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Table 14.3-1

Best-Estimate Large break LOCA Key Parameters and Reference Transient Assumptions

Parameter	**Reference Transient**	**Uncertainty or Bias**
1.0 Plant Physical Description		
a. Dimensions	Nominal	ΔPCT_{MOD}[1]
b. Flow resistance	Nominal	ΔPCT_{MOD}[1]
c. Pressurizer location	Opposite broken loop	Bounded
d. Hot assembly location	Under limiting location	Bounded
e. Hot assembly type	15x15 Upgrade 0.422" OD, ZIRLO™ clad, IFM Grids	Bounded
f. SG tube plugging level	High (10%)	Bounded*
2.0 Plant Initial Operating Conditions		
2.1 Reactor Power		
a. Core average linear heat rate (AFLUX)	Nominal – 100% of RTP (3216 MWt)	ΔPCT_{PD}[2]
b. Peak linear heat rate (PLHR)	FQ = 2.202, Derived from desired Tech Spec (TS) limit FQ = 2.5 and maximum baseload FQ = 2.0	ΔPCT_{PD}[2]
c. Hot rod average linear heat rate (HRFLUX)	FΔH = 1.731, Derived from TSFΔH = 1.7	ΔPCT_{PD}[2]
d. Hot assembly average rate (HAFLUX)	HRFLUX/1.04	ΔPCT_{PD}[2]
e. Hot assembly peak heat rate (HAPHR)	PLHR/1.04	ΔPCT_{PD}[2]
f. Axial power distribution (PBOT, PMID)	Figure 14.3-2	ΔPCT_{PD}[2]
g. Low power region relative power (PLOW)	0.8	Bounded*
h. Hot assembly burnup	BOL	Bounded
i. Prior operating history	Equilibrium decay heat	Bounded
j. Moderator Temperature Coefficient (MTC)	Maximum (0.0)	Bounded
k. HFP boron	800 ppm (at BOL)	Generic

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Table 14.3-1
(Cont.)

Best-Estimate Large Break LOCA Key Parameters and Reference Transient Assumptions

Parameter	Reference Transient	Uncertainty or Bias
2.2 Fluid Conditions		
a. T_{avg}	Nominal T_{avg} = 572.0°F	ΔPCT_{IC}[3]*
b. Pressurizer pressure	Nominal (2250.0 psia)	ΔPCT_{IC}[3]
c. Loop flow	88600 gpm	ΔPCT_{MOD}[1]**
d. T_{UH}	Best-Estimate (~ T_{hot})	0
e. Pressurizer level	Nominal (50.8% of span)	0
f. Accumulator temperature	Nominal (105°F)	ΔPCT_{IC}[3]
g. Accumulator pressure	Nominal (674.7 psia)	ΔPCT_{IC}[3]
h. Accumulator liquid volume (not including line or undeliverable volume)	Nominal (795 ft^3)	ΔPCT_{IC}[3]
i. Accumulator line resistance	Nominal	ΔPCT_{IC}[3]
j. Accumulator boron	Tech Spec Minimum	Bounded
3.0 Accident Boundary Conditions		
a. Break Location	Cold Leg	Bounded
b. Break Type	Guillotine	ΔPCT_{MOD}[1]
c. Break Size	Nominal (cold leg area)	ΔPCT_{MOD}[1]
d. Offsite Power	Not available (RCPs tripped)	Bounded*
e. Safety injection flow	Minimum	Bounded
f. Safety injection temperature	78°F, slightly above nominal (77.5°F)	ΔPCT_{IC}[3]
g. Safety injection delay	Max delay 27.8 sec (LOOP)	Bounded
h. Containment pressure	Minimum based on COCO containment pressure calculation results (Figure 14.3-18) using plant conditions supplied in Tables 14.3-4 & 14.3-5, and the Reference Transient M&E release (Table 14.3-6a)	Bounded

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Table 14.3-1
(Cont.)

Best-Estimate Large Break LOCA Key Parameters and Reference Transient Assumptions

Parameter	Reference Transient	Uncertainty or Bias
i. Single failure	ECCS: Loss of 1 SI train	Bounded
j. Control rod drop time	No control rods	Bounded
4.0 Model Parameters		
a. Critical flow	Nominal (Cd = 1.0)	ΔPCT_{MOD}[1]
b. Resistance uncertainties in broken loop	Nominal (as coded)	ΔPCT_{MOD}[1]
c. Initial stored energy/fuel rod behavior	Nominal (as coded)	ΔPCT_{MOD}[1]
d. Core heat transfer	Nominal (as coded)	ΔPCT_{MOD}[1]
e. Delivery and bypassing of ECC	Nominal (as coded)	Conservative
f. Steam binding/entrainment	Nominal (as coded)	Conservative
g. Non-condensable bases/accumulator nitrogen	Nominal (as coded)	Conservative
h. Condensation	Nominal (as coded)	ΔPCT_{MOD}[1]

Notes:
1.) PCT_{MOD} indicates this uncertainty is part of code and global model uncertainty
2.) PCT_{PD} indicates this uncertainty is part of power distribution uncertainty
3.) PCT_{IC} indicates this uncertainty is part of initial condition uncertainty

* Confirmed to be limiting
** Assumed to be result of loop resistance uncertainty

Table 14.3-2

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Table 14.3-2a

Best-Estimate Large Break LOCA Confirmatory Cases PCT Results Summary

Case	**PCT (°F)**		
	Blowdown	**1st Reflood**	**2nd Reflood**
Initial Transient	1461	1616	1605
Low Nominal RCS T_{avg} (549°F)	1464	1579	1536
No Loss of Offsite Power	1400	1461	1427
Reduced SGTP (0%)	1464	1591	1534
Increased PLOW (0.8) – Reference Transient	1491	1627	1578
Decreased PLOW (0.3)	1456	1607	1573

Table 14.3-2b

Best-Estimate Large Break LOCA Results

Component	**Blowdown**	**1st Reflood**	**2nd Reflood**	**Criteria**
50th Percentile PCT (°F)	<1480	<1568	<1600	N/A
95th Percentile PCT (°F)	<1736	<1904	<1944	<2200
Maximum Local Oxidation (%)	<7.60			<17.0
Maximum Total Hydrogen Generation (%)	<0.62			<1.0

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Table 14.3-3

Plant Operating Range Allowed by the Best-Estimate Large Break LOCA Analysis

Parameter		Operating Range
1.0	**Plant Physical Description**	
	a) Dimensions	No in-board assembly grid deformation during LOCA + SSE
	b) Flow resistance	N/A
	c) Pressurizer location	N/A
	d) Hot assembly location	Anywhere in core interior[1]
	e) Hot assembly type	15x15 OFA with IFMs, ZIRLO™ clad IFBA or Non-IFBA[2]
	f) SG tube plugging level	$\leq$ 10%
2.0	**Plant Initial Operating Conditions**	
	2.1 Reactor Power	
	a) Core average linear heat rate	Core power = 102% of 3216 MWt @ 2% Calorimetric Uncertainty
	b) Peak linear heat rate	$F_Q \leq 2.5$
	c) Hot rod average linear heat rate	$F_{\Delta H} \leq 1.7$
	d) Hot assembly average heat rate	$\bar{P}_{HA} \leq 1.7/1.04$
	e) Hot assembly peak heat rate	$F_{Q,HA} \leq 2.5/1.04$
	f) Axial power dist (PBOT, PMID)	Figure 14.3-17
	g) Low power region relative power (PLOW)	0.3 $\leq$ PLOW $\leq$ 0.8
	h) Hot assembly burnup	$\leq$ 75,000 MWD/MTU, lead rod
	i) Prior operating history	All normal operating histories
	j) MTC	$\leq$ 0 at HFP
	k) HFP boron	800 ppm (at BOL)
	l) Rod power census	Table 14.3-7
	2.2 Fluid Conditions	
	a) T_{avg}	549 – 5.5 $\leq T_{avg} \leq$ 572 +7.5(°F)[3]
	b) Pressurizer pressure	2250 – 55 = Pressurizer Pressure $\leq$ 2250 + 52 psia[4]
	c) Loop flow	$\geq$ 88600 gpm/loop
	d) T_{UH}	Current upper internals, T_{HOT} UH
	e) Pressurizer level	Nomal level, automatic controls
	f) Accumulator temperature	$80 \leq T_{ACC} \leq 130$°F
	g) Accumulator pressure	$555 \leq P_{ACC} \leq 715$ psia
	h) Accumulator volume	$715 \leq V_{ACC} \leq 875$ ft^3

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	i) Accumulator fl/D	Current line configuration
	j) Minimum accumulator boron	$\geq$ 2000 ppm
3.0	**Accident Boundary Conditions**	
	a) Break location	N/A
	b) Break type	N/A
	c) Break size	N/A
	d) Offsite Power	Available or LOOP
	e) Safety injection flow	Table 14.3-8
	f) Safety injection temperature	$35 \leq T_{SI} \leq 100°F$
	g) Safety injection delay	$\leq$ 15.0 seconds (with offsite power) $\leq$ 27.8 seconds (without offsite power)
	h) Containment pressure	Bounded, see Figure 14.3-18; Raw Data Tables 14.3-4 and 14.3-5.
	i) Single Failure	All trains operable[5]
	j) Control rod drop time	N/A

Notes:

1) Peripheral locations will not physically be lead power assembly
2) See Section 14.3.3.3.7 for associated IFBA and fuel evaluations
3) 549°F and 572°F are nominal values. The +/- values reflect bias and uncertainty.
4) 2250 psia is nominal value. The +/- values reflect bias and uncertainty.
5) Analysis considers loss of one train of pumped ECCS.

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TABLE 14.3-3B

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Table 14.3-4

Containment Data (Dry Containment)(Core Calculation)

Net Free Volume, ft^3	2.61×10^6
Initial Conditions	
Pressure, psia	14.7
Temperature, °F	90.0
RWST Temperature, °F	35.0
Service Water Temperature, °F	28.0
Outside Temperature, °F	-20.0
Spray System	
Number of Pumps Operating	2
Total Flow Rate, gpm	6783
Actuation Delay Time, seconds	20
Safeguards Fan Coolers	
Number of Fan Coolers Operating	5
Fastest Post-Accident Initiation of Fan Coolers, seconds	30

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Table 14.3 – 5

Structural Heat Sink Data

Thickness, in		Material	Area, ft^2
1)	0.0065	Paint	
	0.375	Steel	
	36.0	Concrete	49,838
2)	0.0065	Paint	
	0.500	Steel	
	36.0	Concrete	32,072
3)	12.0	Concrete	15,000
4)	0.375	Stainless Steel	
	12.0	Concrete	10,000
5)	12.0	Concrete	61,000
6)	0.0065	Paint	
	0.500	Steel	68,792
7)	0.0065	Paint	
	0.375	Steel	79,904
8)	0.0065	Paint	
	0.250	Steel	27,948
9)	0.0065	Paint	
	0.1875	Steel	69,800
10)	0.125	Steel	3,000
11)	0.138	Steel	22,000
12)	0.0065	Paint	10,000
	0.0625	Steel	
13)	0.0065	Paint	
	0.75	Steel	
	36.0	Concrete	785
14)	0.019	Stainless Steel	
	1.5	Insulation	
	0.375	Steel	
	36.0	Concrete	7,461
15)	0.375	Steel	1,800

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Table 14.3-6

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Table 14.3-6a

Best-Estimate Large Break LOCA Mass and Energy Releases from BCL Used for COCO Calculation at Selected Time Points

Time (sec)	M&E from Loop Side BCL		M&E from Vessel Side BCL	
	Mass Flow (lbm/s)	Energy Flow (Btu/s)	Mass Flow (lbm/s)	Energy Flow (Btu/s)
0.0	9431	5054211	-9	0
0.5	25755	13690582	51011	27128438
1.0	25451	13674828	49339	26235666
1.5	24690	13551964	46268	24604286
2.0	22846	12819397	41524	22096196
4.0	11772	7377433	26953	14467051
6.0	7806	5750281	22101	12098881
8.0	6035	5043209	18040	10422447
10.0	4081	3915462	13792	8234264
12.0	3470	3153564	9823	5990187
14.0	2729	2414386	10345	4907165
16.0	1464	1483439	8215	3265073
18.0	813	884113	7337	2274947
20.0	425	487408	6432	1574623
25.0	47	58072	0	0
30.0	48	60748	-23	0
35.0	34	43605	-54	0
40.0	105	130920	172	21274
45.0	105	131402	3197	485454
50.0	199	245064	2198	762857
60.0	56	70367	167	122120
70.0	46	58072	65	49332
80.0	46	57651	107	76809
90.0	56	70496	127	101991
100.0	50	63717	130	94485
110.0	58	73674	301	158709
120.0	59	74104	280	163947
130.0	57	72174	284	155722
140.0	52	65407	172	111102
150.0	52	65305	169	83030
160.0	53	66233	294	130280
170.0	61	75761	724	251458
180.0	83	92739	1123	292854
190.0	53	65879	590	179045
200.0	47	57883	146	69221
210.0	42	52114	162	77570
220.0	53	65436	257	143531
230.0	62	74610	446	183494
240.0	81	90385	366	163008
250.0	56	69276	345	161995

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Table 14.3-6b

Best-Estimate Large Brake LOCA Mass and Energy Releases from BCL Accumulator and SI

Time (Sec)	Mass Flow (lbm/s)	Energy Flow (Btu/s)	Enthalpy (Btu/lbm)
0.0 – 15.0	1838.7	91696	49.87
15.0 – 30.0	2299.6	93111	40.49
30.0 – end	328.84	1003	3.05

Table 14.3-7

Rod Census Used in Best-Estimate Large Break LOCA Analysis

Rod Group	Power Ratio	% of Core
1	1.0	10
2	0.912	10
3	0.853	10
4	0.794	10
5	0.735	10
6	0.676	10
7	<0.65	40

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Table 14.3-8

Best Estimate Large Break LOCA Total Minimum Injected Flow from HHSI and LHSI

RCS Pressure (psig)	Flow Rate (gpm)
0	3339.2
20	2877.7
40	2401.2
70	1629.7
80	1366.1
90	1016.6
100	718.1
110	544.1
200	529.2
400	441.3
600	324.4
800	174.2
1000	45

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Table 14.3-8a

Initial Parameters For Small Break LOCA Analysis

Licensed Core Power, (MWt)	3216
Total Peaking Factor, F_Q	2.50
Axial Offset, %	13
Hot Channel Enthalpy Rise Factor, $F_{\Delta H}$	1.70
Maximum Assembly Average Power, P_{HA}	1.51
Fuel Assembly Array	15x15 Upgraded w/IFMs
Nominal Accumulator Water Volume, ft^3	795
Accumulator Tank Volume, ft^3	1100
Minimum Accumulator Gas Pressure, psia	555
Loop Flow (gpm)	88600
Vessel Inlet Temperature, °F	540.37
Vessel Outlet Temperature, °F	603.63
RCS Pressure with Uncertainty, psia	2310
Steam Pressure, psia	709.19
Steam Generator Tube Plugging, %	10
Maximum Refueling Water Storage Tank Temperature, °F	110
Maximum Condensate Storage Tank Temperature, °F	120
Non-IFBA Fuel Backfill Pressure, psig	275
$2.0xB^{10}$ IFBA Fuel Backfill Pressure, psig	100
Reactor Trip Setpoint, psia	1748.7
Safety Injection Signal Setpoint, psig	1648.7
Safety Injection Delay Time, s	27.8
Signal Processing Delay and Rod Drop Time, s	4.7
Feedwater Trip Processing Delay Time, s	2
Time for Main Feedwater Flow Coastdown, s	10
Auxiliary Feedwater Pump Start Delay Time, s	60

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Table 14.3-8b

Small Break LOCA Time Sequence of Events

EVENT	Break Size		
	2 inch	3 inch	4 inch
Break Initiation, sec	0.0	0.0	0.0
Reactor Trip Signal, sec.	55.9	22.8	13.0
Safety Injection Signal, sec.	71.2	30.2	16.1
Top of Core Uncovered, sec.	1738	765	601
Accumulator Injection Begins, sec.	NA	1688	890
Peak Clad Temperature occurs, sec.	3518	1954	1053
Top of Core Recovered, sec	N/A	N/A	2560

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Table 14.3-8c

Small Break LOCA Analysis Results

EVENT	BREAK SIZE		
	2 inch	3 inch	4 inch
Peak Clad Temperature, °F	1182	1543	1380
Peak Clad Temperature Location, ft.	11.5	11.75	11.25
Local Zr/H_2O Reaction (max), %	0.12	1.04	0.21
Local Zr/H_2O Reaction Location, ft.	11.25	11.75	11.25
Total Zr/H_2O Reaction, %	<1.0	<1.0	<1.0
Hot Rod Burst Time, seconds	NA	NA	NA
Hot Rod Burst Location, ft.	NA	NA	NA

Tables 14.3-9 & 14.3-10

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Table 14.3-11

Maximum Deflections Allowed For Reactor Support Structures

Component	Allowable Deflections (in)	No-Loss-Of Function Deflections (in)
Upper Barrel		
radial inward	4.1	8.2
radial	1.0	1.0
Upper Package	0.10	0.15
Rod Cluster Guide Tubes	1.00	1.75

NOTE:
The allowable limit deflection values given above correspond to stress levels for the internals structure well below the limiting criteria given by the collapse curves in WCAP-5890 (Reference 60). Consequently, for the internals, the geometric limitations established to assure safe shutdown capability are more restrictive than those given by the failure stress criteria.

Table 14.3-12

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Table 14.3-13

Site Dispersion Factors

	Distance (meters)	(χ/Q) 2 hours (sec/m^3)	(χ/Q) 22 hours (sec/m^3)	(χ/Q) 30 days (sec/m^3)
*	350	10.3×10^{-4}	5.4×10^{-4}	1.35×10^{-4}
	400	9.51×10^{-4}	4.75×10^{-4}	1.03×10^{-4}
	700	5.98×10^{-4}	2.99×10^{-4}	3.87×10^{-5}
	1,000	4.20×10^{-4}	2.10×10^{-4}	2.07×10^{-5}
**	1,100	3.80×10^{-4}	1.90×10^{-4}	1.70×10^{-5}
	2,000	1.90×10^{-4}	9.50×10^{-5}	6.13×10^{-6}
	4,000	7.68×10^{-5}	3.84×10^{-5}	1.82×10^{-6}
	7,000	3.55×10^{-5}	1.77×10^{-5}	6.79×10^{-7}
	10,000	2.14×10^{-5}	1.07×10^{-5}	3.63×10^{-7}
	20,000	7.78×10^{-6}	3.89×10^{-6}	1.07×10^{-7}

These are plotted vs distance on Figure 14.3-73

* Site Boundary
** Low Population Zone

Tables 14.3-14, 14.3-14a, 14.3-14b, 14.3-14c, 14.3-14d, 14.3 14e

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Table 14.3-14f

Containment Sump And Recirculation Pipings Outside Containment Source Strength At Various Times Following A Maximum Credible Accident (TID-14844 Release Fraction)

Source Strength (MeV/cc-sec)

Eγ,MeV	0	0.5 HR.	2.0 HRS.	8.0 HRS.	1 DAY	7 DAYS	30 DAYS
0.2-0.4	3.12+09	1.09+09	8.76+09	7.59+08	5.65+08	2.34+08	2.92+07
0.5-0.9	1.17+10	7.79+09	4.28+09	1.64+09	7.98+08	1.50+08	7.01+07
0.9-1.35	7.01+09	3.31+09	1.95+09	8.57+08	1.95+08	8.76+06	2.34+06
1.35-1.8	6.82+09	3.31+09	1.73+09	6.04+08	1.48+08	4.48+07	1.29+07
1.8-2.2	2.92+09	1.69+09	9.93+08	2.34+08	1.25+07	1.79+06	7.20+05
2.2-2.6	3.31+09	2.14+09	1.19+09	2.53+08	1.31+07	2.73+06	7.79+05
2.6-3.0	1.58+09	2.53+08	1.17+08	1.67+07	3.70+05	4.67+04	1.34+04
3.0-4.0	1.11+09	1.44+08	4.87+07	7.01+06	1.48+05	1.83+04	5.26+03
4.0-5.0	8.18+08	6.23+06	5.06+06	0	0	0	0
5.0-6.0	3.70+06	4.67+04	0	0	0	0	0

Tables 14.3-14g, 14.3-14h, 14.3-15, 14.3-16, 14.3-17

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Table 14.3-18

Assumptions Used In The Analysis Of The
Environmental Consequences Of A Large Break LOCA

Source Term

Plant Power Level		3216 MWt
Core Activity Level		See Table 14C-4
Fraction of activity released from core	Gas Release	Core Melt
Iodines	0.05	0.35
Noble Gases	0.05	0.95
Alkali Metals	0.05	0.25
Tellurium Group	0.0	0.05
Strontium & Barium	0.0	0.02
Noble Metals	0.0	0.0025
Cerium Group	0.0	0.0005
Lanthanide Group	0.0	0.0002
Gap Release Timing (start / end)		30 sec / 30 min
Core Melt Timing (start / end)		30 min / 1.8 hr

Containment Leakage Model

Fraction of airborne Iodine in Containment Atmosphere	
Elemental Form	0.0485
Methyl Form	0.0015
Particulate Form	0.95
Containment Free Volume	2.61×10^6 ft^3
Fraction of Containment Sprayed	0.8
Spray Removal Coefficient (Injection Phase)	
Elemental Iodine	20 hr^{-1}, DF < 200
Methyl Iodine	0 hr^{-1}
Particulate Iodine	4.6hr^{-1}, DF < 50
Spray Injection Phase Timing (Start / End)	67 sec / 45 min
Spray Recirculation Phase Timing (Start / End)	48 min / 4.0 hr
Spray Removal Coefficient (Recirculation Phase)	
Elemental Iodine	10hr^{-1}, DEF < 200
Methyl Iodine	0 hr^{-1}
Particulates, DF <50	2.2 hr^{-1}
Particulates, DF >50	0.22hr^{-1}

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Table 14.3-18
(Cont.)

Assumptions Used In The Analysis Of The
Environmental Consequences Of A Large-Break LOCA

Containment Leakage Model (continued)	
Sedimentation Removal Coefficient	0.1 hr^{-1}, DF < 1000
Containment Leak Rate (0 - 24 hrs)	0.1% per day
(1 – 30 days)	0.05% per day
Fan Cooler Units	
FCU Flow Rate (per unit)	34,000 cfm
Number of units assumed operating	3 of 5
Time Delay to Initiate Operation	60 seconds
Flow Rate Through Filters (per unit)	8,000 cfm
Filter Efficiencies	No filtration credit assumed
Sump Solution Leakage Outside Containment	
Sump Solution Water Volume	374,400 gal
Leakage Through RCP Seal Leakoff Line	
Leak Rate	1.0 gph
Timing of Leakage (start / end)	0.0 hr / 4.0 hr
Iodine Partition Coefficient	0.028
ECCS Recirculation Outside Containment	
Leak Rate	4.0 gph
Timing of Leakage (Start/end)	6.5 hr / 30 days
Iodine Partition Coefficient	0.028
Iodine Species (after release to atmosphere)	
Elemental	0.97
Organic Form	0.03
Control Room Model Parameters	See Appendix 14C
Dose Calculation Inputs and Assumptions	
Nuclide Data	See Table 14C-5
Offsite Breathing Rate	
0 – 8 hours	3.5E-4m^3/sec
8 – 24 hours	1.8E-4m^3/sec
>24 hours	2.3E-4m^3/sec
Offsite Atmospheric Dispersion Factors	See Table 14.3-13

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Table 14.3-18a

Assumptions Used in the Analysis of the Environmental Consequences of a Small-Break LOCA

Source Term

Plant Power Level	3216 MWt
Core Activity	See Table 14C-4
Fraction of Activity Released from Core	
Iodines	0.05
Noble Gases	0.05
Alkali Metals	0.05
Release Timing	Instantaneous

Containment Leakage Release Path

Fraction of airborne Iodine in Containment Atmosphere	
Elemental Form	0.0485
Methyl Form	0.0015
Particulate Form	0.95
Containment Free Volume	2.61E6 ft^3
Fraction of Containment Sprayed	No spray operation
Containment Leak Rate (0 – 24 hrs)	0.1% per day
(1 – 30 days)	0.05% per day
Fan Cooler Units	
FCU Flow Rate (per unit)	34,000 cfm
Number of units assumed operating	3 of 5
Time Delay to Initiate Operation	60 seconds
Flow Rate Through Filters Per Unit	8,000 cfm
Filter Efficiencies	
Elemental Iodine	No credit assumed
Organic Iodine	No credit assumed
Particulates	90%

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Table 14.3-18a
(Cont.)

Assumptions used in the Analysis of the
Environmental Consequences of a Small-Break LOCA

Secondary Side Release Path	
Primary to Secondary Leakage	1.0 gpm
Secondary Side Water Mass	281,600 lb
Steam Release to Atmosphere	
0 -2 hr	405,229 lb
> 2 hr	0.0 lb
Iodine Partition Coefficient for Steaming	0.01
Iodine Form After Release to Atmosphere	
Elemental	97%
Organic	3%
Alkali Metal Partition Coefficient for Steaming	0.001
Control Room Model Parameters	See Appendix 14C
Dose Calculation Inputs and Assumptions	
Nuclide Data	See Table 14C-5
Offsite Breathing Rate	
0 – 8 hours	3.5E-4 m^3/sec
8 – 24 hours	1.8E-4 m^3/sec
>24 hours	2.3E-4 m^3/sec
Offsite Atmospheric Dispersion Factors	See Table 14.3-13

IPEC00036513

Table 14.3-19

Containment Design Parameters

Total Containment Volume	2.61×10^6 ft^3
Lower Containment Volume	0.37×10^6 ft^3
Upper Containment Volume	2.24×10^6 ft^3
Fan Cooler Filtered Flowrate	8,000 cfm per unit
Fan cooler filter efficiency for iodine -	90% elemental
	70% organic
	90% particulate
Containment leak rate after isolation	
0-24 hours	0.1% per day
>24 hours	0.05% per day

IPEC00036514

Table 14.3-20

Reactor Coolant System Equilibrium Activities
(From Table 9.2-5)

Isotope	uc / cc (573 F)
I-131	1.64
I-132	0.605
I-133	2.67
I-134	0.377
I-135	1.44
Xe-133	192.0
Xe-135	4.24
Xe-138	0.46
Kr-85	4.4
Kr-85m	1.43
Kr-87	0.83
Kr-88	2.51

IPEC00036515

Table 14.3-21

Other Parameters Used in Evaluation of Environmental Consequences Of Accident

Meteorology
X/Q (sec/m^3)

Hours After Accident	Site Boundary	Low Population Zone	Breathing* Rate (m/sec)
0-2	1.03×10^{-3}	3.8×10^{-4}	3.47×10^{-4}
2-8	5.4×10^{-4}	1.8×10^{-4}	3.47×10^{-4}
8-24	5.4×10^{-4}	1.9×10^{-4}	1.75×10^{-4}
24-270	1.35×10^{-4}	1.70×10^{-5}	2.32×10^{-4}

*Safety Guide 4 and TID-14844

Table 14.3-22

Doses From Rupture of Pressurizer During Containment Purging

Dose	Containment Isolation Time	Exclusive Radius (0-2 hrs)	Dose (Rem) Low Population Zone (0-30 days)
Thyroid	15 minutes	25.2	9.3
Thyroid	50 minutes	108	39.8
Whole body	15 minutes	0.48	0.18
Whole body	50 minutes	2.05	0.78

IPEC00036516

Table 14.3-23

System Parameters Initial Conditions

Parameters	Value
Core Thermal Power (MWt)	3216
Reactor Coolant System Total Flow Rate (lbm/sec)	37,444.4
Vessel Outlet Temperature (°F)	610.5
Core Inlet Temperature (°F)	548.5
Vessel Average Temperature (°F)	572.0
Initial Steam Generator Steam Pressure (psia)	787.0
Steam Generator Design	Model 44F
Steam Generator Tube Plugging (%)	0
Initial Steam Generator Secondary Side Mass (lbm)	100,668.7
Assumed Maximum Containment Backpressure (psia)	61.7
Accumulator	
Water Volume (ft^3)	807.2
N_2 Cover Gas Pressure (psia)	555.0
Temperature (°F)	130.0
Safety Injection Delay From Beginning of Event (sec)	27.8

Note: RCS Coolant Temperature, and Steam Generator Secondary Side Mass include appropriate uncertainty and/or allowance.

IPEC00036517

Table 14.3-24

Total Pumped ECCS Flow Rate to All Four Loops Diesel Failure (Minimum ECCS)

INJECTION MODE (REFLOOD PHASE)	
RCS Pressure (psia)	**Total Flow (gpm)**
14.7	5252.3
24.7	5115.1
34.7	4975.2
44.7	4832.7
54.7	4687.2
64.7	4536.1
74.7	4367.1
84.7	4192.8
94.7	4012.4
104.7	3825.0
114.7	3630.0
INJECTION MODE (POST-REFLOOD PHASE)	
RCS Pressure (psia)	**Total Flow (gpm)**
61.7	581.4
COLD LEG RECIRCULATION MODE	
RCS Pressure (psia)	**Total flow (gpm)**
61.7	2080.0

IPEC00036518

Table 14.3-25

Total Pumped ECCS Flow Rate to All Four Loops No Failure (Maximum ECCS)

INJECTION MODE (REFLOOD PHASE)	
RCS Pressure (pisa)	**Total Flow (gpm)**
14.7	7815.6
34.7	7479.7
54.7	7129.7
74.7	6745.8
94.7	6330.8
114.7	5885.9
134.7	5403.6
154.7	4866.3
174.7	4215.0
194.7	3414.7
214.7	2180.4
234.7	1332.7
314.7	1290.1
414.7	1234.6
INJECTION MODE (POST-REFLOOD PHASE)	
RCS Pressure (psia)	**Total Flow (gpm)**
61.7	6995.3
COLD LEG RECIRCULATION MODE	
RCS Pressure (psia)	**Total Flow (gpm)**
61.7	4160

IPEC00036519

Table 14.3-26

Decay Heat Curve 1979 ANS Plus 2 Sigma Uncertainty

Time (Sec)	Decay Heat Generation Rate (BTU/BTU)
1.00E+01	0.053876
1.50E+01	0.050401
2.00E+01	0.048018
4.00E+01	0.042401
6.00E+01	0.039244
8.00E+01	0.037065
1.00E+02	0.035466
1.50E+02	0.032724
2.00E+02	0.030936
4.00E+02	0.027078
6.00E+02	0.024931
8.00E+02	0.023389
1.00E+03	0.022156
1.50E+03	0.019921
2.00E+03	0.018315
4.00E+03	0.014781
6.00E+03	0.013040
8.00E+03	0.012000
1.00E+04	0.011262
1.50E+04	0.010097
2.00E+04	0.009350
4.00E+04	0.007778
6.00E+04	0.006958
8.00E+04	0.006424
1.00E+05	0.006021
1.50E+05	0.005323
4.00E+05	0.003770
6.00E+05	0.003201
8.00E+05	0.002834
1.00E+06	0.002580
1.00E+07	0.000808

IPEC00036520

Table 14.3-27

DEHL Break Blowdown M&E Releases (Minimum ECCS)

Time (sec)	Break Path No. 1*		Break Path No. 2 †	
	(lbm/sec)	(1000 BTU/sec)	(lbm/sec)	(1000 BTU/sec)
0.0	0.0	0.0	0.0	0.0
0.001	43216.1	27017.6	43213.1	27014.4
0.002	44433.6	27779.4	44164.6	27604.0
0.1	45538.9	28778	25444.9	15873.6
0.2	32791.7	21154.0	22595.2	14017.6
0.3	32084.8	20646.9	20305.9	12441.9
0.4	31280.4	20117.6	19120.5	11538.7
0.5	31012.0	19936.5	18348.2	10908.8
0.6	30970.6	19912.6	17786.9	10431.7
0.7	30879.0	19877.0	17343.9	10051.4
0.8	30588.4	19730.6	17023.3	9762.1
0.9	30227.7	19552.0	16732.0	9505.9
1.0	29827.8	19360.2	16560.6	9329.1
1.1	29559.7	19264.0	16444.0	9194.1
1.2	29306.8	19187.0	16442.2	9130.6
1.3	29048.7	19106.1	16506.8	9108.9
1.4	28718.0	18974.1	16618.7	9118.1
1.5	28330.4	18796.2	16751.8	9144.7
1.6	27926.6	18603.7	16900.6	9184.9
1.7	27555.8	18429.8	17051.8	9232.4
1.8	27184.5	18255.6	17199.2	9282.7
1.9	26773.1	18050.8	17332.3	9330.3
2.0	26314.9	17808.3	17447.9	9372.9
2.1	25851.9	17556.4	17543.1	9408.5

*mass and energy exiting from the reactor vessel side of the break
†mass and energy exiting from the steam generator side of the break

IPEC00036521

Table 14.3-27
(Cont.)

DEHL Break Blowdown M&E Releases (Minimum ECCS)

Time (sec)	Break Path No. 1*		Break Path No. 2†	
	(lbm/sec)	(1000 BTU/sec)	(lbm/sec)	(1000 BTU/sec)
2.2	25391.2	17303.5	17619.5	9437.1
2.3	24938	17054.4	17679.3	9459.3
2.4	24496.9	16809.6	17722.1	9474.5
2.5	24046.1	46552.9	17750.4	9483.7
2.6	23573.2	16273.0	17766.3	9487.6
2.7	23114.2	15997.0	17771.4	9486.7
2.8	22689.1	15743.8	17768.4	9482.3
2.9	2284.5	15500.2	17756.4	9473.7
3.0	21875.4	15243.8	17734.3	9460.2
3.1	21492.2	15000.3	17702.9	9442.1
3.2	21129.3	14765.0	17663.7	9420.1
3.3	20779.2	14529.5	17615.8	9393.8
3.4	20470.4	14319.2	17561.3	9364.3
3.5	20180.8	14117.6	17501.0	9331.9
3.6	19903.1	13914.9	17434.1	9296.3
3.7	19644.4	13720.7	17361.0	9257.8
3.8	19414.0	13544.5	17283.1	9216.9
3.9	19194.4	13368.4	17199.0	9173.1
4.0	19004.3	13209.8	17109.4	9126.7
4.2	18688.3	12931.8	16914.7	9026.8
4.4	18436.2	12689.9	16696.0	8916.1
4.6	18246.3	12490.3	16458.1	8797.2
4.8	18184.6	12385.8	16197.0	8668.2
5.0	18242.3	12349.7	15914.4	8530.2
5.2	18416.6	12359.7	15630.4	8393.8
5.4	18634.4	12390.6	15313.5	8240.9
5.6	18872.2	12431.3	14943.5	8059.9
5.8	19167.6	12503.3	14553.8	7868.7
6.0	19546.1	12605.6	14202.2	7697.4
6.2	11596.1	9053.2	13865.2	7532.8

*mass and energy exiting from the reactor vessel side of the break
†mass and energy exiting from the steam generator side of the break

IPEC00036522

Table 14.3-27
(Cont.)

DEHL Break Blowdown M&E Releases (Minimum ECCS)

Time (sec)	Break Path No. 1*		Break Path No. 2†	
	(lbm/sec)	(1000 BTU/sec)	(lbm/sec)	(1000 BTU/sec)
6.4	14615.2	10424.6	13543.6	7374.6
6.6	14498.6	10297.0	13191.7	7197.4
6.8	14631.6	10261.7	12824.2	7009.7
7.0	14823.9	10328.0	12479.4	6833.2
7.2	15043.6	10422.7	12161.2	6669.8
7.4	15250.0	10436.2	11845.6	6506.1
7.6	15449.4	10476.1	11532.8	6342.4
7.8	15635.7	10580.4	11228.3	6182.3
8.0	15575.1	10451.1	10941.3	6031.2
8.2	15901.5	10556.6	10677.6	5892.1
8.4	16217.3	10658.8	10419.5	5755.5
8.6	16550.9	10772.6	10167.7	5621.8
8.8	16971.8	10933.5	9920.2	5490.0
9.0	17728.7	11275.9	9678.6	5361.3
9.2	18541.3	11698.2	9442.1	5235.5
9.4	18929.1	11866.3	9206.6	5110.3
9.6	19223.8	11965.3	8971.5	4985.4
9.8	18854.0	11651.4	8726.2	4855.0
10.0	17951.0	11023.8	8481.1	4725.3
10.2	14860.8	9376.5	8233.4	4595.0
10.4	14303.8	9059.4	8004.6	4475.3
10.6	14418.5	9086.4	7784.8	4361.1
10.8	14561.4	9141.4	7592.6	4262.5
11.0	14722.8	9210.1	7417.9	4172.7
11.2	14873.7	9265.9	7245.6	4083.2
11.4	15060.7	9333.2	7079.6	3997.0
11.6	15358.1	9452.3	6918.4	3913.3
11.8	15822.0	9672.5	6752.4	3827.3
12.0	15709.9	9570.8	6587.3	3742.1
12.2	15462.5	9384.3	6419.7	3656.1
12.4	14575.9	8878.8	6246.4	3597.5
12.6	12813.6	7959.4	6076.2	3481.5
12.8	12586.2	7813.6	5909.8	3398.4
13.2	12565.4	7764.9	5614.8	3253.7
13.4	12544.7	7740.0	5481.0	3188.0

*mass and energy exiting from the reactor vessel side of the break
†mass and energy exiting from the steam generator side of the break

IPEC00036523

Table 14.3-27
(Cont.)

DEHL Break Blowdown M&E Releases (Minimum ECCS)

Time (sec)	Break Path No. 1*		Break Path No. 2†	
	(lbm/sec)	(1000 BTU/sec)	(lbm/sec)	(1000 BTU/sec)
13.6	12502.3	7704.5	5356.4	3126.3
13.8	12413.7	7644.8	5235.4	3066.3
14.0	12248.4	7546.3	5115.8	3007.0
14.2	11967.8	7389.8	4998.4	2949.2
14.4	11490.4	7165.4	4884.3	2893.4
14.6	10863.9	6930.2	4767.2	2836.4
14.8	10495.2	6789.9	4653.9	2781.9
15.0	10225.7	6678.1	4544.9	2729.6
15.2	9956.4	6553.4	4433.8	2676.3
15.4	9643.3	6398.4	4323.8	2623.7
15.6	9288.3	6219.5	4213.7	2571.5
15.8	8937.9	6045.4	4099.8	2517.7
16.0	8619.6	5891.9	3979.2	2461.0
16.2	8322.4	5755.6	3846.8	2399.1
16.4	8019.5	5624.2	3698.2	23308
16.6	7695.6	5489.8	3534.0	2256.5
16.8	7342.7	5348.1	3358.3	2177.1
17.0	6962.8	5199.8	3179.0	2094.7
17.2	6557.4	5045.0	3004.6	2012.0
17.4	6136.6	4886.9	2844.0	1932.3
17.6	5701.0	4725.1	2700.7	1857.7
17.8	5260.7	4562.3	2576.3	1789.8
18.0	4822.8	4398.4	2470.4	1729.7
18.2	4368.6	4198.7	2379.7	1677.0
18.4	3966.3	3932.9	2297.4	1628.2
18.6	3702.8	3715.2	2224.0	1584.6
18.8	3502.0	3553.9	2156.6	1544.8
19.0	3337.0	3431.7	2092.0	1508.1
19.2	3175.6	3312.9	2028.2	1473.8
19.4	3006.4	3193.8	1899.5	1441.3
19.6	2833.6	3074.0	1899.5	1410.3
19.8	2659.6	2943.8	1832.2	1378.9
20.0	2476.5	2810.7	1765.5	1348.9
20.2	2279.9	2650.6	1697.5	1319.2

*mass and energy exiting from the reactor vessel side of the break
†mass and energy exiting from the steam generator side of the break

IPEC00036524

Table 14.3-27
(Cont.)

DEHL Break Blowdown M&E Releases (Minimum ECCS)

Time (sec)	Break Path No. 1*		Break Path No. 2†	
	(lbm/sec)	(1000 BTU/sec)	(lbm/sec)	(1000 BTU/sec)
20.4	2105.3	2493.7	1626.0	1289.3
20.6	1969.1	2360.6	1546.6	1258.6
20.8	1835.2	2217.8	1461.5	1230.8
21.0	1703.9	2071.2	1370.8	1202.4
21.2	1582.3	1933.8	1282.9	1171.9
21.4	1471.4	1809.0	1209.5	1142.4
21.6	1373.0	1697.9	1151.9	1116.0
21.8	1298.2	1615.2	1107.2	1095.8
22.0	1257.9	1571.0	1068.6	1074.2
22.2	1194.9	1500.0	1040.6	1054.5
22.4	1119.7	1408.8	1019.6	1036.8
22.6	1047.1	1319.4	1004.2	1019.2
22.8	976.8	1232.5	995.6	1003.8
23.0	928.8	1170.9	988.5	990.1
23.2	860.1	1086.8	975.4	980.3
23.4	770.3	973.1	938.3	975.2
23.6	704.9	893.0	853.3	973.8
23.8	641.7	813.7	661.6	802.7
24.0	588.2	696.1	574.4	703.1
24.2	548.1	595.1	579.4	709.1
24.4	518.7	658.7	512.3	627.2
24.6	500.9	636.1	371.8	456.9
24.8	487.5	618.7	319.8	394.3
25.0	475.6	603.1	247.6	305.8
25.2	54.2	69.9	199.2	247.1
25.4	0.0	0.0	95.2	118.9
25.6	0.0	0.0	0.0	0.0

* mass and energy exiting from the reactor vessel side of the break
† mass and energy exiting from the steam generator side of the break

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Table 14.3-28

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IPEC00036525

Table 14.3-29

DEHL Break Mass Balance (Minimum ECCS)

	Time (sec)	0.0	25.6	25.6
		Mass (1000 lbm)		
Initial	In RCS and Accumulator	732.01	732.01	732.01
Added Mass	Pumped Injection	.00	.00	.00
	Total Added	.00	.00	.00
Total Available		732.01	732.01	732.01
Distribution	Reactor Coolant	527.21	61.26	88.21
	Accumulator	204.80	158.37	131.42
	Total Contents	732.01	219.63	219.63
Effluent	Break Flow	.00	512.36	512.36
	ECCS Spill	.00	.00	.00
	Total Effluent	.00	512.36	512.36
***Total Accountable		732.01	731.98	731.98

Table 14.3-30

Double-Ended Hot Leg Break Energy Balance (Minimum ECCS)

	Time (sec)	.00	25.60	25.60
		Energy (million BTU)		
Initial Energy	In RCS, Accumulator, Steam Generator	775.34	775.34	775.34
Added Energy	Pumped Injection	.00	.00	.00
	Decay Heat	.00	7.72	7.72
	Heat from Secondary	.00	9.96	9.96
	Total Added	.00	17.68	17.68
Total Available		775.34	793.02	793.02
Distribution	Reactor Coolant	305.75	15.57	18.25
	Accumulator	20.35	15.73	13.06
	Core Stored	26.87	10.59	10.59
	Primary Metal	166.26	156.28	156.28
	Secondary Metal	40.98	40.06	40.06
	Steam Generator	215.15	227.53	227.53
	Total Contents	775.34	465.77	465.77
Effluent	Break Flow	.00	326.77	326.77
	ECCS Spill	.00	.00	.00
	Total Effluent	.00	326.77	326.77
Total Accountable		775.34	792.53	792.53

IPEC00036526

Table 14.3-31 & 14.3-32

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Table 14.3-33

DEPS Break Blowdown M&E Release (Minimum ECCS)

Time	Break Path No.1*		Break Path No. 2†	
(sec)	(lbm/sec)	(1000 BTU/sec)	(lbm/sec)	(1000 BTU/sec)
0.0	0.0	0.0	0.0	0.0
0.001	81761.8	44233.2	40576.0	21912.8
0.1	40368.4	21885.1	19793.9	10677.7
0.2	45306.7	24772.6	22451.3	12124.9
0.3	45415.9	25096.5	23512.5	12704.7
0.4	45055.9	25215.1	23517.2	12711.4
0.5	44009.1	24945.6	22969.5	12420.5
0.6	44256.3	25378.4	22403.5	12120.2
0.7	43600.3	25252.0	22049.2	11934.2
0.8	42230.5	24671.9	21887.5	11851.0
0.9	40998.0	24158.5	21772.1	11792.1
1.0	39954.5	23761.7	21676.2	11742.9
1.1	38791.7	23329.2	21567.5	11686.1
1.2	37322.5	22738.0	21464.4	11631.9
1.3	35681.4	22020.7	21381.4	11588.1
1.4	34261.3	21372.7	21325.8	11558.9
1.5	33185.6	20878.9	21311.8	11552.2
1.6	32347.7	20502.1	21355.1	11565.8
1.7	31543.5	20137.7	21256.7	11523.4
1.8	30691.4	19737.1	21078.4	11426.5
1.9	29771.4	19286.1	20897.8	11328.4
2.0	28806.8	18796.0	20735.0	11240.3
2.1	27794.9	18268.3	20582.8	11158.0
2.2	26813.6	17759.7	20406.0	11062.5
2.3	25407.1	16959.1	20198.9	10950.3
2.4	23314.1	15671.6	19979.5	10831.5
2.5	21428.3	14504.8	19781.6	10724.5
2.6	21061.3	14354.1	19588.9	10620.6
2.7	20375.5	13940.6	19405.6	10521.9
2.8	19647.9	13490.3	19200.4	10411.3

*mass and energy exiting the SG side of the break
†mass and energy exiting the pump side of the break

IPEC00036527

Table 14.3-33
(Cont.)

DEPS Break Blowdown M&E Releases (Minimum ECCS)

Time	Break Path No. 1*		Break Path No. 2†	
(sec)	(lbm/sec)	(1000 BTU/sec)	(lbm/sec)	(1000 BTU/sec)
2.9	19327.9	13315.2	19003.6	10305.3
3.0	18990.9	13110.6	18809.5	10201.0
3.1	18936.7	13103.3	18599.8	10088.1
3.2	18708.8	12965.8	18359.2	9958.3
3.3	18376.1	12765.8	18107.3	9822.4
3.4	18036.8	12557.7	17873.1	9696.2
3.5	17593.9	12267.4	17641.1	9571.4
3.6	17069.3	11917.1	17408.8	9446.3
3.7	16489.9	11528.9	17177.4	9321.8
3.8	15903.3	11135.1	16954.4	9202.0
3.9	15363.2	10772.2	16746.0	9090.1
4.0	14881.1	10447.4	16551.1	8985.8
4.2	14051.8	9886.3	16181.4	8787.9
4.4	13368.7	9425.1	15844.1	8607.7
4.6	12849.6	9066.2	15533.4	8442.0
4.8	12412.2	8757.6	15247.7	8289.7
5.0	12013.1	8465.4	14999.3	8157.8
5.2	11703.6	8224.2	14763.6	8032.5
5.4	11586.3	8094.8	14554.2	7921.5
5.6	11514.6	7994.2	14351.2	7813.7
5.8	11482.1	7922.8	14628.1	7971.2
6.0	11522.2	7898.5	14747.6	8037.3
6.2	12194.7	8296.4	14583.2	7949.9
6.4	12141.3	8354.3	14758.0	8049.7
6.6	10307.9	7832.3	14611.9	7971.5
6.8	9121.4	7303.9	14448.4	7884.8
7.0	9073.9	7243.1	14317.3	7815.8
7.2	9131.3	7217.0	14150.0	7726.8
7.4	8251.8	7207.9	13997.8	7646.3
7.6	9481.4	7234.9	13888.3	7588.7

*mass and energy exiting the steam generator side of the break
† mass and energy exiting the pump side of the break

IPEC00036528

Table 14.3-33
(Cont.)

DEPS Break Blowdown M&E Releases (Minimum ECCS)

Time	Break Path No. 1*		Break Path No. 2†	
(sec)	(lbm/sec)	(1000 BTU/sec)	(lbm/sec)	(1000 BTU/sec)
7.8	9840.4	7325.9	13772.0	7525.2
8.0	10359.8	7518.4	13588.0	7423.3
8.2	11037.7	7812.7	13409.2	7324.2
8.4	11803.6	8167.7	13241.4	7231.2
8.6	12593.0	8547.8	13064.4	7133.0
8.8	13245.7	8851.5	12873.9	7027.5
9.0	13483.0	8908.3	12687.9	6924.7
9.2	13276.1	8712.1	12523.4	6834.0
9.4	12965.9	8476.1	12374.3	6751.5
9.6	12574.6	8194.2	12222.6	6667.2
9.8	11645.6	7582.1	12081.1	6588.6
10.0	10593.6	6936.8	12004.7	6546.0
10.2	10208.2	6743.1	11946.2	6512.7
10.4	9933.1	6606.9	11771.6	6414.9
10.6	9558.0	6412.6	11662.9	6354.9
10.8	9363.2	6333.5	11584.3	6312.0
11.0	9110.8	6193.7	11405.8	6213.9
11.2	8843.7	6050.6	11321.2	6168.3
11.4	8611.3	5935.8	11207.4	6105.7
11.6	8331.5	5791.9	11060.1	6024.8
11.8	8094.4	5682.6	10972.5	5976.9
12.0	7841.1	5558.8	10793.8	5878.6
12.2	7631.1	5453.0	10669.0	5810.9
12.4	7447.7	5345.9	10550.1	5746.4
12.6	7300.3	5249.1	10397.4	5663.0
12.8	7176.9	5155.0	10272.0	5594.7
13.0	7066.5	5061.5	10136.5	5520.5
13.2	6964.1	4969.2	10003.1	5447.9
13.4	6860.9	4873.9	9867.7	5374.1
13.6	6756.5	4776.9	9730.9	5299.6
13.8	6652.7	4679.5	9599.0	5227.9

*mass and energy exiting the steam generator side of the break
†mass and energy exiting the pump side of the break

IPEC00036529

Table 14.3-33
(Cont.)

DEPS Break Blowdown M&E Releases (Minimum ECCS)

Time	Break Path No. 1*		Break Path No. 2†	
(sec)	(lbm/sec)	(1000 BTU/sec)	(lbm/sec)	(1000 BTU/sec)
14.0	6551.1	4582.4	9463.3	5154.1
14.2	6454.6	4487.7	9332.5	5083.2
14.4	6364.5	4396.4	9204.0	5013.5
14.6	6286.8	4312.4	9088.1	4950.7
14.8	6229.3	4241.5	8991.4	4898.8
15.0	6169.6	4171.4	8866.3	4830.2
15.2	6106.1	4106.5	8779.1	4783.6
15.4	6042.3	4043.3	8675.6	4727.6
15.6	5976.8	3981.7	8590.2	4682.1
15.8	5911.6	3925.6	8496.0	4631.6
16.0	5840.4	3870.9	8414.4	4588.8
16.2	5773.5	3823.7	8339.7	4550.2
16.4	5698.8	3775.3	8193.0	4472.2
16.6	5626.7	3739.9	8057.1	4403.0
16.8	5544.7	3720.5	7908.6	4327.8
17.0	5421.2	3692.9	7741.4	4242.5
17.2	5275.7	3658.2	7585.2	4161.0
17.4	5130.3	3623.8	7425.0	4062.6
17.6	4987.5	3591.5	7279.3	3956.0
17.8	4848.1	3560.2	7149.2	3845.8
18.0	4712.0	3529.8	7046.1	3743.0
18.2	4578.9	3500.4	6951.5	3642.4
18.4	4447.4	3473.6	6860.8	3545.2
18.6	4317.2	3448.0	6736.5	3434.4
18.8	4184.1	3423.5	6572.9	3308.7
19.0	4049.4	3401.2	6399.0	3183.7
19.2	3911.0	3380.1	6214.5	3060.8
19.4	3769.8	3361.2	6034.1	2949.3
19.6	3623.5	3344.6	5864.7	2853.8
19.8	3472.5	3330.4	5695.8	2768.6
20.0	3285.7	3291.4	5492.9	2673.1

*mass and energy exiting the steam generator side of the break
†mass and energy exiting the pump side of the break

IPEC00036530

Table 14.3-33
(Cont.)

DEPS Break Blowdown M&E Releases (Minimum ECCS)

Time (sec)	Break Path No. 1*		Break Path No. 2†	
	(lbm/sec)	(1000 BTU/sec)	(lbm/sec)	(1000 BTU/sec)
20.2	3023.0	3191.9	5043.3	2439.6
20.4	2764.3	3072.7	4849.5	2304.6
20.6	2543.6	2957.3	4656.1	2205.9
20.8	2393.5	2873.4	4365.2	2058.6
21.0	2180.7	2661.6	4194.5	1968.7
21.2	2028.5	2492.3	3842.0	1786.4
21.4	1885.9	2325.5	3636.1	1644.4
21.6	1764.8	2182.2	3508.4	1551.5
21.8	1659.9	2056.6	3113.1	1339.6
22.0	1549.0	1922.8	2768.0	1141.4
22.2	1453.0	1806.5	2485.3	983.1
22.4	1366.4	1701.7	2269.9	866.9
22.6	1282.0	1598.2	2098.7	777.9
22.8	1194.9	1492.1	2020.9	728.2
23.0	1119.7	1399.9	2062.5	723.2
23.2	1045.0	1307.9	2203.1	754.1
23.4	958.1	1200.9	2404.8	807.2
23.6	868.4	1089.6	2598.9	859.2
23.8	786.1	987.3	2745.0	896.0
24.0	701.2	881.4	2904.0	935.2
24.2	614.4	772.9	3064.9	971.6
24.4	528.0	664.7	3199.8	997.3
24.6	446.8	562.9	3189.0	977.5
24.8	370.2	466.8	2986.6	903.0
25.0	301.9	380.8	2789.6	834.7
25.2	239.4	302.3	2590.3	768.5
25.4	183.9	232.4	2383.7	702.3
25.6	142.6	180.4	2177.3	637.8
25.8	127.2	161.1	1969.4	574.5
26.0	105.9	134.2	1761.3	512.4
26.2	58.4	74.2	1556.4	452.4
26.4	0.0	0.0	1339.4	389.5
26.6	0.0	0.0	1082.8	315.5
26.8	0.0	0.0	724.1	211.6
27.0	0.0	0.0	25.5	7.5
27.2	0.0	0.0	0.0	0.0

*mass and energy exiting the steam generator side of the break
†mass and energy exiting the pump side of the break

IPEC00036531

Table 14.3-34

DEPS Break Reflood M&E Releases (Minimum ECCS)

Time	Break Path No. 1		Break Path No. 2	
(sec)	(lbm/sec)	(1000 BTU/sec)	(lbm/sec)	(1000 BTU/sec)
27.2	0.0	0.0	0.0	0.0
27.8	0.0	0.0	0.0	0.0
27.9	0.0	0.0	0.0	0.0
28.1	0.0	0.0	157.7	12.3
28.2	0.0	0.0	157.7	12.3
28.3	46.8	55.1	157.7	12.3
28.4	31.0	36.5	157.7	12.3
28.6	12.5	14.8	157.7	12.3
28.7	13.3	15.7	157.7	12.3
28.8	15.5	18.2	157.7	12.3
28.9	25.2	29.7	157.7	12.3
29.0	29.2	34.4	157.7	12.3
29.1	35.0	41.2	157.7	12.3
29.2	39.6	46.7	157.7	12.3
29.3	43.8	51.6	157.7	12.3
29.4	47.8	56.4	157.7	12.3
29.5	51.3	60.5	157.7	12.3
29.6	54.5	64.2	157.7	12.3
29.7	58.1	68.5	157.7	12.3
29.8	60.3	71.0	157.7	12.3
29.9	63.8	75.2	157.7	12.3
30.0	66.4	78.3	157.7	12.3
30.1	69.0	81.3	157.7	12.3
30.2	71.5	84.3	157.7	12.3
30.3	73.9	87.1	157.7	12.3
31.3	95.3	112.3	157.7	12.3
32.3	113.0	133.1	157.7	12.3
33.3	128.2	151.1	157.7	12.3
34.3	141.8	167.2	157.7	12.3
34.8	147.5	173.9	157.7	12.3
35.3	154.0	181.6	157.7	12.3
36.6	255.5	301.7	2311.3	359.4
37.3	364.4	431.1	3697.8	614.5
38.3	367.7	435.0	3727.8	628.4

IPEC00036532

Table 14.3-34
(Cont.)

DEPS Break Reflood M&E Releases (Minimum ECCS)

Time	Break Path No. 1		Break Path No. 2	
(sec)	(lbm/sec)	(1000 BTU/sec)	(lbm/sec)	(1000 BTU/sec)
39.3	362.5	428.8	3670.1	622.1
40.0	358.5	424.1	3626.2	616.7
40.3	356.8	422.1	3607.3	614.3
41.3	351.3	415.5	3545.2	606.6
42.3	345.9	409.0	3484.4	598.9
43.3	340.6	402.8	3425.0	591.4
44.3	335.6	396.8	3367.3	584.1
45.3	330.7	391.0	3311.1	576.9
46.1	327.0	386.6	3267.2	571.3
46.3	326.0	385.5	3256.4	569.9
47.3	321.5	380.1	3203.3	563.1
48.3	317.1	374.9	3151.7	566.5
49.3	312.9	369.8	3101.4	550.0
50.3	308.8	365.0	3052.5	543.7
51.3	304.9	360.3	3004.9	537.6
52.3	301.0	355.8	2958.5	531.6
53.3	297.3	351.4	2913.3	525.7
54.3	293.8	347.1	2869.2	520.0
55.3	290.3	343.0	2826.1	514.4
56.3	286.9	338.9	2784.1	509.0
57.3	283.6	335.1	2743.1	503.6
58.3	280.4	331.3	2703.0	498.4
59.3	242.9	286.7	2184.4	434.4
60.3	240.6	284.0	2153.4	430.1
61.3	238.3	281.3	2123.1	425.9
62.3	236.1	278.8	2093.5	421.9
63.3	234.0	276.3	2064.5	417.8
64.3	232.0	273.8	2036.1	413.9
65.3	229.9	271.4	2008.3	410.0
66.3	228.0	269.1	1981.1	406.2
67.3	458.8	543.7	349.1	256.1
68.3	468.5	555.3	353.1	262.1
69.3	460.9	546.2	349.5	257.2
70.3	452.9	536.7	345.8	252.2
71.3	445.0	527.3	342.0	247.2
72.3	437.1	517.8	338.3	242.1

IPEC00036533

Table 14.3-34
(Cont.)

DEPS Break Reflood M&E Releases (Minimum ECCS)

Time (sec)	Break Path No. 1*		Break Path No. 2†	
	(lbm/sec)	(1000 BTU/sec)	(lbm/sec)	(1000 BTU/sec)
73.3	429.2	508.3	334.6	237.2
74.3	421.9	499.6	331.2	232.6
74.7	419.0	496.2	329.9	230.8
75.3	414.7	491.1	327.9	228.2
76.3	407.6	482.6	324.6	223.8
77.3	400.7	474.3	321.4	219.5
78.3	393.8	466.2	318.3	215.3
79.3	387.1	458.1	315.2	211.2
80.3	380.5	450.2	312.2	207.2
81.3	373.9	442.5	309.2	203.2
82.3	367.5	434.8	306.3	199.4
83.3	361.3	427.4	303.5	195.6
84.3	355.1	420.1	300.7	192.0
85.3	349.1	412.9	298.0	188.4
86.3	343.3	406.0	295.4	185.0
87.3	337.6	399.2	292.9	181.6
88.3	332.0	392.5	290.4	178.4
89.4	326.0	385.4	287.7	174.9
90.3	321.3	379.8	285.7	172.2
92.3	311.2	367.8	281.2	166.3
94.3	301.7	356.5	277.1	160.9
96.3	292.7	345.9	273.2	155.8
98.3	284.4	336.0	269.6	151.2
100.3	276.6	326.7	266.2	146.8
102.3	269.3	318.1	263.1	142.8
104.3	262.6	610.1	260.3	139.1
106.3	256.3	302.6	257.6	135.6
107.8	251.9	297.4	255.8	133.2
108.3	250.5	295.8	255.2	132.5
110.3	245.1	289.4	252.9	129.6
112.3	240.2	283.6	250.9	126.9
114.3	235.7	278.2	249.0	124.5
116.3	231.5	273.3	247.3	122.3
118.3	227.7	268.8	245.7	120.3
120.3	224.3	264.7	244.3	118.5
122.3	221.1	261.0	243.0	115.3
126.3	215.7	254.5	240.8	114.0

IPEC00036534

Table 14.3-34
(Cont.)

DEPS Break Reflood M&E Releases (Minimum ECCS)

Time	Break Path No. 1		Break Path No. 2	
(sec)	(lbm/sec)	(1000 BTU/sec)	(lbm/sec)	(1000 BTU/sec)
122.3	221.1	261.0	243.0	116.8
124.3	218.3	257.6	241.8	115.3
126.3	215.7	254.5	240.8	114.0
128.3	213.3	251.7	239.8	112.7
130.2	211.3	249.4	239.0	111.7
130.3	211.2	249.2	239.0	111.6
132.3	209.3	247.0	238.2	110.7
134.3	207.6	245.0	237.5	109.8
136.3	206.1	243.2	236.9	109.0
138.3	204.8	241.6	236.4	108.3
140.3	203.6	240.2	235.9	107.7
142.3	202.6	239.0	235.5	107.2
144.3	201.7	237.9	235.1	106.7
146.3	200.9	237.0	234.8	106.3
148.3	200.2	236.2	234.5	105.9
150.3	199.6	235.5	234.3	105.6
152.3	199.1	235.0	234.1	105.4
154.3	198.7	234.5	233.9	105.1
155.3	198.5	234.2	233.8	105.0
156.3	198.4	234.1	233.7	105.0
158.3	198.1	233.7	233.6	104.8
160.3	197.9	233.5	233.5	104.7
162.3	197.8	233.3	233.5	104.6
164.3	197.7	233.2	233.4	104.5
166.3	197.6	233.2	233.4	104.5
168.3	197.6	233.2	233.4	104.5
170.3	197.7	233.2	233.4	104.5
172.3	197.8	233.3	233.4	104.6
174.3	197.9	233.4	233.4	104.6
176.3	198.0	233.6	233.5	104.6
178.3	198.2	233.8	233.5	104.7
180.3	198.4	234.1	233.6	104.8
182.1	198.6	234.3	233.7	104.9

IPEC00036535

Table 14.3-35
Containment Heat Sinks

No.	Material	Heat Transfer Area ft^2	Thickness ft
1	Paint	41302.	0.000625
	Steel		0.03125
	Concrete		1.0
2	Paint	28613.	0.000625
	Steel		0.04167
	Concrete		1.0
3	Paint	15000.	0.000625
	Concrete		1.0
4	Stainless Steel	10000.	0.03125
	Concrete		1.0
5	Paint	61000.	0.000625
	Concrete		1.0
6	Paint	68792.	0.000625
	Steel		0.0417
7	Paint	81704.	0.000625
	Steel		0.03125
8	Paint	27948.	0.000625
	Steel		0.02083
9	Paint	69800.	0.000625
	Steel		0.015625
10	Paint	3000.	0.000625
	Steel		0.01042
11	Paint	22000.	0.000625
	Steel		0.01152
12	Paint	10000.	0.000625
	Steel		0.0052

IPEC00036536

Table 14.3-36

Thermophysical Properties Of Containment Heat Sinks

Material	Thermal Conductivity (BTU/hr - ft - °F)	Volumetric Heat Capacity (BTU/ft^3 - °F)
Paint	0.2083	36.86
Steel	26.0	56.35
Stainless Steel	8.6	56.35
Concrete	0.8	28.8

IPEC00036537

Table 14.3-37

DEPS Break Principal Parameters During Reflood (Minimum ECCS)

Time (sec)	Flooding		Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	Flow Fraction	Injection			
	Temp (°F)	Rate (in/sec)					Total	Accumulator	Spill	Enthalpy (BTU/lbm)
							(lbm/sec)			
27.2	185.8	.000	.000	.00	.00	.250	.0	.0	.0	.00
28.1	184.0	21.923	.000	.77	1.04	.000	6394.7	5763.9	.0	97.24
28.2	183.6	22.509	.000	.95	1.05	.000	6374.0	5743.2	.0	97.24
28.2	183.4	22.418	.126	1.05	1.06	.225	6353.4	5722.6	.0	97.23
28.6	183.1	2.305	.095	1.31	1.49	.203	6278.0	5647.2	.0	97.20
28.8	183.1	2.493	.117	1.34	1.90	.217	6248.5	5617.7	.0	97.19
28.9	183.2	2.442	.147	1.36	2.15	.270	6209.7	5578.9	.0	97.18
29.1	183.3	2475	.186	1.40	2.56	.295	6171.6	5540.8	.0	97.17
29.8	183.5	2.375	.298	1.50	3.96	.329	6043.0	5412.1	.0	97.12
30.3	183.8	2.321	.364	1.57	5.01	.339	5946.7	5315.9	.0	97.09
34.8	185.8	2.601	.613	2.00	13.40	.359	5290.4	4659.6	.0	96.80
37.3	187.2	3.915	.675	2.24	16.11	.536	4542.5	3949.9	.0	96.56
39.3	188.3	3.760	.698	2.44	16.12	.535	4327.0	3734.5	.0	96.43
40.0	188.7	3.703	.703	2.50	16.12	.533	4268.6	3675.2	.0	96.38
46.1	192.7	3.364	.727	3.01	16.12	.518	3829.7	3228.9	.0	96.00
53.0	197.5	3.127	.735	3.51	16.12	.503	3434.8	2827.8	.0	95.58
60.6	203.0	2.748	.738	4.00	16.12	.459	2559.2	1940.3	.0	94.19
66.3	207.2	2.655	.740	4.34	16.12	.450	2374.9	1754.0	.0	93.77
67.3	208.0	4.011	.748	4.41	16.02	.599	566.8	.0	.0	78.00
68.3	209.0	4.043	.748	4.49	15.84	.600	562.2	.0	.0	78.00

Table 14.3-37
(Cont.)

DEPS Break Principal Parameters During Reflood (Minimum ECCS)

Time (sec)	Flooding		Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	Flow Fraction	Injection			
	Temp (°F)	Rate (in/sec)					Total	Accumulator	Spill	Enthalpy (BTU/lbm)
							(lbm/sec)			
69.3	210.0	3.980	.748	4.58	15.66	.600	564.3	.0	.0	78.00
74.7	215.5	3.642	.749	5.01	14.79	.595	575.5	.0	.0	78.00
82.3	223.3	3.234	.750	5.55	13.87	.587	588.1	.0	.0	78.00
89.4	230.5	2.910	.750	6.01	13.28	.579	597.2	.0	.0	78.00
98.3	238.4	2.589	.750	6.51	12.85	.568	605.5	.0	.0	78.00
107.8	246.3	2.341	.750	7.00	12.66	.556	611.3	.0	.0	78.00
120.3	252.8	2.130	.751	7.58	12.72	.544	615.8	.0	.0	78.00
130.2	257.7	2.028	.753	8.00	12.91	.537	617.7	.0	.0	78.00
144.3	263.8	1.947	.757	8.58	13.32	.531	619.1	.0	.0	78.00
155.3	267.9	1.915	.760	9.00	13.70	.529	619.6	.0	.0	78.00
170.3	272.8	1.895	.765	9.57	14.26	.529	619.7	.0	.0	78.00
182.1	276.2	1.891	.770	10.00	14.72	.530	619.6	.0	.0	78.00

Tables 14.3-38, 14.3-39, 14.3-40, 14.3-41, 14.3-42, 14.3-43, 14.3-44, 14.3-45

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Table 14.3-46

DEPS Break Post-Reflood M&E Releases (Minimum ECCS)

Time	Break Path No. 1		Break Path No. 2	
(sec)	(lbm/sec)	(1000 BTU/sec)	(lbm/sec)	(1000 BTU/sec)
182.2	263.5	324.2	367.2	149.7
187.2	262.8	323.3	368.0	149.5
192.2	262.4	322.9	368.3	149.2
197.2	261.9	322.3	368.8	149.0
202.2	260.9	321.0	369.9	148.9
207.2	260.3	320.3	370.4	148.6
212.2	260.1	320.0	370.7	148.3
217.2	264.3	325.1	366.5	150.5
222.2	263.8	324.6	366.9	150.2
227.2	263.1	323.7	367.7	150.0
232.2	262.5	322.9	368.3	149.7
237.2	262.0	322.3	368.8	149.4
242.2	264.4	321.6	369.3	149.2
247.2	260.8	320.8	370.0	148.9
252.2	259.9	319.8	370.8	148.7
257.2	259.5	319.2	371.3	148.4
262.2	258.5	318.1	372.2	148.3
267.2	257.9	317.3	372.9	148.0
272.2	257.4	316.6	373.4	147.7
282.2	255.9	314.8	374.8	147.3
287.2	255.1	313.9	375.6	147.0
292.2	254.5	313.1	376.3	146.8
297.2	253.9	312.3	376.9	146.5
302.2	93.7	115.3	537.0	188.4
434.5	93.7	115.3	537.0	188.4
434.6	93.5	114.5	537.2	183.2
437.2	93.4	114.4	537.3	183.0
1114.8	93.4	114.4	537.3	183.0
1114.9	76.5	88.0	554.3	48.2
1623.8	69.7	80.2	561.1	49.4
1623.9	69.7	80.2	208.3	48.4
3600.0	56.7	65.3	221.2	50.8
3601.0	49.3	56.7	228.7	39.7
3916.2	47.5	54.7	230.5	40.0
3916.3	47.8	55.0	100.2	17.9

IPEC00036540

Table 14.3-46
(Cont.)

DEPS Break Post-Reflood M&E Releases (Minimum ECCS)

Time	Break Path No. 1		Break Path No. 2	
(sec)	(lbm/sec)	(1000 BTU/sec)	(lbm/sec)	(1000 BTU/sec)
10,000.0	36.0	41.5	112.0	20.0
100,000.0	19.3	22.2	128.7	23.0
1,000,000.0	8.3	9.5	139.8	25.0
10,000,000.0	2.6	3.0	145.4	26.0

IPEC00036541

Table 14.3-47
DEPS Break Mass Balance (Minimum ECCS)

Time (sec)		.00	27.20	27.20	182.14	434.58	1114.8	3600.00
		Mass (thousand lbm)						
Initial	In RCS and Accumulators	732.01	732.01	732.01	732.01	732.01	732.01	732.01
Added Mass	Pumped Injection	.00	.00	.00	94.02	253.21	682.29	1552.63
	Total Added	.00	.00	.00	94.02	253.21	682.29	1552.63
Total Available		732.01	732.01	732.01	826.03	985.21	1414.29	2284.64
Distribution	Reactor Coolant	527.21	40.55	67.50	134.67	134.67	134.67	134.67
	Accumulator	204.80	159.14	132.19	.00	.00	.00	.00
	Total Contents	732.01	199.70	199.70	134.67	134.67	134.67	134.67
Effluent	Break Flow	.00	532.30	532.30	691.35	850.53	1279.61	2149.98
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	532.30	532.30	691.35	850.53	1279.61	2149.98
Total Accountable		732.01	731.99	731.99	826.01	985.20	1414.28	2284.64

IPEC00036542

Table 14.3-48

DEPS Break Energy Balance (Minimum ECCS)

Time (sec)		.00	27.20	27.20	182.14	434.58	1114.84	3600.00
		Energy (Million BTU)						
Initial Energy	In RCS, accumulators, and steam generators	775.34	775.34	775.34	775.34	775.34	775.34	775.34
Added Mass	Pumped Injection	.00	.00	.00	7.33	19.75	53.22	173.60
	Decay Heat	.00	7.65	7.65	25.09	47.82	98.19	235.35
	Heat from Secondary	.00	10.72	10.72	10.72	10.72	10.72	10.72
	Total Added	.00	18.37	18.37	43.15	78.29	162.14	419.68
Total Available		775.34	793.71	793.71	818.49	853.63	937.48	1195.02
Distribution	Reactor Coolant	305.75	9.37	12.05	36.23	36.23	36.23	36.23
	Accumulator	20.35	15.81	13.13	.00	.00	.00	.00
	Core Stored	26.87	14.68	14.68	3.95	3.78	3.55	2.71
	Primary Metal	166.23	158.03	158.03	127.92	94.29	70.05	53.31
	Secondary Metal	40.98	40.83	40.83	36.70	30.14	20.07	15.24
	Steam Generator	215.15	232.85	232.85	205.85	165.18	106.47	80.08
	Total Contents	775.34	471.57	471.57	410.65	329.61	236.37	187.56
Effluent	Break Flow	.00	321.67	321.67	400.48	516.66	698.75	1008.11
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	321.67	321.67	400.48	516.66	698.75	1008.11
Total Accountable		775.34	793.23	793.23	811.13	846.27	935.11	1195.67

Table 14.3-49

DEPS Break Blowdown M&E Releases (Maximum ECCS)

Time	Break Path No. 1*		Break Path No. 2†	
(Sec)	(lbm/sec)	(1000 BTU/sec)	(lbm/sec)	(1000 BTU/sec)
0.0	0.0	0.0.	0.0.	0.0.
0.0	81761.8	44233.2	40576.0	21912.8
0.1	40368.4	21885.1	19793.9	10677.7
0.2	45306.7	24772.6	22451.3	12124.9
0.3	45415.9	25096.5	23512.5	12704.7
0.4	45055.9	25215.1	23517.2	12711.4
0.5	44009.1	24945.6	22969.5	12420.5
0.6	44256.3	25378.4	22403.5	12120.2
0.7	43600.3	25252.0	22049.2	11934.2
0.8	42230.5	24671.9	21887.5	11851.0
0.9	40998.0	24158.5	21772.1	11792.1
1.0	39954.5	23761.7	21676.2	11742.9
1.1	38791.7	23329.2	21567.5	11686.1
1.2	37322.5	22738.0	21464.4	11631.9
1.3	35681.4	22020.7	21381.4	11588.1
1.4	34261.3	21372.7	21325.8	11558.9
1.5	33185.6	20878.9	21311.8	11552.2
1.6	32347.7	20502.1	21335.1	11565.8
1.7	31543.5	20137.7	21256.7	11523.4
1.8	30691.4	19737.1	21078.4	11426.5
1.9	29771.4	19286.1	20897.8	11328.4
2.0	28806.8	18796.0	20735.0	11240.3
2.1	27794.9	18268.3	20582.8	11158.0
2.2	26813.6	17759.7	20406.0	11062.5
2.3	25407.1	16959.1	20198.9	10950.3
2.4	23314.1	15671.6	19979.5	10831.5
2.5	21428.3	14504.8	19781.6	10724.5
2.6	21061.3	14354.1	19588.9	10620.6
2.7	20375.5	13940.6	19405.6	10521.9
2.8	19647.9	13490.3	19200.4	10411.3
2.9	19327.9	13315.2	19003.6	10305.3
3.0	18990.9	13110.6	18809.5	10201.0
3.1	18936.7	13103.3	18599.8	10088.1
3.2	18708.8	12965.8	18359.2	9958.3
3.3	18376.1	12765.8	18107.3	9822.4

*mass and energy exiting the steam generator side of the break
†mass and energy exiting the pump side of the break

IPEC00036544

Table 14.3-49
(Cont.)

DEPS Break Blowdown M&E Releases (Maximum ECCS)

Time	Break Path No. 1*		Break Path No. 2†	
(sec)	(lbm/sec)	(1000 BTU/sec)	(lbm/sec)	(1000 BTU/sec)
3.4	18036.8	12557.7	17873.1	9696.2
3.5	17593.9	12267.4	17641.1	9571.4
3.6	17069.3	11917.1	17408.8	9446.3
3.7	16489.9	11528.9	17177.4	9321.8
3.8	15903.3	11135.1	16954.4	9202.0
3.9	15363.2	10772.2	16746.0	9090.1
4.0	14881.1	10447.4	16551.1	8985.8
4.2	14051.8	9886.3	16181.4	8787.9
4.4	13368.7	9425.1	15844.1	8607.7
4.6	12849.6	9066.2	15533.4	8442.0
4.8	12412.2	8757.6	15247.7	8289.7
5.0	12013.1	8465.4	14999.3	8157.8
5.2	11703.6	8224.2	14763.6	8032.5
5.4	11586.3	8094.8	14554.2	7921.5
5.6	11514.6	7994.2	14351.2	7813.7
5.8	11482.1	7922.8	14628.1	7971.2
6.0	11522.2	7898.5	14747.6	8037.3
6.2	12194.7	8296.4	14583.2	7949.9
6.4	12141.3	8354.3	14758.0	8049.7
6.6	10307.9	7832.3	14611.9	7971.5
6.8	9121.4	7303.9	14448.4	7884.8
7.0	9073.9	7243.1	14317.3	7815.8
7.2	9131.3	7217.0	14150.0	7726.8
7.4	9251.8	7207.9	13997.8	7646.3
7.6	9481.4	7234.9	13888.3	7588.7
7.8	9840.4	7325.9	13772.0	7525.2
8.0	10359.8	7518.4	13588.0	7423.3
8.2	11037.7	7812.7	13409.2	7324.2
8.4	11803.6	8167.7	13241.4	7231.2
8.6	12593.0	8547.8	13064.4	7133.0
8.8	13245.7	8851.5	12873.9	7027.5
9.0	13483.0	8908.3	12687.9	6924.7
9.2	13276.1	8712.1	12523.4	6834.0
9.4	12965.9	8476.1	12374.3	6751.5
9.6	12574.6	8194.2	12222.6	6667.2
9.8	11645.6	7582.1	12081.1	6588.6

*mass and energy exiting the steam generator side of the break
†mass and energy exiting the pump side of the break

IPEC00036545

Table 14.3-49
(Cont.)

DEPS Break Blowdown M&E Releases (Maximum ECCS)

Time	Break Path No. 1*		Break Path No. 2†	
(sec)	(lbm/sec)	(1000 BTU/sec)	(lbm/sec)	(1000 BTU/sec)
9.8	11645.6	7582.1	12081.1	6588.6
10.0	10593.6	6936.8	12004.7	6546.0
10.2	10208.2	6743.1	11946.2	6512.7
10.4	9933.1	6606.9	11771.6	6414.9
10.6	9558.0	6412.6	11662.9	6354.9
10.8	9363.2	6333.5	11584.3	6312.0
11.0	9110.8	6193.7	11405.8	6213.9
11.2	8843.7	6050.6	11321.2	6168.3
11.4	8611.3	5935.8	11207.4	6105.7
11.6	8331.5	5791.9	11060.1	6024.8
11.8	8094.4	5682.6	10972.5	5976.9
12.0	7841.1	5558.8	10793.8	5878.6
12.2	7631.1	5453.0	10669.0	5810.9
12.4	7447.7	5345.9	10550.1	5746.4
12.6	7300.3	5249.1	10397.4	5663.0
12.8	7176.9	5155.0	10272.0	5594.7
13.0	7066.5	5061.5	10136.2	5520.5
13.2	6964.1	4969.2	10003.1	5447.9
13.4	6860.9	4873.9	9867.7	5374.1
13.6	6756.5	4776.9	9730.9	5299.6
13.8	6652.7	4679.5	9599.0	5227.9
14.0	6551.1	4582.4	9463.3	5154.1
14.2	6454.6	4487.7	9332.5	5083.2
14.4	6364.5	4396.4	9204.0	5013.5
14.6	6286.8	4312.4	9088.1	4950.7
14.8	6229.3	4241.5	8991.4	4898.8
15.0	6169.6	4171.4	8866.3	4830.2
15.2	6106.1	4106.5	8779.1	4783.6
15.4	6042.3	4043.3	8675.6	4727.6
15.6	5976.8	3981.7	8590.2	4682.1
15.8	5911.6	3925.6	8496.0	4631.6
16.0	5840.4	3870.9	8414.4	4588.8
16.2	4773.5	3823.7	8339.7	4550.2
16.4	5698.8	3775.3	8193.0	4472.2
16.6	5626.7	3739.9	8057.1	4403.0

*mass and energy exiting the steam generator side of the break
†mass and energy exiting the pump side of the break

IPEC00036546

Table 14.3-49
(Cont.)

DEPS Break Blowdown M&E Releases (Maximum ECCS)

Time	Break Path No. 1*		Break Path No. 2†	
(sec)	(lbm/sec)	(1000 BTU/sec)	(lbm/sec)	(1000 BTU/sec)
16.8	5544.7	3720.5	7908.6	4327.8
17.0	5421.2	3692.9	7741.4	4242.5
17.2	5275.7	3658.2	7585.2	4161.0
17.4	5130.3	3623.8	7425.0	4062.6
17.6	4987.5	3591.5	7279.3	3956.0
17.8	4848.1	3560.2	7149.2	3845.8
18.0	4712.0	3529.8	7046.1	3743.0
18.2	4578.9	3500.4	6951.5	3642.4
18.4	4447.5	3473.6	6860.8	3545.2
18.6	4317.2	3448.0	6736.5	3434.4
18.8	4184.1	3423.5	6572.9	3308.7
19.0	4049.4	3401.2	6399.0	3183.7
19.2	3911.0	3380.1	6214.5	3060.8
19.4	3769.8	3361.2	6034.1	2949.3
19.6	3623.5	3344.6	5864.7	2853.8
19.8	3472.5	3330.4	5695.8	2768.6
20.0	3285.7	3291.4	5492.9	2673.1
20.2	3023.0	3191.9	5043.3	2439.6
20.4	2764.3	3072.7	4849.5	2304.6
20.6	2543.6	2957.3	4656.1	2205.9
20.8	2393.5	2873.4	4365.2	2058.6
21.0	2180.7	2661.6	4194.5	1968.7
21.2	2028.5	2492.3	3842.0	1786.4
21.4	1885.9	2325.5	3636.1	1644.4
21.6	1764.8	2182.2	3508.4	1551.5
21.8	1659.9	2056.6	3113.1	1339.6
22.0	1549.0	1922.8	2768.0	1141.4
22.2	1453.0	1806.5	2485.3	983.1
22.4	1366.4	1701.7	2269.9	866.9
22.6	1282.0	1598.2	2098.7	777.9
22.8	1194.9	1492.1	2020.9	728.2
23.0	1119.7	1399.9	2062.5	723.2
23.2	1045.0	1307.9	2203.1	754.1
23.4	958.1	1200.9	2404.8	807.2
23.6	868.4	1089.6	2598.9	859.2

*mass and energy exiting the steam generator side of the break
†mass and energy exiting the pump side of the break

IPEC00036547

Table 14.3-49
(Cont.)

DEPS Break Blowdown M&E Releases (Maximum ECCS)

Time	Break Path No. 1*		Break Path No. 2†	
(sec)	(lbm/sec)	(1000 BTU/sec)	(lbm/sec)	(1000 BTU/sec)
23.8	786.1	987.3	2745.0	896.0
24.0	701.2	881.4	2904.0	935.2
24.2	614.4	772.9	3064.9	971.6
24.4	528.0	664.7	3199.8	997.3
24.6	446.8	562.9	3189.0	977.5
24.8	370.2	466.8	2986.6	903.0
25.0	301.9	380.8	2789.6	834.7
25.2	239.4	302.3	2590.3	768.5
25.4	183.9	232.4	2383.7	702.3
25.6	142.6	180.4	2177.3	637.8
25.8	127.2	161.1	1969.4	574.5
26.0	105.9	134.2	1761.3	512.4
26.2	58.4	74.2	1556.4	452.4
26.4	0.0	0.0	1339.4	389.5
26.6	0.0	0.0	1082.8	315.5
26.8	0.0	0.0	724.1	211.6
27.0	0.0	0.0	25.5	7.5
27.2	0.0	0.0	0.0	0.0

*mass and energy exiting the steam generator side of the break
†mass and energy exiting the pump side of the break

IPEC00036548

Table 14.3-50

DEPS Break Reflood M&E Releases (Maximum ECCS)

Time	Break Path No. 1		Break Path No. 2	
(sec)	(lbm/sec)	(1000 BTU/sec)	(lbm/sec)	(1000 BTU/sec)
27.2	0.0	0.0	0.0	0.0
27.8	0.0	0.0	0.0	0.0.
27.9	0.0	0.0	0.0	0.0
28.1	0.0	0.0	241.1	18.8
28.2	0.0	0.0	241.1	18.8
28.3	74.4	87.6	241.1	18.8
28.4	22.8	26.8	241.1	18.8
28.5	15.2	17.9	241.1	18.8
28.6	17.3	20.3	241.1	18.8
28.7	22.7	26.7	241.1	18.8
28.8	27.1	31.9	241.1	18.8
28.9	31.9	37.5	241.1	18.8
29.0	37.6	44.3	241.1	18.8
29.1	42.3	49.8	241.1	18.8
29.2	46.5	54.8	241.1	18.8
29.3	50.7	59.7	241.1	18.8
29.5	54.1	63.8	241.1	18.8
29.6	57.9	68.2	241.1	18.8
29.7	61.0	71.9	241.1	18.8
29.8	63.9	75.3	241.1	18.8
29.9	66.7	78.5	241.1	18.8
30.0	69.4	81.7	241.1	18.8
30.1	72.0	84.8	241.1	18.8
30.2	74.5	87.8	241.1	18.8
30.3	77.0	90.8	241.1	18.8
31.3	99.4	117.1	241.1	18.8
32.3	117.4	138.4	241.1	18.8
33.3	133.1	156.9	241.1	18.8
34.3	147.3	173.7	241.1	18.8
34.6	151.0	178.0	241.1	18.8
35.3	159.9	188.6	241.1	18.8
36.6	357.7	423.0	3654.2	570.9
37.9	393.9	466.2	4034.5	656.1
38.3	390.1	461.7	3992.6	653.5
39.3	384.2	454.7	3930.0	646.0
39.6	382.5	452.6	3911.0	643.6

IPEC00036549

IPEC00036549

Table 14.3-50
(Cont.)

DEPS Break Reflood M&E Releases (Maximum ECCS)

Time	Break Path No. 1		Break Path No. 2	
(sec)	(lbm/sec)	(1000 BTU/sec)	(lbm/sec)	(1000 BTU/sec)
39.6	382.5	452.6	3911.0	643.6
40.3	378.4	447.8	3866.8	638.1
41.3	372.7	441.0	3804.6	630.3
42.3	367.2	434.4	3743.9	622.6
43.3	361.9	428.1	3684.7	615.1
44.3	356.7	422.0	3627.1	607.7
45.3	351.8	416.0	3571.2	600.6
45.4	351.3	415.5	3565.7	599.9
46.3	347.0	410.3	3516.8	593.6
47.3	342.3	404.8	3463.9	586.8
48.3	337.9	399.5	3412.5	580.2
49.3	333.6	394.4	3362.5	573.8
50.3	329.4	389.4	3313.9	567.5
51.3	325.4	384.6	3266.6	561.4
52.0	322.6	381.4	3234.2	557.3
52.3	321.4	380.0	3220.5	555.5
53.3	317.7	375.5	3175.6	538.5
54.3	314.0	371.1	3131.8	544.0
55.3	310.4	366.9	3089.0	538.5
56.3	307.0	362.8	3047.4	533.0
57.3	303.6	358.8	3006.6	527.7
58.3	300.3	354.9	2966.9	522.6
59.3	285.7	337.7	2725.6	502.8
59.3	266.9	315.4	2402.0	470.9
60.3	259.8	306.8	2436.2	457.3
61.3	257.4	304.0	2406.0	453.2
62.3	255.2	301.3	2376.5	499.2
63.3	252.9	298.7	2347.6	445.2
64.3	250.8	296.1	2319.4	441.3
65.3	248.7	293.6	2291.7	437.5
66.3	246.6	291.1	2264.6	433.7
67.3	244.6	288.7	2238.1	430.1
68.3	363.1	429.6	374.8	189.2
69.3	363.0	429.4	375.5	189.1
70.3	362.8	429.1	376.5	188.9
71.3	362.5	428.9	377.5	188.7
72.3	362.3	428.5	378.6	188.5

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Table 14.3-50
(Cont.)

DEPS Break Reflood M&E Releases (Maximum ECCS)

Time	Break Path No. 1		Break Path No. 2	
(sec)	(lbm/sec)	(1000 BTU/sec)	(lbm/sec)	(1000 BTU/sec)
73.3	362.0	428.2	379.6	188.4
74.3	361.7	427.9	380.7	188.2
74.5	361.6	427.8	380.9	188.1
75.3	361.4	427.5	381.8	188.0
76.3	361.0	427.1	382.9	187.8
77.9	360.7	426.7	384.0	187.6
78.3	360.3	426.2	385.2	187.4
79.3	359.9	425.7	386.4	187.1
80.3	359.5	425.2	387.7	186.9
81.3	359.0	424.7	389.0	186.7
82.3	358.5	424.1	390.3	186.5
83.3	358.0	423.5	391.7	186.2
84.3	357.4	422.8	393.2	186.0
85.3	356.8	422.1	394.7	185.8
86.3	356.2	421.4	396.2	185.5
87.3	355.6	420.6	397.8	185.2
88.3	354.9	419.8	399.5	185.0
88.9	354.4	419.2	400.5	184.8
90.3	353.4	418.0	403.0	184.5
92.3	351.7	416.0	406.7	183.9
94.3	349.9	413.9	410.6	183.3
96.3	348.0	411.5	414.7	182.8
98.3	345.9	409.0	419.0	182.2
100.3	343.6	406.3	423.5	181.6
102.3	341.2	403.4	428.2	181.0
104.3	338.6	400.4	433.1	180.5
104.4	338.5	400.2	433.3	180.5
106.3	335.9	397.2	438.1	179.9
108.3	333.1	393.8	443.3	179.4
110.3	330.1	390.3	448.7	178.9
112.3	327.0	386.6	454.3	178.4
114.3	323.8	382.8	460.0	177.9
116.3	320.4	378.8	465.9	177.5
118.3	316.9	374.6	472.0	177.1
120.3	313.3	370.3	478.3	176.7
121.3	311.4	368.1	481.5	176.6
122.3	309.5	365.8	484.7	176.4

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Table 14.3-50
(Cont.)

DEPS Break Reflood M&E Releases (Maximum ECCS)

Time	Break Path No. 1		Break Path No. 2	
(sec)	(lbm/sec)	(1000 BTU/sec)	(lbm/sec)	(1000 BTU/sec)
124.3	305.6	361.1	491.3	176.1
126.3	301.5	356.3	498.2	175.9
128.3	297.3	351.3	505.2	175.7
130.3	292.9	346.1	512.4	175.6
132.3	288.3	340.7	519.8	175.5
134.3	283.6	335.0	527.5	175.5
136.3	278.7	329.2	535.4	175.5
138.3	273.6	323.1	543.5	175.6
140.3	268.3	316.9	551.9	175.8
142.3	262.8	310.3	560.6	176.0
144.3	257.0	303.5	569.6	176.4
146.3	251.0	296.4	578.8	176.8
150.3	238.3	281.3	598.3	177.9
152.3	231.4	273.2	608.7	178.6
154.3	224.2	264.7	619.5	179.4
156.3	216.7	255.7	630.7	180.4
158.3	208.7	246.3	642.5	181.5
160.3	200.4	236.4	654.7	182.8
162.3	191.5	225.9	667.6	184.2
163.7	185.0	218.2	677.0	185.3

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Table 14.3-51

DEPS Break Principal Parameters During Reflood (Maximum ECCS)

Time (sec)	Flooding		Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	Flow Fraction	Injection			
	Temp (°F)	Rate (in/sec)					Total	Accumulator	Spill	Enthalpy (BTU/lbm)
							(lbm/sec)			
27.2	185.5	.000	.000	.00	.00	.250	.0	.0	.0	.00
28.1	183.5	23.012	.000	.78	1.06	.000	6770.0	5805.3	.0	96.31
28.2	18.9	24.009	.000	1.08	1.07	.000	6728.0	5763.3	.0	96.29
28.5	182.6	2.385	.100	1.31	1.54	.235	6653.8	5689.1	.0	96.25
29.0	182.8	2.522	.190	1.40	2.60	.306	6550.3	5585.6	.0	96.21
29.7	183.0	2.420	.293	1.50	3.98	.333.	6428.3	5463.6	.0	96.15
30.3	183.2	2.360	.369	1.58	5.23	.342	6321.1	5356.4	.0	96.09
34.6	185.2	2.652	.614	2.00	13.80	.360	5670.4	4705.8	.0	95.72
37.3	186.5	4.078	.680	2.27	16.12	.550	4803.5	3892.8	.0	95.30
39.3	187.6	3.880	.701	2.47	16.12	.547	4615.3	3702.1	.0	95.13
39.6	187.8	3.854.	.704	2.50	16.12	.546	4590.1	3676.5	.0	95.10
45.4	191.4	3.513	.727	3.00	16.12	.533	4164.4	3242.1	.0	94.62
52.0	195.9	3.272	.736	3.50	16.12	.467	2656.4	1708.4	.0	91.73
67.3	206.8	2.742	.741	4.49	16.12	.548	918.5	.0	.0	78.00
68.3	207.6	3.443	.747	4.56	16.12	.548	918.5	.0	.0	78.00
69.3	208.4	3.437	.747	4.63	16.12	.548	918.5	.0	.0	78.00
74.5	213.2	3.406	.749	5.01	16.12	.550	919.0	.0	.0	78.00
82.3	221.0	3.351	.752	5.55	16.12	.552	919.9	.0	.0	78.00
88.9	228.0	3.292	.755	6.00	16.12	.533	922.9	.0	.0	78.00

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Table 14.3-51
(Cont.)

DEPS Break Principal Parameters During Reflood (Maximum ECCS)

Time (sec)	Flooding		Carryover Fraction	Core Height (ft)	Downcomer Height (ft)	Flow Fraction	Injection			
	Temp (°F)	Rate (in/sec)					Total	Accumulator	Spill	Enthalpy (BTU/lbm)
							(lbm/sec)			
98.3	237.6	3.188	.759	6.62	16.12	.555	922.9	.0	.0	78.00
104.4	243.1	3.110	.760	7.01	16.12	.555	924.6	.0	.0	78.00
114.3	250.7	2.967	.763	7.60	16.12	.554	927.8	.0	.0	78.00
121.3	255.4	2.856	.765	8.00	16.12	.553	930.5	.0	.0	78.00
132.3	261.7	2.665	.768	8.60	16.12	.547	935.4	.0	.0	78.00
140.5	265.6	2.507	.769	9.00	16.12	.538	939.6	.0	.0	78.00
152.3	270.4	2.250	.771	9.54	16.12	.517	946.7	.0	.0	78.00
163.7	274.2	1.950	.771	10.00	16.12	.476	954.8	.0	.0	78.00

IPEC00036554

Table 14.3-52

DEPS Break Post-Reflood M&E Releases (Maximum ECCS)

Time	Break Path No. 1		Break Path No. 2	
(sec)	(lbm/sec)	(1000 BTU/sec)	(sec)	(lbm/sec)
163.8	158.0	193.6	806.7	207.2
168.8	157.6	193.1	807.1	206.8
173.8	157.7	193.2	807.0	206.4
178.8	157.7	193.2	807.0	206.0
183.8	157.2	192.7	807.5	205.7
188.8	157.2	192.6	807.5	205.7
193.8	157.2	192.6	807.5	204.8
198.8	156.7	192.0	808.0	204.5
203.8	156.8	192.2	807.8	204.1
208.8	156.6	191.9	808.1	203.7
213.8	156.8	192.1	807.9	206.8
218.8	156.5	191.8	808.2	206.5
223.8	156.2	191.4	808.5	206.1
228.8	156.3	191.6	808.4	205.6
233.8	156.0	191.2	808.7	205.3
238.8	156.1	191.3	808.6	204.8
243.8	156.2	191.4	808.5	204.3
248.8	155.8	190.9	808.9	204.0
253.8	155.8	190.9	808.9	203.5
258.8	155.8	190.9	808.9	203.1
263.8	155.8	190.9	808.9	202.6
268.8	155.7	190.8	809.0	202.2
273.8	155.6	190.7	809.1	201.8
278.8	155.5	190.6	809.2	201.3
283.8	155.4	190.4	809.3	200.9
288.8	155.2	190.2	809.5	200.5
293.8	155.0	189.9	809.7	200.1
298.8	155.2	190.1	809.5	203.0
303.8	154.9	189.8	809.8	202.6
308.8	154.9	189.8	809.8	202.1
313.8	154.6	189.4	810.1	201.7
318.8	154.5	189.4	810.2	201.2
232.8	454.4	189.2	810.2	200.8
328.8	154.3	189.1	810.4	200.3
333.8	154.4	189.2	810.3	199.8
338.8	154.1	188.9	810.5	199.3
343.8	154.1	188.9	810.6	198.8

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Table 14.3-52
(Cont.)

DEPS Break Post-Reflood M&E Releases (Maximum ECCS)

Time	Break Path No. 1		Break Path No. 2	
(sec)	(lbm/sec)	(1000 BTU/sec)	(lbm/sec)	(1000 BTU/sec)
343.8	154.1	188.9	810.6	198.8
348.8	154.0	188.7	810.7	198.4
353.8	153.8	188.5	810.9	197.9
358.8	153.9	188.5	810.8	197.4
363.8	153.8	188.4	810.9	196.9
373.8	153.4	188.0	811.3	199.2
378.8	153.5	188.0	811.2	198.7
383.8	153.3	187.8	811.4	198.2
388.8	153.1	187.6	811.6	197.7
393.8	153.2	187.7	811.5	197.1
398.8	152.9	187.4	811.8	196.7
403.8	153.0	187.5	811.7	196.1
408.8	152.8	187.2	811.9	195.6
413.8	152.9	187.3	811.8	195.1
418.8	152.8	187.2	811.9	197.7
423.8	152.6	187.0	812.0	197.2
428.8	152.5	186.9	812.2	196.6
433.8	152.4	186.7	812.3	196.1
438.8	144.5	177.1	820.2	197.6
443.8	86.1	105.5	878.6	212.4
798.1	86.1	105.5	878.6	212.4
798.2	83.7	102.0	881.0	204.6
798.8	83.7	102.0	881.0	204.6
1033.6	83.7	102.0	881.0	204.6
1033.7	78.5	90.4	896.1	73.4
1172.7	76.6	88.1	888.1	73.8
1172.8	76.6	88.1	474.6	105.6
3119.9	60.4	69.5	233.4	108.5
3120.0	60.4	69.5	233.4	59.3
3600.0	57.7	66.3	236.1	59.7
3600.1	50.5	58.1	243.3	47.9
10000.0	36.7	42.2	257.1	50.6
100000.0	19.6	22.6	274.1	53.9
1000000.0	8.4	9.7	285.4	56.1
10000000.0	2.6	3.0	291.1	57.3

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Table 14.3-53

DEPS Break Mass Balance (Maximum ECCS)

Time (Sec)		.00	27.20	27.20	163.73	798.20	1033.63	3600.00
		Mass (Thousands lbm)						
Initial	In RCS and Accumulators	732.01	732.01	732.01	732.01	732.01	732.01	732.01
Added Mass	Pumped Injection	.00	.00	.00	126.74	738.74	965.85	2314.17
	Total Added	.00	.00	.00	126.74	738.74	965.85	2314.17
***Total Available ***		732.01	732.01	732.01	858.74	1470.74	1697.86	3046.18
Distribution	Reactor Coolant	527.21	40.55	66.31	137.06	137.06	137.06	137.06
	Accumulator	204.80	159.14	133.38	.00	.00	.00	.00
	Total Contents	732.01	199.70	199.70	137.06	137.06	137.06	137.06
Effluent	Break Flow	.00	532.30	532.30	721.67	1333.67	1560.78	2909.14
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	532.30	532.30	721.67	1333.67	1560.78	2909.14
***Total Accountable ***		732.01	731.99	731.99	858.73	1470.73	1697.84	3046.20

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Table 14.3-54

DEPS Break Energy Balance (Maximum ECCS)

Time (sec)		.00	27.20	27.20	163.73	798.20	1033.63	3600.00
		Energy (Million Btu)						
Initial Energy	In RCS, Accumulators and steam generators	775.34	775.34	775.34	775.34	775.34	775.34	775.34
Added Energy	Pumped Injection	.00	.00	.00	9.89	57.62	75.34	321.96
	Decay Heat	.00	7.65	7.65	23.26	76.05	92.67	235.26
	Heat from Secondary	.00	10.72	10.72	10.72	10.72	10.72	10.72
	Total Added	.00	18.37	18.37	43.87	144.40	178.73	567.95
Total Available		775.34	793.71	793.71	819.21	919.74	954.07	1343.29
Distribution	Reactor Coolant	305.75	9.37	11.93	37.02	37.02	37.02	37.02
	Accumulator	20.35	15.81	13.25	.00	.00	.00	.00
	Core Stored	26.87	14.68	14.68	3.95	3.75	3.69	2.71
	Primary Metal	166.23	158.03	158.03	127.18	79.50	71.49	53.32
	Secondary Metal	40.98	40.83	40.83	36.27	23.54	20.26	15.22
	Steam Generator	215.15	232.85	232.85	203.11	125.58	107.48	80.01
	Total Contents	775.34	471.57	471.57	407.52	269.40	239.93	188.27
Effluent	Break Flow	.00	321.67	321.67	404.33	642.97	698.44	1144.08
	ECCS Spill	.00	.00	.00	.00	.00	.00	.00
	Total Effluent	.00	321.67	321.67	404.33	642.97	698.44	1144.08
***Total Accountable ***		775.34	793.23	793.23	811.85	912.38	939.37	1332.35

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Table 14.3-55

LOCA Containment Response Analysis Parameters

SW Temperature (°F)	95
RWST Water Temperature (°F)	110
Initial Containment Temperature (°F)	130
Initial Containment Pressure (psia)	17.2
Initial Relative Humidity	20
Net-Free Volume (ft^3)	2.61×10^6
Reactor Containment Fan Coolers	
Total	5
Minimum ECCS	4
Maximum ECCS	5
Fan Cooler Initiation Setpoint (psig)	5.12
Delay Time(sec)	48.21
Containment Spray Pumps	
Total	2
Minimum ECCS	1
Maximum ECCS	1
Flowrate (gpm) Injection Phase Recirculation Phase	 See Table 14.3-57 970
Containment High setpoint (psig)	24.63
Delay Time (sec)	60
ECCS Recircluation Switchover (sec) Minimum ECCS Maximum ECCS	 1623.4 1172.7
Containment Spray Termination (sec) Minimum ECCS Maximum ECCS	 3355 3119.9
ECCS Flow Rates	
Minimum ECCS	
Injection Alignment (gpm)	2871.2
Recirculation Alignment (gpm)	1864.0
Maximum ECCS	
Injection Alignment (gpm)	5394.5
Recirculation Alignment (gpm)	6320.5
Residual Heat Removal System	
RHR Heat Exchangers	
Total	2
Minimum ECCS	1
Maximum ECCS	2
UA (million BTU/hr °F Hx)	0.62
CCW Flow Through RHR Heat Exchanger (gpm/Hx)	1096
CCW Heat Exchangers	

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Table 14.3-56

Containment Fan Cooler Performance

Containment Temperature (°F)	Heat Removal Rate (BTU/sec/RCFC)
110	674.2
130	1737.0
150	2921.4
170	4162.4
190	5424.6
210	6684.4
230	8836.1
250	10986.4
271	13042.3

Table 14.3-57

Containment Spray Performance

Containment Pressure (psig)	Containment Spray Flow Rate (gpm/pump)
Values for LOCA	
0	2750.8
10	2656.8
20	2558.0
25	2507.4
35	2403.8
45	2296.5
50	2237.9
Values for MSLB	
5.0	2409.7
10.0	2367.5
20.0	2280.9
30.0	2187.5
35.0	2139.3
40.0	2090.1
45.0	2040.1
50.0	1988.9

IPEC00036560

Table 14.3-58

DEPS Break Sequence of Events (Minimum ECCS)

Time (sec)	Event Description
0.0	Break occurs, reactor trip and LOOP power are assumed
0.66	Reactor trip on low pressurizer pressure of 1748.7 psia
1	Fan cooler initiation pressure setpoint reached
4	Low-pressurizer pressure SI setpoint 1648.7 psia reached in blowdown
8	Containment spray initiation pressure setpoint reached
16	Main Feedwater Flow Control Valve closed
16.9	Broken-loop accumulator begins injecting water
17.5	Intact-loop accumulator begins injecting water
27.2	End of Blowdown Phase
27.8	SI begins
48.74	RCFC's actuate
58.4	Broken-loop accumulator water injection ends
66.5	Intact-loop accumulator water injection ends
67.81	Containment spray pump starts
182.1	End of reflood
1118	Peak pressure and temperature occur
1623.8	RHR/HHSI alignment for recirculation
3355	Containment spray is terminated
23400	Hot leg recirculation
1.0E+07	Transient Modeling Terminated

Table 14.3-59
DEPS Break Sequence of Events (Maximum ECCS)

Time (sec)	Event Description
0.0	Break Occurs, Reactor Trip and LOOP power are assumed
0.66	Reactor trip on low pressurizer pressure of 1748.7 psia
1	Fan cooler initiation pressure setpoint reached
4	Low-pressurizer pressure SI setpoint 1648.7 psia reached in blowdown
8	Containment spray initiation pressure setpoint reached
16	Main Feedwater Flow Control Valve closed
16.9	Broken-loop accumulator begins injecting water
17.5	Intact-loop accumulator begins injecting water
27.2	End-of-blowdown phase
27.3	Peak pressure and temperature occur
27.8	SI begins
48.74	RCFCs actuate
59.2	Broken-loop accumulator water injection ends
67.4	Intact-loop accumulator water inject ends
67.81	Containment Spray Pump starts
163.7	End of reflood
1172.7	RHR/HHSI alignment for recirculation
3119.9	Containment spray terminated
1.0E+07	Transient modeling is terminated

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Table 14.3-60

DEHL Break Sequence of Events

Time (sec)	Event Description
0.0	Break Occurs, Reactor Trip and LOOP are assumed
0.6	Reactor trip on low pressurizer pressure of 1748.7 psia
1	Fan cooler initiation pressure setpoint reached
4	Low-presurizer pressure SI setpoint = 1695 psia reached
8	Containment spray initiation pressure setpoint reached
15.2	Broken-loop accumulator begins injecting water
15.5	Intact-loop accumulator begins injecting water
24.2	Peak pressure and temperature occur
25.6	End-of-blowdown phase
25.6	Transient modeling terminated

Table 14.3-61

Deleted

Table 14.3-62

LOCA Containment Response Results (Loss-of-Offsite-Power Assumed)

Case	Peak Pressure (psig)	Peak Steam Temperature (°F)	Pressure at 24 hours (psig)	Steam Temperature at 24 hours (°F)
DEPS Minimum ECCS	42.00 at 1118 sec	260.4 at 1118 sec	13.27	187.8
DEPS Maximum ECCS	38.94 at 23.7 sec	256.2 at 23.7 sec	12.40	183.6
DEHL	40.38 at 24.2 sec	258.6 at 24.2 sec	N/A	N/A

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Table 14.3-63

Post-Accident Containment Temperature Transient
Used In The Calculation Of Aluminum Corrosion

Time Internal (sec)	Water Temperature (°F)
0 – 8	230
8 – 3500	258
3500 – 20,000	228
20,000 – 100,000	220
100,000 – 200,000	195
200,000 – 400,000	185
400,000 – 600,000	175
600,000 – 800,000	165
800,000 – 1,200,000	153
1,200,000 – 3,000,000	140
3,000,000 – 5,000,000	120
5,000,000 – 8,000,000	115
8,000,000 – 8,640,000	110

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Table 14.3-64

Parameters Used To Determine Hydrogen Generation

Core Thermal Power Rating(1)	3281 MWt
Containment Free Volume	2,610,000 ft^3
Containment Temperature at Accident Initiation	130°F
Fuel Cladding Mass Undergoing Zirc-Water Reaction	5.0%
Total Mass of Zirc in the Core	41,002 lbs
RCS Hydrogen Concentration during Normal Operation	50 cc/kg
RCS Mass (normal pressurizer level)	518,182 lbs
Pressurizer Volume	1834.4ft^3
Pressurizer Level (normal operation)	50%
Hydrogen Recombiner Flow Rate	100 scfm

(1) 3216 MWt multiplied by 1.02 to account for source uncertainties.

Inventory of Aluminum Inside the Containment Building		
Item Description	**Weight (lbs)**	**Area (ft^2)**
UFSAR Aluminum Sources		
Source, Intermediate, and Power Range Dectors	472	338
Process Instrumentation and Control Equipment	159	31
Paint	58	7480
Valve Parts inside Containment	230	86
Reactor Vessel Foil	269	10000
Flux Mapping Drive System	1950	335
Reactor Coolant Pump Motor Parts	125	12.8
Other Sources Included in Analysis		
CRDM Cooling Fan Blades	800	131.6
RCP conduit boxes	7.2	4
Rod Position Indicators	10.6	3.7
Others (filters, etc.)	25	25
Total Aluminum	4105.8	18447.1

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Table 14.3-65

Fission Product Decay Energy in the Core

Time After LOCA Days	Energy Release Rate Watts / MWt	Integrated Energy Release Watt-sec MWt
1	5.11E+03	6.01E+08
5	3.41E+03	1.97E+09
10	2.72E+03	3.28E+09
15	2.29E+03	4.36E+09
20	2.00E+03	5.28E+09
25	1.80E+03	6.10E+09
30	1.66E+03	6.84E+09
40	1.47E+03	8.19E+09
50	1.33E+03	9.39E+09
60	1.21E+03	1.05E+10
70	1.12E+03	1.15E+10
80	1.02E+03	1.24E+10
90	9.43 E+02	1.33E+10
100	8.68E+02	1.40E+10

** Considers 50 percent of core halogens, no noble gases and 99 percent of other fission products in the core
n.nnE+yy denotes n.nn x 10^{yy}

Table 14.3-66
FISSION PRODUCT DECAY DEPOSITION IN SUMP SOLUTION

Time After LOCA Days	Sump Fission Product Energy*	
	Energy Release Rate Watts / MWt	Integrated Energy Release Watt-sec / MWt
1	2.56E+02	4.62E+07
5	8.17E+01	8.83E+07
10	5.35E+01	1.17E+08
15	3.80E+01	1.36E+08
20	2.91E+01	1.51E+08
25	2.39E+01	1.62E+08
30	2.06E+01	1.72E+08
40	1.69E+01	1.88E+08
50	1.47E+01	2.01E+08
60	1.30E+01	2.13E+08
70	1.16E+01	2.24E+08
80	1.04E+01	2.33E+08
90	9.34E+00	2.42E+08
100	8.37E+00	2.49E+08

- Considers release of 50 percent of core halogens, no noble gases and 1 percent of other fission products to the sump solution.
- N.nnE+yy denotes n.nn x 10^{y}

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APPENDIX 14A

TURBINE MISSILE PROBABILITIES ANALYSIS

1.0 INTRODUCTION

The analysis of the consequences of a turbine operating (1800 rpm) and overspeed has demonstrated reasonable assurances that missiles would not be generated external to the low pressure turbine casing. The basic assumptions used in the analysis has let to this conclusion were deemed reasonable and conservative and backed by research and development projects especially on the low pressure turbine rotor's material properties.

Indian Point 3 has installed three low pressure turbines in accordance with Modification 90-03-182 MTG. The replacement turbines are significantly improved in design as compared with previous low pressure turbines. This new design reduces the probability of a low pressure turbine rotor failure which generates an external turbine missile. The new designed rotors are of a welded discs type (Figure 3.1 eliminating shrunk on keyed discs and of a material that has high resistance to stress corrosion cracking (SCC). These two major design changes have demonstrated excellent results in operating experiences with no stress corrosion cracking and yields a low probability of external missile generation.

The turbine missile evaluation provided in this Appendix is based on an ASEA Brown Boveri report (Reference 8).

2.0 SUMMARY OF REQUIREMENTS OF THE U.S. NUCLEAR REGULATORY COMMISSION (NRC)

The primary safety objective of the staff of the NRC is the prevention of unacceptable doses to the public from the releases of radioactive contaminants that could be caused by damage to plant safety-related structures, systems and components resulting from missile-generating turbine failures.

The criteria that must be met to demonstrate compliance with regulations is the General Design Criterion 4 of Appendix A to 10 CFR 50, nuclear power plant structures, systems, components important to safety shall be appropriately protected against effects, including the effects of missiles.

Failures of large steam turbines of the main turbine generator have the potential for ejecting large high-energy missiles that can damage plant structures, systems and components. The overall safety objective is to ensure that structures, systems and components important to safety are adequately protected from potential turbine missiles.

The NRC safety objective with regard to turbine missiles is expressed in terms of two sets of criteria applied to the missile generation probability (P1). One set of criteria is to be applied to favorably oriented turbines, and the other is to be applied to unfavorably oriented turbines. (See Table 1.1) The present orientation of the Indian Point 3 Low Pressure turbines places it in the unfavorably oriented category.

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Table 1.1
Turbine System Reliability Criteria

Probability, yr^{-1}		
Favorably oriented turbine	Unfavorably oriented turbine	Required licensee action
(A) $P_1<10^{-4}$	$P_1<10^{-5}$	This is the general, minimum reliability requirement for loading the turbine and bringing the system on line
(B) $10^{-4}<P_1 10^{-3}$	$10^{-5}<P_1<10^{-4}$	If this condition is reached during operation, the turbine may be kept in service until the next schedule outage, at which time the licensee is to take action to reduce P_1 to meet the appropriate A criterion (above) before returning the turbine to service. Exemptions may be granted for valid technical reasons or severe economic hardship.
(C) $10^{-3}<P_1<10^{-2}$	$10^{-4}<P_1<10^{-3}$	If this condition is reached during operation, the turbine is to be isolated from the steam supply within 60 days, at which time the licensee is to take action to reduce P_1 to meet the appropriate A criterion (above) before returning the turbine to service.

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(D) $10^{-2}<P_1$	$10^{-3}<P_1$	If this condition is reached at any time during operation, the turbine is to be isolated from the steam supply within 6 days, at which time the licensee is to take action to reduce P_1 to meet the appropriate A criterion (above) before returning the turbine to service

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3.0 DESCRIPTION OF ABB WELDED LP-ROTOR

3.1 Welded LP-Rotor Design

The welded design used by ABB got large LP-rotors is of the welded type, see Fig. 3.1. It consists of separate relatively small discs welded together to an integral rotor. The welds are positioned at the circumference and are of submerged arc type.

The main design features with respect to the turbine missile generation probability of the welded rotor are:

(1) Low stress level with consequently low yield strength material.

(2) No shrinks fits, no keyways and no central bore.

(3) The small disc forgings used for large LP-rotors can be relatively easily forged resulting in homogenous properties throughout the rotor.

(4) The small forgings with reasonable thickness assure high resolution during ultrasonic inspection.

(5) The welding procedure provides an inert gas atmosphere inside the hollow spheres and around the center of the discs, where the net stresses are highest during operation.

3.2 Description of LP-Rotor Materials

The material employed in the LP-rotors is a tempered, low alloy Cr Ni MO steel. The material used is similar to ASTM 471-65, Class 3, vacuum degassed alloy steel for forgings of turbine discs differing mainly by higher Cr and lower Ni content. The steel does not exactly correspond to ASTM steel due to the requirement of good weldability. The material was introduced in LP-rotor design in 1967 and has since then proven to have sufficient response to heat treatment and good welding properties. In addition, the impact energy, fracture appearance transition temperature FATT50, and fracture toughness are prescribed to exceed the values in the material standard used.

Cross Section of Standard LP-Rotor (Nuclear Power Plant Indian Point Unit 3) Figure 3.1

3.3 Description of Stress and Temperature Distribution in the LP-Rotor

The dominant principal stress in the welded LP-rotors is the circumferential (hoop) stress due to centrifugal forces of the rotor body itself and the blading, see Fig 3.2. The ABB design criteria assures that the maximum circumferential stress at rotor center does not exceed 53% of the minimum specified yield strength at operating temperature. The maximum circumferential stress at nominal speed of 1800 rpm acts at the center of the discs while the values at the outer surface are considerably lower.

The temperature distribution is determined by the steam temperature in the blading path, see Fig. 3.3. As a result of the moderate temperature gradient in the blade path, the temperature gradients in the rotor body are moderate too, resulting in small thermal stresses during operation.

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The stress and temperature distribution in each LP-rotor type is determined by Finite Element Calculations.

Fig. 3.2 Circumferential Stress in Welded LP-Rotor [6]
(Line Number –2) . 50 + σ /MPa 1 ksi = 6,895 MPa

Fig. 3.3 Temperature in Welded LP-Rotor [6]
(Line Number –1) · 10 = T_c T_F = 1,8·T_c + 32
T_c = Temperature in °C, T_F = Temperature in °F

4.0 DESCRIPTION OF OPERATING EXPERIENCE WITH WELDED LP-ROTORS IN NUCLEAR POWER PLANTS

The first turbine generator in a nuclear power plant with welded LP-rotors went into service in 1965. At the end of 1989 there are 59 turbine generators in service with a total of 144 welded LP-rotors. There are no reports to date on rotor failures and no indications of stress corrosion cracking. Table 4.1 summarizes the operating hours with respect to units with more and less than three years of operation.

From Table 4.1 it can be seen that the average operating hours of LP-rotors, which are in service now for more than three years, is approximately 70,000 hours.

Table 4.1

Operating Experience of Welded LP-Rotors in Nuclear Power Plants

	Number of Units	Number of LP-Rotors	Average Operating Hours per LP-Rotor
Total	59	144	
More than 3 years of operation	54	130	≈70,000
Less than 3 years of operation	5	14	≈15,000

5.0 HYPOTHETICAL FAILURE MODES OF WELDED LP-ROTORS

As described in Section 4, there are no failures of welded LP-rotors in nuclear power plants up to now. Therefore the discussion of failure modes is purely hypothetical.

Based on the experience of stress corrosion cracking (SCC) in LP-rotors of the shrunk on disc design, failures due to this type of cracking will be discussed as well as failures due to brittle fracture and fatigue crack growth.

5.1 Failure Modes Due to Stress Corrosion Cracking

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Stress corrosion cracking in LP-rotors is most likely to occur in the region where the transition from dry to wet steam is located, i.e., the region of the Wilson-Line.

It is assumed that a stress corrosion crack is initiated in this area.
The propagation rate of stress corrosion cracks in steam turbine rotor steels depends on the applied stress intensity [2]. This is illustrated in Fig. 5.1. At very low stress intensities, close to the threshold stress intensity, K_{ISCC} cracks grow extremely slowly, i. e., slower than 10^{-11} m/sec (0.01 inch/year). The stress intensity increases from K_{ISCC} and also does the stress corrosion crack growth rate until a plateau is reached where the crack growth rate no longer depends on the stress intensity for quite a range of stress intensity. This "plateau" crack growth rate depends on various other influential variables, for example, on the yield strength of the steel. At higher stress intensities, a further acceleration of stress corrosion cracks is observed, but unfortunately, not well documented. Available stress corrosion crack growth data [2] indicate that the plateau range extends to at least K_I = 100 ksi $\sqrt{\ }$ inch.

Figure 5.1

Effect of stress intensity and yield strength on the growth rate of stress corrosion cracks in a steam turbine rotor steel. Not that K_{ISCC} is not measurably influenced by the change in yield strength; the "plateau" stress corrosion crack growth rate, however, is strongly influenced by the yield strength.

With respect to possible failure modes, this means that once a crack is initiated it will grow in a stable manner until the crack size reaches a value corresponding to 100 ksi $\sqrt{\ }$ inch.

In case of a welded LP-rotor, the maximum principal stress (which is the crack driving stress) is the circumferential stress. This means a possible crack is most likely to be expected in an axial/radial plane, as illustrated in Fig. 5.2.

If the critical crack size is reached, the crack propagation changes from a stable to an accelerated state. When the crack extends to the welds, it will grow in circumferential direction in these areas while in the disc the crack will grow towards the center. Finally, the disc will fracture in three 120° pieces as shown in Fig. 5.2.

It must be pointed out here, that the discussed event is purely hypothetical. It is highly likely that a crack of the considered size at the surface of the rotor will cause a loss of several blades leading to a considerable unbalance and a trip of the unit.

This can be seen easily by comparing the critical crack size, which will lead to an accelerated crack growth, with the size of the blade attachment. The blades, except L-0 and L-1, are fixed in circumferential slots having a depth of maximum 3", while the critical crack size is more than 8".

On the analogy of the "leak before burst criterion" for piping it can be concluded that the welded rotor is protected by a "loss of blades before burst criterion" with respect to SCC. Therefore, this case must be considered purely hypothetical.

5.2 Failure Modes Due to Brittle Fracture

A failure as a result of a brittle fracture in a LP-rotor may occur during a cold-start or an unforeseen overspeed. The prerequisite of such an event is that an existing flow or crack inside the rotor is

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growing up to the critical crack size during operation. This crack growth is due to SCC for surface cracks respective to Low Cycle Fatigue (LCF) for surface and embedded cracks.

ABB assures by stringent requirements on the conditions of forgings for welded rotors that the discs do not have pre-existing flaws or inclusions of unacceptable size. The discs provided for Indian Point 3 were volumetric and surfaced examined and no unacceptable flaws existed.

The fracture toughness of the LP-rotor material is now prescribed to be at least 123 ksi$\sqrt{\ }$ inch at a temperature of 35° C. The actual measured values of K_{IC} values and the yield strength values for the discs of the three LP-rotors of the Indian Point Unit 3. The minimum value is 202 MPa4m (184 ksi$\sqrt{\ }$ in). According to NRC requirements, the ratio between fracture toughness and the maximum circumferential stress at overspeed (= 132% of nominal speed) shall exceed the value 2$\sqrt{\ }$ in.

The maximum stress amounts to:

$$\sigma_{max} = 1.32^2 \bullet \frac{Re_{min}}{1.9} = 84 \text{ ksi} \qquad \text{Equ. 5.14}$$

where:

σ_{max}: Maximum allowable circumferential stress during overspeed

Re_{min}: Minimum value of yield strength, Re_{min} = ksi at room temperature

"1.9": Minimum safety factor to yield strength for welded LP-rotors at nominal speed.

With the minimum measured fracture toughness, one obtains:

$$\frac{184 \text{ ksi} \sqrt{\ } \text{in}}{84 \text{ ksi}} = 2.19 \sqrt{\ } \text{in} > 2 \sqrt{\ } \text{in} \qquad \text{Equ. 5.2}$$

From these facts it can be concluded that a failure due to brittle fracture is much more unlikely than a failure due to SCC.

5.3 Failure Modes Due to Non-SCC

Non-SCC failure in LP-rotors is considered to be caused by fatigue crack growth. It is assumed that a fatigue crack is initiated in a plane perpendicular to the maximum principal stress. The maximum principal stress, which is the crack driving stress, can be the hoop stress or the radial stress (in case of notches only).

Fatigue crack growth in steam turbine rotor steels depends on the applied stress intensity range σK (see Fig. 5.4). It can only occur if stress intensity range σK exceeds threshold value σK_{th}. Above this

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threshold value the relation between the crack growth (da/dN) and the stress intensity range (σK) can be described by the following power-law:

$$\frac{da}{dN} = C \bullet \sigma K^{n} \qquad \text{Equ. 5.3}$$

Equation 5.3 is called "Paris – Equation." The values C and n are dependent on the material used. Once a crack is initiated it will grow in a stable manner until the crack size reaches a value corresponding to σK_{IC}. If this critical crack size is reached, the crack propagation changes from a stable to an accelerated state.

Figure 5.3

Results of K_{IC} -and $R_{\blacksquare}$ - Measurements for the LP-Rotors of Indian Point Unit 3

The items and the test report number (MP.-No) of the forgings of the three LP Indian Point Unit 3 LP-rotors summarized. The actual measure $R_{\blacksquare}$ - (yield strength) and K_{IC} - (fracture toughness) values at room temperature are also tabulated and the fracture toughness statistically analyzed.

Contents:

- Items and MP-numbers of the forgings
- Measured yield strength at room temperature
- Measured fracture toughness at room temperature
- Summary and statistical evaluation

LOW PRESSURE ROTORS

item	No. 1	No. 2	No. 3
1	MP 54912 B	MP 59210 B	MP 59976 B
2	MP 59211 B	MP 59987 B	MP 60220 B
3	MP 59212 B	MP 59977 B	MP 59974 B
4	MP 59213 B	MP 59978 B	MP 65581 B
5	MP 59214 B	MP 59988 B	MP 68147 B
6	MP 54913 B	MP 59215 B	MP 59975 B

MP-No. of the Forgings of the LP-Rotors of Indian Point Unit 3

LOW PRESSURE ROTORS

Item	No.1	No. 2	No. 3
1	97.2	103	107
2	98.9	98.6	98.5
3	103	104	101
4	104	107	99.2
5	99.9	98.2	101
6	99.6	101	106

Yield strength at room temperature in units of ksi.
(Min. value 97.2 ksi, max. value 107 ksi)

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LOW PRESSURE ROTORS

Item	No. 1	No. 2	No. 3
1	670	708	735
2	682	680	679
3	712	718	694
4	715	735	684
5	689	677	693
6	697	698	730

Yield strength at room temperature in units of Mpa
(Min. value 670 Mpa, max. value 735 Mpa)

LOW PRESSURE ROTORS

Item	No. 1	No. 2	No. 3
1	233	232	184
2	251	209	216
3	246	226	229
4	260	243	254
5	255	259	254
6	250	221	188

Fracture toughness at room temperature in units of
ksi$\sqrt{\ }$ in (Min. value 184 ksi$\sqrt{\ }$ in, max. value 260 ksi$\sqrt{\ }$ in)

Figure 5.3 (continued)

LOW PRESSURE ROTORS

Item	No. 1	No. 2	No. 3
1	256	255	202
2	276	230	238
3	270	248	252
4	286	267	279
5	280	285	279
6	275	243	207

Fracture toughness at room temperature in units of
MPa4m (Min. value 202 Mpa $\sqrt{\ }$ in, max. value 286 MPa 4m)

SUMMARY OF RELEVANT MATERIAL PROPERTIES

Yield strength at room temperature is in the range of 97 ksi (670 Mpa)
to 107 ksi (735 Mpa).

Fracture toughness at room temperature is in the range of 184 ksi$\sqrt{\ }$ in.

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(202 MPa 4m) to 260 ksi $\sqrt{\ }$ in (286 MPa 4m).

STATISTICAL EVALUATION

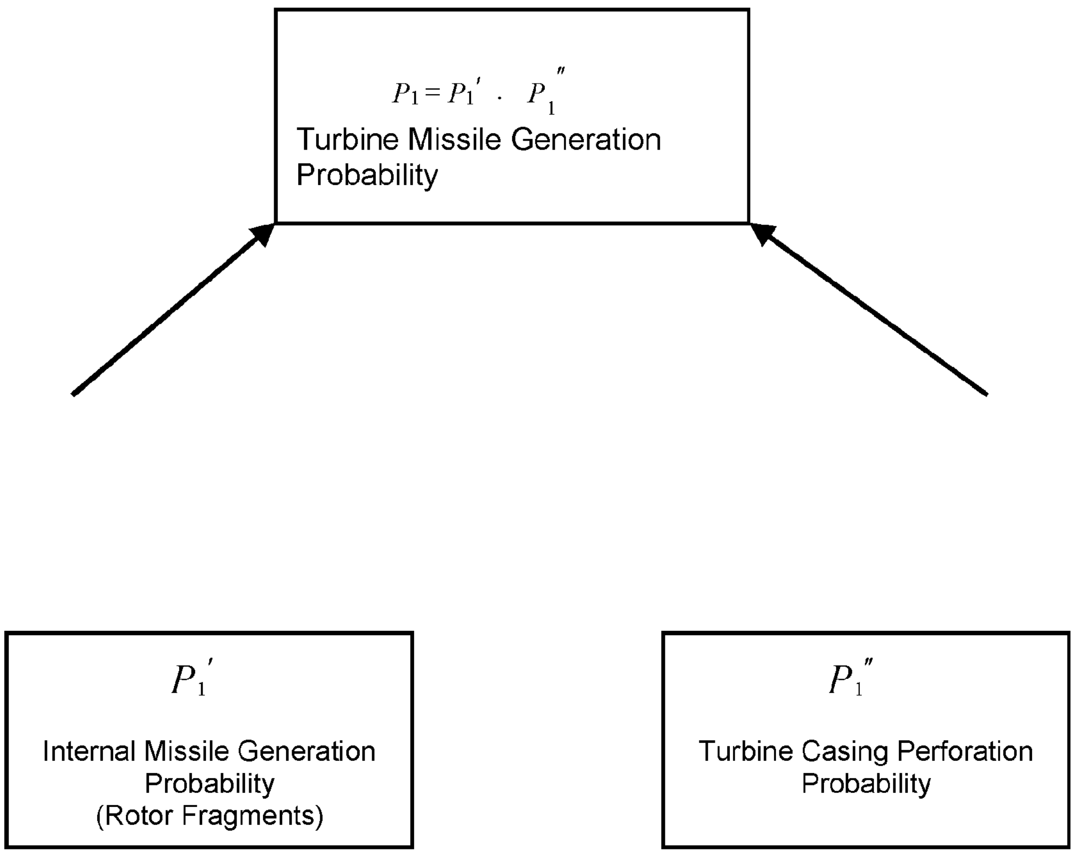
Mean value : $\overline{K}_{IC}$ + 234 ksi $\sqrt{\ }$ in (257 MPa 4m)

Standard deviation: $S_{KIC} + \delta_{KIC} \bullet \overline{K}_{IC} = 0.1 \bullet K_{IC}$

Figure 5.4 - Fatigue Crack Growth

6.0 METHOD FOR CALCULATING TURBINE MISSILE GENERATION PROBABILITY (P_1)

The turbine missile generation probability (P_1) consists of two factors (1) the probabiliy of shaft failure producing an internal turbine missile (P_1') and (2) the probability that this internal missile penetrates the casings and is ejected from out the turbine (P_1'').



The probability P_1' can be determined by means of fracture mechanics such as critical crack sizes, crack growth rates, stresses and temperatures.

These properties and details are very well documented in the case of turbine rotors.

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The procedures for estimating P_1" are not as sophisticated as the procedures for calculating P_1'. The usual method is to compare the kinetic energy of a potential internal turbine missile with the energy necessary to perforate the turbine casing. The result of such an estimation will be either P_1" = 0 or P_1" = 1.0.

Considering these facts, a conservative approach is assumed that the probability P_1 to be one. This means the turbine missile generation probability equals the internal turbine missile generation probability:

$P_1 = P_1$

6.1 Method for Calculating Turbine Missile Generation Probability (P_1) Due to SCC

According to the present knowledge on SCC phenomena, three ranges have to be distinguished.

(1) Crack Initiation or Incubation Phase:

It is commonly accepted that a threshold value K_{ISCC} exist. If the stress intensity K_I is below this threshold SCC is not possible.

(2) Constant Crack Growth Rate:

If the stress intensity K_1 clearly exceeds the threshold value K_{ISCC} the crack growth rate remains constant on a certain plateau value for quite a large range of K_I.

(3) Accelerated Crack Growth Rate, Critical Crack Size:

If K_I exceeds a certain amount, the assumption of a constant plateau-value is no longer valid. Available data [2] indicates that the plateau range extends to at least 100 ksi$\sqrt{\ }$ in is used for the determination of critical crack sizes.

The method for a probabilistic approach to this problem is similar to the proposal of Clark Seth and Shaffer, presented in [3]. However, some modifications we felt necessary from today's point of view, have been introduced.

The probability of generating a missile (P_1) under the conservative assumption $P_1 = P_1$', which was explained previously, is computed as a function of time as follows:

$$P_1(T) = \sum_{i=1}^{M} p_1(T)\ q_1 \qquad \text{Equ. 6.1}$$

(Valid for $p_1\ q_1 << 1$)

Where:

M : Number of flows in the unit

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T : Time in operating years

$P_1(T)$: Probability of missile generation in an individual flow of a LP-rotor.

q_1 :Probability of crack initiation in an individual flow of a LP-rotor

Due to the fact that the ABB LP-rotors in a unit have the same design and the crack will initiate at the same location (Wilson-Line) Equ. 6.1 can be rewritten as:

$P_1 (T) = M . p (T) . q$ Equ. 6.2

6.1.1 Probability of Crack Initiation ,q

The probability of crack initiation in the welded LP-rotors is determined from the operating experience described in Section 4. Only those LP-rotors are considered which have been in a operation for more than three years and which have been inspected at least once a year.

Up to now, there have been no indications of SCC in ABB welded LP-rotors. Therefore, the probability q is conservatively assumed to be the 95% upper confidence bound, differing from the proposal in [3], where the 50% confidence bound was used.

According to usual formulas [4], the 95% upper confidence bound is:

$Q =1 – (0.05)^{1/L}$ Equ. 6.3

Where L denotes the number of inspected LP-flows. As mentioned in Section 4, Table 4.1, L amounts to 2 . 130 =260, leading to:

$q = 0.011$ Equ. 6.3*

6.1.2 Probability of Missile Generation of an Individual LP-Flow, p (T)

The probability of missile generation of an individual LP-flow is defined as the probability that an existing crack will grow rate and critical crack size which is dependent on the loading case (nominal speed and respective overspeed), are the main parameters to be considered.

6.1.3 Influence of Nominal Speed

For nominal speed calculation, it is assumed:

- ° Crack will grow under nominal speed condition due to stress corrosion cracking up to critical crack size.
- ° Critical crack size is fixed by stress intensity K_{IP} at end of plateau range (upper limit of constant crack velocity).

For this case one obtains the following input data:

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a. Crack Growth Rate, r

In accordance with [3] the crack growth rate is treated as log-normal distributed random variable having a mean of :

$$\mathrm{Ln}\,(r) = -4.968 - \frac{7302}{T_F + 460} + 0.0278\,\overline{\mathrm{Re}} \qquad \text{Equ. 6.4}$$

Where:

Ln (r): logarithm of the mean r

T_F: temperature in °F

$\overline{\mathrm{Re}}$: mean of yield strength at room temperature

the physical unit of Equ. 6.4 is in./hour. This equation can be rewritten in the following form:

$$\bar{r} = \exp\left(4.110 - \frac{7302}{1.8T_C + 492} + 0.0278\,\overline{\mathrm{Re}}\right) \text{ in/year} \qquad \text{Equ. 6.4*}$$

Where:

T_C : temperature in °C

The standard deviation Sr equals 0.587 with reference to Ln (r).

In the calculation, the temperature is taken from the Finite Element Analysis and the yield strength is the average value between upper and lower bounds.

b. Critical Crack Size, a_C

The critical crack size a_C for a semi-elliptical surface crack is given by:

$$a_c = G \cdot \frac{1}{1.21 \cdot \pi}\left(\frac{K_{IP}}{\sigma}\right)^2 \qquad \text{Equ. 6.5}$$

Where:

G: Flaw geometry factor

K_{IP}: Stress intensity at the end of the plateau range

Generally, G, K_{IP}, σ are random variables. With respect to this, the following assumptions are made:

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c. Flaw Geometry Factor G

In accordance with [3], G is a uniformly distributed variable ranging from 1.0 to 2.0. The mean is $\overline{G} = 1.5$ and the standard deviation is SG = 0.289.

d. Stress Intensity at End of Plateau Range K_{IP}

According to [2], the plateau values of the constant crack growth rate are only established properly to an upper limit of $\underline{K}_{IC}$ = 100 KSI$\sqrt{\ }$ in. The currently available test results seem to be not sufficient to perform a statistical analysis with respect to the scattering of this plateau limit. Laboratory tests performed by the turbine O.E.M. indicate that the assumed limit K_{IP} = 100 KSI $\sqrt{\ }$ in. is a reasonable conservative value.

It should be noted, that the fracture toughness, K_{IC}, of the material employed in the LP-rotor is specified to be $K_{IC} \geq$ ksi $\sqrt{\ }$ in at 35% C. For these reasons, K_{IP} = 100 ksi $\sqrt{\ }$ in is taken as a constant and not a random variable.

e. Operational Net Stress, σ

The maximum principal stress is the circumferential stress, which is the superposition of centrifugal and thermal stresses during operation. The steady state stresses have to be considered since during startup compressive thermal stresses at the surface are induced.

Due to the fact that all stresses and temperatures are calculated by the Finite element Method, a relative standard deviation $\delta_\sigma = S_\sigma / \overline{\sigma} = \pm 5\%$ is realistic, whereby σ is assumed to be normal distribution.

The mean value σ is determined depending on the location of the Wilson-Line and the critical crack size based on K_{IP}.

f. Determination of Mean $\overline{a}_c$

$$\overline{a}_c = \overline{G} x \frac{1}{1.21\pi}\left(\frac{K_{IP}}{\overline{\sigma}}\right)^2 \qquad \text{Equ. 6.6}$$

g. Determination of Sa_c

For small relative standard deviations the following formula may be used [4]:

$$\frac{Sa_c}{\overline{a}_c} = \sqrt{\left(\frac{SG}{\overline{G}}\right)^2 + 4x\left(\frac{S\sigma}{\overline{\sigma}}\right)^2} \qquad \text{Equ. 6.7}$$

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Using the above given values one obtains:

$$\frac{Sa_c}{\bar{a}_c} = 0.217 \qquad \text{Equ. 6.8}$$

h. Truncation-Factor A of Distribution of a_c

In the distribution of a_c, a truncation is introduced so that the lower bound of a_c corresponds to the lower bound of the random variable G, G = 1.0.

This leads to a truncation of:

$$\frac{Sa_c}{A \bullet \bar{a}_c} = \frac{\bar{G}-1}{\bar{G}} \qquad \text{Equ. 6.9}$$

Whereby a symmetrical distribution was assumed. This means the distribution function of a_c will be truncated at the points $a_c + A \bullet Sa_c$ and $a_c - A \bullet Sa_c$.

6.1.4 Influence of Overspeed (132%)

For overspeed calculation, it is assumed:

°Crack will grow under nominal speed condition due to stress corrosion cracking up to the critical crack size.

°Critical crack size is fixed by fracture toughness K_{IC} (brittle fracture criterion) and stress at overspeed.

Therefore, it is necessary to modify some of the input data.

a) Critical Crack Size, a_C

In this case, the critical crack size for a semi-elliptical surface crack is given by:

$$a_c = G \cdot \frac{1}{1.21 \cdot \pi} \cdot \left(\frac{K_{xc}}{\sigma}\right)^2 \qquad \text{Equ. 6.10}$$

Where:

G : Flaw geometry factor

σ : Operational net stress at overspeed (132% of nominal speed) which is calculated by the finite element method.

K_{IC} : Fracture toughness (critical stress intensity)

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b) Fracture Toughness K_{IC}

Because of test results, fracture toughness K_{IC} can be statistically analyzed:

Mean : $\overline{K}_{IC} = 234 ksi ! in \quad (257 MPa ! m)$

Standard Deviation : $SK_{IC} = \delta K_{IC} \cdot K_{IC} = 0,1 \cdot \overline{K_{IC}}$

For this reason K_{IC} is taken as a random variable in contrast to K_{IP} which is taken as a constant representing a lower bound value.

c) determination of Mean a_c and Standard Deviation Sa_c

Based on random K_{IC} - values, random a_c – values can be calculated. The mean value is given by:

$$\overline{a}_c = \overline{G} \cdot \frac{1}{1.21 \cdot \pi} \cdot \left(\frac{\overline{K}_{IC}}{\overline{\sigma}} \right)^2 \qquad \text{Equ. 6.11}$$

With the standard deviation:

$$sa_\sigma = \overline{a}_\sigma \cdot \sqrt{\left(\frac{SG}{\overline{G}} \right) 2 + 4 \cdot \left(\frac{S\sigma}{\overline{\sigma}} \right) 2 + 4 \cdot \left(\frac{SK_{IC}}{K_{IC}} \right) 2} \qquad \text{Equ. 6.12}$$

Which is available for small relative standard deviations.

Using:

SG/G	=	0.193
$S\sigma / \sigma$	=	0.05
SK_{IC}/K_{IC}	=	0.1

one obtains:

Sa_a/a_c	=	0.295

d) Truncation Factor A

The modified truncation Factor A is calculated with Equ. 6.9.

6.1.5 Computer Code

The procedure for calculating the probability of generating a missile was computerized, the turbine O.E.M. internal computer code HC317.

The program calculates the probability for an individual LP-flow p (T), and generates a plot showing the total probability P_1 (T) according to Equ. 6.2 for a given turbine generator versus operating years.

This program was used for both cases (nominal speed condition, overspeed condition).

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6.2 Method for Calculating Turbine Missile Generation Probability (P_{1LCF}) Due to Non-SCC

LP-rotors of steam turbines experience the highest stresses during cold starts or at overspeed. Figure 6.1 and 6.2 present schematically these two operation cycles with their typical time dependent principal stresses.

Figure 6.1 shows the time dependent stress during a normal operation cycle. The maximal stress occurs during the startup phase due to the thermal expansion of the rotor. During full-load rejection the stresses are higher, caused by the occurring overspeed.

Prior to welding of the rotor, the forgings will be subjected to an ultrasonic (e.g., non-destructive) examination that will locate and scale the majority of flows, though some may escape detection (below the minimum detectable crack size). The probability that such an initial crack grows to critical crack size is calculated as the probability of rotor failure, which means the probability of turbine missile generation due to Non-SCC.

The probability of generating a missile due to Non-SCC (P_{1LCF}) under the conservative assumption:

$$P_{1LCP} \hat{=} P_1 = P_1'$$

P_1 and P_1') is computed as a function of the load cycles as follows:

$$P_{1LCF}(N) = \sum_{J=I}^{M} \qquad \sum_{i=I}^{F} P_{ij}(N) \cdot r_i \qquad \text{Equ. 6.13}$$

Where:

M: Number of flows in the unit

F: Number of different forgings (disks, shaft-end) per flow

N: Load cycle (cold, warm or hot starts, conservatively assumed That all starts are cold starts)

$P_{ij}(N)$: Probability of missile generation of an individual forging (i) of an individual flow (i) of an LP-Rotor

r_i: Probability that a crack with the maximal crack length a_o in forging i is not detected during ultrasonic inspection. It is assumed that r_i=1 for all forgings.

Due to the fact that the turbine O.E.M. LP-rotors in a unit have the same design and with the assumption that $r_i = 1$, Equ 6.13 can be rewritten as:

$$P_{1LCF}(N) = M \cdot \sum_{i=1}^{F} P_1(N) \qquad \text{Equ 6.14}$$

Since one double flow LP-rotor of the Indian Point Unit 3 has six forgings (2 shaft-ends,

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4 disks), three forgings per flow must be considered (F = 3), and so one obtains:

$$P_{ILCF}(N) = M \cdot [P_1 (N) + P_2 (N) + P_3 (N)] \qquad \text{Equ. 6.15}$$

The indices correspond to the forgings (shaft-end, thin disc, thick disc) of a flow.

Figure 6.1: Normal Operation Cycle
Figure 6.2: Full Load Rejection With Overspeed

6.2.1 Probability of Missile Generation of an Individual LP-Rotor Forging p_1 (N)

The probability of missile generation due to Non-SCC for an individual LP-rotor forging is defined as the probability that an initial crack (crack length a_o) grows up to the critical crack length a_c for brittle fracture.

For the determination of this probability some assumptions are made:

- Each forging has an initial crack with the length a_o at the location, where the highest transient stress appears.
- The growth of this initial crack due to low cycle fatigue can be described with the "Paris-Equation."
- The critical crack length a_c is fixed by fracture toughness K_{IC} and maximum principal stress at maximum overspeed.
- Consequently, the crack lengths and the crack growth behavior of the used rotor material are the main parameters to be considered:

a. Initial Crack Length, a_o

Prior to the welding of the LP-rotor, each forging will be subjected to a complete ultrasonic inspection. This examination will locate and scale the majority of existing flows and cracks, though some, which are below the minimum detectable defect size, may escape detection.

For this reason, it is assumed that each forging of an LP-rotor has an initial crack at the location of the highest transient stress. The assumed initial crack length is in the magnitude of 1.25 mm radius (0.049) inch), which is nothing else than the maximum value of minimum detectable defect size of all forgings used for Indian Point Unit 3 Lp-rotors.

For the following calculations:

a_c = 1.27mm – 0.05 inch

was chosen for each of the 4 discs and 2 shaft-ends of the rotors.

The relative standard deviation for a_o is assumed to be log-normal distribution.

b. Critical Crack Length, a_c

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The critical crack size a_c for a semi-elliptical surface crack is given by equation 5.10:

$$a_c = \frac{G}{1.21 \cdot \pi} \cdot \left(\frac{K_{IC}}{\sigma_{MAX}} \right)^2$$

Where:

G : Flaw geometry factor with the mean G = 1.5 and the relative standard deviation $\delta G = 0.193$

K_{IC} : Fracture toughness which is statistically analyzed from the test results of all 18 rotor forgings of Indian Point Unit 2.

σ_{MAX} : Maximum principal stress at maximum overspeed.

c. Maximum principal stress at overspeed. δ_{MAX}

The maximum principal stress at overspeed is the circumferential stress. For the calculation of the critical crack length, the maximum value of $\delta_{\max}$ is taken into account. The maximum values appear adjacent to the rotor axis [7]. The stresses are calculated by the finite Element Method, so a reliable standard deviation of $\delta_{\max} = 0.05$ is realistic.

d. Crack Growth, da/dN

The crack growth due to LCF can be described with the "Paris-Equation." With the relation between the stress intensity range σK and the range $\sigma\sigma$:

$$\Delta K = \sqrt{\frac{1.21 \cdot \pi}{G}} \cdot \Delta\sigma . \sqrt{a} \qquad \text{Equ. 6.16}$$

One obtains the crack growth relation in the following form:

$$da/dN = C \cdot \left[\sqrt{\frac{1.21 \cdot \pi}{G}} \cdot \sigma \cdot \sqrt{a} \right]^n \qquad \text{Equ. 6.17}$$

The integration of Equ. 5.17 from the initial crack length a_o yields to the crack length after N load cycles:

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$$a_N = \left\{ a_o^{(1-n/2)} + (1-n/2) \cdot C \cdot \left(\sqrt{\frac{1.21\pi}{G}} \cdot \Delta\sigma \right)^n \cdot N \right\}^{(1/(1-n/2))}$$

Equ. 6.18

The parameters C and n are material dependent values, which are determined from fatigue crack growth tests. C has a mean value of :

with a relative standard deviation of σ_c = 1.08. The exponent n has a value of n = 3, which is an upper bound value.

σ σ is the maximum stress range during the cycles.

e. Stress range, σ σ

The maximum stress range σ σ results from the different thermal expansion during start conditions, whereby the greatest ranges occur during cold starts.

These stresses are determined by a transient calculation by the Finite element Method [7]. It is assumed that the stress range is equivalent to the maximum appearing stress during a start.

The stress range has the same value for the standard deviation as the other stresses,

$$\delta_{\#\sigma} = 0.05$$

7.0 Low Pressure Rotor Inspection Requirement

7.1 Determination of Inspection Intervals

The maximum allowable inspection intervals are determined evaluating the results for the turbine missile generation probability P_1 (T) for the individual turbine generator.

In the general inspection and overhaul plans, major rotor inspection intervals of 50,000 equivalent operating hours is recommended, see Fig 7.3. If LP-0 UT inspection is successfully performed before 50,000 EOH, an additional 30,000 EOH will be available to Low Pressure Turbine without a major rotor inspection. The results obtained with the probabilistic approach reveal much longer inspection intervals of 14 years. Therefore, the risk of stress corrosion cracking is completely covered by the usual inspection and overhaul programs and no additional measures have to be introduced.

7.2 Recommended LP-Rotor Testing

If a welded LP-rotor is affected by SCC, the cracks will initiate at the outer surface of the rotor body.

The usual recommended LP-rotor testing of welded rotors during major overhauls assures that any indications of SCC will be detected. The testing includes a through visual inspection for erosion and corrosion and a magnetic particle testing at selected areas. In the case of indications, additional ultrasonic examinations will be performed.

Therefore, a complete volumetric ultrasonic inspection for SCC is not necessary in the case of welded LP-rotors.

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In case of sufficient high probability of failure due to fatigue crack growth, 100% volumetric ultrasonic inspection would be necessary. In case of sufficient low probability, no ultrasonic testing is necessary.

Figure 7.3
Recommendations for Inspection Intervals of Large Turbine Generators

7.3 Results for Indian Point Unit 3 Low Pressure Welded Rotor (DS92 Design)

7.3.1 Cross Section of Standard LP-Rotor DS92, see Figure 3.1.

7.3.2 Program Input for SCC-Calculation.

7.3.3 Input Data for Nominal Speed Condition.

The original and determination of the input data for the nominal speed condition are summarized in Appendix 7.1.

The input variables obtained are summarized in Table 7.1.

Table 7.1

Indian Point Unit 3 (Nominal Speed)

a_o inch	Sa_o Inch	Truncation A	r Inch/year	Sr	q	M
8.20	1.78	1.54	0.063	0.587	0.011	6

As a result of the computation, the probability P_1 is plotted versus service life in Figure 7.1.

7.3.4 Input Data for Overspeed Condition

The original and determination of the input data for overspeed condition are summarized in Appendix 7.2.

The input variables obtained are summarized in Table 7.2.

Table 7.2
Indian Point Unit 3 (Overspeed 132%)

a_o inch	Sa_o inch	Truncation A	r inch/year	Sr	q	M
11.20	3.31	1.13	0.063	0.587	.0011	6

As a result of the computation, the probability P_1 for Overspeed Condition is plotted versus service life in Figure 7.1, too (dotted line)

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7.4 Inspection Intervals Because of SCC

The comparison between the two different speed conditions (Figure 7.1) shows that overspeed yields lower time dependent probabilities. This means that nominal speed conditions is dominant for determination of inspection intervals.

As the Indian Point 3 Nuclear Power Plant is in a unfavorable orientation, the 10^{-5} value has to be taken as a minimum limits. See Table 1.1.

From this, a maximum inspection interval of 14 years is allowed see Figure 7.1.

Figure 7.1
LP-Retrofit Indian Point 3 Assessment of the Probability of Steam Turbine Rupture from Stress Corrosion Cracking

PROBABILITY P AS A FUNCTION OF TIME

95% COMF. BOUND: NOMINAL SPEED ——

------- 95% COMF. BOUND: 132% OVERSPEED

7.5 Program Input and Results for Non-SCC

The determination of the input data for non-SCC condition is summarized in Appendix 7.3.

The input data is used for the OEM computer program PROBFRAC, which calculates the probability of missile generation of an individual rotor-disc. For each type of disc (thin disc, thick disc, shaft end), this program was applied to calculate the probabilities P_1 (N). With Equation 6.15, the probability of missile generation of Indian Point 3 was determined and a plot P_{1LCF} (N) versus the numbers of load cycles N was plotted (Figure 7.2).

This figure presents that the probability of missile generation due to Low Cycle Fatigue after N = 250 cycles is in the magnitude of $7 \cdot 10_{-17}$ (here it is assumed that the unit operates approximately 40 years with six starts per year).

In comparison to Figure 7.1, this diagram also presents that SCC is the overly dominant failure mechanism and consequently no additional ultrasonic testing is necessary for detection of fatigue crack growth.

Figure 7.2
LP-Retrofit Indian Point 3 Assessment of the Probability of Steam Turbine Rupture From Low Cycle Fatigue

PROBABILITY P AS A FUNCTION OF LOAD CYCLES

Appendix 7.1

a.) Critical Crack Size a_c and Truncation Factor A

o Mean value $\bar{a}$ See Section 6.1.3. f)

$\bar{G}$ = 1.5 See Section 6.1.3. c)

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K_{IP} = 100 ksi 4in — See Section 6.1.3. d)

$\overline{\sigma}$ =151 MPa = 21.9 ksi — See Section 6.1.3. e) and Fig. 7A

$\overline{a}_c = 8.20$ inch — See Equation 6.6

o Standard deviation S_{ac} — See Section 5.1.2.1. g)

$S_{ac} = 0.217 \cdot \overline{a}_c = 1.78$ inch — See Equation 6.8

o Truncation

$A = 1.54$ — See Section 6.1.3.1. h) Equation 6.9

b) Crack Growth Rate r

o Mean value $\overline{r}$ — See Section 6.1.3 a)

$T_c = 148°C(298°F)$ — See Section 6.1.3 a) and Fig. 7B

$\overline{\mathrm{Re}} = 99 ksi$ (mean value)

Standard deviation $Sr = 0.587$ — See Section 6.1.3 a)

$\overline{r} = 0.063$ inch / year — See Equation 6.4 *

c) Crack Initiation Probability q and Number of Individual Flows per Unit M

95% upper confidence bound $q = 0.011$ — See Section 6.1.1. Equation 6.3 *

$M = 6$

Appendix 7.2

PROGRAM INPUT (OVERSPEED)

a) Critical Crack size a_c and Trucation Factor A

o Mean value $\overline{a}_c$ — See Section 6.1.4 c)

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$\overline{G} = 1.5$ from nominal speed condition	See Section 6.1.3 c)
$\overline{K}_{IC} = 234 ksi$ Ö in	See Section 6.1. .4. b)
$\overline{\sigma} = 303\,MPa = 43.9\,ksi$	See Section 6.1.4 c) and Fig. 7C
$\overline{a}_c = 11.20\,inch$	See Equation 6.11
o Standard deviation S_{ac}	See Section 6.1.4. c)
$S_{ac} = 3.31$ inch	See Equation 6.12
o Truncation	
$A = 1.13$	See Section 6.1.3 h)

b) Other Input Data

All other input data such as

o Crack growth rate $\overline{r} = 0.063$ inch/year (Mean value)
$sr = 0.587$ (standard deviation)

o Crack initiation probability $q = 0.011$

o Number of individual flows per unit

$M = 6$

are the same values as in nominal speed condition

(See Appendix 7.1.)

Appendix 7.3

PROGRAM INPUT (NON-SPEED)

a) Mean and Standard Deviation of Initial Crack Length

It is assumed that each forging has an initial crack with the length:

$\overline{a}_o = 1.27\,mm = 0.05\,inch$ (See Section 6.2.1 a)

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and a standard deviation of:

$\delta_{ao} = 0.41$ (See Section 6.2.1 a)

at the location with the highest transient stress $(\# \sigma)$.

b) Maximal stress at Overspeed $(\sigma_{\max})$ and Transient Stress $(\# \sigma)$

The stress values are taken from IP3's Rotor stress Report TB HTGE52245. There the FE-nodes with the highest stresses of each disk or shaft-end are tabulated. $\# \sigma$ and $\sigma_{\max}$ are at different but adjacent locations in the FE-mesh. For that reason, it is conservatively assumed that maximum transient stress $\# \sigma$ (Fig. 7D) and maximum stress at overspeed $\delta_{\max}$ (Fig. 7C) occur at the same FE-node.

The stress values for each forging, which were used for the calculation, are summarized in the following tables:

Table 7A
Input Stress Values in MPa

	Disk 1/16 Shaft End	Disk 2/5 Thin Disk	Disk 3/4 Thick Disk
σ_{max}	492	416	403
$\Delta \sigma$	294	250	310

Table 7B
Input stress Values in ksi

	Disk 1/16 Shaft End	Disk 2/5 Thin Disk	Disk 3/4 Thick Disk
σ_{Max}	71.4	60.3	58.4
$\Delta \sigma$	42.6	36.3	45.0

The relative standard deviations are:

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$$\delta_{\sigma \max} = \delta_{\# \sigma} = 0.05$$

c) Fracture Toughness and Flaw Geometry Factor

For the fracture toughness, the mean and the standard deviation of the actual measured values form the manufactured disk are taken (see Appendix 2.2):

$$K_{IC} = 257\, MPa\, \text{Ö}\, m = 234 ksi\, \text{Ö}\, in$$
$$\delta_{KIC} = 0.1$$

For the flaw geometry factor G, a surface crack in each disk is conservatively assumed, i.e., G and δ_G are the same values as for SCC-calculation:

$$G = 1.5$$
$$\delta_G = 0.193$$

(See Section 6.1. 3 c)

d) Parameters for the Paris-Equation C and n

These parameters are described in Section 6.2.1 d):

$$\overline{C} = 3.2 \cdot 1^{-12}$$

$$\delta_c = 1.08$$

n = 3 (upper bound value)

Figure 7D - Transient Hoop stress Distribution at Nomial Speed and t=19,200 Sec.

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FSAR APPENDIX 14B

CONSEQUENCES OF A TURBINE MISSILE AT INDIAN POINT 3

1.0 Introduction

This study assesses the possibility of damage due to missiles resulting from steam turbine failure. Turbine blades can fracture and fragments can be ejected at high velocities, breaking through the turbine casing. These turbine missiles could affect the safe operation of the plant. This analysis has been performed to predict the probability of compromising plant safety due to turbine missiles. The method and result of this analysis is discussed below.

2.0 Basis and Assumptions

This analysis is based on the original low pressure (LP) turbines at IP3 manufactured by Westinghouse Corporation. It assumes stress corrosion cracking failure of shrunk-on rotor discs, which break up into large segments. The replacement LP turbines manufactured by ASEA Brown Boveri are of the welded rotor design. They do not have shrunk-on discs and will not produce the large missile segments assumed on turbine failure. The new rotors meet or exceed the design criteria of the Westinghouse rotors including design overspeed. This analysis is re-introduced into the FSAR as it forms the original design basis for 132 % overspeed, the LP Steam Dump system, the Back-up Service Water system, and City Water back-up for Charging Pump cooling.

Westinghouse Corporation, the manufacturer of the original turbines at Indian Point 3, calculated the probability of a turbine failure which generates external missiles as a result of stress corrosion cracking of rotor discs and keyways. In this analysis, this probability is known as P_1, and is a function of crack initiation, subsequent crack growth with time, and critical crack depth. P_1 values have been supplied by Westinghouse for turbine disc failure at rated speed and at 132 % overspeed, for inspection intervals of 18 months, 3 years and 5 years.

Turbine failure produces missiles from the breakup of a turbine disc and other secondary internal impacts from the disc sectors. Missiles of various sizes, shapes, and velocities result which, after leaving the turbine casing, become projected hazards to the remainder of the plant. Therefore, the major objective is to give reasonable assurance of public protection by evaluating the consequences of a turbine disc rupture. That is, determine the probability of turbine missiles causing an offsite release of radiation. As will be discussed below, this analysis shows that the risk of releasing radioactive material due to an accident involving turbine missiles does not violate the limits specified in 10 CFR 100 (1) of 10^{-7} events/year.

Typically, the overall probability of producing a compromise of plant safety, P_4, is factored into the following separate probabilities:

P_1 = probability of turbine failure which results in ejection of external missiles;

P_2 = conditional probability given a turbine failure that missiles from a failed turbine strike each component in a system required for safe shutdown;

P_3 = conditional probability given a missile strike that the function of the struck component is critically impaired.

Therefore, $P_4 = P_1 \times P_2 \times P_3$

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As discussed previously, P_1 is a function of crack initiation, subsequent crack growth with time, and critical crack depth. P_1 is not specifically addressed in this analysis. The Westinghouse determination [(2)] of P_1 is utilized in this analysis for each disc of the low pressure system for determining P_4. Values for P_1 were provided by Westinghouse for several turbine inspection intervals, and are listed in Table 14B-1. The high pressure turbine is assumed to have a negligibly low probability of failure of the type, which produces external missiles.

The other two component probabilities are addressed by the analysis described herein. P_2 is directly calculated by the analysis described in this document. P_3 is addressed indirectly in that the energy range of missiles, which strike critical components, is computed. P_3 is difficult to assess precisely because of the many variables involved. The energy of sticking missiles, however, is probably the most important of these variables and an estimate of it is provided by this analysis. For turbine missiles, P_3 is typically considered to be unity since these very heavy missiles are rather damaging. Furthermore, such an assumption is conservative.

Simulated turbine failures that result is strikes on a sufficient number of components of a critical system are considered to compromise the safety of the plant. The quantity P_2 as defined above accounts for all redundant equipment, and is therefore, the probability given a turbine failure that missiles cause a safety compromise of the plant.

In determining the probability of an offsite release, it is necessary to go beyond P_4 as defined above. In plants such as Indian Point 3 with a protected spent fuel storage, damage to the reactor core is the only mechanism, which can produce an offsite release. Core damage comes from compromise of functions, which are required to achieve and maintain a safe shutdown condition.

Neither an offsite release nor damage to the core is an automatic consequence of damage to equipment, which must ultimately function in order to achieve and maintain a safe shutdown. Generally, there is a significant length of time between damage to such equipment and the requirement that it function in the capacity required for achieving or maintaining a safe shutdown. Repairs can be made or alternative actions taken before such equipment is actually needed. In fact, for most safety-related equipment, there is already in place a procedure for achieving a safe shutdown when the normal function provided by the equipment has been compromised. Those functions, which are predicted by this study to be compromised by turbine missiles, are considered to have a negligible probability of resulting in an offsite release if there is already in place at Indian Point 3 an established procedure for achieving a safe shutdown when plant equipment that normally performs that function has been lost.

2.1 Maximum Design Overspeed

For determination of the maximum design overspeed, the following conservative sequence was assumed:

a) The unit is operating at full load with all turbine valves wide open:
b) The entire turbine load is dropped instantaneously (no credit taken for plant auxiliary load);
c) The auxiliary governor is assumed to operate improperly (i.e., does not respond to turbine load mismatch);
d) Trip is initiated at the emergency overspeed set point:
e) From this point on, the turbine valves operate in the prescribed manner.

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At the instant the load is dropped, the unit is assumed to accelerate at a constant maximum rate corresponding to the initial steam flow and rotational inertia of the unit until the unit reaches the emergency overspeed trip set point plus a pure time delay of 0.1 second. Flow into the turbine is then calculated during valve closure and is modified for flow versus lift characteristics. It takes approximately 0.15 second to fully close all of the turbine valves following the initial 0.1 second delay. Once the valves close completely, additional overspeeding is calculated using the energy stored in the turbine, the moisture separators and the related piping.

The resulting maximum overspeed calculated in this manner was nominally 131%. Westinghouse P_1 values are for 132% overspeed, which considers the uncertainties in valve characteristics and variation in closing times.

3.0 Calculational Methods

3.1 Methods Overview

The general category of turbine missile codes used in the turbine missile analysis are designated MIS (Missile Impact Simulation). The analysis is an extension of the work embodies in the code MIDAS [(3)] written for Offshore Power Systems and the code MISPGE written for Portland General Electric [(4)]. A number of improvements in the options and calculational procedures have been incorporated for this analysis.

For designated convenience, the code used in this study is called MISIP (Missile Impact Simulation of Indian Point). It differs from the other codes primarily in the incorporation of the Indian Point 3 plant model.

The basic procedure involves the tracing of individual missile trajectories by the following sequence:

- Determination by Monte Carlo methods of the initial missile velocity and direction from the respective ranges given;
- Calculation by equations of free-flight ballistics of the missile strike locations on the walls of the plant;
- Determination by Monte Carlo methods of the projected area with which the missile impacts the wall;
- Calculation by empirical relations of the missile-barrier interaction effects;
- Calculation by energy balance and Monte Carlo means (for missile direction) of the missile state following the interaction;
- Termination of the missile trace if the missile ricochets more than three times consecutively in the same room* or exits the plant in a direction which precludes its return or its striking an adjacent plant.

All safety related component rooms (targets) penetrated during the flight of the missile are recorded. Computer output normally lists the number of discs allowed to break up, the number of times each target is struck and penetrated by each breakup, and the characteristics (type and final energy) of the missiles which hit the targets. Detailed traces of each missile flight may be printed out, and any desired interim data are available. From the code output the probability of safety compromise is determined. The specific combination of failures, which result in safety compromises are examined, and compromises of those functions which are covered by procedures for achieving and maintaining a safe shutdown without the given function are considered to have a negligible probability of occurrence.

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3.2 Initial Conditions for Turbine Missiles

Starting conditions required for turbine missile calculations are:

- Missile mass,
- Starting coordinates,
- Initial direction, and
- Initial magnitude of velocity.

In addition, the user inputs the number of disc segments that the simulated disc failure will result in. The analysis looked at breakup into either four 90° segments or three 120° segments because these cases provide the most risk. Also, each segment is assumed to produce two additional fragments as they break through the turbine casting. Figures 14B-1 and 14B-2 show missile shapes for 90° segments. The shapes for 120° segments are similar.

Missile dimensions, mass, and velocity were provided by the Westinghouse Electric Products Division [6, 7] for this analysis. Data was provided for each disc segment and fragment, for breakup into 90° segments and 120° segments, and for breakup at rated speed and 132 percent overspeed. A sample of this data is shown in Table 14B-2, excerpted from Reference 5.

Figure 14B-3 indicated the coordinate system utilized in the calculations and the relationship between the turbine missile initial velocity, the position vector and its components parallel to each of the plant Cartesian coordinates. References 5 and 6 give the azimuthal (ϕ) range and rotational (θ) range of turbine missiles. Inner discs emerge with a θ range uniformly distributed between –5° and +5°. End discs (two per turbine) emerge uniformly between 5° and 25° measured outward from the center of each turbine. $\theta = 0$ corresponds to missile emergence perpendicular to the turbine axis. The rotational range is from 0 to 360 degrees, but the turbine pedestal and condenser stop any missiles emerging in a downward direction. The turbine base prevents missile emergence at angles requiring penetration of this region.

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*This procedure is equivalent to a low-energy termination, but is more convenient as it obviates the need for keeping track of a potential energy reference frame.

This exclusion is justified with the assistance of Figure 14B-4.

In Figure 14B-4, the missiles generated by the breakup of disc 2 are illustrated (two of four sets for a 90° breakup). The disc sectors are located accurately in the radial dimension (prior to breakup). The location of blade ring fragments is assumed. The sizes of fragments are typical for all discs. It is obvious from the figure that only the smallest fragment could possibly exit the base region of the casting without interacting with one or both of the horizontal base plates on the top and bottom of this region. Such as exit is considered impossible for even the smallest fragment because it has a rotational component and could not maintain the precise orientation required to avoid these base plates during the perforation process. These base plants are heavy steel (three inches thick on the top and six inches on bottom). Minimum width (on top) is eight inches. The bottom plates are much wider. These plates are separated by weld gusset plates and are also welded to the outer turbine casing. A missile exiting this region would have to tear these plates or separate them. The added energy required over that to perforate the casting is so great that the probability of emergence in this region is considered negligible.

The range of θ in Figure 14B-3 that excludes the turbine base and the flight paths below horizontal is from 100° to 285°.

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3.3 Missile Trajectories and Strike Locations

The equations of free-flight ballistics (Neglecting air resistance) are used to determine trajectories and strike locations on given plant walls. Exact details are contained in Reference 7. The code keeps track of each cell, or room, in which the given missile is located and determines the next wall struck by solving the velocity equations (in component form) for the minimum time to strike one of the enclosing walls. The minimum time replace in the original equations locates the strike point. Special routines are called if the missile enters "cells" which contain either the containment or the moisture separator reheaters, which are treated as right circular cylinders.

3.4 Missile Interaction With Walls

3.4.1 Concrete Wall Interactions

The only walls which offer significant resistance to the missiles are concrete walls. The effect of walls such as office room partitions and corrugated metal is neglected.

This study used a formula for concrete penetration which was derived on the basis of tests performed by the Commissariat a' l'Energie Atomique-Electricite de France (CEA-EDF) (8). The Electric Power Research Institute recommends this formula as providing the best match to experimental data over a full range of missile velocities (9). The formula is

$$T_P = 0.765\,(\sigma_c)^{-}\left({}^{W}/_{D}\right)^{1/2} V_i^{3/4} \qquad \text{[Equation 3.4-1]}$$

Where:

T_P = thickness of wall that is penetrated 50 percent of the time for given missile (in)
σ_c = concrete compression strength (psi)
W = missile weight (lb)
D = effective diameter (in)
D = $2\sqrt{A/\Pi}$ where A represents an effective impact area
v_i = incident velocity (ft/sec)

The barrier penetration for a given missile depends upon the combination of impact area and velocity. In the analysis, the MISIP code compared the actual thickness (T) of the barrier with T_P to determine whether the missile penetrates the barrier. The velocity (V_P) required to penetrate the barriers is calculated by a rearrangement of equation (3.4-1), if T_P (calculated from equation 3.4-1) exceeds T. The velocity (V_P) required to penetrate the barriers can be found by:

$$V_p = (T_p)^{4/3}\;(\sigma_c)^{1/2}\;(D/W)^{1/3}\;\;0.765 \qquad [3.4\text{-}2]$$

The missile state following penetration is determined by the residual velocity after penetration, V_r :

$$V_r = \left[\frac{W}{W+W_w}\left(V_i^2 - V_p^2\right)\right]^{1/2} \qquad [3.4\text{-}3]$$

Where:

W_w = weight of wall plug removed (lb) = $\pi \dfrac{(1.4\,D)^2}{4} P_c$
P_c = density of concrete = 0.086 lb/in ^{3}

The wall thickness used in Equations 3.4-1, 3.4-2, and 3.4-3 is that parallel to the velocity vector (actual thickness divided by the cosine of the obliquity angle). If the missile penetrates the wall, it is assumed to continue without alteration of its original direction. The containment is a special case, in that the trace is terminated if the containment is penetrated, and the possibility of safety compromise

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examined separately from the program. (Containment penetrations are not observed with simulated missile traces for Indian Point 3).

Following missile ricochet, the MISIP code calculates the missile velocity and angle of obliquity (with the normal to the surface at the point of contact). The ricochet model [10] is based on converting only elastic strain energy stored locally in the concrete wall and in the missile, when the normal velocity becomes 0, to kinetic energy of the rebounding missile. The overall structural response of the wall is ignored. The rebound contribution from the overall structural response of the wall is assumed to be manifested later in time than the local response. The elastic energy available (>) is estimated as:

$$> = \frac{2V\ \sigma_c^2}{6E_s} + \frac{2LA\ \sigma_c^2}{6E_c} \qquad [3.4\text{-}4]$$

Where:

V	= missile volume (in) ^{3}
σ_c	= compressive strength of concrete
$E_s\ E_c$	= modulus of elasticity in steel and in concrete, respectively
L	= thickness of concrete barrier (in)
A	= impact area (in) ^{2}

In experimental work at Calspan [11] the rebound velocities of steel missiles ricocheting form concrete walls were measured. The rebound velocity is:

$$V_{rebound} = \frac{2>}{M}^{1/2} \qquad [3.4\text{-}5]$$

Where > is given by Equation 3.4-4 and M is the missile mass.

From the penetration equations (3.4-1 and 3.4-2) it is obvious that a significant parameter is the projected area of the missile (that projected area perpendicular to the velocity vector). These area values are picked at random from a uniform distribution between an assumed maximum and minimum calculated from data provided by Westinghouse [5, 6].

3.4.2 Steel Wall Interactions

The moisture separator reheaters (MSR's) interact significantly with the missiles because they are located adjacent to the turbine at the same elevations. The MSR's are steel. Steel wall interactions differ significantly from concrete and their specification is somewhat complicated. Briefly, the steel interactions are based on a method derived from experiment by Hagg and Sankey [12] who show that steel perforations can occur in one of two stages. In the first stage, or phase, the resistance of the barrier to perforation is provided only by local compression and shear in the wall because there has been no time for a tensile wave to propagate in the plane of the barrier at significant distances from the interaction point. In the second stage, the barrier "stretches" in its plane perpendicular to the direction of missile penetration and tensile strength contributes significantly to perforation resistance. Perforation can occur in either stage. The perforation conditions and expressions for residual velocity and the ricochet conditions are somewhat unwieldy and hence are not listed here (see Reference 13 for a full derivation). The equations for perforation are based upon the missile mass, velocity, and impact area, and the thickness of the wall. The equations are more complicated than those for concrete because of the inclusion of local compression and shear for the two-phase perforation process.

3.5 Plant Layout and Safety Design

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Figure 14B-15 is an overall layout of Indian Point 3. The plant belongs to the general category designated as an "in-line" plant (Figure 9-6) wherein the turbine orientation allows direct hits from low trajecting missiles on vital plant components in case of turbine failure. Low Trajectory Missiles (LTM's) are whose incident velocity is less than 45° with the horizontal. High Trajectory Missiles (HTM's) emerge at angles greater than 45° with the horizontal. Any given point whose vector from the turbine axis is less than 45° with the horizontal and which is within the azimuthal and distance range of the missile can be struck by either an LTM or HTM. Protection against LTM strikes on critical components can be afforded by geometric arrangement as in the "peninsula" arrangement shown in Figure 14B-6, and this is, the motivation for such plant arrangements. However, for HTM's, any point within the distance range is potentially vulnerable to a missile strike.

Generally, the higher velocity LTM's present the greater risks for this category of missiles because of their greater penetration capacity, but for HTM's, the lower velocity missiles may present the greater risks because their limited distance range produces a greater number of strikes per unit area.

Indian Point 3 is unusual among in-line plants in that it is also less vulnerable to the highest velocity LTM's. This effect is caused by the separation between the turbine and the major safety areas which can be struck by LTM's (the primary auxiliary building and the water intake regions). LTM's must have an initial trajectory above horizontal to clear the turbine pedestal and turbine base.

Those with higher velocities also clear critical safety regions, but the lower velocity ones do not. For typical pants, whose safety regions directly abut the turbine hall, essentially all LTM trajectories intersect these regions and the higher-velocity missiles, of course, cause more damage.

The general model of the Indian Point 3 plant was divided into several rooms or compartments to form the input for the MISIP code. All actual walls were modeled as is. Additionally, many large rooms were divided by imaginary "air" walls into smaller compartments. An example is a large room with a pump on one side. In the model, such a room was divided into two compartments, one containing the pump and the other containing essentially empty space. Thus, if a simulated turbine missile entered this pump room, a determination could be made as to whether the pump itself would have been struck, or if the missile landed harmlessly in the empty part of the room. Also, rooms with redundant equipment were compartmentalized in order to determine whether all or part of the equipment in the room would be struck by missiles.

The plant is modeled in a Cartesian coordinate system with the exception of the reactor containment and the moisture separator reheaters which are modeled (as they truly are) as concrete and steel cylinders, respectively. The code accepts as plant input one line of data for each room of the plant. The line contains the location, thickness and material specifications of each of the six walls. An additional designator for exterior plant walls indicates the potential fates of missiles which exit these walls. These fates are designated:

R – Exit through the roof (roof exits include those missiles which may fall back on plant),
G – Exit through a floor slab,
E – Exit through an exterior wall below grade level,
L – Exit through X_{min} side of plant,
W – Exit through X_{max} side of plant,
D – Exit through Y_{min} side of plant,
U – Exit through Y_{max} side of plant.

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Rectangular parallelepipeds enclose the cylindrical containment vessel and the moisture separator reheaters. These are fictitious cells which serve as a convenient artifice to facilitate calculating missile interactions with a cylindrical surface in an otherwise all-Cartesian system.

The resulting model contained 152 compartments designated as targets or safety regions, and 377 non-safety regions. Table 14B-3 describes each of the safety regions. This table relates each region to regions of the plant fire protection system. Redundancy considerations, as discussed in the next section are also listed.

Figure 14B-7 is an example of a computer-drawn plot of an elevation slice of the plant, showing how the model was compartmentalized. Safety regions are shaded. Each rectangle is assigned a room number. Some of the more important hard-to-read room numbers are described on the side of the figure.

3.6 Redundancy of Plant Shutdown Equipment

The safety of the plant following a turbine missile event is dependent upon the ability to safely shut down and maintain the core in a coolable configuration. In this analysis, it is that the cold or hot shutdown options are to be maintained. For this requirement, the following systems, along with their power sources and power and control activities must be maintained:

1. Reactor Control System (including control room, associated equipment and cabling);

2. Primary Cooling Systems (including boron control and makeup water);

3. Secondary Cooling Systems (following a turbine missile event, either the auxiliary feed system or the turbine bypass systems must be available and for the long term, the residual heat removal system must be available);

4. Component Cooling System.

All of these systems have redundant components, controls, and power circuitry so that no failure of a single item will compromise the ability to maintain a safe shutdown condition. These components and their redundancies are described in Table 14B-3. Some areas, such as the piping penetration area (fire protection region 59A), contain equipment too close together to model redundancy. In this analysis, strikes on such areas were considered to compromise plant safety.

Particular attention was paid to the service water system, which is a common target for turbine missiles. The plant has a backup service water system which can be used when the main service water system is unavailable. In order, to be conservative, P_4 values were first calculated without taking credit for the backup system. Then P_4 values were recalculated assuming the backup system could be used in an emergency. This is an example of an emergency procedure which allows the plant to achieve a safe shutdown despite the loss of a critical system. Results for each of these cases are discussed in Section 4.0.

4.0 Results of turbine Missile Analysis

4.1 Probability of Safety Compromise

Using the MISIP code, a series of computer simulations were performed on the generation of turbine missiles. The 36 turbine discs were analyzed under various conditions. To provide good statistical

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accuracy for the calculation of probabilities, 2000 simulations (trials) per disc were performed. To reduce computer time, only the 18 most important discs were analyzed. These 18 discs were chosen from an examination of the P_1 values and from the results of 100 trials for all discs. After choosing the 18 discs for further study, 2000 trials were performed for the following cases:

1. Breakup into 90° segments at rated speed
2. Breakup into 90° segment at 132 percent overspeed
3. Breakup into 120° segments at rated speed
4. Breakup into 120° segments at 132 percent overspeed

Tables 14B-4 and 14B-5 summarize probabilities of penetration and safety compromise for these discs. Table 14B-4 summarizes the results for the 90° breakup case and Table 14B-5 shows the results for 120° breakup. The probabilities were calculated using P1 values from Table 14B-1 for a 5-year inspection interval. Table 14B-6 summarizes the location of strikes on safety regions according to their designation in the MISIP code and their physical description in the plant.

The overall results of the simulation are given below for 90° and 120° segments, and for failure at rated speed and at 132 percent overspeed. The P_4 values are the probabilities of safety compromise, and utilize the redundancy considerations discussed in Section 3.6. Redundancy was considered in two ways. First, credit was not taken for the backup service water system, so that missile strikes which disabled the main service water system were considered safety compromises. Then the backup service water system was included, resulting in fewer safety compromises. The greatest concern is disc failure at rated speed into 90° segments. The probability of compromising plant safety by this mechanism is 1.70×10^{-8}/year, if no credit is taken for the backup service water system.

<u>P_4 Values with 2000 Trials for Five-Year Inspection Intervals</u>

	No Credit for Backup SW	**Credit Taken for Backup SW**
90° segments, rated speed	1.70E-8/yr	7.47E-9/yr
90° segments, 132 percent overspeed	8.14E-11/yr	4.73E-11/yr
120° segments, rated speed	8.40E-9/yr	1.33E-9/yr
120° segments, 132 percent overspeed	4.57E-11/yr	2.09E-11/yr

4.2 <u>Plant Vulnerability</u>

As noted previously, not all penetrations of safety regions result in compromising the ability of the plant to achieve safe shutdown. The specific strike combinations which produce the safety compromise probability are summarized below for the various cases. In this summary, credit is taken for the backup service water system.

<u>90° segments, rated speed</u>

- Strikes on common electrical penetration area (MISIP target 226):

Disc 8 - 5 strikes
Disc 9 - 2 strikes
Disc 10 – 2 strikes
Disc 16 – 1 strike
Disc 17 – 1 strike
Disc 21 – 1 strike
Disc 25 – 2 strikes

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- Strikes on the common cable spreading room area (MISIP target 529):
  Disc 22 – 1 strike

- Strikes on the switchgear room, buses 3A, 6A:
  Disc 36 – 1 strike

- Strikes on both the main service water pipes under the road and the backup service water valve pit area (MISIP) targets 516 and 198):
  Disc 22 – 1 strike

<u>90° segments, 132 percent overspeed</u>

- Strikes on the common electrical penetration area (MISIP target 226):
  Disc 9 – 1 strike
  Disc10 – 4 strikes
  Disc 16 – 1 strike
  Disc 17 – 1 strike
  Disc 25 – 1 strike

- Strikes on the common cable spreading room area (MISIP target 529):
  Disc 33 – 1 strike
  Disc 36 – 2 strikes

- Strikes on the piping penetration area (MISIP target 135):
  Disc 16 – 1 strike
  Disc 17 – 1 strike
  Disc 20 – 2 strikes
  Disc 22 – 1 strike

- Strikes on both the lower and upper electrical tunnels (MISIP targets 64 and 488):
  Disc 4 – 1 strike

<u>120° segments, rated speed</u>

- Strikes on the common electrical penetration area (MISIP target 226):
  Disc 5 – 1 strike
  Disc 10 – 1 strike
  Disc 17 – 2 strikes

- Strikes on the common cable spreading room area (MISIP target 529):
  Disc 36 – 2 strikes

- Strikes on the switchgear room, buses 3A, 6A
  Disc 10 – 1 strike

<u>120° segments, 132 percent overspeed</u>

- Strikes on the common electrical penetration area (MISIP target 226):
  Disc 5 – 2 strikes
  Disc 8 – 1 strike

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Disc 10 – 2 strikes
Disc 25 – 4 strikes

- Strikes on the common cable spreading room area (MISIP target 529):
  Disc 36 – 2 strikes

- Strikes on the piping penetration area (MISIP target 135):
  Disc 17 –1strike
  Disc 20 – 1 strike

As previously noted, there are several additional safety compromises if credit is not taken for the backup service water system. The vast majority of the cases involve strikes on the service water system piping under the road (MISIP target 516).

For all discs there were 87 such strikes for 90^{o} segments at rated speed, 99 strikes for 90^{o} segments at overspeed, 58 strikes for 120^{o} segments at rated speed, and 43 strikes for 120^{o} segments at overspeed. Since these pipes are under approximately eight feet of dirt, the number of penetrations of target room 516 was somewhat surprising. However, further study showed that the number of missiles which bounded off the road without penetrating the pipes below was much larger than the number of missiles which penetrated to the piping. For the 90^{o} segments, rated speed case, there were only 87 penetrations due to 5465 hits on the road, or 1.6 percent. Given the number of times that turbine missiles strike the roadway, the calculated number of penetrations is felt to be reasonable.

Other safety compromises involved strikes on the pipes inside the concrete bunker downstream from the road (MISIP target 517) and a few cases where as many as five of six service water pump motors were struck. For target 517, there were five strikes for 90^{o} segments at rated speed, seven strikes for 90^{o} segments at overspeed, one strike for 120^{o} segments at rated speed, and one strike for 120^{o} segments at overspeed. With regard to the service water pump motors, the safety compromise resulted from:

- <u>90^{o} segments, rated speed</u>:
  Disc 16 - motors 31, 32, 33, 34, 35
  Disc 17 – all motors
  Disc 20 – motors 31, 32, 33, 35, 36
  Disc 20 – all motors

- <u>120^{o} segments, rated speed</u>:
  Disc 20 – motors 31, 32, 33, 35, 36

Strikes on spent fuel storage were eliminated as possibly safety compromises because only HTM strikes are possible. To produce an offsite release from spent fuel storage a leak below the level of the stored elements would be required. It is concluded that an HTM strike cannot produce such a leak.

The possibility of simultaneous missile strikes on components of the main and backup service water systems was investigated. (Each disc breakup generated at least nine missiles, allowing the possibility of simultaneous strikes on more than one location). However, only one trial out of 144,000 resulted in simultaneous hits, so this is not a safety concern.

4.3 <u>Hazard Due to Turbine Missiles From Unit 2</u>

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An analysis of the hazard to Indian Point 3 due to turbine missiles emanating from Unit 2 was performed. A MISIP model for both units was prepared and the code was executed to track missiles form the Unit 2 turbines. One thousand trials for failure of disc 16 at rated speed, and 1000 trials for disc 17 at 132 percent overspeed were examined. Both cases considered 90° segments because an earlier study [14] found 90° segments to result in the greater hazard in Unit 2. The execution of these two cases resulted in no strikes on IP3. This is considered adequate demonstration that turbine missiles from IP2 do not pose a significant threat to IP3.

5.0 Assessment of Plant Capability to Withstand Postulated Turbine Missile

The previous section discussed the results of the turbine missile analysis in which the probability of compromising the safety of the plant was calculated. The risk was found to be concentrated in a small number of areas of the plant. In this section, the consequences of various turbine missile strikes will be discussed.

5.1 Consideration of Direct Loss of Reactor Coolant

The Reactor Coolant System is contained inside the Reactor Containment structure. This is a reinforced concrete vertical right cylinder with a flat base and hemispherical dome. A welded steel liner with a minimum thickness of ¼ inch is attached to the inside face of the concrete shell to ensure a high degree of leak-tightness. As shown in Figure 14B-8, the wall of the vertical cylinder is 4.5 feet thick and the dome is 3.5 feet thick.

In the turbine missile analysis, there were no simulated turbine failures that resulted in penetration of perforation of the containment. This is considered sufficient evidence that there is adequate protection against direct loss of reactor coolant.

5.2 Considerations to Maintain Plant In A Safe Shutdown Condition

Rupture of a low pressure turbine disc at speeds below the emergency overspeed setpoint will trip the turbine due to loss of condenser vacuum resulting from the damage produced by the ruptured disc. Rupture at or above this setpoint requires that turbine trip has occurred. Since the reactor trips automatically following a turbine trip, both turbine and reactor trip are assured in the event of the turbine missile incident. Hence, maintaining the plant in a safe shutdown condition requires only minimal performance of the decay heat removal, reactor coolant makeup and boration functions.

Before proceeding with the evaluation of the capability of maintaining the plant is a safe shutdown condition, the components related to the normal performance of these functions following a turbine trip and reactor trip will be identified, with due consideration given to system redundancy.

Decay Heat Removal

With sufficient fluid in the Reactor Coolant System, adequate decay heat removal depends on the performance of the steam generator secondary side since the core decay heat removal is assured by the circulating reactor coolant. In the first few minutes, the reactor coolant is circulated by mechanical coastdown of the reactor coolant pumps and subsequently by natural circulation. Decay heat removal from the secondary side depends on the steam relief system and the Auxiliary Feedwater System.

The steam relief system removes thermal energy by releasing steam to the atmosphere via the steam relief valves or the condenser via the turbine bypass. For the turbine missile incident, credit cannot be taken for the turbine bypass since the bypass valves will not open with loss of condenser vacuum.

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The steam dump to the atmosphere consists of five safety valves located on each of the four main steam lines outside the Containment and upstream of the no-return valves as illustrated in Figure 14A-9. The five safety valves in each main steam line are set to relieve at 1065, 1080, 1095, 1110 and 1120 psig, respectively. These twenty valves have a total capacity in excess of the equivalent nominal rated steam flow. In addition, there are four power-operated relief valves which are capable of releasing 10 percent of the equivalent nominal rated steam flow. These valves are automatically controlled by pressure or may be manually operated from the main control board or locally at the valves.

The Auxiliary Feedwater System supplies high pressure feedwater to the steam generators in order to maintain a water inventory for heat removal from the Reactor Coolant System upon inoperability of the Main Feedwater System. Upon loss of condenser vacuum, the valves in the lines supplying steam to the turbine drive of main feed pumps close automatically. Hence, the Auxiliary Feedwater System must come into operation following the turbine missile incident.

The Auxiliary Feedwater System is basically composed of:
1. Two motor-driven feedwater pumps
2. One turbine-driven feedwater pump
3. Auxiliary steam admission to the drive of the turbine-driven feedwater pump
4. Auxiliary feedwater discharge piping
5. Main feedwater lines
6. Auxiliary feedwater suction piping
7. Auxiliary feedwater source.

This system was sized so that any of the auxiliary feedwater pumps can supply the required auxiliary feed. These components, except for the auxiliary steam admission to the drive of the turbine-driven feedwater pump, are illustrated in Figure 14B-10 through 14B-15. Steam to drive the turbine is supplied from two of the main steam lines upstream of the stop valves just outside of the Containment. The turbine is started by the opening of the pressure reducing valve located in the auxiliary feedwater pump room. This valve opens automatically upon loss of power.

Reactor Coolant Makeup

Reactor coolant makeup is required to maintain sufficient fluid in the Reactor Coolant System to guarantee that decay heat is removed continuously from the core. At the same time, however, the boration concentration of the Reactor Coolant System should not be reduced substantially in order to maintain a sufficient shutdown margin. Hence, for the incident under consideration, the makeup source would normally be from the refueling water storage tank. Makeup from the refueling water storage tank involves the following components:

1. Refueling Water Storage Tank,
2. Discharge piping from the Refueling Water Storage Tank to the suction of the charging pumps,
3. Three (3) pumps (one is sufficient),
4. Discharge piping from the charging pumps,
5. Component Cooling system to provide cooling to the charging pump fluid drive, and
6. Service Water System to cool the component cooling water.

Boration

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Boration is required to compensate for the long-term xenon decay transient. The normal Boration system includes the following components:

1. Two boric acid tanks and boric acid batching tank and heaters,
2. Two boric acid transfer pumps (one is sufficient),
3. One boric acid filter,
4. Piping and heat tracing from the tanks to the suction of the charging pumps,
5. Three charging pumps (one is sufficient),
6. Discharge piping from the charging pumps to the RCS,
7. Component Cooling Water System to provide cooling to the charging pump fluid drive coupling, and
8. Service Water System to cool the component cooling water.

The areas of the plant related to the normal performance of decay heat removal, reactor coolant makeup and boration functions that were shown by the missile simulation to be vulnerable to turbine missiles are the service water system, the auxiliary feedwater system, and the electrical penetration area. These areas are evaluated below as to their vulnerability and the identification of available backup systems and appropriate plant personnel actions. Other important areas, such as the steam relief system, the condensate storage tank and piping and the refueling water storage tank and piping, were not struck by simulated missiles, even after 144,000 trials.

5.2.1 <u>Service Water System</u>

The components of the service water system which were found to be vulnerable were the service water pipes as they travel under the roadway and through the concrete bunker in the discharge canal, and to a lesser extent the pump motors. Figure 14B-16 is an overview of the service water system. Figure 14B-17 shows how this system was represented in the MISIP code plant model.

The pipes under the roadway were found to be vulnerable to HTMs although only approximately 1.6 percent of HTMs which struck the road were able to penetrate to the piping. The pump motors were found to have sufficient separation, except in a very small number of trials, to preclude disabling the system. The system can operate with tow of three pumps in either train, so it would be necessary to damage at least two of three pumps in both trains. Figure 14B-18 shows the MISIP model for the pump motor enclosure.

As was discussed earlier, a backup service water system exists to replace the main service water system during emergencies. Only one case in 144,000 resulted in simultaneous missile strikes on components of both systems.

If all sources of service water were unavailable, the affected functions would be reactor coolant makeup and boration. Service water is required for cooling the diesel generators and removing heat from the component cooling system which cools the fluid drive coupling of the charging pumps.

As far as the diesel generators are concerned, they are not considered vital for safe shutdown since there is sufficient amount of time, about 21 hours to restore outside power in case it is lost following turbine disc rupture (see Section 5.3). Cooling to the charging pumps will be accommodated by making up spool piece connections to the charging pump cooling water heater that will allow direct cooling via the city water supply. Initial drainage flow would go to the floor drain and would be eventually piped outside the building. The operators will have sufficient time to open and close the manual valves as required and to make the necessary piping connections (see Section 5.3).

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5.2.2 Auxiliary Feedwater System

As illustrated in Figures 14B-10 through 14B-12, the auxiliary feedwater pumps, located in the auxiliary feedwater pump room, have the four-foot thick shield wall to stop a LTM and two floors of two-foot thick concrete to stop the HTM. In fact, no missile strikes were observed in the pump rooms during the analysis.

The present auxiliary feedwater discharge piping is illustrated in Figures 14B-13 and 14B-14. It is conceivable even though very unlikely, that the HTM could fall between the steam lines and possibly damage all four auxiliary feedwater pipes. The area of concern is the piping run from the point they come up through the second concrete roof to the second main feedwater connection. The MISIP model separated each of the four pipes into separate cells. No cases were observed in which more than two of these cells were penetrated by missiles. If a missile lands on the auxiliary feedwater pipe runs, the ruptured lines can be isolated by closing valves located in the protected auxiliary feedwater pump room. Auxiliary feedwater to two steam generators provides adequate cooling.

A portion of main feedwater lines is required for the introduction of feedwater since the auxiliary feedwater lines are connected individually to these lines near the containment wall. Only two of these main feedwater lines need to be intact for the reasons discussed before, and this will be ensured by their separation.

5.2.3 Electrical Penetration Area

This area is shown in Figures 14B-19 through 14B-21. Missile strikes were observed due to HTMs breaking through the roof, which is light weight concrete with no reinforcement and has a maximum thickness of eight inches. The MISIP model for the upper electrical penetration area is shown in figure 14B-22. It shows that there is an area in which cables from the lower electrical tunnel rise and join cables from the upper tunnel in a common penetration area. Strikes in the area were observed, and could cause some loss of instrumentation and control. However, due to physical separation at the penetration, (see Figure 14B-21) loss of all pressurizer pressure and level channels and loss of all steam generator level channels is unlikely.

5.3 "Worst Case" Turbine Missile Accident

From the standpoint of maintaining the plant in a safe shutdown condition, the worst case accident is considered to be a loss of outside power coincident with the turbine disc rupture in which the missile makes the Service Water System inoperable. This implies that the only power available is from the station batteries (protected by the concrete structures of the control building) since the diesel generators require cooling by the Service Water system for their operation.

This section examines the above accident in two parts. First, conservative assumptions were used to establish estimates of the time available to perform manual actions that will ensure adequate decay heat removal, reactor coolant makeup and boration. The second part deals with defining the specific operator actions.

5.3.1 Time Requirements

In estimating the time available for manual actions it was conservatively assumed that valves which require operator action to change their state initially remain in their state at the time of the accident. However, those possible states that normally do not persist for extended periods were not included as possible states at the time of the accident. For example, the normal and excess letdown isolation

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valves were assumed to be open since they can be opened during periods where the plant is changing power. Alternately, the power operated pressurizer relief valves are normally not exercised during operation, and as such, they were assumed to be closed. The valve status shown in Table 14B-7 was assumed in this evaluation. Within the framework of these assumptions and the corresponding valve status, analyses were performed to establish the time available for manual actions and restoration of outside power. This is summarized as follows:

1. Decay Heat Removal

   To ensure the decay heat removal function, feedwater must be provided to the steam generators within 30 minutes of reactor scram if steam generator boil-off is to be avoided. This will be assured because of the following:
   a. The turbine-driven auxiliary feedwater starts on the automatic opening of the pressure reducing valve in the steam admission line to the drive of this pump;
   b. The auxiliary feedwater control valves are open (normal position); and
   c. The source of feedwater is gravity fed to the suction of the pump from the normally aligned (locked open gate valves) Condensate Storage Tank.

As indicated in Table 14B-7, the steam generator blowdown valves are assumed to be closed and the sample lines is small compared to the outflow from the safety valves and become even less important when the turbine-driven auxiliary feed pump is started. Thus, the open status of these valves is not critical but should be closed in about three or four hours to limit unwanted secondary side losses.

While adequate steam generator level is assured by operation of the turbine-driven auxiliary feedwater pump, there could be carryover from the steam generator turbine drive of this pump if left unattended. It is estimated that full feedwater flow from this pump would not produce water carryover within the first two hours of operation. Hence, within about two hours, it would be prudent to have the Auxiliary feedwater System under control by the operators. Steam generator level will be available via the battery-powered instrument buses.

2. Reactor Coolant Makeup

In estimating the minimum time required to provide reactor coolant makeup, two cases were considered without charging or safety injection: the first corresponding to reactor coolant discharge from the normal letdown line, the excess letdown line and through the seals of the four reactor coolant pumps; and the second case with leakage only through the seals of the reactor coolant pumps. The total initial leakage rate for the first case is approximately 260 gpm and 40 gpm for the second.

The SLAP code employed in this analysis resulted in core uncoverage in approximately seven hours in the first case and 40 hours in the latter.

The rates of decrease of reactor coolant volume in both instances were found to be nearly linear in time. Hence, assuming it took the operators as long as two hours to close the required valves, it would require an additional 28 hours to uncover the core with the seal pump leakage. This means that reactor coolant makeup would not be required for over 30 hours.

2. Boration

To estimate the minimum time required to compensate for the xenon transient, it was assumed that at the time of the turbine incident, the xenon decay is at its maximum rate of approximately 0.13 percent per hour. This would correspond to operating the plant for a long period of time (xenon at its

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equilibrium value), going to a hot shutdown condition for 10 to 14 hour period and then returning to full power.

For the end-of-life (EOL) conditions, there is a 1.72 percent minimum shutdown margin requirement (assuming a stuck rod). At EOL, all rods inserted less the highest worth rod stuck out have a design worth of six percent. This is reduced by ten percent (0.6 percent) to satisfy nuclear design criteria leaving a worth of 5.4 percent.

The required margin of 1.72 percent must be subtracted (5.4-1.72) leaving 3.68 percent worth available at accommodate any transient. Nuclear calculations on the depressurization transient result in a negative reactivity of 2.8 percent. Thus, for the end-of-life core with a stuck rod, a total of 3.68 percent negative reactivity would exist during the period of time without makeup or boration capability. With the maximum positive reactivity addition rate of 0.13 percent per hour associated xenon decay, it would require an increase of 200ppm boron to compensate for the positive xenon reactivity. Borating from the boric acid tanks would take approximately 30 minutes to change the boration concentration by 200 ppm.

Taking into account the decreasing rate of positive reactivity addition at the time of the xenon decay, it would require more than 36 hours for the reactor to return to critical. Prior to this point in time, it would require an increase in boron concentration of about 100 ppm. This change in concentration could be made in about 15 minutes by borating from the boric acid tanks.

While boration capability itself is not required for more than 21 hours, the ability to borate from the boric acid tanks depends on the solubility of the borated water in the boric acid tanks and the piping from these tanks to the suction of the charging pumps. The tanks themselves would not freeze up before four or five days without power to the heaters. The borated water in the piping, on the other hand, could freeze within about an hour without power to the heat tracing. Hence, to assure borating capability at the required time, these lines must be flushed with clean water within this hour period. To accomplish this flushing action, use will be made of the existing primary water flushing provisions. These provisions will be further augmented by installing a cross connection (at the discharge of the boric acid tanks) to the city water supply.

Summarizing, the minimum time requirements are:

FUNCTION	REQUIREMENT	TIME
Decay Heat Removal	Feedwater to steam generators*	½ hour
Boration	Flushing of piping between boric acid tanks and charging pumps	1 hour
Decay Heat Removal	Control of the Auxiliary Feedwater System	2 hours
Decay Heat Removal	Closure of steam generator sample isolation valves	3 hours
Makeup	Closure of normal and excess letdown isolation valves	2 hours
Boration	Boration via boric acid tanks	>21 hours
Makeup	Charging Capability	30 hours

*Feedwater to steam generators via the turbine-driven auxiliary feedwater pump is assured within a few minutes since operator action is not required.

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5.3.2 Operator Actions

Execution of the operator actions to follow will ensure that the decay heat removal, reactor coolant makeup and boration functions can be performed when required and the plant can be maintained in a safe shutdown condition for an excess of 24 hours without any power sources other than the station batteries. However, it should be noted that shutdown could be maintained even in the event these batteries become depleted before outside power is restored.

These actions pertain to those required in the first few hours following the postulated impact of the missile on the Control Room:

1. The operators have approximately 30 minutes to make sure there is sufficient auxiliary feedwater available (approximately 400 gpm). To be assured of this, the following steps are necessary:
    a. Check that the pressure reducing valve on the steam admissions line to the turbine-driven auxiliary feedwater pump is open (located in the Auxiliary Feedwater Pump Room). This valve should automatically open upon loss of power. If this valve is closed, it can be opened manually.
    b. Check that the auxiliary feedwater control valves are open. These are air-operated valves and fail open upon loss of air. These valves, located in the Auxiliary Feed Pump Room, can be opened manually.

2. To make sure that the inventory of water in the steam generators remains sufficient, within about one hour the operators must:
    a. Check that the steam generator blowdown valves are closed (these valves may or may not be open depending on the mode of plant operation). These valves are air-operated and fail closed.
    b. Close the normally open steam generator sample isolation valves (these valves are also air operated and fail closed).

3. To assure that boration from the boric acid tanks can be accomplished when required (>21 hours), the operators have approximately one hour to flush the piping between the boric acid tanks and the suction of the charging pumps. This can be accomplished by connecting the city water supply to the flushing valve at the discharge of the boric acid tanks and opening the drain valves at the suction of the charging pump. Once the flushing operation is complete, these valves can be closed and the lines will be available for the boration operation.

4. To prevent water carryover from the steam generators to the drive of the turbine-driven auxiliary feedwater pump, within about two hours, the operators should have control of this system. Control can be maintained from the Control Room by remote manual operation of the air-operated auxiliary feedwater control valves. Steam generator level indicators, powered by the battery-operated instrument buses, are available in the Control Room. Flow measurement devices are installed in the discharge lines to each steam generator with indicators, also battery-operated, on the control board. The instruments provide the operator with the information necessary to properly route the discharge flow through the remote manual auxiliary feedwater flow control valves.

5. Within about two hours the Reactor Coolant Pressure Boundary must be assured. This can be accomplished from the Control Room by remote manual operation of the air-operated normal and excess letdown line isolation valves. There is no reason why these lines could not be isolated sooner; the two-hour time period is given as an estimate for how long these valves could remain open without presenting any particular problem in maintaining the plant in a safe shutdown condition.

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6. Before boration and reactor coolant makeup can be initiated via a charging pump in the specified time period (21 hours for borating and 30 hours for makeup), cooling to the drive of the charging pumps must be provided. As discussed earlier, this cooling is accommodated by making permanent connections to the charging pump cooling water header that allows drive cooling from the city water supply. To make use of these permanent connections, the operators must attach a spool piece to the permanent flange connection upstream of the new isolation valves on the inlet side of the charging pump cooling water header. Cooling can be initiated by closing the existing cooling water header isolation valves and opening the new isolation and drain valves.

Upon completion of these six steps, the city water supply should be aligned to the suction of the auxiliary feedwater pumps since the Condensate Storage Tanks contain a limited supply of water (24 hours minimum). The air operated valve in the city water line can be opened remotely from the Control Room or from the Auxiliary Feedwater Pump Room with the nitrogen bottles located there. With the restoration of outside power, the plant can now be maintained in the hot shutdown condition almost indefinitely.

The operators may now take steps to align the equipment required for plant cooldown.

5.4 Consequences of Radioactivity Releases

The possibility of turbine missile causing release of fission product activity has been considered. The Reactor Coolant System is protected by the Containment. The gas decay tanks, the volume control tank, and demineralizers are protected by the auxiliary building structure. The liquid holdup tanks are protected by the holdup tank vault structure.

To produce an offsite release from spent fuel storage, a leak below the level of the stored elements would be required. It is concluded that a HTM strike cannot produce such a leak. However, should a HTM land in the spent fuel pit pool from above, damage to some spent fuel assemblies would occur. The impact area of a quarter disc would affect several storage cells. The analysis considered damage to one row around four cells with a maximum of 18 cells damaged. Damage to fuel in these cells would not result in criticality. Although no credible release mechanism is envisioned, doses due to the release of this material were calculated. For the purpose of determining the limiting site boundary dose, it was assumed that these assemblies are all freshly removed from the core having decayed only 100 hours since plant shutdown.

Two cases were considered:
1. An expected case, in which the expected characteristics of the six highest rated assemblies normally to be discharged at end-of-life are assumed, along with best estimate behavior or fission products determined by tests.

2. A design care, in which factors are introduced to allow for uncertainties.

The expected case is summarized is Table 14B-8. The resultant maximum site boundary doses were calculated to be 16.5 rem thyroid and 2.0 rem whole body. The design case is summarized in Table 14B-9. The resultant maximum site boundary doses were calculated to be 57 rem thyroid and 4.2 rem whole body.

Assuming a turbine missile is ejected, the probability of it hitting the fuel pool was calculated to 3.2 x 10^{-4}.

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Should a turbine missile hit the vicinity of the steam lines, no more than two steam lines could be damaged. Activity release would be dependant on RCS activity from operation with fuel defects and steam generator tube leakage, if any, during the period to cool and depressurize the RCS after the accident.

With RCS activity concentration corresponding to operation with one percent clad defects and a 10 gm tube leak for eight hours, the released activity from the RCS leakage for the duration of the accident is 4570 Ci equivalent Xe-133 and 4.6 Ci equivalent I-131. In addition, the iodine activity in the two steam generators which blowdown is 5.6 Ci equivalent I-131. The site boundary dose would be 5.4 rem thyroid and 0.2 rem whole body.

The Refueling Water Storage Tank and the monitor tanks, with low probability, may be struck by a turbine missile. The maximum tritium concentration in the Refueling Water Storage Tank should not exceed 2.5 uc/cc corresponding to a total of 3300 Ci tritium in the tank. The maximum concentration of tritium in the river at Chelsea form a burst release of this tritium would be 7.5×10^{-7} uc/cc which is 2.5×10^{-4} MPC. The release of the activity contained in a monitor tank to the river is given in Table 14B-10. The resultant river concentrations at Chelsea are less than 10^{-7} MPC.

It is concluded that the probability of a turbine missile causing a large release of fission product activity is very low. Further, with worst case assumptions, the turbine missile would not cause offsite exposure in excess of the 10CFR 100 guideline.

6.0 Low Pressure Steam Bypass for Turbine Overspeed Protection

6.1 Description

The Low Pressure Steam Bypass System has been provided to ensure that turbine design overspeed will not be exceeded in the event of a complete loss of electrical load. For this trip, it is necessary to divert directly to the condenser a portion of the heat energy stored within the turbine system. (The design overspeed value would be exceeded if this energy were released through the turbines). The bypass system was designed to nuclear protection system criteria of redundancy, separation, and reliability. Cables associated with dump valves operation and position indication are color-coded with the appropriate color for that channel (i.e., Red – Channel I, White – Channel II) at intervals along the cable. Each cable has a specific path through the raceway system and is provided with permanent markers at each end, cross referencing the cable schedule. In addition, the raceways in the turbine hall were laid out and installed specifically for protection systems are contained in these raceways.

Reliability has been designed into the system, primarily through the separation of the actuating signals, the multiplicity of dump valves and steam dump routes and component redundancies. System performance is assured in the event of a single failure.

In operation, the bypass system takes steam from the moisture separator reheater steam supply lines (cross-under piping), through six 10-inch bypass lines, three on either side of the turbine, which originates from the cross-under piping. Each bypass line has a normally closed 10-inch bypass control upstream. Each dump valves discharges into a 12-inch pipe, which, in turn, communicates with its associated condenser half-section through a breakdown orifice. The total bypass capacity of the Low Pressure Steam Dump System has been designed such that for maximum calculated gross electrical power, any four of the six dump valves will have sufficient capacity to relieve the amounts of steam needed for proper turbine speed control. Each of the dump valves is provided with redundant 3-way solenoid valves (i.e., “A” and “B” solenoids installed in the individual air supply lines).

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6.2 Operation

The low pressure steam bypass is activated on any unit trip signal (86P and 86BU relays) or any mechanical fault that initiates a turbine auto-stop signal (loss of auto-stop oil). The primary unit trip signal (86P) or the back-up unit trip signal (86BU) operate on any electrical fault. The auto-stop oil signals originate from a two-out-of-three matrix made up of contacts on control oil pressure switches.

An 86P or an 86 BU signal will initiate a circuit to energize the "A" solenoid, while an auto-stop signal will initiate a circuit to energize the "A" solenoid, while an auto-stop signal will initiate a circuit to energize the "B" solenoid, thereby causing the dump valve to open.

Through normally open during plant operation, the motor-operated isolation valves are used to isolate each associated dump line when testing of the dump valves is required, or when a dump valve is inoperable, or during that time when the associated main circulation pump is inoperable. This is done to preclude damage to a drained (uncooled) section of the condenser, should a dump valves open spuriously or otherwise. (Note: the turbine condenser is composed of six sections, each of which is separately cooled by its associated main circulating pump for a total of six pumps). The isolating valves are closed individually. Red and green indicating lights are located in the control / test panel to monitor the position of the isolation and dump valves. Also, valve position limit switches will provide an annunciation in the Control Room when any of the isolation valves leave their full open position. For occasions when less than six of the Low Pressure steam dump valves and dump lines are available, limitations on plant gross electrical output are established in the Technical Requirements Manual.

6.3 Testing Provisions

This system can be tested during power operation. The dump valves are to be tested individually and periodically in accordance with the Technical Requirements Manual. The signal reception and logic circuits are provided with test switches and indicating lights to permit individual testing of each channel. The actual operation of each dump valves is to be tested and monitored separately.

7.0 Conclusions

This analysis is based on protection from an offsite release. In this regard, the current design coupled with emergency procedures is certainly adequate. The overall probability of compromising the plant's ability to achieve a safe shutdown due to turbine missiles is less than the 10CFR100 acceptance criteria of 10^{-7} per year.

The area of the plant that is most at risk is the Service Water System. Of those simulated missiles that entered areas of the plant containing safety-related equipment, the majority entered Service Water System components. However, this should not be considered a source of concern because in only a few of these cases were a sufficient number of components damaged as to disable the system. Most of these were due to strikes on the piping under the road, while a few cases resulted in strikes on 2 of 3 pump motors in each safety train. Additional shielding in these areas might reduce the number of safety compromises, but with the low frequency of safety compromises and the number of safety compromises, but with the low frequency of safety compromises and the ability of the backup water system to perform the same functions, such shielding is not considered necessary.

Besides the Service Water System, the other major areas of concern involve locations where electrical cables from both safety trains share a common room. This was found is the cable spreading room, at the start of the electrical tunnels, and in the electrical penetration area. Also, in the

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switchgear room, the four buses are relatively close together. Again, because of the low probability of safety compromise, no remedial action is necessary.

In fact, the model is conservative in that is does not consider the shielding effect that the upper cable trays would have relative to the lower cable trays. Therefore, the probability of safety compromise may actually be lower than was calculated in this study.

In conclusion, it is felt that the results of this study show that Indian Point 3 plant is adequately protected from damage due to missiles generated by turbine failure. The probability that the plant will be unable to maintain a safe shutdown or that an offsite release will occur is well below the 10 CFR100 acceptance criteria.

References

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TABLE 14B-1

P_1 Valves for IP3 Turbine Discs

	100 Percent			132 Percent		
Disc Number	**18-Month***	**3-Year**	**5-Year**	**18-Month**	**3-Year**	**5-Year**
1	5.65E-10	4.60E-9	2.69E-7	1.19E-11	5.20E-10	1.47E-8
2	2.14E-14	3.58E-12	5.08E-10	1.02E-14	9.42E-13	6.48E-11
3	1.05E-13	1.43E-11	1.58E-9	1.27E-14	1.13E-12	7.38E-11
4	1.35E-8	4.84E-7	1.03E-5	4.43E-10	1.07E-8	1.41E-7
5	1.77E-8	6.15E-7	1.27E-5	3.76E-10	9.39E-9	1.29E-9
6	0	0	0	0	0	0
7	0	0	0	0	0	0
8	2.22E-9	2.25E-7	5.61E-6	1.28E-10	3.74E-9	6.21E-8
9	2.39E-8	7.92E-7	1.54E-5	7.30E-10	1.64E-8	1.97E-7
10	4.36E-12	4.06E-10	2.78E-8	3.83E-13	2.33E-11	9.46E-10
11	7.85E-12	6.78E-10	4.31E-18	5.00E-12	1.98E-10	4.97E-9
12	3.80E-13	5.32E-11	6.04E-9	1.02E-12	8.21E-12	4.79E-10
13	2.91E-13	4.24E-11	5.07E-9	6.50E-14	5.66E-12	3.59E-10
14	2.75E-16	6.77E-14	1.55E-11	3.07E-16	3.93E-14	4.04E-12
15	1.22E-14	2.05E-12	2.93E-10	1.95E-15	2.12E-13	1.76E-11
16	2.19E-12	2.38E-10	1.99E-8	1.51E-13	1.02E-11	4.78E-10
17	5.20E-11	4.22E-9	2.41E-7	1.26E-12	6.74E-11	2.36E-9
18	0	0	0	0	0	0
19	0	0	0	0	0	0
20	1.01E-10	7.79E-9	4.14E-7	1.79E-12	9.27E-11	3.12E-9
21	6.70E-12	6.61E-10	4.84E-8	3.79E-13	2.32E-11	9.49E-10
22	3.12E-12	3.00E-10	2.15E-8	2.81E-13	1.77E-11	7.52E-10
23	1.27E-13	1.75E-11	1.96E-9	7.35E-14	5.22E-12	2.62E-10
24	6.65E-11	5.42E-9	3.17E-7	9.65E-11	4.57E-10	1.40E-8
25	1.17E-12	1.46E-10	1.44E-8	2.67E-13	1.94E-11	9.94E-10
26	4.24E-14	6.53E-12	8.39E-10	2.59E-14	2.08E-12	1.21E-10
27	6.30E-15	1.14E-12	1.78E-10	9.75E-16	1.14E-13	1.04E-11
28	2.16E-8	7.26E-7	1.43E-5	6.40E-10	1.46E-8	1.80E-7
29	3.00E-8	9.75E-7	1.84E-5	6.05E-10	1.41E-8	1.78E-7
30	0	0	0	0	0	0
31	0	0	0	0	0	0
32	3.60E-8	1.14E-6	2.09E-5	6.85E-10	1.57E-8	1.93E-7
33	8.75E-9	3.33E-7	7.59E-6	2.98E-10	7.62E-9	1.08E-7
34	2.35E-15	4.66E-13	8.19E-11	3.97E-16	5.09E-14	5.18E-12
35	2.63E-13	3.36E-11	3.42E-9	1.51E-13	9.78E-12	4.40E-10
36	8.90E-11	8.91E-10	6.72E-8	2.10E-12	1.16E-10	4.32E-9
Total	**1.55E-7**	**5.16E-6**	**1.017E-4**	**4.02E-9**	**8.55E-8**	**1.01E-6**

*Length of turbine inspection interval.

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Table 14B-2

Exit Disc and Fragment Missile Properties for
Each Segment for LP-1 & LP-3 (Ref. 6)

90° SEGMENTS

		100 PERCENT RATED SPEED		132 PERCENT OVERSPEED	
MISSILE	DISC or FRAGMENT Weight (lbs.)	EXIT VELOCITY (ft/sec)	EXIT KINETIC ENERGY ($X10^6$ft/sec)	EXIT VELOCITY (ft/sec)	EXIT KINETIC ENERGY ($X10^6$ft/sec)
Disc No. 1 Fragment 1 Fragment 2 Fragment 3	2570 2550 2065 245	Contained	Contained	Contained	Contained
Disc No. 2 Fragment 1a Fragment 2a Fragment 1b Fragment 2b	2705 2825 350 2745 340	156 155 172 158 175	1.02 1.05 0.16 1.07 0.16	238 236 251 240 254	2.38 2.44 0.34 2.45 0.34
Disc No. 3 Fragment 1 Fragment 2 Fragment 2	3725 2955 390 145	217 217 206 -	2.72 2.14 0.26 -	311 311 - 489	5.59 4.43 - 0.54
Disc No. 4 Fragment 1 Fragment 2	3040 310 705	330 330 240	5.15 0.52 0.63	471 471 342	10.47 1.07 1.28
Disc No. 5 Fragment 1 (No Number 2)	3315 345 -	379 379 -	7.38 0.77 -	523 523 -	14.06 1.46 -
Disc No. 6 Fragment 1 Fragment 2	3905 375 1065	412 412 167	10.30 0.99 0.46	560 560 227	19.00 1.82 0.85

a LP-1
b LP-3
* Exit missile of less than 100,000 ft-lb are not reported.

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Table 14B-2
(Cont.)

Exit Disc and Fragment Missile Properties for
Each Segment for LP-1 & LP-3 (Ref. 6)

120^{O} SEGMENTS

		100 PERCENT RATED SPEED		132 PERCENT OVERSPEED	
MISSILE	DISC or FRAGMENT Weight (lbs.)	EXIT VELOCITY (ft/sec)	EXIT KINETIC ENERGY ($X10^{6}$ft/sec)	EXIT VELOCITY (ft/sec)	EXIT KINETIC ENERGY ($X10^{6}$ft/sec)
Disc No. 1	4325	Contained	Contained	Contained	Contained
Fragment 1	3400				
Fragment 2	2755				
Fragment 3	330				
Disc No. 2	3605	122	0.83	195	2.13
Fragment 1a	3825	121	0.87	193	2.21
Fragment 2a	470	142	0.15	211	0.32
Fragment 1b	3720	123	0.88	196	2.23
Fragment 2b	455	144	0.15	214	0.32
Disc No. 3	4970	176	2.39	260	5.21
Fragment 1	3970	176	1.91	260	4.16
Fragment 2	520	*	*	-	-
Fragment 2	195	-	-	*	*
Disc No. 4	4055	256	4.12	368	8.55
Fragment 1	410	256	0.42	368	0.86
Fragment 2	940	202	0.59	291	1.24
Disc No. 5	4415	313	6.71	430	12.66
Fragment 1	460	313	0.70	430	1.32
(No Number 2)	-	-	-	-	-
Disc No. 6	5210	356	10.25	479	18.59
Fragment 1	500	356	0.98	479	1.79
Fragment 2	1425	178	0.70	240	1.27

a LP-1
b LP-3
* Exit missile of less than 100,000 ft-lb are not reported.

IPEC00036618

Table 14B-3

Summary of Indian Point 3 Critical Area Modeling

Fire Protection Area Designation	**Missile Code Designation (System No.)***	**Remarks**
1 Component Cooling Pumps & Cabling	293, 294, 485 (1, 4)	Each pump is redundant
2 Containment Spray Pumps #31, 32	312, 482 (4)	Each pump is redundant
2A Primary Make-up Water System	298, 299, 309 (2)	Protected from missile strikes
3 RHR Pump #31	250 (2)	Redundant with other RHR pump
4 RHR Pump #32	252 (2)	Redundant with other RHR pumps
9A RHR Pump	249 (2)	Redundant with other RGR pumps
69A RHR Piping and Valves	251 (2)	
12A Valve Corridor	259, 260 (2)	
3A Piping Tunnel	295, 296, (2, 3, 4)	Protected from missile strikes
5A, 58A Piping Tunnels	263, 264, 265, 275, 279, 280, 281, 283, 284, 285, 286, 287, 288, 289 (2, 3, 4)	Protected from missile strikes
6A Valve Room	272 (2)	
7A Lower Electrical Tunnel	64, 67, 262, 267, 417, 489 (1)	Redundant with upper electrical tunnel
74A Lower Electrical Penetration Area	90 (1)	Redundant with upper electrical penetration area
60A Upper Electrical Tunnel	184, 185, 391, 418, 488, 490 (1)	Redundant with lower electrical tunnel
73A Upper Electrical Penetration Area	129, 226, 227 (1)	Redundant with lower electrical penetration area
73A Common Electrical Pen Area	226 (1)	Cables from lower tunnel rise and join cables from upper tunnel
8 Boric Acid Transfer Pumps	350 (2)	
8A RHR HXs	235 (2)	Protected from missile strike
9 Safety Injection Pumps #31, 32, 33	235 (2)	Protected from missile strike

*1 is control system
2 is primary cooling system
3 is secondary cooling system
4 is component cooling system

IPEC00036619

Table 14B-3
(Cont.)

Summary of Indian Point 3 Critical Area Modeling

Fire Protection Area Designation	Missile Code Designation (System No.)*	Remarks
10A Valve Corridor	236 (2)	Protected from missile strike
10 Diesel Generator #31 and FD Tank	448, 451 (1)	Redundant with other power supplies
101A D/G #32 and FD Tank	449, 452 (1)	Redundant with other power supplies
102A D/G #33 and FD Tank	450, 453 (1)	Redundant with other power supplies
11 Cable Spreading Room	429, 430, 432, 434, 435, 436, 439, 442, 443, 444, 445 (1)	Redundant with diesels – plant can isolate one diesel
11 Cable Spreading Room Common Area	529 (1)	All cables join before entering electrical tunnels
11 MG Sets #31, 32	431, 433 (1)	Redundant with other power supplies
11 Reactor Trip Breakers	440, 441 (1)	Redundant with control room'
12, 13 Battery Rooms	437, 438 (1)	Redundant with other power supplies
14 Switchgear Room, Bus 3A	420 (1)	Redundant with other buses
14 Switchgear Room, Buses 3A, 6A	421 (1)	Region overlaps the two buses
14 Switchgear Room, Bus 6A	422 (1)	Redundant with other buses
14 Switchgear Room, Buses 3A, 2A	423 (1)	Region overlaps the two buses
14 Switchgear Room, All buses	424 (1)	Region overlaps all buses
14 Switchgear Room, Buses 6A, 5A	425 (1)	Region overlaps the two buses
14 Switchgear Room, 2A	426 (1)	Redundant with other buses
14 Switchgear Room, Bus 2A, 5A	427 (1)	Region overlaps the two buses
14 Switchgear Room, Bus 5A	428 (1)	Redundant with other buses
15 Control Room	446 (1)	Safe shutdown capability from local control stations

*1 is control system
2 is primary cooling system
3 is secondary cooling system
4 is component cooling system

IPEC00036620

Table 14B-3
(Cont.)

Summary of Indian Point 3 Critical Area Modeling

Fire Protection Area Designation	Missile Code Designation (System No.)*	Remarks
17A Large PAB Area: Motor Control Centers Air Receivers N_2 Storage Component Cooling HXs	 314 (1) 357 (1) 359 (1) 346, 352 (4)	 Control valves fail in safe position. Redundant with air system. Each redundant
27A Large PAB Area: Top of Comp. Cooling HXs Comp. Cooling Surge Tanks Boric Acid Tanks	 479, 480 (4) 378, 478 (4) 386, 481 (2)	 Each Redundant Each Redundant Each Redundant
20A Pipe Chase	319 (2)	
23A Pipe Chase	338, 382 (2)	
25A Seal Water HX	373 (2)	
28A Valve Corridor	365, 366 (2)	
29A Volume Control Tank	363 (2)	
32A Non-regenerative HX	377 (2)	
23 Aux. Feedwater Pumps	100 (3)	Protected from missile strikes
52A Aux. Feedwater Piping Cables	101 (3)	Protected from missile strikes
57A Feedwater Stop & Check Valves	99, 102, 232, 233 (3)	All redundant
55A Service Water Pump Motor #33	497 (4)	Two of three pumps in either train required
55A Service Water Pimp Motor #34	498 (4)	Two of three pumps in either train required
55A Service Water Pump Motor #32	500 (4)	Two of three pumps in either train required
55A Service Water Pump Motor #31	501 (4)	Two of three pumps in either train required
55A Service Water Pump Motor #36	502 (4)	Two of three pumps in either train required
55A Service Water Pump Motor #36	504 (4)	Two of three pumps in either train required
55A Service Water Strainer #36	505 (4)	Two of three strainers in either train required

*1 is control system
2 is primary cooling system
3 is secondary cooling system
4 is component cooling system

IPEC00036621

Table 14B-3
(Cont.)

Summary of Indian Point 3 Critical Area Modeling

Fire Protection Area Designation	Missile Code Designation (System No.)*	Remarks
55A Service Water Strainer #35	506 (4)	Two of three strainers in either train required
55A Service Water Strainer #34	507 (4)	Two of three strainers in either train required
55A Service Water Strainer #33	508 (4)	Two of three strainer in either train required
55A Service Water Strainer #32	509 (4)	Two of three strainers in either train required
55A Service Water Strainer #31	510 (4)	Two of three strainers in either train required
55A Service Water Pipes	516, 517	All pipes passing under road and through concrete bunker
55A Service water Valve Pit	524, 525	Essential service water
55A Service water Valves Pit	526	Overlap of essential and non-essential service
55A Service Water Valve Pit Backup Service Water Pump Yard Backup Service Water Valve Pit Area	527 197 (4) 198 (4)	Non-essential service water Redundant with main service water Redundant with main service water
59A Piping Penetration Area	127, 130, 135 (2, 3, 4)	Region 127 is empty space – missile strikes do not cause safety compromise
105A Primary Water Storage Tank	156 (2)	Redundant with RWST
106A Refueling Water Storage Tank	471 (2)	
552, 553 Condensate Storage Tank Offsite Feeder	491 (2) 85 (1)	 Redundant with other power supplies
90A, 91A Spent Fuel Storage	176	Can be struck by HTMs only, but strikes cannot cause offsite release

*1 is control system
2 is primary cooling system
3 is secondary cooling system
4 is component cooling system

IPEC00036622

Table 14B-3
(Cont.)

Summary of Indian Point 3 Critical Area Modeling

Fire Protection Area Designation	Missile Code Designation (System No.)*	Remarks
Additional control cabling between control room and containment	200, 201, 202, 203, 204, 205 (1)	Redundant with additional control cabling between turbine building and containment
Additional control cabling between turbine building and containment	213, 216, 218, 220 (1)	Redundant with additional control cabling between control room and containment

*1 is control system
2 is primary cooling system
3 is secondary cooling system
4 is component cooling system

IPEC00036623

Table 14B-4. Results for 2000 Trials Each for Critical Discs, 90° Segments (24,000 Missile Traces per Disc, for a Grand Total of 432,000 Traces)

Disc Failure at Rated Speed

Disc Number	Number of Missile Penetrations of Safety Regions	Probability Per Year of Penetration	Number of Safety Compromises (No Credit for Backup SW)	Probability of Safety Compromise	Number of Safety Compromises (Credit Taken For Backup SW)	Probability of Safety Compromise
1	107	2.88E-9	0	0	0	0
4	46	4.74E-8	0	0	0	0
5	44	5.59E-8	0	0	0	0
8	47	2.64E-8	5	2.81E-9	5	2.81E-9
9	59	9.09E-8	4	6.16E-9	3	4.62E-9
10	50	1.39E-10	3	8.34E-12	2	5.56E-12
16	138	2.75E-10	13	2.59E-11	1	1.99E-12
17	137	2.93E-9	20	4.28E-10	1	2.14E-11
20	129	5.34E-9	22	9.11E-10	0	0
21	128	6.20E-10	10	4.84E-11	1	4.84E-12
22	109	2.34E-10	14	3.01E-11	2	4.30E-12
24	14	4.44E-10	0	0	0	0
25	112	1.61E-10	17	2.45E-11	2	2.88E-12
28	19	2.72E-8	1	1.43E-9	0	0
29	35	6.44E-8	2	3.68E-9	0	0
32	27	5.64E-8	1	2.09E-9	0	0
33	36	2.73E-8	0	0	0	0
36	10	6.72E-11	1	6.72E-12	1	6.72E-12
Total	1247	4.09E-7	113	1.70E-8	18	7.47E-9

Disc Failure at 132 Percent Overspeed

Disc Number	Number of Missile Penetrations of Safety Regions	Probability Per Year of Penetration	Number of Safety Compromises (No Credit for Backup SW)	Probability of Safety Compromise	Number of Safety Compromises (Credit Taken For Backup SW)	Probability of Safety Compromise
1	100	1.47E-10	0	0	0	0
4	39	5.50E-10	1	1.41E-11	1	1.41E-11
5	53	6.84E-12	1	1.29E-13	0	0
8	59	3.66E-10	0	0	0	0
9	41	8.08E-10	1	1.97E-11	1	1.97E-11
10	46	4.35E-12	4	3.78E-13	4	3.78E-13
16	128	6.12E-12	17	8.13E-13	2	9.56E-14
17	137	3.23E-11	16	3.78E-12	2	4.72E-13
20	156	4.87E-11	27	8.42E-12	2	6.24E-13
21	124	1.18E-11	22	2.09E-12	0	0
22	104	7.82E-12	16	1.20E-12	1	7.52E-14
24	9	1.27E-11	0	0	0	0
25	82	8.15E-12	13	1.29E-12	1	9.94E-14
28	23	4.14E-10	0	0	0	0
29	36	6.41E-10	1	1.78E-11	0	0
32	31	5.98E-10	0	0	0	0
33	20	2.16E-10	1	1.08E-11	1	1.08E-11
36	12	5.18E-9	2	8.64E-13	2	8.64E-13
Total	1200	3.88E-9	122	8.14E-11	17	4.73E-11

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Table 14B-5. Results for 2000 Trials Each for Critical Discs, 120° Segments (18,000 Missile Traces per Disc for a Grand Total 324,000 Traces.

Disc Failure at Rated Speed

Disc Number	Number of Missile Penetrations of Safety Regions	Probability Per Year of Penetration	Number of Safety Compromises (No Credit for Backup SW)	Probability of Safety Compromise	Number of Safety Compromises (Credit Taken For Backup SW)	Probability of Safety Compromise
1	77	2.07E-9	0	0	0	0
4	27	2.78E-8	0	0	0	0
5	22	2.79E-8	2	2.54E-9	1	1.27E-9
8	25	1.40E-8	0	0	0	0
9	38	5.85E-8	0	0	0	0
10	39	1.08E-10	2	5.56E-12	2	5.56E-12
16	77	1.53E-10	14	2.79E-11	0	0
17	102	2.18E-9	14	3.00E-10	2	4.28E-11
20	86	3.56E-9	13	5.38E-10	0	0
21	61	2.95E-10	3	1.45E-11	0	0
22	93	2.00E-10	9	1.94E-11	0	0
24	4	1.26E-10	0	0	0	0
25	54	7.78E-11	4	5.76E-12	0	0
28	11	1.57E-8	0	0	0	0
29	18	3.31E-8	0	0	0	0
32	27	5.64E-8	2	4.18E-9	0	0
33	16	1.21E-8	1	7.58E-10	0	0
36	12	8.06E-11	2	1.34E-11	2	1.34E-11
Total	789	2.54E-7	66	8.40E-9	7	1.33E-9

Disc Failure at 132 Percent Overspeed

Disc Number	Number of Missile Penetrations of Safety Regions	Probability Per Year of Penetration	Number of Safety Compromises (No Credit for Backup SW)	Probability of Safety Compromise	Number of Safety Compromises (Credit Taken For Backup SW)	Probability of Safety Compromise
1	65	9.56E-11	0	0	0	0
4	26	3.67E-10	0	0	0	0
5	40	5.16E-12	2	2.58E-13	2	2.58E-13
8	38	2.36E-10	3	1.86E-11	3	1.86E-11
9	24	4.73E-10	0	0	0	0
10	46	3.97E-12	2	1.89E-13	2	1.89E-13
16	67	3.20E-12	9	4.30E-13	0	0
17	100	2.36E-11	11	2.60E-12	1	2.36E-13
20	101	3.15E-11	10	3.12E-12	1	3.12E-13
21	66	6.26E-12	6	5.69E-13	0	0
22	80	6.07E-12	6	4.51E-13	0	0
24	9	1.27E-11	0	0	0	0
25	46	4.57E-12	8	4.95E-13	4	3.98E-13
28	18	3.24E-10	0	0	0	0
29	28	4.98E-10	1	1.78E-11	0	0
32	25	4.82E-10	0	0	0	0
33	15	1.62E-10	0	0	0	0
36	7	3.02E-12	2	8.64E-13	2	8.64E-13
Total	811	2.74E-9	60	4.57E-11	15	2.09E-11

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Table 14B-6. Safety Regions Struck by Turbine Missiles in Simulation Study.

Target	Description
64, 267	Lower electrical tunnel
85	Offsite power feeder
99, 102, 232, 233	Feedwater stop and check valves
127, 135	Piping penetration area
129, 226, 227	Upper electrical penetration area
156	Primary water storage tank
176	Spent fuel storage
184, 488	Upper electrical tunnel
197	Backup service water pump yard
198	Backup service water valve pit area
213, 216, 218, 200	Cabling (turbine bldg to containment)
420, 421, 422	Control building switchgear room
429, 430, 434, 435, 436, 442, 443, 444, 445, 529	Cable spreading room
446	Control room
497, 498, 500, 501, 502, 504	Service water pump motors
505, 506, 507, 508, 509, 510	Service water strainer pit
516, 517	Service water piping

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Table 14B-7. Assumed Valve Status Just After Turbine Missile Accident.

Valve	Status
Pressurizer Power-Operated Relief Valves	Closed (normally closed)
Oetdown Line Valves (Normal and Excess)	Open
Steam Generator Blowdown Valves	Closed (close automatically on loss of offsite power)
Steam Generator Sample Line Valves	Open
Power Operated Steam Relief Valves (atmospheric dump)	Closed (normally closed)
Turbine Bypass Valves	Closed (normally closed)
Turbine Stop and Control Valves	Closed (automatically close on overspeed and/or loss of condenser vacuum)
Reactor Isolation Valves	Open
Pressure Reducing Valve (in admission line to drive of turbine-driven auxiliary feedwater pump)	Open (opens automatically on loss of offsite power)
Auxiliary Feedwater Control Valves	Open (normally open)

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Table 14B-8. Turbine Missile Accident – Expected Case

Fuel Parameters

Reactor Power (102%)	3086 MWt
No. of Assemblies	193
Fuel Rods per Assembly	204
Normalized Power, 6 Highest Rated Discharged Assemblies	1.16
Normalized Power, Highest Rated Discharged Assembly	1.29
Axial Peak/Avg., Highest Rated Discharged Assembly	1.37

Activity Release Data

Isotope	Release Fraction	Bubble Decontamination Factor	Total Curie Release to Environment
I-131	0.0155	760	42
Xe-133	0.0127	1	48,580
Kr-85	0.227	1	9,520

IPEC00036628

Table 14B-9. Turbine Missile Accident – Design Case

Fuel Parameters

Reactor Power (102%)	3086 MWt
No. of Assemblies	193
Fuel Rods per Assembly	204
Normalized Power, 6 Highest Rated Discharged Assemblies	1.27
Normalized Power, Highest Rated Discharged Assembly	1.29
Axial Peak/Avg., Highest Rated Discharged Assembly	1.72

Activity Release Data

Isotope	Release Fraction	Bubble Decontamination Factor	Total Curie Release to Environment
I-131	0.0322	500	143
Xe-133	0.0259	1	108,470
Kr-85	0.302	1	13,870

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Table 14B-10. Monitor Tank Maximum Activities

Isotope	Activity (uc)
Cs-134	106
Cs-137	625
Mo-99	530
I-131	291
I-133	145
I-135	21

IPEC00036630

APPENDIX 14C

EVALUATION MODELS AND PARAMETERS FOR ANALYSIS OF RADIOLOGICAL CONSEQUENCES OF ACCIDENTS

This appendix contains the parameters and models that form the basis of the radiological consequences analyses for the various postulated accidents.

14C.1 Offsite Dose Calculation Models

Radiological consequences analyses are performed to determine the total effective dose equivalent (TEDE) doses associated with the postulated accidents. The determination of TEDE doses takes into account the committed effective dose equivalent (CEDE) dose resulting from the inhalation of airborne activity (i.e., the long-term dose accumulation in the various organs) as well as the effective dose equivalent (EDE) dose resulting from immersion in the cloud of activity.

14C.1.1 Immersion Dose (Effective Dose Equivalent)

Assuming a semi-infinite cloud, the immersion doses are calculated using the equation:

$$D_{im} = \sum_i DCF_i \sum_j R_{ij} (\chi/Q)_j$$

where:

where:	D_{im}	=	Immersion (EDE) dose (rem)
	DCF_i	=	EDE dose conversion factor for isotope i (rem-m^3/Ci-s)
	R_{ij}	=	Amount of isotope i released during time period j (Ci)
	$(\chi/Q)_j$	=	Atmospheric dispersion factor during time period j (s/m^3)

14C.1.2 Inhalation Dose (Committed Effective Dose Equivalent)

The CEDE doses are calculated using the equation:

$$D_{CEDE} = \sum_i DCF_i \sum_j R_{ij} (BR)_j (\chi/Q)_j$$

where:	D_{CEDE}	=	CEDE dose (rem)
	DCF_i	=	CEDE dose conversion factor for isotope I (rem/Ci)
	R_{ij}	=	Amount of isotope i released during time period j (Ci)
	$(BR)_j$	=	Breathing rate during time period j (m^3/s)
	$(\chi/Q)_j$	=	Atmospheric dispersion factor during time period j (s/m^3)

14C.1.3 Total Effective Dose Equivalent (TEDE)

The TEDE doses are the sum of the EDE and the CEDE doses.

14C.2 Control Room Dose Models

IPEC00036631

Radiological consequences analyses are performed to determine the TEDE doses associated with the postulated accident. The determination of TEDE doses takes into account the CEDE dose resulting from the inhalation of airborne activity (that is, the long-term dose accumulation in the various organs) as well as the EDE dose resulting from immersion in the cloud of activity.

14C.2.1 Control Room Model

The control room is modeled as a discrete volume. The filtered and unfiltered inflow to the control room and the recirculation cleanup flow are used to calculate the activity in the control room. The control room parameters modeled in the analyses are presented in Table 14C-1.

14C.2.2 Immersion Dose Model

Due to the finite volume of air contained in the control room, the immersion dose for an operator occupying the control room is substantially less than it is for the case in which a semi-infinite cloud is assumed. The finite cloud doses are calculated using the geometry correction factor from Murphy and Campe (Reference 1).

The equation is:

$$D_{im} = \frac{1}{GF} \sum_i DCF_i \sum_j (IAR)_{ij}\, O_j$$

where:

D_{im}	=	Immersion (EDE) dose (rem)
GF	=	Control room geometry factor
	=	$1173/V^{0.338}$
V	=	Volume of the control room (ft^3)
DCF_i	=	EDE dose conversion factor for isotope i (rem-m^3/Ci-s)
$(IAR)_{ij}$	=	Integrated activity for isotope i in the control room during time period j (Ci-s/m^3)
O_j	=	Fraction of time period j that the operator is assumed to be present

14C.2.3 Inhalation Dose Model

The CEDE doses are calculated using the equation:

$$D_{CEDE} = \sum_i DCF_i \sum_j (IAR)_{ij} (BR)_j\, O_j$$

where:

D_{CEDE}	=	CEDE dose (rem)
DCF_i	=	CEDE dose conversion factor (rem per curie inhaled) for isotope i
$(IAR)_{ij}$	=	Integrated activity for isotope i in the control room during time period j (Ci-s/m^3)
$(BR)_j$	=	Breathing rate during time period j (m^3/s)
O_j	=	Fraction of time period j that the operator is assumed to be present

IPEC00036632

14C.2.3 Total Effective Dose Equivalent (TEDE)

The TEDE doses are the sum of the EDE and the CEDE doses.

14C.3 General Analysis Parameters

14C.3.1 Source Terms

The sources of radioactivity for release are dependent on the specific accident. Activity may be released from the primary coolant, from the secondary coolant, and from the core if the accident involves fuel failures. The radiological consequences analyses use conservative design basis source terms.

14C.3.1.1 Primary Coolant Source Term

The design basis primary coolant source terms are listed in Table 9.2-5. These source terms are based on continuous plant operation with 1.0-percent fuel defects. The remaining assumptions used in determining the primary coolant source terms are listed in Table 9.2-4.

The accident dose analyses take into account a reduction in the primary coolant source terms for iodines below those listed in Table 9.2-5, consistent with the Tech Spec limit of 1.0 μCi/g dose equivalent I-131 (these iodine concentrations are provided in Table 14C-2).

The radiological consequences analyses for certain accidents also take into account the phenomenon of iodine spiking which causes the concentration of radioactive iodines in the primary coolant to increase significantly. The iodine spike may be a pre-existing spike or a spike that is initiated by the accident transient or associated reactor trip. The pre-existing spike is an iodine spike that occurs prior to the accident and for which the peak primary coolant activity is reached at the time the accident is assumed to occur. The pre-existing spike is assumed to be 60 μCi/g dose equivalent I-131 (Table 14C-2 lists the concentrations of iodine isotopes associated with a pre-existing iodine spike). The probability of this adverse timing of the iodine spike and accident is small.

Although it is unlikely for an accident to occur at the same time that an iodine spike is at its maximum reactor coolant concentration, for many accidents it is expected that an iodine spike would be initiated by the accident or by the reactor trip associated with the accident. Table 14C-3 lists the iodine appearance rates (rates at which the various iodine isotopes are released from the core to the primary coolant by way of the assumed cladding defects) for normal operation. The appearance rates during an iodine spike are assumed to be as much as 500 times the normal appearance rates.

14C.3.1.2 Secondary Coolant Source Term

The secondary coolant source term used in the radiological consequences analyses is conservatively assumed to be 10 percent of the primary coolant equilibrium source term. This is consistent with the Tech Spec limit on iodine in the secondary coolant.

Because the iodine spiking phenomenon is short-lived and there is a high level of conservatism for the assumed secondary coolant iodine concentrations, the effect of iodine spiking on the secondary coolant iodine source terms is not modeled.

IPEC00036633

There is assumed to be no secondary coolant noble gas source term because the noble gases entering the secondary side due to primary-to-secondary leakage enter the steam phase and are discharged via the condenser air removal system.

14C.3.1.3 Core Source Term

Table 14C-4 lists the core source terms at shutdown for an assumed three-region equilibrium cycle at end of life after continuous operation at 2 percent above full core thermal power. In addition to iodines and noble gases, the source terms listed include nuclides that are identified as potentially significant dose contributors in the event of a degraded core accident. The design basis loss-of-coolant accident analysis is not expected to result in significant core damage, but the radiological consequences analysis assumes severe core degradation.

14C.3.2 Nuclide Parameters

The radiological consequence analyses consider radioactive decay of the subject nuclides prior to their release, but no additional decay is assumed after the activity is released to the environment. Table 14C-5 lists the decay constants for the nuclides of concern.

Table 14C-5 also lists the dose conversion factors for calculation of the CEDE doses due to inhalation of iodines and other nuclides and EDE dose conversion factors for calculation of the dose due to immersion in a cloud of activity. The CEDE dose conversion factors are from EPA Federal Guidance Report No. 11 (Reference 2) and the EDE dose conversion factors are from EPA Federal Guidance Report No. 12 (Reference 3).

14C.3.3 Atmospheric Dispersion Factors

Section 14.3.5 lists the off-site short-term atmospheric dispersion factors (χ/Q). Table 14C-6 (Sheet 1 of 2) reiterates these χ/Q values.

The ARCON96 computer code (Reference 4) was utilized to determine the X/Q values at the control room intake. The ARCON96 analysis for Indian Point 3 considered five release locations: a containment surface leak, the side of the Auxiliary Boiler Feed Building, the safety valve discharge (also identified as "organ pipes") located on the Auxiliary Boiler Feed Building, the atmospheric dump valves discharge (also identified as the "silencers") located on the Auxiliary Boiler Feed Building, and the containment vent. These correspond to potential release points for various accident scenarios. Additional conservatisms were added to the calculations:

1. The initial plume standard deviations used were equal to one-sixth of the width and available height of the containment.

2. The initial horizontal plume dimension for vent releases is the equivalent vent diameter divided by six.

3. All vertical velocities were set to zero.

The atmospheric dispersion factors (χ/Q) to be applied to air entering the control room following a design basis accident are specified for each potential activity release location that has been identified. These χ/Q values are listed in Table 14C-7.

The control room χ/Q values do not incorporate occupancy factors.

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References

1. Murphy, K. G., Campe, K. M., "Nuclear Power Plant Control Room Ventilation System Design for Meeting General Criterion 19," paper presented at the 13th AEC Air Cleaning Conference.

2. EPA Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," EPA-520/1-88-020, September 1988.

3. EPA Federal Guidance Report No. 12, "External Exposure to Radionuclides in Air, Water, and Soil," EPA 402-R-93-081, September 1993.

4. NUREG/CR-6331, Ramsdell, J. V. and Simonen, C. A., "Atmospheric Relative Concentrations in Building Wakes," Revision 1, May 1997.

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TABLE 14C-1

ASSUMPTIONS USED FOR ANALYSIS OF CONTROL ROOM DOSES	
Control Room Volume	47,200 ft^3
Unfiltered Inleakage	700 cfm*
HVAC Normal Operating Mode Inflow (unfiltered)	1500 cfm
Time to Switch HVAC from Normal to Emergency Mode After Receipt of Actuation Signal	60 sec
HVAC Emergency Mode Filtered Inflow	1500 cfm
HVAC Emergency Mode Filtered Recirculation Flow	0 cfm
Filter Efficiency	
Elemental	90%
Organic	90%
Particulate	99%
Breathing Rate	3.5E-4 m^3/sec
Atmospheric Dispersion Factors	See Table 14C-7
Occupancy Factors	
0-1 day	1.0
1-4 days	0.6
4-30 days	0.4

* The 700 cfm unfiltered inleakage assumption is applied to all radiological consequences analyses for design basis accidents except for the large-break LOCA (Section 14.3.5.1) which uses a reduced value of 400 cfm.

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Table 14C-2

REACTOR COOLANT IODINE CONCENTRATIONS		
Nuclide	Tech Spec 1.0 µCi/g DE I-131 Equilibrium Operation Limit (µCi/g)	Tech Spec 60 µCi/g DE I-131 48-hour Iodine Spike Limit (µCi/g)
I-130	0.0161*	0.97
I-131	0.7849	47.09
I-132	0.5345	32.07
I-133	1.0555	63.33
I-134	0.1146	6.88
I-135	0.5126	30.76

* While I-130 is included in the dose analyses, it is not included in the definition of Dose-Equivalent I-131 contained in the Technical Specifications.

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Table 14C-3

IODINE APPEARANCE RATES IN THE REACTOR COOLANT TO MAINTAIN A CONCENTRATION OF 1.0 µCI/GRAM DOSE-EQUIVALENT I-131	
Nuclide	**Equilibrium Appearance Rate (Ci/min)**
I-130*	0.0124
I-131	0.4360
I-132	0.9391
I-133	0.7134
I-134	0.4286
I-135	0.4949

* While I-130 is included in the dose analyses, it is not included in the definition of Dose-Equivalent I-131 contained in the Technical Specifications.

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Table 14C-4

REACTOR CORE SOURCE TERM(1)					
	Nuclide	**Inventory (Ci)**		**Nuclide**	**Inventory (Ci)**
Iodines	I-130	3.78E+06	**Sr & Ba**	Sr-89	8.84E+07
	I-131	9.10E+07		Sr-90	8.79E+06
	I-132	1.33E+08		Sr-91	1.11E+08
	I-133	1.88E+08		Sr-92	1.20E+08
	I-134	2.06E+08		Ba-139	1.68E+08
	I-135	1.76E+08		Ba-140	1.60E+08
Noble Gases	Kr-85m	2.44E+07	**Noble Metals**	Mo-99	1.75E+08
	Kr-85	1.11E+06		Tc-99m	1.53E+08
	Kr-87	4.69E+07		Ru-103	1.39E+08
	Kr-88	6.60E+07		Ru-105	9.58E+07
	Xe-131m	9.92E+05		Ru-106	4.84E+07
	Xe-133m	5.45E+06		Rh-105	8.83E+07
	Xe-133	1.79E+08	**Cerium Group**	Ce-141	1.52E+08
	Xe-135m	3.68E+07		Ce-143	1.43E+08
	Xe-135	4.77E+07		Ce-144	1.20E+08
	Xe-138	1.55E+08		Pu-238	4.11E+05
Alkali Metals	Rb-86	2.36E+05		Pu-239	3.50E+04
	Cs-134	2.05E+07		Pu-240	5.21E+04
	Cs-136	5.96E+06		Pu-241	1.17E+07
	Cs-137	1.19E+07		Pu-241	1.17E+07
	Cs-138	1.72E+08		Np-239	1.87E+09
Te Group	Sb-127	9.89E+06		Pu-241	1.17E+07
	Sb-129	2.97E+07		Np-239	1.87E+09
	Te-127m	1.28E+06			
	Te-127	9.83E+06			
	Te-129m	4.28E+06			
	Te-129	2.92E+07			
	Te-131m	1.33E+07			
	Te-132	1.30E+08			

Note (1): The following assumptions apply:

- Core thermal power of 3280.3 MWt (2 percent above the design core power of 3216 MWt)
- Three-region equilibrium cycle core at end of life

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Table 14C-4
(Cont.)
REACTOR CORE SOURCE TERM[1]

	Nuclide	Inventory (Ci)
Lanthanides	Y-90	9.16E+06
	Y-91	1.14E+08
	Y-92	1.21E+08
	Y-93	1.39E+08
	Nb-95	1.56E+08
	Zr-95	1.54E+08
	Zr-97	1.55E+08
	La-140	1.65E+08
	La-141	1.53E+08
	La-142	1.48E+08
	Nd-147	6.07E+07
	Pr-143	1.37E+08
	Am-241	1.44E+04
	Cm-242	3.47E+06
	Cm-244	3.70E+05

Note (1): The following assumptions apply:

- Core thermal power of 3280.3 MWt (2 percent above the design core power of 3216 MWt)
- Three-region equilibrium cycle core at end of life

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TABLE 14C-5 NUCLIDE PARAMETERS			
A. HALOGENS			
Isotope	**Decay Constant (hr^{-1})**	**EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)**	**CEDE Dose Conversion Factor (Sv/Bq)**
I-130	5.61E-02	1.04E-13	7.14E-10
I-131	3.59E-03	1.82E-14	8.89E-09
I-132	3.01E-01	1.12E-13	1.03E-10
I-133	3.33E-02	2.94E-14	1.58E-09
I-134	7.91E-01	1.30E-13	3.55E-11
I-135	1.05E-01	7.98E-14	3.32E-10
B. NOBLE GASES			
Isotope	**Decay Constant (hr^{-1})**	**EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)**	
Kr-85m	1.55E-01	7.48E-15	
Kr-85	7.38E-06	1.19E-16	
Kr-87	5.45E-01	4.12E-14	
Kr-88	2.44E-01	1.02E-13	
Xe-131m	2.43E-03	3.89E-16	
Xe-133m	1.32E-02	1.37E-15	
Xe-133	5.51E-03	1.56E-15	
Xe-135m	2.72E+00	2.04E-14	
Xe-135	7.63E-02	1.19E-14	
Xe-138	2.93E+00	5.77E-14	
C. ALKALI METALS			
Nuclide	**Decay Constant (hr^{-1})**	**EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)**	**CEDE Dose Conversion Factor (Sv/Bq)**
Rb-86	1.55E-03	4.81E-15	1.79E-09
Cs-134	3.84E-05	7.57E-14	1.25E-08
Cs-136	2.20E-03	1.06E-13	1.98E-09
[1]Cs-137	2.64E-06	2.88E-14	8.63E-09
Cs-138	1.29E+00	1.21E-13	2.74E-11

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TABLE 14C-5
(Cont.)
NUCLIDE PARAMETERS

D. TELLURIUM GROUP

Nuclide	Decay Constant (hr^{-1})	EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)	CEDE Dose Conversion Factor (Sv/Bq)
Sb-127	7.50E-03	3.33E-14	1.63E-09
Sb-129	1.60E-01	7.14E-14	1.74E-10
Te-127m	2.65E-04	1.47E-16	5.81E-09
Te-127	7.41E-02	2.42E-16	8.60E-11
Te-129m	8.60E-04	1.55E-15	6.47E-09
Te-129	5.98E-01	2.75E-15	2.42E-11
Te-131m	2.31E-02	7.01E-14	1.73E-09
Te-132	8.86E-03	1.03E-14	2.55E-09

E. STRONTIUM AND BARIUM

Nuclide	Decay Constant (hr^{-1})	EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)	CEDE Dose Conversion Factor (Sv/Bq)
Sr-89	5.72E-04	7.73E-17	2.860E-04
Sr-90	2.72E-06	7.53E-18	2.786E-05
Sr-91	7.30E-02	3.45E-14	1.277E-01
Sr-92	2.56E-01	6.79E-14	2.512E-01
Ba-139	5.03E-01	2.17E-15	8.029E-03
Ba-140	2.27E-03	8.58E-15	3.175E-02

F. NOBLE METALS

Nuclide	Decay Constant (hr^{-1})	EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)	CEDE Dose Conversion Factor (Sv/Bq)
Mo-99	1.05E-02	7.28E-15	1.07E-09
Tc-99m	1.15E-01	5.89E-15	8.80E-12
Ru-103	7.35E-04	2.25E-14	2.42E-09
Ru-105	1.56E-01	3.81E-14	1.23E-10
Ru-106	7.84E-05	0	1.29E-07
Rh-105	1.96E-02	3.72E-15	2.58E-10

Note:
1. The listed average gamma disintegration energy for Cs-137 is due to the production and decay of Ba-137m.

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TABLE 14C-5
(Cont.)
NUCLIDE PARAMETERS

G. CERIUM GROUP

Nuclide	Decay Constant (hr^{-1})	EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)	CEDE Dose Conversion Factor (Sv/Bq)
Ce-141	8.89E-04	3.43E-15	2.42E-09
Ce-143	2.10E-02	1.29E-14	9.16E-10
Ce-144	1.02E-04	8.53E-16	1.01E-07
Pu-238	9.02E-07	4.88E-18	1.06E-04
Pu-239	3.29E-09	4.24E-18	1.16E-04
Pu-240	1.21E-08	4.75E-18	1.16E-04
Pu-241	5.50E-06	7.25E-20	2.23E-06
Np-239	1.23E-02	7.69E-15	6.78E-10

H. LANTHANIDE GROUP

Nuclide	Decay Constant (hr^{-1})	EDE Dose Conversion Factor ($Sv\text{-}m^3/Bq\text{-}s$)	CEDE Dose Conversion Factor (Sv/Bq)
Y-90	1.08E-02	1.90E-16	2.28E-09
Y-91	4.94E-04	2.60E-16	1.32E-08
Y-92	1.96E-01	1.30E-14	2.11E-10
Y-93	6.86E-02	4.80E-15	5.82E-10
Nb-95	8.22E-04	3.74E-14	1.57E-09
Zr-95	4.51E-04	3.60E-14	6.39E-09
Zr-97	4.10E-02	9.02E-15	1.17E-09
La-140	1.72E-02	1.17E-13	1.31E-09
La-141	1.76E-01	2.39E-15	1.57E-10
La-142	4.50E-01	1.44E-13	6.84E-11
Nd-147	2.63E-03	6.19E-15	1.85E-09
Pr-143	2.13E-03	2.10E-17	2.19E-09
Am-241	1.83E-07	8.18E-16	1.20E-04
Cm-242	1.77E-04	5.69E-18	4.67E-06
Cm-244	4.37E-06	4.91E-18	6.70E-05

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Table 14C-6

OFFSITE ATMOSPHERIC DISPERSION FACTORS (χ/Q) FOR ACCIDENT DOSE ANALYSIS

Site boundary χ/Q (sec/m^3)	
0 – 2 hours[(1)]	1.03E-3
Low population zone χ/Q (sec/m^3)	
0 – 2 hours	3.8E-4
2 – 24 hours	1.9E-4
24 – 720 hours	1.7E-5

Note:
1. Nominally defined as the 0 to 2 hour interval but is applied to the 2-hour interval having the highest activity releases in order to address 10 CFR Part 50.67 requirements.

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Table 14C-7

CONTROL ROOM ATMOSPHERIC DISPERSION FACTORS (χ/Q) FOR ACCIDENT DOSE ANALYSIS

χ/Q (s/m^3) at HVAC Intake for the Identified Release Points				
	Plant Vent (1)	**Ground Level Containment Release (2)**	**Atmospheric Dump Valve and Safety Valve Releases(3)**	**Steam Line Break Releases(4)**
0 - 2 hours	6.00E-4	3.57E-4	1.14E-3	9.86E-4
2 - 8 hours	5.20E-4	3.12E-4	1.04E-3	8.74E-4
8 - 24 hours	2.12E-4	1.24E-4	5.05E-4	4.50E-4
1 - 4 days	1.76E-4	1.06E-4	4.01E-4	3.50E-4
4 - 30 days	1.30E-4	7.99E-5	3.21E-4	2.80E-4

Notes:

1. These dispersion factors are used for analysis of the doses for the fuel handling accident, for the volume control tank rupture, for the gas decay tank rupture, for the holdup tank rupture, and for the large-break LOCA (the sump solution leakage to the Plant Auxiliary Building).

2. The listed values represent modeling the containment shell as a diffuse area source and are used for evaluating the doses in the control room resulting from containment leakage of activity for the loss-of-coolant accidents and for the rod ejection accident.

3. The listed values are used for evaluating the doses in the control room for the steam generator tube rupture, for the main steam line break (the intact steam generator steaming releases), for the locked reactor coolant pump rotor, and for the rod ejection accident secondary side activity release pathway. The listed χ/Q values bound both the release point for the atmospheric dump valves and the release point for the safety valves.

4. The listed values are used for evaluating the doses in the control room for the main steam line break (faulted steam generator release path).

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CHAPTER 15

TECHNICAL SPECIFICATIONS AND BASES

The Technical Specifications and Bases were approved by the NRC on 12/12/75 with the issuance of Facility Operating License No. DPR-64. These Technical Specifications are frequently amended. For the latest version of the Technical Specifications for Indian Point 3, refer to Appendix A (Technical) and Appendix B (Environmental) to Facility Operating License No. DPR-64.

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Chapter 16

DESIGN CRITERIA FOR STRUCTURES AND EQUIPEMENT

16.1 SEISMIC DESIGN CRITERIA FOR STRUCTURES AND EQUIPEMENT

16.1.1 Definition of Seismic Design Classifications

All equipment and structures were classified as seismic Class I, Class II, or Class III as recommended in:

a) TID-7024, "Nuclear Reactors and Earthquakes," August 1963,
b) G.W. Housner, "Design of Nuclear Power Reactors Against Earthquakes," Proceedings of the Second World Conference on Earthquake Engineering, Vol. I, Japan 1960, Pg. 133, 134 and 137,
c) 10 CFR 100 Appendix A, "Seismic and Geological Siting Criteria for Nuclear Power plants,"
d) USNRC Reg. Guide 1.29, Rev 3, Sept. 1978, "Seismic Design Classification" and
e) ANSI/ANS-58.14-1993, "Safety and Pressure Integrity Classification Criteria for Light Water Reactors."

Class I

Those structures, systems and components, which must remain functional during or following a Safe Shutdown Earthquake (SSE)* to ensure: (i) The integrity of the reactor coolant pressure boundary, (ii) the capability to shutdown the reactor and maintain it in a safe shutdown condition, and (iii) the capability to prevent or mitigate the consequences of accidents that could result in potential off-site exposures comparable to the guidelines of 10 CFR 100. Also included in this definition are those structures, systems, and components that do NOT perform a safety-related function, but which must maintain structural integrity during or following a SSE to mitigate deleterious effects of system seismic interaction.

*NOTE: The Safe Shutdown Earthquake defines that earthquake which has commonly been referred to as the Design Basis Earthquake (DBE).

Class II

Those structures, systems and components which are necessary for continued operation without undue risk to the health and safety of the public when subjected to the effects of an Operating Basis Earthquake (OBE) in combination with normal operating loads.

Class III

Those structures, systems and components which are not directly related to reactor operation and containment, and which do not have to maintain structural integrity during or following a SSE.

16.1.2 Classification of Particular Structures and Equipment

The following are examples of particular structural and equipment classifications. These classifications are not intended to be all inclusive.

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Item	Class
Buildings and Structures	
Containment (including all penetrations and air locks, the concrete shield, the liner and the interior structures)	I
Fan House	I
Spent Fuel Pit	I
Electrical Tunnels	I
Fuel Storage Building	III
Control Room	I
Waste Holdup Tank	I
Liquid Waste Storage Building	III
Diesel Generator Building	I
Primary Auxiliary Building	I
Containment Access Facility (Structural steel and PAB interfaces only; balance of structure is Class III)	I
Control Building	I
Auxiliary Feedwater System Enclosure (lower portion of Shield Wall area)	I
Intake Structure (The 124 by 58 feet reinforced concrete structure that house the circulating water and service water pumps and their associate equipment)	I
Service Water Pipe Chase and Adjacent Connecting Discharge Canal Wall	I
Turbine Structure	III
Replaced Steam Generator Storage Facility III	
Buildings Containing Conventional Facilities	III
Equipment Piping and Supports*	
Reactor Control and Protection System	I
Radiation Monitoring System	I
Nuclear Process Instrumentation and Controls	I

*NOTE: Class I components (equipment, piping, instrumentation, etc.) located in or supported on a Class II structure are protected from earthquake damage or

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are backed by other Class I components located in or supported by a Class I structure.

Item	Class
Reactor	I
• Vessel and its supports	
• Vessel internals	
• Fuel Assemblies	
• RCC Assemblies and Drive Mechanisms	
• Supporting and positioning members	
• Incore Instrumentation Structure	
Reactor Coolant System	I
• Piping and Valves (including safety & relief valves) Steam Generators	
• Pressurizer	
• Reactor Coolant pumps	
• Supporting and positioning members	
Engineered Safety Features	I
• Safety Injection System (including safety injection and residual heat removal pumps, refueling water storage tank, accumulator tanks, boron injection tank, residual heat exchangers and connecting piping and valving)	
• Containment Spray System (including spray pumps, spray headers, spray additive tank and connecting piping and valving)	
• Containment Air Recirculation Cooling and Filtration System (including fans, coolers, ducts, valves, absolute filters and demisters)	
Primary Auxiliary Building Ventilation System	I
Condensate Storage Tanks	I
Pressurizer Relief Tank (including discharge piping downstream of pressurizer safety relief valves)	II
Residual Heat Removal Loop	I
Containment Penetration and Weld Channell Pressurization System	
Component Cooling Loop	I
Isolation Valve Seal Water System	I

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Item	Class
Sampling System	II
Spent Fuel Pit Cooling Loop	II & III
Fuel Transfer Tube	I
Emergency Power Supply System • Diesel generators and fuel oil storage tank • DC power supply steam system • Vital AC power supply system (instrument bus inverters) • Power distribution lines to equipment required for transformers and switchgear supplying the engineered safety features • Control panel boards • Motor control centers	I
Battery Chargers 31,32, 33 and 35	I
Battery Charger 34	III
Station Service Transformer	III
Control Equipment, facilities and lines necessary for the above seismic Class I items	I
Control Room Air Conditioning System	I
Hot Penetration Cooling System	I, III
Steam Generator Blowdown System	I, II & III
Waste Disposal System • Chemical drain tank • Waste Holdup Tanks • Sump Tank • Gas Decay Tanks • Spent Resin Storage Tank • Reactor Coolant Drain Tank • Compressors • Waste Holdup Tank Pumps • Sump Tank Pumps • Interconnecting Waste Gas Piping	I
All elements not listed as seismic Class I	III
Containment Crane	I
Manipulator, Fuel Storage Building Crane and other cranes	III
Conventional equipment, tanks and piping, other than seismic Class I and Class II	III

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Item	Class
Emergency Feed, and Service Water Pumps and Piping	I
Backup Service Water Pumps and piping	III
Fire Protection System (Piping Supports)* • Diesel Generator Building • Control Building • Primary Auxiliary Building • Fan House • Auxiliary Feed Pump Room • Electrical Tunnel • Containment Building • Fuel Handling Building	I

*Seismically analyzed to Class I criteria in accordance with Section 9.6.2

Primary Makeup Water Storage Tank	I
De-icing Pit and Pumps	III
Plant Vent	I

The Chemical and Volume Control System
is considered seismic Class I except for the items listed below,

Batch Tank	II
Monitor Tanks	II
Monitor Tank Pumps	II
Chemical Mixing Tank	II
Resin Fill Tank	III

The only portions of the plant which might carry substantial radioactivity, and which are seismic Class I, but which are not required because of safeguards operation or the safe shutdown and isolation of the reactor, are portions of the Chemical and Volume Control System and Waste Disposal System.

The specific components in the Chemical and Volume Control System are the volume control tank and holdup tank with associated piping, valves and supports. These components are all seismic Class I. In addition, the design of the system tanks and their location was based upon the commitment that a vessel rupture would not cause doses in excess of 10 CFR 20 limits at the exclusion radius

The specific components in the Waste Disposal System are the gas decay tanks with the associated piping, valves and supports. These components are all seismic Class I. In addition, the gas decay tanks of the Waste Disposal System have been designed such that the failure of any tank will not exceed 10 CFR 20 doses at the exclusion radius.

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The analysis showing that the rupture of the volume control tank or a gas decay tank does not exceed the special dose limits selected for Indian Point Unit 3 is found in Chapter 14.

Those components of the Chemical and Volume Control System that are not seismic Class I are listed above. Those components of the Waste Disposal System which are not seismic Class I are as follows: liquid waste holdup tanks in Liquid Waste Storage Building, regenerant tank, baler, and reactor coolant drain tank. Failure of these components will not result in offsite doses in excess of 10 CFR 20 limits at the site exclusion radius.

16.1.3 General Seismic Design Criteria and Damping Values

The general seismic criteria and methods of analysis described in this section are those which were utilized during the design phase of Indian Point 3. Details of the seismic piping reanalysis for safety related systems conducted by the Authority during 1979 and 1980 are presented in Section 16.3.5.

Class I

All components, systems and structures classified as seismic Class I were designed in accordance with the following criteria:

1. Primary steady state stresses, when combined with the seismic stress resulting from the application of seismic motion with a maximum ground acceleration of 0.05g acting in the vertical and 0.1g acting in the horizontal planes simultaneously, are maintained within the allowable stress limits accepted as good practice and, where applicable, set forth in the appropriate design standards, e.g., ASME Boiler and Pressure Vessel Code, USAS B31.1 Code for Pressure Piping, ACT 318 Building Code Requirements for Reinforced Concrete, and AISC Specifications for the Design and Erection of Structural Steel for Buildings.

2. Primary steady state stresses when combined with the seismic stress resulting from the application of seismic motion with a maximum ground acceleration of 0.10g acting in the vertical and 0.15g acting in the horizontal planes, simultaneously, are limited so that the function of the component, system or structure shall not be impaired as to prevent a safe and orderly shutdown of the plant.

   No loss of function implies that rotating equipment will not freeze, pressure vessels will not rupture, supports will not collapse under the load, systems required to be leak tight remain leak tight and components required to respond actively (such as valves and relays) will respond actively. The criteria for functional adequacy of the structures state that stresses do not exceed yield when subjected to seismic motion with a 0.15g maximum ground acceleration. Seismic Class I equipment associated with the primary reactor coolant loop was designed in accordance with the requirements of Section III of the ASME Boiler and Pressure Vessel Code for Nuclear Vessels for response to a 0.15g maximum ground acceleration earthquake. All seismic Class I piping was designed in accordance with the USAS Code for Pressure Piping B31.1.0 for response to 0.15g maximum ground acceleration earthquake.

For the Containment design, refer to the Containment Design Report (Appendix 5A).

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3. The seismic design criteria and qualification testing employed to assure the adequacy of seismic Class I electrical equipment are discussed in Section 7.2. The control board is not considered protection equipment. Typical switches and indicators for safeguards components were tested to determine their ability to withstand seismic forces without malfunction which would defeat automatic operation of the required component. The control boards are stiff, and past experience indicates that amplification of the board structure and accelerations seen by the devices mounted therein is considerably less than the subsequent acceleration which was shown the device could withstand in testing. Some components, for instance most pumps, required no additional restraints in order to meet the seismic criteria. Tanks generally required thicker walls and/or wall stiffeners and heavier support members and anchor bolts. Battery racks and instrument racks generally required heavier supports, cross bracing and heavier anchor bolts. The protection system equipment racks are bolted to the floor; no other seismic restraints were employed or deemed necessary to meet the seismic criteria. The type testing described in Section 7.2 used the same bolting arrangement as employed in the plant installation.

Seismic analysis of selected seismic Class I components including heat exchangers, pumps, tanks and valves, as well as seismic Class I structures, was performed using one of three methods depending on the relative rigidity of the equipment being analyzed:

(1) Equipment which is rigid and rigidly attached to the supporting structure was analyzed for a g-loading equal to the acceleration of the supporting structure at the appropriate elevation,

(2) Equipment which is not rigid, and therefore a potential for response to the support motion exists, was analyzed for the peak of the floor response curve with appropriate damping values;

(3) In some instances, non-rigid equipment was analyzed using a multi-degree-of-freedom modal analysis including the effect of modal participation factors and mode shapes together with the spectral motions of the floor response spectrum defined at the support of the equipment. The inertial forces, moments, and stresses were determined in each mode. They were then summed using the square-root-sum-of-the-squares method. Where structures were too complex to analyze, testing was performed.

The reactor coolant loop piping and main steam and main feedwater piping inside containment were seismically analyzed by Westinghouse using the computer code WECAN for the Reactor Coolant Loop and the computer code WESTDYN for the main steam and main feedwater lines. Verification of the computer codes for both static, linear and non linear elastic dynamic analysis capability has been performed. Reduced modal analysis method and modal superposition method and modal superposition method are used in the time history seismic analyses.

The reduced modal analysis is used to determine the natural frequencies and mode shapes for a linear, undamped structure. This analysis requires the specification of dynamic or active degrees of freedom (DOF) for the model, which are a subset of the total number of DOF. The selection of dynamic DOF must be such that the low frequency spectrum can accurately be presented while a reduced eigenvalue problem is solved. In other words, the selected or dynamic (or active) DOF should be able to describe the frequency modes at interest.

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The modal superposition method gives a time history solution for the response of an arbitrary structure subjected to known modal forces or ground acceleration time histories. The structure may include linear or non-linear elements. The uncoupled modal equations are integrated analytically.

The input to the time history DBE seismic analysis is in the form of time history seismic motions applied individually at the containment base mat in the north-south, east-west and vertical direction. These time histories seismic motions are based on those used in developing response spectra. The total response is obtained by determining the maximum response from combining absolutely one of the horizontal responses with the vertical seismic response.

Seismic Class I piping having a diameter 6" or larger plus the high head safety injection piping were initially designed statically using spacing tables which reflected the simultaneous application of horizontal and vertical spectral accelerations corresponding to 0.67 and 0.5 times the peak of the amplified floor response spectrum, respectively, developed at the support elevation of the piping system. A multi-degree-of-freedom dynamic analysis using the computer code ADLPIPE employing a dynamic model of the system and the applicable floor response spectrum as input motion was then performed to confirm the static design and analysis. The dynamic analysis successfully confirmed the conservatism of the static design.

Seismic Class I piping less than six inches in diameter was statically analyzed using spacing tables for simultaneously applied horizontal and vertical spectral accelerations corresponding to 2.0 and 1.33 times the peak of the amplified floor response spectrum, respectively, developed at the support elevation of the piping system. The coefficient of two times the peak of the amplified floor response spectrum was selected to account conservatively for modal participation factor effects in each mode and the contribution of higher modes. The design conservatism inherent in such a procedure has been verified by earlier comparative studies (Ginna, H.B. Robinson, and IP-2 Plants) relating seismic design stresses determined by coefficients from the peak of applicable floor or ground response spectrum to those determined by multi-degree-of-freedom detailed modal dynamic analysis.

The six inch diameter was selected as the dividing point because the reduction in pipe support hardware made possible by the more rigorous multi-degree-of-freedom detailed modal dynamic analysis below the six inch size (as opposed to the simplified double-the-peak response) did not warrant its use.

Non-rigid components and equipment components and equipment were only analyzed for an equivalent static load for vertical and horizontal seismic inputs if a dynamic analysis of a multi-degree-of-freedom model of similar component or piece of equipment has shown that the equivalent static load used gives conservative results. It is noted that, as described above, for piping having a diameter less than six inches, twice the peak of the floor response spectrum was used to determine the equivalent static loading. Analytical methods employed in the design of other seismic Class I structures, systems, and components are:

ITEM	METHOD
1. Reactor coolant loop piping and main stream and main feedwater piping inside containment	Multi-degree-of-freedom modal analysis response spectra
2. All other Class 1 Piping	

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	≥ 6" Dia. (including two inch high head safety injection lines)	Equivalent static analysis and confirmatory multi-degree-of-freedom modal analysis response spectra.
	< 6" Dia	Equivalent static analysis
3.	Refueling Water Storage Tank	Multi-degree-of-freedom modal analysis response spectra
4.	Primary Auxiliary Building ventilation system	Equivalent static load
5.	Condensate Storage Tank	Multi-degree-of-freedom modal analysis response spectra
6.	Containment Penetration and Weld Channel Pressurization System	Equivalent static load
7.	Diesel Generators	*See NOTE
8.	Fuel Oil Storage Tanks	No specific seismic design (UL approved, buried, atmospheric design pressure)
9.	DC Power Supply System	Equivalent static load
10.	Power Distribution lines to equipment required for transformers and switch-gear supplying the engi-neered safety features	Equivalent static load analysis on cable tray supports
11.	Control equipment, facili-ties and lines necessary for Items 6 through 9	Equivalent static load
12.	Auxiliary Feedwater System and Building	As outlined in the Authority's response to NRC Generic letter No. 81-14 (IPN-81-66, 8/28/81)
13.	Containment crane	Equivalent static load
14.	Emergency Boiler Feed Pumps and Service Water Pumps	Equivalent static load

*NOTE: No seismic design analysis was provided by the manufacturer of the Emergency Diesel Generator. However, the manufacturer of the diesel engine stated the following: "The diesel engine provided (for IP-2 and IP-3) was originally designed as a prime motive power unit for locomotive service. To meet these

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requirements, all component parts of the engine were designed to withstand minimum shock loads of 2.5g in any direction. This engine when modified for other uses retain this design criteria, as well as all allied equipment required. The engine foundation and sub-base are included."

In addition, the manufacturer of the generator portion of the units stated: "Machines of this type have been transported via rail shipment all over the United States without experiencing difficulty. Rail shipment experience indicate that shock loads of a magnitude of 2G's are common."

The methods utilized to determine the seismic input to these components are stated in the seismic design criteria.

The following criteria and procedures were used in formulating the mathematical model for the reactor coolant loop. Each portion of the piping system (straight runs and elbows) was subdivided into discrete elements. The mass of each of these elements was concentrated at the center of gravity of the elements. The major components were subdivided into discrete elements and the masses were located so as to (1) maintain the proper total mass of the component (2) maintain its moment of inertia about the center of gravity of the component (3) to maintain the position of the center of gravity of the component.

Structures having significant eccentricities between the centers of mass and centers of shear were modeled mathematically so that torsional effects could be considered. These models consisted of lumped masses having effective translational and rotational inertia connected by springs simulating the elastic restraints which included effective torsional stiffness.

A rational basis for the effect of seismic torsion has been developed by N.M. Newmark.[(1) (2)] A key parameter in Newmark's work is the transit time of the soil wave motion to pass over the long dimension of the building.

The shorter the time the less the torsional effect. Because Indian Point is on hard rock, the transit time is quite small. Therefore, the seismic torsional effect is not significant and can be neglected.

For the Containment, the concrete was assumed not to participate in resisting seismic shear even though experimental evidence suggested such contribution is significant, even for biaxially loaded concrete in tension. Therefore, the ductility of the shear resisting mechanism was taken to be provided entirely by the reinforcing steel acting in tension to carry diagonal tension loads.

For all other seismic Class I structures, the standard horizontal and vertical reinforcing in each face of walls and slabs provided the mechanism to resist shear loads which included torsional effects. The design was in accordance with the procedures in ACI-318-63 "Building Code Requirements for Reinforced Concrete," June 1963.

The locations of seismic supports and restraints for all seismic Class I piping, down to ¾ inch in diameter, were determined by the Architect Engineer and shown on installation drawings. In the event that a support or restraint could not be located as specified on the installation drawings, a reanalysis was performed prior to relocation. An as-built verification program has produced revised drawings, which reflect actual field conditions.

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Site Quality Control inspectors verified the final location, correct type of each device, and its proper installation. This verification was made by actual physical check using approved Architect-Engineer drawings, manufacturer drawings, contractor drawings. and benchmarks established for this purpose.

Splice Stagger in the Containment and Other Seismic Class I Structures.

In the Containment, seismic design criteria required that vertical rebar splices be staggered a minimum of 1'-2" and that seismic diagonal bar splices be staggered 1'-2" vertically in each direction . In the dome a 2'-0" stagger pattern was specified throughout for the Cadweld splices as well as the reinforcing splice plates, except for final closure pieces at the apex of the dome. Horizontal rebar splices were specified in elevation and in cross-section (bars or bar pairs) with 2'-4" nominal and 2'-0" minimum stagger.

The above requirements were generally satisfied during construction except in special cases where physical or layout problems occurred in isolated areas in the Containment.

For all seismic Class I structures, other than the containment, rebars were specified to be lap spliced in accordance with the requirements of

ACI-318-63 "Building Code Requirements for Reinforced Concrete." No other specific stagger requirements were formulated. In the Containment, mechanical splices were included in the design because of biaxial tensile stress conditions in the concrete which eliminate bond and require continuous rebar, and because of the ACI-318 requirement that lapped splices in tension cannot be used for bars greater than No. 11.

Splicing of Reinforcing Steel by Welding

Welding of rebar for splicing is not permitted. Strength welding of rebar to structural steel elements or other heavy rebar was not permitted. Tack welding of rebar was not permitted.

Although rebar was not welded it should be pointed out that transition and closure splices in the Containment Dome employed Cadweld splice sleeves welded to structural steel plate (ASTM A 516 GR60). In addition to the destructive testing of random samples employed for all cadwelding, the root and final pass of each weld was magnetic particle inspected.

Class II and III

All seismic Class II* structures and components were designed on the basis of a static analysis for a ground acceleration of 0.05g acting in the vertical and 0.1g acting in the horizontal directions simultaneously. The structural design of all seismic Class III structures met the requirements of the applicable building code which was the "State Building Construction Code," State of New York, 1961. This code does not reference the Uniform Building Code.

The design of seismic Class I piping was subject to loading combination and corresponding stress limits which included loads due to the Design Basis Earthquake and Operating Basis Earthquake while the seismic Class II piping was subject to loads associated only with the Operating Basis Earthquake.

It has been found that in some cases, in the Containment Building the loading combinations and stress limits involving the Operating Basis Earthquake will govern the design of piping systems

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with respect to seismic criteria. In these cases, seismic Class I or Class II piping were designed for Operating Basis Earthquakes. It was therefore designed for the governing condition.

In those cases where it was shown the loading combinations involving the Design Basis Earthquake governed, the adjacent seismic Class II piping and supports were designed to the seismic Class I criteria.

*NOTE: There are no seismic Class II structures.

Effects of Failure of Class III Equipment on Safety-Related Equipment

A review of potential failures of seismic Class III equipment and the potential adverse effects of such failures on safety related equipment was conducted.

The review consisted of determining the seismic Class III lines in the Diesel Generator Building, Vapor Containment, Fuel Handling Building, Service Water Pump Area, Control Building, Turbine Hall, Primary Auxiliary Building and the Auxiliary Boiler feed Pump Room and assessing the flooding potential from each line. This was accomplished by identifying the seismic Class III systems and portions of systems and tracing them through drawings for location and arrangement in the plant. It was determined from the review, that failure of seismic Class III equipment would not potentially adversely affect the performance of safety related equipment in the following buildings: Diesel Generator Building, Vapor Containment, Fuel Handling Building, Service Water Pump Area and Turbine Hall.

The portion of the Indian Point 3 water-medium Fire Protection System in the Diesel Generator Building, although classified as a seismic Class I system, was investigated for the effect of inadvertent actuation. Two 500 gpm sump pumps provided in this building were sized to accept water from the rupture of the diesel generator cooling water system. These pumps are controlled by independent float switch assemblies, each set at a different elevation to start the pumps in sequence. In case of a rupture of the service water pipe supplying water to the diesel generator cooling system, 24" drain lines collect water from the Diesel Generator Building and discharge into the river. The Diesel Generator Rooms are protected with a CO_2 fires suppression system, and the air supply to the engines is via a snorkel directly from the outside. Performance of the Diesel Generators would not be adversely affected by either actuation of the Fire Protection System or rupture of the service water piping in the building.

The original design of the CO_2 fire suppression systems protecting the diesel generator rooms was susceptible to inadvertent operation during a seismic event, seismic interaction, interaction resulting from a tornado generated missile or an adverse environment resulting from a high energy line break in the Turbine Building. This is because the control panels are non-safety-related, seismic Class III and are located in a non-safety-related, seismic Class III structure. Subsequent to the original design, a design change was implemented to install an interfacing safety-related, seismic Class I auxiliary control panel which prevents an inadvertent operation of the systems from resulting in a CO_2 discharge or result in an unacceptable loss of the ventilation systems which serve these rooms.

Essentially, all the equipment in containment is seismic Class I. Flooding in containment would be indicated within a few minutes by various methods, including humidity detectors and sump level sensors. A description of the leak detection systems is provided in Section 6.7.1.

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A portion of the spent fuel cooling loop in the Fuel Storage Building and Primary Auxiliary Building is classified as seismic Class III. The largest source of water in this building is the storage pool. The spent fuel storage pool cooling connections enter near the water level at the top of the pool with this system.

The overhead piping for the new screen wash system has been designed such that in case of failure:

1) All operating pumps will be stopped by a low pressure switch;
2) Failed piping shall drain through the screens and pump back into the bays; and
3) Any water spilled on 15 foot deck shall be insignificant to cause flooding.

No safety related equipment is located in the Turbine Hall. However, flooding from the Turbine Hall could potentially affect the performance of the 480 volt switchgear located in the Control Building at Elevation 15' only if the water reached the elevation of 15'6". Since the Circulating Water System is an open system having absolutely no valves, and therefore no means of producing a high dynamic head, the probability of a failure is practically zero. However, to assure that the 480 volt switchgear would not be adversely affected by flooding , redundant level alarm switches were installed in the pipe tunnel at Elevation 3'3" of the Turbine Hall. These switches sense high water in the pipe tunnel and give an indication to the Control Room. In addition, a barrier was installed at the doorway to the switchgear room to provide protection from flooding up to 19'. The operators have ample time to investigate any flooding problem and take appropriate action by shutting down the circulating pumps to prevent flooding to Elevation 19'.

A DBE seismically induced break of the Turbine Hall Elevation 15'-0" fire protection header could result in flooding of the 480 volt switchgear with a potential loss of all four 480 volt emergency buses. Modification 93-3-433 FRW added six seismic QA Category I supports so that the portion of the QA Category M piping and deluge valves for the water spray systems for Main, Station, Auxiliary, and Unit Auxiliary Transformers, located in the Control Building, will be capable of withstanding a Design Basis seismic event.

Inadvertent actuation of the Fire Protection System in the electrical tunnels will not potentially affect the performance of safety-related equipment in the Control Building. The electrical tunnels are provided with floor drains to handle water from the cable tray fire protection spray system. These drains discharge to grade outside the tunnel.

The original design of the CO_2 fire suppression systems protecting the 480 V switchgear room and cable spreading room was susceptible to inadvertent operation during a seismic event, seismic interaction, interaction resulting from a tornado generated missile or an adverse environment resulting from a high energy line break in the Turbine Building. This is because the control panels are non-safety-related, seismic Class III and are located in a non-safety-related, seismic Class III structure. Subsequent to the original design, a design change was implemented to install an interfacing safety-related seismic Class I auxiliary control panel which prevents an inadvertent operation of the systems from resulting in a CO_2 discharge in either room or results in an unacceptable loss of the ventilation systems which serve these rooms."

The Primary Auxiliary Building was so designed that flooding from any elevation will result in the water settling at the lowest level (Elevation 15') as each room has various floor penetrations which permit drainage to this elevation. In addition, the stairways provide substantial flow area. Performance of the two Residual Heat Removal Pumps located in the 15' elevation of the Primary Auxiliary Building would be affected by flooding only if the water an Elevation of 19'.

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The Systems Interaction Study Determined the effects of the internal flooding in the Primary Auxiliary Building due to failure of Class II and III piping.

The study concluded that Class II and Class III pipe breaks will result in a water level of 18'-5" after 9.9 hours. Approximately 120,000 gallons of water would be required to cause flooding to this elevation. The combined volume, approximately 2,800 gallons, of all non-Class I tanks in the Primary Auxiliary Building would cause negligible flooding if they failed. There are several seismic Class III lines and fire protection systems in the Primary Auxiliary Building that have sufficient capacity to cause flooding of the Residual Heat Removal Pumps. The seismic Class III line in the Primary Auxiliary Building with the largest nominal flow rate would take approximately 6 hours to flood to Elevation 19'. Although it is evident from the above that operators would have sufficient time to discover that a failure in seismic Class III line has occurred and take appropriate actions to prevent flooding to the 19' elevation in the Primary Auxiliary Building, modifications were made to assure that there is adequate drainage area to preclude flooding of the Residual Heat Removal Pumps in the unlikely event that the flooding is not discovered. The drainage area is also adequate to preclude flooding from the Fire Protection System. (See Section 9.6.2)

Evaluation of the Auxiliary Boiler Feed Pump Area, located between the Containment and the Shield Wall, revealed that safety related equipment would not be affected by failure of the seismic Class II portion of the main steam system. Failure of the main feedwater lines, located above and the outside of the Auxiliary Boiler Feed Pump Room, would result in water accumulating at the 18'6" elevation. Performance of the Auxiliary Boiler Feed Pumps could be adversely affected only if the water reached Elevation 19'8" in the Auxiliary Boiler Feed Pump Room. Provisions were made to assure adequate drainage under the worst postulated conditions of the main feedwater line failure.

Installation of level alarm switches in the Turbine Hall and provisions in the Primary Auxiliary Building and the Auxiliary Boiler Feed Pump Area were made during the normal course of construction of the plant.

Also included in the review was the potential effect of chemical releases on safety related equipment. It was determined that chemical releases caused by failure of seismic Class III equipment would have no potential adverse effect on safety related equipment.

<u>Ground Response Spectra</u>

The seismic ground response spectra used in the design of Indian Point 3 are shown in Figure 16.1-1 for the Operating Basis (smaller) Earthquake maximum ground acceleration of 0.10g, and in Figure 16.1-4 for the Design Basis (larger) Earthquake maximum ground acceleration velocity 0.15g. The response spectra were developed from the average acceleration velocity displacement curves presented in TID-7024, Nuclear Reactors and Earthquake, for large-magnitude earthquakes at moderate distances from the epicenter. As such, the curves are made of the combined normalized response spectrum determined from components of four strong-motion ground accelerations: El Centro, California, December 30, 1934; El Centro, California, May 18,1940, Olympia, Washington, April 13, 1949 and Taft, California, July 21, 1952.

Figures 16.1-2 and 16.1-3 are plots of the smoothed site ground response spectra and the ground response spectra derived from the earthquake records from 2 and 5 percent damping.

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The smoothed response spectra plot was taken from Figure 16.1-1. The system period interval where response spectra acceleration values were calculated was 0.02 seconds.

As seen in Figure 16.1-2 and 16.1-3, the computed ground response spectra using a system period interval of 0.02 seconds is equal to or greater than the smooth response spectra for the site.

To assure that the flow response spectra conservatively reflect the effects of variations in assumptions made for structural properties, dampings, and soil structure interactions, the response spectra peaks were widened. This widening effect was applied to peak values and is proportional to the response frequency. The increase in width is greater at high frequencies. The sharp valleys due to discontinuities in the plot were raised by an averaging technique.

In order to reflect in a conservative manner the expected variations of the periods of vibration of the structures in the seismic response curves for Seismic Class I buildings, the response spectra peaks were extended in the period scale by an amount equal to or greater than ± 8.5%.

For seismic Class I structures, having peaks occurring above 5 cps, the peaks were widened by more than ±10%. The only structures which are widened by less than ± 10% are the Containment structure and the Shield Wall. The difference between the widening percentage used for these structures, whose significant peaks occur below 5 cps, and the widening percentage of ±10% was less than 0.1 Hz which is negligible compared to the significant peak frequency. Where this difference can be significant, peaks occurring at frequencies above 5 cps, the peaks were widened by more than ±10%.

Since no strong motion records were available for the Eastern United States, the method used appeared to be the most rational considering the amount of earthquake data currently available. In addition, this method was consistent with the procedure being carried out on the majority of the nuclear plants under construction at that time in the United States.

There was not sufficient data available at that time, particularly in the Eastern United States, to attempt to correlate specific site conditions to a particular response spectrum.

Damping Factors

Table 16.1-1 gives the damping factors used in the design of seismic Class I components and structures.

Combined Horizontal and Vertical Amplified Response Loading

Evaluations were made for the simultaneous occurrence of horizontal and vertical seismic input motions. The results of analyses for each of two orthogonal, horizontal directions of excitation were combined directly with the results for vertical excitation on the basis of absolute sums for piping systems analyzed by Westinghouse and on the basis of algebraic summation for piping systems analyzed by UE&C to verify the static design. Vertical response was assumed to be amplified to the same degree as horizontal motions were amplified in determining floor response motions. Since vertical ground motions were assumed equal to two-thirds of the horizontal ground motions, the resultant vertical floor response spectrum values are two-thirds of those values determined for horizontal floor response. If the combined modal responses for the two horizontal and the vertical directions were combined by the square-root-sum-of-the-squares, bases on statistical independence, the resulting stresses would bot be significantly different

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because of the conservative vertical floor response spectra values assumed on the absolute sum analysis method employed by Westinghouse.

Floor response spectra were generated by plotting maximum dynamic response (acceleration and / or velocity and / or displacement) versus the natural frequency of a series of single-degree-of-freedom oscillators for the floor time history input motion. The time history motion of a floor was determined by dynamic analysis of a multi-degree-of-freedom lumped mass and elastic spring model of the building using a time history ground input motion. The time history ground input was defined such that its ground response spectrum simulates the defined ground response spectrum for the site.

For analysis of mechanical components and piping systems, the modal deflections, forces, and stresses for each mode were computed utilizing the spectral response method for seismic analysis.

The combined total response was obtained by adding the individual modal responses utilizing the square-root-sum-of-the-squares method. Combined total response for closely spaced modal frequencies whose eigen vectors were orthogonal were handled in the above mentioned manner. In the rare event when two significant closely spaced modal frequencies occurred and their eigen vectors were parallel, the combined total response was obtained by adding the RMS values of all other modes to the absolute value of one of the closely spaced modes for the main reactor coolant piping. Since the probability of such a rare event was small, this was disregarded for all systems other than the reactor coolant piping. Forces, moments, deflections, etc., were determined in each mode separately and then combined to determine resultant values. Resultant shears and stresses were computed from these resultant forces and moments.

Natural modes of vibration are normally considered statistically independent and, therefore, a realistic total response was obtained by taking the square root of the sum of the squares of the individual modal responses. However, if significant natural frequencies were closely spaced and their eigen vectors were parallel, the natural modes were assumed to be statistically dependent. Therefore, the absolute value of the response in one of the significant closely spaced modes was added to the square-root-of-the-sum –of-the-squares of all the other modal responses. Two natural frequencies were considered to be closely spaced if their difference was less than ten percent of either value.

16.1.4 Seismic Class I Design Criteria for Vessels and Piping

The loading combinations and stress limits which were employed in the design of seismic Class I piping, vessels, supports, and other applicable components are shown in Table 16.1.-2. The stress limits presented in Table 16.1-2 were used only in conjunction with elastic system dynamic analyses and elastic components analyses.

The emergency condition stress limits were applied to all seismic Class I piping systems outside of the Reactor Coolant Pressure Boundary under load combination of Normal + DBE. This included the steam and feedwater lines inside the Containment, up to and including the isolation valves outside the Containment.

Where restraints on any pipe line were necessary in order to prevent impact on and subsequent damage to neighboring equipment or piping comprising the Reactor Coolant Pressure Boundary, etc., the piping restraint was designed such that a plastic hinge mechanism was not

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formed. For these systems, the stresses due to postulated pipe rupture loads were maintained within the faulted condition limits.

The design criteria in Table 16.1-2 list loading combinations and stress limits for piping and supports for normal, upset, and faulted categories. Criteria for restraints required that stress limits of supported equipment not exceed code limits for the applicable category. For the seismic Class I portion of the main steam and feedwater systems, loading due to pressure, deadweight, thermal, transient pressure, transient temperature, operating basis earthquake, design basis earthquake and pipe break were considered. For the seismic Class III portion of these systems, loadings due to pressure, deadweight, thermal, transient temperature were considered. In addition, operating basis earthquake loads were considered to the extent that they affect the seismic Class I portion and pipe breaks were considered insofar as a break in the seismic Class III portion may not cause a failure of the seismic Class I portion nor cause a violation of the Containment.

The water hammer effect during a postulated loss-of-offsite-power and / or a loss-of-coolant accident were considered in the design of seismic Class I service water piping and pipe supports in containment. This transient (water hammer) could be the result of an earthquake but the effect would be separated in time wherein seismic and transient (water hammer) loading are not combined.

Allowable stress or rated load criteria are contained in the Power Piping Code ANSI B31.1 (1967), the Manufacture's Standardization Society standard MSS-SP-58 for standard supports, or AISC-69 for non-standard supports.

For the seismic Class I portion of the main steam line out to the isolation valves, the restraints at the steam stop valves were designed for a steam pipe break load of 340 kips. Under this load, the maximum applied primary load or stress was limited to the yield strength of the material. The analytical methods used in designing and evaluating the design of the main steam line restraints are in fact steel structures, most of the calculations were based on beam diagrams and formulas for various static loading conditions.

The design of all mechanical supports and restraints of the main steam and feedwater lines was evaluated by an individual other than the designer. Both the designer and the evaluator were graduated structural engineers qualified in structural stress analysis. In the evaluation, consideration was given to the design criteria, allowable stresses and loading combinations, and the analytical methods used in the design.

To perform their function, i.e., allow core shutdown and cooling, the reactor vessel internals must satisfy deformation limits. For this reason the reactor vessel internals were treated separately in Section 14.3.4.

Piping, Vessels and Supports

The reasoning for selection of the above mentioned loading combinations and stress limits was as follows:

1) For the operating basis earthquake, the nuclear steam supply system was designed to be capable of continued safe operation. Equipment and supports needed for this purpose were required to operate within normal design limits as shown in Line 2 of Table 16.1-2.

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2) In the case of the design basis earthquake, it was necessary to ensure that components required to shut the plant down and maintain it in safe shutdown condition do not lose their capability to perform their safety function. This capability was ensured by maintaining the stress limits as shown in Line 3 of Table 16.1-2. No rupture of a seismic Class I pipe can be caused by the occurrence of the design basis earthquake.

3) For the assumed case of a reactor coolant pipe rupture, limit stresses in the unbroken reactor coolant system legs and other seismic Class I vessels and pipes were again as noted in Line 4 of Table 16.1-2.

4) For the extremely unlikely event of the simultaneous occurrence of the design basis earthquake and a reactor coolant system pipe rupture the design of seismic Class I piping and components, excluding the broken pipe, was checked for no loss of function, i.e., the capability to contain fliud and allow fliuid flow. Again this was assured by limiting the various stress combinations within the limits shown in Line 5 of Table 16.1-2.

<u>Reactor Vessel Internals</u>

<u>Design Criteria for Normal Operation</u>

The internals and core were designed for normal operating conditions and subjected to loads of mechanical, hydraulic, and thermal origin. The response of the structure under the operating basis earthquake was included in this category.

The stress criteria established in Section III of the ASME Boiler AND Pressure Vessel Code, Article 4, were adopted as a guide for the design of the internals and core with exception of those fabrication techniques and materials which were not covered by the Code, such as the fuel rod cladding. Seismic stresses were conservatively combined and considered primary stresses.

The members were designed under the basic principles of:

1) Maintaining distortions within acceptable limits,

2) Keeping the stress levels within acceptable limits, and

3) Preventing fatigue failures.

<u>Design Criteria for Abnormal Operation</u>

The abnormal design condition assumed blowdown effects due to a pipe break combined in the most unfavorable manner with the effects associated with the design basis earthquake.

For this condition the criteria for acceptability were that the reactor be capable of safe shutdown and that the engineered safety features be able to operate as designed. Consequently, the limitations established on the internals for these types of loads were concerned principally with the maximum allowable deflections. The deflection and stress criteria for critical components under normal operation, plus the design basis earthquake and blowdown excitation are presented in Section 14.3.4.

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Movement of Reactor Coolant System Components

The criterion for movement of the reactor vessel, under the worst combination of loads, i.e., normal plus the design basis earthquake plus reactor coolant pipe rupture loads, was that movement of the reactor vessel not exceed the clearance between a reactor coolant pipe and the surrounding concrete to prevent excessive shear load on the RCS pipe should this limit be more restrictive than those listed in Table 16.1-2.

The relative motions between reactor coolant system components are controlled by the structures which are used to support the reactor vessel, the steam generators, the pressurizer and the reactor coolant pumps in such a way that the stresses in the carious components and pipes do not exceed the limits established in Table 16.1-2.

Effect of Fabrication and Environment on Materials Properties

The employment of qualified welding procedure and qualified welders and thorough inspections assured that welds on seismic Class I components and piping have little, if any, effect on the tensile properties of base materials. Tests performed by Westinghouse revealed no difference in tensile properties between welded and non-welded pipes.

Accidental imperfections of the order of magnitude of those that pass inspection were also expected to be of no significance. Because of chemistry control of the employed coolants and periodic inspections, corrosion was not anticipated to be a problem.

The only component affected by irradiation is the reactor vessel. Irradiation of the reactor vessel is significant only in the area adjacent to the core. High stress areas, i.e., nozzle to shell junctures, are only slightly affected by irradiation. The neutron exposure to these areas was calculated and its effect on the stress-strain curve evaluated. The corrected stress-strain curve was then used in the development of the limit curves.

Development of the Faulted Condition Stress Limits

The design limit curves that give the allowable piping and vessel stresses for faulted conditions were developed by using the approach presented in WCAP 5890, Rev. 1. [3] This report developed limit curves by using 50 percent of the ultimate strain as the maximum allowable membrane strain. Subsequent to the submission of WCAP 5890, Rev. 1, the allowable membrane strain was limited to 20 percent of the uniform strain. Design limit curves were developed by using the following procedure:

1) Use material data to develop stress-strain curves.

   Stress-strain curves of type 304 stainless steel Inconel 600 and SA 302B low alloy steel at 600 F were generated from tests using graphs of applied load versus cross-head displacement as automatically plotted by the recorder of the tensile test apparatus. The scale and sensitivity of the test apparatus recorder assure accurate measurement of the uniform strain.

   For materials other than these three, stress-strain curves were developed by conservative use of pertinent available material data (i.e., lowest values of uniform strain and initial strain hardening). When the available data was not

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sufficient to develop a reliable stress-strain curve, three standard ASTM tensile tests of the material in question were performed at design temperature. These data were conservatively applied in developing a stress-strain curve.

2) Normalize the ordinate *stress) of the stress-strain curves to the measured yield strength. (Figures 16.1-5, 16.1-6, and 16.1-7)

3) Use 20% of uniform strain as defined on the curve developed under Item 2 as the allowed membrane strain.

4) Establish the normalized stress ratio at 20% of uniform strain on the normalized stress ratio-strain curves developed under Item 2.

5) Establish the value of the absolute membrane stress limit.

   Multiply the normalized stress ratio in Item 4 by the applicable code yield strength at the design temperature to get the membrane stress limit which represents a minimum value. As an alternate, the actual physical properties as determined from standard ASTM tensile tests on specimens from the same heats were used to determine the membrane stress limits. If such an approach was adopted, sufficient documentation was provided to support the actual material properties used.

6) Develop limit curves for the combination of local membrane and bending stresses.

   The limit curves were developed by using the analytical approach presented in WCAP 5890, Rev.1, and the stress-strain curve up to the membrane stress limit as developed under Item 5. Stress and stability analysis results were compared with these limits.

   Examples of design limit curves as developed by using the above procedure are given in Figures 16.1-8 and 16.1-9.

16.1.5 Seismic Design Bases

Design Organization Involved

The design organization which were involved in the seismic design of Indian Point 3 and their responsibilities were as follows:

- Westinghouse

  Responsible for performing a dynamic analysis of the plant seismic Class I structures using a modal analysis approach. Also responsible for preparing response acceleration spectra at selected points in the plant structures for use in the seismic analysis of piping and equipment. Also responsible for the seismic analysis and design of main coolant piping and nuclear steam supply system equipment.

- United Engineers and Constructors

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Responsible for overall coordination of seismic design. Also responsible for seismic design of structures based upon accelerations, shears and moments determined by Westinghouse in their dynamic analysis. Also responsible for analysis and design of balance-of-plant piping and equipment.

The design of all structures and equipment for the plant were either within the scope of supply of Westinghouse or United Engineers and Constructors (UE&C). UE&C had the overall responsibility for the proper execution of the seismic design.

The safety related items of equipment furnished with the nuclear steam supply system underwent seismic analysis by Westinghouse and (where applicable) Westinghouse subcontractors. Westinghouse had the responsibility for approving analysis performed by its subcontractors.

The overall program and the criteria employed were evaluated by Westinghouse. Records of the documented procedures which were followed in this work are applicable to all phases of design, interchange of design information among the involved organizations, revisions thereto, and coordination of all aspects of design (including seismic design), were maintained by the Authority, now Entergy.

All items within the plant were clearly identified as to their importance to overall safety and were classified as seismic Class I, II or III. Major structures, system and components and their respective classifications are listed in Section 16.1.2.

The design engineer utilized the appropriate generated response acceleration spectra to determine the appropriate earthquake loadings.

In order to assure that UE&C responsibilities were met with regard to structural seismic design, each member of the design group was issued a document containing design procedures to convert the analyses results to working drawings for construction. The result of the Westinghouse dynamic analysis, transmitted to UE&C, included shear and moments at critical portions of each Class I Structure. These results were used to design the structural elements to resist these loads in accordance with the criteria and the applicable codes referenced. In those cases where the structure had already been designed by UE&C using the peak of the applicable response curve, the Westinghouse results were checked to insure that they were less than all shears and moments used in design.

Documentation Procedures

The major interface regarding seismic design information was the flow of information between the designer of the structure and designer of the equipment and components which are attached to the structures. The cognizant Structural, Mechanical, Electrical and Instrumentation engineers each had structures and/or equipment for which they had lead responsibility. All of these groups were serviced by a mechanical "analysis" group which performed appropriate analyses which could be translated into loads and stresses and other design information for use by the various designers. The required seismic information was transmitted in writing to the cognizant engineer and this information became part of the design basis information for the structures or components. The design information was reviewed by the designer, the cognizant engineer and the independent second level reviewer.

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Upon completion of the Westinghouse modal analyses of seismic Class I structures, the shear and moments at various elevations were transmitted to UE&C for their use in design. A copy of this information was given to the responsible designer for use in implementing the design. All correspondence was kept in a separate job file by chronological order to insure that the latest information was available. Although there were several revisions to some of the Westinghouse information (received after the initial UE&C designs were completed), this revised Westinghouse information was reviewed to assure adequacy of the original design. Where necessary, revisions were made.

For the Containment Building and Control Building, UE&C performed an independent modal analysis to verify the Westinghouse analysis. The results were sent to Westinghouse for their records and information.

When UE&C drawings were completed or revised, they were issued for construction. Copies were transmitted to WEDCO, Consolidated Edison, and Westinghouse. In the UE&C offices, revised drawings were removed from all files and marked void. The latest revisions were then substituted.

Design Control Measures

Each Engineering Division within United Engineers and Constructors and Westinghouse was responsible for the adequacy of the design produced by those divisions. As such it was the responsibility of the managers of the respective divisions to provide adequate controls to assure satisfactory designs.

After receipt of Westinghouse information, UE&C proceeded with drawing preparation. All completed drawings were independently checked by another designer to insure adequacy of design with regard to design criteria. In addition, the drawings were given an overall check by the design leader and a cursory check by the structural discipline engineer. The drawing was finally signed by the project manager. In addition, all containment structural drawings were transmitted to Westinghouse for approval prior to issue for construction. When the drawing was issued for construction, the letter giving Westinghouse approval was documented in the drawing title block for quick reference.

Purchase Requirements

Specifications issued by both Westinghouse and United Engineers and Constructors included, as a minimum, loading criteria equal to or greater than those developed for a given location in the Indian Point 3 structures. The equipment and component suppliers were required to perform analyses or tests verifying the design and integrity of safety related components, using the appropriate criteria as inputs, or the analysis was done by UE&C or Westinghouse.

All seismic analyses submitted by vendors supplying seismic Class I equipment, which were not under the scope of the Nuclear Steam Supply System Contract, were reviewed by the architect-engineer (UE&C). This review consisted of the following:

1) All seismic calculations or test reports were reviewed by the responsible principal engineer.

2) A separate independent review was performed by the responsible Analytical Division.

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3) The results of the above reviews were coordinated by the principal engineer and comments were returned to the vendor.

4) Final acceptance of the adequacy of the seismic design was confirmed in writing by UE&C after all comments had been resolved.

All seismic analyses submitted by subcontractors to Westinghouse were reviewed by Westinghouse equipment and analytical engineering departments in a manner similar to that described above.

For safety related seismic Class I electrical and control equipment, type tests or analyses were conducted under seismic accelerations based upon the results of a multi-degree-of-freedom, time-history analysis of the structure and applicable frequencies to demonstrate the ability of the equipment to perform its functions.

The analyses, test procedures, and test reports as submitted, whichever were applicable, were reviewed by either Westinghouse or UE&C, depending upon the origin of the specification. The requirements for submittal and approval was included in the specification.

16.1.6 <u>Procedure for Utilization of Station Seismic Monitoring Equipment Following an Earthquake</u>

The purpose of this procedure is to provide a plan for the utilization of data from the seismic monitoring equipment installed at Indian Point 3 following an earthquake.

Use of this data, as specified in the procedure, enables the station operating personnel to determine what course of action to take following an earthquake.

<u>Equipment</u>

The seismic monitoring system consists of equipment located as follows:

<u>Containment</u>

1) Three Engdahl Enterprises Peak Shock Recorders, Model PSR 1200-H-V-12A, installed in a tri-axial mount at Elevation 46'- 0" on the base mat. These provide a plot of eleven points on the 2% damping curve for the vertical axis and two horizontal area. The eleven points are within the frequency range of 2.26Hz.

2) Two Kinematics, Inc. SMA-2 tri-axial strong motion accelero-graphs one installed at Elevation 46'-0" on the base mat and one installed on the Containment Structure Wall at Elevation 99'-0" directly above the lower unit.

3) One Teledyne PRA-103 Peak Recording Accelerograph installed on each of the three following pieces of equipment:
   a) One steam generator
   b) One reactor coolant pump
   c) The pressurizer

<u>Control Room, Elevation 53'</u>

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1) Two Kinemetric, Inc. magnetic tape recorders to receive and record the data from the two SMA-2 accelerographs in containment.

2) A Kinemetrics, Inc. SMP-1 Magnetic Tape Playback Unit.

3) An Engdahl Enterprise Model PSA-1575 Peak Shock Annunciator for visual warning that predetermined acceleration limits making up the 2% damping response spectrum have been exceeded at any or all of the eleven frequencies monitored.

Alarm

In the event of a strong motion earthquake, magnitude 0.01g or greater, an alarm will be annunciated in the Control Room that a seismic event is being recorded by the strong motion accelerographs.

Action Required

The actions required following an alarm are detailed in operating procedures available at the plant site.

16.1.7 Categorization of Structures, Systems and Components

The structures, systems and components of Indian Point 3 can be classified to lie within the following categories:

Category I

A system, part of a system, structure, and/or component shall be deemed Category I if it is necessary to ensure: 1) the integrity of the reactor coolant pressure boundary, or 2) the capability to shut down the reactor and maintain it in a safe, shutdown condition, or 3) the capability to prevent or mitigate the consequences of accidents that could result in potential off-site exposures comparable to the guidelines of 10CFR100. Some Category I structures, systems and components are listed in Table 16.1-4. A detailed listing may be found in the ENN-DC-167 Reference Document and the equipment database.

Non-Category I

A system, part of a system, structure, and/or component shall be deemed Non-Category I, if it is not essential for a safe shutdown, i.e., hot shutdown. Failures of this equipment could result in loss of power generation but would not endanger public safety.

Category M

A Non-Safety Related system, part of a system, structure and/or component shall be deemed Category M if it is a system, structure or component that performs a function which may have some significance to safety with respect to design criteria to which the Quality Assurance Program must be applied as applicable. Some of these systems and components are listed in Table 16.1-5. A detailed listing may be found in the ENN-DC-167 Reference Document and the equipment database.

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16.1.8 Use of Generic Implementation Procedure (GIP) for Seismic Adequacy of Equipment and Parts

The GIP (Reference 4), as modified and supplemented by the U.S. Nuclear Regulatory Commission Supplemental Safety Evaluation Rep[ort No. 2 (Reference 5), may be used as an alternative method to existing methods for the seismic design and verification of existing, modified, new and replacement equipment and parts classified as Seismic Class I.

Only those portions of the GIP listed in "Use of Generic Implementation Procedure (GIP) for New and Replacement Equipment and Parts (NARE)" (Reference 6) shall be used. The other portions of the GIP are not applicable since they contain administrative, licensing, and documentation information which is applicable only to the Unresolved Safety Issue (USI) A-46 program. GIP shall be used with limitations stated in IP3 Nuclear Safety Evaluation NSE 94-3-029 (Reference 7).

References

1) Newmark, N. M., "Torsion in Symmetrical Buildings," Proceedings of the Fourth World Conference on Earthquake Engineering, Santiago, Chile, 2, A-3, 1969, p. 19-32.

2) Newmark, N. M. and Rosenblueth, E. Fundamentals of Earthquake Engineering, Prentice-Hall, Inc., New Jersey, 1971.

3) Wiesemann, R. E., R. E. Tome and R. Salvatori, "Ultimate Strength Criteria to Ensure No Loss of Function of Piping and Vessels under Earthquake Loading", WCAP-5890, Revision 1.

4) Seismic Qualification Utility Group (SQUG), "Generic Implementation Procedure (GIP) for Seismic Verification of Nuclear Plant Equipment", Revision 2, 02/14/1992.

5) Nuclear Regulatory Commission, "Supplement No. 1 to Generic Letter (GL) 87-02 That Transmits Supplemental Safety Evaluation Report No. 2 (SSER No. 2) on SQUG Generic Implementation Procedure Revision 2 As Corrected on February 14, 1992 (GIP-2)", May 22, 1992.

6) Seismic Qualification Utility Group (SQUG), "Use of Generic Implementation Procedure (GIP) for New and Replacement Equipment and Parts (NARE)", Revision 2, October 25, 1999.

7) Indian Point 3 Nuclear Safety Evaluation NSE 94-3-029, "Seismic Verification of Equipment by SQUG Generic Implementation Procedure (GIP)", Rev. 1, 11/23/1999.

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TABLE 16.1-1

DAMPING FACTORS FOR CLASS I COMPONENTS AND STRUCTURES

Component		Per Cent of Critical Damping
Containment Structure:		
(a)	Design Basis Earthquake (larger)	5.0
(b)	Operating Basis Earthquake (smaller)	2.0
Concrete Support Structure of Reactor Vessel:		2.0
Steel Assemblies:		
(a)	Bolted or Riveted	2.5
(b)	Welded	1.0
Concrete Structures above Ground:		
(a)	Shear Wall	5.0
(b)	Rigid Frame	5.0
Piping		0.5

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TABLE 16.1-2

LOADING COMBINATIONS AND STRESS LIMITS

OPERATING CONDITION AND LOADING COMBINATIONS	VESSELS	PIPING	SUPPORTS	PUMPS	VALVES
1. Normal (Deadweight, Thermal and Pressure)	$P_m \leq S_m$ $P_L \leq 1.5\ S_m$ P_m (or P_L) + $P_B \leq 1.5\ S_m$	$P \leq *$	Within stress limits as provided by applicable code either AISC-69 or MSS-SP-58	ASME Section III, 1968 Edition	USAS 16.5 or MSS-SP-66
2. Upset (Normal + Operating Basis Earthquake)	P_m (or P_L) + P_B + Q $\leq 3.0\ S_m$ (See Notes 1 & 2)	$P \leq 1.2 *$			
3. Faulted (Normal + Design Basis Earthquake Loads)	(a) $P_m < 1.25_m$ or S_y' Whichever is larger $P_L < (1.25_m)$ or 1.5 S_y' Whichever is larger P_m (or P_L) + $P_B < 1.5\ (1.25_m)$ Or 1.5 S_y whichever is larger (See Note 3) or (b) Faulted Condintion Stress Limits in Table 16.1-3	Design Limit Curves as discussed in the text (also see Note 4)	Permanent Deflections of Supports Limited to Maintain Supported Equipment Within Faulted Condition Stress Limits.	Maximum Average Membrane Stress ≤ 2.4 S	Maximum Average Membrane Stress ≤ 2.4 S
4. Faulted (Normal + Pipe Rupture Loads)					
5. Faulted (Normal + Design Basis Earthquake + Pipe Rupture Loads).					

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TABLE 16.1-2
(Cont.)

LOADING COMBINATIONS AND STRESS LIMITS

Where	P_m	=	primary general membrane stress intensity
	P_L	=	primary local membrane stress intensity
	P_B	=	primary bending stress intensity
	S	=	allowable value as specified in design codes
	S_m	=	stress intensity value from ASME B&PV Code, Section III
	P	=	piping stress calculated per USAS B31.1 Code for Power Piping.
	s	=	allowable stresses from USAS B31.1 Code for Power Piping. These limits may also apply to ASME Class C vessels
	Q	=	secondary stress intensity
	S_y	=	minimum specified material yield (ASME B&PV Code, Section III, Table N-421 or equivalent)

Note 1: The limits on local membrane stress intensity ($P_L < 1.5S_m$) and primary membrane plus primary bending stress intensity (P_m (or P_L) + $P_B < 1.5S_m$) need not be satisfied at a specific location if it can be shown by means of limit analysis or by tests that the specified loadings do not exceed 2/3 of the lower bound collapse load as per paragraph N-417.6 (b) of the ASME B&PV Code, Section III, Nuclear Vessels.

Note 2: In lieu of satisfying the specific requirements for the local membrane ($P_L < 1.5S_m$) or the primary plus secondary stress intensity (P_m (or P_L) + P_B + $Q < 3S_m$) at a specific location, the structural action may be calculated on a plastic basis and the design will be considered to be acceptable if shakedown occurs, as opposed to continuing deformation, and if the deformations prior to shakedown do not exceed specified limits, as per paragraph N-417.6(a) (2) of the ASME B&PV Code, Section III, Nuclear Vessels.

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TABLE 16.1-2
(Cont.)

LOADING COMBINATIONS AND STRESS LIMITS

Note 3: The limits on local membrane stress intensity ($P_L < 1.8S_m$ or $1.5S_y$) and primary membrane plus primary bending stress intensity (P_M (or P_L) + $P_B < 1.8S_m$ or $1.5S_y$) need not be satisfied at a specific location if it can be shown by means of limit analysis or by tests that the specified loadings meet the requirements of paragraph N-417.10 (c) of the ASME B&PV Code, Section III, Nuclear Vessels; or , for Steam Generators, that the specified loadings do not exceed eighty percent of the lower bound collapse load.

Note 4: As an alternate to the design limit curves which represent a pseudo plastic instability analysis, a plastic instability analysis may be performed in some specific cases considering the actual strain-hardening characteristics of the material, but with yield strength adjusted to correspond to the tabulated value at the appropriate temperature in Table N-424 or N-425, as per paragraph N-417.11 (c) of the ASME B&PV Code, Section III,Nuclear Vessels. These specific cases will be justified on an individual basis.

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TABLE 16.1-3

FAULTED CONDITION STRESS LIMITS FOR CLASS I VESSELS

System (or Subsystem) Analysis	Component Analysis	Stress Limits for Vessels		Test
		P_m	$P_m + P_B$	
ELASTIC	Elastic	Smaller of $2.4\ S_m$ and $0.70\ S_u$	Smaller of (2) $3.6\ S_m$ and $1.05\ S_u$	
	Plastic	Larger of (3) $0.70\ S_u$ or $S_y + 1/3\ (S_u - S_y)$	Larger of (3) $0.70\ S_{ut}$ or $S_y + 1/3\ (S_{ut} - S_y)$	
	Limit Analysis	$0.9\ L_1$ (3) (1)		$0.8\ L_T$ (3) (4)
PLASTIC	Plastic Elastic	Larger of $0.70\ S_u$ or $S_y + 1/3\ (S_u - S_y)$	Larger of $0.70\ S_{ut}$ or $S_y + 1/3\ (S_{ut} - S_y)$	

NOTE:

(1) L_1 = Lower bound limit with assumed yield point equal to $2.3\ S_m$

(2) These limits are based on a bending shape factor of 1.5. For simple bending cases with different shape factors, the limits will be changed proportionally.

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TABLE 16.1-3
(Cont.)

FAULTED CONDITION STRESS LIMITS FOR CLASS I VESSELS

(3) When elastic system analysis is performed, the effect of component plastic deformation on the dynamic system response will be checked. When this method is used, justification will be provided to show that the results of the elastic system analysis are valid.

(4) The limits established for the analysis need not be satisfied if it can be shown from the test of a prototype or model that the specified loads (dynamic or static equivalent) do not exceed 80 percent of L_T, where L_T is the ultimate load or load combination used in the test. In using this method, account shall be taken of the size effect and dimensional tolerances (similitude relationships) which may exist between the actual component and the tested models to assure that the loads obtained from the test are a conservative representation of the load carrying capability of the actual component under postulated loading for faulted conditions.

S_y = Yield stress at temperature

S_u = Ultimate stress from engineering stress-strain curve at temperature

S_{ut} = Ultimate stress from true stess-strain curve at temperature

S_m = Stress intensity from ASME Section III at temperature

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TABLE 16.1-4

CATEGORY 1

SAFETY-RELATED SYSTEMS

- Reactor Coolant System
  Includes: Pressurizer System and Associated Safety and Relief Valves
- Secondary Coolant System up to Second Isolation Valve
  Includes: Secondary Relief, Auxiliary Feedwater and Boiler Blowdown, with the exception of the motor-operated block valves and low-flow bypass valves which are exempted per NUREG-0800 criteria
- Chemical and Volume Control System
- Sampling System
- Containment Ventilation System
  Includes: Containment Air Recirculation Cooling and Filtration System
- Containment Spray
- Waste Disposal System
- Service Water-Essential Header
- Instrument Air System
- Fuel Handling System
- Reactor Protection System
- Engineering Safety Systems Protective System
- Process and Area Radiation Monitoring System
- Emergency Power System
- Containment Penetration and Weld Channel Pressurization System
- Isolation Valve Seal Water System
- Hydrogen Recombiner System
- Safety Injection System
- Component Cooling System
- Residual Heat Removal System
- Spent Fuel Cooling System
- Control Room Ventilation System
- Fuel Building Emergency Exhaust System

SAFETY-RELATED STRUCTURES

- Containment

SAFETY-REALTED COMPONENTS

- Core and Reactor Internals
- Control Rods and Drives
- Incore Thermoncouples
- Temperature Sensors in Auxiliary Feedwater Pump Room

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TABLE 16.1-4
(Cont.)

SAFETY-REALTED CONSUMABLES

- Diesel Generator Fuel Oil
- Boric Acid
- Lubricating Oils for Safety-Related Components
- Demineralizer resins for CVCS
- Sodium Hydroxide for Containment Spray System
- Weld Rod for Safety-Related Items
- Hydraulic Snubber Fluid

SAFETY-REALTED PROGRAM COMMITMENTS

- All itmes designated in Design Specification as ASME Section III, Classes 1, 2, and 3.
- Generic Letter 89-10 Motor Operators

NOTE: A detailed listing of structures and systems is provided in the ENN-DC-167 Reference Document. "QA Category I" denoted the highest classification applicable to the system, structure or component. Lower QA categories may exist within these systems, structures or components. Component QA categorization may be found in the equipment database.

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TABLE 16.1-5

CATEGORY M

NON-SAFETY RELATED STRUCTURES, SYSTEMS AND COMPONENTS

- Packaging of Radioactive Materials for Transport and transportation of Radioactive Materials Under Certain Conditions
- Low Level Radiation Waste Storage Tanks (additions)
- Fire Protection System
- Meteorological Tower
- Temperature Sensors in Penetration Area of Primary Auxiliary Building
- Level Sensors – Lower Level Turbine Building
- Seismic Monitoring System
- Manipulator Crane
- Containment Polar Crane
- Instrumentation (e.g., indicators, recorders, alarms, etc.) not already specifically classified as Category I by other sections of the FSAR that is required for:
    - Executing emergency procedures,
    - Verifying that plant conditions are within limits of Technical Specifications, or
    - Determining the status of Category I equipment including bypasses and permissives
- Level Transmitters: LT-181A, 181B
- Hot Penetration Blower No.'s 31, 32, 33, & 34
- Steam Generator Feed Flow, Steam Flow and Level Recorders
- Spent Fuel Pit Bridge
- Six Pipe Plugs Located on the RCP Motor Flywheel
- Emergency Diesel Generator Starting Air Compressors and Controls
- Manual Handwheel Actuators
- Retainer Clips and Bolts for Closure Head O-Rings on Reactor Vessel
- Turbine Control Oil Auto Stop Trip
- Cotter Pins for VC Airlock and Equipment Hatch
- Changing Pump O-Rings and Gaskets
- AMSAC System
- Fuel Storage Building Crane

NOTE: This lists only a portion of those "category M" non-safety related structures, systems, and components and instrumentation to which the QA Program must be applied, as applicable. A detailed listing may be found in the ENN-DC-167 Reference Document and the equipment database.

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16.2 TORNADO DESIGN CRITERIA

16.2.1 Definition of Design Basis Tornado

The plant is safeguarded from the tornados by the combined use of buildings and structures designed to withstand tornados, and by redundancy of components. All Class I buildings and structures were designed to withstand tornado winds corresponding to 300 mph tangential velocities, traverse velocities of 60 mph and a differential pressure drop of 3 psi in 3 seconds with no loss of function. The exception to this includes areas without safety related equipment or redundant equipment as discussed in FSAR Section 16.2.2.

All Class I buildings and structures were also designed to withstand various postulated tornado-generated missiles, including the following:

Horizontal Missiles

1) 4" x 12" x 12' plank at 300 mph
2) 4000 lb. passenger car at 50 mph less than 25 ft. above the ground.

Vertical Missiles

1) 4" x 12" x 12' plank at 90 mph
2) 4000 lb passenger car at 17 mph less than 25 ft. above the ground.

16.2.2 Tornado-Proof Systems and Equipment

Systems and Equipment Protected by Enclosure

All of the equipment which must be protected from tornados and tornado-generated missiles is contained within structures designed to withstand such loadings. The equipment or systems located within these structures include the following:

Primary Auxiliary Building

1) Safety Injection Pumps

2) Residual Heat Removal Pumps

3) Component Cooling Systems except portions of the piping loop in the Fuel Storage Building (Component Cooling Water operation is assured by the ability to provide make-up from the Primary Water Storage Tank).

4) Waste Disposal System (except for Waste Holdup Tank in Waste Holdup Tank Pit and Reactor Coolant Drain Tank and Pumps in the Containment)

5) Chemical and Volume Control System (except for Excess Letdown and Regenerative Heat Exchangers inside the Containment and Holdup Tanks in the Waste Holdup Tank Pit)

6) Refueling Water Purification Pump

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7) Sampling Systems

8) Auxiliary Building Ventilation System (ducts and supply fans only)

9) Containment Spray Pumps

10) Spray Additive Tanks

11) Pressurization Air Receivers

12) Electrical Tunnels

13) Waste Hold-up Tank Pit

Control Building

1) Instrumentation Readouts and Controls

2) Control Room Ventilation System

3) Batteries and Battery Chargers

4) Instrumentation Air System

5) Additional CCR HVAC Cooling Condenser Units (restrained to the Control Building roof to prevent them from becoming missiles but are not tornado missile protected)

Containment

1) Reactor Vessel, Core, Instrumentation, and Controls

2) Primary Coolant System (including Pressurizer and Pressurizer Relief Tank)

3) Steam Generators

4) Residual Heat Removal Heat Exchangers

5) Reactor Coolant Drain Tank and Pumps

6) Excess Letdown and Regenerative Heat Exchangers

7) Accumulators

8) Recirculation Pumps

9) Containment Air Recirculation Cooling and Filtration System

Diesel Generator Building

Auxiliary Feedwater System Building

The service water pump motors are protected by the service water enclosure, which is surrounded by the Intake Structure Enclosure (ISE) Building. The service water enclosure is

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designed as seismic Class I structure. The sidings and roofings of the ISE are postulated to be airborne during a tornado but will be prevented from coming in contact with the service water pump motors by the service water enclosure.

The potential for damage to spent fuel assemblies stored in the fuel pool from either turbine-generated or tornado-generated missiles is very low. See Appendix 14A for the worst case assumptions of offsite exposures due to turbine missile damaged fuel assemblies. See WCAP-7572, "Effect of Tornado Missiles on Stored Spent Fuel" for analysis of offsite exposures due to tornado missile damaged fuel assemblies. In both cases, the resultant site boundary doses are well below the 10 CFR 100 guidelines.

Service Water Pipe Chase

The two redundant service water supply lines crossing the Discharge Canal are protected by the concrete pipe chase from tornado effects. A postulated tornado generated missile can collapse the 8" concrete slab (at the top of the pipe chase) locally and hit the upper supply line. The pipe is capable to withstand the impact of the missile and the fallen concrete. Pipe stress is still below the allowable stress limit permitted by code.

Systems and Equipment Protected by Redundancy

All components and equipment for safe shutdown and isolation of the reactor are housed within the tornado-proof structures described above, with the following exceptions. For these components and systems, adequate tornado protection is provided by redundancy:

1) Redundancy is provided for the vital 480 volt system by three independent systems. Onsite there are three emergency diesel generators which are redundant and tornado protected; offsite there is a 138 kV above-ground system and a 13.8 kV under-ground system.

2) The emergency feed requirements of the steam generators are assured by tornado protected pumps and redundant water supplies.

3) The water requirements of the primary system are assured by the availability of primary water storage tank, the refueling water tank and the boric acid tanks.

4) Service water supply is assured by redundancy of two supply lines, four screens and six pumps of which only two pumps, one screen and one supply line are required for prolonged shut-down. The intake structure itself is tornado proof. The Backup Service Water System is an additional source of service water independent of the intake structure. The redundant service water supply lines are either buried underground with a minimum of 2'-10" cover or are protected by a minimum of two feed of concrete or a 8 inch thick slab for their entire run. The minimum distance between the headers is one foot. This protection is sufficient for the missiles considered.

Design Procedures

Specific design procedures employed to evaluate the capability for the reinforced concrete structures to withstand tornado loadings were as follows:

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1) The tornado loads were investigated considering overall structural effects. Overturning moments, base shears and toe pressure were checked considering the wind load, missile load, dead load and live load. The tornado loads were investigated considering local structural effects. Concrete and rebar stresses were checked considering wind loads, missile loads, dead loads and live loads.

2) For Items (1) and (2) above, the external wind loads, 3 psi negative pressure, and missile loads were considered in combinations yielding the most conservative load combination and thus the highest stress condition. Only one missile was considered acting at any time simultaneously with the wind loadings.

3) Missile penetrations into the reinforced concrete structure and corresponding loads on the structure were calculated by the following general procedure:

   a) Calculate depth of penetration of the missile using the modified Petry Formula (1).

   b) Calculate the impulsive force considering conservation of energy. The depth of penetration of the missile and the deflection of the structure are considered in calculated the impulsive force.

   c) Calculate the equivalent static force by multiplying the impulsive force by the dynamic load factor considering a rectangular load pulse acting for the duration of missile impact.

   d) Design the structure to resist the equivalent static force using recommended stress indices (2). The tornado protection structures, which are constructed of reinforced concrete, were designed to prevent missile penetration and spalling (by selection of moderate degree of damage allowable stress indices for structural design in accordance with Reference 1) of concrete from the walls, roof slab or dome impacted by the missile. Therefore, secondary missiles are not created which could damage or make inoperable Class I systems which must be protected from tornados.

For a more detailed description of the containment structure tornado analysis refer to Sections 2.2 and 2.4 and to the Containment Design Report in Appendix 5A.

Equipment and systems contained within tornado proof structures are protected from tornados and tornado missiles. Components and systems not housed within tornado-proof structures (but essential for safe shutdown and isolation of the reactor) are provided with protection to that function by component or system redundancy. The prior subheading, “Systems and Equipment Protected by Redundancy” discusses this.

Typical objects that could be postulated as potential tornado missiles were selected. These typical objects were approximated by the shape of simple objects like straight cylinders and slabs.

Assuming 300 mph tornado, an analysis was performed using the modified shapes. The results indicated which objects could be sustained or moved by the winds. Based on the above, the missiles for which plant protection was required were selected. These missiles are listed in Section 16.2.1.

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16.2.3 Tornado Design Criteria

Tornado wind loads are converted to equivalent static structural loadings in accordance with the applicable portions of the wind design methods described in ASCE Paper No. 3269 "Wind Forces on Structures." The provisions for gust factors and variation of wind velocity with height do not apply. The following factored load equation is used for those structures designed to resist tornado wind effects:

$C = (1 \pm 0.05)D + 1.0W'$ For containment structure

$C = 1.0D + 1.0W'$ For all other seismic Class I Buildings and Structures

where:

C = Required load capacity of section.

D = Dead load of the structure plus any normal operating live loads.

W' = Tornado wind load to include pressure drop effect where applicable.

The stress criteria used for this load criterion were for no gross yield of the primary structure with the yield stress levels reduced by the capacity reduction factors as defined in Chapter 5.

Three general criteria were adopted for the design of Indian Point 3 in tornado conditions:

I. A tornado will not cause a Loss-of-Coolant Accident.

II. A tornado will not impair the ability to safety shut the plant down.

III. A tornado, following Loss-of-Coolant Accident, will not impair the long term safety of the plant.

Criterion I

The Reactor Coolant System is contained entirely within the confines of the containment vessel. For the tornado to cause a Loss-of-Coolant Accident the tornado or tornado-produced missiles must penetrate the containment vessel. The design is such that penetration of the containment vessel is not credible.

Criterion II

There are two phases of reactor shutdown that must be considered; a shutdown to hot shutdown condition and a shutdown to cold condition.

Shutdown to Hot Shutdown Condition

The Reactor requires a number of basic services when held for an extended period in the hot standby condition:

a) Residual Heat Removal

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b) Reactivity Control, i.e., as fission poisons decay

c) Pressurizer Pressure and Level Control

d) Auxiliary Building and Control Room Ventilation

e) Electrical Systems

These services require that a number of systems and equipment will continue to operate following a tornado:

a) Residual Heat Removal

Following a normal plant shutdown an automatic steam dump control system bypasses steam to the condenser and maintains the reactor coolant temperature at its no load value. This implies the continued operation of the steam dump system, condensate circuit, condenser cooling water, feedwater pumps and steam generator instrumentation. Failure to maintain water supply to the steam generators would result in steam generator dry out after some 34 minutes and loss of the secondary system for decay heat removal.

Redundancy and full protection where necessary is built into the system to ensure the continued operation of the steam generator units. If the automatic steam dump control system is not available independently controlled relief valves for each steam generator maintain the steam pressure. These relief valves are further backed up by code safety valves for each steam generator. Numerous calculations, verified by startup tests have shown that with the steam generator safety valves operating alone the Reactor Coolant System maintains itself close to the nominal no load condition. The steam relief facility is adequately protected by redundancy and local protection. For decay heat removal, it is only necessary to maintain the control on one steam generator.

For the continued use of the steam generators for decay heat removal, it is necessary to provide a source of water, a means of delivering that water and, finally, instrumentation for pressure and level indication.

The normal source of water supply is the secondary feed circuit; this implies satisfactory operation of the condenser, air ejector, condenser cooling circuit, etc. In addition to the normal feed circuit the plant may fall back on:

1) The condensate storage tanks
2) The city water storage tank
3) The city water supply

Feedwater may be supplied to the steam generators by either the electrical feedwater pumps or by the steam driven feedwater pump; these pumps and associated valves may be controlled both locally and remotely from the Control Room. In the event of loss of compressed air, local operation would be adopted.

For continued operation of the electrical feedwater pumps, the 480 volt system must be assured. This is discussed under item (e).

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In addition, the diesel generators require the continued supply of fuel oil and service water; adequate redundancy and protection exist for this purpose.

Vital instruments and controls are provided both locally and in the Control Room.

b) Reactivity Control

Following a normal plant shutdown to hot shutdown condition, soluble poison is added to the primary system to maintain subcriticality.

For boron addition the Chemical and Volume Control System is used; control may be local or from the Control Room. Routine boration requires the use of:

Charging pumps and volume control tank with associate piping. Boric Acid transfer pumps and tanks and associated piping. Letdown station. Non-regenerative heat exchanger and associated equipment. Component Cooling and Service Water Systems. Periodic operation of one reactor coolant pump for pressurizer homogenization; the auxiliary spray/heaters could be used if necessary. Compressed air for valve operation – manual could be adopted if necessary.

The vital items of this equipment are housed within the containment and the reinforced concrete auxiliary building. The Service Water System is protected by means of redundancy. In order to guarantee the operation of the system the 480 volt system must again be assured.

It is worthy of note that with the reactor held at hot shutdown conditions, boration of the plant is not required immediately after shutdown. The xenon transient does not decay to the equilibrium level until at least 9 hours after shutdown and a further period would elapse before the reactivity shutdown margin provided by the full length control rods have been cancelled. This delay would provide useful time for emergency measures although the essential systems are considered to be adequately protected within the auxiliary building and Containment Building. For loss of CCW due to a missile strike in the Fuel Storage Building, city water is available for hook-up (IPN-02-040).

c) Pressurizer Pressure Level Control

Following a reactor trip, the primary coolant temperature will automatically reduce to the no load temperature condition as dictated by the steam generator conditions. This reduction in the primary water temperature reduces the primary water volume and if continued pressure control is to be maintained primary water makeup is required. The pressurizer pressure level is controlled in normal circumstances by the Chemical and Volume Control System. This requirement implies the charging pump duty referred to for boration plus a guaranteed borated water supply. The facility for boration is safety protected within the Primary Auxiliary Building; it is only necessary to supply water for makeup. Water may readily be obtained from separate sources: that in the volume control tank, boric acid tanks, monitor tanks, primary storage tank, and refueling water storage tank.

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Similarly to the two previous service requirements, the 480 volt system must be assured with the additional electrical load of the pressurizer heaters. Vital instruments and controls are provided both locally and in the Control Room.

d) Ventilation

The most essential ventilation requirements apply to the containment since in order to guarantee the satisfactory operation of the instrumentation and control systems the containment air temperature must be controlled to a tolerable level. This system again requires the satisfactory operation of the Service Water and Electrical Systems.

e) Electrical Systems

Protection from tornado is provided for the 480 volt switchgear and supply redundancy is provided by the diesel generators, gas turbine generator, the two above-ground incoming lines and the one below ground incoming line. The 6.9kV is fed by either the gas turbine generator or by an underground 13.8 kV feeder from the Buchanan substation. The Buchanan substation consists of four buses.

Shutdown to Cold Condition

Plant cooldown is not an immediate requirement following major damage due to a tornado. For a cooldown, the basic services required are:

a) Residual Heat Removal
b) Reactivity Control
c) Pressurizer Pressure Level Control
d) Ventilation
e) Electrical Systems

A cooldown would not be attempted until full equipment facilities had been guaranteed.

Tornado missile damage to a small bore pipe in the Containment Cooling Loop in the Fuel Storage Building (FSB) would require isolation and repair or isolation of piping. Prior to establishing Residual Heat Removal during plant cooldown the CCW System would have to be refilled using operator action. The Primary Water Storage Tank is available to replace lost water inventory.

Criterion III

Following a Loss-of-Coolant Accident the residual heat is removed through internal recirculation conditions with the facility for external recirculation if required. The duty implies the continued operation of the Auxiliary Feedwater System together with the associated electrical and service water supplies. The recirculation systems are protected by the tornado proof containment and auxiliary buildings. The Electrical and Service Water Systems are assured by redundancy as previously discussed.

References:

(1) "Design of Protective Structures" by Arsham Amirikian, Navy Docks P-51, Bureau of Yards and Docks Department of the Navy, Washington, D.C., August 1950.

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(2) TM5-855-1, Department of the Army Technical Manual, “Fundamentals of Protective Design (Non-Nuclear),” 1965.

16.3 DEMONSTRATION OF ADEQUACY OF SELECTED SEISMIC CLASS I ITEMS

16.3.1 Design of Seismic Class I Structures

A multi degree-of-freedom modal analysis was performed on all Class I building structures for Indian Point 3. The results indicate that all except the containment structure are rigid.

16.3.2 Analysis of Seismic Class I Equipment Other Than Reactor Coolant Pressure Boundary*

The ability of Class I equipment including heat exchangers, pumps, tanks, valves, motors, and electrical equipment components to withstand seismic loads was verified using one of the following methods:

(1) Equipment which is rigid and rigidly attached to its support structure was analyzed for a ‘g’ loading equal to the peak acceleration of the supporting structure at the appropriate elevation.

(2) Equipment which is not rigid and therefore potential for response to the support motion exists, was analyzed for the peak of the floor response curve for appropriate damping values.

(3) In some instances non-rigid equipment was analyzed using a multi-degree of freedom modal analysis. All contributing modes are considered. In addition, it should be pointed out that a sufficient number of masses is included in the mathematical models to insure that coupling effects of members within the component are properly considered. The results of these analyses indicate that the models contain more masses than necessary, and that future analyses of comparable equipment could be considerably simplified by considering fewer masses. The method of dynamic analysis uses a proprietary computer code called WESTDYN. This code uses as input, inertia values, member sectional properties, elastic characteristics, support and restraint data characteristics, and appropriate seismic response spectrum. Both horizontal and vertical components of the seismic response spectrum are applied simultaneously. The modal participation factors are combined with the mode shapes and the appropriate seismic response spectra acceleration to give the structural response for each mode. The internal forces and moments are computed for each mode from which the modal stresses are determined. The stresses are then summed using the square root of the sum of squares method.

(4) Type testing of selected electrical equipment has been conducted to demonstrate seismic design adequacy as described in WCAP-7817 and Section 16.3.3.

*NOTE: The analysis of the Reactor Coolant System is discussed in Appendix 4B.

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For the analysis of equipment to resist the vertical seismic component, 2/3 of the horizontal response spectrum curves were used to determine the acceleration appropriate to the vertical frequency.

Engineered Safeguards tanks, e.g., Boric Acid, Accumulator Spray Additive and Surge, were analyzed using method (3) above, for combined horizontal and vertical seismic excitation occurring simultaneously, and in conjunction with normal loads. Hydrodynamic analyses of these tanks were performed using the methods described in Chapter 6 of the U.S. Atomic Energy Commission –TID 7024.

Heat exchangers associated with the Engineered Safeguards Systems, e.g., Component Cooling and Residual Heat Removal, were analyzed using method (3) above, and the results show that stresses and deflections are within allowable limits.

Selected critical Engineered Safeguards valves are analyzed using method (3) above and the results indicate that their fundamental natural frequency is sufficiently separated from the building frequency that they will see little or no amplification of building motion. The results further indicated that the total stresses, considering all modes, is far below the allowable stress limits.

Damping values used for each item of equipment are in conformity with Table 16.1-1.

Non-linearities such as gaps, frictional forces, joint slippage, etc., were not considered explicitly in the model. It was felt that these non-linearities would tend to detune the system, hence act as if to increase the percentage of critical damping thus decreasing the response.

Appendages, such as motors attached to motor operated valves, were included in the mathematical models.

16.3.3 Seismic Testing of Instrumentation and Control Equipment

Mathematical models were not used for seismic design of instrumentation. Ability to withstand the seismic condition was determined by actual vibration type testing of typical instrumentation equipment under simulated seismic accelerations to demonstrate its ability to perform its functions. The seismic testing was reported in Westinghouse reports WCAP-7817, titled "Seismic Testing of Electrical and Control Equipment," by E.L. Vogeding, dated December 1971. The following is a summary:

> In a nuclear power plant, electrical and control equipment that initiates reactor trips, actuates safeguards systems and/or monitors radioactive releases from the plant must be capable of performing their functions during and after an earthquake that has occurred at the plant site. To demonstrate the ability of this equipment to perform under earthquake conditions, selected types of this essential equipment representative of all protection and safeguard circuits and equipment were subjected to vibration tests which simulated the seismic conditions for the "low seismic" class of plants.* During the tests, equipment operation was monitored to prove proper performance of functions. The results show that there were no electrical malfunctions. Based on these results, it is concluded that the equipment will perform their design functions during as well as following a "low seismic" earthquake.

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*NOTE: Those having Design Basis Earthquake horizontal acceleration less than or equal to 0.2g.

The low seismic test envelope is given in WCAP-7817 is appropriate for the locations of this protection and safeguards control and electrical equipment in Indian Point 3. The test curve developed for Indian Point 2 is conservative when applied to Indian Point 3 since the most adverse location, seismically, in Indian Point 2 is steel framed and relatively flexible, while that for Indian Point 3 is of reinforced concrete and therefore relatively rigid.

A typical path taken by a safeguards actuation signal is traced below to show that it is generated, transmitted and conditioned by the through equipment whose seismic adequacy has been demonstrated by test or analysis. A similar exercise may be carried out for reactor protection system signals.

A safeguards signal may be initiated by an instrument or transmitter which has the ability to withstand seismic forces as demonstrated in WCAP-7817, Sec. 4.8. This signal is carried in conduit and cable trays whose supports have been studied for resistance to seismic forces. The signal passes to the process control racks proven as described in WCAP-7817, Sec. 4.2. The signal is sent next to the safeguards actuation racks proven as described in WCAP-7817, Sec. 4.3. The actuation signal proceeds to the appropriate switchgear or active type controller.

The control board is not a Class I component. Typical switches and indicators for safeguards components were tested to determine their ability to withstand seismic forces without malfunction which would defeat automatic operation of the required component. Experience on previous control boards indicated that during shipment, "g" forces considerably greater than those required by the design basis earthquake are applied to the board and no failures of board mounted devices for engineered safeguard circuits had occurred. Past experience also indicated that the amplification due to the board structure can be measured during shipment. WNES instrumented the control boards during shipment to determine this amplification factor. Verification of no loss of function due to switches and indicators in the engineered safeguards circuits was completed by showing that the amplified "g" forces imposed on the devices were considerably less than the devices have shown to be able to withstand testing.

The safeguards circuits employ Westinghouse Series W motor control centers, and type DS circuit breakers and associated metal-enclosed or metal-clad switchgear. Review of these switchgear for proof of adequacy of the seismic resistant design determined that these motor control centers mounted in the metal enclosures, have been shock tested and proven to remain fully operable for shocks of at least 3g in any direction. Proof of resistance of the DS metal-clad switchgear to a seismic response spectrum established to "low seismic" plants have been demonstrated by vibration testing.

The switchgear supplies the power to operate the safeguards equipment completing the actuation train. The seismic design of this equipment is described in Section 16.3.2. The DC power supply may be considered as a branch to this main train of actuation. The source of DC power is the station batteries. The batteries and battery racks present a simple structural problem which was analyzed and found adequate for the forces imparted by the floor upon which they are located. Specially designed styrofoam spacers are installed in the intercell groups to provide additional seismic damping for the cell group. The conduit and cable trays carrying the DC power to the main station train received the same study for seismic support as described above.

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16.3.4 Ability of Service Water Lines to Accept Seismic Ground Displacement

The service water lines consist of two 24" diameter carbon steel pipes. They run in a common trench which is backfilled. Assuming that the ends of a pipe are free to displace vertically but not rotate and that the maximum permissible stress is restricted to 30,000 psi, a parametric study concluded that the following maximum allowable relative displacements may occur during a seismic disturbance without overstressing the pipe:

Length (ft.)	1	10	25	50	75	100
Displacement (inches)	0.002	0.20	1.25	5.01	11.25	20.04

This parametric study consisted of investigating the maximum allowable relative displacements of the ends of the buried 24" service water piping for all lengths of straight pipe segments. The length of pipe was varied as a parameter to ascertain the magnitude of displacement required to stress the pipe to 30,000 psi. The corresponding displacements were then reviewed to determine whether it was feasible that the underlying bedrock could sustain such motion without catastrophic consequence. It was concluded that the displacements required to stress the service water pipe to 30,000 psi were in excess of that which could be reasonably imposed by the bedrock. Pipes entering the containment and other structures are effectively anchored at the points of penetration. When piping was routed from one building to another with restraint at or near entry points, differences in seismic responses between the two buildings were accommodated in the following manner: The floor response curves for the two entry points were overlaid and the envelope of both curves was used as input to the dynamic analysis of the entire piping run between the two buildings.

When a piping system was routed from one building to another, piping and supports arrangements were made in such a way that the relative movement between supports was accommodated by the flexibility of the pipe.

To evaluate the stresses imposed by relative motion, the initial stress analysis utilized the absolute sum of supports displacements found from seismic analysis of structures; using a static approach, the stress was calculated. The resulting stress was combined with other secondary stresses. The total stress was evaluated against B31.1 code allowable stress.

It was concluded that the service water lines can withstand, without being overstressed, relative bedrock displacements associated with the earthquakes defined for the Indian Point site. The Service Water System piping was reanalyzed in the seismic piping reanalysis effort described in Section 16.3.5.

16.3.5 Analysis of Seismic Class I Piping

During the design phase of Indian Point 3, all seismic Class I piping 6 inches in diameter or larger (other than the reactor coolant loop piping and main steam and main feedwater piping inside containment) together with the two inch diameter high head safety injection lines were initially statically designed by UE&C using spacing tables. Subsequently, these lines were dynamically analyzed for seismic response to confirm the static design; all other Class I piping (less than six inches in diameter) was statically designed and analyzed also using spacing tables. During 1979 and 1980 a seismic reanalysis of safety related piping systems was performed. The two design approaches and the reanalysis program outlined below. As indicated earlier, Westinghouse was responsible for seismically analyzing the reactor coolant loop, main

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steam, and main feedwater piping inside containment. Westinghouse was also responsible for other aspects associated with the design of the reactor coolant loop.

The design placement of seismic restraints was predicted on the principle of containing the seismic stresses without restricting the free thermal expansion of the piping system. The systems were designed to have sufficient flexibility to prevent the movements from causing failure of piping or anchors from overstress.

Each of the seismic supports was verified to agree with the as-built location.

Relative displacement between anchor points was considered in the seismic analysis of the main steam lines for Indian Point 3. Analysis indicated that the stresses at the highest stressed point were affected by less than 10% when relative anchor displacements were considered.

Dynamic Analysis of Seismic Class I Piping During Design Phase

Class I piping systems, 6 inches in diameter and larger plus the 2 inch diameter high head safety injection lines were modeled and dynamic flexibility analysis performed. A detailed description of the method of analysis is given below.

The analysis was performed using the proprietary computer code ADLPIPE. The code used as input, system geometry, inertia values, member sectional properties, elastic characteristics, support and restraint data characteristics, and the appropriate Indian Point 3 seismic floor response spectrum for 0.5% critical damping. Both horizontal and vertical components of the seismic response spectrum were applied simultaneously.

With this input data, the overall stiffness matrix of the three dimensional piping system was generated (including translational and rotational stiffness). The modal participation factors were computed and combined with the mode shapes and the appropriate seismic response spectra to give the structural response for each mode.

Each piping run was modeled as a three dimensional system which consisted of straight segments, curved segments, and restraints. Straight segments were distinguished from the curved segments during data output.

The computer code required that the piping be represented by a discrete mass model. Each mass included the contribution of both the steel encasement and conveyed fluid. Where valves or other concreted masses existed in the piping system, these were included in the model.

Restraints were included in the model at their proper location. The directionality of the restraints was also considered of the restraints was also considered.

Some averaging of the response spectra was performed to smooth out the erratic response of the earthquake's random behavior. At the high frequency end of the spectra, the acceleration levels of the smoothed spectra converged to the values of the unsmoothed spectra.

The computer code ADLPIPE utilized an algebraic summation option for intramodal response combination and the square root of the sum of the squares option for intermodal response combinations. The algebraic summation method of combination was later considered unacceptable as it may have predicted nonconservative results in the piping reanalysis. This determination of the inadequacy of the computer code ADLPIPE gave rise to the piping

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reanalysis described later in this section which was performed in accordance with the guidelines provided in IE Bulletin No. 79-07 ("Seismic Stress Analysis of Safety-Related Piping").

The re-analysis was limited to address the concerns cited in the subject IE Bulletin and as such the original analysis criteria (e.g., system modeling) were maintained.

The reactor coolant loop, main steam, and main feedwater piping inside containment were originally analyzed by Westinghouse in a manner acceptable within the requirements of I.E. Bulletin 79-07, and as such the concerns of the subject IE Bulletin were not acceptable to these piping lines.

UE&C Static Analysis of Seismic Class I Piping During Design Phase

Class I piping and supports, other than those dynamically analyzed (i.e., piping less than six inches in diameter except the two inch high head safety injection lines), were analyzed for equivalent static load. With a ground acceleration of 0.15g horizontal and 0.10g vertical, the spectral accelerations corresponding to two times and 1.33 times the maximum point on the 0.5% critical damping amplified response curve was used to calculate an equivalent static force imparted to the pipe and its support points for the horizontal and vertical directions, respectively. The sum of the resulting additional stress plus the normal stresses was limited to 1.2 times the B31.1 code allowable stress for piping. The stresses in the pipe supports and hangers were likewise limited to 1.33 times the allowable stress in accordance with the American Institute of Steel Construction (AISC).

Seismic Reanalysis for Safety Related Piping Systems

As discussed above, the original UE&C confirmatory dynamic analyses for the Indian Point 3 safety related piping systems greater than or equal to six inches diameter plus the high head safety injection piping utilized the computer code ADLPIPE. As discussed in IE Bulletin No. 79-07, the algebraic summation method of combination for intramodal responses was judged unacceptable as it may predict nonconservative results. The following piping system or portions thereof were affected by the subject IE Bulletin and reanalyzed by UE&C:

1) Condensate System
   - Auxiliary feedwater pump suction from condensate storage tank.

2) Auxiliary Feedwater System
   - Turbine driven auxiliary feedwater pump discharge.

3) Service Water System

4) Reactor Coolant System
   - Connections to reactor coolant systems from second check valve
   - Pressurizer surge line
   - Pressurizer relief lines

5) Safety Injection System including
   - Containment spray system
   - Accumulator discharge lines
   - Refueling water

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- Residual heat removal loop
- Boron injection

6) Auxiliary Coolant System
   - Component cooling loop

7) Waste Disposal System
   - Recirculation fan cooling coil drains

Method of Reanalysis

The following method of reanalysis was submitted to and approved by the NRC staff for use in addressing the concerns cited in IE Bulletin No. 79-07. The seismic reanalysis was performed for the Operating Basis Earthquake (OBE) loading condition using the response spectrum analysis approach. The Amplified Response Spectra (ARS) associated with one horizontal (X) component and the vertical (Y) component of the seismic excitation were considered simultaneously. The analysis was repeated for the horizontal (Z) component and the vertical (Y) component. The reanalyses were performed with the UES&C –ADLPIPE- 2 computer code and a computer user option which use the square root of the sum of the squares for both intramodal and intermodal responses.

From these two cases, worst case values for the pipe seismic stresses, support loads and component nozzle loads were multiplied by a factor of 1.38 and then combined with other applicable loadings. The factor of 1.38 was found acceptable by NRC to reflect adequate conservatism in the calculations. Results from loading conditions other than seismic were not recalculated since they were not affected by IE Bulletin No. 79-07.

The factor 1.38, when used in combination with the computer user option, addressed the most conservative interpretation of the FSAR commitments regarding the intramodal response combination (i.e., this factor was utilized to account for the difference between absolute vs. SRSS summations). When a result calculated exceeded the applicable allowable limit, a reanalysis was performed using an equivalent analytical approach which included all three earthquake components and used the square root of the sum of the squares method for both the intermodal and intramodal responses without utilizing the factor 1.38. However, this latter approach was not employed as its use was not deemed necessary.

The results obtained from the OBE seismic reanalyses were multiplied by 1.5 to yield the Design Basis Earthquake (DBE) seismic condition values.

The safety-related lines 15 and 51 from the discharge of the containment spray pumps to the point where they penetrate the containment from Primary Auxiliary Building were further re-analyzed. The results of this re-analysis are presented in Reference 1.

As-Built Configuration of Safety Related Piping System

As part of the analytical effort required to conduct the seismic piping reanalysis program, an "As-Built" verification of those safety related piping systems subjected to the piping reanalysis was performed.

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The field verification program for normally accessible areas consisted of line walks of 194 static span table analyzed lines and 117 dynamically analyzed lines. The program was conducted in accordance with approved plant procedures.

Valve Weight Corrections

A program was conducted to collect information relevant to Velan valve weight data. The program identified the specific swing check valves incorporated into the Indian Point 2 facility, the original weight data used in the piping analysis, the system and line in which the valve is installed, the weight variations, and the present deviation in weight compared with the specific line weight between supports.

The valves range in size from 3" to 12" and are installed in either the Auxiliary Coolant System, the Chemical and Volume Control System or the Safety Injection System. The actual weights of the 18 valves in question were included in the seismic piping reanalysis calculations.

Concrete Expansion Anchor Bolts in Class I Systems

A review of pipe support base plates using concrete expansion anchor bolts demonstrated the existence of QC documentation verifying the compliance with anchor bolt design requirements.

To further verify and complement this documentation, several elements of the program, in addition to the field verification effort, addressed the various criteria and concerns for concrete anchor bolts, as follows:

a) A verification survey incorporating two hundred and fifty (250) base plates with seven hundred (700) anchors was carried out for over twenty (20) normally accessible lines. The survey and sampling effort verified that the engineering, design and installation requirements were carefully carried out.

b) A UT sampling effort for the determination of anchor belt imbedment incorporated more than one hundred seventy five (175) normally accessible (outside Containment) supports. This sampling verified that the design requirements and installation procedures for concrete anchor imbedment were followed.

c) A preload upgrade/test retorque effort subjected to evaluation approximately two thousand (2000) normally accessible supports and more than seven hundred and fifty (750) normally inaccessible supports. This effort insured that supports not addressed in the repair or modification efforts are properly preloaded.

d) On site torque/preload testing was conducted for Hilti Kwik-Bolt Concrete Expansion Anchors. This testing and surveillance verified the appropriate torque values for a corresponding preloading of these anchors.

e) A field inspection effort was run to determine the extent to which anchor bolts on safety related lines were installed in concrete block walls. Seismic Class I system or safety related system supports which utilize concrete anchors in block walls were modified to eliminate the anchor bolt installations in these concrete block walls.

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f) A hanger support repair effort resolved minor variances and problems identified during the line walk and inspection efforts. Approximately eight hundred (800) supports were repaired. Of these, some one hundred and fifty (150) are normally accessible.

g) To further insure proper preloading of concrete anchors, normally inaccessible supports involved in the modification effort had spring disc washers installed.

Results of Piping Reanalysis

All of the piping systems identified previously were reanalyzed for seismic loading using the method of analysis discussed above. The "As-Built" verification was performed in accordance with approved plant procedures. The results of the "As-Built" verification were incorporated in the reanalysis of the piping lines as well as the re-evaluation of valve weights, pipe supports, equipment nozzles and containment piping penetrations.

The results of the line reanalyses show that the total stresses, for both upset and emergency plant operating conditions, are within their respective applicable allowable limits.

Pipe supports, hangers, snubbers and pipe whip restraint components, including the base plate and anchor bolts, were re-evaluated for the new applied piping loads for both upset and emergency conditions. Those not found capable of performing their safety functions within their respective applicable allowable limits were modified as necessary.

Equipment nozzles and containment piping penetrations were reevaluated. The results confirmed that the new applied piping loads, both for the upset and emergency plant operating conditions, are within their respective applicable allowable limits.

16.3.6 Seismic Design of Spent Fuel Pool

Procedures outlined in Section 6.5 of TID-7024, "Nuclear Reactors and Earthquakes," were used for the seismic design of the spent fuel pool. The effects of water in the pool is accounted for in this design approach.

The Fuel Storage Building outside the pool was evaluated for seismic capability to establish that unacceptable damage to the CCW piping would not occur. The methods and criteria were submitted for review in letters IPN-01-034 and IPN-02-040 and established that no unacceptable damage occurs.

16.3.7 Seismic Design of Intake Structure

Procedures outlined in Section 6.5 of TID-7024, "Nuclear Reactors and Earthquakes," were used for the seismic design of the Intake Structure walls. The effect of water sloshing on the walls is accounted for in this design approach. The controlling factor in the design of the Intake Structure was the hydrostatic load, with the worst combination being one chamber empty and the adjacent chamber being filled with water.

References

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1) Letter from C.A. McNeill, Jr. (NYPA) to S.A. Varga (NRC) dated February 8, 1985 entitled "Revision of Results Previously Reported for IE Bulletin No. 79-07 (Seismic Stress Analysis of Safety-Related Piping) Line 51 of Problem 413."

16.4 DETAILS OF STRUCTURAL DESIGN

16.4.1 Design of Containment Interior Structures

The interior structure was designed as five separate main structural components. They are:

- 3' thick fill slab
- 3' thick crane wall
- 4' to 6' thick refueling canal
- 2' thick operating floor slab
- Primary Shield Wall

The method of design, stress analysis, critical stresses and locations were as follows:

3' Thick Fill Slab – The controlling loads on the 3' fill slab occur at the reactions from the primary equipment supports due to various postulated pipe breaks. The slab was designed as a series of radial beams running under the equipment supports and spanning between the reactor support wall and the crane wall. Stresses in reinforcing were limited to fy. Maximum stresses occur immediately below the primary equipment supports.

3' Thick Crane Wall – The crane wall is designed for a 7 psi differential pressure occurring immediately after a primary pipe break and prior to pressure equalization.

Although the stress level associated with this pressure differential were sufficiently low to establish that the concrete could resist the pressure loading, sufficient reinforcing was provided to resist all membrane forces without any contribution from the concrete. Stresses are limited to 0.9 fy. The membrane hoop stress was 13 ksi and the axial vertical rebar stress was 3.13 ksi.

A two dimensional Finite Element Analysis was performed to determine the area which would be affected by the Jet Force. The analysis indicated that in local areas (near the application of the force) some minor yielding of the crane wall rebar occurs. The yielding, which occurs only in the horizontal steel, is very local in nature. There is sufficient steel available in the vertical direction to accommodate any redistribution of load from the horizontal direction. In addition, redistribution will take place with the adjacent understressed facets. The load was assumed to act at the mid-height of the wall, thus causing maximum bending moment.

Further stability of the crane wall was demonstrated by determining the ultimate failure load by means of a yield line analysis. This analysis indicated that the structure has the capacity, through strain energy of structural response, to resist uniform Jet Force load of 2100 kips acting simultaneously with the 7 psi pressure differential without failure.

The containment internal concrete is essentially rigid (fundamental frequency ~17 cps), therefore, seismic loads were calculated using the Design Basis Earthquake maximum ground acceleration (0.15g).

The crane wall was considered as a cantilever beam and the base shear determined by the response spectrum approach. The base shear was distributed to the individual nodes by the formula:

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$F_x = W_x h_x V / \Sigma W_x h_x$

Where: V = Base Shear

W_x = Weight of node under consideration

h_x = Distance from base to section under consideration.

The moment at the base was determined and the uplift calculated by considering a circular ring of thickness equal to the area of steel per inch. This maximum uplift which occurs at the point at the base of the structure stresses the rebar to 1.1 ksi. This load is insignificant when compared with the Jet Force load, therefore, consideration of simultaneous blowdown and earthquake loads do not affect the conclusions above.

The crane wall was also designed to resist steam and feedwater pipe break reactions of 340 kips and 200 kips where supports are connected to the wall. This extra steel provided for pipe break loads is available, in the form of steel buttresses, to resist pressure, jet force and seismic loads; however, it was not considered in the analysis.

4' to 6' Thick Refueling Canal

The refueling canal was designed for the 7 psi pressure differential. The wall resists the pressure by spanning vertically between the refueling floor and the operating floor. Stresses were limited to 0.9 fy.

A Finite Element Analysis was also performed to check the effects of the Jet Force load. Some local yielding was indicated; however, the cross section is sufficient to provide stability since the moment capacity is slightly greater than that of the crane wall. A yield line analysis was performed and provided the basis for the above.

The seismic load was determined by the same procedure used for the crane wall. The average load in kips/ft was distributed over the wall and the vertical span was conservatively assumed to carry the entire load. The resulting bending movement produced a stress of approximately 3 ksi in the rebar. This had an insignificant effect on the conclusions concerning the Jet Force loads when blowdown and earthquake were considered simultaneously.

2' Thick Operating Floor Slab

Because of the many openings in the floor for equipment, the floor is designed as a series of beams. Principal loadings are (D.L. + 500 psf live load) and (7 psi pressure differential + D.L.). The first loading, (D.L. + 500 psf live load), was designed in accordance with Part IV-B of ACI 318. Stresses for the pressure differential case were limited to 0.9 fy.

The operating floor was investigated for Jet Force loads. There appears to be very little area of the operating floor which can be reached by the expanding jet of water from a break in the Reactor Coolant System. The jet is greatly dispersed in the distance between the primary coolant piping and the underside of the operating floor. The only area of the floor which can be struck by a jet spans between the areas of the floor heavily reinforced as beams. The span cross section consists of a T-beam with a 2'-0 thick floor acting as the flange and 7'-0 high

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biological shielding wall as the web. This section resists the jet force load within the 0.9 fy stress limit on the rebar.

Primary Shield Wall

The reactor pressure vessel is enclosed by a 6'-0" thick circular reinforced concrete shield wall which is designed to sustain the internal pressure and provide missile protection for the Containment and liner in the highly unlikely failure of the reactor vessel due to the longitudinal split. All stresses were maintained within 95 percent of specified minimum ultimate rebar tensile stress.

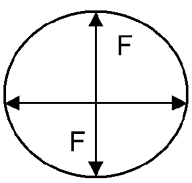
In the event of a circumferential reactor break, the ¼" base mat liner plate at the bottom of the Containment Reactor Cavity Pit, directly under the reactor vessel, is protected by 2'-0" of concrete with a 1" steel liner plate embedded in the top of the concrete. Below the containment base mat liner plate is 4 ½ feet of concrete poured on rock.

The cavity wall was designed to withstand the forces and internal pressurization associated with a longitudinal split without gross damage. The assumed accident condition was a longitudinal split of the cylindrical part of the reactor vessel (i.e., 24.4 feet long) having an average width of 1.9 foot. As a result of the assumed accident, the following two loading cases were considered in the analysis:

Load Condition I

Load on cavity walls at the instant of vessel rupture -

F= 650 kips/ft equivalent static line load at the instant of vessel rupture applied as shown in sketch based on a dynamic load factor of 2 applied to the subcooled pressure of 2250 psi times the average width of the break

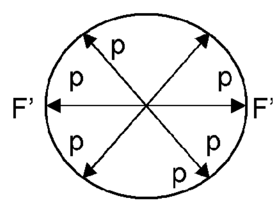


Load Condition 2

Load on cavity walls as shown in sketch -

F' = 188 kips/ft equivalent

P = 600 psi equivalent static pressure



The line load was based on saturated pressure of 1300 psi times the average width of the break and the pressure load was based on energy released and vent area available. The maximum stress level in the rebar under these loading conditions was limited to the 0.95 ultimate strength of the rebar. For Load Condition 1 and Load Condition 2, maximum rebar stresses assuming the concrete to be cracked were 63 ksi and 82.6 ksi, respectively. The rebar used is ASTM A 432 (Revised ASTM 615-63, grade 60) with specified yield of 60 ksi and ultimate tensile strength of 90 ksi.

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