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SUBJECT: Forwards response to NRC 900314 request for addl info re PRA, "Long Term Seismic Program."

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James D. Shiffer
Senior Vice President and
General Manager
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May 3, 1990

PG&E Letter No. DCL-90-118

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Re: Docket No. 50-275, OL-DPR-80
Docket No. 50-323, OL-DPR-82
Diablo Canyon Units 1 and 2
Long Term Seismic Program - Probabilistic Risk Assessment

Gentlemen:

In NRC's letter dated March 14, 1990, the NRC Staff requested additional information regarding the probabilistic risk assessment (PRA) performed for PG&E's Long Term Seismic Program. PG&E's response to the NRC letter is provided as Enclosure 1. Additionally, in a meeting on April 3, 1990, between PG&E and the NRC, the Staff identified several action items for PG&E involving clarification of PRA modeling details. PG&E's response to the action items is provided in Enclosure 2.

Kindly acknowledge receipt of this material on the enclosed copy of this letter and return it in the enclosed addressed envelope.

Sincerely,

J. D. Shiffer

Enclosures

cc: N. Chokshi, NRC (w/enc.)
R. Fitzpatrick, BNL (w/enc.)
M. Bohn, Sandia (w/enc.)
A. P. Hodgdon
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ENCLOSURE 1

PG&E RESPONSE TO NRC'S MARCH 14, 1990 REQUEST

In Enclosure 1 of NRC's letter dated March 14, 1990, the NRC Staff requested additional information relating to the Staff's review of the Diablo Canyon Probabilistic Risk Assessment; the following provides the requested information.

ITEM 1:

In your response dated January 23, 1990 to a question on the fire analysis, you stated that the operator action ZHEF11 was reevaluated with 10 minutes as a time period before the onset of seal damage because of the pump bearing failure. The period of 10 minutes was chosen based on a Westinghouse analysis. Please provide the details of this supporting analysis.

RESPONSE TO ITEM 1:

This statement is based on tests performed on RCP motors which are very similar in construction to those at Diablo Canyon. The tests were performed on the Westinghouse Electro-Mechanical Division test loop at normal reactor coolant system pressure thereby duplicating, very closely, actual thrust bearing loadings. During the tests, the RCP motors were operated for a minimum of 10 minutes with the cooling water isolated from the bearing oil coolers. At test completion, the motors' bearings were inspected and no damage was observed. Since no damage occurred, 10 minutes is the minimum amount of time the RCPs can operate without CCW cooling.

ITEM 2:

In response to questions related to Anticipated Transients Without Trip (ATWT) sequences, please provide the status of the implementation of AMSAC system as required by the ATWS rule.

RESPONSE TO ITEM 2:

The ATWS Mitigation System Actuation Circuitry (AMSAC) has been installed in both Units 1 and 2 at Diablo Canyon; it was installed in Unit 2 during the second refueling outage in late 1988 and in Unit 1 during the third refueling outage in late 1989.

ITEM 3:

With respect to control room fire scenarios, it is not clear how the human actions at the hot shutdown panel and the associated timing are modeled for the cases where the RCPs are not tripped before evacuating the control room. Based on discussions with you, we conclude that reexamination of these scenarios is necessary. Please provide this reexamination and the following information:

a. Discuss the independence between the control room and hot shutdown panel circuitry (both actuation and instrumentation).

b. Discuss whether a LOCA can be mitigated from the hot shutdown panel, and if so, discuss the applicable procedure.

c. Discuss the procedure for tripping the RCPs from outside the control room when component cooling water is lost.

RESPONSE TO ITEM 3:

The reexamination of the control room fire scenarios is provided in the enclosed revised Appendix F Section 3: Diablo Canyon Fire Risk Assessment.

RESPONSE TO ITEM 3a:

The hot shutdown panel (HSDP) is located in the auxiliary building at elevation 104 ft., which is three floors down from the control room. The following controls and indication are available on the HSDP:

AFW pumps, 2 motor-driven and 1 turbine-driven.
AFW pump discharge pressure.
AFW flow.
AFW water source levels.
Steam Generator (SG) level control valves.
SG 10% atmospheric steam dump valves.
SG pressure and level.
Component Cooling Water Pumps (3).
Auxiliary Saltwater Pumps (2).
Containment Fan Coolers (5).
Centrifugal Charging Pumps (2).
Letdown valves (3).
Emergency boric acid valve.
Charging pump flow contol valves (2).
Pressurizer PORVs close only (3).
Pressurizer heaters (2).

Independence from the control room - Controls

Independence from the control room for the controls of the above components is achieved through the use of transfer control relays and cutin/cutout switches. The transfer control relays are actuated by the transfer control switch on the HSDP and the proper positioning of the cutin/cutout switches in the 4.16kV breaker cubicles.

Each control switch on the HSDP has an associated transfer control switch with two positions, Control Room/Local. Control of each component may be individually actuated. When the switch is in the "Control Room" position, the control room portion of the component's controls is activated. When the switch is in "Local" position, the HSDP controls are activated provided that the cutin/cutout switch is in the cutin position; under these conditions, the transfer control relay is energized, which cause contacts to open in the portions of the control circuit which receive input from the control room, and contacts to close in the portions of the circuits which receive input from the HSDP. Once this occurs, the control circuitry which passes through the cable spreading room and the control room is isolated from the rest of the control circuit.

The cutin/cutout switches are located at the switchgear rooms. Depending on the component, some of these switches are normally cutin while others are normally cutout. For those that are normally cutout, the procedure to

establish hot standby from outside the control room (OP AP-8, Section A) instruct the operators to cutin those switches which are normally cutout.

Some safe shutdown equipment is controlled from locations other than the HSDP. For the RHR pumps, switches in the corresponding 4.16kV switchgear cubicles are provided to isloate control room circuitry and allow control of the RHR pumps from the switchgear rooms. The diesel generator controls may be isolated from the control room by operation of local control switch on the diesel generator control panel.

Independence from the control room - Instrumentation

The instrumentation available at the HSDP are not independent from the control room circuits; if an instrumentation circuit should fail, the instrumentation at the HSDP will fail offscale. However, the dedicated shutdown panel (DSP) instrumentation is independent from the control room instrumentation. The following instrumentation is provided on the DSP:

a. Steam Generator Level Indication (4).
b. Reactor Coolant Pressure (1)
c. Pressurizer level (1).
d. Reactor Coolant System Temperature (1).
e. Pressurizer Auxiliary Spray Valve 8145 control.

All of the above instruments except RCS temperature are dedicated for remote indication and are in no way connected with circuitry which passes through the control room or the cable spreading room. Reactor coolant temperature also sends a signal to the control room; however, a switch is provided at the DSP which transfers indication from the control room to the DSP. Additionally, procedures (OP AP-8, Appendix E) instruct the operators to utilize the insturmentation at the DSP if necessary.

RESPONSE TO ITEM 3b:

The hot shutdown panel is not designed for LOCA mitigation and procedures are not written which explicitily describe how to respond to a LOCA from the HSDP. However, a full set of plant procedures are available at the HSDP and, in principle, it would be possible to actuate the required systems and monitor necessary instrumentation from outside the control room at various remote locations.

RESPONSE TO ITEM 3c:

There is no procedure which explicitly instructs the operators to trip the RCPs from outside the control room when control room evacuation is required and when component cooling water is lost. However, there are a variety of annunciators responses and procedures which will instruct the operators to trip the RCPs. Additional discussion of these indications is provided in the revised Appendix F Section 3: Diablo Canyon Fire Risk Assessment.

ADDITIONAL CLARIFYING INFORMATION

In Enclosure 2 of PG&E letter DCL-90-021, dated January 23, 1990, it was stated that "even without CCW cooling to the charging pump package coolers, the charging pumps will continue to operate without damage for a considerable period of time (actual experience data indicates well over one hour)."

The details of the experience data were requested by the Staff verbally and are provided in the following paragraphs.

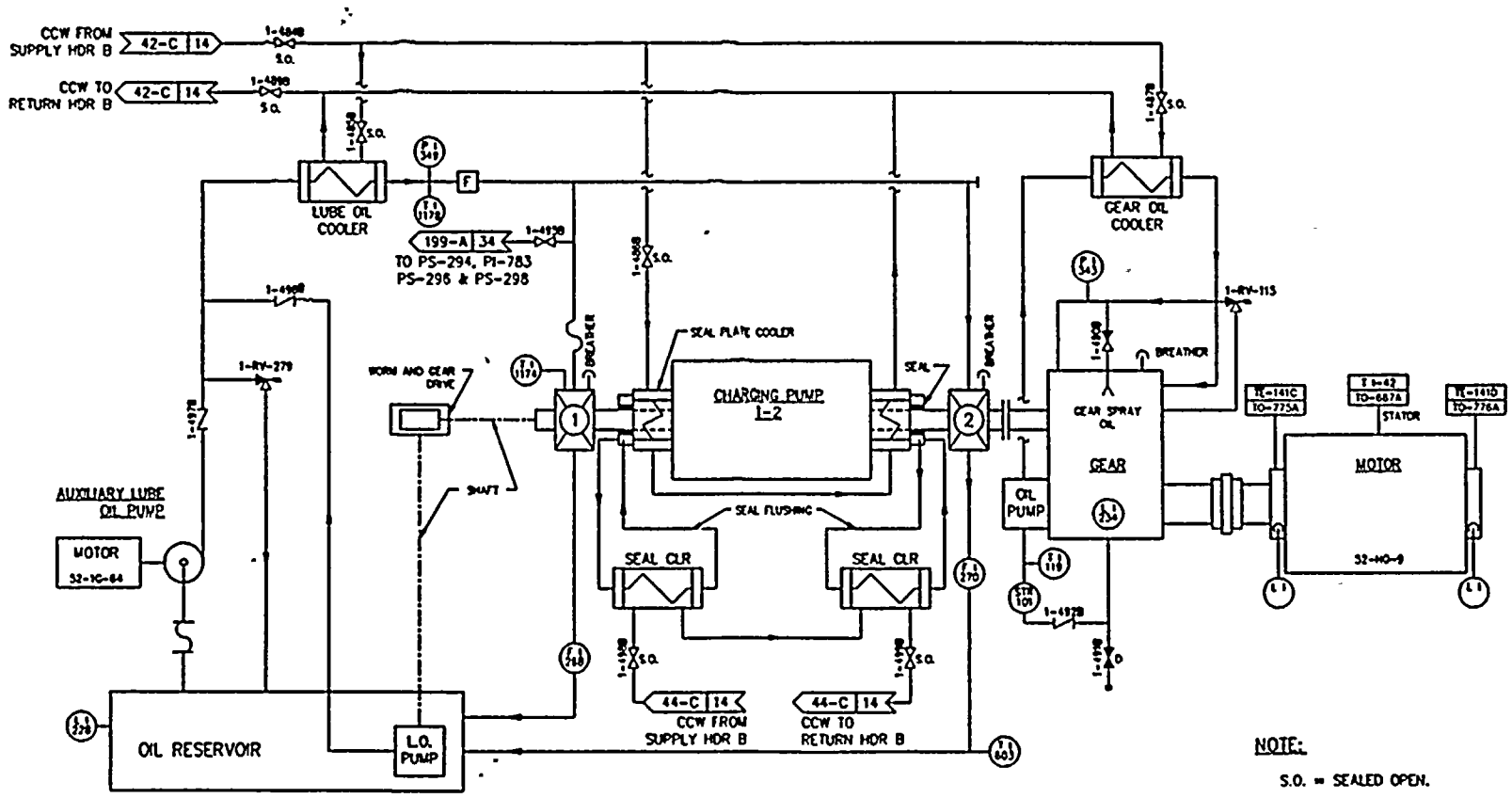
On November 20, 1989 during a Unit 1 refueling outage, surveillance test STP V-15 (ECCS Flow Balance Test), was performed; valve CVCS-1-484B (see Figure 1) was inadvertently left in the closed position, which resulted in the loss of CCW cooling water to the lube oil cooler, gear oil cooler, and the seal plate cooler of CCP 1-2. CCW flow was still provided to the seal flush cooler through valve CVCS-1-498B.

After the pump was run for 1 hour and 7 minutes at 450-500 gpm, a high stator temperature alarm on CCP 1-2 annunciated in the control room. An operator was dispatched to investigate the high temperature alarm; using local indication, the operator identified high lube oil and thrust bearing temperatures and the pump was shut down after 1 hour and 20 minutes of operation. Subsequent alignment checks discovered the incorrect positioning of CVCS-1-484B.

To evaluate potential pump damage, maintenance personnel took lube oil samples; the oil samples did not show signs of any bearing damage and the pump was subsequently tested to record vibration data and oil temperature data; since the data indicated that pump operation was acceptable, the pump was returned to service and subsequent pump operation has been acceptable.

Because the temperature of the pumped water is generally cool (whether taking suction from the volume control tank or the refueling water storage tank), it is believed that the cooling provided to the seal flush coolers was not important in preventing damage to the pump. Therefore, the conditions of operation of this pump were representative of pump operation without any CCW.

Based on this operating experience, it is concluded that the centrifugal charging pumps at DCPP are capable of operating for a significant period of time (on the order of 1 hour) without sustaining damage.



NOTE:

S.O. = SEALED OPEN.

CENTRIFUGAL CHARGING PUMP 1-2
LUBE OIL, GEAR OIL, AND SEAL FLUSH AND COOLER PIPING

Figure 1

Enclosure 1

-5-

UNIT ONE

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ENCLOSURE 2

PG&E RESPONSE TO APRIL 3, 1990 ACTION ITEMS

On April 3 and 4, 1990, a meeting was held between Pacific Gas and Electric Company and the NRC on the review of the Diablo Canyon PRA. As a result of that meeting, 13 items were identified for PG&E's action. The following provides PG&E's responses to these action items.

ITEM 1

It was requested that additional information be provided regarding the treatment of uncertainty in the quantification of the non-seismic dominant sequences, the fire scenarios and the seismic model.

RESPONSE TO ITEM 1:

The development of the non-seismic and seismic dominant sequence models was discussed previously in Enclosure 2 of PG&E letter DCL-90-046, dated February 16, 1990. This response supplements that letter by discussing the treatment of uncertainty using the dominant sequence models.

The non-seismic dominant sequence model accounts for all initiating events other than seismic events which are treated separately. The important sequences for each of the non-seismic initiators are represented explicitly in the non-seismic dominant sequence model. This includes all the so-called internal initiators and a number of spatial hazards such as internal fires and floods. All initiators which were run through the plant event tree models are accounted for in the dominant sequence model, even though some do not explicitly appear.

The control room and cable spreading room fires were quantified separately from the plant event tree models. The uncertainties in these fire scenario frequencies and the non-recovery factors were combined using monte-carlo simulation to arrive at the combined frequency of core damage due to these initiators only. This computation was performed using RISKMAN(1), but in a calculation separate from the dominant sequence model. The resulting uncertainty distribution for the core damage frequency from control room and cable spreading room fires was then added to the dominant sequence model as the term CRFIRE.

The class of events attributed to the release of hazardous chemicals was treated similar to that of control room and cable spreading room fires. Five chlorine and ammonia tank release scenarios were considered. The total frequency of these scenarios leading to hazardous chemicals reaching the control room air intake was computed. The point estimate of the sum of the hazardous chemical scenario frequencies was taken to be the mean of a lognormal distribution with a range factor of 10; this distribution is represented by the variable HAZCHM in the dominant sequence model. For the uncertainty analysis, a factor of 0.1 was used to account for the fraction of time that the wind direction and atmospheric conditions would be unfavorable. The conditional frequency of core damage given a hazardous chemical release that arrives at the control room air intake, was then judged to be limited by the requirement for the operators to replenish the CST within 8 hours. This assumes that the operators have tripped the plant prior to having to evacuate the control room. The human action ZHEHS5 is used to represent the likelihood that the operators fail to prevent core damage given that they have to abandon the control room. The uncertainty distributions for variables ZHEHS5 and HAZCHM are combined in the non-seismic dominant sequence model to estimate the core damage frequency contribution from hazardous chemicals.

The uncertainty in the non-seismic core damage frequency was computed using the top 420, explicitly represented, sequences plus the terms for fires and chemical releases discussed above. The uncertainty in each system split fraction value was computed using monte-carlo uncertainty propagation. The computer code RISKMAN was used to propagate the uncertainty in the failure data through the algebraic equations derived for each of the system models. This process is similar to that used by the SAMPLE program in the Reactor Safety Study.

One important difference is that RISKMAN can sample from any distribution shape that can be represented as a discrete distribution; i.e. it is not restricted to lognormal distributions. Also, the resulting uncertainty distributions for the system unavailabilities are saved as discrete distributions which can also be of any shape. Key statistics for the resulting system unavailability uncertainty distributions are presented in Table 6-37 of the LTSP Final Report(2). These distributions should not be interpreted as being of a particular analytical shape.

The uncertainties in the system unavailabilities are used as input to the computation of the uncertainty in the core damage frequency. Distributions are also used for the uncertainties in the initiating event frequencies and the uncertainties in the non-recovery factors (see Tables 6-33 and 6-53 of the LTSP Final Report). All of the distributions mentioned are represented by discrete distributions.

RISKMAN was then used to compute the uncertainty in the core damage frequency using monte-carlo sampling from these distributions and quantifying the dominant sequence model equations. The result is an uncertainty distribution for the non-seismic core damage frequency.

Table 6-54 of the LTSP Final Report shows the uncertainty in the core damage frequency from groups of initiating events. The distribution presented for internal events is generated from the non-seismic dominant sequence model by zeroing out the frequencies of all other initiators; i.e. fires, floods, and chemical releases. Similarly, the uncertainty in the core damage frequency from other external events is obtained by zeroing out the internal initiating event frequencies (e.g. reactor trip, LOCAs, loss of a DC bus, etc.) and then re-running the monte-carlo simulation.

Table 6-54 also presents the uncertainty distributions for the core damage frequency from seismic events and for the total from all initiators. To compute these distributions, separate uncertainty quantification models were used.

The development of the core damage sequence boolean for seismic events was described previously in PG&E letter DCL-89-283, dated November 13, 1989. This boolean contains terms accounting for both seismic and non-seismic (random failures) of equipment. The uncertainties in the non-seismic equipment failure rates were neglected compared to the much larger uncertainties in the seismic equipment fragilities. This simplified the seismic uncertainty model by permitting the non-seismic equipment failure rates to be treated as constants.

The uncertainties in the seismic fragilities and the hazard curves were input to the seismic uncertainty propagation code SEIS4(3). The family of hazard curves were divided into eight discrete curves presented in Table 6-38 of the LTSP Final Report. The 59 fragility curves were input to the SEIS4 code in terms of three variables; the median acceleration, Beta R, and Beta U. SEIS4 discretizes the continuous family of fragility curves into 18 separate curves and the continuum of seismic levels was divided into 10 discrete ranges for the uncertainty computation. SEIS4 solved the seismic core damage boolean at each seismic level using discrete probability arithmetic. The resulting family of conditional core damage frequency curves were then convolved with the family of seismic hazard curves to arrive at the uncertainty in the seismically initiated core damage frequency, which is the result presented in Table 6-54 of the LTSP Final Report.

To obtain the uncertainty distribution for the total core damage frequency from all initiators, the distribution for seismic initiators was added to a distribution representing the core damage frequency from all non-seismic initiators as computed by the dominant sequence model. RISKMAN was used to conbine these two distributions to obtain the resulting distribution for the total core damage frequency.

REFERENCES:

1. Pickard, Lowe and Garrick Inc., "Riskman - An Interactive Computer Code" March, 1985.

2. Diablo Canyon Long Term Seismic Program Final Report; Pacific Gas and Electric Company, July 1988.

3. Pickard, Lowe and Garrick Inc., "SEIS4-Seismic Risk Assessment Computer Code Users Manual", December, 1985.

ITEM 2:

It was requested that PG&E provide the importance, in terms of the contribution to the core damage frequency, for fire initiating events FS2, FS3, FS4, and FS7. These initiating events do not appear in the non-seismic dominant sequence model.

RESPONSE TO ITEM 2:

The contributions of fire scenarios FS2, FS3, FS4, and FS7 to the total core damage frequency and the highest frequency core damage sequence initiated by each of these fire scenarios are provided below.

Core Damage Frequency(1):

FS2 = 8.88E-8
FS3 = 5.56E-8
FS4 = 7.59E-8
FS7 = 7.08E-8

Highest Frequency Core Damage Sequence(2):

FS2 = 1.39E-8
FS3 = 3.81E-8
FS4 = 9.61E-9
FS7 = 1.66E-8

These fire scenarios do not appear in the non-seismic dominant sequence model because they were not in the top 420 sequences. The lowest frequency sequence in the dominant sequence model, prior to recovery, is the 420th sequence with a frequency of 1.11E-7. Since the highest frequency sequence initiated by the fire scenarios FS2, FS3, FS4, and FS7 were much lower than the 420th sequence, these fire sequences were not explicitly included in the model.

REFERENCES:

1. Computer file <BLACK>PGE.1123>EVENT.TREES>INTERNALS>M40
   Lines 11106, 11107, 11108, and 11111

2. Computer file <BLACK>PGE.1123>EVENT.TREES>INTERNALS>M40
   Lines 6713, 5760, 6724, and 6017.

ITEM 3:

According to BNL, the general transient initiators used in the DCPRA appear to be a factor of two less than the initiators used in other PRAs. A discussion of the development of the generic prior distributions for these types of initiators was requested along with a discussion of the screening criteria used and whether the events screened out could still have occurred during at power conditions.

RESPONSE TO ITEM 3:

The generic initiating event frequencies used to generate the "generic prior" distributions for the DCPRA are based on Reference 1. The two main sources for the generic data used in Reference 1 were:

1. An Idaho National Engineering Laboratory (INEL) study of transients at U.S. nuclear power plants (2). Events selected from this study were those causing forced shutdowns at PWRs from 1980 to 1983.

2. A compilation of License Event Reports by Tennessee Valley Authority. These PWR transients events, from 1984 through July 1987, were used to supplement the INEL study.

The transient events collected from the above sources, which cover a period of 7.6 years from 1980 through July 1987, were further screened to include only those events that are appropriate for use in PRA of a nuclear plant in power operation. The criteria used in the screening process were:

1. Those trip events occurring at or below 25% power were excluded if they occurred during power ascent or during shutdown. Trip events at or below 25% power were included if they occurred during power descent or during prolonged operation at a low power level. It is assumed that the decay heat at low power levels during power ascent is not significant since there is no buildup of fission materials. Consequently, the decay heat is at the same level as at shutdown and these events should not be considered for a full power risk assessment.

2. Trip events occurring between 25% power and 50% power were excluded if they occurred during ascent and were also due to feedwater instability. Feedwater problems are extremely common during power ascent, and it was judged that these events would not be applicable to normal plant operating conditions.

Although no event by event examination was carried out for the events that were screened out, it is believed that the majority of these trips are caused by the activities going on during shutdown and during startup; This is based on experience gained from performing data analysis at almost 20 reactor units and the experience of PLG personnel with background in reactor operations. There may be a few events that have happened at shutdown or during startup that could also happen at power; however, there is so much data for at power conditions that omitting these few events has a negligable impact on the resulting initiating event frequencies. This fact, together with the fact that the initiating event estimate is not an average value but rather the mean of a distribution with a band of uncer-

tainty, supports the position that these few events are adequately represented in the prior distribution.

Many trip events not appropriate to PRA of a nuclear plant in power operation were excluded from the database when the above screening criteria were applied. Moreover, by considering PWR transients from the period 1980 through July 1987, trip events occurring during the first year of operation for many PWR units were excluded from the database. It is evident from Reference 2 that in general nuclear reactors experienced many more trip events in their first few years of operation than in subsequent years. A small number of nuclear plants have come on line in the 80s, and although the first year of their operation is included in the database, it is not typical of their later operating years. The first year data of these plants only tends to make the prior distribution a little wider and the mean a little higher.

The earlier PRAs (Seabrook PSA, TMI-1 PRA, Beznau PRA, Bellefont PRA) used the data developed in Reference 3. This was compiled from information in the EPRI report NP-2230. In the EPRI report, the data was classified into categories, but no description was provided about the event or the plant status when the event occurred. As a result, no screening was done for this data, and the distributions were defined using all the data in each category.

It is expected that the transient initiating event frequencies computed using a screening criteria, as was done for DCPRA , would be lower than the values calculated by using all of the data in a transient events database as was done with NP-2230.

In addition, the DCPRA has factored in plant specific data. The data covered approximately 4.6 years of reactor operation. The updated distributions for the most frequent initiators, reactor trip (RT), turbine trip (TT), and partial loss of main feedwater (PLMF) resulted in lower plant specific mean valves then the generic data. More recently, additional plant specific data (covering a total of 8.5 reactor years) has been collected. This additional data shows a confirmed decreasing trend in the plant specific mean valves below the generic mean.

The total generic mean frequency of plant trips caused by internal initiating events was estimated to be about 4.8 trips per reactor year, (Table 6-33 of the LTSP Final Report (5)). In the April 23, 1990 issue of "Inside NRC" (volume 12, No. 9), the average number of automatic trips while critical at each unit was reported as 2.26 trips per year in 1988 and 1.85 trips per year in 1989. These figures were derived by the NRC's Office for Analysis and Evaluation of Operational Data. This strongly suggests that if more industry data were collected for the year after 1987, the generic priors would likely decrease.

This information combined with the recent 400 day continuous operation of Diablo Canyon Unit 1 and the high lifetime availability factors for both Units 1 and 2 (70% to 80% including refueling outage) support the lower initiating event frequencies for the above categories. Use of the initiator data of the older PRA studies is believed to be unrealistic.

REFERENCES:

1. Database for Probabilistic Risk Assessment of Light Water Nuclear Power Plants, PLG-0500, Volume 6, Revision 1.

2. Idaho National Engineering Laboratory, "Development of Transient Initiating Event Frequencies for Use in Probabilistic Risk Assessments," NUREG/CR-3862, May 1985.

3. Database for Probabilistic Risk Assessment of Light Water Nuclear Power Plants, PLG-0500, Volume 6, Revision 0.

4. A. S. McClymont and B. W. Poehlman, "ATWS: a Reappraisal, Part 3: Frequency of Anticipated Transients", NP-2230, January 1982.

5. Diablo Canyon Long Term Seismic Program Final Report; Pacific Gas and Electric Company, July 1988.

ITEM 4:

Several issues were raised regarding the interfacing systems LOCA analysis; these primarily related to the use of Seabrook data for check valve failure/leak rate. The issue relates to the way data was collected for the Seabrook analysis since this same data was used in the DCPRA.

RESPONSE TO ITEM 4:

The check valve failure rate for disc rupture/gross reverse leakage was derived from earlier work performed as part of the Seabrook Emergency Planning Zone (EPZ) study (1). As part of that study, a review of Nuclear Power Experience for check valve failure events was performed. The initial list of 692 events was screened for those events which involved leakage. A total of 163 leakage events were identified. This list of events was further screened to include only those which involved leakage from check valves at the ECCS and ECCS/RCS boundary of PWRs. No disc ruptures were identified, and the largest leak rate was 200 gpm. Of the initial 692 events, 21 check valve failure events were identified as applicable to the leakage failure mode and involved events in the ECCS and/or RCS systems of PWRs. There were 17 events associated with accumulator check valves and 4 events associated with ECCS/RCS interface valves. For other check valves, there were no failure events reported.

The success or exposure data for the check valves of interest were based on the information provided in NUREG/CR-1363. The total number of check valve hours was estimated at 1E8 hours. Attention was focused on the check valves that are most applicable to the interfacing LOCA events. These include the normally seated check valves in the ECCS systems including the accumulator check valves, those at the interface of the ECCS and the RCS, and those that separate the ECCS system form RWST, containment spray and the containment sump.

In BNL's earlier review of the Seabrook EPZ study, it was questioned whether it is appropriate to use all of the ECCS check valves in the population data base.

It is believed that the total population of valves is appropriate because they are all designed to ASME code, are safety grade, and therefore contain inherent margins of safety for the structural integrity of each valve to remain intact. In regard to this earlier question, a response was prepared (3) that showed that at the Seabrook station roughly 80% of the ECCS check valves were configured in a manner that could be termed interfacing check valves; i.e., namely communicating directly with the RCS or in series with another check valve that does. Moreover, the Seabrook EPZ study utilized experience data only thru November 30, 1984, which totals approximately 424 reactor years of PWR experience. Since that time, there has been over five years of additional experience and it is believed that there have not been any disc rupture/gross leakage events. This suggests that the current success or exposure data is much better than that available at the time of the Seabrook work, so that even if one were to further restrict the check valves which are assumed applicable, the additional success data experience since 1984 would substantially compensate.

That the failure rates for disc rupture/gross leakage reported in the DCPRA are reasonable, is also suggested by the work performed for the NRC (2). The DCPRA used the following disc rupture/gross leakage failure rates for it's assessment of the frequency of interfacing LOCA events:

Leak Rate	Median failure rate (per hour)
>150 gpm	1.9E-8
>800 gpm	4.3E-9
>1700 gpm	2.5E-9

By comparison, the distributions for the five experts (2) are as follows:

Expert ID	Failure Rate
A	A broad distribution whose upper tail ends at 2.0E-8 per hour and whose median is roughly 1E-9 per hour.
B	A broad distribution whose upper tail ends at 2.0E-8 per hour and whose median is roughly 1E-10 per hour.
C	A broad distribution whose upper tail ends at 1.5E-4 per year (1.7E-8 per hour) and whose median is roughly 4E-7 per year (4.6E-11 per hour).
D	A broad distribution whose upper tail ends at 1E-5 per year (1.1E-9 per hour) with a median at roughly 1E-6 per year (1.1E-10 per hour).
E	The same as that presented in the DCPRA (reference 1).

As can be seen from the above, the results presented for Diablo Canyon are comparable or even conservative to those leak rates assumed by the NRC.

REFERENCES:

1. "Seabrook Station Risk Management and Emergency Planning Study", prepared for Public Service Company of New Hampshire by Pickard, Lowe and Garrick, Inc., December, 1985; PLG-0432.

2. "Analysis of Core Damage Frequency From Internal Events: Expert Judgement Elicitation", prepared by Sandia National Laboratories for the USNRC, NUREG/CR-4550, Volume 2.

3. Letter from George S. Thomas of Public Service of New Hampshire to the USNRC dated January 20, 1987, "Comments to BNL Draft Report A-3852", document No. NYN-87-002.

ITEM 5:

BNL questioned whether RCP Seal LOCA's should be considered as initiating events. NUREG/CR-4400 "The Impact of Mechanical- and Maintenance-Induced Failures of Main Reactor Coolant Pump Seals on Plant Safety" was cited as reference.

RESPONSE TO ITEM 5:

NUREG/CR-4400 broadly classified the primary coolant leakages caused by mechanical- and maintenance-induced RCP seal failures into two categories: seal leakages below the normal makeup capacity and leakages in excess of the normal makeup capacity. For mechanical- and maintenance-induced RCP seal failures leading to primary coolant leakages within the normal makeup capacity, the authors of NUREG/CR-4400 concluded qualitatively that this category of events cannot alone lead to any significant safety impact. For a seal LOCA occurring during normal power operation with a leak rate greater than the normal makeup capacity, the transient imposed on the plant and the response of the operators to this event would not be different from the response following a small LOCA event. Therefore, this category of events would be grouped into the non-isolable small LOCA initiating event category in the DCPRA.

An initiating event is defined as an event that eventually either leads to an automatic reactor trip or requires a manual trip. A RCP seal LOCA occurring during normal plant operation and of sufficient magnitude to cause an automatic reactor trip or to require manual reactor trip would be classified (in the generic database for the DCPRA) as either a small break (non-isolable) LOCA or a very small LOCA. In the development of the PLG generic database, there were no non-isolable small LOCA events which occurred during power operation and resulted in reactor trips. However, to account for any possible events that might have been misinterpreted in the data compilation process, two small LOCAs that have occurred at shutdown or at low power levels during startup (without a reactor trip) were included for conservatism. The mean value of the generic prior initiating event frequency for a non-isolable small LOCA is approximately 6E-3 (1).

At DCPP, a seal LOCA would require the operator to make up for the loss of inventory by controlling the charging flow. One charging pump with a capacity of 100 gpm is normally in operation. If the leak is beyond the capacity of one pump, then procedures instruct the operators to start a second charging pump, and isolate letdown. At this point, the makeup capacity is about 250 gpm. If the leak rate is higher than can be made up by two charging pumps, then the operators are instructed to manually initiate SI signal, at which point an automatic trip signal will be generated. For leak rates less than 250 gpm, the operators will start a controlled shutdown without subjecting the plant to the transient caused by a trip from power.

The non-isolable small LOCA initiating event frequency for DCPP should therefore include the frequency of leaks exceeding 250 gpm together with a fraction of the frequency of leaks between 100 gpm (capacity of one charging pump) and 250 gpm that result in trip during the shutdown process.

Appendix A of NUREG/CR-4400 provides frequencies of RCP seal LOCAs in Westinghouse reactors with four loops (Figure 7., pg 24). It is based on leaks that have occurred at reactors, but it is not clear whether all of these leaks lead to reactor trips. Of all the RCP seal failure events analyzed in NUREG/CR-4400, only four cases developed a maximum leak rate in excess of 50 gpm. The Oconee 2 event (January 1974, 90 gpm) occurred prior to the plant's commercial operation, the Robinson event (May 1975, 400-500 gpm) occurred during plant shutdown, and the Indian Point 2 event (July 1977, 75 gpm) occurred during startup. Only the Arkansas 1 event (May 1980, 350 gpm) occurred during power operation and eventually resulted in a plant trip. As such, the frequencies developed for RCP seal LOCAs in NUREG/CR-4400 are inapplicable for the purpose of estimating initiating event frequency. In spite of this, for comparison purposes, from the best estimate curve of Figure 7, the frequency of an RCP leak between 50 and 100 gpm is approximately 6E-3, between 100 and 250 gpm is approximately 6E-3 and above 250 gpm is 6E-3.

Given that the small LOCA initiating event frequency used in the DCPRA was determined conservatively, since it includes events which did not result in reactor trips and since many events in NUREG/CR-4400 did not result in reactor trips, the DCPRA non-isolable small LOCA frequency is deemed adequate. It is noted that the small LOCA frequency used in NUREG/CR-4550 is approximately about 1E-3 (2).

For the case of very small LOCA, these events were modeled with the reactor trip initiator for the DCPRA. The frequency of these events is so low compared to the reactor trip frequency that it would have an insignificant impact on the DCPRA results. Therefore, the RCP very small seal LOCA frequency of 6E-3 would also be insignificant. It must be noted that, in addition to the non-isolable small LOCA initiating event category, the DCPRA also modeled another small LOCA initiating event category, i.e., isolable small LOCA which has a much higher frequency than non-isolable small LOCA.

In summary, after reviewing the assessment of RCP seal LOCAs in NUREG/CR-4400 and reviewing the basis for the generic small LOCA initiating event frequency in the DCPRA (3), it is concluded that RCP Seal LOCA events are adequately represented by the DCPRA non-isolable small LOCA frequency.

REFERENCES:

1. Diablo Canyon Long Term Seismic Program Final Report; Pacific Gas and Electric Company, July 1988.

2. NUREG/CR-4550, Vol. 3, November 1986. Analysis of Core Damage Frequency From Internal Events: Surry, Unit 1. Page IV-18.

3. Database for Probabilistic Risk Assessment of Light Water Nuclear Power Plants, PLG-0500, Revision 1.

ITEM 6:

BNL indicated that they plan to use their own calculated value for loss of auxiliary saltwater as an initiating event instead of the PG&E calculated value. PG&E has already put forth arguments to justify the PG&E determined value; however, if there is any additional information or justification which PG&E believes should be considered, then it should be provided.

RESPONSE TO ITEM 6:

The purpose of this response is to compare the loss of Auxiliary Saltwater (LOSW) initiating event models prepared by BNL and PG&E. The BNL analysis as discussed here is presented in (1). The PG&E LOSW model referenced in this comparison is an updated version of the DCPRA LOSW model (2).

The LOSW initiating event frequencies determined by the BNL and PG&E models are 5.1E-4 and 1.4E-4 per year respecively.

The primary difference between the BNL and PG&E models is that the BNL model assumes that a single Unit 2 ASW train cannot, by itself, supply sufficient cooling water to the CCW heat exchangers of both units. PG&E has determined that a single ASW pump train supplying both units provides adequate flow and heat removal capacity to both units; this is discussed in more detail in the Response to Item 8. Therefore, the PG&E model assumes that one ASW pump can supply the cooling water needs of both units for non-LOCA scenarios such as the loss of auxiliary saltwater initiating event model. This difference in the success criteria used in the two LOSW models accounts for almost all of the difference in the numerical results.

The difference in success criteria also affects the relative importance of other modeling assumptions. For example, the BNL model does not give any credit for recovery from the demusseling configuration even though such an action would be very simple and there would be adequate time available. If it is assumed that 90% of the time recovery from demusseling would be successful, then the results from BNL's loss of auxiliary saltwater model would decrease from 5.1E-4 to approximately 3.8E-4. If BNL were to use the same success criteria as PG&E, the PG&E and BNL initiating event frequency for LOSW would be in agreement.

REFERENCES:

1. Updated BNL Sensitivity Calculation on the Diablo Canyon Initiator Frequency, LOSW; Brookhaven National Laboratory, June 29, 1989.

2. Response to Brookhaven National Laboratory Letter Report-04/Rev. 1, Pacific Gas and Electric Company, June 1989.

ITEM 7:

BNL requsted that a summary of the key assumptions used in determining a battery life of 12 hours during station blackout events be provided.

RESPONSE TO ITEM 7:

The assumptions and calculational methodology used in establishing the 12 hour station battery life (Calculation No. 138-DC) are outlined below:

1. Battery capacity is calculated using IEEE 485-1978, C&D battery curve D-699-1.

2. Battery load current is based on field readings taken over a 2 year period prior to July 1988.

3. AC power is not available to the chargers.

4. Short term loads, such as breaker trip coils, are modeled as being energized for a period of 1 minute even though they may be energized for a much shorter time.

5. The blackout load is calculated by adding short term loads to the measured constant load.

6. The battery room temperature is assumed to be 77 degrees F or higher, since the ventilation system is not available.

7. DC powered equipment will operate with a battery voltage as low as 105V; this is based on the results of the periodic discharge tests which discharge the batteries down to 105V with the equipment operating.

8. Containment isolation occurs soon after blackout.

9. The auxiliary building ventilation system is in normal mode.

10. The steam generator 10% steam dump valves operate continuously.

11. 4kV and 12kV breakers are tripped during the initial attempt to transfer to the diesel.

12. The operators make 3 unsuccessful attempts to start the diesels during the first hour (at 1 min, 30 min, and 60 min).

13. The diesel generator field is flashed (50 amperes) on each start attempt.
14. AC power is restored at the end of 12hrs from the 230kV offsite power source.

15. The battery life was determined for the battery with the heaviest load (Battery 1-1). The battery life for the other two batteries were not evaluated, but would have longer discharge times than battery 1-1. It is likely that the DC powered equipment would continue to operate below 105V, thereby extending the battery life even further. This however, was not taken into account nor was any credit for load stripping assumed.

ITEM 8:

In PG&E Letter DCL-89-283, dated November 13, 1989 the basis for the success criteria for the ASW system and the impact of changing the success criteria for certain scenarios was discussed. BNL requested that PG&E provide the split fraction values used in the PG&E sensitivity analysis.

RESPONSE TO ITEM 8:

The DCPRA assumed that one ASW pump can supply the cooling needs of both units. For most scenarios, this assumption is supported by a review of the existing design calculations for the ASW and CCW systems. For certain scenarios such as LOCA's (including transient induced LOCAs) and Steam Line Break Inside Containment (SLBIC) however, additional heat loads on the containment fan coolers must be removed by the ASW system. Under these conditions it is not known how the CCW system will respond after the loss of Unit 1 ASW and prior to its restoration by crosstying to the other unit. To assess the impact of a change of success criteria for certain scenarios, a sensitivity calculation was performed using a success criteria of one-out-of-two pumps (i.e., credit is given for the Unit 1 pumps only) for the affected sequences.

For non-seismic initiating events, LOCA and SLBIC sequences were re-evaluated giving credit only to Unit 1 ASW pumps. In the calculation of loss of offsite power (LOOP) sequences, detailed power unavailability scenarios were modeled for the vital AC buses of both units. For seismic initiating events, only LOOP sequences were evaluated. The total frequency calculated for these sequences was then used to approximate the increase in the core damage frequency as a result of the more restrictive success criteria used for the LOCA scenarios.

With respect to the transient-induced LOCA sequences in the non-seismic calculations, only initiating events that challenge the RCS pressure relief and which can result in a stuck open PORV were considered in this analysis. Other than LOCA sequences induced by loss of the offsite power initiating event, the non-seismic LOCA sequences were obtained from the DCPRA dominant sequence model. For LOOP sequences, support failures such as the loss of DC power or AC bus failures are assumed negligible compared to diesel generator failures. To determine the LOOP sequences that should be re-quantified using the revised success criteria, an event tree that models the availability of emergency diesel power supply to each of the Unit 1 and Unit 2 vital AC buses was developed. This event tree structure with the appropriate split fraction assignment is shown in Figure 1.

Top events GF, GG, GH, 2G, and 2H model the five diesel generators. Alignment of the swing diesel to Unit 1 or Unit 2 is modeled by top event SW. Also shown in the figure are the split fractions used for; top event AS if one-out-of-two Unit 1 ASW pump success criteria is used, top event AS used in the DCPRA for the corresponding LOOP sequences, and top event PR. Also show in the figure is whether the same LOOP sequences with the same AS and PR split fractions were already included in DCPRA. LOOP sequences for which the ASW success criteria was originally one-out-of-two in the DCPRA model and which use the same split fraction in this evaluation do not need to be re-evaluated.

The assignment of split fraction for top events AS, PR, and SW is described in the following. When power is available to both PORV block valves associated with the PORVs equipped with N2 accumulators, split fraction PR1 is used, otherwise PRD is used. When power is available to both Unit 1 ASW pumps, split fraction ASA is used. If power is available to only one Unit 1 ASW pump, ASC is used. Top Event SW determines which unit the swing diesel will be aligned to. SW1 is used if Unit 1 and Unit 2 have the same number of buses energized. SW2 is used if Unit 2 has more buses energized, and SW3 is used if Unit 1 has more buses energized.

It is noted that recovery actions are developed for non-seismic LOOP sequences that involve diesel generator failures. These recovery actions are added to selected sequences; they are described in the following:

RXBUS1 - Operator action to crosstie buses such that power is available to the PORV block valves for isolating the stuck open PORV. Due to the additional equipment failures experienced (ASW and PORV), the operator failure rate is assumed to be five times worse than operator action ZHERE2 from the DCPRA. ZHERE2 is the human action failure rate for crosstying two vital buses after a diesel generator is recovered following a station blackout event. Since most of offsite power recovery effort is not performed by the plant operators, this recovery action can proceed in parallel with the bus-crosstying action, which would be performed by the plant operators. Offsite power recovery is therefore also considered for a period of one half hour (nonrecovery probability is estimated to be 0.75). The failure frequency equation developed for this recovery (RXBUS1) factors in the appropriate PR split fraction value given power is recovered to the bus. This recovery factor is used when Unit 1 buses HF and HG are energized and bus HH is unavailable.

RXBUS2 - Operator action to crosstie buses such that power is available to both Unit 1 ASW pumps. Offsite power recovery is also considered for a period of one half hour. This recovery factor is used when one Unit 1 bus which powers an ASW pump is de-energized but the other one is not.

RXBUS3 - Operator action to crosstie buses such that power is available to at least one Unit 1 ASW pump and PORV block valves. Offsite power recovery is considered for a period of one half hour. This recovery factor is used when Unit 1 bus HH is energized and buses HF and HG are unavailable.

RXBUS4 - This models recovery of offsite power. This recovery factor is used when all three Unit 1 buses are unavailable. If successful, power to both Unit 1 ASW pumps and PORV block valves would be available.

RESLC4 - This factor considers both offsite power and diesel generator recovery. If the turbine driven AFW pump is available, RESLC2 is used. If the turbine driven AFW pump is unavailable, RESLC1 is used. See Table 6-53 in the LTSP Final Report for RESLC1 and RESLC2. The variable AFTWP is equal to split fraction AW9 from the DCPRA.

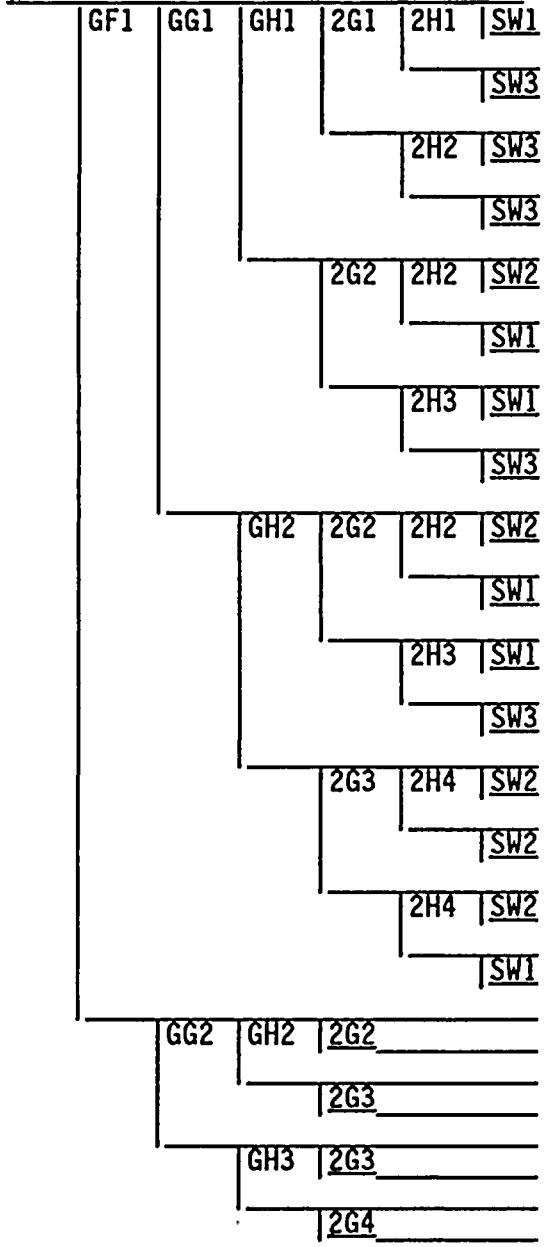
The split fraction value ASW used in the non-seismic calculation of non-LOOP initiated event sequences was generated as part of this calculation. It represents ASW system unavailablity given that a LOOP event has not occurred and that power is available to both Unit 1 ASW pumps and is based on the one-out-of-two pump success criteria. This split fraction is used in non-LOOP sequences only.

In the seismic calculation, only the LOOP sequences were re-evaluated; they were quantified six times corresponding to the six seismic ground acceleration levels. The results of the six seismic quantifications were then summed to calculate the total increase for seismic sequences. The values for the LOOP initiator and split fractions PR1 and PRD are dependent on the ground acceleration level. The LOOP initiator frequency was obtained from the unconditional frequency of seismic loss of offsite power. Split fractions PR1 and PRD used in this calculation include the values of PR1 and PRD presented in DCPRA for seismic initiating event and the conditional seismic failure fraction of reactor coolant pumps. No recovery actions were considered for the calculation of seismic sequences. Seismic failures of diesel generators and the ASW system were included in the DCPRA and are not affected by the revised success criteria of ASW.

Table 1 is a listing of the sequences that were quantified for non-seismic events. Also included in Table 1. is the total contribution of seismic events to the increase in core damage frequency. Table 2 gives the data used for the non-seismic quantification. Table 3 lists the split fraction valves used for the seismic calculation. The seismic calculation use the same LOOP sequences as those used in the non-seismic calculation, except that recovery actions were not considered. The results of seismic calculation are shown in Table 4.

FIGURE 1

EVENT TREE FOR LOOP INITIATING EVENT



ASW S.F. 1/2	ASW S.F. DCPRA	Top Event PR S.F.	Included in DCPRA	Seq No.
A	6	1		1
C	5	1		2
A	6	1		3
C	5	1		4
A	A	1	YES	
C	8	1		5
A	A	1	YES	
C	8	1		6
A	6	1		7
C	5	1		8
A	6	D		9
C	5	D		10
A	A	D	YES	
C	8	D		11
A	A	D	YES	
C	8	D		12
C	9	D		13
F	7	D		14
C	9	D		15
F	7	D		16
C	C	D	YES	
F	B	D		17
C	C		YES	
F	B	D		18
C	9	D		19
F	7	D		20
C	9	D		21
F	7	D		22
C	C		YES	
F	B	D		23
C	C		YES	
F	B	D		24
C	8	1		25
C	C		YES	
C	8	D		26
C	C		YES	
F	B	D		27
F	F		YES	
F	B	D		28
F	F		YES	

TABLE 1

SEQUENCES QUANTIFIED TO EVALUATE THE REVISED ASW SUCCESS CRITERIA

RESULTS SUMMARY

0.93	PERCENT INCREASE IN TOTAL CDF
1.86e-6	TOTAL OF SEISMIC AND NON-SEISMIC
1.29e-6	NON-SEISMIC
5.69e-7	SEISMIC

LOOP SEQUENCES

1	3.34e-9	(LOOP*GF1S*GG1S*GH1S*TG1S*TH1S*SW1S*ASA*PR1)
2	7.22e-9	(LOOP*GF1S*GG1S*GH1S*TG1S*TH1S*SW1*ASC*PR1)*RXBUS2
3	1.83e-12	(LOOP*GF1S*GG1S*GH1S*TG1S*TH1*SW3S*ASA*PR1)
4	6.54e-10	(LOOP*GF1S*GG1S*GH1S*TG1S*TH1*SW3*ASC*PR1)*RXBUS2
5	6.54e-10	(LOOP*GF1S*GG1S*GH1S*TG1*TH2S*SW3*ASC*PR1)*RXBUS2
6	9.93e-10	(LOOP*GF1S*GG1S*GH1S*TG1*TH2*SW3*ASC*PR1)
7	8.88e-10	(LOOP*GF1S*GG1S*GH1*TG2S*TH2S*SW2S*ASA*PRD)*RXBUS1
8	4.38e-9	(LOOP*GF1S*GG1S*GH1*TG2S*TH2S*SW2*ASC*PRD)
9	2.50e-11	(LOOP*GF1S*GG1S*GH1*TG2S*TH2*SW1S*ASA*PRD)*RXBUS1
10	1.79e-9	(LOOP*GF1S*GG1S*GH1*TG2S*TH2*SW1*ASC*PRD)*RXBUS2
11	1.79e-9	(LOOP*GF1S*GG1S*GH1*TG2*TH3S*SW1*ASC*PRD)*RXBUS2
12	6.37e-9	(LOOP*GF1S*GG1S*GH1*TG2*TH3*SW3*ASC*PRD)
13	6.35e-8	(LOOP*GF1S*GG1*GH2S*TG2S*TH2S*SW2S*ASC*PRD)*RXBUS2
14	8.16e-9	(LOOP*GF1S*GG1*GH2S*TG2S*TH2S*SW2*PRD)*RXBUS3
15	1.79e-9	(LOOP*GF1S*GG1*GH2S*TG2S*TH2*SW1S*ASC*PRD)*RXBUS2
16	9.03e-8	(LOOP*GF1S*GG1*GH2S*TG2S*TH2*SW1*PRD)*RXBUS3
17	9.02e-8	(LOOP*GF1S*GG1*GH2S*TG2*TH3S*SW1*PRD)*RXBUS3
18	1.19e-8	(LOOP*GF1S*GG1*GH2S*TG2*TH3*SW3*PRD)*RXBUS3
19	3.57e-9	(LOOP*GF1S*GG1*GH2*TG3S*TH3S*SW2S*ASC*PRD)*RXBUS2
20	1.75e-8	(LOOP*GF1S*GG1*GH2*TG3S*TH3S*SW2*PRD)*RXBUS4
21	6.39e-9	(LOOP*GF1S*GG1*GH2*TG3S*TH3*SW2S*ASC*PRD)
22	1.16e-9	(LOOP*GF1S*GG1*GH2*TG3S*TH3*SW2*PRD)*RXBUS4
23	1.52e-9	(LOOP*GF1S*GG1*GH2*TG3*TH4S*SW2*PRD)
24	2.22e-8	(LOOP*GF1S*GG1*GH2*TG3*TH4*SW1*PRD)
25	6.92e-10	(LOOP*GF1*GG2S*GH2S*TG2S*ASC*PR1)*RXBUS2
26	3.80e-9	(LOOP*GF1*GG2S*GH2*TG3S*ASC*PRD)*RXBUS2
27	1.92e-7	(LOOP*GF1*GG2*GH3S*TG3S*PRD)*RXBUS3
28	6.41e-8	(LOOP*GF1*GG2*GH3*TG4S*PRD)*RESLC4

NON-LOOP SEQUENCES

29	2.29e-8	(LLOCA*ASW)
30	5.25e-8	(MLOCA*ASW)
31	5.96e-7	(SLOCN*ASW)
32	1.40e-8	(SLOCI*PRN*ASW)
33	5.73e-10	(SLBI*SE1*ASW + SLBI*PR5*ASW)
34	1.83e-10	(ISI*PR5*ASW)
35	1.25e-9	(CPEXC*PR1*ASW)
36	4.30e-11	(AMSIV*PR5*ASW)
37	1.37e-11	(SLBO*PR5*ASW)
38	8.87e-12	(MSRV*PR5*ASW)
39	2.64e-10	(IMSIV*PR5*ASW)

TABLE 2

DATA USED IN THE SEQUENCE QUANTIFICATION FOR NON-SEISMIC EVENT

ASW	1.13e-4
RXBUS4	7.60e-1(RELOOP+ASA+PR1/PRD)
RXBUS3	2.00e-2(ZHERE6+ASC*PR1/PRD)*(RELOOP+ASA)
RXBUS1	3.02e-2(ZHERE6*RELOOP+PR1/PRD)
RXBUS2	3.70e-2(ZHERE6*RELOOP+(ASA/ASC))
RELOOP	7.5e-1
ZHERE6	2.65e-2(5*ZHERE2)
ZHERE2	5.3e-3
RESLC4	9.78e-2(AFWTP*RESLC1+(1-AFWTP)*RESLC2)
AFWTP	1.4e-1
TG4S	7.1e-1
SW2S	9.97e-1
SW3S	6e-3
RP2S	4e-3(RP2S=1-RP2)
RF4S	9.45e-1
CI1S	9.96e-1
SI1S	9.97e-1
SW1S	5e-1
OG1S	9.99e-1
SA1S	9.92e-1
SB1S	9.93e-1
GF1S	9.55e-1
GG1S	9.55e-1
GH1S	9.56e-1
TG1S	9.56e-1
TH1S	9.56e-1
GG2S	9.44e-1
TG3S	9.38e-1
TH3S	9.38e-1
GH3S	9.17e-1
TH4S	9.31e-1
TH2S	9.47e-1
TG2S	9.46e-1
GH2S	9.46e-1
REOB1	6.66e-2(REOB1 = (OB1+RF1+LA1+CH2)
RSEQ8	3.45e-3(RSEQ8 = ZHEFO6*RESLC3)
RSEQ10	3.53e-2(RSEQ10= (ZHESW1+AW4)*RESLC1))
RSEQ24	5.75e-2(RSEQ24= (OB1+LA1+RF1+CH1+VA1))
RSEQ25	3.45e-3(RSEQ25= ZHEFO6*RESLC3)
RSEQ34	4.58e-4(RSEQ34= ZHERE2*RESLC3)

TABLE 3

SEISMIC SPLIT FRACTION VALUES

	SEIS1	SEIS2	SEIS3	SEIS4	SEIS5	SEIS6
PR1	2.57e-4	1.37e-3	5.34e-2	6.24e-2	9.05e-2	1.55e-1
PRD	4.8e-2	4.95e-2	5.34e-2	6.23e-2	9.05e-2	1.55e-1
LOOP	1.18e-4	2.52e-4	9.06e-5	9.29e-5	2.62e-5	7.29e-6
SW2	1.75e-3	1.75e-3	1e-2	1e-2	5e-2	5e-2
SW3	9.98e-1	9.98e-1	9.9e-1	9.9e-1	9.5e-1	9.5e-1
SW2S	9.98e-1	9.98e-1	9.9e-1	9.9e-1	9.5e-1	9.5e-1
SW3S	1.8e-3	1.8e-3	1e-2	1e-2	5e-2	5e-2

TABLE 4

RESULTS OF SEQUENCE QUANTIFICATION FOR SEISMIC EVENTS

TOTAL	SEIS1	SEIS2	SEIS3	SEIS4	SEIS5	SEIS6
5.69e-7	8.32e-8	1.85e-7	9.99e-8	1.20e-7	5.49e-8	2.62e-8
SEQ						
1	1.80e-12	2.04e-11	2.87e-10	3.43e-10	1.40e-10	6.70e-11
2	1.05e-10	1.19e-9	1.68e-8	2.01e-8	8.21e-9	3.92e-9
3	5.75e-16	6.53e-15	5.10e-13	6.11e-13	1.25e-12	5.96e-13
4	1.87e-11	2.12e-10	2.95e-9	3.54e-9	1.39e-9	6.62e-10
5	1.87e-11	2.12e-10	2.95e-9	3.54e-9	1.39e-9	6.62e-10
6	1.88e-12	2.13e-11	2.98e-10	3.57e-10	1.40e-10	6.68e-11
7	5.96e-11	1.31e-10	5.04e-11	6.04e-11	2.37e-11	1.13e-11
8	6.11e-12	1.34e-11	2.98e-11	3.57e-11	7.30e-11	3.48e-11
9	3.01e-12	6.62e-12	2.57e-12	3.08e-12	1.26e-12	6.01e-13
10	1.76e-10	3.87e-10	1.50e-10	1.80e-10	7.36e-11	3.51e-11
11	1.76e-10	3.87e-10	1.50e-10	1.80e-10	7.36e-11	3.51e-11
12	3.93e-11	8.64e-11	3.32e-11	3.98e-11	1.56e-11	7.46e-12
13	3.48e-9	7.66e-9	2.95e-9	3.53e-9	1.39e-9	6.62e-10
14	5.71e-10	1.26e-9	2.78e-9	3.33e-9	6.82e-9	3.26e-9
15	1.76e-10	3.87e-10	1.50e-10	1.80e-10	7.36e-11	3.51e-11
16	1.64e-8	3.62e-8	1.40e-8	1.68e-8	6.88e-9	3.28e-9
17	1.64e-8	3.62e-8	1.40e-8	1.68e-8	6.88e-9	3.28e-9
18	3.67e-9	8.07e-9	3.10e-9	3.72e-9	1.46e-9	6.97e-10
19	3.51e-10	7.73e-10	2.97e-10	3.56e-10	1.40e-10	6.68e-11
20	5.75e-11	1.27e-10	2.81e-10	3.36e-10	6.88e-10	3.28e-10
21	3.92e-11	8.64e-11	3.32e-11	3.98e-11	1.56e-11	7.46e-12
22	6.43e-12	1.41e-11	3.14e-11	3.76e-11	7.69e-11	3.67e-11
23	6.43e-12	1.42e-11	3.14e-11	3.76e-11	7.69e-11	3.67e-11
24	2.30e-10	5.06e-10	1.96e-10	2.35e-10	9.62e-11	4.59e-11
25	2.06e-11	2.33e-10	3.28e-9	3.93e-9	1.61e-9	7.68e-10
26	3.91e-10	8.61e-10	3.34e-10	4.00e-10	1.64e-10	7.81e-11
27	3.66e-8	8.05e-8	3.12e-8	3.74e-8	1.53e-8	7.30e-9
28	4.13e-9	9.10e-9	3.53e-9	4.23e-9	1.73e-9	8.26e-10

ITEM 9:

Two split fractions for the Auxiliary Feedwater (AW7 and AW8) system were revised in the uncertainty analysis of the non-seismic dominant sequences. PG&E should provide these values to BNL along with a discussion on how these values were determined. Also, PG&E should consider NRC Information Notice No. 89-58 in regard to whether the PRA model addresses this issue.

RESPONSE TO ITEM 9:

The split fraction values which should be used in the dominant sequence model for AW7 and AW8 are 3.238E-4 and 1.225E-3 respectively. The following discussion describes how these values were determined.

In the original auxiliary feedwater model the following conservative modeling assumptions were made:

1. A support system condition involving failure of one instrument channel was modeled as if both instrument channels that supply the steam generator 10% atmospheric steam dumps are unavailable.

2. No credit was taken for aligning the backup regulated transformer to the failed instrument channel so as to power the 10% steam dumps.

3. Given the unavailability of the 10% steam dumps, the model assumed 20 safety valve challenges and that all 5 safety valves may lift for each challenge. Each safety valve is required to reclose after each challenge.

To reduce the number of minimum cut-sets in the AFW system equation file, certain low frequency cut-sets were removed based on their relative importance; this was done using a point estimate quantification. These cut-sets were related to split fractions AW7 and AW8. Mean values of the split fractions AW7 and AW8, obtained from a monte carlo calculation using the truncated equations, were used in the event tree quantification. These values are presented in column 2 of the table below.

During BNL's review of the AFW system model, BNL identified inconsistencies in the values of the AFW system split fractions. PG&E determined that these inconsistencies were due to the truncated cut-sets. Some of the cut-sets became significant contributors to the AFW system unavailability when the model was quantified using monte carlo quantification because of the products of correlated variables. To reslove the inconsistencies, the cut-sets were restored to the equation file and the equation model requantified. The results are shown in column 3 of the table below. The AFW split fraction values used in the DCPRA event tree model quantification and hence, in the development of the DCPRA dominant sequence model were, however, adequate for determining the important core damage sequences associated with the unavailability of the AFW system. This is because the values of AW7 and AW8 used to develop the dominant sequence model are higher than the revised values of AW7 and AW8 which were used for the uncertainity analysis of the dominant sequences.

The results of event tree quantification revealed that split fractions AW7 and AW8 were in a number of relatively high frequency core damage sequences, and that the conservative assumptions in the original AFW system model contributed to their importance. These sequences involved the failure of one instrument bus which was assumed to fail all four 10% steam dump valves; in actuality, it takes the failure of two instrument buses to render all four 10% steam dumps unavailable. A more realistic analysis of the AFW system model was then performed by assuming that all safety valve failures to re-close were negligible for split fractions AW7 and AW8. This assumption is reasonable because loss of one instrument channel bus does not disable all of the 10% steam dumps, and with the availability of the 10% steam dumps the steam generator safety valves would not be challenged.

The re-analysis involved the revision of an equation in the AFW system model equation file (see Figure E.2-16 of the DCPRA report, Sheet 16) which contained the variable "C." The variable "C" is the failure probability of one or more of the five safety valves on a single steam generator to re-close successfully in each of the 20 cycles. "C" was set equal to zero in the AFW system model equation file, implying that all safety valve failures to re-close were negligible. This resulted in the mean values for AW7 and AW8 as shown in column 4 of the table below: note, the values in Column 5 were used for the uncertainty analysis of the dominant sequence model.

SPLIT FRACTION VALUES FOR AW7 AND AW8 (1)

	Used In (2) Event Tree Quantification (EFT2 Terms Truncated)	EFT2 Terms Restored to AFW Equation File		(5) Used in Dominant Sequence Model
		Variable (3) C not zero	Variable (4) C set to zero	
AW7	6.269-3	3.499-2	3.238-4	3.238-4
AW8	7.759-3	5.318-2	1.225-3	1.225-3

Notes to the table:

(1) Monte Carlo Mean Values.

(2) Computer File PGE.1123>EVENT.TREES>INTERNALS>MFF.RM3
Also see Appendix J, Table J-8.

(3) Computer File PGE.1123>IBM.SYSTEMS>ADDENDUMS>AFW1004M.CTS

(4) Computer File PGE.1123>IBM.SYSTEMS>ADDENDUMS>AFW1006M.CTS

(5) Computer File PGE.1123>EVENT.TREES>INTERNALS>RMODEL>DBF.RM3.SAVE
Also see Appendix J, Table J-3.

The resulting distributions for AFW split fractions AW7 and AW8 from the re-analysis of the AFW system model were used in the uncertainty analysis of the non-seismic dominant sequences. The characteristics of these distributions are provided below (5). These are the values which should be used in the dominant sequence model.

	Mean	5th	50th	95th
AW7	3.238-4	8.715-5	2.234-4	6.907-4
AW8	1.225-3	3.861-4	9.393-4	2.372-3

NRC Information Notice No. 89-58 (1) describes an event that occurred at Diablo Canyon during power operation involving isolation of one of the two steam supply paths to the turbine-drive AFW pump (for maintenance) coincident with removal of a motor-driven AFW pump from service. In the post TMI re-analysis of main feedwater line break for Diablo Canyon, it was assumed that AFW would be supplied to two steam generators in order to prevent the pressurizer from filling with liquid coolant. With one of the two steam supply valves to the AFW pump turbine isolated, only one steam generator may be supplied with AFW if the break occurs on the line feeding the steam generator that provides the remaining steam supply to the turbine-driven AFW pump and if the inoperable motor-driven AFW pump is associated with two steam generators with unbroken feedwater lines. Because of this, the NRC recommended that the turbine-driven AFW pump should be considered inoperable at Diablo Canyon (with respect to the emergency requirements) when one of the two steam paths is isolated.

From a system reliability standpoint, however, the turbine-driven AFW pump is still available even with only one operable steam supply path, since its design function to provide cooling water to all four steam generators can still be accomplished. An additional failure in the remaining steam supply path is required before the turbine-driven AFW pump is rendered unavailable. With the consideration of the joint frequency of maintenance-related isolation of one steam supply path and an additional failure in the remianing path, it was judged in DCPRA that this combined frequency is insignificant compared to the unavailability of turbine-driven AFW pump due to other causes. Maintenance of the steam supply valve(s) was, therefore, not modeled explicitly in DCPRA.

Nonetheless, in the DCPRA any maintenance event contributing to the unavailability of an AFW pump (turbine-driven or motor-driven) train was grouped together with the maintenance events for the corresponding AFW pump. The unavailability of an AFW pump train due to maintenance activities was then calculated based on the frequency and duration of the maintenance events on its respective pump. As such, unavailability of a motor-driven AFW pump due to maintenance (including events related to maintenance on steam generator level control valves as discussed in Information Notice 89-58) was modeled explicitly in the AFW system analysis of DCPRA. The mean frequency and duration of maintenance on a motor-driven AFW pump were estimated in DCPRA to be 5.53E-4 per hour and 21 hours respectively. The mean maintenance unavailability of a motor-driven AFW pump is thus approximately 1.16E-2.

The isolation of one AFW pump turbine steam supply valve described in Information Notice 89-58 is the only occurrence of AFW steam supply isolation at Diablo Canyon since commercial operation. During that event, the steam supply valve was isolated for about 49 hours. Assuming the average frequency and duration of maintenance on steam supply valve are once every three years and 49 hours respectively, the estimated mean unavailability of the AFW pump turbine steam supply valve due to maintenance would be approximately 1.9E-3. Even without considering the frequency of an additional failure in the remaining steam supply path, this is only about 2.5% of the total contribution to the unavailability of the turbine-driven AFW pump train due to other causes, which is approximately 7.3E-2. The impact of the steam supply valve isolation event on the unavailability of the AFW turbine pump train is therefore not significant.

If the unavailability of a motor-driven AFW pump due to maintenance is also considered in conjunction with the isolation of an AFW pump turbine steam supply valve, the combined unavailability is about 2.2E-5. This is less than 2% of the unavailability of one turbine-driven and one motor-driven AFW pump due to all causes, which is 1.2E-3. If the frequency of the additional failure in the remaining AFW turbine steam supply path is also accounted for, the combined frequency should be much less than 1% of the unavailability of two pumps. After the January, 1989 event, PG&E has revised the procedures to make it clear that the turbine-driven pump should be declared inoperable when one AFW pump turbine steam supplies is isolated. If a motor-driven AFW pump also becomes unavailable during the period when one steam supply is isolated, the unit will be shutdown after six hours. This will prevent the recurrence of the January, 1986 event and further reduce its contribution to system unavailability.

REFERENCES:

1. NRC Information Notice No. 89-58: Disablement of Turbine-Driven Auxiliary Feedwater Pump Due to Closure of one of the Parallel Steam Supply Valves; August 3, 1989.

ITEM 10:

BNL requested a summary discussion of top event HS (split fractions HS1, HS2, and HSF) regarding its relative importance to core damage frequency: in particular, if this human error were not considered, how would this change the core damage frequency?

RESPONSE TO ITEM 10:

Top event HS models whether hot standby conditions can be maintained successfully. If Top Event HS fails, it is assumed that long-term actions to control AFW, provide AFW makeup, or establish closed loop RHR cooling have failed and core damage results. Because there is plenty of time for a successful operator response if something does go wrong in the long term after conditions have stabilized, the failure frequency for this event is expected to be very low. Plant operators may decide to cooldown to cold shutdown conditions or to return to power after the cause of the plant trip is identified and resolved. The DCPRA model assumes that they would only attempt to cool down to cold shutdown if a LOCA is in progress. If a LOCA is not in progress, the plant would return to power from hot standby.

Split fraction HS1 is used for stable sequences in which no LOCA has occurred. Split fraction HS2 is used for conditions in which the plant is stable but a small LOCA has occurred. A lower human error rate is used for split fraction HS2 (human action ZHEHS3) than for split fraction HS1, (human action ZHEHS1) because it is believed that the plant operators would be more cautious after responding to a LOCA.

Top event HS is also used as a switch in the frontline event tree models to indicate that core damage may occur if all plant indications are not available following a plant trip, for example, loss of switchgear ventilation resulting in failure of instrument AC; for this case, the split fraction used is HSF.

If all vital instrumentation power is lost as a result of a failure of the 480V switchgear ventilation system (top event SV), seismic failure of the inverters, or a prolonged station blackout, failure of Top Event HS and eventual core damage is assumed. No credit for successfully shutting down without vital instrumentation is given. For this condition, split fraction HSF is used, which has a value of 1.0.

The contribution of failures of top event HS to the non-seismic core damage frequency can be computed from the non-seismic dominant sequence model. The contribution from split fractions HS1 and HS2 can be computed by simply setting their values to 0.0 and calculating the change in core damage frequency. Setting split fraction HS1 to 0.0 reduces the core damage frequency by about 1.9E-5 per year. Split fraction HS2 does not appear in the dominant sequence model hence its contribution is negligable.

Since HSF is a guaranteed failure split fraction and since all guaranteed failed split fractions were suppressed from the sequences for the late event trees, HSF does not appear in the dominant sequence model. The contribution to the core damage frequency for sequences involving split fraction HSF, in which there is a loss of all vital instrument power due to a failure of the switchgear ventilation system, can be seen by setting the

split fraction values for top event SV to 0.0. The reduction in core damage frequency is then found to be 5.4E-6 per year.

Top event HS was also modeled for seismic events. The contribution from HS1 was less than 1% of the total seismically initiated core damage frequency. The contribution from HS2 and from losses of switchgear ventilation were found to be negligible.

ITEM 11

BNL developed a table of conditional core damage frequency (CCDF) for DCPP and other plants. BNL requested PG&E to provide insights regarding the comparison between plants in terms of areas where DCPP is better and areas where DCPP is worse than the other plants.

RESPONSE TO ITEM 11:

BNL presented in Tables 3.9.2a and 3.9.2b of its April 3, 1990 presentation to the NRC, that the conditional core damage frequency computed in the Diablo Canyon study for various initiator groups differ from other PWR plants whose PRA results have been presented previously. Such comparisons are difficult to interpret since, within the group of plants referenced, no two such plants are alike. The following paragraphs present some thoughts on why the Diablo Canyon results may be higher or lower in comparison, although it must be recognized that such differences may be attributable to plant hardware differences and success criteria, which are not immediately apparent without a detailed investigation of the other PRA studies. Often the controlling risk features are related to support system design rather than NSSS vendor.

With regard to the group BNL titles as transients, it is worth noting that this group includes many initiator categories, including loss of offsite power and other support system faults, such as loss of a DC bus. Therefore, such comparisons between studies for different plants is really only meaningful if these other studies considered a similar breakdown of the initiators. For example, it is known that the DCPRA considered both loss of ASW and CCW as initiating events, and as systems which must respond to other initiators. However, the Oconee study did not look at equipment failures intrinsic to the CCW system when evaluating the risks from other initiators. Such equipment failures often have a high conditional core damage frequency.

Three modeling considerations stand out as to why the DCPRA conditional core damage frequency contributed by transients may be higher than for other plants. One is the thorough treatment of common cause events included in the DCPRA system models; the DCPRA models common cause more thoroughly than in earlier, older PRA studies. Secondly, the DCPRA plant model included a simplistic model for failure to maintain hot standby (top event HS) conditions following plant trip in which no LOCA occurs. Such considerations are often neglected in other studies. If this event's contribution is set to zero in the dominant sequence model the core damage frequency decreases by 1.9E-5 per year, or about 16% of the total transient contribution.

A third consideration has to do with the explicit modeling of vital instrument buses in the DCPRA. Many of the older studies which are being compared to the DCPRA did not consider failures of instrument buses. Additionally, given loss of a DC bus at Diablo Canyon , the associated emergency AC bus is assumed unable to transfer to the standby power source after a plant trip regardless of the timing of DC failure. Therefore, loss of a single DC bus can fail both sources of power to an instrument bus. For example, failure of top event DG causes failure of top event I2 and failure of top event DH fails top event I3. The DCPRA model conservatively took no

credit for recovery of the instrument bus by crosstying to an alternate AC supply given both its normal AC and DC power supplies are lost.

If control power to all four 10% atmospheric steam dumps is assumed lost, as was conservatively assumed for loss of an instrument bus, the steam generator safety valves are repeatedly challenged to open and re-close. If safety valves on all four steam generators subsequently fail open, the original AFW system analysis model assumed that the operators would isolate each faulted steam generator causing the loss of all steam generator cooling. This conservative assumption was relaxed for boundary conditions AW7 and AW8 but not for AW5. When this conservatism is also removed to account for the fact that the operators, by procedure, would not isolate the last steam generator (i.e. the new AW5 mean value is 3.4E-5), the core damage frequency would be reduced by 9.4E-6.

For small LOCAs, BNL's comparison indicates that the conditional core damage frequency results for Diablo Canyon are generally lower than those for other PWRs. Except, however, the results for Zion (revised) and for Millstone 3 which are very similar. The small LOCA results may be lower than other studies for two reasons. First, the DCPRA treats the small LOCA category of initiators in two categories: isolable and non-isolable LOCAs. For the isolable LOCAs, credit is then taken for closing the pressurizer PORV block valve on the leaking PORV train. For the case when all support systems are available, this action is accounted for in split fraction PRN.

A smaller proportion of the small LOCA frequency is attributed to non-isolable small LOCAs. For this category, no credit is taken for isolation of the break. Diablo Canyon has four high pressure injection pumps to provide injection for small LOCAs. Consequently, for small LOCAs, the risk is dominated by the two trains available for recirculation from the containment sump. In the event of a failure of recirculation from the containment sump, emergency procedures direct the operators to throttle back on injection flow to preserve RWST inventory. Injection flow is throttled back sufficiently to maintain RCS inventory while removing decay heat by steaming. The operators then provide borated makeup water to the RWST for continued high pressure injection. The DCPRA takes limited credit for this makeup to the RWST, as an alternative to recirculation from the sump. Consequently, the highest frequency small LOCA sequences are likely to be lower than in studies for other PWRs where no credit for this action was given.

The BNL comparative results for medium LOCAs indicate that the Diablo Canyon conditional core damage frequency for medium LOCAs is higher than for any of the other PWR studies, by a factor ranging from 1.4 to 6. The reasons for this may be inferred by a review of the top medium LOCA sequences contributing to core damage. The highest frequency sequence involves a failure of the operators to align for recirculation from the containment sump, given that containment spray successfully actuates to limit containment pressure; this failure is driven by the human error rate. That the error rate used in the DCPRA may differ substantially from other studies is a reflection of the uncertainty in assigning error rates to well proceduralized actions, which have ample time available, but which are being performed under stressful conditions. This sequence accounts for about one-third of the total core damage frequency from medium LOCAs.

Therefore, the difference between studies may be due in part to the human errors rates assigned.

The second highest medium LOCA core damage sequence involves failure of both RHR pumps, preventing recirculation from the sump. For this size LOCA, no credit was given for making up to the RWST and continuing injection using the high pressure pumps. The failure of both RHR pumps may be higher than for studies of other plants because of the thorough treatment of common cause failures in the Diablo Canyon study. Two other reasons tend to make the results higher for Diablo Canyon. One is that the RHR pumps are required to start twice for medium LOCAs; i.e. during the initial actuation on an SI signal, and then again later when aligning for recirculation. Both pump starts were included in the system model for the RHR pumps. Common cause failure of the pumps to start was considered for both pump starts. Secondly, the operators are required by procedure to turn off the RHR pumps after it is determined that RCS pressure is above the RHR injection head. At Diablo Canyon, when an SI signal occurs but RCS pressure is high, the RHR pumps run on miniflow where pump discharge is directed back to the pump suction. CCW cooling is not aligned to the RHR heat exchangers during this time. Therefore, it is assumed that the RHR pumps must be tripped within a reasonable time to prevent overheating. The DCPRA plant model assumes that for medium LOCAs, RCS pressure would remain above the RHR pump shutoff head long enough to require that the pumps be tripped to prevent overheating. Failure of the operators to trip the pumps within about four hours is modeled as a failure of both RHR pumps. This action is included in the RHR system model.

The third highest medium LOCA core damage sequence is associated with vessel failure due to PTS. Most other studies have neglected contributions to core damage caused by PTS.

The above paragraphs provide some insights as to why the conditional core damage frequency from medium LOCAs may be higher than studies for some other plants.

The BNL comparative results for the conditional core damage frequency initiated by large LOCAs also shows that the results are higher for Diablo Canyon by a factor of 1.4 to 9. The highest frequency large LOCA initiated core damage sequence in the DCPRA is due to failure of the accumulators. Three-out-of-three accumulators were assumed required to prevent core damage, with the fourth accumulator inventory assumed to be diverted out the break. This sequence accounts for more than one-third of the total large LOCA contribution to core damage. Other PRAs may have used a less stringent success criteria.

A second large LOCA core damage sequence involves failure of the operators to align for recirculation from the sump. As with the above discussion for medium LOCAs, the human error rate assigned to this action in the DCPRA may be higher than that assumed in other studies. The higher error rate is not attributed to any specific design features at Diablo Canyon, but rather, the differences simply reflect the uncertainty in assigning error rates.

ITEM 12:

It was requested that PG&E develop a list of conservatisms in the DCPRA.

RESPONSE TO ITEM 12:

The following is a list of some of the more significant conservatisms in the DCPRA.

1. The general transient event tree includes top event HS. This event attempts to account for the small chance that core damage could occur even if the plant is successfully brought to hot shutdown with no LOCA present. A detailed analysis of this time period was not performed. Rather, a judgmental factor was used to quantify the likelihood of such events. Most other PRAs have not attempted to quantify such events. Some have modeled in detail the risks from shutdown events (i.e. beginning with the plant on closed loop RHR), but such events were explicitly excluded from consideration in this assessment of Diablo Canyon. It is therefore probably appropriate to eliminate such failures for consideration, particularly when comparing the results against other plants. The contribution of this event to core damage was discussed in the response to Item 10.

2. Some core damage sequences involve the failure of one or two emergency AC power trains and of a pressurizer PORV which has opened but failed to re-close. The model conservatively assumes that the PORV which failed open is associated with the PORV block valve which has lost power due to loss of an emergency AC train. In actuality, if only one emergency AC train is lost (buses HG or HH), the chance is only one in two that the de-energized block valve is associated with the failed open PORV. Even if the de-energized block valve coincides with the failed open PORV, the operators could crosstie emergency power so that the block valve can be closed. No credit was given for the action to crosstie emergency power.

   The impact of this conservative assumption has been estimated by requantifying the dominant sequence model for internal events. When the value for split fraction PRD (which represents the case of one emergency AC power train unavailable) is weighted by the chance that the AC power train lost is associated with the stuck open PORV, the core damage frequency decreases by approximately 3.9E-6 per year.

3. Some sequences involve failure of the turbine-driven AFW pump given DC power for its steam supply valves is initially available. In such cases, about half of the failure-to-start frequency could be easily recovered by simply re-setting the pump trip and re-starting the pump. Credit for such actions has been assumed in other PRAs but was not credited in the DCPRA.

4. To simplify the AFW analysis, failure of one instrument bus was conservatively modeled as failure of instrument power to all four of the steam generator's 10% steam dump valves; in actuality, multiple instrument bus failures would be required. The auxiliary feedwater model then conservatively assumed that if a steam generator safety valve failed open on each of the four steam generators, the operators would

erroneously isolate flow to all four steam generators. Since the procedures caution the operators against isolating the last steam generator available, this assumption is too conservative. For split fractions AW7 and AW8 this assumption was relaxed. However, for split fractions AW5, AW6, AW9 and AWA the results are still conservative. Of the split fractions AW5, AW6, AW9, and AWA the most important one is AW5; in the response to Item 11, it was discussed that when the conservatism for AW5 is reduced, (yielding a new value for AW5 of 3.4E-5) the core damage frequency is reduced by 9.4E-6.

5. For the control room ventilation system model, the human action considered in the system model includes manual switching of the power source to the backup power for the normally operating train or switching to the standby train. However, on failure of the normal HVAC system, no other recovery actions such as operators opening the control room door and installing portable blowers to circulate the air in the control room were considered. This additional recovery action would reduce the conservatism in the model developed for the loss of Control Room ventilation initiating event. The unavailability of the control room ventilation system could be reduced by a factor of 10 based on the human error rate of 0.1 in establishing alternative ventilation for the control room.

6. In the ASW system model, failure of the ASW pump room exhaust fan is modeled as failure of the corresponding pump. However, room heatup calculations show that these rooms will not reach the thermal fragility limit of the ASW pumps within the PRA mission time.

7. When accounting for sequence-specific recovery actions, often the same sequence may be recovered by a variety of different strategies. In such cases, only one recovery strategy was quantified; the alternative strategies were neglected. In many cases, the neglected strategy was that of crosstying two emergency buses. For example, loss of a DC bus was modeled as failing the associated AC bus. However, the associated bus could be recovered by crosstying to an operable AC bus. No credit was given for such actions even though they are proceduralized. Another example is that of recovery from loss of switchgear ventilation. In some sequences, switchgear ventilation fails partly due to a loss of offsite power and failure of one emergency AC power train. No credit was given in such sequences for recovery of offsite power and restart of the second ventilation train.

8. For large LOCAs (up to the design basis LOCA), it is assumed that three -out-of-three accumulators must inject to prevent core damage. The fourth accumulator's inventory is assumed to go out the break. This assumption is believed to be conservative; a less stringent success criteria could be used however, no analysis has been performed to demonstrate this.

9. In modeling the SSPS system for each initiating event, if any required SSPS function fails, all the necessary functions are assumed lost. This conservatism also applies to a single SSPS train; if one SSPS function fails on train A, then all Train A actuation signals are assumed lost. Additionally, no credit was given for master relays with redundant backups.

10. The SPSS model for general transient initiating events is conservative in that it includes all safety activation signals that might be needed in a general transient event even though only a few may be needed depending on the transient and the progression of events after the transient.

11. Recirculation tests for the AFW pumps were assumed to make the pumps unavailable during the tests. This is conservative, since the operators can realign the pumps very quickly.

12. For the loss of DC bus initiating event, failure of any one of the three DC buses as an initiating event was conservatively modeled as failure of the bus with the most severe impact; loss of DC bus 12 was judged to be more severe than the impact associated with failure of either of the other two busses.

13. Pressurized thermal shock of the RCS pressure boundary was included in the DCPRA for medium LOCAs, bleed and feed cooling scenarios, and for sequences involving depressurization of all four steam generators. The probabilities for vessel failure (which lead to an unmitigable LOCA) were derived assuming end-of-life vessel conditions and that there is a pre-existing crack for the start of crack propagation. These assumptions are believed to be very conservative, although the degree of conservatism is difficult to quantify.

14. The diesel fuel transfer system has redundant level controls; to simplify the model, only one valve is modeled.

15. When a diesel generator is removed from service to perform unscheduled maintenance, the DCPP technical specifications require that the remaining operable diesel generators, of the same unit, be demonstrated operable. If diesel generator 1-3 is removed from service, then all the remaining diesels, of both units must be tested. In the modeling of the contribution to diesel generator unavailability due to operator error in not returning the control of the diesel generator or the fuel transfer system to the normal positions after a maintenance-related surveillance test, the detection time for these errors may actually extend beyond the time when the diesel generator in maintenance is restored to operable status. During this fraction of the time, only one diesel generator is actually unavailable; this duration, however, was conservatively treated as contributing to the unavailability of two diesel generators. Further, some surveillance tests do not require the diesel generator control to be changed; for these tests, the diesel generator is not unavailable and the human error to restore the control is not applicable. It was also conservatively assumed that when one diesel is in maintenance, then all diesels will be demonstrated operable by testing. Testing of all diesels is only required when diesel generator 13 is in maintenance. This assumption preserves the symmetry of the five diesels and thus simplifies the modeling of the diesel generators.

16. The seismic analysis treats the failure frequencies of redundant, identical equipment conservatively. A fragility curve was developed for each piece of equipment. If one piece of equipment fails seismically, then the other identical, redundant equipment is also assumed to fail.

For example, a single fragility curve was used to represent the failure of all five diesel generators. However, in reality, the degree of dependence between equipment may be less than the complete dependence modeled in the DCPRA.

17. The analysis of fires uses experience data from all plants throughout the industry. At Diablo Canyon, all of the Class 1E electrical cables, except those in dedicated rooms, are enclosed in conduit. No credit was taken for this design feature when computing the expected fire occurrence/damage frequencies for specific plant equipment and locations. Additionally, the fire analysis is conservative for control room and cable spreading room fires in that control room evacuation is assumed necessary. These scenarios have been re-analyzed to reduce the conservatism (See Appendix F.3).

18. For loss of offsite power sequences, the event sequence models always assume that the pressurizer PORVs will be challenged to open and re-close. This is believed to be conservative. If the reactor trips immediately after the LOOP, RCS pressure would not be expected to rise sufficiently to challenge the PORVs.

19. No credit was given for the positive displacement charging pump for maintaining RCP seal injection even though this pump is powered by a vital bus.

20. For certain transients (e.g., LOOP, SGTR, etc.) that challenge the pressurizer PORVs and/or the safeties it was assumed that as many as two PORVs and three safety valves may open for primary pressure relief even though the success criterion for the primary pressure relief was at least one PORV or safety valve to open. These valves are required to re-close for success of the RCS pressure relief system. Failure of the valves to re-close is assumed to result in a small LOCA. Since the dominant contributors to the RCS pressure relief system unavailability were PORVs and safety valves failure to re-close, the assumption that all PORVs and safety valves may open and must re-close is conservative.

21. For loss of offsite power scenarios involving multiple diesel generator failures, the electric power recovery model did not model diesel generator repair until failure of the last diesel generator. In many cases the failures would be spread out over time; this would allow repairs to be completed on the diesel generator which fails first, prior to the failure of the next diesel generator.

22. Following loss of AC power, portable engine driven DC generators could be aligned for essential DC powered equipment in the twelve hours available before battery depletion. Such recovery was not modeled.

23. No credit was assumed for restoring main feedwater flow to the steam generators following a plant trip, or for blowing down the steam generators so that they may be supplied by the condensate system. Such recovery actions may be important for sequences involving a loss of all auxiliary feedwater. Both of these actions are proceduralized.

24. The bleed and feed success criteria requires two pressurizer PORVs to be operable for 24 hours. Loss of the associated emergency AC power train was assumed to result in PORV failure. However, since the PORVs require DC power to operate for a mission time of 24 hours the battery chargers are required to operate for continued DC power operation. Therefore, it is assumed that AC power was also required for 24 hours. This is conservative because DC power will last for at least 12 hours without the battery chargers and, after 12 hours, it is likely that one PORV is sufficient for continued bleed and feed cooling. No credit was given for this time dependent change in success criteria.

25. For the loss of offsite power frequency, no credit was given for back-feeding from the 500kV switchyard. The generic data used for the loss of offsite power frequency includes events from some plants with only one switchyard. It was conservatively assumed that for all losses of offsite power events, both the 230kV and the 500kV grids would be lost. For events affecting only power supplied from the 230kV grid, a plant trip would not occur; events which affect the 500kV grid would cause a plant trip; however, the 230kV source would still be available. No distinction is made between LOOP events affecting one or both offsite power sources. Additionally, since the weather at Diablo Canyon is very mild compared to many plants in the generic data base, the frequency of LOOP would be expected to be lower.

26. For low pressure injection discharge line interfacing LOCA events, no credit is given for the operators isolating the ruptured low pressure system from the RCS by closing the normally open, motor-operated valve on these lines.

27. No credit is assumed for the inadequate core cooling actions called for by procedure when a small LOCA has developed but all the charging and safety injection pumps are unavailable. It may be possible that the operators could depressurize the RCS sufficiently to permit low pressure injection using the RHR pumps.

28. The DCPRA plant model conservatively assumes that if the containment sump drain line is initially open and fails to isolate in response to a LOCA, sufficient water would be diverted outside containment so as to fail recirculation from the sump in the long term. It is believed that a transient analysis, if performed, would show that this assumption is not necessary, even for large LOCAs. No credit was assumed for manual isolation of the line.

29. The DCPRA plant model takes no credit for the recently installed ATWS Mitigation Actuation Circutry (AMSAC) system. As intended by design, this system should further limit the risks of ATWS events by providing alternate means of automatically actuating AFW and to trip the turbine in the event of a failure of reactor trip.

30. For ATWS events, the initiator is assumed to always occur with the plant at a power level greater than 80%. Main feedwater is assumed unavailable and no credit is given for an automatic initiation of the charging pumps due to the loss of inventory out the pressurizer. For the fraction of time that the reactor is at a lower initial power level, the success criteria for RCS pressure relief would be less

stringent. Credit for success of main feedwater would also minimize the need for RCS pressure relief. If an SI signal were to occur due to the loss of inventory out the pressurizer, this would avoid the need for the operators to manually initiate emergency boration; since thermal hydraulic calculations were not performed, it is not known whether a high containment pressure SI signal would occur within the time available to initiate emergency boration. Therefore, no credit was given in the DCPRA for automatic actuation of SI.

31. Under certain ATWT conditions, failure of RCS pressure relief was assumed to result in excessive LOCA conditions (beyond ECCS capacity). However, split fractions (PR2, PR7, PRE, and PRJ) used in the quantification of this scenario have failure of the PORVs or safety valve to re-close as dominant contributors to the unavailability of the RCS pressure relief system. This is conservative because failure of the valves to re-close after successful pressure relief would only lead to a small LOCA but the event tree model treats it as an excessive LOCA.

32. No credit was taken for operators attempting to shut down the reactor by manually inserting the control rods after initial failure to trip the reactor.

ITEM 13:

The NRC requested PG&E to provide clarification of fire scenario 13-A-FS1 in regards to why this scenario was not considered in the core damage quantification.

RESPONSE TO ITEM 13:

Fire scenario 13-A-FS-1 (Table F.3-3, sheet 65 of 374) is a localized fire in room 13-A which does not propagate due to limited combustible loading. According to PG&E's Appendix R report, the different vital cables are completely separated in different zones. The most serious impact of this fire scenario is the loss of vital AC bus F. This scenario will most likely lead to an orderly plant shutdown, not a plant trip. Therefore, it was not considered as an initiating event. Also, in comparison, a scenario initiated by a plant transient followed by a loss of vital AC bus F has frequency of occurrence of

(~3 trips/year) * (unavailability of AC bus F over a period of 24 hours)

= 2.2E-3 per year.

Fire scenario 13-A-FS-1 has a less severe impact on the plant in that the plant does not trip and has a much lower frequency of occurrence (3.0E-4 per year) compared to the above scenario. Because of these reasons it was concluded that fire scenario 13-A-FS-1 did not warrant treatment in the DCPRA as an initiating event or contribution to system unavailability.

In comparison, fire scenarios 13-B-FS-2 (Table F.3-3, sheet 77 of 374), and the symmetric scenario 13-A-FS-3 (Table F.3-3, sheet 69 of 374), are assumed to fail two vital AC buses because of propagation of fire/smoke, and are treated as initiating events.

F.3 DIABLO CANYON FIRE RISK ASSESSMENT

Revisions to the fire risk assessment have been made in response to comments and questions from the NRC. These revisions affected the control room fire analysis.

F.3.1 Introduction

This study is part of an overall plant risk assessment in which the effects of the various causes for plant equipment failure are investigated by using probabilistic methods. Fire events are one category of these equipment failure causes. In this section, the effects of a fire on plant safety systems are investigated, and the frequency of important fire scenarios is evaluated.

The method used for evaluating the impact of fires and their frequency of occurrence is summarized in the next section. In the sections following Section F.3.2, the process of fire risk assessment and the results of the study are given. In Section F.3.3, the fire scenarios that could generate relatively mild plant transients (i.e., which do not directly lead to core damage) are evaluated. In Sections F.3.4 and F.3.5, the more severe fire-induced events are analyzed, and control room and cable spreading room fires are addressed, respectively. The results of all of these fire analyses are summarized in Section F.3.6.

F.3.2 Methodology

The underlying method used in this study for the evaluation of the fire risk follows the analysis described in References F.3-1 through F.3-4. Evaluation of Diablo Canyon fire events follows the scenario approach in which a large list of scenarios that may potentially take place is envisioned. A scenario is a chain of events starting with the ignition of a combustible. A scenario includes the initiation of a fire, its growth, the ignition of other combustibles, its detection, its suppression, and its impact (by heating, smoke, or activation of the fire suppression systems) on plant equipment.

The starting point for this analysis is based on the results of the spatial interaction study given in Appendix F.2. The spatial interaction study is an integrated effort that identifies physical interactions among various power plant hazards to determine the important scenarios. As part of the spatial interaction study, a large set of internal fire scenarios was generated by that analysis. Final results in Appendix F.2 include an estimation of the scenario frequency, the impact on plant systems of each scenario, and a screening of all scenarios judged to be of significant importance to overall plant risk.

After analyzing the list of important fire scenarios that were screened by the spatial interaction analysis, two separate scenario categories were identified on the basis of their impact on the plant. The first category of scenarios produces a limited amount of equipment damage as a result of the postulated fire. For most scenarios, the extent of its impact on plant systems does not lead to core damage, assuming that the necessary remaining emergency core cooling systems operate. Analysis of this category of scenarios is given in Section F.3.3. The second category is for fire scenarios that have the potential for producing greater levels of plant damage. Fire incidents of this category generally introduce multiple system failures, and could limit the operators' functional capability inside the control room. Core damage might occur in this type of fire scenario if the recovery actions cannot be

conducted successfully. Section F.3.4 describes the control room fire scenarios, and Section F.3.5 gives an evaluation of the cable spreading room fire scenarios.

The analysis in Section F.3.3 is a more detailed evaluation of the results generated by the spatial interaction study. It is conducted by a more in-depth investigation of plant equipment locations and cable routings to better estimate event frequencies and equipment damage for a given fire scenario. In the spatial interaction analysis, it was conservatively assumed that a fire could disable all PRA equipment and cables within a fire area, independent of the severity of the given fire. For the analysis in Section F.3.3, layout and arrangement drawings of plant components and cables were examined, and the appropriate geometry and severity factors for scenario frequency quantification were developed. Both of the factors are defined in the equation below and have a numerical value between 0 and 1. Based on these additional efforts, the analysis provided a more realistic estimate of both the scenario frequency and the plant damage.

The generic expression of fire scenario frequency, R_I, inside a fire area that causes a combination of equipment damage can be expressed as

$$R_I = R_{AREA}\ F_{G,I}\ F_{S,I}\ F_{NS,I}\ F_{HE,I}$$

where

R_{AREA} = the annual frequency of fire of any severity in a given fire area.

$F_{G,I}$ = the conditional frequency of fire scenario I occurring at a specified location, given that a fire has occurred in that fire area (geometry factor).

$F_{S,I}$ = the conditional frequency of fire scenario I that is initiated and has sufficient severity to cause failure of a combination of plant equipment and cables (severity factor).

$F_{NS,I}$ = the conditional frequency of fire scenario I that is not suppressed by the suppression features before it affects equipment.

$F_{HE,I}$ = the conditional frequency of operators' failure to carry out the recovery actions for fire scenario I.

For each scenario in Section F.3.3, the quantification is performed by using the above equation to obtain the frequency of each scenario leading to core damage. Scenarios evaluated in Sections F.3.4 and F.3.5 will be quantified by the above equation and STADIC4 (Reference F.3-5) to get the scenario core damage frequency distribution. As a conservative measure, for most scenarios, $F_{NS,I}$ will be set equal to one in this internal fire analysis.

For fire scenarios that have the potential to introduce more serious plant damage states, such as the control room fires, additional consideration was given to the following aspects of the analysis:

- **Operators' Recovery Actions.** Fire-induced plant hardware damage could be extensive for these types of scenarios; therefore, recovery actions need to be started to prevent core damage. The scenarios may involve evacuation from the main control room (control room fire) or loss of the access to key plant control features inside the control room (cable spreading room fire). Such parameters as the available time window for operator

actions, location of the hot shutdown panel outside the control room, the available indications, and procedural guidance are critical factors considered in the recovery analysis. A detailed evaluation of the operators' response to carry out the necessary recovery tasks was performed in the human actions analysis section (Appendix G), and the results are given in Table F.3-1.

- **Uncertainties of the Frequencies.** To account for the uncertainty associated with the scenario development, probability distributions were assigned for each of the critical parameters. The STADIC4 computer code was used to carry out a Monte Carlo simulation based on these probability distributions. An uncertainty analysis has not been performed for the revised control room fire scenarios.

- **Suppression of the Fire.** For the control room fire scenarios, additional credit was taken for suppression. This credit for suppression of the postulated fires is implicitly considered in the combined geometry and severity factors. A severity factor curve is used that describes the conditional frequency of an electrical fire in the control room propagating a given distance or greater. The DCPRA control room fire severity curve was developed during the analysis of fires at another plant (Reference F.3-8). That study included a major update of the industry fire data based on 405 actual fires that occurred between 1965 and 1987. Most of the fires (354) were previously reported (Reference F.3-9). Data from the SNL study were cross-correlated with data from other sources (References F.3-10 and F.3-11) to extend SNL's 31 data fields to include 13 new fields of information of special interest to risk analysts. In addition, 51 new fires are included in the revised database.

  To develop the control room severity curve, the updated industry fire database was first searched for fires that occurred in control rooms. Only four control room fires were found: numbers 225, 295, 397, and 398 in Reference F.3-8. To improve the breadth of the database, the other fires that were documented in Reference F.3-8 were then examined to determine if any could occur in the control room, giving consideration to similar kinds of equipment, sources, and manning levels. Several such fires were identified and included; these were primarily electrical panel fires.

  For this database of "control room-like" fires, one of the authors of Reference F.3-8, who had been a licensed SRO at a commercial nuclear plant, reviewed the descriptive data. Based on this review, estimates of the area involved in each fire were made. An equivalent radius was then calculated, and these data were used to plot the severity factor curve. The severity factor curve is presented as Figure F.3-11. This severity factor curve was used in the re-analysis of control room fires. Only a point estimate re-analysis of the control room fire scenarios is performed. An uncertainty analysis has not been performed.

F.3.3 Fire Scenario Update

The important scenarios generated from the spatial interaction study form the starting point for this analysis. A sample of a scenario is given in Table F.3-2. Appendix F.2 gives a complete description of each data entry of the scenario. An update of the important spatial interaction scenarios (i.e., scenarios noted by "YES" at data entry 9) is presented in Table F.3-3; data entry 11 of each scenario table gives an abstract of the internal fire analysis. Three subentries were established for this data entry and are titled:

- 11A Considered for Detailed Scenario Assessment
- 11B Justification
- 11C Conclusion

Subentry 11A assigns the level of additional analysis performed for the scenario given in Table F.3-3. After a detailed review of PRA equipment layout, cable routing drawings, and the Appendix R study, the extent of equipment damage by the fire scenario can be determined. This data entry will be registered as "NO" if it was concluded that the plant damage state resulting from the fire event is limited and that no operator recovery action will be required. For most scenarios, the postulated fire would not lead to core damage, assuming the remaining plant safety systems perform their respective mitigation functions. The subentry will be entered as "YES" if the postulated fire scenario could lead to core damage without consideration of operator recovery actions.

Subentry 11B provides a summary of the bases and results developed from the internal fire analysis. As mentioned in Section F.3.2, this evaluation was started on the basis of the final results of the spatial interaction analysis. In lieu of the conservative assumption that all PRA equipment and cables within a fire area will be disabled by the fire scenario, as was postulated by the spatial interaction analysis, a more rigorous evaluation was carried out by a detailed review of the cable and conduit routings and the layout of the plant components to generate their respective geometry and severity factors. Also included in this evaluation is a review of the findings identified by the Diablo Canyon Power Plant Unit 1 Review of 10CFR50 Appendix R (Reference F.3-6).

The estimation of the geometry factor is based on the fraction of floor area covered by the plant equipment of interest in relation to the total area of a given fire area or zone. For the control room fire analysis, since the key fires are only associated with electrical panels, the geometry factor was then related to the fraction of total panel area rather than total floor area. The severity factor can be evaluated from the physical separation between the PRA equipment of interest.

After establishing the geometry and severity factors, an evaluation of the scenario frequency can be performed. Another screening process was carried out based on the comparison of fire scenario frequencies against the likelihood of system failure due to all other causes. If the accumulated fire scenario frequencies for one category of system failure amount to more than 10% of all other causes, then these fire frequencies were incorporated into the event tree models. Otherwise, the fire scenarios are considered to have an insignificant contribution to core damage frequency and were not included in the risk quantification. Table F.3-4 lists all fire scenarios included in the core damage frequency quantification.

Subentry 11C gives a brief statement about the conclusion of the fire scenario. This provides guidance for the final treatment of the analyzed results, to determine whether the scenarios are either included in the risk quantification or screened out during the evaluation. This data entry also identifies scenarios to be analyzed in more detail in Sections F.3.4 and F.3.5.

F.3.4 Analysis of Fires in the Control Room

The control room is continuously manned for each plant operation mode. It is constructed with a 3-hour fire-rated ceiling and floor. Each wall is either 3-hour or 1-hour rated; the latter is for separation of adjacent fire zones where no safety equipment is located. Each fire door is either 3-hour rated or opened to a fire zone that contains no safe shutdown equipment.

There are 28 smoke detectors inside the electrical control panels of the main control room, and there are 4 room smoke detectors. Of the 28 control panel smoke detectors, 13 are located on the main control boards (vertical boards) and 3 are located on the control console. These detectors are ionization detectors that feed into several different annunciator systems; control board annunciator "Fire/Smoke Detector" will provide both an audible alarm and a visual window alarm that informs the control room staff of the actuation of a smoke detector. Each of the 13 smoke detectors on the vertical boards also has a red light associated with it (located on top of the vertical boards) that will provide an indication to the operators of the exact location of the fire.

The control room does not have an automatic fire suppression system; however, within the control room, there are seven Halon fire extinguishers available for manual suppression. The procedures for nonradiological fires (EP M-6) instruct the operators (immediate action) to change the ventilation system to mode 2 to provide 100% outside makeup air during a control room fire. In addition, the control room Fire Fighting Preplan (of EP M-6) instructs the operators to establish additional portable ventilation, if necessary. These actions will ensure the greatest likelihood of maintaining control room habitability.

In addition to a full complement of plant control features inside the main control room, alternate control capabilities are available outside this area. Plant shutdown can be accomplished through the remote hot shutdown panels, the dedicated shutdown panel, breakers inside the switchgear rooms, and via local control features associated with individual pieces of equipment.

F.3.4.1 Critical Fire Scenarios

To establish the potential and significant fire scenarios and to analyze related accident sequences, all of the electrical cabinets and control boards were inspected. This review was based on the consideration of the impact of the fire on the equipment included in the PRA event tree and fault tree models.

The main control boards were analyzed section by section. For each section, it was assumed that fire damaged a contiguous area on the control panel. Different damage areas were considered for each panel, depending on the combination of plant equipment affected.

Fires starting outside the control panels may cause the same extent of damage to the control and instrumentation circuitry but are deemed to be significantly less likely than panel fires because, inside the panels, there are electrical components and cables; therefore, the amount of combustibles is much greater than that outside the panels. Finally, a fire outside the panels is very likely to be caused by the personnel and therefore would be detected and extinguished within a short time.

The following control panels have been identified as potential contributors to risk:

- **Vertical Board VB-1.** Contains controls and instrumentation for ASW, CCW, containment fan cooler, containment spray, safety injection, and RHR systems.
- **Vertical Board VB-2.** Contains controls and instrumentation for the chemical volume control system, pressurizer instrumentation and valves, and reactor coolant pumps.
- **Vertical Board VB-3.** Contains controls and instrumentation for the steam generator, MSIVs, auxiliary feedwater system, and the main turbine.

- **Vertical Board VB-4.** Contains controls for 4-kV buses F, G, and H; containment HVAC; turbine control; and circulating water system.

After identification of critical control boards inside the main control room, four fire scenarios were established as being representative of the most hazardous of all fire scenarios occurring in the control room. All other fires have less impact on plant equipment, require greater severity to inflict the same damage, or have a low frequency of occurrence. The four fire scenarios analyzed are

- **Scenario VB-1.** A fire that affects ASW and/or CCW control circuitry in board VB-1.
- **Scenario VB-2.** A fire that affects the control circuitry of the PORVs and charging pumps in board VB-2. This scenario was divided into two subscenarios in the revised analysis: one that affects only the PORVs, and one that affects both the PORVs and the charging pumps.
- **Scenarios VB-2/3.** A fire at the interface of boards VB-2 and VB-3, affecting PORVs and the auxiliary feedwater system control.
- **Scenario VB-4.** A fire that affects 4-kV buses F, G, and H in board VB-4.

Flow diagrams describing the initiation, propagation, and operator recovery of the fire scenarios are given in Figures F.3-1 through F.3-6. A layout drawing for main control room arrangement is presented in Figure F.3-7.

F.3.4.2 Operator Recovery Actions

One important element of control room fires is the operator response to putting out the fire and bringing the plant to a stable shutdown status. Many different operator-related scenarios can be envisioned. In general, the detection time will be short, given the large number of smoke detectors within the vertical boards and the type of annunciation provided (see previous discussion in Section F.3.4). The operators' immediate response will be to extinguish the fire and follow the appropriate fire response procedures. In addition, the operators will have to respond to equipment failures caused by the fire; this may involve restoring control to the affected equipment from outside the control room (at the hot shutdown panel, 4.16-kV switchgear rooms, 480V switchgear rooms, or at specific equipment locations).

Only in extreme cases would control room evacuation be required; before evacuating the control room, the operators would attempt all means to extinguish the fire and provide adequate ventilation. If necessary, use of self-contained breathing apparatus (SCBA) would also be considered. If control room evacuation is necessary, the operators would follow the Abnormal Operating Procedure OP AP-8 (Control Room Inaccessibility) and establish control of the plant at the hot shutdown panel, dedicated shutdown panel, 4.16-kV switchgear, and 480V switchgear.

To quantify the conditional frequency of failure of the operator recovery action, the following parameters are considered in the evaluation:

- Whether the control room must be evacuated.
- If evacuation is required, whether recovery can be accomplished prior to evacuation.
- The available time window to accomplish the designated operator task.

- The indications available to the operators.
- The stress level on the operator.
- The procedural guidance as it relates to the required mitigation actions.

The following is a discussion of the indications available to the operators and a summary of the proceduralized responses:

- **Indications Available for Equipment Failure Diagnosis.** A variety of indications will be available to the operators for diagnosis of equipment failure. Besides the obvious indication of a fire affecting a portion of the control board, annunciators will alert the operators to actual equipment failure.

- **CCW Indications**

  - **Annunciator PK01-06: CCW Vital Headers A and B.** This annunciator informs the operators that a problem has developed that is resulting in a degradation of CCW heat removal capability for headers A and B. The annunciator response refers the operators to Abnormal Operating Procedure OP AP-11, Malfunction of Component Cooling Water System. Step 1 of Section A (Loss of a Component Cooling Water Pump) instructs the operators to trip the reactor and all RCPs if no CCW pump can be placed in service. Additionally, the operators are instructed to implement Appendix C of the procedure (Backup Cooling to a Centrifugal Charging Pump) to ensure continued seal injection.

  - **Annunciator PK01-08: CCW Header C.** This annunciator informs the operators that a problem has developed with CCW cooling to header C: the primary alarm inputs are low flow for the RCP thermal barrier cooler, the RCP lube oil coolers, or CCW header C. The annunciator response refers the operators to Abnormal Operating Procedure OP AP-11, Malfunction of Component Cooling Water System. Step 1 of Section E (Loss of Component Cooling Water Flow to the RCPs) instructs the operators to trip the reactor and all RCPs if CCW cooling to more than one RCP is affected. Additionally, the operators are instructed to ensure that seal water injection is maintained and to implement Appendix C of the procedure (Backup Cooling to a Centrifugal Charging Pump) to ensure continued seal injection.

  - **Annunciator PK01-09: CCW Pumps.** This annunciator provides an indication of CCW pump trouble. If a pump trips off due to overcurrent, then the annunciator response refers the operators to OP AP-11. The operator response will then be essentially the same as in response to PK01-06, discussed above. This annunciator will also be activated should the operating pump trip off and the standby pump auto start. This, again, will inform the operators of trouble with the CCW system.

- **ASW Indications**

  - **Annunciator PK01-01: CCW System Heat Exchanger Delta-P/Header Pressure.** This annunciator provides an indication of ASW flow problems to the CCW heat exchangers. If loss of the ASW system has occurred, the operators are referred to Abnormal Operating Procedure OP AP-10, Loss of Auxiliary Saltwater. The operators are also instructed to monitor the CCW system temperature. The loss of ASW procedure instructs the operators on the restoration of the ASW system by either placing the standby pump in service or crosstying to the opposite unit.

- **Annunciator PK01-03: Auxiliary Saltwater Pumps.** This annunciator will provide an indication of ASW pump problems to the operators; if the operating pump trips on overcurrent or if the standby pump starts because the normally operating pump has tripped off, then the annunciator will alert the operators of the change in ASW system status.

- **RCP Indications**

  - **Annunciator PK05-01,02,03,04: Reactor Coolant Pumps.** This annunciator informs the operators of developing RCP pump and motor problems. Causes of annunciation include RCP motor and motor bearing temperature, and the RCP seal outlet temperature. The annunciator response lists insufficient or low CCW flow as the probable cause of the problem and instructs the operators to trip the reactor and RCPs if alarms occur on multiple RCPs. Additionally, the annunciator response refers the operators to OP AP-11, Malfunction of Component Cooling Water System (see previous discussion under CCW indication).

Based on the indications available to the operators, it is reasonable to expect prompt diagnosis of CCW and/or ASW system problems; the annunciators will heighten the operator awareness of CCW/ASW system problems and the potential for RCP problems. The operators will act promptly to protect the RCPs by tripping the reactor and the RCPs, and then proceed in the restoration of ASW and CCW. If CCW cannot be readily restored, the operators will follow the procedures to establish fire water cooling to the centrifugal charging pumps to maintain continued RCP seal injection.

For the main control room fire scenarios, operator actions required by the recovery process are discussed next.

For scenario VB-1, the flow diagram in Figure F.3-1 shows that the first operator action, after attempting to extinguish the fire, is to trip the RCPs within 10 minutes to prevent motor bearing damage after the loss of CCW flow. If the pumps are left running, shaft vibrations are assumed to develop when the bearings fail and cause damage to the pump seal assemblies, resulting in a small LOCA. The normally operating charging pump is assumed initially to remain operational, providing RCP seal injection. A second operator action required by scenario VB-1 (ZHEF12) is to restore ASW and CCW cooling from either the control room or the hot shutdown panel. Finally, a third action is to mitigate a LOCA, provided that the control room did not have to be evacuated. The quantification of these actions in scenario VB-1 is discussed in more detail in Section F.3.4.3.

In the analysis of scenario VB-2, a fire-caused hot short leads to the opening of PORVs. ZHEF21 represents the operator failure probability to terminate this small LOCA from the hot shutdown panel. If the operators do not isolate the open PORV from the hot shutdown panel, they may still mitigate the LOCA from the control room if it remains habitable. This action is considered in the analysis.

Operator action for scenario VB-2/3 is similar to that of scenario VB-2, with the addition of an auxiliary feedwater system failure. A review of the shutdown panel layout drawing (Figure F.3-8) shows that both recovery actions can be carried out on the shutdown panel; these operator activities are represented by ZHEF31 and ZHEF32. It should be noted that ZHEF21 and ZHEF31 represent similar operator actions, but each has a different probability due to the respective stress levels associated with the scenarios. Similar to the VB-2

scenarios, if the control room remains habitable, the action to mitigate a LOCA or to initiate feed and bleed cooling is considered for scenario VB-2/3.

Recovery actions for scenario VB-4 are similar to those for scenario VB-1. In addition to extinguishing the fire, they involve tripping the RCPs, restoring power to the 4-kV buses, and if the control room need not be evacuated, mitigating any LOCA that develops. A summary of these operator recovery failure frequencies is given in Table F.3-1.

Having identified the critical fire scenarios and established the required operator response, the following sections analyze the main control room fire scenarios.

F.3.4.3 Fire Scenario VB-1

The sequence of events here is a small fire at the west corner of board VB-1 that is assumed to lead to the failure of the auxiliary saltwater system and/or the component cooling water system. To prevent early RCP seal failure due to bearing failure, the operator would have to trip the RCPs. Continued operability of the charging pumps for RCP seal injection is also jeopardized after the loss of lube oil cooling that is supplied by the CCW. The combination of plant component failures postulated could cause a loss of seal injection and thermal barrier cooling for the reactor coolant pumps. Therefore, to prevent RCP seal failure due to high temperatures, the operators must promptly restore pump seal cooling.

Without recovery, a seal LOCA could occur. If CCW/ASW flow cannot be restored prior to the automatic startup of ECCS, then both high head and low head injection pumps could be damaged due to the lack of CCW cooling. Core damage would then occur due to the failure of safety injection capability.

Figure F.3-1 outlines the various sequences that make up scenario VB-1. First, the diagram indicates that the control room fires of concern affect the west corner of VB-1, which contains the controls for the ASW and CCW systems. The fire must be sufficiently severe to cause complete failure of CCW cooling to the RCPs and the charging pumps, which provide RCP seal injection flow. Interrelated to the severity of the fire, as measured by the amount of combustibles available, are the actions to extinguish the fire before it spreads. If the fire is suppressed before the ASW/CCW controls are affected, the impact of the fire is insignificant, and therefore not considered further. This is indicated in Figure F.3-1 by transferring it to the end state "proceed with plant cooldown procedures."

The next question asked in the event sequence diagram concerns control room habitability. This issue can greatly affect the likelihood of successful operator response to the fire. If the control room remains habitable, it is more likely that the operators will successfully trip the RCPs and restore ASW or CCW, whichever is lost, in time to prevent seal damage. From the hot shutdown panel, it is also possible to prevent a seal LOCA by restoring ASW/CCW. If the control room must instead be evacuated, this has a negative impact on the likelihood of successful operator response. A key consideration then becomes whether the fire fails ASW/CCW before or after the control room is evacuated. If ASW/CCW fail before evacuation, the operators may trip the RCPs before evacuating the control room, as called for by plant abnormal and alarm procedures. If ASW/CCW do not fail until after control room evacuation, no credit is given for the operators tripping the RCPs. The control room evacuation procedures do not instruct the operators to trip the RCPs before leaving the control room, and the RCPs cannot be tripped from the hot shutdown panel. The RCPs can be tripped locally at the switchgear, but no credit was assumed for this action.

The event sequence diagram shown in Figure F.3-1 was converted to an event tree for sequence quantification. This tree is shown as Figure F.3-10. The initiating event frequency is set equal to the frequency of all control room fires. Top Event EF accounts for the conditional frequency of the fire being of insufficient size to fail ASW/CCW or of the fire being extinguished prior to failing ASW or CCW. If the fire is not extinguished prior to ASW/CCW failure, Top Event CR then considers whether the control room must be evacuated. If the control room must be evacuated, Top Event EB then questions whether the ASW/CCW systems fail before or after evacuation. Together, Top Events CR and EB identify the boundary conditions against which the operator recovery actions are quantified. Top Event TP models the likelihood of the operators successfully tripping the RCPs before RCP seal damage once CCW cooling is lost. Top Event RE models the recovery of ASW/CCW prior to eventual seal damage. Top Event RE is only asked if the operators successfully trip the RCPs to prevent early seal damage. Top Event ML models the actions to mitigate a LOCA. LOCA mitigation is assumed to only be possible if the control room does not need to be evacuated; i.e., Top Event CR must be successful.

The event tree shown in Figure F.3-10 was quantified to determine the core damage frequency attributed to scenario VB-1, control room fires.

The annual frequency of control room fires, R_{CR}, which is used as the initiating event, is obtained from Reference F.3-4. The characteristic values of its probability distribution are

$R_{CR,05} = 1.3 \times 10^{-7}$ per reactor-year

$R_{CR,50} = 3.2 \times 10^{-4}$ per reactor-year

$R_{CR,95} = 1.5 \times 10^{-2}$ per reactor-year

$R_{CR} = 4.90 \times 10^{-3}$ per reactor-year

The conditional frequency of fires in board VB-1, given that a control room fire has occurred, is obtained by computing the combined geometry and severity factors.

The combined geometry and severity factor for all locations on a control board is obtained from an integration:

$$F_{G,S} = \int_A F_s(r) F_g(r) dA \qquad (1)$$

where A is the area behind the panel from which a fire can affect the equipment specified in the fire scenario, $F_g(r)$ is the geometry factor expressed in terms of conditional frequency of fire occurrence per unit area, and $F_s(r)$ is the conditional frequency of fire being so severe that an area with radius r is impacted. For $F_s(r)$, the graph in Figure F.3-11 is used, which defines $F_s(r)$ as a function of the radial distance from the fire.

To simplify the computations, $F_s(r)$ is discretized, and the integral is written as

$$F_{G,S} = \sum_i F_{si} \cdot F_{gi} \; \Delta a_i \qquad (2)$$

and for $F_g(r)$, the function used is

$$F_g(r) = \frac{1}{A} \tag{3}$$

where A is the total panel area of the control room.

$$F_{G,S} = \sum_i F_{si} \bullet \frac{\Delta a_i}{A} = \frac{1}{A} \sum_i F_{si} \Delta a_i \tag{4}$$

The integral of the combined geometry and severity factor is performed by taking the summation of the products of conditional fire occurrence probability in each postulated panel area and the associated severity factor that covers the control circuits of ASW and CCW in vertical board VB-1. In the calculation process, the combined geometry and severity factor for scenario VB-1 is obtained by following Equation 4 with some simplifications. This integration is performed by dividing the possible fire area into 1-foot-wide strips (11 strips for this case, as shown in Figure F.3-12), then combining the geometry and severity factors for each strip, and summing the contributions over the selected panel area. The contributions from each strip to the combined geometry and severity factor of scenario VB-1 are listed in Table F.3-6. The geometry factor for each strip is taken as 1.43E-2, which corresponds to the ratio of a 1-foot-wide area on board VB-1 to the total linear width of the Unit 1 main control board panels; i.e., control console and vertical boards. The geometry factor was computed conservatively since only the main vertical control boards and the control console were counted in the determination of total control room panel area.

This approach to combining the geometry and severity factors is one of several interpretations that may have been used. It is believed that this approach is reasonable because it conservatively assumes that all fires in the control room occur in the electrical panels. The partitioning of fires within the electrical panels is based on the linear width of the panels relative to the total width of all such panels in the room. There is no evidence to suggest that the total control room fire frequency should instead be partitioned by some other approach, such as according to control panel volume or projection area onto the floor.

The combined geometry and severity factor for VB-1 is found to be 2.46E-2. This value is used for the split fraction for Top Event EF in the event tree; i.e., Figure F.3-10.

The next top event is CR; i.e., control room habitability. It is judged to be likely that panel fires will be extinguished before the control room would have to be evacuated. Only 5% of such fires is judged to result in the need for control room evacuation. Therefore, 0.05 is used for the split fraction for Top Event CR in the event tree.

Top Event EB represents the failure of equipment in the control room panels before evacuation is required. Since most fires that could impact a given area of VB-1 would have to initiate in close proximity with the affected area; it is believed to be likely that the affected equipment would be failed early in the scenario, before sufficient smoke had accumulated to require evacuation of the control room. Still, there is substantial uncertainty in this value. Therefore, for quantification of the event tree, it is assumed that 50% of the time that the control room must be evacuated, the ASW or CCW controls are failed after the evacuation is completed.

Top Event TP models the operator action to trip the RCPs before the seals are damaged due to the loss of CCW cooling. If the control room must be evacuated before ASW or CCW is failed, no credit is taken for the action. This is because the control room evacuation procedures do not call for tripping the RCPs just before evacuation. If the control room does not need to be evacuated, or if ASW or CCW is lost before it is necessary to evacuate the control room, then plant abnormal and alarm procedures do instruct the operators to trip the RCPs. Human action ZHEF11 (see Table F.3-1) is used to quantify the split fraction for Top Event TP under such conditions.

The restoration of ASW/CCW in the long term, before seal damage, is modeled as human action ZHEF12. Credit for this action is taken if the control room remains habitable, or if ASW/CCW fail before evacuation is required. If ASW/CCW fail only after the control room is evacuated, it is conservatively assumed that no credit is given for restoration of ASW/CCW from outside the control room prior to RCP seal damage.

Top Event ML models the possibility of mitigating a seal LOCA caused by failure of ASW/CCW. The event tree shown in Figure F.3-10 only questions Top Event ML if the control room remains habitable; i.e, no credit is given for mitigating a LOCA from outside the control room. To mitigate the LOCA from the control room, the operators must still restore cooling to the charging or safety injection pumps for high pressure injection and, eventually, restore CCW for cooling to the RHR heat exchangers for sump recirculation. Fire water cooling may be aligned locally to the centrifugal charging pumps. Alternatively, the ASW/CCW systems that failed earlier due to the fire may be recovered once control is switched to the hot shutdown panel. The event tree quantification conservatively assumes no credit for mitigating the LOCA, if the LOCA is caused by seal damage due to failure of Top Event RE. If, however, the LOCA resulted from an earlier failure to trip the RCPs after ASW/CCW failed, then some credit for mitigating the LOCA from the control room after restoring ASW/CCW at the hot shutdown panel is assumed. For this case, a 50% chance that ASW/CCW could be restored in time to mitigate the LOCA is assumed.

To evaluate the operator recovery actions for this scenario, two operator tasks need to be addressed. First, the operators have to stop the reactor coolant pumps from the main control room before severe damage to the pumps occur. Second, the operators need to regain control over the plant from the hot shutdown panels. The time window for severe damage to the RCP bearing is conservatively estimated as 10 minutes after loss of CCW flow is assumed. If the operators trip the reactor coolant pumps within this time, no bearing damage will occur. If the pumps are left running, it is assumed that shaft vibrations caused by bearing failure will occur followed by damage to the seal assemblies. ZHEF11 in Table F.3-1 models actions to trip the RCPs before bearing damage. After the operators trip the reactor coolant pumps, they must also restore at least one source of cooling for the seal assemblies. If seal injection and thermal barrier cooling remain disabled, the flow of hot reactor coolant water through seals will eventually overheat the elastomer O-ring in the assemblies and allow them to deform. Failure of the O-rings will lead to failure of the seals and a small LOCA. Estimates made by Westinghouse indicated that the seals are not expected to fail for at least 1 hour after the loss of all cooling and that they may not experience significant damage for several hours. ZHEF12 models the operator actions to restore CCW and ASW flow from the hot shutdown panels. The point estimate values for recovery actions given in Table F.3-1 are used to quantify the event tree illustrated in Figure F.3-10 for scenario VB-1.

The point estimate core damage frequency from VB-1 scenarios was computed to be 6.47×10^{-6} per year. The highest core damage frequency sequences were found to be 8

and 9 in Figure F.3-10. These both involve a failure to extinguish the fire before ASW and/or CCW is lost, and the need to evacuate the control room. In these sequences, no credit was given for tripping the RCPs or for mitigating the assumed LOCA from outside the control room.

F.3.4.4 Scenario VB-2

This class of fire scenarios includes fires in board VB-2 that affect critical instrumentation and control circuitry of the PORVs. Other pieces of plant equipment that might be affected by this scenario include the charging pumps, pressurizer level and pressure indications, reactor coolant loop temperature, pressure and flow indications, etc. For purposes of core damage frequency quantification, fire scenario VB-2 has been subdivided into two scenarios: VB-2A and VB-2B.

This subdivision of the scenario category was judged to be useful for accurately modeling the impacts of the need for control room evacuation. For VB-2A, the fire is assumed to only fail the PORV via a hot short so that it sticks open causing a LOCA. For VB-2B, the fire is assumed to lead to greater damage, causing both a PORV to stick open and failure of the charging pumps, which are also used for high head safety injection. VB-2B has a lower occurrence frequency than that of VB-2A, but it is conservatively assumed that the additional loss of charging means that if the PORV is not isolated, the LOCA cannot be mitigated. Given the loss of all of the charging pumps, a small LOCA could still be mitigated using the safety injection pumps for high pressure injection. The pressurizer PORV and the safety injection pump controls in the vertical control boards lie on opposite sides of the charging pump controls. Therefore, the assumption used in this analysis (i.e., that the loss of all charging pumps with a stuck-open PORV leads to core damage) is very conservative. In contrast, for VB-2A, if the PORV cannot be isolated but the control room remains habitable, the loss of inventory through the stuck-open PORV may still be mitigated.

Figure F.3-2 presents the event sequence diagram for both VB-2A and VB-2B. The sequence of events is similar to that for VB-1, except, in this case, the RCPs need not be tripped before evacuating the control room. As indicated in the event sequence diagram, credit for closing the associated block valve of the affected PORV from the hot shutdown panel is required, whether or not the control room must be evacuated. If the control room must be evacuated, the event sequence diagram shows that no credit for mitigating a LOCA is assumed. The event sequence diagram is drawn so that it applies to both scenarios VB-2A and VB-2B. For scenario VB-2B, which fails the charging pumps, no credit is given for mitigating the LOCA, even if the control room remains habitable. If the control room must be evacuated, the event sequence diagram questions whether the PORV failed open before evacuation was required. Unlike the assessment for VB-1, this event is not significant for VB-2. The key recovery actions for VB-2 can be diagnosed and then implemented from the hot shutdown panel.

The event sequence diagram shown in Figure F.3-2 was converted to an event tree for quantification. The event tree is presented as Figure F.3-14. Core damage is assumed for all sequences in which Top Event ML fails. The VB-2 fire scenario is assumed to be successfully mitigated if the fire is extinguished prior to failing open the PORV (i.e., Top Event EF success), if the PORV block valve is closed (i.e., Top Event LT success), or if the LOCA is mitigated; i.e., Top Event ML success.

For quantification of scenarios VB-2A and VB-2B, the total control room fire frequency is used as the initiating event. The event tree shown in Figure F.3-14 is then quantified twice, once

for each subscenario. The total control room frequency is specialized to only those that affect the board areas of interest. This is accomplished similar to that presented previously for VB-1 scenarios. The geometry factor and the severity factor are combined by computing the conditional frequency of a fire in one area that propagates sufficiently to fail the indicated equipment and then integrating over the control panel area. The combined geometry and severity factor for VB-2A is .044, and for VB-2B is .0022. Tables F.3-8 and F.3-9 present the calculations for VB-2A and VB-2B respectively. The panel area strips used in the calculations are defined in Figure F.3-16. As expected, the more severe fire, VB-2B, which has to be much larger to fail a PORV and the charging pumps, has a lower combined geometry and severity factor. These differing combined geometry and severity factors are accounted for via the assignment of split fractions for Top Event EF.

As with VB-1 scenarios, for 5% of the VB-2 scenarios, the control room is assumed to have to be evacuated. About 50% of the time that evacuation is required, the equipment is assumed to be failed before the control room must be evacuated. The recovery action to isolate the PORV block valve from the hot shutdown panel is ZHEF21 in Table F.3-1. The recovery action to mitigate a LOCA is ZHEF22 in Table F.3-1. As noted in the event sequence diagram, no credit for mitigating a LOCA from outside the control room, except to close the PORV block valve, is assumed.

The point estimate core damage frequencies for the VB-2A and VB-2B scenarios were computed to be 2.42×10^{-8} and 2.04×10^{-8} per year, respectively. The total of these two contributions (i.e., 4.46×10^{-8} per year) is substantially lower than the earlier total for VB-2 scenarios; i.e., a mean of 1.16×10^{-6} per year. The current results are lower because of the credit taken for mitigating a LOCA from the control room when evacuation is not required and the charging pumps are available. In addition, a lower error rate for ZHEF21 (i.e., 1.88×10^{-3} versus 1.81×10^{-2}) was used in the current analysis, consistent with the analysis presented in Appendix G.

F.3.4.5 Scenarios VB-2/3

This class of fire scenarios includes fires that occur at the interface between boards VB-2 and VB-3. Such fires are of interest if they affect critical instrumentation and control circuitry for the PORVs and for the auxiliary feedwater system. Other pieces of equipment that might be affected by this scenario include pressurizer level and pressure indications. Fire scenarios that only affect the PORVs have already been accounted for in the assessment of VB-2A.

Figure F.3-3 presents the event sequence diagram for the VB-2/3 scenarios. The sequence of events is somewhat more complicated than that for the VB-2 scenarios because of the added failure of the auxiliary feedwater system. Similar to the event sequence diagrams for scenarios VB-1 and VB-2, the event sequence diagram for VB-2/3 scenarios includes events for extinguishing the fire prior to failing the equipment of interest and, if not extinguished, to question whether the control room remains habitable. If the fire fails the equipment of interest, then two recovery actions are considered: closure of the PORV block valve, and restoration of auxiliary feedwater. Both of these actions can be performed from the hot shutdown panel. Therefore, whether the control room must be abandoned has little effect on these actions.

If either of these recovery actions fail, then additional recovery actions are required to prevent core damage. Actions to either mitigate a LOCA or to perform bleed and feed cooling are only credited if control room evacuation is not required. The event sequence

diagram illustrates this for the different sequence paths; i.e., whether the control room must be evacuated. Given that the control room must be evacuated, the timing of the need for evacuation relative to the time of equipment failure is unimportant. This is because there is adequate time for the stuck-open PORV and the loss of auxiliary feedwater to be diagnosed and corrected from the hot shutdown panel. This is not like the assessment for VB-1 scenarios, wherein the timing of the requirement for control room evacuation was more important since the recovery action (i.e., tripping the RCPs) could not be performed from the hot shutdown panel.

The event sequence diagram for scenarios VB-2/3 was converted to an event tree. The event tree is presented Figure F.3-15. This event tree is similar to that for VB-2 scenarios except that an additional top event is added (i.e., Top Event SH), which accounts for the recovery of secondary heat removal. Sequences in which Top Event ML fails are assumed to result in core damage. Sequences in which the PORV block valve is closed and secondary heat removal is recovered are assumed to be successful; i.e., if both Top Events LT and SH succeed. For sequences in which either Top Event LT or SH fails but Top Event ML is successful, no core damage is assumed.

The core damage frequency from VB-2/3 scenarios is computed by quantifying the event tree in Figure F.3-15. As with the other control room fire scenarios, the initiating event frequency is assumed to be the total control room fire frequency. This initiator is then specialized to the specific fires that affect the interface of boards VB-2 and VB-3 causing failure of a PORV and of all auxiliary feedwater by using the combined geometry and severity factor. Similar to scenario VB-1, this combined factor is computed by calculating the probability of fires initiating in a given area to propagate and thereby cause failure of the PORV and auxiliary feedwater, and then integrating over the control panel area. The combined geometry factor for scenario VB-2/3 was computed to be 5.54×10^{-3}. Table F.3-10 presents the calculation for VB-2/3. The panel area strips used in the calculation are defined in Figure F.3-16.

The probability of having to evacuate the control room during such a fire was again assumed to be 5%. The chance that the fire would fail the equipment of interest before the control room had to be evacuated is 0.5. The error rate for failing to isolate the PORV from the hot shutdown panel is ZHEF31, as indicated in Table F.3-1. The failure probability to mitigate a LOCA is judgmentally assigned the value of 1×10^{-2} if the control room remains habitable and the operators successfully restart auxiliary feedwater. If the control room remains habitable but the operators fail to restore auxiliary feedwater, then the failure probability to initiate feed and bleed cooling is assumed to equal the mean value for split fraction OB1 (i.e., 2.89×10^{-2},) consistent with the noncontrol room fire scenario failure probability. If the control room must be evacuated but the PORV block valve is not closed, or auxiliary feedwater is not restored, then no additional credit was assumed for mitigating the LOCA or for bleed and feed cooling.

The point estimate core damage frequency for scenario VB-2/3 was computed to be 1.64×10^{-8} per year. This result is substantially lower than the earlier results, primarily because credit is now being taken for not having to evacuate the control room all of the time, which allows the operators to mitigate a LOCA. Also, the lower human error rates are used, consistent with the final human action analysis documented in Appendix G.

F.3.4.6 Scenario VB-4

This scenario is initiated by a fire in board VB-4. It is assumed to cause the 4-kV breakers to trip open and lead to a loss of feed from startup transformers, auxiliary transformers, and diesel generators. It is postulated that this event could result in deenergization of 4-kV buses F, G, and H. The consequence of this scenario would be the de-energization of ECCS equipment, including charging pumps, component cooling water pumps, auxiliary saltwater pumps, etc. Operation of the nonsafety grade components might not be interrupted; e.g., reactor coolant pumps could continue to run in spite of the loss of seal injection and bearing cooling. The consequence of this event could be similar to that of scenario VB-1. The combination of plant equipment failures could result in damage to the RCP seals if the operators do not trip the reactor coolant pumps and restore seal cooling in time. Failure of the RCP seal leads to a seal LOCA. Core damage might occur with the presence of seal LOCA and failure of safety injection.

The event sequence diagram shown in Figure F.3-4 outlines the various steps that make up scenario VB-4. First, the diagram indicates that the control room fires of concern affect the south portion of VB-4, which contains the controls for the emergency AC breakers. The fire must be sufficiently severe to cause complete failure of emergency AC. Loss of all emergency AC, in turn, leads to loss of all CCW cooling to the RCPs and of all injection to the RCP seals. Interrelated to the severity of the fire, as measured by the amount of combustibles available, are the actions to extinguish the fire before it spreads. If the fire is suppressed before the emergency AC breaker controls are affected, the impact of the fire is insignificant, and therefore not considered further. This is indicated in Figure F.3-4 by transferring to the end state "proceed with plant cooldown procedures."

The next question asked in the event sequence diagram concerns control room habitability. This issue can greatly affect the likelihood of successful operator response to the fire. If the control room remains habitable, it is more likely that the operators will successfully trip the RCPs and restore emergency AC in time to prevent seal damage. From the hot shutdown panel, there is also the possibility of mitigating a seal LOCA by restoring emergency AC in time to permit high pressure injection. If the control room must instead be evacuated, this event has a negative impact on the likelihood of successful operator response. A key consideration then becomes whether the fire fails emergency AC before or after the control room is evacuated. If emergency AC fails before evacuation, the operators may trip the RCPs before evacuating the control room, as called for by plant abnormal and alarm procedures. If emergency AC does not fail until after control room evacuation, no credit is given for the operators tripping the RCPs. The control room evacuation procedures do not instruct the operators to trip the RCPs before leaving the control room, and the RCPs cannot be tripped from the hot shutdown panel. The RCPs can be tripped locally at the switchgear, but no credit was assumed for this action.

Since the scenario impact is similar to the VB-1 scenarios, the frequency of core damage from VB-4 scenarios may also be determined by quantifying the event tree presented in Figure F.3-10. The initiating event frequency is set equal to the frequency of all control room fires. Top Event EF accounts for the conditional frequency of the fire being of insufficient size to fail all emergency AC or of the fire being extinguished prior to failing emergency AC. If the fire is not extinguished prior to failure of emergency AC, Top Event CR then considers whether the control room must be evacuated. If the control room must be evacuated, Top Event EB then questions whether emergency AC fails before or after evacuation. Together,

Top Events CR and EB identify the boundary conditions against which the operator recovery actions are quantified. Top Event TP models the likelihood of the operators successfully tripping the RCPs before RCP seal damage, once all emergency power is lost. Top Event RE models the recovery of emergency AC and CCW cooling prior to eventual seal damage due to high RCS temperatures. Top Event RE is only asked if the operators successfully trip the RCPs to prevent early seal damage. Top Event ML models the actions to mitigate a LOCA, given one has occurred previously. LOCA mitigation is assumed to be possible only if the control room does not need to be evacuated; i.e., Top Event CR must be successful.

The integral of the combined geometry and severity factor is calculated in the same manner as was done for scenario VB-1. One difference is that for VB-4, the postulated panel area of interest covers the control circuits associated with emergency AC rather than those for ASW and CCW. This is demonstrated in Figure F.3-13.

The combined geometry and severity factor for VB-4 is found to be 8.82E-3. This value is used for the split fraction for Top Event EF when quantifying the event tree; i.e., Figure F.3-10.

The next top event is CR; again, the value of 0.05 is used.

Top Event EB represents equipment failure before evacuation is required. Since most fires that could impact a given area of VB-4 would have to initiate in close proximity with the affected area, it is believed to be likely that the affected equipment would be failed early in the scenario before sufficient smoke had accumulated to require evacuation of the control room. Still, there is substantial uncertainty in this value. Therefore, for quantification of the event tree, it is assumed that 50% of the time that the control room must be evacuated, the emergency AC controls are failed after the evacuation is completed.

Top Event TP models the operator action to trip the RCPs before the seals are damaged due to the loss of all emergency AC. If the control room must be evacuated before ASW or CCW is failed, no credit is taken for the action. This is because the control room evacuation procedures do not call for tripping the RCPs before evacuation. If the control room does not need to be evacuated, or if emergency AC is lost before it is necessary to evacuate the control room, then plant abnormal procedures and annunciator responses instruct the operators to trip the RCPs. Human action ZHEF41 (see Table F.3-1) is used to quantify the split fraction for Top Event TP under such conditions.

The restoration of emergency AC in the long term, before seal damage, is modeled as human action ZHEF42. Credit for this action is taken if the control room remains habitable or if emergency AC fails before evacuation is required. If emergency AC fails only after the control room is evacuated, it is conservatively assumed that restoration of emergency AC power is successful.

Top Event ML models the possibility of mitigating a seal LOCA caused by failure of all emergency AC. The event tree shown in Figure F.3-10 only questions Top Event ML if the control room remains habitable; i.e, no credit is given for mitigating a LOCA from outside the control room. To mitigate the LOCA from the control room, the operators must still restore cooling to the charging or safety injection pumps for high pressure injection and, eventually, restore CCW for cooling to the RHR heat exchangers for sump recirculation. Emergency AC and, subsequently, the ASW/CCW systems may be recovered once control is switched to the hot shutdown panel. The event tree quantification conservatively assumes no credit for

mitigating the LOCA, if the operators earlier failed to restore ASW or CCW. If, however, the LOCA resulted from an earlier failure to trip the RCPs after emergency AC failed, then some credit for mitigating the LOCA by restoring emergency AC and ASW/CCW at the hot shutdown panel is assumed. For this case, a 50% chance that emergency AC and ASW/CCW could be restored in time to mitigate the LOCA is assumed.

The point estimate core damage frequency from VB-4 fire scenarios was computed to be 2.36×10^{-6} per year. As with the VB-1 scenarios, the highest frequency core damage sequences were found to be 8 and 9 in Figure F.3-10. These both involve a failure to extinguish the fire before all emergency AC is lost and the need to evacuate the control room. In these sequences, no credit was given for tripping the RCPs or for mitigating the assumed LOCA from outside the control room.

F.3.5 Analysis of Fires in the Cable Spreading Room

In this update of the fire risk assessment no changes were made to the analysis of cable spreading room fires. The revised human action error rates for the cable spreading room fire scenarios from Appendix G are presented in Table F.3-1. Credit for these lower error rates and refined estimates of the severity factors would result in a lower core damage frequency estimate from cable spreading room fires than that presented below.

There is much similarity between the cable spreading room and the control room from the standpoint of analyzing the risk of a fire-induced scenario. The cable spreading room, like the control room, contains the control and instrumentation cables of almost all of the equipment of the plant. It also contains, among other things, control and instrumentation racks associated with plant operation. Figure F.3-9 shows the conduit layout and distribution of the instrumentation racks inside the room. Also identified on this drawing is the location of main control room panels relative to the cable spreading room arrangement.

This fire area is constructed with 3-hour fire-rated ceiling, floor, and walls. Each fire door is either 3-hour rated, or 1½-hour rated for connections to fire areas containing no safety equipment. HVAC dampers of this area have a 1½-hour fire rating.

There are 15 smoke detectors installed throughout this room. The automatic fire suppression feature includes a heat-actuated total flooding CO_2 system, which can also be manually activated from the control room or the cable spreading room. The manual suppression capability consists of a fire water hose reel and portable extinguishers within the room or in adjacent fire zones at the same elevation.

All Class IE circuits are routed in steel conduits or in trays totally enclosed by steel covers. Separation of the enclosed raceway meets the criteria of Institute of Electronics and Electrical Engineers Standard 384 (Reference F.3-7).

The cable spreading room is located directly below the control room. A fire inside this region could disable the control features provided for the main control room. Alternate plant shutdown capabilities are available through the hot shutdown panel, switchgear rooms, or manual control at the equipment.

F.3.5.1 Critical Fire Scenarios

To perform a detailed analysis on the impact of fires in the cable spreading room, information about the exact location of the important cables is needed. All available cable-routing diagrams for the cable spreading room were carefully inspected. However, due to the compactness of the arrangement of the cables inside this room, engineering judgments must be exercised in the development of critical fire scenarios. Therefore, a somewhat conservative approach is followed to establish the frequency of core damage from a cable spreading room fire.

Two fire scenarios were identified as being representative and the most important of all fire scenarios affecting the cable spreading room. All other fires either have the same impact but have to be more severe to inflict damage or fail a more limited amount of equipment. The two fire scenarios analyzed are

- **Cable Spreading Room Fire Scenario One.** A fire affects both the ASW and the CCW system controls.
- **Cable Spreading Room Fire Scenario Two.** A fire affects the PORVs and pressurizer instrumentation.

Flow diagrams that describe the cable spreading room scenarios are given in Figures F.3-5 and F.3-6.

F.3.5.2 Operator Recovery Actions

One of the most important elements of a cable spreading room fire incident is the control room operator's response to the fire. In general, the recovery actions considered are similar to those for the control room fire scenario. However, there are significant differences between the two separate events. For example, the diagnosis of equipment failures due to a cable spreading room fire may be more difficult than that of a control room fire because spurious signals in instrumentation cables or control cables may lead to conflicting indications that might result in misperception of true plant status prior to identification of a fire. For cable spreading room fire scenario one, the required operator actions are similar to those for control room fire scenario VB-1; i.e., ZHEF51 and ZHEF52 of Table F.3-1. Recovery actions for cable spreading room fire scenario two are similar to operator activities in control room fire scenario VB-2. ZHEF61 represents this required action.

F.3.5.3 Cable Spreading Room Fire Scenario One

The sequence of events is a cable spreading room fire leading to the failure of the auxiliary saltwater and component cooling water systems. This fire could have occurred in the region below vertical board VB-1 or another part of the cable spreading room that would lead to failure of these two systems. Due to the uncertainty in cable routing inside this room, a conservative geometry factor will be used. The frequency of this scenario, RCS1, is obtained from

$$R_{CS1} = R_{CS}\ F_{G,CS1}\ F_{S,CS1}\ F_{HE,CS1}$$

where

R_{CS} = the annual frequency of fire of any severity in the cable spreading room.

$F_{G,CS1}$ = the conditional frequency of fire affecting a critical set of cables in such a way that damage to ASW and CCW may occur, given that a cable spreading room fire has occurred.

$F_{S,CS1}$ = the conditional frequency of fire affecting a critical set of cables being of sufficient severity to fail both ASW and CCW.

$F_{HE,CS1}$ = the conditional frequency of operators failing to trip RCPs and restoring CCW and ASW flow within a given time interval.

The annual frequency of cable spreading room fires, R_{CS}, is obtained from Reference F.3-4. Its characteristic values are

$R_{CS,05} = 7.0 \times 10^{-7}$ per reactor-year

$R_{CS,50} = 7.0 \times 10^{-4}$ per reactor-year

$R_{CS,95Z} = 2.2 \times 10^{-2}$ per reactor-year

$< R_{CS} > = 6.7 \times 10^{-3}$ per reactor-year

The conditional frequency of fires affecting a critical set of cables in such a way that failure of ASW and CCW may occur. $F_{G,CS1}$ (given that a cable spreading room fire has occurred) is derived from using judgment based on the evaluation of available cable routing drawings and the results of a detailed cable spreading room fire analysis for other power plants. It is judged that less than 15% of the total cable spreading room floor area would be close to a critical set of cables of these two systems. This term has the following characteristic values:

$F_{G,CS1,05} = 0.075$

$F_{G,CS1,50} = 0.150$

$F_{G,CS1,95} = 0.300$

$< F_{G,CS1} > = 1.64 \times 10^{-1}$

The severity factor $F_{S,CS1}$, that is, the conditional frequency of fire at the critical locations of sufficient severity to cause cable spreading room fire scenario one, is judged to be no greater than 0.5. Therefore, $F_{S,CS1}$ is chosen to be 0.5.

To evaluate the operator recovery actions for this scenario, it is assumed that fire of this magnitude would disable the control features of the auxiliary saltwater system and the component cooling water system from the main control room. Wherever necessary, the operators would supplement their actions to carry out the required shutdown procedures from the hot shutdown panel located at Elevation 104'. Similar to the circumstances confronted by the operators in the control room fire scenario VB-1 (Section F.3.4.3), the recovery actions include tripping the reactor coolant pumps and restoring the RCP seal cooling source to prevent seal LOCA. ZHEF51 models the necessary steps to trip the RCPs before bearing failure causes subsequent seal damage. ZHEF52 models the operator actions to reestablish CCW and ASW flow from the hot shutdown panels. Based on the above analysis, the characteristic values of the recovery actions can be represented as

$F_{HE,CS1,05} = 1.80 \times 10^{-3}$

$F_{HE,CS1,50} = 8.97 \times 10^{-3}$

$F_{HE,CS1,95} = 4.55 \times 10^{-3}$

$F_{HE,CS1} = 1.43 \times 10^{-2}$

All of the parameters needed to evaluate R_{CS1} have been quantified. The characteristic values of cable spreading room fire scenario one are

$R_{CS1,05} = 1.03 \times 10^{-8}$ per reactor-year

$R_{CS1,50} = 5.34 \times 10^{-7}$ per reactor-year

$R_{CS1,95} = 2.48 \times 10^{-5}$ per reactor-year

$< R_{CS1} > = 7.90 \times 10^{-6}$ per reactor-year

F.3.5.4 Cable Spreading Room Fire Scenario Two

The sequence of events is a cable spreading room fire that damages the control circuitry of the PORVs. Other plant equipment that might be affected by this scenario includes the auxiliary relay of the pressurizer pressure control, the auxiliary relay of the pressurizer temperature control, etc. As was shown by the schematic drawing of the PORVs control, either of the above relays can open the power-operated relief valves at the preset level, and both of them are located at the rack nuclear auxiliary safeguard cubicle A (RNASA) inside the cable spreading room. Fire-induced closure of either relays could generate a spurious signal to open the PORVs. This fire could occur in the region below vertical board VB-2 or in another part of the cable spreading room that would lead to failure of the PORVs. Due to the uncertainty in cable routing inside this room, a conservative geometry factor will be used. The frequency of this scenario, R_{CS2}, is obtained from

$$R_{CS2} = R_{CS}\ F_{G,CS2}\ F_{S,CS2}\ F_{HE,CS2}$$

where

$F_{G,CS2}$ = the conditional frequency of fire affecting a critical set of cables so that damage to the PORVs' control circuitry might occur, given that a cable spreading room fire has been initiated.

$F_{S,CS2}$ = the conditional frequency of fire affecting a critical set of cables being of sufficient severity to fail the PORVs.

$F_{HE,CS2}$ = the conditional frequency of operators failing to close PORVs.

The annual frequency of cable spreading room fires, R_{CS}, obtained from Section F.3.5.3, is used here.

The conditional frequency of fires affecting a critical set of cables so that failure to PORVs may occur, $F_{G,CS1}$ (given that a cable spreading room fire has occurred) is derived from using judgment based on the evaluation of available cable-routing drawings and the results of detailed cable spreading room fire analysis for other power plants. It is judged that less than

25% of the cable spreading room floor area would be close to a critical set of cables of the PORVs and the auxiliary relays of the pressurizer pressure and temperature controls. The geometry factor is judged to have these characteristic values:

$F_{G,CS2,05} = 0.125$

$F_{G,CS2,50} = 0.250$

$F_{G,CS2,95} = 0.500$

$< F_{G,CS2} > = 0.276$

The severity factor $F_{S,CS2}$, that is, the conditional fire frequency at the critical locations of sufficient severity to cause cable spreading room fire scenario two, is judged to be no greater than 0.5. Therefore, $F_{S,CS2}$ is chosen to be 0.5.

To evaluate the operator recovery actions for this scenario, it is assumed that fire of this magnitude would disable the operators from closing the PORVs inside the main control room. Thereafter, whenever necessary, the operators would supplement their actions to carry out the required shutdown procedures from the hot shutdown panel located at Elevation 104'. The circumstances facing the operators in this scenario are similar to those of control room fire scenario VB-2 (Section F.3.4.4). The recovery actions include closure of the PORVs and are modeled by ZHEF61. Based on the above analysis, the characteristic values of the recovery actions can be represented as

$F_{HE,CS2,05} = 0.0022$

$F_{HE,CS2,50} = 0.011$

$F_{HE,CS2,95} = 0.055$

$< F_{HE,CS2} > = 0.0181$

All of the parameters needed to evaluate R_{CS2} have been quantified. The characteristic values of cable spreading room fire scenario two are

$R_{CS2,05} = 2.00 \times 10^{-8}$ per reactor-year

$R_{CS2,50} = 1.03 \times 10^{-6}$ per reactor-year

$R_{CS2,95} = 4.88 \times 10^{-5}$ per reactor-year

$< R_{CS2} > = 1.23 \times 10^{-5}$ per reactor-year

F.3.6 Summary of the Results

The results of the Diablo Canyon fire study can be divided into two categories based on the impact of fire-induced plant damage. The first category of fire scenarios leads to limited plant damage. Core damage is not expected for these scenarios if the remaining plant safety systems (i.e., systems not affected by the fire scenario) operate as needed. The methodology used to identify this category of fire scenarios was presented in Section F.3.3, and the results are presented in Table F.3-4. The second category of scenarios has the potential of

introducing more serious plant damage. Evaluation of these fire scenarios was discussed in Sections F.3.4 and F.3.5. Table F.3-5 summarizes these results.

Scenarios given in Table F.3-4 were selected based on the impact on plant systems and the associated scenario frequency. Justifications for the screening process are given by data entry 11 of the individual scenario descriptions in Table F.3-3. Scenarios identified in Table F.3-4 are the fire events included in the plant risk quantification. This is carried out by using the fire scenario frequencies as initiating events for the event tree models. As shown in Table F.3-4, designators FS1 through FS8 represent these fire scenarios. The fire frequencies are assigned to the set of plant equipment failures, as identified in column 2 of Table F.3-4, and summarized below:

- Loss of both motor-driven auxiliary feedwater pumps.
- Loss of all charging pumps.
- Loss of all component cooling water pumps.
- Loss of control room ventilation.
- Loss of both auxiliary saltwater pumps.
- Loss of auxiliary saltwater system and component cooling water system.
- Loss of electric buses F and G.
- Loss of electric buses G and H.
- Loss of electric buses F, G, and H.

Fire scenarios that could lead to more severe plant damage states are summarized in Table F.3-5; control room and cable spreading room fires belong to this category. Scenarios developed from these two critical plant areas require recovery actions to prevent core damage. Plant equipment affected by these fire scenarios were identified and evaluated in Sections F.3.4 and F.3.5; they are shown in column 2 of Table F.3-5 and summarized below:

- Loss of auxiliary saltwater system and component cooling water system control.
- Loss of PORVs and charging pumps control.
- Loss of PORVs and auxiliary feedwater system control.
- Loss of 4-kV buses F, G, and H controls.
- Loss of PORVs control.

As was identified in Table F.3-5, the mean values of the core damage frequencies due to control room and cable spreading room fire are 8.89×10^{-6} per reactor-year and 2.02×10^{-5} per reactor-year, respectively. The total of the core damage frequencies of these two groups of scenarios (i.e., 2.91×10^{-5} per year) is added separately to the dominant sequence model.

The higher frequency for the cable spreading room fire than that of the control room can be explained as follows: first, the higher fire initiating frequency mean value associated with the cable spreading room is 6.7×10^{-3} per reactor-year versus the mean value of 4.9×10^{-3} per reactor-year for the control room. Second, the compact conduit routing inside the cable spreading room leads to the selection of a conservative geometry and severity factors for the cable spreading room fire scenarios. Additionally, no credit is given for the train-to-train separation provided by the conduits on the control cables. Also shown in Table F.3-5 is the mean value of the total core damage frequency introduced from control room and cable spreading room fire events. This value is estimated as 2.91×10^{-5} per reactor-year. This represents a slight decrease (i.e., from 3.18×10^{-5} per reactor-year) from that assessed in the original study.

F.3.7 References

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F.3-2. Siu, N., and G. Apostolakis, "Probabilistic Models for Cable Tray Fires," *Reliability Engineering*, Vol. 3, pp. 213-227, 1982.

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F.3-7. Institute of Electrical and Electronics Engineers, "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits," IEEE Standard 384-1981.

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F.3-10. S. M. Stoller Corporation, Nuclear Power Experience, 1986.

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Table F.3-1 (Page 1 of 2). Operator Recovery Failure Probability for Control Room and Cable Spreading Room Fire							
Fire Scenario	Operator Recovery Action		Mean	5th Percentile	Median	95th Percentile	Point Estimate*
	Action Designator	Action Description					
Scenario VB-1	ZHEF11	Operator trips the RCPs from the control room within 10 minutes.	*				2.33-3
	ZHEF12	Operator restores ASW and CCW flow from hot shutdown panel.	*				2.7-3
Scenario VB-2	ZHEF21	Operator closes PORVs from the hot shutdown panel.	*				1.88-3
	ZHEF22	Operator mitigates LOCA from control room, given charging available.	*				1.0-2
Scenario VB-2/3	ZHEF31	Operator closes PORVs from the hot shutdown panel.	*				1.73-3
	ZHEF32	Operator starts and controls AFW flow from the hot shutdown panel.	*				6.50-3
	ZHEF33	Operator mitigates LOCA from the control room.	*				1.0-2
	ZHEF34	Operator initiates bleed and feed cooling from the control room.	*				2.89-2

Table F.3-1 (Page 1 of 2). Operator Recovery Failure Probability for Control Room and Cable Spreading Room Fire

Fire Scenario	Operator Recovery Action		Mean	5th Percentile	Median	95th Percentile	Point Estimate*
	Action Designator	Action Description					
Scenario VB-4	ZHEF41	Operator trips the RCPs from the control room within 10 minutes.	*				3.46-3
	ZHEF42	Operator restores 4-kV power supply and starts the charging pumps, the CCW pumps, and the ASW pumps.	*				3.17-3
Cable Spreading Room Fire Scenario One	ZHEF51	Operator trips the RCPs within 10 minutes.	2.47-2	0.003	0.015	0.075	7.5-3**
	ZHEF52	Operator restores ASW and CCW flow from hot shutdown panel.	1.43-2	0.0018	0.009	0.045	3.16-3**
Cable Spreading Room Fire Scenario Two	ZHEF61	Operator closes PORVs from the shutdown panel.	1.81-2	0.0022	0.011	0.055	2.60-3**

*The human error rates were revised from those assumed in the original study. The initial human action analysis was not available at the time that the fire analysis was first performed. The point estimates noted in this column are the final point estimates taken from Appendix G, "Human Actions Analysis."

**The revised human error rates for the cable spreading room fires were not used in this revision. Instead, the cable spreading room fire assessment still uses the older, conservative error rates that appeared in the original analysis.

Note: Exponential notation is indicated in abbreviated form; e.g., 2.33-3 = 2.33×10^{-3}.

Table F.3-2. Sample of Scenario Table from the Spatial Interaction Study

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*******                ************************************************************

BUILDING              : Turbine Bldg

LOCATION NAME         :  Cable Spreading Rm EL104

LOCATION DESIGNATOR :  12-C

SCENARIO DESIGNATOR :  12-C-FS-1

 1) HAZARD TYPE :  FS Fire and Smoke

 2) SOURCE TYPE :  CC Control Cable
                   IC Instrumentation Cable
                   PC Power Cable
                   TR Transient Fuel

 3) SCENARIO INITIATION :  FIRE FROM ANY HAZARD SOURCE IN 2).

 4) PATH OF PROPAGATION
    a- PATH TYPE : LOCALIZED          b- PROPAGATION TO : NONE

 5) MITIGATING FEATURES : FIRE HOSES,SMOKE DETECTOR,OTHER FORM OF MITIGATION 1,PORTABLE EXTINGUISHER-CO2

 6) ADDITIONAL SCENARIO DETAIL :  FIRE IS LOCALIZED IN 12-C DUE TO LIMITED
                                  COMBUSTIBLE LOADING.

 7) SCENARIO FREQUENCY : 1.3E-4 /YR

 8) PRA EQUIPMENT AFFECTED :

     EQUIP ID       EQUIP TYPE
     CCW-1--03-CC   CC Control Cable
     DEG-0-TP1-PC   PC Power Cable
     DEG-0DG13-PC   PC Power Cable
     DEG-1DG11-PC   PC Power Cable
     EPV-1-4KVH     DU Electrical Bus
     EPV-1FD43      DA Damper
     EPV-1S-67-PC   PC Power Cable
     FW--1---2-PC   PC Power Cable
     FW--1-110-IC   IC Instrumentation Cable
     FW--1-111-IC   IC Instrumentation Cable
     RHR-1---1-CC   CC Control Cable
     RHR-1---2-CC   CC Control Cable

 9) CONSIDERED FOR FURTHER ANALYSIS : NO

10) REMARKS : NO LOCA DEVELOPED; INTERNAL INITIATING EVENTS WITH FAILURE OF BOTH RHR ARE 1.0E-4 AND THESE DO NOT CONTRIBUTE
              SIGNIFICANTLY; CHR STILL AVAILABLE VIA FC.

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Table F.3-3. Fire Scenario Database

There are no changes to this table.

Table F.3-4. Fire Scenarios for Risk Quantification			
Scenario Designation	Scenario Impact on Plant Equipment	Estimate Frequency	Designator For Support Model Event Tree
3-Q-2-FS-1	Loss of Both Motor-Driven Auxiliary Feedwater Pumps	1.2-4	FS1
14-A-FS-1	Loss of Both Motor-Driven Auxiliary Feedwater Pumps	2.0-5	FS1
14-A-104-FS-1	Loss of Both Motor-Driven Auxiliary Feedwater Pumps	3.0-5	FS1
3-BB-100-FS-1	Loss of Both Motor-Driven Auxiliary Feedwater Pumps	2.1-6	FS1
6-A-5-FS-1	Loss of Both Motor-Driven Auxiliary Feedwater Pumps	8.7-6	FS1
5-A-4-FS-1A	Loss of Both Motor-Driven Auxiliary Feedwater Pumps	4.0-6	FS1
S-3-FS-1	Loss of Both Motor-Driven Auxiliary Feedwater Pumps	3.5-5	FS1
12-A-FS-1	Failure to Start Both Motor-Driven Auxiliary Feedwater Pumps	1.7-5	FS1
5-A-4-FS-1B	Loss of Both Motor-Driven Auxiliary Feedwater Pumps Plus All 10% Dump Valves	2.0-6	FS1
	Total Frequency of FS1	2.39-4	
3-H-1-FS-1	Loss of All Three Charging Pumps	2.0-3	FS2
3-C-FS-5	Loss of All Charging Pumps	4.0-4	FS2
3-AA-FS-1	Loss of All Charging Pumps and Loss of Two MSIVs	3.8-5	FS2
3-J-2-FS-1A	Loss of All Charging Pumps	4.5-4	FS2
	Total Frequency of FS2	2.89-3	
3-J-2-FS-1B	Loss of All CCW Pumps	3.05-6	FS3
8-B-3-FS-1	Loss of Control Room Venting Failing All Three Fans	2.0-3	FS4
4-A-FS-1B	Loss of Both ASW Pumps	3.0-5	FS5
4-B-FS-1	Loss of Both ASW Pumps	2.82-6	FS5
14-E-FS-1	Loss of All ASW and CCW	1.0-5	FS5
	Total Frequency of FS5	4.28-5	
4-A-FS-1A	Loss of Buses F and G	7.5-6	FS6
5-A-1-FS-3	Loss of Buses F and G	1.0-6	FS6
5-A-2-FS-3	Loss of Buses F and G	1.0-6	FS6
12-A-FS-2	Loss of Buses F and G	1.3-7	FS6
12-B-FS-2	Loss of Buses F and G	1.3-7	FS6
13-A-FS-3	Loss of Two Buses (F and G)	6.0-6	FS6
13-B-FS-2	Loss of Two Buses (F and G)	6.0-6	FS6
	Total Frequency of FS6	2.18-5	
5-A-2-FS-4	Loss of Buses G and H	1.0-6	FS7
5-A-3-FS-3	Loss of Buses G and H	1.0-6	FS7
13-C-FS-2	Loss of Two Buses (G and H)	6.0-6	FS7
13-B-FS-3	Loss of Two Buses (G and H)	6.0-6	FS7
	Total Frequency of FS7	1.4-5	
14-D-FS-3	Delayed Failure of All Three Buses F, G, and H	5.0-6	FS8

Table F.3-5. Summary of Fire Scenarios That Lead to Core Damage — Control Room and Cable Spreading Room Fires						
Scenario Designation	Scenario Impact on Plant Equipment	Mean	5th Percentile	Median	95th Percentile	Point Estimate*
Control Room VB-1	Loss of ASW and CCW Control on Board VB-1	*				6.47-6
Control Room VB-2A	Loss of PORVs Only in Board VB-2	*				2.42-8
Control Room VB-2B	Loss of PORVs and Charging Pumps Control in Board VB-2	*				2.04-8
Control Room VB-2/3	Loss of PORVs and Auxiliary Feedwater System Control in Boards VB-2 and VB-3 Interface	*				1.64-8
Control Room VB-4	Loss of 4-kV Buses F, G, and H in Board VB-4	*				2.36-6
	Summation of Control Room Fire Scenario Frequencies	8.89-6				
Cable Spreading Room Scenario 1	Loss of ASW and CCW Control	7.90-6	1.03-8	5.34-7	2.48-5	
Cable Spreading Room Scenario 2	Loss of PORVs Control	1.23-5	2.00-8	1.03-6	4.88-5	
	Summation of Cable Spreading Room Fire Scenario Frequencies	2.02-5	4.08-8	1.87-6	7.93-5	
	Summation of Control Room and Cable Spreading Room Fire Scenario Frequencies	2.91-5				

*The core damage frequency estimates for the control room fire scenarios were revised. However, only point estimates were calculated, not uncertainties. To obtain the revised total contribution to the core damage frequency attributable to control room and cable spreading room fires. The point estimates for the control room fire scenarios were added to the means from the other scenarios in the table; i.e., the point estimates were assumed to be approximately means.

Table F.3-6. Contribution from Each Strip to the Combined Geometry and Severity Factor of VB-1	
Strip	**Combined Geometry and Severity Factor**
Strip 1 to 2 and 3	0.0143 × (0.6) FS ≥ 1 foot = 8.57-3
Strip 2 to 1	0.0143 × (0.6) FS ≥ 1 foot = 8.57-3
Strip 3 to 1	0.0143 × (0.02) FS ≥ 2 feet = 2.86-3
Strip 4 to 1	0.0143 × (0.1) FS ≥ 3 feet = 1.43-3
Strip 5 to 1	0.0143 × (0.06) FS ≥ 4 feet = 8.57-4
Strip 6 to 1	0.0143 × (0.03) FS ≥ 5 feet = 4.29-4
Strip 7 to 1	0.0143 × (0.027) FS ≥ 6 feet = 3.86-4
Strip 8 to 1	0.0143 × (0.024) FS ≥ 7 feet = 3.43-4
Strip 9 to 1	0.0143 × (0.021) FS ≥ 8 feet = 3.00-4
Strip 10 to 1	0.0143 × (0.018) FS ≥ 9 feet = 2.57-4
Strip 11 to 1	0.0192 × (0.015) FS ≥ 10 feet = 2.88-4
Strip 12 to 1	0.0143 × (0.012) FS ≥ 11 feet = 1.71-4
Strip 13 to 1	0.0143 × (0.009) FS ≥ 12 feet = 1.29-4
Combined Geometry and Severity Factor (Total) = 2.46-2	
Note: Exponential notation is indicated in abbreviated form; e.g., 8.57-3 = 8.57×10^{-3}.	

Table F.3-7. Contribution from Each Strip/Area to the Combined Geometry and Severity Factor of Fire Scenario VB-4	
Strip/Area	**Combined Geometry and Severity Factor**
Strip 1	0.0143 × (0.003) FS ≥ 14 feet = 4.29-5
Strip 2	0.0143 × (0.006) FS ≥ 13 feet = 8.57-5
Strip 3	0.0143 × (0.009) FS ≥ 12 feet = 1.29-4
Strip 4	0.0143 × (0.012) FS ≥ 11 feet = 1.71-4
Strip 5	0.0143 × (0.015) FS ≥ 10 feet = 2.14-4
Strip 6	0.0143 × (0.018) FS ≥ 9 feet = 2.57-4
Strip 7	0.0143 × (0.021) FS ≥ 8 feet = 3.00-4
Strip 8	0.0143 × (0.024) FS ≥ 7 feet = 3.43-4
Strip 9	0.0143 × (0.027) FS ≥ 6 feet = 3.87-4
Strip 10	0.0143 × (0.030) FS ≥ 5 feet = 4.29-4
Strip 11	0.0143 × (0.058) FS ≥ 4 feet = 8.29-4
Strip 12	0.0143 × (0.086) FS ≥ 3 feet = 1.23-3
Strip 13	0.0143 × (0.086) FS ≥ 3 feet = 1.23-3
Strip 14	0.0143 × (0.058) FS ≥ 4 feet = 8.29-4
Strip 15	0.0143 × (0.030) FS ≥ 5 feet = 4.29-4
Strip 16	0.0143 × (0.027) FS ≥ 6 feet = 3.86-4
Strip 17	0.0143 × (0.024) FS ≥ 7 feet = 3.45-4
Strip 18	0.0143 × (0.021) FS ≥ 8 feet = 3.00-4
Strip 19	0.0143 × (0.018) FS ≥ 9 feet = 2.57-4
Strip 20	0.0143 × (0.015) FS ≥ 10 feet = 2.14-4
Strip 21	0.0143 × (0.012) FS ≥ 11 feet = 1.71-4
Strip 22	0.0143 × (0.009) FS ≥ 12 feet = 1.29-4
Area 23	0.0196 × (0.006) FS ≥ 13 feet = 1.17-4
Combined Geometry and Severity Factor of VB-4 = 8.82-3	
Note: Exponential notation is indicated in abbreviated form; e.g., 4.29-5 = 4.29×10^{-5}.	

Table F.3-8. Contribution from Each Strip/Area to the Combined Geometry and Severity Factor of Fire Scenario VB-2A

Contributor	Combined Geometric and Severity Factor	
Strip 1	0.0036 x [(0.5) FS≥1ft. — (0.027) FS≥6ft.]	= 1.70-3
Strip 2	0.0036 x [(0.233) FS≥2ft. — (0.03) FS≥5ft.]	= 7.31-4
Strip 3	0.011 x [(1.0) FS≥0ft. — (0.03) FS≥5ft.]	= 1.07-2
Strip 5	0.0143 x [(1.0) FS≥0ft. — (0.027) FS≥6ft.]	= 1.39-2
Strip 7	0.0143 x [(0.5) FS≥1ft. — (0.027) FS≥6ft.]	= 6.76-3
Strip 9	0.0143 x [(0.233) FS≥2ft. — (0.03) FS≥5ft.]	= 2.90-3
Strip 11	0.0143 x [(0.086) FS≥3ft. — (0.058) FS≥4ft.]	= 4.00-4
Strip 4	0.011 x [(0.5) FS≥1ft. — (0.058) FS≥4ft.]	= 4.86-3
Strip 6	0.0143 x [(0.233) FS≥2ft. — (0.086) FS≥3ft.]	= 2.10-3
Combined Geometric and Severity Factor of VB-2A (TOTAL)		= 4.41-2

Table F.3-9. Contribution from Each Strip/Area to the Combined Geometry and Severity Factor of Fire Scenario VB-2B

Contributor	Combined Geometric and Severity Factor	
Strip 7	0.0143 x [(0.027) FS≥6ft. — (0.024) FS≥7ft.]	= 4.29-5
Strip 9	0.0143 x [(0.03) FS≥5ft. — (0.021) FS≥8ft.]	= 1.29-4
Strip 11	0.0143 x [(0.058) FS≥4ft. — (0.018) FS≥9ft.]	= 5.72-4
Strip 13	0.0143 x [(0.03) FS≥5ft. — (0.015) FS≥10ft.]	= 2.15-4
Strip 15	0.0143 x [(0.027) FS≥6ft. — (0.012) FS≥11ft.]	= 2.15-4
Strip 17	0.0143 x [(0.024) FS≥7ft. — (0.009) FS≥12ft.]	= 2.15-4
Strip 19	0.0143 x [(0.021) FS≥8ft. — (0.006) FS≥13ft.]	= 2.15-4
Strip 21	0.0143 x [(0.018) FS≥9ft. — (0.003) FS≥14ft.]	= 2.15-4
Strip 23	0.0143 x [(0.015) FS≥10ft. — (0.0) FS≥15ft.]	= 2.15-4
Strip 25	0.0143 x [(0.012) FS≥11ft.]	= 1.72-4
	Combined Geometric and Severity Factor of VB-2B (TOTAL)	= 2.21-3

Table F.3-10. Contribution from Each Strip/Area to the Combined Geometry and Severity Factor of Fire Scenario VB-2/3

Contributor	Combined Geometric and Severity Factor	
Strip 5	0.0143 x [(0.027) FS≥6ft. — (0.024) FS≥7ft.]	= 4.29-5
Strip 3	0.011 x [(0.03) FS≥5ft. — (0.021) FS≥8ft.]	= 9.90-5
Strip 1	0.0036 x [(0.027) FS≥6ft. — (0.018) FS≥9ft.]	= 3.24-5
Strip 2	0.0036 x [(0.03) FS≥5ft. — (0.015) FS≥10ft.]	= 5.40-5
Strip 4	0.011 x [(0.058) FS≥4ft. — (0.015) FS≥10ft.]	= 4.37-4
Strip 6	0.0143 x [(0.086) FS≥3ft. — (0.015) FS≥10ft.]	= 1.02-3
Strip 8	0.0143 x [(0.086) FS≥3ft. — (0.012) FS≥11ft.]	= 1.06-3
Strip 10	0.0143 x [(0.058) FS≥4ft. — (0.009) FS≥12ft.]	= 7.01-4
Strip 12	0.0143 x [(0.03) FS≥5ft. — (0.006) FS≥13ft.]	= 3.43-4
Strip 14	0.0143 x [(0.027) FS≥6ft. — (0.003) FS≥14ft.]	= 3.43-4
Strip 16	0.0143 x [(0.024) FS≥7ft.]	= 3.43-4
Strip 18	0.0143 x [(0.021) FS≥8ft.]	= 3.00-4
Strip 20	0.0143 x [(0.018) FS≥9ft.]	= 2.57-4
Strip 22	0.0143 x [(0.015) FS≥10ft.]	= 2.15-4
Strip 24	0.0143 x [(0.012) FS≥11ft.]	= 1.72-4
Strip 26	0.011 x [(0.009) FS≥12ft.]	= 9.90-5
Strip 28	0.0036 x [(0.006) FS≥13ft.]	= 2.16-5
	Combined Geometric and Severity Factor of VB- 2/3 (TOTAL)	= 5.54-3

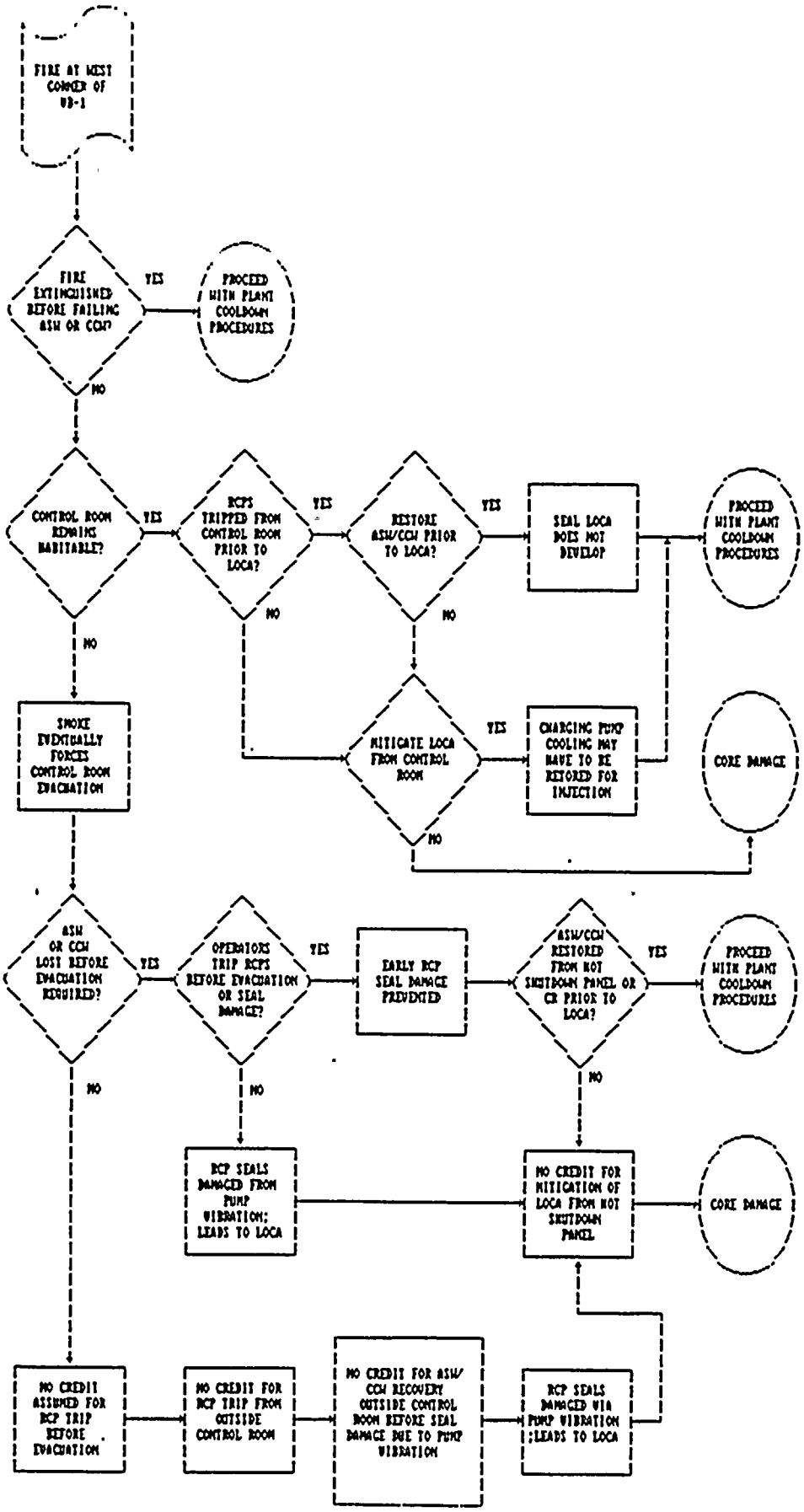


Figure F.3-1. Control Room Fire Scenario VB-1

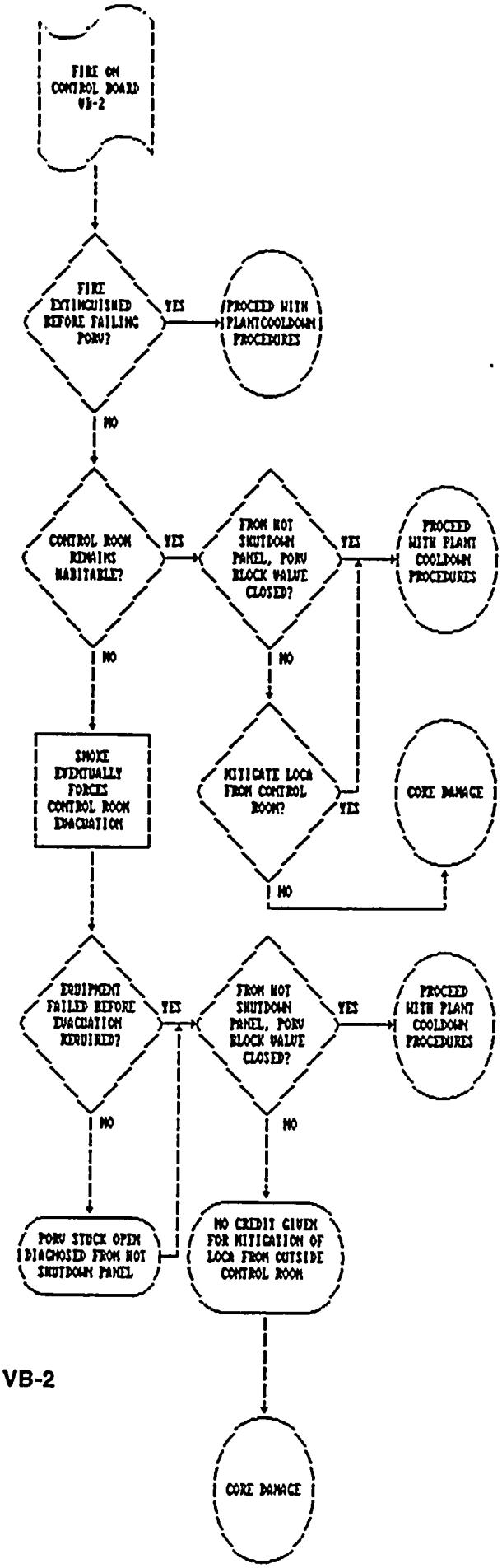


Figure F.3-2. Control Room Fire Scenario VB-2

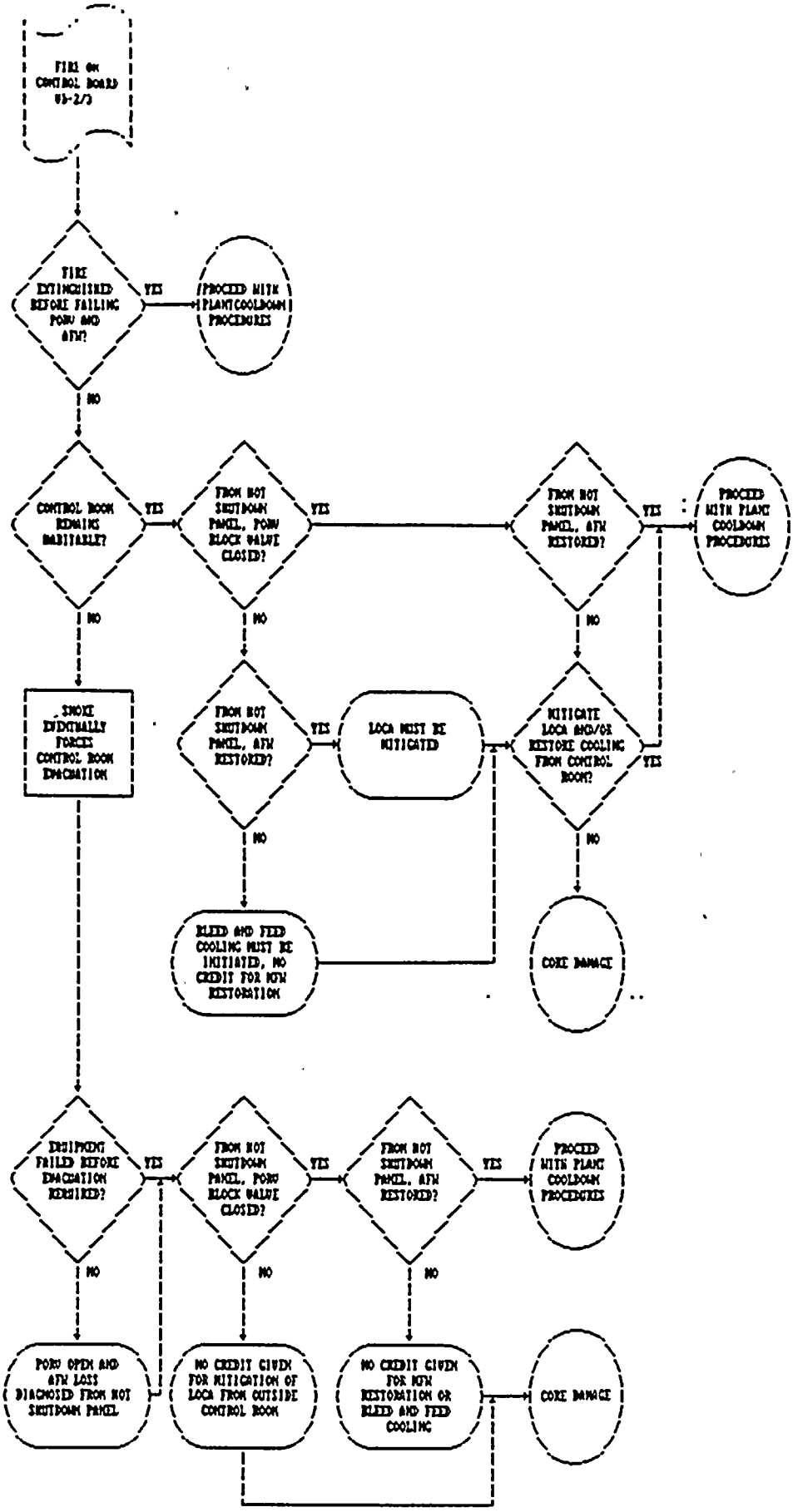


Figure F.3-3. Control Room Fire Scenario VB-2/3

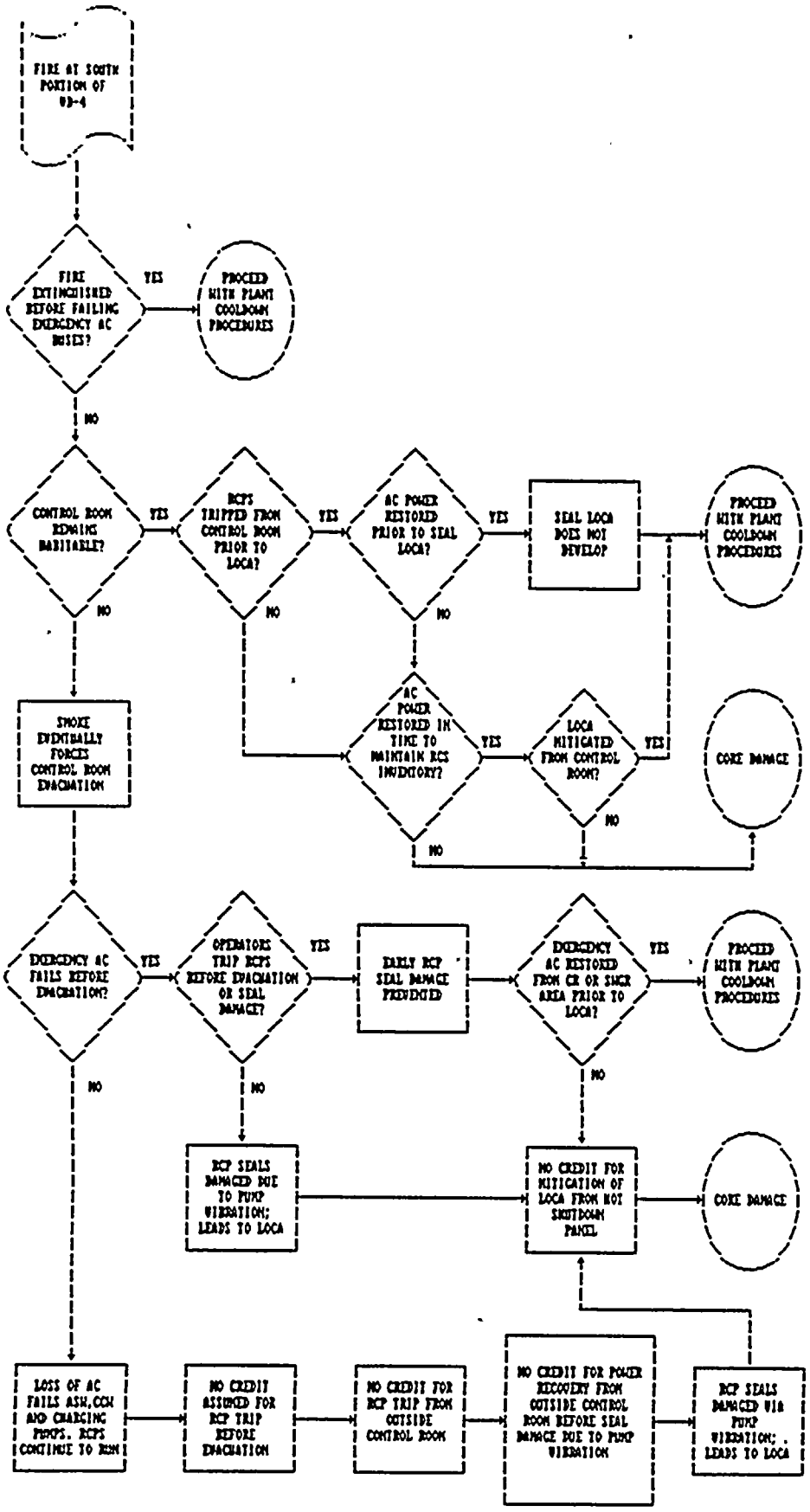


Figure F.3-4. Control Room Fire Scenario VB-4

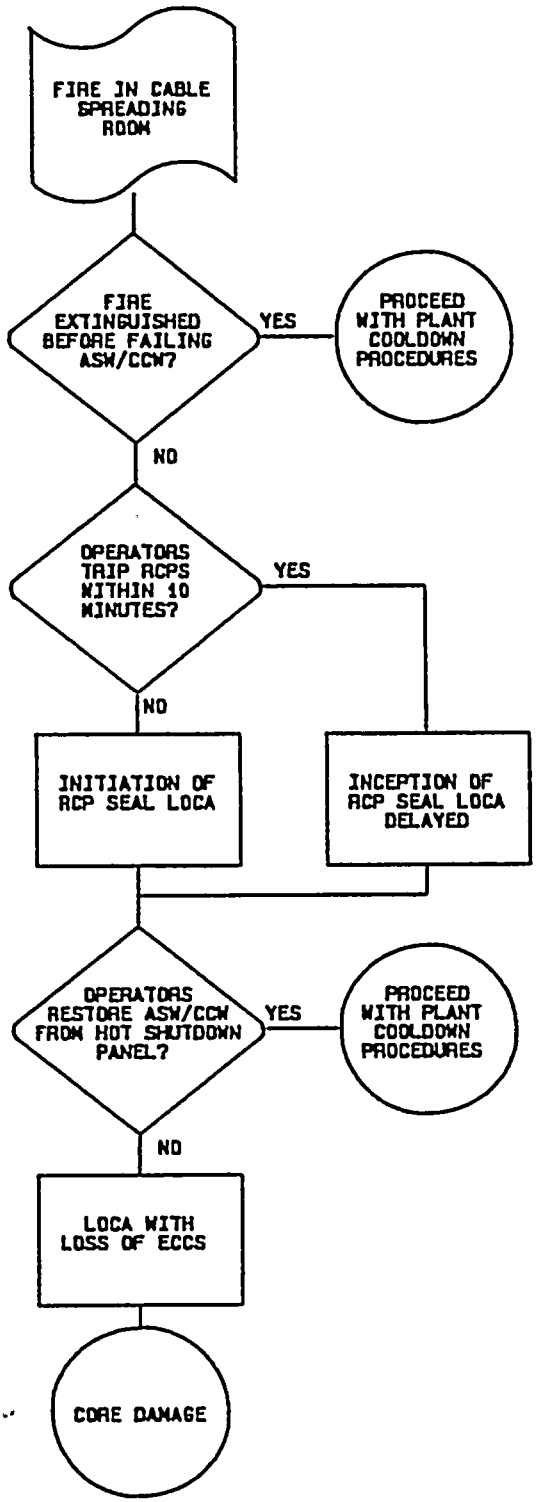


Figure F.3-5. Cable Spreading Room Fire Scenario 1

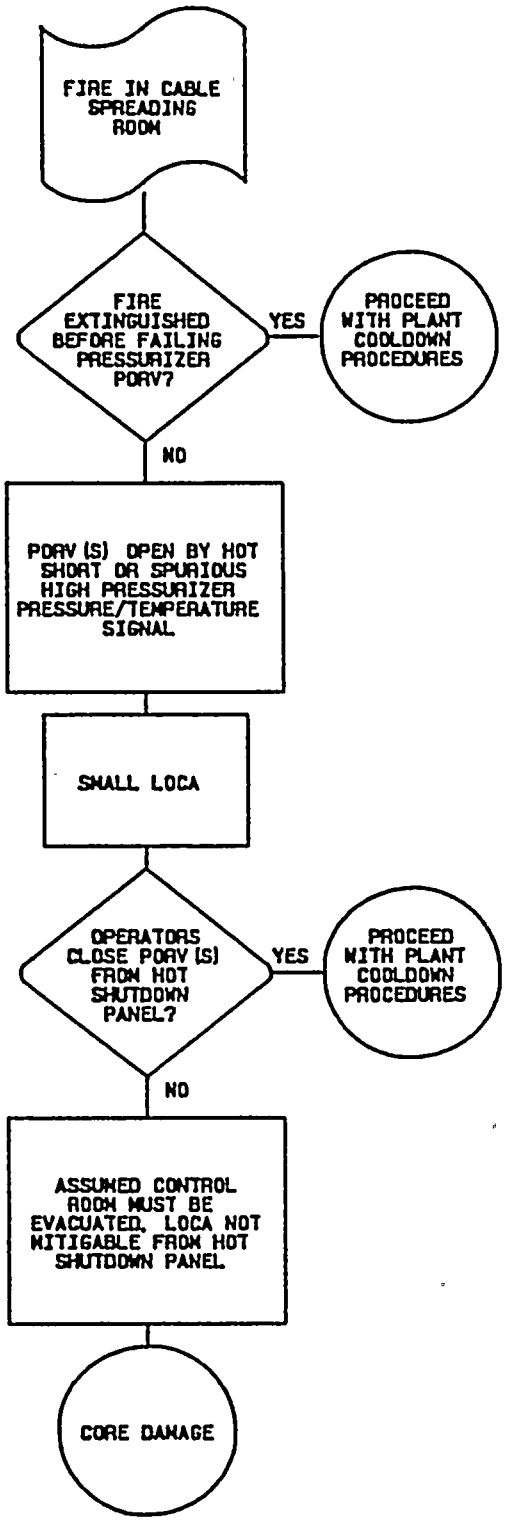


Figure F.3-6. Cable Spreading Room Fire Scenario 2

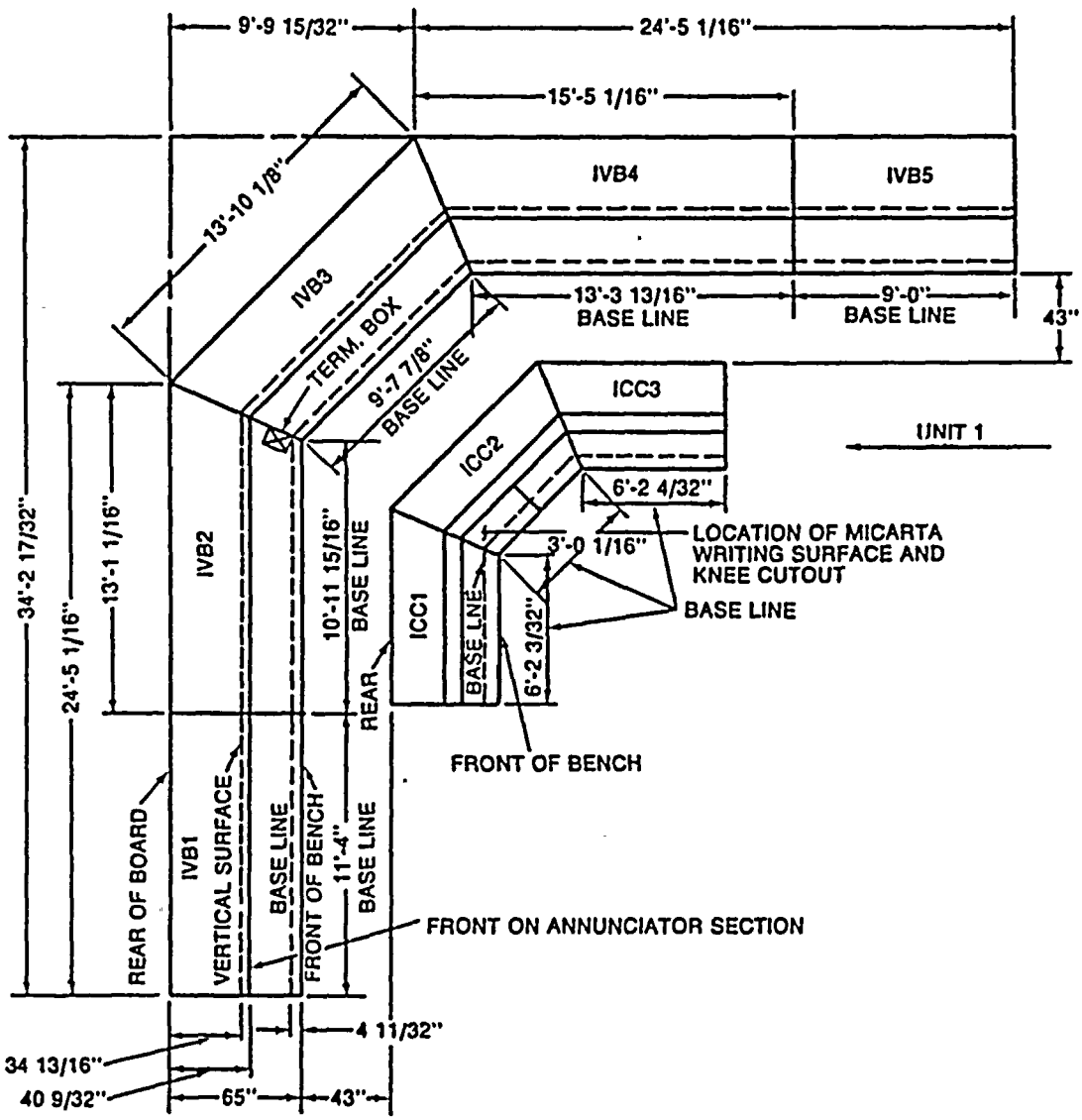


Figure F.3-7. Main Control Room Control Panel Arrangement

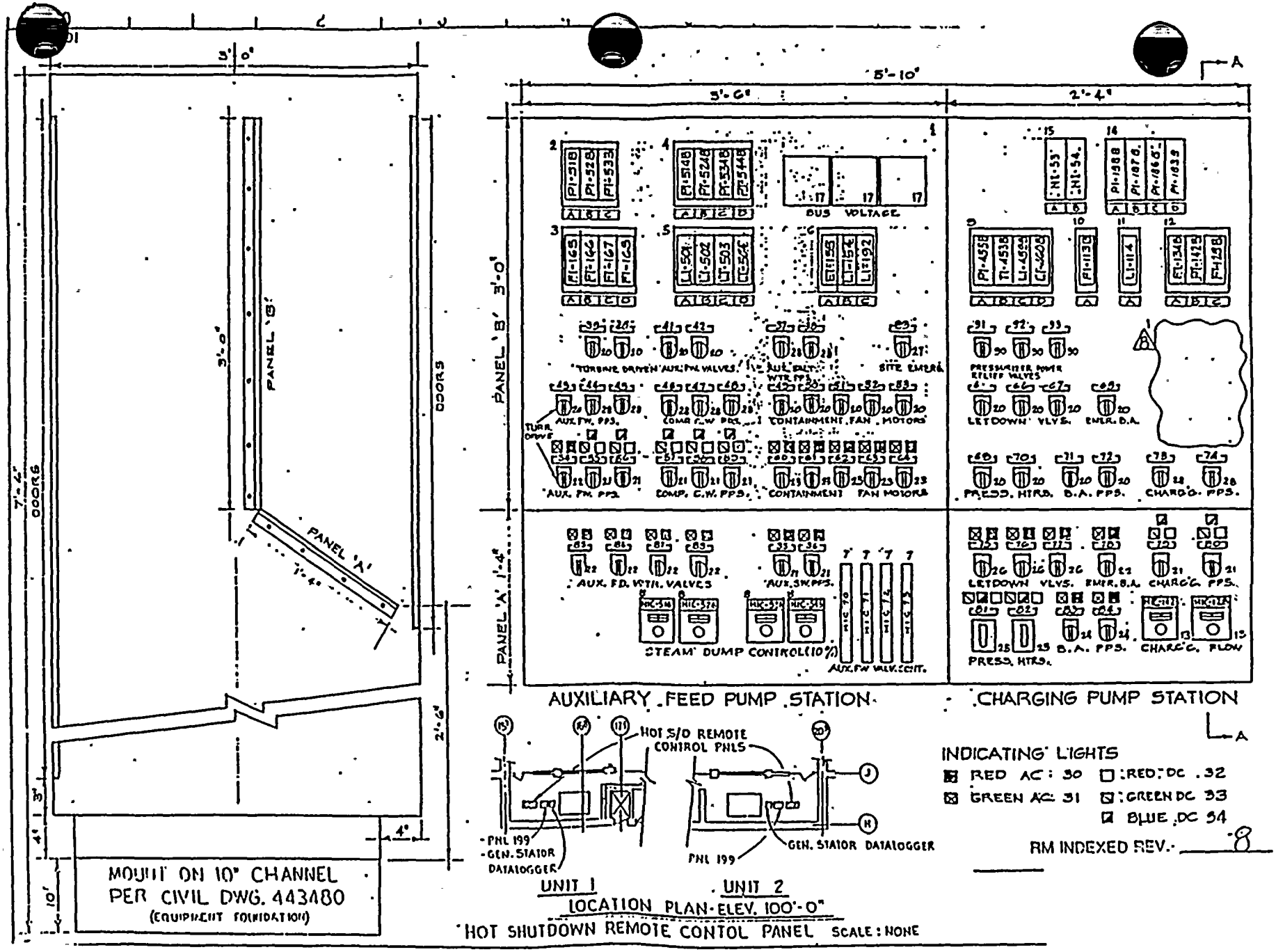


Figure F.3-8. Layout Arrangement of the Hot Shutdown Panel

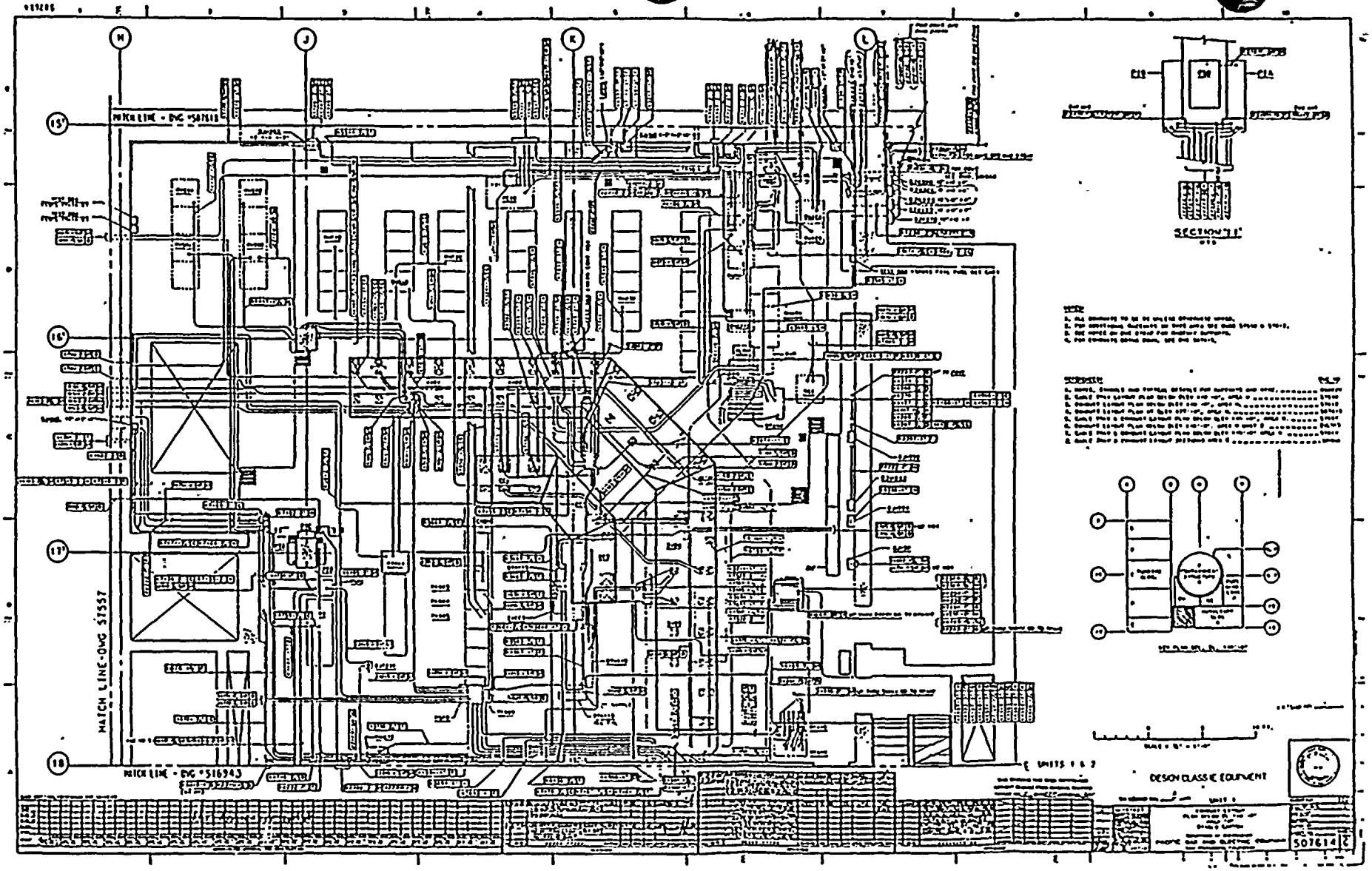
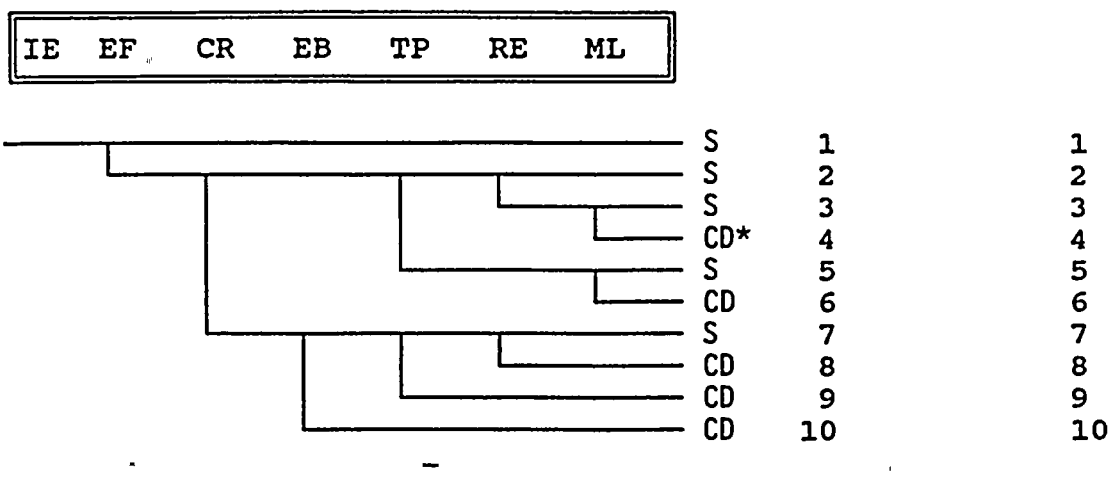


Figure F.3-9. Conduit Layout of the Cable Spreading Room



*CD = Core Damage

Top Event Designator.....	Top Event Description.....
IE	CONTROL ROOM FIRE SCENARIOS VB-1 AND VB-4
EF	EXTINGUISH FIRE BEFORE EQUIPMENT FAILS
CR	CONTROL ROOM REMAINS HABITABLE
EB	EQUIPMENT FAILS BEFORE EVACUTION
TP	RCPS TRIPED BEFORE SEALS DAMAGED
RE	RECOVERY OF EQUIPMENT PRIOR TO LOCA
ML	MITIGATION OF SEAL LOCA

Figure F.3-10. Event Tree for Control Room Fire Scenarios VB-1 and VB-4

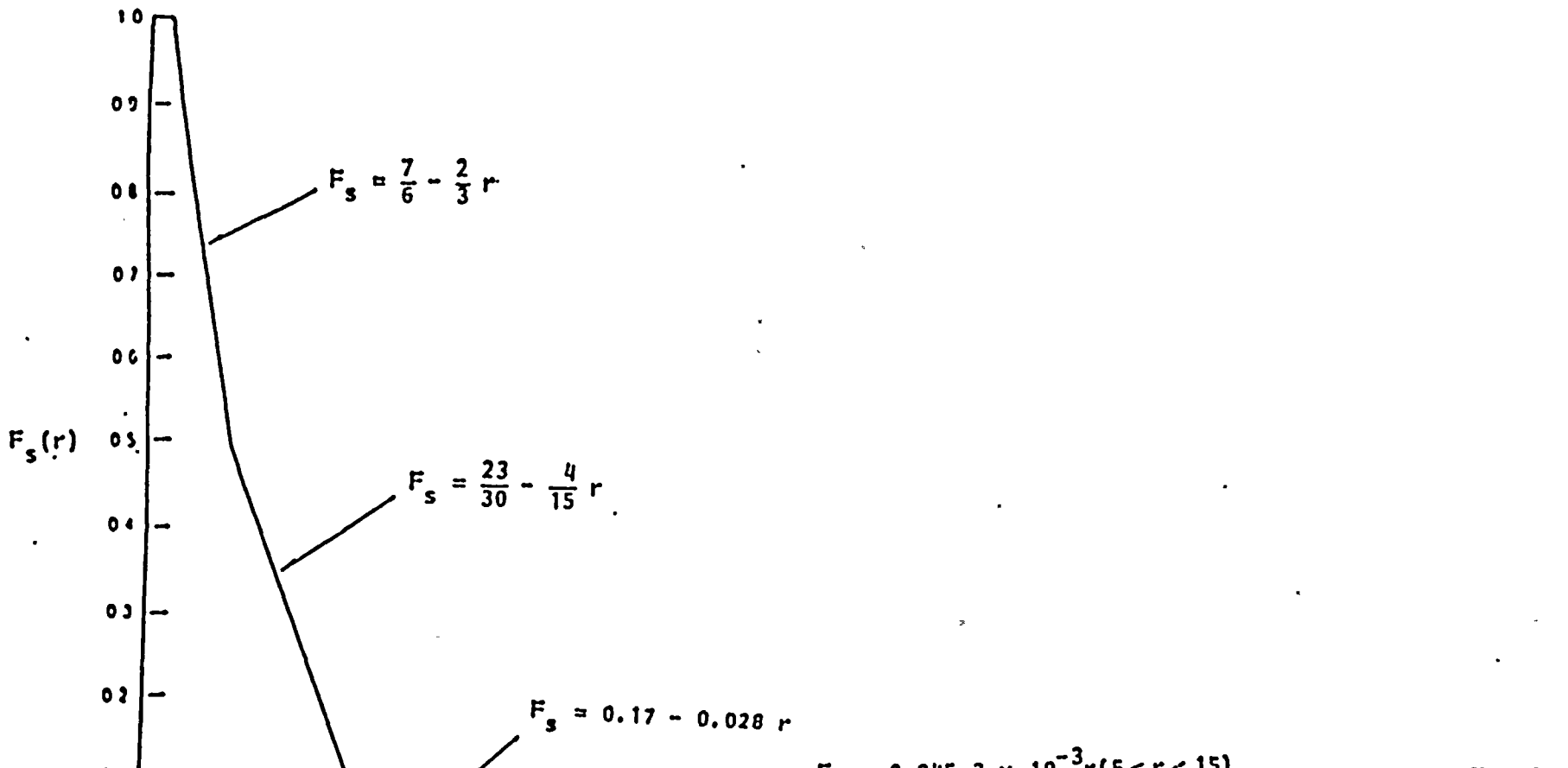


Figure F.3-11. F_s (r) for Control Room Fires

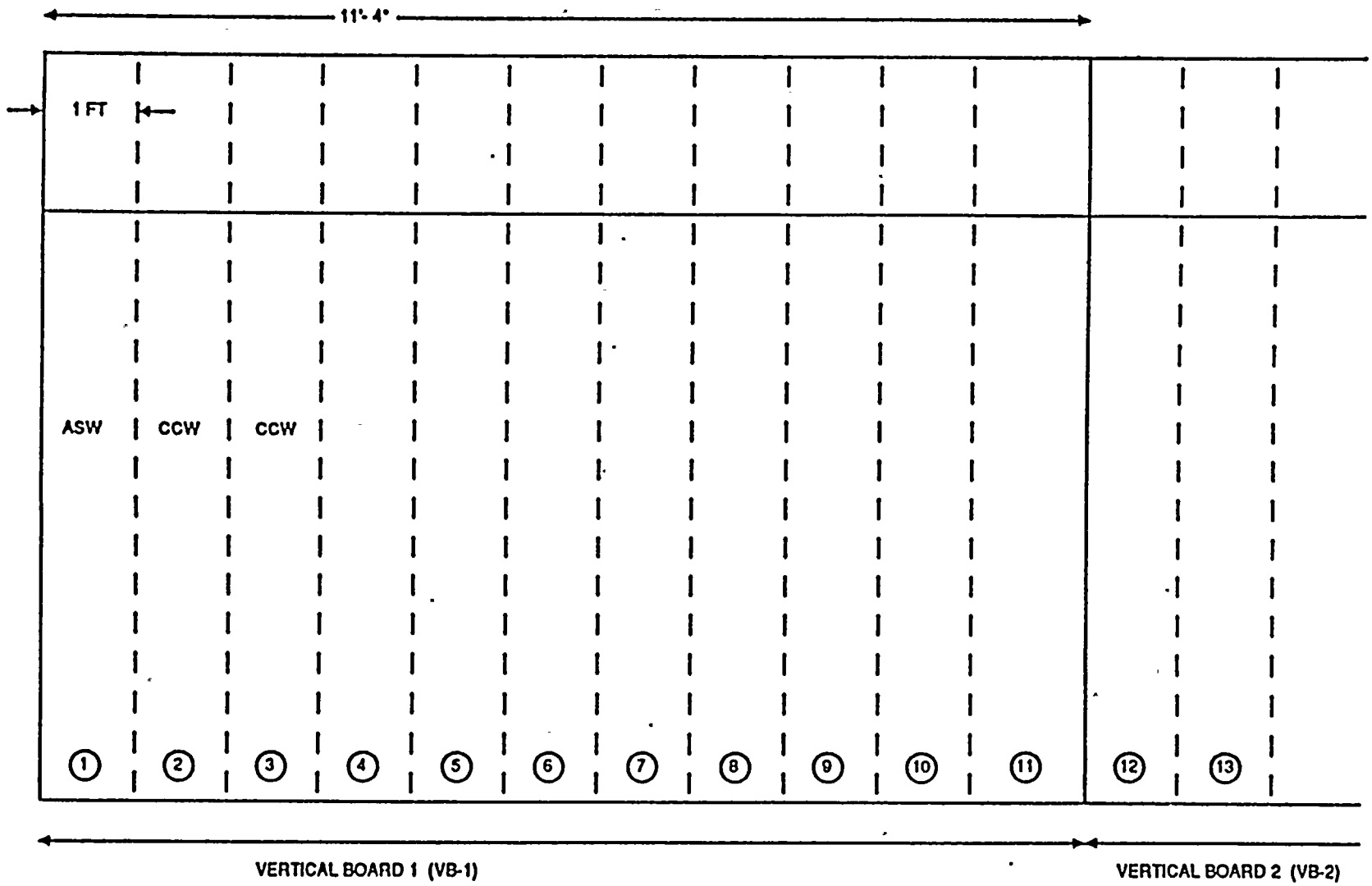


Figure F.3-12. Configuration Used In Calculating the Combined Geometry and Severity Factor for Scenario VB-1

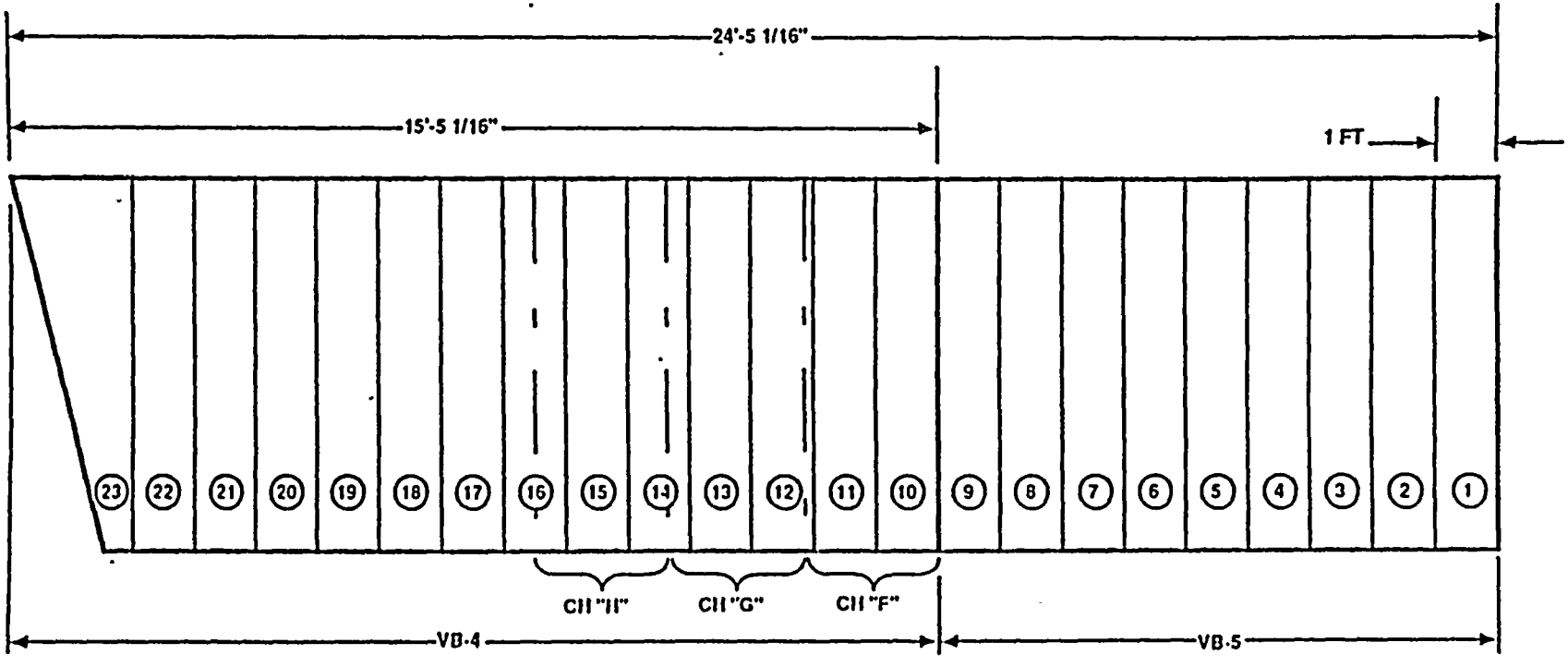
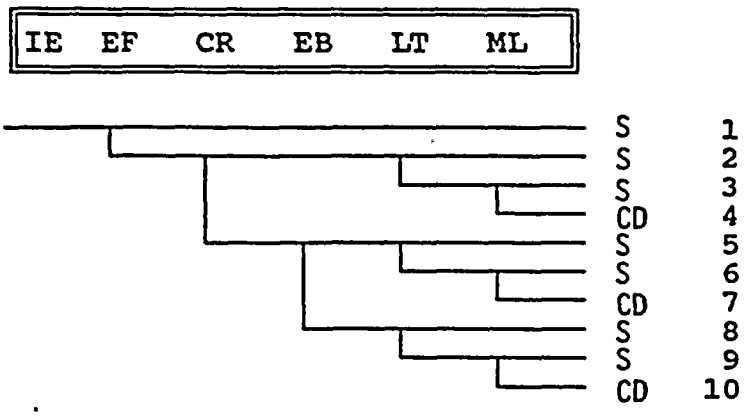


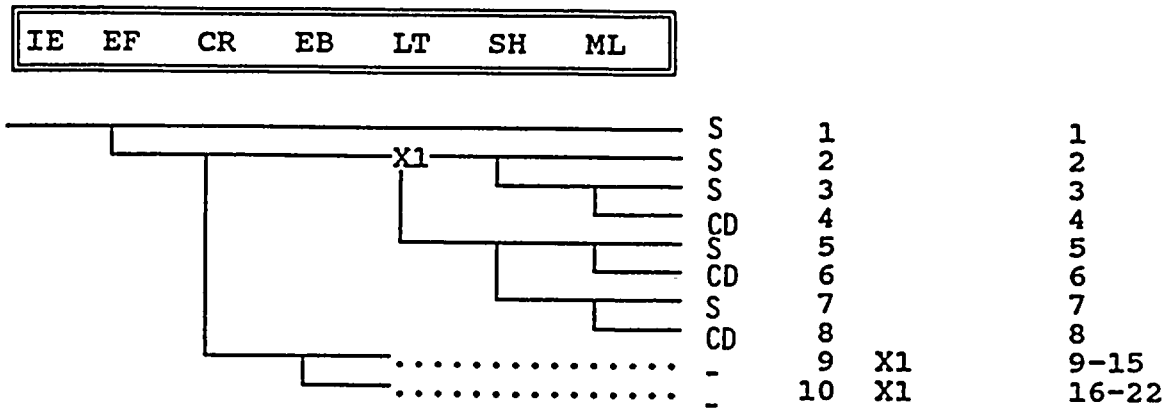
Figure F.3-13. Configuration Used in Calculating the Combined Geometry and Severity Factor for Control Room Fire Scenario VB-4



CD = Core Damage

Top Event Designator.....	Top Event Description.....
IE	CONTROL ROOM FIRE SCENARIOS VB-2A AND VB-2B
EF	EXTINGUISH FIRE BEFORE EQUIPMENT FAILS
CR	CONTROL ROOM REMAINS HABITABLE
EB	EQUIPMENT FAILS BEFORE EVACUATION
LT	LOCA TERMINATED, PORV BLOCK VALVE CLOSED
ML	MITIGATION OF LOCA

Figure F.3-14. Event Tree for Fire Scenario VB-2



CD = Core Damage

Top Event Designator.....	Top Event Description.....
IE	CONTROL ROOM FIRE SCENARIOS VB-2/3
EF	EXTINGUISH FIRE BEFORE EQUIPMENT FAILS
CR	CONTROL ROOM REMAINS HABITABLE
EB	EQUIPMENT FAILS BEFORE EVACUATION
LT	LOCA TERMINATED, PORV BLOCK VALVE CLOSED
SH	SECONDARY HEAT REMOVAL RESTORED
ML	MITIGATION OF LOCA AND/OR LOSS OF HEAT REMOVAL

Figure F.3-15. Event Tree for Fire Scenario VB-2/3

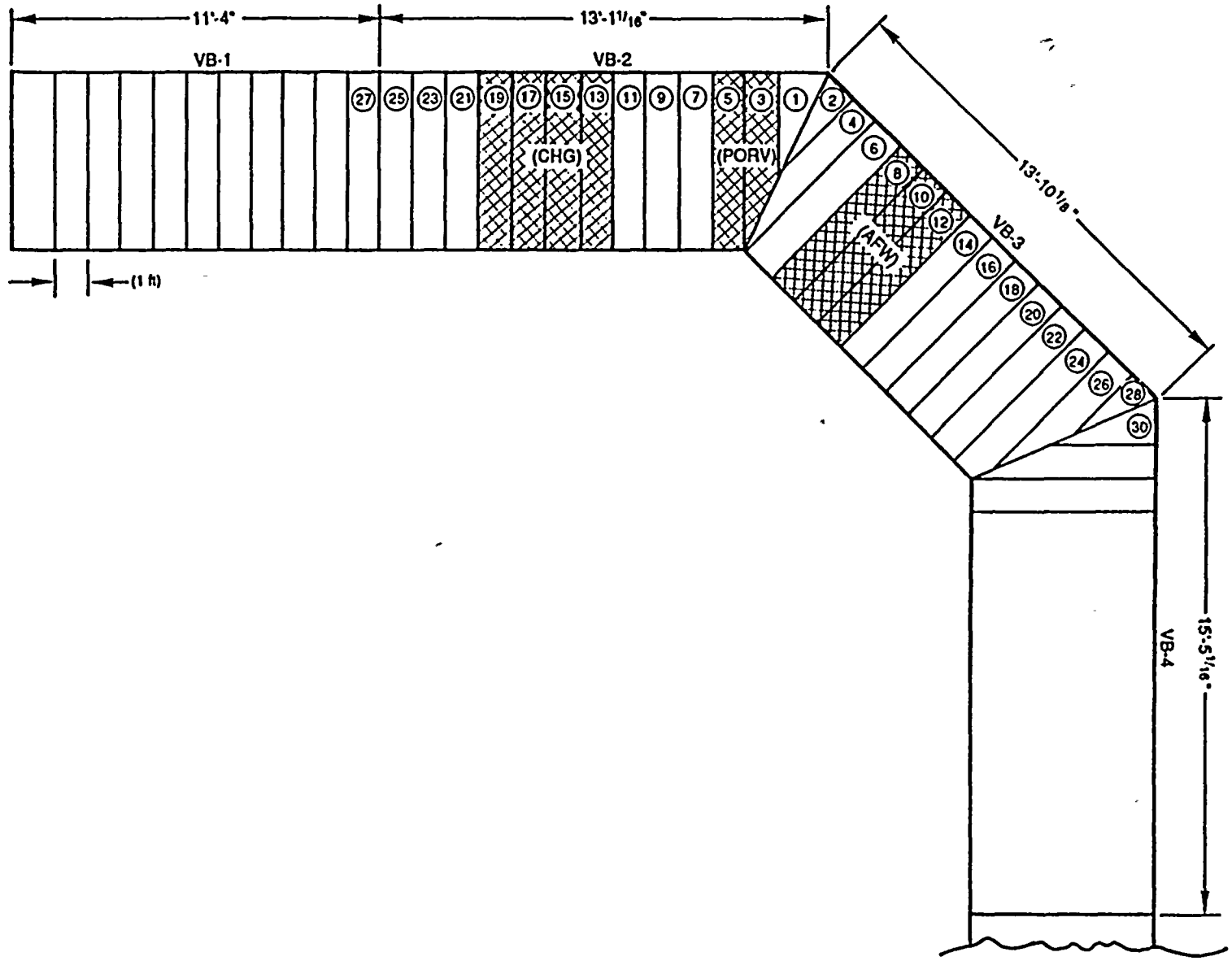


Figure F.3-16. Configuration Used in Calculating the Combined Geometry and Severity Factor for Scenarios VB-2A, VB-2B, and VB-2/3