GE Nuclear Energy

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Chet Poslusny, Senior Project Manager
Standardization Project Directorate
Associate Directorate for Advanced Reactors
and License Renewal
Office of the Nuclear Reactor Regulation

Subject: Submittal Supporting Accelerated ABWR Review Schedule - **Seismic Margins Analysis**

Dear Chet:

Enclosed is a final draft of Section 19I which responds to several NRC questions regarding the ABWR seismic margins analysis. This draft will replace the current Section 19I of the SSAR. Changes made since the last revision of this draft are indicated by vertical bars in the right hand margin. Responses to NRC questions are indicated by vertical bars in the left hand margin. GE believes that all NRC questions regarding the seismic margins analysis are responded to in this final draft.

On Page 19I.1-1, reference is made to Section 19D.7.4. A draft of Section 19D.7 was sent to Bob Palla (by Larry Frederick) on 12/16/92. Another copy is enclosed for your information.

Sincerely,

Jack Fox

Jack Fox
Advanced Reactor Programs

cc: Jack Duncan (GE)
Norman Fletcher (DOE)
Bob Palla (NRC)

See attached distribution

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19I.1 INTRODUCTION

A seismic margins analysis has been conducted for the ABWR using a modification of the Fragility Analysis method of Reference (1) to calculate high confidence low probability of failure (HCLPF) accelerations for important accident sequences and accident classes. HCLPF values were calculated for components and structures using the relationship

$$HCLPF = A_m * \exp(-2.326 * \beta_c)$$

where: A_m = the median peak ground acceleration corresponding to 50% failure probability,

β_c = the logarithmic standard deviation of the component or structure fragility.

The resulting HCLPF acceleration corresponds essentially to the 95th percent confidence level that at that acceleration the failure probability of a particular structure or component is less than 0.05 (5%). HCLPFs for accident sequences were evaluated through use of event trees, and seismic system analysis was performed with fault trees to determine HCLPFs of systems.

The seismic margins analysis evaluates the capability of the plant and equipment to withstand a large earthquake (2*SSE). In this analysis, two alternative methods were used to evaluate the seismic accident sequences—a "convolution" method and a "min-max" method.

In the convolution method, accident sequences are evaluated by combining input fragility curves according to the Boolean expression for each sequence. Seismic and random/human failure probabilities are calculated and combined (convolved) for discrete intervals of ground acceleration, and then integrated over the range of interest.

In the min-max method, input fragilities are combined by using the lowest (minimum) HCLPF value of a group of inputs operating in an OR logic, and by using the highest (maximum) HCLPF value of a group of inputs operating in an AND logic. Random/human failure probabilities are reported in combination with HCLPFs for each accident sequence.

Analysis of the effects beyond core damage (Level 2 PRA analysis) was not a part of this seismic margins analysis. However, event trees were constructed to examine the possibility of loss of containment isolation resulting in a large release given the earthquake and a resulting core damaging accident.

Because of the inclusion of a rupture disk in the ABWR design as an ultimate means of containment heat removal, and because an earthquake would not prevent rupture of the disk, failure of containment heat removal is not modeled in the seismic margins analysis. (There are no Class II sequences in the analysis.) There are two valves in line with the rupture disk; however, these valves are left in an open position, and the earthquake would not cause these valves to close.

There are several operator actions included in the seismic margins analysis. These operator actions are discussed in Section 19D.7.4.

19I.2 COMPONENT AND STRUCTURE FRAGILITY - A_M, β_C

Component and structure fragility values have been established for selected structures and components that have been identified as potentially important to the seismic margins analysis. The fragility values used in the analysis are shown in Table 19I.2-1, together with the calculated component/structure and system HCLPFs. The component fragility values are based on generic components used in operating plants having SSEs of 0.15-0.2 g. These component fragilities are conservative for a plant designed and built to an SSE of 0.3 g. For more information regarding the development of these fragilities and capacities, refer to Appendix 19H.

Table 19I.2-1
ABWR SYSTEMS AND COMPONENTS/STRUCTURES FRAGILITIES

SYSTEM/COMPONENT	MED_CP (A_M)	LOG_STD (β_C)	HCLPF* (in g)
1. Plant Ess. Structures (SI)			
- Reactor Building	3.2	.45	1.12
- Containment	3.1	.44	1.11
- RPV Pedestal	5.0	.44	1.80
- Control Building	4.1	.44	1.47
- Reactor Pressure Vessel Support	5.0	.33	2.32
2. Support Systems (PW)			
a. AC Power (ACP)			
- Diesel Generator	1.8	.46	.62
- Transformer (480 V AC)	1.8	.46	.62
- Motor Control Center	1.8	.46	.62
- Cable Tray	3.0	.60	.74
- Circuit Breaker	2.0	.50	.63
- Inverter	2.2	.46	.75
b. Service Water (SW)			
- Pump (Motor Driven)	1.8	.46	.62
- Heat Exchanger	2.0	.45	.70
- Valve (Motor Operated)	3.0	.60	.74
- Check Valve	3.0	.60	.74
- Room Air Cond. Unit	2.0	.50	.63
- Piping	3.0	.60	.74
- SW Pump House	1.7	.45	.60
- AC Ducting	3.0	.60	.74
c. DC Power (DCP)			
- Batteries <u>AND</u>	3.3	.46	1.13
- Charger	2.2	.46	.75
- Cable Tray	3.0	.60	.74
3. High-Press Core Flooder (UH)			
- Pump (Motor Driven)	1.8	.46	.62
- Injection Valve (Motor Op)	3.0	.60	.74
- HPCF Piping	3.0	.60	.74
- Check Valve	3.0	.60	.74
4. Reactor Core Is. Cooling (UR)			
- Pump (Turbine Driven)	2.0	.45	.70
- Steam Sup. Valve (MO)	3.0	.60	.74
- Discharge Valve (MO)	3.0	.60	.74
- Min Flow Valve (MO)	3.0	.60	.74
- Check Valve	3.0	.60	.74
- RCIC Piping	3.0	.60	.74

* $HCLPF = A_m * \exp(-2.326 * \beta_c)$

Table 19I.2-1
ABWR SYSTEMS AND COMPONENTS/STRUCTURES FRAGILITIES (CONT.)

SYSTEM/COMPONENT	MED_CP (A_M)	LOG_STD (β_C)	HCLPF* (in g)
5. Low-Press Core Flooder (V1)			
- Pump (Motor Driven)	1.8	.46	.62
- Check Valve	3.0	.60	.74
- Injection Valve (MO)	3.0	.60	.74
- Discharge Valve (MO)	3.0	.60	.74
- LPCF Piping	3.0	.60	.74
6. RHR Heat Exchanger (HX)			
- Heat Exchanger	2.0	.45	.70
7. Reactivity Control Sys. (C)			
- Fuel Assemblies	1.4	.35	.62
- CRD Guide Tube	1.8	.36	.78
- CRD Housing	3.5	.46	1.20
- Shroud Support	2.0	.36	.87
- Hydraulic Control Unit	2.0	.50	.63
8. SRVs Close (PC, PC1)			
- Safety Relief Valve	3.0	.60	.74
9. Depressurization (X)			
- Safety Relief Valve	3.0	.60	.74
10. Level & Press. Control (LPL)			
- Safety Relief Valve	3.0	.60	.74
11. Inhibit ADS (PA)			
- Safety Relief Valve	3.0	.60	.74
12. Standby Liq. Cont. Sys. (C4)			
- SLC Tank	1.8	.46	.62
- SLC Pump	1.8	.46	.62
- Valve (Motor Operated)	3.0	.60	.74
- SLC Piping	3.0	.60	.74
13. Condensate Injection (V2)			
- Pump (Motor Driven)	1.8	.46	.62
- Injection Valve (MO)	3.0	.60	.74
- Piping	3.0	.60	.74
14. Firewater System (FW)			
- FW Tank	2.1**	.45	.79
- Pump (Diesel Driven)	1.8	.46	.62
- Injection Valve (Manual)	3.6	.60	.89
- FW Piping	3.0	.60	.74
- Valve (Manual)	3.6	.60	.89

* $HCLPF = A_m * \exp(-2.326 * \beta_c)$

** Firewater tank may be designed and built to a lower capacity if provision is made for a pumper truck and hose to go to an alternate water supply.

19I.3 EVENT TREE ANALYSIS

The event trees used in the ABWR Level 1 seismic margins analysis are shown on Figures 19I.3-1 through 19I.3-3. The individual paths through the event trees represent the accident sequences which are input to the HCLPF analysis. There is essentially only one seismic event tree, but it is presented on three figures representing transfers from Figure 19I.3-1 to Figures 19I.3-2 and 19I.3-3.

The event trees show the random failure probabilities and HCLPFs for each top event. Human error probabilities are included in the random failure probabilities.

19I.3.1 Support State Event Tree

The seismic event tree of Figure 19I.3-1 starts with the spectrum of seismic events, considers whether or not there is a structural failure (node SI), whether or not offsite power is lost (node LOP) and continues from there. Because of the ground rules of the analysis and the relative values of seismic fragilities, loss of structural integrity results in core damage, and survival of offsite power results in successful event termination. Thus, all remaining accident sequences on Figure 19I.3-1 are for cases of no structural failure, but always with loss of offsite power.

The success or failure of emergency AC power and/or service water (node APW), and the emergency DC power (station batteries) (node DP) are taken into consideration in Figure 19I.3-1 to account for support system dependencies. Failure of all DC power results in a high-pressure core melt since all control is lost, the high-pressure systems fail, and the reactor cannot be depressurized. The condition of successful emergency AC and DC power and successful scram is indicated by the ET transfer and is described in detail in Figure 19I.3-2. The condition of successful emergency AC and DC power, but with failure to scram is indicated by the ATWS transfer, and is described in Figure 19I.3-3.

The condition of failure of emergency AC continues on Figure 19I.3-1. The next questions are whether or not there is a loss of DC power (station batteries) and failure to scram (node C). Failure to scram is considered as a Class IV core melt. With successful DC power and scram, RCIC (node UR) and firewater (node FA) are the only available means of water injection into the RPV since all AC power is lost. Since station batteries will eventually discharge resulting in loss of RCIC, or if RCIC fails, the reactor must then be depressurized (node X) to allow firewater injection.

The firewater system has diesel driven pumps and all needed valves can be accessed and operated manually. No support systems are required for firewater operation. The firewater pump is housed in an external building (shed), whose collapse would not prevent the pump from starting and running. The failure probability of firewater is dominated by operator failure to initiate the system. For the upper branch, where RCIC is successful, the operator has 8 hours before the station batteries expire and RCIC trips. The human error probability (HEP) for this case is 1E-3. For the lower branch, where RCIC fails, the operator has only 30 minutes in which to depressurize the reactor and initiate firewater injection. For this case, the HEP is 0.1. In the event that the firewater diesel fails to start, the operator could make use of a fire truck, but this was not modeled.

If the RHR heat exchanger fails (node HX) due to the earthquake, it is presumed that the failure could include a pipe break that could partially drain the suppression pool into the RHR pump room. Fission product scrubbing would still be effective preventing a large release. These sequences are identified with a "P" (e.g., IB2-P).

19I.3.2 LOSP with Emergency Power and Scram Event Tree

In the event tree of Figure 19I.3-2 (ET transfer), there are two similar divisions depending on whether or not there is a stuck-open relief valve (node PC). If there is a stuck-open valve, the reactor will eventually depressurize causing loss of RCIC steam supply. The probability of having a stuck-open valve is 2E-3, based on operating experience. If both high-pressure injection systems fail, the reactor must be depressurized rapidly for low-pressure system use (LPFL -V1, or condensate injection -V2) In ABWR, condensate pumps can be transferred to the emergency bus by the operator. The HEP for this action is 0.1.

19I.3.3 ATWS Event Tree

Figure 19I.3-3 (ATWS transfer) represents failure to scram, and requires standby liquid control (automatic) and operator action to control reactor water level with the injection system(s) that are available. The HEP for this action is 0.01. In this ATWS analysis, if high-pressure systems fail, core damage results. No credit is given to low-pressure injection. For an ATWS, the probability of a stuck-open SRV was conservatively increased to 0.1, on the basis of increased SRV activity.

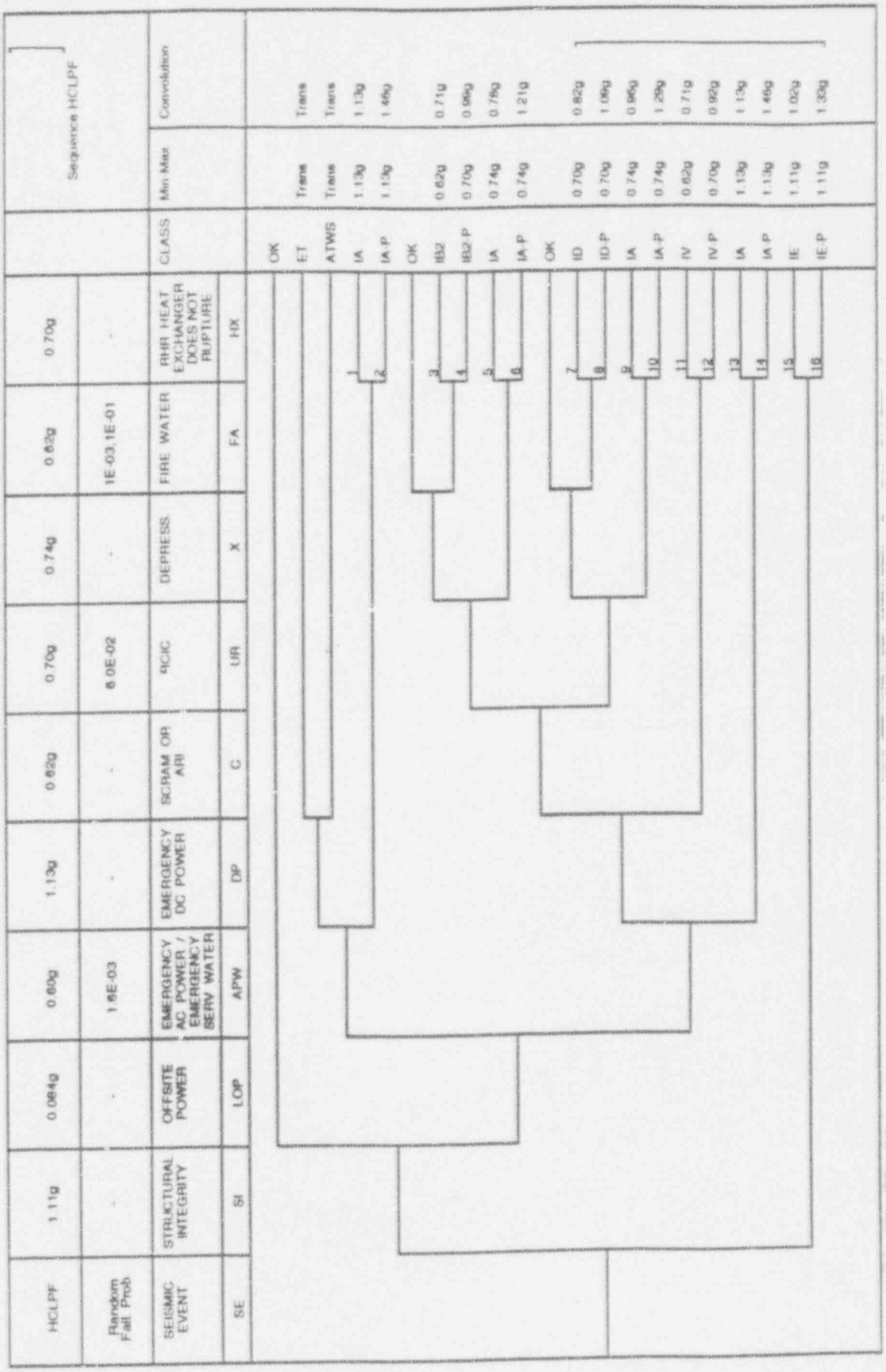

Figure 191.3-1
SEISMIC SUPP^RT STATE EVENT TREE

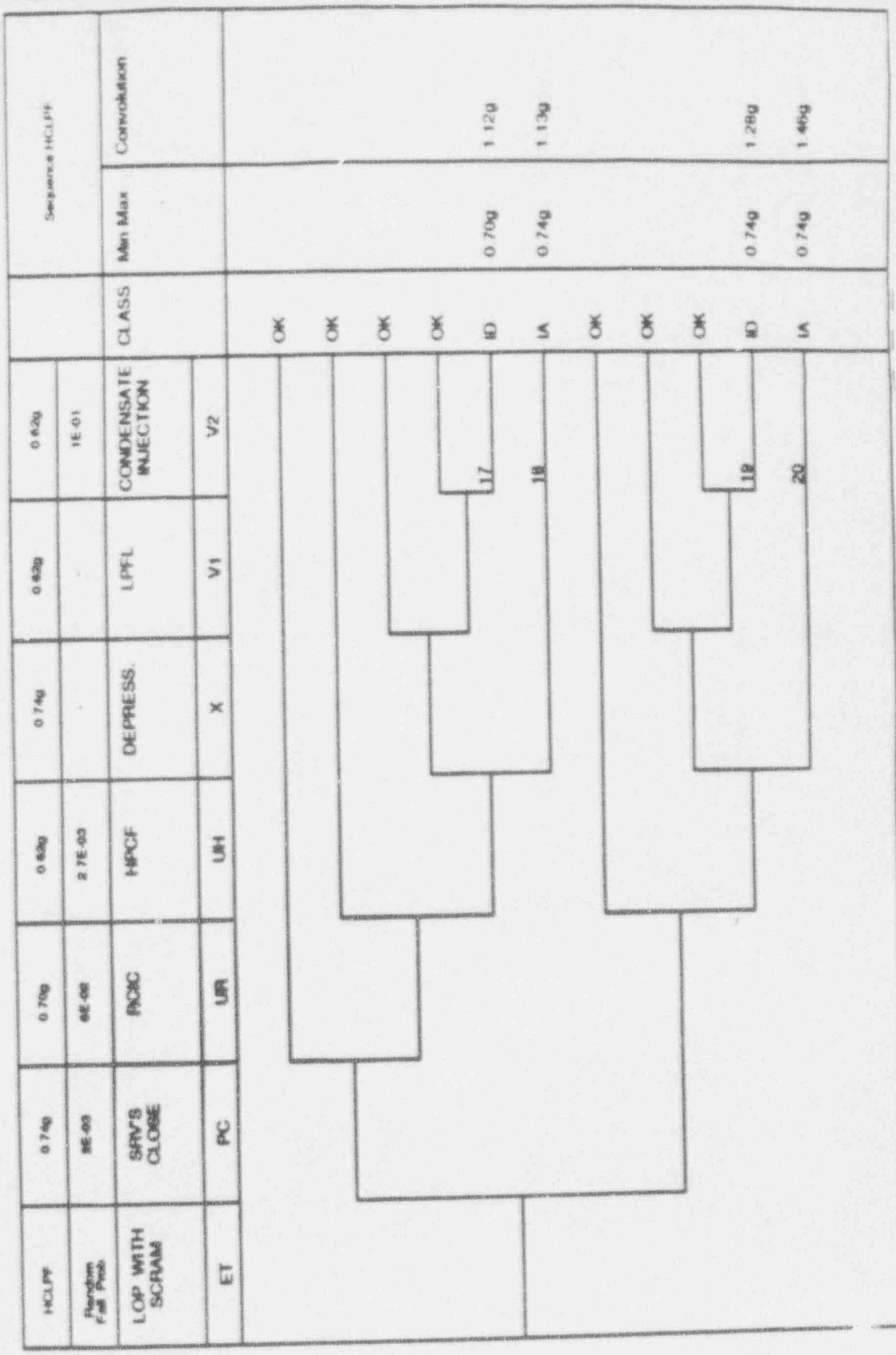

Figure 191.3-2
LOOP WITH SCRAM EVENT TREE

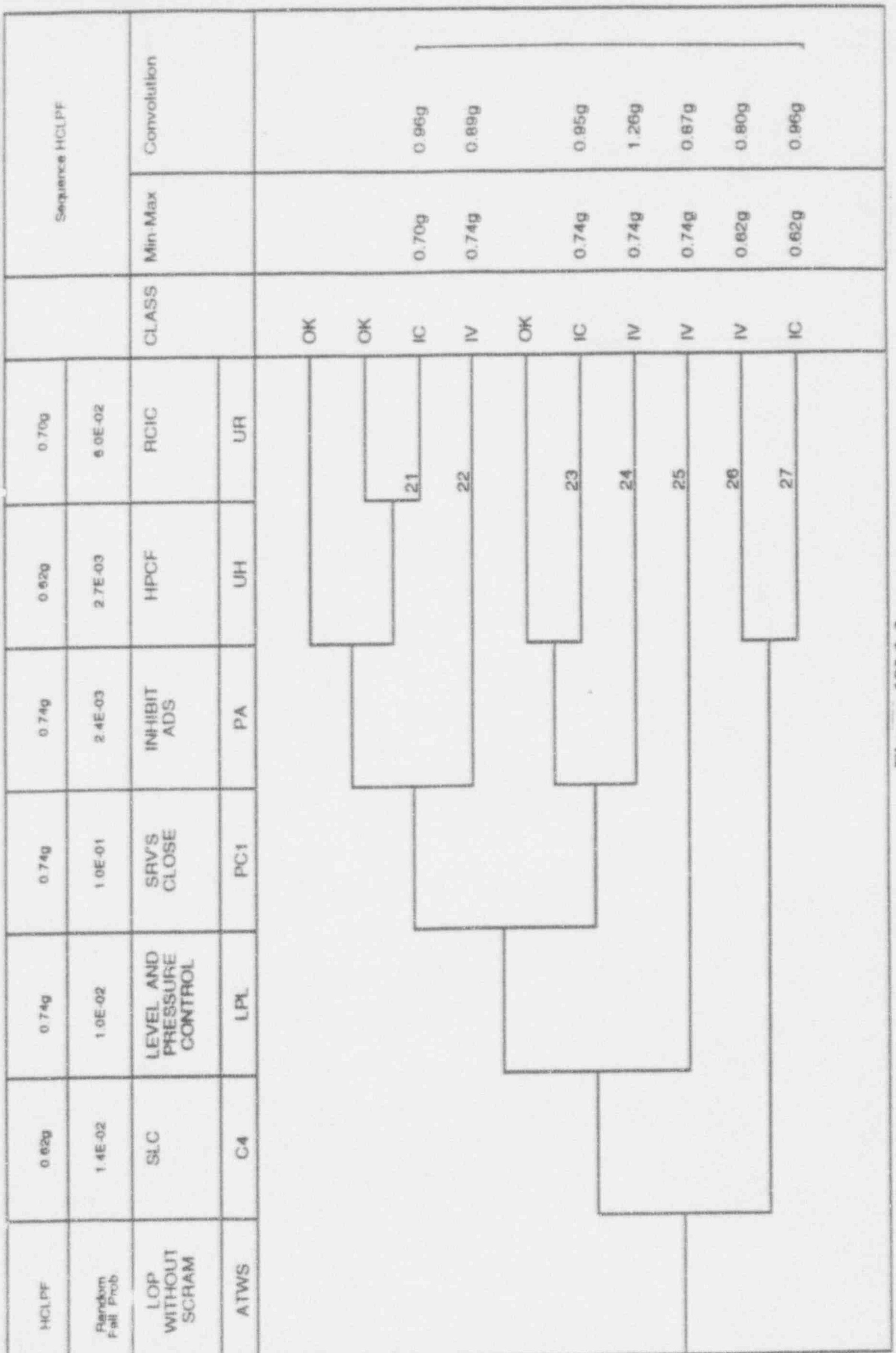

Figure 19I.3-3
ATWS (WITH LOP) EVENT TREE

19I.4 SYSTEM ANALYSIS

The fault trees used in the seismic system analysis are shown on Figures 19I.4-1 through 19I.4-12. The seismic system analysis calculates the probability of seismic failure of each of the important systems throughout the seismic ground acceleration spectrum. The system seismic failure probabilities are then input to the event trees and combined with random system failure probabilities and human errors. The seismic fault trees contain only those components that might be subject to seismic failure. Random system failure probabilities are taken from the internal events analysis and include all other components. One of the important ground rules of the seismic margins analysis is that all like components in a system always fail together.

The reactor protection system, control rod drive system, and alternate rod insertion system were not modeled since the failure of control rods to insert is dominated by the relatively low seismic fragility of the fuel assemblies, control rod guide tubes, and housings. A seismic fault tree for reactivity control is shown on Figure 19I.4-10. The fuel assemblies are the most fragile component.

A seismic fault tree for the standby liquid control system is shown on Figure 19I.4-11. Failure of the standby liquid control system is dominated by failure of two components: the pump and boron supply tank.

Since the most fragile essential component in the plant is the ceramic insulator in the switchyard, the loss of offsite power dominates the analysis and the availability of emergency power becomes very important. The loss-of-power fault tree (Figure 19I.4-7) is for emergency AC power. In the loss of emergency AC power fault tree, the more fragile components are the diesel generator , transformers, motor control centers, inverter and relay switch. The DC power fault tree (Figure 19I.4-8) has two branches: with and without availability of AC power. For the branch with AC power, the batteries <u>and</u> charger must fail.

Systems and equipment which require offsite power, such as the feedwater system, are not modeled since offsite power is presumed to be not available for the core damage sequences. The condensate injection system is modeled on Figure 19I.4-12 since credit is given to the operator for transferring condensate to an emergency bus (see Figure 19I.3-2.) The human error probability (0.01) is much greater than the seismically induced equipment failure probability, therefore, this fault tree has negligible impact on the HCLPF value of the corresponding accident sequences.

Essential service water is as important as emergency power, and its loss would have much the same effect as the loss of emergency power. The loss-of-service-water fault tree is shown on Figure 19I.4-9. The more fragile components in this system are the service water pump, heat exchanger, and room air conditioning unit. The service water pump house, with a HCLPF of 0.60, is also included in this fault tree.

Structure failures that could contribute to seismic core damage are shown on Figure 19I.4-6. In this analysis, any one or more of these structural failures are conservatively presumed to result in core damage. The structures having the lowest seismic capacity are the reactor building and control building.

The remainder of the fault trees are for core cooling and containment cooling (Figures 19I.4-1 through 19I.4-5). The more fragile components in these systems are the pumps, heat exchangers, and the firewater supply tank. The condensate storage tank (CST) is not modeled since the ECCS systems that take suction from the CST have automatic switchover to the suppression pool if CST level is low. Valves for the switchover are included in the fault trees.

Because of the importance of RCIC in station blackout sequences, differences between the seismic RCIC fault tree and the internal events fault tree are explained below:

(1) The internal events fault tree contains basic events that would not be affected by an earthquake, e.g., test and maintenance unavailability. These events contribute to the random failure probability during the seismic event and are included in the random failure part of the seismic analysis. They are deleted from the RCIC seismic fault tree.

(2) The internal events fault tree contains common-cause failure events. These are deleted from the RCIC seismic fault tree since a basic rule of the seismic analysis is that all like components within a system fail together.

(3) The internal events RCIC fault tree contains separate events for the turbine and for the pump. The seismic fault tree uses a combined event, "turbine-driven pump", since that is the assembly for which there is a seismic capacity.

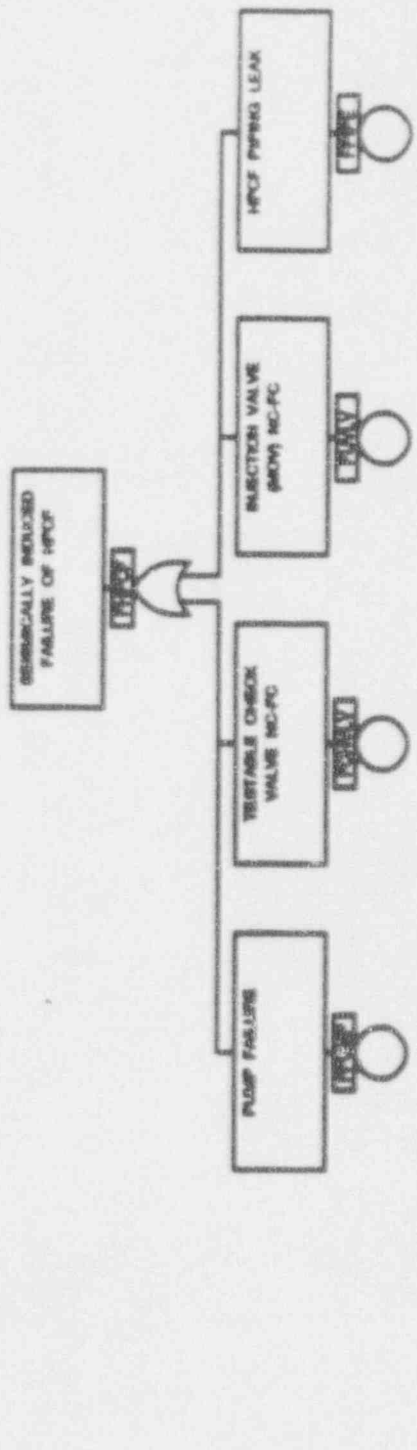

Figure 19I.4-1
HPCF SEISMIC FAULT TREE

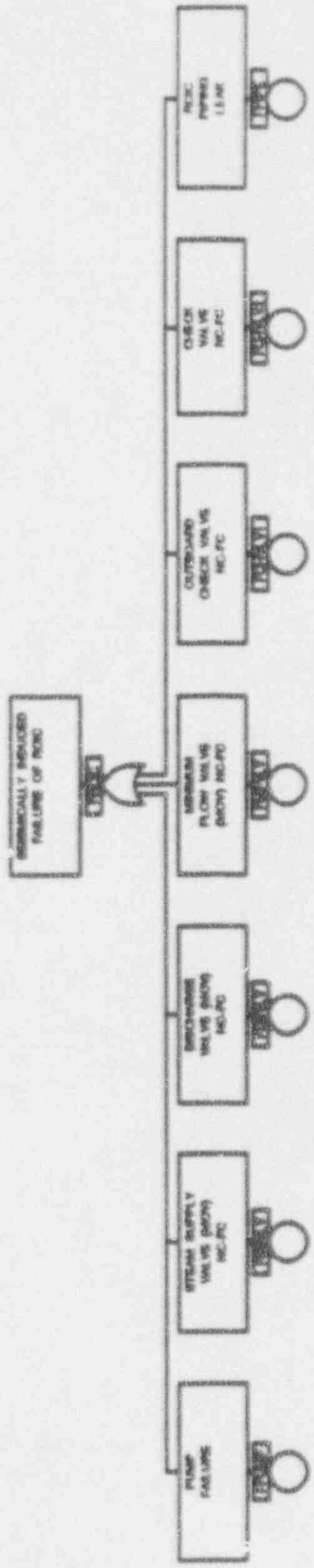

Figure 19I.4-2
RCIC SEISMIC FAULT TREE

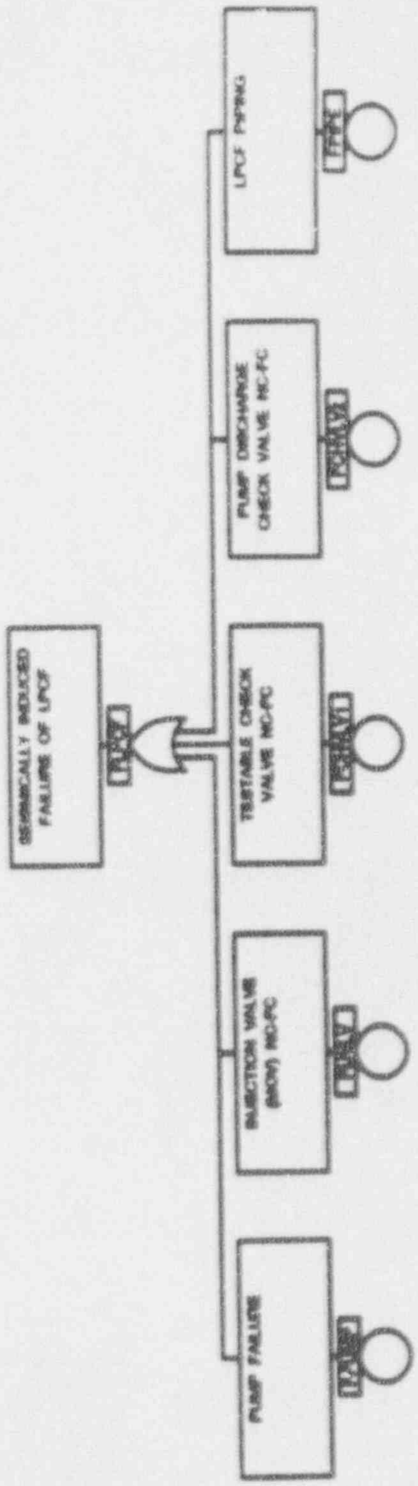

Figure 19I.4-3
LPCF SEISMIC FAULT TREE

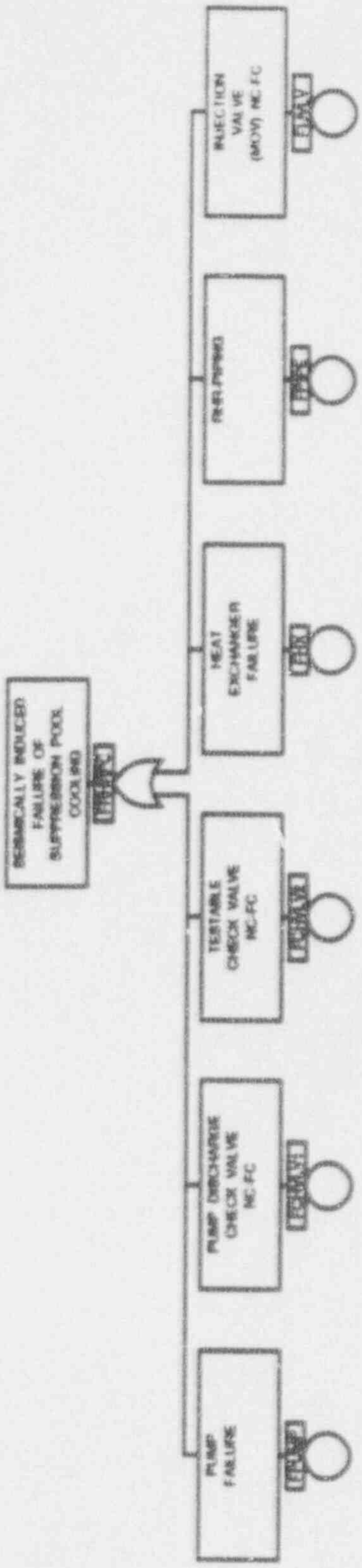

Figure 191.4-4
RHR SEISMIC FAULT TREE

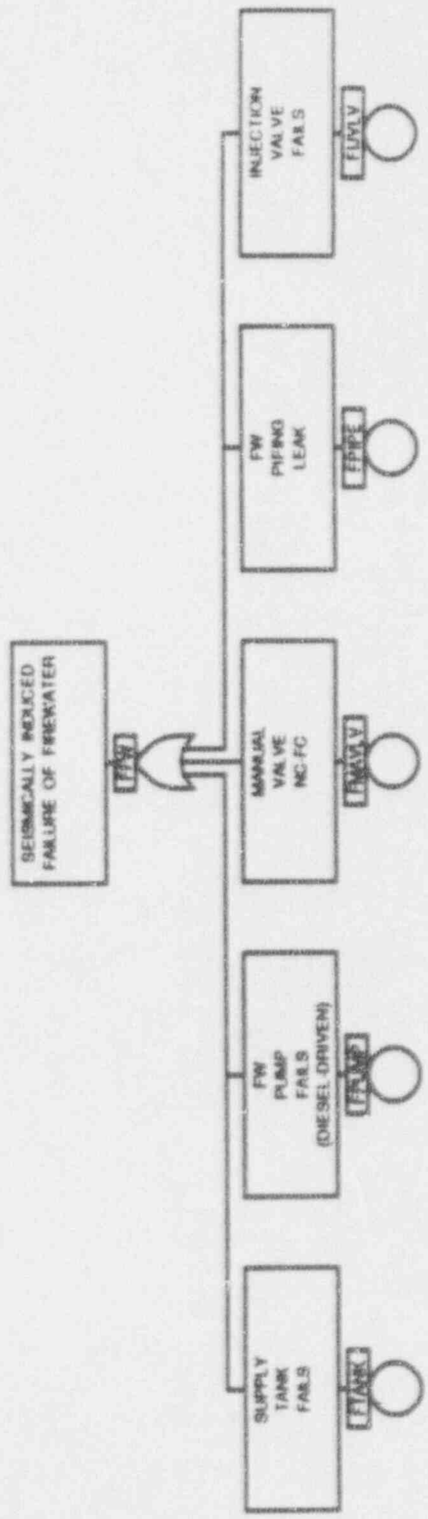

Figure 191.4-5
FW SEISMIC FAULT TREE

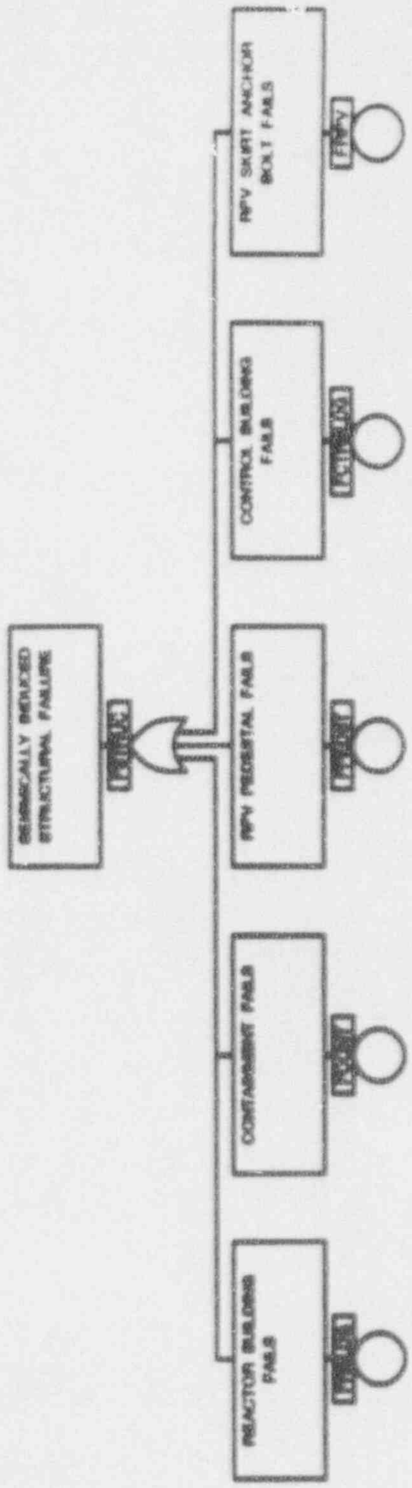

Figure 191.4-6
STRUCT. SEISMIC FAULT TREE

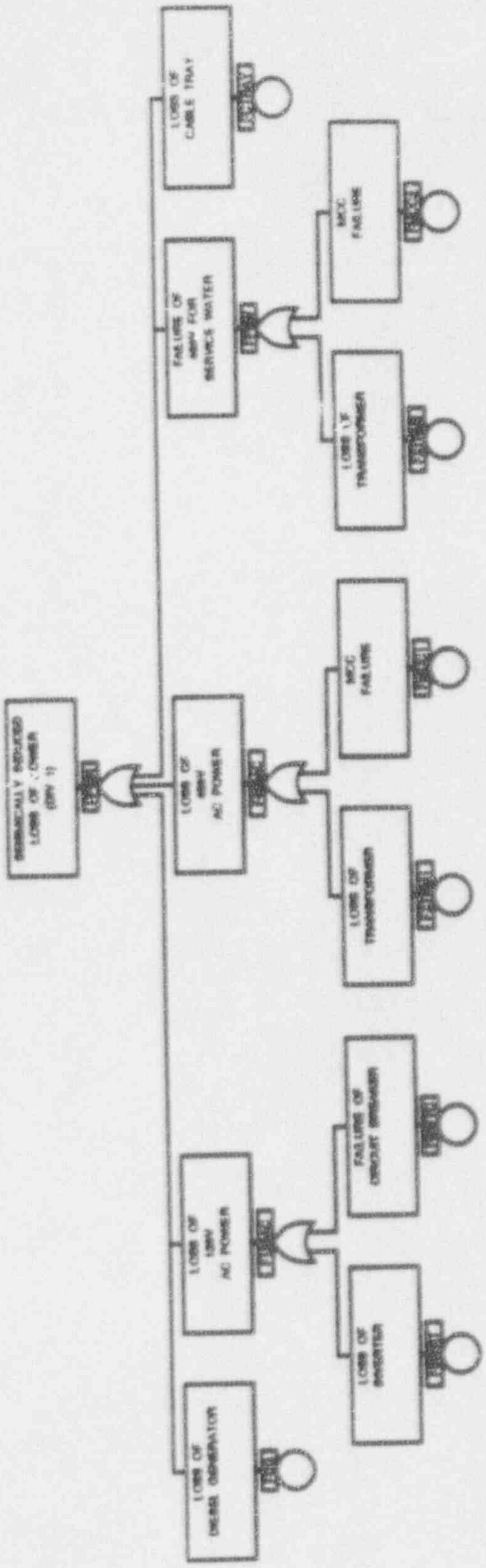

Figure 191.4-7
AC POWER SEISMIC FAULT

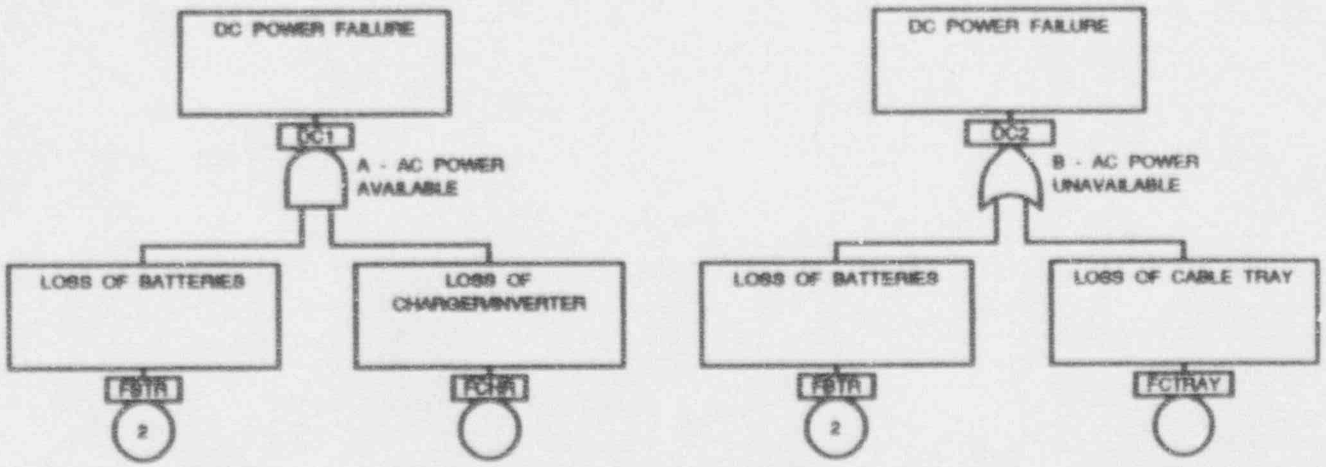

Figure 19I.4-8
DC POWER SEISMIC FAULT TREE

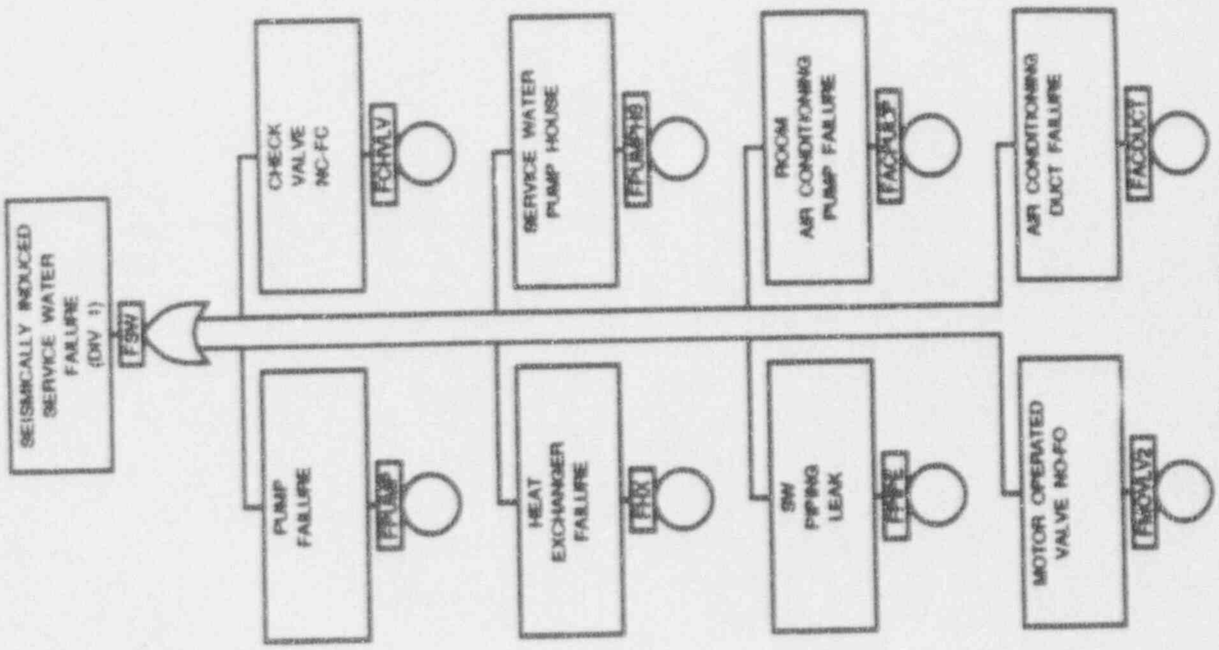

Figure 19I.4-9
SW SEISMIC FAULT TREE

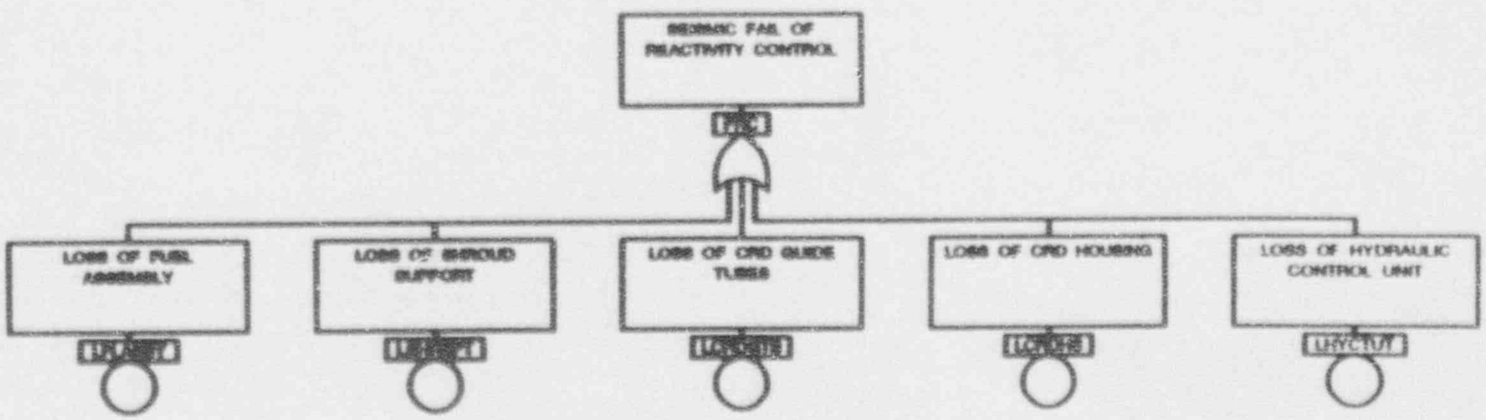

Figure 191.4-10
RC SEISMIC FAULT TREE

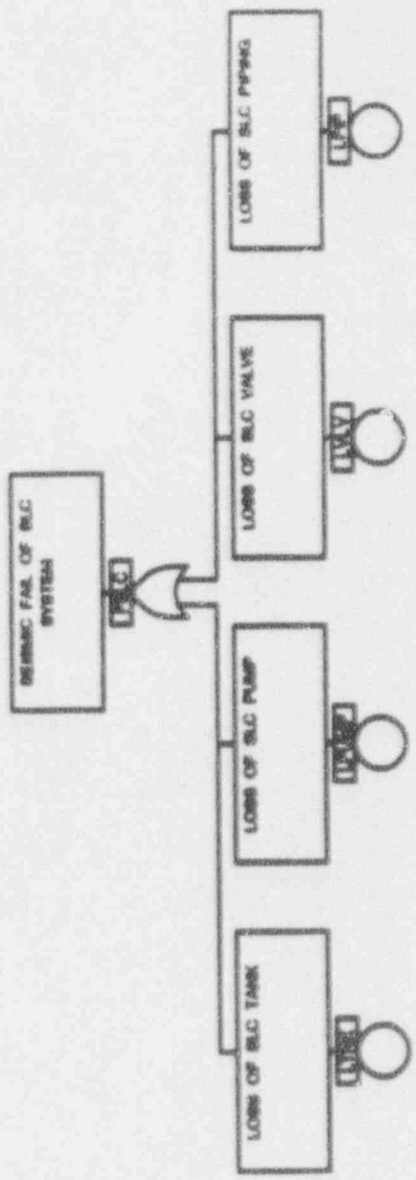

Figure 19I.4-11
SLC SEISMIC FAULT TREE

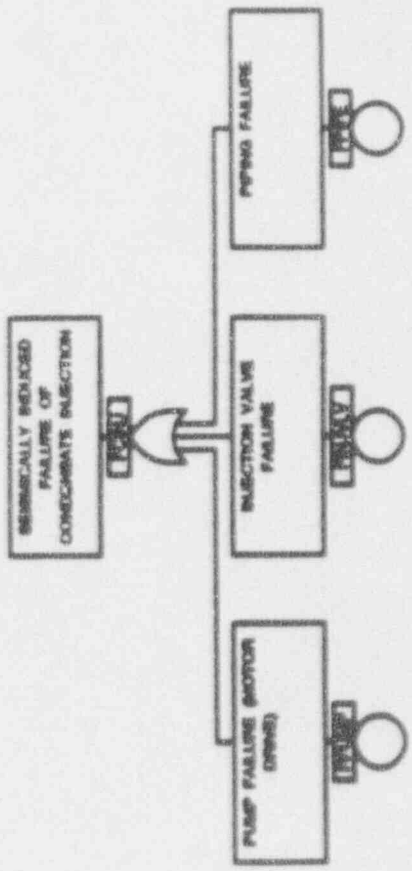

Figure 191.4-12
CI SEISMIC FAULT TREE

19I.5 ACCIDENT SEQUENCE HCLPF ANALYSIS

Seismic fragility of a structure or component is defined as the conditional probability of its failure as a function of peak ground acceleration. The probability model adopted for each component fragility is the log-normal distribution. The density function for the component fragility, f(g), can be written

$$f(g)=\frac{1}{\sqrt{2\pi}*\beta_c*g}\exp\left\{-1/2\left[\ln(g/A_m)/\beta_c\right]^2\right\} \text{ for } g>0$$

where: A_m = median capacity of the component,

β_c = logarithmic standard deviation of the fragility function,

g = peak ground acceleration.

The cumulative distribution of the component fragility, F(g), will then be

$$F(g)=\int_0^g \frac{1}{\sqrt{2\pi}*\beta_c*g_1}\exp\left\{-1/2\left[\ln(g_1/A_m)/\beta_c\right]^2\right\} dg_1$$

19I.5.1 Convolution Analysis

If a system, S, (or sequence) contains two components (A, B) operating in OR logic, the failure of either component will fail the system (S = A + B), and the cumulative fragility distribution of the system is one minus the product of their complementary cumulative fragility distributions :

$$F_S(g)=1-(1-F_A(g))*(1-F_B(g))$$

On the other hand, if two elements operate in AND logic, only the failure of both components will fail the system (S = A * B), and the cumulative fragility distribution of the system is the product of their cumulative fragility distributions:

$$F_S(g)=F_A(g)*F_B(g)$$

Using the two principles above, the distribution function of each system fragility is obtained by combining its component fragility functions based on its Boolean expression derived from the system fault tree.

Then the OR logic methodology is used to convolve the seismic and random/human failure probability of the systems. The combined cumulative fragility distribution of a system, $F_C(g)$, is the OR logic combination of the cumulative seismic fragility distribution, $F_S(g)$, and the cumulative random/human failure distribution, F_r, as follows:

$$F_c(g)=1-(1-F_s(g))*(1-F_r)$$

Similarly, the distribution for each accident sequence is derived from the combined system fragility functions by using the Boolean expression obtained from the seismic accident sequence event trees. The fifth and fiftieth percentiles of the combined cumulative distribution of each accident sequence are used to obtain the A_m and B_c for the corresponding sequence. Then, the HCLPF of each accident sequence is obtained by using the formula presented in the Introduction section as follows:

$$HCLPF = A_m*\exp(-2.326*\beta_c)$$

where the parameters A_m and β_c are the median capacity and logarithmic standard deviation of the lognormal distribution of the accident sequence.

19I.5.2 Min-Max Analysis

If a system, S, (or sequence) contains two components (A,B) operating in OR logic, the failure of any component will fail the system (S = A + B), and the cumulative fragility distribution of the system is governed by the fragility distribution of the weakest component. This principle is applied to the system fault trees, which generally are made up of OR gates.

If two elements operate in AND logic, only the failure of both components will fail the system (S = A * B), and the cumulative fragility distribution of the system is governed by the fragility distribution of the strongest component. This principle is applied to accident sequences, which are composed of ANDed elements.

Random/human failure probabilities greater than 1.0E-3 are combined with HCLPs for elements in an accident sequence as follows:

(HCLPF1 + RHP1)*(HCLPF2 + RHP2) =

HCLPF1*HCLPF2,
HCLPF1*RHP2,
HCLPF2*RHP1,
RHP1*RHP2,

where:	HCLPF1	=	the HCLPF of one event,
	RHP1	=	the random/human failure probability of that event,
	HCLPF2	=	the HCLPF of a second event, and
	RHP2	=	the random/human failure probability of the second event.

The resulting combinations are reduced according to min-max rules.

19I.6 RESULTS OF THE ANALYSES

The results of the convolution analysis are shown on the event trees and in Table 19I.6-1 in terms of HCLPF values for the accident sequences, with and without the inclusion of random failures. As seen in the event trees and the table, the HCLPF values for all accident sequences are greater than 0.60g, which is twice the safe shutdown earthquake (SSE = 0.30g). The results of the convolution analysis in terms of accident classes are shown in Table 19I.6-2.

The HCLPF value of accident sequences obtained from the min-max analysis are printed on the event trees next to the column of accident classes. The combination of HCLPF and random failure probabilities of accident sequences are described in Table 19I.6-3. As can be seen, no accident sequence has a HCLPF lower than 0.60g.

Table 19I.6-1
SEISMIC MARGINS FOR ABWR ACCIDENT SEQUENCES (CONVOLUTION METHOD)

Accident Sequence Number*	With Random Failure			Without Random Failure		
	HCLPF (in g)	MED_CAP (A_m)	LOG_STD (β_c)	HCLPF (in g)	MED_CAP (A_m)	LOG_STD (β_c)
1	1.13	3.30	0.46	1.13	3.30	0.46
2	1.46	3.52	0.38	1.46	3.52	0.38
3	0.72	1.40	0.29	0.72	1.40	0.29
4	0.99	2.13	.33	0.99	2.13	0.33
5	0.79	3.00	0.57	0.79	3.00	0.57
6	1.21	3.34	0.44	1.21	3.34	0.44
7	0.83	1.58	0.28	0.89	1.62	0.26
8	1.09	2.17	0.30	1.12	2.18	0.29
9	0.97	3.01	0.49	0.98	3.01	0.48
10	1.29	3.34	0.41	1.29	3.34	0.41
11	0.72	1.21	0.23	0.72	1.21	0.23
12	0.93	2.01	0.33	0.93	2.01	0.33
13	1.14	3.30	0.46	1.14	3.30	0.46
14	1.46	3.52	0.38	1.46	3.52	0.38
15	1.02	2.30	0.35	1.02	2.30	0.35
16	1.33	2.65	0.30	1.33	2.65	0.30
17	1.12	1.97	0.24	1.15	2.00	0.24
18	1.13	3.04	0.43	1.14	3.04	0.42
19	1.28	3.10	0.38	1.30	3.11	0.37
20	1.46	4.16	0.45	1.46	4.16	0.45
21	0.96	1.68	0.24	0.97	1.69	0.24
22	0.89	3.00	0.52	0.90	3.00	0.52
23	0.95	2.82	0.47	1.08	3.04	0.44
24	1.26	4.05	0.50	1.39	4.16	0.47
25	0.87	2.98	0.53	0.90	3.00	0.52
26	0.80	1.44	0.25	0.80	1.44	0.25
27	0.96	1.69	0.24	0.97	1.69	0.24

* See event trees

Table 19I.6-2
SEISMIC MARGINS FOR ABWR ACCIDENT CLASSES
(CONVOLUTION METHOD)

Accident Class	With Random Failure			Without Random Failure		
	HCLPF (in g)	MED_CAP (A_m)	LOG_STD (β_c)	HCLPF (in g)	MED_CAP (A_m)	LOG_STD (β_c)
IA	0.83	1.76	0.32	0.85	1.76	0.31
IB2	0.71	1.40	0.29	0.72	1.40	0.29
IC	0.90	1.44	0.20	0.92	1.46	0.20
ID	0.83	1.49	0.25	0.90	1.52	0.22
IE	1.02	2.30	0.35	1.02	2.30	0.35
IV	0.69	1.12	0.21	0.70	1.13	0.20
IA-P, IE-P	0.92	1.52	0.22	0.93	1.53	0.21

Table 19I.6-3
HCLPF DERIVATION FOR THE ABWR ACCIDENT SEQUENCES (MIN-MAX METHOD)

Sequence 1 : DP → 1.13g →
: 1.13g

Sequence 2 : DP*HX → 1.13g*0.7g →
: 1.13g

Sequence 3 : APW*FA → (0.60g+1.6E-3)*(0.62g+1.0E-3) →
: 0.62g, 0.60g*1.0E-3

Sequence 4 : HX*APW*FA → 0.70g*(0.60g+1.6E-3)*(0.62g+1.0E-3) →
: 0.70g

Sequence 5 : X*APW → 0.74g*(0.60g+1.6E-3) →
: 0.74g

Sequence 6 : HX*X*APW → 0.70g*0.74g*(0.60g+1.6E-3) →
: 0.74g

Sequence 7 : FA*UR*APW → (0.62g+1.0E-1)*(0.70g+6.0E-2)*(0.60g+1.6E-3) →
: 0.70g , 0.62g*6.0E-2, 0.60g*6.0E-3

Sequence 8 : HX*FA*UR*APW →
: 0.70g*(0.62g+1.0E-1)*(0.70g+6.0E-2)*(0.60g+1.6E-3) →
: 0.70g

Sequence 9 : X*UR*APW → 0.74g*(0.74g+6.0E-2)*(0.60g+1.6E-3) →
: 0.74g

Sequence 10 : HX*X*UR*APW → 0.70g*0.74g*(0.74g+6.0E-2)*(0.60g+1.6E-3) →
: 0.74g

Sequence 11 : C*APW → 0.62g*(0.60g+1.6E-3) →
: 0.62g

Sequence 12 : HX*C*APW → 0.70g*0.62g*(0.60g+1.6E-3) →
: 0.70g

Sequence 13 : DP*APW → 1.13g*(0.60g+1.6E-3) →
: 1.13g

Table 19I.6-3
HCLPF DERIVATION FOR THE ABWR ACCIDENT SEQUENCES
(MIN-MAX METHOD)
(CONTINUED)

Sequence 14 : HX*DP*APW → 0.70g*1.13g*(0.60g+1.6E-3) →
: 1.13g

Sequence 15 : SI →
: 1.11g

Sequence 16 : HX*SI → 0.70g*1.11g →
: 1.11g

Sequence 17 : V2*V1*UH*UR →
: (0.62g+1.0E-1)*0.62g*(0.62g+2.7E-3)*(0.70g+6.0E-2) →
: 0.70g , 0.62g*6.0E-2

Sequence 18 : X*UH*UR → 0.74g*(0.62g+2.7E-3)*(0.70g+6.0E-2) →
: 0.74g

Sequence 19 : V2*V1*UH*PC →
: (0.62g+1.0E-1)*0.62g*(0.62g+2.7E-3)*(0.74g+2.0E-3) →
: 0.74g , 0.62g*2.0E-3

Sequence 20 : X*UH*PC → 0.74g*(0.62g+2.7E-3)*(0.74g+2.0E-3) →
: 0.74g

Sequence 21: UR*UH*C → (0.70g+6.0E-2)*(0.62g+2.7E-3)*0.62g →
: 0.70g , 0.62g*6.0E-2

Sequence 22 : PA*C → (0.74g+2.4E-3)*0.62g →
: 0.74g , 0.62g*2.4E-3

Sequence 23 : UH*PC1*C → (0.62g+2.7E-3)*(0.74g+1.0E-1)*0.62g →
: 0.74g , 0.62g*1.0E-1

Sequence 24 : PA*PC1*C → (0.74g+2.4E-3)*(0.74g+1.0E-1)*0.62g →
: 0.74g

Sequence 25 : LPL*C → (0.74g+1.0E-2)*0.62g →
: 0.74g , C.62g*1.0E-2

Sequence 26 : C4*C → (0.62g+1.4E-2)*0.62g →
: 0.62g

Sequence 27 : UH*C4*C → (0.62g+2.7E- 3)*(0.62g+1.4E-2)*0.62g →
: 0.62g

19I.7 CONTAINMENT ISOLATION AND BYPASS ANALYSIS

In the seismic margins analysis there were no cutsets leading to core damage with HCLPF values lower than 0.6g. A supplemental analysis was conducted to evaluate the HCLPF values for containment isolation for events that could cause containment bypass as a result of an earthquake, with potential for large releases to the environment.

Based on the results of the bypass analysis discussed in Subsection 19E.2.3.3 and shown on Figure 19E.2-19, the events selected for evaluation in this analysis are:

(1) Main steam lines (see Figure 19E.2-19A),

(2) Feedwater or SLC injection lines (see Figure 19E.2-19B),

(3) Reactor instrument, RWCU instrument, LDS instrument/sample or containment atmosphere monitoring lines (see Figures 19E.2-19D, 19E.2-19E, and 19E.2-19F, respectively),

(4) RCIC steam supply or RWCU suction lines (see Figure 19E.2-19E),

(5) Post accident sampling lines (see Figure 19E.2-19J),

(6) Drywell sump drain line (see Figure 19E.2-19J),

(7) SRV discharge lines (see Figure 19E.2-19K),

(8) ECCS lines (see Figure 19E.2-19C),

(9) Drywell inerting/purge lines (see Figure 19E.2-19I),

(10) Wetwell/drywell vacuum breaker lines (see Figure 19E.2-19G).

The bypass paths for atmospheric control system crosstie lines (Figure 19E.2-19H) require inadvertent opening of two normally closed motor operated valves. Since the seismic analysis does not consider a fail-open mode for normally closed valves, these bypass paths are not included in the analysis.

In the bypass analysis of Subsection 19E.2.3.3, several potential bypass pathways were excluded from detailed analysis on the basis of various reasons. The reasons are discussed in Subsection 19E.2.3.3.2 and Table 19E.2.-1. These reasons were reviewed to determine whether they remain valid in regard to seismic events. All but one of the reasons are based on configuration details that would not be affected by an earthquake. RHR wetwell and drywell spray lines were excluded on the basis that the pipes are designed for higher internal pressures than will be seen in actual operation and would thus have a very low probability of breaking. In this case, the seismic event could increase the probability of a break in these lines. However, these pipes have very high seismic capacity (3.0g) with very low probability of breaking due to a seismic event.

An event tree was constructed for each of the above events. These event trees are shown on Figures 19I.7-1 through 19I.7-10. All event trees start with the earthquake as the initiating event followed by a core-damaging accident. If there is no core damage there is no large release. The HCLPF and random failure probability are shown for each branch point, and the sequence HCLPFs using convolution and min-max methods are also shown on the figures.

Figure 19I.7-1 is for suppression pool bypass via main steam lines. Following the earthquake and accident, the question is asked whether or not there is a break in a main steam line outside containment. If there is a break, the question is asked whether or not at least one MSIV in each steam line closes to isolate the break. For the case where there is no break, there could still be a bypass release to the main condenser if a turbine bypass valve is open - unless the MSIVs are closed to isolate the break. The two bypass sequences for this event both have min-max HCLPF capacities of 0.74g.

Figure 19I.7-2 is an event tree for bypass via feedwater or standby liquid control lines. These lines inject into the RPV and are protected from reverse flow by redundant check valves. These check valves provide isolation of upstream breaks provided that one of the valves closes in the line with the break. The two bypass sequences for this event also have min-max HCLPF capacities of 0.74g.

Figure 19I.7-3 is for bypass via reactor instrument, RWCU instrument, LDS instrument, LDS sample or containment atmosphere monitoring lines. These lines are also protected by check valves, a single valve in each line. The bypass sequence for this case also has a min-max HCLPF of 0.74g.

Figure 19I.7-4 is for bypass via either the RCIC steam supply line or the RWCU suction line. Both of these lines are protected by motor operated isolation valves which require power. Since offsite power is lost due to the earthquake, emergency power is required. The two bypass sequences for this event both have min-max HCLPFs of 0.74g.

Figure 19I.7-5 is for bypass via the post accident sampling lines. These lines are also isolated by motor operated valves. The bypass sequences for this event also have min-max HCLPFs of 0.74g.

Figure 19I.7-6 is for bypass via the drywell sump drain line. This line is protected by a motor operated isolation valve and a check valve. Both components have HCLPF capacities of 0.74g and the two bypass sequences have min-max HCLPFs of 0.74g.

Figure 19I.7-7 is for bypass via the SRV discharge lines. If there is a break in an SRV discharge line during a core-damaging accident, and that SRV is open, a bypass pathway will exist. In this analysis, it is assumed that the SRV will be open during the accident. The resulting HCLPF capacity for this sequence is the capacity of the SRV discharge line (0.74g).

Figure 19I.7-8 is for bypass via any of the ECCS lines. The lines of concern are the HPCF and LPFL warm-up and discharge lines. These lines are protected by motor operated isolation valves and check valves. The resulting min-max HCLPF capacity is 0.74g.

Figure 19I.7-9 is for bypass via drywell inerting/purge lines. These lines are protected by air operated valves. The bypass sequence for this case also has a min-max HCLPF value of 0.74g.

Figure 19I.7-10 is for bypass via wetwell/drywell vacuum breaker lines. It requires an inadvertent opening of a vacuum breaker (check valve) to initiate a bypass during a severe accident. The bypass sequence for this case also has a HCLPF of 0.74g.

All sequences for all events in the bypass analysis have min-max HCLPF capacities of 0.74g which is significantly larger than 0.60g (two times SSE) and therefore, no further analysis is needed. (The reason that all bypass sequences have the same HCLPF value (0.74g) is that the failures are always either pipe or valve failures, both of which have the same seismic capacity (3.0g).

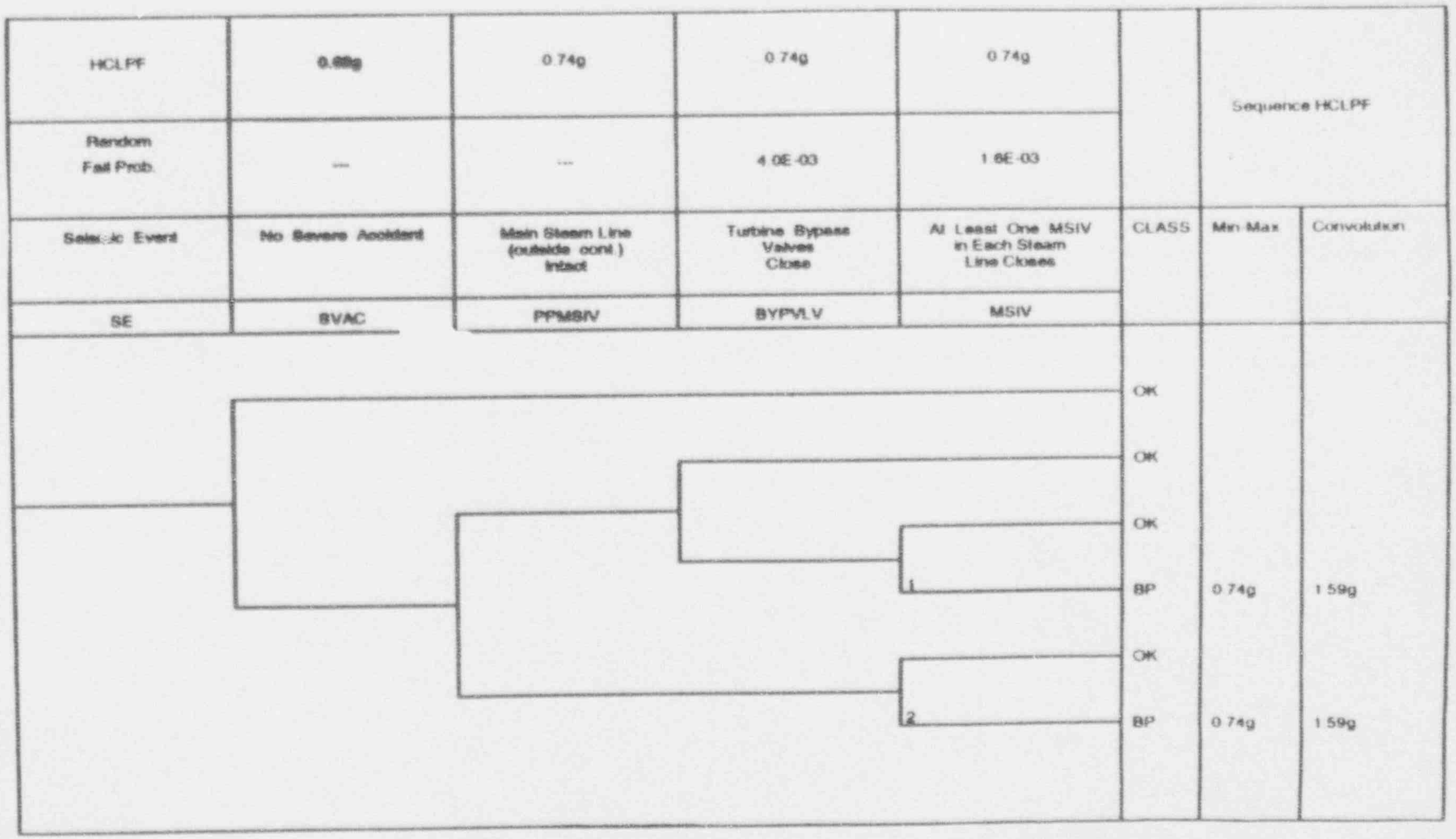

Figure 191.7-1
SUPPRESSION POOL BYPASS VIA MAIN STEAM LINES

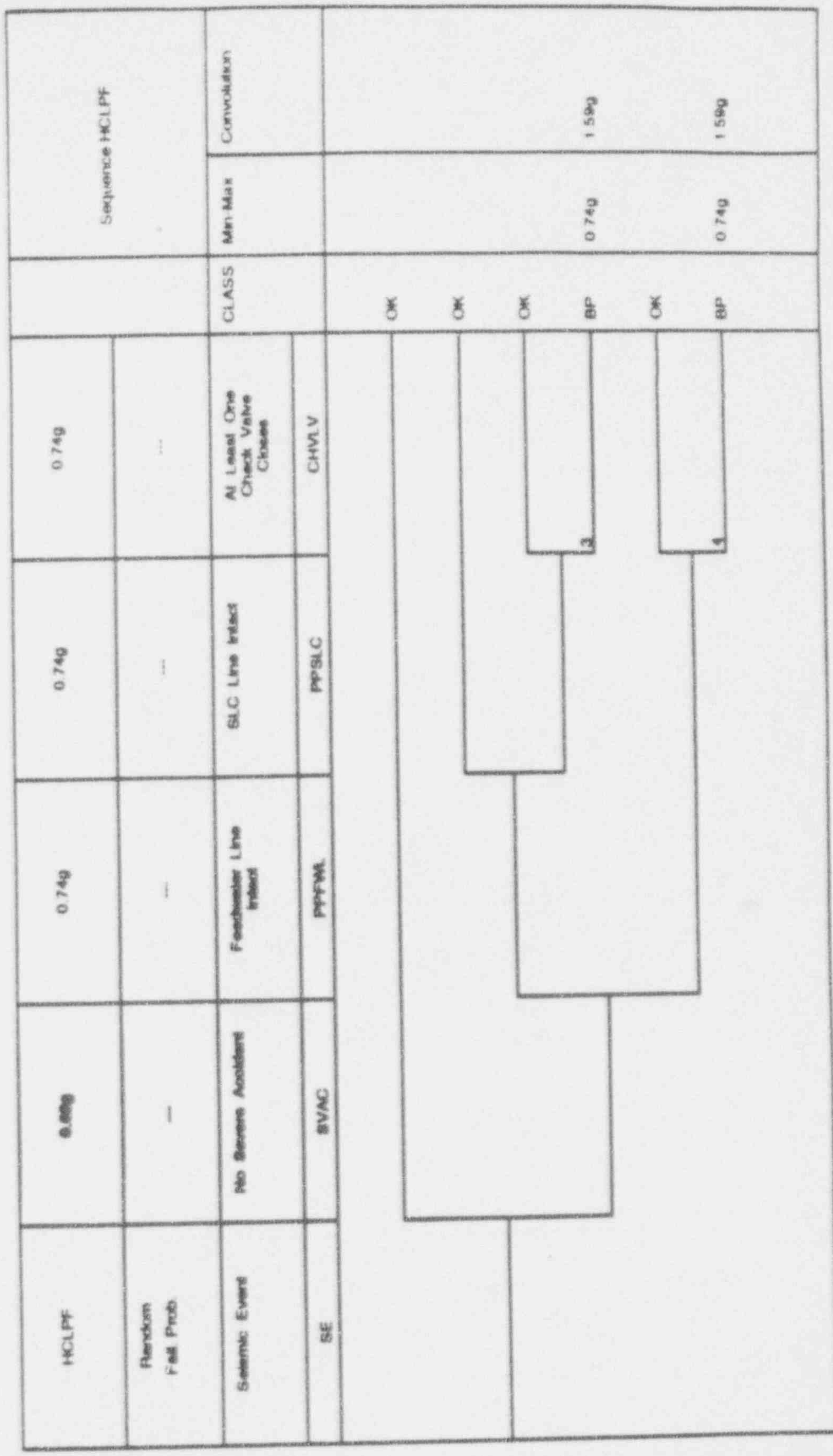

Figure 19I.7-2
SUPPRESSION POOL BYPASS VIA FEEDWATER OR SLC LINES

DRAFT

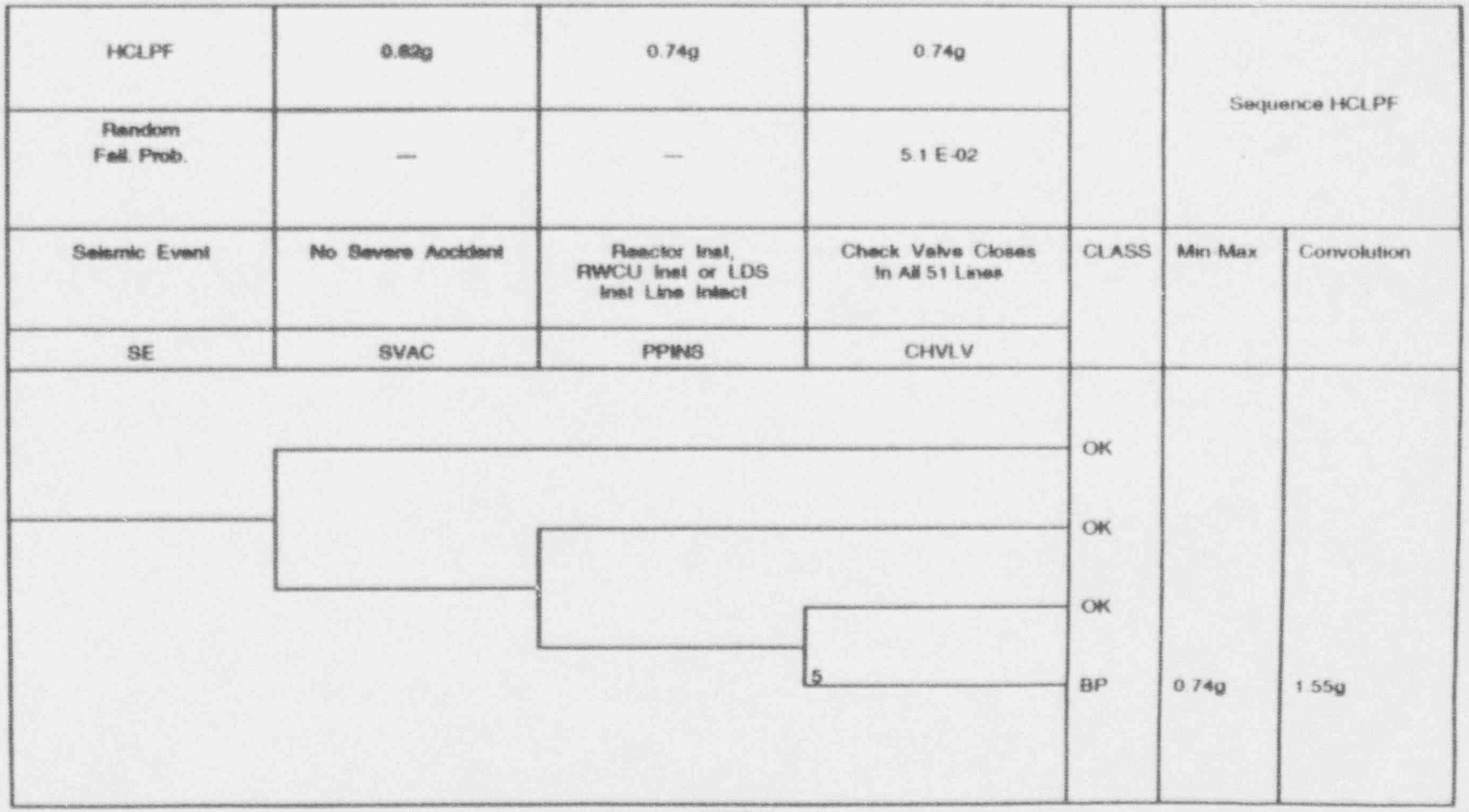

Figure 191.7-3

SUPPRESSION POOL BYPASS VIA REACTOR INST, RWCU INST, LDS INST AND SAMPLE OR CONT. ATMOSPHERE MONIT. LINES

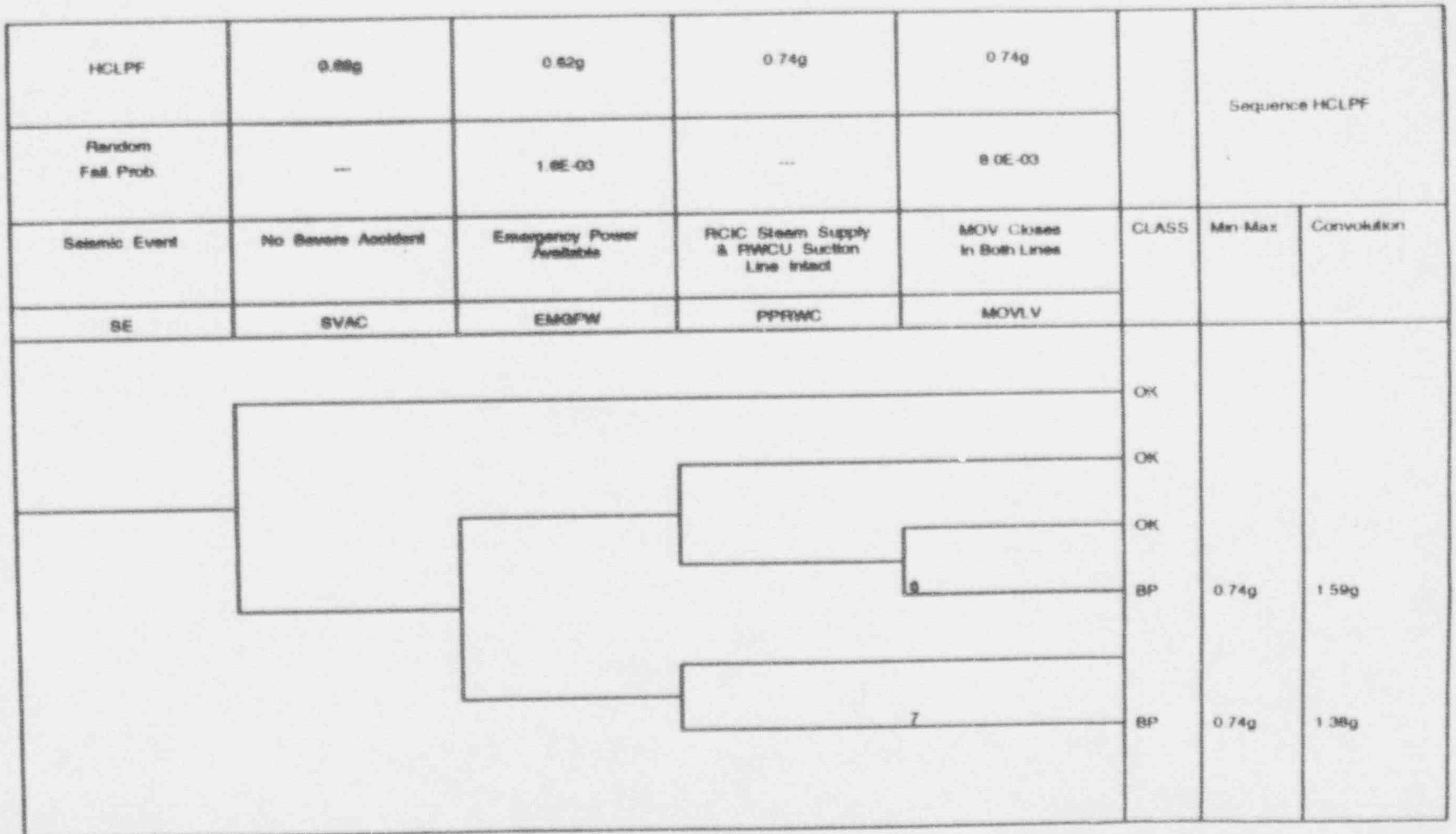

Figure 191.7-4
SUPPRESSION POOL BYPASS VIA RCIC STEAM SUPPLY LINE OR RWCU SUCTION LINE

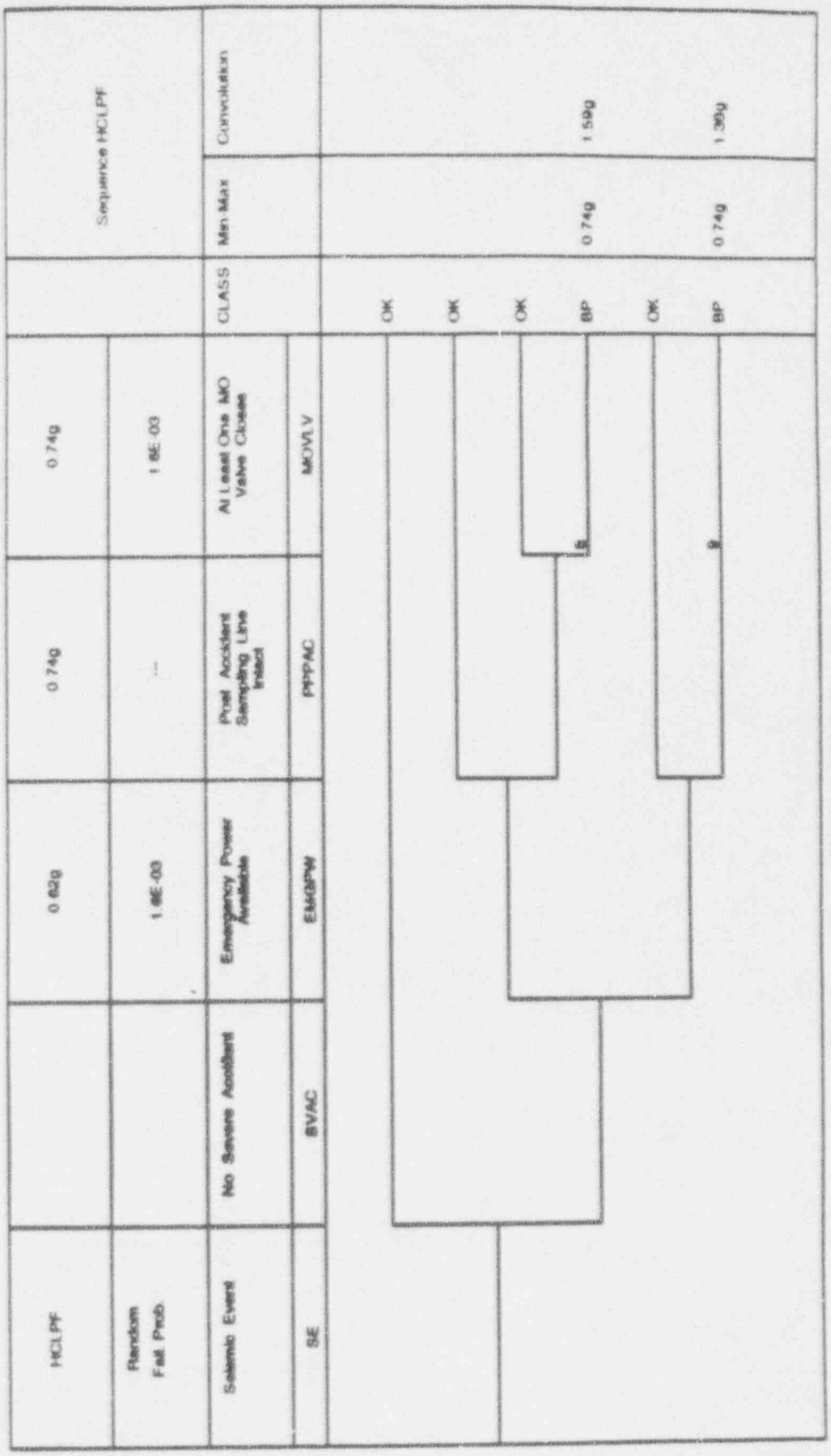

Figure 19I.7-5
SUPPRESSION POOL BYPASS VIA POST-ACCIDENT SAMPLING LINES

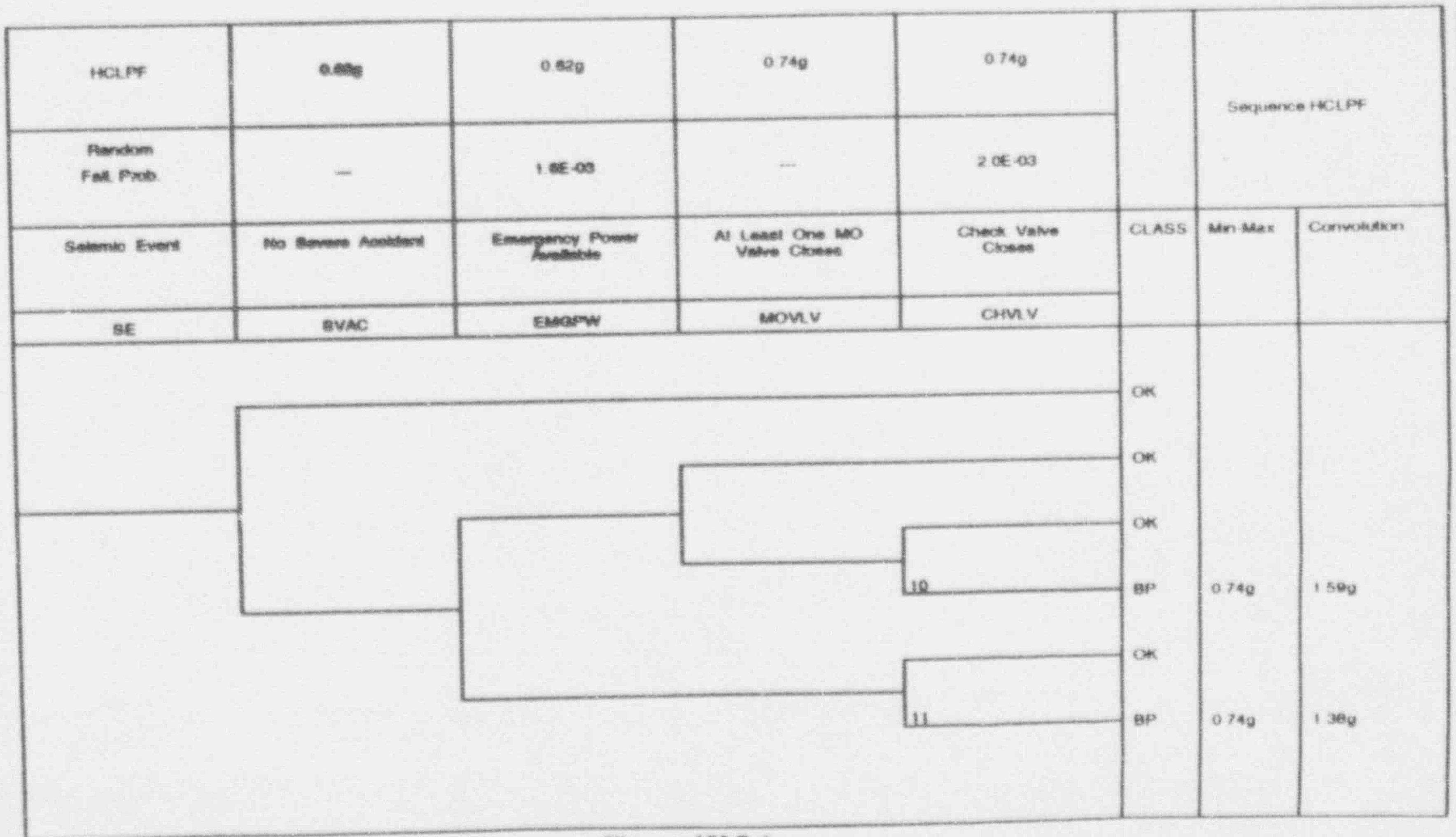

Figure 19I.7-6
SUPPRESSION POOL BYPASS VIA DRYWELL SUMP DRAIN LINE

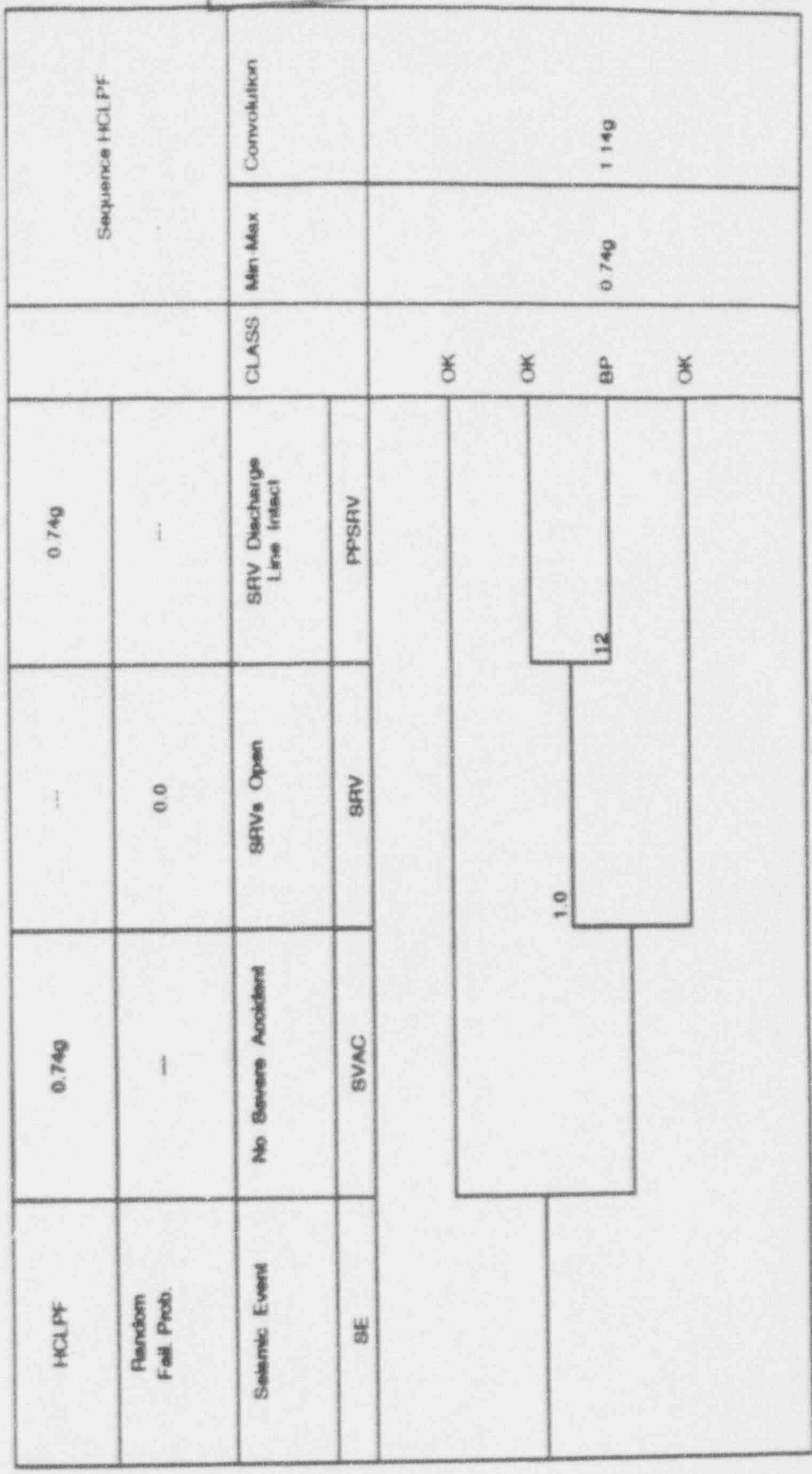

Figure 19I.7-7
SUPPRESSION POOL BYPASS VIA SRV DISCHARGE LINES

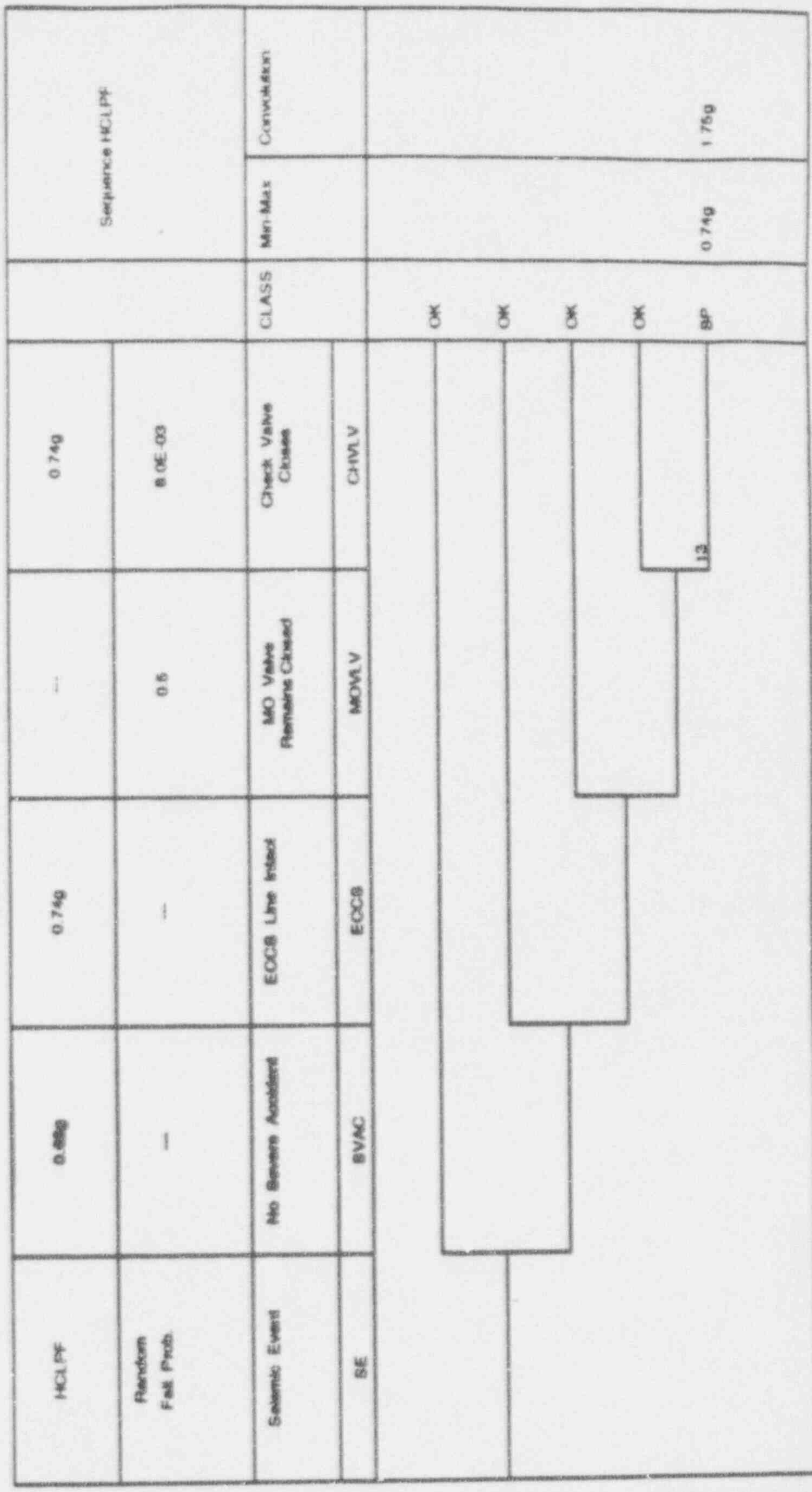

Figure 191.7-8
SUPPRESSION POOL BYPASS VIA ECCS LINES

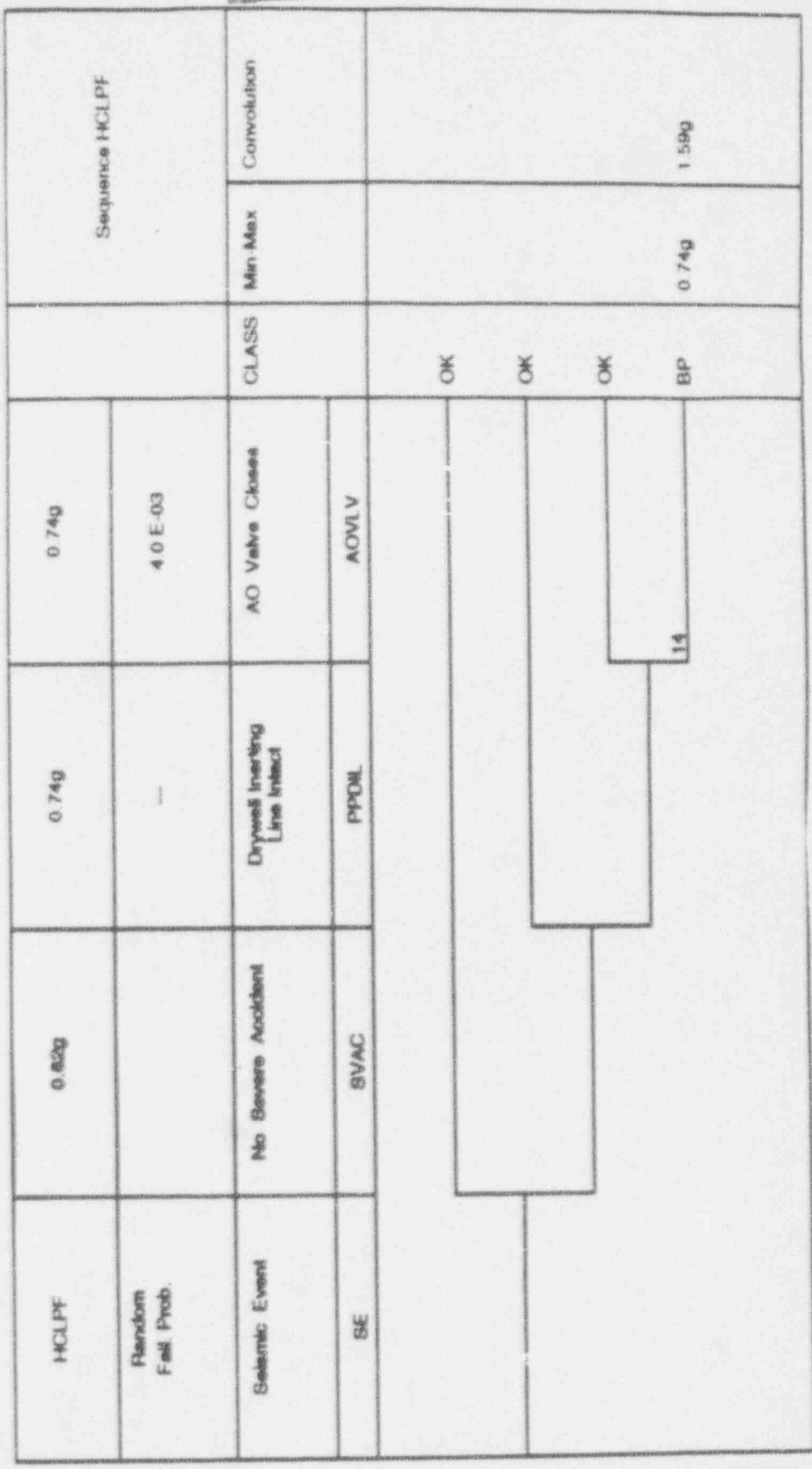

Figure 19I.7-9
SUPPRESSION POOL BYPASS VIA DRYWELL INERTING/PURGE LINES

DRAFT

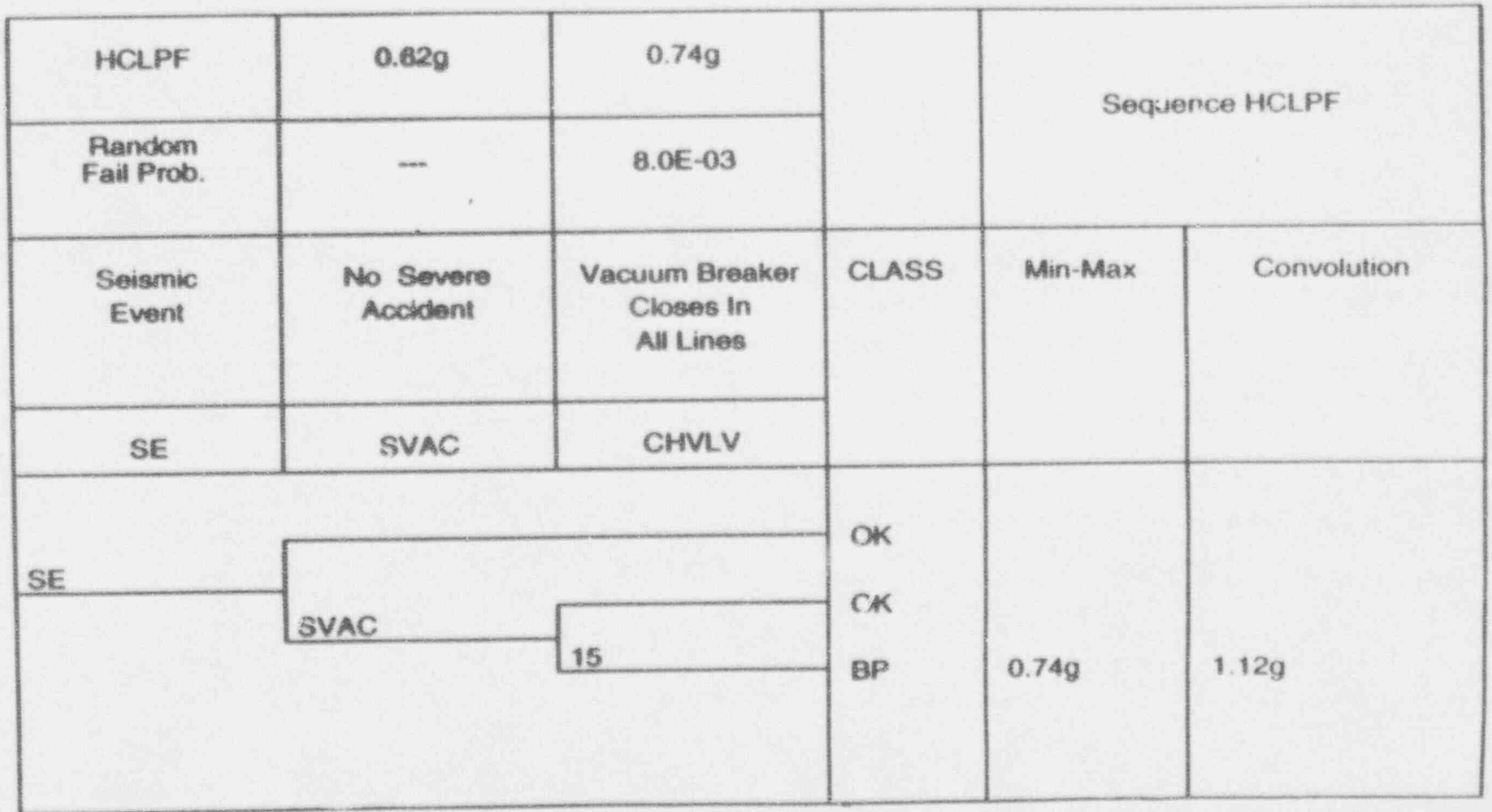

Figure 19I.7-10
SUPPRESSION POOL BYPASS VIA WETWELL/DRYWELL VACUUM BREAKER LINES

19I.8 REFERENCES

1. R.P. Kennedy, et al, *Assessment of Seismic Margin Calculation Methods*, NUREG/CR-5270, Lawrence Livermore National Laboratory, March 1989.

Post-It™ brand fax transmittal memo 7671	# of pages ▸ 35
To BOB PALLA	From LARRY FREDERICK
Co. NRC	Co. G.E.
Dept.	Phone # 408 925-6188
Fax # 301 504-2260	Fax # 408 925-1412

Bob:

Here is the SSAR section on Human Reliability. I believe this writeup now contains our answers to all NRC questions re human factors in the PRA.

I have marked it as a draft to avoid confusion in case there are any changes in the final issue in the SSAR. (I don't foresee any changes.)

Call me if you have any questions.

Larry F.
12/16/98

CC: J. D. Duncan (G.E.)
J. N. Fox (G.E.)

19D.7 HUMAN ERROR PREDICTION

ABWR human reliability analyses were performed by GE personnel. The GE reliability engineering staff has extensive and diverse experience gained through the performance of many significant PRA/PSA programs. These have included three major PRAs that have received regulatory agency review and approval: BWR/6, GESSAR II, Limerick, and Alto Lazio. Performance of human error analyses was an integral part of each of these activities.

An important outcome of these efforts and accumulated experience is the recognition that basic knowledge of BWR plant design, plant procedures, and accident analysis is a key factor in realistically addressing human reliability analysis.

This GE overall BWR knowledge base and direct access to ABWR design engineers and design documentation, in combination with prior BWR human reliability experience, provided the basis for the reliability engineering staff to realistically address human reliability factors in the ABWR PRA analyses.

Results of previous HRAs, which are based upon conventional BWR man-machine interface designs, were used to provide the human reliability assumptions needed for the ABWR PRA. The previous HRA results are considered to be conservative for the ABWR because of the significant improvements in the ABWR man-machine interface design relative to the earlier designs.

19D.7.1 HEPs in the Level 1 PRA

HEPs used in the Level 1 analysis are presented in the applicable component failure rate data tables which accompany each system fault tree presented in Section 19D.6, as well as the tables which document branch point values for each accident sequence event tree in Section 19D.4. Many of these HEPs were taken from the GESSAR II PRA (Reference 1) for which they were collected from various other sources, and modified as appropriate, for the GESSAR application. Many of these values were derived (directly or indirectly) from the Swain and Guttman *Handbook of Human Reliability* (Reference 2). More recent studies suggest that these values may be somewhat conservative. Their application to the ABWR PRA analyses is judged to be acceptable.

Level 1 HEPs also are summarized in Table 19D.7-1, giving the computer designation, the failure probability, identification of the fault tree in which the HEP appears, and a reference for derivation of the HEP value. The first five actions on the list are the most important to the Level 1 analysis because of their effect on core damage frequency. These five actions are discussed in the sensitivity analysis (Section 19D.7.7), including identification of needed provisions in the plant design and procedures related to these operator actions.

The sixth action on the list (LPL - control of water level in an ATWS), although not a significant contributor to CDF (because of the low probability of ATWS), would be very important given an ATWS. Because of this, it is important that the same provisions be made related to this action, as are required for the first five actions, namely:

(1) The operator must have a clear unambiguous indication of the conditions requiring the action.

(2) The operator must have the capability of performing the necessary action from the main control room in a simple straightforward manner.

(3) The operator must have clear written operating procedures regarding the action to be taken.

(4) The operator must have thorough training in the conditions requiring the action.

In general, human errors of both omission and commission are expected to be minimized by operator training and symptom-oriented emergency procedures. In addition, in most cases, substantial opportunity exists for peer and supervisory intervention within the response times available during accident sequences, prior to core damage or loss of containment integrity.

Incorporation of human actions in fault and event trees is relatively straightforward where single overt actions are required to initiate or inhibit system functions. This type of application was predominant in the treatment of human error in the ABWR PRA. Exceptions included the use of screening values based on successful performance of an estimated number of required operations, and the detailed modeling of instrument miscalibration.

The calibration of sensors was identified in WASH-1400 (Reference 3), the *Handbook of Human Reliability*, GESSAR, and other PRAs as being a dominant failure mode for all sensors or instrument systems that are required to initiate typical ECCS functions. The most probable scenario for common mode miscalibration of sensors was identified as that in which a miscalibrated standard is used, the instrument technician fails to recognize the error in the calibration tool, and consequently all sensors measuring that particular parameter are miscalibrated.

The model used to evaluate the above sequence of events in the ABWR PRA is illustrated in Figure 19D.7-1, and includes the assumptions made regarding individual probabilities. This model was initially developed for the GESSAR PRA and is judged applicable to the ABWR analysis. The resulting failure probability for miscalibration of four sensors measuring a single parameter is approximately 2.0E-05. This value was used in the instrumentation system fault trees as appropriate (HFELEBHX, AHPT006, RFE635HX). In cases where one set of sensors is used as initiator or permissive for more than one ECCS system, such commonalities are accounted for in functional fault tree evaluations.

The model of Figure 19D.7-1 was also used to derive the value for the probability of miscalibrating a single sensor (5.0E-5) as applied to the following HEPs:

RPR005CF
RFL007CF
CALN002A
HFE008CF
HPR007CF
REOSSMSC
RPR309MC
RMOSSMSC
RPR303MC

Although miscalibration of sensors is not a significant contributor to CDF (partly because of the low assessed probability), it is an important maintenance action requiring special care.

A maintenance procedure must be established requiring that whenever a sensor is found to be out-of tolerance, before the sensor is recalibrated, the calibration instrument is first checked or an alternate instrument is used to confirm the condition.

19D.7.2 HEPs in the ABWR Containment Event Trees

The human errors used in the ABWR PRA containment event trees (CETs) are listed in Table 19D.7-2. Most of these HEPs are conservative values assigned by engineering judgement with guidance from various HRA reports and analyses. The use of these HEPs in the CETs dictates their relative importance to the PRA, and the importance of the operator action during the hypothesized severe accident.

The most important operator action is OP - the depressurization of the reactor. This action provides the opportunity to recover low-pressure injection systems and prevent or arrest core melt in some sequences. For sequences where core melt cannot be prevented, depressurization provides more benign melt conditions at low reactor pressure. Because of the importance of this action, realistic values (as opposed to conservative estimates) were used in the PRA. The values used (0.006 when 15 minutes were available, and 0.002 when 30 minutes were available) were taken from GESSAR (p. 15.D.3-422) and from an operator time-reliability curve.

The manual depressurization action requires the following:

(1) The operator must have a clear unambiguous indication of the conditions requiring the action.

(2) The operator must have the capability of depressurizing the reactor from the main control room or from the remote shutdown panel if the MCR is uninhabitable.

(3) The operator must have clear written operating procedures regarding depressurization under these conditions.

(4) The operator must have thorough simulator training in the conditions requiring manual depressurization.

ARV - operator initiation of firewater injection into the depressurized RPV following failure of all high- and low-pressure systems - is less important (than OP), partly because of the low frequency of these events. A judgement value of 0.1 was used for two cases where the operator would have 1 hour, and 0.01 for the SBO case where RCIC would be available for the first 8 hours following LOSP. Both values are very conservative.

For all remaining operator actions (HTF, ARC, and RCH), various conditions would exist, as shown in the table; but in all cases, the accident sequence would be well into the final stages following core melt. The operator would have at least 5 hours available, would be very alert and aware of the situation, and presumably would have adequate assistance. None of these operator actions have a significant effect on calculated risk.

19D.7.3 HEPs in the ABWR PRA Level 2 Analysis

In addition to the human actions identified in the Level 1 analysis and in the containment event trees, several additional human actions have been identified and discussed in parts of the Level 2 analysis. All of these operator actions occur late in the accident sequences, mostly well after the beginning of core melt. At those times, the operator is well aware of the situation and has a wealth of assistance in tracking conditions and making decisions regarding needed actions.

An important operator action in the Level 2 analysis is the use of firewater sprays to prevent the upper drywell temperature from exceeding 533^{O}K. This action would be needed within approximately 5 hours of the start of a high-pressure core melt. In the analysis, the HEP used for the probability of operator failure was 1E-3—the same value that was used in the Level 1 analysis for failure to use drywell sprays for RPV injection. The requirements for this action are the following:

(1) The operator must have indications in the control room of upper drywell temperature and pressure.

(2) Emergency operating procedures must provide instructions to the operator to open RHR(C) manual valves E11-F101, -102, and -103.

(3) The operator must have access to those valves under the accident conditions.

In the suppression pool bypass analysis (Subsection 19E.2.3.3), there is an operator action to initiate the wetwell spray in event of failure of the wetwell-drywell vacuum breaker, ACS crosstie, or air-operated inerting line supply valves. With a sufficient amount of time available for this action, a judgement value of 0.01 was used for this HEP. This is an important operator action that requires written emergency procedures and operator training.

Another action in the suppression pool bypass analysis is action by the operator to close ECCS valves that opened as a result of the accident conditions, but should subsequently be closed. A judgement value of 0.5 was used in the analysis for this HEP, with 30 minutes available for the action. Emergency procedures are needed to provide instructions to the operator to close these valves in the event that the associated systems are inoperative.

In the Level 2 analysis, there are several operator actions that are treated deterministically, i.e., the probability and consequences of the operator not performing these actions are not evaluated in the analysis. These actions all occur late in the accident, and there is no reason to consider that they will not be performed. Generally, they are not important actions, and in most cases the consequences of not performing the actions do not have a large effect on risk. These actions are listed in Table 19D.7-3.

19D.7.4 HEPs in the ABWR Seismic Margins Analysis

Human actions modeled in the ABWR seismic margins analysis are listed in Table 19D.7-4. These human actions are modeled in the seismic event trees. There are no human actions in the seismic fault trees. At the time of, and during the duration of an earthquake, equipment that is needed may fail randomly, and some operator actions may be needed. With the exception of FA—initiation of firewater injection, these random failures, and the associated operator actions are included in the seismic analysis identically as modeled in the internal events PRA. For FA, conservative screening values were used in the internal events CETs. Because of the increased importance of firewater in the seismic margins analysis, more realistic values were used for FA. For identification and discussion of other random events, refer to the internal events PRA and the associated HRAs.

In making estimates for HEPs for the seismic analysis, high stress levels were assumed, but no additional factors were applied for the seismic event. One reason for this is that although the earthquake might have been severe, there was no structure (building) failure in the sequences where operator action was credited. Thus, the operator might have been shaken-up, but he would not have been injured or incapacitated. Furthermore, the maximum acceleration expected would not exceed about 1g, which is within the capability of the operator to withstand.

In none of the accident sequences in the analysis was the operator required to perform an action during the earthquake; but only well after the occurrence of the event. It is reasonable to believe that because of the earthquake the operator would be alerted to the possibility of abnormal conditions.

The seismic margins analysis assumes that offsite power will be lost in the event of an earthquake of the magnitude of interest, and the seismic event trees (Section 19I.3) are so constructed.

19D.7.4.1 FA - Firewater Injection

In the first seismic event tree (Figure 19I.3-1), diesel generators fail, so there is no emergency power (station blackout). In this situation, the only means of water injection into the RPV are RCIC and firewater. RCIC is initiated automatically, with manual initiation as a backup action. Reactor depressurization is also automatic, with manual backup. The manual backup actions for RCIC initiation and reactor depressurization were not modeled in the seismic analysis (except for coincidental random events).

In the event sequences of Figure 19I.3-1, RCIC may start and run for 8 hours, or may fail to do so. In the event that RCIC starts and runs for 8 hours, the operator is very much aware that RCIC will trip when the DC power supply fails. It is important that he be prepared to initiate firewater injection into the RPV after RCIC trips and ADS occurs. (He may also be instructed to manually depressurize the reactor prior to automatic depressurization; but this operator action was not modeled.)

For the case where RCIC starts and operates successfully for 8 hours, a judgement value of 0.001 was used for the probability that the operator would fail to accomplish firewater injection. Considering the available time and conditions, this is a somewhat conservative estimate, although more realistic than the value used in the internal events PRA (0.01).

In the event that RCIC fails to start (or run for 8 hours), the operator must take more immediate action to initiate firewater injection. It is assumed that he may have only 30 minutes to accomplish this action. (If RCIC starts, then fails later, he will have additional time.) Allowing 5-10 minutes to perform the action, and using the time-reliability correlation curves from NUREG/CR-4772 (Reference 4, Figure 7-1), a screening value of 0.01 could be used as an estimate of the probability of failure. A more conservative value of 0.1 was used in the analysis. (Use of a less conservative value for this HEP would not change the conclusions of the analysis.)

Operator action to manually initiate firewater injection is an important action in the event of a large earthquake. The following provisions are needed for this action:

(1) The operator must have a clear unambiguous indication of the conditions of electric power, RCIC, and reactor water level.

(2) The operator must have the capability of establishing a firewater injection path into the RPV in a straightforward manner with all necessary tools and equipment readily available.

(3) The operator must have clear written operating procedures regarding the action to be taken.

(4) The operator must have thorough training in the conditions requiring the action, and training in performing the action.

19D.7.4.2 Alternate Boron Injection

In the event tree of Figure 19I.3-1, for the lower branch where scram fails, SLC will not operate due to the absence of AC power. In the seismic event tree and the seismic analysis, no credit was given to the operator for alternate boron injection or controlling the RPV water level to reduce reactivity. This was a simplification, which provided the indication that such operator actions would not be needed to survive the earthquake. Nevertheless, provisions should be made for, and the operator should be instructed and trained to perform these actions.

19D.7.4.3 HX - Heat Exchanger Isolation

In the event tree of Figure 19I.3-1, failure of the RHR heat exchanger (HX) is treated as a rupture resulting in a flooding event; i.e., no credit is given to operator actions to isolate the ruptured heat exchanger. This treatment actually has little effect on the analysis.

19D.7.4.4 V2 - Condensate Injection

In the second seismic event tree (Figure 19I.3-2), the diesel generators operate and there is emergency power. In this event tree, there are operator actions to provide backup manual initiation for RCIC, HPCS, LPFL, and ADS; but these operator actions are not modeled in the seismic analysis. There is only one operator action modeled in the analysis - operator action to recover condensate injection.

Condensate injection is modeled in the internal events PRA, and is discussed in the HRA sensitivity analysis as event COND. In the seismic analysis, the feedwater and condensate pumps will fail or be tripped due to loss of normal AC power. In case of failure of high and low pressure ECCS, the operator should perform a bus transfer for condensate to the plant investment protection power bus, and then restart a condensate pump. With 30 minutes to perform the action, the same HEP value is used that was used in the internal events analysis - 0.1. This is a very low probability sequence, and the V2 HEP has very little effect on the results. Although this cannot rightly be considered as an important human action, the operator should have the means, procedures and training to perform the action, since provisions for the bus transfer have been made in the design.

19D.7.4.5 LPL - Operator Fails to Control Water Level in an ATWS

The third seismic event tree (Figure 19I.3-3) represents a loss of offsite power event followed by failure to scram with control rods. In this tree, the only operator action of interest is the action to control reactor water level.

In this event, the standby liquid control system has operated and injected boron successfully. The operator now should control water level with whichever injection system or systems initiate. This is the same event that was modeled in the internal events analysis (same acronym - LPL). The value assigned to this HEP in the internal events analysis (0.01) was also used in the seismic analysis. This value is a judgement value taken from GESSAR, Table D.1.1-1. With very low Fussel Vesely Importance and Risk Achievement Worth values, this operator action did not show-up in the HRA sensitivity analysis. Given an earthquake, the probability of the combination of loss of offsite power and failure to insert control rods may be somewhat higher than for internal events, but still remains as a low-frequency sequence. It is expected that the operator will have training in ATWS events, and it is important that he have the means of controlling water level; but there is nothing special about this action following an earthquake that requires any additional or different provisions.

19D.7.5 HEPs in the ABWR Fire, Flood, and Shutdown PRA Analyses

19D.7.5.1 Fire PRA

The ABWR fire PRA analysis (Section 19M) showed that it is important that the operators have the capability of initiating ECCS systems from the remote shutdown panel in the event of a fire in the control room. The operators must also have the demonstrated ability to initiate and control RCIC locally. For these two operator actions, the fire analysis uses a bounding value of 0.003, taken from Table G-1 (Reference 2, Appendix G). These are important operator actions that require procedures and training in the use of the remote shutdown panel and local operation of RCIC.

19D.7.5.2 Flood PRA

The human actions specified in the ABWR Flood PRA analysis are listed and described in the Flood Analysis (Section 19R.6). There are three important operator actions:

(1) Isolation of flood sources following detection by sump pump operation and alarms or floor water level detectors,

(2) Closure of watertight doors to prevent damage to equipment in more than one safety division,

(3) Opening of doors or hatches to divert water from safety related equipment (not credited in the PRA).

The PRA used 0.1, 0.05, and 0.01 for the HEPs for actions 1 and 2, above depending on the available time - <30 minutes, 30 minutes to one hour, and >1 hour, respectively. These are conservative values based on engineering judgement.

These operator actions require procedures and training to mitigate the consequences of potential internal floods.

19D.7.5.3 Shutdown Risk Evaluation

The human actions specified in the ABWR shutdown risk evaluation are listed and described in the shutdown analysis (Section 19Q.12). There are five important actions treated probabilistically in the analysis:

(1) Recognition of failure of an operating RHR(SDC) system during shutdown operations.

 (a) Recognition in time to prevent boiling (in Mode 5). HEP = 1E-4 if RHR failure occurs during first three days after shutdown. HEP = 1E-5 if RHR failure occurs later than three days after shutdown.

 (b) Recognition in time to prevent core damage. HEP = 1E-4 if RHR fails when reactor cavity is not flooded. HEP is negligible if cavity is flooded.

(2) Successful startup of a standby RHR(SDC) following loss of the operating division after the operator has successfully diagnosed the problem. The HEP value used was 2E-2 for failing to start up the first of the two standby RHR(SDC) divisions, and 0.1 for failure to start the second division.

(3) Successful use of one of the alternate means of decay heat removal (CUW, FPC, main condenser). The HEP value used was 2E-2 for each system.

(4) Successful use of an alternate means of inventory makeup using one of the non-safety grade systems (CRD, feedwater, or condensate). The HEP value used was 2E-2 for each system.

(5) Utilization of boiling for decay heat removal in Mode 5 (with the RPV head removed), including makeup of inventory lost to boiloff. The HEP value used was 2E-2 for each system.

The above HEP values were calculated conservatively using the procedure for nominal HRA in Table 8-1 (Reference 4).

There are several additional operator actions during shutdown operations that are treated deterministically in the shutdown risk evaluation (i.e., it is assumed that these actions will be taken if needed):

(1) Implementation of fire/flood watches during periods of degraded safety equipment integrity.

(2) Fire fighting during shutdown operations (possibly with part of the fire protection in maintenance).

(3) Use of the remote shutdown panel during shutdown operations.

The above listed shutdown operator actions are important and require procedures, operator training, and necessary instrumentation and alarms. A list of needed instrumentation is given in Subsection 19Q.12.3 of the shutdown risk evaluation.

19D.7.6 Summary of Important Operator Actions from the ABWR PRAs

The following are the important operator actions identified in the ABWR Level 1, Level 2, fire, flood, seismic, and shutdown analyses. They are divided into four categories.

(1) Critical Tasks: These items should be given consideration as being "Critical Tasks" as defined by the human factors evaluation, Design Acceptance Criteria, as noted in Section 18E.2.

(a) Backup manual initiation of HPCF (see 19D.7.1 and 19D.7.7).

(b) Recovery of feedwater following scram with and without isolation (see 19D.7.1 and 19D.7.7).

(c) Use of condensate injection following scram with reactor depressurized (see 19D.7.1, 19D.7.4, and 19D.7.7).

(d) Control of reactor water level in an ATWS (see 19D.7.1, 19D.7.4, and 19D.7.7).

(e) Emergency depressurization of the reactor (see 19D.7.2).

(f) Alignment and initiation of firewater for RPV injection with ECCS failure in an earthquake (see 19D.7.4).

(g) Alignment and initiation of firewater for drywell spray (see 19D.7.3)

(h) Initiation of wetwell spray (see 19D.7.3).

(i) Isolation of water sources in an internal flood (see 19D.7.5).

(j) Initiation of standby RHR in event of failure of operating RHR during shutdown operations (see 19D.7.5).

(2) COL Maintenance Item:

(a) Reopening of HPCF injection valves following maintenance (see 19D.7.1 and 19D.7.7).

(b) Calibration of sensors (see 19D.7.1).

(3) COL System Operating Procedure: Closure of ECCS injection valves following ECCS failure (see 19D.7.3).

(4) COL Procedures and Planning:

(a) Closure of watertight doors in an internal flood (see 19D.7.5).

(b) Opening of doors/hatches to divert water in an internal flood (see 19D.7.5).

(c) Use of non-safety grade equipment for decay heat removal and inventory makeup during shutdown operations (see 19D.7.5).

(d) Use of boiloff for decay heat removal with RPV head removed during shutdown operations (see 19D.7.5).

(e) Fire fighting coordination and establishment of fire/flood watches during shutdown operations (see 19D.7.5).

(f) Use of remote shutdown panel when main control room is uninhabitable (see 19D.7.5).

(g) Local initiation and control of RCIC when control room is uninhabitable (see 19D.7.5).

(h) Outage planning to minimize risk during shutdown operations (see 19D.7.5).

Groups 2, 3 and 4 are included as action items for the COL applicant in 19.9.

19D.7.7 Sensitivity Analysis of HEPs in the ABWR Level 1 PRA

A sensitivity analysis has been conducted of the HEPs in the Level 1 ABWR PRA. The first step in the sensitivity analysis process was to identify and list in rank of importance all human errors included in the Level 1 PRA. That listing is shown in Tables 19D.7-5 and 19D.7-6. Two additional recovery items involving operator action are recovery of offsite power and recovery of diesel generators. Those two items are not included in this sensitivity analysis since the failure probabilities for those items were determined from actual data, not from human reliability analysis, and include factors other than human actions.

The 12 HEPs in Table 19D.7-5 are the only HEPs that show-up in the top 300 cutsets of the analysis, representing 98% of the total core damage frequency. The fourth column in the table gives the HEP value used in the PRA. The fifth column is the error factor (the ratio of the 95th to 50th percentile of the uncertainty distribution) on the HEP, as provided by the PRA uncertainty analysis. In cases where there was no clear basis for determining an error factor, a value of 15 was used.

The sixth column is the Fussell-Vesely Importance (F.V.), which is a measure of the percentage contribution of each item to the total CDF. The items in the table are ranked according to decreasing F.V. The last column is the Risk Achievement Worth (R.A.W.), which is another importance measure, and is the factor by which the total CDF would be multiplied if that specific item had a failure probability of 1.0.

All items below #5 (HBMAER1) contribute much less than 1%, individually, to total CDF. Most of the items in Table 19D.7-5, plus CALN002, HFE008CF, and HPR007CF from Table 19D.7-6 have a relatively high R.A.W., often because these items have relatively low assigned failure probabilities. All items on the list except the 15 items identified above have very low F.V. and R.A.W. measures, and are eliminated from further consideration.

The first screening analysis was made by doubling all the failure probabilities (simultaneously) of all of the 15 items identified above, and then reevaluating core damage frequency. The resulting CDF was 59% higher than the base CDF. This result provided an indication that the CDF was fairly sensitive to one or more of the 15 items.

The next sensitivity run was made by increasing the failure probability of each of the 15 items, individually, by a factor of 4. The factor of 4 includes the 95th percentile of the uncertainty distribution. The results are shown in Table 19D.7-7. The top 5 items each resulted in increases in CDF greater than 1%. The 6th and lower-ranked items each resulted in increases of less than 1/2%, which is considered to be insignificant.

An additional analysis was made, in which the failure probabilities of the 10 items below #5 were increased (simultaneously) by a factor of 4. The result was a 2.33% increase in total core damage frequency, providing a further indication of the relative insensitivity of CDF to variability of the failure probability of these 10 items.

Because of the general uncertainty in theoretical human error analysis, and the involved and labor-intensive nature of the various HRA procedures, the ABWR PRA uses screening methods in several places. Even though some of the HEPs used in the ABWR PRA are screening values and are conservative, no sensitivity runs were made with failure probabilities decreased from the values used in the PRA. The use of more realistic HEPs would reduce total CDF by a small amount, but would require additional more-detailed HRA. Use of more realistic HEPs might also change the relative importance and sensitivity of the individual HEPs, but it is doubtful that any basic conclusions or recommendations would change.

The top 5 items are identified as the most sensitive HEPs in the PRA. The top 4 items are operator actions that are needed after the accident sequence is initiated (Type C actions). Each of the operator actions represented by HEPs #1 - #4 requires the following:

(1) The operator must have a clear unambiguous indication of the conditions requiring the action.

(2) The operator must have the capability of performing the necessary action from the main control room in a simple straightforward manner.

(3) The operator must have clear written operating procedures regarding the action to be taken.

(4) The operator must have thorough simulator training in the conditions requiring the action.

HEP #5 represents a Type A action (occurs prior to initiation of the accident sequence). This error may be an error of omission or an error of commission. To prevent this error from occurring, administrative controls must be in place to require independent verification of the valve position following maintenance, positive control of the key to the valve lock during periods when entry to the containment is possible, and control room verification of the valve position prior to startup.

All five of the important operator actions relate to makeup of reactor inventory - four with the reactor at high pressure, and one (COND) with the reactor at low(er) pressure. One of the items (HOOBOPHL) is an operator action to backup automatic signals that failed to initiate HPCF. Three of the items (Q, Q2, and COND) are actions for recovery of (non-safety) systems that were in normal operation and were lost (tripped) at the time of the event. In cases where failure of the system was the cause of (initiated) the event, no credit was given to the operator for recovery of the system. In some instances, this is a very conservative treatment. The remaining item (HBMAER1) is a Type A operator action resulting in mispositioning of a valve on the HPCF B discharge line.

Discussions of the derivation of the failure probabilities for the five most sensitive actions follow.

19D.7.7.1 HOOBOPHL - Failure to Manually Initiate HPCF

HPCF is automatically initiated if reactor water level decreases to Level 2. The PRA gives credit to the operator for manual backup of the automatic signal. The value used for the probability of failing to provide manual backup initiation is 0.1. (This value for manual backup actions is used throughout the PRA wherever the action required is simple and performed from the control room.)

The action required to manually start the HPCS pumps is simple and is performed directly from the control room with minimal time required for performance of the action. The operator has direct (hardwire) control for initiation of HPCF B. Manual initiation of HPCF C is transmitted through multiplex equipment. Operator action for initiation of HPCF B and C is modeled as a single action. The time available to the operator for cognition and performance of the backup action is at least 30 minutes, except for the ATWS and large LOCA events, where the events proceed more rapidly. For those events, the initiating frequency is low, and the backup manual initiation of HPCF has little effect on CDF.

The estimate of 10% for operator failure probability is made based on a long trail back through GESSAR, the Limerick PRA (Reference 6), Swain and Guttman (Table G-1, p. G-4), and even WASH-1400. In Figures 7-1 and 8-1 (Reference 4) curves for suggested screening values and nominal values for diagnosis HEPs are given. In the case of the ABWR backup manual initiation of HPCF, the operator has at least 30 minutes available, and the actual operation of starting the pumps (after recognition of the need) is simple and requires a minimal amount of time. With at least 30 minutes available for diagnosis, the curves of Figures 7-1 and 8-1 (Reference 4) suggest a failure probability of 0.01. The ABWR PRA uses a conservative screening value of 0.1.

19D.7.7.2 Q - Failure to Inject with Feedwater During a Non-Isolation Event

The ABWR feedwater controller is designed to withstand turbine trips (and other transients) without tripping. Nevertheless, the PRA analysis assumed (conservatively) that 50% of the non-isolation initiating events would result in tripping of the feedwater pumps. It was further postulated that in 10% of these cases, the operator would fail to restart feedwater pumps. (This also is probably conservative, since the FW pumps were in operation just prior to the incident, and only one pump is needed in the accident sequences.)

As in the case of backup initiation of HPCF, the estimate of operator failure probability is made based on GESSAR, the Limerick PRA, and Swain and Guttman. The same curves in Figures 7-1 and 8-1 (Reference 4) for suggested screening values and nominal values for diagnosis HEPs were used. In all cases of FW recovery in the ABWR PRA, the operator has at least 30 minutes available, and the actual operation of restarting a FW pump (after recognition of the need) requires a minimal amount of time. With at least 30 minutes available for diagnosis, the curves of Figures 7-1 and 8-1 (Reference 4) suggest a failure probability of 0.01. The value of 0.1 used in the ABWR PRA is conservative—even more conservative than the value used for initiation of HPCF—because of the higher frequency of, and greater operator familiarity with, startup of feedwater pumps.

Initiation and control of feedwater and condensate are basic routine actions which are performed by the operator repeatedly, from the control room, and under a wide spectrum of varying circumstances and conditions. There are few, if any, actions more familiar to the operator. However, it is essential that the operator have clear indications of the plant conditions (particularly reactor water level and status of ECCS pumps), that he be thoroughly trained under conditions simulating the spectrum of accident sequences of concern, and that the plant EOPs provide clear instructions.

19D.7.7.3 Q2 - Failure to Inject with Feedwater During an Isolation Event

The analysis in the ABWR PRA assumes that 40% of isolation initiating events will be due to loss of feedwater. This is based on operating data from BWRs in the U.S. For events that are initiated by loss of feedwater, the PRA gives no credit for recovery. This is conservative treatment, since many loss-of-feedwater events (in operating plants) are due to spurious trips which are routinely reset.

The ABWR PRA assumes that 60% of the isolation initiating events will be due to closure of the MSIVs. The ABWR feedwater controller is designed to ride-through a MSIV closure event without tripping. Even so, as in the case of non-isolation events, the ABWR PRA analysis assumes that 50% of the MSIV closure events will result in trip of the feedwater pumps. Also, as in the case of the non-isolation events, the probability of failure of the operator to recover feedwater is assigned a value of 0.1 in the PRA. Based on the above factors, the value for Q2 is:

$$0.4 + (0.6 \times 0.5 \times 0.1) = 0.43$$

Since the ABWR feedwater pumps are motor-driven, and the condenser hotwell inventory is automatically replenished from the CST; it is not necessary for the operator to reopen the MSIVs in order to use feedwater for RPV injection. (However, the operator may need to reopen MSIVs to regain the main condenser for decay heat removal.)

19D.7.7.4 COND - Failure to Inject with Condensate (to a Depressurized Reactor)

In the PRA analysis, for transient events with successful scram, and for the small LOCA event, credit is given for operator recovery of condensate following failure of high-pressure injection and depressurization of the reactor on low water level. Actually, in most cases no operator action is required, since condensate pumps will continue to operate and pump through minimum bypass lines so long as power and suction water are available. (If MSIVs close, operator action may be needed to reopen MSIVs to provide recovery of the main condenser for decay heat removal.) The value of 0.1 used for the probability of failure to recover condensate is a very conservative screening value.

19D.7.7.5 HBMAER1 - Valve E22-F005B Closed (NOFC)

Valve E22-F005B is a normally-open valve on the discharge of the B-loop HPCF pump. This valve is a manual locked-open valve located inside of the drywell, and the valve position is indicated in the main control room. The PRA assigns a probability of 0.01 to the possibility of the valve being closed due to human error. Since the valve is inside the containment and is a manual locked-open valve, the human error must be Type A (pre-accident). NUREG/CR-4772 (ASEP) suggests use of a basic HEP of 0.03 for pre-accident errors, which it considers conservative. ASEP and Table 20-22 (Reference 2) suggest application of a factor of 0.1 for recovery. Because of the valve lock and the control room indication of the valve position, application of the recovery factor is reasonable. Plant administrative procedures should also require that the valve position be independently verified following maintenance. The value of 0.01 used in the PRA is conservative.

HCMAER1, which is the operator error for mispositioning the HPCF C discharge valve, also has a HEP value of 0.01 in the PRA; however, it is much less sensitive than HBMAER1. This is because there is no hardwire backup for manual initiation of HPCF C.

19D.7.8 Sensitivity Study of HRA in Comparison to a Brookhaven National Laboratory (BNL) Study

19D.7.8.1 Introduction

This analysis is in response to a request from the NRC for GE to perform a sensitivity analysis on the ABWR PRA comparable to an analysis by BNL (Reference 5). It is a BNL evaluation of the impact of human errors on the internal event risk parameters in the LaSalle PRA.

It should be stated that due to many differences between the ABWR plant and the LaSalle plant, and differences in the structure and methods of the two PRAs, direct comparisons may be misleading. Nevertheless, an attempt has been made to repeat the BNL methods as closely as possible in a similar analysis of the ABWR PRA HEPs.

19D.7.8.2 BNL Analysis

The BNL analysis was performed by multiplying a group of estimated HEPs corresponding approximately to median values of the LaSalle PRA HEPs simultaneously by a factor and computing a new core damage frequency; then multiplying all of these HEPs by a larger factor, and continuing the process to a limit. The same HEPs were also divided by the same factors and the resulting CDF computed. The HEP factors used in the BNL analysis were limited by the 5th and 95th percentiles of the individual input HEP distributions and, of course, the HEP factors were also limited to keep the resulting values within the [0,1] probability interval. The ultimate limit factor in the BNL analysis corresponds to the maximum error factor of 29 of the input HEPs. The result is a curve of CDF vs HEP factor.

The BNL results are shown on Figure 19D.7-2 (Reference 5, Figure 5.7), where BNL showed Type A (pre-accident) and Type C (during accident) HEPs. There were only 4 Type A errors in the LaSalle PRA. The Type C errors were almost completely related to restoration of lost offsite power and recovery of failed diesel-generators.

A statement is made in the BNL report that "...since all human errors were varied simultaneously, the displayed extreme values of core melt frequency should be regarded as hypothetical, resulting from extrapolation of PRA models beyond their originally intended purposes." That caution is equally applicable to this analysis.

19D.7.8.3 ABWR Analysis

In this ABWR analysis, HEP factors were applied directly to the ABWR PRA mean value HEPs. The HEP factor limits corresponded to 5th and 95th percentiles of the HEP distributions, and were also limited to keep the values of the new HEPs within the [0,1] probability interval. The ultimate limit factor in this ABWR analysis corresponds to the maximum ABWR PRA error factor of 15.

One significant difference between the LaSalle and ABWR HEPs is that the LaSalle Type C errors were dominated by recovery of offsite power and diesel-generators; whereas the ABWR PRA did not identify HEPs for these two recovery actions (since the recovery frequencies involved more than operator action and were based on experience data). Another difference is that the LaSalle PRA identified only 4 Type C errors, while the ABWR PRA has 23.

19D.7.8.4 Discussion of ABWR Results

The results of this sensitivity analysis are shown on Figure 19D.7-3, which is a plot similar to the BNL figure. As in the BNL analysis, the HEP factors are shown as ranging from 1/29 to 29. This may be somewhat misleading (in both analyses), since all HEPs are not multiplied by the HEP factor indicated on the x-axis because of upper and lower limits on many of the HEP factors. The shapes of the curves of the two analyses are very similar in that they are both s-curves truncated at both ends, and in both cases the increase in CDF above the base is much greater than the decrease below the base. However, in the LaSalle analysis, the CDF continues to increase out to a HEP factor of 29, whereas the ABWR curve does not increase beyond a HEP factor of about 5, due to lower HEP factor limits and use of means vs medians.

The CDF increase in the ABWR curve is less than in the BNL curve (about a factor of 3 compared to a factor of 10), even though the base CDF for SBWR is about two decades lower than the LaSalle CDF. This is due (at least partially) to the lower HEP factor limits. The difference may also be due in part to the automation of key safety functions in the ABWR design.

19D.7.8.5 Conclusions

The conclusions from this sensitivity analysis are the following:

(1) The sensitivity of CDF to HEP uncertainty is limited by the probability interval [0,1], and the

input HEP error factor (or other measure of dispersion).

(2) Within the limits of the HEP error factors, the maximum effect of HEP uncertainty in the ABWR PRA is an increase in CDF of about a factor of 3.

(3) Decreases in HEP values below the values used in the PRA have very little effect on CDF.

(4) Type A HEPs have very little effect on the results of the PRA.

(5) In spite of significant differences between the ABWR and LaSalle plants and PRA models, the results of the two sensitivity analyses are similar.

(6) The results of this analysis are in agreement with and supplement the previous GE ABWR human error sensitivity analysis.

19D.7.9 Supplemental Sensitivity Analysis

In response to a specific NRC request, all HEPs in the Level 1 internal events analysis were simultaneously increased by factors of +/-5, 10, 15, 20, 25, and 29; and, the resulting CDFs were calculated and plotted. The results are shown in Figure 19D.7-4.

When all ABWR Level 1 HEPs are multiplied simultaneously by 29, the resultant CDF is increased by a factor of approximately 117. This result is not surprising since many low-probability cutsets and accident sequences contain multiple HEPs which are each arbitrarily multiplied by 29.

The results of this analysis should not be compared to the BNL analysis discussed in the previous subsection (19D.7.8) since the basic conditions of the analyses are different in many respects.

19D.7.10 Cutsets and Accident Sequences Containing Human Interactions

The HEPs in the top 300 cutsets of the ABWR internal events analysis were discussed previously in Subsection 19D.7.7 and are listed in Table 19D.7-5. In this subsection, the accident sequences and cutsets containing those HEPs are identified.

19D.7.10.1 Accident Sequences Containing Human Interactions

The accident sequences containing human interactions are listed in Tabled 19D.7-8.

19D.7.10.2 Cutsets Containing Human Interactions

Of the top 300 cutsets in the analysis, 110 contained no human interactions. The total frequency of these cutsets is 1.12E-7, representing 71.6% of the total CDF. There were a total number of 190 cutsets containing hun.an interactions, with a total frequency of 4.28E-8.

Of the 190 cutsets containing human interactions, 128 contained a single HEP. The total frequency of these cutsets was 1.88E-8, representing 12.0% of the total CDF. These cutsets are listed in Table 19D.7-9.

Of the remaining 62 cutsets, 59 contained two HEPs. The total frequency of these cutsets was 2.40E-8, representing 15.4% of the total CDF. These cutsets are listed in Table 19D.7-10.

There were no cutsets containing three HEPs. There was one cutset containing four HEPs. The cutset containing four HEPs has a frequency of 1.20E-11 (<0.1%), and is in sequence TIO-04. The HEPs in this cutset are O, COND, HOOBOPHL, and RHRCFER.

There were two cutsets containing five HEPs. The total frequency of the two cutsets is 3.13E-11 (<0.1%). One of these two cutsets is in sequence TIS-05, with a frequency of 1.86E-11 and containing HEPs Q2, COND, HOOBOPHL, RHRCFER, and ROOIOPHL. The other of these two cutsets in sequence TM-05 with a frequency of 1.27E-11 and containing HEPs Q, COND, HOOBOPHL, RHRCFER, and ROOIOPHL.

19D.7.11 References

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3. Reactor Safety Study, *An Assessment of Accident Risks in U.S. Commercial Nuclear Power Plants*, WASH-1400, NUREG-75/014, United States Atomic Energy Commission, October 1975.

4. A.D. Swain, *Accident Sequence Evaluation Program - Human Reliability Analysis Procedure*, Sandia National Laboratories, NUREG/CR-4772, USNRC, February 1987.

5. *Risk Sensitivity to Human Error in the LaSalle PRA*, Brookhaven National Laboratory, NUREG/CR-5527, March 1990.

6. *Probabilistic Risk Assessment, Limerick Generating Station*, Philadelphia Electric Company, September 1982.

7. J.W. Wreathall, *Operator Action Trees, An Approach to Quantifying Operator Error Probability During Accident Sequences*, NUS Report 4159, NUS Corporation, July 1982.

Table 19D.7-1
HUMAN ACTIONS MODELED IN THE ABWR LEVEL 1 PRA

Name	Description	Failure Prob.	Fault Tree	Basis
HOOBOPHL	Failure to manually initiate HPCF	0.10	HPCF	*
Q	Failure to inject with feedwater	0.05	EVENT TREES	*
Q2	Failure to inject with feedwater (TIS)	0.43	EVENT TREES	*
COND	Failure to inject with condensate	0.10	EVENT TREES	*
HBMAER1	Valve E22-F005B closed (NOFC)	0.01	HPCF SLCS	*
LPL	Oper. fails to control W.L. in an ATWS	0.01	EVENT TREE	GESSAR, Table D.1.1-1
ROERROR4	Oper. fails to attempt manual vlv. op. [Backup for RCIC disch. vlv. (F013)]	0.10	RCIC	GESSAR
CTGMANSW	CTG manual disconnect switch [left] open (Following maintenance on gas turbine gen.)	3E-3	RBCW	NUREG/CR-1278
RPR005CF	Sensor miscalibration	5E-5	RCIC	NEDE-22056, p. 85
RFL007CF	Sensor miscalibration	5E-5	RCIC	NEDE-22056, p. 85
HFELEBHX	Water level 8 sensors miscal. (4 div.)	2E-5	RCIC HPCF	NEDE-22056, p. 85
RHRSPER	Oper. fails to manually initiate [SP cooling initiation (within 20 hours)]	6E-5	RHR	GESSAR, p. 15D.3-465/1
CALN002A	Miscal. of flow xmtrs FT008A, B & C	5E-5	RHR	NEDE-22056, p. 85
RHRCFER	Oper. fails to manually initiate (Backup for RHR core flood A/B/C)	0.10	RHR	NEDC-30936, Appendix H

*See Sensitivity Analysis (Subsection 19D.7.7)

Table 19D.7-1
HUMAN ACTIONS MODELED IN THE ABWR LEVEL 1 PRA (CONTINUED)

Name	Description	Failure Prob.	Fault Tree	Basis
HCMAER1	Valve E22-F005C mispositioned (NOFC)	0.01	HPCF	GESSAR
ROERROR3	Oper. fails to manually open valve	0.01	RCIC	GESSAR
ADSMAN	Failure of ADS manual init. (backup)	2E-3	ADS	GESSAR
ROOIOPHL	Oper. fails to initiate within 30 min. (Backup for RCIC initiation)	0.10	RCIC	GESSAR
NHR	Failure to restore normal heat removal	0.01	EVENT TREES	GESSAR
RWCU	Failure to actuate RWCU	0.10	EVENT TREES	NEDC-30936, Appendix H
RSTTCOPF	Operator fails to reset trip circuit (RCIC internal trips)	0.01	RCIC	NUREG/CR-1278, p. 20-17
SLC000SA	Boron concentration sampling failure	2E-5	SLCS	NEDC-30936, Appendix H
SLC001HE	Operator fails to initiate SLC	0.10	SLCS	NEDC-30936, Appendix H
SLC002HE	Operator fails to initiate SLC tank heater	2E-3	SLCS	NEDC-30936, Appendix H
WOPERR	Oper. fails to perform indicated action (Backup to RBCW initiation)	0.01	RBCW	NUREG/CR-1278, p. 20-17
HUEROR5	Oper. fails to transfer from CST to SP	0.01	HPCF	GESSAR
VOPERRF	Operator fails to start pump	1E-3	RBCW	NUREG/CR-1278, p. 20-17
ASECSNA	Operator fails to backup N2 initiation	0.10	ADS	NEDC-30936, Appendix H
CMAN	Operator fails to backup ARI initiation	0.10	ARI	NEDC-30936, Appendix H

Table 19D.7-1
HUMAN ACTIONS MODELED IN THE ABWR LEVEL 1 PRA (CONTINUED)

delete

Name	Description	Failure Prob.	Fault Tree	Basis
	<u>Electrical</u>			
EHU69C	Operator fails to transfer power	1E-3	Elec.	GESSAR
EHUB1	Operator fails to bypass	1E-3	Elec.	GESSAR
EHUB2	Operator fails to bypass	1E-3	Elec.	GESSAR
EHUB3	Operator fails to bypass	1E-3	Elec.	GESSAR
EHUB4	Operator fails to bypass	1E-3	Elec.	GESSAR
EHUS1AD	Oper. fails to xfer stdby charger to Div. I	1E-3	Elec.	GESSAR
EHUS1BD	Oper. fails to xfer stdby charger to Div. II	1E-3	Elec.	GESSAR
EHUS1CD	Oper. fails to xfer stdby charger to Div. III	1E-3	Elec.	GESSAR
EHUS1DD	Oper. fails to xfer stdby charger to Div. IV	1E-3	Elec.	GESSAR
	<u>Miscalibrations</u>			
HFE008CF	Miscal. of flow xmtrs	5E-5	HPCF	NEDE-22056, p. 85
HPR007CF	Miscal. of pressure xmtrs	5E-5	HPCF	NEDE-22056, p. 85
AHPT006	Miscal. of pressure xmtrs	2E-5	ADS	NEDE-22056, p. 85
RFE635HX	Miscal. of CST level sensors	2E-5	RCIC	NEDE-22056, p. 85
REOSSMSC	Elec. overspeed sensor miscal	5E-5	RCIC	NEDE-22056, p. 85
RPR309MC	High turbine exh. press. xmtr. miscal	5E-5	RCIC	NEDE-22056, p. 85
RMOSSMSC	Mech. overspeed sensor miscal	5E-5	RCIC	NEDE-22056, p. 85
RPR303MC	Low suction press. xmtr. miscal	5E-5	RCIC	NEDE-22056, p. 85
	<u>Valve Mispositions</u>			
ROERROR5	Valve F009 inadvertently left open	0.01	RCIC	GESSAR
HBMAER2	Test valve E22-F009B inadvert. left open	0.01	HPCF	GESSAR
HCMAER2	Test valve E22-F009C inadvert. left open	0.01	HPCF	GESSAR
C001AMOV	Manual override fails initiation signal	1.8E-4	RHR	GESSAR
C001BMOV	Manual override fails initiation signal	1.8E-4	RHR	GESSAR
C001CMOV	Manual override fails initiation signal	1.8E-4	RHR	GESSAR

Table 19D.7-2
HUMAN ACTIONS MODELED IN THE ABWR PRA CONTAINMENT EVENT TREES

Name	Description	Probability	Event Tree (Fig.)	Basis
OP	Operator fails to depressurize reactor	6E-3	19D.5-6	15 min. (Reference 7)
		6E-3	19D.5-9	15 min. (Reference 7)
		2E-3	19D.5-8	30 min. (Reference 7)
ARV	Operator fails to initiate firewater	0.01	19D.5-8	* >8 hrs.
		0.10	19D.5-11	* 1 hr.
		0.10	19D.5-15	* 1 hr.
HTF	Operator fails to initiate drywell spray	0.01	19D.5-4	* 5 hrs.
		0.01	19D.5-5	* 5 hrs.
		0.01	19D.5-7	* 5 hrs.
		0.01	19D.5-10	* 5 hrs.
		0.01	19D.5-13	* 5 hrs.
		0.01	19D.5-14	* 5 hrs.
ARC	Operator fails to recover RHR or firewater	0.01	19D.5-5	* 5 hrs.
	Operator fails to initiate firewater	0.01	19D.5-6	* 5 hrs.
	Operator fails to realign firewater to RPV	0.01	19D.5-7	* 5 hrs.
	Operator fails to initiate firewater	0.01	19D.5-9	* 5 hrs.
	Operator fails to initiate firewater	0.01	19D.5-10	* 5 hrs.
	Operator fails to initiate firewater	0.01	19D.5-11	* >8 hrs.
	Operator fails to initiate firewater	0.01	19D.5-14	* 5 hrs.
	Operator fails to initiate firewater	0.01	19D.5-15	* >8 hrs.

*Judgement - conservative value assigned.

Table 19D.7-2
HUMAN ACTIONS MODELED IN THE ABWR PRA CONTAINMENT EVENT TREES (CONTINUED)

Name	Description	Probability	Event Tree (Fig.)	Basis
RCH	Operator fails to realign RHR after ARC	0.01	19D.5-5	* >8 hrs.
	Operator fails to realign RHR after no ARC	0.10	19D.5-5	* >8 hrs.
	Op. fails to recover RHR or realign firewater	0.01	19D.5-6	* >8 hrs.
	Op. fails to recover RHR after no firewater	0.10	19D.5-6	* >8 hrs.
	Op. fails to recover RHR or realign firewater	0.01	19D.5-7	* >8 hrs.
	Op. fails to recover RHR after no firewater	0.10	19D.5-7	* >8 hrs.
	Op. fails to recover RHR or realign firewater	0.01	19D.5-8	* >8 hrs.
	Op. fails to recover RHR after no firewater	0.10	19D.5-8	* >8 hrs.
	Op. fails to recover RHR	0.01	19D.5-8	* >8 hrs.
	Op. fails to recover RHR or realign firewater	0.01	19D.5-9	* >8 hrs.
	Op. fails to recover RHR after no firewater	0.10	19D.5-9	* >8 hrs.
	Op. fails to recover RHR or realign firewater	0.01	19D.5-10	* >8 hrs.
	Op. fails to recover RHR after no firewater	0.10	19D.5-10	* >8 hrs.
	Operator fails to initiate firewater	0.10	19D.5-10	* 20 hrs.
	Operator fails to realign RHR after ARC	0.01	19D.5-14	* >8 hrs.
	Operator fails to realign RHR after no ARC	0.10	19D.5-14	* >8 hrs.

*Judgement - conservative value assigned.

Table 19D.7-3
OPERATOR ACTIONS TREATED DETERMINISTICALLY IN THE ABWR LEVEL 2 PRA

Assumed Action	Reason for the Assumption
1. RPV W.L. lowered below the shutdown cooling nozzle and RPV depressurized following a break of an RWCU line with failure of isolation valves to close.	Procedures will instruct the operator to maintain the W.L. between TAF and 5 feet above TAF for these conditions (see 19.9.1).
2. Non-essential DC loads shed from station batteries in event of station blackout.	Batteries should be available for at least 8 hours after station blackout without load shedding. The ABWR PRA assumed 8 hours of battery life.
3. Firewater injection stopped if suppression pool W.L. reaches the same elevation as the bottom of the RPV, unless firewater were the only means of RPV injection and the RPV was still intact.	Suppression pool W.L. would be steadily increasing so that the operator would have significant advanced indication that action would be required. This situation would only occur about 16 hours into the accident and at least 12 hours after initiation of firewater. Procedures are needed to provide operator guidance for this situation.
4. Firewater spray is initiated as necessary to maintain the upper drywell temperature below 533°K.	The upper drywell temperature would be slowly and steadily increasing. Thus, the operator would have ample advanced indication that action would be required. This situation would occur about 16 hours into the event.
5. Firewater spray is initiated in the event of drywell failure.	Action needed within 30 minutes of drywell failure, but at least 20 hours into the accident. The operator should be aware of the condition of the drywell at that time.
6. Water supply to the RCIC is switched back to the CST if a high suppression pool temperature alarm occurs.	Suppression pool temperature would be rising slowly and steadily providing the operator with advanced indication that action would be needed. This situation would only occur at least 4.5 hours into the accident.
7. If, at any time, the shutoff head of the firewater system were exceeded, injection would be accomplished by use of a fire pumper truck.	Vessel pressure would be slowly and steadily increasing, thus providing the operator with advanced indication of needed action. This situation would occur no sooner than 15 hours into the accident.

Table 19D.7-4
HUMAN ACTIONS MODELED IN THE ABWR SEISMIC MARGINS ANALYSIS

Name	Description	Probability	Event Tree (Fig.)	Basis
FA	Oper. fails to initiate firewater injection (RCIC operates for 8 hours)	1E-3	19I.3-1	Judgement
FA	Oper. fails to initiate firewater injection (RCIC fails to operate for 8 hours)	0.10	19I.3-1	NUREG/CR 4772, Fig. 7-1
V2	Oper. fails to inject with condensate	0.10	19I.3-2	HRA Sensitivity Anal.
LPL	Oper. fails to control W.L. in an ATWS	0.01	19I.3-3	GESSAR, Table D.1.1-1
W3 - RHR	[The following material was deleted from the seismic margins analysis since the added containment vent is not subject to seismic failure.]			
"W3	Oper. fails to align RHR	6E-5	19I.3-2	GESSAR, p. 15D.3-465/1

For sequences with successful RPV injection, decay heat must be removed. Without normal AC power, normal heat removal with the main condenser is not possible. The RHR system operates on emergency power, and may survive the earthquake in operable condition. The operator has a long time--20 hours--to align the RHR valves for shutdown or suppression pool cooling. This same action was modeled in the internal events analysis as RHRSPER, with a HEP value of 6E-5 (taken from GESSAR, p. 15D.3-465/1). The same value was used in the seismic analysis.

In the HRA sensitivity analysis, RHRSPER did not show up as a very important event. In the seismic analysis, W3 is more important due to the inability to use the main condenser. However, since there is a long time available to perform this action, and since RHR alignment is an action with which the operator should be well trained and familiar, there should be no need for special provisions related to earthquake."

Table 19D.7-5
HUMAN ACTIONS IN THE TOP 300 CUTSETS (98.0% OF CDF)

Rank	Name	Description	Assigned Probability	E.F.	Importance F.V. (%)	Importance R.A.W.
1.	HOOBOPHL	Failure to manually initiate HPCF (Incl. hardwire backup for EMUX failure - HPCF B)	0.10	5	16.0	2.44
2.	Q	Failure to inject with feedwater	0.05	5	12.5	3.38
3.	Q2	Failure to inject with feedwater(TIS)	0.43	5	10.9	1.14
4.	COND	Failure to inject with condensate	0.10	15	1.90	1.17
5.	HBMAER1	Valve E22-F005B closed (NOFC)	0.01	5	1.73	2.71
6.	ROERROR4	Oper. fails to attempt manual vlv. op. [Backup for RCIC disch. vlv. (F013)]	0.10	5	0.15	1.01
7.	CTGMANSW	CTG manual disconnect switch [left] open (Following maintenance on gas turbine gen.)	3E-3	3	0.08	1.28
8.	RHRCFER	Oper. fails to manually initiate (Backup for RHR core flood A/B/C)	0.10	5	0.06	1.01
9.	RPR005CF	Sensor miscalibration	5E-5	10	0.05	11.8
10.	RFL007CF	Sensor miscalibration	5E-5	10	0.05	11.8
11.	HFELEBHX	Water level 8 sensors miscal. (4 div.)	2E-5	10	0.05	27.5
12.	ROOIOPHL	Oper. fails to initiate within 30 minutes (Backup for RCIC)	0.10	5	0.04	1.00

Table 19D.7-6
HUMAN ACTIONS BELOW THE TOP 300 CUTSETS (2.0% OF CDF)

Name	Description	Assigned Probability	E.F.	Importance F.V. (%)	Importance R.A.W.
CALN002A	Miscal. of flow xmtrs FT008A, B & C	5E-5	10	0.15	31.8
RHRSPER	Oper. fails to manually initiate [SP cooling initiation (within 20 hours)]	6E-5	10	0.04	4.08
HCMAER1	Valve E22-F005C mispositioned (NOFC)	0.01	5	0.05	1.06
ROERROR3	Oper. fails to manually open valve	0.01	5	<0.01	1.00
ADSMAN	Failure of ADS manual init. (backup)	2E-3	5	0.01	1.06
NHR	Failure to restore normal heat removal	0.01	15	<.01	1.00
RWCU	Failure to actuate RWCU	0.10	5	<.01	1.00
RSTTCOPF	Operator fails to reset trip circuit (RCIC internal trips)	0.01	5	<.01	1.00
SLC006SA	Boron concentration sampling failure	2E-5	10	<.01	1.07
SLC001HE	Operator fails to initiate SLC	0.10	10	<.01	1.00
SLC002HE	Operator fails to initiate SLC tank heater	2E-3	5	<.01	1.00
WOPERR	Oper. fails to perform indicated action (Backup to RBCW initiation)	0.01	5	*	*
HUEROR5	Oper. fails to transfer from CST to SP	0.01	5	*	*
VOPERRF	Operator fails to start pump	1E-3	5	*	*
ASECSNA	Operator fails to backup N2 initiation	0.10	5	*	*

*Below the cutset cutoff level (E-13)

Table 19D.7-6
HUMAN ACTIONS BELOW THE TOP 300 CUTSETS (2.0% OF CDF) (CONTINUED)

Name	Description	Assigned Probability	E.F.	Importance F.V. (%)	Importance R.A.W.
CMAN	Operator fails to backup ARI initiation	0.10	5	*	*
LPL	Oper. fails to control W.L. in an ATWS	0.01	15	<.01	1.06
	<u>Electrical</u>				
EHU69C	Operator fails to transfer power	1E-3	10	<.01	1.06
EHUB1	Operator fails to bypass	1E-3	10	*	*
EHUB2	Operator fails to bypass	1E-3	10	*	*
EHUB3	Operator fails to bypass	1E-3	10	*	*
EHUB4	Operator fails to bypass	1E-3	10	*	*
EHUS1AD	Oper. fails to xfer stdby charger to Div. I	1E-3	10	*	*
EHUS1BD	Oper. fails to xfer stdby charger to Div. II	1E-3	10	*	*
EHUS1CD	Oper. fails to xfer stdby charger to Div. III	1E-3	10	*	*
EHUS1DD	Oper. fails to xfer stdby charger to Div. IV	1E-3	10	*	*
	<u>Miscalibrations</u>				
HFE008CF	Miscal. of flow xmtrs	5E-5	10	0.01	3.45
HPR007CF	Miscal. of pressure xmtrs	5E-5	10	0.01	3.45
AHPT006	Miscal. of pressure xmtrs	2E-5	10	*	*
RFE635HX	Miscal. of CST level sensors	2E-5	10	*	*
REOSSMSC	Elec. overspeed sensor miscal	5E-5	10	<.01	1.11
RPR309MC	High turbine exh. press. xmtr. miscal	5E-5	10	<.01	1.11
RMOSSMSC	Mech. overspeed sensor miscal	5E-5	10	<.01	1.11
RPR303MC	Low suction press. xmtr. miscal	5E-5	10	<.01	1.11
	<u>Valve Mispositions</u>				
ROERROR5	Valve F009 inadvertently left open	0.01	3	*	*
HBMAER2	Test valve E22-F009B inadvert. left open	0.01	5	*	*
HCMAER2	Test valve E22-F009C inadvert. left open	0.01	5	*	*
C001AMOV	Manual override fails initiation signal	1.8E-4	10	*	*
C001BMOV	Manual override fails initiation signal	1.8E-4	10	*	*
C001CMOV	Manual override fails initiation signal	1.8E-4	10	*	*

*Below the cutset cutoff level (E-13)

Table 19D.7-7

CDF INCREASE WITH ABWR PRA HRAs MULTIPLIED BY 4 (INDIVIDUALLY)

Rank	Name	Description	New Prob.	E.F.	CDF Increase (%)
1.	HOOBOPHL	Failure to manually initiate HPCF (Incl. hardwire backup for EMUX failure - HPCF B)	0.40	5	47.9
2.	Q	Failure to inject with feedwater	0.20	5	37.3
3.	COND	Failure to inject with condensate	0.40	15	5.39
4.	HBMAER1	Valve E22-F005B closed (NOFC)	0.04	5	4.98
5.	Q2	Failure to inject with feedwater (TIS)	0.52	5	2.28
6.	ROERROR4	Oper. fails to attempt manual vlv. op. [Backup for RCIC disch. vlv. (F013)]	0.40	5	0.44
7.	CALN002A	Miscal. of flow xmtrs FT008A, B, & C	2E-4	10	0.41
8.	CTGMANSW	CTG manual disconnect switch [left] open (Following maintenance on gas turbine gen.)	2E-4	3	0.21
9.	RPR005CF	Sensor miscalibration	2E-4	10	0.16
10.	RFL007CF	Sensor miscalibration	2E-4	10	0.16
11.	HFELEBHX	Water level 8 sensors miscal. (4 div.)	8E-5	10	0.15
12.	RHRSPER	Oper. fails to manually initiate [SP cooling initiation (within 20 hours)]	2.4E-4	10	0.06
13.	HFE008CF	Miscal. of flow xmtrs	2E-4	10	0.04
14.	HPR007CF	Miscal. of pressure xmtrs	5E-5	10	0.04

Table 19D.7-8
ABWR PRA ACCIDENT SEQUENCES CONTAINING HUMAN INTERACTIONS

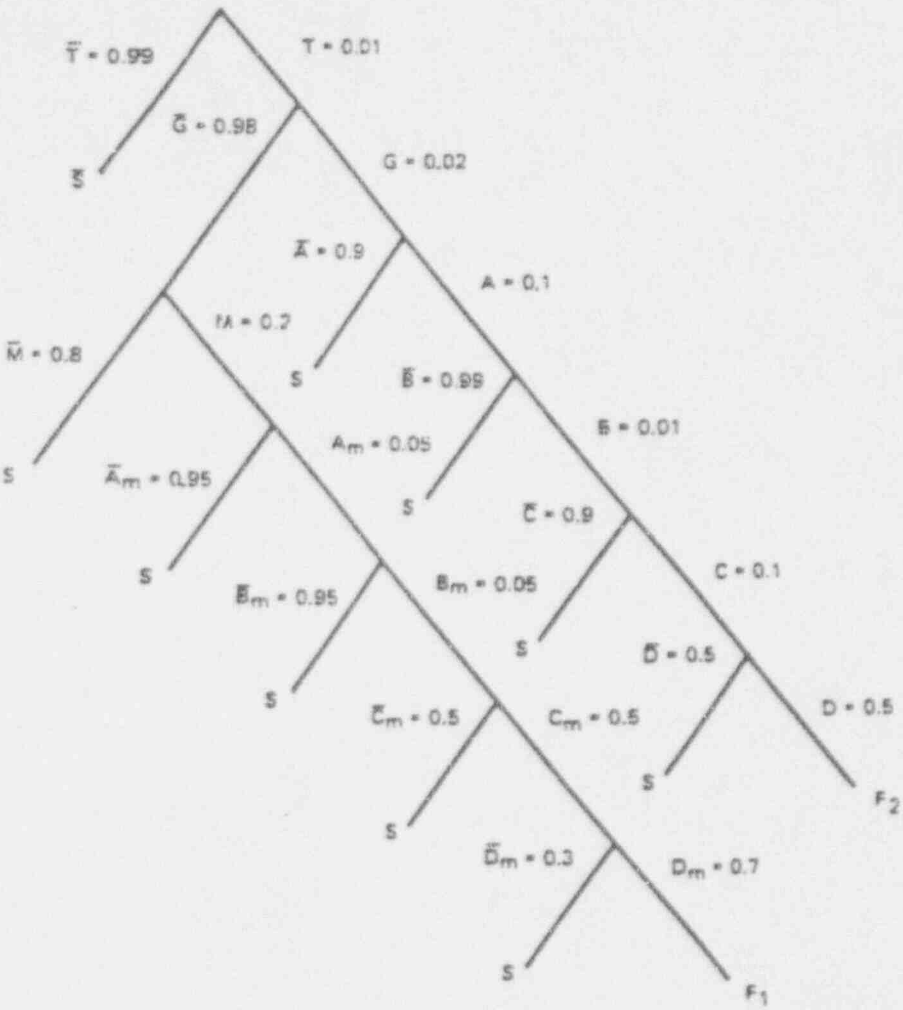
Initiating Event	Sequence	CDF
TM Manual Shutdown		1.15E-8
	TM-05	5.02E-10
	TM-06	1.10E-8
TT Unisolated Transient		6.83E-9
	TT-05	3.44E-10
	TT-06	6.44E-9
TIS Isolated Transient		1.70E-8
	TIS-06	1.61E-8
	TIS-12	4.84E-11
TIO IORV		1.24E-9
	TIO-04	1.54E-10
	TIO-05	1.09E-9
TE2 LOSP <2 hrs		4.47E-9
	TE2-06	4.45E-9
TE LOSP 2-8 hrs		2.88E-9
	TE8-06	2.85E-9
TEO LOSP >8 hrs		1.69E-9
	TEO-05	1.01E-9
	TEO-06	5.61E-10
	TEO-10	5.10E-11
BE2 SBO <2 hrs		6.67E-8
	BE2-02	6.67E-8
BE8 SBO 2-8 hrs		2.57E-8
	BE8-05	2.44E-8
S1 Medium LOCA		3.42E-10
	S1-04	1.42E-10
S2 Small LOCA		2.55E-10
	S2-06	2.45E-10
SO Large LOCA		9.02E-11
	SO-03	9.02E-11
ATWS		2.70E-10
	ATWS-11	1.39E-10

Table 19D.7-9
ABWR PRA CUTSETS CONTAINING A SINGLE HUMAN INTERACTION

HEPs	Sequence	No. Of Cutsets	CDF
Q	TT-06	24	2.28E-9
Q	TM-06	23	3.87E-9
Q	TIO-05	7	2.92E-10
Q	Total	54	6.44E-9
Q2	TIO-05	25	5.70E-9
COND	TEO-05	16	3.94E-10
COND	TEO-010	1	1.44E-11
COND	Total	17	4.08E-10
HOOBOPHL	TEO-06	3	2.06E-10
HOOBOPHL	TE2-06	3	2.90E-9
HOOBOPHL	TE8-06	3	1.86E-9
HOOBOPHL	SO-03	1	1.24E-11
HOOBOPHL	S1-04	3	7.91E-11
HOOBOPHL	S2-06	3	1.42E-10
HOOBOPHL	Total	16	5.21E-9
HBMAER1	TE2-06	3	2.90E-10
HBMAER1	TE8-06	3	1.86E-10
HBMAER1	ATWS-11	1	1.00E-10
HBMAER1	Total	7	5.77E-10
ROERROR4	BE2-02	2	1.66E-10
ROERROR4	BE8-05	2	6.06E-11
ROERROR4	Total	4	2.26E-10
RPR005CF	BE8-05	2	8.33E-11
RFL007CF	BE8-05	2	8.33E-11
HFELEBHX	BE2-02	1	2.44E-11
Total		128	1.88E-8 (12.0%)

Table 19D.7-10
ABWR CUTSETS CONTAINING JUST TWO HUMAN INTERACTIONS

HEPs	Sequence	No. Of Cutsets	CDF
Q and COND	TM-05	12	3.96E-10
Q and COND	TT-05	5	1.65E-10
Q and COND	TIO-04	1	1.38E-11
Q and COND	Total	18	5.75E-10
Q and HOOBOPHL	TM-06	2	4.68E-9
Q and HOOBOPHL	TT-06	5	6.78E-9
Q and HOOBOPHL	TIO-05	3	5.89E-10
Q and HOOBOPHL	Total	10	1.20E-8
Q and HBMAER1	TM-06	3	6.24E-10
Q and HBMAER1	TT-06	3	3.66E-10
Q and HBMAER1	TIO-05	3	5.89E-11
Q and HBMAER1	Total	9	1.05E-9
Q2 and COND	TIS-05	16	5.77E-10
Q2 and HOOBOPHL	TIS-06	3	9.13E-9
Q2 and HOOBOPHL	TIS-12	1	1.37E-11
Q2 and HOOBOPHL	Total	4	9.14E-9
Q2 and HBMAER1	TIS-06	2	6.85E-10
Total		59	2.40E-8 (15.4%)



LEGEND:

S	=	SUCCESS
F	=	FAILURE
T	=	PROBABILITY TOOL ROOM ISSUES BAD CALIBRATION TOOL
G	=	PROBABILITY TOOL IS GROSSLY OUT OF CALIBRATION
M	=	PROBABILITY TOOL IS MODERATELY OUT OF CALIBRATION
A,B,C,D	=	PROBABILITY THAT INSTRUMENT TECHNICIAN FAILS TO RECOGNIZE THE ERROR IN THE CALIBRATION TOOL
m	=	(SUBSCRIPT) INDICATES MODERATE MISCALIBRATION CASE

SOLUTION:

$$P(F) = P(F_1) + P(F_2) = T\,[G M A_m B_m C_m D_m + G\,A\,B\,C\,D]$$
$$= 0.01\,[(0.98)(0.2)(0.5)(0.05)(0.5)(0.7) + (0.02)(0.1)(0.01)(0.1)(0.5)]$$
$$= 0.01\,[1.715 \times 10^{-3} + 1 \times 10^{-6}] = 0.01 \times 1.716 \times 10^{-3}$$
$$= 1.7 \times 10^{-5} \quad (\text{USE } 2 \times 10^{-5})$$

FIGURE 19D.7-1

EVENT SEQUENCE USED TO DERIVE HUMAN ERROR PROBABILITY DURING A CALIBRATION PROCEDURE

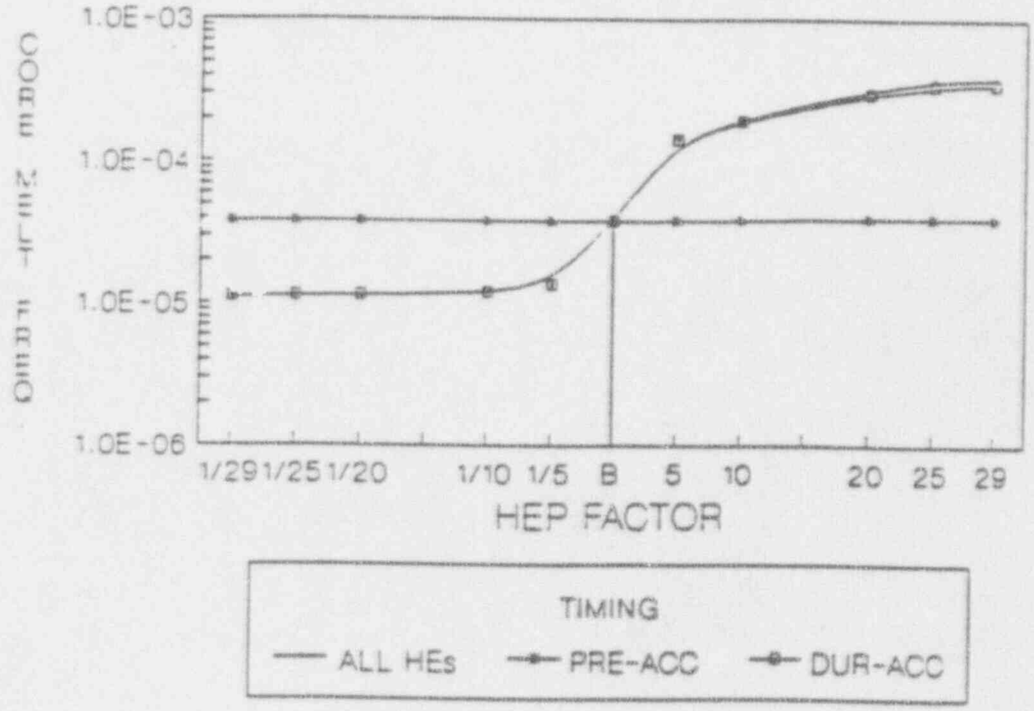
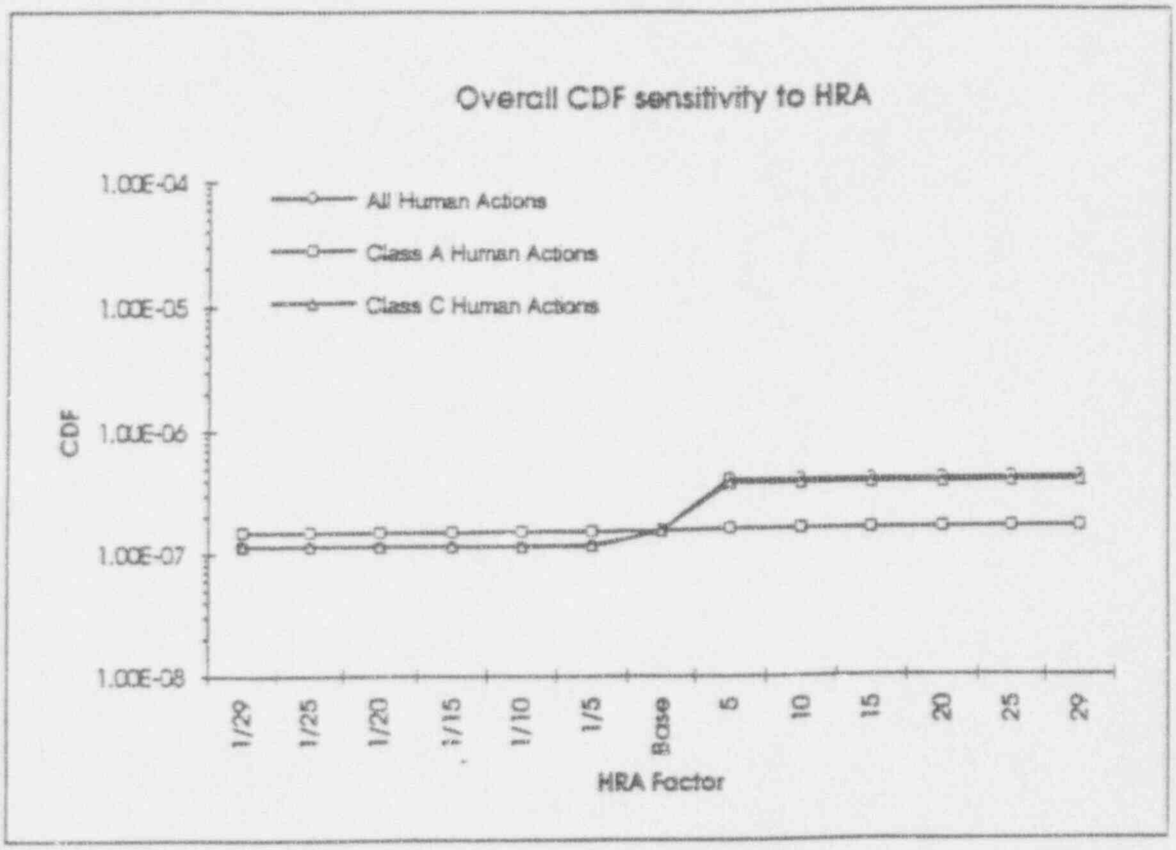
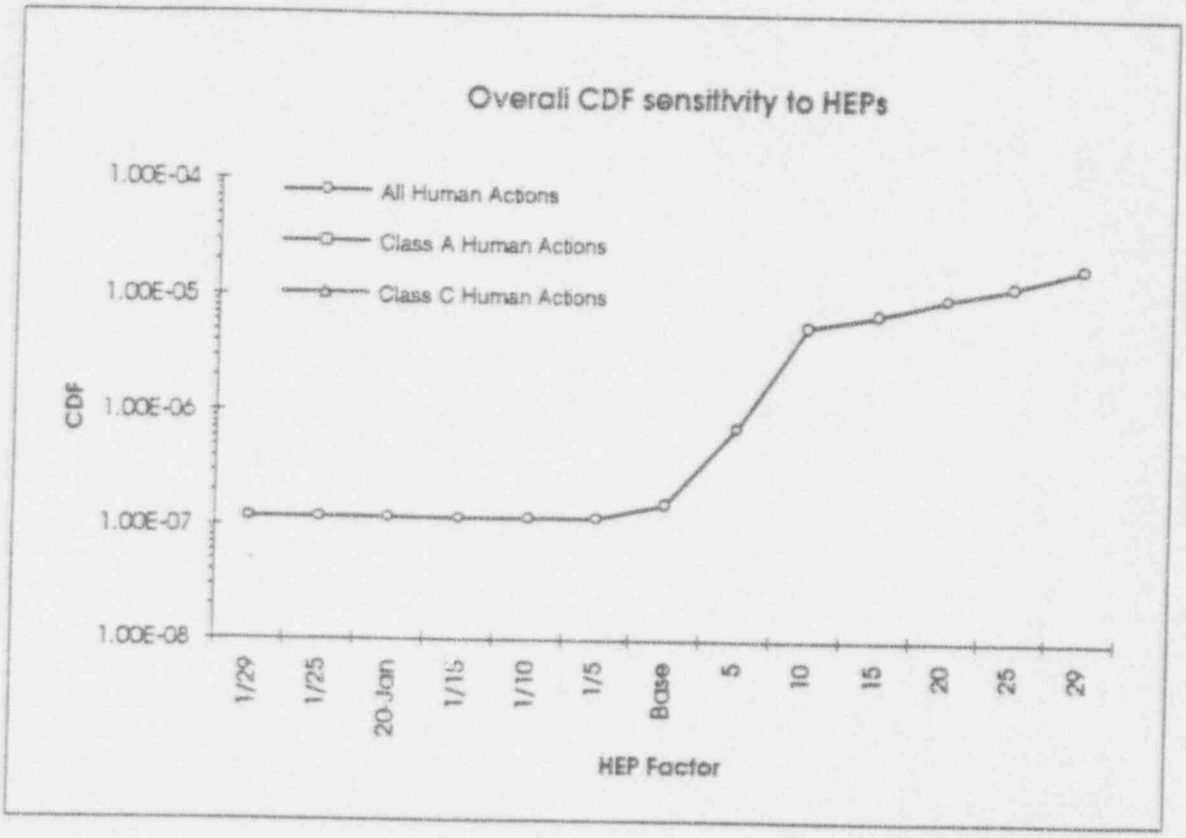


Figure 19D.7-2
SENSITIVITY OF CMF TO PRE-ACCIDENT & DURING-ACCIDENT ERRORS. RESULTS OF THE BNL HRA SENSITIVITY ANALYSIS.
(Reference 5)



Factor	CDF Class A	CDF Class C	CDF ALL
1/29	1.51E-07	1.16E-07	1.15E-07
1/25	1.51E-07	1.16E-07	1.15E-07
1/20	1.51E-07	1.16E-07	1.15E-07
1/15	1.51E-07	1.16E-07	1.15E-07
1/10	1.53E-07	1.16E-07	1.15E-07
1/5	1.54E-07	1.18E-07	1.17E-07
Base	1.56E-07	1.56E-07	1.56E-07
5	1.65E-07	3.86E-07	4.13E-07
10	1.65E-07	3.86E-07	4.13E-07
15	1.65E-07	3.86E-07	4.13E-07
20	1.65E-07	3.86E-07	4.13E-07
25	1.65E-07	3.86E-07	4.13E-07
29	1.65E-07	3.86E-07	4.13E-07

Figure 19D.7-3
RESULTS OF THE ABWR SENSITIVITY ANALYSIS
(In Comparison to the Results of Reference 5)



Factor	CDF Class A	CDF Class C	CDF ALL
1/29			1.17E-07
1/25			1.17E-07
20-Jan			1.17E-07
1/15			1.17E-07
1/10			1.17E-07
1/5			1.18E-07
Base			1.56E-07
5			7.47E-07
10			5.64E-06
15			7.24E-06
20			9.89E-06
25			1.25E-05
29			1.82E-05

Figure 19D.7-4
ABWR PRA CDF WITH ALL HEPs MULTIPLIED SIMULTANEOUSLY BY A FACTOR