NINE MILE POINT NUCLEAR STATION - UNIT 2

INDIVIDUAL PLANT EXAMINATION (IPE)

July 1992
Rev. 0

Prepared by:

Niagara Mohawk Power Corporation
301 Plainfield Road
Syracuse, NY 13212

R.F. Kirchner
Project Manager

Principle Contributors

NMPC

M.W. Cowden
J.A. Fischer
T.J. Gurdziel
G.W. Lapinsky
Dr. S.C. Lin
J.A. Snizek

Contractors

J.H. Moody -- Lead Consultant
Dr. A.N. Beare (General Physics)
Dr. E.T. Burns (ERIN)
T.J. Casey
J.R. Gabor (Gabor, Kenton & Associates)
T.P. Mairs
B. Malinovic (Fauske & Associates)
Dr. G.W. Parry (Haliburton NUS)
J.C. Raines (Fauske & Associates)
D.E. Vanover (Gabor, Kenton & Associates)
Dr. D.A. Wesley (ABB Impell)

Rev 0 (7/92)

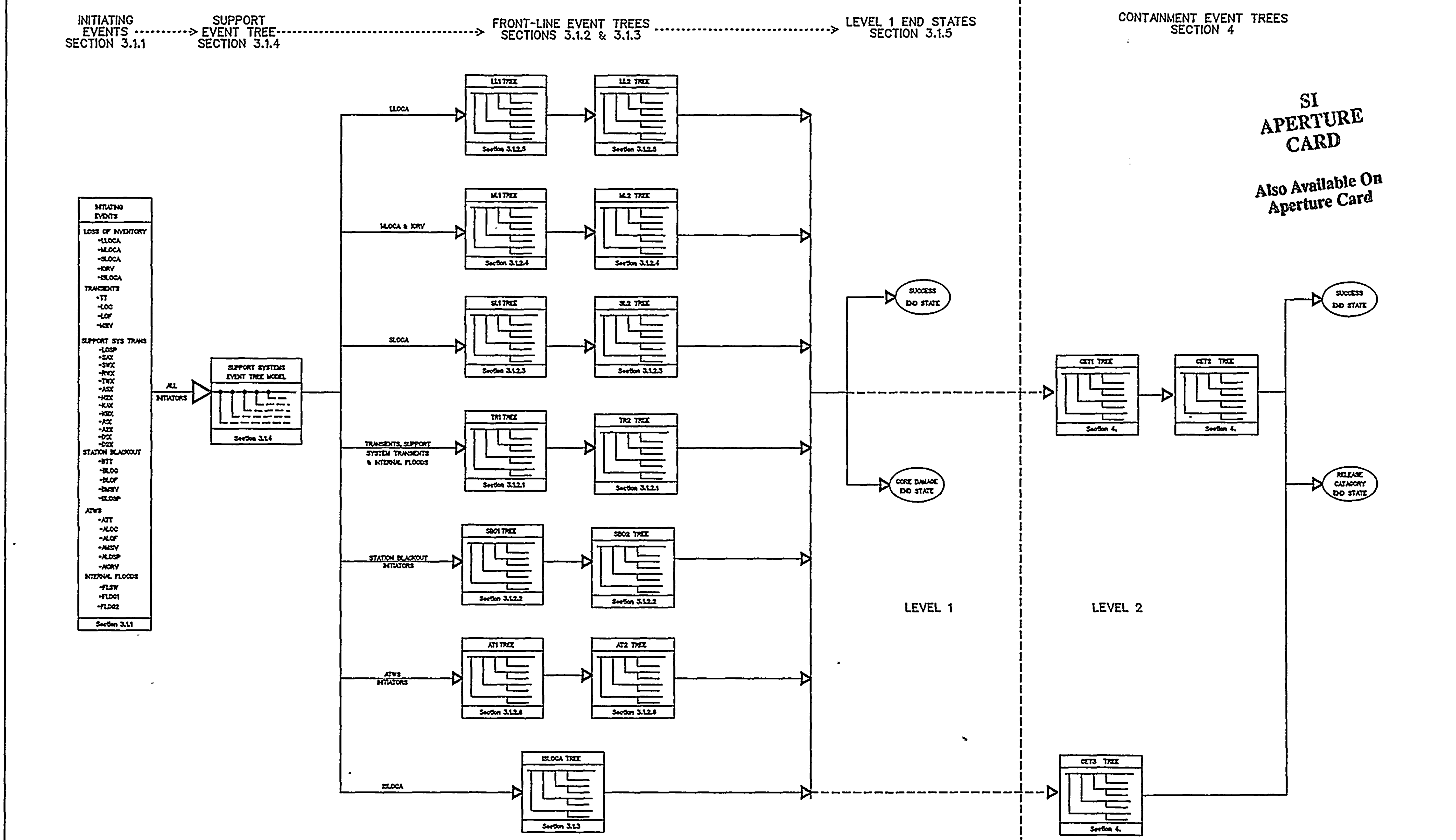


Figure 3.1.2-1
Overview of Event Sequence Model

I:\NWP2IPE\DGN\2F312.DGN Jul. 07, 1992 09:57:13

9208050191 - 01

Rev 0 (7/92)

SI
APERTURE
CARD

Also Available On
Aperture Card

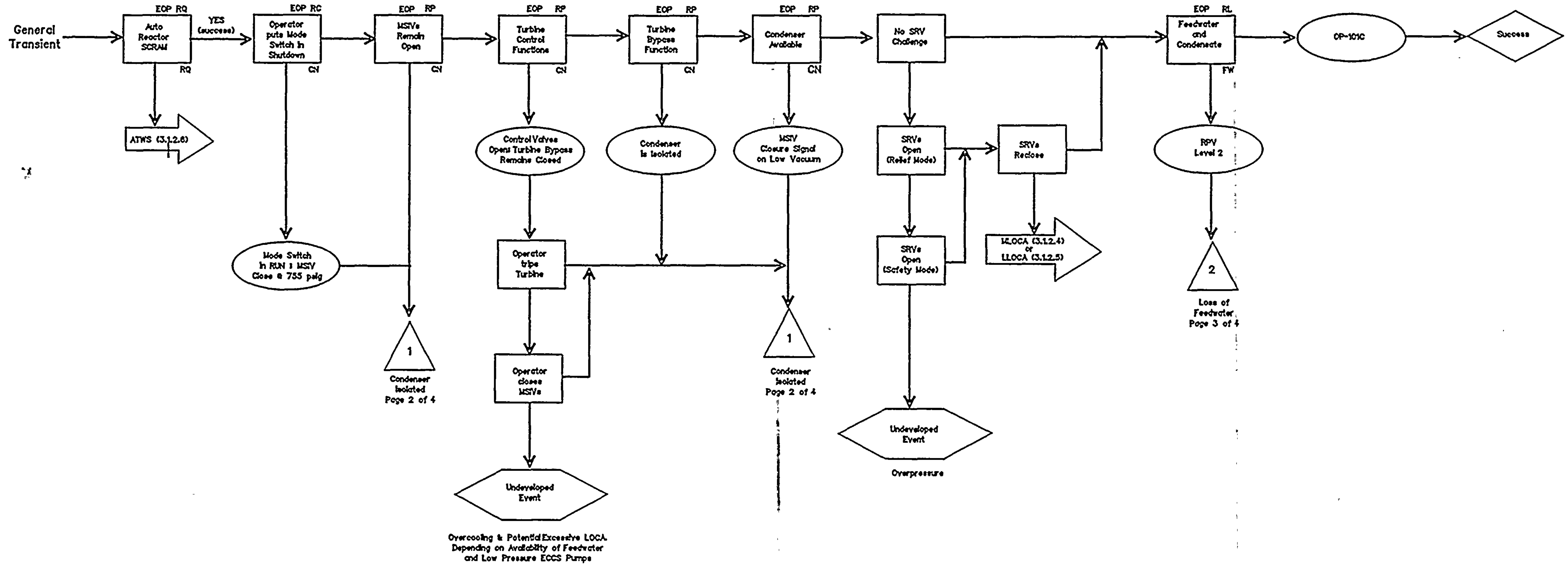


Figure 3.1.2.1-1 (sheet 1 of 4)
General Transients ESD

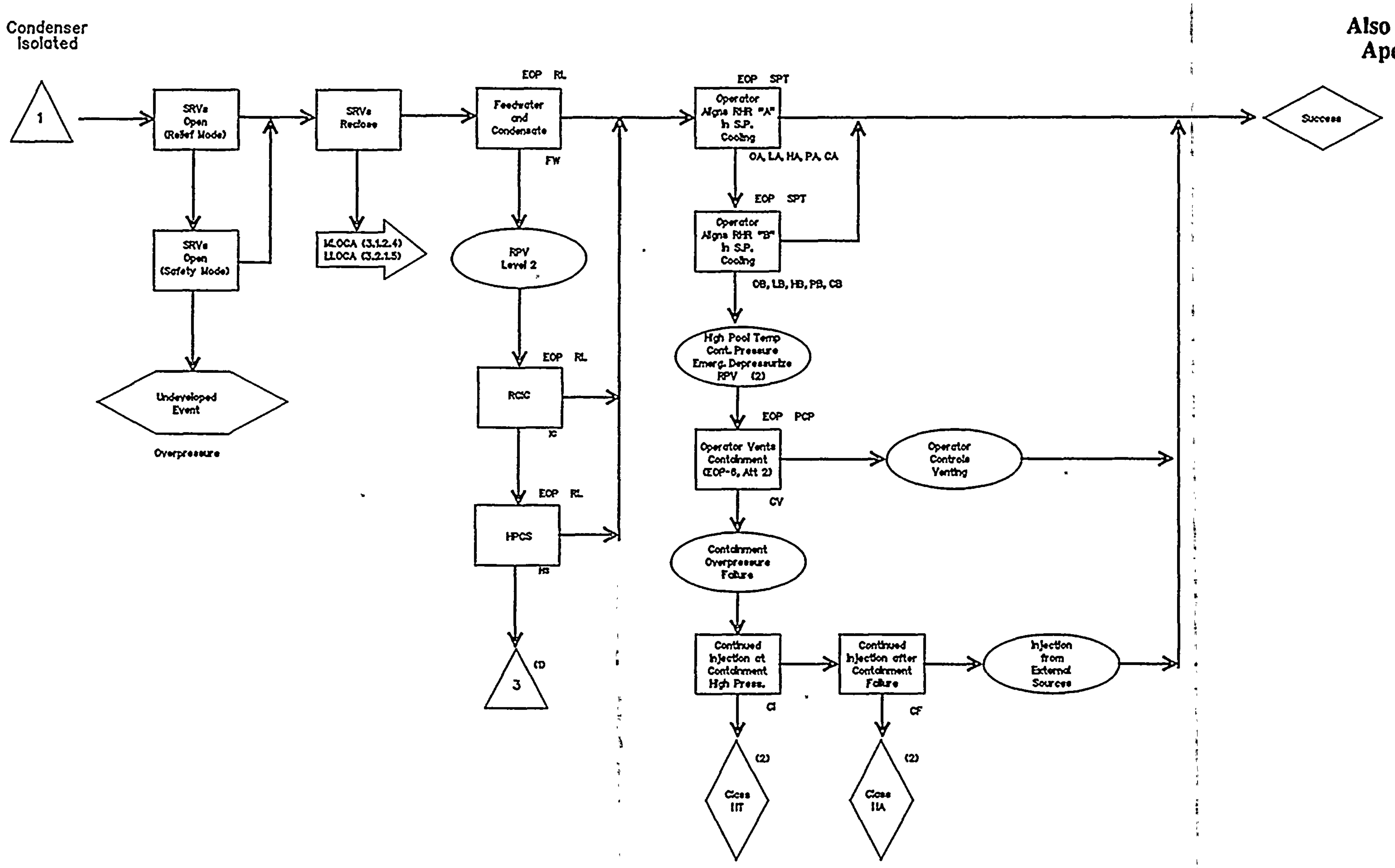
I:\NMP2IPE\DGN\2F3121.DGN Jul. 07, 1992 10:01:51

9208050191-02

Rev 0 (7/92)

SI APERTURE CARD

Also Available On Aperture Card



(1) CRD is not credited as adequate to prevent entry into depressurization conditions or to prevent core damage

(2) The following Heat Removal Capabilities are not modeled:

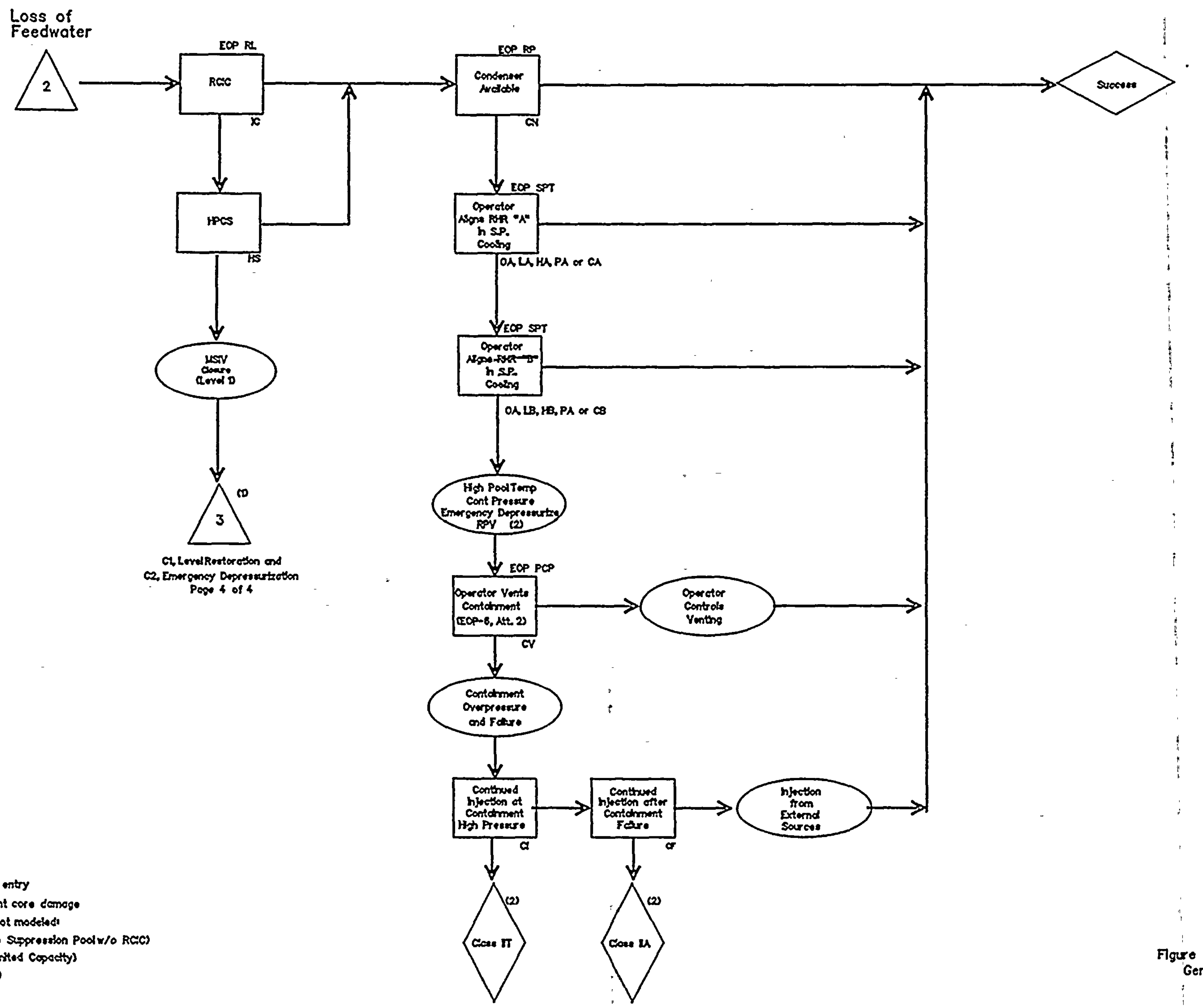
* RHR Steam Condensing Mode (Dumping to Suppression Pool w/o RCIC)
* RWCU Recirculation or Blowdown Mode (Limited Capacity)
* Main Steam Drain Lines (Limited Capacity)

Figure 3.1.2.1-1 (sheet 2 of 4)
General Transients ESD

I:\NMP2IPE\DGN\2F3121.DGN Jul. 07, 1992 10:03:10

9208050191 - 03

Rev 0 (7/92)



SI APERTURE CARD

Also Available On Aperture Card

(1) CRD is not credited as adequate to prevent entry into depressurization conditions or to prevent core damage

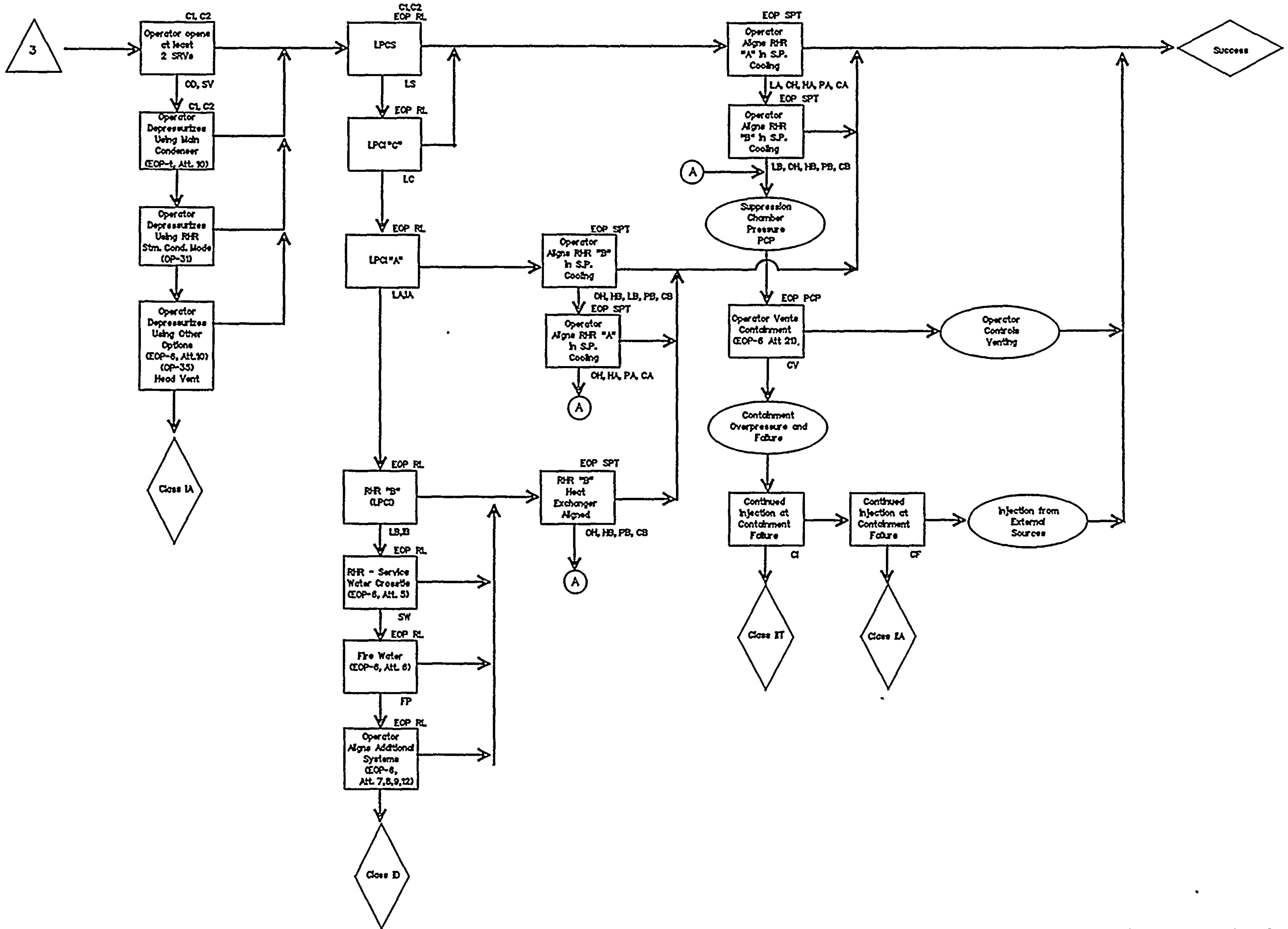
(2) The following Heat Removal Capabilities are not modeled:
- RHR Steam Condensing Mode (Dumping to Suppression Pool w/o RCIC)
- RWCU Recirculation or Blowdown Mode (Limited Capacity)
- Main Steam Drain Lines (Limited Capacity)

Figure 3.1.2.1-1 (sheet 3 of 4)
General Transients ESD

I:\NMP2IPE\DGN\2F3121.DGN Jul. 07, 1992 10:04:38

9208050191 - 04

Rev 0 (7/92)



SI APERTURE CARD

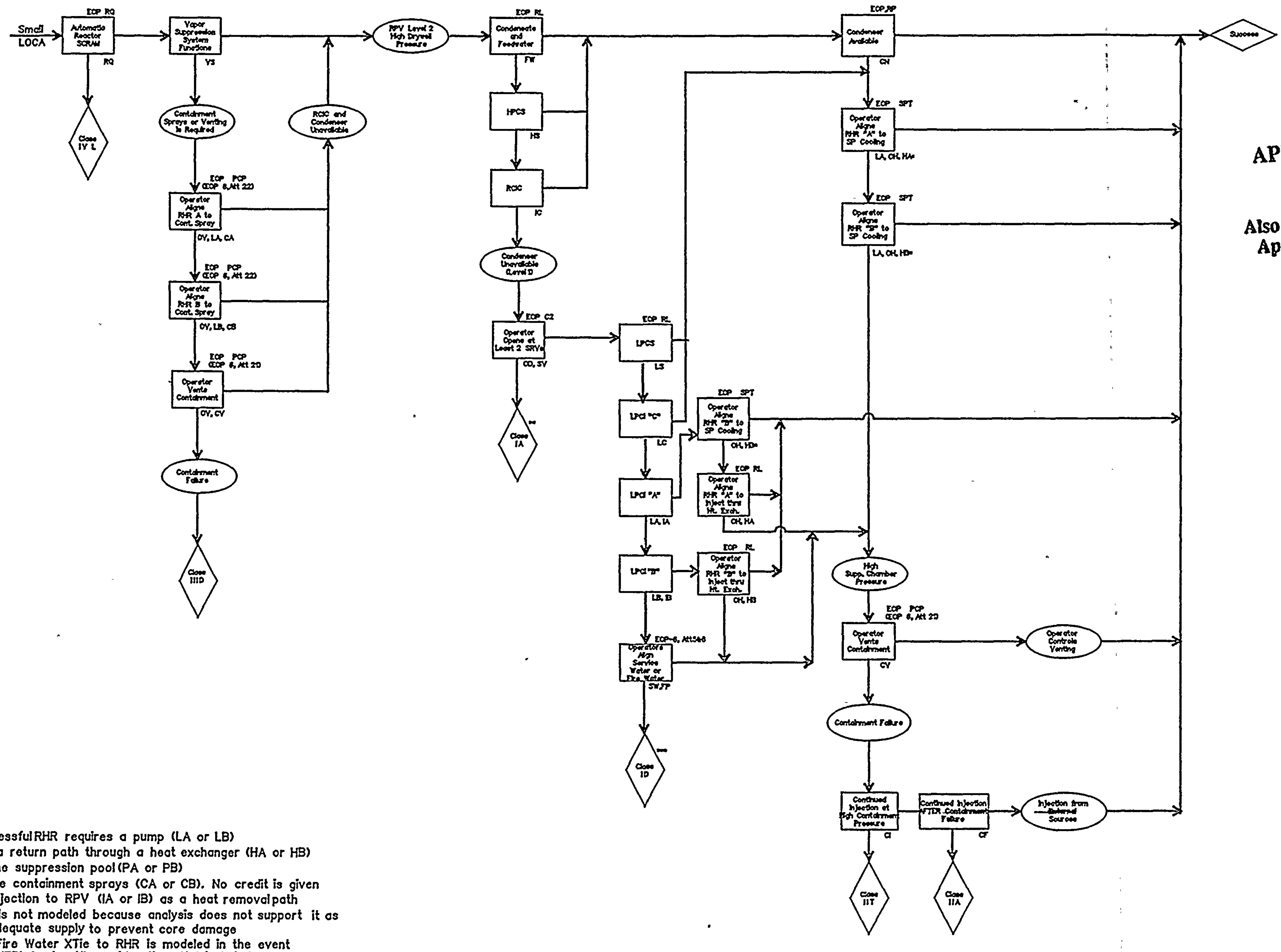
Also Available On Aperture Card

Figure 3.1.2.1-1 (sheet 4 of 4)
General Transients ESD

I:\NMP2IPE\DGN\2F3121.DGN Jul. 07, 1992 10:06:02

9208050191 - 05

Rev 0 (7/92)



* Successful RHR requires a pump (LA or LB) and a return path through a heat exchanger (HA or HB) to the suppression pool (PA or PB) or the containment sprays (CA or CB). No credit is given to injection to RPV (IA or IB) as a heat removal path

** CRD is not modeled because analysis does not support it as an adequate supply to prevent core damage

*** The Fire Water XTie to RHR is modeled in the event tree (FP) for level II consideration. Not included as success in Level I due to uncertainty of its capability.

Figure 3.1.2.3-1
Small LOCA ESD

SI APERTURE CARD

Also Available On Aperture Card

I: \NMP2IPE\DGN\2F3123.DGN Jul. 07, 1992 10: 09: 26

9208050191 - 06

Figure 3.1.2.1-2
Transient Event Tree TR1

MODEL Name: NMP2
Event Tree: TR1

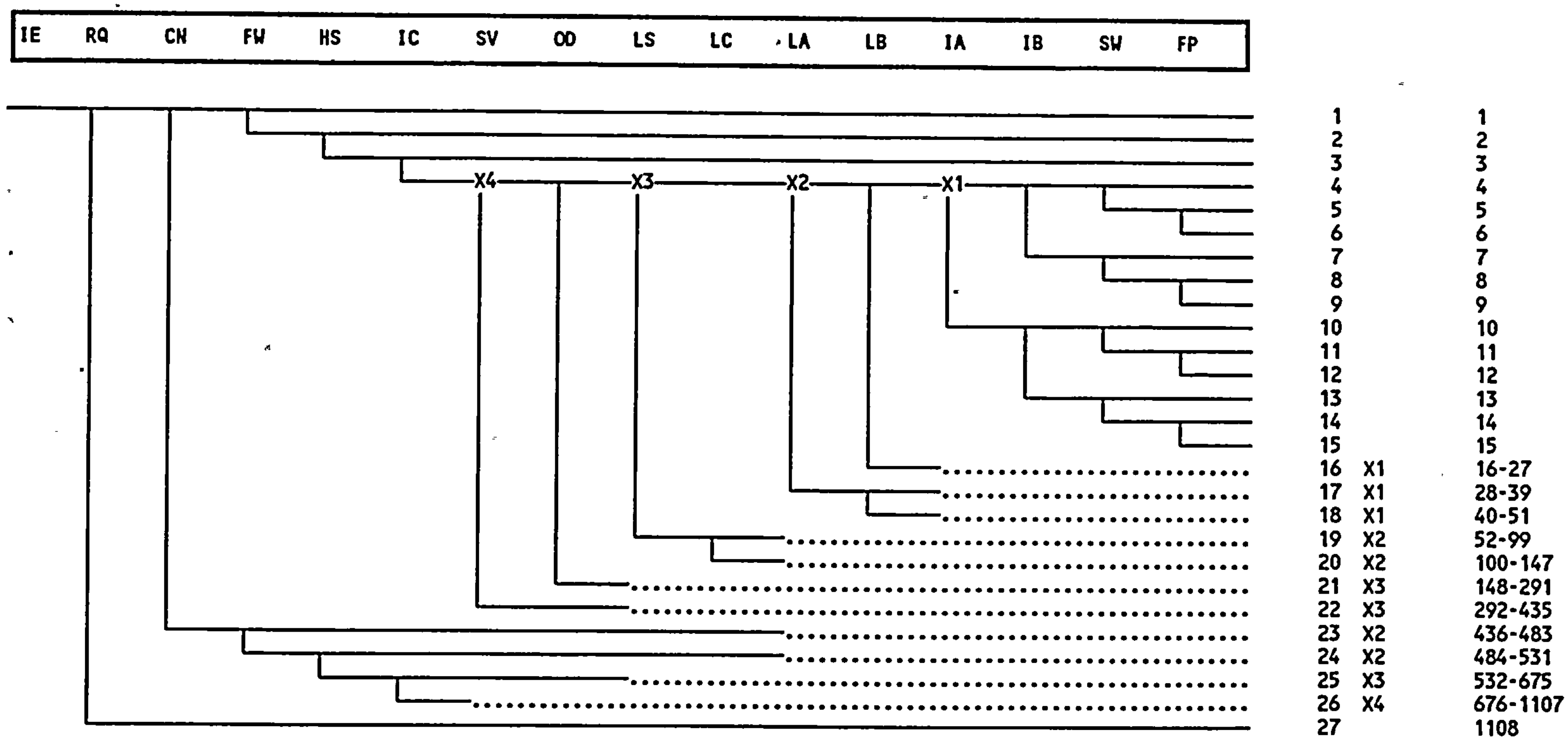


Figure 3.1.2.1-2
Transient Event Tree TR1

MODEL Name: NMP2
Top Event Legend for Tree: TR1

Top Event Designator	Top Event Description
IE	INITIATING EVENT
RQ	REACTOR SCRAM
CN	CONDENSER AVAILABLE
FW	CONDENSATE & FEEDWATER
HS	HPCS
IC	RCIC
SV	SRV/ADS VALVES OPERABLE
OD	OPERATOR DEPRESSURIZES RPV
LS	LPCS
LC	LPCI C
LA	RHR PUMP TRAIN A
LB	RHR PUMP TRAIN B
IA	RHR INJECTION A
IB	RHR INJECTION B
SW	SERVICE WATER CROSSTIE TO RHR
FP	FIRE WATER CROSSTIE TO RHR

Figure 3.1.2.1-2
Transient Event Tree TR1

MODEL Name: NMP2
Split Fraction Logic for Event Tree: TR1

SF	Split Fraction Logic
RQF	BLACK*(INIT=LOSP+INIT=TT+INIT=LOC+INIT=MSIV+INIT=LOF) Rule Comment THIS PREVENTS BLACKOUT SEQUENCES FROM ENTERING TR1
RQ1	1
CNF	NA=F*NB=F+TW=F+AS=F+N2=F + INIT=LOC
CN2	INIT=MSIV
CN1	1
FWF	NA=F*NB=F+TW=F+AS=F+TA=F+TB=F + INIT=LOF
FW1	1
HSF	SA=F*SB=F + TB=F Rule Comment NOTE THAT OPERATORS COULD MANUALLY USE KB
HS1	KA=S*SA=S*SB=S
HS3	KA=S*(SA=F+SB=F)
HS2	KA=F*SA=S*SB=S
HS4	KA=F*(SA=F+SB=F)
HSF	1
ICF	SA=F*SB=F + D1=F + TA=F + TB=F +E1=F*E2=F+UA=F+UB=F
IC1	E1=S*E2=S
IC2	E1=F + E2=F
ICF	1
SVF	D1=F*D2=F + ACA*ACB
SV1	-ACA*-ACB*D1=S*D2=S*N2=S
SV2	-ACA*-ACB*D1=F*D2=S
SV3	-ACA*-ACB*D1=S*D2=F*N2=S
SV9	-ACA*-ACB*D1=S*D2=F*N2=F
SV4	-ACA*-ACB*D1=S*D2=S*N2=F
SV5	ACA*-ACB*D1=S*D2=S
SV6	ACA*-ACB*(D1=F + D2=F)
SV7	-ACA*ACB*D1=S*D2=S
SV8	-ACA*ACB*(D1=F + D2=F)
SVF	1
OD1	1
LSF	ACA + D1=F + E1=F*ME=F + UA=F + MA=F

Figure 3.1.2.1-2
Transient Event Tree TR1

MODEL Name: NMP2
Split Fraction Logic for Event Tree: TR1

SF	Split Fraction Logic
LS1	1
LCF	ACB + D2=F + E2=F*ME=F + UB=F + MB=F
LC1	1
LAF	ACA+D1=F+MA=F
LA1	1
LBF	ACB + D2=F + MB=F
LB3	ACA+D1=F+MA=F
LB1	LA=S
LBA	LA=F
LBF	1
IAF	ACA + E1=F*ME=F + UA=F + MA=F
IA1	1
IBF	ACB + E2=F*ME=F + UB=F + MB=F
IBA	IA=F
IB1	IA=S
IBF	1
SWF	SWGF
SW1	-LPI*OD=S
SW2	1 Rule Comment NO OPERATOR, SWRHR CROSSTIE ONLY
FPF	IA=F*IB=F
FP1	NA=S Rule Comment MOTOR AND DIESEL FIRE PUMP
FP2	1 Rule Comment DIESEL FIRE PUMP ONLY

HPI:= FW=S + HS=S + IC=S

LPI:= LS=S + LC=S + LA=S*IA=S + LB=S*IB=S

NOINJ:= -HPI*(-OD=S + -LPI*-SWRHR)

SWRHR:= SW=S

SWGF:= SB=F + ACB + IB=F

Figure 3.1.2.1-2
Transient Event Tree TR1

MODEL Name: NMP2
Split Fraction Logic for Event Tree: TR1

SF Split Fraction Logic

FPRHR:= FP=S

SPBYP:=RQ=B
Rule Comment
RQ=B ENSURES THAT RULE IS NOT SATISFIED IN CET

CM30:=RQ=B

CM2:=RQ=B

CM8:=RQ=B

CM10:=RQ=B

CM19:=RQ=B

Figure 3.1.2.1-3
Transient Event Tree TR2

MODEL Name: NMP2
Event Tree: TR2

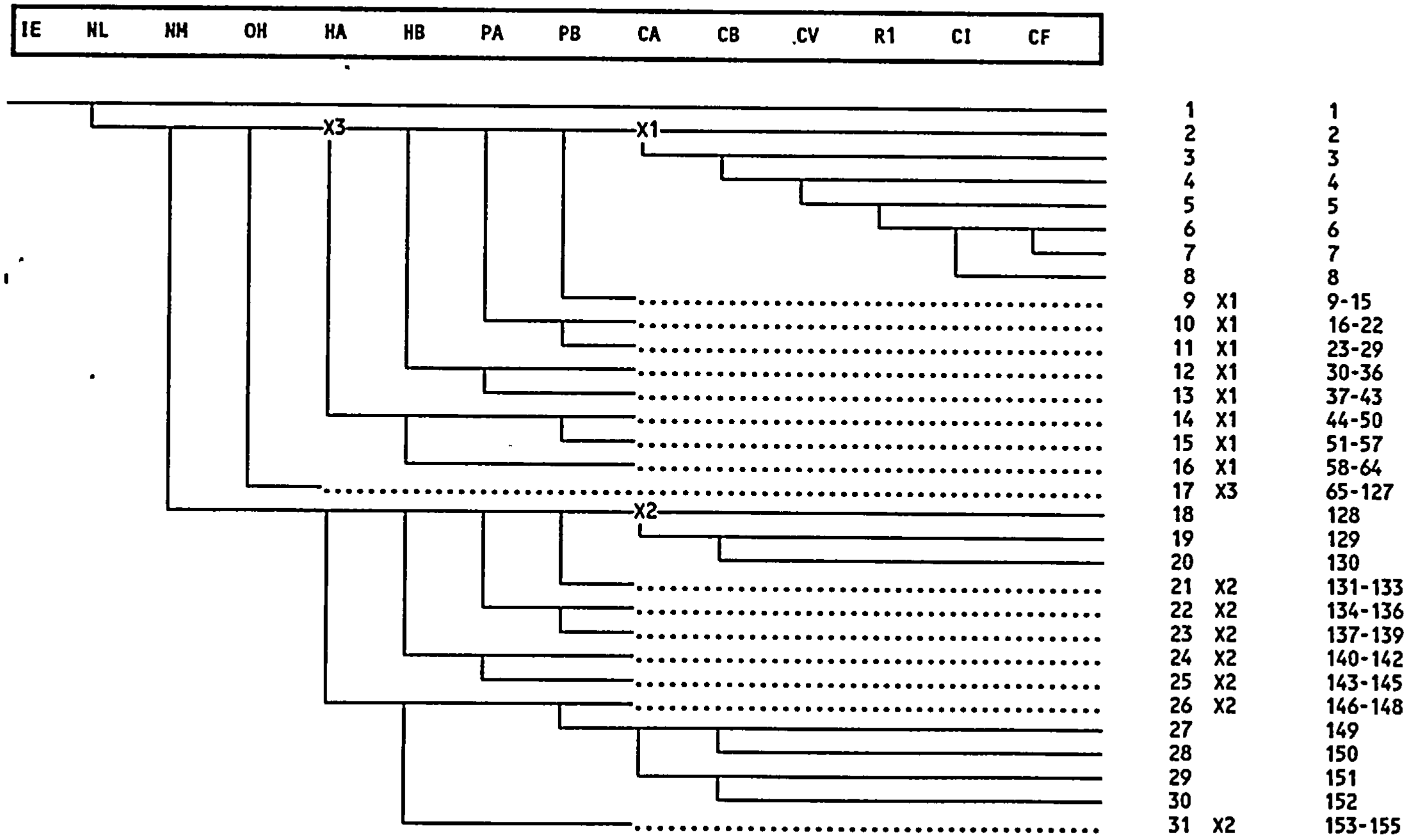


Figure 3.1.2.1-3
Transient Event Tree TR2

MODEL Name: NMP2
Top Event Legend for Tree: TR2

Top Event Designator	Top Event Description
IE	INITIATING EVENT
NL	SUCCESS IN TR1 (NO LATE TREE REQUIRED)
NM	INJECTION SUCCESS (NO CORE DAMAGE IN TR1)
OH	OPERATOR ALIGNS HEAT REMOVAL & PATH
HA	RHR HEAT EXCHANGER A
HB	RHR HEAT EXCHANGER B
PA	SUPPRESSION POOL INJECTION A
PB	SUPPRESSION POOL INJECTION B
CA	CONTAINMENT SPRAY A
CB	CONTAINMENT SPRAY B
CV	CONTAINMENT VENTING
R1	RECOVERY OF OFFSITE POWER FOR LATE SEQUENCES
CI	CONTINUED INJECTION AT HIGH CONTAINMENT PRESS
CF	CONTINUED INJECTION AFTER CONTAINMENT FAILURE

Figure 3.1.2.1-3
Transient Event Tree TR2

MODEL Name: NMP2
Split Fraction Logic for Event Tree: TR2

SF	Split Fraction Logic
NLS	CN=S*(FW=S+HS=S+IC=S) + RQ=F
NLF	1
NMF	NOINJ
NMS	1
OH1	1
HA1	-ACA*SA=S*LA=S
HAF	1
HBF	ACB+SB=F+LB=F
HBB	-ACB*SB=S*LB=S*HA=F*(ACA+SA=F+LA=F)
HB1	-ACB*SB=S*LB=S*HA=S
HBA	-ACB*SB=S*LB=S*HA=F
HBF	1
PAF	NM=S*-HPI*OD=S*LS=F*LC=F*(LB=F+IB=F)*-SWRHR Rule Comment RHR A ALLOWED ONLY WHEN AT LEAST 1 OTHER INJ PUMP AVAIL
PA1	-ACA*LA=S
PAF	1
PBF	ACB+LB=F
PBF	NM=S*-HPI*OD=S*LS=F*LC=F*(LA=F+IA=F)*-SWRHR Rule Comment RHR B ONLY ALLOWED WHEN AT LEAST 1 OTHER INJ PUMP AVAIL
PB1	-ACB*LB=S*-PA=F
PBA	-ACB*LB=S*PA=F
PBF	1
CAS	NM=S*(PA=S*HA=S + PB=S*HB=S)*OH=S Rule Comment NOT REQUIRED SINCE ALREADY HAVE A COOLING PATH
CAF	ACA + LA=F + HA=F + OH=F
CAF	NM=S*-HPI*OD=S*LS=F*LC=F*(LB=F+IB=F)*-SWRHR Rule Comment RHR A ONLY ALLOWED WHEN AT LEAST 1 OTHER INJ PUMP AVAIL
CA1	1
CBF	ACB + LB=F + HB=F + OH=F
CBF	NM=S*-HPI*OD=S*LS=F*LC=F*(LA=F+IA=F)*-SWRHR Rule Comment RHR B ONLY ALLOWED WHEN AT LEAST 1 OTHER INJ PUMP AVAIL

Figure 3.1.2.1-3
Transient Event Tree TR2

MODEL Name: NMP2
Split Fraction Logic for Event Tree: TR2

SF	Split Fraction Logic
CBB	CA=F
CB1	1
CVS	RHR*OH=S Rule Comment NOT REQUIRED SINCE RHR ALREADY SUCCESS
CVF	ACB + N2=F*AS=F
CV1	-ACA*AS=S*N2=S
CV2	(ACA + AS=F)*N2=S
CV5	AS=S*(N2=F + ACA)
CVF	1
R11	OG=F*(A1=F + A2=F)*OH=S
R1F	1
CIS	HS=S
CI1	FW=S*KA=S Rule Comment HPCS REQUIRED BUT NOT ASKED
CI2	FW=S*KA=F
CIF	1
CFF	TA=F + TB=F
CF1	FW=S*RW=S*NA=S*NB=S Rule Comment CRD, HPCS & FEEDWATER POSSIBILITY
CF2	FW=F*RW=S*NA=S*NB=S Rule Comment CRD & HPCS POSSIBLE
CF3	FW=S*(RW=F + NA=F + NB=F) Rule Comment FEEDWATER & HPCS POSSIBLE
CF4	FW=F*(RW=F + NA=F + NB=F) Rule Comment HPCS POSSIBLE
CFF	1

RHR:=LA=S*HA=S*(PA=S+CA=S) + LB=S*HB=S*(PB=S+CB=S)

SPRAY:= LA=S*CA=S + LB=S*CB=S

LOW:= OD=S

Figure 3.1.2.1-3
Transient Event Tree TR2

MODEL Name: NMP2
Split Fraction Logic for Event Tree: TR2

SF	Split Fraction Logic

HIGH:= -LOW

NOSV:= SV=F
Rule Comment
CET USES THIS - HIGH PRESSURE DUE TO EQUIPMENT

SUCCESS:= NL=S + NM=S*(OH=S*RHR + CV=S + R1=S + CF=S)

CLASSIA:= NOINJ*HIGH
Rule Comment
MACROS WITH CLASS.. ARE FOR LEVEL 1 BINS

CLASSID:= NOINJ*LOW

CLASSIC:= RQ=B
Rule Comment
RQ=B ENSURES THAT RULE IS NOT SATISFIED IN CET

CLASSIB:= RQ=B

CLASSIIIB:= RQ=B

CLASSIIIC:= RQ=B

CLASSIIA:= NM=S*(-RHR + OH=F)*CV=F*CF=F*CI=S
Rule Comment
CLASS IIA

CLASSIIT:= NM=S*(-RHR + OH=F)*CV=F*CI=F
Rule Comment
CLASS IIT

CLASSIIL:= RQ=B

CLASSIV:= RQ=B

CLASSIIID:= RQ=B

Figure 3.1.2.1-3
Transient Event Tree TR2

MODEL Name: NMP2
Binning Logic for Event Tree: TR2

Bin	Binning Rules
SUCCESS	SUCCESS
CLASSIA	CLASSIA
CLASSID	CLASSID
CLASSIIA	CLASSIIA
CLASSIIT	CLASSIIT
DEFAULT	1

3.1.2.2 Station Blackout Event Tree Model

As described in Section 3.1.4, all initiating events first pass through the support system event tree. Then, all transients initiating events link to the transient event tree model in Section 3.1.2.1 unless station blackout occurs. Transient station blackout sequences link to the station blackout event tree model described in this section and shown in Figures 3.1.2.2-1 and 2 (event trees SBO1 and SBO2). Station blackout sequences include both loss of offsite power as an initiating event (LOSP) as well as the loss of emergency AC power given that the initiator was not LOSP. The offsite grid is not allowed to be recovered in SBO1 unless it failed. In other words, if station blackout is due to equipment failures at the emergency buses or 115Kv supplies, this situation is not recoverable. Similarly, the emergency diesels are not recoverable in SBO1 unless failure of emergency AC is due to inoperability of the diesel (given that offsite AC fails) and the diesel support systems are available.

Because of the strong time dependency for recovery from station blackout coupled with potential time dependent failure mechanisms of coolant injection systems, it is more accurate to perform a time phased analysis. A coolant injection time phased event tree, SBO1 (Figure 3.1.2.2-1), is developed that minimizes conservative approximations from the treatment of time varying events in a discrete fashion. This time phased approach is used to separate different operator and system responses which are dependent upon the success of other systems. Therefore, the following discussion is provided to explain the spectrum of possible sequences over a 24 hour period for station blackout, and how they are influenced by unavailability of coolant injection (Figure 3.1.2.2-1) and containment heat removal (Figure 3.1.2.2-2).

Event tree SBO1 models a number of time phases where the recovery of offsite AC or a diesel generator is questioned as well as shedding of DC loads to extend the battery life and bypass isolation circuitry to preserve the availability of RCIC. In addition, SBO1 models availability of RCIC, depressurizing the RPV, availability of the diesel fire pump as a source of makeup to the RPV, and whether the RPV remains depressurized. The basis for the time phased model is the results of a station blackout report by GE (GENE-770-04-1290, DRF A00-02336, Rev. 1). The operating staff guidance used are the EOPs and event specific procedure N2-OP-103. The instructions contained in these procedures are discussed below along with their applicability to sequence development.

The following provides a brief summary of the station blackout model:

- If offsite AC power is recovered prior to core damage, the sequence is binned to success. The availability of equipment for recovery increases significantly - the probability of equipment failures is small in comparison to the probability of not recovering AC power.

- If a diesel generator is recovered before core uncovery, the sequence transfers to the second event tree (SBO2) to ask whether containment heat removal is successful - injection is assumed successful. The probability of equipment failures leading to loss of injection given recovery of one emergency diesel is small in comparison to the probability of not recovering a diesel. Containment heat removal is required in SBO2 for success.

- The availability of RCIC extends the time allowed to recover AC power and increases the chances of aligning the diesel fire pump to provide low pressure injection to the RPV.

- Load shedding of DC loads increases the availability of all systems (i.e., RCIC, SRVs and instrumentation) that require DC power beyond two hours.

- The model allows RPV depressurization and injection with the diesel fire pump if RCIC is successful for at least two hours. The RPV must remain depressurized and AC power must be recovered by 19 hours for fire water to remain successful.

The time phased event tree model (Figure 3.1.2.2-1) can be thought of as being divided into the following groups of functional event trees for each time phase:

- Phase 1 (0-30 Minutes)
- Phase 2 (0-2 Hrs.)
- Phase 3 (2-8 Hrs.)
- Phase 4 (8-10 Hrs.)
- Phase 5 (10-19 Hrs.)

The following describes the characteristics of each time phase:

<u>0-30 Minutes</u>
This phase allows AC power to be recovered within the first 30 minutes without considering the availability of RPV makeup. If AC power is recovered, then the station blackout no longer exists. If offsite AC power is recovered, the sequence is binned to success. The availability of equipment for recovery increases significantly and the probability of equipment failures is small in comparison to the probability of not recovering AC power. If a diesel generator is recovered, the sequence transfers to the second event tree (SBO2) to ask whether containment heat removal is successful. The probability of equipment failures leading to loss of injection given recovery of one emergency diesel is small in comparison to the probability of not recovering a diesel.

If AC power is not recovered within 30 minutes, the sequence continues in SBO1, Phase 2.

<u>0-2 Hours</u>
In Phase 2, RCIC must operate for two hours to allow additional recovery time for AC power and for considering use of the diesel fire pump in later phases. If RCIC fails, core damage is assumed to occur at 30 minutes. The model allows the consideration of RPV depressurization, but not alignment of the diesel fire pump because the pump is not judged to be capable of the flowrates necessary for success during this time phase. This discriminates between core damage at high and low pressure.

There are two operator actions that are important to RCIC and DC continued operation beyond two hours:

- Bypassing the RCIC isolation interlock circuitry within two hours.

- Shedding all non-essential loads from the essential battery bus within the first two hours of the transient.

If either of the above operator actions fail, RCIC is assumed to become unavailable at two hours. However, if the RCIC system worked for two hours, recovery of AC power is allowed. Conversely, if AC power is not recovered at two hours, core damage is assumed to occur at two hours due to the above operator failures.

If RCIC is maintained throughout the two hours and both operator actions are accomplished but, AC power is not recovered by two hours, the sequence transfers to the next time phase (2-8 hours). If AC power is recovered, the sequence transfers to success or event tree SBO2 as described above in Phase 1.

2-8 Hours

Upon entering this time phase, RCIC is operational and is being used to control water level. If RCIC fails during the 2-8 hour time phase, the diesel fire pump is the only remaining injection source. If failure of both injection systems occurs anytime during the time phase, the loss of injection is defined to lead to core damage at two hours.

In order to use the diesel fire pump, the operator must depressurize the RPV and maintain the RPV depressurized until AC power is restored. In the 2-8 hour time frame, a principal concern regarding maintaining a depressurized state is the potential depletion of the batteries and nitrogen inventory.

If AC power is not recovered by eight hours and injection has been maintained throughout the time period, the sequence transfers to the next time phase (8-10 hours). If AC power is subsequently recovered, the sequences transfer as described above for Phase 1.

8-10 Hours

If RCIC is operational and is being used to control water level, it can potentially operate in this phase to extend the time for recovering AC power. However, the fault tree model for RCIC in this time phase considers the additional failure mode that the operator depressurizes below the RCIC low steam inlet pressure trip in response to violating the high containment temperature limit (HCTL). The model also considers the conditional failure probability of DC power depletion in the 8-10 hour time phase.

If RCIC fails, the operator must depressurize the RPV and align the diesel fire pump crosstie to RHR. The operator must also maintain the RPV depressurized throughout the time phase. However, the conditional failure probability of DC power is more likely in this time phase compared to the earlier time phase. Failure of low pressure injection is assumed to occur at the beginning of the time phase such that loss of injection leads to core damage at eight hours.

If RCIC or the diesel fire pump successfully provide injection, recovery of AC power at ten hours is questioned. Failure to recover AC power transfers to the 10-19 hour time phase. AC power success transfers to success or SBO2 as discussed above in Phase 1.

10-19 Hours

RCIC is assumed unavailable in this time phase due to guaranteed failure of 125v DC power. Therefore, the event tree does not consider RCIC available in this phase. Consequently, if RCIC was successful in the previous phase, RPV depressurization, diesel fire pump and maintaining the RPV depressurized are required to survive station blackout for 19 hours. If RCIC failed previously, the diesel fire pump had to be successful to enter this phase. In either case, the major issue in this time phase is that of maintaining long-term depressurization of the RPV. With no AC power, no viable means of containment heat removal exists. Eventually, the containment pressure will rise to the point where the SRVs can no longer remain open.

If continued low pressure injection to the RPV is maintained, AC power recovery is questioned. Otherwise, core damage is assumed to occur at ten hours. Successful AC power recovery transfers to success of SBO2 as discussed above in Phase 1. If AC power is not recovered in this time frame, core damage is assumed to occur at 19 hours.

The following describes the event tree top events:

I1, I2, I3, I4, and I5 - Recovery of Offsite Power

These top events model the conditional frequency of recovering offsite power at 30 minutes, 2 hours, 8 hours, 10 hours and 19 hours, respectively. Success of one of these events results in a stable plant condition. It is assumed that this leads to a transient demand similar to MSIV closure events except at lower frequencies. Recovery of offsite power after the first 30 minutes (after I1 failure) is not allowed unless RCIC is successful for the first two hours. The model is based on the quantification of recovery of offsite power which is based on generic information contained in NUREG-1032.

G1, G2, G3, G4, and G5 - Recovery of 1 EDG

These top events model the conditional frequency of recovering at least 1 of 2 emergency diesel generators (EDG) at 30 minutes, 2 hours, 8 hours, 10 hours, and 19 hours, respectively, in the event offsite power is not recovered. Success provides the operator with many options for reestablishing injection (HPCS, RCIC, LPCS, LPCI and CRD). Therefore, the model assumes a source of water to the RPV and the sequence is transferred to event tree SBO2 where containment heat removal is required to achieve a stable plant condition. The model is based on information provided in NUREG-1032.

OA - Operators DC Load Shed and bypass RCIC Isolation Circuitry

Top event OA models the operators shedding DC loads to extend battery life and bypassing the RCIC isolation trip circuitry. Success increases the chances that RCIC and DC power will last at least 8 to 10 hours. Failure results in unavailability of RCIC and all makeup to the RPV at two hours.

U1, U2, and U3 - RCIC Operation

These top events model operation of RCIC during different time phases (0 to 2 hours, 2 to 8 hours, and 8 to 10 hours, respectively). RCIC is designed to start and run without any AC dependence. Although steam driven, RCIC requires DC control power. The DC batteries can supply the load for at least two hours without AC power for charging. Additional time may be available for successful battery operation if load shedding is accomplished by the operator within the first two hours.

RCIC availability during a station blackout is strongly time dependent. The time dependency is principally due to the time varying auxiliary system requirements. The following considerations affect the availability of these systems:

- DC power availability as batteries deplete
- Availability of room cooling and containment heat removal requirements: RCIC isolation on high room temperature and exhaust backpressure.

Failure to shed DC loads or bypass isolation circuitry in top event OA results in unavailability of RCIC beyond two hours (the tree structure does not ask U2).

In the longer term, EOP instructions may lead the operator to depressurize the RPV upon reaching HCTL and cause the loss of RCIC upon the loss of sufficient turbine inlet steam flow. In the 8-10 hour Phase, U3 includes failure modes of the operator depressurizing the RPV below the low steam inlet pressure trip and the conditional probability of DC power depletion.

The following summarizes the RCIC top event models:

- U1 models RCIC start and run for two hours.
- U2 models RCIC operating for six hours (2 to 8 hours).
- U3 models RCIC operating for two hours (8 to 10 hours). No credit is allowed for operating RCIC beyond 10 hours due to guaranteed DC power depletion.

RCIC success at each time phase provides the operator more opportunity for recovering offsite power or an EDG. Success also allows time to align the diesel fire pump and depressurize the RPV which can extend the time to recover AC power. RCIC failure results in core damage if AC power is not recovered or the RPV is not depressurized after the diesel fire pump is aligned for low pressure injection.

O1, O2, and O3 - Operator Depressurizes RPV

Top events O1, O2, and O3 model the operators depressurizing the RPV. The ADS valves, including availability of nitrogen pressure in local valve accumulators, and operator action are included in these top event models. One of the principal functions directed by the symptom based EOPs is to depressurize the RPV during cases in which the suppression pool temperature is rising or when low pressure injection is required for inventory makeup. Several items are of particular importance in evaluating the probability of failure of the operator to depressurize the RPV:

- Automatic ADS would be inhibited in the case of station blackout.

- EOPs direct operator to perform an emergency blowdown of the RPV when HCTL is reached.

- RCIC operation may terminate sooner if depressurization occurs due to the high exhaust back-pressure trip.

- The SRV nitrogen supply is maintained by safety related accumulators. However, these accumulators may be expended over an extended demand period.

- Without suppression pool cooling, the containment pressure may rise sufficiently to compromise the required differential pressure across the SRV pilot valves. This may occur at containment pressure in the range 80 to 85 psi. Adequate differential pressure could be compromised at even lower containment pressures if the accumulator is not fully charged or if they became depleted due to leakage during the extended sequence.

If RCIC is successful for the first two hours (U1 success) and then fails (U2 fails), success of 01 allows an opportunity for the diesel fire pump to be successfully aligned (S1). Similarly, successful depressurization of the RPV in subsequent time phases allows the diesel fire pump to be successfully aligned. Failure of either 01, 02, or 03 guarantees a high pressure core damage sequence. If DC load shedding or RCIC trip bypass fails (OA), and AC power is not recovered or if RCIC is not successful for the first two hours, O1 success implies that core damage occurs while the RPV is at low pressure.

S1, S2 and S3 - Diesel Fire Water to RPV

Top events S1, S2 and S3 model the operators aligning the diesel fire pump to either RHR injection path A or B in accordance with N2-EOP-6 Att. 6. These events are asked if RCIC fails and the operators successfully depressurize the RPV. Failure of either S1, S2, or S3 guarantees a low pressure core damage sequence.

X1, X2, and X3 - RPV Remains Depressurized

These top events model the chances that the RPV remains depressurized through the 8, 10 and 19 hour time periods, respectively. Containment pressure, environmental qualification of the SRVs, and continued availability of DC power and nitrogen are considered in these models. Success allows continued injection with the diesel fire pump, extending the time available for the operator to recover AC power. Failure guarantees a high pressure core damage sequence.

Event Tree SBO2

The second station blackout event tree (SBO2) models containment heat removal if an emergency diesel generator is recovered in SBO1. This tree is similar to the second transient event tree (TR2) described in Section 3.1.2.1 except containment heat removal top events are not asked if core damage sequences transfer to SBO2 (NM failure).

The following summarizes core damage sequences in SBO1 for the different time phases:

- CM30 - RCIC fails to operate for 2 hours and AC power is not recovered in the first 30 minutes.
- CM2 - RCIC succeeded for 2 hours but failed to operate from 2 to 8 hours, the diesel fire pump is not successful or RPV is pressurized, and AC power is not restored at 2 hours. Failure to shed DC loads or to bypass RCIC trip circuitry is modeled as leading to RCIC failure at two hours.
- CM8 - RCIC succeeded for 8 hours but failed to operate from 8 to 10 hours, the diesel fire pump is not successful or RPV is pressurized, and AC power is not restored at 8 hours.
- CM8 - RCIC failed after 2 hours of success, the diesel fire pump was successful from 2 to 8 hours, but the RPV did not remain depressurized after 8 hours and AC power is not restored at 8 hours.
- CM10 - RCIC succeeded for 10 hours, but the diesel fire pump is not successful or RPV is pressurized, and AC power is not restored at 10 hours. RCIC is assumed to fail at >10 hours due to DC battery depletion.
- CM10 - RCIC fails after 2 hours or 8 hours, the diesel fire pump is successful until 10 hours but AC power is not restored at 10 hours and RPV fails to remain depressurized beyond 10 hours.
- CM19 - Injection with the diesel fire pump is successful for 19 hours but AC power is not restored at 19 hours. Since containment heat removal is required by this time, core damage is assumed to occur with RPV pressurized.

Figure 3.1.2.2-1
Station Blackout Event Tree

MODEL Name: NMP2
Event Tree: SBO1

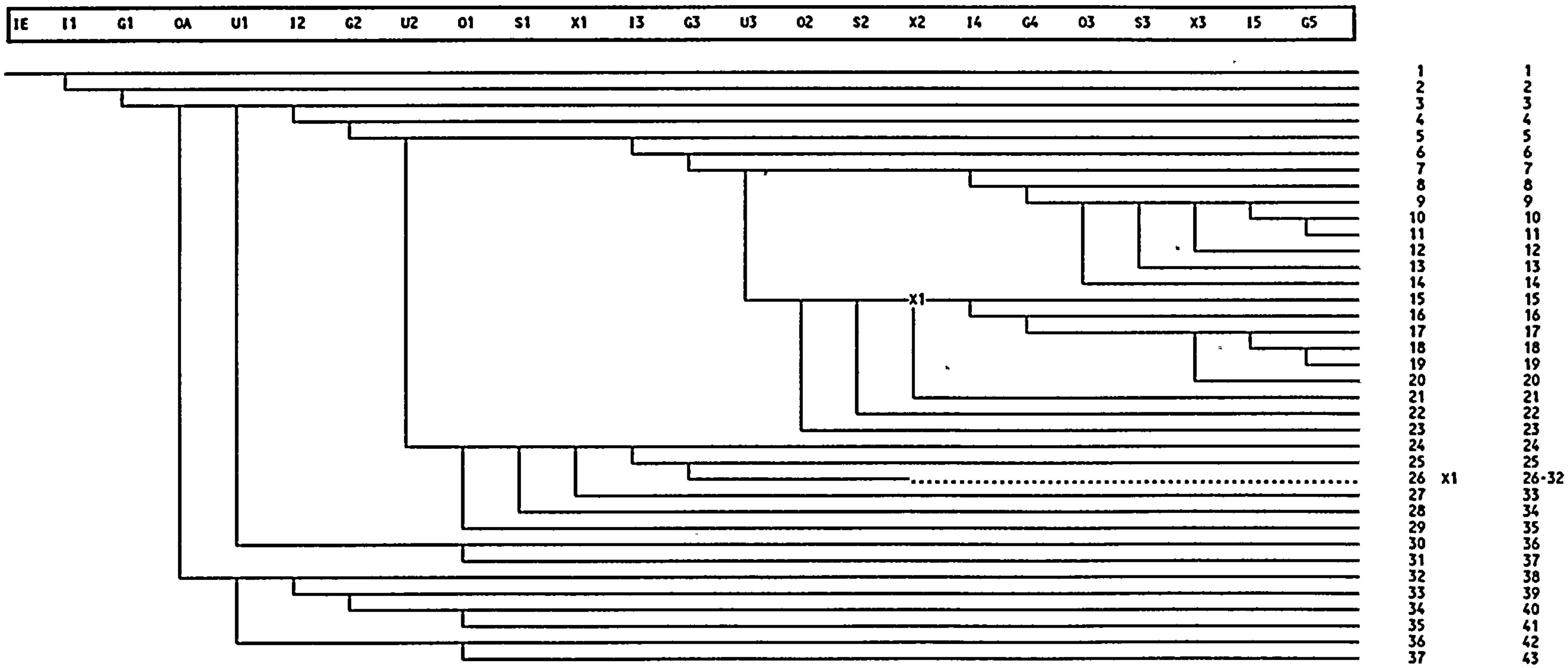


Figure 3.1.2.2-1
Station Blackout Event Tree

MODEL Name: NMP2
Top Event Legend for Tree: SBO1

Top Event Designator	Top Event Description
IE	INITIATING EVENT
I1	RECOVERY OF OFFSITE POWER AT 30 MIN
G1	RECOVERY OF (1) EDG AT 30 MIN
OA	OPERATORS DC LOAD SHEDS & BYPASSES ISOLATION TRIP
U1	RCIC OPERATES FOR T=0-2 HOURS
I2	RECOVERY OF OFFSITE POWER AT 2 HOURS
G2	RECOVERY OF (1) EDG AT 2 HOURS
U2	RCIC OPERATES FOR T=2-8 HOURS
O1	OPERATOR DEPRESSURIZES RPV
S1	FIRE WATER INJECTION TO RPV
X1	RPV REMAINS DEPRESSURIZED UNTIL T=8 HOURS
I3	RECOVERY OF OFFSITE POWER AT 8 HOURS
G3	RECOVERY OF (1) EDG AT 8 HOURS
U3	RCIC OPERATES FOR T=8-10 HOURS
O2	OPERATOR DEPRESSURIZES RPV
S2	FIRE WATER INJECTION TO RPV
X2	RPV REMAINS DEPRESSURIZED UNTIL T=10 HOURS
I4	RECOVERY OF OFFSITE POWER AT T=10 HOURS
G4	RECOVERY OF (1) EDG AT 10 HOURS
O3	OPERATOR DEPRESSURIZES RPV
S3	FIRE WATER INJECTION TO RPV
X3	RPV REMAINS DEPRESSURIZED UNTIL T=19 HOURS
I5	RECOVERY OF OFFSITE POWER AT T=19 HOURS
G5	RECOVERY OF (1) EDG AT 19 HOURS

Figure 3.1.2.2-1
Station Blackout Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: SBO1

SF	Split Fraction Logic
I1S	-BLACK Rule Comment MACRO IN SUPPORT TREE - NONBLACKOUTS ARE MODELED IN TRANSIENT TREE
I11	BLACK1 Rule Comment MACRO IN SUPPORT TREE DEFINES WHEN OFFSITE AC IS RECOVERABLE
I1F	1
G1F	OG=F*A1=S*A2=S*(D1=F+SA=F)*(D2=F+SB=F) Rule Comment NO DIESELS ARE RECOVERABLE DUE TO SUPPORT SYSTEM FAILURES
G11	OG=F*A1=F*A2=F*D1=S*D2=S Rule Comment BOTH DIESELS ARE RECOVERABLE - 1 OF 2 DIESELS BY 30 MINUTES
G12	OG=F*A1=F*D1=S*A2=S*(D2=F+SB=F) + OG=F*A2=F*D2=S*A1=S*(D1=F+SA=F) Rule Comment ONE DIESEL RECOVERABLE - 1 OF 1 DIESEL BY 30 MINUTES
G1F	1
OA1	1
U1F	D1=F + TA=F + TB=F + E1=F*E2=F + UA=F + UB=F
U11	E1=S*E2=S
U12	E1=F + E2=F
U1F	1
I21	BLACK1
I2F	1
G2F	OG=F*A1=S*A2=S*(D1=F+SA=F)*(D2=F+SB=F)
G21	OG=F*A1=F*A2=F*D1=S*D2=S
G22	OG=F*A1=F*D1=S*A2=S*(D2=F+SB=F)+OG=F*A2=F*D2=S*A1=S*(D1=F +SA=F)
G2F	1
U21	1
O1F	D1=F*D2=F
O12	D1=F+D2=F
O11	1
S11	1

Figure 3.1.2.2-1
Station Blackout Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: SBO1

SF	Split Fraction Logic
X11	1
I31	BLACK1
I3F	1
G3F	OG=F*A1=S*A2=S*(D1=F+SA=F)*(D2=F+SB=F)
G31	OG=F*A1=F*A2=F*D1=S*D2=S
G32	OG=F*A1=F*D1=S*A2=S*(D2=F+SB=F) + OG=F*A2=F*D2=S*A1=S*(D1=F+SA=F)
G3F	1
U31	1
O2F	D2=F
O21	1
S21	1
X21	1
I41	BLACK1
I4F	1
G4F	OG=F*A1=S*A2=S*(D1=F+SA=F)*(D2=F+SB=F)
G41	OG=F*A1=F*A2=F*D1=S*D2=S
G42	OG=F*A1=F*D1=S*A2=S*(D2=F+SB=F) + OG=F*A2=F*D2=S*A1=S*(D1=F+SA=F)
G4F	1
O3F	D2=F
O31	1
S31	1
X31	1
I51	BLACK1
I5F	1
G5F	OG=F*A1=S*A2=S*(D1=F+SA=F)*(D2=F+SB=F)
G51	OG=F*A1=F*A2=F*D1=S*D2=S
G52	OG=F*A1=F*D1=S*A2=S*(D2=F+SB=F) + OG=F*A2=F*D2=S*A1=S*(D1=F+SA=F)
G5F	1

CM30:=U1=F

Figure 3.1.2.2-1
Station Blackout Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: SBO1

SF	Split Fraction Logic

CM2:= OA=F*G2=F + U2=F*(O1=F+S1=F+X1=F)

CM8:= U3=F*(O2=F+S2=F+X2=F) + U2=F*G3=F*X2=F

CM10:= X3=F + O3=F + S3=F

CM19:= G5=F

HPI:=I1=B

LPI:=I1=B

NOINJ:= I1=B

SWRHR:=I1=B

SWGF:= I1=B

FPRHR:=I1=B

SPBYP:=I1=B

Figure 3.1.2.2-2
Station Blackout Event Tree

MODEL Name: NMP2
Event Tree: SBO2

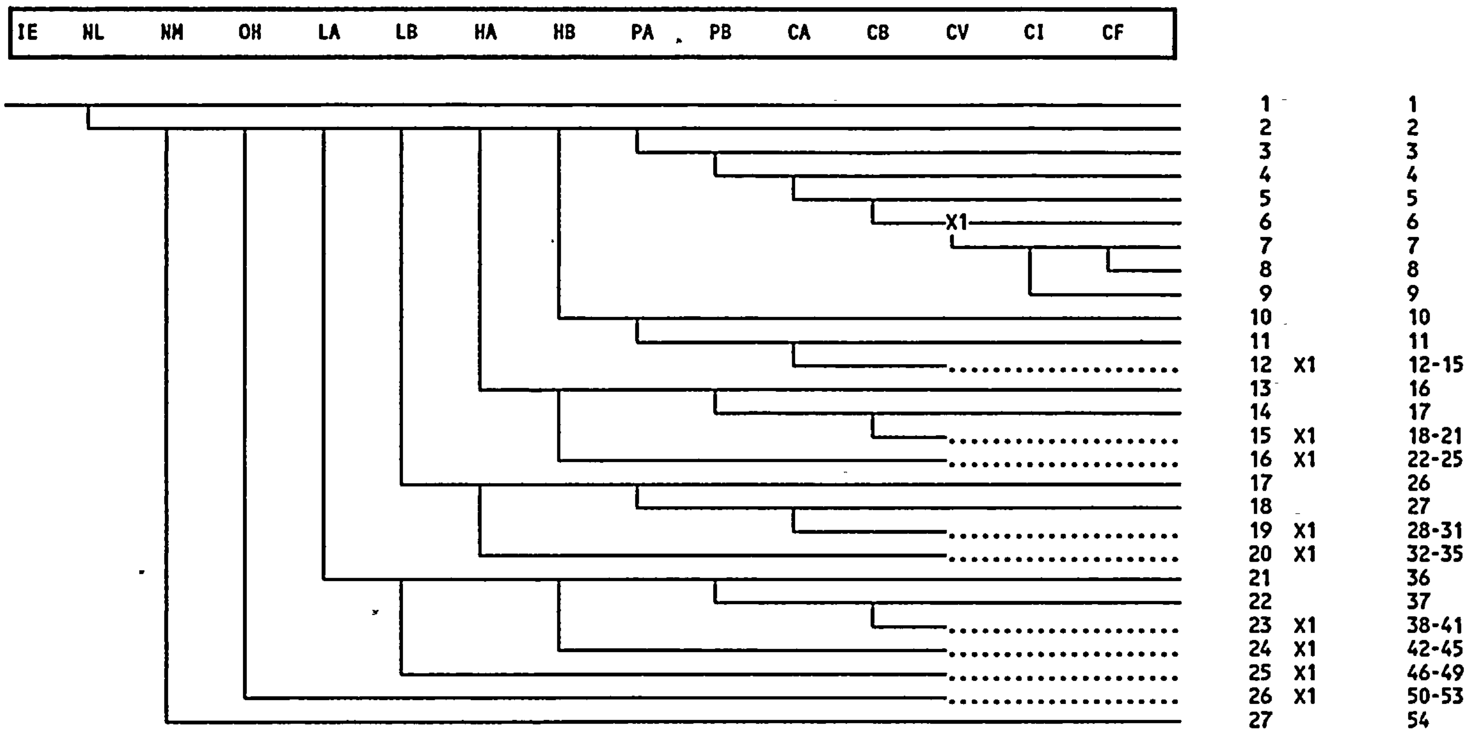


Figure 3.1.2.2-2
Station Blackout Event Tree

MODEL Name: NMP2
Top Event Legend for Tree: SBO2

Top Event Designator	Top Event Description
IE	INITIATING EVENT
NL	SUCCESS IN SBO1 (NO LATE TREE REQUIRED)
NM	INJECTION SUCCESS IN SBO1
OH	OPERATOR ALIGNS HEAT REMOVAL & PATH
LA	RHR PUMP TRAIN A
LB	RHR PUMP TRAIN B
HA	RHR HEAT EXCHANGER A
HB	RHR HEAT EXCHANGER B
PA	SUPPRESSION POOL INJECTION A
PB	SUPPRESSION POOL INJECTION B
CA	CONTAINMENT SPRAY A
CB	CONTAINMENT SPRAY B
CV	CONTAINMENT VENTING
CI	CONTINUED INJECTION AT HIGH CONTAIN PRESS
CF	CONTINUED INJECTION AFTER CONTAINMENT FAILURE

Figure 3.1.2.2-2
Station Blackout Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: SBO2

SF	Split Fraction Logic
NLS	-BLACK+I1=S+I2=S+I3=S+I4=S+I5=S Rule Comment SUCCESS IN SBO1
NLF	1
NMF	CM30+CM2+CM8+CM10+CM19 Rule Comment INJECTION FAILURE IN SBO1
NMS	1
OH1	1
LAF	A1=S*(D1=F+SA=F) Rule Comment A1 IS NOT RECOVERABLE DUE TO SUPPORT SYSTEM FAILURES
LA1	1
LBF	A2=S*(D2=F+SB=F) + OG=F*A1=F*A2=F Rule Comment A2 IS NOT RECOVERABLE DUE TO SUPPORT SYSTEM FAILURES
LB1	1
HA1	1
HB1	1
PA1	1
PB1	1
CA1	1
CB1	1
CVF	A2=S*(D2=F + SB=F) + N2=F Rule Comment A2 NOT RECOVERABLE, A1 IS RECOVERED
CV2	A1=S*(D1=F + SA=F) Rule Comment A1 NOT RECOVERABLE, A2 IS RECOVERED
CV3	A1=F*A2=F Rule Comment EITHER DIESEL COULD BE RECOVERABLE, CV=0.5
CVF	1
CI3	1 Rule Comment HPCS REQUIRED BUT NOT ASKED

Figure 3.1.2.2-2
Station Blackout Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: SBO2

SF	Split Fraction Logic
CF4	1 Rule Comment HPCS AVAILABLE (0.2)

RHR:=LA=S*HA=S*(PA=S+CA=S) + LB=S*HB=S*(PB=S+CB=S)

SPRAY:=LA=S*CA=S + LB=S*CB=S

HIGH:=O1=F+O2=F+O3=F+X1=F+X2=F+X3=F +
G1=S+G2=S+(U2=S*G3=S)+(U3=S*G4=S)

LOW:=-HIGH

NOSV:= D1=F*D2=F + X1=F

SUCCESS:= NL=S + NM=S*(RHR*OH=S + CV=S + CF=S)

CLASSIA:= I1=B

CLASSID:= I1=B

CLASSIC:= I1=B

CLASSIB:= CM30+CM2+CM8+CM10+CM19

CLASSIIIB:= I1=B

CLASSIIIC:= I1=B

CLASSIIA:=NM=S*(-RHR + OH=F)*CV=F*CF=F*CI=S

CLASSIIT:=NM=S*(-RHR + OH=F)*CV=F*CI=F

CLASSIIL:= I1=B

CLASSIV:=I1=B

CLASSIIID:=I1=B

Figure 3.1.2.2-2
Station Blackout Event Tree

MODEL Name: NMP2
Binning Logic for Event Tree: SBO2

Bin	Binning Rules
SUCCESS	SUCCESS
CLASSIB	CLASSIB
CLASSIIA	CLASSIIA
CLASSIIT	CLASSIIT
DEFAULT	1

3.1.2.3 Small LOCA

The small LOCA event sequence diagram in Figure 3.1.2.3-1 summarizes systems and operator actions required to successfully respond to a small LOCA initiating event (SLOCA). The SLOCA break size is defined such that RCIC will provide successful injection on the upper end of the break size. The SLOCA model is similar to the transient model in Section 3.1.2.1 except for the following:

- Timing of events and timing for operator actions is slightly different.
- Vapor suppression (VS) is added to the model as well as the operator actions (OV) required to prevent high containment pressure given failure of VS. Failure to mitigate vapor suppression failure is modeled as an early gross containment failure which causes core damage.
- RCIC is assumed to be unavailable if the vapor suppression system fails due to high turbine exhaust trips.
- The condenser is assumed to be unavailable if the vapor suppression system fails. This forces containment heat removal to be questioned in the event tree model.
- Operator action to restore feedwater is included in the feedwater top event model. It is assumed that high drywell pressure initiates HPCS and causes a Level 8 trip. Thus, the operators have to restore feedwater if necessary.

Success requires SCRAM, injection, and containment heat removal. Success criteria for small LOCA initiating events is summarized in Section 3.1.1. The failure criteria for core damage is the same as described for transients in Section 3.1.2.1 except for the following additional contribution of vapor suppression failure:

- Failure of vapor suppression system (vacuum breakers open) and the operators do not prevent high containment pressure with containment spray (RHR) or containment venting. It is assumed that mitigation is required within 45 minutes before the drywell initiation limit is reached. Failure to mitigate vapor suppression failure is assumed to cause early gross containment failure and core damage.

Two event tree models were developed as shown in Figures 3.1.2.3-2 and 3. The SL1 event tree in Figure 3.1.2.3-2 models reactor SCRAM, containment vapor suppression, and injection. The SL2 event tree in Figure 3.1.2.3-3 models containment heat removal, sprays, venting and continued injection during containment overpressure failure. These two event trees are linked during quantification with the small LOCA initiating event and the support system event tree.

The following summarizes the event tree top events:

<u>RQ - Reactor SCRAM</u>
This top event model is the same as described for transients in Section 3.1.2.1.

VS - Vapor Suppression

Top event VS models the vapor suppression function which is primarily the vacuum breakers between the drywell and wetwell located in the drywell. Two vacuum breakers in series are mounted to each of four short downcomers that communicate with the suppression chamber air space. If two vacuum breakers in series on any one of the downcomers fails (i.e., a path is open while indicating closed), the operators (OV) must align containment sprays or venting in sufficient time to prevent high containment pressure. RCIC (IC) and the condenser (CN) are assumed unavailable if VS fails.

OV - Operator Aligns Containment Sprays or Venting

If vapor suppression system fails, top event OV models the operator actions associated with diagnosing the event and deciding to align containment spray or venting before the drywell initiation limit is reached (45 minutes). The easier task of aligning containment spray from the Control Room is included in this top event. Success of the spray and venting equipment is modeled in event tree TR2 as top events CA, CB, and CV. The operator actions associated with aligning the venting path outside Control Room are modeled in top event CV. Failure of OV is modeled as early gross containment failure and core damage.

CN - Condenser Available

Top event CN models the continued availability of the condenser as described for transients in Section 3.1.2.1 except the condenser is assumed unavailable if vapor suppression (VS) fails.

FW - Condensate and Feedwater

Top event FW models the continued availability of the condensate and feedwater systems as described for transients in Section 3.1.2.1 except the operators are required to restore feedwater after a Level 8 trip. A high drywell pressure signal is assumed to initiate HPCS and cause a Level 8 trip before the operators take manual control.

HS - High Pressure Core Spray (HPCS)

This top event model is the same as top event HS described for transients in Section 3.1.2.1.

IC - RCIC

Top event IC models the RCIC system supplying the RPV for 24 hours from the condensate storage tanks as described for transients in Section 3.1.2.1 except IC is assumed to fail when vapor suppression (VS) fails due to high turbine exhaust pressure trips. IC is also set to failure if all AC power is lost since the system is unlikely to operate for the 24 hour mission time. AC power recovery will be considered later if this assumption is too conservative.

The remaining top events in SL1 and SL2 are the same as described for transients in Section 3.1.2.1, except for top events CA and CB. Containment sprays are modeled to prevent high containment pressure if vapor suppression (VS) fails and the operators (OV) successfully diagnose the event and actuate sprays.

Figure 3.1.2.3-2
Small LOCA Event Tree

MODEL Name: NMP2
Event Tree: SL1

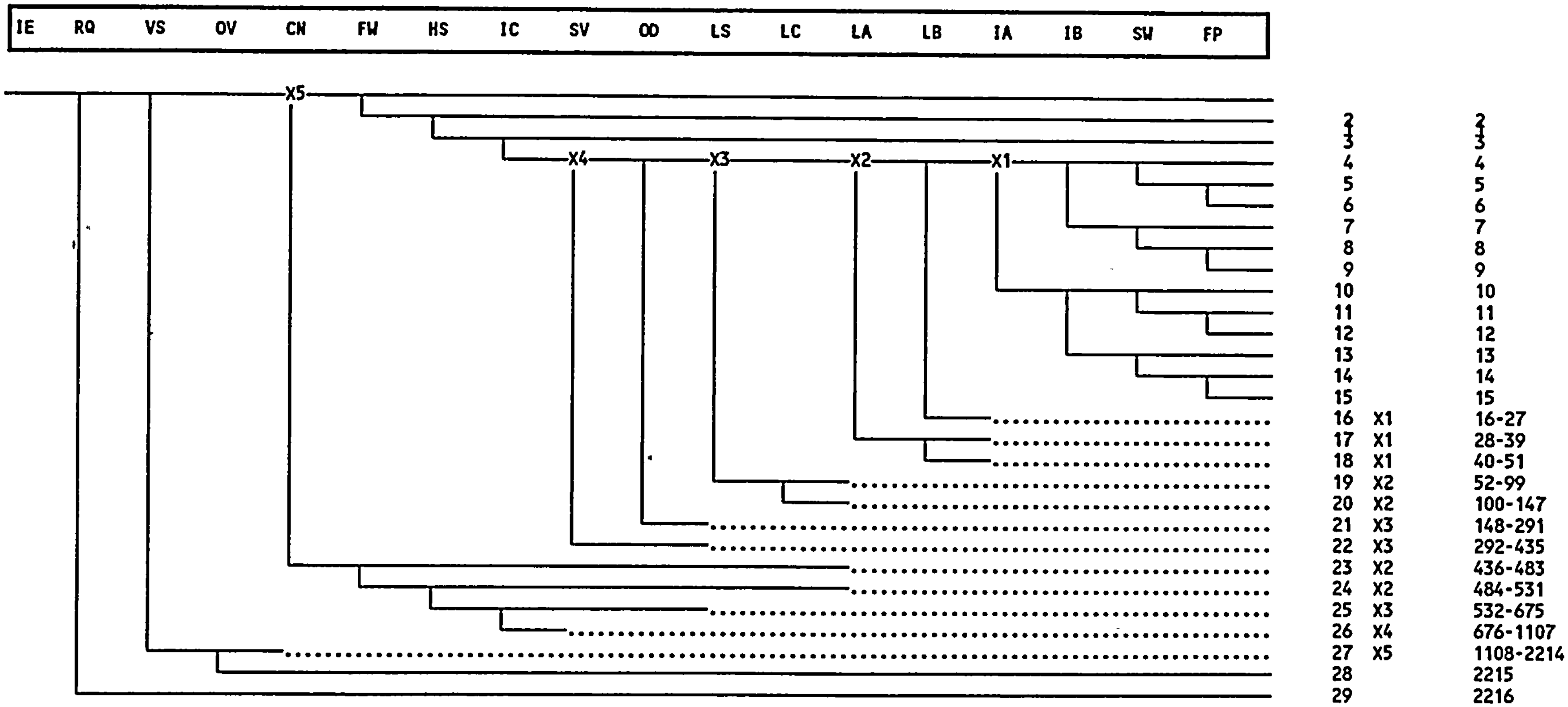


Figure 3.1.2.3-2
Small LOCA Event Tree

MODEL Name: NMP2
Top Event Legend for Tree: SL1

Top Event Designator	Top Event Description
IE	INITIATING EVENT
RQ	REACTOR SCRAM
VS	VAPOR SUPPRESSION
OV	OPERATOR INITIATES SPRAYS OR VENT EARLY
CN	CONDENSER AVAILABLE
FW	CONDENSATE & FEEDWATER
HS	HPCS
IC	RCIC
SV	SRV/ADS VALVES OPERABLE
OD	OPERATOR DEPRESSURIZES RPV
LS	LPCS
LC	LPCI C
LA	RHR PUMP TRAIN A
LB	RHR PUMP TRAIN B
IA	RHR INJECTION A
IB	RHR INJECTION B
SW	SERVICE WATER CROSSTIE TO RHR
FP	FIRE WATER CROSSTIE TO RHR

Figure 3.1.2.3-2
Small LOCA Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: SL1

SF	Split Fraction Logic
RQ1	D1=S*D2=S
RQ3	D1=F*D2=F
RQ2	1
VS1	1
OV1	1
CNF	NA=F+NB=F+TW=F+AS=F+VS=F+N2=F
CN1	1
FWF	NA=F*NB=F+TW=F+AS=F+TA=F+TB=F
FW2	1
HSF	SA=F*SB=F + TB=F
HS1	KA=S*SA=S*SB=S
HS3	KA=S*(SA=F+SB=F)
HS2	KA=F*SA=S*SB=S
HS4	KA=F*(SA=F+SB=F)
HSF	1
ICF	SA=F*SB=F+D1=F+TA=F+TB=F+E1=F*E2=F+UA=F+UB=F+BLACK+VS=F
IC1	E1=S*E2=S
IC2	E1=F + E2=F
ICF	1
SVF	D1=F*D2=F + ACA*ACB
SV1	-ACA*-ACB*D1=S*D2=S*N2=S
SV2	-ACA*-ACB*D1=F*D2=S
SV3	-ACA*-ACB*D1=S*D2=F*N2=S
SV9	-ACA*-ACB*D1=S*D2=F*N2=F
SV4	-ACA*-ACB*D1=S*D2=S*N2=F
SV5	ACA*-ACB*D1=S*D2=S
SV6	ACA*-ACB*(D1=F + D2=F)
SV7	-ACA*ACB*D1=S*D2=S
SV8	-ACA*ACB*(D1=F + D2=F)
SVF	1
OD1	1
LSF	ACA + D1=F + E1=F*ME=F + UA=F + MA=F

Figure 3.1.2.3-2
Small LOCA Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: SL1

SF	Split Fraction Logic
LS1	1
LCF	ACB + D2=F + E2=F*ME=F + UB=F + MB=F
LC1	1
LAF	ACA + D1=F + MA=F
LA1	1
LBF	ACB + D2=F + MB=F
LB3	ACA+D1=F+MA=F
LB1	LA=S
LBA	LA=F
LBF	1
IAF	ACA + E1=F*ME=F + UA=F + MA=F
IA1	1
IBF	ACB + E2=F*ME=F + UB=F + MB=F
IB1	IA=S
IBA	IA=F
IBF	1
SWF	SWGF
SW1	-LPI*OD=S Rule Comment OPERATOR AND SWRHR CROSSTIE
SW2	1 Rule Comment NO OPERATOR, SWRHR CROSSTIE ONLY
FPF	IA=F*IB=F
FP1	NA=S Rule Comment MOTOR AND DIESEL FIRE PUMPS
FP2	1 Rule Comment DIESEL FIRE PUMP ONLY

HPI:= FW=S + HS=S + IC=S

LPI:= LS=S + LC=S + LA=S*IA=S + LB=S*IB=S

NOINJ:= -HPI*(-OD=S + -LPI*-SWRHR)

SWRHR:= SW=S

Figure 3.1.2.3-2
Small LOCA Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: SL1

SF	Split Fraction Logic

SWGF:= SB=F + ACB + IB=F

FPRHR:= FP=S

SPBYP:= VS=F

CM30:=RQ=B
Rule Comment
RQ=B ENSURES THAT RULE IS NOT SATISFIED IN CET

CM2:=RQ=B

CM8:=RQ=B

CM10:=RQ=B

CM19:=RQ=B

Figure 3.1.2.3-3
Small LOCA Event Tree

MODEL Name: NMP2
Event Tree: SL2

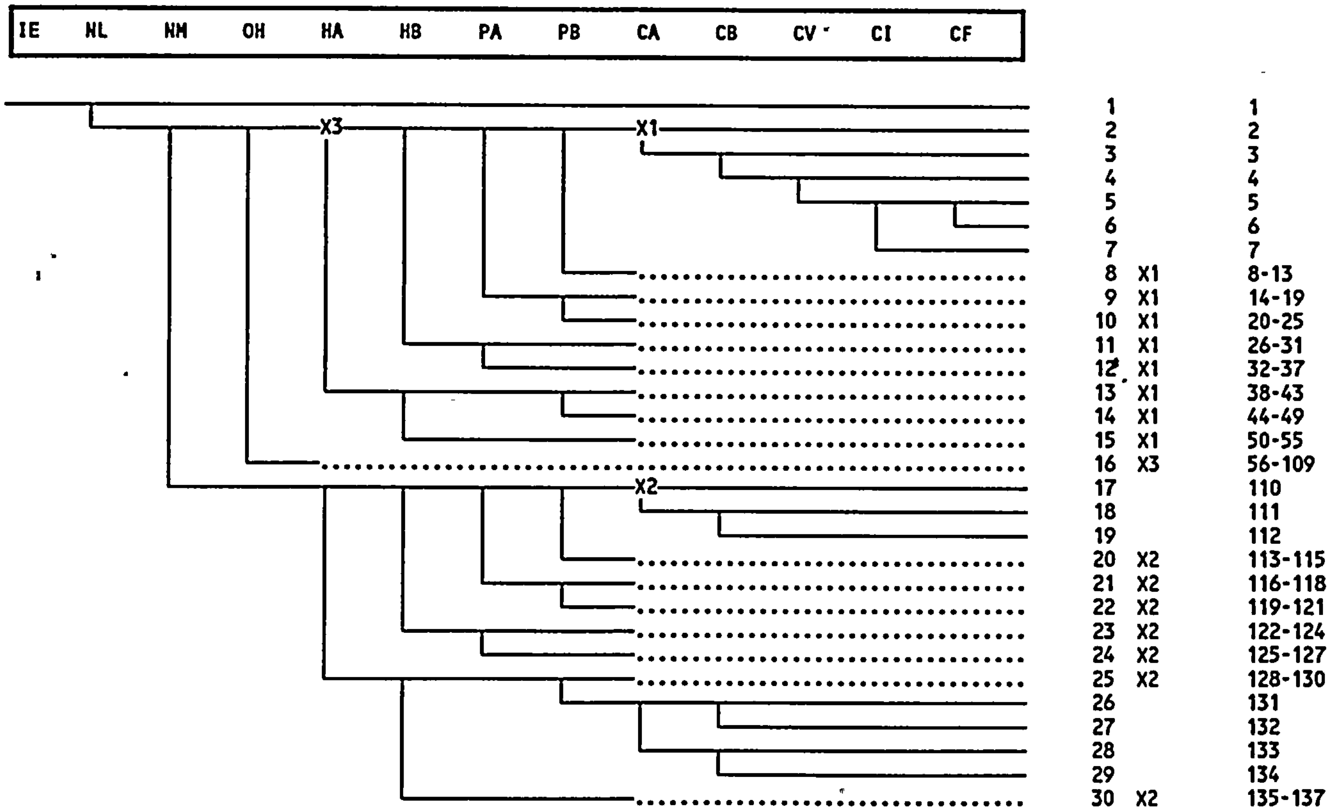


Figure 3.1.2.3-3
Small LOCA Event Tree

MODEL Name: NMP2
Top Event Legend for Tree: SL2

Top Event Designator	Top Event Description
IE	INITIATING EVENT
NL	SUCCESS IN SL1 (NO LATE TREE REQUIRED)
NM	INJECTION SUCCESS (NO CORE DAMAGE IN SL1)
OH	OPERATOR ALIGNS HEAT REMOVAL & PATH
HA	RHR HEAT EXCHANGER A
HB	RHR HEAT EXCHANGER B
PA	SUPPRESSION POOL INJECTION A
PB	SUPPRESSION POOL INJECTION B
CA	CONTAINMENT SPRAY A
CB	CONTAINMENT SPRAY B
CV	CONTAINMENT VENTING
CI	CONTINUED INJECTION AT HIGH CONTAIN PRESS
CF	CONTINUED INJECTION AFTER CONTAINMENT FAILURE

Figure 3.1.2.3-3
Small LOCA Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: SL2

SF	Split Fraction Logic
NLS	CN=S*(FW=S+HS=S+IC=S)
NLF	1
NMF	RQ=F + VS=F*OV=F + NOINJ
NMS	1
OH1	1
HA1	-ACA*SA=S*LA=S
HAF	1
HBF	ACB+SB=F+LB=F
HBB	-ACB*SB=S*LB=S*HA=F*(ACA+SA=F+LA=F)
HB1	-ACB*SB=S*LB=S*HA=S
HBA	-ACB*SB=S*LB=S*HA=F
HBF	1
PAF	NM=S*-HPI*OD=S*LS=F*LC=F*(LB=F+IB=F)*-SWRHR Rule Comment RHR A ALLOWED ONLY WHEN AT LEAST 1 OTHER INJ PUMP AVAIL
PA1	-ACA*LA=S
PAF	1
PBF	ACB+LB=F
PBF	NM=S*-HPI*OD=S*LS=F*LC=F*(LA=F+IA=F)*-SWRHR Rule Comment RHR B ALLOWED ONLY WHEN AT LEAST 1 OTHER INJ PUMP AVAIL
PB1	-ACB*LB=S*-PA=F
PBA	-ACB*LB=S*PA=F
PBF	1
CAS	NM=S*VS=S*(PA=S*HA=S + PB=S*HB=S)*OH=S Rule Comment NOT REQUIRED SINCE ALREADY HAVE A COOLING PATH
CAF	ACA + LA=F + HA=F + OH=F
CAF	NM=S*-HPI*OD=S*LS=F*LC=F*(LB=F+IB=F)*-SWRHR Rule Comment RHR A ALLOWED ONLY WHEN AT LEAST 1 OTHER INJ PUMP AVAIL
CA1	1
CBF	ACB + LB=F + HB=F + OH=F
CBF	NM=S*-HPI*OD=S*LS=F*LC=F*(LA=F+IA=F)*-SWRHR Rule Comment RHR B ALLOWED ONLY WHEN AT LEAST 1 OTHER INJ PUMP AVAIL

Figure 3.1.2.3-3
Small LOCA Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: SL2

SF	Split Fraction Logic
CBB	CA=F
CB1	1
CVS	VS=S*RHR*OH=S + OV=S*SPRAY*RHR*OH=S Rule Comment NOT REQUIRED SINCE RHR ALREADY SUCCESS & VS FAILURE MITIGATED
CVF	N2=F*AS=F + ACB
CVF	OV=S*-SPRAY Rule Comment FORCES FAILURE OF CV IF VS FAILS DUE TO INADEQUATE TIME FOR CV
CV1	(-RHR + OH=F)*-ACA*AS=S*N2=S
CV2	(-RHR + OH=F)*(ACA+AS=F)*N2=S
CV5	(-RHR + OH=F)*(ACA+N2=F)*AS=S
CVF	1
CIS	HS=S
CI1	FW=S*KA=S
CI2	FW=S*KA=F
CIF	1
CFF	TA=F + TB=F
CFF	NM=S*OV=S*-SPRAY*CV=F Rule Comment VAPOR SUPPRESSION FAILURE
CF1	FW=S*RW=S*NA=S*NB=S Rule Comment CRD, HPCS & FEEDWATER POSSIBLE
CF2	FW=F*RW=S*NA=S*NB=S Rule Comment CRD & HPCS POSSIBLE
CF3	FW=S*(RW=F + NA=F + NB=F) Rule Comment FEEDWATER & HPCS POSSIBLE
CF4	FW=F*(RW=F + NA=F + NB=F) Rule Comment HPCS POSSIBLE
CFF	1

RHR:=LA=S*HA=S*(PA=S+CA=S) + LB=S*HB=S*(PB=S+CB=S)

Figure 3.1.2.3-3
Small LOCA Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: SL2

SF	Split Fraction Logic

SPRAY:= LA=S*CA=S + LB=S*CB=S

LOW:= OD=S

HIGH:= -LOW

NOSV:= SV=F

SUCCESS:= NL=S + VS=S*NM=S*(OH=S*RHR + CV=S + CF=S) + OV=S*NM=S*(CV=S + OH=S*RHR*SPRAY)

CLASSIA:= NOINJ*HIGH

CLASSID:= NOINJ*LOW

CLASSIC:= RQ=B
Rule Comment
RQ=B ENSURES THAT RULE IS NOT SATISFIED IN CET

CLASSIB:= RQ=B

CLASSIIIB:= RQ=B

CLASSIIIC:= RQ=B

CLASSIIA:= NM=S*(-RHR + OH=F)*CV=F*CF=F*CI=S

CLASSIIT:= NM=S*(-RHR + OH=F)*CV=F*CI=F

CLASSIIL:= RQ=B

CLASSIVL:= RQ=F

CLASSIVA:= RQ=B

CLASSIV:= CLASSIVA + CLASSIVL

CLASSIIID:= VS=F*OV=F + OV=S*NM=S*-SPRAY*CV=F

Figure 3.1.2.3-3
Small LOCA Event Tree

MODEL Name: NMP2
Binning Logic for Event Tree: SL2

Bin	Binning Rules
SUCCESS	SUCCESS
CLASSIVL	CLASSIVL
CLASSIIID	CLASSIIID
CLASSIA	CLASSIA
CLASSID	CLASSID
CLASSIIA	CLASSIIA
CLASSIIT	CLASSIIT
DEFAULT	1

3.1.2.4 Medium LOCA

The medium LOCA event sequence diagram in Figure 3.1.2.4-1 summarizes systems and operator actions required to successfully respond to a medium LOCA initiating event (MLOCA). The MLOCA break size is defined as too large for RCIC to provide successful injection (greater than small LOCA), but not large enough to depressurize the RPV (large LOCA). The MLOCA model is similar to SLOCA described in Section 3.1.2.3 except for the following:

- Failure to scram is modeled as core damage and not analyzed in the ATWS model.
- The time for operator actions is shorter. Operators must prevent containment failure (OV) within 20 minutes given vapor suppression failure. Depressurization (OD) given HPCS failure is required within 30 minutes.
- Feedwater (FW), RCIC (IC), service water (SW), and fire water (FP) injection sources are not modeled as injection sources in the Level 1 model. The feedwater system is not modeled because of limited makeup capacity to the condenser. In addition, service water and fire water crosstie to RHR supplies are not modeled, because it is assumed that there is insufficient time to align them. SW and FP are included in the event tree model to track availability as a water source later in the containment event tree.
- The condenser (CN) is not modeled. Either low RPV pressure or Level 1 isolation is likely and most of the decay heat would go to the containment for LOCAs larger than a small LOCA.
- An RHR pump injecting to RPV with its heat exchanger is modeled as both successful injection and containment heat removal.
- Top event CI is not modeled for medium and large LOCAs because injection is expected to be transferred to the suppression pool before conditions requiring CI occur, the RPV will be depressurized, and there are no concerns about SRVs reclosing (i.e., guaranteed success of CI given initial injection success).
- Top event CF is included in the event tree model, but is set to guaranteed failure for medium and large LOCAs. This was done to be consistent with the CF model described for transients in Section 3.1.2.1 where high pressure injection and the condensate storage tanks are required. For medium and large LOCAs, it is assumed that the CSTs were injected to the primary containment before switch-over to the suppression pool. As described in Section 3.1.2.1, this is conservative. In addition, this conservative treatment of top event CF in the medium and large LOCA models is not expected to be important.

Success requires scram, injection, vapor suppression or mitigation of its failure, and containment heat removal. Success criteria for medium LOCA initiating events is summarized in Section 3.1.1. The following result in core damage:

- Failure of automatic scram including alternate rod insertion (ARI). No explicit quantitative credit is given to manual scram.

- Failure of vapor suppression system and either the operator fails to mitigate this failure or mitigation equipment fails. The operators have 20 minutes to mitigate vapor suppression failure before the drywell initiation limit is reached.

- Injection Failure: HPCS fails and the RPV is not depressurized or all low pressure ECCS (LPCS and LPCI) pumps fail.

- Loss of Heat Removal: Both RHR heat removal trains fail, containment venting fails and containment overpressure failure causes core damage. For MLOCA and LLOCA, one LPCI pump in the injection mode and injecting through it's heat exchanger provides both successful injection (RPV level control) and heat removal (containment overpressure protection). If RHR fails, containment venting is modeled as a heat removal path. Since the ECCS pumps are designed to operate under saturated conditions, controlled containment venting using the hard piped vent does not fail injection pumps if previously successful. If both RHR and containment venting fail to provide heat removal, continued injection after containment failure is required in order to obtain a success state. However, this is presently set to guaranteed failure.

Two event tree models were developed as shown in Figures 3.1.2.4-2 and 3. The ML1 event tree in Figure 3.1.2.4-2 models reactor scram, containment vapor suppression, and injection. The ML2 event tree in Figure 3.1.2.4-3 models containment heat removal, sprays, venting and continued injection after containment overpressure failure. The two event trees are linked to provide one large front-line response model for a medium LOCA initiating event.

Two types of medium LOCA initiating events are modeled in the medium LOCA event trees.

- A pipe break (steam or liquid) requiring the vapor suppression function. This initiating event is defined in Section 3.1.1 as MLOCA.

- A transient induced medium LOCA due to a stuck open SRV or the inadvertent opening of a SRV as an initiating event. These are grouped and defined as initiating event IORV in Section 3.1.1. Vapor suppression is not required in the short term.

Since most of the top event models are similar to those described for SLOCA and transients in Sections 3.1.2.3 and 3.1.2.1, the top events are not repeated here.

Figure 3.1.2.4-2
Medium LOCA Event Tree

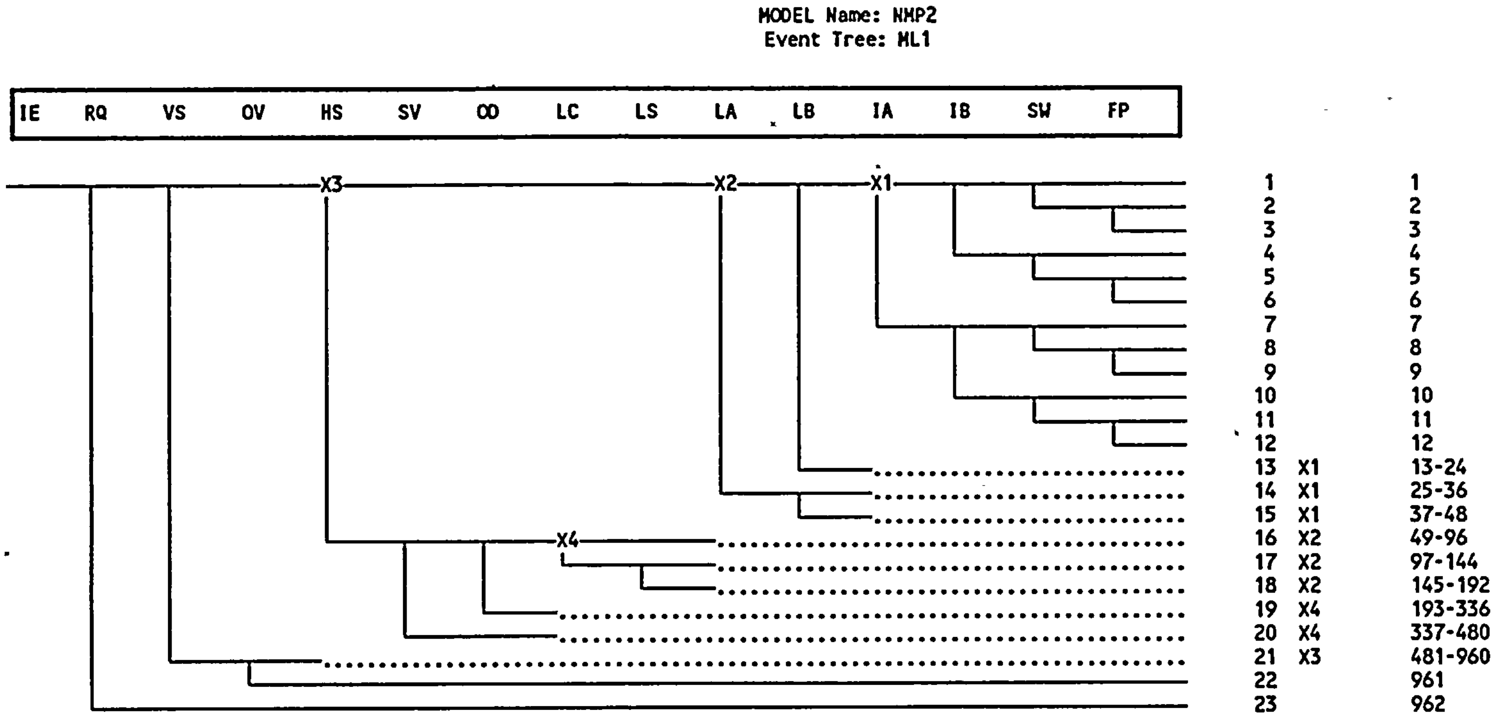


Figure 3.1.2.4-2
Medium LOCA Event Tree

MODEL Name: NMP2
Top Event Legend for Tree: ML1

Top Event Designator	Top Event Description
IE	INITIATING EVENT
RQ	REACTOR SCRAM
VS	VAPOR SUPPRESSION
OV	OPERATOR INITIATES EARLY CONT. SPRAY OR VENT
HS	HPCS
SV	SRV/ADS VALVES OPERABLE
OD	OPERATOR DEPRESSURIZES RPV
LC	LPCI C
LS	LPCS
LA	RHR PUMP TRAIN A
LB	RHR PUMP TRAIN B
IA	RHR INJECTION A
IB	RHR INJECTION B
SW	SERVICE WATER CROSSTIE TO RHR
FP	FIRE WATER CROSSTIE TO RHR

Figure 3.1.2.4-2
Medium LOCA Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: ML1

SF	Split Fraction Logic
RQ1	D1=S*D2=S
RQ3	D1=F*D2=F
RQ2	1
VSS	INIT=IORV
VS1	1
OV2	1
HSF	SA=F*SB=F + TB=F
HSA	INIT=IORV*RW=S*TW=S*AS=S*D1=S*TA=S*TB=S
HS1	KA=S*SA=S*SB=S
HS3	KA=S*(SA=F+SB=F)
HS2	KA=F*SA=S*SB=S
HS4	KA=F*(SA=F+SB=F)
HSF	1
SVF	D1=F*D2=F + ACA*ACB
SV1	-ACA*-ACB*D1=S*D2=S*N2=S
SV2	-ACA*-ACB*D1=F*D2=S
SV3	-ACA*-ACB*D1=S*D2=F*N2=S
SV9	-ACA*-ACB*D1=S*D2=F*N2=F
SV4	-ACA*-ACB*D1=S*D2=S*N2=F
SV5	ACA*-ACB*D1=S*D2=S
SV6	ACA*-ACB*(D1=F + D2=F)
SV7	-ACA*ACB*D1=S*D2=S
SV8	-ACA*ACB*(D1=F + D2=F)
SVF	1
OD2	1
LSF	ACA + D1=F + E1=F*ME=F + UA=F + MA=F
LS1	1
LCF	ACB + D2=F + E2=F*ME=F + UB=F + MB=F
LC1	1
LAF	ACA + D1=F + MA=F
LA1	1
LBF	ACB + D2=F + MB=F

Figure 3.1.2.4-2
Medium LOCA Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: ML1

SF	Split Fraction Logic
LB3	ACA+D1=F+MA=F
LB1	LA=S
LBA	LA=F
LBF	1
IAF	ACA + E1=F*ME=F + UA=F + MA=F
IA1	1
IBF	ACB + E2=F*ME=F + UB=F + MB=F
IB1	IA=S
IBA	IA=F
SWF	SWGF
SW2	1 Rule Comment EQUIPMENT ONLY
FPF	IA=F*IB=F
FP1	NA=S Rule Comment MOTOR AND DIESEL PUMPS
FP2	1 Rule Comment DIESEL PUMP ONLY

HPI:= HS=S

LPI:= LS=S + LC=S + LA=S*IA=S + LB=S*IB=S

NOINJ:= -HPI*(-OD=S + -LPI)

SWRHR:= SW=S

SWGF:= SB=F + ACB + IB=F

FPRHR:= FP=S

SPBYP:= VS=F

CM30:= RQ=B

Rule Comment
RQ=B ENSURES THAT RULE IS NOT SATISFIED IN CET

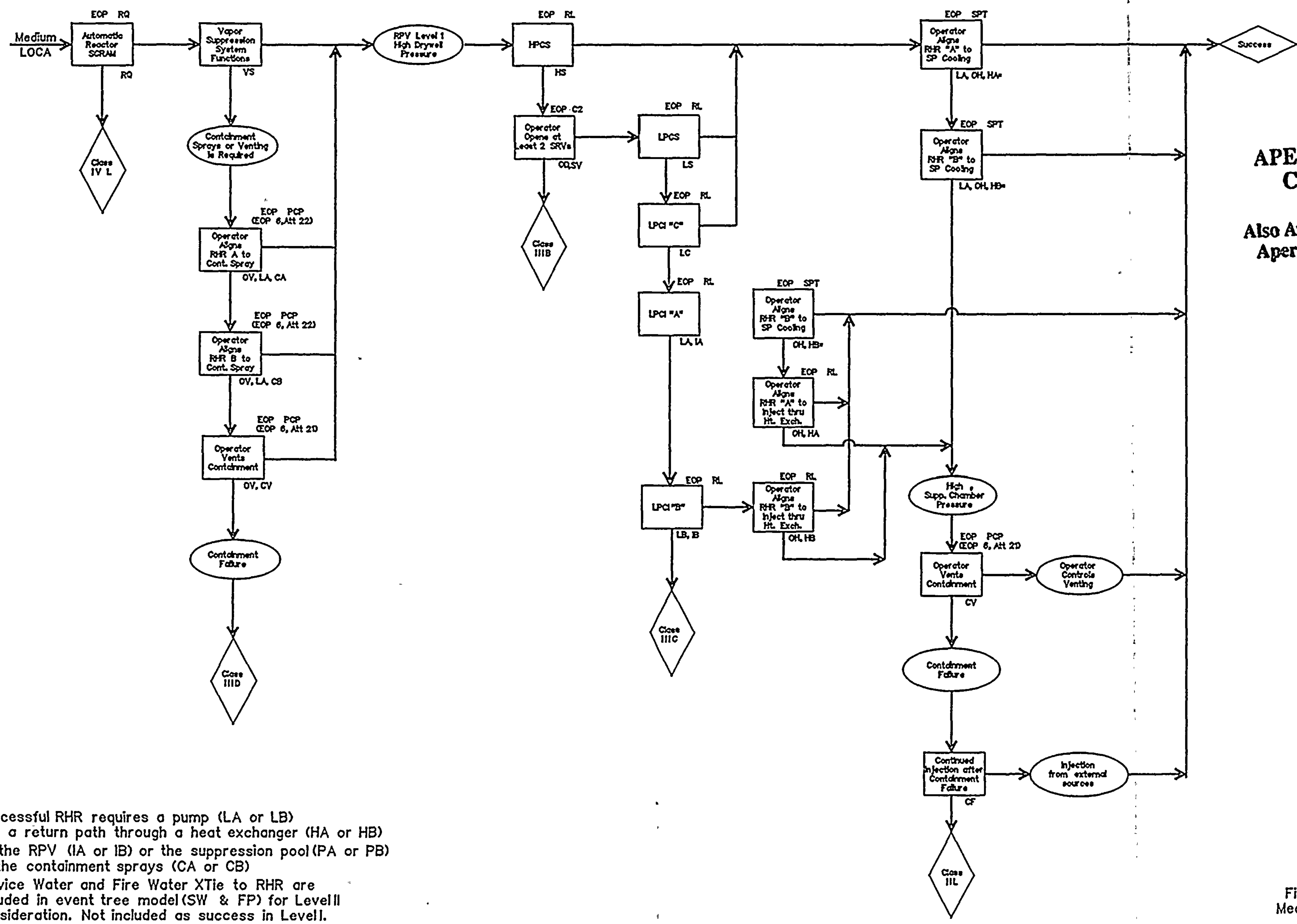
CM2:= RQ=B

CM8:= RQ=B

CM10:= RQ=B

CM19:= RQ=B

Rev 0 (7/92)



* Successful RHR requires a pump (LA or LB) and a return path through a heat exchanger (HA or HB) to the RPV (IA or IB) or the suppression pool (PA or PB) or the containment sprays (CA or CB)

** Service Water and Fire Water XTie to RHR are included in event tree model (SW & FP) for Level II consideration. Not included as success in Level I. Also recovery of Condensate and Feedwater not modeled. RCIC not modeled because MLOCA break is too large for RCIC Success.

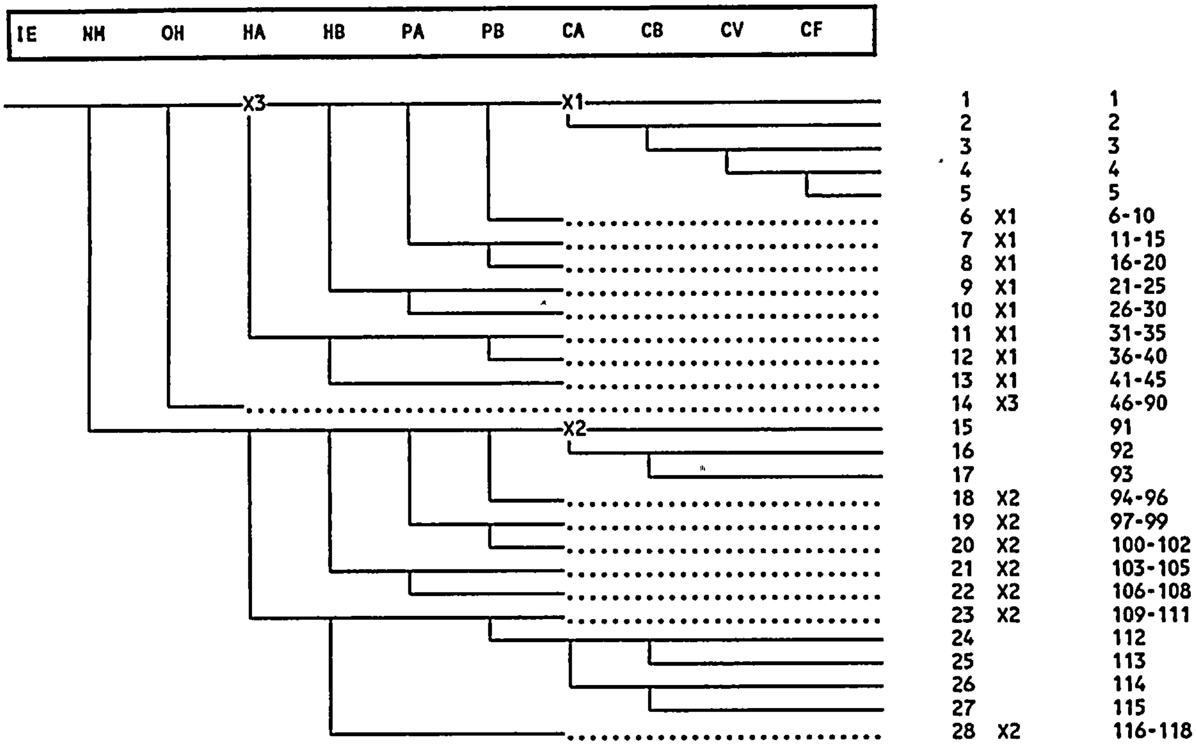
Figure 3.1.2.4-1
Medium LOCA ESD

I:\NMP2IPE\DGN\2F3124.DGN Jul. 07, 1992 10:11:22

9208050191 - 07

Figure 3.1.2.4-3
Medium LOCA Event Tree

MODEL Name: NMP2
Event Tree: ML2



MODEL Name: NMP2
Top Event Legend for Tree: ML2

Top Event Designator	Top Event Description
IE	INITIATING EVENT
NM	INJECTION SUCCESS (NO CORE DAMAGE IN ML1)
OH	OPERATOR ALIGNS HEAT REMOVAL & PATH
HA	RHR HEAT EXCHANGER A
HB	RHR HEAT EXCHANGER B
PA	SUPPRESSION POOL INJECTION A
PB	SUPPRESSION POOL INJECTION B
CA	CONTAINMENT SPRAY A
CB	CONTAINMENT SPRAY B
CV	CONTAINMENT VENTING
CF	CONTINUED INJECTION AFTER CONTAINMENT FAILURE

Figure 3.1.2.4-3
Medium LOCA Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: ML2

SF	Split Fraction Logic
NMF	RQ=F + VS=F*OV=F + NOINJ
NMS	1
OH1	1
HA1	-ACA*SA=S*LA=S
HAF	1
HBF	ACB+SB=F+LB=F
HBB	-ACB*SB=S*LB=S*HA=F*(ACA+SA=F+LA=F)
HB1	-ACB*SB=S*LB=S*HA=S
HBA	-ACB*SB=S*LB=S*HA=F
HBF	1
PAS	NM=S*OD=S*(LA=S*IA=S+LB=S*IB=S) Rule Comment NOT REQUIRED IF INJECTION PATH IS AVAILABLE
PA1	-ACA*LA=S
PAF	1
PBS	NM=S*OD=S*(LA=S*IA=S+LB=S*IB=S) Rule Comment NOT REQUIRED IF INJECTION PATH IS AVAILABLE
PBF	ACB+LB=F
PB1	-ACB*LB=S*-PA=F
PBA	-ACB*LB=S*PA=F
PBF	1
CAS	NM=S*VS=S*(PA=S*HA=S + PB=S*HB=S)*OH=S Rule Comment NOT REQUIRED IF PATH IS AVAILABLE
CAF	ACA + LA=F + HA=F + OH=F
CA1	1
CBF	ACB + LB=F + HB=F + OH=F
CBB	CA=F
CB1	1
CVS	VS=S*RHR*OH=S + OV=S*SPRAY*RHR*OH=S Rule Comment NOT REQUIRED IF RHR SUCCESS AND NO VS OR IT IS MITIGATED
CVF	AS=F*N2=F + ACB

Figure 3.1.2.4-3
Medium LOCA Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: ML2

SF	Split Fraction Logic
CVF	OV=S*-SPRAY Rule Comment FORCES FAILURE OF CV IF VS FAILS DUE TO INADEQUATE TIME
CV1	(-RHR + OH=F)*-ACA*AS=S*N2=S
CV2	(-RHR + OH=F)*(ACA + AS=F)*N2=S
CV5	(-RHR + OH=F)*(ACA+N2=F)*AS=S
CVF	1
CFF	NM=S*OV=S*-SPRAY*CV=F Rule Comment VAPOR SUPPRESSION FAILURE
CFF	1

RHR:=LA=S*HA=S*(OD=S*IA=S+PA=S+CA=S)+LB=S*HB=S*(OD=S*IB=S+PB=S+CB=S)

SPRAY:= LA=S*CA=S + LB=S*CB=S

LOW:= OD=S

HIGH:= -LOW

NOSV:= SV=F

SUCCESS:= VS=S*NM=S*(RHR*OH=S + CV=S + CF=S) + OV=S*NM=S*(CV=S + OH=S*SPRAY*RHR) + INIT=IORV*RQ=F

CLASSIA:= RQ=B
Rule Comment
RQ=B ENSURES THAT RULE IS NOT SATISFIED IN CET

CLASSID:= RQ=B

CLASSIC:= RQ=B

CLASSIB:= RQ=B

CLASSIIIB:= NOINJ*HIGH

CLASSIIIC:= NOINJ*LOW

CLASSIIA:= RQ=B

CLASSIIT:= RQ=B

CLASSIIL:= NM=S*(-RHR + OH=F)*CV=F*CF=F

CLASSIVL:= RQ=F

CLASSIVA:= RQ=B

CLASSIV:= CLASSIVA + CLASSIVL

CLASSIIID:= VS=F*OV=F + OV=S*NM=S*-SPRAY*CV=F

Figure 3.1.2.4-3
Medium LOCA Event Tree

MODEL Name: NMP2
Binning Logic for Event Tree: ML2

Bin	Binning Rules
SUCCESS	SUCCESS
CLASSIVL	CLASSIVL
CLASSIIID	CLASSIIID
CLASSIIIB	CLASSIIIB
CLASSIIIC	CLASSIIIC
CLASSIIL	CLASSIIL
DEFAULT	1

3.1.2.5 Large LOCA

The large LOCA event sequence diagram in Figure 3.1.2.5-1 summarizes the systems and operator actions required to successfully respond to a large LOCA initiating event (LLOCA). The LLOCA break size is defined such that the RPV depressurizes without any requirement to depressurize with the SRVs. The large LOCA model is similar to the medium LOCA model except for the following:

- The timing of events and operator actions is shorter.
- Operator action (OD) and equipment (SV) to depressurize the RPV are not required based on the break size.
- Operator action to prevent containment failure (OV) after vapor suppression system failure is not modeled.

Success requires scram, injection, vapor suppression, and containment heat removal. Success criteria for large LOCA initiating events is summarized in Section 3.1.1. The following result in core damage:

- Failure of automatic scram including Alternate Rod Insertion (ARI). No explicit quantitative credit is given to manual scram.
- Failure of vapor suppression system (vacuum breakers initially open). The containment is assumed to fail before containment sprays or venting can be established. Because this is assumed to lead to a catastrophic containment failure, inventory makeup is also assumed to fail at the time of containment failure.
- All high capacity ECCS pumps fail including High Pressure Core Spray (HPCS).
- Loss of heat Removal is the same as medium LOCA (as described in Section 3.1.2.4).

Two event tree models were developed as shown in Figures 3.1.2.5-2 and 3. The LL1 event tree in Figure 3.1.2.5-2 models reactor scram, containment vapor suppression, and injection. The LL2 event tree in Figure 3.1.2.5-3 models containment heat removal, sprays, venting and continued injection after containment over-pressure failure. These two event trees are linked to provide one large front-line response model for a large LOCA initiating event.

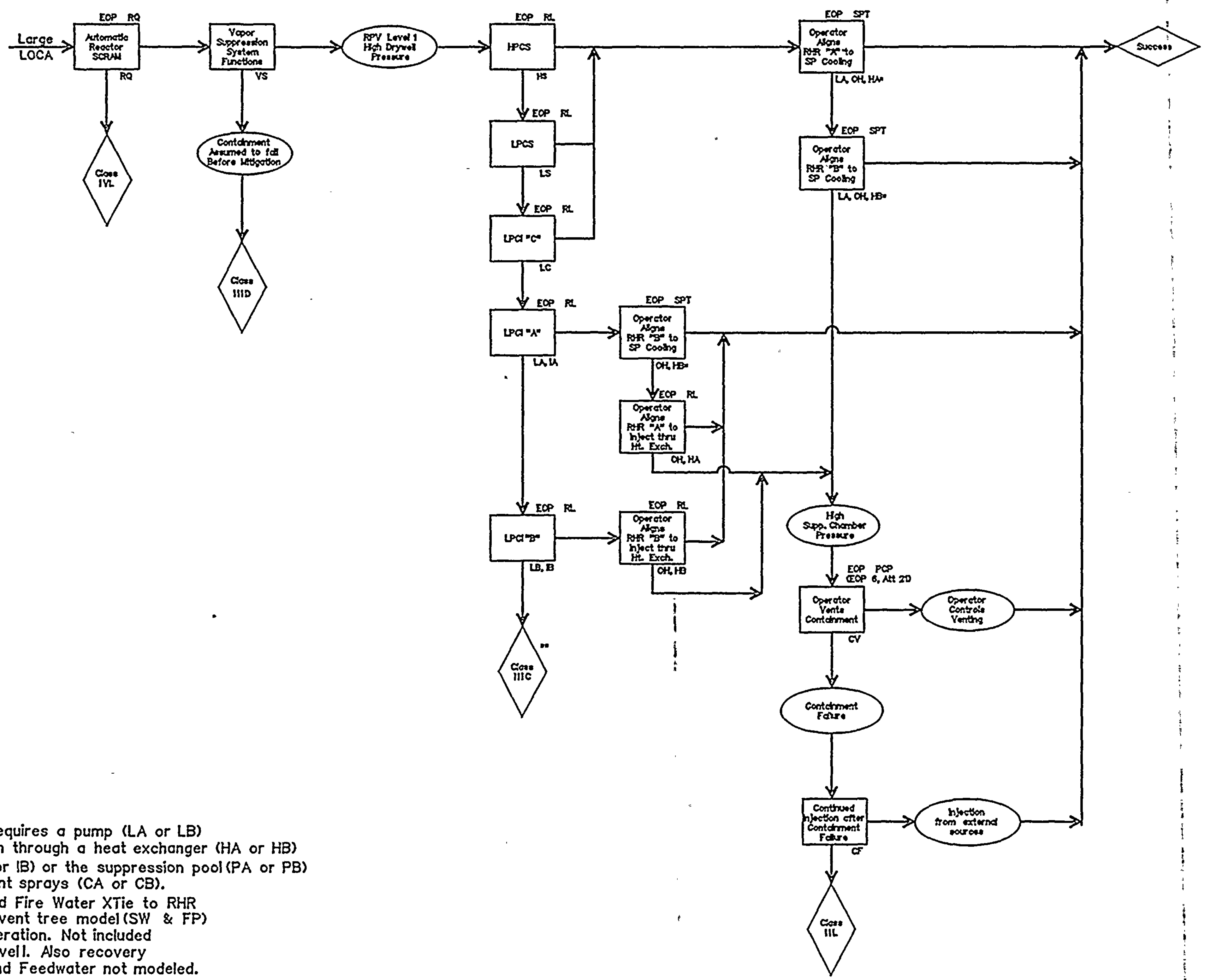
Two types of large LOCA initiating events are modeled in the large LOCA event trees.

- A pipe break (steam or liquid) requiring the vapor suppression function. This initiating event is defined in Section 3.1.1 as LLOCA.

- A transient induced large LOCA due to two or more stuck open SRVs not requiring vapor suppression in the short term. These are judged to be on the same order of magnitude or less than LLOCA. Therefore, explicit modeling of these events was not included.

Since the top event models are similar to those described for MLOCA, SLOCA and transients in Sections 3.1.2.4, 3.1.2.3, and 3.1.2.1, the top events are not repeated here.

Rev 0 (7/92)



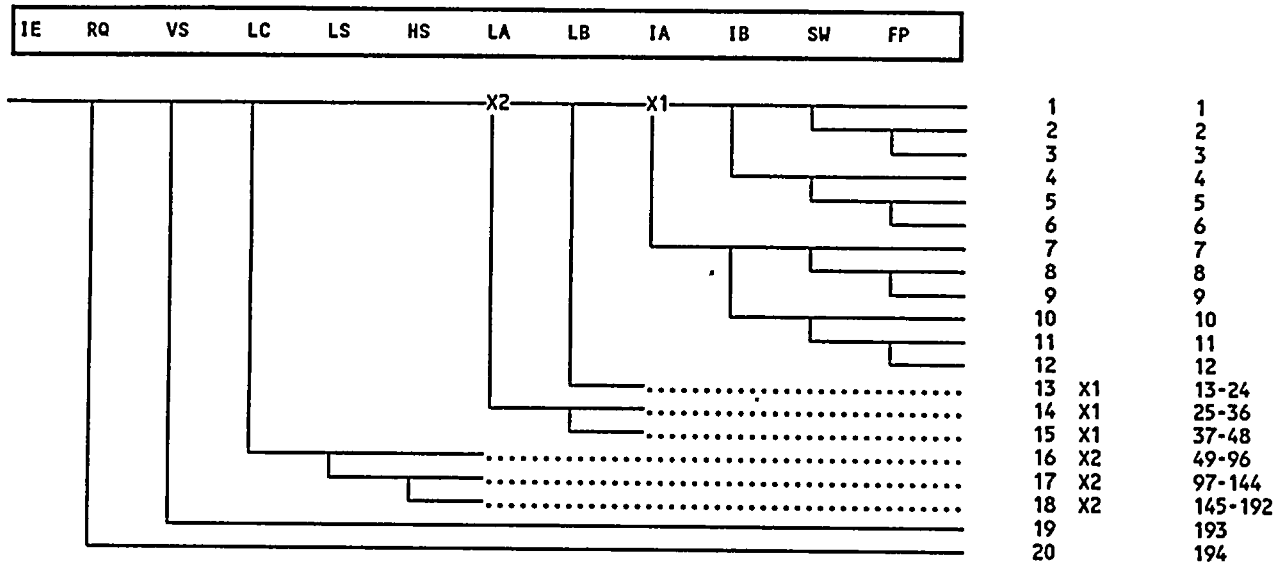
* Successful RHR requires a pump (LA or LB) and a return path through a heat exchanger (HA or HB) to the RPV (IA or IB) or the suppression pool (PA or PB) or the containment sprays (CA or CB).

** Service Water and Fire Water XTie to RHR are included in event tree model (SW & FP) for Level II consideration. Not included as success in Level I. Also recovery of Condensate and Feedwater not modeled.

Figure 3.1.2.5-1
Large LOCA ESD

Figure 3.1.2.5-2
Large LOCA Event Tree

MODEL Name: NMP2
Event Tree: LL1



MODEL Name: NMP2
Top Event Legend for Tree: LL1

Top Event Designator	Top Event Description
IE	INITIATING EVENT
RQ	REACTOR SCRAM
VS	VAPOR SUPPRESSION
LC	LPCI C
LS	LPCS
HS	HPCS
LA	RHR PUMP TRAIN A
LB	RHR PUMP TRAIN B
IA	RHR INJECTION A
IB	RHR INJECTION B
SW	SERVICE WATER CROSSTIE TO RHR
FP	FIRE WATER CROSSTIE TO RHR

Figure 3.1.2.5-2
Large LOCA Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: LL1

SF	Split Fraction Logic
RQ1	D1=S*D2=S
RQ3	D1=F*D2=F
RQ2	1
VS1	1
LCF	ACB + D2=F + E2=F*ME=F + UB=F + MB=F
LC2	1
LSF	ACA + D1=F + E1=F*ME=F + UA=F + MA=F
LS2	1
HSF	SA=F*SB=F + TB=F
HS1	KA=S*SA=S*SB=S
HS3	KA=S*(SA=F+SB=F)
HS2	KA=F*SA=S*SB=S
HS4	KA=F*(SA=F+SB=F)
HSF	1
LAF	ACA + D1=F + MA=F
LA2	1
LBF	ACB + D2=F + MB=F
LBC	ACA+D1=F+MA=F
LB2	LA=S
LBB	LA=F
LBF	1
IAF	ACA + E1=F*ME=F + UA=F + MA=F
IA1	1
IBF	ACB + E2=F*ME=F + UB=F + MB=F
IB1	IA=S
IBA	IA=F
IBF	1
SWF	SWGF
SW2	1
	Rule Comment EQUIPMENT ONLY
FPF	IA=F*IB=F

Figure 3.1.2.5-2
Large LOCA Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: LL1

SF	Split Fraction Logic
FP1	NA=S Rule Comment MOTOR AND DIESEL PUMPS
FP2	1 Rule Comment DIESEL PUMP ONLY

HPI:= HS=S

LPI:= LS=S + LC=S + LA=S*IA=S + LB=S*IB=S

NOINJ:= -HPI*-LPI

SWRHR:= SW=S

SWGF:= SB=F + ACB + IB=F

FPRHR:= FP=S

SPBYP:= VS=F

CM30:= RQ=B
Comment
ENSURES THAT RULE IS NOT SATISFIED IN CET

CM2:= RQ=B

CM8:= RQ=B

CM10:= RQ=B

CM19:= RQ=B

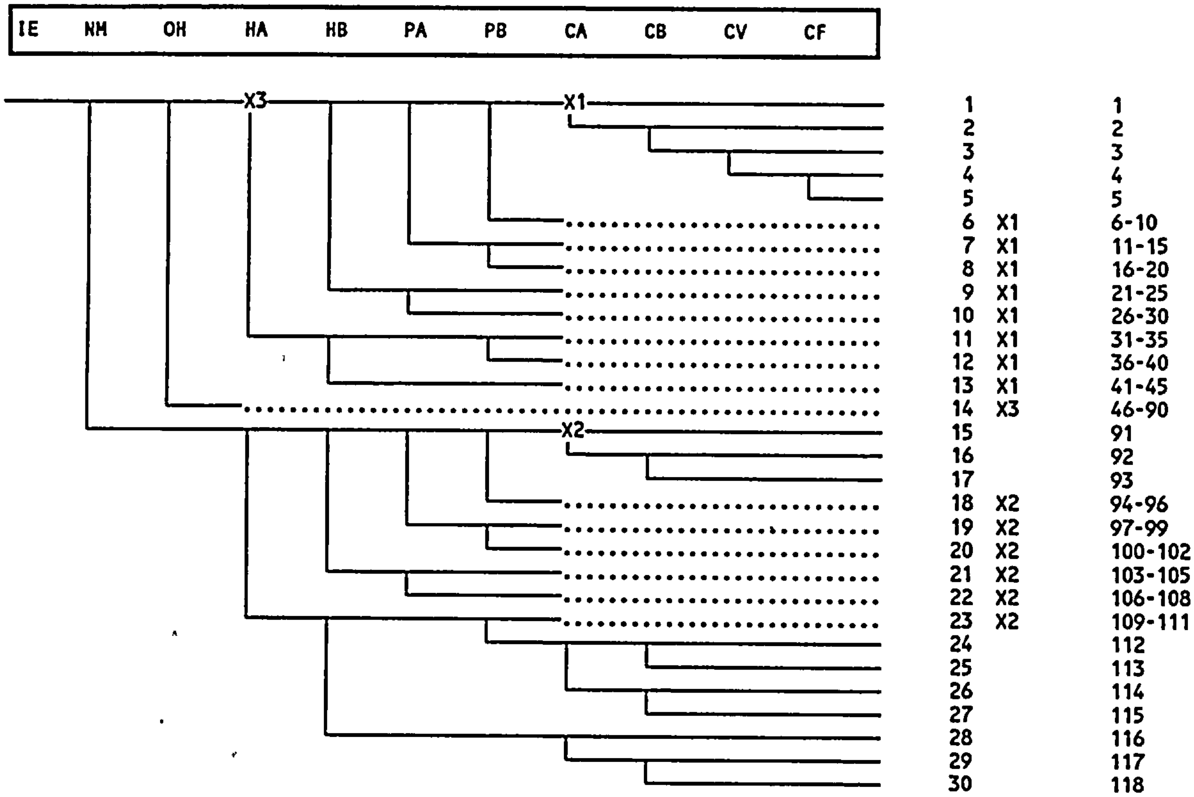
Figure 3.1.2.5-2
Large LOCA Event Tree

MODEL Name: NMP2
Binning Logic for Event Tree: LL1

Bin	Binning Rules
SUCCESS	1

Figure 3.1.2.5-3
Large LOCA Event Tree

MODEL Name: NMP2
Event Tree: LL2



MODEL Name: NMP2
Top Event Legend for Tree: LL2

Top Event Designator	Top Event Description
IE	INITIATING EVENT
NM	INJECTION SUCCESS (NO CORE DAMAGE IN LL1)
OH	OPERATOR ALIGNS HEAT REMOVAL & PATH
HA	RHR HEAT EXCHANGER A
HB	RHR HEAT EXCHANGER B
PA	SUPPRESSION POOL INJECTION A
PB	SUPPRESSION POOL INJECTION B
CA	CONTAINMENT SPRAY A
CB	CONTAINMENT SPRAY B
CV	CONTAINMENT VENTING
CF	CONTINUED INJECTIOIN AFTER CONTAINMENT FAILURE

Figure 3.1.2.5-3
Large LOCA Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: LL2

SF	Split Fraction Logic
NMF	(RQ=F + VS=F + NOINJ)
NMS	1
OH1	1
HA1	-ACA*SA=S*LA=S
HAF	1
HBF	ACB+SB=F+LB=F
HBB	ACA+SA=F+LA=F
HB1	-ACB*SB=S*LB=S*HA=S
HBA	-ACB*SB=S*LB=S*HA=F
HBF	1
PAS	NM=S*(LA=S*IA=S+LB=S*IB=S) Rule Comment NOT REQUIRED IF INJECTION PATH IS AVAILABLE
PA1	-ACA*LA=S
PAF	1
PBS	NM=S*(LA=S*IA=S+LB=S*IB=S) Rule Comment NOT REQUIRED IF INJECTION PATH IS AVAILABLE
PBF	ACB+LB=F
PB1	-ACB*LB=S*-PA=F
PBA	-ACB*LB=S*PA=F
PBF	1
CAS	NM=S*(PA=S*HA=S + PB=S*HB=S)*OH=S Rule Comment NOT REQUIRED IF PATH IS AVAILABLE
CAF	ACA + LA=F + OH=F + HA=F
CA1	1
CBF	ACB + LB=F + OH=F + HB=F
CBB	CA=F
CB1	1
CVS	NM=S*RHR*OH=S Rule Comment NOT REQUIRED IF RHR IS SUCCESS
CVF	AS=F*N2=F + ACB
CV1	(-RHR + OH=F)*-ACA*AS=S*N2=S

Figure 3.1.2.5-3
Large LOCA Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: LL2

SF	Split Fraction Logic
CV2	(-RHR + OH=F)*(ACA + AS=F)*N2=S
CV5	(-RHR + OH=F)*(ACA + N2=F)*AS=S
CVF	1
CFF	1

RHR:= LA=S*HA=S*(IA=S+PA=S+CA=S) + LB=S*HB=S*(IB=S+PB=S+CB=S)

SPRAY:= LA=S*CA=S + LB=S*CB=S

LOW:= INIT=LLOCA

HIGH:= RQ=B
Comment
ENSURES THAT RULE IS NOT SATISFIED IN CET

NOSV:= RQ=B

SUCCESS:= VS=S*NM=S*(RHR*OH=S + CV=S + CF=S)

CLASSIA:= RQ=B

CLASSID:= RQ=B

CLASSIC:= RQ=B

CLASSIB:= RQ=B

CLASSIIIB:= RQ=B

CLASSIIIC:= NOINJ

CLASSIIA:= RQ=B

CLASSIIT:= RQ=B

CLASSIIL:= NM=S*(-RHR + OH=F)*CV=F*CF=F

CLASSIVL:= RQ=F

CLASSIIID:= VS=F

CLASSIVA:=RQ=B

CLASSIV:= CLASSIVA + CLASSIVL

Figure 3.1.2.5-3
Large LOCA Event Tree

MODEL Name: NMP2
Binning Logic for Event Tree: LL2

Bin	Binning Rules
SUCCESS	SUCCESS
CLASSIVL	CLASSIVL
CLASSIIID	CLASSIIID
CLASSIIIC	CLASSIIIC
CLASSIIL	CLASSIIL
DEFAULT	1

3.1.2.6 ATWS

One of the functional requirements for successful accident mitigation is the ability to insert sufficient negative reactivity into the core to bring the reactor subcritical. Low frequency event sequences have been postulated in which an anticipated transient is coupled with a failure to insert sufficient control rods into the reactor, termed ATWS (Anticipated Transient With Failure to Scram) sequences. ATWS sequences typically involve early and large containment failures, leading to substantially different radionuclide release fractions and accident timings than general transients. As such, special accident initiators and event tree models are developed to represent accident sequences.

Two event tree models are shown in Figures 3.1.2.6-1 and 2. The ATWS1 event tree (Figure 3.1.2.6-1) models reactor scram (two top events model the mechanical and electrical contributions to scram failure), and the RRCS and its actuated systems that mitigate ATWS. The ATWS2 event tree (Figure 3.1.2.6-2) models operator actions to inhibit ADS and control RPV coolant inventory and the ability of the operator to mitigate resulting containment overpressurization. These two event trees are linked during quantification with the support system event tree and ATWS initiating events as described in Figure 3.1.2-1.

A number of transient initiating events with failure to scram are modeled with the ATWS event trees. The impact of each initiating event on event tree top events is modeled by using rules that assign the proper conditional split fractions during quantification. For example, the top event for feedwater (FW) would be set to guaranteed failure for the loss of feedwater initiating event or when its support systems are unavailable (i.e., normal AC power).

The following summaries the event tree top events:

IE - Initiating Event

Turbine Trip With Bypass Available

Following an initiating event from high power without control rod insertion (i.e., ATWS), several automatic functions will occur to significantly reduce power production. For example, a high dome pressure signal will initiate an ARI, RPT, and feedwater trip. Given feedwater is recovered and controlled to match the reduced steam production at approximately 50% of full flow, the net steam production could be removed from the system by the condenser via the turbine bypass system, and the suppression pool via the safety relief valves. Because the turbine bypass capacity is 25% of full rated steam flow, the heat load to the suppression pool would also be approximately 25%. This condition has a detrimental effect on containment performance requiring operator action in sufficient time to terminate continued suppression pool temperature rise.

The event tree model considers only transient events initiated from high power. This is considered to be a conservative treatment of these scenarios since the potential effects from an ATWS challenging mitigating system capability is dependent on the amount of residual power being produced in the core. For instance, the turbine trip initiator from low power affords the operator additional time to control reactor power. Specifically, if the initial power is already at 25% or less, feedwater is already balanced to this level, and the turbine bypass is adequate to maintain the condenser as a heat sink for a long period of time without discharging steam to the suppression pool. Conversely, an initiator from high initial power coupled with a failure to scram may initiate a number of automatic protective actions and challenge the operator to mitigate longer term detrimental effects on the plant.

The purpose of this analysis is to establish a best estimate core damage frequency associated with ATWS sequences. To accomplish this end, it is necessary to make engineering judgements regarding systems operability, containment and suppression pool parameters, and operator actions that may be severely challenged during these unusual situations which are far beyond the plant design basis.

MSIV Closure and Loss of Condenser ATWS

The MSIV closure and loss of condenser transient initiators are judged to be important classes of accident initiators because they adversely affect both the normal heat sink and the normal coolant makeup system. Because of the similarity of the effects caused by these isolation initiators on the prevention and mitigating systems, the two events are treated together in this description.

The operator response to an ATWS initiated by a MSIV closure is similar to that explained in the turbine trip case except that:

- Despite the operator's action to reduce power by lowering the reactor water level, a substantial amount of heat (approximately 8 to 18% power) is still being transferred to the suppression pool until boron is injected and sufficient mixing occurs in the core region to reduce the heat load to decay heat levels. Therefore, the operator has less time for establishing heat removal in the MSIV closure initiated transient than in the turbine trip with bypass case involving lowering RPV water level.

- Feedwater flow is terminated as a result of the initiating event. Motor-driven feedwater pumps are assumed incapable of maintaining RPV makeup, given the existing hotwell inventory and the inability of the operator to reopen the MSIVs. The volume in the hotwell is approximately 70,000 gallons, the equivalent of approximately 5 full power minutes of full condensate flow.

- An ATWS event initiated or accompanied by a closure of all MSIVs or loss of condenser will require HPCS or low pressure makeup systems to maintain adequate coolant inventory, as well as the RHR system to remove heat from containment until boron injection is accomplished.

Loss of Feedwater ATWS

Loss of feedwater initiators represent a potentially less severe type of challenge than those associated with isolation events. The key feature of a loss of feedwater event is that the condenser is possibly available as a heat sink. Consequently, the plant response to a loss of feedwater is presumed to be similar to that postulated for the turbine trip with turbine bypass ATWS scenario.

Loss of Offsite Power ATWS

Another accident initiator which can directly affect the availability of mitigating systems is the loss of offsite power. This initiator is assumed to lead to conditions similar to a MSIV closure event, with the added limitation that systems which are to be used may depend on the availability of AC power. For example:

- The recirculation pumps automatically trip.
- SLC requires the EDGs.
- RCIC can be used without AC power.
- Feedwater/condensate and the main condenser are unavailable.
- LPCI, LPCS, and HPCS are dependent on EDGs

The event response for this initiator is virtually identical to that for MSIV closure.

Inadvertent Open Relief Valve ATWS

The IORV accident initiator is treated separately because it is perceived (based upon historical data collected for Target Rock 3 Stage SRVs) to be a relatively frequent event requiring operator intervention to manually initiate reactor shutdown. This is believed to be conservative since NMP2 has a newer improved relief valves (Dikkers). The human error rate contributes significantly to the calculated probability of successful accident mitigation for the IORV initiator.

This event occurs when one of the primary relief valves on the main steam lines inadvertently opens without extraneous influence or challenge from another system (i.e., all reactor pressure and water levels in the reactor coolant pressure boundary are assumed to be at a nominal value before the initiation of the event). When the relief valve opens, there is a momentary depressurization until the turbine pressure control senses the pressure decrease and closes the turbine governor valve in response. In response to the situation, the operator should align and initiate the RHR system in the pool cooling mode to maintain pool temperature. However, the suppression pool temperature will continue to rise until the operator is required to manually scram the reactor. If manual scram or the ARI are ineffective for initiating control rod insertion, the SLC system can insert sufficient negative reactivity to shutdown the reactor and maintain the suppression pool temperature within design limits.

Because of the uncertainty associated with the responses and timing of automatic and operator actions during this sequence of events, it is assumed that the required operator actions and timing for implementing these actions are similarly characterized by those determined for the MSIV closure event. This is considered to be a conservative treatment of the scenario because the condenser may be available for heat removal; thereby, limiting the suppression pool temperature rise.

QM, QE - RPS Mechanical and Electrical Failure

Postulated failures of the reactor scram system may be attributed to combinations of random independent failures or common mode failures. Both types of failures are considered to be of low probability; however, only the common mode failure of scram is considered probabilistically significant.

The mechanical redundancy of the control rod drive mechanisms and the hydraulic system is questioned in top event QM. Similarly, the electrical diversity of sensors, logic, and scram solenoids that reduce the potential for failure of multiple control rods to insert due to electrical common mode failures is questioned in top event QE.

Given electrical induced scram failure and mechanical scram success, the event tree questions the availability of recirculation pump trip (RPT, top event RT) and alternate rod insertion (ARI, top event RI) as functionally redundant to the electrical portion of the scram system. Given mechanical scram failure or electrical scram failure and ARI failure, the event tree questions the availability of the RPT and the standby liquid control (SLC, top event SL) as functionally equivalent to the control drive (CRD) system.

C1, C2 - Redundant Reactivity Control

Top Events C1 and C2 model the two divisions of redundant reactivity control system (RRCS) including input signals required to automatically actuate ATWS mitigating systems. Failure of the RRCS leads to core damage and containment failure. No credit is given to manually initiating reactivity control systems normally actuated by RRCS (i.e., ARI, RPT, feedwater runback, and SLC initiation).

RT - Recirculation Pump Trip

The recirculation pumps provide a method of changing core reactivity without changing control rod position (i.e., positive reactivity can be inserted by increasing the recirculation pump flow). The recirculation pumps automatically trip on either high vessel dome pressure or low reactor water level. The RPT is effective in rapidly inserting sufficient negative reactivity into the core to limit the power and pressure rise following an ATWS to within acceptable limits. At NMP2, the reactor power will drop to approximately 50% following a RPT from a 100% power turbine trip ATWS without feedwater runback or trip.

If feedwater flow to the RPV can be subsequently suspended, the core power will drop even further to approximately 25-30% determined to a large extent by RPV level and pressure. If water level in the core is reduced to near the top of the active fuel, core power can be again reduced because of the resulting higher core void fraction.

Failure of the RPT following a turbine trip from high power is considered to lead to high primary system pressure greater than the service C limit of 1500 psig. It is then assumed a breach in the primary system would occur; low pressure injection would automatically initiate causing recriticality if ARI subsequently failed. Otherwise, the sequence would be evaluated as a large LOCA given ARI operated as designed. However, the model treats failure of RT as causing core damage and containment failure.

RI - Alternate Rod Insertion

ARI is a functionally redundant and diverse system to the electrical portion of the RPS. The system provides added assurance that the postulated electrical failures will not prevent control rod insertion. Failure of RI has the same impact as mechanical scram (QM) failure.

SR - Adequate Pressure Control

The capacity of safety relief valves provide a high level of confidence that there will be sufficient capacity to avoid excessive pressure inside the reactor system following an ATWS. Failure of the vessel pressure relief function is assumed to cause a LOCA. The LOCA would challenge low pressure ECCS to replenish coolant inventory. The injection of cold unborated water is assumed to cause recriticality, eventually leading to containment failure. Successful pressure control is a function of the number of challenges and the number of SRVs required to open per challenge. Failure of top event SR is assumed to cause core damage and containment failure.

FT - Feedwater Runback

Failure to minimize feedwater flow initially during an ATWS scenario could induce RPV overpressure caused by excessive power generation as RPV level remains high post MSIV isolation. Factors affecting the success of feedwater runback are plant specific and include the following:

- Total relief capacity including turbine bypass for turbine trip initiators.
- Presence of an automatic feedwater trip.
- Operator actions to trip feedwater or immediately lower RPV water level.
- Feedwater trip on Level 8 induced by the post turbine trip/MSIV closure response.

Feedwater runback is initiated by RPV high dome pressure for 20 seconds and APRM not downscale trip. Actuation of this circuitry results in the closure of all FW control valve actuators. After 25 seconds, manual control of the feedwater regulating valves is returned to the operator. Failure to minimize feedwater flow initially in an ATWS scenario is expected to induce RPV overpressure caused by excessive power generation. This is assumed to cause core damage and containment failure.

SL - Standby Liquid Control System (SLC)

This top event models automatic injection of SLC to shutdown the reactor before the suppression pool temperature reaches the BIIT limit as prescribed in the EOPs. SLC is initiated by the redundant reactivity control system (top events C1 and C2). Operator action to manually initiate the SLCS is not included in the ATWS model. Failure of top event SL is assumed to result in a containment failure and core damage. For the ATWS model, the ultimate containment capacity is expressed in terms of 240 F maximum suppression pool temperature.

OH, LA, HA, PA, LB, HB, PB - Suppression Pool Cooling

These top events are the same as described for transients in Section 3.1.2.1 except for top event OH which models the operators initiating suppression pool cooling. In the ATWS model, this operator action is required within 20 minutes. Either one train of RHR or the condenser is required for successful heat removal.

MS - Mode Switch Placed in Shutdown

Success of MS (i.e., the up branch) implies that the operator has placed the mode switch in shutdown, as required by the EOPs, immediately after the reactor scram. Should the operator not accomplish this action, it is assumed for this analysis that the operator cannot recover from this error before the RPV is subsequently depressurized. Therefore, upon either ADS actuation or emergency RPV blowdown, the MSIVs are assumed to close, isolating the RCS from the condenser (i.e., top event CN is set to failure).

CN - Condenser Available

This top event model accounts for the potential failure modes that can isolate the RPV and prevent the operator from using the condenser as a heat sink. These failure modes include the condenser and its support equipment. Loss of condenser initiating events are considered unrecoverable by the operator; whereas, some MSIV closure initiating events are potentially recoverable in terms of restoring the condenser as a heat sink. Other operator failures that result in MSIV closure are modeled as separate top events. The operators failing to put mode switch in SHUTDOWN is modeled in top event MS and the operators failing to defeat the low RPV water level isolation interlock is modeled in top event MO.

FW - Feedwater/Condensate Available

Given that the condenser is available, the operator is instructed to maintain water inventory in the RPV using a variety of injection systems, including the feedwater system. The system requirements for adequate feedwater injection are the same as for a general transient event, except that the operator has to restore flow to the RPV post feedwater runback or trip on Level 8. The hotwell inventory is considered inadequate for providing long term high pressure makeup if the condenser is unavailable or the RPV is isolated from the condenser. Otherwise, success implies no core damage given that the operator has initiated boron injection.

SW - Service Water Crosstie to RHR

This top event model is the same as described for transients in Section 3.1.2.1 except SW is only asked for Level 2 containment analysis considerations.

NL - Success In Event Tree AT1

This "switch" is used to bypass the remainder of the ATWS model in event tree AT2 for all sequences not resulting in containment failure or core damage as evaluated in AT1. These sequences include the successful operation of RPS (or ARI in the case of an RPS electrical failure), or automatic actuation of the SLCS and the use of BOP systems for RPV inventory control and heat removal.

NE - No Vessel Failure

This top event is used as a "switch" to identify all event scenarios from AT1 that resulted in overpressurization of the RPV and failure of the reactivity control function. These sequences bypass the AT2 model and are binned in Class IV core damage states.

NM - No SLC Failure

Top event NM is a switch to identify scenarios from AT1 where SLC was required for success. The AT2 model is designed to further investigate sequences that include successful actuation of the SLC system as determined in the AT1 event tree (SL and NM success). All other scenarios (SL and NM failure) bypass the remainder of the model and are binned to Class IV core damage states.

SO - No Stuck Open Relief Valve

Due to large number of relief valves that are expected to actuate during an ATWS event, the likelihood that one SRV will stick open is judged to be more likely than for a general transient. If a stuck open SRV does occur, it is assumed that the capacity of the RCIC system is insufficient to maintain RCS inventory and elevated containment conditions cause containment isolation and unavailability of the MSIVs and condenser.

IC - RCIC Operation

NMP2 EOPs (Rev. 4) instruct the operator to maintain coolant inventory above -45 in. using exclusively the SLCS, CRD, feedwater/condensate, and RCIC systems. Sequences that include the operation of the feedwater system bypass AT2 as a success sequence. This top event considers the probability that RCIC, and the minimal makeup provided by the SLCS, is sufficient to maintain vessel inventory above one-third core height during boron injection. The ability of this limited capacity system to maintain vessel inventory under ATWS conditions is dependent on the equivalent power being generated in the core. For the case where SLC is successful, makeup from RCIC and SLC is considered adequate inventory makeup to maintain level above TAF.

MO - Operator Overrides Low Level MSIV Closure

This top event is used to discriminate between scenarios where the MSIVs are open. For example, given that the MSIVs are initially open post transient initiator, the EOPs instruct the operator to defeat the low RPV water level MSIV isolation interlock. If this action is not accomplished before the water level reaches Level 1 (i.e., assumed to occur as a result of the operator implementing level/power control), the MSIVs are assumed to close. Success at this node implies that the condenser is potentially available as a heat sink throughout the course of the scenario. Succeeding top events address whether the operator can prevent subsequent MSIV closure due to lowering RPV pressure. Failure means that the MSIVs are closed and the condenser is isolated from the RCS.

AI - Operator Inhibits ADS

Success at this top event implies that the operator has inhibited ADS early in the ATWS sequence; thereby, preventing automatic RPV depressurization. The likelihood that the operator will inhibit ADS and avoid rapid vessel depressurization is sequence dependent. On the other hand, the current version of the EOPs unilaterally instruct the operator to inhibit ADS before system conditions develop that would challenge the automatic function. Consequently, failure of the operator to satisfactorily perform this action directly influences the assessment of subsequent operator actions included in the event tree model.

OE - Operator Depressurizes the RPV

There are a large number of symptoms that will direct the operator to depressurize the primary system. Some of these symptoms include:

- high drywell temperature
- high containment pressure
- low RPV water level
- high suppression pool temperature

The need for, and ability of, the operator to correctly depressurize the RPV varies significantly with sequence:

- For turbine trip initiators with SLC actuation and RPV level control, and the main condenser available, emergency depressurization should not be required.

- For cases with SRVs stuck open or MSIV closure, the suppression pool temperature would be the first symptom which would cause the operator to depressurize the RPV. The procedure requires all injection systems (except SLC, CRD, and RCIC) to be shutoff and flow to the vessel minimized during depressurization. Following depressurization, a coolant injection source (i.e., condensate, CRD and RCIC) is used to slowly refill the RPV and control water level above the TAF.

- RPV pressure
- RPV level
- Drywell temperature
- Reactor building temperature

Success at this top event implies that accurate indication of RPV water level would prevent the operator from maintaining level significantly below the MSC and subsequently inducing core damage, believing that level is being controlled per the EOPs.

RH - Containment Heat Removal

The ability to remove heat from containment is a function which is necessary to avoid the release of radioactive material to the environment. The systems available to fulfill this function are:

- The normal heat removal path to the condenser, and
- the RHR heat exchangers.

For turbine trip events in which the turbine bypass valves are open and the condenser available, the heat removal capability is more than adequate for the initial plant power conditions of 25% (i.e., low power initiators). However, regardless of the sequence, if SLC can be injected in a timely manner, ADS inhibited, and water level controlled, then initiation of one train of the RHR system will provide adequate containment heat removal capability and prevent high containment pressure. Therefore, for turbine trip initiated sequences from low power, even with a failure to scram, the ability to remove heat from containment has a very high reliability. Conversely, for high power scenarios, the capability to establish adequate decay heat removal is a function of the core power and heat removal system capacity. Core power (i.e., equivalent steam flow to containment) is dependent upon:

- SLC initiation,
- RPV pressure control, and
- RPV level control

The maximum capacity of the heat removal systems is limited to approximately 29% steam flow. The turbine bypass to the condenser capacity is approximately 25% and the capacity of the two RHR loops is approximately 4%.

For this analysis, this node is used as a switch to denote the availability of either the condenser or one of two RHR trains in the suppression pool cooling mode. Either capability is considered to have sufficient heat removal capacity to maintain acceptable containment conditions in the case where boron injection is accomplished early in the sequence.

CV - Containment Venting

Up to this point in the model, the operator has been successful in controlling the ATWS scenario except for establishing containment heat removal. The containment venting system will be necessary to control containment pressure earlier in an ATWS sequence than for a general transient. Otherwise, the success criteria describing the requirements for successful implementation (i.e., in terms of capability) are the same as for a transient initiated event without ATWS.

CI - Continued Injection at High Containment Pressure

Refer to top event description provided for the transient model in Section 3.1.2.1.

CF - Continued Injection After Containment Failure

Refer to top event description provided for the transient model in Section 3.1.2.1.

- For cases with failures of the high pressure injection systems, the reactor water level may fall below the Level 1 MSIV closure and ADS setpoint. In such cases, the time available to the operator to diagnose the condition and take the correct action is limited to between 5 and 20 minutes, depending upon the drywell pressure, the time of high pressure injection failure, and the rate of reactor water level decrease.

The procedure requires the operator to terminate and prevent injection systems (except SLC and CRD) before emergency depressurizing the RPV. Following depressurization, a low pressure injection source is used to slowly refill the RPV and control water level above the TAF(-45 inches), or -45 inches if level cannot be maintained above TAF.

These three cases describe various operator action models with varying time, stress, and other factors affecting operator performance. However, this ATWS model conservatively assumes that depressurization is always required (i.e., HCTL exceeded or low RPV level) because this requires additional operator actions.

IL - Operator Overrides RCIC Low RPV Pressure Trip

This top event considers the possibility that the operator successfully bypasses the RCIC RPV low pressure trip before the RPV is depressurized as prescribed in the EOPs. However, because success at this node implies that RCIC remains operational throughout the sequence and provides sufficient RPV inventory makeup, the additional failure mode of whether the turbine high exhaust backpressure trip disables RCIC at this point in the scenario is also considered. This top event is assumed to fail if the condenser is unavailable (top event MS, CN or MO failure) since a high exhaust backpressure trip occurs.

CH - RPV Level Not Controlled High

This top event considers the possibility that uncontrolled injection of low pressure makeup to the RPV post depressurization dilutes the core inventory of boron, or flushes boron out of the vessel via the SRVs, causing recriticality. The level/power control contingency of the EOPs instruct the operator to prevent all low pressure ECCS and HPCS before emergency RPV depressurization occurs. It is assumed that if the operator has control of the situation, low pressure makeup can be supplied to the vessel post blowdown and avoid overfilling the RPV. Conversely, if the operator fails to inhibit ADS earlier in the sequence, it is postulated that level control is not possible upon the unexpected actuation of the ADS.

HS - HPCS Operation, LS - LPCS Operation, LC - LPCI C Operation

Refer to the transient event tree description of these top events in Section 3.1.2.1.

The level/power control contingency of the EOPs instruct the operator to avoid using the HPCS, LPCS and LPCI systems for vessel makeup under all conditions during an ATWS, except when coolant inventory cannot be maintained above -45 inches. However, the intentional disabling of the automatic initiation of these makeup systems per the EOPs (refer to top event CH) is not expected to significantly affect the ability of the operator to restore vessel coolant inventory in this situation (i.e., it is assumed that the operator is diligent in monitoring RPV water level given that the decision to emergency depressurize the RPV and prevent subsequent ECCS injection is accomplished). Failure to establish either one of these ECCS systems implies that the operator is unable to maintain and control RPV water level above 1/3 core height.

WL - Water Level Greater Than 1/3 Core Height

Fuel zone level monitors are used when reducing vessel water level during an ATWS event. The EOPs call on the operators to control water level no lower than top of active fuel (TAF).

GE's Nuclear Boiler Spec 761E445AF, sheet 1 requires the fuel zone level instruments to be calibrated for saturated water steam conditions at 0 psig in the vessel and the drywell with no jet pump flow. During an ATWS event, it is assumed that the recirculation pumps have tripped causing reactor power to decrease to approximately 47% with core flow of approximately 29%. The coolant temperature is approximately 545 F. These factors affect the indicated fuel zone vessel level.

First, the temperature (density) effect results in the indicated level reading approximately 40 inches lower than actual in the region near TAF. If the plant operators used this monitor to lower RPV water level below TAF, the actual water level would be 40 inches above the desired level.

Conversely, an increase in fuel zone transmitter sensing line pressure is caused by a power induced convective flow of approximately 29% rated core flow (FSAR Section 4.4, Figure 4.4-1). This causes the indicated level to be higher than actual level. Because these fuel zone level transmitters are calibrated at zero jet pump flow, this unquantified error caused by jet pump flow results in the indicated level reading higher than actual level. If the recirculation pumps are not tripped, the core flow is much higher than in the case above and the error in the indicated level increases even more.

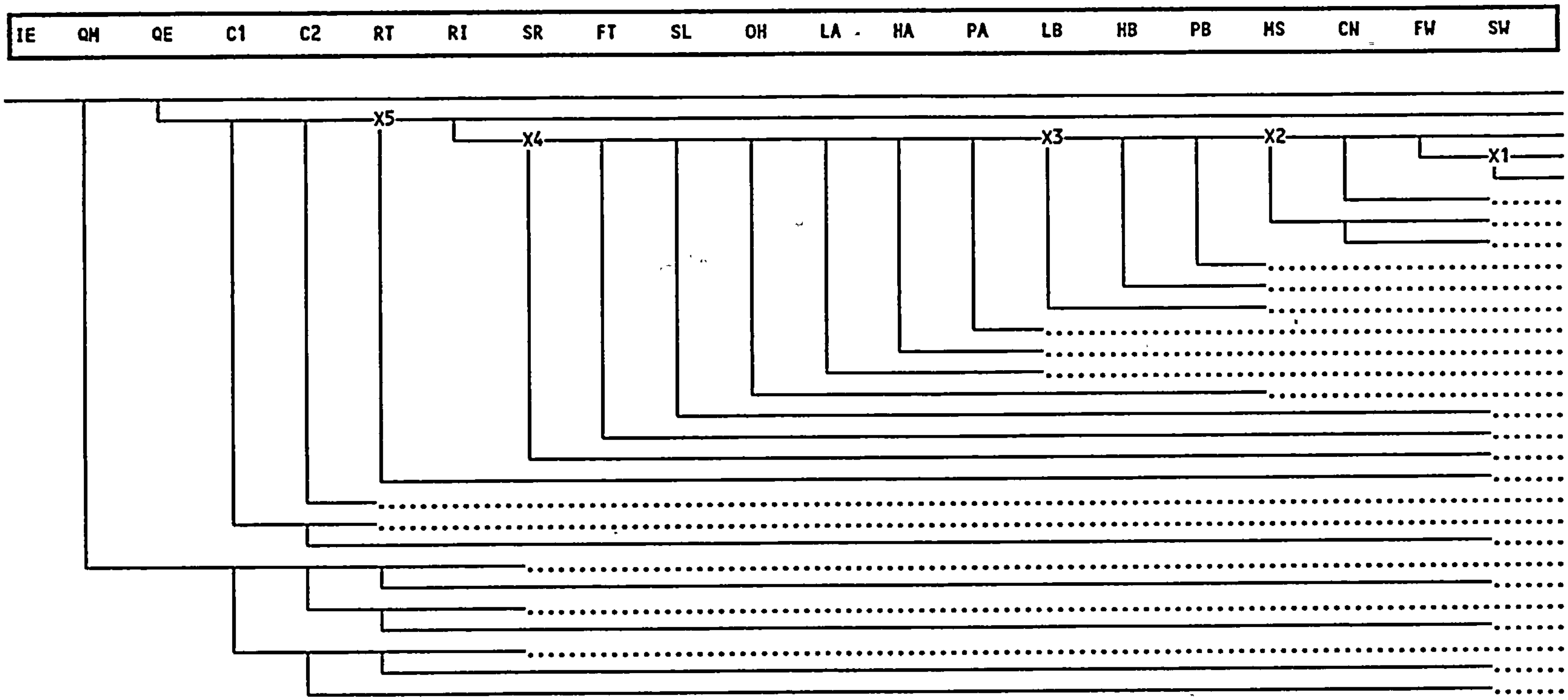
This event considers the possibility that the operator controls RPV water level too low given that the EOP instructions are followed for minimizing reactor power. Under certain situations, the operator is instructed to maintain RPV water level as low as possible, but above -14 inches with reference to the instrument zero (TAF). Therefore, it is possible that the operating crew could be in a situation where RPV water level is below the TAF since the operator is relying on fuel zone level instrumentation to control water inventory.

This top event accounts for the possibility that either the operator fails to control water level above the minimum steam cooling (MSC) level, or the more likely case of the operator establishing a water level indicated as the TAF, but due to level instrument error, is actually lower in the fuel zone region and below the MSC. The level instrument error can be exacerbated by any of the following factors.

- Instrument calibration conditions
- ECCS injection flowrate
- Recirculation pump flow

Figure 3.2.1.6-1
ATWS Event Tree

MODEL Name: NMP2
Event Tree: AT1



1		1
2		2
3		3
4		4
5		5
6	X1	6-7
7	X1	8-9
8	X1	10-11
9	X2	12-20
10	X2	21-29
11	X2	30-38
12	X3	39-74
13	X3	75-110
14	X3	111-146
15	X2	147-155
16	X1	156-157
17	X1	158-159
18	X1	160-161
19	X1	162-163
20	X5	164-325
21	X5	326-487
22	X1	488-489
23	X4	490-648
24	X1	649-650
25	X4	651-809
26	X1	810-811
27	X4	812-970
28	X1	971-972
29	X1	973-974

Figure 3.1.2.6-1
ATWS Event Tree

MODEL Name: NMP2
Top Event Legend for Tree: AT1

Top Event Designator	Top Event Description
IE	INITIATING EVENT
QM	REACTOR SCRAM - MECHANICAL
QE	REACTOR SCRAM - ELECTRICAL
C1	DIV I REDUNDANT REACTIVITY CONTROL
C2	DIV II REDUNDANT REACTIVITY CONTROL
RT	RECIRCULATION PUMP TRIP
RI	ALTERNATE ROD INSERTION
SR	ADEQUATE PRESSURE CONTROL
FT	FEEDWATER RUNBACK
SL	STANBY LIQUID CONTROL SYSTEM
OH	OPERATOR ALIGNS SUPPRESSION POOL COOLING
LA	RHR PUMP TRAIN A
HA	RHR HEAT EXCHANGER A
PA	SUPPRESSION POOL INJECTION A
LB	RHR PUMP TRAIN B
HB	RHR HEAT EXCHANGER B
PB	SUPPRESSION POOL INJECTION B
MS	OPERATOR PUTS MODE SWITCH IN SHUTDOWN
CN	CONDENSER AVAILABLE
FW	CONDENSATE & FEEDWATER AVAILABLE
SW	SERVICE WATER CROSSTIE TO RHR

Figure 3.1.2.6-1
ATWS Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: AT1

SF	Split Fraction Logic
QM1	1
QE1	1
C1F	D1=F
C11	1
C2F	D2=F
C21	1
RIF	(D1=F+C1=F)*(D2=F+C2=F)
RI1	D1=S*D2=S*C1=S*C2=S
RI2	D1=F+D2=F+C1=F+C2=F
RIF	1
RTF	(D1=F+C1=F)*(D2=F+C2=F)
RT1	D1=S*D2=S*C1=S*C2=S
RT2	D1=F+D2=F+C1=F+C2=F
RTF	1
FTS	INIT=ALOSP + INIT=ALOF + INIT=AMSIV + INIT=ALOC
FT1	1
SLF	A1=F+A2=F
SL1	1
SR1	1
OH2	1
LAF	ACA + D1=F + MA=F
LA1	1
LBF	ACB + D2=F + MB=F
LB1	LA=S
LBA	LA=F
LBF	1
HAF	SA=F
HA1	1
HBB	SA=F
HBF	SB=F
HB1	-HA=F
HBA	HA=F

Figure 3.1.2.6-1
ATWS Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: AT1

SF	Split Fraction Logic
HBF	1
PA1	1
PB1	-PA=F
PBA	PA=F
PBF	1
MS1	1
CNF	MS=F
CNF	INIT=ALOSP + INIT=ALOC
CNF	NA=F+NB=F+AS=F+N2=F+TW=F
CN3	INIT=AMSIV
CN1	1
FWF	INIT=ALOSP + INIT=ALOC + INIT=AMSIV + INIT=ALOF
FWF	TW=F+AS=F+TA=F+TB=F+NA=F*NB=F
FW3	1
SWF	SWGF
SW2	1

SWGF:= SB=F + ACB + UB=F

Figure 3.1.2.6-2
ATWS Event Tree

MODEL Name: NMP2
Event Tree: AT2

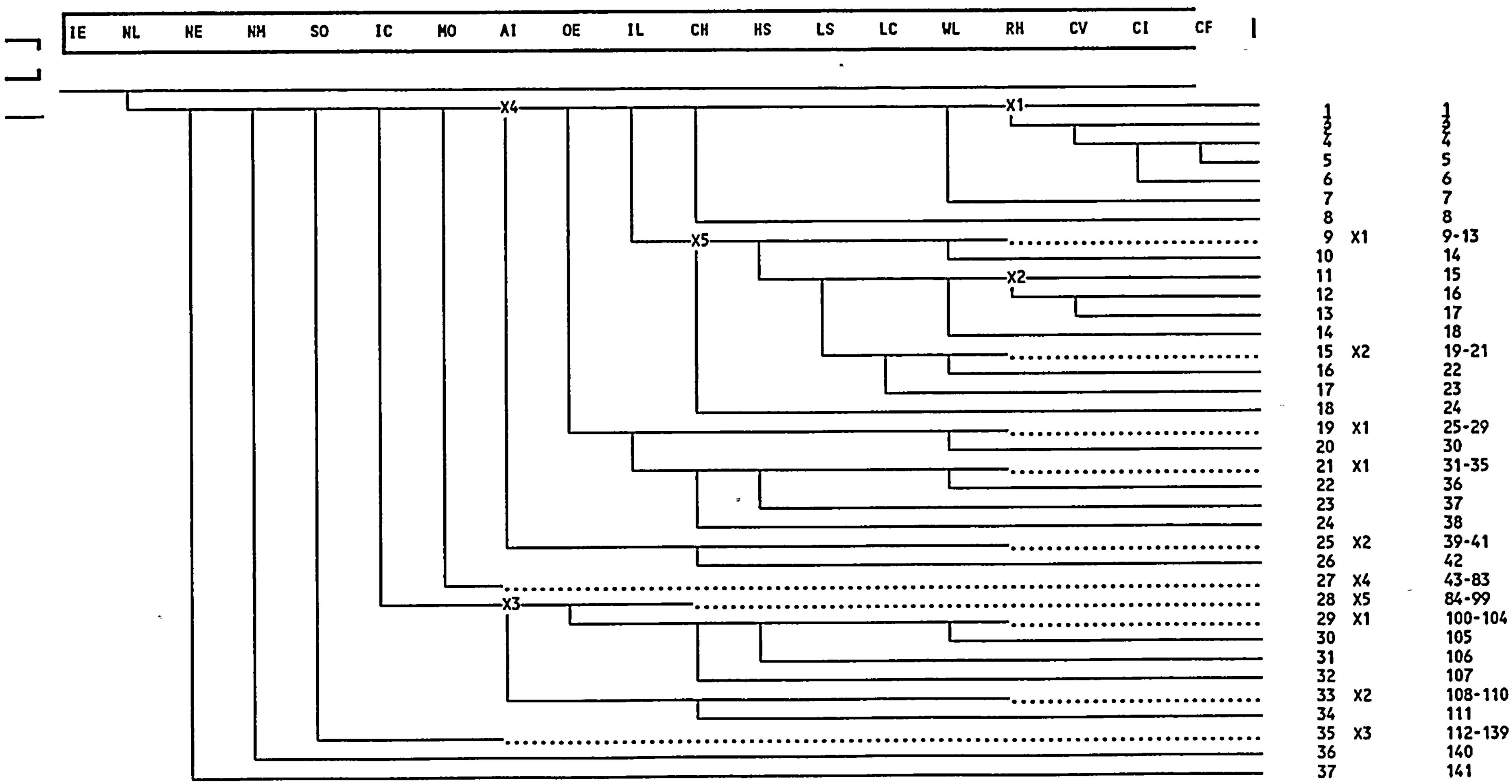


Table 3.1.2.6-2
ATWS Event Tree

MODEL Name: NMP2
Top Event Legend for Tree: AT2

Top Event Designator	Top Event Description
IE	INITIATING EVENT
NL	SUCCESS IN AT1- NO ATWS (TRANSIENT)
NE	NO CATESTROPHIC VESSEL FAILURE
NM	NO SLC FAILURE
SO	STUCK OPEN RELIEF VALVE
IC	RCIC
MO	OPERATOR OVERRIDES LOW LEVEL MSIV CLOSURE
AI	OPERATOR INHIBITS ADS
OE	OPERATOR EMERGENCY DEPRESSURIZES RPV
IL	OPERATOR OVERRIDES RCIC TRIP
CH	RPV LEVEL CONTROL NOT HIGH
HS	HPCS
LS	LPCS
LC	LPCI C
WL	RPV WATER LEVEL > 1/3 CORE HEIGHT
RH	RHR AVAILABLE
CV	CONTAINMENT VENTING
CI	CONTINUED INJECTION AT HIGH CONTAINMENT PRESSURE
CF	CONTINUED INJECTION AFTER CONTAINMENT FAILURE

Table 3.1.2.6-2
ATWS Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: AT2

SF	Split Fraction Logic
NLS	QM=S*(QE=S + RI=S) + SL=S*FW=S
NLF	1
NEF	RT=F + C1=F*C2=F + FT=F + SR=F
NES	1
NMF	SL=F
NMS	1
SOF	INIT=AIORV
SO1	1
ICF	D1=F + TA=F + TB=F + E1=F*E2=F + UA=F + SO=F + UB=F
IC1	E1=S*E2=S
IC2	E1=F + E2=F
ICF	1
MO1	CN=S
MOF	1
AI1	IC=S
AI2	IC=F + SO=F Rule Comment NO HP MAKEUP EXCEPT CRD & SLC
AIF	1
OEF	D1=F*D2=F + ACA*ACB
OE1	-IC=S*-ACA*-ACB*D1=S*D2=S*N2=S
OE2	-IC=S*-ACA*-ACB*D1=F*D2=S
OE3	-IC=S*-ACA*-ACB*D1=S*D2=F*N2=S
OE9	-IC=S*-ACA*-ACB*D1=S*D2=F*N2=F
OE4	-IC=S*-ACA*-ACB*D1=S*D2=S*N2=F
OE5	-IC=S*ACA*-ACB*D1=S*D2=S
OE6	-IC=S*ACA*-ACB*(D1=F+D2=F)
OE7	-IC=S*-ACA*ACB*D1=S*D2=S
OE8	-IC=S*-ACA*ACB*(D1=F+D2=F)
OES	1
ILF	MO=F Rule Comment GUARANTEED RCIC HIGH EXHAUST BACK PRESS TRIP

Table 3.1.2.6-2
ATWS Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: AT2

SF	Split Fraction Logic
ILS	MO=S*OE=F Rule Comment GUARANTEED NO HIGH CONT BACK PRESS & NO RPV LOW PRESS TRIPS FAIL RCIC
IL1	OE=S Rule Comment OPERATOR OVERRIDES LOW LEVEL MSIV CLOSURE INTERLOCK (i.e. MO=S)
ILF	1
CHF	AI=F
CH1	OE=S Rule Comment TERMINATE & PREVENT ALL ECCS, THEN REESTABLISH MAKEUP AFTER DEPRESS
CH2	OE=F Rule Comment INITIATE HPCS GIVEN IT WAS SECURED PER PROCEDURE
CHF	1
HSF	SA=F*SB=F + TB=F
HS1	KA=S*SA=S*SB=S
HS3	KA=S*(SA=F+SB=F)
HS2	KA=F*SA=S*SB=S
HS4	KA=F*(SA=F+SB=F)
HSF	1
LSF	ACA + D1=F + E1=F + UA=F + MA=F
LS1	1
LCF	ACB + D2=F + E2=F + UB=F + MB=F
LC1	1
WL1	IL=S
WL2	1
RHS	RHR + MO=S*OE=F*WL=S Rule Comment NO EOP INSTRUCTION TO BYPASS LOW STM HDR PRESS OR HIGH RAD MSIV CLOSURE
RHF	1
CVF	ACB + N2=F*AS=F
CV1	-ACA*AS=S*N2=S
CV2	(ACA + AS=F)*N2=S

Table 3.1.2.6-2
ATWS Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: AT2

SF	Split Fraction Logic
ILS	MO=S*OE=F Rule Comment GUARANTEED NO HIGH CONT BACK PRESS & NO RPV LOW PRESS TRIPS FAIL RCIC
IL1	OE=S Rule Comment OPERATOR OVERRIDES LOW LEVEL MSIV CLOSURE INTERLOCK (i.e. MO=S)
ILF	1
CHF	AI=F
CH1	OE=S Rule Comment TERMINATE & PREVENT ALL ECCS, THEN REESTABLISH MAKEUP AFTER DEPRESS
CH2	OE=F Rule Comment INITIATE HPCS GIVEN IT WAS SECURED PER PROCEDURE
CHF	1
HSF	SA=F*SB=F + TB=F
HS1	KA=S*SA=S*SB=S
HS3	KA=S*(SA=F+SB=F)
HS2	KA=F*SA=S*SB=S
HS4	KA=F*(SA=F+SB=F)
HSF	1
LSF	ACA + D1=F + E1=F + UA=F + MA=F
LS1	1
LCF	ACB + D2=F + E2=F + UB=F + MB=F
LC1	1
WL1	IL=S
WL2	1
RHS	RHR + MO=S*OE=F*WL=S Rule Comment NO EOP INSTRUCTION TO BYPASS LOW STM HDR PRESS OR HIGH RAD MSIV CLOSURE
RHF	1
CVF	ACB + N2=F*AS=F
CV1	-ACA*AS=S*N2=S
CV2	(ACA + AS=F)*N2=S

Table 3.1.2.6-2
ATWS Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: AT2

SF	Split Fraction Logic
CV5	(ACA + N2=F)*AS=S
CVF	1
CIS	HS=S
CI1	IL=S*KA=S Rule Comment OPERATOR ALIGNS HPCS WHEN NOT PREVIOUSLY ASKED
CI2	IL=S*KA=F
CIF	1
CFF	TA=F + TB=F
CF2	RW=S*NA=S*NB=S*-HS=F
CF4	(RW=F + NA=F + NB=F)*-HS=F
CFF	1

RHR1:= OH=S*LA=S*HA=S*PA=S*LB=S*HB=S*PB=S

RHR:= OH=S*(LA=S*HA=S*PA=S + LB=S*HB=S*PB=S)

SPRAY:= QM=B

LOW:= -HIGH

HIGH:= OE=F + CI=F + CF=F + (HS=F + AI=F*CV=F) + CH=F + NM=F

NOSV:= QM=B

CLASSIVL:= NE=F

CLASSIVA:= NM=F + CH=F + CI=F + CF=F + CV=F*(IL=F*HS=F + IC=F*HS=F + AI=F)

SUCCESS:= NL=S + CV=S + RH=S + CF=S

HPI:= FW=S + IL=S + HS=S

LPI:= LS=S + LC=S + HS=S

SWRHR:= SW=S

FPRHR:= QM=B

SPBYP:= QM=B

CM30:= QM=B

CM2:= QM=B

CM8:= QM=B

CM10:= QM=B

CM19:= QM=B

CLASSIA:= QM=B

Table 3.1.2.6-2
ATWS Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: AT2

SF	Split Fraction Logic
CLASSIB:=	QM=B
CLASSIC:=	LC=F + OE=F*(HS=F+CH=F) + WL=F
CLASSID:=	QM=B
CLASSIIA:=	QM=B
CLASSIIT:=	QM=B
CLASSIIL:=	QM=B
CLASSIIIB:=	QM=B
CLASSIIIC:=	QM=B
CLASSIIID:=	QM=B
CLASSIV:=	CLASSIVA + CLASSIVL

Table 3.1.2.6-2
ATWS Event Tree

MODEL Name: NMP2
Binning Logic for Event Tree: AT2

Bin	Binning Rules
SUCCESS	SUCCESS
CLASSIC	CLASSIC
CLASSIVA	CLASSIVA
CLASSIVL	CLASSIVL
DEFAULT	1

3.1.3 Special Event Trees

3.1.3.1 Interfacing Systems LOCA (ISLOCA) Evaluation

This accident sequence analysis evaluates operator and system response to an Interfacing Systems LOCA (ISLOCA) initiating event. An ISLOCA is initiated by failure of valves that isolate the reactor coolant pressure boundary from the low pressure ECCSs during normal operation. These scenarios can be important to risk because a breach in the ECCSs outside containment can not only lead to failure of the system required to mitigate the LOCA event, but also provide a containment bypass pathway for radionuclide release in the case of core damage.

Previously performed Probabilistic Risk Assessments (PRAs) of BWR reactor plants have identified the possibility of ISLOCA related accident sequences that can result in early core damage and bypass of the containment. These postulated accident scenarios have been estimated to occur at low frequencies. Nevertheless, there have been industry precursors or related events that indicate that the frequencies may not be very low for all plants. Consequently, in the 1980-1981 time frame, there were a number of NRC orders sent out to LWR plants (i.e., 34 PWRs and 2 BWRs) which had similar susceptible Pressure Isolation Valves (PIVs) configurations (i.e., either two check valves or two check valves and an open MOV) directing these utilities to independently leak test these check valves.

Recently, the NRC, under Generic Issue 105, has sponsored two reports by BNL to demonstrate the potential cost-benefit of performing PIV leak testing. The NRC decided, based on analyses performed by BNL, that further study of this issue was required; and therefore, initiated a program with the INEL to perform an ISLOCA examination. This information is used in the investigation of the ISLOCA scenario contribution to the overall risk profile of NMP2.

3.1.3.1.1 Evaluation Overview

The ISLOCA evaluation methodology suggested by EPRI (Ref. 45) was adopted for the performance of the NMP2 assessment, and can be summarized as follows:

1. Develop an event tree sequence diagram which is initiated by the low pressure system overpressurization frequency. The event tree describes the possible accident and core melt progression scenarios that could cause the release of radionuclides bypassing containment and the methods of achieving a stable state.

2. Describe the potential failure modes of PIVs (i.e., initiating events) using a fault tree structure for display and quantification of the low pressure system overpressurization frequency.

3. For each of the event tree top events, use fault trees to display the dominant failure modes that act as a vehicle for quantification.

4. Quantify the results and compare with the IPE reporting criteria.

5. Perform radionuclide release estimates for dominant ISLOCA initiated scenarios contributing to core damage frequency.

Potential contribution to this initiating event frequency from Class II scenarios (e.g. elevated RPV pressure with the low pressure ECCS aligned for coolant makeup), is considered negligible.

3.1.3.1.2 ISLOCA Event Tree Model

The ISLOCA event trees consist of top events that address the following issues:

- Low pressure system overpressurization frequency (Initiating Event)
- Low pressure system failure given overpressurization
  - piping rupture
  - piping leakage
- Immediate isolation of low pressure piping following blowdown
- ECCS makeup capability to the RPV
- Late isolation of low pressure piping

Each of the event tree model top events for the candidate ISLOCA paths are discussed in the following subsections:

- (1) Low Pressure Core Spray discharge line for pump 2CSL*P1
- (3) RHR discharge lines for pumps 2RHS*P1A, B, and C
- (2) RHR Shutdown Cooling return lines to both reactor recirculation loops

The following describes the top events in the event tree. The event tree used is shown in Figure 3.1.3-1.

<u>IE - Initiating Event</u>
The initiating event of an ISLOCA is defined as the coincidental failure of the PIVs separating the RCS pressure boundary from the lower pressure rated piping in an ECCS.

Refer to section 3.1.1 of the main report for additional information concerning the development of the initiating event frequency.

<u>LL - Containment Isolation Breach > 150 gpm</u>
The initiating event fault tree model describes failure modes that can potentially affect PIV integrity and result in loss of containment isolation of the LPCI, SDC, and LPCS systems. A coincidental breach of the inboard and outboard PIVs resulting in less than 150 gpm flow rate into the low pressure rated ECCS piping is considered within the capability of the system to withstand, assuming the proper operation of installed relief valves. This top event

is used to discretize potential ISLOCA scenarios involving minimal leakage of the PIVs from postulated sequences initiated by significant leakage past the PIVs (i.e., in excess of 150 gpm). This split fraction is based on the results of fault tree analysis. Success at top event LL implies that the breach in containment isolation is significant.

ZN - Containment Isolation Breach in the North Auxiliary Bay

Top event ZN is used to distinguish (based on the fault tree analysis for determining the total ISLOCA initiating event frequency) the location of the ISLOCA as being in either the LPCS system or the RHR A system (i.e., LPCI or SDC path), located in the north Auxiliary Bay (i.e., ZN = S). Otherwise, if the system breach is in the south Auxiliary Bay (i.e., ZN=F), the failure of containment isolation PIVs is assumed to have occurred in either the RHR system train B (i.e., either in the LPCI or SDC path) or C.

ZS - Breach in the LPCS or LPCI C System

This top event tree is used to distinguish (again, based on the fault tree analysis for determining the total ISLOCA initiating event frequency), the location of the ISLOCA in either the LPCS or RHR C trains versus the RHR A or B lines (i.e., if ZS=S, then PIV breach is assumed to have occurred in either LPCS or LPCI C; whereas, if ZS=F, then containment isolation failure is defined as being in either RHR train A or B).

ZC - System Breach in RHR SDC Train A or B

This is the final top event in the model used to locate the ECCS in which the containment isolation breach occurred, and describe the magnitude of the failure. Specifically, this node discriminates whether containment isolation occurred in either the RHR shutdown cooling return lines to both reactor recirculation loops versus the RHR LPCI discharge lines (i.e., if ZC=S, then ISLOCA is in either SDC train A or B; whereas, if ZC=F, then ISLOCA is in either LPCI train A or B).

The event tree paths, as defined by nodes LL through ZC, discretize ISLOCA scenarios among the six susceptible ECCS paths, and whether containment isolation breach in these system discharge headers is significant (i.e., > 150 gpm). Subsequent event nodes describe the potential failure of the low pressure system piping, and the capability of operator to maintain RPV inventory and recover from the ISLOCA by isolating the breach.

ZR & ZL - Low Pressure System Integrity

Top events ZR (Low Pressure System does not Rupture) and ZL (Low Pressure System does not Leak) assess whether the low pressure rated system can withstand exposure to normal RCS operating pressure and maintain its integrity.

The ability to maintain system integrity is affected by the design capacity of the system relief valves and the characteristics of the low pressure system piping and in-line components (e.g., flanges, gaskets, seals, bolts).

Following a postulated breach of the pressure isolation interface, the low pressure piping system is assumed to become overpressurized. (Note that system leakage less than 150 gpm is assumed to be within the design capacity of the installed system relief valves in the LPCS and RHR systems; and therefore, these scenarios are judged to be easily mitigated by the operating crew.) For the case of interface leakage greater than 150 gpm, mitigative actions can be taken to prevent the low pressure system from failing; and thereby, avoid subsequent release of the primary coolant outside containment.

Failure of these measures could result in overpressurization and subsequent breach in the ECCS. The probability of system piping rupture (i.e., system component failures are defined as leaks) due to static loading is evaluated using the methodology proposed by the NRC in NUREG/CR-5603. The potential for system piping rupture due to dynamic loading can also occur if the low pressure system is not maintained "full." However, because these low pressure systems are maintained full of water by using jockey pumps while in standby, dynamic effects due to water hammer are not suspected to be significant system failure modes. This is consistent with previous NRC and industry probabilistic evaluations.

If the low pressure system survives the overpressurization, the increased pressure and high reactor temperature could affect the integrity of pump seals, and low pressure rated valves and flanges. Based on NRC evaluation results (as published in NUREG/CR-5603), flanges, valves, and pump seals are considered less fragile than piping. However, if the low pressure system does not rupture, the overpressurization (i.e., greater than the relief capacity in the system), may eventually cause flanges, pump seals, or valves in the low pressure system to leak. The event tree model in this case assumes that system leakage does occur (i.e., if ZR=S, then ZL=F), requiring mitigative actions by the operating crew.

The course of the scenario as defined in these two top events determines the location and the magnitude of the ISLOCA event. For instance, the first four top events define the characteristics of the PIV breach as being either leakage or rupture; whereas, these two top events define the extent of the breach in the low pressure system resulting in a potential containment bypass pathway. However, as far as assessing the crew's capability to mitigate the ISLOCA event, only one combination of PIV and system failures result in an equivalent rupture in the primary system that can challenge the reactor plant and operating crew similar to that postulated for a large break LOCA. This scenario is initiated by a rupture failure of the PIVs and the subsequent rupture failure of the affected low pressure system piping. All other scenarios are assumed to lead to a leakage ISLOCA event that are presumed to affect the primary system similar to a medium break LOCA.

<u>ZI - Isolation Valve Failure is Not Recoverable</u>

This top event models whether the failure mode of the containment isolation MOV is potentially recoverable (e.g., inadvertent mispositioning of the valve during the performance of surveillance of the MOV), and available to the operating crew to isolate the ISLOCA breach either early or late in the scenario.

For instance, if the containment isolation is breached due to mechanical failure of the MOV (i.e., ZI=S), it is assumed that the valve cannot be reclosed to isolate the ISLOCA for the duration of the scenario. Otherwise, if the.cause of the MOV opening is human error or other recoverable failure modes (i.e., ZI=F), the opportunity is available for the operator to isolate the ISLOCA by reclosing the MOV remotely .

Essentially, the split fractions assigned to this event are derived based on inspection of results of the fault tree analysis concerning containment isolation failure (i.e., the split fractions are estimated based on the proportional contribution of human vs. mechanical failure mode contribution to the particular initiating event frequency). These values represent the percentage of scenarios resulting in PIV breach that could subsequently be recovered immediately by the operator.

ZV - System Breach can be Isolated by a Second MOV
LPCI A and B, and SDC A and B return pathways contain an additional remotely operable MOV that could be employed by the operating crew to isolate the ISLOCA given the breach in the affected system is downstream of this valve. Success at this top event implies that the system in question contains another MOV.

ZE - Early Isolation of Low Pressure System Breach
There are numerous indications for determining an ISLOCA-type breach in a low pressure ECCS available to the operators in the control room. These indications include:

- Reactor Building high temperature
- Reactor Building high pressure differential
- Reactor Building high local radiation
- Reactor Building high water level
- Reactor Building high HVAC exhaust radiation
- Refuel Floor high HVAC exhaust radiation
- Interfacing System high pressure.

In addition, the EOPs direct the operating staff to control the secondary containment conditions within a prescribed envelop as defined in N2-EOP-SC, RR.

Based on the operator's ability to recognize that any of the Reactor Building parameters have exceeded threshold values described in the EOP and primary coolant is discharging into, or accumulating in, the compartment, the operator is directed to isolate all systems discharging into the affected area except those that are required to assure safe shutdown, adequate core cooling, and containment integrity.

The capability of the operator to isolate a breach in a low pressure system is a strong function of the pathway design (i.e., number and failure mode of the interface valves) and the low pressure system overpressure failure. The failure types for the interface valves and low pressure systems, as they may affect the assessment for the operating crew isolating the system breach, can be summarized as follows:

Case	Pressure Isolation Valve Failure Modes	Low Pressure System Failure Modes	Characterization of the Rate of Inventory Loss
1	Breach ≤ 150 gpm (LL=F)	System Integrity Assumed (ZR=S and ZL=S)	System leakage ≤ 150 gpm to the Suppression Pool
2	Breach > 150 gpm (LL=S)	Leak (ZL=F)	LOCA Equivalent to Medium Break
3	Breach > 150 gpm (LL=S)	Rupture (ZR=F)	LOCA Equivalent to Large Break

Of the three postulated combined failure modes identified, the one that is most severe in terms of difficulty to isolate is the rupture case (i.e., Case 3), as determined in the quantitative assessment of event ZR.

For scenarios involving the leakage failure of containment isolation PIVs, there may be different effective measures that could mitigate the affect of these failure modes. The effectiveness of these measures is dependent on whether these MOVs can be reclosed to prevent further damage to system piping and components, as considered in node ZI.

The model for this top event is developed to discriminate among leakage and rupture ISLOCA scenarios. For leakage scenarios, the operator has the opportunity to isolate the breach by reclosing the PIVs or another MOV in the system. The air operable, testable check valve in the leakage pathway is assumed conservatively to be ineffective in isolating the release of inventory and is assigned a conditional failure probability of 1.0. On the other hand, the normally open motor-operated valve can potentially provide the isolation function depending on the failure mode that affected it initially. In addition, normally closed MOVs downstream of the containment isolation valves could also be effective in mitigating the ISLOCA. It is conservatively assumed for this evaluation that the operating crew has approximately 15 min. to assess the situation and isolate the breach before core damage results. Of course, this time could be significantly extended if coolant inventory control is established.

For rupture ISLOCA scenarios, normally open MOVs are assumed not to be able to close against differential pressure across the valve during the blowdown; and therefore, the failure probability is conservatively assessed to be 1.0. It should be noted that a rupture ISLOCA is assumed to fully depressurize the RCS within approximately 3-5 minutes. During this time frame the operating crew would be expected to be concerned about establishing RPV inventory control. However, after the plant is depressurized, the operating crew would concentrate on isolating the breach. This second opportunity to restore isolation is further evaluated in node ZT.

Success at top event ZE implies that the operating crew successfully isolated the system upstream of the breach "early" in the scenario.

HS - High Pressure Core Spray (HPCS) Availability

Top event HS provides an assessment of whether the High Pressure Core Spray System is capable of supplying adequate RPV makeup. (Note that the RCIC system is conservatively not included in this assessment.)

The availability of HPCS during postulated leakage ISLOCA scenarios is considered to be high, i.e., equivalent to the model developed for the small break LOCA evaluation. This is because of the compartmentalization of the HPCS and vent capability of the Reactor and Auxiliary Buildings.

If the ISLOCA scenario involves a rupture and is not isolated early, hazardous environmental conditions in the Reactor Building could disable HPCS system equipment. For the time frame of interest, it is assumed that the only, albeit low probability, challenge to the HPCS system is from the high humidity and temperature environment affecting the instrument racks throughout the building.

SV, OD - RPV Depressurization

Another measure to prevent core damage is to depressurize the RPV and provide low pressure makeup. The EOPs clearly direct the operating staff to manually depressurize the RPV under severe environmental conditions in the reactor building or upon low RPV water level. Either of these triggers are assumed to be present during a leakage or rupture scenario. Therefore, the reliability of the depressurization functions is considered to be similar to that previously evaluated for the small or medium LOCA event sequences.

HE, LS, LA, IA, LB, IB, and LC - Low Pressure ECCS Makeup Availability

During an ISLOCA, and if the break is large, the reactor will rapidly depressurize. As RCS pressure decreases, low pressure ECCS systems can inject to the RPV provided the RHR and core spray pumps have not been disabled by potentially severe environmental conditions both in the pump rooms and in the Reactor Building. Essentially, this model uses the same split fractions describing the availability of the LPCI and LPCS systems during large LOCA sequences inside containment. Additionally, dependent failure modes that can potentially affect the availability of several trains of equipment are also evaluated at these nodes. Specifically, top event HE accounts for the possible spatially dependent effects that can fail all ECCSs caused by a severe environment at El. 261' of the Reactor Building during a rupture scenario. Also, dependent failure among trains of ECCSs located in the same Auxiliary Bay is evaluated. This failure mode postulates that sufficient communication exits between adjacent ECCS rooms in the same Auxiliary Bay causing the subsequent failure of both ECCS trains located in these rooms during a rupture ISLOCA event.

Following an ISLOCA, if the coolant makeup cannot be provided using available ECCS, the non-ECCS, such as the cross-connect from main plant service water or control rod drive injection, can be used to provide makeup to prevent core damage. However, the ability of non-ECCSs to inject successfully into the reactor is also dependent on local environment conditions during an ISLOCA and the mitigative measures taken by the operator in response to the accident.

Following failure of ECCS due to severe environmental conditions or lack of mitigative measures, the failure of non-ECCS is judged to be even more likely. Therefore, a conservative conditional failure probability of 1.0 was assigned to non-ECCS makeup capability for both cases of low pressure system leakage and rupture.

ZT - Late Isolation of Low Pressure System Breach

The final top event in the model is the assessment of whether the pathway can be isolated after RPV blowdown. This isolation can result in avoidance of containment bypass event even though core damage may still result.

The only isolation mechanism associated with late isolation of the low pressure system in the pathway is by reclosing remotely operable MOVs after the RPV blowdown within 15 min. following system rupture. Otherwise, for leakage ISLOCA scenarios, core damage and containment bypass can be avoided if either high pressure or low pressure coolant injection can be accomplished within approximately 15 min. from initiation of the event (i.e., success does not require isolation of the system breach if coolant makeup is available to the RPV; the implication being that the operating crew has many hours to shut down the plant and effect repairs).

Success at top event ZT implies that the operating crew isolated the system upstream of the breach within the time frame described above.

Figure 3.1.3-1
ISLOCA Event Tree

MODEL Name: NMP2
Event Tree: ISLOCA

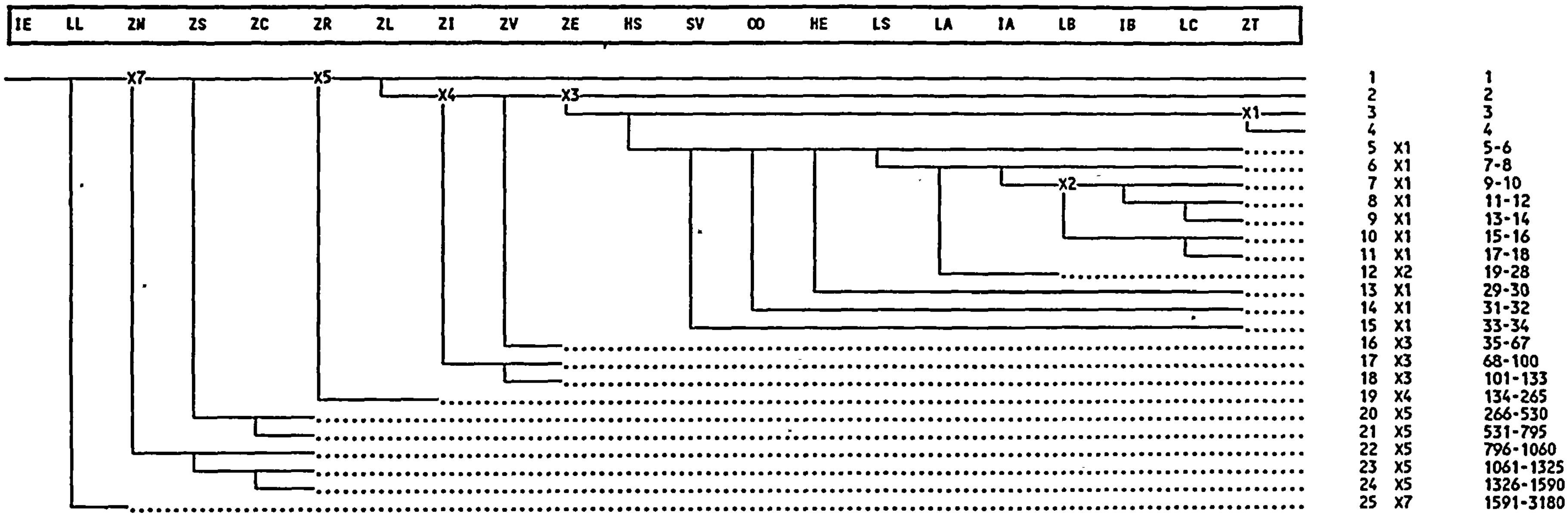


Figure 3.1.3-1
ISLOCA Event Tree

MODEL Name: NMP2
Top Event Legend for Tree: ISLOCA

Top Event Designator	Top Event Description
IE	Initiating Event
LL	CONTAINMENT ISOLATION BREACH > 150 GPM
ZN	CONTAINMENT ISOLATION BREACH IN N AUX BAY
ZS	CONTAINMENT ISOLATION BREACH IN LPCS OR LPCI TRAIN C
ZC	CONTAINMENT ISOLATION BREACH IN RHR SHUTDOWN COOLING TRAIN A OR B
ZR	LOW PRESSURE SYSTEM DOES NOT RUPTURE
ZL	LOW PRESSURE SYSTEM DOES NOT LEAK
ZI	ISOLATION VALVE FAILURE IS NOT RECOVERABLE
ZV	SYSTEM BREACH CAN BE ISOLATED BY A SECOND MOV
ZE	SYSTEM BREACH IS ISOLATED EARLY
HS	HIGH PRESSURE CORE SPRAY
SV	SAFETY RELIEF VALVES - TRAN & SLOCA
OD	OPERATOR DEPRESSURIZES FOR LPI - TRAN & SLOCA
HE	REACTOR BUILDING ENVIRON DOES NOT AFFECT LP ECCS
LS	LOW PRESSURE CORE SPRAY
LA	RHR PUMP 1A TRAIN
IA	LPCI INJECTION PATH A
LB	RHR PUMP 1B TRAIN
IB	LPCI INJECTION PATH B
LC	LPCI TRAIN C
ZT	SYSTEM BREECH IS ISOLATED LATE

Figure 3.1.3-1
ISLOCA Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: ISLOCA

SF	Split Fraction Logic
LL1	1
ZN1	1
ZS1	LL=S
ZS2	LL=F
ZC1	LL=S
ZC2	LL=F
ZRS	LL=F
ZR1	LPCS
ZR2	LPCIA+SDCA
ZR3	LPCIB+SDCB
ZR4	LPCIC
ZLS	LL=F
ZLF	1
ZI2	LL=S*(SDCA+SDCB)
ZI4	LL=S*(LPCIA+LPCIB+LPCIC+LPCS)
ZVF	LPCS+LPCIC
ZVS	1
ZEF	ZR=F Rule Comment ISLOCA RUPTURE ASSUMED UNISOLABLE DURING RPV BLOWDOWN
ZEF	ZI=S*ZV=F Rule Comment NO REMOTE OPERABLE MOVS TO ISOLATE ISLOCA
ZEF	ZI=S*ZV=S*((LPCIA+SDCA)*(A1=F+D1=F)+(LPCIB+SDCB)*(A2=F+D2=F)) Rule Comment 2nd MOV NOT OPERABLE DUE TO LOSS OF POWER
ZEF	ZI=F*ZV=F*(LPCS*(A1=F+D1=F)+LPCIC*(A2=F+D2=F)) Rule Comment ISOLATION MOV NOT OPERABLE DUE TO LOSS OF POWER
ZE1	ZL=F*ZI=S*ZV=S Rule Comment SECOND MOV AVAILABLE TO ISOLATE
ZE2	ZL=F*ZI=F*ZV=S Rule Comment MOV & SECOND MOV AVILABLE TO ISOLATE

Figure 3.1.3-1
ISLOCA Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: ISLOCA

SF	Split Fraction Logic
ZE3	ZL=F*ZI=F*ZV=F Rule Comment MOV AVAILABLE TO ISOLATE
ZEF	1
HSF	SA=F*SB=F + TB=F
HS1	KA=S*SA=S*SB=S*ZL=F
HS3	KA=S*(SA=F+SB=F)*ZL=F
HS2	KA=F*SA=S*SB=S*ZL=F
HS3	KA=S*(SA=F+SB=F)*ZL=F
HS4	KA=F*(SA=F+SB=F)*ZL=F
HS5	ZR=F Rule Comment HAZARDOUS ENVIRONMENT FROM RUPTURE AFFECTS HPCS AUTO START (RACK EL
HSF	1
SVF	D1=F*D2=F + ACA*ACB Rule Comment NO ADVERSE EFFECTS ON ADS FROM HAZARDOUS RB ENVIR
SV1	-ACA*-ACB*D1=S*D2=S*N2=S
SV2	-ACA*-ACB*D1=F*D2=S*N2=S
SV3	-ACA*-ACB*D1=S*D2=F*N2=S
SV4	-ACA*-ACB*D1=S*D2=S*N2=F
SV5	ACA*-ACB*D1=S*D2=S
SV6	ACA*-ACB * (D1=F + D2=F)
SV7	-ACA*ACB*D1=S*D2=S
SV8	-ACA*ACB * (D1=F + D2=F)
SVF	1
OD5	1
HES	ZL=F Rule Comment ISLOCA LEAK
HE1	1 Rule Comment HAZARD ENVIR FROM RUPTURE AFFECTS LPCS AND LPCI (CCF-RACKS EL 261)

Figure 3.1.3-1
ISLOCA Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: ISLOCA

SF	Split Fraction Logic
LSF	ZR=F*(LPCS+LPCIA+SDCA) Rule Comment SPATIAL DEPENDENCY BETWEEN ROOMS
LSF	ACA + D1=F + E1=F*ME=F + UA=F
LS1	1
LAF	ZR=F*(LPCS+LPCIA+SDCA) Rule Comment SPATIAL DEPENDENCY BETWEEN ROOMS
LAF	ACA + D1=F
LA1	1
IAF	ACA + E1=F*ME=F + UA=F
IA1	1
LBF	ZR=F*(LPCIC+LPCIB+SDCB) Rule Comment SPATIAL DEPENDENCY BETWEEN ROOMS
LBF	ACB + D2=F
LB1	1
IBF	ACB + E2=F*ME=F + UB=F
IB1	1
LCF	ZR=F*(LPCIC+LPCIB+SDCB) Rule Comment SPATIAL DEPENDENCY BETWEEN ROOMS
LCF	ACB + D2=F + E2=F*ME=F + UB=F
LC1	1
ZTF	ZR=F * ((LPCIA+SDCA+LPCS)*(A1=F+D1=F)+(LPCIB+SDCB+LPCIC)*(A2=F+D2=F)) Rule Comment ISLOCA RUPTURE AND NO POWER TO MOVs
ZT2	ZR=F*ZI=F Rule Comment RUPTURE, MOV AVAIL TO ISOLATE POST BLOWDOWN (3-15 MIN.)
ZT3	ZR=F*ZV=S Rule Comment RUPTURE, 2nd MOV AVAIL TO ISOLATE ISLOCA POST BLOWDOWN (3-15 MIN.)
ZTF	ZR=F Rule Comment ISLOCA RUPTURES

Figure 3.1.3-1
ISLOCA Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: ISLOCA

SF	Split Fraction Logic
ZTS	ZE=F*(HS=S+LS=S+IA=S+IB=S+LC=S) Rule Comment ISLOCA LEAKS BINNED TO SUCCESS
ZTF	ZV=S*ZE=F*(LC=F+OD=F+SV=F) Rule Comment OPERATOR CAN USE 2nd MOV OR MANUAL VALVES TO ISOLATE LOCALLY W/IN 3
ZTF	ZV=F*ZE=F*(LC=F+OD=F+SV=F) Rule Comment ISLOCA LEAKS W/ NO MANUAL VALVE
ZTF	1

LPCS:= ZN=S*ZS=S
Rule Comment
LPCS DISCHARGE PATH

LPCIA:= ZN=S*ZC=F
Rule Comment
LPCI A DISCHARGE PATH

LPCIB:= ZN=F*ZC=F
Rule Comment
LPCI B DISCHARGE PATH

LPCIC:= ZN=F*ZS=S
Rule Comment
LPCI C DISCHARGE PATH

SDCA:= ZN=S*ZC=S
Rule Comment
RHR A SHUTDOWN COOLING RETURN PATH

SDCB:= ZN=F*ZC=S
Rule Comment
RHR B SHUTDOWN COOLING RETURN PATH

SUCCESS:= ZL=S + ZE=S + (HS=S+LS=S+IA=S+IB=S+LC=S)*ZT=S + CLASSID + CLASSIA

CLASSIA:= (SV=F+OD=F)*ZT=S

CLASSID:= (HE=F + LC=F)*ZT=S

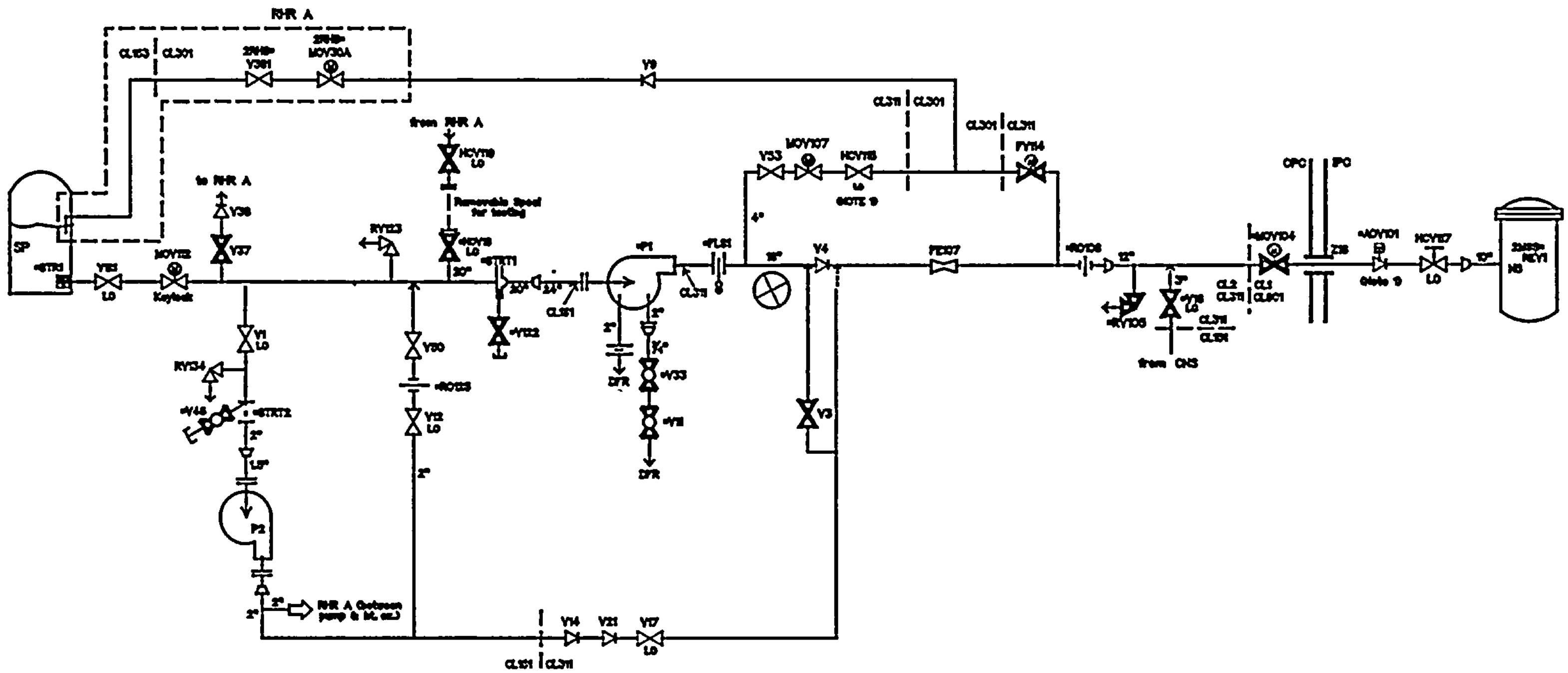
CLASSV:= ZT=F

Figure 3.1.3-1
ISLOCA Event Tree

MODEL Name: NMP2
Binning Logic for Event Tree: ISLOCA

Bin	Binning Rules
SUCCESS	SUCCESS
CLASSV	CLASSV
DEFAULT	1

Rev 0 (7/92)



REFERENCE: PID-32A-8

—— High design pressure and included in ISLOCA model.

—— Pipe class change.

—— Low design pressure.

—— NOT included in ISLOCA model.

⊗ Assumed pipe failure location(s).

Notes: 1. This is an Anchor/Darling testable check valve, there has been an industry wide problem with these valves failing Local Leak Rate Testing (LLRTs).

Figure 3.1.3-2
Low Pressure Core Spray System

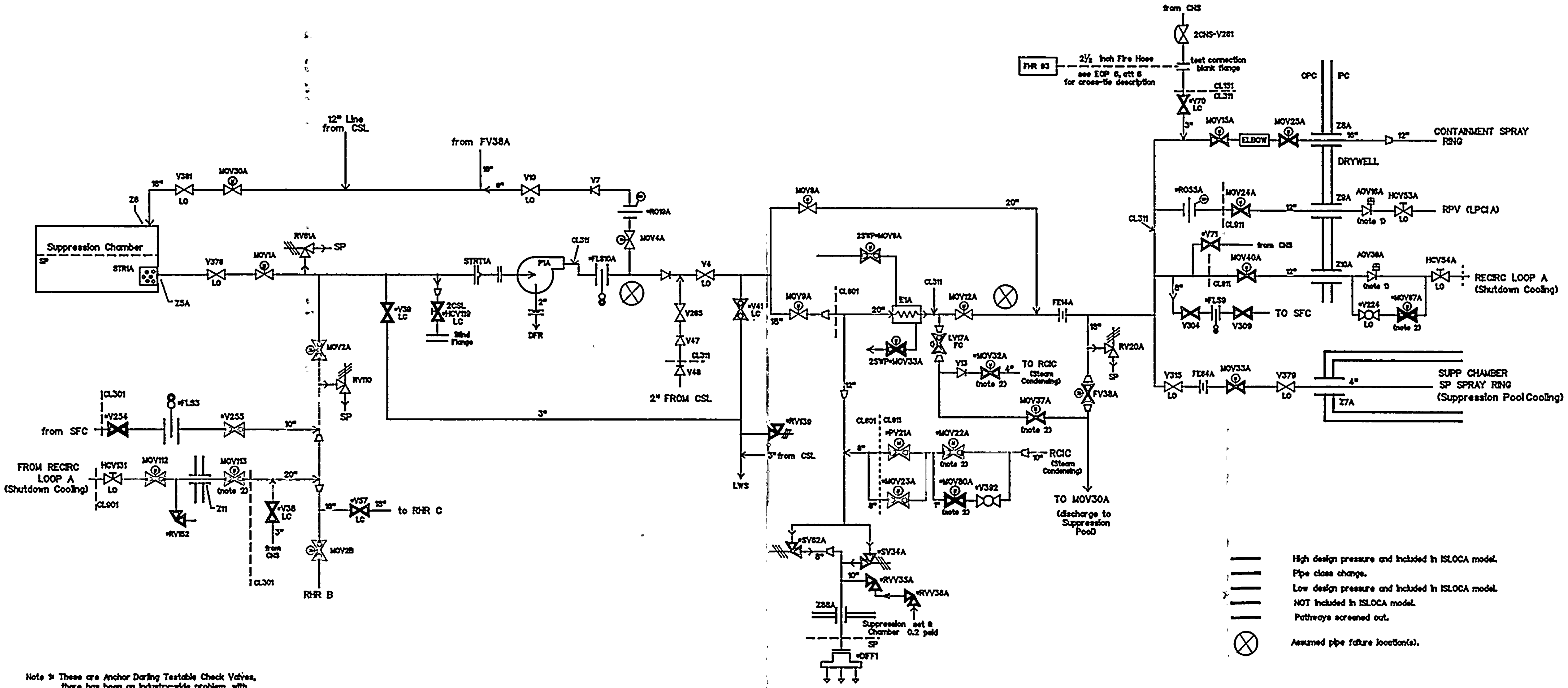
[illegible]

Rev 0 (7/92)

SI APERTURE CARD

Also Available On Aperture Card

References: PID-31A-8
PID-31C-7
PID-31D-9
PID-31F-10



Note 1: These are Anchor Darling Testable Check Valves, there has been an industry-wide problem with these valves failing Local Leak Rate Testing

Note 2: *MOV22A, 32A, 37A, 80A (Steam Condensing) and MOV67A & 113 (Shutdown Cooling) shall be de-energized during normal operation.

Figure 3.1.3-3
Residual Heat Removal "A" Train

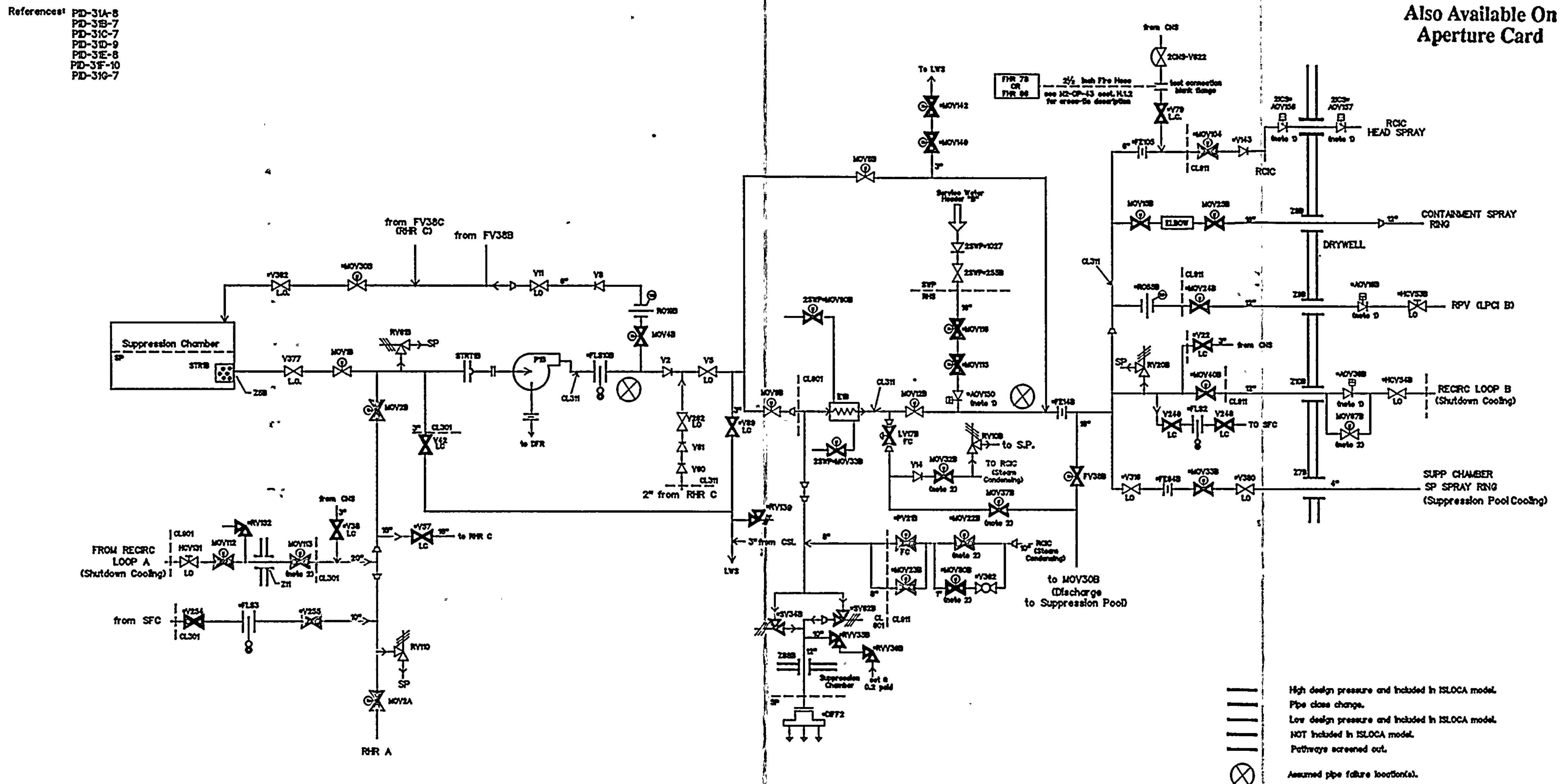
I:\NMP2IPE\DGN\2F313.DGN Jul. 07, 1992 11:28:45

9208050191 - 09

Rev 0 (7/92)

SI APERTURE CARD

Also Available On Aperture Card



Note 1: These are Anchor Darling Testable Check Valves, there has been an industry-wide problem with these valves failing Local Leak Rate Testing

Note 2: *MOV22B, 32B, 37B, 80B (Steam Condensing) and MOV67B & 113 (Shutdown Cooling) shall be de-energized during normal operation.

Figure 3.1.3-4
Residual Heat Removal "B" Train

I:\NMP2IPE\DGN\2F313.DGN Jul. 07, 1992 11:31:45

9208050191 - 10

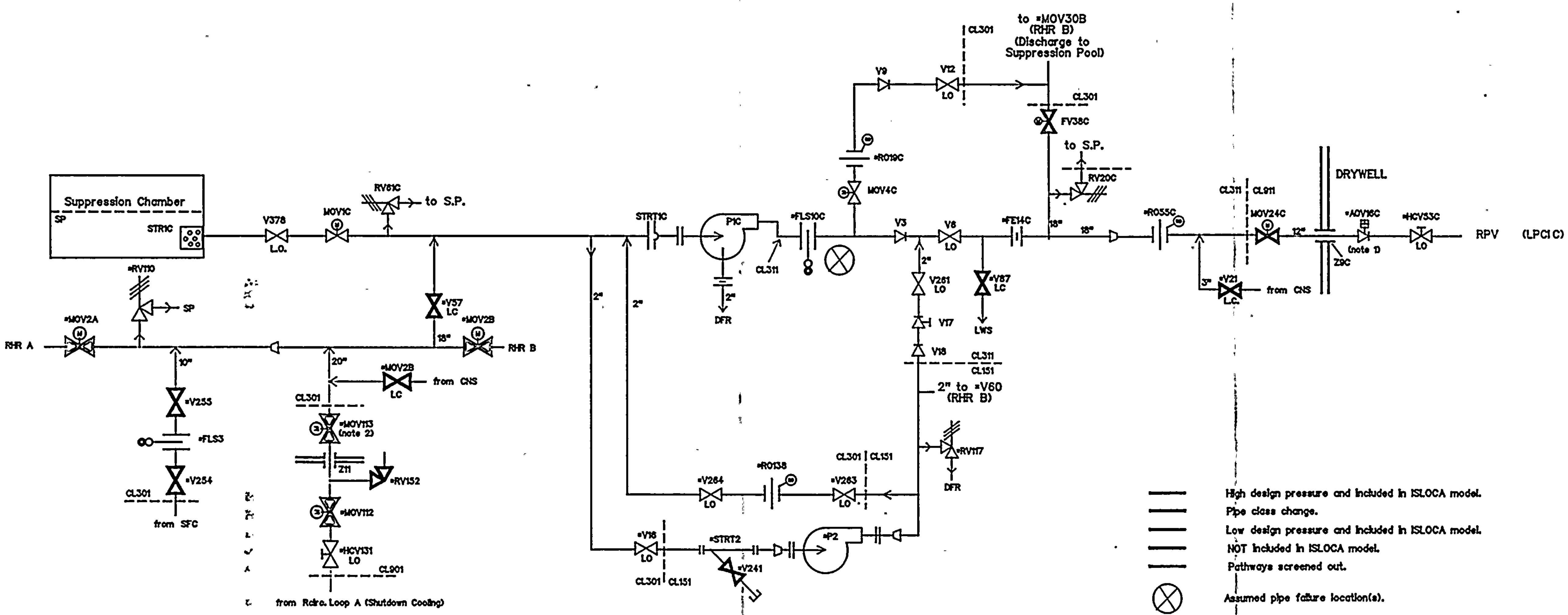
[illegible]

Rev 0 (7/92)

SI APERTURE CARD

Also Available On Aperture Card

References: PID-31A-8
PID-31B-7
PID-31C-7
PID-31F-10
PID-31G-7



Note 1: These are Anchor Darling Testable Check Valves, there has been an industry-wide problem with these valves failing Local Leak Rate Testing

Note 2: *MOV22B, 32B, 37B, 80B (Steam Condensing) and MOV67B & 113 (Shutdown Cooling) shall be de-energized during normal operation.

Figure 3.1.3-5
Residual Heat Removal "C" Train

I:\NMP2IPE\DGN\2F313.DGN Jul. 07, 1992 11:36:30

9208050191 - 11

3.1.4 Support System Event Tree

There are a number of support systems whose failure impacts several front-line systems or functions. These support systems, such as AC power, DC power, ECCS signals, cooling water, and plant air are modeled in the support system event tree. As shown in Figure 3.1.4-1, the support system event tree branches at every top event. The success or failure of the top events are controlled with split fraction rules. For example, the dependency of 115kV Source "A" AC power supply (top event KA) on the availability of offsite AC power (top event OG) is modeled using the following rules.

Split Fraction	Split Fraction Rule
KAF	OG=F + INIT=KAX
KA1	1

which quantitatively sets the failure (split fraction) of top event KA to 1.0 (KAF) if top event OG fails (OG=F) or (+) if the initiating event is loss of 115kV source "A" (INIT=KAX). Otherwise, split fraction KA1 is used which is quantitatively equal to the unavailability of the 115kV Source "A" power supply when offsite power is available.

The support system event tree is linked to the front-line event trees during quantification as shown in Figure 3.1.2-1 and described in Section 3.3.7. The front-line tree top event split fractions are also developed based on their dependency on support systems and the quantification process uses rules as described above. System dependencies are described in Section 3.2.3 and systems analysis summaries are provided in Section 3.2.1.

The following summarizes the top events in the support system event tree:

<u>OG - Offsite AC Grid</u>
Offsite AC power is supplied by the 345kV Scriba/Substation. Failure of this system results in failure of top events KA, KB, NA, and NB (normal AC is failed). Failure also places a demand on the emergency diesels for all three divisions and requires equipment to restart which affects several event tree top events.

<u>KA - 115kV Source A</u>
The 115kV Source A includes the transmission line from the Scriba substation bus "A" at transformer TB1 to Bus 2NNS-SWG016 inside the protected area. This includes connecting switches, circuit breakers, and reserve station service transformer 2RTX-XSR1A. Failure is similar to OG failure except the impact is on emergency AC Division I and III and half of normal AC is failed (NA fails).

<u>KB - 115kV Source B</u>
The 115kV Source B includes the transmission line from the Scriba substation bus "B" at transformer TB2 to Bus 2NNS-SWG017 inside the protected area. This includes connecting switches, circuit breakers, and reserve station service transformer 2RTX-XSR1B. Failure is similar to OG failure except the impact is on emergency AC Division II and half of normal AC is failed (NB fails).

KR - Recovery from a Partial Loss of Offsite Power (PLOSP)
This top event models operator actions required to cross-connect a 115kV source to the opposite emergency switchgear via the auxiliary boiler transformer.

DA - Division I Battery
This top event models availability of the battery on demand which is required when starting large DC equipment and particularly the Div. I diesel generator on loss of offsite power. The failure mode addressed here is from the large, relatively instantaneous, battery drawdown associated with plant upset response. Continued operability of the DC system after the battery demand is modeled in top event D1.

DB - Division II Battery
This top event models availability of the battery on demand which is required when starting large DC equipment and particularly the Div. II diesel generator on loss of offsite power. The failure mode addressed here is from the large, relatively instantaneous, battery drawdown associated with plant upset response. Continued operability of the DC system after the battery demand is modeled in top event D2.

A1 - Division I Emergency AC
This top event includes available power at 2ENS*SWG101, 2EJS*US1, and motor control centers (MCCs). On loss of normal power (OG or KA fails), this also includes operation of the emergency diesel generator (2EGS*EG1) and the opening and closing of associated breakers.

A2 - Division II Emergency AC
This top event includes available power at 2ENS*SWG103, 2EJS*US3, and motor control centers (MCCs). On loss of normal power (OG or KB fails), this also includes operation of the emergency diesel generator (2EGS*EG3) and the opening and closing of associated breakers.

NA - Normal DC and AC from Source A
This includes power available at 2NPS-SWG001 from 115kV Source A (KA). Following a plant trip the event includes the transfer of supply from the main generator to the 115kV supply. This includes the opening and closing of associated circuit breakers and the DC battery (2BYS-BAT1A) required for circuit breaker control.

NB - Normal DC and AC from Source B
This includes power available at 2NPS-SWG003 from 115kV Source B (KB). Following a plant trip the event includes the transfer of supply from the main generator to the 115kV supply. This includes the opening and closing of associated circuit breakers and the DC battery (2BYS-BAT1B) required for circuit breaker control.

D1 - Division I Emergency DC

This top event includes the 2BYS*SWG002A battery board, two battery chargers 2BYS*CHGR2A1 and 2BYS*CHGR2A2, the 2BYS*BAT2A battery, the two AC supply buses (2LAC*PNL100A and 2EJS*PNL100A), and associated breakers.

D2 - Division II Emergency DC

This top event includes the 2BYS*SWG002B battery board, two battery chargers (2BYS*CHGR2B1 and 2BYS*CHGR2B2), the 2BYS*BAT2B battery, the two AC supply buses (2LAC*PNL300B and 2EJS*PNL300B), and associated breakers.

UA - Uninterruptible Power Supply (UPS) - Source A

This top event includes UPS 2VBA*UPS2A and its connections to AC and DC power supplies. Two separate cables supply AC power from Division I AC. Included in the model for this top event are the circuit breakers for each line. One cable supplies DC power from Division I DC. The circuit breaker associated with the cable is modeled in this top event.

UB - Uninterruptible Power Supply - Source B

This top event includes UPS 2VBA*UPS2B and its connections to AC and DC power supplies. Two separate cables supply AC power from Division II AC. Included in the model for this top event are the circuit breakers for each line. One cable supplies DC power from Division II DC. The circuit breaker associated with the cable is modeled in this top event.

E1 - ECCS Logic Division I

This top event models ECCS Division I automatic initiation signals and logic. The model includes drywell pressure and RPV level transmitters and switches.

E2 - ECCS Logic Division II

This top event models ECCS Division II automatic initiation signals and logic. The model includes drywell pressure and RPV level transmitters and switches.

ME - Manual ECCS Actuation

Top event ME models the operators manually starting ECCS pumps from the Control Room as required to provide RPV level control per N2-EOP-RPV, Section RL. This operator action is modeled when either or both ECCS logic signal (E1 or E2) has failed to provide an automatic signal.

SA - Service Water, Train A

This top event models service water pumps and valves supplying the A header (Div. I loop). Valves, heat exchangers and coolers that are supplied off the main header are included in their respective system. For example, emergency diesel coolers and its supply and discharge valves are included in the emergency diesel model.

SB - Service Water Train B

This top event models service water pumps and valves supplying the B header (Div. II loop). Valves, heat exchangers and coolers that are supplied off the main header are included in their respective system. For example, emergency diesel coolers and its supply and discharge valves are included in the emergency diesel model.

RW - Reactor Building Closed Loop Cooling Water

Top event RW models the Reactor Building closed loop cooling water (RBCLC) system. Two of three pumps supplying two of three heat exchangers is required for success.

TW - Turbine Building Closed Loop Cooling Water

Top event TW models the Turbine Building closed loop cooling water (TBCLC) system. Two of three pumps supplying two of three heat exchanges is required for success.

MA - North Aux Bay MCC Room Cooling

Top event MA models redundant unit coolers in the North auxiliary bay MCC room. Failure is assumed to cause failure of the low pressure core spray system and low pressure coolant injection train "A".

MB - South Aux Bay MCC Room Cooling

Top event MB models redundant unit coolers in the South auxiliary bay MCC room. Failure is assumed to cause failure of the low pressure coolant injection train "B" and "C".

AS - Instrument Air

Top event AS models the station air compressors supplying the main header in the Reactor Building. Success requires one of three air compressors supplying the header. Valves, tanks and components supplied from the main header are included in their respective system. For example the valves and accumulators supplying the outside containment vent purge valves are included in the containment venting model.

N1 - High Pressure Instrument

The high pressure instrument nitrogen (gaseous) top event models, the backup nitrogen supply to instrument nitrogen (N2). Included are six gaseous nitrogen storage tanks and associated valves.

N2 - Instrument Nitrogen

The instrument nitrogen (liquid) top event models the conversion of liquid nitrogen to gaseous nitrogen for supply to air operated valves (non-ADS) inside primary containment. Included in the model are the two liquid nitrogen storage tanks, the ambient vaporizers, electric trim heaters, system valves, and the main accumulator. The N2 model includes the availability of high pressure instrument nitrogen (Top Event N1) as a backup to N2.

TA - Condensate Storage Tank A

Top event TA models availability of condensate storage Tank A. The tank is a water source for RCIC and feedwater.

TB - Condensate Storage Tank B

Top event TB models availability of condensate storage Tank B. The tank is a water source for HPCS and feedwater.

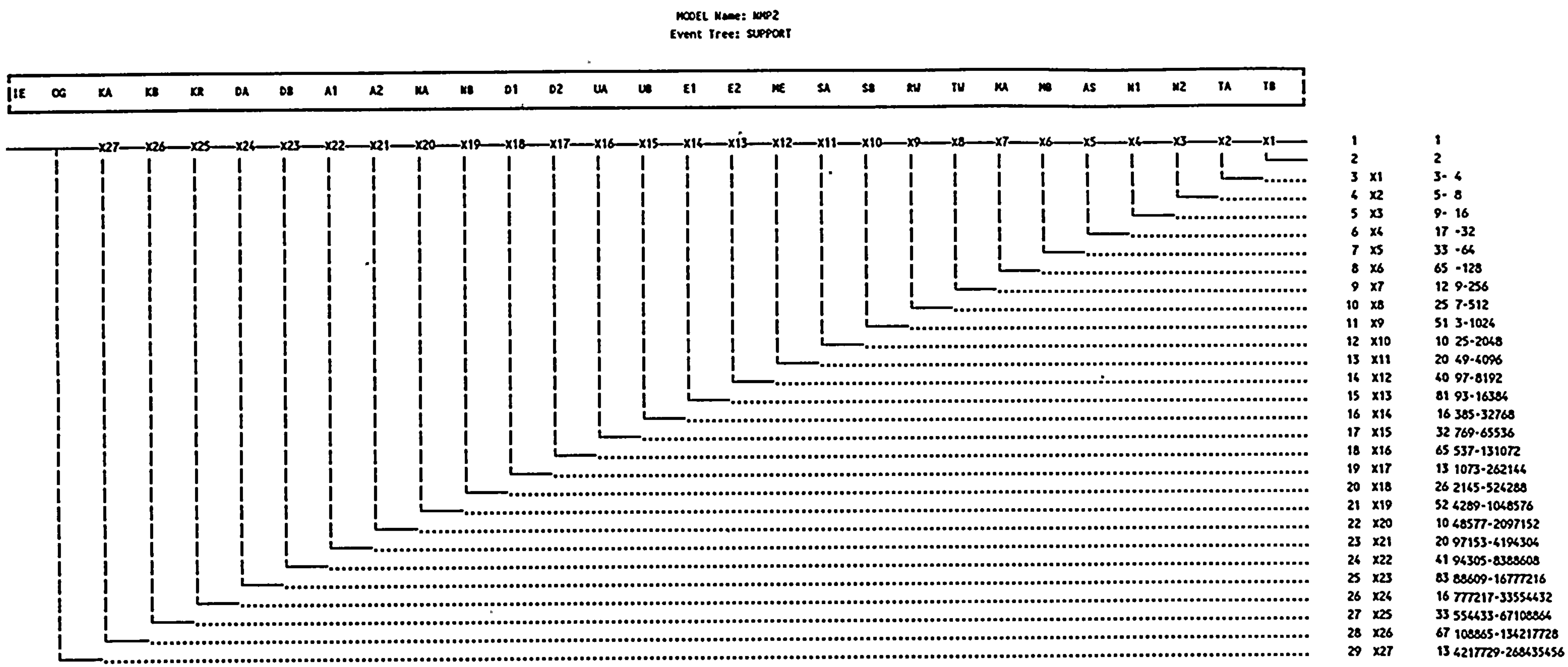


Figure 3.1.4-1

Support System Event Tree

Figure 3.1.4-1
Support System Event Tree

MODEL Name: NMP2
Top Event Legend for Tree: SUPPORT

Top Event Designator	Top Event Description
IE	INITIATING EVENT
OG	OFFSITE AC GRID
KA	115 KV SOURCE "A"
KB	115 KV SOURCE "B"
KR	RECOVERY FROM PARTIAL LOSP (KA OR KB FAILURE)
DA	DIV I DC BATTERY
DB	DIV II DC BATTERY
A1	DIVISION I EMERGENCY AC
A2	DIVISION II EMERGENCY AC
NA	NORMAL AC AND DC SOURCE "A"
NB	NORMAL AC AND DC SOURCE "B"
D1	DIVISION I EMERGENCY DC
D2	DIVISION II EMERGENCY DC
UA	UPS SOURCE "A"
UB	UPS SOURCE "B"
E1	ECCS LOGIC TRAIN 1
E2	ECCS LOGIC TRAIN 2
ME	MANUAL ECCS ACTUATION
SA	SERVICE WATER TRAIN "A"
SB	SERVICE WATER TRAIN "B"
RW	RBCLCW
TW	TBCLCW
MA	NORTH AUX BAY MCC ROOM COOLING
MB	SOUTH AUX BAY MCC ROOM COOLING
AS	INSTRUMENT AIR
N1	HIGH PRESSURE INSTRUMENT NITROGEN (GAS)
N2	INSTRUMENT NITROGEN (LIQUID)
TA	CONDENSATE STORAGE TANK "A"
TB	CONDENSATE STORAGE TANK "B"

Figure 3.1.4-1
Support System Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: SUPPORT

SF	Split Fraction Logic
OGF	INIT=LOSP + INIT=BLOSP + INIT=ALOSP
OG1	1
KAF	OG=F + INIT=KAX
KA1	1
KBF	OG=F + INIT=KBX
KB1	1
KRS	KA=S*KB=S
KRF	KA=F*KB=F
KR1	KA=S*KB=F
KR2	KA=F*KB=S
DA1	1
DB1	DA=S
DB2	1
A1F	INIT=FLDG2 + INIT=A1X + KA=F*KR=F*DA=F
A11	KA=S + KR=S
A12	KA=F*KR=F
A1F	1
A2F	INIT=FLDG2 + INIT=A2X + KB=F*KR=F*DB=F
A21	(KA=S+KR=S)*(KB=S+KR=S)*A1=S
A22	KA=F*KR=F*KB=S*A1=S
A23	KA=S*KB=F*KR=F*A1=S
A24	KA=F*KB=F*A1=S
A25	(KA=S+KR=S)*(KB=S+KR=S)*A1=F
A26	KA=F*KR=F*KB=S*A1=F
A27	KA=S*KB=F*KR=F*A1=F
A29	KA=F*KR=F*A1=F*DA=F
A28	KA=F*KB=F*A1=F
A2F	1
D1F	INIT=D1X + DA=F
D11	A1=S
D12	A1=F
D1F	1

Figure 3.1.4-1
Support System Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: SUPPORT

SF	Split Fraction Logic
D2F	INIT=D2X + DB=F
D21	A1=S*A2=S*D1=S
D22	A1=F*A2=S*D1=S
D23	A1=S*A2=F*D1=S
D24	A1=F*A2=F*D1=S
D25	A1=S*A2=S*D1=F
D26	A1=F*A2=S*D1=F
D27	A1=S*A2=F*D1=F
D28	A1=F*A2=F*D1=F
D2F	1
NAF	KA=F
NA1	1
NBF	KB=F
NB1	1
UAF	A1=F*D1=F + KA=F*KR=F*D1=F
UA1	A1=S*D1=S
UA2	A1=F*D1=S
UA3	A1=S*D1=F
UAF	1
UBF	A2=F*D2=F + KB=F*KR=F*D2=F
UB1	A1=S*A2=S*D1=S*D2=S*UA=S
UB2	A2=F*D2=S
UB3	A1=S*A2=S*D1=S*D2=F*UA=S
UB4	A1=F*A2=S*D2=F
UB5	A1=S*A2=S*D1=F*D2=F*UA=S
UB6	A1=F*A2=S*D2=S
UB7	A1=S*A2=S*D1=F*D2=S*UA=S
UBA	A1=S*A2=S*D1=S*D2=S*UA=F
UBB	A1=S*A2=S*D1=S*D2=F*UA=F
UBC	A1=S*A2=S*D1=F*D2=F*UA=F
UBD	A1=S*A2=S*D1=F*D2=S*UA=F
UBF	1

Figure 3.1.4-1
Support System Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: SUPPORT

SF	Split Fraction Logic
E1F	UA=F + D1=F
E12	INIT=SLOCA + INIT=MLOCA + INIT=LLOCA
E11	1
E2F	UB=F + D2=F
E22	E1=S*(INIT=SLOCA + INIT=MLOCA + INIT=LLOCA)
E2B	E1=F*(INIT=SLOCA + INIT=MLOCA + INIT=LLOCA)
E2A	E1=F
E21	1
MES	E1=S*E2=S
ME2	E1=F*E2=F
ME1	1
SAF	INIT=FLDG1+ INIT=SAX + INIT=SWX + INIT=FLSW*(A2=F+KB=F*D2=F)
SAF	KA=F*KB=F*(A1=F+D1=F) + OG=S*KA=F*KR=F*KB=S*A2=F + OG=S*KA=S*KB=F*KR=F*A1=F
SA1	KA=S*KB=S*A1=S*D1=S*A2=S*D2=S
SA2	KA=S*KB=S*A1=S*D1=F*A2=S*D2=S
SA3	KA=S*KB=S*A1=S*D1=S*A2=S*D2=F
SA4	KA=S*KB=S*A1=S*D1=F*A2=S*D2=F
SA5	KA=F*KB=F*A1=S*D1=S*A2=S*D2=S
SA6	KA=F*KB=F*(A2=F+D2=F)*A1=S*D1=S
SA7	OG=S*KA=F*KB=S*A1=S*D1=S*A2=S*D2=S
SA8	OG=S*KA=F*KB=S*(A1=F+D1=F)*A2=S*D2=S
SAG	OG=S*KA=S*KB=S*A1=F*A2=S*D2=S
SAG	INIT=FLSW*A2=S*D2=S
SA9	OG=S*KA=F*KB=S*A1=S*D1=S*A2=S*D2=F
SAA	OG=S*KA=F*KB=S*(A1=F+D1=F)*A2=S*D2=F
SAA	INIT=FLSW*A2=S*D2=F
SAC	OG=S*KA=S*KB=F*A1=S*D1=S*A2=S*D2=S
SAB	OG=S*KA=S*KB=F*A1=S*D1=S*(A2=F+D2=F)
SAH	OG=S*KA=S*KB=S*A1=S*D1=S*A2=F
SAH	OG=S*KA=F*KR=S*A2=F*D1=S
SAD	OG=S*KA=S*KB=F*A1=S*D1=F*A2=S*D2=S
SAE	OG=S*KA=S*KB=F*A1=S*D1=F*(A2=F+D2=F)

Figure 3.1.4-1
Support System Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: SUPPORT

SF	Split Fraction Logic
SAF	1
SBF	INIT=SWX + INIT=SAX*(A2=F + KB=F*D2=F)
SBF	INIT=FLSW*(A2=F+KB=F*D2=F)
SBF	INIT=FLDG1*(A2=F+KB=F*D2=F)
SBI	INIT=SAX*D2=S + INIT=FLDG1*D2=S
SBF	KA=F*KB=F*(A2=F+D2=F) + OG=S*KA=F*KR=F*KB=S*A2=F + OG=S*KA=S*KB=F*KR=F*A1=F + (INIT=SAX*D2=F)
SB1	KA=S*KB=S*A1=S*D1=S*A2=S*D2=S*SA=S
SBZ	KA=S*KB=S*A1=S*D1=S*A2=S*D2=S*SA=F
SB2	KA=S*KB=S*A1=S*D1=F*A2=S*D2=S*SA=S
SBY	KA=S*KB=S*A1=S*D1=F*A2=S*D2=S*SA=F
SB3	KA=S*KB=S*A1=S*D1=S*A2=S*D2=F*SA=S
SBX	KA=S*KB=S*A1=S*D1=S*A2=S*D2=F*SA=F
SB4	KA=S*KB=S*A1=S*D1=F*A2=S*D2=F*SA=S
SBW	KA=S*KB=S*A1=S*D1=F*A2=S*D2=F*SA=F
SB5	KA=F*KB=F*A1=S*D1=S*A2=S*D2=S*SA=S
SBV	KA=F*KB=F*A1=S*D1=S*A2=S*D2=S*SA=F
SBU	KA=F*KB=F*(A1=F+D1=F)*A2=S*D2=S
SB7	OG=S*KA=F*KB=S*A1=S*D1=S*A2=S*D2=S*SA=S
SBT	OG=S*KA=F*KB=S*A1=S*D1=S*A2=S*D2=S*SA=F
SB8	OG=S*KA=F*KB=S*(A1=F+D1=F)*A2=S*D2=S*SA=S
SBS	OG=S*KA=F*KB=S*(A1=F+D1=F)*A2=S*D2=S*SA=F
SBG	OG=S*KA=S*KB=S*A1=F*A2=S*D2=S*SA=S
SBG	INIT=FLSW*A2=S*D2=S*SA=S
SBL	OG=S*KA=S*KB=S*A1=F*A2=S*D2=S*SA=F
SBL	INIT=FLSW*A2=S*D2=S*SA=F
SB9	OG=S*KA=F*KB=S*A1=S*D1=S*A2=S*D2=F*SA=S
SBR	OG=S*KA=F*KB=S*A1=S*D1=S*A2=S*D2=F*SA=F
SBA	OG=S*KA=F*KB=S*(A1=F+D1=F)*A2=S*D2=F*SA=S
SBA	INIT=FLSW*A2=S*D2=F*SA=S
SBQ	OG=S*KA=F*KB=S*(A1=F+D1=F)*A2=S*D2=F*SA=F
SBQ	INIT=FLSW*A2=S*D2=F*SA=F
SBC	OG=S*KA=S*KB=F*A1=S*D1=S*A2=S*D2=S*SA=S

Figure 3.1.4-1
Support System Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: SUPPORT

SF	Split Fraction Logic
SBO	OG=S*KA=S*KB=F*A1=S*D1=S*A2=S*D2=S*SA=F
SBB	OG=S*KA=S*KB=F*A1=S*D1=S*(A2=F+D2=F)*SA=S
SBP	OG=S*KA=S*KB=F*A1=S*D1=S*(A2=F+D2=F)*SA=F
SBH	OG=S*KA=S*KB=S*A1=S*D1=S*A2=F*SA=S
SBH	OG=S*KA=F*KR=S*A2=F*D1=S*SA=S
SBK	OG=S*KA=S*KB=S*A1=S*D1=S*A2=F*SA=F
SBK	OG=S*KA=F*KR=S*A2=F*D1=S*SA=F
SBD	OG=S*KA=S*KB=F*A1=S*D1=F*A2=S*D2=S*SA=S
SBN	OG=S*KA=S*KB=F*A1=S*D1=F*A2=S*D2=S*SA=F
SBE	OG=S*KA=S*KB=F*A1=S*D1=F*(A2=F+D2=F)*SA=S
SBM	OG=S*KA=S*KB=F*A1=S*D1=F*(A2=F+D2=F)*SA=F
SBF	1
RWF	INIT=RWX + NA=F + NB=F + SA=F
RW1	1
TWF	INIT=TWX + NA=F + NB=F + SA=F
TW2	NB=F*SA=S*NA=S
TW1	1
MAF	ACA
MAA	SA=F
MA1	1
MBF	ACB
MBA	SB=F
MB1	1
ASF	INIT=ASX + NA=F*NB=F + RW=F
AS1	NA=S*NB=S*RW=S
AS2	NA=F*NB=S*RW=S
AS3	NA=S*NB=F*RW=S
ASF	1
N11	1
N2F	NA=F*N1=F + INIT=N2X
N23	NA=F*N1=S
N22	NA=S*N1=F

Figure 3.1.4-1
Support System Event Tree

MODEL Name: NMP2
Split Fraction Logic for Event Tree: SUPPORT

SF	Split Fraction Logic
N21	NA=S*N1=S
N2F	1
TA1	1
TB1	1

NODC:= D1=F*D2=F
Rule Comment
USED IN CETS FOR NO INSTRUMENTATION

ACA:= A1=F + KA=F*KR=F*(D1=F+SA=F)
Rule Comment
LOSS OF EMERGENCY AC DIV I

ACB:= A2=F + KB=F*KR=F*(D2=F+SB=F)
Rule Comment
LOSS OF EMERGENCY AC DIV II

BLACK:= ACA*ACB
Rule Comment
STATION BLACKOUT MACRO

BLACK1:= BLACK*OG=F
Rule Comment
OFFSITE AC IS RECOVERABLE

3.1.5 Sequence Grouping and Back-end Interfaces

Each sequence in the Level 1 model is assigned (binned) to an end state. The two primary end state groups are success and core damage, where core damage is subdivided into several end states. The assignment of sequences to the Level 1 end states is performed by specifying logic rules in terms of the successes and failures of top events in the event trees. The use of logic rules in the event trees is described further in Section 3.3.7.

The Level 2 (Back-end) Containment Event Tree (CET) model depends on the Level 1 model just as the second Level 1 Front-line event trees in Figure 3.1.2-1 depend on the first front-line event tree, the support system event tree, and the initiating event. The Level 2 model required information on support system and front-line system availability, RCS conditions (i.e., high pressure, low pressure and LOCA size), reactor power (ATWS), and containment status. These dependencies are addressed in much the same way as with event trees in the Level 1 model. Logic rules are specified in terms of top event success and failures and the Level 2 event trees are linked directly to the Level 1 model such that sequences are quantified from initiating event to release category (Level 2 end states). The bins are a convenience used in the assessment of Level 1 and to provide a useful method of discussing groups of sequences that may impact Level 2 analyses (e.g., MAAP runs). Howerver, because each Level 1 sequence is linked directly to Level 2, the Level 1 binning does not affect the quantification process. Thus, with this methodology there is no need to define a complete detailed list of Level 1 end states to track all unique dependencies on the Level 2 model.

The following summarizes how the back-end interface (Level 2 dependencies on Level 1) is accomplished:

- The same rules that are used to define Level 1 end states are applied in the Level 2 model. These rules define functional Level 1 sequences based primarily on critical safety function failures. These Level 1 end state rules provide important information about the sequence initiating event type, reactor power, injection systems and containment status. Level 1 end states are described further below.

- Additional rules are applied to identify whether the RPV is at high or low pressure at the time of core damage as well as whether support and front-line systems and functions are available.

A description of the rules applied in the Level 2 model are further described in Section 4.

The definition of core damage end state groups is based primarily on the failure of critical safety functions required to attain a safe stable state as directed in the EOPs. The following summarizes the assignment of sequences to end state groups based on failure of critical safety functions:

```
Init.        Reactor          RPV Level         Primary
------->     Power    ------> & Pressure ------> Containment ----> Success
Event        Control          Control           Control
               |                 |                 |
          Class IVA         Class IA          Class IIA
          Class IVL         Class IB          Class IIT
                            Class IC          Class IIL
                            Class ID          Class IIID
                            Class IIIB        Class V
                            Class IIIC
```

Failure to provide reactor power control has the potential to cause fuel damage and lead to early containment failure due to overpressurization and excessive or dynamic loads with subsequent core damage. Adequate heat removal is unlikely following such a catastrohic failure. RPV level and pressure control includes the high pressure injection function, RPV depressurization and the low pressure injection function. Failure of this function results in core damage and the containment status is addressed in the Level 2 containment event tree model. Failure to provide primary containment control (heat removal or vapor suppression or LOCA outside containment) results in opportunities for the containment to fail first with subsequent core damage. Defining core damage end states in this way provides a convenient interface with the Level 2 (back-end) containment event tree models and allows the dominant core damage sequences to be more readily assessed based on functional failure.

The assessment of core damage sequences for each of the above critical safety function failures is developed further in Figures 3.1.5-1, 2 and 3. Each figure is discussed below:

- **Reactor Power Control Failure (Figure 3.1.5-1).** All reactor power control failures lead to containment failure either before or near the time of core damage. LOCAs with failure to scram (includes failure of alternate rod insertion or recirc. pump trip), are binned to Class IVL. Transients are evaluated in the ATWS event tree model where failure to control power can lead to a RPV LOCA, Class IVL. If not the sequence is binned to Class IVA.

- **Loss of RPV Water Level (Figure 3.1.5-2).** Either a successful scram has occurred or reactivity control is successful in the ATWS model for transients. Loss of injection sequences in the ATWS and Station Blackout models are binned to Class IC and IB, respectively. Loss of injection sequences in the Transient and Small LOCA models are binned to Class IA and ID, depending on whether the RPV is depressurized. Loss of injection sequences in the Medium LOCA model are binned to Class IIIB and IIIC, depending on whether the RPV is depressurized. Loss of injection sequences in the Large LOCA model are binned to Class IIIC as the RPV is guaranteed to be depressurized. Note that overpressure protection (i.e., RPV pressure control) is not modeled except for ATWS since the frequency of severe consequences are very unlikely.

- **Primary Containment Overpressure (Figure 3.1.5-3).** Reactor power control is successful and RPV water level (inventory) is being maintained when containment overpressure conditions exist. For Transients and Small LOCA, if primary containment heat removal is not maintained the containment may fail prior to core damage, subsequently causing core damage (Class IIA) or core damage may occur just prior to containment failure (Class IIT). For all LOCAs, vapor suppression failure is binned to Class IIID where the containment is assumed to fail early causing core damage. For Medium and Large LOCAs, overpressure failure due to loss of heat removal leads to containment failure prior to core damage (Class IIL).

Table 3.1.5-1 provides a summary description of each Level 1 core damage end state as well as the status of the containment at the time of core damage. The following table summarizes the applicable core damage end states for each initiating event type and Level 1 event tree model:

		Loss of Injection		Containment Overpressure		
Initiator/Model	No Power Control	RPV High	Press. Low	Vapor Supp. Fails	Prior to CD	After CD
Transient/Transient	na	IA	ID	na	IIA	IIT
Transient/Blackout	na	IB	IB	na	IIA	IIT
Transient/ATWS	IVA or L	IC	IC	na	IIA	IIT
Small LOCA/SLOCA	IVL	IA	ID	IIID	IIA	IIT
Medium LOCA/MLOCA	IVL	IIIB	IIIC	IIID	IIL	na
Large LOCA/LLOCA	IVL	na	IIIC	IIID	IIL	na
ISLOCA/ISLOCA	na	na	V	na	na	na

Table 3.1.5-1
Summary of Level 1 Core Damage
End States

Core Damage End State	Accident Sequence Definition	Containment[1] Status
Class IA	Loss of inventory makeup with the RPV at high pressure (transient and small LOCA models).	Intact (CET1)
Class IB	Loss of inventory makeup in the station blackout model.	Intact (CET1)
Class IC	Loss of inventory makeup in the ATWS model.	Intact (CET1)
Class ID	Loss of inventory makeup with the RPV at low pressure (transient and small LOCA models).	Intact (CET1)
Class IIA	Loss of containment heat removal and core damage induced post containment failure (transient and small LOCA models).	Failed (CET2)
Class IIT	Loss of containment heat removal and core damage induced prior to containment failure (transient and small LOCA models).	Intact (CET1)
Class IIL	Loss of containment heat removal and core damage induced post containment failure (medium and large LOCA models).	Failed (CET2)
Class IIIB	Loss of inventory makeup in the medium LOCA model with RPV at high pressure.	Intact (CET1)
Class IIIC	Loss of inventory makeup in the medium and large LOCA models with RPV at low pressure.	Intact (CET1)
Class IIID	Vapor suppression failure in the LOCA models challenge containment and causes core damage.	Failed (CET2)
Class IVA	Reactor power control failure in the transient model challenges containment and induces core damage post containment failure.	Failed (CET2)
Class IVL	Reactor power control failure and LOCA conditions challenges containment and induces core damage post containment failure.	Failed (CET2)
Class V	Interfacing System LOCA outside containment.	Bypassed (CET3)

(1) Intact means that containment has not failed in the Level 1 model. Containment isolation and/or structural failure including failure mode is addressed in the Level 2 CET 1 event tree and may transfer to CET2 if containment isolation fails.

Rev 0 (7/92)

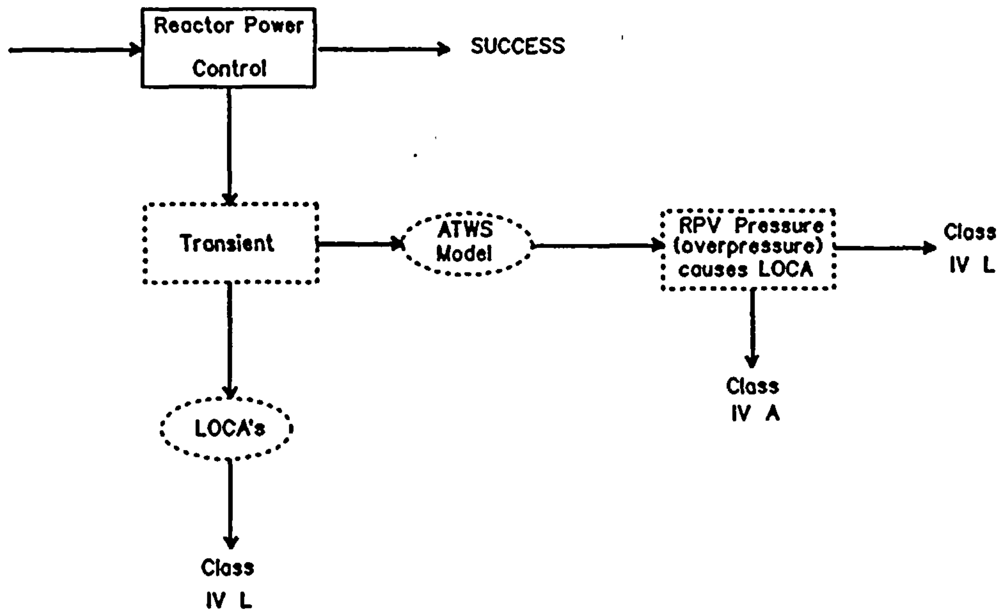


Figure 3.1.5-1
Level 1 End States For
Reactor Power Control Failure

I:\NMP2IPE\DGN\2F315.DGN Jul. 06, 1992 14:29:39

Rev 0 (7/92)

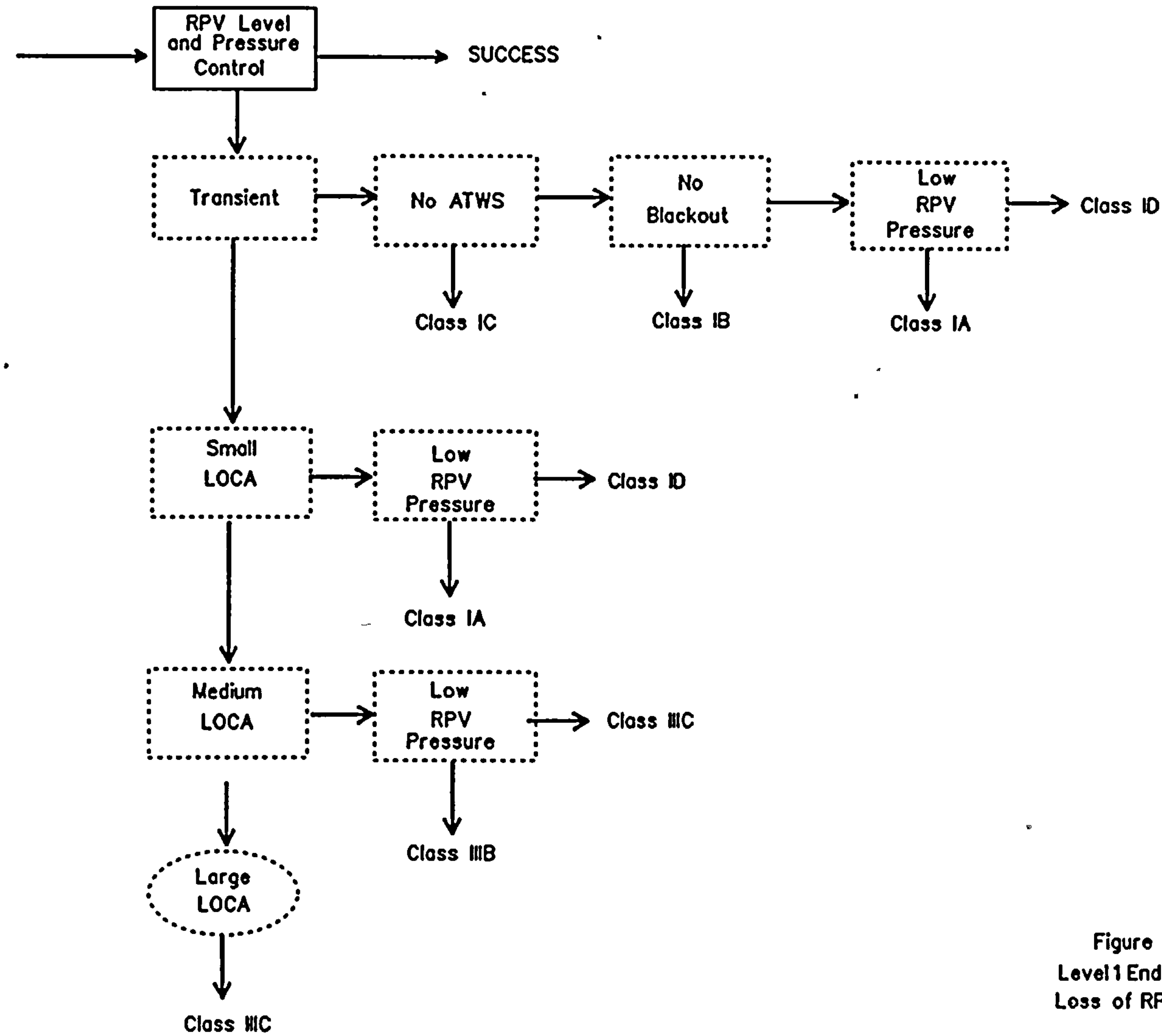


Figure 3.1.5-2
Level 1 End States For
Loss of RPV Inventory

Rev 0 (7/92)

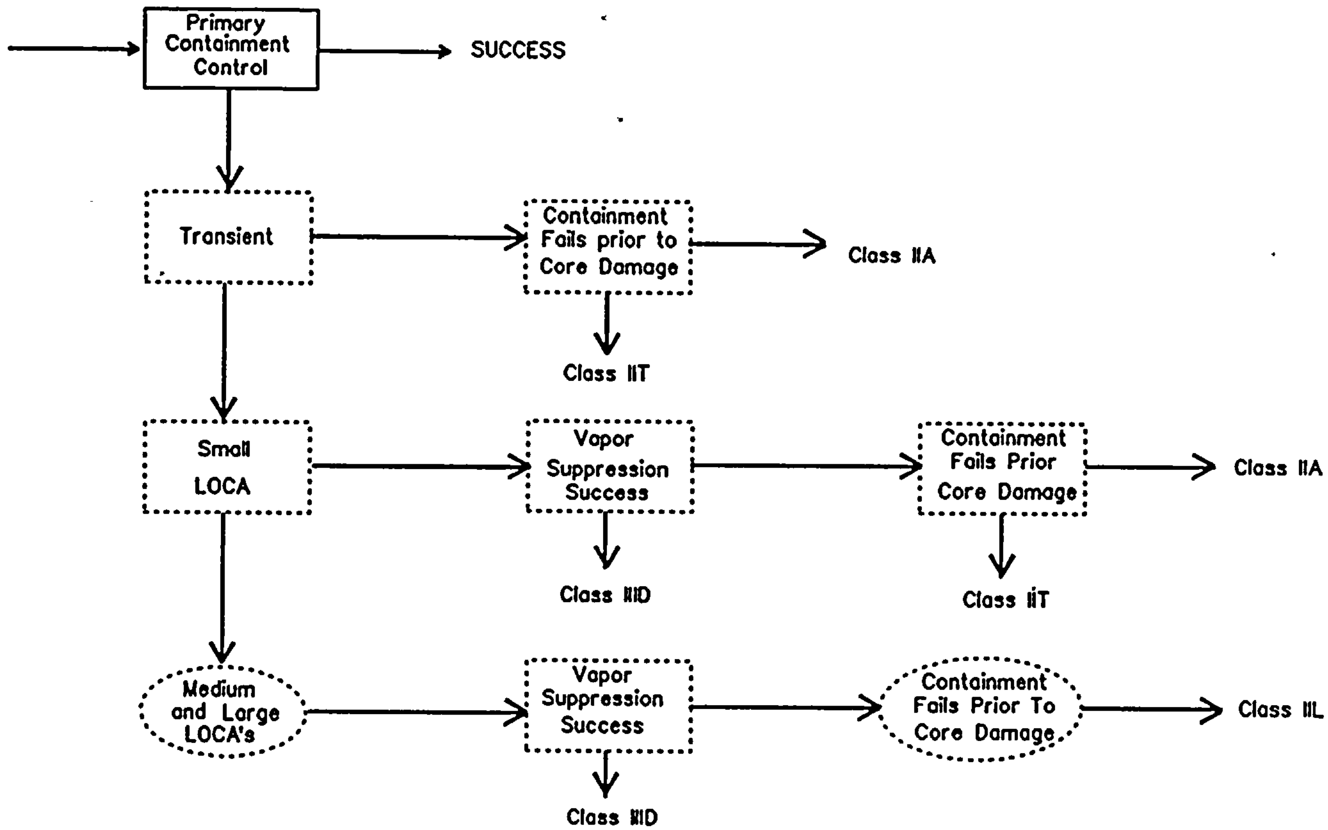


Figure 3.1.5-3
Level 1 End States For
Containment Overpressure

3.2.1 Systems Analysis

Initial System Screening and Familiarization

At the start of the IPE project, plant systems were reviewed to obtain a list of potential systems to include in the IPE. This review is documented in Table 3.2-1. The FSAR and Operating Procedures were reviewed to determine which systems potentially meet one or both of the following conditions:

1. System can potentially cause a plant trip condition and degrade other systems required to respond to the event.

2. System will be required for plant response, recovery or support of safety functions.

The first condition identifies potential common cause initiating events; support systems are important common cause initiating events because they can impact the systems required to mitigate plant transients. The systems identified in Table 3.2-1 under the "Init. Event" column are evaluated further during the initiating event selection task.

The second condition identifies systems required to respond to and mitigate events in the event sequence analysis. The systems identified in the "Plant Response" column of Table 3.2-1 are assessed in more detail while developing dependency tables, systems analysis and event sequence analysis. Table 3.2-2 summarizes the systems and event tree top events that are included in the model. The first column in this table identifies the subsection ("X") in Section 3.2.1 (i.e., 3.2.1.X) which documents the system description. The top events listed under each system are included in the event trees and system description as summarized in the table. Table 3.4.3-3, located at the end of Section 3.4, lists the system top events included in the Level I model. This table is presented as a foldout page so that it can be referenced from throughout Section 3.2.1. Figure 3.2-1 shows a simplified diagram of system actuation logic.

Table 3.2-1 Initial System Screening

Operating Procedure N2-OP-_	System	System Review: Init. Event	System Review: Plant Response	Comment
1	Main Steam	Yes	Yes	Steam bypass, containment integrity
2	Moisture Separator Reheater	No	No	
3	Condensate & Feedwater	Yes	Yes	Core cooling, containment integrity
4	Condensate Storage & Transfer	No	Yes	Condensate tanks, supports feedwater
5	Condensate Demineralizers	No	No	
8	Feedwater Heaters & Extraction Steam	No	No	
9	Condenser Air Removal	Yes	Yes	Condenser availability
10A	Circulating Water (CWS)	Yes	Yes	Condenser availability
10B	Acid Treatment & Hypochlorite	No	No	
11	Service Water	Yes	Yes	Support system, emergency core flood
12	Traveling Water Screens Wash	No	No	
13	Reactor Building Closed Loop Cooling Water (RBCLC)	Yes	Yes	Support system
14	Turbine Building Closed Loop Cooling Water (TBCLC)	Yes	Yes	Support system, loss of Cond. & FW
15	Makeup Water Treatment	No	No	
16	Makeup Water Storage & Transfer	No	No	
17	Process Sampling	No	No	
19	Instrument & Service Air	Yes	Yes	Support system
20	Breathing Air	No	No	
21	Main Turbine	No	No	turbine trip initiator
22A,B,D	Turbine Generator Support	No	No	turbine trip initiator

Table 3.2-1 Initial System Screening

Operating Procedure N2-OP-_	System	System Review		Comment
		Init. Event	Plant Response	
23	Turbine Electrohydraulic Control	No	No	steam bypass, turbine trip initiator
24	Gen. Isolated Phase Bus Direct Cooling	No	No	
25	Aux. Steam, Condensate, Gland Seal	No	No	
26	Generator Stator and Exciter Rectifier	No	No	
27	Cooling Generator Hydrogen & CO_2	No	No	
29	Reactor Recirculation	Yes	Yes	Seal LOCA, Recirc. pump trip
30	Control Rod Drive Hydraulics	Yes	Yes	Reactivity control, CRD injection make-up
31	Residual Heat Removal	No	Yes	Core cooling, containment integrity
32	Low Pressure Core Spray	No	Yes	Core cooling
33	High Pressure Core Spray	No	Yes	Core cooling
34	ADS and Safety Relief Valves	Yes	Yes	Core cooling, RCS integrity
35	Reactor Core Isolation Cooling	No	Yes	Core cooling
36A	Standby Liquid Control	No	Yes	Reactivity Control
36B	Redundant Reactivity Control	Yes	Yes	Reactivity Control, support system
37	Reactor Water Cleanup	No	No	
38	Spent Fuel Pool Cooling & Cleanup	No	No	
39	Fuel Handling & Reactor Servicing	No	No	
40	Liquid Radwaste	No	No	
41	Solid Radwaste	No	No	
42	Offgas	No	No	
43	Fire Protection, Water	No	Yes	Alternate core flood, cooling water
44 - 47	Fire Protection, Other	No	No	

Table 3.2-1 Initial System Screening

Operating Procedure N2-OP-_	System	System Review Init. Event	System Review Plant Response	Comment
48	Auxiliary Boiler	No	No	
49	Hot Water & Glycol Heating	No	No	
50	Domestic Water	No	No	
51	Sanitary Plumbing	No	No	
52	Reactor Building Ventilation	Yes	Yes	Support system
53A	CB Chilled Water, Control & Relay Room Ventilation	Yes	Yes	Support system
53E	Standby Switchgear/Battery Room Ventilation	Yes	Yes	Support system
54A	Normal Switchgear Ventilation	Yes	Yes	Support system
54B	Ventilation - Chilled Water	Yes	Yes	Support system
55	Turbine Building Ventilation	No	No	
56	Radwaste Building Ventilation	No	No	
57	Diesel Generator Building Ventilation	No	Yes	Support system
58	Screenwell Building Ventilation	Yes	Yes	Support system
59A	CB/Reactor Building Electric Tunnel Ventilation	Yes	Yes	Support system
59B	Aux. Building South - Air Conditioning	No	No	
59C	Other Ventilation	No	No	
60	Drywell Cooling	Yes	No	Containment heat removal
61A	Primary Containment Vent Purge & Nitrogen	No	Yes	Containment venting, Nitrogen to ADS, support system
61B	Standby Gas Treatment	No	Yes	Containment venting & source term
62	DBA Recombiner	No	No	

Table 3.2-1 Initial System Screening

Operating Procedure N2-OP-_	System	System Review		Comment
		Init. Event	Plant Response	
63	Reactor Building Drains	No	No	Consider in internal flood analysis
64	Turbine Building Drains	No	No	Consider in internal flood analysis
65	Radwaste Building Drains	No	No	Consider in internal flood analysis
66	Miscellaneous Drains	No	No	Consider in internal flood analysis
67	Drywell Equipment & Floor Drains	No	No	Consider in internal flood analysis
68	Main Generator & 345Kv Switchgear	Yes	Yes	Support system
70	Station Electrical Feed & 115Kv	Yes	Yes	Support system
71	13.8Kv / 4160v / 600v AC Distrib.	Yes	Yes	Support system
72	Standby & Emergency AC Distribution	Yes	Yes	Support system
73A	Normal DC Distribution	Yes	Yes	Support system
73B	24v DC Distribution	Yes	No	Support system to nuclear instrumentation
74A	Emergency DC Distribution	Yes	Yes	Support system
74B	HPCS 125v DC Distribution	Yes	Yes	Support system
75	Station Lighting	No	No	
76	Plant Communication	No	No	
78	Remote Shutdown	No	No	Maybe later for externals and recovery
79	Radiation Monitoring	No	No	
81	Containment Leakage Monitoring	No	No	Containment isolation backup
82	Containment Atmosphere Monitoring	No	No	Containment isolation
83	Primary Containment Isolation	No	Yes	Containment integrity
84	Reactor Building Cranes	No	No	
86	Loose Parts Monitoring	No	No	

Table 3.2-1 Initial System Screening

Operating Procedure N2-OP-_	System	System Review: Init. Event	System Review: Plant Response	Comment
90	Seismic Monitoring	No	No	
91A	Process Computer	No	No	
91B	Safety Parameter Display	No	No	Operator response
92	Neutron Monitoring	No	No	
94	Traversing Incore Probe	No	No	
95A	Rod Worth Minimizer	No	No	
95B	Rod Sequence Control	No	No	
96	Reactor Manual Control/Rod Position	No	No	Reactivity control, ATWS 1 rod at a time
97	Reactor Protection	Yes	Yes	Reactivity control
100A	Standby Diesel Generator	No	Yes	Support system
100B	HPCS Diesel Generator	No	Yes	Support system

Table 3.2-2 Systems & Top Events

Section 3.2.1.x	System & Top Event(s)	Event Trees (x indicates Event Trees that include the Top Event)												
		SUP	TR1	TR2	SL1	SL2	ML1	ML2	LL1	LL2	SBO1	SBO2	AT1	AT2
1	High Pressure Core Spray													
	• HS: HPCS		x		x		x		x					x
2	Reactor Core Isolation Cooling													
	• IC: RCIC		x		x									x
	• U1 - U3: RCIC (Station Blackout)										x			
	• IL: Operator overrides ATWS trips													x
	• OA: Operator sheds DC loads (SBO)										x			
3	Low Pressure Core Spray													
	• LS: LPCS		x		x		x		x					x
4	Residual Heat Removal													
	• LA, LB: RHR A/B pump train		x		x		x		x			x	x	
	• HA, HB: RHR A/B heat exchanger			x		x		x		x		x	x	
	• IA, IB, LC: LPCI injection train		x		x		x		x					x(LC)
	• PA, PB: A/B Supp. Pool Cooling			x		x		x		x		x	x	
	• CA, CB: A/B Containment spray			x		x		x		x		x		
	• OH: Operator aligns RHR cooling			x		x		x		x		x	x	
5	ECCS Actuation													
	• E1, E2: Div. I/II ECCS actuation	x												
	• ME: Manual ECCS actuation	x												
6	AC Power Systems													
	• OG: Offsite power	x												
	• KA, KB: 115kv source A/B	x												
	• NA, NB: Normal AC & DC source A/B	x												
	• A1, A2: Div. I/II Emergency AC	x												
	• UA, UB: Vital UPS source A/B	x												
	• KR: Partial recovery of KA/KB	x												
7	DC Power Systems													
	• D1, D2: Div. I/II Emergency DC	x												
8	Service Water System													
	• SA, SB: Service Water Loop A/B	x												
9	Fire & Service Water Crossties													
	• SW: Service Water - RHR crosstie		x		x		x		x				x	
	• FP: Fire Water - RHR crosstie		x		x		x		x					
	• S1 - S3: Fire Water - RHR crosstie (SBO)										x			

Table 3.2-2 Systems & Top Events

Section 3.2.1.x	System & Top Event(s)	SUP	TR1	TR2	SL1	SL2	ML1	ML2	LL1	LL2	SB01	SB02	AT1	AT2
		Event Trees (x indicates Event Trees that include the Top Event)												
10	Containment Isolation													
	• IS: Containment Isolation (LEVEL 2)													
11	Ventilation Systems													
	• MA/MB: North/South MCC Area Unit Coolers	x												
12	Standby Liquid Control													
	• SL: SLCS												x	
13	Automatic Depressurization													
	• OD: Operator Depressurizes		x		x		x							
	• SV: SRV/ADS Valves		x		x		x							
	• O1 - O3: Operator Depress. (SBO)										x			
	• X1 - X3: SRVs Remain Open (SBO)										x			
	• AI: ADS Inhibit (ATWS)													x
	• OE: Op. Emerg. Depress. (ATWS)													x
	• SR: Adequate Relief (ATWS)												x	
	• SO: Stuck Open Relief Valve (ATWS)													x
14	Control Rod Drive (In top event CF)			x		x		x		x				x
15	Reactor Protection System													
	• RQ: Reactor scram		x		x		x		x					
	• QM: Reactor scram mechanical equip.												x	
	• QE: Reactor scram electrical equip.												x	
	• MS: Mode switch in shutdown												x	
16	Redundant Reactivity Control Systems													
	• C1, C2: Div I/II RRCS												x	
	• CH: Level control not high													x
	• WL: RPV Level > ⅓ core													x
	• MO: Operator overrides level 1													x
	• RI: Alternate Rod Insertion												x	
	• RT: Recirc. Pump Trip												x	
	• FT: Feedwater Runback												x	
17	Containment Venting													
	• CV: Containment Venting			x		x		x		x		x		x
	• GV: Gas Venting (LEVEL 2)													
	• VC: Containment Venting (LEVEL 2)													
	• FB: Drywell Venting (LEVEL 2)													
	• FD: Drywell Venting (Level 2)													

Table 3.2-2 Systems & Top Events

Section 3.2.1.x	System & Top Event(s)	SUP	TR1	TR2	SL1	SL2	ML1	ML2	LL1	LL2	SB01	SB02	AT1	AT2
		Event Trees (x indicates Event Trees that include the Top Event)												
18	Vapor Suppression													
	• VS: Vapor Suppression				x		x		x					
	• OV: Operator Sprays or Vents				x		x							
19	Reactor Building Closed Loop Cooling Water													
	• RW: RBCLC	x												
20	Turbine Building Closed Loop Cooling Water													
	• TW: TBCLC	x												
21	Condensate and Feedwater Systems													
	• FW: Feedwater Available		x		x								x	
	• CW: Condenser as a Heat Sink		x		x								x	
	• TA, TB: A/B Condensate Storage Tank	x												
22	Nitrogen													
	• N1: High Pressure Nitrogen	x												
	• N2: Instrument Nitrogen	x												
23	Instrument Air													
	• AS: IAS	x												
24	Late Containment Failure													
	• CF: Continued Injection after Failure			x		x		x		x		x		x
	• CI: Continued Injection at High Pressure			x		x						x		x
25	Reactor Recirculation -- Seal LOCA													
26	Recovery													
	• R1: AC Power Recovery			x										
	• I1 - I5: Recovery of offsite AC at different time phases (SBO)										x			
	• G1 - G5: Recover of Emergency EDG at different time phases (SBO)										x			
27	RPV Venting (LEVEL 2)													
28	H_2O_2 Analyzers (LEVEL 2)													

Table 3.2-2 Systems & Top Events

Section 3.2.1.x	System & Top Event(s)	Event Trees (x indicates Event Trees that include the Top Event)												
		SUP	TR1	TR2	SL1	SL2	ML1	ML2	LL1	LL2	SBO1	SBO2	AT1	AT2
29	Containment Flood Level Instrumentation (LEVEL 1 & 2)													
XX	Switches (discussion included in the Event Tree Write-ups, Section 3.1.2) • NL: Success • NH: No Injection • NE: No Catastrophic • RH: RHR			x x		x x		x		x		x x		x x x x

Rev 0 (7/92)

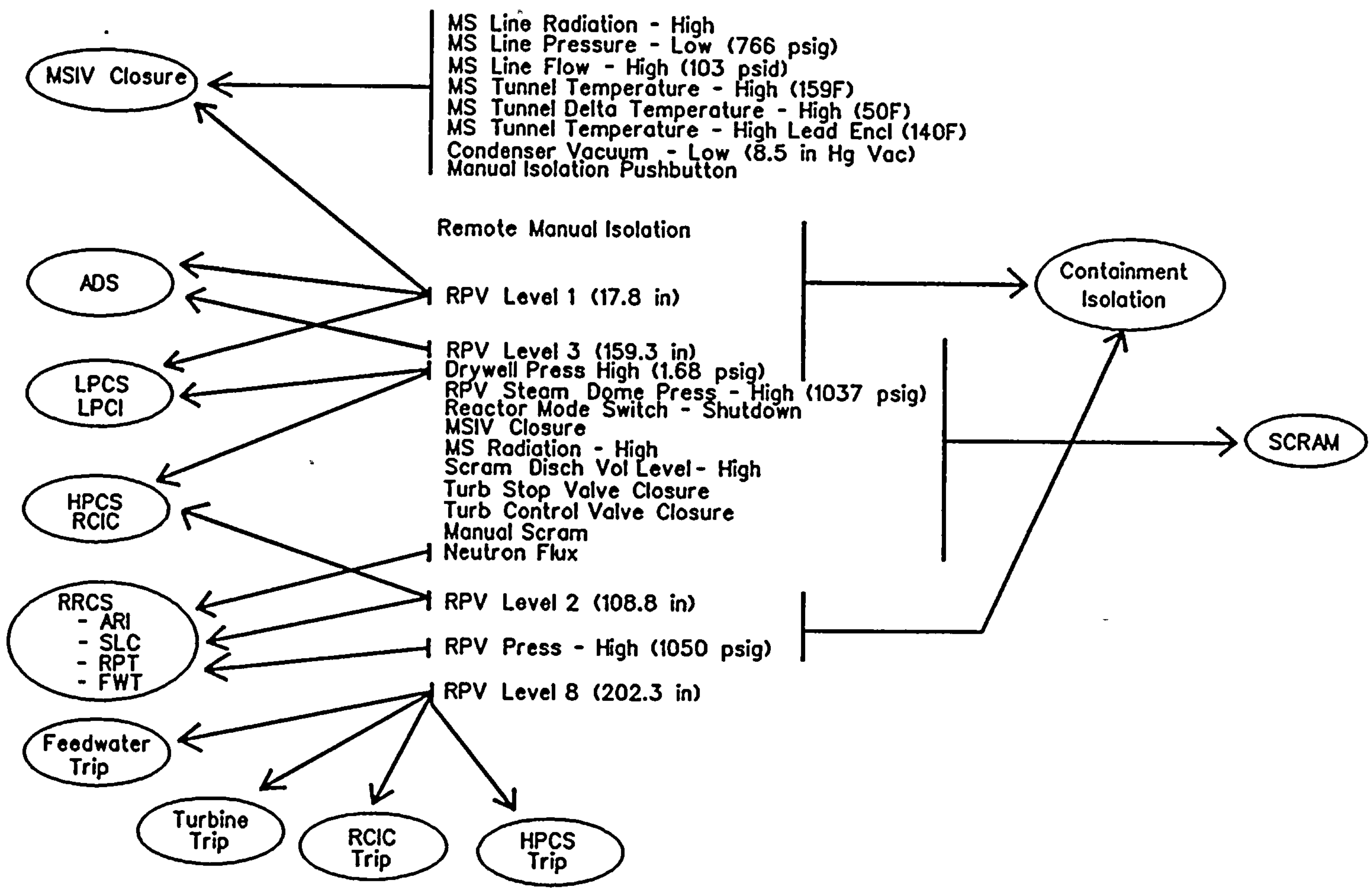
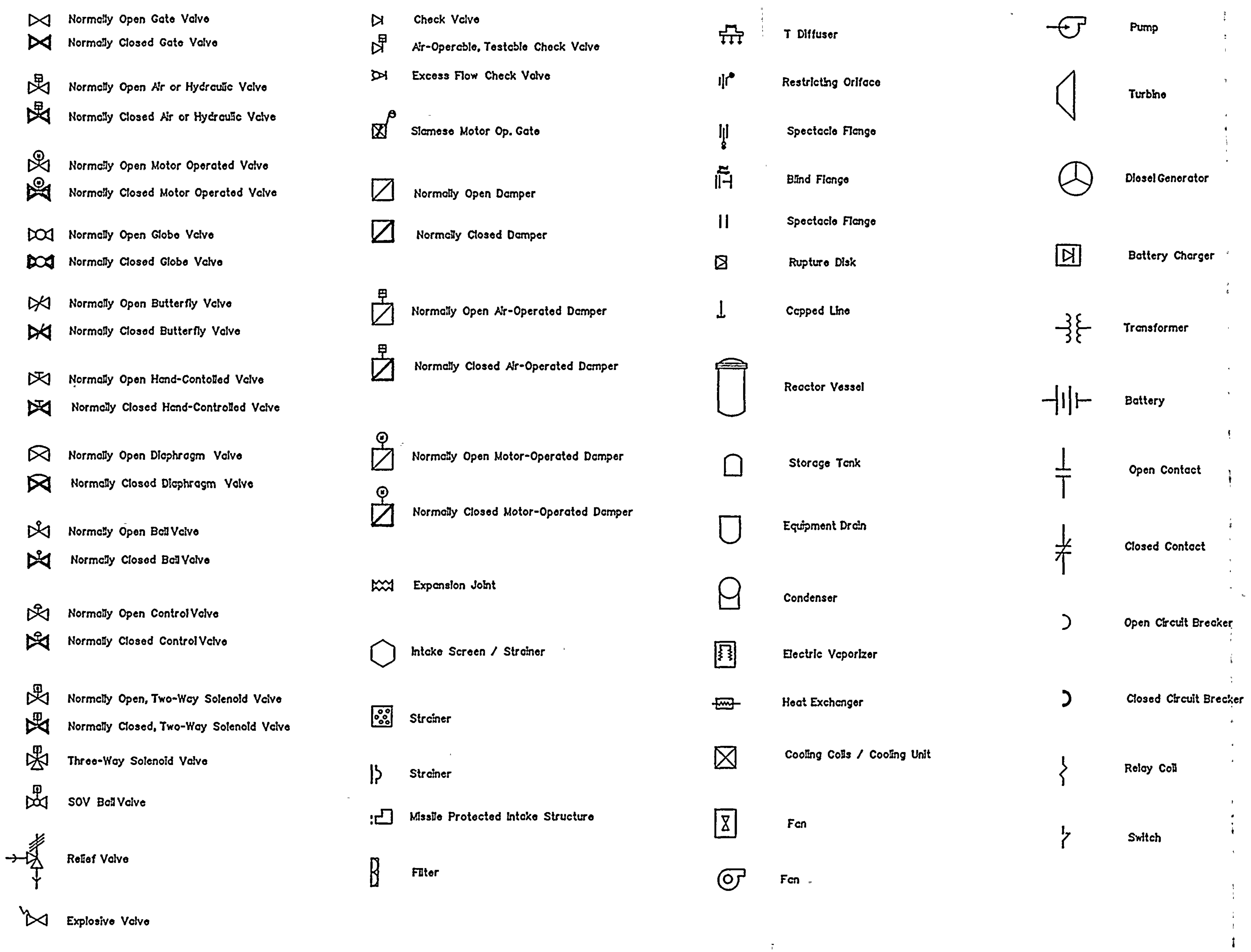


Figure 3.2.1-1
NMP2 Actuation Signals

I:\NMP2IPE\DGN\2F321.DGN Jul. 07. 1992 14:16:29

[illegible]

Rev 0 (7/92)



LO - Locked Open
LC - Locked Closed
FO - Fails Open
FC - Fails Closed
FAI - Fails-As-Is

FE - Flow Element
FT - Flow Transmitter
PT - Pressure Transmitter
PDT - Differetial Press. Xmtr

SI
APERTURE
CARD

Also Available On
Aperture Card

Figure 3.2.1-2
Schematic Symbol Key

I:\NMP2IPE\DGN\2F321.DGN Jul. 07, 1992 10:19:23

9208050191 - 12

System 1

High Pressure Core Spray

3.2.1.1 High Pressure Core Spray

3.2.1.1.1 System Function

The High Pressure Core Spray System (CSH) is a safety system which primarily maintains the reactor vessel inventory after small breaks which do not depressurize the reactor vessel. CSH may also supply makeup water to the reactor vessel in the event of reactor isolation and failure of the Reactor Core Isolation Cooling System.

3.2.1.1.2 Success Criteria

CSH is considered successful if it provides adequate flow to the RPV from Condensate Storage Tank (CST) B for 2 hours and then from the suppression pool for an additional 22 hours. High Pressure Core Spray is modeled in the front line event trees as top event HS. A simplified diagram of the CSH system is provided in Figure 3.2.1.1-1.

3.2.1.1.3 Support Systems

The CSH system model includes the Division III actuation system (switches, pressure transmitters, and level transmitters) and other support equipment (Division III emergency diesel, AC power, and DC power). A complete list of components in the system model and their respective support systems is included in table 3.2.1.1-1.

The system model considers the use of offsite power when it is available and connected to the Division III emergency switchgear. Figure 3.2.1.1-2 provides a simplified one-line electrical diagram.

If offsite power is not available, the emergency diesel generator is required for success of the CSH model. Also, the emergency diesel generator model requires service water to be supplied to the diesel generator in order to be successful. Therefore, Service Water Header A or B is required for EDG cooling when the offsite AC supply to Division III AC is unavailable. The two diesel service water supply MOVs (2SWP*MOV94A and B) are powered from Division III.

The CSH is considered a self supporting system because it has its' own Division III 120V AC, 125V DC, 600V AC, and 4.16kV AC.

The following systems interface or connect to the CSH system:

- Reactor Core Isolation (ICS)
- Reactor Plant Sampling (SSR)
- Condensate Makeup and Drawoff (CNS)
- Nuclear Boiler System Instrumentation (ISC)
- Residual Heat Removal (RHS)
- Standby Liquid Control (SLS)

3.2.1.1.4 System Operation

The CSH System is in the standby mode during normal operation. To facilitate an automatic start, the suction isolation valve, MOV101, is kept in the full open position. This keeps the pump suction line from condensate storage Tank B full of water. The CSH pump is normally in standby; however, the System Pressure Pump is operating to keep the system full of water between check valve V9 and the closed motor-operated injection valve MOV107. The suppression pool suction valve and the test line valves (MOV118, MOV105, MOV110, MOV112, MOV111) are all normally closed also. Test return valves MOV112 and MOV111 receive auto-closure signals on a LOCA signal.

The CSH system will auto-start on either a reactor water level 2 signal or a high drywell pressure signal. The system may also be started manually from the Control Room on Panel P601. If the system is in test when an initiation signal occurs, all equipment returns to a minimum flow position for pump protection.

The CSH system will stop injecting on a reactor high level trip signal (level 8). CSH Injection Valve MOV107 closes and will re-open on a low-low water level signal (level 2). The minimum flow path valve (MOV105) will open while MOV107 closes to protect the pump (< 825 gpm and > 240 psig).

The CSH pump will trip on:

- Motor Phase Overcurrent
- Feed Ground Overcurrent
- Bus Overfrequency

The CSH pumps switch to the suppression pool because of a low CST level or a high suppression pool level. During the switch, MOV118 from suppression pool opens and MOV101 from CST closes when MOV118 is fully open.

3.2.1.1.5 Instrumentation and Controls

The CSH system can be initiated from the Control Room. There are indications and/or alarms in the Control Room for the following:

- Testable Check Valve AOV108
- MOVs 101, 105, 107, 110, 112, 118, 111
- Manual Isolation Valve HCV120
- Pump P1 Run, Amps, Flow, and Pressure
- Pressure of the System Pressure Pump
- High Reactor Vessel Water Level
- HPCS high point low level

Other indications and alarms are available. See the Nine Mile Point Unit 2 OP for a complete list.

3.2.1.1.6 Technical Specifications

If CSH is inoperable, with the RCIC system and the Emergency Core Cooling System (ECCS) Divisions I and II operable; CSH must be restored to OPERABLE status within 14 days. Otherwise, the plant must be in, at least, hot shutdown within the next 12 hours and in cold shutdown within the following 24 hours.

CSH shall be demonstrated operable by the following procedures:

a. At least once per 31 days for the CSH system:

   1. Verify venting water at the high point vents. This demonstrates that the system piping from the pump discharge valve to the system isolation valve is filled with water.

   2. Verify that each valve (manual, power-operated, or automatic) in the flow path is not locked, sealed or otherwise secured in position, and is in the correct position.

b. Verify that when tested pursuant to specification 4.0.5, the CSH pump develops a flow of at least 6350 gpm against a test line pressure greater than or equal to 333 psig.

c. At least once every 18 months, perform a system functional test which includes a simulated automatic actuation of the system throughout its emergency operating sequence and verification that each automatic valve in the flow path actuates to its correct position. Actual injection of coolant into the reactor vessel may be excluded from this test. Verify that the CSH pump will auto-restart on low reactor vessel water level, Level 2, if the pump has been manually stopped.

d. At least once every 18 months, verify that the suction is automatically transferred from the condensate storage tank to the suppression pool on a high water level signal.

e. At least every 12 hours, verify the condensate storage tank required volume is present when the condensate storage tank is required to be operable per Specification 3.5.2.e.

3.2.1.1.7 Surveillance, Testing, and Maintenance

Logic system functional tests and simulated automatic operation of all channels shall be performed at least once per 18 months.

ECCS response times and trip function tests (see Tech Spec Table 3.3.3-3) shall be demonstrated to be within the limits at least once per 18 months.

Each ECCS actuation instrumentation channel shall be demonstrated operable per Table 4.3.3.1-1 in the Unit 2 Tech Specs.

Type C containment isolation valve leak rate test shall be performed once every 18 months.

A functional test and trip unit calibration of HPCS suction transfer on high suppression pool level instrument channels shall be performed monthly.

3.2.1.1.8 References

N2-OP-33, Rev. 4 "High Pressure Core Spray"
FSAR Section 6.3 Emergency Core Cooling Systems
PID 33A and PID 33B
GEK-83335A
Unit 2 Technical Specifications section 3/4.3 and 3/4.5

3.2.1.1.9 Initiating Event Potential

Inadvertent actuation of the CSH system would cause a plant trip. This is a low probability contribution to the plant trip initiating event.

3.2.1.1.10 Equipment Location

AOV108 and HCV120 are located inside containment. The spray sparger is located inside the reactor vessel. The condensate storage tank; expansion joints EJ11B, EJ11D; and valve V37 are in the Condensate Storage Tank Building. All other modeled equipment is in the Reactor Building or pipe tunnels.

3.2.1.1.11 Operating Experience

The Division III battery has been declared inoperable once since commercial operation. The battery required an equalizing charge. Since the event occurred during a shutdown, it does not affect the IPE model. However, there were multiple events prior to startup where the battery required an equalizing charge. It is assumed that the problem has been corrected.

3.2.1.1.12 Modeling Assumptions

1. The suppression pool strainers are sized to provide adequate NPSH to the pump even if the strainer is 50% clogged. There is no insulation in the wetwell and insulation in the drywell is covered with stainless steel. Therefore, the chances of insulation or panels reaching the downcomers on the drywell floor and then getting through the downcomers and reaching a strainer is considered a small contributor.

2. Failure of the system pressure pump (also referred to as water leg pump or line fill pump) is not modeled. Indications and alarms in the Control Room exist for low discharge line pressure, pump operation, and High Point Vent low water level. The system is also verified full monthly. The pressure pump would have to be unavailable for a long period of time before significant drainage would occur. Even with the

discharge piping empty, it is unlikely that the system piping would fail due to water hammer. The delayed delivery of water to the RPV is also assumed insignificant.

3. It is assumed that only one train of engine room ventilation is required for diesel operability. Although Tech Specs require both trains, LER 87-39 states that the diesel would start and run for a considerable amount of time with one ventilation train operable. It is assumed that the operation and/or maintenance personnel would have the other train operable before failure.

4. Pipe connections with double isolation or small lines are not modeled. In addition, pipe failures are not modeled. These failure modes are small contributors particularly for a standby single train system that is not cross-connected to redundant trains or systems.

5. The model includes the pressure transmitters, level transmitters, and switches required to actuate high pressure core spray. In addition, support system dependencies such as 120V AC and 125V DC are modeled in the support event tree. The unavailability of the analog logic circuitry, etc., is not modeled and presently is assumed to be a small contributor. The modeling of low RPV level and high drywell pressure inputs requires both signal trains for success. This is conservative for LOCA initiators.

6. The model includes 2 cycles between level 8, where the minimum flow return valve MOV105 must open, and level 2, where injection MOV107 must re-open. It is assumed that the operators take control of CSH after two cycles.

   Failure of MOV105 to close on level 2 is not modeled since it provides successful cooling and the bypass flow is small. Additionally, failure of MOV107 to close on level 8 is not modeled since overfilling is considered a potential success, and it is believed the operators will terminate flow before the main steam lines are reached. If the main steam lines fill, the condenser will not be available for a heat sink. If water is allowed to enter the RCIC steam line it will cause the RCIC system to become inoperable. The ADS valves may also become inoperative (may not re-close) as a result of water passing through the valves that are designed for steam passage.

7. The Division III emergency diesel including its engine control and field flashing as well as it's skid mounted support systems are modeled as a single component.

8. The failure impact of instrument lines used for Division III instruments were grouped as follows:

   - Break Inside of Drywell. This results in a Small break LOCA (SLOCA) with HPCS actuation, the auto closure of MOV107 fails because the line break prevents level from being sensed. HPCS fills the vessel to Level 8 and Operators have to manually close MOV107 or trip the HPCS pump off. Otherwise, the non-operating RCIC system steam line fills with water thereby preventing use at a future time. If the ADS valves are pushed open by HPCS, they may fail open; but this is considered a success for low pressure injection systems.

- Break outside of drywell at instrument rack. This would not result in a LOCA but a reactor trip may occur due to high flux since this event initiates HPCS on a perceived low water level. Any two of four instrument channels will initiate HPCS, and this break would drain two reference legs. HPCS will fail to auto stop at Level 8. A level 8 signal must be received from each loop of instrumentation, and with this type of break, one loop cannot read level 8. However, main turbine trip and isolation will work automatically.

9. Successful transfer of suction from the condensate tank to the suppression pool requires the closing of valve MOV101 and opening of valve MOV118. The valves are interlocked in such a way that valve MOV118 will not open until valve MOV101 closes. For this reason, the probability of MOV118 failing to open also includes the additional probability of MOV101 failing to close. Also, the high suppression pool level transmitters are not modeled. Any additional failure probability of the level transmitters is insignificant.

10. Division III circuit breakers are named by the ID they are found under in the Nine Mile Point Master Equipment List (MEL). The number in the breaker box on the drawings is not used, rather the breaker cubicle number is used. If information is desired on the breaker, it is listed by cubicle number in document control, and MEL.

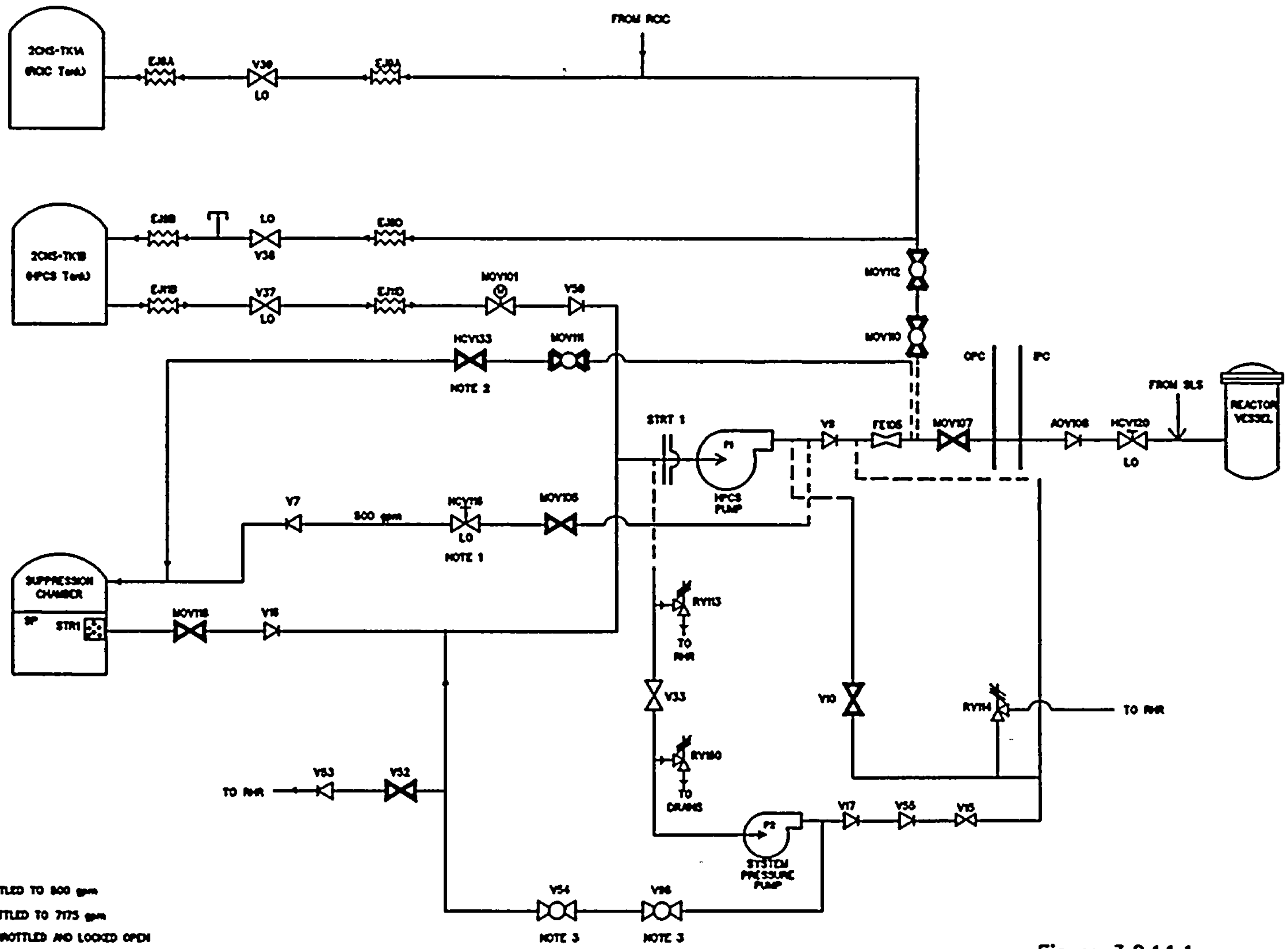
11. Relays K3 and K9 are modeled in a series configuration. K3 controls pump start. K9 controls the initiation seal in, suction valves, and injection valves. The K109 relay, although shown on figure 3.2.1.1-3, is not modeled. This relay returns the system from test configuration to minimum flow recirculation on an initiation signal.

12. The demand mode of the engine room ventilation dampers is opens on demand. On loss of power the dampers fail open. Requiring the dampers to open is conservative, as the dampers may already be open when needed. Once open, the dampers are expected to remain open during the entire event.

3.2.1.1.13 Logic Model and Results

The fault tree(s) is included in Tier 2 documentation, and is available upon request. Quantitative results are summarized in Section 3.3.5 (Tier 1).

Rev 0 (7/92)

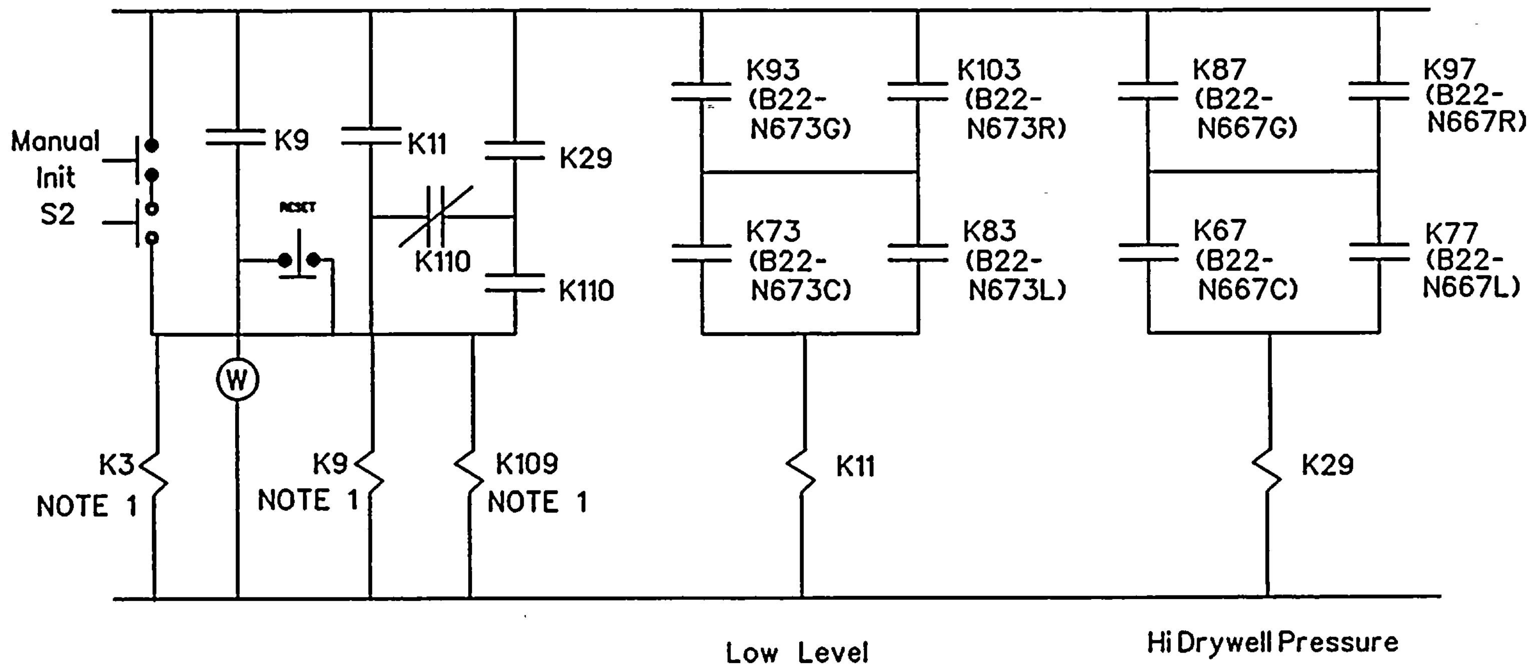


NOTES:

1. HCV116 IS THROTTLED TO 500 gpm
2. HCV133 IS THROTTLED TO 7175 gpm
3. V54, V96 ARE THROTTLED AND LOCKED OPEN

Figure 3.2.1.1-1
High Pressure Core Spray

Rev 0 (7/92)



Note 1: Relays K3 and K5 are modeled, Relay K109 is not modeled. See system summary assumptions for further explaination.

FIGURE 3.2.1.1-3
HPCS Start-up

HIGH PRESSURE CORE SPRAY
Component Block Descriptions

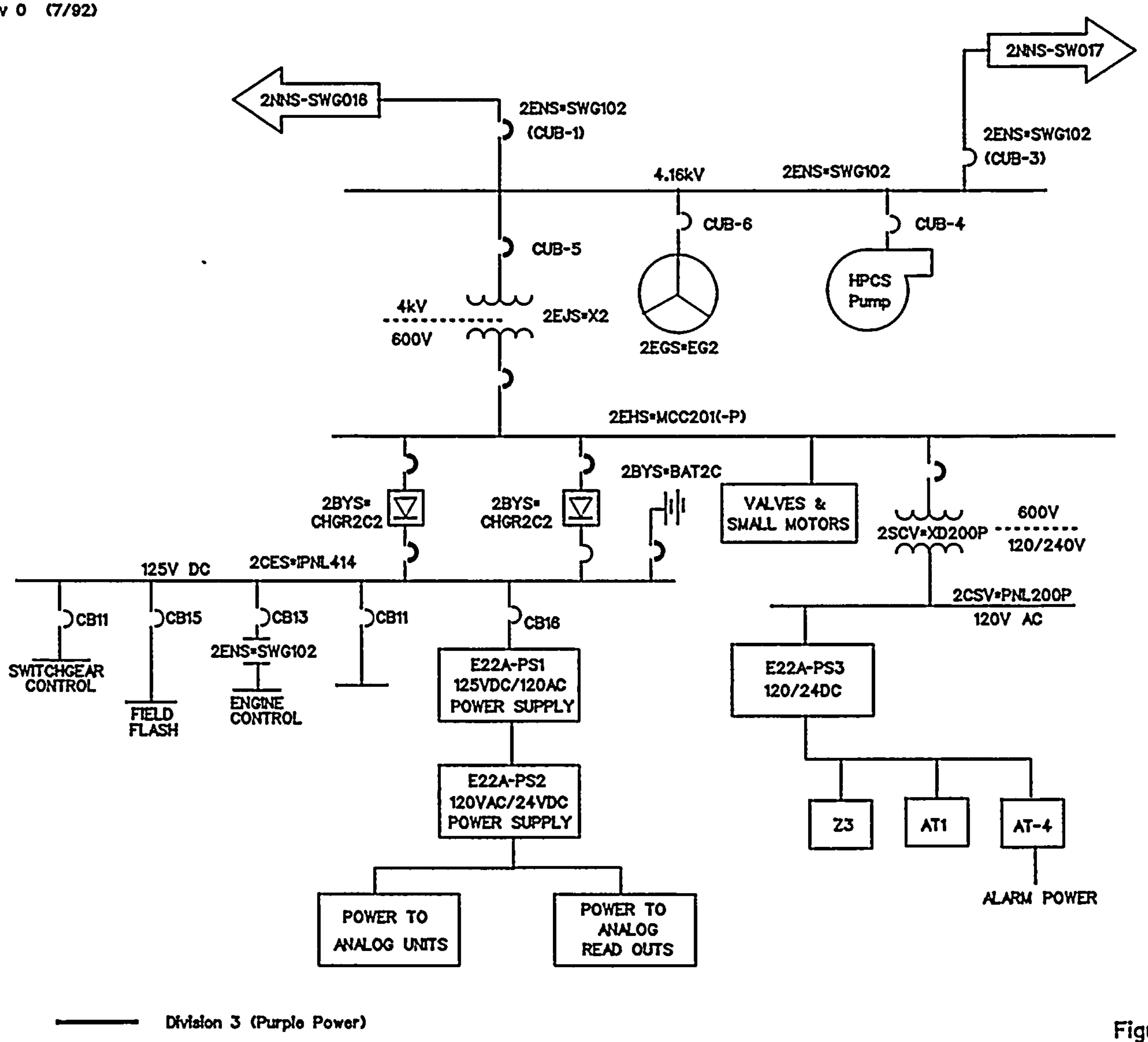
Block	Mark No. (Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
CST SUCTION PATH							
	2CNS-TK1B	CONDENSATE STOR. TK.	RUPTURE	N/A	N/A	N/A	N/A
	2CSH*EJ11B	EXPANSION JOINT	RUPTURES	N/A	N/A	N/A	N/A
	2CSH*V37	CST MANUAL VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CSH*EJ11D	EXPANSION JOINT	RUPTURES	N/A	N/A	N/A	N/A
	2CSH*MOV101	TANK SUCTION MOV	TRANSFER CLOSED	OPEN	CLOSED	2EHS*MCC201 (III	AS-IS
	2CSH*V59	CHECK VALVE	FAILS TO OPEN	CLOSED	N/A	N/A	N/A
	2CSH*LS3A	CST LEVEL SWITCH	FAILS TO TRANSFER	NA	NA	2CEC*PNL625	FAILS
	2CSH*LS3B	CST LEVEL SWITCH	FAILS TO TRANSFER	NA	NA	2CEC*PNL625	FAILS
SP SUCTION PATH							
	2CSH*PT102	PUMP SUCTION P.T.	FAILS LOW	N/A	N/A	N/A	N/A
	2CSH*PSL102	PT102 LOW P.I.S.	FAILS LOW	N/A	N/A	N/A	N/A
	2CSH*STRT1	STRAINER	PLUGGED	N/A	N/A	N/A	N/A
	2CSH*MOV118	SUPP. POOL SUCTION	FAILS TO OPEN	CLOSED	OPEN	2EHS*MCC201 (III	AS-IS
	2CSH*V16	SC CHECK VALVE	FAILS TO OPEN	CLOSED	N/A	N/A	N/A
PUMP							
	2CSH*P1	HPCS PUMP	FAILS TO START	STOPPED	RUNNING	2ENS*SWG102 (III	STOP
	2CSH*P1	HPCS PUMP	FAILS TO RUN	RUNNING	RUNNING	2ENS*SWG102 (III	STOP
	2HVR*UC403A	UNIT COOLER	FAILS	N/A	N/A	2EHS*MCC201	STOP
	2HVR*UC403A	UNIT COOLER	FAILS TO RUN	RUNNING	RUNNING	2EHS*MCC201	STOP
	2HVR*UC403A	UNIT COOLER	FAILS TO START	STOPPED	RUNNING	2EHS*MCC201	STOP
	2SWP*V165A	MANUAL VALVE	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2SWP*V567A	MANUAL VALVE	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2SWP*MOV15A	MOTOR OPERATED VALVE	TRANSFERS CLOSED	OPEN	CLOSED	2EHS*MCC201	AS-IS
	2HVR*UC403B	UNIT COOLER	FAILS	N/A	N/A	2EHS*MCC201	STOP
	2HVR*UC403B	UNIT COOLER	FAILS TO RUN	RUNNING	RUNNING	2EHS*MCC201	STOP
	2HVR*UC403B	UNIT COOLER	FAILS TO START	STOPPED	RUNNING	2EHS*MCC201	STOP
	2SWP*V165B	MANUAL VALVE	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2SWP*V567B	MANUAL VALVE	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2SWP*MOV15B	MOTOR OPERATED VALVE	TRANSFERS CLOSED	OPEN	CLOSED	2EHS*MCC201	AS-IS
MIN FLOW PATH							
	2CSH*MOV105	SUPPLY TO SUPPLY CH	FAILS TO OPEN	CLOSED	OPEN	2EHS*MCC201 (III	AS-IS
	2CSH*FE105	MOV105 FLOW ELEMENT	SETPOINT DRIFT	N/A	N/A	N/A	N/A
	2CSH*HCV116	HAND CONTROL VALVE	TRANSFER CLOSED	OPEN	OPEN	N/A	N/A
	2CSH*V7	CHECK VALVE	FAILS TO CLOSE	OPEN	N/A	N/A	N/A
INJECTION PATH							
	2CSH*V9	CHECK VALVE	FAILS TO OPEN	CLOSED	N/A	N/A	N/A
	2CSH*MOV107	INJECTION SHUTOFF	FAILS TO OPEN	CLOSED	OPEN	2EHS*MCC201 (III	AS-IS
	2CSH*MOV107	INJCT. SHUTOFF MOV	FAILS TO REOPEN	CLOSED	OPEN	2EHS*MCC201 (III	AS-IS
	2CSH*AOV108	TESTABLE CHECK VALVE	FAILS TO OPEN	CLOSED	OPEN	INSTRUMENT AIR	N/A
	2CSH*HCV120	HAND CONTROL VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
MASTER ACT. INST.							
	E22A-K3	MASTER RELAY	FAIL TO PICK UP	N/A	N/A	2CES*IPNL414	FAILS

Table 3.2.1.1-1 REV. 0 (7/92)

HIGH PRESSURE CORE SPRAY Component Block Descriptions							
Block	**Mark No. (Alt. ID)**	**Description**	**Failure Mode**	**Initial State**	**Actuated State**	**Support System**	**Loss of Support**
	E22A-K109	MASTER RELAY	FAIL TO PICK UP	N/A	N/A	2CES*IPNL414	FAILS
	E22A-K9	MASTER RELAY	FAIL TO PICK UP	N/A	N/A	2CES*IPNL414	FAILS
RPV LEVEL INSTRUMENTATION							
	E22A-K11	RPV LEVEL RELAY	FAIL TO PICK UP	N/A	N/A	2CES*IPNL414	FAILS
	2ISC*LT10A	RPV LEVEL XMTR		N/A	N/A	2CES*IPNL414	FAILS
	E22A-K83	LT10A RELAY	FAIL TO PICK UP	N/A	N/A	2CES*IPNL414	FAILS
	2ISC*LIS1673L	LT10A LEVEL I.SWITCH	FAIL TO ACTUATE	N/A	N/A	2CES*IPNL414	FAILS
	2ISC*LT10B	RPV LEVEL XMTR		N/A	N/A	2CES*IPNL414	FAILS
	E22A-K73	LT10B RELAY	FAIL TO PICK UP	N/A	N/A	2CES*IPNL414	FAILS
	2ISC*LIS1673C	LT10B LEVEL I.SWITCH	FAIL TO ACTUATE	N/A	N/A	2CES*IPNL414	FAILS
	2ISC*LT10C	RPV LEVEL XMTR		N/A	N/A	2CES*PNL414	FAILS
	E22A-K103	LT10C RELAY	FAIL TO PICK UP	N/A	N/A	2CES*IPNL414	FAILS
	2ISC*LIS1673R	LT10C L.I.S.	FAIL TO ACTUATE	N/A	N/A	2CES*IPNL414	FAILS
	2ISC*LT10D	RPV LEVEL XMTR		N/A	N/A	2CES*IPNL414	FAILS
	E22A-K93	LT10D RELAY	FAIL TO PICK UP	N/A	N/A	2CES*IPNL414	FAILS
	2ISC*LIS1673G	LT10D L.I.SWITCH	FAIL TO ACTUATE	N/A	N/A	2CES*IPNL414	FAILS
DRYWELL PRESSURE INST.							
	E22A-K29	DRYWELL PRESS. RELAY	FAIL TO PICK UP	N/A	N/A	2CES*IPNL414	FAILS
	2ISC*PT16A	D.W. PRESS XMTR		N/A	N/A	2CES*IPNL414	FAILS
	E22A-K77	PT16A RELAY	FAIL TO PICK UP	N/A	N/A	2CES*IPNL414	FAILS
	2ISC*PIS1667L	PT16A P.I.SWITCH	FAIL TO ACTUATE	N/A	N/A	2CES*IPNL414	FAILS
	2ISC*PT16B	D.W. PRESS XMTR		N/A	N/A	2CES*IPNL414	FAILS
	E22A-K67	PT16B RELAY	FAIL TO PICK UP	N/A	N/A	2CES*IPNL414	FAILS
	2ISC*PIS1667C	PT16B P.I.SWITCH	FAIL TO ACTUATE	N/A	N/A	2CES*IPNL414	FAILS
	2ISC*PT16C	D.W. PRESS XMTR		N/A	N/A	2CES*IPNL414	FAILS
	E22A-K97	PT16C RELAY	FAIL TO PICK UP	N/A	N/A	2CES*IPNL414	FAILS
	2ISC*PIS1667R	PT16C P.I.SWITCH	FAIL TO ACTUATE	N/A	N/A	2CES*IPNL414	FAILS
	2ISC*PT16D	D.W. PRESS XMTR		N/A	N/A	2CES*IPNL414	FAILS
	E22A-K87	PT16D RELAY	FAIL TO PICK UP	N/A	N/A	2CES*IPNL414	FAILS
	2ISC*PIS1667G	PT16D P.I.SWITCH	FAIL TO ACTUATE	N/A	N/A	2CES*IPNL414	FAILS
AC POWER SOURCE							
	2EGS*EG3	HPCS DIES. GENERATOR	FAILS TO START	STOPPED	RUNNING		STOP
	2EGS*EG3	HPCS DIES. GENERATOR	FAILS TO RUN	STOPPED	RUNNING		STOP
	2ENS*SWG102-4	CIRCUIT BREAKER	FAILS TO CLOSE	OPEN	CLOSED	2CES*IPNL414	FAILS
	2ENS*SWG102-4	CIRCUIT BREAKER	TRANSFER OPEN	CLOSED			
	2ENS*SWG102-1	CIRCUIT BREAKER	FAILS TO CLOSE	OPEN	CLOSED	2CES*IPNL414	FAILS
	2NNS-SWG016	SW. GEAR FROM YARD	FAILS	CLOSED		2CES*IPNL414	FAILS
	2ENS*SWG102-6	CIRCUIT BREAKER	TRANSFER OPEN	CLOSED		2CES*IPNL414	FAILS
	2ENS*SWG102-3	CIRCUIT BREAKER	FAILS TO CLOSE	OPEN	CLOSED	2CES*IPNL414	FAILS
	2EJS*X2	STEP DOWN TRANSFORM.					
	2SWP*MOV94A	DIES. COOLING OUTLET	FAILS TO OPEN	CLOSED	OPEN	2EHS*MCC201 (III	AS-IS
	2SWP*MOV94B	DIES. COOLING OUTLET	FAILS TO OPEN	CLOSED	OPEN	2EHS*MCC201 (III	AS-IS
	2SWP*MOV95A	MOTOR OPERATED VALVE	TRANSFERS CLOSED	OPEN	CLOSED	2EHS*MCC103	AS-IS

HIGH PRESSURE CORE SPRAY Component Block Descriptions							
Block	**Mark No. (Alt. ID)**	**Description**	**Failure Mode**	**Initial State**	**Actuated State**	**Support System**	**Loss of Support**
	2SWP*V260	CHECK VALVE	FAILS TO OPEN	CLOSED	N/A	N/A	N/A
	2SWP*942A	MANUAL VALVE	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2SWP*MOV95B	MOTOR OPERATED VALVE	TRANSFERS CLOSED	OPEN	CLOSED	2EHS*MCC303	AS-IS
	2SWP*V259	CHECK VALVE	FAILS TO OPEN	CLOSED	N/A	N/A	N/A
	2SWP*V942B	MANUAL VALVE	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
DC POWER SOURCE							
	2BYS*BAT2C	BATTERY					
	2EHS*MCC201-5C	CIRCUIT BREAKER	TRANSFER OPEN	CLOSED			
	2BYS*CHGR2C2	BAT2C CHARGER2				2EHS*MCC201 (III	FAILS
	2EHS*MCC201-4B	CIRCUIT BREAKER	TRANSFER OPEN	CLOSED			
	2BYS*CHGR2C1	BAT2C CHARGER1				2EHS*MCC201	FAILS
EDG2 ROOM COOLING							
	2HVP*AOD5A	INLET AIR OPERATED DAMPER	FAILS TO OPEN	CLOSED	OPEN	INSTRUMENT AIR	FAILS OPEN
	2HVP*MOD2A	OUTLET MOTOR OPERATED DAMPER	FAILS TO OPEN	CLOSED	OPEN	2SCV*PNL200P	FAILS OPEN
	2HVP*FN2A	EXHAUST FAN	FAILS TO START	STOPPED	RUNNING	2EHS*MCC201	STOP
	2HVP*FN2A	EXHAUST FAN	FAILS TO RUN	RUNNING	RUNNING	2EHS*MCC201	STOP
	2HVP*AOD5B	INLET AIR OPERATED DAMPER	FAILS TO OPEN	CLOSED	OPEN	INSTRUMENT AIR	FAILS OPEN
	2HVP*MOD2B	OUTLET MOTOR OPERATED DAMPER	FAILS TO OPEN	CLOSED	OPEN	2SCV*PNL200P	FAILS OPEN
	2HVP*FN2B	EXHAUST FAN	FAILS TO START	STOPPED	RUNNING	2EHS*MCC201	STOP
	2HVP*FN2B	EXHAUST FAN	FAILS TO RUN	RUNNING	RUNNING	2EHS*MCC201	STOP

Rev 0 (7/92)



References: EE-M01A-5
EE-M01B-4
EE-M01E-5
EE-M01F-4

Figure 3.2.1.1-2
Div. 3 Power (HPCS only)

I:\NMP2IPE\DGN\2S01.DGN Jul. 06, 1992 13:07:04

System 2

Reactor Core Isolation Cooling

3.2.1.2 Reactor Core Isolation Cooling

3.2.1.2.1 System Function

The Reactor Core Isolation Cooling System (RCIC or ICS) is designed to ensure that sufficient reactor vessel inventory and cooling is maintained. Reactor vessel water is maintained or supplemented by the RCIC system during the following conditions:

1. When the reactor vessel is in hot standby, isolated from the condenser, and the feedwater system is not in operation.

2. RCIC can be used for Reactor Coolant System (RCS) makeup while using the steam condensing mode of the Residual Heat Removal (RHR) system. The RCIC pump takes suction from the heat exchanger, with additional makeup from the condensate storage tank. However, steam condensing with RCIC makeup has not been modeled because it is a shutdown function.

In the system model, the RCIC system is considered redundant to the High Pressure Core Spray (CSH) system for transients and small LOCAs.

A simplified drawing of the RCIC system is provided in Figure 3.2.1.2-1.

3.2.1.2.2 Success Criteria

Top Event IC represents the RCIC system and is included in the transient, small LOCA, and ATWS event tree models. To be successful, RCIC must provide an adequate supply of water to the Reactor Pressure Vessel (RPV) from either the Condensate Storage Tanks (CSTs) or the Suppression Pool. However, transfer to the suppression pool is not modeled.

In the ATWS event tree model, top event IL represents the operator actions associated with keeping the RCIC system operable (e.g., defeating RPV low pressure interlock and avoiding the occurrence of RCIC high turbine exhaust backpressure trip).

RCIC is also included in the station blackout (SBO) event tree model. The following summarizes the SBO top events and success criteria:

Top Event	Success Criteria
OA	Operator conducts DC load shedding and bypasses RCIC isolation trip circuits within the first two hours. Otherwise, RCIC and all vessel makeup is assumed to fail at two hours.
U1	RCIC operates for the first two hours of the scenario.
U2	Given U1 success, RCIC continues operating from hour 2 to hour 8 into the scenario (i.e., 6 hours).

Top Event	Success Criteria
U3	Given U2 success, RCIC continues operating from hour 8 to hour 10 (i.e., 2 hours). No credit is given to operating RCIC after 10 hours due to depletion of the DC battery providing system control power.

3.2.1.2.3 Support Systems

Motor operated valves (MOVs) in the RCIC system require 600V MCC power supplies modeled in the A1 & A2 top events, as shown in Table 3.2.1.2-1. However, these valves are normally open and fail as is on a loss of power.

Other auxiliary systems which support the RCIC system are the gland seal system (which prevents turbine steam leakage) and the lube oil cooling water system. Loss of the gland seal system would cause inconsequential steam leakage in the turbine and would not degrade RCIC system operation.

Loss of the lube oil cooling water system would eventually overheat and fail the RCIC pump. This is included in the model. The pressure control valve PCV115 in the lube oil cooling loop fails open on loss of Division I 125V DC (D1) and results in maximum lube oil cooling. Air operated drain valves fail closed and are not required for system success (not modeled). DC motor operated valves fail as is on loss of support.

Automatic initiation of RCIC is dependent on RPV Level 2 instrumentation loops as modeled in the ECCS actuation system (top events E1 and E2 in the Support System event tree). Either division may initiate RCIC.

RCIC will isolate and terminate injection on an ECCS level 8 signal. It will re-inject on a level B2 signal.

RCIC control requires vital bus UPS2A (2VBS*PNL101A). Failure of UPS2A causes flow controls and governor to go to maximum flow and also disables the ability to isolate on RPV Level 8. UPS2B is also required to isolate on Level 8.

3.2.1.2.4 System Operation

The RCIC system is in the standby mode during normal operation. However, the system pressure pump, 2ICS*P2, is operating to maintain system water pressure. The RCIC system is automatically initiated by low water level (RPV Level 2) utilizing a one-out-of-two twice logic. On an RPV Level 8 signal, RCIC stops injecting to the vessel. The system does not isolate because of a Level 8 signal. Valves 2ICS*MOV120 (steam admission valve to turbine) and 2ICS*MOV126 (injection valve) close to stop injection. When the valves are closed the pump stops. The pump may be re-started when a Level 2 signal or an operator re-opens the valves.

All automatic valves in the RCIC system have remote manual capability so the entire system can be operated and tested from the main Control Room.

The system is capable of initiation, independent of AC power, provided the normally open steam supply valves (2ICS*MOV121 and 2ICS*MOV128) are open. 2ICS*MOV121 is controlled by Division I signals, 2ICS*MOV128 is controlled by Division II signals. These valves will automatically close given any of the following conditions:

- High RCIC turbine exhaust diaphragm pressure (> 10 psig),
- RCIC steam supply pressure 75 psig or less,
(The Emergency Operating Procedures (EOPs) may require defeating this isolation signal, this is done by removing relay E51-K86 in P618 and relay E51-K78 in P621 per N2-EOP-6, Att. 2.)
- RCIC steam line differential pressure high (setpoint values are dependant on sensor location),
- RCIC equipment room temperature above 135°F,
- RCIC pipe chase temperature above 135°F,
- RHR equipment area high ambient temperature,
- Reactor building high ambient temperature,
- Reactor building pipe chase high ambient temperature, or
- Manual signals.

On initiation, the RCIC system normally draws water from the condensate storage tanks (2CNS*TK1A and TK1B). The 135,000 gallons of water in TK1A is dedicated to RCIC. In addition, there is a crosstie to 2CNS*TK1B which provides an additional source of water, if necessary. When the water level in the condensate storage tank drops below 6.15 ft. (2ICS*LT3A or 2ICS*LT3B), the system automatically switches to the suppression pool.

Minimum flow to the suppression pool (MOV143 opens) is initiated for RCIC pump protection when the pump discharge flow is less than 100 gpm and the discharge pressure is greater than 125 psig. When 2ICS*MOV120 (RCIC turbine steam supply) is full shut, 2ICS*MOV143 closes.

The turbine steam supply can be shutoff by tripping 2ICS*MOV150 remotely from the control room. The turbine is shutdown (tripped) in the event of any of the following signals:

- Division I or Division II isolation
- Turbine exhaust pressure high
- Pump suction pressure low
- Turbine overspeed

- Level 8 - High reactor water level

The turbine trip valve can be remotely reset for all shutdown signals except turbine overspeed. For turbine overspeed, the overspeed trip latch must be reset at the turbine (locally).

3.2.1.2.5 Instrumentation and Controls

The RCIC system can be manually initiated from the Control Room. This initiation is normally used for testing purposes. Manual controls and status indication are provided in the Control Room for all motor operated valves (MOVs) and air operated valves (AOVs). In addition, the following RCIC parameter indications are provided in the Control Room. The indications alert the operator to a component failure or inoperable status of a component. This allows the operator to control RCIC operation over its full spectrum of operating conditions:

2ICS*FI101	RCIC System Injection Flow Indication
2ICS*PI103	RCIC Steam Supply Pressure Indication
E51-C002-M1	RCIC Turbine Speed Indication
2ICS*PI104	RCIC Pump Discharge Pressure Indication
2ICS-LS221	RCIC High Point Vent Level Low Alarm
2ICS-C1	Gland Seal System Operating Parameters
2ICS-PT1A(B)	RCIC Turbine Exhaust Pressure

The following alarms alert the operator to equipment malfunctions that occur during RCIC operation:

Loss of power to any DC MOV
Actuation of the thermal overloads for any DC MOV
Loss of 24 V DC power
RCIC High Point Vent Low Level

3.2.1.2.6 Technical Specifications

With the RCIC system INOPERABLE, and the High Pressure Core Spray (CSH) system operable, operation may continue provided that the RCIC system be restored to OPERABLE status within 14 days. Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressures to 150 psig or less within the following 24 hours.

3.2.1.2.7 Surveillance, Testing and Maintenance

RCIC is intentionally declared INOPERABLE for monthly system actuation channel, functional tests. Channel instrumentation is tested and calibrated every 31 days. For more information on channel testing and calibration, see Table 4.3.5.1-1 in the Tech Specs.

RCIC system equipment shall be demonstrated OPERABLE at least once every 31 days by checking high point vents, valve alignments, valves which must actuate and are not in a locked or inoperable (i.e., breaker out) position, and verifying the pump flow controller is in the correct position.

RCIC is proven operable, with a simulated cold start, once per quarter. It is also demonstrated operable at least once every 18 months by performing a simulated automatic initiation and restart. The automatic transfers of suction to the suppression pool on low condensate storage tank water level is also verified.

Most RCIC valves which may be required to change position in response to a RCIC actuation are stroke tested quarterly. The remaining valves, 2ICS*MOV126, AOV156 and AOV157 are stroke tested at cold shutdown only.

Isolation system response times are verified once every 18 months. Instrumentation calibration and operation are done according to Tech Spec Table 4.3.2.1-1.

3.2.1.2.8 References

N2-EOP-6, Att. 2, Rev. 0
N2-EOP-RPV, Rev. 4 "RPV Control"
Operating Procedure N2-OP-35, Rev 03, "Reactor Core Isolation Cooling."
USAR section 7.4, "Systems Required for Safe Shutdown," Rev. 1.

Tech Spec:	3/4.3.2
	3/4.3.5
	3/4.6.1
	3/4.6.3
	3/4.7.4
Drawings:	12177-ESK-6ICS01 Rev. 6
	12177-ESK-6ICS02 Rev. 10
	12177-ESK-6ICS03 Rev. 8
	12177-ESK-6ICS04 Rev. 8
	PID's referenced on Figure 3.2.1.2-1

3.2.1.2.9 Initiating Event Potential

Premature actuation of the system could result in a plant SCRAM due to high neutron flux. This is a low probability contributor to the transient initiating event frequency.

A pipe break would result in a plant SCRAM and loss of inventory to the vessel. A steam line break (to the turbine), including the drain pot, would result in a plant SCRAM, a loss of inventory, and isolation of the RCIC steam line.

3.2.1.2.10 Equipment Location

All major equipment is inside the reactor building with the exception of the condensate storage tanks, located in the CST building and valves 2ICS*AOV157 and 2ICS*MOV128, which are inside primary containment.

3.2.1.2.11 Operating Experience

A common failure at Nine Mile Point Unit 2, is to trip due to turbine overspeed on cold quick start. The cause of the overspeed is usually the governor valve binding or water in the turbine. On restart after reset, the turbine may clear the water and succeed on the second try, but if the governor valve is binding, a successful start may take several attempts. The overspeed trip must be reset locally each time.

Extensive Limitorque work was done on the RCIC system with the unit at power. The repair work accounts for much of the system's unavailability. This is not expected to be a normal occurrence, however, the data is included until better availability is demonstrated.

3.2.1.2.12 Modeling Assumptions

1. Lube oil cooling is modeled - it is assumed the RCIC Pump (2ICS*P1) will fail without lube oil cooling.

2. Relief valve 2ICS*RV114 was ignored because of its small size.

3. RCIC is dependant on Division I 125V DC. Failure of Division I 125V DC, after successful RCIC initiation, could result in overfilling the RPV (exceeding Level 8) if additional failures occur. If AC power or vital UPS power is available, operators can prevent overfilling. With AC power available, the operators can manually close steam isolation valves. If vital UPS buses are available, operators can control level if vital UPS buses are available. The frequency of this overfilling condition is considered low. The consequences of overfilling the RPV (i.e., filling steam lines with water) is a potential LOCA outside containment if a pipe break occurs. This is also judged unlikely, therefore these additional failure modes and consequences are not modeled.

4. Failure of instrument lines associated with RCIC break detection is not modeled. An instrument line break inside containment would result in a small LOCA, it is more likely that RCIC would become unavailable due to RCIC isolation. The frequency of this initiator (instrument line break) which potentially fails RCIC is judged to be small in comparison to the frequency of the existing small LOCA initiating event with subsequent failure of RCIC. An instrument line break outside containment would most likely cause automatic isolation and terminate the break. This is a small contributor to system unavailability. An initiating event does not occur unless the instrument line is

not isolated due to additional failures. The frequency and consequences of such an event are also judged insignificant.

5. Closure of 2ICS*MOV150 as the result of a spurious control system actuation was judged to be unlikely. Spurious operation is annunciated immediately and therefore the contribution to standby unavailability is small. Spurious actuation of a normally de-energized relay is considered very unlikely as compared to other system failure modes and is not modeled.

6. Minimum flow to the suppression pool is not modeled. Low flow conditions which require this (pump) protection are considered unlikely and if MOV143 fails open, flow diversion is not significant enough to prevent success.

7. The pressure maintenance system is not modeled. The possibility of the piping not being full is considered low. There is a monthly high point surveillance, a high point vent low level alarm, and pressure pump discharge pressure indication or alarms. The pressure pump would have to be inoperable for a long period of time without being detected to affect system operation.

8. The model requires RCIC to successfully cycle twice between level 8 and level 2. Given this success, it is assumed that the operators would have taken manual control by this time, and mechanical failures dominate. Failure of injection MOV126 to close on Level 8 is not modeled because there are two check valves downstream to prevent backflow. Also, failure of the 2 inch steam start-up line MOV159 to close on Level 8 is not modeled because this would not be a system failure. However, failure of the turbine steam inlet MOV120 to close is included in the model as a failure, and failure of start-up line MOV159 and injection MOV126 to reopen on Level 2 is modeled as a RCIC failure.

9. RCIC is modeled to fail if either of the condensate storage tanks (CST A and B) are not available. 135,000 gallons of CST A are dedicated to RCIC and automatic transfer to the suppression pool occurs on low CST level. This was done because both tanks are connected above the protected 135,000 gallons and the frequency of not having sufficient water from both tanks to continue injecting for 24 hours is small in comparison to the unavailability of RCIC. In addition, N2-EOP-RPV, Section RL instructs operators to continue using the CST source, if available. Because the model requires injection from the CSTs for up to 24 hours, failure modes associated with transfer to the suppression pool are not modeled.

10. Failure of RCIC turbine protective devices are not modeled. The frequency of protective device demand and failure is small in comparison to existing failure modes that are modeled.

3.2.1.2.13 Logic Model and Results

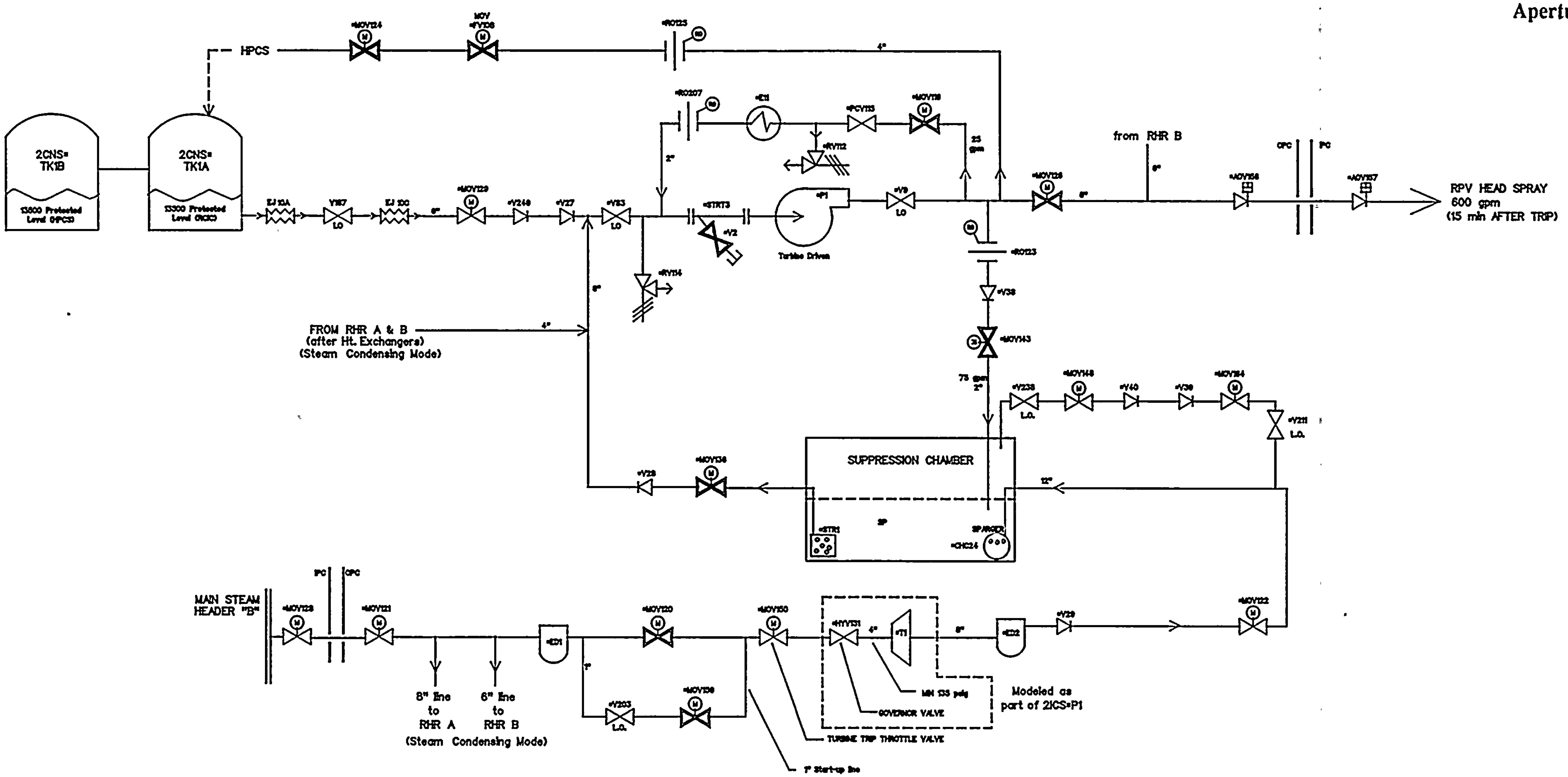
The fault tree(s) is included in Tier 2 documentation, and is available upon request. Quantitative results are summarized in Section 3.3.5 (Tier 1).

Rev 0 (7/92)

SI APERTURE CARD

Also Available On Aperture Card

References: PID-35A-7
PID-35B-9
PID-35C-11
PID-35D-7



These components are modeled in the RCIC Fault Tree.

Figure 3.2.1.2-1
Reactor Core Isolation Cooling

I:\NMP2IPE\DGN\2S02.DGN Jul. 13, 1992 15:53:32

9208050191 - 13

REACTOR CORE ISOLATION COOLING
Component Block Descriptions

Block	Mark No.	(Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
INITIATION SIGNAL								
	E51A-K2	(K2-2ICSN16)	RELAY	FAILS TO PICK UP	DE-ENERG	ENERGIZE	2BYS*PNL201A	DE-ENERGIZED
	E51A-K3	(K3-2ICSN16)	RELAY	FAILS TO PICK UP	DE-ENERG	ENERGIZE	2BYS*PNL201A	DE-ENERGIZED
	B22-N692A	(2ISC*LS1692A)	DIV I LEVEL SWITCH	FAILS TO PICK UP	DE-ENERG	ENERGIZE	2VBS*PNL101A	DE-ENERGIZED
	E51A-K62		LS1692A RELAY	FAILS TO PICK UP	DE-ENERG	ENERGIZE	2VBS*PNL101A	DE-ENERGIZED
	B22-N692E	(2ISC*LS1692E)	DIV I LEVEL SWITCH	FAILS TO PICK UP	DE-ENERG	ENERGIZE	2VBS*PNL101A	DE-ENERGIZED
	E51A-K60		LS1692E RELAY	FAILS TO PICK UP	DE-ENERG	ENERGIZE	2BYS*PNL201A	DE-ENERGIZED
	B22-N692F	(2ISC*LS1692F)	DIV II LEVEL SWITCH	FAILS TO PICK UP	DE-ENERG	ENERGIZE	2VBS*PNL301B	DE-ENERGIZED
	E51A-K16	(K16-2ICSN16)	LS1692F RELAY	FAILS TO PICK UP	DE-ENERG	ENERGIZE	2BYS*PNL201A	DE-ENERGIZED
	E12A-AT8/1		OPTICAL ISOLATOR	FAILS TO PICK UP	DE-ENERG	ENERGIZE	2BYS*PNL201A	DE-ENERGIZED
	B22-N692B	(2ISC*LS1692B)	DIV II LEVEL SWITCH	FAILS TO PICK UP	DE-ENERG	ENERGIZE	2BYS*PNL201A	DE-ENERGIZED
	E51A-K12		LS1692B RELAY	FAILS TO PICK UP	DE-ENERG	ENERGIZE	2BYS*PNL201A	DE-ENERGIZED
	E12A-AT8/2		OPTICAL ISOLATOR	FAILS TO PICK UP	DE-ENERG	ENERGIZE	2BYS*PNL201A	DE-ENERGIZED
MAIN STEAM PATH								
	2ICS*MOV128		INBOARD ISOLATION MOV	TRANSFER CLOSED	OPEN	CLOSED	2EHS*MCC302	AS-IS
	E31-N083A	(2ICS*PDT5A)	DIFF PRESS TRANSMITTER	FAILS HIGH	N/A	N/A	2VBS*PNL101A	FAILS LOW
	E31-N084A	(2ICS*PDT167)	DIFF PRESS TRANSMITTER	FAILS HIGH	N/A	N/A	2VBS*PNL101A	FAILS LOW
	E31A-K2A	(K2A-2RPSA01)	MOV128 RELAY	SPURIOUSLY CLOSES	DE-ENERG	ENERGIZE	2VBS*PNLA103	DE-ENERGIZED
	E31A-K5A	(K5A-2RPSA01)	MOV128 RELAY	SPURIOUSLY CLOSES	DE-ENERG	ENERGIZE	2VBS*PNLA103	DE-ENERGIZED
	2ICS*MOV121		OUTBOARD ISOLATION MOV	TRANSFER CLOSED	OPEN	CLOSED	2EHS*MCC102	AS-IS
	E31-N083B	(2ICS*PDT5B)	DIFF PRESS TRANSMITTER	FAILS HIGH	N/A	N/A	2VBS*PNL301B	FAILS LOW
	E31-N084B	(2ICS*PDT168)	DIFF PRESS TRANSMITTER	FAILS HIGH	N/A	N/A	2VBS*PNL301B	FAILS LOW
	E31A-K2B	(K2B-2RPSB01)	MOV121 RELAY	SPURIOUSLY CLOSES	DE-ENERG	ENERGIZE	2VBS*PNLB103	DE-ENERGIZED
	E31A-K5B	(K5B-2RPSB01)	MOV121 RELAY	SPURIOUSLY CLOSES	DE-ENERG	ENERGIZE	2VBS*PNLB103	DE-ENERGIZED
	2ICS*ED1		DRAIN POT	RUPTURES	N/A	N/A	N/A	N/A
	2ICS*MOV120		TURBINE SUPPLY VALVE	FAILS TO OPEN	CLOSED	OPEN	2DMS*MCCA1	AS-IS
	2ICS*V203		LO GLOBE VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2ICS*MOV159		1" WARM-UP LINE, SUPPLY VALVE	FAILS TO OPEN	CLOSED	OPEN	2DMS*MCCA1	AS-IS
	E51A-K96		WARM-UP LINE RELAY	FAILS TO PICK UP	DE-ENERG	ENERGIZE	2BYS*PNL201A	DE-ENERGIZED
	2ICS*MOV150		TURBINE THROTTLE MOV	TRANSFER CLOSED	OPEN	CLOSED	2DMS*MCCA1	AS-IS
	2ICS*ED2		DRAIN POT	RUPTURES	N/A	N/A	N/A	N/A
	2ICS*V29		CHECK VALVE	FAILS TO OPEN	OPEN	N/A	N/A	N/A
	2ICS*MOV122		TURBINE EXHAUST MOV	TRANSFER CLOSED	OPEN	CLOSED	2DMS*MCCA1	AS-IS
CST SUCTION PATH								
	2CNS-TK1A		COND. STORAGE TANK	RUPTURES	N/A	N/A	N/A	N/A
	2ICS-EJ10A		EXPANSION JOINT	RUPTURES	N/A	N/A	N/A	N/A
	2ICS-V187		L.O. GATE VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2ICS-EJ10C		EXPANSION JOINT	RUPTURES	N/A	N/A	N/A	N/A
	2ICS*MOV129		TANK SUCTION MOV	TRANSFER CLOSED	OPEN	CLOSED	2DMS*MCCA1	AS-IS
	2ICS*V249		CHECK VALVE	FAILS TO OPEN	OPEN	N/A	N/A	N/A
	2ICS*V27		CHECK VALVE	FAILS TO OPEN	OPEN	N/A	N/A	N/A
PUMP								
	2ICS*V83		L.O. GATE VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2ICS*STRT3		TEMPORARY STRAINER	PLUGS	N/A	N/A	N/A	N/A
	2ICS*P1		TURBINE DRIVEN PUMP	FAILS TO START	STOPPED	RUNNING	2ICS*T1	STOP

REACTOR CORE ISOLATION COOLING
Component Block Descriptions

Block	Mark No. (Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
	"	"	FAILS TO RUN	"	"	"	"
	2ICS*V9	L.O. GATE VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2ICS*MOV116	COOLER SUPPLY VALVE	FAILS TO OPEN	CLOSED	OPEN	2DHS*MCCA1	AS-IS
	2ICS*PCV115	LO LUBE OIL COOL PCV	TRANSFER CLOSED	OPEN	CLOSED	2DHS*MCCA1	OPEN
	2ICS*E1	LUBE OIL COOLER	PLUGGED	N/A	N/A	N/A	N/A
	"	"	EXCESSIVE LEAK	N/A	N/A	N/A	N/A
	2ICS*RO207	RESTRICTION ORIFACE	RUPTURES	N/A	N/A	N/A	N/A
RPV INJECTION PATH							
	2ICS*MOV126	INJECT SHUTOFF MOV	FAILS TO OPEN	CLOSED	OPEN	2DHS*MCCA1	AS-IS
	E51A-K20 (K20-2ICSN16)	MOV126 RELAY	FAILS TO PICK UP	DE-ENERG	ENERGIZE	2BYS*PNL201A	DE-ENERGIZED
	2ICS*AOV156	OUTSIDE ISOLATION	FAILS TO OPEN	CLOSED	OPEN	INSTRUMENT AIR	N/A
	2ICS*AOV157	INSIDE ISOLATION	FAILS TO OPEN	CLOSED	OPEN	INST. NITROGEN	N/A
STOP ON RPV LEVEL 8							
	2ICS*MOV120	STEAM SUPPLY TO TURBINE	FAILS TO CLOSE	CLOSED	OPEN	2DHS*MCCA1	AS-IS
	E51A-K95 (K95-2ICSN16)	RELAY	FAILS TO PICK UP	DE-ENERG	ENERGIZE	2BYS*PNL201A	DE-ENERGIZED
	E51-F045 (2ICS*MOV120-33)	LIMIT SWITCH	FAILS TO OPEN	OPEN	CLOSED	NA	N/A
	B22-N693A (2ICS*LS1693A)	DIV I LEVEL SWITCH	FAILS TO PICK UP	DE-ENERG	ENERGIZE	2VBS*PNL101A	DE-ENERGIZED
	E51A-K62 (K62-2ICSN08)	LS1693A RELAY	FAILS TO PICK UP	DE-ENERG	ENERGIZE	2VBS*PNL101A	DE-ENERGIZED
	B22-N693E (2ICS*LS1693E)	DIV I LEVEL SWITCH	FAILS TO PICK UP	DE-ENERG	ENERGIZE	2VBS*PNL101A	DE-ENERGIZED
	E51A-K116 (K116-2CSLN08)	LS1693E RELAY	FAILS TO PICK UP	DE-ENERG	ENERGIZE	2VBS*PNL101A	DE-ENERGIZED
	B22-N693F (2ICS*LS1693F)	DIV II LEVEL SWITCH	FAILS TO PICK UP	DE-ENERG	ENERGIZE	2VBS*PNL101A	DE-ENERGIZED
	E51A-K115 (K115-2ICSN16)	RELAY	FAILS TO PICK UP	DE-ENERG	ENERGIZE	2VBS*PNL301B	DE-ENERGIZED
	E12A-AT8/3	E51A-K115 ISOLATOR	FAILS TO PICK UP	DE-ENERG	ENERGIZE	2VBS*PNL201A	DE-ENERGIZED
	B22-N693B (2ICS*LS1693B)	DIV II LEVEL SWITCH	FAILS TO PICK UP	DE-ENERG	ENERGIZE	2VBS*PNL301B	DE-ENERGIZED
	E51A-K14 (K14-2ICSN16)	RELAY	FAILS TO PICK UP	DE-ENERG	ENERGIZE	2BYS*PNL201A	DE-ENERGIZED
	E12A-AT8/4	E51A-K14 ISOLATOR	FAILS TO PICK UP	DE-ENERG	ENERGIZE	2VBS*PNL201A	DE-ENERGIZED
RESTART ON LEVEL 2							
	E51A-K2 (K2-2ICSN16)	RELAY	FAILS TO PICK UP	DE-ENERG	ENERGIZE	2BYS*PNL201A	DE-ENERGIZED
	E51A-K97 (K97-2ICSN16)	TIMING RELAY	FAILS TO PICK UP	DE-ENERG	ENERGIZE	2BYS*PNL201A	DE-ENERGIZED
	2ICS*MOV159	1" WARM-UP LINE	FAILS TO REOPEN	CLOSED	OPEN	2DHS*MCCA1	AS-IS
	2ICS*MOV120	TURBINE THROTTLE	FAILS TO REOPEN	CLOSED	OPEN	2DHS*MCCA1	AS-IS
	2ICS*P1	RCIC PUMP	FAILS TO RESTART	STOPPED	START	2ICS*T1	STOP
	2ICS*MOV126	INJECTION SHUTOFF	FAILS TO REOPEN	CLOSED	OPEN	2DHS*MCCA1	AS-IS

System 3

Low Pressure Core Spray

3.2.1.3 Low Pressure Core Spray

3.2.1.3.1 System Function

The purpose of the LPCS (CSL) is to provide low-pressure reactor vessel core spray following a LOCA when the vessel has been depressurized and vessel water level has not been restored by the HPCS. The LPCS is functionally diverse from the LPCI mode of the RHR system.

3.2.1.3.2 Success Criteria

To provide adequate flow to the RPV for 24 hours. Low pressure core spray is modeled in the front line event trees as top event LS. A simplified diagram is provided in Figure 3.2.1.3-1.

3.2.1.3.3 Support Systems

Emergency AC Division I, and the suppression pool must be available for system operation. For a complete list of components and their dependencies, see table 3.2.1.3-1, attached.

Actuation of LPCS is common to LPCI "A" actuation. This support system is modeled in the support tree as top event E1.

The following systems interface with or connect to the LPCS system:

- Condensate Makeup and Draw-off (CNS)
- Residual Heat Removal (RHS)

3.2.1.3.4 System Operation

The LPCS system is in standby during normal plant operations.

To facilitate automatic operation, and to operate the pressure maintenance system, the suction isolation valve (MOV112) is normally open. The LPCS pump is normally in standby, but the system pressure maintenance pump is always running to keep the system full and pressurized. The minimum flow valve (MOV107) is normally open, and the test line isolation valve (FV114) is normally closed.

The LPCS system will automatically initiate on either;

- Reactor Vessel Level 1, or
- High Primary Containment Pressure of 1.68 psig.

Automatic actuation causes the following actions:

1. Normally closed test return valve FV114 receives a close signal.

2. When power is available at the pump motor bus, the pump motor breaker closes and the pump starts.

3. Injection valve MOV104 opens when the differential pressure between the pump discharge and RPV is satisfied.

When the pump is running and the discharge flow is low, differential pressure transmitters on the pump discharge signal the minimum flow valve MOV107 to open for pump protection. However, MOV107 is normally in the open position.

For manual initiation a two-position switch, with a disarmed/armed maintained contact collar and an initiation push button, is provided on Panel P601. (This push button provides Division I ECCS initiation. In addition to LPCS, the Division I Diesel and LPCI A will initiate).

Upon initiation, the injection isolation valve (2CSL*MOV104) will not open until the differential pressure across the valve is less than 88 pounds. This corresponds to a vessel pressure of approximately 335 psig.

Once initiated, the LPCS logic seals in and can be reset by the Control Room operator only when the reactor water level and drywell pressure return to normal.

3.2.1.3.5 Instrumentation and Controls

The LPCS can be manually started from the Control Room. In addition there are individual controls for the following equipment also located in the Control Room.

- 2CSL*P1 Core Spray Pump
- 2CSL*P2 Core Spray System Pressure Maintenance Pump
- 2CSL*MOV107 Minimum Flow to Suppression Pool
- 2CSL*FV114 Test return to Suppression Pool
- 2CSL*MOV104 Injection to Reactor Vessel
- 2CSL*MOV112 Suction From Suppression Pool

The following instrumentation provides control signals to various components and monitoring information to plant operators.

- 2CSL*FT107 Pump discharge flow
- 2CSL*FT126 Pump discharge flow
- 2CSL*PDT132 Injection valve differential pressure
- 2CSL*PIS109 Pump discharge pressure
- 2CSL*PIS110 Pump discharge pressure
- 2CSL*PI103 Pump discharge pressure
- 2CSL*PT130 Pump suction pressure

There is a LPCS inoperable alarm in the Control Room. The following alarms are included in the LPCS inoperable alarm:

- 2CSL*MOV112 (Suction valve loss of power)
- 2CSL*MOV104 (Injection valve loss of power)
- 2CSL*MOV107 (Min flow valve loss of power)
- 2CSL*FV114 (Return valve to Suppression Pool loss of power)
- 2RHS*MOV30A (Return valve to Suppression Pool full shut, or control switch in close position)
- 2CSL*MOV112 (Suction valve shut, or switch in close)
- 2CSL*P1 (LPCS pump loss of control power)
- LPCS line break (dP between LPCS and LPCI A > 3.8 psig)
- LPCS Pump 1
  - Overload (trips pump, locked from starting)
  - Auto Trip (Failure to Start)
  - Auto Start
  - Motor Electric Fault

- LPCS relay logic power failure - loss of power or one of the following test push buttons depressed at P629:
  a. Logic Power Monitor (S15)
  b. Power Test (S14)

- LPCS/RHR TEST (Diesel A test jumpered or switch in test P629).
- LPCS TRIP UNIT CALIB/GROSS FAILURE (LPCS trip units being calibrated or sensing a gross failure at P629)
- LPCS TRIP UNIT OOF/POWER FAILURE (LPCS trip units Out Of File or loss of power at P629)
- LPCS MANUALLY OUT OF SERVICE (LPCS manual out of service push button depressed at P601).
- Motor overload on any of the motor operated valves

  NOTE: The valves still align for injection on signal.

There are also alarms for the following conditions:

- LPCS pump room water level 2 inches above the floor (2DFR*LS147).
- LPCS suction greater than 50 psig or < 3.5 psig.
- LPCS/LPCI A injection dP > 3.8 psid.
- LPCS pump discharge > 525 psig or < 62 psig.
- MOV104 differential pressure > 88 psig.
- LPCS system actuated.
- LPCS High Point Vent Level Low.
- LPCS Trip Unit Trouble

Other alarms are listed in N2-OP-32.

The following AUTOMATIC RESPONSES occur in the LPCS system if both channels of drywell pressure sense a pressure greater than 1.68 psig.

1. Division 1 Emergency Diesel Generator 2EGS*EG1 starts.

2. LPCS system aligns in the injection mode and recirculates through the minimum flow valve (MOV107) until the pressure interlocks are satisfied, and injection can occur.

3.2.1.3.6 Technical Specification

Residual Heat Removal technical specifications also apply to the low pressure core spray system, refer to Section 3.2.1.4.6.

3.2.1.3.7 Surveillance, Testing and Maintenance

- Once per month, verify the high point vent is full.
- Once per month, verify positions of all valves not locked in the injection position[1].
- Verify proper flow at 290 pounds pressure is generated.
- At least once per 18 months, perform a system functional test which includes a simulated automatic actuation and verification that all automatic valves actuated correctly. Actual injection of coolant into the reactor may be excluded from this test.
- A listing of surveillance tests is included in the system information notebook.
- A listing of maintenance procedures is included in the system information notebook.

3.2.1.3.8 References

N2-OP-32, Rev. 3 "Low Pressure Core Spray"
USAR Section 6.3, Rev. 1
PID-32A-8
Technical Specifications, Section 3/4.5
GEK-83334A

3.2.1.3.9 Initiating Event Potential

The most significant potential initiating event is opening of the LPCS injection valve (MOV104) at reactor pressure. This could cause a severe over-pressurization of the low pressure piping, damaging the LPCS and nearby equipment, and resulting in an interfacing systems LOCA. If the injection valve transfers open, a stop/check valve in the high pressure piping (AOV101) is intended to prevent over-pressurization of the LPCS piping. Failure of AOV101 and MOV104 will be included in the interfacing LOCA analysis of initiating events.

[1] Except automatic valves capable of automatically returning to the ECCS position upon receipt of an initiation signal.

3.2.1.3.10 Equipment Location

2CSL*AOV101 and hand controlled isolation valve 2CSL*HCV117 are the only equipment inside primary containment. All other LPCS components are in the Reactor Building.

3.2.1.3.11 Operating Experience

There were no particular outstanding operational events relevant to this study. Plant specific component operational data is detailed in section 3.3.2.

3.2.1.3.12 Modeling Assumptions

1. The suppression pool strainers are sized to provide adequate NPSH to the pump even if the strainer is 50% clogged. There is no insulation in the wetwell and insulation in the drywell is covered with stainless steel. Therefore, the chances of insulation or panels reaching the downcomers and plugging a strainer is considered a small contributor.

2. Failure of the pressure maintenance pump is not modeled. Alarms in the control room indicate low discharge pressure and high point low water level. The pressure maintenance pump also has status indication in the control room. The system is also verified full monthly. For these reasons, failure of the piping to be full is considered a low probability event.

3. Failure of MOV107 to open is not modeled because it is normally in the open position.

4. Pipe connections that have double isolation valves or connections which are small lines (typically less than 2 inches) are not modeled. In addition, pipe failures are not modeled. These failures are small contributors, since the LPCS is a standby single train system which is not cross-connected to redundant trains, or to systems throttled to 1,000 gpm.

5. Valve(s) in the suppression pool return line are common to both LPCS and LPCI A. These valves are modeled in the LPCI A System Analysis.

6. Failure of 2VBA*UPS2A prevents opening of 2CSL*MOV104. There is a RPV pressure permissive requirement to open MOV104 which depends on UPS2A. Once MOV104 opens, it will remain open if 2VBA*UPS2A subsequently fails.

7. Switch E21A-S6 (control switch for pump 2CSL*P1) is not modeled. The failure of this and other control equipment (such as the breaker) are included in the Fail To Start category.

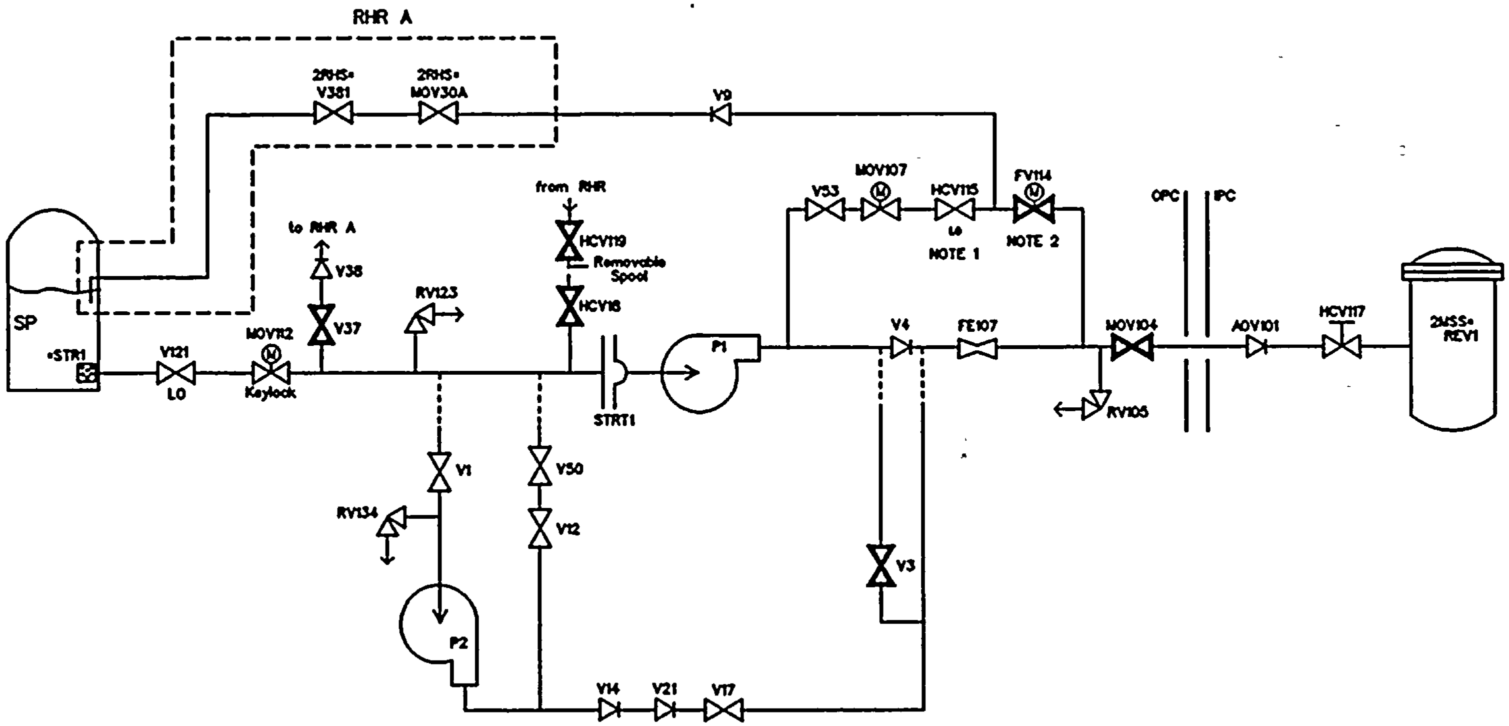
8. Switch E21A-S2 is not modeled in this system. E21A-S2 is the override switch for the injection valve. Given the proper conditions, this could cause the injection valve

to open under high pressure conditions. Therefore, it is modeled in the Interfacing Systems LOCA (ISLOCA) section.

3.2.1.3.13 Logic Model and Results

The fault tree model is included in Tier 2 documentation. Table 3.2.1.3-1 lists components included in the fault tree model, their failure modes, and support systems. Quantitative results are summarized in Section 3.3.5.

Rev 0 (7/92)



Notes: 1. 2CSL*HCV115 is throttled to provide 1000 GPM minimum LPCS flow.

2. 2CSL*FV114 is throttled to 7800 GPM.

Figure 3.2.1.3-1
Low Pressure Core Spray

LOW PRESSURE CORE SPRAY
Component Block Descriptions

Block	Mark No. (Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
SP SUCTION PATH							
	2CSL*V121	PUMP SUCT. ISOLATION	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CSL*MOV112	PUMP SUCTION VALVE	TRANSFER CLOSED	OPEN	CLOSED	2EHS*MCC102C (I	AS-IS
	2CSL*STRT1	STRAINER	PLUGGED	N/A	N/A	N/A	N/A
PUMP							
	E21A-K1	P1 START RELAY	FAIL TO PICK UP	N/A	N/A	N/A	N/A
	E21A-K12	P1 START RELAY	FAIL TO PICK UP	N/A	N/A	N/A	N/A
	2D-2ENSX04	LPCS SEQUENCER	FAIL TO PICK UP	N/A	N/A	N/A	N/A
	2CSL*P1	LPCS PUMP	FAILS TO START	STOPPED	RUNNING	2ENS*SWG101 (I)	STOP
	2CSL*P1	LPCS PUMP	FAILS TO RUN	RUNNING	RUNNING	2ENS*SWG101	STOP
INJECTION PATH							
	2CSL*V4	CHECK VALVE	FAILS TO OPEN	CLOSED	N/A	N/A	N/A
	2CSL*RO106	RESTRICTION ORIFICE	PLUGGED	N/A	N/A	N/A	N/A
	2CSL*MOV104	LPCS INJECTION VALVE	FAILS TO OPEN	CLOSED	OPEN	2EHS*MCC102C (I	AS-IS
	E21A-K50	PDT132 RELAY	FAIL TO PICK UP	N/A	N/A	N/A	N/A
	2CSL*PDT132	D.P. TRANSMITTER		N/A	N/A	N/A	N/A
	2CSL*PDIS132	PDT132 IND.SWITCH	FAIL TO ACTUATE	N/A	N/A	N/A	N/A
	E21A-K14	RELAY	FAIL TO PICK UP	N/A	N/A	N/A	N/A
	2CSL*AOV101	TESTABLE CHECK VALVE	FAILS TO OPEN	CLOSED	N/A	N/A	N/A
	2CSL*HCV117	HAND CTRL VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
MIN FLOW PATH							
	2CSL*FIS107	FLOW SWITCH	FAILS HIGH	N/A	N/A	N/A	N/A
	2CSL*FT107	FLOW TRANSMITTER	FAILS HIGH	N/A	N/A	N/A	N/A
	2CSL*V53	MIN FLOW ISOLATION	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CSL*MOV107	MOTOR OPERATED VALVE	TRANSFERS CLOSE	OPEN	CLOSED	2EHS*MCC102C (I)	AS-IS
	2CSL*HCV115	PUMP MIN FLOW VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CSL*V9	CHECK VALVE	FAILS TO OPEN	CLOSED	N/A	N/A	N/A
	2RHS*MOV30A	RHR RETURN SUPP POOL	TRANSFER CLOSED	OPEN	CLOSED	2EHS*MCC103C (I)	AS-IS
	2RHS*V381	RHS MANUAL VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A

System 4

Residual Heat Removal

3.2.1.4 Residual Heat Removal

3.2.1.4.1 System Function

There are three subsystems, or loops associated with the Residual Heat Removal (RHS) system which are referred to as RHS loops A, B, and C. Simplified diagrams of each loop are provided in Figures 3.2.1.4-1, 2 and 3.

The Residual Heat Removal system has five main functions:

1. The Low Pressure Coolant Injection (LPCI) mode restores water level in the reactor vessel following a Loss Of Coolant Accident (LOCA). RHS Loop C is dedicated to this function. Loop A and B also provide LPCI.

2. The Containment Spray Cooling mode condenses steam and reduces airborne activity in the drywell and the free space of the suppression chamber following a LOCA. (Loop A and B)

3. The Shutdown Cooling mode removes decay heat from the core following a reactor shutdown. (Loop A and B)

4. The RHS Steam Condensing mode condenses reactor steam and returns the condensate to the reactor vessel through the Reactor Core Isolation Cooling System. Since steam condensing is a shutdown function, it is not modeled in the IPE.

5. The Suppression Pool Cooling mode removes heat from the suppression pool water following safety-relief valve blowdown, prolonged RCIC system operation, or during post-accident conditions. (Loop A and B)

The RHS system can be used to flood the containment, if required, for long term post-accident recovery operations. This involves injecting service water through the RHS Loop B piping downstream of the heat exchanger into the RPV and/or containment. In addition, fire water can be injected through RHS Loop A or B by connecting a hose and manually opening valves.

RHS can also augment the Spent Fuel Pool Cooling and Cleanup System if additional cooling capacity is required.

3.2.1.4.2 Success Criteria

Several top event models are developed to cover the injection, sprays, and pool cooling functions in the front line event trees. The following summarizes top event models and success criteria:

Top Event	Success Criteria
LC	RHS Pump C starts and supplies suppression pool water to the RPV for 24 hours.

Top Event	Success Criteria
LA	RHS Pump 1A starts and supplies suppression pool water to heat exchanger E1A for 24 hours.
LB	RHS Pump 1B starts and supplies suppression pool water to heat exchanger E1B for 24 hours.
IA	Injection path from E1A to the RPV opens (MOV24A) and remains open.
IB	Injection path from E1B to the RPV opens (MOV24B) and remains open.
HA	E1A bypass MOV8A closes and service water MOV90A and MOV33A open and remain open.
HB	E1B bypass MOV8B closes and service water MOV90B and MOV33B open and remain open.
PA	Suppression pool cooling path from E1A opens (valve FV38A) and remains open.
PB	Suppression pool cooling path from E1B opens (valve FV38B) and remains open.
CA	Containment spray paths from E1A open (MOV15A and 25A, or 33A) and remain open.
CB	Containment spray paths from E1B open (MOV15B and 25B, or 33B) and remain open.
OH	Operator actions required to provide heat removal with the RHS heat exchangers when there is successful injection.

3.2.1.4.3 Support Systems

RHS loop "A" depends on Division I support systems. RHS loops "B" and "C" depend on Division II support systems. For a complete list of the components and their necessary support systems, see Table 3.2.1.4-1.

Actuation of the LPCI function is dependent on the ECCS actuation system which is modeled in the support event tree as top events E1 and E2.

The discharge headers of loops B and C are maintained full of water and pressurized by the RHS system pressure pump (also called the water leg pump or the line fill pump) to avoid water hammer upon system initiation. The A loop is maintained full and pressurized by the Low Pressure Core Spray System system pressure pump.

The following systems interface with, but are not a part of, the RHS system model:

- Liquid radwaste for reducing suppression pool inventory
- Condensate makeup and draw off
- Spent fuel pool cooling and cleanup
- Reactor core isolation cooling
- Reactor plant sampling
- Reactor Building equipment drain
- Reactor recirculation
- Low pressure core spray system pressure pump (shared with LPCI Loop A)
- Fire Water / Service Water - refer to Section 3.2.1.9

3.2.1.4.4 System Operation

Normal Operation

During normal operation, the RHS system is in standby mode. Both the shell and tube side of each heat exchanger is flushed and filled with pure water to minimize possible corrosion or fouling of heat transfer surfaces. The heat exchanger inlet, outlet, and bypass valves are fully open. Each pump's suppression pool suction valve and minimum flow valve is in the open position. The RHS system pressure pump is running continuously to keep the B and C loop piping filled. The A loop piping is pressurized from the Low Pressure Core Spray system pressure pump. The shutoff valve for return flow to the suppression pool MOV30A and MOV30B is open. All other remotely operated valves in the various subsystem flow paths are closed. Pump A, B, and C's control switches are in the Auto position. The suppression pool is filled to its normal operating level with reactor grade water.

Automatic Operation

The ECCS is actuated automatically (LPCI mode) and requires no operator action during the first 10 minutes following initiation.

Following indication (by high drywell pressure and/or triple low reactor water level) of a Loss Of Coolant Accident (LOCA), the LPCI mode initiates automatically. All three pumps start automatically, taking suction from the suppression pool. In the A and B loops, the normally open heat exchanger bypass valves (MOV8A and B) receive an open signal. The injection valves open when the differential pressure interlock is satisfied. There is a minimum flow valve for each loop (MOV4A, B and C) which opens for pump protection when discharge flow is low.

Once reactor vessel water level has been restored, the LPCI flow must be terminated by the operators by closing the LPCI injection valves.

Manual Operation
During the long-term cooling period (after 10 minutes), the operator takes action to place the suppression pool cooling system in operation. Initiation of the suppression pool cooling system and the containment spray cooling system (if required) are manual actions required by the operator to provide suppression pool, suppression chamber, and containment cooling.

3.2.1.4.5 Instrumentation and Controls

Control switches that allow a safety system bypasse are keylocked. All keylock switches in the Control Room can only be removed when the switches are in the ACCIDENT (SAFE) position. All keys are normally removed from their respective switches during operation and maintained under strict administrative control.

Each automatically initiated Engineered Safety Feature (ESF) system control logic seals-in electrically and remains energized after initial conditions return to normal. Deliberate operator action is required to reset an ESF system logic to normal.

Upon ESF actuation, all components proceed to their safety position. To reset a component, two distinct operator actions are necessary: one to reset the actuation signal, and one to restore each component to it's normal position.

A single pushbutton will initiate both LPCS and LPCI loop A if the Division I diesel generator sequencer permissives are satisfied. A second pushbutton switch initiates LPCI loops B and C via the Division II diesel generator sequencer.

When the reactor water level is at level 1, the LPCI subsystem has priority through the valve control logic over the other RHS subsystems for containment cooling, shutdown cooling, or steam condensing. Immediately following a LOCA, the RHS system is automatically directed to the LPCI mode. Once initiated, LPCI has no automatic shutoff or isolation. System operation continues until manually secured by the Control Room operator.

Indication is provided on Control Room panel P601 for each loop of RHS pump motor amps, service water flow to heat exchangers, RHS pump flow, heat exchanger level, heat exchanger pressure, head spray flow, heat exchanger vent valve position, and heat exchanger outlet conductivity.

Alarms/Annunciation are provided by annunciator panels on Control Room panel 601 and by computer point alarm messages.

Remote controls for pumps; valves; level and pressure controllers are found on control panel 601.

3.2.1.4.6 Technical Specifications

The three LPCI (RHS) loops and the LPCS loop are combined in the Technical Specifications (Tech Specs). All four shall be operable during power operation, otherwise:

Division I LPCS / LPCI A	Division II LPCI B / LPCI C	LCO Length
Any 1 of 4 inoperable		7 Days
Any 2 of 4 inoperable		72 Hours
All Others		Immediate

For the further information, see Tech. Spec. 3.5.1 and 3.5.2.

3.2.1.4.7 Surveillance Testing and Maintenance

If an initiation signal occurs during LPCI testing, the LPCI system automatically returns to the operating mode.

High Pressure Core Spray, Low Pressure Core Spray, and Low Pressure Core Injection testing for functional operability of the control logic relays can be accomplished by use of plug-in test jacks and switches in conjunction with signal sensor tests.

At least once per 31 days LPCI is verified operable by venting at the high point vents. This ensures that the system is filled with water from the pump discharge valve to the isolation valve. It is also verified once per 31 days that each valve (manual, power-operated, or automatic) in the flow path is not locked, sealed, or otherwise secured in position, and is in the correct position. (Except for valves that automatically return to their ECCS position when an ECCS signal is present).

The RHS pumps are flow tested quarterly.

At least once per 18 months, a system functional test is performed which includes simulated automatic actuation of the system throughout its emergency operating sequences. Verification that each automatic valve in the flow path actuates to its correct position is also performed. Actual injection of coolant into the reactor vessel may be excluded from this test.

3.2.1.4.8 References

Nine Mile Point Technology, Residual Heat Removal, Rev. 0
N2-OP-31, Rev. 5 "Residual Heat Removal"
GEK-83337A
Technical Specifications 3/4.5.1, 3/4.5.2
Drawings as referenced on the simplified drawings
USAR section 6.3

3.2.1.4.9 Initiating Event Potential

Spurious initiation of the system could be an initiating event. The RHS system is also considered in the Interfacing Systems LOCA (ISLOCA) write-up.

3.2.1.4.10 Equipment Location

Pump "A" Elev. 175', North Aux. Bay
Pump "B" Elev. 175', South Aux. Bay
Pump "C" Elev. 175', South Aux. Bay

Heat Exchanger "A" Elev. 175', North Aux. Bay
Heat Exchanger "B" Elev. 175', South Aux. Bay

3.2.1.4.11 Operating Experience

There were no particular outstanding operational events relevant to this study. Plant specific component operational data is detailed in section 3.3.2.

3.2.1.4.12 Modeling Assumptions

1. The suppression pool strainers are sized to provide adequate NPSH to the RHS pumps even if the strainers are 50% clogged. There is no insulation in the wetwell and insulation in the drywell is covered with stainless steel. Therefore, the chances of insulation or panels reaching the downcomers on the drywell floor and then getting through the downcomers and reaching a strainer is considered a small contributor.

2. Failure of the system pressure pump is not modeled. Alarms in the Control Room indicate low discharge line pressure, High Point Vent low water level, and pump operation. The pressure pump would have to be unavailable for a long period of time before significant leakage would occur. Even with the discharge piping empty, it is unlikely that the system piping would fail due to water hammer. The delayed delivery of water to the RPV is also assumed insignificant.

3. Relay 2C-ENSY04 is modeled for pump start. There are two sequencing relays, but only one can be energized at any time. The second relay is 2B-ENSY04. One energizes on LOCA only, and the other energizes on a LOCA coincident with a Loss of Offsite Power (LOSP) only. Therefore, as a simplification, only one relay is modeled, as only one will ever be energized at one time.

4. Pipe connections that have double isolation or small lines are not modeled. In addition, pipe failures are not modeled. These failure modes are small contributors particularly for a standby system that is not normally cross-connected to redundant trains or systems.

5. Failure to cool the RHS pump seal coolers does not result in RHS pump failure or excessive leakage.

6. Failure of the ΔP transmitters 2RHS*PDT24A, 24B or 24C sensing lines will result in either; failure to open the respective injection valve (MOV24A, 24B or 24C) or premature attempt to open LPCI injection valves may result in failure of MOV24A or MOV24B & 24C.

7. Failure of 120V AC/24V DC analog system power supply (E21A-PS2) prevents the opening of the RHS injection valves 2RHS*MOV24B & C thus resulting in the failure of LPCI loops B and C. It is modeled in top event UB, UPS Source B. Top event UA models the similar failure of UPS source A.

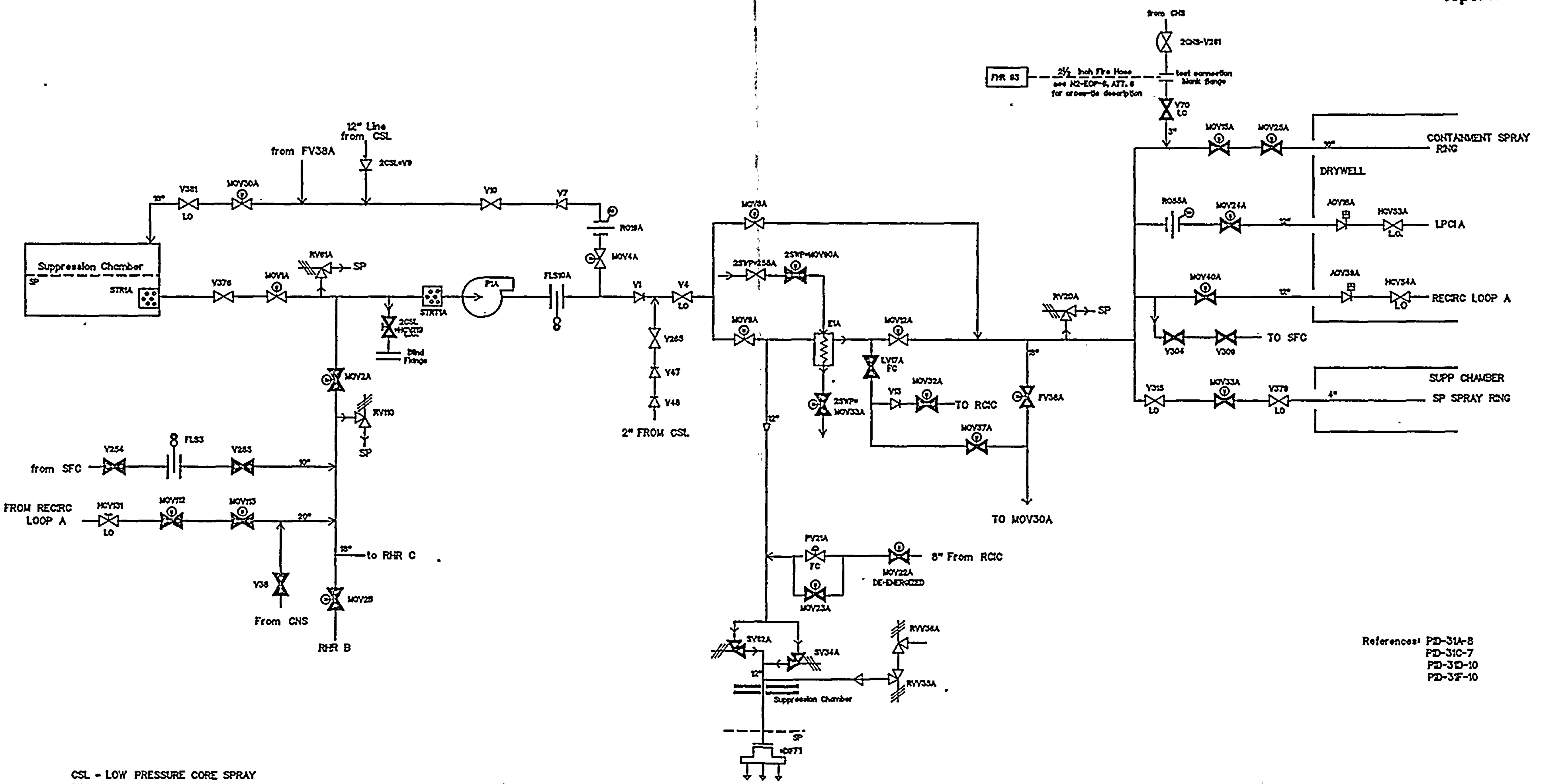
3.2.1.4.13 Logic Model and Results

The fault tree(s) is included in Tier 2 documentation, and is available upon request. Quantitative results are summarized in Section 3.3.4 (Tier 1).

Rev 0 (7/92)

SI APERTURE CARD

Also Available On Aperture Card



References: PD-31A-8
PD-31C-7
PD-31D-10
PD-31F-10

CSL = LOW PRESSURE CORE SPRAY
SFC = SPENT FUEL POOL COOLING
SP = SUPPRESSION POOL

ALL COMPONENTS PREFIXED BY 2RHS* UNLESS NOTED

FIGURE 3.2.1.4-1
RHR Loop A

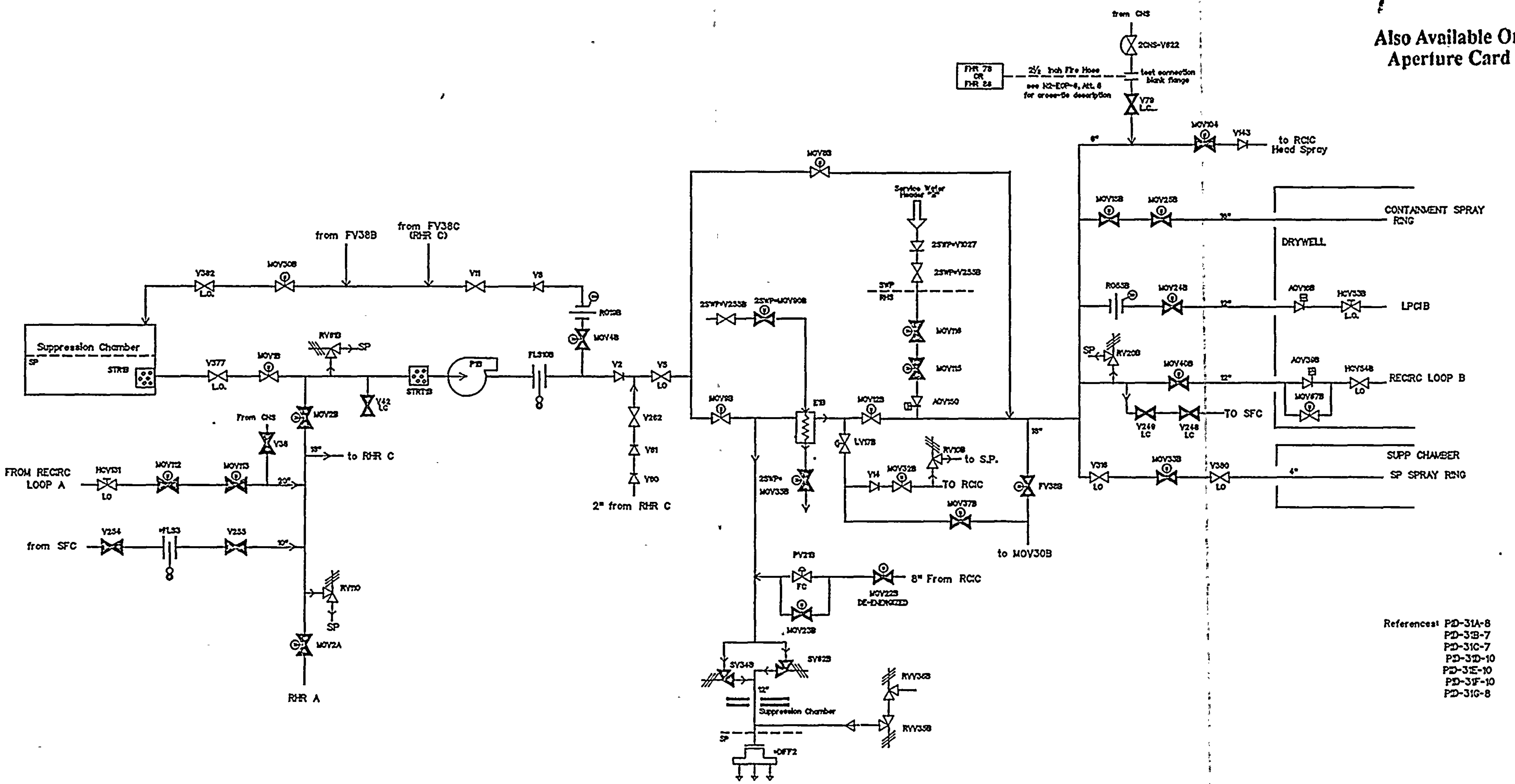
I:\NMP2IPE\DGN\2S04.DGN Jul. 07, 1992 10:27:46

9208050191 - 14

Rev 0 (7/92)

SI APERTURE CARD

Also Available On Aperture Card



SFC = SPENT FUEL POOL COOLING
SP = SUPPRESSION POOL

ALL COMPONENTS PREFIXED BY 2RHS* UNLESS NOTED

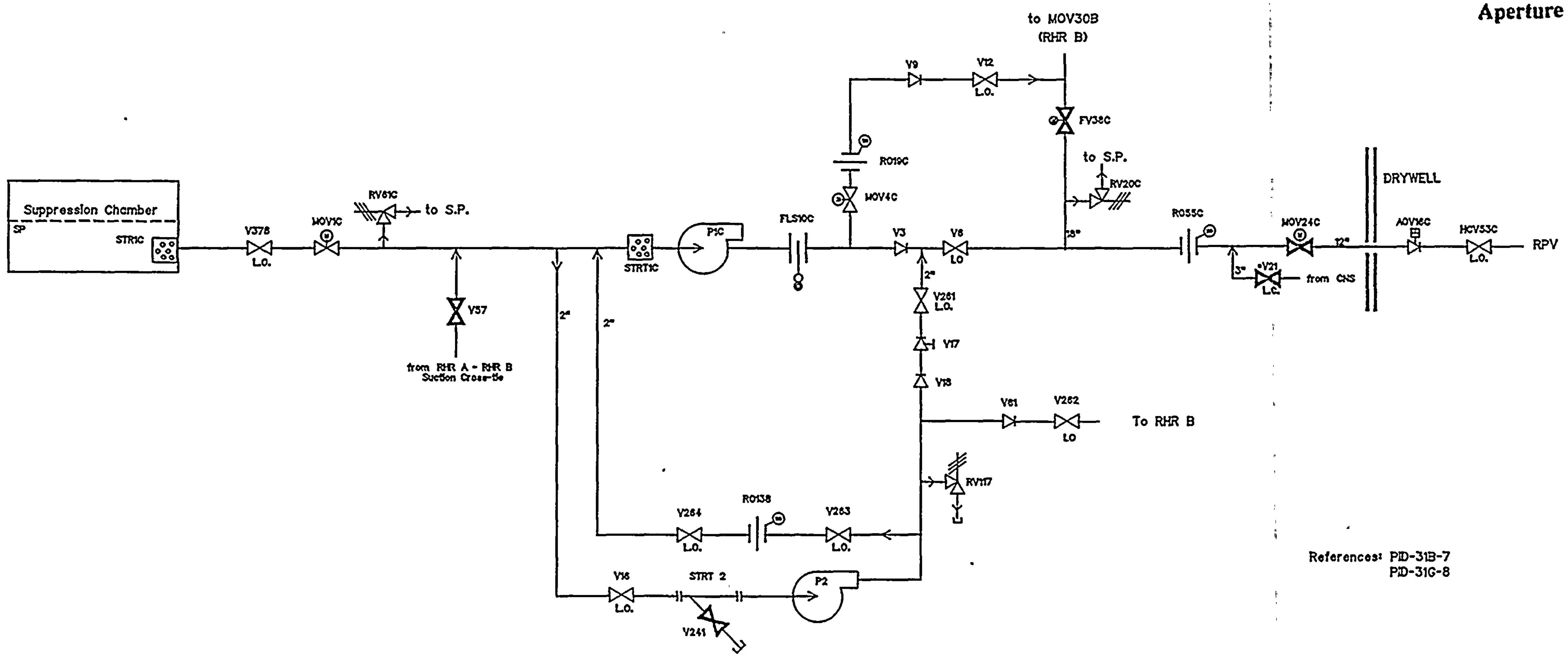
References: PID-31A-8
PID-31B-7
PID-31C-7
PID-31D-10
PID-31E-10
PID-31F-10
PID-31G-8

FIGURE 3.2.1.4-2
RHR Loop B

I:\NMP2IPE\DGN\2S04.DGN Jul. 07, 1992 10:29:50

9208050191-15

Rev 0 (7/92)



SI APERTURE CARD

Also Available On Aperture Card

References: PID-31B-7
PID-31G-8

FIGURE 3.2.1.4-3
RHR Loop C

CNS - Condensate Storage and Transfer System
SFC - Spent Fuel Pool Cooling
SP - Suppression Pool
ALL COMPONENTS PREFIXED BY 2RHS* UNLESS NOTED

I:\NMP2IPE\DGN\2S04.DGN Jul. 07, 1992 10:30:52

9208050191 -16

RESIDUAL HEAT REMOVAL SYSTEM
Component Block Descriptions

Block	Mark No. (Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
TOP EVENT CA							
	2RHS*MOV15A	RHS "A" CONT. SPRAY	FAILS TO OPEN	CLOSED	OPEN	2EHS*MCC103 (I)	AS-IS
	2RHS*MOV25A	RHS "A" CONT. SPRAY	FAILS TO OPEN	CLOSED	OPEN	2EHS*MCC103 (I)	AS-IS
	2RHS*V315	RHS "A" S.P. SPRAY	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2RHS*MOV33A	RHS "A" S.P. SPRAY	FAILS TO OPEN	CLOSED	OPEN	2EHS*MCC103 (I)	AS-IS
	2RHS*V379	SP SPRAY MAN ISO VLV	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
TOP EVENT CB							
	2RHS*MOV15B	RHS "B" CONT. SPRAY	FAILS TO OPEN	CLOSED	OPEN	2EHS*MCC303 (II)	AS-IS
	2RHS*MOV25B	RHS "B" CONT. SPRAY	FAILS TO OPEN	CLOSED	OPEN	2EHS*MCC303 (II)	AS-IS
	2RHS*V316	RHS "B" S.P. SPRAY	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2RHS*MOV33B	RHS "B" S.P. SPRAY	FAILS TO OPEN	CLOSED	OPEN	2EHS*MCC303 (II)	AS-IS
	2RHS*V380	SP SPRAY MAN ISO VLV	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
TOP EVENT HA							
	2RHS*MOV8A	HT EXCH BYPASS MOV	FAILS TO CLOSE	OPEN	CLOSED	2EHS*MCC103 (I)	AS-IS
	2RHS*MOV9A	HEAT EXCH INLET MOV	TRANSFER CLOSED	OPEN	N/A	2EHS*MCC103 (I)	AS-IS
	2RHS*E1A	RHS "A" HEAT EXCH.	FAILS	N/A	N/A	N/A	N/A
	2RHS*MOV12A	HEAT EXCH OUTLET MOV	TRANSFER CLOSED	OPEN	N/A	2EHS*MCC103 (I)	AS-IS
	2SWP*V255A	SW ISOLATION VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2SWP*MOV90A	SW INLET TO RHR E1A	FAILS TO OPEN	CLOSED	OPEN	2EHS*MCC102A (I)	AS-IS
	2SWP*MOV33A	SW OUTLET FROM E1A	FAILS TO OPEN	CLOSED	OPEN	2EHS*MCC102A (I)	AS-IS
TOP EVENT HB							
	2RHS*MOV8B	HT EXCH BYPASS MOV	FAILS TO CLOSE	OPEN	CLOSED	2EHS*MCC303 (II)	AS-IS
	2RHS*MOV9B	HEAT EXCH INLET MOV	TRANSFER CLOSED	OPEN	N/A	2EHS*MCC303 (II)	AS-IS
	2RHS*E1B	RHS B HEAT EXCHANGER	FAILS	N/A	N/A	N/A	N/A
	2RHS*MOV12B	HEAT EXCH OUTLET MOV	TRANSFER CLOSED	OPEN	N/A	2EHS*MCC303 (II)	AS-IS
	2SWP*V255B	SW ISOLATION VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2SWP*MOV90B	SW INLET TO E1B	FAILS TO OPEN	CLOSED	OPEN	2EHS*MCC302 (II)	AS-IS
	2SWP*MOV33B	SW OUTLET FROM E1B	FAILS TO OPEN	CLOSED	OPEN	2EHS*MCC302 (II)	AS-IS
TOP EVENT IA							
	2RHS*MOV24A	LPCI A INJECTION VALVE	FAILS TO OPEN	CLOSED	OPEN	2EHS*MCC103 (I)	AS-IS
	2RHS*PDT24A	DIFF PRESS XMTR					
	2ISC*PT4D	PRESSURE TRANSMITTER PT4D	FAILS				
	E21A-K115A	PDT24A RELAY	FAIL TO PICK UP			2VBS*PNL101A	FAILS
	2RHS*PDIS24A	PDT24A PDIS	FAIL TO ACTUATE				
	2RHS*AOV16A	TESTABLE CHECK VALVE	FAILS TO OPEN	CLOSED	OPEN	N/A	N/A
	2RHS*HCV53A	INSIDE ISO FOR LPCI	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
TOP EVENT IB							
	2RHS*MOV24B	LPCI B INJECTION VALVE	FAILS TO OPEN	CLOSED	OPEN	2EHS*MCC303 (II)	AS-IS
	2RHS*PDT24B	DIFF PRESS XMTR					
	2ISC*PT4D	PRESSURE TRANSMITTER PT4D	FAILS				
	E12A-K115B	PDT24B RELAY	FAIL TO PICK UP			2VBS*PNL301B	FAILS
	2RHS*PDIS24B	PDT24B PDIS	FAIL TO ACTUATE				
	2RHS*AOV16B	TESTABLE CHECK VALVE	FAILS TO OPEN	CLOSED	OPEN	N/A	N/A

RESIDUAL HEAT REMOVAL SYSTEM
Component Block Descriptions

Block	Mark No. (Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
	2RHS*HCV53B	INSIDE ISO FOR LPCIB	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
TOP EVENT LA							
	2RHS*V376	PUMP SUCTION FROM SP	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2RHS*MOV1A	TANK SUCTION MOV	TRANSFER CLOSED	OPEN	CLOSED	2EHS*MCC103 (I)	AS-IS
	2RHS*STRT1A	STRAINER	PLUGGED	N/A	N/A	N/A	N/A
	2RHS*P1A	RHR PUMP 1A	FAILS TO START	STOPPED	RUNNING	2ENS*SWG101 (I)	STOP
	E12A-K3A	P1A START RELAY	FAILS TO PICKUP				
	E12A-K18A	P1A START RELAY	FAILS TO PICKUP				
	2C-2ENSX04	LPCI A SEQUENCER					
	2RHS*P1A	RHR PUMP 1A	FAILS TO RUN	STOPPED	RUNNING	2ENS*SWG101 (I)	STOP
	2RHS*FLS10A	SPECTACLE FLANGE	IN PLACE	N/A	N/A	N/A	N/A
	2RHS*V1	PUMP DISCH CHK VALVE	FAILS TO OPEN	OPEN	N/A	N/A	N/A
	2RHS*V4	PUMP DISCH MAN ISO	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2RHS*MOV4A	MINI FLOW BYPASS	FAILS TO OPEN	CLOSED	OPEN	2EHS*MCC103 (I)	AS-IS
	2RHS*V7	CHECK VALVE	FAILS TO OPEN	CLOSED	N/A	N/A	N/A
	2RHS*V10	MANUAL VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2RHS*MOV30A	RETURN TO SUPP POOL	TRANSFER CLOSED	OPEN	CLOSED	2EHS*MCC103 (I)	AS-IS
	2RHS*V381	RETURN TO SUPP POOL	TRANSFER CLOSED	L.OPEN	N/A	N/A	N/A
	2RHS*MOV8A	HT EXCH BYPASS MOV	TRANSFER CLOSED	OPEN	CLOSED	2EHS*MCC103 (I)	AS-IS
	2RHS*MOV9A	HEAT EXCH INLET MOV	TRANSFER CLOSED	OPEN	CLOSED	2EHS*MCC103 (I)	AS-IS
	2RHS*E1A	RHS A HEAT EXCHANGER	PLUGGED	N/A	N/A	N/A	N/A
	2RHS*E1A	RHS A HEAT EXCHANGER	RUPTURE/LEAKAGE	N/A	N/A	N/A	N/A
	2RHS*MOV12A	HEAT EXCH OUTLET MOV	TRANSFER CLOSED	OPEN	CLOSED	2EHS*MCC103 (I)	AS-IS
	2SWP*V255A	SERVICE W ISOLATION	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2SWP*MOV90A	SW INLET TO E1A	FAILS TO OPEN	CLOSED	OPEN	2EHS*MCC102A (I)	AS-IS
	2SWP*MOV33A	SW OUTLET FROM E1A	FAILS TO OPEN	CLOSED	OPEN	2EHS*MCC102A (I)	AS-IS
TOP EVENT LB							
	2RHS*V377	PUMP SUCTION FROM SP	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2RHS*MOV1B	TANK SUCTION MOV	TRANSFER CLOSED	OPEN	CLOSED	2EHS*MCC303 (II)	AS-IS
	2RHS*STRT1B	STRAINER	PLUGGED	N/A	N/A	N/A	N/A
	2RHS*P1B	RHR PUMP 1B	FAILS TO START	STOPPED	RUNNING	2ENS*SWG103 (II)	STOP
	E12A-K3B	P1B START RELAY	FAIL TO PICKUP				
	E12A-K18B	P1B START RELAY	FAIL TO PICKUP				
	2C-2ENSY04	LPCI B SEQUENCER					
	2RHS*P1B	RHR PUMP 1B	FAILS TO RUN	STOPPED	RUNNING	2ENS*SWG104 (II)	STOP
	2RHS*FLS10B	SPECTACLE FLANGE	IN PLACE	N/A	N/A	N/A	N/A
	2RHS*V2	PUMP DISCH CHK VALVE	FAILS TO OPEN	OPEN	N/A	N/A	N/A
	2RHS*V5	PUMP DISCH MAN ISO	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2RHS*MOV4B	MINI FLOW BYPASS	FAIL TO OPEN	CLOSED	OPEN	2EHS*MCC303 (II)	AS-IS
	2RHS*V8	CHECK VALVE	FAIL TO OPEN	CLOSED	N/A	N/A	N/A
	2RHS*V11	MANUAL VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2RHS*MOV30B	RETURN TO SUPP POOL	TRANSFER CLOSED	OPEN	CLOSED	2EHS*MCC303 (II)	AS-IS
	2RHS*V382	RETURN TO SUPP POOL	TRANSFER CLOSED	L.OPEN	N/A	N/A	N/A
	2RHS*MOV8B	HT EXCH BYPASS MOV	TRANSFER CLOSED	OPEN	CLOSED	2EHS*MCC303 (II)	AS-IS
	2RHS*MOV9B	HEAT EXCH INLET MOV	TRANSFER CLOSED	OPEN	CLOSED	2EHS*MCC303 (II)	AS-IS
	2RHS*E1B	RHS B HEAT EXCHANGER	PLUGGED	N/A	N/A	N/A	N/A

RESIDUAL HEAT REMOVAL SYSTEM
Component Block Descriptions

Block	Mark No. (Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
	2RHS*E1B	RHS B HEAT EXCHANGER	RUPTURE/LEAKAGE	N/A	N/A	N/A	N/A
	2RHS*MOV12B	HEAT EXCH OUTLET MOV	TRANSFER CLOSED	OPEN	CLOSED	2EHS*MCC303 (II)	AS-IS
	2SWP*V255B	SW ISOLATION VALVE	TRANSFER CLOSED	OPEN	CLOSED	N/A	N/A
	2SWP*MOV90B	SW INLET TO E1B	FAILS TO OPEN	CLOSED	OPEN	2EHS*MCC302A (II	AS-IS
	2SWP*MOV33B	SW OUTLET FROM E1B	FAILS TO OPEN	CLOSED	OPEN	2EHS*MCC302A (II	AS-IS
TOP EVENT LC							
	2RHS*V378	PUMP SUCTION FROM SP	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2RHS*MOV1C	TANK SUCTION MOV	TRANSFER CLOSED	OPEN	CLOSED	2EHS*MCC303 (II)	AS-IS
	2RHS*STRT1C	STRAINER	PLUGGED	N/A	N/A	N/A	N/A
	2RHS*P1C	RHR PUMP 1C	FAILS TO START	STOPPED	RUNNING	2ENS*SWG103 (II)	STOP
	E12A-K30B	P1C START RELAY	FAIL TO PICKUP				
	E12A-K21	P1C START RELAY	FAIL TO PICKUP				
	2D-2ENSY04	P1C START RELAY					
	2RHS*P1C	RHR PUMP 1C	FAILS TO RUN	STOPPED	RUNNING	2ENS*SWG103 (II)	STOP
	2RHS*FLS10C	FLANGE	IN PLACE	N/A	N/A	N/A	N/A
	2RHS*V3	PUMP DISCH CHK VALVE	FAILS TO OPEN	OPEN	N/A	N/A	N/A
	2RHS*V6	PUMP DISCH MAN ISO	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2RHS*MOV4C	MINI FLOW BYPASS	FAIL TO OPEN	CLOSED	OPEN	2EHS*MCC303 (II)	AS-IS
	2RHS*V9	CHECK VALVE	FAIL TO OPEN	CLOSED	N/A	N/A	N/A
	2RHS*V12	MANUAL VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2RHS*MOV30B	RETURN TO SUPP POOL	TRANSFER CLOSED	OPEN	CLOSED	2EHS*MCC303 (II)	AS-IS
	2RHS*V382	RETURN TO SUPP POOL	TRANSFER CLOSED	L.OPEN	N/A	N/A	N/A
	2RHS*MOV24C	LPCI C	FAILS TO OPEN	CLOSED	OPEN	2EHS*MCC303 (II)	AS-IS
	2RHS*PDT24C	DIFF PRESS XMTR					
	E12A-K115C	PDT24C RELAY	FAIL TO PICKUP			2VBS*PNL301B	FAILS
	2RHS*PDIS24C	PDT24C PD IND SW	FAIL TO ANNUNC				
	2RHS*AOV16C	TESTABLE CHECK VALVE	FAILS TO OPEN	CLOSED	OPEN	N/A	N/A
	2RHS*HCV53C	INSIDE ISO FOR LPCIC	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
TOP EVENT PA							
	2RHS*FV38A	RETURN TO SP	FAILS TO OPEN	CLOSED	OPEN	2EHS*MCC103 (I)	AS-IS
	2RHS*MOV30A	RETURN TO SP	TRANSFER CLOSED	OPEN	CLOSED	2EHS*MCC103 (I)	AS-IS
	2RHS*V381	RETURN TO SP	TRANSFER CLOSED	CLOSED	N/A	N/A	N/A
TOP EVENT PB							
	2RHS*FV38B	RETURN TO SP	FAILS TO OPEN	CLOSED	OPEN	2EHS*MCC303 (II)	AS-IS
	2RHS*MOV30B	RETURN TO SP	TRANSFER CLOSED	OPEN	CLOSED	2EHS*MCC303 (II)	AS-IS
	2RHS*V382	RETURN TO SP	TRANSFER CLOSED	CLOSED	N/A	N/A	N/A

System 5

ECCS Actuation

3.2.1.5 ECCS Initiation System

3.2.1.5.1 System Function

The ECCS Initiation System is a safety related instrumentation system which detects abnormally low RPV level or high drywell pressure. These conditions indicate that a pipe break or loss of reactor inventory has occurred. The ECCS Initiation System provides the automatic initiation of:

- ADS
- RCIC
- LPCS
- LPCI A, B and C

NOTE: This section models the ECCS initiation logic, not the ECCS systems. HPCS, an ECCS system, has it's own logic, and is modeled in the HPCS system. Although RCIC is not an ECCS system, RCIC uses ECCS instrumentation, and is modeled in this system.

3.2.1.5.2 Success Criteria

Automatic ECCS actuation is modeled in the support event tree as top events E1 and E2 (Division I and Division II). Manual ECCS actuation is modeled as top event ME in the support event tree. An ECCS actuation success diagram is provided in Figure 3.2.1.5-1. As shown, an ECCS signal is generated when one low vessel level or high drywell pressure signal is generated coincidentally with another low vessel level or high drywell pressure signal in a 1 out of 2 taken twice logic. Once initiated, the system safety function has been satisfied because the actuated systems are individually sealed in. There are two redundant trains, each of which is capable of initiating sufficient ECCS systems to protect the reactor core. The Division I ECCS systems are actuated by the Division I ECCS Initiation System, E1. Division II ECCS systems are actuated by Division II ECCS Initiation System, E2.

3.2.1.5.3 Support System

ECCS actuation depends on 120V AC and 125V DC, as follows:

ECCS Actuation	Support
Division I	2VBA*UPS2A
Division I	2BYS*SWG002A
Division II	2VBA*UPS2B
Division II	2BYS*SWG002B

Failure of either AC or DC support to an ECCS actuation division results in loss of auto actuation of that division. Loss of AC can be bypassed by manual actuation as long as DC is available.

The specific support system requirements for the key ECCS actuation system components are shown in Table 3.2.1.5-1.

3.2.1.5.5 Instrumentation and Control

The ECCS actuation signal is developed by two redundant trains (Division I & Division II). Division I receives power from 125V DC Division I and 120V AC UPS-A, and causes initiation of LPCI A, LPCS, ADS I and RCIC. A Division I initiation signal is generated when any one of the following combinations are satisfied:

- Two (2) low vessel level,
- Two (2) high drywell pressures,
- 2ISC*PT17A and 2ISC*LT9C (1 high press and 1 low level), or
- 2ISC*PT17C and 2ISC*LT9A (1 high press and 1 low level).

Division II receives power from 125V DC Division II and 120V AC UPS-B, and causes initiation of LPCI B & C, ADS II and RCIC II. A Division II signal is initiated when any one of the following combinations are satisfied:

- Two (2) vessel water level transmitters,
- Two (2) drywell pressure transmitters,
- 2ISC*PT17B and 2ISC*LT9D (1 high press and 1 low level), or
- 2ISC*LT9B and 2ISC*PT17D (1 high press and 1 low level).

3.2.1.5.6 Technical Specifications

With one ECCS Actuation channel inoperable:

- The inoperable ECCS initiation channel shall be placed in a tripped condition, or the LPCS system shall be declared inoperable if any one of the two vessel level or any one of the two drywell pressure transmitters are inoperable.

- The inoperable LPCI A initiation channel shall be placed in a tripped condition, or the LPCI A system shall be declared inoperable if any one of the vessel level and/or drywell pressure channels are inoperable.

- The inoperable channel for LPCI B or C shall be placed in the tripped condition, or the LPCI B or C system shall be declared inoperable if any one of the 2 vessel level channels or one of the two high drywell pressure channels are inoperable.

3.2.1.5.7 Surveillance, Testing & Maintenance

The ECCS actuation signal operability is monitored as follows:

- Every 12 hours a channel check is performed. This verifies that the channel output is reading its expected value and compares favorably with its redundant channel.

- Every 31 days a functional test is performed which verifies operability of bistables and output relays. Bistables are also calibrated at this time.

- During every refueling a channel calibration is performed which verifies operability of the circuit as a whole.

3.2.1.5.8 References

GEK 83336
GEK 83337
GEK 83334A

GE Elem 807E173TY
GE Elem 807E170TY
GE Elem 807E171TY

USAR Sections 7-3, 7-4, 8-3

12177-PID-28A-9
12177-PID-28B-5
12177-PID-28C-7

Unit 2 Tech Spec Section 3/4.5
Unit 2 Tech Spec Section 3/4.3.3

3.2.1.5.9 Initiating Event Potential

The ECCS Actuation System has failure modes that can result in actuation of one or more ECCS systems. There are two classes of failures; instrumentation and instrument line.

Failure of the instrument sensing line outside of the drywell for 2ISC*LT9A(C) results in initiation of RCIC, failure of the level 8 isolation function and generation of LPCS and LPCI A initiation signal. The excess flow check valve closes and prevents the line break from becoming a small LOCA.

Failure of the instrument sensing line outside of the drywell for 2ISC*LT9B(D) results in initiation of RCIC, failure of level 8 isolation function and generation of LPCI B&C initiation signal. The excess flow check valve closes and terminates a LOCA.

Failure of either of the same line (above) inside the drywell causes a small LOCA and cause RCIC, LPCS and LPCI initiation. A high drywell pressure condition is generated.

Failure of either reference line inside of the drywell causes a high drywell pressure signal, loss of one of the two trains of initiation for RCIC; LPCS; LPCI A, B, C; and 1/2 trip of the RCIC isolation function.

3.2.1.5.10 Equipment Location

Division I Equipment	Location
Relays	H13-P629 Control Room
Bistables	H13-P629 Control Room
Power Supply	H13-P629 Control Room
Switch	H13-P601 Control Room
Level Transmitter LT9A,C	Reactor Building
Pressure Transmitter PT17A,C	Reactor Building

Division II Equipment	Location
Relays	H13-P618 Control Room
Bistables	H13-P618 Control Room
Power Supply	H13-P618 Control Room
Switch	H13-P601 Control Room
Level Transmitter LT9B,D	Reactor Building
Pressure Transmitter PT17B,D	Reactor Building

All instrument lines are located in the drywell and reactor building.

3.2.1.5.11 Operating Experience

There were no outstanding operational events relevant to this study. Plant specific component operational data is detailed in section 3.3.2.

3.2.1.5.12 Modeling Assumptions

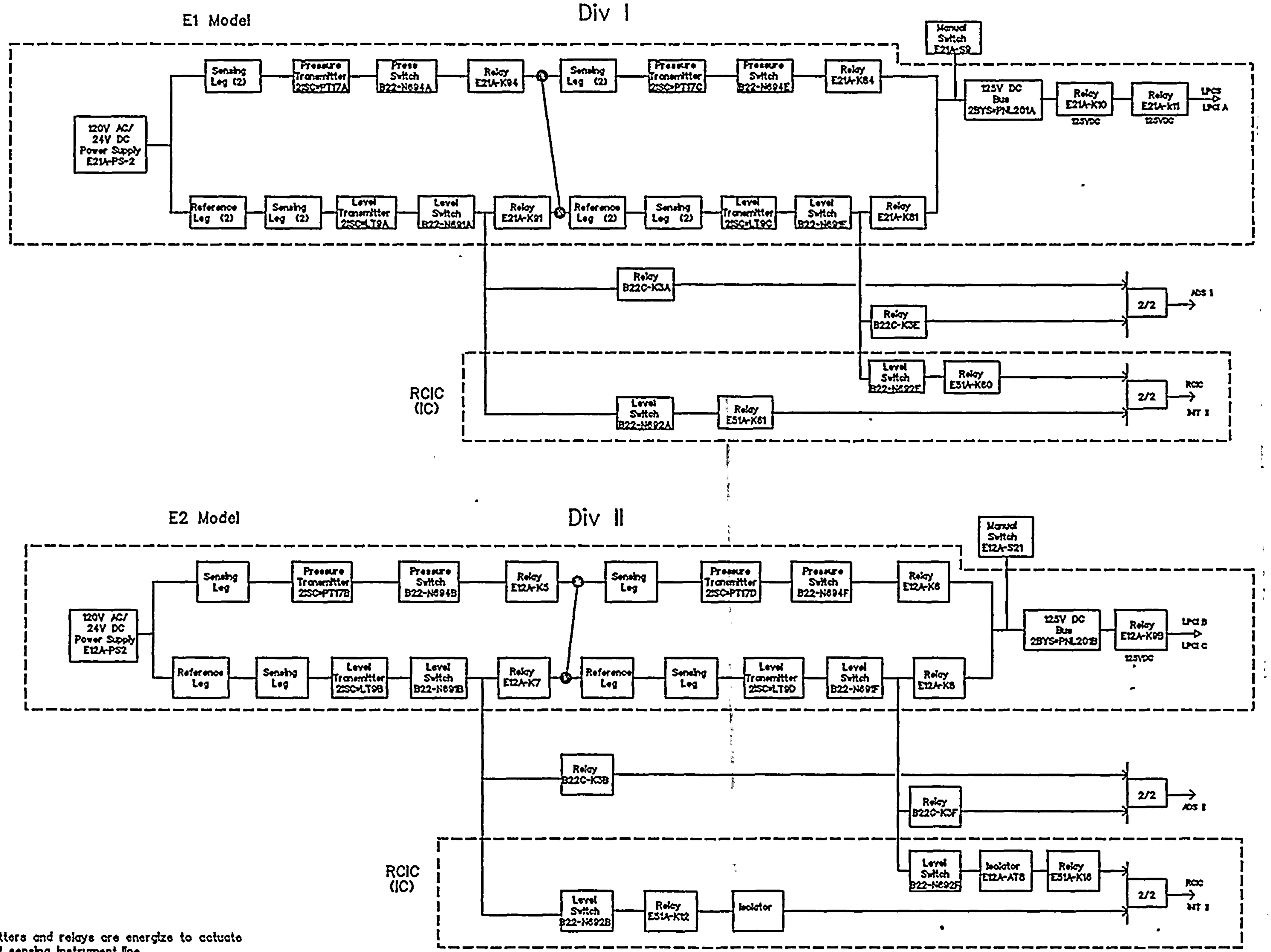
1. Failure of the excess flow check valves resulting in sensing line isolation is included in the line blockage failure mode.

2. The ECCS actuation system starts at the instrument line tap at the vessel and ends at the master relays E21A-K11 for Division I, E12A-K9B for Division II.

3.2.1.5.13 Logic Model and Results

The fault tree(s) is included in Tier 2 documentation, and is available upon request. Quantitative results are summarized in Section 3.3.5 (Tier 1).

Rev 0 (7/92)



1) All I&C transmitters and relays are energize to actuate
2) Reference and sensing instrument line failures are not modelled in the fault tree - will be treated as an initiator if necessary
3) All relays in Div I are powered by UPS "A": 2VBA•UPS2A by way of E21A-PS-2 and all relays in Div II are powered by UPS "B": 2VBA•UPS2B by way of E12A-PS-2 except where noted

SI APERTURE CARD

Also Available On Aperture Card

Figure 3.2.1.5-1
ECCS Actuation
Success Diagram

I:\NMP2IPE\DGN\2S05.DGN Jul. 07, 1992 10:32:41

9208050191 - 17

ECCS ACTUATION
Component Block Descriptions

Block	Mark No.	(Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
TOP EVENT E1								
	2ISC*LT9A	(B22-N091A)	RPV LEVEL TRANSMITTER	FAILS HIGH	N/A	N/A	2CES*RAK004	FAILS LOW
	2ISC*LT9C	(B22-N091E)	RPV LEVEL TRANSMITTER	FAILS HIGH	N/A	N/A	2CES*RAK004	FAILS LOW
	2ISC*PT17A	(B22-N094A)	DW PRESSURE TRANSMITTER	FAILS LOW	N/A	N/A	2CES*RAK004	FAILS HIGH
	2ISC*PT17C	(B22-N094E)	DW PRESSURE TRANSMITTER	FAILS LOW	N/A	N/A	2CES*RAK004	FAILS HIGH
	B22-N691A	(2ISC*LIS1691A)	LT9A LEVEL SWITCH	FAIL TO PICK UP	OPEN	CLOSED	2CEC*PNL629	CLOSED
	B22-N691E	(2ISC*LIS1691E)	LT9C LEVEL SWITCH	FAIL TO PICK UP	OPEN	CLOSED	2CEC*PNL629	CLOSED
	B22-N694A	(2ISC*PIS1694A)	PT17A PRESSURE SWITCH	FAIL TO PICK UP	OPEN	CLOSED	2CEC*PNL629	CLOSED
	B22-N694E	(2ISC*PIS1694E)	PT17C PRESSURE SWITCH	FAIL TO PICK UP	OPEN	CLOSED	2CEC*PNL629	CLOSED
	71X1-2ENSX04		EMERGENCY SEQUENCER RELAY	FAIL TO PICK UP	DE-ENERG	ENERGIZE	2BYS*SWG002A	DE-ENERGIZED
	E21A-K10	(K10-2CSLN07)	RELAY	FAILS TO CLOSE	DE-ENERG	ENERGIZE	2CEC*PNL629	DE-ENERGIZED
	E21A-K11	(K11-2CSLN07)	RELAY	FAILS TO CLOSE	DE-ENERG	ENERGIZE	2CEC*PNL629	DE-ENERGIZED
	E12A-K110A	(K110A-2RHSA32)	EMERGENCY SEQUENCER RELAY	FAIL TO PICK UP	DE-ENERG	ENERGIZE	2CEC*PNL629	DE-ENERGIZED
	E21A-K126A		EMERGENCY SEQUENCER RELAY	FAIL TO PICK UP	DE-ENERG	ENERGIZE	2BYS*PNL201A	DE-ENERGIZED
	E21A-K81	(K81-2CSLN08)	LIS1691E RELAY	FAIL TO PICK UP	DE-ENERG	ENERGIZE	2CEC*PNL629	DE-ENERGIZED
	E21A-K91	(K91-2CSLN08)	LIS1691A RELAY	FAIL TO PICK UP	DE-ENERG	ENERGIZE	2CEC*PNL629	DE-ENERGIZED
	E21A-K94A		PIS1694A & C RELAY	FAIL TO PICK UP	DE-ENERG	ENERGIZE	2BYS*PNL201A	DE-ENERGIZED
TOP EVENT E2								
	2ISC*LT9B	(B22-N091B)	RPV LEVEL TRANSMITTER	FAILS HIGH	N/A	N/A	2CES*RAK027	FAILS LOW
	2ISC*LT9D	(B22-N091F)	RPV LEVEL TANSMITTER	FAILS HIGH	N/A	N/A	2CES*RAK027	FAILS LOW
	2ISC*PT17B	(B22-N094B)	DW PRESSURE TRANSMITTER	FAILS LOW	N/A	N/A	2CES*RAK027	FAILS HIGH
	2ISC*PT17D	(B22-N094F)	DW PRESSURE TRANSMITTER	FAILS LOW	N/A	N/A	2CES*RAK027	FAILS HIGH
	B22-N691B	(2ISC*LIS1691B)	LT9B LEVEL SWITCH	FAIL TO PICK UP	OPEN	CLOSED	2CEC*PNL618	CLOSED
	B22-N691F	(2ISC*LIS1691F)	LT9D LEVEL SWITCH	FAIL TO PICK UP	OPEN	CLOSED	2CEC*PNL618	CLOSED
	B22-N694B	(2ISC*PIS1694B)	PT17B PRESSURE SWITCH	FAIL TO PICK UP	OPEN	CLOSED	2CEC*PNL618	CLOSED
	B22-N694F	(2ISC*PIS1694F)	PT17D PRESSURE SWITCH	FAIL TO PICK UP	OPEN	CLOSED	2CES*PNL618	CLOSED
	71X1-2ENSY04		EMERGENCY SEQUENCER RELAY	FAIL TO PICK UP	DE-ENERG	ENERGIZE	2BYS*SWG002B	DE-ENERGIZED
	E12A-K5	(K5-2RHSB31)	PIS1694B & F RELAY	FAIL TO PICK UP	DE-ENERG	ENERGIZE	2CEC*PNL618	DE-ENERGIZED
	E12A-K7		LIS1691B RELAY	FAIL TO PICK UP	DE-ENERG	ENERGIZE	2BYS*PNL201B	DE-ENERGIZED
	E12A-K8	(K8-2RHSB31)	LIS1691F RELAY	FAIL TO PICK UP	DE-ENERG	ENERGIZE	2CEC*PNL618	DE-ENERGIZED
	E12A-K9B		RELAY	FAILS TO CLOSE	DE-ENERG	ENERGIZE	2BYS*PNL201B	DE-ENERGIZED
	E21A-K110B		EMERGENCY SEQUENCER RELAY	FAIL TO PICK UP	DE-ENERG	ENERGIZE	2BYS*PNL201B	DE-ENERGIZED

System 6

AC Power Systems

3.2.1.6 Plant AC Electric Power

3.2.1.6.1 System Function

The Plant AC Electrical Distribution System consists of the Offsite AC Power System (115kV), the Onsite AC Power System, a 345kV transmission facility, and the Normal Station Service Transformer (Main Transformer). These systems provide sources of power to plant equipment in normal and abnormal plant operating modes.

The Onsite AC Power System consists of a normal or non-safety related (NSR) power system and a three division Emergency AC Power System. The non-safety related AC Power System is normally energized from the unit generator. When the unit generator is unavailable the NSR AC Power System is energized from offsite sources. The NSR AC Power System supplies all non-safety related loads. The Emergency AC Power System supplies power to all class 1E Safety Related equipment. It is normally energized from offsite power sources and onsite emergency generation is available as backup.

The Offsite AC Power System consists of two independent 115kV power sources from the Scriba Station. It supplies power to the Emergency AC Power System during normal and abnormal operation and is a backup for the NSR AC Power System. The 345kV transmission facility connects the unit generator to the Scriba Substation and the Niagara Mohawk Power Grid. The Normal Station Service Transformer steps down the 25kV output of the unit generator to 13.8kV for the plant Normal AC Power System.

3.2.1.6.2 Success Criteria

With the exception of Division III emergency power, AC power systems are modeled in the support system event tree with several top events as described below:

<u>Top Event</u>	<u>Success Criteria</u>
OG	Offsite AC power is available up to and including the Scriba station.
KA	115kV source A power is available from the Scriba station up to the normal AC (NA) supply and emergency AC (A1) supply as shown in Figures 3.2.1.6-1 and 3.2.1.6-3.
KB	115kV source B power is available from the Scriba station up to the normal AC (NB) supply and emergency AC (A2) supply as shown in Figures 3.2.1.6-1 and 3.2.1.6-3.
KR	Operators swap divisional AC sources to the auxiliary boiler transformer during a partial Loss of Off-Site Power (LOSP) as shown in Figures 3.2.1.6-1 and 3.2.1.6-5.

Top Event	Success Criteria
NA	Normal non-safety AC power is available after a plant trip from 115kV source A, through the 13kV bus, to normal AC switchgear (600V) as shown in Figures 3.2.1.6-2 and 3.2.1.6-4. As shown, Normal DC power is included in the model for successful transfer to the 115kV source after plant trip.
NB	Normal non-safety AC power is available after a plant trip from 115kV source B, through the 13kV, to normal AC switchgear (600V) as shown in Figure 3.2.1.6-2 and 3.2.1.6-4. As shown, Normal DC power is included in the model for successful transfer to the 115kV source after plant trip.
A1	Emergency AC power is available at the Division I switchgear and MCCs from either 115kV source A (KA) or emergency diesel EG1 as shown in Figures 3.2.1.6-1 and 3.2.1.6-5.
A2	Emergency AC power is available at the Division II switchgear and MCCs from either 115kV source B (KB) or emergency diesel EG3 as shown in Figures 3.2.1.6-1 and 3.2.1.6-5.

NOTE: Division III emergency AC is dedicated to the high pressure core spray (HPCS) system and therefore is modeled with HPCS in top event HS in the front-line event trees.

Top Event	Success Criteria
UA	120V Vital AC is available at UPS2A from either emergency AC (A1) or emergency DC (D1) as shown in Figures 3.2.1.6-1 and 3.2.1.6-6.
UB	120V Vital AC is available at UPS2B from either emergency AC (A2) or emergency DC (D2) as shown in Figures 3.2.1.6-1 and 3.2.1.6-6.

3.2.1.6.3 Support Systems

Support systems required for success of AC power top events are summarized below:

Top Event	Support System
OG	None modeled
KA	Depends on OG and is modeled in the support event tree. Specific dependencies are summarized in Table 3.2.1.6-1.
KB	Depends on OG and is modeled in the support event tree. Specific dependencies are summarized in Table 3.2.1.6-1.
KR	Depends on KA or KB, which ever is operable during a partial LOSP. Specific dependencies are summarized in Table 3.2.1.6-1.

Top Event	Support System
NA	Depends on KA and normal 125V DC for breaker transfer after a plant trip. The dependency on KA is modeled in the support system event tree. 125V NSR DC is included in the NA model. Specific dependencies are summarized in Table 3.2.1.6-2.
NB	Depends on KA and normal 125V DC for breaker transfer after a plant trip. The dependency on KB is modeled in the support system event tree. 125V NSR DC is included in the NB model. Specific dependencies are summarized in Table 3.2.1.6-2.
A1	Depends on KA as a normal source of power. On loss of offsite AC, A1 depends on Division I 125V DC (DA & D1) for diesel start and control, and breaker control. The diesel also requires service water (SA) for cooling. These dependencies are modeled in the support system event tree. Specific dependencies are summarized in Table 3.2.1.6-3.
A2	Depends on KB as a normal source of power. On loss of offsite AC, A2 depends on Division II 125V DC (DB & D2) for diesel start and control, and breaker control. The diesel also requires service water (SB) for cooling. These dependencies are modeled in the support system event tree. Specific dependencies are summarized in Table 3.2.1.6-3.
UA	Depends on Emergency AC Division I (A1) or 125V DC Division I (D1) as a supply. These dependencies are modeled in the support event tree. Specific dependencies are summarized in Table 3.2.1.6-4.
UB	Depends on Emergency AC Division II (A2) or 125V DC Division II (D2) as a supply. These dependencies are modeled in the support event tree. Specific dependencies are summarized in Table 3.2.1.6-4.

3.2.1.6.4 System Operation

Onsite Non-Safety Related AC Power System

The 13.8kV distribution system has five (5) non-safety related (NSR) and four (4) safety related buses. NSR buses 2NPS-SWG001 and 2NPS-SWG003 are normally powered from the main generator via 2STX-XNS1. They power all 13.8kV NSR motors, all NSR 4.16kV buses, and some 600V normal load centers. Under normal conditions 2NPS-SWG001 supplies safety related buses 2EPS*SWG001 and 2EPS*SWG002. 2NPS-SWG003 supplies safety related buses 2EPS*SWG003 and 2EPS*SWG004 and normally supplies NSR buses 2NPS-SWG004 and 2NPS-SWG005. Buses 2EPS*SWG001, 2, 3 and 4 supply the recirculation pumps. Backup to 2NPS-SWG001 is from the A offsite source with "cubicle only" backup from the B offsite source. Backup to 2NPS-SWG003 is from offsite source B with "cubicle only" backup from offsite source A. 2NPS-SWG002 is fed from offsite power source B via Auxiliary Boiler Transformer 2ABS-XI with backup from offsite A. 2NPS-SWG002 supplies the Auxiliary Boilers.

The plant normal Uninterruptible Power Supply (UPS) provides 120/208V AC 3-phase normal, 120V AC 1-phase normal, and 120V AC 1-phase emergency power to supply plant service, instrumentation, and control loads. The system consists of two 10kVA 120V 1-phase units, five 75kVA 120/208V 3-phase units, and one 5KVA 1 phase unit. Each unit has a normal AC source, a bypass AC source, and a DC source. In the event of a loss of AC power, DC supply power is used.

The two 10kVA UPSs (2VBB-UPS3A and 2VBB-UPS3B) supply Reactor Protection System (RPS) logic trip channel loads, Main Steam Isolation Valve (MSIV) control solenoids, and the Nuclear Monitoring System. Two (2VBB-UPS1A and 2VBB-UPS1B) of the five 75kVA UPSs supply selected NSR instrumentation and control loads. Two UPSs (2VBB-UPS1C and 2VBB-UPS1D) supply selected lighting loads. The remaining 75kVA UPS (2VBB-UPS1G) supplies all plant computer loads. The 5KVA UPS supplies the Gaseous Effluent Monitoring System (GEMS) in the main stack.

Offsite Power System

Two independent sources of offsite power are supplied to Unit 2 from the Scriba substation. The 345kv "A" Scriba bus is connected to the Reserve Station Service Transformer (2RTX-XSR1A) via motor operated disconnect switch 2YUL-MDS1, motor operated circuit switcher 2YUC-MDS3 and transformer T.B.#1, which converts·steps the line voltage from 354 KV to 115 KV. This source is called the "A" source or Line 5. The "B" source, or Line 6, is supplied from the 345kV "B" Scriba bus via motor operated disconnect switch 2YUL-MDS2, motor operated circuit switcher 2YUC-MDS4, and transformer T.B.#2, which steps the line voltage from 345KV to 115 KV. This source is connected to Reserve Station Service Transformer (2RTX-SXR1B).

A five inch bus, called the center bus, cross-connects the A and B buses. A four inch bus taps off the center bus and connects the Auxiliary Boiler Transformer which is normally energized from the 345kV "A" source.

Reserve Station Transformer 2RTX-XSR1A supplies Division I of the Onsite Emergency Power system through its 4.16kV tertiary winding and backs up the normal onsite AC power system through its 13.8kV secondary winding. Reserve Station Transformer 2RTX-XSR1B supplies Division II of the on-site emergency power system through its 4.16kV tertiary winding and backs up the normal onsite AC power system through its 13.8kV secondary winding. The Aux Boiler transformer 2ABS-X1 supplies the aux boiler and associated loads through its 13.8kV secondary winding and its 4.16kV tertiary winding serves as backup to Division I and II of the Onsite Emergency Power System.

Onsite Emergency AC Power System

1. 4.16kV switchgear buses.
2. 600V emergency load centers.
3. 600V Motor Control Centers (MCCs)
4. Distribution Centers.

5. 600-208/120 transformers.
6. 120/240 and 120V distribution panels.
7. 120V emergency UPSs.

Safety related loads are assigned one of three color coded divisions, Division I (Green), Division II (Yellow), and Division III (Purple). Division I and II supply all safety related loads except the High Pressure Core Spray System (CSH). CSH is supplied by Division III.

There are three 4.16kV emergency switchgear buses. Division I is served by 2ENS*SWG101, Division II is served by 2ENS*SWG103, and Division III is served by 2ENS*SWG102. Division I and III are normally energized from offsite source A and Division II is normally energized from offsite source B. Each bus has a dedicated emergency diesel (EDG) as backup to offsite power. EDG output breakers are normally open. The emergency buses can be fed from the alternate Reserve Station transformer as a recovery action. These breakers are normally open. Breakers can also be installed in cubicle only housings to supply the SR buses from the Auxiliary Boiler transformer.

The 600V load centers (2EJS*US1 and 2EJS*US3) are supplied from 4.16kV buses 2ENS*SWG101 and 2ENS*SWG103, respectively. These emergency load centers serve emergency motor loads, MCCs, and 600V emergency distribution panels.

The 120/208V or 120/240V distribution panels of each division supply emergency lighting and emergency instrumentation and control loads.

The plant emergency UPS system consists of two 25kVA, 120V, 1-phase UPSs. UPS 2VBS*UPS2A is normally supplied from Division I 600V power. An alternative AC source exists and DC backup is provided from DC Division I. UPS 2VBS*UPS2B is normally supplied from Division II 600V power. An alternate AC source exists and DC backup is provided from DC Division II. These two UPSs supply the ECCS instrumentation and control panels.

3.2.1.6.5 Instrumentation and Controls

Electrical control board 2CEC*PNL852 in the control room serves as the central instrumentation and control point for the AC system. Control board instruments include: ammeters, voltmeters, and synchroscopes. Control board indications display switch and breaker positions: red - closed and green - tripped. Annunciators are provided for bus overcurrent, loss of DC power, and bus undervoltage. Individual pistol grip control switches are provided for bus supply breakers. Synchronizing switches are provided for synchronizing the reserve station service transformer and the aux boiler transformer to the buses.

Operating procedures N2-OP-70, 71A, 71B and 72 describes individual instruments and controls for the electrical control board.

3.2.1.6.6 Technical Specifications

During normal operation, two physically independent circuits between the offsite transmission network and the class 1E distribution system and diesel generators must be operable. With one circuit unavailable demonstration of the other circuit's operability is made within one hour and every eight hours thereafter. The operable circuit shall be restored within 72 hours or the plant must be shutdown. With both offsite circuits inoperable, there is a one hour LCO. However, the plant will shut down immediately due to the loss of service water pumps, which will isolate RBCLC and TBCLC and cause high drywell pressure and temperature. With both diesels inoperable the plant enters a two hour LCO.

With either Division I or Division II AC distribution inoperable, re-energize in 8 hours or shutdown.

With either Division I or Division II DC distribution inoperable, re-energize in 2 hours or shutdown.

Each independent AC circuit has an operability test every seven days. Each diesel is started and run for at least one hour every month. It is also started and run for 24 hours every 18 months. If failures occur, test frequency is increased up to once every seven days.

3.2.1.6.7 Surveillance, Testing, and Maintenance

The offsite and onsite power distribution systems are determined operable at least once per seven days by verifying breaker position and indicated power availability.

The relays, timers, meters and transducers are calibrated and tested with a frequency between 1 and 6 years.

The MCCs receive Preventive Maintenance on a varying frequency.

The 4.16kV Buses (2ENS*SWG101, 102, 103) are functionally tested monthly. Loss of voltage tests are conducted every 18 months.

The diesels are tested operable monthly (during the monthly run). They are tested for automatic initiation every 18 months, and the load shedding of the emergency busses are also in this test.

3.2.1.6.8 References

N2-OP-100A Rev. 3
N2-OP-71 Rev. 3
N2-OP-72 Rev. 4
Operations Technology "Plant AC Electrical", Rev. 4
Operations Technology "Emergency AC Power System", Rev. 4
USAR Section 8, Rev. 0, 4/89
Technical Specifications 3/4.8.1.1, 3/4.8.1.2, 3/4.8.3.2, 3/4.8.1.1.2

3.2.1.6.9 Initiating Event Potential

A number of initiating events associated with loss of AC Power are included in the model, as described in section 3.1.1.

3.2.1.6.10 Equipment Location

The Division I and II emergency switchgear and 600V panels are located in physically isolated rooms at Elevation 261' of the Control Building.

The Division III switchgear is located on elevation 261' of the Reactor Building.

The three diesel generators are located in physically isolated rooms at elevation 261' in the southern portion of the Control Building. This area is also known as the Emergency Diesel Building which is comprised of the three sections containing each diesel. The normal AC switchgear and 600V load centers are located on elevation 261' of the Normal Switchgear Building. This building is located directly between the Turbine Building and the 115kV switchgear.

3.2.1.6.11 Operating Experience

There were no particular outstanding operational events relevant to this study. Plant specific component operational data is detailed in section 3.3.2.

3.2.1.6.12 Modeling Assumptions

1. Failure of any of the main buses in each division results in a loss of the entire division. This assumption is made to ease modeling. This is conservative in that divisional losses are slightly over estimated.

2. Emergency power diesel generator support equipment, such as fuel and lube oil transfer, jacket water cooling, etc. are assumed in the diesel. The room cooling, DG control room cooling and service water cooling are explicitly modeled. The diesel is also modeled for success in different time frames.

3. For normal DC power (in NA and NB), the charger output breaker is not modeled. The batteries are designed to carry their respective loads, without assistance from the chargers. The charger output breakers are not alarmed, but if one were to open, it would quickly be apparent as battery voltage dropped to the first alarm setpoint. If the AC supply to the charger were to be disrupted, there would be a control room alarm, and OPs could easily reset the charger.

4. The Diesel Building ventilation dampers are modeled only as needing to open on demand, as the dampers fail open on loss of power. This is conservative, as the

dampers may already be open when needed in an event, and are expected to remain open during an event.

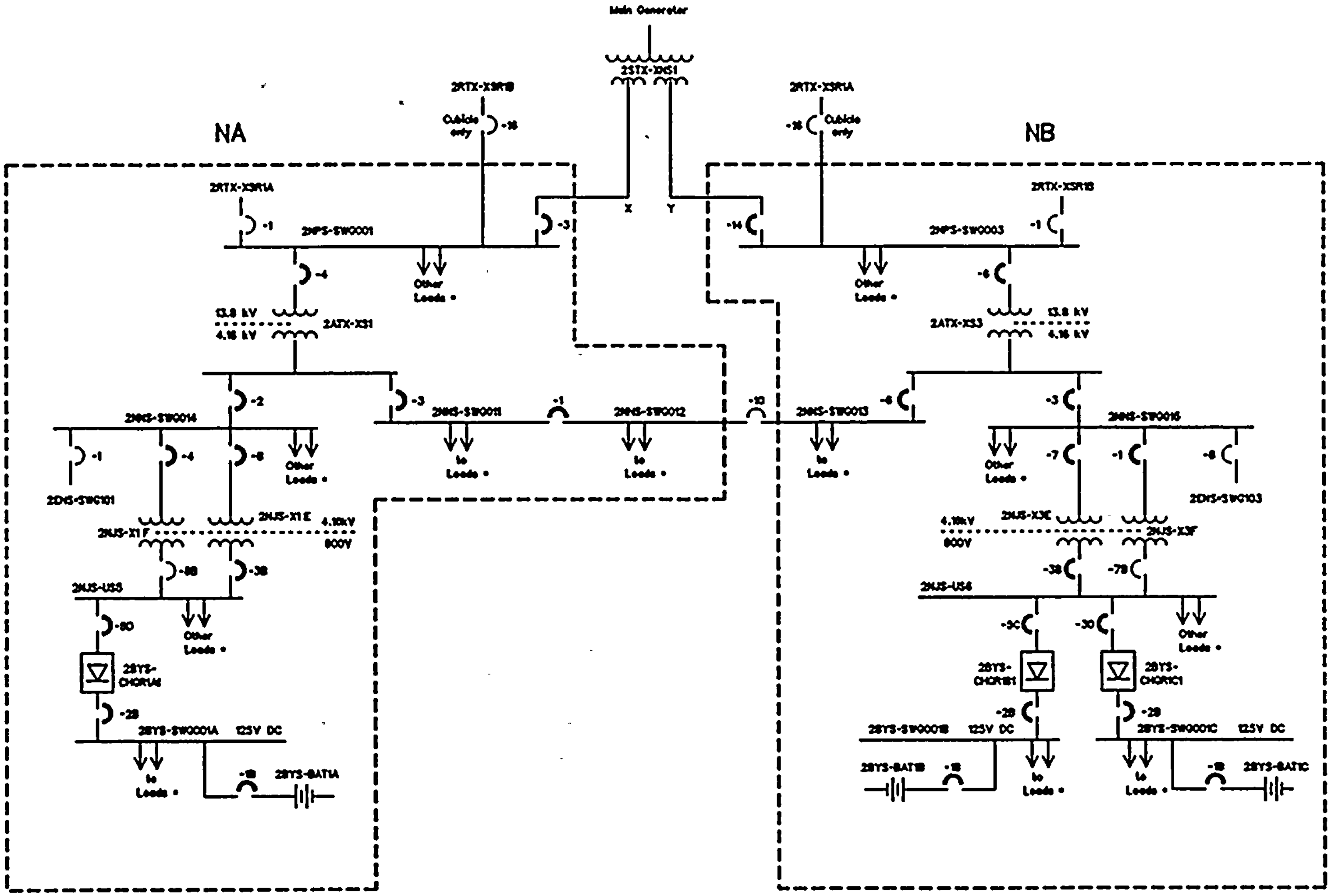
5. It is assumed that only one train of diesel building ventilation is required for diesel operability. Although Tech Specs require both trains, LER 87-39 states that the diesel would start and run for a considerable amount of time with one train operable. It is believed that operations and/or maintenance personnel would have the other train operable before failure.

6. Failure of load sequencing of the diesel generator is included in the failure of the diesel. The failure of the load sequencing is considered a small contributor in comparison to the other failure modes. The worst scenario is that the generator would trip and need to be restarted.

3.2.1.6.13 Logic Model and Results

The fault tree(s) is included in Tier 2 documentation, and is available upon request. Quantitative results are summarized in Section 3.3.5 (Tier 1).

Rev 0 (7/92)



* Loads are not included in the NA and NB models.

References: EE-M01A-5
EE-M01G-5
EE-1C-11

Figure 3.2.1.6-2
AC Power
(Top Events NA and NB)

I:\NMP2IPE\DGN\2S06.DGN Ju 06. 1992 15:30:28

Rev 0 (7/92)

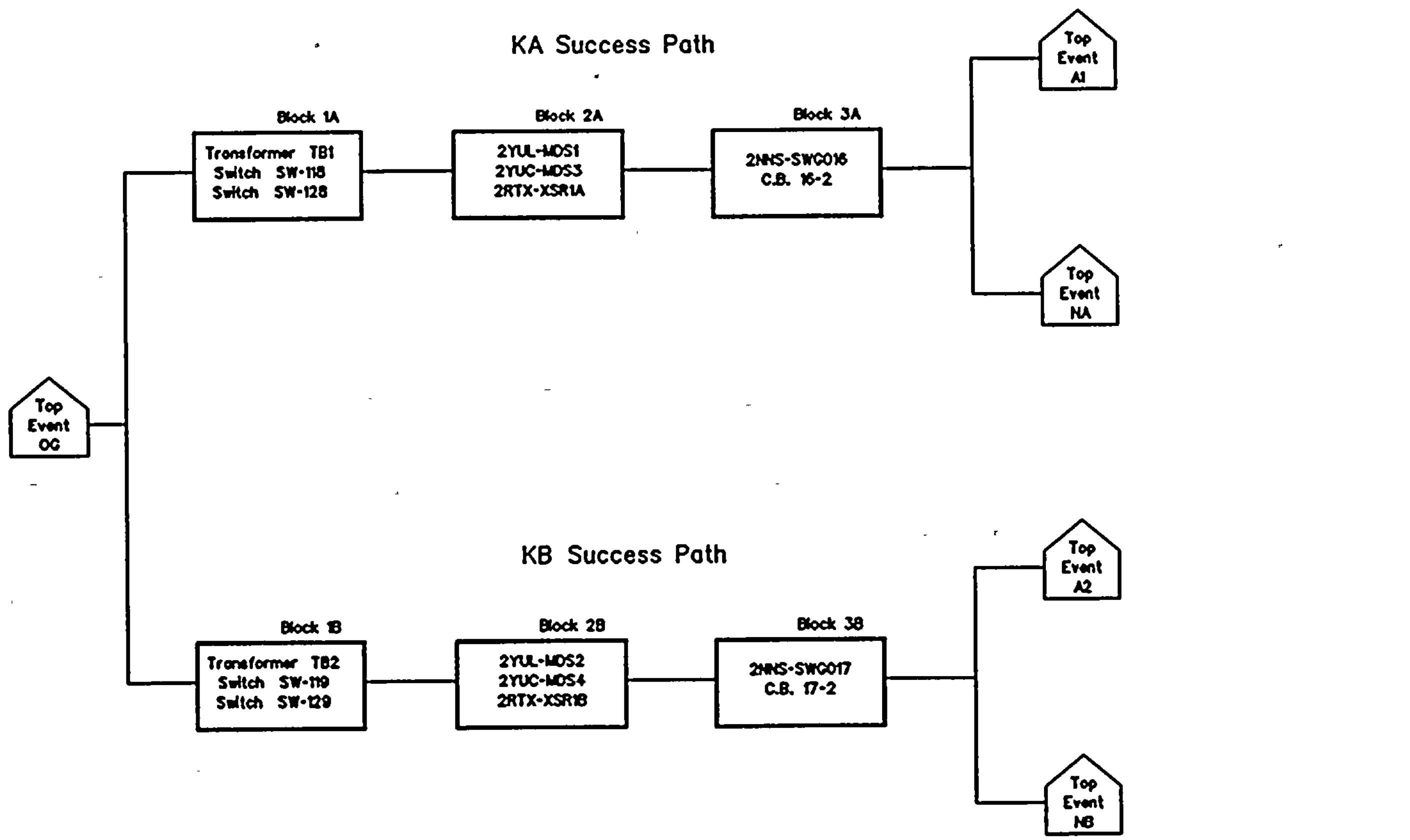


Figure 3.2.1.6-3
Top Event KA & KB Success Diagrams

I:\NMP2IPE\DGN\2S06.DGN Jul. 06, 1992 15:16:02

Rev 0 (7/92)

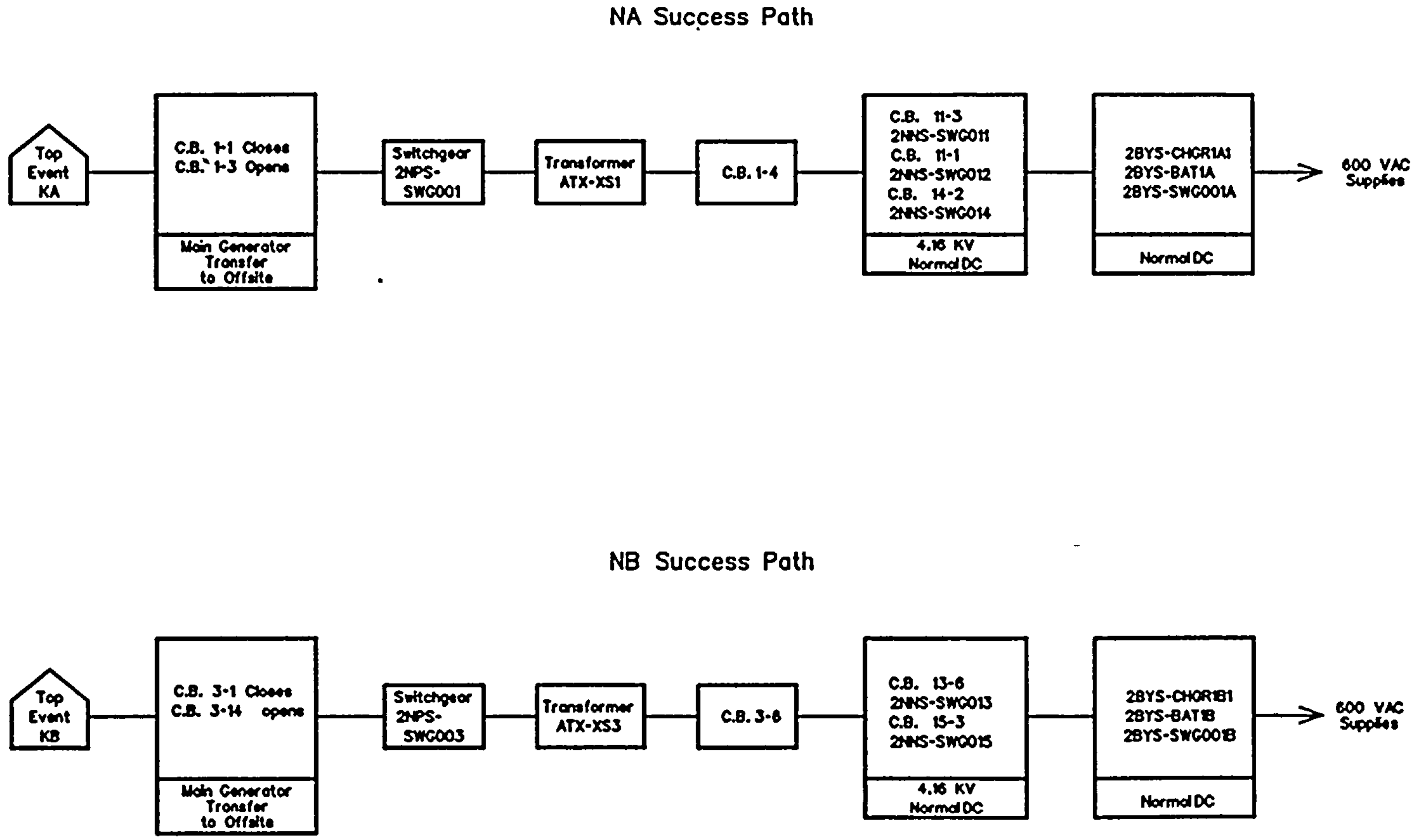
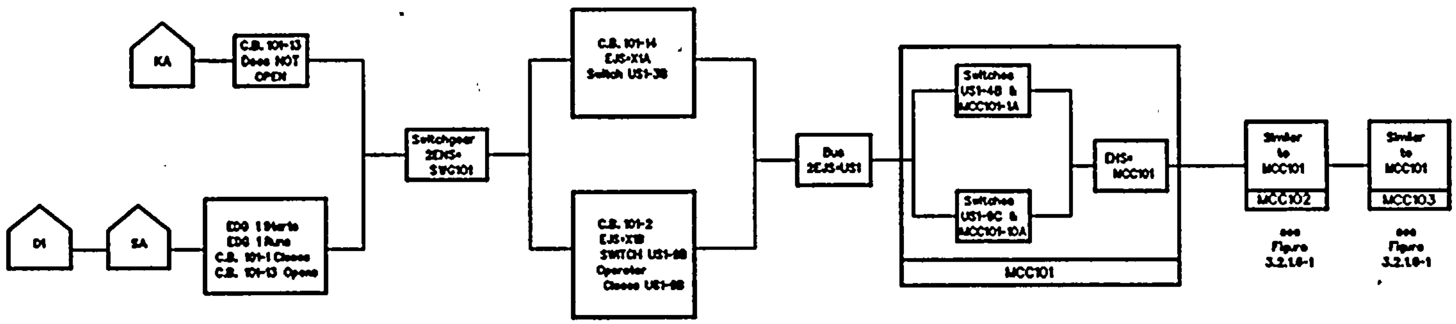


Figure 3.2.1.6-4
Top Event NA & NB Success Diagram

I:\NMP2IPE\DGN\2S06.DGN Jul. 06. 1992 15:19:02

Rev 0 (7/92)

A1 Success Path



A2 Success Path

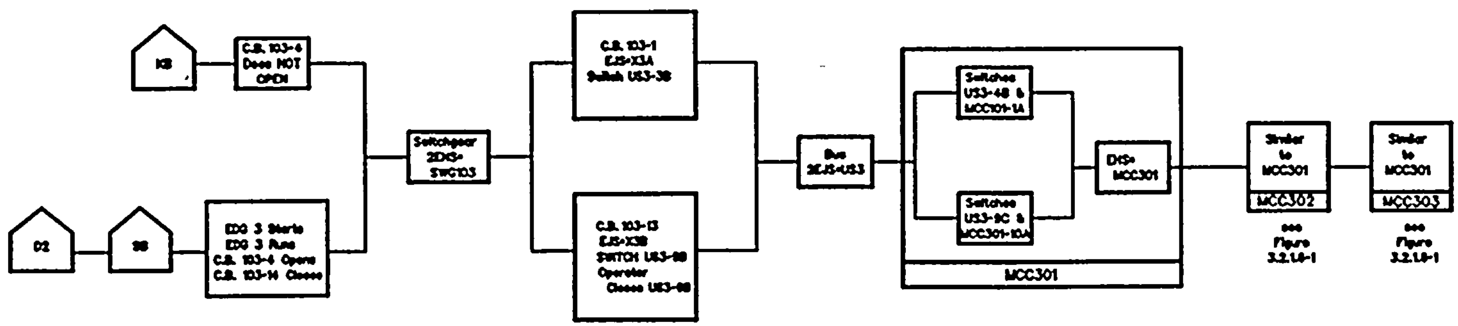
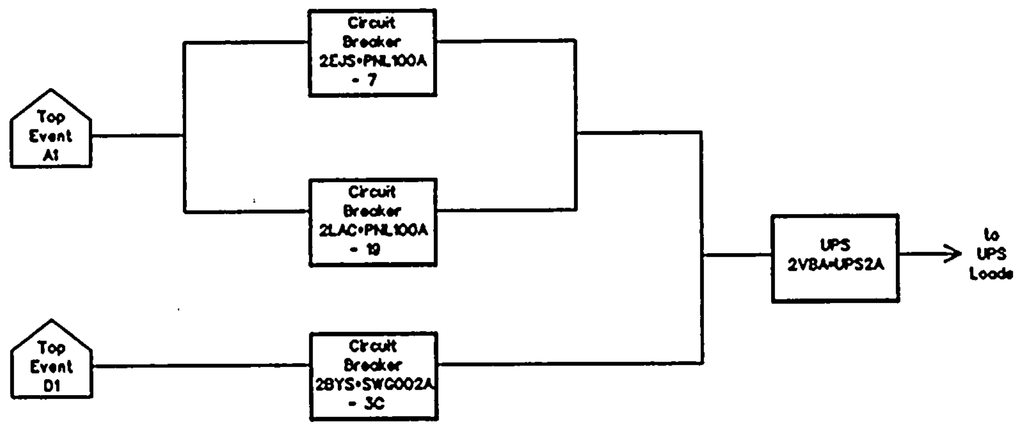


Figure 3.2.1.6-5
Top Event A1 & A2 Success Diagram

I: \NMP2IPE\DGN\2S06.DGN Jul. 06, 1992 15:20:54

Rev 0 (7/92)

UA Success Path



UB Success Path

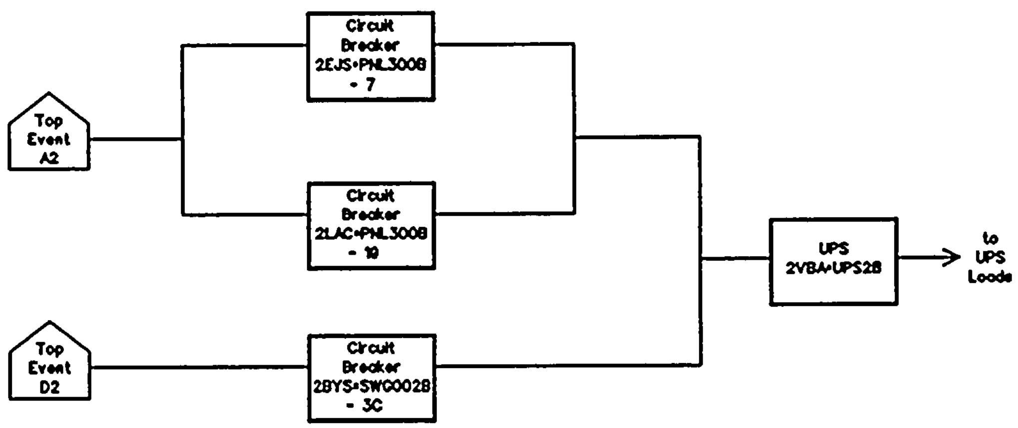


Figure 3.2.1.6-6
Top Event UA & UB Success Diagram

I:\NMP2IPE\DGN\2S06.DGN Jul. 06, 1992 15:33:03

Rev 0 (7/92)

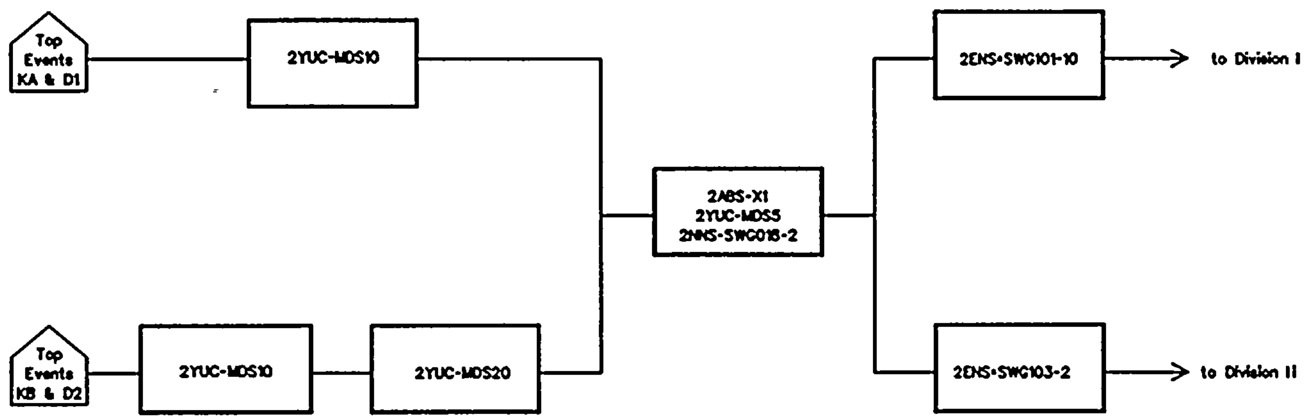
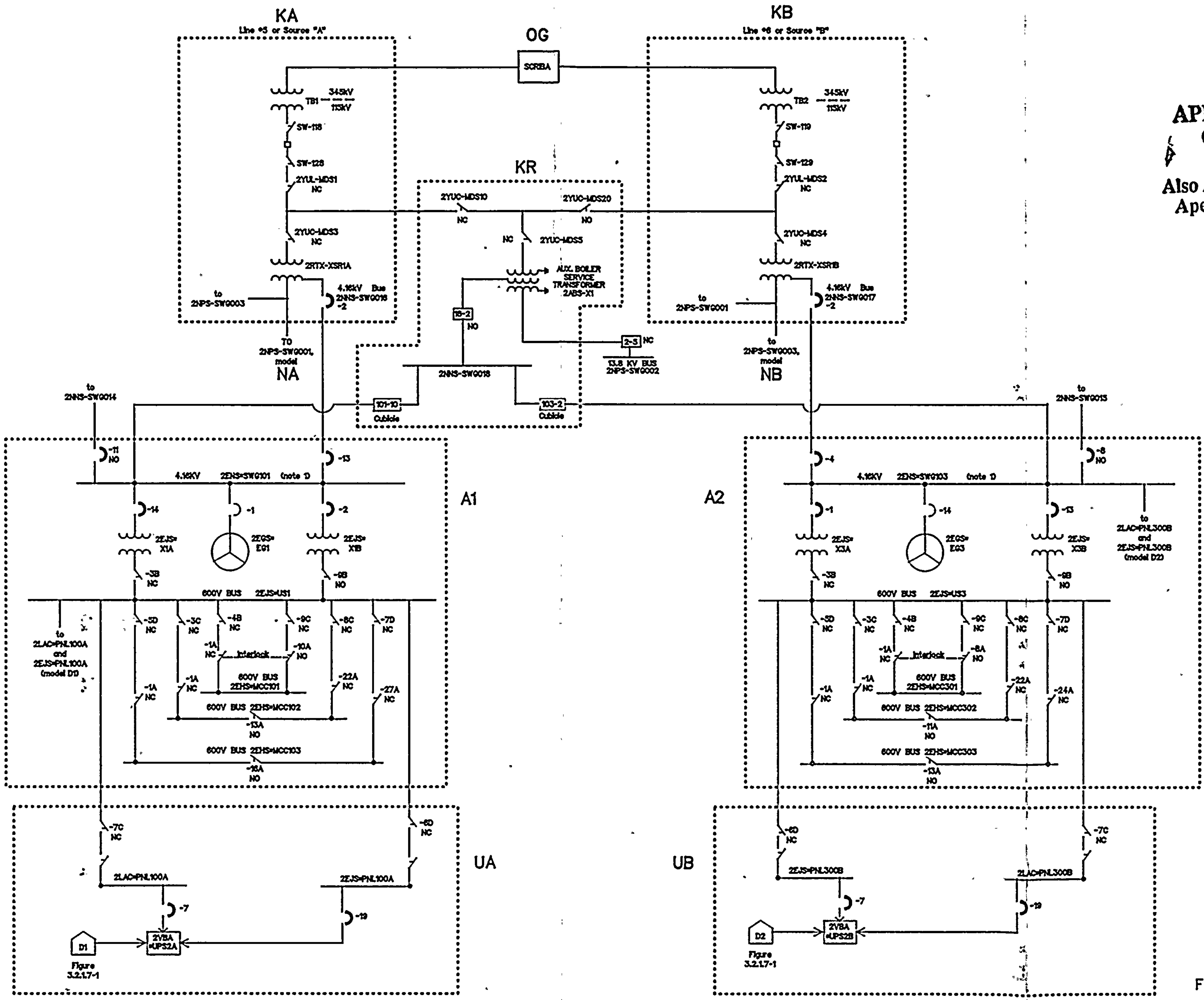


Figure 3.2.1.6-7
Top Event KR Success Diagrams

Rev 0 (7/92)



NC: Normally Closed
NO: Normally Open

1. Switchgear 2ENS*SWG101 (-G) cubicle 101-10 is mechanically KRK KEY interlocked with switchgear 2ENS*SWG103 (-Y) cubicle 103-2.

Division 1 (Green Power)
Division 2 (Yellow Power)

References: EE-M01A-5
EE-M01B-4
EE-M01E-3

SI
APERTURE
CARD

Also Available On
Aperture Card

Figure 3.2.1.6-1
AC Power

I:\NMP2IPE\DGN\2S06.DGN Jul. 07, 1992 11:47:16

9208050191 - 18

Table 3.2.1.6-1 REV. 0 (7/92)

AC POWER (Top Events KA, KB, & KR) Component Block Descriptions							
Block	Mark No. (Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
KA - TRANSFORMER TB1							
	TB1	TRANSFORMER	FAILURE	N/A	N/A	N/A	N/A
	2YUC-MDS10	SWITCH	FAILS TO OPEN	CLOSED	OPEN	2BYS-SWG001A	AS-IS
	SW-118	SWITCH	TRANSFER OPEN	OPEN	CLOSED	2BYS-SWG001A	AS-IS
	2YUC-MDS20	SWITCH	FAILS TO CLOSE	OPEN	CLOSED	2BYS-SWG001B	AS-IS
	2ENS*SWG101-10	CIRCUIT BREAKER	FAILS TO CLOSE	OPEN	CLOSED	2NNS-SWG018	AS-IS
	SW-128	SWITCH	TRANSFER OPEN	OPEN	CLOSED	2BYS-SWG001A	AS-IS
KA - TRANSFORMER XSR1A							
	2RTX-XSR1A	SERVICE TRANSFORMER	FAILURE	N/A	N/A	N/A	N/A
	2YUC-MDS10	SWITCH	TRANSFERS OPEN	CLOSED	CLOSED	2BYS-SWG001A	AS-IS
	2ENS*SWG103-2	CIRCUIT BREAKER	FAILS TO CLOSE	OPEN	CLOSED	2NNS-SWG017	AS-IS
	2YUL-MDS1	SWITCH	TRANSFER OPEN	OPEN	CLOSED	2BYS-SWG001A	AS-IS
	2YUC-MDS3	SWITCH	TRANSFER OPEN	OPEN	CLOSED	2BYS-SWG001A	AS-IS
KR - COMMON EQUIPMENT							
	2ABS-X1	TRANSFORMER	FAILS	N/A	N/A	N/A	N/A
	2NNS-SWG016	SWITCHGEAR	FAILURE	N/A	N/A	N/A	N/A
	2NNS-SWG016-2	CIRCUIT BREAKER	TRANSFER OPEN	CLOSED	OPEN	2BYS-SWG001A	AS-IS
	2YUC-MDS5	SWITCH	TRANSFERS OPEN	CLOSED	CLOSED	2BYS-SWG001A	AS-IS
	2NNS-SWG018-2	CIRCUIT BREAKER	FAILS TO CLOSE	OPEN	CLOSED	2BYS-SWG001A	AS-IS
KB - TRANSFORMER TB2							
	TB2	TRANSFORMER	FAILURE	N/A	N/A	N/A	N/A
	SW-119	SWITCH	TRANSFER OPEN	OPEN	CLOSED	2BYS-SWG001B	AS-IS
	SW-129	SWITCH	TRANSFER OPEN	OPEN	CLOSED	2BYS-SWG001B	AS-IS
KB - TRANSFORMER XSR1B							
	2RTX-XSR1B	SERVICE TRANSFORMER	FAILURE	N/A	N/A	N/A	N/A
	2YUL-MDS2	SWITCH	TRANSFER OPEN	OPEN	CLOSED	2BYS-SWG001B	AS-IS
	2YUC-MDS4	SWITCH	TRANSFER OPEN	OPEN	CLOSED	2BYS-SWG001B	AS-IS
KB - SWITCHGEAR SWG017							
	2NNS-SWG017	SWITCHGEAR	FAILURE	N/A	N/A	N/A	N/A
	2NNS-SWG017-2	CIRCUIT BREAKER	TRANSFER OPEN	OPEN	CLOSED	2BYS-SWG001A	AS-IS

Table 3.2.1.6-2 REV. 0 (7/92)

AC POWER (Top Events NA & NB)
Component Block Descriptions

Block	Mark No. (Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
NA - Main Gen. to Offsite							
	2NPS-SWG001-3	CKT BREAKER FROM MAIN TRANS	FAILS TO OPEN	CLOSED	CLOSED	2BYS-SWG001A	OPEN
	2NPS-SWG001-1	CKT BREAKER FROM RESERVE TRANS	FAILS TO CLOSE	OPEN	CLOSED	2BYS-SWG001A	OPEN
NA - Aux Transformer XS1							
	2NPS-SWG001	13.8KV SWITCHGEAR	FAILURE	N/A	N/A	N/A	N/A
	2NPS-SWG001-4	CIRCUIT BREAKER TO XS1	TRANSFERS OPEN	CLOSED	CLOSED	2BYS-SWG001A	OPEN
	2ATX-XS1	AUX STA SERVICE TRANSFORMER	FAILURE	N/A	N/A	N/A	N/A
NA - 4.16 kV Normal DC							
	2NNS-SWG011-1	CKT BREAKER BETW SWG011 & 012	TRANSFER OPEN	CLOSED	CLOSED	2BYS-SWG001A	OPEN
	2NNS-SWG011	SWITCHGEAR	FAILURE	N/A	N/A	N/A	N/A
	2NNS-SWG011-3	CIRCUIT BREAKER FROM XS1	TRANSFER OPEN	CLOSED	CLOSED	2BYS-SWG001A	OPEN
	2NNS-SWG012	SWITCHGEAR	FAILURE	N/A	N/A	N/A	N/A
	2NNS-SWG014-2	CIRCUIT BREAKER FROM XS1	TRANSFER OPEN	CLOSED	CLOSED	2BYS-SWG001A	OPEN
	2NNS-SWG014	SWITCHGEAR (STUB BUS)	FAILURE	N/A	N/A	N/A	N/A
NA - Normal DC							
	2BYS-SWG001A	BATTERY BOARD	FAILURE	N/A	N/A	2BYS-CHGR1A1	*
	2BYS-SWG001A	BATTERY BOARD	FAILURE	N/A	N/A	2BYS-BAT1A	**
	2BYS-CHGR1A1	BATTERY CHARGER	FAILURE	N/A	N/A	2NJS-US5	DE-ENERGIZED
	2BYS-BAT1A	BATTERY	FAILURE	N/A	N/A	2BYS-SWG001A	DISCHARGE
	2BYS-SWG001A-1B	CIRCUIT BREAKER FROM BATTERY	TRANSFER OPEN	CLOSED	CLOSED	N/A	N/A
NB - Main Gen. to Offsite							
	2NPS-SWG003-1	CIRCUIT BREAKER FROM RES TRANS	FAILS TO CLOSE	OPEN	CLOSED	2BYS-SWG001B	OPEN
	2NPS-SWG003-14	CKT BREAKER FROM MAIN TRANS	FAILS TO OPEN	CLOSED	CLOSED	2BYS-SWG001B	OPEN
NB - Aux Transformer XS3							
	2NPS-SWG003	13.8KV SWITCHGEAR	FAILURE	N/A	N/A	N/A	N/A
	2ATX-XS3	AUX STA SERVICE TRANSFORMER	FAILURE	N/A	N/A	N/A	N/A
	2NPS-SWG003-6	CIRCUIT BREAKER TO XS3	TRANSFER OPEN	CLOSED	CLOSED	2BYS-SWG001B	OPEN
NB - 4.16 kV Normal DC							
	2NNS-SWG013-6	CIRCUIT BREAKER FROM XS3	TRANSFER OPEN	CLOSED	CLOSED	2BYS-SWG001B	OPEN
	2NNS-SWG013	SWITCHGEAR	FAILURE	N/A	N/A	N/A	N/A
	2NNS-SWG015-3	CIRCUIT BREAKER FROM XS3	TRANSFER OPEN	CLOSED	CLOSED	2BYS-SWG001B	OPEN
	2NNS-SWG015	SWITCHGEAR	FAILURE	N/A	N/A	N/A	N/A
NB - Normal DC							
	2BYS-SWG001B	BATTERY BOARD	FAILURE	N/A	N/A	2BYS-CHGR1B1	*
	2BYS-SWG001B	BATTERY BOARD	FAILURE	N/A	N/A	2BYS-BAT1B	**

Table 3.2.1.6-2 REV. 0 (7/92)

AC POWER (Top Events NA & NB) Component Block Descriptions							
Block	Mark No. (Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
	2BYS-CHGR1B1 2BYS-BAT1B 2BYS-SWG001B-1B	BATTERY CHARGER BATTERY CIRCUIT BREAKER FROM BATTERY	FAILURE FAILURE TRANSFER OPEN	N/A N/A CLOSED	N/A N/A CLOSED	2NJS-US6 2BYS-SWG001B N/A	DE-ENERGIZED DISCHARGE N/A

AC POWER (Top Events A1 & A2)
Component Block Descriptions

Block	Mark No. (Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
A1 - Emergency AC							
	2ENS*SWG101-13	CIRCUIT BREAKER FROM SOURCE A	FAILS TO OPEN	CLOSED	CLOSED	2BYS*SWG001A	OPEN
	2EGS*EG1	DIV I EMERG DIESEL GENERATOR	FAILS TO START	STOPPED	RUNNING	N/A	STOP
	2EGS*EG1	DIV I EMERG DIESEL GENERATOR	FAILS TO RUN	STOPPED	RUNNING	N/A	STOP
	2ENS*SWG101-1	CIRCUIT BREAKER FROM EG1	FAILS TO CLOSE	OPEN	CLOSED	2BYS*SWG002A	OPEN
A1 - EDG Cooling							
	2SWP*V941A	MANUAL BLOCKING VALVE	TRANSFERS CLOSED	OPEN	OPEN	N/A	N/A
	2SWP*V231A	MANUAL BLOCKING VALVE	TRANSFERS CLOSED	OPEN	OPEN	N/A	N/A
	2SWP*V76A	CHECK VALVE	FAILS TO OPEN	CLOSED	N/A	N/A	N/A
	2SWP*MOV66A	MOTOR OPERATED VALVE	FAILS TO OPEN	CLOSED	OPEN	2EHS*MCC103	AS-IS
	2SWP*V943A	MANUAL BLOCKING VALVE	TRANSFERS CLOSED	OPEN	CLOSED	N/A	N/A
	2SWP*V944A	MANUAL BLOCKING VALVE	TRANSFERS CLOSED	OPEN	CLOSED	N/A	N/A
	2HVP*UC1A	UNIT COOLER FAN	FAILS TO START	STOPPED	RUNNING	SEJS*PNL102A	STOP
	2HVP*UC1A	UNIT COOLER FAN	FAILS TO RUN	RUNNING	RUNNING	SEJS*PNL102A	STOP
	2HVP*UC1A	UNIT COOLER	FAILS	N/A	N/A	SERVICE WATER	FAILS
	2HVP*AOD4C	AIR OPERATED DAMPER	FAILS TO OPEN	CLOSED	OPEN	INSTRUMENT AIR	FAILS OPEN
	2HVP*FN1C	EXHAUST FAN	FAILS TO START	STOPPED	RUNNING	2EHS*MCC103	STOP
	2HVP*FN1C	EXHAUST FAN	FAILS TO RUN	RUNNING	RUNNING	2EHS*MCC103	STOP
	2HVP*MOD1C	MOTOR OPERATED DAMPER	FAILS TO OPEN	CLOSED	OPEN	2SCM*PNL101A	OPEN
	2HVP*AOD4A	AIR OPERATED DAMPER	FAILS TO OPEN	CLOSED	OPEN	INSTRUMENT AIR	OPEN
	2HVP*FN1A	EXHAUST FAN	FAILS TO START	STOPPED	RUNNING	2EHS*ACC103	STOP
	2HVP*FN1A	EXHAUST FAN	FAILS TO RUN	RUNNING	RUNNING	2EHS*ACC103	STOP
	2HVP*MOD1A	MOTOR OPERATED DAMPER	FAILS TO OPEN	CLOSED	OPEN	2SCM*PNL101A	OPEN
A1 - Switchgear 101							
	2ENS*SWG101	EMERGENCY SWITCHGEAR	LOSS OF POWER	N/A	N/A	N/A	N/A
A1 - Transformer X1A							
	2ENS*SWG101-14	CIRCUIT BREAKER FROM SWG101	TRANSFERS OPEN	CLOSED	CLOSED	2BYS*SWG002A	OPEN
	2EJS*X1A	TRANSFORMER	FAILS	N/A	N/A	N/A	N/A
	2EJS*US1-3B	SUPPLY BREAKER TO US1	TRANSFER OPEN	CLOSED	CLOSED	2BYS*SWG002A	OPEN
A1 - Transformer X1B							
	2ENS*SWG101-2	CIRCUIT BREAKER FROM SWG101	TRANSFERS OPEN	CLOSED	CLOSED	2BYS*SWG002A	OPEN
	2EJS*X1B	EMERGENCY SUBST. TRANSFORMER	FAILS	N/A	N/A	N/A	N/A
	2EJS*US1-9B	SUPPLY BREAKER TO US1	TRANSFER OPEN	OPEN	CLOSED	2BYS*SWG002A	OPEN
A1 - US1							
	2EJS*US1	EMERGENCY BUS	LOSS OF POWER	N/A	N/A	N/A	N/A
A1 - MCC101							
	2EJS*US1-4B	SUPPLY BREAKER	TRANSFER OPEN	OPEN	CLOSED	2BYS*SWG002A	OPEN
	2EHS*MCC101-1A	SUPPLY BREAKER	TRANSFER OPEN	OPEN	CLOSED	2BYS*SWG002A	OPEN
	2EJS*US1-9C	BREAKER	TRANSFER OPEN	OPEN	CLOSED	2BYS*SWG002A	OPEN

Table 3.2.1.6-3 REV. 0 (7/92)

AC POWER (Top Events A1 & A2) Component Block Descriptions							
Block	Mark No. (Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
	2EHS*MCC101-10A	SUPPLY BREAKER	TRANSFER OPEN	OPEN	CLOSED	2BYS*SWG002A	OPEN
	2EHS*MCC101	MOTOR CONTROL CENTER	LOSS OF POWER	N/A	N/A	N/A	N/A
A1 - MCC102							
	2EJS*US1-3C	SUPPLY BREAKER FROM US1	TRANSFER OPEN	CLOSED	CLOSED	2BYS*SWG002A	OPEN
	2EHS*MCC102-1A	SUPPLY BREAKER TO MCC102	TRANSFER OPEN	CLOSED	CLOSED	2BYS*SWG002A	OPEN
	2EJS*US1-8C	BREAKER FROM US1	TRANSFER OPEN	CLOSED	CLOSED	2BYS*SWG002A	OPEN
	2EHS*MCC102-22A	SUPPLY BREAKER TO MCC102	TRANSFER OPEN	CLOSED	CLOSED	2BYS*SWG002A	OPEN
	2EHS*MCC102	MOTOR CONTROL CENTER	LOSS OF POWER	N/A	N/A	N/A	N/A
	2EHS*MCC102-13A	BREAKER BETWEEN BUS A & C	TRANSFER CLOSED	OPEN	CLOSED	2BYS*SWG002A	OPEN
A1 - MCC103							
	2EJS*US1-5D	SUPPLY BREAKER FROM US1	TRANSFER OPEN	CLOSED	CLOSED	2BYS*SWG002A	OPEN
	2EHS*MCC103-1A	SUPPLY BREAKER TO MCC102	TRANSFER OPEN	CLOSED	CLOSED	2BYS*SWG002A	OPEN
	2EJS*US1-7D	BREAKER FROM US1	TRANSFER OPEN	CLOSED	CLOSED	2BYS*SWG002A	OPEN
	2EHS*MCC103-27A	SUPPLY BREAKER TO MCC102	TRANSFER OPEN	CLOSED	CLOSED	2BYS*SWG002A	OPEN
	2EHS*MCC103	MOTOR CONTROL CENTER	LOSS OF POWER	N/A	N/A	N/A	N/A
	2EHS*MCC103-16A	BREAKER BETWEEN BUS A & C	TRANSFER CLOSED	OPEN	CLOSED	2BYS*SWG002A	OPEN
A2 - Emergency AC							
	2ENS*SWG103-4	CIRCUIT BREAKER FROM SOURCE B	FAILS TO OPEN	CLOSED	CLOSED	2BYS*SWG001A	OPEN
	2EGS*EG3	DIV II EMERG DIESEL GENERATOR	FAILS TO START	STOPPED	RUNNING	N/A	STOP
	2EGS*EG3	DIV II EMERG DIESEL GENERATOR	FAILS TO RUN	STOPPED	RUNNING	N/A	STOP
	2ENS*SWG103-14	CIRCUIT BREAKER FROM EG3	FAILS TO CLOSE	OPEN	CLOSED	2BYS*SWG002B	OPEN
A2 - EDG Cooling							
	2SWP*941B	MANUAL BLOCKING VALVE	TRANSFERS CLOSED	OPEN	OPEN	N/A	N/A
	2SWP*231B	MANUAL BLOCKING VALVE	TRANSFERS CLOSED	OPEN	OPEN	N/A	N/A
	2SWP*76B	CHECK VALVE	FAILS TO OPEN	CLOSED	N/A	N/A	N/A
	2SWP*MOV66B	MOTOR OPERATED VALVE	FAILS TO OPEN	CLOSED	OPEN	2EHS*MCC303	AS-IS
	2SWP*V943B	MANUAL BLOCKING VALVE	TRANSFERS CLOSED	OPEN	CLOSED	N/A	N/A
	2SWP*V944B	MANUAL BLOCKING VALVE	TRANSFERS CLOSED	OPEN	CLOSED	N/A	N/A
	2HVP*UC1B	UNIT COOLER FAN	FAILS TO START	STOPPED	RUNNING	2EJS*PNL301B	STOP
	2HVP*UC1B	UNIT COOLER FAN	FAILS TO RUN	RUNNING	RUNNING	2EJS*PNL301B	STOP
	2HVP*UC1B	UNIT COOLER	FAILS	N/A	N/A	SERVICE WATER	FAILS
	2HVP*AOD4D	AIR OPERATED DAMPER	FAILS TO OPEN	CLOSED	OPEN	INSTRUMENT AIR	FAILS OPEN
	2HVP*FN1D	EXHAUST FAN	FAILS TO START	STOPPED	RUNNING	2EHS*MCC303	STOP
	2HVP*FN1D	EXHAUST FAN	FAILS TO RUN	RUNNING	RUNNING	2EHS*MCC303	STOP
	2HVP*MOD1D	MOTOR OPERATED DAMPER	FAILS TO OPEN	CLOSED	OPEN	2SCM*PNL301B	OPEN
	2HVP*AOD4B	AIR OPERATED DAMPER	FAILS TO OPEN	CLOSED	OPEN	INSTRUMENT AIR	OPEN
	2HVP*FN1B	EXHAUST FAN	FAILS TO START	STOPPED	RUNNING	2EHS*MCC303	STOP
	2HVP*FN1B	EXHAUST FAN	FAILS TO RUN	RUNNING	RUNNING	2EHS*MCC303	STOP
	2HVP*MOD1B	MOTOR OPERATED DAMPER	FAILS TO OPEN	CLOSED	OPEN	2SCM*PNL301B	OPEN
A2 - Switchgear 103							
	2ENS*SWG103	EMERGENCY SWITCHGEAR	LOSS OF POWER	N/A	N/A	N/A	N/A

AC POWER (Top Events A1 & A2)
Component Block Descriptions

Block	Mark No. (Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
A2 - Transformer X3A							
	2EHS*SWG103-1	CIRCUIT BREAKER FROM SWG103	FAILS TO OPEN	CLOSED	CLOSED	2BYS*SWG002B	OPEN
	2EJS*X3A	TRANSFORMER	FAILS	N/A	N/A	N/A	N/A
	2EJS*US3-3B	BREAKER TO US3	TRANSFER OPEN	CLOSED	CLOSED	2BYS*SWG002B	OPEN
A2 - Transformer X3B							
	2EHS*SWG103-13	CIRCUIT BREAKER FROM SWG103	TRANSFER OPEN	CLOSED	CLOSED	2BYS*SWG002B	OPEN
	2EJS*X3B	TRANSFORMER	FAILS	N/A	N/A	N/A	N/A
	2EJS*US3-9B	BREAKER TO US3	FAILS TO CLOSE	OPEN	CLOSED	2BYS*SWG002B	OPEN
A2 - US3							
	2EJS*US3	EMERGENCY BUS	LOSS OF POWER	N/A	N/A	N/A	N/A
A2 - MCC301							
	2EJS*US3-4B	SUPPLY BREAKER	TRANSFER OPEN	CLOSED	CLOSED	2BYS*SWG002B	OPEN
	2EHS*MCC301-1A	SUPPLY BREAKER	TRANSFER OPEN	CLOSED	CLOSED	2BYS*SWG002B	OPEN
	2EJS*US3-9C	SUPPLY BREAKER	TRANSFER OPEN	CLOSED	CLOSED	2BYS*SWG002B	OPEN
	2EHS*MCC301-8A	SUPPLY BREAKER	TRANSFER OPEN	OPEN	CLOSED	2BYS*SWG002B	OPEN
	2EHS*MCC301	EMERGENCY MOTOR CONTROL CENTER	LOSS OF POWER	N/A	N/A	N/A	N/A
A2 - MCC302							
	2EJS*US3-3C	SUPPLY BREAKER FROM US3	TRANSFER OPEN	CLOSED	CLOSED	2BYS*SWG002B	OPEN
	2EHS*MCC302-1A	SUPPLY BREAKER TO MCC1A	TRANSFER OPEN	CLOSED	CLOSED	2BYS*SWG002B	OPEN
	2EJS*US3-8C	SUPPLY BREAKER US3	TRANSFER OPEN	CLOSED	CLOSED	2BYS*SWG002B	OPEN
	2EHS*MCC302-22A	SUPPLY BREAKER TO MCC1A	TRANSFER OPEN	CLOSED	CLOSED	2BYS*SWG002B	OPEN
	2EHS*MCC302	EMERGENCY MOTOR CONTROL CENTER	LOSS OF POWER	N/A	N/A	N/A	N/A
	2EHS*MCC302-11A	BREAKER BETWEEN BUS B & D	TRANSFER CLOSED	OPEN	CLOSED	2BYS*SWG002B	OPEN
A2 - MCC303							
	2EJS*US3-5D	SUPPLY BREAKER FROM US3	TRANSFER OPEN	CLOSED	CLOSED	2BYS*SWG002B	OPEN
	2EHS*MCC303-1A	SUPPLY BREAKER TO MCC1A	TRANSFER OPEN	CLOSED	CLOSED	2BYS*SWG002B	OPEN
	2EJS*US3-7D	SUPPLY BREAKER FROM US3	TRANSFER OPEN	CLOSED	CLOSED	2BYS*SWG002B	OPEN
	2EHS*MCC303-24A	SUPPLY BREAKER TO MCC1A	TRANSFER OPEN	CLOSED	CLOSED	2BYS*SWG002B	OPEN
	2EHS*MCC303	EMERGENCY MOTOR CONTROL CENTER	LOSS OF POWER	N/A	N/A	N/A	N/A
	2EHS*MCC303-13A	BREAKER BETWEEN BUS B & D	TRANSFER CLOSED	OPEN	CLOSED	2BYS*SWG002B	OPEN

Table 3.2.1.6-4 REV. 0 (7/92)

AC POWER (Top Events UA & UB)
Component Block Descriptions

Block	Mark No. (Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
TOP EVENT UA							
	2BYS*SWG002A-3C	CIRCUIT BREAKER FROM SWG002A	TRANSFER OPEN	CLOSED	CLOSED	2BYS*SWG002A	OPEN
	2EJS*PNL100A-7	CKT BREAKER FROM 2EJS*PNL100A	TRANSFER OPEN	CLOSED	CLOSED	2BYS*SWG002A	OPEN
	2LAC*PNL100A-19	CKT BREAKER FROM 2LAC*PNL100A	TRANSFER OPEN	CLOSED	CLOSED	2BYS*SWG002A	OPEN
	2VBA*UPS2A	VITAL BUS SYSTEM - UPS	FAILURE	N/A	N/A	2EJS*PNL100A	N/A
TOP EVENT UB							
	2BYS*SWG002B-3C	CIRCUIT BREAKER FROM SWG002B	TRANSFER OPEN	CLOSED	CLOSED	2BYS*SWG002B	OPEN
	2EJS*PNL300B-7	CKT BREAKER FROM 2EJS*PNL300B	TRANSFER OPEN	CLOSED	CLOSED	2BYS*SWG002B	OPEN
	2LAC*PNL300B-19	CKT BREAKER FROM 2LAC*PNL300B	TRANSFER OPEN	CLOSED	CLOSED	2BYS*SWG002B	OPEN
	2VBA*UPS2B	VITAL BUS SYSTEM - UPS	FAILURE	N/A	N/A	2EJS*PNL300B	N/A

System 7

DC Power Systems

3.2.1.7 Plant DC Electric Power

3.2.1.7.1 System Function

The Plant DC Electrical Distribution System provides a reliable source of DC Power to plant DC control power circuits, instrumentation, DC motors, neutron monitoring, and other essential loads.

3.2.1.7.2 Success Criteria

Success is a continuous supply of DC power to essential plant equipment via divisional batteries or battery chargers as shown in Figures 3.2.1.7-1 & 2. Top events D1 and D2 (Division I and II DC power normal load conditions) are modeled in the support event tree. DA and DB are modeled in the support event tree as demand failures on the batteries during the starting of large DC equipment.

NOTE: Normal DC is described here, but is modeled with normal AC power as described in Section 3.2.1.6. Division III DC power is modeled with the HPCS system, in Section 3.2.1.1.

3.2.1.7.3 Support Systems

Each Emergency DC division is fed from an associated Emergency 600V AC power supply as shown in Figure 3.2.1.7-1. Table 3.2.1.7-1 provides specific detail on Emergency DC component support dependencies.

Each normal 125V DC battery charger is supplied by a 600V power supply.

Each battery room is provided with smoke detection, ventilation, and lighting.

3.2.1.7.4 System Operation

The DC Electrical Distribution System consists of the normal or Non-Safety Related (NSR) 125V DC system and the 125V DC Emergency Power system.

<u>NSR 125V DC System</u>
The NSR 125V DC system consists of the NSR 125V DC system, the 24V DC system, and their associated distribution systems. The NSR 125V DC system provides DC power to the normal switchgear, the main transformer, the reserve station service transformers, the auxiliary boiler transformer, and other NSR systems.

The NSR 125V DC system battery chargers receive power from the 600V AC system. The system is comprised of three subsystems. Each subsystem consists of its own battery bank, battery charger, and DC switchgear or distribution panel.

The 24V DC power system provides two redundant DC power sources for the Neutron Monitoring System and the emergency response facility optical isolators. Each redundant subsystem consists of a three wire bus, two 24V DC batteries, and two 24V DC battery chargers.

All 125 and 24V DC loads are powered from manually closed circuit breakers. Loads are normally powered from the battery chargers. During momentary demands in excess of charger capacity, the battery provides the additional capacity required to prevent the chargers from tripping. During loss of offsite power and diesel generator failure, essential DC loads are supplied entirely from the batteries.

Emergency 125V DC System

The Emergency 125V DC system consists of three divisions: Divisions I (Green), II (Yellow), and III (Purple). Each division corresponds to the similarly named Emergency AC division. Each DC division consists of two battery chargers fed from Emergency AC, a battery, and switchgear. The following table lists general equipment supported by each DC division.

Division I

- Protection and Control for 4.16kV Switchgear 2ENS*SWG101
- Protection and Control for 13.8kV Switchgear 2EPS*SWG001, *SWG003
- Protection and Control for 600V Load Center 2EJS*US1
- UPS 2VBA*UPS2A
- Division I EDG Control Panels via panel 2BYS*PNL201A
- Division I EDG Field Flashing
- Division I PGCC Control Circuits via 2BYS*PNL201A
- RCIC system loads via 2DMS*MCCA1

Division II

- Protection and Control for 4.16kV Switchgear 2ENS*SWG103
- Protection and Control for 13.8kV Switchgear 2EPS*SWG002, *SWG004
- Protection and Control for 600V Load Center 2EJS*US3
- UPS 2VBA*UPS2B
- Division II EDG Control Panels via panel 2BYS*PNL201B
- Division II EDG Field Flashing
- Division II PGCC Control Circuits via 2BYS*PNL201B

Division III

- Protection and Control for 4.16kV Switchgear 2ENS*SWG102
- Division III EDG Control Panels via panel 2CES*IPNL414
- Division III EDG Fuel Pump
- Division III EDG Field Flashing
- CSH Solenoid Valves
- CSH Relay Panel
- CSH Control Room Indicator Lamps

Each battery bank has two 100 percent capacity battery chargers capable of maintaining battery charge and providing DC supply power. They are also capable of charging batteries from design minimum charge to full charge in 24 hours. Each charger is fed from its associated 600V AC power source.

The DC switchgear is connected to the battery via a circuit breaker. For Division I and II it is connected to the two battery chargers via a single circuit breaker and for Division III each charger has an associated circuit breaker.

3.2.1.7.5 Instrumentation and Controls

The DC Distribution System has Control Room indication on the back of PNL852. This indication includes battery bus voltage, battery bus amperage, battery charger voltage, and battery charger amperage.

Each battery has a pistol grip control in the Control Room for positive, normal, and negative positions. A test push-button is provided for ground detection.

Division I, II, and III battery chargers have over-voltage protection that disconnects the AC input from the chargers when DC output voltage exceeds a manually set value.

125V DC instrumentation is located on the back of PNL852 in the Control Room.

3.2.1.7.6 Technical Specifications

Divisions I and II are required to be operable with one battery and one battery charger, and have an Allowed Outage Time (AOT) of 2 hours or go to Hot Shut Down (HSD).

Both chargers are required when a UPS is drawing power from DC (eg testing backup supply). If two chargers are not available, there is an AOT of 2 hours, or go to hot shutdown.

Division III must have one charger and battery operable or CSH must be declared inoperable.

3.2.1.7.7 Surveillance, Testing, and Maintenance

Battery and charger parameters must be verified every seven days.

After 92 days, within 7 days of an over-charge, or discharge below 107V verify battery parameters, check for corrosion, and inspect electrolyte.

Every 18 months perform load tests on charger and battery (battery during outage). Also check cell-to-cell resistance.

Battery load testing is completed during planned outages. DC load is maintained entirely from the chargers during the test.

Every 60 months, during shutdown, perform battery discharge test, (From Tech Specs 3/4.8). Tables of battery parameters are shown in the Tech Specs.

3.2.1.7.8 References

N2-OP-73A Rev. 2, "Normal DC Distribution," Operating Procedure
N2-OP-73B Rev. 1, "24V DC Distribution," Operating Procedure
N2-OP-74A Rev. 2, "Emergency DC Distribution," Operating Procedure
N2-OP-74B Rev. 2, "HPCS Emergency DC Distribution," Operating Procedure
AE-100B Rev. 2, "DC Load List"
Tech. Spec. Section 3/4.8.2

3.2.1.7.9 Initiating Event Potential

The DC systems have limited potential for causing an initiating event. However, loss of DC will affect protection and control in vital AC switchgear and could result in station blackout.

Loss of either Divisional DC Battery Bus will result in Reactor Recirc Pumps tripping which requires a manual scram.

3.2.1.7.10 Equipment Location

The batteries, battery chargers, and switchgear for each Safety Related DC division are located in separate battery rooms in the Control Building on elevation 261'. However, the Division III distribution panel (2CES*IPNL414) is located in the Diesel Generator Building at elevation 261'.

The three NSR 125V DC BYS subsystems are located in separate rooms. Battery, battery charger, and switchgear 1A are located on elevation 237' of the switchgear building. The 1B equipment is located in a separate room adjacent to 1A on elevation 237' of the switchgear building. Subsystem 1C is located on elevation 214' of the Control Building.

Subsystems 3A and 3B of the BWS (24V DC) system are also located in separate rooms on elevation 274' of the Control Building.

3.2.1.7.11 Operating Experience

There were no particular outstanding operational events relevant to this study. Plant specific component operational data is detailed in section 3.3.2.

3.2.1.7.12 Modeling Assumptions

1. A demand failure and a continuous duty failure are both included as battery failure modes.

2. Four models are used, DA and DB are for large load demands which require the batteries for supplemental power (the chargers alone do not supply enough power). D1 and D2 model the battery and the chargers during small load conditions, where the chargers can supply all the load.

3. Safety related batteries being unavailable at power is not modeled. Technical Specifications give a two hour LCO, or be shutdown within the next 12 hours. Hence, the likelihood of a battery being unavailable at the onset of an accident is very small. Battery unavailable during normal operation is a small contributor to the overall model.

3.2.1.7.13 Logic Model and Results

The fault tree(s) is included in Tier 2 documentation, and is available upon request. Quantitative results are summarized in Section 3.3.5 (Tier 1).

3.2.1.7.14 Generic Safety Issue

Generic Letter 91-06 requested evaluation of the emergency DC power systems. The IPE project responded to two of the items in the generic letter. These items are discussed below.

QUESTION:

Does the control room at this unit have the following separate, independently annunciated alarms and indications for each division of power?

a.1. Battery charger disconnect or circuit breaker open (both input AC and output DC)?

RESPONSE:

No. The three circuit breakers in question are shown on Figure 3.2.1.7-1, one input to each battery charger and one common output breaker to the DC bus. There are also two breakers internal to each charger (not shown).

If the charger AC input breaker were to open, a division trouble alarm would soon alert operators. The procedure instructs the operator to check which computer point alarmed, in this case a low bus voltage of 125V DC. It then instructs the operators to check various voltage and amp meters to determine what caused the low voltage, and to put the redundant charger on line. The process is explicitly described in the procedure, and the appropriate Tech. Specs. are referenced.

If the charger DC output breaker opened, the same control room window would alarm, and the computer point would alert the operators that the breaker opened. Trouble shooting to quickly determine the cause is explicitly described in the procedure, and the appropriate Tech. Specs. are referenced.

If the charger internal AC breaker were to open, the same alarm would alert the operators to the condition, and the computer point would tell which signal tripped the alarm. Trouble shooting to quickly determine the cause is explicitly described in the procedure, and the appropriate Tech. Specs. are referenced.

If the charger internal DC breaker were to trip, the same window would light and alarm, but this time due to a bus low voltage of 125V DC. The operators are instructed to put the redundant charger on line until the inoperable charger can be repaired. Trouble shooting to quickly determine the cause is explicitly described in the procedure, and the appropriate Tech. Specs. are referenced.

Because of the alarms and procedures, the chargers would not be unavailable for a long period of time. The combination of the low frequency and short duration of a breaker open event results in a small unavailability. Consequently, a breaker open event occurring coincident with a battery demand is a very low frequency event.

Battery maintenance was also quantified in the IPE model, and does not significantly contribute to the results.

QUESTION:

a.4. Does this unit have indication of bypassed and inoperable status of circuit breakers or other devices that can be used to disconnect the battery and battery charger from it's DC bus and the battery charger from it's AC power source during maintenance or testing?

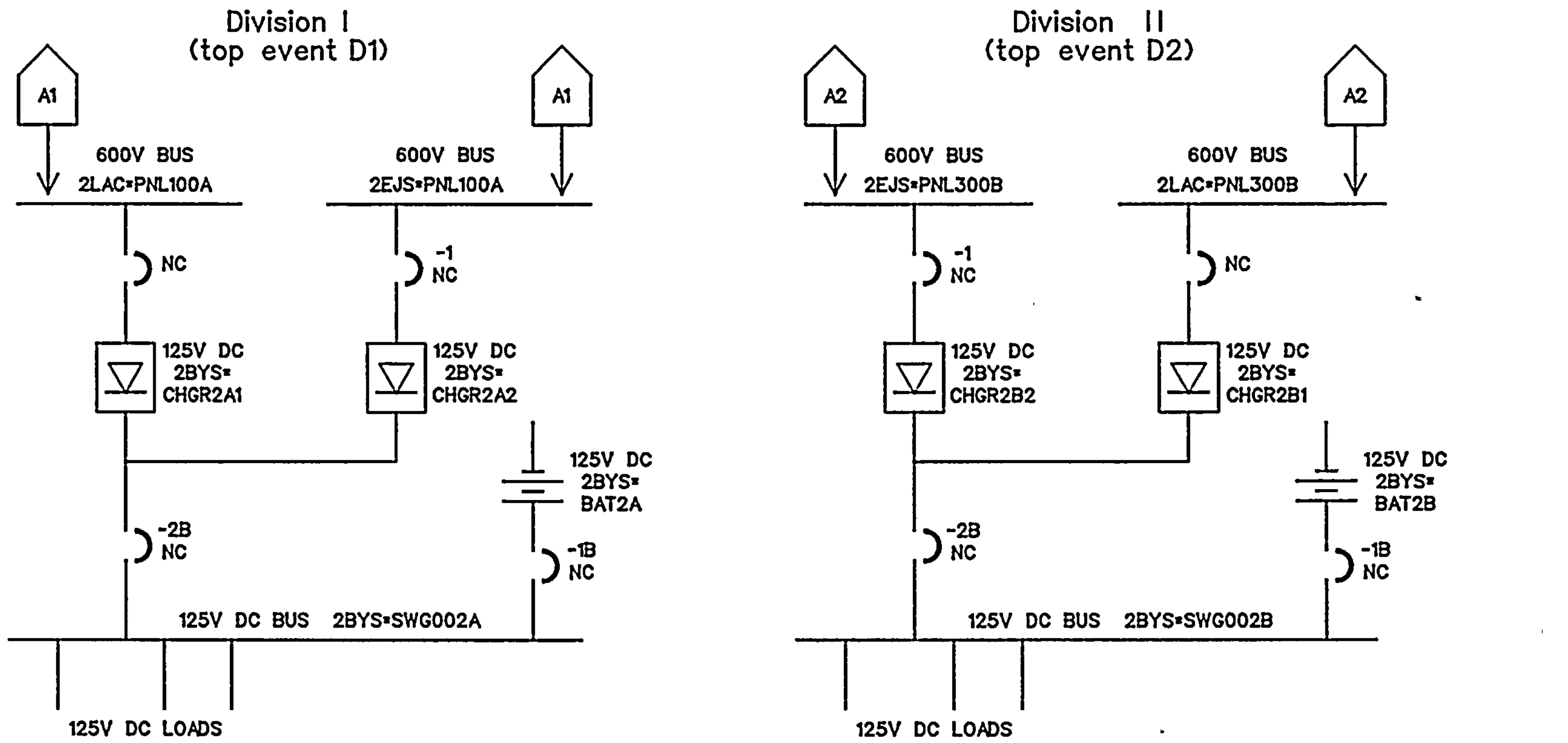
RESPONSE:

No. Preventive maintenance that would disconnect the battery from the bus is only done during shutdown. The only method of disconnecting the battery is by using the circuit breaker (2BYS*SWG002A-1B), which is alarmed. When the alarm comes in, there is both a audible alarm and a flashing light. When the operator acknowledges the alarm, the audible quits, the window stops flashing and goes to a solid light. The alarm windows are checked during each shift. Also, in the operating procedure system line-up, there is an independent verification of breaker position. Finally, if maintenance was performed, and the breaker was left open, the Post Maintenance Test (PMT) should determine this condition.

The batteries can be disconnected from the bus for corrective maintenance while at power, but this is only if it fails a surveillance test or a ground fault is detected. For a ground fault, the battery is momentarily disconnected to determine if it has the fault. If not, it is immediately reconnected, and if it is faulted, maintenance is done and it is returned to service. The control room alarm would remain lit until the breaker was closed, and there would also be a PMT.

There is also no way to disconnect the battery charger from the bus other than with the circuit breakers discussed above. There are no internal fuses which could disconnect it. Again, battery charger and battery unavailability is modeled in the IPE and does not significantly contribute to the results.

Rev 0 (7/92)



Division 1 (Green Power)

Division 2 (Yellow Power)

References: EE-M01F-4

Figure 3.2.1.7-1
Emergency DC Power

Rev 0 (7/92)

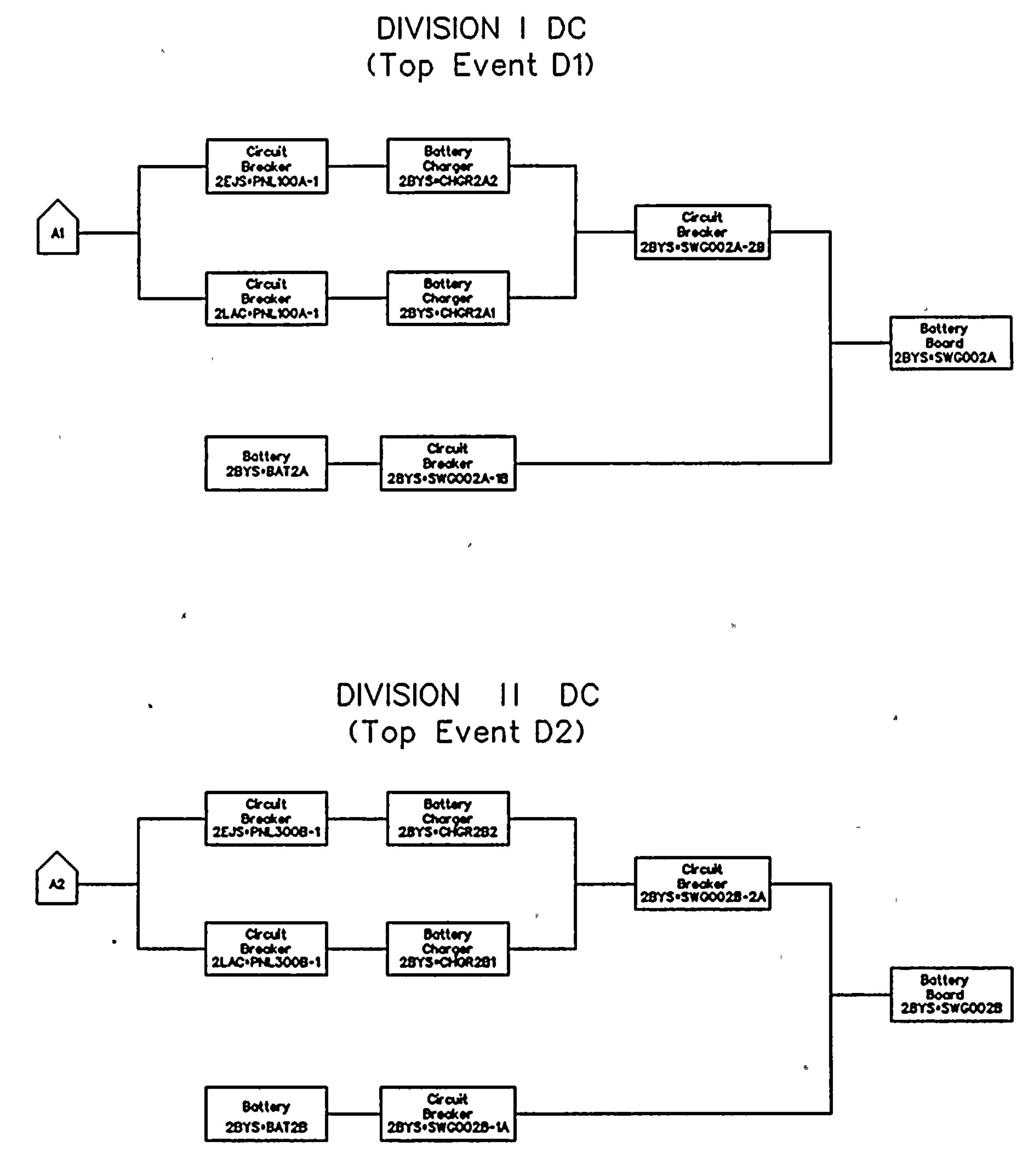


Figure 3.2.1.7-2
Emergency DC (D1 & D2)
Success Diagram

I: \NMP2IPE\DGN\2S07.DGN Jul. 06. 1992 15: 47: 17

DC POWER (Top Events D1, D2, DA, & DB)
Component Block Descriptions

Block	Mark No. (Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
D1 - CHARGER CIRCUIT	2EJS*PNL100A-1 2BYS*CHGR2A1 2LAC*PNL100A-1 2BYS*CHGR2A2 2BYS*SWG002A-2B	CKT BREAKER FROM EJS*PNL100A BATTERY CHARGER 2A1 CKT BREAKER FROM LAC*PNL100A BATTERY CHARGER 2A2 CIRCUIT BREAKER TO SWG002A	TRANSFERS OPEN LOSS OF FUNCTION TRANSFERS OPEN LOSS OF FUNCTION TRANSFERS OPEN	CLOSED N/A CLOSED N/A CLOSED	CLOSED N/A CLOSED N/A CLOSED	2BYS*SWG001A N/A 2BYS*SWG001A N/A 2BYS*SWG001A	N/A N/A N/A N/A N/A
D1 - BATTERY BOARD 2A	2BYS*SWG002A	BATTERY BOARD	LOSS OF FUNCTION	N/A	N/A	N/A	N/A
D1 - BATTERY	2BYS*SWG002A-1B 2BYS*BAT2A	CIRCUIT BREAKER TO BAT2A BATTERY	TRANSFERS OPEN LOSS OF FUNCTION	CLOSED N/A	CLOSED N/A	2BYS*SWG001A N/A	N/A N/A
D2 - CHARGER CIRCUIT	2EJS*PNL300B-1 2BYS*CHGR2B1 2LAC*PNL300B-1 2BYS*CHGR2B2 2BYS*SWG002B-2B	CKT BREAKER FROM EJS*PNL300B BATTERY CHARGER 2B1 CKT BREAKER FROM LAC*PNL300B BATTERY CHARGER 2B2 CIRCUIT BREAKER TO SWG002B	TRANSFERS OPEN LOSS OF FUNCTION TRANSFERS OPEN LOSS OF FUNCTION TRANSFERS OPEN	CLOSED N/A CLOSED N/A CLOSED	CLOSED N/A CLOSED N/A CLOSED	2BYS*SWG001B N/A 2BYS*SWG001B N/A 2BYS*SWG001B	N/A N/A N/A N/A N/A
D2 - BATTERY BOARD 2B	2BYS*SWG002B	BATTERY BOARD	LOSS OF FUNCTION	N/A	N/A	N/A	N/A
D2 - BATTERY	2BYS*SWG002B-1B 2BYS*BAT2B	CIRCUIT BREAKER TO BAT2B BATTERY	TRANSFERS OPEN LOSS OF FUNCTION	CLOSED N/A	CLOSED N/A	2BYS*SWG001B N/A	N/A N/A

System 8

Service Water

3.2.1.8 Service Water System

3.2.1.8.1 System Function

The service water system provides cooling to safety and non-safety related systems. A simplified drawing of the service water system is provided in Figure 3.2.1.8-1. However, the individual cooling loads are not shown.

In addition, the service water system can be used to flood the primary containment and RPV through a cross-connection with RHR loop "B".

3.2.1.8.2 Success Criteria

The service water system is modeled as two trains (loops) in the support event tree (top events SA and SB) where SA represents successful flow from the pump trains in the screenwell to the "A" header (loop) and a return path to the discharge bay. The success for SB requires flow from the pump trains in the screenwell to the "B" header and a return path to the discharge bay. Individual service water cooling loads (i.e. flow from header A to RHR heat exchanger 1A and back to Header A discharge path) are modeled in the respective system analysis.

The number of pumps required for success depends on whether the crosstie between headers (MOV50A and 50B) is open as well as whether the pumps are supplying the reactor and turbine building loads. On a total loss of normal AC power to the pump emergency switchgear, the crosstie between trains and the reactor and turbine building loops receive isolation signals and only one pump will be started on each train when the emergency diesel is supplying the switchgear. Isolation success requires one-of-two crosstie, one-of-three Turbine Building, and one-of-four Reactor Building motor operated isolation valves to close. Loss of the offsite grid (OG in support event tree) or both 115KV sources (KA and KB in the support event tree) would isolate all MOVs. Loss of a single 115KV source would isolate the Reactor Building and Turbine Building isolation valves. The crosstie MOVs between pump trains do not receive an isolation signal unless both normal AC supplies are unavailable. The following summarizes pump success criteria for applicable conditions:

CrossTie	Reactor/Turbine Building	Success Criteria Train A (SA)	Success Criteria Train B(SB)
Open	Open	4 of 6 Pumps	
Open	Closed	2 of 6 Pumps	
Closed	Open	3 of 3 A Pumps,	1 of 3 B Pumps
Closed	Closed	1 of 3 A Pumps,	1 of 3 B Pumps

Note that cases with the crosstie open treat the system as a whole without train separation, this is because different combinations of four pumps can be used to maintain cooling flow.

Since the reactor and turbine building loops are normally supplied by Header A and isolated from Header B, any time the crosstie is closed, Header B requires only 1 of 3 pumps. Failure of turbine or reactor building loops to isolate alone does not affect Header B.

3.2.1.8.3 Support Systems

The Service Water System requires Division I and II Emergency AC for pump operation, and Division I and II DC for pump control. The service water isolation and crosstie MOVs require Division I and II Emergency AC for operation. For a more complete list of major components and dependencies, see Table 3.2.1.8-1.

Figure 3.2.1.8-2 provides a simplified success diagram for actuating 2SWP*MOV50A and MOV93A closure. The Division II MOVs (50B and 93B) are similar, except their input signals depend on Division I AC and DC. Other isolating MOVs are similar to MOV93A and 93B. 120V AC Division II and 125V DC Division II losses are shown in parallel with their portion of the input signals that are de-energized to actuate.

Lake Ontario supplies water to the service water system via two intake structures that supply the screenwell where the water is filtered by trash racks and traveling screens on its way to the service water pump bay. Service water discharges to the discharge bay in the screenwell and then flows to the offshore discharge nozzles.

Service water supports the following systems/equipment with the two safety related loops (Division I and Division II):

- RHS heat exchangers and pump seal coolers (backup),
- Fuel pool cooling and emergency fill,
- Reactor Building emergency recirculation unit coolers (2HVR*UC413A/B),
- Control and relay room chillers & unit coolers,
- Containment spray (emergency drywell flooding),
- Emergency diesel generator and unit cooler,
- HPCS diesel generator and unit cooler,
- HPCS switchgear room unit coolers (2HVC*UC102 & 103A),
- DBA recombiner,
- Control room chilled water (backup), and
- All reactor building unit coolers.

The Turbine Building loop (header) supplies the TBCLC heat exchangers and other non-safety equipment.

The Reactor Building loop (header) supplies the RBCLC heat exchangers and the Reactor building ventilation supply air cooler.

Makeup to the cooling tower is provided from each of two service water discharge headers.

3.2.1.8.4 System Operation

During normal plant operation two of three pumps in each division (four-of-six pumps total) supply the four main loops:

1. Safety Related Loop A (Division I)
2. Safety Related Loop B (Division II)
3. Turbine Building Loop
4. Reactor Building Loop

The service water header crosstie isolation valves (MOV50A & B) are normally open.

On loss of offsite power, all running pumps are stopped and one pump per Division is restarted automatically in timed sequence if the diesel generator has successfully started. If a running pump fails to restart, a standby pump is started automatically (control switch in NORMAL AFTER START). Interlocks prevent a pump from starting unless the associated discharge valve is fully closed (MOV74A through F). In addition, the following automatically occurs after a loss of offsite power.

- Service water supplies cooling water to emergency diesel generators (MOV66A & B and MOV94A & B open). Service water header A supplies the Division I diesel (MOV66A) and header B supplies the Division II diesel (MOV66B). Both headers supply the Division III diesel (MOV94A & B).

- The crosstie between the A and B pump headers isolates (MOV50A & B close).

- Service water to turbine building loop, reactor building loop and circulating water isolates (MOV3A & B, MOV19A & B, MOV93A & B, MOV599, FV47A & B and FV54A & B). MOV3A & B and MOV599 isolate the Turbine Building loop from header A. MOV19A & B and 93A & B isolate the Reactor Building loop from Header A. FV47A and 54A isolate Header A from the circ. water system. FV47B and 54B isolate Header B from the circ. water system.

- Manual start of another service water pump is locked out for 60 seconds.

Differential pressure across the traveling screens initiates an automatic rinse cycle. A low level in the service water pump bay automatically opens the traveling screen bypass valves (MOV77A & B). An excessive high level in the discharge bay automatically closes MOV30A & B to line up the system to discharge through the north intake shaft.

In the service water model, the operators are expected to be able to start a standby pump and ensure that low pressure trip and runout conditions are corrected.

These actions improve success criteria for the number of pumps required. A list of the specific operator actions included follows:

Basic Event	Operator Action and Conditions
HHSA1	Operators start the third standby pump in either Train A or Train B when required.

Basic Event	Operator Action and Conditions
HHSA2	Normal AC power is available and three or more of six pumps are not running or are unavailable. Operators maintain adequate cooling to safety loads by throttling pumps if a runout condition exists and isolating Reactor/Turbine Building loads or loops, as necessary, to supply adequate cooling to safety loads. While isolating non-safety loops, the operators must ensure that low flow conditions are not created that trip pumps. The pumps can be restarted and opening the supply to RHR heat exchangers ensures adequate flow conditions. Note that failure of all three pumps on a single division results in an auto-closure of the Reactor/Turbine Building isolation MOVs. This has been conservatively neglected.
HHSA3	Normal AC power is available and the crosstie inadvertently isolates. On the Division I side, the pumps will initially be in a runout condition requiring the operators to throttle and/or isolate the Reactor/ Turbine Building loops while ensuring that low flow conditions are not created. On the Division II side, a low flow condition may have occurred requiring the operators to open additional flow paths (i.e., RHR heat exchanger) and restarting a pump if they trip. If all three Division II pumps trip, the Reactor/Turbine Building Isolation MOVs receive an isolation signal resulting in a potential low flow trip of all Division I pumps. The model assumes that all six pumps are tripped and the operators must restart pumps.
HHSA4	Partial loss of normal AC power (115KV Source A or B unavailable) and the Turbine/Reactor Building loops successfully isolate. The operators must open an RHR heat exchanger path or the RBCLC path to prevent low flow pump trips and restart a pump if this is not done fast enough.

3.2.1.8.5 Instrumentation and Control

Control switches and indications for individual service water pumps and major valves are located on Panel 601 in the control room.

The following additional indications are provided at Panel 601:

- Pump suction pressure,
- Pump discharge pressure,
- Service water discharge pressure,
- Service water pump bay water level,
- Service water discharge bay water level,
- Service water pump flow,
- Discharge flows to the circulating water system, and
- Tempering water flow.

Annunciators (alarms) are provided in the control room for major equipment trouble, pressure, level and flow.

A service water pump will start automatically only on a valid load sequencing signal if its discharge valve is closed. The discharge valve will automatically open when the pump starts. The service water pump will trip and its discharge valve will shut if any of the following occur:

- Motor/feeder electrical fault,
- Low discharge flow of 1,000 gpm, or
- Respective emergency switchgear sustained under voltage.

3.2.1.8.6 Technical Specifications

Two independent service water system loops with two pumps operable in each loop are required. The following summarizes LCO lengths for different inoperable conditions:

Number of Pumps Available		
Loop A	Loop B	LCO Length
2	1	14 days
1	2	14 days
1	1	7 days (2/loop)
0	2	72 hours (1 pump)
2	0	72 hours (1 pump)
0	1	12 hours (1/loop)
1	0	12 hours (1/loop)

The intake deicing heater system shall be OPERABLE when intake tunnel water temperature is less than 39°F. Each intake structure shall have seven Division I and seven Division II heaters in operation. Otherwise, initiate action to shutdown within 1 hour.

3.2.1.8.7 Surveillance, Testing & Maintenance

Service water supply temperature is checked at least daily for high temperature. The intake temperature is checked at least every 12 hours for low temperature.

Water level at the pump intake is checked at least every 12 hours.

Current and voltage to the intake deicing heaters is checked at least every 7 days.

Valve position verification is performed at least every 31 days. Valves that are locked, sealed or otherwise secured in position are excluded.

At least every 18 months, during shutdown:

- After a simulated test signal, verify each valve servicing non-safety related equipment actuates to its isolation position.
- After a simulated test signal, verify each cross-connect and pump discharge valve actuates to its isolation position.
- After a simulated test signal, verify each pump starts and its associated discharge valve opens.
- Each pump runs and maintains discharge pressure $\geq$ 80 psig with flow $\geq$ 6500 gpm.
- Verify resistance is $\geq$ 28 ohms for each feeder cable and associated heater element in the intake deicing system.
- Perform a LOGIC SYSTEM FUNCTIONAL TEST of the pump starting logic.

3.2.1.8.8 References

N2-OP-11, Revision 4: "Service Water System"
USAR Sections 9.2.1 and 7.3
Technical Specification 3/4.7.1
PIDs that are referenced on the simplified drawing.

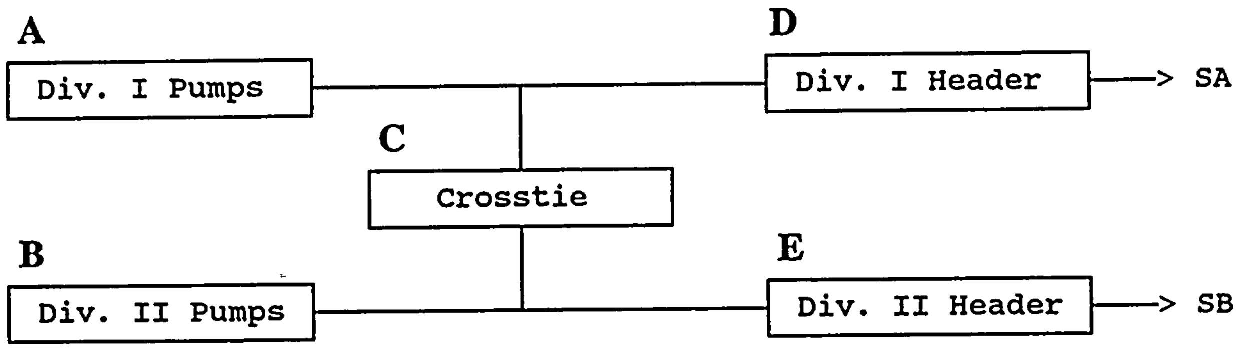
3.2.1.8.9 Initiating Event Potential

Loss of service water will cause a plant trip and is included as an initiating event due to common cause impact on other safety systems. Loss of service water, as an initiating event, is an unlikely event and very difficult to quantify. There is a spectrum of equipment failures leading to partial system failure that requires operator response and common cause failures have significant uncertainties. Many of these intermediate states requiring operator response are neglected since there is time to respond and actions are straight forward and can be performed in the control room. The following two loss of service water initiating events are included in the model and their development is described below:

- SAX - Loss of service water to the Division I header.
- SWX - Total loss of service water.

Normally, balance of plant cooling (reactor building and turbine building component cooling water systems) is supplied by the Division I header. The Division II header supplies ventilation cooling during normal operation and can be manually aligned outside the control room to balance of plant cooling loads. Loss of Division I has the most significant impact relative to initiating a plant trip and requiring operator recovery of balance of plant cooling from outside the control room. Therefore, loss of the Division I header (SAX) is modeled and loss of Division II is neglected as a less likely cause of core damage. Given SAX as an initiating event, the Division II header (top event SB) must be successful until the Division I header is repaired (24 hours is assumed).

The SAX and SWX models are simplified models developed from the service water systems analysis model described in this section. The following provides a simplified sketch of the service water system to explain the development of these models:



Based on the above, the following failure logic can be written for loss of flow at the Division I header, Division II header, and total loss of service water, where "+" stands for "OR" logic and "*" stand for "AND" logic:

- Division I = D + A*C + A*B
- Division II = E + B*C + A*B
- Total = A*B + D*E + A*C*E + B*C*D

Loss of the Division I header (SAX) and total loss of service water (SWX) initiating events are based on the following logic:

- SAX = D + A*C
- SWX = A*B + D*E + A*C*E + B*C*D

The above models are based on the following assumptions:

- Loss of Division I is more significant than loss of Division II as discussed above.

- Loss of all pumps (A*B) is not modeled in SAX because it would cause a loss of all service water, and is included in SWX.

- The Division II header failure model (top event SB) given SAX, is based on the following logic: [SB = B + E]. This is conservative because no credit is given to Division I pumps (A) and the crosstie (C) when the cause of SAX failure is the discharge header (D). In addition, no credit is given in the SB model for operators manually recovering Division II supplies to balance of plant cooling outside the control room. This means that RBCLC, TBCLC, feedwater, and the condenser are assumed unavailable given SAX and SB success.

- Failure of the crosstie (C), the Division I pumps (A) or the Division II pumps (B) alone are not modeled in SAX. These failure modes are similar to a transient. These initiator frequencies are assumed in the general transient category.

3.2.1.8.10 Equipment Location

The service water pumps trains, strainers, discharge valves, and the crosstie MOVs are located in the screen well. The service water system supplies equipment throughout the plant.

3.2.1.8.11 Operating Experience Review

There were no particular outstanding operational events relevant to this study. Plant specific component operational data is detailed in section 3.3.2.

3.2.1.8.12 Modeling Assumptions

1. Plugging of the traveling water screens is a potential common cause failure of the service water system. This failure mode is judged unlikely and is not modeled for the following reasons. An auto rinse cycle occurs: every 24 hours for 20 minutes, on a differential pressure signal, or an intake bay low water level alarm. Furthermore, a system trouble alarm alerts operators of potential blockage. On low intake level, MOV77A and 77B open, bypassing the screens. Sufficient blockage that fails service water is expected to occur over a period of time, allowing corrective actions before total failure.

2. Blockage in the discharge and intake tunnels is not modeled. There are two intakes from the lake and heating at the lake intakes to prevent freezing. A program is in place to ensure that mussels do not become a major problem for the system. Since blockage events are expected to occur over a relatively long period of time it is judged unlikely that both intakes would become completely blocked preventing safe shutdown cooling. The north intake supply will isolate (MOV30A and 30B) on high level in the pump bay. This provides an alternate discharge path through the north intake if the normal discharge path becomes blocked.

3. High point vents that are designed to open on loss of offsite power are not modeled. These vents are in the Reactor Building portion of the system that is isolated on loss of offsite power. In addition, the failure would likely not fail the system.

4. Common cause plugging of pump discharge strainers is not modeled. These strainers have a rinse cycle based on differential pressure and there are system alarms to alert operators of strainer problems. It is unlikely that a significant number of strainers would plug failing service water before corrective actions could be taken.

5. The model contains operator actions within the Control Room to isolate Reactor/Turbine Building loads when required to successfully provide adequate

cooling to safety equipment. Additional operator actions within the Control Room include starting the fifth and sixth pumps (standby pumps) when required, throttling pump discharge during potential runout cases, and restarting a tripped pump after low flow trip. Also, one pump supplying both headers with only one of two RHR heat exchanger loads is a possible success. This is not modeled. No credit is taken for manually closing the crosstie MOVs, thus allowing one of six pumps to be a success on one header. In addition, the fifth and sixth pump are always assumed to require operator action from the control room even though under some conditions they may be automatically started.

6. The model includes conditions where any two pumps are assumed successful if the crosstie is open and the Reactor Turbine Building loads are isolated.

7. The model assumes that pumps A through D are operating - pumps E and F are in standby.

8. Pump check valves are only modeled when the associated pump is stalled. The failure rate of pressurized check valves is assumed to be negligible.

9. Modeling starts at the Service Water Bay, all components prior to this point are 'black boxed'.

10. Valve 2SWP-V8, *MOV599, and *V202B are not modeled as they are in a normally isolated train.

11. The makeup line to the CWS system is not modeled as it is not required for Service Water operation.

12. The dominant failure mode for a check valve in a standby train is assumed to be 'fail to open', therefore, transfers closed is not modeled.

13. Check valve 2SWP*V202B is not modeled as it has no role in the isolation function of the turbine building loop.

14. Service Water system transients (e.q. spurious MOV closure or pump failure) are assumed to require restart of all pumps following low flow trip. In the model, the RBCLC and TBCLC supply MOVs receive a close signal to isolate the non-safety loads. The closure of the valves greatly reduces the flow of water through the Service Water pumps. Because the pumps have the same trip point, it is assumed all of the pumps will trip on low flow. Successful response requires, in most cases, that two pumps restart, thus creating a proper system flowrate.

15. Recovery of spurious MOV closure is not modeled. This is conservative as operator manual recovery is relatively simple and response is not needed for several hours.

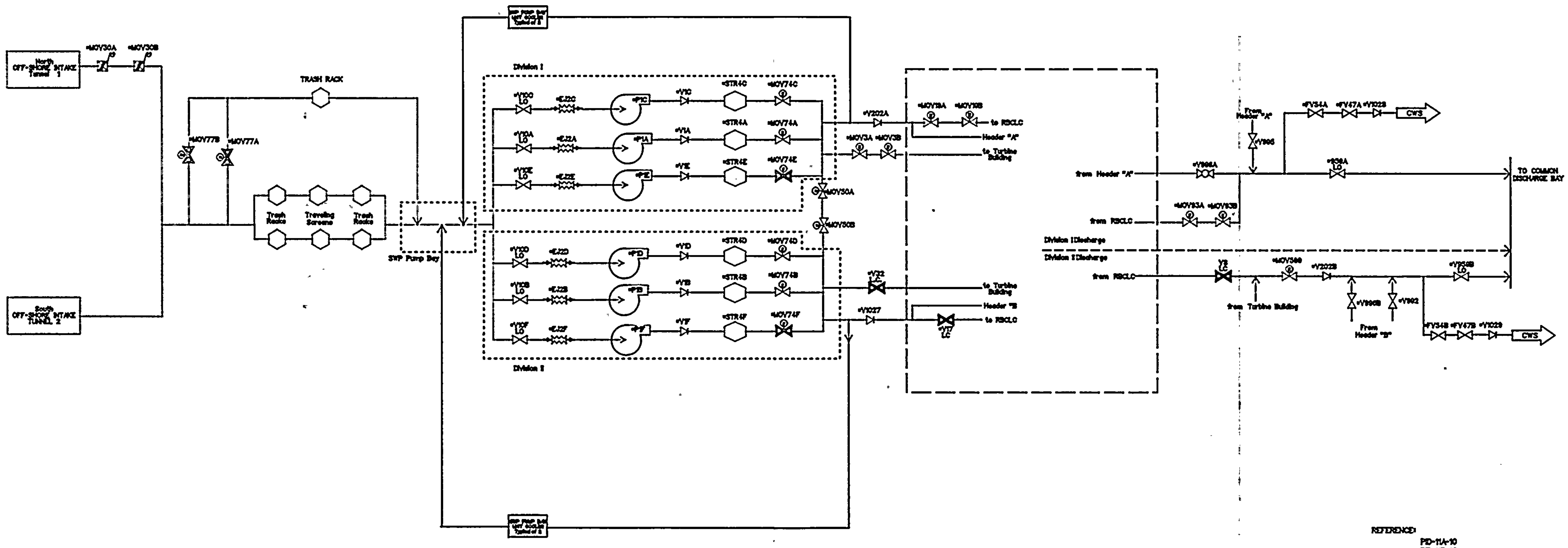
3.2.1.13 Logic Model and Quantification

The fault tree(s) is included in Tier 2 documentation, and is available upon request. Quantitative results are summarized in Section 3.3.4 (Tier 1).

Rev 0 (7/92)

SI APERTURE CARD

Also Available On Aperture Card



NOT Modeled
Division I
Division II
Crosstie
to/from Reactor/Turbine Building Loads
Intake Structure (black box)

REFERENCE:
PID-11A-10
PID-11B-10
PID-11D-8
PID-11H-14
PID-11P-9

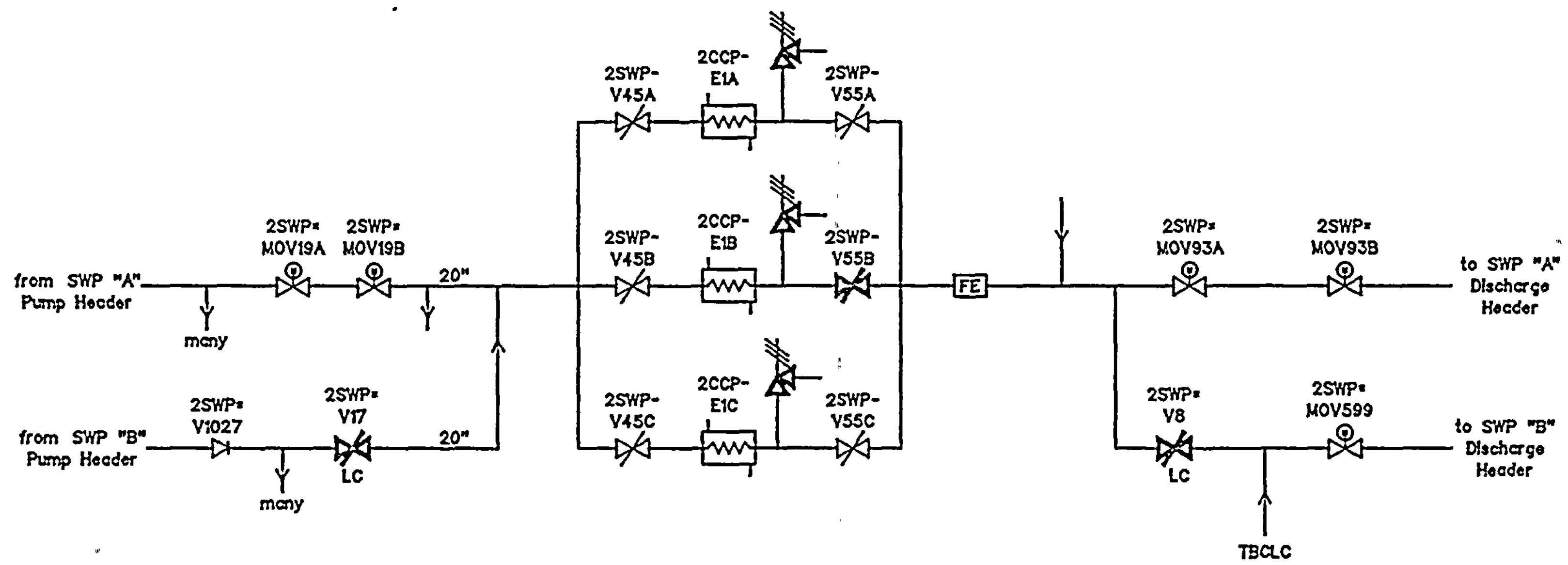
Figure 3.2.1.8-1
Service Water System
(simplified schematic)

I:\NMP2IPE\DGN\2S08.DGN Jul. 07, 1992 11:56:51

9208050191-19

Rev 0 (7/92)

Service Water - Reactor Building Closed Loop Cooling Interface



Service Water - Turbine Building Closed Loop Cooling Interface

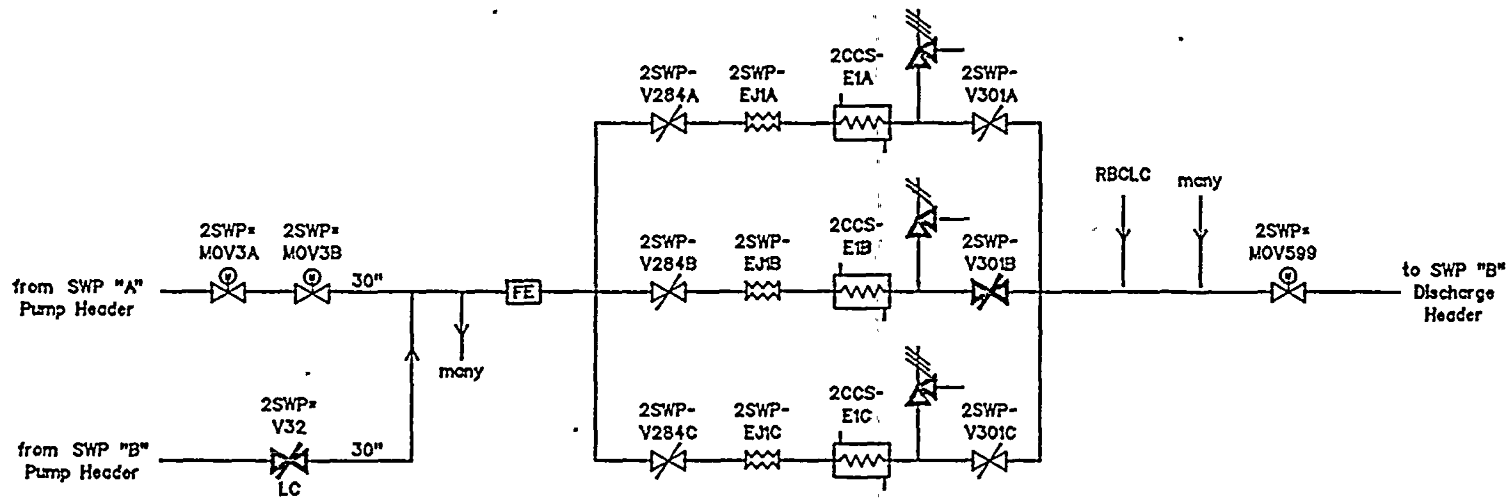


Figure 3.2.1.8-3
Service Water connections
to RBCLC & TBCLC

Rev 0 (7/92)

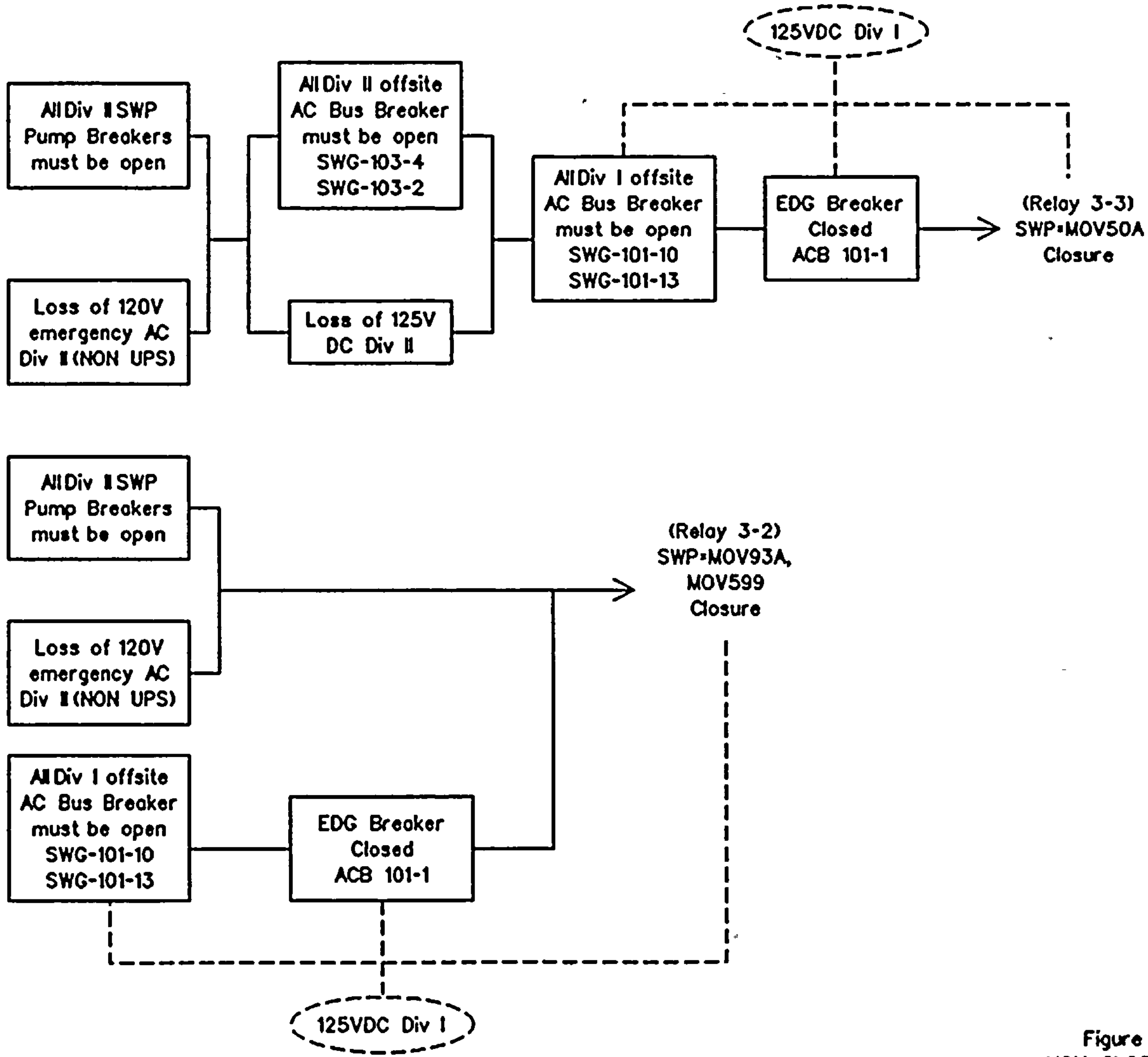


Figure 3.2.1.8-2
MOV CLOSURE SIGNALS
SIMPLIFIED SUCCESS DIAGRAM

Table 3.2.1.8-1 REV. 0 (7/92)

SERVICE WATER SYSTEM
Component Block Descriptions

Block	Mark No. (Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
TRAIN A -- PUMP P1A							
	2SWP*V10A	L.O. Pump Suction Valve	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2SWP*EJ2A	Pump Suction Expansion Joint	RUPTURE	N/A	N/A	N/A	N/A
	2SWP*P1A	"A" Service Water Pump	FAIL TO RUN	RUNNING	RUNNING	2ENS*SWG101	STOP
	2SWP*V1A	P1A Discharge Check Valve	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2SWP*STR4A	P1A Discharge Strainer	PLUGGED	N/A	N/A	N/A	N/A
	2SWP*MOV74A	P1F Discharge Valve	TRANSFER CLOSED	OPEN	OPEN	2EHS*MCC101	AS-IS
TRAIN A -- PUMP P1C							
	2SWP*V10C	L.O. Pump Suction Valve	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2SWP*EJ2C	Pump Suction Expansion Joint	RUPTURE	N/A	N/A	N/A	N/A
	2SWP*P1C	"C" Service Water Pump	FAIL TO RUN	RUNNING	RUNNING	2ENS*SWG101	STOP
	2SWP*V1C	P1C Discharge Check Valve	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2SWP*STR4C	P1C Discharge Strainer	PLUGGED	N/A	N/A	N/A	N/A
	2SWP*MOV74C	P1C Discharge Valve	TRANSFER CLOSED	OPEN	OPEN	2EHS*MCC101	AS-IS
TRAIN A -- PUMP P1E (modeled as standby)							
	2SWP*V10E	L.O. Pump Suction Valve	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2SWP*EJ2E	Pump Suction Expansion Joint	RUPTURE	N/A	N/A	N/A	N/A
	2SWP*P1E	"E" Service Water Pump	FAIL TO START	STOPPED	RUNNING	2ENS*SWG101	STOP
	2SWP*P1E	"E" Service Water Pump	FAIL TO RUN	RUNNING	RUNNING	2ENS*SWG101	STOP
	2SWP*V1E	P1E Discharge Check Valve	FAIL TO OPEN	CLOSED	N/A	N/A	N/A
	2SWP*STR4E	P1E Discharge Strainer	PLUGGED	N/A	N/A	N/A	N/A
	2SWP*MOV74E	P1E Discharge Valve	FAIL TO OPEN	CLOSED	OPEN	2EHS*MCC101	AS-IS
TRAIN A/B CROSSTIE	2SWP*MOV50A	Crosstie Valve	TRANSFER CLOSED	OPEN	OPEN	2EHS*MCC101	CLOSED
	2SWP*MOV50B	Crosstie Valve	TRANSFER CLOSED	OPEN	OPEN	2EHS*MCC301	CLOSED
TRAIN B -- PUMP P1B							
	2SWP*V10B	L.O. Pump Suction Valve	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2SWP*EJ2B	Pump Suction Expansion Joint	RUPTURE	N/A	N/A	N/A	N/A
	2SWP*P1B	"B" Service Water Pump	FAIL TO RUN	RUNNING	RUNNING	2ENS*SWG103	STOP
	2SWP*V1B	P1B Discharge Check Valve	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2SWP*STR4B	P1B Discharge Strainer	PLUGGED	N/A	N/A	N/A	N/A
	2SWP*MOV74B	P1B Discharge Valve	TRANSFER CLOSED	OPEN	OPEN	2EHS*MCC301	AS-IS
TRAIN B -- PUMP P1D							
	2SWP*V10D	L.O. Pump Suction Valve	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2SWP*EJ2D	Pump Suction Expansion Joint	RUPTURE	N/A	N/A	N/A	N/A
	2SWP*P1D	"D" Service Water Pump	FAIL TO RUN	RUNNING	RUNNING	2ENS*SWG103	STOP
	2SWP*V1D	P1D Discharge Check Valve	TRANSFER CLOSED	OPEN	N/A	N/A	N/A

SERVICE WATER SYSTEM
Component Block Descriptions

Block	Mark No. (Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
	2SWP*STR4D	P1D Discharge Strainer	PLUGGED	N/A	N/A	N/A	N/A
	2SWP*MOV74D	P1D Discharge Valve	TRANSFER CLOSED	OPEN	OPEN	2EHS*MCC301	AS-IS
TRAIN B -- PUMP P1F (modeled as standby)							
	2SWP*V10F	L.O. Pump Suction Valve	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2SWP*EJ2F	Pump Suction Expansion Joint	RUPTURE	N/A	N/A	N/A	N/A
	2SWP*P1F	"F" Service Water Pump	FAIL TO START	STOPPED	RUNNING	2EHS*SWG103	STOP
	2SWP*P1F	"F" Service Water Pump	FAIL TO RUN	RUNNING	RUNNING	2EHS*SWG103	STOP
	2SWP*V1F	P1F Discharge Check Valve	FAIL TO OPEN	CLOSED	N/A	N/A	N/A
	2SWP*STR4F	P1F Discharge Strainer	PLUGGED	N/A	N/A	N/A	N/A
	2SWP*MOV74F	P1F Discharge Valve	FAIL TO OPEN	CLOSED	OPEN	2EHS*MCC301	AS-IS
TO SERVICE WATER LOADS							
	2SWP*V202A	Check Valve to Train A Loads	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2SWP*V1027	Check Valve to Train B Loads	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
TO RB/TB LOADS							
	2SWP*MOV19A	MOV to Reactor Building Loads	FAIL TO CLOSE	OPEN	CLOSE	2EHS*MCC102A	AS-IS
	2SWP*MOV19B	MOV to Reactor Building Loads	FAIL TO CLOSE	OPEN	CLOSE	2EHS*MCC302B	AS-IS
	2SWP*MOV3A	MOV to Turbine Building Loads	FAIL TO CLOSE	OPEN	CLOSE	2EHS*MCC101	AS-IS
	2SWP*MOV3B	MOV to Turbine Building Loads	FAIL TO CLOSE	OPEN	CLOSE	2EHS*MCC301	AS-IS
FROM RB/TB LOADS							
	2SWP*MOV93A	Discharge MOV, from RB Loads	FAIL TO CLOSE	OPEN	CLOSE	2EHS*MCC103	AS-IS
	2SWP*MOV93B	Discharge MOV, from RB Loads	FAIL TO CLOSE	OPEN	CLOSE	2EHS*MCC303	AS-IS
	2SWP*MOV599	Discharge MOV, from TB Loads	FAIL TO CLOSE	OPEN	CLOSE	2EHS*MCC103	AS-IS
TRAIN A DISCHARGE							
	2SWP*V996A	Valve from Header A Equipment	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2SWP*V995	Valve from Header A Equipment	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2SWP*V959A	L.O. Valve to Discharge Bay	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
TRAIN B DISCHARGE							
	2SWP*V996B	Valve from Header B Equipment	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2SWP*V992	Valve from Header B Equipment	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2SWP*V959B	L.O. Valve to Discharge Bay	TRANSFER CLOSED	OPEN	N/A	N/A	N/A

System 9

Fire & Service Water Crossties to RHR

3.2.1.9 Fire Water & Service Water Crossties to RHR

3.2.1.9.1 System Function

The capability exists to align fire water and/or service water to the Residual Heat Removal (RHS or RHR) system. Emergency Operating Procedure (EOP) section RL instructs the operators to use this capability for injection, as described in N2-EOP-6, Attachment 5 (Service Water) and Attachment 6 (Fire Water), if RPV water level can not be maintained above 159.3 inches. Other EOP procedures also instruct the operators to use this capability for RPV injection. Figure 3.2.1.9-1 is a simplified diagram of the fire water crosstie to RHR. A success diagram of the crossties to the RHR system is shown in Figure 3.2.1.4-2.

3.2.1.9.2 Success Criteria

Top event SW in the front-line event trees models the service water header B crosstie valves and operator action (when required) to provide injection to the RPV through the RHR injection path. Top event FP in the front-line event trees models the firewater pumps and crosstie valves injecting to the RPV through either RHR A or B injection paths.

Top events S1, S2 and S3 in the station blackout event tree model the diesel fire pump injecting to the RPV through crosstie valves to either RHR A or B injection paths. Operator actions that are required to align this flow path are included.

A success diagram is provided in Figure 3.2.1.9-2. As shown, either RHR injection path A or B (top event IA and IB in the front-line event trees) must be available. Fire water can supply either injection path, whereas only RHR injection path B can be supplied by service water.

3.2.1.9.3 Support Systems

Major components and their support system requirements are summarized in Tables 3.2.1.9-1 and 3.2.1.9-2.

The diesel fire pump has two 24V batteries that can be used for control circuit power. The pump can be controlled from either panel 849 in the control room or from the diesel fire pump room in a station blackout situation.

3.2.1.9.4 System Operation

<u>Service Water Cross-Tie</u>
The Service Water connection is a normally isolated, permanent hardware connection, which ties to the B train of RHS. The isolation consists of two valves, key-locked in the control room (2RHS*MOV115 and 2RHS*MOV116). The connection is from the B train of service water. The Service Water system must provide sufficient flow to the header and be properly aligned via check valve 2SWP*V1027 and manual valve 2SWP*V255B (normally open). In addition to the normal RHS injection path equipment, the two normally closed isolation

valves, 2RHS*MOV115 and 2RHS*MOV116, and check valve 2RHS*AOV150 must open. The service water system is considered an inexhaustible supply of water, and is used as directed in N2-EOP-6, Attachment 5. The connection is downstream of the RHR heat exchanger outlet isolation valve 2RHS*MOV12B.

Fire Water Cross-Tie
Fire water can be aligned to either RHR train A or B, as described in N2-EOP-6, Attachment 6. This requires the removal of a blank flange and the connection of a dedicated hose from a hose reel to the condensate test connection. There are two hose stations accessible for train B, and one accessible for train A. The hoses are stored in locked gang boxes in the north and south aux bay stairwells, 261 foot elevation. The connection locations to RHS are shown on the simplified diagrams for the appropriate RHS train (Figures 3.2.1.4-1 and 3.2.1.4-2).

The fire water system is kept full and has pressure maintained at 120 - 135 psig by two 30 gpm pumps (P3A and P3B). On a loss of normal AC, these pumps are unavailable. Makeup water comes from the service water bay. Large volumes of makeup from the service water bay are supplied by either a diesel operated pump or an electric motor operated pump, or both. The electric motor pump will auto-start at 90 psig, and the diesel pump will auto-start at 80 psig. Once started on auto, the pumps continue to run until manually secured. When the diesel fire pump is running in "AUTO" or "MANUAL", the oil pressure and cooling water trips are bypassed.

3.2.1.9.5 Instrumentation and Controls

The service water system can be aligned from the control room at panel 601. Proper alignment is covered in N2-EOP-6, Attachment 5.

The diesel fire pump can be started from the control room or from a local panel. There is pump running indication in the control room (in case of auto-start). The pump also has trouble indication, 'not in auto-start' indication, and low fuel oil level indication.

The electric motor fire pump can be started from the control room or a local panel. It has indication for the following situations: pump running, trouble, auto-start and failure to start.

3.2.1.9.6 Technical Specifications

FSAR Sections 9A.3.6.2.6 and 9A.3.6.3.4 list the following requirements for the diesel fire water pump and fire hose stations:

Fire water system shall be demonstrated operable:

- at least once per 31 days by verifying that each valve in the flow path is in its correct position;
- at least once per 6 months by performance of a system flush;

- at least once per 12 months by cycling each testable valve through at least one full cycle;

- at least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:
  a. verifying that each automatic valve actuates to its correct position;
  b. verifying each pump develops at least 2500 gpm at a net discharge head of 113 psig;
  c. cycling each valve not testable during plant operation through at least one full cycle;
  d. verifying each pump starts and maintains a system pressure of a least 125 psig; and

- at least once per 3 years by performing a flow test of the system.

The diesel fire pump shall be demonstrated operable:

- at least once per 31 days by:
  a. verifying fuel day tank contains at least 350 gallons of fuel;
  b. starting the pump from ambient conditions and operating for at least 30 minutes of recirc. flow; and

- at least once per 92 days that a sample of the fuel taken from the storage tank is within acceptable limits when checked for viscosity, water and sediment.

Fire hose stations shall be demonstrated operable:

- at least once per 31 days by a visual inspection of those stations accessible during normal plant operation to assure all required equipment is at the station;

- at least once per 18 months by:
  a. visual inspection of those stations not accessible during normal plant operation to assure required equipment is at the station;
  b. removing the hose for inspection and re-racking;
  c. inspecting all gaskets and replacing all degraded gaskets in the couplings;

- at least once per 3 years by:
  a. partially opening each hose station valve to verify valve operability and no flow blockage; and
  b. conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig above the maximum fire main operating pressure, whichever is greater.

3.2.1.9.7 Surveillance, Testing and Maintenance

The diesel fire pump batteries are tested weekly, quarterly and every refueling. These tests do not impact system availability.

There is a 600V MCC and motor maintenance every refueling.

There are functional tests of both the electric and diesel fire pumps every refueling. One pump is put in standby while the other pump is run for at least 30 minutes.

3.2.1.9.8 References

N2-OP-43 Rev. 3: Fire Protection Water
N2-OP-11 Rev. 4: Service Water System

N2-EOP-RPV, Section RL, Rev. 4: RPV Control, RPV Water Level
N2-EOP-6, Attachment 5, Rev. 0: RHR Service Water Crosstie
N2-EOP-6, Attachment 6, Rev. 0: RHR Fire Water System Crosstie

FSAR Appendix 9A

N2-EPM-FPW-Q679, Rev. 0: Quarterly Fire Pump Battery Test
N2-EPM-FPW-R680, Rev. 0: Refuel Cycle Fire Pump Battery Test
N2-EPM-FPW-W678, Rev. 1: Weekly Fire Pump Battery Test
N2-EPM-GEN-R580, Rev. 1: 600V MCC & Motor Preventative Maintenance
N2-FSP-FPW-R001, Rev. 1: Electric/Diesel Fire Pump Functional Test

PIDs are referenced on the appropriate simplified diagrams, Figures 3.2.1.9-1, 3.2.1.4-1 and 3.2.1.4-2

3.2.1.9.9 Initiating Event Potential

Initiating events due to service water and fire water system floods are considered in the internal flood analysis.

3.2.1.9.10 Equipment Location

The fire pumps are in the screenwell building.

All valves considered in the model can be found in the reactor building. The Fire Hose Reels are in the reactor building at elevation 261.

3.2.1.9.11 Operating Experience

There were no particular outstanding operational events relevant to this study. Plant specific component operational data is detailed in section 3.3.2.

3.2.1.9.12 Modeling Assumptions

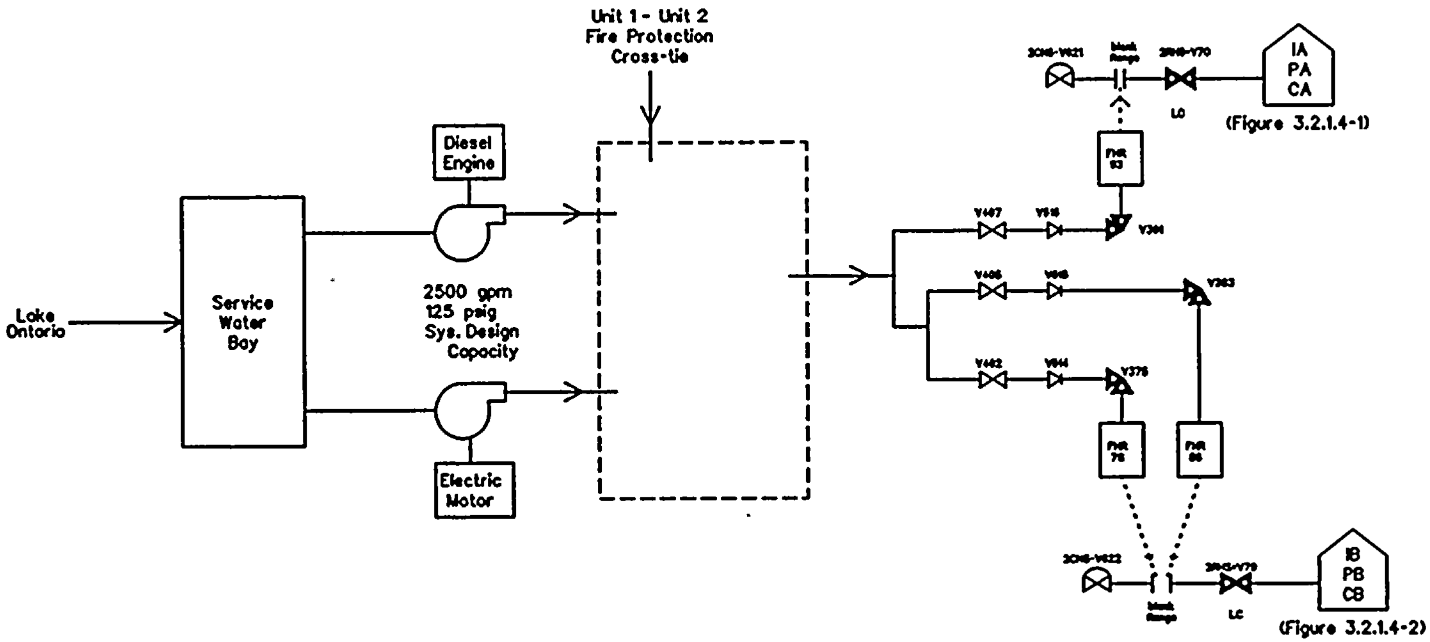
1. The fire water model is a simplified mode. Only major fire water system components are included in the fault trees (i.e., pumps and major valves that interface with the LPCI injection paths).

2. Aligning the diesel fire pump to inject into the RPV during a station blackout includes only the LPCI A injection path. This is conservative since credit is not taken for the B path. However, the failure modes of the injection path are small in comparison with the diesel pump failures and operator actions associated with attempting a second alignment (due to lack of available time).

3.2.1.9.13 Logic Model and Results

The fault tree(s) is included in Tier 2 documentation, and is available upon request. Quantitative results are summarized in Section 3.3.5 (Tier 1).

Rev 0 (7/92)



References: PID-43A-11
PID-43B-10
PID-43F-13
PID-4B-10

Figure 3.2.1.9-1
Fire Water - RHR Crosstie

I:\NMP2IPE\DGN\2S09.DGN Jul. 06. 1992 15:59:10

Rev 0 (7/92)

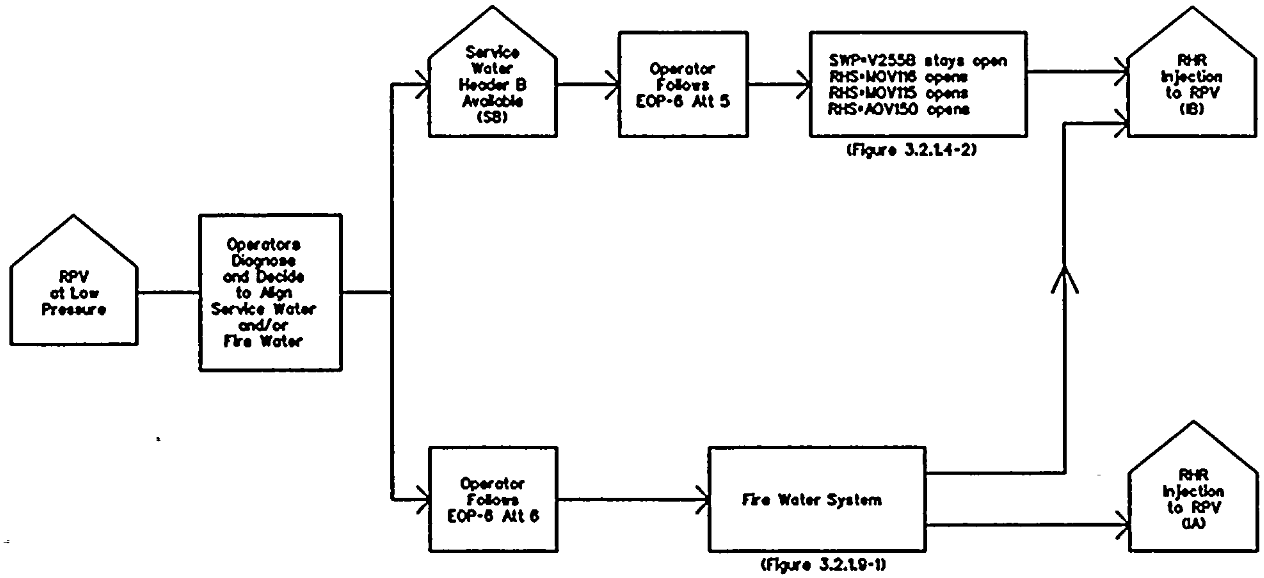


Figure 3.2.1.9-2
Fire Water & Service Water
Connections to RHR, Success Diagram

I:\NMP2IPE\DGN\2S09.DGN Jul. 06. 1992 16:03:40

SERVICE WATER - RHR CROSSTIE Component Block Descriptions								
Block	Mark No.	(Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
SERVICE WATER EQUIPMENT								
	2SWP*V1027		SERVICE WATER CHECK VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2SWP*V255B		SERVICE WATER MANUAL VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
RESIDUAL HEAT REMOVAL EQUIPMENT								
	2RHS*MOV115		SWP-RHS CROSSTIE SHUTOFF MOV	TRANSFER CLOSED	OPEN	OPEN	2EHS*MCC303D (II	AS-IS
	2RHS*MOV116		SWP-RHS CROSSTIE SHUTOFF MOV	TRANSFER CLOSED	OPEN	OPEN	2EHS*MCC303D (II	AS-IS
	2RHS*AOV150		SWP TESTABLE CHECK VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A

FIRE WATER - RHR CROSSTIE
Component Block Descriptions

Block	Mark No. (Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
RHR A CROSSTIE							
	2CNS-V621	VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	N/A	BLANK FLANGE	IN PLACE	N/A	N/A	N/A	N/A
	2RHS*V70	VALVE	FAILS TO OPEN	CLOSED	N/A	N/A	N/A
	2FPW*V407	VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2FPW*V516	CHECK VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2FPW*V391	VALVE	FAILS TO OPEN	CLOSED	N/A	N/A	N/A
	2FPW*FHR93	FIRE HOSE REEL	N/A	N/A	N/A	N/A	N/A
FPW TO RHR B PATH A							
	2FPW*V405	VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2FPW*V515	CHECK VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2FPW*V383	VALVE	FAILS TO OPEN	CLOSED	N/A	N/A	N/A
	2FPW*FHR86	FIRE HOSE REEL	N/A	N/A	N/A	N/A	N/A
FPW TO RHR B PATH B							
	2FPW*V402	VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2FPW*V514	CHECK VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2FPW*V375	VALVE	FAILS TO OPEN	CLOSED	N/A	N/A	N/A
	2FPW*FHR78	FIRE HOSE REEL	N/A	N/A	N/A	N/A	N/A
RHR B Injection							
	2CNS-V622	VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	N/A	BLANK FLANGE	IN PLACE	N/A	N/A	N/A	N/A
	2RHS*V79	VALVE	FAILS TO OPEN	CLOSED	N/A	N/A	N/A
PUMP							
	2FPW-P1	DIESEL FIRE PUMP	FAILS TO START	STOP	STOPPED	N/A	STOP
	"	"	FAILS TO RUN	"	"	"	"
	2FPW-P2	MOTOR DRIVEN FIRE PUMP	FAILS TO START	RUNNING	STOPPED	2NNS-SWG012	STOP
	"	"	FAILS TO RUN	"	"	"	"

System 10

Containment Isolation

3.2.1.10 Primary Containment Isolation

3.2.1.10.1 System Function

The primary containment isolation system isolates lines that penetrate the containment structure from the RPV, the containment atmosphere, and the suppression pool. This isolation protects the public in the case of a severe accident by preventing the release of radioactivity and by protecting safety equipment outside the containment that may further mitigate an accident.

3.2.1.10.2 Success Criteria

The containment isolation function is modeled in the containment event tree as top event IS. In general, at least one valve must close in each line penetrating the containment. All penetrations are screened to determine if it is suitable for inclusion in the containment isolation model as documented in Table 3.2.1.10-1 and discussed further below.

Exceptions to the isolation function are the Emergency Core Cooling Systems that operate during a severe accident and are considered a closed system outside containment or an extension of the containment boundary.

3.2.1.10.3 Support Systems

The support system requirements for each valve is documented in Table 3.2.1.10-1. Also included is whether the valve fails as is (FAI) or fails closed on loss of support systems. Support system dependencies for the modeled containment isolation penetrations are listed in Table 3.2.1.10-2.

The containment isolation function is implemented in two groups. The first group is the MSIVs/MSIV Bypass Valves, which depend on independent actuation signals, nitrogen, instrument air, and 2VBB-UPS3A or 2VBB-UPS3B for operation. The other group consists of all other containment isolation valves, normally referred to as the containment isolation system. This group is actuated by a separate logic scheme and involves additional support systems.

MSIVs are closed with a one (1) out of two (2) twice logic. Each logic channel can be actuated if any one of the following conditions are met:

- Low Reactor Water Level (Level 1)
- Low Condenser Vacuum
- High Steam Line Flow
- High Steam Line Radiation
- High Main Steam Line Area Temperature
- Low Steam Line Pressure

Containment isolation valves are closed in a one (1) out of two (2) twice logic on the following signals:

1. Low Reactor Water Level (Level 2)
2. High Drywell Pressure

Containment isolation actuation is normally energized by 120V AC power (2VBB-UPS3A and UPS3B) as presented in Figures 3.2.1.10-1 and 2. Since containment isolation is de-energized to actuate, loss of either UPS or its panel generates a signal to actuate isolation of one division. The actuation relays are normally energized by Division I or II 125V DC power, as shown in Figure 3.2.1.10-3. Loss of DC power closes the appropriate division isolation valve.

3.2.1.10.4 Screening of Penetrations

All penetrations were screened to identify the probable failure paths which are included in the model. Table 3.2.1.10-1 summarizes the results of the screening process. The table catalogs each penetration in the following columns:

- Pen. (Dia.) - This column identifies the penetration identification number as in the FSAR and provides the penetration diameter in parentheses.

- Description - This column provides a brief description of the system associated with the penetration.

- Valves Support - This column lists valves by mark number that could isolate the subject penetration. Prior to the valve, an "I" in parentheses indicates the valve is inside containment and an "O" in parentheses indicates the valve is outside containment. Following the valve mark number, its' support systems are identified.

- Normal - This column indicates the normal position of the valve during power operation.

- Signal - This column lists major isolation signals that the valve receives.

- Fails - This column identifies whether the valve fails as is (FAI) or fails closed when it loses its' support system. N/A is used in this column for check valves since they require no support system.

- Screening - This column indicates whether the line penetrating containment connects to reactor coolant boundary (RPV), drywell (DW), suppression chamber (SC), suppression pool (SP) or is a closed system inside containment (Closed). In addition, this column contains references to notes that further explain why penetrations are screened out of the model.

- Model - This column documents whether the penetration is included in the model (yes/no). This decision is based on the normal position of the valves, whether they fail closed, the number of valves, the size of the penetration, and the notes provided in the screening column.

- FSAR Figure - This column references the FSAR figure and sheet and the PID that shows the penetration and its isolation valves, if applicable.

Screening penetrations out of the model ("NO" in the "Model" column) is provided for two reasons. The first is to identify the penetrations that are more likely to fail open. The second reason is to exclude the penetrations that contribute insignificantly to the frequency of containment isolation failure. The most important penetrations selected can be open during operation, have only two valves in series, and the line size is 2 inches in diameter or greater. Other penetrations were judged to have additional levels of redundancy such as line size (less than 2 inch diameter), additional valves, normally closed valves, indication in control room or closed system.

As shown in Table 3.2.1.10-1, the following penetrations were judged to dominate containment isolation failure:

Penetration	Description
39	Drywell Floor Drain - Normally Open FAI MOVs
40	Drywell Equip. Drain - Normally Open FAI MOVs
43	Drywell Floor Drain Tank Vent - Normally Open FAI MOVs
45	Drywell Equipment Drain Tank Vent - Normally Open FAI MOVs
48	Purge Exhaust from DW - Fail Closed AOVs
49	Purge Inlet to DW - Fail Closed AOVs
50	Purge Inlet to SC - Fail Closed AOVs
51	Purge Exhaust from SC - Fail Closed AOVs
58	Cont. Purge to DW - Fail Closed SOVs
59	Cont. Purge to SC - Fail Closed SOVs

3.2.1.10.5 System Operation

The containment isolation system sensor channels, channel relays, division relays, and the relays that drive the actual isolation valves are all de-energized on a containment isolation signal. Thus on loss of 120V AC that initiats the signal, or 125V DC that operates the relay, a containment isolation signal will be generated. This is fail safe for most component failures. The simplified schematics in Figures 3.2.1.10-1 and 2 show Division I input signals and actuation for RPV Level 2 and high drywell pressure inputs. Many different relay failures and some multiple sensor channel component failures will result in isolation of one or all containment isolation paths. Loss of the electrical power sources for the relays will result in containment isolation.

To fail the containment isolation function, one or more relays must fail to have their contacts open, or several transmitters have to fail to sense high drywell pressure or low vessel level. Failure of the containment isolation function may be caused by the following situations:

- Both vessel level transmitters fail high, indicating a high vessel level.
- Both drywell pressure transmitters fail low indicating a low drywell pressure.
- One drywell pressure and one reactor vessel level transmitter fails.
- Relays fail to open because of mechanical failure or contact welding.

Containment isolation may be manually actuated in the Control Room. Individual valves have control switches and indication in the Control Room. N2-EOP-RPV, Section RL instructs the operators to initiate isolation of any path that failed to isolate. All of the valves in the Containment Isolation system model are included in N2-EOP-6, Attachment 1.

3.2.1.10.6 Technical Specifications, Surveillance, Testing

The primary containment isolation valves and reactor instrumentation line excess flow check valves shown in Technical Specification Table 3.6.3-1 shall be OPERABLE with isolation times less than or equal to those shown in Technical Specification Table 3.6.3-1.

Each primary containment isolation valve (see Tech. Spec. Table 3.6.3-1) shall be demonstrated OPERABLE before returning the valve to service after maintenance, repair or replacement work is performed on the valve or its associated actuator, control or power circuit. This is done by cycling the valve through at least one complete cycle of full travel and verifying the specified isolation time.

Each automatic isolation valve (see Tech. Spec. Table 3.6.3-1) shall be demonstrated OPERABLE during COLD SHUTDOWN or REFUELING at least once per 18 months by verifying that on a containment isolation test signal each automatic isolation valve actuates to its isolation position.

The isolation time of each power operated or automatic valve (see Technical Specification Table 3.6.3-1) shall be determined to be within its limit when tested pursuant to Specification 4.0.5.

Each reactor instrumentation line excess flow check valve (see Technical Specification Table 3.6.3-1) shall be demonstrated OPERABLE at least once per 18 months by verifying that the valve checks flow.

Each TIP system explosive isolation valve (VEX) shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying the continuity of the explosive charge.

b. At least once per 18 months by removing at least one explosive squib from at least one explosive valve and initiating the squib, such that each explosive squib in each explosive valve will be tested at least once per 36 months. The replacement charge for the exploded squib shall be from the same manufactured batch as the one fired or from another batch which has been certified by having at least one of that batch successfully fired. No squib shall remain in use beyond the expiration of its shelf-life and operating life, as applicable.

3.2.1.10.7 References

FSAR Section 6.2.4: Primary Containment Isolation System
Table 6.2-56
Figures 6.2-70, Sheets 1 through 43b
FSAR Section 7.3.1.1.2: Primary Containment and Reactor Vessel Isolation Control System (PCRVICS) - Instrumentation and Controls
FSAR Section 5.4.5: Main Steam Isolation System

N2-OP-83, Rev. 1: Primary Containment Isolation System
N2-OP-1, Rev 7: Main Steam System
N2-OP-101C, Rev. 6: Plant Shutdown Operating Procedure

N2-EOP-RPV, Section RL, Rev. 4: RPV Control, RPV Water Level
N2-EOP-6, Att. 1, Rev. 0: RPV Water Level/High Drywell Pressure Associated ESF Actuation

PIDs listed in Table 3.2.1.10-1

Tech. Spec. Section 3.6.2
Tech. Spec. Sections 3/4.6.3, including Table 3.6.3-1

3.2.1.10.8 Initiating Event Potential

Spurious containment isolation would cause loss of cooling to the reactor recirculation pumps and drywell coolers. However, the frequency and impact of this event is enveloped by other initiating events such as loss of an emergency DC bus, loss of offsite power, and loss of RBCLC.

3.2.1.10.9 Operating Experience

There were no particular outstanding operational events relevant to this study. Plant specific component operational data is detailed in section 3.3.2.

3.2.1.10.10 Modeling Assumptions

1. The present model does not credit loss of support systems (AOVs and SOVs fail safe) as redundant to actuation signals.

2. Only RPV Level 2 and drywell pressure signals are modeled. These signals apply to penetrations judged most likely to be open and included in the model. In addition, loss of the isolation function due to loss of input signals is not expected to dominate. There is also the capability to manually actuate the valves.

3. Pre-existing leaks, as a failure mode, are included in the model based on generic data as discussed in the containment Level 2 study (C.2.1.5). The probability of the pre-existing leaks is based on the Pacific Northwest Laboratory Study for the NRC.

3.2.1.10.11 Logic Model and Results

The fault tree(s) is included in Tier 2 documentation, and is available upon request. Quantitative results are summarized in Section 3.3.5 (Tier 1).

Table 3.2.1.10-1
Containment Isolation Screening

Pen. (Dia.)	Description	Valves	Support (a)	Normal	Signal (b)	Fails	Screening (c)		Model	FSAR Figure
1A (26)	Main Steam Line A	(I)MSS*AOV6A (O)MSS*AOV7A	N2,VBS Air,VBS	Open	L1,RM	Closed	RPV	Note 1	NO	6.2-70 sh 1 PID-1E-12 PID-1F-12
(3/4)	Drain Line	(O)MSS*SOV97A	SCI	Closed	L1,RM	Closed	RPV	Note 5	NO	
1B (26)	Main Steam Line B	(I)MSS*AOV6B (O)MSS*AOV7B	N2,VBS Air,VBS	Open	L1,RM	Closed	RPV	Note 1	NO	6.2-70 sh 1
(3/4)	Drain Line	(O)MSS*SOV97B	SCI	Closed	L1,RM	Closed	RPV	Note 5	NO	
1C (26)	Main Steam Line C	(I)MSS*AOV6C (O)MSS*AOV7C	N2,VBS Air,VBS	Open	L1,RM	Closed	RPV	Note 1	NO	6.2-70 sh 1
(3/4)	Drain Line	(O)MSS*SOV97C	SCI	Closed	L1,RM	Closed	RPV	Note 5	NO	
1D (26)	Main Steam Line D	(I)MSS*AOV6D (O)MSS*AOV7D	N2,VBS Air,VBS	Open	L1,RM	Closed	RPV	Note 1	NO	6.2-70 sh 1
(3/4)	Drain Line	(O)MSS*SOV97D	SCI	Closed	L1,RM	Closed	RPV	Note 5	NO	
1A/B/C/D (2)	Main Steam Lines A/B/C/D Drain Line	(O)MSS*MOV208	EHS	Closed	L1,RM	FAI	RPV	Note 5	NO	6.2-70 sh 1 PID-1F-11
2 (6)	Main Steam Drain Line	(I)MSS*MOV207 (I)MSS*MOV189 (I)MSS*MOV111 (O)MSS*MOV112 (O)MSS-MOV187 (O)Condenser	NHS NHS EHS EHS NHS 	Closed Closed Closed Closed Closed N/A	RM RM L1,RM L1,RM RM N/A	FAI FAI FAI FAI FAI N/A	RPV	Note 6	NO	6.2-70 sh 2 PID-1E-12
3	Spare								NO	
4A (24)	Feedwater Line A to RPV	(I)FWS*V12A (O)FWS*AOV23A (O)FWS*MOV21A (O)FWS-V104A	CV CV EHS CV	Open Open Open Open	RF RF RM RF	N/A N/A FAI N/A	RPV	3 Check Valves	NO	6.2-70 sh 3 PID-6B-10
4B (24)	Feedwater Line B to RPV	(I)FWS*V12B (O)FWS*AOV23B (O)FWS*MOV21B (O)FWS-V104B	CV CV EHS CV	Open Open Open Open	RF RF RM RF	N/A N/A FAI N/A	RPV	3 Check Valves	No	6.2-70 sh 3

Table 3.2.1.10-1
Containment Isolation Screening

Pen. (Dia.)	Description	Valves	Support (a)	Normal	Signal (b)	Fails	Screening (c)	Model	FSAR Figure
4A/B (8)	Feedwater Lines	(O)WCS*HOV200	EHS	Open	RM	FAI	RPV 4 Check Valves	No	6.2-70 sh 3
	A & B to RPV	(O)WCS*V346	CV	Open	RF	N/A			PID-6B-10
		(O)WCS*V47	CV	Open	RF	N/A			PID-37B-9
5A (24)	RHS Pump A	(I) None		--	--	--			
	Suction from	(O)RHS*HOV1A	EHS	Open	RM	FAI	SP Note 2	NO	6.2-70 sh 4
	Suppression Pool								
5B (24)	RHS Pump B	(I) None		--	--	--			
	Suction from	(O)RHS*HOV1B	EHS	Open	RM	FAI	SP Note 2	NO	6.2-70 sh 4
	Suppression Pool								
5C (24)	RHS Pump C	(I) None		--	--	--			
	Suction from	(O)RHS*HOV1C	EHS	Open	RM	FAI	SP Note 2	NO	6.2-70 sh 4
	Suppression Pool								
6A (18)	RHS Test Line	(I) None		--	--	--			
	Loop B to	(O)RHS*HOV30B	EHS	Open	RM	FAI	SP Note 2	NO	6.2-70 sh 6
	Suppression Pool								
6B (18)	RHS Test Line	(I) None		--	--	--			
	Loop A to	(O)RHS*HOV30A	EHS	Open	RM	FAI	SP Note 2	NO	6.2-70 sh 6
	Suppression Pool								
7A (4)	RHS Containment	(I) None		--	--	--			
	Spray Loop A to	(O)RHS*HOV33A	EHS	Closed	L1,DP,RM	FAI	SC Note 2	NO	6.2-70 sh 7
	Suppression Pool								
7B (4)	RHS Containment	(I) None		--	--	--			
	Spray Loop B to	(O)RHS*HOV33B	EHS	Closed	L1,DP,RM	FAI	SC Note 2	NO	6.2-70 sh 7
	Suppression Pool								
8A (16)	RHS Containment	(I) None		--	--	--			
	Spray Loop A	(O)RHS*HOV25A	EHS	Closed	RM	FAI	DW Note 2	NO	6.2-70 sh 8
	to Drywell	(O)RHS*HOV15A	EHS	Closed	RM	FAI			
8B (16)	RHS Containment	(I) None		--	--	--			
	Spray Loop B	(O)RHS*HOV25B	EHS	Closed	RM	FAI	DW Note 2	NO	6.2-70 sh 8
	to Drywell	(O)RHS*HOV15B	EHS	Closed	RM	FAI			
9A (12)	RHS/LPCI	(I)RHS*AOV16A	CV	Closed	RF	Closed	RPV Note 2	NO	6.2-70 sh 9
	Loop A to RPV	(O)RHS*HOV24A	EHS	Closed	RM	FAI			
9B (12)	RHS/LPCI	(I)RHS*AOV16B	CV	Closed	RF	Closed	RPV Note 2	NO	6.2-70 sh 9
	Loop B to RPV	(O)RHS*HOV24B	EHS	Closed	RM	FAI			
9C (12)	RHS/LPCI	(I)RHS*AOV16C	CV	Closed	RF	Closed	RPV Note 2	NO	6.2-70 sh 9
	Loop C to RPV	(O)RHS*HOV24C	EHS	Closed	RM	FAI			

Table 3.2.1.10-1
Containment Isolation Screening

Pen. (Dia.)	Description	Valves	Support (a)	Normal	Signal (b)	Fails	Screening (c)	Model	FSAR Figure
10A (12)	RHS SD Loop A to Rx Recirc Loop A	(I)RHS*AOV39A (O)RHS*MOV40A	CV EHS	Closed Closed	RF L3,RP,RM	Closed FAI	RPV Note 2	NO	6.2-70 sh 13
(2)	RHS SD Cooling Return Line Inboard Valve Bypass Line	(I)RHS*MOV67A	EHS	Closed	L3,RP,RM	FAI			
10B (12)	RHS SD Loop B to Rx Recirc Loop B	(I)RHS*AOV39B (O)RHS*MOV40B	CV EHS	Closed Closed	RF L3,RP,RM	Closed FAI	RPV NOTE 2	NO	6.2-70 sh 13
(2)	RHS SD Cooling Return Line Inboard Valve Bypass Line	(I)RHS*MOV67B	EHS	Closed	L3,RP,RM	FAI			
11 (20)	RHS SD Supply from Rx Recirc	(I)RHS*MOV112 (O)RHS*MOV113 (O)RHS*MOV2A (O)RHS*MOV2B	CV EHS EHS EHS	Closed Closed Closed Closed	L3,RP,RM L3,RP,RM	FAI FAI FAI FAI	RPV Note 2	NO	6.2-70 sh 14
		(I)RHS*RV152		Closed	N/A	Closed			
12 (20)	CSH Suction from Suppression Pool	(I) None (O)CSH*MOV118	 EHS	-- Closed	-- RM	-- FAI	SP Note 2	NO	6.2-70 sh 5
13 (12)	CSH Test Return to Suppression	(I) None (O)CSH*MOV111	 EHS	-- Closed	-- L2,DP,RM	-- FAI	SP Note 2	NO	6.2-70 sh 15
(4)	CSH min Flow BPass	(O)CSH*MOV105	EHS	Closed	RM	FAI		NO	
14 (12)	CSH to RPV	(I)CSH*AOV108 (O)CSH*MOV107	CV EHS	Closed Closed	RF RM	Closed FAI	RPV Note 2	NO	6.2-70 sh 9
15 (20)	CSL Suction from Suppression Pool	(I) None (O)CSL*MOV112	 EHS	-- Open	-- RM	-- FAI	SP Note 2	NO	6.2-70 sh 4
16 (12)	CSL to RPV	(I)CSL*AOV101 (O)CSL*MOV104	CV EHS	Closed Closed	RF RM	Closed FAI	RPV Note 2	NO	6.2-70 sh 10
17 (6)	ICS Suction from Suppression Pool	(I) None (O)ICS*MOV136	 DHS	-- Closed	-- RM	-- FAI	SP Note 2	NO	6.2-70 sh 5
18 (2)	ICS Min Flow to Suppression Pool	(I) None (O)ICS*MOV143	 DHS	-- Closed	-- RM	-- FAI	SP Note 2	NO	6.2-70 sh 11

Table 3.2.1.10-1
Containment Isolation Screening

Pen. (Dia.)	Description	Valves	Support (a)	Normal	Signal (b)	Fails	Screening (c)	Model	FSAR Figure
19 (12)	ICS Turbine Exh to Suppression Pool	(I) None		--	--	--			6.2-70 sh 12
		(O)ICS*HOV122	DHS	Open	RM	FAI	SP Note 2	NO	
20 (3/4)	Spare							NO	
21A (10)	Steam to ICS Turb & RHS Heat Exchangers	(I)ICS*HOV128	EHS	Open	RM	FAI	RPV Note 2	NO	6.2-70 sh 16
		(O)ICS*HOV121	EHS	Open	RM	FAI			
(1)	ICS Turb Steam Supply, Bypass Inboard Iso Valve	(I)ICS*HOV170	EHS	Closed	RM	FAI		NO	
21B (4)	Spare							NO	
22 (6)	ICS to RPV	(I)ICS*AOV157	CV	Closed	RF	Closed	RPV Note 2	NO	6.2-70 sh 17
		(O)ICS*AOV156	CV	Closed	RF	Closed			
		(O)ICS*HOV126	DHS	Closed	RM	FAI			
	RHR Rx Head Spray	(O)RHS*HOV104	EHS	Closed	L1,RP,RM	FAI	RPV Note 2	NO	
		(O)RHS*V143	CV	Open	RF	N/A			
23 (8)	WCS Supply from RCS and RPV	(I)WCS*HOV101	NHS	Open	RM	FAI	RPV Note 8	NO	6.2-70 sh 18 PID-37A-9
		(I)WCS*HOV104	NHS	Open	RM	FAI			
		(I)WCS*HOV102	EHS	Open	L2,RM	FAI			
		(O)WCS*HOV112	EHS	Open	L2,RM	FAI			
24 (3)	Spare							NO	
25	RDS Lines to RPV	(I) None		--	--	--			
(1)	53 Insert	(O) multiple		See FSAR	Table 6.2-56	note 17	RPV NOTE 3	NO	N/A
(3/4)	53 Withdrawal	(O) multiple							
26	RDS Lines to RPV	(I) None		--	--	--			
(1)	39 Insert	(O) multiple		See FSAR	Table 6.2-56	note 17	RPV NOTE 3	NO	N/A
(3/4)	39 Withdrawal	(O) multiple							
27	RDS Lines to RPV	(I) None		--	--	--			
(1)	54 Insert	(O) multiple		See FSAR	Table 6.2-56	note 17	RPV NOTE 3	NO	N/A
(3/4)	54 Withdrawal	(O) multiple							
28	RDS Lines to RPV	(I) None		--	--	--			
(1)	39 Insert	(O) multiple		See FSAR	Table 6.2-56	note 17	RPV NOTE 3	NO	N/A
(3/4)	39 Withdrawal	(O) multiple							
29 (1.5)	SLCS to RPV	(I)SLS*V10	CV	Closed	RF	N/A	RPV NOTE 2	NO	6.2-70 sh 43
		(O)SLS*HOV5A	CV, EHS	Closed	RF	Closed			
		(O)SLS*HOV5B	CV, EHS	Closed	RF	Closed			

Table 3.2.1.10-1
Containment Isolation Screening

Pen. (Dia.)	Description	Valves	Support (a)	Normal	Signal (b)	Fails	Screening (c)	Model	FSAR Figure
30A (3)	Spare							NO	
30B (3)	Spare							NO	
31A (1.5)	TIP Drive Guide Tube to RPV	(I) None		--	--	--			
		(O)NMS*SOV1A	SCA	Closed	L2,DP,RM	Closed	Closed	NO	6.2-70 sh 19
		(O)NMS*VEX1A	BYS	Open	RM	Open			
31B (1.5)	TIP Drive Guide Tube to RPV	(I) None		--	--	--			
		(O)NMS*SOV1B	SCA	Closed	L2,DP,RM	Closed	Closed	NO	6.2-70 sh 19
		(O)NMS*VEX1B	BYS	Open	RM	Open			
31C (1.5)	TIP Drive Guide Tube to RPV	(I) None		--	--	--			
		(O)NMS*SOV1C	SCA	Closed	L2,DP,RM	Closed	Closed	NO	6.2-70 sh 19
		(O)NMS*VEX1C	BYS	Open	RM	Open			
31D (1.5)	TIP Drive Guide Tube to RPV	(I) None		--	--	--			
		(O)NMS*SOV1D	SCA	Closed	L2,DP,RM	Closed	Closed	NO	6.2-70 sh 19
		(O)NMS*VEX1D	BYS	Open	RM	Open			
31E (1.5)	TIP Drive Guide Tube to RPV	(I) None		--	--	--			
		(O)NMS*SOV1E	SCA	Closed	L2,DP,RM	Closed	Closed	NO	6.2-70 sh 19
		(O)NMS*VEX1E	BYS	Open	RM	Open			
32 (1.5)	N2 Purge to TIP	(I)GSN*V170	CV	Open	RF	N/A	Closed	NO	6.2-70 sh 42
		(O)GSN*SOV166	SCM	Open	L2,DP,RM	Closed			
33A (4)	CCP Supply to RCS Pump A	(I)CCP*MOV94A	EHS	Open	L2,DP,RM	FAI	Closed	NO	6.2-70 sh 20
		(O)CCP*MOV17A	EHS	Open	L2,DP,RM	FAI			
		(O)CCP*V117	CV	Open	RF	N/A			
		(O)CCP*MOV93A	NF	Open	N/A	FAI			
33B (4)	CCP Supply to RCS Pump B	(I)CCP*MOV94B	EHS	Open	L2,DP,RM	FAI	Closed	NO	6.2-70 sh 20
		(O)CCP*MOV17B	EHS	Open	L2,DP,RM	FAI			
		(O)CCP*V59	CV	Open	RF	N/A			
		(O)CCP*MOV93B	NF	Open	N/A	FAI			
		(I)CCP*RV170	-	Closed	N/A	N/A			
34A (4)	CCP Return From RCS Pump A	(I)CCP*MOV16A	EHS	Open	L2,DP,RM	FAI	Closed	NO	6.2-70 sh 21
		(O)CCP*MOV15A	EHS	Open	L2,DP,RM	FAI			
		(O)CCP*MOV22A	NF	Open	N/A	N/A			
34B (4)	CCP Return From RCS Pump B	(I)CCP*MOV16B	EHS	Open	L2,DP,RM	FAI	Closed	NO	6.2-70 sh 21
		(O)CCP*MOV15B	EHS	Open	L2,DP,RM	FAI			
		(O)CCP*MOV22B	NF	Open	N/A	N/A			
		(I)CCP*RV171	-	Closed	N/A	N/A			
35 (4)	Spare							NO	

Table 3.2.1.10-1
Containment Isolation Screening

Pen. (Dia.)	Description	Valves	Support (a)	Normal	Signal (b)	Fails	Screening (c)		Model	FSAR Figure
36 (2)	Service Air to Drywell	(I)SAS*HCV163	SCM	Closed	LMC	N/A	DW	NOTE 4	NO	6.2-70 sh 22
		(O)SAS*HCV161	SCM	Closed	LMC	N/A				PID-19J-11
37 (2)	Breathing Air to Drywell	(I)AAS*HCV136	-	Closed	LMC	N/A	DW	NOTE 4	NO	6.2-70 sh 22
		(O)AAS*HCV134	-	Closed	LMC	N/A				PID-20E-6
38A (3/4)	RDS to Recirc Pump A Seal	(I)RCS*V60A	CV	Open	RF	N/A	RPV	3 CHECK VALVES	NO	6.2-70 sh 23
		(O)RCS*V90A	CV	Open	RF	N/A				
		(O)RCS*V59A	CV	Open	RF	N/A				
38B (3/4)	RDS to Recirc Pump B Seal	(I)RCS*V60B	CV	Open	RF	N/A	RPV	3 CHECK VALVES	NO	6.2-70 sh 23
		(O)RCS*V90B	CV	Open	RF	N/A				
		(O)RCS*V59B	CV	Open	RF	N/A				
39 (6)	Drywell Floor Drain To Tank	(I)DFR*MOV121	EHS	Open	L2,DP,RM	FAI	DW		YES	6.2-70 sh 24
		(O)DFR*MOV120	EHS	Open	L2,DP,RM	FAI				PID-63E-11
40 (4)	Equipment Drains from Drywell	(I)DER*MOV119	EHS	Open	L2,DP,RM	FAI	DW		YES	6.2-70 sh 24
		(O)DER*MOV120	EHS	Open	L2,DP,RM	FAI				PID-67A-9
41 (3/4)	Rx Coolant Recirc to Sample Cooler	(I)RCS*SOV104	SCI	Closed	L2,RM	Closed	RPV		NO	6.2-70 sh 25
		(O)RCS*SOV105	SCI	Closed	L2,RM	Closed				
		(O)SST*AOV150	AIR			Closed				
42A (2)	Fire Protection for Rx Recirc Pump	(I)FPW*SOV219	SCM	Closed	L2,DP,RM	Closed	Closed, capped		NO	6.2-70 sh 26
		(O)FPW*SOV218	SCM	Closed	L2,DP,RM	Closed				
42B (2)	Fire Protection for Rx Recirc Pump	(I)FPW*SOV221	SCM	Closed	L2,DP,RM	Closed	Closed, capped		NO	6.2-70 sh 26
		(O)FPW*SOV220	SCM	Closed	L2,DP,RM	Closed				
43 (6)	Drywell Floor Drain Tank Vent to Drywell	(I)DFR*MOV140	EHS	Open	L2,DP,RM	FAI	DW		YES	6.2-70 sh 27
		(O)DFR*MOV139	EHS	Open	L2,DP,RM	FAI				PID-63E-11
44A (3)	Capped Spare								NO	
44B (3)	Capped Spare								NO	
44C (3)	Capped Spare								NO	
44D (3)	Capped Spare								NO	
44E (2)	Service Air to Drywell	(I)SAS*HCV162	SCM	Closed	LMC	N/A	DW	NOTE 4	NO	6.2-70 sh 22
		(O)SAS*HCV160	SCM	Closed	LMC	N/A				PID-19J-11
44F (2)	Breathing Air to Drywell	(I)AAS*HCV137	-	Closed	LMC	N/A	DW	NOTE 4	NO	6.2-70 sh 22
		(O)AAS*HCV135	-	Closed	LMC	N/A				PID-20E-6

Table 3.2.1.10-1
Containment Isolation Screening

Pen. (Dia.)	Description	Valves	Support (a)	Normal	Signal (b)	Fails	Screening (c)		Model	FSAR Figure
45 (2)	Equipment Drain Tank (2DER-TK1) Vent to Drywell	(I)DER*MOV130 (O)DER*MOV131	EHS EHS	Open Open	L2,DP,RM L2,DP,RM	FAI FAI	DW		YES	6.2-70 sh 27 PID-67A-9
46A (8)	CCP Supply to Drywell Space Cooler	(I)CCP*MOV273 (O)CCP*MOV265	EHS EHS	Open Open	L2,DP,RM L2,DP,RM	FAI FAI	Closed		NO	6.2-70 sh 28
46B (4)	Capped Spare								NO	
46C (4)	Fire Protection H20 for Containment Hose Reel Standpipe	(I)FPW-V629 (I)10 FHRs (O)Spool piece removed during normal operation (Open)	- -	Closed Closed	N/A N/A	N/A N/A	DW	NOTE 4	NO	PID-43G-11
46D (4)	Capped Spare								NO	
47 (8)	CCP Return from Drywell Space Cooler	(I)CCP*MOV122 (O)CCP*MOV124	EHS EHS	Open Open	L2,DP,RM L2,DP,RM	FAI FAI	Closed		NO	6.2-70 sh 28
48 (14)	Purge Exhaust from Drywell	(I)CPS*AOV108 (O)CPS*AOV110	Air, SCM Air, SCM	Closed Closed	L2,DP,RM L2,DP,RM	Closed Closed	DW	NOTE 7	YES	6.2-70 sh 29 PID-61A-8
49 (14)	Purge Inlet to Drywell	(I)CPS*AOV106 (O)CPS*AOV104	Air, SCM Air, SCM	Closed Closed	L2,DP,RM L2,DP,RM	Closed Closed	DW	NOTE 7	YES	6.2-70 sh 29
50 (12)	Purge Inlet to Wetwell	(I)CPS*AOV107 (O)CPS*AOV105	Air, SCM Air, SCM	Closed Closed	L2,DP,RM L2,DP,RM	Closed Closed	SC	NOTE 7	YES	6.2-70 sh 29
51 (12)	Purge Exhaust from Wetwell	(I)CPS*AOV109 (O)CPS*AOV111	Air, SCM Air, SCM	Closed Closed	L2,DP,RM L2,DP,RM	Closed Closed	SC	NOTE 7	YES	6.2-70 sh 29
52A (1)	Capped Spare								NO	
52B (1)	Capped Spare								NO	
53A (1.5)	Instrument Air to ADS Vlv Accumulators	(I)IAS*V448 (O)IAS*SOV164	CV SCM	Open Open	RF L2,DP,RM	N/A Closed	Closed		NO	6.2-70 sh 30 PID-19D-11
53B (1.5)	Instrument Air to ADS Vlv Accumulators	(I)IAS*V449 (O)IAS*SOV165	CV SCM	Open Open	RF L2,DP,RM	N/A Closed	Closed		NO	6.2-70 sh 30 PID-19F-8
53C (1.5)	Instrument Air to SRV Accumulator Tank	(I)IAS*SOV184 (O)IAS*SOV166	SCM SCM	Open Open	L2,DP,RM L2,DP,RM	Closed Closed	Closed		NO	6.2-70 sh 30 PID-19D-11
54A (3)	Capped Spare								NO	
55A (3)	Hydrogen Recombiner 1A Supply to Wetwell	(I)HCS*MOV4A (O)HCS*MOV1A	EHS EHS	Closed Closed	L2,DP,RM L2,DP,RM	FAI FAI	SC	NOTE 2	NO	6.2-70 sh 31a PID-62A-10

Table 3.2.1.10-1
Containment Isolation Screening

Pen. (Dia.)	Description	Valves	Support (a)	Normal	Signal (b)	Fails	Screening (c)		Model	FSAR Figure
55B (3)	Hydrogen Recombiner 1B Supply to Wetwell	(I)HCS*MOV4B	EHS	Closed	L2,DP,RM	FAI	SC	NOTE 2	NO	6.2-70 sh 31a
		(O)HCS*MOV1B	EHS	Closed	L2,DP,RM	FAI				
56A (3)	Hydrogen Recombiner 1A Return from Drywell	(I)HCS*MOV6A	EHS	Closed	L2,DP,RM	FAI	DW	NOTE 2	NO	6.2-70 sh 31a
		(O)HCS*MOV3A	EHS	Closed	L2,DP,RM	FAI				PID-62D-10
		(O)HCS*MOV25A	EHS	Closed	RM	FAI				PID-62B-8
56B (3)	Hydrogen Recombiner 1B Return from Drywell	(I)HCS*MOV6B	EHS	Closed	L2,DP,RM	FAI	DW	NOTE 2	NO	6.2-70 sh 31a
		(O)HCS*MOV3B	EHS	Closed	L2,DP,RM	FAI				
		(O)HCS*MOV25B	EHS	Closed	RM	FAI				
57A (3)	Hydrogen Recombiner 1A Return from Wetwell	(I)HCS*MOV5A	EHS	Closed	L2,DP,RM	FAI	SC	NOTE 2	NO	6.2-70 sh 31
		(O)HCS*MOV2A	EHS	Closed	L2,DP,RM	FAI				
		(O)HCS*MOV25A	EHS	Closed	RM	FAI				
57B (3)	Hydrogen Recombiner 1B Return from Wetwell	(I)HCS*MOV5B	EHS	Closed	L2,DP,RM	FAI	SC	NOTE 2	NO	6.2-70 sh 31
		(O)HCS*MOV2B	EHS	Closed	L2,DP,RM	FAI				
		(O)HCS*MOV25B	EHS	Closed	RM	FAI				
58 (2)	Containment Purge to Drywell	(I)CPS*SOV122	SCM	Closed	L2,DP,RM	Closed	DW	NOTE 7	YES	6.2-70 sh 29
		(O)CPS*SOV120	SCM	Closed	L2,DP,RM	Closed				PID-61A-8
59 (2)	Containment Purge to Wetwell	(I)CPS*SOV121	SCM	Closed	L2,DP,RM	Closed	SC	NOTE 7	YES	6.2-70 sh 29
		(O)CPS*SOV119	SCM	Closed	L2,DP,RM	Closed				
60A (3/4)	CMS from Drywell	(I)CMS*SOV61A	SCM	Open	L2,DP,RM	Closed	DW, small/closed outside		NO	6.2-70 sh 32
		(O)CMS*SOV60A	SCM	Open	L2,DP,RM	Closed				
60B (3/4)	CMS from Drywell	(I)CMS*SOV24A	SCM	Open	L2,DP,RM	Closed	DW, small/closed outside		NO	6.2-70 sh 32
		(O)CMS*SOV24C	SCM	Open	L2,DP,RM	Closed				
60C (3/4)	CMS from Drywell	(I)CMS*SOV63A	SCM	Open	L2,DP,RM	Closed	DW, small/closed outside		NO	6.2-70 sh 32
		(O)CMS*SOV62A	SCM	Open	L2,DP,RM	Closed				
60D (3/4)	CMS from Drywell	(I)CMS*SOV33A	SCM	Open	L2,DP,RM	Closed	DW, small/closed outside		NO	6.2-70 sh 32
		(O)CMS*SOV32A	SCM	Open	L2,DP,RM	Closed				
60E (3/4)	CMS from Drywell	(I)CMS*SOV61B	SCM	Open	L2,DP,RM	Closed	DW, small/closed outside		NO	6.2-70 sh 32
		(O)CMS*SOV60B	SCM	Open	L2,DP,RM	Closed				
60F (3/4)	CMS from Drywell	(I)CMS*SOV24B	SCM	Open	L2,DP,RM	Closed	DW, small/closed outside		NO	6.2-70 sh 32
		(O)CMS*SOV24D	SCM	Open	L2,DP,RM	Closed				
60G (3/4)	CMS from Drywell	(I)CMS*SOV63B	SCM	Open	L2,DP,RM	Closed	DW, small/closed outside		NO	6.2-70 sh 32
		(O)CMS*SOV62B	SCM	Open	L2,DP,RM	Closed				
60H (3/4)	CMS from Drywell	(I)CMS*SOV33B	SCM	Open	L2,DP,RM	Closed	DW, small/closed outside		NO	6.2-70 sh 32
		(O)CMS*SOV32B	SCM	Open	L2,DP,RM	Closed				
61A (3/4)	Capped Spare								NO	

Table 3.2.1.10-1
Containment Isolation Screening

Pen. (Dia.)	Description	Valves	Support (a)	Normal	Signal (b)	Fails	Screening (c)	Model	FSAR Figure
61B (3/4)	CHS from Wetwell	(I)CHS*SOV26A (O)CHS*SOV26C	SCH SCH	Open Open	L2,DP,RM L2,DP,RM	Closed Closed	SC, small/closed outside	NO	6.2-70 sh 32
61C (3/4)	CHS to Wetwell	(I)CHS*SOV34A (O)CHS*SOV35A	SCH SCH	Open Open	L2,DP,RM L2,DP,RM	Closed Closed	SC, small/closed outside	NO	6.2-70 sh 32
61D (3/4)	Capped Spare								
61E (3/4)	CHS from Wetwell	(I)CHS*SOV26B (O)CHS*SOV26D	SCH SCH	Open Open	L2,DP,RM L2,DP,RM	Closed Closed	SC, small/closed outside	NO	6.2-70 sh 32
61F (3/4)	CHS to Wetwell	(I)CHS*SOV34B (O)CHS*SOV35B	SCH SCH	Open Open	L2,DP,RM L2,DP,RM	Closed Closed	SC, small/closed outside	NO	6.2-70 sh 32
67 (10)	Spare							NO	
68 (10)	Capped Spare							NO	
69 (6)	Spare							NO	
70 (6)	Capped Spare							NO	
71 (3)	Spare							NO	
72 (14)	Capped Spare							NO	
73 (6)	RHS Relief Valve Discharge to Suppression Pool	(I) None (O)RHS*RV108 (O)RHS*RV20C		-- N/A	-- None	-- N/A	SP NOTE 2	NO	6.2-70 sh 33
74 (6)	Flanged Spare							NO	
75 (3)	Capped Spare							NO	
76 (3)	Capped Spare							NO	
77 (1.5)	Capped Spare							NO	
78 (1.5)	Capped Spare							NO	
79 (1.5)	Capped Spare							NO	
80 (1.5)	Spent Fuel Pool Cooling	(I)SFC*V204 (I)SFC-V265 (O)SFC*V203 (O)SFC-V395		Closed Closed Closed Closed	N/A N/A N/A N/A	N/A N/A N/A N/A	RPV, seal drain	NO	6.2-70 sh 40 PID-38C-8
81 (1.5)	Capped Spare							NO	
82 (1)	Capped Spare							NO	
83 (1)	Capped Spare							NO	
85 (1)	Capped Spare							NO	
86 (1)	Capped Spare							NO	
87 (1)	Capped Spare							NO	
88A (3/4)	RHS Safety Valve Discharge to Suppression Pool	(I) None (O) See Note 33 of FSAR Table 6.2-56		--	--	--	SP NOTE 2	NO	6.2-70 sh 34

Table 3.2.1.10-1
Containment Isolation Screening

Pen. (Dia.)	Description	Valves	Support (a)	Normal	Signal (b)	Fails	Screening (c)	Model	FSAR Figure
88B (3/4)	RHS Safety Valve Discharge to Suppression Pool	(I) None (O) See Note 33 of FSAR Table 6.2-56		--	--	--	SP NOTE 2	NO	6.2-70 sh 34
89A (3/4)	LMS from Drywell	(I)LMS*SOV152 (O)LMS*SOV153	SCH SCH	Closed Closed	L2,DP,RM L2,DP,RM	Closed Closed	DW, small/closed outside	NO	6.2-70 sh 35
89B (3/4)	Capped Spare							NO	
89C (3/4)	LMS from Wetwell	(I)LMS*SOV156 (O)LMS*SOV157	SCH SCH	Closed Closed	L2,DP,RM L2,DP,RM	Closed Closed	SC, small/closed outside	NO	6.2-70 sh 35
90 (1.5) (3/4)	ICS Vacuum Breaker	(I) None (O)ICS*MOV148 (O)ICS*MOV164 (O)RHS*V192	 DHS DHS 	-- Open Open Closed	-- DP,RM DP,RM LC	-- FAI FAI N/A	SC NOTE 2	NO	6.2-70 sh 36
91A (1.5)	Instrument Air to Drywell	(I)IAS*SOV185 (O)IAS*SOV167	SCH SCH	Open Open	L2,DP,RM L2,DP,RM	Closed Closed	Closed	NO	6.2-70 sh 37 PID-19G-14
91B (1.5)	Instrument Air to Drywell	(I)IAS*SOV180 (O)IAS*SOV168	SCH SCH	Open Open	L2,DP,RM L2,DP,RM	Closed Closed	Closed	NO	6.2-70 sh 37 PID-19G-14
91C (1.5)	Capped Spare							NO	
91D (1.5)	Capped Spare							NO	
92 (1)	N2 Supply to Actuators for 2CPS*AOV109	(I)CPS*V51 (O)CPS*SOV133	CV SCH	Closed Closed	RF L2,DP,RM	Closed Closed	Closed	NO	6.2-70 sh 43a PID-61A-8
96 (1)	N2 Supply to Actuators for 2CPS*AOV107	(I)CPS*V50 (O)CPS*SOV132	CV SCH	Closed Closed	RF L2,DP,RM	Closed Closed	Closed	NO	6.2-70 sh 43b PID-61A-8
98A (3)	RHR Relief Valve Discharge to Suppression Pool	(I) None (O)CSL*RV123 (O)CSL*RV105 (O)RHS*RV61A (O)RHS*RV110 (O)RHS*RV139 (O)RHS*RV20A		-- N/A	-- None	-- N/A	SP NOTE 2	NO	6.2-70 sh 38
98B (3)	RHR Relief Valve Discharge to	(I) None (O)CSH*RV114 (O)CSH*RV113 (O)RHS*RV61B (O)RHS*RV61C (O)RHS*RV20B		-- N/A	-- None	-- N/A	SP NOTE 2	NO	6.2-70 sh 38

Table 3.2.1.10-1
Containment Isolation Screening

Pen. (Dia.)	Description	Valves	Support (a)	Normal	Signal (b)	Fails	Screening (c)	Model	FSAR Figure
99A (3/4)	Hydraulic Unit from Recirc Flow Cntl Vlv HYV 17A (Drain Line)	(I)RCS*SOV82A (O)RCS*SOV68A	VBS VBS	Open Open	L2,DP,RM L2,DP,RM	Closed Closed	Closed	NO	6.2-70 sh 39
99B (1)	Hydraulic Unit from Recirc Flow Cntl Vlv HYV 17A (Open Line)	(I)RCS*SOV81A (O)RCS*SOV67A	VBS VBS	Open Open	L2,DP,RM L2,DP,RM	Closed Closed	Closed	NO	6.2-70 sh 39
99C (1)	Hydraulic Unit from Recirc Flow Cntl Vlv HYV 17A (Pilot Line)	(I)RCS*SOV80A (O)RCS*SOV66A	VBS VBS	Open Open	L2,DP,RM L2,DP,RM	Closed Closed	Closed	NO	6.2-70 sh 39
99D (1)	Hydraulic Unit from Recirc Flow Cntl Vlv HYV 17A (Closed Line)	(I)RCS*SOV79A (O)RCS*SOV65A	VBS VBS	Open Open	L2,DP,RM L2,DP,RM	Closed Closed	Closed	NO	6.2-70 sh 39
100A (3/4)	Hydraulic Unit from Recirc Flow Cntl Vlv HYV 17B (Drain Line)	(I)RCS*SOV82B (O)RCS*SOV68B	VBS VBS	Open Open	L2,DP,RM L2,DP,RM	Closed Closed	Closed	NO	6.2-70 sh 39
100B (1)	Hydraulic Unit from Recirc Flow Cntl Vlv HYV 17B (Open Line)	(I)RCS*SOV81B (O)RCS*SOV67B	VBS VBS	Open Open	L2,DP,RM L2,DP,RM	Closed Closed	Closed	NO	6.2-70 sh 39
100C (1)	Hydraulic Unit from Recirc Flow Cntl Vlv HYV 17B (Pilot Line)	(I)RCS*SOV80B (O)RCS*SOV66B	VBS VBS	Open Open	L2,DP,RM L2,DP,RM	Closed Closed	Closed	NO	6.2-70 sh 39
100D (1)	Hydraulic Unit from Recirc Flow Cntl Vlv HYV 17B (Closed Line)	(I)RCS*SOV79B (O)RCS*SOV65B	VBS VBS	Open Open	L2,DP,RM L2,DP,RM	Closed Closed	Closed	NO	6.2-70 sh 39
multiple	All Instrument Lines from Reactor Vessel	(I) None (O)EF Check Valves		-- Open	-- Excess Flow	-- Open	RPV Note 9	NO	6.2-70 sh 41
multiple	All Instrument Lines Penetrating Primary Containment	(I) None (O)EFV		-- Open	-- Excess Flow	-- Open	RPV Note 9	NO	6.2-70 sh 41

TABLE 3.2.1.10-1 NOTES

a. Key to Valve Support System Requirements (Valve Motive Power)

AIR	Instrument Air
N_2	Nitrogen
NHS	Normal AC - MCC
EHS	Emergency AC - MCC
VBS	Vital UPS 120V AC
SCI	Normal 120V AC (Non-UPS)
SCM	Emergency 120V AC (Non-UPS)
DMS	Emergency 125V DC
CV	Check Valve
NF	Non Function Valve Operator

b. Key to Valve Isolation Signals

L1	RPV Level 1
L2	RPV Level 2
L3	RPV Level 3
RM	Remote Manual
RF	Reverse Flow (check valve)
DP	Drywell Pressure High
RP	RPV Pressure High
LMC	Local Manual Control, Locked Closed, Indication in Control Room
LC	Locked Closed

Screening Notes

1. The main steam lines will remain open after a plant trip unless an automatic signal or manual action causes isolation. Also, loss of support systems cause isolation valves to fail closed. MSIV closure would be of interest given a break outside containment. For example, steam line breaks with MSIV failure would be a LOCA outside containment. The significance of MSIV failures to close are evaluated as potential LOCA outside containment initiators.

2. ECCS connections to the suppression pool, suppression chamber and drywell are considered closed systems outside containment or extensions to the containment boundary. Connections to the RPV are evaluated as potential LOCA outside containment initiators.

3. Penetrations 25 through 28, which are the 185 control rod drive insert and withdrawal lines, are considered as initiating events and LOCA outside containment.

4. Normally closed penetrations that open into the drywell or suppression chamber air spaces are administratively locked with indication in the Control Room. In addition,

the containment monitoring system would provide redundant indication of an open penetration since purge and nitrogen addition would be abnormal.

5. These are small, normally closed drains from outer MSIV with indication in Control Room. First valve (97A, B, C, D) fails closed and common second valve (MOV208) fails as is.

6. Three levels of redundancy isolate the Main Steam Line drains. The valves; MOV111, MOV112, and either MOV207 or MOV189; are normally closed and have Control Room indication. MOVs 207 and 189 are in parallel upstream of MOV111

7. Penetrations 48 through 51 are allowed to be open for purging (inerting) 90 hours per 365 days during Modes 1, 2, and 3 (N2-OP-101C, Rev. 06 and Tech. Spec. 3.6.1.7). This is approximately 1 percent of the time. Purge exhaust (penetrations 48 and 51) are allowed to be open for pressure control through SOV102 (2" line) as long as AOV101 (20" line) is closed. At present, this appears to be an infrequent event.

   During power operation with the containment previously inerted, nitrogen makeup is provided through penetrations 58 and 59. At present, this appears to be an infrequent event.

   All AOV penetrations are conservatively assumed to be open 10% of the time.

8. The reactor water cleanup system is designed for high pressure, normally connected to RPV and therefore considered a closed system. Considered as a potential LOCA outside containment path.

9. Instrument lines are small and their failures are considered as initiating events and LOCA outside containment.

Table 3.2.1.10-2

PENETRATIONS MODELED IN TOP EVENT IS

Penetration	Valves [1]	Motive Support [2]	Signal [3]
39	(I) 2DFR*MOV121	2EHS*MCC302	RM, L2(II), DP(II)
	(O) 2DFR*MOV120	2EHS*MCC102	RM, L1(I), DP(I)
40	(I) 2DER*MOV119	2EHS*MCC302	RM, L2(II), DP(II)
	(O) 2DER*MOV120	2EHS*MCC102	RM, L1(I), DP(I)
43	(I) 2DFR*MOV140	2EHS*MCC302	RM, L2(II), DP(II)
	(O) 2DFR*MOV139	2EHS*MCC102	RM, L1(I), DP(I)
45	(I) 2DER*MOV130	2EHS*MCC302	RM, L2(II), DP(II)
	(O) 2DER*MOV131	2EHS*MCC102	RM, L1(I), DP(I)
48	(I) 2CPS*AOV108	Nitrogen, FC	
	2CPS*SOV108[4]	SCM (A2), FC	RM, L2(II), DP(II)
	2IAS*SOV180[4]	SCM (A2), FC	RM, L2(II), DP(II)
	2IAS*SOV168[4]	SCM (A1), FC	RM, L1(I), DP(I)
	(O) 2CPS*AOV110	Inst. Air, FC	
	2CPS*SOV110[4]	SCM (A1), FC	RM, L1(I), DP(I)
49	(I) 2CPS*AOV106	Nitrogen, FC	
	2CPS*SOV106[4]	SCM (A2), FC	RM, L2(II), DP(II)
	2IAS*SOV180[4]	SCM (A2), FC	RM, L2(II), DP(II)
	2IAS*SOV168[4]	SCM (A1), FC	RM, L1(I), DP(I)
	(O) 2CPS*AOV104	Inst. Air, FC	
	2CPS*SOV104[4]	SCM (A1), FC	RM, L1(I), DP(I)
50	(I) 2CPS*AOV107	Nitrogen, FC	
	2CPS*SOV107[4]	SCM (A2), FC	RM, L2(II), DP(II)
	2CPS*SOV132[4]	SCM (A2), FC	RM, L2(II), DP(II)
	(O) 2CPS*AOV105	Inst. Air, FC	
	2CPS*SOV105[4]	SCM (A1), FC	RM, L1(I), DP(I)
51	(I) 2CPS*AOV109	Nitrogen, FC	
	2CPS*SOV109[4]	SCM (A2), FC	RM, L2(II), DP(II)
	2CPS*SOV133[4]	SCM (A2), FC	RM, L2(II), DP(II)
	(O) 2CPS*AOV111	Inst. Air, FC	
	2CPS*SOV111[4]	SCM (A1), FC	RM, L1(I), DP(I)
58	(I) 2CPS*SOV122	SCM (A2), FC	RM, L2(II), DP(II)
	(O) 2CPS*SOV120	SCM (A1), FC	RM, L1(I), DP(I)
59	(I) 2CPS*SOV121	SCM (A2), FC	RM, L2(II), DP(II)
	(O) 2CPS*SOV119	SCM (A1), FC	RM, L1(I), DP(I)

TABLE 3.2.1.10-2 NOTES

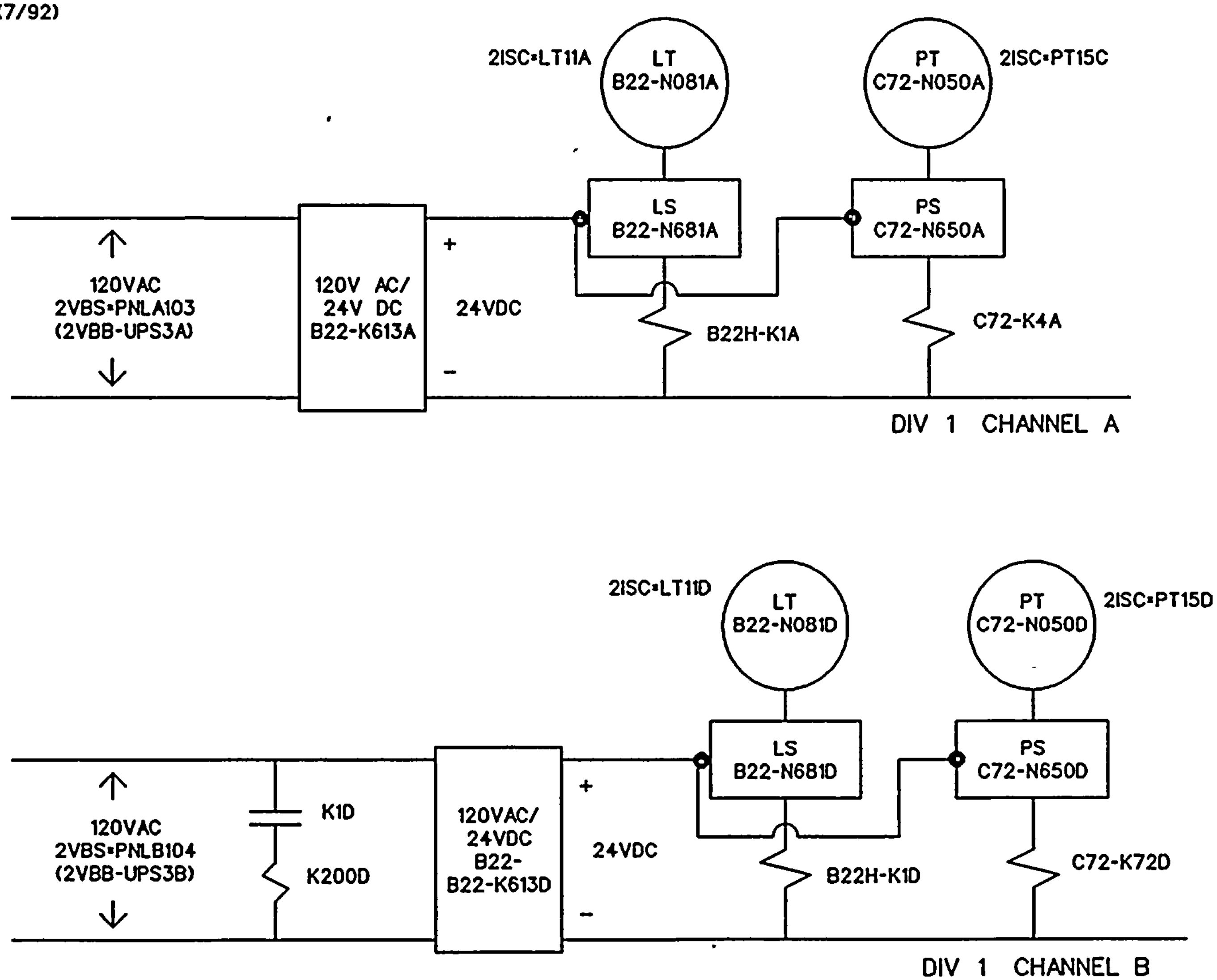
1. (I) = Inside Containment
   (O) = Outside Containment

2. FC = Fail closed on loss of support
   SCM(A2) = Emergency 120V AC Division II (Non-UPS)
   SCM(A1) = Emergency 120V AC Division I (Non-UPS)

3. RM = Remote Manual in Control Room
   L2(I) = RPV Level 2 Division I
   L2(II) = RPV Level 2 Division II
   DP(I) = High Drywell Press Division I
   DP(II) = High Drywell Press Division II

4. These SOVs are containment isolation vales that supply nitrogen to the inside AOV. If they successfully close, nitrogen will bleed off the AOV resulting in their eventual closure. The model neglects multiple SOV failures regarding the inside AOVs (i.e., only the AOV is modeled).

Rev 0 (7/92)

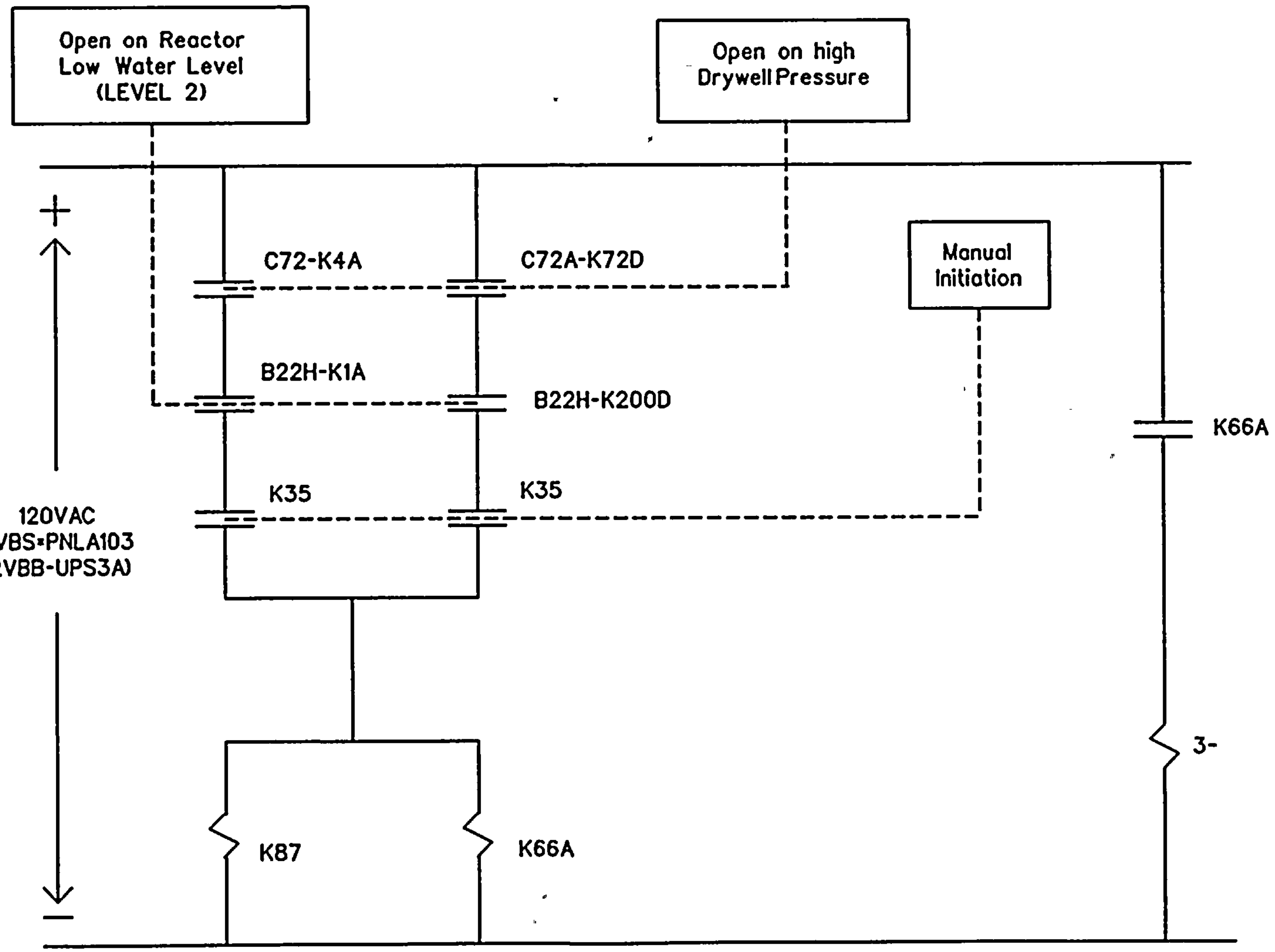


Containment Isolation Initiation Signals
All relays are de-energized on trip

Figure 3.2.1.10-1
CIS Initiation Logic

I:\NMP2IPE\DGN\2S10.DGN Jul. 06. 1992 16:19:53

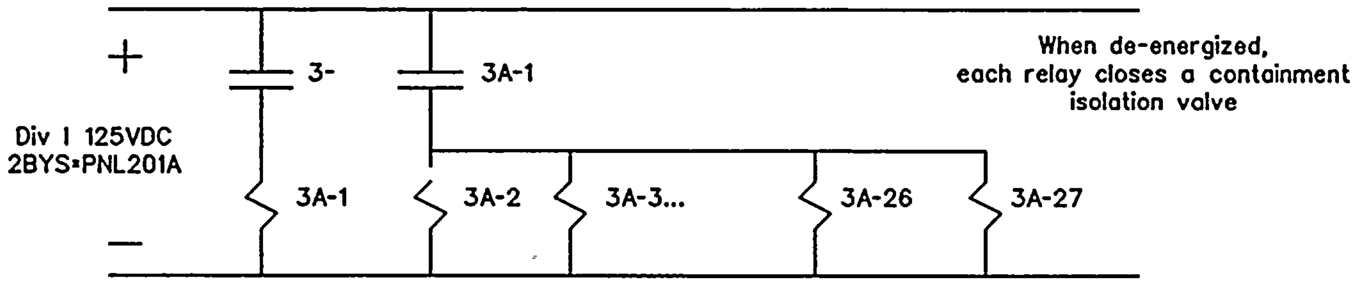
Rev 0 (7/92)



Typical Div I Containment Isolation Signal
all relays are de-energized on trip

Figure 3.2.1.10-2
Containment Isolation Logic

Rev 0 (7/92)



Div I Containment Isolation Actuation relays

Figure 3.2.1.10-3
CIS Actuation Logic

I: \NMP2IPE\DGN\2S10.DGN Jul. 06. 1992 16:21:30

System 11

Ventilation Systems

3.2.1.11 Ventilation Systems

This section documents the evaluation of ventilation systems as a spatial, environmental dependency on other plant equipment. The importance of these spatial dependencies depends on:

- Heat loads in the area,
- The size of the area versus heat loads,
- Sensitivity of equipment in the area, and
- The potential for successful recovery actions.

Usually, loss of ventilation results in long term failure of equipment as the room heats up and exceeds equipment temperature limits. Given instrumentation and procedures, there should be a high likelihood of recovery in most cases. Even the most sensitive equipment may survive if doors are opened or portable fans are used.

Table 3.2.1.11-1 documents an initial review of plant buildings, rooms within the buildings, ventilation systems that support the buildings and rooms, and ventilation equipment and support systems. This review was used to provide an initial screening of ventilation systems and areas based on their potential impact on important safety equipment in the PRA model. The following ventilation systems were identified as requiring additional evaluation:

- Control Building - this area contains the emergency switchgear and relay rooms, solid state protection equipment, and other sensitive electrical equipment.

- Emergency Diesel Generator Rooms - these rooms are expected to heat up very quickly without ventilation and impact emergency diesel operability.

- Reactor Building - this area contains important equipment including the RCIC and HPCS pump rooms.

- North & South Auxiliary Bays - these areas contain RHR and LPCS pumps as well as the supporting motor control centers (MCCs).

- Screenwell - this area contains the service water and fire water pumps.

Each of the above areas is evaluated in the subsections below. Based on this evaluation, the following ventilation dependencies were included in the IPE model:

- The diesel control room unit coolers are included in the emergency diesel generator model for each division (Top Events A1, A2 and HS).

- Redundant unit coolers in the HPCS pump room are included in the HPCS model (Top Event HS).

- Redundant unit coolers in the North and South Auxiliary Bay MCC rooms are modeled as top events MA and MB. Failure of MA is modeled as failure of the LPCS and RHR A systems. Failure of MB is modeled as failure of the RHR B and C systems.

3.2.1.11.1 Control Building

3.2.1.11.1.1 Control Building Ventilation Description

The Control Building ventilation system can be divided into the following subsystems:

- Control Room Ventilation
- Relay Room Ventilation
- Control Building Special Filter Train
- Control Building Chilled Water
- Standby Switchgear/Battery Room Ventilation

<u>Control Room Ventilation</u>
The Control Room ventilation subsystem supplies tempered, recirculated air and outdoor air to areas on elevation 306' including the Control Room. The subsystem is served by two full capacity redundant air conditioning units (2HVC*ACU1A and 1B) as shown on the simplified diagram in Figure 3.2.1.11-1A. During normal operation one unit is operating with the other in standby. The standby unit will auto-start on low flow or high temperature of the operating unit.

ACU1A and 1B depend on Division I and Division II AC power. Normal cooling is provided by the Control Building chilled water system. In the event of failure of both chiller compressors, the ACUs can be manually lined up to the service water system for backup cooling.

As shown in Figure 3.2.1.11-1A, there are makeup and exhaust fans (2HVC-HVU1 and 2HVC-FN3) that are manually aligned when smoke removal is required.

<u>Relay Room Ventilation</u>
The relay room ventilation subsystem supplies tempered, recirculated air and outdoor air to areas on elevation 288' including the relay room. The subsystem is served by two full capacity redundant air conditioning units (2HVC*ACU2A and 2B) as shown on the simplified diagram in Figure 3.2.1.11-1A. The computer room is also provided with two air conditioning units (2HVC-ACU4A and 4B) with booster fans (2HVC-FN17 and FN18). During normal operation, one relay room unit is operating and one computer room unit is operating with another in standby in each room. The standby unit will auto-start on low flow or high temperature of the operating unit.

ACU2A and 2B depend on Division I and Division II AC power while the computer room ventilation depends on normal AC power. Normal cooling is provided by the Control Building chilled water system. In the event of failure of both chiller compressors, the ACUs may be manually lined up to the service water system for backup cooling.

As shown in Figure 3.2.1.11-1A, there are makeup and exhaust fans (2HVC-HVU1 and 2HVC-FN3) that are manually aligned when smoke removal is required.

Control Building Special Filter Trains
Outside air makeup is provided to the Control Room, relay room, and computer room via two normally open intake dampers (2HVC*AOD61A and 61B) as shown in Figure 3.2.1.11-1A. This figure also shows the normally open special filter train bypass valves (2HVC*MOV1A and 1B) and the filter trains themselves. The MOVs close and the filter trains open and operate on a high radiation signal or LOCA signal. The filter trains and MOVs depend on Division I and II emergency AC power.

Control Building Chilled Water
The Control Building chilled water system provides chilled water (approximately 45°F) to each of the following ACUs as shown in Figure 3.2.1.11-1D:

Room	Unit Cooler IDs
Control Room	2HVC*ACU1A and 1B
Relay Room	2HVC*ACU2A and 2B
Computer Room	2HVC-ACU4A and 4B
Remote Shutdown Room	2HVC*ACU3A and 3B

As shown in the figure, the system is a closed loop piping system consisting of two independent, redundant chilled water loops where one loop cools the "A" ACUs and the other cools the "B" ACUs. In the event of failure of both chiller units (2HVK*CHL1A and 1B) and/or both circulating pumps (2HVK*P1A and 1A), service water can be manually valved in to provide cooling directly to the ACU cooling coils.

Standby Switchgear/Battery Room Ventilation
The standby switchgear/battery room ventilation system supplies tempered, outdoor air to all areas of the 261', 237', and 214' elevations of the Control Building. The following areas are served by this subsystem:

- Division I, II, & III cable chase
- Division I, II, & III switchgear rooms
- Division I, II, & III battery rooms
- Remote shutdown rooms A & B
- Division I & II chiller equipment rooms
- Division I, II, & III cable areas
- 24V battery rooms
- Halon storage area

Two redundant air supply fans (2HVC*FN11A and FN11B) supply air to the 261' elevation corridor where it becomes makeup for the elevation 261' areas and rooms. This is shown on Figures 3.2.1.11-1A and B along with the switchgear room unit coolers described below.

Unit coolers 2HVC*UC101A and 108A circulate and provide cooling of air in the Division I switchgear and battery rooms, and unit coolers 2HVC*101B and 108B provide cooling to the Division II switchgear and battery rooms. The unit coolers depend on Division I and II AC power and service water. Unit cooler 2HVC*UC102 provides cooling to the Division III switchgear and cable areas. This cooler is powered from Division III AC power and depends on service water. Two redundant battery room exhaust fans (2HVC*FN4A and 4B) provide for exhausting air from the Division I, II and III battery rooms and switchgear rooms to the outside.

The basement cable spreading unit coolers (2HVC*UC106 and UC107) provide ventilation and cooling to the Division I and Division II basement areas, respectively (Figure 3.2.1.11-1C). Service water and Division I and II AC power are required for support.

The Division I and II chiller equipment rooms are each supplied with a unit cooler (2HVC*UC103A and 103B) for recirculation and cooling as required. The unit coolers depend on service water and Division I and II AC power.

Remote shutdown rooms A and B are supplied with individual air conditioning units (2HVC*ACU3A and ACU3B). Control Building chilled water normally supplies the cooling water to these air conditioning units with service water as a manual backup.

Two fans (2HVC-FN21A and FN21B) exhaust air from the 24V and computer battery rooms to the outside.

A fan (2HVC-FN6) is provided for smoke removal in the Division I, II, and III switchgear rooms, cable chases and Division III cable areas (Figure 3.2.1.11-1C). Two fans (2HVC-FN12 and FN14) provide smoke removal in the Division I and II basement areas, respectively.

3.2.1.11.1.2 Control Building Evaluation

The following Control Building areas are judged to be most important to supporting plant response and they contain the most sensitive electrical equipment:

- Control room (elevation 306')
- Relay room (elevation 288')
- Standby switchgear rooms (elevation 261')

<u>Control Room</u>
Loss of air conditioning to the Control Room would be detected early since the operators are in the Control Room and there are both trouble and inoperability alarms associated with the systems. Other cooling possibilities include forcing air through the Control Room with the special filter trains and/or smoke removal fans and/or opening doors and allowing air conditioning from the lower elevations to provide some cooling. Even if the Control Room became very hot and started to impact electrical equipment and systems, the operators still have the option of taking control at the remote shutdown rooms at elevation 261'. These rooms have their own safety related air conditioners. Given these capabilities and the presence of operators, the likelihood of ventilation failures leading to core damage is judged unlikely and is not modeled.

<u>Relay Room</u>

Loss of air conditioning to the relay room would be detected early since there are both trouble and inoperability alarms associated with the systems including high temperature in the room. Other cooling possibilities include forcing air through the relay room with the special filter trains and/or smoke removal fans and/or opening doors to the adjoining computer room (separate safety related air conditioning units) or upper and lower elevations to provide some cooling. Even if the relay room became very hot and started to impact electrical equipment and systems, the operators still have the option of taking control at the remote shutdown rooms at elevation 261'. These rooms have their own safety related air conditioners. Given these capabilities and the presence of operators on the next elevation, the likelihood of ventilation failures leading to core damage is judged unlikely and is not modeled.

Standby Switchgear Rooms

Loss of air conditioning to a standby switchgear room would be detected early since there are both trouble and inoperability alarms associated with the systems including high temperature in the room. In addition, loss of cooling water procedures (N2-OP-53E.H.2) instruct the operators to monitor the area temperature and establish means of temporary ventilation as necessary to control the area temperature. There is additional air conditioning in adjoining rooms and the higher elevations, therefore, opening doors is a potential recovery. There are supply fans to the elevation 261' corridor and a smoke removal fan, which could be another potential recovery. The design basis heat load is 180,000 BTU/hour.

The standby switchgear rooms are important because if the heat loads are high enough and recovery actions are not taken, unrecoverable failures of emergency AC is a possibility. However, metal clad switchgear and molded case breakers with electro-mechanical relays are not as sensitive to high temperature as molded case circuit breakers with thermal elements. The most sensitive equipment in the switchgear rooms are battery chargers and uninterruptible power supplies (UPS). The following failure impacts are expected:

- Charger failure would transfer the 125V DC bus to its battery supply.
- UPS failure would prevent transfer to DC power if AC power is lost.

The room ceilings are very high (27 feet), there are doors into a corridor from each end of the room, and there are doors to both the outside and into very high stairwells. Based on room design, which is large in comparison to heat load, it appears that there will be adequate time for operator diagnosis and there are several recovery operations. Therefore, loss of ventilation is judged to be a small contributor to core damage and is not modeled.

3.2.1.11.2 Emergency Diesel Building

Ventilation for the three emergency diesel generator rooms and their associated Control Rooms is provided by two separate systems, a normal system in operation when the diesels are not running and a standby system to support diesel operation. In addition, a unit cooler is provided for each of the three diesel control rooms. A simplified diagram of the ventilation systems is provided in Figure 3.2.1.11-2.

The normal ventilation system, exhaust fans 2HVP-FN3A, 3B & 4 and makeup air fan 2HVP-FN5, are not required to support emergency diesel operation.

The standby ventilation system in each diesel room is required to support diesel operation. Each system contains two 50% capacity (45,500 cfm) vane axial fans and associated dampers, including two inlet dampers as shown in Figure 3.2.1.11-2.

Failure of the standby ventilation system would heat up the room very quickly during diesel operation and therefore, should be included in the emergency diesel system models. One of two exhaust fans and one of two intake dampers will provide adequate heat removal from the emergency diesel rooms. Since redundancy exists, these failures would be small contributors to diesel unavailability in comparison to the diesel itself. This is discussed in detail in LER 50-410-87-39. For this reason, the ventilation failure model is included in the emergency diesel model.

Failure of the diesel control room unit cooler is assessed to cause a diesel failure. Temperature measurements have shown that there is approximately a 20°F temperature difference between the ambient temperature (77°F) and the temperature inside the Diesel Generator Control Panel. Without the cooling effect of the unit cooler, the ambient temperature will continue to rise and the temperatures within the control panels will continue to rise even more. The failure of the unit cooler is included in the emergency diesel model as a failure of the diesel controls.

3.2.1.11.3 Reactor Building

3.2.1.11.3.1 Reactor Building Ventilation Description

The following three modes of Reactor Building area cooling are described below:

- Normal ventilation system which supplies the general areas of the Reactor Building as shown in Figure 3.2.1.11-3.
- Emergency recirculation mode which initiates if the normal ventilation system fails (low air flow signal) or due to a high radiation signal or a LOCA signal.
- Local area unit coolers are provided throughout the Reactor Building.

The normal and emergency recirculation modes provide limited air cooling to the RCIC and HPCS rooms. Therefore, these two areas are assumed to be dependent on local unit coolers and are evaluated further below. The same is true for the North and South Auxiliary Bays which are evaluated in Sections 3.2.1.11.4 and 3.2.1.11.5.

<u>Normal Reactor Building Ventilation Mode</u>
Normally, the Reactor Building is cooled by supplying 140,000 cfm to various areas via a network of ductworks and ductwork accessories. Supply air is supplied through two of three, 50% vane axial fans (FN1A, 1B & 1C) while the third fan is in standby. The air passes

through a cooler (CLC2) prior to distribution in the Reactor Building. The spent air is then exhausted from two separate duct systems, one taking suction from above the refueling floor and the other from the duct system below the refueling floor. Each exhaust duct is equipped with two 100% vane axial fans (FN2A, 2B, 5A & 5B). Each fan is capable of passing 70,000 cfm. The power supplies for these fans are from normal 600V AC power sources (2NJS-US2) and the system isolates on any of the following:

- A high Reactor Building radiation signal,
- A low Reactor Building flow signal, or
- A LOCA signal.

As shown in Figure 3.2.1.11-3, air operated dampers (AOD1A, 1B, 9A, 9B, 10A, & 10B) isolate the normal ventilation system on these signals and fail closed on loss of support systems. In addition, these signals start the emergency recirculation ventilation mode.

Emergency Recirculation Mode

As described above, this mode of operation can occur due to a high Reactor Building radiation signal, a low Reactor Building air flow signal or a LOCA signal. These signals isolate the normal ventilation system and recirculates Reactor Building air through two unit coolers (2HVR*UC413A & 413B). These unit coolers depend on Division I and II Emergency AC power and Service Water, respectively.

Local Area Unit Coolers

Area cooling is provided by unit coolers at specific locations within the Reactor Building. These unit coolers are dependent on emergency AC and service water. Both Division I and Division II unit coolers are provided at each elevation. The unit coolers for each Reactor Building elevation area and the RCIC and HPCS pump rooms are summarized in Table 3.2.1.11-1. Forced flow through each unit cooler varies from 4400 cfm on elevations 215' and 240' to 7400 cfm on elevations 196' and 261'. Each unit cooler flow rate in the HPCS room is 27,200 cfm. Each unit cooler flow rate in the RCIC room is 7400 cfm. There are no unit coolers at elevation 175, however, there are four unit coolers at elevation 196 that provide adequate cooling.

3.2.1.11.3.2 Reactor Building Evaluation

Reactor Building General Areas

When normal ventilation is lost (emergency recirculation mode functions successfully and/or local unit coolers are successful), several areas can heat up to the point where manual operator actions are needed to prevent an excessive temperature and possible plant shutdown. Within 10 min, an operator must be dispatched to the US2 room (2NJS-US2) to open both doors so that the load center will not overheat. The Reactor Water Clean-Up (RWCU) pump room temperature should be monitored and temporary ventilation should be provided. The frequency and consequences of losing normal ventilation and not recovering this loss prior to a plant trip or shutdown is judged to be enveloped by other initiating events such as loss of normal AC power.

There are several high temperature alarms in the Control Room, one measuring the exit temperature at the recirculation unit coolers and the supply air temperature. Local temperature detectors are located at many areas in the Reactor Building. All safety related unit cooler motor trips are annunciated in the Control Room.

Failure of all general area unit coolers will not cause severe temperature conditions as long as one emergency recirculation cooler is available and operating. The emergency recirculation coolers (2HVR*UC413A or UC413B), will provide adequate cooling to the general Reactor Building areas. Failure of the general area unit coolers along with failure of the emergency recirculation coolers will not cause severe temperature conditions to occur. The normal ventilation system is judged to be adequate to cool components located in the general areas. Loss of both normal ventilation and emergency reactor building recirculation coolers will not cause a severe temperature condition if all unit coolers from either division are available. This is because the sum total forced flow of all unit coolers in each division is greater than 50,000 cfm. This is comparable to the 70,000 cfm from each emergency recirculation cooler.

Even if all cooling is lost to the general areas of the Reactor Building, the heat up rates are expected to be slow on the lower elevations where the more important sensitive equipment is located. The Reactor Building general area is a very large area with floor gratings that communicate to the upper elevations. Therefore, loss of all cooling in the Reactor Building general areas is not judged significant enough to model.

HPCS Room

HPCS Room cooling is judged to be lost if both area unit cooler, 2HVR*UC403A and 2HVR*UC403B, are inoperable. This is based the fact that the HPCS pump motor is a 3050 hp motor. This generates 470,000 btu/hr in the room. This heat load without cooling causes the HPCS motor and the general HPCS area to increase in temperature. This temperature increase affects the motor in several ways and the fire suppression features by simulating the environment that would be encountered during a fire.

As the suppression pool temperature increases to 212°F, the heat is added to the room by the suction piping at a rate of approx. 230,000 btu/hr. After the area temperature exceeds 212°F, the pipe becomes a heat sink and removes a significant amount of heat. A small amount of heat can also be removed from this enclosure if the Reactor Building Emergency Recirc. Cooler is operating. However the air flow is only 120 CFM, while a single HPCS unit cooler circulates 27,200 CFM of cooled air. This cooling feature has been neglected.

Preliminary calculations using data from HVR-32, the "Reactor Building & Aux Bay Heat Gain," and the methodology from EHV-10 Rev. 1, "Rate of Change of the Zone Temperature After an Abnormal Event," shows that the steady state room temperature would reach approx. 387°F if the motor continued to operate without cooling. The time constant is approximately 5.7 hrs. In 11.4 hrs, the room temperature would be approx. 346°F.

The only HPCS component that is directly affected by high temperature is the pump motor. Obviously the motor will be hotter than the room ambient temperature since it is generating all of the heat. Actual measurements have shown that the maximum winding temperature (hot spot) will be 80°C (144°F) above ambient temperature of the room.

If the motor were operated at these elevated temperatures for even a short time (several hours), the motor would catastrophically fail because of insulation damage resulting in an electrical fault. The insulation used in the motor windings is Class "F". This insulation can be operated at higher temperature than Class "A", "B" or "D". For class "F" insulation, the "hot spot temperature" can reach 150°C (302°F). This translates to a normally loaded motor operating in an ambient temperature of 75°C (165°F).

The circuit breaker for the HPCS pump motor will trip on overload if the motor temperature increases. The time that the motor will trip at is dependent on the initial motor heatup and the setting at which the overload relays will trip. For the HPCS pump, 2CSH*P1, the relay will trip at 127% of full load current. The resistance of the motor windings (stator and rotor) is an increasing function of temperature, it increases linearly with increasing temperature. A typical an increase in resistance occurs at a rate of 10% / 47°F increase in ambient temperature. As temperature increases, rotor resistance increases, current decreases reducing rotor power and developed torque (torque is proportional to current squared). A 10% decrease in rotor current causes a 29% decrease in torque. This process continues until the developed torque equals the required torque. Once this point is reached, an increase in temperature results in an increase in rotor slip, (the rotor slows down to develop the needed torque). This increase in slip causes an increase in current. This in turn heats up the motor and reduces cooling.

Insulation resistance decreases on increasing temperature. As the motor windings increase in temperature, the frequency of insulation failure leading to an electrical fault increases. This effect, when added to the winding resistance increase, promotes increased insulation degradation. This process continues until the motor trips out on either overload or electrical fault due to an insulation failure.

The manufacturer's data sheet recommends that the motor be shutoff if the stator temperature reaches 160°C (339°F) or the bearing temperature reaches 100°C. This stator temperature will normally be reached if the ambient temperature reaches 85°C (180°F).

The high temperatures that can occur with loss of area cooling will result in closure of the fire dampers for unit cooler 2HVR*UC403B and loss of exhaust air via normal or recirculation paths. These dampers are designed to close when the area temperature reaches approx 170°F. This will complicate restoration of cooling at a later time if recovery is contemplated.

<u>RCIC Room</u>

RCIC Room cooling is judged to be acceptable even if both RCIC enclosure unit coolers, 2HVR*UC412A and 2HVR*UC412B, are inoperable. This is based on the condition that the operators bypass the high temperature trips and open the water tight door to allow hot air to circulate via the top of the door while cooler air is induced through the lower part of the open door. The max heat load from the RCIC turbine and piping is less than 40,000 btu/hr.

The most temperature sensitive items in this area are the governor and it's electronic control box, the EGM box. The lube oil used for the turbine bearings is also temperature sensitive. If the EGM circuit fails to a zero output signal then RCIC flow control goes to full flow. The governor uses turbine lube oil as it's hydraulic fluid. Because it is cooled by recirculating RCIC fluid through the oil cooler, it is not affected by loss of area cooling. Preliminary calculations show that the max temperature is approx 211°F with a time constant of 3.15 hrs (no credit for operator action to re-establish cooling). Other components that must remain operable are the steam admission valves, 2ICS*MOV159 and 2ICS*MOV120, and lube oil cooling valve 2ICS*PCV115. It is expected that these valves will survive this environment, especially since the operator has been alerted to the problem. The above operator action also ensures that the fire protection features are not actuated.

3.2.1.11.4 North Auxiliary Bay

3.2.1.11.4.1 North Auxiliary Bay Ventilation Description

The North Auxiliary Bay contains the following six HVAC designated zones:

Zone 1	RBCLC heat exchanger area
Zone 2	LPCS pump room
Zone 3	RHR pump A room
Zone 4	RHR heat exchanger room
Zone 60	Electric cable tray area
Zone 20	Electric MCC area

Zone 1 is cooled by supplying 660 cfm of outside air and removing it via the exhaust duct system. There is no unit cooler in this zone.

Zone 2 has two unit coolers each rated at 10,000 cfm, 2HVR*UC402A & 402B. Outside air is supplied via the normal vent system at a rate of 250 cfm.

Zone 3 has two unit coolers each rated at 15,600 cfm, 2HVR*UC401A & 401D. Outside air is supplied via the normal vent system at a rate of 250 cfm.

Zone 4 has one unit cooler, rated at 15,600 cfm, 2HVR*UC405. Outside air is not supplied to this area.

Zone 60 is cooled by supplying 500 cfm of outside air and removing it via the exhaust duct system. There is no unit cooler in this zone.

Zone 20 has two unit coolers each rated at 10,000 cfm, 2HVR*UC408A & 408B. Outside air is not supplied to this area.

3.2.1.11.4.2 North Auxiliary Bay Evaluation

The North Aux. Bay MCC area was evaluated for the effects of loss of area cooling. The ^Eredundant unit coolers are the only source of cooling for this area. Both unit coolers are powered from the same electrical bus. The north bay MCC area was chosen because it has a slightly higher heat load than the south bay MCC area. Preliminary analysis indicates that on loss of area cooling, the area temperature will increase to 145°F with a time constant of 1.8 hrs. The MCC area has numerous electrical devices (thermal magnetic breakers and motor overloads) that need to operate to support some of the ECCS systems. In a molded case circuit breaker there are two electrical sensitive elements, a thermal element which senses an overload condition and a magnetic element which senses a short circuit.

The thermal element which consists of a heater and a temperature sensitive switch, can be affected by a small increase in ambient temperature. For one specific breaker, the TEB 240V, an increase of 10°C in ambient temperature can cause a shift in setpoint that can range from 9% in open air to 28% in large switch panels. Information obtained from industry sources suggests a time related shift in trip setpoint also. This shift can be as much as 20% over ten (10) years and can be in either direction. These combined effects can result in either spurious tripping or loss of electrical coordination. The sequence of events cannot be predicted because it depends on several factors. One breaker may trip while another identical breaker does not trip. The reason for this is that the margin between the full load current and the long time trip setting is different for identical breakers. Also the proximity to equipment carrying large loads can cause the local temperature to increase disproportionately.

The setpoints at which magnetic elements trip are not temperature sensitive in the sense that a temperature increase will not cause a premature trip.

Thermal overloads used to protect motors from overload are affected by an increase in ambient temperature. The setpoint is normally set at 125% full load. Setpoint drift also occurs over time and this causes premature depowering. For safety related applications, the thermal overloads are either an alarm only or the overload trip is bypassed on an accident signal. For applications in areas where the ambient temperatures can swing over 40°C, temperature compensated thermal elements are employed. However, temperature compensation is used only in areas where the temperature is expected to routinely exceed 40°C.

In view of the above considerations, all electrical equipment that has a thermal element in a Molded Case Breaker (MCB) or an uncompensated or unbypassed thermal overload is postulated to fail if cooling is lost for 9.0 hrs or more.

Failure of the MCC area coolers, 2HVR*UC408A and 2HVR*UC408B, results in the following:

- RCIC must be in operation at t = 9 hr for it to be credited for times greater than t = 9 hr to 24 hr. This is because the power source for the RCIC steam admission valve and bypass valves are powered by 2DMS*MCCA1 which is considered lost at t > 9 hrs. It is assumed that RCIC will be in operation.

- LPCS and RHR "A" systems must be in operation at t = 9 hrs if they are to be credited for later times. Failure of the power source 2EHS*MCC102 (600V AC) causes loss of powered operation. However, if the necessary MOVs are already positioned for injection/cooling, then the loss of 600V AC will not result in valve closure or repositioning.

- The standby liquid control pump 2SLS*P1A is also assumed to fail after a period of 9 hours without HVAC to the MCC area. This pump is only required to operate for 2 hours.

Based on the above long-term impacts, loss of cooling to the MCC rooms could be neglected.

Low Pressure Core Spray Room
Failure of area unit coolers 2HVR*UC402A and 2HVR*UC402B, would eventually cause failure of the LPCS pump, if recovery actions are not taken. This failure process is similar to that described in the HPCS area analysis. Analysis of the area heatup on loss of all HVAC has results similar to those for the HPCS room. The heatup in this room is less severe than the heatup in RHR A pump room and pump failure is conservatively assessed to be the same as for the RHR A pump.

RHR A Pump Room
Failure of area unit coolers, 2HVR*UC401A and 2HVR*UC401D, would eventually cause failure of the RHR A pump, if recovery actions are not taken.

Analysis of the heatup of this room as a result of loss of area cooling with the RHR A pump operating results in high area temperature. This temperature was calculated to reach a steady state value of 440°F. The method used and data gathered was the same as that used in the HPCS area heatup analysis. The time constant was calculated to be 2.3 hours.

The maximum temperature will not be reached because the fire suppression system will be automatically initiated when the area temperature heats up. The sprinkler leads fuse at 186°F. Once the water sprays start, the area temperature will decrease. Water droplets and vapor will be pulled into the hot motor and this will place a severe stress, thermal and resistive, on the insulation system of the motor stator. As insulation resistance decreases or even fails, phase-to-phase or phase-to-ground leakage or fault currents increase and the motor will trip on over current.

Based on the above evaluation, it takes a failure of redundant unit coolers in each pump area to fail a single pump, however, failure of redundant unit coolers in the MCC area could lead to failure of both systems. As described above, this is conservative if the system is aligned properly within 5 hours. Rather than evaluate the pump rooms at this time, the Division I MCC area cooling system will be included in the support system event tree. Failure of this system is modeled as failure of LPCS and RHR "A" systems. RCIC and SLC are assumed to be operable prior to the 5 hours.

3.2.1.11.5 South Auxiliary Bay

3.2.1.11.5.1 South Auxiliary Bay Ventilation Description

The South Auxiliary Bay contains the following five HVAC zones:

Zone 11	RHR Heat Exchanger Room B
Zone 12	RHR Pump Room B
Zone 13	RHR Pump Room C
Zone 61	Electric cable tray area
Zone 21	Electric MCC area

Zone 11 has one unit cooler, rated at 15,600 cfm, 2HVR*UC405. Outside air is not supplied to this area.

Zone 12 has two unit coolers each rated at 10,000 cfm, 2HVR*UC401C & 401E. Outside air is supplied via the normal vent system at a rate of 250 cfm.

Zone 13 has two unit coolers each rated at 10,000 cfm, 2HVR*UC401CB & 401F. Outside air is supplied via the normal vent system at a rate of 250 cfm.

Zone 61 is cooled by supplying 500 cfm of outside air and removing it via the exhaust duct system. There is no unit cooler in this zone.

Zone 21 has two unit coolers each rated at 10000 cfm, 2HVR*UC409A & 409B. Outside air is not supplied to this area.

3.2.1.11.5.2 South Auxiliary Bay Evaluation

The most important areas regarding impact on the IPE model include the RHR pump "B" area, RHR pump "C" area and MCC area. The major heat loads in these areas during RHR B and RHR C pump operation include the following:

Room	Heat Load	Volume (ft^3)
RHR Pump B	231,128 btu/hr	20,830
RHR Pump C	240,338 btu/hr	20,830
RHR Heat Exchanger B	33,546 btu/hr	27,360
MCC Area	99,484 btu/hr	62,300

See the description of MCC electrical heat up analysis in Section 3.2.1.11.4.2. Failure of the MCC area coolers 2HVR*UC409A and 2HVR*UC409B results in the following:

> RHR B and RHR C - these systems are considered failed if they are not properly aligned for injection at times later than t = 5 hrs. If they are properly aligned, failure of the associated MCC, 2EHS*MCC302 (600V AC), has no effect. The 4kV breakers needed for the RHR pumps B and C are not affected by failure of the above MCC or 2DMS*MCCB1 (125V DC). This failure will cause loss of operability of the RHR valves. However, since the valves are already positioned there is no effect.

<u>RHR B Pump Room</u>
Failure of area unit coolers, 2HVR*UC401C and 2HVR*UC401F, would eventually cause failure of the RHR B pump, if recovery actions are not taken. The analysis of the heatup in this room is enveloped by the analysis for RHR A pump room.

<u>RHR C Pump Room</u>
Failure of area unit coolers, 2HVR*UC401B and 2HVR*UC401E, would eventually cause failure of the RHR C pump, if recovery actions are not taken. The analysis of the heatup in this room is enveloped by the analysis of RHR A pump room.

Based on the above evaluation, it takes a failure of redundant unit coolers in each pump area to fail a single pump. Failure of redundant unit coolers in the MCC area could lead to failure of both systems. If the system is not aligned properly, within 5 hrs, the Division II MCC area cooling system will be included in the support system event tree. Failure of this system is conservatively modeled as failure of the RHR "B" and RHR "C" systems.

3.2.1.11.6 Service Water Pump Bay

There are two independent service water bays with the Division I pumps located in one bay and the Division II pumps in the other bay. Failure of the unit coolers in the respective screenwells 2HVY*UC2A & C (for Division I) and 2HVY*UC2B & D results in loss of area cooling to the above areas. The service water pumps are located at the lowest point at elevation 244'. In addition to these pumps, each service water discharge line has a revolving screen strainer powered from the respective 600V AC motor control center, 2EHS*MCC101 and 2EHS*MCC301, located at elevation 263' in each screenhouse. Service water system isolation valves are also powered by these MCCs.

Loss of area cooling causes area temperatures to rise especially in the upper region of the building (elevations 263' to 275'). 2EHS*MCC101 and 2EHS*MCC301 contain motor starters with thermal elements and motor overloads. Both of these components are eventually expected to actuate prematurely resulting in loss of the strainer backwash and loss of power operation to various MOVs. However, these MOVs can be manually operated if the need arises. Loss of strainer backwash is expected to eventually result in a flow blockage resulting in a total loss of service water. The length of time to flow blockage is highly dependent on the lake water condition and time of year.

The service water pumps are located at elevation 244'. Since they pump lake water, the pipes, pumps and adjoining steal are kept cool by conduction. The loss of HVAC is not expected to cause loss of any service water function within the first 24 hours following an event. Even the failure scenario that causes loss of all strainers will result in an alarm situation. This alarm initiates on loss of control circuit voltage. Once alerted to this loss of HVAC, the plant operators can either restore cooling or open a door at elevation 279'. This will allow any trapped hot air at elevation 263' to vent out to the higher elevations of the screenhouse, thus cooling the MCCs. The strainer motors can be restored and operability can be restored to the affected MOVs.

3.2.1.11.7 Fire Pump Rooms

The electric driven fire pump and its unit cooler require normal AC power for success. Since this pump is not significant to the PRA results and a simplified model is used, the cooling dependency is not considered a significant contributor, and is not modeled.

The diesel driven fire pump is important to the station blackout model and room fans depend on normal AC power. Therefore, the operators are assumed to be required to open doors into the diesel fire pump room during a station blackout.

3.2.1.11.8 References

LER 50-410-87-39	Licensee Event Report, 07/30/87
N2-OP-52, Rev. 3:	Reactor Building Ventilation
N2-OP-53A, Rev. 5:	Control Building Ventilation
N2-OP-53E, Rev. 2:	Standby Switchgear/Battery Room Ventilation
FSAR Section 9.4	

HVAC Heat Up Calculations:

HVR-28, Rev. 0:	Secondary Containment Gross Volume
HVR-32, Rev. 2:	Reactor Building
HVC-58, Rev. 2:	Control & Relay Rooms
HVC-62, Rev. 2:	Control & Relay Rooms
HVC-64, Rev. 2:	Switchgear Rooms (elev. 261')
EC-26, Rev. 4:	Control Building, elevations 237' & 244'
EC-28, Rev. 4:	Switchgear Rooms (elev. 261')

Table 3.2.1.11-1
VENTILATION SCREENING

Building	Location or Room	HVAC System	Ventilation Equipment	Support Systems	Comments	Screening
Control Building	General	2HVC	*AOD54A; *AOD54B; *FN11A; *FN11B; *FLT3A; *FLT3B	EJS, IAS	to stairwell, cable chases, corridors; air from the 261' corridors is used as make-up for elev. 261' areas/rooms	Important area requiring further evaluation.
	Kitchen	2HVC	-FN7	SCA		This area does not support safety equipment in the IPE model.
	Lavatories	2HVC	-FN1	SCA		This area does not support safety equipment in the IPE model.
	Make-up to Elev. 288'6" & 306'	2HVC	-HVU1; *AOD142	NHS, IAS	smoke removal mode, cannot be used if airborne radiation hazard is present	Not modeled.
	Discharge, Elev. 288'6" & 306'	2HVC	*AOD148; *AOD145; *AOD120; *AOD117; -FN3	NHS, IAS	smoke removal mode, cannot be used if airborne radiation hazard is present	Not modeled.
	Special Filter Trains	2HVC	*AOD61A; *AOD61B; *MOV1A; *MOV1B; *FLT2A; *FLT2B; *FN2A; *FN2B; *DMP1A; *DMP1B; *RE18A; *RE18B; *RE18C; *RE18D	EJS, IAS	The special filter trains are normally bypassed. On a rad monitor trip signal from *RE18A & C (18B & D) or a Div I (Div II) LOCA Signal, *MOV1A (MOV1B) will close and divert air to special filter train FLT2A (FLT2B).	Not modeled.
	Control Building Chilled Water	2HVK	*CHL1A; *CHL1B; *P1A; *P1B; *SOV36A; *SOV36B; *TK1A; *TK1B	EJS, EHS, EHA, SCM, SWP	HVK provides cooling water to: *ACU1A & B (Control Room), *ACU2A & B (relay room), -ACU4A & B (computer room), and *ACU3A & B (remote shutdown room). If HVK fails, service water can be manually valved in to provide cooling directly to the ACUs' cooling coils.	Important system requiring further evaluation.
	Elevation 288'6"	2HVC	Special Filter Trains (above); Control Building Chilled Water (above); Make-up to/Discharge from Elev. 288' & 306'6" (above); *ACU2A; *ACU2B; -ACU4A; -ACU4B; -FN17; -FN18; *AOD12A; *AOD12B; -AOD122; -AOD176	EHS, SCA, IAS, HVK	elevation 288'6" has: relay room, computer room, cable chases, HVAC equipment rooms & instrument shop -- temperature in Relay Room and computer room must be maintained below 140°F	Important area requiring further evaluation.

Building	Location or Room	HVAC System	Ventilation Equipment	Support Systems	Comments	Screening
	Elevation 306'0"	2HVC	Special Filter Trains (above); Control Building Chilled Water (above); Make-up to/Discharge from Elev. 288' & 306'6" (above); *FLT1A; *FLT1B; *ACU1A; *ACU1B; *AOO6A; *AOO6B	EHS, IAS, HVK	elevation 306'0" has: Control Room, cable chases, HVAC equipment rooms & instrument shop -- temperature in Control Room must be maintained below 140°F	Important area requiring further evaluation.
	Remote Shutdown Rooms	2HVC	Control Building Chilled Water (above); *ACU3A; *ACU3B	EHS, HVK	A temperature switch starts/stops the ACUs as required.	Important area requiring further evaluation.
	Records Storage Vault	2HVC	-ACU5		A thermostat switch starts/stops the ACU as required.	This area does not support safety equipment in the IPE model.
	Div 1 & 2 Stby Swgr/Bat Rooms	2HVC	Control Building elev. 261' (above); *UC101A; *UC101B; *UC108A; *UC108B	EJS, SWP	Unit coolers are normally running with service water supplied to the cooling coils as required -- temperature indicating switches.	Important area requiring further evaluation.
	Div 1, 2 & 3 Battery Rooms	2HVC	Control Building elev. 261' (above); *FN4A; *FN4B	EJS	exhaust for battery rooms, one fan normally running with the other in auto	Important area requiring further evaluation.
	Div 1, 2 & 3 Cable Areas	2HVC	Control Building elev. 261' (above); *UC102; *UC106; *UC107	EJS, SWP	Unit coolers are normally running with service water supplied to the cooling coils as required -- temperature indicating switches.	Important area requiring further evaluation.
	24V & Computer Battery Rooms	2HVC	Control Building elev. 261' (above); -FN21A; -FN21B	NHS	exhaust for battery rooms, one fan normally running with the other in auto	This area does not support safety equipment in the IPE model.
	Div 1 & 2 Chiller Equip. Rooms	2HVC	Control Building elev. 261' (above); *UC103A; *UC103B; -FN8	EHS, NHS, SWP	A temperature switch starts/stops the ACUs as required. The fan exhausts the rooms and trips in the smoke removal mode.	Important area requiring further evaluation.
	Smoke Removal	2HVC	Control Building (above); -FN6; -FN12; -FN14	NHS	-FN6 exhausts: the Div 1,2 & 3 switchgear rooms and cable chases and the Div 3 cable areas -- the fan will stop on CO^2 discharge to these areas. -FN12 & 14 exhaust the Div 1 & 2 cable areas -- associated UCs must be stopped before starting the fans.	Not modeled.

Building	Location or Room	HVAC System	Ventilation Equipment	Support Systems	Comments	Screening
Normal Switchgear Building	Ventilation Equip Penthouse	2HVC	-FN20A; -FN20B; -FN20C	NHS	The fans (FN20A/B/C) start when the temperature in the penthouse reaches 85°F. They take suction from outside, discharge to a common plenum -- the air is distributed to the penthouse via 6 dampers. Air is discharged from the penthouse via louvers at the east and west ends of the penthouse.	Ventilation failures leading to loss of Normal AC are assumed to be enveloped or contained in the loss of Normal AC frequency.
	General Building Ventilation	2HVC	-CLC1; -FN13A; -FN13B; -FN13C; -FN19A; -FN19B; -FN19C; -AO0218; -AO0226; -AO0219	NHS; IAS	Normally, 2 of 3 fans are in operation on both the supply and the discharge side of the system, with the third fan in auto. The cooling coils (CLC1) on the intake filter are cooled by chilled water (HVN).	Ventilation failures leading to loss of Normal AC are assumed to be enveloped or contained in the loss of Normal AC frequency.
	Battery Rooms Exhaust	2HVC	-FN15A; -FN15B; -DMP62A; -DMP62B	NHS	Either of the 2 100% fans (FN15A/B) is used to keep atmospheric pressure slightly lower in the battery rooms than in adjacent areas, to prevent exfiltration of hydrogen. The fans discharge through a common vent to the atmosphere.	Ventilation failures leading to loss of Normal AC are assumed to be enveloped or contained in the loss of Normal AC frequency.
	LF MG Set Penthouse	2HVC	-FN16A; -FN16B	NHS	The LFMG ventilation system consists of two supply fans (FN16A/B) which draw outside air through two air louvers and discharge directly over the LFMG sets. The fans start when the room temperature reaches 85°F. Air exhausts outdoor from the penthouse via two air louvers.	Ventilation failures leading to loss of Normal AC are assumed to be enveloped or contained in the loss of Normal AC frequency.
CB-RB Electrical Tunnels		2HVN	Control Building (above); Reactor Building; -FN9; -FN10; *UC104; *UC105	NHS, EJS, SWP	HVN provides cooling an smoke removal for the North & South Electrical Tunnels. The unit coolers provide cooling and ventilation. The fans provide smoke removal; the UCs MUST be shutdown in smoke removal mode because smoke removal is achieved by reverse flow through normal air supply ducts.	The heat loads are not high and cables are assumed to have high capacities.

Building	Location or Room	HVAC System	Ventilation Equipment	Support Systems	Comments	Screening
Service Building Area	General	2HVE	-FN1, -FN2, -FN3, -FN4, -CH1, -UHE501, -UHE502, -UHE503, -UHE506, -UHE507, -UHE508, -UHE509, -UHE510, -UHE511, -UHE512, -UHE513, -UHE514, -UHE515, -UHE516, -UHE517	NHS, NJS	HVE maintains temperatures between 65° & 85°F in the following areas: service room, foam room, service building entrance corridor, service building access passageway and the valve pit areas.	These areas do not support safety equipment in the IPE model.
Auxiliary Boiler Room	General	2HVI	-HVU1A; -HVU1B; -FN2A; -FN2B; UHE701; UHE702; UHE703; UHE704	NHS, NJS	HVI is used to maintain room temperature between 65° & 85°F in the auxiliary boiler room. Normally, one supply and one exhaust fan are operating with the other equipment in standby.	This area does not support safety equipment in the IPE model.
Aux. Service Bldg. South		2HVL	-ACU1, -FN1, -FN2, -FN4	NJS, SCA, NHS, SWP	Mixed outside air and recirculated air is tempered by -ACU1 and circulated throughout the building. Air is exhausted to the outdoors by -FN2 and potentially contaminated air is exhausted to the HVR system via -FN4. -FN1 exhausts the carbon dioxide tank room when temperatures exceed 85°F.	This area does not support safety equipment in the IPE model.
Diesel Generator Building	Common Makeup Air Assembly	2HVP	-FN5, -CH1	NHS, NJS	The common makeup air assembly heats and filters outside air. FN5 supplies the air to the building corridors where it is drawn into the diesel control rooms and then the diesel generator rooms by the normal exhaust fans (below).	Not required for emergency diesel operation.
	Div 1 EDG/EDG Control Rooms	2HVP	Common Makeup Air Assembly (above), *FN1A, *FN1C, -FN3A, *UC1A	EHS, NHS, NJS, EJS, SWP	Normal exhaust fans (FN3A & B, FN4) provide cooling when the diesel generators are NOT running (above). When an EDG starts, the associated fan stops and must be manually restarted when the EDG stops.	Important area requiring further evaluation.
	Div 2 EDG/EDG Control Rooms	2HVP	Common Makeup Air Assembly (above), *FN1B, *FN1D, -FN3B, *UC1B	EHS, NHS, NJS, EJS, SWP	Standby exhaust fans (FN1A,B,C,D & 2A,B) maintain an air temperature of approximately 80°F in the diesel generator room(s) using outside air. When an EDG starts, the associated fans start and must be manually stopped when the EDG stops.	Important area requiring further evaluation.

Building	Location or Room	HVAC System	Ventilation Equipment	Support Systems	Comments	Screening
	Div 3 EDG/EDG Control Rooms	2HVP	Common Makeup Air Assembly (above), *FN2A, *FN2B, -FN4, *UC2	EHS, NJS, NHS, SWP	The unit coolers are used to keep diesel generator control room temperatures below 140°F.	Important area requiring further evaluation.
Reactor Building	General	2HVR	-CLC2, -FN1A, -FN1B, -FN1C, -FN2A, -FN2B, -FN5A, -FN5B, -FN6, -AOD1A, -AOD1B, -AOD9A, -AOD9B, -AOD10A, -AOD10B	NHS, SWP, IAS	Supply Unit Cooling Coil (-CLC2), Supply Fans (-FN1A, B & C), Exhaust Fans to Atmosphere (-FN2A & B below refuel floor, -FN5A & B at/above refuel floor), Reactor Head Evacuation Fan (-FN6), Reactor Building Isolation Dampers close	Important area requiring further evaluation.
	Emergency Recirculation UCs	2HVR	*UC413A, *UC413B	EJS, SWP	UC413A or B will auto-initiate on the following: 1) RB HIGH radiation; 2) RB LOW air flow; 3) LOCA (high drywell pressure or RPV Level 2). These signals also close RB isolation dampers -AOD1A/B, -AOD9A/B AND -AOD10A/B.	Important area requiring further evaluation.
	General, elevation 175′ & 196′	2HVR	*UC404A, *UC404B, *UC404C, *UC404D	EJS, SWP	major equipment on elev. 175′: HPCS, LPCS, RHS & RCIC instrument racks; RBCLC heat exchanger; seismic instrumentation; RB/DW equip./floor drain pumps/sumps	Important area requiring further evaluation.
	General, elevation 215′	2HVR	*UC407A, *UC407B, *UC407C, *UC407D, *UC407E	EJS, SWP	major equipment on elev. 215′: Recirc pump, Main steam flow & jet pump instrument racks; SFPC heat exchangers; Control rod drive pumps & filters; 2NHS-MCC005; 2NHS-MCC014; suppression chamber access hatch	Important area requiring further evaluation.
	General, elevation 240′	2HVR	*UC410A, *UC410B, *UC410C	EJS, SWP	major equipment on elev. 240′: hydrogen recombiners; SRM/IRM pre-amp racks; TIP purge equipment; TIP drive mechanisms; TIP indexing mechanisms	Important area requiring further evaluation.
	General, elevation 261′	2HVR	*UC411A, *UC411B, *UC411C, *UC414A, *UC414B	EJS, SWP	major equipment on elev. 261′: RPV level & pressure instrument racks (4); RDS & RPV temp. recorder; RDS hydraulic units; RDS master control station; RDS hyd. power unit accumulators; 2NHS-MCC012; 2NHS-MCC011; equipment hatch; equip. & personnel hatch; hydrogen cooler	Important area requiring further evaluation.

Building	Location or Room	HVAC System	Ventilation Equipment	Support Systems	Comments	Screening
	Drywell Cooling	2HVR	2DRS-UC1A, -UC1B, -UC1C, -UC1D, 2DRS-UC2A, -UC2B, -UC2C, -UC2D, 2DRS-UC3A, -UC3B		Presently not included in the NMP2 model.	Not modeled.
	Primary Containment Purge	2HVR	2CPS-FN1		Presently not included in the NMP2 model.	Not modeled.
	HPCS Pump Room	2HVR	Reactor Building (above), Emergency Recirculation UCs, *UC403A, *UC403B	EHS, SWP	major equipment: 2CSH*P1; 2CHS*P2	Important area requiring further evaluation.
	RCIC Room	2HVR	Reactor Building, Emergency Recirculation UCs, *UC412A, *UC412B	EJS, SWP	major equipment: 2ICS*P1; 2ICS*P2; 2ICS*T1; 2ICS*C1	Important area requiring further evaluation.
North Aux Bay	LPCS Pump Room	2HVR	Reactor Building, Emergency Recirculation UCs, *UC402A, *UC402B	EHS, SWP	major equipment: 2CSL*P1; 2CSL*P2	Important area requiring further evaluation.
	RHR Pump Room A	2HVR	Reactor Building, Emergency Recirculation UCs, *UC401A, *UC401D	EJS, SWP	major equipment: 2RHS*P1A	Important area requiring further evaluation.
	RHR Heat Exchanger Room A	2HVR	Reactor Building, Emergency Recirculation UCs, *UC405	EJS, SWP	major equipment: 2RHS*E1A	Important area requiring further evaluation.
	North MCC Area	2HVR	Reactor Building, Emergency Recirculation UCs, *UC408A, *UC408B	EJS, SWP	major equipment: 2EDA*XD100A; 2SCV*XD100A; 2LAR-XLN15; 2EPS*SWG001; 2EPS*SWG002; 2EHS*MCC102; 2DHS*MCCA1; 2EJA*PNL100A; 2EJS*PNL101A; 2EJS*PNL103A; 2EJS*PNL104A; 2VBS*PNL103A; 2VBS*PNLA103; 2VBS*PNLB105; 2SCV*PNL101A; 2MSS*IPNL91A thru D; 2HCS*PNL22A; 2CHS*PNL66A; 2CHS*PNL73A	Important area requiring further evaluation.
South Aux Bay	RHR Pump Room B	2HVR	Reactor Building, Emergency Recirculation UCs, *UC401C, *UC401F	EJS, SWP	major equipment: 2RHS*P1B	Important area requiring further evaluation.
	RHR Heat Exchanger Room B	2HVR	Reactor Building, Emergency Recirculation UCs, *UC406	EJS, SWP	major equipment: 2RHS*E1B	Important area requiring further evaluation.

Building	Location or Room	HVAC System	Ventilation Equipment	Support Systems	Comments	Screening
	RHR Pump Room C	2HVR	Reactor Building, Emergency Recirculation UCs, *UC401B, *UC401E	EJS, SWP	major equipment: 2RHS*P1C, 2RHS*P2	Important area requiring further evaluation.
	South MCC Area	2HVR	Reactor Building, Emergency Recirculation UCs, *UC409A, *UC409B	EJS, SWP	major equipment: 2EJA*XD301B; 2SCV*XD301B; 2LAR-XLN16; 2EPS*SWG003; 2EPS*SWG004; 2EHS*MCC302; 2DMS*MCCB1; 2EJA*PNL300B; 2EJS*PNL302B; 2EJS*PNL303B; 2EJS*PNL304B; 2VBS*PNLA106; 2VBS*PNLB106; 2SCV*PNL301B; 2HSS*IPNL90A thru D; 2HCS*PNL22B; 2CMS*PNL66B; 2CMS*PNL73B	Important area requiring further evaluation.
GTS Building	General	2HVR	Reactor Building, Emergency Recirculation UCs, *UC415A, *UC415B, -FN7, -FN8	EJS, SWP, NHS	major equipment: HVR supply equipment; HVR chillers; HVR filters; 2NJS-PNL733; 2CES-IPNL101; 2SCI-PNLC104; Glycol heat exchanger/tanks; GTS filters; GTS fans	This area does not support safety equipment in the IPE model.
Turbine Building	General	2HVT	-FN1A, -FN1B, -FN1C, -FN2A, -FN2B, -FN2C, -FN10A, -FN10B, -ACUS2, -ACUS4, -UC204, -UC224, -UC225, -UC226, -CLC1	NJS, NHS, SWP	Supply Fans (-FN1A/B/C) 2 operate, 1 standby; Exhaust to Stack (-FN2A/B/C) 2 operate, 1 standby; Stack ventilation (-FN10A/B) 1 operates, 1 standby	The turbine building is a large area. Ventilation failures leading to plant trip and additional balance of plant system failures is assumed to be enveloped by other initiating events such as loss of normal AC.
	Misc. Local Area Ventilation	2HVT	Turbine Building, General (above); -ACUS1; -ACUS2; -ACUS4; -UC206A; -FN8; -UC205; -UC219; -UC220; -UC221A; -UC221B; -UC223A; -UC223B; -FN5; -FN6; -FN7	TBCLC, SWP, NHS, NJS, SCA	Charcoal Decay Bed Room: -ACUS1; T.B. Sample Room: -ACUS2; HVAC Exh. Equip. Area: -ACUS4; 2NJS-US10 Load Center Room: -UC206A, -FN8; Offgas Area: -UC205; Clean Steam Reboiler Room: -UC219, -UC220; Moisture Sep./Reheat Area: -UC221A/B, -UC223A/B; Elevator Rooms: -FN5/6/7	
	Main Steam Tunnel	2HVT	Turbine Building (above), -FN11, -UC210A, -UC210B, -UC211	NHS, NJS, SWP	Failure could cause a MSIV closure.	
	Generator Area	2HVT	Turbine Building (above), -UC217A, -UC217B, -UC222A, -UC222B, -UC222C, -UC222D, -UC222E, -UC222F	NJS, SWP		

Building	Location or Room	HVAC System	Ventilation Equipment	Support Systems	Comments	Screening
	Heater Bays	2HVT	Turbine Building (above), -UC203A, -UC203B, -UC215A, -UC215B, -UC202A, -UC202B, -UC214A, -UC214B, -UC201A, -UC201B, -UC213A, -UC213B	NJS, SWP	"A" Bay: -UC203A/B, -UC215A/B; "B" Bay: -UC202A/B, -UC214A/B; "C" Bay: -UC201A/B, -UC213A/B	
	Feed Pump Area	2HVT	Turbine Building (above), -UC206A, -UC206B, -UC206C, -UC206D, -UC206E, -UC206F	NJS, SWP		
	North Area of Condenser	2HVT	Turbine Building (above), -UC216A, -UC216B, -UC216C, -UC216D, -UC216E	NJS, SWP		
	Condenser Tube Removal Area	2HVT	Turbine Building (above), -UC212A, -UC212B, -UC218A, -UC218B, -UC218C, -UC218D, -UC218E,	NJS, SWP		
	Condensate Pump Area	2HVT	Turbine Building (above), -UC207A, -UC207B, -UC208A, -UC208B, -UC209A, -UC209B	NJS, SWP		
Radwaste Building	General	2HVW	-FN1A, -FN1B, -FN2A, -FN2B, -FN3A, -FN3B, -FN4, -FN5, -FN10A, -FN10B, -FN11A, -FN11B, -FN12A, -FN12B, -FN13, -ACU1A, -ACU1B, -ACU2A, -ACU2B, -CND1A, -CND1B, -CND2A, -CND2B	NJS, NHS, SCA, VBS	HVW uses a "once-through" system to maintain temp. between 65° & 110°F with these sub-systems: supply air, general area exhaust, equipment exhaust, unit heaters, smoke exhaust, liner filling hood exhaust, PASS equip. exhaust, Radwaste Cntl Rm ventilation & decontamination general area ventilation.	This area does not support safety equipment in the IPE model.
Screenwell Building	General	2HVY	-FN3A, -FN3B, -FN9A, -FN9B, -FN9C, -FN10	NHS, NJS	Supply fans (-FN9A/B/C) supply outside air to the ventilation system, the exhaust fans (-FN3A/B) exhaust the used air to the atmosphere. -FN10 provides air from the ventilation system to the screen backwash area.	This area does not support safety equipment in the IPE model.

Table 3.2.1.11-1
VENTILATION SCREENING

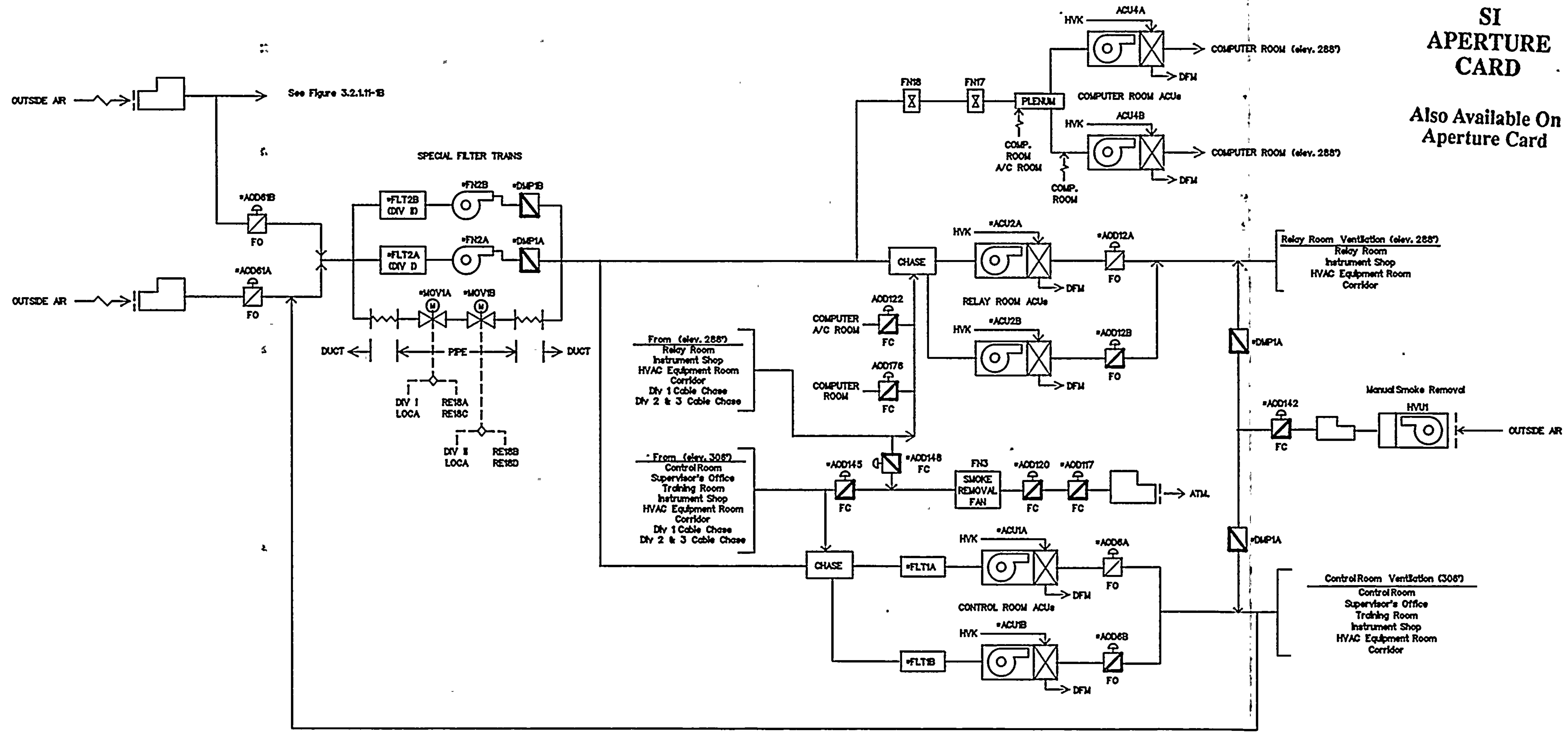
Building	Location or Room	HVAC System	Ventilation Equipment	Support Systems	Comments	Screening
	Circulating Water Pump Area	2HVY	Screenwell Building (above), -FN5A, -FN5B	NHS, NJS	The circ water pump area has two exhaust booster fans (-FN5A/B), a roof vent (-RFV1) and the associated instrumentation and ductwork.	This area does not support safety equipment in the IPE model.
	Fire Pump Rooms	2HVY	Screenwell Building (above), -FN11, -FN7, -UC1	NHS, NJS	The diesel fire pump room has a supply (-FN11) and an exhaust fan (-FN7) which circulate outside air. The electric fire pump room is cooled by -UC1, and heated by unit heaters.	Important area requiring further evaluation.
	Service Water Pump Bays	2HVY	Screenwell Building (above), *UC2A, *UC2C, *UC2B, *UC2D, -FN17	EHS	Each bay has two 100% unit coolers. The bays exhaust to a common airshaft which has a smoke removal fan (-FN17). The airshaft exhaust to the yard through the Aux. Boiler Building.	Important area requiring further evaluation.
Electric Bay	General	2HVY	-FN15A, -FN15B, -FN16	NHS, NJS	Cooling and ventilation are achieved in the electrical bay by one inlet fan (-FN15) and two exhaust fans (-FN15A/B). Temperature is maintained above 65°F by nine unit heaters.	This area does not support safety equipment in the IPE model.
Chiller Building	General	2HVY	-FN19A, -FN19B		Cooling/ventilation is provided by two exhaust fans (-FN19A/B). Three unit heaters provide area heating.	This area does not support safety equipment in the IPE model.
Condensate Storage Tank Bldg	General	2HVY	-FN4A, -FN4B, -FN18A, -FN18B	NJS, NHS	Two exhaust fans (-FN4A/B) use outside air to maintain temperature in the CST Building below 95°F. Unit heaters maintain temperature above 65°F. Tank overflow ventilation fans (-FN18A/B) are used to exhaust gases from the DFM-TK1 sump & CNS-TK1A/B overflow to the stack.	Not required to support the IPE model.
Demineralized Storage Tnk Bldg	General	2HVY	-FN8	NHS, NJS	The demineralized storage tank building ventilation system provides heating and ventilation for:the waste-neutralizing tank, the acid storage tank and the sodium-hypochlorite skid areas on elev. 261' in the screenwell building.	This area does not support safety equipment in the IPE model.

[illegible]

Rev 0 (7/92)

SI APERTURE CARD

Also Available On Aperture Card



References:
PID-53B-12
PID-53C-11
PID-53D-10
PID-53E-11
PID-53F-9

— Relay Room HVAC Equipment

— Control Room HVAC Equipment

— Computer Room HVAC Equipment

— Common Equipment

Figure 3.2.1.11-1A
Control Building Ventilation
Elev. 288' & 306'

I:\NMP2IPE\DGN\2S11.DGN Jul. 07, 1992 12:08:18

9208050191 - 21

Rev 0 (7/92)

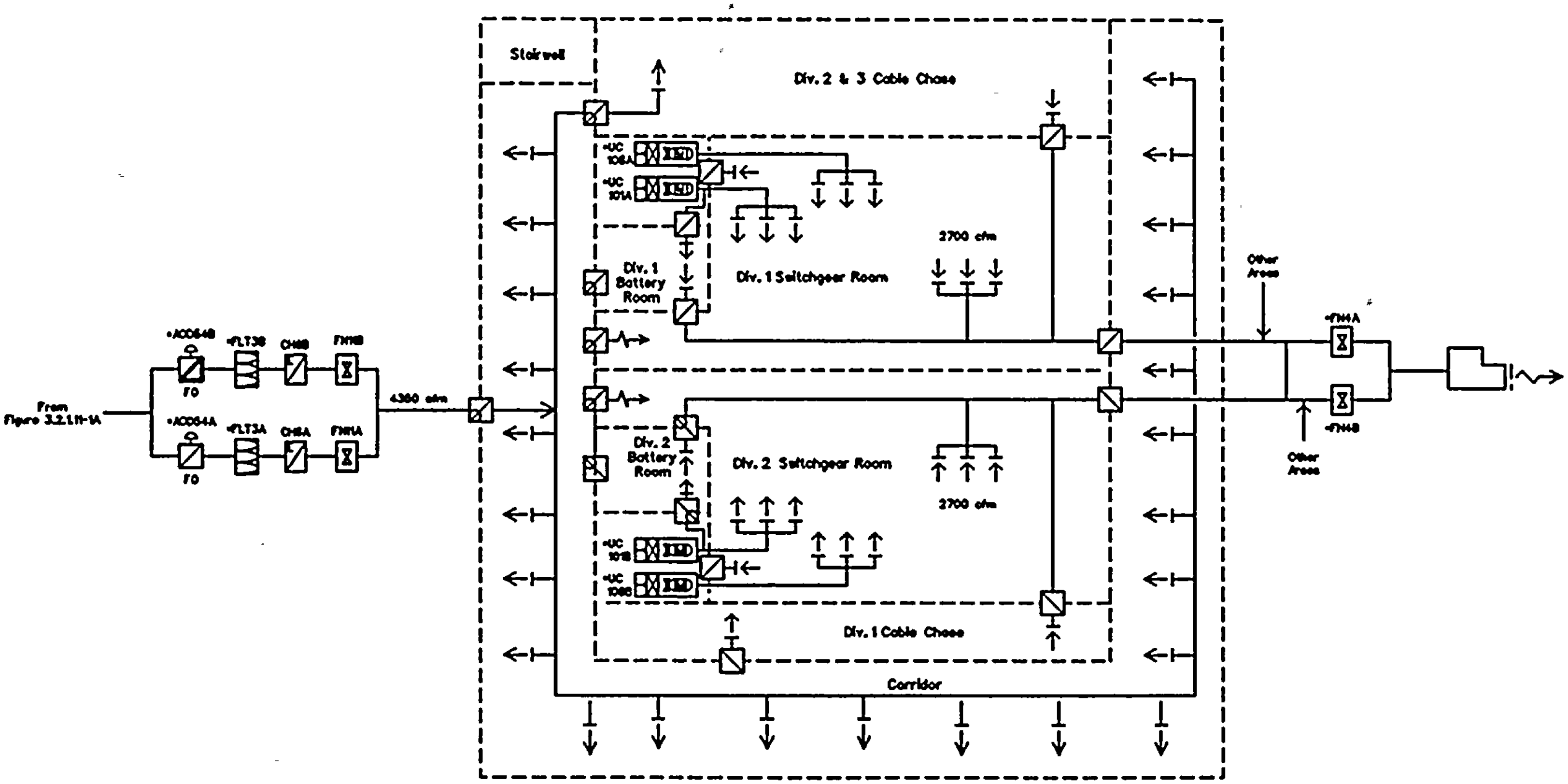
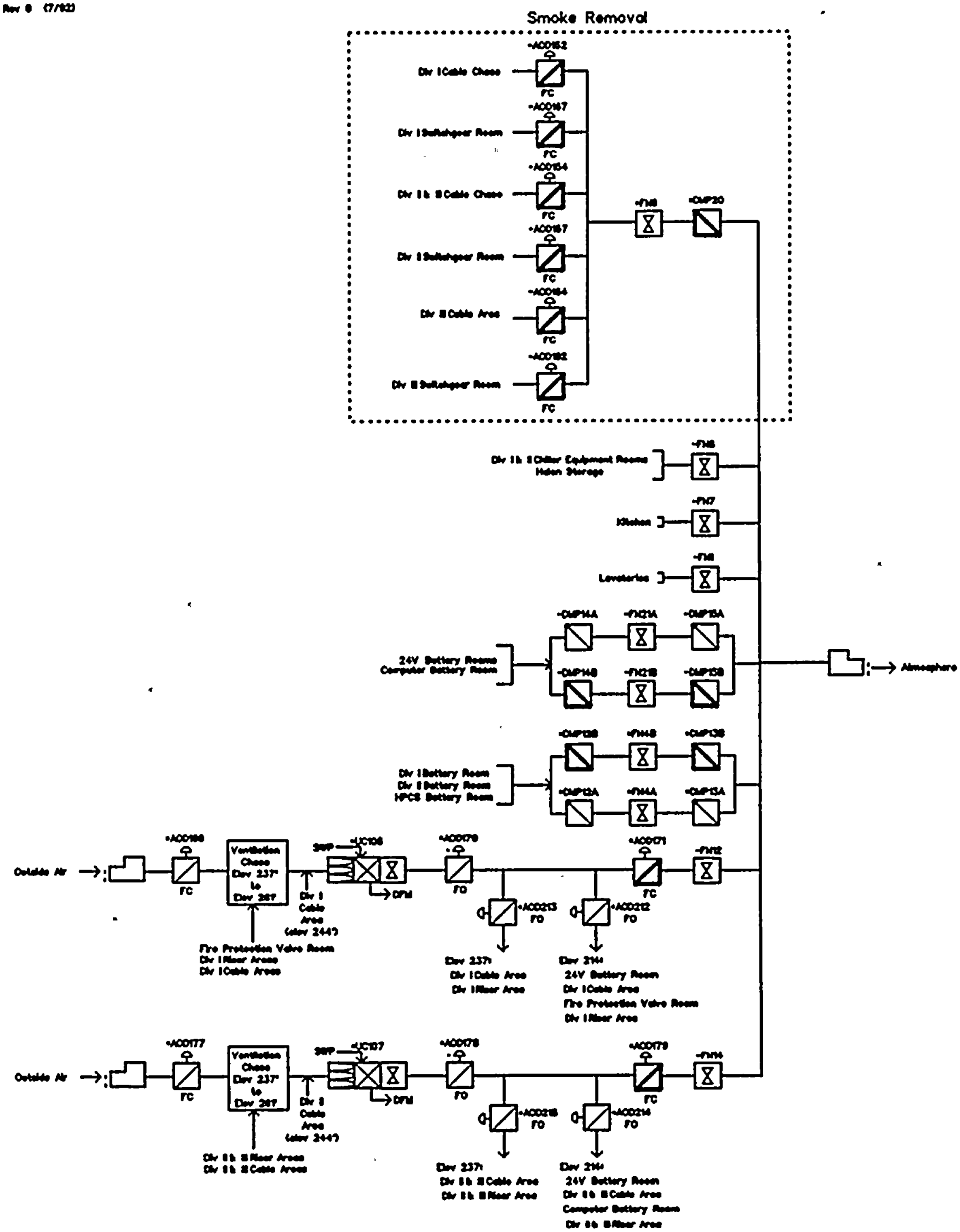


Figure 3.2.1.11-1B
Control Building HVAC
Switchgear/Battery Rooms

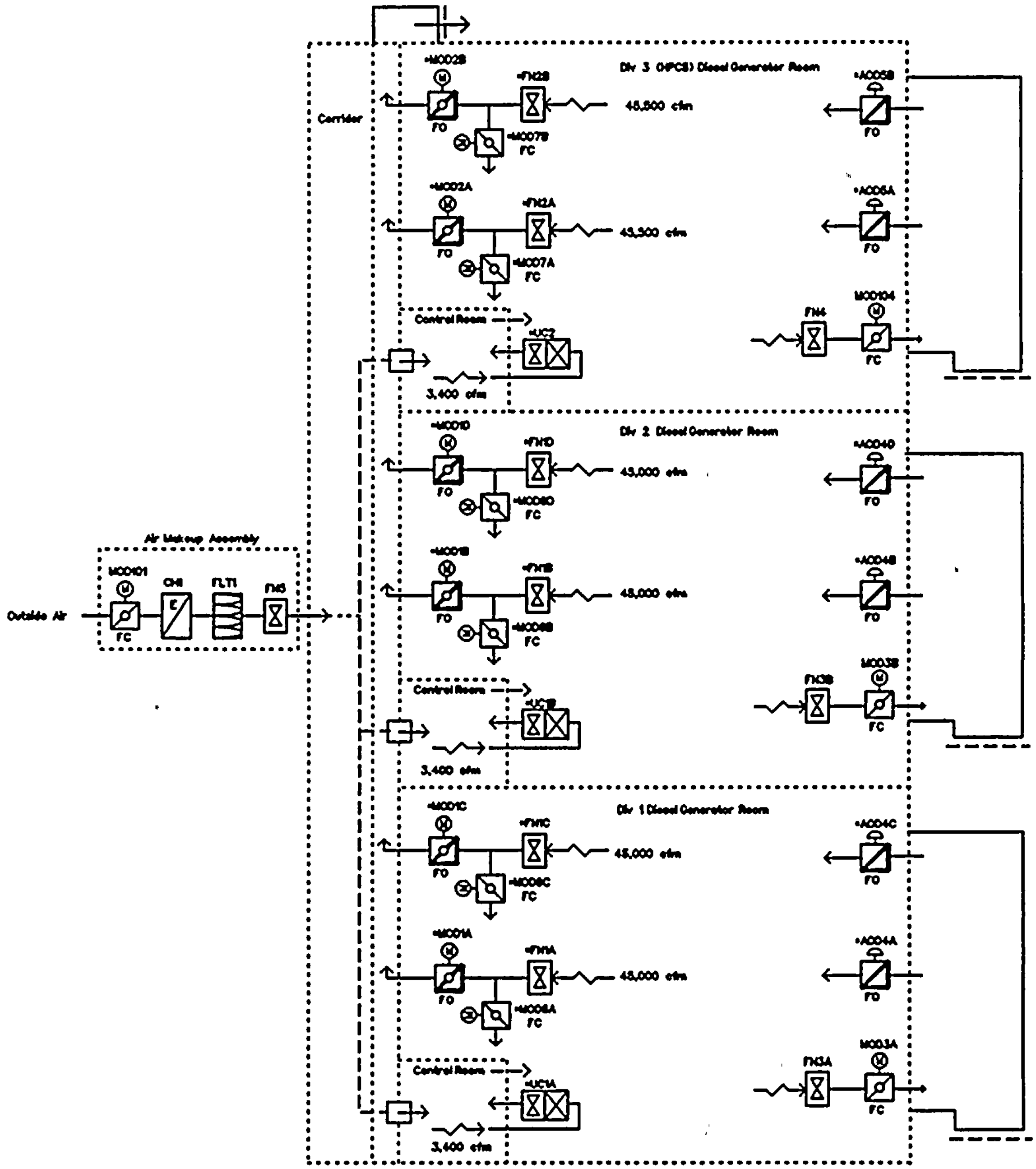
Rev 0 (7/92)



References:
PO-830-12
PO-83C-11
PO-830-10
PO-83E-11
PO-83F-9

Figure 3.2.1.11-1C
Control Building Ventilation
Standby Switchgear/
Battery Room

Rev 0 (7/93)



Reference: PID-57A-7

Figure 3.2.1.11-2
Diesel Generator Building Ventilation

Rev 0 (7/92)

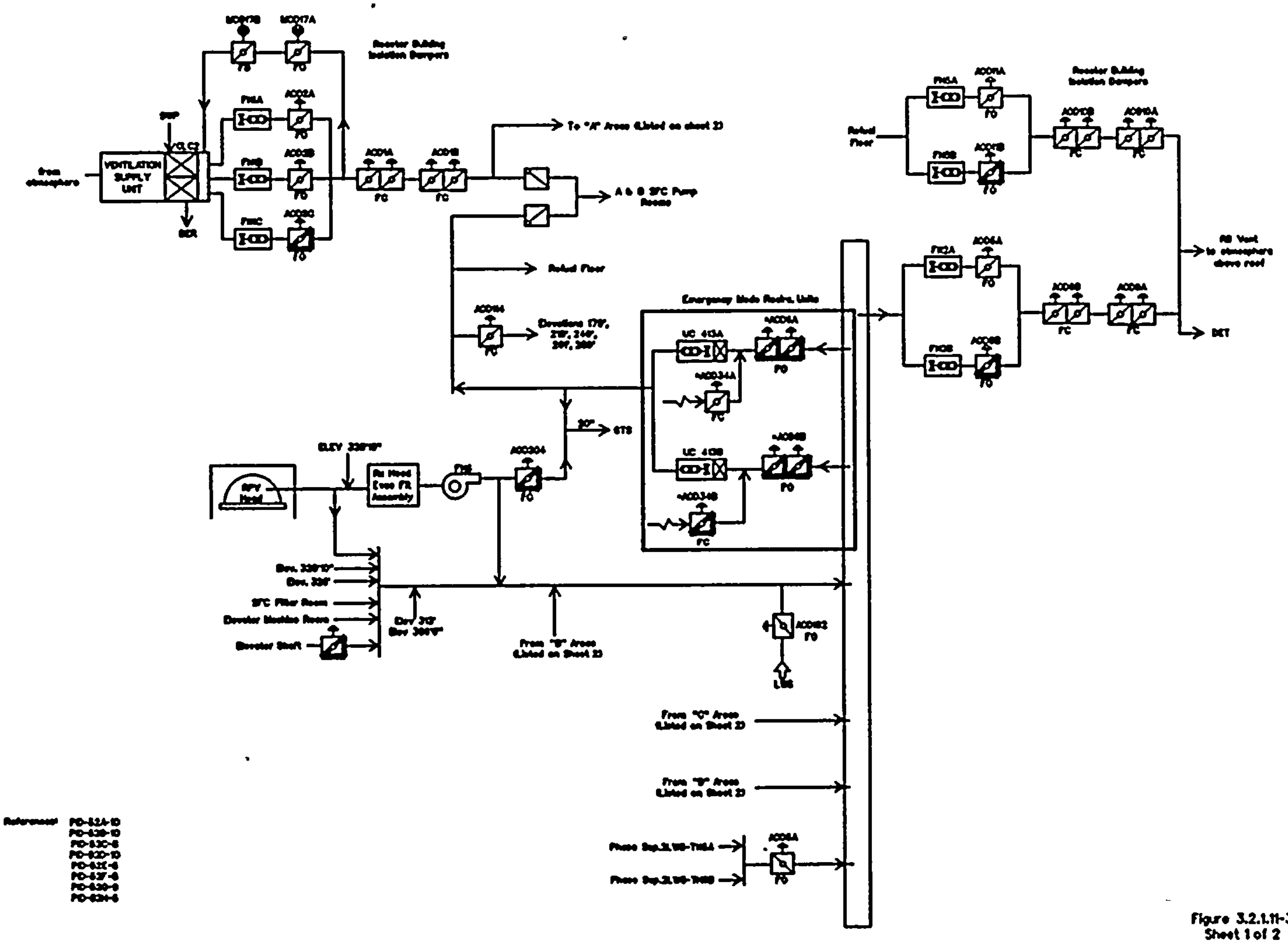


Figure 3.2.1.11-3
Sheet 1 of 2
Reactor Building Ventilation

I:\NMP2IPE\DGN\2S11.DGN Jul. 07. 1992 08:15:52

Figure 3.2.1.11-3 sheet 2 of 2
Reactor Building Ventilation

To A Areas	From B Areas	From C Areas	From D Areas
RCIC Pump Room	Elevation 240'	RCIC Pump Room (120 cfm)	Elevation 289'
LPCS Pump Room	Elevation 261'	HPCS Pump Room (120 cfm)	Elevation 306'
RHR Pump Rooms (A, B & C)	Elevation 289'	LPCS Pump Room	Elevation 328'
RBCLC Heat Exchanger Room	Control Rod Drive Maint. Facility Room	RHR Pump Rooms (A, B & C)	CRD Facility
North & South Elect. Cable Tray Area	Phase Separator Pump Rooms 6A & 6B	RBCLC Heat Exchanger Room	RWCU Heat Exchanger Rooms
Elect. Room NJC Load Center	South Shielding Area	Electric Cable Tray Area	RWCU FLTR Hold-up Pump Room
Elevation 175'	FLTR Backwash Tank Room	Elevation 175'	Phase Separator Tank Rooms 6A & 6B
Elevation 215'	Change Room	Elevation 215'	Contaminated Equipment Storage Room
Elevation 240'	A Spent Fuel Pool Cooling Pump Room	Elevation 240'	SFC Pump Room 1B
Elevation 261'	Spent Fuel Pool Demin. Room	Elevation 261'	Resin Storage Area
Elevation 286'		North Shielding Area	Precoat Pump Room
Elevation 328'10"		Valve Guide Room	
Elevation 353'10", Refuel Floor		A & B RWCU Pump Rooms	
Change Room		Rad Pipe Chase	
Contaminated Equipment Storage Area			
East Shielding Area			
Elevator Machine Room			
North Stair Tower			
Phase Separator Pump Room 6B			
Rad Pipe Chase			
A & B RWCU Pump Rooms			
A & B Spent Fuel Pool Cooling Heat Exchanger Rooms			

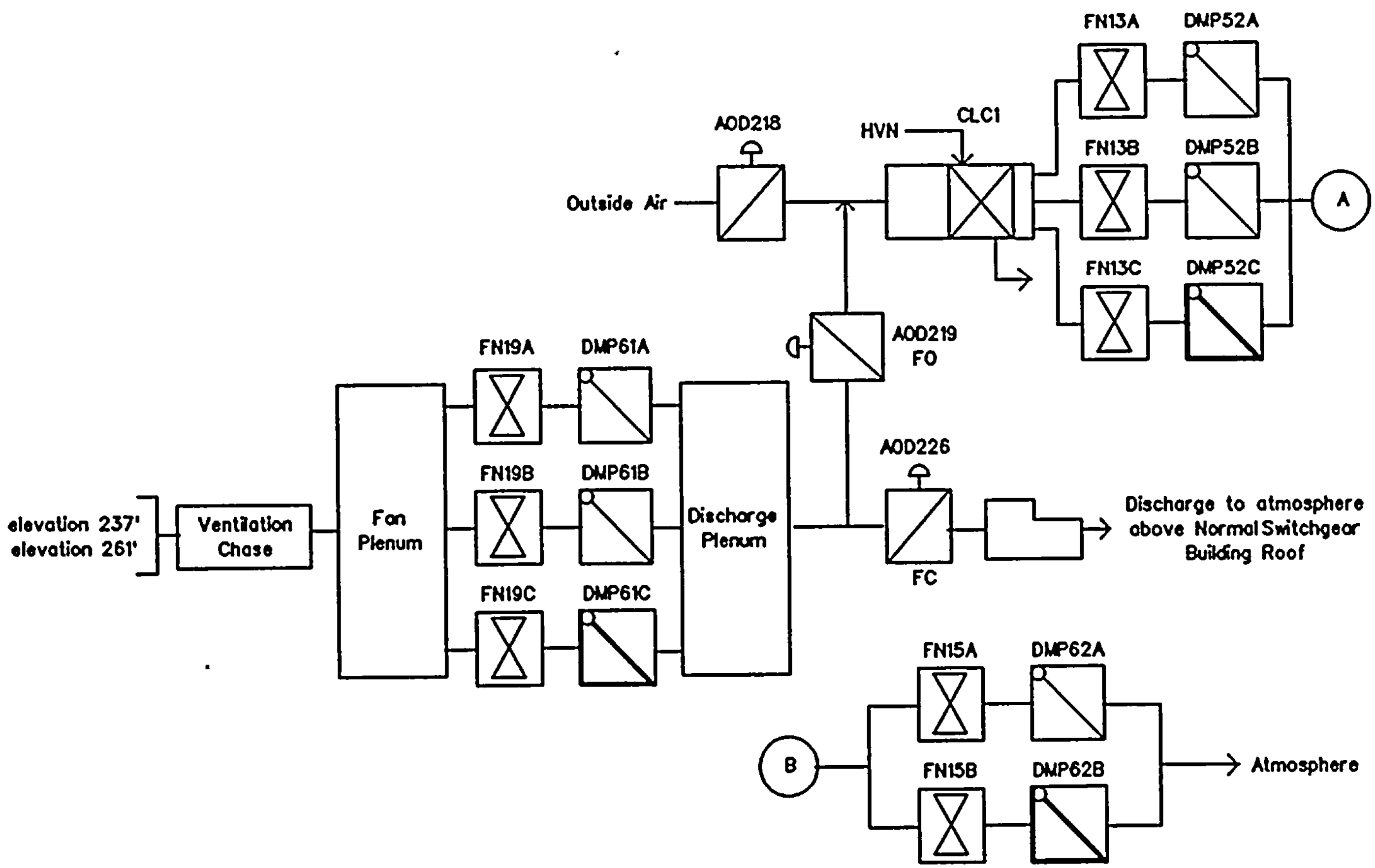
To A Areas	From B Areas	From C Areas	From D Areas
A & B Spent Fuel Pool Cooling Pump Rooms			

Rev 0 (7/92)

Intentionally
Left Blank

Figure 3.2.1.11-4

Rev 0 (7.92)



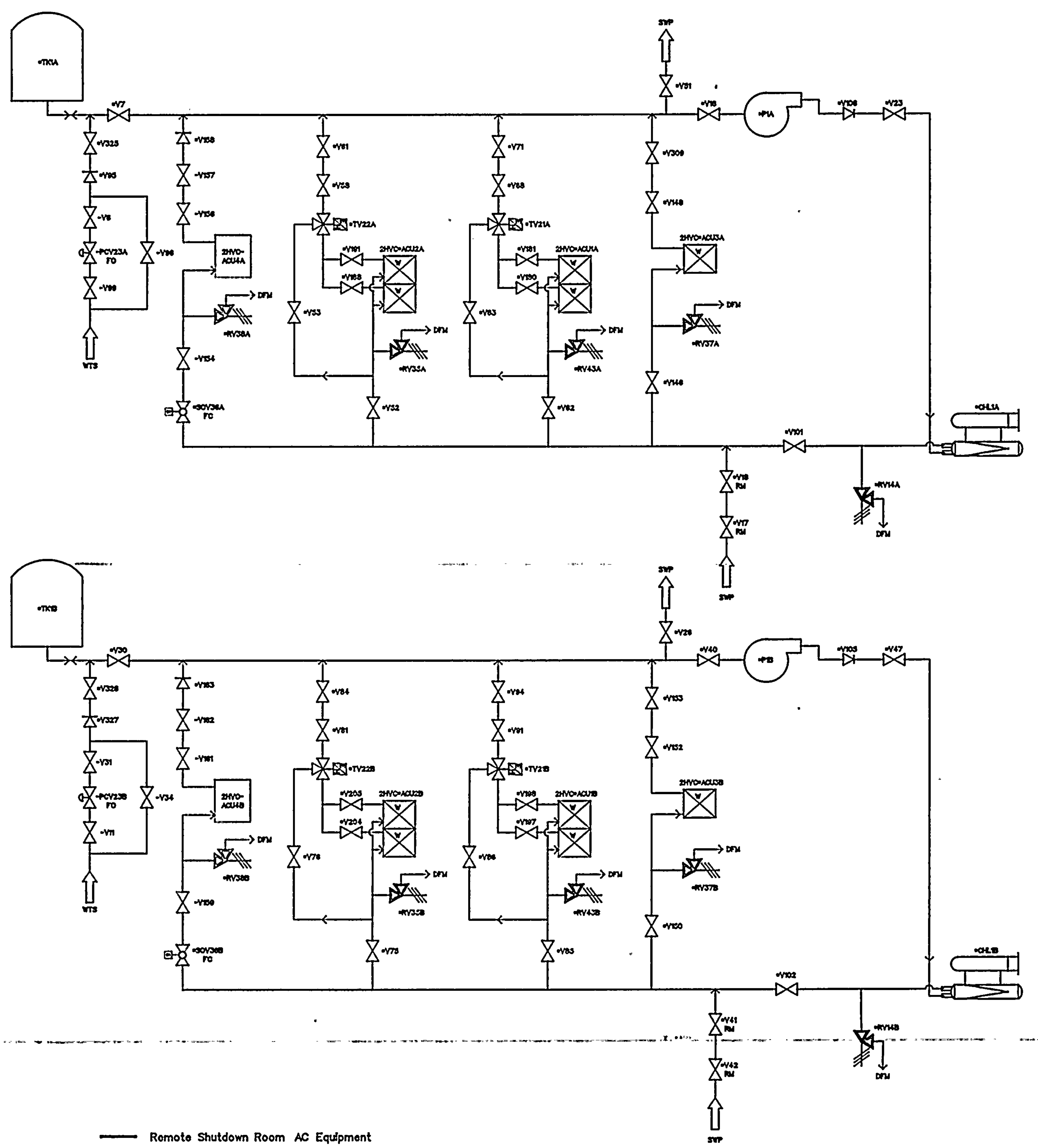
Reference: PID-54A-9

Figure 3.2.1.11-5
Normal Switchgear Building Ventilation

I:\NMP2IPE\DGN\2S11.DGN Jul. 07. 1992 08:17:31

Rev 0 (7/92)

Reference: PID-53A-10



—— Remote Shutdown Room AC Equipment

—— Control Room ACU Equipment

—— Relay Room ACU Equipment

—— Computer Room ACU Equipment

—— Chilled Water / WTS / Service Water Equipment

Figure 3.2.1.11-1D
Control Building Chilled Water

System 12

Standby Liquid Control

3.2.1.12 Standby Liquid Control (SLS) System

3.2.1.12.1 System Function

The Standby Liquid Control system injects sodium pentaborate solution into the reactor if sufficient control rods do not insert when required. A simplified drawing of the SLS system is provided in Figure 3.2.1.12-1.

3.2.1.12.2 Success Criteria

Success of the SLS system is considered to require two SLS pump trains injecting approximately 86 gpm into the reactor for one hour given auto-initiation from the redundant reactor control system. Top event SL in the ATWS event tree models the SLS system.

3.2.1.12.3 Support Systems

The SLS system requires Emergency AC power for operation. Table 3.2.1.12-1 summarizes major equipment, failure modes, and support equipment.

The following systems interface with or connect to the SLS system:

- The redundant reactivity control system (RRCS) automatically initiates SLS (see system operation below).
- The instrument air system connects to SLS providing air for storage tank mixing and level indication. Loss of instrument air results in level indication decreasing to zero.
- Demineralized water supplies water for testing and flushing.
- The SLS system shares the HPCS injection piping to the reactor vessel.

3.2.1.12.4 System Operation

The SLS system is in standby during normal operations.

If a sufficient number control rods do not insert into the core, the redundant reactivity control system (RRCS) automatically initiates SLS on high dome pressure or Low-Low water level (a 98 second timer is started). If at the end of 98 seconds, the signal is still present (Low-Low level is not sealed in) and sufficient power remains (APRM "not downscale" or "INOP") then RRCS will initiate SLS. Two RRCS actuation signals (Division I & II) are provided and each division starts both pumps. Each pump control circuit includes pump start, storage tank MOV open signal, and the squib valve open signal.

The SLS system is initiated manually from the main control room (panel P601) by turning a keylocked switch for system A or a different keylocked switch for system B to the pump RUN position.

3.2.1.12.5 Instrumentation and Controls

SLS can be initiated from the main control room. There is indication in the control room for the following:

- Storage tank outlet valves MOV1A & MOV1B (Close/Normal/Open)
- Pump 1A & 1B (Stop/Normal/Run)
- Squib Valve VEX3A & VEX3B continuity
- Testable check valve MOV5A & MOV5B (Open/Close/Off Normal)
- Storage tank level
- Pump test throttle valve HCV116 (Open/Close)
- Test tank outlet valve HCV111 (Open/Close)
- Common injection line valve HCV114 (Open/Close)

Annunciator alarms are provided in the control room for the following:

- Both Division I & II System Inoperable - Each division includes the pump, storage tank outlet MOV1, squib VEX3, isolation stop check MOV5, and manual out of service.

- SLSS Storage Tank Temp High/Low

- SLSS Storage Tank Level High/Low

- Both Division I & II SLSS Storage Tank Level Low

- Both Division I & II RRCS RWCU Isolated

- SLSS Pump 1A Valve 1A/5A Motor Overload

- SLSS Pump 1B Valve 1B/5B Motor Overload

Storage tank temperature indication is available locally.

3.2.1.12.6 Technical Specifications

With one pump and/or one explosive valve inoperable, restore to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours. With the SLS system otherwise inoperable, restore system to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours.

3.2.1.12.7 Surveillance, Testing, and Maintenance

The temperature and volume of the storage tank is checked daily.

The following are performed monthly:

- Verify continuity of explosive valves

- The weight and boron concentration of sodium pentaborate is determined. This procedure could cause cavitation if the aerator is running and the pumps start. However, there is a note in the procedure stating this, and there are holdout tags on the pump control switches to indicate the off normal condition and to remind the operator to stop the air flow if SLS initiation is required.

- Verify valve positions

The following are performed Quarterly:

- Pump test demonstrating minimum 41.2 gpm per pump at 1220 psig. Demineralized water from the test tank is used.

  NOTE: The train being tested is unavailable and there is the potential for misalignment. However, the procedure has all the valves returned to operable positions, and there is an independent, double, valve position verification after the procedure is completed. Therefore, the probability of a system failure due to misalignment is considered very small.

- MOV operability test on MOV1A, MOV1B, MOV5A and MOV5B. MOV1A and MOV1B are isolated during the test. If this test is performed before the pump test, valve misalignment would be detected during the pump test.

The following are performed during refueling:

- One SLS loop is manually initiated including explosive valve and the flow path from pumps to reactor vessel is verified by pumping demineralized water from the test tank.

- The piping from the storage tank is verified unblocked by pumping from the storage tank to the test tank for 1 minute.

- Valve position indication is verified.

- The heaters are verified to be OPERABLE.

- Channel functional test of SLS initiation. Squib valve fuses are removed during test.

Maintenance is performed as required. There is no planned maintenance performed during power operation that would result in unavailability of either loop.

3.2.1.12.8 References

N2-OP-36A, Rev. 2 "Standby Liquid Control"
N2-OP-36B, Rev. 1 "Redundant Reactivity Control System"

USAR Section 9.3.5, Revision 1
USAR Section 7.4.1.2, Revision 1

ESK-6SLS01 Sheet 1 Rev. 11
Drawings as referenced on the simplified drawing

N2-CSP-3M
N2-EPM-GEN-V531
N2-EPM GEN-V532
N2-EPM-GEN-V582
N2-ICP-GEN-@001
N2-IMP-GEN-038
N2-ISI-SLS-@001
N2-ISI-SLS-@003
N2-MPM-SLS-R143
N2-OSP-SLS-M001 Rev. 0
N2-OSP-SLS-Q001 Rev. 2
N2-OSP-SLS-Q002 Rev. 0
N2-OSP-SLS-R001 Rev. 1
N2-OSP-SLS-R002 Rev. 1
N2-OSP-SLS-R004 Rev. 0

3.2.1.12.9 Initiating Event Potential

Inadvertent operation of the SLS would shutdown the reactor over time, until the operators trip the pumps. Although this event would cause a cleanup problem, its frequency is small compared to other transients that trip the reactor.

3.2.1.12.10 Equipment Location

As shown in Figure 3.2.1.12-1, the common injection path (V10 & HCV114) is inside the primary containment. The storage tank, pumps and other valves are inside the reactor building.

3.2.1.12.11 Operating Experience

There were no particular outstanding operational events relevant to this study. Plant specific component operational data is detailed in section 3.3.2.

3.2.1.12.12 Modeling Assumptions

1. Success criteria requires both pump trains including the pump suction MOV, squib valve, and MOV check valve to be successful. That is, no redundancy in the system is modeled. This is based on the NMP2 response to the ATWS rule. As a result of this success criteria, the suction crosstie (V28 & 29) and the discharge crosstie (V52 & 53) are not modeled.

2. If the demineralized water isolation valve 2SLS*V5 were open during a SLSS actuation, this could result in a boron dilution of 50% or more. This is modeled as a failure. The failure mechanism is failure to close the valve after quarterly testing.

3. Instances of two or more locked closed independently verified valves that must be open to cause SLS failure are not modeled. The probability of this failure occurrence is small, and insignificant when compared to the other single point failures.

4. Unavailability of the system due to storage tank temperature, level and concentrations are neglected. Temperature and volume are checked daily and the concentration is determined monthly. In addition, temperature and level alarms are provided in the control room. To cause system failure, one of these parameters must be out of specification and go undetected. In addition, the out of specification condition must be very severe to realistically fail the system.

5. Failure of the system due to misalignment of HCV111, HCV114 and HCV116 is neglected because of independent verification after test and valve position indication in the control room.

6. Misaligning the instrument air connection to the pump suction is assumed to be an unlikely system failure.

7. Misalignment of the pump suction drain lines to the 55 gallon drum are neglected. The drains on the pump suction, if left open, would be detected since demineralized water would continue to drain into the drum.

8. Pipe failures are not modeled. The suction side is at low pressure and storage tank rupture & expansion joint ruptures are modeled as system failures. Exclusion of piping is assumed insignificant.

9. The explosive valves and storage tank isolation MOVs are initiated from the pump start circuit, but are assumed independent in the model. This dependency is insignificant since the model requires both trains for success.

10. Misalignment of V50 and V51 are presently not modeled since they are closed during tests performed at refueling. The frequency is less than that for V45 and 46, the impact is similar, and V45 and 46 are included in the model.

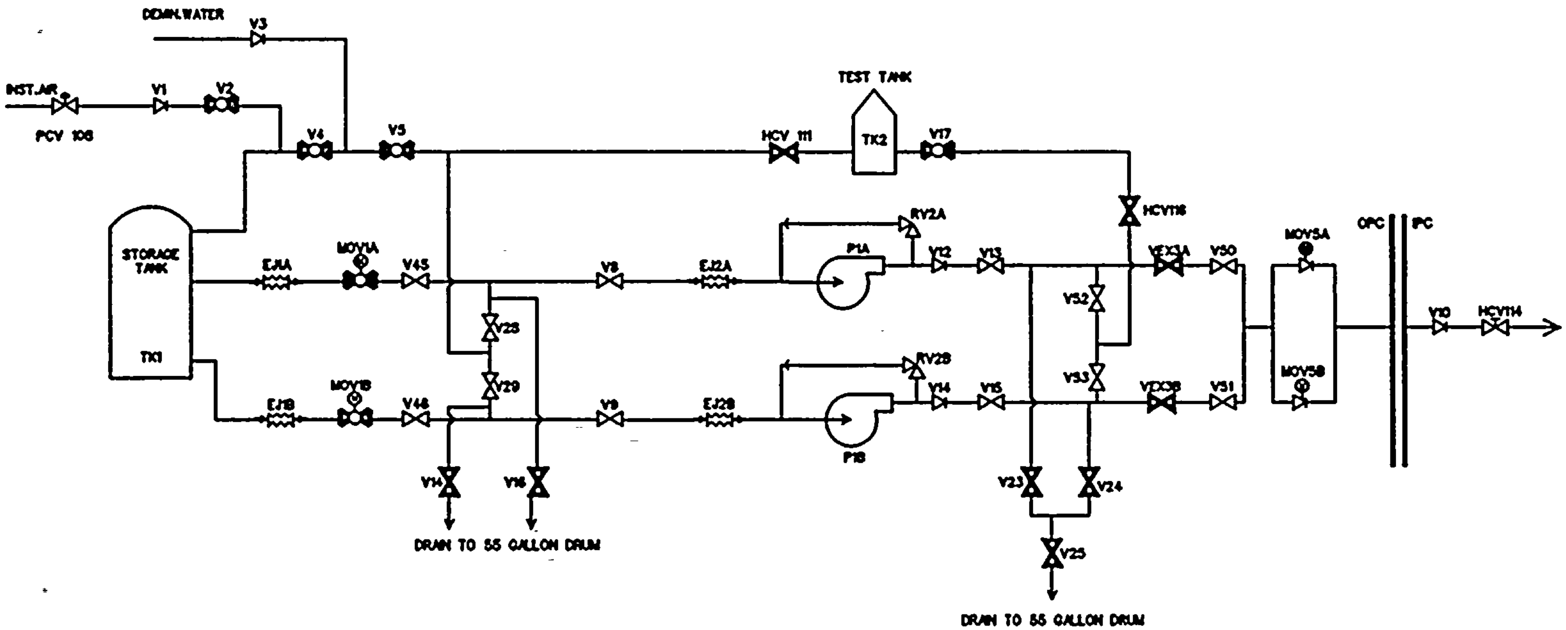
11. Staggered testing is used to ensure that one pumping path is always available for born injection into the RPV, except when reverse flow leak testing is performed on the pump discharge check valves, when both trains are momentarily inoperable.

12. The failure of HPCS check valve 2CSL*AOV108 (upstream of SLS injection point) to close is not modeled. This valve is counter-weighted to ensure closure, and the pipe should not drain unless a relief valve opens and sticks open (3 failures).

3.2.1.12.13 Logic Model and Results

The fault tree is included in the Tier 2 documentation. Quantitative results are summarized in section 3.3.5 (Tier 1).

Rev 0 (7/92)



References: PID-36A-10
FSK-27-16, Sh. A
FSK-27-16, Sh. B

Figure 3.2.1.12-1
Standby Liquid Control

Table 3.2.1.12-1 REV. 0 (7/92)

STANDBY LIQUID CONTROL
Component Block Descriptions

Block	Mark No. (Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
TANK/EXPANSION JOINTS	2SLS*TK1	Storage Tank	RUPTURES	N/A	N/A	N/A	N/A
	2SLS*EJ1A	Tank Expansion Joint	RUPTURES	N/A	N/A	N/A	N/A
	2SLS*EJ1B	Tank Expansion Joint	RUPTURES	N/A	N/A	N/A	N/A
	2SLS*EJ2A	Pump Expansion Joint	RUPTURES	N/A	N/A	N/A	N/A
	2SLS*EJ2B	Pump Expansion Joint	RUPTURES	N/A	N/A	N/A	N/A
MAKE-UP (DEMIN) WATER	2SLS*V5	Make-up Water Shutoff Valve	TRANSFERS OPEN	CLOSED	N/A	N/A	N/A
TRAIN A SUCTION	2SLS*MOV1A	Storage Tank MOV	FAILS TO OPEN	CLOSED	OPEN	2EHS*MCC102C(1)	AS-IS
	2SLS*V45	Manual Valve	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
PUMP A	2SLS*V8	Manual Valve	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2SLS*P1A	Pump 1A	FAIL TO START	STOP	RUNNING	2EHS*MCC102C(I)	STOP
	"	"	FAIL TO RUN	"	"	"	"
	2SLS*RV2A	Relief Valve	TRANSFER OPEN	CLOSED	N/A	N/A	N/A
	2SLS*V12	Check Valve	FAIL TO OPEN	CLOSED	OPEN	N/A	N/A
	2SLS*V13	Manual Valve	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
TRAIN A DISCHARGE	2SLS*VEX3A	Explosive Valve	FAIL TO OPEN	CLOSED	OPEN	2EHS*MCC102 (I)	CLOSED
	2SLS*V50	Manual Valve	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
TRAIN B SUCTION	2SLS*MOV1B	Storage Tank MOV	FAIL TO OPEN	CLOSED	OPEN	2EHS*MCC302D(II)	AS-IS
	2SLS*V46	Manual Valve	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
PUMP B	2SLS*V9	Manual Valve	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2SLS*P1B	Pump 1B	FAIL TO START	STOP	RUNNING	2EHS*MCC302D(II)	STOP
	"	"	FAIL TO RUN	"	"	"	"
	2SLS*RV2B	Relief Valve	TRANSFER OPEN	CLOSED	N/A	N/A	N/A
	2SLS*V14	Check Valve	FAIL TO OPEN	CLOSED	OPEN	N/A	N/A
	2SLS*V15	Manual Valve	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
TRAIN B DISCHARGE	2SLS*VEX3B	Explosive Valve	FAIL TO OPEN	CLOSED	OPEN	2EHS*MCC302	CLOSED
	2SLS*V51	Manual Valve	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
ISOLATION VALVES	2SLS*MOV5A	Isolation Stop Check Valve	FAIL TO OPEN	CLOSED	N/A	N/A	N/A
	2SLS*MOV5B	Isolation Stop Check Valve	FAIL TO OPEN	CLOSED	N/A	N/A	N/A

Table 3.2.1.12-1

STANDBY LIQUID CONTROL
Component Block Descriptions

Block	Mark No. (Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
COMMON INJECTION PATH							
	2SLS*V10	Check Valve	FAIL TO OPEN	CLOSED	OPEN	N/A	N/A
	2SLS*HCV114	Manual Valve	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
REACTOR WATER CLEANUP SYSTEM ISOLATION							
	2WCS*MOV102	RWCS Isolation Valve	FAILS TO CLOSE	OPEN	CLOSED	2EHS*MCC302	AS-IS
	2WCS*MOV112	RWCS Isolation Valve	FAILS TO CLOSE	OPEN	CLOSED	2EHS*MCC102	AS-IS

System 13

Automatic Depressurization

3.2.1.13 Automatic Depressurization System

3.2.1.13.1 System Function

The Automatic Depressurization System (ADS) lowers Reactor Pressure Vessel (RPV) pressure to allow the low pressure Emergency Core Cooling Systems, (Low Pressure Core Spray (LPCS) and Injection (LPCI)), to provide cooling water to the core. These systems serve as a backup to the high pressure coolant systems when a low reactor water level occurs with no significant loss of pressure and high pressure systems are either unavailable or insufficient to maintain vessel water level. A simplified diagram showing the seven ADS valves and nitrogen supply is provided in Figure 3.2.1.13-1. There are 18 safety relief valves (SRVs) capable of depressurizing the RPV. Seven of the 18 SRVs have an additional ADS accumulator and redundant ADS solenoids.

3.2.1.13.2 Success Criteria

RPV depressurization (or emergency RPV depressurization) is modeled in the front-line event trees with several top events.

Transient & LOCA Event Trees (NON-ATWS and NON-BLACKOUT)

Top event SV models the ADS/SRV valves and top event OD models the operator action. Since the Emergency Operating Procedures (EOPs) instruct the operators to inhibit ADS and manually depressurize the RPV if level is not restored, the front-line model assumes that manual depressurization (top event OD) is required for success.

Success of top event SV requires two of seven valves to open and remain open for 24 hours. The simplified success diagram in Figure 3.2.1.13-2 applies to the short-term (time < 15 hours) - each ADS valve will remain open for at least 15 hours with its respective accumulator tank if it is opened and left opened (i.e., the valve is not cycled open and closed several times). Beyond 15 hours, additional nitrogen supply tanks outside containment are assumed to be required.

ATWS Event Tree

Event tree top event OE models the operators emergency depressurizing the RPV in ATWS scenarios. The operator must be successful and three of seven of ADS valves must open. Therefore, the model is similar to SV discussed above except that OE requires three of seven valves and the operator is included.

Event tree top event AI models the operators inhibiting ADS. This is modeled for ATWS but not transient and LOCA because inhibiting ADS must be accomplished relatively quickly for ATWS scenarios. Also, the consequences of ADS inhibit failure in an ATWS are far greater than in transient or LOCA cases.

Event tree top event SR models adequate pressure control by the safety relief valves in the ATWS model. The number and capacity of relief valves gives a high level of confidence that pressure will be maintained at a safe level. It is assumed that approximately 90% of the relief capacity is required to mitigate the RCS pressure transient upon MSIV closure (i.e., 16 of 18 SRVs are required to open). Therefore, failure of 3 of the safety relief valves would cause a LOCA, thereby requiring the injection of cold, unborated water from low pressure ECCS. The subsequent initiation of ECCS is assumed to cause recriticality, leading to core damage and containment failure.

Event tree top event SO models the probability that 2 SRVs fail to reclose after the pressure excursion during a MSIV closure or loss of condenser vacuum ATWS event. Due to the large number of demands on safety relief valves to operate, the occurence two valves failing to reclose is considered more probable than for an anticipated transient event without ATWS. It is assumed that, with stuck open SRVs, the RCIC system would not be capable of maintaining RCS inventory, even with early initiation of the SLC system. Additionally, the containment would be severely challenged by sustained addition of heat to the suppression pool. This would also provide more stress on an operator attempting to shutdown the reactor.

Station Blackout Event Tree

Event tree top events O1, O2, and O3 model operator actions to provide emergency depressurization of the RPV in a station blackout scenario. The operator must be successful and two of seven ADS valves must open.

Event tree top events X1, X2, and X3 model SRVs remaining open after they successfully open.

3.2.1.13.3 Support Systems

The ADS dependency on support systems varies as a function of time. In the short-term (within 15 hours of the event, time < 15 hrs.) each ADS valve requires Division I or Division II 125V DC for actuation (top events D1 and D2 in support tree model). Each ADS valve contains two solenoid valves ("A" and "B") supplied by top events D1 and D2 - both solenoids are supplied by the same accumulator tank (see Figure 3.2.1.13-3). All seven ADS valves and all remaining safety relief valves (11 additional valves) each contain a "C" solenoid powered from Division I 125V DC as shown.in Figure 3.2.1.13-3.

In the long-term (time > 15 hrs.), Nitrogen make up to the individual accumulators is obtained from two dedicated ADS tanks via a containment isolation valve and an automatic supply valve actuated on low nitrogen gas pressure. Operator action is required to bypass the containment isolation signal in order to open the isolation valve to recharge the ADS accumulators. The following ADS valves rely on the long-term nitrogen supply from 2IAS*TK4 and 2IAS*TK5 as follows (See Figure 3.2.1.13-1):

ADS Valve	Accumulators	Isolation Valve	Makeup Valve	N_2 Supply

2MSS*PSV121	2IAS*TK32	2IAS*SOV164	2IAS*SOVX181	2IAS*TK4
2MSS*PSV126	2IAS*TK34	2IAS*SOV164	2IAS*SOVX181	2IAS*TK4
2MSS*PSV127	2IAS*TK33	2IAS*SOV164	2IAS*SOVX181	2IAS*TK4
2MSS*PSV129	2IAS*TK38	2IAS*SOV165	2IAS*SOVX186	2IAS*TK5
2MSS*PSV130	2IAS*TK37	2IAS*SOV165	2IAS*SOVX186	2IAS*TK5
2MSS*PSV134	2IAS*TK36	2IAS*SOV165	2IAS*SOVX186	2IAS*TK5
2MSS*PSV137	2IAS*TK35	2IAS*SOV165	2IAS*SOVX186	2IAS*TK5

As long as at least one 125V DC power source and nitrogen gas is available, then each ADS valve can be manually opened. Each ADS valve can be automatically initiated by Division I or II Emergency Core Cooling System (ECCS) signals (E1 or E2 in the support system event tree). Figure 3.2.1.13-4 provides a sketch of simplified ADS logic. The relationship between the support systems and the ADS is depicted on Figure 3.2.1.13-5. The effects of various support system failures on ADS operation are described in Table 3.2.1.13-1.

The following equipment or systems interface with ADS:

- Main Steam
- Instrument Air System & Nitrogen Systems
- Remote Shutdown Control Panel
- RHR In Suppression Pool Cooling Mode
- Suppression Pool
- Primary Containment

3.2.1.13.4 System Operation

The ADS is automatically actuated when certain conditions are sensed and other permissives are satisfied. The conditions and permissive that actuate ADS Division I (Division II is similar) are (refer to Figure 3.2.1.13-4).

- RPV Level 1 (twice) and
- RPV Level 3 and
- ADS not inhibited and
- LPCS or LPCI A pump operating and
- 105 Sec Time Delay

Automatic ADS initiation is blocked per N2-EOP-RPV which requires the plant operator to actuate the ADS Inhibit Switch (S34). When conditions (vessel level, suppression pool temperature, drywell pressure) reach levels stated in the EOPs, the operator opens all seven ADS valves using each individual valve control switch. A typical valve control scheme is shown on Figure 3.2.1.13-3 for 2MSS*PSV129. Switch S16A or S16B when actuated bypasses the auto-open contacts K8A and K8E or K8B and K8F causing the valve to open and stay open.

Operation of the ADS will increase suppression pool temperatures. The operators must start the RHR system in the suppression pool cooling mode to ensure that the steam quenching capacity of the pool is not reduced below acceptable levels.

Additional Manual Actions

There are manual operations needed to ensure availability of nitrogen after 15 hours (N2-EOP-RPV pressure control). The operator must override any LOCA signal to valves 2IAS*SOV164 for Division I and 2IAS*SOV165 for Division II so that nitrogen can be supplied from 2IAS*TK4 and TK5. This is performed by turning a key locked switch in the control room to the override position to enable the SOVs (2IAS*164 and 165) to be opened. This override switch also allows SOVs (2IAS*SOV166 and 184) that supply nitrogen (N2) to all 18 SRV accumulators to be opened.

3.2.1.13.5 Instrumentation and Controls

The safety/relief valves (all 18) are spring-loaded with external pneumatic operating cylinders which permit remote or automatic opening at pressures below the set point of their spring actuators. Pneumatic operation is initiated by remote-manual switches, by signals from the reactor vessel pressure transmitters, or, in the case of the ADS valves, by a signal from ADS logic channels. Each of the 18 safety/relief valves is equipped with a "C" solenoid operated pilot valve which, when energized, admits nitrogen to the pneumatic cylinder actuator. The ADS function is accomplished with 7 of the 18 valves. These ADS valves have two additional solenoids (A and B), actuated by the ADS relay logic.

Controls for ADS valves are located on panels P628 (Division I logic) and P631 (Division II logic). Each panel has 7 key-locked switches with "AUTO" and "OPEN" positions, 7 sets of open and shut indication lights, a logic power test push-button and logic test jacks. On panel P601 there are "Armed" push-buttons for manual initiation, logic resets and a Manual-Out-of-Service push-button (S37A/S37B) is provided for Division I and Division II logic respectively, with lights for indicating system status. Seven white lights from Division I ADS and seven white lights from Division II ADS on panel P601 provide indication of any SRV which operates in the ADS mode.

Each ADS safety/relief valve has an additional nitrogen accumulator which is normally supplied from the gaseous nitrogen storage system via an ADS storage tank, pressure regulator and primary containment isolation valves. The nitrogen from the individual accumulator is then routed to the "A" and "B" solenoids which port the nitrogen to operate the safety/relief valve actuator when ADS relay logic is satisfied.

The ADS safety/relief valves are provided with individual actuating control circuits for solenoids "A" and "B" as shown on Figure 3.2.1.13-3. The circuit for solenoid "A" is called ADS Logic Channel "A" or ADS Division I, and the circuit for solenoid "B" is called Channel "B" or ADS Division II. Each relay logic contains two subchannels, "A" and "E" for Division I, and "B" and "F" for Division II. Either Division logic can automatically actuate the ADS function when both subchannels of that Division are energized.

Subchannels "E" and "F" are energized when all of the following conditions exist: a) low-low-low reactor water level of 17.8" (Level 1); and b) either LPCS or "A" RHR Pump running ("E" subchannel), "B" or "C" RHR Pump ("F" subchannel) running. A simplified elementary is shown on Figure 3.2.1.13-4.

Subchannels "A" and "B" are energized when the following conditions exist: a) low-low-low reactor water level of 17.8" (level 1); b) confirmatory low reactor water level of 159.3 inches (Level 3); c) either LPCS or "A" RHR Pump running ("A" subchannel) - either "B" or "C" RHR Pump running ("B" subchannel) and; d) 105 second time delay.

With subchannels "A" and "E" energized, the "A" solenoids of the ADS safety/relief valves are energized. With subchannels "B" and "F" energized, the "B" solenoids of the ADS safety/relief valves are energized. Either of the "A" or "B" solenoids for any ADS safety/relief valve will cause the valves to pneumatically actuate and remain open until initiation signals and logic are reset.

Division initiation logic can be disabled by use of each Division ADS AUTO INITIATION DISABLE switches.

These are keylocked switches located on panel 601 in the Control Room. Each division relies on its associated 125V DC power supply for indication only, the inhibit function requires only operator action to turn the switch and disconnect the contacts. When operated, the automatic depressurization function is disabled, but can be manually initiated. The valves can still be opened one at a time manually, and will still operate in the safety mode for vessel over-pressurization.

Depressing the ADS logic initiated seal-in reset push-buttons for Division I and Division II will reset the logic when the initiation signals have cleared or will reset the 105 second timer when initiation signals are still present. Manual initiation of ADS can be accomplished by arming and depressing the ADS LOGIC MANUAL INITIATION push-buttons, A & E logic for Division I or the B & F logic for Division II.

Four (4) ADS valves can be transferred and operated from the Remote Shutdown Panel.

3.2.1.13.6 Technical Specifications

With either ADS Trip System "A" or "B" inoperable, the inoperable trip system must be returned to operable status within:

1. Seven days, provided that the HPCS and RCIC systems are operable, or

2. Seventy-two hours, provided either the HPCS or RCIC systems are inoperable.

Otherwise the plant is placed in Hot Shutdown within the next 12 hours and dome pressure is reduced to less than 100 psi.
With up to two ADS valves inoperable, restore the inoperable ADS valve(s) to OPERABLE status within 14 days of the first ADS valve becoming inoperable or be in at least HOT SHUTDOWN within the next 12 hours and reduce reactor steam dome pressure to less than or equal to 100 psig within the next 24 hours.

With three or more ADS valves inoperable, be in at least HOT shutdown within 12 hours and reduce reactor steam dome pressure to less than or equal to 100 psig within the next 24 hours.

ADS System:

At least once per 31 days, perform a channel functional test of the accumulator backup compressed gas system, low pressure alarm system.

At least once per 18 months:

a. Perform a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence, excluding actual valve actuation.

b. Manually open each ADS valve when the reactor steam dome pressure is greater than or equal to 100 psig.

c. Perform a CHANNEL CALIBRATION of the accumulator backup compressed gas system, low-pressure alarm system, and verify an alarm setpoint of 163.5 ± 3.2 psig on decreasing pressure.

d. Perform a leak rate test for ADS SRV pneumatic operators.

3.2.1.13.7 Surveillance Testing and Maintenance

Testing of the ADS will not interfere with automatic ADS operation on an initiation signal, due to the redundancy of the channels.

Four test jacks are provided to allow ADS logic testing, one for each logic subchannel. During testing, only one logic subchannel should be actuated at a time. However, when the test plug is plugged into one test jack, the complimentary subchannel of that division is automatically rendered inoperative. Inadvertent ADS actuation cannot occur even if both channels are improperly placed in the test mode at the same time.

An annunciator for a faulty test is sounded if test plugs are inserted in both subchannels of a division at the same time. Annunciation is also provided in the main control room whenever a test plug is inserted into a jack.

3.2.1.13.8 References

N2-OP-34, Rev. 4, Nuclear Boiler, Automatic Depressurization & Safety Relief Valves
N2-EOP-RPV, Rev. 4, RPV Control

GEK-833329A

FSAR 6.3.2.2.2
Tech Spec 3/4.6.2
Tech Spec 3/4.5.1
Tech Spec 3/4.3.3

0007.212-001-021N (GE: 807E155TY Sh. 4)
0007.212-001-026P (GE: 807E155TY Sh. 9)
0007.212-001-027N (GE: 807E155TY Sh. 10)
0007.212-001-028T (GE: 807E155TY Sh. 11)
Piping & Inst. Drawings referenced on Figure 3.2.1.13-1

3.2.1.13.9 Initiating Event Potential

The spurious opening of any safety/relief valve is an initiating event that causes a LOCA. This is included in the model as a medium break LOCA initiating event.

3.2.1.13.10 Equipment Location

The seven ADS valves are located inside the Primary Containment on the four main steam lines upstream of the inboard main steam isolation valve. Three main steam lines have two ADS valves and one main steam line has one. They were selected on the basis of their distribution among the four main steam lines and the location of their discharge quenchers in the suppression pool.

3.2.1.13.11 Operating Experience

There were no outstanding operational events relevant to this study. Plant specific component operational data is detailed in section 3.3.2.

3.2.1.13.12 Modeling Assumptions

1. The automatic actuation features of the ADS were not modeled. The plant EOPs call for inhibiting the ADS initially and manually opening each valve later on. Therefore, the auto-actuation features are not expected to be used.

2. The supply of nitrogen in the long-term (time > 15 hours) requires the operator to open containment isolation valves 2IAS*SOV164 and 165. This is accomplished by

turning a key-locked switch to the override position which overrides the containment isolation signal. The frequency of failure to perform this action is assessed to be small because:

a. The operators will have a substantial amount of available time.
b. Failure to open the valve can be reversed very easily.
c. The EOPs instruct operators to perform this action.

This operator action is modeled in top event OD which models the operator action to open SRVs in the short-term and to use the key-lock switch as required in the long-term.

3. The common nitrogen path that supplies all 18 SRV "C" solenoids is modeled when support systems are available. The failure of all 18 valves to operate is neglected since the common supply path will dominate.

4. The fault tree model assumes that all seven ADS valves are opened by operators and success requires an external nitrogen source in the long-term (16-24 hours). This could be conservative if the operators open only three or four valves at a time. Since only two SRVs are required to depressurize, opening a few rather than all seven ADS valves would allow the remaining valves to operate later without the need for an external nitrogen source. Opening fewer than seven ADS valves is not modeled.

5. The opening of valve 2IAS*V890 by the operator in the event of 2IAS*SOVX186 and 2IAS*SOVY186 failure is not modeled. This conservatism is made to simplify the human reliability analysis and because the small probability of redundant SOV failure does not affect results. A similar assumption is made for 2IAS*V889.

6. Pipe connections that have double isolation valves or connections which are small lines (typically less than 2 inches) are not modeled. In addition, pipe failures are not modeled. These failure modes are small contributors due to the redundancy in the system.

7. The emergency nitrogen connections in the missile protected area of the nitrogen yard are shown on the simplified drawing, but are not modeled. Use of these connections are not described in the EOPs, but it could be a source of makeup.

8. The rupture disks on the pressure relief line for nitrogen tanks 2IAS*TK4 and 2IAS*TK5 are not modeled. The system is tested to the high pressure alarm setpoint every 18 months at a pressure of 385 psig. This disk will also preclude a depressurization of the system due to a relief valve opening early.

9. The containment isolation valves are modeled as "Fail to Open and Remain Open". These valves are assumed to close on a containment isolation immediately after the initiating event.

10. The loss of the SRV nitrogen charging line is not modeled. We model the use of the SRV manual open once, and the accumulators will accomplish this. We do not model recharging and re-use of these accumulators.

3.2.1.13.13 Logic Model and Results

The fault tree models are attached. Quantitative results are summarized in Section 3.3.5. (Tier 1).

Table 3.2.1.13-1
ADS/SRV Valve Dependencies

State (1)	Support Failures (2)					Impact on ADS Valves (3)			Impact on SRVs, (4) N2 Path
	A1	A2	D1	D2	N2	Valves Available	TK4 Path	TK5 Path	
1	All Support Available					7	A	A	A
2			X		X	7*	A	A	U
3				X		7*	A	A	A
4					X	7	A	A	U
5	X				X	7	U	A	U
6	X		X		X	7*	U	A	U
6	X			X	X	7*	U	U	U
7		X			X	7	A	U	U
8		X	X		X	7*	A	U	U
8		X		X	X	7*	A	U	U
9				X	X	7*	A	A	U
F			X	X		0	———	———	F
SB +	X	X				7	U	U	U
SB +	X	X	X			7*	U	U	U
SB +	X	X		X		7*	U	U	U

* 1 of 2 solenoids on each ADS valve is unavailable
\+ Denotes Station Blackout

TABLE 3.2.1.13-1 NOTES
ADS/SRV VALVE DEPENDENCIES

1. The support state number refers to the split fractions used for top events SV and OE in the event tree quantification. For example, "1" refers to split fraction SV1 where the fault tree was quantified with all support systems available. "F" refers to guaranteed failure. A separate top event (O1, O2, and O3) models the station blackout, "SB", state.

2. X - Denotes support system unavailable

   A1 - Division I Emergency AC
   A2 - Division II Emergency AC
   D1 - Division I 125 V DC
   D2 - Division II 125 V DC
   N2 - Nitrogen Supply to 18 SRV "C" Solenoids

3. Impact on ADS valves includes availability of the valves and solenoids as well as the long-term nitrogen supply from 2IAS*TK4 and 2IAS*TK5.

   A - Denotes that containment isolation valves from TK4 or TK5 can be opened which would extend the availability of applicable ADS valves from 15 hours to > 24 hours.

   U - Denotes that containment isolation valves from TK4 or TK5 cannot be opened. The applicable ADS valves are capable of remaining open for approximately 15 hours with their individual ADS accumulator.

4. Impact on SRV N2 path due to support system failures are as follows:

   A - Denotes that containment isolation valves from N2 to the 18 SRVs (the "C" solenoid and accumulator) can be opened which allows their operation for 24 hours.

   U - Denotes that containment isolation valves cannot be opened or N2 is unavailable. The SRVs can operate on their non-ADS accumulator and solenoid for a short time period.

   F - Denotes that SRVs are unable to operate due to loss of DC power.

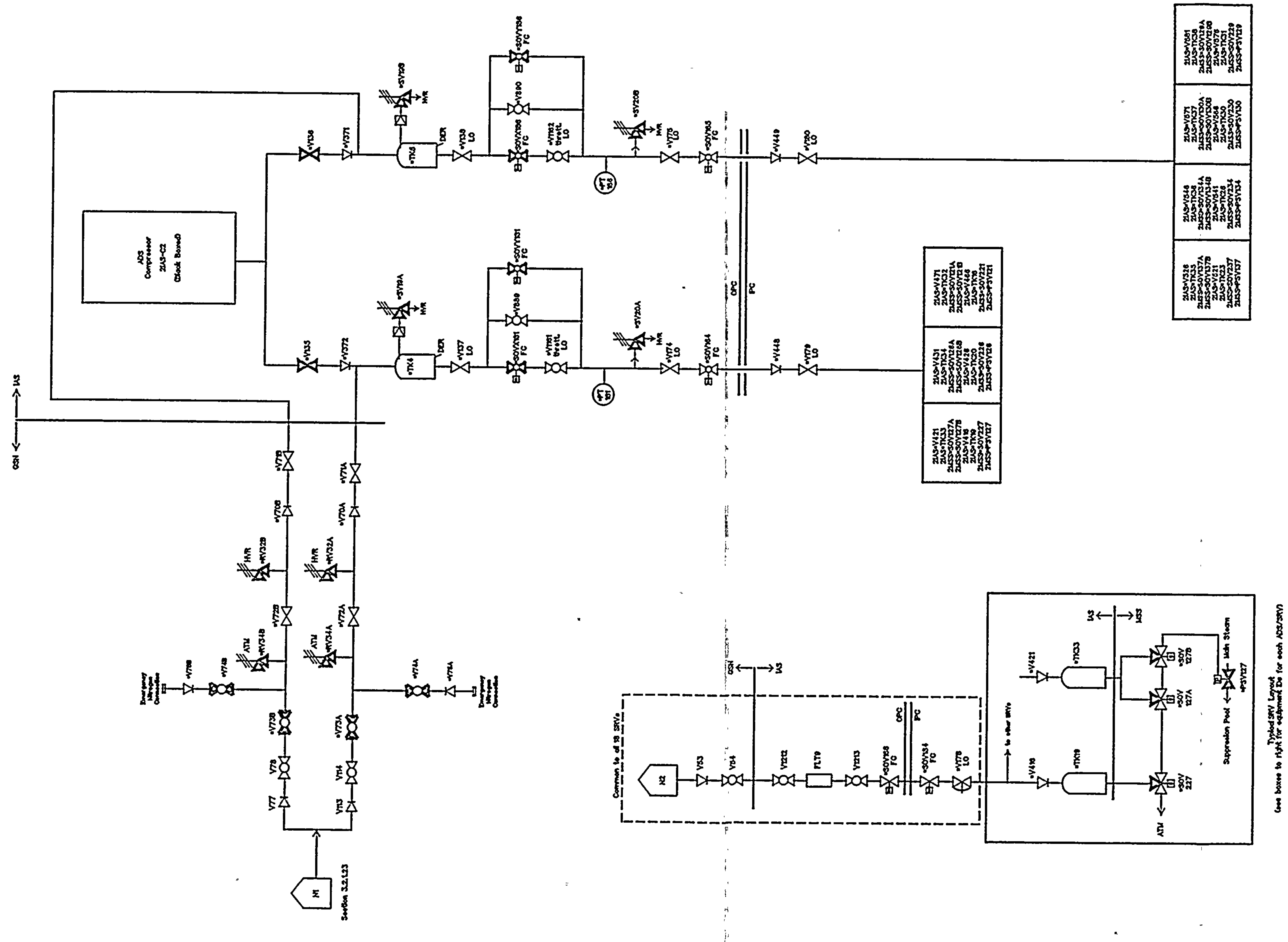


FIGURE 3.2.1.13-1
AUTOMATIC DEPRESSURIZATION (SV)

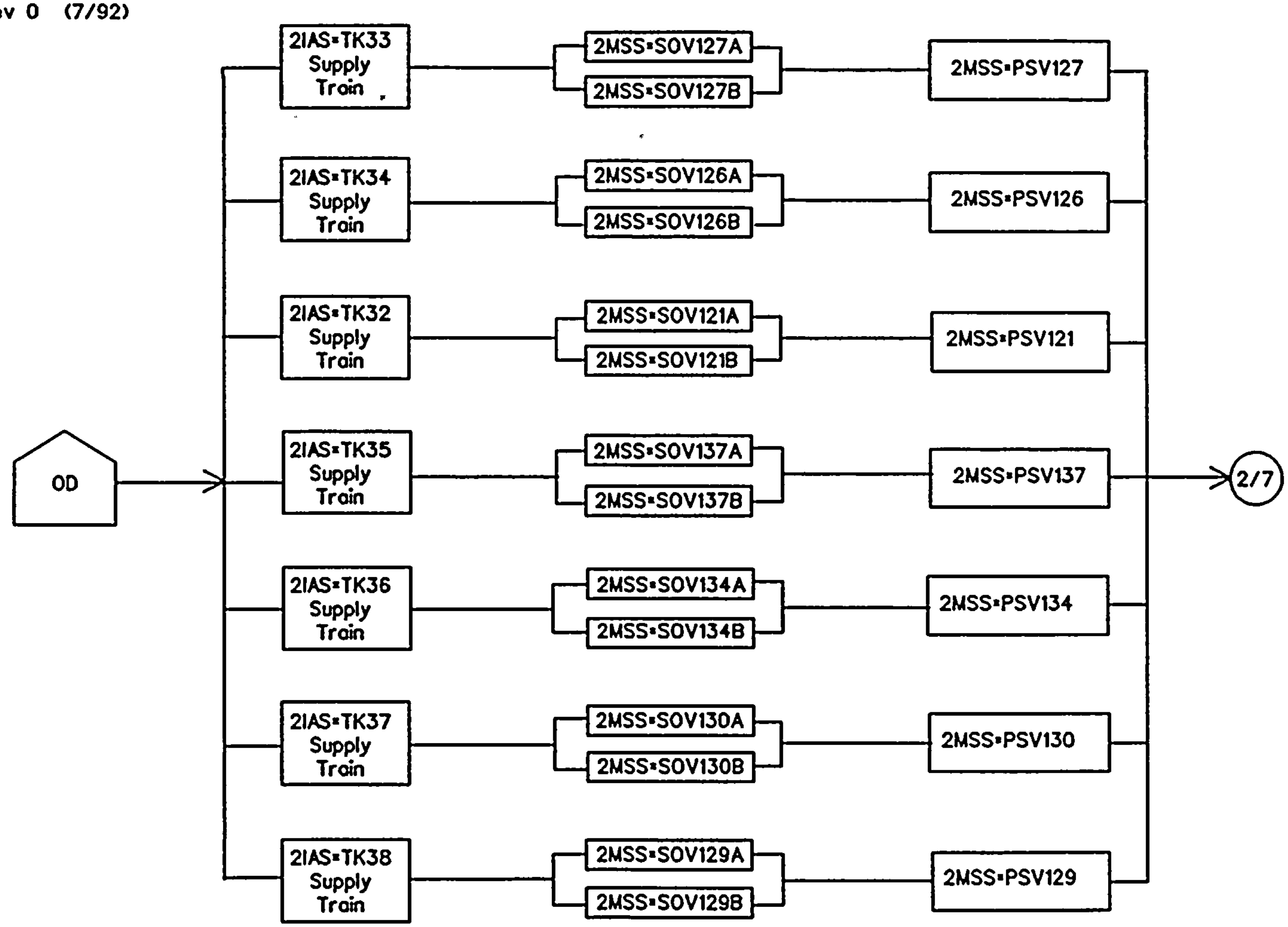
References: PID-1A-4
PID-1B-4
PID-1C-4
PID-1D-4
PID-18D-11
PID-19E-9
PID-19F-8
PID-105B-11

Rev 0 07/92

SI
APERTURE
CARD

Also Available On
Aperture Card

Rev 0 (7/92)



Note 1: "A" solenoids depend on Div I 125V DC (D1)
"B" solenoids depend on Div II 125V DC (D2)

Figure 3.2.1.13-2
SV SUCCESS DIAGRAM

I:\NHP2IPE\DGN\2S13.DGN Jul. 07. 1992 08:28:38

Rev 0 (7/92)

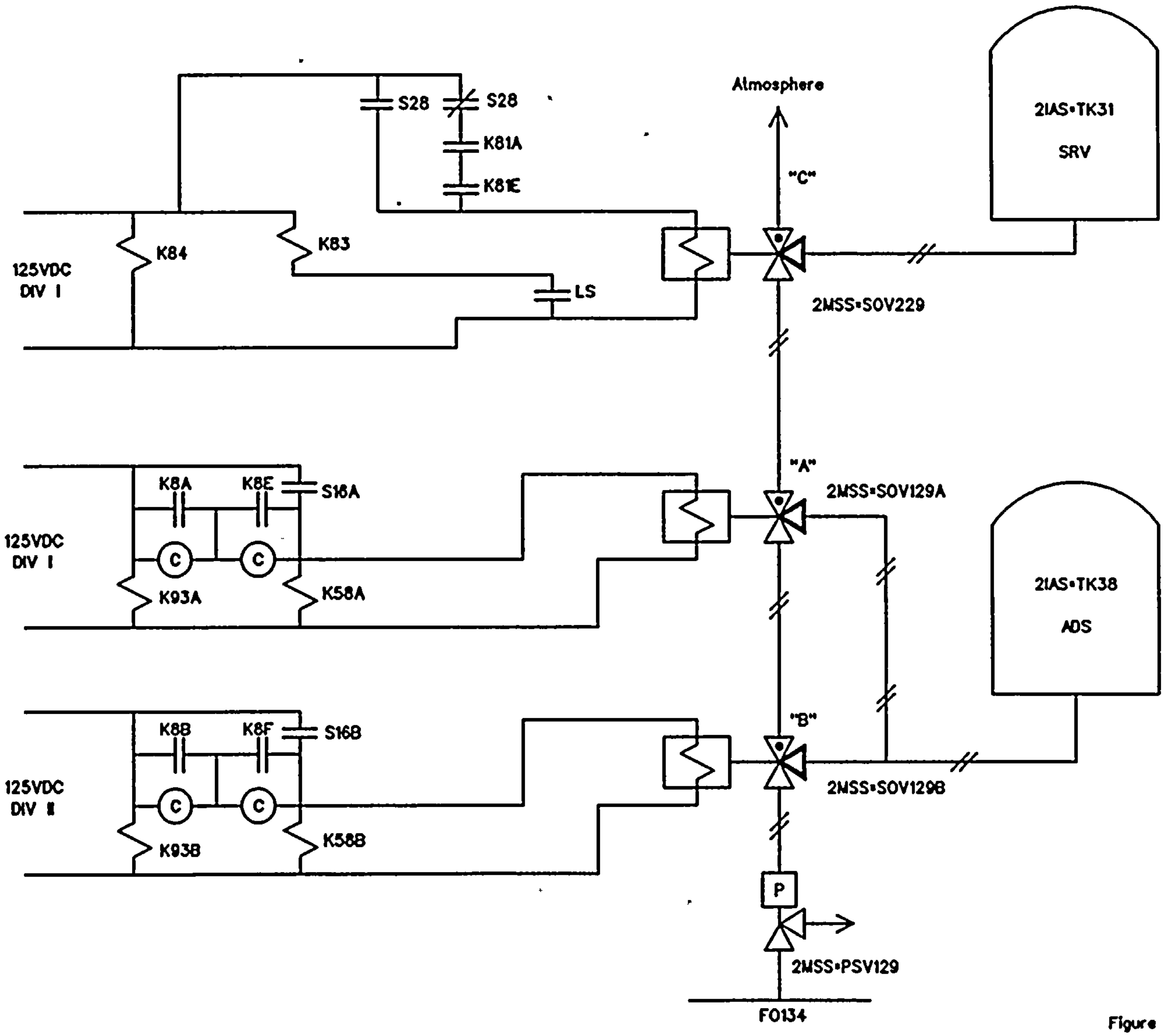


Figure 3.2.1.13-3
Typical ADS Valve

I: \NMP2IPE\DGN\2S13.DGN Jul. 07. 1992 08:31:40

Rev 0 (7/92)

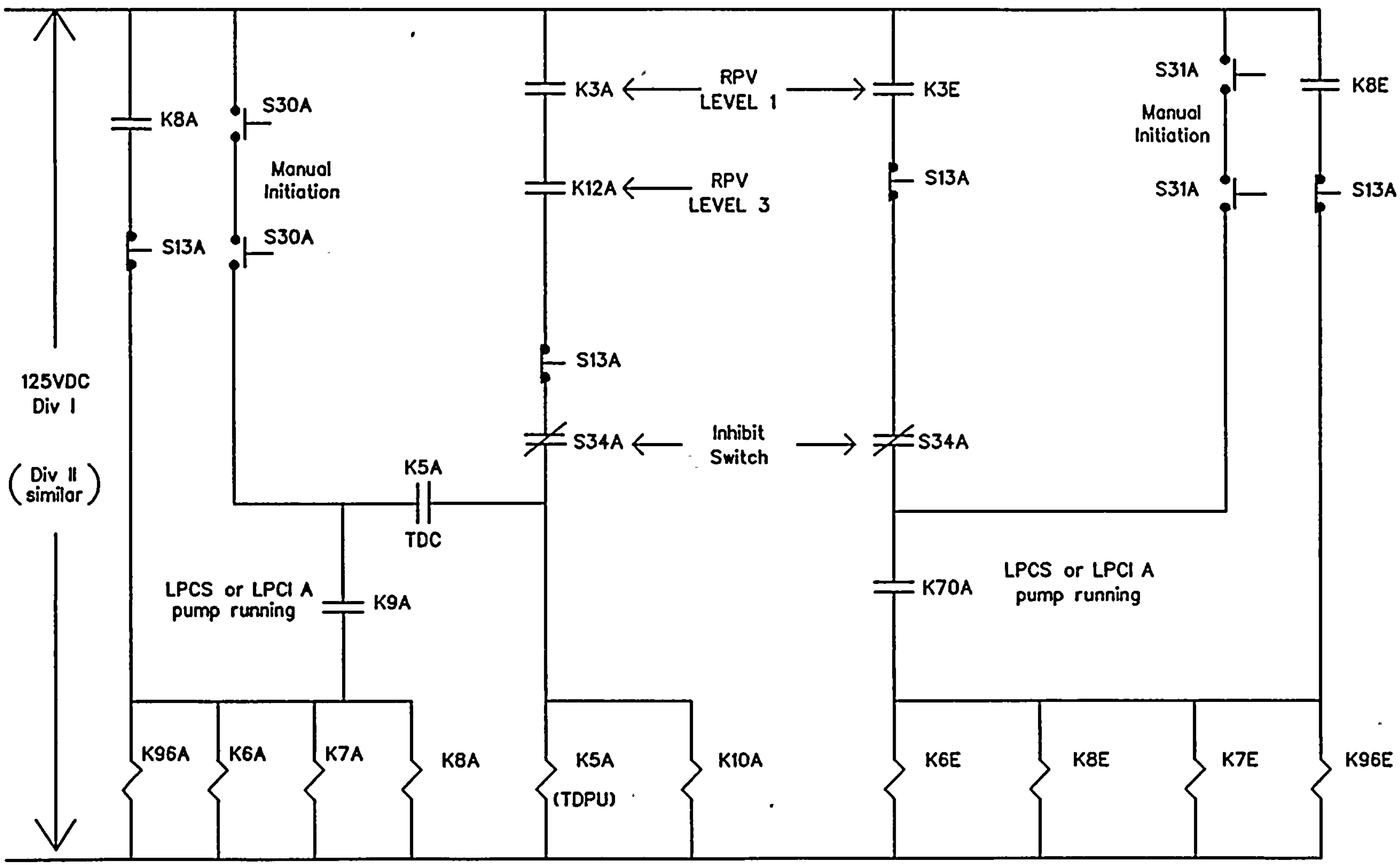


Fig 3.2.1.13-4
Simplified ADS Logic

I: \NMP2IPE\DGN\2S13.DGN Jul. 07. 1992 08:34:35

Rev 0 (7/92)

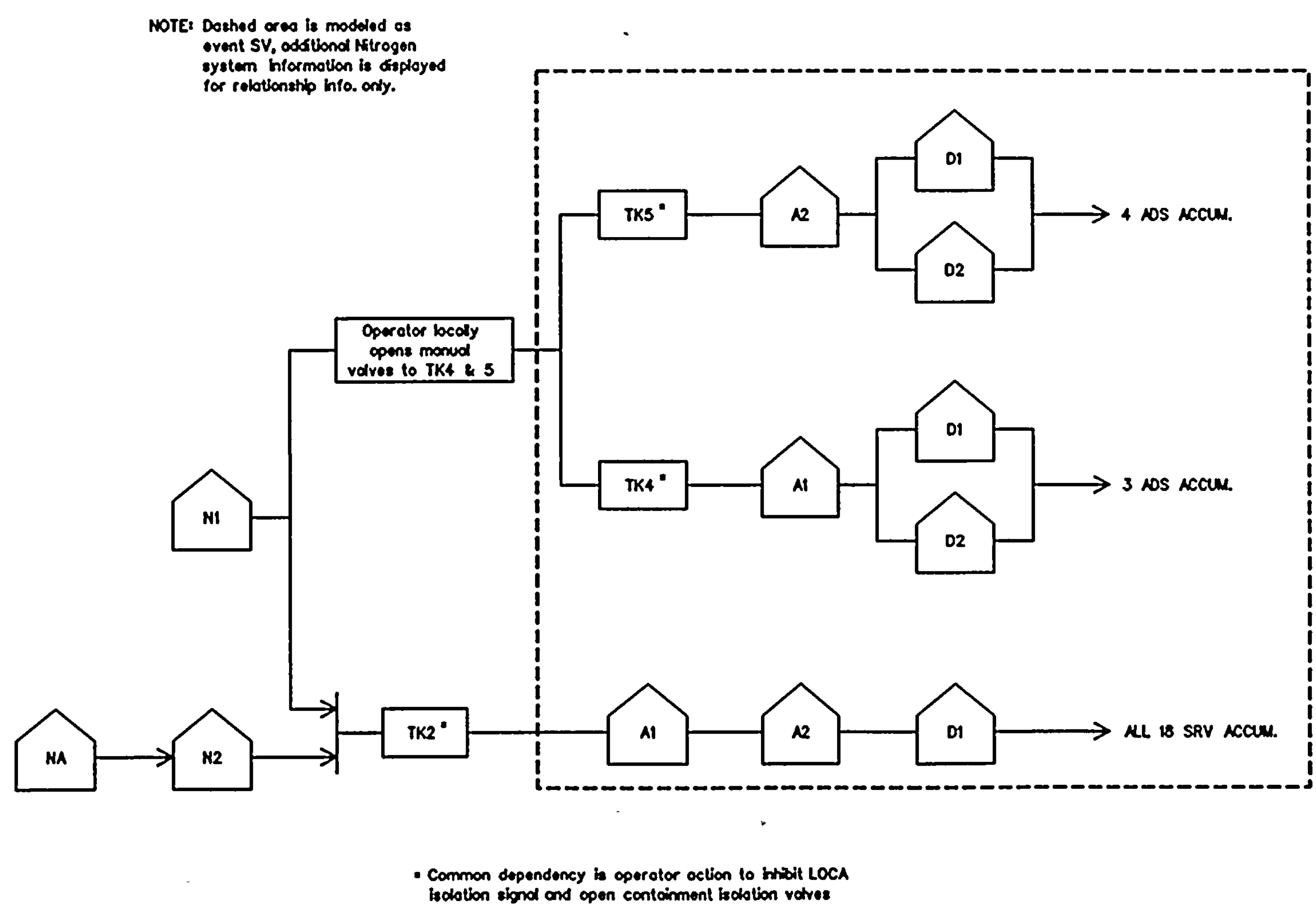


FIGURE 3.2.1.13-5
SRV DEPRESSURIZATION
SUPPORT SYSTEM SUCCESS DIAGRAM

I:\NMP2IPE\DGN\2S13.DGN Jul. 07. 1992 08:36:22

AUTOMATIC DEPRESSURIZATION
Component Block Descriptions

Block	Mark No. (Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
Common Nitrogen Supply							
	2GSN-V53	CHECK VALVE	FAILS TO OPEN	OPEN	N/A	N/A	N/A
	2GSN-V54	MANUAL VALVE	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2IAS-V1212	MANUAL VALVE	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2IAS-FLT9	FILTER	PLUGGED	N/A	N/A	N/A	N/A
	2IAS-V1213	MANUAL VALVE	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2IAS*SOV166	SOLENOID VALVE	FAILS TO OPEN	CLOSED	OPEN	2SCM*PNL102A	CLOSED
	2IAS*SOV166	SOLENOID VALVE	TRANSFERS CLOSED	OPEN	OPEN	2SCM*PNL102A	CLOSED
	2IAS*SOV184	SOLENOID VALVE	FAILS TO OPEN	CLOSED	OPEN	2SCM*PNL302B	CLOSED
	2IAS*SOV184	SOLENOID VALVE	TRANSFERS CLOSED	OPEN	OPEN	2SCM*PNL302B	CLOSED
	2IAS*V178	L.O. MANUAL VALVE	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
Tank TK4 & Assoc. Valves							
	2IAS*TK4	NITROGEN RECEIVER TANK	RUPTURE/EXCESS. LEAK	N/A	N/A	N/A	N/A
	2IAS*SV19A	RELIEF VALVE	TRANSFERS OPEN	CLOSED	N/A	N/A	N/A
	2IAS*V137	MANUAL VALVE	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2IAS*SOVX181	SOLENOID VALVE	FAILS TO OPEN	CLOSED	OPEN	2SCM*PNL102A	CLOSED
	2IAS*V1161	MANUAL VALVE	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2IAS*SOVY181	SOLENOID VALVE	FAILS TO OPEN	CLOSED	OPEN	2SCM*PNL102A	CLOSED
	2IAS*PT181	PRESSURE TRANSMITTER	FAILS LOW	N/A	N/A	2CEC*PNL731	FAILS HIGH
	2IAS*SV20A	RELIEF VALVE	TRANSFERS OPEN	CLOSED	N/A	N/A	N/A
	2IAS*V174	MANUAL VALVE	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2IAS*SOV164	SOLENOID VALVE	FAILS TO OPEN	CLOSED	OPEN	2SCM*PNL102A	CLOSED
	2IAS*SOV164	SOLENOID VALVE	TRANSFERS CLOSED	OPEN	OPEN	2SCM*PNL102A	CLOSED
	2IAS*V448	CHECK VALVE	FAILS TO OPEN	OPEN	N/A	N/A	N/A
	2IAS*V179	VALVE	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
SRV/ADS Valve PSV121							
	2IAS*V471	CHECK VALVE	FAILS TO OPEN	CLOSED	N/A	N/A	N/A
	2IAS*TK32	ACCUMULATOR TANK	RUPTURE/EXCESS. LEAK	N/A	N/A	N/A	N/A
	2MSS*SOV121A	SOLENOID VALVE	FAILS TO OPEN	CLOSED	OPEN	2BYS*PNL201A	CLOSED
	2MSS*SOV121B	SOLENOID VALVE	FAILS TO OPEN	CLOSED	OPEN	2BYS*PNL201B	CLOSED
	2MSS*PSV121	PNEUMATIC VALVE	FAILS TO OPEN	CLOSED	OPEN	NITROGEN	CLOSED
SRV/ADS Valve PSV126							
	2IAS*V431	CHECK VALVE	FAILS TO OPEN	CLOSED	N/A	N/A	N/A
	2IAS*TK34	ACCUMULATOR TANK	RUPTURE/EXCESS. LEAK	N/A	N/A	N/A	N/A
	2MSS*SOV126A	SOLENOID VALVE	FAILS TO OPEN	CLOSED	OPEN	2BYS*PNL201A	CLOSED
	2MSS*SOV126B	SOLENOID VALVE	FAILS TO OPEN	CLOSED	OPEN	2BYS*PNL201B	CLOSED
	2MSS*PSV126	PNEUMATIC VALVE	FAILS TO OPEN	CLOSED	OPEN	NITROGEN	CLOSED
SRV/ADS Valve PSV127							
	2IAS*V421	CHECK VALVE	FAILS TO OPEN	CLOSED	N/A	N/A	N/A

AUTOMATIC DEPRESSURIZATION Component Block Descriptions							
Block	Mark No. (Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
	2IAS*TK33	ACCUMULATOR TANK	RUPTURE/EXCESS. LEAK	N/A	N/A	N/A	N/A
	2MSS*SOV127A	SOLENOID VALVE	FAILS TO OPEN	CLOSED	OPEN	2BYS*PNL201A	CLOSED
	2MSS*SOV127B	SOLENOID VALVE	FAILS TO OPEN	CLOSED	OPEN	2BYS*PNL201B	CLOSED
	2MSS*PSV127	PNEUMATIC VALVE	FAILS TO OPEN	CLOSED	OPEN	NITROGEN	CLOSED
Tank TK5 & Assoc. Valves							
	2IAS*TK5	NITROGEN RECIEVER TANK	RUPTURE/EXCESS. LEAK	N/A	N/A	N/A	N/A
	2IAS*SV19B	RELIEF VALVE	TRANSFERS OPEN	CLOSED	N/A	N/A	N/A
	2IAS*V138	MANUAL VALVE	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2IAS*SOVX186	SOLENOID VALVE	FAILS TO OPEN	CLOSED	OPEN	2SCM*PNL302B	CLOSED
	2IAS*V1162	MANUAL VALVE	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2IAS*SOVY186	SOLENOID VALVE	FAILS TO OPEN	CLOSED	OPEN	2SCM*PNL302B	CLOSED
	2IAS*PT186	PRESSURE TRANSMITTER	FAILS LOW	N/A	N/A	2CEC*PNL730	FAILS HIGH
	2IAS*SV20B	RELIEF VALVE	TRANSFERS OPEN	CLOSED	N/A	N/A	N/A
	2IAS*V175	MANUAL VALVE	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2IAS*SOV165	SOLENOID VALVE	FAILS TO OPEN	CLOSED	OPEN	2SCM*PNL302B	CLOSED
	2IAS*SOV165	SOLENOID VALVE	TRANSFERS CLOSED	OPEN	OPEN	2SCM*PNL302B	CLOSED
	2IAS*V449	CHECK VALVE	FAILS TO OPEN	OPEN	N/A	N/A	N/A
	2IAS*V180	VALVE	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
SRV/ADS Valve PSV129							
	2IAS*V581	CHECK VALVE	FAILS TO OPEN	CLOSED	N/A	N/A	N/A
	2IAS*TK38	ACCUMULATOR TANK	RUPTURE/EXCESS. LEAK	N/A	N/A	N/A	N/A
	2MSS*SOV129A	SOLENOID VALVE	FAILS TO OPEN	CLOSED	OPEN	2BYS*PNL201A	CLOSED
	2MSS*SOV129B	SOLENOID VALVE	FAILS TO OPEN	CLOSED	OPEN	2BYS*PNL201B	CLOSED
	2MSS*PSV129	PNEUMATIC VALVE	FAILS TO OPEN	CLOSED	OPEN	NITROGEN	CLOSED
SRV/ADS Valve PSV130							
	2IAS*V571	CHECK VALVE	FAILS TO OPEN	CLOSED	N/A	N/A	N/A
	2IAS*TK37	ACCUMULATOR TANK	RUPTURE/EXCESS. LEAK	N/A	N/A	N/A	N/A
	2MSS*SOV130A	SOLENOID VALVE	FAILS TO OPEN	CLOSED	OPEN	2BYS*PNL201A	CLOSED
	2MSS*SOV130B	SOLENOID VALVE	FAILS TO OPEN	CLOSED	OPEN	2BYS*PNL201B	CLOSED
	2MSS*PSV130	PNEUMATIC VALVE	FAILS TO OPEN	CLOSED	OPEN	NITROGEN	CLOSED
SRV/ADS Valve PSV134							
	2IAS*V546	CHECK VALVE	FAILS TO OPEN	CLOSED	N/A	N/A	N/A
	2IAS*TK36	ACCUMULATOR TANK	RUPTURE/EXCESS. LEAK	N/A	N/A	N/A	N/A
	2MSS*SOV134A	SOLENOID VALVE	FAILS TO OPEN	CLOSED	OPEN	2BYS*PNL201A	CLOSED
	2MSS*SOV134B	SOLENOID VALVE	FAILS TO OPEN	CLOSED	OPEN	2BYS*PNL201B	CLOSED
	2MSS*PSV134	PNEUMATIC VALVE	FAILS TO OPEN	CLOSED	OPEN	NITROGEN	CLOSED
SRV/ADS Valve PSV137							
	2IAS*V526	CHECK VALVE	FAILS TO OPEN	CLOSED	N/A	N/A	N/A

AUTOMATIC DEPRESSURIZATION Component Block Descriptions							
Block	Mark No. (Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
	2IAS*TK35 2MSS*SOV137A 2MSS*SOV137B 2MSS*PSV137	ACCUMULATOR TANK SOLENOID VALVE SOLENOID VALVE PNEUMATIC VALVE	RUPTURE/EXCESS. LEAK FAILS TO OPEN FAILS TO OPEN FAILS TO OPEN	N/A CLOSED CLOSED CLOSED	N/A OPEN OPEN OPEN	N/A 2BYS*PNL201A 2BYS*PNL201B NITROGEN	N/A CLOSED CLOSED CLOSED

System 14

Control Rod Drive

3.2.1.14 Control Rod Drive

3.2.1.14.1 System Function

The control rod drive hydraulic system (CRD) is primarily intended to be the source of the driving force for insertion or withdrawal of control rods, including scram drive. It must also maintain the rod positions.

CRD can also be used as an alternate method of coolant injection in an emergency scenario, which is the function of interest in this system summary. A simplified diagram of the CRD hydraulic system is provided in Figure 3.2.1.14-1.

3.2.1.14.2 Success Criteria

The CRD Pumps can inject to the RPV through seal leakage (approximately 230 gpm with both pumps running) which can be an effective source for RPV makeup several hours after a successful scram. CRD is presently modeled as a potentially successful injection source in event tree top event CF (see Section 3.2.1.24) under conditions where the containment overpressure failure has occurred due to loss of heat removal. This would be approximately 20 or more hours after the reactor is shutdown and when one pump would provide successful injection. Because containment failures can impact the availability of this system and there are large uncertainties with regard to CRD availability and operator actions, a point estimate value is used for the CRD equipment (i.e., no fault tree was developed since other factors dominate availability of the CRD).

3.2.1.14.3 Support Systems

The following summarizes support equipment required for major CRD equipment to function:

CRD Equipment	Support
CRD Feedpump 1A, 2RDS-P1A	2NNS-SWG014
CRD Feedpump 1B, 2RDS-P1B	2NNS-SWG015
CRD Feedpump 1A Heater, 2RDS-H1A	2SCA-PNL201
CRD Feedpump 1B Heater, 2RDS-H1B	2SCA-PNL201
CRD Water Throttle MOV, 2RDS-PV101	2NHS-MCC008
CRD Instrumentation Power	2VBS-PNLA102
CRD Indication Power	2VBS-PNLA101
CRD Temperature Recorder, 2RDSN07	2VBS-PNLA101

Also required are the condenser or condensate storage tanks as a water supply, as shown in Figure 3.2.1.14-2.

Reactor Building Closed Loop Cooling Water (RBCLC) supplies the cooling water for the CRD bearing and seal cooler.

3.2.1.14.4 System Operation

There are 2, 100% capacity pumps, with one pump normally running and the other in standby.

When all the control rods are inserted, the system can inject water into the Reactor Pressure Vessel (RPV), if necessary. The pumps can supply at total of 115 gpm each. Under an emergency condition, both pumps can be used simultaneously for a total injection of approximately 230 gpm.

3.2.1.14.5 Instrumentation and Controls

There are indications/alarms in the control room for the following:

CRD pump 1A/1B auto-trip
CRD pump 1A/1B suction pressure low
CRD pump 1A/1B trouble
CRD pump 1A/1B motor electrical fault
CRD pump 1A/1B motor overload
CRD pump 1B suction pressure low
CRD pump discharge header pressure low
CRD pumps suction filter differential pressure high
CRD charging water pressure low
CRD drive water filter differential pressure high
Control Rod temperature high
Control Rod drift
Control Rod over-travel
Control Rod Drive Accumulator trouble
SDV (scram discharge volume) drain AOV123 closed
SDV drain AOV130 closed
SDV vent AOV124 closed
SDV vent AOV132 closed

There is position indication for all rods.

The following CRD equipment can be operated from the control room:

Equipment Description, ID	Panel
Control Rod Drive Hydraulic pump 1A, 2RDS-P1A	P603
Control Rod Drive Hydraulic pump 1B, 2RDS-P1B	P603
Drive water pressure control valve, 2RDS-PV101	P603

There are also controls for each control rod and a scram insert control.

3.2.1.14.6 Technical Specifications

With no running CRD pumps or more than 1 inoperable scram accumulator with its rod withdrawn, shutdown immediately. Otherwise, insert the rod and isolate it from moving.

With a CRD pump running and an inoperable scram accumulator, repair the scram accumulator within 8 hours or be in hot shutdown within 12 hours.

3.2.1.14.7 Surveillance Testing and Maintenance

As mentioned above, no fault trees are being developed for the Control Rod Drive System. Testing and maintenance frequencies are considered to have a minor effect on system availability.

3.2.1.14.8 References

PID's as referenced on simplified drawing
Technical Specifications
USAR
N2-OP-30

3.2.1.14.9 Initiating Event Potential

Unnecessary scrams or shutdowns are included in the transient initiating event frequency as described in Section 3.1.1.

3.2.1.14.10 Equipment Location

The hydraulic control units (HCU) are located under the reactor vessel, one for each control rod (185 in all). The HCUs consist of directional control valves and piping, scram inlet and outlet valves, scram accumulator, charging and exhaust water header, cooling water header, and instrument air header. The CRD pumps and associated equipment are in the reactor building on elevation 215'.

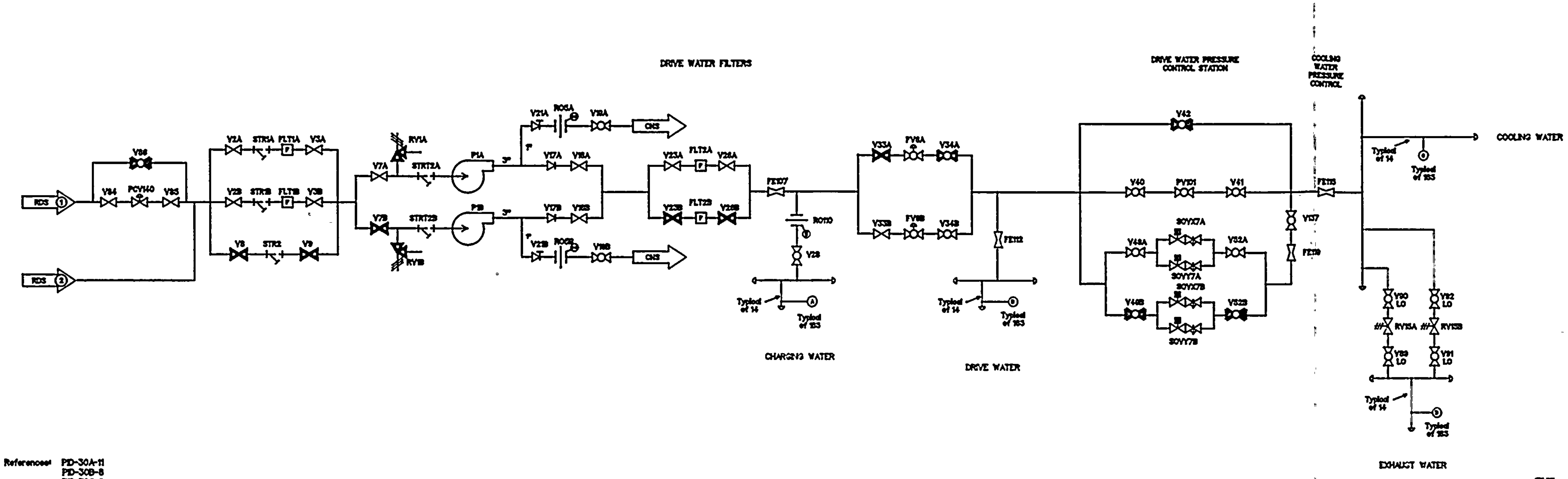
3.2.1.14.11 Operating Experience

There were no particular outstanding operational events relevant to this study. Plant specific component operational data is detailed in section 3.3.2.

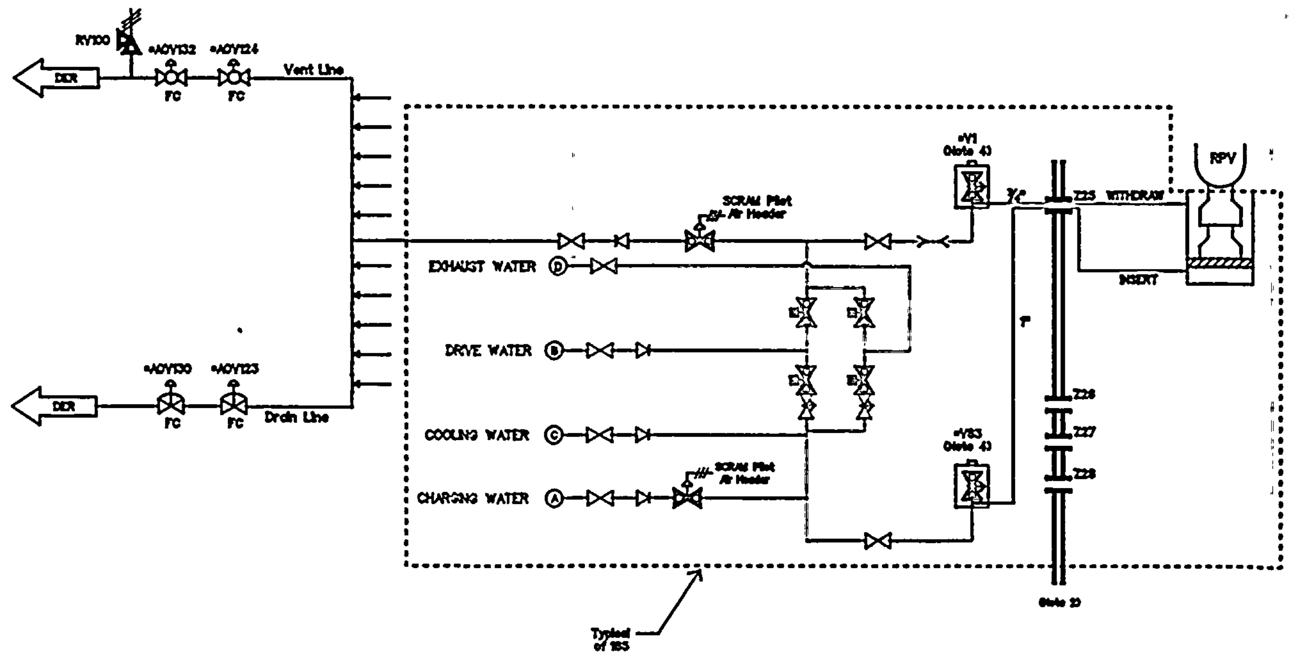
3.2.1.14.12 Modeling Assumptions

1. As described above under success criteria, a point estimate of system availability over 24 hours is used. Development of fault trees and their quantification is not deemed necessary at this time.

Rev 0 (7/93)



References: PID-30A-11
PID-30B-8
PID-30C-9



Note 1. All equipment IDs start with the prefix 2RDS- unless otherwise specified

Note 2. The penetration show is typical of 370, All enter through one of the following: Z25, Z26, Z27 or Z28.

Note 3. A is Charging Water; B is Drive Water; C is Cooling Water; D is Exhaust Water; E is Insert; F is Withdraw; G is SCRAM Discharge; H is SCRAM Pilot Air Header

Note 4. There are 185 each of V1 and V83, they are Block type ANGLE, NEEDLE, VENT valves.

- Common Equipment
- Charging Water Lines
- Drive Water Lines
- Cooling Water Lines
- Exhaust Water Lines
- Insert / Withdraw Lines

SI
APERTURE
CARD

Also Available On
Aperture Card

Figure 3.2.1.14-1
ControlRod Drive Hydraulic System

I:\NMP2IPE\DGN\2S14.DGN Jul. 07, 1992 12:56:07

9208050191-24

Rev 0 (7/92)

SI APERTURE CARD

Also Available On Aperture Card

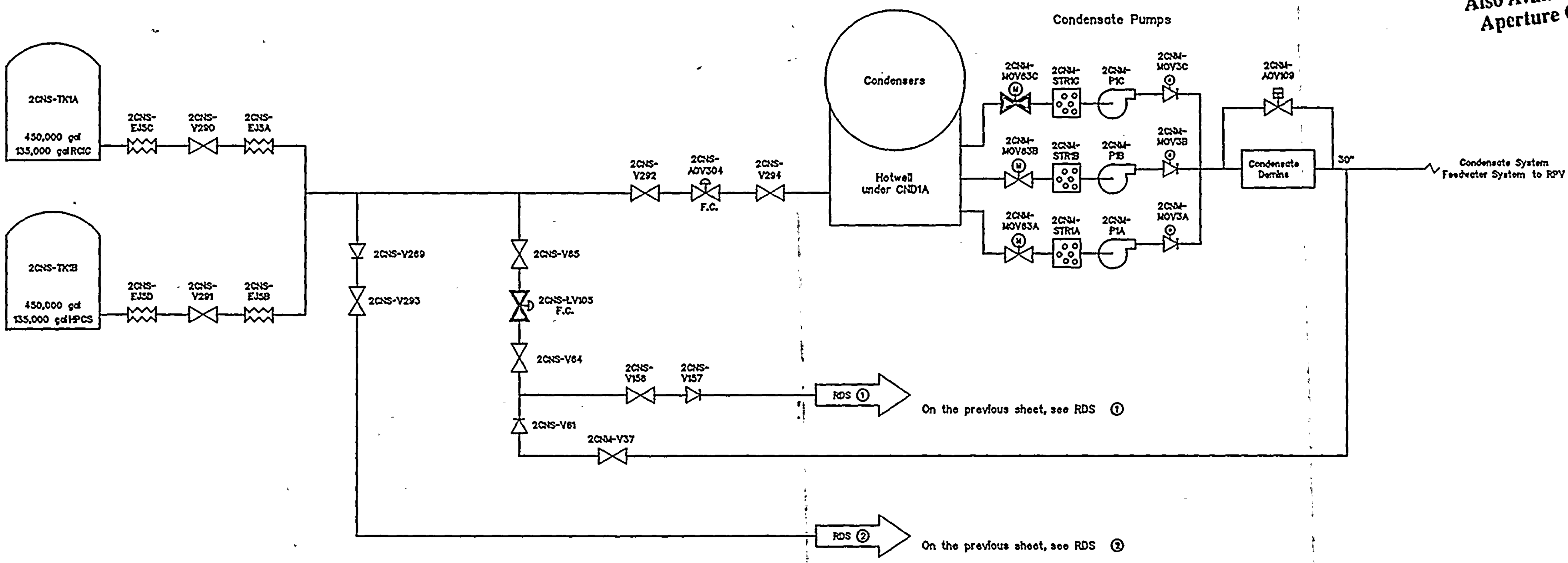


Figure 3.2.1.14-2
Water to the Control Rod Drive System from Condensate and Condensate Transfer Systems

I:\WP2IPE\DGN\2S14.DGN Jul. 07, 1992 10:43:06

9208050191-25

System 15

Reactor Protection

3.2.1.15 Reactor Protection System

3.2.1.15.1 System Function

The Reactor Protection System (RPS) is designed to prevent excessive fuel cladding or Reactor Coolant Pressure Boundary (RCPB) damage following abnormal reactor plant operating transients.

3.2.1.15.2 Success Criteria

The reactor protection system monitors selected reactor plant process parameters. When these parameters exceed a preselected setpoint, the RPS generates signals to automatically shutdown the reactor by rapidly inserting all reactor control rods (scram), which will drive the reactor subcritical.

Event tree top event RQ models automatic scram in the LOCA event tree models. The Redundant Reactivity Control System (RRCS), Alternate Rod Insertion (ARI), and reactor recirculation pump trip functions provide redundancy to the electrical portion of RPS in top event RQ.

Event tree top events QM and QE model the mechanical and electrical portions of RPS in the ATWS model. Transient sequences are evaluated in the ATWS model. RRCS and its actuated functions are included in the ATWS model as separate top events as described in Section 3.2.1.16.

Success of top event MS is that the operator has placed the mode switch in shutdown, as required by the EOPs, immediately after the reactor scram. Should the operator fail to accomplish this action, it is assumed that the operator cannot recover from this error before the RPV is subsequently depressurized. Therefore, upon either ADS actuation or emergency RPV blowdown, the MSIVs are assumed to close, isolating the RCS from the condenser (i.e., top event CN is set to failure).

3.2.1.15.3 Support Systems

Power supplies for the RPS trip channels consists of two 120V AC, 1-phase, Uninterruptible Power Supply (UPS) Systems. Trip system A is fed from normal UPS 2VBB-UPS3B which is normally powered from 2NJS-PNL402. Trip system B is fed from normal UPS 2VBB-UPS3A, which is normally energized from 600V AC non-safety-related lighting panel 2LAT-PNL100. In case of a loss of the normal supply, power is automatically provided by 125V DC non-safety related switchgear. The batteries are capable of feeding the UPS for at least 2 hours. The RPS input signals are de-energized to actuate and are therefore fail-safe on loss of 120V AC.

3.2.1.15.4 System Operation

The reactor protection system consists of two independent, functionally identical trip systems (A and B). Each trip system is divided into two independent, functionally identical trip channels (A1, A2; B1, B2). These four channels consist of the sensors, relays, contacts, switches, and trip units which initiate an automatic scram to prevent the reactor from operating under unsafe, or potentially unsafe conditions.

Each RPS channel receives an input from at least one independent sensor for each critical reactor parameter. When a parameter is determined to be out of its normal operating or transient range (e.g., reaches its trip setpoint), and a sufficient number of sensors reach this unsafe condition, a scram signal will be generated from the RPS logic. The scram signal will cause electrical power to be interrupted to the scram pilot solenoid valves on each Control Rod Drive (CRD) Hydraulic Control Unit (HCU), and all control rods will be rapidly inserted into the reactor core, shutting down the reactor.

The following variables are monitored to provide protective input to the RPS indicating the need for reactor scram:

- Neutron flux,
- Reactor vessel high pressure,
- Reactor vessel low water level,
- Turbine stop valve closure,
- Turbine control valve fast closure,
- Main steam line isolation,
- Scram discharge volume high level,
- Drywell high pressure, and
- Main steam line high radiation.

Note that the MSIVs are closed automatically from measured parameters such as high steam flow, high steam line radiation, low reactor water level, low vacuum, high steam tunnel temperature, and low steam line pressure.

The RPS is designed (including logic and actuated devices) to be fail safe. A loss of electrical power or air supply will not prevent a reactor scram.

Two methods are provided for manually causing a reactor scram. Four pushbuttons are provided, each of which will de-energize its respective RPS channel when pushed. Placing the Reactor Mode Switch into the SHUTDOWN position will also cause a manually actuated reactor scram, by de-energizing all RPS channels. The Reactor Mode Switch, located on the Reactor Control Panel (P603), is a 5-position key-lock switch. The position of this switch establishes the conditions under which the reactor may operate.

3.2.1.15.5 Instrumentation and Controls

There are 2 groups of four white indicating lights on the vertical section of the reactor control panel P603, on either side of the rod and detector display. Each pilot scram valve solenoid light is associated with one of the four rod groups in trip system A or B. The group on the left is associated with the "A" solenoids for the scram pilot valves, and the group on the right with the "B" solenoids. The lights are normally lit, and extinguish as a result of the de-energization of the associated channel sensor relays K14A through H, J-N, P, R, and S. This shows which RPS trip system has tripped and is sending a scram signal to the control rods.

There are no other indications directly associated with the RPS system. However, there are many indications available to the operator which have a direct bearing on RPS operation. For example, the reactor vessel instrumentation system provides indication of reactor vessel water level and pressure and the control rod drive hydraulic system provides scram discharge volume levels, etc.

The following RPS indicating lights are on reactor control panel P603:

- Pilot Scram Valve Solenoid Lights
- Scram Discharge Volume Vent Valves Position
- Scram Discharge Volume Drain Valves Position
- Trip Unit A/C (B/D) in Calib. / Gross Failure
- RPS A(B) Manually Out of Service
- Reactor Scram Trip Logic A1/A2 (B1/B2)
- Turbine Stop Valve Closure
- Turbine Control Valve Closure
- Recirculation Pump Trip System A (B)
- 24V DC Power Division 1, 3 (2, 4)

The following RPS Indicating Lights are on Control Rod Test Panel P610:

- Rod Scram Timing Test Lights (185)
- Generator A Feed Available
- Generator B Feed Available
- Alternate A Feed Available
- Alternate B Feed Available
- RPS Annunciators
- Division 1(2) Drywell Pressure High Scram
- Division 1(2) Neutron Monitoring System Scram
- Division 1(2) Reactor Pressure High Scram
- Division 1(2) Turbine Control Valve Fast Closure Scram
- Division 1(2) Reactor Water Level Low Scram
- Division 1(2) Turbine Stop Valve Closure Scram
- Division 1(2) Main Steam Line Radiation High Scram
- Division 1(2) MSIV Closure Scram
- Division 1(2) Scram Dump Volume High Level Scram
- Division 1(2) Reactor Scram
- Division 1(2) Manual Scram
- Division 1(2) RPS System Inoperable
- Division 1(2) Reactor Recirc. Pump Trip Inoperable
- Division 1(2) Turbine Control & Stop Valve Closure Bypassed
- Division 1(2) Manual Switch Scram Permissive

- Division 1(2) MSIV Closure Scram Bypassed
- RPS A1(B1) Trip Unit Card Out of File/Power Failed
- RPS A2(B2) Trip Unit Card Out of File/Power Failed
- RPS A1(B1) Isolator Power Fail/Input Card Out
- RPS A2(B2) Isolator Power Fail/Input Card Out
- RPS A2(B2) Isolator Output Card Out of File
- RPS A2(B2) Isolator Output Card Out of File
- Division 1(2) Mode Switch Shutdown Bypassed
- RPS A(B) Trip Unit in Calibrate or Gross Failure
- Division 1(2) Scram Dump Volume High Water Level Trip Bypassed
- RPS Non-Divisional Trip Unit in Calibrate or Gross Failure
- Drywell Pressure High/Low

The RPS Motor-Generator (M-G) controls are located on a local panel in the same room with the respective M-G set. The controls consist of a START-STOP switch and voltage regulator.

The RPS power source select switch is located on the control rod test panel, P610, in the control room. The switch may be operated at any time, but the operator must be aware that a half-scram will occur whenever this switch is operated, as it is a dead-bus transfer.

On trip system A(B) protection system panels P609 (P611), located in the control room, are several RPS controls. The MSIV test switches allow testing of the MSIV scram sensor logic circuits. The turbine stop valve closure test switches allow testing of the turbine stop valve scram sensor logic circuits.

The reactor control panel P603 contains the remaining RPS controls. The reactor mode switch is located on the desk section of P603 to the right of the Rod Select Module. The four reactor scram manual trip switches will de-energize the respective reactor scram trip logic circuits.

If the scram signal(s) which initiated the scram has cleared, and no other scram signal exists, the scram reset switches will permit re-energization of their respective logic channels.

The four discharge volume high water level bypass switches are used to bypass the SDV high level scram signal so the scram trip logic can be reset. Due to the 10 second time delay for enabling the scram reset logic, with the reactor vessel pressurized, the SDV always fills to greater than its scram setpoint before the scram trip logic can be reset. One bypass switch exists for each RPS channel and logic requires the reactor mode switch to be in either SHUTDOWN or REFUEL for this bypass to function.

The two discharge volume isolation test switches allow testing the discharge volume isolation valve solenoids.

The bypass selector switches for the neutron monitoring system (1 for SRM, 2 for IRM, 2 for APRM) allow bypassing a neutron monitor channel in RPS trip system A or B.

Located on each control rod drive hydraulic control unit (local) are two toggle switches. These rod scram test switches are used to test the individual coil of the scram pilot solenoid valve, and to perform surveillance testing of the individual control rod scram times.

3.2.1.15.6 Technical Specifications

Technical specification requirements for reactivity control systems and RPS instrumentation are described in technical specification sections 3.1 and 3.3.1.

3.2.1.15.7 Surveillance Testing and Maintenance

Technical specification surveillance requirements are provided in technical specification sections 4.1 and 4.3.1.

3.2.1.15.8 References

Burns, E.T., "Reassessment of the BWR Scram Failure Probability." June 4-8, 1989. ANS Transaction Volume 59, TANSA059 1-366 (1989), ISSN: 0003-018X.

N2-OP-97, Rev. 3: RPS Operating Procedure

Drawings:

GE 732E103AF, Rev. 7:	Nuclear Boiler
GE 761E291AF, Rev. 2:	Reactor Recirculation System
GE 761E952AF, Rev. 9:	Control Rod Drive Hydraulic System
GE 761E354, Rev. 5:	Control Rod Drive Hydraulic System
GE 732E118A, Rev. 9:	Neutron Monitoring System
GE 807E162TY, Rev. 2:	Startup Range Neutron Monitoring
GE 761E596C, Rev. 2:	Neutron Monitoring System Elem.
GE 807E163TY, Rev. 11:	Power Range Neutron Monitoring System Elem.
GE 807E166TY, Rev. 9:	Reactor Protection System Elem.
GE DL807E166TY, Rev. 10:	Reactor Protection System Elem.
GE 732E170A, Rev. 5:	Reactor Protection System Elem.
GE 115D6268TY, Rev. 7:	Reactor Protection System MG Set Control Elem.

3.2.1.15.9 Initiating Event Potential

Spurious scrams are included as initiating events.

3.2.1.15.10 Equipment Location

Equipment is located in the Reactor Building. Power supplies and control logic are in the Reactor and Control Buildings.

3.2.1.15.11 Operating Experience

There were no outstanding operational events relevant to this study. Plant specific component operational data is detailed in Section 3.3.2.

3.2.1.15.12 Modeling Assumptions

1. A simplified model is presently used to model failure of scram. Reference 1 indicates that the unavailability of the RPS scram has not changed from NUREG-0460 based on precursor data. The following electrical and mechanical contributions are suggested based on precursor events:

Electrical	2.6E-5
Mechanical	4.3E-6
TOTAL	3.0E-5/Demand

   The above scram failure estimates are treated as mean values and these values are assumed to be independent of support systems. This should be a reasonable assumption since the scram system is a de-energize to actuate system (fail-safe) and utilizes, for the most part, dedicated input devices. At NMP2, the alternate rod insertion (ARI) system is independent of the scram system and is automatic. The redundant reactivity control system (RRCS) automatically actuates ARI (energize to actuate) as described in section 3.2.1.16.

3.2.1.15.13 Logic Model and Results

The fault tree(s) is included in Tier 2 documentation, and is available upon request. Quantitative results are summarized in Section 3.3.5 (Tier 1).

System 16

Redundant Reactivity Control

3.2.1.16 Redundant Reactivity Control System (RRCS)

3.2.1.16.1 Function

The redundant reactivity control system (RRCS) mitigates ATWS events by providing a backup to the Control Rod Drive (CRD) system and Reactor Protection System (RPS). The RRCS is a two division (redundant) protection system, each capable of performing all RRCS initiation functions. Each division also has two channels. RRCS initiates the following functions:

- Alternate Rod Insertion (ARI) depressurizes the scram discharge air header through valves separate from the RPS scram valves.
- Reactor Recirculation Pump Trip (RPT) either downshifts pumps to a Low Frequency Motor Generator (LFMG) or trips power to the pump motors. This reduces core flow thereby reducing power generation.
- Feedwater Runback stops feedwater flow into the reactor vessel which reduces core subcooling thereby reducing power generation.
- Standby Liquid Control System (SLS) injects a neutron absorbing solution (sodium pentaborate) into the core to stop power generation. The SLS system is described in Section 3.2.1.12.

3.2.1.16.2 Success Criteria

The RRCS and the functions initiated by RRCS are included in the ATWS event tree model, as detailed below. The following summarizes success criteria:

At least 1 of 2 divisions of RRCS must actuate for automatic operation. Both channels in a division must be actuated for division actuation. Either high RPV pressure or low RPV water level (Level 2) signals will automatically initiate RRCS (Top events C1 and C2). A RRCS permissive, Average Power Range Monitor above downscale trip (APRM not downscale permissive), is required for feedwater runback, Low Frequency Motor Generator (LFMG) trip, SLS, and Clean-up system isolation. A simplified success diagram is provided in Figure 3.2.1.16-1.

ARI (Top Event RI) - One of two valves at each of four air vent locations open when energized to cause ARI actuation. The ARI valves are designed to vent the scram air header to cause all rods to begin scramming within 15 seconds. A success diagram is provided in Figure 3.2.1.16-2. A simplified diagram of the ARI valves and interface with scram discharge volume is provided in Figure 3.2.1.16-3.

RPT (Top event RT) - Both pumps will transfer from high-to-low speed (LFMG) on high RPV pressure. Either RRCS division will transfer both pumps. Both pumps are tripped from their normal power supply by a RPV Level 2 signal or high dome pressure. If the transfer to the low frequency generator is made on high dome pressure, the LFMG will trip after 25 seconds if the APRM permissive is satisfied. Either RRCS division will trip both pump motors. An electrical diagram is depicted on Figure 3.2.1.16-4 and a success diagram is provided in Figure 3.2.1.16-5.

Feedwater Runback (Top event FT) - In this event, either the level control valve closes or the minimum bypass valve opens for each main feedpump discharge path. Figure 3.2.1.16-6 provides a simplified sketch showing the feedwater control valves and minimum flow valve to the condenser. Figure 3.2.1.16-7 shows the success diagram.

In addition, the following operator action models are included in the ATWS event tree model:

Top Event	Success Criteria
CH	Operator controls level and does not flush boron out of the core.
WL	Operator, instructed to lower level, drops level too low, and core damage results.
MO	Operator bypasses low RPV level MSIV isolation.

3.2.1.16.3 Support Systems

RRCS depends on 125V DC to operate:

- 125V DC Division I (2BYS*PNL202A) is required for RRCS Division I operation.
- 125V DC Division II (2BYS*PNL202B) is required for RRCS Division II operation.

2VBS*PNL301A and 301B (120V AC) provide power to RRCS indication and status, but are not required for automatic operation.

ARI solenoid valves depend on the following for actuation:

- 125V DC Division I (2BYS*PNL202A) and RRCS Division I for Division I valves
- 125V DC Division II (2BYS*PNL202B) and RRCS Division II for Division II valves

Feedwater runback depends on non-divisional 125V DC, normal AC, and RRCS:

- Either division of RRCS is capable of activating feedwater runback.

- Non-divisional 125V DC (2BYS-PNLB101) is required to actuate feedwater control valves closure.

- Normal AC power (2NHS-MCC003 and 2SCI-PNLA101) is required.

Recirc pump trip depends on 125V DC and RRCS:

- Division I pump trips (1 breaker per pump) depends on 125V DC Division I (2DMS*MCCA1) and RRCS Division I.

- Division II pump trips (1 breaker per pump) depend on 125V DC Division II (2DMS*MCCB1) and RRCS Division II.

The low frequency motor generator trip depends on 125V DC (non-divisional) for tripping of the LFMG motor breakers and the LFMG generator breakers. Either division of RRCS will trip both breakers.

3.2.1.16.4 System Operation

There are three initiators: reactor high pressure of 1050 psig, reactor vessel low water level of 108.8 inches (Level 2), or manual. The results differ, depending on the initiator.

Reactor High Pressure Initiation
Alternate Rod Insertion is initiated, the 60 Hz. circuit breakers to the Recirc. Pumps are tripped, and the 15 Hz circuit breakers to the Recirc. Pumps are closed. After 25 seconds of high reactor pressure and with the APRM not downscale permissive satisfied, the 15 Hz. circuit breakers are tripped, feedwater runback is initiated and the feedwater minimum flow valves are failed open. After an additional 73 seconds with continued reactor high pressure and the same APRM permissive, Standby Liquid Control Injection is initiated, and Reactor Water Clean-up system is isolated.

Reactor Low Water Level Initiation
Alternate Rod Insertion is initiated and all 60 Hz. and 15 Hz. circuit breakers to the Recirc. Pumps are tripped. After 98 seconds and with continued reactor low water and with the APRM not downscale permissive satisfied, Standby Liquid Control injection is initiated and Reactor Water Clean-up system is isolated. There are no feedwater system signals generated for reactor low water.

Manual Initiation
Alternate Rod Insertion is initiated. After 98 seconds and with APRMs not downscale, Standby Liquid Control initiation and Reactor Water Clean-up system isolation are initiated. The manual initiation of RRCS does not result in automatic Recirc. Pump trip from either 60 Hz or from 15 Hz. circuit breakers, nor does it result in any automatic feedwater system action being taken.

Standby Liquid Control (SLC)
Both Standby Liquid Control trains are activated by either RRCS division, RRCS being actuated by RPV low water level (level 2), RPV high pressure, or manual RRCS initiation. After 98 seconds have elapsed and the APRM permissive is successful, SLC will automatically inject into the RPV.

3.2.1.16.5 Instrumentation and Controls

The RRCS, being essentially an interfacing logic system, has no indications directly associated with the system, except for indicating lights and alarms.

There are a number of amber indicating lights on Reactor Control Panel P603. They show:

- RRCS ARI Initiated, Division I (2)
- RRCS Manual Initiation, Division I (2)
- RRCS ARI Ready for Reset, Division II, (2)
- RRCS Ready for Reset, Division II, (2)
- RRCS Test Fault (Essential Logic Failure), Division I (2)
- LFMG Transfer, Division I (2)
- RRCS FW Runback Initiated, Division I (2)

There are a number of amber indicating lights on Reactor Core Cooling Control Board-P601, Redundant Reactivity Control (RRCS Logic) Panel C22-P001(2). They show:

- RRCS Manual Initiation Armed, Division I (2)
- RRCS Manual Initiation Division I (2), Channel A, or Division I (2), Channel B
- High Dome Pressure, Division I (2), Channel A or Division I (2), Channel B
- Low-Low Water Level (Level 2) Trip, Division I (2) Channel A, or Division I (2) Channel B
- RRCS Potential ATWS, Division I (2)
- RRCS ARI Initiated, Division I (2)
- RRCS RWCU (WCS) Isolated, Division I (2)
- RRCS Confirmed ATWS, Division I (2)
- RRCS ARI Ready for Reset, Division I (2)
- RRCS Ready for Reset, Division I (2)
- RRCS Test Fault (Self-Test System Failure), Division I (2)
- RRCS ATM Calibration or Gross Failure, Division I (2)
- RRCS Trouble, Division I (2)
- LFMG Transfer, Division I (2)
- RRCS Recirc. Pumps Tripped, Division I (2)
- RRCS FW Runback Initiated, Division I (2)
- SLS Storage Tank Low Level, Division I (2)
- RRCS Out of Service, Division, (2)

There are three colors of RRCS Annunciators on Reactor Control Panel P603. They are red, amber and white. The annunciators are:

- RRCS Potential ATWS, Division I (2)
- RRCS Recirc. Pumps Tripped, Division I (2)
- RRCS FW Runback Initiated, Division I (2)
- RRCS Confirmed ATWS, Division I (2)
- SLCS Storage Tank Low Level, Division I (2)
- RRCS Out of Service, Division I (2)
- RRCS Manual Initiation, Armed, Division I (2)
- RRCS Trouble, Division I (2)

- RRCS Potential ATWS, Division I (2)
- RRCS Confirmed ATWS, Division I (2)
- RRCS Manual Initiation, Division I (2)
- SLS Storage Tank Low Level, Division I (2)
- RRCS Out of Service, Division I (2)
- RRCS Manual Initiation Armed, Division I (2)

The RRCS has the capability of being manually initiated from the Reactor Control Panel (P603). There are 4 pushbuttons that have a collar that needs to be armed first. Both pushbuttons in either RRCS Division must be armed and depressed to initiate the system.

In order to test the RRCS-ARI function, test switches are provided on the RRCS Control Panels C22-P001 and P002 located in the Relay Room.

Resetting the RRCS-ARI logic is accomplished with the ARI Reset switches. They are pushbutton-type switches and are located at the RRCS portion of P603.

3.2.1.16.6 Technical Specifications

There are two channels per trip system. With one OPERABLE channel per trip system for one or both trip systems, place the inoperable channel(s) in the tripped condition within 1 hour. With no OPERABLE channel per trip system:

- If the inoperable channels consist of one reactor vessel water level channel and one reactor vessel pressure channel, declare the Trip system INOPERABLE if not restored in 2 hours.

- If the inoperable channels include two reactor vessel water level channels or two reactor vessel pressure channels, declare the Trip system INOPERABLE.

With one Trip system INOPERABLE, restore to OPERABLE within 72 hours or be in at least STARTUP within the next 6 hours.

With both Trip systems INOPERABLE, restore at least one Trip system to OPERABLE within 1 hour or be in at least STARTUP within the next 6 hours.

3.2.1.16.7 Surveillance Testing and Maintenance

The RRCS is continuously checked by a solid state microprocessor-based self-test system. This system checks RRCS sensors, logic, and protective devices.

Surveillance requirements for water level and for pressure instrumentation are:

- Channel checks every 12 hours
- Channel functional tests every 31 days
- Trip unit setpoint calibration every 31 days
- Channel calibration every 550 days are required.

3.2.1.16.8 References

N2-OP-36B, Rev. 1 Redundant Reactivity Control System
FSAR Sections 7.6.1.8
FSAR Section 7.6.2.8

Technical Specifications 3/4.3.4.1 ATWS Recirculation Pump Trip System Instrumentation.
GEK 83282A, 90428
Drawings: RRCS 944E309TY May 1985 Sheets 1 - 41
Reactor Recirc. F61E791TY Sheets 1 - 30
ESK - 5RCS05-10

3.2.1.16.9 Initiating Event Potential

Spurious operation of RRCS or its functions could cause a plant trip. This frequency is low in comparison to existing initiation.

3.2.1.16.10 Equipment Location

The instrumentation and controls are located in the Control Room and in the relay Room on Panels P603 and C22-P001(2).

3.2.1.16.11 Operating Experience

A modification, apparently to address electrical spiking, (PN2Y87MX146) was canceled. There appears to be a problem with spiking which can initiate this system when a Division, taken out for maintenance, is restored to service. Actions are described in N2-OP-36B.

3.2.1.16.12 Modeling Assumptions

1. The redundant reactivity control system is modeled as two independent redundant subsystems identified as top events C1 and C2. Some RRCS activation functions such as ARI, Recirc Pump Trip or LFMG trip are initiated by either low vessel level or a high dome pressure, while LFMG transfer and Feedwater Runback (FWRB) are initiated solely on high dome pressure.

   To simplify the model the following assumptions are made:

   a. RRCS model has only one transmitter for each channel. This is conservative for the case where either a level or pressure signal is sufficient for RRCS initiation.

   b. The APRM not down scale trip permissive is not modeled. This permissive is normally satisfied during operation. If this function failed to operate during a normal scram the RRCS would be initiated. If it failed low (removing RRCS permissive) it would inhibit standby liquid control system and Feedwater Runback initiation, while all other RRCS functions are initiated. However, for this to occur four out of eight (4/8) neutron monitoring channels would have to fail low. Since these failures have control room indication and alarms, it is highly unlikely that four out of eight neutron monitoring channels could go downscale causing a downscale trip without prompt operator action.

2. The LFMG transfer is not modeled because any signal that initiates LFMG transfer or trip would also cause a recirc pump trip at the same time.

   The LFMG transfer is not modeled because:

   - Failure to transfer is equivalent to a recirc pump trip.

   - The transfer signal (low vessel level time delayed with APRM permissive) results in a LFMG trip after a time delay.

   Failure of Non-Divisional 125V DC is not modeled as a failure of the LFMG to trip. This is because if the 125V DC bus were not available the LFMG would not start at all. Then the recirc pumps would trip from normal speed and coast down to speed.

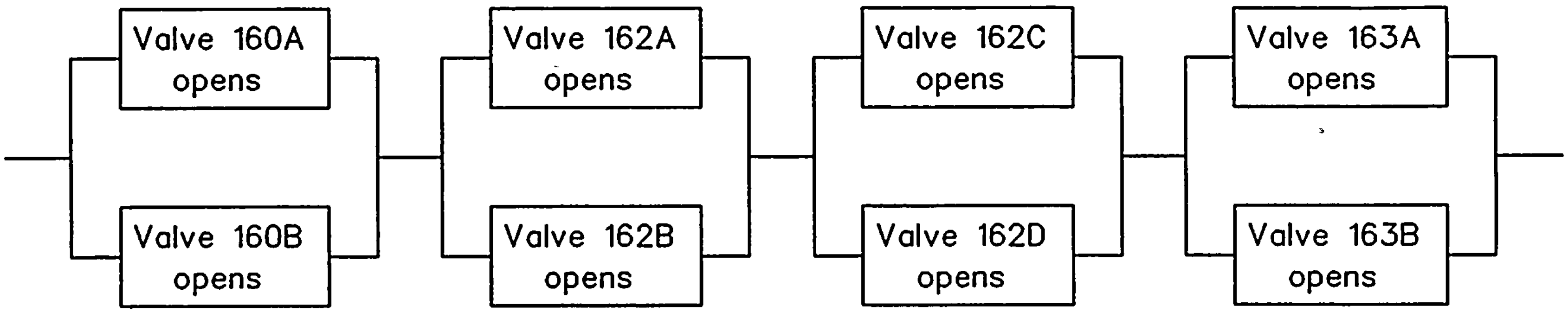
3. The Feedwater Runback is modeled as requiring flow to the reactor from all three (3) feedwater injection paths to be terminated by either closure of the level control valve or opening the bypass valve to the condenser.

4. The Low Frequency Motor Generator (LFMG) trip is modeled as being successful if both MG sets 2RCS-MG1A and 2RCS-MG1B are tripped. Each MG set can effectively be tripped if either the generator output breaker (2NPS-SWG004 or SWG005) or MG motor breaker (2NNS-SWG011 or SWG013) are opened. Each breaker can be tripped by either division of RRCS signals.

5. Loss of normal AC fails the feedwater runback function. However, it also causes loss of feedwater flow (pumps) and therefore is considered a guaranteed success of feedwater runback.

3.2.1.16.13 Logic Model and Results

The fault tree(s) is included in Tier 2 documentation, and is available upon request. Quantitative results are summarized in Section 3.3.5 (Tier 1).



ARI

Valves are energized to actuate

Valves	RRCS	125VDC
160A, 162A, 162C & 163A	C1	Div I
160B, 162B, 162D & 163B	C2	Div II

Figure 3.2.1.16-2
Alternate Rod Insertion

Rev 0 (7/92)

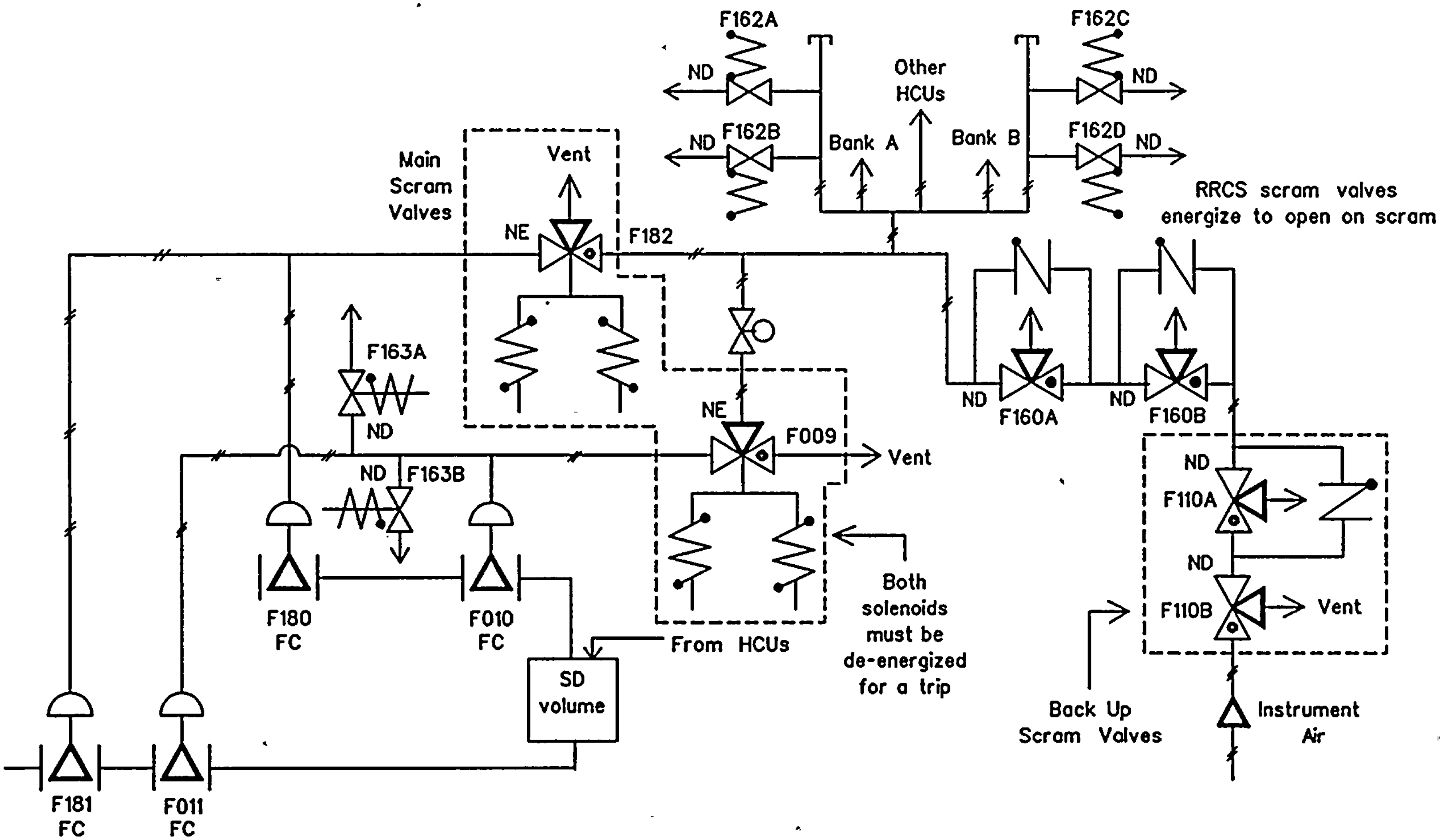
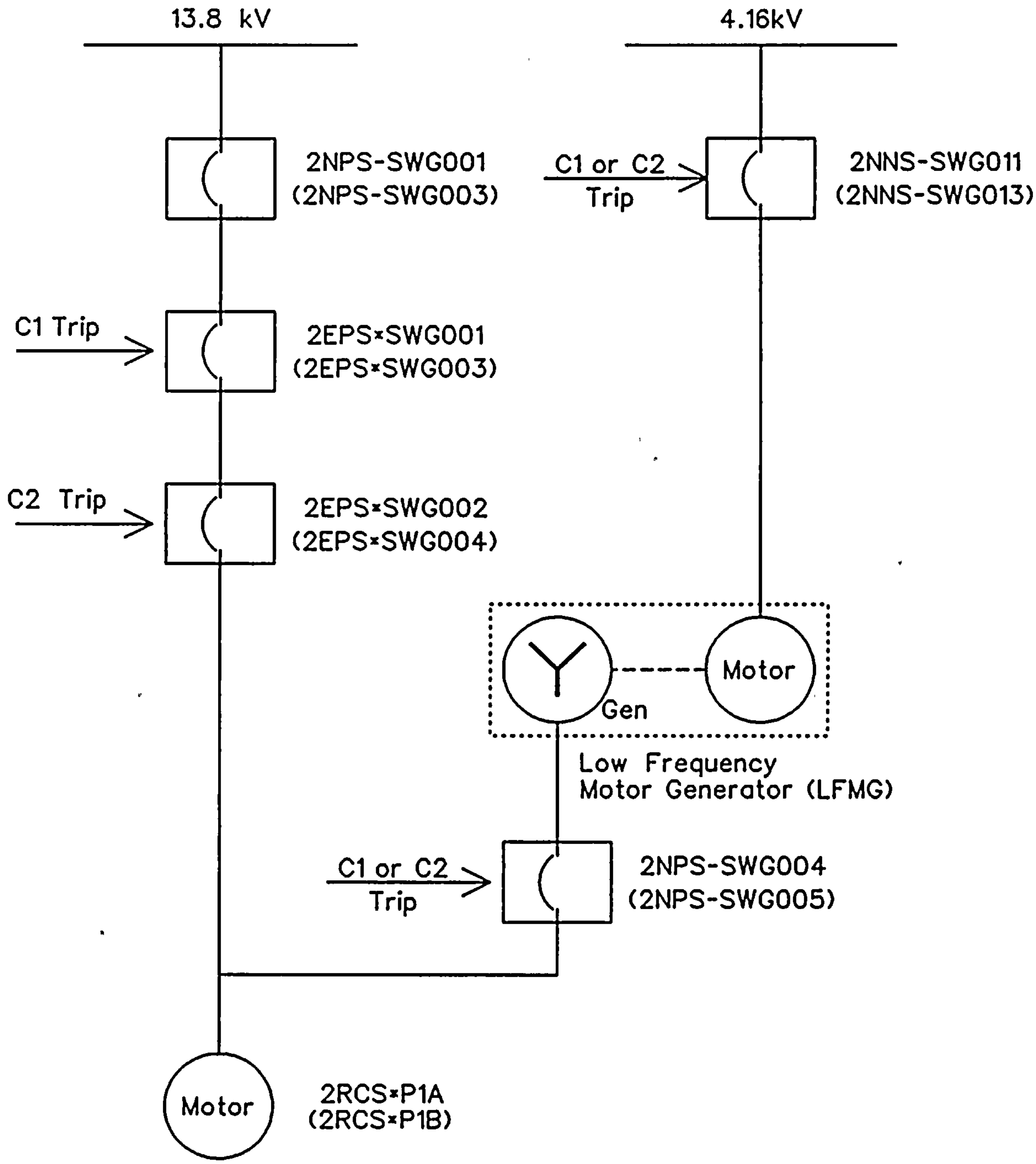


Figure 3.2.1.16-3
ARI Interface with
SCRAM Discharge Volume

ND - SOV normally de-energized
NE - SOV normally energized

I:\NMP2IPE\DGN\2S16.DGN Jul. 07. 1992 08:42:07

Rev 0 (7/92)

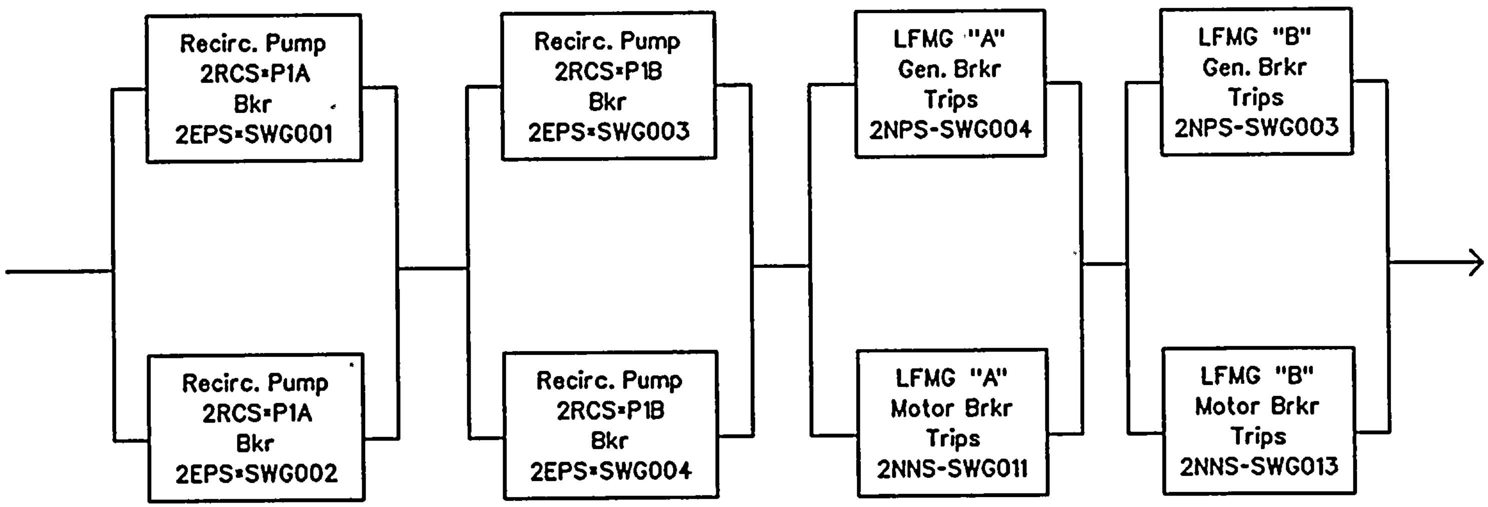


System "A" shown and equipment listed
System "B" similar, equipment in parenthesis

Figure 3.2.1.16-4
Recirculation Pump
Electrical Diagram

I:\NHP2IPE\DGN\2S16.DGN Jul. 07. 1992 08:43:46

Rev 0 (7/92)

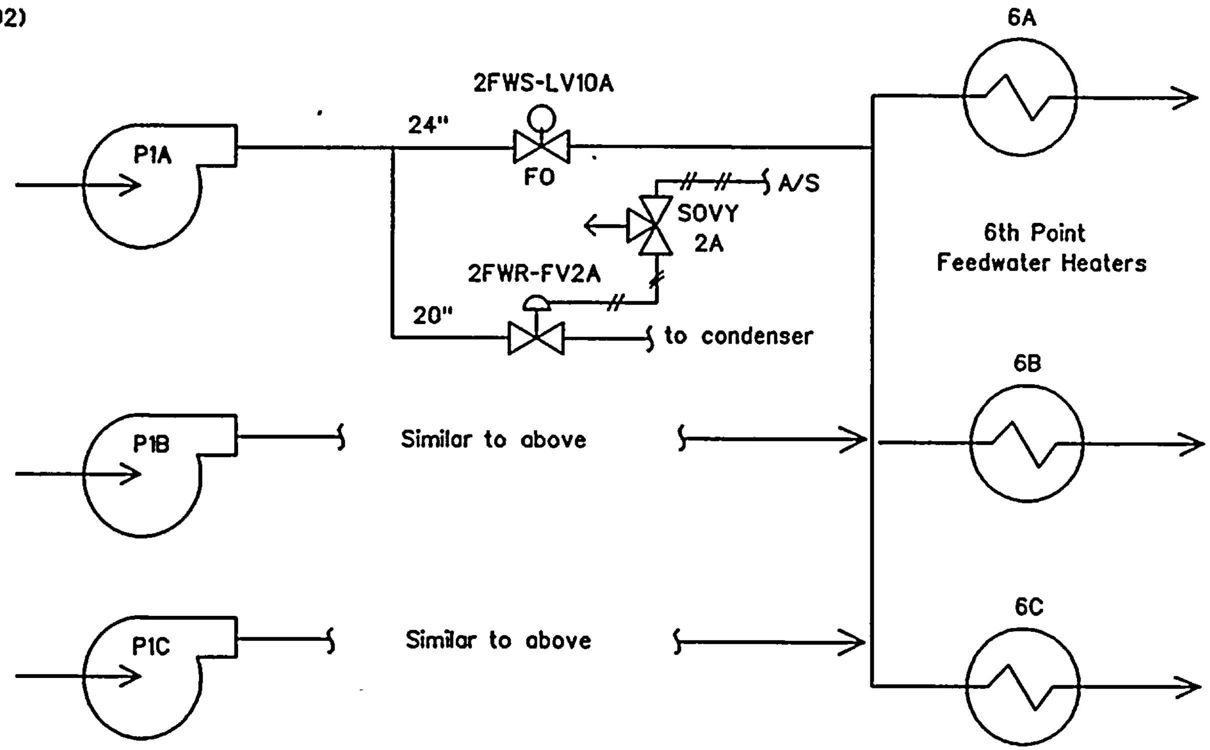


Both recirc pumps must trip 2RCS*P1A, P1B
2RCS*P1A can be tripped by either Bkr 2EPS*SWG001 or 2EPS*SWG002
2RCS*P1B can be tripped by either Bkr 2EPS*SWG003 or 2EPS*SWG004
The generator or motor breaker for each MG set must open

Figure 3.2.1.16-5
Recirc Pump Trip

I: \NMP2IPE\DGN\2S16.DGN Jul. 07. 1992 08: 45: 25

Rev 0 (7/92)



FT is a success if any combination of either LV10(A/B/C) closure and/or FV2(A/B/C) opening for each pump discharge

If Instrument Air fails - all FV2's (A/B/C) will open and it is a success

If Instrument Air is available then the failure of 125VDC and/or 600VAC or 120VAC (all non-divisional) results in FT failure

If all support is available then 2FWS-LV10(A/B/C) and FV2(A/B/C) fail for each pump discharge

If C1 and C2 fail, then FT fails. (All valves recieve C1 and C2 signals)

Figure 3.2.1.16-6
Feedwater Runback

I:\NMP2IPE\DGN\2S16.DGN Jul. 07. 1992 08:47:04

Rev 0 (7/92)

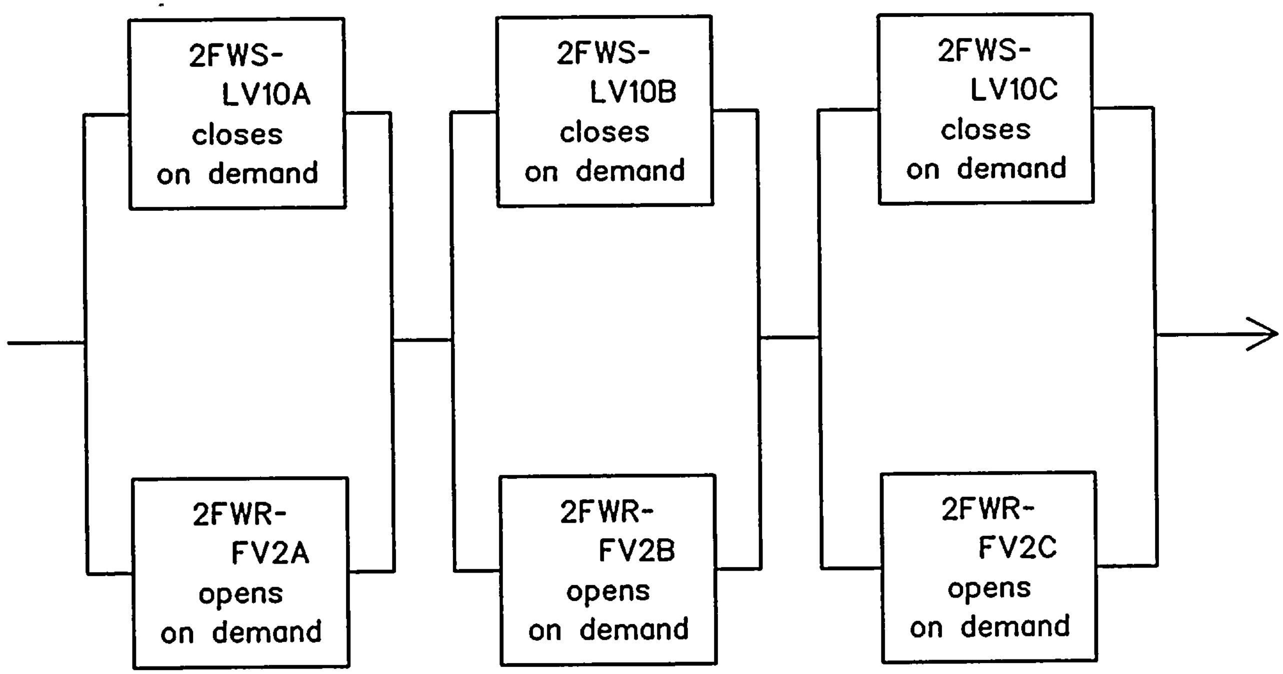


Figure 3.2.1.16-7
Feedwater Trip
Success Diagram

I:\NMP2IPE\DGN\2S16.DGN Jul. 07. 1992 08:48:55

Table 3.2.1.16-1

REV. 0 (7/92)

REDUNDANT REACTIVITY CONTROL (Top Event C1 & C2) Component Block Descriptions							
Block	Mark No. (Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
C1 - POWER SUPPLIES							
	CHANNEL A POWER SUPPLY	to circuit 24VDC-PS-2RRSA02	FAIL TO OPERATE	N/A	N/A	N/A	N/A
	CHANNEL B POWER SUPPLY	to circuit 24VDC-PS-2RRSC02	FAIL TO OPERATE	N/A	N/A	N/A	N/A
C1 - TRANSMITTERS							
	2ISC*PT2A (B22-N403A)	Channel A Pressure Transmitter	FAIL TO OPERATE	N/A	N/A	2CES*RAK004	FAILS
	2ISC*PT2B (B22-N403E)	Channel B Pressure Transmitter	FAIL TO OPERATE	N/A	N/A	2CES*RAK005	FAILS
C2 - POWER SUPPLIES							
	CHANNEL A POWER SUPPLY	to circuit 24VDC-PS-2RRSB02	FAIL TO OPERATE	N/A	N/A	N/A	N/A
	CHANNEL B POWER SUPPLY	to circuit 24VDC-PS-2RRSD02	FAIL TO OPERATE	N/A	N/A	N/A	N/A
C2 - TRANSMITTERS							
	2ISC*PT2C (B22-N403B)	Channel A Pressure Transmitter	FAIL TO OPERATE	N/A	N/A	2CES*RAK027	FAILS
	2ISC*PT2D (B22-N403F)	Channel B Pressure Transmitter	FAIL TO OPERATE	N/A	N/A	2CES*RAK026	FAILS

ALTERNATE ROD INSERTION
Component Block Descriptions

Block	Mark No. (Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
VALVES							
	2RDS*SOV156	ARI Solenoid Valve	FAILS TO OPEN	CLOSED	OPEN	2BYS*PNL202A	AS-IS
	2RDS*SOV157	ARI Solenoid Valve	FAILS TO OPEN	CLOSED	OPEN	2BYS*PNL202B	AS-IS
	2RDS*SOV158	ARI Solenoid Valve	FAILS TO OPEN	CLOSED	OPEN	2BYS*PNL202A	AS-IS
	2RDS*SOV159	ARI Solenoid Valve	FAILS TO OPEN	CLOSED	OPEN	2BYS*PNL202B	AS-IS
	2RDS*SOV160	ARI Solenoid Valve	FAILS TO OPEN	CLOSED	OPEN	2BYS*PNL202A	AS-IS
	2RDS*SOV161	ARI Solenoid Valve	FAILS TO OPEN	CLOSED	OPEN	2BYS*PNL202B	AS-IS
	2RDS*SOV162	ARI Solenoid Valve	FAILS TO OPEN	CLOSED	OPEN	2BYS*PNL202A	AS-IS
	2RDS*SOV163	ARI Solenoid Valve	FAILS TO OPEN	CLOSED	OPEN	2BYS*PNL202B	AS-IS

REACTOR RECIRC. PUMP TRIP
Component Block Descriptions

Block	Mark No.	(Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
RECIRC. PUMP 1A TRIP								
	2EPS*SWG001-1	(BRK 3A)	Pump 1A Breaker	FAIL TO OPEN	CLOSED	CLOSED	2EPS*SWG001	OPEN
	2EPS*SWG002-1	(BRK 4A)	Pump 1A Breaker	FAIL TO OPEN	CLOSED	CLOSED	2EPS*SWG002	OPEN
RECIRC. PUMP 1B TRIP								
	2EPS*SWG003-1	(BRK 3B)	Pump 1B Breaker	FAIL TO OPEN	CLOSED	CLOSED	2EPS*SWG003	OPEN
	2EPS*SWG004-1	(BRK 4B)	Pump 1B Breaker	FAIL TO OPEN	CLOSED	CLOSED	2EPS*SWG004	OPEN
LOW FREQUENCY MOTOR GENERATOR TRIP								
	2NPS-SWG004-1		"A" LFMG Generator Breaker	FAIL TO CLOSE	OPEN	CLOSED	2NPS-SWG004	OPEN
	2NNS-SWG011-9		"A" LFMG Motor Breaker	FAIL TO CLOSE	OPEN	CLOSED	2NPS-SWG011	OPEN
	2NPS-SWG005-1		"B" LFMG Generator Breaker	FAIL TO CLOSE	OPEN	CLOSED	2NPS-SWG005	OPEN
	2NNS-SWG013-1		"B" LFMG Motor Breaker	FAIL TO CLOSE	OPEN	CLOSED	2NPS-SWG013	OPEN

Table 3.2.1.16-4 REV. 0 (7/92)

FEEDWATER PUMP TRIP
Component Block Descriptions

Block	Mark No.	(Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
POWER SOURCE	2BYS-PNLB101		125V DC Power Source	DE-ENERGIZED	ENERGIZE	ENERGIZE	2BYS-SWG001B	DE-ENERGIZED
2FWS-LV10A	2NHS-MCC003 2BYS-PNLB101 K23-2FWSN33 2FWS-LV10A	 (C33A-K23)	"Normal" AC Power Source 125V DC Power Source Relay Feedwater Level Control Valve	DE-ENERGIZED DE-ENERGIZED DE-ENERGIZED FAILS TO CLOSE	ENERGIZE ENERGIZE ENERGIZE OPEN	ENERGIZE ENERGIZE ENERGIZE CLOSE	2NJS-US1 2BYS-SWG001B 2BYS-PNLB101 2NHS-MCC003	DE-ENERGIZED DE-ENERGIZED DE-ENERGIZED AS-IS
2FWR-FV2A	2SCI-PNLA101 K30-2FWSN33 2FWR-SOVY2A 2FWR-FV2A	 (C33A-K30)	120V AC Power Source Relay Solenoid Valve to 2FWR-FV2A Flow Control Valve	DE-ENERGIZED DE-ENERGIZED FAILS TO OPEN FAILS TO OPEN	ENERGIZE ENERGIZE CLOSED CLOSED	ENERGIZE ENERGIZE OPEN OPEN	2NJS-US4 2SCI-PNLA101 2SCI-PNLA101 Instrument Air	DE-ENERGIZED DE-ENERGIZED AS-IS CLOSED
2FWS-LV10B	2NHS-MCC003 2BYS-PNLB101 K23-2FWSN33 2FWS-LV10B	 (C33A-K23)	"Normal" AC Power Source 125V DC Power Source Relay Feedwater Level Control Valve	DE-ENERGIZED DE-ENERGIZED DE-ENERGIZED FAILS TO CLOSE	ENERGIZE ENERGIZE ENERGIZE OPEN	ENERGIZE ENERGIZE ENERGIZE CLOSE	2NHS-MCC003 2BYS-SWG001B 2BYS-PNLB101 2NHS-MCC003	DE-ENERGIZED DE-ENERGIZED DE-ENERGIZED AS-IS
2FWR-FV2B	2SCI-PNLA101 K30-2FWSN33 2FWR-SOVY2B 2FWR-FV2B	 (C33A-K30)	120V AC Power Source Relay Solenoid Valve to 2FWR-FV2B Flow Control Valve	DE-ENERGIZED DE-ENERGIZED FAILS TO OPEN FAILS TO OPEN	ENERGIZE ENERGIZE CLOSED CLOSED	ENERGIZE ENERGIZE OPEN OPEN	2NJS-US4 2SCI-PNLA102 2SCI-PNLA102 Instrument Air	DE-ENERGIZED DE-ENERGIZED AS-IS CLOSED
2FWS-LV10C	2NHS-MCC003 2BYS-PNLB101 K24-2FWSN33 2FWS-LV10C	 (C33A-K24)	"Normal" AC Power Source 125V DC Power Source Relay Feedwater Level Control Valve	DE-ENERGIZED DE-ENERGIZED DE-ENERGIZED FAILS TO CLOSE	ENERGIZE ENERGIZE ENERGIZE OPEN	ENERGIZE ENERGIZE ENERGIZE CLOSE	2NJS-US1 2BYS-SWG001B 2VBS-PNLB101 2NHS-MCC003	DE-ENERGIZED DE-ENERGIZED DE-ENERGIZED AS-IS
2FWR-FV2C	2SCI-PNLA101 K30-2FWSN33 2FWR-SOVY2C 2FWR-FV2C	 (C33A-K30)	120V AC Power Source Relay Solenoid Valve to 2FWR-FV2C Flow Control Valve	DE-ENERGIZED DE-ENERGIZED FAILS TO OPEN FAILS TO OPEN	ENERGIZE ENERGIZE CLOSED CLOSED	ENERGIZE ENERGIZE OPEN OPEN	2NJS-US4 2SCI-PNLA101 2SCI-PNLA101 Instrument Air	DE-ENERGIZED DE-ENERGIZED AS-IS CLOSED

System 17

Containment Venting

23.2.1.17 Containment Venting

3.2.1.17.1 System Function

Containment venting uses the containment purge system to vent the primary containment during a severe accident. This function supports the Emergency Operating Procedures. A simplified diagram is provided in Figure 3.2.1.17-1.

3.2.1.17.2 Success Criteria

There are several event tree top events that model venting the containment through a containment purge exhaust path. Top event CV, in the front-line event trees, models the suppression chamber purge exhaust path and its alignment via the Standby Gas Treatment System (SGTS) filter/fan bypass to the stack in accordance with N2-EOP-6, Attachment 21. This action is directed in the EOP to be accomplished before containment pressure reaches 45 psig, and must, for the purposes of the PRA model, be maintained for 24 hours. The following top events are included in the backend model (Level 2 containment event trees):

Top Event	Success Criteria
GV	Combustible gas venting is directed by the EOPs (per N2-EOP-6, Attachment 27) when 6% by volume of H_2 concentration and 5% by volume O_2 concentration is realized inside containment, and the recombiners are inoperable. The model is the same as CV except for actions required by the operator.
VC	Containment venting is directed by the EOPs, per N2-EOP-6 Attachment 21, to be accomplished before containment pressure reaches the Primary Containment Pressure Limit (PCPL). The model is the same as CV, except for actions required by the operator (i.e., the CV model is used only for scenarios in which the operator has several hours before this action is invoked).
FB/FD	Drywell venting, in support of containment flooding, is required by the EOPs to be accomplished regardless of whether the C6 contingency EOP is in force to maintain containment pressure less than 45 psig. The model is similar to CV except the drywell purge exhaust path is modeled instead of the wetwell purge exhaust valves and the operator action is to accomplish this action while the containment is being flooded.

3.2.1.17.3 Support Systems

The instrument air system supplies air pressure to control air operated valves. The instrument nitrogen system supplies gas pressure to operate air operated valves inside primary containment. Emergency AC supplies motor operated valves and solenoid operated valves associated with containment isolation AOVs. Table 3.2.1.17-1 gives a more detailed list of components, support systems, and failure modes.

3.2.1.17.4 System Operation

The containment depressurization process is done in two phases, both of which are accomplished manually.

The process is initially configured such that the containment gas flows via a 2" pipe (PV104 and SOV102) into the SGTS inlet gas stream and is subsequently processed through a designated SGTS filter train assembly, and directed out the stack.

However, under most accident conditions defined in the PRA, the flow via a 2" pipe is inadequate to prevent containment overpressurization. The system configuration is subsequently manually aligned to augment containment gas flows via 20" pipe (AOV101) into the opposite SGTS filter train inlet piping (MOV2A or 2B). Blind flanges at the SGTS filter/fan assembly ensure the diversion of flow around the SGTS filter/fan assembly and through the 14" diameter bypass pipe (PV5A or 5B). Effluent then goes out the stack (MOV3A or 3B).

3.2.1.17.5 Instrumentation and Controls

Containment venting is manually implemented using a combination of fuse removal, breaker deactivation, jumper installation, cable lifting, controller adjustment, and switch operation.

3.2.1.17.6 Technical Specifications

Valves 2CPS*AOV105, 2CPS*AOV107, 2CPS*AOV109, 2CPS*AOV111, 2CPS*AOV104, 2CPS*AOV106, 2CPS*AOV108, and 2CPS*AOV110 may be open up to 90 hours per 365 days for the purpose of venting OR purging in conditions 1, 2, and 3.

With drywell or suppression chamber purge supply or exhaust isolation valve (with resilient seats) not passing leak rate, repair in 24 hours, or be in mode 3 in next 24 hours.

Once per refuel, isolation valves of specification 3.6.1.7.b, verify that the open limit is blocked to 70° or 60°, as applicable.

Once per 92 days, isolation valves shall be leak rate tested.

3.2.1.17.7 Surveillance Testing and Maintenance

Isolation valves are leak rate tested every 92 days.

Open limits are checked every refuel.

3.2.1.17.8 References

Operating Procedure N2-OP-61A, Rev. 3
N2-EOP-6 Attachment 21, Rev. 00
Attachment 27, Rev. 00
PID-61A-7

Technical Specifications section 3.6.1.7

Final Report, Containment Venting for Emergency
Decay Heat Removal NM2-19105, Access #07226 2262

3.2.17.9 Initiating Event Potential

Failure of the Containment Venting function does not result in any potential initiating events.

3.2.1.17.10 Equipment Location

The two inboard isolation valves are located inside the containment. The outboard isolation valves and process valves are located in the Reactor Building and in the SGTS filter/fan rooms. Refer to Table 3.2.1.17-1.

3.2.1.17.11 Operating Experience

There were no outstanding operational events relevant to this study. Plant specific component operational data is detailed in section 3.3.2.

3.2.1.17.12 Modeling Assumptions

1. The containment venting function (CV) in the front-line event trees models the suppression chamber purge exhaust path and neglects the redundant drywell purge exhaust path. This is conservative, but it is considered insignificant given the present design and implementation difficulties associated with the manual installation of blind flanges (requires a maintenance crew).

2. Venting via the 20" pipe is assumed since the 2" pipe is expected to be inadequate for severe accidents being modeled. In fact, attempts to use the 2" pipe are expected to contribute to containment venting failure. This action could make flange installation impractical due to radiological concerns associated with venting potential radioactive effluent using either train of the SGTS.

3. SGTS valves downstream of the purge exhaust air operated valves are neglected in the hardware model (i.e., there are redundant paths, valves can be manually opened locally, and human performance associated with the present design is expected to dominate the operator's ability to align the system as required by the EOPs).

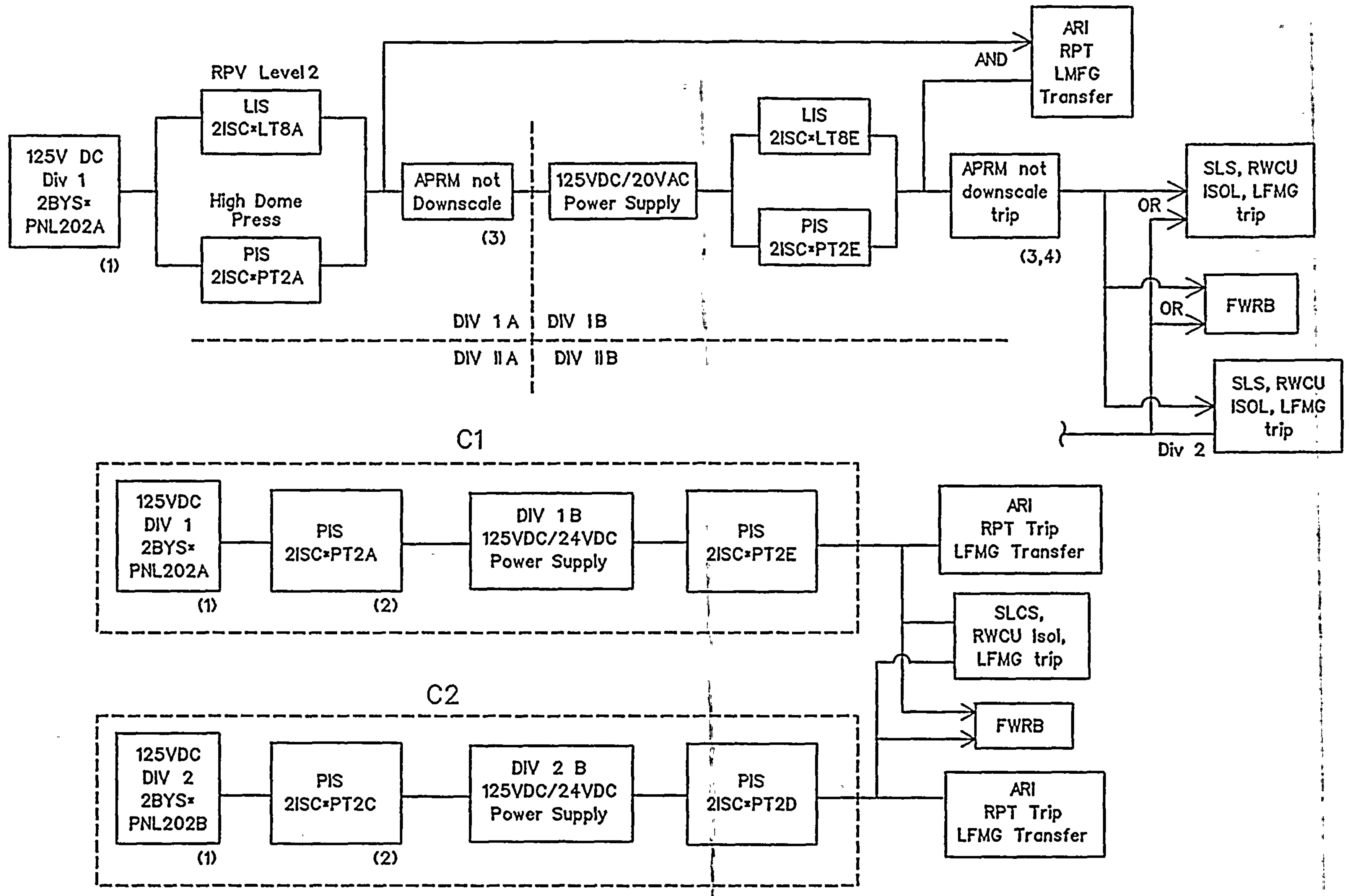
4. The manual recovery action of aligning instrument air to inboard containment purge valves, upon failure of the N_2 gas supply, is conservatively excluded from all containment venting models.

5. The failure mode of valve 2CPS*SOV109 in the CV, GV, and VC trees is modeled as FAILS TO OPEN. The valve is assumed to have received an isolation signal to close the valve from its normal open state. The failure mode of FAILS TO CLOSE upon the isolation signal is not included in the models because a failure of SOV109 to isolate would contribute to successful containment venting. Successful containment venting requires that this valve re-open after receiving the isolation signal to supply Nitrogen to 2CPS*AOV109.

3.2.1.17.13 Logic Model and Results

The fault tree(s) is included in Tier 2 documentation, and is available upon request. Quantitative results are summarized in Section 3.3.4 (Tier 1).

Rev 0 (7/92)



SI APERTURE CARD

Also Available On Aperture Card

Note 1: 125V DC/24V DC power source is included.

Note 2: LFMG transfer and FWRB occur only on high dome pressure while SLCS, LFMG trip, recirc pump trip and ARI are initiated by high dome pressure or low vessel level. To simplify the model while maintaining conservatism, a single transmitter (pressure) is used as the initator of each RRCS Div 1 channel. PIS is a xmtr and a bistable.

Note 3: APRM - Not-Downscale trip is not required for ARI, RPT and LFMG transfer. It is a required permissive to fire RRCS, SLCS, RWCU and LFMG trip. It is normally satisfied during power operation (Logical 1). If it fails to detect that the RPS has been successful, the end result is an unnecessary RRCS actuation. If it is spuriously actuated (Logical 0) at the time of an ATWS the SLCS, RWCU isolation, FWRB and LFMG trip would be failed while ARI and RPT would be successful. For a spurious (Logical 0) actuation of APRM - Not-Downscale trip to occur, it would require four our of eight neutron monitoring channels to fail and therefore the frequency of this event is considered to me small and is not modeled.

Note 4: The relays for the APRM - Not-Downscale trip are powered by the RPS 120V AC buses, failure of a bus results in a success for RRCS and enables APRM - Not-Downscale trip to SLCS.

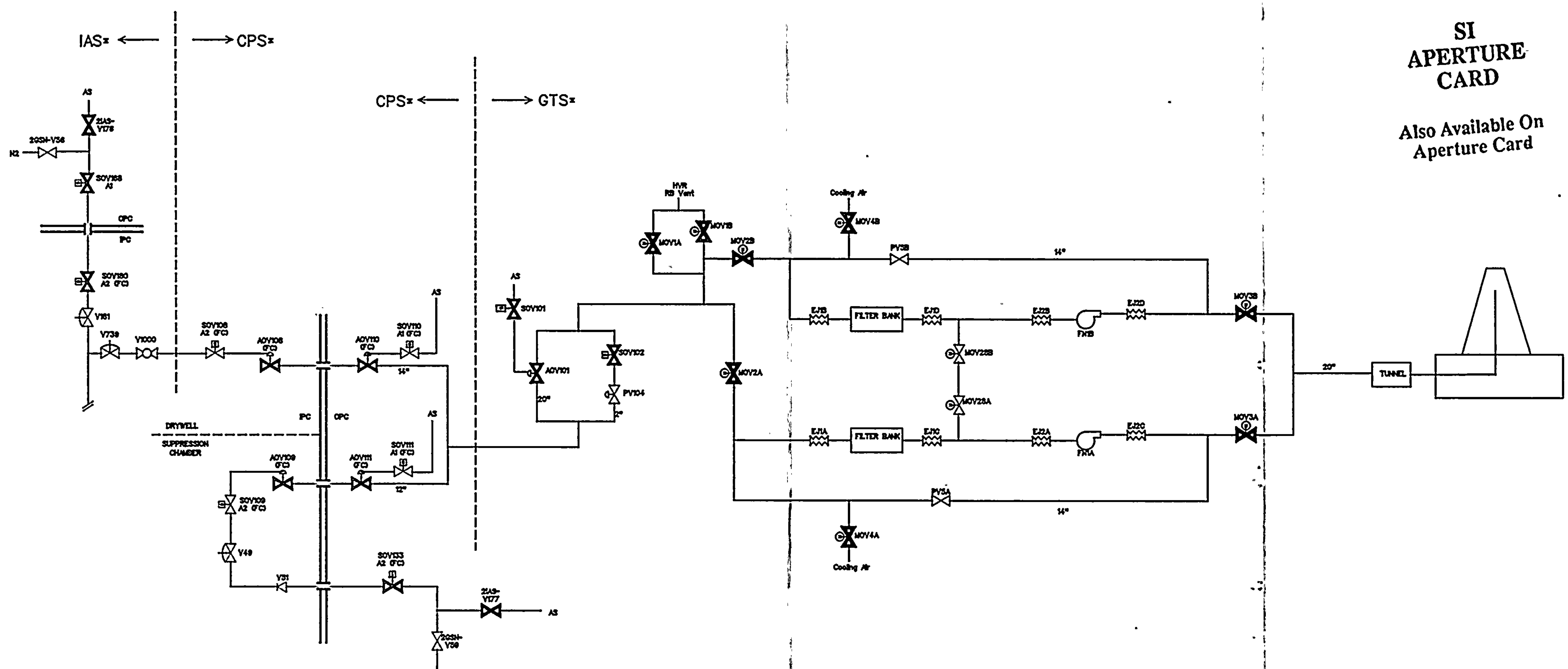
Note 5: 120V AC power supplies 2VBS*PNL101A & 101B provide power to RRCS but only for indication and equipment status.

Figure 3.2.1.16-1
Redundant Reactivity
Control System

I:\NMP2IPE\DGN\2S16.DGN Jul. 07, 1992 10:46:00

9208050191 ~ 26

Rev 0 (7/92)



A1: Emergency AC, Div I
A2: Emergency AC, Div II
N2: Nitrogen
AS: Instrument Air
FC: Fails Closed

modeled in top events CV, GV & VC
modeled in top events FB & FD
presently NOT modeled

SI APERTURE CARD

Also Available On Aperture Card

FIGURE 3.2.1.17-1
CONTAINMENT VENTING

I:\NMP2IPE\DGN\2S17.DGN Jul. 07, 1992 12:59:49

9208050191 - 27

CONTAINMENT VENT Component Block Descriptions							
Block	Mark No. (Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
SUPPRESSION CHAMBER PURGE (CV, GV & VC)							
	2GSN-V59	Nitrogen Supply	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CPS*SOV133	Nitrogen Containment Isolation	FAILS TO OPEN	CLOSED	OPEN	2SCH*PNL302B	CLOSED
	2CPS*V51	Nitrogen Supply Check Valve	FAILS TO OPEN	CLOSED	OPEN	N/A	N/A
	2CPS*V47	Nitrogen Supply	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2CPS*SOV109	Nitrogen to 2CPS*AOV109	FAILS TO OPEN	CLOSED	OPEN	2SCH*PNL302B	CLOSED
	2CPS*AOV109	S.Chamber Purge Isolation	FAILS TO OPEN	CLOSED	OPEN	NITROGEN	CLOSED
	2CPS*SOV111	Air to 2CPS*AOV111	FAILS TO OPEN	CLOSED	OPEN	2SCH*PNL101A	CLOSED
	2CPS*AOV111	S.Chamber Purge Isolation	FAILS TO OPEN	CLOSED	OPEN	INSTRUMENT AIR	CLOSED
DRYWELL PURGE VALVES (FB & FD)							
	2GSN-V56	Nitrogen Supply	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2IAS*SOV168	Nitrogen Containment Isolation	FAILS TO OPEN	CLOSED	OPEN	2SCH*PNL102A	CLOSED
	2IAS*SOV180	Nitrogen Containment Isolation	FAILS TO OPEN	CLOSED	OPEN	2SCH*PNL302B	CLOSED
	2IAS*V181	Nitrogen Supply	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2IAS*V739	Nitrogen Supply	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2IAS*V1000	Nitrogen Supply	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2CPS*SOV108	Nitrogen to 2CPS*AOV108	FAILS TO OPEN	CLOSED	OPEN	2SCH*PNL302B	CLOSED
	2CPS*AOV108	Drywell Purge Isolation Valve	FAILS TO OPEN	CLOSED	OPEN	NITROGEN	CLOSED
	2CPS*SOV110	Air to 2CPS*AOV110	FAILS TO OPEN	CLOSED	OPEN	2SCH*PNL101A	CLOSED
	2CPS*AOV110	Drywell Purge Isolation Valve	FAILS TO OPEN	CLOSED	OPEN	INSTRUMENT AIR	CLOSED
VENT PATH TO STACK (presently NOT modeled)							
	2GTS*SOV101	Air to Isolation Valve AOV101	FAILS TO OPEN	CLOSED	OPEN	2SCI-PNLA101	CLOSED
	2GTS*AOV101	CPS to GTS Isolation Valve	FAILS TO OPEN	CLOSED	OPEN	INSTRUMENT AIR	CLOSED
	2GTS*MOV1A	SBGT inlet from Rx Bldg. HVAC	FAILS TO CLOSE	CLOSED	OPEN	2EHS*MCC102A	AS-IS
	2GTS*MOV1B	SBGT Inlet from Rx Bldg. HVAC	FAILS TO CLOSE	CLOSED	OPEN	2EHS*MCC302B	AS-IS
	2GTS*MOV2A	SBGT "Train A" Inlet Valve	FAILS TO OPEN	CLOSED	OPEN	2EHS*MCC102A	AS-IS
	2GTS*PV5A	Rx Bldg Pressure Control Valve	FAILS TO OPEN	CLOSED	OPEN	2EJS*PNL103A	AS-IS
	2GTS*MOV3A	SBGT Fan 1A Discharge Valve	FAILS TO OPEN	CLOSED	OPEN	2EHS*MCC102A	AS-IS
	2GTS*MOV2B	SBGT "Train B" Inlet Valve	FAILS TO OPEN	CLOSED	OPEN	2EHS*MCC302B	AS-IS
	2GTS*PV5B	Rx Bldg Pressure Control Valve	FAILS TO OPEN	CLOSED	OPEN	2EJS*PNL303B	AS-IS
	2GTS*MOV3B	SBGT Fan 1B Discharge Valve	FAILS TO OPEN	CLOSED	OPEN	2EHS*MCC302B	AS-IS

System 18

Vapor Suppression

3.2.1.18 Vapor Suppression

3.2.1.18.1 Function

The Primary Containment, (Mark II), provides a barrier to limit the release of radioactive materials to the environment after a Design Basis Accident (DBA). The drywell houses the reactor vessel, the reactor recirculation system, branch connections of the Reactor Coolant Pressure Boundary (RCPB) and other support systems required for reactor operation. A series of downcomer vent pipes, penetrating the drywell floor, connects the drywell atmosphere with the stored water within the suppression pool. The suppression pool provides rapid condensation and cooling of the steam-air-water mixture during a LOCA or safety relief valve (SRV) actuation. The containment atmosphere is inerted with nitrogen gas during reactor operation to minimize the possibility of a flammable hydrogen-air mixture following an accident.

Drywell: The functions of the drywell are:

1. Contain the radioactivity and steam resulting from a break of the Reactor Coolant Pressure Boundary (RCPB) in the drywell and direct the steam through the downcomer pipes to the suppression pool which condenses the steam and limits the pressure excursion following a LOCA.

2. Provide radiation shielding for the secondary containment.

3. Provide structural support for the refueling pool.

4. Provide protection for the reactor vessel from missiles and pipe whip.

Downcomer Pipes: The drywell floor is penetrated by 121 downcomer pipes, four vacuum relief lines, 18 SRV lines and vents, and 20 drain lines. The pipes provide a flow path for uncondensed steam from the drywell to the suppression pool. The downcomer pipes are opened to the drywell and extend into the suppression pool 9.5 feet below the minimum suppression pool level. The downcomer pipes project 3 to 6 inches above the drywell floor where they are shielded by steel deflector plates. The plates prevent overloading any single vent pipe by direct flow from a pipe break at a particular vent and minimizes the potential for downcomer blockage by debris. The drywell floor provides the anchor support for the downcomers.

Suppression Chamber: The suppression chamber is supported and anchored on the reinforced concrete mat. The suppression chamber contains the suppression pool.

Suppression Pool: The suppression pool stores sufficient water to condense the steam released from blowdown of the Reactor Coolant System (RCS) after a LOCA or SRV actuation. Approximately 1,122,000 gallons of water are contained within the suppression pool. The suppression pool also serves as a reservoir of water for the Emergency Core Cooling Systems (ECCS), a condensing medium for the Reactor Core Isolation Cooling (ICS) turbine exhaust and a backup water supply for the ICS. It also provides a back-up water supply for the High Pressure Core Spray System (HPCS). The water used to fill the pool comes from the Condensate Transfer and Storage System.

The suppression pool water volume maintains the pool water temperature below design limits during the blowdown phase of a LOCA. The Suppression Pool Cooling mode of the Residual Heat Removal System (RHS) is used to maintain the long term, post-LOCA pool temperatures.

Downcomer Vent Vacuum Breakers: Four vacuum relief lines provide a return flow path from the suppression chamber gas space to the drywell. The relief lines limit the negative differential pressure between the drywell and the suppression chamber to a maximum value of 4.70 psid to maintain structural integrity of primary containment during conditions of large differential pressure.

The vacuum breakers prevent drawing water from the suppression pool up into the downcomer pipes. If this was to occur, the resultant forces upon the suppression chamber during a design basis LOCA would be greater than design due to the increased mass of water being expelled into the suppression pool.

The vacuum breakers also provide drywell floor relief protection during the vessel re-flood phase after a LOCA. The drywell and the suppression chamber pressures will equalize after a LOCA. When the cool ECCS water begins to flood into the vessel, the water will pour out the break and condense the steam in the drywell. This will cause a vacuum to be produced in the drywell and place stresses on the drywell floor. The drywell vacuum breakers function to equalize the differential pressure.

The vacuum breakers are located on the drywell floor. Each vacuum breaker assembly consists of two check valves in series mounted in piping that connects the drywell and suppression chamber. The vacuum breakers have the capability for remote manual testing from panel 628 in the control room.

SRV Vacuum Breakers: All 18 safety relief valves (SRVs) penetrate the drywell floor and discharge into the suppression pool with the line terminating in a T-quencher connections. The SRV discharge lines are arranged to provide an evenly distributed heat load to the pool when a group of SRVs open.

Two vacuum relief valves on each SRV discharge line serve to admit drywell atmosphere to the SRV discharge line - preventing siphoning of water into the SRV discharge line as it cool off (steam condensation) after an opening cycle. Otherwise, water in the line more than a few feet above the suppression pool may cause excessive pressure at the SRV discharge when the valve reopens.

Containment Spray: The Containment Spray System is provided to quickly reduce containment pressure during the post accident period of a LOCA. The Containment Spray

System consists of two subsystems: the drywell spray and the suppression chamber spray. The drywell spray consists of two independent loops and spray headers. The suppression chamber spray consists of one spray header supplied from two otherwise independent loops.

3.2.1.18.2 Success Criteria

Vapor suppression is modeled in the LOCA event trees as top event VS. Initially, all four vacuum breaker equipped short downcomers must isolate the drywell from the suppression chamber. At least one of two check valves in each line must be closed. Otherwise, the vapor suppression system is bypassed and containment could overpressurize during a LOCA if sprays or venting is not initiated.

Top event OV in the LOCA event trees models operator actions associated with mitigation of vapor suppression failure (VS = F). The operators must actuate containment spray or align containment venting to suppress drywell pressure rise. The operators have 20 minutes for medium LOCA and 45 minutes for small LOCA.

3.2.1.18.3 Support Systems

The vacuum breakers and downcomers require no support systems to function. Testing of the vacuum breakers requires 120V AC (2SCI-PNLB102) power and Nitrogen or Instrument Air.

3.2.1.18.4 System Operation

The suppression pool provides for rapid condensation and cooling of the steam-air-water mixture during a LOCA or safety relief valve (SRV) actuation. The water stored in the suppression pool is capable of condensing the steam displaced into the pool through the downcomer vents, and the amount of water is sufficient that no operator action is required for at least ten (10) minutes immediately following initiation of a LOCA. In addition, the design allows any significant amount of water from pipe breaks within the primary containment to drain back to the suppression pool. This closed loop ensures a continuous, adequate supply of water for core cooling.

Vacuum breakers provide a return flow path from the suppression gas space to the drywell. The vacuum breakers are designed to limit the negative differential pressure between the drywell and the suppression chamber to a maximum value of 4.70 psid. Each vacuum breaker flow path has two relief valves in series to ensure a leak tight boundary under positive drywell-to-suppression chamber differential pressure conditions. Three flow paths are required for the vacuum breaker design basis. One additional flow path is provided to accommodate the postulated single failure of one vacuum breaker.

The vacuum breakers have the capability for remote manual testing. The design provides assurance of limiting the differential pressure between the drywell and suppression chamber and ensures proper valve operation and testing during normal plant operation.

3.2.1.18.5 Instrumentation and Controls

Temperature elements sense containment atmosphere and suppression pool water temperatures. Drywell and suppression chamber pressure is monitored by the Containment Monitoring System (CMS) and provide a containment isolation signal for the Primary Containment Isolation System. Suppression pool level is also monitored by CMS. The pressure between the inner and outer seals of electrical penetrations is monitored by the Leakage Detection System to provide an input to the computer to determine leak location.

3.2.1.18.6 Technical Specifications

If one suppression chamber/drywell vacuum breaker is open, verify the other in the pair is closed within 2 hours and restore the vacuum breaker to the closed position within 72 hours.

If position indication is inoperable, verify the other vacuum breaker in the pair to be closed within 2 hours and at least once per 24 hours thereafter; otherwise shutdown is required.

Suppression pool technical specification water level limits must be restored within 1 hour; otherwise shutdown is required.

3.2.1.18.7 Surveillance Testing and Maintenance

A monthly operability test of the drywell-to-suppression pool vacuum breakers is conducted by cycling each vacuum breaker through at least one complete cycle of full travel. Verification of actual position is determined by the full open position limit switch and the full shut position limit switch. The suppression pool level is verified at least once every 24 hours.

3.1.2.18.8 References

Unit 2 Operations Technology, Chapter 19, Revision 4
Unit 2 USAR, Section 5.2
Unit 2 USAR, Section 6.2

PID-28A-11
PID-19G-15
ESK-7ISC03
FSK-27-19J
FSK-12-1.0

Technical Specifications 3/4.5.3 and 3/4.6.4

3.1.2.18.9 Initiating Event Potential

There is no significant initiating potential.

3.1.2.18.10 Equipment Location

The primary containment is located inside the Reactor Building. The drywell to suppression chamber vacuum breakers are located inside primary containment on the drywell floor. The SRV vacuum breakers are located inside containment at elevation 254' (2SVV*RVV101 through 118) and elevation 251' (2SVV*RVV201 through 218).

3.1.2.18.11 Operating Experience

There were no particular outstanding operational events relevant to this study. Plant specific component operational data is detailed in section 3.3.2.

3.2.1.18.12 Modeling Assumptions

Failure of the vapor suppression function is believed to be a low probability event for the following reasons:

1. Failure of two vacuum breakers in series (in open position) while indicating closed position is unlikely given testing, indications and technical specifications requirements described above.

2. Failure of a sufficient number of the 121 downcomer pipes due to blowdown loads or defects are passive pipe failures and considered unlikely. Also unavailability of the suppression pool (inadequate level) is considered a low probability contributor.

3. Failure of vacuum breakers to open and re-close (fail open during cycle) is considered less significant in the Level I analysis, given the initial blowdown is successful. These failure modes are considered in the Level II containment analysis model.

4. Success criteria is believed to be conservative for the large LOCA initiating event. One open suppression chamber/drywell vacuum breaker path may provide successful vapor suppression for the Level I analysis.

Since the above failure modes are judged to be on the order of 1×10^{-5} or lower, a point estimate value of 1×10^{-4} is used for top event VS (vapor suppression failure).

3.2.1.18.13 Logic Model and Results

The fault tree(s) is included in Tier 2 documentation, and is available upon request. Quantitative results are summarized in Section 3.3.5 (Tier 1).

Rev 0 (7/92)

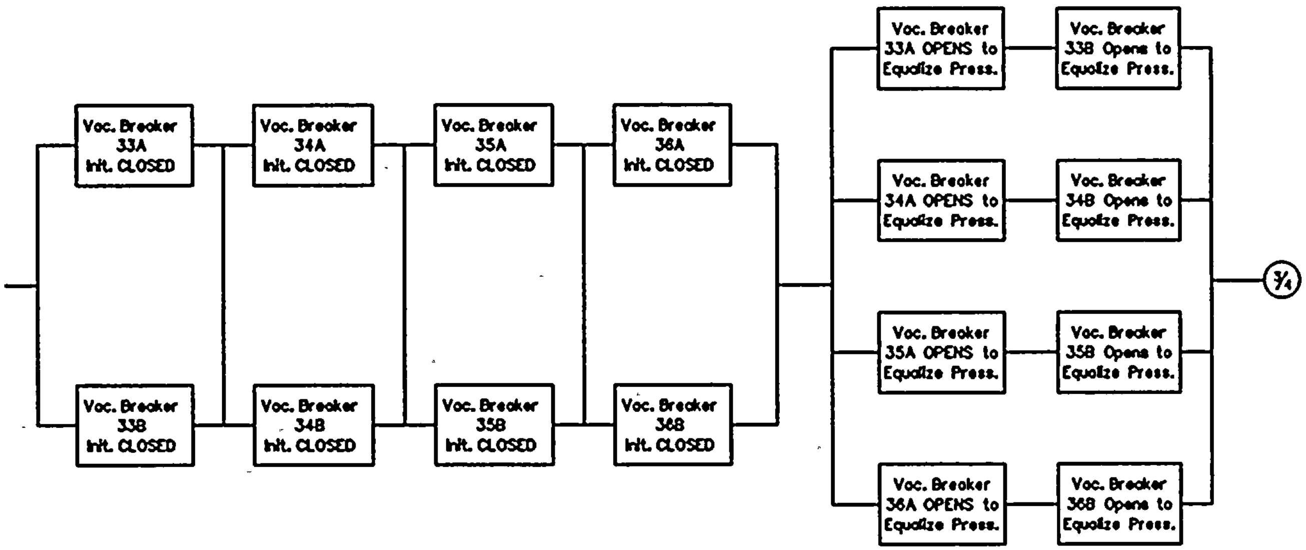


Figure 3.2.1.18-1
Top Event VS Success Diagram

I:\NMP2IPE\DGN\2S18.DGN Jul. 07, 1992 08:51:07

System 19

Reactor Building Closed Loop Cooling

3.2.1.19 Reactor Building Closed Loop Cooling Water

3.2.1.19.1 System Function

The Reactor Building Closed Loop Cooling (RBCLC or CCP) system has two major functions. It is a heat sink for reactor auxiliary equipment and accessories. It is also an intermediate system to prevent a release of contaminated cooling water from reaching the service water system. A simplified diagram of the RBCLC system is provided in Figure 3.2.1.19-1.

3.2.1.19.2 Success Criteria

The top event in the support system event tree, RW, models RBCLC maintaining adequate cooling water flow to plant support equipment. Two of three RBCLC pumps, two of three booster pumps, and two of three heat exchangers are required for success. A success diagram is provided in Figure 3.2.1.19-2.

3.2.1.19.3 Support Equipment

The RBCLC system depends on normal AC power, service water, instrument air, and emergency power supplies. Normal AC provides power to the six pumps. Service water provides the heat sink for the system. Instrument air is used to manipulate the temperature control valves. Emergency power provides power to the safety related components. Table 3.2.1.19-1 provides additional detail on RBCLC component support dependencies.

3.2.1.19.4 System Operation

The RBCLC system is normally operating. It has three 50% capacity trains, each including a pump, a booster pump and a heat exchanger, which can be put in and taken out of service from the Control Room. Two trains, including the associated booster pumps and heat exchangers, are normally in service. The standby booster pump will auto-start if either of the following conditions exist:

- Motor over-current on either of the two other booster pumps.
- Low discharge header pressure.

The RBCLC has an automatic temperature control valve (2CCP-TV108) which is set at 86°F. This valve bypasses the appropriate amount of water around the heat exchangers to keep the system water at the proper temperature. On loss of air pressure to the valve or power to the associated temperature element, the temperature control valve fails in the maximum cooling position.

A subsystem of the RBCLC system is the instrument air compressor cooling loop. This subsystem is modeled in the instrument air system model, section 3.2.1.23.

3.2.1.19.5 Instrumentation and Controls

Major RBCLC components can be controlled from the Control Room. The controls include pump and booster pump start and stop, seal and bearing coolers for RHR, and recirculation pump coolers.

The Control Room alarms include Division I/II isolation valves inoperable; pump auto start, trip, and overload; low system pressure; and cooling system trouble.

3.2.1.19.6 Technical Specifications

All isolation valves are tested in accordance to Tech. Spec. Table 3.6.3-1. If test results are not acceptable, the affected penetration must be placed in an "isolated" condition within 4 hours, or be in hot shutdown within the next 12 hours, and cold shutdown within 24 hours.

3.2.1.19.7 Surveillance, Testing and Maintenance

The Reactor Building Closed Loop Cooling valves undergo operability testing quarterly and when the plant goes into cold shutdown.

Valve position indication is verified at least once per fuel cycle (every 18 months).

All automatic isolation valves are tested for full automatic isolation at least once every 18 months.

A Leak Rate Test is run on the containment isolation valves at least once every two years.

3.2.1.19.8 References

OP N2-OP-13 Rev. 2
PID's as referenced on simplified drawing
Tech Spec 3/4.6.3, 3/4.6.1
FSAR Section 9.2.2

3.2.1.19.9 Initiating Event Potential

A loss of RBCLC would fail the instrument air compressors. The instrument air tanks would bleed down quickly and cause individual control rods to insert as pressure is lost at the SCRAM valve.

Also, recirculation pump motor and seal coolers would heat up and render the pumps inoperable per unit technical specifications.

A loss of RBCLC would fail the Drywell Unit Coolers. Because of loss of drywell cooling, the pressure in the drywell would increase quickly, causing a LOCA signal and reactor SCRAM.

3.2.1.19.10 Equipment Location

The heat exchangers are in the North Auxiliary Bay at elevation 198'. The RBCLC pumps are inside secondary containment at elevation 328' and the booster pumps are at elevation 198'.

3.2.1.19.11 Operational Experience

There were no particular outstanding operational events relevant to this study. Plant specific component operational data is detailed in section 3.3.2.

3.2.1.19.12 Modeling Assumptions

1. Expansion Tank Level Control was not modeled.

2. The instrument air (IAS) dependency is not modeled. An IAS failure causes temperature control valves to go to maximum cooling. This would not effect RBCLC operability in an emergency.

3. Bypass valve 2CCP-V173 is not modeled because it is a manual valve. An operator would have to go into the field to manipulate it. This is considered a recovery action, and is not credited.

4. Failure of service water header A is assumed to fail RBCLC. The present model does not take credit for manually aligning service water header B.

5. Opening and sticking of a relief valve is not modeled as an initiating event or as a system failure. Due to the multiple failures required, and the volume of makeup available, it is believed that makeup will exceed losses.

6. Failure of the heat exchanger bypass valves was not modeled. On loss of support, the temperature control valve fails in the maximum cooling condition. Consequently, the probability of these valves failing full open is very small, and is subsumed by failures of the individual heat exchanger trains (e.g., HX fouling).

7. Pump trains A and B are considered to be operating and pump train C is in standby mode.

8. Failure of one expansion joint is modeled as a system failure. It is assumed that the loss of cooling water inventory exceeds makeup.

9. The auto start function of the standby train; including circuitry, flow, and transmitters; is not specifically modeled. These failures are included in the Pump Fails to Start failure mode.

10. Check valves on the running trains, A and B, are not modeled. The statistical likelihood of a open check valve transferring closed is insignificant in relation to the rest of the pump train. However, on a standby system, failure of a check valve to open is significant, and is included in the model.

3.2.1.19.13 Logic Model and Results

The fault tree(s) is included in Tier 2 documentation, and is available upon request. Quantitative results are summarized in Section 3.3.5 (Tier 1).

Rev 0 (7/92)

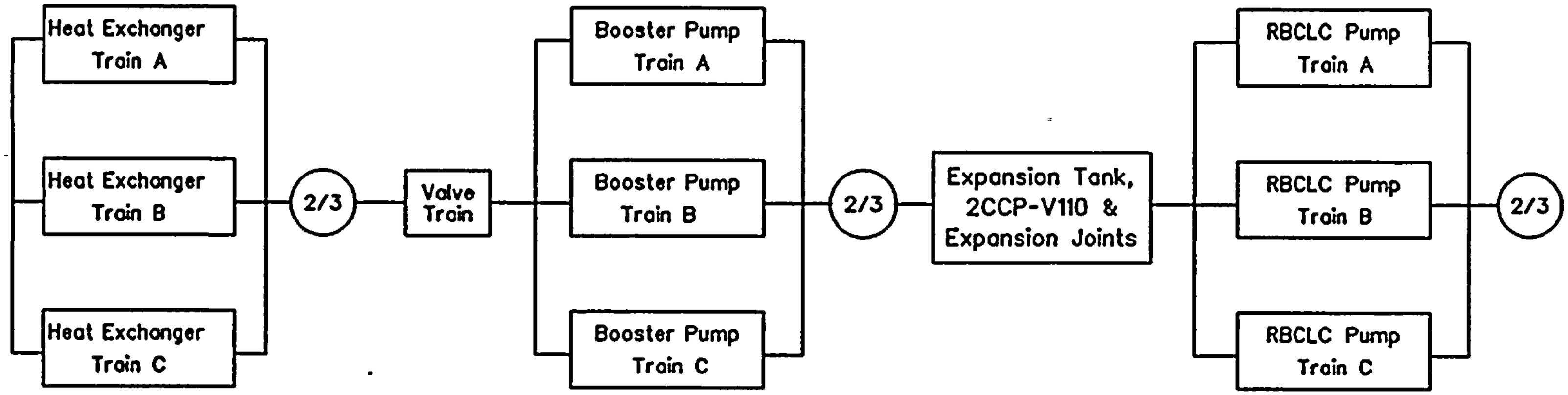


Figure 3.2.1.19-2
Reactor Building Closed
Loop Cooling Water

I: \NMP2IPE\DGN\2S19.DGN Jul. 07. 1992 08:53:34

REACTOR BUILDING CLOSED LOOP COOLING Component Block Descriptions								
Block	Mark No.	(Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
HEAT EXCH TRAIN A (RUNNING)								
	2CCP-V166		2CCP-E1A INLET VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2SWP-V55A		SERVICE WATER FROM 2CCP-E1A	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CCP-E1A		RBCLC HEAT EXCHANGER	RUPTURES	N/A	N/A	SERVICE WATER	FAILS
	2SWP-V45A		SERVICE WATER TO 2CCP-E1A	TRANSFER CLOSED	OPEN	N/A	.N/A	N/A
	2CCP-V169		2CCP-E1A OUTLET VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
HEAT EXCH TRAIN B (RUNNING)								
	2CCP-V167		2CCP-E1B INLET VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2SWP-V55B		SERVICE WATER FROM 2CCP-E1B	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CCP-E1B		RBCLC HEAT EXCHANGER	RUPTURES	N/A	N/A	SERVICE WATER	FAILS
	2SWP-V45B		SERVICE WATER TO 2CCP-E1B	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CCP-V170		2CCP-E1B OUTLET VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
HEAT EXCH TRAIN C (STANDBY)								
	2CCP-V168		2CCP-E1C INLET VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2SWP-V55C		SERVICE WATER FROM 2CCP-E1C	FAILS TO OPEN	CLOSED	N/A	N/A	N/A
	2SWP-V55C		SERVICE WATER FROM 2CCP-E1C	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CCP-E1C		RBCLC HEAT EXCHANGER	RUPTURES	N/A	N/A	SERVICE WATER	FAILS
	2SWP-V45C		SERVICE WATER TO 2CCP-E1C	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CCP-V171		2CCP-E1C OUTLET VALVE	FAILS TO OPEN	CLOSED	N/A	N/A	N/A
COMMON VALVES								
	2CCP-V172		TV108 INLET BLOCKING	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CCP-TV108		TEMPERATURE CONTROL VALVE	TRANSFER CLOSED	OPEN	OPEN	INSTRUMENT AIR	OPEN
	2CCP-V174		TV108 OUTLET BLOCKING	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
BOOSTER PUMP TRAIN A (RUNNING)								
	2CCP-V786		P3A SUCTION VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CCP-STRT1D		STRAINER	PLUGGED	N/A	N/A	N/A	N/A
	2CCP-P3A		BOOSTER PUMP 3A	FAILS TO RUN	RUNNING	RUNNING	2N[illegible]SWG013	STOP
	2CCP-V789		P3A DISCHARGE STOP CHECK VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
BOOSTER PUMP TRAIN B (RUNNING)								
	2CCP-V787		P3B SUCTION VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CCP-STRT1E		STRAINER	PLUGGED	N/A	N/A	N/A	N/A
	2CCP-P3B		BOOSTER PUMP 3B	FAILS TO RUN	RUNNING	RUNNING	2NNS-SWG015	STOP
	2CCP-V790		P3B DISCHARGE STOP CHECK VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
BOOSTER PUMP TRAIN C (STANDBY)								
	2CCP-V788		P3C SUCTION VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CCP-STRT1F		STRAINER	PLUGGED	N/A	N/A	N/A	N/A

REACTOR BUILDING CLOSED LOOP COOLING
Component Block Descriptions

Block	Mark No. (Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
	2CCP-P3C	BOOSTER PUMP 3C	FAILS TO START	STOPPED	RUNNING	2NNS-SWG014	STOP
	2CCP-P3C	BOOSTER PUMP 3C	FAILS TO RUN	STOPPED	RUNNING	2NNS-SWG014	STOP
	2CCP-V791	P3C DISCHARGE STOP CHECK VALVE	FAILS TO OPEN	CLOSED	N/A	N/A	N/A
TANK & EXPANSION JOINTS							
	2CCP-EJ1A	P1A SUCTION EXPANSION JOINT	RUPTURES	N/A	N/A	N/A	N/A
	2CCP-EJ1B	P1B SUCTION EXPANSION JOINT	RUPTURES	N/A	N/A	N/A	N/A
	2CCP-EJ1C	P1C SUCTION EXPANSION JOINT	RUPTURES	N/A	N/A	N/A	N/A
	2CCP-EJ2A	P1A DISCHARGE EXPANSION JOINT	RUPTURES	N/A	N/A	N/A	N/A
	2CCP-EJ2B	P1B DISCHARGE EXPANSION JOINT	RUPTURES	N/A	N/A	N/A	N/A
	2CCP-EJ2C	P1C DISCHARGE EXPANSION JOINT	RUPTURES	N/A	N/A	N/A	N/A
	2CCP-EJ3A	P3A SUCTION EXPANSION JOINT	RUPTURES	N/A	N/A	N/A	N/A
	2CCP-EJ3B	P3B SUCTION EXPANSION JOINT	RUPTURES	N/A	N/A	N/A	N/A
	2CCP-EJ3C	P3C SUCTION EXPANSION JOINT	RUPTURES	N/A	N/A	N/A	N/A
	2CCP-EJ4A	P3A DISCHARGE EXPANSION JOINT	RUPTURES	N/A	N/A	N/A	N/A
	2CCP-EJ4B	P3B DISCHARGE EXPANSION JOINT	RUPTURES	N/A	N/A	N/A	N/A
	2CCP-EJ4C	P3C DISCHARGE EXPANSION JOINT	RUPTURES	N/A	N/A	N/A	N/A
	2CCP-TK1	EXPANSION TANK	RUPTURES	N/A	N/A	N/A	N/A
	2CCP-V110	MANUAL VALVE FROM 2CCP-TK1	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
PUMP TRAIN A (RUNNING)							
	2CCP-V1	P1A SUCTION VALVE	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2CCP-STRT1A	STRAINER	PLUGGED	N/A	N/A	N/A	N/A
	2CCP-P1A	RBCLC PUMP 1A	FAILS TO RUN	RUNNING	RUNNING	2NNS-SWG012	STOP
	2CCP-V7	P1A DISCHARGE STOP CHECK VALVE	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
PUMP TRAIN B (RUNNING)							
	2CCP-V2	P1B SUCTION VALVE	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2CCP-STRT1B	STRAINER	PLUGGED	N/A	N/A	N/A	N/A
	2CCP-P1B	RBCLC PUMP 1B	FAILS TO RUN	RUNNING	RUNNING	2NNS-SWG015	STOP
	2CCP-V8	P1B DISCHARGE STOP CHECK VALVE	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
PUMP TRAIN C (STANDBY)							
	2CCP-V3	P1C SUCTION VALVE	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2CCP-STRT1C	STRAINER	PLUGGED	N/A	N/A	N/A	N/A
	2CCP-P1C	RBCLC PUMP 1C	FAILS TO START	STOPPED	RUNNING	2NNS-SWG014	STOP
	2CCP-P1C	RBCLC PUMP 1C	FAILS TO RUN	STOPPED	RUNNING	2NNS-SWG014	STOP
	2CCP-V9	P1C DISCHARGE STOP CHECK VALVE	FAILS TO OPEN	CLOSED	N/A	N/A	N/A

System 20

Turbine Building Closed Loop Cooling

3.2.1.20 Turbine Building Closed Loop Cooling Water

3.2.1.20.1 System Function

The Turbine Building Closed Loop Cooling Water (TBCLC) system provides demineralized cooling water to designated equipment in the Turbine Building and the Radwaste Building. The system is an intermediate cooling distribution loop that transfers heat from designated equipment to the service water system. A simplified diagram of the TBCLC system is provided in Figure 3.2.1.20-1.

3.2.1.20.2 Success Criteria

The top event in the support system event tree, TW, models TBCLC maintaining flow through two of three pump trains and two of three heat exchanger trains to provide adequate cooling to turbine building equipment. A success diagram is provided in Figure 3.2.1.20-2.

3.2.1.20.3 Support Equipment

TBCLC requires normal AC power and Service Water (SWP) for successful operation. Instrument Air is required for temperature control, but not for operation. Table 3.2.1.20-1 summarizes support systems required for all components that are modeled.

3.2.1.20.4 System Operation

The TBCLC system consists of three 50% capacity circulating pumps, three 50% capacity heat exchangers, and a makeup tank. The system is normally in operation with two pumps running and two heat exchangers in service. The remaining pump and heat exchanger are in standby. The surge and makeup tank water level is controlled automatically by the tank water level control valve, with additional water available from the makeup water system. System temperature is automatically maintained at 87.5°F by 2CCS-TV104; which bypasses flow around the heat exchangers as required.

3.2.1.20.5 Instrumentations and Controls

<u>Level Instrumentation</u>
The level (inches of water) in the surge tank provides signals to automatically open and close the system makeup water valve to maintain a proper system water inventory.

<u>Controls</u>
The TBCLC pumps have trip signals on low suction pressure. They will auto-start on low system pressure. These functions are not modeled. Individual four position control switches for the TBCLC pumps are located on Panel 601.

The TBCLC heat exchanger bypass temperature control station on Panel 601 has a push-button to select automatic or manual control, a slide bar to change valve position in manual control, and an indication of output signal strength.

In the Control Room, indications are provided for:

- TBCLC pump discharge pressure, and
- TBCLC system heat exchanger cooling water outlet temperature.

In the Control Room, alarms are provided for:

- TBCLC system trouble,
- Radiation monitor trouble/manually out of service/radiation status, and
- Process liquid radiation monitor activated.

3.2.1.20.6 Technical Specifications

None

3.2.1.20.7 Surveillance, Testing and Maintenance

Pumps in the TBCLC system are proven operable by their use during normal plant operations. The standby heat exchanger and pump are placed in service periodically to ensure their operability and to allow all pumps and heat exchangers to wear evenly.

3.2.1.20.8 References

N2-OP-14 Rev. 3
FSAR Section 9.2.7
12177-PID-14A-8

3.2.1.20.9 Initiating Event Potential

A rupture of the SWP-TBCLC heat exchanger could cause a flooding threat to plant equipment.

3.2.1.20.10 Equipment Location

All the major equipment, including the pumps and heat exchangers, are located in the Turbine Building at elevation 250'.

3.2.1.20.11 Operational Experience

There were no particular outstanding operational events relevant to this study. Plant specific component operational data is detailed in section 3.3.2.

3.2.1.20.12 Modeling Assumptions

1. Loss of instrument air to the temperature control valves is not modeled. The valves are designed to fail in the maximum cooling mode, which is considered acceptable.

2. Failure of any TBCLC expansion joint is considered a failure of the system. An operator would be required to manually isolate the failed train. For this reason, all TBCLC expansion joints are modeled under the same branch of the fault tree, not in the individual trains.

3. The heat exchanger bypass loop, including valve 2CCS-V106 is not modeled. On loss of support, the temperature control valve fails in the maximum cooling condition.

4. The Makeup Water System is not included in the TBCLC model. Because of the large quantities of water available in the makeup system tanks and piping of the closed system, it is assumed that adequate makeup is available for TBCLC.

5. Opening and sticking of a relief valves is not modeled. Due to the multiple failures required, and the volume of makeup available, it is believed that makeup will exceed losses.

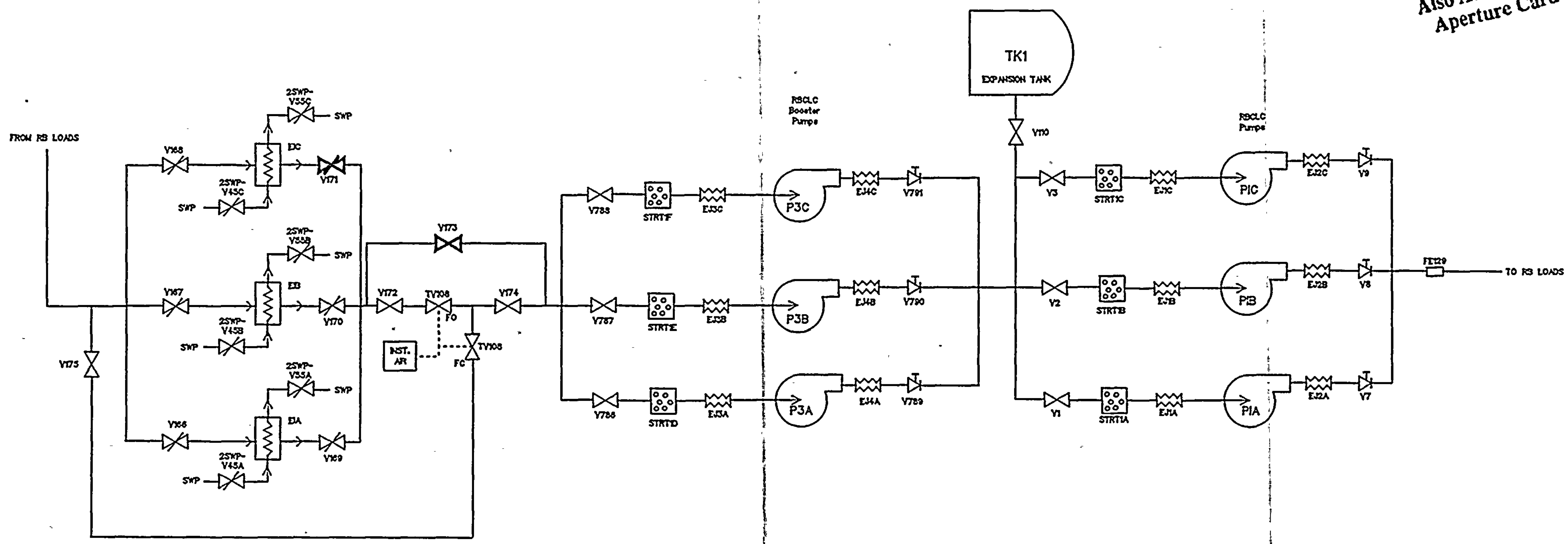
6. Flow elements are not modeled individually. They are considered part of the pump model.

7. TBCLC pump trains B and C are assumed to be operating. Pump train A is in standby mode.

3.2.1.20.13 Logic Model and Results

The fault tree(s) is included in Tier 2 documentation, and is available upon request. Quantitative results are summarized in Section 3.3.5 (Tier 1).

Rev 0 (7/92)



References: PID-13A-10
PID-13B-8
PID-13C-7
PID-13D-9
PID-13E-6
PID-13F-6
PID-11D-8

Figure 3.2.1.19-1
Reactor Building Closed
Loop Cooling Water

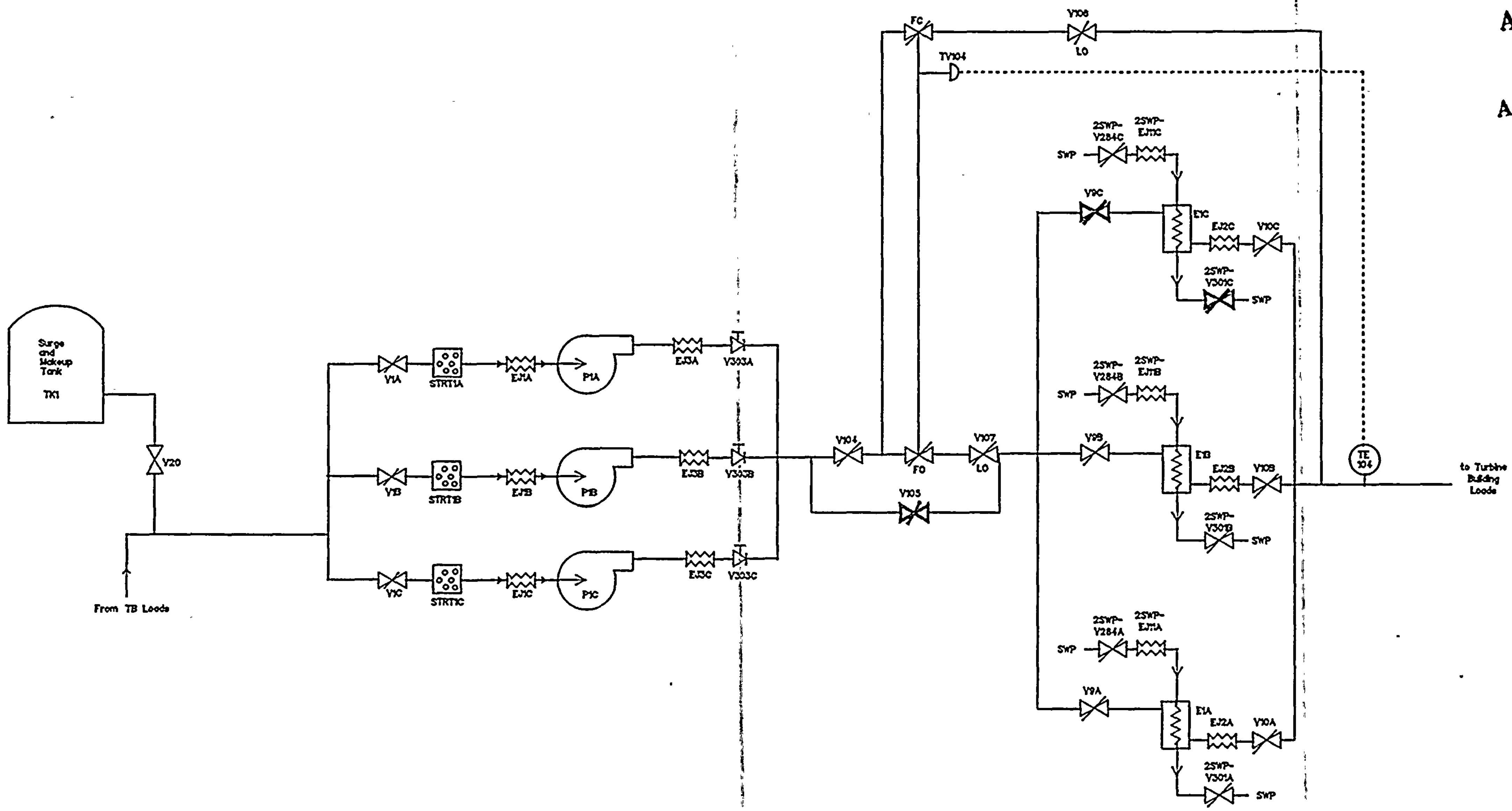
SI APERTURE CARD

Also Available On Aperture Card

I:\NMP2IPE\DGN\2S19.DGN Jul. 07, 1992 10:48:14

9208050191-28

Rev 0 (7/92)



References: PID-14A-8
PID-14B-7
PID-14C-7
PID-14D-8
PID-14E-7
PID-14F-7
PID-11J-9

Figure 3.2.1.20-1
Turbine Building Closed
Loop Cooling System

I:\NMP2IPE\DGN\2S20.DGN Jul. 07, 1992 10:50:20

9208050191-29

Rev 0 (7/92)

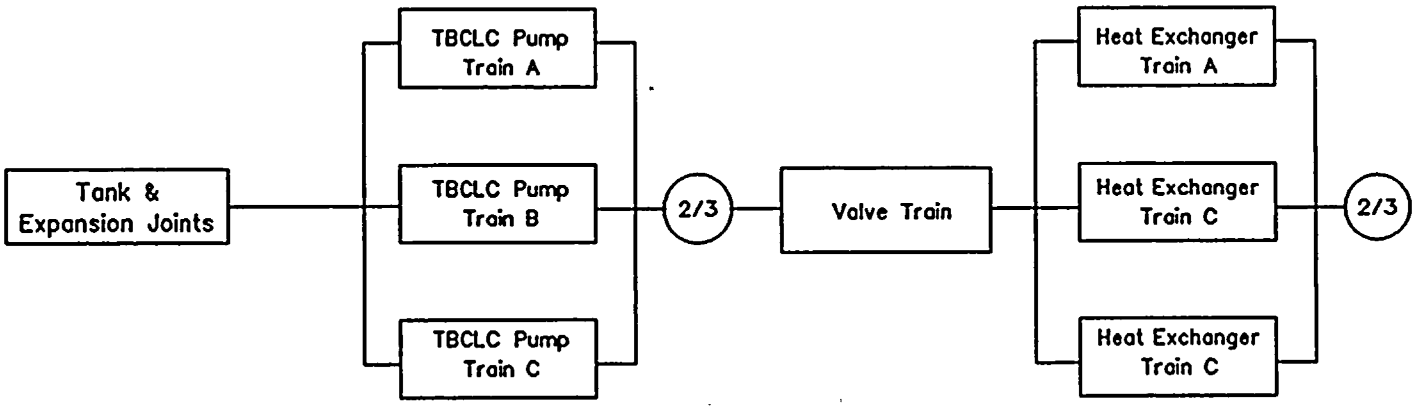


Figure 3.2.1.20-2
Turbine Building Closed
Loop Cooling Water

I:\NMP2IPE\DGN\2S20.DGN Jul. 07. 1992 08:55:43

TURBINE BUILDING CLOSED LOOP COOLING Component Block Descriptions							
Block	Mark No. (Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
TANK & EXP JOINTS							
	2CCS-TK1	SURGE TANK	RUPTURES	N/A	N/A	N/A	N/A
	2CCS-V20	SURGE TANK OUTLET ISOLATION	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CCS-EJ1A	P1A SUCTION EXPANSION JOINT	RUPTURES	N/A	N/A	N/A	N/A
	2CCS-EJ1B	P1B SUCTION EXPANSION JOINT	RUPTURES	N/A	N/A	N/A	N/A
	2CCS-EJ1C	P1C SUCTION EXPANSION JOINT	RUPTURES	N/A	N/A	N/A	N/A
	2CCS-EJ2A	E1A OUTLET EXPANSION JOINT	RUPTURES	N/A	N/A	N/A	N/A
	2CCS-EJ2B	E1B OUTLET EXPANSION JOINT	RUPTURES	N/A	N/A	N/A	N/A
	2CCS-EJ2C	E1C OUTLET EXPANSION JOINT	RUPTURES	N/A	N/A	N/A	N/A
	2CCS-EJ3A	P1A DISCHARGE EXPANSION JOINT	RUPTURES	N/A	N/A	N/A	N/A
	2CCS-EJ3B	P1B DISCHARGE EXPANSION JOINT	RUPTURES	N/A	N/A	N/A	N/A
	2CCS-EJ3C	P1C DISCHARGE EXPANSION JOINT	RUPTURES	N/A	N/A	N/A	N/A
	2SWP-EJ11C	SERVICE WATER TO E1C EXP JOINT	RUPTURES	N/A	N/A	N/A	N/A
	2SWP-EJ11B	SERVICE WATER TO E1B EXP JOINT	RUPTURES	N/A	N/A	N/A	N/A
	2SWP-EJ11A	SERVICE WATER TO E1A EXP JOINT	RUPTURES	N/A	N/A	N/A	N/A
PUMP TRAIN A (STANDBY)							
	2CCS-V1A	PUMP 1A SUCTION ISOLATION	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CCS-STRT1A	STRAINER	PLUGGED	N/A	N/A	N/A	N/A
	2CCS-P1A	TBCLC PUMP 1A	FAILS TO START	STOPPED	RUNNING	2NNS-SWG011	STOP
	2CCS-P1A	TBCLC PUMP 1A	FAILS TO RUN	STOPPED	RUNNING	2NNS-SWG013	STOP
	2CCS-V303A	P1A DISCHARGE STOP CHECK VALVE	TRANSFER CLOSED	CLOSED	N/A	N/A	N/A
PUMP TRAIN B (RUNNING)							
	2CCS-V1B	PUMP 1B SUCTION ISOLATION	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CCS-STRT1B	STRAINER	PLUGGED	N/A	N/A	N/A	N/A
	2CCS-P1B	TBCLC PUMP 1B	FAILS TO RUN	RUNNING	RUNNING	2NNS-SWG013	STOP
	2CCS-V303B	P1B DISCHARGE STOP CHECK VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
PUMP TRAIN C (RUNNING)							
	2CCS-V1C	PUMP 1C SUCTION ISOLATION	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CCS-STRT1C	STRAINER	PLUGGED	N/A	N/A	N/A	N/A
	2CCS-P1C	TBCLC PUMP 1C	FAILS TO RUN	RUNNING	RUNNING	2NNS-SWG012	STOP
	2CCS-V303C	P1C DISCHARGE STOP CHECK VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
VALVE TRAIN							
	2CCS-V107	TEMP CNTL VALVE OUTLET ISOLATE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CCS-V104	TEMP CNTL VALVE INLET ISOLATE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CCS-TV104	TEMPERATURE CONTROL VALVE	TRANSFER CLOSED	OPEN	N/A	INSTRUMENT AIR	OPEN
	2CCS-V105	TEMP CNTL VALVE ISOL. BYPASS	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
HEAT EXCH TRAIN A (RUNNING)							
	2CCS-V9A	2CCS-E1A INLET ISOLATION	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CCS-E1A	TBCLC HEAT EXCHANGER	RUPTURES	N/A	N/A	SERVICE WATER	N/A

TURBINE BUILDING CLOSED LOOP COOLING
Component Block Descriptions

Block	Mark No. (Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
	2CCS-V10A	2CCS-E1A OUTLET ISOLATION	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2SWP-V284A	SERVICE WATER TO 2CCS-E1A	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2SWP-V301A	SERVICE WATER FROM 2CCS-E1A	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
HEAT EXCH TRAIN B (RUNNING)							
	2CCS-V9B	2CCS-E1B INLET ISOLATION	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CCS-E1B	TBCLC HEAT EXCHANGER	RUPTURES	N/A	N/A	SERVICE WATER	N/A
	2CCS-V10B	2CCS-E1B OUTLET ISOLATION	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2SWP-V284B	SERVICE WATER TO 2CCS-E1B	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2SWP-V301B	SERVICE WATER FROM 2CCS-E1B	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
HEAT EXCH TRAIN C (STANDBY)							
	2CCS-V9C	2CCS-E1C INLET ISOLATION	FAILS TO OPEN	CLOSED	N/A	N/A	N/A
	2CCS-E1C	TBCLC HEAT EXCHANGER	RUPTURES	N/A	N/A	SERVICE WATER	N/A
	2CCS-V10C	2CCS-E1C OUTLET ISOLATION	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2SWP-V301C	SERVICE WATER FROM 2CCS-E1C	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2SWP-V301C	SERVICE WATER FROM 2CCS-E1C	FAILS TO OPEN	CLOSED	N/A	N/A	N/A
	2SWP-V284C	SERVICE WATER TO 2CCS-E1C	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A

System 21

Condensate and Feedwater

3.2.1.21 Condensate and Feedwater Systems

3.2.1.21.1 System Function

The main condenser is designed to provide a heat sink for condensing steam from the turbine, turbine bypass, and miscellaneous vents and drains. If the main steam system is isolated, the condensate storage tanks provide makeup to the condenser. Makeup from the condensate storage tanks to the condenser is shown in Figure 3.2.1.21-2. Figure 3.2.1.21-3 shows the paths under consideration, into and out of the condenser. Figure 3.2.1.21-4 shows the condenser air removal system.

The feedwater system is designed to maintain adequate reactor water level to compensate for 115% of nuclear boiler rated steam flow. The condensate system supports the feedwater system by transporting condensate from the main condenser hotwell to the reactor feed pump suction header. The condensate and feedwater systems are shown in Figure 3.2.1.21-1.

3.2.1.21.2 Success Criteria

Successful feedwater operation provides adequate heat removal and core flow to replace water lost to nuclear boil off. In the event tree, four top event models are developed to cover continued availability of the condenser (CN) the condensate storage tanks (TA & TB) and the feedwater supply (FW) to the RPV after a plant trip. The following summarizes the top event model success criteria:

Top Event	Success Criteria
CN	The condenser, its support systems (Circulating Water, MSIVs, Turbine Bypass, etc.) are available after a plant trip to provide a heat sink for 24 hours. Operators must put the mode switch in SHUTDOWN prior to MSIV closure signal due to low RPV pressure.
FW	Condensate and Feedwater systems, and their support systems, continue to be available for 24 hours after a plant trip. Makeup to the condenser hotwell from the condensate storage tanks is required. Operators must restore feedwater after a small LOCA induced feedwater isolation and RPV Level 8 feedwater trip (high drywell pressure is assumed to initiate HPCS and raise the RPV to Level 8).
TA	Condensate Storage Tank A available with initial inventory at 80%.
TB	Condensate Storage Tank B available with initial inventory at 80%.

3.2.1.21.3 Support Systems

3.2.1.21.3a Condensate and Feedwater Systems (FW)

The following summarizes support equipment required for feedwater system operation.

- Instrument Air System The condensate demineralizer bypass valve (AOV109) and the heater and drain cooler bypass valve (AOV101) open automatically on a turbine trip with reactor power over 80%. These valves fail as is on a loss of instrument air.

  Each feedwater pump has a minimum flow recirculation line. The flow control valves (2FWR-FV2A, B & C) fail open on loss of instrument air. Each feed pump is interlocked with its respective flow control valve so that a pump will not start if the flow control valve doesn't open.

- Normal AC and DC Power Distribution Required to power pumps and motor operated valves as shown below in the list of major components.

- Turbine Building Closed Loop Cooling Provides cooling water to the pumps. TBCLC is dependent on normal AC and DC power.

- Condensate Storage and Transfer Required for condenser makeup to the hotwell. Pumps depend on normal AC and DC Power. Level control valve (LV103) fails open on loss of support.

- Standby and Emergency AC Distribution Containment isolation valves 2FWS*MOV21A and 2FWS*MOV21B fail as is on a loss of power.

- Containment Isolation Signals that Isolate FW Remote manual signal operates 2FWS*MOV21A and 2FWS*MOV21B.

Major components are:

Component	Support Equipment
Condensate Transfer Pump A	2NJS-US9A
Condensate Transfer Pump B	2NJS-US9B
Cond. Pump Suction (MOV63A)	2NHS-MCC010A
Cond. Pump Suction (MOV63B)	2NHS-MCC010B
Cond. Pump Suction (MOV63C)	2NHS-MCC010C
Condensate Pump 2CNM-P1A	2NNS-SWG011
Condensate Pump 2CNM-P1B	2NNS-SWG013
Condensates Pump 2CNM-P1C	2NNS-SWG011, 2NNS-SWG013
Cond. Pump Discharge (MOV3A)	2NHS-MCC010A
Cond. Pump Discharge (MOV3B)	2NHS-MCC010B
Cond. Pump Discharge (MOV3C)	2NHS-MCC010C

Component	Support Equipment
Cond. Demin. Bypass (AOV109)	Instrument Air
2CNM-SOV109A	2SCA-PNL403 (NA & NB)
2CNM-SOV109B	2SCA-PNL403 (NA & NB)
Cond. Booster Pump Suction (MOV7A)	2NHS-MCC010A
Cond. Booster Pump Suction (MOV7B)	2NHS-MCC010B
Cond. Booster Pump Suction (MOV7C)	2NHS-MCC010C
Condensate Booster Pump P2A	2NPS-SWG001
Condensate Booster Pump P2B	2NPS-SWG003
Condensate Booster Pump P2C	2NPS-SWG001, 2NPS-SWG003
1st-5th Pt. Heater Bypass (AOV101)	Instrument Air
2CNM-SOV101A	2SCA-PNL403 (NA & NB)
2CNM-SOV101B	2SCA-PNL403 (NA & NB)
2CNM-SOV101C	2SCA-PNL403 (NA & NB)
2CNM-SOV101D	2SCA-PNL403 (NA & NB)
Feedwater Pump Bypass (MOV122)	2NHS-MCC010
Feedpump Suction Valve (2CNM-MOV84A)	2NHS-MCC010A
Feedpump Suction Valve (2CNM-MOV84B)	2NHS-MCC010B
Feedpump Suction Valve (2CNM-MOV84C)	2NHS-MCC010C
Reactor Feedwater Pump (2FWS-P1A)	2NPS-SWG001
Reactor Feedwater Pump (2FWS-P1B)	2NPS-SWG003
Reactor Feedwater Pump (2FWS-P1C)	2NPS-SWG001, 2NPS-SWG003
2FWR-FV2A	Instrument Air
2FWR-SOVX2A	2SCI-PNLA101 (NA & NB)
2FWR-SOVY2A	2SCI-PNLA101 (NA & NB)
2CNM-FT68A	2CEC*PNL731
2FWR-FV2B	Instrument Air
2FWR-SOVX2B	2SCI-PNLA102 (NA & NB)
2CNM-FT68B	2CEC*PNL731
2FWR-FV2C	Instrument Air
2FWR-SOVX2C	2SCI-PNLA101 (NA & NB)
2FWR-SOVY2C	2SCI-PNLA101 (NA & NB)
2CNM-FT68C	2CEC*PNL731
FW Level Control (2FWS-LV10A)	2NHS-MCC003A
FW Level Control (2FWS-LV10B)	2NHS-MCC003B
FW Level Control (2FWS-LV10C)	2NHS-MCC003C
Feedpump Discharge (2FWS-MOV47A)	2NHS-MCC003A
Feedpump Discharge (2FWS-MOV47B)	2NHS-MCC003B
Feedpump Discharge (2FWS-MOV47C)	2NHS-MCC003C
6th Point Heater Bypass (2FWS-MOV102)	2NHS-MCC0003
FW Outside Isolation (MOV21A)	2EHS*MCC102A
FW Outside Isolation (MOV21B)	2EHS*MCC102C

3.2.1.21.3b Condenser as a Heat Sink (CN)

The following summarizes support equipment required for system operation.

- Main and Auxiliary Steam: MSIV closure makes the condenser unavailable as a heat sink. MSIVs will close on a loss of instrument air/nitrogen or on a loss of both vital buses (2VBB-UPS3A and UPS3B). The MSIVs receive the following key closure signals:

  Reactor Water Level LO-LO-LO (17.8", decreasing)
  Low Condenser Vacuum (8.5" Hg)
  Main Steam Line Low Pressure (766 psig)
  Main Steam Line High Flow (103 psid, 140%)
  Main Steam Line High Radiation
  Manual Isolation

  Auxiliary Steam is required by the SJAEs for condenser air removal. The steam for this system comes directly from the main steam system. The steam inlet valves (2ASS-MOV148 and 2ASS-MOV152) are motor operated and are driven by normal AC power. Pressure control valves (2ASS-PV107 and 2ASS-PV139) fail open on loss of instrument air, they can also be bypassed with manually operated valve 2ASS-HCV151.

- 120V AC Vital Bus 2VBB-UPS3A & 2VBB-UPS3B: Loss of both UPSs will result in MSIV closure, loss of only one UPS will not actuate the MSIV trip solenoids.

- Instrument Air: Outboard MSIVs are supplied instrument air to open, each outboard MSIV has a 4 ft^3 accumulator (enough air to open the MSIV once in case of pneumatic supply failure). Check valves on the accumulators guard against air leakage.

- Nitrogen: Inboard MSIVs are supplied nitrogen to open, each inboard MSIV has a 4 ft^3 accumulator (enough nitrogen to close the MSIV once). Check valves on the accumulators guard against air leakage.

- Turbine Bypass: Five turbine bypass valves provide a path for 25% of the rated steam flow into the condenser. These valves depend on the turbine EHC system. Failure of this path makes the condenser unavailable as a heat sink.

- Turbine Electro-Hydraulic Control System: On loss of hydraulic pressure, the accumulators for the turbine bypass valves will hold the valves open for approximately one minute. The hydraulic fluid is pumped from the reservoir through one of the two 100% trains. The pumps are powered by Normal AC power, so on a loss of power, the turbine bypass valves will loose pressure and close.

- Condenser Air Removal System (SJAE Trains): These are required to maintain condenser vacuum and are dependent on auxiliary steam for support. Loss of condenser vacuum isolates MSIVs making the condenser unavailable as a heat sink.

- Circulating Water System: At least three of six trains are required to maintain condenser vacuum. Circulating Water provides the cooling water to the condenser and depends on normal AC and DC Power and TBCLC.

- Service Water System: Provides 25,000 gpm make-up water to the Circulating Water System and provides cooling water to TBCLC.

- Turbine Building Closed Loop Cooling: Provides cooling water to the circulating water pumps and coolers for the turbine Electro Hydraulic Control (EHC) system.

- Condensate Transfer and Storage: Required for condenser makeup to the hotwell. The pumps depend on normal AC and DC Power. Level control valve (LV103) fails open on loss of support.

Major components are:

Component	Support Equipment
Inboard MSIVs 2MSS*AOV6 (A-D)	Instrument Nitrogen
Outboard MSIVs 2MSS*AOV7 (A-D)	Instrument Air
Inboard MSIV Trip Solenoid A	2VBS*PNLB106
Inboard MSIV Trip Solenoid B	2VBS*PNLA106
Outboard MSIV Trip Solenoid A	2VBS*PNLA105
Outboard MSIV Trip Solenoid B	2VBS*PNLB105
Circ. Water Pump Suction (MOV2A)	2NHS-MCC015A
Circ. Water Pump Suction (MOV2B)	2NHS-MCC015B
Circ. Water Pump Suction (MOV2C)	2NHS-MCC015C
Circ. Water Pump Suction (MOV2D)	2NHS-MCC015A
Circ. Water Pump Suction (MOV2E)	2NHS-MCC015B
Circ. Water Pump Suction (MOV2F)	2NHS-MCC015C
Circulating Water Pump A	2NPS-SWG001
Circulating Water Pump B	2NPS-SWG003
Circulating Water Pump C	2NPS-SWG001
Circulating Water Pump D	2NPS-SWG003
Circulating Water Pump E	2NPS-SWG001
Circulating Water Pump F	2NPS-SWG003
Condenser A Outlet (MOV5A)	2NHS-MCC010B
Condenser A Outlet (MOV5B)	2NHS-MCC010A
Condenser B Outlet (MOV5C)	2NHS-MCC010B
Condenser B Outlet (MOV5D)	2NHS-MCC010A
Condenser c Outlet (MOV5E)	2NHS-MCC010B
Condenser C Outlet (MOV5F)	2NHS-MCC010A
2ASS-MOV148	2NHS-MCC010 (NA & NB)
2ASS-MOV152	2NHS-MCC010 (NA & NB)
2ASS-PV107	
2ASS-SOV107	2SCI-PNLB101(NA & NB)
2ASS-PV139	
2ASS-SOV139	2SCI-PNLB101(NA & NB)

3.2.1.21.4 System Operation

The condenser is normally operating providing a heat sink for reactor steam. The steam flows through the main steam isolation valves (MSIVs) and into the condenser through either the turbine or the turbine bypass valves (which can bypass up to 25% rated steam flow).

The operators must put the mode switch in SHUTDOWN to prevent MSIV closure on a LOW RPV PRESSURE SIGNAL (P < 755 psig). The MSIVs will isolate on a RPV Level 1 signal (17.8 in. and dropping). MSIV closure isolates the steam flow path between the reactor and the condenser, disabling the condenser's function as a heat sink.

The condenser air removal system, specifically the SJAE system, maintains a vacuum of greater than 28.5 in Hg in the condenser. One of the two SJAE trains must be operational to maintain condenser vacuum.

The Circulating Water system provides cooling water to the condenser. There are six circulating water pumps, of which only three are required for full power operation. Each condenser outlet valve will open or close when the associated circulating water pump starts or stops. Each pump is interlocked with its respective condenser outlet valve, and will stop upon valve closure.

As water leaves the condensers, it collects in the hotwells and the water collection box under condenser A. The hotwells are designed to provide water for five minutes of full power operation. The condensate transfer system normally provides make-up water to the condenser as needed (up to 500 gpm); there is emergency make-up to the hotwells on a condenser hotwell Lo-Lo signal.

There are two 450,000 gallon, reinforced fiberglass, condensate storage tanks. Each tank has a reserve of 135,000 gallons; tank A uses this reserve to supply RCIC and Tank B supplies HPCS. The tanks are cross-tied above the 135,000 gallon mark so they have the same amount of water. There is sufficient water to supply RPV injection for a 24-hour period (supplying HPCS, RCIC and the feedwater system), in the event of transients and small LOCAs, so the only failure mode considered is a tank rupture. If this water supply is depleted, the operator must manually start make-up to the CSTs.

The condensate system is normally operating. Three 50% capacity condensate pumps take suction from the collection box under condenser A. The pumps can be controlled either manually or automatically. There are also three 50% condensate booster pumps. These pumps can be controlled either manually or automatically.

The feedwater system is normally operating. There are three feedwater pumps, each capable of providing 68% NBR flow at runout conditions. On loss of control signal, the feedwater flow control valves fail as is. The valves are locked in the position the valve was in just prior to the loss of control signal. On an ATWS signal the valves are driven to the closed position and held there for 30 seconds and valve control is switched from automatic to manual. The feedwater flow is recirculated to the condenser via valves 2FWR-FV2A,2B, and 2C. After the 30 second time delay has timed out the operator can control feedwater flow either manually or switch back to automatic control.

The feedwater pumps must be started and stopped manually. Interlocks prevent operation of the feed pumps when reactor water level is high; the pumps will trip on a RPV Level 8 signal. When the Level 8 condition has been cleared, the pumps can be reset and started from the control room.

3.2.1.21.5 Instrumentation and Controls

There are controls in the Control Room for both manual and automatic operation of each system.

Indication is provided for the following:

Condensate:

- Condenser vacuum
- Condensate pump discharge header and flow, recirculation valve position and motor current.
- Condensate low-pressure system flow.
- Condensate booster pump discharge header and flow, recirculation valve position and motor current.
- Feedwater low flow control valve flow.
- Reactor feed pump suction pressure.
- Local differential pressure across the condensate pump suction strainers.

Feedwater

- Reactor feed pump discharge pressure
- Feedwater cycle cleanup flow
- Reactor feed pump suction flow
- Reactor feed pump recirculation valve position
- Reactor feed pump bearing seal water ΔP

Condensate Storage and Transfer

- Condensate Transfer Pumps 1A/1B status
- Condensate Transfer Pumps header flow
- Condensate Storage Tank 1A/1B level

Condenser Air Removal System

- Main Condenser vacuum
- Main Condenser vacuum breakers current
- Seal Water pumps current
- Vacuum Pump current
- Pre-coolers inlet isolation valves position and flow
- AOV105 position & flow
- MOV5A, 5B, 5C vacuum breakers position and flow

Circulating Water System

- Circulating water pump discharge valve position
- Circulating water pump various temperatures

Alarms are provided for:

Condensate

- Condensate Pump parameters
- Condensate system trouble, and no backup pump available
- Condensate booster pump parameters
- Condensate booster pump system trouble, and no backup pump available.
- Condensate system low flow

Feedwater

- Reactor feed pump discharges pressures
- Feedwater cycle cleanup flow
- Reactor feed pump suction flow
- Reactor feed pump recirculation valve position
- Reactor feed pump bearing seal water ΔP

Condensate Storage and Transfer

- Condensate Transfer Pump 1A/1B auto-trip/fail to start
- Condensate Transfer Pump 1A/1B auto-start
- Condensate Transfer Pump 1A/1B motor electric fault
- Condensate Transfer Pump discharge header demand flow low
- Condensate Transfer Pump discharge header pressure flow
- Condensate Transfer Pump discharge header flow high
- Condensate Storage Tank 1A/1B level high-high
- Condensate Storage Tank 1A/1B level high
- Condensate Storage Tank 1A/1B level lo-lo
- Condensate Storage Tank 1A/1B level lo
- Condenser Hotwell high/lo-lo

Air Removal System

- Air Ejector 2A/2B auxiliary steam flow low
- Air Ejector 2A/2B inlet steam Header pressure high/low
- Condenser air removal recovery tank level hi-hi/lo-lo

Circulating Water System

- Circulating Water Pump auto-trip
- Circulating Water Pump motor overload/electric fault
- Circulating Water Pump suction pressure low
- Condenser Discharge eater boxes level low
- Circulating Water Pump bearing/winding temperature high
- Cooling Tower Basin water temperature high
- Cooling Tower Basin water level high/low
- Cooling Tower Screens differential level >6"

3.2.1.21.6 Technical Specification

The only technical specifications on this system are for Gaseous Effluents monitoring. They will not be summarized here, the specific sections are 3/4.11.2 and 3/4.3.7.11.

3.2.1.21.7 Surveillance, Testing and Maintenance

The type "C" feedwater valves and the "C" containment isolation valves are leak rate tested at least every 18 months, position indication is also verified every 18 months.

3.2.1.21.8 References

N2-OP-1, Rev. 7: Main and Auxiliary Steam System
N2-OP-3, Rev. 4: Condensate and Feedwater Systems
N2-OP-4, Rev. 1: Condensate Storage and Transfer System
N2-OP-5, Rev. 2: Condensate Demineralizer System
N2-OP-9, Rev. 3: Condenser Air Removal System
N2-OP-10A, Rev. 2: Circulating Water System
N2-OP-11, Rev. 4: Service Water System
N2-OP-21, Rev. 2: Main Turbine System
N2-OP-23, Rev. 1: Turbine Electro-Hydraulic Control System

FSAR Section 10.2: Turbine Generator
FSAR Section 10.3: Main Steam Supply System
FSAR Section 10.4: Other Features of the Steam and Power Conversion System

PIDs are listed on Figures 3.2.1.21-1 through 3.2.1.21-4 as appropriate.

3.2.1.21.9 Initiating Event Potential

Loss of condenser, loss of feedwater, MSIV closure and loss of support systems are included as initiating events.

3.2.1.21.10 Equipment Location

The outboard feedwater isolation valves are in the main steam tunnel. The inboard valves are in primary containment. All other components considered in the condensate and feedwater system, including the condensers, condensate pumps and feedwater pumps, are located in the Turbine Building or the heater bay.

The MSIVs are located inside primary containment and in the main steam tunnel. The turbine bypass valves are in the Turbine Building. The circulating water components are found in either the Turbine Building or the screenwell area. All components from the condenser air removal system are located in the Turbine Building.

The components that have been modeled from the condensate storage and transfer systems are in either the Condensate Storage Tank Building or the Turbine Building.

3.2.1.21.11 Operating Experience

Plant specific data was used to determine whether average storage tank levels exceeded those necessary to assure a 24 hour supply of water in case of a plant trip. Based on 4-1/2 months of operating data, average tank levels were 419,000 gals. A derived worst case scenario considered a tank level of 318,845 gallons, (70% capacity) which would provide sufficient inventory to remove decay heat for a 24 hour period.

3.2.1.21.12 Modeling Assumptions

1. Condenser vacuum can be maintained with only 25% of the rated steam flow in from the turbine bypass valves.

2. The hogging system cannot be used to maintain condenser vacuum, it is used only during start-up to initiate the vacuum (it draws a vacuum of 23 in Hg.).

3. The fault tree models are simplified models with equipment failures grouped into one basic event. Given that an initiating event or support system does not cause failure, the basic event represents unavailability of the specific function. Unavailability of the function (the basic event) is judgmentally based since no accurate data exists on balance of plant reliability after a plant trip. For example, condensate and feedwater supply to the RPV (top event FW) would be set to guaranteed failure given any of the following conditions:

    - Initiating event is a loss of feedwater.
    - Initiating events or support system failure such as normal AC power, instrument air, RBCLC or CST failure.

    Given the above conditions do not exist, the availability of condensate and feedwater depend on continued operation of each system and makeup from the CSTs. In the event of small LOCA, an operator action is included which recovers a pump trip occurring at level 8. It is assumed that high drywell pressure will initiate HPCS.

4. The dependency on UPS supplies is not considered since two non-divisional UPS supplies must fail to cause MSIV closure, and this is considered an unlikely event. Loss of normal AC power is a much more likely cause of MSIV closure and loss of condenser.

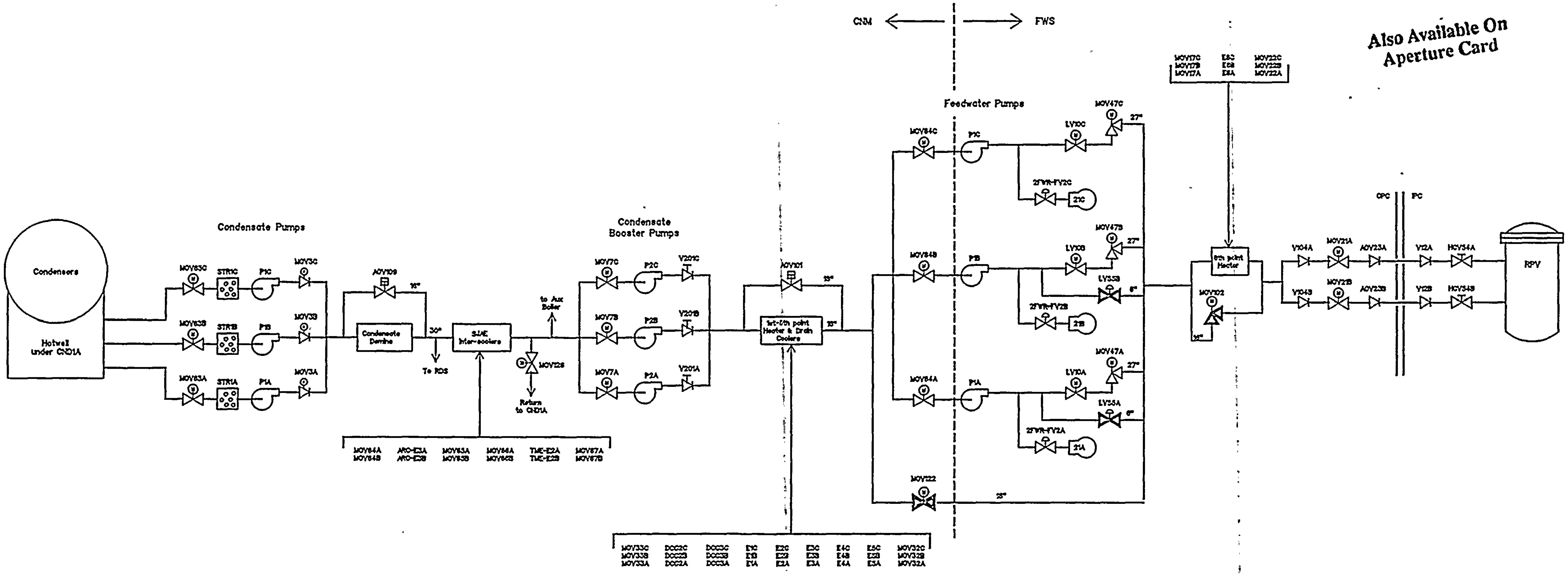
3.2.1.21.13 Logic Model and Results

The fault tree(s) is included in Tier 2 documentation, and is available upon request. Quantitative results are summarized in Section 3.3.5 (Tier 1).

Rev 0 (7/92)

SI
APERTURE
CARD

Also Available On
Aperture Card



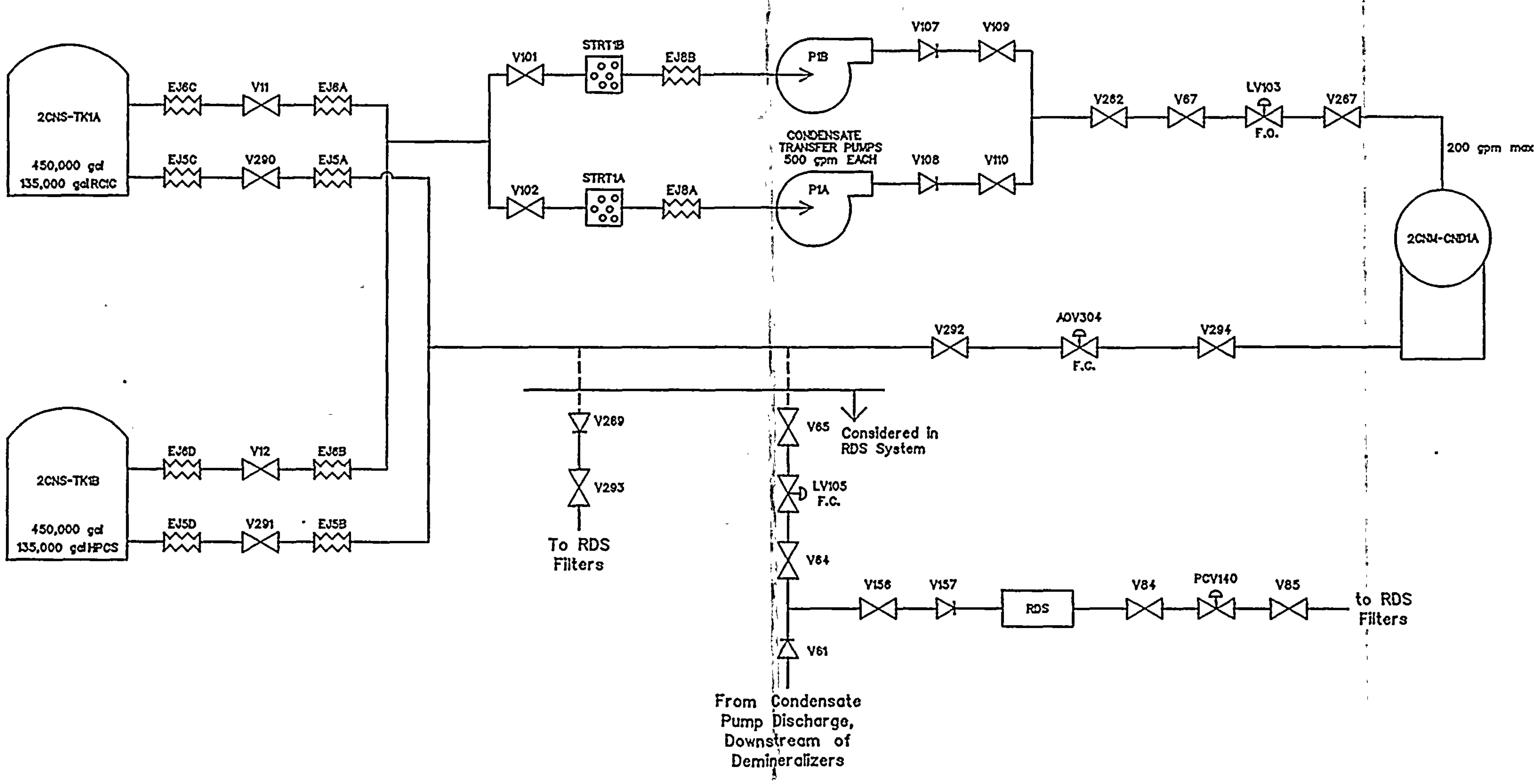
References: PID-3A-10
PID-3B-9
PID-3C-9
PID-3D-6
PID-3E-7
PID-6A-10
PID-6B-9
PID-6C-10
PID-6D-8
PID-6E-5

Figure 3.2.1.21-1
Condensate and Feedwater

I:\NMP2IPE\DGN\2S21.DGN Jul. 07, 1992 11:09:34

9208050191-30

Rev 0 (7/92)



References: PID-4A-8
PID-4B-9
PID-4C-7

Figure 3.2.1.21-2
Condensate Transfer - Makeup to Condenser and Control Rod Drive System

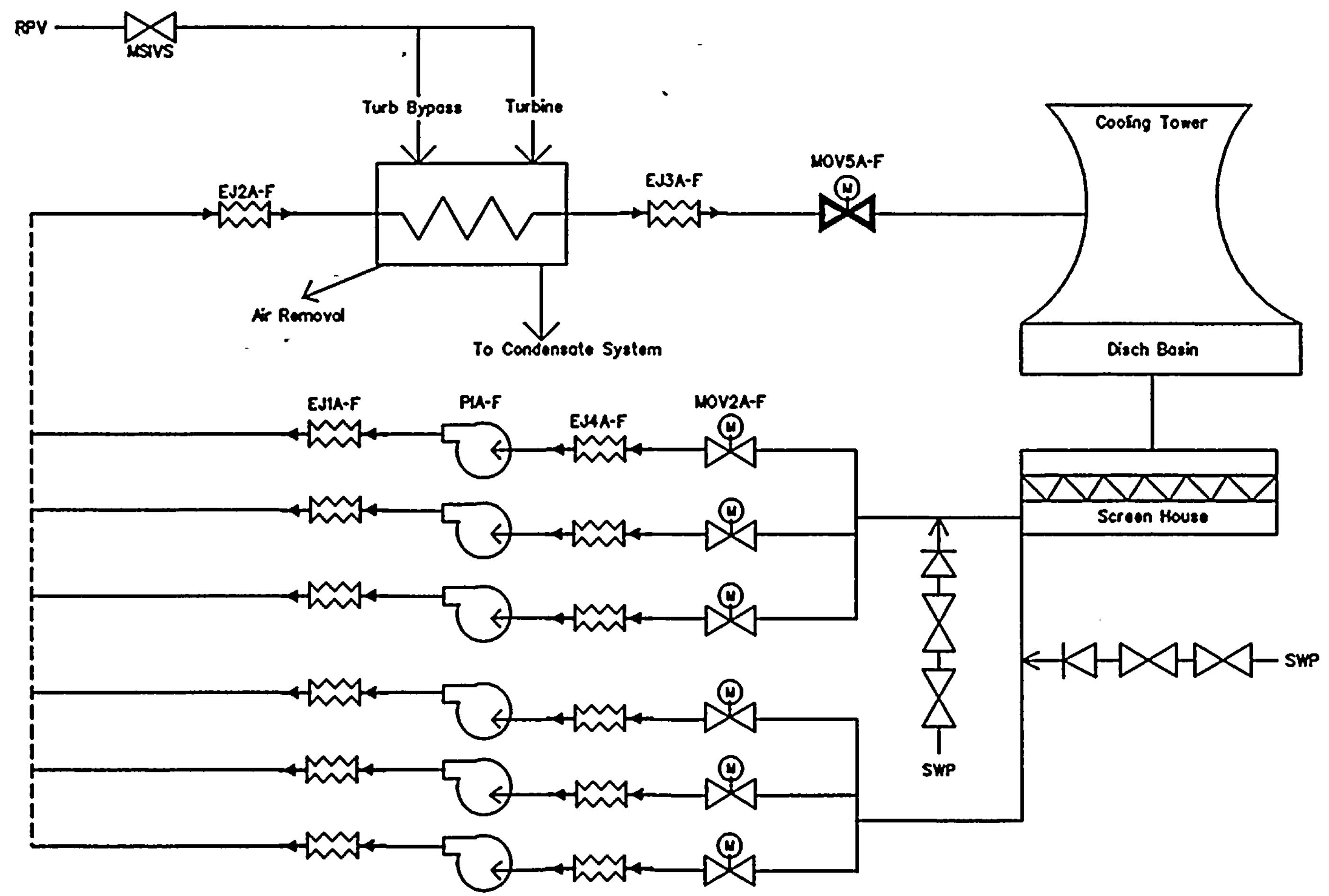
I:\NMP2IPE\DGN\2S21.DGN Jul. 07, 1992 11:10:46

SI APERTURE CARD

Also Available On Aperture Card

9208050191-31

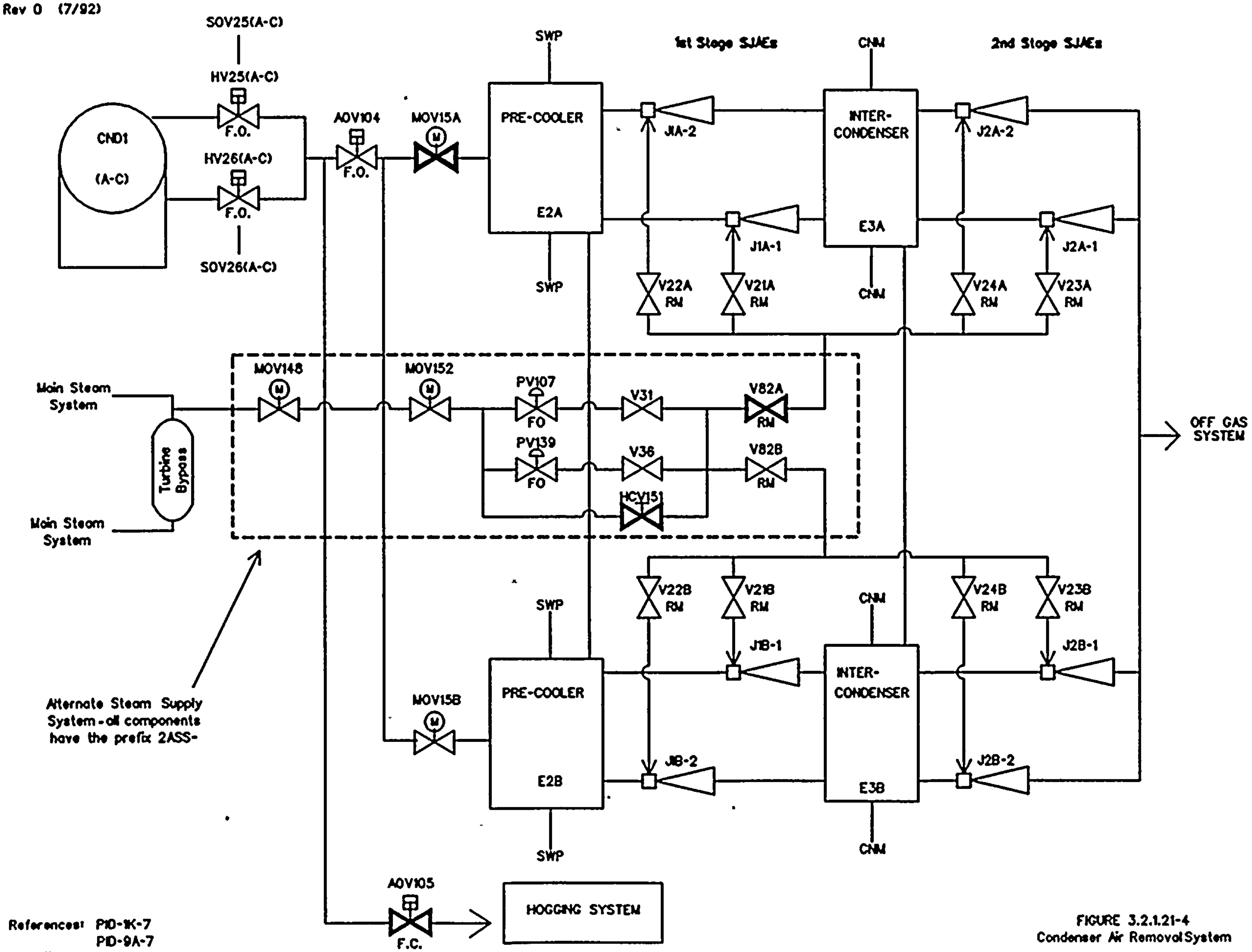
Rev 0 (7/92)



References: PID-10A-5
PID-10B-5
PID-10C-9
PID-10D-10

Figure 3.2.1.21-3
Condenser (CN)
Circulating Water

Rev 0 (7/92)



References: PID-1K-7
PID-9A-7
PID-25A-9

FIGURE 3.2.1.21-4
Condenser Air Removal System

I: \NHP2IPE\DGN\2S21.DGN Jul. 07. 1992 09:02:04

System 22

Nitrogen Systems

3.2.1.22 Nitrogen Systems

3.2.1.22.1 System Function

The primary purposes of the nitrogen system is to provide a source of instrument nitrogen to pneumatic valves inside containment, inert primary containment, and maintain containment pressure. A simplified diagram of the nitrogen system is provided in Figure 3.2.1.22-1.

3.2.1.22.2 Success Criteria

The nitrogen system is considered successful if an uninterrupted supply of nitrogen is delivered to plant loads. The nitrogen systems are modeled in the support system event tree as top events N1 and N2. Success diagrams for these top events are provided in Figure 3.2.1.22-2.

3.2.1.22.3 Support Systems

Instrument and Service Air (IAS) lines and valves serve the nitrogen system during normal operation. Normal AC power (2NHS-MCC016 via 2NJS-US9) is required for operation of the trim heaters. Tables 3.2.1.22-1 and 2 provide additional detail on component dependencies.

3.2.1.22.4 System Operation

The nitrogen system is comprised of two systems: the instrument nitrogen system (N2) and the high-pressure instrument nitrogen system (N1).

The instrument nitrogen system consists of two cross-connected liquid nitrogen storage tanks and two branches of supply nitrogen. One branch consists of electric vaporizers and associated valves that supply purge nitrogen to the primary containment. The second branch consists of redundant ambient vaporizers, redundant trim heaters, associated valves, and a gaseous nitrogen storage tank that provides nitrogen to containment loads such as MSIVs, containment vent valves, SRVs, and drywell vacuum breakers. Each MSIV and SRV has an associated accumulator tank.

The high-pressure instrument nitrogen system consists of six gaseous nitrogen storage tanks that supply the ADS accumulator tanks and the instrument nitrogen system. Three of the six storage tanks are in stand by and the system provides low flow backup to the instrument nitrogen system upstream of the gaseous nitrogen storage tank.

3.2.1.22.5 Instrumentation and Controls

The system has pressure and flow instrumentation to monitor nitrogen gas supply for indication and alarm. Containment isolation supply valves are controlled from P851 in the control room. Keylock override switches are provided to override a containment isolation signal for nitrogen supply inside primary containment.

3.2.1.22.6 Technical Specifications

The nitrogen system is considered non-safety related. However, the accumulators associated with safety-related equipment are considered safety related and are treated under the technical specifications for each system.

3.2.1.22.7 Surveillance, Testing, and Maintenance

Preventative Maintenance is done on the relief valves every 3 years. A complete nitrogen system inservice test is done every 40 months. Nitrogen system containment isolation valves are checked for leak rate every 18 months. A valve operability test is performed every quarter and valve position indication is checked every 18 months.

3.2.1.22.8 References

N2-OP-61A Rev 3
FSAR Section 9.3.1: Compressed Air Systems
N2-POT-34
N2-POT-19-2

PIDs are referenced on Figure 3.2.1.22-1

3.2.1.22.9 Initiating Event Potential

Loss of instrument nitrogen could lead to MSIV closure, as the accumulators leak-off.

3.2.1.22.10 Equipment Location

2GSN-V52, TK2 and V202 (in top event N2) are located in secondary containment. All other equipment modeled (in N1 and N2) is located in the Nitrogen Area (refer to boundaries indicated on Figure 3.2.1.22-1).

3.2.1.22.11 Operating Experience

There were no particular outstanding operational events relevant to this study. Plant specific component operational data is detailed in section 3.3.2.

3.2.1.22.12 Modeling Assumptions

1. The two ADS branches from the gaseous storage tanks (2GSN*TK4 and TK5) are included in the ADS model since they serve only that system.

2. Containment isolation valves are modeled with the component being provided with instrument nitrogen.

3.2.1.22.13 Logic Model and Results

The fault tree(s) is included in Tier 2 documentation, and is available upon request. Quantitative results are summarized in Section 3.3.5 (Tier 1).

HIGH PRESSURE NITROGEN SYSTEM (N1) Component Block Descriptions								
Block	Mark No.	(Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
TANK A,B,C TRAIN (Active)								
	2GSN-TK3A		Nitrogen Storage Tank	RUPTURES	N/A	N/A	N/A	N/A
	2GSN-PSV30A		TK3A Relief Valve	STICKS OPEN	CLOSED	N/A	N/A	N/A
	2GSN-V101C		TK3A Outlet Valve	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2GSN-TK3B		Nitrogen Storage Tank	RUPTURES	N/A	N/A	N/A	N/A
	2GSN-PSV21A		TK3B Relief Valve	STICKS OPEN	CLOSED	N/A	N/A	N/A
	2GSN-V101B		TK3B Outlet Valve	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2GSN-TK3C		Nitrogen Storage Tank	RUPTURES	N/A	N/A	N/A	N/A
	2GSN-PSV20A		TK3C Relief Valve	STICKS OPEN	CLOSED	N/A	N/A	N/A
	2GSN-V101A		TK3C Outlet Valve	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2GSN-V103		Tank Train Outlet Valve	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
NORMAL VALVE TRAIN								
	2GSN-V119		Manual Globe Valve	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2GSN-SV26A		Relief Valve	STICKS OPEN	CLOSED	N/A	N/A	N/A
	2GSN-PCV24A		Pressure Control Valve	TRANSFERS CLOSED	OPEN	CLOSED	N/A	N/A
	2GSN-V117		Manual Globe Valve	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
TANK D,E,F TRAIN (Reserve)								
	2GSN-TK3D		Nitrogen Strorage Tank	RUPTURES	N/A	N/A	N/A	N/A
	2GSN-PSV30B		TK3D Relief Valve	STICKS OPEN	CLOSED	N/A	N/A	N/A
	2GSN-V102C		TK3D Outlet Valve	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2GSN-TK3E		Nitrogen Storage Tank	RUPTURES	N/A	N/A	N/A	N/A
	2GSN-PSV21B		TK3E Relief Valve	STICKS OPEN	CLOSED	N/A	N/A	N/A
	2GSN-V102B		TK3E Outlet Valve	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2GSN-TK3F		Nitrogen Storage Tank	RUPTURES	N/A	N/A	N/A	N/A
	2GSN-PSV20B		TK3F Relief Valve	STICKS OPEN	CLOSED	N/A	N/A	N/A
	2GSN-V102A		TK3F Outlet Valve	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2GSN-V104		Tank Train Outlet Valve	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
STANDBY VALVE TRAIN								
	2GSN-V116		Manual Globe Valve	TRANSFERS CLOSED	OPEN *	N/A	N/A	N/A
	2GSN-SV26B		Relief Valve	STICKS OPEN	CLOSED	N/A	N/A	N/A
	2GSN-PCV24B		Pressure Control Valve	TRANSFER CLOSED	OPEN	CLOSED	N/A	N/A
	2GSN-V118		Manual Globe Valve	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A

OPEN * : Normally closed valve, operator must open to align the standby train

Table 3.2.1.22-2 REV. 0 (7/92)

INSTRUMENT NITROGEN SYSTEM (N2)
Component Block Descriptions

Block	Mark No. (Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
TANK 1A LEG	2GSN-TK1A	Liquid Nitrogen Storage Tank	RUPTURE/LEAKAGE	N/A	N/A	N/A	N/A
	2GSN-V13A	TK1A Outlet Valve	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
TANK 1B LEG	2GSN-TK1B	Liquid Nitrogen Storage Tank	RUPTURE/LEAKAGE	N/A	N/A	N/A	N/A
	2GSN-V13B	TK1B Outlet Valve	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
VAPORIZER 1A TRAIN	2GSN-V20A	EV1A Inlet Valve	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2GSN-EV1A	Ambient Vaporizer	LOSS OF FUNCTION	N/A	N/A	N/A	N/A
	2GSN-V21A	EV1A Outlet Valve	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
VAPORIZER 1B TRAIN	2GSN-V20B	EV1B Inlet Valve	TRANSFERS CLOSED	OPEN *	N/A	N/A	N/A
	2GSN-EV1B	Ambient Vaporizer	LOSS OF FUNCTION	N/A	N/A	N/A	N/A
	2GSN-V21B	EV1B Outlet Valve	TRANSFERS CLOSED	OPEN *	N/A	N/A	N/A
TRIM HEATER 1A TRAIN	2GSN-RV102	Relief Valve	STICKS OPEN	CLOSED	N/A	N/A	N/A
	2GSN-V29A	E1A Inlet Valve	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2GSN-E1A	Trim Heater	LOSS OF FUNCTION	OFF	ON	2NHS-MCC016	OFF
	2GSN-RV29A	Relief Valve	STICKS OPEN	CLOSED	N/A	N/A	N/A
	2GSN-V25A	E1A Outlet Valve	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2GSN-RV105	Relief Valve	STICKS OPEN	CLOSED	N/A	N/A	N/A
TRIM HEATER 1B TRAIN	2GSN-V29B	E1B Inlet Valve	TRANSFERS CLOSED	OPEN *	N/A	N/A	N/A
	2GSN-E1B	Trim Heater	LOSS OF FUNCTION	OFF	ON	2NHS-MCC016	OFF
	2GSN-RV29B	Relief Valve	STICKS OPEN	CLOSED	N/A	N/A	N/A
	2GSN-V25B	E1B Outlet Valve	TRANSFERS CLOSED	OPEN *	N/A	N/A	N/A
SUPPLY VALVE LEG	2GSN-TCV108	Temperature Controlled SOV	TRANSFERS CLOSED	OPEN	OPEN	** 2GSN-IPNL166	CLOSED
	2GSN-V27	Manual Valve to PCV109	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2GSN-PCV109	Pressure Control Valve	TRANSFERS CLOSED	OPEN	CLOSED	N/A	N/A
	2GSN-V29	Manual Valve from PCV109	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2GSN-V33	Manual Bypass Around PCV109	FAILS TO OPEN	CLOSED	N/A	N/A	N/A
	2GSN-V35	Check Valve	FAILS TO OPEN	CLOSED	N/A	N/A	N/A
HIGH PRESSURE NITROGEN SUPPLY	2GSN-V120	Manual Valve to PCV144	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2GSN-PCV144	Pressure Control Valve	TRANSFERS CLOSED	OPEN	CLOSED	N/A	N/A
	2GSN-V121	Manual Valve from PCV144	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2GSN-RV147	Relief Valve	STICKS OPEN	CLOSED	N/A	N/A	N/A

OPEN * : Normally closed valve, operator must open to align the standby train
**: Power source for 2GSN-TC108, valve controller

INSTRUMENT NITROGEN SYSTEM (N2)
Component Block Descriptions

Block	Mark No. (Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
	2GSN-V41	Check Valve	FAILS TO OPEN	CLOSED	N/A	N/A	N/A
GASEOUS NITROGEN RECEIVER TANK							
	2GSN-V42	Manual Valve	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2GSN-V50	TK2 Inlet Valve	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A
	2GSN-V52	Check Valve	FAILS TO OPEN	OPEN	N/A	N/A	N/A
	2GSN-TK2	Nitrogen Reciever Tank	RUPTURE/LEAKAGE	N/A	N/A	N/A	N/A
	2GSN-SV135	TK2 Relief Valve	STICKS OPEN	CLOSED	N/A	N/A	N/A
	2GSN-V202	TK2 Outlet Valve	TRANSFERS CLOSED	OPEN	N/A	N/A	N/A

OPEN * : Normally closed valve, operator must open to align the standby train
**: Power source for 2GSN-TC108, valve controller

Rev 0 (7/92)

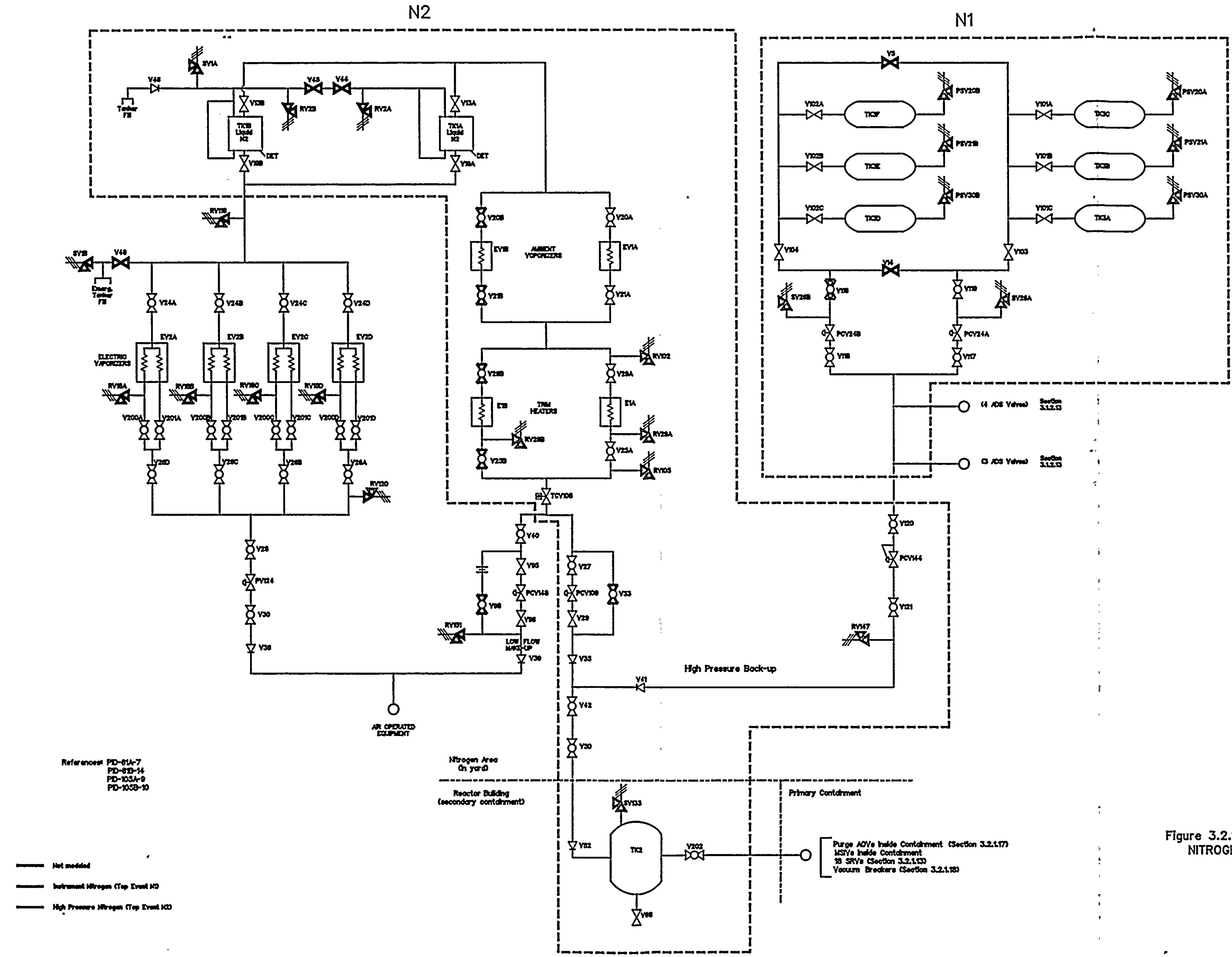


Figure 3.2.1.22-1
NITROGEN

Rev 0 (7/92)

SI APERTURE CARD

Also Available On Aperture Card

High Pressure Instrument Nitrogen Success Diagram (N1)

Instrument Nitrogen Success Diagram (N2)

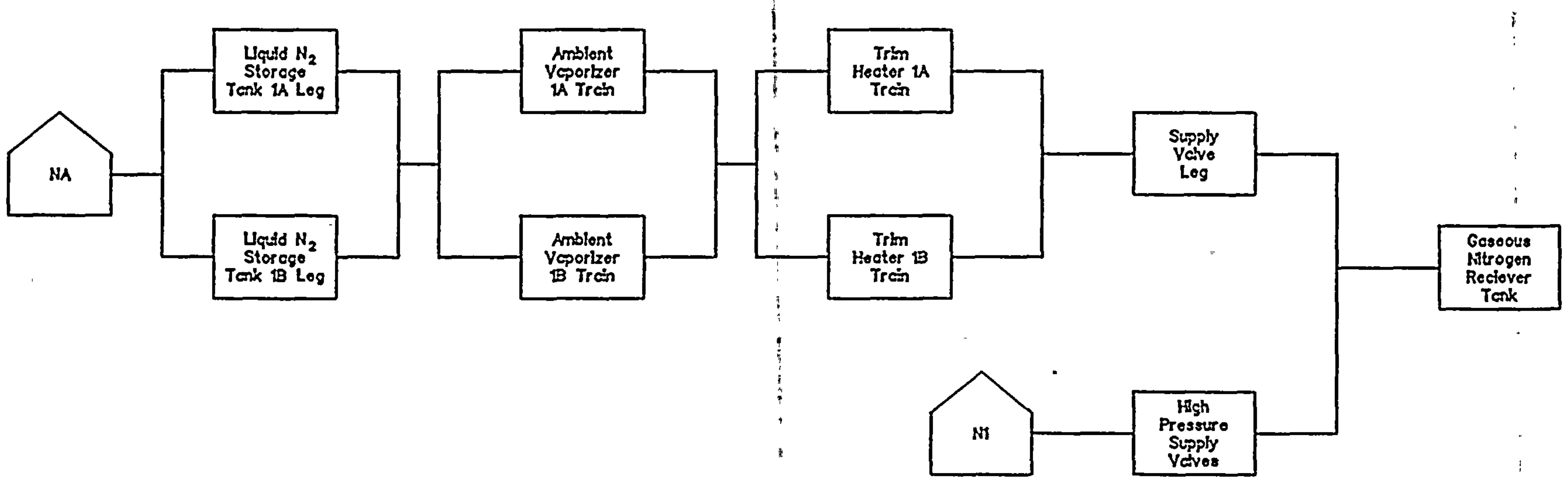


Figure 3.2.1.22-2
Instrment Nitrogen Success Diagrams

I:\NMP2IPE\DGN\2S22.DGN Jul. 07, 1992 11:13:16

9208050191-33

System 23

Instrument Air

3.2.1.23 Instrument, Service, and Breathing Air System

3.2.1.23.1 System Function

The Instrument, Service, and Breathing Air System supplies compressed air for plant instrumentation, control, processes, and breathing air stations. A simplified diagram of the Instrument Air System is provided in Figure 3.2.1.23-1.

3.2.1.23.2 Success Criteria

One of three compressors operating to maintain system pressure at the discharge of the instrument air receiver tank is modeled in the support system event tree as top event AS. A success diagram for top event AS is provided in Figure 3.2.1.23-2.

3.2.1.23.3 Support Systems

The plant Normal AC System supplies AC power to the air compressors. The plant Normal DC system supplies DC power to the air compressors for control and indication. Table 3.2.1.23-1 lists support systems for each component modeled in the instrument air system.

A subsystem of the Reactor Building Closed Loop Cooling system (RBCLC) cools the instrument air compressors. A simplified diagram is provided in Figure 3.2.1.23-3. This is a closed loop which uses RBCLC as a heat sink. Major equipment in the subsystem includes two 100% capacity pumps (125 GPM each), two 100% capacity heat exchangers, and a surge tank. Support systems for equipment in this loop are shown in Table 3.2.1.23-2.

3.2.1.23.4 System Operation

The Instrument, Service, and Breathing Air System supplies compressed air to a variety of plant systems and components. On loss of compressed air, a number of components fail resulting in several critical plant events. Key failures and events are:

1. Feedwater pump minimum flow bypass valves fail open, resulting in partial loss of feedwater flow to the reactor vessel.

2. MSIV pilot valve accumulators will eventually leak-off causing the MSIVs to close.

3. Scram inlet and outlet valves will open causing a scram. In addition, CRD flow control valves will close to approximately 2% open and drain and vent valves for the discharge volume will close.

4. Containment atmosphere control valves, containment ventilation isolation valves, and ventilation supply isolation dampers fail closed.

5. Steam supply valves to the Steam Jet Air Ejector (SJAE) close resulting in an eventual loss of condenser vacuum.

Three separate air compressor trains supply compressed air to the instrument and service air systems. Each train is identical. The compressors receive air from intake filters located on the turbine building roof. The compressed air is routed through an intercooler and after-cooler, then discharged into an air receiver tank. The compressors and tanks are equipped with pressure indication. Normally, one compressor is in lead mode, another in lag mode, and the third is in backup mode. The lag compressor starts when demand is in excess of the lead compressor's capability. Each compressor has a three step regulator which allows the compressor to operate at full, one-half, or zero load while maintaining rated speed.

The three compressor trains join to form a common header. This header divides into two branch headers. One branch serves the instrument air system and the other serves the service air system.

The Instrument Air System consists of a receiver tank, two prefilters, two dryers, two after-filters, piping, valves, and controls. A common header is formed downstream of the compressed air receiver tanks. From the common air header, compressed air flows through the prefilters, the air dryer, the after-filter, enters the reactor building, passes through a check valve, and into the instrument air receiver tank. A line taps off the receiver tank and branches to serve the various system loads.

Loads inside containment are supplied from the air receiver tank when the containment is not inerted. These lines are supplied by the nitrogen system when the containment is inerted.

The Service Air System branches from the instrument air system upstream of the prefilters. The service air line passes through a header stop valve and splits to supply various branches and components.

A separate air compressor with two air receiving tanks is dedicated to the automatic depressurization system (ADS). This compressor is not used during normal operation in preference to the nitrogen system. The ADS air compressor is only used during shutdown conditions to maintain ADS relief pressure.

3.2.1.23.5 Instrumentation and Controls

Pressure instrumentation, located in the header upstream of the instrument and service air system branch point, is used to control the compressors. The lag compressor is controlled by 2IAS-PS223 and starts at 100 psig. The backup compressor is controlled by 2IAS-PS104, and starts at 90 psig.

System air pressure is monitored at various locations to identify and locate ruptures for manual isolation of portions of the system, as required. Each compressor has an automatic three step regulator for free air unloading. This allows an operating compressor to operate at full, one-half, or zero load while maintaining rated speed. A pressure switch controls the regulator. Each compressor has a local pressure switch that will stop the compressor on high pressure.

3.2.1.23.6 Technical Specifications

If one or more of the primary containment isolation valves becomes INOPERABLE, it must be returned to operable within 4 hours, or be isolated.

3.2.1.23.7 Surveillance, Testing, and Maintenance

The Instrument air system is tested every 40 months. The following equipment has maintenance or surveillance testing with the specified frequencies:

Equipment	Frequency
Relief Valves	3 years
Pre-Filters	1 year
Air Dryer	Semi-Annually
Compressors	Semi-Annually
Isolation Valves	18 months

3.2.1.23.8 References

N2-OP-19 Rev.. 2
PID's as referenced on Figures 3.2.1.23-1, 3.2.1.23-3
FSAR Section 9.3.1
Technical Specifications Section 3.6.3
Calculation 2IPE-1

3.2.1.23.9 Initiating Event Potential

Partial or complete loss of Instrument Air could result in several initiating events. Most fit in the overall category of General Transient, these include:

- Loss of air to CRD SCRAM inlet and outlet valves causing a plant SCRAM.

- Loss of air to the feedwater minimum flow bypass valves or condensate flow control valves resulting in a loss of feedwater.

- Loss of air to the steam admission valves of the moisture-separator reheater could cause a turbine trip. Loss of air to the steam supply valves of the SJAE causes a loss of condenser vacuum.

- Loss of air to the MSIVs will cause them to close.

3.2.1.23.10 Equipment Location

The common instrument air compressors and associated equipment are located on the North end of the Turbine Building on Elevation 250'. The Instrument Air receiver and the main air header are also located on TB 250'. From this location the Instrument and Service Air branches route to lines and smaller receiver tanks that serve systems and components throughout the plant.

The compressor and tanks for the ADS dedicated Air System are located on the 289' elevation of the Reactor Building.

3.2.1.23.11 Operating Experience

There were no particular outstanding operational events relevant to this study. Plant specific component operational data is detailed in section 3.3.2.

3.2.1.23.12 Modeling Assumptions

1. Instrument air failure is treated as an instantaneous loss of air pressure to equipment requiring air for operation. In the event of compressor failure, the air pressure will gradually leak off. The model does not distinguish between failure modes that lead to loss of air pressure (i.e., ruptures) and compressor failures.

2. Loss of makeup water to the instrument air compressor cooling loop expansion tank (2CCP-TK2) is not modeled. This is considered a low probability event since there is automatic level control and a control room alarm on low tank level.

3. RBCLC Expansion tank drain valve 2CCP-V847 is not modeled. This is considered a low probability event since it is normally closed, there is automatic level make-up, and a control room trouble alarm for expansion tank level. Accordingly, this failure is considered a low probability occurrence.

4. Drain lines are not modeled. These lines are generally double isolated and have a negligible contribution to overall system failure.

5. Loss of the Reactor Building Closed Loop Cooling water loop to the instrument air compressor(s) will fail the compressor(s). The compressors will quickly overheat and trip.

6. Loss of the main RBCLC loop will fail the compressor subloop, resulting in the loss of all 3 air compressors.

7. The instrument air compressor intercoolers, after-coolers, and related valving are "black boxed" for modeling purposes.

8. The redundant train of prefilters, air dryers, and after filters are modeled only to stay in service. Because all of the valving is manual, the dominant failure mechanism is failure of the operator to properly align the system. However, the manual valves are modeled for transfers closed failures.

3.2.1.23.13 Logic Model and Results

The fault tree(s) is included in Tier 2 documentation, and is available upon request. Quantitative results are summarized in Section 3.3.5 (Tier 1).

Rev 0 (7/92)

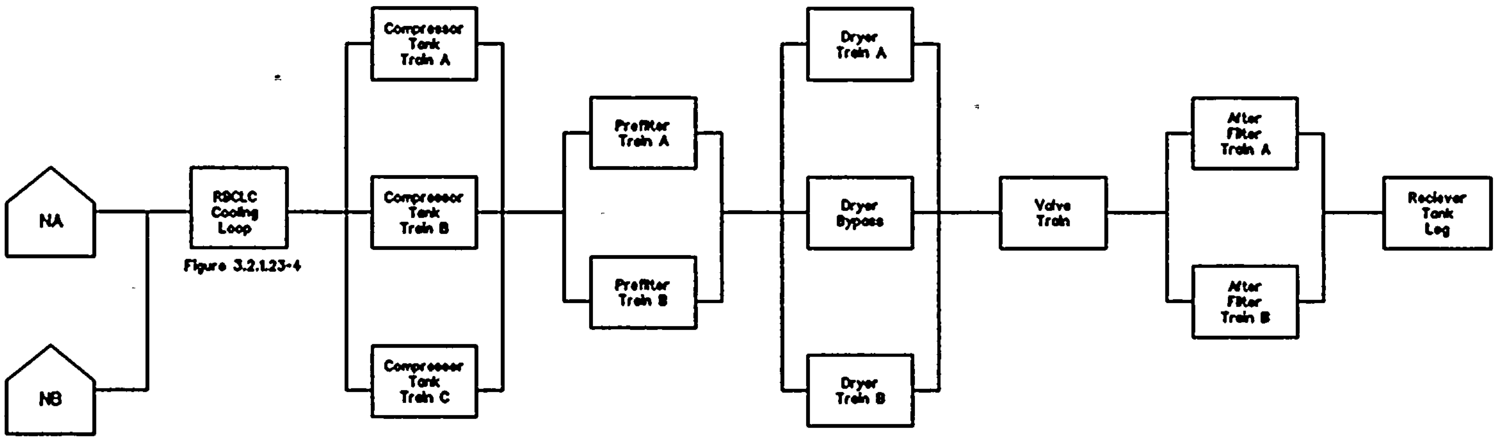
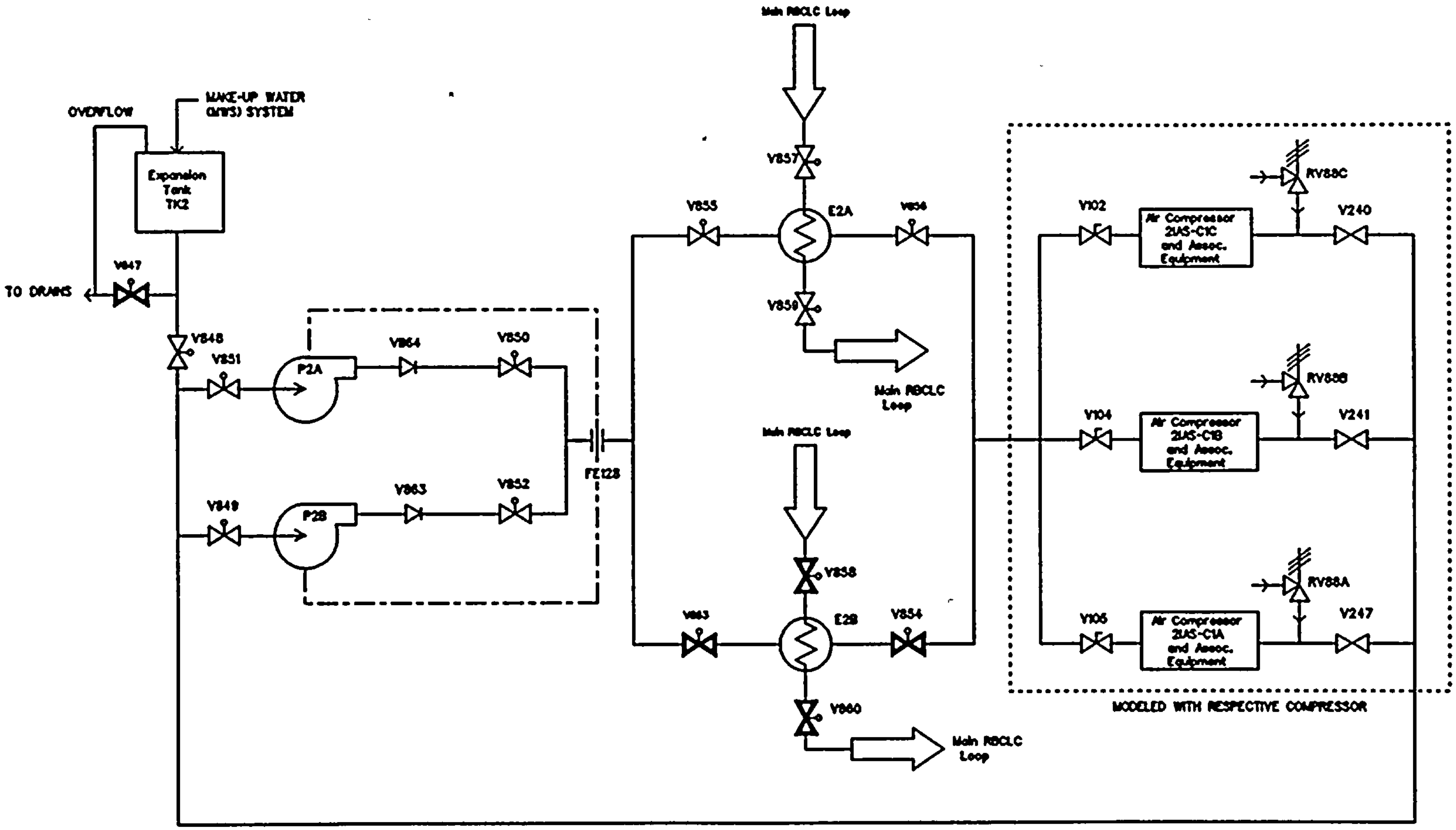


Figure 3.2.1.23-2
Instrument Air Success Diagram (AS)

I:\NWP2IPE\DGN\2S23.DGN Jul. 07. 1992 09:12:25

Rev 0 (7/92)



REFERENCE DRAWINGS: PID-13B-10
PID-13G-7

3.2.1.23-3
IAS COMPRESSOR, RBCLC COOLING LOOP

I: \NMP2IPE\DGN\2S23.DGN Jul. 07. 1992 09:14:45

Rev 0 (7/92)

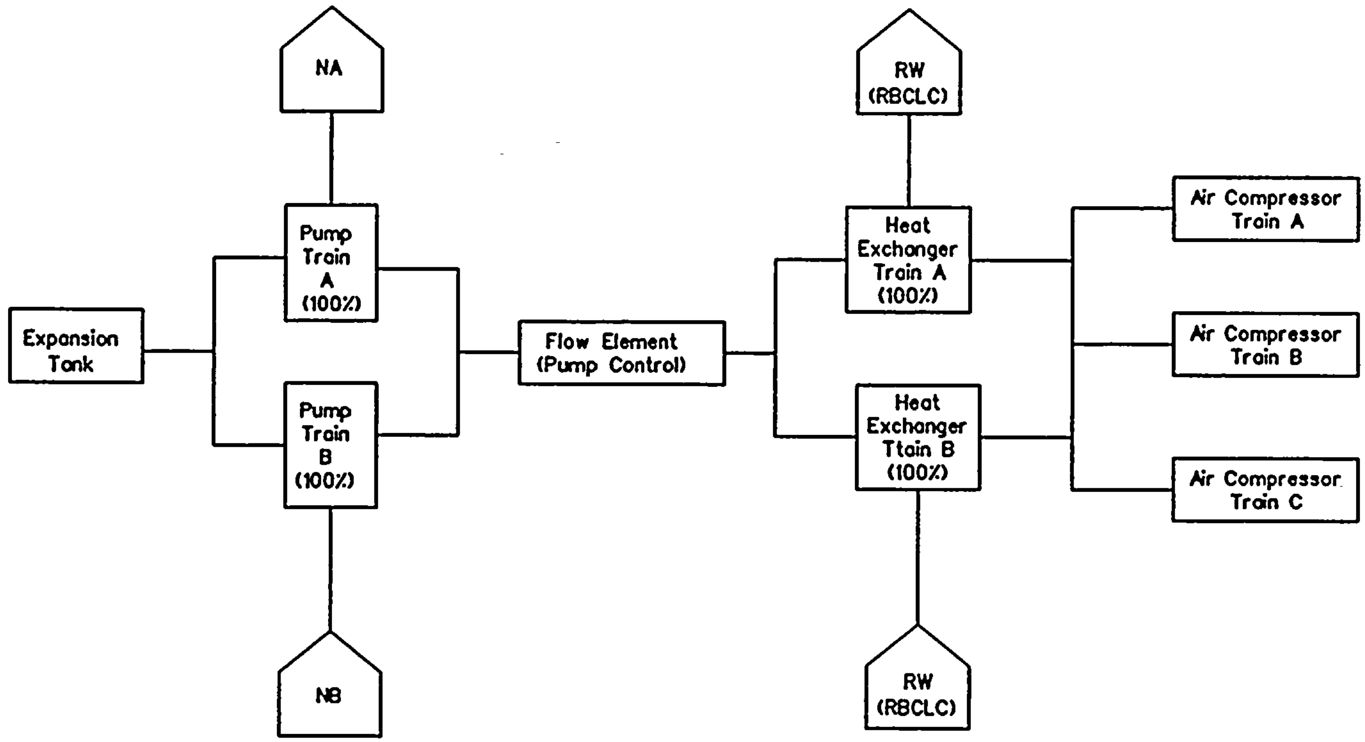


Figure 3.2.1.23-4
IAS Compressor Cooling
Success Diagram

INSTRUMENT AIR
Component Block Descriptions

Block	Mark No. (Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
Compressor 1A (Modeled as Lead)							
	2IAS-C1A	AIR COMPRESSOR	FAILS TO RUN	RUNNING	RUNNING	2NJS-US5	STOP
	2IAS-V101	CHECK VALVE	FAILS TO OPEN	OPEN	N/A	N/A	N/A
	2IAS-V502	MANUAL TK1A INLET VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2IAS-TK1A	RECIEVER TANK	RUPTURES	N/A	N/A	N/A	N/A
	2IAS-V102	MANUAL TK1A OUTLET VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2IAS-FLT1A	AIR INTAKE FILTER	CLOGGED / PLUGGED	N/A	N/A	N/A	N/A
Compressor 1B (Modeled as Lag/Backup)							
	2IAS-C1B	AIR COMPRESSOR	FAILS TO START	STOPPED	RUNNING	2NJS-US6	STOP
	2IAS-C1B	AIR COMPRESSOR	FAILS TO RUN	RUNNING	RUNNING	2NJS-US6	STOP
	2IAS-V201	CHECK VALVE	FAILS TO OPEN	CLOSED	N/A	N/A	N/A
	2IAS-V602	MANUAL TK1B INLET VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2IAS-TK1B	RECIEVER TANK	RUPTURES	N/A	N/A	N/A	N/A
	2IAS-V202	MANUAL TK1B OUTLET VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2IAS-FLT1B	AIR INTAKE FILTER	CLOGGED / PLUGGED	N/A	N/A	N/A	N/A
Compressor 1C (Modeled as Lag/Backup)							
	2IAS-C1C	AIR COMPRESSOR	FAILS TO START	STOPPED	RUNNING	2NJS-US10	STOP
	2IAS-C1C	AIR COMPRESSOR	FAILS TO RUN	RUNNING	RUNNING	2NJS-US10	STOP
	2IAS-V301	CHECK VALVE	FAILS TO OPEN	CLOSED	N/A	N/A	N/A
	2IAS-V702	MANUAL TK1C INLET VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2IAS-TK1C	RECIEVER TANK	RUPTURES	N/A	N/A	N/A	N/A
	2IAS-V302	MANUAL TK1C OUTLET VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2IAS-FLT1C	AIR INTAKE FILTER	CLOGGED / PLUGGED	N/A	N/A	N/A	N/A
Prefilter Train A							
	2IAS-V216	MANUAL FLT2A INLET VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2IAS-FLT2A	PREFILTER	PLUGGED	N/A	N/A	N/A	N/A
	2IAS-V217	MANUAL FLT2A OUTLET VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
Prefilter Train B							
	2IAS-V218	MANUAL FLT2B INLET VALVE	TRANSFER CLOSED	*OPEN	N/A	N/A	N/A
	2IAS-FLT2B	PREFILTER	PLUGGED	N/A	N/A	N/A	N/A
	2IAS-V219	MANUAL FLT2B OUTLET VALVE	TRANSFER CLOSED	*OPEN	N/A	N/A	N/A
Dryer Train A							
	2IAS-V280	MANUAL DRY1A INLET VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2IAS-DRY1A	AIR DRYER	LOSS OF FUNCTION	N/A	N/A	N/A	N/A
	2IAS-V281	MANUAL DRY1A OUTLET VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
Dryer Bypass							

INSTRUMENT AIR
Component Block Descriptions

Block	Mark No. (Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
	2IAS-V298	MANUAL BYPASS FOR DRYER TRAINS	FAILS TO OPEN	*OPEN	N/A	N/A	N/A
Dryer Train B							
	2IAS-V282	MANUAL DRY1B INLET VALVE	TRANSFER CLOSED	*OPEN	N/A	N/A	N/A
	2IAS-DRY1B	AIR DRYER	LOSS OF FUNCTION	N/A	N/A	N/A	N/A
	2IAS-V283	MANUAL DRY1B OUTLET VALVE	TRANSFER CLOSED	*OPEN	N/A	N/A	N/A
Air Supply Valves							
	2IAS-V503	MANUAL VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2IAS-V504	MANUAL VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2IAS-V505	MANUAL VALVE (Bypass Valve)	TRANSFER CLOSED	*OPEN	N/A	N/A	N/A
Afterfilter Train A							
	2IAS-V235	MANUAL FLT3A INLET VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2IAS-FLT3A	AFTERFILTER	PLUGGED	N/A	N/A	N/A	N/A
	2IAS-V236	MANUAL FLT3A OUTLET VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
Afterfilter Train B							
	2IAS-V237	MANUAL FLT3B INLET VALVE	TRANSFER CLOSED	*OPEN	N/A	N/A	N/A
	2IAS-FLT3B	AFTERFILTER	PLUGGED	N/A	N/A	N/A	N/A
	2IAS-V238	MANUAL FLT3B OUTLET VALVE	TRANSFER CLOSED	*OPEN	N/A	N/A	N/A
Receiver Tank							
	2IAS-TK2	RECEIVER TANK	RUPTURES	N/A	N/A	N/A	N/A
	2IAS-V711	MANUAL TK3 INLET VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2IAS-V641	CHECK VALVE	FAILS TO OPEN	OPEN	N/A	N/A	N/A
	2IAS-TK3	RX BLDG RECIRC TANK	RUPTURES	N/A	N/A	N/A	N/A

INSTRUMENT AIR -- COMPRESSOR COOLING Component Block Descriptions							
Block	Mark No. (Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
EXPANSION TANK							
	2CCP-TK2	EXPANSION TANK	RUPTURES	N/A	N/A	N/A	N/A
	2CCP-V848	EXPANSION DRAIN BALL VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
PUMP TRAIN A (RUNNING)							
	2CCP-V851	P2A INLET BALL VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CCP-P2A	PUMP 2A	FAILS TO RUN	RUNNING	RUNNING	2NHS-MCC011	STOP
	2CCS-V864	P2A OUTLET CHECK VALVE	FAILS TO OPEN	N/A	N/A	N/A	N/A
	2CCP-V850	P2A OUTLET BALL VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
PUMP TRAIN B (STANDBY)							
	2CCP-V849	P2B INLET BALL VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CCP-P2B	PUMP 2B	FAILS TO START	STOPPED	RUNNING	2NHS-MCC012	STOP
	2CCP-P2B	PUMP 2B	FAILS TO RUN	RUNNING	RUNNING	2NHS-MCC012	STOP
	2CCP-V863	P2B OUTLET CHECK VALVE	FAILS TO OPEN	CLOSED	N/A	N/A	N/A
	2CCP-V863	P2B OUTLET CHECK VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CCP-V852	P2B OUTLET BALL VALVE	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
FLOW SWITCH (PUMP CONTROL)							
	2CCP-FISX128	PUMP DISCHARGE FLOW SWITCH	FAIL TO IND LOW FLOW	N/A	N/A	2NHS-MCC011	FAILS LOW
HEAT EXCHANGER TRAIN A							
	2CCP-V855	HEAT EXCHANGER INLET BALL VLV	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CCP-V857	HEAT EXCHANGER INLET BALL VLV	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CCP-V856	HEAT EXCHANGER OUTLET BALL VLV	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CCP-V859	HEAT EXCHANGER OUTLET BALL VLV	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CCP-E2A	HEAT EXCHANGER 2A	RUPTURE	N/A	N/A	N/A	N/A
HEAT EXCHANGER TRAIN B							
	2CCP-V853	HEAT EXCHANGER INLET BALL VLV	FAIL TO OPEN	CLOSED	N/A	N/A	N/A
	2CCP-V853	HEAT EXCHANGER INLET BALL VLV	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CCP-V858	HEAT EXCHANGER INLET BALL VLV	FAILS TO OPEN	CLOSED	N/A	N/A	N/A
	2CCP-V858	HEAT EXCHANGER INLET BALL VLV	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CCP-V854	HEAT EXCHANGER OUTLET BALL VLV	FAIL TO OPEN	CLOSED	N/A	N/A	N/A
	2CCP-V854	HEAT EXCHANGER OUTLET BALL VLV	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CCP-V860	HEAT EXCHANGER OUTLET BALL VLV	FAIL TO OPEN	CLOSED	N/A	N/A	N/A
	2CCP-V860	HEAT EXCHANGER OUTLET BALL VLV	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CCP-E2B	HEAT EXCHANGER 2B	RUPTURE	N/A	N/A	N/A	N/A
AIR COMPRESSOR TRAIN A							
	2CCP-V105	COMPRESSOR INLET PLUG COCK VLV	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CCP-C1A	COMPR 1A COOLING (BLACK BOX)	RUPTURE	N/A	N/A	N/A	N/A

Table 3.2.1.23-2 REV. 0 (7/92)

INSTRUMENT AIR -- COMPRESSOR COOLING
Component Block Descriptions

Block	Mark No. (Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
	2CCP-V247	COMPRESSOR OUTLET GATE VLV	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
AIR COMPRESSOR TRAIN B							
	2CCP-V104	COMPRESSOR INLET PLUG COCK VLV	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CCP-C1B	COMPR 1B COOLING (BLACK BOX)	RUPTURE	N/A	N/A	N/A	N/A
	2CCP-V241	COMPRESSOR OUTLET GATE VLV	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
AIR COMPRESSOR TRAIN C							
	2CCP-V102	COMPRESSOR INLET PLUG COCK VLV	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CCP-C1C	COMPR 1C COOLING (BLACK BOX)	RUPTURE	N/A	N/A	N/A	N/A
	2CCP-V240	COMPRESSOR OUTLET GATE VLV	TRANSFER CLOSED	OPEN	N/A	N/A	N/A

System 24

Late Containment Failure

3.2.1.24 Injection During Late Containment Failure

The Level 1 event trees model the possibility that injection will continue to be successful during containment overpressure conditions and after containment failure. The following top events model continued injection when the condenser, RHR, and containment venting fail to control primary containment pressure:

CI - Continued injection at high containment pressure
CF - Continued injection after containment failure

Top event CI models continued injection to the RPV from the suppression pool during accident sequences involving the loss of long term containment heat removal.

Non-LOCA Sequences
For loss of decay heat removal sequences, at approximately 18 hours after the initiating event, containment pressure will reach the "Maximum Primary Containment Water Level Limit" (MPCWLL) in N2-EOP-PCC Section SPL which instructs the operators to terminate injection into the primary containment from sources external to the primary containment. At approximately 24 hours, containment pressure reaches 100 psia where the SRVs are assumed to shut for non-LOCA sequences. The model assumes that the operators follow procedures and use only injection sources from the suppression pool (i.e., HPCS from the suppression pool). This assumption forces the sequence model to core damage before containment failure if HPCS is unavailable. Except for the MPCWLL EOP limit, continued high pressure injection from external sources (CRD, feedwater) would tend to be successful until the containment failed. CI is set to guaranteed success if HPCS (HS) was previously successful. If feedwater was successful and HPCS had not been demanded previously in the sequence, the availability of HPCS and the operator action to start and align HPCS to the suppression pool are evaluated. If HS has failed previously, CI is set to guaranteed failure.

If CI is found to fail or be intentionally terminated, then core damage occurs with the containment intact, but containment pressure is very high (i.e., above 100 psig).

LOCA Sequences
Top event CI is not included in the medium and large LOCA event tree models because it is assumed to be successful. This is because low pressure injection has already been successful, the reactor coolant system will be depressurized (no concern about safety relief valves closing), and injection will already be available from the suppression pool, thus, continued injection is assumed to continue.

Top event CF models whether injection remains successful after containment failure in the case where CI is success. This probability depends on the containment failure size (small or large) and failure location (upper drywell or lower drywell or wetwell), availability of injection systems given the containment failure size and location, and operator actions.

Judgements made regarding the impact of containment failure modes on high pressure injection systems capable of taking suction from the condensate storage tanks are summarized in the table below. The three systems capable of providing high pressure injection include control rod drive (CRD), feedwater (FW), and HPCS (HS). The containment failure modes:

"High" (e.g. drywell head), "Low" (e.g., wetwell) and size which is shown as "Large" and "Small" are described further in Section 4.

Injection Failure Modes	Containment Failure Modes			
	Size		Location	
	Large	Small	High	Low
Environmental	HS (1)			CRD Fails HS (1)
NPSH	HS (2)			
Structural (Piping)	CRD (3) FW (3) HS (3)			CRD (3) FW (3) HS (3)

(1) Large containment failures in the lower drywell or wetwell could affect the environment in the HPCS room.

(2) HPCS suction must be transferred to the CSTs if pump is not damaged.

(3) Large containment failures in the lower drywell or wetwell could structurally fail injection piping.

The following table summarizes injection system failure designators that represent failure probabilities based on the above judgmental impacts. The CF fault tree includes these probabilities as well as system unavailability due to normal conditions and support systems.

Injection System	Containment Failure Conditions			
	Small		Large	
	High	Low	High	Low
CRD	CRD1	Failed	CRD2	Failed
Feedwater	FW1	FW1	FW2	FW2
HPCS	HS1	HS1	HS2	Failed

HPCS: Since HPCS was required (by procedure) in top event CI to be on the suppression pool, it must now be transferred to the condensate storage tanks (CSTs) for large containment failures that impact NPSH. Small failures are less likely to impact either NPSH or the HPCS environment since the pump is located in a protected separate room. HPCS is assumed unavailable for the large low containment failure mode due to the combination of environmental, NPSH, and structural failure concerns.

CRD: A CRD pump must be operating and its support systems available. Any containment failure in the lower drywell or wetwell is assumed to fail these pumps because they are unprotected and located in the open reactor building.

Feedwater: Feedwater with condensate transfer pumps and the necessary support systems can provide injection from the CSTs. Since this equipment is located outside the Reactor Building, it is given an opportunity for success for all containment failure modes.

The following table summarizes assumptions made in the CF model (post containment failure) regarding RPV pressure, the availability of the suppression pool as a source of water, and the availability of alternate low pressure injection systems (i.e., service water and fire water systems):

	Sequence Type	
	Transient & SLOCA	MLOCA & LLOCA
RPV Pressure	Not Depressurized	Depressurized
Suppression Pool	Not a Source	Not a Source
Alternate LPI	Not Considered	Not Considered

The model conservatively requires high pressure injection from the condensate storage tanks (i.e., source external to the primary containment). As shown above for transients and small LOCA events, the RPV is not assumed to be depressurized (no credit is given for the operator using SRVs to blowdown RPV post containment failure). Thus, alternate low pressure injection systems are not considered. Additionally, no credit is given to the suppression pool as a makeup source due to concerns about the pool conditions after containment failure (e.g., pump cavitation). In the case of medium and large LOCA scenarios, top event CF is set to guaranteed failure in the event tree models. This is because it is assumed that the CSTs were exhausted by HPCS earlier in the sequence. In addition, the availability of the suppression pool and alternate LPI systems are not considered in either model consistent with the transients and small LOCA event tree models.

The above assumptions are conservative because no credit is given to low pressure injection being used in the following situations:

- Given a small containment failure and the safety relief valves are available, low pressure injection (LPCS, LPCI) from the suppression pool is a possibility.

- The service water and fire water crossties to the RHR LPCI injection path (alternate LPI) is possible if the safety relief valves are available.

- For medium and large LOCAs, neglecting the suppression pool and alternate LPI systems is very conservative.

System 25

Reactor Recirculation Seal LOCA

3.2.1.25 Reactor Recirculation Pump (RRP) Seals

This section documents a review of the Reactor Recirculation Pump (RRP) seal design and the potential significance of a RRP seal LOCA event. Based on a number of factors, RRP seal LOCAs are judged to have a minor impact on overall core damage frequency. These factors include the RRP seal design, instrumentation and alarms available for detection of seal problems, and procedures to respond to these problems. This conclusion is discussed below. Figures 3.2.1.25-1 through 5 provide simplified diagrams of a typical reactor recirculation loop and the associated RRP seals.

3.2.1.25.1 Significance of RRP Seal LOCA

As an initiating event, RRP seal LOCA is assumed to be enveloped by the small LOCA initiating event in both frequency and consequence. This is based on the following:

- To contribute significantly, the frequency of an unisolated RRP seal LOCA would have to be on the same order of magnitude as the small LOCA initiating event. The operators have the capability to isolate the RRP and the LOCA with motor operated valves. Therefore, the frequency of an unisolated RRP seal LOCA would be based on failure to isolate the LOCA (operator, equipment, and normal AC power failures). This failure probability is considerably less than the small LOCA initiating event frequency. Moreover, when compared with the other transients in the model, isolated RRP seal LOCA would have minimal consequences.

- There are adequate alarms, instrumentation and procedures to protect the pump, ensure detection of seal problems, and require plant shutdown.

- Maximum seal LOCA is 50 gpm/pump (total of 100 gpm for both pumps). This is a relatively small break size and would be on the lower end of the small LOCA size definition. Based on this break size, the success criteria for plant response would not likely be affected in a negative way.

In summary, the frequency of RRP seal LOCA transient (small leaks requiring shutdown or isolated small LOCA) is judged to be small when compared to transients with similar or more severe consequences. Similarly, the likelihood of RRP seal LOCA (unisolated small LOCA) initiator is judged to be small, particularly when compared with the existing small LOCA initiating event frequency.

RRP seal LOCA subsequent to other transient initiating events is also deemed unlikely. This is based on the following:

- The likelihood of a RRP seal LOCA concurrent with another initiating event would be quantified based on a 24 hour mission time, rather than the per/year frequency for initiating events. Therefore, the frequency of an initiating event concurrent with RRP seal LOCA is small in comparison to the RRP seal LOCA initiating event discussed above. This is certainly true for the independent failure of seals while support systems are available. RRP seal failures due to loss of support are discussed below.

- Loss of support system initiating events are a significant contributor to RRP seal LOCA. A loss of RBCLC initiating event would result in a loss of RRP seal jacket cooling and loss of cooling to CRD pumps required for RRP seal purge. Loss of both seal jacket cooling and seal purge could cause a RRP seal LOCA. During a loss of RBCLC, the operators can isolate the RRP with motor operated valves if normal AC power is available. The operators can also align service water as a backup cooling source. A loss of normal AC initiating event would result in loss seal purge (CRD) and loss of seal jacket cooling (RBCLC). To restore these functions, the operators can align CRD and RBCLC to an emergency diesel. However, the ability to isolate a RRP seal LOCA with motor operated valves would be lost. The frequency of these initiating events are on the same order of magnitude or larger than small LOCA. However, additional failures such as operator failure to recover, service water failure, or emergency diesel failure result in frequencies less than small LOCA.

- Accident sequence success criteria for transients and small LOCAs are essentially the same. One significant exception is vapor suppression which is required for small LOCA success. However, vapor suppression failure is not considered a risk significant sequence. The more likely accident sequences are loss of normal AC station blackout sequences. The impact of a potential RRP seal LOCA is considered in the station blackout analysis. This analysis considers how long RCIC can operate from the CSTs with a RRP seal LOCA.

In summary, the success criteria for transients and small LOCA are essentially the same. Small LOCA sequences which have notably different success criteria than existing transients are not significant contributors to risk. The more likely sequences (e.g., station blackout) already consider the impact of RRP seal LOCA.

3.2.1.25.2 RRP Seal Design Review

<u>Reactor Recirculation Pumps</u>
The pumps are single-stage centrifugal pumps (driven flow). Each pump is designed to deliver a rated flow of 47,200 gpm at a discharge pressure head of 805 feet. The 8900 HP pump motors can receive 60 Hz power from the 13.8kV buses or 15 Hz power from the associated Low Frequency Motor Generator (LFMG) set. The 15 Hz motor generator set is used to power the pumps at 25% speed during reactor startup and low power operation to minimize cavitation of the pumps, jet pumps, and flow control valves.

<u>Reactor Recirculation Pump Shaft Seals</u>
Each pump is equipped with a dual mechanical shaft seal assembly. Each assembly consists of two full pressure seals and integral pressure controls. These components are built into a cartridge which can be replaced without removing the motor from the pump. Each individual seal in the cartridge is designed for full pump design pressure. The pump shaft passes through a breakdown bushing in the pump casing to reduce leakage to approximately 50 gpm in the event of gross failure of both shaft seals. The cavity temperature and pressure of each seal are monitored to indicate seal performance and condition. Each seal provides approximately 500 psid across its surface. The staging flow sets up an equal pressure drop across each seal.

The mechanical seals are kept clean and cool by a seal purge system. The seal purge provides a continuous flow of clean, cool water from the Control Rod Drive Hydraulic (CRDH) System. The water is injected into the vent connection of the pressure instrument tap, at about 3 to 5 gpm. One gallon flows through seal No. 1 as staging flow, while the remainder flows around the pump shaft and throttle bushing into the impeller cavity. Loss of the seal purge alone is not expected to cause a seal problem as long as the jacket cooling is available. Procedures instruct operators to monitor pump temperatures, and take appropriate actions if parameters in Table 3.2.1.25-1 are exceeded.

Cooling water, provided by the Reactor Building Closed Loop Cooling Water (CCP or RBCLC) System, flows through a cooling water jacket around the seal assembly. The primary seal water is routed through the tube side of the heat exchanger by an auxiliary impeller on the main pump impeller shaft. Loss of cooling water will cause seal temperatures to rise 40° over injection water temperature and loss of pump motor winding coolers will cause an increase in winding temperature. Procedures instruct the operator to monitor motor winding temperatures and shutdown the affected pump if temperatures exceed 248°F.

If both seal injection and cooling water are lost, procedures instruct the operator to trip the affected pump.

Seal failures are indicated as follows:

- Failure of No. 1 Seal Only (Lower Seal) No. 2 seal pressure would approach No. 1 seal pressure, resulting in FS-N007 (FS 40A/B) alarming. Alarm window 602115 (602116) would light, "RECIRC PUMP 1A(B) SEAL STAGING FLOW HIGH/LOW," and a slight temperature increase in the seal cavity would occur.

- Failure of No. 2 Seal Only (Upper Seal) No. 2 seal pressure would drop, dependant on the magnitude of the failure. Leakage through FS-N002 will exceed the setpoint and alarm. Alarm window 602109 (602110) would light, "RECIRC PUMP 1A(B) OUTER SL LEAKAGE HIGH." Annunciator 602115 (602116) "RECIRC PUMP 1A(B) SEAL STAGING FLOW HIGH/LOW" would alarm due to low flow, computer point RCSFC05 (RCSFC06).

- Failure of Both Seals Pressure in both cavities would drop, depending on the magnitude of the failures; lower seal cavity pressure may not drop significantly unless the failure was large. Both FS-N002 and FS-N007 would alarm (annunciator 602115 (602116) and 602109 (602110)), seal cavity temperature would increase.

Major leakage requires the pump to be tripped and plant shutdown.

As shown in Table 3.2.1.25-1, there is instrumentation available to the operators in the control room to monitor the pump parameters shown.

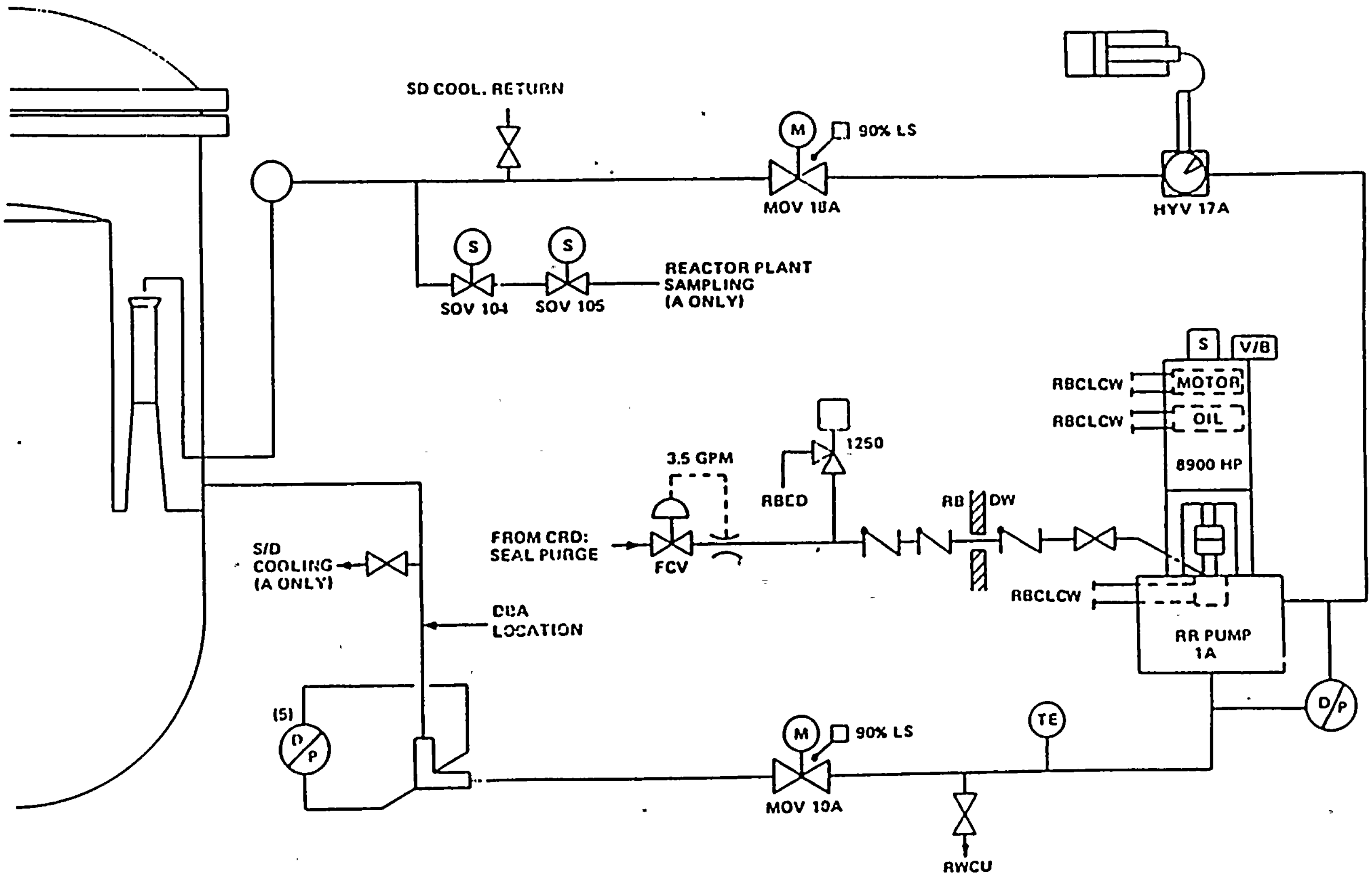
3.2.1.25.3 References

N2-OP-29, Revision 06, Reactor Recirculation System.

Table 3.2.1.25-1 Pump Instrumentation Parameters

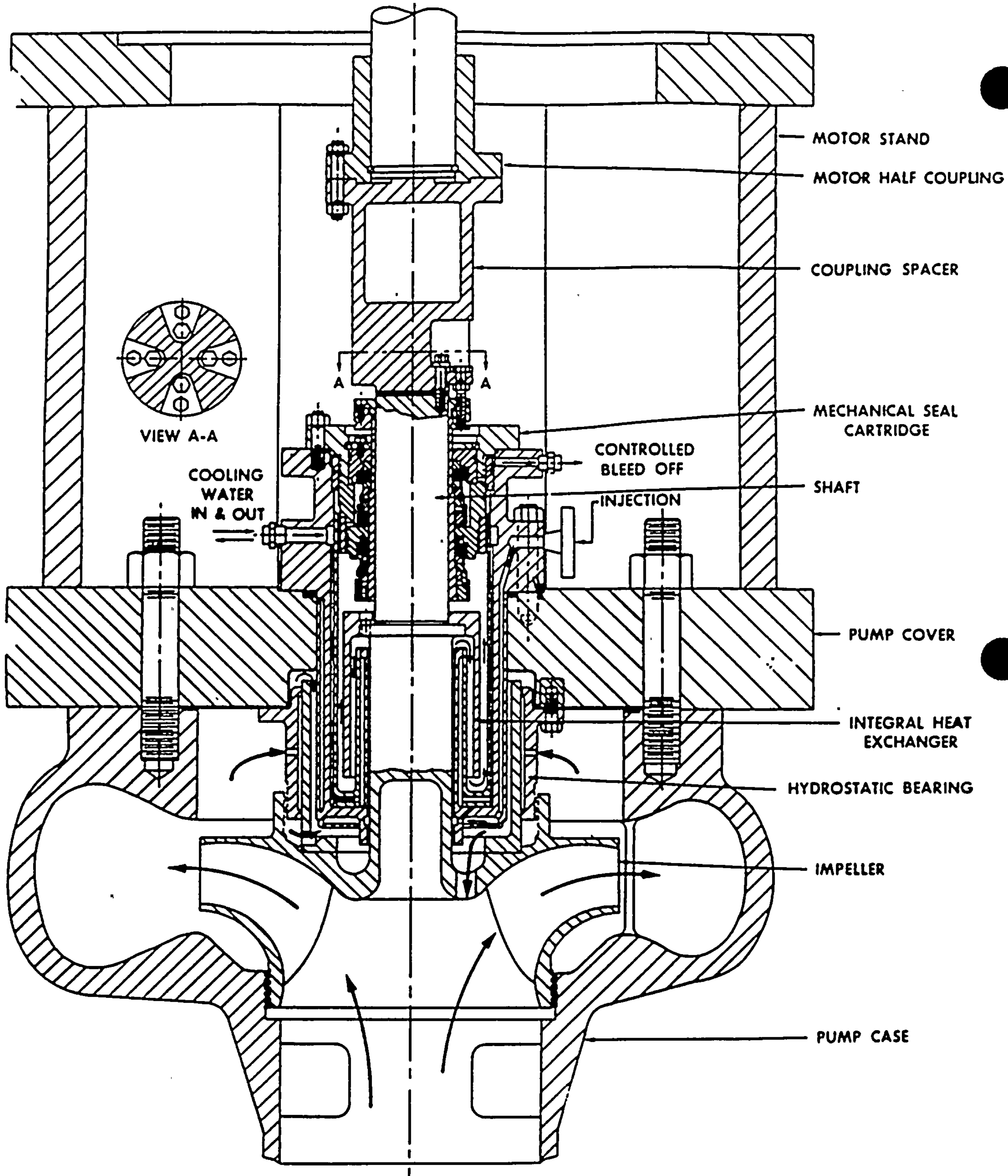
Description	Normal	Without Injection	Alert	Danger
Injection Fluid Inlet Pressure, P1	1020 psig	< 1020 psig	N/A	1200 psig
Upper Seal Staging Press., P2	510 psig	255 - 765 psig	< 255 psig (3) or > 765 psig	< 100 psig or > 920 psig
Seal Recirculation Outlet Temperature, T1 (TE29A/B, RCSTA17/18)	88°F (2) and 115°F (2)	150°F (1)	> 185°F	N/A
Upper Seal Fluid Recirculation Temperature, T3 (TE28A/B, RCSTA15/16)	96°F (2) and 127°F (2)	153°F (1)	> 185°F	> 200°F
Cooling Water (CCP) Design Inlet Temperature	60 - 105°F	60 - 105°F	N/A	110°F
Seal Staging Design Flow, Q1	1.3 gpm	1.3 gpm	0.8 gpm Lo 1.6 gpm Hi	1.8 gpm
Seal Heat Exchanger to Lower Seal Chamber Temperature, T2 (TE30A/B, RCSTA19/20)	86°F	N/A	> 185°F	> 200°F
Seal Leakage, Q2	< 0.3 gpm	0.3 gpm	> 0.8 gpm	1.2 gpm
Motor Frame Vibration (NBS88A/B, RSCNC03/04)	2 mils (2)	2 mils (2)	> 3 mils (2)	> 5 mils (2)
Shaft Vibration (NBS86A/B, RSCNC03/04)	6 mils (2)	N/A	> 13.5 mils	> 15 mils

(1) Expected temperatures at 533°F pumpage temperature, minimum cooling water flow and seals staged.
(2) At design condition.
(3) Based on injection fluid inlet pressure (P1) of 1020 psig.



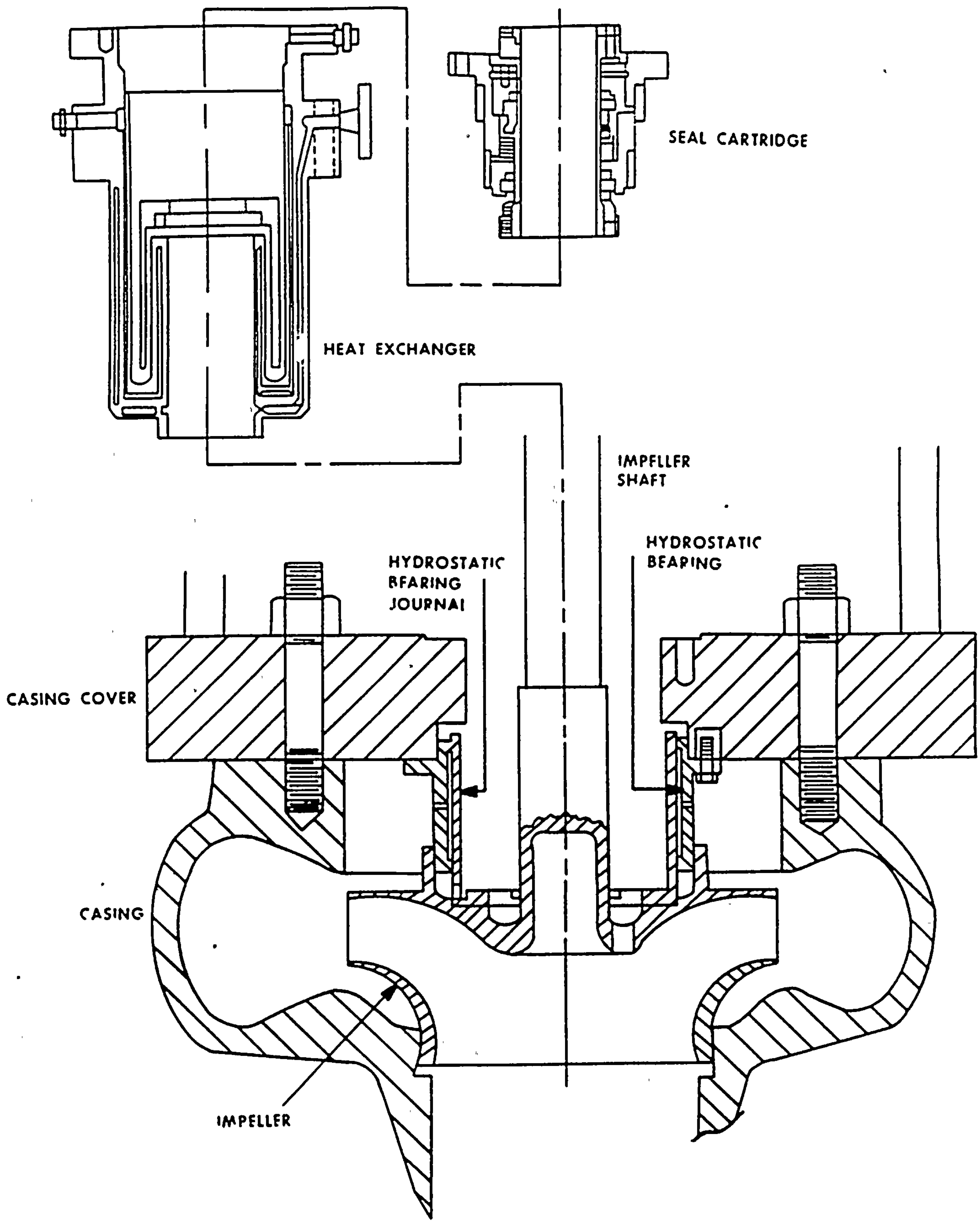
Taken from the NMP-2 Operations Technology, Section N2-RCS-08, Rev. 4

Figure 3.2.1.25-1
Reactor Recirculation
(Loop A)



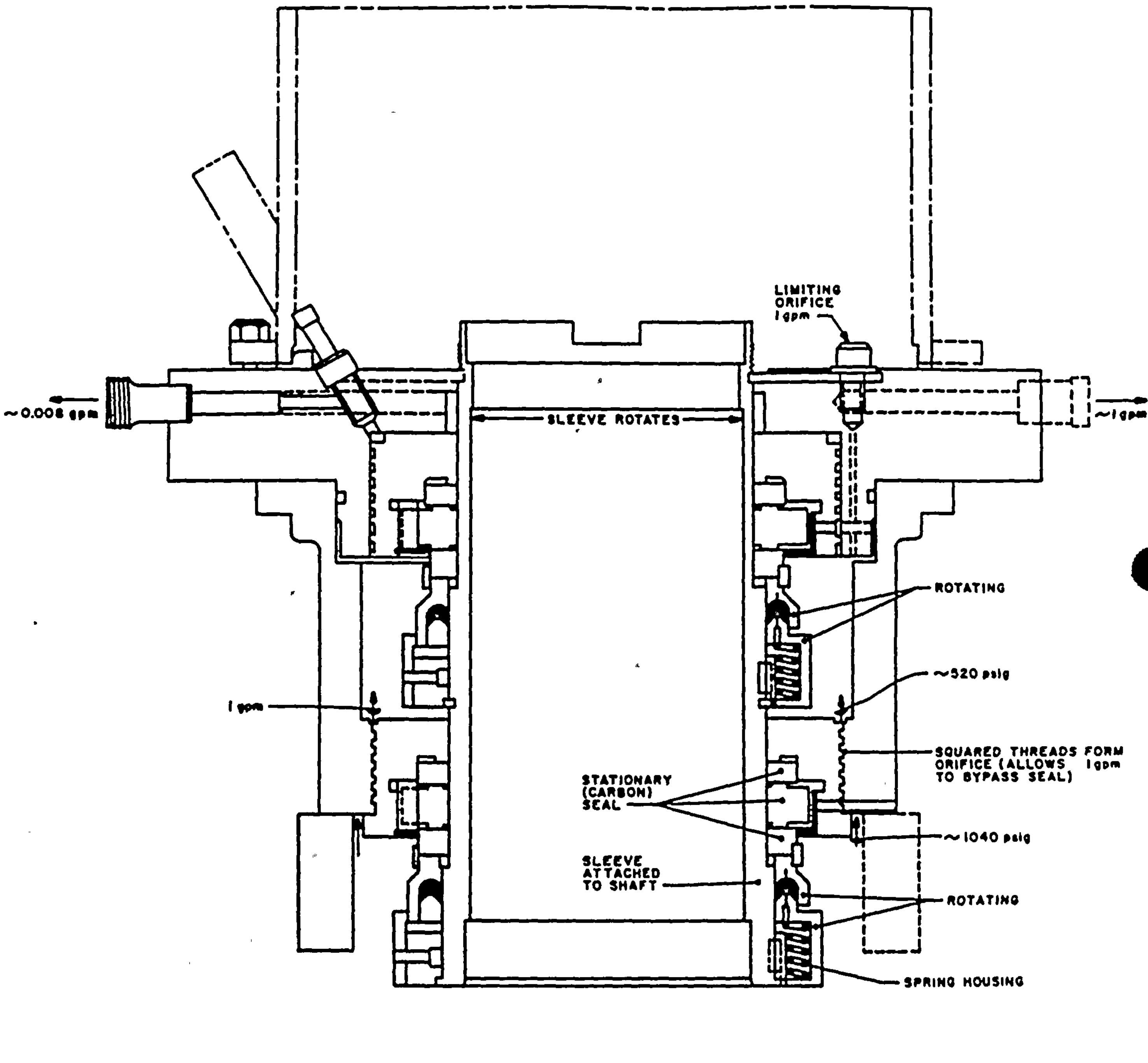
Taken from the NMP-2 Operations Technology,
Section N2-RCS-08, Rev. 4

Figure 3.2.1.25-2
Recirculation Pump
Cross Section



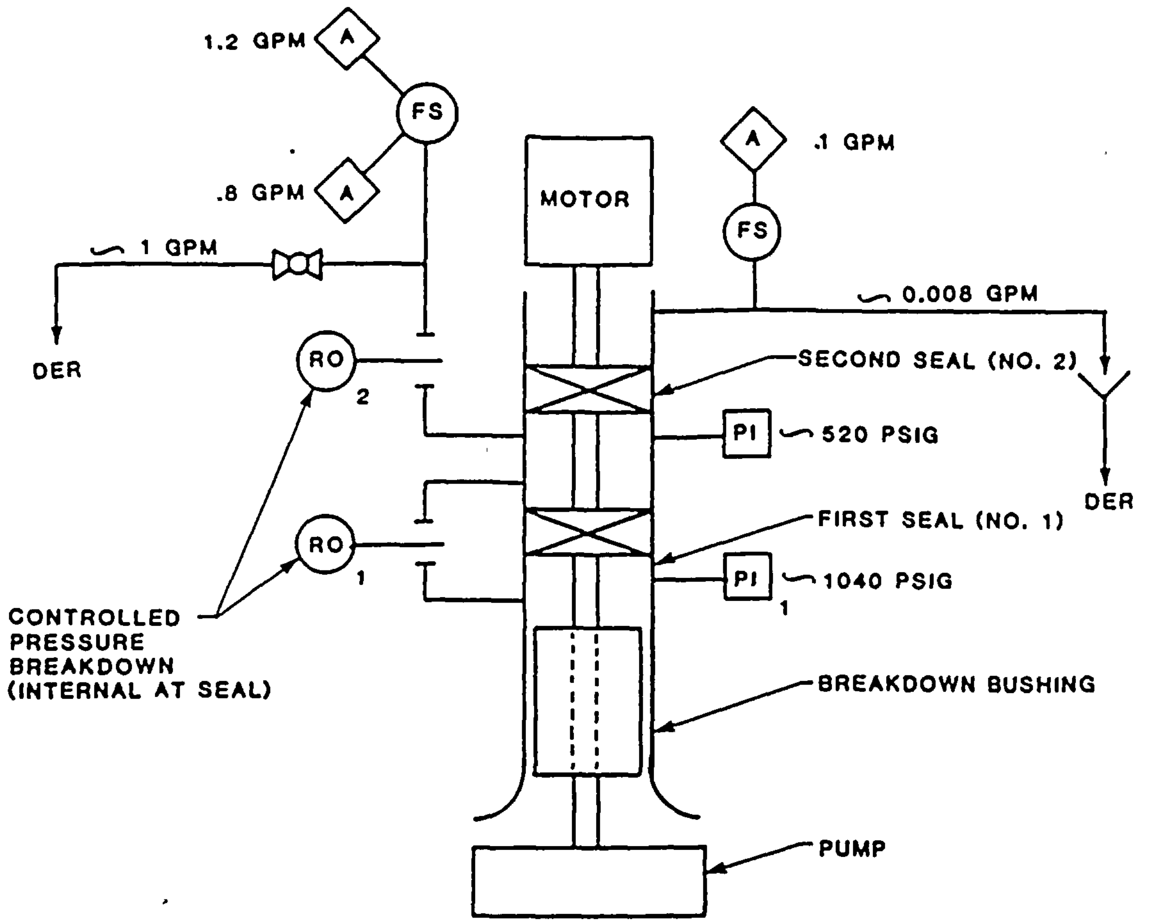
Taken from the NMP-2 Operations Technology,
Section N2-RCS-08, Rev. 4

Figure 3.2.1.25-3
Recirculation Pump
Cross Section



Taken from the <u>NMP-2 Operations Technology</u>,
Section N2-RCS-08, Rev. 4

Figure 3.2.1.25-4
Mechanical Seal Cartridge



FAILURE OF NO.1 SEAL ONLY:	NO.2 SEAL PRESSURE WOULD APPROACH NO.1 SEAL PRESSURE. LEAKAGE THROUGH NO.2 ORIFICE WILL GO TO ~1.1GPM AND FS "A" WILL ALARM HI AT >1.2GPM. SEAL TEMPERATURES DECREASE.
FAILURE OF NO.2 SEAL ONLY:	NO.2 SEAL PRESSURE WOULD DROP DEPENDENT UPON MAGNITUDE OF FAILURE LEAKAGE THROUGH FS "B" WOULD EXCEED 0.1GPM AND ALARM HI NO.2 SEAL TEMPERATURE DECREASES.
FAILURE OF BOTH SEALS:	TOTAL LEAKAGE OUT OF THE SEAL ASSEMBLY WOULD APPROACH 50GPM AS LIMITED BY THE BREAKDOWN BUSHING. BOTH FS"A" AND FS"B" WOULD ALARM HIGH. PRESSURE IN BOTH SEALS WOULD DROP DEPENDING UPON MAGNITUDE OF FAILURE. (NO.1 PRESSURE MIGHT NOT DROP SIGNIFICANTLY UNLESS FAILURE WAS LARGE.)
PLUGGING OF NO.1 INTERNAL "RO":	NO.2 PRESSURE WOULD GO TOWARD ZERO AND FLOW THRU FS"A" WOULD APPROACH ZERO AND ALARM LOW AT 0.8GPM. BOTH SEAL TEMPERATURES INCREASE.
PLUGGING OF NO.2 INTERNAL "RO":	NO.2 SEAL PRESSURE WOULD APPROACH NO.1 SEAL PRESSURE. CONTROLLED LEAKAGE WOULD APPROACH ZERO AND ALARM LOW AT 0.8GPM. BOTH SEAL TEMPERATURES WOULD INCREASE.

Taken from the NMP-2 Operations Technology, Section N2-RCS-08, Rev. 4

Figure 3.2.1.25-5
Recirc. Pump Seals
(Functional)

System 26

Recovery

3.2.1.26 Recovery

3.2.1.26.1 Success Criteria

Top event KR models operator actions required to cross-connect a 115kV source to the opposite emergency switchgear via the auxiliary boiler transformer.

Top event R1 recovers offsite power for those sequences with successful injection. However, offsite power, 1 emergency diesel, the opposite train of RHR, and containment venting has failed. For these sequences, there is at least 15 hours until severe containment conditions are reached. Successful recovery is modeled as a successful containment heat removal sequence. Failure means that the other RHR train and venting could not be recovered. Whether core damage occurs is determined in top events CI and CF.

Top events I1, I2, I3, I4, and I5 model the conditional probability of recovering offsite power to the normal electrical distribution system during various time frames of a station blackout scenario (i.e., 30 minutes, 2 hours, 8 hours, 10 hours, and 19 hours after the LOSP initiating event). Successful restoration of offsite AC power is assumed to result in a plant transient similar to a MSIV closure event, except at a significantly lower frequency.

Similarly, top events G1, G2, G3, G4, and G5 model the conditional probability of recovering at least 1 of 2 emergency diesel generators (EDGs) during a station blackout scenario in the event offsite power is not recovered. Successful restoration of at least one division of emergency AC power provides the operator with many options for re-establishing ECCS makeup to the RPV. Therefore, LOSP initiated scenarios involving the recovery of emergency AC power are further evaluated to determine whether containment heat removal is available long term to achieve a stable plant condition.

3.2.1.26.2 Development of AC Power Recovery Data

Recovery of Offsite AC Power

The cumulative probability of not restoring offsite power during station blackout events was calculated using information provided in NUREG-1032. Specifically, the generic offsite AC power recovery model, presented in the NUREG, was modified (by accounting for NMP2 design features and geographical characteristics) to develop a plant specific recovery curve describing the weighted cumulative failure probability to recover offsite power as a function of time[1].

The following table presents the derivation of the weighted cumulative failure probability to recover offsite power for the five time frames of interest:

---

[1] The frequency of loss of offsite power events (as calculated using the methodology described in NUREG-1032) due to plant-centered, grid related, and severe weather causes are 0.087/year, 0.020/year, and 0.013/year, respectively. Based on these frequencies, plant-centered events contribute approximately 72.5% to the overall initiating event frequency for loss of offsite power; whereas, the contribution or grid related and weather induced events to the LOSP initiating event frequency is approximately 16.7% and 10.8%, respectively. This information was used to "weight" the generic recovery model provided in NUREG-1032.

CUMULATIVE FAILURE PROBABILITIES
TO RECOVER OFFSITE POWER

Time (Hours)	Cumulative Failure Probability to Recover Offsite Power			Weighted Cumulative Failure Probability to Recover Offsite Power
	Plant-Centered	Grid Related	Severe Weather	
0.5	0.13	0.55	0.89	0.28
2	0.01	0.22	0.68	0.12
8	--	0.04	0.21	0.03
10	--	0.03	0.15	0.03
19	--	--	0.02	0.02

Figure 3.2.1.26-1 presents the NMP2 offsite AC power recovery curve developed using the methodology provided in NUREG-1032.

Recovery of Emergency AC Power

The cumulative probability of not recovering emergency AC power for station blackout scenario of durations 0.5, 2, 8, 10, and 19 hours was similarly calculated using information provided in NUREG-1032 concerning EDG repair time.

NUREG-1032 states that approximately 50% of all diesel generator failures reported in NUREG/CR-2989 were repaired within 8 hours. Further, if two diesel generators failed as a result of independent causes and operators could diagnose the problems to select the quickest means of repair, in 50% of these cases one of two diesel generators would have been returned to service within approximately 4 hours. Additionally, it was assumed that the cumulative failure probability of restoring a single EDG is 1.0 at 0.1 hours, and < 0.1 for an SBO duration of 30 hours for the 2 EDG configuration, and approximately 0.1 for an event duration of 70 hours with only 1 EDG potentially recoverable.

Based on these data, the cumulative failure probability as a function of time was estimated for both emergency AC system configurations.

The following table presents the results of this analysis:

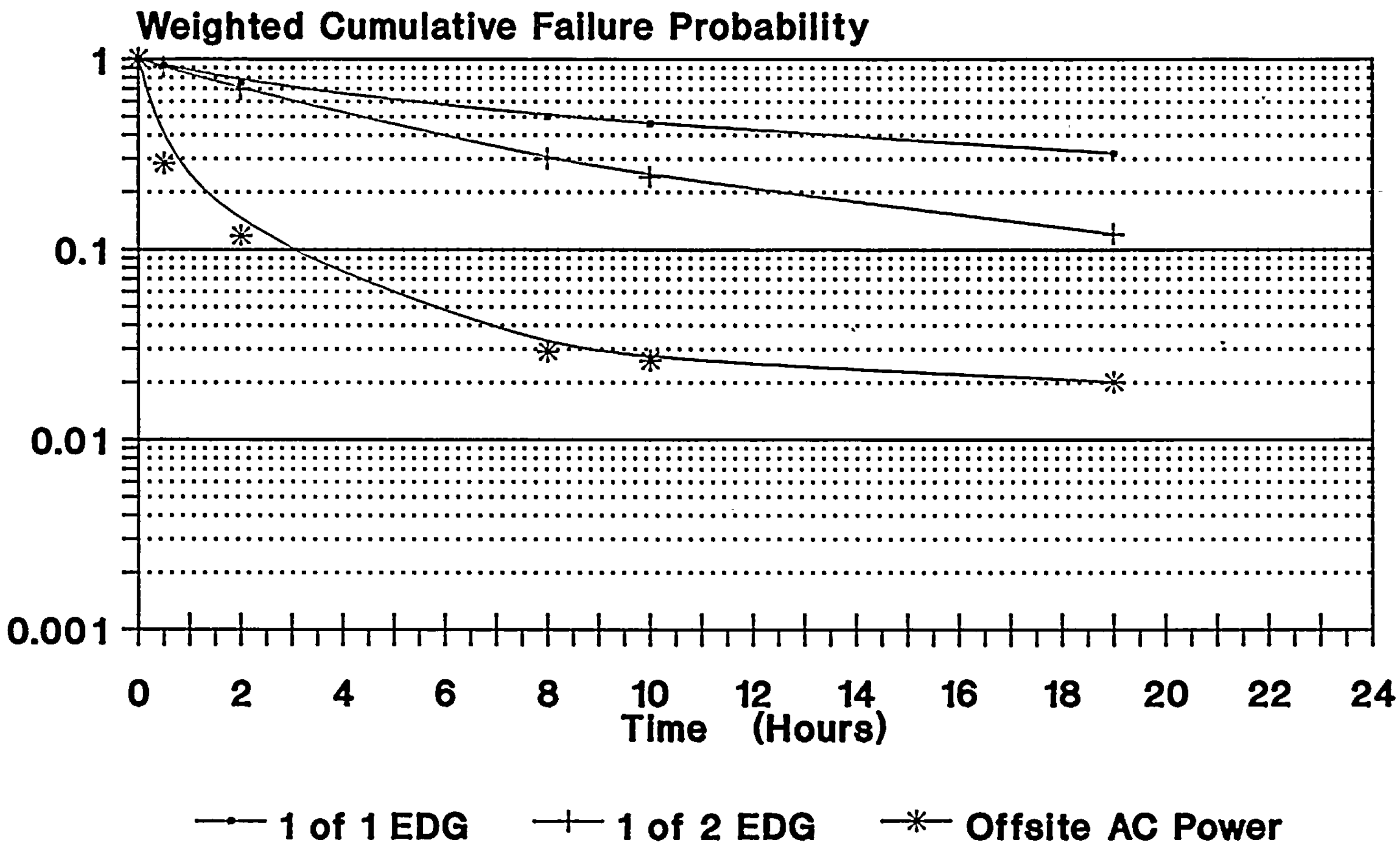
CONDITIONAL FAILURE PROBABILITIES
TO RECOVER EMERGENCY AC POWER

Time (Hours)	Cumulative Failure Probability to Recover 1 of 2 EDGs	Conditional Failure Probability to Recover 1 of 2 EDGs	Cumulative Failure Probability to Recover 1 of 1 EDG	Conditional Failure Probability to Recover 1 of 1 EDG
0.5	0.91	0.91	0.93	0.93
2	0.69	0.76	0.78	0.84
8	0.30	0.40	0.50	0.60
10	0.24	0.61	0.46	0.77
19	0.12	0.20	0.32	0.41

Figure 3.2.1.26-1 presents the NMP2 emergency AC power recovery curves developed using the methodology provided in NUREG-1032.

Rev 0 (7/92)

Figure 3.2.1.26-1
AC Power Recovery Curve



Developed based on NUREG-1032
methodology and plant-specific data.

System 27

RPV Venting

3.2.1.27 RPV Venting

3.2.1.27.1 Function

In the event that containment flooding is required per EOPs, the Reactor Pressure Vessel (RPV) will need to be vented to reduce RPV pressure and allow injection to raise the water level above the fuel zone. Emergency Operating Procedure N2-EOP-C6 (attached) covers this event. It will refer the operator to N2-EOP-6, Att. 12 (also attached). The only vent path mentioned by the EOPs is through the main steam system to the condenser.

3.2.1.27.2 Success Criteria

N2-EOP-6, Att. 12, states that the preferred method of venting the RPV during a required containment flooding scenario is through the MSIVs, steam bypass chest, and turbine bypass valves to the condenser. This path is drawn in blue on Figure 3.2.1.27-1.
If a pair of MSIVs cannot be opened, or the turbine bypass valves cannot be opened, alternate paths are given, and shown on Figure 3.2.1.27-1. Table 3.2.1.27-1 shows the equipment required for all vent paths.

Theoretically, the RPV can be vented through the ECCS piping. However, this path is not modeled because it is not explicitly stated in the EOPs.

3.2.1.27.3 Support Systems

A summary of the systems that are required for RPV venting follows. A more complete listing of the components, their support systems, and other pertinent information is in Table 3.2.1.27-1.

- Instrument Air
- 120V AC (non-divisional)
- Normal AC
- Emergency AC Division I and Division II
- Turbine Electro-hydraulic Control (EHC) system

NOTE: The breaker for valve 2MSS*MOV112 is racked out, and must be engaged before the valve can be opened. The breaker is at panel 2EHS*MCC102A, in the North Auxilary Bay, and it is believed that the environment in this area would allow for this action in an emergency.

3.2.1.27.4 System Operation

The preferred method of venting the RPV during a required containment flooding scenario is through the MSIVs, to the steam bypass chest, and through the turbine bypass valves to the condenser. This path is drawn in blue on Figure 3.2.1.27-1.

All required equipment can be operated from the control room. If any isolation signal exists, it must be overridden from the control room by jumpering certain relays, as explained in the EOPs. LOCA signals can also be overridden from the control room using the LOCA override switches for valves 2IAS*SOV166 and 2IAS*SOV184.

If a pair of MSIVs in one line cannot be opened, then 5 drain valves should be opened to bypass the MSIVs. The particular valves are listed in Table 3.2.1.27-1, along with their dependencies, and are drawn in red on the simplified drawing.

To use drain valve 2MSS*MOV112 however, a breaker must be racked in at panel 2EHS*MCC102A in the North Auxilary Bay. All other equipment can be operated from the control room.

If the turbine bypass valves cannot be opened, the EOP directs an operator to open numerous main steam line drain valves. These valves are listed in Table 3.2.1.27-1, along with their associated dependencies. These valves are drawn in green on the attached simplified drawing.

3.2.1.27.5 Instrumentation and Controls

Except for 2MSS*MOV112, all necessary equipment has controls and indication in the control room. As stated previously, the breaker for this is in the North Auxilary Bay and must be racked in before it can be opened

3.2.1.27.6 Technical Specifications

Two main steam isolation valves (MSIVs) per main steam line shall be operable with closing times greater than 3 seconds and less than or equal to 5 seconds.

With at least one MSIV operable in each main steam line, 4 hours is given to either restore valve operation or isolate the affected main steam line by use of a deactivated MSIV in the closed position. Otherwise, be in at least hot shutdown within the next 12 hours and in cold shutdown within the following 24 hours.

The main turbine bypass system shall be operable when running in operational condition 1 and when thermal power is 25% or more of rated thermal power.

With the main turbine bypass inoperable, restore the system to operable status within 1 hour

At least 16 of the reactor coolant system safety/relief valves shall operate with the specified code safety function lift settings; the acoustic monitor for each operable valve shall be operable.

With less than 16 safety/relief valves operating, be in at least hot shutdown within 12 hours and cold shutdown within the next 24 hours.

With one or more safety/relief valves stuck open, provided that the suppression pool temperature is less than 110°F, close the stuck valve; if unable to do so within 5 minutes or if the average temperature of the suppression pool water is above 110°F, place the reactor mode switch in shutdown position.

The acoustic monitors are to be restored to operable status within 7 days or be in at least hot shutdown within the next 12 hours and in cold shutdown within the next 24 hours.

3.2.1.27.7 Surveillance Testing and Maintenance

In-service inspection shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable addenda required by 10CFR 50.55a(g), except where written relief has been granted by the NRC.

Each primary containment isolation valve shall be demonstrated operable before returning the valve to service, during cold shutdown or refueling, once per 18 months by cycling the valve at least once and verifying that the valve actuates to its isolation position.

The isolation time of each primary containment power operated or automatic valve shall be determined within its limit when tested per Technical Specification Table 3.6.3-1.

The acoustic monitor for each safety/relief valve shall be demonstrated operable with the setpoint verified to be 0.25 of the full-open noise level by performance of a channel functional test at least once per 31 days and a channel calibration test at least once per 18 months.

3.2.1.27.8 References

N2-EOP-C6, Rev. 4
N2-EOP-6, Attachment 12, Rev. 0
PID-23A-10
LSK 27-19
LSK 31-01.25
Other PIDs as referenced on drawing

3.2.1.27.9 Initiating Event Potential

The RPV venting function does not result in any potential initiating events.

3.2.1.27.10 Equipment Location

Equipment location is listed in Table 3.2.1.27-1

3.2.1.27.11 Operating Experience

There were no outstanding operational events relevant to this study. Plant specific component operational data is detailed in section 3.3.2.

3.2.1.27.12 Modeling Assumptions

1. 2MSS-AOV87A, B, C, and D are not modeled because they are normally open, fail open, and for this scenario are required to be opened.

2. 2MSS-AOV92A and B, which fail closed on loss of air or normal AC, are assumed to close in the initiation of the event, and must be opened for RPV venting.

3. 2MSS-AOV88A and B are not modeled. They are normally open, fail open on loss of air, and are for this scenario are required to be open.

4. For the purpose of bypassing the turbine bypass chest, either all MOVs or all AOVs must open and remain open. This is a simplification because the exact number of required valves is not known.

5. Event tree top events NA and NB are modeled at the top of the tree, since the MSIVs require either NA or NB, and 2MSS-MOV187 requires NA. Therefore, failure of NA and NB will fail the system.

3.2.1.27.13 Logic Model and Results

Results are summarized in the Level 2 documentation.

Table 3.2.1.27-1
RPV Vent Equipment

Comp ID	Motive Power	Indication	PNL	CR SW?	Norm	Fail	Bld	El
2MSS-AOV87A	Instrument Air	Local / CR	P824	Y	O	FO	MST	238
2MSS-SOV87A	2SCI-PNLB101 120V AC N						MST	245
2MSS-AOV87B	Instrument Air	Local / CR	P824	Y	O	FO	MST	238
2MSS-SOV87B	2SCI-PNLB101 120V AC N						MST	245
2MSS-AOV87C	Instrument Air	Local / CR	P824	Y	O	FO	MST	238
2MSS-SOV87C	2SCI-PNLB101 120V AC N						MST	245
2MSS-AOV87D	Instrument Air	Local / CR	P824	Y	O	FO	MST	238
2MSS-SOV87D	2SCI-PNLB101 120V AC N						MST	245
2MSS*AOV6A	Instrument Air	Local / CR	P602	Y	O	FC	PC	251
2MSS*SOV6A-2	2VBS*PNLB106 120V AC N						PC	252
2MSS*SOV6A-3	2VBS*PNLA106 120V AC N						PC	252
2MSS*AOV6B	Instrument Air	Local / CR	P602	Y	O	FC	PC	251
2MSS*SOV6B-2	2VBS*PNLB106 120V AC N						PC	252
2MSS*SOV6B-3	2VBS*PNLA106 120V AC N						PC	252

Table 3.2.1.27-1
RPV Vent Equipment

Comp ID	Motive Power	Indication	PNL	CR SW?	Norm	Fail	Bld	El
2MSS*AOV6C	Instrument Air	Local / CR	P602	Y	O	FC	PC	251
2MSS*SOV6C-2	2VBS*PNLB106 120V AC N						PC	252
2MSS*SOV6C-3	2VBS*PNLA106 120V AC N						PC	252
2MSS*AOV6D	Instrument Air	Local / CR	P602	Y	O	FC	PC	251
2MSS*SOV6D-2	2VBS*PNLB106 120V AC N						PC	252
2MSS*SOV6D-3	2VBS*PNLA106 120V AC N						PC	252
2MSS*AOV7A	Instrument Air	Local / CR	P602	Y	O	FC	MST	251
2MSS*SOV7A-2	2VBS*PNLB105 120V AC N						MST	251
2MSS*SOV7A-3	2VBS*PNLA105 120V AC N						MST	251
2MSS*AOV7B	Instrument Air	Local / CR	P602	Y	O	FC	MST	251
2MSS*SOV7B-2	2VBS*PNLB105 120V AC N						MST	251
2MSS*SOV7B-3	2VBS*PNLA105 120V AC N						MST	251
2MSS*AOV7C	Instrument Air	Local / CR	P602	Y	O	FC	MST	251
2MSS*SOV7C-2	2VBS*PNLB105 120V AC N						MST	251

Table 3.2.1.27-1
RPV Vent Equipment

Comp ID	Motive Power	Indication	PNL	CR SW?	Norm	Fail	Bld	El
2MSS*SOV7C-3	2VBS*PNLA105 120V AC N						MST	251
2MSS*AOV7D	Instrument Air	Local / CR	P602	Y	O	FC	MST	251
2MSS*SOV7D-2	2VBS*PNLB105 120V AC N						MST	251
2MSS*SOV7D-3	2VBS*PNLA105 120V AC N						MST	251
2MSS-MOV187	2NHS-MCC003	Local / CR	P602	Y	C	AS IS	MST	250
2IAS*SOV166	2SCM*PNL102A I	Local / CR	P851	Y	O	FC	SC	294
2IAS*SOV184	2SCM*PNL302B II	Local / CR	P851	Y	O	FC	SC	294
2MSS-PSV89A	EHC	Local / CR	P851	Y	C	FC[1]	TB	299
2MSS-PSV89B	EHC	Local / CR	P851	Y	C	FC[1]	TB	299
2MSS-PSV89C	EHC	Local / CR	P851	Y	C	FC[1]	TB	299
2MSS-PSV89D	EHC	Local / CR	P851	Y	C	FC[1]	TB	299
2MSS-PSV89D	EHC	Local / CR	P851	Y	C	FC[1]	TB	299
2MSS*MOV207	2NHS-MCC012	Local / CR	P824	Y	C	FC	PC	249
2MSS*MOV111	2EHS*MCC302	Local / CR	P602	Y	C	FC	PC	249
2MSS*MOV112	2EHS*MCC102	Local / CR	P602	Y	C	FC	MST	246
2EHS*MCC102A	2EJS*US1						ABN	240
2MSS-MOV187	2NHS-MCC003	Local / CR	P602	Y	C	AS IS	MST	250
2MSS-MOV21A	2NHS-MCC003	Local / CR	P824	Y	C	FC	TB	277

Table 3.2.1.27-1
RPV Vent Equipment

Comp ID	Motive Power	Indication	PNL	CR SW?	Norm	Fail	Bld	El
2MSS-MOV21B	2NHS-MCC003	Local / CR	P824	Y	C	FC	TB	277
2MSS-MOV21C	2NHS-MCC003	Local / CR	P824	Y	C	FC	TB	277
2MSS-MOV21D	2NHS-MCC003	Local / CR	P824	Y	C	FC	TB	277
2MSS-MOV10A	2NHS-MCC003	Local / CR	P824	Y	C	FC	TB	250
2MSS-MOV10CA	2NHS-MCC003	Local / CR	P824	Y	C	FC	TB	264
2MSS-AOV10B	Instrument Air	Local / CR	P824	Y	C	FC	TB	263
2MSS-AOV10D	Instrument Air	Local / CR	P824	Y	C	FC	TB	263
2MSS-MOV199	2NHS-MCC003	Local / CR	P824	Y	C	FO	TB	277
2MSS-AOV201	Instrument Air	Local / CR	P824	Y	C	FO	TB	281
2MSS-MOV147	2NHS-MCC003	Local / CR	P824	Y	C	FC	TB	277
2MSS-AOV191	Instrument Air	Local / CR	P824	Y	C	FO	TB	286
2MSS-AOV194	Instrument Air	Local / CR	P824	Y	C	FO	TB	286
2MSS-AOV203	Instrument Air	Local / CR	P824	Y	C	FO	TB	286
2MSS-AOV205	Instrument Air	Local / CR	P824	Y	C	FO	TB	286
2MSS-AOV209	Instrument Air	Local / CR	P824	Y	C	FO	TB	281
2MSS-AOV85A	Instrument Air	Local / CR	P824	Y	C	FO	MST	238
2MSS-AOV85B	Instrument Air	Local / CR	P824	Y	C	FO	MST	238
2MSS-AOV85C	Instrument Air	Local / CR	P824	Y	C	FO	MST	238
2MSS-AOV85D	Instrument Air	Local / CR	P824	Y	C	FO	MST	238

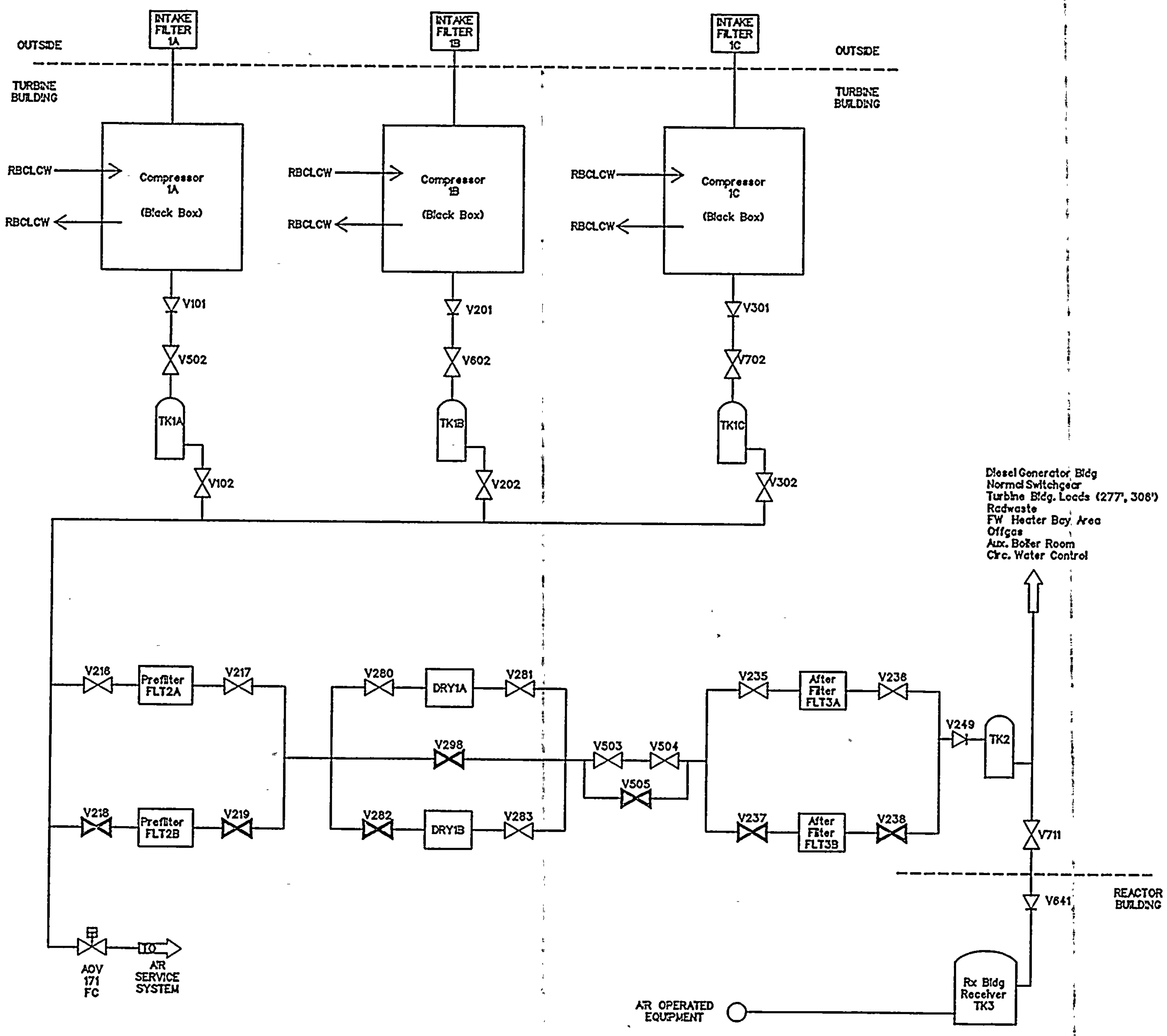
Table 3.2.1.27-1 Legend and Notes

•	'Motive Power'	'N' indicates non-divisional
•	'Indication'	Local indicates local indication CR indicates control room indication
•	'PNL'	What panel in control room
•	'CR SW'	Control Room control switch Yes or No
•	'Norm'	Normal Operating position
•	'Fail'	Position on loss of motive power
•	'Bld'	Building equipment is located in: MST - Main Steam Tunnel PC - Primary Containment ABN - Auxilary Building North SC - Secondary Containment
•	'El'	Elevation in feet

1. These valve's have accumulators, and will temporarily remain open. However, these valves are assumed to eventually close.

[illegible]

Rev 0 (7/92)



References: PID-19A-8
PID-19B-12
PID-19C-9
PID-19D-10
PID-19E-9
PID-19F-7
PID-19G-13
PID-19H-10
PID-19J-10
PID-19K-7
PID-19L-4
PID-19M-4

FIGURE 3.2.1.23-1
INSTRUMENT AIR

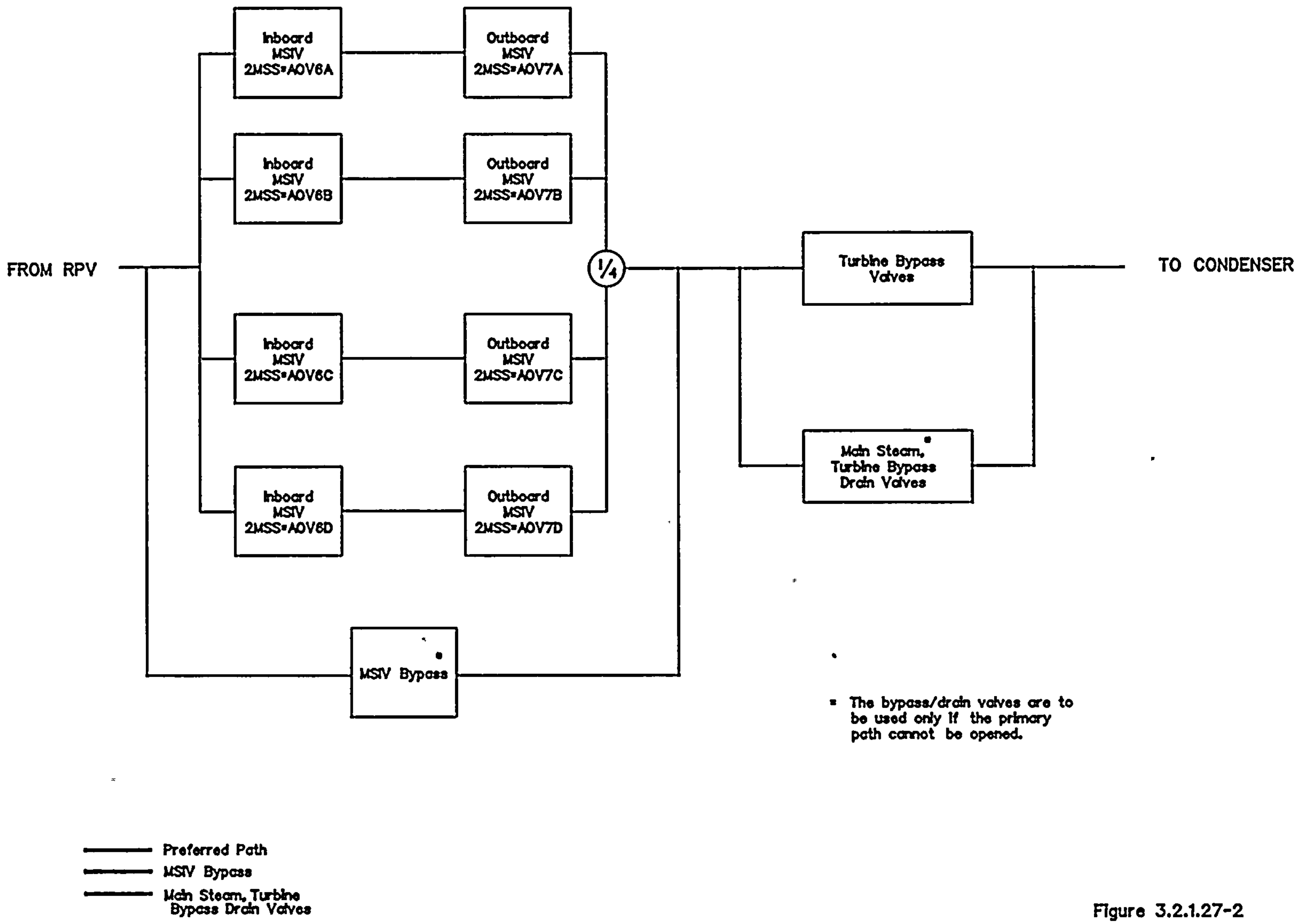
SI
APERTURE
CARD

Also Available On
Aperture Card

I:\NMP2IPE\DGN\2S23.DGN Jul. 07, 1992 11:15:36

9208050191 - 34

Rev 0 (7/92)



* The bypass/drain valves are to be used only if the primary path cannot be opened.

Figure 3.2.1.27-2
RPV Venting Success Diagram

I:\NMP2IPE\DGN\2S27.DGN Jul. 06, 1992 13:39:38

System 28

Vessel and Primary Containment Instrumentation

3.2.1.28 Vessel and Primary Containment Instrumentation

3.2.1.28.1 System Function

The reactor vessel instrumentation (ISC) and containment atmospheric monitoring (CMS) systems supply information about vessel and containment parameters during normal operating and post accident conditions. These parameters include:

- Vessel level and pressure,
- Drywell pressure and temperature,
- Suppression chamber pressure and temperature,
- Suppression pool level and temperature,
- Hydrogen and oxygen concentration in the drywell,
- Hydrogen and oxygen concentration in the suppression chamber, and
- Gaseous and particulate radiation levels.

3.2.1.28.2 Support Systems

<u>Vessel Monitoring Instrumentation</u>

For each parameter listed below, with the exception of shutdown and upset level, there is redundancy in both the instrument power sources and sensing lines.

Parameter	Transmitter	Power Supply
Fuel Zone Level	2ISC*LT13A	2VBS*PNL101A
	2ISC*LT13B	2VBS*PNL301B
Wide Range Level	2ISC*PDT110	2BYS-PNLB101
	2ISC*LT11A	2VBS*PNLA103
	2ISC*LT11B	2VBS*PNLB103
	2ISC*LT11C	2VBS*PNLA104
	2ISC*LT11D	2VBS*PNLB104
	2ISC*LT10A	2CES*IPNL414
	2ISC*LT10B	2CES*IPNL414
	2ISC*LT10C	2CES*IPNL414
	2ISC*LT10D	2CES*IPNL414
	2ISC*LT9A	2VBS*PNL101A
	2ISC*LT9B	2VBS*PNL301B
	2ISC*LT9C	2VBS*PNL101A
	2ISC*LT9D	2VBS*PNL301B
Narrow Range Level	2ISC*LT7A	2VBS*PNLA104
	2ISC*LT7B	2VBS*PNLB103
	2ISC*LT7C	2VBS*PNLA103
	2ISC*LT7D	2VBS*PNLB104
	2ISC*LT8A	2BYS*PNL202A
	2ISC*LT8B	2BYS*PNL202A
	2ISC*LT8C	2BYS*PNL202B

Parameter	Transmitter	Power Supply
	2ISC*LT8D	2BYS*PNL202B
	2ISC*PDT14A	2VBS-PNLB102
	2ISC*PDT14B	2VBS-PNLB101
	2ISC*PDT14C	2VBS-PNLA102
	2ISC*PTD12A	2VBS*PNL101A
	2ISC*PDT12B	2VBS*PNL301B
Upset Range Level	2ISC*PDT110	2BYS-PNLB101
Shutdown Range Level	2ISC*LT105	2VBS*PNLB104
Reactor Pressure	2ISC*PT2A	2BYS*PNL202A
	2ISC*PT2B	2BYS*PNL202A
	2ISC*PT2C	2BYS*PNL202B
	2ISC*PT2D	2BYS*PNL202B
	2ISC*PT4A	2VBS*PNLA104
	2ISC*PT4B	2VBS*PNLB103
	2ISC*PT4C	2VBS*PNLA103
	2ISC*PT4D	2VBS*PNLA104
	2ISC*PT5A	2VBS*PNL101A
	2ISC*PT5D	2VBS*PNL101A
	2ISC*PT6A	2VBS*PNL101A
	2ISC*PT6B	2VBS*PNL301B
	2ISC*PT108	2VBS-PNLB101
	2ISC*PT109	2VBS-PNLB101
	2ISC*PT172	2VBS-PNLB101

Drywell Monitoring Instrumentation

Parameter	Transmitter	Power Supply
Drywell Pressure	2CMS*PT1A	2VBS*PNL101A
	2CMS*PT1B	2VBS*PNL301B
	2CMS*PT2A	2VBS*PNL101A
	2CMS*PT2B	2VBS*PNL301B
	2ISC*PT15A	2VBS*PNLA104
	2ISC*PT15B	2VBS*PNLB103
	2ISC*PT15C	2VBS*PNLA103
	2ISC*PT15D	2VBS*PNLB104
	2ISC*PT16A	2CES*IPNL414
	2ISC*PT16B	2CES*IPNL414
	2ISC*PT16C	2CES*IPNL414
	2ISC*PT16D	2CES*IPNL414
	2ISC*PT17A	2VBS*PNL101A
	2ISC*PT17B	2VBS*PNL301B
	2ISC*PT17C	2VBS*PNL101A
	2ISC*PT17D	2VBS*PNL301B

Parameter	Transmitter	Power Supply
Drywell Temperature	2CMS*TE101	2VBS*PNL101A
	2CMS*TE102	2VBS*PNL101A
	2CMS*TE103	2VBS*PNL101A
	2CMS*TE104	2VBS*PNL101A
	2CMS*TE105	2VBS*PNL101A
	2CMS*TE106	2VBS*PNL101A
	2CMS*TE116	2VBS*PNL301B
	2CMS*TE117	2VBS*PNL301B
	2CMS*TE118	2VBS*PNL301B
	2CMS*TE119	2VBS*PNL301B
	2CMS*TE120	2VBS*PNL301B
	2CMS*TE121	2VBS*PNL301B

Suppression Chamber/Pool Monitoring Instrumentation

Parameter	Transmitter	Power Supply
Suppression Chamber Pressure	2CMS*PT7A	2VBS*PNL101A
	2CMS*PT7B	2VBS*PNL301B
Suppression Chamber Temperature	2CMS*TE107	2VBS*PNL101A
	2CMS*TE108	2VBS*PNL101A
	2CMS*TE109	2VBS*PNL101A
	2CMS*TE122	2VBS*PNL301B
	2CMS*TE123	2VBS*PNL301B
	2CMS*TE124	2VBS*PNL301B
Suppression Pool Level	2CMS*LT9A	2VBS*PNL101A
	2CMS*LT9B	2VBS*PNL301B
	2CMS*LT11A	2VBS*PNL101A
	2CMS*LT11B	2VBS*PNL301B
Suppression Pool Temperature	2CMS*TE67A	2VBS*PNL101A
	2CMS*TE67B	2VBS*PNL301B
	2CMS*TE68A	2VBS*PNL101A
	2CMS*TE68B	2VBS*PNL301B
	2CMS*TE69A	2VBS*PNL101A
	2CMS*TE69B	2VBS*PNL301B
	2CMS*TE70A	2VBS*PNL101A
	2CMS*TE70B	2VBS*PNL301B
	2CMS*TE50B	2VBS*PNL301B
	2CMS*TE51B	2VBS*PNL301B
	2CMS*TE52B	2VBS*PNL301B
	2CMS*TE53B	2VBS*PNL301B
	2CMS*TE54B	2VBS*PNL301B
	2CMS*TE55B	2VBS*PNL301B
	2CMS*TE56B	2VBS*PNL301B

Parameter	Transmitter	Power Supply
	2CMS*TE57B	2VBS*PNL301B
	2CMS*TE58B	2VBS*PNL301B
	2CMS*TE59B	2VBS*PNL301B

Hydrogen/Oxygen Monitoring

The following summarizes Emergency AC dependencies (Division I Emergency AC MCCs and panels are supplied by 2EJS*US1, Division II by 2EJS*US3):

Function	Equipment ID	Power Supply
Analyzer Control	2CMS*PNL66A	2EHS*MCC102 (I)
	2CMS*PNL66B	2EHS*MCC102 (I)
Analyzer/Pump Remote Control	2CMS*PNL73A	2EHS*MCC302 (II)
	2CMS*PNL73B	2EHS*MCC302 (II)
Drywell Isolation		
to A Analyzer	2CMS*SOV24A,C	2SCM*PNL102A (I)
from A Analyzer	2CMS*SOV33A,32A	2SCM*PNL102A (I)
to B Analyzer	2CMS*SOV24B,D	2SCM*PNL302B (II)
from B Analyzer	2CMS*SOV33B,32B	2SCM*PNL302B (II)
Suppression Chamber Isolation		
to A Analyzer	2CMS*SOV26A,C	2SCM*PNL102A (I)
from A Analyzer	2CMS*SOV34A,35A	2SCM*PNL102A (I)
to B Analyzer	2CMS*SOV26B,D	2SCM*PNL302B (II)
from B analyzer	2CMS*SOV34,35B	2SCM*PNL302B (II)
Drywell Collection Lines		
to A Analyzer	2CMS*SOV23A,C,E	2SCM*PNL102A (I)
to B Analyzer	2CMS*SOV23B,D,F	2SCM*PNL302B (II)
Suppression Chamber Collection Lines		
to A Analyzer	2CMS*SOV25A,C	2SCM*PNL102A (I)
to B Analyzer	2CMS*SOV25B,D	2SCM*PNL302B (II)
A Analyzer Inlet Valve	2CMS*SOV64A	2SCM*PNL102A (I)
H_2/O_2 Analyzer 2A Power		2SCV*PNL101A (I)
H_2/O_2 Analyzer Pump A	2CMS*P2A	2EHS*MCC102 (I)
A Analyzer Outlet Valve	2CMS*SOV65A	2SCM*PNL102A (I)
B Analyzer Inlet Valve	2CMS*SOV64B	2SCM*PNL302B (II)
H_2/O_2 Analyzer 2B Power		2SCV*PNL301B (II)
H_2/O_2 Analyzer Pump B	2CMS*P2B	2EHS*MCC302 (II)
B Analyzer Outlet Valve	2CMS*SOV65B	2SCM*PNL302B (II)

The containment isolation valves get an isolation signal on: reactor water level LO-LO, HIGH drywell pressure and manual isolation. These valves have LOCA override capability in the Control Room.

3.2.1.28.4 System Operation

The systems should be operating in all modes of plant operation. More complete information is discussed below for each type of instrumentation.

3.2.1.28.5 Instrumentation and Controls

Vessel Monitoring Instrumentation
Vessel level indicators can measure vessel level over the range of -165" through 545". The instrument zero is set at 380.69" above the vessel reference zero. This instrument zero corresponds to the top of the upper core support plate. Top of Active Fuel (TAF) is at -14.4" below this instrument reference point.

In the control room, there are five (5) types of water level indicators and trip levels. They are:

Indication	Range	Trip Level
Fuel Zone Level	- 165" → + 35"	*
Wide Range Level	- 5" → +205"	*
Narrow Range Level	+145" → +205"	*
Upset Range Level	+145" → +325"	*
Shutdown Range Level	+145" → +545"	*

* See GEK 761E4454F sh. 1 for the various trip levels and functions.

Within each of these groups, there are redundant power sources and instrument sensing lines.

Reactor pressure monitors, located in the control room, have a range of 0 through 1500 psig. There is both redundancy in instrument power sources and in sensing lines.

Drywell Monitoring Instrumentation
The Division I drywell pressure transmitters supply Panel 601 for narrow range (-5 psig to +5 psig) and wide range indication (0-150 psig). The narrow range supplies computer points and a high drywell pressure alarm at 1.5 psig. The wide range supplies a computer point in addition to Control Room indication. The Division II narrow range drywell pressure transmitters provide indication (-5 to +5 psig) on Panel 601, a recorder on Panel 898, and a high drywell pressure alarm (1.5 psig) on Panel 875. The wide range transmitter provides a signal to a recorder on Panel 898.

The highest and lowest drywell air temperatures, of the six, are displayed on Panel 873 for Division I and Panel 875 for Division II. Temperature elements provide signals to alarms and analog computer points. A high drywell temperature alarm actuates annunciators at 150°F.

Suppression Chamber/Pool Monitoring Instrumentation
Parameters measured in the suppression chamber include:

- Suppression chamber pressure,
- Suppression chamber temperature,
- Suppression pool water level, and
- Suppression pool temperature.

One pressure transmitter for each division monitors suppression chamber pressure. Division I supplies wide range (0-150 psig) indication on Panel 601; Division II supplies wide range to a recorder on Panel 898. Division I narrow range (0-5 psig) displays on Panel 601, as does Division II.

The suppression chamber air temperature is displayed in the Control Room in a similar fashion on Panel 873 for Division II and Panel 875 for Division I. Temperature elements provide signals to alarms and analog computer points. High suppression chamber temperature alarm annunciators alarm at 83.5°F.

Suppression pool level is monitored by two level transmitters per division. The Division I and Division II narrow range (198 to 202 feet) indicates on Panel 601. Division I wide range (197 to 217 feet) indicates on Panel 601 and Division II supplies a recorder on Panel 898. The Division I and Division II transmitters also supply computer points.

Suppression pool temperature can be monitored on Panel 601 from 10 different locations for Division I and from 10 different locations for Division II. The temperatures are monitored at elevation 199' in the pool using thermocouples. A selector switch is on Panel 601. Temperature switches actuate annunciators at 82.5°F and at 101°F.

Four Division I and Division II suppression pool water temperature elements at the 197' elevation provide post accident monitoring of temperature to a computer point and a temperature selector switch on Panel 601. In addition, temperature elements supply signals to recorders on Panel 898. These temperature elements do not actuate alarms.

Hydrogen/Oxygen Sample Analyzer
The sample analyzers monitor the drywell and suppression chamber on a rotating basis in five locations. The Division I analyzer provides indication for Hydrogen (0-30%) on Panel 601 and Division II supplies a recorder on Panel 898. An alarm comes off each division at greater than 3.7% Hydrogen concentration. Each division also supplies computer points.

Oxygen concentration information is supplied by each division. Division I supplies indication on Panel 601 (0-10%) and Division II supplies a recorder on Panel 898. Each division generates alarms at greater than 3.5% oxygen concentration and each division supplies computer points.

Alarm annunciators for both hydrogen and oxygen are located on Panel 873 for Division I and Panel 875 for Division II.

Radiation Monitoring

Division I and II radiation monitors are provided. Each monitor draws a sample from two elevations in the drywell and measures for gaseous and particulate indication. The monitors provide indications on Panel 880. If a setpoint is exceeded, it alarms an annunciator on Panel 851.

Controls

The controls for the Containment Atmospheric Monitoring System are located on Panels 873 for Division I and Panel 875 for Division II. There are LOCA override switches located on the respective panels. These are used to sample the containment following an isolation.

To manually isolate the system, there are two arm and depress pushbuttons on Panel 602. One for Division II is for inboard valves. The other, for Division I, is for outboard valves. When actuated, an amber light is illuminated.

The hydrogen/oxygen sample systems may be controlled for single stream sampling in the control room at Panel 873 (Division I) and Panel 875 (Division II).

3.2.1.28.6 Technical Specifications

Applicable sections include:

Isolation Actuation Instrumentation	3.3.2.	4.3.2
Accident Monitoring Instrumentation	3.3.7.5	4.3.7.5
Suppression Pool	3.5.3	4.5.3
Primary Containment Leakage	3.6.1.2	4.6.1.2
Drywell Average Air Temperature	3.6.1.6	4.6.1.6
Primary Containment Isolation Valves	3.6.3	4.6.3
Drywell and Suppression Chamber Oxygen Concentration	3.6.6.2	4.6.6.2

3.2.1.28.7 References

Operations Technology

Chapter 23C, CMS, Containment Atmospheric Monitoring System
Chapter 62, RMS, Radiation Monitoring System
Chapter 63, PASS, Post Accident Sampling System

GEK 761E4454F sh. 1
TL2CMS series test loop diagrams
TL2ISC series test loop diagrams
N2-OP-82, Containment Atmospheric Monitoring

3.2.1.28.8 Initiating Event Potential

In the event of a tubing system leak during a LOCA, the inside of the CMS panels may become filled with radioactive, explosive, or very hot (greater than 270 degree) gases.

3.2.1.28.9 Equipment Location

The valves on the collection lines are located between elevation 290' and 297' in the drywell, and at elevations 219' and 220' in the suppression chamber. The containment isolation valves are at elevations 295' and 298' leaving the drywell, and return at elevation 265'. In the suppression chamber the isolation valves are at elevations 219' and 226' leaving and at elevations 218' and 228' returning. Corresponding inboard and outboard isolation valves are at the same elevations.

The analyzer inlet and outlet valves are in the auxiliary bays (A train in the North Bay, B train in the South Bay) at elevation 249'. The pumps and panels 66A,B and 73A,B are located at elevation 240' in the auxiliary bays (same convention as above).The LOCA override keylock switches are in the Control Room on panels P870 and P875.

System 29

Containment Flood Instrumentation

3.2.1.29 Containment Flood Level Instrumentation

3.2.1.29.1 Function

The purpose of the containment flood instrumentation evolution is to determine primary containment water level when fuel zone instrumentation is not available or is off-scale.

3.2.1.29.2 Success Criteria

Success would be to maintain primary containment water level below the maximum limit as shown on EOP Table C6-1 during a containment flood scenario. In the event that containment flooding is required per EOPs, the water level inside containment will need to be determined per N2-EOP-6, Attachment 23. Determination of water level is accomplished by opening specific valves to pressure transmitters, finding the differential pressure, and then finding the level on Figure 23.1 in Attachment 23. This determination is necessary to prevent exceeding the Maximum Primary Containment Water Level Limit, Figure C6-1, in N2-EOP-C6. This write-up only evaluates the equipment and actions needed for level determination above 224 ft.

A simplified diagram of the system, Figure 3.2.1.29-1, shows the flow path and valve positions required.

3.2.1.29.3 Support Systems

A summary of the systems that are required follows. A more complete listing of the components, including their individual dependencies is in table 3.2.1.29-1. Required systems include:

- Emergency AC Division I and II
- Normal AC source B
- 2VBB-UPS1A
- 125V DC Division I

3.2.1.29.4 System Operation

N2-EOP-6, Att. 23 describes the method of determining containment level. For determination of level between 224 and 298.5 ft, operators first close Nitrogen Makeup Flow Control Valve 2CPS-FV125 (this valve is normally closed).

Containment Purge System (CPS) isolation valves 2CPS*SOV119 and 2CPS*SOV121 must close and remain closed. During normal operation, these valves are closed, and they isolate on a LOCA signal.

CPS isolation valves 2CPS*SOV120 and 2CPS*SOV122 must open to allow pressure to transmitter 2CPS-PT127.

Pressure transmitter 2CMS*PT7A must also remain operational.

The operators read the pressure at each point, find the difference, and locate this value on the chart in N2-EOP-6, Att. 23, page 3. This value is then referenced on EOP Figure C6-1 to verify that maximum water level relative to pressure is not being exceeded.

3.2.1.29.5 Instrumentation and Controls

All necessary equipment has both indication and control in the control room.

3.2.1.29.6 Technical Specifications

With any purge system valve inoperable, isolate affected penetrations within 4 hours, or be in hot shutdown within next 12 hours, and cold shutdown within the following 24 hours.

With one or more excess flow check valves inoperable, isolate the affected instrument and declare inoperable, or be in hot shutdown in 12 hours, and cold shutdown in the next 24 hours.

3.2.1.29.7 Surveillance Testing and Maintenance

<u>Equipment tested quarterly</u>
- 2CPS*SOV119
- 2CPS*SOV120
- 2CPS*SOV121
- 2CPS*SOV122

<u>Equipment tested every 18 months</u>
- 2CMS*PT7A
- 2CMS*EFV5A

<u>Equipment tested on a variable or as required basis</u>
- 2CPS-PT127
- 2CPS-FV125
- 2CPS-V25[1]
- 2CMS*V6A[1]

3.2.1.29.8 References

N2-EOP-C6, Rev. 4.0
N2-EOP-6, Attachment 23, Rev. 0
NMP2 Technical Specifications
PIDs as referenced on simplified drawing

[1] These are manual isolation valves, used only when necessary.

3.2.1.29.8 Initiating Event Potential

The Containment Flood Level Instrumentation does not lead to any potential initiating events.

3.2.1.29.10 Equipment Location

Valves 2CPS*SOV121 and 2CPS*SOV122 are located inside primary containment, all other equipment is located in the reactor building.

3.2.1.29.11 Operating Experience

There were no outstanding operational events relevant to this study. Plant specific component operational data is detailed in section 3.3.2.

3.2.1.29.12 Modeling Assumptions

1. Isolation valves 2CPS*SOV119 and 2CPS*SOV121 are assumed closed initially. These valves are normally closed, close on an isolation signal, and fail closed on loss of power.

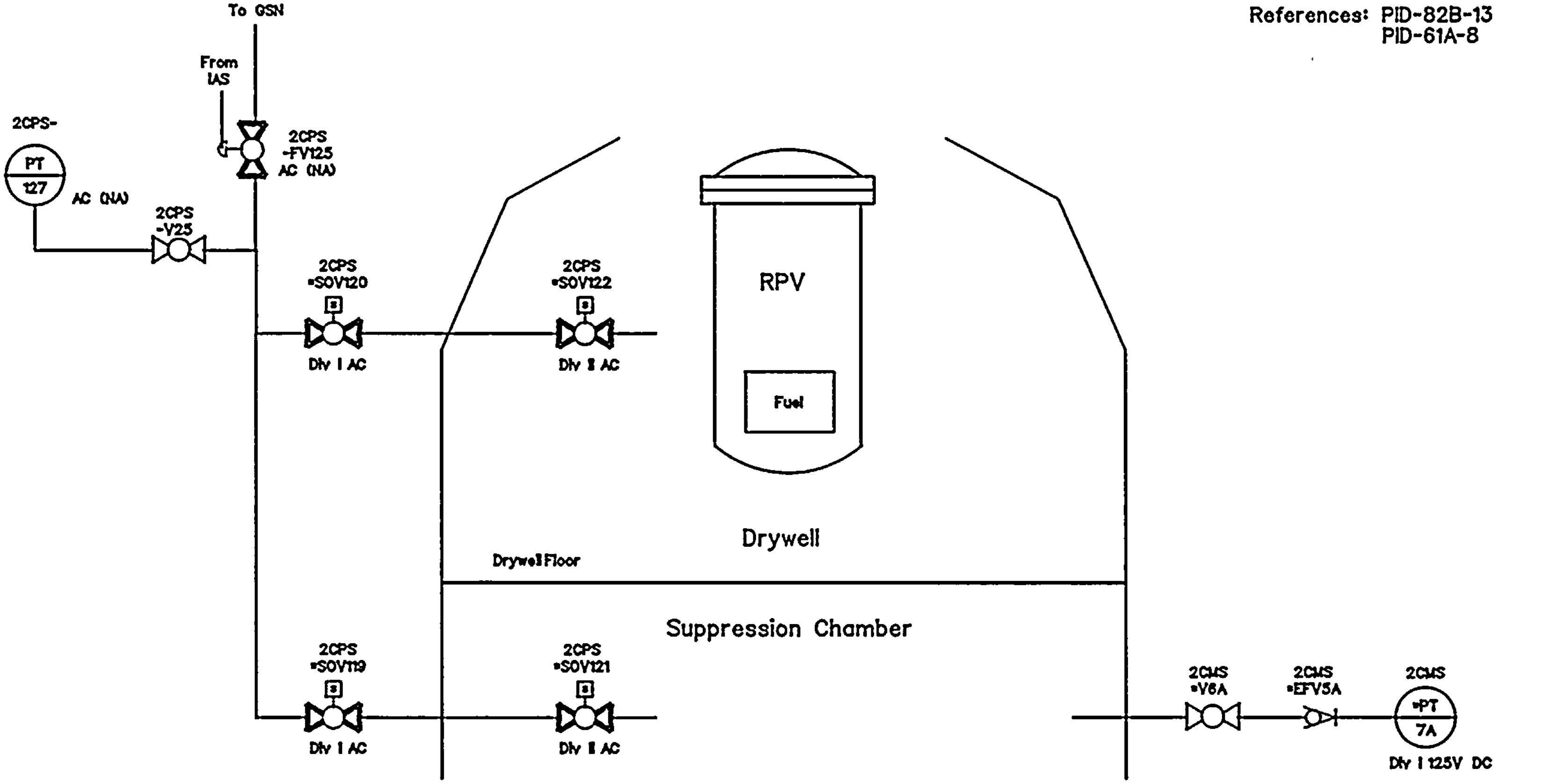
3.2.1.29.13 Logic Model and results

The fault tree(s) is included in Tier 2 documentation, and is available upon request. Quantitative results are summarized in the Level 2 documentation.

CONTAINMENT FLOOD INSTRUMENTATION
Component Block Descriptions

Block	Mark No. (Alt. ID)	Description	Failure Mode	Initial State	Actuated State	Support System	Loss of Support
INSTRUMENTS							
	2CPS-PT127	DW presure transmitter	FAILS	N/A	N/A	2VBB-UPS1A	FAILS
	2CMS*PT7A	SC presure transmitter	FAILS	N/A	N/A	2BYS*SWG002A	FAILS
VALVES							
	2CPS*SOV119	Outboard isolation valve	TRANSFER OPEN	CLOSED	CLOSED	2EJS*US1	CLOSED
	2CPS*SOV120	Outboard isolation valve	FAIL TO OPEN	CLOSED	OPEN	2EJS*US1	CLOSED
	2CPS*SOV120	Outboard isolation valve	TRANSFER CLOSED	OPEN	OPEN	2EJS*US1	CLOSED
	2CPS*SOV121	Inboard isolation valve	TRANSFER OPEN	CLOSED	CLOSED	2EJS*US3	CLOSED
	2CPS*SOV122	Inboard isolation valve	FAIL TO OPEN	CLOSED	OPEN	2EJS*US3	CLOSED
	2CPS*SOV122	Inboard isolation valve	TRANSFER CLOSED	OPEN	OPEN	2EJS*US3	CLOSED
	2CPS-FV125	Flow valve	TRANSFER OPEN	CLOSED	N/A	INSTRUMENT AIR	AS-IS
	2CPS-V25	Manual isolation valve	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CMS*V6A	Manual isolation valve	TRANSFER CLOSED	OPEN	N/A	N/A	N/A
	2CMS*EFV5A	Excess flow check valve	TRANSFER CLOSED	OPEN	N/A	N/A	N/A

Rev 0 (7/92)



Path REQUIRED to be open.

Path REQUIRED to be closed.

Figure 3.2.1.29-1
Containment Flood Instrumentation

I:\NMP2IPE\DGN\2S29.DGN Jul. 06, 1992 13:21:05

Section 3.2.2

Not Used

3.2.2 Not Used

NUREG-1335 suggests that this section be used for system analysis. However, it was deemed more appropriate for NMP2 to include all system information in Section 3.2.1. As such this section is not used. NUREG-1335 also suggests that fault trees could be included here. The large amount of fault tree pages developed for the NMP2 IPE preclude their inclusion here. They have been classified as Tier II documents for the NMP2 IPE and are available on request.

Section 3.2.3

System Dependencies

3.2.3 System Dependencies

Identification of dependencies between systems is one of the most important tasks in developing an accurate model of the plant. The two tables in this section describe important physical and functional dependencies between plant systems. Table 3.2.3-1 documents how support systems depend on other support systems. Table 3.2.3-2 documents how front-line systems depend on support systems. These tables focus on critical dependencies that can impact the successful operation of systems. Non-critical dependencies or interfaces, in general, are not included although they may be described in the system description.

In developing the tables, system divisions or trains are shown to help explain dependencies and only direct dependencies are shown. That is, secondary or cascading dependencies are not shown in the tables, but are included in the logic model. Failure of a system on the left column has an impact on the systems on the top row as explained by the notes in the table. For example, failure of Service Water Train A in Table 3.2.3-1 will fail emergency diesel 2EGS*EG1 and emergency diesel 2EGS*EG2 will depend solely on Service Water Train B. In addition, the normal service water supply to RBCLC and TBCLC will be unavailable, although the "B" train of service water can be manually aligned if available. Note the cascading impact of emergency AC, TBCLC, and RBCLC failures are not shown for the Service Water Train A failure. The impact of these failures are shown for these systems.

Additional descriptions of dependencies are provided in the system descriptions (Section 3.2.1) and the initiating event selection documentation (Section 3.1.1).

Equipment dependencies on ventilation and cooling is described in Section 3.2.1.11.

Table 3.2.3-1 SUPPORT-TO-SUPPORT SYSTEM DEPENDENCIES

	Normal AC	Emergency AC (I)	Emergency AC (II)	Emergency AC (III)	125V DC (I)	125V DC (II)	125V DC (III)	Normal DC	120V AC UPS 2A	120V AC UPS 2B
Normal AC	-	2	3	2				14		
Emergency AC, Div. I		-			11				15	
Emergency AC, Div. II			-			12				16
Emergency AC, Div. III				-			13			
125V DC, Div. I		4			-				15	
125V DC, Div. II			5			-				16
125V DC, Div. III				6			-			
Normal DC	1							-		
120V AC, UPS 2A									-	
120V AC, UPS 2B										-
RRCS, Div. I										
RRCS, Div. II										
ECCS Signal, Div. I										
ECCS Signal, Div. II										
Instrument Air										
Nitrogen										
Service Water "A"		7		9						
Service Water "B"			8	9						
Reactor Building Closed Loop Cooling										
Turbine Building Closed Loop Cooling										

Table 3.2.3-1 SUPPORT-TO-SUPPORT SYSTEM DEPENDENCIES

	RRCS Div. I	RRCS Div. II	ECCS Signal I	ECCS Signal II	Instrument Air	Nitrogen	Service Water A	Service Water B	Reactor Bldg CLC	Turbine Bldg CLC
Normal AC					21	24	26	26	31	35
Emergency AC, Div. I						25	27	10		
Emergency AC, Div. II						25	10	28		
Emergency AC, Div. III										
125V DC, Div. I	17		19				29	10		
125V DC, Div. II		18		20			10	30		
125V DC, Div. III										
Normal DC					21	24			32	32
120V AC, UPS 2A	17		19							
120V AC, UPS 2B		18		20						
RRCS, Div. I	-									
RRCS, Div. II		-								
ECCS Signal, Div. I			-							
ECCS Signal, Div. II				-						
Instrument Air					-	22			33	33
Nitrogen					22	-				
Service Water "A"							-		34	36
Service Water "B"								-	34	36
Reactor Building Closed Loop Cooling					23				-	
Turbine Building Closed Loop Cooling										-

TABLE 3.2.3-1 SUPPORT-TO-SUPPORT SYSTEM DEPENDENCIES

1. The normal AC supplies depend on normal 125v DC power for control and protection.

   - Battery 2BYS-BAT1A provides 125V DC control power to:

     a) The primary protection of transformers 2RTX-XSR1A, 2RTX-XSR1B and 2ABS-XS1

     b) 115KV preferred power source A switchyard motor operated disconnect switches and circuit switchers

     c) Protection of Normal Station Service transformer 2STX-XS1

     d) 345KV switchyard and generator step up transformer

     e) 13.8KV and 4.16KV main supply breakers

   - Battery 2BYS-BAT1B provides 125V DC control power to:

     a) The backup protection of transformers 2RST-XSR1A, 2RST-XSR1B and 2ABS-XS1

     b) 115KV preferred power source B switchyard motor operated disconnect switches and circuit switchers

     c) Protection of 345KV switchyard and generator step up transformer

     d) 13.8KV and 4.16KV feeder supply breakers

   Given a plant trip, 125V normal DC is required to support fast transfer and slow transfer of AC power from the main generator and normal station transformer to the reserve transformers. On slow transfer, normally operating components must be manually restarted.

2. Emergency AC Division I (2ENS*SWG101) and Division III (2ENS*SWG102) are normally supplied from offsite AC power through preferred source A and reserve station transformer 1A (2RTX-XSR1A). On loss of offsite AC power, or preferred source A, an emergency diesel (2EGS*EG1 and 2) must start and load its respective emergency switchgear.

3. Emergency AC Division II (2ENS*SWG103) is normally supplied from offsite AC power through preferred source B, reserve station transformer 1B (2RTX-XSR1B). On loss of offsite AC power or preferred source B, an emergency diesel (2EGS*EG3) must start and load its emergency switchgear.

4. Emergency AC Division I depends on 125V DC Division I for emergency Diesel starting, breaker control and generator control (4.1KV 2ENS*SWG101 and 2EJS*US1 breakers).

5. Emergency AC Division II depends on 125V DC Division II for emergency Diesel starting, breaker control, and generator control (4.1KV 2ENS*SWG103 and 2EJS*US3 breakers).

6. Emergency AC Division III depends on 125V DC Division III for emergency Diesel starting and breaker control (4.1KV 2ENS*SWG102). Once the diesel has started and loaded, 125V DC is only needed for automatic trip functions. The diesel will not trip on loss of 125V DC Division III.

7. Service water Loop A supplies cooling to emergency diesel 2EGS*EG1 and its space cooler 2HVP*UC1A.

8. Service water Loop B supplies cooling to emergency diesel 2EGS*EG3 and its space cooler 2HVP*UC1B.

9. Either service water Loop A or B can supply.cooling to emergency diesel 2EGS*EG2 and its space cooler 2HVP*UC2.

10. Service water Div. I motor operated valve control input signals are generated by loss of power on the opposite Div. II. For example, loss of 120V emergency AC (non-UPS) Div. II will generate an isolation signal to the Div. I MOVs that isolate the reactor building and Turbine building loops. This loss would also generate at least a partial input signal for Div. I MOV50A closure which isolates the service water loops. Also, loss of 125V DC Div. II would generate at least a partial input signal for Div. I MOV50A closure. Div. II MOVs are similar in that there are Div. I interfaces.

11. 125V DC Division I is normally supplied by emergency AC Division I through battery chargers 2BYS*CHG2A1 and 2A2. On loss of the emergency AC supply, the battery (2BYS*BAT2A) is capable of supplying loads for 19.5 hours if load shedding is implemented.

12. 125V DC Division II is normally supplied by emergency AC Division II through battery chargers 2BYS*CHG2B1 and 2B2. On loss of the emergency AC supply, the battery (2BYS*BAT2B) is capable of supplying loads for 19.5 hours if load shedding is implemented.

13. 125V DC Division III is normally supplied by emergency AC Division III through battery chargers 2BYS*CHG2C1 and 2C2. On loss of the emergency AC supply, the battery (2BYS*BAT2C) is capable of supplying loads for 4 hours.

14. Normal DC power is normally supplied by AC power through battery chargers 2BYS-CHGR1A1, 2BYS-CHGR1B1, and 2BYS-CHGR1C1. On loss of normal AC, batteries carry the DC loads.

15. 120V AC vital UPS Division I (2VBA*UPS2A) is normally supplied by emergency AC Division I with backup from 125V DC Division I. Loss of both AC and DC results in failure of 120V AC. On loss of emergency AC, 120V AC is available until the batteries discharge. (See Note 11)

16. 120V AC vital UPS Division II (2VBA*UPS2B) is normally supplied by emergency AC Division II with backup from 125V DC Division II. Loss of both AC and DC results in failure of 120V AC. On loss of emergency AC, 120V AC is available until the batteries discharge. (See Note 12)

17. 120V AC Vital UPS Division I (2VBS*PNL101A from 2VBA*UPS2A) is required for RRCS Division I operation, and 125v DC is required for control logic. RRCS I depends on Normal 125V DC for operation of the LFMG motor and generator breakers 1A, 1B, and 2A. It also depends on Div. I 125V DC for operation of the Recirculation pump motor breakers 3A and 3B. Breakers 3A and 3B each have redundant trip coils, one being tripped via RRCS I and the other via RPS Bus A. Tripping of either trip coil will open the breaker.

18. 120V AC Vital UPS Division II (2VBS*PNL301B from 2VBA*UPS2B) is required for RRCS Division II operation, and 125v DC is required for control logic. RRCS II depends on normal 125V DC for LFMG motor and generator breakers 1A, 1B, and 2B for operation. It also depends on Div. II 125V DC for operation of Recirc. pump motor breakers 4A and 4B. Breakers 4A and 4B each have redundant trip coils, one being tripped via RRCS II and the other via RPS Div. II. Either trip coil is capable of tripping the breaker.

19. Loss of UPS 2VBA*UPS2A (2VBS*PNL101A) results in failure of Division I automatic ECCS actuation. Division II 125V DC is required for manual initiation only. ECCS Signal I depends on 125V DC Div. I for powering relays that provide EDG loading permissives for LPCS and LPCI-A.

20. Loss of UPS 2VBA*UPS2B (2VBS*PNL101B) results in failure of Division II automatic ECCS actuation. Division II 125V DC is required for manual initiation only. ECCS Signal II depends on 125v DC Div. II for powering relays that provide EDG loading permissives for LPCI B&C.

21. Normal AC and DC is required for instrument air compressors. AC for supply power and DC for control and indication.

22. The plant nitrogen system provides nitrogen to the instrument air system loads inside primary containment.

23. A sub-loop of RBCLC provides air compressor cooling. Loss of the RBCLC system would cause failure of the instrument air compressors.

24. Nitrogen trim heaters (GSN-E1A and B) and electric vaporizers (GSN-EV2A through D) are powered by normal AC through 2NHS-MCC016 and 2NJS-US9, respectively.

25. Nitrogen supply solenoid operated containment isolation valves fail closed on loss of 120V emergency AC Div. I or Div. II (Non UPS). These lines supply the SRVs, inboard MSIVs, and the inboard drywell and suppression chamber vent valves. Containment isolation signal and instrument dependencies are described in Table 3.2.3-2.

26. On loss of preferred AC sources, the normally running service water pumps trip, and one must restart on its' respective emergency AC supply after the motor operated discharge valves close. The valves must reopen after pump start.

27. Loop A (Division I) pumps 2SWP*P1A, C, E are powered by emergency AC Division I (2ENS*SWG101).

28. Loop B (Division II) pumps 2SWP*P1B, D, F are powered by emergency AC Division II (2ENS*SGW103).

29. Loop A (Division I) pump control power is from 125V DC Division I (2BYS*SWG002A).

30. Loop B (Division II) pump control power is from 125V DC Division II (2BYS*SWG002B).

31. RBCLC pumps and the booster pumps are powered by normal AC power as shown below. On loss of offsite AC the RBCLC system fails and service water supply to the RBCLC heat exchangers is isolated.

    Booster Pump A 2NNS-SWG013
    Booster Pump B 2NNS-SWG015
    Booster Pump C 2NNS-SWG014
    Pump A 2NNS-SWG012
    Pump B 2NNS-SWG015
    Pump C 2NNS-SWG014

32. Normal DC power is required for pump start (AC power breaker control at switchgear). Some components used to allow services water to cool the RHR seal coolers and spent fuel cooling are safety related and are powered by emergency power sources.

33. Instrument air provides air to RBCLC controls, valves, and indication. All air operated valves will fail resulting in super cooling of certain loads and loss of cooling to other loads.

    a) Make-up fails open resulting in tank overflows to drains

    b) Tempering flow of CCP flow through CCP/SWP heat exchanger fails open resulting in CCP system temperature dropping below design

    There is a similar dependency on loss of IAS to TBCLC system. All air operated valves fail resulting in super cooling of certain loads

    a) Make-up fails open resulting in tank overflow to drains

    b) Full flow to Generator, Lube oil, an EHC coolers

34. Normally, service water Train A supplies the three RBCLC heat exchangers through a common header. Service water to the heat exchangers is isolated on loss of preferred source AC (MOV19A and B). Service water Train B can supply the heat exchangers by opening a locked closed manual valve.

35. TBCLC pumps are powered by the following normal AC switchgear:

    Pump A 2NNS-SWG011
    Pump B 2NNS-SWG013
    Pump C 2NNS-SWG012

36. Normally, service water Train A supplies the three TBCLC heat exchangers through a common header. Service water to the heat exchangers is isolated on loss of preferred source offsite AC (MOV3A and B). Service water Train B can supply the heat exchangers by opening a locked closed valve.

Table 3.2.3-2 SUPPORT-TO-FRONTLINE SYSTEM DEPENDENCIES

					RHR A		RHR B		RHR C				
	RCIC	HPCS	SRV/ADS	LPCS	LPCI	SPC	LPCI	SPC	LPCI	CRD	Fire Water	Conden-sate	Feed-water
Normal AC				10	10	10	10	10	10	19	26	23	23
Emergency AC, Div. I			5	7	11	11							
Emergency AC, Div. II			5				12	12	12				
Emergency AC, Div. III		4											
125V DC, Div. I	1		6	8	13	13							
125V DC, Div. II			6				14	14	14				
125V DC, Div. III		4											
Normal DC										20	27	24	24
120V AC, UPS 2A	2			60	60								
120V AC, UPS 2B							61		61				
RRCS, Div. I													
RRCS, Div. II													
ECCS Signal, Div. I	3		3,6	3	3								
ECCS Signal, Div. II	3		3,6				3		3				
Instrument Air	52				53	53	53	53	53	22		51	54
Nitrogen			5										
Service Water "A"		4			15	15,16							
Service Water "B"		4					17	17,18	17				
Reactor Building Closed Loop Cooling					10	10	10	10	10	21			
Turbine Building Closed Loop Cooling												25	

Table 3.2.3-2 SUPPORT-TO-FRONTLINE SYSTEM DEPENDENCIES

	RPS	Cont. Vent	Cont. Isol.	ATWS					MSIVs	Turbine Bypass	RRP Seal Integrity	Drywell Cooling
				ARI	RPT	FWR	SLS	Man. Rods				
Normal AC	28		32		47	57		42	49	59	58	46
Emergency AC, Div. I		31	33				40		43			46
Emergency AC, Div. II		31	33				40		43			46
Emergency AC, Div. III			33									
125V DC, Div. I	29		34	36	48						58	
125V DC, Div. II	29		34	36	48						58	
125V DC, Div. III												
Normal DC	28		32		48					59	58	
120V AC, UPS 2A												
120V AC, UPS 2B												
RRCS, Div. I				37	38	39	41					
RRCS, Div. II				37	38	39	41					
ECCS Signal, Div. I												
ECCS Signal, Div. II												
Instrument Air	30	50	35				55		56			
Nitrogen		31	35						43			
Service Water "A"												
Service Water "B"												
Reactor Building Closed Loop Cooling											44	45
Turbine Building Closed Loop Cooling												

TABLE 3.2.3-2 SUPPORT-TO-FRONTLINE SYSTEM DEPENDENCIES

1. RCIC depends on 125V DC Division I for initiation and MOV operation (2DMS*MCCA1) and also gland seal air compressor.

2. 120V AC Division I (2VBS*PNL101A) is required for flow control (2ICS*FC101). On loss of 120V AC vital UPS Div. I, RCIC fails to maximum flow.

3. ECCS input actuation to LPCS, ADS, LPCI (RHR), and RCIC depends on operation of common transmitters and trip units as described below:

Transmitters	Trip Units	Systems	Div
LT N091A, E	LIS N691A, E	LPCS, ADS(A), RHR(A)	I
	LIS N692A, E	RCIC	I
LT N091B, F	LIS N691B, F	ADS(B), RHR(B), RHR(C)	II
	LIS N692B, F	RCIC	II
PT N094A	PIS N694A	LPCS, RHR(A), RCIC	I
PT N094E	PIS N694E	LPCS, RHR(A)	I
PT N094B	PIS N694B	RHR(B), RHR(C), RCIC	II
PT N094F	PIS N694F	RHR(B), RHR(C)	II

Level transmitters (LT) are RPV water level
Pressure transmitters (PT) are drywell pressure
Division I, II are UPS2A, UPS2B, respectively

4. HPCS is dependent on Division III Emergency 4KV, 600V AC, 120V AC and 125V DC. Emergency AC is required for pump and MOV operation, and normally supplies 120V AC and 125V DC which are required for control, system initiation, and instrumentation. HPCS depends on Service Water A or B for diesel cooling, diesel room cooling and HPCS pump room cooling.

5. Nitrogen is supplied to the ADS (accumulators) through isolation valves 2IAS*SOV164 and 165.These valves fail closed on loss of non-UPS 120V AC Division I and Division II, respectively.

6. Each of the 7 ADS valves has three solenoids (A, B, and C). Any solenoid can open the valve by admitting nitrogen pressure. Solenoids A and B receive ADS initiation signals from Division I and II, respectively. Failure of both signals would require manual actuation to open the valves. Also, solenoids A and B depend on 125V DC Division I and II (2BYS*SWG002A and B), respectively. Failure of both DC power supplies would fail ADS. Solenoid C operates in the relief mode (all 18 SRVs), requires 125V DC Division I to operate, and utilizes different pressure transmitters.

7. The LPCS pump is powered from Division I emergency AC 2ENS*SWG101 and motor operated valves are powered through 2EHS*MCC102C.

8. The LPCS pump requires Division I 125V DC 2BYS*PNL201A from 2BYS*SWG002A for pump start (breaker control).

9. Later.

10. RBCLC normally provides RHR pump seal cooling with service water as a backup. On loss of RBCLC, service water must be manually aligned to RHR pump seal cooling. On loss of instrument air or 120V AC control power, service water supply valves to RHR seal cooling are opened and RBCLC is isolated.

11. RHR pump A (2RHS*P1A) requires emergency AC Division I power (2ENS*SWG101). Division I emergency AC MCCs (2EHS*MCC102A and 103C) are required to operate MOVs.

12. RHR pump B and C (2RHS*P1B and C) require emergency AC Division II power (2ENS*SWG103). Division II emergency AC MCCs (2EHS*MCC303D) are required to operate MOVs.

13. 125V DC Division I (2BYS*SWG002A) is required to start RHR pump 1A (breaker control).

14. 125V DC Division II (2BYS*SWG002B) is required to start RHR pump 1B and 1C (breaker control).

15. Service water Loop A is required for RHR pump A seal cooling when RBCLC is unavailable.

16. Service water Loop A is required to heat exchanger 2RHS*E1A for suppression pool cooling.

17. Service water Loop B is required for RHR pump B and C seal cooling when RBCLC is unavailable.

18. Service water Loop B is required to heat exchanger 2RHS*E1B for suppression pool cooling.

19. The CRD pumps are normally powered by normal AC power as follows:

Pump 1A	2NNS-SWG014
Pump 1B	2NNS-SWG015

NOTE: In the event of a loss of offsite power without a LOCA, Pump P1A can be powered from the Division I 4Kv Bus via Switchgear 101-11 and 14-1. Pump P1B can be powered from Division II 4KV Bus via Switchgear 103-6 and 15-8.

20. Normal DC power is required for pump start (breaker control).

21. RBCLC is required for CRD pump bearing and seal coolers.

22. On loss of air, the following valves fail as stated:

- Scram inlet and outlet (AOV126 and 127 are typical of 1 HCU) fail open to scram reactor.
- Flow Control Valves (FV6A and 6B) close to approximately 2% open.
- Scram discharge drain and vent valves fail closed.

23. Condensate and feedwater pumps depend on normal AC as follows:

Condensate Pump 1A	2NNS-SWG011
Condensate Pump 1B	2NNS-SWG013
Condensate Pump 1C	2NNS-SWG011 or 13
Condensate Boost Pump 1A	2NPS-SWG001
Condensate Boost Pump 1B	2NPS-SWG003
Condensate Boost Pump 1C	2NPS-SWG001 or 3
Feedwater Pump 1A	2NPS-SWG001
Feedwater Pump 1B	2NPS-SWG003
Feedwater Pump 1C	2NPS-SWG001 or 3

Valves:

2FWS-MOV110	2NHS-MCC003
2FWS-MOV112	2NHS-MCC003
2FWS-MOV17A	2NHS-MCC003
2FWS-MOV17B	2NHS-MCC003
2FWS-MOV17C	2NHS-MCC003
2FWS-MOV22A	2NHS-MCC003
2FWS-MOV22B	2NHS-MCC003
2FWS-MOV22C	2NHS-MCC003
2FWS-MOV47A	2NHS-MCC003
2FWS-MOV47B	2NHS-MCC003
2FWS-MOV47C	2NHS-MCC003
2FWS*MOV21A (Outside IV)	2EHS*MCC102A (I)
2FWS*MOV21B (Outside IV)	2EHS*MCC102C (I)

24. Condensate and feedwater depend on normal DC for pump start.

25. TBCLC is required for condensate and feedwater pump cooling.

26. The motor driven fire water pump (2FPW-P2) is powered by normal AC 2NNS-SWG012. The pressure maintenance pumps (P3A and B) are powered from 2NHS-MCC0015A and B, respectively.

27. The motor driven fire water pump (2FPW-P2) depends on normal 125V DC for pump start.

28. Power to RPS trip channels is from normal UPS 2VBB-UPS3A and 3B. These UPS are supplied from both normal AC and normal DC power. Loss of both normal AC and DC or the UPS's will result in RPS trip actuation ("fail safe"). Loss of a single UPS results in a half scram. In addition, normal AC supplies power to the MG sets (2PRM-MG1A and B) that supply the solenoid operated scram pilot valves. Loss of offsite AC or both MGs results in de-energizing (and opening) of scram pilot valves

29. 125V DC Division I and II (2BYS*SWG002A and 2B) provide power to the backup scram valve solenoids which are energize to actuate. When actuated, either valve blocks the instrument air header -- as air bleeds off the rods insert. Failure of both 125V DC Divisions would fail the backup scram function.

30. Loss of instrument air causes a scram as air bleeds from the scram valves.

31. Nitrogen is required to open the inboard drywell vent valve 2CPS*AOV108. The nitrogen supply through containment isolation valves IAS*SOV168 and 180 will isolate on loss of either non-UPS 120V AC Division I or II, respectively. Nitrogen is required to open the inboard suppression chamber vent valve 2CPS*AOV109. The nitrogen supply through containment isolation valves 2CPS*SOV133 will isolate on loss of non-UPS 120V AC Division II.

32. Power to Containment Isolation System logic is from normal UPS 2VBB-UPS3A and 3B. These UPS are supplied by both normal AC and DC power, as shown below.

    2VBB-UPS3A

    DC Supply: 2BYS-SWG001C
    AC Supply: 2NJS-US1 (2LAT-PNL100) normal
    2NJS-US5 (2NJS-PNL500) alternate

    2VBB-UPS3B

    DC Supply: 2BYS-SWG001B
    AC Supply: 2NJS-US4 (2NJS-PNL402) normal
    2NJS-US6 (2NJS-PNL600) alternate

    Loss of both normal AC and DC or the UPSs will result in closure signals to the isolation valves (i.e., the logic fails safe).

    UPS 3A and 3B alternate AC supplies (2NJS-US5 and 2NJS-US6 respectively) are stub buses that can be supplied by Div. I and II Emergency AC Power, respectively.

33. There are some Containment Isolation Valves that are motor operated and Fail-As-Is on loss of Emergency AC Power (Div. I, II, III).

    In addition, there are solenoid operated valves that fail closed on loss of Emergency AC Power (Div. I, II).

    Upon loss of Emergency AC Power (Div. I, II, III), there are air operated valves that have their air/nitrogen isolated and fail close.

    The TIP System solenoid operated, ball valves apparently fail closed on a loss of 120V AC Power, if the TIP cable has been inserted into containment. It is not clear which train powers these valves.

34. There are Containment Isolation Valves that are solenoid operated and Fail Closed on loss of Div. I or II 125V DC Power. In addition there are DC operated MOVs that Fail-As-Is on loss of Div. I 125V DC Power.

    The TIP Shear (VEX) Valves fail open on a loss of Emergency DC Power (it is not clear which 125V DC train powers the VEX valves).

35. Loss of Station Air Pressure results in the closure of the Outboard AOVs in the Containment Isolation System. Loss of the Service Nitrogen results in the closure of the Inboard AOVs in the Containment Isolation System.

36. 125V DC is required for ARI actuation. 2BYS*PNL202A and 2B supply Division I and II solenoid valves, respectively. Four solenoid valves per division are energized to actuate..

37. Either RRCS Division will actuate ARI. There are four solenoid valves per Division that are actuated.

38. Either RRCS Division will trip both reactor recirc pumps. There are three trip functions as described below:

    - Transfer-high vessel dome pressure trips pumps from their normal power supplies transferring the pumps to the low frequency motor-generator (LFMG) sets.

    - RPT-low low water level opens the pump motor breakers with no transfer to the LFMG.

    - LFMG Trip-high vessel dome pressure trips LFMG power supply breakers to the pumps after a 25 sec delay if the APRM downscale trip has not occurred.

39. Either RRCS Division (high dome pressure) will automatically initiate feedwater run back. RRCS starts a 25 sec timer, after which, if the APRM downscale trip has not actuated to trip the RRCS logic, feedwater run back will be initiated.

40. Emergency AC Division I (2EHS*MCC102C) supplies power to SLC Pump 1A explosive valve 3A, MOV1A and MOV5A and Div. II (2EHS*MCC302D) supplies power to Pump 1B, explosive valve 3B, MOV1B and MOV5B.

41. Either RRCS Division can automatically initiate both SLC trains by starting both pumps, opening both suction MOVs, and causing all squib actuated valves to open. RRCS starts a 98 sec timer, after which, if the APRM downscale trip is not actuated to trip the RRCS logic, SLC will be actuated.

42. Manual rod control requires 120V AC normal UPS power and CRD system.

43. Nitrogen is required to open the inboard MSIVs. The nitrogen supply through containment isolation valves IAS*SOV166 and 184 will isolate on loss of either non-UPS 120V AC Div. I or II, respectively. An accumulator provides backup nitrogen at each MSIV.

44. RBCLC provides normal cooling to the reactor recirc pump seal cooling jacket, motor bearing oil cooling, and motor winding air-water heat exchangers. Service water provides backup to RBCLC. Note: CRD supplies purge water to pump seals.

45. RBCLC is required for drywell cooling. Containment isolation valves supplying RBCLC will close on a LOCA signal. LOCA keylock switches can be used to over ride this signal in the Control room.

46. Drywell unit coolers require normal AC power from 2NHS-MCC011 and 12. Except when LOCA signal is present, stub buses can be connected to the emergency diesels.

47. The reactor recirculation pumps (RRP) are normally supplied AC power from breakers 2EPS*SWG001, 2, 3, and 4 which are supplied by 2NPS-SWG001 and 3. RRPs will trip on loss of normal AC.

48. 125V DC is required to trip and close breakers supplying AC power to the RRPs.

2EPS*SWG001	2DMS*MCCA1
2EPS*SWG002	2DMS*MCCB1
2EPS*SWG003	2DMS*MCCA1
2EPS*SWG004	2DMS*MCCB1
2NPS-SWG001	2BYS-SWG001B
2NPS*SWG003	2BYS-SWG001B

49. The inboard MSIV trip solenoid A is powered from 120V AC RPS UPS B (2VBS*PNLB106) and the inboard trip solenoid B is powered from 120V AC RPS UPS A (2VBS*PNLA106). The outboard MSIV Trip Solenoid A is powered from 120V AC RPS UPS A (2VBS*PNLA105). The outboard MSIV Trip Solenoid is powered from 120V AC RPS UPS B (2VBS*PNLB105).

Loss of both UPS will result in MSIVs closure.

50. On loss of instrument air, 2CPS*AOV110 and 111 fail closed, failing containment vent capability.

51. Steam supply valves to steam jet air ejectors close on loss of instrument air. The result is loss of condenser vacuum (eventually). Condenser makeup valves fail open - level control is lost. The condensate return valves from the hot well fail closed.

52. RCIC steamline drain valves close on loss of instrument air.

53. RHR heat exchanger steam supply control valves close on loss of instrument air. There are motor operated bypass valves, which depend on Emergency AC Div. I, or II.

54. Minimum flow bypass valves open, bypassing feedwater to the condenser, resulting in a loss of feedwater flow.

55. Level <u>indication</u> in SLS tank fails to zero.

56. MSIVs or loss nitrogen close on loss of instrument air.

57. On loss of normal AC, Feedwater flow is terminated because feedwater pumps are de-energized. This results in a FWR success.

58. Purge water is supplied from CRD. Refer to CRD for dependencies.

59. Turbine bypass depends on both 115V AC and normal 125V DC to power the EHC system. Bypass valve operation requires that condenser vacuum be maintained.

60. LPCS and LPCI-A depend on 2BYS*UPS2A for the opening permissive for injection valve. Also it is needed for system process monitoring.

61. LPCI-B and C depend on 2BYS*UPS2B for the opening permissive for their respective injection valve. It is also needed for system process maintaining.

SI
APERTURE
CARD

Also Available On
Aperture Card

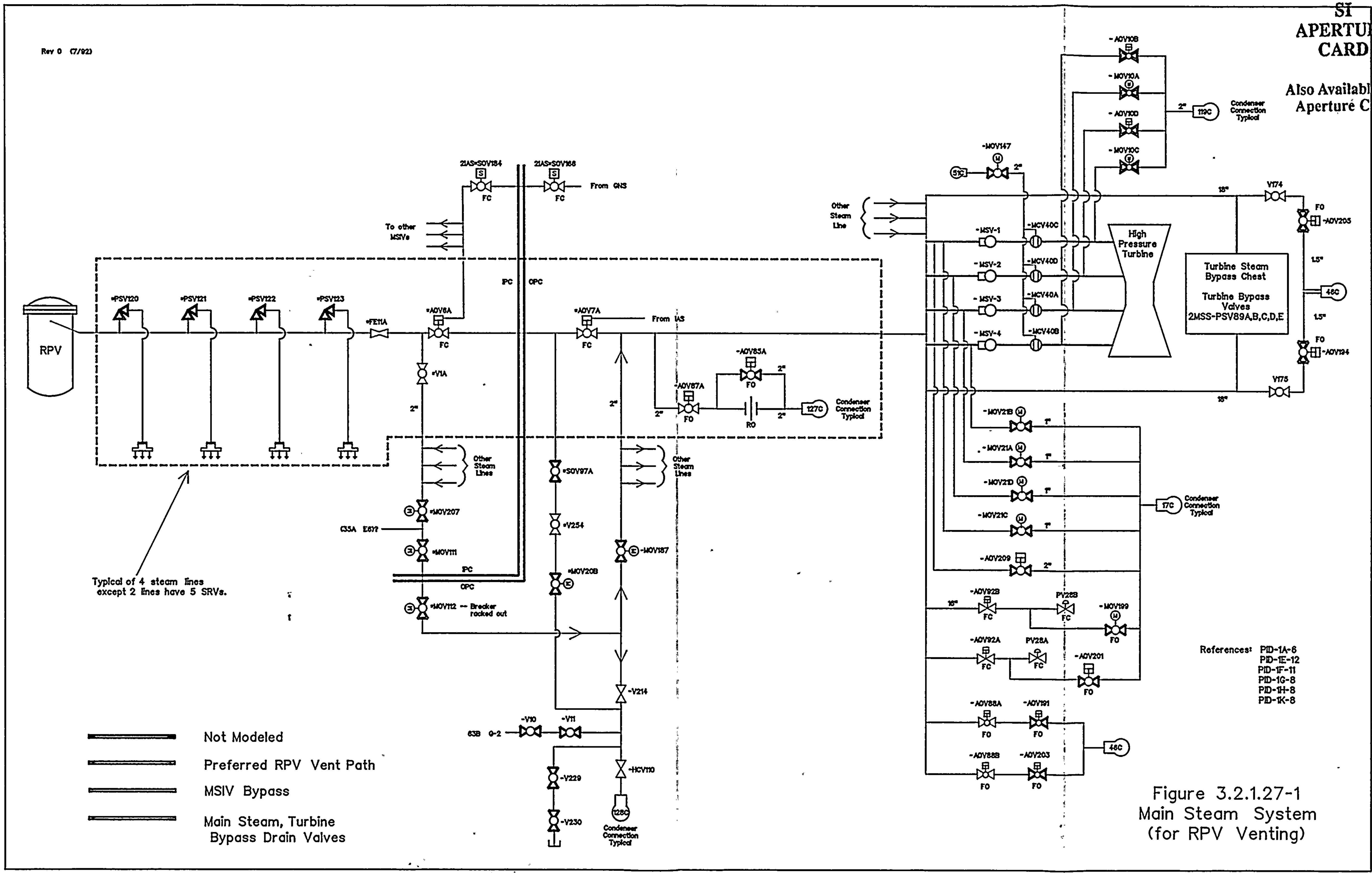


Figure 3.2.1.27-1
Main Steam System
(for RPV Venting)

I:\NMP2IPE\DGN\2S27.DGN Jul. 07, 1992 13:25:45

9208050191 - 35

Section 3.3

Sequence Quantification

3.3 Sequence Quantification

The quantification of event sequences requires initiating event frequencies and top event split fraction probabilities. Initiating event frequencies are quantified as events per year and represent the amount of plant challenges expected. Top event split fractions represent the probability that systems and operators function as required to respond to an initiating event. Split fraction probability is a function of system unavailability due to test and maintenance, equipment failure rate, common cause failure frequency, and human failure rate. This probability is quantified by constructing a fault tree that uses Boolean algebra to develop cutsets that denote system failure combinations. Sequence probability is then quantified by multiplying initiating event frequency and the split fractions representing each failed top event.

The basic building block of the sequence quantification is the basic event. A basic event is the probability of an individual component or operator failure. The basic event probabilities are input to the top event cutsets where they are combined to total the top event split fractions.

The development of basic event probability is based on the statistical analysis of numerous sources of data. These sources can be categorized as expert opinion generic data (Type 1), industry-wide generic data based on plant experience (Type 2), and Nine Mile Point Unit 2 plant-specific data (Type 3). Combining data from these three source types is done using Bayesian analysis. One-stage Bayesian analysis is used when two of the source types have available data and two stage Bayesian analysis is used when all three source types have available information. Tables presented later in this section list exact data values and source for each category and the Bayesian result used for quantification.

Bayesian analysis provides a statistical tool to combine data from various sources. The state of knowledge of each source is represented by a distribution of possible values for a given quantity. In this case, component failure rates are estimated by various sources with corresponding uncertainty regarding the true value. Rather than selecting a given source, the Bayesian methodology is used to arrive at an estimate for each given component failure rate by adequately considering each source in a consistent manner. The approach to Bayesian analysis developed by PLG, inc. is used in this study. [Reference 37] The basic equation of the Bayes method is:

$$P(x|E,E_0) = k^{-1} L(E|x,E_0)\, P(x|E_0) \qquad (1)$$

where:

$P(x \mid E_0)$ is the probability of x being the true value of the unknown quantity based on a state of knowledge represented by E_0, received prior to any new evidence.

$L(E \mid x,E_0)$ is the likelihood of observing new evidence represented by E, assuming that the true value is equal to the prior quantity represented by x.

$P(x \mid E,E_0)$ is the probability of x being the true value of the unknown quantity in light of new evidence E and prior knowledge E_0.

and

K^{-1} is a normalizing factor given by:

$$k=\int_x L(E|x,E_0)\,P(x|E_0)\,dx \tag{2}$$

The distribution $P(x \mid E_0)$ is established by the analyst using a range of possible means and error factors based on values given by available data sources. The likelihood function can assume two different distributions depending on the value being studied. If the component failure mode is measured as failures per unit time, then the likelihood function is considered to follow a Poisson distribution given by:

$$P(k,T|\lambda)=\frac{(\lambda T)^k}{k!}e^{-\lambda T} \tag{3}$$

If the component failure mode is measured as failures per demand, then the likelihood function is considered to follow a binomial distribution given by:

$$P(k,D|\lambda)=\frac{D!}{(D-k)!\,k!}(1-\lambda)^{D-k}\lambda^k \tag{4}$$

However, as the number of component demands become large, the Binomial distribution can be represented by the Poisson distribution [Reference 38] such that:

$$\lambda_D D=\lambda_T T \tag{5}$$

where λ_D is the failure rate per demand and λ_T is the failure rate per unit time. This simplifies the methodology such that the Poisson model can be used for per demand and per unit time cases.

Type 1 generic data is assumed to be normally distributed. The likelihood function for multiple sources of expert opinion is:

$$L(\lambda_1,\ldots,\lambda_N|\lambda)=\prod_{I=1}^{N}\left\{\frac{1}{\sqrt{2\pi}\sigma_I}\exp\left[-\frac{1}{2}\left(\frac{\lambda_I-(\lambda_t+b_I)}{\sigma_I}\right)^2\right]\right\} \tag{6}$$

where b_i is the bias assigned to the source by the analyst. The bias term represents the analyst's confidence in the source being representative of the failure mode under study. λ_t + b_i is the mean corrected for the ith source bias and σ_i is the variance of the ith source. For this study the bias term is typically zero indicating no bias toward individual Type 1 sources of data.

For Type 2 data from N available sources of plant data the likelihood equation is:

$$p(I_1|\Theta,I_0)=\prod_{I=1}^{N}\int_0^{\infty}\phi(\lambda|\theta)\,\frac{(\lambda T_I)^{K_I}}{K_I!}\exp(-\lambda T_I)\,d\lambda \tag{7}$$

where K_i represents the number of failures at plant i and T_i represents the time period or number of demands at plant i. $\phi(\lambda\,|\,\theta)$ is assumed lognormal and given by:

$$\phi(\lambda|\mu,\sigma)=\frac{1}{\sqrt{2\pi}\sigma\lambda}\exp\left\{-\frac{1}{2}\left(\frac{\ln\lambda-\ln\mu}{\sigma}\right)^2\right\} \tag{8}$$

When Type 1 data is available then equation 6 would be inserted in equations 1 and 2. When Type 2 data is available then equation 7 would be inserted in equations 1 and 2. When Type 1 and Type 2 data are available then the individual likelihood equations are combined as:

$$L(I_1|\Theta,I_0)=\prod_{I=1}^{N}P_I(k_I,T_I|\Theta,I_0) \tag{9}$$

For this case equation 9 would then be inserted in equations 1 and 2.

This process forms the first stage of the two stage Bayesian method. Equation 1 is used again for the second stage where the posterior of the first stage is taken as the prior and plant-specific data is used to develop the second stage posterior. The term $P(x\,|\,E,E_0)$ in equation 1, representing the result of the first stage is substituted as the term $P(x\,|\,E_0)$ in the second stage use of Equation 1.

The likelihood equations representing plant-specific data are given by the Poisson distribution, for failures per unit time, and the Binomial distribution, for failures per demand.The Poisson likelihood is given by:

$$P(k, T|\lambda) = \frac{(\lambda T)^k}{k!} e^{-\lambda T} \tag{10}$$

where k is the number of NMP2 failures and T is the exposure hours over which the failures occurred. The Binomial likelihood is given by:

$$P(k, D|\lambda) = \frac{D!}{(D-k)!k!} (1-\lambda)^{D-k} \lambda^k \tag{11}$$

where k is the number of NMP2 failures and D is the number of demands over which the failures occurred. However, when the number of demands is large the Poisson can be substituted for the Binomial as above.

The plant-specific likelihood is then substituted into the second stage use of Equation 1 and the posterior represents the value used for quantification of system unavailability.

Section 3.3.1

List of Generic Data

3.3.1 List of Generic Data

The following two areas are presented in this section:

- Component Failure Rate,
- Component Maintenance Frequency and Duration,

The generic database is primarily based on data developed by PLG and documented in a proprietary database [Reference 39] and the Nuclear Computerized Library for Assessing Reactor Reliability (NUCLARR) [Reference 40]. The PLG generic database is based on and evolved from PRAs performed by PLG. In addition, the database is based on data collected from U.S. reliability data sources. The PLG database provides the basis for Type 1 generic data. NUCLARR is a database developed by EG&G Idaho, Inc. that compiles actual component failure records in the nuclear industry. NUCLARR provides the basis for Type 2 generic data in the form of total failures over unit time or demands for representative data records. The Type 1 and Type 2 data is combined as discussed above to form the NMP2 generic database.

This generic database is updated using NMP2 plant specific data as described in Section 3.3.2.

3.3.1.1 Component Failure Rates

Table 3.3.1-1 lists the generic component failure rates and the main characteristics of the generic distribution. This table also identifies the database designator that is used in quantifying systems. The source column shows the references used for each component. As discussed above, PLG and NUCLARR are the primary sources. However, each source lacks information for a few components. In this case the generic data value is based on the source with available information.

3.3.1.2 Component Maintenance Frequency and Duration

The probability that a component or system is unavailable due to maintenance when called upon is an important contributor to overall system unavailability during an event. The PLG database is used for determining generic maintenance unavailability. This unavailability is then updated as above with plant specific data. Table 3.3.1-2 shows generic maintenance frequency and duration by component.

3.3.1.3 Internal Caused Initiating Event Frequencies

The development of initiating event frequencies is described in Section 3.1.1. Loss of support system initiating event frequencies and internal flood initiating event frequencies are presented in Sections 3.3.5 and 3.3.8, respectively.

3.3.1.4 Phenomenological Events Database

For events in the IPE model that were of a phenomenological nature or otherwise were impractical to measure using data, an analysis was performed to determine appropriate failure probabilities. In many cases expert judgement was used along with reference to an appropriate source on the matter. Table 3.1-3 lists these events and includes the failure distribution parameters along with a description of the analysis performed.

Table 3.3.1-1
Generic Component Failure Rate Distributions

Database Variable	Component and Failure Mode	Source	Mean	Percentile 5th	Percentile Median	Percentile 95th
AAZB1	Press. Liquid Storage Tank Ruptures	1,2	5.15E-7	1.06E-8	1.73E-7	1.65E-6
ABZB1	Unpress. Liquid Storage Tank Ruptures	1,2	5.15E-7	1.06E-8	1.73E-7	1.65E-6
ACZB1	Gas Storage Tank Ruptures	1,2	6.54E-7	1.05E-8	1.13E-7	1.99E-6
BAZL1	Battery Charger Fails During Operation	1,2	1.87E-5	1.09E-6	8.69E-6	4.87E-5
BBZD1	Battery Fails on Demand	1,2	6.58E-4	1.33E-5	1.59E-4	1.63E-3
BBZL1	Battery Fails During Operation	1,2	2.93E-6	2.75E-8	8.64E-7	6.10E-6
CAZN1	Circuit Breaker Transfers Open	1,2	5.03E-7	1.36E-7	5.32E-7	8.29E-7
CAZO1	Circuit Breaker Fails to Open	1,2	1.34E-3	1.09E-4	7.06E-4	3.06E-3
CAZP1	Circuit Breaker Fails to Close	1,2	1.30E-3	1.27E-4	7.15E-4	2.59E-3
DGZE1	Air Dryer Plugs During Operation	1,2	6.12E-5	2.81E-7	9.99E-6	2.12E-4
EBZL1	Electrical Bus Fails During Operation	1,2	5.11E-7	3.65E-8	3.12E-7	1.09E-6
EUZD1	UPS Fails to Transfer on Demand	ISZD1 Value Used				
FCZR1	Air Compressor Fails During Operation	1,2	9.39E-5	4.48E-6	3.15E-5	2.06E-4
FCZS1	Air Compressor Fails to Start	1,2	3.65E-3	1.61E-4	1.58E-3	1.01E-2
FFZR1	Ventilation Fan Fails to Run	1,2	5.14E-5	1.15E-7	2.51E-6	2.02E-4
FFZS1	Ventilation Fan Fails to Start on Demand	1,2	1.11E-3	2.42E-6	1.80E-4	2.79E-3
GAZR1	Div I,II EDG Fails to Run Dur First Hr.	1,2	3.19E-3	1.49E-4	5.95E-4	1.21E-2

Table 3.3.1-1
Generic Component Failure Rate Distributions

Database Variable	Component and Failure Mode	Source	Mean	Percentile 5th	Percentile Median	Percentile 95th
GAZR2	Div I,II EDG Fails to Run After First Hr.	1,2	3.25E-3	1.41E-4	1.13E-3	6.86E-3
GAZS1	Div I,II EDG Fails to Start	1,2	1.10E-2	1.11E-3	8.01E-3	1.73E-2
GBZR1	Div III EDG Fails to Run Dur First Hr.	1,2	3.19E-3	1.49E-4	5.95E-4	1.21E-2
GBZR2	Div III EDG Fails to Run After First Hr.	1,2	3.25E-3	1.41E-4	1.13E-3	6.86E-3
GBZS1	Div III EDG Fails to Start	1,2	1.10E-2	1.11E-3	8.01E-3	1.73E-2
HCZL1	Heat Exchanger Ruptures/Plugs Dur Oper	1,2	1.72E-6	4.54E-8	7.15E-7	5.97E-6
IBZD1	Bistable Fails on Demand	1,2	1.06E-6	4.65E-8	4.45E-7	2.13E-6
ISZD1	Switch Fails on Demand	1,2	1.4E-6	6.63E-8	5.81E-7	1.12E-6
ITZL1	Transmitter Fails During Operation	1,2	2.77E-6	4.16E-7	1.35E-6	6.26E-6
IWZL1	Sensor Fails During Operation	1,2	1.63E-6	5.50E-8	3.16E-7	2.20E-6
KSZE1	Strainer Plugs During During Operation	1,2	8.74E-6	3.86E-7	2.09E-6	1.47E-5
LEZL1	Heater Fails During Operation	2	7.63E-6	9.85E-7	6.94E-6	9.89E-6
PCZR1	Standby Centrifugal Pump Fails Dur Oper	1,2	4.95E-5	5.68E-6	2.52E-5	1.35E-4
PCZS1	Standby Centrifugal Pump Fails to Start	1,2	5.22E-3	4.87E-4	2.08E-3	1.03E-2
PDZR1	Norm Oper Centrifugal Pump Fails Dur Oper	1,2	3.70E-5	3.18E-6	1.67E-5	1.06E-4
PDZS1	Norm Oper Centrifugal Pump Fails to Start	1,2	3.54E-3	2.19E-4	1.75E-3	7.88E-3
PRZR1	Standby Reciprocating Pump Fails Dur Oper	PCZR1 Value Used				

Table 3.3.1-1
Generic Component Failure Rate Distributions

Database Variable	Component and Failure Mode	Source	Mean	Percentile 5th	Percentile Median	Percentile 95th
PRZS1	Standby Reciprocating Pump Fails to Start	PCZS1 Value Used				
PTZR1	Turbine Driven Pump Fails Dur Oper (RCIC)	1,2	3.53E-3	4.10E-5	8.71E-4	9.08E-3
PTZS1	Turbine Driven Pump Fails to Start (RCIC)	1,2	4.27E-2	1.38E-3	2.11E-2	1.15E-1
RAZD1	Relay Fails on Demand	1	2.08E-4	1.05E-5	4.23E-5	7.76E-4
TBZL1	Transformer Fails During Operation	1,2	1.06E-6	1.14E-7	3.99E-7	2.57E-6
VAZD1	Air Oper Valve Fails on Demand	1,2	2.78E-3	7.07E-5	3.16E-4	3.87E-3
VAZM1	Air Oper Valve Transfers Closed	1,2	4.30E-7	1.18E-8	1.06E-7	1.66E-6
VCZM1	Check Valve Transfers Closed	1,2	3.05E-8	5.87E-10	8.72E-9	7.78E-8
VCZO1	Check Valve Fails to Open on Demand	1,2	2.07E-4	5.52E-6	8.09E-5	4.69E-4
VHZM1	Manual Valve Transers Closed	1,2	1.16E-7	6.58E-10	2.19E-8	2.68E-7
VMZD1	Motor Oper Valve Fails on Demand	1,2	5.49E-3	6.06E-4	2.52E-3	1.48E-2
VMZM1	Motor Oper Valve Transfers Closed	1,2	2.21E-7	6.59E-9	5.52E-8	6.13E-7
VPZN1	Safety/Relief (SRV) Sticks Open After Demand	1,2	3.16E-3	6.48E-5	7.28E-4	7.93E-3
VPZO1	Safety/Relief (SRV) Fails to Open on Dem	1,2	8.21E-3	7.01E-4	6.19E-3	1.55E-2
VRZN1	Relief Valve Transfers Open	1,2	4.26E-6	4.08E-7	7.42E-6	1.04E-5
VSZD1	Soleniod Oper Valve Fails to Open On Dem	1,2	5.84E-3	9.33E-5	1.58E-3	2.02E-2
VSZM1	Solenoid Oper Valve Transfers Closed	1,2	2.14E-6	3.81E-7	1.29E-6	2.64E-6

Table 3.3.1-1
Generic Component Failure Rate Distributions

Database Variable	Component and Failure Mode	Source	Mean	Percentile		
				5th	Median	95th
VTZM1	Temp Control Valve Transfers Closed	1,2	1.27E-7	7.30E-10	2.40E-8	2.74E-7
VUZM1	Pressure Control Valve Transfers Closed	VTZM1 Value Used				
VXZO1	Explosive Valve Fails to Open on Demand	3	2.77E-3	6.00E-5	9.42E-4	6.41E-3
WJZB1	Expansion Joint Ruptures	1,4	1.33E-7	7.70E-9	4.86E-8	8.94E-8

Notes on Table 3.3.1-1

Sources

1) PLG [Reference 39]
2) NUCLARR [Reference 40]
3) Peach Bottom Unit 2 PRA NSAC 152 October 1991
4) IEEE STD-500

Table 3.3.1-2
Generic Maintenance Unavailability

Database Variable	Component or System	Mean	Percentile 5th	Percentile Median	Percentile 95th
		Maintenance Unavailability			
BAZMU	Battery Charger	1.57E-4	3.20E-6	6.66E-5	1.29E-3
BCAMU	Battery	1.57E-4	3.20E-6	6.66E-5	1.29E-3
CCPMU	RBCLC Standby Pump Train	8.89E-2	1.89E-4	1.45E-2	9.0E-1
CCSMU	TBCLC Standby Pump Train	8.89E-2	1.89E-4	1.45E-2	9.0E-1
CSHMU	High Pressure Core Spray	4.37E-3	7.11E-5	1.62E-3	4.45E-2
CSLMU	Low Pressure Core Spray	3.36E-3	2.85E-5	1.02E-3	3.37E-2
FASMU	Air Compressor	3.25E-3	1.64E-5	6.57E-4	2.41E-2
GAZMU	Div I,II Diesel Generator	1.35E-2	1.32E-4	3.59E-3	8.60E-2
GBZMU	Div III Diesel Generator	1.35E-2	1.32E-4	3.59E-3	8.60E-2
ICSMU	Reactor Core Isolation Cooling	4.37E-3	7.11E-5	1.62E-3	4.45E-2
RHCMU	Low Pressure Coolant Injection (RHR C)	3.36E-3	2.85E-5	1.02E-3	3.37E-2
RHSMU	Residual Heat Removal A,B	3.36E-3	2.85E-5	1.02E-3	3.37E-2
SLSMU	Standby Liquid Control	8.7E-4	9.55E-6	2.44E-4	5.95E-3
SWPMU	Standby Service Water Pump Train	1.79E-3	1.50E-5	4.55E-4	1.19E-1
VZDMU	Safety/Relief Valve (SRV)	5.17E-4	9.43E-6	2.48E-4	4.37E-3

Table 3.3.1-3
Data Base for Phenomenologic Events

Variable (Event Designation)	Description	Distribution Parameter		Basis
		Mean & Std Deviation	Median & Error Factor	
ZZZ01	SPURIOUS ADS INITIATION	μ=1.0E-5 σ=4.59E-5		Engineering judgement based on information published in previously performed BWR PRAs
ZZZ02 (XXXZZZENVIRONM02)	(X1) ENVIRONMENTALLY INDUCED FAILURE OF SRVS, t=2-8 hrs		M=1.0E-2 EF=10	Engineering judgement based on information contained in NMP2 SBO Study (GENE-770-04-1290)
ZZZ03 (XXXZZZBLEEDDOWN03)	(X1) INVENTORY IN N_2 ACCUMULATORS BLEEDS DOWN, t=2-8 hrs		M=1.0E-2 EF=5	Engineering judgement based on information contained in NMP2 SBO Study (GENE-770-04-1290)
ZZZ04 (XXXZZZENVIRONM04)	(X1) ENVIRONMENTALLY INDUCED FAILURE OF SRVS, t=8-10 hrs		M=5.0E-2 EF=10	Engineering judgement based on information contained in NMP2 SBO Study (GENE-770-04-1290)
ZZZ05 (XXXZZZBLEEDDOWN05)	(X2) INVENTORY IN N_2 ACCUMULATORS BLEEDS DOWN, t=8-10 hrs		M=1.0E-1 EF=5	Engineering judgement based on information contained in NMP2 SBO Study (GENE-770-04-1290)
ZZZ06 (XXXZZZENVIRONM06)	(X3) ENVIRONMENTALLY INDUCED FAILURE OF SRVS, t=10-19 hrs		M=2.0E-1 EF=10	Engineering judgement based on information contained in NMP2 SBO Study (GENE-770-04-1290)
ZZZ07 (XXXZZZBLEEDDOWN07)	(X3) INVENTORY IN N_2 ACCUMULATORS BLEEDS DOWN, t=10-19 hrs		M=5.0E-1 EF=5	Engineering judgement based on information contained in NMP2 SBO Study (GENE-770-04-1290)
ZZZ08	CONDENSATE - FEEDWATER SYSTEM FAILURE		M=1.4E-4 EF=3	Engineering judgement based on estimaing unavailability of system from LOF initiating event frequency

Table 3.3.1-3
Data Base for Phenomenologic Events

Variable (Event Designation)	Description	Distribution Parameter		Basis
		Mean & Std Deviation	Median & Error Factor	
ZZZ09	LOSS OF CONDENSER AND/OR SUPPORT EQUIPMENT		M=1.1E-3 EF=3	Engineering judgement based on estimaing unavailability of system from LOC initiating event frequency
ZZZ18 (ICSZZZU2PRESS018)	(U2) INSUFFICIENT STEAM INLET PRESSURE TO RCIC TURBINE, t=2-8 hrs		M=1.0E-3 EF=5	Engineering judgement based on MAAP calculations
ZZZ19 (ICSZZZU2BATTER19)	(U2) RCIC FAILS, BATTERY DEPLETION, t=2-8 hrs		M=1.0E-2 EF=10	Engineering judgement based on information contained in NMP2 SBO Study (GENE-770-04-1290)
ZZZ21 (ICSZZZU2EQUIP021)	(U2) EXTREME ENVIRONMENT FAILS RCIC EQUIPEMENT, t=2-8 hrs		M=1.0E-2 EF=10	Engineering judgement based on plant specific room heat-up calculation
ZZZ23 (ICSZZZU3PRESS023)	(U3) INSUFFICIENT STEAM INLET PRESSURE TO RCIC TURBINE, t=8-10 hrs		M=5.0E-3 EF=5	Engineering judgement based on MAAP calculations
ZZZ24 (ICSZZZU3BATTER24)	(U3) RCIC FAILS, BATTERY DEPLETION, t=8-10 hrs		M=5.0E-2 EF=10	Engineering judgement based on information contained in NMP2 SBO Study (GENE-770-04-1290)
ZZZ26 (ICSZZZU3EQUIP026)	(U3) EXTREME ENVIRONMENT FAILS RCIC EQUIPMENT, t=8-10 hrs		M=5.0E-2 EF=10	Engineering judgement based on plant specific room heat-up calculation
ZZZ27 (FPWWZZZFLOW00027)	(S1) INADEQUATE FIRE WATER FLOW TO RPV, t=2-8 hrs		M=2.0E-1 EF=10	Engineering judgement based on preliminary assessment of system capability
ZZZ29 (FPWWZZZFLOW00029)	(S2) INADEQUATE FIRE WATER FLOW TO RPV, t=8-10 hrs		M=1.0E-1 EF=10	Engineering judgement based on preliminary assessment of system capability

Table 3.3.1-3
Data Base for Phenomenologic Events

Variable (Event Designation)	Description	Distribution Parameter		Basis
		Mean & Std Deviation	Median & Error Factor	
ZZZ31 (FPWWZZZFLOW00031)	(S3) INADEQUATE FIRE WATER FLOW TO RPV, t=10-19 hrs		M=1.0E-1 EF=10	Engineering judgement based on preliminary assessment of system capability
ZZZ33 (ICSZZZICSSTALL33)	(IL) RCIC STALLS DUE TO OPERATOR FULLY DEPRESSURIZING THE RPV		M=3.0E-1 EF=5	Engineering judgement assuming RPV pressure remains high during the ATWS
ZZZ34	(IS) PATH THROUGH Z48 OPEN	μ=1.0E-1 σ=5.0E-2		Table 3.2.1.10-1, Note 7
ZZZ35	(IS) PATH THROUGH Z49 OPEN	μ=1.0E-1 σ=5.0E-2		Table 3.2.1.10-1, Note 7
ZZZ36	(IS) PATH THROUGH Z50 OPEN	μ=1.0E-1 σ=5.0E-2		Table 3.2.1.10-1, Note 7
ZZZ37	(IS) PATH THROUGH Z51 OPEN	μ=1.0E-1 σ=5.0E-2		Table 3.2.1.10-1, Note 7
ZZZ38	(IS) PATH THROUGH Z58 OPEN	μ=1.0E-1 σ=5.0E-2		Table 3.2.1.10-1, Note 7
ZZZ39	(IS) PATH THROUGH Z59 OPEN	μ=1.0E-1 σ=5.0E-2		Table 3.2.1.10-1, Note 7
ZZZ40	(IS) SMALL PRE-EXISTING LEAK	μ=5.0E-3 σ=2.5E-3		Level 2, Vol. 2 (Section C.2), and PNL study for NRC
ZZZ41	(IS) LARGE PRE-EXISTING LEAK	μ=1.0E-4 σ=4.0E-4		Level 2, Vol. 2 (Section C.2), and PNL study for NRC
ZZZ42	(QM, RQ) MECHANICAL FAILURE OF AUTO-SCRAM FUNCTION	μ=4.3E-6 σ=2.0E-5		Section 3.2.1.15

Table 3.3.1-3
Data Base for Phenomenologic Events

Variable (Event Designation)	Description	Distribution Parameter		Basis
		Mean & Std Deviation	Median & Error Factor	
ZZZ43	(QE, RQ) ELECTRICAL FAILURE OF AUTOMATIC SCRAM FUNCTION	μ=2.6E-5 σ=1.0E-4		Section 3.2.1.15
ZZZ45 (CFXZZZOOCRDOOO45)	(CF) CRD SYSTEM UNAVAILABLE		M=1.0E-3 EF=3	Engineering judgement (Section 3.2.1.24)
ZZZ48 (CFXZZZHSFAILAX48)	(CF) HPCS FAILS DUE TO SMALL CONTAINMENT FAILURE		M=1.0E-1 EF=5	Engineering judgement (Section 3.2.1.24)
ZZZ49 (CFOZZZCRDFAILA49)	(CF) CRD FAILS DUE TO SMALL UPPER DRYWELL FAILURE		M=1.0E-2 EF=5	Engineering judgement (Section 3.2.1.24)
ZZZ50 (CFOZZZFWFAILAX50)	(CF) FEEDWATER FAILS DUE TO SMALL CONTAINMENT FAILURE		M=2.0E-2 EF=5	Engineering judgement (Section 3.2.1.24)
ZZZ52 (CFOZZZCRDFAILB52)	(CF) CRD FAILS DUE TO LARGE UPPER DRYWELL FAILURE		M=1.0E-1 EF=5	Engineering judgement (Section 3.2.1.24)
ZZZ53 (CFOZZZFWFAILBX53)	(CF) FEEDWATER FAILS DUE TO LARGE CONTAINMENT FAILURE		M=1.0E-1 EF=5	Engineering judgement (Section 3.2.1.24)
ZZZ54 (CFXZZZHSFAILBX54)	(CF) HPCS FAILS DUE TO LARGE CONTAINMENT FAILURE		M=5.0E-1 EF=5	Engineering judgement (Section 3.2.1.24)
ZZZ56	(WL) IL=S, WATER LEVEL INDUCES CORE DAMAGE		M=1.0E-3 EF=30	Engineering judgement based on suspected irratic fuel zone RPV water level indication as level approaches TAF during an ATWS
ZZZ57	(WL) IL=F, WATER LEVEL INDUCES CORE DAMAGE		M=1.0E-2 EF=30	Engineering judgement based on suspected irratic fuel zone RPV water level indication as level approaches MSC during an ATWS

Table 3.3.1-3
Data Base for Phenomenologic Events

Variable (Event Designation)	Description	Distribution Parameter		Basis
		Mean & Std Deviation	Median & Error Factor	
BCAL3 (BYSBCABAT2AB00L3)	(X1) BATTERY DEPLETION DURING t=2-8 hrs		M=5.0E-3 EF=10	Engineering judgement based on information contained in NMP2 SBO Study (GENE-770-04-1290)
BCAL4 (BYSBCABAT2AB00L3)	(X1) BATTERY DEPLETION DURING t=8-10 hrs		M=1.0E-2 EF=10	Engineering judgement based on information contained in NMP2 SBO Study (GENE-770-04-1290)
BCAL5 (BYSBCABAT2AB00L3)	(X1) BATTERY DEPLETION DURING t=10-19 hrs		M=1.0E-2 EF=10	Engineering judgement based on information contained in NMP2 SBO Study (GENE-770-04-1290)

Section 3.3.2

Plant-Specific Data and Analysis

3.3.2 Plant Specific Data and Analysis

Consideration of plant specific equipment failure and maintenance unavailability data is necessary to accurately calculate risk measures (eg. Core Melt frequency) for NMP2. The NMP2 Inservice Testing (IST) database was used for standby equipment and failure modes classified as failure per demand. This database represents the periodic testing of various plant equipment. The IST database tabulates the results of each test performed. For the components included in the IPE an aggregate number of test failures and tests was taken and is tabulated below. The plant specific failure rate is given by the number of failures divided by the total number of tests.

In general, the types of components tested are standby pumps, emergency diesel generators, and valves that must change state to perform a safety function. Pump success is measured by obtaining a given flow-rate and discharge pressure in a specified time following a start signal. Emergency diesel generator success is measure by obtaining rated voltage and frequency in a specified time following a start signal. Valve success is measured by the time for an entire change of state to occur (Stroketime).

For the remainder of plant equipment the plant maintenance history was reviewed. The most effective means for this was by using the NMP2 Nuclear Plant Reliability Data System (NPRDS) administered by the Institute of Nuclear Plant Operations (INPO). This database consists of information on equipment failures that has been screened by trained plant staff. This ensures consistency of the information and results in a more focused set of data records. This resulted in a much more efficient IPE data collection task than would be expected with raw, unscreened plant data. NPRDS searches for individual components were reviewed by the IPE team and failures representative of system models were tabulated along with the total operating history of the component population.

For maintenance unavailability the INPO Quarterly Performance Indicator Data report for NMP2 was used. This provided unavailable hours for emergency diesel generators and ECCS systems. For the remainder of system unavailability, plant status logs were used.

Table 3.3.2-1 lists the plant specific failure records for each component and Table 3.3.2-2 lists plant maintenance unavailability. Some variables have their value fields marked "N/A" to denote that information was either unavailable or otherwise not collected by the IPE Team. Table 3.3.2-3 shows the Bayesian updated failure distribution for each component. The values in this table are used in system quantification. For variables with one or two "N/A" fields in Table 3.3.2-1, the values in Table 3.3.2-3 are simply generic values indicating that no plant specific updating occurred. Table 3.3.2-4 shows Bayesian updated maintenance unavailability. These values are also used in system quantification.

Table 3.3.2-1
NMP2 Plant Specific Failures by Component

Database Variable	Component and Failure Mode	Number of Failures	Total Comp Demands or Cum. Oper Time
AAZB1	Press. Liquid Storage Tank Ruptures	0	4.68E+4 hr
ABZB1	Unpress. Liquid Storage Tank Ruptures	0	1.42E+5 hr
ACZB1	Gas Storage Tank Ruptures	0	1.42E+5 hr
BAZL1	Battery Charger Fails During Operation	0	1.71E+5 hr
BBZD1	Battery Fails on Demand	0	N/A
BBZL1	Battery Fails During Operation	0	1.39E+5 hr
CAZN1	Circuit Breaker Transfers Open	3	1.38E+7 hr
CAZO1	Circuit Breaker Fails to Open	1	N/A
CAZN1	Circuit Breaker Fails to Close	11	N/A
DGZE1	Air Dryer Plugs During Operation	0	5.54E+4 hr
EBZL1	Electrical Bus Fails During Operation	0	1.05E+5 hr
EUZD1	UPS Fails to Transfer on Demand	6	N/A
FCZR1	Air Compressor Fails During Operation	2	2.96E+4 hr
FCZS1	Air Compressor Fails to Start	1	N/A
FFZR1	Ventilation Fan Fails to Run	N/A	N/A
FFZS1	Ventilation Fan Fails to Start	N/A	N/A
GAZR1	Div I,II EDG Fails to Run During First Hr.	5	258 hr
GAZR2	Div I,II EDG Fails to Run After First Hr.	0	264 hr
GAZS1	Div I,II EDG Fails to Start	7	265 d
GBZR1	Div III EDG Fails to Run During First Hr.	0	144 hr
GBZR2	Div III EDG Fails to Run After First Hr.	0	96 hr
GBZS1	Div III EDG Fails to Start	5	149 d
HCZL1	Heat Exchanger Ruptures/Plugs During Operation	5	9.35E+5 hr
IBZD1	Bistable Fails on Demand	0	N/A
ISZD1	Switch Fails on Demand	22	N/A
ITZL1	Transmitter Fails During Operation	9	1.17E+7 hr
IWZL1	Sensor Fails During Operation	4	7.28E+6 hr

Table 3.3.2-1
NMP2 Plant Specific Failures by Component

Database Variable	Component and Failure Mode	Number of Failures	Total Comp Demands or Cum. Oper Time
KSZE1	Strainer Plugs During Operation	0	1.80E+05 hr
LEZL1	Heater Fails During Operation	1	1.31E+5 hr
PCZR1	Standby Centrifugal Pump Fails During Operation	0	654 hr
PCZS1	Standby Centrifugal Pump Fails to Start	0	26 d
PDZR1	Norm Operating Centrifugal Pump Fails During Operation	20	4.76E+6 hr
PDZS1	Norm Operating Centrifugal Pump Fails to Start	1	301 d
PRZR1	Standby Reciprocating Pump Fails During Operation	0	55 d
PRZS1	Standby Reciprocating Pump Fails to Start	1	61 d
PTZR1	Turbine Driven Pump Fails During Operation (RCIC)	1	121 hr
PTZS1	Turbine Driven Pump Fails to Start (RCIC)	0	21 d
RAZD1	Relay Fails on Demand	20	N/A
TBZL1	Transformer Fails During Operation	1	1.57E+6 hr
VAZD1	Air Operated Valve Fails on Demand	5	1395 d
VAZM1	Air Operated Valve Transfers Closed	0	2.02E+6 hr
VCZM1	Check Valve Transfers Closed	0	3.03E+6 hr
VCZO1	Check Valve Fails to Open on Demand	1	1258 d
VHZM1	Manual Valve Transers Closed	0	7.51E+06 hr
VMZD1	Motor Operated Valve Fails on Demand	7	4578 d
VMZM1	Motor Operated Valve Transfers Closed	0	3.94E+6 hr
VPZN1	Safety/Relief (SRV) Sticks Open After Demand	N/A	N/A
VPZO1	Safety/Relief (SRV) Fails to Open on Demand	0	N/A
VRZN1	Relief Valve Transfers Open	3	5.81E+5 hr
VSZD1	Soleniod Operated Valve Fails to Open On Demand	17	1703 d
VSZM1	Solenoid Operated Valve Transfers Closed	0	1.81E+6 hr
VTZM1	Temperature Control Valve Transfers Closed	0	5.13E+5 hr
VUZM1	Pressure Control Valve Transfers Closed	0	2.14E+5 hr

Table 3.3.2-1
NMP2 Plant Specific Failures by Component

Database Variable	Component and Failure Mode	Number of Failures	Total Comp Demands or Cum. Oper Time
VXZO1	Explosive Valve Fails to Open on Demand	0	6
WJZB1	Expansion Joint Ruptures	0	4.36E+5 hr

Table 3.3.2-2
NMP2 Plant Specific Maintenance Unavailability

Database Variable	Component or System	Maintenance Unavailability
BAZMU	Battery Charger	0
BCAMU	Battery	0
CCPMU	RBCLC Standby Pump Train	4.08E-02
CCSMU	TBCLC Standby Pump Train	3.17E-04
CSHMU	High Pressure Core Spray	1.56E-02
CSLMU	Low Pressure Core Spray	1.30E-02
FASMU	Air Compressor	5.09E-02
GAZMU	Div I,II Diesel Generator	1.18E-3
GBZMU	Div III Diesel Generator	4.04E-03
ICSMU	Reactor Core Isolation Cooling	6.39E-02
RHCMU	Low Pressure Coolant Injection (RHR C)	8.30E-03
RHSMU	Residual Heat Removal (RHR) A,B	7.50E-03
SLSMU	Standby Liquid Control	1.10E-02
SWPMU	Standby Service Water Pump Train	2.85E-01
VZDMU	Safety/Relief Valve (SRV)	0

Table 3.3.2-3
Updated Failure Rate Distributions

Database Variable	Component and Failure Mode	Mean	Percentile 5th	Percentile Median	Percentile 95th
AAZB1	Press. Liquid Storage Tank Ruptures	4.82E-7	1.04E-8	1.67E-7	1.55E-6
ABZB1	Unpress. Liquid Storage Tank Ruptures	4.27E-7	1.00E-8	1.55E-7	1.33E-6
ACZB1	Gas Storage Tank Ruptures	5.19E-7	1.02E-8	9.37E-8	1.80E-6
BAZL1	Battery Charger Fails During Operation	4.73E-6	5.78E-7	2.14E-6	1.15E-5
BBZD1	Battery Fails on Demand	6.58E-4	1.33E-5	1.59E-4	1.63E-3
BBZL1	Battery Fails During Operation	1.32E-6	2.24E-8	5.73E-7	2.45E-6
CAZN1	Circuit Breaker Transfers Open	3.31E-7	1.41E-7	2.78E-7	4.40E-7
CAZO1	Circuit Breaker Fails to Open	1.34E-3	1.09E-4	7.06E-4	3.06E-3
CAZN1	Circuit Breaker Fails to Close	1.30E-3	1.27E-4	7.15E-4	2.59E-3
DGZE1	Air Dryer Plugs During Operation	8.63E-6	1.59E-7	1.89E-6	2.47E-5
EBZL1	Electrical Bus Fails During Operation	4.92E-7	3.35E-8	2.95E-7	1.08E-6
EUZD1	UPS Fails to Transfer on Demand	ISZD1 Value Used			
FCZR1	Air Compressor Fails During Operation	5.98E-5	1.26E-5	3.52E-5	7.77E-5
FCZS1	Air Compressor Fails to Start	3.65E-3	1.61E-4	1.58E-3	1.01E-2
FFZR1	Ventilation Fan Fails to Run	5.14E-5	1.15E-7	2.51E-6	2.02E-4
FFZS1	Ventilation Fan Fails to Start on Demand	1.11E-3	2.42E-6	1.80E-4	2.79E-3
GAZR1	Div I,II EDG Fails to Run During First Hr.	1.35E-2	2.62E-3	1.09E-2	1.57E-2

Table 3.3.2-3
Updated Failure Rate Distributions

Database Variable	Component and Failure Mode	Mean	Percentile		
			5th	Median	95th
GAZR2	Div I,II EDG Fails to Run After First Hr.	2.98E-3	1.39E-4	1.10E-3	6.36E-3
GAZS1	Div I,II EDG Fails to Start	1.80E-2	9.74E-3	1.38E-2	1.78E-2
GBZR1	Div III EDG Fails to Run During First Hr.	1.18E-2	1.39E-3	9.18E-3	1.26E-2
GBZR2	Div III EDG Fails to Run After First Hr.	3.32E-3	1.15E-4	8.17E-4	9.86E-3
GBZS1	Div III EDG Fails to Start	1.81E-2	9.64E-3	1.38E-2	1.80E-2
HCZL1	Heat Exchanger Ruptures/Plugs During Oper	4.95E-6	1.24E-6	3.56E-6	7.16E-6
IBZD1	Bistable Fails on Demand	1.06E-6	4.65E-8	4.45E-7	2.13E-6
ISZD1	Switch Fails on Demand	1.40E-6	6.63E-8	5.81E-7	1.12E-6
ITZL1	Transmitter Fails During Operation	7.16E-7	2.44E-7	4.65E-7	6.87E-7
IWZL1	Sensor Fails During Operation	3.22E-7	4.59E-8	1.74E-7	3.02E-7
KSZE1	Strainer Plugs During Operation	2.76E-6	1.50E-7	1.34E-6	5.03E-6
LEZL1	Heater Fails During Operation	7.65E-6	1.02E-6	6.97E-6	9.87E-6
PCZR1	Standby Centrifugal Pump Fails During Operation	4.95E-5	5.68E-6	2.52E-5	1.35E-4
PCZS1	Standby Centrifugal Pump Fails to Start	4.74E-3	4.48E-4	2.00E-3	1.00E-2
PDZR1	Norm Operating Centrifugal Pump Fails During Operation	4.07E-5	2.08E-5	3.50E-5	5.26E-5
PDZS1	Norm Operating Centrifugal Pump Fails to Start	3.21E-3	3.40E-4	1.79E-3	6.58E-3
PRZR1	Standby Reciprocating Pump Fails During Operation	6.57E-5	1.48E-6	2.80E-5	2.06E-4

Table 3.3.2-3
Updated Failure Rate Distributions

Database Variable	Component and Failure Mode	Mean	Percentile		
			5^{th}	Median	95^{th}
PRZS1	Standby Reciprocating Pump Fails to Start	7.6E-3	5.58E-4	2.03E-3	2.39E-2
PTZR1	Turbine Driven Pump Fails During Operation (RCIC)	4.07E-3	3.74E-4	2.24E-3	7.42E-3
PTZS1	Turbine Driven Pump Fails to Start (RCIC)	2.04E-2	9.82E-4	1.36E-2	5.05E-2
RAZD1	Relay Fails on Demand	2.08E-4	1.05E-5	4.23E-5	7.76E-4
TBZL1	Transformer Fails During Operation	6.08E-7	1.18E-7	3.60E-6	6.20E-6
VAZD1	Air Operated Valve Fails on Demand	3.76E-3	1.46E-3	2.60E-3	3.75E-3
VAZM1	Air Operated Valve Transfers Closed	2.44E-7	1.68E-8	1.06E-7	5.67E-7
VCZM1	Check Valve Transfers Closed	2.45E-8	5.64E-10	8.07E-9	7.31E-8
VCZO1	Check Valve Fails to Open on Demand	2.07E-4	5.52E-6	8.09E-5	4.69E-4
VHZM1	Manual Valve Transers Closed	2.94E-8	4.79E-10	9.62E-9	7.53E-8
VMZD1	Motor Operated Valve Fails on Demand	1.63E-3	7.97E-4	1.13E-3	2.49E-3
VMZM1	Motor Operated Valve Transfers Closed	6.10E-8	4.28E-9	3.80E-8	9.69E-8
VPZN1	Safety/Relief (SRV) Sticks Open After Demand	3.16E-3	6.48E-5	7.28E-4	7.93E-3
VPZO1	Safety/Relief (SRV) Fails to Open on Demand	8.21E-3	7.01E-4	6.19E-3	1.55E-2
VRZN1	Relief Valve Transfers Open	4.13E-6	6.76E-7	3.47E-6	6.10E-6
VSZD1	Soleniod Operated Valve Fails to Open On Demand	4.90E-3	2.48E-3	3.62E-3	4.77E-3
VSZM1	Solenoid Operated Valve Transfers Closed	6.61E-7	1.41E-7	5.13E-7	7.04E-7

Table 3.3.2-3
Updated Failure Rate Distributions

Database Variable	Component and Failure Mode	Mean	Percentile		
			5th	Median	95th
VTZM1	Temperature Control Valve Transfers Closed	8.78E-8	6.98E-10	2.17E-8	2.56E-7
VUZM1	Pressure Control Valve Transfers Closed	9.48E-5	4.74E-6	4.74E-5	9.00E-5
VXZO1	Explosive Valve Fails to Open on Demand	2.71E-3	5.94E-5	9.14E-4	6.11E-3
WJZB1	Expansion Joint Ruptures	1.19E-7	7.63E-9	4.78E-8	8.81E-8

Table 3.3.2-4
Updated Maintenance Unavailability

Database Variable	Component or System	Mean	Percentile 5th	Percentile Median	Percentile 95th
		Maintenance Unavailability			
BAZMU	Battery Charger	5.08E-5	8.07E-6	3.58E-5	1.02E-4
BCAMU	Battery	2.03E-5	9.76E-7	9.05E-6	6.24E-5
CCPMU	RBCLC Standby Pump Train	3.59E-2	2.20E-2	2.86E-2	3.52E-2
CCSMU	TBCLC Standby Pump Train	3.12E-4	1.63E-4	2.33E-4	3.04E-4
CSHMU	High Pressure Core Spray	1.41E-2	6.99E-3	1.04E-2	1.39E-2
CSLMU	Low Pressure Core Spray	1.09E-2	4.56E-3	7.30E-3	1.60E-2
FASMU	Air Compressor	4.49E-2	2.19E-2	3.28E-2	4.37E-2
GAZMU	Div I,II Diesel Generator	1.13E-3	4.61E-4	7.84E-4	1.49E-3
GBZMU	Div III Diesel Generator	3.98E-3	2.90E-3	3.51E-3	4.67E-3
ICSMU	Reactor Core Isolation Cooling	3.98E-2	1.53E-2	2.36E-2	6.25Q-2
RHCMU	Low Pressure Coolant Injection (RHR C)	6.81E-3	2.48E-3	4.87E-3	8.47E-2
RHSMU	Residual Heat Removal A,B	5.97E-3	2.27E-3	4.04E-3	8.21E-3
SLSMU	Standby Liquid Control	8.53E-3	2.92E-5	4.68E-4	1.40E-2
SWPMU	Standby Service Water Pump Train	5.06E-2	3.37E-3	3.50E-2	7.21E-2
VZDMU	Safety/Relief Valve (SRV)	1.64E-4	3.18E-6	5.45E-5	5.23E-4

Section 3.3.3

Human Failure Data (Generic and Plant-Specific)

3.3.3 Human Reliability Analysis

The objective of the human reliability analysis (HRA) is to provide qualitative and quantitative assessments of the human interactions addressed in the Nine Mile Point, Unit 2 IPE model. The logic model has been constructed to include basic events that represent failures of the operating crew to perform certain required functions in response to upset conditions and equipment failures. The quantitative assessments provided are in the form of probabilities of these human interaction basic events. The qualitative insights are in the form of identification of those plant and scenario specific factors that have an impact on the reliability of operating crew performance.

Section 3.3.3.1 presents an overview of the HRA process, and section 3.3.3.2. describes the quantification approach and presents two examples. Section 3.3.3.3 summarizes the results. Section 3.3.3.4 discusses pre-initiator event errors. Sections 3.3.3.1 to 3.3.3.3 deal entirely with post-initiator event errors.

3.3.3.1 Overall Approach to Human Reliability Analysis

The plant logic model, the event trees and fault trees, were constructed to include human interaction basic events. To define the plant logic, these events are adequately defined in terms of the failure mode they represent e.g., operators fail to depressurize the reactor following a loss of high pressure injection. However, in order to quantify these events, i.e., estimate their probabilities, it is essential to define them in much greater detail. For example, it is necessary to understand what cues and procedures the operators use to guide them to perform the required function, what they have to do to successfully accomplish that function, the time available, and other factors that might influence their probability of success or failure. These factors are all scenario specific. The first step in the HRA was therefore to define the events as clearly as possible in preparation for the quantification. This was done by studying the scenarios to which the human interaction events contribute, and understand, among other things, the time line of the events.

Another function of this step of the HRA task is an identification of potential dependence between the human interaction (HI) events that occur in the model. Functional dependencies of the type, "if event A occurs, event B cannot be successful," are handled in the overall structure of the model, i.e., they are hardwired into the event tree structure. What is principally of concern here is the influence of success or failure in a preceding event on the probability of success or failure of another event. There are a variety of reasons why the events may be probabilistically dependent; one important issue is that the cognitive processes needed to recognize the need for multiple actions may have common elements. To assist in the identification of such cognitively correlated HIs, the following groundrules were adopted:

(a) If two HI events are associated with responses to the same plant status (e.g., initiate HPCS pump, initiate RCIC on failure of auto initiation at Level 2), the cognitive part of the failure probabilities are considered to be totally dependent.

(b) As a corollary to this, if, in the chronological development of the scenarios, an HI failure event follows a successful HI, and the procedural instructions for both events are closely related, the cognitive failure probability of the second HI should be very small and can be neglected, since the success in the first event implies a successful recognition of the scenario.

(c) If human interactions are i) separated by a significant time (i.e., time between cues or required responses is long), or ii) separated chronologically in the sequence by a successful action, or iii) responses to different cues in different parts of the EOPs, they may be regarded as being independent.

(d) In addition, the early memorized responses may be regarded as independent from these actions for which the procedures are expected to be providing the direction.

Other types of dependency, such as the fact that performing one function may take resources away from another is also considered by addressing, in the evaluation of the HEPs, the role of crew personnel, both in performing the actions called for, and in recovering from failure to execute correctly.

Since there is a very large number of possible scenarios, this process of review could be a very time-consuming task. However, many scenarios can in fact be grouped functionally in terms of cues, procedures, and key operator responses, so that a limited number of different variations of the functionally similar human interaction events is adequate to represent the human reliability aspects. In general, the grouping of scenarios for HI purposes was done conservatively, i.e., the HEP was evaluated for the most demanding scenario in a functionally similar grouping. These limiting or bounding scenarios were identified through discussions with the event sequence analysts.

Relatively few dependent HI basic events were identified, partly because of the assumptions upon which the logic model was constructed, and partly because of the design features of the plant. For example, one of the most difficult HRA tasks in a BWR PRA is the treatment of ATWS, where there are many inter- dependent actions. However, one of the human interactions usually found to be significant in BWR PRAs, namely initiation of boron injection via the SLC system, is not a concern at Nine Mile Point, Unit 2, since initiation of SLC is automatic. Nevertheless, some of the human interactions in the ATWS model were identified as being dependent. In addition, some of the events within the fault tree models for the service water system and the high pressure nitrogen system were found to be dependent. These dependencies are reflected in the probabilities used.

3.3.3.2 Quantification Approach

The model of human interactions used for the evaluation of a human error probabilities is the simple one that splits the response into two components, a detection, diagnosis and decision (DDD) phase, and an execution phase. This is compatible with the ASEP methodology [Ref.42], the more recently published EPRI methodology [Ref. 43], and the HRA Handbook [Ref. 44], all of which were used in the quantification. Reference is made to these documents for details.

The way in which these methods were used is as follows. The "nominal approach" of ASEP was used to generate the probability of failure in the DDD phase for many of the HEPs and in the execution phase for most of the HEPs. The ASEP method was used for estimating the HEPs in the DDD phase for those human interactions that are time limited, or involve recognizing an appropriate procedure. The decision tree approach of Reference 43 was used for the DDD contribution to the HEPs for those human interactions that are associated with missing a certain step in a procedure, or can otherwise be characterized as conditional upon having made a correct choice of initial response. In some cases, the annunciator response model of reference 44 was considered an appropriate model for the DDD phase.

While it was generally used for the execution phase of the HIs, it is well known that in certain circumstances, the ASEP approach is conservative. So for particular important actions which are well practiced, with significant chance for recovery, the elemental HEPs of the Handbook [Ref. 44] were used to provide a more realistic estimate of the execution phase HEPs.

To apply these methods, it was necessary to understand the temporal content for the actions, and identify the cues, procedural directions and detailed steps required to achieve success. In addition, opportunities to recover from an error were identified to give a basis for applying recovery factors to the initial base value HEP. These details are documented in the HRA analysis file. Examples of the quantification are given below.

3.3.3.2.1. Analysis of HI Event HHOD-1

The event is defined in the logic model as "operators fail to depressurize the RPV on loss of high pressure injection and maintain it depressurized for 24 hours". This event occurs in the transient, small LOCA, and medium LOCA event trees. The action is necessary to allow the low pressure systems to inject to prevent core damage. The success criterion of the model is that depressurization and subsequent injection should be established before the level drops substantially below the top of active fuel. This success criterion defined by thermal hydraulic analysis results, is used as the basis for determining the time window within which the action of depressurization should be accomplished. The first indication that the depressurization may be required is when, at the Level 2 actuation setpoint, the HPCS and RCIC systems fail to automatically initiate. Since Level 2 is reached relatively quickly, the time windows were measured from the time of this cue (Level 2). These time windows are estimated from thermal hydraulic analysis as about 20 minutes for a medium LOCA and about 45 minutes for the small LOCA or a transient. The path to reach the depressurization instruction is via EOP-RPV path RL, through the procedure EOP-C1, developed from the BWROG contingency 1, and into EOP-C2. The procedural instructions direct the operators to manually open each of the ADS valves when the level in the reactor vessel reaches the top of active fuel. It is assumed in the PRA model that the automatic depressurization signal has been inhibited in accordance with procedure and that manual depressurization is indeed necessary. This assumption implies that the operating crew has already entered into the EOPs, and suggests that the HEP should be relatively low.

While the action is only required when level reaches TAF, the crew has ample time to anticipate the required action and be prepared for its execution. This is a frequently practiced action, and very simple to do. The EOP-C2 even has an explicit confirmatory step to verify that the SRVs are open, and the impact on the plant is obvious. The explicit confirmatory step is credited as a recovery mechanism for the execution phase. The monitoring of the pressure could also be credited as a recovery mechanism for the execution phase, but is not considered in this calculation.

Both the ASEP and the EPRI approach could be used for the evaluation. In the end the ASEP approach was used because it allows a differential between the medium and small LOCA cases on the basis of the difference in time windows. The EPRI approach addresses the time available only by assessing whether there is time available for recovery from an initial error, and it is judged that in both these cases enough time is indeed available. Therefore, there is no difference between the HEPs estimated for the medium and small LOCA cases using the EPRI approach.

The ASEP methodology was applied as follows for the transient or small LOCA case. The time window is 45 minutes, and the time necessary to accomplish the execution of the task is estimated to be 2 minutes. Thus, the time available for DDD is 43 minutes. Using rule 9e in Table 8-1 of Reference 43, the lower bound curve of the nominal diagnosis model (Figure 7-1) is considered appropriate and given an HEP for the DDD phase of 2×10^{-4}. The HEP for the execution phase is estimated from the probability of failing to open ADS valves and failing to recover from that error when asked by the procedure if the valves are open. This results in an HEP of 5×10^{-4}.

The sum of the errors in the DDD and execution phase is 7×10^{-4}. This is assumed to be a median value for a lognormal distribution with an error factor of 5. The correction factor to convert this median into a mean value is 1.6. Therefore, the HEP for the small LOCA or transient case is $7 \times 10^{-4} \times 1.6 = 1.1. \times 10^{-3}$.

In a similar manner to HEP for the medium LOCA was estimated to be 3×10^{-3}.

In addition, this human interaction basic event includes failure to maintain low pressure in the RPV, which essentially involves re-establishing nitrogen supply to the safety relief valves if and when the containment is isolated. The SRVs would be expected to remain open on their accumulators for many hours, at which time the decay heat is low enough that it would take a long time to repressurize the RPV, in the event that reduced nitrogen pressure caused the SRVs to fail closed. Therefore, there is ample time to recover, and this portion of the HEP is considered negligible.

3.3.3.2.2. Analysis of HI Events Associated with the Service Water System

In the fault tree model for the service water system, there are four basic events SWPZHHHSA1(2,3,4)00091 (HHSA1,HHSA2,HHSA3,HHSA4) associated with different operator actions to start additional pumps, isolate the reactor building and turbine building loads, and stop a pump or open a flow path. The events are included to model the actions necessary to maintain adequate flow through the system and either prevent pump trip on low flow, and prevent pump runout, or to restart pumps if they have tripped, under a variety of scenarios (for details, see the system description, section 3.2.1.8).

In reviewing the structure of the model, it was determined that event HHSA1, (starting pump E or F given that one pump has failed), and event HHSA2 (isolating reactor building and turbine building loads given only three pumps are operating) were highly dependent when they appeared in the same cutset. The dependence was conservatively assumed to be complete and the basic event identifier for SA2 subsequently changed to that for event SA1. In this way the failure represented by event SA2 is assumed to always occur if the failure represented by event SA1 occurs.

Even though all four events were initially intended to describe different responses, they were quantified in the same way, as a response to a pump trip annunciator. Given that there is adequate time to restart a pump even if the pump should first trip,(events HHSA3 and HSA4), and that the operation of the SW system is a high priority function in which the operators are skilled, it is judged that the probability is dominated by failure to notice the annunciators. Since, in the majority of cutsets, the failures in the SW system occur at times different from those of the other failures in the cutset, service water events HHSA1,2,3 are independent of other HIs. Since several annunciators may be triggered sequentially, but all pointing to the same problem, the implication that their impact is reinforcing rather than distracting. A value of $1.6x10^{-4}$/demand based on the annunciator response model of Reference 44 is considered appropriate. This is based on a basic HEP of 1×10^{-4}, and an assumed lognormal distribution with an EF of 5. This error factor is smaller than that proposed by Swain [Ref. 44], but is chosen so that the mean value is somewhat lower than his value to represent the re-enforcing nature of the multiple alarms.

The exception to being able to argue that the service water related HI is independent of other HIs is event HHSA4, which occurs as a result of a loss of one of the offsite sources. In that case, there will be many annunciators and they will be indicating different problems. However, the operators are highly conscious of the need for SW, and establishing an appropriate flow is considered a top priority action, and its probability is estimated as being the same as for the other service water related events. The other HI event considered in this scenario, (i.e. cross connecting the offsite buses), is a longer term recovery action and is considered independent of the event HHSA4. Establishing service water is not expected to detract significantly from the time available to achieve the cross-connect of the buses..

3.3.3.3 <u>RESULTS</u>

The Level I human interaction basic events analyzed, the HEP values and some comments are presented in Table 3.3.3-1.

The HEP evaluation for the NMP2 Level II actions has been performed in a manner similar to that described above. Consultation between the Level I and Level II analysts was required to ensure consistancy. Section 4.6.2.5 summarizes the actions considered in the Level II IPE and the HEPs developed for use in the Level II assessment.

3.3.3.4 Pre-initiating Event Human Errors

The potential contribution to system unavailability due to the improper performance of maintenance and testing procedures was evaluated. Generally, this investigation relied on the use of industry and plant specific data, and theoretical HRA techniques to identify systems that may be especially susceptible to human induced failure modes as a result of system maintenance or testing, and quantify the affect of these failure modes on system unavailability. However, since there is significant industry data to accurately represent actual equipment unavailability as a result of numerous causes, more weight was given to results supported by data analysis rather than HRA.

An important assumption implicit in this analysis is that human induced failure modes affecting single train systems, caused by a maintenance related error, are subsumed by estimates of equipment unavailability due to test and maintenance activities and equipment failure rates. This assumption appears to be supported by industry and plant specific data. Therefore, this analysis concentrates on identifying common cause human induced errors that can potentially disable several trains of a system; and thereby, fail a critical safety function contained in the IPE model.

The procedure to assess system unavailability caused by the improper performance of maintenance and testing procedures is discussed below:

1. Review industry and plant data, and similar analyses performed in other PRAs for BWR reactor plants to identify potential maintenance related events (i.e., misalignment of equipment or miscalibration of system electronic components), that could defeat normal system operation.

2. Screen these data to select only those events that could affect system performance and appreciably contribute to core damage frequency. The screening criteria used by the analyst to exclude human induced system failures, caused by the improper performance of maintenance activities, from further consideration include:

    - Single trained systems that contain independent instrumentation and control logic (i.e., the unavailability of the system caused by the operator error is assumed to be accounted for in another failure or initiating event),
    - Systems that contain components that are monitored and annunciated in the control room if misaligned or disabled (e.g., breaker left open),
    - Systems that contain MOVs that would be automatically repositioned upon actuation if the valves were misaligned,
    - Mechanical systems that are normally operating, or
    - System failures that could be caused by a maintenance or testing activity that includes double verification of system configuration during system restoration, post maintenance operability check, or inspection of system configuration during routine operability checks (e.g., shiftly or daily checks).

3. Perform an analysis of the NMP2 maintenance and testing procedures performed on the equipment that is selected to determine the probability of human induced system failures caused by the improper performance of these activities. (Note that this probability is synonymous to the unavailability of the equipment.)

System Selection Process

An evaluation of industry and plant experience was conducted to ensure that important system unavailability contributions from maintenance induced system failures are accounted for in the IPE model. A query of the INPO LER industry data base was performed to identify potential maintenance related events involving miscalibration of I&C equipment and misalignment of system components that occurred during the performance of a scheduled surveillance test or maintenance activity at a BWR plant from January 1980 through June 1992. The criteria for selecting events from the data base include the following:

- The misalignment of a system or the miscalibration of an electronic component must have been caused by operator error;
- the resulting system fault prevented equipment operation as defined in the functional and system success criteria; or
- the fault is not realized until a subsequent system challenge (i.e., system operation or performance of another test or maintenance activity), results in an obvious system failure.

Table 3.3.3-2 describes the events selected from the data base that were further evaluated as potential precursor events. A review of these data resulted in the identification of one pre-accident human induced event (i.e., failure to restore the SLC system to a normal configuration post test), that could result in the failure of the reactivity control function during a postulated ATWS scenario. Additionally, other events suggested a potential common cause failure of ECCSs caused by miscalibration of dP transmitters. Due to the potential severe consequence of common cause failure of these instruments (e.g., RPV pressure dP transmitters), this event was also selected for further consideration.

A description of the testing procedures performed on both systems at NMP2 is provided in the following subsections.

3.3.3.4.1 Standby Liquid Control System Analysis

The SLC system is an automatic initiation system that could be left misaligned upon completion of testing or maintenance.

The following surveillance tests are performed on the SLCS:

- N2-CSP-3M, Standby Liquid Control Chemistry Surveillance at Unit 2
- N2-OSP-SLS-Q002, Standby Liquid Control Motor Operated Valve Operability Test
- N2-OSP-SLS-Q001, Standby Liquid Control Pump, Check Valve, and Relief Valve Test

When surveillance tests are performed on this system, the operator initials each step upon its completion, including returning the system to its normal configuration (using a checklist).

After the test is completed, an independent verification of the restoration valve line-up is performed by an independent operator.

Additionally, a post maintenance test is performed by Operations to verify system operability. An independent verification of system restoration is also performed. Therefore, it is judged that these activities pose little risk of contributing to SLC system unavailability.

3.3.3.4.2 ECCS Initiation Instrumentation Analysis

There are several human induced system failure mechanisms that can render a pressure instrument loop inoperable, causing the failure of low pressure ECCS injection valves to fail to open on demand.

There are four (4) separate instrument loops, one for each low pressure ECCS injection path. Calibration of these instruments is performed using procedure N2-ISP-ISC-R103. This procedure was reviewed to determine the potential for miscalibration.

These instrument loops contain differential pressure transmitters with a range of 0-700 psid. The RHR loops have a setpoint at 127 psid and the Core Spray loop has a setpoint of 84.8 psid. Actuation at these setpoints cause permissive logic to be satisfied that allow low pressure injection MOVs to open if an ECCS signal is present.

The following discussion describes how calibration errors and improper restoration of the instrument pressure transmitter can impact instrumentation loop operation and low pressure injection MOV permissive interlock logic:

- The effect of a miscalibration caused by using an incorrect head correction or zero correction will, at worst, cause a shift in setpoint in the non-conservative direction (i.e., low). If the head correction is 60 feet and it was applied incorrectly, it could result in the opening permissive being reduced by approximately 30 psig.

- Utilization of a defective calibration source would require significant adjustment of the pressure instrument loop. However, if the loop setpoint is discovered to be out of specification, it must be reported to the Shift Supervisor.

- The setpoint for the open permissive could be reduced to a low value if the technician incorrectly adjusted the setpoint to the minimum trip setting. This means that the injection valve would be prevented from opening until the reactor pressure reaches an extremely low value. This is not expected to result in injection failure, but rather a delay establishing injection.

- Another situation that would prevent the injection valve from opening is mispositioning the high pressure instrument isolation valve while the line is pressurized. This condition would not be detected during routine observation of the instrument loop readout because the loop readout is normally at 100% scale or offscale high. However, inadvertent misalignment of the instrument block valve can be recognized during the calibration of the instrument by comparing the transmitter reading in psid before and after the procedure (i.e., if either valve is mispositioned, open or closed, the loop delta P will change), and performing an independent verification of valve position before returning the instrument to service.

- Instrument loop failures caused by using the wrong calibration procedure, is not expected to be important contributor to system unavailability for the following reasons:

  - The procedure specifies 0-700 psid and the loop readout scale is specified 0-700 psid, not 0-100% scale. It would not be easily confused with a procedure that specified a 0-100% scale.

  - This procedure cannot be easily confused with a reactor vessel level calibration procedure again because the level calibration procedure specifies either 0-100% scale or 0-150 inches of H_20, not 0-700 psid.

  - This procedure cannot be easily confused with a reactor pressure calibration procedure since this instrument does not contain a high pressure sensing line (i.e., these instruments are usually referenced to atmosphere).

The opportunity for common cause error that could render all four (4) injection valve permissives inoperable and prevent valve opening on demand is limited. Most of the errors described above result in a situation that is detectable during normal operation. The failure mode that has the most significant impact on operability of the low pressure injection valve pressure permissive is the failure mode that isolates the high pressure sensing line while it is at high pressure. This failure mode would require that several operator errors be committed, including independent verification of valve position post calibration, in the proper sequence for an instrument loop to be rendered inoperable.

The potential for common cause failure of multiple ECCS instrument loops due to human interaction was investigated. Factors that affect the performance of this procedure that could mitigate the potential for common cause failure of this equipment is described below:

- When ECCS instrumentation is calibrated, it is generally not accomplished by the same crew. Instead, calibrations are commonly performed by crews on different shifts. If available time is short, multiple crews on one shift may be used to accomplish the work.

- Additionally, the work is well proceduralized, and is independently reviewed before the work is closed out. "As found" and "as left" conditions are recorded and reviewed by a Supervisor to determine whether the results of the test are reasonable.

- Finally, if a piece of meter and test equipment is found out of calibration all instrumentation calibrated with this test equipment since the last test when the instrument was certified to be in calibration is evaluated for accuracy and recalibration if necessary.

Results of Analysis

The potential for human induced common cause failure of the SLCS and the ECCS instrumentation due to miscalibration is considered unlikely.

Unavailability of the SLCS is assessed to be approximately 3.0E-3 with an error factor of 3. This estimation is based on the data analysis, and does not assume the incorporation of recommended changes to Operations and Chemistry surveillance procedures, and the operator rounds checklist described in Section 6. (Application of the ASEP HRA methodology, assuming these procedural changes, would yield a median point estimate of 3.0E-4 with an error factor of approximately 16.)

Unavailability of the ECCS loop instruments is assessed to be 1.0E-5 with an error factor of 10. This mean point estimate is based on expert judgement in interpreting existing industry and plant specific data. This assessment is in close agreement with the lower bound mean value provided in an analysis performed for NUREG-1150.

TABLE 3.3.3-1
TABLE OF HUMAN ERROR PROBABILITIES

EVENT NAME	SCENARIO DESCRIPTION	OPERATOR ACTION	HEP (Mean, EF)	COMMENTS
HHOE1 (1)	ATWS, ADS and HPCS inhibited, FW and RCIC unavailable	Emergency Depressurization on loss of RPV inventory	.16, 5	
HHOE1 (2)	ATWS, ADS and HPCS inhibited, RCIC operating	Emergency Depressurization on HCTL	0.0	Assumed success. This is conservative with respect to demand for events HHIL1, and HHCH1
HHIC1	Loss of service water, unit tripped operating on RCIC	Prevent or recover from RCIC isolation on high room temperature	1.0	No procedure
HHIL1	ATWS, depressurization required	Override RCIC low pressure trips	.57, 0+	Control room resource limitation leads to high HEP
HHOA1	Station Blackout	Isolate RCIC backpressure and high temperature trips and shed DC loads	10^{-2}, 5	Assumes SBO procedure implemented
HHMO-1	ATWS	Bypass MSIV closure signal	1.0	Not high priority, and resource limitation make this very unlikely.
HHAI-1	ATWS	Inhibit ADS	5.6×10^{-3}, 5	
HHME-1	Transient or small LOCA, auto-initiation of ECCS	Manually initiate ECCS	.032, 5	Assumed very conservative but value retained since not of high importance
HHCN1	Scram, MISVs don't close	Mode switch in shutdown to to prevent MSIV closure	3×10^{-4}, 5	
HHCN2	MSIV closure during transient	Reopen MISV to provide path to condenser	.07, 3	
HHCN3	MSIV ATWS event	Reopen MISV to provide path to condenser	1.0	

+ No error factor is assigned to HEPs on the order of .5

TABLE 3.3.3-1
TABLE OF HUMAN ERROR PROBABILITIES

EVENT NAME	SCENARIO DESCRIPTION	OPERATOR ACTION	HEP(Mean,EF)	COMMENTS
HHSA1	One or more service water pumps trips	Start pump E of F when required	$2.0x10^{-4}$, 5	All service water events are modeled as annunciator responses.
HHSA2	As above	Isolate RB/TB loads to prevent pump runout	$2.0x10^{-4}$, 5	This event should be renamed the same as HHSA1 to address dependency
HHSA3	SW cross-tie inadvertently goes closed	SW loads to prevent loss of service water	$2.0x10^{-4}$, 5	
HHSA4	115KV source A or B fails	Control service water flow to prevent loss of service water	$2.0x10^{-4}$, 5	
HHMS1	ATWS	Put mode switch in shutdown	$3x10^{-4}$, 5	This is the same as for a normal transient with scram
HHN11	Failure of nitrogen system	Valve in high pressure nitrogen system	$3.6x10^{-3}$, 5	
HHFW1	Small LOCA	Restore motor-driven feedwater	$5.5x10^{-3}$, 5	
HHFW2	ATWS, feedwater runback	Restore motor-driven feedwater before MSIVs go closed on low level	.5, 0	
HHOD1	Transient or small LOCA, loss of high pressure injection	Manually emergency depressurize	$1.1x10^{-3}$, 5	
HHOD2	Medium LOCA, loss of high pressure injection	Manually emergency depressurize	$3x10^{-3}$, 5	
HHOV1	Small LOCA, vapor suppression system fails	Initiate containment sprays	$4x10^{-2}$, 5	
HHOV2	Medium LOCA, vapor suppression system fails	As above	$4x10^{-2}$, 5	

TABLE 3.3.3-1
TABLE OF HUMAN ERROR PROBABILITIES

EVENT NAME	SCENARIO DESCRIPTION	OPERATOR ACTION	HEP(Mean, EF)	COMMENTS
HHCI1	Transient or LOCA with high containment pressure and high suppression pool level	Start HPCS from suppression pool	$2.5x10^{-2}$, 5	
HHCH1	ATWS, depressurized RPV	Terminate and prevent low pressure injection to prevent flushing out boron	$4.6x10^{-2}$, 5	
HHSW1	Loss of all ECCs	Align SW for low pressure injection	.04, 5	
HHS11	Station blackout (short term)	Align FW pump for injection	.5	
HHS21	Station blackout (medium term)	Align FW pump for injection	.5	
HHS31	Station blackout (long term)	Align FW pump for injection	.5	
HHO11	Station blackout (short term)	Depressurize reactor	$1x10^{-3}$, 5	Use ODI
HHO21	Station blackout (medium term)	Depressurize reactor	$1x10^{-3}$, 5	Use ODI
HHO31	Station blackout (long term)	Depressurize reactor	$1x10^{-3}$, 5	Use ODI
HHHA1	Failure of RHR due to valve failure	Manually open valve	$1x10^{-2}$, 5	
HHMA1	Loss of service water	Open LPI pump room doors to establish cooling	.1, 3	Assumes procedure in place

TABLE 3.3.3-1
TABLE OF HUMAN ERROR PROBABILITIES

EVENT NAME	SCENARIO DESCRIPTION	OPERATOR ACTION	HEP(Mean, EF)	COMMENTS
HHKR1	Loss of offsite source A	Cross-connect remaining source to bus A	.02, 5	
HHRK2	Loss of offsite source B	Cross-connect remaining source to bus B	.03, 5	
HHOH1	Transient/LOCA	Align containment heat removal	$1x10^{-5}$, 5	
HHOH1	ATWS	Align containment heat removal	$9.6x10^{-3}$, 5	
HHCV-1	Loss of RHR, all support systems available	Vent containment	$6x10^{-3}$, 5	
HHCV-2	Loss of RHR, no instrument air or AC	Vent containment	$1.4x10^{-2}$, 5	
HHCV-3	Loss of RHR, no instrument nitrogen	Vent containment	$8.7x10^{-3}$, 5	
HHCH2	ATWS, RPV not depressurized	Restart HPCS	1.0	Conservatively assumed dependent on OE
HHU21	Blackout, RCIC	Stop depressurization before RCIC stalls	$1.0x10^{-2}$, 5	see HHOA1

Table 3.3.3-2
Potential Common Cause Human Errors

Event Date	Docket LER No.	Event Description	Cause Description	Comments
1/22/82	298 82001-00	During routine surveillance testing, reactor vessel level switch NBI-LIS-72C failed to trip at its tech. spec. setpoint. The -145.5" low level switches in the redundant logic were operational and would have responded to a low level.	A Yarway model 4418C level indicating switch was sluggish in actuating at its setpoint. The cause of the occurrence is misalignment of the switch mechanism.	No indication of common cause failure affecting RPV level indication. Independent failure modes are considered to be satisfactorily treated in existing data base.
1/6/82	259 82002-00	During normal operation, drywell high pressure switch, 1-PS-64-58C, was found by NRC inspector to be isolated. No hazard to the public existed because 1-PS-64-58B and D were fully operable during the time that 1-PS-64-58C was isolated.	SI 4.2.B.5 was performed on 1-PS-64-58C on 1/4/82 and apparently left it isolated. 1-PS-64-58C was immediately returned to normal. All other valving involved in SI 4.2.B-5 was verified correct.	No indication of common cause failure affecting drywell pressure indication. Independent failure modes are considered to be satisfactorily treated in existing data base.
6/19/79	324 79050-00	During normal plant operation, a clearance was given to the mechanics to uncouple 2B RHR Service Water Pump for an alignment check. The mechanics uncoupled 2A RHR Service Water Pump by mistake, making both loops of RHR Service Water inoperable for approximately 7 hours.	When mistake was discovered, 2A pump immediately recoupled & tested to insure operability.	Although both RHRSW trains were unavailable, it is assumed that if the operating crew and maintenance personnel didn't immediately notice the error, the time frame is limited by the Technical Specification LCO applicable to RHR availability. The NMP2 PRA model assumes that administrative procedures are effective in preventing both of the RHR heat exchanger trains from being disabled due to errors in performing maintenance.
6/23/76	324 76000-00	With reactor at power in run mode and performance of HPCI steam line high differential pressure periodic test in progress, two high steam line differential pressure switches (E41-DPIS-N004 & N005) were found out of calibration at 245 & 314 inches of water, respectively. Shift foreman failed to declare an LCO, therefore, the required isolation was not done.	Personnel error: "N004" had drifted out of calibration and "N005" had been miscalibrated before by using a "PLUS" instead of a "MINUS" head correction factor. Switches immediately recalibrated to 219 & 292 inches.	No indication of common cause failure. Additionally, this failure mode is considered to have a negligible affect on the PRA since the probability of steam line break scenarios is sufficiently low. Otherwise, this failure mode would not prevent normal operation of the system.
5-21-79	277 79027-00	During a PCIS logic system functional test with the unit shutdown RWCU inboard isolation valve MO-2-12-15 failed to automatically isolate. The cause, discovery, and resolution of the event all occurred during a unit outage. The RWCU isolation valves are required to be operable only during power operation, therefore, there is no safety significance.	Limit switch LS-7 on the motor operated valve was misaligned during replacement of the limit switch assembly. The switch was readjusted and the interlocks retested with acceptable results. The MO valve maintenance procedure was revised strengthening administrative controls to assure performance of outgoing interlock checks prior to return of service.	No indication of common cause failure of the RWCU system. Additionally, this failure mode is considered to have a negligible affect on the PRA, since the probability of steam line break scenario during the time period between surveillance testing is sufficiently low.

Table 3.3.3-2
Potential Common Cause Human Errors

Event Date	Docket LER No.	Event Description	Cause Description	Comments
3/21/78	409 78006-01	Controls for both Alternate Low Pressure Core Spray Pumps were aligned for manual rather than "AUTO" start. Reactor startup was in progress with less than 1% power attained.	Operator action had placed pump controls in the control room to "OFF" to discontinue backup operation for the HPSW system. Controls were not promptly returned to "AUTO" position. Upon discovery of the misalignment, controls were shifted to "AUTO." Modification to annunciator system will be accomplished to improve alarm display for vital system or components.	A similar occurrence of an improper system alignment not being immediately recognized by the operating crew in the control room is prevented by control board annunciator system and alarm response procedures. Additionally, there are at least 4 control room operators surveying the control boards routinely during the shift. The probability of this failure is considered low for NMP2.
3-4-76	321 76000-00	While performing surveillance per procedure HNP-1-3106, the B21-N009C Main steam line instrument flow rack was not correctly valved back into service. As a result, the instrument was out of service for 22 hours. The misalignment was discovered because the instrument went upscale and tripped half of the logic for the group 1 isolation.	Personnel error: technicians relied upon gage readings, not actual valve lineups. Upon discovery, the instrument was returned to service.	With the instrument flow rack valved out, high steam flow protection for the affected steam line is disabled. However, the probability that a HELB in the affected line over a time period of a shift is considered extremely remote (i.e., the operating crew is likely to recognize the apparent abnormal indication within one shift).
2/20/74	277 74000-00	During "A" RHR logic testing, relay 10A-K33A energized, but did not conduct. During "B" RHR logic testing relay 10A-K113B failed to open.	The "A" relay contacts were dirty and the "B" relay cover was on upside down and misaligned preventing movement.	No indication of common cause failure affecting the RHR initiation logic. Independent failure modes are considered to be satisfactorily treated in existing data base.
7/18/84	331	Standby Liquid Control System Misalignment	While in normal full power operation, improper manipulation of the common suction valve from the standby liquid control system (SBLC) tank to two SBLC pumps resulted in SBLC being isolated for nearly 5 hours. The manual valve, which had been erroneously unlocked and cycled by a chemistry technician while performing a portion of a surveillance test, was observed to be in the incorrect position by licensed operators while walking by the SBLC. The valve was immediately restored to full open and the SBLC lineup verified.	Common cause failure mode affecting the SLC system treated explicitly in fault tree model.

Table 3.3.3-2
Potential Common Cause Human Errors

Event Date	Docket LER No.	Event Description	Cause Description	Comments
3-17-86	458	Diesel Generator Fuel Oil Valve Misalignment	On 3/17/86 at 1526 with the unit at 44% power and during a surveillance test of the Division II Diesel Generator, the diesel generator began to lose speed and was manually tripped. Investigation revealed that a misaligned fuel oil strainer valve restricted fuel oil flow to the engine. Further investigation revealed that the diesel generator may have been inoperable since 2/17/86. Tech. Specs requires a plant shutdown if one diesel generator is inoperable for greater than 72 hours. Immediate corrective action was taken to restore operability to the diesel generator by 3/18/86. Because of a similar problem earlier with the Division I Diesel Generator, additional corrective action was taken to prevent recurrence. There was no adverse effect on the health and safety of the public.	This event is assumed to describe a potential common cause failure mode that affected both EDGs. The NMP2 LOOP/SBO event model considers the unavailability of the EDGs due to common cause failure to start and run modes. The data base used to determine the applicable common cause factor is assumed to include this event.
8/26/86	277	Reactor Water Level Transmitter Out-of-Service	On August 26, 1986 at approximately 1600 hours, it was discovered that the two instrument rack isolation valves for reactor water level transmitter LT-2-2-3-72B (LT-72B) were closed, thereby rendering the transmitter inoperable. Unit 2 was in startup mode at approximately 7% thermal power at the time of the discovery. LT-72B provides reactor water level signals to several logics including Core Standby Cooling System (CSCS) initiations. Loss of LT-72B inputs to these logics would not have defeated the capability of the CSCS to initiate automatically due to abnormal reactor water level because redundant channels were operable. The valve misalignment resulted in a failure to comply with the plant Technical Specifications for a 9½ hour time period. Instrument technicians performing a surveillance test had neglected to return the instrument tack isolation valves to their proper post-test position.	No indication of common cause failure affecting RPV level indication, since only one level transmitter was isolated. Independent failure modes are considered to be satisfactorily treated in existing data base.

Table 3.3.3-2
Potential Common Cause Human Errors

Event Date	Docket LER No.	Event Description	Cause Description	Comments
1/4/89	341	Failure of recirculation system field breaker due to mechanical binding.	On January 4, 1989, recirculation pump B's field breaker failed to trip when the motor generator set was shutdown during a controlled shutdown. An operator verified that the MG Set had stopped rotating locally, but found that the trip coil had burned out. The field breakers for both divisions of the recirculation system were quarantined until an action plan was developed for troubleshooting. Based on analysis of the breaker, it was determined that the failure was caused by mechanical binding of the linkage. This is attributed to the failure to lubricate the breaker during preventative maintenance in March of 1988 and the misalignment of the contacts. A conclusive cause for the misalignment of the contacts could not be determined.	No indication of common cause failure affecting RPT protective circuitry. Independent failure modes are considered to be satisfactorily treated in existing data base. Additionally, the NMP2 ATWS event models explicitly consider the failure of the RPT safety function failure due to CCF of these field breakers.
6/16/89	461	Inadequate procedure leads to miscalibration of reactor water clean-up leak detection modules resulting in operation prohibited by Tech. Spec.	On June 19, 1989, Riley Point Modules 1E31-N621A, the Division I Reactor Water Clean-Up System (RWCU) pump room A area temperature module and 1E31-N620A, the west RWCU heat exchanger room temperature module failed their channel checks. Investigation into the failures revealed that the modules had been miscalibrated on June 16, 1989. The control and instrumentation (C and I) technician who performed the calibration on June 16, 1989, used an incorrect mode setting on the potentiometer used to perform the calibration. Tech. Spec. 3.3.2.C.1 requires that when either of these modules be inoperable they be restored to operable status within two hours or that the RWCU system be isolated. This Tech. Spec. was not met for approximately seventy-two (72) hours. The cause of this event is attributed to an inadequate procedure. Contributing to the event was a personnel error and the lack of a criteria for determining if the results of channel checks are satisfactory.	No indication of common cause failure of the RWCU system. Additionally, this failure mode is considered to have a negligible affect on the PRA, since the probability of steam line break scenarios during the time period between surveillance testing is sufficiently low.

Table 3.3.3-2
Potential Common Cause Human Errors

Event Date	Docket LER No.	Event Description	Cause Description	Comments
6/17/89	458	Mispositioned pressure transmitter isolation valve found misaligned causing inability to sense drywell pressure a condition prohibited by Tech. Spec. 3.0.4.	On 6/17/89, with the unit in operational condition 4 (cold shutdown), while performing a safety system valve line-up, a pressure transmitter root valve for the Penetration Valve Leakage Control System (PVLCS) was found to be closed. This caused one division of the PVLCS to be inoperable. Investigation determined that this valve had probably been mispositioned since the conclusion of the primary containment integrated leak rate test on 5/30/89. At 2230 on 6/15/89, the plant entered a mode of operation with both divisions of PVLCS required to be operable. This was a violation of Tech. Spec. 3.0.4. This condition would not have prevented the PVLCS from being initiated. However, this may have prevented this division from supplying adequate sealing pressure to the operable branch lines in the event a branch line experienced a low pressure condition. The probable cause for this misalignment was a personnel error in conjunction with a misapplication of a test tagging procedure.	No indication that this event could be a precursor for potential common cause failure of the PVLCS.
11/20/89	278	Miscalibration of reactor level transmitters result in technical specification violation.	On 9/11/90 was discovered during the performance of a surveillance test that level transmitter (LT) 3-2-3-99D was out of calibration causing level indicating switch (LIS) 3-2-3-99D trip setpoint to exceed technical specification limits. On 9/25/90, LT 3-2-3-99C was found similarly out of calibration causing LIS 3-2-3-99C trip setpoint to exceed technical specification limits. LT/LIS 3-2-3-99C and D are two of four instruments loops which provide a group 1 Primary Containment Isolation System (PCIS) signal on triple low reactor water level. The other two instrument loops were functional. LT/LIS 3-2-3-99C and D are believed to have been out of calibration since their last calibration during the Unit 3 seventh refueling outage.	This event indicates a potential precursor for common cause failure of the RPV level indication for PCIS actuation. However, it should be noted that not all channels were disabled. Refer to additional discussion of this failure mode in the associated text of this report section.

Table 3.3.3-2
Potential Common Cause Human Errors

Event Date	Docket LER No.	Event Description	Cause Description	Comments
3/20/90	293	General Electric type AK-2A-50 circuit breaker did not open during planned bus transfer while shutdown.	On March 20, 1990 at 1750 hours, a 480V AC load center circuit breaker that is part of a safety-related transfer scheme did not open automatically as designed during a planned bus B6 transfer. Breaker 52-202, type AK-2A-50 modified with a micro-versa trip unit, was manufactured by the General Electric Company. The failure of 52-202 to open resulted in the failure of its trip coil. In response, bus B2 was intentionally de-energized and re-energized at 1825 hours after breaker 52-202 was tripped using its local trip button after the breaker's latch prop was manually realigned. Because bus B2 was de-energized, portions of the primary and secondary containment isolation systems isolated, and shutdown cooling and salt service water cooling were interrupted for approximately 37 minutes. Breaker 52-202 failed to open because its latch mechanism was misaligned due to the absence of a retainer ring. The cause of the missing retainer ring could not be determined with certainty.	No indication of common cause failure of this type of breaker. It is assumed that the existing data base includes this event in the development of failure rate data for this component.
16/90	271	APRM miscalibration due to personnel error.	On October 29, 1990 an engineering review of APRM (EIIS=IG) calibration data obtained during plant startup identified a miscalibration at 1156 and 1254 hours on October 16, 1990 with the plant at 20% power. The Average Power Range Monitors (APRMs) were miscalibrated lower than required in Tech. Spec. sections 2.1.A.1.A, 2.1.B.1 and 3.1.B. The root cause of the miscalibration was due to personnel error on the part of the technician performing the calibrations.	This event indicates a potential precursor for common cause failure of the APRMs. However, the failure of this system in the mode suggested is not explicitly considered in the NMP2 PRA.

Section 3.3.4

Common Cause Failure Data

3.3.4 Common Cause Failure Data

This section describes the common cause modeling of similar components within a system. The general methodology associated with identifying and modeling inter-system dependencies is described in section 2.3. The evaluation of common cause failure due to equipment misalignment and miscalibration is considered as described in Section 3.3.3.

Common cause failures of multiple redundant equipment occurs from three main causes:

1) Inadequate design or equipment qualification

2) Improper maintenance or testing

3) Equipment aging.

Each of the above common causes of equipment failure has occurred in industry. These occurrences and their effect on eliminating redundancy make common cause failure modeling an important consideration in IPE. The selection of common cause failure modes to model in the NMP2 and subsequent quantification of their probability is discussed in the remainder of this section.

The identification of systems and components to model for common cause is based on the following interpretation of common cause screening presented in NUREG/CR-4780 [Reference 41]:

- Components within a system that are identical and represent redundancy in the failure logic model are considered.

- Common cause passive failures are neglected and considered to be insignificant particularly when there are active components in the system.

- Common cause modeling is neglected when it is expected to provide an insignificant contribution. For example, common cause failures of redundant actuation devices in a single train system that requires a pump to start and motor operated valves to open based on initiation signals from the redundant sources would be neglected. In this example common cause modeling of redundant actuation devices would be neglected because failure would be dominated by failure of the pump and MOV, each in single element cutsets.

- Common cause failure models for a system may be limited to major components. For example, if pump failure to start and run, and motor operated valve failure modes exist, then other less frequent events may be neglected.

Using the above considerations, the systems modeled in the NMP2 IPE were reviewed for potential common cause failure modes. Review of the standby liquid control system (SLS) resulted in no common cause failure modeling. Since the IPE model requires each train of SLS to operate, common cause failure of the trains is unimportant. System failure is dominated by independent equipment failures. Review of the redundant trains of residual heat removal (RHR) produced a number of common caüse failure modes. Redundant pumps, MOVs, pressure switches, check valves, and relays all present the potential for common cause. Similar consideration was given to the remainder of IPE scope systems.

Table 3.3.4-1 summarizes failure modes modeled for systems where common cause was deemed important. The table also identifies the number of components in the common cause group and data variables used in the quantification. The parameter model used to quantify common cause failures within a system is the Multiple Greek Letter (MGL) method. The MGL method is described in the PLG Common Cause database [Reference 39]. Basically, the MGL method assigns a common cause parameter to groups of like components. The MGL parameter represents the portion of component failure probability that is shared among components. This establishes a relationship between independent component failure probability and component group failure probability.

The first four parameters of the MGL model are:

- **Total** = Total failure frequency due to both independent common cause failures.
- **Beta** = conditional probability that the cause of a component failure will be shared by one or more additional components given that one component in a group has failed.
- **Gamma** = conditional probability that the cause of a component failure will be shared by one or more additional components given that two components in a group has failed.
- **Delta** = conditional probability that the cause of a component failure will be shared by one or more additional components given that three components in a group has failed.

The above model has different effects depending on the number of components in the common cause group and modeling assumptions. For example, the BETA factor when relating to the two emergency diesel generators models a 2 of 2 failure mode indicating that both EDGs fail in common cause. The BETA factor when referring to service water pumps refers to a 2 of 6 common cause failure mode indicating that a group of 2 components fail among a population of six pumps. In cases where several components can fail in common cause, a simplification is made to ensure that the model can be quantified. In these cases, less than the ideal number of MGL parameters is used. When this occurs the failure of all components is equated to the highest order MGL parameter. For example, service water pump modeling only employs BETA, GAMMA, and DELTA factors indicating that groups of two, three and four pumps are modeled. In this case, the DELTA factor is used for common cause quantification of four or more pumps failing in common cause. Thus, the conditional probability associated with five and six pumps failing is not modeled. This type of conservatism is not considered to significantly alter results.

The source of data for generic event descriptions and classification was the PLG generic common cause database [Reference 39]. This reference describes the development of the distributions and individual sources for each variable. The PLG generic events database covers several hundred years of operating experience for components in PWRs and BWRs. The generic screening was performed by a team of PLG PRA experts having a broad range of expertise and background including operation, systems analysis, data analysis and common cause failure experts. NMP2, being a relatively new plant, has very little common cause failure experience and, as such, there is no reason to believe that the PLG CCF Database does not apply. Table 3.3.4-2 shows the distributions for common cause parameters used for the system analysis.

Table 3.3.4-1
Common-Cause Modeling Summary

System	Event Tree Top Events	Component Group	Failure Mode	Number of Components	Data Base Variables			
					Total	Beta	Gamma	Delta
AC Power	A1/A2	Emergency Diesels	Fail to Start	2	GAZS1	GAZSB		
			Fail to Operate	2	GAZR1	GAZRB		
			Ventilation Fan Run	2	FFZRB			
			Vent Fan Start	2	FFZSB			
		Circuit Breakers	Fail to Open	2	CAZO1	CAZDB		
			Fail to Close	2	CAZP1	CAZDB		
		SWP Motor Op. Valves	Fail to Open	2	VMZD1	VMZDB		
		SWP Check Valves	Fail to Open	2	VCZO1	VJZOB		
	UA/UB	UPS	Fails to Switch	2	EUZD1	EUZDB		
ECCS Actuation	E1/E2	Relays	Fail to Close	3	RAZD1	RAZDB	RAZDG	
Service Water	SA/SB	Pumps	Fail to Start	6	PDZS1	PDZSB	PDZSG	PDZSD
			Fail to Run	6	PDZR1	PDZRB	PDZRG	PDZRD
		Pump Motor Op. Valves	Fail to Close	4	VMZD1	VMZDB	VMZDG	
			Fail to Open	6	VMZD1	VMZDB	VMZDG	
		Pump E/F Check Valve	Fails to Open	2	VCZO1	VCZOB		
RBCLC	RW	Main Pumps	Fail to Run	3	PDZR1	PDZRB	PDZRG	
		Booster Pumps	Fail to Run	3	PDZR1	PDZRB	PDZRG	
TBCLC	TW	Pumps	Fail to Run	3	PDZR1	PDZRB	PDZRG	
Instrument Air	AS	Compressors	Fail to Start	2	FCZS1	FCZSB		
			Fail to Run	3	FCZR1	FCZRB	FCZRG	

Table 3.3.4-1
Common-Cause Modeling Summary

System	Event Tree Top Events	Component Group	Failure Mode	Number of Components	Data Base Variables			
					Total	Beta	Gamma	Delta
SRVs	SV/OE	SRVs	Fail to Open	7	VPZO1	VPZOB	VPZOG	VPZOD
		Nitrogen SOVs	Fail to Open	8	VSZD1	VSZDB		
		SRV Solenoid Valves	Fail to Open	14	VSZD1	VSZDB		
	O1/O2/O3	SRVs	Fail to Open	7	VPZD1	VPZDB	VPZDG	VPZDD
		SRV Solenoid Valves	Fail to Open	14	VSZD1	VSZDB		
RHR	LA/LB	Pumps	Fail to Start	2	PCZS1	PCZSB		
			Fail to Run	2	PCZR1	PCZRB		
		Pump Check Valves	Fail to Open	4	VCZO1	VCZOB		
	HA/HB	Motor Op. Valves	Fail to Close	2	VMZD1	VMZDB		
		SWP Motor Op. Valves	Fail to Open	4	VMZD1	VMZDB	VMZDG	VMZDD
	IA/IB	Motor Op. Valves	Fail to Open	2	VMZD1	VMZDB		
		Check Valves	Fail to Open	2	VCZO1	VCZOB		
		Pressure Switches	Fail on Demand	2	ISZD1	ISZDB		
		Relays	Fail on Demand	2	RAZD1	RAZDB		
	PA/PB	Motor Op. Valves	Fail to Open	2	VMZD1	VMZDB		
	CA/CB	Motor Op. Valves	Fail to Open	6	VMZD1	VMZDB		
RRCS	RT	Breakers	Fail to Open	8	CAZD1	CAZDB		
	FT	Relays	Fail to Close	6	RAZD1	RAZDB		
	RI	Solenoid Valves	Fail to Open	8	VSZD1	VSZDB	VSZDG	VSZDD

Table 3.3.4-1
Common-Cause Modeling Summary

System	Event Tree Top Events	Component Group	Failure Mode	Number of Components	Data Base Variables			
					Total	Beta	Gamma	Delta
Containment Isolation	IS	Air Op. Valves	Fail to Close	8	VMZD1	VMZDB		
		Solenoid Valves	Fail to Close	4	VSZD1	VSZDB		
		Motor Op. Valves	Fail to Close	8	VMZD1	VMZDB		

Table 3.3.4-2
MGL Parameter Distributions

Database Designator	Component and Failure Mode	Mean	5th Percentile	Median	95th Percentile
CAZDB	Circuit Breaker fails on demand -- BETA	6.99E-2	5.86E-4	5.46E-2	1.57E-1
FCZRB	Air Compressor fails to run -- BETA	1.00E-3	1.07E-5	5.93E-4	2.79E-3
FCZRG	Air Compressor fails to run -- GAMMA	6.99E-2	5.86E-4	5.46E-2	1.57E-1
FCZSB	Air Compressor fails to start -- BETA	1.00E-3	8.83E-4	7.60E-4	2.30E-3
FFZRB	Ventilation Fan Fails to Run -- BETA	1.00E-2	8.83E-4	7.60E-3	2.30E-2
FFZSB	Ventilation Fan Fails to Start -- BETA	7.00E-2	5.86E-4	5.46E-2	1.57E-1
GAZRB	Diesel Generator fails to run -- BETA	1.00E-2	8.83E-4	7.60E-3	2.30E-2
GAZSB	Diesel Generator fails to start -- BETA	1.00E-3	1.073E-5	5.93E-4	2.79E-3
ISZDB	Switch fails on demand -- BETA	7.00E-2	5.85E-4	5.46E-2	1.57E-1
PCZSB	Normally Standby Centrifugal pump fails to start -- BETA	1.14E-2	1.68E-4	8.94E-3	2.52E-2
PCZRB	Normally Standby Centrifugal pump fails to run -- BETA	7.00E-2	5.86E-4	5.46E-2	1.57E-1
PDZSB	Norm. Operating Centrifugal pump fails to start -- BETA	1.00E-2	8.83E-4	7.60E-3	2.30E-2
PDZSG	Norm Operating Centrifugal pump fails to start -- GAMMA	1.40E-2	3.40E-2	1.27E-1	2.45E-1
PDZSD	Norm Operating Centrifugal pump fails to start -- DELTA	1.78E-1	1.32E-2	1.49E-1	3.68E-1
PDZRB	Normally Operating Centrifugal pump fails to run -- BETA	1.00E-3	1.07E-5	5.94E-4	2.79E-3
PDZRG	Norm Operating Centrifugal pump fails to run -- GAMMA	7.00E-2	5.85E-4	5.46E-2	1.57E-1
PDZRD	Normally Operating Centrifugal pump fails to run -- DELTA	1.78E-1	1.32E-2	1.49E-1	3.68E-1
RAZDB	Relay fails on demand -- BETA	7.00E-2	5.85E-4	5.46E-2	1.57E-1

Table 3.3.4-2
MGL Parameter Distributions

Database Designator	Component and Failure Mode	Mean	5th Percentile	Median	95th Percentile
VCZOB	Check valve fails to open on demand -- BETA	1.00E-2	8.83E-4	7.60E-3	2.30E-2
VMZDB	MOV fails on demand -- BETA	7.00E-2	5.85E-4	5.46E-2	1.57E-1
VMZDG	MOV fails on demand -- GAMMA	1.40E-1	3.40E-2	1.27E-1	2.45E-1
VMZDD	MOV fails on demand -- DELTA	1.78E-1	1.32E-2	1.49E-1	3.68E-1
VPZDB	SRV fails to open on demand -- BETA	1.00E-2	8.83E-4	7.60E-3	2.30E-2
VPZDG	SRV fails to open on demand -- GAMMA	7.00E-2	5.85E-4	5.46E-2	1.57E-1
VPZDD	SRV fails to open on demand -- DELTA	1.78E-1	1.32E-2	1.49E-1	3.68E-1
VSZDB	Solenoid valve fails on demand -- BETA	7.00E-2	5.85E-4	5.46E-2	1.57E-1
VSZDG	Solenoid valve fails on demand -- GAMMA	7.00E-2	5.85E-4	5.46E-2	1.57E-1
VSZDD	Solenoid valve fails on demand -- DELTA	2.84E-1	7.88E-2	2.65E-1	4.71E-1

Section 3.3.5

Quantification of Unavailability of Systems and Functions

3.3.5 Quantification of Unavailability of Systems and Functions

The system models are quantified using the RISKMAN code and the database described in Sections 3.3.1, 3.3.2, 3.3.3 and 3.3.4. The method is a standard Boolean reduction quantification. Split fractions represent the probability that a system works under given conditions when called upon in an accident. Results for each split fraction are presented in Table 3.3.5-1. RISKMAN stores the mean value of each split fraction in the master frequency file. This file is used as input for the event tree quantification. The results of systems analysis for initiating event frequency are discussed in Section 3.1.1.

There are dummy top events in Table 3.3.5-1 that do not appear in event trees. These are created when there are shared dependencies between top events and quantification of the second top is conditional on success or failure of the first. To perform these conditional calculations, a system model containing both top events is required and becomes the dummy top event. The following is an example:

Top Event	Description
A1	Emergency AC Div. I
A2	Emergency AC Div. II
A3	Emergency AC Div. I and II

Top events A1 and A2 are modeled in the SUPPORT event tree. Top event A3 is a dummy top event used to quantify top event A2. A3 contains common cause failures of equipment in A1 and A2. The shared dependency between A1 and A2 is common cause. A fault tree is developed and quantified for both A1 and A3. Then A2 is quantified conditional to A1 success or failure. If there is symmetry between A1 and A2, then the following is an example of how A2 is quantified for the split fraction boundary condition A11:

$$A21_{(\text{given A1 success})} = \frac{A11 - A31}{1 - A11}$$

$$A22_{(\text{given A1 failure})} = \frac{A31}{A11}$$

Table 3.3.5-1
System Split Fractions

SF Name	Top	SF Value	Split Fraction Description
A11	A1	9.0039E-05	KA=S
A12	A1	5.2668E-02	KA=F
A1F	A1	1.0000E+00	GAR. FAILURE
A21	A2	9.0030E-05	KA=S*KB=S*A1=S
A22	A2	9.0020E-05	KA=F*KB=S*A1=S
A23	A2	5.2670E-02	KA=S*KB=F*A1=S
A24	A2	5.1990E-02	KA=F*KB=F*A1=S
A25	A2	8.9750E-05	KA=S*KB=S*A1=F
A26	A2	9.0130E-05	KA=F*KB=S*A1=F
A27	A2	5.2720E-02	KA=S*KB=F*A1=F
A28	A2	6.4910E-02	KA=F*KB=F*A1=F
A29	A2	5.2670E-02	KA=F*KR=F*A1=F*DA=F
A2F	A2	1.0000E+00	GAR FAILURE
A31	A3	8.0787E-09	KA=S*KB=S
A32	A3	4.7463E-06	(KA=S*KB=F)+(KA=F*KB=S)
A33	A3	3.4189E-03	KA=F*KB=F
AI1	AI	5.6100E-03	IC=S
AI2	AI	5.6100E-03	IC=F
AIF	AI	1.0000E+00	GAR FAILURE
AS1	AS	2.9810E-04	ALL SUPPORT AVAILABLE
AS2	AS	1.5991E-02	NA=F*NB=S*RW=S
AS3	AS	1.4750E-03	NA=S*NB=F*RW=S
ASF	AS	1.0000E+00	NA=F*NB=F+RW=F
C11	C1	2.6158E-04	D1=S
C1F	C1	1.0000E+00	D1=F
C21	C2	2.6158E-04	D2=S
C2F	C2	1.0000E+00	D2=F
CA1	CA	4.9064E-03	A1=S
CAF	CA	1.0000E+00	A1=F
CAS	CA	0.0000E+00	GUARANTEED SUCCESS
CB1	CB	4.7040E-03	A2=S*CA=S
CBB	CB	4.5860E-02	A2=S*CA=F
CBF	CB	1.0000E+00	A2=F
CC1	CC	2.2500E-04	A1=S*A2=S
CE1	CE	5.1000E-04	CONTAINMENT ISOLATED AND INTACT
CE2	CE	1.5000E-03	CONTAINMENT ISOLATED AND INTACT
CE3	CE	5.0000E-04	CONTAINMENT ISOLATED AND INTACT
CE4	CE	4.6000E-03	CONTAINMENT ISOLATED AND INTACT
CEF	CE	1.0000E+00	CONTAINMENT ISOLATED AND INTACT
CES	CE	0.0000E+00	CONTAINMENT ISOLATED AND INTACT
CF1	CF	8.6076E-02	FW=S*NA=S*NB=S*RW=S*TA=S*TB=S
CF2	CF	5.1452E-01	FW=F*NA=S*NB=S*RW=S*TA=S*TB=S
CF3	CF	8.8144E-02	FW=S*(NA=F+NB=F+RW=F)*TA=S*TB=S
CF4	CF	5.2734E-01	FW=F*(NA=F+NB=F+RW=F)*TA=S*TB=S
CFF	CF	1.0000E+00	TA=F+TB=F
CH1	CH	4.6000E-02	OE=S
CH2	CH	1.0000E+00	OE=F
CHF	CH	1.0000E+00	AI=F
CI1	CI	5.6470E-02	KA=S
CI2	CI	1.7350E-01	KA=F

Table 3.3.5-1
System Split Fractions

SF Name	Top	SF Value	Split Fraction Description
CI3	CI	1.7350E-01	STATION BLACKOUT
CIF	CI	1.0000E+00	GUARANTEED FAILURE
CIS	CI	0.0000E+00	GUARANTEED SUCCESS
CL1	CL	1.0961E-02	ALL SUPPORT
CL2	CL	1.0971E-02	NB=F
CLF	CL	1.0000E+00	A1=F+NA=F+A2=F+DC1=F
CN1	CN	1.6750E-03	NON-ATWS, NON-MSIV
CN2	CN	1.0168E-01	NON-ATWS, MSIV CLOSURE
CN3	CN	1.0000E+00	ATWS, MSIV CLOSURE
CNF	CN	1.0000E+00	AS=F+N2=F+NA=F+NB=F+CN=F(IE)
CTL	CT	5.0000E-01	CET3-CONT REMAINS INTACT
CV1	CV	2.8422E-02	A1=S*A2=S*AS=S*N2=S
CV2	CV	3.6422E-02	(A1=F+AS=F)*A2=S*N2=S
CV3	CV	5.1820E-01	BLACKOUT, 1 EDG RECOVERED
CV5	CV	3.1122E-02	AS=S*(N2=F+A1=F)
CVF	CV	1.0000E+00	A2=F+N2=F*AS=F
CVS	CV	0.0000E+00	GUARANTEED SUCCESS
CX1	CX	4.5000E-01	CONTAIMENT FLOODING
CX2	CX	1.0000E-06	CONTAIMENT FLOODING
CY1	CY	4.5000E-01	CONTAIMENT FLOODING
CY2	CY	1.0000E-06	CONTAIMENT FLOODING
CYF	CY	1.0000E+00	CONTAIMENT FLOODING
CZA	CZ	5.1000E-04	CONTAINMENT ISOLATED AND INTACT
CZB	CZ	5.5000E-03	CONTAINMENT ISOLATED AND INTACT
CZC	CZ	1.5000E-03	CONTAINMENT ISOLATED AND INTACT
CZD	CZ	6.5000E-03	CONTAINMENT ISOLATED AND INTACT
CZE	CZ	5.0000E-04	CONTAINMENT ISOLATED AND INTACT
CZF	CZ	1.0000E+00	CONTAINMENT ISOLATED AND INTACT
CZG	CZ	5.4000E-03	CONTAINMENT ISOLATED AND INTACT
CZH	CZ	4.6000E-03	CONTAINMENT ISOLATED AND INTACT
CZJ	CZ	9.6000E-03	CONTAINMENT ISOLATED AND INTACT
CZM	CZ	4.6000E-03	CONTAINMENT ISOLATED AND INTACT
CZN	CZ	1.0000E-01	CONTAINMENT ISOLATED AND INTACT
CZS	CZ	0.0000E+00	CONTAINMENT ISOLATED AND INTACT
D11	D1	1.1821E-05	ALL SUPPORT AVAILABLE
D12	D1	7.1877E-05	A1 FAILED
D1F	D1	1.0000E+00	GF
D21	D2	1.1820E-05	A1=S*A2=S*D1=S
D22	D2	1.1820E-05	A1=F*A2=S*D1=S
D23	D2	7.1880E-05	A1=S*A2=F*D1=S
D24	D2	7.1880E-05	A1=F*A2=F*D1=S
D25	D2	1.1820E-05	A1=S*A2=S*D1=F
D26	D2	1.1820E-05	A1=F*A2=S*D1=F
D27	D2	7.1880E-05	A1=S*A2=F*D1=F
D28	D2	6.6120E-05	A1=F*A2=F*D1=F
D2F	D2	1.0000E+00	GUARANTEED FAILURE
D31	D3	1.3973E-10	A1=S*A2=S
D32	D3	8.4979E-10	A1=F+A2=F
D33	D3	4.7525E-09	A1=F*A2=F
DA1	DA	6.5810E-04	BATTERY DEMAND FAILURE

Table 3.3.5-1
System Split Fractions

SF Name	Top	SF Value	Split Fraction Description
DB1	DB	6.5810E-04	DIV II BATTERY DEMAND FAILURE
DB2	DB	6.5810E-04	DIV II BATTERY
DC1	DC	1.0000E-02	LOCATION OF CONTAINMENT BREACH -- DRYWELL
DC2	DC	3.8000E-01	LOCATION OF CONTAINMENT BREACH -- DRYWELL
DC3	DC	1.0000E-03	LOCATION OF CONTAINMENT BREACH -- DRYWELL
DC4	DC	3.8000E-01	LOCATION OF CONTAINMENT BREACH -- DRYWELL
DCF	DC	1.0000E+00	LOCATION OF CONTAINMENT BREACH -- DRYWELL
DCS	DC	0.0000E+00	LOCATION OF CONTAINMENT BREACH -- DRYWELL
DI1	DI	0.0000E+00	LOCATION OF CONTAINMENT BREACH -- DRYWELL
DI2	DI	3.8000E-01	LOCATION OF CONTAINMENT BREACH -- DRYWELL
DI3	DI	9.9000E-01	LOCATION OF CONTAINMENT BREACH -- DRYWELL
DI4	DI	4.2000E-01	LOCATION OF CONTAINMENT BREACH -- DRYWELL
DIF	DI	1.0000E+00	LOCATION OF CONTAINMENT BREACH -- DRYWELL
DIS	DI	0.0000E+00	LOCATION OF CONTAINMENT BREACH -- DRYWELL
E11	E1	1.5025E-03	TRANSIENT
E12	E1	1.0501E-03	LOCA
E1F	E1	1.0000E+00	GAR. FAILURE
E21	E2	1.4790E-03	TRANSIENT*E1=S
E22	E2	1.0260E-03	LOCA*E1=S
E2A	E2	1.7380E-02	TRANSIENT*E1=F
E2B	E2	2.3970E-02	LOCA*E1=F
E2F	E2	1.0000E+00	GUARANTEED FAILURE
E31	E3	2.6115E-05	TRANSIENT
E32	E3	2.5174E-05	LOCA
ELF	EL	1.0000E+00	CET2 SWITCH FOR PASSING CET1 SEQUENCES
ELS	EL	0.0000E+00	CET2 SWITCH FOR PASSING CET1 SEQUENCES
FB1	FB	1.1000E-01	CET2 CONT FLOOD & VENT
FB2	FB	1.5000E-01	CET2 CONT FLOOD & VENT
FBA	FB	1.3000E-01	CET2 CONT FLOOD & VENT
FBF	FB	1.0000E+00	GUARANTEED FAILURE
FBS	FB	0.0000E+00	GUARANTEED SUCCESS
FC1	FC	5.0000E-02	CONTAIMENT FLOODING
FC2	FC	.1.0000E+00	CONTAIMENT FLOODING
FC3	FC	5.0000E-02	CONTAIMENT FLOODING
FCB	FC	4.0000E-02	CONTAIMENT FLOODING
FCC	FC	1.0000E+00	CONTAIMENT FLOODING
FCD	FC	5.0000E-02	CONTAIMENT FLOODING
FCF	FC	1.0000E+00	CONTAIMENT FLOODING
FCS	FC	0.0000E+00	CONTAIMENT FLOODING
FD1	FD	1.5000E-01	CET1 CONT FLOOD & VENT
FD2	FD	2.0000E-01	CET1 CONT FLOOD & VENT
FD3	FD	1.5000E-01	CET1 CONT FLOOD & VENT
FDF	FD	1.0000E+00	GUARANTEED FAILURE
FDS	FD	0.0000E+00	GUARANTEED SUCCESS
FI1	FI	2.0000E-02	CONTAIMENT FLOODING
FI2	FI	2.0000E-01	CONTAIMENT FLOODING
FI3	FI	1.0000E+00	CONTAIMENT FLOODING
FI4	FI	2.0000E-02	CONTAIMENT FLOODING
FI5	FI	1.0000E+00	CONTAIMENT FLOODING
FI6	FI	2.0000E-02	CONTAIMENT FLOODING

Table 3.3.5-1
System Split Fractions

SF Name	Top	SF Value	Split Fraction Description
FI7	FI	1.0000E-01	CONTAIMENT FLOODING
FIF	FI	1.0000E+00	CONTAIMENT FLOODING
FIS	FI	0.0000E+00	CONTAIMENT FLOODING
FP1	FP	9.9429E-06	NA=S
FP2	FP	1.6621E-03	NA=F
FPF	FP	1.0000E+00	IA=F*IB=F
FT1	FT	1.1421E-04	C1=S*C2=S
FT2	FT	1.1421E-04	C1=F+C2=F
FTF	FT	1.0000E+00	GUARANTEED FAILURE
FTS	FT	0.0000E+00	GUARANTEED SUCCESS
FW1	FW	1.7500E-04	TRANSIENT
FW2	FW	5.6750E-03	SLOCA
FW3	FW	5.0018E-01	ATWS
FWF	FW	1.0000E+00	GUARANTEED FAILURE
FWS	FW	0.0000E+00	GUARANTEED SUCCESS
G11	G1	9.1000E-01	RECOVERY OF EDG (SBO TIME PHASE 1)
G12	G1	9.3000E-01	RECOVERY OF EDG (SBO TIME PHASE 1)
G1F	G1	1.0000E+00	RECOVERY OF EDG (SBO TIME PHASE 1)
G21	G2	7.6000E-01	RECOVERY OF EDG (SBO TIME PHASE 2)
G22	G2	8.4000E-01	RECOVERY OF EDG (SBO TIME PHASE 2)
G2F	G2	1.0000E+00	RECOVERY OF EDG (SBO TIME PHASE 2)
G31	G3	4.0000E-01	RECOVERY OF EDG (SBO TIME PHASE 3)
G32	G3	6.0000E-01	RECOVERY OF EDG (SBO TIME PHASE 3)
G3F	G3	1.0000E+00	RECOVERY OF EDG (SBO TIME PHASE 3)
G41	G4	6.1000E-01	RECOVERY OF EDG (SBO TIME PHASE 4)
G42	G4	7.7000E-01	RECOVERY OF EDG (SBO TIME PHASE 4)
G4F	G4	1.0000E+00	RECOVERY OF EDG (SBO TIME PHASE 4)
G51	G5	2.0000E-01	RECOVERY OF EDG (SBO TIME PHASE 5)
G52	G5	4.1000E-01	RECOVERY OF EDG (SBO TIME PHASE 5)
G5F	G5	1.0000E+00	RECOVERY OF EDG (SBO TIME PHASE 5)
GV2	GV	9.9000E-01	CET1 COMB GAS VENT
GV3	GV	9.9000E-01	CET1 COMB GAS VENT
GVF	GV	1.0000E+00	GUARANTEED FAILURE
GVS	GV	0.0000E+00	GUARANTEED SUCCESS
HA1	HA	1.6058E-04	RHR HEAT EXCHANGER FAILS
HAF	HA	1.0000E+00	A1=F+SA=F
HB1	HB	1.5770E-04	A1=S*A2=S*SA=S*SB=S*HA=S
HBA	HB	1.8020E-02	A1=S*A2=S*SA=S*SB=S*HA=F
HBB	HB	1.6060E-04	HA NOT ASKED
HBF	HB	1.0000E+00	GAR FAILURE
HC1	HC	2.8937E-06	A1=S*A2=S*SA=S*SB=S
HE1	HE	1.0000E-02	Rx Bldg Env IMPACT ON LP ECCS
HEF	HE	1.0000E+00	GUARANTEED FAILURE
HES	HE	0.0000E+00	GUARANTEED SUCCESS
HR1	HR	1.0000E-02	CONTAINMENT HEAT REMOVAL -- RHR
HR2	HR	1.0000E-02	CONTAINMENT HEAT REMOVAL -- RHR
HRF	HR	1.0000E+00	CONTAINMENT HEAT REMOVAL -- RHR
HRS	HR	0.0000E+00	CONTAINMENT HEAT REMOVAL -- RHR
HS1	HS	2.8007E-02	(KA=S*KR=S)*SA=S*SB=S
HS2	HS	1.4322E-01	(KA=F*KR=F)*SA=S*SB=S

Table 3.3.5-1
System Split Fractions

SF Name	Top	SF Value	Split Fraction Description
HS3	HS	3.1473E-02	(KA=S*KR=S)*(SA=F+SB=F)
HS4	HS	1.4848E-01	(KA=F*KR=F)*(SA=F+SB=F)
HS5	HS	1.0000E+00	ISLOCA RUPTURE
HSA	HS	3.2000E-03	HPCS & FW CRD POSSIBLE FOR IORV
HSF	HS	1.0000E+00	TB=F+(SA=F*SB=F)
I11	I1	2.8200E-01	OFFSITE POWER RECOVERY (SBO TIME PHASE 1)
I1F	I1	1.0000E+00	OFFSITE POWER RECOVERY (SBO TIME PHASE 1)
I1S	I1	0.0000E+00	OFFSITE POWER RECOVERY (SBO TIME PHASE 1)
I21	I2	4.1500E-01	OFFSITE POWER RECOVERY (SBO TIME PHASE 2)
I2F	I2	1.0000E+00	OFFSITE POWER RECOVERY (SBO TIME PHASE 2)
I31	I3	2.4800E-01	OFFSITE POWER RECOVERY
I3F	I3	1.0000E+00	GAR FAILURE
I41	I4	8.9700E-01	OFFSITE POWER RECOVERY (SBO TIME PHASE 4)
I4F	I4	1.0000E+00	OFFSITE POWER RECOVERY (SBO TIME PHASE 4)
I51	I5	7.6100E-01	OFFSITE POWER RECOVERY (SBO TIME PHASE 5)
I5F	I5	1.0000E+00	OFFSITE POWER RECOVERY (SBO TIME PHASE 5)
IA1	IA	2.0694E-03	A1=S*UA=S*E1=S
IAF	IA	1.0000E+00	A1=F+UA=F+E1=F
IB1	IB	1.9550E-03	A2=S*UB=S*E2=S*IA=S
IBA	IB	5.7130E-02	A2=S*UB=S*E2=S*IA=F
IBF	IB	1.0000E+00	A2=F+UB=F+E2=F
IC1	IC	1.6209E-01	E1=S*E2=S*UB=S*D1=S*UA=S*TA=S*TB=S*SA=S*SB=S
IC2	IC	1.6249E-01	(E1=F+E2=F)*UB=S*D1=S*UA=S*TA=S*TB=S*SA=S*S
ICF	IC	1.0000E+00	D1=F+UA=F+TA=F+TB=F+(E1=F*E2=F)+UB=F+(SA=F*SB=F)
IL1	IL	9.4500E-01	ALL SUPPORT AVAIL
ILF	IL	1.0000E+00	GUAR. FAILURE
ILS	IL	0.0000E+00	GUARANTEED SUCCESS
IR1	IR	1.0000E+00	COOLABILITY OF CORE DEBRIS WITHIN THE RPV
IR2	IR	2.0000E-01	COOLABILITY OF CORE DEBRIS WITHIN THE RPV
IR3	IR	1.0000E-01	COOLABILITY OF CORE DEBRIS WITHIN THE RPV
IR4	IR	1.0000E+00	COOLABILITY OF CORE DEBRIS WITHIN THE RPV
IR5	IR	1.0000E-01	COOLABILITY OF CORE DEBRIS WITHIN THE RPV
IR6	IR	2.0000E-04	COOLABILITY OF CORE DEBRIS WITHIN THE RPV
IR7	IR	2.0000E-04	COOLABILITY OF CORE DEBRIS WITHIN THE RPV
IRA	IR	5.6000E-01	COOLABILITY OF CORE DEBRIS WITHIN THE RPV
IRB	IR	7.0000E-01	COOLABILITY OF CORE DEBRIS WITHIN THE RPV
IRC	IR	6.3000E-01	COOLABILITY OF CORE DEBRIS WITHIN THE RPV
IRD	IR	5.2000E-01	COOLABILITY OF CORE DEBRIS WITHIN THE RPV
IRE	IR	4.0000E-01	COOLABILITY OF CORE DEBRIS WITHIN THE RPV
IRF	IR	1.0000E+00	COOLABILITY OF CORE DEBRIS WITHIN THE RPV
IRS	IR	0.0000E+00	COOLABILITY OF CORE DEBRIS WITHIN THE RPV
IS1	IS	5.2536E-03	A1=S*A2=S
IS2	IS	1.3205E-02	A1=F+A2=F
IS3	IS	1.0500E-01	STATION BLACKOUT
ISF	IS	1.0000E+00	GUARANTEED FAILURE
ISS	IS	0.0000E+00	GUARANTEED SUCCESS
IX1	IX	1.1822E-04	ALL SUPPORT AVAIL
KA1	KA	8.0670E-05	NORMAL
KAF	KA	1.0000E+00	LOSP
KB1	KB	4.8940E-05	ALL SUPPORT AVAILEQ

Table 3.3.5-1
System Split Fractions

SF Name	Top	SF Value	Split Fraction Description
KBF	KB	1.0000E+00	GAR FAILURE
KC1	KC	3.1738E-05	NORMA
KCF	KC	1.0000E+00	XXOGF=F
KR1	KR	2.1363E-02	KA=F*KB=S
KR2	KR	3.4006E-02	KA=S*KB=F
KRF	KR	1.0000E+00	KA=F*KB=F
KRS	KR	0.0000E+00	KA=S*KB=S
L31	L3	4.7591E-04	NO LLOCA
L32	L3	4.6978E-04	LLOCA
LA1	LA	1.4016E-02	NO LLOCA
LA2	LA	1.3806E-02	LLOCA
LAF	LA	1.0000E+00	A1=F+D1=F+E1=F
LB1	LB	1.3730E-02	NO LLOCA*LA=S
LB2	LB	1.3520E-02	LLOCA*LA=S
LB3	LB	1.4020E-02	LAF*NO LLOCA
LBA	LB	3.3960E-02	NO LLOCA*LA=F
LBB	LB	3.4030E-02	LLOCA*LA=F
LBC	LB	1.3810E-02	LAF*LLOCA
LBF	LB	1.0000E+00	A2=F+D2=F+E2=F
LC1	LC	1.5799E-02	NO LLOCA
LC2	LC	1.7632E-02	LLOCA
LCF	LC	1.0000E+00	MB=F+A2=F+D2=F+E2=F+UB=F
LL1	LL	5.1000E-02	BREACH > 150 GPM
LS1	LS	2.3264E-02	NOLLOCA=S*A1=S*D1=S*E1=S*UA=S*MA=S
LS2	LS	2.3488E-02	NOLLOCA=F*A1=S*D1=S*E1=S*UA=S*MA=S
LSF	LS	1.0000E+00	A1=F+D1=F+E1=F+UA=F+MA=F
MA1	MA	1.7643E-04	ALL SUPPORT AVAIL
MAA	MA	1.0018E-01	SA=F
MAF	MA	1.0000E+00	GAR. FAILURE
MB1	MB	1.7643E-04	ALL SUPPORT AVAIL
MBA	MB	1.0018E-01	SB=F
MBF	MB	1.0000E+00	GAR. FAILURE
ME1	ME	3.2000E-02	MANUAL ECCS ACTUATION
ME2	ME	3.9720E-01	E1=F*E2=F
MEF	ME	1.0000E+00	GUARANTEED FAILURE
MES	ME	0.0000E+00	GUARANTEED SUCCESS
MO1	MO	1.0000E+00	ALL SUPPORT AVAIL.
MOF	MO	1.0000E+00	GUAR. FAILURE
MS1	MS	3.0000E-04	ALL SUPPORT AVAIL.
MSF	MS	1.0000E+00	GUAR. FAILURE
MU1	MU	2.0000E-03	MAKEUP REMAINS AVAIL
MU2	MU	5.0000E-03	MAKEUP REMAINS AVAIL
MU3	MU	1.0000E-03	MAKEUP REMAINS AVAIL
MU4	MU	2.0000E-03	MAKEUP REMAINS AVAIL
MU5	MU	1.0000E-03	MAKEUP REMAINS AVAIL
MU6	MU	6.1000E-01	MAKEUP REMAINS AVAIL
MUF	MU	1.0000E+00	MAKEUP REMAINS AVAIL
MUS	MU	0.0000E+00	MAKEUP REMAINS AVAIL
N11	N1	6.6567E-03	HIGH PRESSURE NITROGEN
N1F	N1	1.0000E+00	GAR. FAILURE

Table 3.3.5-1
System Split Fractions

SF Name	Top	SF Value	Split Fraction Description
N21	N2	3.2822E-04	NA=S*N1=S
N22	N2	8.7854E-04	NA=S*N1=F
N23	N2	1.3519E-02	NA=F*N1=S
N2F	N2	1.0000E+00	NA=F
N2IE1	N2IE	3.6254E-04	ALL SUPPORT AVAIL.
NA1	NA	3.5519E-03	NORMAL NA SYSTEM
NAF	NA	1.0000E+00	GAR. FAILURE
NB1	NB	3.5321E-03	NORMAL SYSTEM NB
NBF	NB	1.0000E+00	GAR. FAILURE
NC1	NC	7.4000E-01	CONTAINMENT BREACH SIZE -- LEAKAGE
NC2	NC	1.9000E-01	CONTAINMENT BREACH SIZE -- LEAKAGE
NCF	NC	1.0000E+00	CONTAINMENT BREACH SIZE -- LEAKAGE
NCS	NC	0.0000E+00	CONTAINMENT BREACH SIZE -- LEAKAGE
NEF	NE	1.0000E+00	AT2 SWITCH FOR VESSEL FAILURE
NES	NE	0.0000E+00	AT2 SWITCH FOR NO VESSEL FAILURE
NF1	NF	1.5000E-01	CONTAINMENT BREACH SIZE -- OVERPRESSURE FAILURE
NF2	NF	1.1000E-01	CONTAINMENT BREACH SIZE -- OVERPRESSURE FAILURE
NF3	NF	7.8000E-03	CONTAINMENT BREACH SIZE -- OVERPRESSURE FAILURE
NFA	NF	7.4000E-01	CONTAINMENT BREACH SIZE -- OVERPRESSURE FAILURE
NFF	NF	1.0000E+00	CONTAINMENT BREACH SIZE -- OVERPRESSURE FAILURE
NFS	NF	0.0000E+00	CONTAINMENT BREACH SIZE -- OVERPRESSURE FAILURE
NLF	NL	1.0000E+00	SWITCH -- SUCCESS (LATE TREES)
NLS	NL	0.0000E+00	SWITCH -- SUCCESS (LATE TREES)
NMF	NM	1.0000E+00	SWITCH -- NO INJECTION (LATE TREES)
NMS	NM	0.0000E+00	SWITCH -- NO INJECTION (LATE TREES)
O11	O1	1.0010E-03	ALL SUPPORT AVAILABLE
O12	O1	1.0010E-03	D1=F+D2=S
O1F	O1	1.0000E+00	D1=F*D2=F
O21	O2	1.0010E-03	ALL SUPPORT AVAILABLE
O22	O2	1.0010E-03	D1=F+D2=F
O2F	O2	1.0000E+00	D1=F*D2=F
O31	O3	1.0010E-03	ALL SUPPORT AVAILABLE
O32	O3	1.0010E-03	D1=F+D2=F
O3F	O3	1.0000E+00	D1=F*D2=F
OA1	OA	9.9000E-03	ALL SUPPORT AVAIL.
OAF	OA	1.0000E+00	GUAR. FAILURE
OO1	OO	1.0000E-03	OPERATORS INITIATE
OO2	OO	3.0000E-03	ADS (MLOCA)
OO5	OO	1.0000E-02	ADS (ISLOCA)
OE1	OE	1.6005E-01	ALL SUPPORT AVAILABLE
OE2	OE	1.6133E-01	A1=S*A2=S*D1=F*D2=S*N2=F
OE3	OE	1.6005E-01	A1=S*A2=S*D1=S*D2=F*N2=S
OE4	OE	1.6125E-01	A1=S*A2=S*D1=S*D2=S*N2=F
OE5	OE	1.6694E-01	A1=F*A2=S*D1=S*D2=S*N2=F
OE6	OE	1.6757E-01	A1=F*A2=S*(D1=F+D2=F)*N2=F
OE7	OE	1.9114E-01	A1=S*A2=F*D1=S*D2=S*N2=F
OE8	OE	2.0577E-01	A1=S*A2=F*(D1=F+D2=F)*N2=F
OE9	OE	1.6133E-01	A1=S*A2=S*D1=S*D2=F*N2=F
OEF	OE	1.0000E+00	GAR FAILURE
OES	OE	0.0000E+00	GAR SUCCESS

Table 3.3.5-1
System Split Fractions

SF Name	Top	SF Value	Split Fraction Description
OG1	OG	1.7300E-04	LOSP
OGF	OG	1.0000E+00	GUARANTEED FAILURE
OH1	OH	1.0000E-05	OPERATOR FAILS TO ALLIGN RHR
OH2	OH	9.6000E-03	OPERATOR FAILS TO ALLIGN RHR IN ATWS
OHF	OH	1.0000E+00	GUARANTEED FAILURE
OHS	OH	0.0000E+00	GUARANTEED SUCCESS
OI1	OI	4.5500E-01	REACTOR PRESSURE STATUS
OI2	OI	5.1000E-03	REACTOR PRESSURE STATUS
OI3	OI	5.9000E-02	REACTOR PRESSURE STATUS
OI4	OI	2.0000E-01	REACTOR PRESSURE STATUS
OIF	OI	1.0000E+00	REACTOR PRESSURE STATUS
OIL	OI	5.0000E-01	REACTOR PRESSURE STATUS
OIS	OI	0.0000E+00	REACTOR PRESSURE STATUS
OP1	OP	4.5500E-01	REACTOR PRESSURE STATUS
OP2	OP	5.1000E-03	REACTOR PRESSURE STATUS
OP3	OP	5.9000E-02	REACTOR PRESSURE STATUS
OP4	OP	2.0000E-02	REACTOR PRESSURE STATUS
OP6	OP	4.5500E-01	REACTOR PRESSURE STATUS
OP8	OP	4.5500E-01	REACTOR PRESSURE STATUS
OP9	OP	8.0000E-03	REACTOR PRESSURE STATUS
OPA	OP	5.0000E-03	REACTOR PRESSURE STATUS
OPB	OP	3.0000E-03	REACTOR PRESSURE STATUS
OPF	OP	1.0000E+00	REACTOR PRESSURE STATUS
OPS	OP	0.0000E+00	REACTOR PRESSURE STATUS
OV1	OV	4.0000E-02	OPERATOR MITIGATES VAPOR SUPP FAILURE
OV2	OV	4.0000E-02	OPERATOR MITIGATES VAPOR SUPP FAILURE
P31	P3	1.1672E-04	A1=S*A2=S
PA1	PA	1.6372E-03	A1=S
PAF	PA	1.0000E+00	GUARANTEED FAILURE
PAS	PA	0.0000E+00	GUARANTEED SUCCESS
PB1	PB	1.5230E-03	A2=S*PA=S
PBA	PB	7.1300E-02	A2=S*PA=F
PBF	PB	1.0000E+00	GUARANTEED FAILURE
PBS	PB	0.0000E+00	GUARANTEED SUCCESS
QE1	QE	2.6000E-05	ALL SUPPORT AVAIL.
QEF	QE	1.0000E+00	GUAR. FAILURE
QH1	QH	4.3000E-06	ALL SUPPORT AVAIL.
QHF	QH	1.0000E+00	GUAR. FAILURE
R11	R1	2.2000E-02	AC POWER RECOVERY
R1F	R1	1.0000E+00	AC POWER RECOVERY
RB1	RB	3.0000E-01	REACTOR BUILDING EFFECTIVENESS
RB2	RB	9.0000E-01	REACTOR BUILDING EFFECTIVENESS
RB3	RB	1.0000E-02	REACTOR BUILDING EFFECTIVENESS
RB4	RB	3.0000E-01	REACTOR BUILDING EFFECTIVENESS
RB5	RB	9.5000E-01	REACTOR BUILDING EFFECTIVENESS
RB6	RB	1.0000E-02	REACTOR BUILDING EFFECTIVENESS
RB7	RB	9.9000E-01	REACTOR BUILDING EFFECTIVENESS
RBF	RB	1.0000E+00	REACTOR BUILDING EFFECTIVENESS
RBS	RB	0.0000E+00	REACTOR BUILDING EFFECTIVENESS
RHF	RH	1.0000E+00	SWITCH -- RHR (ATWS TREE)

Table 3.3.5-1
System Split Fractions

SF Name	Top	SF Value	Split Fraction Description
RHS	RH	0.0000E+00	SWITCH -- RHR (ATWS TREE)
RI1	RI	4.2584E-04	D1=S*D2=S*C1=F*C2=F
RI2	RI	1.8567E-02	(D1=F+D2=F)*C1=S*C2=S
RIF	RI	1.0000E+00	(D1=F+C1=F)*(D2=F+C2=F)
RM1	RM	3.0000E-01	REACTOR BUILDING EFFECTIVENESS
RM2	RM	9.5000E-01	REACTOR BUILDING EFFECTIVENESS
RM3	RM	1.0000E-02	REACTOR BUILDING EFFECTIVENESS
RM4	RM	3.0000E-01	REACTOR BUILDING EFFECTIVENESS
RM5	RM	9.5000E-01	REACTOR BUILDING EFFECTIVENESS
RM6	RM	1.0000E-02	REACTOR BUILDING EFFECTIVENESS
RM7	RM	9.9000E-01	REACTOR BUILDING EFFECTIVENESS
RM8	RM	1.0000E+00	REACTOR BUILDING EFFECTIVENESS
RMF	RM	1.0000E+00	REACTOR BUILDING EFFECTIVENESS
RML	RM	3.0000E-01	REACTOR BUILDING EFFECTIVENESS
RMM	RM	5.0000E-02	CET3 Rx Bldg
RMN	RM	1.0000E-01	CET3 Rx Bldg
RMS	RM	0.0000E+00	REACTOR BUILDING EFFECTIVENESS
RQ1	RQ	4.8593E-06	D1=S*D2=S
RQ2	RQ	3.0850E-05	D1=F+D2=F
RQ3	RQ	3.0300E-05	D1=F*D2=F
RQF	RQ	1.0000E+00	GUAR. FAILURE
RT1	RT	9.9782E-05	D1=S*D2=S*C1=S*C2=S
RT2	RT	2.5847E-03	(D1=F+D2=F)*C1=S*C2=S
RTF	RT	1.0000E+00	(D1=F+C1=F)*(D2=F+C2=F)
RV1	RV	5.1656E-01	ALL SUPPORT
RV2	RV	1.0000E+00	NA=F
RV3	RV	5.1656E-01	NB=F
RV4	RV	1.0000E+00	NA=F*NB=F
RVF	RV	1.0000E+00	GAR FAIL
RW1	RW	6.1812E-04	ALL SUPPORT AVAILABLE
RWF	RW	1.0000E+00	NA=F+NB=F
RX1	RX	1.0000E-01	COOLABILITY OF CORE DEBRIS WITHIN THE RPV
RX2	RX	2.0000E-01	COOLABILITY OF CORE DEBRIS WITHIN THE RPV
RX3	RX	1.0000E-02	COOLABILITY OF CORE DEBRIS WITHIN THE RPV
RX4	RX	6.0000E-04	COOLABILITY OF CORE DEBRIS WITHIN THE RPV
RXA	RX	5.6000E-01	COOLABILITY OF CORE DEBRIS WITHIN THE RPV
RXB	RX	7.0000E-01	COOLABILITY OF CORE DEBRIS WITHIN THE RPV
RXC	RX	6.3000E-01	COOLABILITY OF CORE DEBRIS WITHIN THE RPV
RXD	RX	5.2000E-01	COOLABILITY OF CORE DEBRIS WITHIN THE RPV
RXE	RX	4.0000E-01	COOLABILITY OF CORE DEBRIS WITHIN THE RPV
RXF	RX	1.0000E+00	COOLABILITY OF CORE DEBRIS WITHIN THE RPV
RXG	RX	1.0000E-02	COOLABILITY OF CORE DEBRIS WITHIN THE RPV
RXS	RX	0.0000E+00	COOLABILITY OF CORE DEBRIS WITHIN THE RPV
RXX	RX	6.0000E-04	COOLABILITY OF CORE DEBRIS WITHIN THE RPV
S11	S1	4.8045E-01	ALL SUPPORT AVAIL.
S1F	S1	1.0000E+00	GUAR. FAILURE
S21	S2	3.7165E-01	SUPPORT AVAIL.
S2F	S2	1.0000E+00	GUAR. FAILURE
S31	S3	3.7200E-01	SUPPORT AVAIL.
S3F	S3	1.0000E+00	GUAR. FAILURE

Table 3.3.5-1
System Split Fractions

SF Name	Top	SF Value	Split Fraction Description
SA1	SA	3.0120E-06	ALL SUPPORT AVAILABLE
SA2	SA	3.0300E-06	OG=S*KA=S*KB=S*A1=S*A2=S*D1=S*D2=S
SA3	SA	3.0300E-06	OG=S*KA=S*KB=S*A1=S*A2=S*D1=S*D2=F
SA4	SA	3.0750E-06	OG=S*KA=S*KB=S*A1=S*A2=S*D1=F*D2=F
SA5	SA	1.9120E-05	KA=F*KB=F*A1=S*A2=S*D1=S*D2=S
SA6	SA	1.8670E-03	KA=F*KB=F*A1=S*A2=F*D1=S*D2=F
SA7	SA	2.1070E-04	OG=S*KA=F*KB=S*A1=S*A2=S*D1=S*D2=S
SA8	SA	2.1290E-03	OG=S*KA=F*KB=S*A1=F*A2=S*D1=F*D2=S
SA9	SA	2.1210E-04	OG=S*KA=F*KB=S*A1=S*A2=S*D1=S*D2=F
SAA	SA	4.2530E-03	OG=S*KA=F*KB=S*A1=F*A2=S*D1=F*D2=F
SAB	SA	2.8970E-04	OG=S*KA=S*KB=F*A1=S*A2=F*D1=S*D2=F
SAC	SA	2.1070E-04	OG=S*KA=S*KB=F*A1=S*A2=S*D1=S*D2=S
SAD	SA	2.1210E-04	OG=S*KA=S*KB=F*A1=S*A2=S*D1=F*D2=S
SAE	SA	2.4130E-03	OG=S*KA=S*KB=F*A1=S*A2=F*D1=F*D2=F
SAF	SA	1.0000E+00	GUARANTEED FAILURE
SAG	SA	2.1290E-03	OG=S*KA=S*KB=S*A1=F*A2=S*D1=S*D2=S BUT QUANTIFIED AS IF KA=F SINCE A1=F IS SAME CONDITIONS
SAH	SA	2.8970E-04	OG=S*KA=S*KB=S*A1=S*A2=F*D1=S*D2=S BUT QUANTIFIED AS IF KB=F SINCE A2=F IS SAME CONDITIONS
SB1	SB	2.4110E-06	ALL SUPPORT AVAILABLE
SB2	SB	2.4110E-06	OG=S*KA=S*KB=S*A1=S*A2=S*D1=F*D2=S*SA=S
SB3	SB	5.0450E-06	OG=S*KA=S*KB=S*A1=S*A2=S*D1=S*D2=F*SA=S
SB4	SB	5.0450E-06	OG=S*KA=S*KB=S*A1=S*A2=S*D1=F*D2=F*SA=S
SB5	SB	1.5560E-05	KA=F*KB=F*A1=S*A2=S*D1=S*D2=S*SA=S
SB7	SB	2.1340E-06	OG=S*KA=F*KB=S*A1=S*A2=S*D1=S*D2=S*SA=S
SB8	SB	3.0120E-06	OG=S*KA=F*KB=S*A1=F*A2=S*D1=F*D2=S*SA=S
SB9	SB	2.1200E-06	OG=S*KA=F*KB=S*A1=S*A2=S*D1=S*D2=F*SA=S
SBA	SB	2.8500E-06	OG=S*KA=F*KB=S*A1=F*A2=S*D1=F*D2=F*SA=S
SBB	SB	3.7300E-06	OG=S*KA=S*KB=F*A1=S*A2=F*D1=S*D2=F*SA=S
SBC	SB	2.1350E-06	OG=S*KA=S*KB=F*A1=S*A2=S*D1=S*D2=S*SA=S
SBD	SB	2.1140E-06	OG=S*KA=S*KB=F*A1=S*A2=S*D1=F*D2=S*SA=S
SBE	SB	3.5740E-06	OG=S*KA=S*KB=F*A1=S*A2=F*D1=F*D2=F*SA=S
SBF	SB	1.0000E+00	GUARANTEED FAILURE
SBG	SB	3.0120E-06	OG=S*KA=S*KB=S*A1=F*A2=S*D1=S*D2=S*SA=S
SBH	SB	3.7300E-06	OG=S*KA=S*KB=S*A1=S*A2=F*D1=S*D2=S*SA=S
SBI	SB	2.1400E-04	INIT=SAX*D1=S
SBK	SB	9.8710E-01	OG=S*KA=S*KB=S*A1=S*A2=F*D1=S*D2=S*SA=F
SBL	SB	9.9990E-01	OG=S*KA=S*KB=S*A1=F*A2=S*D1=S*D2=S*SA=F
SBM	SB	1.0000E+00	OG=S*KA=S*KB=F*A1=S*A2=F*D1=F*D2=F*SA=F
SBN	SB	1.0000E+00	OG=S*KA=S*KB=F*A1=S*A2=S*D1=F*D2=S*SA=F
SBO	SB	1.0000E+00	OG=S*KA=S*KB=F*A1=S*A2=S*D1=S*D2=S*SA=F
SBP	SB	1.0000E+00	OG=S*KA=S*KB=F*A1=S*A2=F*D1=S*D2=F*SA=F
SBQ	SB	9.9930E-01	OG=S*KA=F*KB=S*A1=F*A2=S*D1=F*D2=F*SA=F
SBR	SB	9.9000E-01	OG=S*KA=F*KB=S*A1=S*A2=S*D1=S*D2=F*SA=F
SBS	SB	9.9860E-01	OG=S*KA=F*KB=S*A1=F*A2=S*D1=F*D2=S*SA=F
SBT	SB	9.8990E-01	OG=S*KA=F*KB=S*A1=S*A2=S*D1=S*D2=S*SA=F
SBU	SB	1.8600E-03	KA=F*KB=F*A1=F*A2=S*D1=F*D2=S*SA=F
SBV	SB	1.9200E-01	KA=F*KB=F*A1=S*A2=S*D1=S*D2=S*SA=F
SBW	SB	3.1140E-01	OG=S*KA=S*KB=S*A1=S*A2=S*D1=F*D2=F*SA=F
SBX	SB	3.0110E-01	OG=S*KA=S*KB=S*A1=S*A2=S*D1=S*D2=F*SA=F

Table 3.3.5-1
System Split Fractions

SF Name	Top	SF Value	Split Fraction Description
SBY	SB	3.0090E-01	OG=S*KA=S*KB=S*A1=S*A2=S*D1=F*D2=S*SA=F
SBZ	SB	2.9660E-01	OG=S*KA=S*KB=S*A1=S*A2=S*D1=S*D2=S*SA=F
SC1	SC1	8.9323E-07	ALL SUPPORT AVAILABLE
SC2	SC1	9.1201E-07	OG=S*KA=S*KB=S*A1=S*A2=S*D1=F*D2=S
SC3	SC1	9.1260E-07	OG=S*KA=S*KB=S*A1=S*A2=S*D1=S*D2=F
SC4	SC1	9.5762E-07	OG=S*KA=S*KB=S*A1=S*A2=S*D1=F*D2=F
SC5	SC2	3.6716E-06	KA=F*KB=F*A1=S*A2=S*D1=S*D2=S
SC6	SC2	1.8670E-03	KA=F*KB=F*A1=S*A2=F*D1=S*D2=F
SC7	SC3	2.0866E-04	OG=S*KA=S*KB=S*A1=S*A2=S*D1=S*D2=S
SC8	SC3	2.1261E-03	OG=S*KA=F*KB=S*A1=F*A2=S*D1=F*D2=S
SC9	SC3	2.1007E-04	OG=S*KA=F*KB=S*A1=S*A2=S*D1=S*D2=F
SCA	SC3	4.2497E-03	OG=S*KA=F*KB=S*A1=F*A2=S*D1=F*D2=F
SCB	SC4	2.8604E-04	OG=S*KA=S*KB=F*A1=S*A2=F*D1=S*D2=F
SCC	SC4	2.0866E-04	OG=S*KA=S*KB=F*A1=S*A2=S*D1=S*D2=S
SCD	SC4	2.1008E-04	OG=S*KA=S*KB=F*A1=S*A2=S*D1=F*D2=S
SCE	SC4	2.4096E-03	OG=S*KA=S*KB=F*A1=S*A2=F*D1=F*D2=F
SCF	SC1	1.0000E+00	GAR FAILURE
SCG	SC3	2.1261E-03	OG=S*KA=S*KB=S*A1=F*A2=S*D1=S*D2=S BUT QUANTIFIED AS IF KA=F BECAUSE A1=F IS SAME COND
SCH	SC4	2.8604E-04	OG=S*KA=S*KB=S*A1=S*A2=F*D1=S*D2=S BUT QUANTIFIED AS IF KB=F SINCE A2=F IS SIMILAR COND
SCU	SC2	1.8597E-03	OG=F*KA=F*KB=F*A1=F*A2=S*D1=F*D2=S
SD1	SD1	3.3040E-06	ALL SUPPORT AVAILABLE
SD2	SD1	3.3227E-06	OG=S*KA=S*KB=S*A1=S*A2=S*D1=F*D2=S
SD3	SD1	5.9580E-06	OG=S*KA=S*KB=S*A1=S*A2=S*D1=S*D2=F
SD4	SD1	6.0030E-06	OG=S*KA=S*KB=S*A1=S*A2=S*D1=F*D2=F
SD5	SD2	1.9235E-05	KA=F*KB=F*A1=S*A2=S*D1=S*D2=S
SD6	SD2	1.0000E+00	KA=F*KB=F*A1=S*A2=F*D1=S*D2=F
SD7	SD3	2.1079E-04	OG=S*KA=F*KB=S*A1=S*A2=S*D1=S*D2=S
SD8	SD3	2.1291E-03	OG=S*KA=F*KB=S*A1=F*A2=S*D1=F*D2=S
SD9	SD3	2.1219E-04	OG=S*KA=S*KB=S*A1=S*A2=S*D1=S*D2=F
SDA	SD3	4.2525E-03	OG=S*KA=F*KB=S*A1=F*A2=S*D1=F*D2=F
SDB	SD4	2.8977E-04	OG=S*KA=S*KB=F*A1=S*A2=F*D1=S*D2=F
SDC	SD4	2.1079E-04	OG=S*KA=S*KB=F*A1=S*A2=S*D1=S*D2=S
SDD	SD4	2.1219E-04	OG=S*KA=S*KB=F*A1=S*A2=S*D1=F*D2=S
SDE	SD4	2.4132E-03	OG=S*KA=S*KB=F*A1=S*A2=F*D1=F*D2=F
SDF	SD1	1.0000E+00	GAR FAILURE
SDG	SD3	2.1291E-03	OG=S*KA=S*KB=S*A1=F*A2=S*D1=S*D2=S BUT QUANTIFIED AS IF KA=F SINCE A1=F IS SIMILAR COND
SDH	SD4	2.8977E-04	OG=S*KA=S*KB=S*A1=S*A2=F*D1=S*D2=S BUT QUANTIFIED AS IF KB=F SINCE A2=F IS SIMILAR
SDU	SD2	1.8600E-03	OG=F*KA=F*KB=F*A1=F*A2=S*D1=F*D2=S
SE1	SE	2.1178E-06	ALL SUPPORT AVAILABLE
SF1	SF	1.2268E-08	ALL SUPPORT AVAILABLE
SG1	SG	2.1399E-04	ALL SUPPORT AVAILABLE
SL1	SL	2.8355E-02	C1=S*C2=S*A1=S*A2=S
SL2	SL	2.9987E-02	(C1=F+C2=F)*A1=S*A2=S
SLF	SL	1.0000E+00	(C1=F*C2=F)+A1=F+A2=F
SN1	SN	2.1100E-03	SUPPRESSION POOL BYPASS
SN2	SN	2.1000E-04	SUPPRESSION POOL BYPASS

Table 3.3.5-1
System Split Fractions

SF Name	Top	SF Value	Split Fraction Description
SN3	SN	8.1000E-02	SUPPRESSION POOL BYPASS
SNF	SN	1.0000E+00	SUPPRESSION POOL BYPASS
SNS	SN	0.0000E+00	SUPPRESSION POOL BYPASS
SO1	SO	1.1980E-03	RELIEF VALVE STUCK OPEN
SO2	SO	9.9840E-06	RELIEF VALVE STUCK (2)
SOF	SO	1.0000E+00	GUAR. FAILURE
SP1	SP	2.1000E-03	SUPPRESSION POOL BYPASS
SP2	SP	2.1000E-04	SUPPRESSION POOL BYPASS
SP3	SP	8.1000E-02	SUPPRESSION POOL BYPASS
SPF	SP	1.0000E+00	SUPPRESSION POOL BYPASS
SPS	SP	0.0000E+00	SUPPRESSION POOL BYPASS
SR1	SR	4.5150E-04	ADEQUATE PRESSURE RELIEF
SR2	SR	5.5330E-07	ADEQUATE PRESSURE RELIEF (3)
SRF	SR	1.0000E+00	GUAR. FAILURE
SV1	SV	1.1290E-05	ALL SUPPORT AVAILABLE
SV2	SV	9.1408E-05	A1=S*A2=S*D1=F*D2=S*N2=F
SV3	SV	1.1318E-05	A1=S*A2=S*D1=S*D2=F*N2=S
SV4	SV	8.9774E-05	A1=S*A2=S*D1=S*D2=S*N2=F
SV5	SV	5.3309E-03	A1=F*A2=S*D1=S*D2=S*N2=F
SV6	SV	5.3385E-03	A1=F*A2=S*(D1=F+D2=F)*N2=F
SV7	SV	5.5709E-03	A1=S*A2=F*D1=S*D2=S*N2=F
SV8	SV	5.8826E-03	A1=S*A2=F*(D1=F+D2=F)*N2=F
SV9	SV	9.1408E-05	A1=S*A2=S*D1=S*D2=F*N2=F
SVF	SV	1.0000E+00	GAR FAILURE (D1=F*D2=F) NOTE (A1=F*A2=F) GOES TO SBO IN TOP EVENTS 01,02,03 & X1,X2,X3
SW1	SW	4.3685E-02	A2=S*IB=S*SB=S*(DEMAND EXIST FOR OP ACTION)
SW2	SW	3.6849E-03	A2=S*IB=S*SB=S*(NO DEMAND EXIST FOR OP ACTION)
SWF	SW	1.0000E+00	A2=F+IB=F+SB=F
TA1	TA	1.0270E-05	CST A
TAF	TA	1.0000E+00	GUARANTEED FAILURE
TB1	TB	1.0270E-05	CST B
TBF	TB	1.0000E+00	GUARANTEED FAILURE
TD1	TD	2.0000E-02	COOLANT INJECTION FOR TEMP. CONTROL OF MOLTEN DEBRIS
TD2	TD	5.0000E-02	COOLANT INJECTION FOR TEMP. CONTROL OF MOLTEN DEBRIS
TD3	TD	1.0000E+00	COOLANT INJECTION FOR TEMP. CONTROL OF MOLTEN DEBRIS
TD4	TD	2.0000E-02	COOLANT INJECTION FOR TEMP. CONTROL OF MOLTEN DEBRIS
TD5	TD	1.0000E+00	COOLANT INJECTION FOR TEMP. CONTROL OF MOLTEN DEBRIS
TD6	TD	1.0000E-01	COOLANT INJECTION FOR TEMP. CONTROL OF MOLTEN DEBRIS
TD7	TD	6.3000E-04	COOLANT INJECTION FOR TEMP. CONTROL OF MOLTEN DEBRIS
TDA	TD	5.3000E-01	COOLANT INJECTION FOR TEMP. CONTROL OF MOLTEN DEBRIS
TDB	TD	2.8000E-01	COOLANT INJECTION FOR TEMP. CONTROL OF MOLTEN DEBRIS
TDC	TD	2.4000E-01	COOLANT INJECTION FOR TEMP. CONTROL OF MOLTEN DEBRIS
TDD	TD	7.6000E-02	COOLANT INJECTION FOR TEMP. CONTROL OF MOLTEN DEBRIS
TDE	TD	3.6000E-02	COOLANT INJECTION FOR TEMP. CONTROL OF MOLTEN DEBRIS
TDF	TD	1.0000E+00	COOLANT INJECTION FOR TEMP. CONTROL OF MOLTEN DEBRIS
TDJ	TD	6.3000E-04	COOLANT INJECTION FOR TEMP. CONTROL OF MOLTEN DEBRIS
TDS	TD	0.0000E+00	COOLANT INJECTION FOR TEMP. CONTROL OF MOLTEN DEBRIS
TR1	TR	6.0000E-03	COOLANT INJECTION FOR TEMP. CONTROL OF MOLTEN DEBRIS
TR2	TR	2.0000E-01	COOLANT INJECTION FOR TEMP. CONTROL OF MOLTEN DEBRIS
TR3	TR	1.0000E+00	COOLANT INJECTION FOR TEMP. CONTROL OF MOLTEN DEBRIS

Table 3.3.5-1
System Split Fractions

SF Name	Top	SF Value	Split Fraction Description
TR4	TR	1.0000E-01	COOLANT INJECTION FOR TEMP. CONTROL OF MOLTEN DEBRIS
TR5	TR	1.0000E+00	COOLANT INJECTION FOR TEMP. CONTROL OF MOLTEN DEBRIS
TR6	TR	6.0000E-04	COOLANT INJECTION FOR TEMP. CONTROL OF MOLTEN DEBRIS
TR7	TR	2.0000E-02	COOLANT INJECTION FOR TEMP. CONTROL OF MOLTEN DEBRIS
TR8	TR	5.0000E-02	COOLANT INJECTION FOR TEMP. CONTROL OF MOLTEN DEBRIS
TRA	TR	5.3000E-01	COOLANT INJECTION FOR TEMP. CONTROL OF MOLTEN DEBRIS
TRB	TR	2.8000E-01	COOLANT INJECTION FOR TEMP. CONTROL OF MOLTEN DEBRIS
TRC	TR	2.4000E-01	COOLANT INJECTION FOR TEMP. CONTROL OF MOLTEN DEBRIS
TRD	TR	7.6000E-02	COOLANT INJECTION FOR TEMP. CONTROL OF MOLTEN DEBRIS
TRE	TR	3.6000E-02	COOLANT INJECTION FOR TEMP. CONTROL OF MOLTEN DEBRIS
TRF	TR	1.0000E+00	COOLANT INJECTION FOR TEMP. CONTROL OF MOLTEN DEBRIS
TRG	TR	1.0000E-02	COOLANT INJECTION FOR TEMP. CONTROL OF MOLTEN DEBRIS
TRI	TR	1.0000E-01	COOLANT INJECTION FOR TEMP. CONTROL OF MOLTEN DEBRIS
TRS	TR	0.0000E+00	COOLANT INJECTION FOR TEMP. CONTROL OF MOLTEN DEBRIS
TRZ	TR	1.0000E-01	COOLANT INJECTION FOR TEMP. CONTROL OF MOLTEN DEBRIS
TW1	TW	6.6828E-05	ALL SUPPORT AVAILABLE
TW2	TW	5.6625E-03	NB=F*SA=S*NA=S
TWF	TW	1.0000E+00	SA=F+NA=F
U11	U1	7.5686E-02	D1=S*UA=S*TA=S*TB=S*E1=S*E2=S*UB=S
U12	U1	7.6487E-02	D1=S*UA=S*TA=S*TB=S*E1=F*E2=S*UB=S
U1F	U1	1.0000E+00	D1=F+UA=F+TA=F+TB=F+(E1=F*E2=F)+UB=F
U21	U2	1.7873E-01	ALL SUPPORT AVAIL.
U2F	U2	1.0000E+00	GUAR. FAILURE
U31	U3	5.2635E-01	ALL SUPPORT AVAIL.
U3F	U3	1.0000E+00	GUAR. FAILURE
UA1	UA	1.1820E-05	A1=S*D1=S
UA2	UA	1.9754E-05	A1=F*D1=S
UA3	UA	1.1820E-05	A1=S*D1=F
UAF	UA	1.0000E+00	GAR. FAILURE
UB1	UB	1.1820E-05	ALL SUPPORT AVAILABLE
UB2	UB	1.9760E-05	A2=F*D2=S
UB3	UB	1.1820E-05	A1=S*A2=S*D1=S*D2=F*UA=S
UB4	UB	1.1820E-05	A1=F*A2=S*D2=F
UB5	UB	1.1820E-05	A1=S*A2=S*D1=F*D2=F*UA=S
UB6	UB	1.1820E-05	A1=F*A2=S*D2=F
UB7	UB	1.1820E-05	A1=S*A2=S*D1=F*D2=S*UA=S
UBA	UB	1.1820E-05	ALL SUPPORT AVAILABLE BUT UA=F
UBB	UB	1.1820E-05	A1=S*A2=S*D1=S*D2=F*UA=F
UBC	UB	1.1820E-05	A1=S*A2=S*D1=F*D2=F*UA=F
UBD	UB	1.1820E-05	A1=S*A2=S*D1=F*D2=S*UA=F
UBF	UB	1.0000E+00	A2=F*D2=F
UC1	UC	1.3980E-10	ALL SUPPORT AVAILABLE
UC2	UC	1.3980E-10	A1=S*A2=S*D1=S*D2=F
UC3	UC	1.3980E-10	A1=S*A2=S*D1=F*D2=S
UC4	UC	1.3980E-10	A1=S*A2=S*D1=F*D2=F
VC1	VC	0.0000E+00	CET CONT VENTING
VC2	VC	3.8400E-02	CET CONT VENTING
VC3	VC	3.8400E-02	CET CONT VENTING
VCF	VC	1.0000E+00	GUARANTEED FAILURE
VCS	VC	0.0000E+00	GUARANTEED SUCCESS

Table 3.3.5-1
System Split Fractions

SF Name	Top	SF Value	Split Fraction Description
VS1	VS	9.5145E-06	ALL SUPPORT AVAIL.
VSF	VS	1.0000E+00	GUARANTEED FAILURE
VSS	VS	0.0000E+00	GUARANTEED SUCCESS
WL1	WL	1.0000E-04	IL=S
WL2	WL	1.0000E-03	IL=F
WLF	WL	1.0000E+00	GUAR. FAILURE
WR1	WR	0.0000E+00	LOCATION OF CONTAINMENT BREACH -- WETWELL
WR2	WR	8.7000E-01	LOCATION OF CONTAINMENT BREACH -- WETWELL
WR3	WR	1.0000E-06	LOCATION OF CONTAINMENT BREACH -- WETWELL
WR4	WR	9.1000E-01	LOCATION OF CONTAINMENT BREACH -- WETWELL
WRF	WR	1.0000E+00	LOCATION OF CONTAINMENT BREACH -- WETWELL
WRS	WR	0.0000E+00	LOCATION OF CONTAINMENT BREACH -- WETWELL
WW1	WW	1.0000E-01	LOCATION OF CONTAINMENT BREACH -- WETWELL
WW2	WW	0.0000E+00	LOCATION OF CONTAINMENT BREACH -- WETWELL
WW3	WW	5.4000E-01	LOCATION OF CONTAINMENT BREACH -- WETWELL
WWF	WW	1.0000E+00	LOCATION OF CONTAINMENT BREACH -- WETWELL
WWS	WW	0.0000E+00	LOCATION OF CONTAINMENT BREACH -- WETWELL
X11	X1	5.2780E-02	BLACKOUT SRVs CLOSE
X1F	X1	1.0000E+00	GAR FAILURE
X21	X2	3.2124E-01	BLACKOUT SRVs CLOSE
X2F	X2	1.0000E+00	GAR FAILURE
X31	X3	8.9900E-01	BLACKOUT SRVs CLOSE
X3F	X3	1.0000E+00	GAR FAILURE
ZC1	ZC	5.0570E-01	SYSTEM BREACH IN SHUTDOWN COOLING TRAIN
ZC2	ZC	5.3570E-01	SYSTEM BREACH IN SHUTDOWN COOLING TRAIN
ZE1	ZE	5.6000E-01	EARLY ISOLATION OF SYSTM BREACH
ZE2	ZE	5.1000E-02	EARLY ISOLATION OF SYSTM BREACH
ZE3	ZE	5.1000E-02	EARLY ISOLATION OF SYSTM BREACH
ZEF	ZE	1.0000E+00	EARLY ISOLATION OF SYSTM BREACH
ZI2	ZI	8.8800E-02	ISOLATION VALVE FAILURE IS NOT RECOVERABLE
ZI4	ZI	1.1200E-01	ISOLATION VALVE FAILURE IS NOT RECOVERABLE
ZLF	ZL	1.0000E+00	LOW PRESSURE SYSTEM INTEGRITY -- LEAKAGE <= 150 GPM
ZLS	ZL	0.0000E+00	GUARANTEED SUCCESS
ZN1	ZN	5.0000E-01	CONTAINTHENT ISOLATION BREACH IN NORTH AUX. BAY
ZR1	ZR	1.0000E-04	LOW PRESSURE SYSTEM INTEGRITY -- RUPTURE
ZR2	ZR	1.0000E-04	LOW PRESSURE SYSTEM INTEGRITY -- RUPTURE
ZR3	ZR	1.0000E-04	LOW PRESSURE SYSTEM INTEGRITY -- RUPTURE
ZR4	ZR	1.0000E-04	LOW PRESSURE SYSTEM INTEGRITY -- RUPTURE
ZRS	ZR	0.0000E+00	GUARANTEED SUCCESS
ZS1	ZS	6.6410E-01	BREACH IS IN LPCI C OR LPCS SYSTEM
ZS2	ZS	6.5120E-01	BREACH IS IN LPCI C OR LPCS SYSTEM
ZT2	ZT	5.1000E-02	LATE ISOLATION OF SYSTEM BREACH
ZT3	ZT	5.5000E-01	LATE ISOLATION OF SYSTEM BREACH
ZTF	ZT	1.0000E+00	LATE ISOLATION OF SYSTEM BREACH
ZTS	ZT	0.0000E+00	LATE ISOLATION OF SYSTEM BREACH
ZVF	ZV	1.0000E+00	SYSTEM BREACH CAN BE ISOLATED USING A SECOND MOV
ZVS	ZV	0.0000E+00	SYSTEM BREACH CAN BE ISOLATED USING A SECOND MOV

Section 3.3.6

Generation of Support System States and Quantifications of Their Probability

3.3.6 Generation of Support System States and Quantification of their Probabilities

The event tree quantification process links the support system event tree directly to the front-line event trees. As described in Section 3.3.7, rules are used to define support system success and failure where each front-line system is quantified under its applicable support system boundary conditions. Therefore, the concept of "support system states" is not applicable to the quantification process.

Section 3.3.7

Quantification of Sequence Frequencies

3.3.7 Quantification of Sequence Frequencies

The quantification of sequences is performed by linking initiating event with event trees as illustrated in Figure 3.1.2-1. Initiating events and event trees are described in Section 3.1. As shown, all initiating events pass through the SUPPORT event tree, and then to the appropriate front-line event tree(s). The frequency of each initiating event, a sequence cutoff frequency, and the logic for linking to event trees for each initiator are input to the RISKMAN code for quantification of sequences. For example, the large LOCA quantification and a list of event trees in order of linking (SUPPORT, LL1, LL2 including LLOCA initiating event frequency). The linking of these three event trees is similar to creating one very large event tree for the large LOCA initiator. A cutoff frequency of 1E-10 was used for core damage sequence quantification of all initiating events.

In each event tree, the success or failure of each top event (branch point) depends on the tree structure, top event rules, and the quantitative value assigned to the top event failure. Each failure value for a top event is referred to as a split fraction. A top event may have several split fractions due to top event dependencies on initiating events and success or failure of systems asked previously in the event tree models. The choice of split fraction for each top event during sequence quantification is based on logic rules. This is described further below.

Inter-system dependencies are summarized in Section 3.2.3. Split fraction rules are one way to account for these dependencies during sequence quantification. Another way that dependencies are included is through the tree structure itself. In this case, an earlier top event failure guarantees failure of top events later in the same tree. Therefore there is no branch point for these later top events (a pass through). This is described further below.

The large LOCA event tree (LL1) is used as an example to illustrate the use of rules and the quantification of sequences. The LL1 event tree, its top event description and the split fraction rules used to quantify the LL1 tree are provided in Figure 3.1.2.5-2. Note that the split fraction rules have a three-letter code where the first two are the same as the event tree top event and the last one identifies the specific split fraction. When the third letter is an "F," this usually represents a guaranteed failure of the top event. Within the logic rules, the symbols "+" "*" and "-" represent "OR" "AND" and "NOT" logic, respectively. "INIT=" is used to represent initiating events in the rules. "S" "F" and "B" are used to represent success, failure and bypass of the top events.

<u>Tree Structure Dependencies</u>
As shown in Figure 3.1.2.5-2, failure to scram (top event RQ failure) or vapor suppression failure (top event VS) bypass the remaining top events. These sequences are modeled as core damage events with containment failure and questioning the success of other top events can be considered not necessary or their guaranteed failure.

In addition, when top event LC (LPCI "C") is success, top events LS (Low Pressure Core Spray) and HS (High Pressure Core Spray) are bypassed because they are not necessary if LPCI "C" is successful. Likewise, if LS is success, HS is bypassed. However, note that LPCI "A" (LA and IA) and LPCI "B" (LB and IB) are questioned. The reason for this is the dual function of both injection and containment heat removal can be provided by RHR A and B. Containment heat removal is questioned in event tree LL2 and depends on the success or failure of these top events.

Split Fraction Rules
As an example, the following rules are used for top event LC:

Split Fraction	Split Fraction Logic (RULE)
LCF	ACB + D2=F + E2=F*ME=F + UB=F + MB=F
LC1	1

Top event LC has two split fractions; the first logic rule directs the quantification to use the value LCF (guaranteed failure of LPCI C) if any of the following support system failures occur in the SUPPORT event tree along the sequence being quantified:

- Unavailability of AC power train B (ACB) to the pump and valves. ACB is defined as a macro in the SUPPORT event tree which introduces another method of using logic rules. For example, ACB is defined as follows in the SUPPORT event tree:

  ACB:= A2=F + KB=F*KR=F*(D2=F + SB=F)

  where top event A2 models the emergency bus and emergency diesel when normal offsite AC power is unavailable to the bus. Loss of offsite AC power to the emergency bus (KB=F * KR=F) and failure of emergency diesel support systems; Div. II DC power (D2) or service water (SB).

- Unavailability of Div II 125V DC (D2=F) which is required to start the pump (breaker control).

- Unavailability of Div II ECCS actuation system (E2=F) and the operators fail to manually start the pump (ME=F).

- Unavailability of Div II 120V AC vital uninterruptible power (UB=F) which is required to support analog logic to open the injection MOV.

- Unavailability of room cooling (MB=F).

For each sequence, if the above support systems are available to support LPCI C operation, then the LCF split fraction rule is not satisfied. In this case, the quantification code passes to the next rule which is LC1. The split fraction LC1 represents unavailability of LPCI C given all support systems are unavailable. The "1" logic rule says to always use split fraction LC1 if previous rules did not apply.

The initiating event frequencies used in the quantification are given in Table 3.1.1-1. The event trees and logic rules used to quantify the event trees are provided in Section 3.1 as shown in Figure 3.1.2-1. The event tree top event failure fractions (split fractions) are based on human failure analysis results in Section 3.3.3 and systems analysis results in Section 3.3.5. A list of top event split fractions used to quantify the event trees is provided in Table 3.3.5-1.

The binning of sequences to SUCCESS or plant damage states is based on binning logic rules defined for the last event tree (LL2,.ML2, etc. as shown in Figure 3.1.2-1) that is linked to the initiating event. The rules are provided with the appropriate event tree (i.e., LL2).

When the containment event trees (CET) are linked on the end of each initiating event and level 1 event trees to obtain Level 2 results, the plant damage state bins are no longer required. Actually, the plant damage state binning rules are converted to macros and are used in the last frontline event tree split fraction rules, such that the CET's top event rules can use them. Binning rules are defined to collect CET sequences, which begin with the initiating event, in the appropriate release category. The CET event trees and rules are provided in Section 4.

Section 3.3.8

Internal Flood Analysis

3.3.8 Internal Flood Analysis

3.3.8.1 Introduction

An analysis has been performed to identify potential accident scenarios involving internal floods at Nine Mile Point Unit 2. Several probabilistic risk assessments have shown that spatially dependent events, such as internal floods, can contribute to core damage frequency, since these events can potentially cause an initiating event and subsequent spatially or functional dependent failure of critical systems.

The internal flood analysis methodology and model is described in this section. Additionally, supporting analysis is documented in Appendix C, including the quantification of flooding initiating event frequencies for scenarios that are included in the overall risk model.

3.3.8.2 Methodology

The methodology employs a conservative screening analysis technique to determine potential flood sources and locations, and the associated impacts on continued plant operation and the ability of the operating crew to safely shut down the plant. Postulated flood scenarios are defined in terms of the flood source, the extent of propagation to adjacent locations, and the extent of equipment damage. The frequencies of these scenarios are then quantified. Important scenarios are considered in the overall risk model, during which a more detailed analysis may be performed if the risk results are significant. The methodology is summarized below:

Plant Familiarization
Key plant design information that provides details of the plant layout is reviewed to familiarize the analysts with the location of potential flood sources and pathways available for the propagation of a flood. This includes arrangement drawings, the MELB and FHA assessments discussed in the FSAR, and internal documentation of related events that have occurred at NMP2. The PRA models are also reviewed to ensure familiarity with systems models, initiating events, intersystem dependencies, success criteria, and plant response.

Flood Experience Review
Industry data concerning actual occurrences of flooding at nuclear power facilities are collected from Nuclear Power Experience (Reference 39) are reviewed to ensure familiarization with actual flooding scenarios; specifically, their causes and effects. These data are used to postulate internal flooding scenarios that could be important for consideration in the NMP2 analysis, and to develop initiating event frequencies for those scenarios that are explicitly quantified in the plant specific evaluation.

Flood and Equipment Locations
Using this information, potential flood sources throughout the plant, and systems located in areas that, if flooded, could affect plant operation are postulated.

The following information on flood sources and locations is considered for this task:

- Fluid system piping diameter and lengths,
- Drainage systems' tank and sump configuration and capacity,
- Area flooding instrumentation,
- Potential for isolating the source,
- Equipment locations, and
- Assessment of flood propagation pathways.

Plant Walkdown

A plant walkdown is conducted to collect additional information and to confirm previous documentation and analysis of flood sources, and their potential impact, propagation pathways, and detection capability. Additionally, the relative location , with respect to flood, of plant equipment located in these areas was also confirmed during this task. A detailed scaled model was used to familiarize the team with the plant layout before the walkdown.

Scenarios and Screening

The potential for an initiating event, the propagation of the flood to other locations, and system impact associated with postulated flooding are considered in judging whether a flood scenario should be quantitatively evaluated, since flooding scenarios are initially screened using conservative assumptions about flood size and system impacts.

Usually, a flooding event must either cause an initiating event and impact an important system or cause significant system failures that could jeopardize the ability of the operating crew to safely shut down the plant (i.e., due to flood propagation), for a scenario to be modeled. For example, a flood that fails one train of ECCS, but does not cause an initiating event is not considered further; the low frequency of the flood, combined with the other required failures to cause core damage, makes it a highly improbable event. Similarly, a flood that causes a loss of feedwater, but no other failures is insignificant because the plant model already considers failures of BOP systems due to other causes at a much higher frequency.

Quantification

A point estimate of flood initiating event frequencies is developed using industry data without taking credit for operator intervention. The operating crew's ability to mitigate the event, and any consequential system damage are then evaluated in the model. The scenarios that result in successful mitigation of the flooding event, but not without equipment damage are subsequently transferred to the appropriate the plant model to evaluate their impact further. Flooding scenarios that cannot be mitigated are compared with the frequency of similar plant damage states in the overall risk model. An uncertainty analysis may be performed on dominant scenarios in the overall risk model.

Flooding initiating event frequency can be calculated using either of two methods. One method sums rupture failure rates for valves, tank, piping, and expansion joints in a location. this requires detailed information of the number of valves, type of valves, and piping sections. Another, more preferable method uses historical data on the total annual frequency of floods. The total flood frequencies for various plant buildings are apportioned to specific locations. This apportionment is based on the relative density of flood sources (e.g., piping, tank, etc.) in each plant location. The following general equation is used for the second method:

$$R_x = F_i \, f_{x,i} \, f_{S,X} \, f_{P,x}$$

where,

R_x = Annual frequency of a flood scenario in Location x.

F_i = Total annual frequency of the flood of any severity in Building i.

$f_{x,i}$ = Conditional frequency of the flood occurring in Location x of Building i, given that the flood has occurred in Building i.

$f_{S,x}$ = Severity factor; conditional frequency of the flood being of a severity to cause equipment failure.

$f_{P,x}$ = Propagation factor; conditional frequency of the flood propagating to the adjacent locations, given that the flood occurred at Location x with the severity specified to cause equipment failure. (For localized cases, $f_{P,x}$ = 1.0.)

The above equation provides the approach used in quantifying a scenario, starting with the historical data for the annual frequency of a flood occurring in a particular building and then considering the fraction of that flood frequency that might occur in the postulated location. The conditional factors, such as the severity of the flood and the ability of the operating crew to mitigate its propagation, are taken into account during the quantification of these postulated scenarios.

The following general equation is used to quantify scenarios in which recovery actions are included:

$$S_z = R_z(D_x + I_x)$$

where,

S_x = The annual frequency of the scenario and recovery failure.

R_x = The initiating event frequency of the scenario.

D_x = The conditional probability that the operating crew fails to detect the flood.

I_x = The conditional probability of the operating crew not isolating or mitigating the flood prior to the dependent failure of critical systems, given detection of the flood.

3.3.8.3 Identification of Flood Scenarios

3.3.8.3.1 Flood Sources

The following major flood sources were identified:

- Systems Connected to Lake Ontario
  - Service Water
  - Fire Water System
- Cooling Tower and circulating Water System
- Condensate Storage Tank (CST)
- Suppression Pool
- Reactor Building and Turbine Building Component Cooling Water Systems (i.e., RBCLC and TBCLC)

Circulating water and service water systems are considered to be potentially large flood sources, owing to their large pipe diameter, flow rates, and unless isolated, virtually infinite sources. In addition, large floods have been experienced within the industry with these systems as cited in the data base.

The circulating water system circulates water from the cooling tower to the main condensers in the Turbine Building and back to the cooling tower. The system contains large diameter fiberglass and carbon steel piping, and expansion joints. Motor operated butterfly valves provide system isolation capability at the suction side of the circulating water pumps and discharge sides of the water box.

The service water system provides cooling water to RBCLC and TBCLC heat exchangers, RHR heat exchangers, room unit coolers, EDGs, and other loads throughout the plant. Remotely operable MOVs provide system isolation capability both at the pumps and heat exchangers.

The fire protection water distribution source is Lake Ontario. The fire water pumps are located in the Service Water Building. Appendix C generally describes the fire suppression systems installed at NMP2.

The condensate Storage Tank (capacity of 720,000 gallons) is located in the Radwaste Building. This system provides condensate makeup to the condenser hotwell, and is the primary source of water for high pressure ECCSs and the CRDH system.

The suppression pool is located in the containment wetwell and normally contains greater than a million gallons of water (i.e., approximately 1.22E6 gallons) during normal operation.

The secondary component cooling water systems (i.e., RBCLC and TBCLC) are located in the Reactor Building and Turbine Building, respectively. However, these systems contain a limited water inventory in comparison with the circulating water and service water sources; and therefore, their rupture is assumed to pose little potential challenge to the operator's ability to safely shut down the plant.

3.3.8.3.2 Postulated Flood Scenarios

Table 3.3.8-1 presents a discussion of each flooding scenario considered as part of this assessment. Additionally, the disposition of each of these events is also discussed.

Those flooding scenarios selected for explicit quantification are discussed below:

- FLOODING OF THE TURBINE BUILDING FROM THE CIRCULATING WATER SYSTEM (Initiating Event "FLTB-CW")

  Description of Flooding Scenario
  The rupture of a circulating water supply line to a water box inlet can cause significant flooding of the turbine building. Generally, the cause of the breach in the system is random rupture of a water line, since maintenance on a water box is not performed while the plant is operating. If the operating crew can immediately isolate the line by securing the pump train and close the suction and discharge isolation valves, the plant can continue to operate at reduced power. If the condensate pit is flooded so that the discharge isolation valve cannot be closed, the operating crew must isolate the affected circulating water system train. In the extremely improbable situation that the operating crew fails to isolate the flood source (i.e., cooling tower reservoir), there is a possibility that the reactor building could become flooded to the point that all ECCSs are rendered inoperable.

  Potential Recovery Actions
  The operating crew must either isolate the individual circulating water train, or secure the entire system to prevent catastrophic flooding of the piping tunnel, and potentially, the reactor building basement elevation.

- FLOODING OF THE TURBINE BUILDING FROM THE SERVICE WATER SUPPLY TO TBCLC (Initiating Event "FLTB-SW")

  Description of Flooding Scenario
  Given that the operating crew can recognize the source of the flooding as the service water cooling supply to TBCLC heat exchangers, it is expected that isolation of the service water supply header to TBCLC can be expeditiously isolated remotely from the Control Room. Eventually, loss of cooling for TBCLC will require that the operating crew to manually shut down the plant because of its affect on BOP systems.

  Potential Recovery Actions
  The operating crew can close one of two MOVs to isolate the service water supply header from the TBCLC system.

- FLOODING OF THE CONTROL BUILDING El. 261' FROM SERVICE WATER OR FIRE WATER (Initiating Event "FLDG")

<u>Description of Flooding Scenario</u>
The largest source of water that poses a potential threat of flooding the Control Building El. 261' is the service water system that provides cooling water to the EDGs. Although the exposure of 8" piping above the 261' level is limited to the EDG rooms and the piping trough that communicates with these rooms, an unmitigated rupture of this piping could result in significant damage to the plant, since there are no structural barriers to prevent flooding of both emergency switchgear rooms located on the same elevation of building. Should the operating crew be unable to stop the flooding, the emergency switchgears will be adversely affected (i.e., assuming the accumulation of greater than 6" of water in the rooms), eventually causing a short of all the 4160 Vac buses.

<u>Potential Recovery Actions</u>
In the case of FLDG, the appropriate immediate action is to first open the missile door on El. 261' and drain the water that has accumulated in the Control Building outside. While this action is being accomplished, the operating crew could be preparing to align the service water system in anticipation of isolating the header supplying the affected EDG. However, the affected service water divisional header should be isolated as soon as possible by splitting out the affected header and securing all service water pumps in that division. (Note that it is presumed that the operating crew would be unable to safely enter the affected EDG room if significant amount of water existed on the floor of control building El. 261'.) Upon isolation of the service water division, the loss (at least temporarily) of the TBCLC and the RBCLC will result in the shutdown of the plant. Once the flood is terminated and the affected service water header is isolated locally at the EDG, the operating crew can realign service water to the turbine and reactor building loads.

- FLOODING OF THE SERVICE WATER PUMP BAYS (Initiating Event "FLSW")

<u>Description of Flooding Scenario</u>
Initiating Event "FLSW" assumes a large breach in a service water pump train (presumable during maintenance on that train based on inspection of data), that requires immediate action by the operating crew or maintenance personnel to isolate the breach before the entire pump bay becomes flooded. Early isolation of the affected pump train allows the continued operation of the service water system without any affect on the plant. Otherwise, the loss of a service water pump train (i.e., flooding of the pumps causes the respective power breakers to open on an electrical short), causes the isolation of the RBCLC and TBCLC service water supply headers. Subsequently, the operating crew is required to deal with the impending plant shutdown and restore service water to the reactor and turbine building loads.

<u>Potential Recovery Actions</u>
The immediate recovery actions require personnel within the vicinity of the system breach to isolate the flood source. However, if the service water pumps become

affected, the operating crew will be involved with restoring service water to equipment necessary to shut down the plant.

3.3.8.4 Internal Flooding Event Tree Model

The following sections describe the event tree model developed to analyze the four internal flooding events at NMP2. Figure 3.3.8-1 presents the event tree model and split fraction logic rules.

3.3.8.4.1 Assumptions

Several assumptions were considered in the development of the internal flooding analysis. These are described below:

- Certain potential flooding scenarios were evaluated to be less significant in terms of impact on the plant, and therefore, were not explicitly quantified. These events include any leak from a pipe that is within the screening criteria adopted for the MELB analysis (and therefore, treated separately in an unrelated but more prescriptive analysis), breach in a system resulting from maintenance activities on equipment that requires the manual local isolation of small water lines (e.g., room unit coolers), or larger breaches in a water system caused by an improperly performed maintenance activity if the maintenance is performed under close supervision (e.g., flow test of a fire water system).

- Flooding caused by a breach in limited volume systems (e.g., RBCLC and TBCLC), was evaluated and determined to offer no significant challenge to the operating crew that would seriously complicate their ability to safely place the plant in a stable shut down condition.

- Flooding in the Radwaste Building or from radwaste systems were evaluated and determined to be inconsequential to safe plant operation.

- Flooding inside containment, from a breach in a system not connected to the RCS, is considered to be adequately contained so that even if plant shutdown is required, the operating crew is not hampered in using BOP and ECCS systems to place the plant in a safe stable condition. Of course, LOCAs inside containment are treated separately in the PRA..

- Of the various scenarios considered in this analysis, LOCA events outside containment are explicitly not considered in this portion of the PRA, since the emphasis of the two analyses is different. Specifically, an internal flooding event, from a source other than the primary coolant, does not pose an immediate threat to the ability of the operating crew to maintain safe primary plant conditions post reactor shutdown (e.g., water inventory, decay heat removal).

- Some flood scenarios are postulated to affect entire pump bays. The common cause failure of all pumps located in the flooded bay is assumed to result in the automatic closure of isolation valves that are interlocked with the pump motor breaker. Additionally, it is further assumed that the subsequent opening or breach of these valves is unlikely, and not considered a viable failure mode. Therefore, flooding in a service water pump bay or a circulating water pump bay is assumed to result in the failure of all pumps in that division and subsequent isolation of the pump from the discharge header, and in the case of the circulating water pump, the suction header also.

- The maximum flood height that can occur in a service water bay from any flood source is determined to be approximately El. 257' (i.e., equivalent to the maximum water level of Lake Ontario); thereby, preventing the flooding of the adjacent bay. Similarly, the normal water level of the circulating water tower is approximately El. 257'. Therefore, flooding of the turbine building from the circulating water source is expected to be confined to the lower elevation of the building and the piping tunnel.

- Currently, NMP2 has no event-oriented procedures for mitigating a flooding scenario. Instead, N2-EOP-SC provides minimal direction for the operating crew to isolate the flood source if the reactor building is the location of the flood. Additionally, the appropriate annunciation procedures for tank and sump high water alarms instruct the operating crew to investigate the cause of the alarm. Therefore, any discussion of operator response provided in an event scenario description assumes that a procedure that outlines effective mitigating actions is available to the operating crew (e.g., flooding of the emergency diesel generator room from the service water system).

- Each of the auxiliary bays house low pressure ECCSs that are isolated from potential external flooding hazards by submarine type doors. The only means for water to enter any of these rooms is either through the doorways or via the piping trough, in the case where the reactor building is flooded up to El. 201'. For this analysis, it is assumed that the submarine doors can withstand a static pressure associated with approximately 20 ft. of water, as prescribed in the MELB analysis. Therefore, flooding of all high and low pressure ECCS compartments in the auxiliary and reactor buildings is assumed to occur when the water level in the reactor building reaches El. 195'.

- There are numerous diverse tank and sump level indicators to alert the operating crew of an accumulation of water in all areas of the plant considered in the internal flooding analysis. Therefore, the conditional probability of all water level indication within an area coincidentally failing to indicate in the Control Room a potential flooding condition is assumed to be remote.

- Flooding events that are postulated to result in an automatic or manual shutdown of the plant are not accounted for in the development of accident initiating event frequencies. Instead, it is assumed that the data base used to calculate these initiator frequencies includes all shutdown events regardless of the cause.

- The potential for improperly performed maintenance activities is considered in the quantification of all postulated internal flooding scenarios, except for maintenance of water box during power operation. Although this activity is not prohibited from being performed while the plant is at power, it is assumed that the activity would not be undertaken.

3.3.8.4.2 Discussion of Event Tree Top Events

Top Event FLI

The initiating event is defined as significant flooding of plant locations that both result in automatic or manual shutdown of the plant and equipment damage that can potentially challenge the operating crew's ability to safely achieve a stable plant condition. Table 3.3.8-2 provides the derivation of the initiating event frequencies for the four scenarios quantified in the model. Appendix C provides the basis for this derivation.

Top Event FLA

Top event "FLA" considers the possibility that a postulated flooding scenario can be isolated automatically (i.e., isolation logic circuitry), and mitigated without operator action. Success at this node implies that flooding is suspended without significant damage to other equipment in the vicinity of the affected area; whereas, failure to isolate the flooding source automatically requires an assessment of whether the operating crew can manually isolate the flooding before induced equipment damage results. Note that, for this assessment, no postulated flooding events can be automatically isolated; therefore, all scenarios are assigned a guaranteed failure at this node.

Top Event FLR

Immediately upon initiation of the flooding event, it is expected that the operating crew will have sufficient indication of plant conditions to determine the location and extent of the flooding, and the water source. (Appendix C provides a listing of the diverse and redundant sump and tank water level indications for all areas of the plant considered in this analysis.) Top event "FLR" models the likely action that the operating crew recognizes the situation and mitigates the flooding event before damage to other equipment, including the opposing train of the same system, further complicates the operator's ability to safely shut down the plant. Success at this top event implies that the operating crew has successfully mitigated the flooding event and limited the extent of plant damage to equipment in the immediate vicinity of the flood without propagation of the flood into adjacent spaces or damaging other equipment located in the same area.

Top Event FLL

Although the operating crew failed to immediately isolate the flood source early in the scenario, additional actions may still be implemented to prevent further plant damage. Therefore, top event "FLL" models the conditional probability that the operating crew fails to mitigate the flooding event and further damage to plant equipment within 1 hour from the initiation of the event. Primarily of interest are the scenarios that can result in excessive flooding of the reactor building El. 175' and the control building El. 261', because both of these events require that a considerable amount of water accumulate inside the plant before considerable damage to vital plant equipment is realized. A conservative time frame of 1 hour is selected because: 1) it conservatively bounds the minimum quantity of water necessary to cause catastrophic flooding of the two areas that can result in loss of all ECCS or unrecoverable station blackout, respectively; and 2) it is considered highly likely that the operating crew will be able to isolate the flood source remotely given the numerous primary and secondary indications of flooding in the control room. Failure at this top event implies that catastrophic flooding of either the reactor building basement or the control building poses a significant threat to continued safe plant operation. On the other hand, success at event node "FLL" implies that the operating crew has successfully isolated the flood source, but must now turn their attention to return the plant to a safe condition by restoring cooling water to safety related equipment and BOP systems.

Top Event FLS

Of particular concern in this analysis is the scenario involving significant flooding of El. 261' of the control building resulting in damage to all emergency switchgears. For the purposes of this analysis, the impending station blackout condition is considered a unrecoverable plant condition. Therefore, this top event evaluates the conditional probability that the operating crew can prevent the flooding from affecting the emergency switchgear buses by 1.5 hours into the accident scenario. (The time frame is defined by the accumulation of 6" of water at El. 261' assuming 850 gpm flowrate and no loses from the designated volume.) Success at this event node implies that flooding of the control building is mitigated within 1.5 hours from the initiation of the event, either by isolating the source of water or diverting the flooding outside the building. Failure to mitigate flooding is assumed to lead to extensive flooding of the emergency switchgear rooms, resulting in the inability on the part of the operating crew to maintain the plant in a safe and stable condition.

Top Event FLW

Degradation of the service water system can have a profound affect on subsequent plant operation. For instance, in the case of flooding in either service water pump bay, or any event that requires the operating crew to isolate a division of service water (e.g., to isolate flooding of the EDG room remotely), a temporary loss of all RBCLC and TBCLC would occur. Consequently, the long term operation of BOP systems or ECCSs would require the operating crew to reestablish the affected cooling water system before a stable plant condition could be achieved. Success at this top event implies that the TBCLC and RBCLC systems are unaffected by the flooding scenario, both directly by the accumulation of water and in terms of potential operator response to the event during which the system is isolated. Failure at node "FLW" implies that both cooling water systems have been isolated during the course of the flooding scenario.

Top Event FLB

Similarly, the BOP systems can be permanently adversely affected by flooding of the turbine building. Additionally, the isolation of the TBCLC would result in the temporary failure of the BOP systems requiring the operating crew to restore these systems if coolant makeup or decay heat removal were subsequently necessary. Success at this event node implies that the BOP is unaffected, and presumably operational, during the course of the flooding scenario. Conversely, failure at top event "FLB" implies that the BOP systems are adversely affected by flooding.

Top Event FLX
In the case of severe flooding in the turbine building, the water will flow into the piping tunnel. As the tunnel fills, the potential for flooding of the reactor building via the piping penetrations increases as the static head on the inflatable seals requires their proper functioning to maintain physical separation. Should the seals fail, it is assumed that the resulting water level in the reactor building will exceed El. 195' (refer to assumptions) and cause flooding of all high and low pressure ECCS rooms located in the reactor building basement. Success at this top event implies that no flooding in the reactor building occurs during the scenario; whereas, failure at this event node is assumed to result in severe flooding of the reactor building that disables all ECCSs. These scenarios are presumed to lead to core damage conditions.

3.3.8.4.3 Intermediate End States

As described above, the methodology employs a conservative screening analysis technique to determine potential flood sources and locations, and the associated impacts on continued plant operation and the ability of the operating crew to safely shut down the plant. Postulated flood scenarios are defined in terms of the flood source, the extent of propagation to adjacent locations, and the severity of equipment damage. The frequencies of these scenarios are then quantified. With the exception of those sequences that result in core damage, and categorized as Class IA scenarios, the remaining sequences are transferred to the transient model for further consideration. The definition of the intermediate end states that describe these sequences are defined below:

- End state "TRTT" (i.e., transfer to turbine trip transient model) assumes that flooding events that were isolated immediately required the operating crew to trip the turbine as a result of lowering condenser vacuum. End state "TRLOC" (i.e., transfer to loss of condenser vacuum transient model) assumes that the entire circulating water system had to be secured by the operating crew to completely isolate the flood source.

- End state "TRFLTBSW" includes scenarios where the service water to TBCLC has been isolated.

- The end state "TRFLSW" is defined as the condition in which all 3 pump trains in the same service water division are isolated.

- The end state "TRFLDG" is defined as the condition where the flooding is mitigated, damage to the emergency switchgears has been avoided, and the plant is shutdown due to the necessary isolation of a service water division. Additionally, the operating crew has isolated the rupture locally so that the service water system can be realigned to its normal configuration.

Table 3.3.8-1
Description of Potential Flooding Scenarios

Initiating Event Designation	Water Source	Flood Location	Flood Scenario and Disposition
Not Applicable	Fire Water	EDG Room	Aside from damage to the diesel engine, which would force the operating crew to shut down the plant, inadvertent actuation of the compartment preaction sprinkler or improperly performed maintenance on the sprinkler header could result in flooding of this compartment and the 261' elevation of the Control Building. However, the flooding capacity of the fire water system is considerably less than the service water system. Additionally, the data indicate that all industry events were either maintenance related or spurious actuation caused by a high temperature profile in the engine room, resulting in localized flooding to which the operators or maintenance personnel properly responded before the capacity of the drain system was exceeded. Therefore, this event is not modeled, even though the location of the flood is in close proximity of the emergency switchgear rooms. Instead, potential flooding of the Control Building from fire water is considered bounded by the analysis performed for initiating event FLDG. This treatment is considered appropriate for three reasons: 1) improperly performed maintenance tasks usually can be mitigated due to the proximity of personnel during the activity; 2) the control room has several redundant indication if such a flood event were to occur (i.e., drains and preaction sprinkler system initiation); and 3) relatively limited flowrate of the system.
FLDG	Fire Water	Control Building El. 261'	El. 261' of the Diesel Generator Building contains one 6" fire suppression system header that supplies the 3 preaction water systems for the engine rooms. The location of the header is in the hallway outside the EDG cubicle where the pipe enters the building from the yard. The random rupture of this header could cause considerable damage to the plant by flooding the emergency switchgear rooms. The affect of this scenario on the ability of the operator to mitigate the flood and safely shut down the plant is similar to that described for service water flooding in the EDG rooms (i.e., initiating event FLDG). Therefore, this event is explicitly quantified and included in the analysis of initiating event FLDG.

Table 3.3.8-1
Description of Potential Flooding Scenarios

Initiating Event Designation	Water Source	Flood Location	Flood Scenario and Disposition
Not Applicable	Fire Water	Reactor Building El. 175'	Flooding the basement elevation of the Reactor Building to the point at which plant operation is threatened requires an extraordinary amount of water. Even assuming that the drain system is incapable of removing the water from the area, there is sufficient redundant indication in the Control Room (i.e., sump level annunciation and actuation of the water deluge system), to allow the operator many hours to respond without damaging low pressure ECCSs. (Note that significant flooding of the basement elevation of the reactor building up to El. 195' is considered to fail all ECCS located in the auxiliary and reactor buildings. Additionally, any flooding in the lower elevation is considered to render RCIC inoperable because of suspected damage to 2 instrument racks, i.e., 2CES*RAK029 and 2CES*RAK017.) It is assumed that under the most extreme circumstances where the operating crew would fail to take any action to mitigate the flooding within many hours, the BOP systems are expected to remain unaffected. Therefore, this event is not modeled in the PRA.
Not Applicable	Fire Water	Turbine Building	The El. 250' of the turbine building contains the 5 deluge and 3 fire water preaction fire water systems. However, as noted in the analysis assumptions, the probability that significant flooding could occur as a result of improperly performed maintenance is considered to be remote. In fact, the most probable flooding scenario involves the rupture of any major supply header. The potential consequences of flooding the turbine building can be severe, except that in this case it would take on the order of hours for a sufficient amount of water to accumulate and cause the failure of BOP systems and flood the reactor building via the piping tunnel. Therefore, this flooding event is not explicitly modeled. Instead, flooding of the turbine building from the circulating water system or service water to the TBCLC system is considered to dominate all evaluations of flooding in this area.
Not Applicable	Fire Water	Cable Raceways	The cable fire spray system is designed to extinguish a fire in a local and confined area of a raceway. The probability of that the system floods either the fire area or propagate into an adjacent area is considered very limited, and is considered as part of the Fire Hazards Analysis (FHA) (i.e., in which the flow rate of the system, the configuration of the area, and the capacity of any drainage system in the area are considered). The possibility that subjecting any raceway to water spray could induce electrical shorts is a function of the condition of the cabling; the potential impact of suppression system actuation on plant systems is also evaluated separately in the FHA. Therefore, this event is not explicitly modeled in the PRA.

Table 3.3.8-1
Description of Potential Flooding Scenarios

Initiating Event Designation	Water Source	Flood Location	Flood Scenario and Disposition
Not Applicable	Fire Water	Piping Tunnel	The piping tunnel contains numerous 4" and 6" fire suppression headers. The amount of water required to flood the piping tunnel and the lower elevation of the reactor building would require that the largest pipe rupture and no response by the operating crew to isolate the fire water suppression system for many hours. The probability that the operating crew would be unaware of the situation and unable to secure the fire water system or isolate the ruptured line is considered negligible, especially since there are numerous redundant indications available to notify the operator of a system instability and water accumulation in the many sumps and drain tanks in the tunnel. Additionally, it is not expected that even the most severe flooding event could affect the BOP systems and the ability of the operating crew from utilizing these systems to safely shut down the plant. Therefore, instead of explicitly modeling this event, it is considered that any low frequency events that could potentially result in a challenge to the plant are subsumed by the modeling of other similar events that involve severe flooding of the turbine building (i.e, initiating event FLTB).
Not Applicable	Fire Water	Control Room and Relay Rooms	These areas that contain sensitive electronic equipment do not rely on water sprinkler systems for fire protection. Therefore, flooding in these areas is not treated in the PRA.
FLDG	Service Water	EDG Room	The largest source of water that poses a potential threat of flooding the Control Building El. 261' is the service water system, which provides cooling water to the EDG engine jacket. Although the expos[illegible] of 8" piping above the 261' level is limited to the EDG rooms and the piping trough that communicates with all EDG rooms, an unmitigated rupture of this piping could result in significant damage to the plant, since there are no physical barriers to prevent flooding of both emergency switchgear rooms. Specifically, the flooding of the emergency buses would cause the immediate failure of the service water system, eventually resulting in station blackout. Therefore, this flooding event is explicitly modeled in the PRA.

Table 3.3.8-1
Description of Potential Flooding Scenarios

Initiating Event Designation	Water Source	Flood Location	Flood Scenario and Disposition
FLTB-SW	Service Water	Turbine Building El. 250'	Significant flooding in El. 250' of the turbine building can develop if there is a breach in the service water piping supplying the TBCLC heat exchangers. Although there is limited exposure of service water piping in the turbine building (i.e., the piping tunnel is adjacent to this part of the turbine building), the potential exists for either one of six 24" pipe segments to rupture or a system breach occurring during maintenance. The consequential flooding of the turbine building would be significant both in terms of the amount of water that can accumulate in the building and its impact on BOP systems. Left unmitigated, such a flooding event could over the course of hours fill the lower elevation of the building and the piping tunnel. However, the operating crew has a large time window to isolate the flood source by remotely isolating the turbine building supply header from the main service water header (i.e., MOV 3A or 3B), which would disable all BOP systems. Therefore, the likelihood that the operating crew can mitigate the flooding is explicitly modeled in the PRA.
Not Applicable	Service Water	Piping Tunnel	Although there is a potential for a rupture in a service water header in the piping tunnel, there are no similar events mentioned in the data base. Since rupture of the piping or a valve is considered the only reasonable failure mode that could affect this portion of the service water system while the plant is at power, the possibility of such a low frequency event causing significant flooding of the tunnel and possibly the Reactor Building is considered bounded by the analysis of turbine building flooding (i.e., initiating event FLTB-SW).

Table 3.3.8-1
Description of Potential Flooding Scenarios

Initiating Event Designation	Water Source	Flood Location	Flood Scenario and Disposition
Not Applicable	Service Water	Auxiliary Building - ECCS or RBCLC rooms	Significant flooding of one Auxiliary Bay and the Reactor Building that could both cause plant shutdown is considered remote, and is assumed to be dominated by failure to maintain proper system isolation (i.e., on the service water side) during maintenance on a RBCLC heat exchanger. Under the worst circumstances (i.e., 2 divisions of low pressure ECCSs damaged from a rupture in the RHR service water header), over 2 million gallons of water would have to flood the Reactor Building basement to disable all the high and low pressure ECCSs. This event, however severe, would provide a large time window (i.e., many hours) for the operating crew to isolate the RBCLC service water header (i.e., by closing either MOV 19A or 19B) before flooding of the ECCSs would occur. Additionally, the isolation of the RBCLC supply header would not affect the supply to the turbine building, EDGs, and the ECCSs. Therefore, the operating crew would have all safety equipment available to place the plant in a stable shutdown condition after the plant tripped on loss of instrument air. The flooding event is not explicitly modeled in the PRA because it is considered highly improbable that the operating crew would fail to isolate one of three MOVs within 3 hours (i.e., conservative estimate assuming 10,000 gpm system flow rate). Instead, the contribution to the total loss of RBCLC initiating event frequency is considered in the support system transient model.
FLSW	Service Water	Service Water Pump Bay	Flooding of a pump bay can have serious consequences with respect to continued plant operation. For instance, any breach in a service water header upstream of a pump discharge isolation valve would cause the subsequent failure of all 3 pumps in that division, and their isolation from the main supply header. Additionally, if flooding is not terminated and the opposite pump bay becomes affected (i.e., in the case where the system breach occurs downstream of the pump discharge isolation valves), a total loss of service water could result. This severely degraded plant condition could jeopardize the ability of the operating crew to safely shut down the plant. Based on inspection of the data base, the events affecting the service water system were limited to a particular pump train, and were caused by leakage failures of equipment or maintenance related activities. This event is explicitly modeled (i.e., initiating event FLSW).
Not Applicable	Service Water	Service Water Building	The frequency for a breach in any service water piping contained in the service water building, and not in the pump bays, is considered negligible compared with the frequency of flooding of the system associated with maintenance activities performed on the pumps and strainers. Additionally, the impact of flooding inside the building, in terms of propagation, is also considered negligible since the water would be expected to flow into the screen well or outside. Therefore, this flooding event is not explicitly modeled; it is considered bounded by the analysis of flooding in the pump bays.

Table 3.3.8-1
Description of Potential Flooding Scenarios

Initiating Event Designation	Water Source	Flood Location	Flood Scenario and Disposition
Not Applicable	Condensate Storage Tank (CST)	Reactor Building El. 175'	NMP2 has experienced a CST rupture that subsequently flooded the Reactor Building; rupture of the tank resulted in flooding of the piping tunnel through backing-up of an interconnected drain system, and subsequently the Reactor Building basement via a failed isolation water-tight seal. However, even though the operating crew would be required to shut down the plant as directed in Technical Specifications due to loss of CST inventory, the resulting flood in El. 175' of the Reactor Building would equilibrate at a level of approximately 6 ft. It is judged that this amount of water in the basement would neither interfere with plant operation or the performance of a normal shutdown of the plant, nor affect ECCSs in the Reactor or Auxiliary Buildings. Therefore, this event is not explicitly modeled in the PRA.
Not Applicable	Circulating Water	Circulating Water Pump Bay	Flooding initiated in the circulating water pump bays would cause a complete loss of the system due to the draining down of the tower water supply into the affected pump bay. As the pump bay fills with water, it is assumed that the motors will trip on high current, and subsequently cause the pump isolation suction and discharge valves to close; thereby, isolating the flood source. However, even if the isolation valve failed to automatically isolate, the maximum water level that could be achieved in the circulating water pump bay would be the normal water level of the system reservoir (i.e., El 257'). This water level is approximately 3 ft. higher than the level of the feedwater heater bay pipe chases that provide a communication pathway between the pump bay and the turbine building. Therefore, under the most extreme conditions, insignificant flooding of the turbine building would result. This flooding scenario is not explicitly modeled in the PRA.
FLTB-CW	Circulating Water	Turbine Building El. 250'	The rupture of a circulating water line in the Turbine Building is potentially the most severe flooding event postulated in this analysis. For instance, if the entire inventory of the circulating water system were flooded into the building, it is expected that in addition to the lower elevation of the turbine building, the piping tunnel would also be flooded. Even though the Control Building would not be affected, it is possible that the Reactor Building could be flooded if the building separation seals were to fail under the hydrostatic head of the water accumulated in the piping tunnel. Therefore, the BOP equipment in the Turbine Building and the ECCSs in the Reactor Building could be severely affected. This event is explicitly modeled (i.e., initiating event FLTB).

Table 3.3.8-1
Description of Potential Flooding Scenarios

Initiating Event Designation	Water Source	Flood Location	Flood Scenario and Disposition
Not Applicable	Suppression Pool	Auxiliary Building - ECCS Room	Another postulated scenario involves the damage of 1 division of low pressure ECCS in an Auxiliary Bay caused by a breach in the ECCS and the subsequent flooding of the room from the suppression pool. (Note that high pressure ECCSs are normally aligned to the CST.) The suppression pool water level would equilibrate at a level low enough to prevent flooding of the adjacent ECCS room and the Reactor Building (i.e., water level would be less than the height of ECCS piping trough). Although the ECCS suction lines would be uncovered, high pressure ECCS (i.e., normal suction path from the CST and RCIC turbine exhaust remains submerged), and the BOP would be unaffected by the event. Additionally, the SRV tail pipes remain intact and submerged. Therefore, the operating crew would be able to perform a normal shutdown of the reactor. This event is not modeled in the PRA.
Not Applicable	Condensate - Primary Coolant	Turbine Building El. 250'	Leaks in the feedwater and condensate systems are not considered in the flooding analysis. Instead, these events are treated in the Break Outside Containment LOCA evaluation. This is considered an appropriate treatment of these events for the following reasons: • The events cited in the data base are all leakage events where it is assumed that the conditional probability that the operating crew fails to prevent propagation of the flood is considered very remote. • In the case of a large LOCA event outside containment, automatic response of the pressure regulator system and the NSSS is considered a very reliable means to isolate the breach from the RPV. • The total inventory of the RCS is much less than the other water sources (i.e., circulating water and service water systems), considered in the turbine building flooding assessment. Therefore, LOCA outside containment events are not explicitly treated in the flooding analysis.

Table 3.3.8-2
Summary of Flood Initiating Event Frequencies

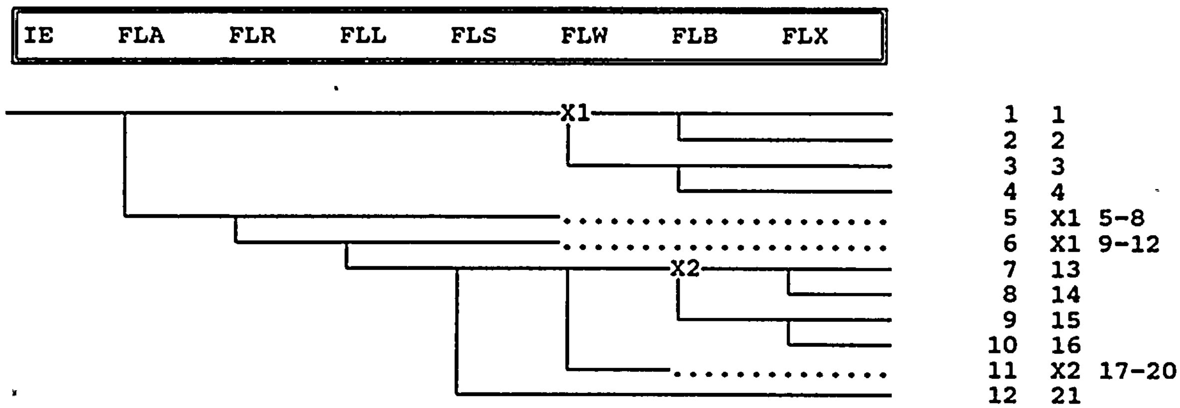
Initiating Event Designation	Initiating Event Description	Significant System Breach (Referenced by Event Number)			Total No. of Events in Data Base (per 1081 Reactor Years)	Adjusted No. of Events (per 740 Reactor Years)	Total Initiating Event Frequency
		Maintenance Related (1)	Inadvertent Action	Random Breach			
FLSW	Flooding of a Service Water Bay	34, 62			2	1.4	1.9E-3 / Reactor Yr
FLTB-CW	Turbine Building flooding, from the Circulating Water System	94, 96		4, 93	4	2.8	3.8E-3 / Reactor Yr
FLTB-SW	Turbine Building flooding, from the Service Water System				1 (2)	0.7	9.3E-4 / Reactor Yr
FLDG	Flooding of a Diesel Generator Room				1 (2)	0.7	9.3E-4 / Reactor Yr

1. Either loss of system isolation established as a pre-requisite condition for a maintenance activity or failure to restore the system to proper configuration upon completion of the activity.

2. One (1) incipient event assumed to occur within the time frame associated with the data base (i.e., conservative estimate since limited industry experience cannot be cited to support a flooding initiating event of significantly lesser frequency).

Figure 3.3.8-1

MODEL Name: NM2FLOOD
Event Tree: FLOOD



Top Event Designator	Top Event Description
IE	Initiating Event
FLA	AUTOMATIC ISOLATION OF FLOOD SOURCE
FLR	EARLY MANUAL ISOLATION OF FLOOD SOURCE
FLL	LATE MANUAL ISOLATION OF FLOOD SOURCE
FLS	STATION BLACKOUT NOT INDUCED BY FLOOD EVENT
FLW	SERVICE WATER SYSTEM NOT AFFECTED
FLB	BOP NOT AFFECTED BY FLOOD EVENT
FLX	NO FLOODING OF REACTOR BUILDING

Figure 3.3.8-1

MODEL Name: NM2FLOOD
Split Fraction Logic for Event Tree: FLOOD

SF	Split Fraction Logic
FLAF	INIT=FLDG + INIT=FLSW + INIT=FLTBSW + INIT=FLTBCW
FLAS	1
FLR1	INIT=FLTBCW Rule Comment ------------ NO MAINTENANCE PERFORMED ON CIRC WATER PUMP WHILE OPERATING; OP NEEDS TO TRIP PUMP BEFORE WATER LEVEL SUBMERGES DISCHARGE MOV
FLR2	INIT=FLSW Rule Comment ------------ OPERATOR FAILS TO TERMINATE FLOODING BEFORE AN ENTIRE DIVISION OF SW IS AFFECTED; 0.5 PERSONNEL WOULD RECOGNIZE FAILURE OF ISOLATION FOR MAINTENANCE AND 0.5 FAILURE TO SUBSEQUENTLY TERMINATE FLOODING W/IN 5 MIN.
FLR3	INIT=FLTBSW Rule Comment ------------ FLOODING ASSUMED TO BE DOMINATED BY MAINTENANCE ACTIVITIES; ISOLATION OF FLOODING BEFORE BOP IS AFFECTED
FLRF	INIT=FLDG Rule Comment ------------ ALL OPERATOR ACTIONS FOR FLDG SCENARIO TREATED IN "FLS"
FLLS	INIT=FLSW Rule Comment ------------ FLOODING CONFINED TO (1) SERVICE WATER BAY; RUPTURE IN SUPPLY HEADER CONSIDERED REMOTE PROBABILITY COMPARED TO INITIATING EVENT BASED ON DATA
FLL1	INIT=FLTBCW Rule Comment ------------ BOP AFFECTED; NO DAMAGE TO CONTROL BUILDING; POSSIBLE FLOODING OF REACTOR BUILDING

Figure 3.3.8-1

MODEL Name: NM2FLOOD
Split Fraction Logic for Event Tree: FLOOD

SF	Split Fraction Logic
FLL2	INIT=FLTBSW Rule Comment ------------ BOP AFFECTED; POSSIBLE FLOODING OF REACTOR BLDG AND EL. 261' OF CONTROL BLDG
FLLF	1
FLSS	INIT=FLSW Rule Comment ------------ SERVICE WATER BAY SEPARATED FROM CONTROL BUILDING
FLSS	INIT=FLTBCW Rule Comment ------------ WATER FORM CIRC WATER SYSTEM CAN'T FLOOD TURBINE BUILDING UP TO ELEV. 261'
FLS1	INIT=FLDG Rule Comment ------------ CONDITIONAL PROB THAT SW FLOODING CAN'T BE TERMINATED WITHIN 1.5 HRS, EITHER BY MANUALLY OPENING DOORS TO OUTSIDE OR SPLITTING-OUT THE SWS
FLS2	INIT=FLTBSW Rule Comment ------------ CONDITIONAL PROB THAT SW FLOODING CAN'T BE TERMINATED WITHIN 4 HOURS
FLSF	1
FLWS	INIT=FLSW * FLR=S Rule Comment ------------ FLOODING EVENTS THAT WERE ISOLATED EARLY W NO DAMAGE TO OTHER SW PUMPS IN BAY

Figure 3.3.8-1

MODEL Name: NM2FLOOD
Split Fraction Logic for Event Tree: FLOOD

SF	Split Fraction Logic
FLWS	INIT=FLTBCW Rule Comment ------------ SWS UNAFFECTED BY FLOODING IN TURBINE BUILDING
FLWF	INIT=FLDG + INIT=FLSW + INIT=FLTBSW Rule Comment ------------ FLOODING CAUSES LOSS OF 1 DIVISION OF SWS AT LEAST TEMPORARILY ASSUMING THHAT THE FLOODING WAS ISOLATED REMOTELY
FLBS	INIT=FLSW * FLW=S Rule Comment ------------ FLOODING EVENTS THAT WERE ISOLATED EARLY W/ NO DAMAGE TO OTHER SW PUMPS IN BAY, AND THEREFORE, NO AFFECT ON TBCLC
FLBS	INIT=FLTBCW * FLR=S Rule Comment ------------ LIMITED FLOODING IN CONDENSATE PIT; 1 WATER BOX AFFECTED; CONDENSER AVAILABLE
FLBF	INIT=FLSW + INIT=FLTBSW Rule Comment ------------ TBCLC ISOLATED; NO RECOVERY CONSIDERED
FLBF	INIT=FLTBCW Rule Comment ------------ TBCLC ISOLATED; NO RECOVERY CONSIDERED
FLBF	INIT=FLDG Rule Comment ------------ TBCLC ISOLATED; NO RECOVERY CONSIDERED
FLBF	1

Figure 3.3.8-1

MODEL Name: NM2FLOOD
Split Fraction Logic for Event Tree: FLOOD

SF	Split Fraction Logic
FLX1	INIT=FLTBCW * FLL=F + INIT=FLTBSW * FLL=F Rule Comment ------------ SEVERE FLOODING OF REACTOR BUILDING VIA PIPE PENETRATIONS THAT DISABLES ALL ECCSs; CONDITIONAL PROB THAT SEALS FAIL
FLXS	1

Figure 3.3.8-1

MODEL Name: NM2FLOOD
Binning Logic for Event Tree: FLOOD

Bin	Binning Rules
CLASSIA	FLS=F + FLX=F
TRTT	INIT=FLTBCW * FLB=S
TRLOC	INIT=FLTBCW * FLB=F
TRFLSW	INIT=FLSW * FLB=F
TRFLDG	INIT=FLDG * FLB=F
TRFLTBSW	INIT=FLTBSW * FLB=F
FLOODSAFE	FLB=S

Section 3.4

Results and Screening Process

3.4 Results and Screening Process

The Level 1 (Front-end) results of the Nine Mile Point Unit 2 (NMP2) IPE model are presented in this section. The Level 2 (Back-end) results are presented in Section 4.6.

The total mean core damage frequency is calculated as 3.1E-5 per year. As described in Section 3.1.5, the definition of core damage end states is based primarily on critical safety functions required to attain a safe stable state. The contribution from each core damage end state is summarized below:

End State	Safety Function Failure	Sequence Type	Frequency	Fraction of Total
Class ID	Injection (LP)	Transient/SLOCA	9.1E-6	0.29
Class IA	Injection (HP)	Transient/SLOCA	5.8E-6	0.19
Class IB	Injection	Blackout	5.5E-6	0.18
Class IIA	Heat Removal	Transient/SLOCA	4.7E-6	0.15
Class IIT	Heat Removal	Transient/SLOCA	4.2E-6	0.13
Class IVA	Power Control	ATWS	8.0E-7	0.03
Class IIIB	Injection (HP)	MLOCA	4.2E-7	0.01
Class IIL	Heat Removal	MLOCA/LLOCA	3.0E-7	0.01
Class IC	Injection	ATWS	2.2E-7	<0.01
Class IVL	Press Control	ATWS/LOCAs	7.2E-8	<0.01
Class V	Containment	Bypass (ISLOCA)	2.5E-8	<0.01
Class IIID	Vapor Suppression	LOCA	1.1E-8	<0.01
Class IIIC	Injection (LP)	MLOCA/LLOCA	6.4E-9	<0.01
TOTAL			3.1E-5	1.00

The above results are examined a number of ways in this section including the importance or contributions from initiating events, systems (event tree top events), and human actions (event tree top events). As described in the following sections, support system failures are important contributors to core damage risk with AC power the more important system.

3.4.1 Application of Generic Letter Screening Criteria

The IPE reporting requirements are described in Generic Letter 88-20 and NUREG-1335 for both functional and systemic sequences. The NMP2 IPE model provides results in terms of both functional and systemic sequences. As described above, the core damage end state grouping approach applied to this study represents functional sequence grouping. Thus, the results summarized above address the functional sequence reporting requirements. The reporting guidelines state that the total number of most significant sequences to be reported for systemic sequences should not exceed 100 and that the mean frequency should be reported. The following addresses how the 100 sequence limit and the other reporting requirements for systemic sequences are addressed:

- Any systemic sequence that contributes 1E-7 or more per reactor year to core damage. Table 3.4.1-1 lists sequences with a frequency of 4.0E-8 or more per reactor year to core damage. Additional sequences can be provided upon request.

- All systemic sequences within the upper 95% of the total core damage frequency. The top 100 sequences are provided in Table 3.4.1-1 which accounts for about 83% of the total core damage frequency. Additional sequences can be provided upon request.

- All systemic sequences within the upper 95% of the total containment failure frequency. This reporting requirement is addressed in Section 4.6.

- Systemic sequences that contribute to containment bypass frequency in excess of 1E-8 per reactor year. The IPE model includes containment bypass sequences due to interfacing LOCA events. There are no sequences greater than 1E-8. The total contribution from interfacing LOCA sequences is about 2.5E-8 as described above for core damage end state CLASS V.

- Any other systemic sequence that the utility determines to be important to core damage frequency or to poor containment performance. No additional important systemic sequences that contribute to core damage frequency have been identified. Section 4 further addresses containment performance and containment isolation importance.

In summary, Table 3.4.1-1 provides the top 100 core damage sequences. The initiating event frequencies and event tree top event unavailabilities used to quantify sequences are based on mean values. The sequences in Table 3.4.1-1 are presented in terms of the initiating event, the top event failures (split fractions) that occurred, and the core damage end state where the sequence belongs. Table 3.1.4-2 lists the initiating events, their frequency and a brief description. Table 3.2-2 describes event tree top events and provides a road map to the applicable system description and event tree models where additional detailed descriptions can be found. Table 3.4.3-3 lists the system top events included in the Level I model. This table is presented as a foldout page so that it can be referenced from throughout Section 3.2.1. The top event failures (split fractions) identified in Table 3.4.1-1 are described in Section 3.3.5 where the first two characters represent the top event in the event tree and the third character represents quantitative unavailabilities of the top event under several boundary conditions. An "F" in the third character location represents guaranteed failure (split fraction frequency = 1.0) of the top event because of dependencies on other top events. The core damage end states are described above and in Section 3.1.5. The top 10 core damage sequences are described in detail in Section 1.4.

3.4.2 Vulnerability Screening

Generic Letter 88-20 requests Licensee's to conduct a systematic search for severe accident vulnerabilities. With regard to core damage frequency, there are no unusual or plant unique contributors to core damage that appear different in comparison to other BWR plants and the results indicate that NMP2 can meet the NRC safety goal for core damage. Although no vulnerabilities were identified, this section provides an in-depth examination of the core damage frequency results in a number of ways to derive plant specific insights on the importance of systems, functions, and human actions. In addition, plant improvements identified during the study are summarized in Section 6.

In addition, no vulnerabilities were identified from the containment (Level 2) examination. This conclusion is based on the following:

- The NMP2 Mark II containment structure was evaluated in detail to assess the containment capability for pressure, temperature, and dynamic loads. At low temperatures, the ultimate capability of NMP2 was found to exceed 140 psig. This capability is similar to other published evaluations of BWR containment capability.

- A large number of deterministic phenomenological analyses were performed to test whether the NMP2 containment response to severe accident phenomena presents any unusually poor performance. The calculations indicate that the NMP2 containment provides a substantial benefit in the mitigation of severe accidents.

- The containment isolation system was explicitly evaluated to determine whether containment isolation failure could contribute to radionuclide release magnitude or timing. The results indicate that the NMP2 containment isolation system has a high reliability except for the case of station blackout sequences where the operators have to close motor operated valves locally. These sequences do contribute to the frequency of radionuclide release.

- Potential containment bypass sequences were also investigated to ascertain whether these sequences could result in defeating containment capability. The containment bypass sequences resulted in a frequency of approximately 3E-8/year. This frequency and its impact on radionuclide release is considered not to represent an unusual containment performance.

- The containment mitigating systems were included in the containment evaluation process to ensure that the full capability of containment is included in the assessment. The NMP2 containment mitigating systems are typical of BWR/5 Mark II plants and these systems provide capability to mitigate severe accidents and no unusually poor performance was found.

3.4.2.1 Contribution of Functionally Grouped Sequences

As summarized on the first page of this section, core damage end states provide a functional grouping of core damage sequences. This functional grouping of end states is described further in Section 3.1.5. The following table combines the results on page 3.4-1 into more general groups where ATWS includes Class IC and IV, Blackout includes Class IB, Loss of Heat Removal includes Class II, and Loss of Injection includes the remaining Class I and Class III.

Functional Group	Frequency	Fraction of Total
ATWS	1.1E-6	0.04
Blackout	5.5E-6	0.18
Loss of Heat Removal	9.1E-6	0.29
Loss of Injection	1.5E-5	0.50
Internal Floods *	1.5E-6	0.05

* The internal flood contribution is not mutually exclusive from the other functional groups.

Station blackout sequences are actually injection failures, therefore, the effective loss of injection total is approximately 68 percent. Station blackout is dominated by loss of offsite AC power as an initiating event (99 percent), unavailability of both Division I & II emergency diesels, and failure to recover offsite AC or a diesel before RCIC failure. The dominant sequence is a station blackout sequence (see Sequence 1 description in Section 1.4).

Loss of injection is primarily dominated by loss of support system initiating events and then additional support system failures after the initiating event. Dominant sequences include loss of Divisional AC power as an initiating event and then subsequent loss of the opposite Division of DC power (see description of Sequences 2 and 3 in Section 1.4).

Loss of long term heat removal contributes approximately 29 percent and is discussed further in the following sections and in Section 3.4.3. The dominant sequence (see Sequence 4 description in Section 1.4) is due to a loss of Division II AC power initiating event which fails containment venting because nitrogen cannot be supplied to the inside containment vent valve. Subsequently, service water fails such that RHR heat removal is unavailable.

ATWS sequences contribute approximately 4 percent with the reliability of the standby liquid control system (top event SL) and operator actions to control RPV level after emergency depressurization (top event CH) most important. Approximately 36 percent of the ATWS total is from failure of standby liquid control (SL) and 56 percent from operator action CH. At NMP2, the standby liquid control system, alternate rod insertion, feedwater runback and reactor recirculation pump trip functions are automatically actuated by the redundant reactivity control system. Overall, these functions are generally more reliable than the standby liquid control system (SL) and there contribution to ATWS is only about 1 percent.

Internal flooding events provide a minor contribution and are dominated by one service water system flood. Service water is important because it can be a potentially large flood source and it is an important support system. The dominant sequence (66 percent) is a flood in an emergency diesel room and because it is not isolated within 2 hours, is assumed to flood and fail all emergency AC power (see Sequence 5 description in Section 1.4). As shown in the next section, this initiating event (FLDG2) contributes to loss of injection end state Class IA.

3.4.2.2 Initiating Event Importance

Each initiating event that contributes at least 2% to the total core damage frequency is summarized below:

Initiator	Core Damage Frequency	Fraction of Core Damage
BLOSP *	5.4E-6	0.17
A2X	4.9E-6	0.16
A1X	4.6E-6	0.15
LOSP *	2.7E-6	0.09
KAX	2.6E-6	0.08
KBX	2.2E-6	0.07
LOC	1.0E-6	0.03
FLDG2	9.3E-7	0.03
TT **	9.0E-7	0.03
ATT **	6.7E-7	0.02
ASX	5.4E-7	0.02

* BLOSP and LOSP are the same initiating event. BLOSP is linked to the station blackout model. LOSP is linked to the general transient model.

** TT and ATT are the same initiating event. ATT is linked to the ATWS model. TT is linked to the general transient model.

The contribution of each of the above initiating events to loss of decay heat removal (DHR) and loss of injection end states is summarized in a table on the next page.

The above results indicate that AC power and, in general, support system initiating events are important (more than 75 percent). AC power is important because most of the other support and frontline systems depend on AC power to operate. Loss of normal offsite AC power and sequences leading to station blackout is the most important initiating event (BLOSP). Station blackout results in loss of all injection systems except RCIC and AC power must be restored before RCIC fails.

Loss of either Division I or Division II Emergency AC (A1X or A2X) is an important initiator for similar reasons discussed above. Although less likely, these initiators cause a loss of all safety systems on one division and lead to the isolation of non-safety systems from the service water system. Isolation of reactor building and turbine building component cooling systems results in loss of the condenser and the feedwater system, as well as loss of instrument air. A2X results in the unavailability of containment venting (nitrogen supply isolation valves fail closed and can not be manually opened) and thus is important to loss of containment heat removal. Both A2X and A1X are important contributors to loss of injection sequences because loss of the opposite division of DC or AC power causes loss of all ECCS pumps. Their contributions to the functional core damage groups are summarized below in a table.

A partial loss of normal AC power (KAX and KBX) is also important because it leads to a diesel challenge, challenges the reliability of service water, and isolates the non-safety cooling systems from service water. Their contributions to the functional core damage groups are summarized below.

The LOSP initiating event and the resulting non-station blackout sequences are relatively important. As shown in the table below, approximately 55 percent of the sequences are associated with loss of high pressure injection and failure of the RPV depressurization function (Class IA). LOSP guarantees failure of feedwater (top event FW) and high pressure core spray requires its diesel for success which reduces system reliability compared to the case with all AC power available. Approximately 37 percent of the LOSP sequences are associated with loss of decay heat removal (Class II). LOSP guarantees failure of the condenser as a heat sink and challenges the RHR and containment venting functions for successful heat removal.

Initiating Event Contribution to Class I and II End States

Initiating Event	Loss of DHR	Loss of Injection		
	Class II	Class IA	Class ID	ClassIB
BLOSP	<1E-7	-	-	5.4E-6
A2X	1.4E-6	4.7E-7	3.0E-6	-
A1X	3.1E-7	5.2E-7	3.8E-6	-
LOSP	1.0E-6	1.5E-6	1.8E-7	-
KAX	1.2E-6	1.1E-6	2.2E-7	-
KBX	1.4E-6	3.3E-7	4.8E-7	-
LOC	9.2E-7	<1E-7	<1E-7	-
FLDG2	<1E-7	9.3E-7	<1E-7	-
TT	4.1E-7	1.3E-7	3.6E-7	-
ATT	<1E-7	-	-	-
ASX	3.5E-7	1.7E-7	<1E-7	-

Note: Calculated frequencies less than 1E-7 are not reported because the contribution from the unaccounted event sequence cutoff frequency begins to contribute to the calculated value.

3.4.2.3 Event Tree Top Event Importance

One top event importance measure can be provided by calculating the contribution to total core damage of sequences that contain the top event failure. For each top event, the total is calculated and contributions from guaranteed failure (GF) of the top event and non-guaranteed failures (NGF or probabilistic) are provided.

Table 3.4.2-1 ranks the 50 most important top events based on total importance. As shown the guaranteed failure contribution is important. Guaranteed failure means the top event failure was set to 1.0 because of dependencies (i.e., support systems) on other top events that have failed in the model. The contribution from guaranteed failure is not related to reliability of the system, but is due to dependencies on other systems. Therefore, reducing the importance of these top events requires an improvement in the system whose failure caused the guaranteed failure or a design change to remove the dependency. Non-safety systems such as instrument air (AS), TBCLC (TW), RBCLC (RW), the condenser (CN), and feedwater (FW) systems show up at the top because of the importance of AC power systems as initiating events. Loss of AC initiators lead to loss of motive power to equipment and isolation of service water cooling to non-safety equipment as described previously.

Table 3.4.2-2 ranks the 50 most important top events based on all other sequences that contain failure of the top event except for the guaranteed failure (probabilistic or NGF). In this case, it is possible to change core damage frequency by changing the reliability of the system. The results in Table 3.4.2-2 suggest the following ranking of systems:

- Emergency AC Power (A2 and A1)
- AC Power Recovery (I1, G1, I2, G2)
- RHR System (LA and LB)
- Containment Venting (CV, includes operator action)
- HPCS (HS)
- Service Water System (SA and SB, includes operator)
- Containment Failure Causes Core Damage (CF)
- RCIC System (IC, U1, U2)
- DC Power (DA and DB)
- ECCS Pump Room Cooling (MB and MA, includes operator)

AC power top events and RCIC top events U1 & U2 impact the station blackout sequence frequencies. Containment venting, RHR, and service water impact long term decay heat removal sequences. Service water, RCIC (IC), HPCS, DC power and ECCS room cooling impact the loss of injection sequence frequencies. The important split fractions for these top events are identified in the next section.

3.4.2.4 Split Fraction Importance

Event tree top event failures (unavailabilities) are quantified under several boundary conditions (i.e., inter-system dependencies). These unavailabilities are referred to as split fractions which are identified by a three character code. The first two characters define the event tree top event and the third defines unique split fractions for the top event. Split fraction "Importance" has been calculated as the fractional contribution to total core damage frequency of sequences that contain the split fraction. Table 3.4.2-3 ranks split fractions by "Importance" of approximately 0.01 or greater and excludes the guaranteed failure split fractions from the calculations. This ranking of split fractions also satisfies another importance measure referred to as "Risk Reduction Worth" which is defined as the factor decrease in core damage frequency when the top event split fraction is set to guaranteed success (0.0). "Risk Reduction Worth" is equal to [1 - "Importance"]. These split fractions in Table 3.4.2-3 coincides with the important top events identified in Table 3.4.2-2.

A third importance measure is "Risk Achievement Worth" which is the factor increase in core damage frequency when the top event split fractions are set to guaranteed failure (1.0). Table 3.4.2-4 ranks the top 50 split fractions by risk achievement worth. The most important systems or operator actions in this ranking typically have high availabilities and have a direct impact on core damage frequency such that a significant reduction in availability can have a significant impact on core damage frequency.

3.4.2.5 Human Action Importance

Operator actions are included in several of the event tree top events discussed in the previous sections. Table 3.4.2-5 summarizes top events and split fractions that contain operator actions. This table also indicates whether the split fraction contains a contribution from equipment failures in addition to the human actions. When both human and equipment contributions exist, it is important to evaluate both of these contributions before drawing conclusions about the importance of human actions. Table 3.4.2-6 provides an "Importance" ranking of split fractions that contain operator actions. In addition, "Risk Reduction Worth" and "Risk Achievement Worth" importance measures are provided.

3.4.2.6 Important Contributors to Split Fractions

Each split fraction may contain one or more of the following contributions to unavailability of the system:

- Equipment failures including common cause
- Test & maintenance unavailability
- Human error

Contributions to each split fraction can be easily obtained from the computer model including basic event importance. This is particularly useful in evaluating the importance of maintenance unavailability of systems and provides an input to the maintenance program. Contributions from the systems analysis for selected important split fractions are provided in Table 3.4.2-7. The split fractions were selected based on a review of Section 3.4.2.4.

3.4.3 Decay Heat Removal Evaluation

NRC has requested plant specific evaluations and resolution of unresolved safety issue A-45 to be contained in the IPE. The importance of decay heat removal can be derived from the

NMP2 IPE results as described in the previous sections above and further described in this section.

The following systems/functions can provide successful decay heat removal in the IPE model:

- Main Condenser - Event tree top event CN models the availability of the main condenser which is the preferred method of decay heat removal.

- RHR System - Given condenser unavailability, event tree top events OH, LA, HA, PA, CA and IA model alignment of RHR train A in the heat removal mode. OH models the operator actions, LA models the pump train, HA models the heat exchanger, PA models the suppression pool return path, CA models the containment spray return path, and IA models the injection path. As a minimum, top events OH, LA and HA are required for success and at least one of the paths back to the RPV or suppression pool is required. The success criteria for one of the three return paths depends on the initiating event. Similarly, event tree top events OH, LB, HB, PB, CB and IB model RHR train B. Either train provides successful decay heat removal.

- Containment Venting - Given loss of RHR, event tree top event CV models opening the suppression chamber purge exhaust and venting through the stack to prevent severe containment overpressure conditions and a heat removal path.

- Containment Failure Mode - Given that all the above systems are unavailable, event tree top events CI and CF model continued injection during severe containment overpressure conditions (CI) and whether containment failure causes core damage (CF). If either top event fails, core damage occurs.

Failure of the above systems/functions results in core damage and the sequences are grouped in the following core damage end states also described in Section 3.1.5:

- CLASS IIA: Loss of containment heat removal and core damage induced post containment failure (CF failure) given transient or small LOCA.

- CLASS IIT: Loss of containment heat removal and core damage induced prior to containment failure (CI failure) given transient or small LOCA.

- CLASS IIL: Loss of containment heat removal and core damage induced post containment failure (CF failure) given medium or large LOCA.

In general, the model questions loss of injection and bins these sequences prior to the binning of loss of heat removal sequences. Therefore, it is possible to have failures (i.e., support system failures) that cause both loss of injection and decay heat removal. However, loss of injection is an immediate concern and is more likely to cause core damage much earlier in time relative to loss of long term decay heat removal. These loss of injection sequences are neglected in this evaluation. It should be noted that loss of injection results in loss of heat removal from the core and it is assumed that "loss of decay heat removal" is concerned with the long term loss of decay heat removal and the systems identified above. The percent contribution to total core damage is provided below for each of the three end states and the total of all three:

Core Damage End State	Frequency	Fraction of Total
CLASS IIA	4.7E-6	0.15
CLASS IIT	4.2E-6	0.13
CLASS IIL	3.0E-7	0.01
TOTAL	9.2E-6	0.29

As described in previous sections, station blackout and loss of injection make up most of the remaining core damage frequency.

The importance of event tree top events to the total decay heat removal core damage frequency described above is provided in Table 3.4.3-1. In addition, important support systems that impact loss of decay heat are included in the table. Containment venting (CV) and RHR pump trains (LA and LB) are most important which is to be expected. Any improved availability to these systems would provide the greatest reduction in the frequency of loss of decay heat removal sequences. Note that if CV was made perfect, 41 percent of the sequences would remain due to support system failures that guarantee CV failure. Service water (SA and SB) and emergency AC Division II (A2) are the most important support systems. Service water impacts pump room cooling, heat removal from the RHR heat exchangers, and condenser support systems. Emergency AC Division II guarantees failure of containment venting, guarantees failure of RHR Division II (LB), and condenser support systems. Another observation from Table 3.4.3-1 has to do with the importance of the condenser which is the preferred heat removal source. Note that the condenser is guaranteed to be unavailable in 95 percent of the sequences. Thus, the importance of the condenser has little to do with the condenser itself. This again points to the importance of support system failures as described throughout this section. Split fraction importance for the above top events is provided in Table 3.4.3-2. The alignment and cutset importance to split fractions are discussed in Section 3.4.2.6. The contribution to containment venting split fractions is provided below.

SPLIT FRACTION	DESCRIPTION	UNAVAILABILITY		
		EQUIPMENT	OPERATORS	TOTAL
CV1	All support avail	0.022	0.006	0.028
CV2	Loss of air or AC	0.022	0.014	0.036
CV5	Loss of Nitrogen	0.022	0.009	0.031

As shown in Table 3.4.3-2, split fraction CV2 is the most important and then CV1. The frequency of loss of decay heat removal can be reduced by improving the reliability of the equipment assessed in this study.

3.4.4 USI and GSI Screening

3.4.4.1 Adequacy of Safety-Related DC Power Supplies

NMPC's response to NRC Generic Letter 91-06 regarding the "Adequacy of Safety-Related DC Power Supplies", Generic Issue A-30, committed to address two NRC questions through the IPE submittal. The following summarizes the two questions:

1. Does the control room have a separate, independent annunciated alarms and indications for each division of DC power, for battery charger disconnect, or circuit breaker open (both input AC and output DC)? NMPC response was NO indicating the need for this provision would be evaluated in the NMP2 IPE submittal.

2. Does the unit have indication of bypass and inoperable status of circuit breakers or other devices that can be used to disconnect the battery charger from its DC bus and the battery charger from its AC power source during maintenance or testing? NMPC response was NO indicating the need for this provision would be evaluated in the NMP2 IPE submittal.

The above considerations were included in the DC power systems analysis and assessed to have a minor contribution to system unavailability. The following summarizes the reasons for this conclusion:

1. Each division of DC power contains redundant chargers. If the AC input circuit breaker to the active charger or the charger internal circuit breakers were to open, a charger trouble annunciator would alarm in the control room. This alarm is common to both chargers in a division, but is separate between divisions. There is a common output circuit breaker from the redundant chargers in each division. If the output breaker was open on a division, a low voltage alarm would occur within a few hours as the battery voltage depletes. The likelihood of this occurring is small given the independent check of system alignment requirements and the short period of time that this misalignment would be undetectable. Thus, this contribution was judged to be insignificant.

2. Each battery supply has a circuit breaker and an annunciator alarms in the control room indicating breaker status. Computer points indicate the specific breaker that is open. This alarm is not bypassed or made inoperable during system testing or maintenance. Thus, it is considered unlikely that a circuit breaker could be left open for long without being noticed in the control room.

The Division I and II DC power supplies are modeled in event tree top events DA, DB, D1, and D2. DA and DB model the availability of the battery on demand for Division I and II, respectively. D1 and D2 model the battery and charger availability along with the switchgear for Division I and II, respectively. The importance of these top events is included in the results described in the previous sections. As shown in Table 3.4.2-2, 3, and 4, DC power is an important system, however, the contribution from the above concerns were not assessed to contribute significantly to system unavailability.

3.4.4.2 Other USIs and GSIs

NMPC has reviewed the USIs and GSIs in NUREG-0933, "A Prioritization of Generic Safety Issues", and agree that many can and should be resolved on a plant-specific basis using the IPE as an input. However, no other USIs or GSIs are currently being addressed at this time.

Table 3.4.1-1
Top 100 Core Damage Sequences

Rank.	Initiator.	Index....	Frequency......	Failed and Multi-State Split Fractions.	End State.
1	BLOSP	117	2.5627E-06	/OGF*KAF*KBF*KRF*A12*A28*NAF*NBF*SAF*SBF *RWF*TWF*MAF*MBF*ASF/I11*G11*U11/NLF*NMF	CLASSIB
2	A2X	94	2.3667E-06	/DA1*A2F*D1F*E1F*SAF*SBF*RWF*TWF*HBF*ASF /CNF*FWF*HSF*ICF*LSF*LCF*LAF*LBF*IBF*SWF /NLF*NMF*HAF*HBF*CAF*CBF	CLASSID
3	A1X	71	2.2735E-06	/DB1*A1F*D2F*E2F*SAF*SBF*RWF*TWF*HAF*ASF /CNF*FWF*HSF*ICF*LSF*LCF*LAF*LBF*IAF*SWF /NLF*NMF*HAF*HBF*CAF*CBF	CLASSID
4	A2X	25	1.0377E-06	/A2F*SAH*SBK*RWF*TWF*HBF*ASF/CNF*FWF*HSF *ICF*LBF*IBF*SWF/NLF*HAF*HBF*CAF*CBF*CVF *R1F*CIF	CLASSIIT
5	FLDG2	1	9.1257E-07	/A1F*A2F*SAF*SBF*RWF*TWF*HAF*HBF*ASF/CNF *FWF*HSF*ICF*SVF*LSF*LCF*LAF*LBF*IAF*IBF *SWF*FPF/NLF*NMF*HAF*HBF*CAF*CBF	CLASSIA
6	A1X	27	8.9517E-07	/A1F*SAG*SBL*RWF*TWF*HAF*HBA*ASF/CNF*FWF *HSF*ICF*LSF*LCF*LAF*LBF*IAF*IBF*SWF*FPF /NLF*NMF*HAF*HBF*CAF*CBF	CLASSID
7	BLOSP	115	8.4764E-07	/OGF*KAF*KBF*KRF*A12*A28*NAF*NBF*SAF*SBF *RWF*TWF*MAF*MBF*ASF/I11*G11*I21*G21*U21 *S11/NLF*NMF	CLASSIB
8	KAX	23	8.2098E-07	/KAF*NAF*RWF*TWF*ASF/CNF*FWF*HS2*IC1*OO1 /NLF*NMF	CLASSIA
9	LOSP	127	7.6014E-07	/OGF*KAF*KBF*KRF*NAF*NBF*RWF*TWF*ASF/CNF *FWF*HS2*IC1*OO1/NLF*NMF	CLASSIA
10	LOC	13	4.1658E-07	//CNF*LA1*LBA/NLF*HAF*HBF*CAF*CBF*CV1*R1 F*CF1	CLASSIIA
11	A2X	85	3.8035E-07	/A11*A2F*SAF*SBF*RWF*TWF*MAF*MBF*ASF/CNF *FWF*HSF*ICF*SVF*LSF*LCF*LAF*LBF*IAF*IBF *SWF*FPF/NLF*NMF*HAF*HBF*CAF*CBF	CLASSIA
12	A1X	64	3.7913E-07	/A1F*A25*SAF*SBF*RWF*TWF*HAF*HBF*ASF/CNF *FWF*HSF*ICF*SVF*LSF*LCF*LAF*LBF*IAF*IBF *SWF*FPF/NLF*NMF*HAF*HBF*CAF*CBF	CLASSIA
13	BLOSP	2	3.6599E-07	/OGF*KAF*KBF*KRF*DB1*A12*A2F*NAF*NBF*D2F *UBF*E2F*SAF*SBF*RWF*TWF*MAF*MBF*ASF/I11 *G1F*U1F/NLF*NMF	CLASSIB
14	BLOSP	17	3.6018E-07	/OGF*KAF*KBF*KRF*DA1*A1F*A29*NAF*NBF*D1F *UAF*E1F*SAF*SBF*RWF*TWF*MAF*MBF*ASF/I11 *G1F*U1F/NLF*NMF	CLASSIB
15	KBX	109	3.4837E-07	/KBF*A11*NBF*SAF*SBF*RWF*TWF*MAF*HBA*ASF /CNF*FWF*HSF*ICF*LSF*LCF*LAF*LBF*IAF*IBF *SWF*FPF/NLF*NMF*HAF*HBF*CAF*CBF	CLASSID
16	KBX	6	3.4252E-07	/KBF*NBF*RWF*TWF*ASF/CNF*FWF*LA1*LBA/NLF *HAF*HBF*CAF*CBF*CV2*R1F*CF4	CLASSIIA
17	KBX	168	3.2022E-07	/KBF*KR1*A23*NBF*RWF*TWF*HBF*ASF/CNF*FWF *LA1*LBF*IBF*SWF/NLF*HAF*HBF*CAF*CBF*CVF *R1F*CF4	CLASSIIA
18	D1X	25	3.1274E-07	/A21*D1F*E1F*SAF*SBF*RWF*TWF*HBF*ASF/CNF *FWF*HSF*ICF*LSF*LCF*LAF*LBF*IBF*SWF/NLF *NMF*HAF*HBF*CAF*CBF	CLASSID
19	D2X	18	3.0045E-07	/A11*D2F*E2F*SAF*SBF*RWF*TWF*MAF*ASF/CNF *FWF*HSF*ICF*LSF*LCF*LAF*LBF*IAF*SWF/NLF *NMF*HAF*HBF*CAF*CBF	CLASSID
20	KAX	6	2.9410E-07	/KAF*NAF*RWF*TWF*ASF/CNF*FWF*LA1*LBA/NLF *HAF*HBF*CAF*CBF*CV2*R1F*CF4	CLASSIIA

Table 3.4.1-1
Top 100 Core Damage Sequences

Rank.	Initiator.	Index....	Frequency......	Failed and Multi-State Split Fractions.	End State.
21	LOC	14	2.8965E-07	//CNF*LA1*LBA/NLF*HAF*HBF*CAF*CBF*CV1*R1F*CI1	CLASSIIT
22	ASX	6	2.7829E-07	/ASF/CNF*FWF*LA1*LBA/NLF*HAF*HBF*CAF*CBF*CV2*R1F*CF2	CLASSIIA
23	ATT	37	2.7346E-07	//QM1*SL1/NLF*NMF	CLASSIVA
24	LOSP	110	2.7230E-07	/OGF*KAF*KBF*KRF*NAF*NBF*RWF*TWF*ASF/CNF*FWF*LA1*LBA/NLF*HAF*HBF*CAF*CBF*CV2*R1F*CF4	CLASSIIA
25	A1X	80	2.6846E-07	/DB1*A1F*D2F*E2F*SAF*SBF*RWF*TWF*HAF*MBA*ASF/CNF*FWF*HSF*ICF*LSF*LCF*LAF*LBF*IAF*IBF*SWF*FPF/NLF*NMF*HAF*HBF*CAF*CBF	CLASSID
26	A1X	6	2.6794E-07	/A1F*SAG*SBL*RWF*TWF*MAF*ASF/CNF*FWF*HSF*ICF*LSF*LAF*IAF*SWF/NLF*HAF*HBF*CAF*CBF*CV2*R1F*CIF	CLASSIIT
27	LOSP	230	2.6458E-07	/OGF*KAF*KBF*KRF*A24*NAF*NBF*SBF*RWF*TWF*MBF*ASF/CNF*FWF*LA1*LBF*IBF*SWF/NLF*HAF*HBF*CAF*CBF*CVF*R11*CF4	CLASSIIA
28	A2X	100	2.6404E-07	/DA1*A2F*D1F*E1F*SAF*SBF*RWF*TWF*MAA*MBF*ASF/CNF*FWF*HSF*ICF*LSF*LCF*LAF*LBF*IAF*IBF*SWF*FPF/NLF*NMF*HAF*HBF*CAF*CBF	CLASSID
29	LOSP	250	2.4508E-07	/OGF*KAF*KBF*KRF*A24*NAF*NBF*SBF*RWF*TWF*MBF*ASF/CNF*FWF*HS4*IC1*SV7*LBF*IBF*SWF/NLF*NMF*HBF	CLASSIA
30	MLOCA	8	2.3282E-07	//HS1*OD2/NMF	CLASSIIIB
31	KBX	35	2.2822E-07	/KBF*NBF*SAC*SBO*RWF*TWF*ASF/CNF*FWF*HSF*ICF*SWF/NLF*HAF*HBF*CAF*CBF*CV2*R1F*CIF	CLASSIIT
32	KAX	100	2.1970E-07	/KAF*NAF*SA7*SBT*RWF*TWF*ASF/CNF*FWF*HSF*ICF*SWF/NLF*HAF*HBF*CAF*CBF*CV2*R1F*CIF	CLASSIIT
33	LOF	2	2.1583E-07	//FWF*HS1*IC1*OD1/NLF*NMF	CLASSIA
34	LOSP	27	2.1367E-07	/OGF*KAF*KBF*KRF*A12*NAF*NBF*SAF*RWF*TWF*MAF*ASF/CNF*FWF*HS4*IC1*SV5*LSF*LAF*IAF/NLF*NMF*HAF*CAF	CLASSIA
35	FLDG1	4	2.0423E-07	/SAF*RWF*TWF*ASF/CNF*FWF*LB1/NLF*HAF*HBF*CAF*CBF*CV2*R1F*CF4	CLASSIIA
36	RWX	6	1.9761E-07	/RWF*ASF/CNF*FWF*LA1*LBA/NLF*HAF*HBF*CAF*CBF*CV2*R1F*CF4	CLASSIIA
37	IORV	6	1.9669E-07	//LA1*LBA/HAF*HBF*CAF*CBF*CV1*CFF	CLASSIIL
38	ATT	6	1.7187E-07	//QM1*FW3/NLF*MO1*ILF*CH1	CLASSIVA
39	KBX	14	1.6481E-07	/KBF*NBF*RWF*TWF*ASF/CNF*FWF*HS1*IC1*OD1/NLF*NMF	CLASSIA
40	BLOSP	107	1.5732E-07	/OGF*KAF*KBF*KRF*A12*A28*NAF*NBF*SAF*SBF*RWF*TWF*MAF*MBF*ASF/I11*G11*I21*G21*I31*G31*U31*S21/NLF*NMF	CLASSIB
41	SAX	4	1.4580E-07	/SAF*RWF*TWF*ASF/CNF*FWF*LB1/NLF*HAF*HBF*CAF*CBF*CV2*R1F*CF4	CLASSIIA
42	KBX	152	1.4550E-07	/KBF*KR1*NBF*SAC*SBO*RWF*TWF*MBF*ASF/CNF*FWF*HSF*ICF*LBF*IBF*SWF/NLF*HAF*HBF*CAF*CBF*CVF*R1F*CIF	CLASSIIT
43	A1X	72	1.3776E-07	/DB1*A1F*D2F*E2F*SAF*SBF*RWF*TWF*MAF*ASF/CNF*FWF*HSF*ICF*LSF*LCF*LAF*LBF*IAF*IBA*SWF*FPF/NLF*NMF*HAF*HBF*CAF*CBF	CLASSID

Table 3.4.1-1
Top 100 Core Damage Sequences

Rank.	Initiator.	Index....	Frequency......	Failed and Multi-State Split Fractions.	End State.
44	ASX	14	1.3724E-07	/ASF/CNF*FWF*HS1*IC1*OD1/NLF*NMF	CLASSIA
45	IORV	11	1.3289E-07	//HSA*OD2/NMF	CLASSIIIB
46	A2X	36	1.2022E-07	/A2F*SAH*SBK*RWF*TWF*MAA*MBF*ASF/CNF*FWF *HSF*ICF*LSF*LCF*LAF*LBF*IAF*IBF*SWF*FPF /NLF*NMF*HAF*HBF*CAF*CBF	CLASSID
47	BLOSP	100	1.1767E-07	/OGF*KAF*KBF*KRF*A12*A28*NAF*NBF*SAF*SBF *RWF*TWF*MAF*MBF*ASF/I11*G11*I21*G21*I31 *G31*I41*G41*X31/NLF*NMF	CLASSIB
48	ATT	10	1.1568E-07	//QM1*FW3/NLF*IC1*OE1*CH2	CLASSIC
49	TT	189	1.1384E-07	/DA1*A21*D1F*E1F*SAF*SBF*RWF*TWF*MBF*ASF /CNF*FWF*HSF*ICF*LSF*LCF*LAF*LBF*IBF*SWF /NLF*NMF*HAF*HBF*CAF*CBF	CLASSID
50	KAX	275	1.1367E-07	/KAF*KR2*A22*NAF*SAF*SBF*RWF*TWF*MAF*MBF *ASF/CNF*FWF*HSF*ICF*SVF*LSF*LCF*LAF*LBF *IAF*IBF*SWF*FPF/NLF*NMF*HAF*HBF*CAF*CBF	CLASSIA
51	KAX	51	1.1066E-07	/KAF*NAF*RWF*TWF*ASF*N23/CNF*FWF*LA1*LBA /NLF*HAF*HBF*CAF*CBF*CVF*R1F*CF4	CLASSIIA
52	TT	166	1.0937E-07	/DB1*A11*D2F*E2F*SAF*SBF*RWF*TWF*MAF*ASF /CNF*FWF*HSF*ICF*LSF*LCF*LAF*LBF*IAF*SWF /NLF*NMF*HAF*HBF*CAF*CBF	CLASSID
53	FLSW	1	1.0815E-07	/A21*SAF*SBF*RWF*TWF*MBF*ASF/CNF*FWF*HSF *ICF*LBF*IBF*SWF/NLF*HAF*HBF*CAF*CBF*CVF *R1F*CIF	CLASSIIT
54	TWX	6	1.0770E-07	/TWF/CNF*FWF*LA1*LBA/NLF*HAF*HBF*CAF*CBF *CV1*R1F*CF2	CLASSIIA
55	A2X	53	1.0730E-07	/A2F*NB1*RWF*TWF*MBF*ASF/CNF*FWF*LA1*LBF *IBF*SWF/NLF*HAF*HBF*CAF*CBF*CVF*R1F*CF4	CLASSIIA
56	A2X	68	1.0630E-07	/A2F*NA1*RWF*TWF*MBF*ASF/CNF*FWF*LA1*LBF *IBF*SWF/NLF*HAF*HBF*CAF*CBF*CVF*R1F*CF4	CLASSIIA
57	KBX	94	1.0427E-07	/KBF*A11*NBF*SAF*SBF*RWF*TWF*MAF*ASF/CNF *FWF*HSF*ICF*LSF*LAF*IAF*SWF/NLF*HAF*HBF *CAF*CBF*CV2*R1F*CIF	CLASSIIT
58	LOSP	155	1.0246E-07	/OGF*KAF*KBF*KRF*NAF*NBF*RWF*TWF*ASF*N23 /CNF*FWF*LA1*LBA/NLF*HAF*HBF*CAF*CBF*CVF *R1F*CF4	CLASSIIA
59	BLOSP	119	9.8701E-08	/OGF*KAF*KBF*KRF*A12*A28*NAF*NBF*SAF*SBF *RWF*TWF*MAF*MBF*ASF/I11*G11*OA1*I21*G21 /NLF*NMF	CLASSIB
60	RWX	14	9.5088E-08	/RWF*ASF/CNF*FWF*HS1*IC1*OD1/NLF*NMF	CLASSIA
61	BLOSP	105	8.8805E-08	/OGF*KAF*KBF*KRF*A12*A28*NAF*NBF*SAF*SBF *RWF*TWF*MAF*MBF*ASF/I11*G11*I21*G21*I31 *G31*U31*I41*G41*X31/NLF*NMF	CLASSIB
62	BLOSP	106	8.5442E-08	/OGF*KAF*KBF*KRF*A12*A28*NAF*NBF*SAF*SBF *RWF*TWF*MAF*MBF*ASF/I11*G11*I21*G21*I31 *G31*U31*X21/NLF*NMF	CLASSIB
63	KAX	110	8.2660E-08	/KAF*NAF*SA7*SBT*RWF*TWF*ASF*N23/CNF*FWF *HSF*ICF*SWF/NLF*HAF*HBF*CAF*CBF*CVF*R1F *CIF	CLASSIIT
64	KBX	62	8.2103E-08	/KBF*NBF*SAC*SBO*RWF*TWF*MAA*MBA*ASF/CNF *FWF*HSF*ICF*LSF*LCF*LAF*LBF*IAF*IBF*SWF *FPF/NLF*NMF*HAF*HBF*CAF*CBF	CLASSID

Table 3.4.1-1
Top 100 Core Damage Sequences

Rank.	Initiator.	Index....	Frequency......	Failed and Multi-State Split Fractions.	End State.
65	A1X	84	7.9711E-08	/DB1*A1F*D2F*E2F*HE1*SAF*SBF*RWF*TWF*MAF *ASF/CNF*FWF*HSF*ICF*LSF*LCF*LAF*LBF*IAF *IBF*SWF*FPF/NLF*NMF*HAF*HBF*CAF*CBF	CLASSID
66	KAX	144	7.9170E-08	/KAF*NAF*SA7*SBT*RWF*TWF*MAA*MBA*ASF/CNF *FWF*HSF*ICF*LSF*LCF*LAF*LBF*IAF*IBF*SWF *FPF/NLF*NMF*HAF*HBF*CAF*CBF	CLASSID
67	A2X	104	7.8401E-08	/DA1*A2F*D1F*E1F*ME1*SAF*SBF*RWF*TWF*MBF *ASF/CNF*FWF*HSF*ICF*LSF*LCF*LAF*LBF*IAF *IBF*SWF*FPF/NLF*NMF*HAF*HBF*CAF*CBF	CLASSID
68	SWX	1	7.7825E-08	/SAF*SBF*RWF*TWF*ASF/CNF*FWF*HSF*ICF*SWF /NLF*HAF*HBF*CAF*CBF*CV2*R1F*CIF	CLASSIIT
69	BLOSP	101	7.7533E-08	/OGF*KAF*KBF*KRF*A12*A28*NAF*NBF*SAF*SBF *RWF*TWF*MAF*MBF*ASF/I11*G11*I21*G21*I31 *G31*I41*G41*S31/NLF*NMF	CLASSIB
70	KAX	15	7.6298E-08	/KAF*NAF*RWF*TWF*ASF/CNF*FWF*HS2*LA1*LBA /NLF*HAF*HBF*CAF*CBF*CV2*R1F*CIF	CLASSIIT
71	KBX	189	7.2372E-08	/KBF*KR1*A11*NBF*SAF*SBF*RWF*TWF*MAF*MBF *ASF/CNF*FWF*HSF*ICF*SVF*LSF*LCF*LAF*LBF *IAF*IBF*SWF*FPF/NLF*NMF*HAF*HBF*CAF*CBF	CLASSIA
72	LOSP	236	7.1600E-08	/OGF*KAF*KBF*KRF*A24*NAF*NBF*SBF*RWF*TWF *MBF*ASF/CNF*FWF*HS4*LA1*LBF*IBF*SWF/NLF *HAF*HBF*CAF*CBF*CVF*R11*CIF	CLASSIIT
73	LOSP	119	7.0644E-08	/OGF*KAF*KBF*KRF*NAF*NBF*RWF*TWF*ASF/CNF *FWF*HS2*LA1*LBA/NLF*HAF*HBF*CAF*CBF*CV2 *R1F*CIF	CLASSIIT
74	TT	57	7.0512E-08	/NB1*RWF*TWF*ASF/CNF*FWF*LA1*LBA/NLF*HAF *HBF*CAF*CBF*CV2*R1F*CF4	CLASSIIA
75	BLOSP	51	7.0450E-08	/OGF*KAF*KBF*KRF*A24*NAF*NBF*SA6*SBF*RWF *TWF*MAF*MBF*ASF/I11*G12*U11/NLF*NMF	CLASSIB
76	FLDG1	46	7.0276E-08	/A21*SAF*SBF*RWF*TWF*MBF*ASF/CNF*FWF*HSF *ICF*LBF*IBF*SWF/NLF*HAF*HBF*CAF*CBF*CVF *R1F*CIF	CLASSIIT
77	N2X	10	7.0224E-08	/N2F/CNF*LA1*LBA/NLF*HAF*HBF*CAF*CBF*CV5 *R1F*CF1	CLASSIIA
78	BLOSP	78	7.0182E-08	/OGF*KAF*KBF*KRF*A12*NAF*NBF*SAF*SBU*RWF *TWF*MAF*MBF*ASF/I11*G12*U11/NLF*NMF	CLASSIB
79	TT	84	6.9973E-08	/NA1*RWF*TWF*ASF/CNF*FWF*LA1*LBA/NLF*HAF *HBF*CAF*CBF*CV2*R1F*CF4	CLASSIIA
80	TWX	14	6.8065E-08	/TWF/CNF*FWF*HS1*IC1*OD1/NLF*NMF	CLASSIA
81	ALOC	6	5.8369E-08	//QM1*CNF/NLF*MOF*ILF*CH1	CLASSIVA
82	TT	30	5.6188E-08	/SA1*SBZ*RWF*TWF*ASF/CNF*FWF*HSF*ICF*SWF /NLF*HAF*HBF*CAF*CBF*CV2*R1F*CIF	CLASSIIT
83	KAX	75	5.4853E-08	/KAF*NAF*RWF*TWF*ASF*N11*N2F/CNF*FWF*LA1 *LBA/NLF*HAF*HBF*CAF*CBF*CVF*R1F*CF4	CLASSIIA
84	KAX	161	5.4311E-08	/KAF*NAF*E11*E2A*HE2*RWF*TWF*ASF/CNF*FWF *HS2*ICF*LSF*LCF*IAF*IBF*SWF*FPF/NLF*NMF	CLASSID
85	N2X	61	5.0821E-08	/NA1*RWF*TWF*ASF*N2F/CNF*FWF*LA1*LBA/NLF *HAF*HBF*CAF*CBF*CVF*R1F*CF4	CLASSIIA
86	LOSP	178	5.0788E-08	/OGF*KAF*KBF*KRF*NAF*NBF*RWF*TWF*ASF*N11 *N2F/CNF*FWF*LA1*LBA/NLF*HAF*HBF*CAF*CBF *CVF*R1F*CF4	CLASSIIA

Table 3.4.1-1
Top 100 Core Damage Sequences

Rank.	Initiator.	Index....	Frequency......	Failed and Multi-State Split Fractions.	End State.
87	N2X	47	5.0537E-08	/NB1*RWF*TWF*ASF*N2F/CNF*FWF*LA1*LBA/NLF *HAF*HBF*CAF*CBF*CVF*R1F*CF4	CLASSIIA
88	LOSP	214	5.0286E-08	/OGF*KAF*KBF*KRF*NAF*NBF*E11*E2A*ME2*RWF *TWF*ASF/CNF*FWF*HS2*ICF*LSF*LCF*IAF*IBF *SWF*FPF/NLF*NMF	CLASSID
89	SAX	41	5.0172E-08	/A21*SAF*SBF*RWF*TWF*MBF*ASF/CNF*FWF*HSF *ICF*LBF*IBF*SWF/NLF*HAF*HBF*CAF*CBF*CVF *R1F*CIF	CLASSIIT
90	TT	125	4.9915E-08	/A21*SAH*SBK*RWF*TWF*MBF*ASF/CNF*FWF*HSF *ICF*LBF*IBF*SWF/NLF*HAF*HBF*CAF*CBF*CVF *R1F*CIF	CLASSIIT
91	N2X	11	4.8828E-08	/N2F/CNF*LA1*LBA/NLF*HAF*HBF*CAF*CBF*CV5 *R1F*CI1	CLASSIIT
92	BLOSP	114	4.8379E-08	/OGF*KAF*KBF*KRF*A12*A28*NAF*NBF*SAF*SBF *RWF*TWF*MAF*MBF*ASF/I11*G11*I21*G21*U21 *X11/NLF*NMF	CLASSIB
93	A1X	122	4.7159E-08	/OG1*KAF*KBF*KRF*A1F*A28*NAF*NBF*SAF*SBF *RWF*TWF*MAF*MBF*ASF/CNF*FWF*HSF*ICF*SVF *LSF*LCF*LAF*LBF*IAF*IBF*SWF*FPF/NLF*NMF *HAF*HBF*CAF*CBF	CLASSIA
94	ALOC	24	4.6374E-08	//QM1*SL1/NLF*NMF	CLASSIVA
95	AMSIV	4	4.4899E-08	//QM1*CN3/NLF*MOF*ILF*CH1	CLASSIVA
96	LOSP	246	4.3748E-08	/OGF*KAF*KBF*KRF*A24*NAF*NBF*SBF*RWF*TWF *MBF*ASF/CNF*FWF*HS4*IC1*OO1*LBF*IBF*SWF /NLF*NMF*HBF	CLASSIA
97	TT	149	4.3062E-08	/A11*SAG*SBL*RWF*TWF*MAF*MBA*ASF/CNF*FWF *HSF*ICF*LSF*LCF*LAF*LBF*IAF*IBF*SWF*FPF /NLF*NMF*HAF*HBF*CAF*CBF	CLASSID
98	A2X	44	4.2483E-08	/A2F*D11*E1F*SAF*SBF*RWF*TWF*MBF*ASF/CNF *FWF*HSF*ICF*LSF*LCF*LAF*LBF*IBF*SWF/NLF *NMF*HAF*HBF*CAF*CBF	CLASSID
99	KAX	117	4.0974E-08	/KAF*NAF*SA7*SBT*RWF*TWF*ASF*N11*N2F/CNF *FWF*HSF*ICF*SWF/NLF*HAF*HBF*CAF*CBF*CVF *R1F*CIF	CLASSIIT
100	A1X	37	4.0807E-08	/A1F*D22*E2F*SAF*SBF*RWF*TWF*MAF*ASF/CNF *FWF*HSF*ICF*LSF*LCF*LAF*LBF*IAF*SWF/NLF *NMF*HAF*HBF*CAF*CBF	CLASSID

Table 3.4.1-2
Initiating Events

Initiator	Description	Frequency
A1X	LOSS OF EMERGENCY AC DIV. I	4.3E-3
A2X	LOSS OF EMERGENCY AC DIV. II	4.3E-3
AIORV	ATWS IORV	1.5E-2
ALOC	ATWS LOC	3.9E-1
ALOF	ATWS LOF	5.2E-2
ALOSP	ATWS LOSP	4.0E-2
AMSIV	ATWS MSIV	3.0E-1
ASX	INSTRUMENT AIR FAILURE	3.3E-2
ATT	ATWS TT	2.3
BLOC	BLACKOUT LOC	3.9E-1
BLOF	BLACKOUT LOF	5.2E-2
BLOSP	BLACKOUT LOSP	4.0E-2
BMSIV	BLACKOUT MSIV	3.0E-1
BTT	BLACKOUT TT	2.3
D1X	FAILURE OF DIVISION I DC POWER	4.2E-3
D2X	DIV. II DC FAILURE	4.2E-3
FLDG1	FLOOD IN DIESEL ROOM ISOLATED	9.3E-4
FLDG2	FLOOD IN DIESEL ROOM UNISOLATED	9.3E-7
FLSW	FLOOD IN THE SERVICE WATER PUMP HOUSE	1.4E-3
IORV	STUCK OPEN RELIEF VALVE	1.5E-2
ISLOCA	INTERFACING SYSTEM LOCA	1.1E-4
KAX	LOSS OF 115KV SOURCE A	4.0E-2
KBX	LOSS OF 115KV SOURCE B	4.0E-2
LLOCA	LARGE LOCA	7.0E-4
LOC	LOSS OF CONDENSER	3.9E-1
LOF	LOSS OF FEEDWATER	5.2E-2
LOSP	LOSS OF OFFSITE AC	4.0E-1
MLOCA	MEDIUM LOCA	3.0E-3
MSIV	ISOLATION EVENT	3.0E-1
N2X	LOSS OF NITROGEN INITIATING EVENT	6.0E-2
RWX	RBCLC FAILURE	2.3E-2
SAX	LOSS OF SERVICE WATER HEADER A	6.6E-4
SLOCA	SMALL LOCA	8.0E-3
SWX	LOSS OF ALL SERVICE WATER	2.8E-6
TT	TURBINE TRIP	2.3
TWX	TBCLC FAILURE	1.6E-2

Table 3.4.2-1
Top Event Importance (Top 50)
Sorted by TOTAL Importance

Rank..	Top....	Guar. Event....	Probabilistic..	Total..........	Frequency......
1.	NL	9.7506E-01	0.0000E+00	9.7506E-01	3.0266E-05
2.	AS	8.9009E-01	8.9898E-04	8.9099E-01	2.7657E-05
3.	RW	8.7155E-01	1.9950E-03	8.7354E-01	2.7115E-05
4.	TW	8.6691E-01	1.2996E-04	8.6704E-01	2.6913E-05
5.	CN	7.5817E-01	2.9089E-03	7.6108E-01	2.3624E-05
6.	FW	7.2844E-01	1.2064E-02	7.4050E-01	2.2985E-05
7.	NM	6.8511E-01	0.0000E+00	6.8511E-01	2.1266E-05
8.	CA	6.7317E-01	4.9984E-04	6.7367E-01	2.0911E-05
9.	HA	6.6751E-01	1.5599E-03	6.6907E-01	2.0768E-05
10.	SA	5.3039E-01	1.3630E-01	6.6669E-01	2.0694E-05
11.	HB	6.6493E-01	1.3117E-03	6.6625E-01	2.0681E-05
12.	SB	5.4477E-01	1.1865E-01	6.6342E-01	2.0593E-05
13.	CB	6.5590E-01	9.8599E-04	6.5688E-01	2.0390E-05
14.	LB	4.7176E-01	1.4446E-01	6.1622E-01	1.9128E-05
15.	HS	4.6087E-01	1.5311E-01	6.1399E-01	1.9058E-05
16.	IC	4.7156E-01	1.2134E-01	5.9289E-01	1.8404E-05
17.	LA	4.0158E-01	1.7067E-01	5.7225E-01	1.7763E-05
18.	MB	4.7495E-01	6.7903E-02	5.4286E-01	1.6850E-05
19.	SW	5.2803E-01	9.0552E-04	5.2894E-01	1.6418E-05
20.	MA	4.4185E-01	3.1917E-02	4.7377E-01	1.4706E-05
21.	A2	2.0087E-01	2.5886E-01	4.5973E-01	1.4270E-05
22.	A1	1.9452E-01	2.3437E-01	4.2889E-01	1.3313E-05
23.	LS	4.0043E-01	6.2937E-03	4.0672E-01	1.2625E-05
24.	IB	3.7719E-01	9.8695E-03	3.8706E-01	1.2014E-05
25.	NA	3.6095E-01	1.5743E-02	3.7669E-01	1.1693E-05
26.	LC	3.6932E-01	2.6857E-03	3.7201E-01	1.1547E-05
27.	NB	3.4964E-01	1.3632E-02	3.6327E-01	1.1276E-05
28.	KA	3.5961E-01	1.3377E-03	3.6095E-01	1.1204E-05
29.	KB	3.4806E-01	1.5798E-03	3.4964E-01	1.0853E-05
30.	KR	2.7829E-01	3.0835E-02	3.0912E-01	9.5952E-06
31.	IA	3.0777E-01	8.5433E-04	3.0863E-01	9.5800E-06
32.	CV	1.2448E-01	1.6905E-01	2.9354E-01	9.1115E-06
33.	R1	2.6892E-01	1.3650E-02	2.8257E-01	8.7709E-06
34.	OG	2.6279E-01	1.4826E-02	2.7762E-01	8.6174E-06
35.	FP	1.8109E-01	5.3095E-05	1.8115E-01	5.6229E-06
36.	I1	7.4834E-04	1.7667E-01	1.7742E-01	5.5071E-06
37.	G1	2.7599E-02	1.4929E-01	1.7689E-01	5.4908E-06
38.	CF	9.6689E-03	1.5014E-01	1.5981E-01	4.9605E-06
39.	E2	1.3366E-01	1.1444E-02	1.4511E-01	4.5042E-06
40.	E1	1.3355E-01	6.8872E-03	1.4044E-01	4.3593E-06
41.	CI	1.1908E-01	1.4653E-02	1.3373E-01	4.1510E-06
42.	D2	1.3132E-01	2.2158E-03	1.3353E-01	4.1449E-06
43.	D1	1.3134E-01	2.0550E-03	1.3340E-01	4.1407E-06
44.	U1	2.7106E-02	9.1416E-02	1.1852E-01	3.6790E-06
45.	DA	0.0000E+00	1.1736E-01	1.1736E-01	3.6430E-06
46.	DB	0.0000E+00	1.1736E-01	1.1736E-01	3.6428E-06
47.	OO	0.0000E+00	1.0452E-01	1.0452E-01	3.2442E-06
48.	SV	7.3957E-02	2.3418E-02	9.7375E-02	3.0226E-06
49.	I2	6.9116E-04	5.7678E-02	5.8370E-02	1.8118E-06
50.	G2	8.1853E-04	5.7071E-02	5.7890E-02	1.7969E-06

Table 3.4.2-2
Top Event Importance (Top 50)
Sorted by PROBABILISTIC Importance

Rank..	Top....	Probabilistic..	Guar. Event....	Total..........	Frequency......
1.	A2	2.5886E-01	2.0087E-01	4.5973E-01	1.4270E-05
2.	A1	2.3437E-01	1.9452E-01	4.2889E-01	1.3313E-05
3.	I1	1.7667E-01	7.4834E-04	1.7742E-01	5.5071E-06
4.	LA	1.7067E-01	4.0158E-01	5.7225E-01	1.7763E-05
5.	CV	1.6905E-01	1.2448E-01	2.9354E-01	9.1115E-06
6.	HS	1.5311E-01	4.6087E-01	6.1399E-01	1.9058E-05
7.	CF	1.5014E-01	9.6689E-03	1.5981E-01	4.9605E-06
8.	G1	1.4929E-01	2.7599E-02	1.7689E-01	5.4908E-06
9.	LB	1.4446E-01	4.7176E-01	6.1622E-01	1.9128E-05
10.	SA	1.3630E-01	5.3039E-01	6.6669E-01	2.0694E-05
11.	IC	1.2134E-01	4.7156E-01	5.9289E-01	1.8404E-05
12.	SB	1.1865E-01	5.4477E-01	6.6342E-01	2.0593E-05
13.	DA	1.1736E-01	0.0000E+00	1.1736E-01	3.6430E-06
14.	DB	1.1736E-01	0.0000E+00	1.1736E-01	3.6428E-06
15.	OO	1.0452E-01	0.0000E+00	1.0452E-01	3.2442E-06
16.	U1	9.1416E-02	2.7106E-02	1.1852E-01	3.6790E-06
17.	MB	6.7903E-02	4.7495E-01	5.4286E-01	1.6850E-05
18.	I2	5.7678E-02	6.9116E-04	5.8370E-02	1.8118E-06
19.	G2	5.7071E-02	8.1853E-04	5.7890E-02	1.7969E-06
20.	U2	3.4027E-02	0.0000E+00	3.4027E-02	1.0562E-06
21.	QH	3.3007E-02	0.0000E+00	3.3007E-02	1.0246E-06
22.	MA	3.1917E-02	4.4185E-01	4.7377E-01	1.4706E-05
23.	KR	3.0835E-02	2.7829E-01	3.0912E-01	9.5952E-06
24.	S1	3.0113E-02	0.0000E+00	3.0113E-02	9.3473E-07
25.	SV	2.3418E-02	7.3957E-02	9.7375E-02	3.0226E-06
26.	I3	2.1895E-02	6.2582E-04	2.2521E-02	6.9905E-07
27.	G3	2.1611E-02	7.1068E-04	2.2322E-02	6.9289E-07
28.	N2	2.0344E-02	2.2104E-02	4.2448E-02	1.3176E-06
29.	ME	1.7995E-02	0.0000E+00	1.7995E-02	5.5858E-07
30.	NA	1.5743E-02	3.6095E-01	3.7669E-01	1.1693E-05
31.	OG	1.4826E-02	2.6279E-01	2.7762E-01	8.6174E-06
32.	CI	1.4653E-02	1.1908E-01	1.3373E-01	4.1510E-06
33.	R1	1.3650E-02	2.6892E-01	2.8257E-01	8.7709E-06
34.	NB	1.3632E-02	3.4964E-01	3.6327E-01	1.1276E-05
35.	N1	1.3255E-02	0.0000E+00	1.3255E-02	4.1144E-07
36.	U3	1.2532E-02	0.0000E+00	1.2532E-02	3.8901E-07
37.	SL	1.2115E-02	6.3161E-04	1.2747E-02	3.9566E-07
38.	FW	1.2064E-02	7.2844E-01	7.4050E-01	2.2985E-05
39.	I4	1.1840E-02	4.3752E-04	1.2277E-02	3.8109E-07
40.	G4	1.1757E-02	4.9511E-04	1.2252E-02	3.8030E-07
41.	CH	1.1545E-02	8.2810E-03	1.9826E-02	6.1540E-07
42.	E2	1.1444E-02	1.3366E-01	1.4511E-01	4.5042E-06
43.	IB	9.8695E-03	3.7719E-01	3.8706E-01	1.2014E-05
44.	X3	9.0715E-03	0.0000E+00	9.0715E-03	2.8158E-07
45.	E1	6.8872E-03	1.3355E-01	1.4044E-01	4.3593E-06
46.	OE	6.8766E-03	0.0000E+00	6.8766E-03	2.1345E-07
47.	LS	6.2937E-03	4.0043E-01	4.0672E-01	1.2625E-05
48.	S2	5.8430E-03	0.0000E+00	5.8430E-03	1.8137E-07
49.	OA	4.6433E-03	0.0000E+00	4.6433E-03	1.4413E-07
50.	X2	4.1882E-03	0.0000E+00	4.1882E-03	1.3000E-07

Table 3.4.2-3
Split Fractions
Ranked by Importance and Risk Reduction Worth

SF Name...	Importance.....	Achievement..	SF Value.......	Frequency......
A12	1.8108E-01	4.1668E+00	5.2668E-02	5.6207E-06
I11	1.7667E-01	1.4498E+00	2.8200E-01	5.4839E-06
LA1	1.7034E-01	1.2748E+01	1.4016E-02	5.2874E-06
A28	1.4774E-01	3.1098E+00	6.4910E-02	4.5858E-06
G11	1.4019E-01	1.0134E+00	9.1000E-01	4.3515E-06
CV2	1.2333E-01	4.2627E+00	3.6422E-02	3.8281E-06
LBA	1.2228E-01	4.4759E+00	3.3960E-02	3.7957E-06
IC1	1.2091E-01	1.5988E+00	1.6209E-01	3.7530E-06
DA1	1.1736E-01	1.7834E+02	6.5810E-04	3.6430E-06
DB1	1.1689E-01	1.7774E+02	6.5810E-04	3.6283E-06
CF4	1.1376E-01	1.1020E+00	5.2734E-01	3.5313E-06
U11	9.1156E-02	2.0550E+00	7.5686E-02	2.8295E-06
OD1	9.0878E-02	9.1390E+01	1.0000E-03	2.8209E-06
HS2	7.3690E-02	1.4075E+00	1.4322E-01	2.2874E-06
MBA	6.6813E-02	1.4490E+00	1.0018E-01	2.0739E-06
I21	5.7678E-02	1.0813E+00	4.1500E-01	1.7904E-06
A11	5.3288E-02	5.9229E+02	9.0039E-05	1.6541E-06
G21	5.3096E-02	1.0163E+00	7.6000E-01	1.6481E-06
SBL	4.4162E-02	1.0000E+00	9.9990E-01	1.3708E-06
SAG	4.4162E-02	2.1699E+01	2.1290E-03	1.3708E-06
HS1	4.3918E-02	2.4517E+00	2.8007E-02	1.3632E-06
SAH	4.2473E-02	1.4755E+02	2.8970E-04	1.3184E-06
SBK	4.2191E-02	1.0003E+00	9.8710E-01	1.3096E-06
CV1	4.0288E-02	2.3772E+00	2.8422E-02	1.2505E-06
A21	3.7145E-02	4.1327E+02	9.0030E-05	1.1530E-06
U21	3.4027E-02	1.1360E+00	1.7873E-01	1.0562E-06
OM1	3.3007E-02	0.0000E+00	4.3000E-06	1.0246E-06
A24	3.1403E-02	1.5226E+00	5.1990E-02	9.7475E-07
MAA	3.0623E-02	1.0786E+00	1.0018E-01	9.5056E-07
S11	3.0113E-02	1.0287E+00	4.8045E-01	9.3473E-07
HS4	2.9787E-02	1.1604E+00	1.4848E-01	9.2459E-07
I31	2.1895E-02	1.0664E+00	2.4800E-01	6.7963E-07
KR1	2.1152E-02	1.9187E+00	2.1363E-02	6.5655E-07
CF1	2.0638E-02	1.2191E+00	8.6076E-02	6.4062E-07
G31	1.9635E-02	1.0293E+00	4.0000E-01	6.0947E-07
N23	1.9467E-02	2.0761E+00	1.3519E-02	6.0425E-07
SA7	1.8686E-02	8.9614E+01	2.1070E-04	5.8003E-07
SBT	1.8686E-02	1.0002E+00	9.8990E-01	5.8003E-07
SBO	1.8230E-02	1.0000E+00	1.0000E+00	5.6585E-07
SAC	1.8230E-02	8.7482E+01	2.1070E-04	5.6585E-07
NA1	1.5743E-02	4.7931E+00	3.5519E-03	4.8866E-07
CF2	1.5736E-02	1.0148E+00	5.1452E-01	4.8844E-07
LB1	1.4853E-02	1.9148E+00	1.3730E-02	4.6104E-07
OG1	1.4826E-02	8.5962E+01	1.7300E-04	4.6020E-07
CI1	1.4312E-02	1.2185E+00	5.6470E-02	4.4425E-07
R11	1.3650E-02	1.6068E+00	2.2000E-02	4.2369E-07
NB1	1.3632E-02	4.2089E+00	3.5321E-03	4.2313E-07
N11	1.3255E-02	1.9913E+00	6.6567E-03	4.1144E-07
A25	1.3234E-02	1.4827E+02	8.9750E-05	4.1079E-07
OD2	1.3038E-02	5.3325E+00	3.0000E-03	4.0469E-07
A29	1.2767E-02	1.2275E+00	5.2670E-02	3.9628E-07
U31	1.2532E-02	1.0036E+00	5.2635E-01	3.8901E-07
A23	1.2470E-02	1.2184E+00	5.2670E-02	3.8708E-07
SL1	1.2115E-02	1.3951E+00	2.8355E-02	3.7606E-07
FW3	1.1967E-02	1.0120E+00	5.0018E-01	3.7146E-07
I41	1.1840E-02	1.0014E+00	8.9700E-01	3.6751E-07
CH1	1.1545E-02	1.2392E+00	4.6000E-02	3.5836E-07
SV5	1.0854E-02	2.9609E+00	5.3309E-03	3.3693E-07
G41	1.0573E-02	1.0067E+00	6.1000E-01	3.2818E-07
E2A	1.0492E-02	1.4643E+00	1.7380E-02	3.2568E-07

Table 3.4.2-4
Split Fractions (Top 50)
Ranked by Risk Achievement Worth

Rank..	SF Name...	Importance.....	Achievement..	SF Value.......	Frequency......
1.	A11	5.3288E-02	5.9229E+02	9.0039E-05	1.6541E-06
2.	A21	3.7145E-02	4.1327E+02	9.0030E-05	1.1530E-06
3.	DA1	1.1736E-01	1.7834E+02	6.5810E-04	3.6430E-06
4.	DB1	1.1689E-01	1.7774E+02	6.5810E-04	3.6283E-06
5.	A25	1.3234E-02	1.4827E+02	8.9750E-05	4.1079E-07
6.	SAH	4.2473E-02	1.4755E+02	2.8970E-04	1.3184E-06
7.	OO1	9.0878E-02	9.1390E+01	1.0000E-03	2.8209E-06
8.	SA7	1.8686E-02	8.9614E+01	2.1070E-04	5.8003E-07
9.	SAC	1.8230E-02	8.7482E+01	2.1070E-04	5.6585E-07
10.	OG1	1.4826E-02	8.5962E+01	1.7300E-04	4.6020E-07
11.	A22	3.7640E-03	4.2806E+01	9.0020E-05	1.1684E-07
12.	SAG	4.4162E-02	2.1699E+01	2.1290E-03	1.3708E-06
13.	KA1	1.3377E-03	1.6942E+01	8.0670E-05	4.1522E-08
14.	LA1	1.7034E-01	1.2748E+01	1.4016E-02	5.2874E-06
15.	HA1	1.5599E-03	1.0573E+01	1.6058E-04	4.8419E-08
16.	MA1	1.2932E-03	7.9988E+00	1.7643E-04	4.0142E-08
17.	MB1	1.0901E-03	6.8714E+00	1.7643E-04	3.3837E-08
18.	OO2	1.3038E-02	5.3325E+00	3.0000E-03	4.0469E-07
19.	HBB	6.5666E-04	5.0708E+00	1.6060E-04	2.0383E-08
20.	NA1	1.5743E-02	4.7931E+00	3.5519E-03	4.8866E-07
21.	E11	6.8481E-03	4.7039E+00	1.5025E-03	2.1257E-07
22.	LBA	1.2228E-01	4.4759E+00	3.3960E-02	3.7957E-06
23.	CV2	1.2333E-01	4.2627E+00	3.6422E-02	3.8281E-06
24.	NB1	1.3632E-02	4.2089E+00	3.5321E-03	4.2313E-07
25.	A12	1.8108E-01	4.1668E+00	5.2668E-02	5.6207E-06
26.	RW1	1.9950E-03	4.0990E+00	6.1812E-04	6.1924E-08
27.	AS1	8.9898E-04	3.9058E+00	2.9810E-04	2.7905E-08
28.	RT1	2.8842E-04	3.8572E+00	9.9782E-05	8.9527E-09
29.	SBU	5.0190E-03	3.6791E+00	1.8600E-03	1.5579E-07
30.	SA6	5.0017E-03	3.6477E+00	1.8670E-03	1.5525E-07
31.	A26	2.0383E-04	3.2590E+00	9.0130E-05	6.3270E-09
32.	SBI	4.8523E-04	3.2521E+00	2.1400E-04	1.5062E-08
33.	A28	1.4774E-01	3.1098E+00	6.4910E-02	4.5858E-06
34.	N21	8.7711E-04	3.0621E+00	3.2822E-04	2.7226E-08
35.	SV5	1.0854E-02	2.9609E+00	5.3309E-03	3.3693E-07
36.	TW1	1.2996E-04	2.8116E+00	6.6828E-05	4.0340E-09
37.	SAB	4.7529E-04	2.6273E+00	2.8970E-04	1.4753E-08
38.	SV7	8.9407E-03	2.5349E+00	5.5709E-03	2.7752E-07
39.	HS1	4.3918E-02	2.4517E+00	2.8007E-02	1.3632E-06
40.	HSA	4.6233E-03	2.4326E+00	3.2000E-03	1.4351E-07
41.	CV1	4.0288E-02	2.3772E+00	2.8422E-02	1.2505E-06
42.	N23	1.9467E-02	2.0761E+00	1.3519E-02	6.0425E-07
43.	U11	9.1156E-02	2.0550E+00	7.5686E-02	2.8295E-06
44.	SV4	9.3733E-05	2.0350E+00	8.9774E-05	2.9095E-09
45.	N11	1.3255E-02	1.9913E+00	6.6567E-03	4.1144E-07
46.	D24	8.3538E-05	1.9575E+00	7.1880E-05	2.5930E-09
47.	KR1	2.1152E-02	1.9187E+00	2.1363E-02	6.5655E-07
48.	LB1	1.4853E-02	1.9148E+00	1.3730E-02	4.6104E-07
49.	D12	8.9772E-05	1.8354E+00	7.1877E-05	2.7866E-09
50.	R11	1.3650E-02	1.6068E+00	2.2000E-02	4.2369E-07

Table 3.4.2-5
Top Events and Split Fractions with Operator Actions

Top Event and Split Fractions	Equipment Unavailability Included
AI: ATWS ADS inhibit • AI1	 NO
CF: Continued injection after cont fails • All CF split fractions	 YES
CH: ATWS control of low pressure injection • CH1, control low pressure injection • CH2, restart HPCS	 NO NO
CI: Continued injection at MPCWLL • CI1, align HPCS to suppression pool	 YES
CN: Condenser • CN1, mode switch in SHUTDOWN • CN2, CN1 plus open MSIV path given MSIV IE • CN3, ATWS open MSIV path given MSIV IE	 YES YES YES
CV: Containment venting • CV1, all support available • CV2, loss of air • CV5, loss of nitrogen	 YES YES YES
FW: Feedwater • FW2, feedwater given SLOCA • FW3, ATWS restore feedwater	 YES YES
HA/HB: Locally open RHR Heat Exchanger MOV • HA1, HB1 & HBA	 YES
IC: RCIC • ICF, disable high temp trip for loss of SW	 YES
IL: ATWS RCIC low pressure trip • IL1	 NO
KR: Recover partial loss of offsite AC • KR1, loss of 115Kv source A • KR2, loss of 115Kv source B	 YES YES
MA/MB: Aux. Bay pump room cooling (open door) • MA1, MB1 & MBA	 YES
ME: Manual ECCS actuation • ME1	 NO
MO: ATWS prevent MSIV low level trip • MO1	 NO
MS: ATWS mode switch in SHUTDOWN • MS1	 NO
N2: Nitrogen, valve in high press bottles • N21, N22 & N23	 YES

Table 3.4.2-5
Top Events and Split Fractions with Operator Actions

Top Event and Split Fractions	Equipment Unavailability Included
OA: Blackout shed DC loads & prevent RCIC trip • OA1	 NO
OD: RPV depressurization • OD1, transient & SLOCA • OD2, MLOCA	 NO NO
OE: ATWS emergency Depressurization • OES, RCIC Success • OE1 through 8, RCIC Failure	 NO YES
OV: Mitigate vapor suppression failure • OV1, SLOCA • OV2, MLOCA	 NO NO
OH: Align containment heat removal • OH1, non-ATWS • OH2, ATWS	 NO NO
O1/O2/O3: Blackout emergency depressurize • All split fractions	 YES
S1/S2/S3: Blackout align diesel fire water • All split fractions	 YES
SA/SB: Service water, start pump & control • All split fractions	 YES
SW: Align Service water for LPI • SW1	 YES
U2: Blackout, RCIC (2 to 8 hrs) • U21	 YES

Table 3.4.2-6
Importance of Split Fractions with Human Actions

SF Name...	Importance.....	Achievement..	SF Value.......	Frequency......
CV2	1.2333E-01	4.2627E+00	3.6422E-02	3.8281E-06
CF4	1.1376E-01	1.1020E+00	5.2734E-01	3.5313E-06
OD1	9.0878E-02	9.1390E+01	1.0000E-03	2.8209E-06
MBA	6.6813E-02	1.4490E+00	1.0018E-01	2.0739E-06
SBL	4.4162E-02	1.0000E+00	9.9990E-01	1.3708E-06
SAG	4.4162E-02	2.1699E+01	2.1290E-03	1.3708E-06
SAH	4.2473E-02	1.4755E+02	2.8970E-04	1.3184E-06
SBK	4.2191E-02	1.0003E+00	9.8710E-01	1.3096E-06
CV1	4.0288E-02	2.3772E+00	2.8422E-02	1.2505E-06
U21	3.4027E-02	1.1360E+00	1.7873E-01	1.0562E-06
S11	3.0113E-02	1.0287E+00	4.8045E-01	9.3473E-07
KR1	2.1152E-02	1.9187E+00	2.1363E-02	6.5655E-07
CF1	2.0638E-02	1.2191E+00	8.6076E-02	6.4062E-07
N23	1.9467E-02	2.0761E+00	1.3519E-02	6.0425E-07
SA7	1.8686E-02	8.9614E+01	2.1070E-04	5.8003E-07
SBT	1.8686E-02	1.0002E+00	9.8990E-01	5.8003E-07
SBO	1.8230E-02	1.0000E+00	1.0000E+00	5.6585E-07
SAC	1.8230E-02	8.7482E+01	2.1070E-04	5.6585E-07
CF2	1.5736E-02	1.0148E+00	5.1452E-01	4.8844E-07
CI1	1.4312E-02	1.2185E+00	5.6470E-02	4.4425E-07
OD2	1.3038E-02	5.3325E+00	3.0000E-03	4.0469E-07
FW3	1.1967E-02	1.0120E+00	5.0018E-01	3.7146E-07
CH1	1.1545E-02	1.2392E+00	4.6000E-02	3.5836E-07
KR2	9.6832E-03	1.2021E+00	3.4006E-02	3.0057E-07
ME1	8.2846E-03	9.9633E-01	3.2000E-02	2.5716E-07
MO1	6.9852E-03	1.0000E+00	1.0000E+00	2.1682E-07
CH2	6.8766E-03	1.0000E+00	1.0000E+00	2.1345E-07
OE1	6.8534E-03	1.0343E+00	1.6005E-01	2.1273E-07
SA1	5.9253E-03	0.0000E+00	3.0120E-06	1.8392E-07
SBZ	5.8517E-03	1.0138E+00	2.9660E-01	1.8164E-07
S21	5.8430E-03	1.0032E+00	3.7165E-01	1.8137E-07
SBU	5.0190E-03	3.6791E+00	1.8600E-03	1.5579E-07
SA6	5.0017E-03	3.6477E+00	1.8670E-03	1.5525E-07
OA1	4.6433E-03	1.2921E+00	9.9000E-03	1.4413E-07
CV5	4.2866E-03	1.1335E+00	3.1122E-02	1.3306E-07
CN3	3.0431E-03	1.0000E+00	1.0000E+00	9.4459E-08
S31	2.9907E-03	1.0004E+00	3.7200E-01	9.2831E-08
OH1	2.7768E-03	0.0000E+00	1.0000E-05	8.6194E-08
CN2	2.1223E-03	1.0187E+00	1.0168E-01	6.5878E-08
HA1	1.5599E-03	1.0573E+01	1.6058E-04	4.8419E-08
MA1	1.2932E-03	7.9988E+00	1.7643E-04	4.0142E-08
AI1	1.1788E-03	1.1988E+00	5.6100E-03	3.6591E-08
MB1	1.0901E-03	6.8714E+00	1.7643E-04	3.3837E-08
SA8	1.0426E-03	1.4858E+00	2.1290E-03	3.2363E-08
SBS	1.0426E-03	1.0000E+00	9.9860E-01	3.2363E-08
N21	8.7711E-04	3.0621E+00	3.2822E-04	2.7226E-08
SBX	7.9536E-04	1.0016E+00	3.0110E-01	2.4688E-08
CN1	7.8659E-04	1.4469E+00	1.6750E-03	2.4416E-08
HBA	6.4703E-04	1.0353E+00	1.8020E-02	2.0084E-08
SBI	4.8523E-04	3.2521E+00	2.1400E-04	1.5062E-08
SAB	4.7529E-04	2.6273E+00	2.8970E-04	1.4753E-08
SBP	4.7529E-04	1.0000E+00	1.0000E+00	1.4753E-08
SA5	2.1864E-04	0.0000E+00	1.9120E-05	6.7866E-09
SBV	2.1488E-04	1.0009E+00	1.9200E-01	6.6700E-09
OH2	1.9023E-04	9.9975E-01	9.6000E-03	5.9049E-09
SB5	1.5527E-04	0.0000E+00	1.5560E-05	4.8196E-09
O11	1.3968E-04	1.0105E+00	1.0010E-03	4.3356E-09
OV1	9.6203E-05	1.0023E+00	4.0000E-02	2.9862E-09
OV2	3.6076E-05	1.0009E+00	4.0000E-02	1.1198E-09
O12	2.3441E-05	9.9656E-01	1.0010E-03	7.2763E-10
OE2	2.3145E-05	1.0001E+00	1.6133E-01	7.1844E-10
SBR	2.3062E-05	1.0000E+00	9.9000E-01	7.1585E-10
SA9	2.3062E-05	1.1082E+00	2.1210E-04	7.1585E-10
SA3	2.1182E-05	0.0000E+00	3.0300E-06	6.5750E-10
SAD	1.9743E-05	1.0927E+00	2.1210E-04	6.1284E-10
SBN	1.9743E-05	1.0000E+00	1.0000E+00	6.1284E-10
SBY	1.7280E-05	1.0000E+00	3.0090E-01	5.3638E-10
SA2	1.7280E-05	0.0000E+00	3.0300E-06	5.3638E-10
O21	1.3664E-05	1.0011E+00	1.0010E-03	4.2414E-10
SW1	1.3459E-05	1.0003E+00	4.3685E-02	4.1777E-10
HB1	8.0464E-06	9.2653E-01	1.5770E-04	2.4976E-10
O31	6.7280E-06	9.9909E-01	1.0010E-03	2.0884E-10
FW2	6.0762E-06	1.0011E+00	5.6750E-03	1.8861E-10

Table 3.4.2-7
Contributors to Important Split Fractions

System/ Top Event	Split Fraction	Failure Description (CutSet)	% Contrib to SF
AC POWER	A11 (Div. 1 All Sup. Avail.)	Bus Failure Transformer Failure Circuit Breaker Opens Spur.	65 17 17
	A21 (Div. 2 All Sup. Avail.)	Similar to Above (Symmetrical Divisions)	
	A12 (Div. I Fail Dur. LOSP)	EDG Fails to Run EDG Fails to Start EDG HVAC Fails MOV Fails to Open on Demand Circ Break Fails to Close Circ Break Fails to Open Maintenance Unavailability	48 34 7 3 3 2 2
	A28 (Div. 2 Fail Dur. LOSP)	Similar to Above (Symmetrical Divisions)	
	A33 (Both Div. Fail Dur. LOSP)	Ind. Failure EDGs Start-Run (Mult. Combinations) Multiple independant failure combinations (i.e., 1 EDG & MOV), includes maintenance Common Cause Failure (CCF) of EDGs CCF of EDG HVAC CCF Service Water Supply MOVs CCF of Circuit Breakers to Close CCF of Circuit Breakers to Open	60 17 8 4 3 3 3
HPCS	HS1 (all Sup)	Maintenance Unavailability MOVs Fail on Demand (1 of 4) Pump Fails to Start Pump Fails to Run	50 22 17 4
	HS2 (LOSP)	EDG Fails to Run Maintenance Unavailability EDG Fails to Start Pump fails to Start MOVs Fail on Demand (1 of 4) Room Cooling Fails	60 12 12 3 5 1
RCIC	IC1 (All Sup)	Maintenance Unavailability Pump Fails to Start MOVs Fail on Demand (1 of 4) Pump Fails to Run	57 25 12 4
	U11 (SBO Phase 1)	Similar to above	

Table 3.4.2-7
Contributors to Important Split Fractions

System/ Top Event	Split Fraction	Failure Description (CutSet)	% Contrib to SF
RHR	LA1 (Div. I Fail)	Pump Fails to Start Pump Fails to Run Maintenance Unavailability Relay Fails-Demand (1 of 3) Check Valve Fails on Demand	49 34 8 5 3
	LBA (Div. 2 Fail)	Similar to Above (Symmetrical Divisions)	
	L31 (Both Div. Fail)	Common Cause Fail-Pump Start Failure of Pumps-Start & Run Common Cause Fail-Pump Run A pump Fail or Maintenance Start-B Pump Maintenance	69 7 7 3
Service Water	SCG (SAG/SBL)	RBCLC Supply MOV fail to Close RBCLC Supply MOV Relay Fails on Demand Operator Fails to Start Pump	74 9 9
Containment Vent	CV2	SOVs Fail on Demand (1 of 3) Operator Fails to Align AOVs Fail on Demand (1 of 2)	41 37 21

Table 3.4.2-7
Contributors to Important Split Fractions

Top...	Probabilistic..	Guar. Event....	Frequency......
CV	5.7589E-01	4.2411E-01	9.1107E-06
LA	5.7363E-01	8.8419E-02	6.0317E-06
CF	5.1144E-01	3.2942E-02	4.9597E-06
LB	4.8563E-01	3.4917E-01	7.6056E-06
CI	4.9923E-02	4.0570E-01	4.1510E-06
CN	9.7337E-03	9.5247E-01	8.7663E-06
OH	9.4607E-03	0.0000E+00	8.6194E-08
HA	5.2882E-03	9.7650E-01	8.9447E-06
IB	4.8840E-03	3.3431E-01	3.0903E-06
HB	4.4417E-03	9.8231E-01	8.9899E-06
IA	1.3987E-03	7.8730E-02	7.3002E-07
CB	8.9664E-04	9.9491E-01	9.0724E-06
PB	8.9664E-04	0.0000E+00	8.1690E-09
PA	3.6266E-05	4.6454E-03	4.2653E-08
CA	3.6266E-05	9.9577E-01	9.0724E-06

Important Support Systems

Top...	Probabilistic..	Guar. Event....	Frequency......
SA	2.8569E-01	1.1169E-01	3.6204E-06
SB	2.3557E-01	1.5570E-01	3.5647E-06
A2	1.5190E-01	1.5470E-01	2.7932E-06
N2	5.6343E-02	6.3771E-02	1.0943E-06
R1	4.6505E-02	9.1621E-01	8.7709E-06
NA	4.2305E-02	2.5523E-01	2.7108E-06
NB	3.4943E-02	2.7515E-01	2.8252E-06
N1	3.0232E-02	0.0000E+00	2.7544E-07
A1	2.7796E-02	3.4970E-02	5.7184E-07
RW	5.6417E-03	7.8594E-01	7.2118E-06
AS	2.5617E-03	8.2992E-01	7.5845E-06

Table 3.4.3-2
Important Decay Heat Removal Split Fractions

SF Name...	Importance.....	Achievement..	SF Value.......	Frequency......
LA1	5.7252E-01	4.0937E+01	1.4016E-02	5.2161E-06
CV2	4.2018E-01	1.2116E+01	3.6422E-02	3.8281E-06
LBA	4.1651E-01	1.2845E+01	3.3960E-02	3.7947E-06
CF4	3.8760E-01	1.3474E+00	5.2734E-01	3.5313E-06
CV1	1.3717E-01	5.6892E+00	2.8422E-02	1.2497E-06
SAH	1.2951E-01	4.4787E+02	2.8970E-04	1.1799E-06
SBK	1.2855E-01	1.0007E+00	9.8710E-01	1.1712E-06
CF1	7.0316E-02	1.7466E+00	8.6076E-02	6.4062E-07
A21	6.0324E-02	6.7043E+02	9.0030E-05	5.4959E-07
N23	5.3651E-02	4.7043E+00	1.3519E-02	4.8879E-07
CF2	5.3523E-02	1.0505E+00	5.1452E-01	4.8763E-07
SBT	5.0627E-02	1.0005E+00	9.8990E-01	4.6125E-07
SA7	5.0627E-02	2.4116E+02	2.1070E-04	4.6125E-07
SBO	5.0014E-02	1.0000E+00	1.0000E+00	4.5566E-07
SAC	5.0014E-02	2.3828E+02	2.1070E-04	4.5566E-07
CI1	4.8762E-02	1.7444E+00	5.6470E-02	4.4425E-07
R11	4.6505E-02	3.0674E+00	2.2000E-02	4.2369E-07
A24	4.5378E-02	1.7597E+00	5.1990E-02	4.1342E-07
LB1	4.5240E-02	4.1497E+00	1.3730E-02	4.1217E-07
NA1	4.2305E-02	1.2166E+01	3.5519E-03	3.8542E-07
A23	4.2121E-02	1.7399E+00	5.2670E-02	3.8375E-07
SAG	3.6144E-02	1.7939E+01	2.1290E-03	3.2930E-07
SBL	3.6144E-02	1.0000E+00	9.9990E-01	3.2930E-07
NB1	3.4943E-02	1.0168E+01	3.5321E-03	3.1835E-07
N11	3.0232E-02	4.5417E+00	6.6567E-03	2.7544E-07
LB3	2.2867E-02	2.5466E+00	1.4020E-02	2.0833E-07
SA1	1.6821E-02	0.0000E+00	3.0120E-06	1.5325E-07
A11	1.6784E-02	1.8658E+02	9.0039E-05	1.5292E-07
SBZ	1.6570E-02	1.0390E+00	2.9660E-01	1.5096E-07
CV5	1.4605E-02	1.4547E+00	3.1122E-02	1.3306E-07
A12	1.1011E-02	1.0806E+00	5.2668E-02	1.0032E-07
OH1	9.4607E-03	0.0000E+00	1.0000E-05	8.6194E-08
CN2	7.1994E-03	1.0636E+00	1.0168E-01	6.5591E-08
RW1	5.6417E-03	9.9131E+00	6.1812E-04	5.1400E-08
HA1	5.2882E-03	3.3912E+01	1.6058E-04	4.8179E-08
A28	4.0580E-03	1.0561E+00	6.4910E-02	3.6971E-08
IBA	4.0140E-03	9.9251E-01	5.7130E-02	3.6570E-08
CV3	3.9337E-03	1.0037E+00	5.1820E-01	3.5839E-08
N21	2.6924E-03	8.5351E+00	3.2822E-04	2.4529E-08
AS1	2.5617E-03	9.4232E+00	2.9810E-04	2.3338E-08
CN1	2.5343E-03	2.5098E+00	1.6750E-03	2.3089E-08
HBB	2.2373E-03	1.4927E+01	1.6060E-04	2.0383E-08
HBA	2.2045E-03	1.1201E+00	1.8020E-02	2.0084E-08
SAB	1.3998E-03	5.7870E+00	2.8970E-04	1.2753E-08
SBP	1.3998E-03	1.0000E+00	1.0000E+00	1.2753E-08
IA1	1.3987E-03	7.5881E-01	2.0694E-03	1.2743E-08
SBI	1.2584E-03	6.8297E+00	2.1400E-04	1.1465E-08
CI3	1.1610E-03	1.0025E+00	1.7350E-01	1.0578E-08
LA2	1.1076E-03	1.0791E+00	1.3806E-02	1.0091E-08
SBS	1.0891E-03	1.0000E+00	9.9860E-01	9.9220E-09
SA8	1.0891E-03	1.5054E+00	2.1290E-03	9.9220E-09
LBB	9.9809E-04	1.0283E+00	3.4030E-02	9.0932E-09
PB1	8.9664E-04	1.5797E+00	1.5230E-03	8.1690E-09
CBB	8.9664E-04	1.0187E+00	4.5860E-02	8.1690E-09
IB1	8.6999E-04	8.6126E-01	1.9550E-03	7.9262E-09
SBX	6.4418E-04	1.0006E+00	3.0110E-01	5.8689E-09
SB5	5.2901E-04	0.0000E+00	1.5560E-05	4.8196E-09
SBU	1.3589E-04	1.0707E+00	1.8600E-03	1.2380E-09
PA1	3.6266E-05	1.0128E+00	1.6372E-03	3.3041E-10
CA1	3.6266E-05	1.0074E+00	4.9064E-03	3.3041E-10
SA3	2.6264E-05	0.0000E+00	3.0300E-06	2.3929E-10
LBC	1.7976E-05	1.0013E+00	1.3810E-02	1.6378E-10
SBN	1.6184E-05	1.0000E+00	1.0000E+00	1.4745E-10
SAD	1.6184E-05	1.0755E+00	2.1210E-04	1.4745E-10
SBR	1.5715E-05	1.0000E+00	9.9000E-01	1.4317E-10
SA9	1.5715E-05	1.0728E+00	2.1210E-04	1.4317E-10
A25	1.5169E-05	1.1173E+00	8.9750E-05	1.3820E-10
SA5	1.2799E-05	0.0000E+00	1.9120E-05	1.1661E-10
SA2	1.1098E-05	0.0000E+00	3.0300E-06	1.0111E-10
SBY	1.1098E-05	1.0000E+00	3.0090E-01	1.0111E-10

Intentionally
Left
Blank

Intentionally
Left
Blank

INTENTIONALLY LEFT BLANK

Table 3.4.3-3 Level 1 Systems & Top Events

Section 3.2.1.x	System Name	Top Event Designation	Top Event Description
6	AC Power Systems	A1	Division I Emergency AC
6	AC Power Systems	A2	Division II Emergency AC
13	Automatic Depressurization	AI	ADS Inhibit (ATWS)
23	Instrument Air	AS	Instrument Air
16	Redundant Reactivity Control	C1	Division I RRCS
16	Redundant Reactivity Control	C2	Division II RRCS
4	Residual Heat Removal	CA	Containment Spray A
4	Residual Heat Removal	CB	Containment Spray B
24	Late Containment Failure	CF	Continued Injection after Containment Failure
16	Redundant Reactivity Control	CH	Level Control not High
24	Late Containment Failure	CI	Continued Injection at High Pressure
21	Condensate & Feedwater	CN	Condenser Available (heat sink)
17	Containment Venting	CV	Containment Venting
7	DC Power Systems	D1	Division I Emergency DC
7	DC Power Systems	D2	Division II Emergency DC
5	ECCS Actuation	E1	Division I ECCS Actuation
5	ECCS Actuation	E2	Division II ECCS Actuation
17	Containment Venting	FB	Drywell Venting (Level 2)
17	Containment Venting	FD	Drywell Venting (Level 2)
9	Fire & Service Water Crossties	FP	Fire Water Crosstie to RHR
16	Redundant Reactivity Control	FT	Feedwater Runback
21	Condensate & Feedwater	FW	Feedwater Available
26	Recovery	G1	Recovery of Emergency EDG (SBO w/o high pressure makeup, time = 0-30 min)
26	Recovery	G2	Recovery of Emergency EDG (SBO, time = 0-2 hrs)
26	Recovery	G3	Recovery of Emergency EDG (SBO, time = 2-8 hrs)
26	Recovery	G4	Recovery of Emergency EDG (SBO, time = 8-10 hrs)
26	Recovery	G5	Recovery of Emergency EDG (SBO, time = 10-19 hrs)
17	Containment Venting	GV	Gas Venting (Level 2)
4	Residual Heat Removal	HA	RHR A Heat Exchanger
4	Residual Heat Removal	HB	RHR B Heat Exchanger
1	High Pressure Core Spray	HS	HPCS
26	Recovery	I1	Recovery of Offsite AC (SBO w/o high pressure makeup, time = 0-30 min)
26	Recovery	I2	Recovery of Offsite AC (SBO, time = 0-2 hrs)
26	Recovery	I3	Recovery of Offsite AC (SBO, time = 2-8 hrs)

INTENTIONALLY LEFT BLANK

Table 3.4.3-3 Level 1 Systems & Top Events

Section 3.2.1.x	System Name	Top Event Designation	Top Event Description
26	Recovery	I4	Recovery of Offsite AC (SBO, time = 8-10 hrs)
26	Recovery	I5	Recovery of Offsite AC (SBO, time = 10-19 hrs)
4	Residual Heat Removal	IA	LPCI A Injection Train
4	Residual Heat Removal	IB	LPCI B Injection Train
2	Reactor Core Isolation Cooling	IC	RCIC
2	Reactor Core Isolation Cooling	IL	Operator Overrides ATWS Trips
10	Containment Isolation	IS	Containment Isolation (Level 2)
6	AC Power Systems	KA	115 kV Source A
6	AC Power Systems	KB	115 kV Source B
6	AC Power Systems	KR	Partial Recovery of KA and/or KB
4	Residual Heat Removal	LA	RHR A Pump Train
4	Residual Heat Removal	LB	RHR B Pump Train
4	Residual Heat Removal	LC	RHR C Pump Train & LPCI C Injection Train
3	Low Pressure Core Spray	LS	LPCS
11	Ventilation Systems	MA	North MCC Area Unit Coolers
11	Ventilation Systems	MB	South MCC Area Unit Coolers
5	ECCS Actuation	ME	Manual ECCS Actuation
16	Redundant Reactivity Control	MO	Operator Overrides Level 1
15	Reactor Protection System	MS	Mode Switch in Shutdown
22	Nitrogen	N1	High Pressure Nitrogen
22	Nitrogen	N2	Instrument Nitrogen
6	AC Power Systems	NA	Normal AC & DC Source A
6	AC Power Systems	NB	Normal AC & DC Source B
XX	Switches	NE	No Catastrophic (ATWS)
XX	Switches	NL	Success (Late Trees)
XX	Switches	NM	No Injection (Late Trees)
13	Automatic Depressurization	O1	Operator Depressurizes RPV (SBO, time = 0-2 hrs)
13	Automatic Depressurization	O2	Operator Depressurizes RPV (SBO, time = 2-8 hrs)
13	Automatic Depressurization	O3	Operator Depressurizes RPV (SBO, time = 8-10 hrs)
2	Reactor Core Isolation Cooling	OA	Operator sheds DC (SBO)
13	Automatic Depressurization	OD	Operator Depressurizes RPV
13	Automatic Depressurization	OE	Op. Emergency Depressurizes (ATWS)
6	AC Power Systems	OG	Offsite Power
4	Residual Heat Removal	OH	Operator aligns RHR Cooling
18	Vapor Suppression	OV	Operator Sprays or Vents

INTENTIONALLY LEFT BLANK

Table 3.4.3-3 Level 1 Systems & Top Events

Section 3.2.1.x	System Name	Top Event Designation	Top Event Description
4	Residual Heat Removal	PA	Suppression Pool Cooling A
4	Residual Heat Removal	PB	Suppression Pool Cooling B
15	Reactor Protection System	QE	Reactor Scram Electrical Equipment
15	Reactor Protection System	QM	Reactor Scram Mechanical Equipment
26	Recovery	R1	AC Power Recovery (Transient Event Tree TR2)
XX	Switches	RH	RHR (ATWS)
16	Redundant Reactivity Control	RI	Alternate Rod Insertion
15	Reactor Protection System	RQ	Reactor Scram
16	Redundant Reactivity Control	RT	Reactor Recirculation Pump Trip
19	Reactor Building Closed Loop Cooling Water	RW	RBCLC
9	Fire & Service Water Crossties	S1	Fire Water to RHR (SBO, time = 2-8 hrs)
9	Fire & Service Water Crossties	S2	Fire Water to RHR (SBO, time = 8-10 hrs)
9	Fire & Service Water Crossties	S3	Fire Water to RHR (SBO, time = 10-19 hrs)
8	Service Water	SA	Service Water Loop A
8	Service Water	SB	Service Water Loop B
12	Standby Liquid Control	SL	SLCS
13	Automatic Depressurization	SO	Stuck Open Relief Valve (ATWS)
13	Automatic Depressurization	SR	Adequate Relief (ATWS)
13	Automatic Depressurization	SV	SRV/ADS Valves
9	Fire & Service Water Crossties	SW	Service Water Crosstie to RHR
21	Condensate & Feedwater	TA	Condensate Storage Tank A
21	Condensate & Feedwater	TB	Condensate Storage Tank B
20	Turbine Building Closed Loop Cooling Water	TW	TBCLC
2	Reactor Core Isolation Cooling	U1	RCIC (SBO, time = 0-2 hrs)
2	Reactor Core Isolation Cooling	U2	RCIC (SBO, time = 2-8 hrs)
2	Reactor Core Isolation Cooling	U3	RCIC (SBO, time = 8-10 hrs)
6	AC Power Systems	UA	Vital UPS Source A
6	AC Power Systems	UB	Vital UPS Source B
17	Containment Venting	VC	Containment Venting (Level 2)
18	Vapor Suppression	VS	Vapor Suppression
16	Redundant Reactivity Control	WL	RPV Level > ⅓ core
13	Automatic Depressurization	X1	SRVs Remain Open (SBO, time = 2-8 hrs)
13	Automatic Depressurization	X2	SRVs Remain Open (SBO, time = 8-10 hrs)
13	Automatic Depressurization	X3	SRVs Remain Open (SBO, time = 10-19 hrs)

4.0 Back-end Analysis

The purpose of this section is to describe the approach for the performance of the containment analysis for the Nine Mile Point 2 Individual Plant Examination. This section outlines a complete Level 2 PRA methodology which satisfies the request made by the NRC in the IPE Generic Letter 88-20 and its companion guidance document, NUREG-1335. This section also develops a framework within which future questions regarding Nine Mile Point 2 containment performance can be addressed.

The Nine Mile Point 2 IPE evaluation includes consideration of severe accident behavior recognizing the Nine Mile Point 2 containment capability and incorporating the role of Nine Mile Point 2 mitigating systems in responding to an accident. This information can furnish input for the future development of accident management procedures and/or the revision of a current emergency operating procedures.

4.0.1 Historical Background

In April of 1983, the NRC published a "Proposed Commission Policy Statement on Severe Accidents and Related Views on Nuclear Reactor Regulation." A major focus of the NRC issues arising out of the TMI-2 accident [Ref. 48] and the proposed Severe Accident Policy Statement related to containment performance under postulated severe accident conditions.

The NRC Severe Accident Policy Statement [Ref. 51] states that on the basis of current information existing plants pose no undue risk to the health and safety of the public. Therefore, the NRC sees no justification to take immediate action on generic rulemaking or other regulatory changes for existing plants because of issues related to severe accidents. Justification for this position is paraphrased below:

- Operating nuclear power plants require no further regulatory action to deal with severe accident issues unless significant new safety information arises to question whether there is adequate assurance of no undue risk to public health and safety.

- Recognizing that plant specific PRAs have yielded valuable insights regarding unique plant vulnerabilities to severe accidents leading to low cost modifications, licensees of each operating reactor will be expected to perform a limited scope, accident safety analysis designed to discover instances (i.e., outliers) of particular vulnerability to core melt or to unusually poor containment performance, given core-melt accidents. These plant specific studies will serve to verify that conclusions developed from intensive severe accident safety analyses of reference or surrogate plants can be applied to each of the individual operating plants.

The policy continues to state that if new information becomes available that changes the Commission's perception of "no undue risk", then the underlying technical issues would be evaluated to determine if the issue is plant specific or generic. Options for reducing the vulnerability would be identified and any decision regarding implementation would be consistent with the cost effectiveness criteria of the NRC backfit policy. If the issue is generic in nature and goes beyond the scope of current regulatory requirements, then a generic rulemaking could be initiated.

The Severe Accident Policy is a guide to regulatory decision making (as opposed to a rulemaking, which is the process of issuing, deleting, or changing a regulation). The policy provides general procedures for staff approval of items related to severe accidents. It is important to differentiate the issuance of this policy from a generic rulemaking. A rulemaking on the severe accident issues presumably would have required existing plants to add additional capability to prevent, mitigate, or manage severe accidents. The NRC does not feel that this is necessary, due in part to the improvements in safety that have resulted from the post-TMI items and the published results of nuclear power plant PRAs. The NRC also believes that if new safety issues are identified, they will likely be unique to a specific plant, and therefore, may not warrant major generic design changes.

Even though the NRC has stated that it believes existing plants pose no undue risk to the health and safety of the public, the policy statement indicates that all plants will be required to perform a limited scope safety analysis to verify that this conclusion is true and to identify any potential outliers that might be plant specific.

Implicit in these statements is a commitment to "defense-in-depth" that seeks to establish reliable barriers to prevent fission product escape. The first of these barriers is to prevent any core damage from occurring. This aspect of the prevention of severe accidents has been addressed through Level 1 Probabilistic Risk Assessments (PRA) [Ref. 52], and may continue to be addressed in this manner or through Individual Plant Evaluations [Ref. 53].

A second aspect of the defense-in-depth is in the integrity of the containment in controlling and mitigating the release of radionuclides. A widely accepted, detailed analysis framework for this intermediate defense barrier has not been established to date. This is the subject of this section of the report.

A third "barrier" to public risk is the dispersion of radionuclides coupled with plant location and population evacuation to mitigate the impact of radionuclide releases on the population. Thus far, this aspect of risk analysis has not been considered for implementation as part of the Severe Accident Policy Statement.

4.0.2 IPE Overview

In the Commission policy statement on severe accidents in nuclear power plants issued on August 8, 1985 (50 FR 32138), the Commission concluded, based on available information, that existing plants pose no undue risk to the public health and safety and that there is no present basis for immediate action on generic rulemaking or other regulatory requirements for these plants. However, the Commission recognizes, based on NRC and industry experience with plant-specific probabilistic risk assessments (PRAs), that systematic examinations are beneficial in identifying plant-specific vulnerabilities to severe accidents that could be fixed with low cost improvements. Therefore, each existing plant should perform a systematic examination to identify any plant-specific vulnerabilities to severe accidents and report the results to the Commission.

The general purpose of this examination, defined as an Individual Plant Examination (IPE), is for each utility:

- To develop an appreciation of severe accident behavior,
- To understand the most likely severe accident sequences that could occur at its plant,
- To gain a more quantitative understanding of the overall probabilities of core damage and fission product releases, and
- If necessary, to reduce the overall probabilities of core damage and fission product releases by modifying, where appropriate, hardware and procedures that would help prevent or mitigate severe accidents.

It is expected that the achievement of these goals will help verify that at U.S. nuclear power plants severe core damage and large radioactive release probabilities are consistent with the Commission's Safety Goal Policy Statement. Besides the Individual Plant Examinations, closure of severe accident concerns will involve future NRC and industry efforts in the areas of accident management and generic containment performance improvements.

The NRC expects the utility's staff participating in the IPE to:

- Examine and understand the plant emergency procedures, design, operations, maintenance, and surveillance to identify potential severe accident sequences for the plant;
- Understand the quantification of the expected sequence frequencies;
- Determine the leading contributors to core damage and unusually poor containment performance, and determine and develop an understanding for their underlying causes;
- Identify any proposed plant improvements for the prevention and mitigation of severe accidents;
- Examine each of the proposed improvements, including design changes as well as changes in maintenance, operating and emergency procedures, surveillance, staffing, and training programs; and
- Identify which proposed improvements will be implemented and their schedule.

Next, the NRC has formulated a plan for the "closure" of severe accident issues involving six main elements [Ref. 68, 70]:

- Examination of existing plants for severe accident vulnerabilities (IPE)
- Development of Containment Performance Improvements [Ref. 69, 75, 76]
- Improvement of plant operations

- Development of a severe accident research program at the NRC
- Implementation of IPEEE
- Development and implementation of a severe accident management program for each licensee

The NRC staff has also provided the BWR utilities a set of Accident Management Strategies which could be voluntarily considered in the IPE process [Ref. 75].

The NRC staff has provided recommendations to the Commission on how to assess the IPE with respect to the Commission's safety goal [Ref. 71] in the following ways:

- No direct risk comparisons of individual plant IPE results with the Safety Goal Policy are currently planned. This is consistent with the primary purpose of the IPE process which is to identify and eliminate plant-specific vulnerabilities, rather than to develop plant-specific risk profiles.
- However, as stated in SECY-88-147, there will be comparisons of IPE results in the aggregate with the safety goal subsidiary objectives, proposed in SECY-89-102 in order to provide information by which the adequacy of the NRC's regulations can be judged.
- As described in SECY-88-147, indirect comparison of the IPEs and other available PRAs with the Safety Goals, focusing on the insights gained and the adequacy of regulations, is planned. However, this comparison will be highly dependent upon the content of the IPE submittals and, therefore, development of specific guidance in this area is not recommended at this time. Rather, it is recommended that the staff evaluate the IPE results as a whole and summarize any conclusions and recommendations for the Commission at the completion of the IPE review process.

As part of the review process, the NRC staff has developed for Commission consideration a definition for "large release" [Ref. 72] which is a term used in the safety goal definition and in the IPE Generic Letter. This definition is discussed in this report when deriving the radionuclide release categories in Sections 4.7.

4.0.3 Application of the IPE Requirements by Niagara Mohawk

This section documents the Niagara Mohawk Power Corporation effort to establish a thorough, traceable, and technically sound methodology for examining the Nine Mile Point 2 containment capability under severe accident conditions in response to Generic Letter 88-20. It has involved the development of a detailed set of containment event trees as a framework for examining severe accident phenomena including both active and passive mitigation functions of the Nine Mile Point 2 Mark II containment. This effort is based upon previous methods used in the Shoreham PRA [Ref. 59], the Limerick IPE [Ref. 57], the Peach Bottom Containment Evaluation [Ref. 64], and the Vermont Yankee Containment Safety Study [Ref. 63].

Figure 4.0-1 provides a simple flow chart that describes the relationship between the Level 1

PRA and the Level 2 containment evaluation. The interface between the two evaluations requires the transfer of information describing the key aspects of the postulated severe accident scenarios.

The principal technical advances that have been incorporated into the Nine Mile Point 2 containment evaluation effort include the following:

- Providing a containment event tree that includes sufficient detail to quantify the effects of plant modifications and changes in procedures.

- Establishing added success paths for recovery of degraded core conditions within the reactor vessel (e.g., TMI-2 events), including those that involve recovery actions during in-vessel core melt progression accidents.

- Incorporating the latest emergency procedures at Nine Mile Point 2. This includes containment flooding which is a major model perturbation from previous studies.

- Interfacing with the BWROG/NUMARC containment safety study to incorporate the latest input on severe accident phenomenological issues as they affect containment response (e.g., direct containment heating, heat management, seal performance).

- Establishing plant specific deterministic calculations to support the improved success criteria using MAAP calculations as the basis.

- Providing a traceable documentation path through the containment event tree so that both qualitative and quantitative insights can be developed. This facilitates both communication with the NRC and internal use within NMPC.

- Incorporating responses to issues raised by the NRC contractors in NUREG-1150 in a more visible manner.

4.0.4 Objectives of the Level 2 Analysis

Individual Plant Examinations (IPEs), as directed by NRC Generic Letter 88-20, are to be performed to identify whether there are plant specific vulnerabilities that require hardware or procedural modifications. These IPEs are to be performed recognizing that there are large uncertainties in the current state of knowledge regarding generic plant response to certain phenomena. In other words, it is recognized that the IPEs are being performed despite the lack of perfect knowledge regarding the details of core melt progression models and postulated rare physical phenomena. These generic uncertainties were recognized at the time the NRC Severe Accident Policy Statement was written; specifically, the Commission concluded that, despite the uncertainties, operating plants were safe and no generic safety modifications were necessary. Therefore, the IPEs are to determine if plant specific features would alter this generic conclusion, i.e. that current plants are safe.

The primary objective of this analysis is to perform a comprehensive containment evaluation of the Nine Mile Point 2 plant. The approach addresses the requirements of Generic Letter 88-20 and NUREG-1335, utilizing plant-specific analyses and referenceable calculations (i.e., previously performed analyses and available data), which have been determined to be appropriate for the application to the Nine Mile Point 2 study. The second, but equally important, objective is to achieve technology transfer of the methodology to NMPC personnel. This is achieved by integrating team members through formalized training and task-by-task joint working sessions to implement the program plan.

Another objective of this plant specific evaluation is to provide a framework within which the following items can be understood:

- Do any unusual containment vulnerabilities exist in the Nine Mile Point 2 containment which are required to be modified by procedural or hardware changes?

- Can severe accident behavior information be presented to engineering, operations, and maintenance in a way that will assist these organizations in preventing or mitigating severe accidents through forward thinking approaches?

- Are there severe accident management techniques that should be developed?

Other objectives of this plant-specific evaluation are to allow the following:

- Provide a consistent interface with the accident sequence frequencies assessed in the updated, Level 1 internal events Nine Mile Point 2 IPE.

- Represent possible containment failure mechanisms

- Represent uncertainties in severe accident phenomenology

- Identify controlling plant features

- Incorporate technical information from many sources

- Provide a methodology that may be updated with new plant information and severe accident technology

- Highlight the time windows for recovery actions to be integrated into an accident management program.

As a result of meeting these objectives, the Nine Mile Point 2 Level 2 containment safety analyses are responsive to the NRC-requested Individual Plant Examination (IPE) for containment and severe accident evaluations as identified in Generic Letter 88-20 and NUREG-1335.

4.0.5 Approach to the Level 2 Analysis

The process of performing the containment analysis begins with an evaluation of the Nine Mile Point 2 Level 1 sequences. These sequences are categorized in terms of the type of challenge to containment posed by each sequence and the operability of systems that could mitigate these effects. Since risk is additive, it is possible to bin or group similar sequences based on these criteria, and consider each bin collectively as representing one challenge type to the containment. While each Level 1 accident sequence is explicitly treated in the computer model of the Nine Mile Point 2 plant, the rules and split fractions used in the assessment take advantage of the recognition of similar accident challenges or classes from the Level 1 analysis.

Plant structural and physical information is required in order to evaluate the response of the containment systems to the core damage event. This information is used to perform the plant-specific analyses, as well as, to characterize or modify the results of studies from other similar plants for use within the Nine Mile Point 2 study.

The determination of ultimate containment failure capability is required to assess the timing, size, and location of possible failure modes. A containment analysis of the Nine Mile Point 2 Mark II containment by ABB Impell is included in the analysis to provide insights on the containment failure pressure, temperature, and location.

An integrated deterministic evaluation of the physical response within the containment to accident challenges uses the calculated thermal-hydraulic response of containment compared with the ultimate containment capability to identify possible containment failure points for a particular accident scenario. Therefore, an assessment of the physical response of plant and containment systems to each challenge is performed using a deterministic code.

The containment event tree (CET) is a device for representing these various accident scenarios in terms of system capability and human interaction to arrest core damage and prevent an undesirable outcome.

The objective of obtaining a realistic, analytical result can be achieved by including necessary detail regarding system capability and human intervention. In this context, the containment event tree allows for the consideration of the operating staff implementation active mitigation strategies which might reduce the severity of release or delay the time of the release. Such actions would allow additional time for the implementation of other actions which might terminate the event. Actions which prevent major releases, reduce the consequences, or delay a radionuclide release are effective in reducing the overall plant risk. Consequently, these operating staff actions are treated explicitly in the containment event tree for each sequence type.

The containment event tree nodes are quantified by developing functional fault trees to describe the various factors which influence the nodal failure probability. These detailed, plant-specific nodal fault trees are then solved to obtain system Boolean equations for incorporation directly into RISKMAN®. This facilitates the ability to utilize the RISKMAN® code throughout the IPE and assures consistency depicting accident scenarios and communicating results.

The Nine Mile Point 2 IPE also includes an assessment of phenomenological matters considering NRC positions on these issues and related uncertainties including the issues as summarized in IDCOR Technical Report 86.1 (i.e., letters from T. Speis, NRC, to A. Buhl, ITC, "Position

Papers for the NRC/IDCOR Technical Issues," dated September 22, 1986; November 26, 1986; and March 11, 1987).

The Nine Mile Point 2 Level 1 system analysis is integrated with the containment analysis so that initiating events and system failures (resulting in core damage) that also impair containment systems are accounted for through the direct coupling of the Level 1 and Level 2 event trees. This direct linking on a sequence-by-sequence basis of the front-end to back-end portions ensures that the support state conditions (e.g., dependencies) are properly accounted for throughout the front-end and back-end trees. These trees and their direct linking include preventive or mitigative features as well as timing considerations. Three different containment event tree structures, each linked to the appropriate front-end event tree sequence, are used to properly handle the various combinations of accident sequences that may occur, (e.g., containment failure before core damage cases as well as vice-versa and containment bypass sequences).

Figure 4.0-2 provides a simplified overview of the Nine Mile Point 2 Level 1 and Level 2 PRA model identifying the nomenclature of the various elements of the event tree models and their interfaces.

Figure 4.0-3 provides a simplified flow chart of the major technical tasks involved in the Nine Mile Point 2 Level 2 evaluation and where each of these elements is discussed in this report.

DEVELOPMENT OF AN INTEGRATED RISK MODEL

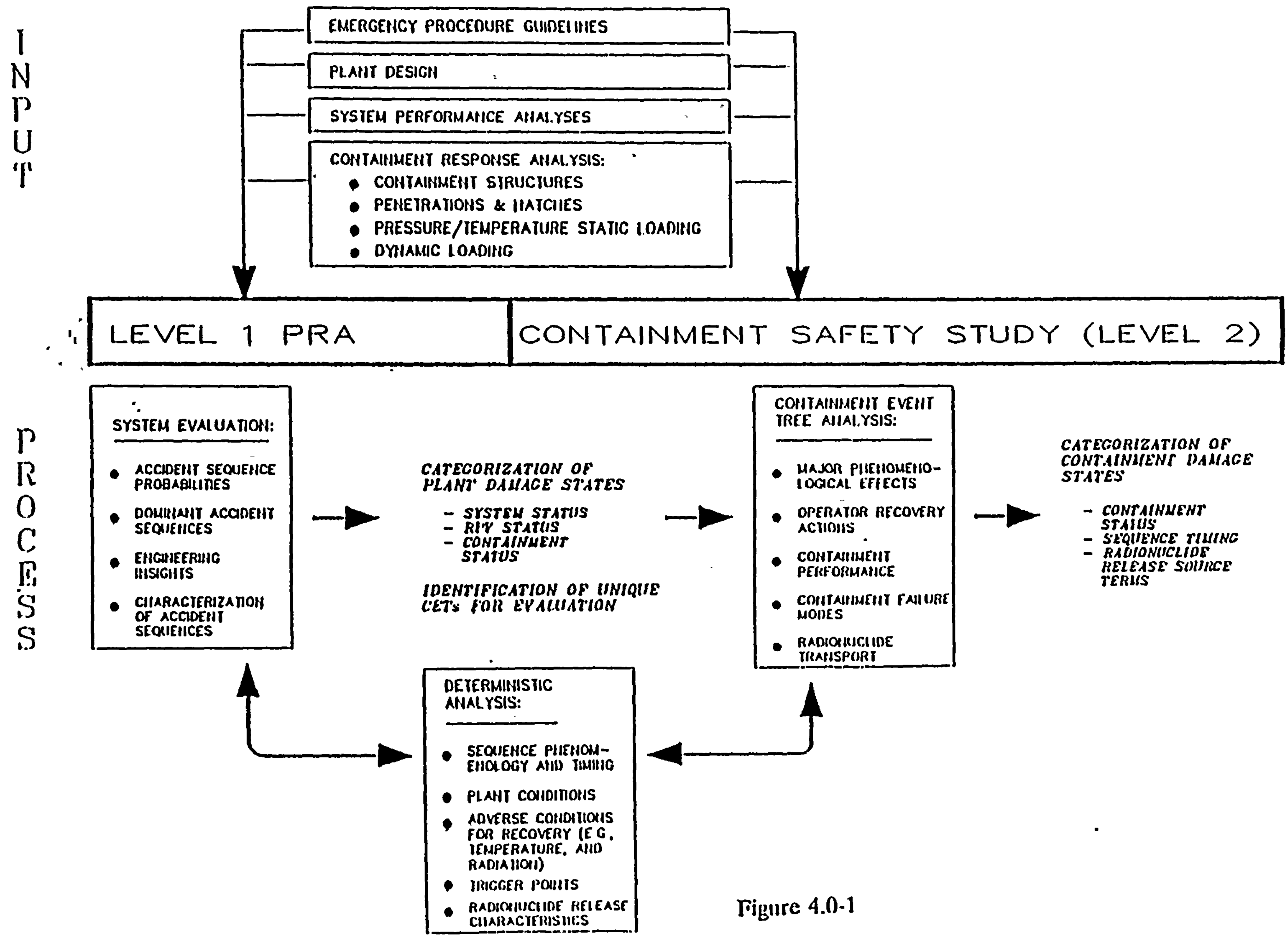


Figure 4.0-1

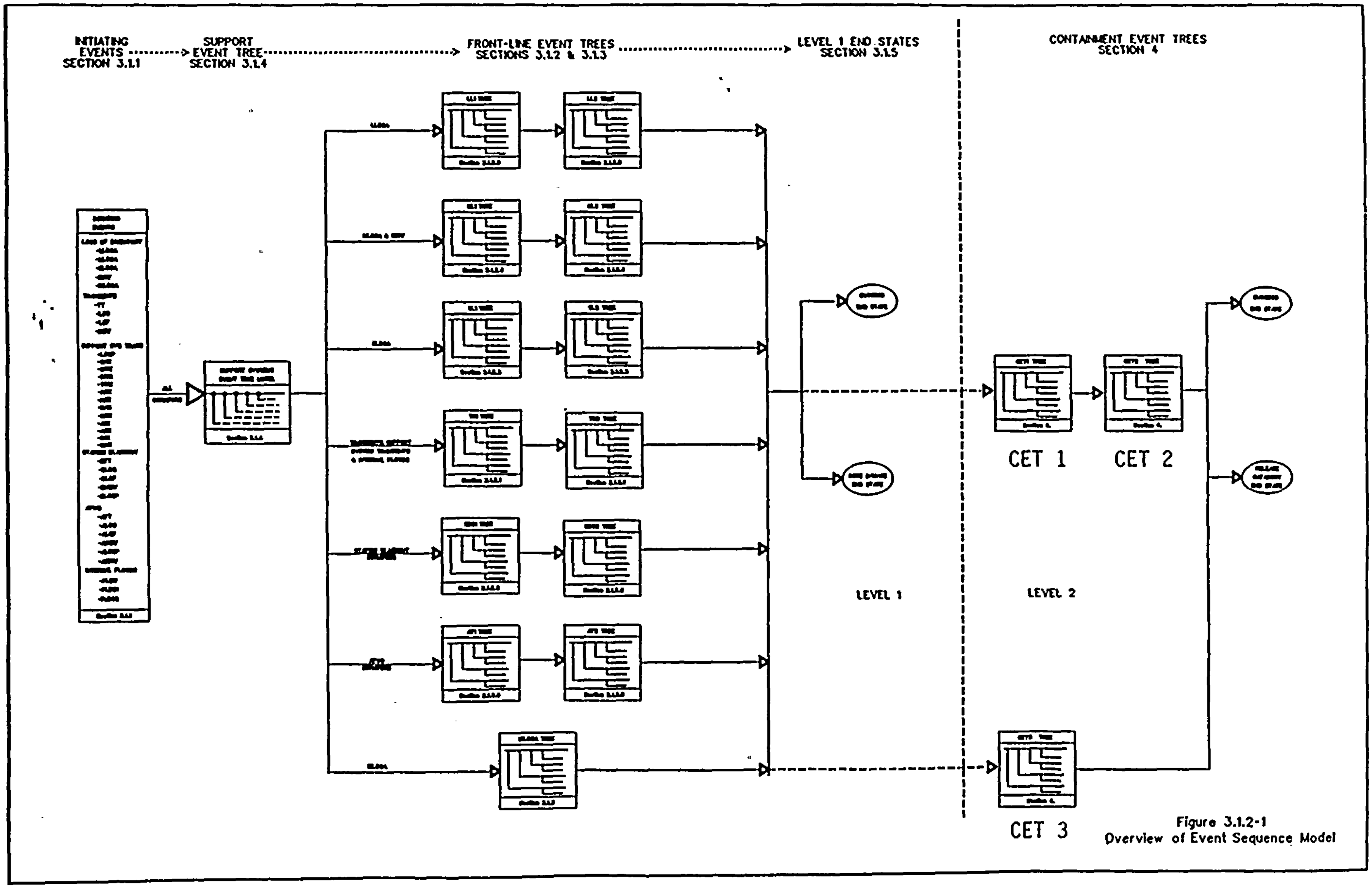


Figure 4.0-2 Nine Mile Point Unit 2 PRA Model

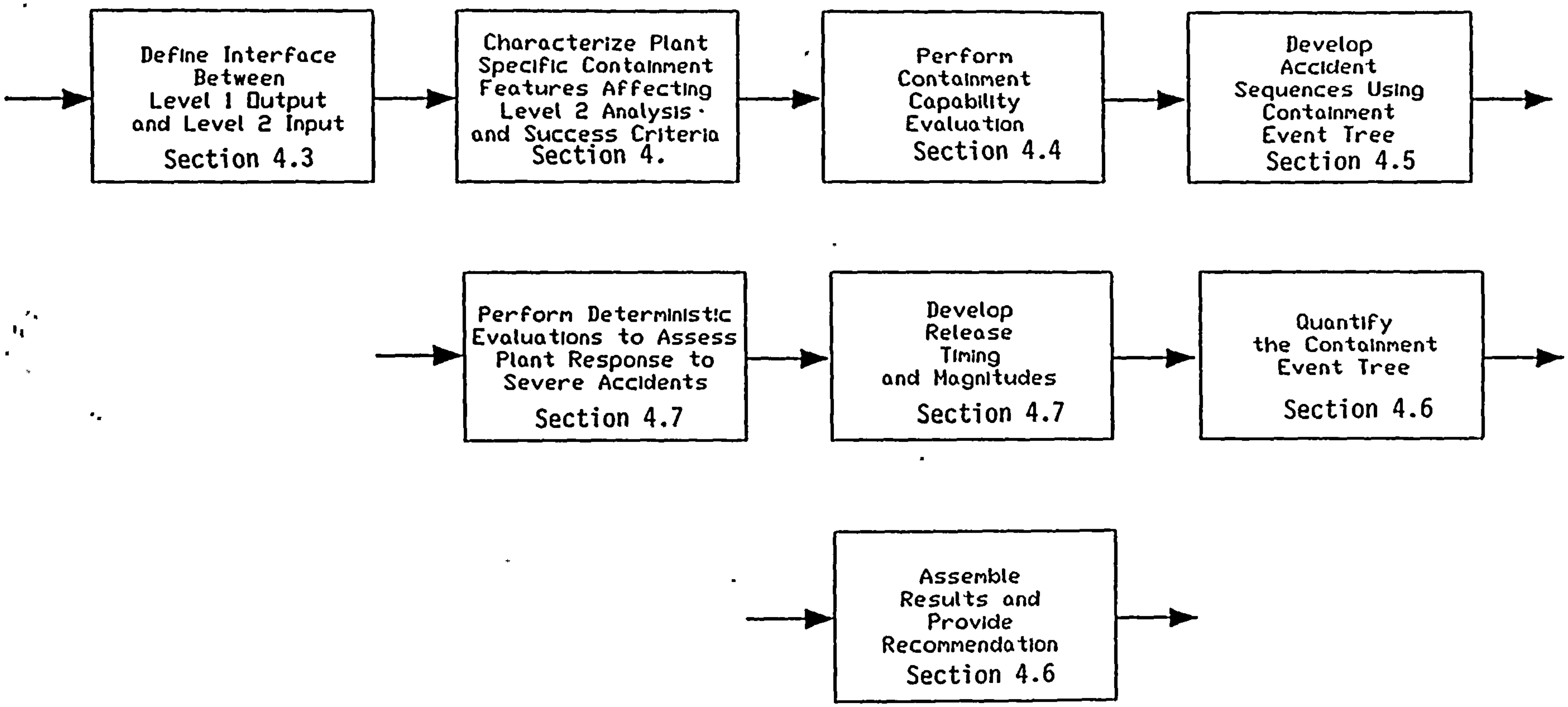


Figure 4.0-3 Major Technical Elements in Level 2 Evaluation

4.1 Plant Design and Plant Description

This subsection provides the following data and design descriptions:

- Primary containment
- Secondary containment
- Functional capability of containment.

4.1.1 Summary of Primary Containment Features

The primary containment (PC) structure is a low leakage, pressure suppression system. It forms a fission product barrier which, in conjunction with the secondary containment system, will contain the radioactive fission products generated during all modes of plant operation and any postulated design basis accident so that off-site doses will not exceed the requirements of 10CFR100. Nine Mile Point Unit 2 employs a Mark II pressure suppression containment system which houses the reactor vessel, the reactor coolant recirculation loops, and other branch connections of the reactor coolant pressure boundary (RCPB). The Mark II primary containment is of the pressure suppression type in the form of a conical frustrum over a cylindrical section. The drywell is in the upper conical section and the suppression chamber is in the lower cylindrical section. The suppression chamber is separated from the drywell by the drywell floor and contains a large reservoir of water called the suppression pool.

The general configuration of the primary and secondary containments are shown in Figure 4.1-1. The principal design parameters and characteristics are given in Table 4.1-1.

The main functions of the primary containment system as stated in the Nine Mile Point Unit 2 FSAR [Ref. 86] are:

- To withstand the pressures and temperatures resulting from a loss-of-coolant accident (LOCA).
- To withstand the peak environmental transient pressures and temperatures associated with the postulated spectrum of line breaks.
- To withstand jet forces associated with the flow from the postulated rupture of any pipe within it.
- To withstand missiles from internal sources and excessive motion of pipes that could directly or indirectly endanger the integrity of the containment.
- To withstand hydrodynamic loads.

In addition to those functions specified above, the containment also provides the following functions:

- A heat sink using the suppression pool

- A barrier to prevent radionuclide release to the environment

- A potential scrubbing mechanism in the radionuclide release path using the suppression pool and the drywell sprays.

4.1.1.1 Component Description

The following containment system components will be discussed in this subsection:

- Drywell

- Reactor Pedestal

- Biological Shield Wall

- Drywell Head

- Suppression Pool

- Downcomer Vent System

- Primary Containment Penetrations.

Drywell [Ref. 86]

The drywell houses the reactor vessel and associated equipment. The primary function of the drywell is to contain the radioactivity and withstand pressures and temperatures resulting from a breach of the reactor coolant pressure boundary (RCPB) and to provide a hold up time for decay of any radioactive material released.

The drywell is a steel lined reinforced concrete vessel in the shape of a truncated cone having a base diameter of approximately 91 ft and a top diameter of approximately 34 ft (see Figure 4.1-2). The internal design pressure for the drywell is 45 psig and the maximum external design pressure differential is 4.7 psid.

The drywell floor which separates the drywell from the suppression chamber is constructed of steel lined reinforced concrete. It serves both as a pressure barrier between the drywell and the suppression chamber and as the support structure for the reactor pedestal and downcomers. The design downward and upward differential pressures on the drywell floor are 25 and 10 psid, respectively. In addition, the drywell floor is structurally designed to withstand a pressure load equal to 1.5 times the design downward and upward pressures, thereby increasing the safety margin.

Reactor Pedestal [Ref. 86]

The reactor pedestal is a vertical cylindrical shell type with a reinforced concrete foundation. (The pedestal wall thickness is 5.125' at drywell floor.) This foundation supports the reactor pressure vessel (RPV) and the biological shield wall. The reactions transmitted by the shield wall and the RPV to the pedestal are due mainly to seismically induced loads, pipe break pressures and pipe rupture restraints attached directly to the shield wall and the loads transmitted to the shield wall by the radial beam systems. Other pipe rupture restraints are attached directly to the reactor pedestal.

In plan, the reactor pedestal is located in the centerline of the RPV and, therefore, on the centerline of the primary containment vessel. In elevation, the reactor pedestal is located directly under the RPV and shield wall.

The bottom of the RPV skirt and the shield wall are connected directly to the top of the reactor pedestal. The bottom of the reactor pedestal is keyed into the reinforced concrete containment basemat.

Biological Shield Wall [Ref. 86]

The biological shield wall located in the drywell adjacent to the RPV reduces neutron and gamma radiation from the reactor in order to permit drywell access and maintenance with minimum radiation exposure to personnel. It also extends the life of drywell components that may be damaged from gamma radiation.

The biological shield is a high density, steel reinforced concrete cylindrical structure surrounding the vessel. It is 1 foot 8 1/2 inches thick and has an outside diameter of 31 feet 6 1/2 inches. The biological shield extends from the reactor pedestal elevation to elevation 314 ft 1 1/2 inches.

The biological shield wall provides lateral support for the RPV to accommodate both seismic forces and jet forces resulting from the breakage of any pipe attached to the RPV. Lateral support is provided in compression only, thereby eliminating tension forces that would pull the containment inward.

Drywell Head [Ref. 86]

The drywell head surrounds the RPV head. The top of the drywell is capped with a bolted/gasketed head (Figure 4.1-2). The drywell head is constructed of 1 1/8 inch thick steel plate, semi-ellipsoidal shape with a cylindrical flange which mates with a flange on the drywell. It is sealed with two replaceable O-rings.

Suppression Pool [Ref. 86]

The pressure suppression pool (see Figure 4.1-2) has approximately 154,400 ft^3 of demineralized water contained in the pressure suppression chamber at high water level and approximately 145,200 ft^3 at low water level. Air volume above the pool changes as water volume changes, therefore, air volume is 190,600 ft^3 and 199,800 ft^3 for high and low water level, respectively. The pressure suppression chamber is a stainless steel clad steel lined, reinforced concrete vessel in the shape of a cylinder, having an inside diameter of 91 ft. The foundation mat, to which the vessel is anchored, is lined with steel plates within the inside diameter of the cylinder. The steel plates are welded to each other and to steel embedments to maintain the primary containment function of a gastight enclosure.

The suppression chamber is designed to serve many purposes, including the following:

- The suppression pool acts as the heat sink for all of the following: A Loss of Coolant Accident (LOCA) within the drywell; a safety valve or safety relief valve lift; or, the RCIC turbine exhaust. Energy is transferred to the suppression pool by the discharge piping from the reactor pressure relief valves, the drywell vent system, the RCIC system turbine exhaust pipe, and the RHR heat exchangers relief valves. The exhaust steam is discharged below the water surface and is condensed. The SRV discharge piping is used as the energy transfer path for any condition which requires relief valve operation. The drywell vent system is the energy transfer path for energy released to the drywell during a LOCA.

- The suppression pool is the primary source of water for the low pressure core spray (LPCS) and low pressure coolant injection (LPCI) system, provides a safety-related source of water for the reactor core isolation cooling (RCIC) and high pressure core spray (HPCS) systems.

- The suppression pool acts as an intermediate heat sink for transferring heat from the reactor and then to the RHR system in the suppression pool cooling mode.

The reactor coolant recirculation piping instantaneous circumferential rupture represents the most rapid design basis accident energy addition to the pool. For this accident, the vent system, which connects the drywell and suppression chamber, conducts a flow from the drywell to the suppression chamber without excessive resistance and distributes this flow effectively and uniformly in the pool. The pressure suppression pool receives this flow, condenses the steam portion, and releases the non-condensible gases to the pressure suppression chamber air space. An additional benefit, not part of the design basis, is that the suppression pool acts as an effective scrubber of fission products other than nobles gases when the release pathway is through the suppression pool.

The suppression pool receives steam and water energy from the reactor relief valve discharge piping or the drywell vent system downcomers which discharge under water. The steam, and any water carryover, cause an increase in pool volume and temperature. Energy can be removed from the suppression pool cooling mode.

The suppression pool water level and temperature are continuously monitored in the control room and are maintained within strict limits imposed by technical specification requirements.

Downcomer Vent System [Ref. 86]

The downcomer vent system connects the drywell to the suppression chamber and is used to conduct steam and non-condensible flow into the pool following a postulated primary system pipe rupture inside the drywell (i.e., LOCA). The downcomers consist of 121 pipes open to the drywell and submerged 9.5 ft below the low water level (operating minimum) of the suppression pool. The internal diameter of each downcomer is 23.25 inches. The downcomers project 3 to 6 in above the drywell sloping (or sloped) floor so that small quantities of water leakage flow past the downcomers and are collected in the drywell floor drain system.

Each downcomer opening is shielded by a 2 1/4 inch thick steel deflector plate to prevent overloading any single vent pipe by direct flow from a pipe break to that particular vent. The deflector plate also minimizes the potential for downcomer blockage by debris.

As part of the downcomer system, there are four sets of wetwell to drywell vacuum breakers (2 vacuum breakers in series for each set). Vacuum breakers provide a return flow path from the suppression chamber gas space to the drywell. The vacuum breakers are designed to limit the negative differential pressure between the drywell and the suppression chamber to a maximum value of 4.7 psid. Each vacuum breaker set consists of two relief valves (i.e., check valves) in series to ensure a leak tight boundary under positive drywell-to-suppression pool chamber differential pressure conditions. The vacuum breakers are located inside the primary containment drywell and do not, therefore, form an extension of the primary containment boundary. These valves are mounted in piping that connects the drywell and suppression chamber. This location removes the vacuum breakers from the direct effects of chugging transients.

The elevation of the vacuum breakers and the elevation of some other key locations within the containment is shown in Figure 4.1-3. A schematic of a "straight-pipe" downcomer is shown in Figure 4.1-4.

Primary Containment Penetrations

In order to maintain design containment integrity, containment penetrations have the following characteristics:

- They are capable of withstanding peak transient pressures which could occur due to the postulated DBA rupture of any pipe inside the drywell.

- They are capable of withstanding forces caused by impingement of fluid from the rupture of the large local pipe or connection without failure.

- They are capable of accommodating the thermal and mechanical stresses which may be encountered during all modes of operation without failure.

The types of containment penetrations are as follows:

- Pipe penetrations

- Instrumentation penetrations

- Electrical penetrations
- Traveling In-Core Probe (TIP) penetrations
- Personnel and Equipment Access Locks
- Access to the Pressure Suppression Chamber
- Access for refueling operations.

4.1.2 Summary of Secondary Containment [Ref. 86]

The secondary containment, consisting of the reactor building and auxiliary bay structures, completely encompasses the primary containment and provides a radionuclide barrier to trap fission products if they escape from the primary containment. Figure 4.1-2 shows a cross section of the reactor building as it surrounds the containment. The secondary containment is maintained at a negative pressure of 0.25 in water gauge to ensure that while the systems are operating any leakage is into the reactor building.

The reactor building houses the refueling and reactor servicing equipment, the new and spent fuel storage facilities, and other reactor auxiliary or service equipment, including RCIC system, RWCU demineralizer system, SLC system, CRD system equipment, HPCS, and electrical equipment components. The reactor building auxiliary bays house the LPCS system, the RHR system heat exchangers and pumps, the RBCLCW system heat exchangers, and electrical equipment components.

The reactor building structure is designed to meet the following design bases:

- The reactor building is designed to meet Category I requirements.
- The reactor building is designed and constructed in accordance with the structural design criteria given in Section 3.8 of Nine Mile Point Unit 2 FSAR.
- The reactor building is designed to provide low inleakage and outleakage during normal plant operation.
- The reactor building is designed to withstand applied wind pressures resulting from the design basis wind velocity.
- The reactor building is designed to withstand pipe whip loads plus jet impingement loads due to high energy pipe breaks outside of the primary containment.
- The reactor building is designed to allow for periodic inspection and functional tests of the penetrations.
- The reactor building is designed to withstand tornado-generated missiles.

- The reactor building is designed for all probable combinations of the design basis wind, tornado velocities, and associated difference of pressure within the structure and atmospheric pressure outside of the structure.

- All entrances to the reactor building are through double door air lock systems.

Figure 4.1-5 shows the model nodalization of the reactor building, used in the deterministic code, MAAP.

Ventilation System - The normal reactor building ventilation system (HVRS) is designed to automatically shut down and isolate and to automatically start the SGTS and safety-related unit coolers upon receipt of any of the following signals that indicate either a LOCA or a refueling accident:

- High drywell pressure

- Reactor vessel low water level

- High radiation level in exhaust ducts above or below the refueling floor.

The NMP-2 secondary containment pressure control function utilizes the HVRS (normal operation) and the SGTS (emergency operation) instrumentation and controls to maintain a negative pressure of 0.25 in water gauge with respect to the atmosphere. This ensures that while the systems are operating any leakage is into the reactor building. All reactor building air is either exhausted through the exhaust air plenum, where it is constantly monitored, or discharged through the filtration units of the SGTS.

Table 4.1-1

PRINCIPAL DESIGN PARAMETERS AND CHARACTERISTICS OF NINE MILE POINT UNIT 2 PRIMARY CONTAINMENT [Ref. 86]

Pressure Suppression Chamber Internal Design Pressure External Design Pressure	 45 psig 4.7 psid
Drywell Internal Design Pressure External Design Pressure Differential	 45 psig 4.7 psid
Drywell Net Free Volume	306,200 $ft.^3$ (1)
Pressure Suppression Chamber Free Air Volume (Minimum)	190,600 $ft.^3$ (2)
Pressure Suppression Chamber Free Air Volume (Maximum)	199,800$ft.^3$ (1)
Pressure Suppression Pool Water Volume (Maximum)	154,200 $ft.^3$ (1)
Pressure Suppression Pool Water Volume (Minimum)	145,200 ft^3 (1)
Design Temperature of Drywell	340°F
Design Temperature of Pressure Suppression Chamber	270°F

(1) Calculated value obtained from USAR Rev. 0, April 1989.

(2) Calculated valve obtained from USAR Rev. 0, April 1989.

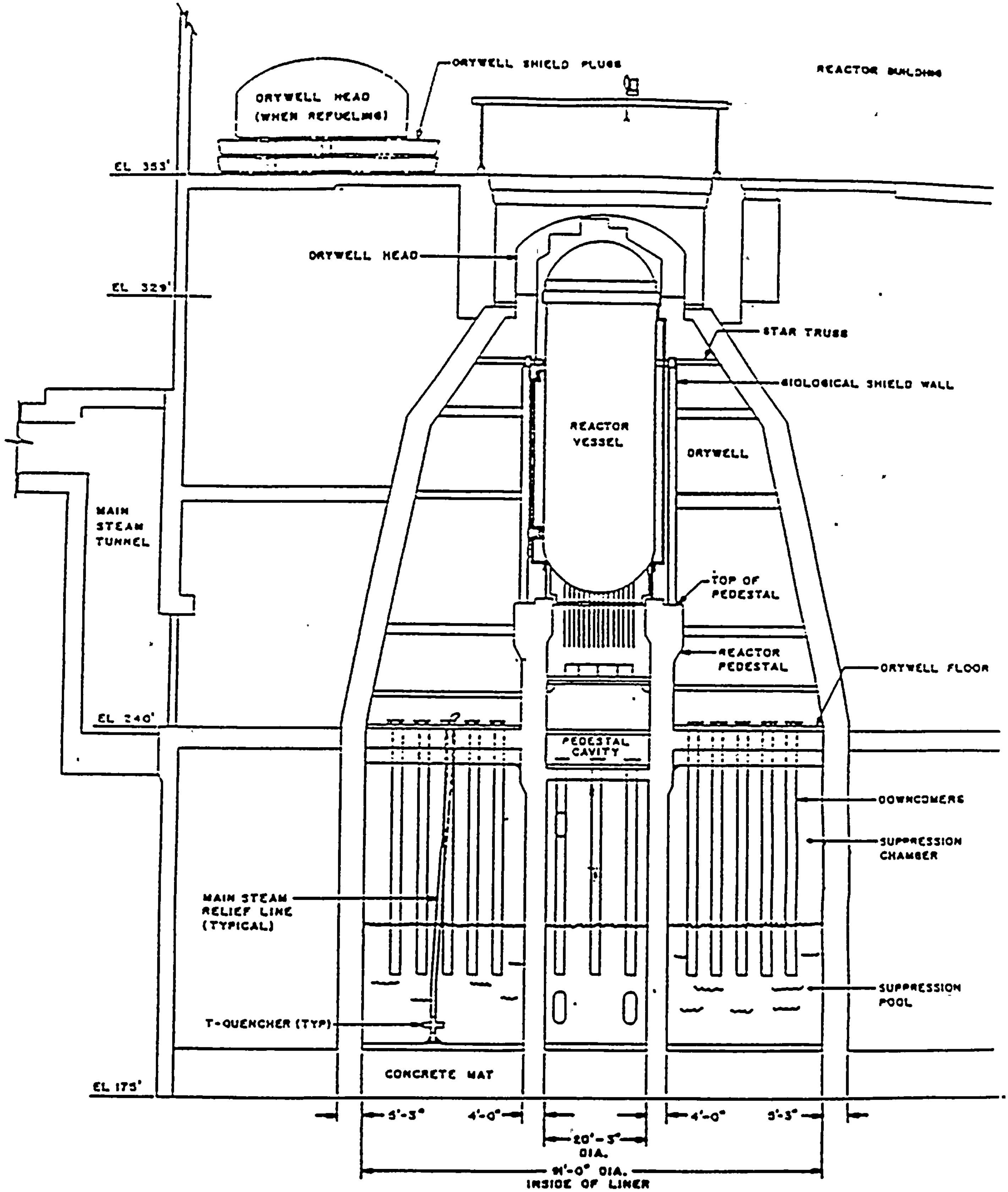


Figure 4.1-1

Nine Mile Point Unit 2 Mark II Containment

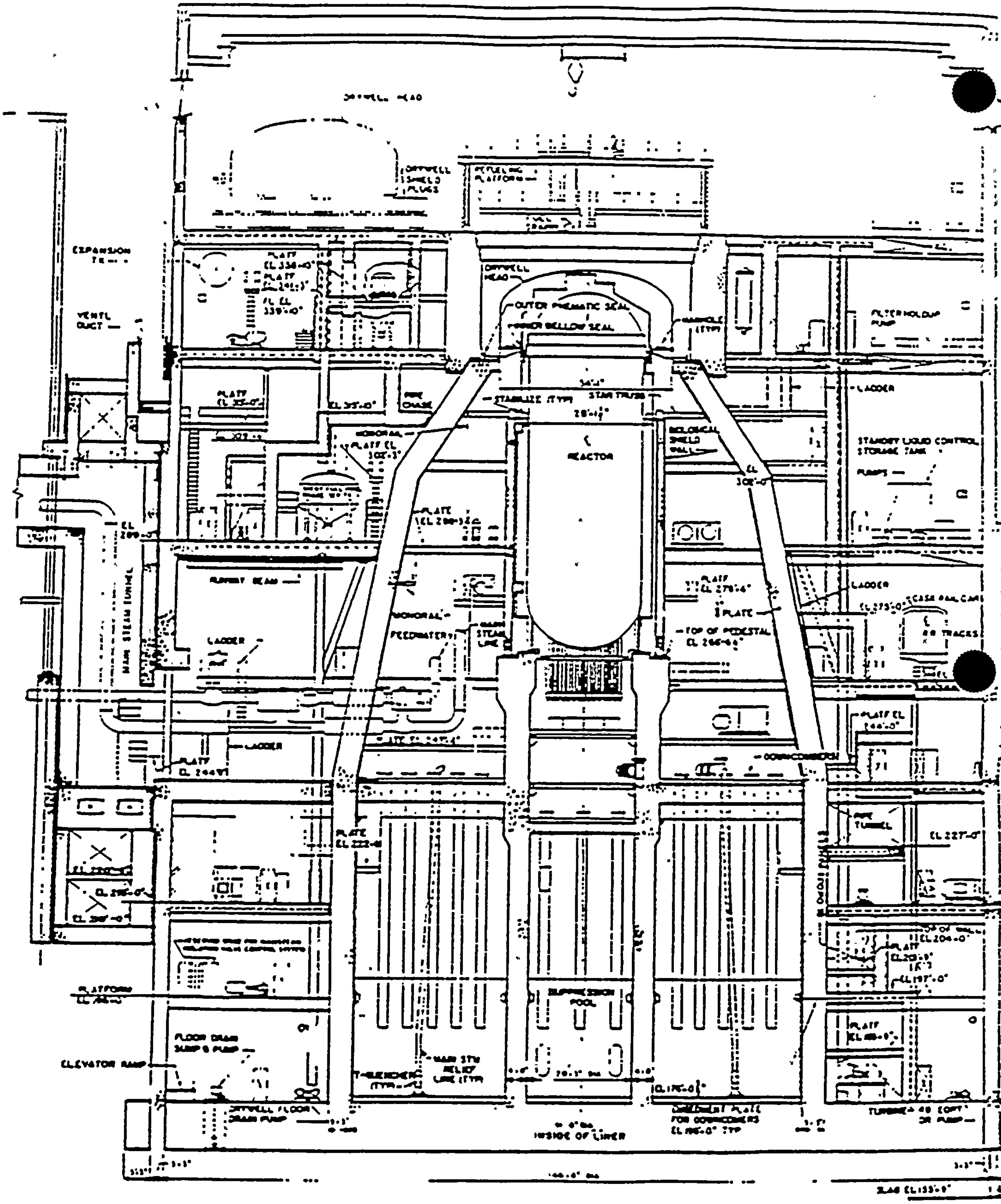


Figure 4.1-2

Nine Mile Point Unit 2 Containment Details

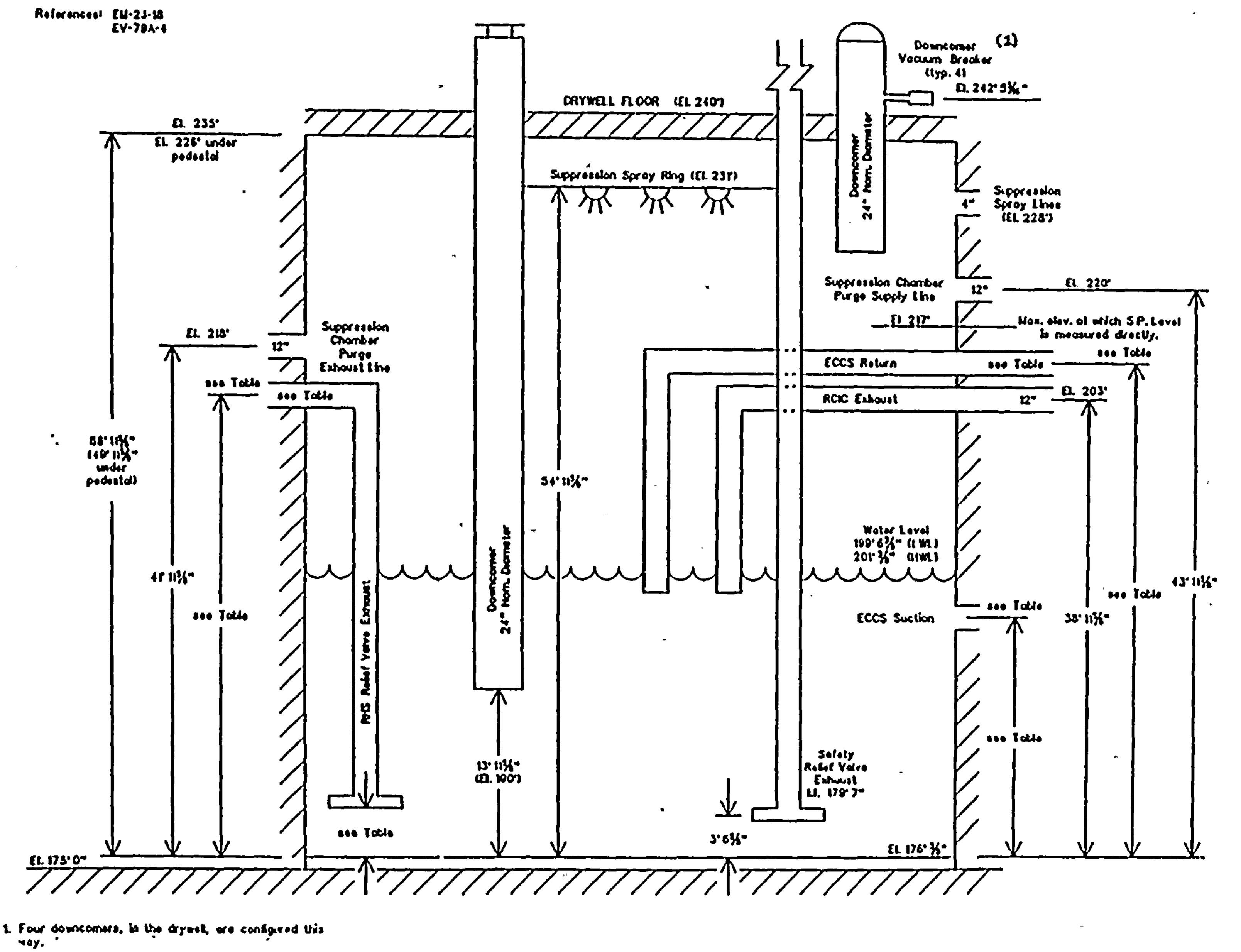


Figure 4.1-3 Nine Mile Point Unit 2 Suppression Pool Component Locations

Notes to Table 4.1-3

Description	Pen.	(Sub)System(s)[2]	Elevation[3]	Height above S.P. Floor	Nom.[4] Pipe Size	T-Quencher[5] Height above S.P. Floor
ECCS Suction[1]	Z5A	RHR A	195′	18′ 11⅝"	24"	N/A
	Z5B	RHR B	195′	18′ 11⅝"	24"	N/A
	Z5C	RHR C	195′	18′ 11⅝"	24"	N/A
	Z12	HPCS	195′	18′ 11⅝"	20"	N/A
	Z15	LPCS	195′	18′ 11⅝"	20"	N/A
	Z17	RCIC	199′	22′ 11⅝"	3"	N/A
ECCS Return[1]	Z6A	RHR A & LPCS	206′	29′ 11⅝"	18"	N/A
	Z6B	RHR B & C	206′	29′ 11⅝"	18"	N/A
	Z13	HPCS & RCIC	218′	41′ 11⅝"	12"	N/A
RCIC Exhaust	Z19	RCIC	203′	26′ 11⅝"	12"	N/A
	Z90	RCIC (Vac.-Br)	216′	39′ 11⅝"	1.5"	N/A
Suppression Chamber Purge Supply	Z50	CPS	220′	43′ 11⅝"	12"	N/A
	Z59	CPS	217′	40′ 11⅝"	2"	N/A
Suppression Chamber Purge Exhaust	Z51	CPS	218′	41′ 11⅝"	12"	N/A
Suppression Spray	Z7A	RHR A	228′	51′ 11⅝"	4"	N/A
	Z7B	RHR B	228′	51′ 11⅝"	4"	N/A
Relief Valve Exhaust Lines	Z73	RHR	218′	41′ 11⅝"	6"	N/A
	Z88A	RHR A	202′	25′ 11⅝"	12"	19′ 11⅝" (196′)
	Z88B	RHR B	202′	25′ 11⅝"	12"	19′ 11⅝" (196′)
	Z98A	RHR A & LPCS	202′	25′ 11⅝"	3"	N/A
	Z98B	RHR B, C & HPCS	202′	25′ 11⅝"	3"	N/A

Notes to Table 4.1-3 (cont)

SRV Number	T-Quencher	T-Quencher[6] Elevation	T-Quencher Height above S.P. Floor
2MSS*PSV120	2SVV*DIFF120	179' 7"	3' 6¼"
2MSS*PSV121	2SVV*DIFF121	179' 7"	3' 6¼"
2MSS*PSV122	2SVV*DIFF122	179' 7"	3' 6¼"
2MSS*PSV123	2SVV*DIFF123	179' 7"	3' 6¼"
2MSS*PSV124	2SVV*DIFF124	179' 7"	3' 6¼"
2MSS*PSV125	2SVV*DIFF125	179' 7"	3' 6¼"
2MSS*PSV126	2SVV*DIFF126	179' 7"	3' 6¼"
2MSS*PSV127	2SVV*DIFF127	179' 7"	3' 6¼"
2MSS*PSV128	2SVV*DIFF128	179' 7"	3' 6¼"
2MSS*PSV129	2SVV*DIFF129	179' 7"	3' 6¼"
2MSS*PSV130	2SVV*DIFF130	179' 7"	3' 6¼"
2MSS*PSV131	2SVV*DIFF131	179' 7"	3' 6¼"
2MSS*PSV132	2SVV*DIFF132	179' 7"	3' 6¼"
2MSS*PSV133	2SVV*DIFF133	179' 7"	3' 6¼"
2MSS*PSV134	2SVV*DIFF134	179' 7"	3' 6¼"
2MSS*PSV135	2SVV*DIFF135	179' 7"	3' 6¼"
2MSS*PSV136	2SVV*DIFF136	179' 7"	3' 6¼"
2MSS*PSV137	2SVV*DIFF137	179' 7"	3' 6¼"

Notes

1. RCIC has been included despite the fact that is NOT a true ECCS system.
2. Common acronyms were used rather than the NMP2 system codes, for easier use.
3. Elevation given on MEL (Master Equipment List on PRIME computer). It has been assumed that this value is given to the nearest inch. This value is assumed to be the elevation of the CENTER of the penetration.
4. Value given on MEL.
5. It has been assumed that the elevation of the T-Quencher (given on MEL) is the elevation of the CENTER of the T-Quencher.
6. Value given on drawing EM-2J, Rev. 18.

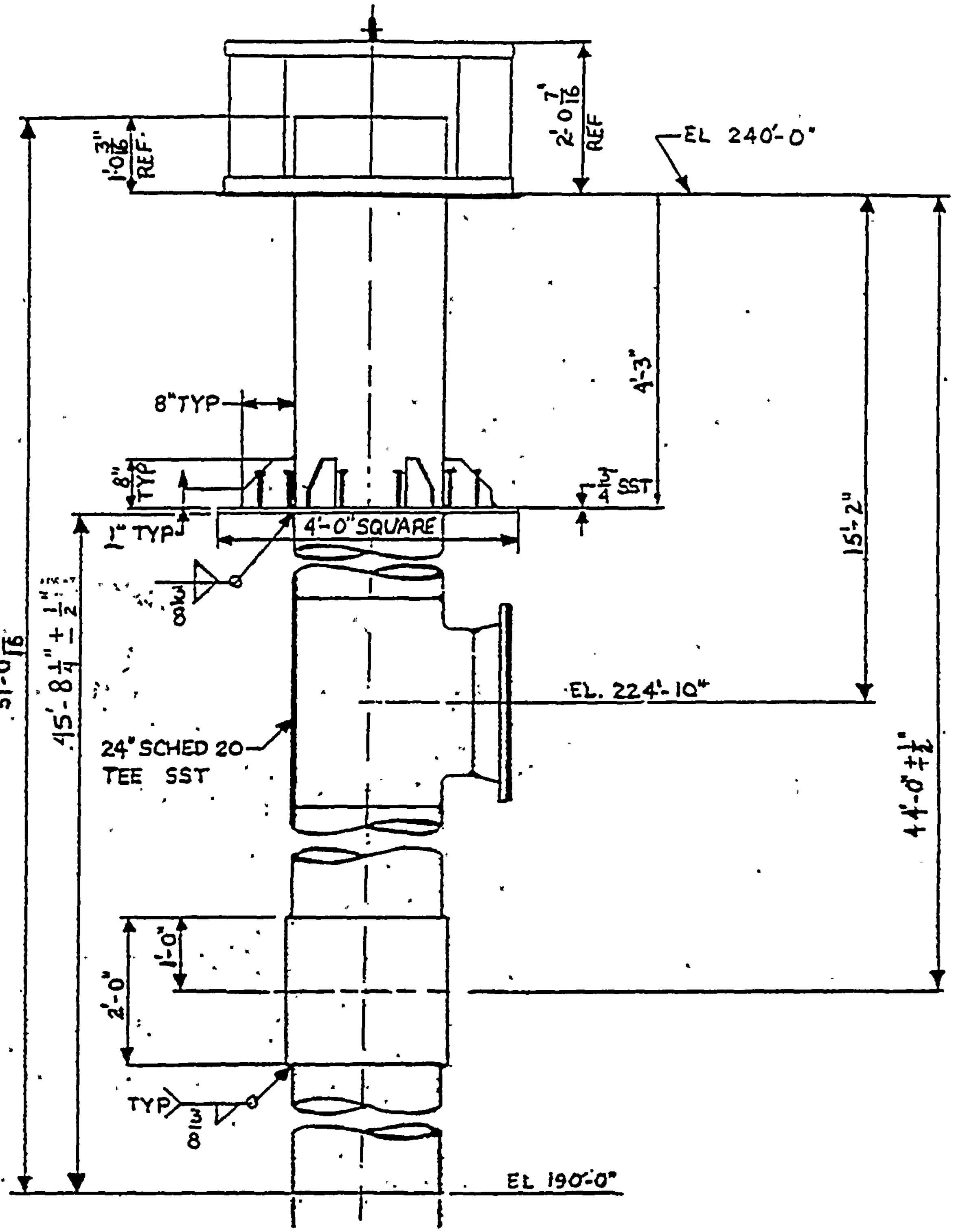
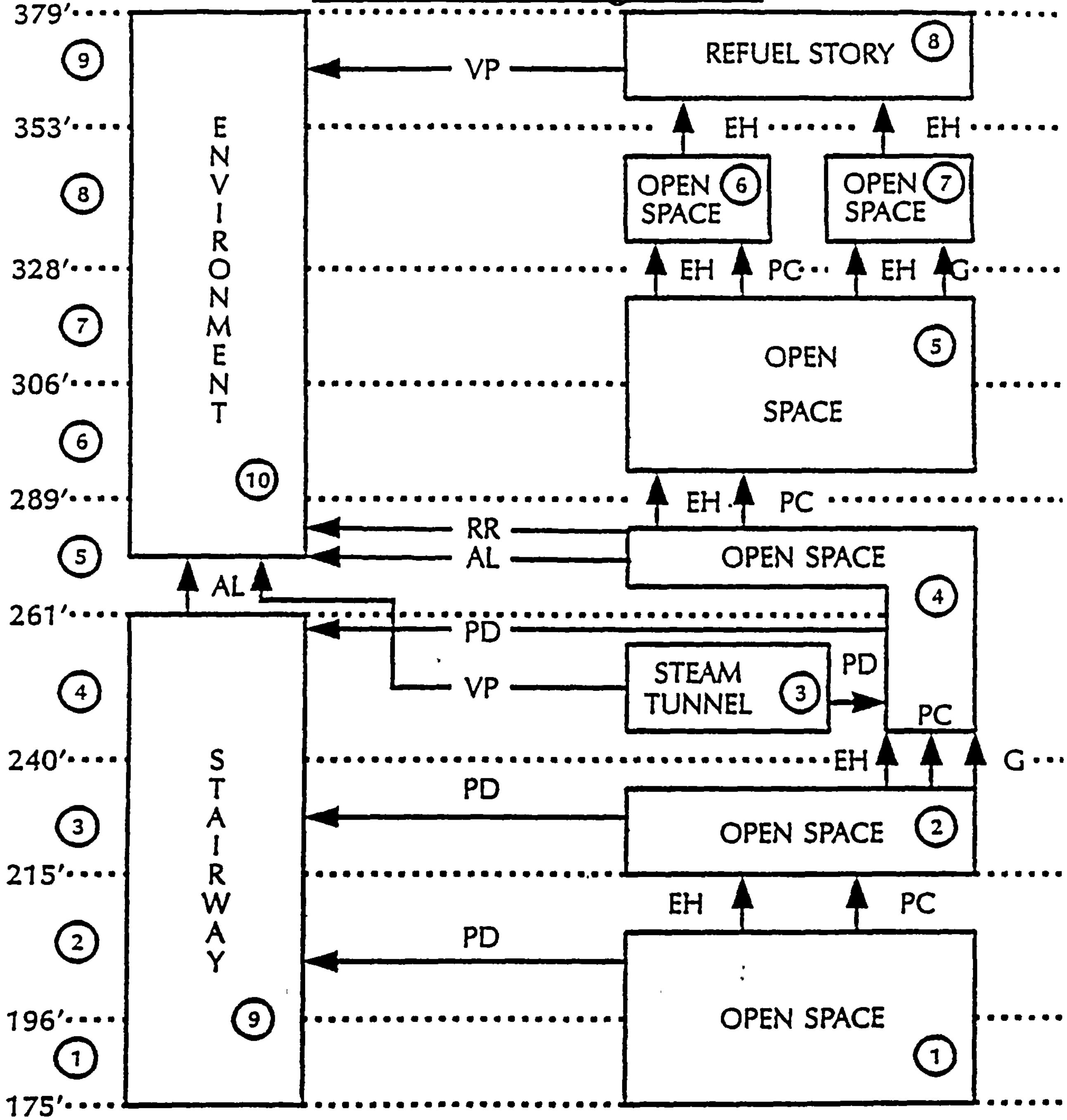


Figure 4.1-4

Schematic of Downcomer

Nine Mile Point 2
Drywell and Wetwell Containment Failure
Reactor Building Model



Nomenclature

VP - Vent Panel
RR - Rail Road
AL - Air Lock

EH - Equipment Hatch
PC - Pipe Chase
G - Grating

PD - Pressure Differential Failures

JCR43

Figure 4.1-5

4.2 PLANT MODELS AND METHODS FOR PHYSICAL PROCESSES

The modeling of physical processes and phenomena is carried out in two principal ways:

- First, computer code models, principally MAAP, are used to calculate plant response and radionuclide releases

- Secondly, generic issues which are not modeled by such codes are treated in a probabilistic manner within the containment event tree.

Subsection 4.2.1 documents the approach taken in the deterministic calculations to support the modeling of core melt progression in the containment event trees:

- Key event timing

- Containment pressure and temperatures

- Radionuclide releases.

The MAAP code is the primary tool used in the analysis; extensive MAAP calculations were performed to gain insights from possible variations in modeling or assumptions.

Subsection 4.2.2 documents the phenomena not included explicitly in the MAAP model.

Subsection 4.2.3 provides a list of key assumptions used in the analysis.

Subsection 4.9 summarizes some of the formal sensitivity evaluation performed in support of the IPE.

4.2.1 Deterministic Evaluation in Support of Severe Accident Analysis

Primary and secondary containment response to pressures, temperatures, flow rates, and timing of actions was evaluated using the BWR Mark II version of the MAAP thermal hydraulic code (version 3.0B, revision 7.01). This code, using Nine Mile Point 2 specific parameters as input, provided reactor and containment pressures, levels (water and radiation), and temperatures. Also calculated were the time windows between key events such as core damage and containment failure.

The MAAP results are used in the Level 2 analysis to determine success criteria, release timing and magnitude, and the location of the containment failure, and time available for critical accident management actions.

4.2.1.1 Purpose of Using Deterministic Analysis

The assessment of plant response under postulated severe accident scenarios is a complex integrated evaluation. The primary and secondary containment building responses are sensitive to pressures, temperatures, flows, and event timings. These parameters also affect the operator action timings, the radionuclide release timings, and the mitigating system performance assessments. Therefore, the proper Nine Mile Point 2 characterization of the severe accident progression is important to the realistic representation of the plant's response during a severe accident. These deterministic calculations are performed to estimate the time varying functions of pressures and temperatures in the RPV, the drywell, the wetwell, and the reactor building and the source term magnitude and timing of an impending radionuclide release.

This information is critical to the determination of the benefit associated with postulated recovery actions that could be implemented to mitigate specific effects of a severe accident. In addition, performing NMP2 deterministic analysis helps to maximize the understanding of severe accident progression within NMPC Engineering and Operations Departments to support future accident management efforts. It also develops in-house expertise in order to address other NRC issues.

4.2.1.2 Tools (Codes) Available

There are several codes available which can be utilized to determine a plant specific response. Included in the list of codes are; MELCOR, STCP, BWRSAR, LTAS, and MAAP. Among these, only MAAP, MELCOR, and STCP are fully integrated codes capable of modeling all aspects of a severe accident while representing all important interrelationships between phenomena.

The computer codes required for such analyses must be able to address several fundamental needs. These fundamental needs include the following:

1. Quantification and refinement of system success criteria (primary and containment systems);

2. Quantification of containment response to severe accident phenomena including the performance of containment systems, mission times, and response intervals;

3. Quantification of fission product releases (radionuclide magnitude, release timing, or co-current energy release);

4. Quantification of operator/recovery actions;

5. Ability to integrate the systems (frontend) and containment (backend) assessments.

Several codes are available that have been suggested for use for containment performance analyses. These codes are MAAP, MELCOR, STCP (Source Term Code Package), and BWRSAR (in combination with CONTAIN). In order to select the appropriate code to accomplish these tasks, their various attributes must be compared. The attributes of each code are generally described below. Additionally, Table 4.2-1 summarizes such a comparison.

The MAAP code compares well when considering capability of such a tool for accomplishing the tasks described above. Furthermore, user support, QA requirements, NRC "acceptability," and the required user's knowledge of severe accident phenomena are other attributes that should be considered when choosing a tool. MAAP is unique in that EPRI supports it and provides direct user support via the MAAP Users Group (MUG) and will maintain an archived and controlled version in support of QA. However, most importantly, it is an integrated code package that can most completely model the widest spectrum of severe accidents.

4.2.1.3 Advantages of MAAP

MAAP is judged to be the most appropriate tool to use in support of the IPE. Additional factors reinforce this decision:

- According to NUMARC and EPRI estimates, approximately 40 utilities representing 60 plants are expected to use MAAP for IPEs or PRAs to meet the requirements of the generic letter. A program is currently underway by NUMARC, EPRI, and DOE to bring NRC up to speed on MAAP and thus to make the IPE submittal process more orderly.

- Among the competing tools, MAAP has the highest level of QA documentation. This documentation includes two EPRI-sponsored efforts: a recently-completed formal design review by respected independent authorities and an independent validation and verification program which included a line-by-line review of the source code.

- MAAP is being aggressively developed and maintained. Continued development is being funded by EPRI and US DOE, and included in the development was an extensive thermal-hydraulic benchmarking activity sponsored by EPRI.

- In comparison to some of the new NRC tools, the MAAP code is fast-running and relatively mature, with a considerable history of successful use at utilities. It is quite practicable and very common to run the code on 386-type personal computer.

- An active MAAP User's Group consisting of over 40 members exist through which helpful information is shared between utilities and other MAAP users.

- EPRI has developed a guideline document to provide the users with recommendations on selected parameter values. These recommendations will assist the user in addressing many of the key areas of uncertainty.

- EPRI has performed numerous sensitivity analyses using MAAP to better address some of the NRC questions on important phenomenology.

4.2.1.4 Nine Mile Point 2 Unique Features Incorporated Into MAAP

The Nine Mile Point 2 MAAP model includes several plant specific features that could not be handled by the generic parameter file. Several of these Nine Mile Point 2 specific features were incorporated into the MAAP assessment methodology. These features include the following:

- Containment failure was specified as a function of drywell pressure and drywell gas temperature. This required multiple MAAP runs to first determine when the containment failure limits would be exceeded and then a run to force the containment to fail at the time the pressure-temperature containment failure "limit" curve was projected to be exceeded.

- The SW cross-tie was used as the alternate injection source to the reactor vessel. Alternate injection was included in the engineered safeguard section of the parameter file.

4.2.1.5 Deterministic Results

Section 4.7 describes the deterministic results that are used in the CET evaluation to determine the following:

- Success criteria
- Timing of release
- Failure location
- Radionuclide release magnitude.

4.2.2 Phenomena Not Included in MAAP Model

While the MAAP code is the primary deterministic assessment tool used in the containment evaluation, there are accident sequences and phenomena that the MAAP code is judged not to be effective in treating. For these sequences and phenomena, separate effects analyses, experiments, or expert judgement are used in the evaluation process. This subsection is a brief review of these sequences and phenomena that fall into this category and the disposition of them for the Nine Mile Point 2 IPE.

First, the Nine Mile Point 2 IPE includes the assessment of phenomenological matters considering NRC positions on these issues and related uncertainties including the issues as summarized in IDCOR Technical Report 86.1 (e.g., letters from T. Speis, NRC, to A. Buhl, ITC, "Position Papers for the NRC/IDCOR Technical Issues," dated September 22, 1986; November 26, 1986; and March 11, 1987). The IDCOR regulatory interaction program was devoted to the definition and resolution of open technical issues related to the assessment of severe accidents. Great progress has been made between the NRC and the Industry in resolving these issues through a variety of technical exchange meetings. Many of these issues manifest themselves as NRC concerns with specific MAAP models. NUREG-1335 stated that the industry should be aware of the NRC positions on unresolved issues when performing their IPE.

Table 4.2-1 lists the current status of each of the issues. While the status reflects the "resolution" of the issue for the NMP2 IPE baseline calculations, NMPC has expended significant resources to investigate possible sensitivities to the results that may occur as a result of these issues. Section 4.9 summarizes the results of the sensitivity evaluations.

In addition to the issues formally identified in the IDCOR NRC issues, there are other phenomenological issues that are not treated by MAAP. These issues are identified below along with their disposition in the Nine Mile Point 2 IPE:

Sequence or Phenomena	Disposition in Nine Mile Point 2 IPE
Ex-vessel Steam Explosion	Treated probabilistically in the IPE assessment
Mark I Shell Failure	Not applicable to the NMP2 Mark II containment
Direct Impingement Induced Failure	Treated probabilistically
Direct Containment Heating	Separate effects analysis and treated probabilistically
Reactivity Insertion during Core Melt Progression	Separate effects analysis and treated probabilistically

Two known errors in BWR, MAAP-Rev. 8.0 required special handling for some of the cases.

1) Wetwell venting sequences were modeled to force "wetwell failure" at the time of venting to properly track fission product releases.

2) ADS valves were manually reclosed at 85 psia in the drywell since the use of the input parameter to do this did not work properly.

4.2.3 Assumptions in the Modeling

In the course of a complex analysis, it is usually necessary to make assumptions or interpretations in order to model a system or group of systems.

Assumptions can introduce effects into the analysis that are:

- Realistic
- Conservative
- Non-conservative.

The assumptions made in the analysis are meant to provide a realistic, best estimate basis for the evaluation. However, because of uncertainties, assumptions that may not be known to be best estimate may be used. In general, these assumptions will take a conservative approach.

Finally, there may be exceptions to the above two rules in which apparent non-conservatisms have been included in the analysis. These apparent non-conservatisms are generally present because of modeling simplicity. This section discusses both the conservatisms and non-conservatisms in the analysis. Analysis not highlighted here are considered to be "best estimate".

4.2.4.1 Assumptions

This subsection provides a list of general assumptions that have been used in the Level 2 PRA analysis.

Level 1 Interface

- Each of the Level 1 end states represents a core damage situation in which the RPV water level is below 1/3 core height and decreasing as a result of insufficient coolant makeup to the RPV.

Level 2

- The containment event tree has been structured to be as concise as possible, but at the same time sufficiently detailed to represent important functional events that can result in significant differences in containment survivability, the magnitude of radionuclide releases, or timing of radionuclide releases.

- The list of containment failure modes considered in the Level 2 assessment is believed to be comprehensive, including all published failure modes; however, there may be other failure modes not currently postulated or known that could also compromise the containment. Section 4.4 summarizes the failures modes and their treatment.

- The containment capability has been assessed based on extrapolation of detailed deterministic calculations at "low" temperatures. The further extrapolation of the containment capability to high temperatures and pressures has been performed using separate effects assessments and engineering judgement.

- The response of the containment to severe accidents (i.e., calculated pressures and temperatures) is modeled using the MAAP code. The results have been checked against other published deterministic codes from similar plants.

- The calculated source terms (i.e., radionuclide release magnitude and timing) have been determined using the MAAP code. These accident source terms have been compared with other deterministic code calculations for similar plants.

- Generally, the treatment of hardware repair and recovery is explicitly treated in the Level 1 analysis. The Level 2 model considers repair and recovery of systems that primarily affect the ability of the operator to maintain RPV coolant inventory (e.g., high pressure and low pressure injection systems, offsite power, RHR, and EDGs). Level 1 scenarios that result in the loss of containment integrity prior to core damage usually do not include any additional opportunity to restore injection systems in the Level 2 analysis to prevent vessel failure (i.e., due to minimum time frames and severe secondary containment conditions).

- MAAP is not yet capable of modeling the response during containment flood sequences without some modeling intervention to mock-up plant features that MAAP can treat. Therefore, changes in the volumes of gas space available in the wetwell were attempted to allow the MAAP code to converge. These changes were unsuccessful and inferences from other published analyses were used to characterize the response.

- The DF of the suppression pool was modified for saturated pools to result in a factor of 10 decrease in particulate releases. This is considered consistent with existing scrubbing data from GE and Battelle. However, the current MAAP code assigns values of DF = 1.0 for saturated pool cases. A pool DF of 1.0 is considered unrealistic and this result has been modified in the results reported here. This change has not been implemented for cases in which suppression pool bypass occurs.

- The Nine Mile Point 2 containment is normally inerted; therefore hydrogen combustion is not a dominant contributor to the release However, it was assumed that hydrogen combustion occurs due to the presence of numerous electrical components whenever the core is damaged and the containment is not inerted and combustible gas control actions are not taken. Furthermore, it is assumed that hydrogen deflagration always produces a containment and secondary containment failure and a large release of radioactivity (i.e., no credit is taken for frequent periodic burning of small amounts of combustible gases to limit the pressure rise).

- The containment vent valves remain operational after containment venting.

- Representative sequences for MAAP evaluation are chosen conservatively, and a number of sequences are calculated to lead to similar release bins.

- Retention of debris in-vessel even after substantial core degradation has been included in the IPE assessment. This assessment has been included based on the time availability for adequate recovery, and the insights from NRC sponsored computer models, e.g., BWRSAR, MARCH, MELCOR.

- The Nine Mile Point 2 IPE treats phenomenological uncertainties through sensitivity studies performed with MAAP as well as using insights from other studies. Selection of the sensitivity runs are generally consistent with those given in the EPRI draft report "Recommended Sensitivity Analyses for an Individual Plant Examination using MAAP 3.0B."

- The containment response assessment and the evaluation of radionuclide release timing and magnitude is performed with NMP2 specific MAAP calculations.

- The Nine Mile Point 2 IPE considers the possible outcomes resulting from the potential of direct containment heating. This flexibility allows a baseline quantification and sensitivities for different accident management actions.

- The potential for hydrogen combustion in the reactor building has been assessed using the MAAP code. In addition, a probability that reactor building hydrogen deflagration occurs regardless of MAAP calculations has also been accounted for in the reactor building effectiveness node of the containment event tree.

- Decontamination factors for the secondary containment in the analyses consider the possibility of natural circulation and localized hydrogen burns causing loss of secondary containment building effectiveness.

- The Nine Mile Point 2 IPE includes a sufficient number of release categories to adequately account for the potential individual source terms, taking into account severity and timing.

4.2.4.2 Conservatisms

This subsection provides a list of known conservative assumptions that have been used in the Level 2 analysis:

- The containment failure curve is more limiting than calculated by ABB Impell on a plant specific basis for Nine Mile Point 2. This could result in slightly shorter times to containment failure than if the ABB Impell plant specific curve is used. In addition, there may be sequences that would be considered as resulting in "no containment failure" if the ABB Impell curve were used to assign a containment ultimate capability in this analysis. However, because a more conservative curve is used, containment failure would be assigned in this evaluation.

- Dynamic containment failures are postulated at a calculated bulk temperature of 260°F in the pool for ATWS (included explicitly in the analysis to determine failure location and timing).

- The mass of the DW equipment appears to be underestimated resulting in fewer heat sinks and shorter times to high drywell temperature, i.e., which affects both the containment failure timing and the revaporization source term contributions.

- Little credit is allowed for the reactor building DF. The reactor building DF is limited to no more than a factor of 10 and is determined by MAAP. It is also applied probabilistically such that for drywell head failure, no credit is given to the reactor building.

- No credit is allowed for lower release due to small containment failures.

- Drywell venting as part of the containment flood process is evaluated to include the coupled effects of RPV venting to the condenser and the drywell vent. As noted by the MAAP sensitivity cases in Section 4.9, the radionuclide release for each individual case is a medium (M) as long as the condenser provides reasonable retention of radionuclides. For situations in which the condenser is ineffective (noted by RM failure cases), the release is assumed to be a high (H). Because the deterministic containment flood modeling has been developed as a first of a kind model, conservative models have been used resulting in an over estimate of the radionuclide release.

- During RPV breach and RPV blowdown, when a calculated threshold gas velocity is exceeded within the pedestal region, core debris is entrained and transported to the drywell. MAAP does not entrain debris to the wetwell even though there are large openings via the in-pedestal downcomers. Therefore, the amount of debris entrained to the drywell is overestimated for NMP2. This adversely impacts long term heatup of the drywell region resulting in a decreased calculated time to containment failure.

- The radionuclide release for a given sequence may have releases which occur over a long period of time. For the bin scheme used in the Nine Mile Point 2 IPE, the releases of sequences are grouped such that the earliest time of release (even if that is only noble gas) is used to set the <u>time</u> of release (e.g., early or late). This conservative bin scheme may result in some overestimation of the releases associated with bins such as Early/High which have been referred to by the NRC as a "large" release.

4.2.4.3 Non-conservatisms

This subsection provides a list of potential non-conservative assumptions that could influence the Level 2 results.

- Residual debris remaining in the RPV could result in high DW temperatures. MAAP cases indicate that even when water injection is available to the RPV (via LPCI injection) that high drywell temperatures are possible. This is considered to be an analysis anomaly because the flow to the vessel would provide vessel cooling (currently not accounted for in the MAAP runs) and therefore reduce temperatures in the drywell. Additionally, the use of core spray or drywell spray would provide the desired drywell cooling.

Table 4.2-1

PHENOMENA DISCUSSED BY NRC AND IDCOR

ISSUE	NRC POSITION AND CURRENT STATUS	Nine Mile Point 2 IPE RESOLUTION
1. Fission product release prior to vessel failure	No substantial differences between NRC and industry models; no compelling evidence for volatile iodine release [Ref. 126]	MAAP model used
2. Recirculation of coolant in the RPV	Mostly an issue for PWR sequences at very high pressure (e.g., blackout) [Ref. 126]	N/A
3. Release models for control rod material	No substantial differences between NRC and industry positions; no compelling evidence that B_4C affects iodine chemistry or hydrogen production. [Ref. 126]	MAAP model used
4. Fission product and aerosol deposition in primary system and containment	NRC concerned that MAAP aerosol correlations might be inadequate when transport times are short, e.g., when early containment failure occurs. Subsequent EPRI and USDOE sponsored comparisons of the model to detailed methods indicate that model is suitable for IPE use. [Ref. 126, 127]	MAAP model used
5. In-Vessel Hydrogen Generation	NRC concerned that MAAP "blockage" model may seriously under-predict hydrogen production; EPRI is currently recommending that the model that may under predict H_2 not be used for base-case IPE calculations. [Ref. 126, 128]	H_2 production maximized in MAAP base calculations; sensitivity performed
6. Core melt progression and vessel failure	NRC and IDCOR agreed that the mass of molten material available at vessel failure was uncertain and sensitivities to this quantity should be investigated when calculations are performed. [Ref. 126]	Sensitivity performed
7. In-vessel steam explosions leading to Alpha mode failure of containment	Steam Explosion Review Group as well as Industry experts subscribe to the view that steam explosions sufficient to fail containment do not contribute significantly to risk; this is consistent with NUREG-1150; issue considered largely resolved for IPEs. [Ref. 126]	Probabilistically treated
8. Direct containment heating	Considered primarily a PWR issue.	Probabilistically treated
9. Ex-vessel fission product release	MAAP model improved to include more chemical species; not likely to be major issue for IPEs. [Ref. 126]	MAAP model used

Table 4.2-1

PHENOMENA DISCUSSED BY NRC AND IDCOR

ISSUE		NRC POSITION AND CURRENT STATUS	Nine Mile Point 2 IPE RESOLUTION
10.	Ex-vessel heat transfer models from molten core to concrete	While uncertainties exist, MAAP model compares relatively well to experiment; unlikely to be major issue for IPEs [Ref. 126]	MAAP model used for base calc; sensitivities performed.
11.	Revaporization of deposited fission products	IDCOR and NRC agree that MAAP uncertainty calculations should be performed to treat possibility of chemical reactions between volatile fission products and steel surfaces. [Ref. 126]	MAAP model used
13a.	Amount and timing of suppression pool bypass	Mainly an issue for Mark IIII; use of Vaughan model to assess plugging of leakage path by aerosols (as in MAAP) acceptable for flowpaths less than 1 cm wide. [Ref. 126]	N/A
13b.	Retention of fission products in ice beds	PWR ice condenser issue. [Ref. 126]	N/A
14.	Modeling of emergency response	Not an issue for IPEs; issue resolved if analyst assumes that a fraction (e.g., 5 percent) of the population does not evacuate. [Ref. 126]	N/A
15.	Containment performance	IDCOR and NRC agreed that a spectrum of failure sizes should be considered to address spectrum of failure pressures be considered; recent EPRI report lends credence to IDCOR leak-before-break assumption. [Ref. 126]	Spectrum of Failures included in both MAAP assessment and probabilistically in CET evaluation.
16.	Secondary containment performance	In response to NRC concerns, MAAP model was made much more detailed. While NRC concerns focused on dependence of aerosol residence time, hydrogen burns in the secondary containment, and the rate of concrete off-gas production in containment, in most plants it appears that fission product retention will mainly depend on failure modes of secondary containment and scrubbing. [Ref. 126]	MAAP model and probabilistic assessment used
17.	Hydrogen Ignition and Burning	Not an issue for Mark IIs. NRC concerned with MAAP models for global burns and burns at igniters. MAAP models were significantly updated to address NRC concerns; not likely to be a major issue for IPEs. [Ref. 126]	N/A

4.3 BINS AND PLANT DAMAGE STATES

The interface between the Level 1 and Level 2 analyses is important to ensure that the information and data from the Level 1 analysis (e.g., dependencies) are properly transferred and interpreted in the Level 2 CET.

In some PRA analyses (e.g., WASH-1400 [Ref. 89], NUREG-1150 [Ref. 122]), the coupling of the front-end analysis to the back-end is through the binning of the multitude of front-end sequences into groups of plant damage states with similar back-end characteristics. For such analyses, it is important that the bins be justified on the basis of such factors as timing of important events or operability of key features.

The Nine Mile Point 2 assessment involves the direct coupling of <u>each</u> sequence from the Level 1 to the CET evaluation. Specifically, the Nine Mile Point 2 IPE directly links the front-end to back-end portions of severe accident sequences through directly linked event trees. These trees ensure that the support state conditions and other dependencies are properly accounted for throughout the front-end and back-end trees.

Therefore, the Level 1 end state bins are reduced in importance, but still have a valuable use as a summary point. In addition, they are used in the probabilistic assessment in the Level 2 CETs to simplify rule writing and split fraction assignment.

The CETs and their direct linking to each Level 1 sequence include preventive or mitigative features as well as timing considerations. Three different containment event tree structures, each linked to the appropriate front-end event sequence, are used to properly handle the various combination of accident sequences that may occur (e.g., containment failure before core damage cases as well as vice-versa and containment bypass sequences).

4.3.1 Input to CET: Interface Between Accident Sequence Classes (Level 1 End States and Containment Challenges)

Binning may have three purposes:

- First, as a necessity in order to perform the probabilistic evaluation (e.g., WASH-1400) by transferring information from the Level 1 to Level 2 analysis;

- Second, as a method of allowing discrete evaluations using deterministic codes;

- Third, as a method to display Level 1 results.

The Nine Mile Point 2 evaluation has used the bin scheme for the second and third purposes.

Three approaches are available for transferring the appropriate Level 1 information to the Level 2 assessment. These three approaches are the following:

APPROACH	DESCRIPTION
1	Use a functional failure description of the Level 1 plant damage states. Additional information on individual system status can be inferred from the dominant accident sequences
2	Use a sophisticated all encompassing plant damage state matrix to identify each sequence.
3	Use RISKMAN capability to link the appropriate CET for <u>each</u> Level 1 accident sequence.

The method chosen for the Nine Mile Point 2 Level 2 analysis is Approach 3. This method allows the appropriate containment event tree model to be coupled directly with the Level 1 event trees and directly account for containment system dependencies with each sequence end state using rules and split fractions that define the Level 1 accident scenario and system status.

Because each Level 1 accident sequences is explicitly transferred into the Level 2 containment event tree and is then evaluated within the Level 2 CET, there is no "binning" <u>required</u> at the end of Level 1. This approach avoids the problems that have arisen in other analysis in which representative sequences or bins are used as entry states into the CET. The Nine Mile Point 2 approach of explicitly transferring all Level 1 sequences into the Level 2 CET allows the analysis to proceed on a sequence-by-sequence basis with binning occurring only at the end of the Level 2 evaluation.

While this choice precludes the necessity of formally defining functionally related plant damage states, it is useful for the purposes of display and for assuring completeness of the CET derivation (i.e., requires the analyst to examine the functional basis of accident sequence types and portray Level 1 results in terms of specific plant damage states).

Each of these aspects are discussed in this section:

- The Level 1 results have been chosen to be usefully displayed in functional groupings having similar challenges to containment and operator response.
- The deterministic calculations used to calculate pressure and temperature responses of containment are also conveniently characterized as related to these types of challenges.
- Finally, the Level 1 end states are used for simplification in writing rules that are used in the CET to process each Level 1 sequence.

The formulation of a Probabilistic Risk Assessment (PRA) has three general phases:

- Systems evaluation (Level 1)
- Containment response and source term evaluation (Level 2)

- Ex-plant public risk evaluation (Level 3)

The "systems" evaluation (i.e., Level 1 IPE) involves the assessment of those scenarios that could lead to core damage. The subsequent treatment of mitigative actions and the inter-relationship of the plant systems with the containment after core damage is then treated in the Containment Event Tree (i.e., Level 2 IPE).

An offsite consequence analysis can also be performed within the PRA structure (i.e., Level 3) to determine the impact of severe accidents on public safety. This portion of a PRA is not required as part of the IPE process and is currently not included in the Fermi IPE.

Figure 4.3-1 shows the three phases of a PRA and the relationship of the IPE-required reporting criteria for each of the Level 1 and Level 2 results.

In the NMP2 Level 1 PRA, a broad spectrum of accident sequences have been postulated that could lead to core damage and potentially challenge containment. The NMP2 Level 1 PRA has calculated the frequency of those accident sequences that contribute to the core damage frequency using system oriented (systemic) event trees. Each of these sequences may result in a challenge to containment. However, many of these challenges to containment have similarities in their functional failure characteristics. This observation has been confirmed in individual BWR PRAs [Ref. 87 through 93], including NUREG-1150. The result is that these studies have categorized these containment challenges into a finite, discrete group of accident sequence bins which have similar functional failures. The definition of the functional accident sequences are derived to describe the relationship between specific accident sequences and BWR critical safety functions and accomplish this information transfer to the Level 2 portion of the IPE.

As pointed out in past BWR PRAs [Ref. 87, 88], different portions of the spectrum of postulated core damage accidents pose substantially different challenges to the containment depending upon the system failures and phenomena that have contributed to the sequence. Therefore, the containment event tree response must be capable of reflecting the entire spectrum of challenges to ensure that the following are explicitly incorporated:

- System failures in the Level 1 evaluation (including support systems)
- Phenomenological interaction due to the type of core melt progression
- RPV conditions
- Containment conditions
- Timing of the sequence of events (i.e., core damage and containment failure (if applicable).

4.3.2 Level 1 PRA End States Classification Scheme

A plant damage state classification into five accident sequence functional classes can be performed using the functional events as a basis for selection of end states. The description of functional classes is presented here to introduce the terminology to be used in characterizing the basic types of challenges to containment. The reactor pressure vessel condition and containment condition for each of these classes at the time of initial core damage is noted below:

Core Damage Functional Class	RPV Condition	Containment Condition
I	Loss of effective coolant inventory (includes high and low pressure inventory losses)	Intact
II	Loss of effective containment pressure control, e.g., heat removal	Breached or Intact
III	LOCA with loss of effective coolant inventory makeup	Intact
IV	Failure of effective reactivity control	Breached or Intact
V	LOCA outside containment	Breached (bypassed)

In assessing the ability of the containment and other plant systems to prevent or mitigate radionuclide release, it is desirable to further subdivide these general functional categories. In the second level binning process, the similar accident sequences grouped within each accident functional class are further discriminated into subclasses such that the potential for system recovery can be modeled. The interdependencies that exist between plant system operation and the core melt and radionuclide release phenomena are represented in the release frequencies through the binning process involving these subclasses, as shown in past PRAs and PRA reviews. The binning process, which consolidates information from the systems' evaluation of accident sequences leading to core damage in preparation for transfer to the containment-source term evaluation, involves the identification of 12 classes and subclasses of accident sequence types. Table 4.3-3 provides a description of these subclasses that are used to summarize the Level 1 PRA results.

Published BWR PRAs have identified that there may be a spectrum of potential contributors to core melt or containment challenge that can arise for a variety of reasons. In addition, sufficient analysis has been done to indicate that the frequencies of these sequences are highly uncertain; and therefore, the degree of importance on an absolute scale <u>and</u> relative to each other, depends upon the plant specific features, assumptions, training, equipment response, and other items that have limited modeling sophistication.

This uncertainty means that the analyst can neither dismiss portions of the spectrum from consideration nor emphasize a portion of the spectrum to the exclusion of other sequence types. This is particularly true when trying to assess the benefits and competing risks associated with a modification of a plant feature.

This end state characterization of the Level 1 PRA in terms of accident subclasses is usually sufficient to characterize the CET entry states for most purposes. However, when additional refinement is required in the CET quantification, it may be useful to further discriminate among the contributors to the core damage accident classes. This discrimination can be performed through the use of the individual accident sequence characteristics.

For NMP2, functional based plant damage states are used to summarize Level 1 results and to ensure that the Level 2 CETs are sufficient to allow each functional sequence to be addressed.

4.3.3 Summary of Specific Aspects of the Level 1 - Level 2 Interface

This subsection provides a brief summary of particular aspects of the interface that are useful to highlight.

- Equipment failures in Level 1: Equipment failures that have been assessed in Level 1 are carried by the computer into the Level 2 analysis. Therefore, failed equipment cannot be used in the Level 2 assessment unless an explicit evaluation has been performed as part of the Level 2 to support repair or recovery. This would include consideration of adverse environments where appropriate. This includes support systems, accident prevention systems, and mitigation systems.

- Human errors: There is a check performed on all sequences to ensure that Level 1 sequences that result from human errors have only those recoveries that can be justified as consistent with operating staff recoveries given human failures in the Level 1 analysis.

- RPV status: The RPV pressure condition is explicitly transferred from the Level 1 analysis to the CET.

- Containment status: The containment status is explicitly transferred from the Level 1 analysis to the CET. This includes recognition of whether the containment has previously failed, is intact, or is at elevated pressure conditions.

- Containment isolation: All support system dependencies are transferred as part of the individual Level 1 sequences such that the containment isolation evaluation is performed on a sequence-by-sequence basis.

- Differences in accident sequence timing are also transferred with the Level 1 sequences. These timings affect such sequences as:

  - station blackout
  - loss of containment heat removal
  - ATWS
  - vapor suppression failure.

  This allows the timing to be properly assessed in the Level 2 CET.

- Thermal hydraulic deterministic assessments:

  The use of deterministic codes in the characterization of accident sequences has been discretized in a manner similar to the functional sequence binning classification scheme identified above. These functional sequences then have various additional failures applied to determine containment response for various postulated scenarios through the CET. Variations in timing and assumptions regarding subtle sequence variations have been explicitly calculated to ensure that the sequence representations using the thermal-hydraulic code is representative.

- Dual Usage: Because the Level 1 and Level 2 models are directly coupled on a sequence basis the accountability of common water sources or common power sources falls out of the combined sequence analysis when it is run from initiating event to release point.

- Mission Times: The mission times for the entire sequence from initiating event to release point are considered.

- Timing of Recovery: Equipment or power recovery is accounted for at various phases in the Level 1 and 2 analyses. Each sequence includes a consistent recovery model to ensure no double counting.

4.3.4 Level 2 End States/Bins

In a manner similar to the Level 1 evaluation, there are reasons to also bin or group end states of the Level 2 evaluation into similar groups. Because of the emphasis of the IPE on assessing possible accident management procedural or hardware modifications, it has been decided that an adequate binning scheme for the Level 2 results wold encompass both of the following:

- Radionuclide release magnitude.

- Radionuclide release timing.

Section 4.7 summarizes the derivation of the Level 2 end state bins. Table 4.3-4 provides the results of the Level 2 end state bin derivation.

Table 4.3-3

SUMMARY OF THE CORE DAMAGE ACCIDENT SEQUENCE SUBCLASSES

Accident Class Designator	Subclass	Definition	WASH-1400 Designator Example
Class I	A	Accident sequences involving loss of inventory makeup in which the reactor pressure remains high.	TQUX
	B	Accident sequences involving a station blackout and loss of coolant inventory makeup.	T_EQUV
	C	Accident sequences involving a loss of coolant inventory induced by an ATWS sequence with containment intact.	T_TC_MQU
	D	Accident sequences involving a loss of coolant inventory makeup in which reactor pressure has been successfully reduced to 200 psi.; i.e., accident sequences initiated by common mode failures disabling multiple systems (ECCS) leading to loss of coolant inventory makeup.	TQUV
Class II	A	Accident sequences involving a loss of containment heat removal with the RPV initially intact; core damage induced post containment failure	TW
	L	Accident sequences involving a loss of containment heat removal with the RPV breached but no initial core damage; core damage after containment failure.	AW
	T	Accident sequences involving a loss of containment heat removal with the RPV initially intact; core damage induced post high containment pressure	N/A
	V	Class IIA or IL except that the vent operates as designed; loss of makeup occurs at some time following vent initiation. Suppression pool saturated but intact.	TW
Class III (LOCA)	A	Accident sequences leading to core damage conditions initiated by vessel rupture where the containment integrity is not breached in the initial time phase of the accident.	R
	B	Accident sequences initiated or resulting in small or medium LOCAs for which the reactor cannot be depressurized prior to core damage occurring.	S_1QUX
	C	Accident sequences initiated or resulting in medium or large LOCAs for which the reactor is a low pressure and no effective injection is available.	AV
	D	Accident sequences which are initiated by a LOCA or RPV failure and for which the vapor suppression system is inadequate, challenging the containment integrity with subsequent failure of makeup systems.	AD

Table 4.3-3

SUMMARY OF THE CORE DAMAGE ACCIDENT SEQUENCE SUBCLASSES

Accident Class Designator	Subclass	Definition	WASH-1400 Designator Example
Class IV (ATWS)	A	Accident sequences involving failure of adequate shutdown reactivity with the RPV initially intact; core damage induced post containment failure.	$T_T C_M C_2$
	L	Accident sequences involving a failure of adequate shutdown reactivity with the RPV initially breached (e.g., LOCA or SORV); core damage induced post containment failure.	N/A
	T	Accident sequences involving a failure of adequate shutdown reactivity with the RPV initially intact; core damage induced post high containment pressure.	N/A
	V	Class IV A or L except that the vent operates as designed; loss of makeup occurs at some time following vent initiation. Suppression pool saturated but intact.	N/A
Class V	--	Unisolated LOCA outside containment	N/A

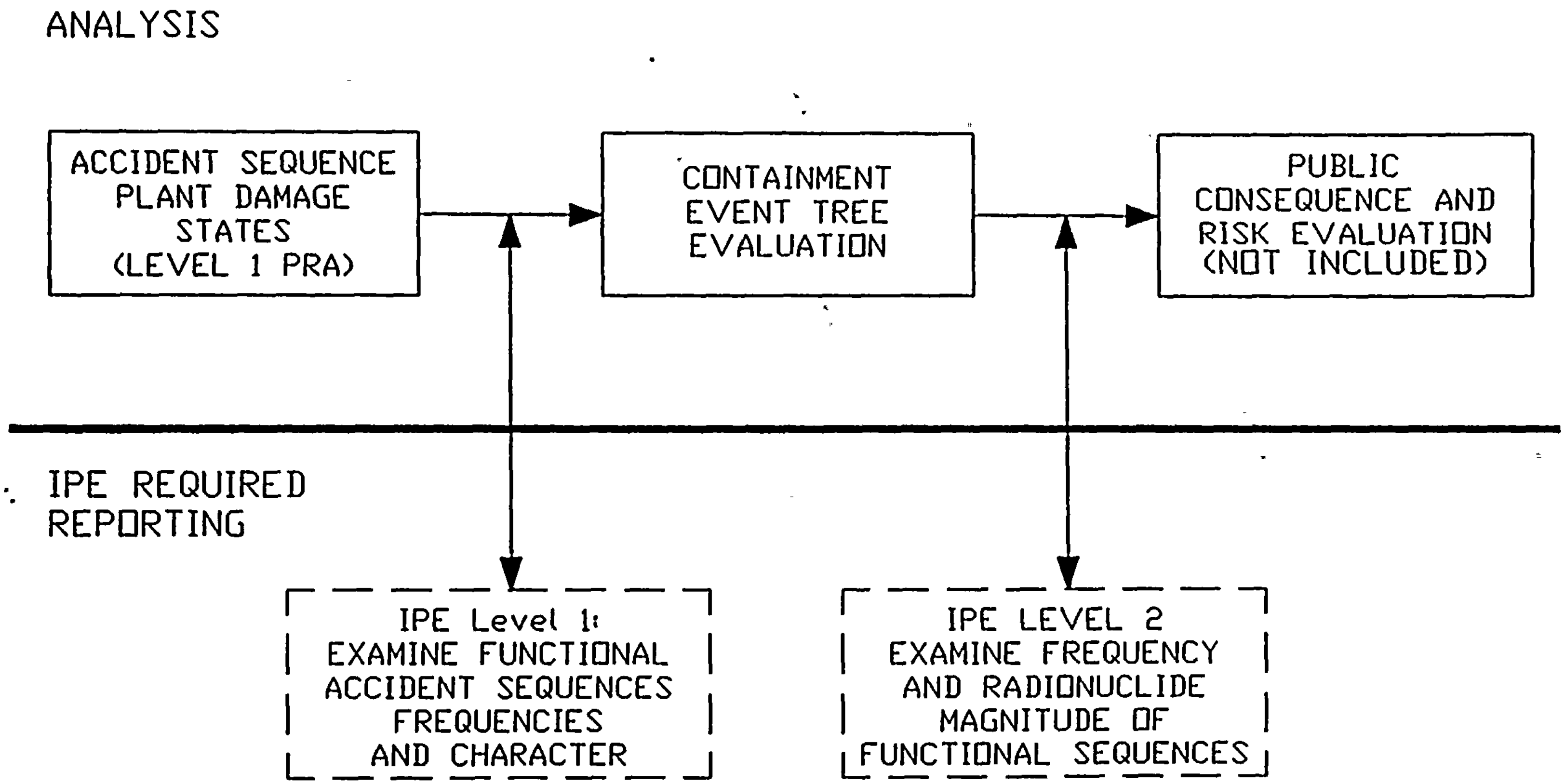


Figure 4.3-1 IPE Required Reporting Criteria and Event Tree Analysis Interrelationship

Table 4.3-4

LEVEL 2 ENDSTATE BINS:

RADIONUCLIDE RELEASE SEVERITY & TIMING CLASSIFICATION SCHEME
(SEVERITY, TIMING)

Release Severity		Release Timing	
Classification Category	Cs Iodide % in Release	Classification Category	Time of Initial Release
High (H)	greater than 10	Late (L)	greater than 24 hours
Moderate (M)	1 to 10	Intermediate (I)	6 to 24 hours
Low (L)	0.1 to 1	Early (E)	less than 6 hours
Low-low (LL)	less than 0.1		
No iodine (OK)	0		

Thirteen (13) Level 2 Endstate Bins

H/E
H/I
H/L
M/E
M/I
M/L
L/E
L/I
L/L
LL/E
LL/I
LL/L
OK

4.4 CONTAINMENT FAILURE CHARACTERIZATION

A knowledge of the pressure and temperature capability of the containment, as well as the probable location and size of a containment failure, is fundamental in determining the timing and magnitude of a potential radionuclide release under postulated severe accident conditions.

The Level 2 analysis is strongly influenced by the containment failure modes and their timing. Therefore, this subsection includes an assessment of:

- Primary containment failure modes (Section 4.4.2)
  - pressure and temperature dependent
  - dynamic failure modes
  - phenomenological induced failures
- Containment vent induced failure modes (Section 4.4.3)
- Primary containment ultimate capability (Section 4.4.4)
  - low temperatures
  - moderate temperatures
  - high temperatures
  - dynamic load induced
- Summary of containment failure assessment (Section 4.4.5)
- Secondary containment failure modes (Section 4.4.6)

4.4.1 Nine Mile Point 2 Mark II Containment

Although the frequency of core damage events is very low, severe accidents may present a threat to the integrity of the containment. Primary containment (see Figure 4.4-1) is one of the boundaries preventing the release of radionuclide fission products to the environment. Therefore, to assess accident management actions that could be implemented as part of contingency planning for severe accidents, containment response under severe accident conditions must be considered.

Section 4.1 has briefly described the key features of the containment. This subsection discusses postulated containment failure modes along with the ultimate capability of containment to withstand these failure modes.

4.4.2 Postulated Containment Challenges/Failure Modes

To determine the effectiveness of the requirements in assuring adequate containment performance, a systematic review of the containment challenges associated with a spectrum of severe accident types has been assembled. Radionuclide releases are associated with a containment failure (location and size) and accident sequence.

NMP2 specific MAAP evaluations are used to determine the release magnitude and timing for the various release categories corresponding to some of the postulated containment failure modes (e.g., slow containment overpressure).

The containment capability is combined with the deterministic MAAP calculations described in Sections 4.2 and 4.7 to determine the timing and location of many of the containment failure modes. Figure 4.4-2 is a simplified information flow chart showing the combination of MAAP deterministic results for one sequence overlaid on the containment capability curve determined by the containment structural analysis. This comparison of calculated deterministic results with the capability curve is used to identify the time it takes for containment conditions to degenerate to the point at which the containment integrity may be jeopardized.

However, as will be seen in this section, not all postulated containment failure modes are easily calculated with existing deterministic codes. For such cases, separate effects analyses or engineering judgement is used to provide the magnitude of the threat and its timing. MAAP can still be used to bracket the general time frame by characterizing when the necessary and sufficient conditions are present (e.g., for when steam explosions might be possible).

One of the basic reasons for focusing on containment failure mode and timing is that it can immediately make obvious the type of response that can either mitigate or reduce the containment failure probability. Accident management actions will be dependent primarily on the containment failure assessment and supplemented by the radionuclide release magnitude.

Table 4.4-1 presents the various functions associated with plant response to accident and transient conditions which either preclude expected challenges, or allow the containment to accommodate challenges.

Table 4.4-1 IMPORTANT FUNCTIONS FOR PREVENTION AND ACCOMMODATION OF CONTAINMENT CHALLENGES
• Reactivity Control • Reactor Pressure Control • Fuel/Debris Coolant Inventory Control • Containment Pressure/Temperature Control • Combustible Gas Control • Containment Isolation • Vapor Suppression • Containment Structural Capability for External Loading

Associated with combinations of success or failure of each of these functions during transient or accident conditions are potential challenges to the integrity of the fuel, reactor coolant system and containment. These functions are assessed probabilistically and deterministically in the containment event tree analysis.

Table 4.4-2 identifies either postulated containment challenges or the corresponding failure modes that have previously been identified in severe accident or Design Basis Accident (DBA) analyses. These challenges/failure modes span a range from those historically considered by regulations to those beyond traditional design bases, including severe accident conditions. To ensure that a comprehensive list of challenges is investigated, the important containment functions listed in Table 4.4-1 were reviewed and an assessment made of their impact on containment integrity. In addition, challenges/failure modes were selected to encompass the following:

- initiating events which by definition result in bypass of containment,

- random system or equipment failures which could lead to breach of the containment boundary independent of any severe accident challenges, and

- potential dependent failures that could be caused by phenomena which challenge the structural integrity of containment as a result of the accident. Challenges to containment integrity as identified in the General Design Criteria of the Standard Review Plan are incorporated into the study. The list also covers those postulated accident initiators or phenomena that have been identified as important in past industry and NRC studies, such as the Industry Degraded Core Rulemaking (IDCOR) program, NUREG-1150, and various BWR probabilistic risk studies. Also cited are the failure modes provided in the PRA Procedures Guide regarding potential containment failure modes. This last item was explicitly suggested in the IPE Generic Letter 88-20.

Table 4.4-3 summarizes the disposition of these failure modes in the Nine Mile Point 2 Level 1 and 2 assessments after extensive evaluations of the Nine Mile Point 2 containment, the severe accident spectrum, and current published information,

The containment failure modes have specific characteristics that allow them to be associated with the critical parameters governing radionuclide release determination. Three of these critical parameters are:

- Time of containment failure
- Size of containment failure
- Location of containment failure.

Table 4.4-4 summarizes the general relationships of these three characteristics considered in the Level 2 analysis to represent the NMP2 applicable failure modes from Table 4.4-3. Note that multiple failure sizes and locations are possible for many of the failure modes. These relationships are better defined by MAAP analyses (see Section 4.7) and are used to construct the containment event trees and the functional fault tree for each node.

In addition to identified containment failure modes, there are also a number of related phenomenological issues. The IDCOR regulatory interaction program was devoted to the definition and resolution of open technical issues related to the assessment of severe accidents.

These are discussed in Section 4.2. A consideration of each of these issues has been included in the development of the containment capability curves and the CET nodal evaluations. Specific sensitivity studies have been performed on selected issues, (see Section 4.9).

4.4.3 Containment Vent Induced Failure Modes

Containment venting represents a controlled radionuclide release pathway that can be used as an accident mitigation measure to prevent other uncontrolled failure modes or assist in maintaining core/debris cooling. For both purposes, venting is an alternative to an uncontrolled release.

Containment venting is modeled as a hard-piped release pathway from either the wetwell or the drywell directly to the environment. No adverse reactor building harsh environment (except for local radiation shine) is expected to occur during the venting process. Therefore, the modeling of containment venting is quite straightforward. It involves no reactor building decontamination factor (DF), and as stated above, no ill-effects on reactor building equipment is modeled.

4.4.4 Containment Ultimate Capability

The primary containment ultimate structural integrity is important in severe accident analysis due to its key role as a fission product barrier. Section 4.4.2 identified in tabular form the individual containment failure modes identified from the literature.

For the purposes of this analysis, the entire spectrum of accident conditions that could challenge containment integrity can be categorized into four regimes. The following four regimes are used to determine those areas where containment ultimate capability is assessed against the postulated containment failure modes:

1) Pressure Induced Containment Challenge: Containment pressures may increase from normal operating pressure along a saturation curve to very high pressures (i.e., beyond 100 psi), during accidents involving:

   - Insufficient long term decay heat removal; and
   - inadequate reactivity control and consequential inadequate containment heat removal.

2) Temperature Induced Containment Challenge: Containment temperatures can rise without substantial pressure increases if containment pressure control measures (e.g., venting) are available, but debris temperature control is inadequate. In such cases, containment temperature at less than design pressure may occur during accidents involving core melt progression.

3) Combined Pressure and Temperature Induced Containment Challenge: Containment pressures and temperatures can both rise during a severe accident due to molten debris effects following RPV failure and subsequent core concrete interaction. For instance:

   - Drywell temperatures can rise from approximately 300°F at core melt initiation to above 1000°F in time frames on the order of 10 hours.

- Pressure can rise due to non-condensible gas generation and RPV blowdown in the range of 40 psig to 100 psig over this same time frame.

4) Dynamic Loads:
In addition to these "steady state" challenges, failure modes associated with dynamic loading resulting from high steam flow to a saturated pool or from energetic phenomena (e.g., steam explosions) are also postulated.

It is clear from analyses of the severe accident challenges to containment, that the containment response and capability both vary substantially over a spectrum of possible challenges in terms of temperature and pressure. Therefore, the definition of adequate containment performance proposed explicitly considers these regimes. The following subsection addresses the containment ultimate capability for the four regimes.

A probabilistic evaluation of the Nine Mile Point Nuclear Station Unit 2 containment performance has been conducted by ABB-Impell [Ref. 121]. Potential failure modes (i.e., structural capacity and seal degradation) of the containment structure and penetrations due to temperature and pressure conditions well beyond its design basis were considered in this evaluation. Based on this information and supplemental generic information concerning seal and structural material performance at extremely high temperature and under dynamic loadings, the criteria for describing the NMP2 containment performance during severe accidents have been developed.

Figure 4.4-1 shows the Nine Mile Point Unit 2 primary containment relative to various rooms and compartments in the reactor building.

Figure 4.4-2 provides a simplified schematic showing the process of combining the calculated ultimate plant containment capability established in this section with the deterministic containment conditions during postulated severe accidents. Combining the two curves results in the assessment of the time of containment failure, i.e., the time when the containment internal conditions violate the assessed ultimate plant capability.

Features of the NMP2 Mark II containment that were investigated include the concrete containment structure (drywell, drywell head, and wetwell), liner, containment hatches, hatch seals, penetrations and isolation valves.

Before examining the containment failure modes in detail, it is useful to describe importance of the containment failure size and how it is to be characterized in the NMP2 Level 2 model.

Containment Failure Size

Because the containment failure size and location are influential in quantifying the radionuclides released to the reactor building and subsequently to the environment, the size and location are important features to identify in the analysis and include in the probabilistic CET evaluation. The failure size can be divided into three size ranges: negligible, small and large. In order to define more quantitatively the terms negligible, small, and large containment failures, Figure 4.4-3 is provided to show the spectrum of sizes included in each category and the point estimate used to model the size.

The discrete failure sizes used in the accident modeling and shown in Figure 4.4-3 are determined considering the following issues:

- Containment leak size versus leak rate.
- Leak rate versus impact on risk.
- Heuristic classifications of rupture sizes from other studies.

These issues are discussed below as they pertain to each modeled failure size.

Negligible Leak

Stone and Webster evaluated the effect of containment leak size on the containment leak rate [Ref. 129]. A sampling of some of the results is shown in the table below:

Table 4.4-5		
Containment Leak Size (in.)		Approximate Containment Leak Rate at Design Pressure
DIA (in.)	Area (in^2)	(wt%/day)
.25	.05	.5
.34	.09	1.0
.50	.2	2.4
1.25	1.2	14.4
2.00	3.1	31.0
3.4 (Estimated)	9.1	100.0 (Estimated)

In addition, ORNL [Ref. 130] evaluated the impact of leak rates on public risk. The study uses information from WASH-1400 as the basis for its risk sensitivity calculations and finds that Figure 4.4-4 reflects the fractional impact on risk associated with containment leak rates. ORNL concluded that the impact of "leakage" rates on LWR accident risks is indicated to be relatively small.

Based upon the information in Table 4.4-5 and Figure 4.4-4, it is judged that small leaks (i.e., those which are not judged to dominate the risk to the public) can be defined to be those that modify risk by less than 5%. Such a definition would include leaks of less than 35%/day. Based on Table 4.4-5, a 35%/day containment leak rate equates to an equivalent diameter leak of slightly greater than 2 inches. Therefore, the probability of having a leak size with equivalent diameter less than 2 inches is the basis for describing the "negligible" leakage regime.

Large Containment Failure

For NUREG-1150, while the question of containment failure location is largely specific to the individual containment geometries, the approach to characterizing potential failure sizes was more generic. Three possible failure sizes were distinguished: leak, rupture, and catastrophic rupture. Working quantitative definitions of each failure size were based on the thermal-hydraulic evaluation of containment depressurization times.

- A leak was defined as a containment breach that would arrest a gradual pressure buildup but would not result in containment depressurization in less than 2 hours. The typical leak size was evaluated for all plants to be on the order of 0.1 $ft.^2$.

- A rupture was defined as a containment breach that would arrest a gradual pressure buildup and would depressurize the containment within 2 hours. For all plants, a rupture was evaluated to correspond to a hole size in excess of approximately 1.0 $ft.^2$.

- A catastrophic rupture was defined as the loss of a substantial portion of the containment boundary with possible disruption of the piping systems that penetrate or are attached to the containment wall.

In contrast to NUREG-1150, earlier IDCOR studies classified large rupture as approximately 5 $ft.^2$. The failure size for large breaks used in the NMP2 analysis (2 $ft.^2$) was estimated to be between these values with a tendency more towards the NUREG-1150 estimate. This size is sufficiently large to rapidly depressurize the containment.

Small Containment Failure Size

The failure size for small containment failures was logically estimated to be between the 2" diameter point that denotes the upper bound of the negligible leakage spectrum and the 1.0 $ft.^2$ point (NUREG-1150 estimate) that denotes the lower bound of the large failure spectrum. The 27 $in.^2$ point estimate used in the model was selected because it is approximately the size at which the primary containment will slowly depressurize for most accidents.

Types of Containment Failure Modes

The containment failure modes identified by ABB Impell in the analysis are the following:

- Wall/Basemat Junction Failure
- Wetwell Liner Tearing
- Wall/Basemat Junction Radial Shear
- Hoop Membrane @ Elev. 324'
- Wetwell Hoop Membrane
- Drywell Flexure/Tension @ Elev. 240'
- Radial Shear @ Elev. 324'
- Basemat Shear

It should be noted that of the top 11 NMP2 containment failure modes identified by ABB Impell for low temperature challenges, all but one were structural failures which are assumed here to result in a large containment failure. The one failure that is classified as a small failure is that associated with liner tearing in the wetwell airspace. ABB Impell also evaluated the possibility of seal and penetration failures and found that any seal leakage failures would occur at much higher pressures and therefore were not contributing to the containment failure probability at low temperatures.

The remainder of this subsection provides a summary of the containment performance for the following regimes of challenge:

- Capability at low temperature (below 400°F)
- Capability at intermediate temperature (between 400°F and 900°F)
- Capability at high temperature (above 900°F)
- Capability for high suppression pool temperature <u>and</u> high SRV discharge flow rates
- Capability of hatches and penetrations at elevated temperatures.

4.4.4.1 Containment Temperature less than 400°F (Structural Capability Only)

Table 3-1 of reference [Ref. 121] presents the median containment structural pressure capacities associated with 11 non-exclusive failure modes for the situation where the containment interior temperature is less than 400°F. The probability of any one failure mode resulting in containment breach includes: (1) the probability of only that failure mode occurring; and (2) the probability of that mode occurring simultaneously with one or more other modes. Based on the distribution parameters given in Reference [Ref. 121] a log normal probability function can be derived to predict the cumulative probability of each failure mode as a function of pressure. Additionally, the total probability of one or more of these failures modes occurring at any quasi-static pressure condition can be adequately described by the following expression:

$$P_{TOTAL} = 1 - (P_1 * P_2 * P_3 * P_4 * P_5 * P_6 * P_7 * P_8 * P_9 * P_{10} * P_{11}) \qquad (4\text{-}1)$$

Where P_i = Probability of Failure mode i

$$P_{TOTAL} = 1 - \prod (P_i)$$

Table 4.4-6 summarizes the following:

- The assessed cumulative probability of each individual failure mode occurring if the containment were exposed to pressures ranging from 120 to 220 psig, and
- The total cumulative probability of any one or more of these failure modes occurring at each pressure interval. This is calculated using equation (4-1).

Figure 4.4-5 summarizes the cumulative failure probabilities for each failure mode and the calculated total cumulative failure probability if all failure modes are considered. Based on inspection of the total failure probability distribution, the median failure probability is approximately 141 psig. Using the ABB Impell assessment that a log normal function describes these results, and that the distribution has an error factor of approximately 1.22, the mean containment failure pressure is calculated to also be 141 psig.

The dominant failure modes presented in Table 4.4-6 are subsequently combined into five categories based on the affected containment location and severity of the structural breach. The relative probability of these five categories is approximated by comparing the contribution of each constituent failure mode probability to the sum of their total. No leakage at closures is predicted by ABB Impell for these cases. Therefore, for containment temperatures below 400°F; the conditional containment failure probabilities used in the NMP2 Level 2 evaluation is as follows:

NMP2 Containment Failure Location and Size: Below 400°F		
Location	Containment Breach	Relative Failure Probability
Drywell	Rupture	0.28
Wetwell below water line	Rupture	0.40
Wetwell above water line	Rupture	0.06
Drywell	Leakage	<< 0.01 (Estimate)
Wetwell above water line	Leakage	0.26

4.4.4.2 Containment Temperature Approximately 600°F (Structural Capability Only)

Similarly, the results of the analysis regarding structural capacity of the containment at drywell temperatures of approximately 600°F have been developed using the ABB Impell containment evaluation.

Table 4.4-7 summarizes the following:

1) The assessed cumulative probability of each failure mode occurring if the containment were exposed to pressures ranging from 120 to 220 psig. It is important to recognize that a combination of Tables 3-1 (wetwell) and 3-2 (drywell) from Reference [Ref. 121] is used to represent the containment temperature profile corresponding to the postulated severe accident conditions. This is because the wetwell is assumed to remain at 400°F even though the drywell will be exposed to 600°F temperatures.

2) The total cumulative probability of any one or more of these failure modes occurring at these pressures.

Figure 4.4-6 summarizes the cumulative failure probabilities for each failure mode and the calculated total cumulative failure probability if all failure modes are considered. Using the ABB Impell assessment that a log normal function describes the resulting distribution, the mean containment failure pressure is calculated to equal 138 psig.

The assessment for this portion of the analysis only considers the structural failures; leakage at closures is not included at this point in the analysis (see Sections 4.4.4.5 and .6).

The relative probabilities of the five containment failure categories for temperatures of 600°F in the drywell and 400°F in the wetwell, are presented in the following table:

NMP2 Containment Failure Location and Size: Approximately 600°F in DW		
Location	Containment Breach	Relative Failure Probability
Drywell	Rupture	0.32
Wetwell below water line	Rupture	0.38
Wetwell above water line	Rupture	0.06
Drywell	Leakage	<< 0.01 (Estimate)
Wetwell above water line	Leakage	0.24

4.4.4.3 Containment Temperature of Approximately 800°F (Structural Capability Only)

Similarly, the results of the analysis regarding structural capacity of the containment at a drywell temperature of approximately 800°F can be developed.

Table 4.4-8 summarizes the following:

- The assessed cumulative probability of each failure mode occurring if the containment were exposed to pressures ranging from 120 to 220 psig. It is important to recognize that a combination of Tables 3-1 (wetwell) and 3-2 (drywell) from Reference [Ref. 121] is used to represent the containment temperature profile corresponding to the postulated severe accident conditions. This is because the wetwell is assumed to remain at 400°F even though the drywell will be exposed to 800°F temperatures.

- The total probability of any one or more failure modes occurring at the same pressures.

Figure 4.4-7 summarizes the cumulative failure probabilities for each failure mode and the calculated total if all failure modes are considered. Assuming that a log normal function describes these results, the mean containment failure pressure is calculated to be approximately 136 psig.

The assessment for this portion of the analysis only considers the structural failures; leakage at closures is not included at this point in the analysis (see Sections 4.4.4.5 and .6).

The relative probabilities of the five containment failure categories for temperatures of 800°F in the drywell and 400°F in the wetwell, are presented in the following table:

NMP2 Containment Failure Location and Size: Approximately 800° in DW		
Location	Containment Breach	Relative Failure Probability
Drywell	Large	0.43
Wetwell below water line	Large	0.32
Wetwell above water line	Large	0.05
Drywell	Leakage	<< 0.01 (Estimate)
Wetwell above water line	Leakage	0.20

The conclusion from these assessments is that the mean probability of containment structural failure remains relatively constant at approximately 140 psig, but the relative contribution among the various failure modes changes significantly, as a function of temperature. For instance, at 400°F, 40% of the total containment failure probability is associated with rupture below the wetwell water line, while at 800°F this contribution is reduced to 32% and the large drywell failure becomes the dominant mode for containment structural failure.

4.4.4.4 High Pool Temperatures, High Containment Pressures and High SRV Discharge Rates

Based on the limited amount of data to support containment integrity at high SRV discharge rates and at elevated containment temperatures, pressures and water levels, the containment wetwell is considered to be failed if temperatures exceed 260°F in the suppression pool and substantial power is being produced in the core and discharged to the pool. This is further supported by a number of issues each of which identifies potential topics effecting containment failure when subjected to these conditions.

These issues are the following:

- Condensation phenomena
- Temperature profile at the quencher device
- Limitation of calculational models
- Vacuum breaker performance with cycling drywell sprays
- Containment structural capability under hydrodynamic loads
- Cyclic pressure effects
- Elevated wetwell water levels affecting hydrodynamic loads.

Based upon the uncertainty associated with each of these effects, the containment integrity is not judged to be capable of sustaining prolonged blowdown from SRVs with power levels substantially above decay heat, i.e., greater than 4% when the pool temperature is above 260°F.

4.4.4.5 Failure of Hatches and Penetrations at High Temperature

Just as with the main structural failure modes of the containment, the potential failure modes that affect the containment hatches are defined as breaches in the pressure boundary. However, the failure modes evaluated in this section do not necessarily result in gross failures. This is especially true for hatches which are pressure unseating (i.e., outward opening). Among the hatches at Nine Mile Point Unit 2, two equipment hatches, the control rod drive removal hatch, and the two suppression pool access hatches are pressure unseating closures. In addition, the drywell closure head which is secured by taper-pins loaded in shear is also a pressure unseating penetration.

For these outward opening hatches, leakage is not expected to occur prior to metal-to-metal flange separation. Even at high temperature, the E-P rubber seals are expected to prevent leakage provided the preload is not exceeded by the internal pressure load. After the bolt preload is exceeded, metal-to-metal separation of the flanges will occur. However, leakage will not occur until after the flange opening exceeds the O-ring pinch which is highly temperature dependent. At temperatures above 200°F, E-P rubber shows considerable residual set, and above about 350°F, essentially 100% residual set can be expected. Therefore, at temperatures in excess of 400°F, leakage in hatches is included in the assessment of containment structural capability.

The ABB Impell report [Ref. 121] concluded that, for material temperatures greater than about 400°F, leakage was found to occur at lower pressures for the pressure unseating hatches. Therefore, the progression of the cumulative leak areas in the drywell and the wetwell were evaluated as a function of pressure for material temperatures of 400°and 800°F. Since it is expected that the wetwell airspace temperature will remain significantly cooler than the drywell airspace (i.e., a maximum temperature of approximately 400°F), the drywell head penetration is determined to provide the single largest potential leakage site in containment as a result of seal degradation and flange distortion. Additionally, as drywell temperature approaches 800°F, a greater proportion of containment leakage is expected from the head flange region rather than the two equipment hatches. At temperatures greater than 800°F, time-dependent creep effects are possible to enter into the response of the controlling hatches. As a result, material creep can further increase the leakage areas with time.[1]

4.4.5 Summary of Probabilistic Evaluation of Nine Mile Point Unit 2 Containment Performance

It is possible to identify the dominant NMP2 containment failure modes in three pressure-temperature regimes based on the ABB Impell report. The following discussion describes the relevant containment failure modes for the three regimes of containment performance as denoted on Figure 4.4-8:

(1) At low temperature (i.e, below approximately 400°F), the dominant failure mode corresponds to a large failure in the wetwell below the water line at the junction of the primary containment wall and the basemat. This failure mode corresponds to a ductile flexure-tension failure in which the maximum strength of the wall section is reached, resulting in a large failure of containment. The second most dominant failure mode, having a median pressure capacity only slightly higher than the basemat failure mode, corresponds to a leakage failure mode in the wetwell air space induced by local liner tearing. For this failure mode, the liner tear is also a ductile failure. The exact size of the breach and water level in the wetwell corresponding to the accident conditions is uncertain; therefore, this failure is assumed to occur above the water line. The third

[1] High temperature creep effects were not evaluated in the ABB Impell analysis.

most critical containment failure mode affects the structural integrity of the drywell head region. The resultant large breach of containment in this case is primarily caused by hoop membrane failure. Table 4.4-9 summarizes the conditional probability of containment failure as a function of size and location if the capability curve (Figure 4.4-8) is exceeded.

(2) At intermediate temperatures and pressures, a general reduction in the pressure capacities of various structures is indicated by the Impell analysis. Additionally, the Impell analysis postulates some re-ordering of these failure modes other than the two wetwell failure modes described above. Most importantly, at temperatures greater than 400°F, seal leakage failure modes govern the failure capability of the containment. These failures are predominantly in the drywell and are associated with pressure unseating flanges; specifically, the two equipment hatches, the drywell closure head, and the control rod drive removal hatch. The elastomeric seal material of these four major containment openings is assumed to be severely degraded when subjected to temperatures above 400°F (i.e., 100% compression set of the seals). (However, at lower temperatures, there is sufficient O-ring pinch as to preclude significant leakage around these hatches and penetrations.)

The progression of the cumulative leak areas in the drywell and the wetwell were evaluated as a function of pressure for material temperatures of 400°F and 800°F. Since it is expected that the wetwell air space temperature will remain significantly cooler than the drywell air space (i.e., a maximum temperature of approximately 400°F), the drywell head penetration is determined to provide the single largest leakage area in containment as a result of seal degradation and flange distortion. Additionally, the leakage area from the head flange is a strong function of temperature; so as drywell temperature approaches 800°F, a greater proportion of containment leakage is expected from the head flange region rather than the two equipment hatches.

Probabilistically, the cumulative mean leakage area from these three drywell penetrations are judged to be the dominant containment failure modes affecting containment performance and the transport of radionuclides to the environment over this intermediate temperature range. Implicit in this conclusion is the assumption that the total cumulative leakage area is sufficient to mitigate containment pressurization from non-energetic events (i.e., steam vaporization and non-condensible gas generation). Considering the degree of uncertainty stated in the Impell analysis regarding leakage size, this conclusion seems reasonable and justifiable.

Qualitatively, the conclusion of the Impell report regarding seal performance is substantiated by other industry studies performed on similar pressure-unseating structures that contain like seal material. Additionally, the premise that the leakage is sufficient to depressurize containment during all but ATWS scenarios upon degradation of these seals is consistent with generic studies applicable to a Mark II containment. Table 4.4-10 summarizes the conditional probability of containment failure as a function of size and location if the capability curve (Figure 4.4-8) is exceeded on the sloping portion of the curve.

(3) For extremely high temperatures, direct high temperature induced failures of the containment boundary may occur at:

- Penetration and hatch seals (personnel airlock, equipment hatches, purge and vent valves),
- Drywell head flange,
- Electrical penetration assemblies (EPAs),
- The pedestal[1], causing loss of support to the RPV and the disruption of piping attached to the RPV,
- The RPV steel skirt, causing loss of support to the RPV and the disruption of piping attached to the RPV,

The leakage area increases at extremely high temperatures due primarily to severe seal deterioration and material creep rupture effects[2].

Additionally, at extremely high containment gas temperatures (i.e., >900°F), industry studies have postulated that material (i.e., steel and concrete) properties may change sufficiently to cause containment structural degradation to the extent that its integrity cannot be maintained at any internal pressure. Although the Impell analysis did not specifically consider scenarios during which the containment would be subjected to such extreme temperature conditions, the assumption of a large drywell head failure is considered a reasonable and possibly conservative estimate of the containment performance in such a situation.

The purpose of assimilating the information contained in the Impell report is to develop an integrated containment performance profile describing the pressure and temperature conditions (considered in the structural analysis to be quasi-static) inside the containment that can cause a larger than negligible breach from structural or seal failure in the containment. Figure 4.4-8 provides the best estimate containment failure pressure as a function of the containment drywell temperature.

Once the analyst has estimated the mean performance capability of the containment, the probabilistic split fractions, describing in general terms the size and location of the containment breach, are developed for severe accident scenarios postulated in the Level 2 assessment. This task requires that the analyst understand the accident signature of each scenario before the capability of the containment to respond to deteriorating conditions can be assessed

---

[1] Preliminary calculations indicate that because of the large amount of rebar in the pedestal, that the pedestal integrity is considered only a secondary or late contributing failure mode.

[2] Creep rupture failures on bolts that have been preloaded can occur at close to zero internal pressure. Calculations on other BWRs have indicated that creep rupture is a potentially dominant failure mode if the shell or bolts are exposed to temperatures of 1200°F and 60 psig. In addition, if leakage occurs through deteriorated seals and exposes the bolts directly to hot gases, this failure mode could be accelerated.[E-8]

probabilistically. Therefore, the MAAP code is used to determine the accident signature (i.e., containment pressure and drywell gas temperature) by modeling the scenario initially assuming that the containment has infinite capacity to remain intact. The analyst then superimposes the scenario signature onto the containment performance profile to estimate the event node split fractions describing the probable location(s) of the impending containment breach. Figure 4.4-2 provides a simplified diagram describing this process.

Tables 4.4-9 through 4.4-12 summarize the conditional probabilities of containment failure under various postulated accident conditions described in Figure 4.4-8 based on an interpretation of the Impell report and industry research. Figure 4.4-9 illustrates the breach locations that represent failure in a particular containment "zone". These locations are assigned to each zone to facilitate the calculation of the source term associated with a particular accident sequence that results in containment failure and radionuclide release to the reactor building.

Containment failure size, as defined in the preceding tables as being either "Rupture or Leak," is relevant only in the context of determining the source term associated with each accident scenario that results in containment breach. Therefore, the following area dimensions are selected to represent containment breach size:

$$\text{RUPTURE} = 2\ \text{ft}^2$$
$$\text{LEAK} = 27\ \text{in}^2$$

This information is subsequently used in MAAP calculations to determine the severity of radionuclide release to the environment during any postulated severe accident involving containment failure.

4.4.6 Reactor Building Failure Mode

The NMP2 secondary containment consists of the reactor building which surrounds the Mark II containment. The failure modes and locations of the NMP2 reactor building are as follows:

> Failure Location: The reactor building is a concrete structure with blowout panels located in the refuel floor roof. Therefore, overpressurization of the reactor building has been found to result in failure of the blowout panels and a release path through one of these blowout panel pathways.
>
> Failure Modes: Reactor building overpressure failure at the blowout panels is the dominant failure mode.
>
> Nevertheless, the Level 2 assessment has assigned a high probability of a lack of reactor building effectiveness (i.e., a DF = 1.0) due to a number of possible phenomena and assumptions such as:
>
> - Hydrogen burning
> - High flow rates
> - Model inadequacy
>
> These phenomena can result in the reactor building having essentially no significant radionuclide retention capability at critical times in the release scenario.

Table 4.4-2

POSTULATED CONTAINMENT CHALLENGES/FAILURE MODES

		Challenges/Failure Modes That May Precede a Severe Accident
1.	*	Containment Isolation Failure
2.	*	Interfacing System Loss of Coolant Accident (LOCA)
3.	*	Blowdown Forces Due to RPV Rupture or Containment Overpressure Due to Catastrophic Reactor Pressure Vessel (RPV) Failure
4.	*	Pipe Whip/Steam Jet Impingement
5.	*	Containment Overpressure Due to Anticipated Transient Without Scram (ATWS) or Loss of Containment Heat Removal
6.		Containment Overpressure from Pool Bypass (BWR)
7.		External Pressure Loading Due to Partial Vacuum Conditions
8.	*	Missiles from Internal (Plant) Sources
9.	*	Tornado and Tornado Missiles[1]
10.		Seismic Induced Failure[1]
11.		Containment Venting
		Challenges/Failure Modes Potentially Resulting from a Severe Accident
12.		High Pressure Core Melt Ejection
13.	*	Hydrogen Related Issues (Deflagration/Detonation)
14.	*	In-vessel Steam Explosion
15.	*	Ex-vessel Steam Explosion
16.	*	Reactor Pressure Vessel Support Failure and Containment Basemat Penetration
17.		Containment Sump Failure from Core Debris
18.	*	Containment Shell Failure from Core Debris
19.		Containment Overtemperature Due to Debris
20.	*	Containment Overpressurization Due to Core Debris Decay Heat Steam Generation
21.	*	Noncondensible Gas Generation
22.		Reactivity Insertion During Core Melt Progression
23.		N_2 Pressure
24.		Direct Impingement

* Identified from PRA Procedures Guide

[1] These failure modes are treated in the IPEEE and are not addressed in this report.

Table 4.4-3

SUMMARY OF TREATMENT OF CHALLENGES IN THE NINE MILE POINT 2 CONTAINMENT SAFETY STUDY

Postulated Containment Challenges	Disposition
Containment Initial Conditions	
1. Containment Isolation Failure	Included in CET Treatment assumes inerting has substantial benefit in assuring isolation Plant specific drain line failures (bypass) are of interest because Nine Mile Point 2[1] drain line design is a "fail as is".
Sequence Dependent Failure Modes	
2. Interfacing System LOCA	Included in Level 1 evaluation
3. RPV Rupture Overpressure	Included in Level 1 evaluation
4. Pipe Whip/Steam Jet Impingement	Dismissed based on low probability
5. ATWS - Overpressure TW - Overpressure	Included in Level 1 evaluation
6. Vapor Suppression Failure (Suppression Pool Bypass)	Included in Level 1 evaluation
7. Containment Implosion Due to Drywell Sprays	Low Probability Due to Mark II Structural Capability and EPG procedural guidance
8. Missiles from Internal Sources	To be evaluated in IPEEE
9. Tornado and Tornado Missiles	To be evaluated in IPEEE
10. Seismic Induced	To be evaluated in IPEEE
11. Containment Venting and Combustible Gas Vents	Included in Level 1 and 2 Event Trees

[1] MOV do not fail closed on loss of air or control power.

Table 4.4-3

SUMMARY OF TREATMENT OF CHALLENGES IN THE NINE MILE POINT 2 CONTAINMENT SAFETY STUDY

Postulated Containment Challenges	Disposition
Phenomenological Failure Modes	Addressed In Level 2 CET
12. Direct Containment Heating	Included (low probability)
13. Hydrogen Effects:	
- Quantity of H_2 Produced In-Vessel	Range of values examined
- H_2 + O_2 Deflagration Effects	Conditional probability of deflagration included
- Introduction of O_2	None considered possible except operation deinerted
- RPV Blowdown Failure + H_2 Causes containment failure	Calculated not to cause containment failure at Nine Mile Point 2
14. In-vessel Steam Explosions	Included in Level 2 analyses
15. Ex-vessel Steam Explosions	Included in Level 2 analyses
16. Structural Failure Due to RPV Collapse and Tear Out of Penetration	Included in high temperature induced pedestal/skirt failures
17. Containment Sump Line Failure	Not applicable to Nine Mile Point 2 by design
18. Direct Contact of Molten Material W/Steel Shell	Not a NMP-2 Mark II issue because of design differences.
19. DW Head Seal Performance at Elevated Temperature (High Temp Failure)	Included as a potential leak path
20. Containment overpressure due to decay heat	Included in Level 2 analyses
21. Noncondensible Gas Generation (Core Concrete Attack)	Included (range of modeling assumptions examined)
22. Reactivity Insertion during Core Melt Progression	Included in Level 2 quantification
23. N_2 Overpressurization During Accident	Included in Level 2 analyses (low probability)
24. Direct Impingement	Included in Level 2 quantification

Table 4.4-4

SUMMARY OF TIMING, SIZE, AND LOCATION FOR POSTULATED CONTAINMENT FAILURE MODES

Postulated Containment Challenge	Timing	Size	Location[1]
Sequence Dependent Failure Modes			
• ATWS Without Mitigation	Early	Large	DW, WW
• RPV Rupture Large Enough to Cause Containment Failure	Early	Large	DW
• TW-Overpressure	Late	Small, Large	DW, WW
• Vapor Suppression Failure + LOCA	Early	Large	DW, WW
• N_2 Overpressurization	Intermediate	Small, Large	DW, WW[2]
• Combustible Gas Vent	Early[3]	Large	WW
• Containment Implosion Due to DW Spray Initiation	Early	Large	DW
• Containment Overpressure Vent	Late	Small	WW
Phenomenological Failure Modes			
• Noncondensible Gas Generation	Intermediate[3]	Small, Large	DW, WW
• Direct Containment Heating	Early[3]	Large	DW
• DW Temperature Rise	Intermediate[3]	Small, Large	DW
• Steam Explosions	Early[3]	Large	DW
• Hydrogen Explosions	Early[3]	Large	WW, DW[2]
• Structural Failure due to Penetration Tearout	Intermediate[3]	Large	DW
• Vessel Thrust Forces	Early[3]	Large	DW
Containment Initial Conditions			
• Containment Isolation Failure	Early	Large	DW
• Containment Leakage	Early/Late[3]	Small	WW

[1] WW = Wetwell, DW = Drywell

[2] Always treated as a drywell failure in the simplified CET evaluation.

[3] These times are relative to RPV breach, which of course may be delayed significantly from accident initiation depending on the accident sequence.

Table 4.4-6

CUMULATIVE CONTAINMENT FAILURE PROBABILITY AS A FUNCTION OF FAILURE MODE AND PRESSURE[1,2]
(400°F in the drywell and wetwell)

Containment Failure Mode	Cumulative Probability of Each Failure Mode						
	120	140	160	170	180	200	220
Wall/Basemat Junction Failure	0.036	0.16	0.39	0.51	0.63	0.81	0.92
Wetwell Liner Tearing	0.048	0.16	0.34	0.44	0.54	0.71	0.83
Wall/Basemat Junction Radial Shear	0.021	0.09	0.23	0.32	0.42	0.61	0.76
Hoop Membrane @ Elev. 324'	0.035	0.11	0.24	0.32	0.4	0.56	0.7
Wetwell Hoop Membrane	0.006	0.037	0.12	0.19	0.28	0.46	0.64
Drywell Flexure/Tension @ Elev. 240'	0.004	0.028	0.1	0.16	0.24	0.42	0.6
Radial Shear @ Elev. 324'	0.007	0.03	0.084	0.13	0.18	0.3	0.44
Basemat Shear	0	0.002	0.01	0.021	0.038	0.098	0.19
Cumulative Total Failure Probability	0.15	0.48	0.83	0.93	0.98	0.99	~1.0

1 The eight most dominant failure modes are considered.

2 Structural failures dominate the Impell assessment at these temperatures. No closure leakage induced failure modes are considered possible by Impell.

Table 4.4-7

STRUCTURAL FAILURE PROBABILITY AS A FUNCTION OF FAILURE MODE AND PRESSURE[1,2]
(600°F in the Drywell and 400°F in the Wetwell)

Failure Mode	Cumulative Probability of Each Failure Mode						
	120	140	160	170	180	200	220
Wall/Basemat Junction Failure	0.036	0.16	0.39	0.51	0.63	0.81	0.92
Wetwell Liner Tearing Shear	0.048	0.16	0.34	0.44	0.54	0.71	0.83
Wall/Basemat Junction Radial	0.021	0.09	0.23	0.32	0.42	0.61	0.76
Hoop Membrane @ Elev. 324'	0.042	0.13	0.27	0.35	0.43	0.59	0.73
Wetwell Hoop Membrane	0.006	0.037	0.12	0.19	0.28	0.46	0.64
Drywell Flexure/Tension @ Elev. 240'	0.007	0.043	0.14	0.21	0.3	0.49	0.67
Radial Shear @ Elev. 324'	0.009	0.037	0.1	0.15	0.21	0.34	0.48
Basemat Shear	0	0.002	0.01	0.021	0.038	0.098	0.19
Cumulative Failure Probability Total	0.16	0.51	0.85	0.94	0.98	0.99	~1.0

[1] The eight most dominant failure modes are considered.

[2] Leakage through closure seals is considered separately and included in Table 4.4-10. This table is based solely on structural failures of containment.

Table 4.4-8

CUMULATIVE STRUCTURAL CONTAINMENT FAILURE PROBABILITY AS A FUNCTION OF FAILURE MODE AND PRESSURE[1,2]
(800°F in the Drywell and 400°F in the Wetwell)

Containment Failure Mode	Cumulative Probability of Each Failure Mode						
	120	140	160	170	180	200	220
Wall/Basemat Junction Failure	0.036	0.16	0.39	0.51	0.63	0.81	0.92
Wetwell Liner Tearing Shear	0.048	0.16	0.34	0.44	0.54	0.71	0.83
Wall/Basemat Junction Radial	0.021	0.09	0.23	0.32	0.42	0.61	0.76
Hoop Membrane @ Elev. 324'	0.069	0.18	0.34	0.42	0.51	0.66	0.78
Wetwell Hoop Membrane	0.006	0.037	0.12	0.19	0.28	0.46	0.64
Drywell Flexure/Tension @ Elev. 240'	0.022	0.094	0.24	0.33	0.43	0.62	0.77
Radial Shear @ Elev. 324'	0.017	0.062	0.15	0.21	0.27	0.41	0.56
Basemat Shear	0	0.002	0.01	0.021	0.038	0.098	0.19
Cumulative Failure Probability Total	0.2	0.57	0.88	0.95	0.99	0.99	~1.0

[1] The eight most dominant failure modes are considered.

[2] This table is based solely on structural failures of containment. Leakage through closure seals is considered separately and included in Table 4.4-11.

Table 4.4-9

SUMMARY OF THE NMP2 MARK II CONTAINMENT CONDITIONAL FAILURE PROBABILITY[1] AT INTERNAL TEMPERATURES < 400°F

CONTAINMENT LOCATION	FAILURE TYPE	LEVEL 2 PRA EXPERT ASSESSMENT
Zone C1 DW Head	Leak	Epsilon
	Rupture	0.28
Zone C2 DW Upper Body	Leak[2]	(2)
	Rupture[2]	(2)
Zone C3 DW Main Body	Leak[2]	(2)
	Rupture[2]	(2)
Zone C4 WW Above Water Line	Leak	0.26
	Rupture	0.06
Zone C5 WW Below Water Line	Leak	Epsilon
	Rupture	0.40
Parameter		Mean

[1] Containment pressurization caused by water vaporization and non-condensible gas generation post RPV breach. Containment interior conditions: T ≤ 400°F, P ~ 141 psig.

[2] Drywell failures in these zones are considered less likely, and are treated conservatively as failures in Zone C1.

Table 4.4-10

SUMMARY OF THE NMP2 MARK II CONTAINMENT CONDITIONAL FAILURE PROBABILITY[(1),(3),(4)] AT DRYWELL TEMPERATURE 700°F AND WETWELL TEMPERATURES OF 400°F

CONTAINMENT LOCATION	FAILURE TYPE	LEVEL 2 PRA EXPERT ASSESSMENT
Zone C1 DW Head	Leak	0.8
	Rupture	0.08
Zone C2 DW Upper Body	Leak[(2)]	(2)
	Rupture[(2)]	(2)
Zone C3 DW Main Body	Leak[(2)]	(2)
	Rupture[(2)]	(2)
Zone C4 WW Above Water Line	Leak[(3)]	< 0.01
	Rupture	< 0.01
Zone C5 WW Below Water Line	Leak	Epsilon
	Rupture	0.1
Parameter		Mean

[1] Containment pressurization caused by water vaporization and noncondensible gas generation post RPV breach. Containment drywell interior conditions: T = 700°F, P ~ 80 psig.

[2] Drywell failures in these zones are considered less likely, and are treated conservatively as failures in Zone C1.

[3] Suppression pool airspace temperature is estimated to be < 400°F, which is judged not to affect the integrity of hatches and penetrations inside the suppression chamber.

[4] This estimate of containment failure probability includes an assessment of the containment leakage contribution due to closure seal failures.

Table 4.4-11
SUMMARY OF THE NMP2 MARK II CONTAINMENT
CONDITIONAL FAILURE PROBABILITY[1],[3],[4]
AT DRYWELL TEMPERATURES > 900°F

CONTAINMENT LOCATION	FAILURE TYPE	LEVEL 2 PRA EXPERT ASSESSMENT
Zone C1 DW Head	Leak	Epsilon
	Rupture	1.0
Zone C2 DW Upper Body	Leak[3]	(3)
	Rupture[3]	(3)
Zone C3 DW Main Body	Leak[3]	(3)
	Rupture[3]	(3)
Zone C4 WW Above Water Line	Leak	Epsilon
	Rupture	Epsilon
Zone C5 WW Below Water Line	Leak	Epsilon
	Rupture	Epsilon
Parameter		Mean

[1] Containment pressurization caused by the water vaporization and non-condensible gas generation post RPV breach. Containment drywell interior conditions: T$\geq$ 900°F.

[2] Suppression pool airspace temperature is estimated to be <400°F, which is judged not to affect the integrity of hatches and penetration inside the suppression chamber.

[3] Drywell failures in these zones are considered less likely, and are treated conservatively as failures in Zone C1.

[4] This estimate of containment failure probability includes an assessment of the containment leakage contribution due to closure seal failures.

Table 4.4-12

SUMMARY OF THE NMP2 MARK II CONTAINMENT CONDITIONAL FAILURE PROBABILITY[1]

Containment Location	Failure Type	LEVEL 2 PRA EXPERT ASSESSMENT		
		Unmitigated ATWS[2]	High Pressure RPV Blowdown[3]	Failure at CET node CZ/CE or CX/CY[4]
Zone C1 DW Head	Leak	Epsilon	Epsilon	Epsilon
	Rupture	0.01	0.28	1.0
Zone C2 DW Upper Body	Leak[5]	(5)	(5)	(5)
	Rupture[5]	(5)	(5)	(5)
Zone C3 DW Main Body	Leak[5]	(5)	(5)	(5)
	Rupture[5]	(5)	(5)	(5)
Zone C4 WW Above Water Line	Leak	Epsilon	0.26	Epsilon
	Rupture	0.99	0.06	Epsilon
Zone C5 WW Below Water Line	Leak	Epsilon	Epsilon	Epsilon
	Rupture	Epsilon	0.40	Epsilon
Parameter		Mean	Mean	Mean

[1] Containment pressurization caused predominantly by energetic effects post RPV breach.

[2] Suppression pool water temperature > 260°F. Refer to Section E.3.

[3] Class IA, IB, and IC scenarios in which RPV breach occurs with the vessel at high pressure. The resulting containment pressure spike (with little increase in drywell airspace temperature) is assumed to immediately fail the containment structure.

[4] Containment interior conditions: $T_{Drywell}$ ~600°F, $P_{Drywell}$ ~95 psig.

[5] Drywell failures in these zones are considered less likely, and are treated conservatively as failures in Zone C1.

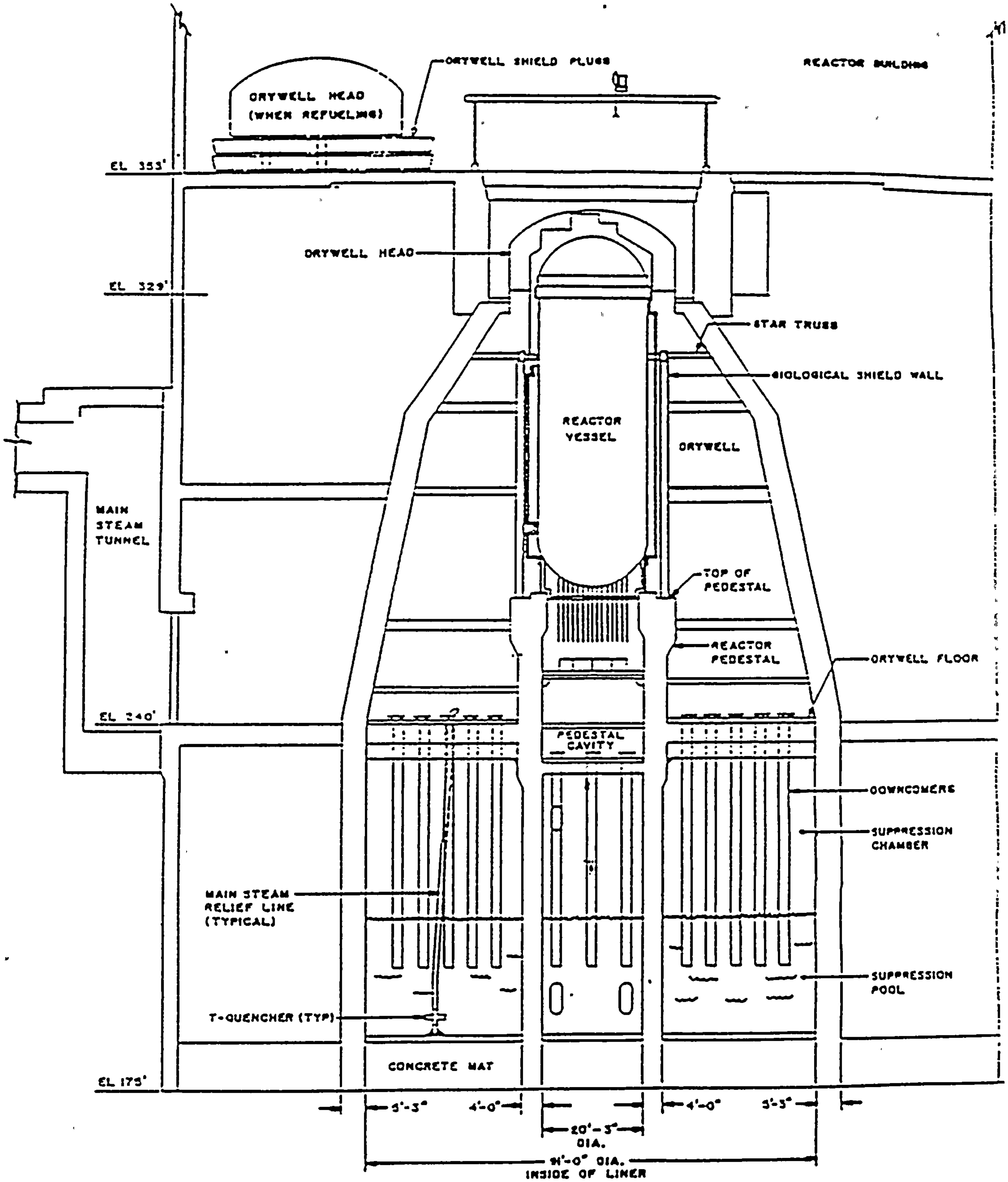


Figure 4.4-1 Nine Mile Point Unit 2 Containment

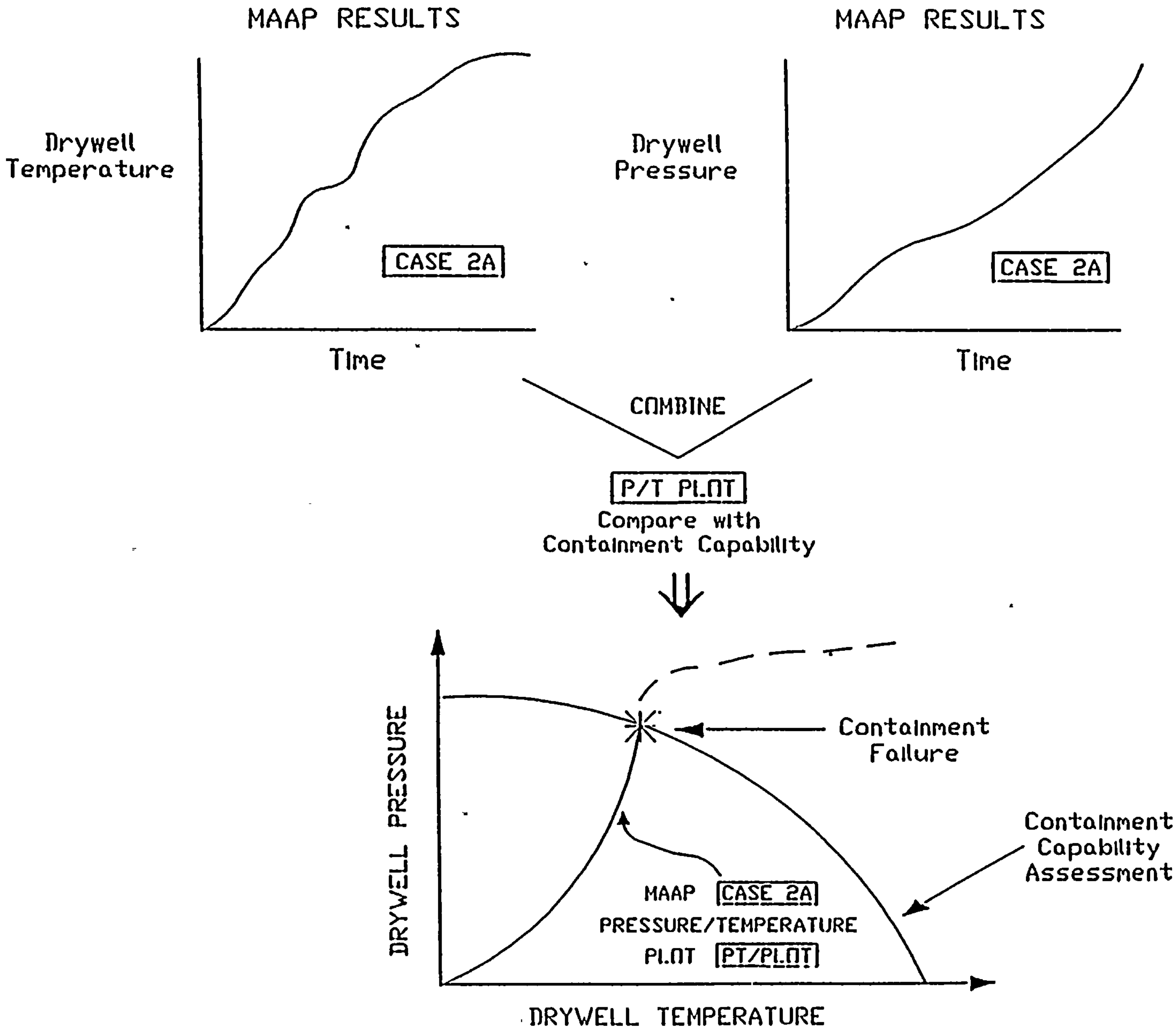


Figure 4.4-2 Simplified Flow Chart Showing the Relationship of MAAP Deterministic Results Compared with the Ultimate Containment Capability

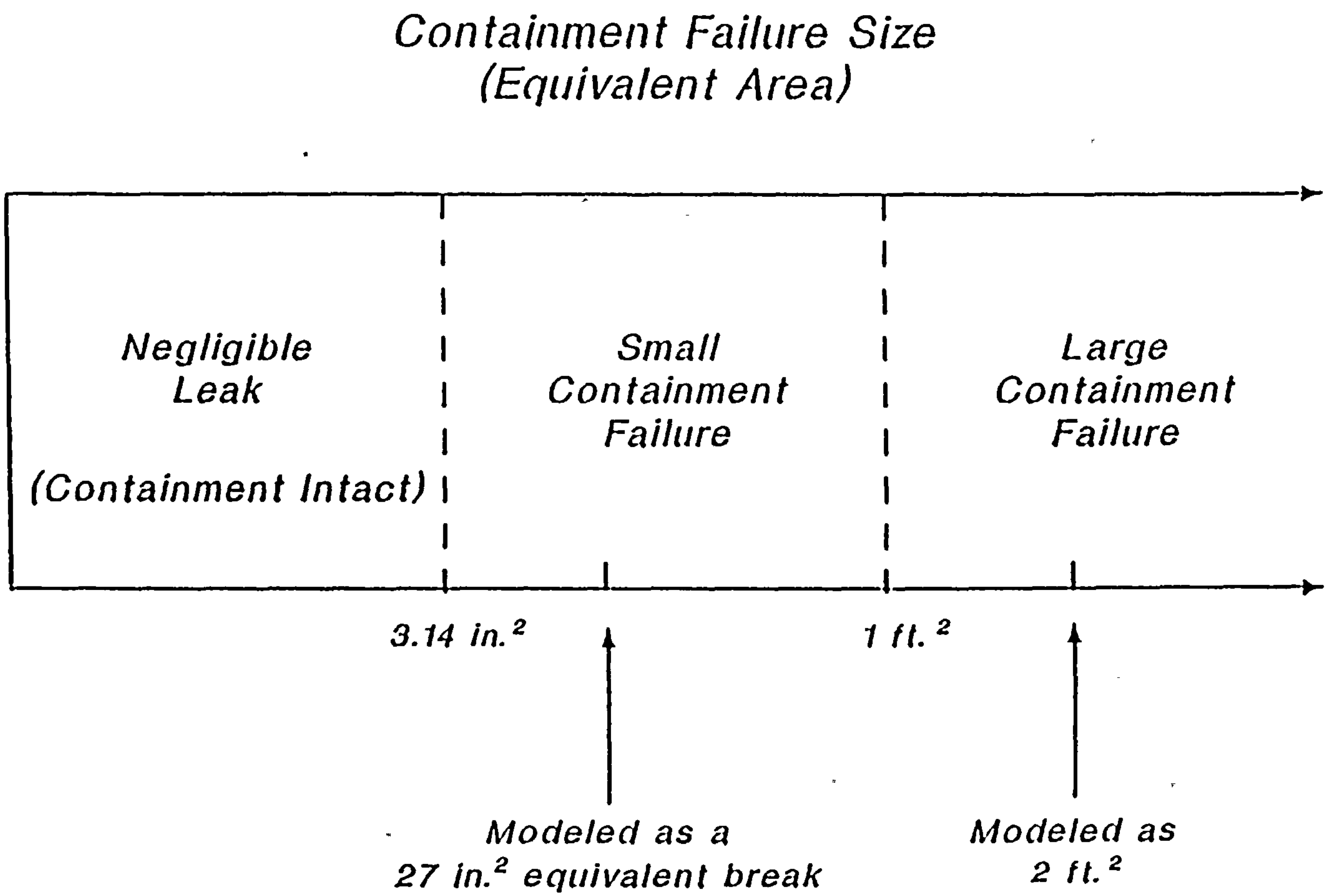


Figure 4.4-3

Characterization of the Containment Failure in the Level 2 Model

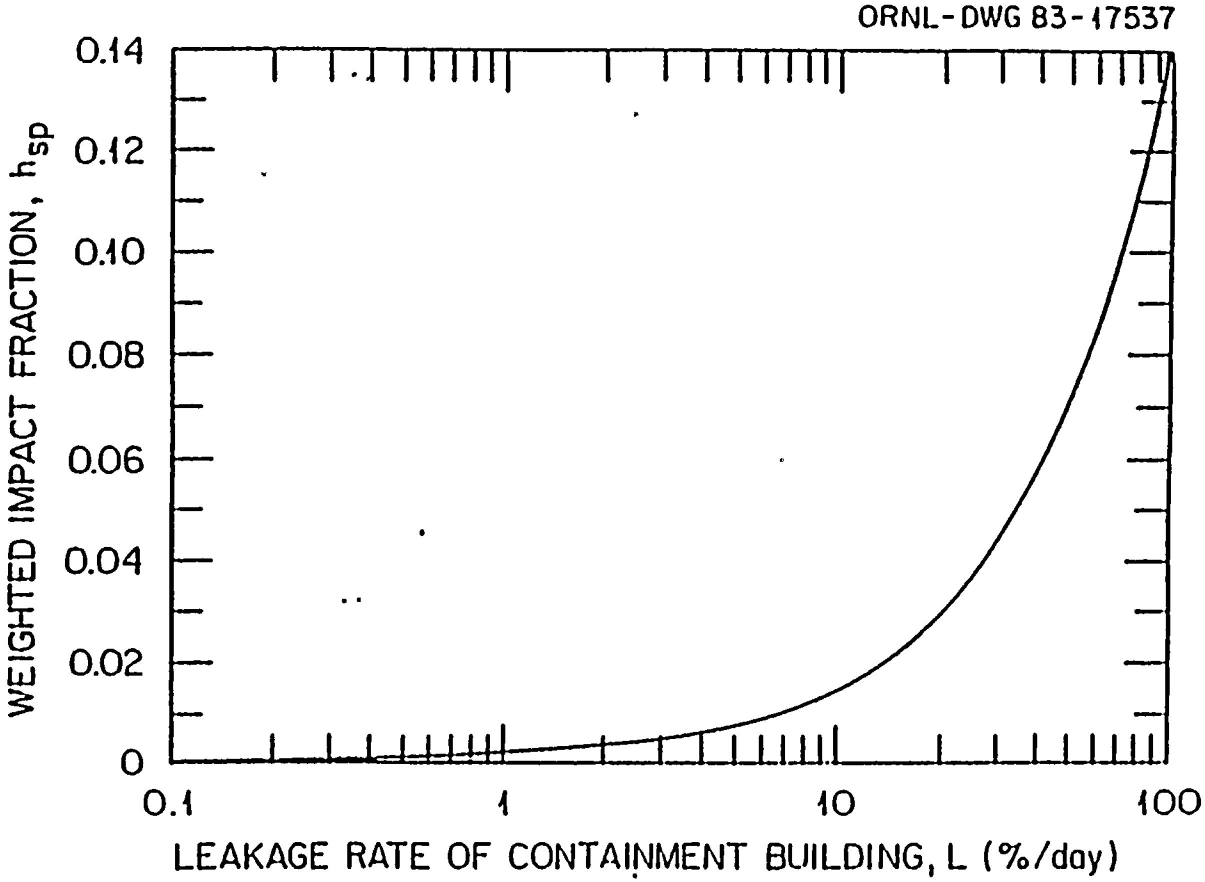


Figure 4.4-4

Impact on Accident Spectrum on Risk From Variation in Containment Building Leakage Rate

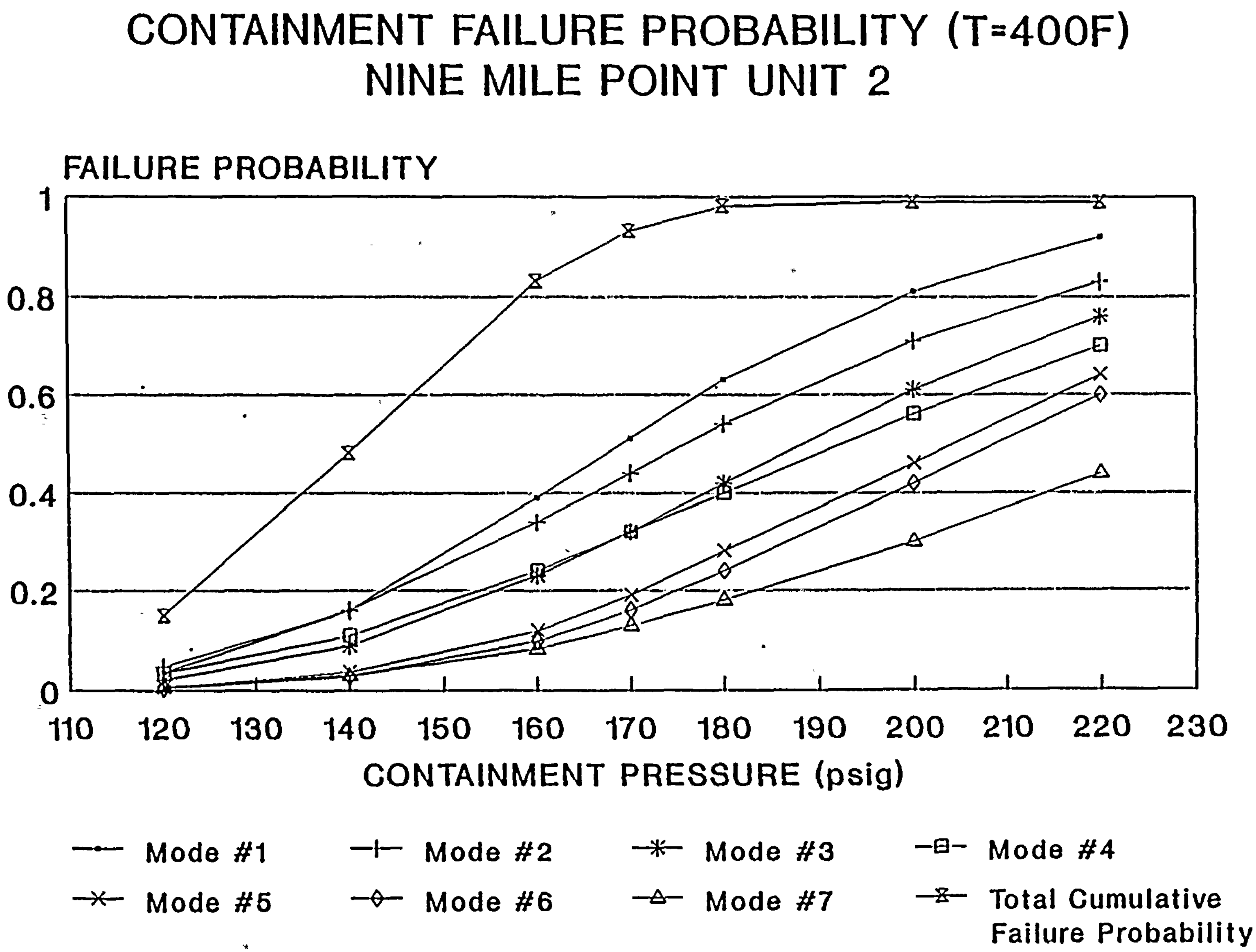

CONTAINMENT FAILURE PROBABILITY (T=400F)
NINE MILE POINT UNIT 2
FAILURE PROBABILITY
1
0.8
0.6
0.4
0.2
0
110
120
130
140
150
160
170
180
190
200
210
220
230
CONTAINMENT PRESSURE (psig)
Mode #1
Mode #2
Mode #3
Mode #4
Mode #5
Mode #6
Mode #7
Total Cumulative Failure Probability

Figure 4.4-5

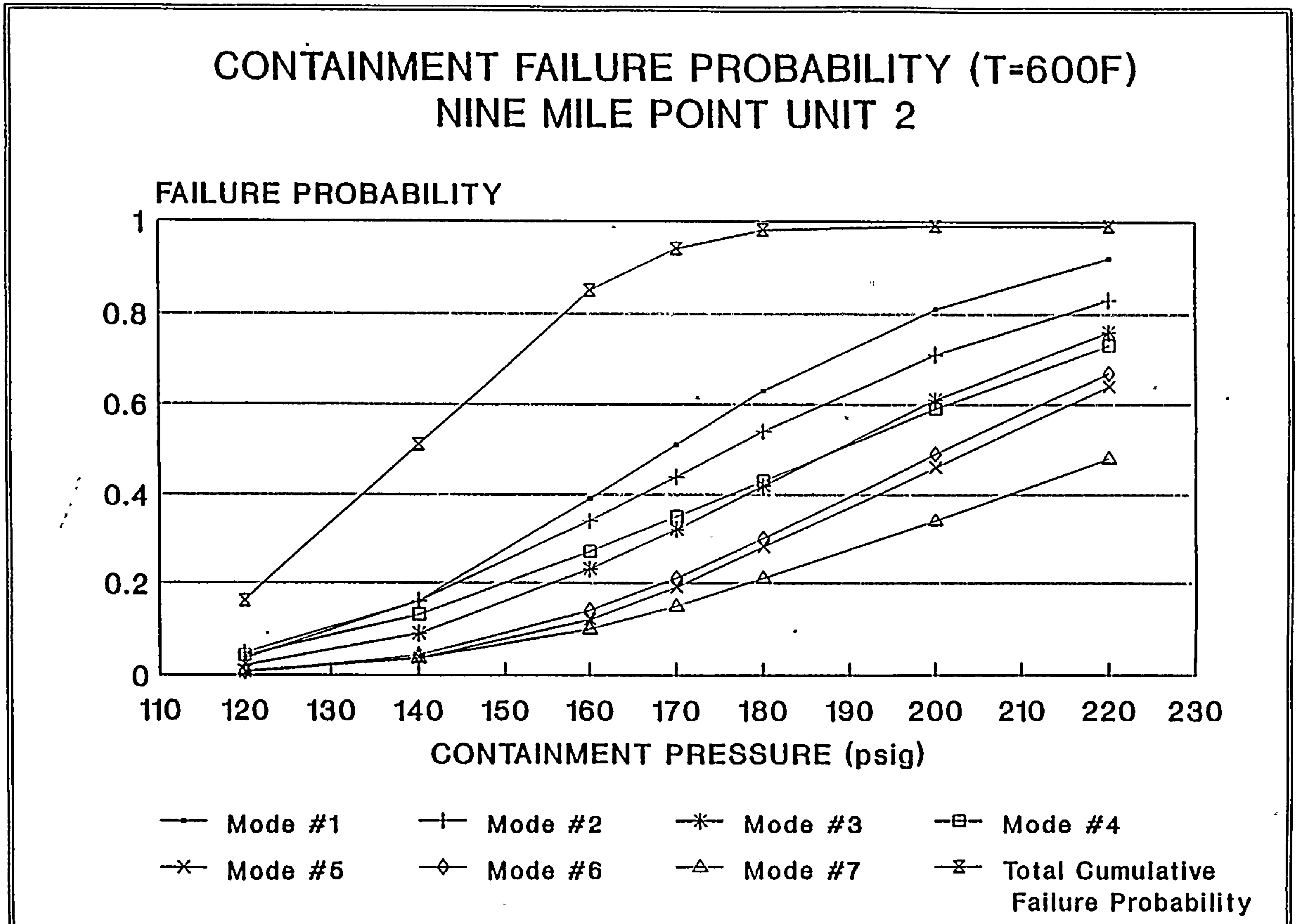

CONTAINMENT FAILURE PROBABILITY (T=600F)
NINE MILE POINT UNIT 2
FAILURE PROBABILITY
1
0.8
0.6
0.4
0.2
0
110
120
130
140
150
160
170
180
190
200
210
220
230
CONTAINMENT PRESSURE (psig)
Mode #1
Mode #2
Mode #3
Mode #4
Mode #5
Mode #6
Mode #7
Total Cumulative Failure Probability

Figure 4.4-6

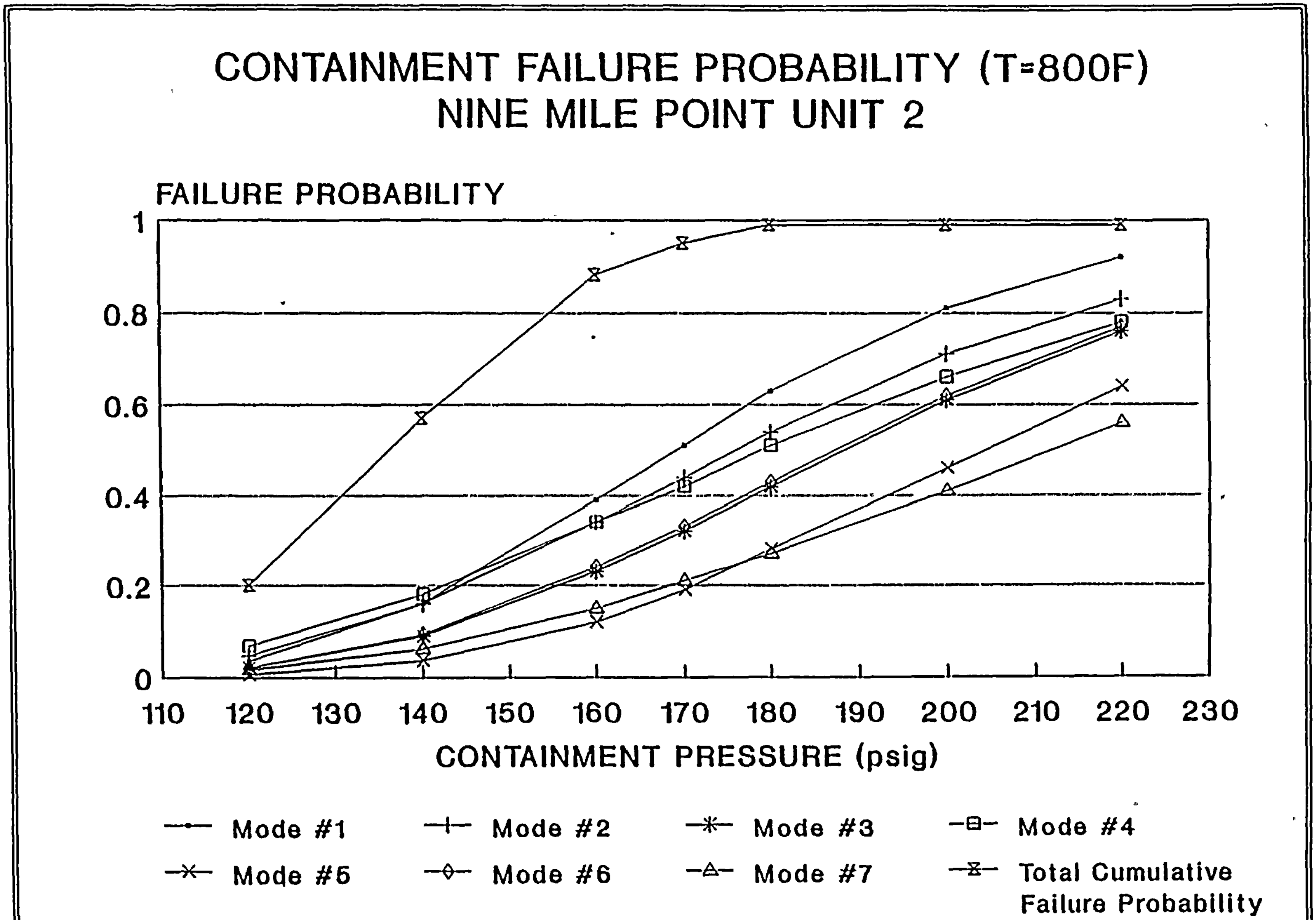
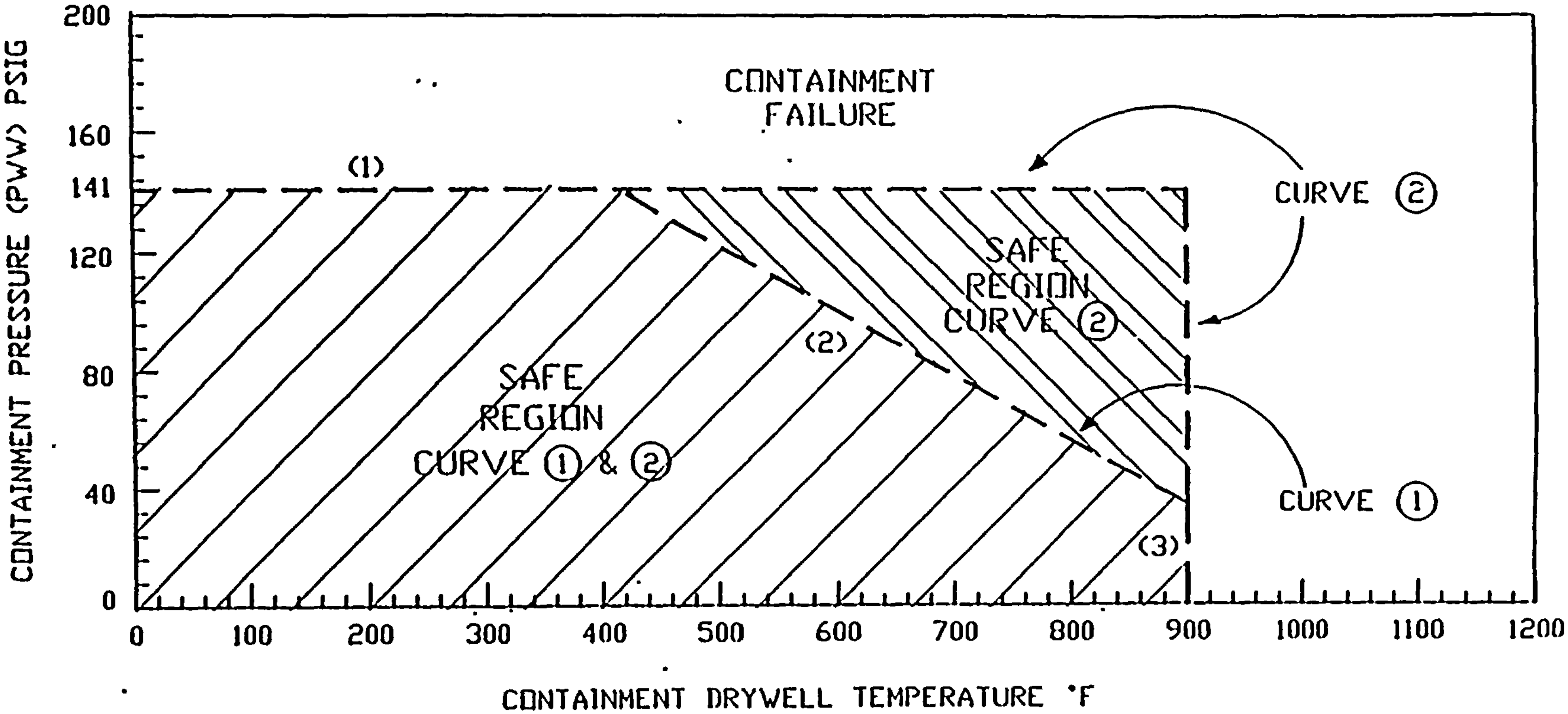

CONTAINMENT FAILURE PROBABILITY (T=800F)
NINE MILE POINT UNIT 2
FAILURE PROBABILITY
1
0.8
0.6
0.4
0.2
0
110
120
130
140
150
160
170
180
190
200
210
220
230
CONTAINMENT PRESSURE (psig)
Mode #1
Mode #2
Mode #3
Mode #4
Mode #5
Mode #6
Mode #7
Total Cumulative Failure Probability

Figure 4.4-7



NINE MILE POINT UNIT 2, MARK II CONTAINMENT PRESSURE
AND TEMPERATURE PERFORMANCE PROFILE

Figure 4.4-8

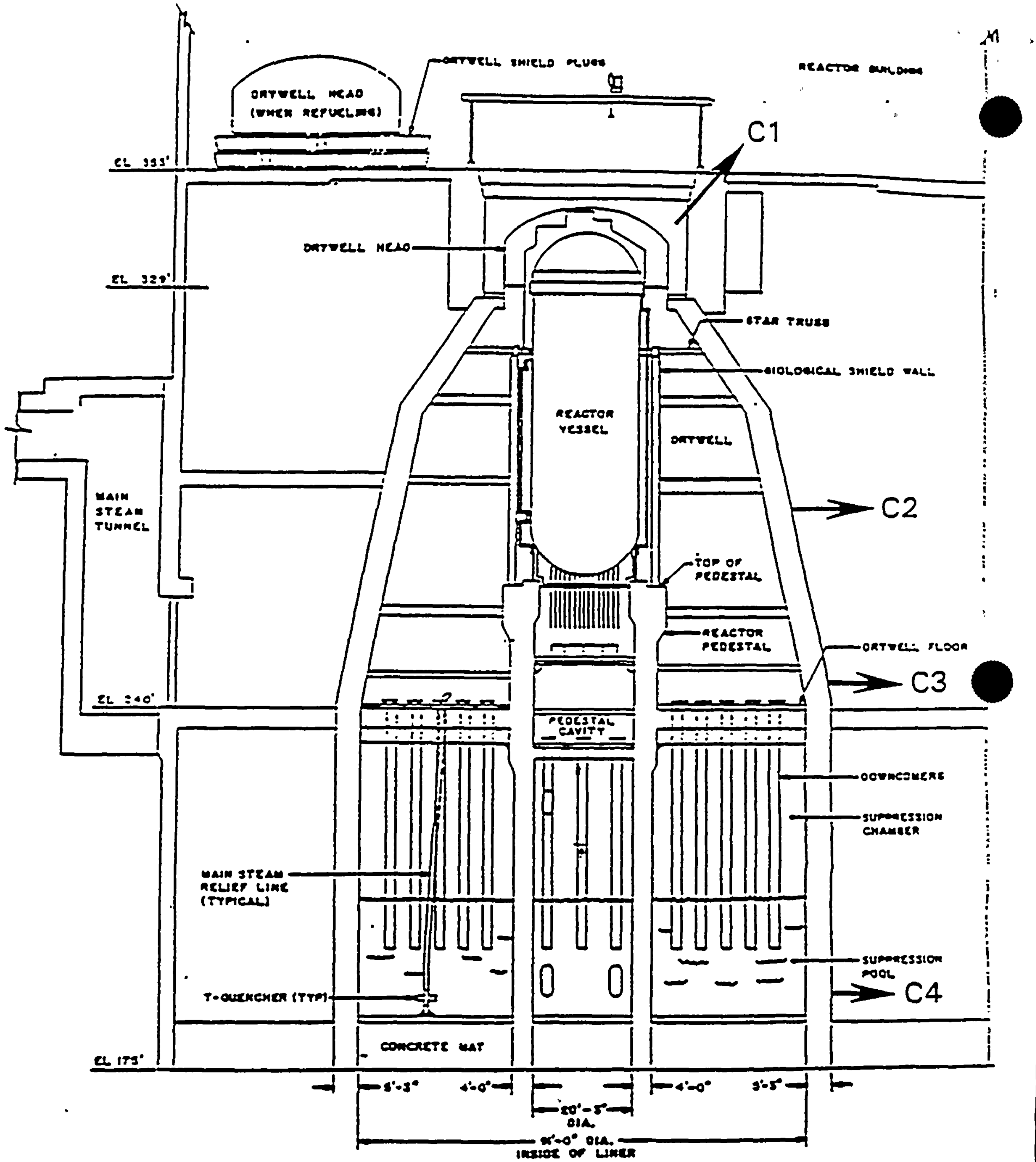
Notes To Figure 4.4-8

Curve 1:

1) Curve 1 represents a best estimate of containment performance under quasi-static, extreme temperature and pressure conditions. Therefore, the timing of containment failure can be determined for scenarios that do not involve rapid and acute pressurization caused by energetic effects (e.g., steam explosions, DCH, hydrogen combustion, high pressure melt ejection). Therefore, this curve is applicable to scenarios where the operating crew is unable to establish containment heat removal or prevent extreme internal temperatures (due to radiant energy from core debris inside the vessel or debris that is exposed on the pedestal floor.) Because the temperature induced failures (e.g., seal degradation and creep rupture) are time at temperature dependent phenomena, accident conditions are not considered to fail the containment unless the transient exceeds the failure point for longer than 3 hours. However, if the containment pressure exceeds approximately 140 psig, it is assumed to fail immediately due to structural failure.

2) In the case where containment is challenged post core damage due to energetic effects (i.e., containment pressure exceeds 141 psig or temperature exceeds 900°F). Curve 2 is judged to best represent containment failure conditions.

3) An additional failure mode due to hydrodynamic loads at elevated pool temperatures and high SRV discharge flow rates is not represented in Figure E.5-1. Based on the limited amount of data to support containment integrity at high SRV discharge rates and at elevated containment temperature and pressures (i.e., unmitigated ATWS events), the containment wetwell is considered to fail if temperatures exceed 260°F in the suppression pool and substantial power continues to be produced in the core and discharged to the pool.



Containment Failure Zones

Figure 4.4-9

4.5 Containment Event Trees

4.5.1 Containment Evaluation Process

Nine Mile Point 2 Containment Event Trees (CETs) are developed to provide the link between: (1) the Level 1 event tree core damage end states; and (2) safe shutdown or radionuclide release end states that describe release magnitude and timing. The CET is used to map out the possible containment conditions affecting the radionuclide releases associated with a given core damage sequence. The Nine Mile Point 2 IPE uses containment event trees that integrate system and human responses with phenomenological aspects of a severe accident. The potential for recovery actions based on the accident management philosophy of the EOPs is included. Additionally, these models describe the various potential radionuclide release paths to the environment and provide an estimate of their relative likelihoods.

The approach chosen focuses on the containment failure mechanisms and the timing of such failures. Application of this approach makes use of a number of deterministic and probabilistic risk assessment tools to establish a framework for radionuclide release evaluation. The spectrum of radionuclide releases which could result from these end states is then calculated for the postulated discrete end states of the CET.

Figure 4.5-1 is a simplified schematic that indicates the Level 2 Nine Mile Point 2 model uses:

- A containment event tree

- A set of functional fault trees to describe the failure modes at each event tree node.

The Nine Mile Point 2 containment event tree structure has been developed with the following objectives:

- Represents the time sequence of events and divides the CET into major time periods;

- Incorporates all important system, human and phenomenological occurrences including possible recovery;

- Maintains a simplified representation;

- Avoids the necessity of intermediate binning;

- Preserves the nature of the challenge throughout the analysis;

- Divides sequence treatment based on whether the RPV is at high or low pressure and whether the accident progression is in-vessel or ex-vessel;
- Explicitly recognizes the effect of postulated containment failure modes;

- Allows the identification of recovery and repair actions that can terminate or mitigate the progression of a severe accident (note that prevention measures have been addressed in the system evaluation of core damage frequency); and

- Categorizes the end states of the resulting sequences into groups that can be assessed for their effect on radionuclide release magnitude and timing.

The first objective was achieved by representing the containment event tree as a series of chronological occurrences based upon MAAP runs and NRC code timing results. Some compromise to time phasing occurs where two events are mutually dependent upon each other. These are minimized, however, and the event tree generally represents the timed sequence of events from initiator to sequence end state.

Therefore, the analysis implements the containment event tree assessment in a time phased approach. The first time phase involves occurrences up to vessel breach i.e., including opportunities for in-vessel recovery. The second time phase covers the period from vessel failure or arrest in-vessel until the intermediate term phenomena have occurred. This can be visualized as being approximately 3 to 15 hours after vessel challenge. The third time phase includes longer term phenomena such as containment heat removal and reactor building response. These time phases may overlap in certain accident scenarios.

The remaining objectives were satisfied by using a sufficiently large number of top events and the use of functional fault trees to describe qualitatively and quantitatively the interelationship among mitigating systems, operator actions, and the resulting end states.

The desire to use a thorough approach that emphasizes functional response and uses fault trees to develop the detail of each node was reinforced by NUREG-1150 (Draft) peer review comment that stated:

> *By using the large event tree approach (of NUREG-1150) the containment event tree becomes less of a tool for understanding containment functions. The fact that there are front line containment functions supported by various individual systems and features becomes somewhat lost. The use of a smaller functional event tree with supporting fault tree logic would probably provide a more manageable product capable of providing greater insights.*
>
> *This comment was further expanded in the Special Committee report.*
>
> *It seemed to us that this level of detail exceeded understanding of the phenomena involved, and implied greater insight into the processes assumed to be taking place than was justified. When confronted by the need to quantify poorly understood phenomena, it is certainly necessary to dissect the problem carefully to ensure that important aspects are not overlooked. But this practice should be restricted to assisting the thought process, and the final quantification should be at a scale commensurate with the overall understanding.*

In addition to the need to have detail commensurate with the available data and the ability to communicate that data, another important insight has been gleaned from the NUREG-1150 peer Review [Ref. 85]:

> *The containment event tree should make allowance for accident management actions, for example, attempting to recover cooling and to protect vessel and containment integrity. Such actions are likely to substantially change the course of events and could significantly affect risk.*

The containment event tree (CET) is a tool for identifying and analyzing the spectrum of accident scenarios which may evolve following postulated core damage accidents. By considering the active and passive mitigating functions which can occur after a significant amount of core degradation, end states are identified in which the primary containment maintains its integrity or functionability. It has been recognized, since the publication of WASH-1400, that there can be a significant conservatism in the reactor plant risk estimates if the containment functionability is automatically assumed to be ineffective following postulated core degradation or melt sequences. Therefore, CETs are developed and quantified in order to provide a realistic and systematic assessment of:

- The relative possibility of successfully mitigating postulated accidents
- The severity and timing of associated radionuclide releases from a degraded core accident.

The following were used as input to the CET models:

- Containment Walkdown Results
- P&IDs of Containment Control Systems
- Drawings of Containment Structure and Penetrations
- Technical Specifications
- Containment Leak Data
- Operating Experience
- Containment Structural Analyses
- EOPs (Including Containment Control)
- Level 1 Analysis and Results
- Deterministic Model (e.g., MAAP).[1]

---

[1] The NMP2 IPE utilizes the MAAP code for plant specific analyses of containment challenges. However, industry experience and staff positions on phenomenological uncertainties are also taken into account.

The CET provides a characterization of the state of the containment from the time of the initial core damage to either mitigation of the accident within the RPV or penetration of the RPV. The core melt progression sequences are also followed through their potential interaction with the containment to states involving either: (a) successful mitigation within the containment; or, (b) a radionuclide release.

Given the entry states, the CETs model the containment response (i.e., the core and containment conditions which could affect the accident progression paths and challenge the containment), plus the active and passive mitigative capability of the plant systems to terminate or reduce the radionuclide release.

In the development of the CET, the important factors which affect the consequences for an accident are considered. Consequences in this context are measured in terms of the magnitude and timing of the radionuclide release. The primary focus of the "back-end" analyses is on containment failure mode and release timing rather than on source term analysis. The release and transport of both radioactive material inside containment and that released to the environment are tracked for future input to the accident management process. The identification of the containment failure mode and timing is generally used as an indicator of the type of response that can either mitigate or reduce containment failure probability.

Containment Event Tree analytic approaches consist of two basic approaches:

Approach 1: Segregate all containment active systems from the core melt progression phenomena and bin the results of the analysis of all containment active systems before entering the containment phenomena trees.

Approach 2: Treat both active containment systems and phenomena in the same tree.

The approach chosen could be a function of the type of containment and the emergency operating procedures used. The following discussion and conclusions are for Nine Mile Point 2 which has the BWROG EPGs implemented at the site. It is judged most useful to treat both the systems and required operator actions together in the CET.

4.5.2 Description of CET1, CET2, and CET3 Models

Three types of containment event trees are used to characterize various containment challenges. The Nine Mile Point 2 IPE directly links the front-end to back-end portions of severe accident sequences through directly linked event trees. These trees transfer the support state conditions from the front-end to the back-end trees and include considerations of preventive or mitigative features, as well as timing considerations. Three different containment event tree structures, each linked to the front-end trees, are used to properly handle cases with containment failure before core damage, core damage before containment failure, and containment bypass sequences. The CETs include:

- Class I and III CETs: Containment initially intact. These sequences are characterized by an initial loss of coolant makeup to the reactor vessel that leads to core damage.

- Class II and IV CETs: Containment initially failed or seriously challenged before core damage. For these classes of accidents, the primary containment boundary would fail before or at the time the molten core penetrates the reactor vessel. In Class II accident sequences, the inability to remove heat from the containment results in heat up of the suppression pool and a gradual containment pressurization. A more rapid pressurization is expected for Class IV accidents (e.g., ATWS). Reactor power remains above decay heat levels so that the amount of energy transferred to the suppression pool exceeds its heat removal capacity.

- Class V: Containment bypassed and a direct release path established from the RPV to the reactor building. The Class V CET is used to evaluate two general types of core melt scenarios: (1) LOCAs outside containment for which coolant makeup to the reactor vessel has failed and leads to a core melt event with a direct release pathway from the vessel to the reactor building; and, (2) an interfacing LOCA.

Examples of the three generalized types of CETs are given in the following figures:

Initiating Accident	CET Characteristic	Example CET Figure
Class I & IIIA; B, C & IIT	Containment initially intact	4.5-3
Class IIA, IIL, IIV, IIID & IV	Containment initially failed at time of core melt initiation	4.5-4
Class V	Containment bypassed	4.5-5*

* See Section 4.5.4 for the description of the containment bypass event tree.

Figure 4.5-2 summarizes the RISKMAN CET framework and the key elements of the CET documentation: (1) Top events; (2) Rules for selecting split fractions; (3) Split fractions. .

4.5.3 Description of CET 1 and 2 Functional Nodes

The CET structure includes event tree nodes that address the following four aspects of severe accidents that are considered important in characterizing a radionuclide release:

- Core damage accident class (i.e., the entry state to the CET);

- Mitigating system response including operator actions (post core melt);

- Containment response, including pressures, temperatures and possibly failure location, path, and size, if appropriate;

- Reactor building response including failure size and location which are sequence dependent;

- Phenomenological effects that can alter any of the above.

Because of the large number of interrelated degraded core accident phenomena which must be considered, the process of evaluating the severe core damage events and their effects on containment can be a complex and iterative task. Given the entry states, (i.e., core damage sequences from Level 1), the CETs model the containment response (i.e., the core and containment conditions which could affect the accident progression paths and challenge the containment), plus the active and passive mitigative capability of the plant systems to terminate or reduce the radionuclide release.

The functional event nodes of the CET which are used to describe these accident management opportunities include the following:

- Containment Isolation (IS)
- Reactor pressure status (OI/OP)
- Coolability of core debris within the RPV (IR/RX)
- Combustible Gas Venting (GV)
- Containment Isolated and Intact (CZ/CE)
- Coolant injection for temperature control of molten debris (TD/TR)
- Passive mitigation: containment of the debris
- Containment Flooding (FI/FC, CX/CY, FD/FB)
- Containment heat removal (HR, VC)
    - RHR
    - Venting
- Suppression Pool Bypass (SP/SN)
- Containment breach size (NC/NF)
    - Leakage
    - Overpressure failures
- Location of containment breach (DI/DC, WR/WW)
    - Drywell (DI/DC)
    - Wetwell airspace (WR/WW)
- Reactor Building effectiveness (RM/RB).

These top level functional events are described in more detail below.

Containment Isolation - IS

Consistent with the NRC preference indicated in NUREG-1335, containment isolation is the first system nodal decision point of the CET. This node considers the following as they effect the status of containment isolation prior to core damage:

- The pathways that could significantly contribute to containment isolation failure,

- The signals required to automatically isolate the penetration,
- The potential for generating the signals for all initiating events,
- Consideration of testing and maintenance, and
- The quantification of each containment isolation mode (including common-mode failure).

The "IS" node is used to assess whether the NMP-2 containment has been successfully isolated given the core damage challenge identified in the Level 1 PRA. Because the NMP-2 containment isolation system has high reliability and the containment is required to be inerted, there is high confidence that the containment is also isolated.

Initiating events that include containment breach in the Level 1 are binned as part of the Level 1/Level 2 interface process and are transferred to CETs that bypass the IS node (i.e., Class IIA, IIL, IIV, IV, IIID, and V sequences fall into this category).

Operator Depressurizes the Reactor Vessel - OI/OP

This heading represents the manual or automatic action of depressurizing the RPV. The operator recovery action to depressurize the reactor allows low pressure system access to the RPV. The upward path at this node represents successful depressurization and the down path models failure.

The status of RPV pressure can have a profound impact on the ability to successfully mitigate a severe accident and the subsequent containment response. Therefore, the determination of the RPV pressure is a key to understanding subsequent active and passive mitigation capability.

Core Melt Progression Arrested In-vessel - IR/RX

The containment event tree node (IR/RX) addresses the ability to arrest core melt within the reactor vessel. Specifically, success requires recovery of coolant makeup to the reactor vessel so that cooling may be re-established to prevent further degradation of the fuel integrity. The time window for recovery (if successful) occurs between core damage initiation and the time when the core melt progression cannot be halted within the RPV. Recall that any recovery actions prior to the onset of significant core damage are treated in the Level 1 analysis. The likelihood of in-vessel recovery is dependent on the characterization of core degradation as well as the time available for in-vessel recovery. The time window is dependent on the specific core damage scenario. This can be one hour to several hours depending upon the analytic model used.

The IR/RX nodal assessment addresses:

- The operator action to inject to the RPV
- The equipment availability to support injection

- Phenomena which may preclude successful arrest of the core melt progression in-vessel.

The makeup sources to ensure debris cooling in-vessel consist of the same sources examined in the Level 1 system evaluation. Therefore, the "IR/RX" node is primarily an examination of repair and recovery actions that can occur in the time window of in-vessel core melt progression. Note that the success of "IR/RX" is also strongly dependent on the successful RPV depressurization of the previous node, (OI/OP). In turn, IR/RX also has strong influences on subsequent CET nodes such as "TD/TR", availability of water injection to the containment after RPV breach. The "TD/TR" node examines water recovery over the much longer time frame of 2 hours up to 10-20 hours.

Combustible Gas Venting - GV

This node addresses the possibility that both the containment may have a combustible gas mixture and no operator actions would be taken to mitigate the condition.

This CET heading characterizes the potential for venting the containment during accident sequences in which combustible gases may be present. The upward branch defines the condition where the vent has been opened to control combustible gas mixtures, given the unlikely situation that the containment is deinerted. The downward path represents cases in which the containment remains inerted or the vent is not otherwise opened.

Early Containment Failure - CZ/CE

This node addresses postulated severe accident phenomena that can result in an energetic failure of containment during the core melt progression. Energetic containment failure modes resulting from the core melt accident sequence initiator and the subsequent core melt phenomena at the time of initial RPV breach due to debris attack are estimated to have potentially high radionuclide releases. These can be considered also "early" for Classes I and III. Event heading (CZ/CE) describes the condition of the containment after a failure of the primary system. In the upward path the containment has remained intact during the initial stages of core melt progression up through RPV blowdown, while the downward path depicts an energetic failure of the drywell induced at approximately the time of a loss of primary system integrity.

The containment is a primary defense in retaining core melt fission products. The failure modes included in the early containment failure include the following:

- Containment pressurization due to RPV blowdown causes rapid containment pressure rise above capability
- Steam explosion
- Direct containment heating
- Recriticality
- Hydrogen deflagration in a deinerted containment.

The structure of the CZ/CE model divides these phenomena into in-vessel and ex-vessel phenomena, depending upon the success or failure of the IR/RX node.

Wherever possible, the NMP2 specific MAAP model is used to describe the boundary conditions of containment challenges. Engineering analyses regarding the capability of the NMP-2 containment to withstand the various energetic accident phenomena were not performed. Rather, industry studies and staff positions on phenomenological uncertainties were taken into account to assign failure probabilities that are deemed representative of the NMP2 Mark II containment. An assessment of the NMP-2 containment capability in response to slower developing overtemperature and overpressure scenarios (e.g., loss of debris cooling, loss of containment heat removal) was performed and is documented in Section 4.4. Those scenarios are inherently different than those modeled by the CZ/CE node, and thus, are modeled under a separate node (NC/NF and TD/TR) so that the different potential releases can be accounted for.

Active Mitigation Temperature Control - TD/TR

If core melt progression results in RPV bottom head breach, a portion of the debris would be discharged and distributed to the following locations:

- The drywell pedestal floor: A small residual amount will end up on the floor.

- The drywell outside the pedestal: Because of possible debris entrainment during the RPV blowdown process, molten debris can be transported to the drywell even though the sunken pedestal is present. (This occurs for cases with depressurization failure.)

- The wetwell inside the pedestal: The molten debris will be both discharged into the suppression pool via the RPV blowdown through the in-pedestal downcomers (not explicitly modeled in MAAP (conservative)), and debris on the pedestal floor will overflow into the downcomers into the wetwell.

Some residual material may also be left behind in the RPV.

Subsequent to debris discharge from the RPV, containment challenge may occur from high temperatures in the drywell or a combination of high temperatures coupled with high pressures due to steam generation and noncondensible gas generation. The high drywell temperatures may result from two potential sources:

- The entrained debris that reaches the ex-pedestal portion of the drywell

- The residual fuel debris in the RPV.

If water can be directed to each of these heat sources, then drywell temperatures can be controlled.

Injection of water into the containment and/or the RPV can mitigate the consequences of a core melt and assist in preventing either of these failure modes if containment pressure control is also achieved. Each of these are discussed below:

- Drywell Sprays

  Drywell sprays can mitigate the consequences of a potential core melt accident. The sprays can perform three functions, the two most important of which are: (1) scrubbing fission products that are not otherwise scrubbed (i.e., in the case where the suppression pool is bypassed); and, (2) providing water to cool the entrained core debris on the drywell floor. In this mode of operation, containment failure could be prevented by termination of 1) drywell wall heating and the associated temperature induced containment failure, and 2) noncondensible gas generation due to core concrete reaction.

- Vessel Water Injection

  RPV water injection can perform some of the same functions as spray operation mentioned above, i.e., scrub fission products from the debris, prevent containment overtemperature failure, and reduce the core concrete reaction by quenching the debris. The RPV injection systems can also perform another function as well: water injected into the vessel will generally cool the RPV structures. This cooling may prevent fission product revaporization from the RPV.

  The systems that might perform the function of coolant injection post core melt at NMP-2 include:

  - Fire System - To RPV
  - Control rod drive - To RPV
  - Low pressure coolant injection - To RPV or Sprays
  - Low Pressure Core spray - To RPV
  - High Pressure Core Spray - To RPV
  - Condensate/Feedwater - To RPV
  - Condensate transfer - To RPV
  - Service water cross-tie - To RPV or Sprays
  - ECCS keep-full systems to RPV.

Containment failure size and location is dependent on the status of this CET function (see also the discussion of NC/NF and DI/DC).

Operation of the vessel water injection systems after vessel failure will act to cool the core debris that remains in the RPV and on the pedestal floor (small amount). The post-core melt water injection will cool the RPV and prevent the drywell from reaching very high temperatures due to RPV radiative heating. For Class II, IIID, IV and V with the containment already failed, preventing the drywell from overheating will prevent overtemperature failure of the drywell head seal, the drywell liner, or the penetrations. An added benefit for vessel water injection after vessel breach is the potential to scrub ex-vessel fission products via the water overburden.

Containment Flood - FI/FC, CX/CY, FD/FB

NMP2 Procedures (based on the BWROG EPG Revision 4) specify containment flooding from external water sources when the RPV water level cannot be restored.

These nodes address the question of whether operator actions will be taken to flood the containment with external water during the core melt progression or whether the actions will be to maintain suppression pool level at approximately LCO limits.

Containment Heat Removal - HR

This node addresses the availability of the RHR system and the operator action to initiate the system for containment heat removal.

The NMP-2 Mark II containment system is provided with significant heat capacity and heat management capabilities. The management of heat in the containment prior to, during, and following a severe core damage event directly affects containment response. The NMP-2 containment heat capacity can be classified as both active and passive. The passive capacities include the suppression pool and the containment structure. The active heat management capabilities include the RHR system, the RWCU system, venting, and drywell coolers. This event tree node addresses all heat management capabilities (except venting and use of the main condenser), but the dominant influence on successful containment heat removal post core melt is the RHR system. Severe accident effects on the performance of the RHR system (e.g., steam binding) are considered in the model. (Note containment venting is discussed separately below.)

The RHR system, operating in the suppression pool cooling mode, can maintain long term containment integrity through adequate containment heat removal if other failure modes can also be mitigated. With the RHR system operating during the course of a core melt accident, the containment pressures and temperatures can be maintained within the structural failure criteria of the containment. As a result, the consequences of a radioactive release to the environment can be prevented.

The upward branch at this event tree node represents successful containment heat removal via the RHR system operating in the suppression pool cooling mode. The downward branch models failure of containment heat removal.

Wetwell Vent - VC

This event heading characterizes use of the wetwell vent to relieve containment pressure, and provide an alternate path for containment pressure control. Venting provides the operator a means of removing decay heat and non-condensibles and maintaining the integrity of the containment. At this node, the upward path represents successful use of the vent, while the downward path represents venting failure due to mechanical faults, inadequate procedures, or operator error. Severe accident effects on the performance of the wetwell vent (e.g., high differential pressure prevents valve operation) are considered in the model.

Suppression Pool Bypass - SP/SN

This node is an assessment of hardware availability to preserve the suppression function of the wetwell.

If the operator is unsuccessful in maintaining the heat management functions as described in the preceding section, wetwell venting would be required to maintain containment integrity. As a result, this event heading examines the potential for suppression pool bypass that would allow the release of radionuclides from the reactor vessel to pass directly from the drywell to the wetwell air space without the benefit of suppression pool scrubbing during venting. The upward branch at this event tree node represents no bypass, while the downward branch models a scenario in which releases pass directly to the wetwell air space and out the wetwell vent or a wetwell air space failure.

Containment Response Integrity - NC/NF, DI/DC, WR/WW

These nodes address the size and location of the containment failure.

For the purposes of a containment performance evaluation under severe accident conditions, it is useful to have a criterion to describe the adequacy of containment integrity as a function of pressure and temperature within containment. Using severe accident profiles from published Mark II severe accident analyses and the evaluations of the NMP2 containment integrity under such conditions, criteria for the containment remaining intact can be established. These criteria are based upon published BWR containment assessments using the following priorities:

- NMP-2 specific
- Mark II specific
- BWR specific.

A containment capability profile which represents a reasonable interpretation of the plant specific analyses for characterizing the severe accident performance of the NMP-2 Mark II containment is presented in Section 4.4. An NMP2 containment capability assessment has been performed which identifies the likely containment failure modes and conditions causing these failures for three distinct extreme cases: (1) high temperature, (2) high pressure, and an (3) intermediate point for each. These three cases are used to establish the realistic performance limits estimated for the NMP-2 containment.

The containment failure location and its size will impact the calculated radionuclide releases. Failure location and size also depend on the core melt accident sequence and the operability of mitigating systems. The NMP2 specific analysis considers the effects of high temperatures and pressures on seals, valves, hatches, and other key areas of the containment structure (e.g., drywell head area).

A more complex CET could examine the possibility of a small containment breach subsequent to any severe accident, even those that are adequately mitigated by coolant injection and containment heat removal. The simplification that is used here is that the resultant leakage would be relatively small and within the capacity of the SGTS, such that little if any release greater than the DBA would be calculated. NUREG-1150 studies have shown that such small leakages make no measurable contribution to the assessed public risk.

Continued Inventory Makeup - MU

This node considers the effect of harsh environment (e.g., humidity, temperature) following containment failure or venting on the availability and survivability of injection systems and components.

Reactor Building - RM/RB

BWRs have a secondary containment. The secondary containment at NMP-2 consists of the reactor building and auxiliary bay structure. The reactor building can act to retain a significant fraction of the radionuclides released from the primary containment for certain severe accident scenarios. Time averaged decontamination factors for the reactor building vary between 1.0 and much greater than 10. The determination of whether the reactor building is effective is made at this node.

Contributors to the determination of reactor building effectiveness include the following:

- Reactor building integrity after containment failure
- Standby gas treatment system (SGTS) operation
- Fire sprinkler operation
- Hydrogen combustion in the reactor building
- Reactor building integrity after hydrogen combustion.

The down branch of the reactor building node implies minimal effectiveness of the reactor building on the retention of fission products due to failures such as:

- Combustion of gases in the reactor building causing high temperature and minimum or zero retention.

- Direct pathway from the containment failure location to the blowout panels with minimal interaction within the reactor building.

4.5.4 Level 2 PRA Success Criteria

The Level 2 containment event tree (CET) describes accident progression from initiation of core damage to successful mitigation or release. Each node within the CET requires a definition of what success implies. As part of the containment event tree development and the Level 2 evaluation of core melt progression and mitigation there will also be successful end states in which radionuclide releases from the reactor building will be prevented or substantially contained. As part of the evaluation, these success states require a consistent criteria to be applied in order to allow the quantitative assessment of the radionuclide release frequency and magnitude.

For these reasons it is important to establish the success criteria to be used in the qualitative and quantitative analysis. Failure to meet these criteria may result in extreme containment conditions (e.g., excessive temperature and/or pressure) that could challenge containment structural integrity.

4.5.4.1 Overview of Level 2 Success Criteria

The overall success criteria for the Level 2 evaluation can be defined in terms of successful achievement of the following safety functions:

- RPV integrity,
- Containment integrity, and
- Reactor Building effectiveness.

These overall success criteria describe the bases upon which each protective barrier is examined to determine its effectiveness in radionuclide release mitigation. Table 4.5-1 summarizes the success criteria used for these top level safety functions.

4.5.4.2 Functional Success Criteria

Using the overall success criteria established above, the next step is to interpret these success criteria in terms of key functions that can be explicitly defined and quantified within the Level 2 model.

The key functions contained in the CET model as top events include the following:

- Containment Isolated (IS)
- RPV Depressurization (OI/OP)
- Core Melt Arrested In-Vessel (IR/RX)
- Combustible Gas Venting (GV)
- Containment Remains Intact (CZ/CE)
- Injection Established to RPV or Drywell (TD/TR)
- Containment Flooding Initiated (FI/FC)
- Containment Intact During Flooding (CX/CY)
- Containment Flooded and Drywell Vented (FD/FB)
- Containment Heat Removal Maintained (HR)
- Containment Venting (VC)
- Suppression Pool Not Bypassed (SP/SN)
- No Large Containment Failure (NC/NF)
- Makeup Remains Available (MU)
- Drywell Intact (DI/DC)
- Wetwell Airspace Failure (WR/WW)
- Reactor Building Effectiveness (RM/RB).

Table 4.5-2 summarizes the success criteria for each functional node in the containment event tree (CET).

Table 4.5-1

OVERALL LEVEL 2 SUCCESS CRITERIA

Protective Barrier	Success Criteria	Reference
RPV Bottom Head Integrity	RPV bottom head integrity is assured if the following criteria are met: a) RPV internal temperature must be maintained below 4000°F; if not, then RPV structural failure and depressurization is assumed at the location of high temperature. b) Bottom head temperatures and local penetration are maintained below their melting temperature. If molten debris is calculated to fail instrument tubes or CRD penetrations, then RPV structural failure and depressurization is assumed.	 [94]
Containment Integrity	The containment integrity is assured if the following criteria are met: • No energetic, early containment failure modes occur;(1) • Pressure and temperature within the containment must be maintained below the best estimate containment capability (see Section 4.4); • Containment is isolated (i.e., no unisolated openings greater than 2 inches in diameter); and If these criteria are not met the containment is assumed to fail. The size and location of the failure are treated probabilistically and are discussed in Section 4.4, containment capability.	 [95] [95] [95]

Table 4.5-1

OVERALL LEVEL 2 SUCCESS CRITERIA

Protective Barrier	Success Criteria	Reference
Reactor Building Integrity and Effectiveness	The reactor building is considered effective in mitigating radionuclide releases, if the reactor building isolates and either of the following two criteria are met:	
	• Reactor building integrity is maintained if the reactor building pressurizes to no more than 0.25 psig. If not, then the reactor building blowout panels floor are assumed to have opened.	[96]
	• The reactor building is assumed effective in removing radionuclides if the following criteria are met. (The estimated decontamination factor DF, is to be determined by MAAP.):	
	- No structural breach in the reactor building, allowing free communication with the environment, caused by events such as hydrogen detonation in the reactor building.	
	- No natural circulation paths with chimney effects are established within the reactor building that could drastically reduce residence time and retention within the reactor building.	[94]
	- No direct release pathways are created that cause a release from the containment or RPV to the environment with little or no holdup.	

(1) Early, energetic containment failure modes may include the following:

- In-vessel steam explosion
- Ex-vessel steam explosion
- Direct containment heating (DCH)
- Hydrogen detonation
- Pressure spike at blowdown following RPV breach.

Table 4.5-2

SUCCESS CRITERIA

CET Functional Node	Success Criteria
Containment Isolation (IS)	The successful (i.e., the up branch) of the containment isolation node (IS) is satisfied if the containment penetrations that communicate between the drywell (or wetwell) atmosphere and the reactor building (or environment) are "closed or isolated." The criteria used to satisfy this requirement of "closed and isolated" is that no line, hatch, or penetration has an opening equivalent to 2 inches or greater in diameter. This implies that all containment penetrations are adequately sealed and isolated during the entire accident progression through the end of Level 1 until (1) a safe stable state is reached; or, (2) the accident conditions exceed the ultimate capability of containment, or (3) containment is vented per EOPs.
RPV Depressurization (OI/OP)	This function questions whether the operator depressurizes the RPV after core damage but before vessel breach has occurred. Success of this action would allow low pressure injection, if available, and would minimize the challenge to containment due to a high pressure RPV rupture. The functional success criterion for this node is defined as having the RPV depressurized (i.e., less than 100 psig) until core melt is arrested in-vessel or until the RPV is breached by debris attack. The success of the depressurization function for the RPV following core damage initiation is similar to the criterion established in the Level 1 analysis, i.e., prior to core damage. However, there are additional phenomena (i.e., non-condensible gas generation contributing to a high containment pressure that prevents SRV operation, and potentially very high containment temperatures of the SRVs) which can occur during the accident progression beyond core damage and pose further challenge to the operator's ability to depressurize the RPV. The success criteria is to depressurize the RPV to less than 100 psig. The success criteria, in terms of systems, is the same as that used prior to core damage, i.e., • Any 2 SRVs [Ref. 95, 97] or • Failure of the primary system due to high temperature during core melt progression.[1] or • A large or medium LOCA. Other alternatives, such as the RCIC steam lines and reopening the MSIVs, may be available but are not credited in this analysis.

Table 4.5-2

SUCCESS CRITERIA

CET Functional Node	Success Criteria
Arrest Core Melt Progression In-vessel (IR/RX)	In-vessel recovery or arrest of core melt progression addresses the ability of the operating staff to restore adequate core cooling from the time the end state of the Level 1 PRA occurs (i.e., RPV water level less than 1/3 core height and decreasing) until restoration of water injection make-up cannot prevent the breach of the RPV bottom head by debris. As part of the definition of success, it is also useful to define what constitutes failure to maintain the RPV intact. The two primary failure modes that have been identified in the literature include: • Local penetration seal failure due to debris heat up and local failure at welds, [Ref. 94] • Creep rupture failure of the entire bottom head. [Ref. 98] The MAAP evaluation calculates that the RPV integrity would be challenged by debris contact with local penetration welds. This is supported by experiments by R. Leahey (RPI) which indicate for PWRs that drain plug configurations are susceptible to failure [Ref. 99]. This configuration correlates to the BWR instrument tubes or CRD seals. The base quantification assumes that RPV failure occurs at local penetrations. The large, bottom head failure scenario is treated as a sensitivity case. Preventing the core melt from progressing outside the reactor pressure vessel requires the timely introduction of water onto the debris and intact fuel assemblies. Both timing and system requirements must be defined as part of the success criteria. There are differences in core melt progression models regarding the ability to recover adequate cooling under different circumstances. These vary from no credit for retention of debris in-vessel after core <u>melting</u> has begun (MAAP), to substantial credit for recovery even after debris has accumulated in the bottom head (MARCH). The best estimate success criteria used in this evaluation are based on the time available from the initiation of core degradation until just before substantial core relocation occurs. This typically is on the order of 30-40 minutes. In terms of system requirements, coolant injection is assumed necessary to re-flood the RPV to above top of active fuel. It is judged, based on deterministic calculations, that this can be accomplished using makeup systems (identified in the EOPs) with capability greater than approximately 1000 gpm.[5]

Table 4.5-2

SUCCESS CRITERIA

CET Functional Node	Success Criteria
Combustible Gas Venting (GV)	The functional success criterion at this node is that the containment vent and purge lines are opened to allow combustible gas mixtures to be removed from containment. The downward path of GV in the CET implies that combustible gas venting has not been initiated. Therefore, on the downward path either of two conditions may exist: • The containment is inerted[3] or • A combustible gas mixture is present and venting was not effectively initiated. The probabilistic evaluation of which of these two states may occur on the downward branch are treated in the Containment Remains Intact Early (CZ) node. Note that hydrogen recombiners are of such low through-put capacity that the amount of hydrogen potentially generated during a severe accident cannot be effectively processed by the recombiners. Specifically, hydrogen fractions of the containment atmosphere of greater than 12% can be anticipated within the first 2 hours of core melt progression. Therefore, the hydrogen recombiner system is considered not to be effective in preventing a hydrogen deflagration in a severe accident situation. Hydrogen combustion that could lead to containment failure is prevented by either of the following: • Inerted operation with no oxygen intrusion during the accident [Ref. 101] • Combustible gas purging and venting through the purge and vent lines [Ref. 100]. If both of these success paths fail, hydrogen detonation is assumed to occur, resulting in containment failure. The location of the failure is assumed to be in the drywell head region and is classified as a large failure.

Table 4.5-2

SUCCESS CRITERIA

CET Functional Node	Success Criteria
Containment Remains Intact (CZ/CE)	The functional success criterion for the containment intact node is that the containment drywell retains its pressure capability and that no early containment failure modes compromise the containment integrity. The early containment failures modeled by the CZ/CE node are characterized by phenomenological events (e.g., steam explosions, missile generation, direct containment heating) that are believed to challenge containment integrity relatively quickly following core melt. Late containment failures, modeled in nodes, are characterized by extreme pressure and temperature conditions that develop slowly over the course of the accident due to inadequate containment heat removal. Note that successful prevention of early containment failure does not necessarily preclude late containment failure. Therefore, successful prevention of early containment failure requires the following: • No direct containment heating (direct containment heating is precluded if the RPV is already depressurized) [Ref. 101] • No ex-vessel steam explosion [Ref. 120] • No failure of vapor suppression prior to RPV blowdown (i.e., the suppression pool is not bypassed and if no more than one set of Drywell to Wetwell vacuum breaker fails open) • No in-vessel steam explosion (i.e., in-vessel steam explosions are precluded if either the RPV is at high pressure, greater than 100 psig, or the core does not fragment into fine particles before dropping onto the bottom head) [Ref. 102, 103] • No high pressure spike sufficient to cause containment failure occurs at the time of vessel melt-through (i.e., extreme pressure spikes are precluded if the RPV bottom head penetration fails locally;or the RPV remains at low pressure) • No hydrogen deflagration or detonation (i.e., if the containment remains inert or effective combustible gas vent was operated successfully, then, hydrogen detonation or deflagration is guaranteed not to occur) • No recriticality due to an unusual core configuration that may be achieved during the melt progression; [Ref. 105] If these failure modes cannot be prevented, containment failure is assumed to occur. The failure location is assumed to be in the drywell head region and is classified as a large failure.

Table 4.5-2

SUCCESS CRITERIA

CET Functional Node	Success Criteria
Ex-vessel Debris Coolability (TD/TR)	Ex-vessel core debris coolability can be considered to be successful if very high containment temperatures, can be prevented. These are considered preventable if on a best estimate basis a continuous water supply is available to the debris with a flow rate of greater than 1000 gpm. Table 4.5-3 is developed to demonstrate the basis for the success paths in TD/TR. As shows in the Table 4.5-3, any water injection source (CS, LPCI, DWSPRAY) alone will result in prevention of a drywell temperature failure. (Prevention of other failure modes such as overpressure require operation of RHR or venting in addition.) For the ATWS cases (TR), containment is assumed to have previously failed due to the Level 1 sequence definition. If the failure is in the wetwell, then no subsequent water injection source is necessary to prevent a drywell overtemperature failure. The two methods that may provide adequate coolant injection to the debris bed include continued make-up to the RPV and initiation of drywell sprays.[a] Failure at this node could result in either of the following occurring: • High temperatures in the drywell, or • Excessive concrete ablation causing pedestal structural failure or basemat penetration. These effects would influence the integrity of containment. Note that inadequate water injection will be modeled for the purposes of consequence evaluation as inducing a drywell failure high in the containment. [Ref. 89, 101] However, there are some models that indicate that concrete attack and non-condensible gas generation will not be terminated even if substantial water injection is available to the debris. The temperatures in the drywell will be acceptable, but continued non-condensible gas generation will occur. In this case, venting would be an adequate mitigation measure. The base case model includes debris cooling when water can be supplied to the debris or debris enters the suppression pool.
Containment Flooding (FI/FC)	Success at this node implies that the containment flooding contingency procedure has been initiated by the operating staff and that a system of adequate flow capacity from external sources is available to implement the procedure. In addition to these two requirements, the instrumentation must be available to initiate the flood operation.

Table 4.5-2

SUCCESS CRITERIA

CET Functional Node	Success Criteria
Containment Remains Intact (CX/CY)	The success branch of the CX/CY node occurs if two situations can be prevented: • Blowdown of the RPV into a reduced free volume (i.e., the increased water level creates a reduced free volume that results in a decreased capability of the containment to accept blowdown loads.) and • Core melt progression causing RPV failure and a large steam vaporization. These two failure modes are somewhat dependent upon the relative timing of containment fill versus core melt progression. In addition, the effects are dependent on the following: • Whether the RPV is depressurized allowing injection of external water sources (Node OI/OP), and • Whether containment flooding is accomplished through injection nozzles outside of the RPV (i.e., drywell sprays and RHR suppression pool return lines).
Containment Flooded Above Debris (FD/FB)	This node evaluates the possibility that the operator suspends containment flooding because the staff is unable to maintain containment conditions within prescribed limits described in the EOPs. Success at FD/FB includes either RPV venting or drywell venting. Since it is presumed that containment pressurization will occur during the latter stage of flooding as a result of a diminishing drywell volume, the operator will be required to establish a drywell or RPV vent path (i.e., > 8 inch equivalent diameter). Drywell or RPV venting can have varying degrees of releases associated with it depending on the following: • When in the containment flood process venting is required, and • Whether success of RHR and injection is effective in controlling containment pressure Success at this juncture in the model is defined as the continuation of the flooding evolution with containment conditions remaining within the limits of the Maximum Primary Containment Water Level Limit (MPCWLL).
Containment Pressure Control (see node descriptions HR and VC below)	Successful containment pressure control is achieved if either of two functional nodes are successfully satisfied; (1) RHR containment heat removal or (2) Containment venting. Because these have different potential impacts on the radionuclide releases they are treated in separate nodes.

Table 4.5-2

SUCCESS CRITERIA

CET Functional Node	Success Criteria
(1) RHR Containment Heat Removal (HR)	Successful containment pressure control is unattainable using RHR[4] suppression pool cooling if either of the following conditions occur: • No debris cooling (in-vessel or ex-vessel) • "Early" containment failure modes.
(1) RHR Containment Heat Removal (HR) (con't)	RHR has the capability to remove heat from containment through the RHR heat exchangers. This capability requires: • A flow path from the suppression pool • One RHR pump • One RHR pump heat exchanger • Service water to cool the heat exchanger • A return flow path to: - The suppression pool - The RPV - The drywell spray (wetwell spray flow rate is considered to low). • Bypass of the low RPV water level (2/3 core height) interlock if not using RPV return Failure at this juncture in the sequence implies insufficient containment heat rejection to the environment and that the continued decay heat generation could subject the containment to continued pressurization. This condition may eventually cause structural failure, which could subsequently threaten continued successful core coolant injection. Note that RHR success is a moot point if adequate injection to the core or debris has failed. This is because of high temperatures from debris radiative heating or high pressure from non-condensible gases will cause drywell failure.

Table 4.5-2

SUCCESS CRITERIA

CET Functional Node	Success Criteria
(2) Containment Venting (VC)	The capability to vent the wetwell is a valuable supplement to the containment pressure control systems. As pressure and temperature increase, there is decreasing confidence in the ability to maintain the integrity of the containment pressure boundary. By instituting a controlled vent of the containment atmosphere, it is possible to maintain long term containment integrity by providing a viable means of containment pressure control and heat removal. Venting also constitutes a viable mitigative action to minimize the source term released to the environment. Containment venting is successful if it can remove the excess heat and non-condensible gases from the containment and, thereby, maintain the containment pressure within acceptable limits. Adequate pressure control can be obtained by containment venting if the following conditions are satisfied: • Reactivity control exists • No "early" containment failure modes occur • Containment flooding does not eliminate the venting pathways • Vent pathways can be opened and controlled. Based upon deterministic calculations, a containment vent of approximately 8 inches in diameter will provide sufficient vent capability to prevent containment failure for sequences involving the loss of containment heat removal or severe accidents [Ref. 95]. Currently, no vent capability is considered successful for unmitigated or failure to scram events.
No Suppression Pool Bypass (SP/SN)	This node in the CET is used to characterize the magnitude of radionuclides that may escape the containment if wetwell failure or venting occurs. Success means that radionuclides are directed through the suppression pool. Subsequent headings address specific release paths. Success in preventing suppression pool bypass requires that: • No complete set of vacuum breakers remains stuck open; • The suppression pool water level remains above the bottom of the downcomers; and • The downcomers do not rupture or failure to debris attack.

Table 4.5-2

SUCCESS CRITERIA

CET Functional Node	Success Criteria
No Large Containment Failure (NC/NF)	This event examines the size of containment leakage that may be induced by extreme pressure and temperature conditions. The downward path at this event tree node is defined as large leakage or failure, while the upward path depicts either no leakage or the existence of drywell leak paths that prevent further containment pressurization. Any failure of the containment structure greater than 1 $ft.^2$ is considered to be a large containment failure and is modeled as a 2 ft^2 break in the MAAP runs. A small break is assumed to be 1 ft^2 or less in size, and is modeled in MAAP with a leak size of 27 $in.^2$. A small containment breach may be characterized by any of the following breach of containment: • Electrical penetration leak, • Hatch seal leak, • Bellows seal leak, or • Drywell head seal leak: - Thermal degradation - Inadequate pre-load Leak sizes up to 3 $in.^2$ in equivalent area are assumed to present a negligible impact on the course of the accident. The downward branch of the "No Large Containment Failure" node is probabilistically based on the NMP2 specific structural analysis. However, there are certain cases in which failure (i.e., large break) is guaranteed. These cases include the following: • Failure to scram sequences with continued injection and no effective SLC, • No injection to containment, causing high temperature induced failure, • Any early containment failure (e.g., steam explosion, etc.), or • LOCAs plus failure of vapor suppression

Table 4.5-2

SUCCESS CRITERIA

CET Functional Node	Success Criteria
Coolant Makeup Remains Available Post Containment Failure (MU)	This event node is used to examine the availability of water injection to the drywell and RPV following containment failure. Failure of coolant makeup to the debris results in delayed fission product release due to heat up and revaporization of fission products on the RPV internals and containment structures. Releases are reduced if coolant injection can be maintained. The success of coolant makeup following containment failure may be compromised by any of the following: • Harsh environment in reactor building • Steam binding of pumps. • Disruption of injection pathways due to catastrophic containment failure. The same success criteria established for accomplishing the ex-vessel debris coolability (node "TI/TD") influence the analysis of whether functional success is achieved at this node. Alignment of the following injection sources external to the reactor building may be used to achieve success (these systems are not hindered by steam binding or harsh conditions in the reactor building): • SW crosstie • Condensate transfer • Condensate
Drywell Intact (DI/DC)	Containment failure has already been asked in the CET. If containment failure has not occurred, this node is bypassed. If containment failure is determined to have occurred, then, the "DI/DC" node is included to distinguish whether the failure occurred in the drywell (downward branch) or wetwell (upward branch). The probabilistic determination of the location of the failure is determined based on the NMP2 structural analysis for slow overpressure event. Additional guidance is also provided for other accident scenarios as follows: • High temperature induced failures result in drywell failures • Rapid or energetic failure modes are assumed to occur in the drywell (e.g., steam explosions, etc.)
Wetwell Airspace Failure (WR/WW) (Scrubbed Release)	This node appears after the Drywell Intact (DI/DC) node. If the DI/DC node determines that the containment failure occurred in the drywell this node is bypassed. If containment failure occurred in the wetwell, this node distinguishes whether the wetwell failure occurred above or below the wetwell water line. As in the previous node, successfully avoiding a large containment failure requires successful containment heat removal. The probabilistic determination of the location of the failure is determined based on the NMP2 structural analysis for slow overpressurization events.

Table 4.5-2

SUCCESS CRITERIA

CET Functional Node	Success Criteria
Reactor Building Effectiveness (RM/RB)	The reactor building provides a substantial capability to remove particulate fission products from the release pathway for scenarios where the containment has failed. Success of the reactor building to provide a substantial radionuclide reduction (i.e., a factor of 5 to 10 reduction in the radionuclide release magnitude) is based upon any of the following: • Very small containment failures for which the reactor building remains substantially intact • Primary containment failures low in the reactor building for which the release pathway consists of a tortuous route through the reactor building <u>and</u> no other failures are induced due to hydrogen conbustion. • Cases in which substantial fire protection spray is occurring during the release (not credited due to limited area coverage by the fire sprays in the Reactor Building, i.e., cable trays only) [Ref. 52, 107]

Notes to Table 4.5-2

1 Primary system failure may be induced by very high internal temperatures generated by molten debris in an uncooled state within the RPV. Such high temperatures coincident with high RPV pressures may lead to localized failures at weak points high within the RPV.

2 Opening MSIVs or the use of RCIC steam line are not credited because these are not directed by the EOPs, or are of insufficient capacity to lead to depressurization, respectively.

3 For this situation the containment remained inerted and venting was not required. Therefore, in this case, the down branch is not considered as a failure of combustible gas venting but as a continuation of the sequence.

4 Other modes of containment heat removal are not considered effective because interlocks or procedural restrictions under severe accident conditions (e.g., RWCU, Main Condenser)

5 The 1000 gpm criterion is an approximation. There is a comparatively large degree of uncertainty surrounding this issue. However, ORNL calculations seem to indicate that an injection rate close to 1000 gpm initiated at 30 minutes may be sufficient.

6 The drywell sprays for NMP-2 may be ineffective in debris cooling due to the drywell pedestal configuration. This has not been proven but is conservatively not included as successful for direct debris cooling in the model development.

Table 4.5-3

TD/TR SUCCESS AND FAILURE SUMMARY TABLE EXAMINING THREE KEY SYSTEMS AND THREE KEY ACCIDENT TYPES

(MAAP Cases Used as Basis are Shown in Parentheses)

	Injection Source			
Accident Class	LPCI	CS	DW Spray	No Injection
Class IA*	S (Inferred)	S (IAILDNP)	S (Inferred)	F (IAILDNP)
Class ID/III*	S (ID2LD (DW/H 50 hrs.))	S (ID6LD (Inferred))	S (Inferred)	F (IDL8D[2])
Class IV*	S (Inferred[1])	S (Inferred[1])	S (Inferred[3])	S (IVAI-LW)

[1] See MAAP Case IVA-1-LWNP

[2] 37 Hrs. after V Fail.

LEGEND

S = No containment failure induced within 24 hours due to lack of injection. Covers cases with RHR pool cooling and without RHR pool cooling.

F = Containment failure due solely to the status of the injection system identified; i.e., pool cooling or vent has operated successfully.

CONTAINMENT EVENT TREE MODEL DEVELOPMENT

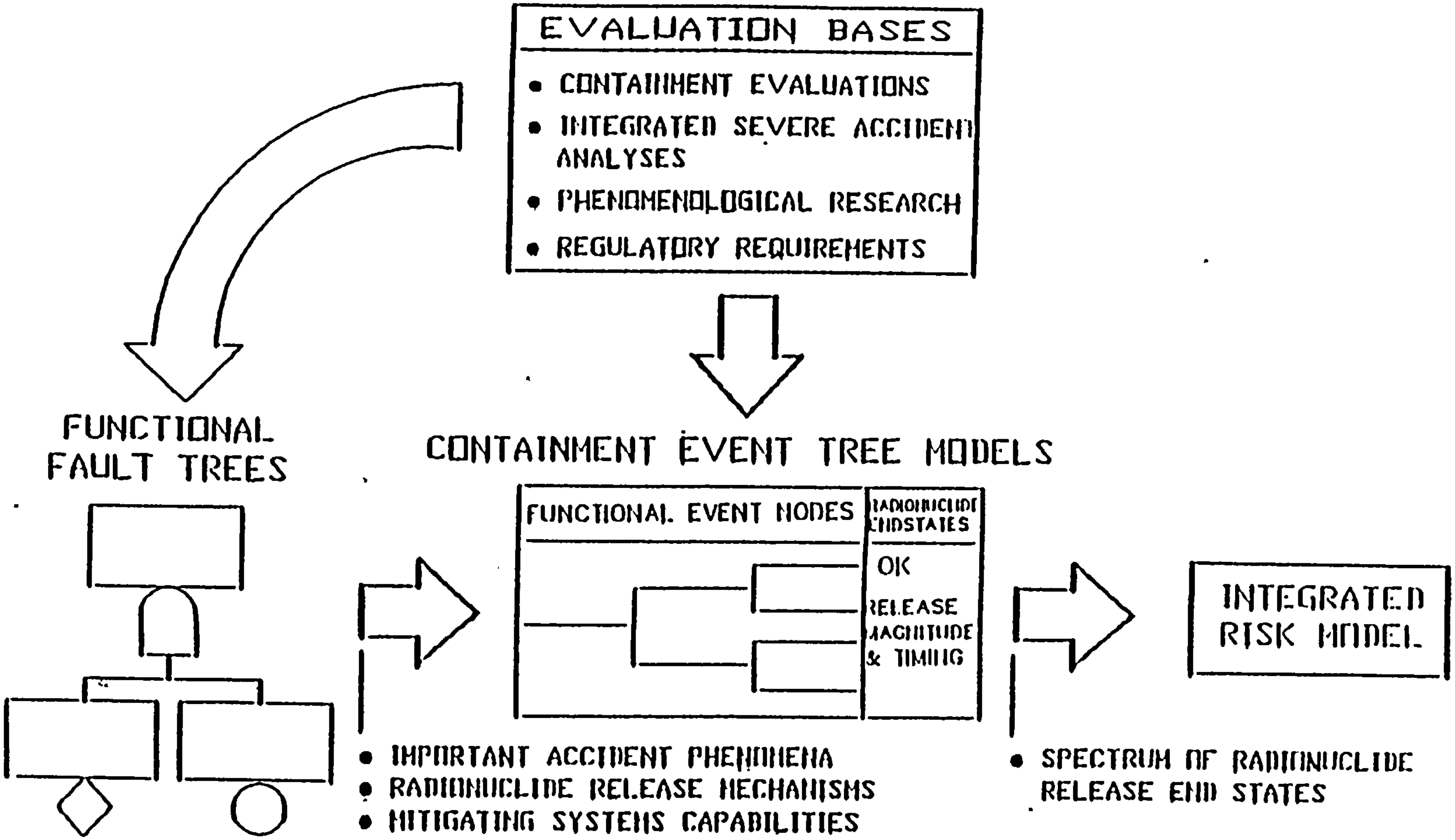


Figure 4.5-1

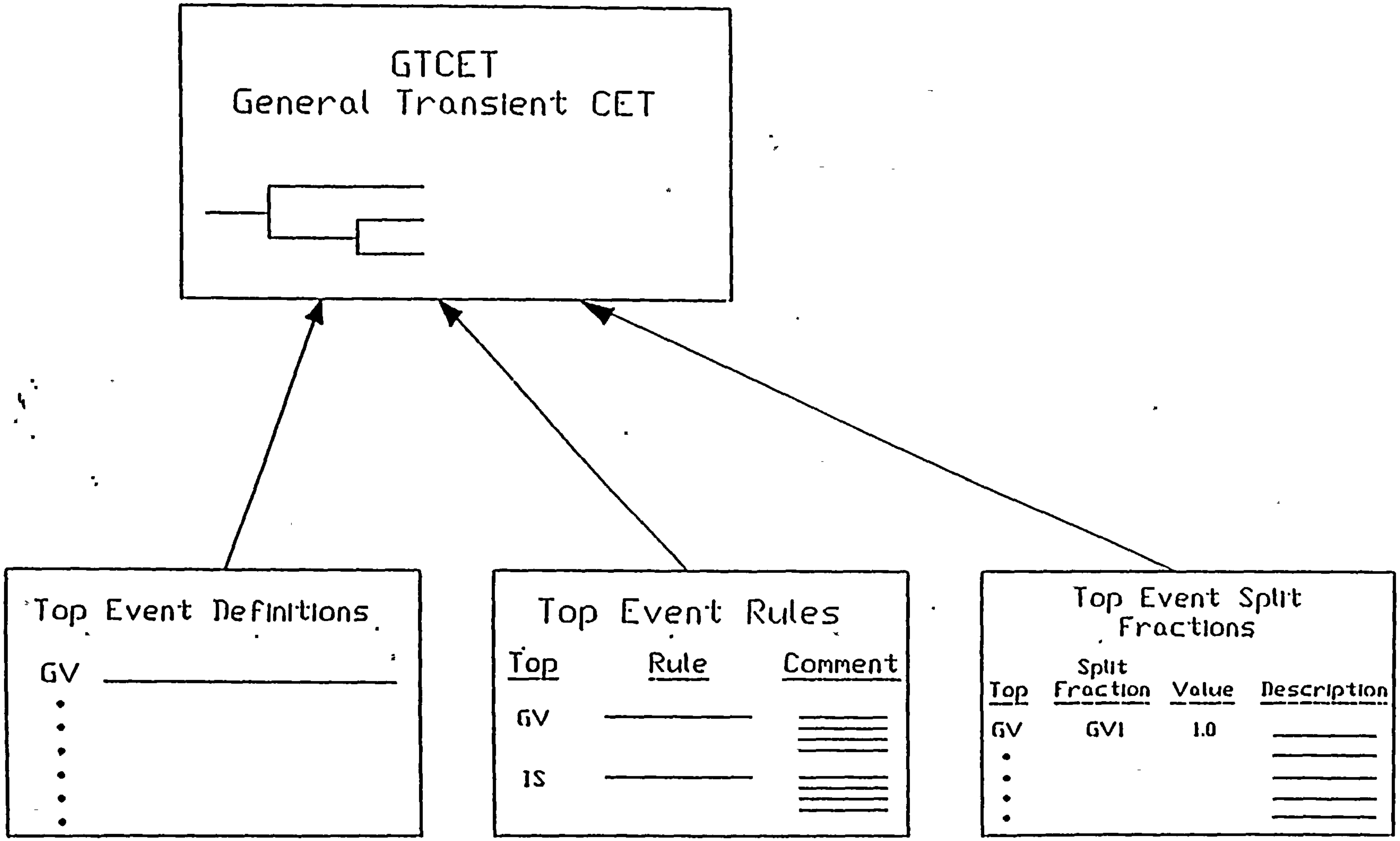


Figure 4.5-2 Summary of Containment Event Tree Documentation and Quantification Required for Input to RISKMAN

MODEL Name: NMP2
Event Tree: CET1

09:22:00 20 JAN 1992

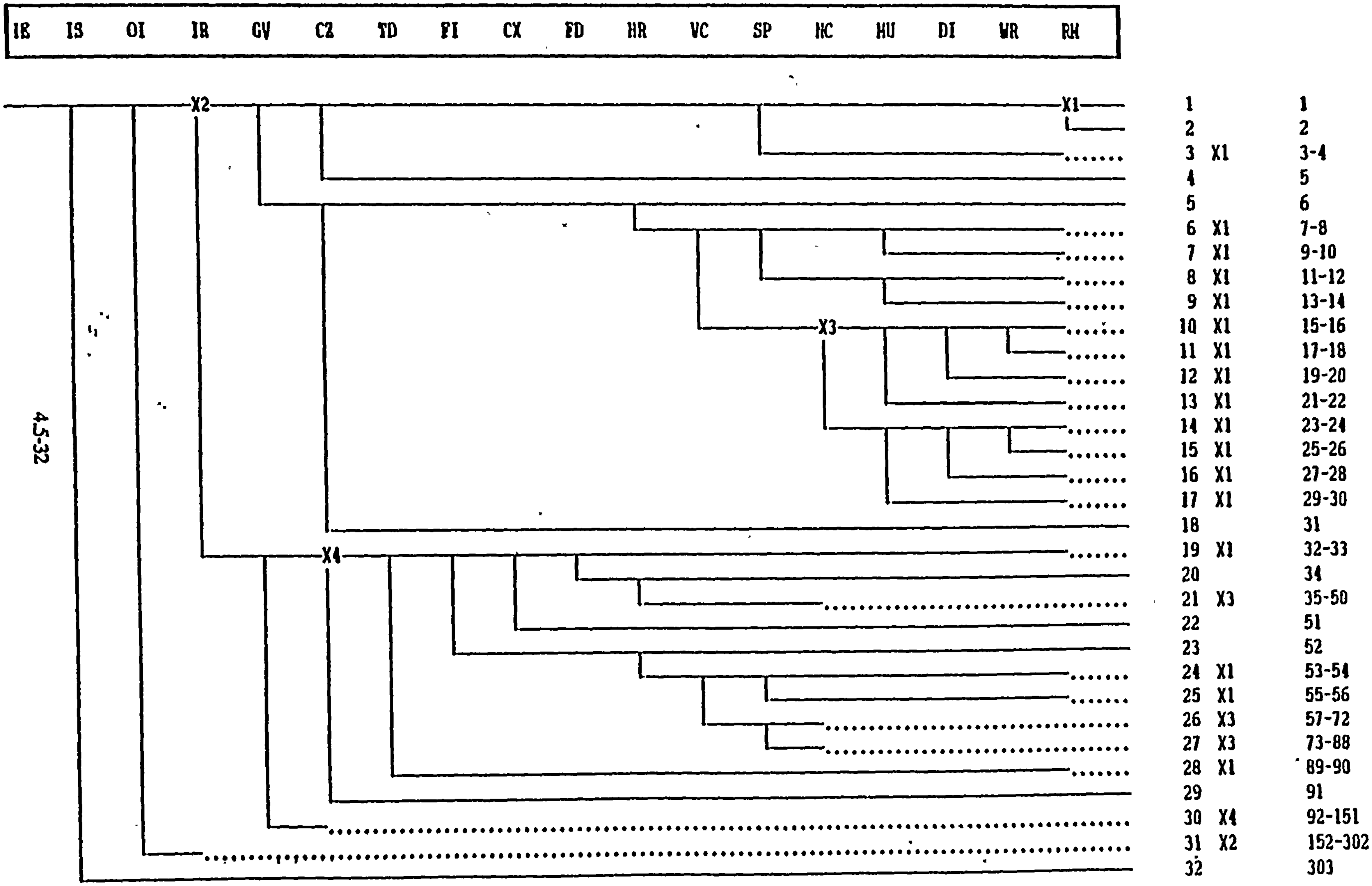


Figure 4.5-3

Nine Mile Point 2 Containment Event Tree for Class I and III Sequences (CET1)

Legend for Figure 4.5-3

09:22:16 20 JAN 1992
Page 1

Top Event Designator.....	Top Event Description............................
IE	INITIATING EVENT
IS	ISOLATED CONTAINMENT & NO INJECTION
OI	OPERATOR RPV DEPRESSURIZATION
IR	IN-VESSEL RECOVERY
GV	COMBUSTIBLE GAS VENTING INITIATED
CZ	CONTAINMENT ISOLATED AND INTACT
TD	INJECTION ESTABLISHED
FI	CONTAINMENT FLOODING INITIATED
CX	CONTAINMENT INTACT DURING FLOODING
FD	CONTAINMENT FLOODED & DRYWELL VENTED
HR	RHR CONTAINMENT HEAT REMOVAL
VC	CONTAINMENT VENTING
SP	SUPPRESSION POOL NOT BYPASSED
NC	NO LARGE CONTAINMENT FAILURE
MU	MAKEUP REMAINS AVAILABLE
DI	DRYWELL INTACT
WR	WETWELL AIR SPACE FAILURE
RM	REACTOR BUILDING EFFECTIVENESS

Page No. 1

MODEL Name: NHP2
Event Tree: CET2

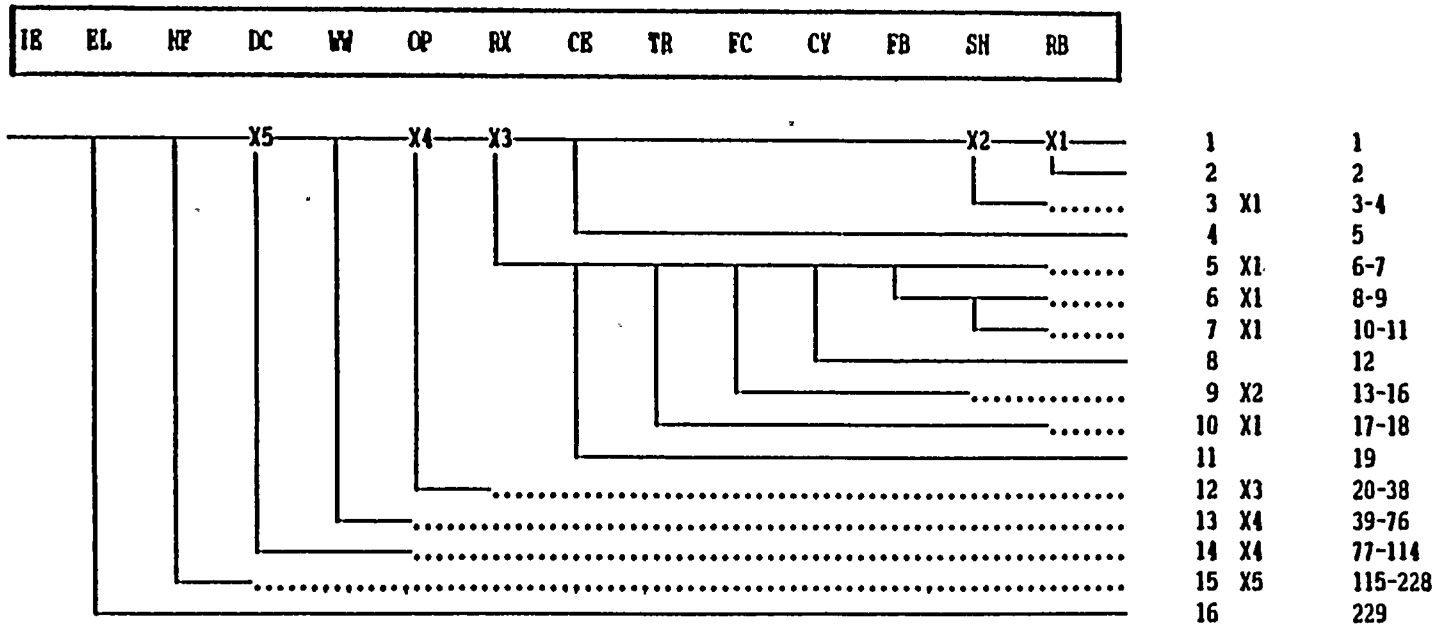
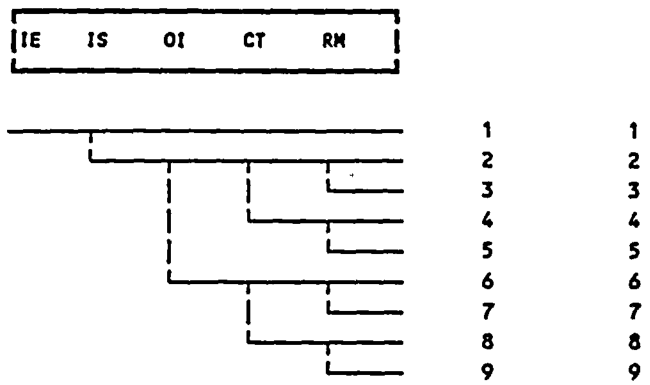


Figure 4.5-4

Legend for Figure 4.5-4

Top Event Designator....,	Top Event Description...........................
IE	INITIATING EVENT
EL	CONTAINMENT UNISOLATED OR FAILED FIRST
NF	NO LARGE CONTAINMENT FAILURE
DC	DRYWELL INTACT
WW	WETWELL AIR SPACE FAILURE
OP	OPERATORS RPV DEPRESSURIZATION
RX	IN-VESSEL RECOVERY
CE	CONTAINMENT ISOLATED AND INTACT
TR	INJECTION ESTABLISHED
FC	CONTAINMENT FLOODING INITIATED
CY	CONTAINMENT INTACT DURING FLOODING
FB	CONTAINMENT FLOODED AND DRYWELL VENTED
SN	SUPPRESSION POOL NOT BYPASSED
RB	REACTOR BUILDING EFFECTIVENESS

Event Tree: CET3



Legend

Top Event Designator.....	Top Event Description..............................
IE	Initiating Event
IS	CONTAINMENT ISOLATION
OI	RPV DEPRESSURIZATION
CT	CONTAINMENT REMAINS INTACT LONG TERM
RM	REACTOR BUILDING EFFECTIVENESS

Figure 4.5-5 ISLOCA CET

4.6 ACCIDENT PROGRESSION AND CET QUANTIFICATION

This subsection summarizes two important features of the Level 2 modeling:

- The accident progression description
- The CET Quantification.

4.6.1 Deterministic Calculations

The Nine Mile Point 2 IPE considers a full spectrum of severe accidents that have been postulated and which may challenge accident management actions in unique or special ways. These unique challenges include:

- Core melt progression with containment intact
  - At high RPV pressure
  - At low RPV pressure
  - With and without adequate reactivity insertion
- Core melt progression with the containment breached or not isolated

The Nine Mile Point 2 deterministic MAAP calculations provide the technical baseline for the determination of:

- Success criteria and plant response at each node in the CET
- Containment survivability under the postulated severe accident
- The source terms
- Sensitivities to key parameters reflecting the uncertainty range.

As part of this summary write-up, it is useful to indicate the general trends that have been observed for the different postulated accident classes that are encountered in the evaluation of accident management actions under severe accident conditions. Therefore, a small sample of the more than 50 calculated MAAP cases for the Nine Mile Point 2 containment are included here for reference.

Included in this section are MAAP calculated containment response pressures and temperatures. These conditions are compared with the ultimate containment capability to ascertain the status of containment. Figure 4.6-1 shows conceptually how this comparison is performed.

The representative severe accidents that are discussed here for example include the following:

- Class IA: Loss of adequate makeup at high RPV pressure with the containment initially intact
- Class ID: Loss of adequate makeup at low RPV pressure with the containment initially intact
- Class II: Loss of adequate containment heat removal

- Class IV: ATWS event with containment failure preceding the occurrence of core damage.

Table 4.6-1 provides the designators for these sequences. Many other postulated accident scenarios are performed to determine the variations in timing associated with changes in the sequence (see Section 4.7 for a summary of MAAP runs). These examples are shown only for illustration. There are in excess of 50 MAAP runs that are used as part of the Nine Mile Point 2 IPE to characterize plant response and radionuclide release. The following is a brief discussion of the representative sequences shown here for illustration.

Class IA: Loss of Makeup at High RPV Pressure with the Containment Initially Intact.
No containment injection is assumed during the accident sequence. The key events for this sequence can be summarized as follows:

Event	Timing (Hrs.)
Core Uncovered	0.5
Initiation of Core Melt	1.06
RPV Failure/Breach	2.69
Containment Failure Location: DW Head Size: Large	31.6
Radionuclide Release Magnitude: High	31.6

Figures 4.6-2 and 4.6-3 demonstrate that a high pressure core melt sequence (e.g., TQUX) can result in a high containment pressure spike while containment is at relatively low temperatures. This spike will not in and of itself cause containment failure. However, the subsequent heat-up of containment and pressure inside containment will lead to failure at approximately 31.6 hours into the accident. This can be seen by comparing the pressures and temperatures at 31.6 hours with the containment capability curve from Section 4.4.

Class ID: Loss of Adequate RPV Makeup at Low RPV Pressure
The key events in this postulated sequence involve the plant response when the RPV has been successfully depressurized, but no injection is available to the RPV or containment for the duration of the sequence:

Event	Timing (Hrs.)
RPV Depressurization	.42
Core Uncovered	.44
Initiation of Core Melt (peak core temp.)	1.01
RPV Failure/Breach	1.53
Containment Failure	23.7
Radionuclide Release	23.7

Figures 4.6-4 and 4.6-5 provide the drywell pressure and temperature traces for the case in which the RPV fails at low pressure and no injection or heat removal capability exists. There is still a containment pressure spike at RPV failure, but it is approximately 35 psig less than the pressure increase observed for a high RPV pressure blowdown (Class IA above). The containment is calculated to fail at approximately 24 hours due to the combination of high pressure and temperature. (See also Section 4.9 for the discussion of core melt progression models on this pressure rise.)

Class II: Loss of Adequate Containment Heat Removal
This accident sequence involves core damage only after containment failure occurs. The HPCS pump is assumed available as an injection source to the RPV throughout the accident and up to the time that the containment is beached. This is substantially different in timing and response from the Class I sequences:

Event	Timing (Hrs.)
Containment Fails	31.56
Core Uncovered	34.33
Onset Core Melt	35.92
Radionuclide Release To Environment	35.97
RPV Breach	39.53

Figures 4.6-6 and 4.6-7 provide the drywell pressure and temperature for a sequence in which containment heat removal is postulated to fail, but coolant injection to the RPV remains available until containment failure occurs. This sequence is similar to the WASH-

1400 "TW" sequence. The containment failure occurs at relatively low containment temperatures at a time of 31 hours after scram and loss of containment heat removal. This represents an exceedingly long time.

Class III: LOCAs with Inadequate Makeup
No example sequences are presented. The MAAP results for Nine Mile Point 2 demonstrate that the characteristics are similar to those of Class I for similar system availability cases.

Class IV: ATWS Induced Containment Failure Followed by Core Damage
In this accident sequence, containment failure is induced by a rapid increase in containment pressure which precedes core damage. The key events for this postulated scenario are the following:

Event	Timing (Hrs.)
Containment Failure	1.13
Initiation of Core Melt (peak core temp.)	1.66
Radionuclide Release Magnitude: Medium	1.66
RPV Failure/Breach	2.01

Figures 4.6-8 and 4.6-9 provide the containment drywell pressure and temperature traces for a postulated ATWS. For this "worst case" scenario containment failure occurs "early", i.e., in the 1 to 2 hour time frame and it is assumed that core damage follows immediately due to loss of effective injection at containment rupture.

4.6.2 Quantification Process

The quantification of the Level 2 IPE model merges the deterministic thermal hydraulic calculations, the postulated containment failure modes, the assessment of the containment ultimate strength, the assessment of mitigation and the probabilistic assessment of the likelihood of each. This subsection describes the overview of this process.

4.6.2.1 Postulated Containment Failure Modes

A comprehensive list of containment failure mechanisms is developed and presented in Section 4.4. The CET was used to structure these failure mechanisms so they could be probabilistically assessed given the severe accident challenges which are determined from the Level 1 analysis and considering the recovery and mitigation in the Level 2 analysis.

This process resulted in the identification of the most probable potential containment failure sequences for Nine Mile Point 2.

4.6.2.2 Containment Ultimate Strength

As described in Section 4.4 the ultimate containment capability for the spectrum of severe accidents is determined. This ultimate capability is then overlaid on top of the containment pressure and temperature response determined (see Section 4.6.1) to assess the probability of containment failure. The ultimate containment capability is based on a plant specific assessment of CB & I for static loading and a separate effects analysis for dynamic loads.

4.6.2.3 Equipment Survivability

The quantification process includes an examination of the impact of severe accident conditions on equipment required for accident prevention and mitigation. However, formal environmental qualification requirements are not applicable to the IPE and accident management process. When credit is taken for equipment in severe accidents, an assessment is made of the ability of the equipment to perform the function for a specific period of time considering exposure to temperature, pressure, aerosol loading, radiation, and moisture. The degree of credit is based upon a review of studies concerning the capacity of equipment to survive or operate in various environments [Ref. 4-72 through 4-89]. If the available data do not cover the range of conditions expected during a severe accident, then the data are extrapolated. The NMP2 IPE considers the survivability/operability of equipment, systems, structures relied upon in a severe accident using principally engineering judgment coupled with some limited data.

Research studies and tests of equipment survivability were reviewed for the following components:

- Cables
- Electrical penetration assemblies
- Electrical connections
- Solenoid valves
- Motor-operated valves
- Motor-driven pumps
- MCCs.

In general, components located in the reactor building have a fairly high reliability rate. The reactor building is estimated to experience temperatures of 100 to 200°F in worst cases, and most components can survive in this type of environment for tens of hours. Cable connections (specifically terminal blocks) appear to be the weakest links, exhibiting high failure rates in steam environments of approximately 200°F. However, plants such as NMP2 have removed terminal blocks (as a result of Information Notice 84-47) from safety systems and selected systems that may see harsh environments.

Susceptibility of individual components was not modeled; for example, injection systems were grouped into a single basic event that considered failure due to harsh environment.

Due to like components among systems, the assumption was made that if components failed in one system due to harsh environment then so did components in the other injection systems. Injection systems, the RHR system, the depressurization system, and the vent valves were modeled considering environmental effects.

4.6.2.4 Containment Isolation

Consistent with NUREG-1335, containment isolation is modeled as the first node in the containment event tree. The modeling of containment isolation is based on a fault tree model. The fault tree for containment isolation incorporates modeling of containment hatches and large lines that penetrate the containment and are open to the containment atmosphere (e.g., purge and vent lines). The fault tree considers automatic isolation signal failure, pre-existing open pathways, manual isolation, and component failures.

Failure of containment isolation is modeled as a failure in the drywell. Containment isolation failure is conservatively characterized as a high radionuclide release at the time of initial core damage (i.e., H/E release categorization) based upon a representative worst case MAAP calculation.

4.6.2.5 Human Intervention

One of the important attributes of a Probabilistic Risk Assessment (PRA) is that the analysis provides an integrated picture of the design, maintenance, and operational factors that influence plant safety. Human reliability in the operation of a nuclear power plant plays an extremely important role in assuring its safety. Human actions that can affect safety include: operator actions (e.g., control room manipulation, diagnosis of plant conditions, recovery actions, and system manipulation); maintenance actions (e.g., preventive maintenance, and corrective calibration and inspection); and management actions (e.g., problem solving and decision making). Human Reliability Analysis (HRA) is an important tool for analyzing the human element that is common to each of these factors, and integrating this information with the plant system and accident sequence models.

The Level 2 PRA considers important human interaction events that can affect containment performance and radionuclide release frequency, magnitude, or timing, and establish the risk profile of NMP2. Therefore, it is necessary to consider the human tasks that are performed under normal operating conditions, and those actions performed in response to accidents or abnormal occurrences. These are actions that may occur during the course of an accident as the operator interprets the incoming diagnostic information and implements the task determined to be appropriate. However, whether during normal operation or during responses to an accident situation, only human errors, defined as mistakes in the performance of assigned tasks, are modeled. It is assumed that any intentional deviation from operating procedures is made because of misdiagnosis or misleading indication for which the operators believe their method of operation to be safer or more efficient.

The Level 2 analysis incorporates the consideration of the key operator actions. In general, the actions considered in the analysis are confined to those that are proceduralized (i.e., actions directed by current EOPs), although, occasionally operator actions are included that are not explicitly directed by the EOPs. Refer to Table 4.6-2 for a list of types of operator actions included in the Level 2 analysis. Those actions that are EOP-directed are quantified considering the level of operator training for each action. The quantification method for operator actions is the same as that employed in the Level 1 analysis.

The purposes for performing the HRA are: (1) to provide a qualitative understanding of the specific operator actions and the dominant influences that could alter the assessment of successfully performing these actions (e.g., sequence dependencies); (2) to determine the appropriate HRA method to quantify the EOP steps and actions to which the CET model is sensitive; and (3) to provide the best estimate quantitative values for these events based on state-of-the-technology models.

The HRA has been developed in cooperation with analysts responsible for conducting the Level 1 HRA to ensure that the IPE logic models accurately reflect the instructions contained in the EOPs and the training that operations personnel receive on the implementation of these directions. Additionally, close cooperation among the HRA analysts ensured the consistent application of the various methodologies used in the analysis, and the proper treatment of potential dependencies that could influence the quantification of actions contained in both the Level 1 and Level 2 models.

The HRA included the performance of the following tasks:

- Task 1 - Identification of Key PRA Human Error Elements
  - Task 1.1 - Identification of the key operator actions that affect prevention and mitigation of severe accidents and location of these interfaces within the PRA model.
  - Task 1.2 - A review of the PRA and similar BWR PRAs, and the identification of the accident sequences and associated human errors that may contribute to containment challenge or recovery actions that may prevent or mitigate these states.
- Task 2 - A review of the plant specific EOPs to identify areas of possible ambiguity or potential for confusion.
- Task 3 - Development of a set of questions and postulated accident scenarios to ascertain if the PRA accurately reflects the EOPs and associated training.
- Task 4 - Interviewing Operators and Staff
  - Task 4.1 - A set of interviews with trainers, operating staff, and EOP developers to ensure that the responses to the questions are understood for incorporation into the model. An accident sequence will be established with timing and symptoms to allow the staff to walk-through the EOPs.
  - Task 4.2 - Observation of selected actions on the simulator to confirm the responses in the interview.

- Task 4.3 - Confirmation that the simulator instrumentation layout is adequate, and that there are no unusual features of the plant to compromise the qualitative and quantitative conclusions.

- Task 5 - Application of the information from the interviews into the available quantification models, ensuring that dependencies are properly incorporated.

- Task 6 - Document the process providing qualitative insights and identification of modeling techniques to be used to model the operator actions.

The model of human interactions used to evaluate human error probabilities discretizes operator response into three components, a detection, diagnosis and decision phase, and an execution phase. This is compatible with the following methodologies used for this HRA: (Reference is made to these documents for details.)

- EPRI Methodology [Ref. 43],
- Technique for Human Error Rate Prediction (THERP) [Ref. 44], and
- Accident Sequence Evaluation Program (ASEP) [Ref. 42].

The EPRI methodology is an HRA procedure for application of a simulator measurement-based approach to estimate operator non-response probability due to misdiagnosis of a situation that requires the implementation of a particular action. (Note that this approach requires the analyst to estimate the HEP associated with implementing the action using another appropriate methodology.) Consequently, this methodology is applicable to the quantification of post-accident human interaction events initiated from the control room as directed by abnormal or emergency operating procedures. However, because the use of simulator data to estimate non-response probabilities can require considerable extrapolation, an alternate approach to quantify the HEP was also used. This approach is based on identifying failure mechanisms and the factors that impact their probability, and the results of the study of operator errors performed as part of the EPRI Operator Reliability Experiments (ORE) program. The application to the high stress environment of post core damage situations is performed recognizing that available HRA techniques are not developed explicitly for these conditions; and therefore require, judicious application.

THERP is a method to predict human error probabilities and to evaluate the degradation of a man-machine system likely to be caused by human errors alone or in connection with equipment functioning, operational procedures and practices, or other system and human characteristics that influence system behavior. THERP is used to develop the probability of task failure and correcting improper task performance based on estimates of various factors influencing these events.

The ASEP methodology is a technique that can be used for evaluating both pre-accident and post accident operator actions. The intent of the ASEP methodology is to enable system analysts, with minimal support from experts in human reliability analysis, to make estimates of human error probabilities, and other human performance characteristics associated with pre-accident and post-accident type operator actions, that are sufficiently accurate for many probabilistic risk assessments. This methodology is based on the THERP technique.

For the pre-accident HRA, the emphasis of the analysis is to quantitatively estimate the potential impact on system safety caused by the improper performance of primarily rule-based procedures before the occurrence of an event that may challenge the system to operate and fulfill its intended function. The approach used in the ASEP methodology to estimate the HEPs associated with post-accident tasks requires the analyst to determine two separate time-dependent probabilities: the probability of performing a correct diagnosis within its allowable time; and the probability of performing the post-diagnosis execution actions within the time remaining for acceptable system responses to the abnormal event.

In addition to these HRA methodologies, one operator action was quantified using results from HRAs performed in other PRAs of similar reactor plant designs. However, this approach was limited to the quantification of that human interaction event (i.e., suppression pool cooling initiation), that could not be adequately evaluated using more prescriptive methods due to the unusually long time frame to implement this action.

As part of the HRA process, the quantification of the Human Error Probabilities (HEPs) are performed, using the various quantification methods, to provide a point estimate. Table 4.6-2 summarizes the recommended human error probabilities for use as best estimate values in the NMP2 IPE.

Table 4.6-2 provides the following information:

- The designator used in the Level 2 IPE model.
- A brief description of the action as it applies to a specific function.
- The description of the sequence under which the action applies. Note the sequence classes are described in Tables 4.6-3.
- The HRA model that is used in the quantification process.
- The point estimate HEP used in the quantification.

4.6.3 Quantification Results

This subsection includes the following summaries:

- The quantification of the plant damage states from the Level 1 PRA for input to the CET.
- The output radionuclide release frequencies from the CET quantification for the baseline evaluation.
- Graphical comparison of the radionuclide release magnitudes and timing and their major contributors.
- Summary of Top 100 systemic sequences associated with containment failure.

- Discussion of the Top 10 sequence contributors to the "large" release category.

4.6.3.1 Input

Each accident sequence defined in the Level 1 IPE that is above the culling limit is transferred to the Level 2 evaluation model. This provides the input for the Level 2 assessment. As discussed earlier, the different accident types represent substantially different challenges to containment, containment mitigating systems, and the operating staff. Therefore, each accident sequence has been treated separately in the containment event tree evaluation.

This treatment consists of:

- Using the CET structure that best describes the chronology of events

- Using "rules" that logically include the appropriate dependencies as a function of the sequence type.

Table 4.6-3 summarizes the core damage frequency contributions from the Level 1 PRA by subclass. This, in turn, provides a useful display of the aggregate output of Level 1 input to the Level 2 containment evaluation.

4.6.3.2 Output Summary

The Level 2 quantification can be summarized briefly in two complementary tables. These tables provide a wealth of quantitative information that is useful in the interpretation of the current NMP2 containment capability given the spectrum of core damage sequences calculated in the Level 1 IPE. These tables also allow interpretation of the impact of possible changes.

Table 4.6-4 summarizes the following results:

- <u>Input</u>: The Nine Mile Point 2 Level 1 PRA accident sequence frequencies are used as input to the containment event tree evaluation.

- <u>Radionuclide Release End States</u>: The release categories used to discriminate among the CET end states are identified.

- <u>Output</u>: The output frequencies of the CETs as a function of the end state bins are identified.

The individual accident class contributors to the radionuclide release frequency may also provide insights into the containment performance as a function of the type of severe accident. The contributors to each of the end states can be broken down by the type of accident class from the Level 1 analysis as shown in Table 4.6-5. This is discussed further in Section 4.6.3.7

The CET model quantification provides a yardstick to measure the best estimate containment performance given that severe accidents could progress to beyond core damage. The quantification may include some conservatism to account for the inability of current models and experiments to predict certain severe accident related phenomena.

A substantial fraction (26%) of the accidents transferred from the Level 1 PRA are substantially mitigated such that releases are essentially contained within an intact containment (i.e, OK release bin). Approximately 97% of the postulated accidents do not have "large" releases occurring before protective action can be taken.

The individual accident class contributors to the radionuclide release frequency may also provide insights into the containment performance as a function of the type of severe accident. The contributors to each of the end states can be broken down by the type of severe accident class from the Level 1 analysis as shown in Table 4.6-5. The definitions of Classes IA through V are provided in Section 4.3.

Table 4.6-5 shows that the largest contributors to the worst release category, high (H) release magnitude and early release (E), are due to loss of makeup at high pressure, (Class IA), with no depressurization recovery, loss of makeup at low pressure (Class ID), ATWS (Class IV) and ISLOCA and breaks outside containment (Class V).

For the late (L) high (H) release magnitude category, loss of containment heat removal (Class II type events) are the dominant contributors.

The evaluation of possible sensitivities about the baseline evaluation has also been included. These sensitivity evaluations are discussed in detail in Section 4.9.

4.6.3.3 Examination of The Baseline Quantified Results of The Nine Mile Point 2 CET

Different organizations may have different opinions on what are the most important issues related to the protection of the public health and safety. For example, FMEA may consider the understanding, prevention, and mitigation of accidents that could result in changing evacuation plans or evacuation effectiveness as the key issue. For such determinations both the magnitude and timing of the accident sequence is of importance. On the other hand, some organizations may consider latent health effects to be the dominant contributor to public risk and therefore the magnitude of the release is of principal importance regardless of the timing. To account for these different viewpoints, the radionuclide release binning is summarized in different ways such that various organizations can make the most effective use of the information for their specific purpose.

Therefore, this subsection examines the quantitative results of the Nine Mile Point 2 Level 1 and Level 2 PRA evaluations from a number of different viewpoints.

4.6.3.3.1 Plant Damage States

The input to the Level 2 PRA evaluation comes from the output of the Level 1 Nine Mile Point 2 PRA. Each of the accident subclasses represents different challenges to containment and therefore will have different impacts on public safety. The characteristics of the dominant contributing classes can be summarized as follows:

Dominant Contributing Plant Damage States from Level 1 PRA

Accident Class	% of Core Damage Frequency	Characteristic
IA	19%	Accident sequences involving loss of inventory makeup in which the reactor pressure remains high.
IB	18%	Station Blackout
ID	29%	Loss pressure core melt sequences
IIA	15%	Accident sequences involving a loss of containment heat removal with the RPV initially intact; core damage induced post containment failure.
IIT	13%	
IV	3%	Failure of adequate reactivity control results in overpressure failure of containment before core melt.
V	< 1%	Unisolated LOCA outside containment
Other III	2-3%	Includes station blackout, various classes of LOCAs, loss of coolant makeup and loss of containment heat removal accidents.

The impact of these subclasses on public safety are summarized in the following subsections.

4.6.3.3.2 Radionuclide Release Magnitude Frequency

The frequency of radionuclide release is characterized by the quantification of the Level 1 and Level 2 PRA models. The Level 2 containment event tree end states are further delineated by the magnitude and timing of the calculated radionuclide release. Using the end state release magnitude and timing, a comparison can be developed to identify the overall frequency of the various end state release magnitudes, from very low to high. The two term matrix (release magnitude, timing) is shown in Table 4.6-6.

Figure 4.6-10 summarizes in bar-graph form a comparison of the total core damage frequency (i.e., the results of the Level 1 IPE) with the end state frequencies of the Level 2 analysis, i.e., High (H), Moderate (M), Low (L) and Low-low (LL) release magnitudes plus those severe accident sequences that result in an intact containment (OK). A substantial fraction of the core damage end states are either of low release or the containment remains intact and no substantial release (42%).

These radionuclide release results can also be plotted in a pie-chart format to show the relative contributions from the various Level 2 release magnitude end states (see Figure 4.6-11).

It is important that the above selection of release bins be flexible enough to be used to answer important questions that may be raised by the NRC in IPE evaluation or application. Therefore, a review of available published NRC directives and staff recommendations was performed. It appears that one of the items of interest in the assessment (either on a plant specific or a generic basis is a comparison with the safety goal general performance guideline stated by the Commission as follows:

> *Consistent with the traditional defense-in-depth approach and the accident mitigation philosophy requiring reliable performance of containment systems, the overall mean frequency of large release of radioactive materials to the environment from a reactor accident should be less than 1 in 1,000,000 per year of reactor operation.*

The difficulty is in clearly defining the "large" release. The following discussion provides the basis for connecting this "large" release with the release bin characterization in the Peach Bottom PRA as High/Early.

The primary purpose of the IPE is to perform a systematic evaluation of each plant for vulnerabilities to severe accidents, not to assess nuclear power plant risk relative to the safety goals. In addition, the strength of PRAs or similar examinations is not in determining absolute risk, but in better understanding plant operations and in determining relative risks and how to reduce them.

The Commission's Severe Accident Policy Statement stated:

> *...formulate an integrated systematic approach to an examination of each nuclear power plant not operating or under construction for possible significant risk contributors (sometimes called "outliers") that might be missed absent a systematic search.*

In SECY-88-205 dated July 15, 1988, which forwarded the IPE program to the Commission for approval, the staff indicated how the IPE results would be used with the Safety Goal Policy.

> *. . . we intend to review the IPE results as an aggregate to identify severe accident vulnerabilities generic to a class or several classes of plants. Such generic vulnerabilities would be used to determine if deficiencies in the regulations existed. If deficiencies were identified, the benefits of modifying the regulations would be assessed against the safety goal policy as part of determining whether modifications to the regulations were needed.*

The use of the IPE results in this fashion is consistent with the Commission's Safety Goal Policy.

The Commission recognizes that the safety goal can provide a useful tool by which the adequacy of regulations or regulatory decisions regarding changes to the regulations can be judged.

The NRC staff recommendation on the definition of "large" release (SECY 90-405 dated December 14, 1990) is:

A large release is a release of radioactivity from the containment to the environment of a magnitude equal to or greater than: (An amount, to be determined by the staff, expressed in curies or fraction of the core inventory, which has the potential, based on representative site characteristics, for causing one or more offsite prompt fatalities.)

This definition of "large" release is based on offsite consequences. However, rather than comparing plant specific offsite consequences, the staff proposes that a spectrum of sites be considered to establish representative site characteristics. These site characteristics would take into account factors such as meteorology and population distribution. From these site characteristics, the staff will determine a value for an accidental radioactive release to the environment that would have the potential for causing doses high enough that one or more prompt fatalities are probable at the representative site. In other words, Safety Goal Objective Level Three would define a large release to be a release of predetermined magnitude.

In this definition, the magnitude of the source term release may be expressed as curies (or "equivalent curies") or fraction of the core inventory of chemical elements that represent the radionuclides present at full power operation. Appropriate provision will need to be made to address significant variations in power levels, if the definition is stated in terms of fraction of core inventory released.

The effort to determine the release magnitude would focus on highly exposed individuals to determine the release required for a prompt fatality in a fashion identical to that used in NUREG-1150. That is, the weighted probability of a prompt fatality over the exposed population, given site and source term factors, would be determined. The source term factors include the timing of the release, its path to the environment and energy content, and the biological effectiveness of the various radionuclides. The site factors include population distribution and meteorology. It is expected that the assumptions used for emergency planning early in the accident sequence will not be critical and that the magnitude selected for a large release will be independent of emergency planning assumptions early in the accident sequences. The staff intends to confirm this by evaluating the effect of various emergency planning assumptions as part of the analysis.

Therefore, the definition of "large" release can be correlated to the High/Early Nine Mile Point 2 release category because of the following:

High: Releases less than High have been shown to have little chance of causing prompt fatalities.

Early: The issue is to define those releases that can lead to prompt fatalities before effective emergency planning can be supplemental.

The question of "large" release to the environment can then be answered by examining those releases that are both early and of a high magnitude.

Because of the emphasis in the NRC Safety Goal Policy on the objective of maintaining "large" releases below 1E-6/reactor year on a generic basis, it may be useful to compare the Nine Mile Point 2 results with this objective (see Figure 4.6-12). This comparison, while not required, provides some indication of the acceptability of the resulting frequency of release. From Table 4.6-4 and Figure 4.6-12, it can be seen that, if "large" release is defined as any release to the environment of sufficient radionuclide material to be life threatening and within a time frame too short to allow protective action (i.e., H/E), then the frequency of such "large" releases would be 8.0E-7/year which is below the NRC staff defined objective [SECY-90-405].

4.6.3.3.3 Timing of Radionuclide Releases

As noted above in the discussion of "large" release, another parameter in the evaluation of the impact of radionuclide releases is related to the timing of the release. This parameter is of importance to identify the time available for:

- Accident management response actions
- Public safety measures, such as sheltering or evacuation.

Three categories of timing used in the end state quantification are as follows:

- Late: Releases are initiated more than 24 hours after the accident initiation (EAL Trigger).

- Intermediate: Releases are initiated between 6 hours and 24 hours after accident initiation (EAL Trigger).

- Early: Releases are initiated within 6 hours of the accident initiation (EAL Trigger).

Figures 4.6-13 and 4.6-14 summarize the frequency of radionuclide release for the Nine Mile Point 2 accident analysis as a function of the timing of the radionuclide initiation. It is important to note that the radionuclide release occurs over an extended time and the time noted here is the initial release time of the release. In addition, note that some other definitions of "early" release have used release times with windows of time as small as two hours.

Even with these definitions of timing, the early radionuclide release is only 7% of the total calculated severe accident frequency (i.e., frequency of a core damage event).

4.6.3.4 Containment Integrity

Figure 4.6-15 provides an interesting division of the releases.

In the assessment of radionuclide release, the mechanisms for releases include:

- Containment failures
- Containment venting.

The reason for separating the release modes between containment venting and containment failure is to provide an indication of those release pathways that are controllable, and therefore for which containment integrity can be restored.

Based on this quantification, releases associated with venting represent a small fraction of the releases.

Another informative division of releases is based on containment failure modes. Figure 4.6-16 provides a pie graph illustrating the division of releases and no release sequences; the release sequences are subdivided into various containment "failure" modes:

- Overtemperature/Overpressure
- Vent Containment
- Energetic
- Other.

4.6.3.5 Combination of Release Magnitude and Timing

In Sections 4.6.3.2 and 4.6.3.3, the radionuclide release as a function of magnitude and timing were examined separately. These two viewpoints can be combined to determine if there are relationships or impacts associated with the release magnitude and timing that may influence accident management decisions.

Figure 4.6-17 summarizes graphically the radionuclide release magnitude in a manner similar to Figure 4.6-10 except that it is augmented to also show the contributions to each release magnitude associated with the time of the release.

Figure 4.6-18 summarizes the radionuclide release timing in a manner similar to Figure 4.6-12 except that it is augmented to also show the contributions to each release time phase associated with the magnitude of the release. No obvious trends are elicited by these plots, except that no definitive correlation exists among timing and magnitude.

4.6.3.6 Containment Radionuclide Releases as a Function of Accident Type

Important insights into possible accident management strategies can be obtained by identifying the types of accident sequences that are contributing to the radionuclide release bins.

Figure 4.6-19 provides a graphical summary of the "large"[1] release contributors by accident class. As can be seen from the figure, loss of makeup at high pressure (Class IA), loss of coolant inventory (Class ID), ATWS, loss of AC power with isolation failure, and containment bypass sequences (Class V accidents) are the dominant contributors to High-Early releases, i.e., large releases.

Figure 4.6-20 provides a graphical summary of "High" release contributors by accident class. High releases consist of the High-Early, High-Intermediate and High-Late release categories. The added information here presents accident sequences with potentially high health effects, if no evacuation occurs. The contributing sequence types are noticeably different.

It is also useful to examine the difference between the contributors to core damage and those to large releases. Figure 4.6-21 compares the contributors to core damage frequency and those that contribute to "large" releases. This comparison graphically shows that the sequences that dominate core damage frequency are not necessarily those which dominate the high release. For example, ATWS (Class IV) or ISLOCA (Class V) plus other LOCA accidents are small contributors to core damage frequency, but make up a significant fraction of the "large" release category.

The "large" release can be compared in pie-chart format with the total core damage frequency that is transferred from the Level 1 analysis i.e., shown in Figure 4.6-21. The conditional probability of a large release given a core damage accident is calculated to be 0.03. Therefore, it should be pointed out that the relative sizes of the two pie-charts are not proportioned to their total frequencies.

4.6.3.6.1 Top 100 Level 2 Sequences

NUREG-1335 Page 27 states that one of the reporting criteria for systemic sequences is:

> "All systemic sequences within the upper 95 percent of the total containment failure frequency."

However, Page 2-6 of NUREG-1335 also states:

> "The total number of unique sequences to be reported should be determined by the criteria listed below, or by the criteria in Appendix 2 to the Generic Letter, but in any case should not exceed the 100 most significant sequences.

Based on this guidance, the RISKMAN quantified model was searched to identify the top 100 sequences (i.e., the most limiting of the criteria) that are associated with containment failure. Table 4.6-7 summarizes this list of the top 100 sequences that are derived from the Level 2 end states.

---

[1] "Large" releases are defined as those releases which are of high magnitude and occur sufficiently early such that no effective public action can be taken. This is conservatively estimated using the HIGH-EARLY (H-E) release category from the Peach Bottom Level 2 results and is consistent with the NRC staff definition of "large" release in SECY-90-405.

It should be noted that the end state bin is identified for each of the sequences. In some cases no release occurs.

The explanation of the Level 1 sequence designators is provided in the Section 3 summary of top 100 core damage systemic sequences. Individual failures for the containment evaluation portion of the sequence are defined by the top events of the CET which are found in Section 4.5.

Dominant Sequences Contributing to the "Large" Radionuclide Release Category (Early/High)

The calculation of radionuclide releases and their assignment to specific release categories (e.g., Early/High (E/H) which is equivalent to a "large" release) requires an integrated assessment of the containment response using the containment event trees, the probabilistic assessment of failure modes, the incorporation of containment integrity, and the thermal hydraulic and fission product transport deterministic evaluation. Because of the complexity of this process, there have been conservatisms that are used in certain cases. The result of these conservatisms is that the frequency of "large" releases may be conservatively estimated.

The following discussion of the top ten sequence contributors to the Early/High ("large") release category will identify some of these conservatisms. Table 4.6-8 lists the top sequence contributors to the Early/High radionuclide release category.

<u>E/H Sequences 1 & 2 (9.4E-8/yr)</u>
These two sequences are initiated by an internal flood and are identical to CDF Sequence 5 where the Level I end state is Class 1A, loss of high pressure injection and core damage occurs at high RPV pressure (RPV not depressurized). The flood causes a loss of all divisional AC power, which leads to a loss of all injection and nitrogen supplies to the safety relief valves. In both sequences, the operators fail to locally close AC powered containment isolation valves (IS3 = 0.1) before core damage. The only difference between these two E/H sequences is whether the RPV depressurizes before RPV failure (OP1 = 0.46). However, the assignment to the E/H release end state is not dependent on this condition. There are no injection sources available to recover the core in-vessel (RXF) or to restore injection after RPV failure (TRF). Because there is no injection into the containment to cool core debris, a high temperature induced drywell failure is assumed with little credit given to the reactor building for source term reduction (RB7 = 0.99). The failure to isolate the containment results in the assignment of these sequences to an early (E) release. The magnitude of the release is assigned based on several considerations:

- MAAP calculations indicate releases of Low or Medium for this sequence (Reactor Building Decontamination Factor DF=10 to 100).

- Hydrogen generation and subsequent deflagration in the reactor building could reduce the DF in the reactor building to approximately 1.0.

- Drywell head failure due to high drywell temperatures could cause a high release later in the sequence. This high release from the drywell head would be assured if the isolation were to occur later due to: 1) intermittent power recovery or heroic action to close the valves; or 2) The drain lines from the sumps could also be assumed to be plugged during the core melt progression, resulting in containment pressurization and eventual drywell failure. Specifically, the delayed isolation of containment (i.e., the procedurally directed action) would actually result in a potentially higher release than if containment isolation fails and the reactor building is effective in reducing the release.

As can be seen, the assignment to a high release category may be conservative.

E/H Sequence 3 (4.2E-8/yr)
The initiating event is a loss of offsite AC power (LOSP = 0.04/yr) which disables normal operating non-safety systems such as the condenser, feedwater, RBCLC, TBCLC, and instrument air. In addition, normally operating safety systems such as service water must restart on demand after the emergency diesels start and load. Given the LOSP initiating event, the following additional failures lead to core damage:

- Division I emergency diesel generator fails or is in maintenance (A12 = 5.3E-2).
- High pressure core spray (HPCS) which must operate from its diesel is unavailable (HS4 = 0.15).
- RCIC fails either due to equipment failure or it is in maintenance (IC1 = 0.16).
- The safety relief valves (SRV) are closed due to equipment failures (SV5 = 5.3E-3).

This leads to a Class 1A core damage end state which is a loss of injection at high RPV pressure. In the Level II model, containment isolation is successful, but the RPV does not depressurize prior to RPV failure (OI1 = 0.46), thus, there is no chance of low pressure injection success and recovery of the core in-vessel (IRF). Because of SRV equipment failures, there is no opportunity given for operator depressurization in this scenario. Containment flooding is successfully initiated, but the containment does not remain intact during flooding (CX1 = 0.45). The EOPs direct containment flooding using available sources (e.g. service water) to either the RPV or containment. The service water cannot be injected to the RPV therefore it is directed to containment. The time to flood containment is of the same order as the time of core melt progression. If containment vapor suppression is defeated during the flood process (e.g., wetwell is filled with water) before the RPV is breached at high pressure, then containment failure is assured. It is assumed that such an induced containment failure is both catastrophic and energetic.

E/H Sequence 4 (2.7E-8/yr)
This sequence is identical to CDF Sequence 2 where the Level I end state is Class 1D, loss of all injection and core damage occurs at low RPV pressure (RPV is depressurized). The initiating event is a loss of Division II Emergency AC power (A2X = 4.3E-3/yr) which disables all safety systems that depend on Division II emergency AC. In this sequence, the Division I battery also fails (DA1 = 6.6E-4) on demand which prevents the restart of Division I service water pumps, fails RCIC, and prevents the start of Division I safety

systems such as RPV injection. All service water is unavailable, feedwater is unavailable, HPCS fails due to loss of room cooling, and thus, all injection is unavailable.

In the Level II model, containment isolation fails under the degraded power state (IS2 = 1.3E-2) and the core recovery in-vessel is guaranteed to fail (RXF) since there is no injection source available and no recovery is taken for the equipment failures. Restoring injection after RPV failure is not allowed to be successful (TR3 = 1.0) since all injection is unavailable and there is uncertainty with regard to the capacity of the diesel fire pump providing success. Because there is no injection into the containment to cool core debris, a high magnitude radionuclide release is assumed with little credit given to the reactor building for source term reduction (RB7 = 0.99). (See discussion under E/H Sequences 1 & 2.)

<u>E/H Sequence 5 (2.6E-8/yr)</u>
This sequence is identical to CDF Sequence 3 where the Level I end state is Class 1D, loss of all injection and core damage occurs at low RPV pressure (RPV is depressurized). The initiating event is a loss of Division I Emergency AC power (A1X = 4.3E-3/yr) which disables all safety systems that depend on Division I emergency AC. In this sequence, the Division II battery fails (DB1 = 6.6E-4) on demand which prevents the restart of Division II service water pumps, fails RCIC, and prevents the start of Division II safety systems such as RPV injection. All service water is unavailable, feedwater is unavailable, HPCS fails due to loss of room cooling, and thus, all injection is unavailable.

In the Level II model, containment isolation fails under degraded power conditions (IS2 = 1.3E-2) and the core recovery in-vessel is guaranteed to fail (RXF) since there is no injection source available and no recovery is taken for the equipment failures. Restoring injection after RPV failure is not allowed to be successful (TR3 = 1.0) since all injection is unavailable and there is uncertainty with regard to the capacity of the diesel fire pump providing success. Because there is no injection into the containment to cool core debris, a high magnitude radionuclide release is assumed with little credit given to the reactor building for source term reduction (RB7 = 0.99). (See discussion under E/H Sequences 1 & 2.)

<u>E/H Sequence 6 (2.1E-8/yr)</u>
The Level I end state is Class 1A, loss of high pressure injection and core damage occurs at high RPV pressure (RPV not depressurized). The initiating event is a loss of Division II Emergency AC power (A2X = 4.3E-3/yr) which disables all safety systems that depend on Division II Emergency AC. In this sequence, Division I AC power subsequently fails (A11 = 9.0E-5) resulting in a total loss of all emergency AC power. Service Water fails due to loss of AC power which causes the loss of condenser and feedwater as well as room cooling. RCIC is assumed to be unavailable due to high temperature trips and no procedures for preventing this trip given loss of all service water. HPCS is assumed to be unavailable due to loss of room cooling. Thus, there is a total loss of injection to the core. Procedures for preventing RCIC trip under loss of service water conditions are being developed, thus this assumption is conservative and will be considered further in future analyses and updates.

In the Level II model, containment isolation fails as the operators fail to locally close AC powered containment isolation valves (IS3 = 0.1) before core damage. Core recovery in-vessel is guaranteed to fail (RXF) since there is no injection source available and no recovery is taken for the equipment failures. Restoring injection after RPV failure is not allowed to be successful (TRF) since all injection is unavailable and there is uncertainty with regard to the capacity of the diesel fire pump providing success. Because there is no injection into the containment to cool core debris, a high magnitude radionuclide release is assumed with little credit given to the reactor building for source term reduction (RB7 = 0.99). (See discussion under E/H Sequences 1 & 2.)

E/H Sequence 7 (2.1E-8/yr)

This sequence is similar to E/H sequence 6 above, where the Level I end state is Class 1A, loss of high pressure injection, and core damage occurs at high RPV pressure (RPV not depressurized). The initiating event is a loss of Division I Emergency AC power (A1X = 4.3E-3/yr) which disables all safety systems that depend on Division I Emergency AC. In this sequence, Division II AC power subsequently fails (A25 = 9.0E-5) resulting in a total loss of all emergency AC power. Service water fails due to loss of AC power which causes the loss of condenser and feedwater as well as room cooling. RCIC is assumed to be unavailable due to high temperature trip, and there are no procedures for preventing this trip given loss of all service water. HPCS is assumed to be unavailable due to loss of room cooling. Thus, there is a total loss of injection to the core. Procedures for preventing RCIC trip under loss of service water conditions are being developed, thus this assumption is conservative and will be considered further in future analyses and updates.

In the Level II model, containment isolation fails as the operators fail to locally close AC powered containment isolation valves (IS3 = 0.1) before core damage. Core recovery in-vessel is guaranteed to fail (RXF) since there is no injection source available and no recovery is taken for the equipment failures. Restoring injection after RPV failure is not allowed to be successful (TRF) since all injection is unavailable, and there is uncertainty with regard to the capacity of the diesel fire pump providing success. Because there is no injection into the containment to cool core debris, a high magnitude radionuclide release is assumed with little credit given to the reactor building for source term reduction (RB7 = 0.99). (See discussion under E/H Sequences 1 & 2 above.)

E/H Sequence 8 (2.1E-8/yr)

This sequence is identical to CDF Sequence 8 where the Level 1 end state is Class IA, loss of high pressure injection and core damage occurs at high RPV pressure (RPV not depressurized). The initiating event is a partial loss of offsite power (KAX = 4.0E-2) to the Division I emergency AC. All Division I service water pump breakers open which causes isolation of RBCLC and TBCLC. Loss of cooling to the condenser, feedwater, and turbine generator equipment requires an immediate shutdown by the operators. Isolation of TBCLC and RBCLC is assumed to result in a low flow trip of the opposite Division service water pumps, thus requiring them to restart. In this sequence, HPCS fails (HS2 = 0.15) and RCIC fails (IC1 = 0.16) resulting in a loss of high pressure injection. Then, the operators fail to depressurize the RPV (OD1 = 1E-3) to allow low pressure injection systems success.

In the Level II model, containment isolation is successful, but the operators fail to depressurize RPV (OI3 = 5.9E-2), thus, there is no chance of low pressure injection success and recovery of the core in-vessel (IRF). Containment flooding is successfully initiated, but the containment does not remain intact during flooding (CX1 = 0.45). The EOPs direct containment flooding using available sources (e.g. service water) to either the RPV or containment. The service water cannot be injected to the RPV therefore it is directed to containment. The time to flood containment is of the same order as the time of core melt progression. If containment vapor suppression is defeated during the flood process (e.g., wetwell is filled with water) before the RPV is breached at high pressure, then containment failure is assured. It is assumed that such an induced containment failure is both catastrophic and energetic.

E/H Sequence 9 (1.9E-8/yr)
This sequence is identical to CDF Sequence 9 where the Level 1 end state is Class IA, loss of high pressure injection and core damage occurs at high RPV pressure (RPV not depressurized). The initiating event is a loss of offsite AC power (LOSP = 0.04/yr) which leads to unavailability of feedwater and the condenser. In this sequence, HPCS fails (HS2 = 0.14) and RCIC fails (IC1 = 0.16) resulting in a loss of high pressure injection. Then, the operators fail to depressurize the RPV (OD1 = 1E-3) to allow low pressure injection systems success.

In the Level II model, containment isolation is successful, but the operators fail to depressurize RPV (OI3 = 5.9E-2), thus, there is no chance of low pressure injection success and recovery of the core in-vessel (IRF). Containment flooding is successfully initiated, but the containment does not remain intact during flooding (CX1 = 0.45). The EOPs direct containment flooding using available sources (e.g. service water) to either the RPV or containment. The service water cannot be injected to the RPV therefore it is directed to containment. The time to flood containment is of the same order as the time of core melt progression. If containment vapor suppression is defeated during the flood process (e.g., wetwell is filled with water) before the RPV is breached at high pressure, then containment failure is assured. It is assumed that such an induced containment failure is both catastrophic and energetic.

E/H Sequence 10 (1.9E-8/yr)
This sequence is a turbine trip initiating event (ATT = 2.3/yr) which turns into an ATWS sequence due to mechanical failures in the Scram system (QM1 = 4.3E-6). The redundant reactivity control system, recirculation pump trip, and feedwater runback are all successful. In this sequence feedwater is not restarted by the operators (FW3 = 0.50), RCIC fails (IC1 = 0.16), the operators fail to emergency depressurize in time for low pressure injection (OE1 = 0.16), and the operators fail to restart HPCS (CH2 = 1.0). This leads to a Class IC core damage end state. In the Level II model, the RPV did not depressurize (OP1 = 0.46), thus, there is no chance of low pressure injection success and recovery of core in-vessel (RXF). Containment flooding is successfully initiated, but the containment does not remain intact during flooding (CY1 = 0.45). The EOPs direct containment flooding using available sources (e.g. service water) to either the RPV or containment. The service water cannot be injected to the RPV therefore it is directed to containment. The time to flood containment is of the same order as the time of core melt progression. If containment vapor suppression is defeated during the flood process (e.g., wetwell is filled with water) before the RPV is breached at high pressure, then containment failure is assured. It is assumed that such an induced containment failure is both catastrophic and energetic.

4.6.3.7 Impact of Accident Class on Releases

Each individual functional accident class (Class IA, IB, ... Class V) has unique impacts on the containment and containment response. These impacts can be shown in graphical form by examining the release category distribution for each accident class. This data comes from Table 4.6-5.

Figure 4.6-22 shows that for Class IA (core damage accident caused at high RPV pressure) the dominant radionuclide release mode is high release in the intermediate or long term. There is also a substantial fraction of postulated IA sequences that can be recovered with no release or very low releases because of recovery from core damage with the core still within the RPV.

It should be noted that a substantial fraction of the Class IA sequences have multiple hardware failures (front line or support system) that result in defeating the ability of the operator to depressurize. These hardware failures preclude in-vessel recovery and in many cases these same hardware failures preclude subsequent debris cooling.

In addition, it is found from Figure 4.6-19 that there is a substantial fraction of the "large" radionuclide releases attributed to certain Class IA sequences that also have isolation failures induced by power failures to normally open MOV isolation valves. These types of Class IA sequences that preclude recovery and have coincident isolation failures result in the "large" release.

Figure 4.6-23 shows the Class IB accident sequence (Station Blackout) contributions to release categories. The Class IB examination shows that the AC power recovery both in-vessel and ex-vessel result in a substantial fraction of recovery scenarios for the postulated Class IB accident sequences. Therefore, Class IB does not contribute substantially to radionuclide release frequency.

Figure 4.6-24 shows Class IC accident sequences (ATWS with loss of makeup) contributions to release categories. Because the sequence frequency for Class IC is relatively low, these sequences have a relatively small impact on the overall conclusions regarding radionuclide release.

Figure 4.6-25 provides the Class ID (loss of makeup at low RPV pressure) accident sequence contributions to radionuclide release category.

Class ID accident sequences are a dominant contributor to core damage and are made up of multiple support system failures that preclude both in-vessel recovery and debris cooling. The results of the MAAP calculations coupled with the ABB containment capability evaluation shows that for such sequences the NMP2 containment has a substantial capability to maintain its integrity over the initial period of the core melt progression including 10-30 hours after vessel failure. This passive feature of the NMP2 containment represents a substantial benefit in that it provides a great deal of time for emergency response actions and for innovative solutions to the multiple hardware failures postulated. Neither of these beneficial aspects have been quantitatively included in the assessment. For example, no recovery of failed support system hardware is included in the Level 2 analysis. This is judged to be conservative.

Figures 4.6-26 and 4.6-27 show the Class IIA and IIL (Loss of containment heat removal with containment failure before core damage) contributions to the radionuclide release categories. These releases are all late, i.e., it takes more than 24 hours for this sequence to develop such that a radionuclide release occurs. Therefore, these sequences, while they have the potential to produce high releases, the timing of the release makes them of less interest for early health effects. Note that system recovery and containment failure location determine the radionuclide release magnitude category.

Figure 4.6-28 shows the Class IIT contributions to the radionuclide release categories (loss of containment heat removal sequence with the operator termination of injection and core damage before containment failure). These releases are all late, i.e., it takes more than 24 hours for this sequence to develop from the time when EALs are exceeded. These sequences result in a containment failure at the time when RPV blowdown from high RPV pressure occurs. Because the containment failure is induced at precisely the time that the RPV is blowing down and radionuclides are airborne and energetically released -- the result is a relatively high fraction of High (H) releases.

Figures 4.6-29 and 4.6-30 show the Class IIIB and IIIC contributions to radionuclide releases. These contributions to radionuclide release are small and represent negligible contributions to the release categories. Substantial recovery is possible for Class IIIB during in-vessel core melt progression and therefore there is very little contribution to release.

Figure 4.6-31 shows the Class IV (ATWS with a failed containment) contribution to the radionuclide release category. The ATWS sequences produce early radionuclide releases, therefore the ATWS scenarios can be important contributors, especially to the High/Early category (H/E).

Figure 4.6-32 shows the Class V (ISLOCA) contribution to the radionuclide release categories. The "large" release appears to be a substantial segment of the graph, but because the overall ISLOCA frequency is low, the net impact is small.

4.6.4 Overview Summary of Level 2 Results

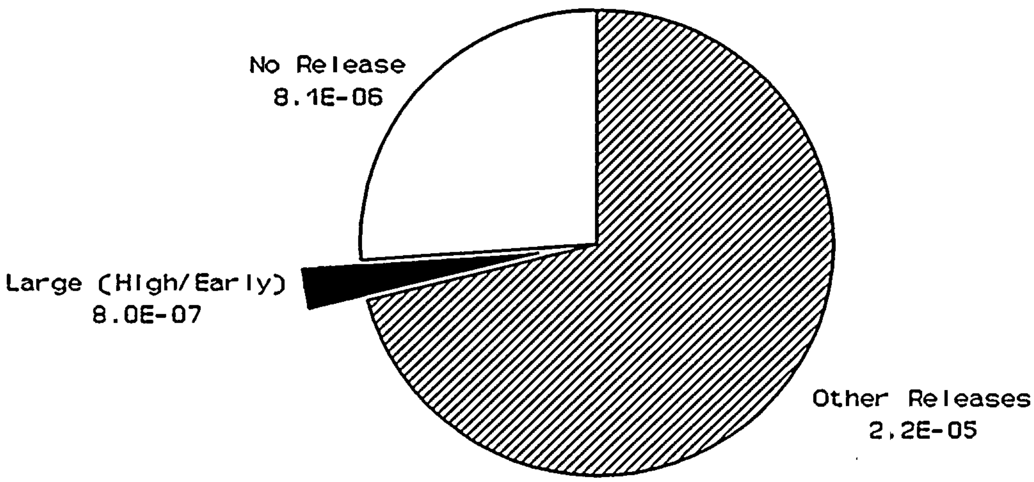
The NMP2 IPE has also evaluated the containment performance by examining the frequency, magnitude, and timing of possible radionuclide releases. The containment evaluation indicates that NMP2 does not have any unusually poor containment performance. A figure of merit that is associated with radionuclide releases is the frequency of a "large" release. The "large" radionuclide release frequency calculated in the NMP2 IPE is 8.0×10^{-7} per year. This frequency confirms that NMP2 poses no undue risk to the public. Figure 4.6-33 shows the relative comparison of large releases compared with other releases.

The NRC in their Severe Accident Policy Statement (1985) stated that:

> *On the basis of current available information, the Commission concludes that existing plants pose no undue risk to the public health and safety and sees no present basis for immediate action on a generic rule making or other regulatory changes for these plants because of severe accident risk.*

The NMP2 IPE has determined that there are no plant specific or unique features of NMP2 that would alter this generic conclusion.

The NMP2 Mark II containment has shown robustness in the face of a wide spectrum of severe accidents. Deterministic MAAP calculations have shown that with no active mitigation, the NMP2 containment has substantial margin to failure and has unique containment features that allow radionuclide release mitigation in magnitude and/or a substantial delay of release. As an example of this robust nature, the following figure shows the spectrum of level II end states. This figure indicates that the "large" radionuclide release, which is a primary IPE figure of merit proposed by the Severe Accident Statement, represents a small fraction of all end states.



These results are comparable to proposed safety goals that have been presented by industry and the NRC. The proposed safety goal for core damage is less than 1.0×10^{-4} per year and a common proposed goal for large release is less than 1.0×10^{-6} per year.

No unusually poor containment performance has been identified, including during core melt progression. In fact, the Mark II containment has many features that provide substantial capability to mitigate against releases -- especially against early releases. Accident management insights have been derived from the model construction, the application of the methodology, the baseline quantification, and the exercising of the model to derive the sensitivity evaluations. These assessments will be considered in the future in establishing accident management plans at NMP2.

The IPE process has identified a number of possible accident management items that may be useful to fold into either the EOPs or into Accident Management guidance. Future NMPC efforts on applying the IPE will make use of these insights and the IPE framework to determine the tradeoffs involved in incorporating these insights as procedural enhancements. Section 4.9 summarizes the principal insights identified as part of the Level 2 IPE process that should be further considered as the NMP2 accident management strategies are developed.

Table 4.6-1

EXAMPLE REPRESENTATIVE ACCIDENT SEQUENCES

MAAP Case	Accident Class	Description	Figure Numbers
LII-IA4-LDNP	IA	- High Pressure Core Melt - RPV Fails at high pressure - No injection available to the RPV or the drywell - No containment venting	4.6-2 (Pressure) 4.6-3 (Temperature)
LII-ID8LDNPV	ID	- Core melt at low RPV pressure - No injection available - No containment venting - No RHR available - Drywell head failure induced by high pressure and temperature	4.6-4 (Pressure) 4.6-5 (Temperature)
LII-2A-2LW	II	- Containment failure due to loss of containment heat removal - Continued core injection until containment fails	4.6-6 (Pressure) 4.6-7 (Temperature)
LII-4A-ILD	IV	- Failure to scram - No effective boron injection - Inadequate containment heat removal - Failure of core makeup after containment fails	4.6-8 (Pressure) 4.6-9 (Temperature)

Table 4.6-2
SUMMARY OF LEVEL 2 OPERATOR ACTIONS INCLUDED IN HUMAN RELIABILITY ANALYSIS

HRA Designator	HRA Description	Sequence	HRA Model	HEP
ISHU-FD--24X	Operator fails to isolate path given isolation signal fails	Class I, III	ASEP	1.0E-1[17]
ISHU-ED-24X	Operator fails to isolate path given isolation signal fails	Class I, III	ASEP	1.0E-1[17]
ISHUDWPEL24X	Operator opens line during normal operations	Class I, III	N/A[1]	1.1E-2[1, 17]
ISHUDWPI-24X	Operator fails to isolate path given isolation signal fails	Class I, III	ASEP	1.0E-1[17]
ISHUDWPSL24X	Operator opens line during normal operations	Class I, III	N/A[1]	1.0E-2[1, 17]
ISHUDWPS-24X	Operator fails to isolate path given situation signal fails	Class I, III	ASEP	1.0E-1[17]
ISHUDWPC-24X	Operator fails to isolate path given situation signal fails	Class I, III	ASEP	1.0E-1[17]
ISHUWWPSL24X	Operator opens line during normal operations	Class I, III	ASEP	5.0E-2[1]
ISHUWWPI-24X	Operator fails to isolate path given isolation signal fails	Class I, III	ASEP	1.0E-1[17]
	Manual operation of containment isolation valves for the DW drain lines	SBO	EPRI 6560L 12/89	0.1
OIHUB4RPV00X	Emergency depressurization during in-vessel core degradation	Class IA, IC	ASEP Lower Bound (used in Level 1 as for OD-1, 2)	0.2[2]
OIHUNODPR00X	Operator fails to maintain depressurization	Class ID	Swain	1E-3[3]

Table 4.6-2
SUMMARY OF LEVEL 2 OPERATOR ACTIONS INCLUDED IN HUMAN RELIABILITY ANALYSIS

HRA Designator	HRA Description	Sequence	HRA Model	HEP
GLPB1A24	Alignment for RHRSW for RPV injection	Class IA, ID • RX/IR - IA, ID - IIIC - IV	ASEP	(4) 0.2 0.1[20] 1.0
RXHUEXC1-00X	Operator intervenes and terminates injection	Class IA, IB, IC, ID, IIIB, IIIC	Swain	1.0E-4[5]
RXHUHWINJ24X	Failure to makeup to condenser hotwell	All	ASEP	0.1[6]
CZHUPOOL-24Y	Plant operated with pool level high (pre-accident)	Core Melt Progression and Subsequent RPV Bottom Head Breach	EPRI (Pre-Accident)	1.0E-5[7]
CZHUCOINJ00X	Operator restores cooling injection after control rods are melted	Core Melt Progression In-Vessel	Time Window	1.0E-4[8]
CTHUPHSLC01X	Failure to inject SLC with boron for low water level	Core Melt Progression In-Vessel	ASEP	1.0[9]
TDNUPROCD24X	Procedure precludes use of drywell sprays	Core Melt Progression In-Vessel	Not Required	1.0[10]
TDHURSWDN24X	Operator fails to align RHRSW for drywell spray	Core Melt Progression In-Vessel	Not Required	1.0[10]
TDHUMPREC24X	Operator fails to recover high pressure injection systems	Long Term Recovery	Recovery	0.9
TDHULPREC24X	Operator fails to recover low pressure injection system	Long Term Recovery	Recovery	0.9

Table 4.6-2
SUMMARY OF LEVEL 2 OPERATOR ACTIONS INCLUDED IN HUMAN RELIABILITY ANALYSIS

HRA Designator	HRA Description	Sequence	HRA Model	HEP
TDHURHRSW24X	Operator fails to provide makeup to the RPV using RHRSW	Core Melt Progression Ex-Vessel	ASEP	0.1[13]
GVHUVENT-24X	Combustible Gas Control	De-Inerted + Core Damage Sequence	ASEP	1E-2
FCHUNDEOP00X	Flood Containment	Core Damage and Failure to Contain In-Vessel	ASEP	0.1[14]
	RPV Vent	Containment Flood Scenario	ASEP	1E-1[16]
FDHUDWVP-00X	Drywell Vent	Containment Flood Scenario	ASEP	1E-1[15]
VCHUNDIMP00X	Containment Vent	Severe Accident Overpressure	ASEP	1E-2
RNHUNOIS000X	Continue Flooding Reactor Building Compartment to Mitigate Release	V-SEQUENCE	EPRI	1.0
RNHUFLUND00X	AC Power Recovery (19)	(19)	(19)	(19)
HRHURHRCL00X	Operator Fails to Initiate Suppression Pool Cooling	Class IA, IB, IC, ID, IIIB, IIIC	PRA	1E-6

NOTES TO TABLE 4.6-2

(1) Although this action could be construed as a potential error of commission, the event is estimated to be the percentage of time that the purge/vent valve is normally open during operation as allowed by Technical Specifications and as supported by operating experience.

(2) This event is a "conditional" action given that the operator has been unable to accomplish RPV depressurization before the onset of core damage, as evaluated in the Level 1 PRA event tree models.

The conditional probability is derived be determining the cumulative failure probability to depressurize the RPV based on the time from the first trigger until RPV breach. This conditional failure probability is that fraction of the HEP not accounted for in the Level 1 analysis. However, it should be noted that due to the potentially long time frame, during which the operating crew can accomplish RPV depressurization, the available HRA methodologies cannot provide reliable cumulative HEPs from which a conditional probability can be derived. Therefore, this HEP is based primarily on expert judgement.

(3) Given that the operator had initially depressurized the RPV using either the SRVs or the TBVs, it is assumed that the dominant failure mode affecting the operator's ability to maintain the RPV depressurized is not human error related. Instead, the operability of depressurization function is more dependent on hardware failures due to random causes or phenomenological effects (e.g., high containment pressure affecting the Dikker SRVs), over the course of the scenario mission time. In this instance, the failure probability accounts for the random failure of the SRVs over the 24 hr. mission time. The operator component is judged to be a negligible contributor because it is expected that the ADS SRVs would be left in the manual open position, as directed in the RPV emergency depressurization EOP, for the duration of the accident.

(4) One of the last resort methods of RPV injection to arrest melt progression in-vessel can be accomplished by the use of the RHR service water cross-tie. This alignment allows service water to be injected into the RPV via the LPCI lines.

The HEP associated with failing to make the alignment assumes that:

- The operator has approximately 40 minutes after RPV water level reaches TAF to recover coolant inventory,
- The action is clearly proceduralized in the EOPs and in Attachment 5, and
- All actions prescribed in the attachment can be accomplished remotely in the Control Room.

Notes to Table 4.6-2 (con't)

(5) This is the crucial mistake which the operators at Three Mile Island made in the 1979 accident. It is considered an unlikely action; operator training has improved greatly since that time. In addition, this is an act of commission, typically not included in a PRA.

However, since it may be considered a "classic" or highly visible action, it is included in the analysis and assigned a low probability similar to SHARP "skill" based assessment, but using the ASEP as the source for the mean estimate (i.e., a time frame of approximately 2 hours is assumed).

(6) This scenario requires the operator to establish make-up to the condenser for long term RPV coolant injection using the condensate system. Therefore, the operator must recognize that the automatic hotwell level control has failed to maintain condensate inventory while controlling RPV water level. It is assumed for this HEP quantification that the operator has at least 1 hour to correct the situation and establish makeup to the hotwell before the condensate inventory is depleted. The operator can accomplish this function by either overriding the valve controller and manually opening the valve, or bypassing the line. The HEP is conservatively assessed to be 1.0E-1/demand, and is applicable only for general transient events, since the rate of hotwell inventory makeup is presumed to be insufficient in the case of LOCAs, or severe accidents (i.e., with the RCS isolated).

(7) This containment wetwell parameter is subject to strict administrative controls that require the operator to maintain suppression water level during power operation in accordance with Technical Specifications. It is judged highly unlikely that the operating crew would establish a suppression pool water level very high out of the normal operating band as to defeat the vapor suppression function as a precondition to the accident. Instead, it is judged more probable that a miscalibrated instrument could provide a false low indication to which the operators would properly respond. Therefore, the conditional probability that the containment is in such an abnormal configuration as the time of the accident is considered remote and is derived using the ASEP method for pre-accident HRA evaluation.

(8) A small time window exists between the time when the control rods begin to melt until the fuel rods also begin to melt. Injection of water during this time frame could create a large reactivity excursion.

This event models the possibility that the operator restores injection within this small time frame. Expert judgement is used to assign a low probability of coincidental occurrence.

Notes to Table 4.6-2 (con't)

(9) Although the NMP2 EOPs direct the operator to flood the RPV for this condition using all available injection sources (including the SLCS), it is conservatively assumed that the operating crew would be unable to prevent recriticality of the core (in its abnormal configuration) upon recovering other low pressure injection systems. This judgement is based on the understanding that if such a vulnerable core configuration could be achieved, the operator would be inclined to use other high capacity systems and restore coolant inventory to near normal band as the primary objective, given that there is no explicit direction in the EOPs to inject boron into the RPV under such circumstances.

(10) It is assumed that, for all sequences, the operator would be procedurally or physically precluded from initiating drywell sprays. This potentially conservative treatment (specifically, int he case of some station blackout scenarios, small and medium LOCAs, and Class ID events), is consistent with the DWSI limit provided in the EOPs.

(11) Since the RPV is breached and at low pressure, it is assumed that the feedwater and the HPCS systems are the only high pressure injection sources capable of providing adequate makeup to the vessel to avoid containment failure. However, if the severe accident has progressed to this point with the operating crew being unable to provide makeup to the RPV using either system, it is judged that the dominant failure mode is equipment failure and not operator inaction. Therefore, although the operating crew has approximately 6 - 10 hours to restore makeup to the RPV, the HEP estimate of 0.9/demand is based on the premise that the operators will be contained in their efforts to effect repairs of the system either because of limited access to the reactor and turbine buildings, or because of limited access to the reactor and turbine buildings, or because the cause of failure is not repairable.

(12) If the accident has progressed to this point with injection systems failed, it is assumed that some major problem exists (e.g., equipment failure, debris plugging of suction lines, during AC power), precluding the operator from repairing these systems in the time frame of interest.

This estimate is also supported by the fact that many areas of the reactor enclosure may be inaccessible at this point in the accident.

The probability of failure is conservatively estimated assuming that there are many hours (i.e., 6 - 10 hours) available prior to the postulated containment over-temperature failure.

Notes to Table 4.6-2 (con't)

(13) Unlike the quantification of TDHULPREC24X, use of the RHRSW system to provide RPV makeup is considered a diverse enough means (i.e., with respect to the low pressure ECCS, BOP, and the fire water system) to warrant separate treatment. Specifically, for this time frame in the accident defined as (6 to 10 hours post vessel breach), it is judged that the dominant failure mode for not establishing vessel makeup using the RHRSW system is operator performance related. Therefore, the quantification of this HEP accounts for the conditional probability that the operating crew fails to align the system for injection, given that this action was not accomplished in the previous time frame (i.e., as defined in node SI). Using the ASEP methodology, it is estimated that this probability is 0.1/demand.

(14) Revision 4 of the EPGs, which have been implemented in the current version of the NMP2 EOPs, direct that containment flooding be undertaken for all postulated severe accidents resulting in RPV breach caused by debris interaction.

Numerous actions are required of the operating crew to accomplish this action; including assessing damage to the RCS and determining the operability of instrumentation necessary to perform containment flooding, aligning systems for external makeup to the RPV and/or containment, and venting the RPV and drywell containment. This basic event models both the operating crew failing to diagnose or initiate flooding before containment conditions (i.e., primarily high drywell temperature) deteriorate to the point that structural breach results.

The evaluation of the HEP assumes that the operating crew has the authority to implement this contingency without consulting with plant management or the TSC, and that adequate instrumentation is operable to accurately indicate the status of the RCS and containment. Based on the ASEP methodology, it is judged that there would be high dependence between the two senior control room personnel in both recognizing the plant conditions and making the decision to flood the containment. Additionally, it is conservatively assumed that flooding must be initiated within 30 minutes from the time that injection is recovered post vessel breach to cool the core debris. Considering that the operating crew would be under high stress to implement the containment flooding contingency procedure (knowing that RPV venting to the environment would subsequently be required), the HEP is assigned a value of 0.1/demand.

Notes to Table 4.6-2 (con't)

(15) Implementation of the containment flooding contingency procedure does not alleviate the responsibility of the operator from maintaining containment conditions within acceptable limits throughout the evolution. In fact, as containment water level rises, the possibility that non-condensable gases become concentrated in the drywell to the point where overpressure becomes a concern also increases. Therefore, this action is defined as the operation of drywell vent path to relieve containment overpressure and maintain containment integrity during the course of the flooding evolution.

The time frame available to the operator to successfully implement containment pressure control is defined by the point at which unmitigated overpressure conditions result in containment breach. This time period is conservatively estimated to be 30 min. (Note that combustible gas concentration and the potential for hydrogen combustion was not considered when determining the allowable time frame for operator action, since the containment is assumed to have remained isolated.)

Again, given the dependence on two senior control room operators to recognize the conditions and initiate this action under stressful conditions, it is determined that the conditional probability for failing to vent the drywell is 1.0E-2/demand. Additionally, the quantification of this HEP is based on the assumption that a remotely operable containment venting system is installed.

(16) This operator action is an essential part of the containment flood contingency procedure. The containment flood procedure and the associated RPV vent procedure (i.e., Attachment 12) is implemented with the goal of preventing any residual fuel and core debris inside the RPV from remaining uncovered due to the vessel being at a greater pressure than the drywell as the containment is flooded to the height of TAF.

The EOP bases instruct the operator to vent the RPV to allow the water level in the containment drywell and inside the RPV to equalize and cool any fuel or debris remaining in the vessel. The options available to the operator include using the condenser by directing steam from the RPV via the main steam lines or steam line drains to the condenser.

The time frame for accomplishing this action is defined by the rate at which the containment is being flooded. Successful venting of the RPV implies that the RPV internal pressure is less than or equal to containment drywell pressure by the time that the lower head of the vessel (the presumed location of a postulated breach) is submerged. Assuming that the torus water level was within Technical Specification prescribed limits upon initiation of the flooding procedure, it is conservatively estimated that the operator has 2 hours after entering the flood contingency procedure to establish RPV venting.

Notes to Table 4.6-2 (con't)

The quantification of the HEP was performed assuming the following information:

- EOP Attachment 12 prescribes the appropriate actions to vent the RPV.
- The operating crew has clear authority to undertake this action if directed by the EOPs.
- All actions for opening a main steam line can be accomplished remotely from the control room.
- All support systems (e.g., AC power, N_2) are available to operate the MSIVs, and the condenser intact.

By applying the ASEP, the failure probability to vent the RPV is conservatively estimated at 1.0E-2/ demand.

(17) Screening value used as conservative estimate. Value not critical for IPE assessment.

(18) Intentionally Left Blank

(19) This is not an operator error in the same context as other actions being evaluated. Rather this is a combination of both hardware recovery, hardware availability, and operator action to successfully align AC power. It is based on experiential data and not derived with HRA techniques.

(20) No recovery taken in the Level 1 for use of RHRSW for injection. Therefore this includes all recovery.

Table 4.6-3
Summary of the Core Damage Accident Sequence Subclasses

Accident Class Designator	Subclass	Definition	Level 1 Frequency (per Rx Yr)
Class I	A	Accident sequences involving loss of inventory makeup in which the reactor pressure remains high.	5.8E-6
	B	Accident sequences involving a station blackout and loss of coolant inventory makeup.	5.5E-6
	C	Accident sequences involving a loss of coolant inventory induced by an ATWS sequence with containment intact.	2.2E-7
	D	Accident sequences involving a loss of coolant inventory makeup in which reactor pressure has been successfully reduced to 200 psi.	9.1E-6**
Class II	A	Accident sequences involving a loss of containment heat removal with the RPV initially intact; core damage induced post containment failure.	4.7E-6
	L	Accident sequences involving a loss of containment heat removal with the RPV breached but no initial core damage; core damage induced post containment failure.	3.0E-7
	T	Accident sequences involving a loss of containment heat removal with the RPV initially intact; core damage induced post high containment pressure.	4.2E-6
	V	Class IIA or IIL except that the vent operates as designed; loss of makeup occurs at some time following vent initiation. Suppression pool saturated but intact.	---
Class III (LOCA)	A	Accident sequences leading to core damage conditions initiated by vessel rupture where the containment integrity is not breached in the initial time phase of the accident.	N/A
	B	Accident sequences initiated or resulting in small or medium LOCAs for which the reactor cannot be depressurized prior to core damage occurring.	4.2E-7
	C	Accident sequences initiated or resulting in medium or large LOCAs for which the reactor is a low pressure and no effective injection is available.	6.4E-9
	D	Accident sequences which are initiated by a LOCA or RPV failure and for which the vapor suppression system is inadequate, challenging the containment integrity with subsequent failure of makeup systems.	1.1E-8

** Principally composed of sequences with loss of AC or DC power on both buses; little recovery available.

Table 4.6-3
Summary of the Core Damage Accident Sequence Subclasses

Accident Class Designator	Subclass	Definition	Level 1 Frequency (per Rx Yr)
Class IV (ATWS)	A	Accident sequences involving failure of adequate shutdown reactivity with the RPV initially intact; core damage induced post containment failure.	8.0E-7
	L	Accident sequences involving a failure of adequate shutdown reactivity with the RPV initially breached (e.g., LOCA or SORV); core damage induced post containment failure.	7.2E-8
Class V	--	Unisolated LOCA outside containment	2.5E-8
		TOTAL	3.1E-5

Table 4.6-4

SUMMARY OF CONTAINMENT EVALUATION

INPUT | OUTPUT

Nine Mile Point 2 PRA LEVEL 1		CET EVALUATION	
Core Damage Frequency	Characterize Release	Release Bin	Release Frequency (per Year)
3.1E-5/year	Little or No Release	OK	8.1E-6
	Low Public Risk Impact	LL & Late	1.9E-6
		LL & I	5.8E-8
		LL & E	3.6E-7
		L & Late[1]	2.2E-6
		L & I	2.6E-8
		L & E	3.3E-7
	Moderate Release	M & Late[1]	7.8E-7
		M & I	1.4E-7
		M & E	8.0E-7
	High Release	H & Late[1]	4.4E-6
		H & I	1.1E-5
		H & E	8.0E-7
TOTAL			3.1E-5

[1] One of the areas that PRA tools are somewhat limited is in the estimation of recovery or repair during extended times such as 24 hours. Some estimates would indicate that response over such an extended time could be very extensive and highly successful. Therefore, it can be argued that virtually no accidents that take beyond 24 hours to release should be considered to be a significant potential contributor to public risk.

Table 4.6-5
Summary Table of Release vs. Accident Class[1, 7]

Class	NOREL	LL[6]	L/E	L/I	L/L	M/E	M/I	M/L	H/E	H/I	H/L	Total Release[7]	Total[7]
IA	2.4E-6	ε	6.0E-10	9.6E-11	2.5E-9	3.6E-8	4.5E-8	3.6E-8	3.9E-7	2.1E-6	3.9E-8	2.6E-6	5.0E-6
IB	5.1E-6	7.3E-7	1.9E-7	1.6E-10	5.5E-10	9.4E-9	3.0E-10	3.6E-10	5.0E-8	3.4E-8	9.8E-11	1.0E-6	6.1E-6
IC	1.5E-9	ε	9.8E-8	---	ε	6.0E-9	4.7E-9	ε	3.2E-8	6.9E-10	ε	1.4E-7	1.4E-7
ID	ε	ε	ε	---	ε	9.7E-10	9.0E-8	ε	1.7E-7	9.1E-6	1.7E-10	9.3E-6	9.3E-6
IIA	---[5]	9.0E-7	ε	---	1.6E-6	---	---	3.7E-7	---	---	8.3E-7	3.7E-6	3.7E-6
IIL	---	1.3E-7	ε	---	8.3E-8	---	---	1.4E-8	---	---	2.1E-8	2.5E-7	2.5E-7
IIT	---	5.1E-7	ε	---	1.5E-7	---	---	2.4E-7	---	---	2.9E-6	3.8E-6	3.8E-6
IIIB	4.1E-7	ε	ε	2.6E-8	ε	1.2E-8	9.2E-11	ε	8.2E-9	1.0E-8	1.0E-11	5.6E-8	4.6E-7
IIIC	1.7E-11	ε	ε	---	ε	ε	6.6E-11	ε	7.6E-9	4.0E-9	---	1.2E-8	1.2E-8
IIID	---	8.2E-9	ε	---	---	2.8E-9	---	---	ε	---	---	1.1E-8	1.1E-8
IV	---	6.0E-8	ε	---	---	7.5E-7	---	---	1.1E-7	---	---	9.2E-7	9.2E-7
V	---	ε	ε	---	---	2.2E-8	---	---	5.0E-9	---	---	2.7E-8	2.7E-8
Total from Summing Column[3, 7]	7.9E-6	2.3E-6	2.9E-7	2.6E-8	1.8E-6	8.4E-7	1.4E-7	6.6E-7	7.7E-7	1.1E-5	3.8E-6	2.2E-5	3.0E-5[7]

Notes:

(1) Sequence quantification cutoff frequency was set to 1E-11. End State Frequencies between 1E-11 and 1E-8 are presented as < 1E-8 due to potential contributions of sequences less than the cutoff (1E-11). The code ε is used to indicate that no sequence greater than 1E-11 were quantified in the event tree model.

(2) This table is used to determine appropriate contributions for purposes of assessing dominant contributors.

(3) This sum is approximately the same as that developed for the RISKMAN run at 1E-11 cutoff. The assumption is that the distribution of contributors is the same for each case.

(4) Sum of column (i.e., Release Category) from RISKMAN (1E-9 cutoff)/RISKMAN (1E-11 cutoff)

(5) Dashes imply that this release category and accident class is not feasible by definition.

(6) LL column calculated by summing all other releases for a class and subtracting from total release column.

(7) This table was generated from the model quantification immediately prior to the final quantification. As such, results depicted in this table are approximate and are presented to give the reader an appreciation for Level I end-state effects on the Level II end-states. Very minor differences in the totals would be noticed if this table were generated from the final quantification. Therefore, there is no reason to believe that this table is not representative.

Table 4.6-6

RELEASE SEVERITY AND TIMING CLASSIFICATION SCHEME
(SEVERITY, TIMING)

Release Severity Source Term Release Fraction		Release Timing	
Classification Category	Cs Iodide % in Release	Classification Category	Time of Release
High (H)	greater than 10	Late (L)	greater than 24 hours
Moderate (M)	1 to 10	Intermediate (I)	6 to 24 hours
Low (L)	0.1 to 1	Early (E)	less than 6 hours
Low-low (LL)	less than 0.1		
No iodine (OK)	0		

Table 4.6-7
TOP 100 Level II Sequences

Rank.	Initiator.	Index....	Frequency......	Failed and Multi-State Split Fractions.	End State.
1	A2X	235	2.2853E-06	/DA1*A2F*D1F*E1F*SAF*SBF*RWF*TWF*MBF*ASF /CNF*FWF*HSF*ICF*LSF*LCF*LAF*LBF*IBF*SWF /NLF*NMF*HAF*HBF*CAF*CBF/IRF*GVF*TD3*RM7 /ELF	IHI
2	A1X	509	2.1953E-06	/DB1*A1F*D2F*E2F*SAF*SBF*RWF*TWF*MAF*ASF /CNF*FWF*HSF*ICF*LSF*LCF*LAF*LBF*IAF*SWF /NLF*NMF*HAF*HBF*CAF*CBF/IRF*GVF*TD3*RM7 /ELF	IHI
3	BLOSP	627	1.3345E-06	/OGF*KAF*KBF*KRF*A12*A28*NAF*NBF*SAF*SBF *RWF*TWF*MAF*MBF*ASF/I11*G11*U11/NLF*NMF /GV2/ELF	NOREL
4	A1X	671	8.6440E-07	/A1F*SAG*SBL*RWF*TWF*MAF*MBA*ASF/CNF*FWF *HSF*ICF*LSF*LCF*LAF*LBF*IAF*IBF*SWF*FPF /NLF*NMF*HAF*HBF*CAF*CBF/IRF*GVF*TDF*RM7 /ELF	IHI
5	BLOSP	321	8.4814E-07	/OGF*KAF*KBF*KRF*A12*A28*NAF*NBF*SAF*SBF *RWF*TWF*MAF*MBF*ASF/I11*G11*U11/NLF*NMF /IRE*GVF*FDF/ELF	NOREL
6	KAX	1807	7.5646E-07	/KAF*NAF*RWF*TWF*ASF/CNF*FWF*HS2*IC1*OD1 /NLF*NMF/GVF/ELF	NOREL
7	LOSP	391	7.0040E-07	/OGF*KAF*KBF*KRF*NAF*NBF*RWF*TWF*ASF/CNF *FWF*HS2*IC1*OD1/NLF*NMF/GVF/ELF	NOREL
8	A2X	118	4.7218E-07	/A2F*SAH*SBK*RWF*TWF*MBF*ASF/CNF*FWF*HSF *ICF*LBF*IBF*SWF/NLF*HAF*HBF*CAF*CBF*CVF *R1F*CIF/OI1*IRF*GVF*CZF/ELF	LHI
9	FLDG2	2	4.3782E-07	/A1F*A2F*SAF*SBF*RWF*TWF*MAF*MBF*ASF/CNF *FWF*HSF*ICF*SVF*LSF*LCF*LAF*LBF*IAF*IBF *SWF*FPF/NLF*NMF*HAF*HBF*CAF*CBF/IRF*GVF *TDF*RM7/ELF	IHI
10	A2X	899	3.7494E-07	/A2F*SAH*SBK*RWF*TWF*MBF*ASF/CNF*FWF*HSF *ICF*LBF*IBF*SWF/NLF*HAF*HBF*CAF*CBF*CVF *R1F*CIF/GVF*HRF*VCF*NC1*MUF*RMF/ELF	LHI
11	FLDG2	5	3.6437E-07	/A1F*A2F*SAF*SBF*RWF*TWF*MAF*MBF*ASF/CNF *FWF*HSF*ICF*SVF*LSF*LCF*LAF*LBF*IAF*IBF *SWF*FPF/NLF*NMF*HAF*HBF*CAF*CBF/OI1*IRF *GVF*TDF*RM7/ELF	IHI
12	BLOSP	562	3.5313E-07	/OGF*KAF*KBF*KRF*A12*A28*NAF*NBF*SAF*SBF *RWF*TWF*MAF*MBF*ASF/I11*G11*I21*G21*U21 *S11/NLF*NMF/GV2/ELF	NOREL
13	BLOSP	570	3.4956E-07	/OGF*KAF*KBF*KRF*A12*A28*NAF*NBF*SAF*SBF *RWF*TWF*MAF*MBF*ASF/I11*G11*I21*G21*U21 *S11/NLF*NMF/IRD*GVF*FDF/ELF	NOREL
14	KBX	562	3.3639E-07	/KBF*A11*NBF*SAF*SBF*RWF*TWF*MAF*MBA*ASF /CNF*FWF*HSF*ICF*LSF*LCF*LAF*LBF*IAF*IBF *SWF*FPF/NLF*NMF*HAF*HBF*CAF*CBF/IRF*GVF *TDF*RM7/ELF	IHI
15	D1X	317	3.0199E-07	/A21*D1F*E1F*SAF*SBF*RWF*TWF*MBF*ASF/CNF *FWF*HSF*ICF*LSF*LCF*LAF*LBF*IBF*SWF/NLF *NMF*HAF*HBF*CAF*CBF/IRF*GVF*TD3*RM7/ELF	IHI
16	D2X	157	2.9012E-07	/A11*D2F*E2F*SAF*SBF*RWF*TWF*MAF*ASF/CNF *FWF*HSF*ICF*LSF*LCF*LAF*LBF*IAF*SWF/NLF *NMF*HAF*HBF*CAF*CBF/IRF*GVF*TD3*RM7/ELF	IHI
17	A1X	562	2.5923E-07	/DB1*A1F*D2F*E2F*SAF*SBF*RWF*TWF*MAF*MBA *ASF/CNF*FWF*HSF*ICF*LSF*LCF*LAF*LBF*IAF *IBF*SWF*FPF/NLF*NMF*HAF*HBF*CAF*CBF/IRF *GVF*TDF*RM7/ELF	IHI

Table 4.6-7
TOP 100 Level II Sequences

Rank.	Initiator.	Index....	Frequency......	Failed and Multi-State Split Fractions.	End State.
18	A2X	284	2.5497E-07	/DA1*A2F*D1F*E1F*SAF*SBF*RWF*TWF*MAA*MBF *ASF/CNF*FWF*HSF*ICF*LSF*LCF*LAF*LBF*IAF *IBF*SWF*FPF/NLF*NMF*HAF*HBF*CAF*CBF/IRF *GVF*TDF*RM7/ELF	IHI
19	MLOCA	88	2.2570E-07	//HS1*OD2/NMF/GV3/ELF	NOREL
20	ATT	74	2.1070E-07	//QM1*SL1/NLF*NMF/ISF/NFF*RXF*FCF*RBF	EMED
21	LOF	152	1.9688E-07	//FWF*HS1*IC1*OD1/NLF*NMF/GV3/ELF	NOREL
22	BLOSP	943	1.9059E-07	/OGF*KAF*KBF*KRF*DB1*A12*A2F*NAF*NBF*D2F *UBF*E2F*SAF*SBF*RWF*TWF*MAF*MBF*ASF/I11 *G1F*U1F/NLF*NMF/GV2/ELF	NOREL
23	BLOSP	458	1.8757E-07	/OGF*KAF*KBF*KRF*DA1*A1F*A29*NAF*NBF*D1F *UAF*E1F*SAF*SBF*RWF*TWF*MAF*MBF*ASF/I11 *G1F*U1F/NLF*NMF/GV2/ELF	NOREL
24	A2X	405	1.8248E-07	/A11*A2F*SAF*SBF*RWF*TWF*MAF*MBF*ASF/CNF *FWF*HSF*ICF*SVF*LSF*LCF*LAF*LBF*IAF*IBF *SWF*FPF/NLF*NMF*HAF*HBF*CAF*CBF/IRF*GVF *TDF*RM7/ELF	IHI
25	A1X	467	1.8189E-07	/A1F*A25*SAF*SBF*RWF*TWF*MAF*MBF*ASF/CNF *FWF*HSF*ICF*SVF*LSF*LCF*LAF*LBF*IAF*IBF *SWF*FPF/NLF*NMF*HAF*HBF*CAF*CBF/IRF*GVF *TDF*RM7/ELF	IHI
26	A2X	408	1.5187E-07	/A11*A2F*SAF*SBF*RWF*TWF*MAF*MBF*ASF/CNF *FWF*HSF*ICF*SVF*LSF*LCF*LAF*LBF*IAF*IBF *SWF*FPF/NLF*NMF*HAF*HBF*CAF*CBF/OI1*IRF *GVF*TDF*RM7/ELF	IHI
27	KBX	1574	1.5186E-07	/KBF*NBF*RWF*TWF*ASF/CNF*FWF*HS1*IC1*OD1 /NLF*NMF/GVF/ELF	NOREL
28	A1X	470	1.5138E-07	/A1F*A25*SAF*SBF*RWF*TWF*MAF*MBF*ASF/CNF *FWF*HSF*ICF*SVF*LSF*LCF*LAF*LBF*IAF*IBF *SWF*FPF/NLF*NMF*HAF*HBF*CAF*CBF/OI1*IRF *GVF*TDF*RM7/ELF	IHI
29	BLOSP	349	1.4333E-07	/OGF*KAF*KBF*KRF*A12*A28*NAF*NBF*SAF*SBF *RWF*TWF*MAF*MBF*ASF/I11*G11*U11/NLF*NMF /IS3/DCF*RB2	ELLO
30	A1X	523	1.3302E-07	/DB1*A1F*D2F*E2F*SAF*SBF*RWF*TWF*MAF*ASF /CNF*FWF*HSF*ICF*LSF*LCF*LAF*LBF*IAF*IBA *SWF*FPF/NLF*NMF*HAF*HBF*CAF*CBF/IRF*GVF *TDF*RM7/ELF	IHI
31	ATT	353	1.3243E-07	//QM1*FW3/NLF*MO1*ILF*CH1/ISF/NFF*RXF*FC F*RBF	EMED
32	LOC	28	1.3179E-07	//CNF*LA1*LBA/NLF*HAF*HBF*CAF*CBF*CV1*R1 F*CI1/OI1*IRF*GVF*CZF/ELF	LHI
33	LOSP	1594	1.2974E-07	/OGF*KAF*KBF*KRF*A24*NAF*NBF*SBF*RWF*TWF *MBF*ASF/CNF*FWF*HS4*IC1*SV7*LBF*IBF*SWF /NLF*NMF*HBF/GVF/ELF	NOREL
34	IORV	162	1.2882E-07	//HSA*OD2/NMF/GV3/ELF	NOREL
35	ASX	500	1.2646E-07	/ASF/CNF*FWF*HS1*IC1*OD1/NLF*NMF/GVF/ELF	NOREL
36	A1X	170	1.2191E-07	/A1F*SAG*SBL*RWF*TWF*MAF*ASF/CNF*FWF*HSF *ICF*LSF*LAF*IAF*SWF/NLF*HAF*HBF*CAF*CBF *CV2*R1F*CIF/OI1*IRF*GVF*CZF/ELF	LHI
37	BLOSP	1159	1.2113E-07	/OGF*KAF*KBF*KRF*DB1*A12*A2F*NAF*NBF*D2F *UBF*E2F*SAF*SBF*RWF*TWF*MAF*MBF*ASF/I11 *G1F*U1F/NLF*NMF/IRE*GVF*FDF/ELF	NOREL

Table 4.6-7
TOP 100 Level II Sequences

Rank.	Initiator.	Index....	Frequency......	Failed and Multi-State Split Fractions.	End State.
38	BLOSP	464	1.1920E-07	/OGF*KAF*KBF*KRF*DA1*A1F*A29*NAF*NBF*D1F *UAF*E1F*SAF*SBF*RWF*TWF*MAF*MBF*ASF/I11 *G1F*U1F/NLF*NMF/IRE*GVF*FDF/ELF	NOREL
39	A2X	206	1.1609E-07	/A2F*SAH*SBK*RWF*TWF*MAA*MBF*ASF/CNF*FWF *HSF*ICF*LSF*LCF*LAF*LBF*IAF*IBF*SWF*FPF /NLF*NMF*HAF*HBF*CAF*CBF/IRF*GVF*TDF*RM7 /ELF	IHI
40	LOSP	1145	1.1311E-07	/OGF*KAF*KBF*KRF*A12*NAF*NBF*SAF*RWF*TWF *MAF*ASF/CNF*FWF*HS4*IC1*SV5*LSF*LAF*IAF /NLF*NMF*HAF*CAF/GVF/ELF	NOREL
41	TT	671	1.0992E-07	/DA1*A21*D1F*E1F*SAF*SBF*RWF*TWF*MBF*ASF /CNF*FWF*HSF*ICF*LSF*LCF*LAF*LBF*IBF*SWF /NLF*NMF*HAF*HBF*CAF*CBF/IRF*GVF*TD3*RM7 /ELF	IHI
42	LOSP	1600	1.0783E-07	/OGF*KAF*KBF*KRF*A24*NAF*NBF*SBF*RWF*TWF *MBF*ASF/CNF*FWF*HS4*IC1*SV7*LBF*IBF*SWF /NLF*NMF*HBF/OI1*IRF*GVF*FIF/ELF	NOREL
43	TT	1601	1.0561E-07	/DB1*A11*D2F*E2F*SAF*SBF*RWF*TWF*MAF*ASF /CNF*FWF*HSF*ICF*LSF*LCF*LAF*LBF*IAF*SWF /NLF*NMF*HAF*HBF*CAF*CBF/IRF*GVF*TD3*RM7 /ELF	IHI
44	KBX	650	1.0384E-07	/KBF*NBF*SAC*SBO*RWF*TWF*ASF/CNF*FWF*HSF *ICF*SWF/NLF*HAF*HBF*CAF*CBF*CV2*R1F*CIF /OI1*IRF*GVF*CZF/ELF	LHI
45	KAX	254	9.9965E-08	/KAF*NAF*SA7*SBT*RWF*TWF*ASF/CNF*FWF*HSF *ICF*SWF/NLF*HAF*HBF*CAF*CBF*CV2*R1F*CIF /OI1*IRF*GVF*CZF/ELF	LHI
46	A1X	152	9.6807E-08	/A1F*SAG*SBL*RWF*TWF*MAF*ASF/CNF*FWF*HSF *ICF*LSF*LAF*IAF*SWF/NLF*HAF*HBF*CAF*CBF *CV2*R1F*CIF/GVF*HRF*VCF*NC1*MUF*RMF/ELF	LHI
47	A2X	896	9.2122E-08	/A2F*SAH*SBK*RWF*TWF*MBF*ASF/CNF*FWF*HSF *ICF*LBF*IBF*SWF/NLF*HAF*HBF*CAF*CBF*CVF *R1F*CIF/GVF*HRF*VCF/ELF	LLLO
48	RWX	192	8.7616E-08	/RWF*ASF/CNF*FWF*HS1*IC1*OD1/NLF*NMF/GVF /ELF	NOREL
49	BLOSP	353	8.7442E-08	/OGF*KAF*KBF*KRF*A12*A28*NAF*NBF*SAF*SBF *RWF*TWF*MAF*MBF*ASF/I11*G11*U11/NLF*NMF /IS3/DCF*RXE*RB2	ELO
50	KBX	632	8.2457E-08	/KBF*NBF*SAC*SBO*RWF*TWF*ASF/CNF*FWF*HSF *ICF*SWF/NLF*HAF*HBF*CAF*CBF*CV2*R1F*CIF /GVF*HRF*VCF*NC1*MUF*RMF/ELF	LHI
51	LOC	869	8.0933E-08	//CNF*LA1*LBA/NLF*HAF*HBF*CAF*CBF*CV1*R1 F*CF1/ISF/NFA*WW3*RXF	LLLO
52	KBX	815	7.9920E-08	/KBF*NBF*SAC*SBO*RWF*TWF*MAA*MBA*ASF/CNF *FWF*HSF*ICF*LSF*LCF*LAF*LBF*IAF*IBF*SWF *FPF/NLF*NMF*HAF*HBF*CAF*CBF/IRF*GVF*TDF *RM7/ELF	IHI
53	KAX	236	7.9379E-08	/KAF*NAF*SA7*SBT*RWF*TWF*ASF/CNF*FWF*HSF *ICF*SWF/NLF*HAF*HBF*CAF*CBF*CV2*R1F*CIF /GVF*HRF*VCF*NC1*MUF*RMF/ELF	LHI
54	KBX	1200	7.7797E-08	/KBF*KR1*A23*NBF*RWF*TWF*MBF*ASF/CNF*FWF *LA1*LBF*IBF*SWF/NLF*HAF*HBF*CAF*CBF*CVF *R1F*CF4/ISF/NFA*WW3*RXF*TRF*RB7	LHI

Table 4.6-7
TOP 100 Level II Sequences

Rank.	Initiator.	Index....	Frequency......	Failed and Multi-State Split Fractions.	End State.
55	KAX	896	7.7064E-08	/KAF*NAF*SA7*SBT*RWF*TWF*HAA*HBA*ASF/CNF *FWF*HSF*ICF*LSF*LCF*LAF*LBF*IAF*IBF*SWF *FPF/NLF*NMF*HAF*HBF*CAF*CBF/IRF*GVF*TDF *RM7/ELF	IHI
56	A1X	236	7.6972E-08	/DB1*A1F*D2F*E2F*ME1*SAF*SBF*RWF*TWF*MAF *ASF/CNF*FWF*HSF*ICF*LSF*LCF*LAF*LBF*IAF *IBF*SWF*FPF/NLF*NMF*HAF*HBF*CAF*CBF/IRF *GVF*TDF*RM7/ELF	IHI
57	A2X	303	7.5706E-08	/DA1*A2F*D1F*E1F*ME1*SAF*SBF*RWF*TWF*MBF *ASF/CNF*FWF*HSF*ICF*LSF*LCF*LAF*LBF*IAF *IBF*SWF*FPF/NLF*NMF*HAF*HBF*CAF*CBF/IRF *GVF*TDF*RM7/ELF	IHI
58	LOC	833	7.4706E-08	//CNF*LA1*LBA/NLF*HAF*HBF*CAF*CBF*CV1*R1 F*CF1/ISF/	LLLO
59	KBX	118	7.1946E-08	/KBF*NBF*RWF*TWF*ASF/CNF*FWF*LA1*LBA/NLF *HAF*HBF*CAF*CBF*CV2*R1F*CF4/ISF/NFA*WW3 *RXF*FBF	LLO
60	KBX	1196	6.6272E-08	/KBF*KR1*A23*NBF*RWF*TWF*MBF*ASF/CNF*FWF *LA1*LBF*IBF*SWF/NLF*HAF*HBF*CAF*CBF*CVF *R1F*CF4/ISF/NFA*RXF*TRF*RB7	LHI
61	KBX	1323	6.6201E-08	/KBF*KR1*NBF*SAC*SBO*RWF*TWF*MBF*ASF/CNF *FWF*HSF*ICF*LBF*IBF*SWF/NLF*HAF*HBF*CAF *CBF*CVF*R1F*CIF/OI1*IRF*GVF*CZF/ELF	LHI
62	LOSP	763	6.4279E-08	/OGF*KAF*KBF*KRF*A24*NAF*NBF*SBF*RWF*TWF *MBF*ASF/CNF*FWF*LA1*LBF*IBF*SWF/NLF*HAF *HBF*CAF*CBF*CVF*R11*CF4/ISF/NFA*WW3*RXF *TRF*RB7	LHI
63	TWX	280	6.2089E-08	/TWF/CNF*FWF*HS1*IC1*OD1/NLF*NMF/GV3/ELF	NOREL
64	KAX	1978	6.1776E-08	/KAF*NAF*RWF*TWF*ASF/CNF*FWF*LA1*LBA/NLF *HAF*HBF*CAF*CBF*CV2*R1F*CF4/ISF/NFA*WW3 *RXF*FBF	LLO
65	KBX	1123	6.1424E-08	/KBF*NBF*RWF*TWF*ASF/CNF*FWF*LA1*LBA/NLF *HAF*HBF*CAF*CBF*CV2*R1F*CF4/ISF/	LLLO
66	ASX	338	5.8454E-08	/ASF/CNF*FWF*LA1*LBA/NLF*HAF*HBF*CAF*CBF *CV2*R1F*CF2/ISF/NFA*WW3*RXF*FBF	LLO
67	LOSP	1028	5.7198E-08	/OGF*KAF*KBF*KRF*NAF*NBF*RWF*TWF*ASF/CNF *FWF*LA1*LBA/NLF*HAF*HBF*CAF*CBF*CV2*R1F *CF4/ISF/NFA*WW3*RXF*FBF	LLO
68	LOC	4	5.5773E-08	//CNF*LA1*LBA/NLF*HAF*HBF*CAF*CBF*CV1*R1 F*CI1/GVF*HRF*VCF*NC1*WR2/ELF	LLO
69	LOSP	759	5.4756E-08	/OGF*KAF*KBF*KRF*A24*NAF*NBF*SBF*RWF*TWF *MBF*ASF/CNF*FWF*LA1*LBF*IBF*SWF/NLF*HAF *HBF*CAF*CBF*CVF*R11*CF4/ISF/NFA*RXF*TRF *RB7	LHI
70	KAX	1607	5.4535E-08	/KAF*KR2*A22*NAF*SAF*SBF*RWF*TWF*MAF*MBF *ASF/CNF*FWF*HSF*ICF*SVF*LSF*LCF*LAF*LBF *IAF*IBF*SWF*FPF/NLF*NMF*HAF*HBF*CAF*CBF /IRF*GVF*TDF*RM7/ELF	IHI
71	KAX	2286	5.2867E-08	/KAF*NAF*E11*E2A*ME2*RWF*TWF*ASF/CNF*FWF *HS2*ICF*LSF*LCF*IAF*IBF*SWF*FPF/NLF*NMF /IRF*GVF*TDF*RM7/ELF	IHI
72	KAX	1949	5.2741E-08	/KAF*NAF*RWF*TWF*ASF/CNF*FWF*LA1*LBA/NLF *HAF*HBF*CAF*CBF*CV2*R1F*CF4/ISF/	LLLO

Table 4.6-7
TOP 100 Level II Sequences

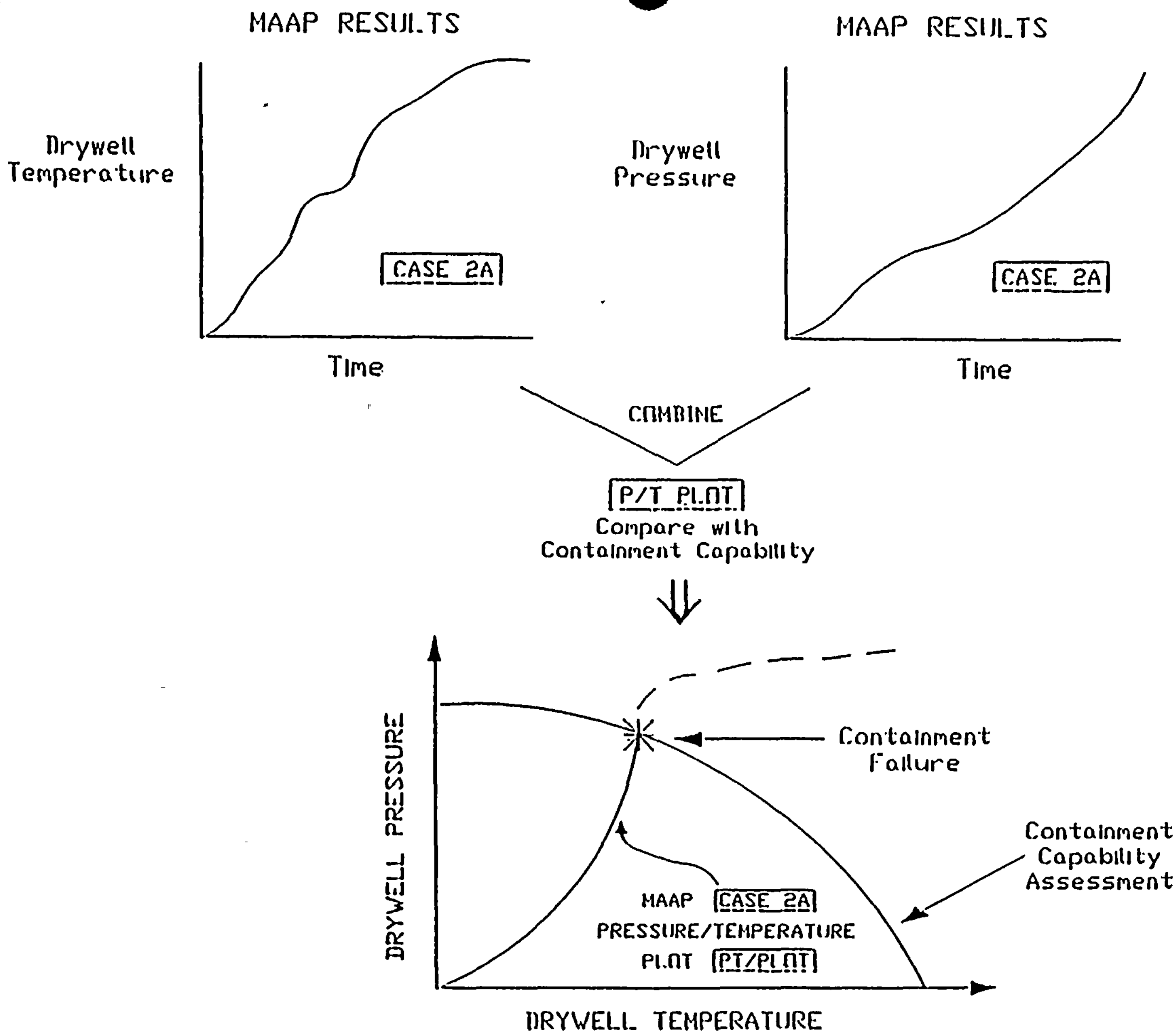
Rank.	Initiator.	Index....	Frequency......	Failed and Multi-State Split Fractions.	End State.
73	KBX	1306	5.2568E-08	/KBF*KR1*NBF*SAC*SBO*RWF*TWF*MBF*ASF/CNF *FWF*HSF*ICF*LBF*IBF*SWF/NLF*HAF*HBF*CAF *CBF*CVF*R1F*CIF/GVF*HRF*VCF*NC1*MUF*RMF /ELF	LHI
74	FLDG2	8	5.1220E-08	/A1F*A2F*SAF*SBF*RWF*TWF*MAF*MBF*ASF/CNF *FWF*HSF*ICF*SVF*LSF*LCF*LAF*LBF*IAF*IBF *SWF*FPF/NLF*NMF*HAF*HBF*CAF*CBF/IS3/DCF *RXF*TRF*RB7	EHI
75	LOC	887	5.0707E-08	//CNF*LA1*LBA/NLF*HAF*HBF*CAF*CBF*CV1*R1 F*CF1/ISF/NFA*DC2*RXF*RB5	LLO
76	LOSP	1845	5.0669E-08	/OGF*KAF*KBF*KRF*A12*NAF*NBF*SAF*RWF*TWF *MAF*ASF/CNF*FWF*HS4*IC1*SV5*LSF*LAF*IAF /NLF*NMF*HAF*CAF/OI1*IRF*GVF*FDF/ELF	NOREL
77	ATT	401	5.0620E-08	//QM1*FW3/NLF*IC1*OE1*CH2/ISF/NFF*RXF*RB F	ELO
78	ASX	309	4.9905E-08	/ASF/CNF*FWF*LA1*LBA/NLF*HAF*HBF*CAF*CBF *CV2*R1F*CF2/ISF/	LLLO
79	FLSW	26	4.9207E-08	/A21*SAF*SBF*RWF*TWF*MBF*ASF/CNF*FWF*HSF *ICF*LBF*IBF*SWF/NLF*HAF*HBF*CAF*CBF*CVF *R1F*CIF/OI1*IRF*GVF*CZF/ELF	LHI
80	LOSP	1685	4.8949E-08	/OGF*KAF*KBF*KRF*NAF*NBF*E11*E2A*ME2*RWF *TWF*ASF/CNF*FWF*HS2*ICF*LSF*LCF*IAF*IBF *SWF*FPF/NLF*NMF/IRF*GVF*TDF*RM7/ELF	IHI
81	LOSP	999	4.8832E-08	/OGF*KAF*KBF*KRF*NAF*NBF*RWF*TWF*ASF/CNF *FWF*LA1*LBA/NLF*HAF*HBF*CAF*CBF*CV2*R1F *CF4/ISF/	LLLO
82	LOC	849	4.8748E-08	//CNF*LA1*LBA/NLF*HAF*HBF*CAF*CBF*CV1*R1 F*CF1/ISF/NFA*RXF	LLLO
83	KBX	1204	4.8512E-08	/KBF*KR1*A23*NBF*RWF*TWF*MBF*ASF/CNF*FWF *LA1*LBF*IBF*SWF/NLF*HAF*HBF*CAF*CBF*CVF *R1F*CF4/ISF/NFA*DC2*RXF*TRF*RB7	LHI
84	KBX	1183	4.7939E-08	/KBF*KR1*A23*NBF*RWF*TWF*MBF*ASF/CNF*FWF *LA1*LBF*IBF*SWF/NLF*HAF*HBF*CAF*CBF*CVF *R1F*CF4/ISF/RXF*FCC	LLO
85	KBX	1034	4.7444E-08	/KBF*A11*NBF*SAF*SBF*RWF*TWF*MAF*ASF/CNF *FWF*HSF*ICF*LSF*LAF*IAF*SWF/NLF*HAF*HBF *CAF*CBF*CV2*R1F*CIF/OI1*IRF*GVF*CZF/ELF	LHI
86	KAX	1610	4.5387E-08	/KAF*KR2*A22*NAF*SAF*SBF*RWF*TWF*MAF*MBF *ASF/CNF*FWF*HSF*ICF*SVF*LSF*LCF*LAF*LBF *IAF*IBF*SWF*FPF/NLF*NMF*HAF*HBF*CAF*CBF /OI1*IRF*GVF*TDF*RM7/ELF	IHI
87	ALOC	70	4.4974E-08	//QM1*CNF/NLF*MOF*ILF*CH1/ISF/NFF*RXF*FC F*RBF	EMED
88	IORV	97	4.3928E-08	//LA1*LBA/HAF*HBF*CAF*CBF*CV1*CFF/ISF/NF A*DC2*RXF*RB5	LLO
89	KBX	1136	4.3334E-08	/KBF*NBF*RWF*TWF*ASF/CNF*FWF*LA1*LBA/NLF *HAF*HBF*CAF*CBF*CV2*R1F*CF4/ISF/NFA*RXF *FBF	LLO
90	FLDG1	292	4.2898E-08	/SAF*RWF*TWF*ASF/CNF*FWF*LB1/NLF*HAF*HBF *CAF*CBF*CV2*R1F*CF4/ISF/NFA*WW3*RXF*FBF	LLO
91	FLDG2	11	4.2629E-08	/A1F*A2F*SAF*SBF*RWF*TWF*MAF*MBF*ASF/CNF *FWF*HSF*ICF*SVF*LSF*LCF*LAF*LBF*IAF*IBF *SWF*FPF/NLF*NMF*HAF*HBF*CAF*CBF/IS3/DCF *OP1*RXF*TRF*RB7	EHI

Table 4.6-7
TOP 100 Level II Sequences

Rank.	Initiator.	Index....	Frequency......	Failed and Multi-State Split Fractions.	End State.
92	LOSP	1850	4.1875E-08	/OGF*KAF*KBF*KRF*A12*NAF*NBF*SAF*RWF*TWF *MAF*ASF/CNF*FWF*HS4*IC1*SV5*LSF*LAF*IAF /NLF*NMF*HAF*CAF/OI1*IRF*GVF*CX1/ELF	EHI
93	KBX	134	4.1692E-08	/KBF*NBF*RWF*TWF*ASF/CNF*FWF*LA1*LBA/NLF *HAF*HBF*CAF*CBF*CV2*R1F*CF4/ISF/NFA*DC2 *RXF*RB5	LLO
94	TT	538	4.1582E-08	/A11*SAG*SBL*RWF*TWF*MAF*MBA*ASF/CNF*FWF *HSF*ICF*LSF*LCF*LAF*LBF*IAF*IBF*SWF*FPF /NLF*NMF*HAF*HBF*CAF*CBF/IRF*GVF*TDF*RM7 /ELF	IHI
95	RWX	353	4.1509E-08	/RWF*ASF/CNF*FWF*LA1*LBA/NLF*HAF*HBF*CAF *CBF*CV2*R1F*CF4/ISF/NFA*WW3*RXF*FBF	LLO
96	BLOSP	393	4.1119E-08	/OGF*KAF*KBF*KRF*A12*A28*NAF*NBF*SAF*SBF *RWF*TWF*MAF*MBF*ASF/I11*G11*OA1*I21*G21 /NLF*NMF/GV2/ELF	NOREL
97	A2X	11	4.1023E-08	/A2F*D11*E1F*SAF*SBF*RWF*TWF*MBF*ASF/CNF *FWF*HSF*ICF*LSF*LCF*LAF*LBF*IBF*SWF/NLF *NMF*HAF*HBF*CAF*CBF/IRF*GVF*TD3*RM7/ELF	IHI
98	BLOSP	396	4.0703E-08	/OGF*KAF*KBF*KRF*A12*A28*NAF*NBF*SAF*SBF *RWF*TWF*MAF*MBF*ASF/I11*G11*OA1*I21*G21 /NLF*NMF/IRD*GVF*FDF/ELF	NOREL
99	KBX	1207	4.0375E-08	/KBF*KR1*A23*NBF*RWF*TWF*MBF*ASF/CNF*FWF *LA1*LBF*IBF*SWF/NLF*HAF*HBF*CAF*CBF*CVF *R1F*CF4/ISF/NFA*DC2*OP8*RXF*TRF*RB7	LHI
100	LOSP	767	4.0082E-08	/OGF*KAF*KBF*KRF*A24*NAF*NBF*SBF*RWF*TWF *MBF*ASF/CNF*FWF*LA1*LBF*IBF*SWF/NLF*HAF *HBF*CAF*CBF*CVF*R11*CF4/ISF/NFA*DC2*RXF *TRF*RB7	LHI

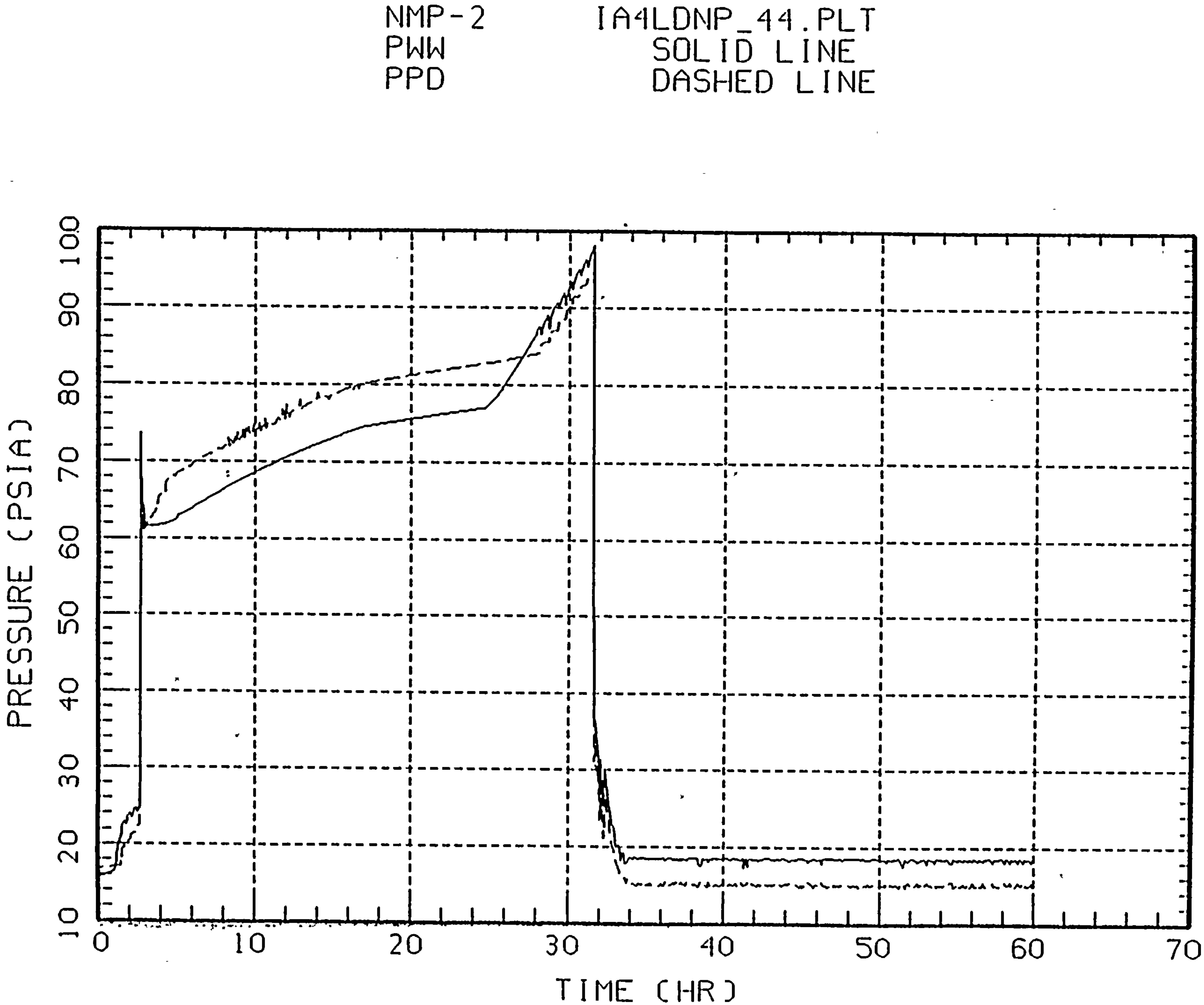
Table 4.6-8
Top Early/High Sequences

E/Hi Rank.	Total Rank.	Initiator.	Frequency......	Failed and Multi-State Split Fractions....	End State.
1.	74	FLDG2	5.1220E-08	/A1F*A2F*SAF*SBF*RWF*TWF*MAF*MBF*ASF/CNF *FWF*HSF*ICF*SVF*LSF*LCF*LAF*LBF*IAF*IBF *SWF*FPF/NLF*NMF*HAF*HBF*CAF*CBF/IS3/DCF *RXF*TRF*RB7	EHI
2.	91	FLDG2	4.2629E-08	/A1F*A2F*SAF*SBF*RWF*TWF*MAF*MBF*ASF/CNF *FWF*HSF*ICF*SVF*LSF*LCF*LAF*LBF*IAF*IBF *SWF*FPF/NLF*NMF*HAF*HBF*CAF*CBF/IS3/DCF *OP1*RXF*TRF*RB7	EHI
3.	92	LOSP	4.1875E-08	/OGF*KAF*KBF*KRF*A12*NAF*NBF*SAF*RWF*TWF *MAF*ASF/CNF*FWF*HS4*IC1*SV5*LSF*LAF*IAF /NLF*NMF*HAF*CAF/OI1*IRF*GVF*CX1/ELF	EHI
4.	151	A2X	2.7355E-08	/DA1*A2F*D1F*E1F*SAF*SBF*RWF*TWF*MBF*ASF /CNF*FWF*HSF*ICF*LSF*LCF*LAF*LBF*IBF*SWF /NLF*NMF*HAF*HBF*CAF*CBF/IS2/DCF*RXF*TR3 *RB7	EHI
5.	155	A1X	2.6278E-08	/DB1*A1F*D2F*E2F*SAF*SBF*RWF*TWF*MAF*ASF /CNF*FWF*HSF*ICF*LSF*LCF*LAF*LBF*IAF*SWF /NLF*NMF*HAF*HBF*CAF*CBF/IS2/DCF*RXF*TR3 *RB7	EHI
6.	196	A2X	2.1348E-08	/A11*A2F*SAF*SBF*RWF*TWF*MAF*MBF*ASF/CNF *FWF*HSF*ICF*SVF*LSF*LCF*LAF*LBF*IAF*IBF *SWF*FPF/NLF*NMF*HAF*HBF*CAF*CBF/IS3/DCF *RXF*TRF*RB7	EHI
7.	197	A1X	2.1279E-08	/A1F*A25*SAF*SBF*RWF*TWF*MAF*MBF*ASF/CNF *FWF*HSF*ICF*SVF*LSF*LCF*LAF*LBF*IAF*IBF *SWF*FPF/NLF*NMF*HAF*HBF*CAF*CBF/IS3/DCF *RXF*TRF*RB7	EHI
8.	199	KAX	2.1032E-08	/KAF*NAF*RWF*TWF*ASF/CNF*FWF*HS2*IC1*OO1 /NLF*NMF/OI3*IRF*GVF*CX1/ELF	EHI
9.	213	LOSP	1.9473E-08	/OGF*KAF*KBF*KRF*NAF*NBF*RWF*TWF*ASF/CNF *FWF*HS2*IC1*OO1/NLF*NMF/OI3*IRF*GVF*CX1 /ELF	EHI
10.	217	ATT	1.8958E-08	//QM1*FW3/NLF*IC1*OE1*CH2/ISF/NFF*OP1*RX F*CY1	EHI
11.	222	ATT	1.8571E-08	//QM1*SL1/NLF*NMF/ISF/NFF*RXF*FCF*SN3*RB F	EHI
12.	233	A2X	1.7767E-08	/A11*A2F*SAF*SBF*RWF*TWF*MAF*MBF*ASF/CNF *FWF*HSF*ICF*SVF*LSF*LCF*LAF*LBF*IAF*IBF *SWF*FPF/NLF*NMF*HAF*HBF*CAF*CBF/IS3/DCF *OP1*RXF*TRF*RB7	EHI
13.	236	A1X	1.7710E-08	/A1F*A25*SAF*SBF*RWF*TWF*MAF*MBF*ASF/CNF *FWF*HSF*ICF*SVF*LSF*LCF*LAF*LBF*IAF*IBF *SWF*FPF/NLF*NMF*HAF*HBF*CAF*CBF/IS3/DCF *OP1*RXF*TRF*RB7	EHI
14.	266	A2X	1.5103E-08	/DA1*A2F*D1F*E1F*SAF*SBF*RWF*TWF*MBF*ASF /CNF*FWF*HSF*ICF*LSF*LCF*LAF*LBF*IBF*SWF /NLF*NMF*HAF*HBF*CAF*CBF/IRF*GVF*CZD/ELF	EHI
15.	280	A1X	1.4508E-08	/DB1*A1F*D2F*E2F*SAF*SBF*RWF*TWF*MAF*ASF /CNF*FWF*HSF*ICF*LSF*LCF*LAF*LBF*IAF*SWF /NLF*NMF*HAF*HBF*CAF*CBF/IRF*GVF*CZD/ELF	EHI



Simplified Flow Chart Showing the Relationship of MAAP Deterministic Results Compared with the Ultimate Containment Capability

Figure 4.6-1



Wetwell (PWW) and Drywell (PPD) Pressures for Loss of Makeup at High RPV Pressure

Figure 4.6-2

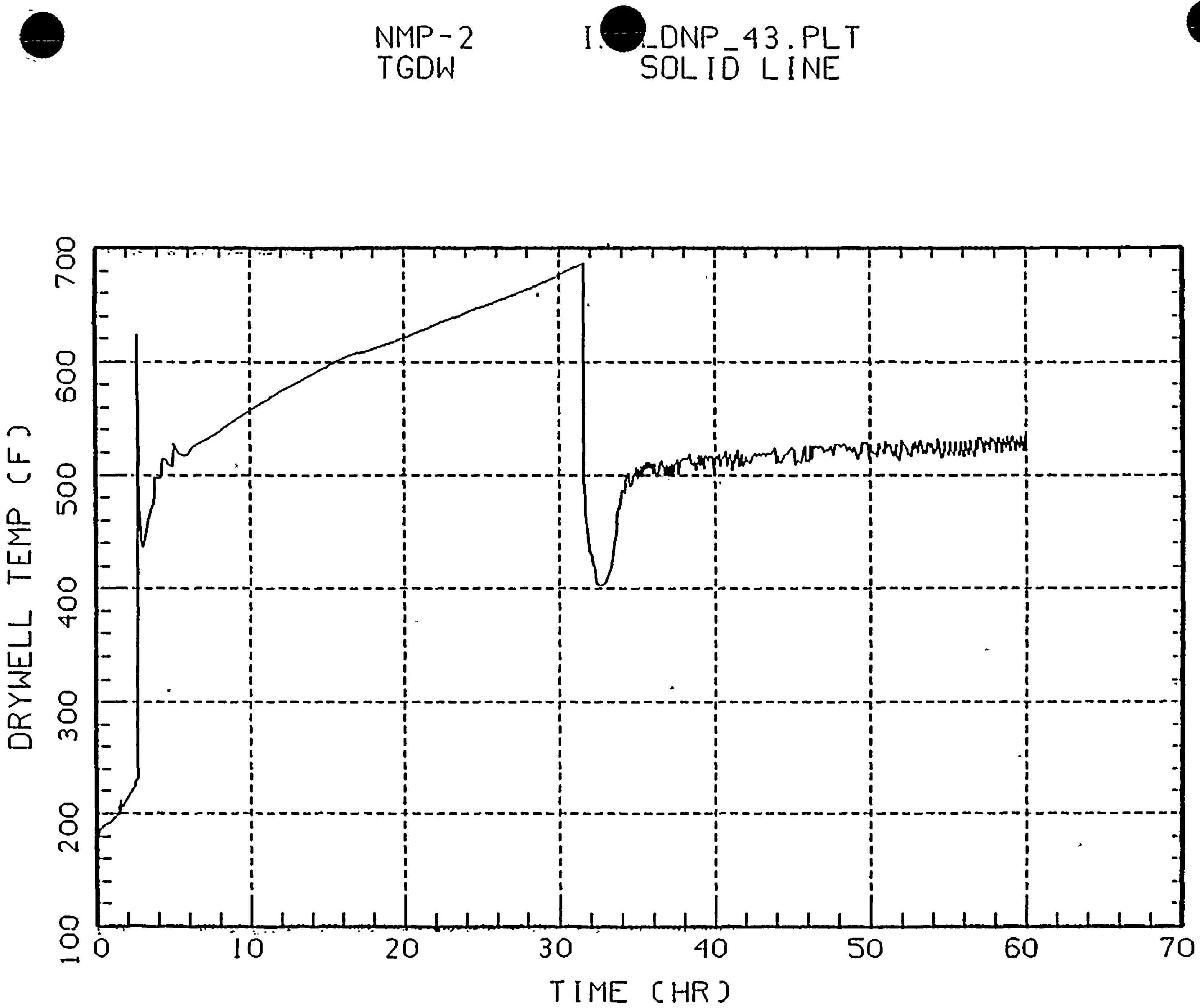


Figure 4.6-3

Drywell Temperature for Loss of Makeup at High RPV Pressure

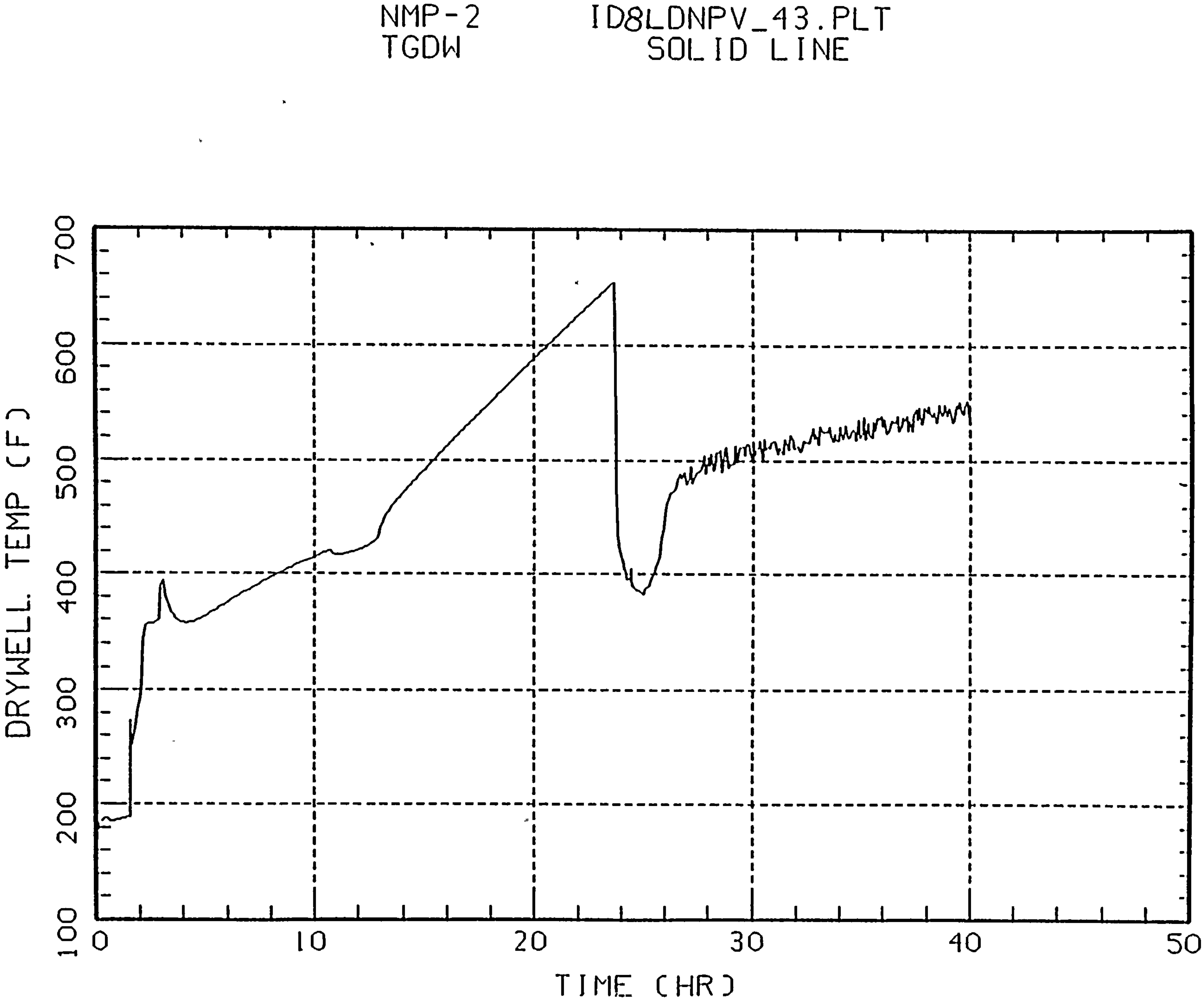


Figure 4.6-4 Drywell Temperature for Class ID Response

C135-90-02-713

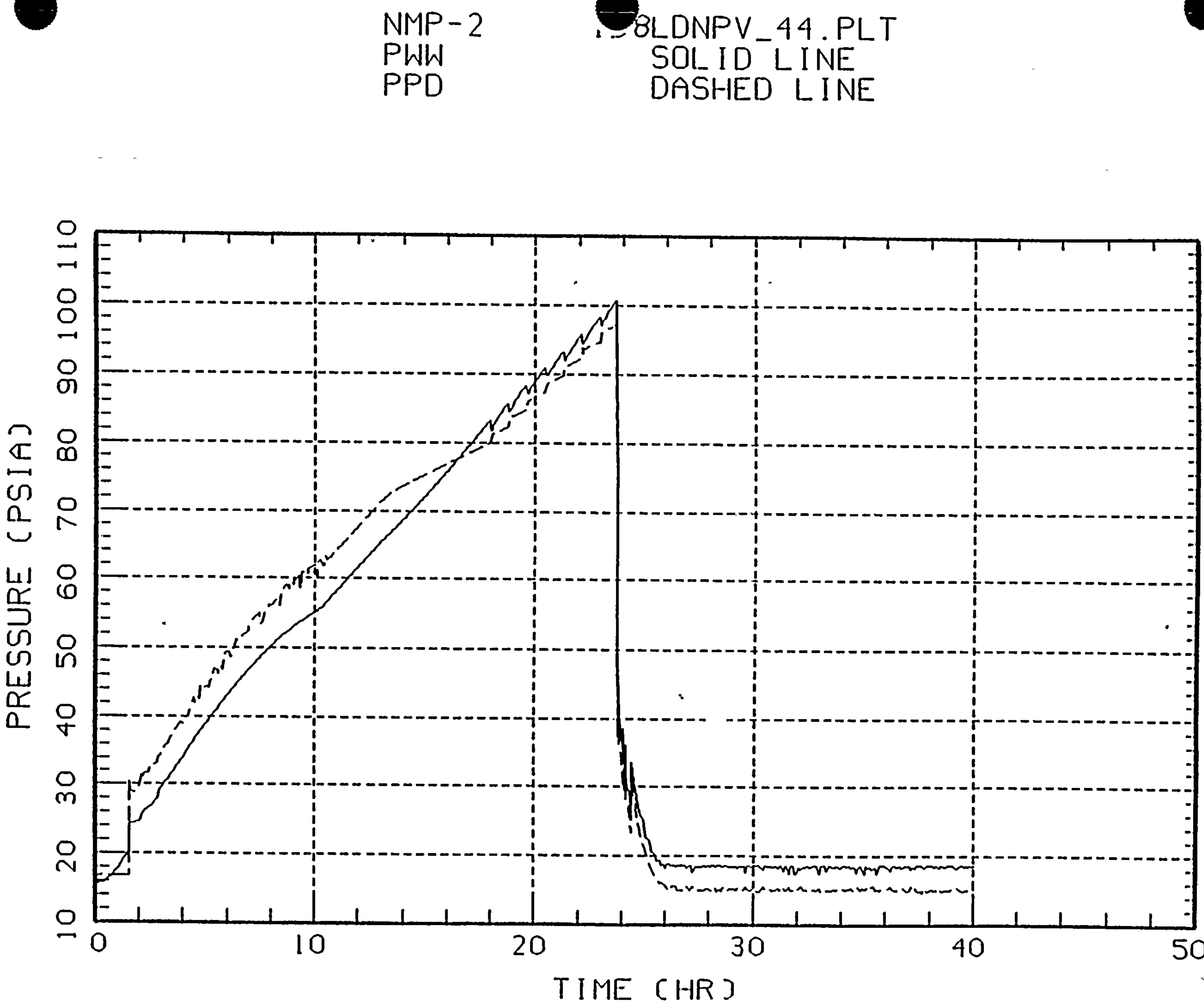
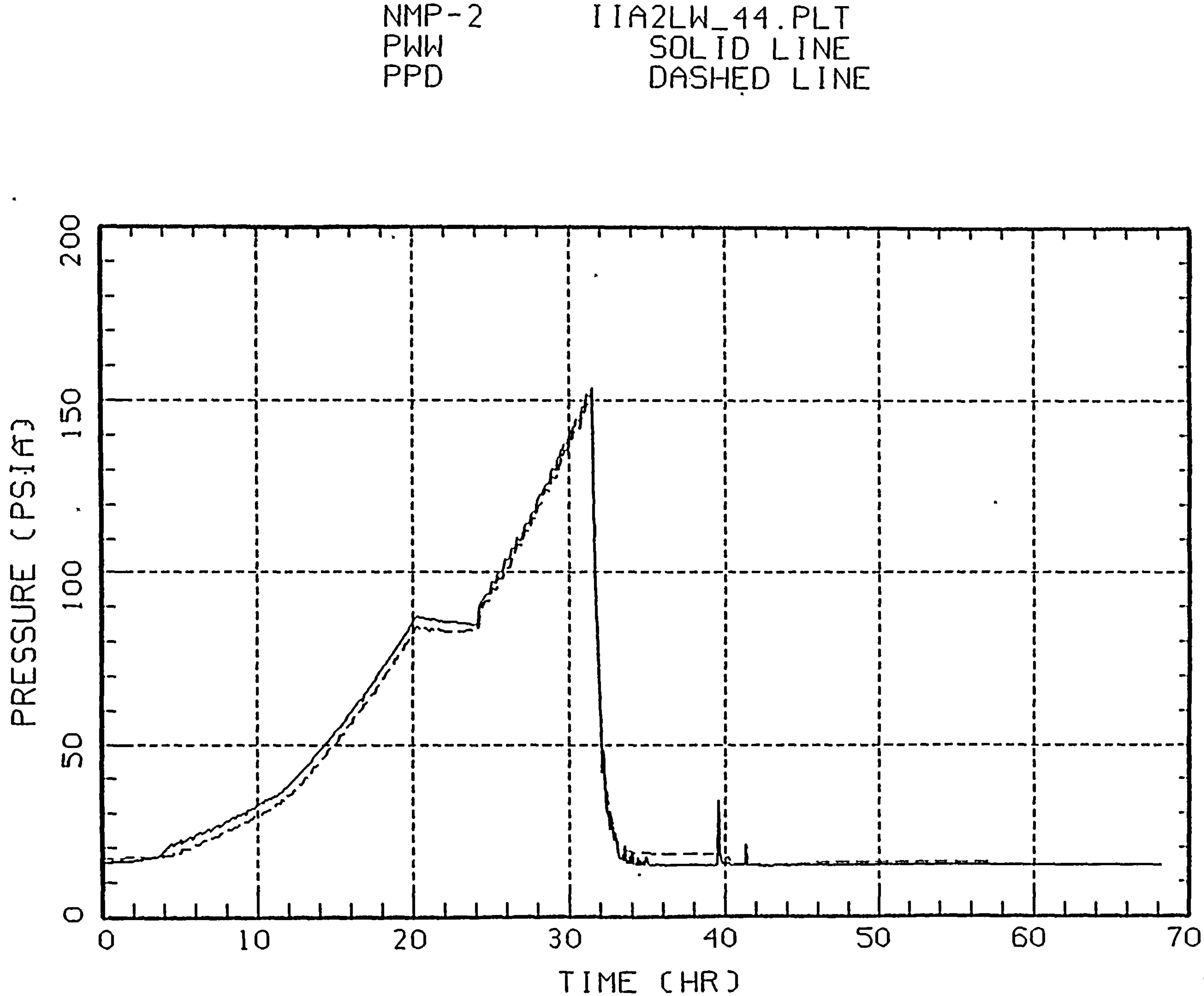


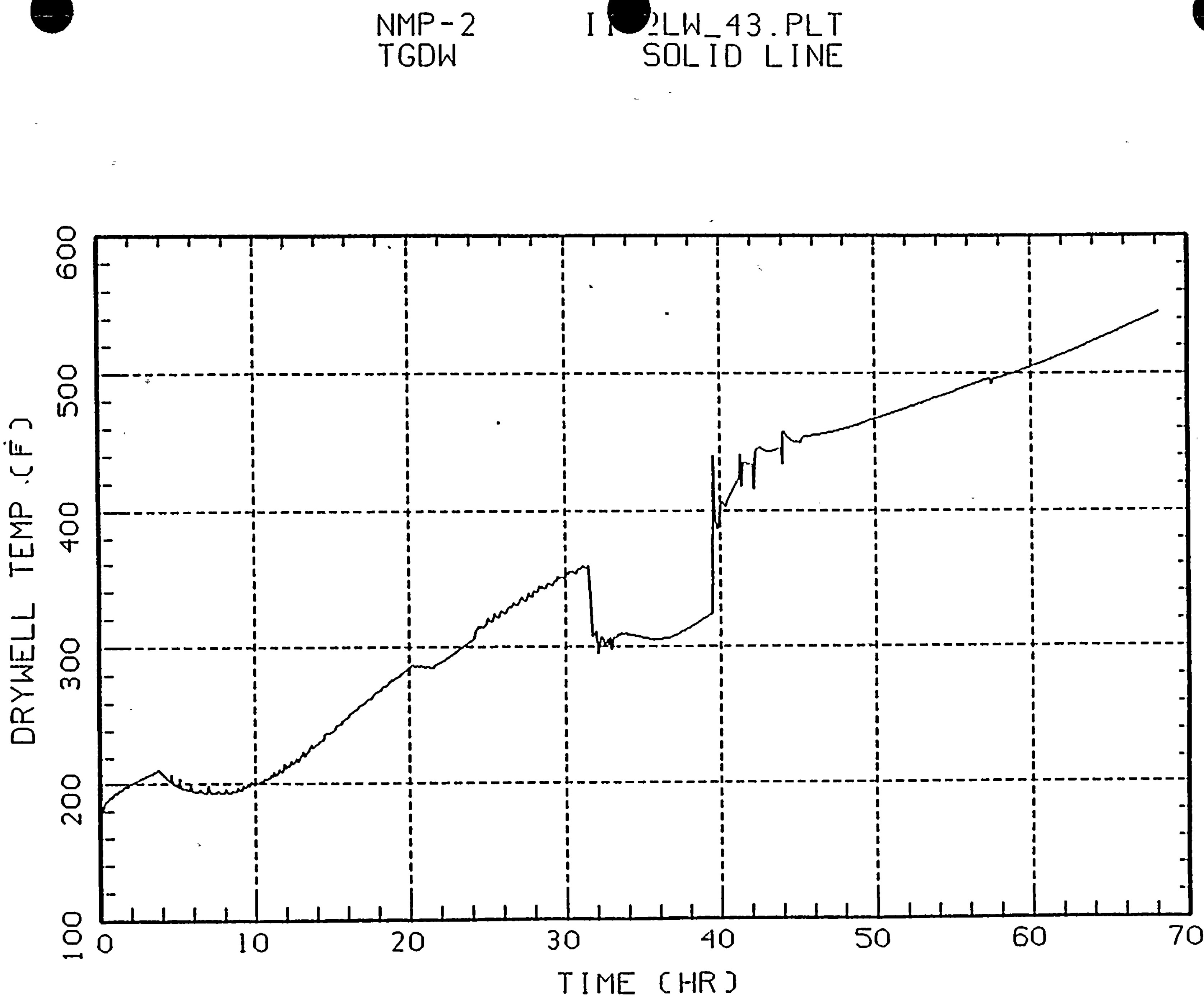
Figure 4.6-5 Drywell Pressure for Class ID Response

C135-90-02-713



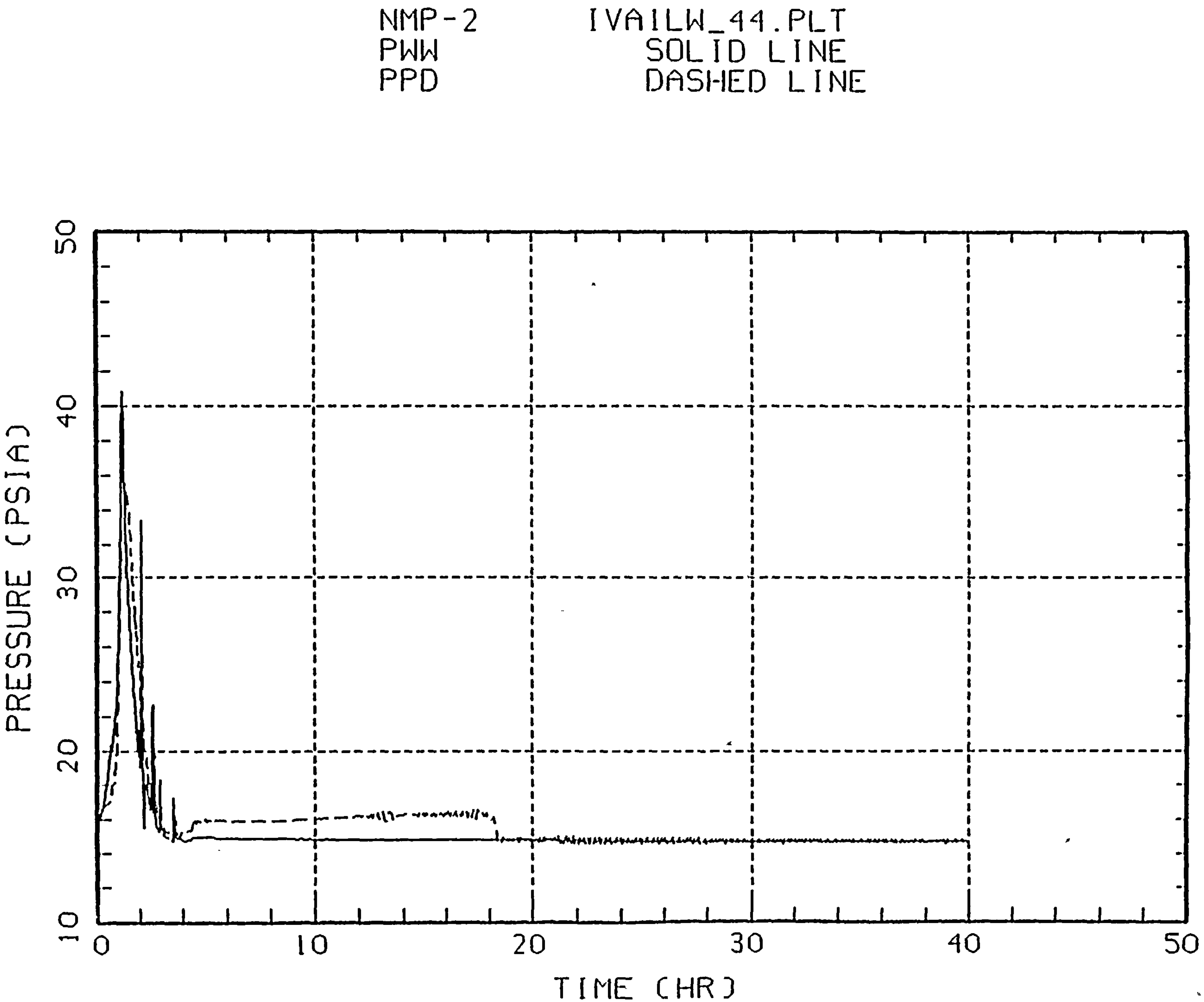
Wetwell (PWW) and Drywell (PPD) Pressures for Loss of Containment Heat Removal (IIA2LW)

Figu[illegible]6-6



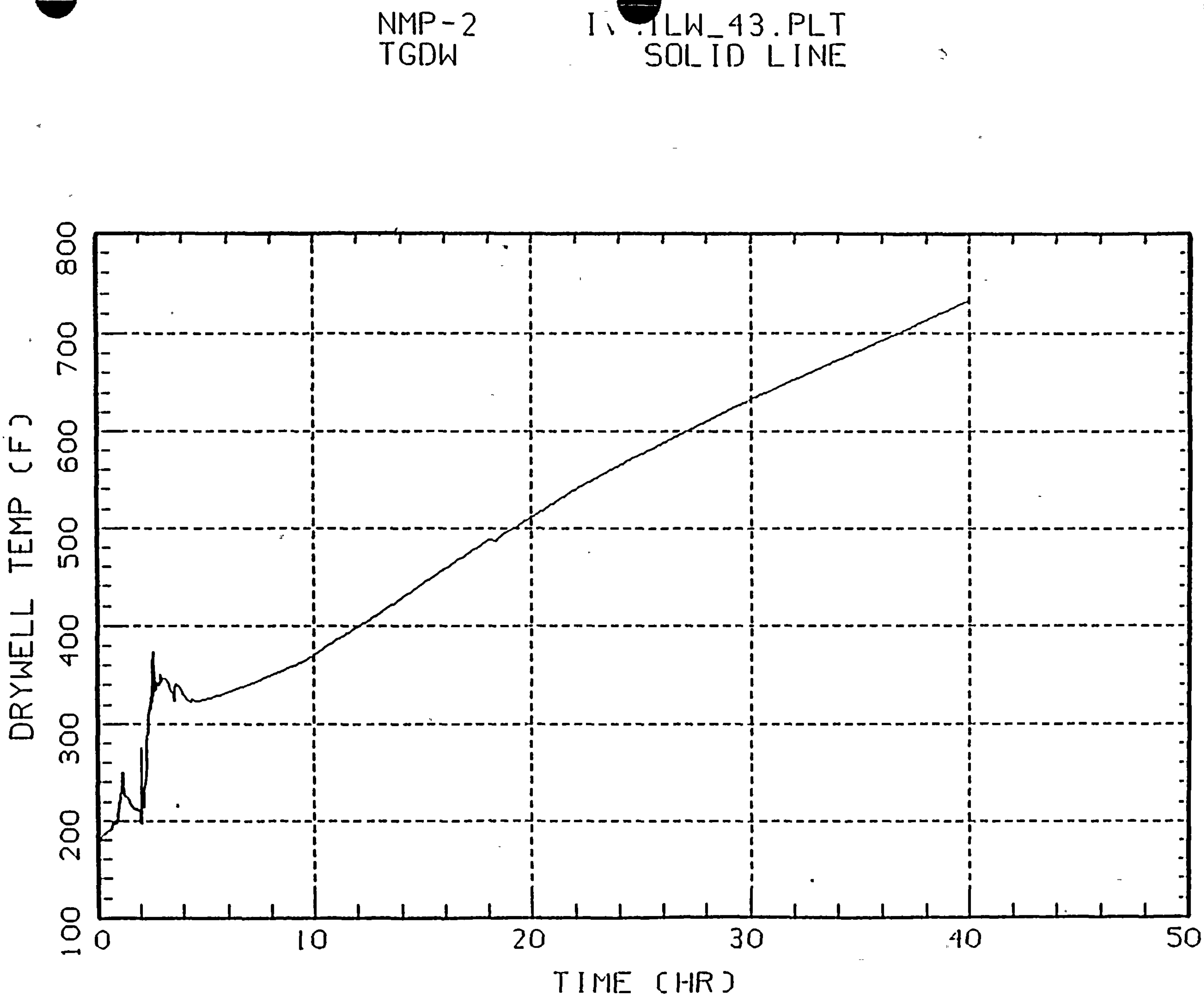
Drywell Gas Temperature for Loss of Containment Heat Removal (IIA2LW)

Figure 4.6-7



Wetwell (PWW) or Drywell (PPD) Pressure for ATWS (IVA1LW)

Fig[illegible]4.6-8



Drywell Gas Temperature For ATWS (IVA1LW)

Figure 4.6-9

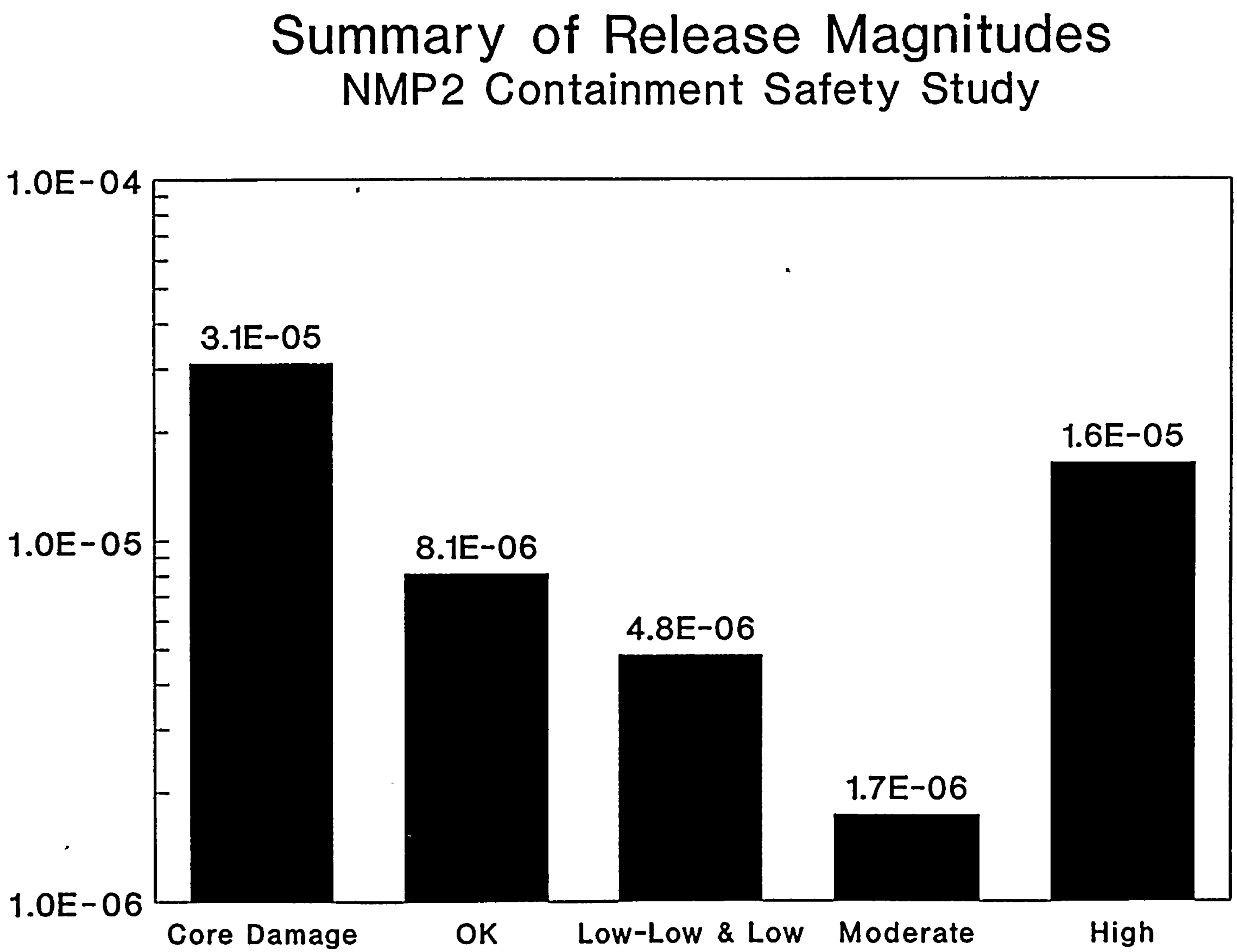


Figure 4.6-10

Rev. 0 (7/92)

Summary of Radionuclide Release Magnitudes

NMP2 Containment Safety Study

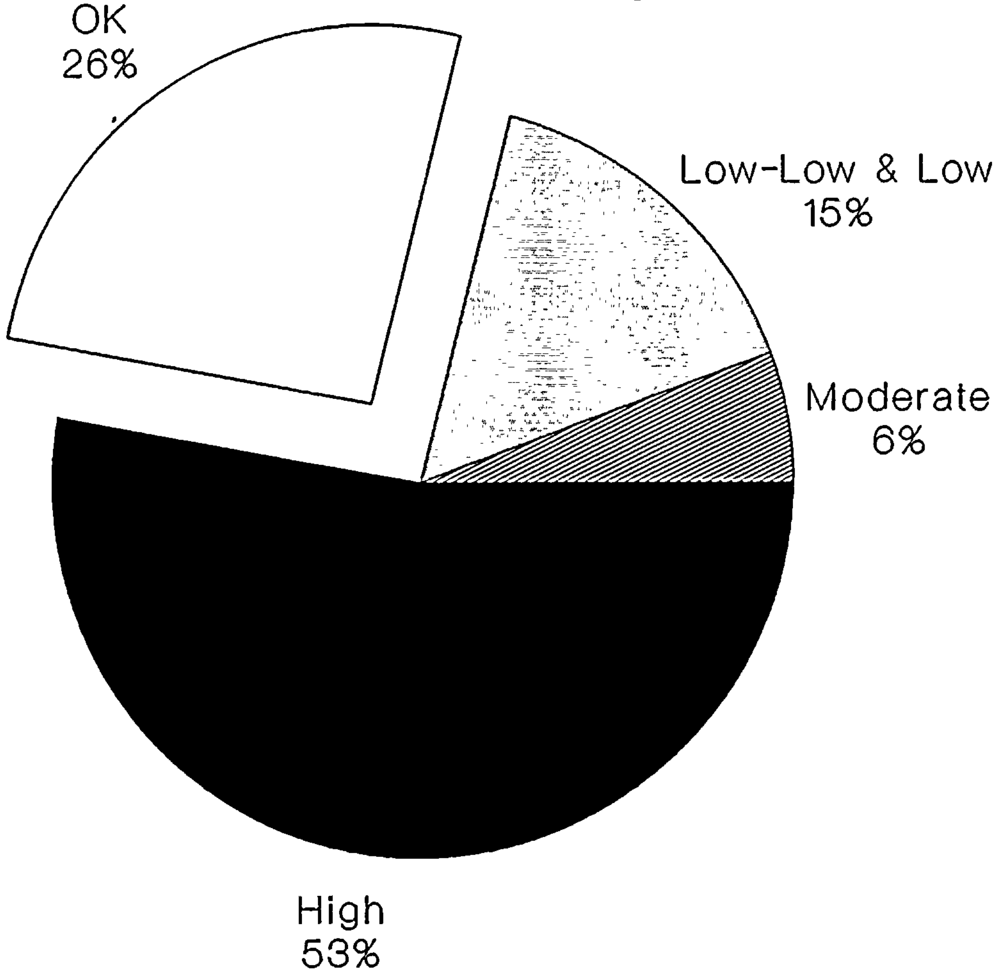


Figure 4.6-11

Rev. 0 (7/92)

Comparison of "Large" Release Category to NRC Proposed Generic Goal

NMP2 Containment Safety Study

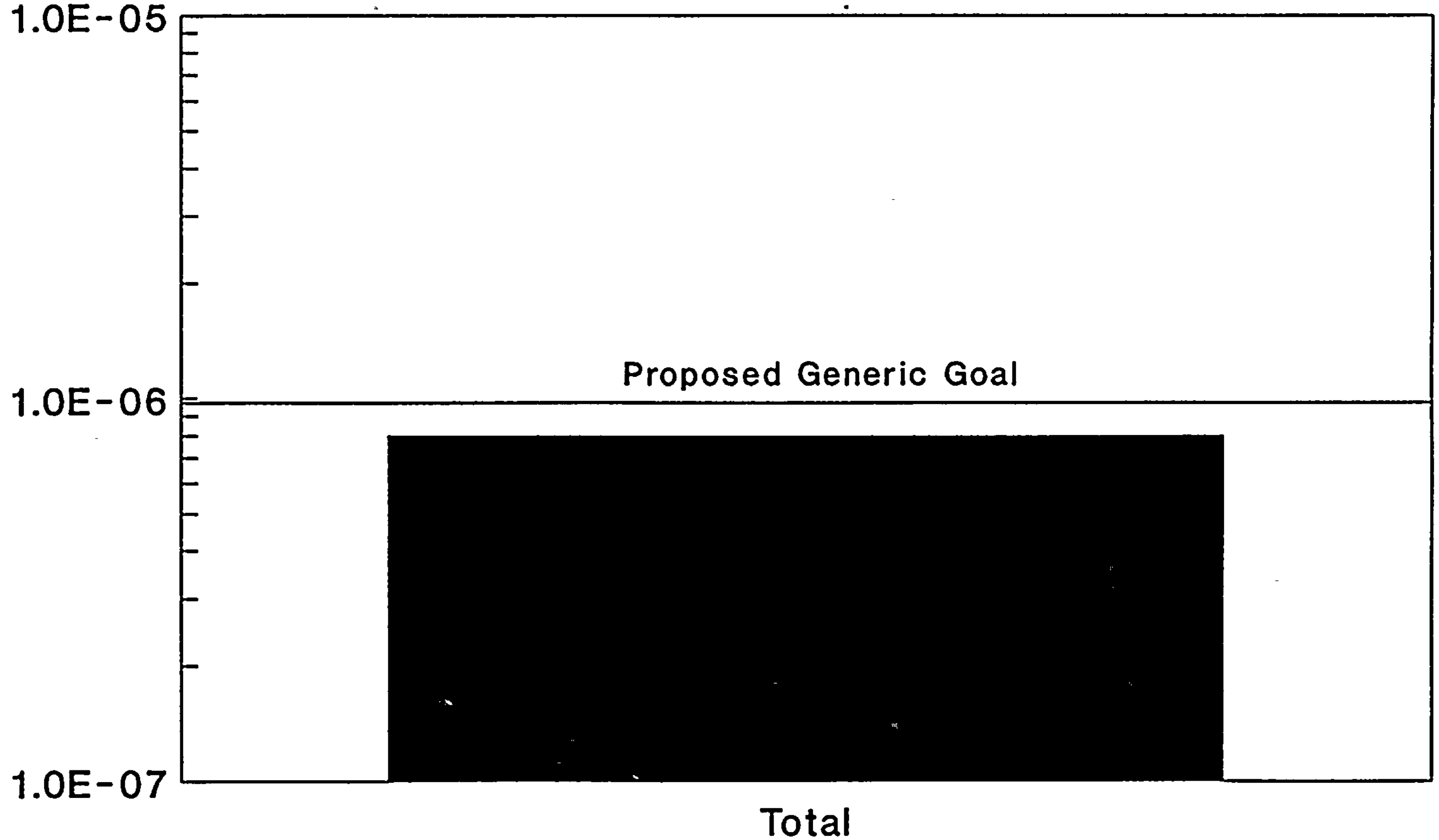


Figure 4.6-12

Rev. 0 (7/92)

Summary of Radionuclide Release Timings
NMP2 Containment Safety Study

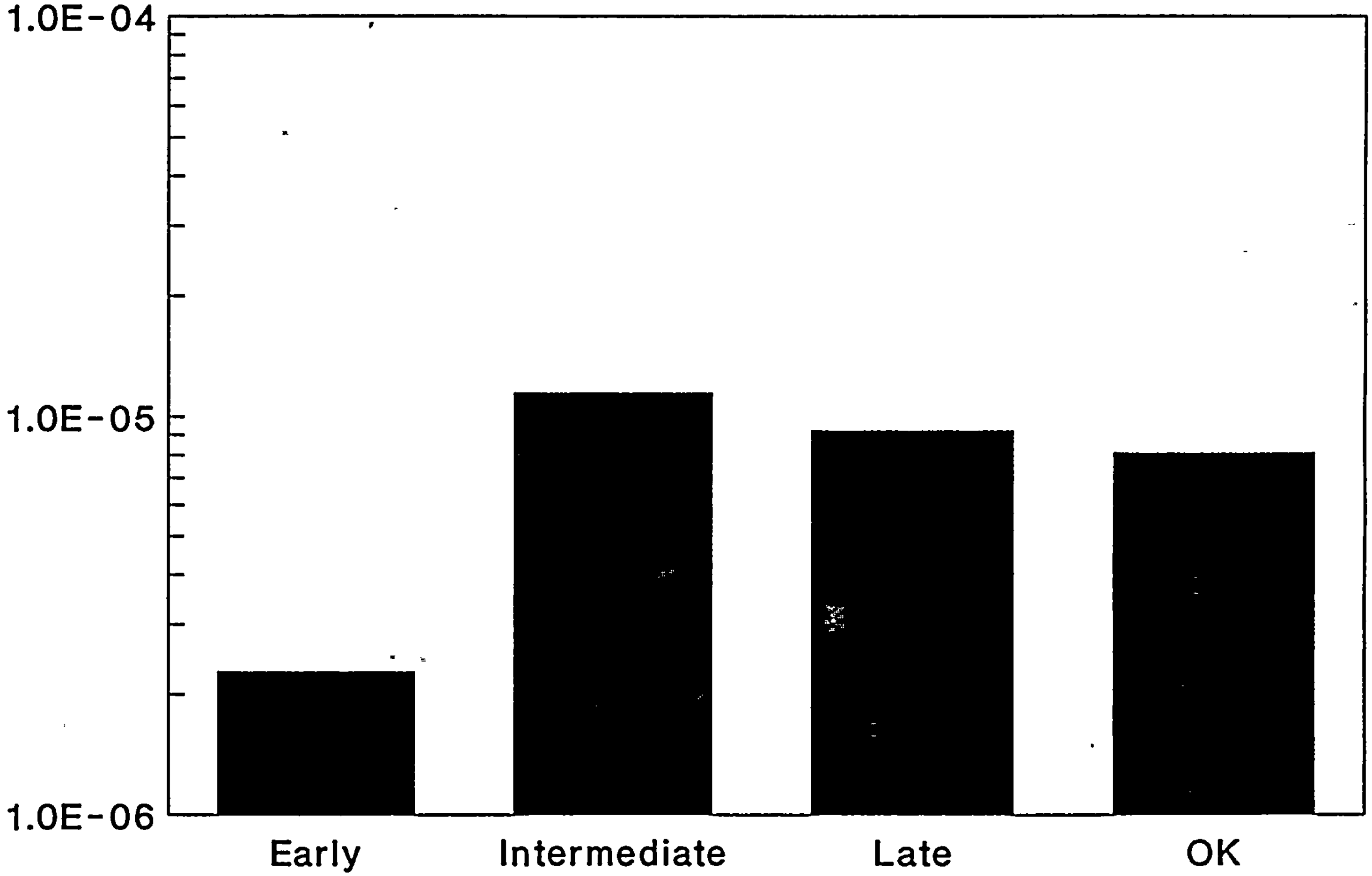


Figure 4.6-13

Rev. 0 (7/92)

Summary of Radionuclide Release Timings
NMP2 Containment Safety Study

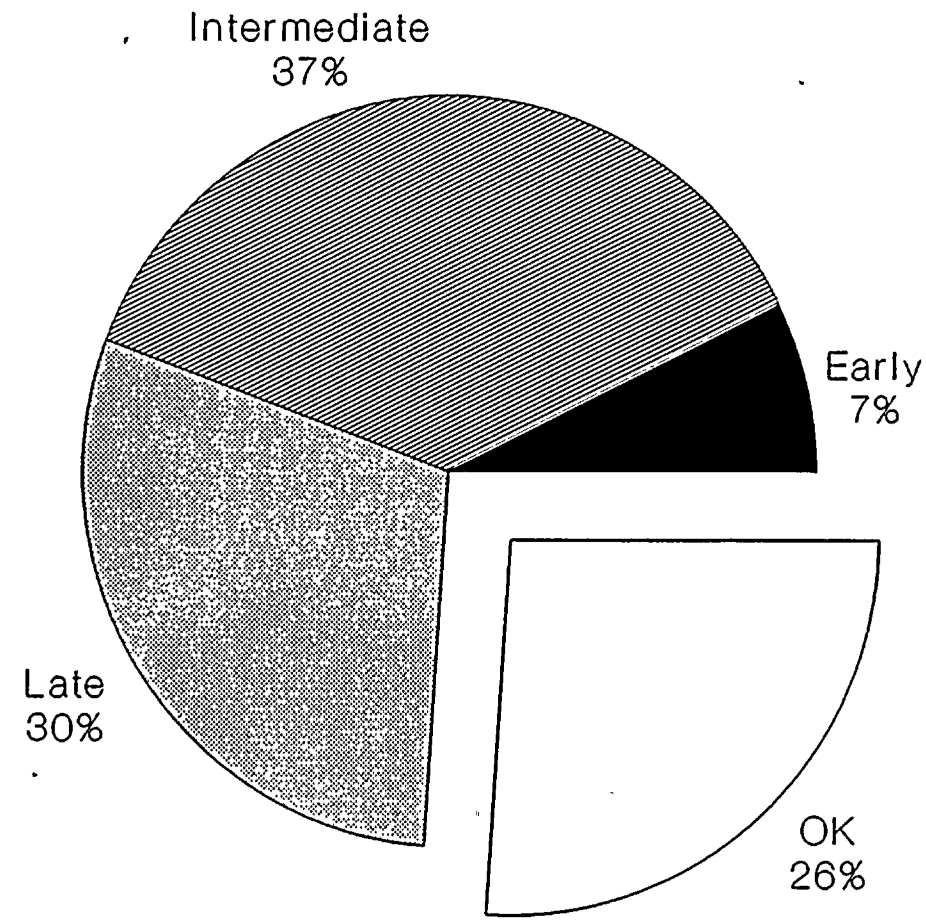


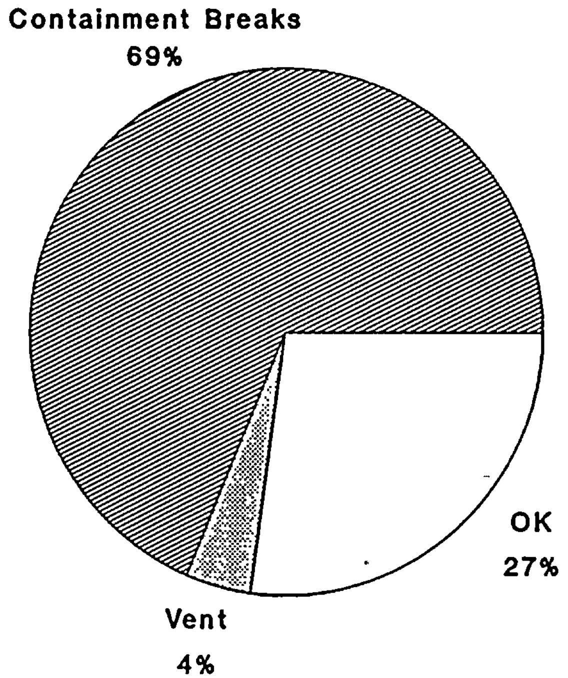
Figure 4.6-14

Rev. 0 (7/92)

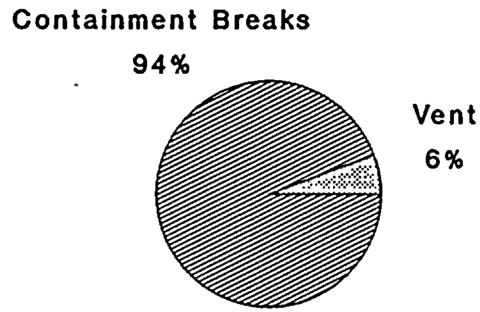
Ratio of Vent Sequences to Containment Failure Sequences

NMP2 Containment Safety Study

Total Frequency of Release = 3.0E-5 Per RX Year



All CET End States



All Releases

Figure 4.6-15

Rev. 0 (7/92)

Refer to Table 4.6-5, Note 7

Relative Contribution of Containment Failure Modes to Radionuclide Release

(All Releases : 2.2E-5/yr)

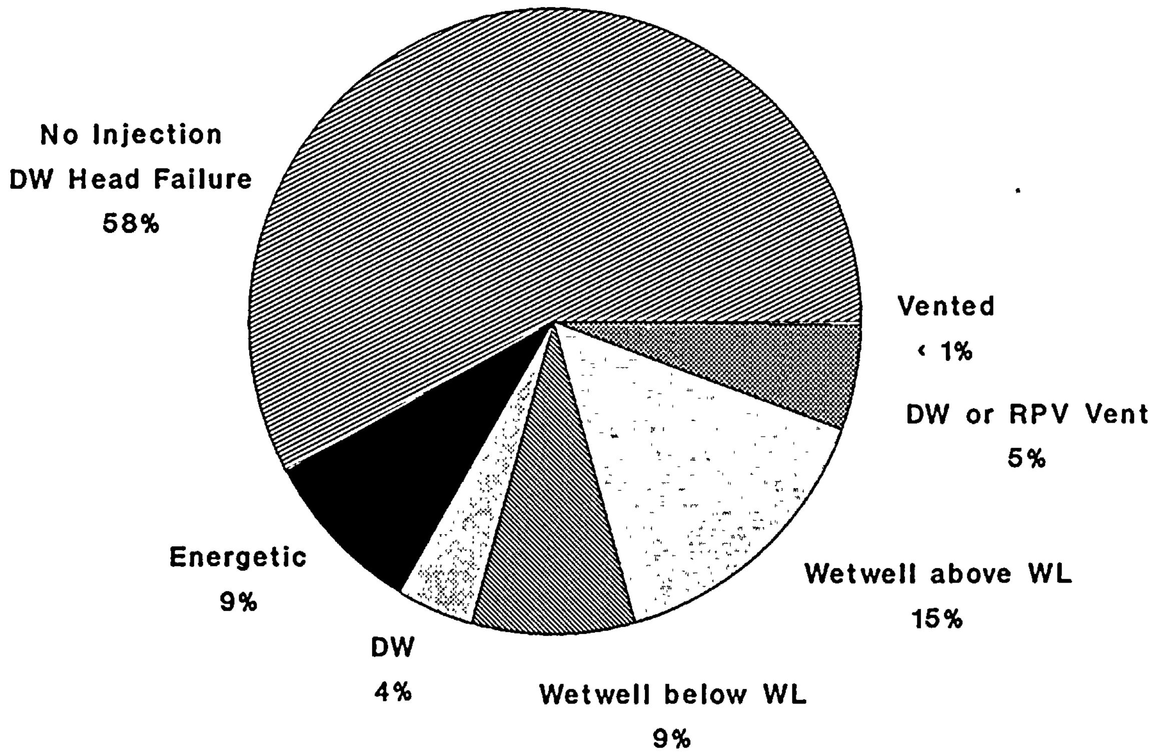


Figure 4.6-16

Refer to Table 4.6-5, Note 7

Rev. 0 (7/92)

Release Magnitude Contributions
NMP2 Containment Safety Study

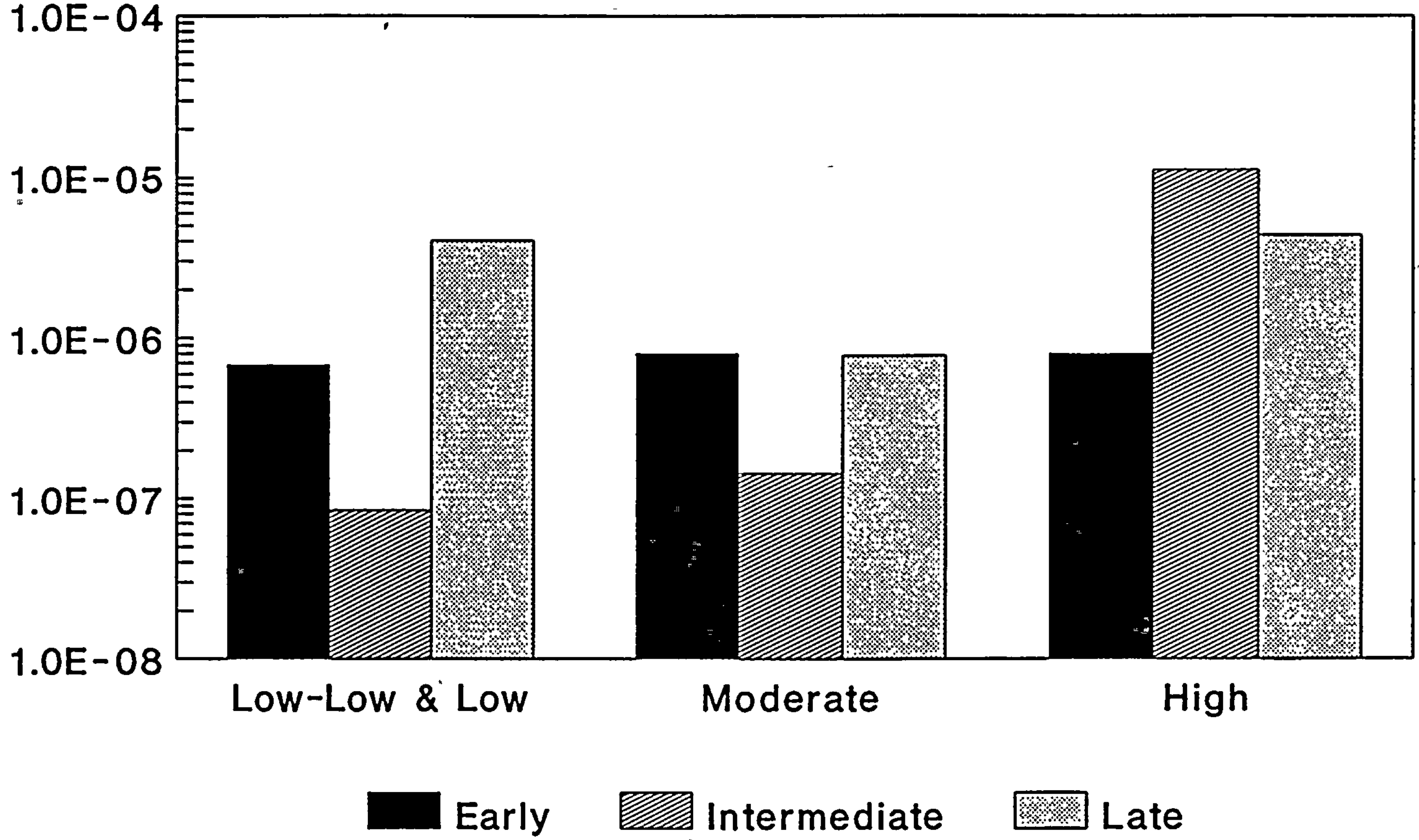


Figure 4.6-17

Refer to Table 4.6-5, Note 7

Rev. 0 (7/92)

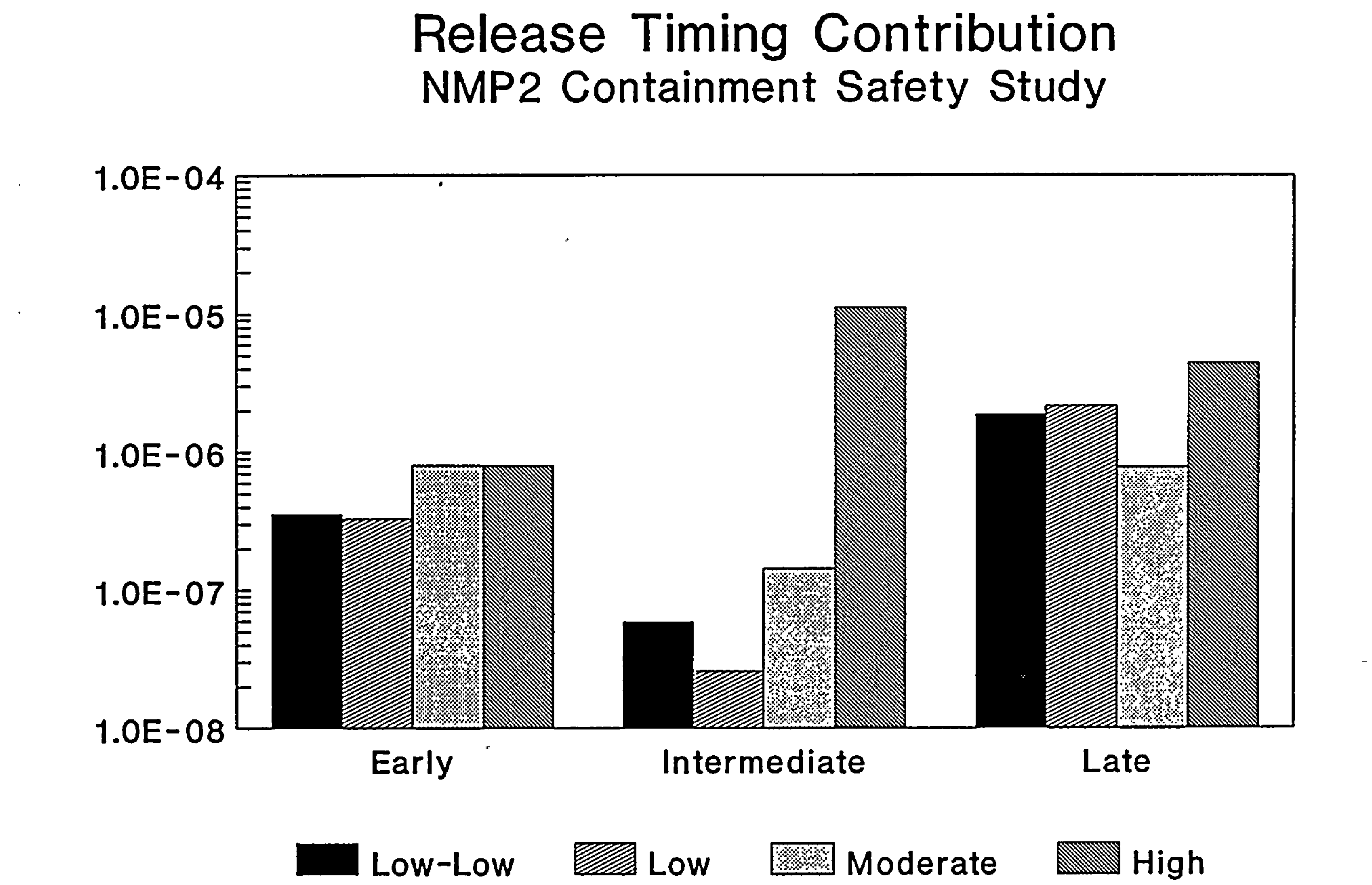


Figure 4.6-18

Refer to Table 4.6-5, Note 7

Rev. 0 (7/92)

Contributors to Large Release Frequency

NMP2 Containment Safety Study

Total Large (High/Early) 7.7E-7 Per RX Year

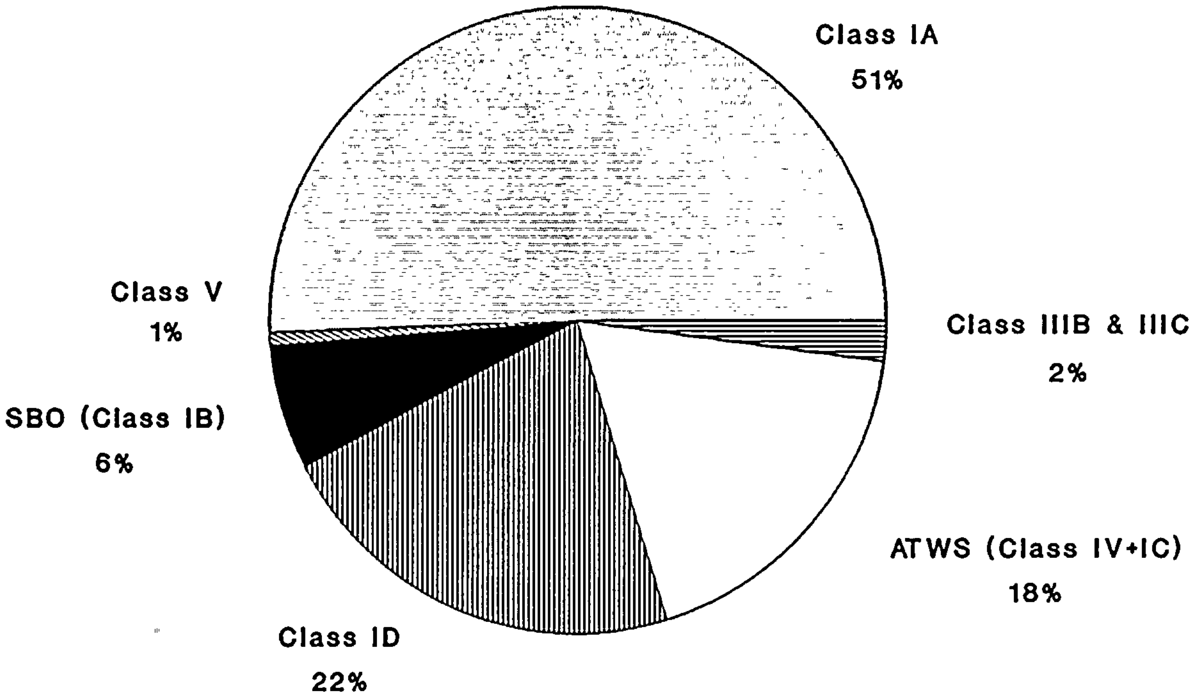


Figure 4.6-19

Refer to Table 4.6-5, Note 7

Rev. 0 (7/92)

Contributors to High Release Frequency

Nine Mile Point Unit 2

(1.6E-5/yr)

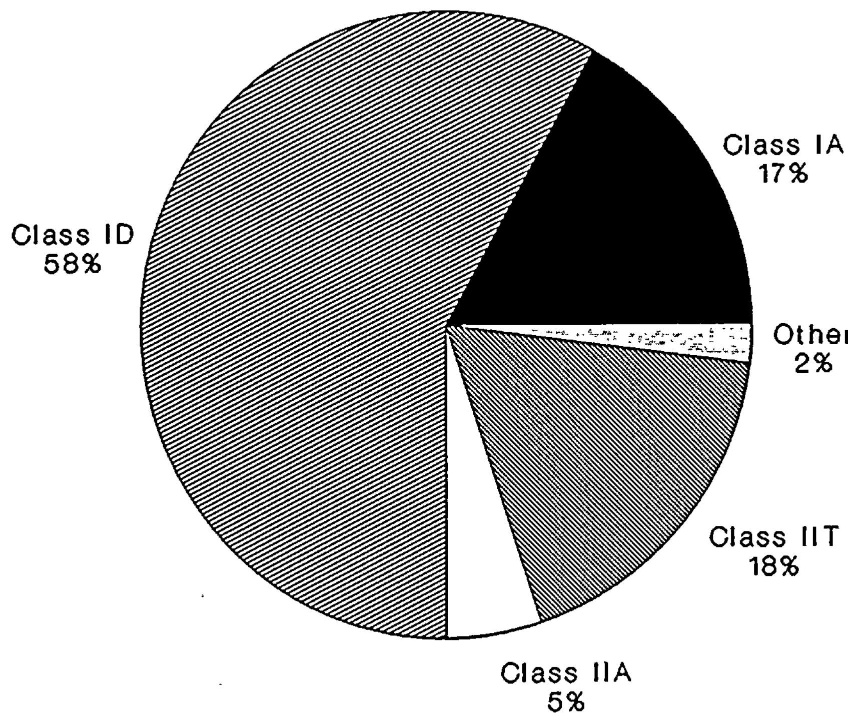
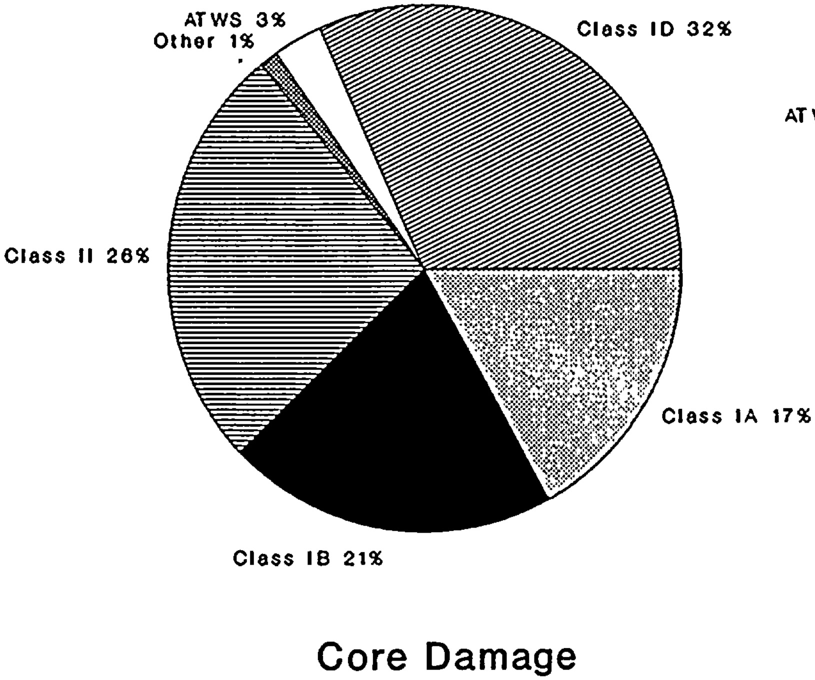


Figure 4.6-20

Refer to Table 4.6-5, Note 7

Rev. 0 (7/92)

NMP2 Containment Safety Study
Comparison of Core Damage and Large Release Contributors



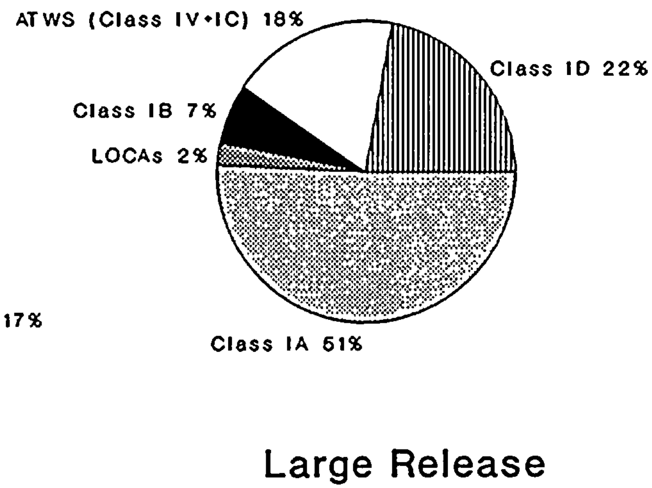


Figure 4.6-21

Refer to Table 4.6-5, Note 7

Rev. 0 (7/92)

Contribution from Class IA Categories
Nine Mile Point Unit 2

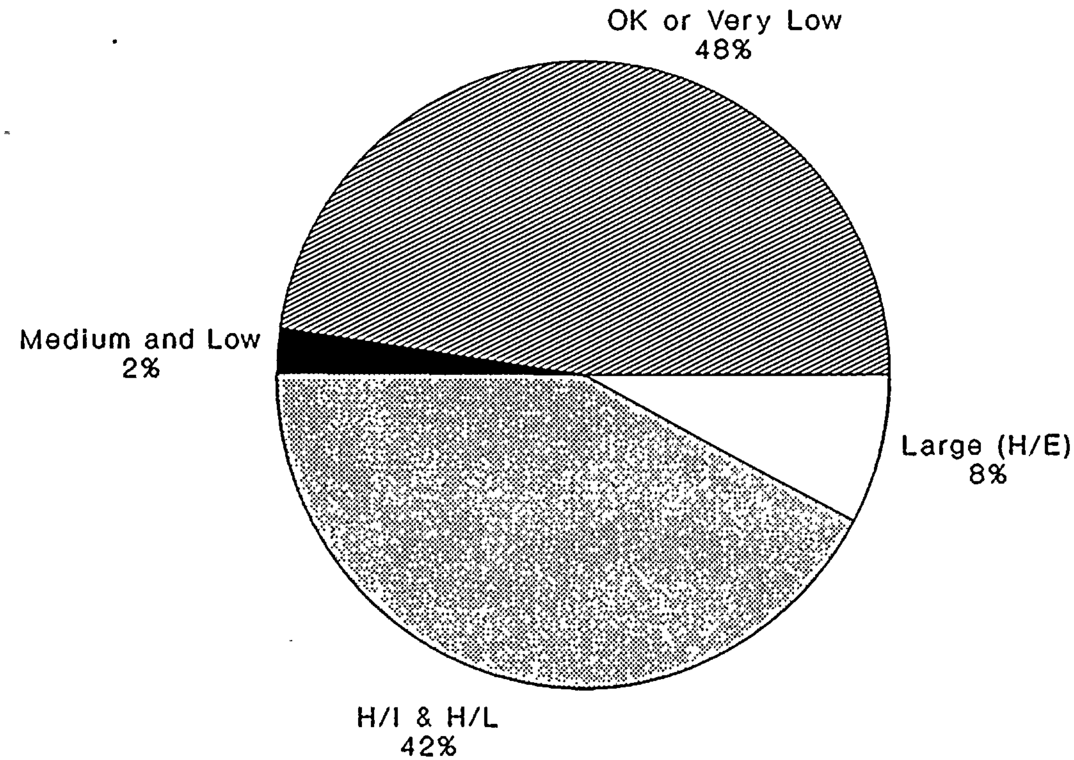


Figure 4.6-22

Refer to Table 4.6-5, Note 7

Rev. 0 (7/92)

Contribution from Class IB to Release Categories

Nine Mile Point Unit 2

(6.1E-6/yr)

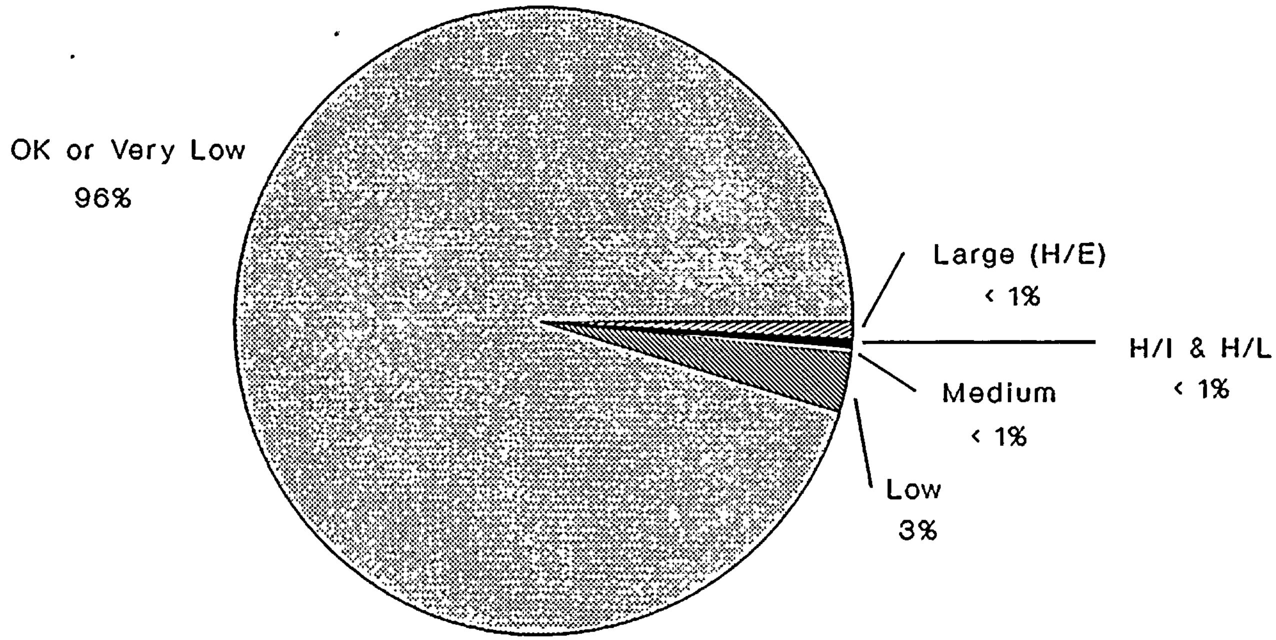


Figure 4.6-23

Refer to Table 4.6-5, Note 7

Rev. 0 (7/92)

Contribution from Class IC to Release Categories

Nine Mile Point Unit 2

(1.4E-7/yr)

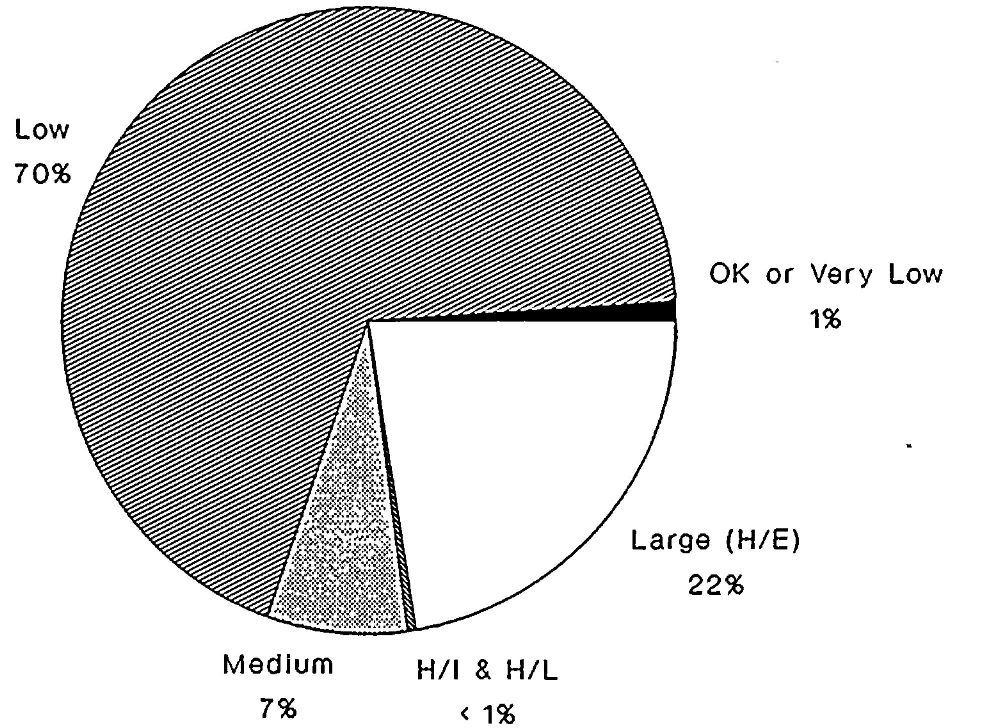


Figure 4.6-24

Refer to Table 4.6-5, Note 7

Rev. 0 (7/92)

Contribution from Class ID to Release Categories

Nine Mile Point Unit 2

(9.3E-6/yr)

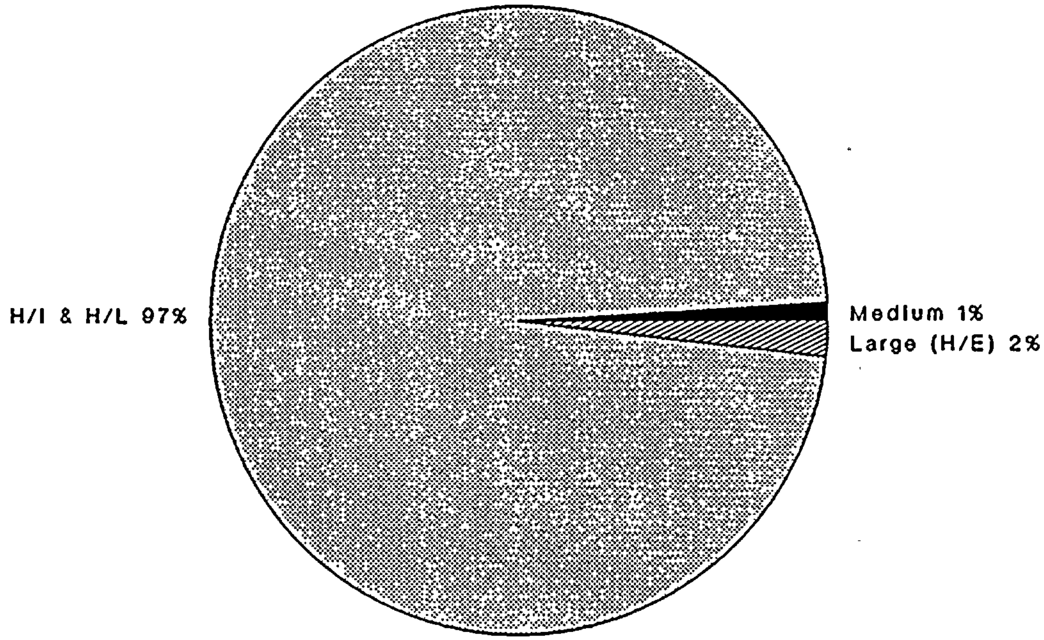


Figure 4.6-25

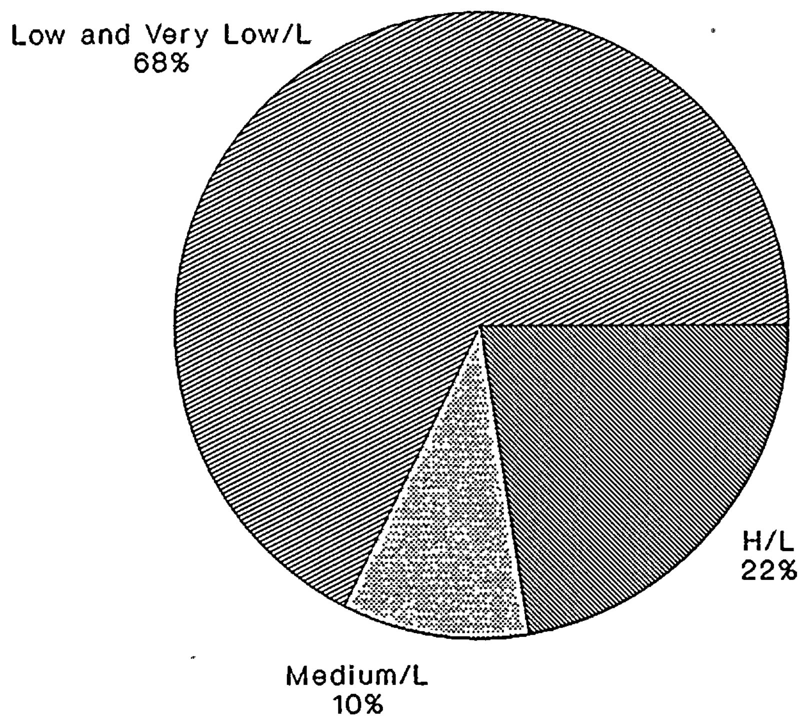
Refer to Table 4.6-5, Note 7

Rev. 0 (7/92)

Contribution from Class IIA to Release Categories

Nine Mile Point Unit 2

(3.7E-6/yr)



* These are all Late releases

Figure 4.6-26

Refer to Table 4.6-5, Note 7

Rev. 0 (7/92)

Contribution from Class IIL to Release Categories

Nine Mile Point Unit 2

(2.5E-7/yr)

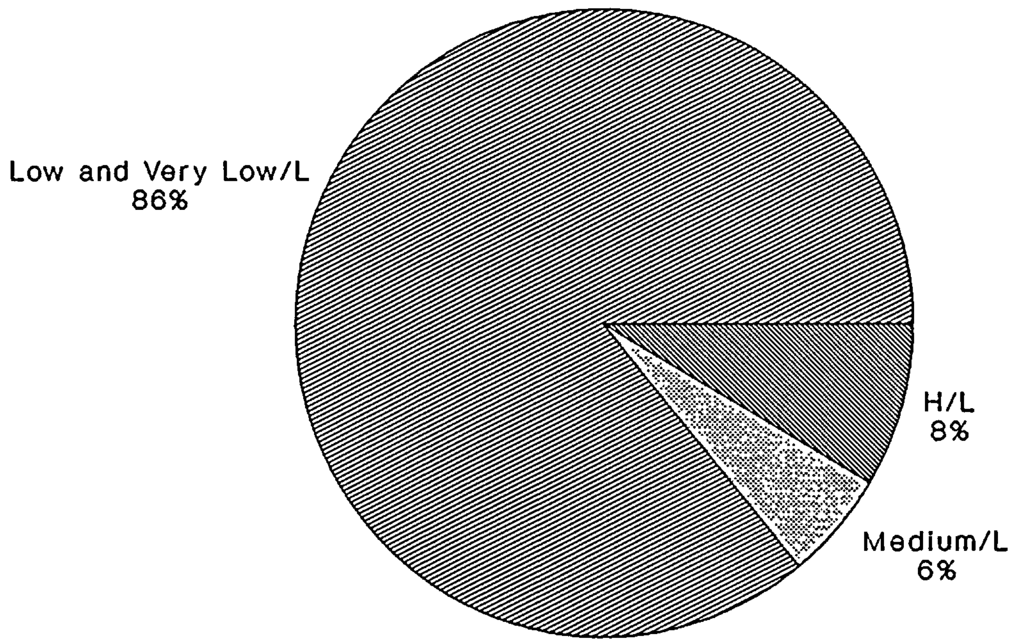


Figure 4.6-27

Refer to Table 4.6-5, Note 7

Rev. 0 (7/92)

Contribution from Class IIT to Release Categories

Nine Mile Point Unit 2

(3.8E-6/yr)

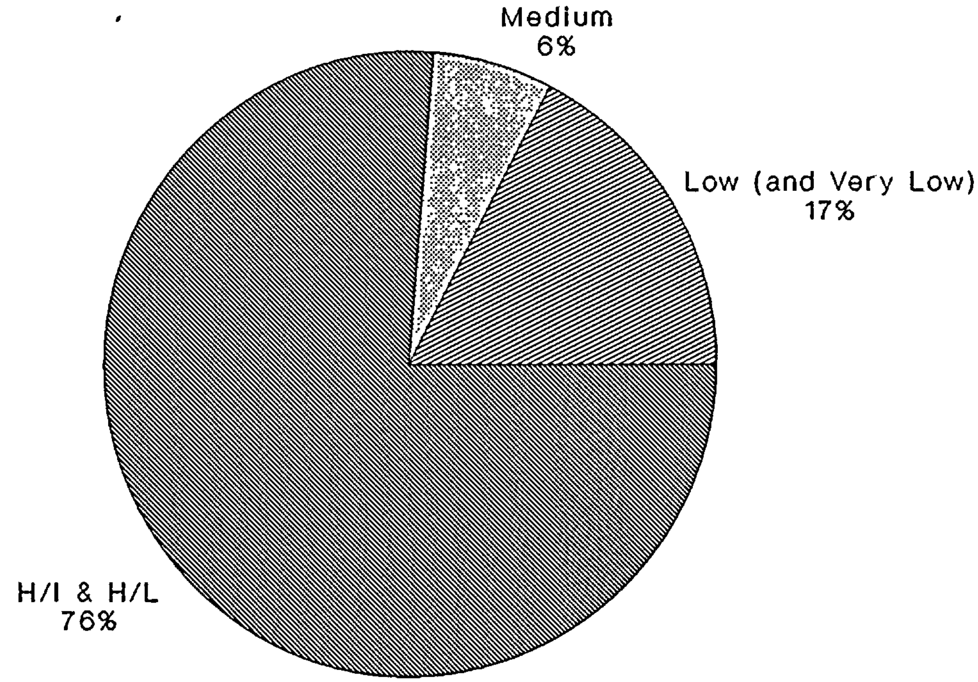


Figure 4.6-28

Refer to Table 4.6-5, Note 7

Rev. 0 (7/92)

Contribution from Class IIIB to Release Categories

Nine Mile Point Unit 2

(4.6E-7/yr)

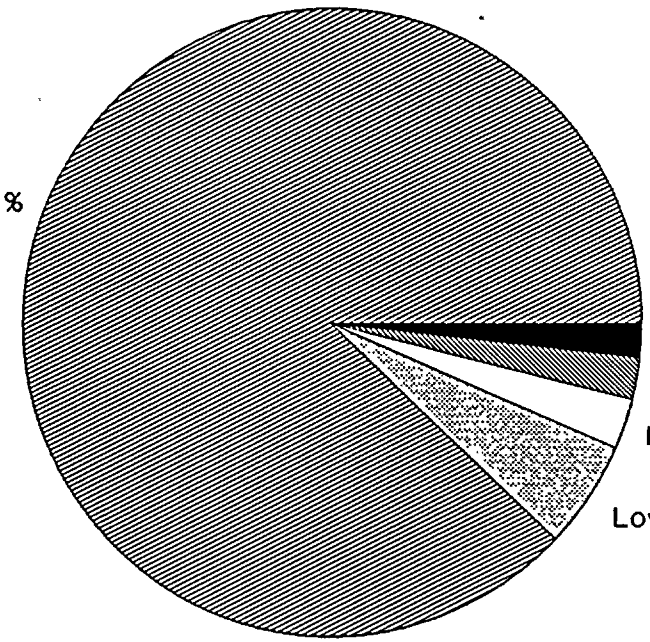


Figure 4.6-29

Refer to Table 4.6-5, Note 7

Rev. 0 (7/92)

Contribution from Class IIIC to Release Categories

Nine Mile Point Unit 2

(1.2E-8/yr)

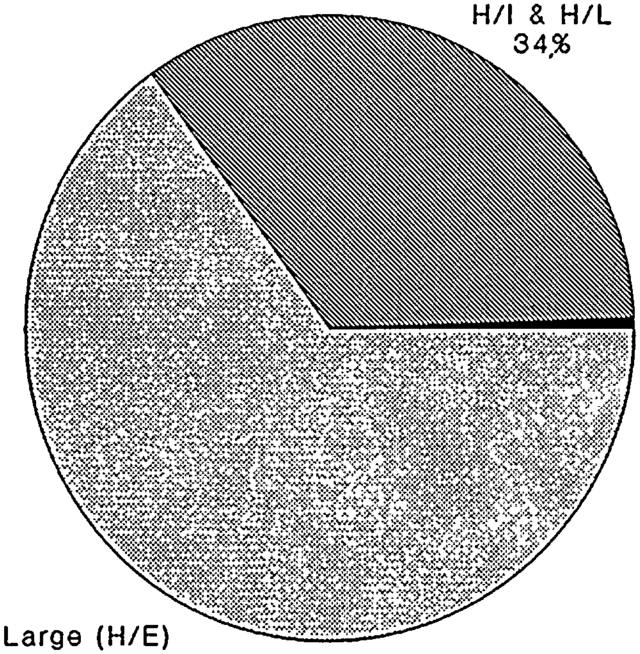


Figure 4.6-30

Refer to Table 4.6-5, Note 7

Rev. 0 (7/92)

Contribution from Class IV to Release Categories

Nine Mile Point Unit 2

(9.2E-7/yr)

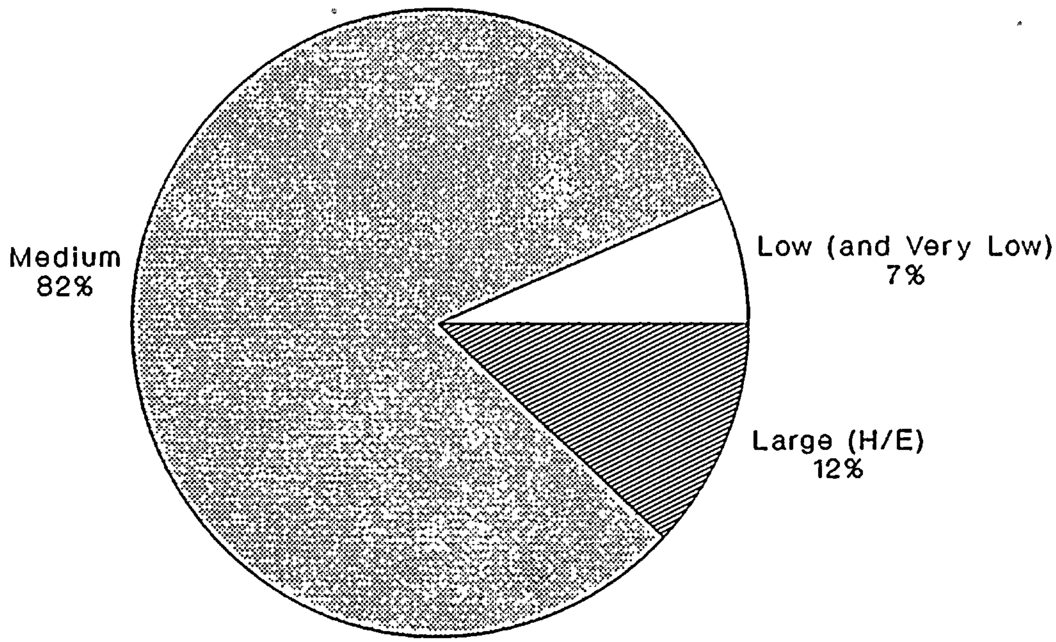


Figure 4.6-31

Refer to Table 4.6-5, Note 7

Rev. 0 (7/92)

Contribution from Class V to Release Categories

Nine Mile Point Unit 2

(2.7E-8/yr)

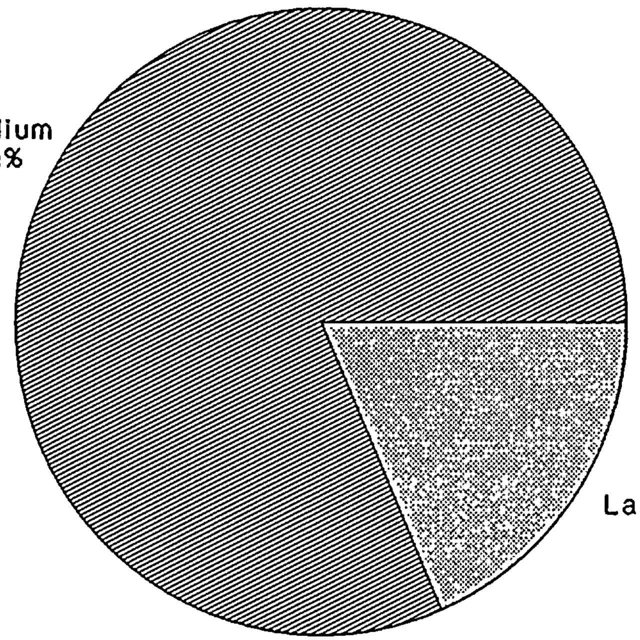


Figure 4.6-32

Refer to Table 4.6-5, Note 7

Rev. 0 (7/92)

Comparison of High/Early (Large) Release

Total High/Early (Large) Release = 7.70E-7/Rx Year

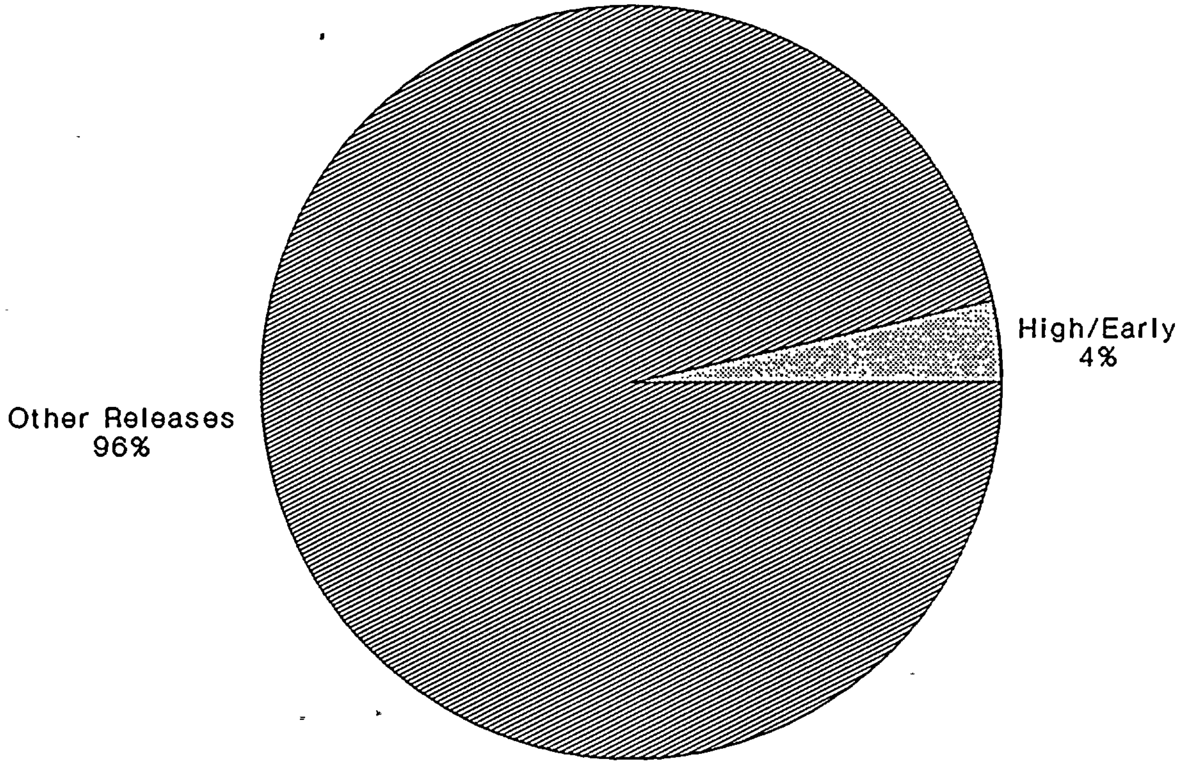


Figure 4.6-33

Refer to Table 4.6-5, Note 7

Rev. 0 (7/92)

4.7 RADIONUCLIDE RELEASE CHARACTERIZATION

The radionuclide release sequences determined from the CET evaluation that exceed the screening criteria frequency (i.e., reporting criteria) have been assessed to determine their radionuclide release magnitude and timing.

The determination of the radionuclide release magnitude for the Nine Mile Point 2 IPE has taken two approaches both of which were identified as viable options in NUREG-1335.

The two approaches are:

- Use of plant specific Nine Mile Point 2 calculations to both confirm the surrogate plant calculations and to fill in missing sequence calculations.

- Use of existing Mark II radionuclide releases for a similar plant to augment the characterization of some release sequences.

This subsection includes the following important discussion items regarding radionuclide release characterization:

- Overview (Section 4.7.1)
- Governing features (Section 4.7.2)
  - Removal processes and pathways
  - Containment failure modes
  - Phenomenology
  - Timing
- Release Bins (CET End States) (Section 4.7.3)
- Criteria for Release Bins (Section 4.7.4)
  - Magnitude
  - Timing
- MAAP calculational results for the Release Bins (Section 4.7.5)
- Sensitivities are discussed in Section 4.9 because of the rather extensive provided.

4.7.1 Overview of Potential Release Characterization

Each CET end state can be associated with a radionuclide source term bin which covers a spectrum of similar potential scenarios and timing. Theoretically, it would be desirable in determining the point estimates of risk to evaluate the source terms for each sequence or each accident plant damage state. However, for purposes of risk presentation, the CET end states can also be characterized in such a manner as to combine similar "consequence impact" release event sequences within a CET end state.

This technique is considered an appropriate approach for characterizing severe accidents postulated in the IPE. For instance, the NRC staff in NUREG-1335 (p A-13) states that:

> It is common practice to reduce the large number of possible source terms to a smaller number of representative release categories.

The number of representative release categories selected is left to the discretion of the analyst; however, it is essential to select a sufficient number of representative release categories so that each of the individual source terms can be adequately represented by one of the representative release categories.

After a set of representative source terms has been established, each of the CET endpoints can be allocated to an appropriate representative source term. The frequency of any given representative source term can then be determined by summing all the CET endpoint frequencies for each of the plant damage states that are allocated to it. When this process is complete, the relative importance (magnitude of source term or frequency) of each containment challenge can be determined.

Additionally, the NUREG-1150 Peer Review of the second draft identified a number of areas where important methodology insights could be gained from past probabilistic analysis. One such area included observations regarding the treatment of plant damage states to be used in the Level 2 containment evaluation. Some of the committee observations are as follows:

It is well-established practice in reactor safety in general, and in PSA in particular, to consider families of events and plant damage states. This practice greatly reduces the likelihood of omission of accident sequences that should be included. In the front-end analysis, initiating events are usually grouped into families based upon the similarity of physical phenomena or the response needed from plant systems. Depending on the system's failure modes, different sequences of events within a single family may finally lead to different physical phenomena and consequences. Therefore, it is appropriate to provide a different grouping at the back end of the analysis.

It is helpful for that purpose to define plant damage states that include all sequences leading to a physical condition of the plant with common attendant outside consequences (source terms).

In each plant damage state, the families of events with high probability of occurrence dominate the calculated contribution to risk. Therefore, excluding low-probability sequences from the analysis will not change results significantly. Which cutoff frequency is appropriate depends on the classification of event families and on the frequency of the dominant risk contributors. Experience shows that neglecting sequences with a frequency about two orders of magnitude below the calculated mean core-damage frequency does not noticeably change the overall core damage frequency. Thus, for plants that have a mean core-damage frequency of 10^{-5}/yr, a cutoff frequency of 10^{-7}/yr seems appropriate.

The situation is different if entire plant damage states are neglected. Dropping an entire plant damage state might cause an entire class of consequences to be dropped from the analysis. But then it is not reasonable to analyze in detail plant damage states whose frequency is below that of catastrophic failures like that of the reactor pressure vessel, for which the conditional probability of severe offsite consequences could be high. Present understanding sets the upper bound for the frequency of such a failure at about 10^{-7} per plant year for a pressure vessel that has an acceptably low nil ductility temperature, including the region of the welds.

PSA is increasingly used for decision making, in particular, for identifying means for further risk reduction. The consideration of small contributions to risk is not helpful in this context, in particular if their calculation is influenced by large uncertainties. Therefore, decision making normally includes a de minimis concept providing a clear-cut distinction between a substantiated real risk which is to be limited and reduced, and insignificant risks that are not reliably assured.

4.7.2 Governing Features in Radionuclide Release Characterization

There are a number of plant features or accident progression features that can substantially increase or decrease the ability to retain fission products or mitigate their release. This subsection reviews some of the more important of these features in the following areas:

- Removal processes
- Containment failure modes
- Phenomenology
- Timing.

4.7.2.1 Removal Processes in Containment

Radionuclide release processes are initiated when the core overheats and melts. These release processes involve transport from the fuel, from the RPV, and from primary and secondary containment. These release processes when categorized into end states can indicate the amounts and types of radionuclide material that could potentially be released to the environment. It should be noted that, depending on the kind of accident in progress, there are inherent removal mechanisms that can occur to remove and retain these fission products. These deposition mechanisms include plateout and retention on the vessel surfaces (at least as long as RPV temperatures remain relatively low).

Once the fission products are airborne in the containment, there are removal mechanisms that reduce the magnitude of the source terms that are available for leakage to the environment. These removal mechanisms include plateout and settling in containment. The degree of attenuation is determined to a large extent by the time available for these processes to occur. The time between fission product release from the fuel to containment failure determines the residence time of the radionuclides within containment. The containment failure modes and failure location also contribute to determining the radionuclide removal mechanisms that are operating along the exit path to the environment.

Given that radionuclides are released from the fuel, the removal of fission products from any leakage pathway varies with the kind of accident sequence in progress, the containment failure mode, and the type of fission product being transported. These removal mechanisms may be categorized in terms of the following:

- Natural removal - Radionuclides may be removed by natural deposition (plateout) or settling mechanisms.

- Active Safety System - The following systems can potentially "wash-out" or filter particulate radionuclides.
  - Containment sprays
  - RPV and containment injection
  - SGTS.

- Passive Safety System - The suppression pool provides a removal mechanism for radioactivity during a core melt progression accident. The effectiveness of pool decontamination depends on the characteristic of the aerosol source (e.g., particle size distribution), the temperature of the water, and whether pool bypass pathways exist. In addition, the reactor building provides another possible deposition site for passive retention of radionuclides.

These removal mechanisms are each included in the Nine Mile Point 2 specific deterministic modeling usint the MAAP code.

4.7.2.2 Containment Failure Modes

For each of the accident sequence classes, there is a set of containment failure modes and release pathways that affect the magnitude of the radionuclide releases. Briefly, the principal methods in which the containment failure modes affect the radionuclide release are:

- Size of the Containment Breach - The size of the postulated containment failure can contribute to determining the usefulness of the reactor building with regards to the capability of the structure and systems to affect the release source term.

- Location of the Breach - The location of the postulated containment failure affects the degree of radionuclide release decontamination along the path. The more tortuous the pathway for release, the greater the likelihood that deposition reduces the radionuclide release. One of the most important aspects of the failure location is its relation to the suppression pool. For some sequences that include drywell failure, the radionuclide release after containment failure could bypass the suppression pool, thus eliminating this valuable fission product removal mechanism.

4.7.2.3 Phenomenology

The CET includes an assessment of the probability of occurrence of energetic phenomenological effects that can result in containment failure and add energy to the radionuclide release. Examples of such phenomena include the following:

- Steam explosions
- Hydrogen detonation

- Direct containment heating
- Excessive blowdown pressure.

Such phenomena, while of low probability even given a severe accident, may have a substantial influence on the containment integrity, radionuclide removal processes, and the radionuclide release source term. Therefore, the end states of the Level 2 PRA are also influenced directly by the occurrence of these phenomena.

4.7.2.4 Timing

Two aspects of radionuclide release timin are of imporance in characterizing both accident management actions and public protective response. These tow aspects of timing are:

- The time release set by the time when containment is breached or bypassed <u>and</u> core damage has occurred

- The duration of the radionuclide release.

The time of release is straightforward and is set by the time when both core damage and containment bypass have occurred.

The length of time over which the accident progresses can influence the degree of retention and the pathway through which the release propagates.

The assessment of radionuclide release duration for the purposes of calculating release magnitudes and the assignment of accident sequences to release categories includes two considerations:

1) The compensatory measures that can be taken to significantly reduce or prevent dose to the public, and

2) The characteristics of radionuclide release.

It is incumbent upon the PRA analyst to determine the end point of deterministic calculations that describe the impact of an accident scenario, with respect to potential off-site consequences after all measures prescribed in the EOPs are postulated to be ineffective in mitigating the accident.

These two principal considerations are discussed below along with the conclusion regarding the selection of an appropriate release duration used to determine the magnitude of the source term assigned to the severe accident end state.

4.7.2.4.1 Compensatory Measures

The consideration of MAAP calculational results for determining and assigning a radionuclide release category to an accident sequence is based in part on off-site accident response which is not examined directly in a Level 2 PRA. Some of these response actions are prescribed in the facility's emergency response plan, however, these actions, which are routinely practiced, are geared to mobilizing utility resources to implement emergency procedures, assessing the potential off-site consequence of an accident, and recommending to government officials appropriate action for protecting the public. Usually, an emergency plan does not include direction for the emergency response organization to effect supplemental actions (i.e., in addition to the EOPs) to prosecute the accident and return the reactor plant to a stable, albeit damaged condition. Niagara Mohawk is currently actively pursuing programs to develop severe accident management strategies for implementing such actions to determine if there might be any in addition to the EOPs.

The scenario end point might represent the time at which the accident poses minimal additional off-site dose to the population. This time frame is difficult to precisely assess because the analyst must extrapolate beyond existing emergency procedures to forecast the utility and government's ability to implement effective mitigation measures to either terminate radionuclide release to the environment or to remove the affected population.

There are primarily two ways to minimize the accident's impact on the population and regain control of any offsite releases:

1) Mitigate the radionuclide release from the facility; and
2) Evacuate the affected public.

The mitigation of releases through the achievement of safe stable states usually requires the restoration or repair of equipment. The evaluation of safe stable states in a PRA has also generally involved the assessment of equipment operation and operator actions over an extended period of time. This time frame is nominally taken to be sufficient to marshall additional resources and mitigate the accident progression.

Therefore, the use of equipment and the associated considerations that dominate the choice of mission time are as follows:

- Beyond the time frame of 24 hours, "ad hoc" procedures can be developed to utilize additional hardware and personnel resources, and implement system recovery and alignments that are not presently considered part of plant practices or training for such extreme and unlikely situations.

- The emergency planning organization and procedures could potentially accelerate the time to implement such heroic actions if a catastrophic event were to occur at a nuclear facility in the United States.

- During the course of the accident, the TSC and EOF would become operational, and additional expertise could be available to successfully prosecute the accident.

- It is considered highly likely that off-site resources (e.g., equipment, power, vehicles) would become available.

The considerations associated with evacuation are more difficult to enumerate. From a risk perspective, actual data from natural and man-caused disasters indicate that public evacuations can be effectively carried out well within time frames of less than 36 hours.

For instance, even during the disaster at Chernobyl, effective protective measures (i.e., evacuating the population and undertaking heroic actions to mitigate the radionuclide release) were being implemented within 36 hours after the explosion disrupted the reactor. Figure 4.7-1 shows the approximate timeline of events to regain control of the facility and minimize the off-site effects from the radionuclide release.

It is expected that if a similar catastrophic event were to develop at a U.S. facility, that the governing Emergency Plan would be implemented soon after the declaration of an emergency. The Chernobyl example can only be used to illustrate the "worst-case" bound on timeliness of emergency response as a result of unplanned, but heroic actions by numerous individuals and agencies. The incident at TMI-2 and subsequent enhancements in the US Emergency Planning indicates that the emergency command structure and on-site and off-site resources would be available to the government to protect the public and the utility to mitigate the accident within the time frame of 36 hours.

4.7.2.4.2 Important Characteristics of Radionuclide Release Timing

The radionuclide release characterization has been postulated to have two primary characteristics:

- An early release component that occurs at or near RPV and/or containment breach

- A very long duration release component that is characteristic of either:

  - revaporization of radionuclides for "hot" internal deposition surfaces; or,

  - re-evolution of material deposited in water pools.

Using the MAAP code, it has been found that the revaporization term can occur over time frames of many days. However, it is also found that the predominant release term generally can be found to occur within a time frame of 36 hours past RPV breach for "dry" cases. (Cases with no water injection.) These observations from MAAP are based primarily on BWR MAAP assessments where temperatures of internal surfaces may be substantially higher than in PWRs. Therefore, for PWRs, the 36 hour duration may be too long.

Examples of this characteristic time frame are shown in the CsI release for a number of different accident types shown in Figure 4.7-2.

- Class IA: Loss of Makeup at High Pressre
- Class IB: Station Blackout loss of Makeup (SBO)
- Class IIIC: LOCA with Inadequate Makeup
- Class IVA: Failure to Scram with Containment Failure Induced

While there is still a residual revaporization source term emanating from the containment at 36 hours, the continual release trend is such that it is not expected that there will be a shift to a higher release bin by using a 36 hour calculational cut-off.

The re-evolution of fission products form the suppression pool is not accounted for with the MAAP code. It is judged that this effect is small for the following reasons:

- Fission products previously deposited in the suppression pool are not expected to be released as the pool flashes or boils. Salt water boiling experiments typically result in a large salt deposit on the heated surface after the water has been boiled away. This same type of phenomena would be expeced to occur in the suppression pool as water flashed during containment depressurization.

- Pool decontamination experiments performed by EPRI for saturated pools have resuled in a net decrease in the throughour aerosol mass indicating that the saturated pool is not re-releasing previously deposited fission products as the pool continues to boil.

- As the pool flashes, water droplets containing deposited fission products could be entrained into the gas stream and swept out of containment. One analogy that has been made relates to the presence of salt on an automobile that has been parded next to the ocean. It is true that gase moving across a large body of water will entrain small droplets of water, however, the important consideration here is that this behavior would not be significant in the case of the limited pool area of a Mark II wetwell.

- I_2 and organic iodine may evolve from water pools with high rediation <u>and</u> low p.h. GE calculations indicate the suppression pool remains basic over an extended period of time minimizing the release of I_2 or organic iodine.

Therefore, no additional radionuclide release term from suppression pool re-evolution is included in the assessment. This is consistent with experiments by ORNL in which the pH of the pool remains basic. An accident management insight from the suppression pool re-evolution issue is that a pH control substance may be needed to be injected into the suppression pool over the long term to inhibit the formation of I_2 or organic iodine.

4.7.2.4.3 Conclusion Regarding Release Duration

Based on this information, the scenario end point in the Level 2 PRA is defined as 36 hours after RPV breach.

4.7.3 Radionuclide Release Categories (CET End States)

The spectrum of possible radionuclide release scenarios is represented by a discrete set of categories or bins. The end states of the containment and phenomenological event sequences may be characterized according to certain key quantitative attributes that affect offsite consequences. These attributes include two important factors:

1) Timing (e.g., early or late releases); and,
2) Total quantity of fission products released.

Therefore, the containment event tree end states are meant to represent the source term magnitude and relative timing of the radionuclide release. The number of categories (thirteen) to be used in the source term characterization of magnitude and timing offers a level of discrimination similar to that included in numerous published PRAs.

The IPE process has received extensive guidance from the NRC staff to identify areas of special emphasis. There are a number of issues regarding the definition CET end states that are summarized below:

- Timing of radionuclide release, per se, does not appear to be a parameter requested by either the Generic Letter 88-20 or NUREG-1335 (the guidance document). However, the guidance document (p.A-11) does indicate that the time of containment failure would be of interest.

- It is stated in NUREG-1335 that for accident management evaluations the timings of accident sequence events (presumably containment failure and release are key items) are important to include.

- Generic Letter 88-20 refers to a source term magnitude greater than 10% I and 10% Cs as a sequence which is to be reported (i.e., WASH-1400, BWR-3 Category or higher releases).[1, 2]

- The NRC staff in NUREG-1335 (p. A-12) states that:

  During the last several years, there have been extensive evaluations of fission product release (source terms) during severe accidents for a variety of reactor designs. The staff encourages the use of these existing calculations whenever they can be shown to be applicable. Consideration must be given to the types of sequences in the release category, however, and the timing on release characteristics for each before selecting release characteristics to represent the category.

- The NRC staff (NUREG-1335 p. C-14) states that "a source term" should be reported for all functional sequences with core damage frequency at or above 1E-6/year (1E-7 per year for systemic sequences). In addition, IPE review documents from the NRC state that the release magnitude of up to 100 sequences with frequencies above 1E-6/year should be estimated. These can be estimated using:

---

[1] BWR 3 Release Category:

Species	Release Fraction
Noble Gases (NG)	1.0
I	0.1
CS	0.1
Te	0.3

[2] Note that in subsequent discussions, these releases are denoted as the High (H) category of the classification scheme devised in this evaluation.

- Code calculations, or
- Past published calculations.

The radonuclide release magnitude, the release timing, and the implications of each are determined using the results of MAAP calculations and past PRA evaluations. The information developed in previous studies has been used in making subjective assessments for these source term characterizations. The event sequences contributing to a radionuclide release are grouped. Those that are similar in timing and release fractions are sorted into groups of release categories to assist in summarizing the results. As requested by the NRC, the Nine Mile Point 2 IPE includes a cross check of accident sequences for: frequency, containment bypass, containment isolation, containment system availability, and approximate source term. The cross checking sheet is a direct output of RISKMAN and is included as part of the Tier 2 documentation for each of the top 200 accident sequences. Further, as requested by the NRC the IPE also examines in detail the status of the containment systems and related systems prior to core melt.

The next section identifies the criteria used to define the release bins used in the Nine Mile Point 2 IPE analysis.

4.7.4 Criteria Used in Timing and Release Magnitude Assignments

The release categories are defined based on two parameters: timing and severity.

4.7.4.1 Timing Bins

Timing of the release for each sequence is based on MAAP calculations of the sequence chronology and refers to the relative time of radionuclide release compared to the time that Emergency Action Levels (EALs) are exceeded. Three timing categories are used, as follows:

1. Early (E): Less than 6 hours from accident initiation[1]
2. Intermediate (I): Greater than or equal to 6 hours, but less than 24 hours
3. Late (L): Greater than or equal to 24 hours.

The definition of the categories is based upon past experience with Level 3 PRAs such as Limerick and Shoreham concerning offsite accident response:

- 0-6 hours is conservatively assumed to include cases in which minimal offsite protective measures have been observed to be performed in non-nuclear accidents.

- 6-24 hours is a time frame in which much of the offsite nuclear plant protective measures can be assured to be accomplished.

- > 24 hours are times at which the offsite measures can be assumed to be fully effective.

---

[1]Essentially the time when Emergency Action Levels (EALs) are entered.

Figures 4.7-3 and 4.7-4 are example applications of the determination of radionuclide release time categorizations as a function of:

- The accident type
- The time of release relative to the Emergency Action Level.

The Emergency Action Level is used as the trigger for interaction and is generally considered to occur essentially at the time of initial perturbation, i.e., within 20-30 minutes of accident initiation.

4.7.4.2 Release Magnitude Bins

The five severity classifications associated with volatile or particulate releases[1] are defined as follows:

1) High (H) - A radionuclide release of sufficient magnitude to be reasonably likely to cause early fatalities.

2) Moderate (M) - A radionuclide release of sufficient magnitude to cause near-term health effects.

3) Low (L) - A radionuclide release with the potential for latent health effects.

4) Low-Low (LL) - A radionuclide release with undetectable or minor health effects.

5) Negligible (OK) - A radionuclide release that is less than or equal to the containment design basis leakage.

The quantification of the source terms associated with each of these release severity categories was accomplished through plant specific NMP2 deterministic calculations. In addition, a review of existing consequence analyses performed in previous IDCOR studies, PRAs, and NRC studies containing detailed consequence modeling was used as a cross check on the results. To date, no single consequence analysis has evaluated all of the release paths identified in this study. Therefore, it was necessary to identify a common factor that could be used to allow the results of consequence analyses from different studies to be used in this study. The review of previous studies revealed an assumption that could be made relating

---

[1] The effects of noble gases may be quite dramatic, causing substantial health effects if released at a time in an accident before protective action can be initiated and if the associated plume is directed at an occupied location. The noble gases themselves may result in "prompt" (also called "early") injuries or fatalities. In the definition of the above timing categories, the effects of nobal gases have been implicitly included in included in the assignment of radionuclide release categories. Substaintial weight is given to the dominant term in cost benefit evaluations from past assessments, i.e., the latent health effects for which the above formulation of release magnitude adequately encompasses the effects of noble gases on the release. This approach is adequate for the purpose of comparing source terms form other PRAs on a common basis using CsI as the principal comparative parameter. The principal caution, then is to recognize that "Early" releases will include noble gases and therefore may have some potential to induce early or prompt health effects event for cases in which the CsI release fraction is less that 0.1

release characteristics based on CsI release fraction to off-site consequences. That is, an approximate relationship exists between the fraction of cesium and iodine released and the whole-body population dose. Based on the compilation of of a number of consequence analyses, however, one method [Ref. 138] has been developed that provides an approximate relationship for the minimum fractions of radionuclides released that result in "early fatalities or "early" injuries. For the release of iodine, for example, the thresholds for early fatalities and early injuries occur at releas fractions of the core inventory of approximately 0.1 or 0.01, respectively. Figure 4.7-5 shows the conditional mean number of latent cancer fatalities as a function of the cesium release fraction. Cesium is chosen as a measure of the source term magnitude because it delivers a substantial fraction of the total whole body population dose.

A significant feature of Figure 4.7-5 is that a reduction in the source term magnitude by a given factor does not lead to a reduction in the number of latent cancer fatalities by the same factor. For low source terms, the population dose tends to be dominated by the noble gases because for the source terms considered here, the noble gas release fraction remains approximately equal to unity even when the cesium release fraction becomes very small. This is why the curve shown in Figure 4.7-5 tends to flatten out at the left-hand end. Therefore, in the release fractions of 10^{-3} to 10^{-4} Cs the number of latent fatalities are found to be less than 1% of the latent fatalities for the highest release. The latent fatalities are dominated by the noble gas release. This grouping of releases is referred to in this analysis as the LL grouping.

Figure 4.7-6 summarizes the impacts of release magnitude on another health effects measure, i.e., the early fatalities. The line drawn through the results is a representation of where the base case results of a typical PRA might lie given "reasonable" assumptions about evacuation and the availability of medical treatment.

The wide range of uncertainties shown in Figure 4.7-6 indicates that drawing conclusions about the effect of variations in the source term magnitude on public risk is not always simple. However, the most significant feature of Figure 4.7-6 is that, once the average release fraction (Cs, I) falls below ~ 0.1, the conditional mean number of early fatalities is very small or zero except for a few outliers that correspond to some pessimistic assumptions.

Once the source term climbs above 0.1, however, the mean increases very rapidly because the source terms are large enough to ensure that doses above the early fatality threshold can sometimes occur among centers of population a few miles (kilometers) from the site. These conclusions seem to be very robust with respect to uncertainties. Therefore, releases with Cs or I fractions above 0.1 are included as the high (H) release category.

Moderate and low release categories are simple interpolations between H and LL using the approximate 1 to 1 relationship observed in latent health effects (Figure 4.7.4-3) over this range of CsI release.

Using these insights, a relationship was developed with the five release severity categories. The results of this partitioning are as follows:

Release Severity	Fraction of Release CsI Fission Products
High	greater than 10%
Moderate	1 to 10%
Low	0.1 to 1.0%
Low-Low[1]	less than 0.1%
Negligible	much less than 0.1%

[1] This category includes some venting sequences where only the nobel gases are released.

This relationship allows the use of results of many consequence analyses in providing source terms from the breadth of release paths analyzed in this study. Understanding the plant specific influences on each sequence source term as affected by the various release paths allows the assignment of release severity to each of the sequences.

Because timing can be an important parameter in assessing accident management and emergency response actions, the timing of the release is carried along with the end state definition. This release timing is a surrogate for the containment failure timing and is judged to be a more useful parameter.

Therefore, the containment event tree end states are characterized using a two-term matrix (i.e., severity and timing) as shown in Table 4.7-0.

4.7.5 MAAP Calculational Results for Release Bins

The extensive MAAP evaluations performed for NMP2 are used to enhance the knowledge of accident progression modeling at NMPC, to characterize the radionuclide release and states for the CETs, and to provide input on the success criteria to be applied for each CET node. Tables 4.7-1, 2, and 3 provide Nine Mile Point 2 specific MAAP sequence calculations for radionuclide release fractions. The information contained in these tables summarizes the key parameters for the classification of each Nine Mile Point 2 CET sequence in terms of radionuclide release timing and severity.

Table 4.7-1 identifies the boundary conditions that can be used to distinguish between successful stable cases with the containment intact (denoted as "S" in the last column). This table is useful in establishing the timing for key accident sequences and identifying the minimum set of equipment necessary to preserve the containment boundary, given a severe accident.

Table 4.7-2 is used to characterize the containment status at the failure condition (pressure, temperature and failure size) plus whether the pool has been bypassed, i.e., are the downcomers failed or are there stuck open Ww-DW vacuum breakers. This table also identifies whether water has been supplied to debris ex-vessel for any substantial time and what is the fraction of the core remaining in the RPV and the ex-pedestal portion of the drywell at long times into the accident.

Table 4.7-3 summarizes the CsI distribution at the time declared as the end of the calculation (36 hours past the RPV beach by the debris). The release characterization magnitude and timing is also presented in this table.

From these tables, many of the key insights that can be used in the severe accident evaluation can be developed. There are however, subtleties associated with each computer run that make a careful scrutiny of the input data and output graphical results a prudent step. Therefore, NMPC has developed a complete set of reference books (7 volumes) containing the MAAP analyses.

Making use of these deterministic calculations, a simplified matrix can be assembled to define the end state radionuclide release magnitude. The following is a summary of that approach and provides the general guidance for disposition of the CET sequences.

<u>Timing</u>

The timing of the release bins are dependent on both the Level 1 accident sequence timing and the status of the CET functional events.

First, the Level 1 accident sequences have the following effects on release timing:

	SEQUENCE	INFERRED TIMING
•	TW (Class II)	L
•	ATWS (Class IV)	E
•	SBO (Late)[1]	I or L[3]
•	SBO (Early)[2]	E or I or L[3]
•	TQUX (Class IA)	E or I or L[3]
•	TQUV (Class ID)	E or I or L[3]
•	LOCA plus vapor suppression failure	E

Overlaid on top of these Level 1 characteristic times are effects resulting from the CET mitigation systems. Based on the plant specific Nine Mile Point 2 MAAP calculations, Table 4.7-4 summarizes the Level 2 CET top event and the Level 1 accident class dependencies that can alter the timings for individual CET sequences. Using these dependencies, the RISKMAN binning rules for radionuclide release categories at the end of every sequence can be written.

<u>Magnitude of Radionuclide Release</u>

The rules for assigning release magnitude categories are described below:

1. There are three fundamental variables

---

[1] SBO related to initial successful injection but subsequent loss of injection due to battery depletion, high pool temperature, etc.

[2] SBO related to loss of all injection in 0-2 hour time frame.

[3] Timing dependent upon subsequent CET top events.

- Initial containment failure mode,
- Water availability, and
- Reactor building effectiveness.

An evaluation of these variables, to a large degree, determines the magnitude. Table 4.7-5 summarizes these deterministic calculated release magnitudes.

In addition to the containment failure modes identified in Table 4.7-5, it is also necessary to estimate the source term for severe accidents for which the containment remains substantially intact. A recent estimate of source terms by members of NRC AEOD and NRR staffs [4.5-2] indicates (with the containment intact and leaking at its maximum technical specification leakage rate) that the escape fraction would be 2E-4 for the initial one hour of the release.

Therefore, with no reactor building filtration or holdup effectiveness, the leakage escape fraction could translate into a release fraction of 0.0048 to the environment over 24 hours (i.e., the low (L) category). If the reactor building remains effective in removing some of the radionuclides through condensation, inertial deposition, or gravitational settling, then the release fraction is estimated to be between .0001 and .001, i.e., the low-low (LL) category. If SGTS is operational essentially no release is expected.

There are exceptions and variations that can be important in the assessment that also create variability in the release magnitude. The remaining rules are these exceptions.

2. There are energetic failures of the containment drywell at approximately the time of RPV failure. It is assumed based on a spectrum of MAAP analyses that a sufficient fraction of CsI is airborne to result in the ejection of a large CsI release. Therefore, sequences involving CZ/CE and CX/CY functional node failures are ranked as high (H) release categories.

3. Containment isolation failure is treated conservatively in the assignment of radionuclide release end states, sequences assigned a high (H) release in the case of IS failure even though:

   - The failures could be relatively small,
   - The failures could be from the wetwell airspace, or
   - The failures could be into closed or filtered systems (e.g., SGTS).

   This is particularly true for the containment DW sump and equipment sump drain lines. A NMP2 MAAP calculation indicated that isolation failures in these lines were medium magnitude releases. Nevertheless, they are treated conservatively in the NMP2 model, i.e., as if the reactor building is ineffective in mitigating the release.

   Therefore, the failure to isolate that bypasses the reactor building is assumed to lead to a high release.

4. A review of the NMP2 MAAP evaluations indicates that there are no containment failures (including ATWS failures) with initial wetwell failure that subsequently degraded into a drywell head failure even with the loss of injection makeup. Specifically, drywell temperatures remain below 800°F for the calculated assessment time and with the containment depressurized (i.e., failed). No other drywell failure mode is induced.

5. Events during which the containment flood contingency is successfully implemented and completed (i.e., FC = S * CX = S * FD = S), are found to have the possibility of direct releases from the RPV to the condenser and from the drywell through the drywell vent. Conservative estimates based on Nine Mile Point 2 MAAP calculations are used to characterize the release categories as follows:

   - Successful containment flood, but ineffective reactor building, condenser, or turbine building effectiveness results in a high (H) release

   - With effective mitigation by the secondary buildings and the condenser/ turbine building, the release is classified as moderate (M).

6. Scenarios involving wetwell airspace or wetwell vent failure are treated as scrubbed releases when there is no suppression pool bypass, and are assigned a severity class of low-low (LL) if the suppression pool remains subcooled.

7. Wetwell failures below the water line result in the water level equilibrating inside and outside the wetwell to cover the breach (i.e., assumed at the basemat juncture as determined in the ABB Impell structural analysis). Therefore, the RB node for wetwell failures below the water line is used to distinguish whether the scrubbing of the source term was effective to reduce to release through the breach. The magnitude of the release for the scrubbed case is the same as a wetwell vent case without suppression pool bypass.

8. One of the most complex conditions is that related to sequential containment failures associated with the termination of injection following initial containment failures. In such situations, containment failure will eventually progress to a drywell failure. Consequently, the release is usually relatively small during the period in which the water and saturation conditions exist inside containment. At the time of containment dryout, which tends to be late in such a sequence, the release magnitude could increase if sufficient material is located on containment structural surfaces that can be revaporized as the airspace temperature increases.

9. Use of the hard pipe vent results in bypassing the reactor building. Therefore, the reactor building node is not considered in sequences where successful wetwell venting has occurred (CV = success).

10. Suppression pool bypass is modeled in two ways in the CET:

    - First, as a method of potential containment challenge during blowdown (see CZ node). Failure of this function results in a high release.

- First, as a method of potential containment challenge during blowdown (see CZ node). Failure of this function results in a high release.

- Second, it is modeled as a leakage failure of the vacuum breaker in the wetwell to drywell interface that allows some radionuclides to bypass the suppression pool. The impact of bypassing the suppression pool is modeled as an increase of a factor of 10 in the radionuclide release.

11. Failure of reactor depressurization and failure of containment heat removal have not been determined, based on these analyses or other referenced analyses, to result in substantial changes to the release fraction unless other containment failure modes are encountered, that is, the release is determined by the specific containment failure mode not depressurization (success or failure) or conteinment heat removal (success or failure)

12. Re-evolution of fission products from the suppression pool is not explicitly modeled in the MAAP code. (See Section 4.7.2.4.2 for additional discussion of this modeling feature.)

Table 4.7-0

LEVEL 2 END STATE BINS:

RADIONUCLIDE RELEASE SEVERITY AND TIMING CLASSIFICATION SCHEME
(SEVERITY, TIMING)(2)

Release Severity		Release Timing	
Classification Category	Cs Iodide % in Release	Classification Category	Time of Initial Release(1) Relative to Accident Initiation
High (H)	greater than 10	Late (L)	greater than 24 hours
Moderate (M)	1 to 10	Intermediate (I)	6 to 24 hours
Low (L)	0.1 to 1	Early (E)	less than 6 hours
Low-low (LL)	less than 0.1		
No iodine (OK)	0		

(1) The accident initation is used as the surrogate for the time when EALs are exceeded which would dictate the time of release

(2) Thirteen (13) Level 2 End State Bins:
H/E
H/I
H/L
M/E
M/I
M/L
L/E
L/I
L/L
LL/E
LL/I
LL/L
OK

TABLE 4.7-1
MAAP Thermal Hydraulic Summary - Success States

Sequence Designator	Sequence Timing (hrs.) RPV Depressurization	Core Uncovered	Below 1/3 Core Height	Onset Core Melt	Peak Core Temperature	Vessel Failure	Containment Events Vent or Fail	Time / Location	Containment Peak Conditions Pressure (psia)	Suppression Pool Temperature (°F)	Drywell Temp. (°F)	Success or Failure (1)
IA1LD	2.74	0.5	0.78	1.06	2.74	2.74	F	2.79 /DWH	118.2	154.2	596.3	F
IA1LDDB	3.43	0.5	0.78	1.06	2.79	3.43	NA	_ /DWH	73.55	169.1	571.5	S
IA1LDFC	2.74	0.5	0.78	1.06	2.74	2.74	F	2.78 /DWH	121.2	152.8	595.7	F
IA1LDNP	2.74	0.5	0.78	1.06	2.74	2.74	NA	_ /_	74.08	158.9	560.5	S
IA1SD	2.74	0.5	0.78	1.06	2.74	2.74	F	2.79 /DWH	120.3	155.7	596.3	F
IA4DN10	2.7	0.5	0.78	1.06	2.7	2.7	NA	_ /_	98.09	295.7	615.	S
IA4DN10A	2.81	0.5	0.78	1.06	2.81	2.81	NA	_ /_	110.7	300.5	732.1	S
IA4I3NRR	2.84	0.5	0.78	1.06	2.84	2.84	F	0. /DWH	61.5	242.	593.7	F
IA4IS3NP	2.78	0.5	0.78	1.06	2.78	2.78	F	0. /DWH	61.46	241.7	589.5	F
IA4LDNCP	2.69	0.5	0.78	1.06	2.69	2.69	F	34.18 /DWH	123.3	315.6	599.2	F
IA4LDNP	2.69	0.5	0.78	1.06	2.69	2.69	F	31.61 /DWH	98.11	281.4	686.4	F
IA4LDXNC	2.69	0.5	0.78	1.06	2.69	2.69	F	24. /DWH (3)	74.73	220.1	741.9	F
IA4LWXN7	2.69	0.5	0.78	1.06	2.69	2.69	F	6.99 /WWA	74.12	211.3	740.2	F
IA4LWXNC	2.69	0.5	0.78	1.06	2.69	2.69	F	11.16 /WWA	74.73	211.4	787.5	F
IA6LD	1.06	0.5	0.78	1.06	2.32	2.32	NA	_ /_	52.03	169.1	392.6	S
IA7LD	1.06	0.5	0.77	1.06	2.27	2.27	F	0. /DWH	38.66	164.6	367.7	F
IA7LW	1.07	0.5	0.78	1.06	2.28	2.28	F	0. /WWA	41.91	164.3	373.5	F
ID1LD	0.42	0.44	0.46	1.02	1.53	1.53	V	18.89 /WWA	59.69	278.7	420.4	S*
ID2LD	0.42	0.44	0.46	1.01	1.52	1.52	NA	_ /_	54.83	165.2	745.	S
ID2LDCP	0.42	0.44	0.46	1.01	1.52	1.53	NA	_ /_	42.57	168.3	426.5	S
ID2LW	0.42	0.44	0.46	1.01	1.52	1.52	NA	_ /_	54.83	165.2	745.	S
ID8LD	0.42	0.44	0.46	1.02	1.53	1.53	V	6.43 /WWA	61.08	233.	788.5	S*
ID8LDC1	0.42	0.44	0.46	1.01	1.61	1.61	V	7.46 /WWA	61.13	233.9	799.4	S*
ID8LDC2	0.42	0.44	0.46	1.02	1.53	1.53	V	6.81 /WWA	60.99	241.	577.5	S*

TABLE 4.7-1
MAAP Thermal Hydraulic Summary - Success States

Sequence Designator	Sequence Timing (hrs.)						Containment Events		Containment Peak Conditions			Success or Failure (1)
	RPV Depressurization	Core Uncovered	Below 1/3 Core Height	Onset Core Melt	Peak Core Temperature	Vessel Failure	Vent or Fail	Time / Location	Pressure (psia)	Suppression Pool Temperature (°F)	Drywell Temp. (°F)	
ID8LDCP	0.42	0.44	0.46	1.02	1.53	1.53	V	7.61 /WWA	60.43	245.4	433.3	S*
ID8LDNPV	0.42	0.44	0.46	1.02	1.53	1.53	F	23.71 /DWH	100.7	291.2	654.6	F
ID8LDXC	0.42	0.44	0.46	1.02	1.53	1.53	V	5.41 /WWA (5)	62.51	228.2	799.9	S*
ID8LWN7V	0.42	0.44	0.46	1.02	1.53	1.53	F	9.88 /WWA	97.97	213.1	785.7	F
ID8LWNPV	0.42	0.44	0.46	1.02	1.53	1.53	F	23.71 /WWA	100.7	292.6	770.2	F
IIA1LD	3.73	31.72	32.14	33.25	34.95	35.	F	31.56 /DWH	153.7	360.4	536.3	F
IIA1LW	3.73	31.86	32.3	33.45	35.12	35.17	F	31.56 /WWA	153.7	360.4	632.2	F
IIA1SD	3.73	34.33	35.33	37.11	37.13	38.02	F	31.56 /DWH	153.7	360.8	551.1	F
IIA1SW	3.73	34.35	35.33	35.96	38.56	38.61	F	31.56 /WWA	153.7	360.6	684.1	F
IIA2LD	3.73	34.35	35.31	35.96	39.74	39.81	F	31.56 /DWH	153.7	360.4	590.7	F
IIA2LW	3.73	34.33	35.26	35.92	39.49	39.53	F	31.56 /WWA	153.7	360.4	581.5	F
IIA2SD	3.73	34.29	35.28	35.89	39.61	39.69	F	31.56 /DWH	153.7	360.8	566.8	F
IIA2SW	3.73	34.34	35.33	35.94	39.94	39.94	F	31.56 /WWA	153.7	360.6	776.	F
IIIC1LD	0.	0.02	0.01	0.28	0.67	0.67	V	1.24 /WWA	60.25	229.3	647.3	S*
IIIC1V10	0.	0.02	0.01	0.29	0.66	0.67	V	1.46 /WWA	63.47	228.5	665.	S*
IIIC1V56	0.	0.02	0.01	0.29	0.66	0.67	V	1.27 /WWA	59.92	292.1	645.	S*
IIT1LD	0.29	25.61	26.51	27.08	30.34	30.34	F	30.34 /DWH	147.7	355.1	447.2	F
IIT1LW	0.29	25.61	26.51	27.08	30.34	30.34	F	30.34 /WWA	147.7	354.8	464.6	F
IIT1SD	0.29	25.61	26.51	27.08	30.34	30.34	F	30.34 /DWH	226.5	362.	522.4	F
IIT1SW	0.29	25.61	26.51	27.08	30.34	30.34	F	30.34 /WWA	209.1	364.7	543.3	F
IVA1LD	1.13	0.14	1.15	1.69	1.69	2.05	F	1.13 /DWH	40.48	268.9	582.8	F
IVA1LDDB	1.13	0.15	1.15	1.68	1.69	6.29	F	1.13 /DWH	40.51	268.7	496.8	F
IVA1LW	1.13	0.14	1.15	1.66	2.01	2.01	F	1.13 /WWA	40.87	268.6	713.1	F
IVA1LWNP	1.13	0.14	1.15	1.66	2.01	2.01	F	1.13 /WWA	40.87	268.6	751.2	F

TABLE 4.7-1
MAAP Thermal Hydraulic Summary - Success States

Sequence Designator	Sequence Timing (hrs.)						Containment Events		Containment Peak Conditions			Success or Failure (1)
	RPV Depressurization	Core Uncovered	Below 1/3 Core Height	Onset Core Melt	Peak Core Temperature	Vessel Failure	Vent or Fail	Time / Location	Pressure (psia)	Suppression Pool Temperature (°F)	Drywell Temp. (°F)	
IVA2LD	3.52	0.14	1.22	1.56	3.52	3.52	F	1.13 /[illegible]	89.14	260.8	662.6	F
IVA2LW	3.45	0.14	1.2	1.56	3.45	3.45	F	1.13 /WWA	86.95	260.4	751.2	F
IVA2LWA6	3.51	0.14	1.21	1.57	3.51	3.51	F	1.13 /WWA	89.63	260.5	753.4	F
IVA2LWN7	3.45	0.14	1.2	1.56	3.45	3.45	F	1.13 /WWA	85.65	260.4	751.3	F
IVA2LWNP	3.45	0.14	1.2	1.56	3.45	3.45	F	1.13 /WWA	85.65	260.4	768.5	F
V1LD	0.	0.02	0.02	0.38	0.72	0.72	F	__ /__	19.74	222.2	528.	F
V2LD	0.	0.02	0.02	0.37	0.67	0.72	F	__ /__	19.57	222.4	523.5	F

TABLE 4.7-2
MAAP Sequence Thermal Hydraulic Analysis Summary

Sequence Designator	Containment Conditions at Failure/Vent					Water on Debris Ex-Vessel	Note	Core Fraction Long Term	
	Pressure (psia)	Suppression Pool Temperature (°F)	Drywell Temp. (°F)	Failure Size (ft^2)	Suppression Pool Bypass			In RPV	In DW
IA1LD	119.57	138.97	561.74	2.	BP	CS	ON FOR ENTIRE RUN	0.44	0.25
IA1LDDB	—	—	—	2.	BP	—	—	0.37	0.24
IA1LDFC	119.59	138.81	562.95	2.	BP	CS	ON FOR ENTIRE RUN	0.44	0.25
IA1LDNP	—	—	—	—	I	CS	ON FOR ENTIRE RUN	0.44	0.25
IA1SD	119.57	138.97	561.74	0.194	BP	CS	ON FOR ENTIRE RUN	0.44	0.25
IA4DN10	—	—	—	—	I	NONE	—	0.17	0.25
IA4DN10A	—	—	—	—	I	NONE	—	0.17	0.25
IA4I3NRB	15.45	84.81	135.	0.175	I	NONE	—	0.17	0.25
IA4IS3NP	15.45	84.81	135.	0.175	I	NONE	—	0.2	0.24
IA4LDNCP	123.36	315.64	599.24	2.	I	NONE	—	0.	0.31
IA4LDNP	98.25	281.3	686.56	2.	I	NONE	—	0.11	0.25
IA4LDXNC	73.98	170.11	674.51	2.	I	NONE	—	0.11	0.43
IA4LWXN7	74.26	157.08	604.24	2.	BP	NONE	—	0.12	0.43
IA4LWXNC	74.02	170.1	675.5	2.	I	NONE	—	0.12	0.43
IA6LD	—	—	—	—	BP	CS	ON FOR ENTIRE RUN	0.57	0.14
IA7LD	15.45	84.81	135.	1.069	BP	CS	ON FOR ENTIRE RUN	0.57	0.14
IA7LW	15.45	84.81	135.	0.785	BP	CS	ON FOR ENTIRE RUN	0.57	0.15
ID1LD	59.69	278.3	420.31	0.244	I	CS	ON FOR ENTIRE RUN	0.44	0.15
ID2LD	—	—	—	—	I	LPCI	ON FOR ENTIRE RUN	0.34	0.
ID2LDCP	—	—	—	—	I	LPCI	ON FOR ENTIRE RUN	0.	0.
ID2LW	—	—	—	—	I	LPCI	ON FOR ENTIRE RUN	0.34	0.
ID8LD	61.51	187.33	383.02	0.244	BP	NONE	—	0.29	0.
ID8LDC1	61.27	193.78	386.67	0.244	BP	NONE	—	0.29	0.
ID8LDC2	61.27	197.68	356.67	0.244	BP	NONE	—	0.23	0.
ID8LDCP	60.5	210.99	343.19	0.244	BP	NONE	—	0.	0.

TABLE 4.7-2
MAAP Sequence Thermal Hydraulic Analysis Summary

Sequence Designator	Containment Conditions at Failure/Vent					Water on Debris Ex-Vessel	Note	Core Fraction Long Term	
	Pressure (psia)	Suppression Pool Temperature (°F)	Drywell Temp. (°F)	Failure Size (ft^2)	Suppression Pool Bypass			In RPV	In DW
ID8LDNPV	98.8	291.01	653.43	2.	I	NONE	—	0.34	0.
ID8LDXC	65.14	171.19	403.99	0.244	BP	NONE	—	0.29	0.
ID8LWN7V	98.31	210.57	424.97	2.	BP	NONE	—	0.28	0.
ID8LWNPV	98.38	291.01	654.07	2.	I	NONE	—	0.31	0.
IIA1LD	155.45	361.24	360.77	2.	BP	NONE	—	0.34	0.
IIA1LW	155.45	361.24	360.77	2.	BP	NONE	—	0.34	0.
IIA1SD	155.45	361.24	360.77	0.194	BP	NONE	—	0.37	0.
IIA1SW	155.45	361.24	360.77	0.194	BP	NONE	—	0.37	0.
IIA2LD	155.45	361.24	360.77	2.	BP	NONE	—	0.23	0.23
IIA2LW	155.45	361.24	360.77	2.	BP	NONE	—	0.17	0.23
IIA2SD	155.45	361.24	360.77	0.194	BP	NONE	—	0.23	0.22
IIA2SW	155.45	361.24	360.77	0.194	BP	NONE	—	0.34	0.23
IIIC1LD	58.99	127.36	358.87	0.244	BP	NONE	—	0.29	0.
IIIC1V10	64.14	129.98	391.21	0.244	BP	—	—	0.29	0.
IIIC1V56	59.75	128.39	386.1	0.244	BP	—	—	0.34	0.
IIT1LD	147.7	353.2	384.25	2.	BP	LPCI	TRIPPED AT ≈ 38 HRS	0.	0.21
IIT1LW	147.7	353.2	384.25	2.	BP	LPCI	TRIPPED AT ≈ 40 HRS	0.	0.21
IIT1SD	147.7	353.2	384.25	0.194	BP	LPCI	TRIPPED AT ≈ 46 HRS	0.	0.21
IIT1SW	147.7	353.2	384.25	0.194	BP	LPCI	TRIPPED AT ≈ 50 HRS	0.	0.21
IVA1LD	35.75	260.12	231.26	2.	BP	NONE	—	0.29	0.
IVA1LDDB	36.26	259.71	239.12	2.	BP	—	—	0.23	0.
IVA1LW	35.64	260.12	231.56	2.	BP	NONE	—	0.29	0.
IVA1LWNP	35.64	260.12	231.56	2.	I	NONE	—	0.29	0.
IVA2LD	34.99	260.36	237.26	2.	BP	NONE	—	0.29	0.25
IVA2LW	34.99	260.36	237.26	2.	BP	NONE	—	0.29	0.23

TABLE 4.7-2
MAAP Sequence Thermal Hydraulic Analysis Summary

Sequence Designator	Containment Conditions at Failure/Vent					Water on Debris Ex-Vessel	Note	Core Fraction Long Term	
	Pressure (psia)	Suppression Pool Temperature (°F)	Drywell Temp. (°F)	Failure Size (ft^2)	Suppression Pool Bypass			In RPV	In DW
IVA2LWA6	34.37	260.	239.02	2.	BP	NONE	—	0.29	0.19
IVA2LWN7	34.99	260.36	237.26	2.	BP	NONE	—	0.29	0.24
IVA2LWNP	34.99	260.36	237.26	2.	I	NONE	—	0.29	0.24
V1LD	—	—	—	1.78	BP	NONE	—	0.29	0.
V2LD	—	—	—	1.78	BP	NONE	—	0.29	0.

TABLE 4.7-3
MAAP Sequence Calculations for Radionuclide Release Fractions

Sequence Designator	Containment Conditions at Time of Failure/Vent: Pedestal Concrete Attack Depth (ft.)	Total H2 Produced RPV Failed (lbm) / Containment Vent/Fail (lbm)	Ex-Vessel H2 Fraction (mole fraction): Drywell	Pedestal	Wetwell	CsI Distribution (VF+36 hrs): Initial Release Time (Hrs)	Release Duration (Hrs)	Mass Fraction to the Reactor Bldg.	Mass Fraction to the Environment	Release Characterization: Severity	Timing
IA1LD	0.	941. / 1021.	0.0294	0.005	0.1554	2.79	35.95	0.0824	0.0469	M	E
IA1LDDB	—	1160. / —	—	—	—	—	—	0.	0.	NA	NA
IA1LDFC	0.	933. / 1020.	0.0301	0.005	0.1541	2.78	35.96	0.0832	0.0474	M	E
IA1LDNP	—	861. / —	—	—	—	—	—	0.	0.	NA	NA
IA1SD	0.	941. / 1021.	0.0294	0.005	0.1554	2.79	35.95	0.0503	0.0306	M	E
IA4DN10 (1)	—	857. / —	—	—	—	42. (4)	-3.3	0.	0.	NA	NA
IA4DN10A	—	862. / —	—	—	—	43.62	-4.81	0.	0.	M	NA
IA4I3NRR	0.	949. / 0.	0.	0.	0.	1.25	37.59	0.1641	0.0034	L (7)	E
IA4IS3NP	0.	877. / 0.	0.	0.	0.	1.25	37.53	0.1626	0.0347	M (6),(7)	E
IA4LDNCP	0.0167	858. / 863.	0.0426	0.0425	0.1062	34.18	4.51	0.002	0.0009	LL	L
IA4LDNP	0.0347	858. / 1037.	0.0393	0.0393	0.1906	31.61	7.079999	0.118	0.0756	M	L
IA4LDXNC	0.3357	858. / 1134.	0.0416	0.0393	0.2913	11.16	27.53	0.0586	0.0403	M	I
IA4LWXN7	0.3044	858. / 1124.	0.0584	0.0571	0.243	6.99	31.7	0.0149	0.0113	M	I
IA4LWXNC	0.3357	858. / 1134.	0.0416	0.0394	0.2913	11.16	27.53	0.0002	0.0002	LL	I
IA6LD	—	441. / —	—	—	—	—	—	0.	0.	NA	NA
IA7LD	0.	492. / 0.	0.	0.	0.	1.06	37.21	0.0136	0.0006	LL	E
IA7LW	0.	473. / 0.	0.	0.	0.	1.07	37.21	0.0002	1.E-4	LL	E
ID1LD	0.	142. / 142.	0.0184	0.017	0.0261	18.89	18.64	0.001	0.001	L (2)	I
ID2LD	—	149. / —	—	—	—	49.86	-12.34	0.	0.	NA	NA
ID2LDCP	—	149. / —	—	—	—	—	—	0.	0.	NA	NA
ID2LW	—	149. / —	—	—	—	49.86	-12.34	0.	0.	NA	NA
ID8LD	0.0245	142. / 266.	0.0078	0.0074	0.0808	6.44	31.09	0.0745	0.0745	M	I
ID8LDC1	0.0226	131. / 150.	0.0045	0.0043	0.0454	7.46	30.15	0.0689	0.0689	M	I

TABLE 4.7-3
MAAP Sequence Calculations for Radionuclide Release Fractions

Sequence Designator	Containment Conditions at Time of Failure/Vent						CsI Distribution (VF+36 hrs)				Release Characterization	
	Pedestal Concrete Attack Depth (ft.)	Total H2 Produced RPV Failed (lbm)	Total H2 Produced Containment Vent/Fail (lbm)	Ex-Vessel H2 Fraction (mole fraction) Drywell	Ex-Vessel H2 Fraction (mole fraction) Pedestal	Ex-Vessel H2 Fraction (mole fraction) Wetwell	Initial Release Time (Hrs)	Release Duration (Hrs)	Mass Fraction to the Reactor Bldg.	Mass Fraction to the Environment	Severity	Timing
ID8LDC2	0.0189	143. /	287.	0.0084	0.008	0.0869	6.81	30.72	0.0551	0.0551	M	I
ID8LDCP	0.0151	142. /	147.	0.0028	0.0028	0.0473	7.62	29.91	0.0077	0.0077	L	I
ID8LDNPV	0.0243	142. /	269.	0.007	0.0064	0.0522	23.71	13.82	0.3586	0.1668	H	I
ID8LDXC	0.6007	142. /	459.	0.0392	0.04	0.1022	5.41	32.12	0.1897	0.1897	H (5)	E
ID8LWN7V	0.0236	142. /	268.	0.0029	0.0027	0.0554	9.88	27.65	0.0453	0.0103	M	I
ID8LWNPV	0.0243	142. /	269.	0.007	0.0064	0.0522	23.71	13.82	0.0649	0.031	M	I
IIA1LD	0.	573. /	0.	0.	0.	0.	33.25	37.75	0.4674	0.3677	H	L
IIA1LW	0.	589. /	0.	0.	0.	0.	33.46	37.71	0.28	0.07	M (2)	L
IIA1SD	0.	440. /	0.	0.	0.	0.	37.12	36.9	0.48	0.3875	H	L
IIA1SW	0.	891. /	0.	0.	0.	0.	36.01	38.6	0.12	0.03	M (2)	L
IIA2LD	0.	1175. /	0.	0.	0.	0.	35.98	39.83	0.6433	0.4333	H	L
IIA2LW	0.	1205. /	0.	0.	0.	0.	35.97	39.56	0.35	0.12	H (2)	L
IIA2SD	0.	1147. /	0.	0.	0.	0.	35.94	39.75	0.4637	0.376	H	L
IIA2SW	0.	1035. /	0.	0.	0.	0.	35.96	39.98	0.13	0.02	M (2)	L
IIIC1LD	0.0024	279. /	423.	0.0734	0.0745	0.0222	1.24	35.43	0.218	0.218	H	E
IIIC1V10	0.002	308. /	405.	0.0646	0.0652	0.0262	1.46	35.21	0.2067	0.2067	H	E
IIIC1V56	0.002	308. /	405.	0.0715	0.0724	0.0231	1.27	35.4	0.0373	0.0373	M	E
IIT1LD	0.	1069. /	1069.	0.0027	0.0027	0.1408	30.34	36.	0.4828	0.208	H	L
IIT1LW	0.	1069. /	1069.	0.0027	0.0027	0.1408	30.34	36.	0.407	0.314	H (2)	L
IIT1SD	0.	1069. /	1069.	0.0027	0.0027	0.1408	30.34	36.	0.2369	0.1642	H	L
IIT1SW	0.	1069. /	1069.	0.0027	0.0027	0.1408	30.34	36.	0.106	0.03	M (2)	L
IVA1LD	0.	381. /	0.	0.	0.	0.	1.69	36.36	0.4965	0.264	H	E
IVA1LDDB	0.	736. /	0.	0.	0.	0.	1.69	40.6	0.186	0.0728	M	E

TABLE 4.7-3
MAAP Sequence Calculations for Radionuclide Release Fractions

Sequence Designator	Containment Conditions at Time of Failure/Vent: Pedestal Concrete Attack Depth (ft.)	Total H2 Produced RPV Failed (lbm) / Containment Vent/Fail (lbm)	Ex-Vessel H2 Fraction (mole fraction): Drywell	Pedestal	Wetwell	CsI Distribution (VF+36 hrs): Initial Release Time (Hrs)	Release Duration (Hrs)	Mass Fraction to the Reactor Bldg.	Mass Fraction to the Environment	Release Characterization: Severity	Timing
IVA1LW	0.	399. / 0.	0.	0.	0.	1.66	36.35	0.23	0.08	M (2)	E
IVA1LWMP	0.	399. / 0.	0.	0.	0.	1.66	36.35	0.032	0.01	M (2)	E
IVA2LD	0.	914. / 0.	0.	0.	0.	1.58	37.94	0.4728	0.2665	H	E
IVA2LW	0.	898. / 0.	0.	0.	0.	1.58	37.87	0.28	0.07	M (2)	E
IVA2LWA6	0.	881. / 0.	0.	0.	0.	1.59	37.92	0.28	0.07	M (2)	E
IVA2LWN7	0.	898. / 0.	0.	0.	0.	1.58	37.87	0.08	0.02	M (2)	E
IVA2LWMP	0.	898. / 0.	0.	0.	0.	1.58	37.87	0.024	0.006	L (2)	E
V1LD	—	243. / —	—	—	—	0.38	36.34	0.8595	0.725	H	E
V2LD	—	242. / —	—	—	—	0.38	36.34	0.822	0.704	H	E

Notes to Table 4.7-1, 2 or 3

(1) The * in the last column of Table 4.7-1 signifies that success means continment did not fail but the continment was vented.

(2) These sequences involve radionuclide release through a saturated suppression pool. The MAAP model (Rev. 7) calculates a DF = 1.0 for such cases. This is inconsistent with available data and therfore the releases have been modified to a more realistic radionuclide release. A DF = 10 has been applied to releases through a saturated pool.

DF ADJUSTMENTS FOR NMP2

	ORIGINAL		ADJUSTED	
	To RB	To ENV	To RB	To ENV
ID1LD	0.011	0.011 (M)	0.001	0.001 (L)
IIA1LW	0.446	0.091 (M)	0.28	0.07 (M)
IIA1SW	0.186	0.030 (M)	0.12	0.03 (M)
IIA2LW	0.403	0.131 (H)	0.35	0.12 (H)
IIA2SW	0.194	0.029 (M)	0.13	0.02 (M)
IIT1LW	0.407	0.314 (H)	0.407	0.314 (H)
IIT1SW	0.106	0.030 (H)	0.106	0.030 (M)
IVA1LW	0.425	0.137 (H)	0.23	0.08 (M)
IVA2LW	0.411	0.116 (H)	0.28	0.07 (M)
IVA1LWNP	0.321	0.105 (H)	0.032	0.010 (M)
IVA2LWNP	0.244	0.062 (M)	0.024	0.006 (L)
IVA2LWN7	0.278	0.069 (M)	0.08	0.02 (M)
IVA2LW4A	0.412	0.120 (H)	0.28	0.07 (M)

(3) Incorrect failure point used on input to MAAP; the calculated value with this input was 11.16 hr. A best estimate of the failure time is approximately 24 hours.

(4) The MAAP calculation was expected to fail at a pressure corresponding to 49 hours while the actual time to failure is estimated at 42 hours.

(5) This case is a sensitivity case only and should not be used as a baseline case. It includes the following assumptions:

- 1 ft of core debris is held up in the pedestal (contrary to belief that it will be transported to the pool).

- drywell failure is assumed before DW temperature exceeds the drywell failure curve.

Because of these conservations, this case does not represent a baseline case.

(6) The containment isolation failure analyses have been performed assuming the downcomers remain intact throughout the accident sequence. Because the downcomers are expected to fail, and bypass the suppression pool, it is judged that a conservative assessment of a high relese is prudent for these cases.

(7) The case with the railroad door failed open is considered to be the best estimate case based on the parameter file compiled by FAI. The non-failure of the railroad door is considered as a sensitivity case if it can be determined that the railroad door will not be forced open.

Table 4.7-4

NMP2 RADIONUCLIDE RELEASE SEQUENCE TIMING SUMMARY

Sequence	Sequence Timing Alone	FAILED CONTAINMENT CONDITIONS[1],[2]					
		$\overline{GV}$	CZ	TD[3]	CX	HR[6]	VC[7]
IA	—	E IA-1-LD	E IA-1-LD	I IA-4-LDNP IA-4-LDXNC	E[5] ID-9-LDFLD	I ID-1-LD ID-8-LDCP	I ID-1-LD ID-8LDNDV
IB Early	—	E IA-1-LD	E IA-1-LD	I IA-4-LDNP IA-4-LDXNC	E ID-9-LDFLD	I ID-1-LD ID-8-LDCP	I ID-1-LD ID-8-LDNPV
IB Late	—	I (EST.)	I (EST.)	I IA-4-LDNP IA-4-LDXNC	E ID-9-LDFLD	L (EST.) ID-8-LDCP	I (EST.)
IC	—	E IA-1-LD	E IA-1-LD	I IA-4-LDNP IA-4-LDXNC	E ID-9-LDFLD	I ID-1-LD ID-8-LDCP	I ID-1-LD
ID	—	E ID-1-LD	E ID-1-LD	I ID-8-LDNPV IA-4-LDXNC	E ID-9-LDFLD	I ID-1-LD ID-8-LDCP	I ID-1-LD ID-8-LDNPV
II A	LATE[4] IIA-1-LD; IIA-2-LD						
II V	INTERMEDIATE ID-1-LD						
II T	LATE[4] IIT-1-LD						
III A (Not Used in NMP2)	—	E ID-1-LD	E ID-1-LD	I ID-8-LDNPV	E ID-9-LDFLD	I ID-1-LD ID-8-LDCP	I ID-1-LD
III B	—	E IIIC-1-LD	E IIIC-1-LD	I ID-8-LDNPV	E ID-9-LDFLD	I ID-1-LD ID-8-LDCP	I ID-1-LD
III C	—	E IIIC-1-LD	E IIIC-1-LD	I ID-8-LDNPV	E ID-9-LDFLD	I ID-1-LD ID-8-LDCP	I ID-1-LD
III D	EARLY[4] IIIC-1-LD						
IV A	EARLY[4] IVA-1-LD; IVA-2-LD						
IV L	EARLY[4] IVA-1-LD; IVA-2-LD						
IV V	EARLY[4] IVA-1-LD						
V	EARLY[4] V-1-LD; V-2-LD						

LEGEND	
Timing →	E
MAAP Case →	I-A-7

Notes to Table 4.7-4

(1) GV - Combustible Gas Venting
CX/CY, CZ/CE - Energetic Failures
TD/TR - Failure of debris cooling
VC & HR - Failure of containment heat removal
E - Early 0 - 6 Hours
I - Intermediate 6 - 24 Hours
L - Late > 24 Hours

(2) This table is interpreted as if each containment failure mode is treated separately; i.e., it is assumed that one, and only one, failure mode occurs. The timing then reflects this particular failure mode.

(3) These cases are not relevant because without injection it is assumed shell failure will precede other drywell failure modes.

(4) The timing is set by definition of these accident sequences.

(5) The assignment of this failure mode to early is based on a very conservative timing, assuming that containment flooding can occur very rapidly.

(6) The HR failures leading to a successful vent condition are found to occur over a whole range of times from approximately 6 hours through very late times.

(7) No explicit MAAP cases available. The time for TD = S * HR = F * VC = F could be a late (L) failure. An intermediate failure time is considered a conservative estimate.

Table 4.7-5

CET NODAL EFFECTS ON SOURCE TERM MAGNITUDE
(RPV Breached by Molten Debris)

Initial Containment Failure Mode	Water Availability to the Molten Debris	Reactor Building Effectiveness	Source Term Magnitude	MAAP Reference Case
Drywell				
Large DW	Yes	Yes	L[1]	IA-4-LD-NCP[2]
Large DW	Yes	No	M	IA-4-LD-NCP[2]
Large DW	No	Yes	M[1]	IA-4-LD-NP
Large DW	No	No	H	IA-4-LD-NP ID-8-LD-NPV IVA-1-LD IVA-2-LD
Small DW	Yes	Yes	L	IA-4-LD-NCP[2] IA1-SD[3]
Small DW	Yes	No	M	IA-4-LD-NCP[2] IA1-SD[3]
Small DW	No	Yes	M	IA-4-LD-NP
Small DW	No	No	H	IA-4-LD-NP IA-1-SD[3]
Wetwell				
Wetwell Vent or Failure w/no Suppression Pool Bypass[4]	Yes	Yes	LL[1]	EST.
Wetwell Vent or Failure w/no Suppression Pool Bypass[4]	Yes	No	LL	EST.
Wetwell Vent or Failure w/no Suppression Pool Bypass[4]	No	Yes	L	IA-4-LWXN7 (WW Failure Only)
Wetwell Vent or Failure w/no Suppression Pool Bypass[4]	No	No	M	IA-4-LWXN7
Wetwell Vent or Failure with Suppression Pool Bypass	Yes	Yes	L	ID-1-LD
Wetwell Vent or Failure with Suppression Pool Bypass	Yes	No	L	ID-1-LD
Wetwell Vent or Failure with Suppression Pool Bypass	No	Yes	M	ID-8-LD-XC IVA-1-LW IVA-2-LW (WW Failure Only)
Wetwell Vent or Failure with Suppression Pool Bypass	No	No	H	ID-8-LD-XC IVA-1-LW IVA-2-LW

Notes to Tables 4.7-5

(1) Reactor building was calculated as ineffective in the MAAP case performed. Therefore, under the low probability case of the reactor building being effective the release is approximately 1 order of magnitude lower.

(2) Drywell shell failure occurs in this MAAP case because no debris cooling is available.

(3) Reactor Building is calculated as effective in the referenced MAAP case; if it is in fact ineffective the release would increase by a factor of 10.

(4) Engineering judgement and existing MAAP calculations (without venting) are used to estimate the source terms.

(5) This case is used as a surrogate for effective coolant injection. For this case, all debris is transferred to the wetwell. This "looks" similar to a case with effective core spray or drywell spray cooling of debris.

(6) A few "small drywell failure" cases were performed. These few cases tended to show that the releases were comparable to the large drywell failure cases.

(7) No suppression pool bypass means that the downcomers stay intact for at least 7 minutes (i.e., a large fraction of the time of RPV blowdown).

Table 4.7.6-1

NRC IDENTIFIED PARAMETERS FOR SENSITIVITY STUDY (NUREG-1335)

Parameters	Sensitivity Method Used	
	Probabilistic	Deterministic
• Performance of containment heat removal systems during core meltdown accidents	X	
• In-vessel phenomena (primary system at high pressure)		
- H_2 production and combustion in containment		X
- Induced failure of the reactor coolant system pressure boundary	X	
- Core relocation characteristics		X
- Mode of reactor vessel melt-through	X	
• In-vessel phenomena (primary system at low pressure)		
- H_2 production and combustion in containment	X	
- Core relocation characteristics		X
- Fuel/coolant interactions	X	
- Mode of reactor vessel melt-through	X	
• Ex-vessel phenomena (primary system at high pressure)		
- Direct containment heating concerns	X	
- Potential for early containment failure due to pressure load		X
- Long-term disposition of core debris (coolable or not coolable)		X
• Ex-vessel phenomena (primary system at low pressure)		
- Potential for early containment failure due to direct contact by core debris	X	
- Long-term core-concrete interactions:		
-- Water availability		X
-- Coolable or not coolable		X

Table 4.7.6-2

List of Sensitivity Items

Sensitivity Item	Required by GL88-20 or NUREG-1335	Deemed Useful In Nine Mile Point 2 IPE Response	Proposed Cases for Accident Management Investigations
In-vessel Core Melt Progression			
- Hydrogen Production	X	X	
- Temperature of Melt			
- Model for control rods			
- Model for candling			
- RPV breach model and assumptions	X		
- In-vessel steam explosion		X(P)	
- Induced primary system LOCAs	X	X(P)	
- In-vessel recovery		X(P)	X
- In-vessel reactivity excursion		X(P)	X

(P) - Probabilistic sensitivities performed.

Table 4.7.6-2

List of Sensitivity Items

Sensitivity Item	Required by GL88-20 or NUREG-1335	Deemed Useful In Nine Mile Point 2 IPE Response	Proposed Cases for Accident Management Investigations
Ex-vessel Core Melt Progression			
- Debris Temperature	X		
- Amount of debris discharged from vessel			
- DW sump coolability		X	
- Coolability with water present	X	X	
- Effective DW floor area		X	
- Pool Bypass			
-- Vacuum Breaker		X	
-- Downcomers		N/A	
-- Other		N/A	
- Quenching Model in Pool (MKII)	X	N/A	
- DCH		X(P)	
- Amount of Material			
-- Retained in drywell			
-- Retained in pedestal			

Table 4.7.6-2

List of Sensitivity Items

Sensitivity Item	Required by GL88-20 or NUREG-1335	Deemed Useful In Nine Mile Point 2 IPE Response	Proposed Cases for Accident Management Investigations
Containment Failure			
- Size	X	X	
- Location	X	X	
- Pressure (Ultimate Capability)			
- Temperature			
- ATWS induced dynamic containment failure mode			
- Containment venting			X
- Pool bypass			
- Aerosol Plugging		X	
- Direct contact of debris	X		
- Pressure Rise	X		
Reactor Building Effectiveness	X	X	
- Hydrogen Burn			
- Circulation Established			
- Direct Release			

Table 4.7.6-2

List of Sensitivity Items

Sensitivity Item	Required by GL88-20 or NUREG-1335	Deemed Useful In Nine Mile Point 2 IPE Response	Proposed Cases for Accident Management Investigations
Critical Safety Functions			
- Reactivity Control			
- Pressure Control			X
- High Pressure Makeup			X
- Depressurization			X
- Low Pressure Makeup			X
- Containment Heat Removal	X	X	X
- Containment Temperature Control			X
- Containment Pressure Control			X
- Combustible Gas Control			X
- Containment Water Level Control			X
- Containment Flooding			X
- Drywell Spray Use		X	X

Table 4.7.6-2

List of Sensitivity Items

Sensitivity Item	Required by GL88-20 or NUREG-1335	Deemed Useful In Nine Mile Point 2 IPE Response	Proposed Cases for Accident Management Investigations
Other Actions			
- Accident Management Actions			X
- Disregard DWSI Curve			X
- Containment Flood Always by Procedure			
- Containment Flood With no RPV Vent		X	
- Containment Flood Only Late in Sequence		X	X
- Fill DW with water (MKI)		X	
- Vent to 0 psig			X
- Vent to control 40-60 psig			
- Vent to control 60-90 psig			X

Table 4.7.6-3

DEFINITION OF REACTOR BUILDING SENSITIVITY CASES

Cases	Description
Base Cases LII1D11 + LII1D12	MG Set Sprays initiate on Fourth Floor Node Temperature > 165°F (i.e., shortly after containment failure) SGTS node flow isolates on node temperature > 310°F Torus Room Sprays do not operate
"A"	MG Set Sprays do not operate SGTS node flow isolates on node temperature 310°F Torus Room Sprays do not operate
"B"	MG Set Sprays do not operate SGTS flow remains on for duration Torus Room Sprays do not operate
"C"	MG Set Sprays initiate on Fourth Floor Node Temperature > 165°F (i.e., shortly after containment failure) SGTS flow remains on for duration Torus Room Sprays do not operate
"D"	MG Set Sprays initiate on Fourth Floor Node Temperature > 165°F (i.e., shortly after containment failure) SGTS node flow isolates on node temperature > 310°F Torus Room Sprays initiate on Torus Room Node Temperature > 286°F (at about 11.8 hours in case LII1D11D)
"E"	Same as base cases, but assumed SGTS DF=1000 instead of 100
"F"	MG Set Sprays initiate on Fourth Floor Node Temperature > 165°F (i.e., shortly after containment failure SGTS system unavailable Torus Room Sprays do not operate

Table 4.7.6-4

REACTOR BUILDING EFFECTIVENESS FOR
DRYWELL SHELL FAILURE TO TORUS ROOM

Assumptions	Expected Reactor Building DFs
Sprays Effective/SGTS Isolates due to High Temperatures	> 25
Sprays Effective/SGTS Effective for Duration of Sequence	> 25
Sprays Ineffective/SGTS Available	> 10*
Sprays Ineffective/SGTS Unavailable	> 10

* > 25 for small breaks to reactor building

Table 4.7.6-5

CsI SCRUBBING DATA -- HOT POOL*

Run Number	Injected Gas Steam Mass Fraction	Injected Gas Mass Flow Rate (g/s)	Injected Aerosol Mass Flow Rate (mg/min)	Injected Gas Temperature (°K)	$T_{tank\ gas}$ (°K)		$T_{tank\ wall}$ (°K)		T_{water} (°K)		DF
					Before	After	Before	After	Before	After	
101	0	2.05	38.5	500	375	368	372	369	372	368	7.7
102	0	2.14	44.3	445	375	369	373	370	373	369	8.6
103	0	2.07	27.5	470	373	369	373	370	373	369	5.0
104	0.47	2.06	13.4	524	374	370	373	371	372	370	13
105	0.83	1.83	33.2	410	375	374	373	372	373	371	22
106	0	2.05	64.8	472	372	370	372	370	373	369	12
107	0.98	12.51	28.1	416	373	374	373	372	373	370	31
108	0	2.05	75.5	437	373	369	373	369	373	369	6.2

* All runs performed at a submergence of 1.65 m.

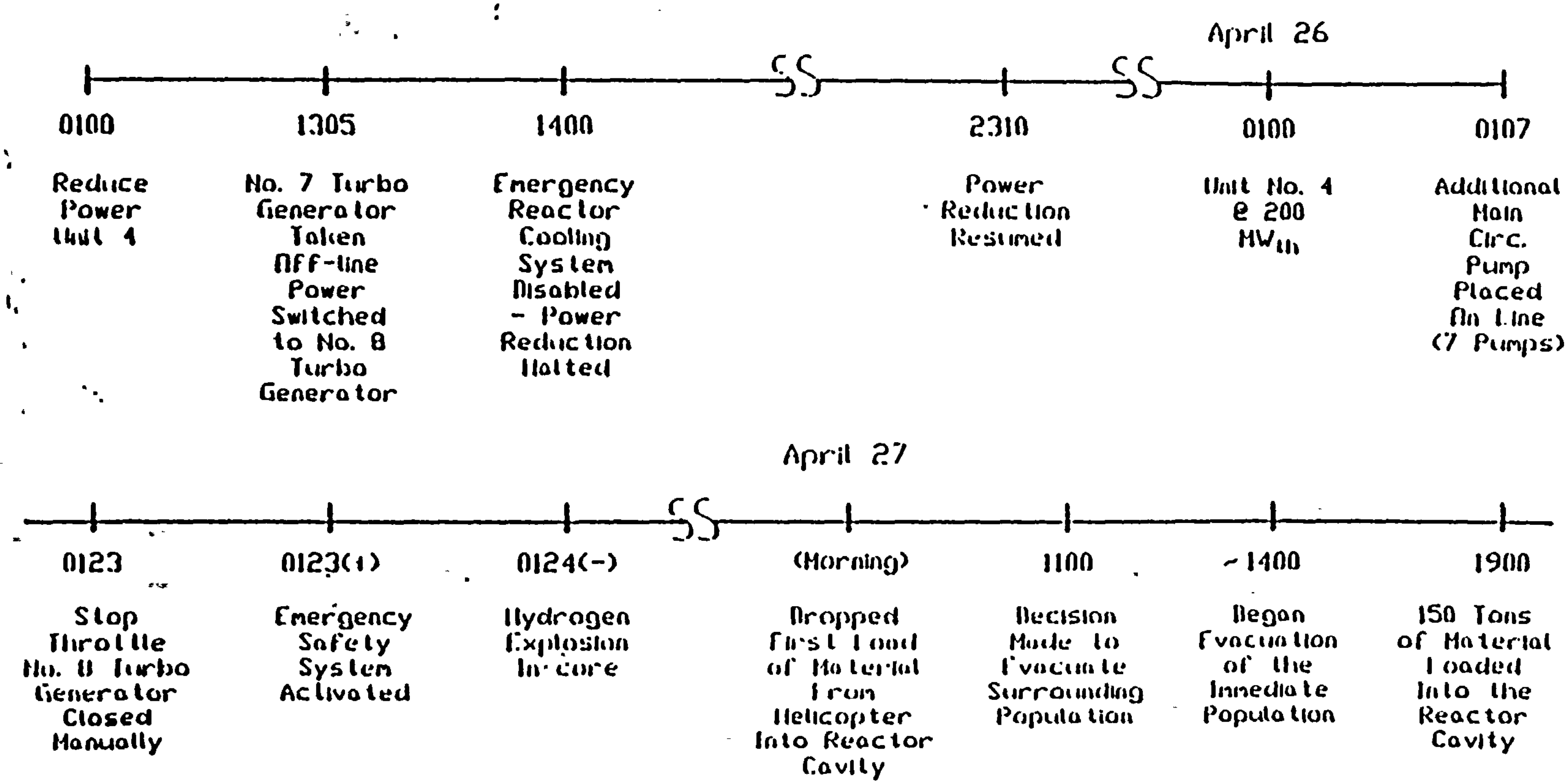
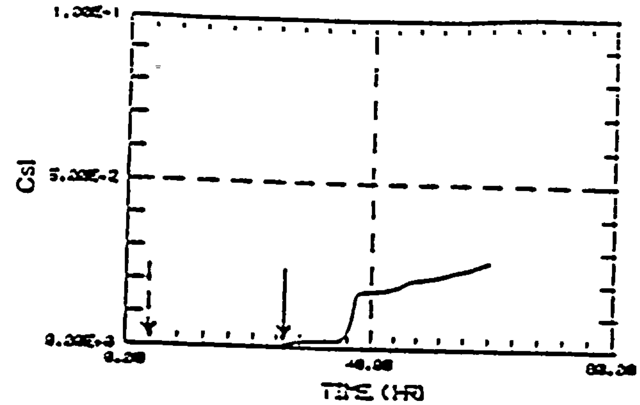


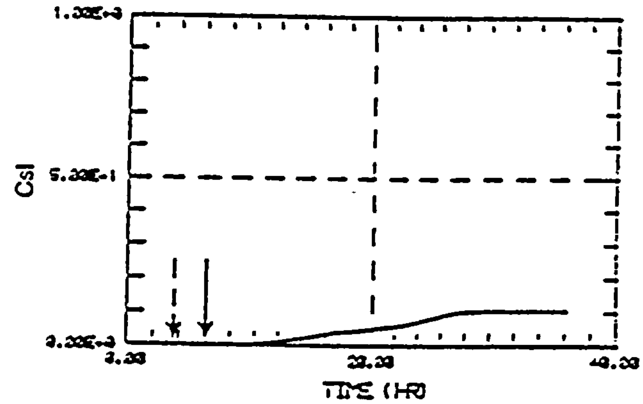
Figure 4.7-1

Chernobyl Unit 4 Accident Timeline

Class IA:

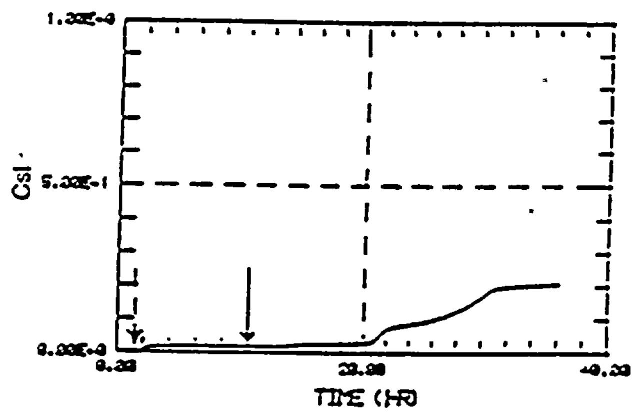


SEO:

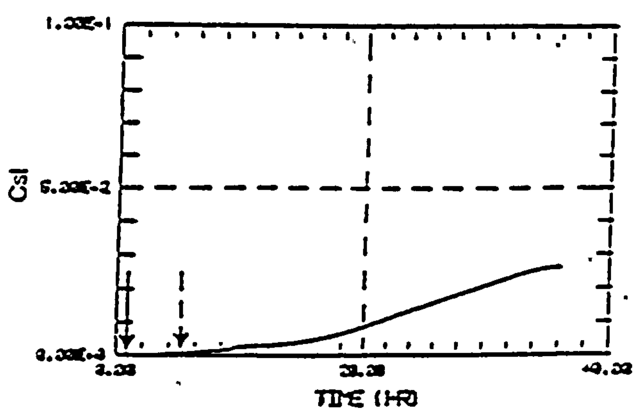


Class IIIC: LOCA

(Vent at RPV Failure)



ATWS: Class IV

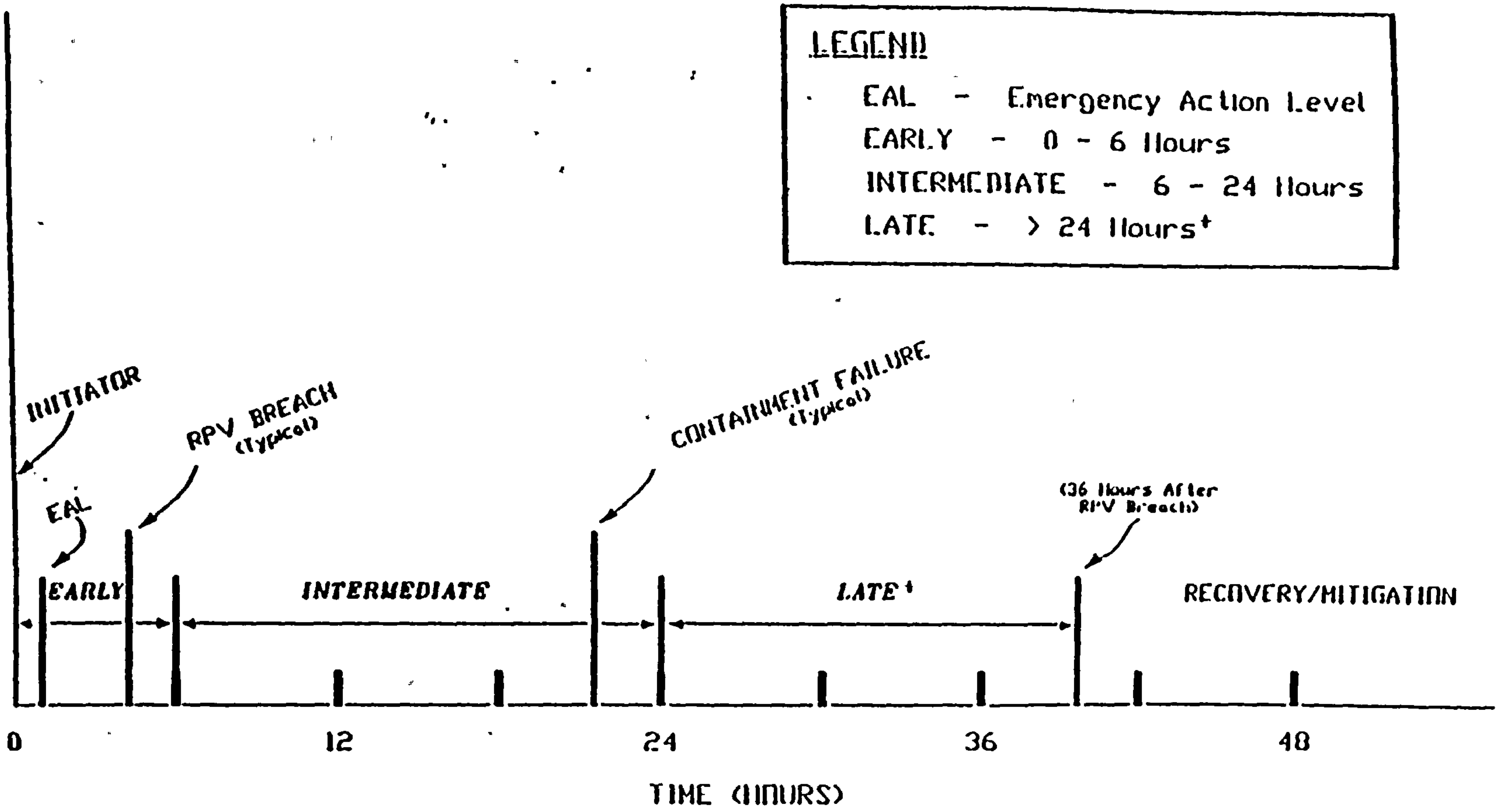


LEGEND:

RPV Failure

Containment Failure

Figure 4.7.2-2 Comparison of the CsI Release Profile as a Function of Timing Following RPV and Containment Failure



+ For purposes of radionuclide release calculations, this includes 36 hours past RPV breach by molten debris.

Figure 4.7-3

Example of the Definition of Timing Used in the Radionuclide Release Categorization (Class I, II, and III)

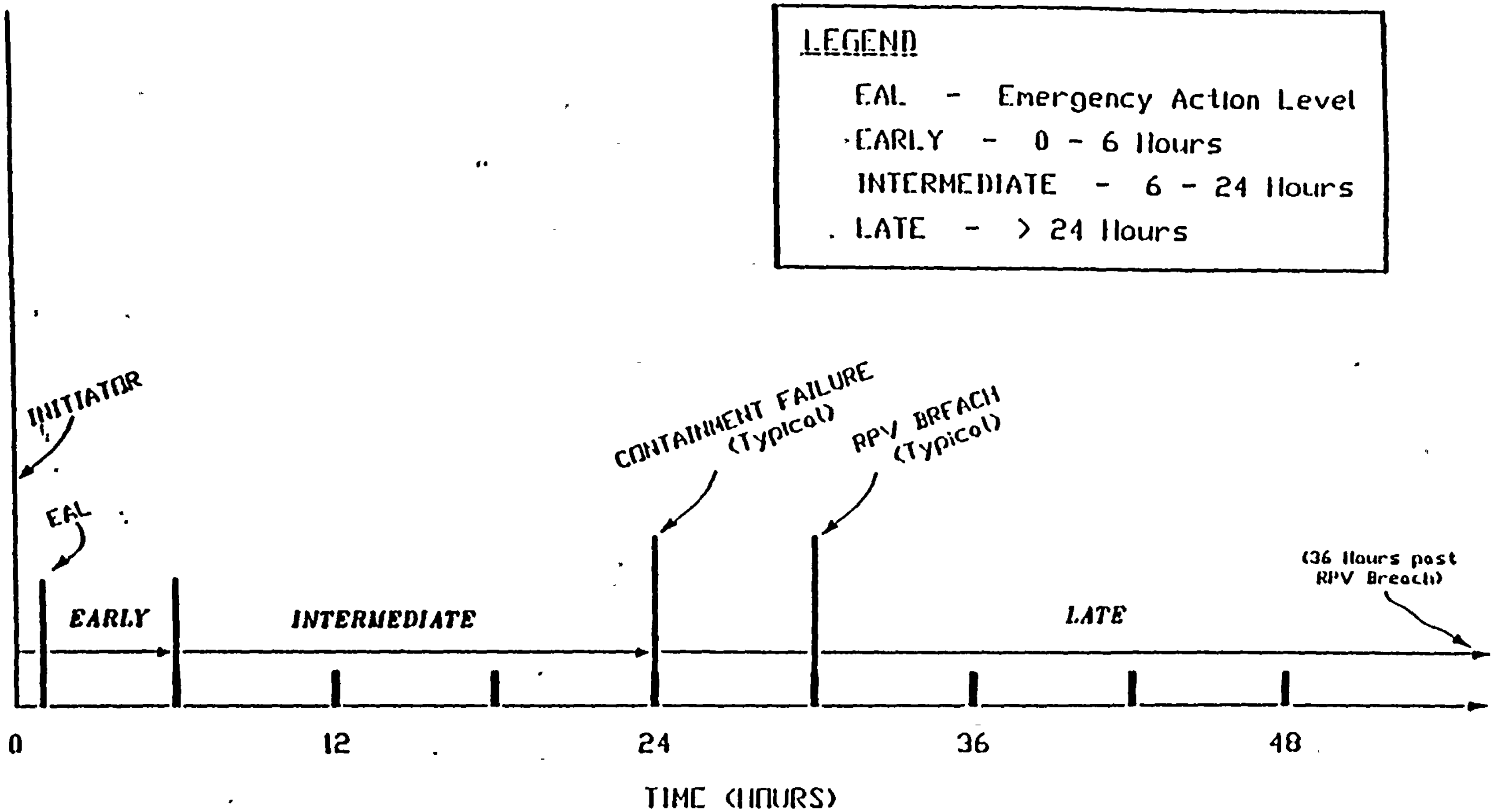
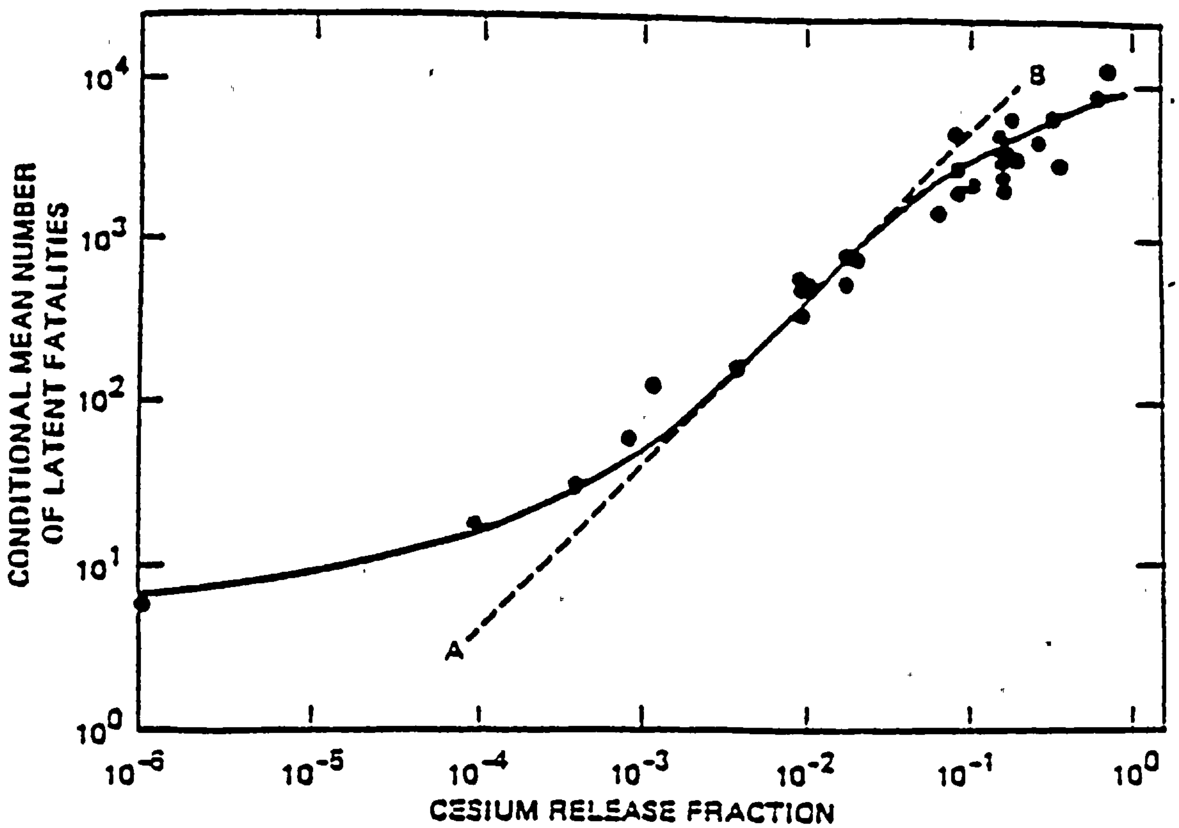


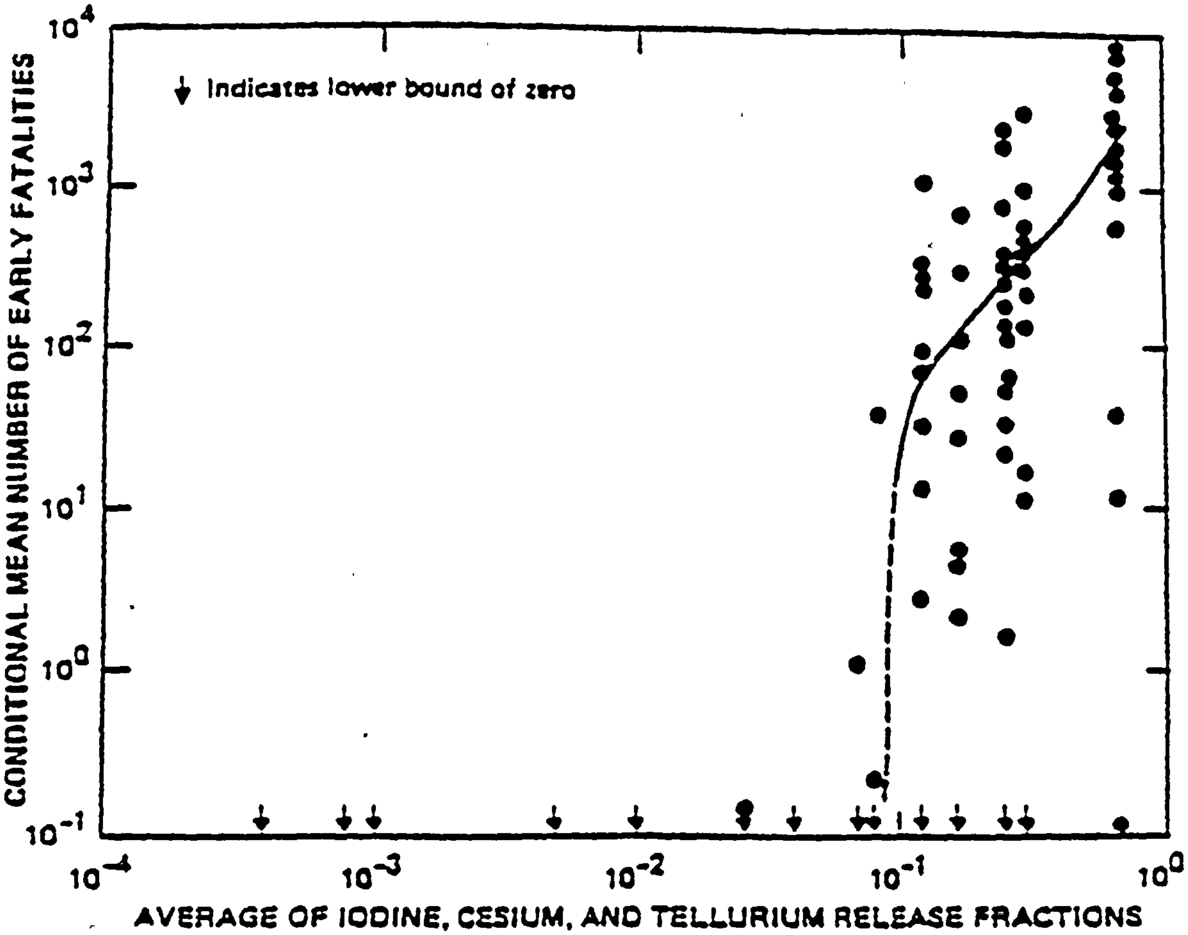
Figure 4.7-4

Example of the Definition of Timing Used in the Radionuclide Release Categorization (Class II)



Sensitivity of Number of Latent Cancer Fatalities to the Cesium Release Fraction

Figure 4.7-5



Sensitivity of Mean Number Early Fatalities
to Source Term Magnitude

Figure 4.7-6

4.8 Accident Management Insights: Qualitative Summary of Impact on Radionuclide Release Due to Functional Top Events

This section summarizes the insights and generalizations based on the Nine Mile Point 2 specific containment evaluation using MAAP deterministic calculations. These insights also include consideration of features that impact accident management response under postulated core melt progression.

Specifically, this section discusses those accident management actions and insights that may be useful to consider as Niagara Mohawk continues to develop its accident management programs. These actions relate specifically to those low frequency accidents for which core damage has occurred and continued core melt progression may lead to containment challenge or radionuclide release.

Some of the special features that are incorporated into the discussion of the accident management actions are the following:

- Drywell sunken pedestal under the RPV
- Downcomers located in the drywell floor directly below the RPV
- Containment hardpiped vent capability
- Automated SLC for redundant reactivity control.
- Thick and substantial pedestal walls that support the RPV.
- Containment flooding systems and procedures
- Communication of drywell sprays with the pedestal.

As part of the deterministic assessment of containment response, MAAP calculations of a wide spectrum of postulated accident scenarios have been performed to support the containment event tree evaluation. These postulated accident sequence calculations are prescribed based on past experience with other BWR PRAs and the knowledge of dominant phenomenological and system effects on containment response or radionuclide release severity or timing.

The key phenomena and functional effects for which insights are derived and are discussed include the following:

- RPV Depressurization

- Water Injection
  - To RPV
  - To Containment
  - Flooding
  - Debris Cooling

- Energetic failures

- Combustible gas control

- Containment Heat Removal/Containment Pressure Control
  - RHR
  - Venting

- Suppression Pool Bypass

- Containment Flooding
  - Drywell Vent
  - RPV Vent

- Reactor Building Effectiveness

- Containment Failure Mode
  - Location
    - -- Drywell
    - -- Wetwell
  - Size

- Reactivity Control

- Containment Isolation

4.8.1 RPV Depressurization

The ability to depressurize the RPV during core melt progression, i.e., prior to RPV breach by molten debris can be a major influence on:

1) the determination of the accident sequence timing;
2) phenomena that occur; and,
3) the challenge applied to the containment.

These effects are reflected in the Level 2 model in three principal ways:

- RPV Phenomena:

  The sequence can be completely altered by modifying the conditional probability of subsequent event tree nodes dependent on the pressure status of the RPV. For example, use of low pressure injection systems.

  Depressurization is an important action to take because it affects the following important issues and phenomena:

  - Steam explosion likelihood based on initial conditions

  - Vapor suppression success criteria

  - Recovery of RPV injection capability (e.g., LPCI) to terminate core melt progression in-vessel.

The probablistic modification of the sequences due to the pressure status of the RPV is treated in the CET rules. This is based on previous separate effects evaluations as a function of RPV pressure, such as the possibility of steam explosions or vapor suppression bypass.

- <u>Containment Interactions</u>:

  The challenge to containment can cause actions or failures not otherwise implemented.

  The challenges to containment as a result of core melt progression without depressurization are investigated in MAAP cases:

  - With vapor suppression
  - With degraded vapor suppression
  - With inadequate vapor suppression
  - With RPV depressurization successful.

  The results of the investigation confirm that containment challenges are minimized when the RPV is depressurized during the core melt progression. Accident management actions could be implemented to further reinforce that this is the case.

  Only the low probability of in-vessel steam explosion is increased slightly by depressurization during core melt progression, other effects of depressurization are beneficial for accident mitigation.

- <u>Releases</u>:

  Radionuclide release end states may be altered as a result of the status of RPV depressurization because the depressurized state results in the lowest energy state for the primary system. This energy state precludes an RPV depressurization due to a core melt progression induced RPV breach from causing containment failure. Alternatively, the RPV blowdown at RPV failure could lead to containment failure and energetic radionuclide release. MAAP calculations indicate this to result in a very high radionuclide release potential.

  Therefore, the direct impact of the depressurization node on radionuclide release and timing is most pronounced if RPV depressurization occurs simultaneously with an RPV breach and through the RPV breach with an open containment or is the cause of a containment drywell failure.

Therefore, the accident management action is that RPV depressurization is to be encouraged to minimize radionuclide releases for all of the above reasons.

4.8.2 Water Injection to the RPV or Containment

One of the most important mitigating system actions that can be implemented as part of accident mitigation is the injection of water into containment or into the RPV. The adequacy of this injection for minimizing radionuclide releases can be evaluated for different combinations of other functional and phenomenological events.

Different methods of water injection are available from a wide variety of sources. These water sources are clearly defined in the EOPs and in training. They include the following:

- FW/Condensate
- HPCS
- RCIC
- RHR/LPCI
- CS
- SW Crosstie
- CRD
- Smaller capacity systems.

This section discusses the following related issues for injection:

- RPV injection (prioritization of injection methods)
- RPV injection versus drywell sprays
- Debris cooling
- Injection during vent
- Injection to prevent containment failure.

RPV Injection (Core Spray Versus LPCI Injection to RPV)

Water injection to the RPV has a number of beneficial features which include:

- Cooling residual core material in the bottom head.
- Cooling fuel rods that remain intact in the core region
- Cooling by steaming the fission products that are plated out on RPV internal surfaces (dryer/separator).

The ability to provide all of these cooling benefits varies with the water source, i.e., the injection source and its flow rate. This preference of injection to the RPV can also be compared with injection to the containment via drywell sprays or containment flood.

The following RPV injection sources are considered viable and have the following benefits or disadvantages (if a choice needs to be made among injection sources):

Core Spray (HPCS and LPCS): This appears to be the most desirable[1] injection source for severe accident mitigation and minimizing radionuclide releases. The core spray systems have a relatively high flow rates and produce a spray pattern that is most conducive to cooling material in the RPV given that the RPV bottom head has been breached during core melt progression.

Water will also run out the bottom head of the vessel through the breach and fall on the debris on the drywell floor or through the pedestal downcomers to debris. This results in the potential to also cool the debris on the drywell floor, in the downcomers, or in the suppression pool.

The use of CS in lieu of LPCI appears to be most useful in response to degraded core conditions in which water level cannot be restored to within the fuel zone instrument range regardless of flow rate. This conclusion is based on MAAP calculations which indicate the potential for increased drywell temperatures for LPCI injection cases when debris remains in the RPV. Such conditions could lead to premature failure of containment or release of excess radionuclides due to revaporization. The prioritization of injection systems may be an action that could be included in future accident management development.

LPCI: This is the next most desirable injection source. It has all the advantages cited for Core Spray except that it is injected in the downcomer region and results in the possibility of being short circuited past the core region and directly out the bottom head breach. This has the possibility of allowing revaporization in the extremely long term as one of its disadvantages. This could be most important in containment flood scenarios when RPV venting is directed by the EOPs and where the revaporization source term may escape directly through the RPV vent. Therefore; an AM strategy might be to prioritize the use of core spray systems over LPCI given that bottom head breach may have occurred or if RPV water level cannot be restored and a choice of injection system must be made.

SW Crosstie: This has identical attributes to LPCI except a continuous supply of cool water is available; LPCI recirculates water from the suppression pool. This results in containment flooding.

CRD: This water source is desirable, but is of limited flow rate. In addition, after RPV breach the flow path may not allow delivery to the RPV or to the drywell. This system is not considered here as an effective mitigating system for severe accidents that have progressed outside the RPV, AM actions should not <u>rely</u> on its use. Nevertheless, using CRD <u>can</u> be of benefit.

---

[1] Note that conflicting conclusions may be reached using current T&H codes for sequences in which there is a failure to scram and the RPV is intact. Note also there may be a small window of cases in which current procedures would discourage the use of LPCS and HPCS if water level were <u>perceived</u> to be hung up above -14 inches and a reactivity control problem in progress.

MAAP has limited modeling capability to examine the subtle differences in various in-vessel injection methods, even after RPV failure. Therefore, the above qualitative assessments are based on engineering judgement using MAAP guidance and inferences from the MAAP cases where appropriate, along with insights from separate effects analyses where available.

RPV Injection Versus Drywell Sprays

Core spray (HPCS and LPCS) injection to the RPV has all the advantages that were discussed above in the RPV injection discussion.

Drywell sprays have many of the advantages of the core spray injection method including maintaining low drywell temperatures; however, the use of drywell sprays would be marginally effective in cooling debris that was retained in the vessel or in the sunken pedestal, but may be of vital importance to cool debris outside the sunken pedestal.

The drywell sprays have the ability to cool most if not all drywell surfaces effectively such that the containment boundary can be protected from high temperature induced failures. This is particularly true of core debris that may be entrained during RPV blowdown and resides in the drywell. However, the RPV internal surfaces may remain hot if no injection to the RPV can be restored. Such high temperatures may challenge containment penetrations that go to the RPV or result in radionuclide revaporization.

In addition, because of the NMP-2 sunken pedestal design the drywell spray flow does not have easy access to the pedestal region. Low flows would likely reach the sunken pedestal floor and therefore be of assistance in cooling any debris that remained on the pedestal floor. However, flow from the RPV breach would likely be more directly available.

In addition to core spray, drywell spray offers an additional alternative to the control of drywell temperature to avoid premature containment failure. Therefore, an accident management strategy may seek the initiation of drywell sprays - this may require the relaxation of the restrictions on the use of the drywell sprays in the Drywell Spray Initiation (DWSI) curve of the EOPs.

In addition, if the operators were able to enter into containment flooding, then RPV venting would be directed and the use of drywell sprays during RPV venting may also have a minimal beneficial effect on reducing the release directly from the RPV.

LPCI, SW Crosstie, and CRD discussions are similar to those above.

Debris Cooling (see also above discussion on RPV injection versus Drywell Sprays)

Coolant injection to the drywell via either the RPV or the drywell sprays has the benefit of providing debris cooling. This cooling will have the following beneficial effects:

- Limit the temperature increase in the drywell during the core melt progression

- Limit the non-condensible gas generation in the containment and, thereby, prevent reaching the critical containment failure pressure and temperature.

Because of certain phenomena identified on some experiments (i.e., debris entrainment during high pressure blowdown), a fraction of molten debris may be swept out of the pedestal and into the drywell. Because of the special NMP2 sunken pedestal configuration, no RPV injection water will be available to cool the debris in the drywell outside the pedestal. Therefore, the drywell sprays are potentially important in accident management to ensure that the drywell is not challenged by temperature or temperature and pressure conditions created by entrained debris that resides outside the pedestal. Entrainment is currently included in the MAAP model.

Injection During Containment Failure or Vent

As part of the evaluation of containment failure or vent and the impact on releases, it is important to assess the volume or flow rate of makeup to the debris during the melt progression. The greater the cooling (from any source), the lower the radionuclide releases. Because of the potential for suppression pool bypass during venting, sprays (drywell or wetwell) during venting may be important.

Injection To Prevent Containment Failure

One of the principal benefits of water injection is that when coupled with containment pressure control (nearly equivalent to containment heat removal) that successful water injection can prevent containment failure. This can prevent the following postulated containment failure modes:

- High temperature and pressure induced containment failure
- Large non-condensable gas generation.

The most effective of the methods of water injection are:

- Core Spray
- Drywell Sprays

Therefore, actions to ensure that these systems can be aligned and operated under severe accident conditions would appear to have high priority.

4.8.3 Combustible Gas Control

The EOPs specify the containment is part of accident mitigation to minimize the possibility of a combustible gas mixture. The vent process should be undertaken when symptoms are met.

For the postulated severe accidents considered in the Nine Mile Point 2 PRA, these conditions would include cases for which the containment is deinerted and radionuclides have been released from the fuel.

For such cases the combustible gas venting has the following influences:

- The release of radionuclides begins early in the sequence.
- The vent path is assumed to be the wetwell vent. (This may be non-conservative)
- The suppression pool is subcooled for the majority of the release and, therefore, radionuclide releases are found to be substantially reduced by suppression pool scrubbing. The pool is considered subcooled because of the availability of RHR pool cooling for most of the combustible gas venting cases. If drywell venting is undertaken, then the releases increase in to the high category.

Insights for combustible gas venting are the following:

- Rapid venting of large quantities of the containment atmosphere should be undertaken if H_2 and O_2 fractions are above 6%. Priority should be set high to get the containment vented.
- Use of the wetwell vent path is preferred.

4.8.4 Energetic Phenomena

There are a large number of energetic phenomena that have been postulated during core melt progression accidents. These phenomena include, among others:

- Steam explosions
- Direct containment heating
- Hydrogen detonation.

While the MAAP code can provide insights regarding sufficiency of conditions to cause these phenomena, it is not believed that MAAP provides a means to calculate the results of such phenomena. Therefore, consistent with past PRA work (e.g., WASH-1400, NUREG-1150, Limerick PRA, Shoreham PRA), when these phenomena are probabilistically and deterministically considered to occur (i.e., see CET end states for CZ and CE failed), they are assigned a high release category. The release category timing is still determined by the sequence specific core melt progression timing. No Nine Mile Point 2 specific MAAP calculations are performed to further refine this binning.

The primary method of accident management to mitigate these events is RPV depressurization and use of drywell sprays.

4.8.5 Containment Wetwell Venting or Wetwell Breach With Continued Injection

If suppression pool cooling is not available and the main condenser is not used, containment venting provides a useful method of containment pressure control and containment heat removal.

Despite the core melt progression outside the vessel if continued coolant injection to the containment can be maintained containment venting is used, then radionuclide releases can be minimized if much of this discussion also applies to situations in which the wetwell airspace is the discharge pathway for any radionuclide releases.

Different cases of containment venting are found to result in substantially different estimates of the radionuclide release:

	Maintain Injection to RPV or Containment	Wetwell Vented	Suppression Pool Bypass
Case 1	YES	YES	NO
Case 2	YES	YES	YES
Case 3	NO	YES	NO
Case 4	NO	YES	YES

The results of the MAAP calculations indicate that:

1) Case 1: The radionuclide releases are very low (LL) for the case in which water injection, wetwell venting, and no suppression pool bypass are present.

2) Case 2: Releases are approximately 10-100 times larger for the case in which suppression pool bypass is present.

3) Case 3: Releases are approximately 100-1,000 times larger than case 1.

4) Case 4: Releases are more than 1000 times larger than Case 1.

The purpose of venting is to avoid containment over-pressurization and protect the containment structural integrity. Functionally, this can be accomplished by using the system designed for containment venting or combustible gas control. Additionally, the containment can be successfully vented through a breach in the structure.

The impact of venting on a potential environmental source term is dependent primarily on two factors:

1) Timing for establishing the vent pathway; and

2) The suppression pool effectiveness, i.e., the availability of a pathway that routes the radionuclides through the suppression pool, the suppression pool condition (e.g., temperature), potential suppression pool bypass.

These conditions are further discussed below.

4.8.5.1 Timing of Radionuclide Release

The timing of containment venting can influence the radionuclide release by:

- Releasing material early in an accident scenario
- Minimizing any blowdown flow from the containment during the vent process.

The effects on radionuclide release magnitude and the effect assigned to timing of release can both be very important from an accident management standpoint.

MAAP calculations indicate that the special NMP2 containment design has a feature that can affect the timing of radionuclide release via containment vent. This feature is the in-pedestal downcomers below the RPV which results in debris entering the suppression pool at RPV breach. Thus, core melt progression causes quenching of debris in the suppression pool. The EOP gives direction to vent the containment, and such action would lead to the release of radionuclides via the vent. MAAP results have indicated that containment venting (which is specified prior to 45 psig) would occur significantly before predicted containment failure (which is predicted at approximately 140 psig). There could be tens of hours difference in the release time.

The factors that affect the decision to vent containment can be categorized as follows (also refer to Table 4.8.5-1):

- Containment Structural Capability
    - Static
    - Dynamic
- Containment Vent Valve Capability
- System Operability
    - SRV
    - EQ in Drywell
    - RCIC
    - MSIV Open
    - LPCI
- Radionuclide Activity
- Plant Availability

- Containment Leakage/Reactor Building Environment
- Deinerting
- Depletion of Non-Condensibles
- Loss of NPSH

Early venting is currently included in those cases where we can vent and inject (e.g. flood cases). The only time when we do not include an early vent consequence is for cases with venting operable, but no outside injection.

Table 4.8.5-1

Summary of Plant Considerations for Optimization of Containment Vent Pressure

Parameter for Consideration	Less Than Design (40 PSIA)	Design (60 PSIA)	Design + Margin	Design + .5 X Design (90 PSIA)	Ultimate (120 PSIA)
Containment Structural Capability[1]					
- Static	OK	OK	OK	OK	OK
- Dynamic	OK	OK	OK	Marginal	Not Acceptable
Containment Vent Valve Capability[1]	OK	OK	OK	OK	Not Acceptable
System Operability					
- SRV	OK	OK	OK	Marginal	Not Acceptable
- EQ in Drywell	OK	L	L	L	L
- RCIC	OK	L	L	L	L
- MSIV Open	OK	L	L	L	L
- LPCI	OK	OK	L	L	L
Radionuclide Activity[1]	H	H	H	H/OK	OK
Plant Availability	OK	OK	OK	L	L
Containment Leakage/ Reactor Building Environment	OK	OK	OK	L	L
Deinerting	H	H	H	H	H
Depletion of Non-Condensibles	H	H	H	H	H
Loss of NPSH	H	H	H	H	H

OK: Means that through the consideration of this parameter alone it appears acceptable to vent at pressure up to and including the value cited.

L: Means that a tentative conclusion regarding this parameter alone has been reached which would indicate it prudent to vent at lower pressures.

H: Means that a tentative conclusion regarding this parameter alone has been reached which would indicate it prudent to vent at higher pressures.

[1] Heavily Weighted

Two examples of the extremely unlikely failure sequences for which containment venting can result in substantial releases are the following:

- Sequence #1

  - Large LOCA
  - No injection available
  - Vacuum breakers stuck open during the process
  - Pedestal drain plate melt-through by debris
  - Venting initiated at PCPL

  The resulting radionuclide release as calculated by MAAP is (1) initiated early; and, (2) the consequential release is high.

  This impact could be substantially mitigated if no venting occurs and the containment is allowed to absorb a substantial amount of the severe accident energy. If this occurs, the release time can be delayed substantially from several hours to nearly a day depending on whether systems can be recovered during the core melt progression.

- Sequence #2

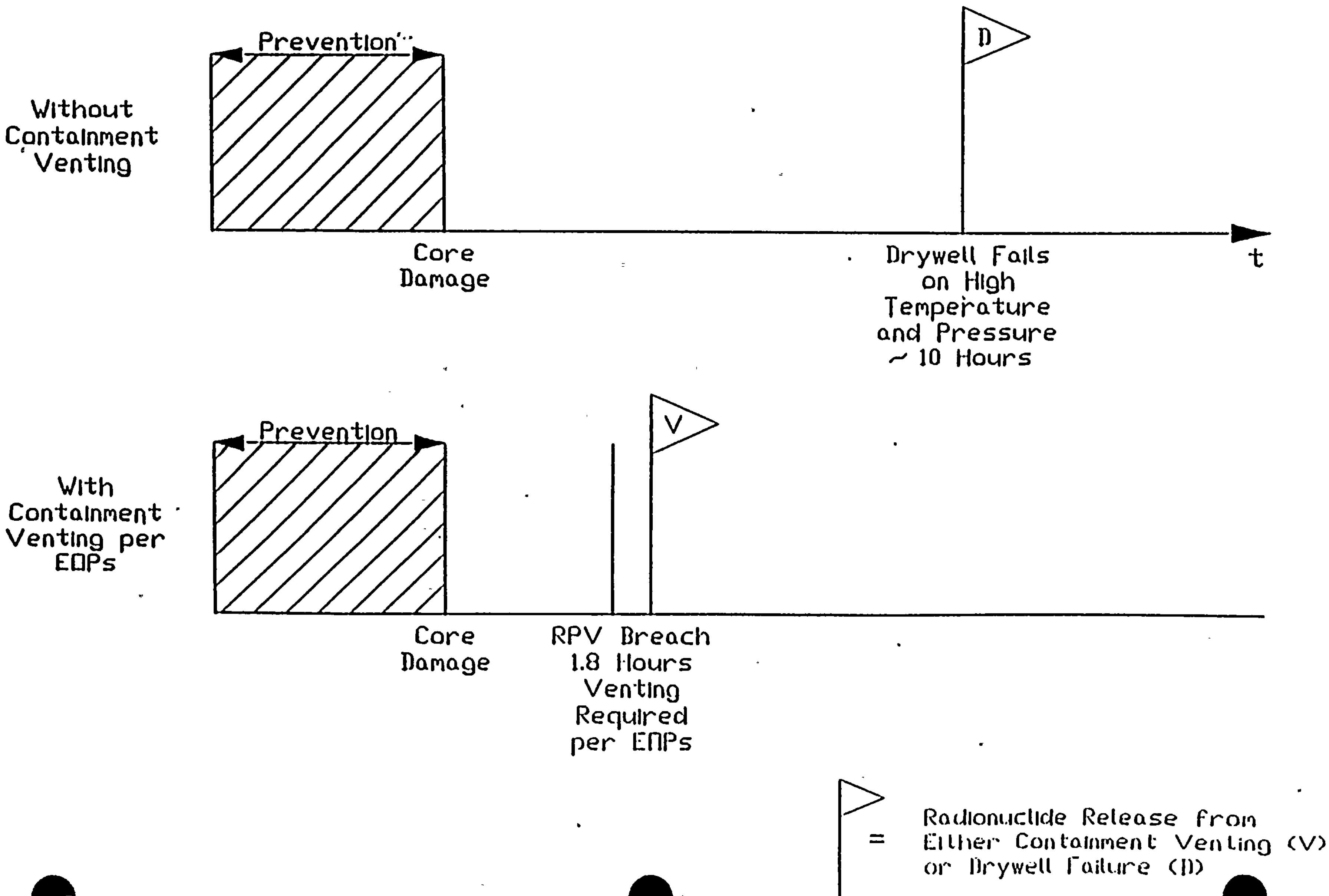
  There are accident sequences that can involve severe core damage and result in coincident, transient high containment pressure near, or slightly above the containment design pressure. If the containment design pressure is chosen as the vent pressure, then containment venting will be initiated automatically or by operator action at the time of the initial pressure rise.

  Figure 4.8.5-1 shows a comparison of two postulated accident scenarios - one with containment venting implemented and one without containment venting implemented.

  The containment venting is seen to result in a very early radionuclide release (~ 1.8 hours) compared with the no venting case (~ 10 hours). This means that the vent strategy, as implemented in the EOPs, results in premature releases to the environment relative to the use of a delayed containment vent to 90 or 100 psia.

Figure 4.8.5-1

Comparison of the Timing of Radionuclide Release with EOP Required Vent Action Versus no Vent Action



Observations Regarding Containment Venting for Overpressure Protection

Observations concerning containment venting as specified in the NMP2 EOPs include the following:

- Venting is a strategy to provide a defense-in-depth approach to accident management using existing BWR configurations and equipment. As such, it provides a graded response to accidents.

- Venting can be a useful part of an integrated strategy to prevent accident types that challenge the capability of the containment by overpressurization. This would allow the operating staff to maintain coolant injection makeup by avoiding coolant injection failures that may be induced by an uncontrolled containment failure at an undefined location.

- Venting can be a useful part of strategy for severe accident mitigation to preserve the multiple containment functions.

- Competing phenomena that could reduce the positive safety influence of venting have been identified, but their contribution appears to be substantially less than the potential positive aspects for most sequences.

- Delays in venting may be justified to beyond the plant design pressure when containment temperatures are relatively low.

- Another insight derived from this evaluation is that containment failure is predicted to occur due to the high combination of high pressure and high temperature - a condition for which venting has not been designed to combat. Specifically, the containment failure is predicted to occur below current EOP vent pressure when temperatures in the drywell exceed 650°F. Therefore, the second accident management insight is that for high drywell temperatures the vent pressure may need to be reduced to prevent uncontrolled releases due to drywell failures.

- Situations that direct the containment to be vented as a means to prevent containment failure by overpressurization are conditions far beyond the plant's design basis and are restricted to very specific and low frequency circumstances. Venting actions are among the last resort actions, i.e., taken only after the primary methods of performing the protective functions associated with containment heat removal and pressure control have failed. Venting is intended to prevent more serious or uncontrolled failures that are judged likely to occur should venting activities not be performed.

- Venting permits a gradual reduction of a containment pressure rise as opposed to a potentially uncontrolled depressurization associated with containment rupture.

- Venting from the wetwell maximizes suppression pool scrubbing essentially limiting releases to noble gases when the suppression pool remains in the pathway for fission products.

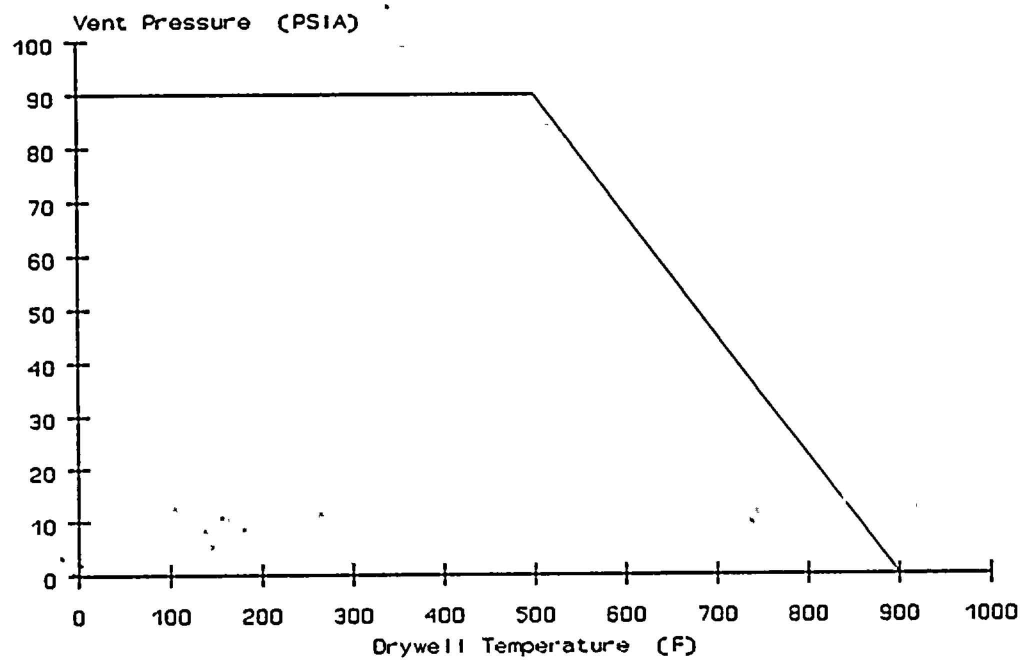
Conclusion

The conclusion is that optimization of containment venting under severe accident conditions would require some analyses and rethinking of the current EPG instructions, particularly related to:

- The timing of radionuclide releases from the drywell or during bypass sequences.

- Temperature dependent failures of containment.

A proposed temperature dependent venting pressure is as follows:



This use of containment venting (when containment pressure is only 45 psig) results in a radionuclide release which is substantially earlier than if containment vent is delayed until containment pressure is 60 to 70 psig. Accident Management actions may consider delaying venting or recalculating the curve that requires venting at 45 psig. This Accident Management insight is considered very important in avoiding early radionuclide release under severe accident conditions at NMP2.

4.8.5.2 Suppression Pool Scrubbing of Effluent

Suppression pool water temperature (i.e., degree of subcooling) may affect the characteristic of the pool to retain aerosols during the vent. It is postulated that as the bulk temperature of the pool approaches saturation temperature, the effective DF of the pool decreases. In fact, current MAAP calculations [Rev. 7.01] indicate that upon reaching saturation temperature, the pool DF becomes unity (i.e., all aerosol radionuclides pass through the pool).

As alluded to in the discussion above, it appears that the suppression pool DF with respect to the retention of radionuclides decreases as its temperature increases. General Electric in NEDO-25420 dated June 1981 found the following:

> Suppression pool decontamination factors appropriate for use in BWR risk assessments are presented in Table 4.8.5-2. Based on the data presented and the expected BWR transport conditions, suppression pool decontamination factors of at least 10^2 for elemental iodine and particulates, and 10^3 for cesium iodide are justifiable for subcooled pools. For saturated pools, decontamination factors of at least 30 for elemental iodine and 10^2 for particulates and cesium iodide are currently justifiable.
>
> Natural processes such as the agglomeration of solids, plateout, deposition, washout, etc., also pay an important role in limiting the quantity of fission products available for leakage to the environment. The overall attenuation factor applicable to BWR degraded core accident scenarios includes both the effects of pool scrubbing and of such natural removal processes that will occur in the various volumes of the BWR process systems and its multiple containment system.

This effect is further discussed in the following section concerning the effect on source term with the availability of RHR system heat exchangers in the suppression pool cooling mode.

4.8.5.3 Suppression Pool Bypass

Suppression pool can be bypassed if molten core debris causes the in-pedestal downcomers to fail on high temperature. Under conditions with suppression pool bypass, there are accident management actions that can be taken to minimize the impact of such bypass on fission product release. These AM actions include:

- Use of suppression pool sprays to scrub fission products from the wetwell atmosphere
- Use of drywell sprays to scrub fission products from containment.

Currently, NMP-2 EOPs indicate that sprays are useful, but are not used unless - adequate core cooling is assured. AM may be focused in a different direction if RPV breach has occurred and venting is imminent. Therefore, given these phenomena and the special containment design of NMP2, it is judged useful to consider a more liberal use of sprays prior to initiating venting in order to minimize radionuclide release.

Generally, assuming that the suppression pool provides a means to scrub aerosols, maintaining a positive differential pressure between the drywell and the wetwell has a beneficial effect in reducing the magnitude of a source term by directing a portion of the radionuclides into the pool. If the pool becomes bypassed due to downcomer failure or vacuum breaker failure (i.e., radionuclides can be transported from either the RPV or drywell to the wetwell air space), then scrubbing of aerosols cannot be less effective. Consequently, a release following containment failure in the wetwell or containment wetwell venting will contain a larger fraction of radionuclide aerosols and particulate.

4.8.6 Suppression Pool Cooling Mode of RHR

The RHR system heat exchangers are placed on-line by the operator to maintain the containment within the pressure and temperature boundary conditions prescribed in the EOPs. Containment heat removal affects both the magnitude and timing of a potential source term release to the environment. Timing (and magnitude) of an impending release can be extended by controlling containment pressure below the point at which structural failure occurs. The magnitude of the release can be affected by two phenomena:

1) maintaining the suppression pool temperature less than the NPSH and vortex limits of ECCSs taking suction off the pool; and

2) controlling suppression pool water temperature below saturation.

Each of these phenomena are briefly discussed below.

4.8.6.1 Timing Of Containment Failure

It is postulated that under certain conditions the timing of containment failure after the development of a source term inside the containment can affect the magnitude of any subsequent release to the environment. This effect is further discussed in the section addressing the timing of vent initiation.

4.8.6.2 Controlling Suppression Pool Water Temperature

Maintaining suppression pool water temperature as low as possible extends the time before containment pressure and temperature challenges may occur. Plant specific MAAP calculations have shown that the availability of water to molten fuel debris (given that in-vessel recovery was unsuccessful) reduces the magnitude of the pressure and temperature challenge to the containment, as well as the source term that accumulates inside the drywell air space. Therefore, the use of RHR suppression pool cooling which is already part of the EOP directions is confirmed to be an important accident management action.

4.8.6.2.1 Suppression Pool Cooling

There is a special NMP2 pedestal design with limited communication between the in-pedestal water and the ex-pedestal water in the wetwell. Under postulated severe accidents, the molten debris from the RPV is predicted to deposit within the pedestal. This means that the pedestal water will heat up relatively rapidly and the communication with the ex-pedestal water will be limited. Therefore, suppression pool cooling may have limited effectiveness in maintaining a subcooled pool and thereby preventing a long term containment pressurization challenge. The best return paths to use during suppression pool cooling under severe accidents with the RPV breached is to take suction from the suppression pool and return flow to the RPV. An alternative may be to return to the drywell sprays, but this is considered desirable only if drywell temperature and/or pressure are rising. The return flow to the RPV would fall to the pedestal and thereby provide the necessary circulation between in-pedestal and ex-pedestal water to prevent an overpressure challenge, i.e., operation of LPCI plus an in-line heat exchanger would prevent containment failure. This can be an important AM insight. This is in general already directed by the EOPs.

4.8.6.2.2 Scrubbing (DF)

The suppression pool water temperature also affects the potential for "scrubbing" aerosols if the source term is directed through the pool before egress from the containment. MAAP calculations [Rev. 7.01] indicate that there is a correlation (i.e., and inverse relationship) between the water temperature and the effective pool DF. Presently, these analyses indicate that the suppression pool is ineffective for scrubbing radionuclide aerosols once the water temperature achieves its saturation temperature. This assumption does not appear consistent with NEDO-24250 and recent experiments. In fact, due to bubble dynamics in a saturated pool, the DF may actually increase at saturation. It is the judgement of the IPE team that a DF of at least 10 for a saturated pool is reasonable. The MAAP results will be adjusted accordingly based on this judgement. Of course, this adjustment will only apply to the pool scrubbing portion of the source term for events with late drywell failure and no suppression pool bypass.

4.8.7 Water Injection Post Containment Failure (MU)

In the plant specific Nine Mile Point 2 MAAP calculations [Rev. 7.01], it appears that the impact of continued water injection to either the RPV or drywell after containment failure (or venting) can be considered to have two possible effects:

- For cases with drywell head failures it is found that the reduction in total CsI radionuclide release to the environment is reduced at most by approximately a factor of 2.
- For cases in which the containment failure is in the wetwell the availability of MU or post containment water injection to the RPV or drywell will result in minimizing the releases.

4.8.8 Reactivity Control

Because failure to scram sequences from the Level 1 analysis are postulated to challenge containment integrity early in a sequence, measures to control reactivity are extremely important for <u>both</u>:

- Core damage prevention
- Avoidance of early containment failures.
- Automated SLC has reduced the ATWS contribution to core damage frequency and containment challenges. Thus, this has reduced the frequency of early containment failures.

The use of the automatic SLC system is a significant difference from other BWRs. This feature allows automatic logic to initiate the SLC system and avoids the need for operation intervention for successful boron injection.

In addition, the latest BWROG Rev. 4 EPGs have been implemented for level/power control to further enhance response to ATWs.

4.8.9 Containment Isolation

Because drywell failures to isolate have the potential of leading to high early releases from containment, containment isolation is considered an important containment feature. NMP2 has a special feature of the containment isolation system that involves the use of AC powered MOVs on the drywell equipment and floor drains. The use of AC powered MOVs as isolation valves means that under a station blackout conditions, that these valves would not be able to close automatically.

The NMP2 EOPs have clear direction early in the response that the operating staff is to initiate isolation if it has not occurred. Further, specific directions are provided by auxiliary procedure EOP-6 Attachment 1.

In this auxiliary procedure, Section 1.2g provides explicit direction to the operating staff to ensure that the drywell equipment and drywell floor drain isolation valves are closed.

These valves are containment isolation valves but are not highlighted in any manner within the procedure.

It may be useful during implementation of the SBO specific procedure, in operator training or in future AM guidance to emphasize the need to go locally to these MOVs to close the valves and provide containment drywell isolation.

4.8.10 Containment Flooding

Given the current state of knowledge regarding severe accident phenomenology, the Fermi EOPs have established a near optimum balance among the contingency procedures which the operator can implement.

The EOPs generally define one of the following: the optimum procedural path; a procedural pathway that is close to optimum; or, a pathway for which insufficient analytical (and experimental) information is available to more precisely define the optimum pathway. Changes in the current understanding of severe accident phenomena or in the philosophy of dealing with severe accidents may impact some of the EOP steps and contingency actions.

A possible improved response for current containment flood types of sequences for which the EPG directions result in the highest potential consequences at the earliest time, is to provide the operators guidance on protecting containment and cooling debris using methods that do not require opening the RPV vent and avoid using the DW vent unless no other alternative exists. Alternate actions have been shown to produce substantially lower releases and much longer <u>times to failure</u> if no action is taken, i.e., even no action is better than action directed by the EPGs.

4.8.11 Containment Injection At High Containment Pressure

There is a set of very low frequency severe accidents for which the containment may be at elevated pressures (i.e., above the containment vent pressure) and for which the EOPs would dictate that injection to the RPV be terminated when containment pressure exceeds MPCWLL.

Because such a strategy can lead directly to core damage and a subsequent containment challenge it is judged prudent to not terminate water injection to the containment under any circumstances for which core degradation may be aggravated by the termination of injection. This can be addressed in Accident Management investigations.

4.8.12 Summary of Accident Management Insights

Table 4.8.13-1 summarizes the insights from the Level 2 portion of the NMP2 IPE.

Table 4.8.13-1
SUMMARY TABLE OF ACCIDENT MANAGEMENT INSIGHTS

Special Containment Failure	Impact		Accident Management Strategy Insight
	Positive	Negative	
Ex-Vessel Recovery	X		The use of CS or DW spray in lieu of LPCI appears to be most useful in response to degraded core conditions. This conclusion is based on MAAP calculations which indicate the potential for increased drywell temperatures for LPCI injection cases when debris remains in the RPV. Such conditions could lead to premature failure of containment. The prioritization of injection systems may be an action that could be included in future accident management development.
Drywell Sunken Pedestal	X		The containment drywell sunken pedestal under the RPV results in directing virtually all of the molten debris to be initially collected in the pedestal region and prevents migration of substantial quantities of debris outside the pedestal to attack the liner. Therefore, the AMS insight is to ensure cooling injection can be effectively provided to inside the pedestal region for downcomer and debris cooling.
Downcomers Located in the Pedestal Floor	X		The sunken pedestal has downcomers in the floor that result in virtually all the debris in the pedestal to go to the suppression pool. Within the suppression pool there will be a continuous steaming source from the basement floor resulting in wetwell pressurization relative to the drywell.
Containment Hardpiped Vent	X		Make maximum use of the hardpiped vent system to control containment pressure by optimizing: - when the vent is used - how long it is opened - how many cycles are required
Containment Hardpiped Vent			Containment venting per the EOPs provides the expected benefit in prevention of core damage and additional benefit in the containment overpressure protection under severe accidents. Wetwell venting has profoundly greater potential for radionuclide scrubbing than if the drywell vent is used. There is essentially no DF for drywell venting. (See also discussion regarding downcomers in pedestal). Therefore, drywell venting should be a last resort vent method.
Phenomenological Effects	X		DCH, steam explosions, vapor suppression failure, etc. are found to have the potential to lead to relatively high releases, but the net effect is a relatively small impact on risk (i.e., frequency of large release). These phenomenological effects are being pursued by NRC research on a generic basis. Current EOP directions are considered optimized to combat these phenomena. No additional plant specific actions are recommended at this time. Continuing effort as part of accident management implementation will be to follow these research efforts.
Ex-Vessel Recovery	X		The use of CS or DW spray in lieu of LPCI appears to be most useful in response to degraded core conditions. This conclusion is based on MAAP calculations which indicate the potential for increased drywell temperatures for LPCI injection cases when debris remains in the RPV. Such conditions could lead to premature failure of containment. The prioritization of injection systems may be an action that could be included in future accident management development.
Drywell Floor and Equipment Drain Isolation Valve Power Supplies		X	Minimize the time that the valves are open. Provide a procedure to rapidly manually close the valves given a loss of power to the isolation MOVs.
Automated SLC	X		Ensure that its use is optimized (not defeated by manual intervention). Ensure that the EOPs/AM strategies are optimized to take advantage of the automatic feature.
Thick Pedestal Walls	X		RPV support does not appear to be a problem during severe accident core melt progression accidents.

Table 4.8.13-1
SUMMARY TABLE OF ACCIDENT MANAGEMENT INSIGHTS

Special Containment Failure	Impact		Accident Management Strategy Insight
	Positive	Negative	
Containment Hardpiped Vent			The timing of radionuclide release can be substantially affected by containment venting. In fact, releases may occur through venting when the release may otherwise be prevented. For other cases, the release may occur 20 hours earlier than otherwise releasing noble gases and scrubbed release if a DF can be assured. MAAP calculations indicate that the special NMP2 containment design has a feature that can affect the timing of radionuclide release via containment vent. This feature is the in-pedestal downcomers below the RPV which results in debris entering the suppression pool at RPV breach. Thus, core melt progression causes quenching of debris in the suppression pool, pressurization of containment with steam, and the EOP direction to vent the containment. Such action would lead to the release of radionuclides via the vent. This use of containment venting (when containment pressure is only 45 psig) results in a radionuclide release which is substantially early than if containment vent is delayed until containment pressure is 60 to 70 psig. The Accident Management insight is considered very important in avoiding earlier radionuclide release under severe accident conditions at NMP2.
Relax DWSI Curve		X	In addition to core spray, drywell spray offers an additional alternative to the control of drywell temperature to avoid premature containment failure. This can be very important if debris is entrained to the ex-pedestal drywell or if venting is required. Therefore, an accident management strategy may seek the initiation of drywell sprays, this may require the relaxation of the restrictions on the use of the drywell sprays in the Drywell Spray Initiation (DWSI) curve of the EOPs. Discussion: Training should cover the use of containment sprays for multiple purposes during severe accident conditions. Not only do sprays provide the ability to maintain containment temperature and pressure within limits but they can provide a means of debris cooling when vessel injection is not possible. Sprays also permit scrubbing of the containment atmosphere to reduce potential fission product releases from containment. Emphasis should be placed on initiation of sprays on the limits specified in the EOPs to assure reliability of debris cooling during conditions in which core damage and vessel penetration may have occurred. Evaluation of the effects of spray operation on vessel injection rate may indicate that adequate vessel injection and spray operation can occur simultaneously. A second aspect of the spray initiation is to: • Anticipate vessel failure and err on the side of drywell spraying, even if current containment pressures and temperatures are not above initiation limits. This may require symptoms that would allow initiation under such postulated circumstances. • Provide a more flexible DWSI curve that would allow drywell spray initiation when containment pressures and temperatures are quite adverse, i.e., exactly when the sprays may do the most good.
Containment Bypass (Downcomers In-Pedestal)		X	The probability that the downcomers inside the pedestal may fail and create a suppression pool bypass leads to AM considerations to establish drywell or wetwell sprays to scrub fission products from the atmosphere. (See also DWSI curve discussion).
Containment Failure Modes			The evaluated containment failure modes indicate that large structural failures are expected to dominate the containment failure modes at low internal temperature (i.e., less than 400°F). Above these temperatures, leakage is expected to be the failure mode over the intermediate temperatures. Then at very high temperatures, creep rupture and other failure modes are expected to lead to larger failure modes in the drywell.

Table 4.8.13-1
SUMMARY TABLE OF ACCIDENT MANAGEMENT INSIGHTS

Special Containment Failure	Impact		Accident Management Strategy Insight
	Positive	Negative	
Containment Injection at High Containment Pressure		X	There is a set of very low frequency severe accidents for which the containment may be at elevated pressures (i.e., above the containment vent pressure) and for which the EOPs would dictate that injection to the RPV (from external water sources) be terminated when containment pressure exceeds MPCWLL. Because such a strategy can lead directly to core damage and a subsequent containment challenge, it is judged prudent to not terminate water injection to the containment under any circumstances for which core degradation may be aggravated by the termination of injection. This can be addressed in Accident Management Investigations.
Containment Configuration of Suppression Pool			For accident sequences in which the pressurization source is in the wetwell airspace,(e.g., any ex-vessel core melt progression) <u>with</u> the containment failure in the wetwell airspace, the wetwell pressurization source prevents material from the drywell migrating to the wetwell airspace even if the downcomers are failed. This keeps the releases from the wetwell lower for such cases compared with other plants (given suppression pool bypass). <u>OR</u>, stated in another way, Suppression pool bypass may be more likely at NMP2, but its consequences are mitigated against by the nature of the core melt progression phenomena.
Containment Isolation (general)	X		Containment isolation is highly reliable. The operating experience of NMP2 and the other inerted BWRs indicates that containment isolation is reliable. Because drywell failures to isolate have the potential of leading to high early releases from containment, containment isolation is considered an important containment feature. NMP2 has a special feature of the containment isolation system that involves the use of AC powered MOVs on the drywell equipment and floor drains. The use of AC powered MOVs as isolation valves means that under station blackout conditions, these valves would not be able to close automatically. The NMP2 EOPs have clear direction early in the response that the operating staff is to initiate isolation if it has not occurred. Further, specific directions are provided by auxiliary procedure EOP-6, Attachment 1. In this auxiliary procedure, Section 1.2g provides explicit direction to the operating staff to ensure that the drywell equipment and drywell floor drain isolation valves are closed. These valves are containment isolation valves, but are not highlighted in any manner within the procedure. It may be useful during implementation of the SBO specific procedure, in operator training or in future AM guidance, to emphasize the need to locally close these MOVs to provide containment drywell isolation.

Table 4.8.13-1
SUMMARY TABLE OF ACCIDENT MANAGEMENT INSIGHTS

Special Containment Failure	Impact		Accident Management Strategy Insight
	Positive	Negative	
Containment Isolation (SBO)		X	Because drywell failures to isolate have the potential of leading to high early releases from containment, containment isolation is considered an important containment feature. NMP2 has a special feature of the containment isolation system that involves the use of AC powered MOVs on the drywell equipment and floor drains. The use of AC powered MOVs as isolation valves means that under a station blackout conditions, that these valves would not be able to close automatically. The NMP2 EOPs have clear direction early in the response that the operating staff is to initiate isolation if it has not already occurred. Further, specific directions are provided by auxiliary procedure EOP-6, Attachment 1. In this auxiliary procedure, Section 1.2g provides explicit direction to the operating staff to ensure that the drywell equipment and drywell floor drain isolation valves are closed. These valves are containment isolation valves but are not highlighted in any manner within the procedure. It may be useful during implementation of the SBO specific procedure, in operating training or in future AM guidance to emphasize the need to locally close these MOVs to provide containment drywell isolation.
Containment Flood Capability and Procedures	X	X	The maximum primary containment water level limit (MPCWLL) has some important effects on the PRA evaluation. The specific effects discussed here are related to the impact on the frequency and magnitude of radionuclide releases. Background The EPGs were developed primarily to prevent and mitigate events prior to core damage. No calculations were performed for severe accidents to demonstrate that radionuclide releases were minimized by the actions directed in the EPGs. In fact, minimizing radionuclide releases is not even an objective of the EPGs except as it derives from preventing core damage or containment failure. Discussion One of the areas of the EPGs which may have a strong impact on the IPE assessment relates to the MPCWLL treatment. The MPCWLL implications for the IPE are discussed as follows: • The MPCWLL has associated with it directions to terminate all injection from external water sources if MPCWLL is exceeded. • When such injection is terminated, the EOPs may, therefore, have eliminated the only injection source capable of preventing core damage. This can then lead to: a. High RPV pressure, if containment pressure exceeds the point at which the compressed gas system can maintain SRVs open (e.g., 100 psig inside containment). b. Core damage, if all injection to the RPV is terminated or becomes unavailable. • The results of such an event have been shown (using integrated severe accident codes such as MAAP) to lead to the failure of the RPV and containment simultaneously. This is calculated to cause the energetic release of radionuclides at the time when the highest flow rates are present and result in sweeping fission products to the environment.

Table 4.8.13-1
SUMMARY TABLE OF ACCIDENT MANAGEMENT INSIGHTS

Special Containment Failure	Impact		Accident Management Strategy Insight
	Positive	Negative	
(continued)			Therefore, the plant specific implementation of the EPGs will lead to higher releases at earlier times than previously calculated by NRC, IDCOR, or individual utilities for certain low frequency events modeled in the PRA. The Level 2 evaluation explicitly calculated these effects on radionuclide release such that the magnitude of the impact was shown to be potentially high. It may be useful to establish liaison with the EOP writers for purposes of examining alternatives in the accident management phase of IPE closure to the termination of injection. <u>Interface with RPV Venting in the Case of Containment Flooding</u> The containment flooding contingency procedure requires the operating crew to vent the RPV to allow the ingress of water into the RPV as the containment water level rises. Instead of venting the RPV to the condenser, this action could be accomplished by preferentially opening the RPV head vent using the same procedure as described for RPV-ED. There are potential benefits in using this path: Using the drywell vent only. Additionally, if the breach in the RPV were submerged, the drywell vent could be as effective in controlling containment conditions and minimizing the release source term as a wetwell vent with the suppression pool intact. Given the current state of knowledge regarding severe accident phenomenology, the NMP2 EOPs have established a near optimum balance among the contingency procedures which the operator can implement. The EOPs generally define one of the following: the optimum procedural path; a procedural pathway that is close to optimum; or, a pathway for which insufficient analytical (and experimental) information is available to more precisely define the optimum pathway. Changes in the current understanding of severe accident phenomena or in the philosophy of dealing with severe accidents may impact some of the EOP steps and contingency actions. A possible improved response for current containment flood types of sequences for which the EPG directions result in the highest potential consequences at the earliest time, is to provide the operators guidance on protecting containment and cooling debris using methods that do not require opening the RPV vent and avoid using the DW vent unless no other alternative exists. Alternate actions have been shown to produce substantially lower releases and much longer <u>times to failure</u> if no action is taken, i.e, even no action is better than action directed by the EPGs.

Table 4.8.5-2

Minimum Supportable and Potentially Attainable Suppression Pool Decontamination Factors For Iodine and Particulates

Transport Pathway and Associated Event(s)	Minimum Supportable DFs		Potentially Attainable DFs[3]
	Subcooled Pool[1]	Saturated Pool[2]	
Reactor pressure vessel to pool via safety relief valve and quencher (Transients)	10^3 CsI, I^-, HI 10^2 particulates 10^2 I_2	10^2 particulates[4] 30 I_2	10^5-10^6 CsI, I^-, III 10^3-10^6 particulates 10^2-10^3 I_2
Reactor pressure vessel to pool via vents (Transients following RPV depressurization, or LOCA post blowdown period)	10^3 CsI, I^-, HI 10^2 particulates 10^2 I_2	10^2 particulates[4] 30 I_2	10^4-10^6 CsI, I^-, III 10^3-10^4 particulates 10^2-10^3 I_2
Aerosol Transport to Pool via Vents (Core-Concrete Vaporization Release)	10^2 particulates 10^2 I_2	10^2 particulates[4] 30 I_2	10^3-10^6 particulates 10^2-10^3 I_2

Notes:

(1) During these conditions, complete condensation is expected when the pool is subcooled.

(2) A subcooled pool is at a temperature below the saturation temperature corresponding to the pressure in the containment, while in a saturated pool steady state boiling "steaming" is occurring.

(3) Potentially attainable by further testing (saturated-subcooled pools).

(4) Includes CsI

4.9 Sensitivity Evaluation for NMP2

As part of the containment evaluation there are phenomenological and probabilistic (e.g., system reliability, operator action) issues that can have a large impact on the course of the events or the radionuclide release magnitude and timing. Both types of issues become candidates for sensitivity analysis. The NMP2 CET provides a structure to perform sensitivity studies on issues for which a large uncertainty may exist.

Probabilistic and phenomenological uncertainties are addressed in this section to ensure that appropriate accident management actions which may be strongly influenced by these uncertainties are identified. These uncertainties are, in general, addressed quantitatively using either ranges of probabilities or deterministic computer calculations to simulate alternative modeling assumptions. In a few selected cases, the uncertainties are discussed qualitatively to ascertain their impact on accident management actions.

This section includes the following information:

- Approaches to sensitivity (Section 4.9.1)

- Overview of the issues for which an uncertainty or sensitivity study is desirable (Section 4.9.2)

- Deterministic sensitivity studies (Section 4.9.3)

- Probabilistic sensitivity studies (Section 4.9.4)

Table 4.9-1 (Table A.5 from NUREG-1335) identified parameters for which sensitivity cases may be performed. From these parameters, the phenomena and assumptions used in MAAP that are subject to the most uncertainty for NMP2 have been investigated. Most of the resources of the NMP2 IPE back-end analysis effort is devoted to treating uncertainties which could directly influence accident management strategies, in general, and containment failure time, in particular. Stated more narrowly from the standpoint of accident management, the principal goal in sensitivity studies should be to identify and understand physical phenomena which put a premium on specific operator actions. In addition, accident management actions have been identified to be effective for controlling or preventing postulated phenomena under certain accident sequence conditions or assuming certain modeling conditions. These phenomena may not be physically possible or may behave differently than the modeling assumptions. It is judged that it may also be prudent to investigate the impact of the accident management actions over a range of postulated physical models on phenomenological assumptions.

Fewer resources should be devoted to phenomena which are to varying degrees: (1) generic rather than plant-specific; (2) being studied elsewhere on a generic basis; or, (3) which do not impact accident management strategies directly even though they could affect the source term from a given sequence. For such phenomena, only best-estimate treatments are recommended here.

The results of the sensitivity cases are described in Sections 4.9.3 and 4.9.4. The following section identifies possible approaches to performing the sensitivity analysis and identifies the method chosen for NMP2. In addition, Section 4.9.2 identifies the issues to be examined and the method used.

4.9.1 Sensitivity Approaches

The approaches for investigating key sensitivities can take on a wide spectrum of breadth and depth.

This section identifies three optional approaches that could be used to satisfy different objectives:

Resource Intensive Approach

The resource intensive approach is a comprehensive attempt to identify all parameters or modeling assumptions that have uncertainties of larger than an error factor of three and to include a sensitivity of varying these. In addition, the approach identifies coupled parameters that also need to be varied.

IPE Approach

The IPE approach is designed to satisfy the requirements of IPE Generic Letter 88-20 for the Level 2 portion of the IPE.

- Address the phenomenological issues posed by the NRC
  - Probabilistically, or
  - Deterministically
- Identify a limited sample of additional containment or plant specific issues that should be addressed.

The assessment of the NRC identified sensitivity items is performed probabilistically in some cases, and deterministically in other cases. These two approaches are used as follows:

- Probabilistic sensitivity assessment requires the analyst to use a range of point estimate values to describe the frequency of occurrence for system performance and operator recovery, and phenomena considered in the model. The resulting change in release frequency then reflects the model (i.e., the plant) sensitivity to these issues.
- Deterministic sensitivity assessment considers the extremes of the physical models used to represent the accident phenomena. The results of these deterministic calculations indicate the influence on the physical plant response associated with variations in the phenomenological modeling. By varying the model within postulated parameter ranges, the thermal hydraulic calculation provides insights into the magnitude of effect on key event timing, containment response, and radionuclide release.

Accident Management Sensitivity Approach

This group of sensitivities would be developed to support additional investigations to attempt to optimize accident management actions or hardware use that could be implemented as part of an accident management response to severe accidents.

Conclusion

As part of the IPE report, Niagara Mohawk has selected the IPE Approach. Therefore, this section will present the results of sensitivity assessments on a group of selected issues and plant specific features. Further sensitivities for accident management considerations will be performed in the future after the model has been accepted by the NRC for application to NMP2 specific investigation.

In performing the sensitivity evaluations it is important to note that a number of the items are coupled and the varying of individual parameters may not capture the complete impact of the coupling among variations in groups of parameters.

4.9.2 Sensitivity Runs Overview

To ensure that a broad scope of possible severe accident progressions is considered in the NMP2 IPE, several sensitivity analyses were performed using the MAAP code. Fifty-four MAAP cases were selected to evaluate the key functional events for mitigating radioactive releases associated with severe accidents at the NMP2 plant. This set of MAAP calculations represents a best estimate of how the plant will respond under severe accident conditions. However, it is recognized that considerable uncertainty exists in the modeling of the complex phenomena associated with such accidents. One should recognize that MAAP cannot and does not contain detailed models for all phenomena. Indeed, there are more mechanistic codes available such as CONTAIN and SCDAP/RELAP. These are generally used in a research setting and are not considered by us to be suitable for use in IPEs due to long run times and the much greater requirements they impose on the user for specialized knowledge of severe accident phenomena. An alternative code whose scope is similar to MAAP is MELCOR. However, less experience has been accumulated with the MELCOR code than with MAAP. Therefore, MAAP was chosen as the best available tool to perform the plant specific evaluation. However, in selected cases, MELCOR results on similar plants are also utilized.

Table 4.9-2 summarizes an extensive list of possible sensitivity calculations that could be performed to support a full PRA. Within Table 4.9-2 are identified those phenomena or items that:

a. Are required by GL-88-20 or NUREG-1335 to be addressed as part of the IPE.

b. Other items that are deemed sufficiently important to address.

c. Items that are deferred until the accident management program at NMP2 is fully implemented.

4.9.3 Deterministic Sensitivity Results

The MAAP model parameters generally represent inputs to phenomenological models in which significant uncertainties exist. Variations in the values of these parameters can be made to assess the impact of uncertainties in important physical models. The best estimate values used in the NMP2 IPE are provided in the NMP2 IPE MAAP Parameter File. These best estimate values were directly taken from the "Recommended Sensitivity Analyses for an Individual Plant Examination Using MAAP 3.0B" Gabor, Kenton and Associates, EPRI 1990.

The MAAP sensitivity cases address the uncertainties in the following phenomena:

- In-Vessel Hydrogen Generation
- Core Melt Progression
  - Amount of Debris Retained In-Vessel
  - RPV Breach Model
- Ex-Vessel Debris Coolability
- Debris Distribution In Containment
  - Effective Area of Drywell Floor
  - Amount of Material Retained in Pedestal
- Suppression Pool Mixing
- Pool Bypass
  - Pedestal Downcomer Failure
  - Vacuum Breaker Failure
- Containment Failure Size
- Containment Failure Location
- Drywell Equipment Mass
- Saturated Pool Decontamination Factor
- Containment Venting
- Reactor Building Modeling Assumptions
- Drywell Spray Usage

- Containment Flooding

- Coupling of DW Equipment Mass and Amount of Debris Retained in the RPV.

4.9.3.1 In-Vessel Hydrogen Generation (Core Blockage)

Hydrogen production during core damage events is found to be a small contributor to potential radionuclide release for inerted BWRs (see NUREG-1150). Nevertheless, there are short periods of time when BWRs (Mark I & II containment designs) may operate deinerted. For such deinerted conditions, it has been assumed in the NMP2 model that, without operator intervention, a core damage event would lead to containment energetic failure and a high radionuclide release. The NMP2 IPE has found that using this bounding assumption, the hydrogen generation and consequential containment failures do not control either the overall radionuclide release frequency or the "large" radionuclide release frequency.

Therefore, the NMP2 approach is considered to bound the effects of the hydrogen production model, and therefore, a detailed analysis is not considered necessary. Nevertheless, a brief discussion of the phenomenon is useful because it also influences other areas of core melt progression phenomena.

For loss of inventory accidents, as the core becomes uncovered, the fuel cladding will begin to oxidize producing hydrogen as a byproduct.

Eventually, melting and relocation of the core material will ensue with the potential for blocking steam flow and reducing additional cladding oxidation. Three options are available in MAAP for treating the resulting effects from melting and relocation of core material:

1. No Blockage Model: For this model, melting and relocation of the core will have negligible impact on the hydrogen generation, gas flows, and fission product release rates. (MAAP parameter FCRBLK = -1)

2. Local Blockage Model: For this model, melting and relocation of cladding away from the melting region will terminate oxidation of the Zircaloy in that node. Relocation will have a negligible impact on the gas flows or fission product release. (MAAP parameter FCRBLK = 0.0)

3. Channel Blockage Model: For this model, relocation of core material will seal off and pressurize the fuel channel. The increased pressure would force out the remaining water in the channel and terminate the flow of gasses. Without steam, oxidation of the cladding would stop. (MAAP parameter FCRBLK = 1.0). This was the IDCOR MAAP model and is no longer recommended.

Considerable uncertainty and controversy has historically been associated with trying to decide which of these core melt scenarios is the most realistic. For now, we note only that MAAP enables the user to select any of these scenarios.

While the actual amount of hydrogen generation may not always be of primary importance in inerted BWR containments, the increased core exit temperatures that typically occur with the no-blockage and local-blockage options will tend to result in RPV fission products being swept to the suppression pool early in a sequence. Those fission products would then not be available for revaporization later in the sequence; thus, smaller source terms will typically be predicted than if channel blockage is assumed. If very large amounts of hydrogen are produced, containment failure could occur even in inerted BWR containments due to the partial pressure of the hydrogen. This has not been observed in MAAP calculations performed to date.

The local blockage option (FCRBLK = 0.0) was selected for all of the NMP2 base cases. The "no blockage" model will typically result in the largest quantity of hydrogen generation in vessel. Case ID8LD was rerun with the two variations for the blockage model. The channel blockage model (FCRBLK=1.0) was used in case ID8LDC1, and the 'no blockage' model (FCRBLK=-1.0) was used in case ID8LDC2. Only minor variations in the results for these three MAAP runs are apparent as seen in Tables 4.7-1, 4.7-2, and 4.7-3. The principal differences are the slight variations in the amount of hydrogen generation for each case. The "no blockage" model (ID8LDC2) increased the hydrogen generation only slightly (< 1%), and the "channel blockage" option decreased the in-vessel hydrogen generation by about 10%. The selection of the local blockage option (FCRBLK=0.0) was made for all other NMP2 MAAP cases to provide a consistent best estimate base point for all further sensitivities.

4.9.3.2 Core Melt Progression

The core melt progression model can have strong influences on the containment response. The above sub-section has discussed the minimal impact of the hydrogen generation and core blockage model on the results for an inerted containment. This subsection discusses two potentially strong impacts on the containment response:

- The location of debris during the core melt progression
- RPV breach model.

4.9.3.2.1 Amount of Debris Retained In-Vessel

There are a number of influences due to the location of the fuel debris during core melt progression - characterized in this brief summary as "residual debris" in the RPV. Some of these influences include:

- Core-concrete interaction
- Quenching and pressurization of the wetwell
- Drywell temperature effects.

Core-concrete interaction: The amount of core-concrete interaction is a function of the amount of debris released and its temperature. Because of the NMP2 containment design, molten debris has direct access to the suppression pool via pedestal downcomers after RPV breach. Therefore, little core-concrete interaction on the drywell pedestal floor is predicted.

Quenching and pressurization of the wetwell: The larger the release of debris from the RPV to the suppression pool, the higher the pressurization of the containment. Therefore, the release of all debris from the RPV would result in the highest containment pressurization at the earliest time.

Drywell temperature: The retention of debris or fuel bundles in the RPV would result in a decay heat source being maintained in the drywell. This could result in high drywell temperatures and eventual drywell failure unless drywell sprays, core spray, or another effective cooling source becomes available. The quenching of debris in the suppression pool is, in general, adequate to provide sufficient drywell cooling to preclude temperature induced failures. The drywell temperature rise is a function of both the in-core model and the accident sequence.

The amount of core material remaining in the RPV is calculated by MAAP. As the core begins to melt, fuel relocates into lower regions of the core. This continues until the lowest core node in any radial region becomes completely molten at which time all molten core material exits the core region and moves into the lower plenum.

In past analyses, it has been observed that the amount of material molten at the onset of fuel movement into the lower head is strongly dependent on the amount of in-vessel Zircaloy oxidation. More oxidation tends to heat up the core and results in a larger mass of molten material moving out of the core region. Due to various modeling assumptions and a general lack of detail in representing core melt progression, within MAAP, there are two possibilities that can be evaluated. One involves all material exiting at vessel failure, and the other involves some of the core material remaining behind in the RPV.

It is important to understand the impact of each of these core melt progression scenarios. If all of the core material exits the RPV, it will provide more mass for core/concrete and core/water interactions. If core material remains behind in the vessel, it may contribute to late fission product revaporization and drywell heat-up. With the default MAAP input parameters, the majority of the cases predicted some material remaining in-vessel after vessel failure. This turned out to be the cases with the highest radionuclide release.

The MAAP parameter FMAXCP specifies the minimum amount of core material capable of supporting the remainder of the core. When the fractional amount of core material remaining in the vessel is less than FMAXCP, the remaining core material is forced out of the vessel and into the pedestal region. The default value of FMAXCP was set equal to 0.1. As stated previously, this leads to the majority of the cases with residual material in RPV core region following RPV bottom head breach.

To investigate the effects of the uncertainty in this phenomenon, cases ID2LD, ID8LD, and IA4LDNP were rerun with FMAXCP set equal to 0.8 (ID2LDCP, ID8LDCP, and IA4LDNCP). This larger value of FMAXCP forces all of the core material out of the vessel at vessel failure compared to some material remaining behind in the default cases. A summary of key results is shown in Table 4.9-3.

As a result of the reduction in the heat-up of the drywell, containment failure is delayed or does not occur, in each of the three sensitivity cases. The assumption that all of the core material left the vessel upon failure removed the dominant heat source from the vessel (i.e., the remaining core material), and minimized the threat to containment. For cases with injection (ID2LDCP), or wetwell venting (ID8LDCP) available, containment failure is not predicted to occur within 60 hours of sequence initiation. With no injection or containment heat removal available (IA4LDNCP), containment failure is delayed by only a few hours, but the fission product releases are reduced by a factor of ten.

From these cases, it is apparent that the assumption of allowing core material to remain in vessel can tend to maximize the threat to containment. It should be noted that in the MAAP analysis, no cooling of the core barrel occurs since LPCI is modeled as being injected directly into the lower head. In reality, the colder water flowing through the recirculation piping and down through the jet pumps will have a cooling effect on the core barrel and should minimize or eliminate this apparently large effect on containment failure. In addition, if core spray is available the RPV temperature and debris temperature would certainly be controlled. In any event, the MAAP base calculations (which allow debris to remain in the vessel) were done to indicate the potential for such a failure mode to exist and the desire in prescribing accident management action to avoid such situations.

A sensitivity was performed by ORNL in NUREG/CR-5565. The detailed analysis of BWR Mark II short-term station blackout sequence involved loss of all AC power with concurrent failure of the high pressure reactor vessel injection systems. Containment venting is assumed not to be implemented in these sequences because of support system or hardware unavailability even though the NMP2 EOPs would direct venting under these high containment pressure conditions.

Seven short-term station blackout sensitivity cases were conducted. The description of these seven scenarios is given in Table 4.9-4. The BWR-LTAS, BWRSAR, and MELCOR codes were employed to provide an analysis of the accident sequence from its inception until several hours after reactor vessel failure. The first three calculations were intended to investigate the role of automatic depressurization system (ADS) and drywell spray activation (a dedicated alternate power supply system is assumed) on Mark II severe accident containment performance. These calculations were halted at the time of drywell floor burn-through (due to the core-concrete interaction), because of MELCOR code limitations. The last four calculations were intended to investigate the potential implications of early entry of core-concrete debris into the wetwell via downcomers.

Table 4.9-4 summarizes the results of the calculations performed for this sensitivity study in terms of the estimated time to containment failure. Containment failure via the traditional over-pressure failure mode (at approximately 135 psig) was predicted to occur in the depressurized reactor vessel scenario when 100% of the debris is assumed to directly enter the wetwell pool. The predicted time to containment failure for this sequence is 8.5 hrs. This failure mode was also predicted for the scenario in which the reactor vessel is not depressurized and 100% of the debris remains in the drywell reactor pedestal. The predicted time to containment failure for this sequence is 10 hrs.

Four short-term blackout calculations investigated the potential implications of early interaction of core-concrete debris with the water contained in the in-pedestal wetwell region. The results of these preliminary calculations (Fig. 4.9-1) indicate that the early entry of very large fractions (80 - 100%) of the core-concrete debris into the suppression pool could result in over-pressure failure of the containment within 9 hrs. of accident inception if no mitigation is available. Specifically, if no RHR suppression pool cooling or containment venting is available for containment heat removal, then containment failure is anticipated. This is modeled in the base case NMP2 IPE as an intermediate time failure of containment as confirmed by these sensitivity runs.

Conclusion

The results from NUREG/CR-5565 indicate that the minimum time to reach a given containment pressure (e.g., ultimate failure pressure) is when 100% of the molten debris is relocated to the wetwell. This plot also shows that 40% to 70% of the debris in the wetwell results in the longest times to reach high containment pressures. The ORNL assessment also indicates that the time to containment failure for a pressurized RPV with debris discharge to the drywell only would be on the order of 10 hours for an unmitigated short SBO. One caution in the interpretation of these results is the apparent neglect of the containment failure pressure on drywell temperature. The NMP2 MAAP analysis has factored this additional variable into the containment failure evaluation.

4.9.3.2.2 RPV Breach Model

Two aspects of the RPV breach model are investigated here:

- The impact of debris relocation assumptions on the timing and subsequent containment pressurization.
- The impact of breach size.

Without recovery of ECCS, the core will continue to melt and eventually relocate into the lower head. In the MAAP model, this relocation involves a relatively large mass of molten material.

Considering that all BWRs have lower head penetrations, it is likely in this model that rapid heatup and failure of a penetration will occur. The BWR MAAP model calculates the heat-up and failure of the lower head penetrations. However, the rate and thermodynamic state of the material entering the lower head is uncertain. Other scenarios have been postulated in which core debris remains coolable within the lower head until all of the remaining water is boiled away [NUREG/CR-5565]. The debris then heats up and eventually the lower head fails. These scenarios can be simulated in MAAP-BWR with a minor code change to delay vessel failure in the manner described above. This was done for Cases IA-I-LD and IV-A-1LD. The sensitivity results are represented by cases IA1LDDB and IVA1LDDB, respectively.

As Table 4.9-5 indicates, delaying vessel failure can have a dramatic impact on the results. In case IA1LDDB, it takes about 45 minutes to boil away the water in the lower plenum and to reheat the debris bed to above its melting point. This extended time period allows more molten material to transport to the lower plenum before vessel failure occurs. Therefore, about 10% more core material is available in the lower plenum at the time of vessel failure, but the reduced amount of energy associated with the debris (no superheat in this case) reduces the containment pressure spike after vessel failure to below the pressure-temperature thresholds shown in Figure 4.4-8 which were exceeded in the base case (IA1LD). With containment heat removal and injection restored long term, containment failure would be precluded as shown in case IA1LDDB. For the case IAILD the containment integrity may be considered marginal because of the higher pressure/temperature spike.

In case IVAILDDB, since only about 31% of the core material is initially transported to the lower plenum, the delay to vessel failure becomes even more extended (from 2.1 to 6.3 hours). Over this time period an additional 26% is eventually added to the lower plenum before vessel failure. More importantly, this delay allows more fission products to be swept to the suppression pool prior to vessel failure. About 65% of the CsI inventory is left in the pool in case IVAILDDB whereas only 10% is left in the pool in case IVAILD. The end result is about a factor of four reduction (from 27% to 7%) in the total CsI release to the environment if the non-baseline assumption of a delayed bottom head breach is used.

Therefore, it can be concluded that the default assumption of allowing the vessel to fail shortly after debris slumps to the lower plenum is conservative relative to both radionuclide release magnitude and timing.

In addition, NRC contractors have performed a sensitivity study of the containment pressure response for a surrogate BWR due to a high pressure blowdown (NUREG/CR-5331). Sensitivity to the RPV lower head breach size was determined by performing calculations in which the RPV break area was varied between of 0.005, and 1.0 m^2. The base case used an area of 0.0079 m^2, which corresponds to an effective hole diameter of 0.1 m (3.65 in.).

These analyses suggest that containment peak pressure increased with the RPV break area up to break areas of approximately 0.1 m^2 (14 in. diameter). For larger RPV break areas the peak pressure was nearly independent of break size. For the case with an area of 1 m^2 (3.7 ft. diameter or more than 10 ft^2), the peak pressure occurred before the vent downcomers cleared of water and increased the peak pressure about 10% over that of the base case. The highest peak pressure calculated for this sensitivity study was 127 psig for the 1.0 m^2 RPV breach size.

4.9.3.3 Debris Coolability

Without continued water injection, the core debris will dryout and begin to heat up. Eventually, the debris begins to interact with the concrete basemat. There is also a possibility that core-concrete attack can occur in the presence of an overlying water pool.

Prior to containment failure, any fission products that are evolved by core-concrete attack or by long term revaporization will be deposited in the drywell or an overlying water pool, or will be transported to the suppression pool. At containment failure, the amount of fission product release will be dictated by the airborne mass at failure and the subsequent rate of revaporization from the drywell and RPV.

MAAP assumes that debris will transfer heat to an overlying water pool at a rate given by the critical heat flux. A user input parameter (FCHF) controls the magnitude of the heat flux assumed. The nominal value used for FCHF was 0.10. Case IA1LD was rerun with FCHF reduced to 0.02; this new case is labeled IA1LDFC. The lower value of FCHF greatly reduces the amount of heat transferred from a debris bed to an overlying water pool.

There are unique features of NMP2 that influence the effects of possible variations in FCHF and the resultant modeling of reduced heat transfer to the overlying water. First, NMP2 has only a slight lip (~ 3") on the downcomers located in the pedestal. Therefore, only a small amount of debris can be retained in the pedestal using most core melt progression models. Also, the ex-pedestal drywell floor area is sufficiently large (5200 $ft.^2$) to cool debris in that region even without water available. (Sensitivities to these assumptions are explored in Section 4.9.3.4.) Consequently, the choice of FCHF influences only the behavior of the debris which transports to the suppression pool. Figure 4.9-2 clearly shows the effects of the reduced heat transfer rate on the calculated debris temperature in the wetwell. Although, the debris is not cooled in case IA1LDFC until about 25 hours compared to almost immediately in case IA1LD, the peak temperature of the debris is still limited to about 1300F. Therefore, the overall results are not significantly influenced by the choice of FCHF for NMP2 as can be seen in Tables 4.7-1, 4.7-2, and 4.7-3.

4.9.3.4 Debris Distribution in Containment

4.9.3.4.1 Effective Area of Drywell Floor

MAAP assumes that as debris is entrained out of the pedestal it spreads uniformly across the entire drywell floor. In reality, this may not be a true representation. To investigate the sensitivity to this assumption, a case was run with the drywell floor area reduced by a factor of four to 1300 $ft.^2$. The sensitivity case IVA2LWA4, represents a sequence with debris uniformly spreading across one quadrant of the drywell with no injection or containment heat removal available after RPV failure. As Figure 4.9-3 indicates, the long term debris temperature is about 600°F higher in case IVA2LWA4 than in the base case IVA2LW. However, the overall effects on containment pressurization and heatup are negligible as is apparent by examining the gas temperatures in the drywell (see Figure 4.9-4). In addition, since only a minimal amount of concrete attack is predicted in case IVA2LWA4, the fission product releases are also approximately the same.

4.9.3.4.2 Amount of Material Retained in Pedestal

In the NMP2 pedestal configuration, the downcomers do not extend significantly above the floor elevation. Consequently, in the MAAP base calculations, a negligible amount of material is retained in the pedestal region (i.e., most of the debris from the vessel as it enters the pedestal is either entrained in a gas stream that enters the ex-pedestal drywell or is drained into the suppression pool). This assumption seems reasonable as a best estimate for NMP2. However, if more debris were retained in the pedestal region, the potential for increased concrete attack and the resulting increased non-condensibles would affect the pressurization rate could significantly influence the results. Therefore, as a sensitivity to this assumption, two cases (IA4LDNP and ID8LD) were modified and rerun with the MAAP input parameter XCMC set equal to 1.0 ft. The resulting sensitivities are cases IA4LDXNC and ID8LDXC which are summarized in Table 4.9.6. The modification allows one foot of debris material to be retained in the pedestal before draining to the suppression pool. Entrainment to the drywell is not affected by the choice of this parameter other than to keep debris in the pedestal longer (therefore making more available for entrainment).

Table 4.9-6 shows the influence of the increased concrete attack which is allowed to occur in the sensitivity cases IA4LDXNC and ID8LDXC. The threat to containment is more rapid as inert aerosols and non-condensible gases generated from the concrete attack process increase the pressurization rate. However, the presence of inert aerosols also tends to increase the rate at which fission product aerosols are removed from the gas by gravitational settling. This leads to a slight reduction in the amount of CsI that is ultimately released to the environment in the sensitivity cases. However, since the end releases are dominated by primary system revaporization after containment failure, the magnitude of the CsI releases can be considered to be approximately the same for these cases.

4.9.3.5 Suppression Pool Mixing

MAAP assumes that all of the water in the suppression pool (both inside and outside of the pedestal wall) is well-mixed, with the pool response characterized by one average pool temperature. This may lead to a reduced containment pressurization rate and a delay in the calculated times to vent or fail containment. Due to the four large flow openings (approximately 3 ft. x 6 ft.) located in the pedestal wall four feet above the suppression pool floor, significant pool mixing should be expected. However, bounding separate effects hand calculations were performed to examine the impact if no pool mixing is considered. First, a hand calculation was done to compare with MAAP case IA4LDNCP (to verify the validity of the hand calculation) assuming a well mixed pool; then a calculation was performed for no-mixing and partial-mixing cases. The details are included below:

<u>Prediction of Containment Failure Time for NMP2: Case IA4-LD-NCP</u>

<u>CASE 1: Entire Wetwell Available - Well Mixed</u>

Input:

Volume Drywell (free):	8592 m^3
Volume Wetwell (free):	9324 m^3 - 3878 m^3 = 5446 m^3
Volume Pedestal (free):	278 m^3
Initial Pool Temp:	38°C (100°F)

Calculated by MAAP:

In-Vessel H_2:	390 kg. (860 lbs.)
Fraction of Core in DW:	.31
Fraction of Core in PED:	.03
Fraction of Core in WW:	.66
Total pool mass:	3.88 E6 kg.
Pool Mass in Pedestal:	2.00 E5 kg.

The total containment pressure can be calculated as:

$$P_c = P_i + P_{H2} + P_{sat}$$

where:

P_c = containment pressure
P_i = initial pressure
P_{H2} = H_2 pressure due to Zr oxidation
P_{sat} = pressure of saturated steam evaluated at the suppression pool temperature
P_i = 15 psia

The additional H_2 represents the following pressure increase:

$$P_{H2} = \frac{NRT}{V} = \frac{(195)(8314)(320)}{14316} = 36 \text{ kpa} = 5 \text{ psia}$$

where:

N = number of moles of H_2
R = gas constant
T = gas temperature
V = total volume

The approximate RPV initial water inventory is 3E5 kg of saturated water at 1000 psia. If this water is cooled to saturation at 15 psia, the energy involved is

$$Q = 3E5 \text{ kg } (H_{f1000} - H_{f15})$$
$$Q = 3E5\ (1{,}267{,}000 - 417{,}000)$$
$$Q = 2.6E11 \text{ j}$$

The increase in suppression pool temperature due to this energy dump is:

$$\Delta T = \frac{Q}{MC_p} = 2.6E11/(3.88E6 \times 4200)$$
$$= 16°C\ (30°F)$$

Quenching the core debris that falls into the wetwell is equal to the following increase in pool temperature:

$$\Delta T = \frac{MC_p\Delta T(\text{Debris})}{MC_p(\text{Pool})} = 15°C(26°F)$$

The initial pool temperature just after vessel failure is then
30 + 26 = 56°F + 100°F or 156°F (69°C)

For the containment pressure to be 123 psia

$$P_{sat} = 123 - 5 - 15 = 103 \text{ psia}$$

The saturation temperature corresponding to this pressure is 166°C (331°F). This is an increase of 175°F (97°C) over the pool temperature just after vessel failure. Assuming .7% decay heat, the following computes the time after vessel failure (2.7 hrs) that it would take to increase the pool temperature to 331°F.

$$t = \frac{MC_p\Delta T}{Q}$$

Q = decay heat = (.007)(.66)(33E8)
where .66 = fraction of core in wetwell
= 15 MW
M = Mass of pool = 3.88E6 kg
ΔT = 97°C
C_p = specific heat of water = 4200 J/kg

therefore t = 105,000 sec = 29 hrs

Since vessel failure occurs at 2.7 hrs, the approximate time to reach 123 psia is 32 hrs.

CASE 2: Pedestal Water Only Used as Heat Sink

If we were to assume that this increase of 97°C only occurred in the pedestal region and did not involve the outer wetwell pool then the time to reach 123 psia will be:

$$t = \frac{MC_p\Delta T}{Q} + 2.7 \text{ hrs}$$

where

M = pedestal water mass, 2E5 kg
C_p = specific heat of water, 4200 J/kg
ΔT = 97°C
Q = 15MW

The pressure would reach 123 psia in just over 4 hours.

CASE 3: Partial Mixing Case Between Inside and Outside the Pedestal

If we assume a delay in the bulk pool heatup such that the outer pool is 50°F cooler than the inner pool then the time to reach 123 psia in containment can be computed by:

$$Q\,t = (MC_p\,\Delta T)_{ped} + (MC_p\,\Delta T)_{wetwell}$$

$$t = \frac{(2E5)(4200)(97) + (3.68E6)(4200)(69)}{15MW}$$

$$\Delta T = 76{,}500 \text{ sec} = 21 \text{ hrs}$$

The time to reach 123 psia would be approximately 24 hrs.

It is appropriate to expect natural circulation to occur between the water inside the pedestal and in the wetwell region outside the pedestal. Based on the above calculations, it appears that an intermediate (I) (6-24 hrs) release timing is justifiable. A summary of the results is presented in Table 4.9-7 which also presents cases with containment heat removal available for illustration purposes.

4.9.3.6 Pool Bypass

Suppression pool bypass can have a dramatic impact on radionuclide releases. Two dominant pool bypass mechanisms have been identified during severe accidents:

- Downcomer failure due to debris interaction
- Vacuum breaker failure

4.9.3.6.1 Downcomer Failure Within the Pedestal

The NMP2 containment configuration includes 8 steel downcomers that penetrate the drywell floor within the pedestal below the RPV. These downcomers also form part of the boundary between the drywell and the wetwell airspace. A breach of this boundary represents a bypass of the suppression pool. This can result in an increase in radionuclide releases if a wetwell airspace failure or wetwell venting occurs during a postulated core melt progression accident. However, it was discovered that the timing of the downcomer failure relative to the time of RPV breach can have a strong influence on the transport of radionuclides to the environment.

As part of the assessment of the NMP2 pedestal downcomer integrity, it was initially assumed that the in-pedestal downcomer failure occurred immediately at the time of vessel failure for the majority of the MAAP calculations. Given failure of these pipes, a direct path from the pedestal region to the wetwell airspace would be created resulting in suppression pool bypass. Because extensive studies by the NRC have identified significant delays between initial molten debris contact and downcomer failure, it was judged important to characterize accident progression with a best estimate of downcomer response. Sensitivities were made testing various combinations of failing, not failing, or failing with a delay of seven minutes after vessel failure.

As part of the assessment of in-pedestal downcomer integrity and the timing of its failure, three separate time phases may be considered:

- Transport of debris from the RPV to the floor,
- Transport of debris within the pedestal to the downcomer pipe, and
- Attack or heat up time.

A detailed set of calculations was performed by ORNL in NUREG/CR-5623: BWR Ex-Vessel Corium Interaction Studies, S.R. Greene, et al., ORNL, dated November 1991.

Based upon this reference and other expert opinions, experts have concluded that the timing of downcomer failure time is strongly dependent on the following:

- The debris superheated,
- The debris depth on the pedestal floor, and
- The melt to crust heat transfer coefficient.

The transport times are shown below to be relatively small contributors to the total estimate of delay time. Therefore, little effort has been expended to attempt to search for available experimental or analytic work.

Transport Time From RPV

First, sufficient debris material is required to form a melt flow to the downcomers.

The timing of the transport of sufficient material from the RPV to the drywell floor is estimated to be on the order of 30 seconds based on NMP2 MAAP calculations.

Transport Time From Pedestal to Shell

NUREG/CR-5423 in Appendix B indicates that once debris is on the floor, debris can then reach the downcomer in less than 11 seconds (see Appendix B of NUREG/CR-5423).

Next the timing of the downcomer failure itself is evaluated. However, in this case a second report on debris attack of steel material is used to infer the timing for the debris induced failure of the steel once it is in contact with the surface.

Attack Time

The time for attack is the most crucial of the times evaluated.

ORNL work by S. Greene indicated that for a situation such as shown in Figure 4.9-5 the time from debris release until the pipe failure (time when the steel exceeds the melt temperature) is 1.5 hours. In a similar manner, Figure 4.9-6 shows debris attack of a vent pipe. In this case, a different code calculation with the modeling shown in Figure 4.9-6 indicated failure occurs in less than 30 minutes. These are for ½ inch pipe thicknesses

Finally, in a third situation, modeled as shown in Figure 4.9-7 the calculated time for the plate failure is approximately 7 minutes. Specifically, the 2DKO program was used by ORNL to analyze the response of a stainless steel plate to falling metallic debris. The debris pour history calculated by BWRSAR was employed. For minimal structural material in the drain pipe, it was evident that the drain pipe atmosphere would quickly heat up to a high enough value such that radiation heat transfer from the heated plate to the pipe region would be small. Thus, for the sake of conservatism, the plate surface-to-ambient heat transfer coefficient was set to zero. The thickness of the plate was taken as 0.5 in.

ORNL (S. Greene, et al.) calculated a plate failure time of 7 minutes based on this configuration.

Conclusion

In summary, the molten debris attack of the downcomer causing failure is estimated to take:

- Less than 30 seconds for blowdown of sufficient material to challenge the downcomers
- Less than 11 seconds for transport to the downcomers
- More than 7 minutes for melting of the downcomer

The total time from RPV breach until downcomer breach is therefore conservatively estimated at 7 minutes.

This is considered to be the delay time between the initial RPV breach due to molten debris and the time when downcomer integrity is considered to be compromised.

The seven minutes is based on the work at ORNL for Mark II containments (S. Greene, et al.) which calculated that seven minutes would be required to heat up and fail certain pedestal steel configurations, i.e., drain plates. The configuration is not exactly the same for NMP2 and the time to fail the downcomers is believed to be longer. However, the radionuclide release may not be affected. A choice of seven minutes was used for the sensitivity cases shown here. Key results for these cases are shown in Table 4.9-8.

These analyses and sensitivities appear to indicate the following:

- A realistic assessment that the downcomers will survive for at least 7 minutes after vessel failure, results in:

  - Medium release if the containment failure is in the wetwell airspace
  - High release if the containment failure is in the drywell head.

  Note that no specific sensitivity of drywell head failure release path to the integrity of the downcomers has been performed because cases with no downcomer failure (the most optimistic case) produces a high release.

In the base case (IA1LD), with pedestal downcomer failure right at vessel failure, the resulting load on the drywell quickly exceeded the pressure-temperature limits established in Figure 4.4-8. If downcomer failure is not allowed or delayed long enough to maintain pressure suppression capabilities, then the dynamic load in the drywell is less severe making it highly unlikely that the containment will fail shortly after vessel failure for this type of scenario. For example, in case IA1LDNP, containment failure is precluded immediately after vessel failure.

The IA4 and ID8 sensitivities show the large impact of not having pool bypass (no downcomer failure and containment failure in the wetwell, IA4LWXNC and ID8LWNPV) versus cases with "complete" pool bypass in the form of drywell head failure (IA4LDXNC and ID8LDNPV). Maintaining the release path such that it does not bypass the pool is extremely beneficial in reducing source term. Delaying downcomer failure by seven minutes also leads to a significant reduction in the reported source term release compared with the drywell failure case with pool bypass. However, there are still fission products retained in the vessel for these cases which would be available for revaporization and release should the vessel and drywell continue to heat up in an unmitigated fashion leading to failure in the drywell. Consequently, the seven minute delay is judged not to have a significant impact on the releases for cases that are dominated by late revaporization (Class I and III).

The Class IV (ATWS) sensitivities to pedestal downcomer failure show the first order effect of whether or not the pedestal downcomers fail (IVA1LW vs. IVA1LWNP and IVA2LW vs. IVA2LWNP). A factor of ten reduction in source term for these cases is caused by MAAP modeling limitations calculating decontamination factors in saturated pools. This is discussed more fully in Section 4.9.3.10. The case with a seven minute delay (IVA2LWN7) shows that a substantial reduction in the release can occur for cases with containment failure before vessel failure since the majority of the release occurs shortly after vessel failure. Consequently, the base case results for releases in Class II or IV cases with pedestal downcomer failure immediately at the time of vessel failure is considered conservative and is not used as the base case.

4.9.3.6.2 Vacuum Breaker Failure

The effects of pool bypass were clearly demonstrated in the pedestal downcomer failure discussion above. Since pedestal downcomer failure is more likely to occur for core damage sequences at NMP2 than vacuum breaker failure, no explicit sensitivity runs were made involving vacuum breaker failure.

4.9.3.7 Containment Failure Size

Several sensitivity cases were run to evaluate the impact of a small (0.194 ft^2) containment failure versus a large (2.0 ft^2) containment failure. Table 4.9-9 summarizes the containment failure size sensitivity cases. In general, reducing the assumed break area will slightly reduce the calculated fission product release. This can be seen from the following sensitivity run comparisons. Cases that are dominated by late primary system revaporization (e.g. - IA4 and ID8) will probably not be affected as much by the assumed containment failure size. In that case, the revaporization rate will dictate the source term rather than the rate of efflux from containment.

4.9.3.8 Containment Failure Location

The magnitude of fission product releases resulting from a severe accident is strongly influenced by the location of the containment failure. This dependence is compounded by the effects of the reactor building. For NMP2, a failure of the upper drywell head would lead to a release into the reactor building refueling floor. The blowout panels located in the refueling floor elevation would quickly open to relieve pressure, resulting in a direct release to the environment. A release to a lower region in the reactor building (via wetwell airspace failure) would have a more tortuous route to the environment resulting in additional residence time and deposition as the release is transported through the reactor building. Additionally, if pool bypass does not occur, a significant reduction in source term release will occur due to the effects of pool scrubbing. Table 4.9-10 summarizes the various results.

It should be noted that wetwell failures below the water line in the suppression pool can be characterized recognizing the ABB Impell primary failure modes. The principal failure mode is a failure at the junction of the wetwell wall and the base rate. This failure mode results in the equilibration of the water level inside and outside the containment and therefore a release pathway that has a scrubbed release through water. Therefore, the release can be considered similar in magnitude to the wetwell airspace breaks with <u>no</u> bypass.

4.9.3.9 Drywell Equipment Mass

The equipment mass in the drywell may impact the rate of temperature rise in the drywell. The assumed NMP2 drywell equipment mass is conservatively estimated at 220,000 kg. A more reliable estimate for NMP2 has been recently estimated as twice this conservative value of 220,000 kg. Consequently, sensitivity cases were run for a case where containment failure is predicted to occur late in the sequence due to high drywell temperatures (IA4LDNP). The sensitivity cases, IA4DN10A and IA4DN10, examine the influence of two cases. The cases represent drywell equipment masses that are two (2) and ten (10) times the conservative estimate used in most of the base case runs. Key results are show in Table 4.9-11. As can be seen, even a dramatic increase in the assumed drywell equipment mass does not have a significant effect on the particular sequence results for the NMP2 Mark II.

4.9.3.10 Saturated Pool Decontamination Factor

Maintaining suppression pool water temperature as low as possible extends the time that the operator can establish makeup to either the RPV or the drywell upon its breach. Plant specific MAAP calculations have shown that the availability of water to fuel debris (given that in-vessel recovery was unsuccessful) reduces both the impact to the containment, as well as the source term that accumulates inside the drywell air space. The suppression pool water temperature also affects the potential for "scrubbing" aerosols if the source term is directed through the pool before egress from the containment. The MAAP code [Rev. 8.0] assigns a pool DF of 1.0 for all saturated pools. This assumption does not appear consistent with NEDO-24250 and recent experiments.

There are three computer codes currently in use which model the aerosol removal process in a suppression pool; (1) SPARC, written by Battelle Northwest Laboratory, (2) SUPRA, written for EPRI, and (3) DECON, written by General Electric. All three of these codes are based on the three fundamental particle removal mechanisms. These mechanisms are Brownian diffusion, gravitational settling, and inertial deposition. In addition, each model includes other particle removal mechanisms which are not necessarily included by the other models.

The parameters affecting aerosol scrubbing have been categorized by several sources as follows: [Ref. 135]

- Most important:
  - Particle size
  - Particle density
  - Bubble size and shape
  - Volume fraction of steam in inlet gas

- Intermediate importance:
  - Pool depth
  - Pool temperature
  - Percent of soluble material in particles

- Least important:
  - Noncondensible gas composition
  - Pressure above pool

The principal obstacle to obtaining accurate experimental measurements of decontamination factors for saturated or nearly saturated pools is the significant condensation on the surfaces of the pool compartment above the water level. Condensing steam in the pool compartment drives the aerosol to the walls (a mechanism known as diffusiophoresis), and the condensation laden with aerosol then drips back down into the pool. The effect of this phenomena has been studied at Battelle Columbus Laboratory and is presented in the EPRI report "Scrubbing of Aerosols by Water Pools; Volume 1" [Ref. 136] and EPRI RP 2117 [Ref. 137]. Several other studies have examined the problem of aerosol scrubbing by suppression pools, but few have done experiments to specifically test the effect of saturated pools on decontamination factors.

The results of the initial BCL experiments show very high DF values for saturated pools. These unexpectedly high DF values were obtained, however, without consideration of diffusiophoretic deposition of aerosol to the tank walls. In order to isolate the magnitude of this effect and determine a true DF value, BCL ran a series of experiments in which the exterior of the tank was insulated and heating blankets were installed at several positions on the exterior of the tank surfaces above the water line to insure the wall temperature was approximately equal to the gas temperature. A trough was installed just above the water level to measure the amount of condensate runoff. For this series of runs the deposited mass on the wall was measured to be no more than 3% of the total mass escaping the pool, and thus it was concluded that the DFs measured in these tests were not significantly affected by diffusiophoresis in the air space. The results of these tests indicate a minimum DF of 5.0 and a trend of increasing DF with increasing steam mass fraction of the injected gas. Since the injected gas in a BWR severe accident would consist primarily of steam, these runs with large steam mass fractions are most representative of an actual accident scenario.

Based on careful review of the Nine Mile Point 2 Level II MAAP runs, it is justifiable to take credit for at least a DF of 10 for time phase when the pool is saturated and fission products are transported through the pool. In actuality, a DF greater than 10 is considered likely for the time periods of interest, but a DF of 10 will be used as a conservative approach to adjust the release categories. The assumption of a DF of at least 10 for all time periods prior to pool bypass has been applied to the affected cases. The original results and the adjusted results are shown in Table 4.9-12.

4.9.3.11 Containment Venting

Several of the base cases involved the successful operation of containment venting. This was nominally modeled as opening the vent and holding it open upon first exceeding the NMP2 containment vent pressure (45 psig). However, a better interpretation of the NMP2 EOPs would have the operator venting the containment in a fashion sufficient to maintain the containment pressure below the Primary Containment Pressure Limit (PCPL). This case was run with the vent open at 45 psig and closed at 35 psig, simulating the operator's desire to maintain the pressure below the PCPL curve. Case IIIC1-LD was chosen as the base case and the sensitivity was performed in case IIIC1V56. An additional sensitivity was performed in case IIIC1V1O in which the vent was held fully open after the PCPL curve had been exceeded for ten minutes. The key results are presented in Table 4.9-13.

Clearly, the IIICV156 results indicate that a better representation of following the NMP2 EOP procedures will significantly reduce the reported fission product releases. For this case, the vent is calculated to cycle 17 times over the first 14 hours as the water from the initiating LOCA in the drywell boils away; then a gradual depressurization occurs until the suppression pool is saturated at about 24 hours. Only 4 vent cycles are required to maintain the pressure below the PCPL curve for the remainder of the simulation out to 40 hours. The fact that the total time period in which the vent is actually open is greatly reduced contributes to significantly reducing the fission product release. Delaying the vent opening by 10 minutes, as in case IIIC1V10, does not substantially affect the results. Also, for cases where late containment failure occurs due to high temperatures in the drywell, a reduction in the source term during the time period of venting will be insignificant compared to the magnitude of the release due to late primary system revaporization and direct release through the drywell.

4.9.3.12 Reactor Building Modeling Assumptions

Secondary containment configurations differ among BWRs. Speaking generally, however, for the secondary containment to retain a significant quantity of fission products, one of two conditions must occur. Either active decontamination measures or natural removal processes must be successful.

"Active" decontamination processes include scrubbing due to the passage of fission products through deep water pools, decontamination by ventilation system filters, or scrubbing due to extensive area coverage from fire sprays. If such measures are functional, they will generally overwhelm the natural settling processes and result in relatively small environmental releases of all fission products except for noble gases. A few qualifications to this statement must be offered, however. First, ventilation filters are not usually designed for large aerosol loadings: Significant aerosols would be present in a severe accident.

Consequently, filters have been postulated to tear, overheat, or clog. Second, ventilation filters may not cover all the volume of all the affected secondary containment regions. Finally, while aerosol behavior is relatively well understood, there are significant uncertainties associated with the effectiveness of scrubbing of fission product vapors in water pools; these might impact the release when the source of fission products is at a very high temperature.

If no such active measures are at work, we must rely on natural settling processes. For these to be effective, the fission products must have a relatively long residence time in the secondary containment before they can be swept to the environment. This, in turn, generally requires that the ventilation systems be secured, that the flowrate from the primary system or containment be relatively small, and that vigorous natural circulation be avoided between the secondary containment and the environment. The last of these requirements is often the most difficult to confirm. Vigorous natural circulation between the secondary containment and the environment can be set up if one large hole is opened (leading to large counter-current flows through the one opening), or if two holes are opened, one low in the building and one higher up. This latter configuration gives rise to a "chimney-like" flow pattern. Since it is often difficult to know the precise failure pressures and failures modes of the myriad of openings in the secondary containment, and since the pressure differentials between rooms are typically quite small, it can be difficult to establish precisely which doors open and which stay closed. For this reason, one must evaluate carefully any prediction of large decontamination factors due to natural settling processes in secondary containments.

With all of that in mind, and examining the base results for NMP2, it was apparent that relatively low decontamination factors were being calculated. Table 4.9-14 summarizes some of the results. For drywell failures to upper regions of the reactor building, DFs of only 1.5-2.0 were predicted. Given the unavailability of automatically initiated spray systems in this region and the close proximity of the reactor building release path (blowout panels) to the containment failure location, the DF calculated is minimal and not much can be done to increase the DF in the reactor building for these cases.

For the wetwell failure cases, DFs of about 2-5 are predicted. (The only exception is the IIT1-LW case which fails containment immediately after vessel failure, resulting in a large flow rate through the reactor building and a low DF of approximately 1.3.) Upon examining the output closely, it becomes apparent that the wetwell failure results are influenced by the assumption of a low pressure differential forcing the opening of the railroad door on the ground floor. When this door opens and the blowout panels at the refuel floor elevation also open, the "chimney-like" flow pattern described previously increases the throughput in the reactor building. Consequently, the residence time and resulting chance for increased fission product retention is minimized. The containment isolation failure case with failure low into the reactor building (IA4IS3DP) was rerun to examine the influence of the assumed railroad door failure. If this door does not open (case IA4I3NRR), the fission product retention in the reactor building is greatly increased as is shown in Table 4.9-15. Although it appears reasonable that the railroad door may fail at a low pressure differential, whether or not it actually fails given the presence of the blowout panels on the refuel floor is still in doubt. This case indicates the <u>potential</u> for increased fission product retention for cases with failures low in the reactor building if the railroad door does not fail.

Sensitivities on the effects of reactor building sprays and operation of the Standby Gas Treatment System were not deemed necessary for NMP2 due to the prevalence of low decontamination factors in the reactor building for the majority of sequences.

4.9.3.13 Use of Drywell Spray

The use of drywell sprays to provide beneficial effects on minimizing radionuclide releases has several important restrictions. These restrictions make it unlikely that drywell spray can be successfully used in severe accident mitigation given the current restrictive conditions in the EOPs (based on BWROG Rev. 4 EPGs). The principal restrictions are the following:

- The current Drywell Spray Initiation Curve (DWSI) for NMP2 provides restrictions on when the drywell sprays can be initiated. Given this curve and the conditions that exist for many of the severe accidents investigated in the Level 2 analysis, the curve will be satisfied only a short amount of time during which the sprays could be beneficial.

- Because RHR is probably unavailable as a vessel injection source, RHR pumps may therefore be unavailable as a pumping source for DW sprays.

- The EOPs direct the systems be used to preferentially to ensure adequate core cooling over a wide spectrum of accidents. Reactor water level is given as the primary method of indication of adequate core cooling. Therefore, for degraded states with the RPV breached resulting in no water level indication, if RHR is reestablished, the RHR must be used for injection to the vessel first, unless the Primary Containment Pressure Limit (PCPL) has been exceeded, and if the PCPL is exceeded it is likely that the DWSI limit is exceeded as well, thereby precluding DW spray initiation.

- No explicit directions are given to begin spraying from external sources regardless of the suppression pool temperature on containment pressure. In fact, Revision 4 of the EOPs implicitly assumes that sprays will only be used from suppression pool sources.

4.9.3.14 Containment Flooding

Given the current state of knowledge regarding severe accident phenomenology, the NMP2 EOPs have established a near optimum balance among the contingency procedures which the operator can implement.

The EOPs generally define one of the following: the optimum procedural path; a procedural pathway that is close to the optimum; or, a pathway for which insufficient analytical (and experimental) information is available to more precisely define the optimum pathway. Changes in the current understanding of severe accident phenomena or in the philosophy of dealing with severe accidents may impact some of the EOP steps and contingency actions.

The specific issue that is addressed here is the decision regarding containment flooding versus possible alternatives.

Postulated Scenario

The specific accident sequence investigated has the following elements:

- Core damage occurs due to the loss of injection makeup to the RPV.
- The containment is initially intact and inerted
- The RPV water level continues to drop below 2/3 core height
- Eventually, the RPV bottom head is breached and core melt progression continues with debris released to the containment.
- In the mean time, power or injection sources are reestablished. Two choices exist: water from LPCI to the RPV or drywell sprays, or RHR service water to the RPV.
- For the following it can be assumed that RHRSW will be used to flood the containment.

The containment flooding actions are specified in the NMP2 EOPs whenever the RPV water level is below approximately 2/3 core height and cannot be restored, or when the RPV water level is indeterminant and RPV pressure cannot be maintained 61 psig greater than the wetwell pressure. Two actions of specific interest are:

- RPV venting when the containment water level reaches the bottom of the recirculation lines (i.e., near the bottom of the RPV and above any debris on the floor).
- Drywell venting if the containment pressure cannot be maintained below the Primary Containment Pressure Limit (i.e., approximately containment design pressure).

These two actions would be specified in the accident response actions to a severe accident in which core melt had progressed to cause the breach of the bottom head of the RPV and from which subsequent recovery actions allowed the injection of external water (e.g., RHRSW or fire water or condensate) to the RPV or to the drywell sprays.

Analysis

After performing thermal hydraulic analysis for various BWR IPEs (Mark I and IIs), it has become clear that, under certain postulated severe accidents, the BWR EPGs direct operators to perform actions that may have the most adverse potential impact on the public (namely, containment flooding and the associated RPV and drywell venting).

MAAP 3.0B was not designed to calculate all of the proper thermal-hydraulic conditions in a flooded containment; and input "tricks" that were successful for other plants were not successful for NMP2. Consequently, the conclusions regarding flooding must rely on these previous analysis. In general, it was found that opening the RPV vent would allow for a high release of fission products through the condenser into the turbine building and potentially to the environment. This would also be the case for NMP2 since the base MAAP calculations showed up to 50% of the CsI could be remaining in the reactor vessel at the time of RPV vent actuation. The opening of the vent and the resulting dynamic flow through the hot vessel would make revaporization of the fission products quite likely. As these fission products re-evolved, they would be released directly to the turbine building through the condenser. The actuation on the drywell vents could have the same effect on fission products in the drywell in that they would be released direct to the environment.

A possible improved response for these types of sequences for which the EPG directions result in the highest potential consequences at the earliest time is to provide the operators guidance on protecting containment and cooling debris using methods that do not require opening the RPV vent and avoid using the drywell vent unless no other alternative exists. Alternate sequence calculations have been shown to produce substantially lower releases and much longer times to failure if no action is taken, i.e., even no action is better than this specific action directed by the EPGs. This issue should be examined in future accident management investigations.

4.9.3.15 Coupling of Drywell Equipment Mass and Amount of Debris Retained in the RPV

The Mark II response to situations with no injection (e.g., due to support system failures) results in the following possibilities:

- Residual debris in the RPV and drywell may cause temperature increases in the drywell.

- Pressurization of the containment could result due to non-condensible gas generation and steaming of the suppression pool due to molten debris falling into the suppression pool.

The current base model Level 2 quantification for NMP2 has a large percentage of Intermediate/High (IHI) releases.

The reason for these results are the following:

- The types of sequences that remain in the Level 1 IPE and therefore lead to core damage are sequences with support system failures and therefore little or no recovery available in the Level 2.

- The result of no system recovery is that a drywell head failure can be induced due to high temperatures and pressures. This failure mode at elevated drywell temperature is based on the expert opinion evaluation of containment failure modes.

- The radionuclide releases (consequences) for this sequence type as determined by MAAP are right on the boundary between High and Medium. The sensitive parameters relative to these releases are the following:

  - DW equipment mass
  - Fuel mass remaining in the RPV long term
  - Duration of the accident calculation time
  - Size of the failure

  See Table 4.9-16 for the summary of impacts.

DW Equipment Mass

The DW equipment mass in the NMP2 MAAP model is believed to be underestimated by a factor of 2. This results in slightly shorter calculated times to high drywell temperature and therefore shorter calculated times to containment failure in the drywell. This is a relatively small effect on containment failure size and CsI released.

Fuel Mass Remaining in RPV and DW Long Term

The residual material in the RPV and DW is the principal driving force for elevated drywell temperatures above saturation. The larger the fraction of material in the drywell and RPV the more rapid the temperature rise and the higher the long term drywell temperatures.

The temperature of the drywell following vessel failure is a relatively strong function of the debris retained in the pedestal and drywell. Case IA-4-LD-XNC retained nearly twice the material of Case IA-4-LDNP and resulted in a temperature in the drywell at long times of 750°F for the former case versus 500°F for the latter case.

Time Duration of Release

The MAAP cases performed for the conditions of no injection lead to the assessment that the release tends to be a medium release during the time period specified in our projected evaluation (36 hours after RPV failure).

However, because of revaporization effects this release becomes a high release within about 5 additional hours beyond the 36 hours for the conservative case of (220,000 Kg of DW equipment mass).

Size of Failure

The treatment of the TD/TR failure as a large drywell failure is somewhat conservative based on the ABB-Impell assessment that even at high drywell temperatures the wetwell could still fail.

This conservatism can be addressed in a sensitivity case which specifies that the release be through the wetwell 10% of the time. This has been shown by MAAP calculations to reduce the radionuclide release from a High (H) to a Low for those 10% of the cases that are wetwell failures.

Conclusion

It appears reasonable for all these reasons to consider the TD cases to be of medium magnitude rather than high (H). However, the release magnitude for all TD cases have been left as High. This is judged to be a conservatism in the assignment of release magnitude.

The interpretation of the MAAP results are that:

- The mass of steel in the DW is judged to be underestimated in the baseline NMP2 MAAP model. A more realistic estimate would result in slight improvements in release characteristics.

- If no residual material is in the RPV, the DW will not fail on temperature, i.e., essentially zero consequence instead of IHI.

- Wetwell failure as an alternate location has not been included in the TD/TR model assessment.

The combination of these three "conservatisms" means that the I/HI category may be overestimated in the NMP2 IPE.

4.9.3.16 Summary of Deterministic Sensitivities

The formulation of the deterministic sensitivity evaluation was based heavily on the EPRI sponsored study that recommended which parameters in MAAP should be varied and over what expected range [Ref. 128]. Table 4.9-17 summarizes those recommended sensitivity cases performed for NMP2. Table 4.9-18 summarizes the additional sensitivity cases, beyond those identified in the EPRI report, which were performed explicitly to support NMP2-specific containment event tree evaluations.

4.9.4 Probabilistic Uncertainty Evaluations

In addition to those sensitivities that are most appropriately assessed using deterministic evaluations, there are also sensitivities that are more appropriately treated in a probabilistic framework. This subsection includes the results of the probabilistic sensitivities for the following postulated events:

- Induced failure of the reactor coolant boundary
- Mode of vessel failure
- Fuel coolant interactions
- Direct containment heating

4.9.4.1 Induced Failure of the Reactor Coolant System Pressure Boundary by In-vessel Phenomena

Temperature induced failures of the primary system leading to system depressurization may include either of the following mechanisms:

1. Induced LOCA Effects: This implies induced primary system failure associated with high internal temperatures during the core melt progression. Temperatures of 2000°F to 4000°F are calculated in available deterministic codes due to core degradation and Zircaloy oxidation. Primary system boundary components that are potentially vulnerable to such high temperatures and pressures (i.e., 1000 psig) are:

   - Recirculation pump seals
   - Instrument lines
   - Welded attachments of piping
   - Valves
   - RPV head seal.

   Preliminary PWR analysis indicates that in-vessel natural circulation produces more uniform core temperature and also transfers more of the core heat to the upper-plenum structure and walls. There is some indication that this wall heating might cause early failure of the primary system pressure boundary and reactor coolant system depressurization before the melt-through of the reactor coolant system in high pressure sequences for PWRs.

   Natural circulation and the resultant heat transfer between the uncovered region of the intact core (before collapse) and the upper plenum, can have a significant effect upon in-vessel severe accident behavior. If the reactor vessel remains pressurized, stresses on primary system components (such as pump seals, head seals, CRD seals, etc.) and structural members are greater than if the vessel had been depressurized. Hence, the likelihood of primary system failure is greater for accident sequences in which the reactor has not been depressurized and high temperatures (>1500°F) due to core melt progress are present.

   If the reactor coolant system pressure has fallen sufficiently at the time of corium melt-through of the vessel lower head, pressurized ejection and dispersion of the melt will not occur, eliminating the threat of pressurized melt ejection (i.e., direct containment heating) to the integrity of the containment.

   It is acknowledged that PWR analysis is a useful input but may not be directly applicable to the BWR design. Because of the differences in design and lack of BWR analyses, the NMP2 quantification has only included a relatively small probability that depressurization would occur due to a high temperature induced breach of the primary system.

2. Stuck Open SRVs: Detailed analyses of the in-vessel accident progression for high pressure core damage scenarios indicate that (for the in-vessel quenching model) very high temperature gases pass through the SRVs over about a 30 minute time period prior to vessel breach. The SRVs experience a maximum of about 100 open-close cycles during this period. (Note: this is based upon

plant specific MAAP analyses for other BWRs.) Because of the high temperature of gases flowing through the open SRVs, as well as possible vapors and aerosols, it is possible that SRVs could fail to reseat leading to stuck open SRVs and a depressurized RPV.

The RPV safety relief valves are cycling during the core melt progression due to high non-condensible gas generation and steam generation. During these cycles under extremely adverse temperature conditions the probability of a stuck open SRV is estimated to be substantially increased from that during normal operation. However, for this quantification a conservative characterization of the probability at these conditions far beyond the EQ envelope is estimated to be only a factor of 8 higher than the normal operation failure probability of 3.7E-3/valve/d. Assuming that three valves are activated five times (versus the 100 times predicted by some code calculations) over the 2 hours of core melt progression leads to a stuck open probability of approximately 0.45, or a probability of 0.55 that the safety relief valves do not stick open.

<u>Sensitivity Results</u>

Sensitivity to the baseline NMP2 assumptions regarding temperature induced LOCAs have been performed. The two sensitivity cases and the baseline assumptions are the following:

INPUT			
Parameter Affecting Sensitivity	Failure to Depressurize Probability		
	NMP2 Baseline	Sensitivity #1 Optimistic	Sensitivity #2 Pessimistic
Temperature Induced Primary System Failure	.7	.01	1.0
Induced SORV	.45	.01	1.0

The quantitative impact on the "large" release category is related to those portions of the spectrum associated with failure to depressurize and energetically induced failures during RPV blowdown. The sensitivity results are provided below:

OUTPUT RESULTS			
Scenario End State	NMP2 Baseline Frequency	Sensitivity #1 Frequency (Optimistic)	Sensitivity #2 Frequency (Pessimistic)
Large Release (High/Early)	7.7E-7/yr	3.8E-7/yr	1.27E-6/yr

4.9.4.2 In-vessel and Ex-vessel Phenomena

There are a number of potentially energetic phenomena that have been postulated to occur during the course of core melt progression. The sensitivity of the model to these phenomena are treated here in a probabilistic manner. The baseline probabilistic characterization is derived from published separate effects analyses and was quantified with phenomenological basic event estimates that are judged to be the most realistic for the NMP2 plant design. Nevertheless, in addition to the realistic base case, two sensitivity cases were run in order to investigate the effect of phenomenological uncertainties on the Level II results. In these two sensitivity cases the phenomenological events are varied to the 90% confidence interval values of probability distribution ranges determined from the review of industry studies. Sensitivity Case #1 represents the optimistic case in which the likelihood of these phenomenological containment failure modes are estimated to be at the low end of the probability distribution. Sensitivity Case #2 represents the pessimistic case in which the likelihood of the events are estimated to be at the high end of the probability distribution.

The values used in the base case for the various failure modes included in the sensitivity studies are summarized in Table 4.9-19. Table 4.9-19 also shows the range of probability estimates used in other studies. Note that data for some failure modes could not be located, and in one case are not directly applicable to NMP2. In these cases, the probability range of energetically induced failure modes is considered to be 1E-4 to 1E-2; 1E-4 is considered to be the optimistic value and 1E-2 is considered to be the pessimistic value.

Note that two of the failure modes, pedestal failure and direct containment heating, have a low probability of occurrence assigned to them. This is because these failure modes are judged to be insignificant almost to the point of being not applicable to NMP2. In the case of pedestal failure, the issue of concern is that substantial pools of debris will remain in the pedestal and attack the pedestal walls to the point of failure. However, a large fraction of the debris that falls to the pedestal floor is predicted to melt-through the downcomers and drop into the suppression pool, precluding significant degradation of the pedestal walls.

The second failure mode, direct containment heating, is a postulated event in which the melted core breaches the vessel wall (most likely at an instrument tube location) while the RPV is at high pressure. The combination of the nozzle size breach with the RPV at high pressure has been postulated to result in rapid ejection of particulated debris such that rapid heating of the atmosphere and containment failure occurs. Direct containment heating is widely believed to be a PWR issue and not a BWR issue, primarily due to the fact that BWR reactor vessels operate at much lower pressure, and are equipped with additional components (e.g., CRDs, "shoot-out steel") in the bottom head area. The components and other structures in and below the bottom head are predicted to prevent finely particulated dispersal of the melted core as it exits the vessel. Refer to Figures 4.9-8 and 4.9-9 for drawings of the bottom head and the shoot-out steel.

The values used in the two sensitivity cases are summarized in Table 4.9-20. In the case of the pedestal failure, the high value is not set to 1.0 because a large majority of the debris is believed to be discharged directly to the pool through the in-pedestal downcomers, thereby precluding substantial pedestal concrete attack. In addition, the concrete walls of the pedestal are over four feet thick.

A comparison of the results of the base case and each sensitivity case is shown in Table 4.9-21. The asterisks next to certain totals indicate the release category totals that changed compared to the base case. In Sensitivity Case #2 (the pessimistic case), note that the high magnitude release categories increased in value. This is due to the fact that sequences with phenomenologically induced large containment failure modes result in high magnitude releases. Conversely, Sensitivity Case #1 (the optimistic case) results in lower frequencies of high magnitude releases because the likelihood of phenomenological failure modes is less.

Based on these sensitivity runs the frequency of large releases ranges from 6.1E-7 to 6.8E-6. The base case (i.e., the best estimate case) results indicate that the frequency of large releases at NMP2 is closer to the lower end of the spectrum.

The results of the sensitivity studies are graphically summarized in Figure 4.9-10. This figure shows the range of the release frequencies for certain release categories. For H/E releases, the uncertainty range is seen to be approximately an order of magnitude. The high magnitude release frequency is more tightly clustered, and shows very little dispersion. The reason for the large dispersion in the high-and-early (H/E) releases is due to large variations in the frequency of the energetic containment challenge phenomena. Primary containment failures due to large energetic phenomena are modeled in most cases as resulting in a H/E release; therefore, varying the frequency of these phenomena has a direct effect on the frequency of H/E releases.

Releases for Moderate (M) or Low (L, LL) releases are not substantially affected by changes in the energetic failure modes that primarily influence the high/early category.

Table 4.9-1

NRC IDENTIFIED PARAMETERS FOR SENSITIVITY STUDY
(NUREG-1335)

- Performance of containment heat removal systems during core meltdown accidents.

- In-vessel phenomena (primary system at high pressure).

  - H_2 production and combustion in containment
  - Induced failure of the reactor coolant system pressure boundary.
  - Core relocation characteristics
  - Mode of reactor vessel melt-through

- In-vessel phenomena (primary system at low pressure)

  - H_2 production and combustion in containment
  - Core relocation characteristics
  - Fuel/Coolant interactions
  - Mode of reactor vessel melt-through

- Ex-vessel phenomena (primary system at high pressure)

  - Direct containment heating concerns
  - Potential for early containment failure due to direct contact by core debris
  - Long-term core-concrete interactions:

    - Water availability
    - Coolable or not coolable

- Ex-vessel phenomena (primary system at low pressure)

  - Potential for early containment failure due to direct contact by core debris.
  - Long-term core-concrete interactions:

    - Water availability
    - Coolable or not coolable

Table 4.9-2

LIST OF SENSITIVITY ITEMS

Sensitvity Item	Required by GL88-20 or NUREG-1335	Deemed Useful in NMP2 IPE Response	Proposed Cases for Accident Management Investigations
IN-VESSEL CORE MELT PROGRESSION			
- Hydrogen Production	X	X	
- Temperature of Melt			
- Model for Control Rods			
- Model for Candling			
- RPV Breach Model and Assumptions	X	X	
- Amount of Debris Retained In-vessel		X	
- In-vessel Steam Explosion	X	X(P)	
- Induced Primary System LOCAs	X	X(P)	
- In-vessel Recovery		X(P)	
- In-vessel Reactivity Excursion		X(P)	X
- Revaporization of Deposited Fission Products	X	X	
EX-VESSEL CORE MELT PROGRESSION			
- Debris Temperature	X	(1)	
- Amount of Debris Discharged From Vessel		X	
- DW Sump Coolability		N/A	
- Coolability with Water Present	X	X	X
- Effective DW Floor Area		X	
- Pool Bypass			
-- Vacuum Breaker		X	
-- Downcomers		X	
-- Other		N/A	
- Quenching Model in Pool (MKII)	X	X	
- DCH		X(P)	
- Amount of Material		X	
-- Retained in Drywell			
-- Retained in Pedestal			
- Suppression Pool Mixing		X	

(P) Probabilistic Sensitivities Performed.

Table 4.9-2

LIST OF SENSITIVITY ITEMS

Sensitivity Item	Required by GL88-20 or NUREG-1335	Deemed Useful in NMP2 IPE Response	Proposed Cases for Accident Management Investigations
CONTAINMENT FAILURE			
- Size	X	X	
- Location	X	X	
- Pressure (Ultimate Capability)			
- Temperature			
- DW Equipment Mass		X	
- ATWS Induced Dynamic Containment Failure Mode			X
- Containment Venting		X	X
- Aerosol Plugging			
- Direct Contact of Debris	X	N/A	
- Pressure Rise	X	X	
REACTOR BUILDING EFFECTIVENESS	X	X	
- Hydrogen Burn			
- Circulation Established			
- Direct Release			
CRITICAL SAFETY FUNCTIONS			
- Reactivity Control			
- Pressure Control			X
- High Pressure Makeup			X
- Depressurization			X
- Low Pressure Makeup			X
- Containment Heat Removal	X	X	X
- Containment Temperature Control			X
- Containment Pressure Control			X
- Combustible Gas Control			X
- Containment Water Level Control			X
- Containment Flooding			X
- Drywell Spray Use			X

Table 4.9-2

LIST OF SENSITIVITY ITEMS

Sensitivity Item	Required by GL88-20 or NUREG-1335	Deemed Useful in NMP2 IPE Response	Proposed Cases for Accident Management Investigations
OTHER ACTIONS			
- Accident Management Actions			X
- Disregard DWSI Curve			X
- Containment Flood Always by Procedure			X
- Containment Flood with No RPV Vent		X	
- Containment Flood Only Late in Sequence		X	X
- Fill DW with Water (MKI)		N/A	
- Vent to 0 psig		X	X
- Vent to Control 40-60 psig		X	X
- Vent to Control 60-90 psig			X

(1) The debris temperature being discharged from the RPV is higher than the initial debris discharge calculated using codes such as BWSAR (see NUREG/CR-5565). Therefore, the MAAP assessment is judged to be sufficiently conservative to provide an acceptable basis for these thermal hydraulic response of containment.

Intentionally
Left
Blank

Table 4.9-3

SENSITIVITY TO DEBRIS RETAINED IN-VESSEL

	FMAXCP	Vessel Failure Time (Hrs.)	Core Fraction In RPV at 60 Hrs.	Time at Which Containment Reaches 45 psig (Hrs)	Vent	Containment Failed (Hrs)[1]	CSI Released at 60 Hrs.
ID2LD	0.1	1.5	0.34	N/A	No	49.9 (RHR Cooling)	0.16
ID2LDCP	0.8	1.5	0.0	> 60.0	No	> 60.0 (RHR Cooling)	0.00
ID8LD	0.1	1.5	0.29	6.4	Yes	38.6	0.32
ID8LDCP	0.8	1.5	0.0	7.6	Yes	>60.0	0.008
IA4LDNP	0.1	2.7	0.11	3.0	No	31.6	0.15
IA4LDNCP	0.8	2.7	0.0	4.0	No	34.2	0.015

[1]Based on exceeding drywell pressure and temperature limitations as shown in Figure 4.4-8.

Table 4.9-4

MARK II SHORT-TERM STATION BLACKOUT SCENARIO SUMMARY

Case	ADS Actuated	Drywell Sprays Activated?	Debris Split Fraction		Time to Containment Failure (hr)	Time to Drywell Floor Failure	Wetwell Pressure at Time of Floor Failure (psia)
			% Drywell	% Wetwell			
ST-1	yes	no	100	0	N/A	13.5	117
ST-2	no	no	100	0	10.0	12.2	173
ST-3	yes	yes	100	0	N/A	16.6	145
ST-4	yes	no	95	5	N/A	14.1	120
ST-5	yes	no	80	20	> 20.4	> 20.4	134[1]
ST-6	yes	no	60	40	> 26.7	> 26.7	140[2]
ST-7	yes	no	0	100	8.5	> 12.9	145
LT-1	no	no	100	0	20.8	> 22.1	165[3,4]

(1) Wetwell pressure at 20.4 h.
(2) Wetwell pressure at 26.7 h.
(3) Wetwell pressure at 22.1 h.
(4) Note: All containment failure mechanisms were disabled for the long-term station blackout calculation.
(5) Containment failure due to overpressure at 135 psi is not modeled in these computer simulations.

Table 4.9-5

SENSITIVITY TO DELAYED VESSEL BREACH

	Vessel Failure Time (Hrs.)	Containment Failure Time (Hrs.)	Core Fraction Retained In-Vessel Immediately After VF	Injection After Vessel Failure	Containment Heat Removal Available	CSI Release to Environment
IA1LD	2.7	2.7	0.45	Yes	Yes	0.05
IA1LDDB	3.4	N/A	0.37	Yes	Yes	0.00
IVAILD	2.1	1.1	0.69	No	No	0.27
IVAILDDB	6.3	1.1	0.43	No	No	0.07

Table 4.9-6

SENSITIVITY TO DEBRIS RETAINED IN PEDESTAL

	Core Fraction Long Term				Concrete Attack in Pedestal (Ft.)	Vessel Failure Time (Hrs.)	Containment at 45 psig/Vent ? (Hrs.)	Containment Failed[1] (Hrs.)	Injection After Vessel Failure?	CSI Released to Environment at 60 Hrs.
	In RPVQ	In PD	In DW	In WW						
IA4LDNP (Base)	0.11	0.13	0.25	0.51	0.03	2.7	2.7/No	31.6	No	0.11
IA4LDXNC	0.11	0.25	0.43	0.21	0.34	2.7	2.7/No	24.0	No	0.09
ID8LD (Base)	0.29	0.11	0.0	0.60	0.02	1.5	6.4/Yes	38.6	No	0.20
ID8LDXC	0.29	0.17	0.0	0.54	0.60	1.5	5.4/Yes	32.0	No	0.15

[1] Based on exceeding drywell pressure and temperature limitations as shown in Figure 4.4-8.

Table 4.9-7

SENSITIVITY TO SUPPRESSION POOL MIXING AND CONTAINMENT HEAT REMOVAL STATUS

MARK II CONTAINMENT FAILURE TIMING

CASE	System Status				Time to Containment Failure	
	Wetwell Cooling RHR	Drywell Spray Thru HX	RPV Injection	Mixing of Wetwell and Pedestal Water	MAAP	Hand Calculation
1	No	No	No	Yes	34 Hrs. IA-4-LD-NCP	32 Hrs.
2	No	No	No	No/Partial	---	4 Hrs./24 Hrs.
3	Yes	No	No	Yes	No Failure ID-2-LD-CP	---
4	No	No	No	Yes	No Failure ID-2-LD-CP	---

TABLE 4.9-8

SENSITIVITY TO PEDESTAL DOWNCOMER FAILURE (POOL BYPASS)

	Vessel Failure Time(hrs)	Pedestal Downcomer Failure Time(hrs)	Containment Failure Time (hrs)[1] Location	Injection After Vessel Failure	CsI Retained In Vessel	CsI in Wetwell	CsI in Drywell	CsI Release from Containment
IA1LD	2.7	@VF	2.7/DWH	YES	0.38	0.45	0.09	0.08
IA1LDNP	2.7	No Failure	N/A	YES	0.38	0.51	0.11	0.00
IA4LDXNC	2.7	No Failure	11.2/DWH[2]	NO	0.0	0.52	0.34	0.14
IA4LWXNC	2.7	No Failure	11.2/WWA[2]	NO	0.06	0.55	0.38	0.0003[3]
IA4LWXN7	2.7	@VF+7 min.	7.0/WWA[2]	NO	0.10	0.55	0.34	0.015
ID8LDNPV	1.5	No Failure	23.7/DWH	NO	0.06	0.28	0.30	0.359
ID8LWNPV	1.5	No Failure	23.7/WWA	NO	0.25	0.35[3]	0.39	0.007[3]
ID8LWN7V	1.5	@VF+7 min.	10.0/WWA	NO	0.41	0.27	0.27	0.045
IVA1LW	2.0	@VF	1.1	NO	0.25	0.34[3]	0.18	0.23[3]
IVA1LWNP	2.0	No Failure	1.1	NO	0.29	0.42[3]	0.26	0.032[3]
IVA2LW	3.5	@VF	1.1	NO	0.10	0.51[3]	0.11	0.28[3]
IVA2LWNP	3.5	No Failure	1.1	NO	0.10	0.67	0.20	0.024[3]
IVA2LWN7	3.5	@VF+7 min.	1.1	NO	0.10	0.64	0.18	0.08[3]

[1] Based on exceeding drywell pressure and temperature limitations as shown in Figure 4.4-8.
[2] Containment Failure was forced to occur at 74.0 psia in wetwell and 680°F in the drywell (inadvertently below limits in Figure 4.4-8).
[3] After adjustment to account for Saturated Pool DFs (see Section 4.7.3.10).

Table 4.9-9

SENSITIVITY TO CONTAINMENT FAILURE SIZE

	Containment Failure Size (ft^2)	Vessel Failure Time (hrs)	Containment Failure Time (hrs)	CsI to Reactor Building
IA1LD	2.0	2.8	2.8	0.08
IA1SD	0.194	2.8	2.8	0.05
IIA1LD	2.0	35.0	31.6	0.47
IIA1SD	0.194	38.0	31.6	0.48
IIA2LD	2.0	39.8	31.6	0.64
IIA2SD	0.194	39.6	31.6	0.46
IIT1LD	2.0	30.3	30.3	0.48
IIT1SD	0.194	30.3	30.3	0.24
IIA1LW	2.0	35.2	31.6	0.45
IIA1SW	0.194	38.6	31.6	0.19
IIA2LW	2.0	39.5	31.6	0.40
IIA2SW	0.194	39.9	31.6	0.19
IIT1LW	2.0	30.3	30.3	0.41
IIT1SW	0.194	30.3	30.3	0.11

Table 4.9-10

SENSITIVITY TO CONTAINMENT FAILURE LOCATION

	Containment Failure Location[1]	Pool Bypass	Vessel Failure Time (hrs)	Containment Failure Time (hrs)	CsI Released	
					To Reactor Building	To The Environment
IA7LD	DWH	YES	2.3	0.0	0.0136	0.0006
IA7LW	WWA	YES	2.3	0.0	0.0002	0.0001
ID2LD	DWH	NO	1.5	49.9	0.16	0.12
ID2LW	WWA	NO	1.5	49.9	0.012	0.003
IIA1LD	DWH	YES	35.0	31.6	0.47	0.37
IIA1LW	WWA	YES	35.2	31.6	0.45	0.09
IIA2LD	DWH	YES	39.8	31.6	0.64	0.43
IIA2LW	WWA	YES	39.5	31.6	0.40	0.13
IIT1LD	DWH	YES	30.3	30.3	0.48	0.21
IIT1LW	WWA	YES	30.3	30.3	0.41	0.31
IA4LDXNC	DWH	NO	2.7	11.2	0.06	0.04
IA4LWXNC	WWA	NO	2.7	11.2	4E-4	2E-4
ID8LDNPV	DWH	NO	1.5	23.7	0.356	0.167
ID8LWNPV	WWA	NO	1.5	23.7	0.007[2]	0.003[2]

(1) DWH = Drywell Head Failure
WWA = Wetwell Airspace Failure

(2) After reduction to account for saturated pool decontamination factors as described in Section 4.9.3.10.

Table 4.9-11

SENSITIVITY TO DRYWELL EQUIPMENT MASS

	Assumed Drywell Equipment Mass (lb)	Containment Failure Time (hrs)[1]	Containment Failure Conditions		CsI Released To Reactor Building
			Pressure	Temperature	
IA4LDNP	.5 E6	31.6	95 psig	680°F	0.12
IA4DN2	1 E6	32.5	96	675°F	0.12
IA4DN10	5 E6	42.0	102 psig	625°F	0.06

[1] Based on exceeding pressure and temperature limitations as shown in Figure 4.4-8, and the interpolation of the MAAP run graphical traces.

Table 4.9-12

ADJUSTED RELEASES FOR CASES WITH SATURATED POOL CONDITIONS

	MAAP UNADJUSTED		MODIFIED MAAP[1]	
	To Reactor Building	To Environment	To Reactor Building	To Environment
ID1LD	0.011	0.011 (M)	0.001	0.001 (L)
ID8LWNPV	0.065	0.031 (M)	0.007	0.003 (L)
IA4LWXNC	0.003	8E-4 (LL)	3E-4	8E-6 (LL)
IIA1LW	0.446	0.091 (M)	0.28	0.07 (M)
IIA1SW	0.186	0.030 (M)	0.12	0.03 (M)
IIA2LW	0.403	0.131 (H)	0.35	0.12 (H)
IIA2SW	0.194	0.029 (M)	0.13	0.02 (M)
IIT1LW	0.407	0.314 (H)	0.407	0.314 (H)
IIT1SW	0.106	0.030 (H)	0.106	0.030 (M)
IVA1LW	0.425	0.137 (H)	0.23	0.08 (M)
IVA2LW	0.411	0.116 (H)	0.28	0.07 (M)
IVA1LWNP	0.321	0.105 (H)	0.032	0.010 (M)
IVA2LWNP	0.244	0.062 (M)	0.024	0.006 (L)
IVA2LWN7	0.278	0.069 (M)	0.08	0.02 (M)
IVA2LWA4	0.412	0.120 (H)	0.28	0.07 (M)

(1) The MAAP results are modified to take into account a minimum suppression pool DF of 10 even for saturated pool conditions.

Table 4.9-13

SENSITIVITY TO CONTAINMENT VENT MODELING

	Vent Strategy	Vessel Failure Time (hrs)	Vent Time (hrs)	CsI Release
IIIC1LD	Full Open at 45 psig	0.7	1.3	0.22
IIIC1V10	Full Open at 45 psig + 10 minutes	0.7	1.5	0.21
IIIC1V56	Cycled to maintain Pressure between 35 and 45 psig	0.7	1.3 initially	0.04

Table 4.9-14

SENSITIVITY TO CONTAINMENT FAILURE LOCATION IN THE REACTOR BUILDING

	Containment Interface Failure[(1)]	Vessel Failure Time (hrs)	Containment Failure Time (hrs)	CsI Released		Reactor Building DF
				To Reactor Building	To Environment	
DRYWELL FAILURE CASES						
IA1LD	7	2.8	2.8	0.08	0.05	1.6
IA4LDXNC	7	2.7	11.2	0.06	0.04	1.5
IIA1LD	7	35.0	31.6	0.47	0.37	1.3
IIA2LD	7	39.8	31.6	0.64	0.43	1.5
IIT1LD	7	30.3	30.3	0.48	0.21	2.3
IVA1LD	7	2.0	1.1	0.50	0.26	1.9
IVA2LD	7	3.5	1.1	0.47	0.27	1.7
ID8LDNPV	7	1.5	23.7	0.36	0.17	2.1
WETWELL FAILURE CASES						
IA4LWXNC	1	2.7	11.2	4E-4	2E-4	2.0
IIA1LW	1	35.2	31.6	0.45	0.09	5.0
IIA2LW	1	39.5	31.6	0.40	0.13	3.1
IIT1LW	1	30.3	30.3	0.41	0.31	1.3
ID8LWNPV	1	1.5	23.7	0.007 0.003		2.3
IA4IS3NP	1	2.8	0.0	0.018	0.04	4.5

(1) Node 1 (RHR Room - usually from wetwell airspace failure)
Node 7 (328' elevation - usually from drywell head failure)

Table 4.9-15

SENSITIVITY TO REACTOR BUILDING MODELING ASSUMPTIONS
(RAILROAD DOOR FAILURE)

	Containment Failure Time (hrs)	Vessel Failure Time (hrs)	Blowout Panels Fail (hrs)	Railroad Doors Fail (hrs)	Other Failures to Environment	CsI Release		Reactor Building DF
						To Reactor Building	To Environment	
IA4IS3NP	0.0	2.8	1.3	1.3	No	0.18	0.04	4.5
IA4I3NRR	0.0	2.8	1.3	N/A	No	0.16	0.003	47.6

Table 4.9-16

INFLUENCES ON THE RADIONUCLIDE RELEASE FOR TD = F SEQUENCES

Case	Release Severity/Time	DW Equipment Mass	Residual Fuel Mass in RPV/DW (long term)	Duration of Accident	Size of Failure[2]
IA4LDXNC	M/I	Base 220,000 kg	~ 11%	36 Hrs.	Large
IA4LDXNC[1]	H/I	Base 220,000 kg	~ 11%	48 Hrs.	Large
IA4LDNP	H/I	Base 220,000 kg	11/25%	36 Hrs.	Large
IA4LDNCP	LL/L	Base 220,000	0%	36 Hrs.	Large
IA4DN10	M/L	2,200,000	> 10%	60 Hrs.	Large
ID8LDNPV	H/I	Base	34%	36 Hrs.	Large
IA4DN10A	H/I	440,000		60 Hrs.	Large

[1] Same as 1st case except calculation covers 48 hrs. instead of 36 hrs.

[2] Inferences from other runs indicate that even the smaller containment failure size will yield similar consequential results.

Table 4.9-17

EPRI Recommended Senditivity Cases

Parameter Name	NMP2 Baseline Value	Parameter Purpose	EPRI Sensitivity Recommendation	
			Baseline	Range
FCRBLK	0.0	Local core blockage model used ot model core damage. This produces a best estimate of the hydrogen production	0.0	-1, +1
FMAXCP	0.1	Defines the minimum amount of retained debris in the RPV allowed before the model assumes all materia is discharged from the RPV.	0.1	0.1, 0.8
FCHF	0.1	MAgnitude of the heat flux from the debris to an overlying pool of water	0.1	0.02, 2.0
ADWF	5200 ft^2	Effective Drywell Area	Plant Specific	¼ - 1 Total Area
---	Range	Containment Failure Location	Range	Range Includes: - Small 0.194 ft^2 - Large 2 ft^2 (These are NMP2 unique values)
	Range	Containment Failure Location	Range	Range Includes: - Drywell Head - Wetwell Airspace - Wetwell below water line
	Plant Specific	Reactor Building Model		

Table 4.9-18

SUPPLEMENTAL SENSITIVITY CASES

Issue	Parameter Purpose	EPRI Sensitivity Recommendation	
		Baseline	Range
Instrument Tube Penetration Failure Upon Melting	RPV Breach Model - 2 NMP2 cases with delayed breach until lower plenim boiloff	Instrument Penetration	Delayed Vessel Breach
CCI	Amount of Debris Retained in the Mark II Pedestal	0.16 ft.	1.0 ft.
Wetwell Pool Volume	Suppression Pool Mixing	Total Wetwell Pool Volume	Pedestal Only Hand Calculation
Pool Bypass	Delay in Downcomer Failure	0 min.	Downcomer failure included as follows: - No Downcomer Failures - Downcomers Fail at RPV Breach - Downcomers Fail at RPV Breach + 7 min.
Drywell Heat Up	Uncertainty in Drywell Equipment Mass	220,000 kg (NMP2 Specific)	Range Includes: - 440,000 kg - 1,100,000 kg (NMP2 Sensitivity)

Table 4.9-19

SUMMARY OF PROBABILITIES OF PHENOMENOLOGICAL CONTAINMENT FAILURE MODES

Phenomena	Best Estimate Value	Range
H_2 Deflagration	5E-3	0.0 - 1E-2
In-Vessel Steam Explosion	1E-4	1E-4 - 1E-2
Ex-Vessel Steam Explosion	1E-3	1E-4 - 1E-2
Direct Containment Heating (DCH)	1E-3	1E-2 (upper bound)
Debris Impingement	1E-6	0 - 1.0[1]
High Pressure Blowdown Overwhelms Vapor Suppression	1E-3	1E-3
Pedestal Failure Causes RPV Collapse	1E-6	0.0 - 1.0
Missiles Generated During Blowdown Pierce Drywell	1E-4	N/A
Downcomers Rupture During High Pressure Blowdown	3.74E-5	N/A
Containment Flooding Induced Containment	0.5	1E-3 - 1.0

[1] Does not apply to the NMP2 containment configuration. This range, taken from other studies, applies to Mark I primary containments.

Table 4.9-20

SUMMARY OF PROBABILITIES USED IN THE SENSITIVITY CASES

Phenomena	Optimistic Sensitivity Case #1	Pessimistic Sensitivity Case #2
H_2 Deflagration	1E-3	5E-3
In-Vessel Steam Explosion	1E-4	1E-2
Ex-Vessel Steam Explosion	1E-4	1E-2
Direct Containment Heating (DCH)	1E-3	1E-2
Debris Impingement	1E-6	1E-2
High Pressure Blowdown Overwhelms Vapor Suppression	1E-3	1E-2
Pedestal Failure Causes RPV Collapse	1E-6	1E-2
Missiles Generated During Blowdown Pierce Drywell	1E-4	1E-2
Downcomers Rupture During High Pressure Blowdown	3.74E-5	1E-2
Containment Flooding Induced Containment Failure	1E-3	1.0

Table 4.9-21

COMPARISON OF RELEASE FREQUENCIES OF BASE CASE AND SENSITIVITY CASES

Release Category	Base Case	Sensitivity Case #1 (Optimistic)	Sensitivity Case #2 (Pessimistic)
M/L	6.6E-7	6.6E-7	6.6E-7
M/I	1.4E-7	1.4E-7	1.4E-7
M/E	8.4E-7	8.4E-7	8.4E-7
L/L	1.8E-6	1.8E-6	1.8E-6
L/E	2.9E-7	2.9E-7	2.9E-7
L/I	2.6E-8	2.6E-8	2.6E-8
H/E	7.7E-7	6.1E-7	6.8E-6
H/L	3.8E-6	3.1E-6	5.2E-6
H/I	1.1E-5	1.1E-5	1.1E-5
LL	2.3E-6	2.3E-6	2.3E-6
Total Release	2.2E-5	2.1E-5	2.9E-5

* Release category totals that changed when compared to Base Case.

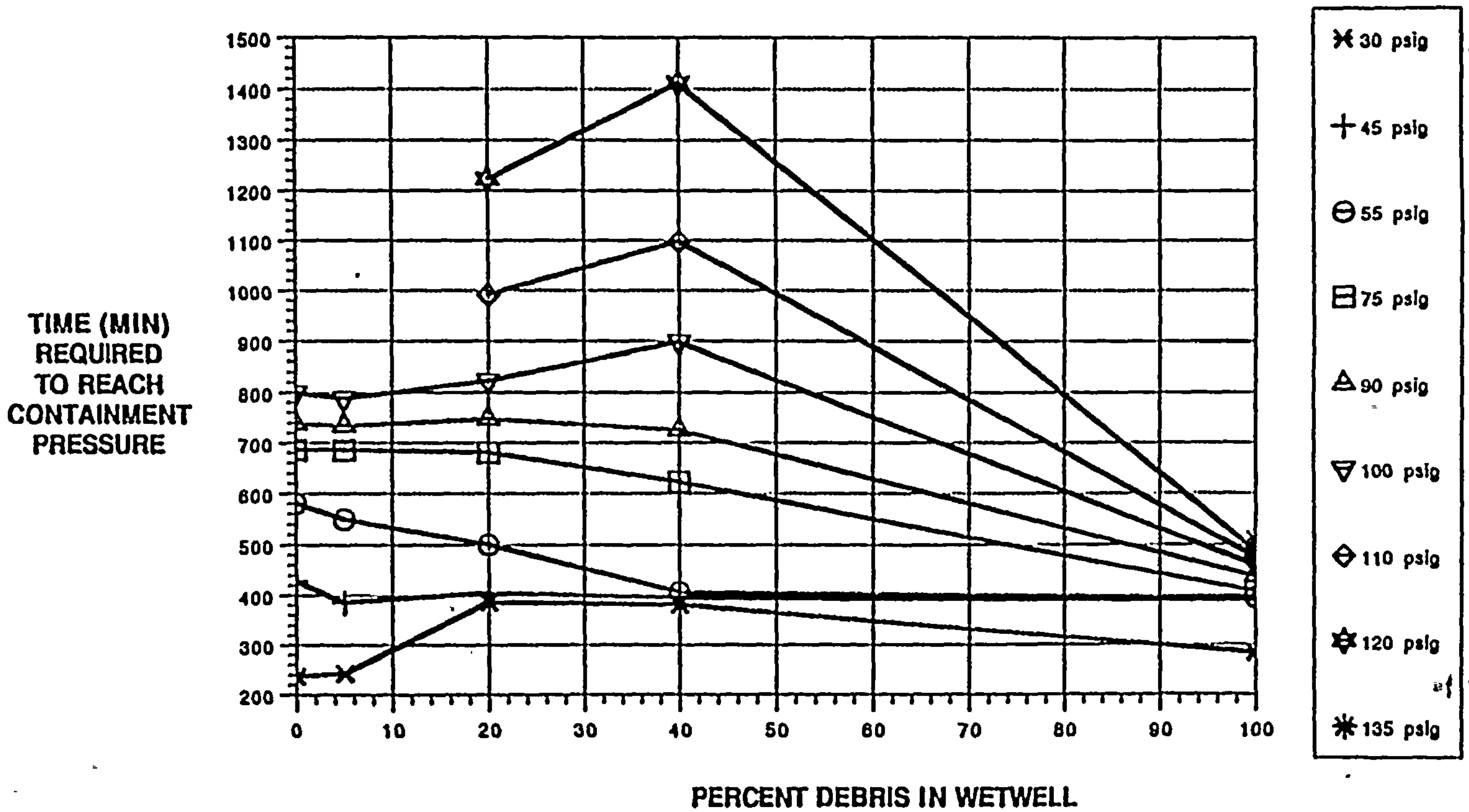


Figure 4.9-1 Effect of Debris-Wetwell Pool Interactions on Mark II Primary Containment (wetwell) Pressure Response

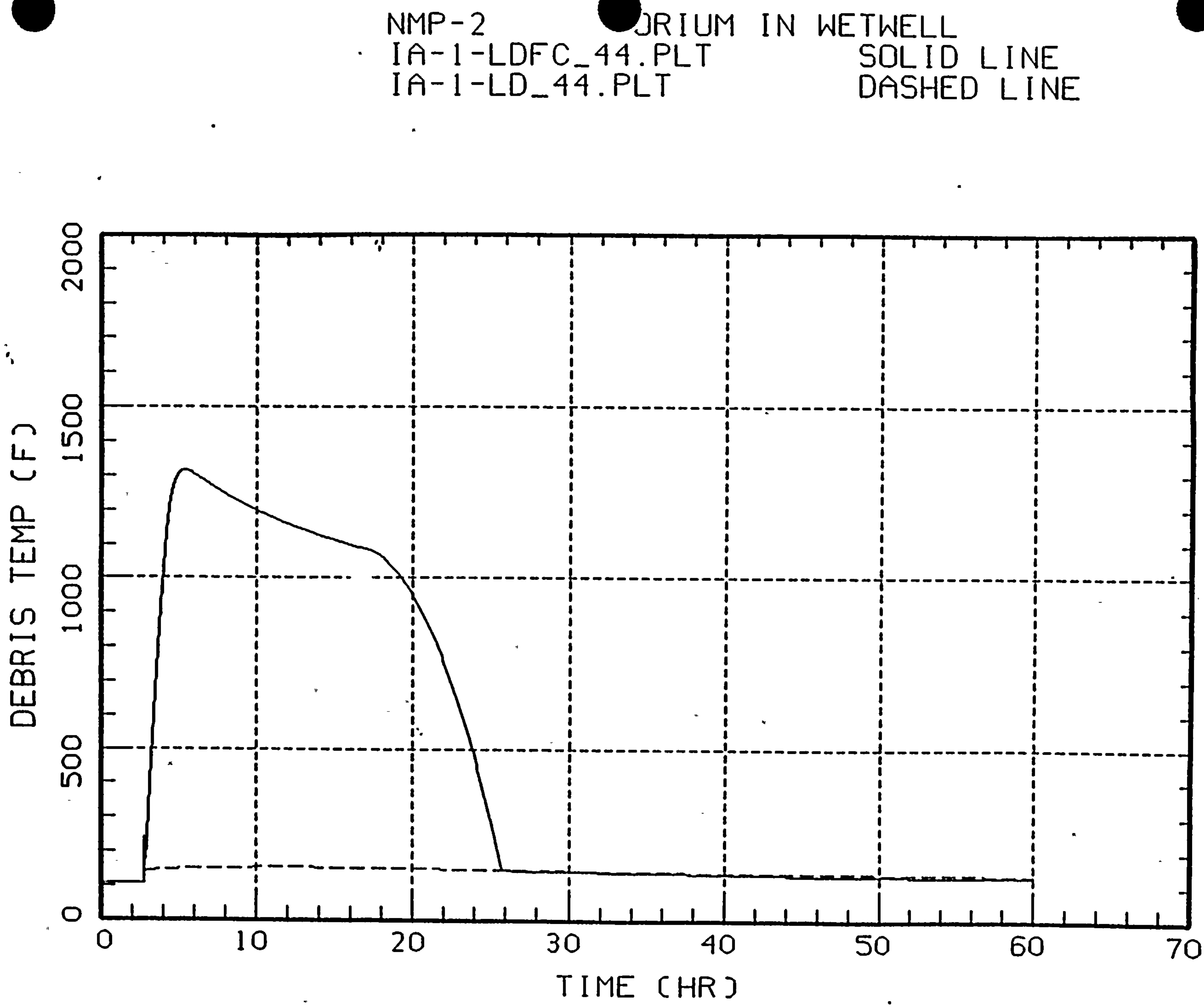


Figure 4.9-2 Comparison of the Debris Temperature in the Wetwell Using Two Different Heat Transfer Models for Debris Cooling

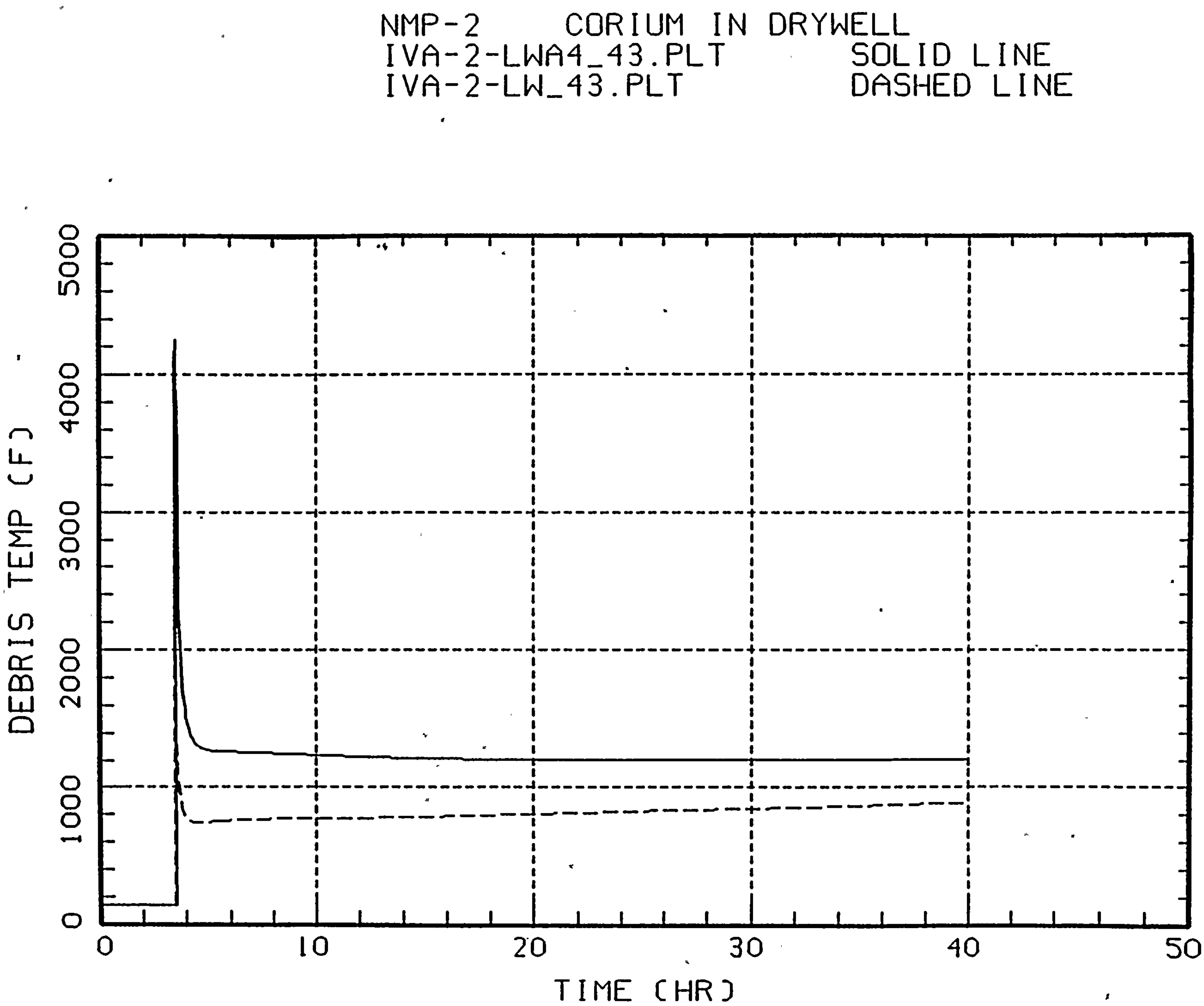


Figure 4.9-3 Comparison of the Debris Temperature in the Drywell as a Function of Amount of Drywell Floor Assumed Available for Entrained Debris

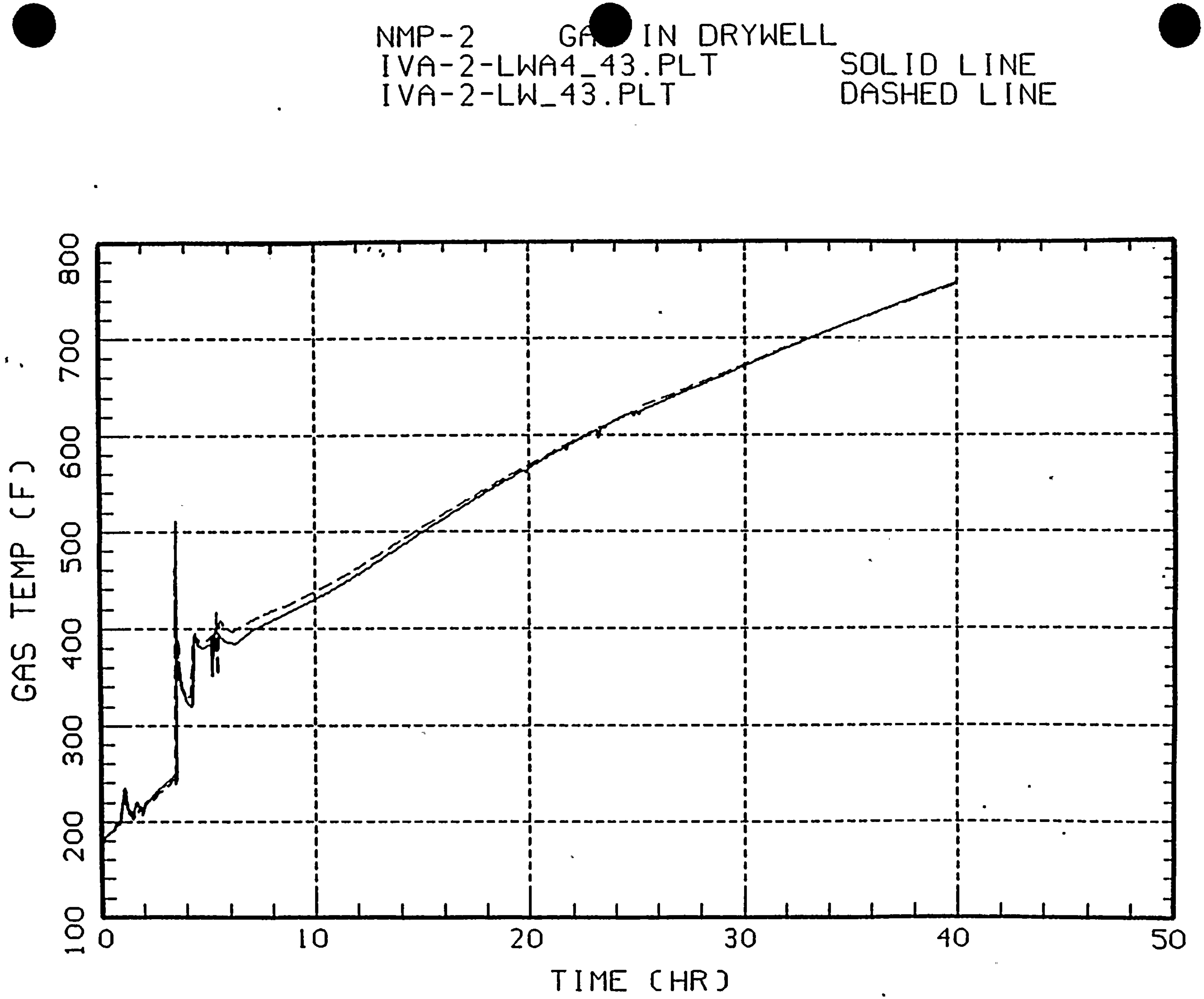
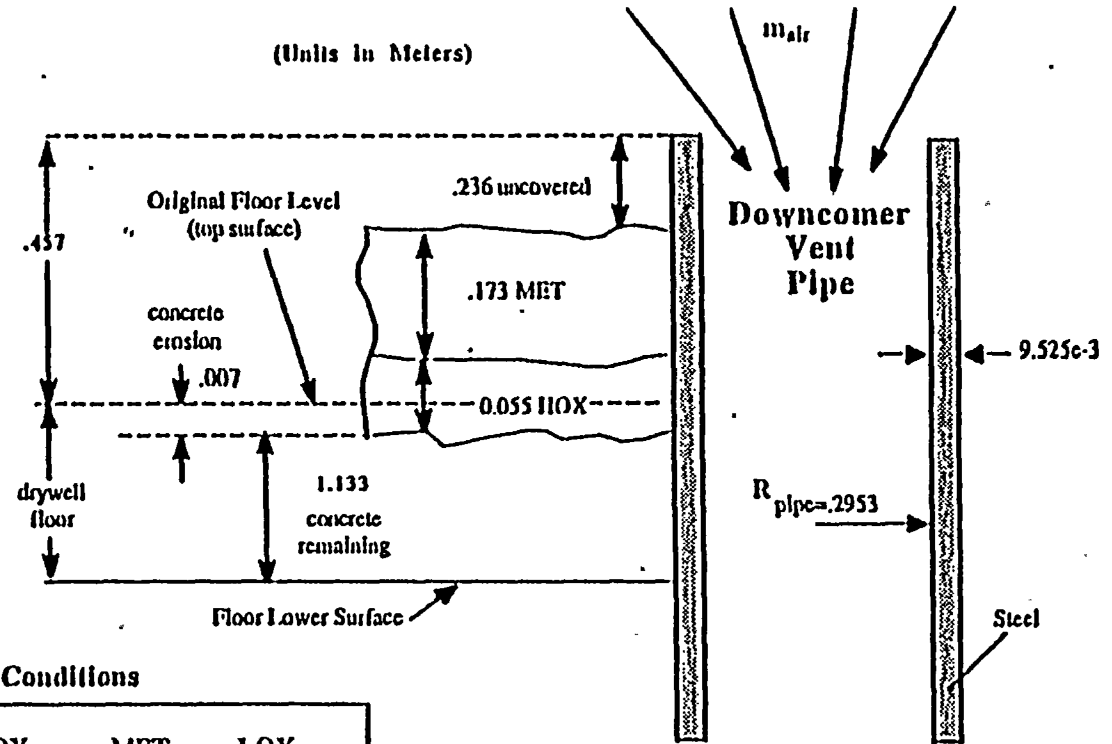


Figure 4.9-4 Comparison of the DW Gas Temperature for Two Cases of Entrained Debris Outside the Pedestal



Debris Conditions

	HOX	MET	LOX
Cp(J/Kg-K)	5.5313e2	6.2494e2	-
K(w/m-K)	2.9800e0	4.3105e1	-
m(Kg)	5.3620e4	1.3034e5	-
density(Kg/m3)	8.4455e3	6.4802e3	-
Qdecay(w)	4.8828e6	9.9649e4	-
Qchem(w)	0.0000e0	7.5678e6	-
Hside(w/m2-K)	0.0000e0	1.8300e3	-
crust side (m)	6.0835e0	3.4630e-1	-
crust top (m)	5.5307e-2	5.5268e-3	-
crust bot (m)	5.5305e-2	0.0000e0	-
Temp (K)	1.7619e3	1.8864e3	-
T solidus (K)	2.8483e3	1.8851e3	-

Concrete Temp. Distribution
X=distance from upper surface

X (m)	T (k)
0.0	425.8
0.0064	424.9
0.0224	375.3
0.0427	353.8
0.0833	337.6
0.1646	324.7
0.3272	318.3
0.5386	313.8

Containment Conditions

P = 3.8744e5 pa
$T_{pipe\ wall}$ = 758.53 K
T_{ATM} = 758.5 K
$\dot{m}_{air}$ = .0171 kg/sec

Figure 4.9-5 Drywell Floor Debris and Containment Conditions at 420 Minutes Without Drywell Floor Flooding for Short-Term Station Blackout with ADS

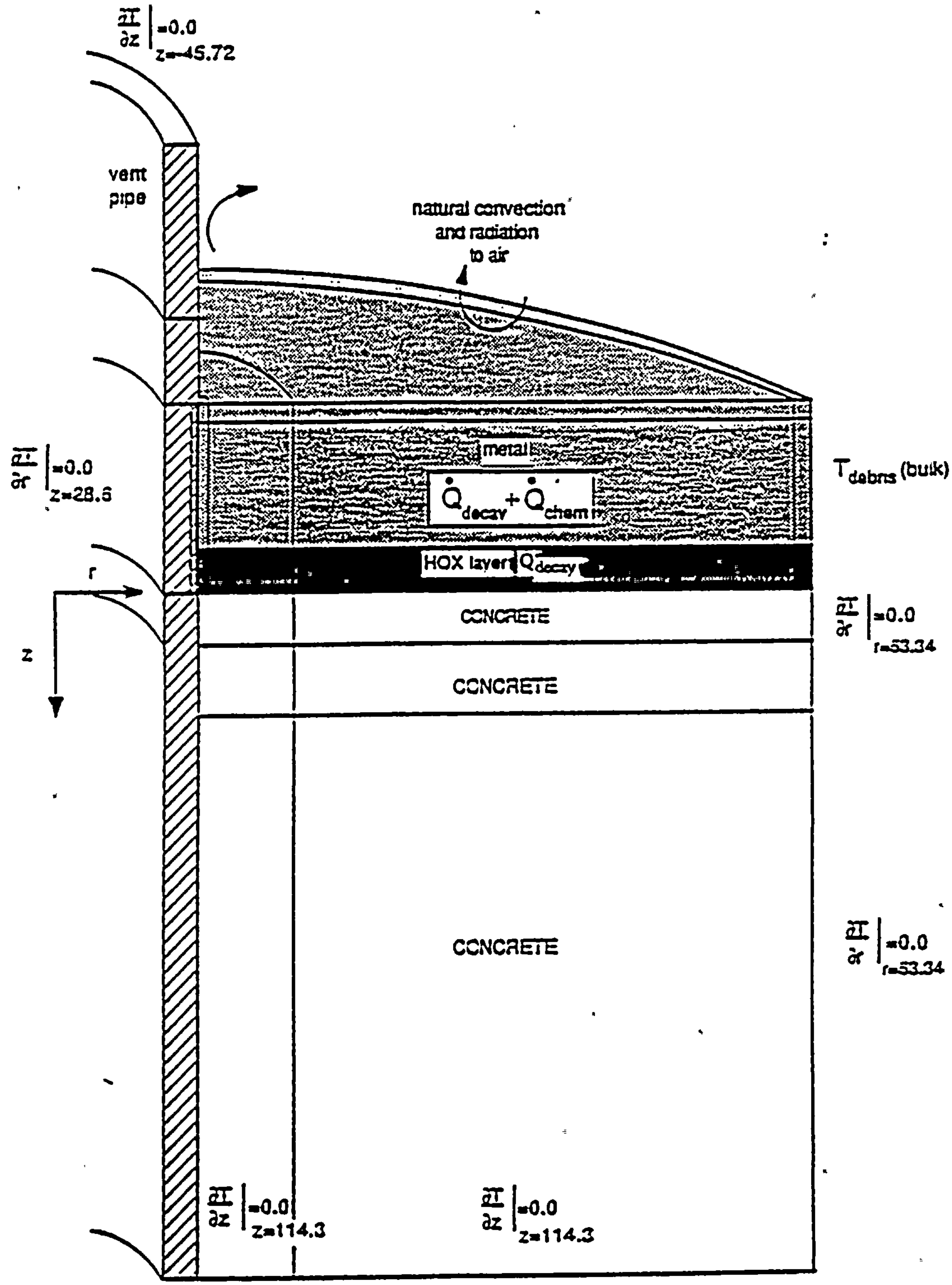


Figure 4.9-6 HEATING-7 Downcomer/Debris Model

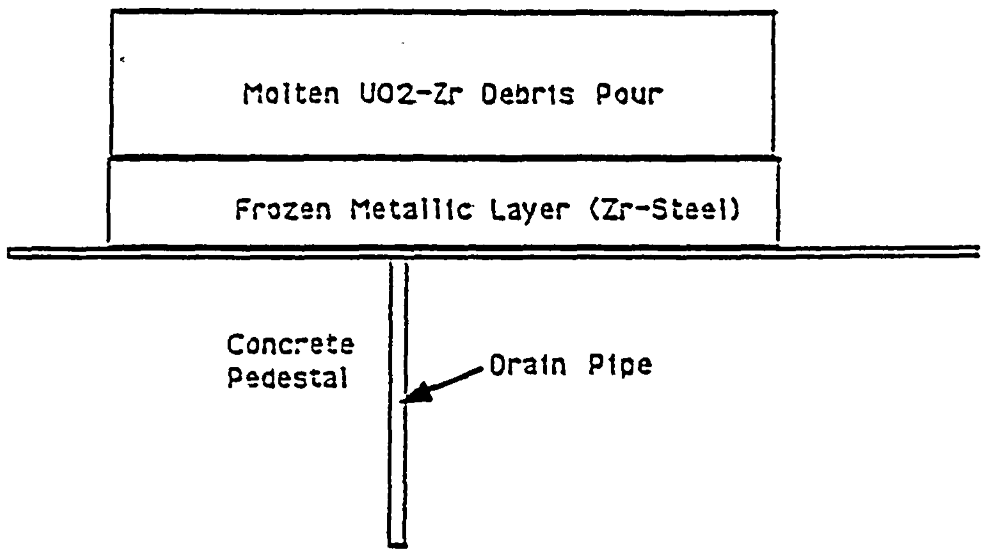


Figure 4.9-7 Stage 2 -- Scenario of Debris Drain Attack

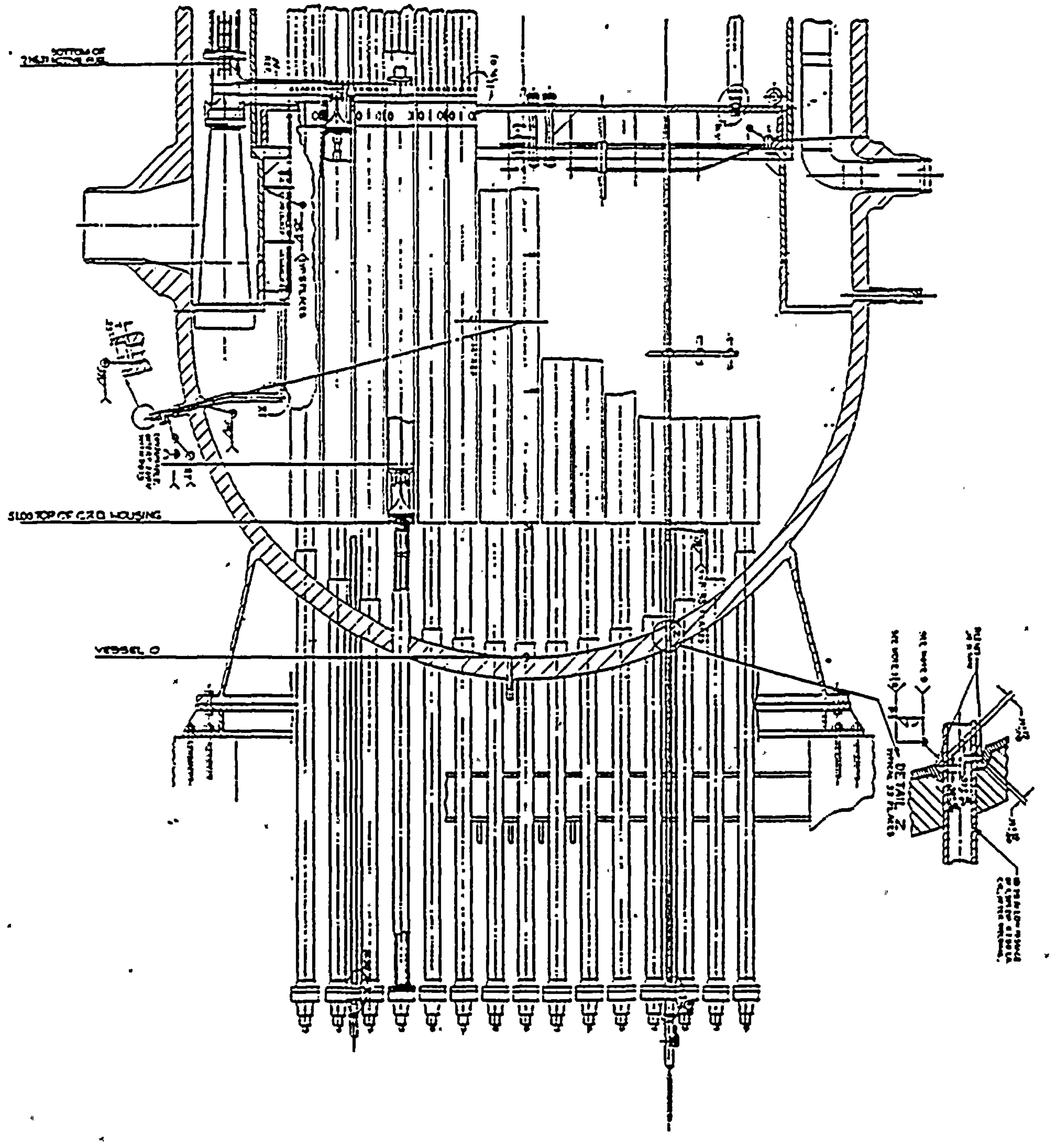


Figure 4.9-8 Diagram of the Reactor Vessel Bottom Head

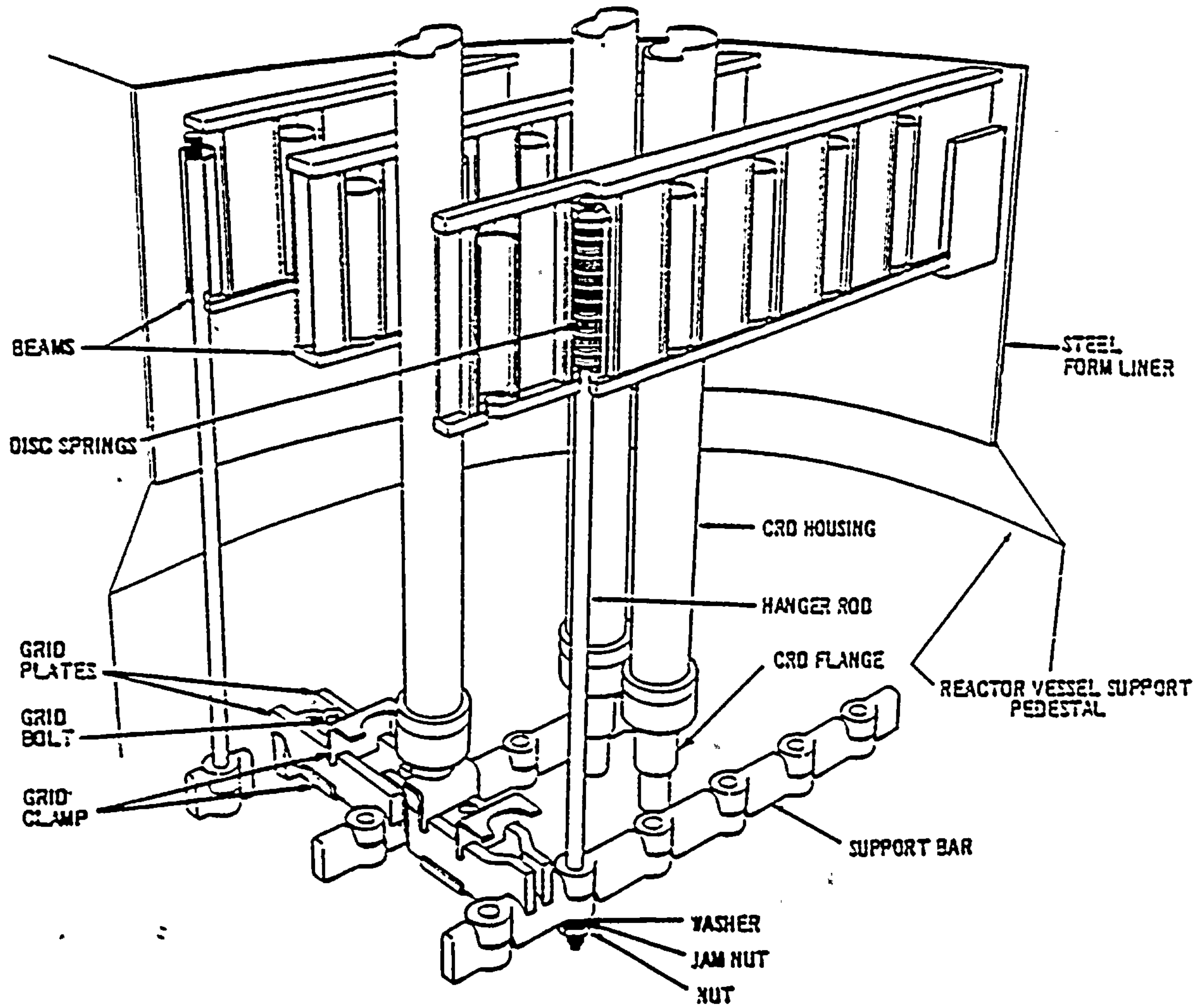


Figure 4.9-9 Diagram of the CRD Shoot-Out Steel

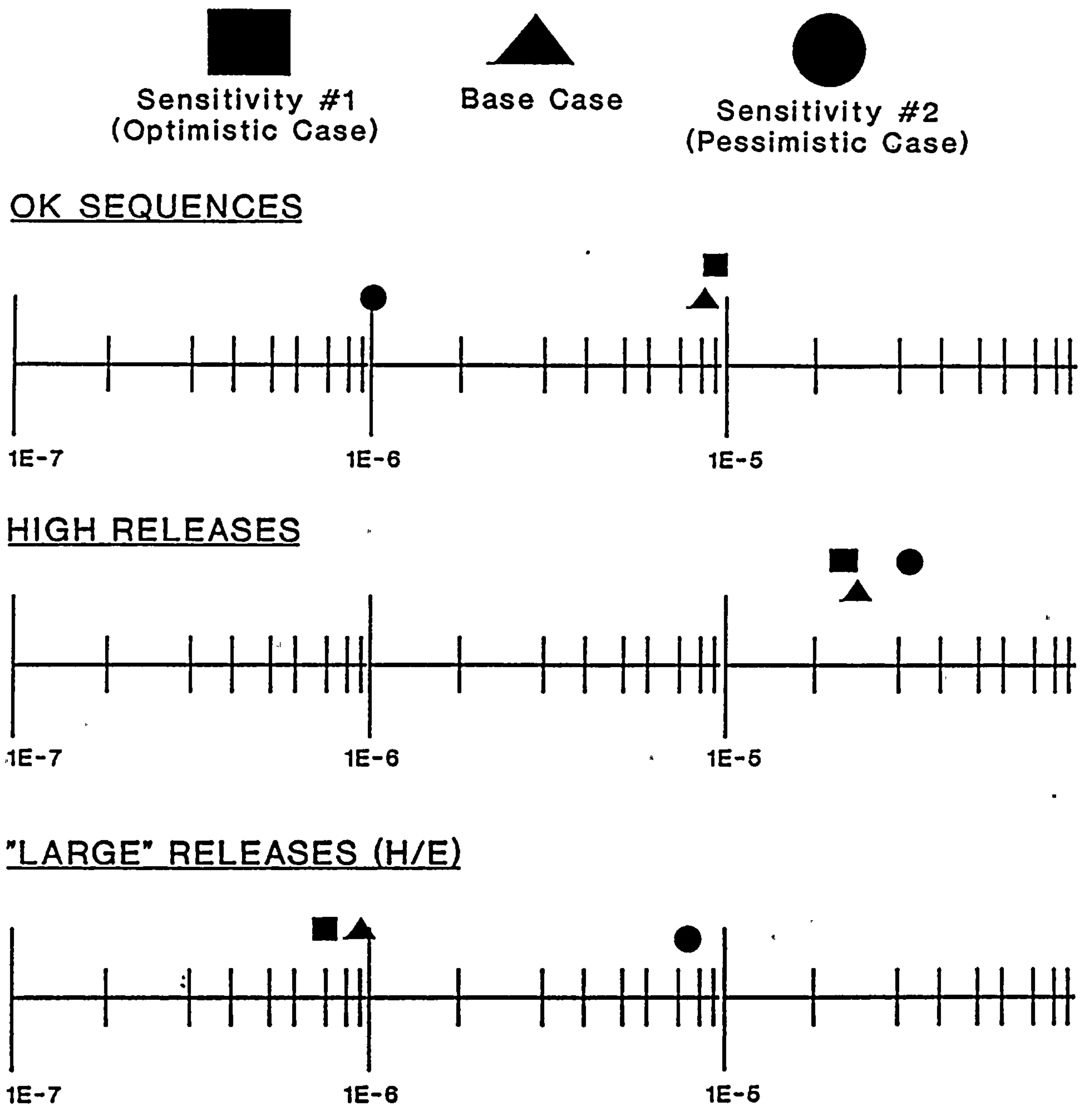


Figure 4.9-10 Results of NMP2 Sensitivity Studies

C1359106FX

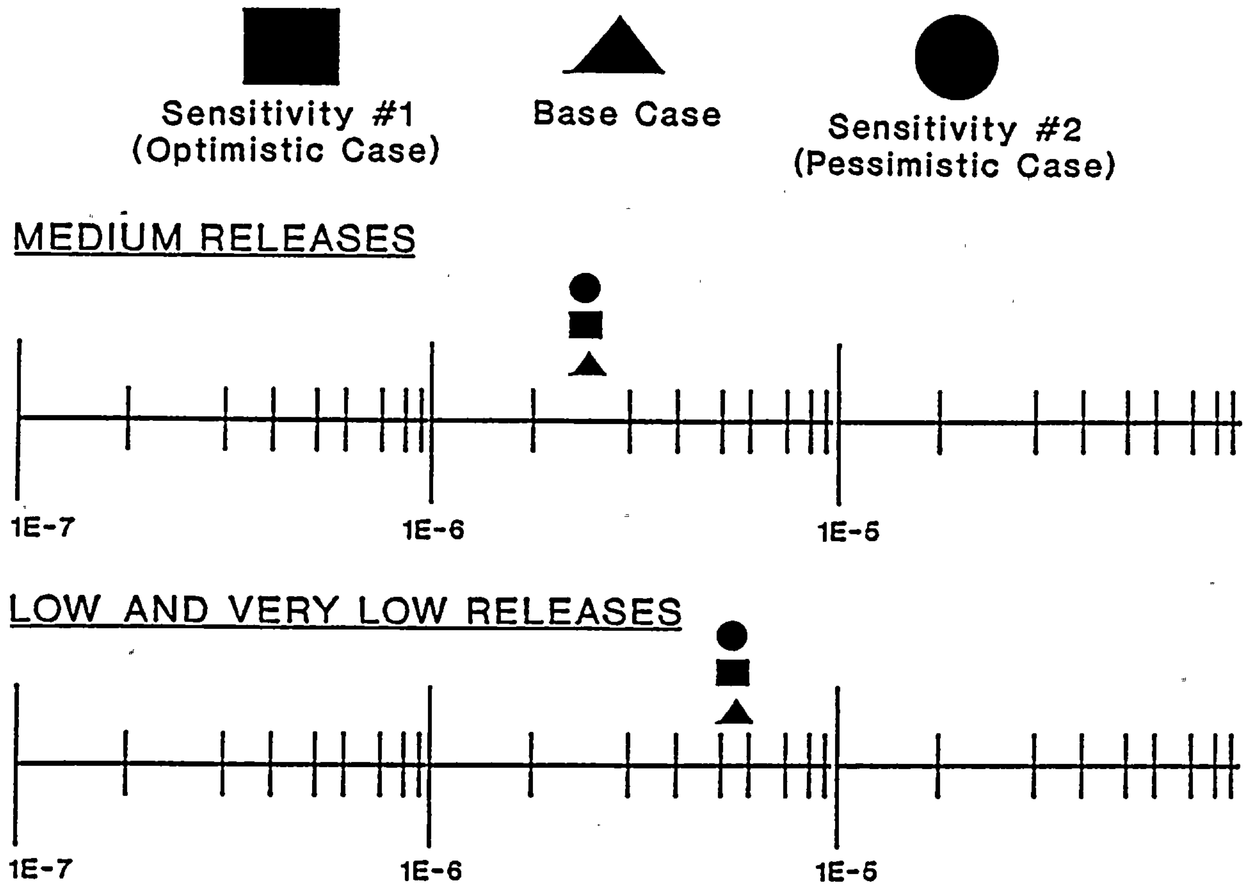


Figure 4.9-10 (con't) Results of NMP2 Sensitivity Studies

C1359106FX

5 UTILITY PARTICIPATION AND INTERNAL REVIEW

5.1 IPE Program Organization

Plans to develop Probabilistic Risk Assessments (PRAs) on both Nine Mile Point Units 1 and 2 began in 1989. Niagara Mohawk Power Corporation (NMPC) recognized that PRAs would be a major investment, but the benefits to the organization could also be significant. In addition, NMPC realized the most effective approach to developing a long term PRA capability to support future risk management and accident management activities is to commit a full time staff to developing the PRAs and ensure involvement from the entire organization. For these reasons, the following were established in 1990:

- 5 full time engineers were assigned to the PRA team.
- A project plan was developed to address the need for technical support from throughout the NMPC Nuclear Division. This support is crucial to ensuring that the PRA realistically represents the plant design and operation. At the same time, this involvement develops PRA awareness and knowledge throughout the organization.
- Quality Assurance (QA) and the Independent Safety Evaluation Group (ISEG) were identified as the primary departments responsible for ensuring independent review. Additional reviews further ensure technical accuracy and PRA awareness throughout the organization.

The Nuclear Technology Department was assigned primary responsibility for developing the PRAs and ensuring NMPC is involved in all aspects of the IPE. Five full time NMPC engineers were assembled in this department to carry out this responsibility. One of these full time members has operating experience. In addition, cognizant engineers from other departments were identified to provide technical support. Technical support included responding to the IPE team when additional details or knowledge of design, engineering, and operations were required.

Table 5-1 shows the overall organization of the IPE Program and Table 5-2 lists the PRA team cognizant engineers including their department and technical areas of support. A number college students participated in the study. These co-op students are listed in Table 5-3. Consultants used for the project are listed in Table 5-4. This table also delineates the area of support provided by each.

5.2 Composition of the Independent Review Team

An independent in-house review team was assembled to further assure technical accuracy, develop additional awareness and knowledge, and provide quality assurance to the process. The Quality Assurance Department (QA) and the Independent Safety Engineering Group (ISEG) were assigned responsibility for the independent in-house review. QA took the lead in organizing, planning, and documenting the reviews. ISEG provided an important technical resource that was, for the most part, independent of the PRA development process.

Most of the system analysis documentation received at least two reviews since early drafts were reviewed and then the final draft IPE report was reviewed. Reviewing early drafts supported the goal of developing awareness and knowledge throughout the organization by starting earlier in the process. In addition, the PRA team received informal input and support from other departments while developing the PRA and desired to have an early technical check. For these reasons, the list of reviewers included engineers involved with supporting the PRA development. However, there is no reason to believe their review was not objective and critical. In addition, there were sufficient technical reviewers to ensure an independent review. It should be pointed out that the tasks associated with developing a PRA are iterative as most tasks are dependent. Therefore, many of the early drafts were rough, incomplete and changed significantly during the PRA development.

Table 5-5 provides a list of the in-house, independent review team.

5.3 Areas of Review and Major Findings

Table 5-6 indicates the report sections and responsibilities for review. The NMP2 IPE Review began early in the project and continued until completion. Sections of the analysis were reviewed as completed. The majority of comments dealt with the operation of plant systems under upset conditions and the actions required by EOPs. These comments usually dealt with plant details such that listing of individual comments here would be unwieldy. Assumptions used to enable modeling of the plant's complex systems were of particular interest to the reviewers. Many comments in the form of verbal conversations, "marked-up" IPE pages, and memorandums were received. Resolution of the comments were often reviewed with the commenter. No major comments were received; where a major comment is loosely defined as one that would effect results substantially. The main result of the review was a better understanding of plant operation and design. As such, the review improved the accuracy of the IPE relative to the plant as designed and operated by NMPC. Issues relating to PRA theory were generally treated within the IPE Team. Consultants from various companies and backgrounds as well as the experience developed by the NMPC IPE Staff assured that PRA issues were treated adequately.

A particular benefit of the review that did not effect the quality of this report was the interaction between the IPE team and individuals from other disciplines. Many comments from reviewers were actually questions about PRA. Answering these comments became, for the most part, mini-tutorials on IPE. This interaction improved overall NMPC understanding of PRA and should enhance our ability to continually apply IPE to plant issues.

5.4 Resolution of Comments

Comments were generally input directly into the study and reviewed with the commenter. No comment resolution by the IPE Team was disputed by the Review Team.

TABLE 5-1
Nine Mile Point Unit 2 IPE Team

Staff Member	Responsibilities
Robert F. Kirchner	Project Management and Level I Analysis
Thomas J. Gurdziel	Level I Analysis
James A. Snizek	Level I Analysis and Review Comment Resolution
Julie A. Fischer	Level I Analysis
Michael W. Cowden	Transient Analysis
Sheng-Chi Lin	Transient Analysis
G.W. Lapinsky	Human Reliability Analysis (HRA)

TABLE 5-2
Nine Mile Point Unit 2 IPE Support Team

Name	Group	Area of Support
R.A. Cushman	Nuclear Technology	Project management, project plan development
A.T. Denny	Operations	Plant familiarization, sequence development, HRA
J. Helker	Operations	EOPs, sequence development
J. Neyhard	IST	Equipment test data
R. Green	System Engineering	Service water system operation
D. Willis	Operations	Plant operation
R. Deuvall	Mechanical Design	System design basis
A. Julka	Electrical Design	System design basis
P. O'Brien	Electrical Design	System design basis
U. Buiva	Electrical Design	System design basis
J. Cushman	Structural Design	Structural design basis
R.K. Slade	Training	HRA
J.G. Reid	Training	HRA
P. Walsh	Training	HRA
J. Toothaker	Training	HRA
D. Holt	Training	HRA
R. Bigelow	Training	HRA

TABLE 5-2
Nine Mile Point Unit 2 IPE Support Team

B. Hennigan	Training	HRA
G. Pitts	Operations	HRA, EOP Response
B. Moore	Operations	HRA

TABLE 5-3
Nine Mile Point Unit 2 IPE Co-op Students

Co-op Student	School	IPE Assignment
Vicki Chan	Rensselaer Polytechnic Institute	System analysis
Grace Sun	Rensselaer Polytechnic Institute	System analysis
Robert Scherfling	Rochester Institute of Technology	Data analysis
Paul Grimes	University of Cincinatti	Data and system analysis
Gregg Nichols	Rensselaer Polytechnic Institute	MAAP transient analysis I/O
Amanda Wohlleber	Clarkson University	Data analysis
Steven Ainsworth	Rensselaer Polytechnic Institute	MAAP code parameter file development

Table 5-4
Nine Mile Point Unit 2 IPE Consultants

Consultant	Company	Area of Support
J.H. Moody	Independent	Consultant Coordination, LI Analysis, LI/LII Interface
Dr. A.N. Beare	General Physics	Human Reliability Analysis (HRA)
Dr. E.T. Burns	ERIN Engineering and Research, inc.	Level II Analysis, Containment Performance Analysis
T.J. Casey	Independent	Systems Analysis
J.J. Euto	XESS	Data Analysis
J.R. Gabor	Gabor, Kenton & Associates	MAAP Thermal Hydraulic (T/H) Analysis
T.P. Mairs	Independent	LI Analysis, LII Analysis Support
B. Malinovic	Fauske and Associates	MAAP T/H Analysis
Dr. G.W. Parry	Haliburton NUS	HRA
J.C. Raines	Fauske and Associates	MAAP T/H Analysis
W.P. Sullivan	General Electric	Review
D.E. Vanover	Gabor, Kenton & Associates	MAAP T/H Analysis
Dr. D.A. Wesley	ABB Impell	Containment Performance Analysis

TABLE 5-5

Nine Mile Point Unit 2 IPE Review Team

Reviewer	Group
P.J. O'Brien - IPE Review Lead	Independent Safety Evaluation Group (ISEG)
S. Barber - IPE Review lead	Quality Assurance (QA)
G. Thompson	System Engineering
R.K. Slade	Training
A. Vierling	Nuclear Technology
K. Ward	Design
L. Smith	QA
J. Ting	System Engineering
D. Flood	System Engineering
A. Julka	Electrical Design
R. Deuvall	Mechanical Design
F. Gerardine	System Engineering
J. Helker	Operations
G. Moyer	Operations
A. Sassani	Plant Evaluation
T. Sullivan	System Engineering
R. Mahwhinney	System Engineering

TABLE 5-5

Nine Mile Point Unit 2 IPE Review Team

J. Kaminski	Emergency Preparedness
J. Thuotte	Licensing
J.P. Cushman	Structural Design
R.E. Watson	Maintenance
W.P. Sullivan	General Electric

TABLE 5-6
Nine Mile Point Unit 2 PRA Responsibilities

Report Sections	Management	ISEG	QA	Engineering	Operations
1. Executive Summary	X	X	X	X	X
2. Examination Description	X	X			
3.1 Accident Sequence Delineation		X	X	X	X
3.2 System Analysis		X	X	X	
3.3.1 Generic Data		X	X	X	
3.3.2 Plant Specific Data		X	X	X	
3.3.3 Human Failure Data		X	X		X
3.3.4 Common Cause Data		X	X	X	
3.3.5 Quantification of Systems		X	X		
3.3.6 Support System States (Not Applicable)	-	-	-	-	-
3.3.7 Quantification of Sequences		X	X		
3.3.8 Internal Flooding Analysis		X	X	X	X
3.4 Results and Screening	X	X	X	X	X
4. Back-end Analysis		X	X	X	
5. Utility Participation and Review	X	X	X	X	X
6. Plant Improvements and Unique Features	X	X	X	X	X
7. Summary and Conclusions	X	X	X	X	X

6.0 Plant Improvements and Unique Safety Features

Performing an IPE leads to a unique perspective on the plant under study. Section 6.1 discusses NMP2 features that were noted to be of particular interest during the study. A number of improvements were identified during the study that resulted in specific improvement initiatives which are discussed in Section 6.2. In addition to these initiatives, the study developed some insights that are discussed in Section 6.3. These insights, for a number of reasons, did not result in immediate action. These insights will continue to be studied by NMPC and as more information and research becomes available specific action may be initiated.

6.1 Noteworthy NMP2 Safety Features

Some interesting design features were identified during the IPE and are summarized below:

- <u>ATWS</u>
  The redundant reactivity control system (RRCS) at NMP2 automatically actuates standby liquid control (SLC), reactor recirculation pump trip, alternate rod insertion, and feedwater runback. This system was assessed to be reliable and negated the need to model operator actions associated with these functions. Other operator actions associated with level control are not dependent on manual initiation of SLC or the other functions.

- <u>Spatial Considerations</u>
  The spatial arrangement and separation of safety divisions at NMP2 appears to be very good, although the IPE evaluation of other hazards such as fires has not been initiated yet. The separation of the auxiliary bays, submarine type doors to the auxiliary bay pump rooms, HPCS, and RCIC provide substantial protection from floods and other hazards. On the other hand, this spatial protection provides difficulties for equipment when room cooling is lost. However, redundancy in room cooling units provides reliable cooling in comparison to a single pump. It takes a total loss of service water to require recovery actions associated with opening doors and protecting pumps. Several air conditioning systems in the control building provides significant redundancy with regard to opening doors to other areas with separate air conditioning.

- <u>HPCS</u>
  The HPCS system is completely independent including actuation system inputs and emergency AC (referred to as Division III), yet during a station blackout the HPCS is unavailable because the HPCS diesel depends on service water which is unavailable during a station blackout. A potential improvement discussed below has been identified for consideration which would allow a chance for HPCS success.

- Interfacing Systems LOCA
  The frequency of an interfacing systems LOCA was assessed to be of low frequency at NMP2 due primarily to extra strength pipe used for piping diameters greater than 12 inches. Piping less than or equal to 12 inches is standard or extra strength pipe. Thus the probability of pipe rupture is unlikely. The RHR shutdown cooling suction path has power removed from the motor operated valves during power operation and there is a third normally closed motor operated valve in each pump suction path. The low pressure injection paths are not stroke tested during power operation and procedural precautions are being added to logic testing procedures to reduce the likelihood of inadvertent opening.

- Offsite Power Connections
  The 345 kV and 115 Kv connections are physically separate outside the plant located in the switchyard. There are cross-tie capabilities inside the plant which allow one 115kV source to supply all divisions of emergency AC power.

- DC Power
  The DC power system is divided between a non-safety related subsystem and a safety related subsystem. The safety related subsystem is completely independent of the nonsafety subsystem. This greatly improves DC reliability as load-shedding of numerous non-safety loads is not necessary to protect the safety related loads. Some DC load shedding is warranted, but because of the separation, it is limited to a relatively few loads.

- Hardened Containment Vent Capability
  The availability of containment vent systems and procedures gives NMP2 an additional set of mitigation actions to take in an emergency

- Motor Driven Feedwater Pumps
  With the exception of support system failures, feedwater (injection) and condenser (heat removal) were determined to be highly independent. That is, loss of condenser initiating events do not cause loss of feedwater at NMP2.

- Drywell Sunken Pedestal Under RPV
  The sunken drywell pedestal located directly under the RPV will contain any corium released from the RPV. This protects the outside walls of the containment from direct attack by corium.

- Downcommers Located in Drywell Floor Below RPV
  The downcommers located in the sunken drywell pedestal greatly enhance the coolability of debris and enhance the scrubbing of releases.

- Thick and Substantial Pedestal Walls Support RPV
  The pedestal is constructed such that it supports the RPV even after substantial attack from corium.

- Containment Flooding Capability
  Containment flooding systems and procedures are available which give NMP2 and additional set of mitigation actions to take in an emergency.

- Drywell Sprays Communicate with the Pedestal
  Any corium released from the RPV will be contained in the sunken drywell pedestal. Coolability of the contained debris is enhanced by the ability of drywell sprays to reach the sunken pedestal. However, no credit is currently taken in the IPE for this feature because initiation of drywell spray is dependent on the drywell spray initiation (DWSI) curve. This curve limits the conditions where this feature would be useful. See related discussions in Sections 6.3 and 4.8.

6.2 IPE Based Improvements

A number of benefits are derived from the IPE. An appreciation of the range of severe accidents that could occur at NMP2, the more likely sequences that contribute to risk, and the importance of components, systems, and human actions that determine the risk are an immediate value. In addition, cost beneficial improvements are typically identified during these studies. The following improvements were identified and included in the IPE model as they are being implemented (Table 6.1-1 also describes the recommended actions and the expected symptoms):

- Containment Venting
  Loss of long term decay heat removal is an important class of sequences. The IPE model includes a design modification that will be installed during the 1993 refueling outage. This modification allows the standby gas treatment filters to be isolated with valves rather than requiring operations and maintenance personnel to remove expansion joints and install blind flanges. As part of this modification, the EOP procedure associated with aligning containment venting will be revised. The IPE model takes credit for this procedure change to add guidance on locally opening the outside containment purge valve when instrument air or Division I emergency AC is unavailable. It is also assumed that guidance will be provided to align instrument air to the nitrogen supply if nitrogen is unavailable. This allows the operators to open the inside containment purge valve when nitrogen is unavailable and air is available. The above improvements provide higher confidence and more reliable human action assessments for successfully venting the containment when required.

- Auxiliary Bay Pump Room Cooling
  Loss of service water scenarios, although low frequency events, lead to loss of room cooling to HPCS, RCIC, and the low pressure injection pump rooms in the auxiliary bays. Failure of all these pumps would lead to a total loss of injection. It was judged that the pumps in the auxiliary bays could be protected by opening doors from the auxiliary building (El 175) into the pump rooms. There is a return path high in the pump room back to the auxiliary building (El 196) through a pipe chase. There is currently no explicit procedural guidance for performing this action but it is being added to procedures.

- Station Blackout Procedures
  As a result of the station blackout rule making, NMPC has committed to develop station blackout specific emergency operating procedures. The IPE station blackout model assumes this procedure has been developed, trained on by the operators, and includes insights from the IPE station blackout analysis. Specifically, the following will be addressed in the procedure:

  - The GE station blackout analysis will be referenced and the IPE blackout model will be used as the framework for developing the procedures.

  - Bypassing RCIC isolation interlock circuitry within 2 hours. This includes high room temperature isolation and turbine exhaust backpresssure isolation.

  - Shedding all non-essential DC loads within the first 2 hours of the event. This increases the time that DC will be available for RCIC, relief valve operation, and instrumentation.

  - Remote operability of RHR injection MOVs without AC power to allow diesel fire pump injection (EOP-6 Att. 6). Manual hookup of diesel fire water should be performed within the first 2 hours. The present model does not take credit for the diesel fire pump being aligned within the first 2 hours.

  - Instructions on how SRVs should be operated to minimize depletion of nitrogen and DC power. In addition, if RCIC is available and emergency depressurization is required, it is important that depressurization not cause a low RPV pressure isolation of RCIC or cause RCIC to stall. If RCIC fails, then it is important to depressurize sufficiently to allow the diesel fire pump to inject.

  - Closure of outside containment isolation valves dependent on AC power is addressed by EOPs and EOP-6 Attachment 1. The station blackout procedure should explicitly include local closure of these valves as this is important to the frequency of severe accident release.

As of this printing Draft Station Blackout (SBO) procedures have been developed and are under review.

- <u>Internal Flood Analysis</u>
  A large service water system flood or fire water system flood in an emergency diesel room or the control building was considered potentially important because all emergency AC is located in the area. The floor area is large and water would pass under doors to lower elevations and adjacent areas, however, the potential does exist for water to accumulate over time given a large flood. There are adequate sumps and alarms to alert the operators of flooding conditions, but it was decided that additional guidance to the operators was appropriate. This additional guidance includes opening doors from outside that will remove water from the building and isolation of the flood.

- <u>Interfacing Systems LOCA</u>
  Test and maintenance procedural precautions were identified to ensure that inadvertent opening of low pressure injection paths during power operation are unlikely.

6.3 IPE Insights

There were additional insights identified during the IPE that may be considered in the future. These insights are summarized below:

- <u>RCIC Room Cooling</u>
  The IPE model presently takes no credit for disabling RCIC high temperature trips and opening its room door given a loss of service water. RCIC was assessed to be capable of operation at fairly high temperatures, however, the alarm response procedures do not presently provide the necessary guidance to allow credit for these actions. Although the IPE takes no credit for these actions, the procedural improvements have been initiated.

- <u>Station Blackout</u>
  Station blackout is an important contributor to core damage frequency as assessed in the present IPE results. The following were qualitative insights and potential improvements that surfaced during the station blackout analysis:

  - There is uncertainty about the capability of diesel fire water to provide successful injection (EOP-6 Att. 6) through a 100 feet of 2.5 inch canvas hose. Test data and additional information or tests are being pursued to establish a system injection flow profile. This system is a backup to RCIC and requires the SRVs to remain open (RPV depressurized) which depends on nitrogen and DC power.

  - The HPCS system would become another recovery option if fire water could be used to cool the HPCS emergency diesel. There may be a relatively inexpensive modification that would provide this capability. Use of existing service water piping and/or Unit 1 connections are being considered.

- The RCIC backpressure trip set point (10 psig) and the RCIC high temperature trip set point (135 F) appear to be unnecessarily low. Whether the set points can be set higher may be investigated.

- <u>Partial Loss of Offsite AC</u>
  The IPE model includes recovery from loss of one 115kV offsite source. However, the human reliability analysis and interviews at the plant identified that the procedures are somewhat difficult and could be improved.

- <u>Service Water Recovery</u>
  Service water is an important contributor to core damage frequency. A more careful analysis of system capability with less stringent success criteria may be investigated as well as recovery actions for equipment failures. In addition, more procedural guidance may be warranted. For example, no credit is given to using one service water pump to supply one train of safety equipment when the crosstie between divisions is open.

- <u>Reliability Program</u>
  The IPE identifies the importance of systems which can be used as an input to the Reliability Centered Maintenance Program. Based on the IPE results, the following systems were identified as the more important:

  - AC power with emphasis on emergency AC and the diesels.
  - Containment venting air operated valves
  - RHR system with emphasis on heat removal function.
  - Service water with emphasis on pump trains.
  - RCIC
  - HPCS

- <u>Operator Action Insights</u>
  As described above, procedural improvements are included in the IPE and are being incorporated by plant operations. Additional potential improvements in procedures are also described in this section for future consideration. In addition, development of the IPE required interaction with operations and training personnel which provided further insights. The IPE team will be ensuring that the operator training department is aware of the IPE results, the dominant sequences and insights, and support development of any training material.

- ATWS

  The RCIC turbine backpressure trip set point was noted to be low (10 psig) as above under station blackout. EOP-C5 implies operators can only keep MSIVs open (i.e., if they close leave them closed). EOP section RPV RP states if boron injection..... open MSIVs. No credit is given to the operators keeping MSIVs open or reopening them except when feedwater is restored before level reaches Level 1. Therefore, particularly as HPCS is terminated, most ATWS scenarios turn into isolation scenarios because of feedwater runback. MAAP calculations indicated that allowing HPCS to operate may be beneficial, thus the competing risks associated with alternative may be investigated. Also, reliability of instrumentation such as fuel zone instruments used when dropping water level should be investigated to ensure that operators understand potential differences in readings.

- Water Injection

  During a postulated core damage accident, molten debris could be located in a number of areas beside the RPV. As such, strategies to cool the debris where it resides could be valuable. However, current EOPs focus on RPV injection and prohibit diversion of flow from the RPV unless adequate core cooling is assured; likely not the case if core relocation has occurred. In addition the drywell spray initiation (DWSI) curve prohibits initiation in some cases where spray could be useful. Therefore, severe accident strategies could be developed that address corium location.

- Containment Venting

  The NMP2 IPE noted some severe accident sequences where venting at containment design pressure might not be the best alternative (note Section 4.8). Consideration of a number of possible, but low probability, events raises the issue that containment venting might benefit from using other symptoms in addition to design pressure for initiation. In particular, venting at design pressure results in a relatively early release in some cases whereas waiting until some later time results in a smaller release. In addition, more time is available for mitigation actions and/or evacuation if containment vent is delayed.

- Containment Flooding

  A possible improved response for current containment flood types of sequences for which the EPG directions result in the highest potential consequences at the earliest time, is to provide the operators guidance on protecting containment and cooling debris using methods that do not require opening the RPV vent and avoid using the DW vent unless no other alternative exists. Alternate actions have been shown to produce substantially lower releases and much longer times to failure if no action is taken, i.e., even no action may be better than action directed by the EPGs.

- <u>Containment Injection at High Containment Pressure</u>
  There is a set of very low frequency severe accidents for which the containment may be at elevated pressures (i.e., above the containment vent pressure) and for which the EOPs would dictate that injection to the RPV be terminated when containment pressure exceeds MPCWLL. Because such a strategy can lead directly to core damage and a subsequent containment challenge it may be prudent to not terminate water injection to the containment under at least some circumstances for which core degradation may be aggravated by the termination of injection.

- <u>Standby Liquid Control (SLS) recovery actions</u>
  Three procedures which either disable or could disable the SLS system have been identified, two surveillance procedures and one chemistry procedure.

  N2-OSP-SLS-Q001, "Standby Liquid Control pump, check valve, and relief valve test", causes one train to be inoperable and, for a short time causes the redundant train to be inoperable. As an outcome of LER 91-15, restoration actions are explicitly stated if an actuation signal is received while both trains are inoperable. These actions only restore the redundant train, there are no actions describing the proper restoration of the train originally in test. The IPE model requires both trains of SLS to ensure adequate flow of borated water to the core for all ATWS scenarios. A future consideration may be to develop a method (procedure or guideline) to restore the inoperable train to operable.

  N2-OSP-SLS-Q002, "Standby Liquid Control motor operated valve operability test", also makes the SLS train in test inoperable. Again, should a SLS initiation occur during execution of this procedure, there are no proceduralized steps for recovery of the inoperable train. As discussed above, the IPE model requires both trains of SLS to be operable for ATWS scenarios, and future revisions of this procedure may want to consider a train restoration guideline as stated above.

  Also, a change be considered that the "checker" in the system restoration section (specifically regarding valve line-up) not be the same person that completed the procedure.

  N2-CSP-3M, "Standby Liquid Control chemistry surveillance", operates an air sparger in the borated water tank. The procedure states that if a SLS actuation occurs while the sparger is on, pump cavitation could occur. Hold out tags are placed on the SLS pump start switches to alert an operator to secure the air sparger if required. There is however, no independent verification of the air supply valves being returned to the closed position in section 5.1 of the procedure. If the boron concentration is acceptable, no other work is done, and the valves are not verified in this procedure. If the boron concentration is unacceptable, the procedure is continued, and when the valve is closed in these sections, it is independently verified closed. However, at the end of the corrective steps, the technician is directed to repeat the procedure to verify boron concentration, exiting the procedure from the step with no independent verification. As of this writing, a Procedure Change Evaluation has been initiated.

Table 6.1-1
Recommended Plant Improvements

Scenario Description	Affected System	Recommended Actions	Expected Symptoms
Transient event initiated by the loss of either the instrument air, division I AC, or instrument nitrogen	Hardened wetwell vent	The IPE credits an improved hardened wetwell vent that is currently scheduled to be installed in NMP2 during Refueling Outage 3 in 1994. Although it is expected that the direction in the current revision of the EOPs is applicable to this vent system configuration, the implementing procedure N2-EOP-6, Att. 21 could be revised to include contingency actions in the case of loss of instrument air or division 1 emergency AC, and failure of the instrument nitrogen system. For instance, in the existing vent configuration, an operator could manually open the outboard valve AOV111 locally upon loss of air to the valve operator or loss of AC power. Additionally, the inboard purge valve AOV109 could be remotely operated upon loss of instrument nitrogen if the operator can manually align the cross-connect between the instrument air and nitrogen systems via 2IAS-V1203.	The symptom prompting the operating crew to implement containment venting is clearly delineated in the EOP Section PCP. The proposed procedure changes instead require that the operator recognize the loss of air, nitrogen, Div. I AC power.
Transient event initiated by, or involving subsequent failure of, loss of service water to the Auxiliary and Reactor Buildings	All ECCSs	Among the numerous challenges facing the operating crew in responding to an event involving the loss of service water, additional actions must be implemented to prevent excessive heat-up of the ECCS rooms from seriously degrading pump motors and electronic control circuitry. A strategy for avoiding this situation, which is credited in the IPE, is to augment natural circulation in these rooms by blocking open the doors leading to El. 175' of the Reactor Building. It is recommended that the following actions be incorporated into N2-OP-11, section H for each ECCS running during the course of the event: (Note that these actions are equally applicable in the case where local high temperature develops in the ECCS room while the system is operating. Therefore, these actions could be included in the OPs for each ECCS.) • Align the ECCS required for makeup to the RPV; • immediately open the door(s) to the Reactor Building while the system is operating (e.g., all doors in the lower elevation of the appropriate auxiliary bay for low pressure ECCS); • restore room isolation when the system is no longer operational. By maintaining the areas isolated during the event, sufficient room ventilation should be available to prevent damage to the ECCS equipment. Additionally, any concern about the potential for flooding the basement of the Reactor Building are also alleviated.	The symptoms for the total loss of service water are delineated in N2-OP-11. In addition, it is expected that these actions would be implemented in the case of a station blackout or partial loss of service water to area coolers.

Table 6.1-1
Recommended Plant Improvements

Scenario Description	Affected System	Recommended Actions	Expected Symptoms
Loss of offsite power with the subsequent failure of emergency AC power (i.e., station blackout)	DC Power	Load shedding of non-essential DC loads is assumed to be an action that the operating crew undertakes immediately (i.e., the PRA model assumes the action must be accomplished within 2 hours), upon recognizing the station blackout condition. The purpose of the action is to extend the operational life of the 125V DC power supplies to vital equipment necessary to mitigate the event (e.g., instrumentation, SRVs, RCIC), and maintain the habitability of the Control Room. Appendix B of the NMP2 Station Blackout Study (GENE-770-04-1290) describes the various candidate loads that could be potentially shed from their power source in the case of a station blackout. It should be emphasized, however, that further analysis is required to verify that the de-energization of these loads don't further complicate plant recovery.	There are numerous obvious symptoms indicating a station blackout event available to the operating crew in the Control Room. The four recommended actions should be accomplished within 2 hours from the initiation of the event. It is also expected that these actions will require considerable resources to implement these tasks in difficult conditions caused by no ac and limited dc power in the plant.
	RCIC	The same study cites numerous potential failure modes that could challenge the RCIC system during a station blackout event. Generally, the following actions are required to be performed by the operating crew to prevent an automatic trip of the system (i.e., the PRA model assumes that these actions must also be accomplished within the first 2 hours of the event): • Defeat all area high temperature isolation logic (i.e.,piping area, RCIC room, and RHR room); • defeat the RPV low pressure isolation interlocks per N2-EOP-6, Att. 2; and • defeat the RCIC turbine high exhaust back-pressure trip. There are also operational dependencies to consider in the development of the station blackout procedure. For instance, the EOPs should contain explicit instruction prohibiting the operator from fully depressurizing the RPV below approximately 200 psig (i.e., sufficient RPV pressure to maintain turbine inlet steam pressure to operate the turbine greater than 1500 rpm), upon reaching the HCTL. Instead, the crew should manually control RPV pressure below the HCTL by using the RCIC and a single SRV to conserve dc power and N2 gas supply. In addition, there are temperature dependent failure modes that can affect the operation of the RCIC system. The most important of which is the temperature affect of electronic control circuitry located in the room. The operating crew should augment RCIC room ventilation by opening the door into the Reactor Building El. 175'. (Please refer to the loss of service water scenario description.)	

Table 6.1-1
Recommended Plant Improvements

Scenario Description	Affected System	Recommended Actions	Expected Symptoms
	Diesel Powered Fire Pump (DFP)	The DFP is a potential makeup source that can be aligned by the operating crew to provide RPV injection. However, there are many dependencies that must be overcome to use the system in this mode during a station blackout, aside from the fact that the system may be incapable of providing sufficient makeup to the RPV (i.e., internal pressure ≥ 50 psig and an elevation head of > 100 ft.). There are several actions that need to be accomplished to align the system per N2-EOP-6, Att. 6: • Manually align the RHR system for injection, preferably the division B due to the redundancy of the hose path; • manually gag the pump relief to increase the pump head; and (This action requires verification.) • depressurize the RPV completely after failure of the RCIC system. Given that AC power has not been recovered and, by definition, the RCIC system has failed, there is considerable uncertainty that this procedure can be effectively implemented under these conditions. Therefore, development of this procedure should involve evaluation as to the potential effectiveness of these actions (i.e., with regards to system capability and limitations in aligning the system), to accomplish the makeup function.	
	Contain ment Isolation	Although, the failure of containment isolation is generally not a concern relative to preventing core damage, the failure to isolate lines may pose significant consequences should the operating crew be unable to provide RPV makeup long term during a station blackout. The Level 2 model accounts for the probability that the operating crew fails to manually isolate the penetrations that remain open upon loss of AC power. Table 4-4 in the above referenced station blackout study (GENE-770-04-1290), identifies the penetrations that must be manually isolated before the onset of core damage. Specifically, the valves that require local manipulation are designated by footnote (2) in the table.	

Table 6.1-1
Recommended Plant Improvements

Scenario Description	Affected System	Recommended Actions	Expected Symptoms
An ISLOCA event in the LPCI or LPCS low pressure rated piping inside the Auxiliary Bay	LPCI and LPCS injection isolation valves	The IPE postulates the improbable event that the isolation between the RCS and a low pressure rated system (i.e., specifically, the LPCI and LPCS injection paths) is breached, resulting in the subsequent failure (i.e., leak or rupture) of the system outside containment. However, key to this assessment is the assumption that an important potential precursor event is avoided because of precautions in surveillance procedure N2-ISP-ISC-M003 to prohibit a technician from inadvertently opening the injection isolation MOV as a result of improper performance of the procedure. There are two situations that can result in the MOV opening, on the differential pressure/valve open permissive signal, while performing the monthly surveillance test: 1) The coincidental manual initiation of the low pressure ECCS that actuates the S2 contact; and 2) the coincidental actuation of the K14 contact. The quantification of the probability that this precursor event occurs is based on the presumption that the surveillance test procedure has been revised to add the following precautions: • Tag out the key-locked switch S2 for the MOV during the test; and • prohibit the concurrent performance of all other procedures that can potentially generate a signal that closes the K14 contact.	There are no symptoms applicable to this recommendation. Instead, the recommendation is to add precautions to surveillance test procedure N2-ISP-ISC-M003.

Table 6.1-1
Recommended Plant Improvements

<table>
<tr><th>Scenario Description</th><th>Affected System</th><th>Recommended Actions</th><th>Expected Symptoms</th></tr>
<tr><td>Extensive internal flooding of Control Building El. 261' that affects the emergency switchgear buses</td><td>Cooling water to the EDG</td><td>Although internal flooding is evaluated to be a low frequency event, it has received considerable visibility in the industry due to plant specific analyses that have indicated a higher than expected vulnerability as a result of improperly performed maintenance. At NMP2, the close proximity of the emergency AC switchgears to the EDG rooms could pose a threat to safe plant operation if severe flooding in an EDG room were to occur. A scenario in the IPE postulates that the 8" service water header in either EDG room is breached that results in the flooding of the Control Building at a rate of approximately 850 gpm. Currently, the annunciation response procedure for the EDG room drain tanks direct the operating crew to investigate the cause of the alarm. However, more explicit instruction has been assumed to be available to the crew to successfully mitigate the event during the quantification of the sequence in the IPE. Specifically, the actions include, upon recognition that the source of the flooding is the service water system:

• Secure and isolate the affected service water division;
• prevent extensive flooding of El. 261' by opening two doors (i.e., on the east and west side of the building) to divert the water outside the building; and
• isolate the breach locally, and restore the service water division to safely shutdown the plant.

It is estimated that the operating crew has approximately 60 min. to mitigate the flooding of the Control Building and avert damage to the emergency switchgears. However, the consequences of damaging this equipment are so severe (i.e., station blackout with no service water), it warrants a procedure to ensure that the responding operator can take immediate effective action to terminate the flooding.</td><td>The water level indication 2DFD-LS101, 102, or 103 (annunciator panel 851-357) for the drain tanks in the EDG rooms are considered to be the most timely and reliable indication of flooding in the area.</td></tr>
</table>

Table 6.1-1
Recommended Plant Improvements

Scenario Description	Affected System	Recommended Actions	Expected Symptoms
Internal flooding of a service water pump bay that affects 1 division of service water	Service water pumps in 1 division	Another important internal flood initiating event involves the loss of all service water pumps in a division. Industry experience indicates that maintenance related activities, during which system isolation is compromised, are the primary cause of flooding. The source of the flood can be either from the service water side (e.g., loss of isolation during strainer cleaning), or Lake Ontario (e.g., during pump maintenance). Although the water level is assumed to remain below the top of the bay in either situation, the automatic response of the service water system upon the loss of all three pumps in a division can pose significant challenge to the operating crew to balance service water loads and restore cooling water to TBCLC and RBCLC. It is recommended that the appropriate actions to isolate the system breach and restore service water to vital loads be included in a SOP for the loss of a division of the service water system.	The water level indication 2DFM-LS136 in the "A" service water bay, or 2DFM-LS137 in the "B" service water bay (annunciator panel 851-341) for the sumps in the Service Water Building are judged to be the most timely and reliable indication of flooding in the area.

7.0 Summary and Conclusions

The NMP2 IPE set out with a number of goals and objectives. These were met by forming a capable inhouse team and performing a state-of-the-art PRA analysis.

Quantitative results show the NMP2 poses no undue risk to the heath and safety of the public. As a snapshot, the IPE gives confidence in the ability of NMP2 to safety produce electricity. However, the study suggests that improvement is possible. Qualitative results delineate possible improvement actions which are discussed in detail in this report. The improvement initiatives will continue, based on the IPE and its updates, until the plant is retired. Clearly, the IPE, as a living program, will continue to benefit the plant until decommissioning.

During the IPE a number of unresolved issues were studied. Based on the IPE, these issues can be resolved. Per NUREG-1335, these issues are summarized here with analysis detail presented in Section 3.4.

> **Unresolved Safety Issue A-45** Based on the IPE evaluation, as discussed in Section 3.4, the NMP2 decay heat removal is adequate. While, loss of decay heat removal sequences are important to IPE results, no specific vulnerabilities exist. Cost-beneficial improvements have been initiated by and incorporated in the IPE and will continue to be evaluated as part of accident management development. Therefore, Unresolved Safety Issue A-45 can be resolved for NMP2.
>
> **Generic Letter 91-06** addressed the adequacy of onsite DC power systems. NMP2 has adequate DC capability and the IPE found no issues requiring correction. The generic letter raised several questions which are answered and discussed in Section 3.2.1.7. Section 3.4 discusses the DC issue in terms of overall result. Based on this, the IPE suggests that Generic Letter be resolved for NMP2.

8.0 List of References

1 PRA Procedures Guide. Washington DC: U. S. Nuclear Regulatory Commission. January 1983. NUREG/CR-2300.

2 ATWS: A Reappraisal. Palo Alto, California: Electric Power Research Institute, January 1982. EPRI NP-2230.

3 Development of Transient Initiating Event Frequencies for Use in Probabilistic Risk Assessment. Washington, DC: U. S. Nuclear Regulatory Commission, June 1984. NUREG/CR-3862.

4 Brunswick Steam Electric Plant Probabilistic Risk Assessment. Raleigh, North Carolina: Carolina Power & Light Co., April 1988.

5 Probabilistic Risk Assessment, Shoreham Generating Station. Hicksville, New York: Long Island Lighting Company, June 1983.

6 Probabilistic Risk Assessment, Limerick Generating Station. Pittsburgh, Pennsylvania: Philadelphia Electric Company, June 1982.

7 Analysis of Core Damage Frequencies from Internal Events: Peach Bottom, Unit 2. Washington, DC: U. S. Nuclear Regulatory Commission October 1986. NUREG/CR-4550/Vol. 4.

8 Analysis of Core Damage Frequencies from Internal Events: Grand Gulf-1. Washington, DC: U. S. Nuclear Regulatory Commission, April 1987. NUREG/CR-4550/Vol. 6

9 Reactor Safety Study - An Assessment of Accident Risk in U. S. Commercial Nuclear Power Plants. Washington, DC: U. S. Nuclear Regulatory Commission, October 1975, WASH-1400 NUREG-75/014.

10 Additional Information Required for NRC Staff, Generic Report on Boiling Water Reactors. San Jose, California: General Electric Corporation, December 1980. NEDO-24708A, Rev. 1, Vol. 1.

11 Licensed Operating Reactors, Status Summary Report. Washington, DC: U. S. Nuclear Regulatory Commission, February 1989. NUREG-0020.

12 Seabrook Station Probabilistic Safety Assessment. Manchester, New Hampshire: Public Service Company of New Hampshire, December 1983.

13 Loss of Off-Site Power at Nuclear Power Plants: Data and Analysis. Palo Alto, California: Electric Power Research Institute, March 1982. EPRI NP-2301.

14 Overpressurization of Emergency Core Cooling Systems in Boiling Water Reactors. Washington, DC: U. S. Nuclear Regulatory Commission, Office for Analysis and Evaluation of Operational Data, September 1985. AEOD/C502.

15 EPRI NP-438, Characteristics of Pipe System Failures in Light Water Reactors. EPRI, August 1977.

16 Evaluation of Station Blackout Accidents at Nuclear Power Plants, Technical Findings Related to Unresolved Safety Issues A-44. Washington, DC: U. S. Nuclear Regulatory Commission, June 1988. NUREG-1032, Final Report.

17 Hope Creek Generating Station Status Blackout Technical Report. Public Service Electric & Gas Company, April 14, 1989.

18 E. T. Burns et al., Inadequate Core Cooling Detection in Boiling Water Reactors. S. Levy, Inc. SLI-8218, November 1982.

19 Calvert, John et al., Review of BWR Reactor Vessel Water Level Measurement Systems. SLI-8211, May 1982.

20 IDCOR Technical Report 10.1: Containment Structural Capability of Light Water Nuclear Power Plants. Technology for Energy Corporation, July 1983.

21 Containment Performance Working Group Report. NUREG-1037, U. S. Nuclear Regulatory Commission, May 1985.

22 Estimates of Early Containment Loads from Core Melt Accidents. NUREG-1079, U. S. Nuclear Regulatory Commission, December 1985.

23 A. A. Sonin and J. H. Chun, "Small Scale Simulation of Vapor Discharges into Subcooled Liquid Pools," Nuclear Engineering and Design. 5 (1985), 353-362.

24 A. A. Sonin, Nuclear Engineering and Design. 65 (1981), 17-21.

25 T. M. Su, Suppression Pool Temperature Limits for BWR Containments. U.S. NRC, NUREG-0783 (September 1981).

26 Letter from A. A. Sonin, (MIT), to D. J. Hanson, (EG&G Idaho, Inc.), dated March 21, 1986.

27 E. T. Burns, T. P. Mairs, V. M. Andersen, Shoreham Nuclear Power Station Full Power PRA Update: Supplemental Containment System Implementation. Long Islands Lighting company, February 1988.

28 Assessment of BWR Mitigation of ATWS, Volume II (NUREG-0460 Alternate No. 3). NEDE-24222, General Electric Co., December 1979.

29 Shoreham Nuclear Power Station Probabilistic Risk Assessment. Long Island Lighting Company, June 1983.

30 Yue, D.D., Containment Electrical Penetrations Failures During Degraded Core Accidents. IDCOR Containment Structural Capacity Workshop, Fauske & Associates, Burr Ridges, Illinois, February 24-25, 1982.

31 Sebrell, W., The Potential for Containment Leak Paths through Electrical Penetration Assemblies Under Severe Accident Conditions. NUREG/CR-3234, July 1983.

32 Severe Accident Testing of Electrical Penetrations Assemblies. Prepared by D. B. Clauss at Sandia National Laboratories for Nuclear Regulatory Commission, NUREG/CR-5334, November 1989.

33 Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors. General Electric, NEDO-24708A, Rev. 1, December 1980.

34 Ahl, T. J. et al., Analysis of a Mark I Containment Vessel for Severe Accident Conditions. NUREG/CR-0095, June 1988.

35 Information on Scrams and Scram-Reduction Practices at U.S. Nuclear Generating Units (1986 - 1990). INPO 91-009, Institute of Nuclear Power Operations, September 1991.

36 Probabilistic Evaluation of Nine Mile Point 2 Containment Performance for Beyond Design Basis Conditions. MV-1120-136-R001, Rev. 0, ABB-Impell, June 1991.

37 PLG, Inc., RISKMAN: PRA Workstation Software Users Manual. Newport Beach, CA, 1990.

38 Milton, J.S. and Arnold, J.C., Probability and Statistics in the Engineering and Computing Sciences. McGraw-Hill, NY, 1986

39 PLG, Database for Probablistic Risk Assessment of Light Water Nuclear Power Plants. PLG-0500, 1989.

40 Gertman, D.I., et.al., Nuclear Computerized Library for Assessing Reactor Reliability (NUCLARR). prepared for the U.S. Nuclear Regulatory Commission, NUREG/CR-4639, 1988.

41 Mosleh, A., et.al., Procedures for Treating Common Cause Failures in Safety and Reliability Studies. prepared for the Electric Power Research Institute and the U.S. Nuclear Regulatory Commission, NUREG/CR-4780, Volumes I and II, 1988.

42 Swain, Alan D., Accident Sequence Evaluation Program Human Reliability Analysis Procedure. NUREG/CR-4772, February 1987.

43 An Approach to the Analysis of Operator Actions in Probabilistic Risk Assessment. EPRI-TR-100259, December 1991.

44 Swain, A.D. and Guttman, H.E., Handbook or Human Reliability Analysis with Emphasis on Nuclear Power Plant Application. NUREG/CR-1278, August 1983.

45 E.T. Burns, et.al., ISLOCA Evaluation Guidlines. Prepared for Nuclear Safety Analysis Center, NSAC-154, September 1991.

46 E.T. Burns, et.al., Interfacing System Isolation Experience Review. Prepared for Nuclear Safety Analysis Center, NSAC-155, August 1991.

47 E.T. Burns, et.al., ISLOCA Prevention and Mitigation Measures. Prepared for Nuclear Safety Analysis Center, NSAC-167, September 1991.

48 Kemeny, J.G., Chairman, Report of the President's Commission on the Accident at Three Mile Island, October 1979.

49 "Nuclear Power Plant Response to Severe Accidents," IDCOR Technical Summary Report, AIF, December 1984.

50 Safety Goals to the Operation of Nuclear Power Plants; Policy Statement, Nuclear Regulatory Commission, 10CFR Part 50, dated July 30, 1986.

51 Policy Statement on Severe Reactor Accidents Regarding Future Designs and Existing Plants, Nuclear Regulatory Commission, 10CFR Part 50, August 1985.

52 "PRA Procedures Guide," NUREG/CR-2300, U.S. Nuclear Regulatory Commission, January 1983.

53 BWR Individual Plant Evaluation Methodology, Delian Corporation, March 1987.

54 "Zion Probabilistic Safety Study," Commonwealth Edison Company, 1981.

55 "Probabilistic Safety Study: Enrico Fermi 2, Detroit Edison Company.

56 "Probabilistic Risk Assessment: Big Rock Point Plant," Consumers Power Company, March 1981.

57 "Severe Accident Risk Assessment: Limerick Generating Station," Philadelphia Electric Company, April 1983.

58 "Indian Point Probabilistic Safety Study," Power Authority of the State of New York and Consolidated Edison Company, 1982.

59 Shoreham Nuclear Power Station Probabilistic Risk Assessment, Long Island Lighting Company, Docket 50-322, June 1983.

60 Technical Support for the Utility Group on ATWS, SAI-012-082-55, Science Application, Incorporated, December 31, 1981.

62 IDCOR Task 4.1, Containment Event Trees, AIF.

63 Vermont Yankee Containment Safety Study, Vermont Yankee Nuclear Power Corporation, August 1986.

64 Peach Bottom Containment Analysis Evaluation, NUREG/CR-4700.

65 Limerick Generating Station, Individual Plant Examination, Philadelphia Electric Company, Docket 50-352, 353, 1992.

66 Integration Plan for Closure of Severe Accident Issues, SECY-88-147, Victor Stello, Jr., dated May 25, 1988.

67 "Proposed Criteria to Accommodate Severe Accidents in Containment Design," ACRS Letter from David A. Ward to Chairman Carr, May 17, 1991.

68 "Integration Plan for Closure of Severe Accident Issues," Stello, V. Jr., SECY-88-147, May 25, 1988.

69 "Mark I Containment Performance Improvement Program," Stello, V. Jr., SECY-89-017, January 23, 1989.

70 "Status of Implementation Plan for Closure of Severe Accident Issues and Status of the Individual Plant Examination Program," Taylor, J.M., SECY-91-084, March 28, 1991.

71 "Role of Individual Plant Examinations (IPE) In Assessing Industry Status with Respect to the Commission's Safety Goal Policy," Taylor, J.M., SECY-90-104, March 20, 1990.

72 "Formulation of a Large Release Definition and Supporting Rationale," Taylor, J.M., SECY-90-405, December 14, 1990.

73 "Individual Plant Examination for Severe Accident Vulnerabilities-10 CFR 50.54(f)," Generic Letter 88-20, November 23, 1988.

74 "Initiation of the Individual Plant Examination for Severe Accident Vulnerabilities-10 CFR 50.54(f), Generic Letter 88-20, Supplement No. 1, August 29, 1989.

75 "Accident Management Strategies for Consideration In the Individual Plant Examination Process, Generic Letter 88-20, Supplement No. 2, April 4, 1990.

76 "Completion of Containment Performance Improvement Program and Forwarding of Insights for Use in the Individual Plant Examination for Severe Accident Vulnerabilities, Generic Letter 88-20, Supplement No. 3, July 6, 1990.

77 "Individual Plant Examination (IPE) for Severe Accident Vulnerabilities," Generic Letter 88-20, February 28, 1989.

78 "Accident Management," Congel, F.J., Workshop on the Individual Plant Examinations, Fort Worth, TX, February 2 through March 2, 1989.

79 Safety Goal, SECY-89-102.

80 Nine Mile Point Unit 2 FSAR, Rev. 0. April 1989.

81 "IDCOR Technical Report 10.1 Containment Structural Capability of Light Water Nuclear Power Plants", AIF, July 1983.

82 Evaluation of Secondary Containment Radionuclide Release Transport, FAI/EPRI, to be published.

83 Pelto, P.S., K. Ames and R. Gallucci, Reliability Analysis of Containment Isolation Systems, U.S. Nuclear Regulatory Commission, NUREG/CR-4220, PNL-5432, June 1985.

84 Kouts, H.J.C., Special Committee Review of the Nuclear Regulatory Commission's Severe Accident Risks Report (NUREG-1150), NUREG-1420, dated August 1990.

85 Kastenberg, W.E., Chairman, Findings of the Peer Review Panel on the Draft Reactor Risk Reference Document, NUREG-1150, NUREG/CR-5113, dated May 1988.

86 Nine Mile Point Unit 2 FSAR, Rev. 0, April 1989.

87 Limerick Generating Station Probabilistic Risk Assessment, Philadelphia Electric Company, 1981.

88 Shoreham Nuclear Power Station Probabilistic Risk Assessment, Long Island Lighting Company, June 1983.

89 Reactor Safety Study: An Assessment of Accident Risks on U.S. Commercial Nuclear Power Plants, U.S. Nuclear Regulatory Commission, NUREG-75/014, WASH-1400, October 1975.

90 Limerick Generating Station Probabilistic Risk Assessment, Philadelphia Electric Company, Docket 50-352, 50-353, September 1982.

91 Probabilistic Risk Assessment Shoreham Nuclear Power Station, Unit 1, Science Applications, Inc., SAI-372-83-PA-01, June 1982.

92 Garcia, A.A. et. al., Interim Reliability Evaluation Program: Analysis of the Millstone Point Unit 1 Nuclear Power Plant, Science Applications, Inc., NUREG/CR-3085, February 1983.

93 Mays, S.E. et. al., Interim Reliability Evaluation Program: Analysis of the Browns Ferry, Unit 1, Nuclear Plant, EG&G Idaho, Inc., NUREG/CR-2802, July 1982.

94 MAAP 3.0B

95 Shoreham Nuclear Power Station Probabilistic Risk Assessment, Long Island Lighting Company, Docket 50-322, June 1983.

96 Theofanous, T.G., Yan, H. Ratnam, U., and W.H. Amarasooriya, The Probability of Liner Failure in a Mark I Containment, NUREG/CR-5423, July 1989.

97 Greene, G.A., "Discussion of the Technical bases for Evaluation of the Vulnerability of the Mark I BWR Containment Shell to Melt-Through by Core Debris," presented at Mark I BWR Workshop, Baltimore, M.D., February 24-26, 1988.

98 Limerick Generating Station Updated Final Safety Analysis Report (LGS UFSAR).

99 Leahey, R.T., Experiments on Bottom Head Melt-Through, Rensselaer Polytechnic Institute, 1982-1988.

100 BWR Owners Group Emergency Procedure Guidelines (EPGs), Revision 4, NEDO 31331, March 1987.

101 Payne, A.C., et. al., Evaluation of Severe Accident Risks: Peach Bottom, Unit 2, NUREG/CR-4551, SAND86-1309, Vol. 4, Rev. 1, Part 1, December 1990

102 Estimates of Early Containment Loads from Core Melt Accidents, NUREG-1079, Containment Loads Working Group, December 1985.

103 Theofanous, T.G. et al., An Assessment of Steam-Induced Containment Failure, NUREG/CR-5030, February 1989.

104 Steam Explosion Review Group, A Review of the Current Understanding of the Potential for Containment Failure Arising from In-Vessel Steam Explosions, U.S. Nuclear Regulatory Commission, NUREG-1116, June 1985.

105 A.A. Sonin, Nuclear Engineering and Design, 65 (1981), 17-21.

106 IDCOR Technical Summary Report, 1984.

107 S.R. Greene, "The Role of BWR Mark I Secondary Containments in Severe Accident Mitigation," presented at the Fourteenth Water Reactor Safety Information Meeting, Gaithersburg, Maryland, October 28, 1986.

108 Swenson, D.V. and Corradini, M.L., "Monte Carlo Analysis of LWR Steam Explosions", SAND81-1092, NUREG/CR-2037, 1981.

109 Bohl, W.R., "An Investigation of Explosion Loadings Using SIMMER-II," LA-UR-85-3200 (in preparation for publication).

110 M. Berman, D.V. Swenson, and A.J. Wickett, "An Uncertainty Study of PWR Steam Explosions," SAND83-1438, NUREG/CR-3369, May 1984.

111 D.E. Mitchell, M.L. Corradini, and W.W. Tarbell, "Intermediate Scale Steam Explosion Phenomena: Experiments and Analysis," Sandia National Laboratories, NUREG/CR-2145, SAND81-0124, October 1981.

112 T.M. Su, Suppression Pool Temperature Limits for BWR Containments, U.S. NRC, NUREG-0783, September 1981.

113 Letter from A.A. Sonin, MIT to D.J. Hanson, EG&G Idaho, Inc. dated March 21, 1986.

114 A.A. Sonin and J.H. Chun, "Small Scale Simulation of Vapor Discharges into Subcooled Liquid Pools," Nuclear Engineering and Design, 5 (1985) 353-362.

115 J.A. Block, "Condensation - Driven Fluid Motions," International Journal of Multiphase Flow, Vol. 6, No. 1-2, February/April, 1980.

116 Small Break LOCA - Additional Information, General Electric, NEDO-24708A, December 1979.

117 Denning, R.S., et al., Recriticality in a BWR Following a Core Damage Event, NUREG/CR-5653, December 1990.

118 Nine Mile Point Unit 2 IPE Walkdown Summary, Rev 0, 11/20/90.

119 E.T. Burns, J.R. Gabor, and T.P. Mairs, BWR Containment Vulnerability Analysis, ANS Summary Meeting Vol. 59 (pg. 223), Atlanta, GA, June 1989.

120 Nuclear Power Plant Response to Severe Accident, IDCOR Technical Summary Report, November 1984.

121 "Probabilistic Evaluation of Nine Mile Point Unit 2 Containment Performance for Beyond Design Basis Conditions," MV-1120-136-R001, Rev. 0, ABB-Impell, dated June 1991.

122 Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants, U.S. NRC, NUREG-1150, May 1989.

123 Precursors to Potential Severe Core Damage Accidents, Prepared by Minarick, J.W., et. al. at the Oak Ridge National Laboratory for the U.S. Nuclear Regulator Commission. NUREG/CR-2497.

124 Precursors to Potential Severe Core Damage Accidents, Prepared by Cottrell, W.B., et. al. at the Oak Ridge National Laboratory for the U.S. Nuclear Regulator Commission. NUREG/CR-3591.

125 Precursors to Potential Severe Core Damage Accidents, Prepared by Minarick, J.W., et. al. at the Oak Ridge National Laboratory for the U.S. Nuclear Regulator Commission. NUREG/CR-4674.

126 Supplement to the Technical Summary Report: Nuclear Power Plant Response to Severe Accidents, TENERA, December 1988.

127 "Evaluation of Empirical Aerosol Correlations," EPRI NP-4974, J.M. Otter and E.V. Vaughan, December 1986.

128 "Recommended Sensitivity Analyses for an Individual Plant Examination using MAAP 3.0B, EPRI, (Not Yet Published).

129 ALWR Severe Accident Dose Analysis, DE-ACOG-87RL11313, March 1989.

130 Burns, T.J., Impact of Containment Building Leakage on LWR Accident Risk, Oak Ridge National Laboratory, NUREG/CR-3539, April 1984.

131 P.K. Mast, D.R. Bradley, J.E. Brockmann, R.J. Lipinski, and D.A. Powers, "Uncertainty in Radionuclide Release Under Specific LWR Accident Conditions, Volume IV", Sandia National Laboratories, SAND84-0410 Vol 4, Appendix E, December, 1985.

132 "Scrubbing of Aerosols by Water Pools, Volume 1: Quencher Orifice Injection", EPRI NP-4890SP, Volume 1, Project 2117-1, October 1986.

133 "Removal of Radionuclides by Water Pools Under Severe Accident Conditions", EPRI RP 2117, October 1983.

134 Devonrue, "Guidelines and Technical Bases for NUMARC Initiatives Addressing Station Blackout at Light Water Reactors," NUMARC 87-00, Rev. 1., August 1991.

135 P.K. Mast, D.r. Bradley, J.E. Brockmann, R.J. Lipinski, and D.A. Powers, "Uncertainty in Radionuclide Release Under Specific LWR Accident Conditions, Volume IV," Sandia National Laboratories, SAND84-0410 Vol. 4, Appendix E, December 1985.

136 "Scrubbing of Aerosols by Water Pools, Volume 1: Quencher Orifice Injection," EPRI NP-4890SP, Volume 1, Project 2117-1, October 1986.

137 "Removal of Radionuclides by Water Pools Under Severe Accident Conditions," EPRI RP 2117, October 1983.

138 G.D. Kaiser, "The Implications of Reduced Source Terms for Ex-Plant Consequence Modeling," Executive Conference on the Ramifications of the Source Term (Charleston, SC), March 12, 1985.