides

Actinides are the heavy elements produced from neutron capture by uranium and plutonium isotopes. The actinide concentration following shutdown is calculated assuming that at shutdown the concentration of each actinide is at its equilibrium value. This equilibrium value is determined assuming no neutron captures by the radioactive nuclide. These assumptions conservatively result in a higher actinide concentration.

Decay of Activated Structures

The principal structural material in the reactor core is zirconium and it is therefore the principal source of decay heat from activated structures. Other materials in and around the reactor core are the steel in the control blades, shroud, and bottom support plate. However, the contribution to the total decay heat from activation of materials outside the fuel is negligible and is therefore not included.

Fission Heat Induced by Delayed Neutrons

When a reactor shuts down, the power does not drop to zero immediately. Instead it decays away with time due to fissions caused by delayed neutrons. The contribution from delayed neutrons is conservatively determined by assuming a slow blowdown rate, which results in a smaller void negative reactivity feedback, and hence a slower decrease in the neutron flux following a LOCA.

Summary

The ANSI/ANS-5.1 (1979) model of decay heat was believed to provide a more accurate representation than previous models. In developing the decay heat power curve for GE 7 fuel the ANSI/ANS-5.1 (1979) standard was conservatively applied. Therefore, at the time of the certification of the ABWR it was believed that the GE 7 fuel based decay heat curve used in the

long-term containment analysis was conservative, so the uncertainty (2-sigma adder) was not included. This is commonly referred to as the best estimate decay heat curves.

Further analysis done based on ANSI/ANS–5.1 (1994), including the uncertainty, has determined that the decay heat curves using best estimate ANSI/ANS–5.1 (1979) were non-conservative for long-term analysis due to additional actinides and activation products that were not included.

2.2.3 Containment Vents

The ABWR drywell is divided into upper and lower drywells, connected by rectangular drywell connecting vents. Large break LOCAs [e.g., Main Steam Line Break (MSLB), FWLB] occur in the upper drywell. In order to model the drywell as a single volume, it is necessary to include the drywell connecting vent (DCV) loss coefficients to determine the total vent flow loss coefficients. See NUREG-1503, Section 6.2.1.2, for additional details. In the containment analysis for the certified ABWR DCD, the main vent system model did not capture some of the key features that impact the short-term containment response and thus the pool swell loads.

The ABWR vent system is similar to the Mark III design. The Mark III has a weir annulus in the drywell and 3 rows of horizontal vents to connect the drywell to the suppression pool. Instead of a weir annulus, ABWR has 10 vertical vents and 30 (3 each) horizontal vents. The certified DCD model did not properly simulate the horizontal vent portion of the vent system and incorrectly modeled the vent clearing time. These deficiencies are the major contributor to the difference between the certified ABWR and the ABWR revised containment analysis results.

2.3 Revised Containment Analysis Assumptions

2.3.1 FWLB

The feedwater system side of the FWLB was again modeled (Figure 4) by using a revised time variant feedwater mass flow rate and enthalpy directly to the drywell airspace. As shown in this figure, however, flow is assumed to continue until the condensate discharge and feedwater portions of the systems are drained. This eliminates the certified ABWR DCD baseline assumption of feedwater flow step changing to zero at 120 seconds. The time histories of the mass flow and enthalpy (Figure 5) were determined from the predicted characteristics of a typical feedwater system performance. The conservatism of the assumed mass flow and enthalpies will be confirmed after the detailed condensate and feedwater system designs and procurement of major equipment are completed (e.g., piping lengths and pump characteristics).

In addition, to provide added assurance of acceptable results, FWLB mitigation is added to the ABWR design. A break of the feedwater line is detected by instrumentation that measures the differential pressure between the two-feedwater lines and then confirmation of high drywell pressure will enable the logic to trip the condensate pumps (Figure 6). The logic and breakers will be safety-related to add assurance that only safety-related equipment is credited in the analysis.

With this logic, the trip of the condensate pumps will prevent the addition of the condenser water volume to the drywell. The mass that is assumed to flow from the FWLB will be limited to the volume in the condensate discharge and feedwater system piping.

In terms of plant safety the consequences of spurious operation of this additional logic is bounded by the existing DCD analysis. A spurious trip that initiates a trip of all condensate pumps would be equivalent to the DCD analysis in subsection 15.2.7 "Loss of Feedwater Flow". A failure of a single breaker would result in a trip of a single condensate pump, which is a normal operational transient and results in the control systems initiating corrective actions. During a FWLB a failure of a breaker to trip (i.e., the single failure) can be mitigated by operator actions at 30 minutes without exceeding the containment design values.

2.3.2 Decay Heat

Subsequent analysis performed by GE after certification determined that additional actinides and activation products that were not included in the ANSI/ANS-5.1 (1979) standard affect the decay heat curves. The products, while individually negligible, when summed together are non-negligible. These summation calculations determined that the inclusion of the actinides other than U^{239} and Pu^{239} and activation products does not significantly affect short-term decay heat calculations. However, for time after shutdown greater than 10^4 seconds (~ 3 hours), the effect on total decay heat can be significant. It was determined for best estimate analyses (e.g. ABWR DCD containment analysis) that do not include the 2-sigma uncertainty adder; the following guidance should be applied:

$T < 10^3$ seconds - no evaluation is required

$10^3 < T < 10^6$ seconds – evaluation recommended – decay heat can increase by 3% to 6%

$T > 10^6$ seconds – evaluation required

Table 1 shows the long-term decay heat inputs for both the certified ABWR DCD and revised ABWR containment analysis. The certified DCD decay heat was based on the GE 7 fuel best estimate analyses. In the revised analysis the decay heat is based on a GE 14 core using the ANSI/ANS-5.1 (1994), which includes contributions from additional actinides and activation products, with a 2-sigma uncertainty applied.

As shown in Table 1, the revised ABWR decay heat values are more conservative than the certified ABWR DCD values at all time intervals. A summary of the GE 7 and GE 14 fuel parameters used for determining the decay heat tables is provided in Table 2.

The decay heat inputs for the short-term analysis are based on ANSI/ANS 5 (1971) with 20%/10% margin. This input is the same for both the certified and revised ABWR analysis and is not impacted by the required changes for the long-term decay heat inputs.

2.3.3 Containment Vents

The revised ABWR containment analysis was performed using the DCV loss coefficients and considering the horizontal vents. The total DCV loss coefficient is based on a summation of losses. The entrance loss coefficient takes into account for ABWR the biological shield wall is next to the entrance. The flow loss coefficient accounts for trash racks at the entrance to the vents to block insulation from entering the vents and flowing into the suppression pool. The friction loss through the DCV is the maximum pressure loss coefficient due to piping, cabling and supports routed in the DCV. The exit loss coefficient can be neglected since each DCV is directly above a Drywell-Wetwell (DW-WW) vertical vent. These flow losses are then summed and included in the containment analysis model for the DCV.

The dimensions of the horizontal vents were included in the revised analysis and confirmation of the vent clearing was performed to ensure the revised model was correct.

These modifications were the major contributors to the revised analysis results for the wetwell pressure and drywell-to-wetwell differential pressures, as shown in Table 3.

2.3.4 Additional Changes

The following additional changes are included in the revised ABWR containment analysis. These changes were made to incorporate additional lessons learned from operating plant experience.

Suppression Pool Volume

The water volume in the suppression pool including the vents is required to be equal to or greater than 3,580 cubic meters, as stated in the Tier 1 Section 2.14.1. The ABWR revised containment analyses of scenarios with low initial suppression pool water level were performed with a smaller water volume (3,455 cubic meters) to ensure analysis/operational margin. This smaller volume is based on a suppression pool water level of 6.9 meters. The 7-meter water level is equivalent to the volume of 3,580 cubic meters. The technical specification for suppression pool water level (LCO 3.6.2.2) is greater than or equal to 7 meters and less than or equal to 7.1 meters. This is a very tight band to control the suppression pool water level; so additional margin (0.1 meters) has been built-in to the safety analysis. It is conservative to base the safety analysis for scenarios with low initial suppression pool water level on a smaller water volume since this results in higher pool temperatures.

RHR Heat Exchanger

The RHR heat exchanger heat transfer coefficient was increased from 3.7×10^5 W/°C to 4.27×10^5 W/°C (approximately 15% increase). The increase was to accommodate the RHR shutdown cooling requirements needed to support the shorter refueling outages that operating plants are achieving. The containment analysis used the larger heat transfer coefficient since this size will be standard for ABWR.

Wetwell Design Temperature

The certified ABWR wetwell gas space design temperature was 104 °C. The containment structural analysis design value is 124 °C; therefore the Tier 2 DCD is proposed to be revised to

reflect the higher value. As shown in Table 3, the analysis results are still below the 104°C and designing the wetwell to a higher temperature is conservative.

3.0 Justification for Changes

The proposed changes to the containment analysis correct some of the assumptions and ensure that the calculated pressures and temperatures for LOCA will be conservative. This containment analysis has confirmed that the containment functional capability/integrity, as described in Section 2.1 item 1, will be maintained.

As shown in Table 3 the results for the Lungmen FSAR, that also implemented these changes, are very similar to the revised ABWR results. A contributor to the differences between the Lungmen and revised ABWR results is the higher initial reactor dome pressure for Lungmen.

The addition of the FWLB mitigation to the ABWR design will provide added assurance that the revised containment analysis results will remain conservative when detailed feedwater and condensate system design and procurement work is completed.

4.0 Equipment Qualification

There is no impact on environmental qualification of equipment due to the higher predicted drywell temperatures and pressures. The qualification of equipment is based on the containment design pressures and temperatures. The predicted airspace temperature, based on bounding MSLB flows, exceeds the design temperature for 1.2 seconds and then quickly decreases to below the design temperature. This is shown in Figure 7. Since it would take much longer than 1.2 seconds for the temperature of the structure and equipment in the drywell (e.g. valves) to increase significantly, the drywell equipment and structure remain below the design temperature. Consequently, qualification for equipment in the drywell to containment design temperatures and pressures is bounding.

The equipment that is being added for the FWLB mitigation (instruments and circuit breakers) will be environmentally qualified for the predicted environment associated with the FWLB.

5.0 Operating Experience

The addition of the FWLB mitigation logic is the only plant system hardware change resulting from the revised ABWR containment analysis that could potentially impact plant operations. The other changes are associated with the analytical model and will not impact plant operations. To minimize the operational impact of the FWLB mitigation logic, confirmatory high drywell pressure signal is required to prevent spurious trips of the condensate pumps. In the unlikely event of a false trip, the DCD transient analysis of a loss of the feedwater flow (Tier 2 DCD subsection 15.2.7) reflects the ABWR response to the trip. The circuit breakers that will be used will be similar to the End-of-Cycle Recirculation Pump Trip (EOC RPT) breakers that have been used on BWRs for more than 30 years.

6.0 Nuclear Safety Review

As discussed in Section 2.3, the containment analysis in the DCD has three major areas that must be revised:

FWLB flow changes,

Decay heat using 2 sigma uncertainty, and

Containment vents model.

The revised ABWR containment analysis described in this LTR addresses each of these concerns.

The proposed changes are consistent with GDC requirements and recommendations based on operating experience. The proposed changes are based on more detailed analyses and the current licensing methodology for decay heat. The containment analysis decay heat methodology changes have been reviewed and approved by the NRC for operating plants. As described above, the analyses that justified adoption of these changes for operating plants are also applicable to the ABWR. The FWLB mitigation (safety related logic and breakers) is very similar to the EOC-RPT that has been approved by the NRC for all BWR 4/5 & 6 plants.

The proposed changes do not involve a departure from Tier 1 or Tier 2* information. Generic changes to the ABWR Generic TS are necessary to implement the revised containment analysis. NRC approval is required to implement the TS changes. The containment analysis conforms to all SRP requirements (March 2007 Sections 6.2.1, 6.2.1.1.C and 6.2.1.3) except for the deviations described in DCD section 6.2.1.1.5.6.

Appendix A provides the justification for changes to the DCD.

7.0 Consistency with ABWR Design Control Document (DCD)

There is no design departure from the Tier 1 DCD. The design changes described in this LTR are generic changes to the ABWR Generic TS and Tier 2 certified design information. The proposed changes incorporate the revised containment analysis into the ABWR design certification. The DCD Tier 2 and the Generic Technical Specifications markups are shown in Appendix B.

8.0 Descriptions of DCD Markups

The DCD markups provided in Appendix B identify the specific changes proposed by this LTR. The proposed changes are at the same level of detail as the original DCD.

The bracketed information [ ] in the Technical Specifications is preliminary, pending design detailing. These brackets are similar to the brackets that are contained in the generic Technical Specifications in the current DCD.

8.1 Tier 2 DCD Markups

Tier 2 Appendix 3B and associated table and figures

Tier 2 Subsection 6.2.1 and associated tables and figures

8.2 Generic TS DCD Markups

TS 16.3.6

TS Bases 16.3.6

9.0 Conclusions

The proposed design changes are consistent with the recommendations for containment analysis performed on operating plants (i.e., 2 sigma adder) and confirm the acceptability of the containment design, and system performance requirements, and improve nuclear safety.

As discussed more fully in Appendix A to this LTR, the proposed changes will promote increased standardization. Therefore, GE requests that the NRC amend the ABWR DCD to incorporate the changes. NRC approval is required for these generic changes to the Generic Technical Specifications.

10.0 References

10.1 GE14 Compliance With Amendment 22 of NEDE-24011-P-A (GESTAR II), NEDC-32868P, September 2000.

10.2 "American National Standard for Decay Heat Power in Light Water Reactors", ANSI/ANS-5.1-1979.

10.3 Ashok Thadani (USNRC) to Gary Sozzi (GE), "Use of SHEX Program and ANSI/ANS 5.1-1979 Decay Heat Source for Containment Long-Term Pressure and Temperature Analysis", July 13, 1993.

10.4 NEDO-23785-1-A Volume III, "The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident," October 1984.

Table 1 Long-Term Decay Heat

Shutdown Time (secs)	Certified ABWR Shutdown Power (fraction of rated power)	Revised ABWR Shutdown Power (fraction of rated power)
0	1	1
0.1	0.983	0.9893
0.2	0.924	0.9598
0.4	0.739	0.9304
0.6	0.583	0.7453
0.8	0.487	0.4928
1	0.332	0.3382
2	0.149	0.1546
4	0.0685	0.07344
6	0.0561	0.06049
8	0.0522	0.0562
10	0.0483	0.052
20	0.0422	0.04524
40	0.0371	0.03973
60	0.0345	0.0368
80	0.0324	0.03462
100	0.0311	0.03318
150	0.0289	0.0371
200	0.0274	0.02909
400	0.0241	0.0255
600	0.0221	0.02347
800	0.0207	0.02197
1000	0.0196	0.02078
2000	0.016	0.01706
4000	0.0127	0.01369
6000	0.0112	0.01208
8000	0.0103	0.01114
10000	0.00972	0.01048

Table 2 GE 7 and GE 14 Fuel

Parameter	Certified ABWR DCD	Revised ABWR DCD
Fuel	GE 7	GE 14
Number of fuel assemblies	872	872
Fuel rod array	8 x 8	10 x 10
Overall length (inches)	176	176
EOC Core Average Exposure	27.4 GWd/MT	27.4 GWd/MT
Core Average Time at Power	3.36 years	3.5 years

Table 3 Analysis Results

Design Parameter	Design Value	Certified ABWR DCD Calculated Value	Lungmen FSAR Calculated Value	Revised ABWR Calculated Value
1. Drywell pressure	309.9 kPaG	268.7 kPaG	278.5 kPaG	279.6 kPaG
2. Drywell temperature	171.1°C	170°C	176.3°C	177.3°C *
3. Wetwell pressure	309.9 kPaG	179.5 kPaG	210.4 kPaG	205.6 kPaG
4. Wetwell temperature				
Gas Space	124°C †	98.9 °C	101.7 °C	94.5°C
Suppression pool	97.2°C	96.9 °C	92.8 °C	97.1° C
5. Drywell-to-wetwell differential pressure	+172.6 kPaD – 13.7 kPaD	+ 109.8 kPaD -10.7 kPaD	+ 170.3 kPaD -3.72 kPaD	+ 172.4 kPaD – 3.86 kPaD

* Design value is exceeded at 1.2 seconds into the event and then temperature decreases as shown in Figure 7.

† The design value of 124°C is used for the containment structural analysis design basis. The revised analysis results are still below the certified ABWR DCD value of 104°C.

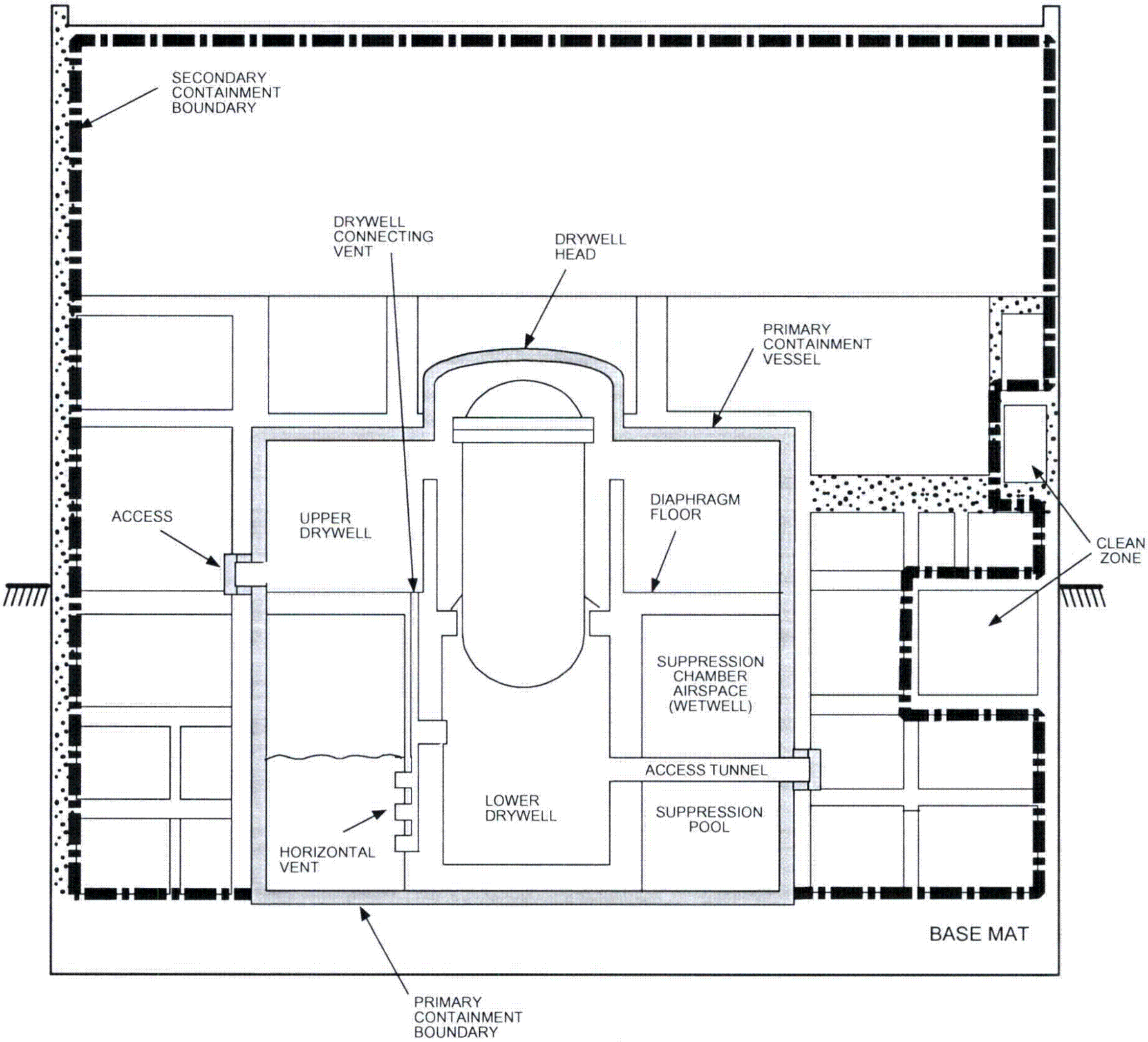
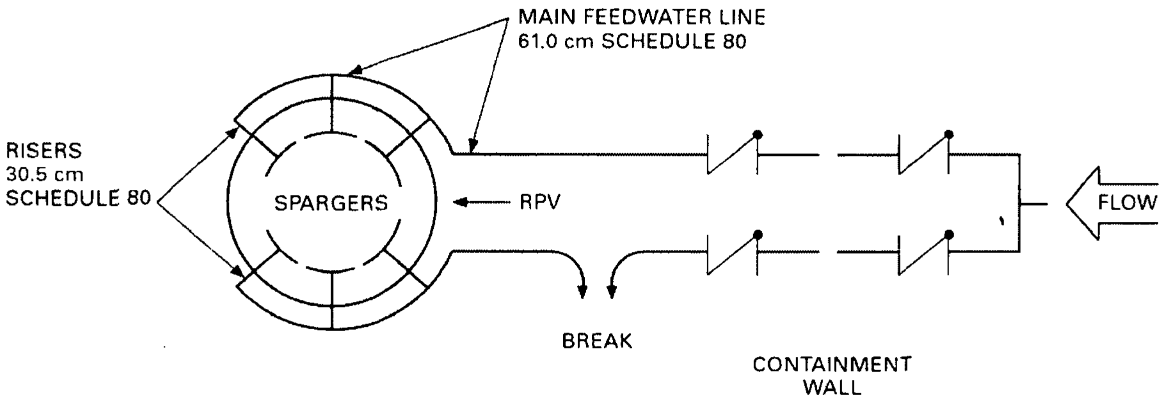


Figure 1 ABWR Containment



FLOW AREAS: MAIN FEEDWATER LINE = 0.24 m^2
RPV NOZZLE = 0.12 m^2
RISERS = 0.07 m^2
SPARGER OPEN AREA = 0.03 m^2

Figure 2
A Break in the Feedwater Line

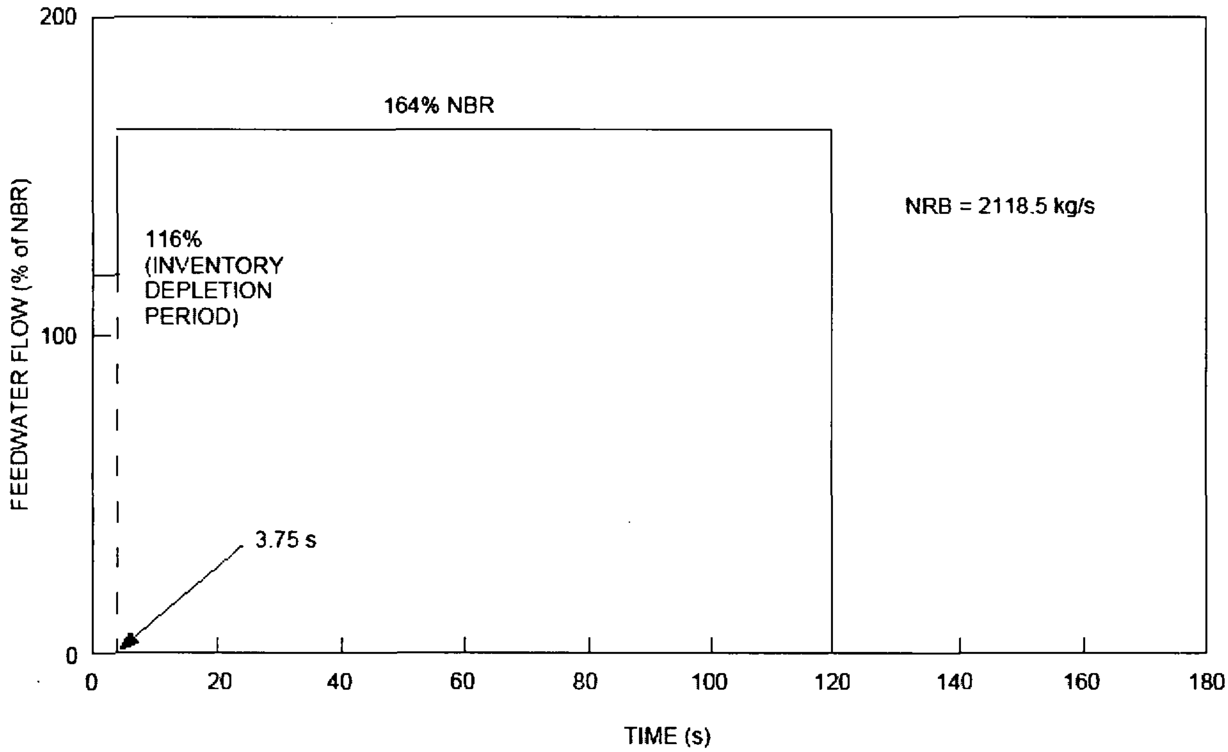


Figure 3 Certified ABWR Feedwater Line Break Flow – Feedwater System Side of Break

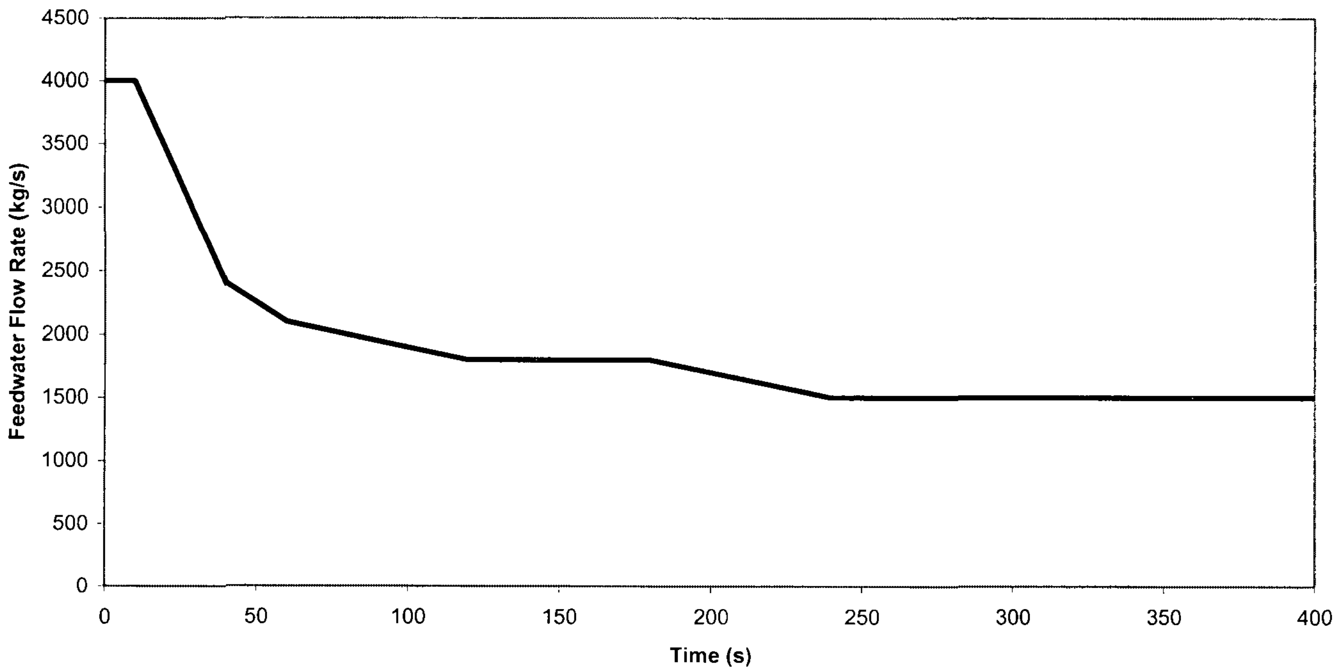


Figure 4 Revised Feedwater Line Break Flow – Feedwater System Side of Break

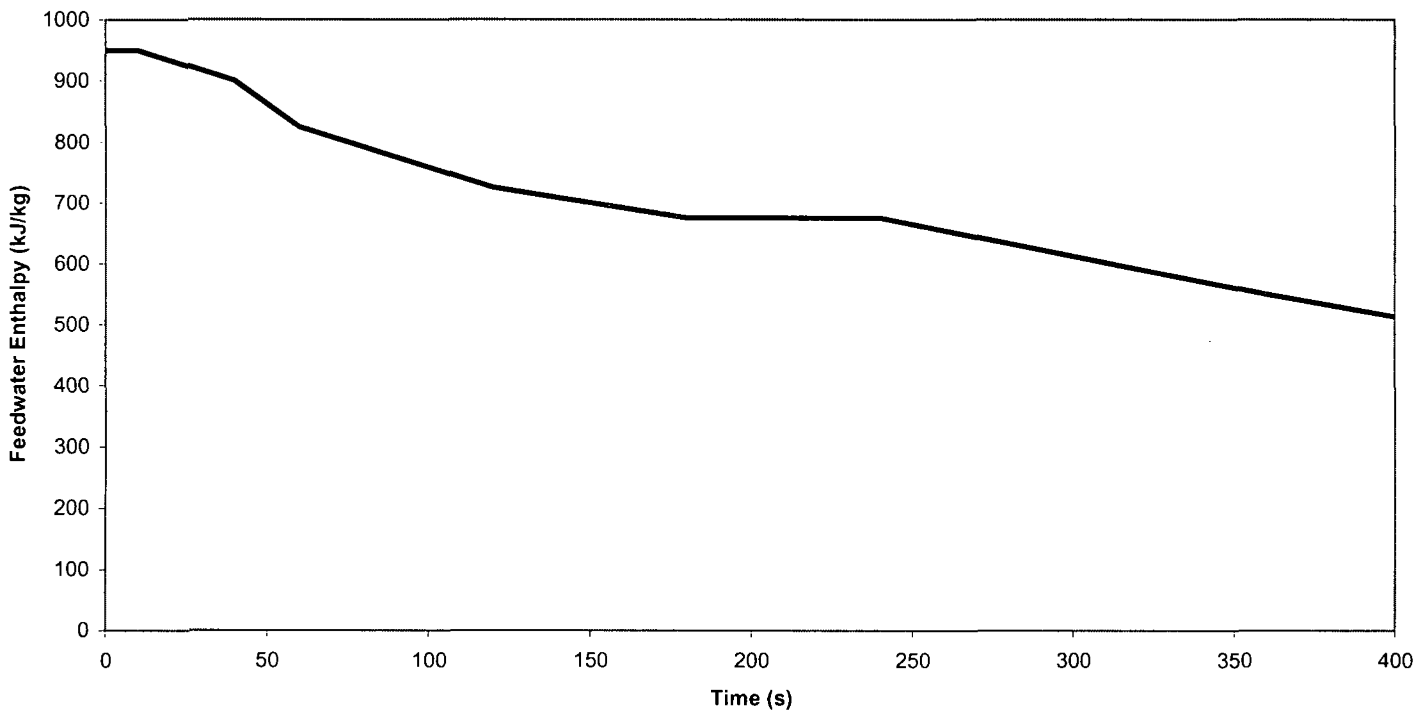


Figure 5 Revised Feedwater Line Break Enthalpy—Feedwater System Side of Break

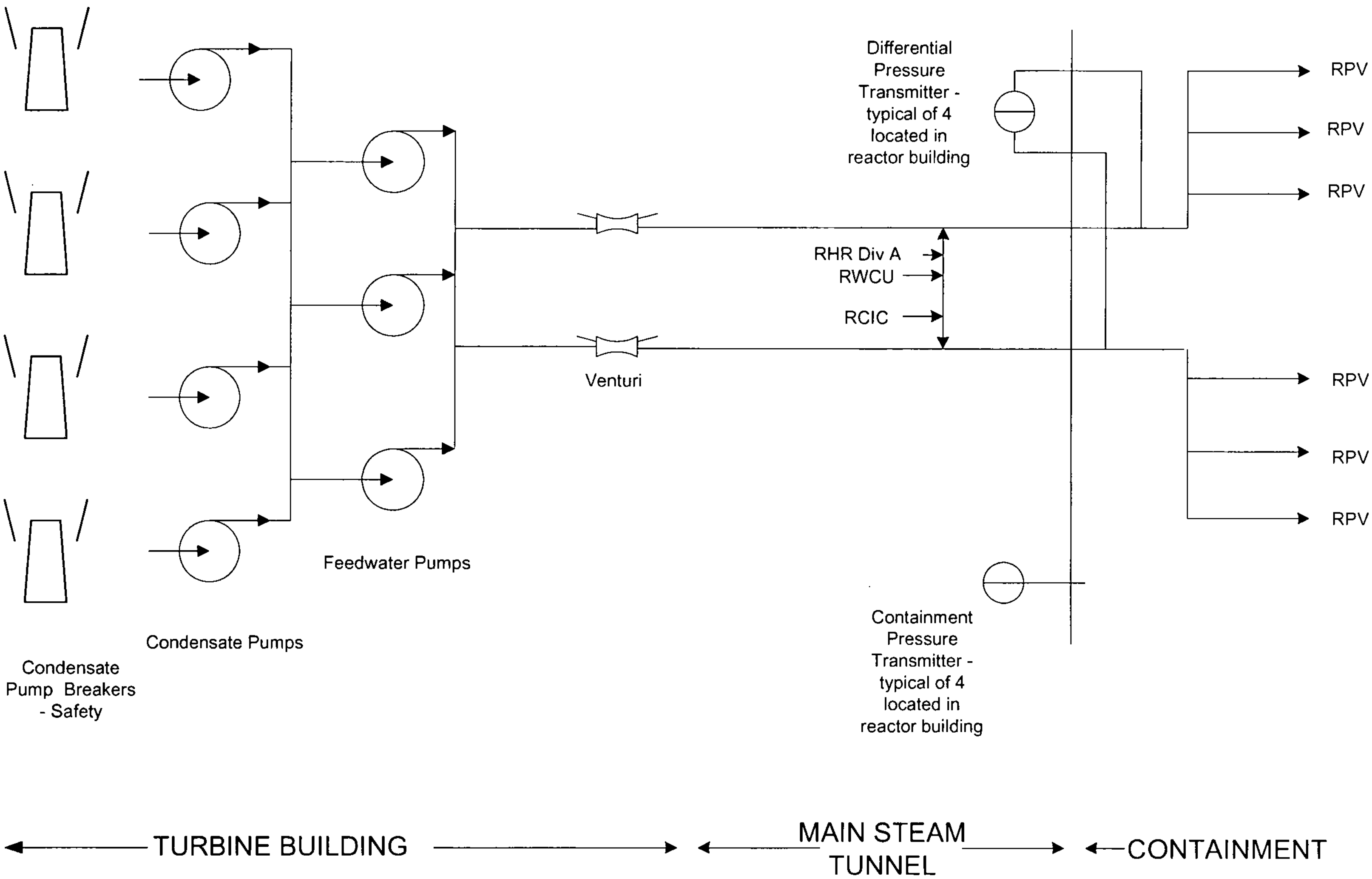


Figure 6 Feedwater Line Break Mitigation

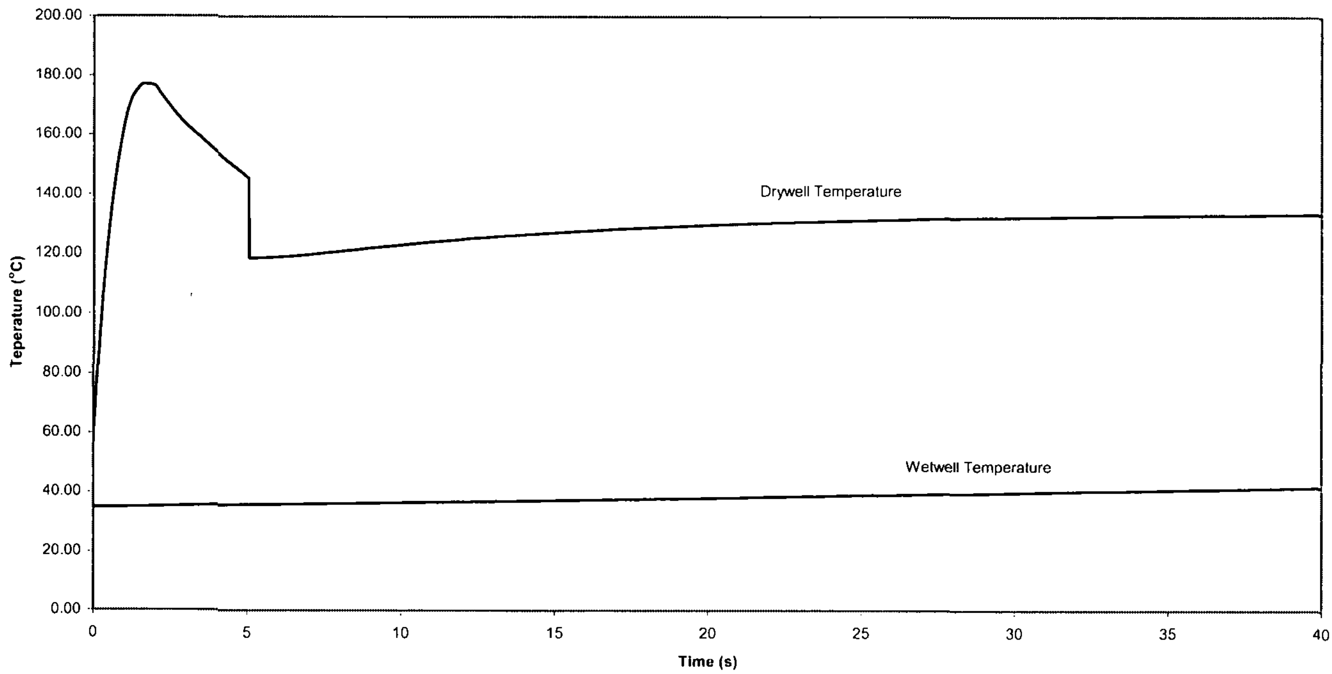


Figure 7 MSLB Short-term Temperature Response

Appendix A

Justification of Changes to the ABWR DCD

Justification for Changes to the ABWR DCD

This LTR proposes an exemption from the generic Technical Specifications (TS) in the ABWR DCD. The exemption to generic TS requires NRC approval. This LTR demonstrates that the proposed changes meet the requirements for a design certification amendment per 10 CFR 52.63(a).

10 CFR 52.63(a)(1) provides for NRC approval of changes to a standard certification necessary to bring either the certification or the referencing plants into compliance with the NRC regulations applicable and in effect at the time the certification was issued. In addition, 10 CFR 52.63(a)(1)(vi), as approved by the Commission in an affirmation session on April 17, 2007, allows for a change to a generic DCD if the change "Contributes to increased standardization of the certification information." As discussed below, the proposed changes to the generic DCD satisfy these criteria.

The proposed changes involve the implementation of a revised containment analysis that corrects identified deficiencies in the containment analysis in the ABWR DCD and are intended to be generic and applicable to all COL applicants that reference the ABWR design certification. The methodology for decay heat is has been reviewed and approved by the NRC. The FWLB mitigation (safety related logic & breakers) is similar to the EOC-RPT function that has been approved by the NRC on all BWR4/5 & 6 plants. Therefore, the proposed changes satisfy 10 CFR 52.63(a)(1).

At least one prospective COL applicant (i.e., the COL applicant for South Texas Project Units 3 and 4) intends to implement the proposed departures from the ABWR DCD. Furthermore, it may be expected that other COL applicants will also desire to implement the proposed departures.

Given the generic nature of these proposed changes and the fact that at least one COL applicant intends to make the changes, it would contribute to increased standardization if the NRC were to make a generic change to the DCD to incorporate these proposed changes. Therefore, the proposed changes satisfy the criteria in draft final 10 CFR 52.63(a)(1)(vi).

Appendix B

ABWR DCD Significant Tier 2 Marked Changes

3B Containment Hydrodynamic Loads

3B.4.1 Pressure and Temperature Transients

A LOCA causes a pressure and temperature transient in the drywell and wetwell due to mass and energy released to the drywell. The severity of this transient loading condition depends upon the type and size of LOCA. ~~*Section 6.2 provides pressure and temperature transient data in the drywell and wetwell for the most severe LOCA case [design basis accident (DBA)]. This transient data establish the structural loading conditions in the containment.*~~ Bounding pressure and temperature envelope curves for large, intermediate, and small break LOCAs are used to establish the structural loading conditions in the containment.

3B.4.2.1 Pool Boundary Loads

Structures located between 0 and ~~*7m*~~ 8.3m *above the initial surface will be subjected to impact load by an intact water ligament, where the* ~~*7m*~~ 8.3m *value corresponds to the calculated maximum pool swell height. The load calculation methodology will be based on that approved for Mark II and Mark III containments (NUREG-0487 and NUREG-0978).*

Structures located at elevations between ~~*7m*~~ 8.3m *and* ~~*10.3m*~~ 11.6m *will be subjected to froth impact loading. This is based on the assumption that bubble breakthrough (i.e., where the air bubbles penetrate the rising pool surface) occurs at* ~~*7m*~~ 8.3m *height, and the resulting froth swells to a height of 3.3m. This froth swell height is the same as that defined for Mark III containment design, and this is considered to be a conservative value for the ABWR containment design. Because of substantially smaller wetwell gas space volume (about 1/5th of the Mark III design), the ABWR containment is expected to experience a froth swell height substantially lower than that in Mark III design. The wetwell gas space is compressed by the rising liquid slug during pool swell, and the resulting increase in the wetwell gas space pressure will decelerate the liquid slug before the bubble break-through process begins. The load calculation methodology will be based on that approved for the Mark III containment*

As shown in Figure 3B-13 the gas space above the ~~*10.3m*~~ 11.6m *elevation will be exposed to spray condition* ~~*including*~~, which is expected to induce *no significant loads on structures in that region.*

As drywell air flow through the horizontal vent system decreases and the air/water suppression pool mixture experiences gravity-induced phase separation, pool upward movement stops and the "fallback" process starts. During this process, structures between the bottom vent and the ~~*10.3m*~~ 11.6m *elevation can experience loads as the mixture of air and water fall past the structure. The load calculation methodology for the defining such loads will be based on that approved for Mark III containment (NUREG-0978).*

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Table 3B-1 Pool Swell Calculated Values

Description	Value
1. Air bubble pressure (maximum)	~~133.37~~ **185.0** kPaG
2. Pool swell velocity (maximum)	6.0 m/s
3. Wetwell airspace pressure (maximum)	~~107.87~~ **154.0** kPaG
4. Pool swell height (maximum)	~~7m~~ **8.3m**

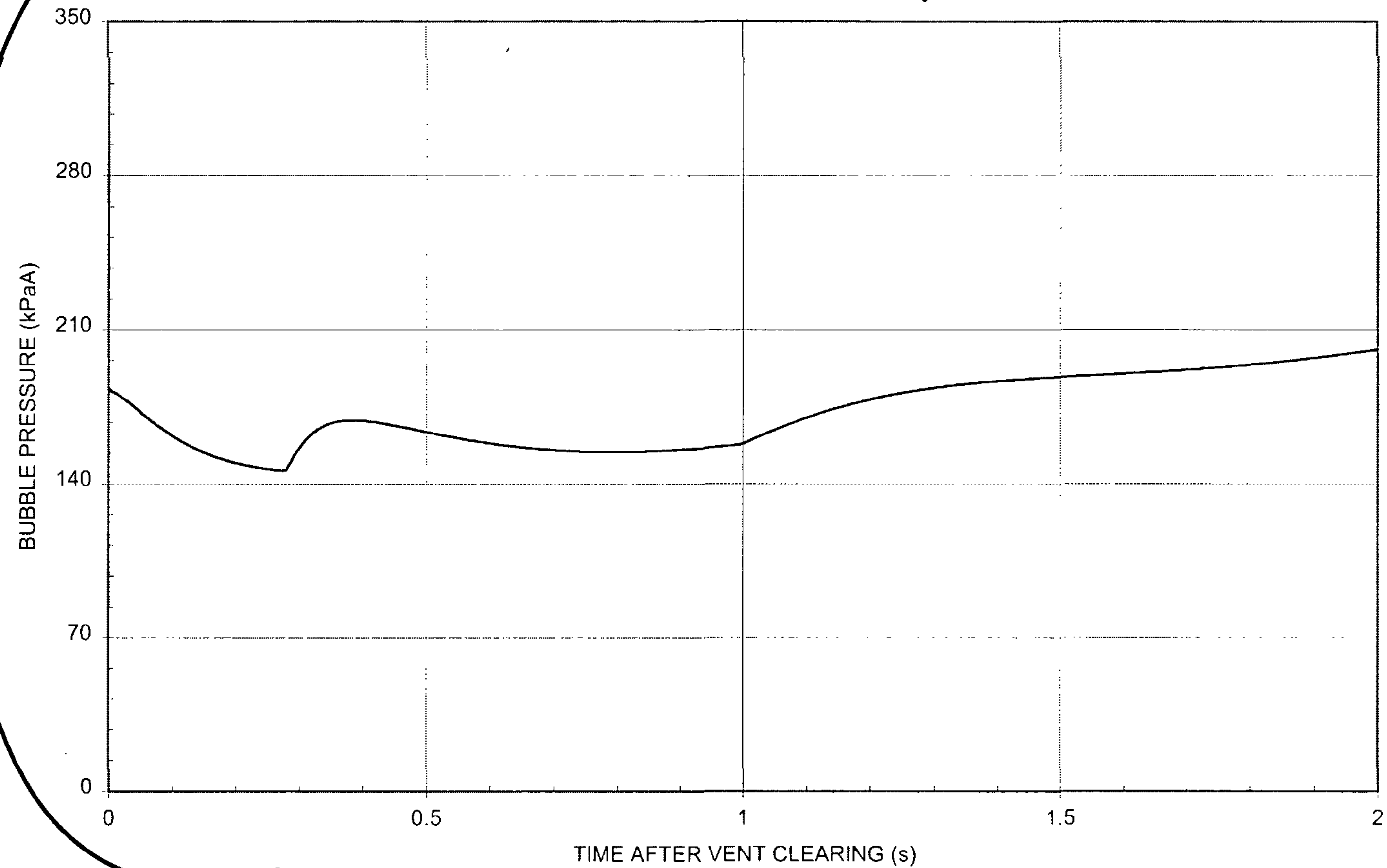

Figure 3B-12 Time History of Air Bubble Pressure

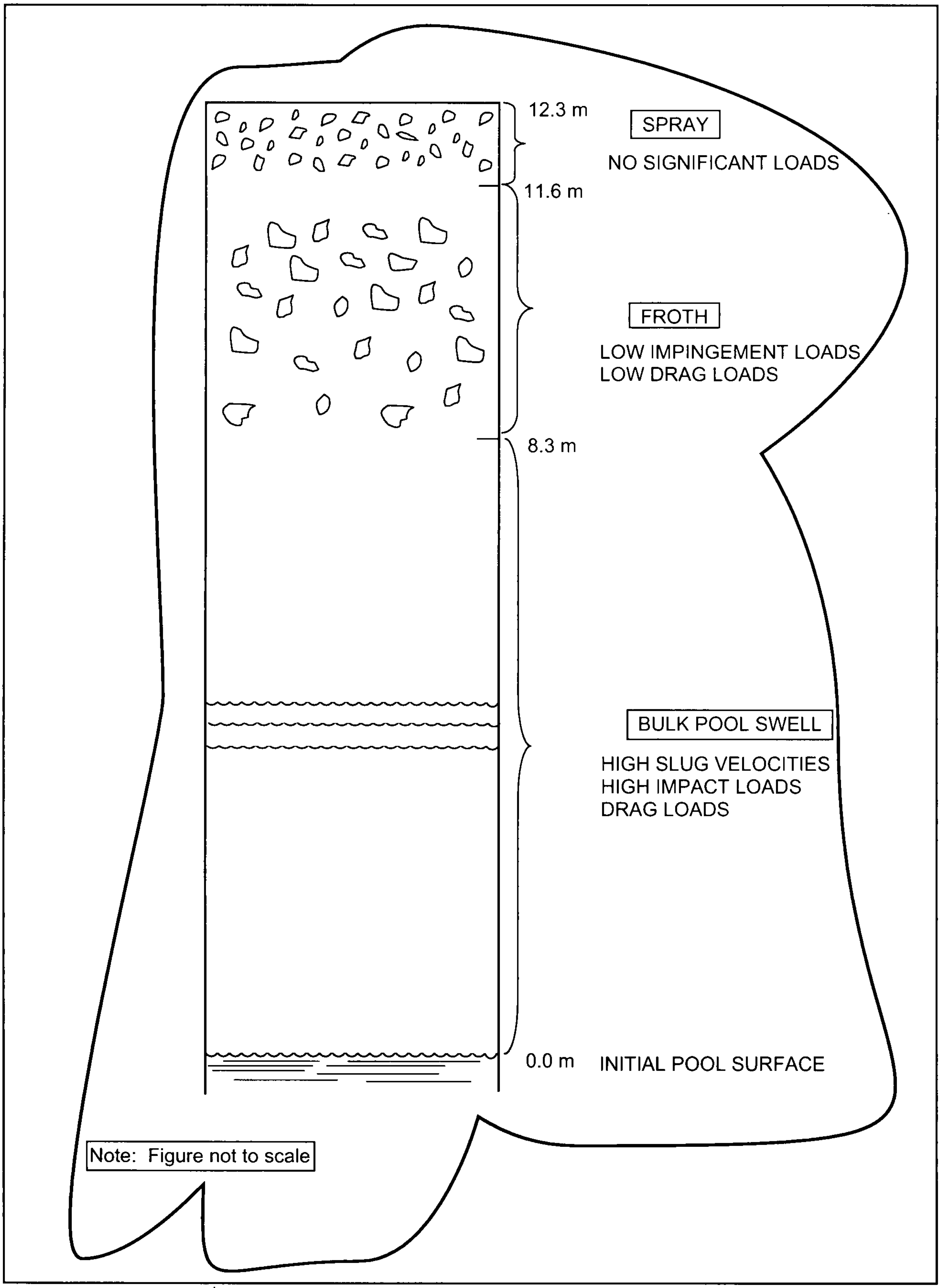

Figure 3B-13 Schematic of the Pool Swell Phenomenon

6.2 Containment Systems

6.2.1 Containment Functional Design

6.2.1.1 Pressure Suppression Containment

6.2.1.1.2 Design Features

6.2.1.1.2.1 Drywell

The drywell is designed to withstand the pressure and temperature transients associated with the rupture of any primary system pipe inside the drywell and also the rapid reversal in pressure when the steam in the drywell is condensed by the ECCS flow following post- LOCA flooding of the RPV.

A vacuum breaker system has been provided between the drywell and wetwell. The purpose of the wetwell-to-drywell vacuum relief system is to prevent backflooding of the suppression pool water into the lower drywell and to protect the integrity of the diaphragm floor (D-F) slab between the drywell and wetwell, and the drywell structure and liner. Redundant vacuum relief systems are provided to protect against failure of a single system. The design drywell-to-wetwell pressure difference is + 172.6 kPaD and – 13.73 kPaD. The vacuum breaker system is also designed to withstand the high temperature associated with the break of a small line in the drywell which does not result in rapid depressurization of the RPV.

The maximum drywell temperature occurs in case of a steamline break (177.3°C). Even though the maximum airspace temperature is above the design value (Table 6.2-1), it is only for a short time (about 1.2 seconds). Because it takes a much longer time for the drywell structural materials to increase in temperature, the drywell structural materials remain below the design temperature.~~(169.7°C) and is below the design value (171.1°C).~~

The maximum drywell pressure occurs in case of a feedwater line break (~~268.7~~ 279.6 *kPaG). The design pressure for the drywell (309.9 kPaG) includes* 11% ~~16~~*% margin.*

No vacuum breaker system is required for the primary containment-to-Reactor Building negative pressure, which is predicted to be maximum ~~11.8~~ 8.76 *kPaG, between the wetwell and the Reactor Building, compared to the design negative pressure of 13.7 kPaG.*

6.2.1.1.2.2 Wetwell

The suppression pool water is located inside the wetwell annular region between the cylindrical RPV pedestal wall and the outer wall of the wetwell. The horizontal vent system communicates the drywell to the suppression pool. The nominal submergence to the centerline of the top row of horizontal vents is 3.5m. The vertical spacing between the centerlines of the horizontal vents is 1.37m. The centerline of the bottom horizontal vent is 0.76m above the bottom of the suppression pool.

In the event of a pipe break within the drywell, the increased pressure inside the drywell forces a mixture of air, steam and water through the drywell connecting vents (DCVs) and horizontal vents into the suppression pool, where the steam is rapidly condensed. The noncondensable gases transported with the steam escape to and are contained in the free air volume of the wetwell. There is sufficient water volume in the suppression pool to provide a minimum of 0.61 meters of submergence over the top to the upper row of horizontal vents when water is removed from the pool during post-LOCA drawdown by the ECCS. This drawdown floods the RPV to the steamlines, floods the lower drywell to its drain to the DCV, and provides for water in transit from the break on its gravity drain back to the suppression pool.

The wetwell chamber design pressure is 309.9 kPaG and design temperature is ~~103.9~~ 124°C.

Performance of the pressure suppression pool concept in condensing steam under water (main steamlines through the SRVs) has been demonstrated by the horizontal vent system tests as described in Appendix 3B.

The SRVs discharge steam from the relief valves through their exhaust piping and quenchers into the suppression pool. The quencher locations within the suppression pool are identified in Figures 1.2-3c, 1.2-13i and 3B-3. Operation of the SRVs is intermittent and closure of the valves with subsequent condensation of steam in the exhaust piping can produce a partial vacuum, thereby sucking suppression pool water into the exhaust pipes. Vacuum relief valves are provided on the exhaust piping to control the maximum SRV discharge bubble pressure resulting from high water levels in the SRV discharge pipe.

Under normal plant operating conditions, the maximum suppression pool water and wetwell airspace temperature is 35°C or less. Under blowdown conditions following an isolation event or LOCA, the initial pool water temperature may rise to a maximum of 91°C at 30 minutes ~~76.7°C~~*. The continued release of decay heat after the initial blowdown may result in suppression pool temperatures as high as* 97.1 ~~97.2~~*°C. The Residual Heat Removal (RHR) System is available in the Suppression Pool Cooling mode to control the pool temperature. Heat is removed via the RHR heat exchanger(s) to the Reactor Building Cooling Water (RCW) System and finally to the Reactor Service Water (RSW) System. The RHR System is described in Subsection 5.4.7.*

6.2.1.1.3 Design Evaluation

6.2.1.1.3.2 Containment Design Parameters

Table 6.2-2 provides a listing of the key design parameters of the primary containment system including the design characteristics of the drywell, suppression pool and the pressure suppression vent system.

Table 6.2-2a provides the performance parameters of the related ESF systems which supplement the design conditions of Table 6.2-2 for containment cooling purposes during post-blowdown long-term accident operation. Performance parameters given include those applicable to full capacity operation and reduced capacities assumed for containment analyses. Analyses calculating long-term containment response following a main steamline break credited containment cooling system only, and containment sprays were not used. Analyses calculating long-term containment response following a feedwater line break used both containment cooling system and containment sprays.~~Analyses calculating long-term containment response following a feedwater line break and main steamline break used containment cooling system only, and containment sprays were not used.~~

6.2.1.1.3.3 Accident Response Analysis

The containment functional evaluation is based upon the consideration of several postulated accident conditions which would result in the release of reactor coolant to the containment. These accidents include:

(1) *An instantaneous guillotine rupture of a feedwater line*

(2) *An instantaneous guillotine rupture of a main steamline*

(3) *Small break accidents*

The containment design pressure and temperature were established based on enveloping the results of this range of analyses. ~~plus providing NRC prescribed margins.~~

For the ABWR pressure suppression containment system, the peak containment pressure following a LOCA is very insensitive to variations in the size of the assumed primary system rupture. This is because the peak occurs late in the blowdown and is determined in very large part by the transfer of the noncondensible gases from the drywell to the wetwell airspace. This process is not significantly influenced by the size of the break. In addition, there is ~~a~~ an *11% ~~15%~~ margin between the peak calculated value and the containment design pressure that will easily accomodate small variations in the calculated maximum value.*

Tolerances associated with fabrication of the RPV nozzles have been taken into account in this analysis.

~~*Tolerances associated with fabrication and installation may result in the as built size of the postulated break areas being 5% greater than the values presented in this chapter. Based on the above, these as built variations would not invalidate the plant safety analysis presented in this chapter and Chapter 15.*~~

6.2.1.1.3.3.1 Feedwater Line Break

Immediately following a double-ended rupture in one of the two main feedwater lines just outside the vessel (Figure 6.2-1), the flow from both sides of the break will be limited to the maximum allowed by critical flow considerations. The effective flow area on the RPV side is given in Figure 6.2-2. Reverse RPV flow in the second FW line is prevented by check valves shown in Figure 6.2-1. During the inventory depletion period, subcooled blowdown occurs and the effective flow area at saturated condition is much less than the actual break area. The detailed calculational method is provided in Reference 6.2-1.

The feedwater system side of the feedwater line break (FWLB) was modeled by adding a time variant feedwater mass flow rate and enthalpy directly to the drywell airspace. The time histories of the mass flow and enthalpy were determined from the operating characteristics of a typical feedwater system.

~~*The maximum possible feedwater flow rate was calculated to be 164% of nuclear boiler rated (NBR), based on the response of the feedwater pumps to an instantaneous loss of discharge pressure. Since the Feedwater Control System will respond to decreasing RPV water level by demanding increased feedwater flow, and there is no FWLB sensor in the design, this maximum feedwater flow was conservatively assumed to continue for 120 seconds (Figure 6.2-3).*~~ *This is very conservative because:*

~~*(1) All feedwater system flow is assumed to go directly to the drywell.*~~

~~*(2) Flashing in the broken feedwater line was ignored.*~~

~~*(3)*~~ *(1) Initial* reactor power is 102% NBR. ~~*feedwater flow was assumed to be 105% NBR.*~~

(2) FWLB mitigation has been added to the ABWR design as described in Section 7.3.1.1.2.

~~*(4) The feedwater pump discharge flow will coastdown as the feedwater system pumps trip due to low suction pressure. During the inventory depletion period, the flow rate is less than 164% because of the highly subcooled blowdown. A feedwater line length of 100m was assumed on the feedwater system side.*~~

The specific enthalpy time history, assuming the break flow of Figure 6.2-3, is shown in Figure 6.2-4.

6.2.1.1.3.3.1.1 Assumptions for Short-Term Response Analysis

The response of the Reactor Coolant System and the Containment System during the short-term blowdown period of the accident has been analyzed using the following assumptions:

(1) *The initial conditions for the FWLB accident are such that ~~system energy~~* the containment pressure response *is maximized ~~and the system mass is minimized~~. That is:*

   (a) *The reactor is operating at 102% of the rated thermal power, which maximizes the post-accident decay heat.*

   (b) *The initial suppression pool mass is at the* high *~~low~~ water level.*

   (c) *The initial wetwell air space volume is at the high water level.*

   (d) *The suppression pool temperature is the operating maximum temperature.*

(2) *The feedwater line is considered to be severed instantaneously. This results in the most rapid coolant loss and depressurization of the vessel, with coolant being discharged from both ends of the break.*

(3) *Scram occurs in less than one second from receipt of the high drywell pressure signal.*

(4) *The main steam isolation valves (MSIVs) start closing at 0.5 s after the accident. They are fully closed in the shortest possible time (at 3.5 s) following closure initiation. 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.*

(5) *The vessel depressurization flow rates are calculated using Moody's homogeneous equilibrium model (HEM) for the critical break flow (Reference 6.2-2). The break area on the RPV side for this study is shown in Figure 6.2-2. During the inventory depletion period, subcooled blowdown occurs and the effective break area at saturated conditions is much less than the actual area. The detailed calculational method is provided in Reference 6.2-1.*

   Reactor vessel internal heat transfer is modeled by dividing the vessel and internals into six metal nodes. A seventh node ~~depends on the fluid (saturated or subcooled liquid, saturated steam) covering the node at the time~~ models the reactor fuel. *The assumptions include:*

   (a) *The center of gravity of each node is specified as the elevation of that node.*

   (b) *Mass of water in system piping (except for feedwater) is included in initial vessel inventory.*

   (c) *Initial thermal power is 102% of rated power at steady-state conditions with corresponding heat balance parameters which correspond to turbine control valve constant pressure of* 6.85 *~~6.75~~ MPaA.*

   (d) *~~Pump heat,~~* Fuel *~~fuel~~ relaxation and metal-water reaction heat are added to the ANSI/ANS-5.1* (1971) *decay heat curve plus 20% margin.*

   (e) *Initial vessel pressure is ~~7.31~~* 7.27 *MPaA.*

(6) *There are two HPCF Systems, one RCIC System, and three RHR Systems in the ABWR. ~~One HPCF System, one RCIC System and two RHR Systems are assumed to be available. HPCF flow cannot begin until 36 seconds after a break, and then the flow rate is a function of the vessel to wetwell differential pressure. Rated HPCF flow is 182~~*

~~m^3/h per system at 8.12 MPaD and 727 m^3/h, per system at 0.69 MPaD. Rated RHR flow is 954 m^3/h at 0.28 MPaD with shutoff head of 1.55 MPaD. Rated RCIC flow is 182 m^3/h with reactor pressure between 8.12 MPaG and 1.04 MPaG, and system shuts down at 0.34 MPaG.~~These systems are not modeled since the time interval analyzed for short-term is approxiamtely the same time as the response time of associated systems injections into the RPV.

(7) *Drywell and wetwell airspaces are homogeneous mixtures of inert atmosphere, vapor and liquid water.*

(8) The wetwell airspace temperature remains in equilibrium with the suppression pool temperature ~~*The wetwell airspace temperature is allowed to exceed the suppression pool temperature as determined by a mass and energy balance on the airspace.*~~

(9) *Wetwell and drywell wall and structure heat transfer are ignored.*

(10) *Actuation of SRVs is modeled.*

(11) *Wetwell-to-drywell vacuum breakers are not modeled.*

(12) *Drywell and wetwell sprays and RHR cooling mode are not modeled.*

(13) *The dynamic backpressure model is used.*

(14) *Initial drywell conditions are 0.107 MPa, 57°C, and 20% relative humidity.*

(15) *Initial wetwell airspace conditions are 0.107 MPa, 35°C and 100% relative humidity.*

(16) *The drywell is modeled as a single node. All break flow into the drywell is homogeneously mixed with the drywell inventory.*

(17) *Because of the unique containment geometry of the ABWR, the inert atmosphere in the lower drywell would not transfer to the wetwell until the peak pressure in the drywell is achieved. Figure 6.2-5 shows the actual case and the model assumption. Because the lower drywell is connected to the drywell connecting vent, no gas can escape from the lower drywell until the peak pressure occurs. This situation can be compared to a bottle whose opening is exposed to an atmosphere with an increasing pressure. The contents of the lower drywell will start transferring to the wetwell as soon as the upper drywell pressure starts decreasing. A conservative credit for transfer of 50% of the lower drywell contents into the wetwell was taken.*

6.2.1.1.3.3.1.2 Assumptions for Long-Term Cooling Analysis

Following the blowdown period, the ECCS discussed in Section provides water for core flooding, containment spray, and long-term decay heat removal. The containment pressure and temperature response during this period was analyzed using the following assumptions:

(1) There are two HPCF Systems, one RCIC System, and three RHR Systems in the ABWR. All motor operated pump systems (HPCF and RHR) are assumed to be available. HPCF flow cannot begin until 47 seconds after a break, and then the flow rate is a function of the vessel-to-wetwell differential pressure. Rated HPCF flow is 182 m^3/h per system at 8.12 MPaD and 727 m^3/h, per system at 0.69 MPaD. Rated RHR flow is 954 m^3/h at 0.28 MPaD with shutoff head of 1.55 MPaD. ~~*The ECCS pumps are available as specified in Subsection*~~

~~6.3.1.1.2 (except one low pressure flooder feeding a broken feedwater line, in case of a FWLB).~~ *A single failure of one RHR heat exchanger was assumed for conservatism.*

(2) *The ANSI/ANS-5.1-*1994 *decay heat* plus 2-sigma uncertainty *is used. Fission energy, fuel relaxation heat, and pump heat are included.*

(3) *The suppression pool* ~~is the only heat sink available in the containemtn system~~ volume corresponds to the low water level; however, the wetwell airspace volume used corresponds to the suppresion pool at the high water level.

(4) After 30 minutes, one RHR heat exchanger is activated to remove energy via recirculation cooling of the suppression pool and one RHR heat exchanger is activated to remove energy via drywell sprays with the RCW System and ultimately to the RSW System. ~~After 10 minutes, the RHR heat exchangers are activated to remove energy via recirculation cooling of the suppression pool with the RCW System and ultimately to the RSW System. This is a conservative assumption, since the RHR design permits initation of containemnt cooling well before a 10 minute period (see response to Question 430.26).~~

(5) *The maximum service water temperature is assumed to be 35°C. This is a conservative assumption that maximizes the suppression pool temperature.*

(6) *The lower drywell flooding* ~~of 815 m³ was assumed to occur 70 seconds after scra. During the blowdown phase, a portion of break flow flows into the lower drywell. This is conservative, since lower drywell flooding will probably occur at approximately 110 to 120 second time period~~ is not modeled.

(7) Structural heat sinks are modeled in the containment system. ~~At 70 seconds, the feedwater specific enthalphy becomes 418.7 J/g (100°C saturation fluid enthalphy).~~

6.2.1.1.3.3.1.3 Short-Term Accident Responses

The calculated containment pressure and temperature responses for a feedwater line break are shown in Figures 6.2-6 and 6.2-7, respectively. ~~The peak pressure (268.7 kPaG) and temperature (140°C) occur in the drywell. The containment design pressure of 309.9 kPaG is 115% of the peak pressure.~~

~~The drywell pressurization is driven by the wetwell pressurization for stable peaks. The wetwell pressurization is a function of three major parameters:~~

~~(1) The increased wetwell air mass caused by the addition of drywell air~~

~~(2) Compression of the airspace volume due to increased suppression pool volume~~

~~(3) Increased vapor partial pressure from increasing suppression pool temperature~~

~~The suppression pool volume increase is caused by the liquid addition to the containment system from the broken feedwater line. Contribution of these parameters to wetwell pressurization is about 80% by the increased air mass, 15% by the compression effects, and 5% by the increased vapor partial pressure. Once air carryover from the drywell is completed, the wetwell and, subsequently, the drywell pressure peak occurs as the volumetric compression is completed and the pool volume begins to decrease due to the drawdown effects of the ECCS flow. Since the suppression pool volume continues to decrease as the ECCS flow continues, the short term pressure peak is the peak pressure for the transient.~~ The containment pressure response (Figure 6.2-6) covers the pool swell phase of the short-term containment response. The drywell pressure peaks

soon after bubble breakthrough as the break flow continues to push the drywell air to the wetwell. The wetwell pressure also continues to climb after this phase as the air carryover from the drywell continues.

The peak values shown in Table 6.2-1 are not based on the short-term FWLB event results.

6.2.1.1.3.3.1.4 Long-Term Accident Responses

In order to assess the adequacy of the containment system 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 in Subsection 6.2.1.1.3.3.1.2.

The ~~short~~long-term pressure peak (~~268.7~~ 279.6 kPaG) of Figure 6.2-8a is the peak pressure for the whole transient. Figure 6.2-8 shows temperature time histories for the suppression pool, wetwell, and drywell temperatures. The peak pool temperature (~~96.9~~ 97.1 °C) is reached at ~~15,350~~ 8596 seconds (~~4.264~~ 2.39 hours) and remains below the 97.2°C limit.

6.2.1.1.3.3.2 Main Steamline Break

A schematic of the ABWR main steamlines, with a postulated break in one of the main steamlines, is shown in Figure 6.2-9. The main steamline (MSL) break is a double-ended break with one end fed by the RPV directly through the broken line, and the other fed by the RPV through the unbroken main steamlines until the MSIVs are closed. Once the MSIVs are closed, the break flow is only from the RPV through the broken line.

The effective break area used for the MSL is shown in Figure 6.2-10. More detailed descriptions of the MSL break model are provided in the following:

(1) *Each MSL contains a flow limiter built into the MSL nozzle on the RPV with a throat area of 0.0983m^2, as shown in Figure 6.2-9.*

(2) *The break is located in one MSL at the inboard MSIV.*

(3) *During the inventory depletion period, the flow multiplier of 0.75 is applied (Reference 6.2-1).*

(4) *The flow resistance of open MSIVs is considered. A conservative value of 2.062 for pressure loss coefficient for two open MSIVs was taken. The nominal value is approximately 3.0. When the open MSIV resistance is considered, the flow chokes at the MSIV on the piping side as soon as the inventory depletion period ends. The effective flow area on the piping side reduces to 70% of a frictionless piping area. The value of 70% applies to flow of steam and two- phase mixture with greater than 15% quality.*

    This assumption is quite conservative because all other resistances in piping are ignored and the flow in the steamline within a one to two second period is either all steam or a two-phase mixture of much greater than 15% quality.

(5) *MSIVs are completely closed at a conservative closing time of ~~5.5~~ 5.0 seconds ~~(0.5 seconds greater than the maximum closing time plus instrument delay),~~ in order to maximize the break flow.*

6.2.1.1.3.3.2.1 Assumptions for Short-Term Response Analysis

The response of the reactor coolant system and the containment system during the short-term

blowdown period of the MSLB accident is analyzed using the assumptions listed in the above subsection and Subsection 6.2.1.1.3.3.1.1 for the feedwater line break, with the following exceptions:

~~*(1) The vessel depressurization flow rates are calculated using the Moody's HEM for the critical break flow.*~~

~~*(2) The turbine stop valve closes at 0.2 second. This determines how much steam flows out of the RPV, but does not affect the inventory depletion time on the piping side.*~~

~~*(3)*~~*(1)* The break flow is saturated steam until the two-phase level swell reaches the main steam nozzle in two seconds, thereby changing the flow quality to the RPV average quality. This provides the highest drywell pressure and temperature. ~~*The break flow is saturated steam if the RPV collapsed water level is below the MSL elevation; otherwise, the flow quality is the vessel average quality. This case provides the limiting drywell temperature.*~~

~~*Another case was evaluated with the assumption that the two phase level swell would reach the main steam nozzle in one second, thereby changing the flow quality to the RPV average quality after one second. This case provides a higher drywell pressure but a lower drywell temperature than the first assumption.*~~

~~*(4) The feedwater mass flow rate for a MSL break was assumed to be 130% NBR for 120 seconds. This is a standard MSL break containment analysis assumption based on a conservative estimate of the total available feedwater inventory and the maximum flow available from the feedwater pumps with discharge pressure equal to the RPV pressure. The feedwater enthalpy was calculated as described for the FWL break (Subsection) for 130% NBR flow, and is shown in Figure 6.2-11.*~~

~~(5)~~(2) *The SRVs are not actuated.*

6.2.1.1.3.3.2.2 Assumption for Long-Term Cooling Analysis

The containment pressure and temperature response during the period following blowdown is analyzed using the assumptions listed in Subsection 6.2.1.1.3.3.1.2 with the following exceptions:

(1) After 30 minutes, the RHR heat exchangers are activated to remove energy via recirculation cooling of the suppression pool with one division of RHR and suppression pool cooling with another division of RHR. This is a conservative assumption, since the RHR design permits initiation of containment cooling well before the 30-minute time.

(2) Feedwater flow coast down ends when the inlet temperature drops below the expected peak suppression pool temperature. The integrated enthalpy is as described in Figure 6.2-11. This break-off is conservative in order to limit water into the system that would tend to lower the peak suppression pool temperature.

6.2.1.1.3.3.2.3 Short-Term Accident Response

Figures 6.2-12 ~~*through 6.2-15*~~ and 6.2-13 *show the pressure and temperature responses of the drywell and wetwell during the blowdown phase of the steamline break accident.*

The MSLB with two-phase blowdown starting at two seconds provides the highest peak drywell temperature. The peak drywell temperature is 177.3°C, above the design value of 171.1°C. Even though the maximum airspace temperature is above the design value, it is only a short time (about 1.2 seconds). Because it takes a much longer time for the drywell structural materials to increase in temperature, the drywell structural materials remain below the design temperature. The MSLB is the limiting event for peak drywell temperature. The FWLB is the most limiting for drywell pressure.

~~*The MSLB with two-phase blowdown starting when the RPV collapsed level is at the main steamline nozzle provides the highest peak drywell temperature. The peak drywell temperature is 169.7°C, below the design value of 171.1°C, and is the limiting one as compared to the FWLB peak temperature. The peak drywell pressure for the MSLB remains below that for the FWLB, which becomes the most limiting.*~~

6.2.1.1.3.3.2.4 Long-Term Accident Response

Figures 6.2-14 and 6.2-15 show the pressure and temperature responses of the drywell and wetwell for the long-term phase of the steamline break accident.

The long-term MSLB provides the highest peak wetwell pressure. This peak value remains below the design limit (Table 6.2-1).

~~*The long-term containment pressure and temperature responses following the MSLB accident remain below those for the feedwater line break, which is the most limiting event.*~~

6.2.1.1.3.4 Accident Analysis Models

6.2.1.1.3.4.1 Short-Term Pressurization Model

The analytical models, assumptions and methods used to evaluate the containment response during the reactor blowdown phase of a LOCA are described in References ~~*s*~~ *6.2-1 and* ~~*6.2-2*~~ **6.2-3**.

6.2.1.1.3.4.2 Long-Term Cooling Model

Once the RPV blowdown phase of the LOCA is over, a fairly simple model of the drywell and wetwell may be used. During the long-term post-blowdown transient, the RHR cooling system flow path is a closed loop and the suppression pool mass will be constant.

The analytical models, assumptions and methods used to evaluate the containment response during the long-term cooling phase of a LOCA are described in Reference 6.2-3.

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6.2.1.1.4 Negative Pressure Design Evaluation

During normal plant operation, the inerted wetwell and the drywell volumes remain at or slightly above atmospheric conditions. However, certain events in the containment cause depressurization transients that can create a negative pressure differential across the diaphragm floor and lower drywell access tunnels (negative means the wetwell pressure is greater than the drywell pressure) and a negative pressure differential across the drywell and the wetwell walls (negative means the Reactor Building pressure is greater than the containment pressure). Vacuum relief function is necessary in order to limit these negative pressure differentials within design values. The events which cause the containment depressurization are:

(1) The drywell/wetwell sprays are inadvertently actuated during normal operation.

(2) The drywell is depressurized following a LOCA.

(3) The wetwell spray is actuated subsequent to a stuck open relief valve (SORV).

However, the design/expected operating conditions preclude the first event (inadvertent DW spray during normal operation). There are features on the ABWR that prevent the initiation of the RHR mode of the drywell spray(s) during normal plant operation. The first is an interlock on the drywell spray injection valves that requires high drywell pressure to be present before the valves are allowed to be opened. Also, there is a time delay in the logic that will allow initiation of drywell spray 60 seconds after the drywell high pressure signal (2 psig) is received. In addition, the RHR system can only be manually initiated in the drywell spray mode from the main control room by two methods, both requiring two independent actions. Therefore, the probability of a spurious initiation of drywell spray during normal plant operation is very remote.

Drywell depressurization following a ~~FWLB~~ LOCA results in the severest pressure transient in the drywell; this transient is therefore used in sizing the Wetwell-to-Drywell Vacuum Breaker System (WDVBS). The most severe depressurization in the wetwell is caused by wetwell spray actuation subsequent to a stuck open relief valve. The analysis of this transient shows that the Primary Containment Vacuum Breaker System (PCVBS) is not required.

6.2.1.1.4.1 Wetwell-to-Drywell Negative Differential Pressure

The WDVBS is sized to keep the differential pressure between the drywell and wetwell within the negative design values for the PCV, diaphragm floor, and tunnels during all operating and accident transients.

Without the WDVBS, the post-LOCA drywell pressure may decrease to the saturation pressure (20.6-27.5 kPaA) of the drywell spray flow or the break flow out of the RPV, and the wetwell pressure may be still around 275.6 kPaA, creating the negative pressure differential close to 275.6

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kPaD. The primary purpose of the WDVBS is to prevent such a large negative pressure differential between the drywell and wetwell. In addition, the WDVBS can hold the drywell pressure above the negative design pressure of the PCV liner. This is achieved by the transfer of air from the wetwell to the drywell.

Two LOCA events are analyzed to check the adequacy of the WDVBS design. They are:

Event 1: ECCS reflood following a LOCA (FWLB) and actuation of the DW sprays without actuation of the WW sprays.

Event 2: Small Steam Line Break (0.00093 m^2) with DW spray actuation and without actuation of the WW sprays.

During a LOCA (FWLB or SSLB) event, the air initially in the drywell will be purged into the wetwell air space and the drywell will be filled with mostly steam. During this period, the wetwell will be pressurized due to air flow from the drywell, and the drywell will experience a rapid depressurization due to steam condensation when the drywell spray is initiaied. Without WDVBS, relatively cold break flow which occurs for the FWLB when the RPV is flooded with ECCS wateror during drywell spray flow could result in a negative pressure differential exceeding the design value of 13.7 kPaD. The primary purpose of the WDVBS is to keep the negative pressure differential between the drywell and wetwell within its design value. In addition, the WDVBS can hold the drywell pressure above the negative design pressure of the PCV liner. This is achieved by the transfer of gas from the wetwell to the drywell through the WDVBS.

For each of the two LOCA events identified above, an event scenario is developed such that the WW-to-DW negative pressure differential is maximized.

For Event 1, the following assumptions are made:

(1) The RPV is initially in a hot standby condition with negligible decay heat generation.

(2) The initial pressure for both the DW and WW is 101.1 kPaA.

(3) The initial relative humidity for both the DW and WW is 100%.

(4) The initial temperature for the DW and WW is 57.2°C and 35°C, respectively.

(5) The service water temperature for the RHR is 15.6°C.

(6) HPCF and RCIC flow taken from the condensate storage tank at 15.6°C.

(7) The ECCS is comprised of 2 HPCF, 1 RCIC, and 3 RHR (2 containment sprays – Divisions B and C, and 1 LPFL) - no single failure in ECCS.

(8) The DW spray is initiated when the WW pressure peaks. Since peak WW occurs before 2 minutes into the event, the DW spray is assumed to initiate at 2 minutes. The design precludes initiation of the DW spray before 2 minutes.

(9) The vacuum breakers are fully open when the WW pressure is 3.45 kPa higher than the DW.

For Event 2, the assumptions are the same as Event 1, except for the following:

(1) The RPV is initially at 102% of rated power.

(2) Scram occurs, concurrent with occurrence of 0.00093 m^2 small steam line break.

(3) The DW spray is initiated when the WW pressure peaks. Since peak WW pressure does not occur within 30 minutes into the event, the DW spray is assumed to initiate at 30 minutes. This assumption is based on the premise that the operator is expected to initiate containment cooling with the DW spray at latest at 30 minutes into the event.

(4) For this event, an additional sensitivity case was analyzed by assuming 106.52 kPaA for the initial DW and WW pressure and 20% relative humidity for the DW.

The two LOCA events were analyzed by assuming that one of eight vacuum breakers in the WDVBS is out of service. The total vacuum breaker flow area is characterized by the ratio $A/\sqrt{k}$, where A is the actual flow area of the vacuum breaker and k is its pressure loss coefficient. The value of $A/\sqrt{k}$ is calculated to be 0.82 m^2 with one vacuum breaker out of service. The calculated negative pressure differential between the wetwell and drywell for the two events is 3.86 kPaD which is well below the design value of 13.7 kPaD; the adequacy of the WDVBS design is confirmed. The pressure-time histories for Event 1 are shown in Figure 6.2-17.

~~*The specific requirements to be met by the WDVBS are:*~~

~~*(1) The drywell to wetwell negative differential pressure shall be less than 13.7 kPaG. This limits the negative pressure differential across the diaphragm floor, tunnels, and the pedestal.*~~

~~*(2) The drywell to Reactor Building negative pressure shall be less than 13.7 kPaG. This requirement protects the PCV liner portion on the drywell portion of the containment.*~~

~~*Drywell depressurization is caused by two major events:*~~

~~*(1) Post LOCA drywell depressurization*~~

~~*(2) Inadvertent drywell spray actuation during normal operation.*~~

~~*The former causes a much larger depressurization in the drywell; this depressurization would become the most severe if a break occurred in a feedwater line. Hence, the feedwater line break post LOCA transient is the limiting event for the sizing of the WDVBS. Following the break, the pressurization of the drywell causes the air in the drywell to be purged into the wetwell airspace, leaving the drywell full of steam. Subsequent condensation of this steam by cold ECCS flow through the break results in the depressurization of the drywell. This depressurization follows the general trend shown in Figure .*~~

~~*As the RPV is reflooded, the ECCS flow begins to cascade down through the break and into the drywell, causing the initial drywell depressurization (Region I in Figure). The pressure differential between the drywell and wetwell causes suppression pool water to flow into the drywell, accelerating the drywell depressurization rate evenfurther (Region II). When the pressure difference between the drywell and wetwell reaches a predetermined setpoint, the WDVBS opens, allowing the flow of air back into the drywell, thus slowing down its depressurization, and eventually reaching a steady state (Region III). As can be observed, the maximum negative pressure differential between the wetwell and drywell occurs during the depressurization of the drywell and can be controlled by proper sizing of the WDVBS.*~~

~~*Drywell to Reactor Building negative pressure differentials can exist during both drywell*~~

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~~*depressurization and the steady state condition which takes place as the drywell pressure approaches the wetwell pressure. The drywell and wetwell pressures decrease slightly below the initial containment pressure because the steam condenses due to the drywell spray or the cold break flow as the air is evenly distributed in the PCV.*~~

~~*Limiting conditions are selected such that the initial drywell depressurization is the most severe. This determines the WDVBS size to meet the drywell to wetwell negative design pressure requirement. The following case was found to be the limiting one:*~~

~~*(1) No decay heat in the RPV, as would be the case during a hot standby condition (not reactor isolation)*~~

~~*(2) No drywell sprays.*~~

~~*(3) Maximum break flow.*~~

~~*(4) No pool cooling.*~~

~~*(5) Maximum allowable wetwell temperature prior to LOCA.*~~

~~*(6) ECCS flow taken from the condensate storgae tank at 15.6°C.*~~

~~*(7) The ECCS is comprised of 2 HPCFs, 1 RCIC, and 3 RHR LPFLs (no single failure in ECCS).*~~

~~*Additionally, the limiting event and conditions were considered for the PCV negative design pressure requirement on the drywell part during steady state operations. The limiting event is the same as the one above and all conditions are the same except that the wetwell spray was activated.*~~

~~*The following assumptions were made during the analysis model:*~~

~~*(1) Suppression pool and wetwell airspace temperatures prior to the LOCA are 35°C.*~~

~~*(2) Minimum condensate storage tank temperature is 15.6°C.*~~

~~*(3) Maximum combination of HPCF, RCIC and LPFL fows is 4316 m³/h and remains at this value throughout the event.*~~

~~*(4) Any liquid flow into the drywell remains in the drywell airspace..*~~

~~*When the drywell pressure first drops below the wetwell pressure, the following conditions exist in the containment:*~~

~~*(1) Pressure in the drywell is 271.6 kPaA.*~~

~~*(2) Pressure in the wetwell is 273.6 kPaA.*~~

~~*(3) Drywell Temperature is 130.1°C.*~~

~~*(4) Wetwell temperature is 98.4°C.*~~

~~*(5) Relative humidity in the drywell is 100%.*~~

~~*(6) Relative humidity in the wetwell is 12.9%.*~~

~~*(7) Height of water in the suppression pool is 7.62m.*~~

~~(8) Suppression pool temperature is 51.5°C.~~

~~(9) Height of water in the horizontal vent vertical pipes is 7.5 m.~~

~~Other physical parameters of importance to this transient are:~~

~~(1) Surface area of the suppression pool (wetwell side) = 506.6m^2.~~

~~(2) Total flow area of drywell connecting vents – 11.3 m^2.~~

~~(3) Lower drywell volume – 1644.4 m^3.~~

~~(4) Upper drywell volume – 5493.7 m^3.~~

~~(5) Air volume ratio (wetwell/drywell) – 0.81.~~

~~(6) Vacuum breakers start opening at 0.69 kPaD, and fully open at 3.43 kPaD.~~

~~The vaccum break size is characterized by the ratio~~

$$\cancel{A/\sqrt{k}}$$

~~, where A is the actual flow area of the vacuum breaker and k its pressure loss coefficient. When~~

$\cancel{A/\sqrt{k} \geq}$

~~the calculated negative pressure differential is 9.8 kPaD between the wetwell and drywell. The pressure time histories are shown in Figure . Thus, a WDVBS effective area of 0.77m^2 is adequate to satisfy the drywell to wetwell negative design pressure requirements of 13.7 kPaD.~~

~~With the WDVBS size determined above, the PCV negative design pressure on the drywell side is checked. This analysis utilizes the wetwell spray in order to minimize the wetwell/drywell pressure. Figure shows the pressure time histories for the wetwell and drywell. It should be noted that no drastic depressurization occurs because the WDVBS has sufficient size to prevent the initial rapid depressurization in the drywell. In addition, the wetwell airspace contains a large amount of air and the wetwell spray capacity is less than 15% of the drywell break flow capacity. The lowest containment pressure, and thus the maximum PCV to Reactor Building negative pressure, occurs during the steady state end of the transient. The final pressure becomes lower than the initial containment pressure because the drywell/wetwell sprays decrease the vapor partial pressure and cool the air in the PCV as the WDVBS equalizes the pressure in the drywell to that in the wetwell.~~

~~The maximum negative pressure is 5.9 kPaG for the drywell and wetwell, which satisfies the PCV negative design pressure requirement of 13.7 kPaG.~~

~~With a typical vacuum breaker diameter of 50.8 cm and a flow loss coefficient (k) of 3, the required number of wetwell to drywell vacuum breakers is eight, which considers one single failure in the WDVBS. The total flow area for eight vacuum breakers is 1.53m^2.~~

~~Vacuum breakers are intended to be swing check type valves which open passively due to negative differential pressure (wetwell gas space pressure greater than the drywell pressure) across the valve dish, and require no external power to actuate them. These valves are installed horizontally locating in wetwell gas space, one valve per penetration (through pedestal wall) opening into lower~~

~~*drywell. This position location protects vacuum breaker valves from being subjected to cyclic pressure loading during LOCA steam condensation period. Position location of these valves, both axially and azimuthally, are shown in Figures 1.2.3c and 1.2.13k.*~~

~~*In view that these vacuum breaker valves are located in the wetwell gas space, they can be subjected to loads due to pool swell during early phase of a loss of coolant accident. The containment design will provide features, as appropriate, which will protect these valves from applicable loads due to pool swell. For example, the design may include features which protect the valves by designing catwalk structure below the valves as a solid plate of sufficient area assuring complete structural shielding of vacuum breakers which are located (approximately) 1m above the catwalk platform from possible direct pool swell impact loads, as well as protection from possible water fallback associated with flow around edges of solid catwalk area. The COL applicant will review the issue of providing appropriate structural features protecting these valves from pool swell loads and propose to the NRC staff an appropriate design for assuring that these valves are protected adequately. The structural shielding will be designed for pool swell loads determined based on the methodology approved for Mark II/III designs. For design of structural shielding features, pool swell loads to the maximum practical extent will be defined. See Subsection for COL license information.*~~

6.2.1.1.4.2 Wetwell-to-Reactor Building Negative Differential Pressure

Since the WDVBS meets the PCV negative design pressure requirement on the drywell, additional analyses were performed to determine need for the PCVBS to satisfy the PCV negative design pressure requirement on the wetwell.

The wetwell-to-Reactor Building negative pressure shall be less than 13.7 kPaG to protect the PCV liner in the wetwell.

The ABWR plant does not have vacuum breakers between the containment and the reactor building. Three limiting containment depressurization events are analyzed to confirm that the above requirement is met without vacuum breakers between the primary containment and reactor building. Inadvertent actuation of drywell sprays is not included as a limiting event because of the design features discussed in Section 6.2.1.1.4.1. The limiting events analyzed are:

Event 3: LOCA (FWLB) event with containment (DW and WW) sprays on.

Event 4: SSLB with containment (DW and WW) sprays on.

Event 5: Stuck Open Relief Valve Event with DW and WW sprays on

To develop the scenarios for these three events the following are considered:

During any of the above events, the suppression pool temperature increases as the event progesses. As a result, the DW and WW spray temperature will also increase as the event progresses. Since a colder spray temperature produces a greater and more rapid pressure drop, the analyses simulate spray initiation at the earliest possible times.

Plant emergency operation procedures call for termination of containment spray when the containment pressure decreases below 101.35 kPaA. Therefore, this operation procedure is considered in the event scenario, assuming that it would take 30 minutes for the operator to terminate the spray after a reading of 101.35 kPaA containment pressure.

The event scenarios and results for the three analyses used to evaluate Containment-to-Reactor Building negative differential pressure are described as follows:

Event 3: LOCA (FWLB) with DW and WW sprays on

This event is identical to Event 1 described in Section 6.2.1.1.4.1, except that:

> The WW spray is turned on at one minute into the event and the DW spray is turned on at two minutes into the event.
>
> Both DW and WW sprays are turned off 30 minutes after either DW or WW airspace pressure decreases to 101.35 kPaA.

For this event, the containment pressure and temperature initially increase due to the break flow from the reactor. There is an initial rapid decrease in the drywell and wetwell pressures when the containment sprays are turned on. After the reactor blowdown period is completed and the ECCS flow floods the vessel to the break location, relatively colder break flow occurs. This results in a further decrease in the containment pressures and temperatures. The containment pressure ultimately decreases to 101.35 kPaA around 20 minutes into the event. Thirty minutes later, the containment sprays are terminated by the operator. At that time, the containment pressure reaches its minimum value. The calculated maximum Reactor Building -to-Primary Containment differential pressure for this event is 2.62 kPaD which is less than the containment negative design value of 13.7 kPaD.

Event 4: SSLB with DW and WW sprays on

This event is identical to Event 2 described in Section 6.2.1.1.4.1, except that:

> The WW spray is turned on at one minute into the event and the DW spray is turned on at two minutes into the event.
>
> Both DW and WW sprays are turned off 30 minutes after either DW or WW airspace pressure decreases to 101.35 kPaA.

For this event, the containment experiences a rapid depressurization (below 101.35 kPaA) shortly after initiation of the containment spray which levels out at about 5 minutes into the event. Afterwards the drywell and wetwell pressure increase gradually for the remainder of the event. Therefore, only the initial containment depressurization shortly after spray initiation is of concern for this event.

The calculated maximum RB-to-DW and RB-to-WW differential pressure for this event are 9.10 kPaD and 5.86 kPaD, respectively, which are less than the containment negative differential pressure design value of 13.7 kPaD.

Event 5: SORV with DW and WW Sprays On

During an SORV event, SRV discharge to the suppression pool heats up the suppression pool and also increases the wetwell airspace pressure and temperature. When the pressure in the wetwell becomes greater than the drywell pressure, the WDVBS allows the flow of air from the wetwell to the drywell,. This results in a gradual increase in both the drywell and wetwell pressure. For this analysis, it is assumed that the reactor will scram when the pool temperature reaches 43.3°C. The analysis for this event is performed from this point; namely from the time of 43.3°C suppression pool temperature when a reactor scram occurs. It is assumed for the analysis that the drywell temperature is kept at 57.2°C operating temperaure due to

the operation of the drywell cooler. The drywell and wetwell pressure may be higher than the minimum operating pressure of 101.35 kPaA due to the SRV discharge to the suppression pool prior to reactor scram. However, for this analysis, it is conservatively assumed that the drywell and wetwell pressure is 101.35 kPaA at the time of reactor scram Thus, the following initial conditions are assumed for this event:

- SP temperature is 43.3°C.
- WW airspace temperature is at 43.3°C.
- DW temperature is 57.2°C.
- DW relative humidity is 100%.
- DW pressure is 101.35 kPaA.
- WW pressure is 101.35 kPaA.

With the initial conditions above, the following is also assumed:

(1) The RPV is initially at 102% of rated power.

(2) Reactor scram occurs, concurrent with occurrence of stuck-open relief valve.

(3) The WW spray is turned on at one minute into the event and the DW spray is turned on at two minutes into the event.

(4) Both DW and WW sprays are turned off 30 minutes after either DW or WW airspace pressure decreases to 101.35 kPaA.

Similar to Event 4, the containment experiences a rapid depressurization (below 101.35 kPaA) shortly after initiation of the containment spray which levels out at about 6 minutes into the event. Afterwards the drywell and wetwell pressures increase gradually for the remainder of the event duration. Therefore, only the initial containment depressurization shortly after spray initiation is of concern for this event.

The calculated maximum Reactor Building (RB)–to-DW and RB-to-WW differential pressure for this event is 12.13 kPaD and 8.76 kPaD, respectively, which are less than the containment negative design value of 13.7 kPaD.

As shown above, for all three events the containment-to-reactor building differential pressure without vacuum breakers is less than the design pressure of 13.7 kPaD. Figure 6.2-18 shows the pressure-time histories for the wetwell and drywell for Event 5, which is the limiting event with respect to the minimum containment pressure.

~~*The following wetwell depressurization Events [(1),(2) and (3)] which may result in negative differential pressure in the wetwell were considered:*~~

~~*(1) Drywell and wetwell spray actuation during normal operation*~~

~~*(2) Wetwell spray actuation subsequent to stuck open relief valve (SORV)*~~

~~*(3) Drywell and wetwell spray actuation following a LOCA*~~

~~*The depressurization results presented in the previous subsection indicate that maximum negative*~~

~~*pressure in the wetwell for the Event (3) conditions is expected to be 5.9 kPaG, which satisfies the PCV negative design pressure requirement of 13.7 kPaG without the PCVBS. Events (1) and (2) were analyzed to determine the limiting maximum negative pressure in the wetwell, and conclude whether or not the PCVBS is required.*~~

~~*(1): Drywell and wetwell spray actuation during normal operation*~~

~~*The key assumptions and initial conditions used in analyzing this event are:*~~

~~*(1) Inert gas behaves as a perfect gas.*~~

~~*(2) Initial drywell temperature is 57.2°C.*~~

~~*(3) Initial wetwell temperature is 35°C.*~~

~~*(4) Initial containment (drywell and wetwell) pressure is 101.1 kPaA.*~~

~~*(5) Initial drywell relative humidity is 20%.*~~

~~*(6) Initial wetwell relative humidity is 100%.*~~

~~*(7) Wetwell and drywell spray water source is the suppression pool.*~~

~~*(8) Drywell spray flow rate is 954 m^3/h.*~~

~~*(9) Wetwell spray flow rate is 160 m^3/h.*~~

~~*(10) Initial suppression pool temperature is 35°C.*~~

~~*(11) WDVBS area is 0.77m^2.*~~

~~*(12) . No PCVBS modeled.*~~

~~*Recognizing that drywell initial relative humidity and suppression pool initial temperature (suppression pool is the water source for sprays), an additional case representing a non-mechanistic and conservative combination of these two input parameters was also analyzed. The two cases which were analyzed for this event are:*~~

~~*(a) Initial conditions and assumptions as listed above.*~~

~~*(b) Same as case a, except drywell initial relative humidity of 60%, and suppression pool initial temperature of 23.9°C.*~~

~~*The calculated maximum negative differential pressure in the wetwell for cases a and b is found to be 6.9 kPaG and 11.8 kPaG, respectively. These results show that the containment design satisfies the PCV negative design pressure requirement of 13.7 kPaG, without PCVBS.*~~

~~*Event (2): Wetwell Spray Actuation Subsequent to SORV.*~~

~~*The effect of SRV discharge to the suppression pool is to heat the wetwell airspace, thus increasing its pressure. When the pressure in the wetwell becomes greater than the drywell pressure, the WDVBS allows the flow of air from the wetwell to the drywell, thereby pressurizing both volumes. The wetwell pressure and temperature peak when the reactor decay heat decreases below the heat removal from the continued pool cooling and wetwell spray. The wetwell temperature and pressure decrease, but the drywell pressure remains at its peak value. When the pressure difference between the two volumes becomes greater than the hydrostatic head of water above the top vent, air flows back into the wetwell airspace, slowing down the wetwell depressurization rate. The pressure differential between the drywell and wetwell is maintained constant at the hydrostatic head above the top row of horizontal vents. The final pressure in the wetwell is lower than the Reactor*~~

~~Building (R/B) pressure because more air is transferred to the drywell during wetwell pressurization than is received during wetwell depressurization.~~

~~The following assumptions are made in analyzing the above event:~~

~~(1) Inerted gas behaves as a perfect gas.~~

~~(2) Temperature in the drywell remains at 57.2°C throughout the transient by means of the drywell cooler.~~

~~(3) Initial wetwell temperature is 35°C.~~

~~(4) Initial containment pressure is 101.1 kPaA~~

~~(5) Maximum suppression pool temperature is 97.2°C.~~

~~(6) Wetwell spray is from the suppression pool.~~

~~(7) Initial wetwell spray temperature is 35°C.~~

~~(8) Capacity of the RHR heat exchanger is 0.371 MJ/s°C.~~

~~(9) Maximum wetwell temperature is determined by the maximum wetwell spray temperature and the pool surface heat transfer to the wetwell airspace.~~

~~(10) Convective heat transfer coefficient between the suppression pool and the wetwell airspace is 41.0 kJ/hm²°C.~~

~~(11) Mixture of steam and air in the drywell is homogeneous such that the ratio of its partial pressures remain constant after the peak pressure is attained.~~

~~(12) Air content of the horizontal vent flow mixture increases the wetwell pressure.~~

~~(13) Drywell pressure is equal to the wetwell pressure when the peak pressure is reached.~~

~~(14) Wetwell vapor pressure is equal to the saturation pressure at the wetwell temperature due to the wetwell spray.~~

~~(15) Initial relative humidity in the drywell is 20%.~~

~~(16) Initially, the suppression pool is at the High Water Level point.~~

~~(17) Wetwell spray flow rate is 114 m³/h.~~

~~An analysis was conducted with no PCVBS, and the maximum negative differential pressure between the wetwell and the Reactor Building was determined to be 11.8 kPaD. This shows that the SORV is a much more severe event than the Event (3) (during which the maximum negative differential pressure is 5.9 kPaG) and Event 1 (during which the maximum differential pressure is 9.8 kPaG) transients. Therefore, the PCV negative pressure requirement of 13.7 kPaG on the wetwell side can be met without PCVBS.~~

6.2.1.1.5 Steam Bypass of the Suppression Pool

6.2.1.1.5.4 Bypass Capability With Containment Spray and Heat Sinks

An analysis has been performed which evaluates the bypass capability of the containment for a spectrum of break sizes considering containment sprays and containment structural heat sinks as means of mitigating the effects of steam bypass of the suppression pool.

The containment system design provides two RHR spray loops, and each loop consists of both wetwell and drywell sprays. In operation of RHR in spray mode, the wetwell and drywell sprays activate simultaneously. Per loop, the design flow rate of drywell spray is about ~~800~~ 840 *m^3/hour, and that of wetwell spray is about 114 m^3/hour. In this analysis it is assumed that spray is to be initiated no sooner than 30 minutes after the wetwell gas space pressure is reached to ~~103.0~~* 103.4 *kPaG. This assumed value of spray initiation pressure set point, which is higher than the EPGs pressure set point of 71.6 kPaG, is expected to produce slightly conservative results. The suppression pool water passes through the RHR heat exchanger and is then injected into the drywell and wetwell spray headers located respectively in the upper region of drywell and wetwell gas space. The spray will rapidly condense the stratified steam, creating a homogeneous air-steam mixture in the containment. Structural heat sinks (drywell and wetwell boundary surfaces) were considered with variable convective heat transfer coefficients based on Uchida correlation. The reactor vessel shutdown rate was assumed to be 55.6°C/h, and the maximum design service water temperature was used. This shutdown rate corresponds to the maximum rate which does not thermally cycle the reactor vessel. This analysis results in an allowable maximum steam bypass leakage capability of*

$A/\sqrt{K}$

of 50 cm^2, meeting the criterion that calculated maximum containment pressure remain below the containment design pressure. Allowable leakage capacity vs primary system break area is shown in Figure 6.2-42.

The key assumptions for allowable steam bypass calculations utilizing structural heat sinks are summarized as follows:

(1) *Following the occurrence of a pipe line break within the drywell, air is purged through the vents into the wetwell.*

(2) *Flow through the postulated leakage path is pure steam. For a given leakage path, if the leakage flow consists of mixture of liquid and vapor, the total leakage mass flow rate is higher, but the steam flowrate is less than for the case of pure steam leakage. Since the steam entering the wetwell air space results in the additional pressurization, this is considered as a conservative assumption.*

(3) *The containment sprays are manually actuated 30 minutes after the wetwell airspace pressure reaches to ~~103.0~~* 103.4 *kPaG.*

(4) Credit for both drywell and wetwell sprays was taken. ~~*Credit for wetwell spray only was taken. Considering that wetwell spray is more effective in mitigating consequences of steam bypass leakage, credit for drywell spray was not taken to produce conservative results.*~~

(5) *The efficiency of the sprays is dependent upon the local steam-to-air ratio. A conservative constant value of 0.7 was used in this analysis.*

(6) *Heat is transferred to exposed drywell/wetwell concrete walls (with steel liner) in the drywell and wetwell gas space regions. The Uchida convective heat transfer coefficients used are based on the local steam-to-air ratio.*

(7) *No energy is assumed to leave the containment except through the RHR heat exchangers.*

The following is an illustration of the methods employed in calculating steam condensing capability under typical post-LOCA conditions. The condensation capability is calculated using the following equation:

$$M_c = M_s \times N_s \times [(T_c - T_s)/H_{fg}] \times C_p$$

where

M_c = *steam condensation rate*

M_s = *spray flow rate*

N_s = *spray efficiency*

T_c = *containment temperature*

T_s = *spray temperature at the spray nozzles*

H_{fg} = *latent heat of vaporization of water*

C_p = *constant pressure specific heat of water*

The spray water temperature is calculated from:

$$T_s = T_p - KHX \times [(T_p - T_{sw}) / (M_s \times C_p)]$$

where

T_p = *suppression pool temperature*

KHX = *RHR heat exchanger effectiveness*

T_{sw} = *service water temperature*

Containment sprays have a significant effect on the allowable steam bypass capability. Use of sprays increases the maximum allowable bypass leakage by an order of magnitude and represents an effective backup means of condensing bypass steam. See Appendix 6E for additional bypass consideration.

6.2.1.1.5.6 Justification for Deviation From SRP Requirements

6.2.1.1.5.6.1 Actuation of Wetwell Sprays

It is recognized that provision of manual, and not automatic, spray actuation of wetwell sprays in the ABWR design is a deviation from the SRP requirement (Appendix A to SRP Section 6.2.1.1.C) of automatic actuation of sprays. The SRP states that the wetwell spray should be automatically actuated 10 minutes following a LOCA signal and an indication of pressurization of the wetwell to quench steam bypassing the suppression pool. However, in determining

maximum allowable steam bypass leakage area for ABWR design, analyses assume and take credit for operator actuation of wetwell sprays 30 minutes (instead of 10 minutes) following a LOCA signal and after the wetwell gas space pressure reaches to ~~103.0~~ 103.4 *kPaG, though ABWR EPGs permit actuation of wetwell sprays when wetwell gas space pressure reaches to 71.6 kPaG.*

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

The environmental conditions created by any high-energy line break (HELB) are analyzed according to Regulatory Guide 1.89. The first step in such analysis is to calculate the mass and energy release rate from the high-energy line break (HELB).

Figure 6.2-22 shows the break flow rate and specific enthalpy for the feedwater line break flow coming from the feedwater system side. Figure 6.2-23 shows the same information for the feedwater line break flow coming from the RPV. Figures 6.2-24 and 6.2-25 show the same information for the main steamline break flow with two-phase blowdown starting when the collapsed water level reaches the main steamline nozzle and when t = 2.0 seconds~~1.0 second~~.

6.2.2 Containment Heat Removal System

6.2.2.3 Design Evaluation of the Containment Cooling System

6.2.2.3.1 System Operation and Sequence of Events

In the event of the postulated LOCA, the short-term energy release from the reactor primary system will be dumped to the suppression pool. Subsequent to the accident, fission product decay heat will result in a continuing energy input to the pool. The RHR SPC mode will remove this energy which is released into the primary containment system, thus resulting in acceptable suppression pool temperatures and containment pressures.

In order to evaluate the adequacy of the RHR System, the following is assumed:

(1) *With the reactor initially operating at 102% of rated power, a LOCA occurs.*

(2) *A single failure of a RHR heat exchanger is the most limiting single failure.*

(3) *The ECCS flows assumed available are 2 HPCF, 1 RCIC, and 2 LPFL (RHR).*

(4) Containment cooling is initiated after 30 minutes. This is a conservative assumption given that the RHR system design provides pool cooling during the LPFL mode of RHR which, for a large pipe break, can occur in 3 to 5 minutes.~~Containment cooling is initiated after 10 minutes (see Response to Question 430.26).~~

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Table 6.2- 1 Containment Parameters

	Design Parameter	Design Value	Calculated Value
1.	Drywell pressure	309.9 kPaG	~~268.7~~ **279.6** kPaG
2.	Drywell temperature	171.1°C	~~177.0~~ **177.3**°C [1]
3.	Wetwell pressure	309.9 kPaG	~~179.5~~ **205.6** kPaG
4.	Wetwell temperature		
	• Gas Space	~~103.9~~ **124**°C	~~98.9~~ **94.5**°C
	• Suppression pool	97.2°C	~~96.9~~ **97.1**°C
5.	Drywell-to-wetwell differential pressure	+172.6 kPaD – 13.7 kPaD	~~+109.8~~ **+172.4** kPaD ~~–10.7~~ - **3.86** kPaD

[1] Design value is exceeded at 1.2 seconds into the event and then temperature decreases as shown in Figure 6.2-13.

Table 6.2- 2 Containment Design Parameters

	Drywell	Wetwell
A. Drywell and Wetwell[1]		
1. Internal Design Pressure (kPaG)	309.9	**309.9** ~~309.6~~
2. Negative Design Pressure (kPaG)	−13.7	−13.7
3. Design Temperature (°C)	171.1	**124** ~~103.9~~
4. Net Free Volume (m^3)	7350	5960
5. Maximum allowable leak rate[2] (%/day)	0.5	0.5
6. Minimum Suppression Pool Water Volume (m^3)	—	**3455** ~~3580~~
7. Suppression pool depth (m)		
Low Level	—	**6.9** ~~7~~
High Level	—	7.1
B. Vent System		
1. Number of Vents		30
2. Nominal Vent Diameter (m)		0.7
3. Total Vent Area (m^2)		11.6
4. Vent Centerline Submergence Low Level, (m)		
Top Row		**3.4** ~~3.5~~
Middle Row		**4.8** ~~4.9~~
Bottom Row		**6.1** ~~6.2~~
5. Vent Loss Coefficient (Varies with number of vents open)		2.5– ~~3.5~~ **5.0**

1 Items A.1, A.2, A.3 and A.5 apply to related structures including lower drywell access tunnels, drywell equipment hatches, drywell personnel locks and drywell head.

2 Corresponds to calculated peak containment pressure related to the design basis accident conditions. Excludes MSIV leakage.

Table 6.2- 2a Engineered Safety Systems Information for Containment Response Analyses

	Full Capacity	Containment Analysis Value
A. Containment Spray		
1. Number of RHR Pumps	**2**(1)	1(1)
2. Number of Lines	**2**(1)	1(1)
3. Number of Heat Exchangers	**2**(2)	1 (2)
4. Drywell Flow Rate (kg/h)	0.84×10^6	0.84×10^6
5. Wetwell Flow Rate (kg/h)	1.14×10^5	1.14×10^5
B. Containment Cooling System		
1. Number of RHR Pumps	3	2
2. Pump Capacity (m^3/h/pump)	954	954
3. RHR Heat Exchangers		
a. Type–U-tube,		
b. Number	3	2
c. Heat Transfer Area (m^2/unit)	(3)	(3)
d. Overall Heat Transfer Coefficent (Btu/h – m^2-°C/unit)	(3)	(3)
e. Reactor Cooling Water Flowrate (~~kg~~m^3/h)	~~1.2 x 10^6~~ **1200**	~~1.2x 10^6~~ **1200**
f. Maximum Cooling Water Inlet Temperature (°C)	~~37.8~~**35**	~~37.8~~**35**

1 *Two redundant loops available with one pump each.*

2 ~~*One header each for drywell and wetwell.*~~ The heat exchanger is shared with both the wetwell and drywell sprays.

3 *The RHR heat exchanger characteristic has been defined by an overall K coeficient based on a temperature difference and the heat rate. The defining equation is:*

$$Q = (K)\,(\Delta T)$$

$$\cancel{Q, \frac{\text{kcal}}{\text{s}} = \left(K, \frac{\text{kcal}}{(\text{s}°\text{C})}\right)\left(\acute{y}T, °\text{C}\right)}$$

~~*The K value is 370.5 kJ/s°C.*~~ The K value is 4.27×10^5 W/°C

The applicable temperature difference occurs form the RHR heat exchanger's reactor side inlet to the ultimate heat sink temperature. Thus, K is a characteristic of the combined RHR and reactor cooling water system's heat exchangers.

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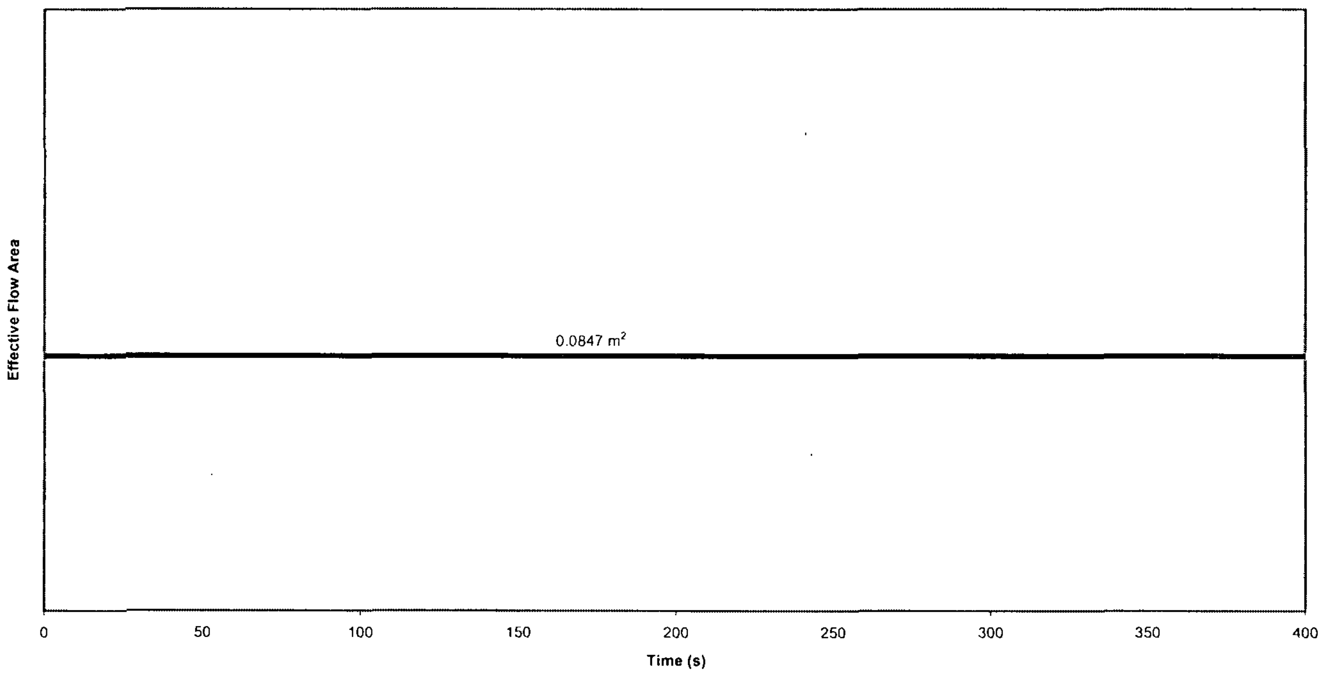

Figure 6.2-2 Feedwater Line Break—RPV Side Break Area

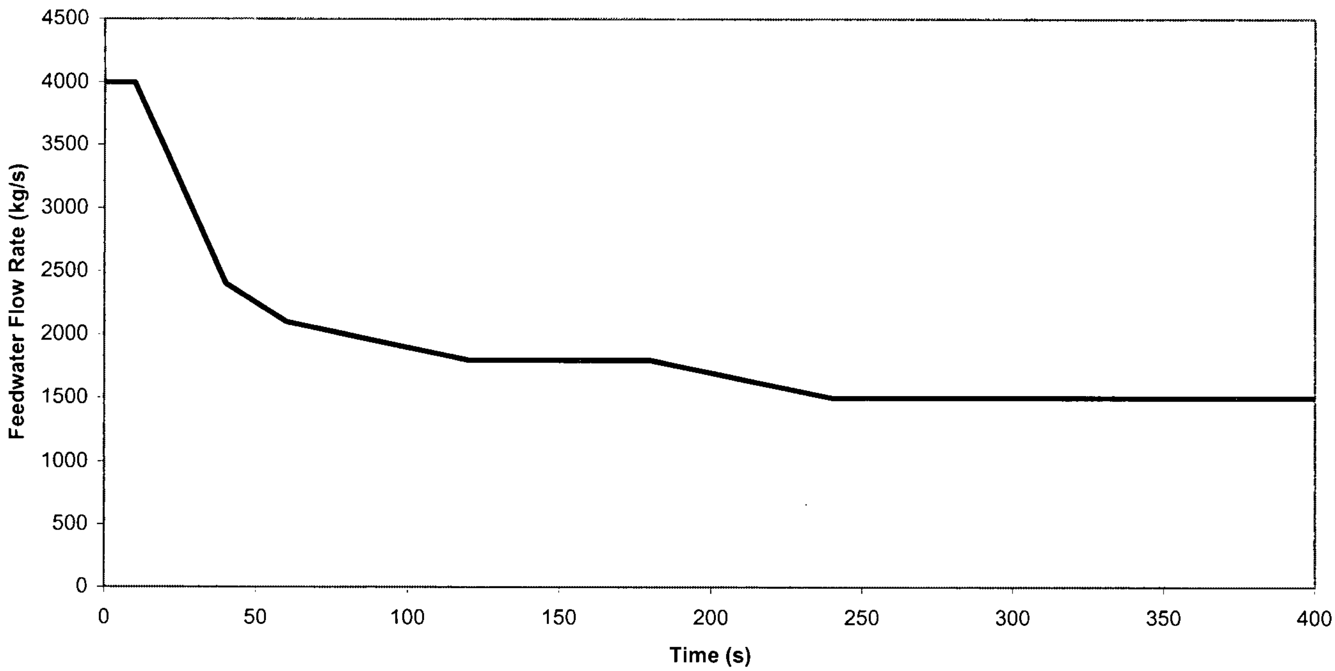

Figure 6.2- 3 Feedwater Line Break Flow—Feedwater System Side of Break

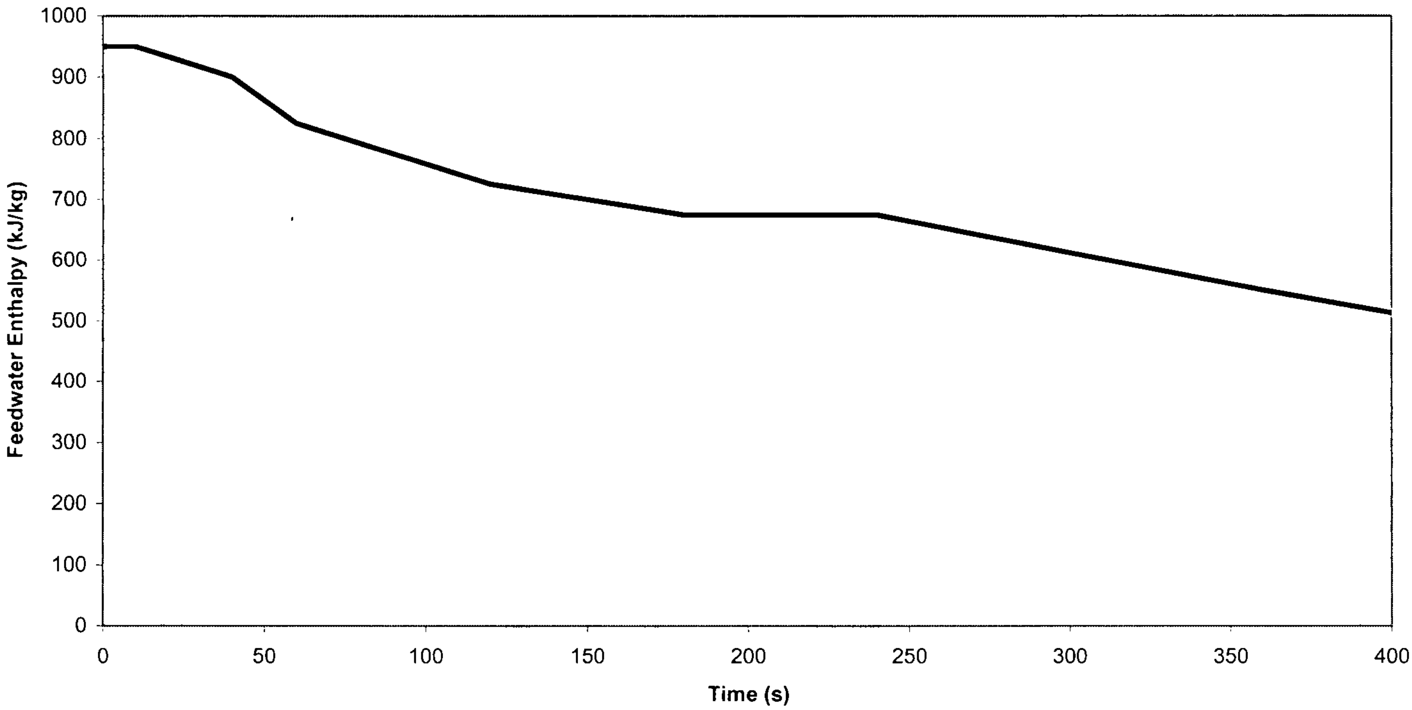

Figure 6.2- 4 Feedwater Line Break Flow Enthalpy—Feedwater System Side of Break

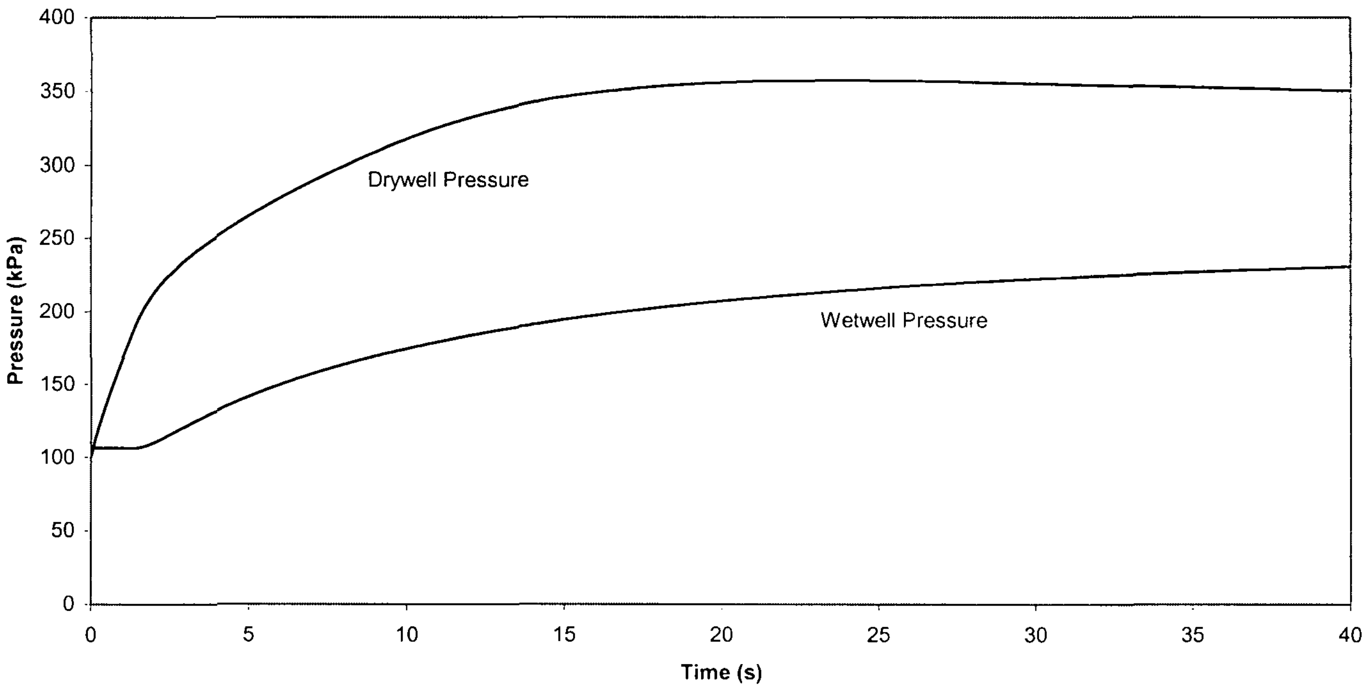

Figure 6.2-6 Pressure Response of the Primary Containment for Feedwater Line Break

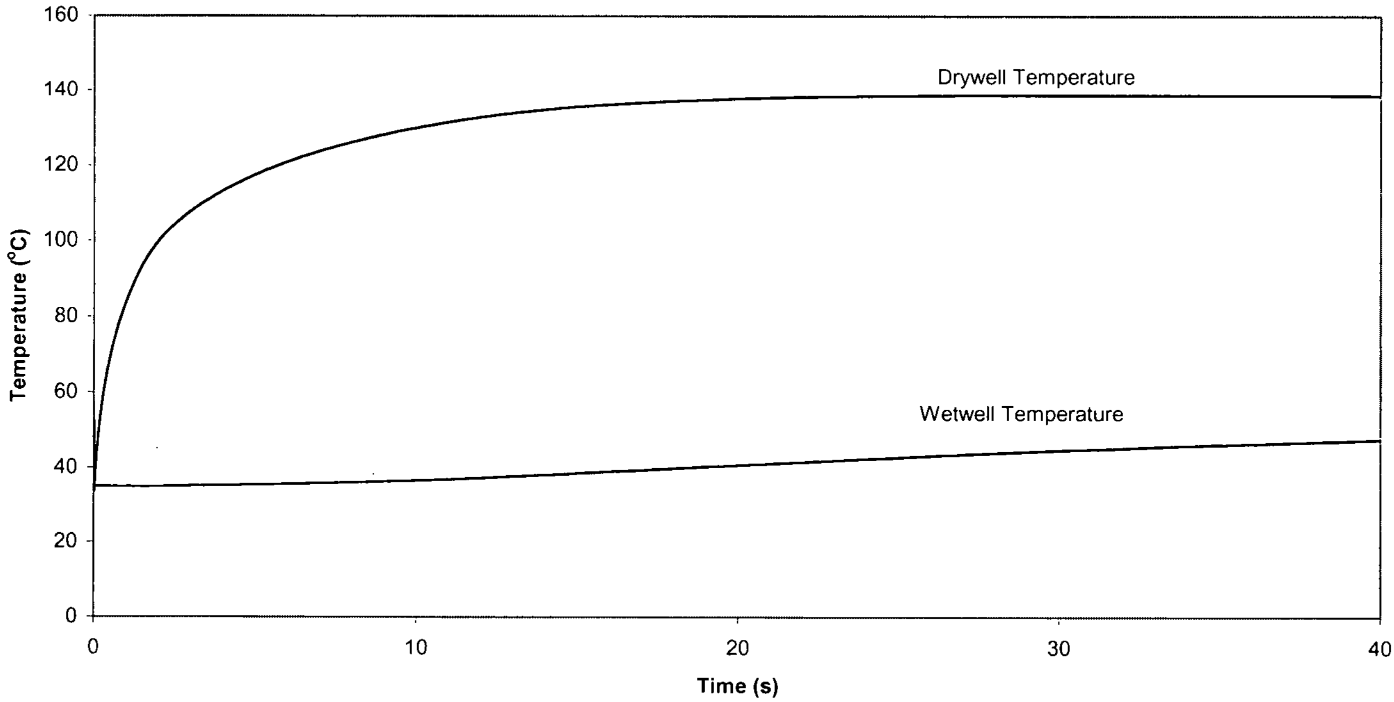

Figure 6.2- 7 Temperature Response of the Primary Containment for Feedwater Line Break

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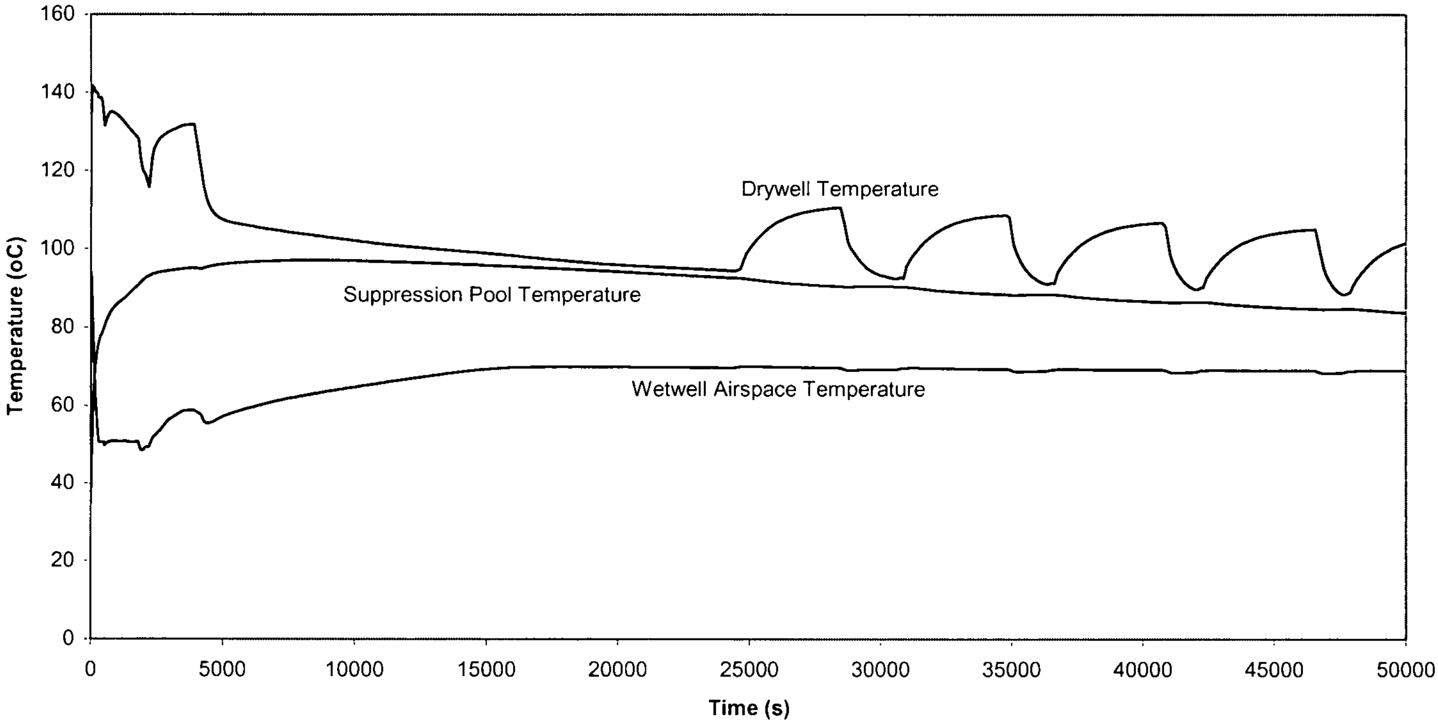

Figure 6.2- 8 Temperature Time History After a Feedwater Line Break

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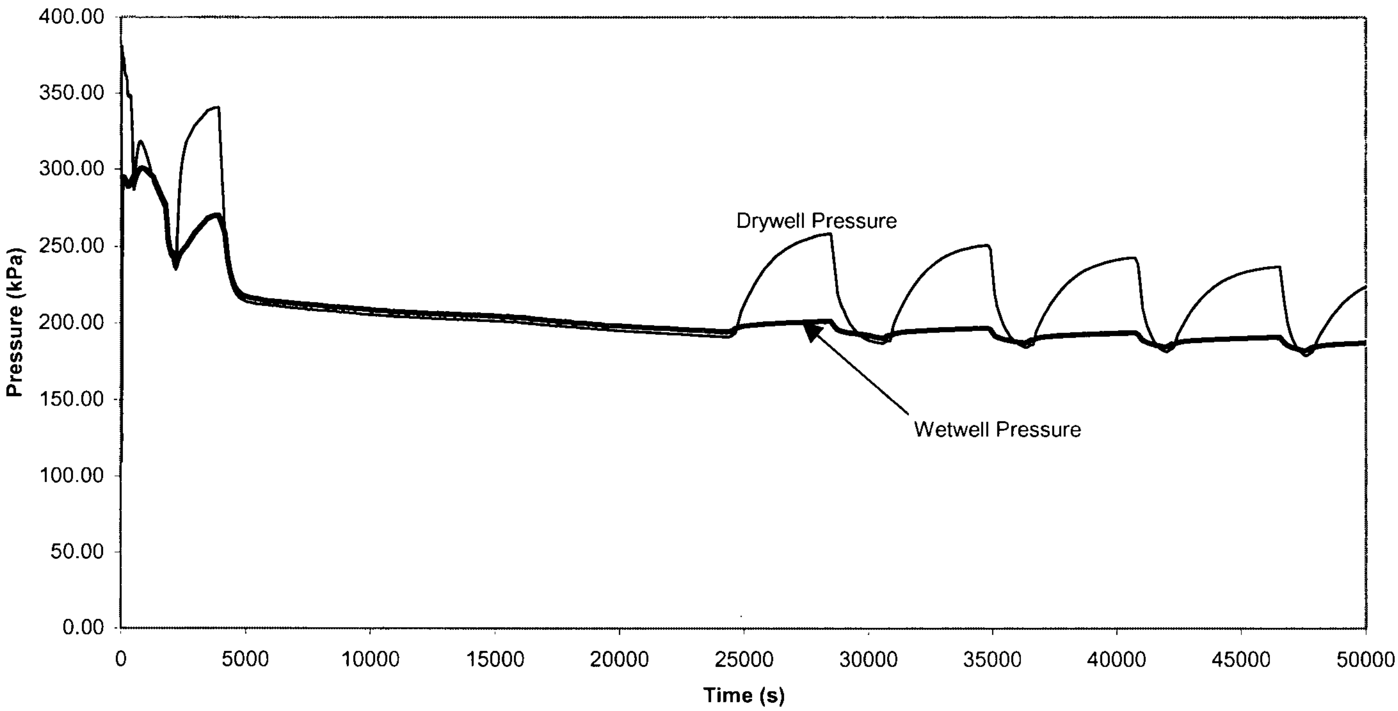

Figure 6.2- 8a Pressure Time History After a Feedwater Line Break

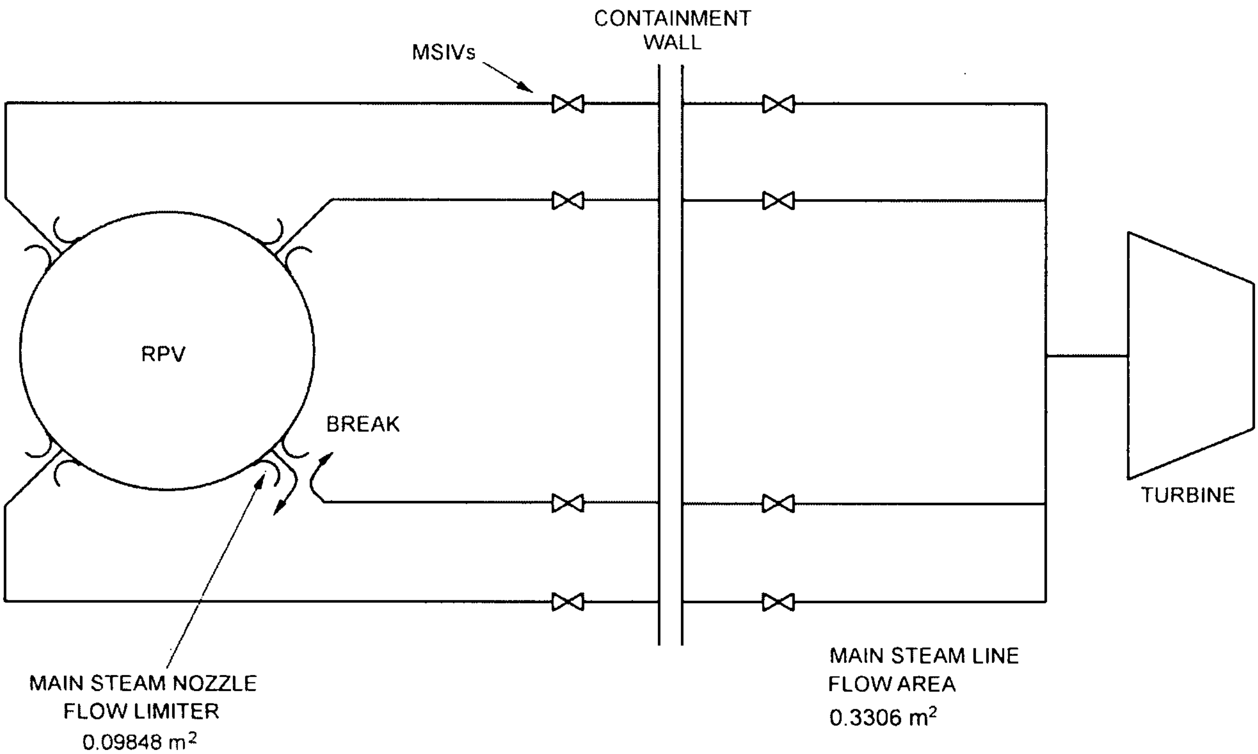

Figure 6.2- 9 ABWR Main Steamlines with a Break

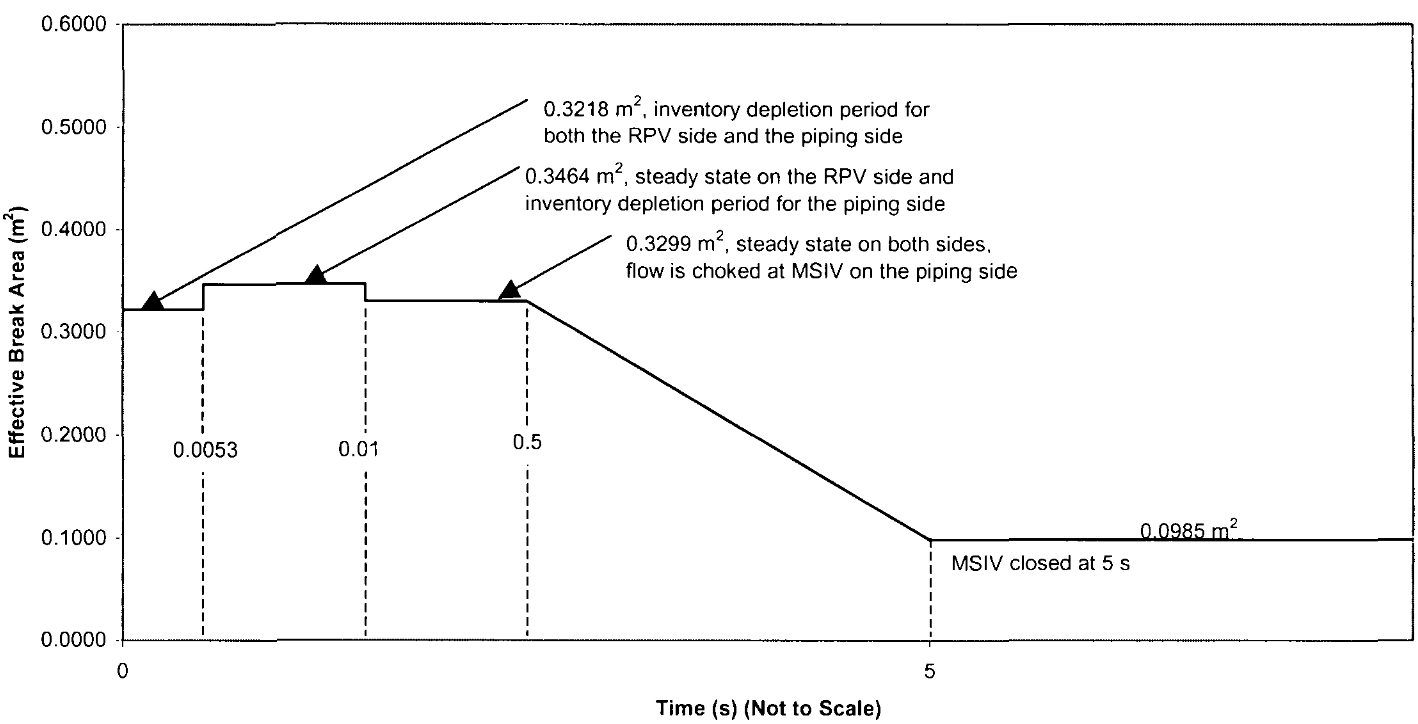

Figure 6.2- 10 MSLB Area as a Function of Time

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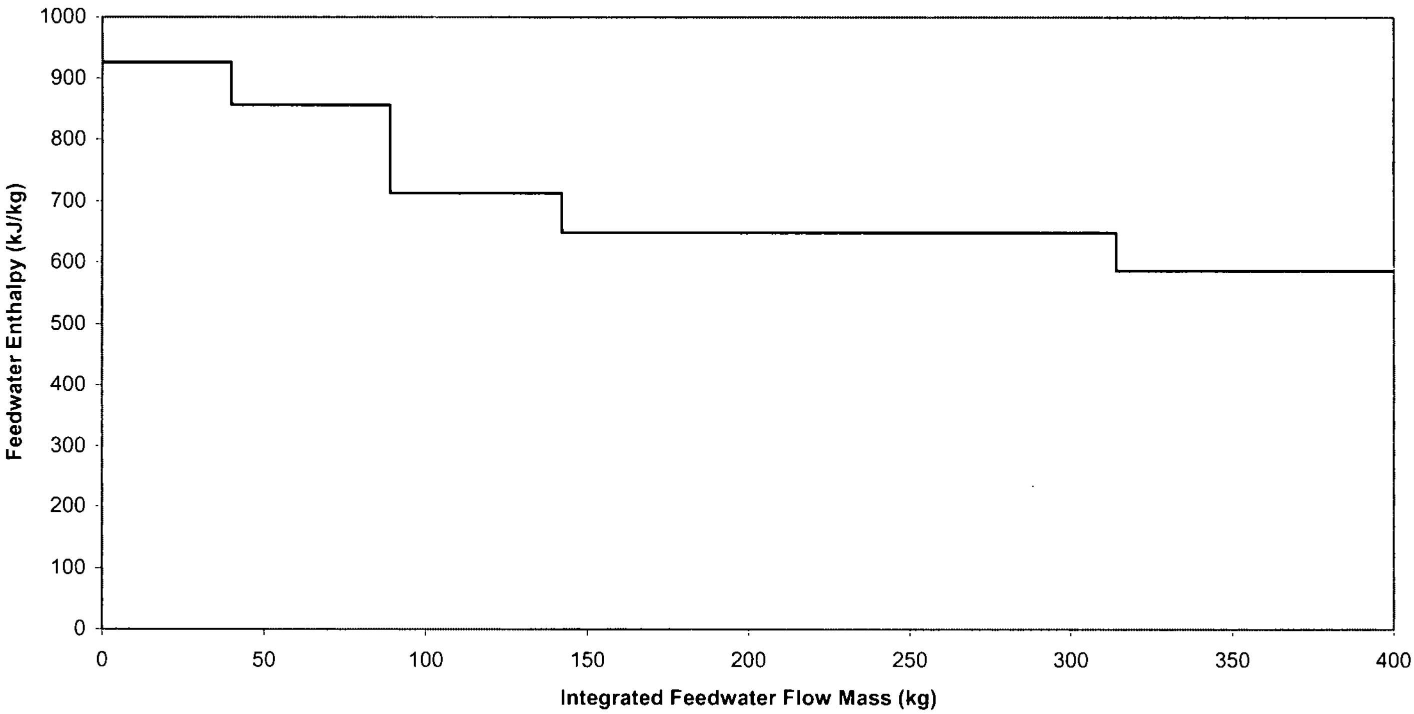

Figure 6.2-11 Feedwater Specific Enthalpy as a Function of Integrated Feedwater Flow Mass

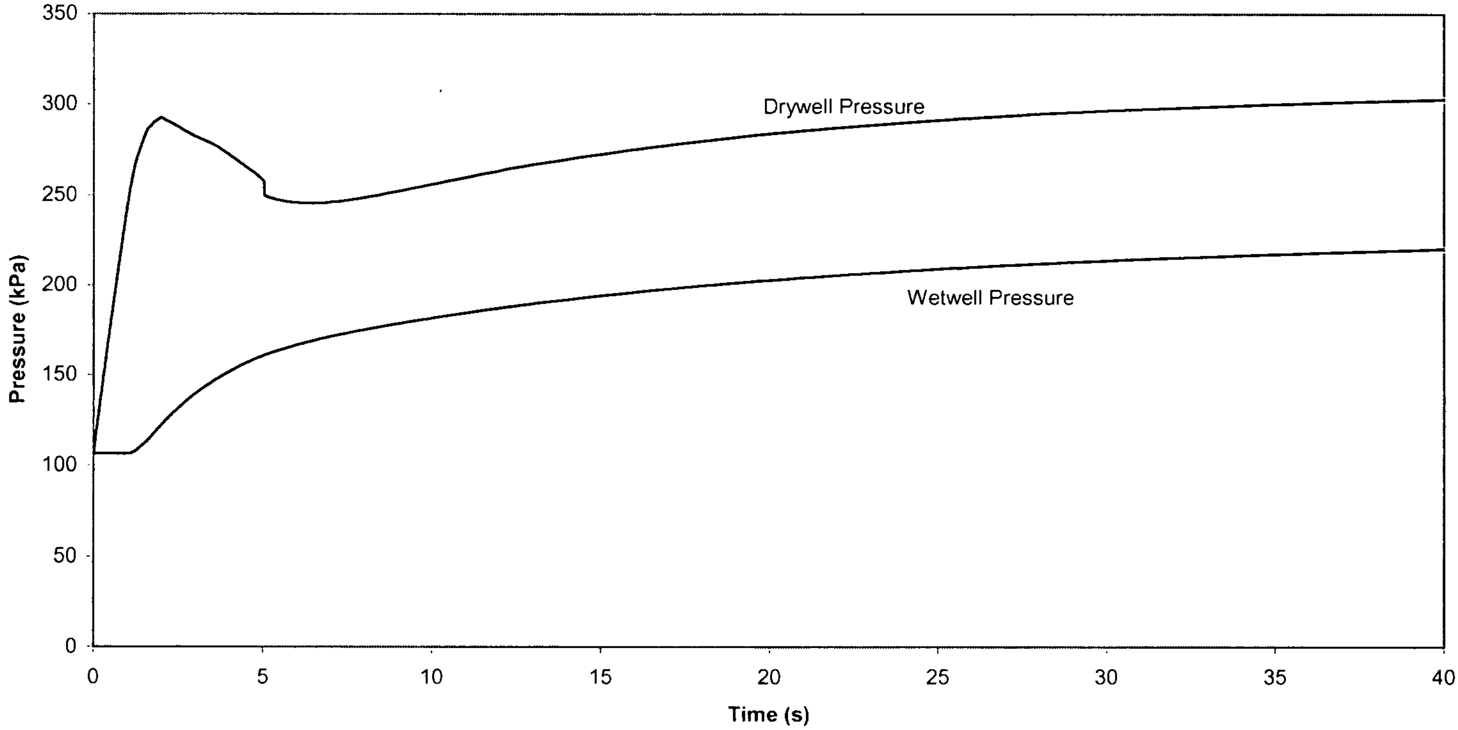

Figure 6.2- 12 Pressure Time History for MSLB with Two-Phase Blowdown Starting ~~When the RPV Collapsed Level Reaches the Main Steam Nozzle~~ at 2 Seconds

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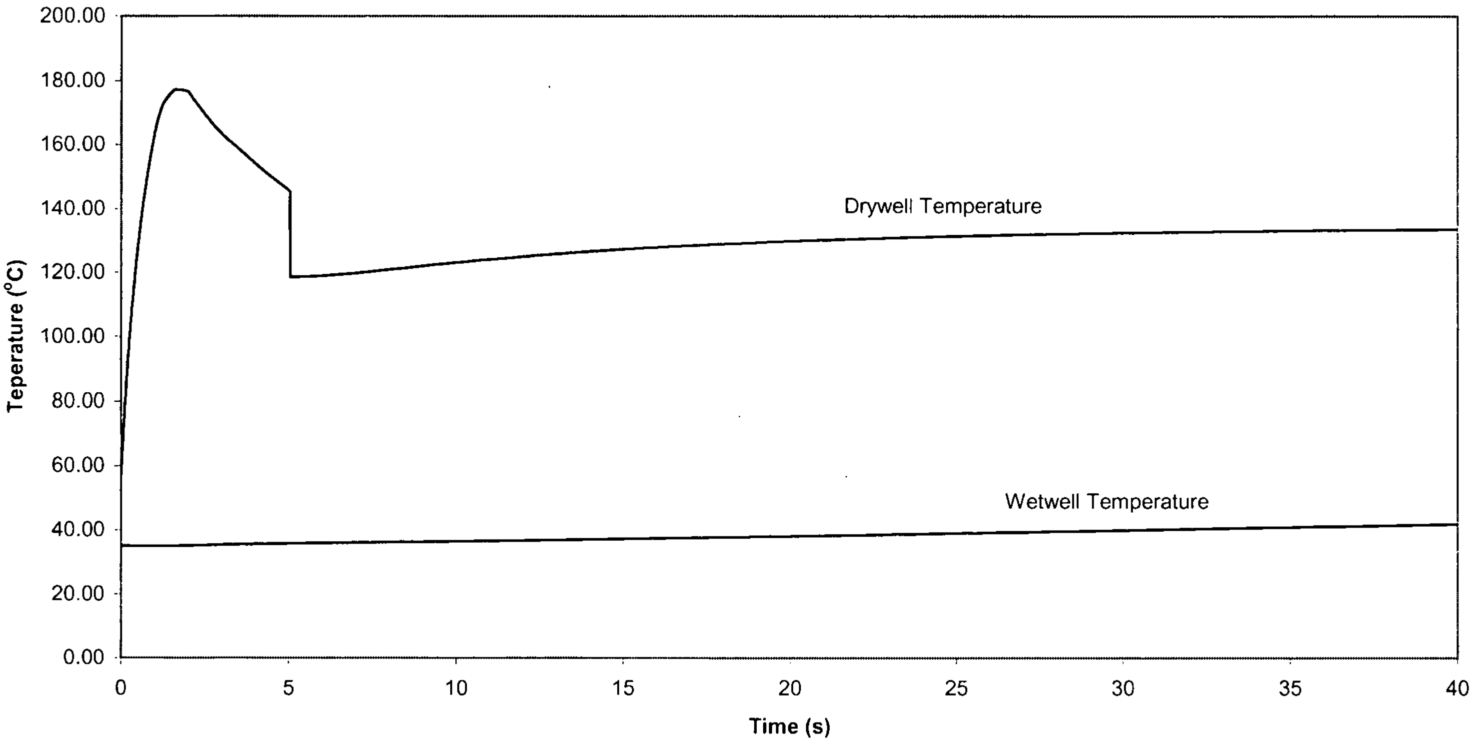

Figure 6.2- 13 Temperature Time History for MSLB with Two-Phase Blowdown Starting ~~When the RPV Collapsed Level Reaches the Main Steam Nozzle~~ at 2 Seconds

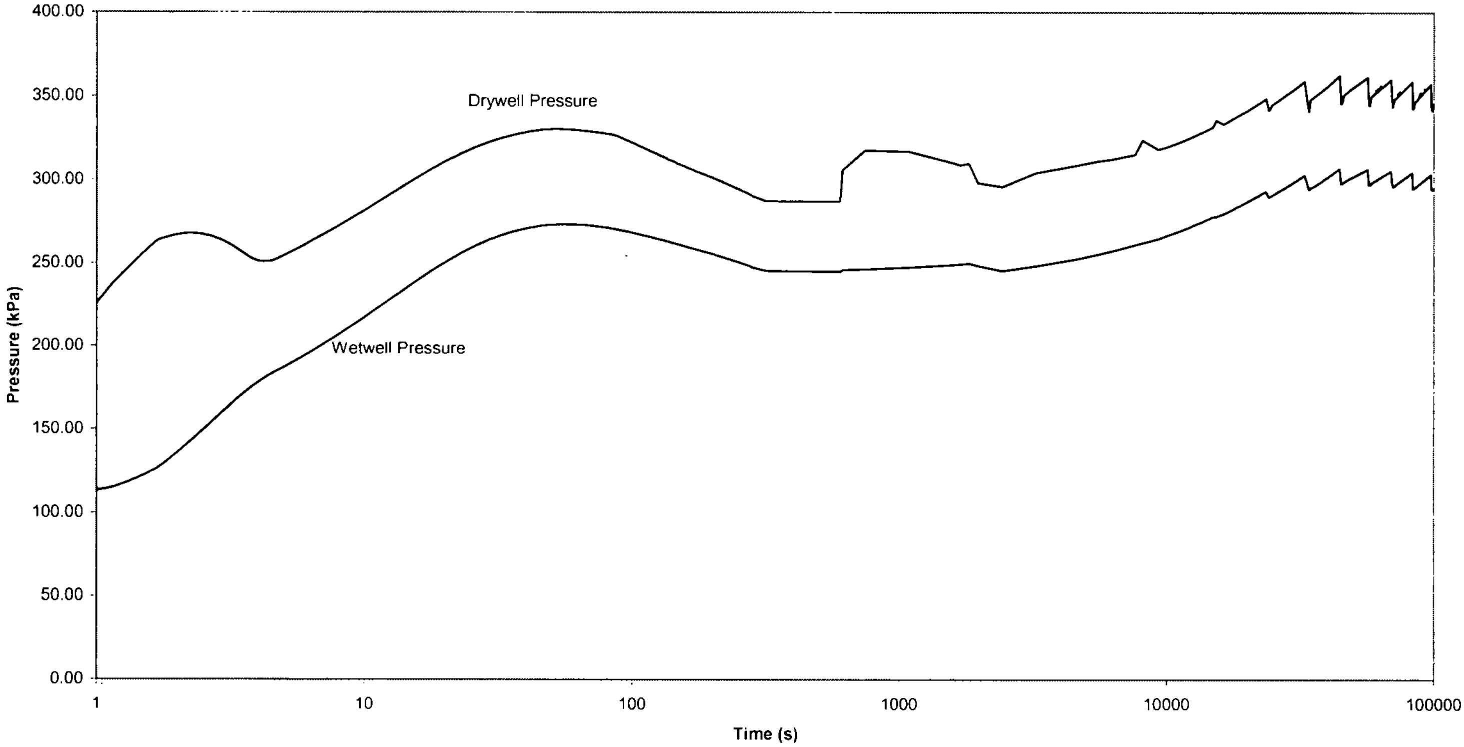

Figure 6.2- 14 Pressure Time History for Long-term MSLB ~~with Two-Phase Blowdown Starting at One Second~~

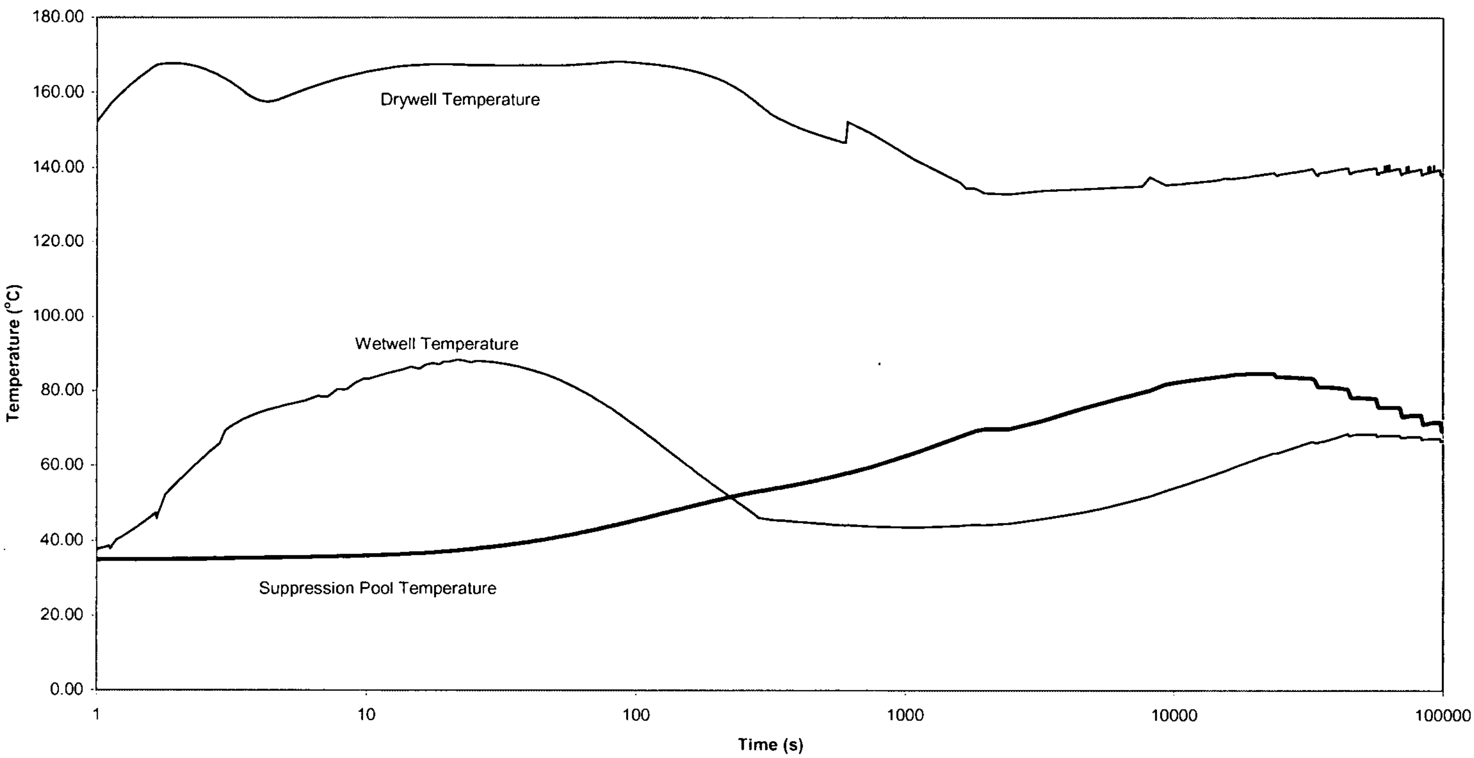

Figure 6.2- 15 Temperature Time History for Long-term MSLB ~~with Two-Phase Blowdown Starting at One Second~~

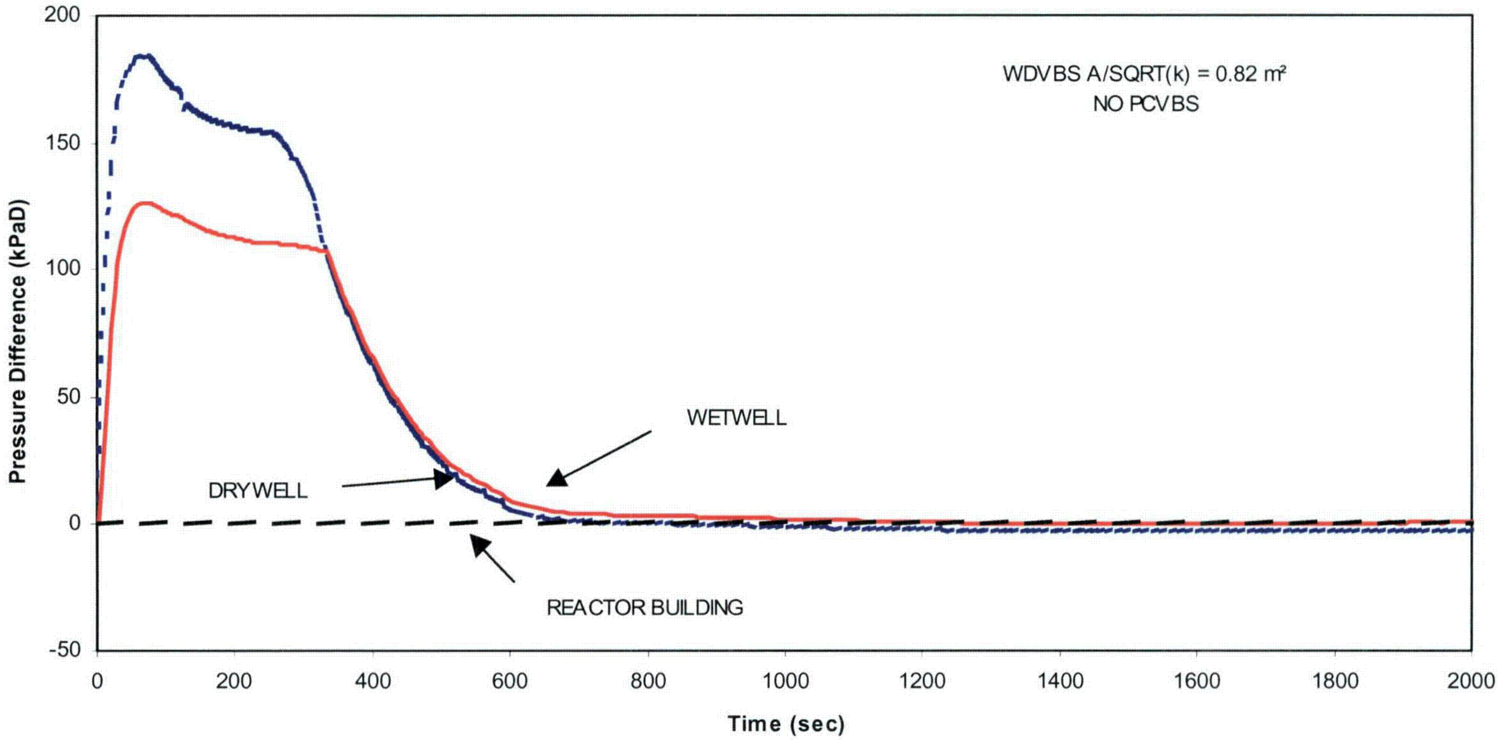

Figure 6.2- 17 Differential Pressures in Wetwell and Drywell Relative to Reactor Building for Vacuum Breaker Size of ~~.771~~ .82 m^2

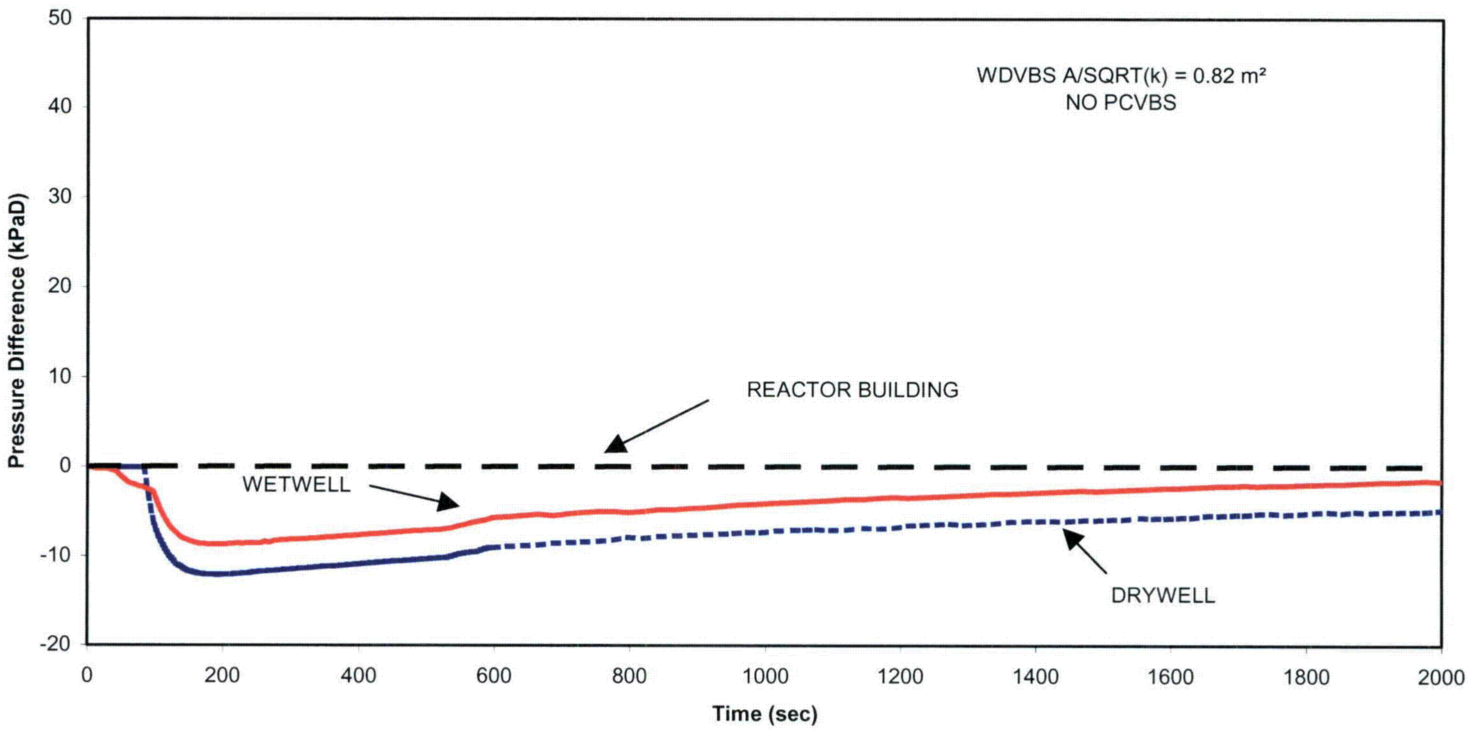

Figure 6.2- 18 Differential Pressures in Wetwell and Drywell Relative to Reactor Building with Wetwell Spray for Vacuum Breaker Size of ~~.771~~ .82 m²

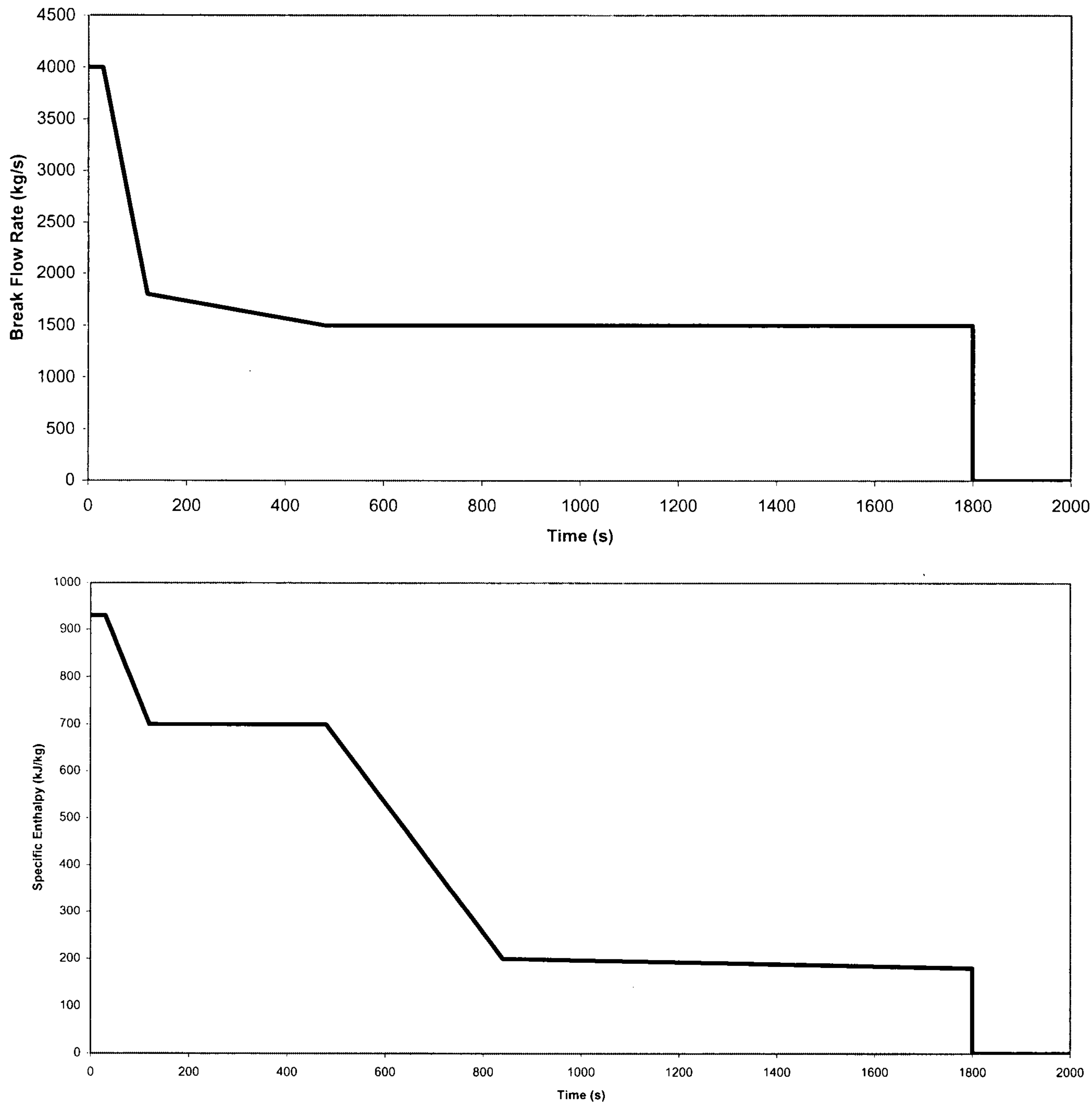

Figure 6.2- 22 Break Flow Rate and Specific Enthalpy for the Feedwater Line Break Flow Coming from the Feedwater System Side

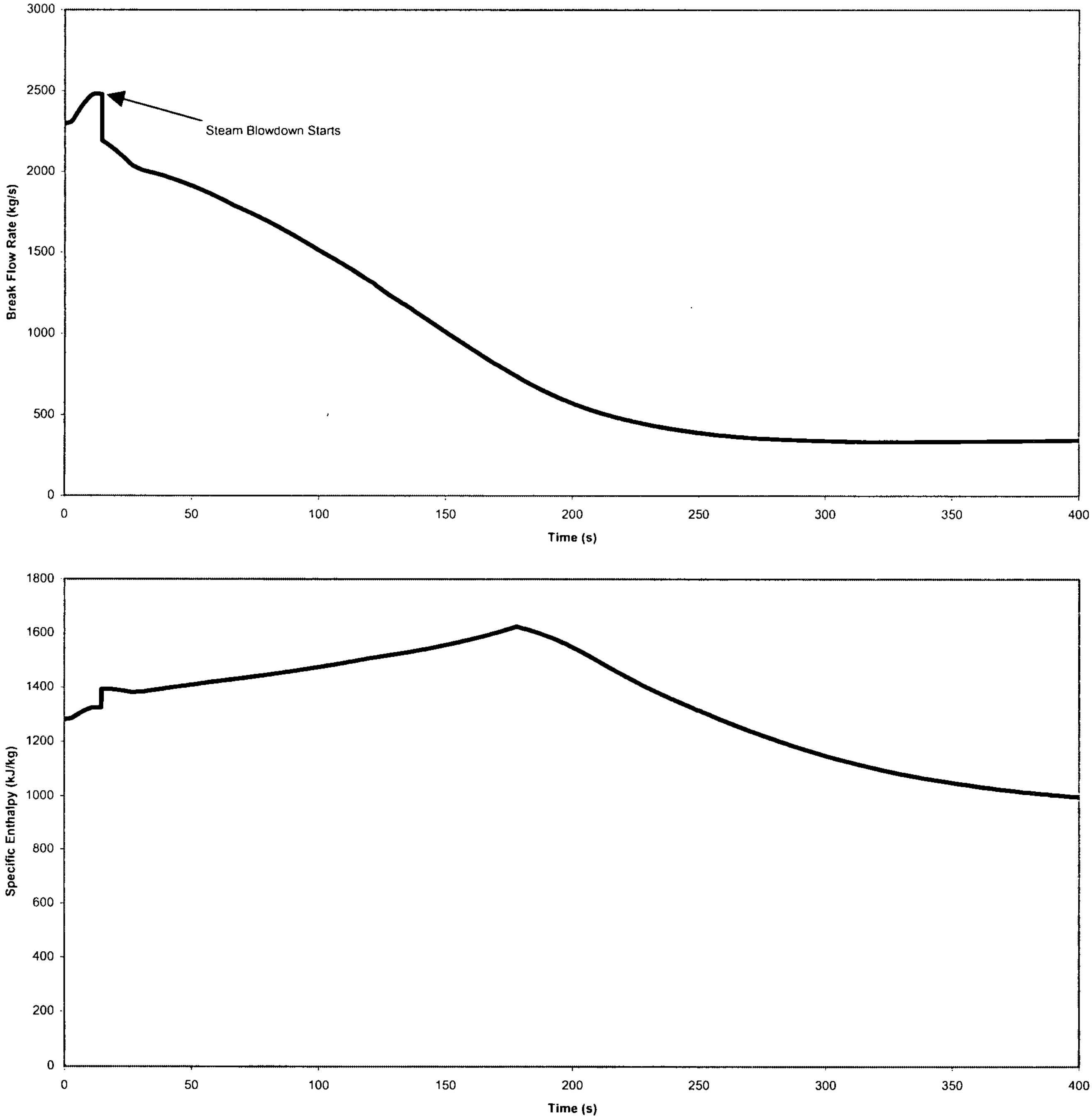

Figure 6.2-23 Break Flow Rate and Specific Enthalpy for the Feedwater Line Break Flow Coming from the RPV Side

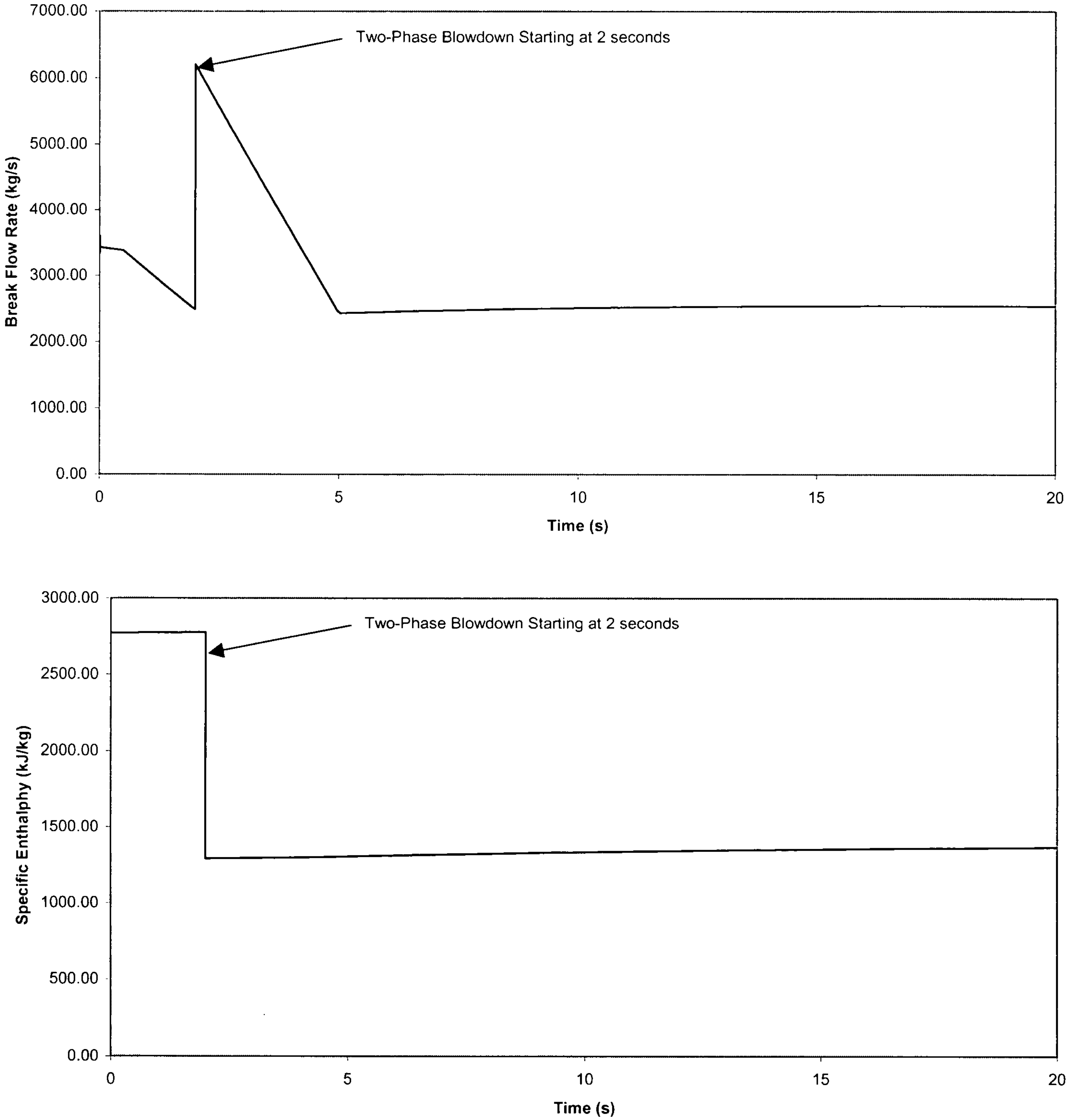

Figure 6.2- 24 Break Flow Rate and Specific Enthalpy for the Main Steamline Break with Two-Phase Blowdown Starting When the Collapsed Water Level Reaches the Steam Nozzle

Rev. 0

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~~Figure 6.2-25 Break Flow Rate and Specific Enthalpy for the Main Steamline Break with Two-Phase Blowdown Starting at One Second~~

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.1.6.1	-------------------NOTE------------------- Not required to be met for vacuum breakers when performing their intended function. --- Verify each vacuum breaker is closed.	14 days <u>AND</u> Within 2 hours after any discharge of steam to the wetwell from the safety/relief valves (S/RVs) or any operation that causes the wetwell-drywell differential pressure to be reduced by ≥ 0.69 kPaD.
SR 3.6.1.6.2	Perform a functional test of each vacuum breaker.	18 months
SR 3.6.1.6.3 3.45	Verify each required vacuum breaker fully opens at ≤ ~~3.43~~ kPaD.	18 months
SR 3.6.1.6.4	Perform CHANNEL CALIBRATION of vacuum breaker position indication channel.	18 months

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.6.2.4.1	Verify each RHR containment spray subsystem manual, power operated, and automatic valve in the flow path that is not locked, sealed, or otherwise secured in position is in the correct position or can be aligned to the correct position.	31 days
SR 3.6.2.4.2	Verify each associated (i.e., in subsystems B & C) RHR pump develops a flow rate ≥ 114 m^3/h ~~and < 160 m^3/h~~ through the wetwell spray sparger.	92 days

BASES

BACKGROUND (continued)

conformance with 10 CFR 50, Appendix J (Ref. 3), as modified by approved exemptions.

APPLICABLE SAFETY ANALYSES

The safety design basis for the primary containment is that it must withstand the pressures and temperatures of the limiting DBA without exceeding the design leakage rate.

The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE such that release of fission products to the environment is controlled by the rate of primary containment leakage.

Analytical methods and assumptions involving the primary containment are presented in References 1 and 2. The safety analyses assume a nonmechanistic fission product release following a DBA, which forms the basis for determination of offsite doses. The fission product release is, in turn, based on an assumed leakage rate from the primary containment. OPERABILITY of the primary containment ensures that the leakage rate assumed in the safety analyses is not exceeded.

calculated
279.6 kPaG

The maximum allowable leakage rate for the primary containment (L_a) is 0.5% by weight of the containment air per 24 hours at the ~~maximum~~ peak containment pressure (P_a) of ~~0.269 MPaG~~ or [ ]% by weight of the containment air per 24 hours at the reduced pressure of Pt of [ ] MPaG (Ref. 1).

Primary containment satisfies Criterion 3 of the NRC Policy Statement.

LCO

Primary containment OPERABILITY is maintained by limiting leakage to within the acceptance criteria of 10 CFR 50, Appendix J (Ref. 3). Compliance with this LCO will ensure a primary containment configuration, including equipment hatches, that is structurally sound and that will limit leakage to those leakage rates assumed in the safety analyses. Individual leakage rates specified for the primary containment air lock are addressed in LCO 3.6.1.2.

(continued)

BASES

BACKGROUND (continued)

The primary containment air locks form part of the primary containment pressure boundary. As such, air lock integrity and leak tightness are essential for maintaining primary containment leakage rate to within limits in the event of a DBA. Not maintaining air lock integrity or leak tightness may result in a leakage rate in excess of that assumed in the unit safety analysis. SR 3.6.1.1.1 leakage rate requirements conform with 10 CFR 50, Appendix J (Ref. 2), as modified by approved exemptions.

APPLICABLE SAFETY ANALYSES

The DBA that postulates the maximum release of radioactive material within primary containment is a LOCA. In the analysis of this accident, it is assumed that primary containment is OPERABLE, such that release of fission products to the environment is controlled by the rate of primary containment leakage. The primary containment is designed with a maximum allowable leakage rate (L_a) of 0.5%(excluding MSIV leakage) by weight of the containment air per 24 hours at the calculated maximum peak containment pressure (P_a) of ~~0.269 MPaG~~ 279.6 kPaG (Ref. 3). This allowable leakage rate forms the basis for the acceptance criteria imposed on the SRs associated with the air lock.

Primary containment air lock OPERABILITY is also required to minimize the amount of fission product gases that may escape primary containment through the air lock and contaminate and pressurize the secondary containment.

The primary containment air lock satisfies Criterion 3 of the NRC Policy Statement.

LCO

As part of the primary containment, the air lock's safety function is related to control of containment leakage rates following a DBA. Thus, the air lock's structural integrity and leak tightness are essential to the successful mitigation of such an event.

The primary containment air locks are required to be OPERABLE. For each air lock to be considered OPERABLE, the air lock interlock mechanism must be OPERABLE, the air lock must be in compliance with the Type B air lock leakage test, and both air lock doors must be OPERABLE. The interlock

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.4 Drywell Pressure

BASES

BACKGROUND

The drywell pressure is limited during normal operations to preserve the initial conditions assumed in the accident analysis for a Design Basis Accident (DBA) or loss of coolant accident (LOCA).

APPLICABLE SAFETY ANALYSES

Primary containment performance is evaluated for the entire spectrum of break sizes for postulated LOCAs (Ref. 1). Among the inputs to the DBA is the initial primary containment internal pressure (Ref. 1). Analyses assume an initial drywell pressure of 5.20×10^{-3} MPaG. This limitation ensures that the safety analysis remains valid by maintaining the expected initial conditions and ensures that the peak LOCA drywell internal pressure does not exceed the maximum allowable of 0.310 MPaG.

long term

The maximum calculated drywell pressure occurs during the ~~reactor blowdown~~ phase of the DBA, which is determined to be a feedwater line break. The calculated peak drywell pressure for this limiting event is ~~0.269 MPaG~~ (Ref. 1).

279.6 kPaG

Drywell pressure satisfies Criterion 2 of the NRC Policy Statement.

LCO

In the event of a DBA, with an initial drywell pressure $\leq 5.20\times10^{-3}$ MPaG, the resultant peak drywell accident pressure will be maintained below the drywell design pressure.

APPLICABILITY

In MODES 1, 2, and 3, a DBA could cause a release of radioactive material to primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations of these MODES. Therefore, maintaining drywell pressure within limits is not required in MODE 4 or 5.

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.5 Drywell Air Temperature

BASES

BACKGROUND

The drywell contains the reactor vessel and piping, which add heat to the airspace. Drywell coolers remove heat and maintain a suitable environment. The average airspace temperature affects the calculated response to postulated Design Basis Accidents (DBAs). The limitation on the drywell average air temperature was developed as reasonable, based on operating experience. The limitation on drywell air temperature is used in the Reference 1 safety analyses.

APPLICABLE SAFETY ANALYSES

Primary containment performance is evaluated for a spectrum of break sizes for postulated loss of coolant accidents (LOCAs) (Ref. 1). Among the inputs to the design basis analysis is the initial drywell average air temperature (Ref. 1). Analyses assume an initial average drywell air temperature of 57°C. This limitation ensures that the safety analysis remains valid by maintaining the expected initial conditions and ~~ensures that the peak LOCA drywell temperature does not exceed the maximum allowable temperature of 171°C (Ref. 2). Exceeding this design temperature may result in the degradation of the primary containment structure under accident loads~~. Equipment inside primary containment, required to mitigate the effects of a DBA, is designed to operate and be capable of operating under environmental conditions expected for the accident.

the primary containment structural materials remain below the design temperature

The most severe drywell temperature condition occurs as a result of a small Reactor Coolant System rupture above the reactor water level, which results in the blowdown of reactor steam to the drywell. The drywell temperature analysis considers main steam line breaks occurring inside the drywell and having various break areas. The maximum calculated drywell average temperature for the worst case break area is provided in Reference 2.

Drywell air temperature satisfies Criterion 2 of the NRC Policy Statement.

(continued)

B 3.6 CONTAINMENT SYSTEMS

B 3.6.1.6 Wetwell-to-Drywell Vacuum Breakers

BASES

BACKGROUND

The function of the wetwell-to-drywell vacuum breakers is to relieve vacuum in the drywell. There are eight internal vacuum breakers between the drywell and the wetwell, which allow gas and steam flow from the wetwell to the drywell when the drywell is at a lower pressure than the wetwell. Therefore, the wetwell-to-drywell vacuum breakers prevent an excessive negative differential pressure across the wetwell/drywell boundary. Each vacuum breaker is a self actuating valve, similar to a check valve, and requires no external power for actuation.

A negative pressure inside the drywell is caused by rapid depressurization of the drywell. Events that cause this rapid depressurization are cooling cycles, ~~inadvertent drywell spray actuation and~~ steam condensation from sprays or subcooled water spilling out of a break in reflood stage of a primary system rupture. Cooling cycles result in minor pressure transients in the drywell that occur slowly and are normally controlled by heating and ventilation equipment. Spray actuation or the spill of subcooled water out of a break results in more significant pressure transients and are important in sizing the internal vacuum breakers.

In the event of a primary system rupture, steam condensation within the drywell results in the most severe pressure transient. Following a primary system rupture, gas in the drywell is purged into the wetwell free airspace, leaving the drywell full of steam. Subsequent condensation of the steam can be caused in ~~two~~ three possible ways, namely, Emergency Core Cooling System flow from a ruptured pipe, feedwater flow from a ruptured pipe, or containment spray actuation following a loss of coolant accident (LOCA). These ~~two~~ three cases determine the maximum depressurization rate of the drywell.

In addition, the waterleg in the vertical vents of the vent system is controlled by the drywell-to-wetwell differential pressure. If the drywell pressure is less than the wetwell pressure, there will be an increase in the vent waterleg. This will result in an increase in the water clearing

(continued)

BASES

APPLICABILITY

(feedwater line break or main steam line break)

and due to actuation of drywell sprays

In MODES 1, 2, and 3, a DBA could result in excessive negative differential pressure across the drywell wall, caused by the rapid depressurization of the drywell. The event that results in the limiting rapid depressurization of the drywell is the primary system rupture that purges the drywell of gas and fills the drywell free airspace with steam. Subsequent condensation of the steam (due to cold water spilling out of the ruptured pipe) would result in depressurization of the drywell. The limiting pressure and temperature of the primary system prior to a DBA occur in MODES 1, 2, and 3. Also, inadvertent actuation of the drywell spray could result in rapid depressurization of the drywell. The vacuum breakers, therefore, are required to be OPERABLE in MODES 1, 2, and 3.

In MODES 4 and 5, the probability and consequences of these events are reduced by the pressure and temperature limitations in these MODES; therefore, maintaining wetwell-to-drywell vacuum breakers OPERABLE is not required in MODE 4 or 5.

ACTIONS

A.1

With one of the eight vacuum breakers inoperable for opening (e.g., the vacuum breaker is not open and may be stuck closed or not within its opening differential pressure limit, so that it would not function as designed during an event that depressurized the drywell), the remaining seven OPERABLE vacuum breakers are capable of providing the vacuum relief function. However, overall system reliability is reduced because a single failure in one of the remaining vacuum breakers could result in an excessive wetwell-to-drywell differential pressure during a DBA.

Therefore, with one of the eight required vacuum breakers inoperable, 72 hours is allowed to restore the inoperable vacuum breaker to OPERABLE status so that plant conditions are consistent with those assumed for the design basis analysis. The 72 hour Completion Time is considered acceptable due to the low probability of an event in which the remaining vacuum breaker capability would not be adequate.

(continued)

BASES

ACTIONS (continued)

B.1

One or more open vacuum breakers allow communication between the drywell and wetwell airspace, and, as a result, there is the potential for wetwell overpressurization due to this bypass leakage if a LOCA were to occur. Since the vacuum breakers are normally biased closed by gravitational force, Condition B mostly like be entered due to inaccurate position indication.

If vacuum breaker position indication is not reliable, an alternate method of verifying that the vacuum breakers are closed is by checking the position indication instrumentation. Another alternate method of verifying that the vacuum breakers are closed is by increasing the drywell

3.45

pressure by $\boxed{3.43}\times10^{-3}$ MPa above the wetwell pressure and verifying that the pressure differential does not fall below 2.06×10^{-3} MPaD for 15 minutes without makeup. The required 12 hour Completion Time is considered adequate to perform this test. If the stated criteria of this test is not met, Condition C must be entered.

C.1 and C.2

If the inoperable wetwell-to-drywell vacuum breaker cannot be closed or restored to OPERABLE status within the required Completion Time, the plant must be brought to a MODE in which the LCO does not apply. To achieve this status, the plant must be brought to at least MODE 3 within 12 hours and to MODE 4 within 36 hours. The allowed Completion Times are reasonable, based on operating experience, to reach the required plant conditions from full power conditions in an orderly manner and without challenging plant systems.

SURVEILLANCE REQUIREMENTS

SR 3.6.1.6.1

Each vacuum breaker is verified closed (except when being tested in accordance with SR 3.6.1.6.2 or when performing its intended function) to ensure that this potential large bypass leakage path is not present. This Surveillance is performed by observing the vacuum breaker position

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.6.1 (continued)

indication or by increasing the drywell pressure by 3.45 $\boxed{3.43}\times10^{-3}$ MPa above the wetwell pressure and verifying that the pressure differential does not fall below 2.06×10^{-3} MPaD for 15 minutes without makeup. This criteria was developed assuming ideal gas behavior, a leakage area corresponding to 10% of the allowable leakage area, the average temperatures in the wetwell and drywell remained within ± 0.5°C throughout the testing interval, and that adequate instrumentation exists to measure the pressure decay. Basing the test criteria on 10% of the allowable leakage area provides a large degree of margin in demonstrating that the vacuum breakers are adequately closed and sealed. Additionally, if the allowable leakage area were to exist, a 3.45 pressure differential of $\boxed{3.43}\times10^{-3}$ MPa would decay completely within 15 minutes. Maintaining the average temperatures of the wetwell and drywell is important because the pressure differentials in this test are relatively small and can be significantly impacted by small temperature changes. (However, if temperature control is a problem, new test parameters should be developed which take into account the normal temperature variations.)

The 14 day Frequency is based on engineering judgment and is considered adequate in view of the fact that the vacuum breakers are normally biased closed by gravitational forces. Verification of vacuum breaker closure is also required within 2 hours after any discharge of steam to the wetwell from the safety/relief valves or any operation that causes the drywell-to-wetwell differential pressure to be reduced by ≥ 6.86×10^{-4} MPaD.

SR 3.6.1.6.2

Each vacuum breaker must be cycled to ensure that it opens adequately to perform its design function and returns to the fully closed position. This ensures that the safety analysis assumptions are valid. The 18 month Frequency of this SR is based on the need to perform the surveillance during an outage. The vacuum breakers can only be manually actuated and are only accessible during an outage.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.1.6.3

3.45

Verification of the vacuum breaker opening pressure is necessary to ensure the validity of the safety analysis assumption that the vacuum breakers are fully open when the wetwell pressure exceeds the drywell pressure by ~~3.43~~ $\times 10^{-3}$ MPa. The 18 month Frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage. The 18 month Frequency is acceptable based on the passive design of the vacuum breakers (no actuator required for opening).

SR 3.6.1.6.4

A CHANNEL CALIBRATION is a complete check of the instrument loop and the sensor. The test verifies that the channel responds to the measured parameter with the necessary range and accuracy. The 18 month frequency is based on the ABWR expected refueling interval and the need to perform this Surveillance under the conditions that apply during a plant outage.

REFERENCE

1. DCD Tier 2, Section 6.2.

BASES

BACKGROUND (continued)

condense the steam from bypass leaks from the drywell to the wetwell airspace during the postulated LOCA.

APPLICABLE SAFETY ANALYSES

Reference 1 contains the results of analyses that predict the primary containment pressure response for a LOCA with the maximum bypass leakage effective area. The effective flow path area for bypass leakage has been calculated to be 5 cm^2, assuming no spray operation. With operation of one

containment

~~wetwell~~ spray subsystem, the effective bypass leakage area was calculated to be 50 cm^2.

The intent of the analyses is to demonstrate that the pressure reduction capacity of the RHR containment spray system operating in the wetwell spray mode is adequate to maintain the primary containment conditions within the design limit.

The RHR containment spray system satisfies Criterion 3 of the NRC Policy Statement.

LCO

In the event of a LOCA, a minimum of one RHR containment spray subsystem is required to mitigate potential bypass leakage paths and maintain the primary containment peak pressure below the design limits (Ref. 1). To ensure that these requirements are met, two RHR containment spray subsystems must be OPERABLE with power from two safety related independent power supplies. Therefore, in the event of an accident, at least one subsystem is OPERABLE, assuming the worst case single active failure. An RHR containment spray subsystem is OPERABLE when the pump, the heat exchanger, and associated piping, valves, instrumentation, and controls for both wetwell and drywell spray modes are OPERABLE.

APPLICABILITY

In MODES 1, 2, and 3, a LOCA could cause heatup and pressurization of the primary containment. In MODES 4 and 5, the probability and consequences of these events are reduced due to the pressure and temperature limitations in

(continued)

BASES (continued)

SURVEILLANCE REQUIREMENTS (continued)

SR 3.6.2.4.2

Verifying each associated RHR pump develops a flow rate ≥ 114 m^3/h ~~and less than 160 m^3/h~~ while operating in the wetwell spray mode with flow through the heat exchanger (operating in the suppression pool cooling mode) ensures that pump performance has not degraded during the cycle. Flow is a normal test of centrifugal pump performance required by Section XI of the ASME Code (Ref. 2). This test confirms one point on the pump design curve and is indicative of overall performance. Such inservice inspections confirm component OPERABILITY, trend performance, and detect incipient failures by indicating abnormal performance. In addition, verifying that the wetwell spray flow ensures that the assumptions for minimum flow for bypass leakage mitigation and the maximum flow for wetwell negative pressure evaluation in the Reference 1 analyses remain valid. The Frequency of this SR is 92 days.

REFERENCES

1. DCD Tier 2, Section 6.2.1.1.5.

2. ASME, Boiler and Pressure Vessel Code, Section XI.