ATTACHMENT 2 TO AEP:NRC:1082K

Donald C. Cook Nuclear Plant
Individual Plant Examination
for External Events

Revised Fire Probabilistic Risk Assessment

DONALD C. COOK NUCLEAR PLANT UNITS 1 AND 2

FIRE ANALYSIS NOTEBOOK

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REVISION 1

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NOTE

This notebook has been prepared in accordance with the applicable sections of 10 CFR 50, Appendix B, "Quality Assurance for Nuclear Power Plants and Fuel Reprocessing Plants." The documentation and the analysis reported in this notebook utilize design and plant information contained in reference documents applicable to Donald C. Cook Nuclear Plant. A list of these references is presented in this notebook.

Note that the signatures on the preceding page identify the individuals with primary preparation and review responsibilities.

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1.0 INTRODUCTION

An analysis was performed to determine fire vulnerabilities at the Donald C. Cook Nuclear Plant. A fire in the plant can potentially damage equipment and cabling which are necessary to provide decay heat removal and safe shutdown of the reactor. Many rooms or zones in the plant contain equipment or cabling required for these functions. Depending on the amount and type of equipment in the zones, these zones may have a high probability of a fire. This analysis provides assumptions, initial zones considered, fire frequency evaluation, screening assessment of zones before the walkdowns, additional screening based on walkdown findings and further evaluations, and the detailed analysis and quantification of fire-induced core damage from the remaining zones. This analysis was performed to address the requirements of Reference 1.

1.1 ACRONYMS

- AEPSC - American Electric Power Service Corporation
- AFW - Auxiliary Feedwater
- ATWS - Anticipated Transient Without SCRAM
- CCP - Centrifugal Charging Pump
- CCW - Component Cooling Water
- CDF - Core Damage Frequency
- COMPBRN - Compartmental fire modelling computer program
- EDG - Emergency Diesel Generator
- ENG - Engineered
- EPRI - Electric Power Research Institute
- EPS - Emergency Power Supply
- ESW - Essential Service Water
- FHA - Fire Hazards Analysis
- HVAC - Heating, Ventilation and Air Conditioning
- Hx - Heat Exchanger
- IEEE - Institute of Electrical and Electronic Engineers
- IEF - Initiating Event Frequency
- IPE - Individual Plant Examination
- IPEEE - Individual Plant Examination of External Events

- LER - Licensee Event Report
- LOSP - Loss of Offsite Power
- LSI - Local Shutdown Indication
- MCC - Motor Control Center
- MDAFP - Motor Driven Auxiliary Feedwater Pump
- MFW - Main Feedwater
- MOV - Motor Operated Valve
- MSIV - Main Steam Isolation Valve
- NESW - Nonessential Service Water
- PORV - Power Operated Relief Valve
- PRA - Probabilistic Risk Assessment
- PRZ - Pressurizer
- RCP - Reactor Coolant Pump
- RHR - Residual Heat Removal
- RP - Reactor Protection
- SCE - System Cutset Editor (Westinghouse software)
- SENS - Sensitivity Analysis Code (Westinghouse software)
- SI - Safety Injection
- SSCA - Safe Shutdown Capability Assessment
- SSSA - Safe Shutdown System Analysis
- TDAFP - Turbine Driven Auxiliary Feedwater Pump

2.0 BACKGROUND

Fire prevention for nuclear power plants is a major concern. The fire that occurred at the Brown's Ferry Nuclear Plant in 1975 revealed fire vulnerabilities of some plant designs. As a result of this event, the Appendix R section of 10 CFR 50 (Reference 2) was issued and required that all domestic utilities perform a fire hazards analysis, determine the consequences of a fire, and establish a comprehensive fire protection program for the plant.

As a result of the fire protection program, plants provide equipment and methods for detection, suppression, and prevention of fires. Detection can take the form of heat and ionization-type sensors located in various parts of the plant as well as fire watch personnel. Suppression can take the form of automatic water sprinkler systems, CO_2, or Halon flooding systems located in the area of concern, hose stations located near critical areas, and portable extinguishers. Also, included in the fire suppression plan is the fire brigade which is a vital part of the fire protection plan. Finally, prevention can exist in the form of methods set forth by the plant in controlling the fixed and transient combustible loadings that are located in each area of the plant. Also, the arrangement of safe shutdown equipment and cabling relative to major heat sources can prevent the start or spread of a severe consequence fire.

Initiation of a fire in any type of facility has the potential for severe consequences. The impact of a fire depends on the type and size of the fire, the means of detecting and suppressing a fire once it has started, and the possibility of the fire spreading to cause additional damage. This analysis was limited to the quantification of fire-induced core damage resulting from damage to equipment critical to decay heat removal or safe shutdown for the Donald C. Cook Nuclear Plant.
The protection of a commercial nuclear power plant from the adverse impact of fires is governed in the United States by the Code of Federal Regulations Section 10.50 (10 CFR 50 Appendix R) (Reference 2), which defines the set of rules which must be followed in fire protection design measures and the implementation of a fire protection plan. The U.S. NRC has issued Supplement 4 to GL-88-20 (Reference 1) and the associated NUREG-1407 (Reference 3), which require utilities to perform an IPEEE for internal fire events. Requirements include:

- Description of the treatment of dependencies between remote shutdown panels and control room circuitry
- Use of plant specific data
- Evaluation of suppression-agent-induced damage to equipment
- Evaluation of fire barrier integrity
- Verification of as-built cable routing
- Treatment of transient combustibles
- Determination of uncertainty sources in core damage frequency calculations
- Determination of seismic/fire interactions

This notebook discusses the above issues which are incorporated in the methodology used for this fire PRA, the prescreening process, the determination of the zonal fire initiation frequencies, general fire walkdown findings, detailed evaluation of the remaining critical fire zones, and the quantification to determine fire-induced contribution to core damage.

3.0 METHODOLOGY

This fire analysis employs both qualitative and quantitative assessments, as described below.

3.1 INITIAL ASSESSMENT

The initial assessment involved determining the zones that had safe shutdown equipment or cabling or components modelled in the IPE, and determining possible ignition sources in each zone.

Information regarding fire prevention equipment, combustible heat loadings, safe shutdown equipment, safe shutdown cabling, and fire barrier construction was reviewed. The documents containing the majority of this information are the "Fire Hazards Analysis" (Reference 4) and the "Safe Shutdown Systems Analysis" (Reference 5).

3.1.1 Initial Qualitative Screening

Table 1 identifies the fire zones that have either safe shutdown equipment and/or cabling, or any zones which have components modelled in the IPE. This table is actually Table 1 from Revision 0 of the Fire PRA (Reference 6). Of the 120 fire zones listed in Table 1, 47 were eliminated because they had no modelled equipment or safe shutdown equipment or cabling, and 8 more were eliminated because they were inside containment. Containment fire zones were not considered in this analysis because previous fire PRAs did not show that containment fires are risk significant (Reference 7). The remaining 65 zones were evaluated in greater detail.

This qualitative screening is found in Section 4.3.

3.1.2 Zone Fire Frequencies

The fire initiation frequencies were then determined for major fire areas (usually major buildings) from the generic FIREDATA data base developed by the Sandia National Laboratories (Reference 8). The plant-specific fire data for the Cook Nuclear Plant was considered but was not used to supplement the generic data because only one out of the 40 internal fire reports was issued as a Licensee Event Report (LER). The majority of the additional data was not utilized since it described fires which were self-extinguishing and did not result in damage or large monetary losses. For this reason, LERs or insurance reports were not issued. The one LER that was issued was not considered because its addition would not have significantly affected the frequencies. The additional data was not comparable to the rest of the database entries. The plant-specific data is available in Reference 6.

This major area fire initiation frequency was then divided among the fire zones based on similarities between the fire initiators in the database and equipment in the fire zone. Although only 65 zones are reviewed for risk significance, fire initiation frequencies were calculated for 100 fire zones to properly distribute the plant wide fire database information. As described above, containment fires were not considered.

Since most of the fire zones are eliminated from detailed consideration in a screening process, a minimum fire initiation frequency of .001 was chosen to avoid detailed evaluation of low significance zones. When a zone which has this minimum fire initiation frequency passes the initial screening process, a more detailed evaluation of the fire initiation frequency in this zone may be performed.

The calculation of the fire initiation frequency for each zone is found in Section 4.4.

3.2 INITIATING EVENT SCREENING

The Cook Nuclear Plant IPE analyzes several separate event trees designed to model the success and failure paths that are potential from a given set of initial equipment unavailabilities. Each of the remaining 65 zones were reviewed to determine which event initiation frequencies could be impacted if a fire damaged all equipment in a zone. At a minimum, all zones are assumed to initiate a plant trip (typical transient).

A set of conditional initiating event frequencies were calculated due to random failures, assuming important equipment was disabled in a fire. These calculations are described in Section 4.5, and the results are listed in Table 10. These calculations are based on the IPE initiating event frequency calculations (Reference 9).

If the event initiation frequency for sequences other than a typical transient could be affected, the fire initiation frequency for that zone was divided among the sequences. The event initiation frequency for other sequences was calculated by multiplying the zone's fire initiation frequency by the conditional (random) initiating event frequency determined to be possible for that zone. For example, if a zone contained the cables for both component cooling water (CCW) pumps, the event initiation frequency for a Loss of CCW would be the fire initiation frequency for that zone (conditional probability of a Loss of CCW with both pumps gone = 1). If only one train of CCW was in the zone, however, the fire frequency would be divided among a Loss of CCW and a typical transient. The value used for the Loss of CCW would be the fire initiation frequency multiplied by the conditional initiating event frequency for a Loss of CCW, with one train lost due to fire (see Table 10). The initiating event frequency for the typical transient would be the difference between the fire initiation frequency and the Loss of CCW initiating event frequency found for this zone.

The review and identification of event sequences to be considered is found in Section 4.2. Twenty one zones were found to have an impact on event initiation frequencies other that a typical transient.

3.3 INITIAL QUANTITATIVE ASSESSMENT

Assuming fire induced failure of all equipment and cabling in a fire zone, the core damage frequency was calculated for each sequence initiator for the 65 remaining zones. If the calculated core damage frequency was greater than the FIVE methodology cutoff of 1.E-6/yr (Reference 7), the zone was identified for further review. If the calculated core damage frequency was between 1.E-7/yr and 1.E-6/yr, it was identified for reporting purposes only. If the zone core damage frequency was less than the reporting criteria of 1.E-7/yr (Reference 10), it was screened from any further consideration.

For fire zones where the TRA initiating event applied, an estimated CDF was calculated three different ways. First, the transient screening performed in the original analysis is still valid. In those screenings, the Westinghouse System Cutset Editor (SCE) (Reference 11) was used to calculated estimated CDF by damaging all of the SSSA equipment and cabling as well as components modelled in the PRA in a given fire zone. A review of those earlier results found them to be consistently conservative. The SCE results were adjusted for the new fire initiation frequency used in this analysis.

The original SCE runs used an early version transient quantification that was conservative compared to the final transient quantification. For fire zones where the SCE analysis did not

result in screening, a new Westinghouse Windows based software called SENS (Reference 12) was used to calculate the estimated CDF with all the SSSA equipment and cabling, as well as, PRA modelled equipment unavailable in a given fire zone. The final transient quantification was used as input to this code. Like SCE, the SENS code is a cutset editor. In this code, combinations of specific basic events are given a failure value of one to calculate a change in core damage frequency. This sensitivity analysis code handles more cutsets and provides a simpler user interface than the earlier version. To use this software with the TRA initiating event, a specific quantification run (Reference 13) was performed by decreasing the cutset cutoff value by two orders of magnitude, which produced a file TRA.OUT with approximately 4000 cutsets to use as input to the SENS code. This special TRA.OUT was developed to prevent truncating important basic events. Using the TRA.OUT with the SENS code, conditional core damage frequencies for a fire zone were determined by setting SSSA components, cable and PRA modelled components associated basic events to 1.0 which fails the basic event. The fire induced core damage frequency is determined by normalizing this results by dividing the TRA initiating event frequency of 3.8 and multiplying by the zone fire initiation frequency.

The last method used in performing the screening assessment on fire zones where TRA initiating events applied was based on engineering judgment. These were zones where no Westinghouse SCE runs from the original analysis had been performed. For each of the fire zones evaluated (Appendix B), all of the SSSA components, cabling and PRA components were identified and reviewed for potential impact on the TRA event based components located in the zone and on previous experience performing quantitative screenings with SENS. Fire zones were then screened out in this portion of the analysis by inspection if the estimated CDF would be well less than cutoff of 1.0E-07/yr.

This zone screening assessment is found in Section 4.6.

3.4 DETAILED REVIEW OF SIGNIFICANT ZONES

Fire zones in the initial quantitative screening that did not meet the FIVE Methodology screening criteria of <= 1.0E-06 were considered to be high priority and were evaluated in this portion of the analysis. Of the fire zones evaluated in the initial quantitative screening, only 23 fire zones required further detailed evaluation.

As part of the detailed evaluation, walkdowns were performed for each of the fire zones to determine locations of transient combustibles, important SSSA cables and components, and the potential of fire propagation from zone to zone.

The detailed evaluation of these zones consists primarily of more accurate assessments of the probable extent of fire damage and the impact of the fire damage on the power and control cables (i.e., evaluation of the failure mode, such as fail to open or loss of control). When critical equipment is found not to be damaged by a fire, or could not be damaged by the same fire that damages a second important piece of equipment in a given zone, the core damage frequency is reassessed. This process is continued until the core damage frequency is sufficiently low to remove the zone from further consideration, or further reduction is not obtainable.

Assumptions commonly used for this stage of the evaluation are summarized in Section 4.1, and the techniques used are summarized in Section 4.7. The actual evaluation of each zone is found in Section 4.8.

3.5 DUAL UNIT CONSIDERATIONS

Dual unit items and actions which were credited in the internal events analysis were considered in the fire analysis. These are described in the individual notebooks comprising the internal events analysis. It was found that there were no additional considerations beyond those described in the internal events analysis.

3.6 STATUS OF APPENDIX R MODIFICATIONS

Cook Nuclear Plant had an Appendix R audit in 1990. All pertinent documentation was updated to support this audit. All modifications made as a result of and exemptions to the Appendix R ruling are included in this updated documentation, which was used in this analysis.

4.0 ANALYSIS

4.1 ASSUMPTIONS

The following assumptions were used in this analysis:

1. It was conservatively assumed that a fire in any zone would cause a reactor trip.

2. Fire zones were considered for investigation in this analysis only if they contained safe shutdown equipment and/or cabling, components necessary for decay heat removal, or other components modeled in the IPE. Zones containing neither of these were not considered critical to core integrity.

3. The safe shutdown equipment and cable routing information was taken from the Safe Shutdown Systems Analysis, SSSA (Reference 5). The SSSA was revised in late 1990 in preparation for the 1990 Appendix R audit. AEPSC reviewed a sample of the cables, and found the cable routings to be acceptable. The results of this verification are documented in Reference 14.

4. A 24-hour period was assumed as the base mission time for this analysis. This time is consistent with the time assumed in the internal events analysis, per NUREG-1335 (Reference 10).

5. The fire frequencies used in this analysis were determined using the generic FIREDATA data base developed by the Sandia National Laboratories (Reference 8).

6. Fire-induced disabling of the control room HVAC was not assumed to result in control room inhabitability because the control room is constantly manned and a cooling failure would be noticed and corrective action initiated in a timely manner. Likewise, component failures arising from failure of the control room HVAC was not considered credible. This assumption is supported by information in Reference 15 which states that, with a complete failure of control room HVAC, operator actions such as opening doors and configuring portable fans would delay control room temperature increase sufficiently to allow safely taking the plant to Mode 5 before the control room becomes uninhabitable.

7. 1E-6/yr was selected as a screening value to identify conservatively calculated sequence frequencies which could lead to core damage. (Reference 7)

8. Propagation via items which were granted exemptions to the Appendix R ruling (Reference 16) (i.e., unrated ceiling and floor hatches) was qualitatively examined. After careful review of these exemptions, it was concluded that they would have no significant impact on the Fire PRA.

9. It was assumed that fire barriers will remain intact for fires of less than rated duration. (e.g., a 3-hour fire wall will withstand a one hour fire.)

10. The delineations and boundaries employed in the Appendix R analysis (fire areas and zones) were used in this analysis.

11. Fire-induced opening of the Pressurizer PORVs was not assumed to result in a non-recoverable small LOCA because the failure of the PORV's control cable due to fire can be recovered in a relatively short period of time. This assumption is supported by information in Reference 17, which states that, with a failure of the PORV due to "hot shorts" and a resultant "open circuit" which would close the air operated PORVs, termination of the small LOCA would occur within 30 minutes of its occurrence. The "open circuit" is the dominant failure mode because all conductors will eventually contact the grounded cable tray. The "hot short" cannot exist with grounded wires. Therefore, it will eventually lead to "open circuit" as more insulation decomposes or melts away.

    The small LOCA event was analyzed as part of the IPE PRA Level II containment analysis (Reference 18). The results of the analysis showed that, for a small break LOCA with nothing available, the core uncovered at about 1 hour.

    In addition, procedures are in place that would direct operators to the take the required actions necessary to close the PORVs (Reference 19).

12. The COMPBRN IIIe code (Reference 20) was used in this analysis to verify engineering judgement of extent and timing of damage. This computer code addresses NRC concerns identified in the Fire Risk Scoping Study (Reference 21).

13. Fires originating in electrical cabinets, including switchgear, transformers, inverters, distribution panels, fire protection panels, I&C panels, etc. are assumed to stay in the cabinet of origin. This is consistent with the EPRI Fire Events DataBase, and Sandia National Lab tests (Reference 7). Thus, cabinet fires do not propagate and the loss of equipment function due to a cabinet fire is limited to the loss of the equipment supported by the cabinet. Similarly, if a cabinet contains internal partitions which completely separate cabinet sections without through-wall penetrations, a fire originating in one section is assumed not to disable equipment located in adjacent sections. This approach is used in NUREG-1150 (Reference 22).

14. Given that the industry fire experience shows real transients (i.e. combustibles not permanently or semi-permanently stored in an area) are insignificant as fire sources (mops, notebooks, single cardboard boxes) and the Cook Nuclear Plant has a very strong housekeeping/combustible material control process, true transient combustibles are judged to be insignificant for fire risk. This is consistent with NRC observations in SECY-93-143, May 21, 1993. Cook Nuclear Plant fire zones are inspected on a weekly basis with only 5 to 6 condition reports issued per year as result of transient combustibles being in inappropriate locations.

It is assumed that the only sources with risk significance are those which are semi-permanently stored in fixed locations, and whose presence is evaluated and tracked by permits issued by the fire protection engineer.

15. For cable in conduit and trays that meet IEEE 383 standards, cable function is lost if the cable temperature reaches 700°F (Reference 7).

16. Cables that meet IEEE 383 standards are assumed to ignite at 931°F (Reference 22).

17. No fire propagation for cable in conduit is considered reasonable.

18. Cables in cabinets and control panels are assumed to ignite and fail. This assumption is considered to be very conservative based on industry experience with cabinet fires.

19. "Hot Shorts" are not considered in power cables. Three phase hot shorts across the proper phases must occur for inappropriate motor operation.

20. Hot Shorts are considered significant for control cables both in cable trays and control cabinets, unless the cable tray/cabinet wiring layout make shorts unlikely. For Cook Nuclear Plant, hot shorts were considered on a case by case basis, depending on the fire scenario. For example, the double break circuitry design philosophy was used to take credit for valves that would not spuriously operate due to fire damage of control cables (Reference 23). This is consistently used in the Appendix R analysis for motor operated valves (MOVs) and air operated valves with the double break control circuitry design.

21. In the normal transient event, both main feedwater and the cross-tie to Unit 2 auxiliary feedwater are assumed as potentially available should the Unit 1 auxiliary feedwater should fail. Primary feed and bleed actions also constitute a success path. Since the cable routing for main feedwater is not explicitly determined, credit for main feedwater should not generally be given for this system. Quantifications were initially performed assuming the availability of main feedwater. These results were subsequently reviewed for the impact of this assumption. Given the availability of Unit 2 auxiliary feedwater, the lack of main feedwater has little numerical impact on the screening evaluation, and this issue is only addressed for zones where this may be of particular concern.

4.2 EVALUATION OF FIRE-INDUCED INITIATING EVENTS

Initiating events which theoretically may be induced by a fire include loss-of-offsite power, MSIV closures, opening of steam relief valves, loss of component cooling water (and subsequent RCP seal LOCA), inadvertent opening of pressurizer PORVs leading to a small LOCA, off-normal pressurizer pressure, loss of feedwater, loss of condensate pumps, loss of circulating water, turbine trip, loss of control air, loss of essential service water, anticipated transient without SCRAM (ATWS), or loss of 250 V DC power. Each of these was examined to determine if a fire could induce the event at Cook Nuclear Plant or if components could be damaged that would impact the initiating event frequency in the Cook PRA.

- o Loss-of-offsite power

From the original analysis, it was determined that a fire-induced loss of offsite power at the Cook Nuclear Plant is not a credible event. Per the walkdown notebooks (References 24 and 25), it was determined that outside the auxiliary building, there is adequate spatial separation to preclude the possibility of a fire causing the loss of more than one source of offsite power. It was also found that inside the auxiliary building, there is adequate spatial separation of cabling and components to avoid fire-induced loss of offsite power.

- o MSIV closures, Opening of steam relief valves

MSIV closures or steam relief valve openings would initiate a plant trip. The responses required to mitigate these events are identical to those required to mitigate a transient (with power conversion). This event was adequately analyzed in the internal events portion of the Cook Nuclear Plant PRA, therefore, no further analysis is performed in the fire analysis.

- o Loss of component cooling water

A loss of component cooling water could possibly be induced by a fire disabling the CCW pumps or their cabling. This statement was based on an initial review of the SSSA cabling and components for each of the significant fire zones. This could cause a loss of CCW or substantially impact the initiating event frequency for this event. Thus, it was determined that the potential exists for a fire to cause a loss of CCW and, therefore, would require further detailed evaluation.

- o Inadvertent opening of pressurizer PORVs leading to a small LOCA

Prolonged opening of the pressurizer PORVs could lead to a small LOCA. The pressurizer PORVs are located inside containment. It was determined that the potential does exist for a PORV spuriously failing open due to fire damage to control cable (Hot Shorts) which in turn would initiate a small break LOCA. However, it is likely to be terminated with 30 minutes because the Hot Shorts will become open circuits. When this happens, the air-operated PORVs would close, thus terminating the LOCA (Reference 17). The small LOCA event was analyzed as part of the IPE PRA Level II containment analysis (Reference 18). The results of the analysis showed that, for a small break LOCA with nothing available, the core uncovered at about 1 hour.

In addition, procedures are in place that would direct operators to take the required actions necessary to close the PORVs (Reference 19).

Based on the above review, it is concluded that failing open a PORV due to a "hot short" to the control cable would not uncover the core and result in a non-recoverable small break LOCA.

- o Off-normal pressurizer pressure

Pressurizer pressure signal cables were not traced as part of the Appendix R effort. It was conservatively assumed that a fire will damage these cables. The

consequence of more than two pressurizer pressure signal cables being damaged is a reactor trip, since a trip is generated on 2/3 logic (Reference 26). The ensuing event would proceed as a transient (Reference 9). The transients accident sequence (with power conversion) can be used to model this fire-induced event.

- Loss of feedwater, loss of condensate pumps, loss of circulating water

  Cables for these systems were not traced as part of the Appendix R effort. Fire-induced damage to the components (or their cables) could initiate a reactor trip. The ensuing responses required for safe shutdown are identical to those modelled in the transients (without power conversion) accident sequence.

- Turbine trip

  A fire could initiate a turbine trip, and the ensuing responses required for mitigation are identical to those modelled in the transients accident sequence (with power conversion).

- Loss of control air

  Loss of control air would result in the feedwater regulating valves closing, thus causing a reactor trip. Thus, a loss of control air is actually a transient with steam conversion unavailable (the feedwater system is considered unrecoverable). Either auxiliary feedwater or primary bleed and feed is required to prevent core damage. Although control air is lost, the pressurizer PORVs are supplied by air bottles to perform their required function. Therefore, unique consideration was not given to the loss of control air.

- Loss of essential service water

  A loss of essential service water could possibly be induced by a fire disabling the SSSA ESW cables and components. This statement was based on an initial review of the SSSA cabling and components for each of the significant fire zones. Zones were identified where fire could cause a loss of ESW or substantially impact the initiating event frequency. It was concluded that the potential exists for a fire to cause a loss of ESW and, therefore, would require further detailed evaluation.

- ATWS

  ATWS is not considered credible as occurring concurrent with a fire, or being induced by a fire. This can be justified with the following arguments:

  Assume that a fire occurs. The question may be posed, "What is the probability of a concurrent ATWS?" Since a fire and an ATWS are independent events,

  f(ATWS) = 3.86E-6/year (Reference 9)
  p(ATWS) in 24-hour time period = f(ATWS) x 24/8760hr/yr = 1.06E-8

  conservatively estimate p(core melt) to be 1.0

  conservatively estimate f(fire) to be 1.0

Therefore,

f(fire) x p(ATWS) x p(core melt) = 1.06E-08/year

This conservative value is very low, and well below the screening value of 1E-7/year.

Consider the case where a fire induces an ATWS. This would be possible if a fire disabled reactor trip cabling or logic. Trip cabling is segregated and two concurrent and independent fires which would disable all trains of reactor trip cabling is not considered credible. In addition, a simple loss of power to the breakers will cause a reactor trip.

- o Loss of 250 V DC Power

  This event could occur if a fire were to disable 250 V DC components in the switchgear room (zone 41), a battery room, or a cable vault. This event was modelled as part of this analysis.

- o Station Blackout

  Based on an initial review of the SSSA cables and components, the potential for impacting the initiating event frequency of an SBO exists due to a loss of one train of emergency diesel generator. As a result, the SBO initiating event requires further detailed evaluation.

Conclusions:

Within a reasonable probability, the potential exists that a fire could possibility induce the following initiating events or substantially impact the initiating event frequency of these events at Cook Nuclear Plant:

- Transient (with power conversion) - TRA
- Transient (without power conversion) - TRS
- Loss of 250 V DC Power - VDC
- Loss of Component Cooling Water - CCW
- Loss of Essential Service Water - SWS
- Station Blackout - SBO

4.3 INITIAL ZONES CONSIDERED - PRESCREENING

Table 1 identifies the fire zones that have either safe shutdown equipment and/or cabling, or any zones which have components modelled in the IPE. This table is the same as Table 1 from Revision 0 of the Fire PRA (Reference 6). This determination was made based on the Safe Shutdown Systems Analysis document (Reference 5) and discussions with the systems analysts (Reference 6). Of the 120 fire zones listed in Table 1, 47 were eliminated because they had no modelled equipment or safe shutdown equipment or cabling, and 8 more were eliminated because they were inside containment. Containment fire zones were not considered in this analysis because previous Fire PRAs did not show that containment fires are risk significant (Reference 7). The remaining 65 zones were evaluated further, as described in Sections 4.5 and 4.6, for an impact on initiating events and to determine their core damage frequency contribution.

4.4 DETERMINATION OF ZONE FIRE INITIATION FREQUENCY

4.4.1 Determination of Fire Initiation Frequency for Major Areas

The major area fire frequency was determined by using the FIREDATA data base (Reference 8) developed by Sandia National Laboratories. This data base contains 354 fires from pressurized water reactor, boiling water reactor, and high temperature gas reactor nuclear plants which occurred from 1965 to May 1985. This data base separates the events by fire location into the following five general locations (The value after each area represents the total number of industry years for which data exists for that area.):

o	Reactor Building	902.2 years
o	Auxiliary Building	717.1 years
o	Turbine Building	689.4 years
o	Cable Spreading Room	799.6 years
o	Control Room	721.0 years

The Cook Nuclear Plant was divided into 6 areas for evaluation. The areas are:

- Containment
- Electrical Switchgear - Basement
- Electrical Switchgear - Upper Level
- Auxiliary Building (Including Control Room)
- Cable Spreading Areas
- Turbine Building

The FIREDATA data base was accessed to determine fires that would fit into these 6 locations. Power operation as well as some hot and cold shutdown fires were considered. The shutdown fires were considered if they were judged to be possible during power operation.

The fire frequency was determined by taking the number of fires found for each area and dividing them by the years of operating experience. Since only 5 general locations were given in the data base, they had to be 'fitted' into the six Cook Nuclear Plant zones.

In general, the operational experience for the reactor building was used for the Cook containment location, the experience for the auxiliary building was used for the Cook electrical switchgear locations and the Cook auxiliary building, the experience for the turbine building was used for the Cook turbine building, and the experience for the cable spreading room was used for the Cook cable spreading room. Some fire experience was reallocated to other locations based on the type of equipment involved.

From the data base findings, the fire initiation frequencies for each area were determined and are summarized in Table 2. They range from 2.50E-03/yr for the cable spreading room to 5.37E-02/yr for the basement of the electrical switchgear building. As described in (Reference 6), plant-specific data was not used. The detailed calculation is found in Reference 27.

4.4.2 Fire Initiation Frequency Distribution Among Fire Zones

The fire initiation frequency for each location is distribute to the various zones in that location by the type of fire initiator. All fires that were assumed to be possible in the fire location are examined and placed into an appropriate category, such as electrically initiated, welding initiated, pump initiated, etc. The equipment in each zone is then reviewed, and the fire initiation frequency for each category of database fires is distributed to the zones based on the relative amount of that equipment type in each zone.

Since the first use of the fire initiation frequency by zone is for screening of the fire zones for core damage risk significance, a minimum fire initiation frequency of 1.e-3/year is chosen. This value is consistent with the cutoff range of the fire database (between 689 and 902 years of fire experience was assumed in the original analysis (Reference 27)). If the screening indicates that any zone that uses this minimum initiation frequency is risk significant, this assumption can then be refined when the zone is evaluated in more detail. Note that the minimum value is in the accuracy range of the database for the entire location, not a specific zone.

4.4.2.1 Diesel Generator Rooms in Switchgear Basement

38.5 fires were allocated to this fire location. Note that some fires are applicable to more than one location, resulting in fractional fires being allocation to some locations. By examining diesel building fires on Table 5 (of Reference 27), 31 of the 32 fires in this table are related directly to the diesel equipment. Most of these are oil fires ignited by the hot diesels. In addition, one fire from Other Buildings (Table 7 of Reference 27) is appropriate for the diesel generator rooms. Therefore, these 32 fires will be allocated to the two diesel rooms. Other fires included in this location are 4.5 fires from transformer yard fires (Table 6 of Reference 27) and 2 additional transformer fire from Other Buildings (Table 7 of Reference 27). These will be allocated to the diesel transformer room. No diesel oil pump fires are evident in the database, so a minimum initiation frequency is used for this zone.

ZONE	IDENTIFICATION	Calculation	FIRE FREQUENCY (per year)
13	DIESEL OIL PUMP ROOM	minimum	1.0e-3
14	TRANSFORMER ROOM	6.5/717	9.1e-3
15	1CD DIESEL GENERATOR ROOM	32/2/717	2.2e-2
16	1AB DIESEL GENERATOR ROOM	32/2/717	2.2e-2
	TOTAL	38.5/717	5.4e-2

The results for the diesel generator room are summarized on Table 5.

4.4.2.2 Switchgear Rooms - Upper Level

19.17 fires were allocated to these zones. These are categorized as follows (Tables refer to Reference 27):

Switchgear Room - Table 8 - 11 fires.
6 due to welding
2 due to switchgear
3 due to buses

Transformers - Table 4 - 4.5 fires (all transformers)

Battery Room - Table 9 - 4 fires, 1 of 2 battery rooms in this location (x1/2)

* Note - the battery room in zone 55 was neglected in the original split. For this evaluation, it will be counted in that area, even though this will be double counting.

Other Buildings - Table 7 - 1 2/3 fires
1 due to transformers
2/3 due to welding/grinding

Location totals

2 - switchgear
3 - buses
6.67 - welding/grinding
5.5 - transformers
2 - battery rooms

total 19.17

Zone allocation assumptions (by reviewing Reference 4)
Switchgear is in zones 41, 42A, 42C - split by approximate floor area of gear, say 3/2/1, respectively.
Buses - Significant Buses are in 40A, 40B and to a lesser extent 41 and 42A. Split 2/2/1/1, respectively.
Welding could occur anywhere in this location, split by approximate floor area Split (40A/40B/41/42A/42B/42C/42D) (1/1/2/1/.5/.25/.25) respectively.
Transformers are in 41 and 42A - split evenly
Battery room - all to 42D

40A 4KV AB SWITCHGEAR ROOM

Buses 3 x 2/6= 1.
Welding 6.67 x 1/6= 1.1

Total 2.1
IEF = 2.1/717 = 2.9e-3

40B 4KV CD SWITCHGEAR ROOM

same as 40A

41 ENG SAFETY SYSTEMS AND MCC ROOM

Switchgear 2 x 3/6 = 1
Buses 3 x 1/6 = .5
welding 6.67 x 2/6 = 2.2
transformers 5.5 x 1/2 = 2.8

Total 6.5
IEF = 6.5/717 = 9.1e-3

42A E.P.S. TRANSFORMER ROOM

Switchgear 2 x 2/6 = .7
Buses 3 x 1/6 = .5
welding 6.67 x 1/6 = 1.1
transformers 5.5 x 1/2 = 2.8

Total 5.1
IEF = 5.1/717 = 7.1e-3

42B E.P.S. CONTROL AND DRIVE ROOM

welding 6.67 x .5/6 = .55

Total .55
IEF = .55/717 = 7.7e-4 use minimum 1.e-3

42C E.P.S. MOTOR ROOM

Switchgear 2 x 1/6 = .3
welding 6.67 x .25/6 = .3

Total .6
IEF = .6/717 = 8.4e-4 use minimum 1.e-3

42D E.P.S. (AB) BATTERY ROOM

welding 6.67 x .25/6 = .3
battery 2 x 1/1 = 2.

Total 2.3
IEF = 2.3/717 = 3.2e-3

The results for the switchgear room - upper level are summarized on Table 6.

4.4.2.3 Auxiliary Building

29.67 fires were allocated to these zones. These are categorized as follows (Tables refer to Reference 27):

Auxiliary Building - Table 10 - 20 fires.
1 due to welding or other work
12 due to electrical equipment
6 due to pumps
1 due to radwaste gas

Battery Room - Table 9 - 4 fires, 1 of 2 battery rooms in this location (x1/2)

Control Room - Table 11 . fires

Other Buildings - Table 7 - 2 2/3 fires
2 due to electrical equipment
2/3 due to welding or other work

Reactor Building - Table 2 - 2 fires
1 due to welding or other work
1 due to pumps

Location totals
2 2/3 due to welding or other work
14 due to electrical equipment
7 due to pumps
1 due to radwaste gas
2 battery rooms
3 control rooms

total 29.67

Zone allocation assumptions (by reviewing Reference 4).

Welding or other work - Since equipment is relatively uniformly distributed through the area, and work is potentially to be performed on any equipment, distribution by area is a good approximation. Since a minimum IEF of 1.e-3 will be used at least for the screening, calculation of small numbers is not needed. The total frequency due to this contributor is 2.67/717=3.7e-3. This contributor can be neglected for any zone not comprising 10% or greater of the auxiliary building area. The only zone with greater than 10% of the total analyzed area of 150,000 ft^2 is zone 69. See Table 3.

Electrical equipment - in the auxiliary building, the electrical equipment is found in hallways or vestibule areas. Again, it is relatively evenly distributed in these areas. The frequency can be approximately split by the area of these zones. Zones marked with an E on Table 3 are included in this split.

Pumps - a count of the pumps is the auxiliary building can be made, and the IEF split by this count. 55 pumps were counted in the auxiliary building zones. Since a minimum screening IEF of 1.e-3 is used, zones with fewer than 3 pumps can neglect this initiator (7 x 3/55/717= 5e-4).

Zone 1 / 5 / 6M/ 36/ 44S/ 69/
pumps 15 / 5 / 4 / 3 / 3 / 6 /

The remaining three items (radwaste gas, battery, and control rooms) can be allocated to the appropriate zones (5/106/53).

1 AUX BLDG

Pumps	7 x 15/55	= 1.9
Electrical	14 x 4.5/98	= .6

Total 2.5
IEF = 2.5/717 = 3.5e-3

5 AUX BLDG (EAST END)

Pumps	7 x 5/55	= .6
Electrical	14 x 8.6/98	= 1.2
Radwaste gas	1 x 1/1	= 1

Total 2.8
IEF = 2.8/717 = 3.9e-3

6M AUX BLDG (MIDDLE SECTION OF WEST END)

Pumps	7 x 4/55	= .5
Electrical	14 x 6.1/98	= .9

Total 1.4
IEF = 1.4/717 = 2.0e-3

6N AUX BLDG (NORTH SECTION OF WEST END)

Electrical 14 x 4.2/98 = .6

Total .6
IEF = .6/717 = 0.8e-3 use minimum 1.e-3

12 QUADRANT 2 PIPING TUNNEL

Electrical 14 x 7.8/98 = 1.1

Total 1.1
IEF = 1.1/717 = 1.5e-3

33A MAIN STEAM LINE AREA, EAST

Electrical 14 x 3.3/98 = .5

Total .5
IEF = .5/717 = .7e-3 use minimum 1.e-3

36 SPENT FUEL PIT HEAT EXCHANGER PUMP ROOM

Pumps	7 x 3/55 =	.4
Electrical	14 x 1.6/98 =	.2

Total .6
IEF = .6/717 = .8e-3 use minimum 1.e-3

38 QUADRANT 2 PENETRATION CABLE TUNNEL

Electrical 14 x 2.6/98 = .4

Total .4
IEF = .4/717 = .6e-3 use minimum 1.e-3

43 ACCESS CONTROL AREA

Electrical 14 x 4.6/98 = .4

Total .4
IEF = .4/717 = .6e-3 use minimum 1.e-3

44N AUX BLDG NORTH

Electrical 14 x 7.4/98 = 1.0

Total 1.
IEF = 1./717 = 1.4e-3

44S AUX BLDG SOUTH

Pumps	7 x 3/55 =	.4
Electrical	14 x 9.4/98 =	1.3

Total 1.7
IEF = 1.7/717 = 2.4e-3

49 HVAC VESTIBULE

Electrical 14 x 3.2/98 = .6

Total .6
IEF = .6/717 = 0.8e-3 use minimum 1.e-3

51 AUX BLDG (EAST END)

Electrical 14 x 5.4/98 = .8

Total .8
IEF = .8/717 = 1.1e-3

52 AUX BLDG (WEST END)

Electrical 14 x 11./98 = 1.6

Total 1.6
IEF = 1.6/717 = 2.2e-3

53 UNIT 1 CONTROL ROOM

Control Room 3 x 1/1 = 3

Total 3
IEF = 3/717 = 4.2e-3

69 AUX BLDG

Pumps	7 x 6/55 =	.8
Electrical	14 x 17.9/98 =	2.6
Welding	2.67 x 17.9/150=	.3

Total 3.7
IEF = 3.7/717 = 5.1e-3

106 AUX FEED WATER BATTERY ROOM #1

Battery Room 2 x 1/1 = 2

Total 2.
IEF = 2./717 = 2.8e-3

144 UNIT 1 HOT SHUTDOWN PANEL ENCLOSURE

Electrical 14 x .9/98 = .13

Total .13
IEF = .13/717 = .2e-3 use minimum 1.e-3

Remaining Auxiliary Building Zones use minimum 1.E-3 fire frequency. The Auxiliary building results are summarized on Table 7.

4.4.2.4 Cable Spreading Area

2 fires were originally allocated to these zones. One was due to a transient source (candle) and the other was due to a breaker. A room of 250 VDC batteries is located in the switchgear cable spreading area (Zone 55) (by reviewing (Reference 4)). This should have been included in this area, for an additional two fires.
Zones 56 and 57 have no electrical equipment, just cabling. The transient source will be evenly split to the three zones. The breaker and battery fires to zone 55 (see 4.4.2.3).

The original analysis used 799 experience years for cable spreading. The auxiliary building experience years of 717 will be conservatively used here since fires were added from auxiliary building battery rooms.

55 SWITCHGEAR ROOM CABLE VAULT

Total 3.33
IEF 3.33 /717 = 4.6e-3

56 AUXILIARY CABLE VAULT

Total .33
IEF .33 /717 = .5e-3 use minimum value of 1.e-3

57 CONTROL ROOM CABLE VAULT

same as 56

The results for the cable spreading rooms are summarized on Table 8.

Note that potential ignition sources such as candles are carefully controlled at the Cook Nuclear Plant. There is no significant, credible ignition source for zones 56 and 57, or for the region of Zone 55 outside of the battery area. Since significant important cabling is in these zones, the fire initiation frequency is refined here for these areas for use after the screening. The screening will use the screening minimum value above for consistency.

The transient source (candle) assumed in the original analysis is not plausible at the Cook Nuclear Plant. By reviewing the FIVE fire initiation frequency methodology, only two potential fire initiators were identified, general transient (which would include candles), and unqualified cabling.

Most cabling in the plant is qualified to either IEEE-383 or IPCEA standards (Reference 60). The fire ignition frequency due to the remaining small amount of unqualified cabling would provide a negligible fire initiation frequency.

Access to the cable spreading zones is strictly controlled, and none of the transient ignition sources (cigarettes, heaters, etc.,) are allowed in the zones during power operation. However, clearly some fire initiation is plausible in any area. Therefore, to establish a plausible fire initiation frequency, one weighing factor of the transient initiator is assumed for each zone. Thus, the fire initiation frequency is the plant wide frequency times the weighing factor divided by the number of evaluated zones, or 1.3E-3*1/100=1.3E-5/yr for each of the three cable spreading areas. This evaluation is sufficiently conservative to cover any additional contribution from the unqualified cabling discussed above.

4.4.2.5 <u>Turbine Building</u>

28.67 fires were allocated to these zones. These are categorized as follows (Tables refer to Reference 27):

This calculation uses 689 experience years for these zones consistent with the original calculation.

Turbine Building - Table 13 - 21 fires.
- 3 due to welding or other work
- 3 due to electrical equipment
- 7 due to hydrogen or gas systems
- 8 due to oil leaks/ pumps

Other Buildings - Table 7 - 3 2/3 fires
- 1 due to oil on hot piping
- 1 due to hydrogen
- 1 2/3 due to welding or other work

In addition, 1 fire was included from a security building fire, and 1 from a pump room fire.

Reactor Building - Table 2 - 2 fires
- 2 due to oil leaks/hot equipment

Location totals
- 4 2/3 due to welding or other work
- 3 due to electrical equipment
- 12 due to oil/ pumps
- 8 due to hydrogen or gas systems
- 1 due to security equipment

total 28.67

Zone allocation assumptions (by reviewing Reference 4).

Generally, welding/work is assumed to be distributed by floor area. The water intake area (zone 143) was ignored since it is a large open area. Also, the large turbine building floor (Zone 129) was scaled down by 1/5 to adjust for less equipment. With these adjustments, equipment and therefore work is assumed to be well distributed in these areas. Zone 77, the welding shop, is assumed to have 1 2/3 of the 4 2/3 fires.

Electrical equipment fires are assumed to be distributed in areas judged to have significant equipment of that type, by floor area since the equipment is generally evenly distributed in those areas.

Oil/pump fires are distributed by floor area in areas containing significant oiled equipment, including turbine/generator bearing and the auxiliary boiler.

Hydrogen fires dominate the hydrogen or gas systems fire initiator. Areas with hydrogen or waste gas equipment are selected.

The 1 security equipment fire is assumed to occur in zone 91, which has a small security room which may have similar equipment.

The spreadsheet used to calculate the fire frequencies by zone is found on Table 3. The results for the turbine building are summarized on Table 9.

4.5 ACCIDENT SEQUENCE INITIATION FREQUENCY FOR ZONE SCREENING

Each of the 65 zones that had safe shutdown equipment or cabling, or equipment modeled in the IPE was reviewed to identify any zones that could induce or substantially impact the initiating event frequency for any of the following events: normal transients, transients without power conversion systems, loss of 250VDC power, loss of component cooling water, loss of essential service water or station blackout (see Section 4.2).

4.5.1 Accident Sequences Considered for Each Zone

The lists of safe shutdown components and cables from the SSSA (Reference 5) were reviewed, as well as system cutset editor runs performed for Revision 0 of the Fire PRA, to determine if a fire in any of the 65 fire zones could adversely affect the component cooling water, essential service water, 250VDC, or emergency diesel generator systems. Of the 65 fire zones reviewed, 21 zones were identified that have components or cables critical to the operation of at least one of the above systems. The components or trains that could be affected by fires in the 21 zones are summarized in Table 11, with more detail given in Appendix E. The applicable initiating event frequencies for these 21 zones was determined next.

4.5.2 Frequencies of Sequence Initiation

The values used for the initiating event frequency calculations can be found in Table 10. These values are calculated in Appendix G, except for the SBO initiating event frequency for Zone 13, which is calculated in Appendix E. These were calculated based on the IPE initiating event frequency calculations (Reference 9).

For zones where more than one non-TRA initiating event was considered, event trees were used to determine the initiating event frequencies. Event trees were used for zones 6N, 15, 16, 29G, 40A, 40B, 42A, 42C, 42D, and 79. These event trees are included as Figures 1 through 10. The frequencies determined in these event trees are zone specific, as they include the probability of a fire in that zone.

For the 11 zones that did not require an event tree, the initiating event frequencies were found based on the critical components in the zone. For example, if a fire in the zone could disable one train of component cooling water, the initiating event frequency is 1.0E-02 (from Table 10). The zone specific frequency would be (1.0E-02) * (zone specific fire initiation frequency).

Table 11 is a summary of the zone specific frequencies for initiating events for all 21 zones. The fire frequencies for the zones were taken from Tables 4 through 9. See Appendix E for more information. For the 44 zones not listed in Table 11 (i.e., zones with only TRA initiating event), their zone specific TRA initiating event frequencies are equal to the fire initiation frequencies.

4.6 QUANTITATIVE SCREENING ASSESSMENT

Following the initial screening (Table 1), which left 65 zones, and the identification of zones that could affect non-TRA IEFs, the core damage frequency was estimated for each of the zones. These core damage frequencies were used for the next screening, as described in Section 4.7. The estimated core damage frequencies and the corresponding methods used are included in Table 12. Initiating event hand calculations (Appendix E), computer runs (SCE or SENS), and engineering judgement were used to determine core damage frequency. These methods are described below.

4.6.1 Events Other than Transients Initiated

Following the calculation of the zone specific frequencies of initiating events for the 21 zones impacting component cooling water, essential service water, 250VDC or the emergency diesel generators (as shown in Table 11), the core damage frequency for each zone was estimated. These calculations of core damage frequency can be found in Appendix E, and the results of these calculations are in Table 12. When TRA was also a credible initiating event (see Table 11), its corresponding core damage frequency was estimated as described in Section 4.6.2.

4.6.2 Transient Initiated Events

The assessment of transient events conservatively considered the fire initiation frequency (without consideration of fire detection or suppression) and assumed the failure of all components located in the fire zone under investigation. After quantification those zones having a core melt contribution lower than 1.0E-06/year were eliminated from a thorough inspection.

For instance, fire zone 1C, which contains the East RHR train, was eliminated from a thorough inspection because the low fire initiating frequency in conjunction with the one RHR train being disabled resulted in a core damage frequency considerably smaller than the screening value of 1E-06/year.

In performing the initial quantitative screening, a large number of zones were looked at for their impact on CDF. Many of the fire zones that were analyzed in Revision 0 of the Fire PRA using the TRA event have not changed as a result of this revision. To identify the necessary changes for this revision, each of the SCE runs from the original analysis was reviewed and baseline runs were performed using the SENS code (Reference 12) to verify the accuracy of the runs. The original SCE runs used an earlier version of the transient quantification that was conservative to the final transient quantification. This comparison showed a 1.0E-02 decrease in the TRA.OUT conditional probability for the current analysis compared to the original analysis. Therefore, the SCE runs (based on the early version of TRA.OUT) were considered to be very conservative and no new SENS runs (based on the current version of TRA.OUT) were needed for these fire zones. The final transient quantification for Revision 0 of the D. C. Cook Nuclear Plant PRA was used as input to this code.

In situations during this revision where no SCE run was found from the original analysis, a couple of different methods were used to address that specific fire zone. Initially, reviewing the fire zones, all SSSA cable and components were identified and matched with an associated modelled PRA component so that a SENS run could be performed. The initial review focused on fire zones that were felt to be dominate contributors to CDF. The results of the SENS runs on TRA events are presented in Table 12. With the first round of the SENS runs completed, the remaining TRA fire zones were identified along with the zone's associated SSSA cables and components. From the components identified, engineering judgement was used on these fire zones. Based on the similarity between fire zones from the earlier SENS runs, it could be judged that the estimated CDF would be much smaller than the 1.0E-07 reporting criteria. Appendix B provides a list of the zones that were evaluated using engineering judgement along with justification for screening out the zones. Table 12 also lists the fire zones that used these assumptions.

Of the 53 fire zones that were evaluated using the TRA event, all but two zones were found to have an estimated core damage frequency lower than 1.0E-06. These remaining zones (Zone 6M and 17C) contained the cables and components associated with the Auxiliary Feedwater System,

requiring a walkdown and further detailed evaluation. Documentation of all the estimated core melt frequency estimations for this screening exercise are located in Appendix B. Table 12 lists the TRA zones that were eliminated from a thorough inspection.

4.6.3 Screening Summary

Table 12 lists the estimated core damage frequency for each of the 65 zones and the corresponding method used to determine this value. Some zones had more than one relevant initiating event sequence (CCW, ESW, SBO or TRA). The most significant non-TRA initiating event was considered, as well as TRA. As a result, some zones have two values listed in Table 12. The more accurate value of CDF was used for the priority determination (i.e., the values in bold print in Table 12). As described in Section 4.6.2.1, the SCE runs are not believed to be as accurate as other methods.

The following guidelines were used for this priority determination:

High priority zones $CDF \geq 1E\text{-}06$
Low priority zones $1E\text{-}07 \leq CDF < 1E\text{-}06$
No priority zones $CDF < 1E\text{-}07$

Table 12 also lists the priority determination for the 65 zones. The 23 high priority zones were evaluated further in the analysis, as described in Section 5.0. The 38 no priority zones were screened out and not considered further. The remaining four low priority zones were not evaluated any further, and they were not added into the final core damage frequency due to fire. The low priority zones will be reported to the NRC in our Fire PRA submittal, however, consistent with the reporting criteria of Reference 10.

4.7 OVERVIEW OF THE EVALUATION OF SIGNIFICANT FIRE ZONES

The 23 high priority zones were evaluated further using engineering judgement, walkdowns and COMPBRN runs. In addition, the evaluation of some fire zones (e.g., control room) required the calculation of human reliability values.

4.7.1 Walkdown Findings

Walkdowns were performed at the Cook Nuclear Plant on September 8 and September 22, 1994. The walkdown participants were J. M McNanie and M. A. Wilken from AEPSC. Thirteen of the 23 high priority zones were walked down. The walkdown findings for these thirteen zones can be found in Appendix C.1. These walkdown findings were used for the final screening and quantification, as described in Section 5.3. Zone 6M was walked down at a later date, as described below. The nine zones that were not walked down were zones 15, 16, 17C, 29G, 55, 56, 57, 112 and 144. For zones 15, 16, 17C, 29G and 112, it was not necessary to have detailed walkdowns for these zones. The existing walkdown information, analyst familiarity and drawings were used to substitute the fire analysis methods used. When needed, discussions with plant personnel or fire protection engineers supplemented this information. Zones 55, 56, 57 and 144 include the control room, the auxiliary cable vault, the control room cable vault and the hot shutdown panel enclosure. They were not walked down because no helpful information would have resulted.

A walkdown was also performed of fire zone 6M, on November 10, 1994, by R. B. Bennett from AEPSC. This walkdown was done to address a concern regarding taking credit for Unit 2 AFW.

Cabling for all trains of AFW for both units pass through this zone. However, it was found that the Unit 1 cabling is well separated from the Unit 2 cabling, with no significant combustible source present. These walkdown findings are included as Appendix C.2.

4.7.2 COMPBRN Run

A COMPBRN run was performed to support the analysis assumption that the oil stored in the fire storage cabinet (in Zone 52) would not get to a high enough temperature to spontaneously ignite, even with a 1 gallon spill of oil burning adjacent to the cabinet. Appendix F contains additional information on this COMPBRN run, including the input and output files.

4.7.3 Engineering Judgement

In performing the detailed review of the remaining fire zones, engineering judgement was used to determine what the worst case fire would be and what components would fail. In most cases, a specific methodology was used such as FIVE or COMPBRN to provide verification of the components or cables that would be damaged in a given fire scenario. Section 4.1 lists the assumptions that were used in performing the detailed evaluation. Some assumptions used in this analysis were: only fire sources of risk significance are those which are semi-permanently stored in fixed locations, fires originating in electrical cabinets are assumed to stay in the cabinet of origin, and MOVs that have double break control circuitry will not spuriously operate a motor operated valve. The combination of the above techniques was used in evaluating fire zones presented in Section 5.3.

4.7.4 Human Reliability Values

Human reliability values were necessary for the evaluation of zones 53 (Unit 1 control room) and 55, 56 and 57 (cable vaults). These calculations can be found in Appendix D. Appendix D.1 contains the human reliability calculation of the operators failing to cooldown and depressurize following a loss of CCW, due to a control room fire in the service water panel. This human error probability was determined to be .025. Appendix D.2 contains the human reliability calculation of the operators failing to crosstie Unit 2 AFW and CVCS using the emergency remote shutdown procedure series (Reference 28), following a loss of all Unit 1 power and control and evacuation of the control room, due to a cable vault fire. This human error probability was determined to be 0.11.

4.8 FINAL QUANTIFICATION AND SCREENING

The following sections provide a detailed evaluation of the fire zones that did not meet the FIVE Methodology screening criteria of 1.0E-06 during the initial screening assessment. Each fire zone along with the postulated fire scenario is described in detail. The walkdown notes as well as fire zone layout drawing for each of the fire zones below are located in Appendix C. The assumptions made in the detailed evaluation are described more thoroughly in Section 4.1.

4.8.1 Fire Zone 6M - Auxiliary Building - Middle Section of West End - Both Units

Fire zone 6M contains the Boric Acid Storage tank room as well as the corridor surrounding the room. This fire zone required additional evaluation since Unit 2 auxiliary feedwater cabling passes through the zone and main feedwater cabling is not explicitly traced (see Assumption 21). Thus, all secondary cooling could fail as a result of a zone wide fire. Walkdowns performed in this zone determined that the conduit containing AFW cables enters the BAST room at

approximately 10 feet elevation, lowers to about 8 feet, and turns immediately to exit the room. The wall penetrations where each unit's conduit enters the room are about 20 feet apart, with a concrete cable tunnel separating them. The pipe tunnel extends about 6 feet into the room. This is the closest the two sets of cables get, since the cables turn toward their respective units. No combustion sources were noted in the room and, by discussion with the CVCS system engineer, no combustibles are typically ever stored in the tank room. No transient combustibles semi-permanent or permanent were found to exist in this fire zone and, based on the walkdowns, there are no significant fire ignition sources located in the vicinity of the AFW SSSA cables. The Boric Acid Transfer pumps located in this room are small enough not to be considered a significant fire risk. A review of the both unit's AFW cable conduit during the walkdown of fire zone 6M showed good separation between Unit 1 and Unit 2 auxiliary feedwater cable conduit.

Based on the above detailed evaluation, it is concluded that fire zone 6M is not a fire risk due to the limited amount of combustibles located in the fire zone (see Assumption 14). Therefore, this zone can be screened out from further analysis.

4.8.2 Fire Zone 6N - Auxiliary Bldg - North Section of West End - Unit 1

Fire zone 6N contains several MCCs that supply components that were modelled in the Cook Nuclear Plant PRA. The systems that were affected were CCW, CVCS (Charging), and ECCS (High Pressure Recirculation). In reviewing the SSSA cables and components for this fire zone, it was determined through an event tree analysis that loss of CCW was the critical initiating event for fire zone 6N (see Figure 1).

Based on walkdowns in fire zone 6N, it was concluded that a transient combustible fire in the area of MCC 1-AB-A was not a credible scenario. The transient combustibles found in this zone consisted of binders with paper around the RP desk and a covered garbage can a substantial distance from the MCC in question. No other permitted transient combustibles, such as, oils, aerosol, and cleaning fluids were found in this zone. The nearest critical component to MCC 1-AB-A would be 1-AB-D which contains the 600VAC starter for the east CCP lube oil pump which is approximately 18 feet away. Thus, it was concluded that fire damage would be confined to the critical MCC.

A fire in MCC 1-AB-A results in a loss of power to 1-AZV-A, failing WMO-737, a valve that provides ESW flow to West CCW Hx. This valve would fail as is, so the normally operating train would not be impacted. Another critical component affected in MCC 1-AB-A would be the west CCP lube oil pump. Loss of this component would result in loss of one train of high pressure recirculation. Thus, the worst case fire in this zone would result in a loss of one train of component cooling water in standby and the west train of charging for high pressure recirculation. Based on this evaluation, an analysis was performed failing the west (standby) train of CCW (WMO-737TM) and the west train of ECCS high pressure recirculation (PP-50WPS). The variables needed to calculate an estimated core damage frequency consist of fire initiation frequency (1.0E-03), initiating event frequency of loss of one train of CCW in standby (2.34E-04) and the conditional probability for a loss of CCW analysis failing the charging pump (4.54E-02). Using the above values, the results of the analysis showed that failure of these component provide a estimated CDF of 1.06E-08, which is less than the EPRI Five methodology cutoff of 1.0E-06. Therefore, this zone is not risk significant.

4.8.3 Fire Zone 15 - 1CD Diesel Generator Room

Fire zone 15 contains the Unit 1 CD emergency diesel generator (EDG), as well as the SSSA cabling traveling through the zone. The cabling located in the zone that is not directly associated with the EDG consisted of the both RHR and charging pumps, West ESW, East AFW and several MCCs of both safety trains. Discussion with Appendix R personnel and a review of each of the cables in the fire zone determined that the majority of the west train safety-related cables were embedded in concrete pilaster and surrounded by a layer of Thermolag. By way of engineering judgement, it was assumed that for any given fire scenario the cables embedded in concrete would not be damaged in a fire. Excluding these cables, a worst case fire in this zone would result in a loss of a standby train of CCW due to loss of control to pump 1-PP-10E. This is considered to be the dominate initiating event for this analysis.

Based on this review, a sensitivity analysis run was performed to determine the failure probability of loss of a standby train of CCW along with the failure other critical cables located in zone that are not in the concrete pilaster. Using the event tree analysis for loss of CCW, a sensitivity analysis was performed, failing the east train of CCW (1-PP-10E), to determine the estimated CDF. The loss of Unit 1 AB EDG was not included in this analysis since it would have little impact on the CCW event tree. The variables needed to calculate the estimated core damage frequency consist of fire initiation frequency (2.2E-02), initiating event frequency of loss of one train of CCW in standby (2.34E-04) and the output of the sensitivity analysis run failing components (5.9E-02). Using the above values, the estimated CDF was calculated to be 3.04E-07. The results of this calculation show that the estimated CDF is less than the FIVE methodology cutoff of 1.0E-06. Therefore, this zone is of low fire risk significance, but will be reported.

4.8.4 Fire Zone 16 - 1AB Diesel Generator Room

Fire zone 16 contains the Unit 1 AB emergency diesel generator (EDG), as well as the SSSA cabling traveling through the zone. The cabling located in the zone that is not directly associated with the EDG consisted of the control cabling for West RHR pump, West charging pump, West ESW, and West AFW pump. A review of each of the cables determined that the worst case fire for this zone (assuming damage to all the cables) would be a loss of a standby train of CCW, as well as the EDG. This is due to loss of control to pump 1-PP-10W. This is considered to be the dominate initiating event for this analysis.

Based on this review, a sensitivity analysis run was performed to determine the failure probability of loss of a standby train of CCW along with the failure of other critical cables located in zone that are not in the concrete pilaster. Using the event tree analysis for loss of CCW, a sensitivity analysis was performed, failing the east train of CCW (1-PP-10W), to determine the estimated CDF. The loss of Unit 1 AB EDG was not included in this analysis since it would have little impact on the CCW event tree. The variables needed to calculate the estimated core damage frequency consist of fire initiation frequency (2.2E-02), initiating event frequency of loss of one train of CCW in standby (2.34E-04) and the output of the sensitivity analysis run failing components (6.8E-02). Using the above values, the estimated CDF was calculated to be 3.50E-07. The results of this calculation shows that the estimated CDF is less than the FIVE methodology cutoff of 1.0E-06. Therefore, this zone is of low fire risk significance, but will be reported.

4.8.5 Fire Zone 17C - Corridor to AFW Pump Rooms - Both Units

Fire zone 17C is the corridor to the AFW pump rooms and contains SSSA cabling associated with the AFW pumps and valves for both Unit 1 and 2. Review of the SSSA cable failure modes for

this fire zone determined that for both Units 1 and 2 all trains of AFW would be lost in a worst case fire assuming all cables are damaged in the fire zone. Assuming an additional failure of the Main Feedwater System, since this cabling is not explicitly traced, results in an estimated CDF greater than the 1.0E-06 cutoff.

No permanent or semi-permanent transient combustibles are located in the fire zone. Therefore, the assumption that the only sources with risk significance are those which are semi-permanently stored in fixed locations, and whose presence is evaluated and tracked by permits issued by the fire protection engineer, can be used for this fire zone. It was also noted that the door to fire zone 17E is open to 17C due to steamline break concerns for the turbine-driven AFW pump room. There is a fusible link located on the door which closes at approximately 375°F (Tech Evaluation 11.40, Reference 29) which would be well below the cable damage temperatures of 700°F. Also, there are no permanent or semi-permanent combustibles stored in fire zone 17E.

Based on the above detailed evaluation, it is concluded that fire zone 17C is not a fire risk due to the limited amount of combustibles located in the fire zone. Therefore, this zone can be screened out from further analysis.

4.8.6 Fire Zone 29G - Screenhouse Motor Control Room for ESW - Both Units

Fire Zone 29G contains SSSA cabling for both units ESW systems which traverse through protected and unprotected pull boxes and conduit in this zone. For this detailed evaluation, the fire zone documentation was reviewed, as well as walkdowns performed by plant personnel to determine the approximate location of the components in the fire zone. Two non-safety related MCCs are also located in the center of this fire zone. The pull boxes are located approximately 10 feet from the floor.

Since no transient semi-permanent or permanent combustibles were found in this fire zone, it is assumed that a fire in this zone would start in one of the non-safety related MCCs. Based on the fact that fire propagation outside of a cabinet is not credible as indicated earlier, this fire would not cause damage to the ESW SSSA cabling within the zone. Therefore, based on this detailed evaluation, this fire zone is screened out from further review.

4.8.7 Fire Zone 40A - 4kV AB Switchgear Room

Fire zone 40A contains the electrical safety buses that provide 4KV power to west train components modelled in the Cook Nuclear Plant PRA. Of significance, fire zone 40A contains electrical safety bus T11A which provides power to the ESW, CCW, SI, CCP, RHR pumps, as well as, other components important to safety. In evaluating this fire zone, an event tree analysis (Figure 5) was performed to determine what the critical initiating event would be. The analysis showed that the dominant contributor would be a loss of CCW initiating event. Since no transient combustibles were found to exist in this fire zone, it is assumed that a fire in this zone starts in one of the 4KV buses. For this evaluation, it was assumed that the worst case scenario is ignition and destruction of the west CCW pump 4KV bus. In the electrical safety bus T11A, each of the individual safety buses is in a cabinet separated by metal partitions to prevent fire from spreading from one safety bus to another (Reference 30). Therefore, it is assumed that a fire within a cabinet will not create enough heat outside of the cabinet to damage cables in the adjacent safety buses (Reference 59).

Based on these assumptions, this fire scenario would result in only failing the west CCW pump bus on T11A. Based on this evaluation and, using the event tree analysis for loss of CCW, an

analysis was performed failing the west train of CCW (PP-10WPS) to obtain an estimated CDF. The variables needed to calculate the estimated core damage frequency consist of fire initiation frequency (3.2E-03), initiating event frequency of loss of one train of CCW (1.0E-02) and the output of the sensitivity analysis run failing the above components (1.65E-02). Using the above values, the estimated CDF was calculated to be 5.28E-07. Since the fire is equally likely to occur in any of the four cabinets located in this fire zone, this fire initiation frequency can further be divided since only the T11A cabinet is critical, for an estimated CDF of 1.32E-07.

4.8.8 Fire Zone 40B - 4kV CD Switchgear Room

Fire zone 40B contains the electrical safety buses that provide 4KV power to east train components modelled in the Cook Nuclear Plant PRA. Of significance, fire zone 40A contains electrical safety bus T11D which provides power to the ESW, CCW, SI, CCP, and RHR pumps, as well as other components important to safety. In evaluating this fire zone, it was determined, based on walkdowns, that the assumptions performed for fire zone 40A hold true for fire zone 40B.

Based on this evaluation and using the event tree analysis for loss of CCW, an analysis was performed failing the west train of CCW to determine the estimated CDF. The variables needed to calculate the estimated core damage frequency consist of fire initiation frequency (3.2E-03), initiating event frequency of loss of one train of CCW (1.0E-02) and the output of the analysis run failing the above components (2.33E-02). Using the above values, the estimated CDF was calculated to be 7.45E-07. Since the fire is equally likely to occur in any of the four cabinets located in this fire zone, this fire initiation frequency can further be divided since only the T11D cabinet is critical, for an estimated CDF of 1.86E-07.

4.8.9 Fire Zone 41 - Engineered Safety System & MCC Room - Unit 1

Fire zone 41 contains the 600VAC buses, transformers, MCCs and AB battery chargers used to support safety and non-safety related electrical equipment to support the Cook Nuclear Plant. Many of these components were modelled in the PRA. During the walkdown of this fire zone, no transient combustibles, permitted or otherwise, were found in the area. Based on these findings, it was assumed that, if a fire would occur in this zone, it would be located in one of the electrical cabinets and would not propagate outside of the cabinet.

Based on reviewing the Cook Nuclear Plant PRA, it was determined that the worst case fire in this zone would take out one the 600VAC safety buses or its associated transformer. The 600VAC bus 11D was found to be the critical component in modeling a fire in this zone since it supports CCW components. Since these components are typically motor operated valves, the failure of power to these valves would result in a "fail as is" failure mode. Loss of power to a normally running train of CCW would not affect the system. The train of CCW in standby would be affected by this failure mode. As a result, this analysis will assume that bus 11D is supporting the CCW train that is in standby. The CCW initiating event is considered to be the dominant contributor to CDF in this scenario. Bus 11D, as modelled in the Cook PRA, provides 600VAC power to west train motor operated valves. The worst case fire scenario in this bus would be a fire that damages the whole bus 11D, resulting in a loss of 600VAC (1 train) and loss of one train of CCW. This failure would only impact the train that is in standby. The CCW initiating event frequency for a loss of one train of CCW in standby was calculated to be 2.34E-04 (Table 10).

Using the above information, an analysis can be performed to determine the impact that a loss of bus 11D would have on the core damage frequency. The fire initiation frequency was calculated to be 9.1E-03. The conditional probability for a loss of CCW analysis failing bus 11D is 5.27E-02. The estimated CDF for fire zone 41 is as follows.

Est. CDF = 2.34E-04*9.1E-03*5.27E-02

= 1.12E-07

4.8.10 Fire Zone 42A - Electrical Power System Transformer Room - Unit 1

Fire zone 42A is very similar to fire zone 41 in that 42A contains the opposite train 600VAC buses, 11A and 11C, and transformers. Most of the components that these electrical buses support were modelled in the Cook Nuclear Plant PRA. During the walkdown of this fire zone, no transient combustibles, permitted or otherwise, were found in the area. Based on these findings, it was assumed that if a fire would occur in this zone, it would be located in one of the electrical cabinets and would not propagate out of the cabinet.

As with fire zone 41, the worst case fire in this zone would in the 600VAC electrical bus 11A which supports CCW components. The same assumptions will be used in the analysis that were used in fire zone 41. The CCW initiating event is considered to be the dominant contributor to CDF in this scenario. Bus 11A, as modelled in the Cook PRA, provides 600VAC power to east train motor operated valves. The worst case fire scenario in this bus would be a fire that damages the whole bus 11A, resulting in a loss of 600VAC (1 train) and the loss of one train of CCW. This failure would only impact the train that is in standby. The CCW initiating event frequency for a loss of one train of CCW in standby was calculated to be 2.34E-04 (Table 10).

Using the above information, an analysis was performed to determine the impact that a loss of bus 11A would have on the core damage frequency. The fire initiation frequency was calculated to be 7.1E-03. The conditional probability for a loss of CCW analysis failing bus 11D is 5.02E-02. The estimated CDF for fire zone 42A is as follows.

Est. CDF = 2.34E-04*7.1E-03*5.02E-02

= 8.34E-08

The results of this calculation show that the estimated CDF is less than the reporting cutoff of 1.0E-07. Therefore, fire zone 42A will be screened out.

4.8.11 Fire Zone 42C - Electrical Power System Transformer Room - Unit 1

Fire zone 42C contains the 250VDC distribution panel 1-MCAB which provides a switch and fuse between the battery loads and the 250VDC AB battery, as modelled in the Cook Nuclear Plant PRA. With no transient combustible typically found in this fire zone, it can be assumed that the worst case fire in this zone would start in the distribution cabinet and damage all of the critical cables. Based on reviewing the one-line electrical drawing (Reference 31), the worst case fire would result in a loss of one train of 250VDC. The event tree analysis for initiating events for this fire zone also showed that a loss of CCW was the dominant contributor to CDF in the event of a fire. This was due to the fact that loss of the 250VDC distribution panel will result in a loss of control power for valves on the CCW train in standby. Using the event tree analysis results, the estimated CDF can be further reduced from the value calculated in the first screening. The

initiating event frequency for a CCW train in standby was calculated to be 2.34E-04, the fire initiating frequency (1.0E-03), and the conditional probability for a loss of a standby train of CCW analysis failing a train of 250VDC is 5.06E-02. Multiplying these three numbers together results in an estimated CDF for fire zone 42C of 1.18E-08. Since the estimated CDF in less than the reporting criteria of 1.0E-07, this fire zone screens out.

4.8.12 Fire Zone 43 - Access Control Area - Both Units

Fire zone 43, located in the auxiliary building, only contains two SSSA cables which provide power (1-AM-A) to CMO-420, the west CCW heat exchanger outlet shutoff valve. The cables in this fire zone are not directly visible since this fire zone contains office and laboratory areas which have ceiling tiles. These cables are located in conduit above the ceiling tiles. There is another cable in the SSSA listing associated with the east train of CCW, but it was discovered that this cable is a spare and would not impact the CCW system.

In analyzing this fire zone, failure of the cables discussed above would only impact one train of CCW. Failure of the two power cables would result in a "fail as is" for the CCW Hx discharge valve CMO-420. Therefore, failure of the CCW train in standby would be the worst case scenario. The CCW initiating event for this fire scenario is 2.34E-04. The fire initiation frequency for this zone is 1.0E-03 and the conditional probability for a loss of a standby train of CCW with a failure of 1-AMA-A is 4.39E-02. Based on multiplying these three values together the estimated CDF was calculated to 1.02E-08. This value is less than the reporting criteria of 1.0E-07. Therefore, this fire zone can be screened out.

4.8.13 Fire Zone 44N - Auxiliary Building North - Both Units

Fire zone 44N contains valves, heat exchangers and SSSA cables associated with the Component Cooling Water system, as well as other SSSA cabling associated with various safe shutdown systems. Many of the components and cables were modelled in the Cook Nuclear Plant PRA. In reviewing the other SSSA cabling located in this zone, it was identified that both Unit 1 and 2 AFW cabling resided in fire zone 44N. For Unit 2 AFW, cabling for only one train of the AFW system resides in this zone. Since cabling for the Main Feedwater System was not traced, the Main Feedwater System was not credited in this analysis. In evaluating this fire zone, each of the SSSA cables and components were reviewed to determine what impact failure of the cable or component would have on the CCW system. Based on the review of the cables and components, the failure of CCW system would be the dominate contributor to CDF.

The walkdowns conducted on this zone identified several transient combustibles. A RP desk with paper and plastics was in the vicinity, a dress out area was off to the side of the corridor with a substantial amount of anti-c's, boots, gloves located in metal caged bins, and in the corridor was a covered garbage can.

Based on the amount of transient combustibles and the number of cables in this fire zone, it is assumed that a worst case fire would damage all of the cables. Evaluation of the SSSA cables and components concluded that damaging these cables would only result in a loss of one train of CCW in standby. The explanation holds true for this zone as with several other zones in this analysis in that the MOVs would fail "as is", resulting in a failure of the CCW train in standby. The normally running train would be affected by the loss of power or control of the valves in this zone. As with the other zones, hot shorts are not assumed to cause the CCW MOVs to spuriously operate. This assumption is based on the double break control circuitry design and the Appendix R analysis.

Based on this evaluation and using the event tree analysis for loss of CCW, an analysis was performed failing the west train of CCW (CMO-420), all three trains of Unit 1 AFW, and the MFW system to determine the estimated CDF. The Unit 2 AFW system still has two trains available to provide decay heat removal if a fire would occur. The other train of the Unit 2 AFW did not make cutoff for CCW.OUT. The variables needed to calculate the estimated core damage frequency consist of fire initiation frequency (1.4E-03), initiating event frequency of loss of one train of CCW in standby (2.34E-04) and the output of the analysis run failing the above components (4.4E-02). Using the above values, the estimated CDF was calculated to be 1.44E-08, which is lower than the reporting criteria of 1.0E-07, therefore, this zone screens out.

Since the loss of Unit 1 AFW system could significantly impact the TRA event tree, an analysis was performed failing the Unit 1 AFW system, 1 train of Unit 2 AFW, Unit 1 MFW system and a standby train of CCW. The variables needed to calculate the estimated core damage frequency consist of fire initiation frequency (1.4E-03), the normalized TRA initiating event frequency (1/3.8) and the output of the analysis failing the above components (2.70E-05). Using the above values, the estimated CDF was calculated to be 9.95E-09, which is lower than the reporting criteria of 1.0E-07. Therefore, this zone screens out.

4.8.14 Fire Zone 44S - Auxiliary Building South - Both Units

Fire zone 44S contains the CCW pumps and associated SSSA cables for both units. Only the Unit 1 CCW system was modelled for the Cook Nuclear PRA. As part of the original Fire PRA developed during revision 0 of the IPEEE, COMPBRN runs were performed simulating a lube oil spill around one of the CCW pumps. Each of the CCW pumps has a concrete lip built around the pumps to prevent an oil spill from spreading out to the opposite train pump. As part of this fire scenario it was assumed that the pump with the oil spill has failed. The COMPBRN run was used to confirm that no damage would occur to the opposite train pump, as a result of this fire. The results of the COMPBRN run confirmed that the opposite train pump would not be damage (Reference 32).

Based on this evaluation and, using the event tree analysis for loss of CCW, an analysis was performed failing the normally running train to obtain an estimated CDF. The variables needed to calculate the estimated core damage frequency are the fire initiation frequency (2.4E-03), initiating event frequency of loss of the running train of CCW (1.0E-02) and the output of the run failing the above components (1.6E-02). Using the above values, the estimated CDF was calculated to be 3.8E-07.

4.8.15 Fire Zone 51 - Auxiliary Building - East End - Both Units

Fire zone 51 contains the control cables associated with the west CCW Hx discharge valve CMO-420 and the east CCW Hx inlet and outlet ESW supply valves. These components were modelled in the Cook Nuclear Plant PRA. The cables run through covered cable trays which travel directly through the fire zone on the wall next to the passenger elevator. During the walkdowns, the only transient combustible identified in this zone was a 55 gallon drum of used oil which is approximately 30 feet away from the cable trays. In a discussion with a fire protection engineer at the plant, it was explained that the 55 gallon drum was used as a temporary collection site for used oil. This allowed for the oil to be disposed in large quantities rather than a gallon at a time (Reference 33). The oil drum resides in a spill proof container which goes up to about half the height of the barrel and is chained to the floor. The cap on the oil drum is also locked in place. There is an automatic sprinkler system in this zone that would prevent the pressure inside the barrel to increase to a point which would cause it to implode (Reference 33). Based on PMI-2270,

Fire Protection, no more than a gallon can of oil should be transferred through the zone at any one time. Therefore, a fire in the vicinity of the barrel would consist of no more than one gallon of spilled oil.

The FIVE methodology quantitative screening analysis can be used to determine if damage would occur to the cable trays if one gallon of oil by the barrel of oil would ignite and start on fire. Table 13 shows the work sheet used to perform this calculation. Based on the results of the FIVE methodology screening analysis, which showed no damage to the critical cables located in this zone. Therefore, fire zone 51 can be screened out from further analysis.

4.8.16 Fire Zone 52 - Auxiliary Building - West End - Both Units

Fire zone 52 contains the same SSSA control cables that go through fire zone 51. This includes the west CCW Hx discharge valve CMO-420 and the east CCW Hx inlet and outlet ESW supply valves. These components were modelled in the Cook Nuclear Plant PRA. As identified during the walkdowns, the critical cables carrying the SSSA cables important to the Fire PRA come into fire zone 52 at approximately 13 feet above the floor until they get close to the MCCs and then they drop to about 7.5 feet. There is a flammable storage cabinet about 21' from the MCCs. The cabinet is designated as a 10 ft^3 cabinet with oils and solvents, and is tied down with a thick metal strap. There is nothing directly above the cabinet, but cable tray 1AU-C4 comes within approximately 12 feet of the cabinet. The MCCs 1-AM-A and 1-AM-D are about 21 feet apart with 1-AM-A being little more than 40 feet away from the flammable storage cabinet. MCCs 1-AM-A and 1-AM-D provide 600VAC power to motor operated valves associated with both trains of CCW, as well as other components not as critical to Cook Nuclear Plant PRA.

Based on the above information and assuming that no one fire could take out both trains of MCCs (1-AM-A and AM-D), this analysis will assume that a spill of one gallon of lube oil in the vicinity of the flammable storage would cause a fire and take out the nearest MCC, 1-AM-D. COMPBRN runs showed that the remaining oil inside the cabinet would not ignite and burn if the doors are closed (Appendix F). This would result in a loss of one train of CCW in standby due to loss of power to motor operated valves in the CCW system. The MOVs would "fail as is" and would not affect the train that is normally running.

Using the above information and the event tree analysis for loss of CCW, an analysis was performed failing the standby train of CCW to determine an estimated CDF. The variables needed to calculate the estimated core damage frequency are the fire initiation frequency (2.2E-03), initiating event frequency of loss of one train of CCW in standby (2.34E-04) and the output of the sensitivity analysis run failing the above components (4.82E-02). Using the above values, the estimated CDF was calculated to be 2.48E-08, which is lower than the reporting criteria of 1.0E-07. Therefore, this zone was screened out.

4.8.17 Fire Zones 53 and 144 - Unit 1 Control Room and Hot Shutdown Panel Enclosure

Since the control room has cabling for all the important equipment in the plant, a fire in the control room has special significance to a fire PRA. To establish a core damage frequency for a fire in this room, the comparative study found in "Fire PRA Requantification Studies" (Reference 34) was followed. Previous control room studies have found that control room fires are typically electrical fires in cabinets. For typical cabinets, these are small, localized fires with no spreading potential. The concerns arise from the potential destruction of the controls for multiple trains of important equipment that share a cabinet, and the potential that the control room must be evacuated if the fire is not quickly suppressed.

Although in a separate fire zone, the Unit 1 hot shutdown panel enclosure has essential indication and controls for shutting down the plant from the Unit 2 control room. This cabling is not electrically isolated from the control room cabling, meaning a fault in the Unit 1 control room could disable the instrumentation and controls on the Unit 1 hot shutdown panel. Therefore, this evaluation considers both zones as one, and used the fire initiation frequency of the control room.

Seabrook is a Westinghouse four loop pressurized water reactor, similar to the Cook Nuclear Plant. In the Seabrook portion of the study, three critical fires were identified, resulting in (in Cook Nuclear Plant terminology) loss of CCW, loss of ESW, and a station blackout. By reviewing the Cook Nuclear Plant control room layout, these initiating events were considered to be appropriate for review.

For the probability of fire suppression and control room evacuation, the probability cited in the requantification study (Reference 34) was used (.0034). This is the probability that the fire will not be manually suppressed before the smoke from an electrically-initiated cabinet fire obscures the control board. This value was based on detailed human reliability studies and timing from cabinet fire tests. The configuration for the Cook Nuclear Plant control room is similar to that of the study, so the result is deemed applicable here.

The initiating event frequency for critical cabinet fires is typically determined by some ratio of the amount of critical cabinetry to the total amount of cabinetry in the control room. This has been by floor area around the cabinets, area of the cabinets, or number of cabinets. All give somewhat similar results. For this study, the length of cabinets is taken as an approximate measure of the cabinet area. This is reasonably accurate since the depths of the various cabinets are similar, and none of the measures take into account the amount or type of equipment within a cabinet, which is more likely an appropriate ratio method.

Loss of CCW

A detailed review of the back of the service water panel reveals that it is separated into five sections by interior partitions. These partitions are constructed of 10 gauge steel and 1/2" Marinite boards with no penetrations (Reference 35). Therefore, these partitioned areas can be treated as separate fire initiator areas. The CCW partitioned area is approximately half the length of the full service water panel. Based on a panel length scaling (using drawing no. 12-5976-6 of Reference 4), the probability of a fire in the CCW portion of this panel is .015.

Plant specific human reliability studies were performed for the recovery actions needed to respond to the loss of CCW. Due to the initial confusion caused by the fire, the operators are assumed to not trip the reactor coolant pumps quickly. The reactor coolant pump seals are assumed to catastrophically fail, and recovery requires depressurization of the reactor and low pressure injection within about 100 minutes.

If the control room need not be evacuated, the recovery can be accomplished from the control room. This human action evaluation is found in Section 4.7.4, giving a failure value of .025.

Loss of CCW quantification, control room recovery

Fire initiation frequency = 4.2E-3/year
Fire in CCW panel = .015
Control room not evacuated = .997
Recovery action = .025
Total = 4.2E-3*.015*.997* .025 = 1.6E-6

Loss of CCW quantification, control room evacuation required

Fire initiation frequency = 4.2E-3/year
Fire in critical cabinet = .015
Control room evacuated = .0034
Recovery action (assumed failed for convenience)
Total = 4.2E-3*.015*.0034 = 2.1E-7

The control room loss of CCW failure frequency is equal to 1.8E-06/yr (1.6E-6 + 2.1E-07).

Loss of ESW

A single panel fire could remove both Unit 1 trains from service. The ESW has automatic alignment to the alternate unit's ESW pump (which would not fail due to the fire), and the crosstie valve to Unit 2 is normally open. The control for this crosstie valve is on the ESW panel, but the double break wiring leads only to a loss of valve control, which will not interrupt Unit 2 ESW flow. A complete loss of ESW is only possible with the random failure of the Unit 2 ESW system to provide flow (6.6E-3) (Table 10).

The loss of ESW event is virtually identical to the loss of CCW event. If ESW is lost, the CCW system would continue to operate for a period of time, slowing the heatup of critical equipment and leading to more time for recovery. Thus, the human action evaluation is conservatively applicable to this scenario. The ESW panel is half the length of the CCW panel, leading to half the probability of a fire in this portion of the service water panel (.008).

Loss of ESW quantification, control room recovery

Fire initiation frequency = 4.2E-3/year
Failure of Unit 2 ESW to start = 6.6E-3
Fire in critical cabinet = .008
Control room not evacuated = .997
Recovery action = .025
Total = 4.2E-3*.0066*.008*.997* .025 = 5.5E-9

Loss of ESW quantification, control room evacuated

Fire initiation frequency = 4.2E-3/year
Failure of Unit 2 ESW to start = 6.6E-3
Fire in critical cabinet = .008
Control room evacuated = .0034
Recovery action (assumed failed for convenience)
Total = 4.2E-3*.0066*.008*.0034 = 7.5E-10

The control room loss of ESW failure frequency is sufficiently small that it can be ignored.

Station Blackout

The electric power panel is partitioned into three sections by Marinite and steel, separating the normal power controls from each diesel generator's controls. Therefore, a station blackout would result if a panel fire would fail the normal power supply and both diesel generators would fail to start (3.5E-4, Reference 9). The probability, given a fire, that the fire would be in this critical portion of the panel is .025 based on scaling the total length of panels (using drawing no. 12-5976-6 of Reference 4). For the station blackout scenario, restoration requires RCP seal cooling and continuation of auxiliary feedwater after 4 hours, when the control batteries are assumed to be discharged. In addition to recovery of the affected unit's equipment, equipment and operators from the unaffected unit (which has normal power) are available to support equipment cross-ties.

Station Blackout

Fire initiation frequency = 4.2E-3/year
Failure of both diesel generators = 3.5E-4
Fire in critical cabinet = .025
Recovery action (not considered)
Total = 4.2E-3*3.5E-4*.025 = 3.7E-8

The control room station blackout frequency is sufficiently small that it can be ignored.

For the control room fires evaluated, only the loss of CCW was determined to result in a significant core damage frequency of 1.8E-06/yr.

4.8.18 Fire Zones 55, 56, and 57 - Switchgear, Auxiliary and Control Room Cable Vaults

Due to the similarities, the three cable spreading zones are evaluated together. Other than a limited amount of lighting and similar cable, there is no significant fire ignition source in these zones. In the fire initiation frequency calculation (Section 4.4.2.4), a fire initiation frequency of 1.3E-5/yr was estimated for each of these zones.

The only significant source of combustibles in these zones (excluding the battery area, which is physically separated from the cable spreading area) is cable insulation (Reference 4). Qualified cabling will not propagate a fire (Reference 7), so any fire in these zones must be localized, minimizing damage.

If a fire was to damage all (or a significant portion) of the cabling in a spreading area, plant procedures call for evacuation of the control room and for shutdown of the reactor using local shutdown indication panels and Unit 2 equipment. Since Unit 2 equipment will not be impacted by a fire in these zones, the only significant failure potential is the relatively complex, high stress human actions to follow the remote shutdown procedure. This human action is described in detail and modeled in Section 4.7.4. The failure probability is calculated to be 0.11.

Since the only significant combustion source is cable insulation, which will not support propagation, fire suppression would be very effective in minimizing damage. Automatic fire suppression for these zones is CO_2 (Zones 55 and 56) or Halon (Zone 57), using ionization detectors which respond rapidly (Reference 7). Based on these factors, it is judged that if the fire protection system actuates, damage will be limited and successful shutdown from the control

room is possible. It is judged that the limiting local fire would be in the area underneath the CCW panel, since the loss of CCW is generally the limiting event. The loss of CCW recovery value of 0.025 is valid for this scenario (see previous section). The location scaling of the control room panel evaluation (.015) above should also be a reasonable estimate, as well as the value for the control room recovery action. Thus, the core damage frequency is (fire suppression successful) for each zone,

1.3E-5/yr * .015 * .025 = 5.E-9/yr.

If the automatic fire suppression fails, the control room must be evacuated and the remote shutdown procedure used. The failure rate of a typical halon system (.05) (Reference 7) will be used, which bounds the failure rate of a CO_2 system (.04) (Reference 7). The human failure value for evacuation of the control room and establishing crossties to the other unit is .11 (Section 4.7.4). The core damage frequency for each cable spreading zone is then

1.3E-5/yr * (.05) * (.11) = 7.2E-8/yr.

Thus, none of the cable spreading zones are considered risk significant from a fire perspective.

4.8.19 Fire Zone 79 - Turbine Room - Northeast Portion - Unit 1

Fire zone 79 contains some the SSSA control cables to the EDGs and ESW valves. These components were modelled in the Cook Nuclear Plant PRA. As identified during the walkdowns, the cable tray containing these cables is located in the hallway between the two Unit 1 EDG rooms 15 feet from the floor and running horizontally between the EDG rooms. The cable tray is fire wrapped with Thermolag material. No semi-permanent or permanent combustibles (other than the cable insulation itself) were identified during the walkdown of this fire zone. Therefore, the assumption that the only fire sources with risk significance are those which are semi-permanently stored in fixed locations, and whose presence is evaluated and tracked by permits issued by the fire protection engineer can be applied to this fire zone. Based on this assumption, no fire sources are available in this fire zone that could cause damage to cables in the hallway. As a result, fire zone 79 was screened out from further review.

4.8.20 Fire Zone 112 - ESW Pipe Tunnel - Unit 1

Fire zone 112 contains two SSSA control cables and the ESW header cross-tie valves WMO-705 and WMO-707. The two control cables are for the ESW pressures switches WPS-701 and WPS-705. These pressure switches will automatically start a standby ESW pump should the header pressure drops below 40psig (Reference 36). Walkdowns by plant personnel and review of the Fire Hazard Analysis (References 25 and 4) determined that there are no semi-permanent or permanent transient combustibles located in this fire zone. As well, no significant fire ignition sources are available in this zone to cause damage to the ESW SSSA cabling. Based on this review, the assumption that the only fire sources with risk significance are those which are semi-permanently stored in fixed locations, and whose presence is evaluated and tracked by permits issued by the fire protection engineer can be applied to this fire zone. Based on this assumption, no fire sources are available in this fire zone that could cause damage to cables in the raceway. As a result, fire zone 112 was screened out from further review.

4.8.21 Combined Zones 41, 42A, 42B, 42C, 42D for Large Turbine Fire

In response to an NRC request for additional information regarding a turbine building fire, this analysis was performed to determine the potential for a turbine building fire which could damage cabinets in fire zones 41 and 42A, simultaneously. Further, review of this fire area concluded that fire zones 42B, 42C, and 42D would also need to be included in this evaluation since the fire barriers between these zones would not provide adequate protection in the fire scenario described below. This concern is a result of the fact that the open roll-up door separating zone 42A from the turbine building is normally open. The NRC requested that an analysis be provided of the plant response given simultaneous damage to all cables and equipment in these fire zones.

Following a review of the potential fires that could occur in the Turbine Building, it was concluded, that the only potentially credible fire scenario that could cause this type of damage would be a turbine missile creating a massive turbine lube oil spill. The oil would ignite and be driven by steam or explosions into the 4kV Switchgear rooms. Identification of the SSSA cabling and components within these fire zones indicates a substantial potential impact on components modelled in the PRA. Such a large number of components would be affected, given an assumption of widespread damage, that it was concluded that the use of Unit 2 equipment would be required to successfully shutdown the unit. This is incorporated in the analysis using remote shutdown procedures and their associated human failure rates.

Based on these assumptions, the likelihood of a turbine missile at the Cook Nuclear Plant was first determined. A review of the PRA external events notebook (Reference 37) provided the details of the failure probability of Unit 1 General Electric turbine (Unit 1 turbine had the greater failure probability compared to Unit 2). During the Unit 1 outage in November 1990, an inspection was performed on the shrunk-on wheels of the Low Pressure Turbine rotor for the Unit 1 turbine. Based on this inspection, it was determined that the annual probability of turbine failure resulting in the ejection of turbine disc fragments through the turbine casing was 1.6E-06 (Reference 38). The human failure value for evacuation of the control room and establishing crossties to the other unit is .11 (Appendix D). The core damage frequency for this type of fire scenario is then

1.6E-06/yr * (.11) = 1.76E-07/yr.

This core damage frequency is quite conservative since the probability that a turbine missile could cause a large oil spill and a fire that would enter the combined fire zones was not evaluated.

Thus, the Turbine Building fire scenario damaging the 4kV Switchgear area is not considered to be risk significant from a fire perspective. This core damage frequency is not included in the total for fire damage.

4.9 CONTAINMENT PERFORMANCE

Plant responses arising from a fire are identical to those initiated by other internal events. Containment performance is also identical to that modelled in the Level 2 analysis. Refer to the Level 2 analysis for more detail (Reference 18). Based on the Level 2 analysis, this section provides a qualitative evaluation of the potential for containment damage after a fire induced core damage event.

To prevent containment failure after core damage, both hydrogen igniters and containment spray capability are potentially required (Reference 18). These requirements will be evaluated separately.

In evaluating the potential for hydrogen damage as a result of a station blackout (Reference 18), it was determined that hydrogen induced containment failure was possible, but a hydrogen ignition only for specific containment conditions was required. In the evaluation, these conditions were reached in the SBO-50 (which assumes six hours of auxiliary feedwater) at about 16 hours into the accident. If ignition had occurred significantly before those conditions were reached, insufficient hydrogen would be present to damage the containment. This station blackout accident sequence is very similar in timing and containment impact to a loss of CCW accident sequence, although electric power for fans, hydrogen igniters, containment spray pumps and other electrical equipment is available during a loss of CCW. The fire damage scenarios are dominated by the loss of component cooling water accident sequence.

The control cables for hydrogen igniters are not traced in the SSCA (Reference 5), so they cannot be credited for most fire damage scenarios. In the control room fire cases, the critical panels that contribute to the core damage frequency do not have the hydrogen igniter controls. Thus, for the core damage frequency that results from control room fires (about half of the total), hydrogen igniters are available and hydrogen is not a containment damage concern. For the remainder of the fire zones that contribute to the core damage frequency, the igniters may not be available. However, other electric equipment could (unintentionally) ignite the hydrogen before the critical conditions are reached, or alternately the hydrogen may not ignite when the conditions are critical. Considering the low core damage frequency contribution that could be impacted by the lack of hydrogen igniters, the significant chance that hydrogen igniter controls would not be impacted in many of the fire scenarios, the large amount of time available to recover hydrogen igniters (8 to 16 hours), and the relatively low probability (Reference 18) that containment would fail if the igniters do not function, hydrogen is not considered to be a significant concern for fire risk scenarios.

Spray capability is required to protect the containment from damage after any type of loss of coolant sequence, including those initiated by a loss of CCW induced reactor coolant pump seal failure. In the Level II analysis (Reference 18), containment failure was predicted in 30 hours due to overpressurization for the similar station blackout sequence (SBO-50). Since the containment pressure rise is driven almost solely by decay heat, and containment sprays (or RHR sprays) are capable of removing decay heat a few hours after reactor trip, recovery of either spray system will prevent containment failure if started at any time before containment failure. Since over a day is available for these actions, it is estimated that a human action failure rate in the 0.1 range could be calculated for this high stress scenario. Therefore, fire induced containment failure frequency for fire scenarios is estimated to be a factor of 10 lower than the core damage frequency, or about 4E-7 (see Section 7.0).

5.0 TREATMENT OF FIRE RISK SCOPING STUDY AND OTHER ISSUES

These issues were addressed in the original analysis (Reference 6). Following is a summary of the findings, and their treatment in the original analysis.

5.1 DEPENDENCIES BETWEEN CONTROL ROOM AND REMOTE SHUTDOWN PANEL CIRCUITRY

The functions of the control room and associated remote shutdown panel both rely on the same cable spreading room, making interactions between the two possible. This issue was raised by the NRC and AEPSC responded to this in page 10 of AEP:NRC:0692BT. This is documented in Appendix E of the original analysis (Reference 6) and in Reference 39. The LSI (Local Shutdown Indication) panels can be used to achieve safe shutdown following a fire, in the event of a fire disabling both the control room and its associated remote shutdown panel.

5.2 USE OF PLANT-SPECIFIC DATA (MANUAL FIRE FIGHTING EFFECTIVENESS)

The plant-specific fire brigade training and response times are documented in Appendix E of the original analysis (Reference 6) and in Reference 40. A conservative time for fully turned-out fire brigade response anywhere in the plant is 10 minutes.

5.3 SUPPRESSION AGENT-INDUCED DAMAGE

The fire brigade is trained to avoid "pushing" the fire or flame plume into areas containing safety-related equipment. Due to the use of an E-type nozzle, which has a 30 degree spray pattern, the fire brigade is instructed to keep away from any energized electrical equipment. In the case of an electrical (class C) fire, the fire brigade is trained to first de-energize the panel to enable the class C fire to be treated as a class A or B fire. When experiencing a "fullblown" fire (i.e., a room completely filled with flames), instead of using the method of "surround and drown" or "flood and find out", the fire brigade is trained to use short bursts to knock the fire down, thus allowing the fire brigade the ability to observe the fire and locate its base with a minimal amount of water damage. See Appendix E of the original analysis (Reference 6) and Reference 41 for more details.

5.4 FIRE BARRIER INTEGRITY

AEPSC has programs in place to maintain fire barrier integrity, as described in the procedures listed in Appendix E of the original analysis (Reference 6) and Reference 42.

5.5 VERIFICATION OF AS-BUILT CABLING

Of the 2675 cables deemed necessary to achieve an Appendix R safe shutdown, 60 cables were randomly selected in the original analysis (Reference 6) to verify cable routing against what was described in the SSSA. Once selected, the AEPSC Nuclear Design Department reviewed the appropriate drawings and identified the actual cable routing. This was compared against the SSSA to determine the impact on AEPSC Appendix R compliance. Based on the 60 cable random sample, at a 95% confidence level, the as-built safe shutdown cable routing did not adversely affect Appendix R compliance. Based on these results, the SSSA cable routing, for all practical purposes, was judged to represent the as-built condition. This is described more fully in Appendix E of the original analysis (Reference 6) and in Reference 14.

5.6 TREATMENT OF TRANSIENT COMBUSTIBLES

Procedure 12 SHP 2270 FIRE.012 describes the AEPSC treatment of transient combustibles. They are monitored by two methods:

- Tours which estimate transient combustibles in a fire zone
- Scaffolding log

A computer program updates the fire zone fire loading to ensure that FHA fire loading estimates are not exceeded. If they are, then an hourly roving fire watch is posted. Appendix E of the original analysis (Reference 6) and Reference 43 describe the procedure.

5.7 TREATMENT OF UNCERTAINTIES

GL-88-20, Supplement 4 (Reference 1), requires an identification of all sources of uncertainties. They are listed below:

- Determination of fire-initiation frequencies
- Fire propagation probabilities
- Fire suppression probabilities, automatic and manual
- Human error estimations
- Random failure probabilities
- Barrier failure probabilities

These are described in Appendix E of the original analysis (Reference 6) and in Reference 44.

5.8 SEISMIC-FIRE INTERACTIONS

It was determined following on-site discussions with EQE after the original seismic walkdowns (Reference 45) that the 17-ton CO_2 tank is vulnerable to a seismic event. A seismic event could move the tank, severing pipe connections and expel all CO_2. If the seismic event also induced a fire, fire suppression in those zones with automatic CO_2 suppression could be limited to manual suppression. Further seismic analysis concluded that these tanks will survive a design basis earthquake. Problems will not arise unless a much larger earthquake occurs (Reference 45). Thus, it was concluded that these tanks do not pose a significant seismic/fire interaction concern. Manual suppression efforts may also be hampered by other seismic effects. Reference 25 (original walkdown notebook) describes this. Other seismic/fire interactions of a lesser degree, which also do not pose a significant threat to safe shutdown of the plant, are discussed in Reference 45 which provided input to the seismic fragility analysis for the Cook Nuclear Plant Seismic PRA (Reference 46).

6.0 AREAS OF CONSERVATISM

The following lists areas of conservatism are present in this analysis:

- It was assumed that components would fail in the worst possible way. It may be possible to assign a statistical distribution to the failure rate of a component due to fire (fire fragilities).

- Heroic actions and recovery actions beyond those considered in the original IPE were not credited in this analysis other than those actions specifically modeled for unique control room and cable spreading area fires scenarios.

- Human actions specifically modeled for the fire PRA used primarily the simplified ASEP methodology (Reference 49). The use of the more detailed THERP methodology (Reference 50) would remove conservatism from the human action failure rates.

- The core damage frequency is dominated by the control room fires. It was assumed that all controls in an entire cabinet would fail due to the fire. However, given the flame retardancy of the cables, the low electrical power in the cables, and the effectiveness of quick fire detection and suppression, actual damage would be expected to be far less severe.

- In general, fire detection and suppression is not credited in the analysis.

- In general, cables protection by fire barrier material was not credited.

7.0 SUMMARY OF KEY FINDINGS

The following lists those zones for which the core damage frequency was calculated to be greater than 1.0E-07. The CDF values for these fire zones include low priority zones in the initial screening analysis, as well as the fire zone in the detailed evaluation whose CDF is greater than 1.0E-07.

Zone	Contribution
15	3.04E-07
16	3.50E-07
29B	1.07E-07
29E	1.07E-07
40A	1.32E-07
40B	1.86E-07
41	1.12E-07
42D	1.68E-07
44S	3.80E-07
53	1.81E-06
91	1.02E-07
Total	3.76E-06

The dominant accident sequence that resulted in core damage for these cases was a loss of component cooling water.

8.0 REFERENCES

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4. "Fire Hazards Analysis," Donald C. Cook Nuclear Plant Units No. 1 and 2, American Electric Power, Rev. 6, January 31, 1992.

5. "Safe Shutdown Systems Analysis," AEP D.C. Cook Power Plant Units 1 and 2, American Electric Power, Calculation 0121-090-CAL005 Rev. 3, October 3, 1990, and subsequent references and drawings.

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25. "Fire Analysis Outside Containment Walkdown Notebook," Donald C. Cook Nuclear Plant Units 1 and 2, Westinghouse Electric Company, June 1991, Rev. 0.

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28. Emergency Remote Shutdown Procedures

    a) 01-OHP 4025.001.001, Rev. 0, "Emergency Remote Shutdown."

    b) 01-OHP 4025.LS-1, Rev. 0, "Process Monitoring from LSI Panels", Section LS-1-1.

    c) 01-OHP 4025.LS-2, Rev. 0, "Start-Up AFW", Section LS-2-1.

    d) 01-OHP 4025.LS-2, Rev. 0, "Start-Up AFW", Section LS-2-2.

    e) 01-OHP 4025.LS-6, Rev. 0, "RCS Make-Up, Seal Injection, and Boration with CVCS Cross-tie", Section LS-6-1.

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Table 1
Unit 1 Fire Zones

Fire Zone	Area	Zone Description	Safe Shutdown Equipment	Safe Shutdown Cabling
1	A	Auxiliary Building - both units	No	Yes
1A	A	Containment Spray Pump East - Auxiliary Building - Unit 1	No	No
1B	A	Containment Spray Pump West - Auxiliary Building - Unit 1	No	No
1C	A	Residual Heat Removal Pump East - Auxiliary Building - Unit 1	Yes	Yes
1D	A	Residual Heat Removal Pump West - Auxiliary Building - Unit 1	Yes	Yes
2	B	Pump Bay - Turbine Building - both units	No*	No*
3	C	Drumming / Drum Storage - both units	No	No
4	D	Sampling Room - Auxiliary Building - both units	No	Yes
5	E	Auxiliary Building - both units	Yes	Yes
6A	E	Auxiliary Building Pipe Tunnel - both units	No	No
6N	E	Auxiliary Building (N. section of West end) - Unit 1	Yes	Yes
6M	E	Auxiliary Building (Middle section of West end) - both units	No	Yes
7	F	Quadrant 1 Cable Tunnel - Unit 1	No	Yes
8	G	Quadrant 4 Cable Tunnel - Unit 1	No	Yes
9	H	Quadrant 3N Cable Tunnel - Unit 1	No	Yes
10	H	Quadrant 3M cable Tunnel - Unit 1	No	Yes
11	I	Quadrant 3S Cable Tunnel - Unit 1	No	Yes
12	J	Quadrant 2 Piping Tunnel - Unit 1	Yes	Yes
13	K	Diesel Oil Pump Room - Unit 1	Yes	Yes
14	L	Transformer Room - Unit 1	No	Yes
15	M	1CD Diesel Generator Room - Unit 1	Yes	Yes
16	N	1AB Diesel Generator Room - Unit 1	Yes	Yes
17A	O	West Auxiliary Feed Pump Room - Unit 1	Yes	Yes
17C	Q	Corridor to Auxiliary Feed Pump Rooms - both units	Yes	Yes
17D	R	East Auxiliary Feed Pump Room - Unit 1	Yes	Yes
17E	S	Turbine Auxiliary Feed Pump Room - Unit 1	Yes	Yes
28	B	Diesel Fire Pump Room - Unit 1	No	No
29A	EE	Essential Service Water Pump PP-1E - Unit 1	Yes	Yes
29B	EE	Essential Service Water Pump PP-1W - Unit 1	Yes	Yes
29E	EE	Motor Control Center for ESW Pumps - Unit 1	Yes	Yes
29G	EE	Screen House Motor Control Room for ESW - both units	No	Yes
31	C	Concrete Mixing Building / Drumming Area - both units	No	No
32	C	Cask Handling Area - both units	No	Yes
33	FF	Main Steam Valve Enclosure, East - Unit 1	Yes	Yes
33A	FF	Main Steam Line Area - Unit 1	No	No
33B	FF	Non Essential Service Water Valve Area, West - Unit 1	No	Yes
35	C	Instrument Calibration Room - both units	No	No
36	C	Spent Fuel Pit Heat Exchanger Pump Room - both units	No	No
37	HH	Valve Gallery - both units	No	No

Table 1
(continued)

Fire Zone	Area	Zone Description	Safe Shutdown Equipment	Safe Shutdown Cabling
38	II	Quadrant 2 Penetration Cable Tunnel - Unit 1	Yes	Yes
40A	KK	4kV AB Switchgear Room	Yes	Yes
40B	KK	4kV CD Switchgear Room	Yes	Yes
41	LL	Eng Safety System & MCC Room (& Underfloor) - Unit 1	Yes	Yes
42A	MM	E.P.S. Transformer Room - Unit 1	Yes	Yes
42B	MM	E.P.S. Control Rod Drive Room - Unit 1	No	Yes
42C	MM	E.P.S. Motor Control Room - Unit 1	Yes	Yes
42D	MM	E.P.S. (AB) Battery Room -	Yes	Yes
43	HH	Access Control Area - both units	No	Yes
44N	HH	Auxiliary Building North - both units	Yes	Yes
44S	HH	Auxiliary Building South - both units	Yes	Yes
44A	HH	Containment Spray Heat Exchanger Room #18E, Auxiliary Building - Unit 1	No	No
44B	HH	Containment Spray Heat Exchanger Room #18W, Auxiliary Building - Unit 1	No	No
44C	HH	Residual Heat Removal Heat Exchanger Room #17E, Auxiliary Building - Unit 1	Yes	No
44D	HH	Residual Heat Removal Heat Exchanger Room #17W, Auxiliary Building - Unit 1	Yes	No
48	C	New Fuel Storage Tank - both units	No	No
49	C	HVAC Vestibule - Unit 1	No	Yes
51	C	Auxiliary Building (East End) - both units	No	Yes
52	C	Auxiliary Building (West End) - both units	Yes	Yes
53	QQ	Unit 1 Control Room - Unit 1	Yes	Yes
55	SS	Switchgear Room Cable Vault - Unit 1	Yes	Yes
56	TT	Auxiliary Cable Vault - Unit 1	No	Yes
57	UU	Control Room Cable Vault - Unit 1	No	Yes
61	E	Spray Additive Tank Room - both units	No	No
62A	YY	Reciprocating Charging Pump - Unit 1	Yes	Yes
62B	YY	Centrifugal Charging Pump - Unit 1	Yes	Yes
62C	YY	Centrifugal Charging Pump - Unit 1	Yes	Yes
64A	E	Safety Injection Pump North - Unit 1	No*	No*
64B	E	Safety Injection Pump North - Unit 1	No*	No*
66	AAA	Containment Piping Annulus	Yes	Yes
67	AAA	Containment Lower Volume	Yes	Yes
68	AAA	Containment Upper Volume	Yes	Yes
69	C	Auxiliary Building - both units	Yes	Yes
70	BBB	Control Room HVAC Equipment - Unit 1	No	No
71	BBB	Unit 1 Computer Room - Unit 1	No	No
77	B	Welding Shop - Turbine Building - Unit 1	No	No

Table 1
(continued)

Fire Zone	Area	Zone Description	Safe Shutdown Equipment	Safe Shutdown Cabling
78	B	Heating Boiler Room - Turbine Building - Unit 1	No	No
79	B	Turbine Room Unit 1 (N.E. Portion) - Unit 1	No	Yes
80	B	Turbine Room Unit 1 (S.E. Portion) - Unit 1	No	Yes
81	B	Turbine Room Unit 1 (S.W. Portion) - Unit 1	No	No
82	B	Turbine Room Unit 1 (N.W. Portion) - Unit 1	No	No
83	B	Turbine Room Unit 1 Lube Oil Room - Unit 1	No	No
90	B	Turbine Room Unit 1 (N.E. Portion) - Unit 1	No	No
91	B	Turbine Room Unit 1 (S.E. Portion) - Unit 1	No*	Yes
92	B	Turbine Room Unit 1 (S.W. Portion) - Unit 1	No*	No*
93	B	Turbine Room Unit 1 (N.W. Portion) - Unit 1	No	No
94	B	Turbine Room Unit 1 Office Space - Unit 1	No	No
95	B	Turbine Room Unit 1 Turbine Oil Tank Room - Unit 1	No	No
101	AAA	Containment Accumulator Enclosure West	Yes	Yes
103	AAA	Reactor Head Enclosure	Yes	Yes
105	FF	Contractor Access Control Building - both units	No	No
106	C	Auxiliary Feed Water Battery Room #1 Auxiliary Building - Unit 1	Yes	Yes
108	B	West Steam Valve Enclosure - Unit 1	No	No
110	B	Main Steam Accessway - Unit 1	No	Yes
112	B	Essential Service Water Pipe Tunnel Unit 1	Yes	Yes
114	B	Essential Service Water Pipe Tunnel Unit 1	Yes	Yes
116	DDD	RW, CS, PW Tank Area Pipe Tunnel - Unit 1	No	No
118	AAA	Containment Regen Heat Exchanger Room	No	Yes
120	AAA	Containment Accumulator Enclosure East	Yes	Yes
122	AAA	Containment Instrumentation Room	Yes	Yes
124	B	UPS Invertor Room Security - both units	No	No
125	B	CAS Security - both units	No	No
126	B	Tech Support Center - both units	No	No
127	B	TSC, UPS Invertor and Battery Rooms - both units	No	No
128	B	UPS Battery Room Security - both units	No	No
129	B	Turbine Deck - Unit 1	No	Yes
131	B	Service and Office Bldgs - both units	No	No
132	AAA	Unit 1 Ice Condenser	No	No
134	AAA	Unit 1 Reactor Vessel	No	No
136	A	Unit 1 Pipe Tunnel - Unit 1	No	No
138A	A	CVCS Hold-up Tank Area N. - both units	No	No
138B	A	CVCS Hold-up Tank Area Mid- both units	No	No
138C	A	CVCS Hold-up Tank Area S. - both units	No	No
139	B	Turbine Room Sump - both units	No	No
140	B	Turbine Caustic and Acid Storage Tank Area - both units	No	No
141	B	Turbine Pump Pit - both units	No	No
142	B	Screenhouse - both units	No	No

Table 1
(continued)

Fire Zone	Area	Zone Description	Safe Shutdown Equipment	Safe Shutdown Cabling
143	B	Water Intake and Discharge System both units	No	No
144	UU	Unit 1 Hot Shutdown Panel Enclosure Unit 1	No	Yes
146	C	Auxiliary Building Unloading Platform both units	No	No
147	FFF	Containment Access Building - both units	No	No

* The components located in these fire zones were not considered to be safe shutdown components in the SSSA but were used in the Level 1 PRA. These components are affiliated with nonessential service water, safety injection, control air, and feedwater systems.

Table 2
Summary Table of D.C. Cook Fire Frequency Evaluation

PLANT LOCATION	FIRE FREQUENCY (per year)
CONTAINMENT	3.33E-02
ELECTRICAL SWITCHGEAR BASEMENT	5.37E-02
ELECTRICAL SWITCHGEAR UPPER LEVEL	2.67E-02
AUXILIARY BUILDING	4.14E-02
CABLE SPREADING AREA	2.50E-03
TURBINE BUILDING	4.16E-02

Table 3
Floor Area of Fire Zones in the Auxiliary Building (Unit 1)

Zone	Area	Electrical	Zone	Area	Electrical
1	4500	E	44B	220	
1A	324		44C	270	
1B	324		44D	270	
1C	284		44N	7580	E
1D	284		44S	9360	E
3	2657		48	1650	
4	1025		49	3200	E
5	8635	E	51	5386	E
6A	10890		52	11085	E
6M	6095	E	53	4410	Note 1
6N	4212	E	61	1000	
7	960		62A	405	
8	2050		62B	416	
9	539		62C	416	
10	800		64A	288	
11	840		64B	288	
12	7812	E	69	17914	E
31	986		70	1715	
32	4240		71	430	
33	1040		105	2380	
33A	3316	E	106	180	
35	323		108	897	
36	1624	E	110	1776	
37	2730		112	1229	
38	2650	E	114	539	
43	4630	E	116	1724	
44A	220		136	300	
			144	89	E

Total Area = 149,407 ft²
Area designated as significant electrical equipment = 98,088 ft²

Note 1 - Specific initiating event frequency for this zone, n/a for electrical.

Table 4
Spreadsheet for Calculation of Turbine Building Fire Zones
Fire Initiation Frequencies

Column 1	2	3	4	5	6	7	8	9	10
Zone	Area	pp/oil	Area	elect	Area	Gas	Area	adds	IEF
2	9342	1	9342		0		0		1.9E-03
17A	252	1	252		0		0		5.2E-05
17C	328		0	1	328		0		3.6E-05
17D	219	1	219		0		0		4.5E-05
17E	219	1	219		0		0		4.5E-05
28	400	1	400		0		0		8.2E-05
29A	332	1	332		0		0		6.8E-05
29B	402	1	402		0		0		8.3E-05
29E	92		0	1	92		0		1.0E-05
29G	1554		0	1	1554		0		1.7E-04
77	1740		0		0		0	1.67	2.5E-03
78	2160	1	2160		0		0		4.5E-04
79	11140	1	11140	1	11140	1	11140		6.0E-03
80	14418	1	14418	1	14418		0		4.2E-03
81	12812	1	12812	1	12812		0		3.7E-03
82	11212	1	11212	1	11212	1	11212		6.0E-03
83	897	1	897		0		0		1.8E-03
90	10998		0		0	1	10998		3.0E-03
91	15400		0		0		0	1	1.9E-03
92	13825	1	13825		0		0		2.8E-03
93	12705		0		0	1	12705		3.5E-03
94	890		0		0		0		2.5E-05
95	590	1	590		0		0		1.2E-04
127	1035		0	1	1035		0		1.2E-04
129	10000		0		0		0		2.8E-04
139	139		0		0		0		3.9E-06
140	880		0		0	1	880		2.4E-04
141	1161	1	1161		0		0		2.4E-04
142	18608	1	18608		0		0		3.8E-03
143	0		0		0		0		0.0E-00
Totals	153750		97989		52591		46935		0.042

Column 2 lists the zone area (see text for Zones 129 and 143)
Columns 3, 5, 7 has a one if significant equipment of this type is in the Zone
Columns 4, 6, 8 copies the area of the zone if it is identified in the prior column
Totals for the area columns 2, 4, 6, 8 are on the bottom line
Column 9 has specific fires from the database allocated to the zone
Column 10 shows the result of the fire initiation frequency by zone, and the total at the bottom. The calculation sums the fires allocated to each ignition source, ratio by the zone to total area, and divided the total by the experience years (689). See text for further details.

Table 5
Fire Frequencies for Fire Zones
in Basement of Switchgear Building
(based on .0537 fires/year for switchgear building basement)

ZONE IDENTIFICATION	FIRE INITIATION FREQUENCY (per year)
13 DIESEL OIL PUMP ROOM	1.0E-3
14 TRANSFORMER ROOM	9.1E-3
15 1CD DIESEL GENERATOR ROOM	2.2E-2
16 1AB DIESEL GENERATOR ROOM	2.2E-2
TOTAL	5.4E-2

Note: The total exceeds the location frequency since minimum values were used.

Table 6
Fire Frequencies for Fire Zones in the Upper Level of the Switchgear Building
(based on .0267 fires/year for upper level of switchgear building)

ZONE IDENTIFICATION	FIRE INITIATION FREQUENCY (per year)
40A 4KV AB SWITCHGEAR ROOM	2.9E-3
40B 4KV CD SWITCHGEAR ROOM	2.9E-3
41 ENG SAFETY SYSTEMS & MCC ROOM	9.1E-3
42A E.P.S. TRANSFORMER ROOM	7.1E-3
42B E.P.S. CONTROL AND DRIVE ROOM	1.0E-3
42C E.P.S. MOTOR ROOM	1.0E-3
42D E.P.S. (AB) BATTERY ROOM	3.2E-3
TOTAL	2.7E-3

Note- Total increased slightly for new values due to use of a minimum value.

Table 7
Fire Frequencies for Fire Zones in the
Auxiliary Building
(based on .0414 fires/year for the auxiliary building)

ZONE IDENTIFICATION	FIRE INITIATION FREQUENCY (per year)
1 AUX BLDG	3.5E-3
1A CONTAINMENT SPRAY PUMP EAST	
1B CONTAINMENT SPRAY PUMP WEST	
1C RESIDUAL HEAT REMOVAL PUMP EAST	
1D RESIDUAL HEAT REMOVAL PUMP WEST	
3 DRUMMING/DRUM STORAGE	
4 SAMPLING ROOM, AUX BLDG	
5 AUX BLDG (EAST END)	3.9E-3
6A AUX BLDG PIPE TUNNEL	
6M AUX BLDG (MID SEC. OF WEST END)	2.0E-3
6N AUX BLDG (N. SEC. OF WEST END)	1.0E-3
7 QUADRANT 1 CABLE TUNNEL	
8 QUADRANT 4 CABLE TUNNEL	
9 QUADRANT 3N CABLE TUNNEL	
10 QUADRANT 3M CABLE TUNNEL	
11 QUADRANT 3S CABLE TUNNEL	
12 QUADRANT 2 PIPING TUNNEL	1.5E-3
31 CONCRETE MIXING /DRUMMING AREA	
32 CASK HANDLING AREA	
33 MAIN STEAM VALVE ENCLOSURE, E.	
33A MAIN STEAM LINE AREA, EAST	1.0E-3
33B NON-ESS. SERV. WTR VALVE AREA,W	
35 INSTRUMENT CALIBRATION ROOM	
36 SPENT FUEL PIT HT EXCH. PUMP ROOM	1.0E-3
37 VALVE GALLERY	
38 QUADRANT 2 PENE. CABLE TUNNEL	1.0E-3
43 ACCESS CONTROL	1.0E-3
44A CONTAINMENT SPRAY HX ROOM #18E	
44B CONTAINMENT SPRAY HX ROOM #18W	
44C RHR HX ROOM #17E, AUX BLDG	
44D RHR HX ROOM #17W, AUX BLDG	
44N AUX BLDG NORTH	1.4E-3
44S AUX BLDG SOUTH	2.4E-3
48 NEW FUEL STORAGE AREA	
49 HVAC VESTIBULE	1.0E-3
51 AUX BLDG (EAST END)	1.1E-3
52 AUX BLDG (WEST END)	2.2E-3
53 UNIT 1 CONTROL ROOM	4.2E-3
61 SPRAY ADDITIVE TANK ROOM	
62A RECIPROCATING CHARGING PUMP	
62B CENTRIFUGAL CHARGING PUMP	
62C CENTRIFUGAL CHARGING PUMP	

Table 7 (continued)
Fire Frequencies for Fire Zones in the
Auxiliary Building
(based on .0414 fires/year for the auxiliary building)

ZONE IDENTIFICATION		FIRE INITIATION FREQUENCY (per year)
64A	SAFETY INJECTION PUMP NORTH	
64B	SAFETY INJECTION PUMP SOUTH	
69	AUX BLDG	5.1E-3
70	CONTROL ROOM HVAC EQUIPMENT	
71	UNIT 1 COMPUTER ROOM	
105	FORMER CONTR ACCESS CONTROL	
106	AUX FEED WATER BATTERY ROOM #1	2.8E-3
108	WEST STEAM VALVE ENCLOSURE	
110	MAIN STEAM ACCESSWAY	
112	ESS. SERVICE WATER PIPE TUNNEL	
114	ESS. SERVICE WATER PIPE TUNNEL	
116	RW, CS, PW TANK AREA PIPE TNL	
136	UNIT 1 PIPE TUNNEL	
144	UNIT 1 HOT SHUTDOWN PANEL ENCL	1.0E-3
	TOTAL	**

* New values not listed are assumed to be a minimum value of 1.0E-3.
** 3.8E-2 using calculated values only, 7.5E-2 including non-calculated minimums.

Table 8
Fire Frequencies for Fire Zones in
Cable Spreading Rooms

ZONE IDENTIFICATION	FIRE INITIATION FREQUENCY (per year)
55 SWITCHGEAR ROOM CABLE VAULT	6.0E-3
56 AUXILIARY CABLE VAULT	1.0E-3
57 CONTROL ROOM CABLE VAULT	1.0E-3
TOTAL	8.0E-3

Note - Total frequency increased due to the inclusion of battery room fires in Zone 55 and minimum fire frequencies.

Table 9
Fire Frequencies for Fire Zones in
the Turbine Building
(based on .0416 fires/year for the turbine building)

ZONE IDENTIFICATION		FIRE INITIATION FREQUENCY (per year)
2	PUMP BAY, TURBINE BLDG	1.9E-3
17A	WEST AUX FEED PUMP ROOM	
17C	CORRIDOR TO AUX FEED PUMP	
17D	EAST AUX FEED PUMP ROOM	
17E	TURBINE AUX FEED PUMP ROOM	
28	DIESEL FIRE PUMP ROOM	
29A	ESW PUMP PP-1E	
29B	ESW PUMP PP-1W	
29E	MOTOR CONTROL CNTR, ESW PUMPS	
29G	MOTOR CONTROL ROOM, ESW	
77	WELDING SHOP, TURBINE BLDG	2.5E-3
78	HEATING BOILER ROOM, TURBINE	
79	TURBINE ROOM (N.E. PORTION)	6.0E-3
80	TURBINE ROOM (S.E. PORTION)	4.2E-3
81	TURBINE ROOM (S.W. PORTION)	3.7E-3
82	TURBINE ROOM (N.W. PORTION)	6.0E-3
83	TURBINE ROOM LUBE OIL ROOM	
90	TURBINE ROOM (N.E. PORTION)	3.0E-3
91	TURBINE ROOM (S.E. PORTION)	1.9E-3
92	TURBINE ROOM (S.W. PORTION)	2.8E-3
93	TURBINE ROOM (N.W. PORTION)	3.5E-3
94	TURBINE ROOM OFFICE SPACE	
95	TURBINE ROOM TURBINE OIL TANK	
127	TSC, UPS INVERTOR AND BATTERY	
129	UNIT 1 TURBINE DECK	
139	TURBINE ROOM SUMP	
140	TURBINE CAUSTIC/ACID STORAGE TANK	
141	TURBINE PUMP PIT	
142	SCREENHOUSE	3.8E-3
143	WATER INTAKE/DISCHARGE SYSTEM	
	TOTAL	**

* New values not listed are assumed to be a minimum value of 1.0E-3.
** 3.8E-2 using calculated values only, 5.7E-2 including non-calculated minimums.

Table 10
Initiating Event Frequencies

System	Components/Trains Lost Due to Fire	Initiating Event Frequency*
CCW	A) 1 operating train B) 1 standby train	A) 1.0E-02 B) 2.3E-04
ESW	A) 1 operating train B) 1 standby train C) Both operating trains D) Both standby trains E) Both U1 trains F) Both U2 trains G) Both trains/header aligned to U1 loads H) Both trains/header aligned to U2 loads	A) 4.5E-05 B) 1.2E-05 C) 6.6E-03 D) 3.4E-04 E) 6.6E-03 F) 3.4E-04 G) 6.9E-03 H) 2.4E-05
250VDC	1 train	1.0
SBO	A) 1 diesel generator B) 2 of 4 ESW supply valves to diesel generators	A) 2.2E-06 B) 5.2E-08

* These values are calculated in Appendix G.

Table 11
Summary of Zone Specific Frequencies for Initiating Events

Zone	Zone Description	Trains of CCW, ESW, D/G's or 250VDC in zone	IE's to Consider and their frequencies (frequencies are product of zone fire frequency and zone IEF)
6N	Aux bldg - N section of W end - U1	- 1 train of CCW (1W) - both trains of ESW (U1) - 1 D/G (1AB)	CCW: 1.0E-05 TRA: 9.9E-04
13	Diesel oil pump rm - U1	- 1 D/G (1CD)	SBO: 2.8E-09 TRA: 1.0E-03
15	1 CD diesel generator room - U1	- 1 train CCW (1E) - total loss of ESW (U1) (lose U1 pumps & crosstie WMO-707) - 1 D/G (1CD)	CCW: 2.2E-02
16	1AB diesel generator room - U1	- 1 train CCW (1W) - 2 trains/1 header ESW (1W header) - 1 D/G (1AB)	CCW: 2.2E-04 TRA: 2.2E-02
29A	ESW pump PP-1E - U1	- 1 train of ESW (1E)	ESW: 4.5E-08 TRA: 1.0E-03
29B	ESW pump PP-1W - U1	- both trains of ESW (U1)	ESW: 6.6E-06 TRA: 1.0E-03
29E	MCC for ESW pumps - U1	- both trains of ESW (U1)	ESW: 6.6E-06 TRA: 1.0E-03

Table 11
Summary of Zone Specific Frequencies for Initiating Events

Zone	Zone Description	Trains of CCW, ESW, D/G's or 250VDC in zone	IE's to Consider and their frequencies (frequencies are product of zone fire frequency and zone IEF)
29G	Screen house motor control rm for ESW - both units	- all 4 trains of ESW - both D/G's (U1)	CCW: 1.0E-03 SBO w/ CCW: 1.1E-07
40A	4kV AB switchgear room	- 1 train CCW (1W) - 1 train ESW (1W) - 1 D/G (1AB) - 1 train 250VDC (CD)	250VDC: 2.9E-03 CCW w/ 250VDC: 2.9E-05
40B	4kV CD switchgear room	- 1 train CCW (1E) - 1 train ESW (1E) - 1 D/G (1CD)	CCW: 2.9E-05 TRA: 2.9E-03
42A	EPS transformer rm - U1	- 1 train CCW (1W) - 2 trains/1 header ESW (1W header) - 1 D/G (1AB)	CCW: 7.1E-05 TRA: 7.0E-03
42C	EPS motor control rm - U1	- 1 train CCW (1W) - 1 train ESW (1W) - 1 D/G (1AB) - 1 train 250VDC (AB)	250VDC: 9.9E-04 CCW w/ 250VDC: 1.0E-05
42D	EPS AB battery rm	- 1 D/G (1AB) - 1 train 250VDC (AB)	250VDC: 3.2E-03

Table 11
Summary of Zone Specific Frequencies for Initiating Events

Zone	Zone Description	Trains of CCW, ESW, D/G's or 250VDC in zone	IE's to Consider and their frequencies (frequencies are product of zone fire frequency and zone IEF)
43	Access control area - both units	- 1 train of CCW (1W)	CCW: 1.0E-05 TRA: 1.0E-03
44N	Aux bldg N - both units	- total loss of CCW (U1) - 1 train of ESW (1E)	CCW: 1.4E-03
44S	Aux bldg S - both units	- total loss of CCW (U1)	CCW: 2.4E-03
51	Aux bldg - E end - both units	- total loss of CCW (U1) (lose W train & ESW cooling to E train)	CCW: 1.1E-03
52	Aux bldg - W end - both units	- total loss of CCW (U1) (lose W train & ESW cooling to E train)	CCW: 2.2E-03
79	Turbine rm - U1 - NE portion	- both trains of CCW (total loss of CCW) - all 4 trains of ESW (lose U1 pumps plus crossties) - both D/G's (U1)	CCW: 6.0E-03 SBO w/ CCW: 6.6E-07
112	ESW pipe tunnel - U1	- all 4 trains of ESW (lose U1 pumps plus crossties)	CCW: 1.0E-03
114	ESW pipe tunnel - U1	- 2 of 4 ESW supply valves to D/G's	SBO: 5.2E-11 TRA: 1.0E-03

Table 12
Summary of Estimated Core Damage Frequencies for the 65 Zones

Zone	Prior -ity?	Zone Description	Init. Event	Fire Freq.	Method Used to Calculate CDF**	Estimated CDF
1	no	Aux bldg both units	TRA	3.5E-03	SCE	5.64E-09
1C	no	RHR pump E - aux bldg	TRA	1.0E-03	SCE	1.61E-09
1D	no	RHR pump W - aux bldg	TRA	1.0E-03	SCE	1.61E-09
2*	no	Pump bay - turb bldg - both units	TRA	1.9E-03	SENS	9.00E-09
4	no	Sampling room - aux bldg - both units	TRA	1.0E-03	SCE	1.61E-09
5	no	Aux bldg - both units	TRA	3.9E-03	SCE	1.85E-08
6N	Hi	Aux bldg - N section of W end - U1	TRA CCW	1.0E-03	SCE IE	1.50E-03 > 1.58E-07
6M	Hi	Aux bldg - middle section of W end - both units	TRA	2.0E-03	SENS	2.82E-06
7	no	Quadrant 1 cable tunnel - U1	TRA	1.0E-03	EJ	< 1.0E-07
8	no	Quadrant 4 cable tunnel - U1	TRA	1.0E-03	EJ	< 1.0E-07
9	no	Quadrant 3N cable tunnel - U1	TRA	1.0E-03	EJ	< 1.0E-07
10	no	Quadrant 3M cable tunnel - U1	TRA	1.0E-03	EJ	< 1.0E-07
11	no	Quadrant 3S cable tunnel - U1	TRA	1.0E-03	EJ	< 1.0E-07
12	no	Quadrant 2 piping tunnel - U1	TRA	1.5E-03	SENS	1.83E-10
13	no	Diesel oil pump rm - U1	TRA SBO	1.0E-03	EJ IE	< 1.0E-07 2.22E-10
14	no	Transformer rm - U1	TRA	9.1E-03	SCE	1.47E-08
15	Hi	1 CD diesel generator room - U1	CCW	2.2E-02	IE	3.49E-04
16	Hi	1AB diesel generator room - U1	TRA CCW	2.2E-02	SCE IE	4.73E-07 3.49E-06
17A	no	W AFW pump rm - U1	TRA	1.0E-03	SCE	4.73E-09
17C	Hi	Corridor to AFW pump rms - both units	TRA	1.0E-03	SENS	1.41E-06

Table 12
Summary of Estimated Core Damage Frequencies for the 65 Zones

Zone	Prior -ity?	Zone Description	Init. Event	Fire Freq.	Method Used to Calculate CDF**	Estimated CDF
17D	no	E AFW pump rm - U1	TRA	1.0E-03	SENS	8.82E-11
17E	no	Turbine AFW pump rm - U1	TRA	1.0E-03	SCE	8.59E-09
29A	no	ESW pump PP-1E - U1	TRA ESW	1.0E-03	SCE IE	1.60E-09 7.29E-10
29B	Low	ESW pump PP-1W - U1	TRA ESW	1.0E-03	SCE IE	1.60E-09 1.07E-07
29E	Low	MCC for ESW pumps - U1	TRA ESW	1.0E-03	SENS IE	7.76E-11 1.07E-07
29G	Hi	Screen house motor control rm for ESW - both units	CCW	1.0E-03	IE	1.58E-05
32	no	Cask handling area - both units	TRA	1.0E-03	SCE	1.61E-09
33	no	E main steam valve enclosure - U1	TRA	1.0E-03	SENS	6.53E-10
33B	no	W NESW valve area - U1	TRA	1.0E-03	EJ	< 1.0E-07
38	no	Quadrant 2 penetration cable tunnel - U1	TRA	1.0E-03	SCE	9.61E-09
40A	Hi	4kV AB switchgear room	CCW	2.9E-03	IE	> 4.59E-07
40B	Hi	4kV CD switchgear room	TRA CCW	2.9E-03	SCE IE	2.32E-08 > 4.59E-07
41	Hi	Eng safety system & MCC room (& under floor) - U1	n/a	9.1E-03	n/a	9.10E-03
42A	Hi	EPS transformer rm - U1	TRA CCW	7.1E-03	SENS IE	4.63E-09 1.12E-06
42B	no	EPS control rod drive rm - U1	TRA	1.0E-03	SCE	4.73E-09
42C	Hi	EPS motor control rm - U1	CCW	1.0E-03	IE	> 1.58E-07
42D	Low	EPS AB battery rm	250V	3.2E-03	IE	1.68E-07
43	Hi	Access control area - both units	TRA CCW	1.0E-03	SENS IE	8.91E-11 > 1.58E-07

Table 12
Summary of Estimated Core Damage Frequencies for the 65 Zones

Zone	Prior -ity?	Zone Description	Init. Event	Fire Freq.	Method Used to Calculate CDF**	Estimated CDF
44N	Hi	Aux bldg N - both units	CCW	1.4E-03	IE	> 2.22E-05
44S	Hi	Aux bldg S - both units	CCW	2.4E-03	IE	3.80E-05
44C	no	RHR Hx rm #17E - aux bldg - U1	TRA	1.0E-03	SCE	1.61E-09
44D	no	RHR Hx rm #17W - aux bldg - U1	TRA	1.0E-03	SCE	1.61E-09
49	no	HVAC vestibule - U1	TRA	1.0E-03	SCE	1.75E-09
51	Hi	Aux bldg - E end - both units	CCW	1.1E-03	IE	> 1.58E-05
52	Hi	Aux bldg - W end - both units	CCW	2.2E-03	IE	> 3.49E-05
53	Hi	U1 control rm	n/a	4.2E-03	n/a	4.20E-03
55	Hi	Switchgear rm cable vault - U1	n/a	6.0E-03	n/a	6.00E-03
56	Hi	Auxiliary cable vault - U1	n/a	1.0E-03	n/a	1.00E-03
57	Hi	Control rm cable vault - U1	n/a	1.0E-03	n/a	1.00E-03
62A	no	Reciprocating charging pump - U1	TRA	1.0E-03	EJ	< 1.0E-07
62B	no	CCP- U1	TRA	1.0E-03	SCE	1.65E-09
62C	no	CCP - U1	TRA	1.0E-03	EJ	< 1.0E-07
64A*	no	SI pump N - U1	TRA	1.0E-03	SCE	1.75E-09
64B*	no	SI pump N - U1	TRA	1.0E-03	SCE	1.75E-09
69	no	Aux bldg - both units	TRA	5.1E-03	EJ	< 1.0E-07
79	Hi	Turbine rm - U1 - NE portion	CCW	6.0E-03	IE	> 9.48E-05
80	no	Turbine rm SE portion - U1	TRA	4.2E-03	SCE	1.99E-08
91	Low	Turbine rm SE portion - U1	TRA	1.9E-03	SENS	1.02E-07
92*	no	Turbine rm SW portion - U1	TRA	2.8E-03	SCE	4.62E-09
106	no	AFW battery rm #1 - aux bldg - U1	TRA	2.8E-03	SENS	2.25E-10
110	no	Main steam accessway - U1	TRA	1.0E-03	EJ	< 1.0E-07

Table 12
Summary of Estimated Core Damage Frequencies for the 65 Zones

Zone	Prior -ity?	Zone Description	Init. Event	Fire Freq.	Method Used to Calculate CDF**	Estimated CDF
112	Hi	ESW pipe tunnel - U1	CCW	1.0E-03	IE	1.58E-05
114	no	ESW pipe tunnel - U1	TRA SBO	1.0E-03	SCE IE	1.60E-09 4.20E-12
129	no	Turbine deck - U1	TRA	1.0E-03	SENS	8.16E-11
144	Hi	U1 hot shutdown panel enclosure	n/a	1.0E-03	n/a	1.0E-03

* Per Table 1, the components located in these fire zones were not considered to be safe shutdown components in the SSSA, but were modelled in the Level 1 PRA.

** Methods used to calculate CDF:
- IE - initiating event hand calculation (see Appendix E)
- SCE - System Cutset Editor computer run (see Reference 47)
- SENS - SENS computer code run (see Appendix B)
- EJ - engineering judgement (see Section 4.6.2.2)

Totals:
23 high priority zones (estimated CDF ≥ 1E-06)
4 low priority zones (estimated CDF between 1E-06 and 1E-07)
38 zones screened out (estimated CDF < 1E-07)

Table 13
Fire Zone 51
(Reference 7)

WORKSHEET 3: RADIANT EXPOSURE SCENARIOS ENGLISH UNITS VERSION

1	CRITICAL RADIANT FLUX TO TARGET (LOOK UP VALUE FROM TABLE 1E)	1.00	$Btu/s/ft^2$
2	PEAK FIRE INTENSITY (USE TABLE 2E FOR GUIDANCE)	7425*	Btu/s
3	RADIANT FRACTION OF HEAT RELEASE (REPRESENTATIVE VALUE = 0.4)	0.4	
4	RADIANT HEAT RELEASE RATE ([BOX 2]X[BOX 3])	2970	Btu/s
5	CRITICAL RADIANT FLUX DISTANCE (LOOK UP VALUE FROM TABLE 10E)	< 30**	ft
	IF THE EXPOSURE FIRE IS LOCATED WITHIN THIS DISTANCE (INDICATED IN BOX 5) OF THE TARGET, CRITICAL CONDITIONS CAN OCCUR. OUTSIDE THIS RANGE, CRITICAL CONDITIONS ARE NOT INDICATED FOR THE SCENARIO UNDER CONSIDERATION.		

Notes:

* Assume transformer oil with unit heat release rate = 135 $Btu/s\text{-}ft^2$ and spill specific area = 55 ft^2/gal (Pennzoil 30-HD)

Peak Fire Intensity = 135 $Btu/s\text{-}ft^2$ X 55 ft^2/gal = 7425 Btu/s

** Estimated by following the graph in Table 10E of Reference 7.

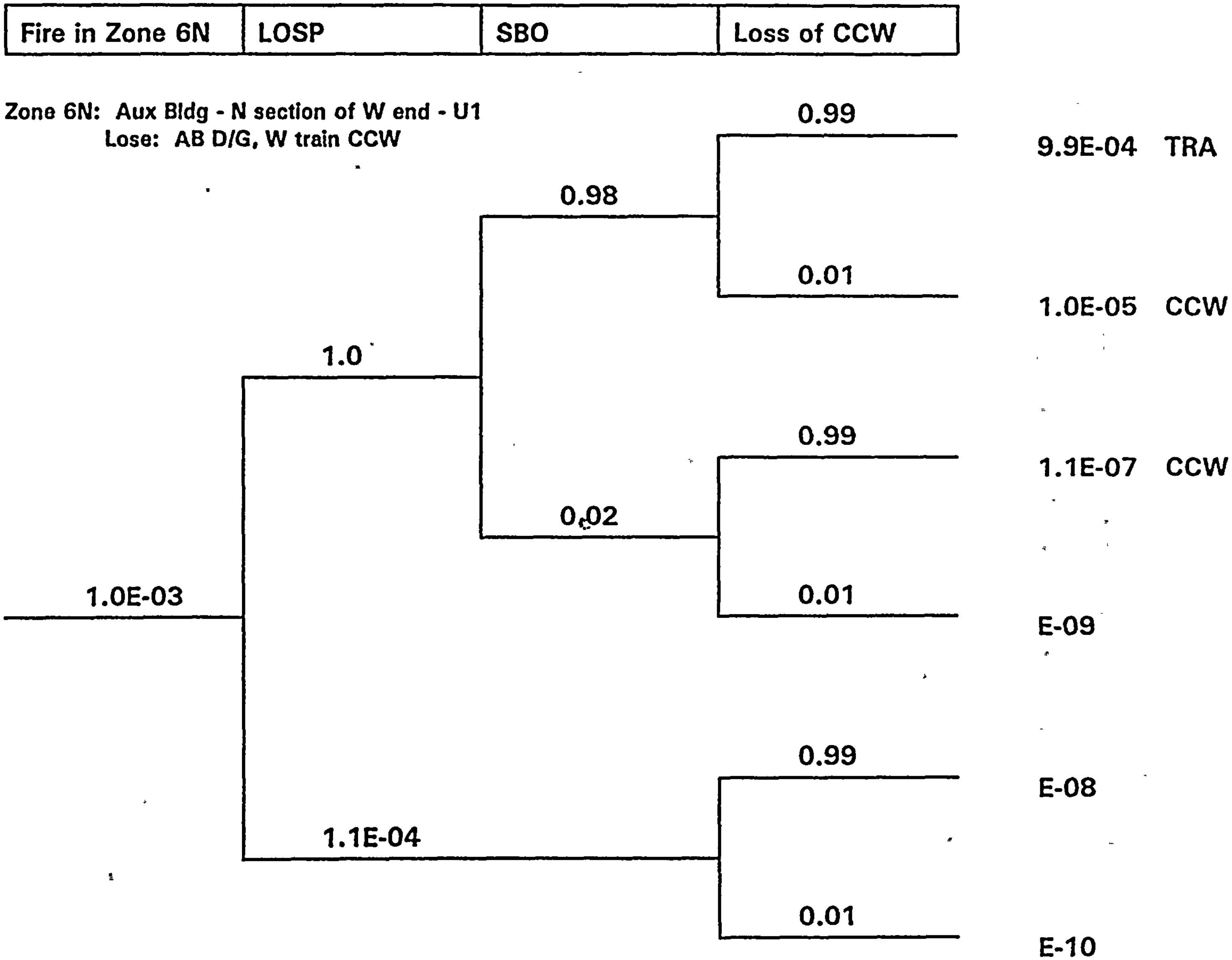

Figure 1 - Event Tree for Zone 6N

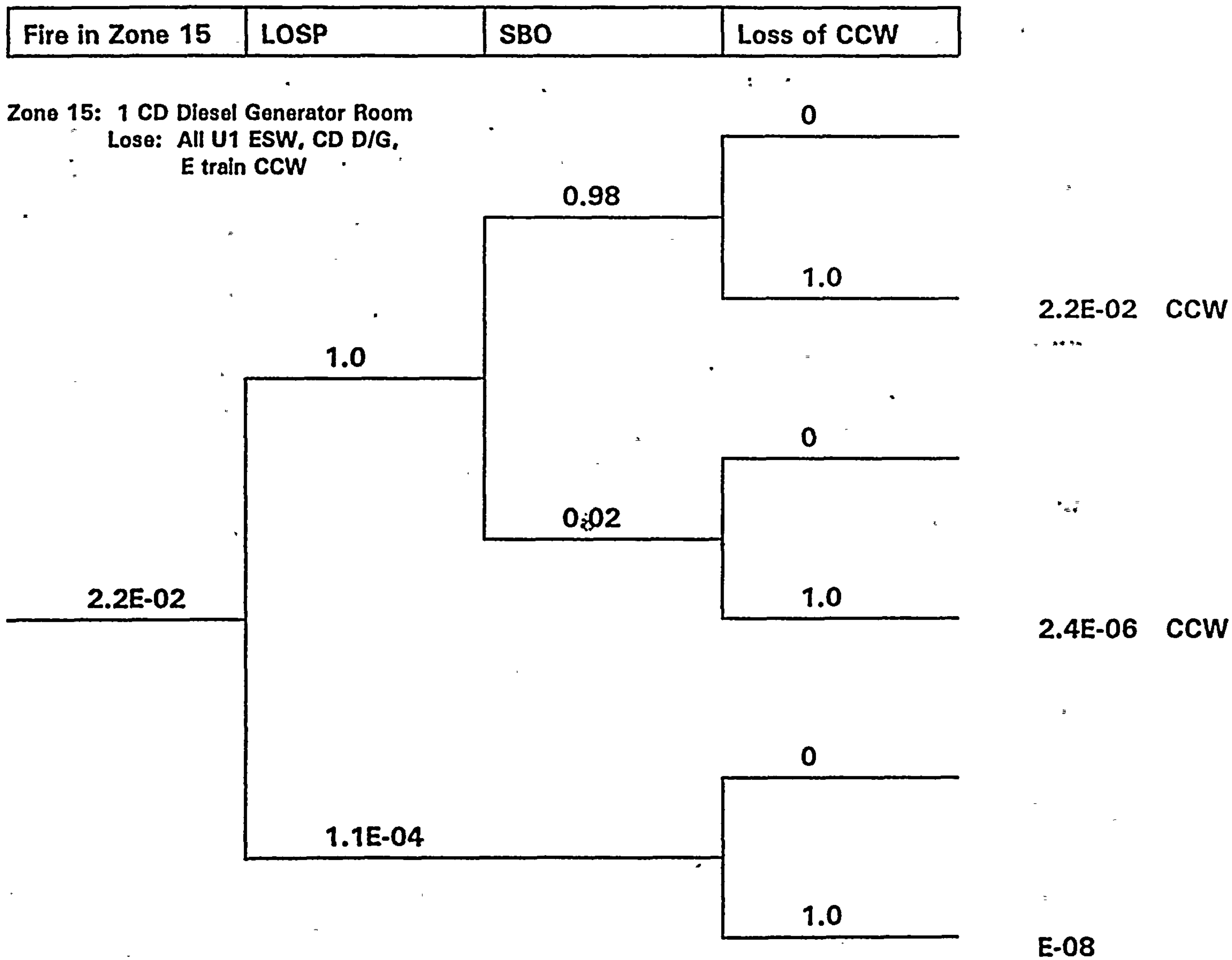

Figure 2 - Event Tree for Zone 15

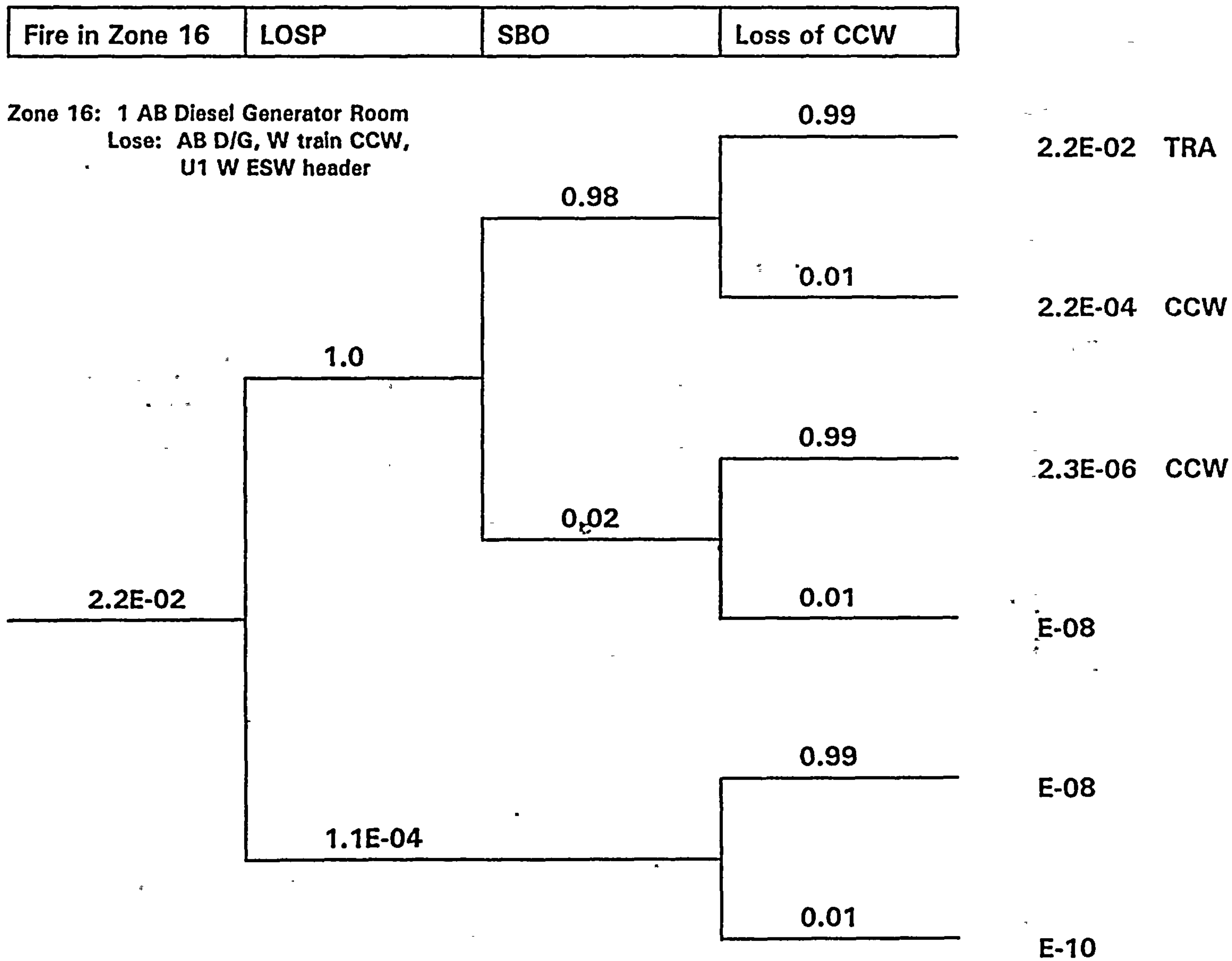

Figure 3 - Event Tree for Zone 16

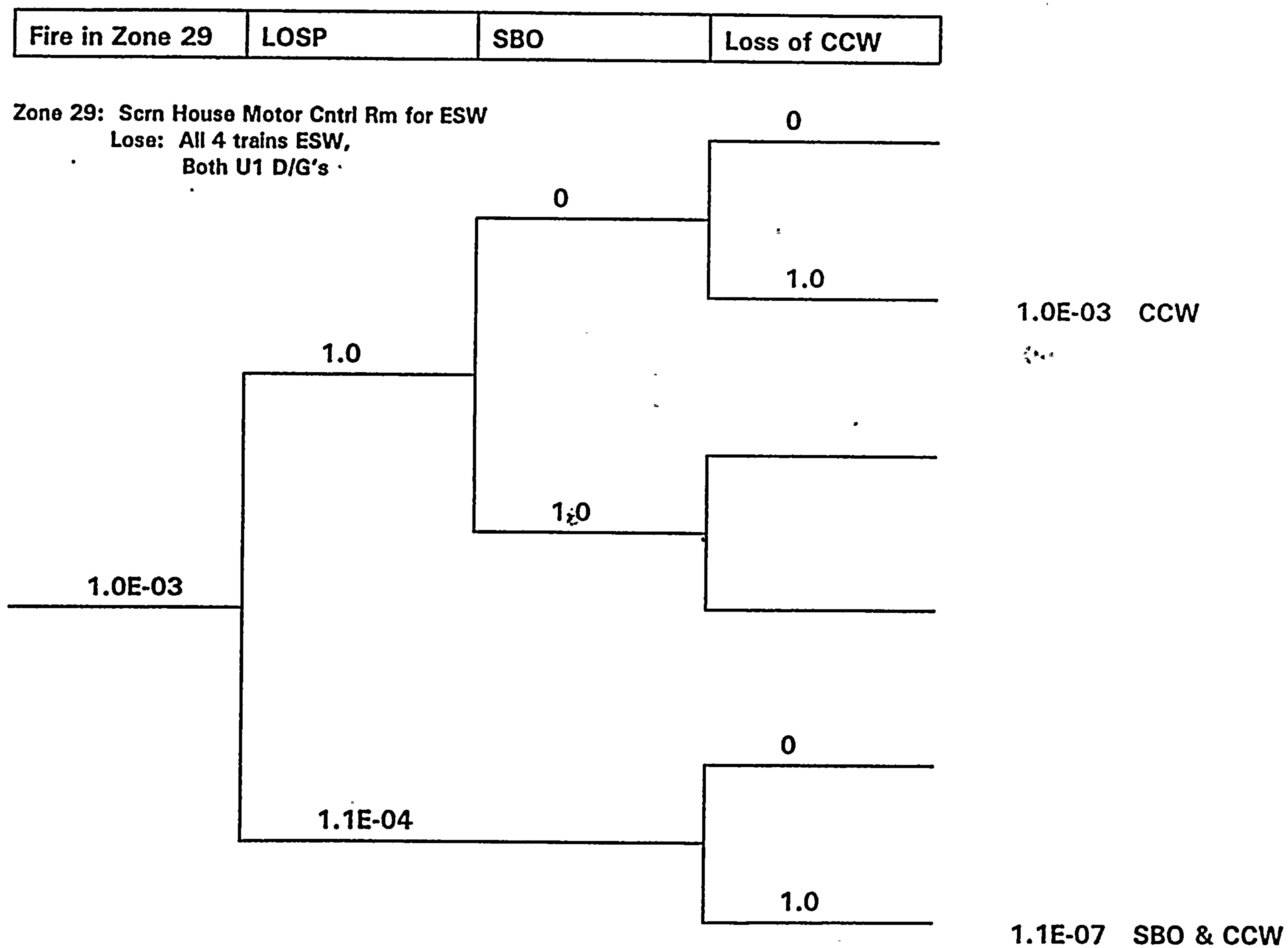

Figure 4 - Event Tree for Zone 29

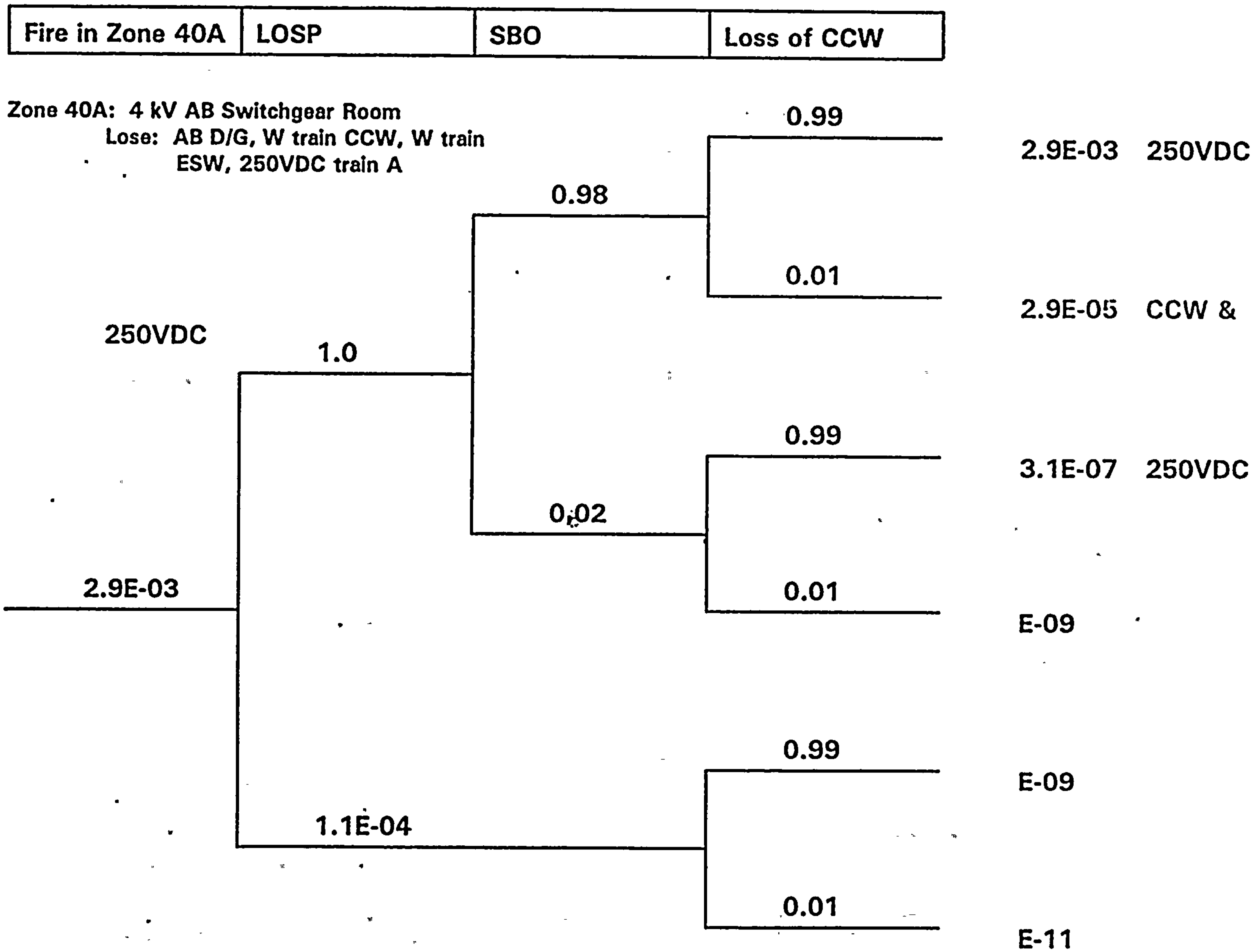

Figure 5 - Event Tree for Zone 40A

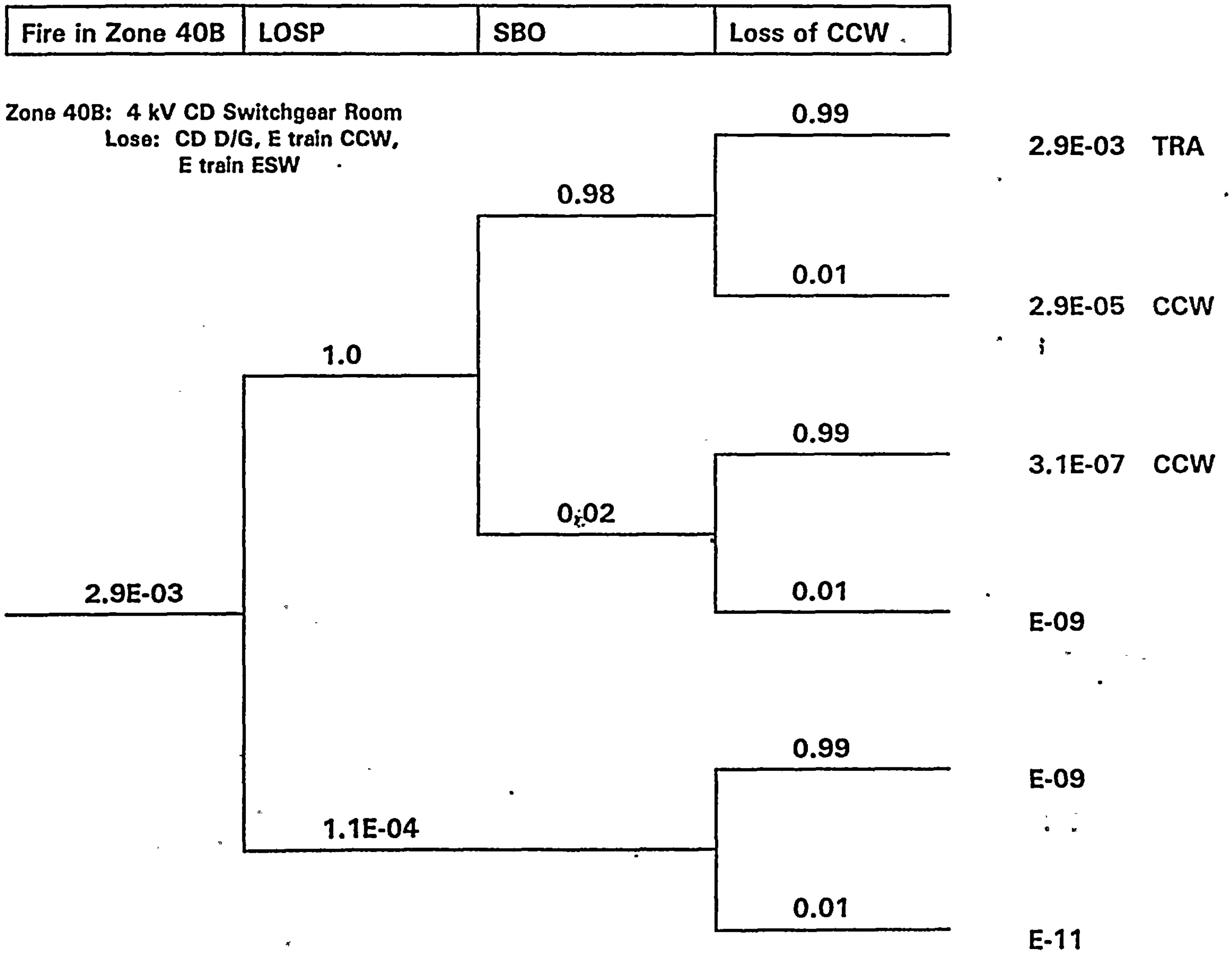

Figure 6 - Event Tree for Zone 40B

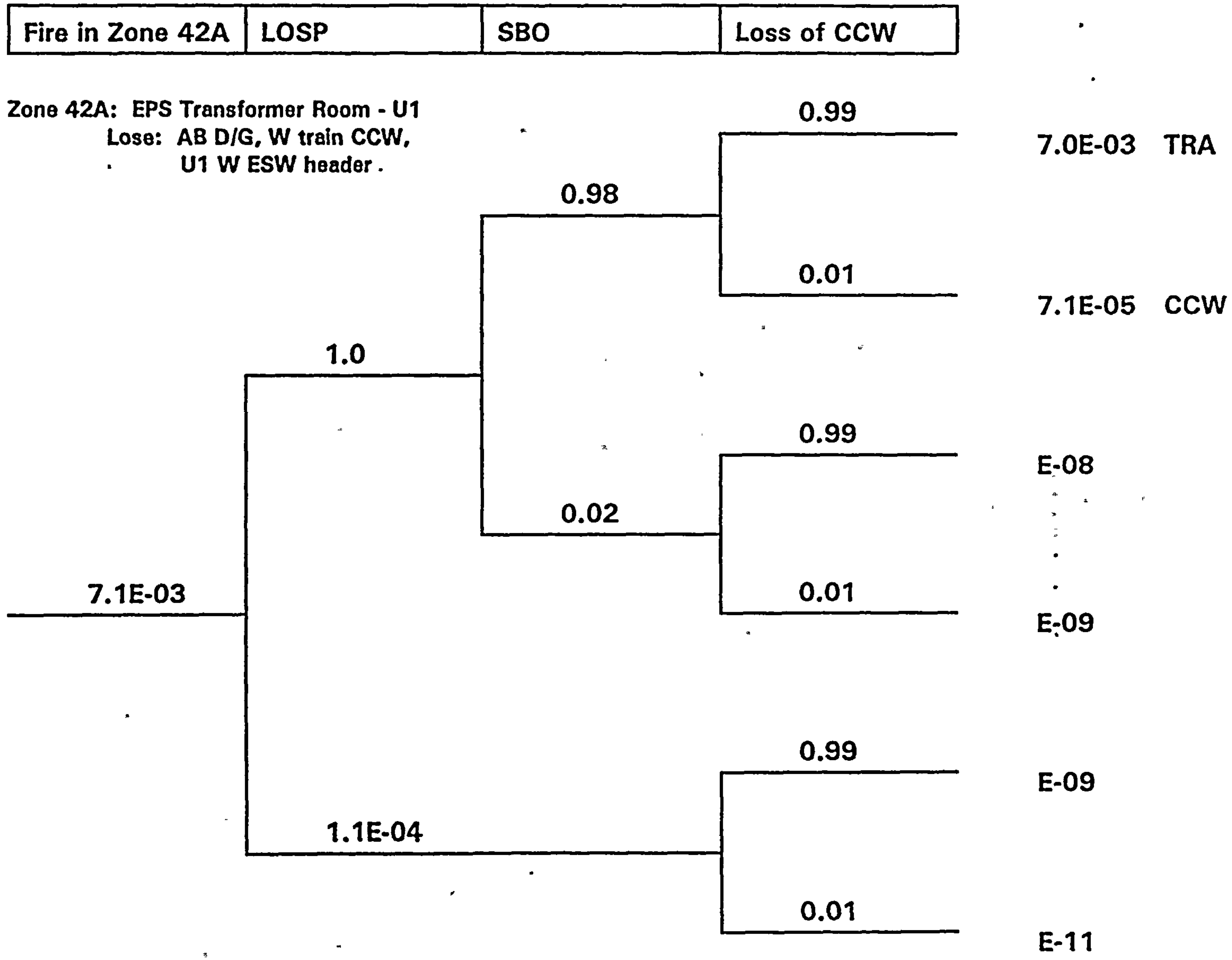

Figure 7 - Event Tree for Zone 42A

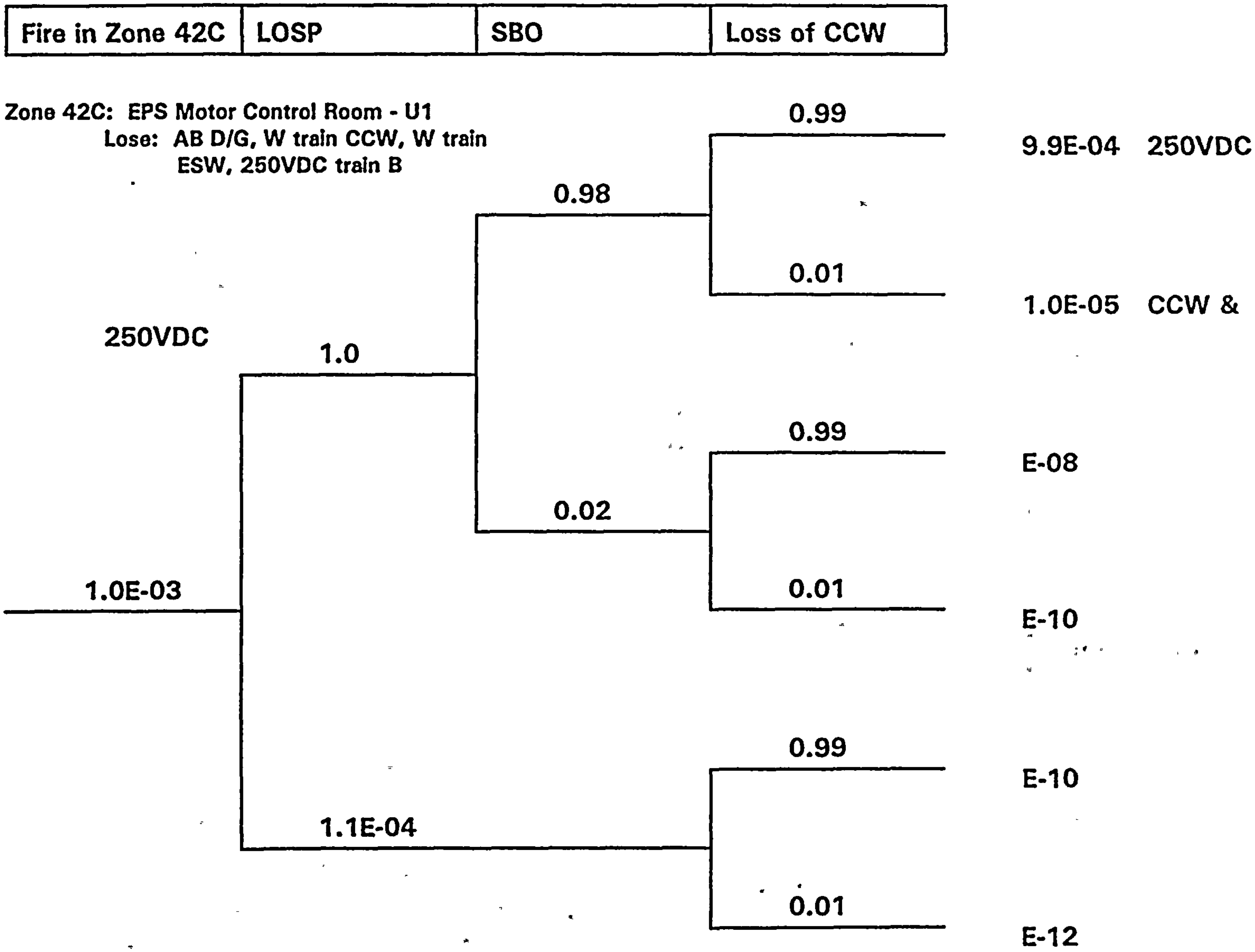

Figure 8 - Event Tree for Zone 42C

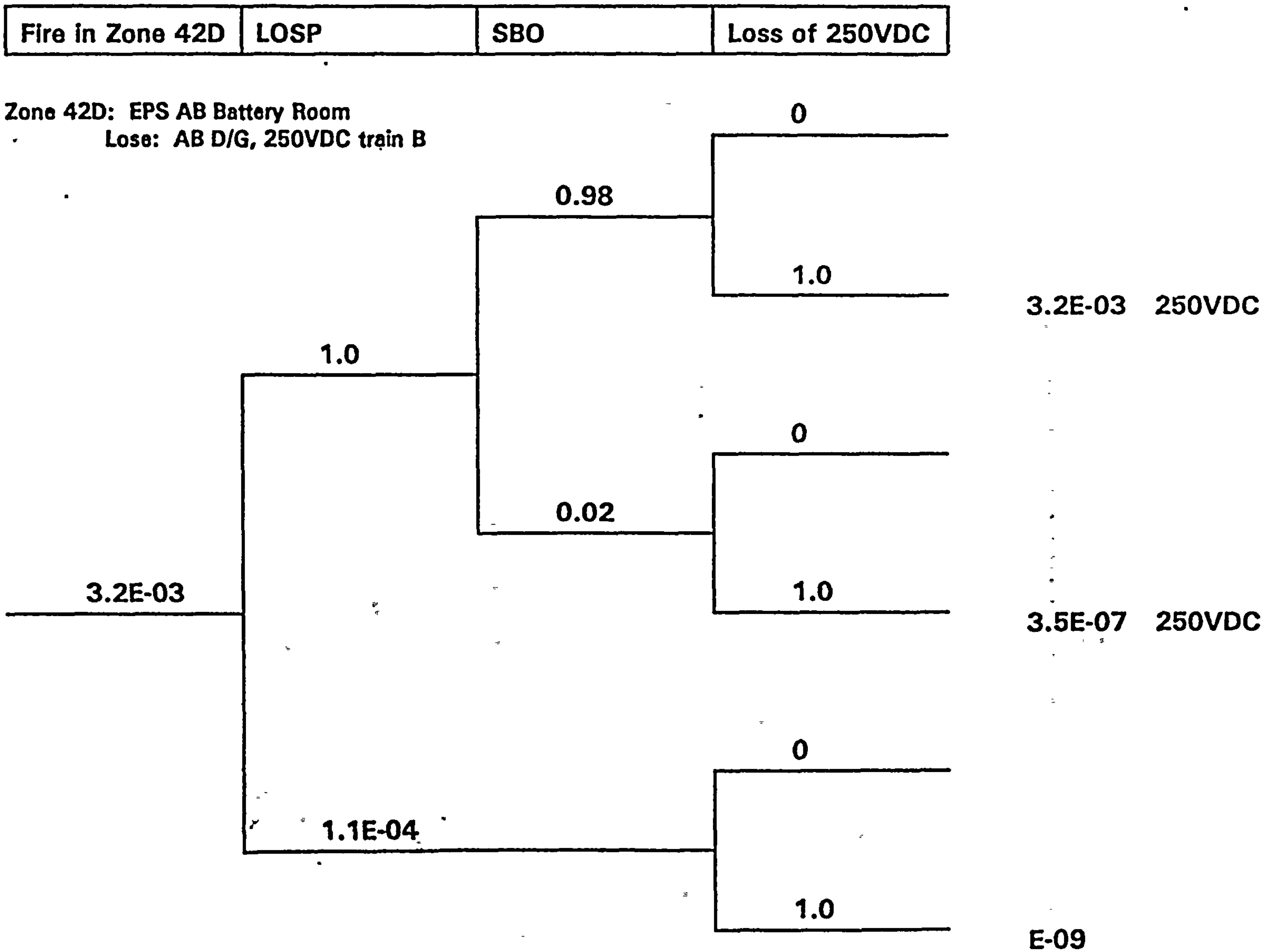

Figure 9 - Event Tree for Zone 42D

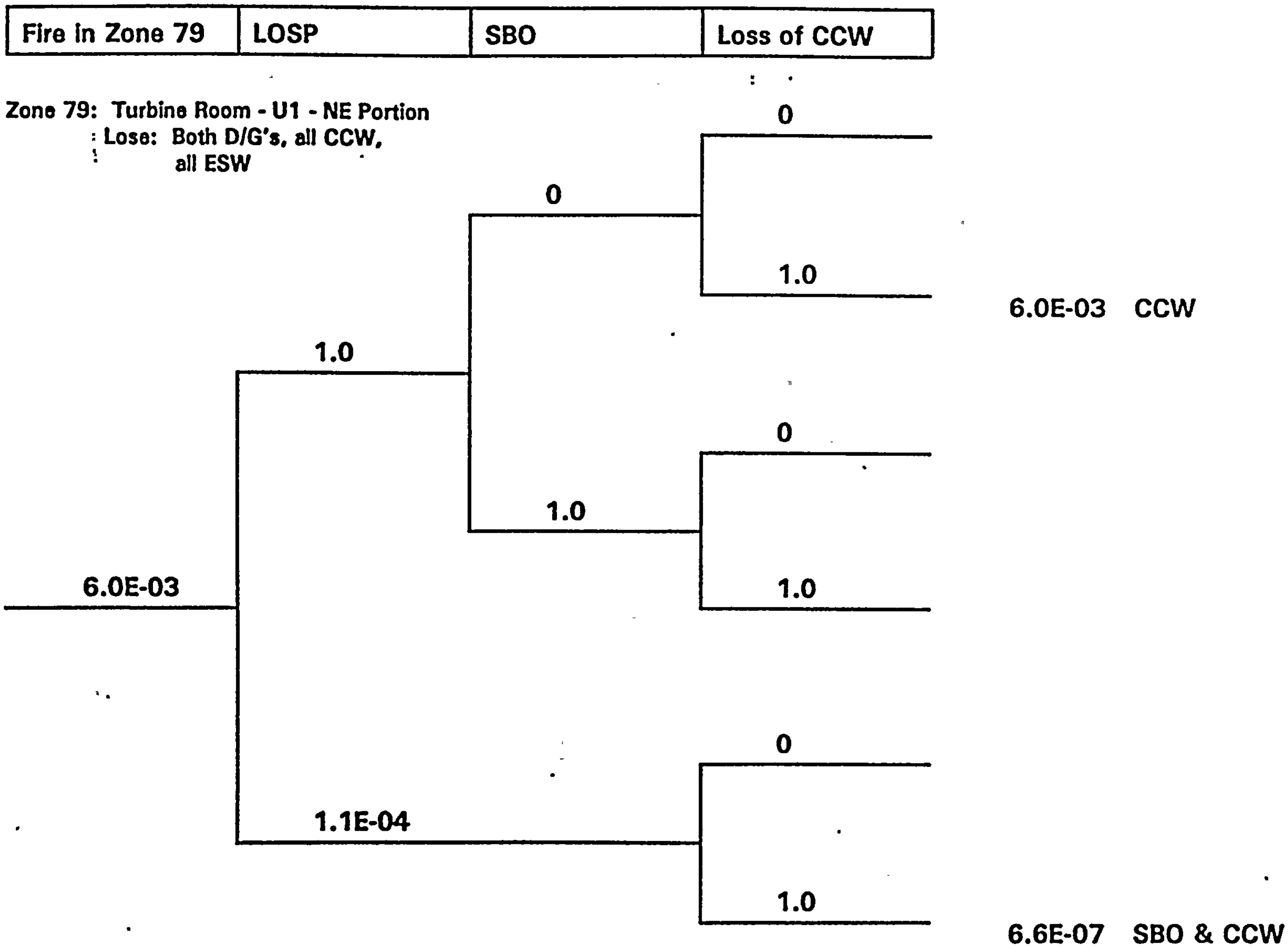

Figure 10 - Event Tree for Zone 79

APPENDIX A
NRC CONCERNS ON FIRE PROBABILISTIC RISK ASSESSMENT FROM JULY, 1994 AUDIT

On July 27-29, 1994, a team from the NRC reviewed the Revision 0 of the IPEEE at the Cook Nuclear Plant site. At the exit, several concerns on the fire PRA were expressed in the exit. This appendix summarizes and explains those concerns. Exit notes can be found in AEP:NRC:1082K.

1) Fire initiation frequency should address ignition sources. Revision 0 appropriately calculated the fire initiation frequency for large areas based on the fire database. However, revision 0 distributed this initiation frequency to the various fire zones by combustible loading. The more appropriate method would be to distribute this by the type of equipment, i.e. the equipment that caused the fires described in the database. The FIVE (Reference 7) methodology uses this approach.

2) Premature Screening (use of normal transient for all screens). Revision 0 incorrectly assumed that all equipment initiated accident sequences were responded to in the transient event tree. This is incorrect. For example, a LOCA can be initiated from a loss of component cooling water because of reactor coolant pump seal failure, which is not addressed in the transient event tree. Therefore, fire induced failure of one train of component cooling water combined with the random failure of the second train would show significantly higher failure frequencies than the transient event tree would indicate.

3) Potential premature screening of control room and cable vault. Revision 0 assumes that evacuation of the control room and use of the auxiliary feedwater crosstie to unit 2 alone is sufficient to avoid core damage. The requirement for continued reactor coolant pump seal cooling and the high failure rate of outside of control room human actions was not considered.

4) Possible concern with our taking credit for auxiliary feedwater crosstie. See 3).

5) Fire Propagation between zones not adequately addressed. Revision 0 only looked at the adequacy of fire barriers. Consideration was not given to fire sources which could be in two zones at once.

6) Fire suppression was credited with eliminating all damage. However, limited damage will occur before fire protection system actuates. The extent of fire damage before the fire can be suppressed should be calculated.

7) It was observed in the walkdowns that a couple of sprinkler heads were upside down, calling into question the fire protection system. This was addressed outside of the scope of the PRA, and was found to be a limited problem.

In October, 1994, a draft revision to the fire PRA addressing these major concerns was presented to the NRC at their offices in Washington. The following is a list of additional concerns identified at that meeting. The general consensus was that the major concerns at the audit were being appropriately addressed. These concerns are summarized in the attached letter from the NRC dated November 14, 1994 (Reference 48).

NOV 21 1994

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

November 14, 1994

cc: P. A. Barrett
S. J. Brewer
E. E. Fitzpatrick
J. A. Kobyra
D. M. H. Malin
B. R. Signet
W. G. Smith, Jr.

Mr. E. E. Fitzpatrick, Vice President
Indiana Michigan Power Company
c/o American Electric Power
Service Corporation
1 Riverside Plaza
Columbus, OH 43215

SUBJECT: REQUEST FOR ADDITIONAL INFORMATION REGARDING THE IPEEE FOR DONALD C. COOK NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2 (TAC NOS. M83609 AND M83610)

Dear Mr. Fitzpatrick:

A meeting was held at our offices on October 25, 1994, between members of your staff, NRC and contractor reviewers to discuss the D. C. Cook IPEEE. Based on that meeting, you have made significant improvements in the IPEEE since our initial audit of the IPEEE at the Cook site in July. A few questions came up at the October meeting for which your staff did not have immediate answers. Therefore, enclosed is a list of additional requests for information based on the discussion at that meeting. Please advise me if you anticipate it will take more than 90 days to respond to these questions. Please call me at (301) 504-3017, if you have any comments or questions.

Sincerely,

John B. Hickman, Project Manager
Project Directorate III-1
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

Enclosure: RAI

cc w/encl: See next page

A-2

cc:

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, Illinois 60532-4351

Attorney General
Department of Attorney General
525 West Ottawa Street
Lansing, Michigan 48913

Township Supervisor
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Post Office Box 818
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Al Blind, Plant Manager
Donald C. Cook Nuclear Plant
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Resident Inspector Office
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Stevensville, Michigan 49127

Gerald Charnoff, Esquire
Shaw, Pittman, Potts and Trowbridge
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Washington, DC 20037

Mayor, City of Bridgman
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Bridgman, Michigan 49106

Special Assistant to the Governor
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Department of Public Health
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Lansing, Michigan 48909

Mr. S. Brewer
American Electric Power Service
Corporation
1 Riverside Plaza
Columbus, Ohio 43215

Additional Requests for Information

Regarding the Individual Plant Examination

of Externally Initiated Events (IPEEE)

Based on Meeting with Licensee on October 25, 1994

Fire:

1. Assuming a zero frequency of fire for a compartment (as was done for the three cable vaults) is not acceptable practice. The conditional core damage frequency from loss of all cables in any of the three cable vaults is equal to 1.0, without any recovery actions. Please provide an analysis of the recovery actions assuming complete loss of all cables in each of the three cable vaults (individually).

2. There is a potential for turbine building fire to damage cabinets in fire zones 41 and 42A, simultaneously. This is due to the normally open roll-up door separating zone 42A from the turbine building. Please provide an analysis of the plant response given simultaneous damage to all cables and equipment in these two rooms.

3. Human error analysis from Seabrook was used for the control room fire analysis in the IPEEE fire addendum. Since it is important for understanding the ability of the plant operating staff to deal with fire events in areas where a large number of safety-related systems may be affected, please provide the basis for using the Seabrook human error analysis for D. C. Cook scenarios.

4. The main feedwater cables have not been traced at D. C. Cook. The assumption that for some zones/areas, the Power Conversion System (PCS) may survive is not well founded. Please provide clarification as to the contribution of the main feedwater to the screened out fire zones/areas and the significant core damage scenarios.

Seismic:

1. Please describe the rationale for selection of the 11 components for seismic fragility re-evaluation. Justify how this rationale insures that all components likely to control plant seismic capability and severe accident risk (both for core damage and for radioactive release) have been considered in the re-evaluation.

2. Please provide the data, calculations, and results for the seismic fragility re-evaluations that were performed for the 11 selected components.

3. Please provide a description of the development of soil-structure interaction (SSI) margin factors used in the fragility re-evaluations. Justify the basis for not generating new in-structure spectra for development of the SSI margin factors. Explain how these SSI margin factors have been applied.

4. Please provide an explanation as to why the 600 VAC transformers, RPS [reactor protection system] panels, turbine building pedestal and 250 VDC system have dropped out of the dominant contributor list, whereas cable trays have been introduced to this list.

5. Please provide a discussion of how the IPEEE seismic addendum addresses/impacts the containment performance assessment. Other than the list of dominant contributors identified for core damage frequency, what are the dominant contributors to containment failure (i.e., early release or large late release) and failure of accident mitigation systems? Do the new fragilities alter the containment performance insights presented in the original IPEEE submittal?

6. Please provide a discussion of the peer review process and its results as applied to the seismic addendum/re-evaluation.

7. For all recommended actions/fixes identified in the seismic IPEEE walkdowns (including all items documented by the licensee's walkdown contractor), please provide a table delineating the recommended action/item, its analysis and/or treatment in the seismic IPEEE process, and its disposition status.

APPENDIX B
SENSITIVITY ANALYSIS TRA RUN SUMMARIES

Section B.1 of this appendix contains the cable listings, sensitivity analysis run outputs and estimated core damage frequency calculation for the initial screening assessment on the TRA only fire zones. Section B.2 contains TRA fire zones that were evaluated using engineering judgement. Section B.3 contains the sensitivity analysis run inputs and outputs for some of the fire zones which required detailed evaluations. Section B.4 contains the quantification outputs for CCW and TRA events which were used in the fire screening evaluations and also the Risk Achievement rankings for these events.

APPENDIX B
SENSITIVITY ANALYSIS RUN SUMMARIES

(Note: This appendix contains computer output.
It was not included to reduce the volume of the submittal.)

APPENDIX C
WALKDOWN FINDINGS

Section C.1 of this appendix contains the notes from the walkdowns performed on September 8 and September 22, 1994. Section C.2 contains the notes from the walkdown of Zone 6M, performed on November 10, 1994.

Appendix C.1
Notes from Fire Walkdowns
Performed on 9/8/94 & 9/22/94

Zones 6N, 40A, 40B, 41, 42A, 42C, 43, 44N, 44S, 51, 52, 55 and 79 were walked down on September 8 and September 22, 1994. The notes from these walkdowns are included below:

Zone 6N

Critical cable trays in zone: 1AZ-C21, 23, 25, 27, 64, 70, 72.

1-ABV-A and 1-ABV-D are far from each other (≈ 60-70').

The lowest elevation, and most conservative location, of critical cables is above the busses (at ≈ 7'). The critical cable trays travelled (at most) ≈ 3' past the busses at this elevation, then they ran vertical against the wall until they reached an elevation of ≈ 15-20', where the trays ran horizontal away from the wall.

The critical cable trays seemed to be all closed.

Transient combustibles in this zone include an RP desk with bookshelves next to it and a garbage can. (There is also an RP monitor cabinet and a frisking station in this zone that are fairly close to MCCs.)

1AZ-C64 runs vertical at wall for about 6' with open (grated) cable tray 1AZ-P11 right next to it, and 1AZ-C62 on the left side (about 1.5' away).

1AZ-C27 and 1AZ-C30 run next to each other vertical against the wall for about 6', and then go up and up.

There is metal conduit below many of the cable trays, however, this is not considered to be an intervening combustible.

The layout of C70 (red) is similar to that of C27 (green).

The sketched walkdown notes for this zone are included as Figures C.1-1 and C.1-2.

Zone 40A

Critical cable trays in zone: 1EI-C23, 24, 25, 26, 27, 28, 29, 30

1EI-C30 comes into zone about 2' above fire door (≈10' up), and ≈3' from edge of wall

Bus T11A has dimensions: 26'0" (l) x 4'10" (w) x 6'8" (h). (It has 12 compartments, that are 2'2" wide, which gives a total bus length of 26'.)

The critical cable trays are mostly closed (i.e., small sections, about 17" long, are grated). The lowest elevation, and the worst location of the trays, is above the busses (at ≈ 6' 9").

The sketched walkdown notes for this zone are included as Figure C.1-3.

Zone 40B

Critical cable trays in zone: 1EI-C1, 2, 3, 4, 5

The critical cable trays are mostly closed (i.e., small sections, about 17" long, are grated). The lowest elevation, and the worst location of the trays is above the busses (at ≈ 6' 10").

Bus T11D has dimensions: 26'0" (l) x 4'10" (w) x 6'8" (h). (It has 12 compartments, that are 2'2" wide, which gives a total bus length of 26'.)

1EI-C1 and 1EI-C54 run right next to each other for the last several feet before the ceiling (≈5').

Several open (i.e., grated) non-critical cable trays run through upper portion of zone.

The only fire protection headers are located ≈7' above bus T11D. There are other red supply lines.

1EI-C4 is only about 2' from the edge of the wall where it enters the ceiling. 1EI-C5 comes into room from about 9' up and ≈ 2' from edge of wall.

The sketched walkdown notes for this zone are included as Figure C.1-4.

Zones 41 and 42A

Critical cable trays in zone: 1EM-C6, 1EI-C5, 6, 7, 13, 15, 30, 33, 34, 35, 36, 37, 40 and 1CT-P22 and P31 (in cable spreading area underneath 613' elevation).

Busses 11A, 11B, 11C and 11D are all 4'10" deep and 7'6" tall. Busses 11A and 11B are 7' long (two 2' compartments and two 1.5' compartments). Bus 11C is 8' long (four 1.5' compartments and one 2' compartment). Bus 11D is 6'6" long (three 1.5' compartments and one 2' compartment.)

Busses 11A, 11B, 11C and 11D are 2' back from the end of the fire wall.

Critical cables in conduit that were found: 1-8356G, 1-8789G, 1-8862G.

1-8356G is in conduit, with its lowest and most vulnerable position being above 1-EZC-D (at 7' 10").

Many of the AB battery charger control cables are in cable trays 1CT-P22 and 1CT-P31, which are located in the cable spreading area underneath the floor.

The sketched walkdown notes for this zone are included as Figures C.1-5 and C.1-6.

Zone 42C

Critical cable trays in zone: 1EI-C93.

1EI-C93 exits 1-MCAB (at ≈ 8'), runs horizontal to opposite wall (about 2' above fire door), and then vertical up wall, next to 1EI-D1.

The sketched walkdown notes for this zone is Figure C.1-7.

Zone 43

The critical cables that are supposed to be in this zone could not be found (1-8501R and 1-8502R). They might run above tiled ceiling.

The sketched walkdown notes for this zone are included as Figure C.1-8.

Zone 44N

The critical cable trays in this zone were not identified due to their large number and due to the large size of the zone.

There is a large dress out area in the zone, with a large volume of PC clothing. There is safety related conduit (green and red) ≈ 3' above this area (elevation ≈ 7').

There is a garbage can with a lid in this zone. Cable tray 1AZ-C20 runs vertical along wall, only 2" from garbage can, and it looks like it is wrapped in Thermolag. Green safety related conduit runs about 3'6" above the top of the can (elevation ≈ 7').

Cable trays 1AI-P2 and 1AI-C5 run ≈ 5' above the RP desk (≈ 10' elevation). Cable trays 1AI-P1 and 1AI-C1, which run above VCC 1-AZV-A, are about 5' from the edge of the RP desk at an elevation of ≈ 8', and are right over the edge of the desk at an elevation of ≈ 10' (≈ 5' from top of desk).

The sketched walkdown notes for this zone are included as Figures C.1-9.

Zone 44S

No transient combustibles were identified in this zone, therefore, the information from Revision 0 of the Fire PRA is sufficient. (This zone was analyzed in detail in Revision 0 of the Fire PRA.)

The sketched walkdown notes for this zone are included as Figure C.1-10.

Zone 51

Critical cable trays in zone: 1AU-C4, C13

The only transient combustibles located was a 55 gallon barrel of used oil. The barrel was locked and chained to the floor, and the opening to the barrel was locked. The lower half of the barrel was surrounded by some type of fire proof oil retainer.

Cable trays 1AU-C4 and C13 run along wall from floor to ceiling (elevation ≈ 15'), and then run horizontally along ceiling. They are mostly closed, except for sections where they are cross-tied to another cable tray.

There is ≈ 40' between the critical cable trays and the MCCs.

The sketched walkdown notes for this zone are included as Figure C.1-11.

Zone 52

Critical cable trays in zone: 1AU-C3, 4, 7, 8, 10, 11.

Critical cable trays are metal, about 7.5' up, with good separation between them. There was no noticeable combustibles around critical cable trays.

There is a flammable storage cabinet about 21' from the MCCs. The cabinet is designated as a $10ft^3$ cabinet with oils and solvents, and is tied down with a thick metal strap. There is nothing directly above the cabinet, but cable trays 1AU-C4 comes within ≈ 12' of the cabinet (≈ 9' horizontal distance from cabinet, at an elevation of 13'). Other critical cable trays come within 12' (horizontal) of the cabinet, at an elevation of 7.5'.

1-AM-A and 1-AM-D are about 21' apart. The N-train battery charger is within 12' of 1-AM-D.

The sketched walkdown notes for this zone are included as Figure C.1-12.

Zone 55

This zone was walked down to examine the walls that separate the charger and battery rooms from the rest of Zone 55. Thick concrete walls and an asbestos wall are used to separate the batteries and chargers from the critical cable trays and conduit in the zone.

The sketched walkdown notes for this zone are included as Figure C.1-13.

Zone 79

Critical cable trays in zone: 1AZ-C34 (and conduit).

There were a lot of non-safety related cable trays (i.e., not green or red) in the main portion of zone. They started with "1TZ".

Cable tray 1AZ-C34 runs horizontally across the ceiling (elevation ≈ 15') in the diesel generator corridor. There is nothing below it and it is fire wrapped.

The red critical cables are also in this corridor, wrapped in conduit. Conduit (safety related and non-safety related) ran horizontal and vertical in the corridor. None of the safety related vertical conduit was identified to be critical, so it is assumed that the red critical conduit ran along the ceiling of the corridor.

The sketched walkdown notes for this zone are included as Figure C.1-14.

Figure C.1-1
Zone 6N

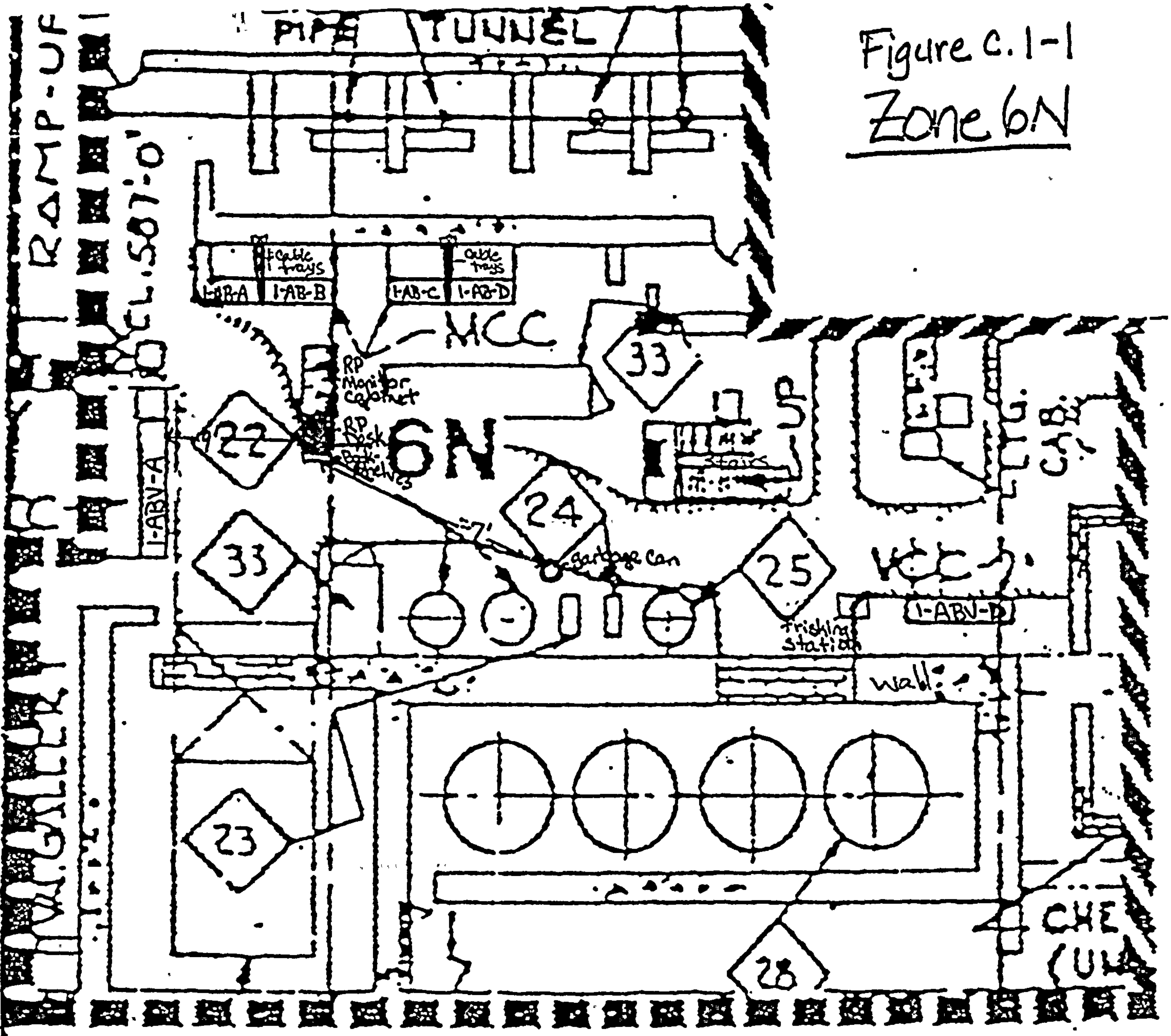


Figure C.1-2
Zone 6

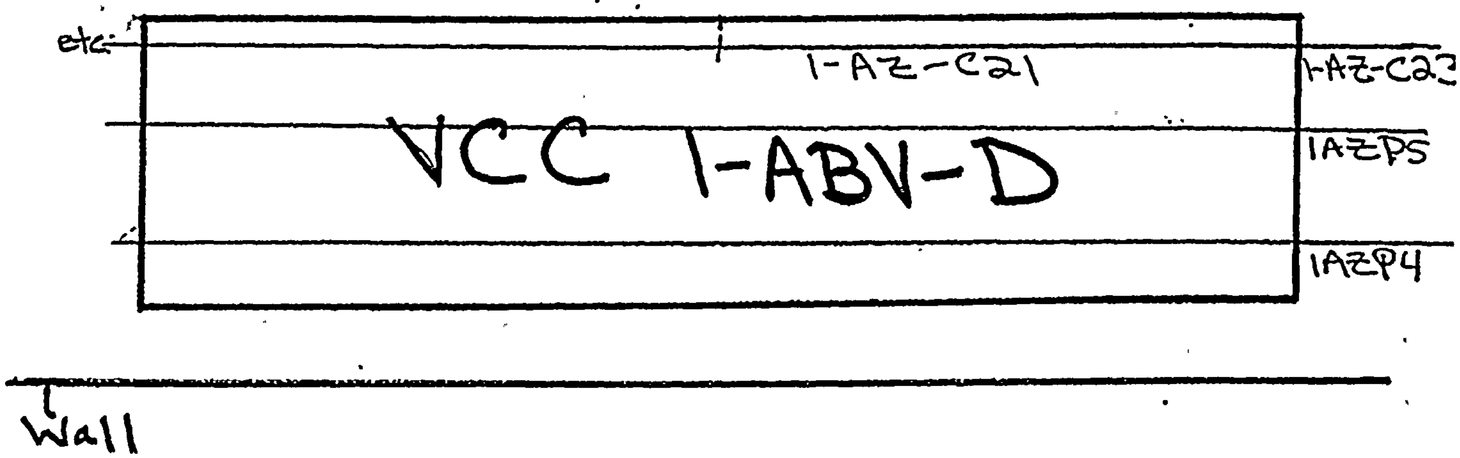
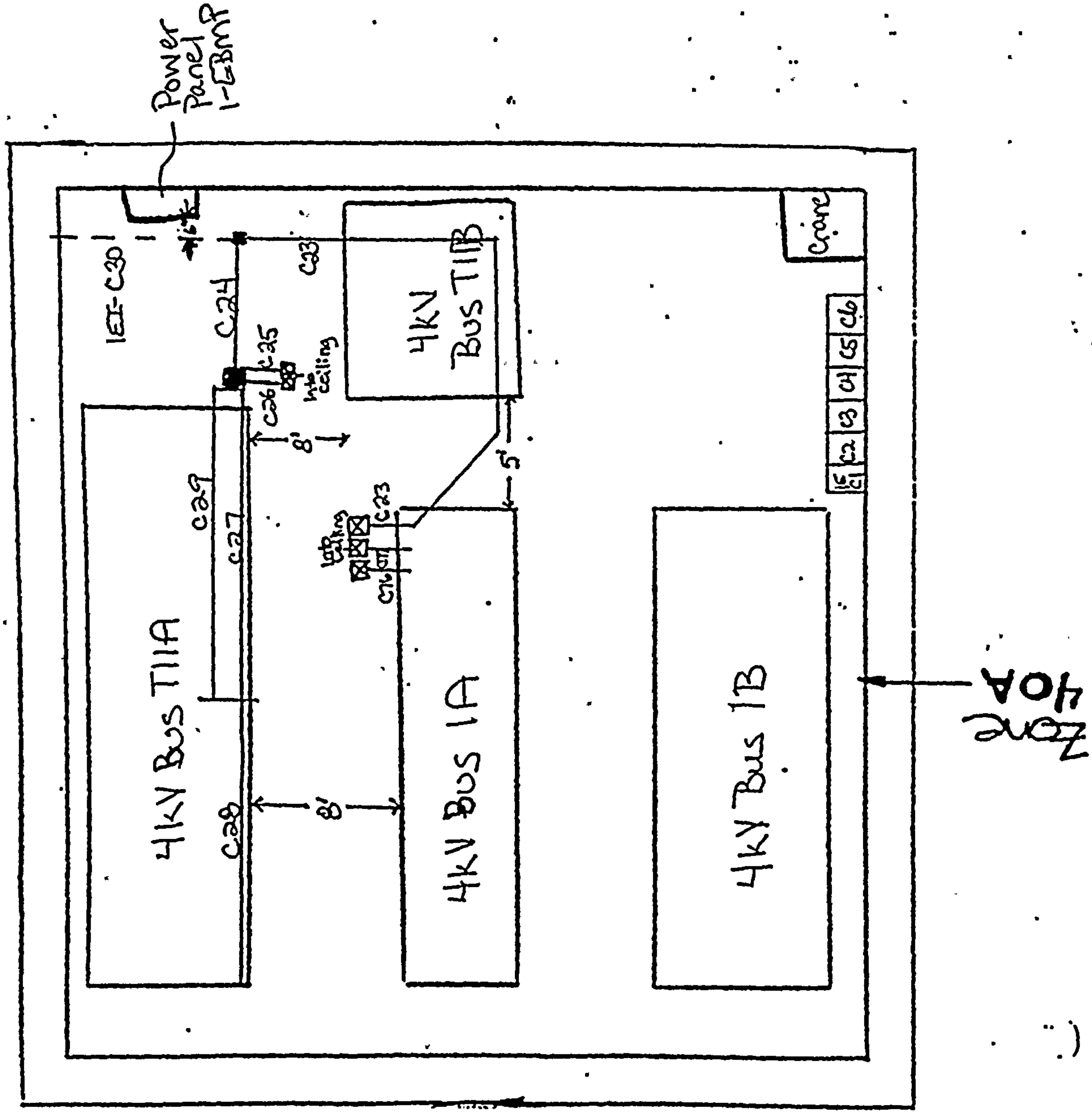


Figure C.1-3
Zone 40A



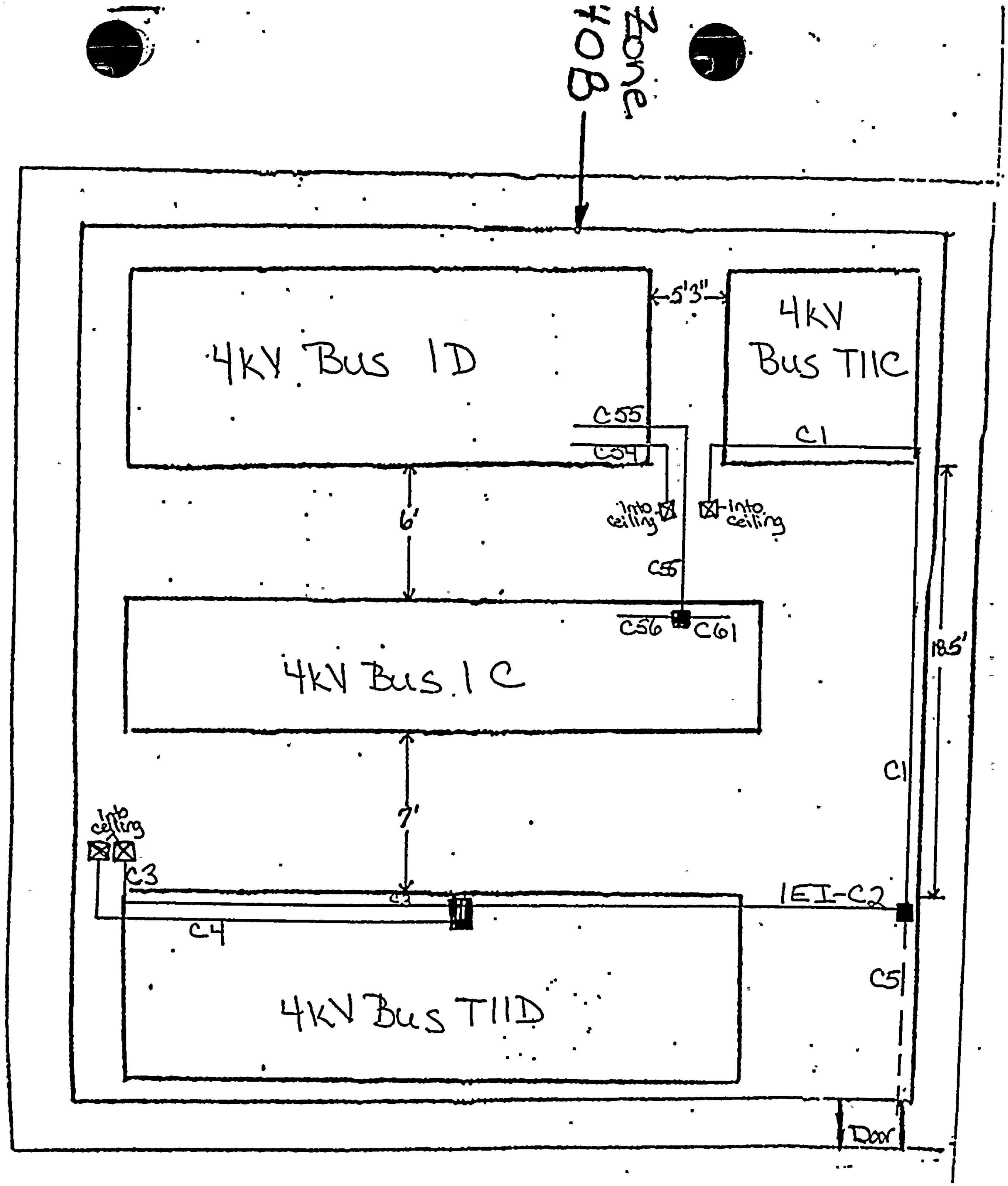
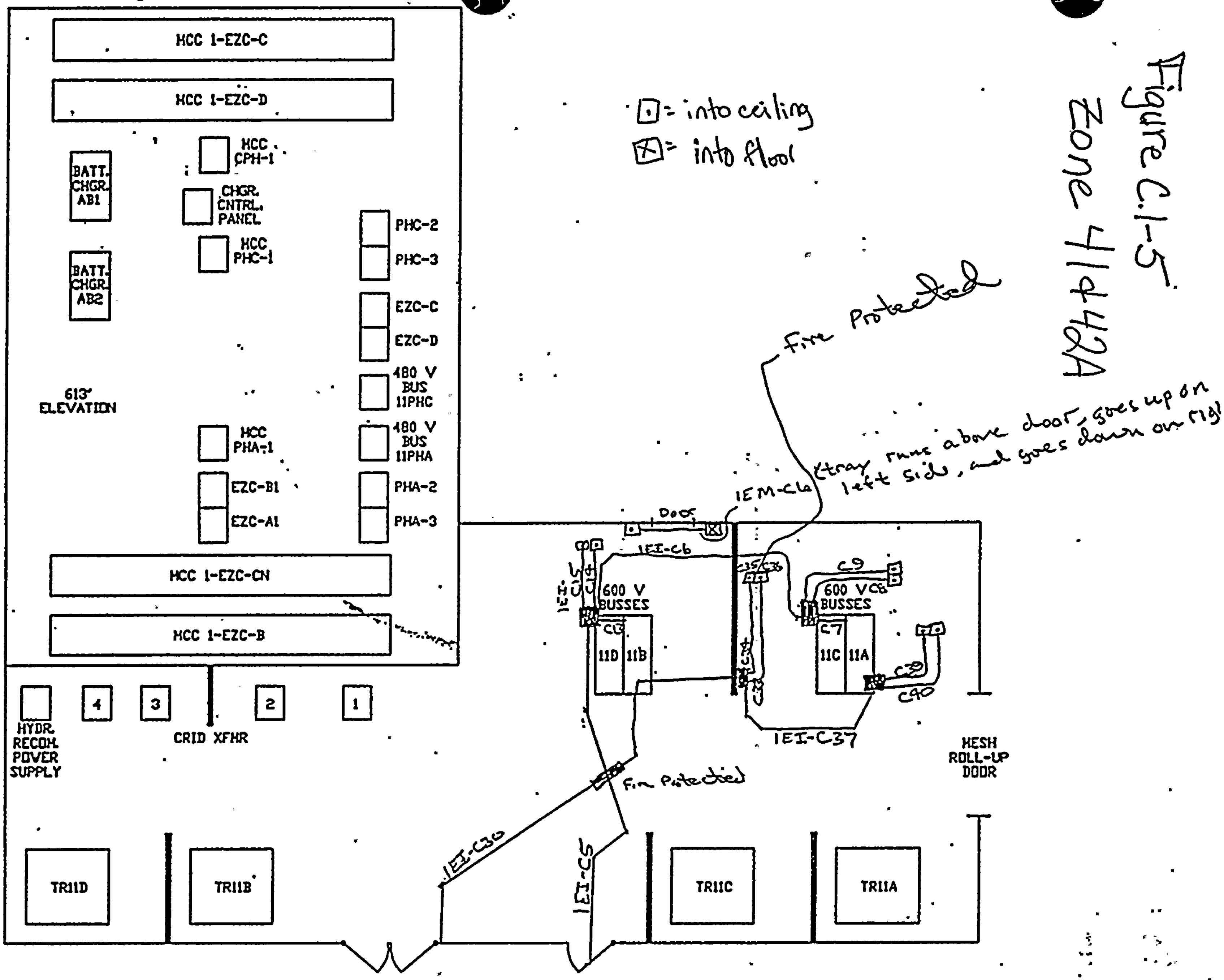


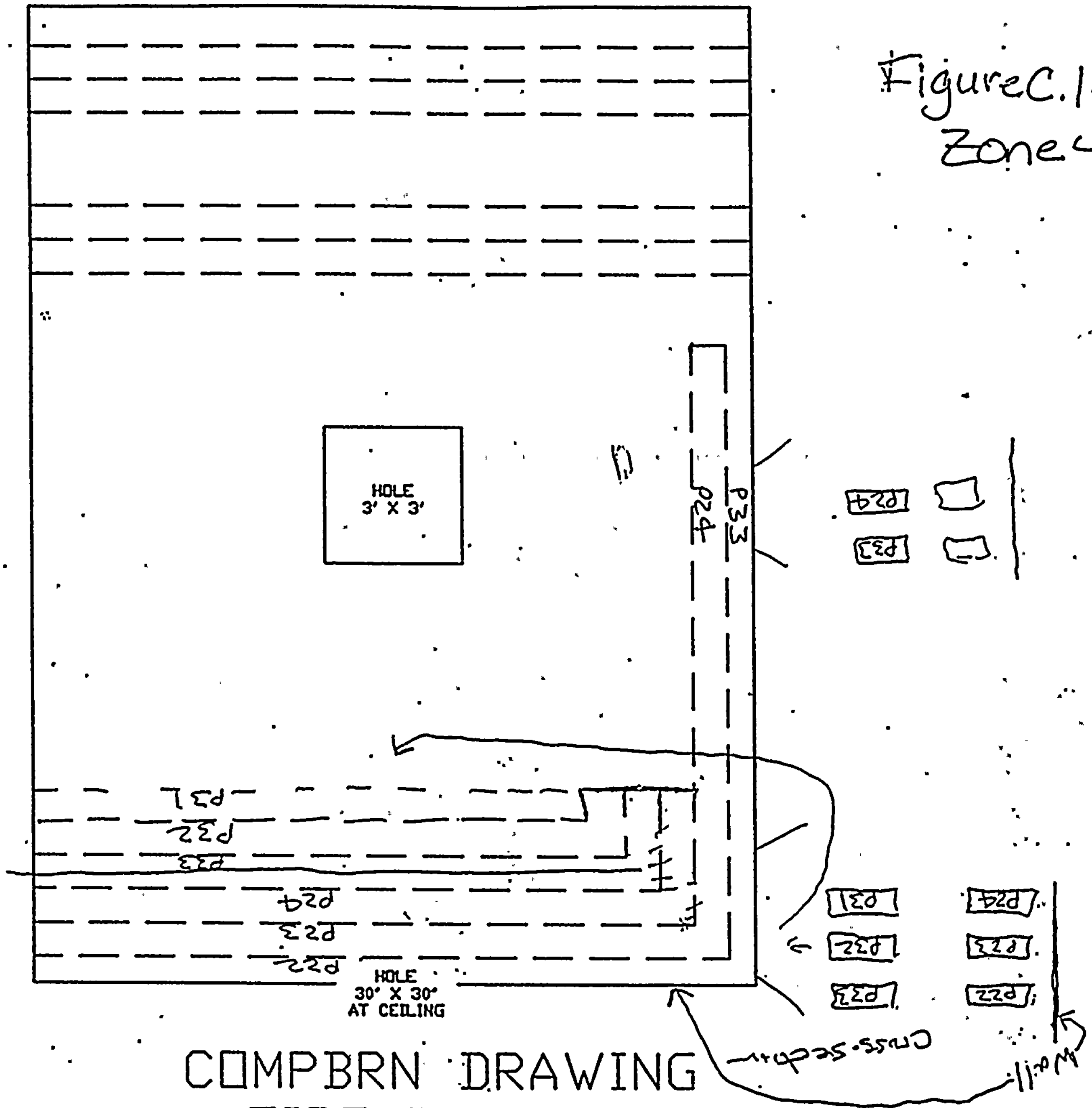
Figure C.1-4
Zone 40B

Figure C.1-5
Zone 41&42A

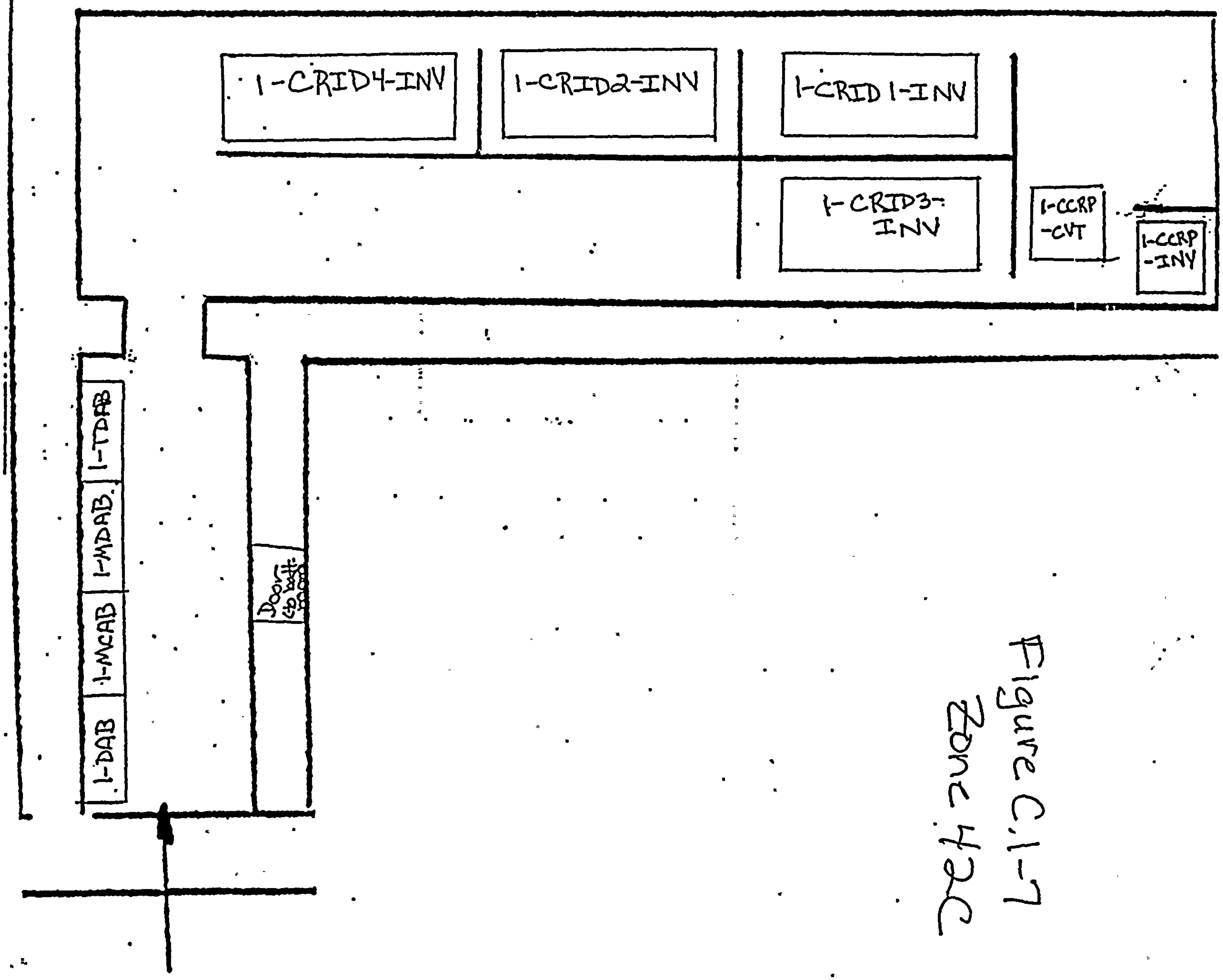


COMPBRN DRAWING
FIRE ZONES 41&42A

Figure C.1-6
Zone 41



COMPBRN DRAWING
FIRE ZONE 41
CABLE SPREADING AREA
UNDERNEATH 613' ELEVATION

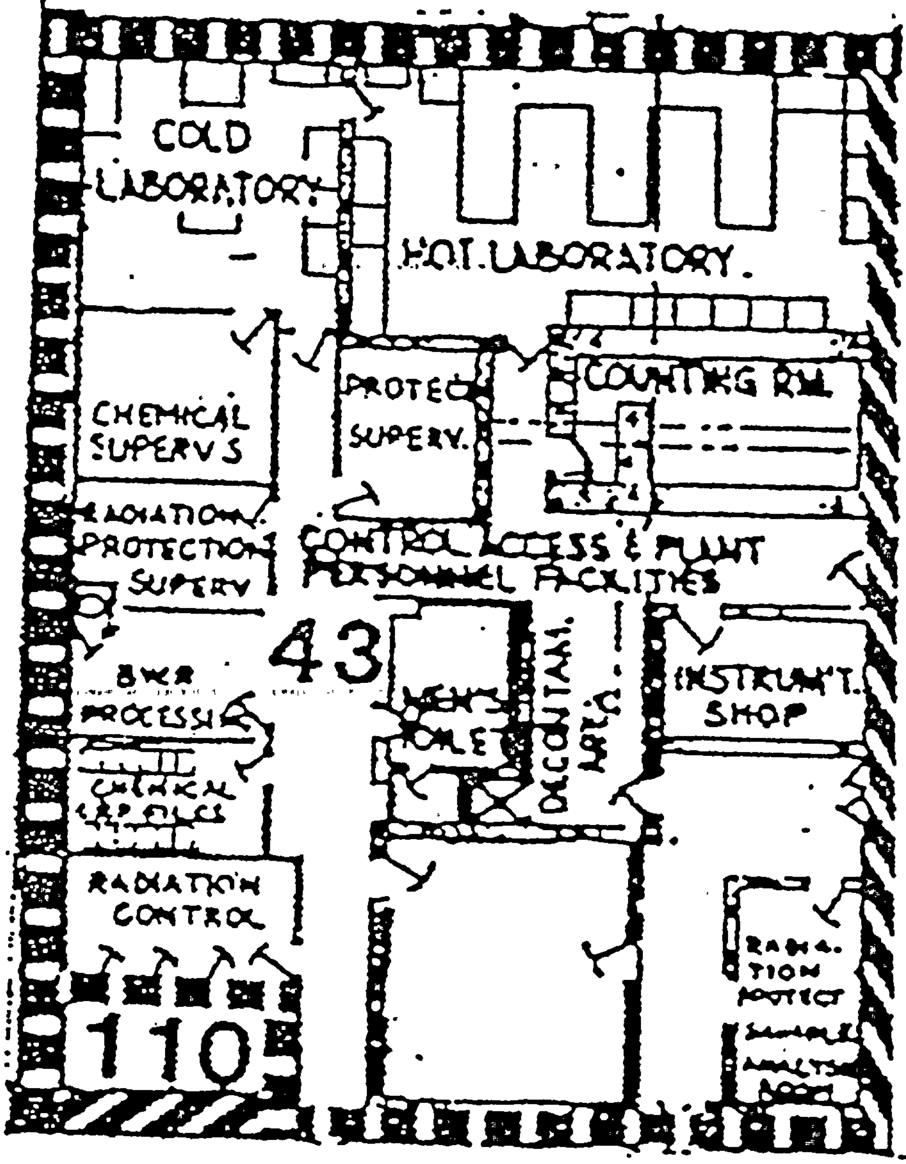


Zone 42C

Figure C.1-7 Zone 42C

Zone 43

Figure C.1-8
Zone 43



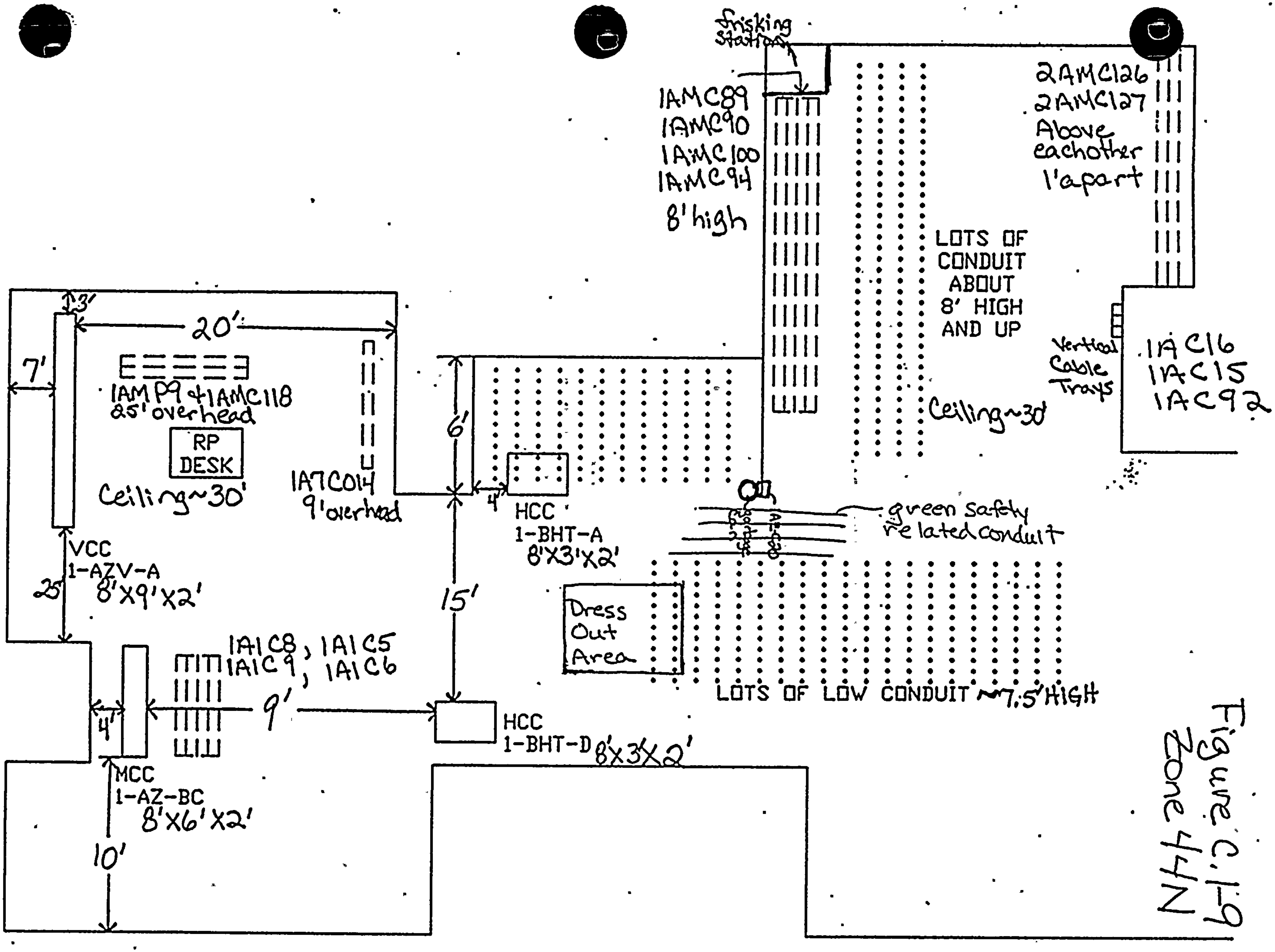
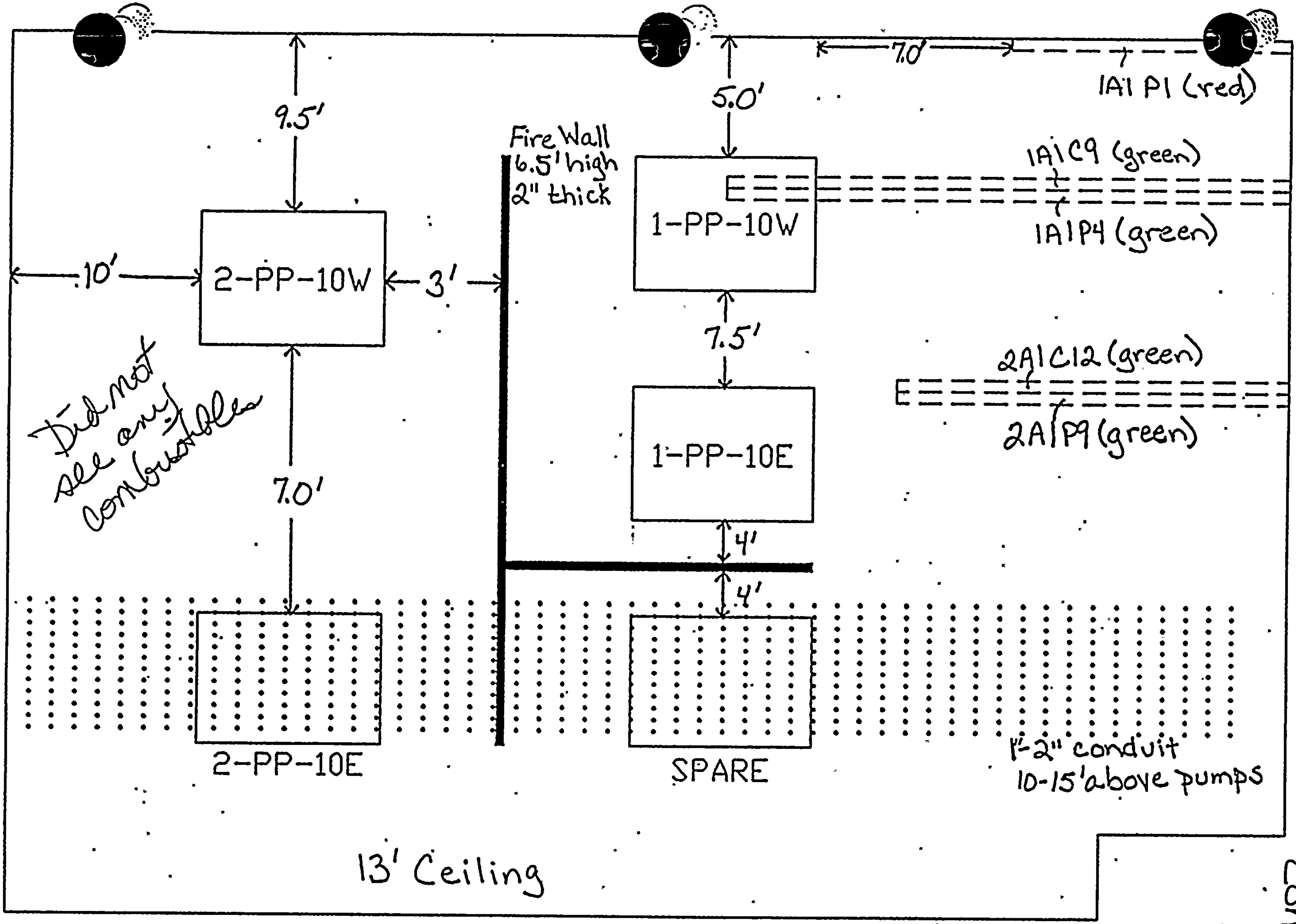


Figure C.19 Zone 44N

NOTES:

1. ▭▭▭ = CABLE TRAY
2. · · · · · · = CONDUIT

COMPBRN DRAWING
FIRE ZONE 44N



NOTES:
1. ⊏⊐⊏⊐⊏⊐ = CABLE TRAY
2. • • • • • • = CONDUIT
3. Fire protection everywhere around pumps/motors

COMPBRN DRAWING
FIRE ZONE 44S

(from Revision 0 of the Fire PRA)

Figure C.1-10 Zone 44S

Zone 51

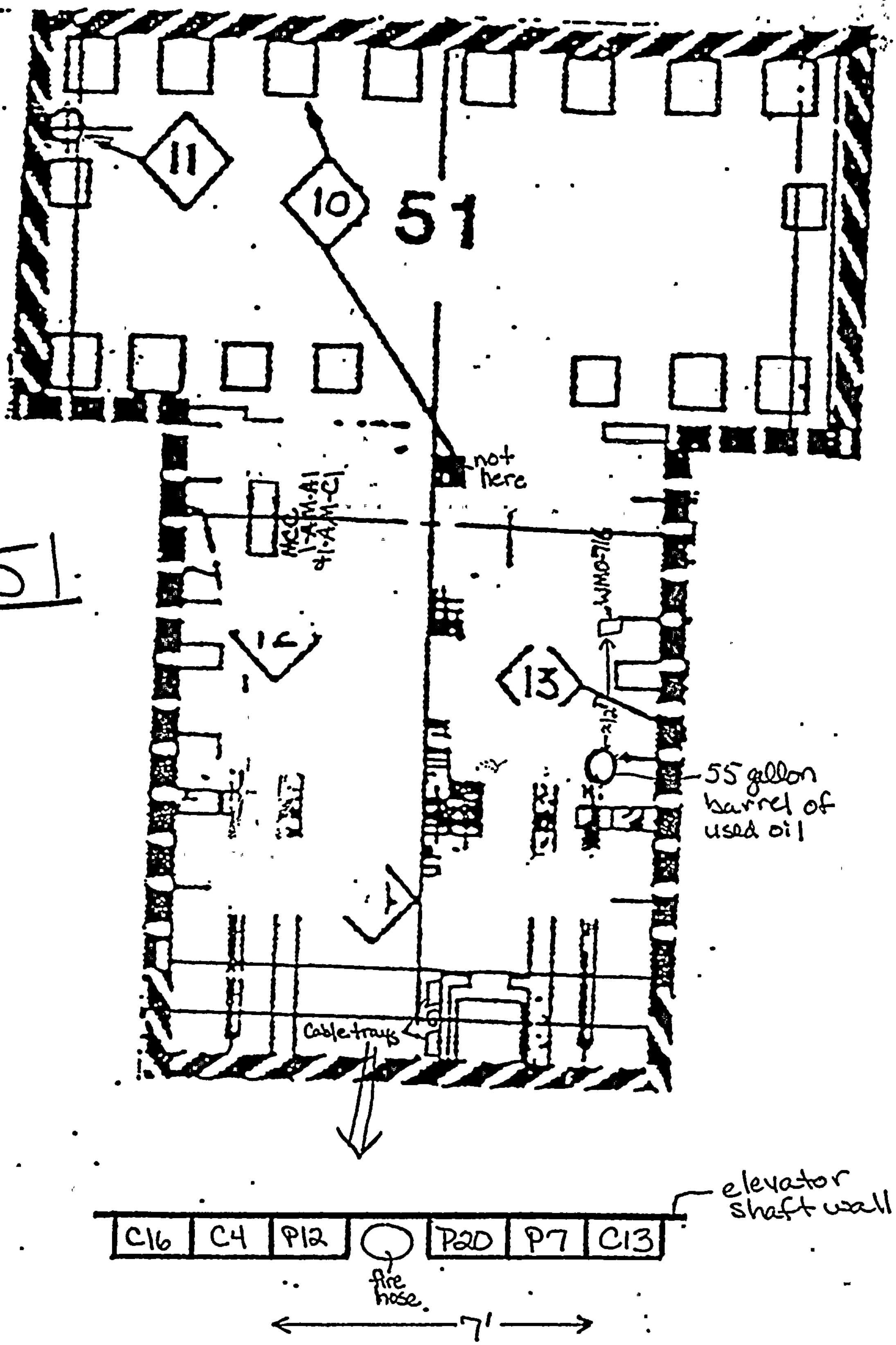
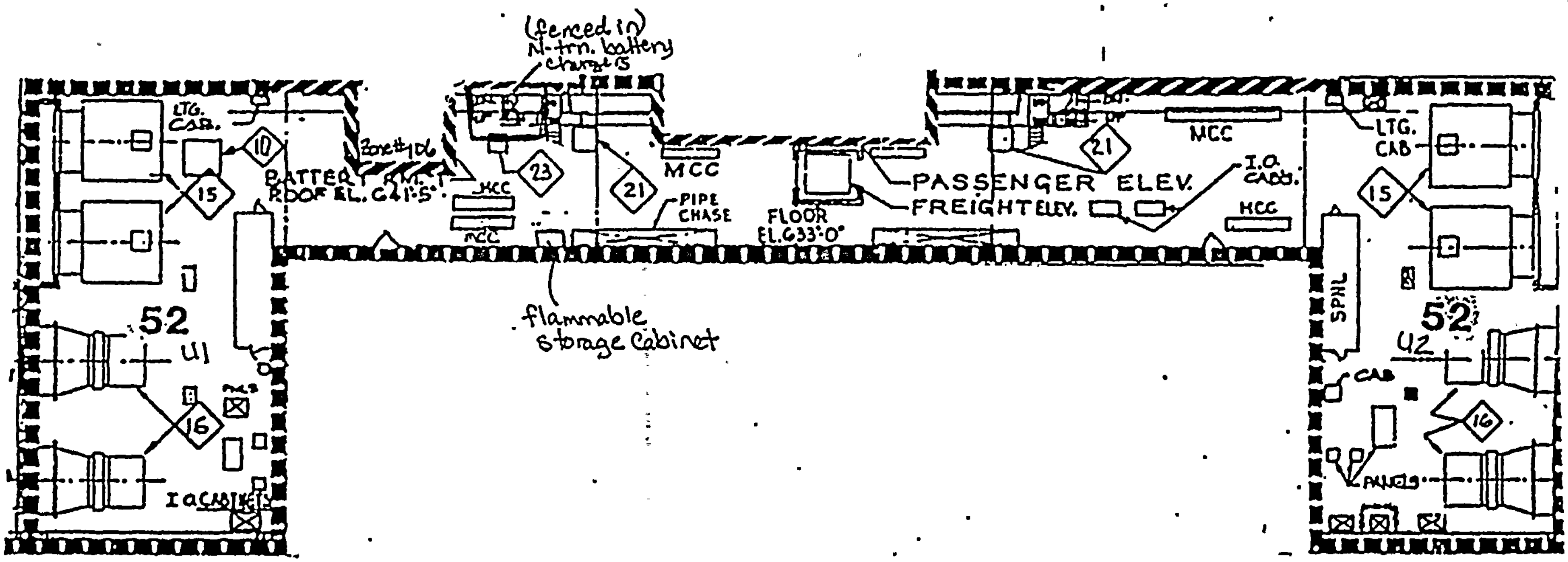


Figure C.1-11
Zone 51



Zone 52

Figure C.1-12
Zone 52

Figure C.1-13
Zone 55

All critical conduit and cable trays are outside battery room & charger room walls.

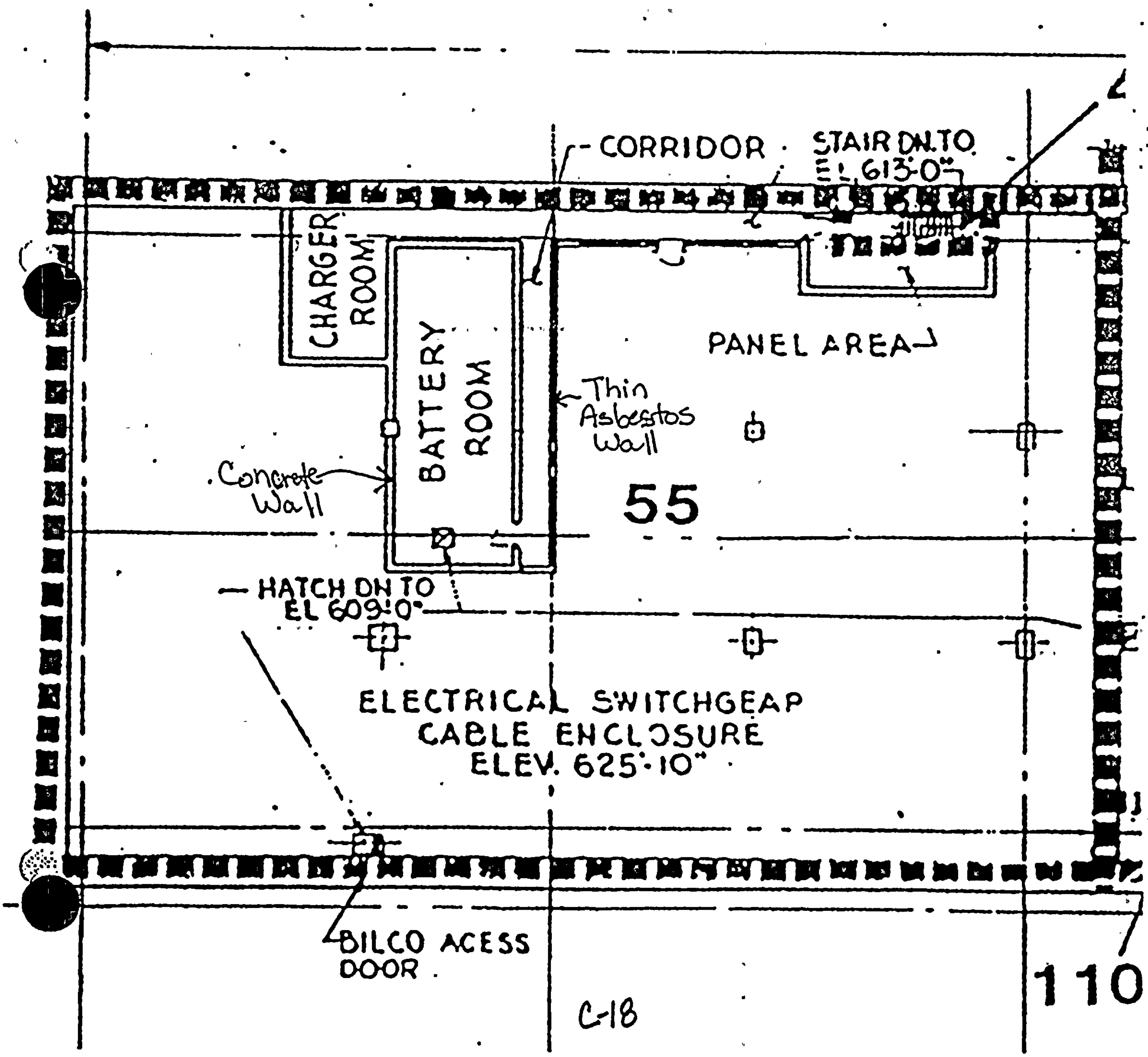
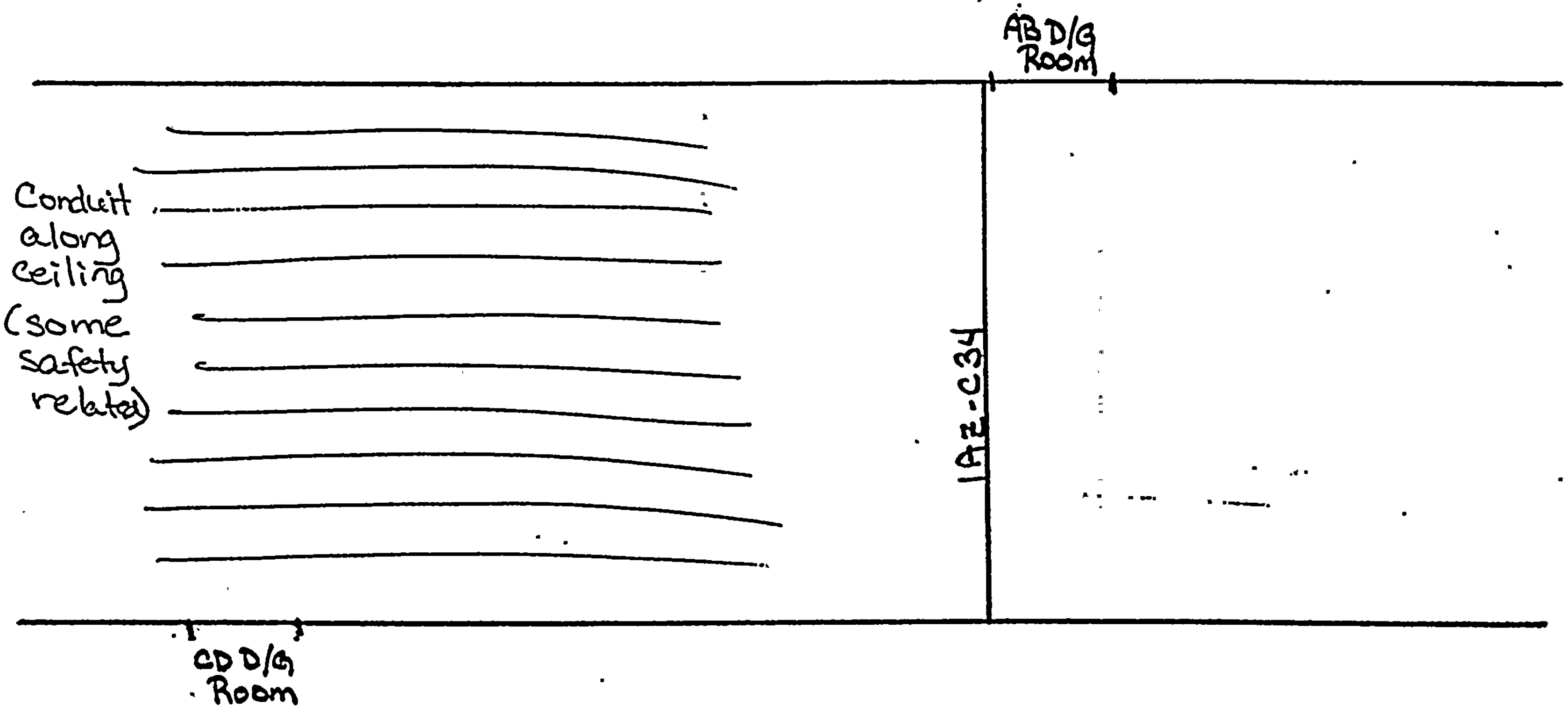


Figure C.1-79

Zone 79 -

Diesel Generator Corridor Section



Appendix C.2
Notes from Fire Walkdown
Performed on 11/10/94

This walkdown was performed for Zone 6M only, on November 10, 1994. The notes from this walkdown are included below:

Zone 6M

Cabling for all three trains of auxiliary feedwater for both units pass through this zone according to the SCC. The primary interest in this walkdown is the location of those cables, and potential fire sources and combustibles in the zone.

The zone is comprised of the boric acid tank room, the section of hallway east of that room including the elevator shafts, and the seal water filter rooms. The cabling of interest was located at the west wall of the boric acid tank room. The cables entered in conduit at about 10 feet elevation, lower to about 8 feet, and turn immediately to exit the room. One conduit, 80180G-2, was only at 6' elevation. For the auxiliary feedwater cabling for the two units, the wall penetrations are about 20 feet apart, with a concrete cable tunnel separating them. The pipe tunnel extends about 6 feet into the room. This is the closest the two sets of cables get, since the cables turn toward their respective units.

There is miscellaneous electrical equipment about four feet in front of the cable penetrations. All the equipment is in typical electrical cabinets, and a small (3'high) enclosed transformer is on the Unit 1 side. No combustion sources were noted in this room, and by discussion with R. Leonard, the CVCS system engineer, no combustibles are ever stored in the tank room. Four small boric acid transfer pumps are in the room, separated from the cables by the tanks.

Small transient sources (anti-Cs in 3' tall wire mesh bins) were found in the hallway area near the elevator shafts.

The sketched walkdown notes for this zone are included as Figure C.2-1.

Figure C.21
Zone 6M

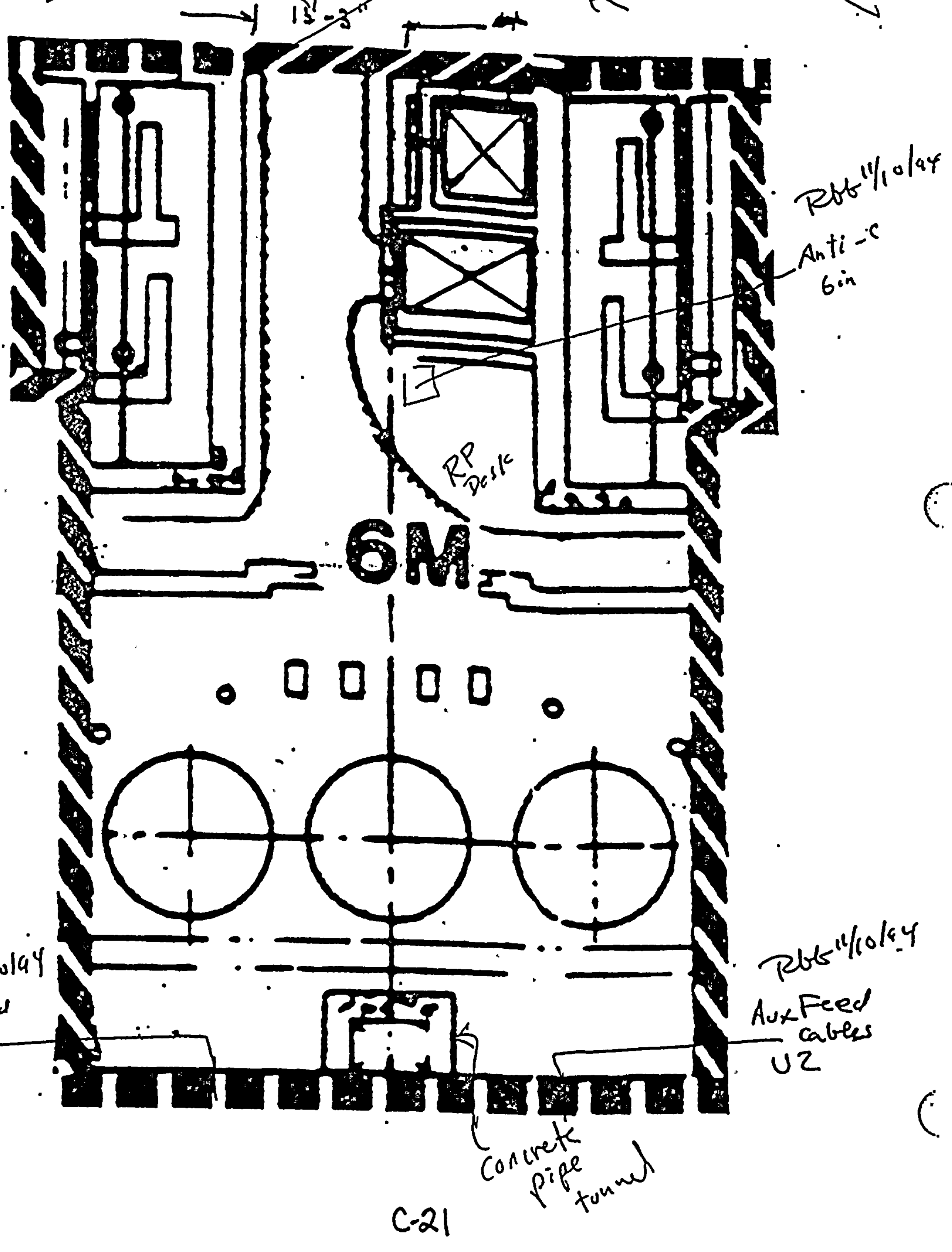
JM 3-21-91 BDM 3/21/91 JCK 3/21/91

Fire Analysis Walkdown Data

Reactor D.C. Cook Unit 1 + 2 Building Aux. Bldg (Middle Section of W. End)

Fire Zone 6M Fire Area E Elevation 587'-0"

Inspector [illegible] Verifier Date 3/21/91



APPENDIX D
HUMAN RELIABILITY CALCULATIONS

This appendix contains the human reliability calculations of the operators failing to cooldown and depressurize following a loss of component cooling water, due to a control room fire (Appendix D.1), and the operators failing to crosstie Unit 2 auxiliary feedwater and chemical and volume control systems following a loss of all Unit 1 power and control, due to a cable vault fire (Appendix D.2).

Appendix D.1
Loss of Component Cooling Water due to Control Room Fire
Failure to Cooldown and Depressurize

This analysis calculates the human error probability that the operators will not successfully cooldown and depressurize the reactor coolant system, following a loss of component cooling water and, subsequently, loss of the reactor coolant pump seals.

Background and Assumptions

This analysis addresses the critical human actions necessary to prevent core damage, following a total loss of the component cooling water system for Unit 1. This loss is caused by a fire in the control room service water panel. It is assumed that the operators will fail to identify the loss of CCW early enough to trip the reactor coolant pumps. As a result of this assumption, a LOCA equivalent to 480 gpm through each of the reactor coolant pump seals is postulated. Eventual recovery of CCW is not evaluated for the Fire PRA, as the scope only extends to the initiation of RHR.

This analysis is based on the ASEP Nominal HRA for Post-Accident Tasks (Chapter 8 of Reference 49), and Reference 50 was used to model recovery actions and dependence. Insights and class handouts from the Process Safety Institute's Human Reliability Analysis class were also used for this analysis (Reference 51).

The basic assumptions used in this analysis are listed below:

1) There is very little time for the operators to trip the reactor coolant pumps before seal failure occurs, therefore, it was conservatively assumed that the operators would not trip them in time to prevent seal failure (Reference 34).

2) The reactor coolant pumps' seal failures result in the maximum postulated leak rate (480 gpm/pump).

3) It is assumed that the fire is suppressed within 15 minutes, and the operators are able to remain in the control room (Reference 34).

4) The loss of CCW is conservatively considered a second event occurring closely in time with the control room fire (Table 8-2, Reference 49).

5) High head emergency core cooling is not available, as charging and safety injection pumps require CCW for cooling.

6) Low dependence was assumed between operator errors and the shift technical advisor (STA) correctly monitoring the status trees and identifying when a red path has been reached. The function of the STAs is to monitor the critical plant parameters using the status trees, and not to concentrate on the specific actions performed by the operators (Table 20-4 and Table 21-1 (E$_{1}$) of Reference 50).

7) All other systems are assumed to work properly (i.e., auxiliary feedwater works as designed).

8) An extremely high stress level is assumed for recovery actions when a red path has been reached, as the red paths indicate very serious conditions that must be addressed immediately.

9) Based on THERP (Reference 50, Items 9d and 10b of Table 8-1), the critical actions were considered dynamic because the diagnosis HEP was not adjusted downwards, as the EOPs do not specifically address this scenario of a control room fire causing a loss of component cooling water. This rule (from 10b of Table 8-1, Reference 49) is very conservative, however, because once the operators are past the diagnosis stage and into the appropriate procedure (ES-1.2, Reference 52), the EOPs are very good.

10) As there is 1.5 hours available for diagnosis, a moderately high level of stress is assumed for the critical actions following this time. The operators had plenty of time to extinguish the fire, to distinguish that a loss of CCW was the problem, and to start stepping through the procedures. All systems that are not dependent on CCW are assumed to function properly.

The Analysis

This analysis was performed by reviewing the event, success criteria and corresponding procedures, talking with training and operations personnel (see References 53a-f), performing a task analysis, performing a timing analysis, and then developing and quantifying an HRA event tree. The task analysis identified the critical actions and the recovery actions, as described below. The timing analysis identified the amount of time available to diagnose and perform the actions, such that core melt will be prevented. The HRA event tree is included as Figure D-1. Tables D-1 and D-2 contain timing information. Table D-3 includes the following information for each failure limb: person performing action, estimated HEP and source, corresponding procedure step (from task analysis) and an explanation of the action. Table D-4 is the quantification of the event tree, which resulted in a failure probability of .025.

Task Analysis

The three critical actions to be performed by the operators are steps 5, 7 and 32, listed below, from "POST LOCA COOLDOWN AND DEPRESSURIZATION" (Reference 52a). The critical actions include ensuring that the RHR pumps are not running (when RCS pressure is greater than 300 psig), initiating RCS cooldown and starting RHR pumps when the appropriate RCS conditions are met. Step 34 is also listed below, as it directs the operators to return to step 5 if the RCS temperature is ≥ 200°F.

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED
__ 5.	Check if RHR Pumps Should Be Stopped:	
	a. Check RHR pumps - ANY RUNNING	a. Go to Step 6.
	b. Check ECCS - ALIGNED FOR INJECTION MODE	b. Go to Step 6.
	c. Check RCS pressure:	c. Go to Step 6.
	• Pressure - GREATER THAN 300 PSIG (590 PSIG FOR ADVERSE CONTAINMENT)	
	• Pressure - STABLE OR INCREASING	

d. Stop RHR pumps and place in NEUTRAL

__ 7. Initiate RCS Cooldown To Cold Shutdown

a. Maintain cooldown rate in RCS cold legs - LESS THAN 100°F/HR

c. Transfer condenser steam dump to steam pressure mode

d. Using steam pressure controller, dump steam to condenser from intact SG(s) — d. Dump steam using intact SG(s) steam relief valve.

__ 32. Check if RHR System Can Be Placed In Service:

a. Check the following: — a. Go to Step 33.

- RCS temperature - LESS THAN 350°F
- RCS pressure - LESS THAN 363 PSIG (SEE SUPPLEMENT FOR ADVERSE CONTAINMENT)

b. Consult Plant Evaluation Team to determine if RHR System should be place in Service

__ 34. Check RCS Temperature - LESS THAN 200°F — Return to step 5.

If the operators fail at the above actions, a critical red or orange path will be reached on the STA status trees. The STA would then inform the operators that they are on a critical path, and they would switch to procedure FR-C.1 (Reference 52b) or FR-C.2 (Reference 52c), depending on the reactor vessel water level. These procedures will guide them to cooldown the reactor coolant system by dumping steam to the condenser, either at a maximum rate (Step 13 of Reference 52b) or at a limit of 100°F/hr (Step 11 of Reference 52c). They will continue the cooldown until at least two RCS hot leg temperatures are < 350°F and the reactor vessel level narrow range indication is > 60% (Step 18 of Reference 52b or Step 16 of Reference 52c). Then, if RCS pressure is not < 300 psig, or if RHR flow is not sufficient (Step 14 of Reference 52d), they will return to Step 1 of the "POST LOCA COOLDOWN AND DEPRESSURIZATION" procedure (Reference 52a). As these recovery actions are equivalent to those listed above, they were not included in the above listing.

Timing Analysis

Due to the modelling of possible recovery once a STA red path is reached, the time relationships from Figure 6-3 of Reference 49 (i.e., To, Tm, Td and Ta) have been modified, as defined in Table D-1. A MAAP 3.0b (Reference 54) run was performed to determine some of these critical times. The output from this MAAP run is included as Table D-2.

Table D-1
Timing Analysis Table

Term	Time (from To)	Definition Used
To	0	Fire in control room, annunciation of CCW and reactor trip.
Tm'	1.65 hours (99 minutes)	Time when enter red path on STA status tree, if steam dump had not been initiated. See Table D-2.
Ta	7 minutes	Time to initiate steam dump. See #5a and #5b of Table 8-1 of Reference 49: 5 minute delay assumed, and 2 one minute actions (stop RHR and initiate steam dump, performed on primary operating panels in control room)
Td See	90 minutes (conservative)	Time available for diagnosis of loss of CCW. Td = Tm'-Ta. Figure 6-3 of Reference 49.
Tm	1.85 hours (111 minutes)	Must have initiated steam dump by now to save core. See Table D-2.
Tr	0.2 hours (12 minutes)	Time available to perform recovery actions (i.e., to initiate steam dump). There is ample time, as only one action is required in this time. Tr = Tm - Tm'

Table D-2
Results of MAAP Run

5
ccw9 - depressurize sg porvs only - at 1000 F tgup

DC. COOK loss of ccw:

TIME HR	ZWV FT	TGUP *** K	PPS PSI	TCRHOT F
0.000E+00	2.620E+01	5.593E+02	2.112E+03	1.224E+03
1.667E-02	2.620E+01	5.523E+02	1.790E+03	5.659E+02
1.045E-01	2.620E+01	5.643E+02	1.100E+03	6.130E+02
2.057E-01	2.620E+01	5.680E+02	1.158E+03	6.123E+02
3.063E-01	2.620E+01	5.677E+02	1.153E+03	6.071E+02
4.053E-01	2.620E+01	5.673E+02	1.147E+03	6.033E+02
5.097E-01	2.620E+01	5.671E+02	1.143E+03	6.002E+02
6.052E-01	2.620E+01	5.669E+02	1.140E+03	5.978E+02
7.110E-01	2.620E+01	5.667E+02	1.137E+03	5.959E+02
8.000E-01	2.620E+01	5.665E+02	1.134E+03	5.941E+02
9.130E-01	2.620E+01	5.662E+02	1.129E+03	5.925E+02
1.101E+00	2.620E+01	5.660E+02	1.126E+03	5.903E+02
1.205E+00	2.620E+01	5.644E+02	1.090E+03	5.855E+02
1.303E+00	2.620E+01	5.624E+02	1.073E+03	5.844E+02
1.407E+00	2.360E+01	5.635E+02	1.084E+03	5.850E+02
1.507E+00	2.016E+01	5.687E+02	1.076E+03	6.935E+02
1.605E+00	1.672E+01	6.082E+02	1.063E+03	1.198E+03
1.708E+00	1.408E+01	6.929E+02	1.069E+03	1.466E+03
1.725E+00	1.372E+01	7.063E+02	1.063E+03	1.477E+03
1.743E+00	1.337E+01	7.202E+02	1.050E+03	1.493E+03
1.759E+00	1.303E+01	7.315E+02	1.041E+03	1.540E+03
1.776E+00	1.272E+01	7.444E+02	1.030E+03	1.607E+03
1.793E+00	1.243E+01	7.571E+02	1.019E+03	1.676E+03
1.809E+00	1.215E+01	7.694E+02	1.006E+03	1.751E+03
1.827E+00	1.187E+01	7.789E+02	9.895E+02	1.825E+03
1.844E+00	1.164E+01	7.979E+02	9.735E+02	1.892E+03
1.860E+00	1.142E+01	8.120E+02	9.562E+02	1.972E+03
1.869E+00	1.135E+01	8.100E+02	9.312E+02	2.015E+03
1.872E+00	1.139E+01	8.561E+02	9.061E+02	2.018E+03
1.877E+00	1.140E+01	8.635E+02	8.561E+02	2.008E+03
1.880E+00	1.134E+01	8.934E+02	8.376E+02	2.010E+03
1.884E+00	1.120E+01	8.627E+02	7.799E+02	2.021E+03
1.901E+00	1.106E+01	8.755E+02	6.476E+02	2.079E+03
1.909E+00	1.106E+01	8.579E+02	5.966E+02	2.113E+03
1.920E+00	1.219E+01	8.986E+02	5.427E+02	2.177E+03
1.932E+00	1.355E+01	9.284E+02	5.020E+02	2.142E+03
1.945E+00	1.526E+01	9.843E+02	4.758E+02	1.982E+03
1.956E+00	1.649E+01	9.623E+02	4.537E+02	1.809E+03
1.968E+00	1.787E+01	9.565E+02	4.397E+02	1.678E+03
1.978E+00	1.911E+01	8.644E+02	4.219E+02	1.587E+03
1.987E+00	2.029E+01	7.709E+02	4.104E+02	1.407E+03
2.000E+00	2.218E+01	6.100E+02	4.019E+02	7.749E+02
2.101E+00	2.620E+01	5.624E+02	2.567E+02	4.279E+02
2.201E+00	2.620E+01	5.168E+02	1.979E+02	4.035E+02
2.302E+00	2.620E+01	5.043E+02	1.811E+02	3.942E+02
2.400E+00	2.620E+01	5.167E+02	1.842E+02	3.935E+02
2.514E+00	2.620E+01	5.243E+02	1.894E+02	3.926E+02
2.860E+00	2.402E+01	5.259E+02	1.860E+02	3.846E+02
3.210E+00	2.392E+01	5.259E+02	1.858E+02	3.806E+02
3.562E+00	2.378E+01	5.256E+02	1.858E+02	3.727E+02
3.912E+00	2.355E+01	5.250E+02	1.859E+02	3.635E+02
3.995E+00	2.355E+01	5.248E+02	1.859E+02	3.625E+02
4.000E+00	2.356E+01	5.249E+02	1.859E+02	3.624E+02

Table D-3
Explanation of Terms and Values in Figure 1

Failure Limb & (Person*)	Estimated HEP* and Source {Reference}	Procedure Step {Reference}	Explanation
A_1 (crew)	.003 T8-2 (between #12 & #13) and UB T8-3 #1 {49}	n/a	Fail to diagnose the second abnormal event, total loss of Unit 1 CCW. The fire was considered the first abnormal event. The upper bound was also used due to the added confusion and stress of the fire.
B_1 (LO)	.001 ϵ {51}	5 {52a}	Fails to stop RHR pumps. The operators are very well trained on the conditions when RHR (i.e., low pressure injection) can be used, and every time the operators reach step 5 of ES-1.2 (Reference 52a), they re-examine if RHR pumps should be stopped (see the *Task Analysis* Section).
C_1 (LO)	.05 T8-5 #4 {49}	7 {52a}	Fail to dump steam using steam pressure controller. (See Assumptions 9 and 10.)
C_2 (US)	.5 T8-5 #7 {49}	n/a	Unit Supervisor fails to correct the LO's error.
C_3 (LO)	.05 T8-5 #4 {49}	7 {52a}	Same as C1.
C_4 (LO)	.05 T8-5 #4 {49}	32 {52a}	Fails to place RHR in service when conditions are right.
C_5 (STA)	.05 ϵ with LD T20-21 #2a {50}	Red Path {55}	STA fails to notice red path conditions. (See Assumption 6)
C_6 (LO)	.25 T8-5 #5 {49}	13 {52b} or 11 {52c}	Fail to dump steam using steam pressure controller. (See Assumptions 8 and 9.)
C_7 (LO)	.25 T8-5 #5 {49}	32 {52a}	Fails to place RHR in service when conditions are right. (See Assumptions 8 and 9.)
D_1 (LO)	.05 T8-5 #4 {49}	32 {52a}	Same as C4.
D_2 (US)	.5 T8-5 #7 {49}	n/a	Unit Supervisor fails to correct the LO's error.
D_3 (LO)	.05 T8-5 #4 {49}	32 {52a}	Same as C4.
D_4 (STA)	.05 ϵ with LD T20-21 #2a {50}	Red Path {55}	STA fails to notice red path conditions. (See Assumption 6)

Failure Limb & (Person*)	Estimated HEP* and Source {Reference}	Procedure Step {Reference}	Explanation
D5 (LO)	.25 T8-5 #5 {49}	32 {52a}	Same as C7.

* Key

crew - entire control room crew
LO - licensed operator
STA - shift technical advisor
US - unit supervisor
UB - upper bound of probability
LD - low dependence

Table D-4
Event Tree Quantification

$F_1 = A_1 = .003$

$F_2 = a_1B_1 = .001$

$F_3 = a_1b_1C_1C_2C_5 = .001$

$F_4 = a_1b_1C_1c_2C_3 = .001$

$F_5 = a_1b_1C_1c_2c_3C_4 = .001$

$F_6 = a_1b_1C_1C_2c_5C_6 = .006$

$F_7 = a_1b_1C_1C_2c_5c_6C_7 = .004$

$F_8 = a_1b_1c_1D_1D_2D_4 = .001$

$F_9 = a_1b_1c_1D_1d_2D_3 = .001$

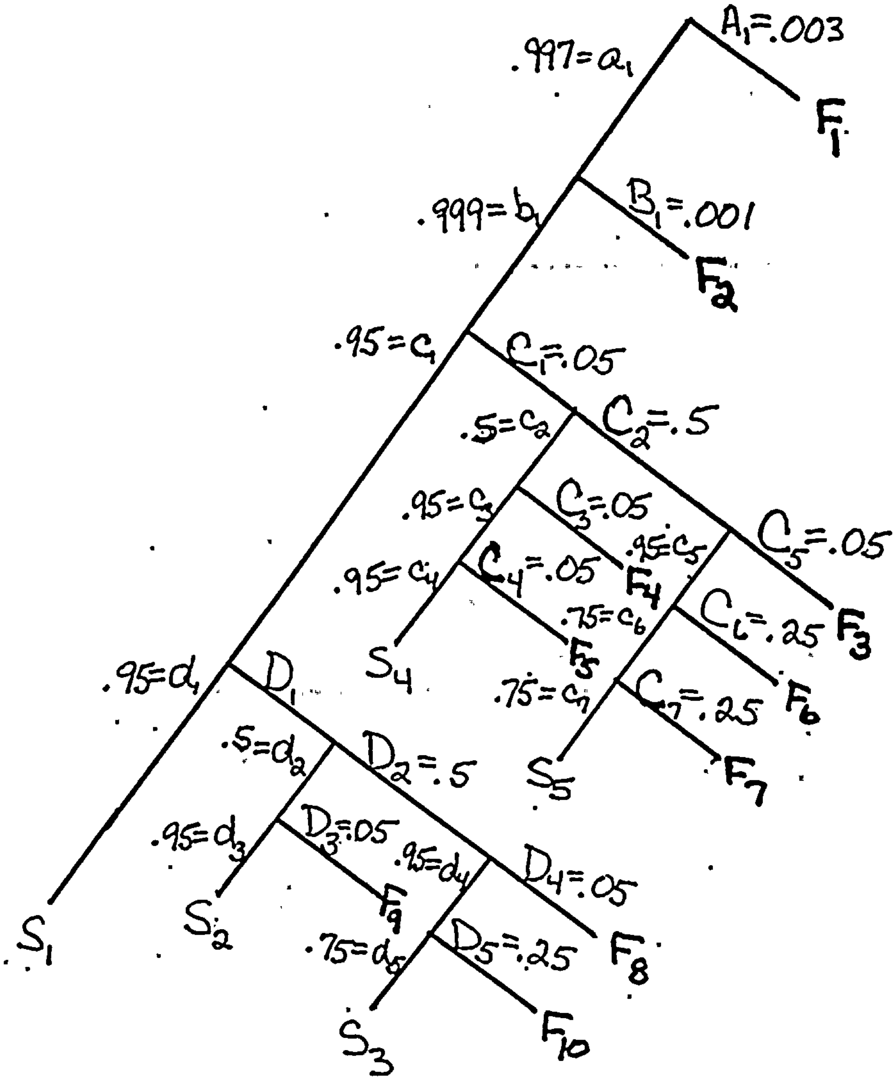
$F_{10} = a_1b_1c_1D_1D_2d_4D_5 = .006$

$F_T = F_1 + F_2 + F_3 + F_4 + F_5 + F_6 + F_7 + F_8 + F_9 + F_{10}$

$F_T = .003 + 6(.001) + 2(.006) + .004$

$F_T = .025$

Figure D-1
Loss of CCW Due to Fire Event Tree



Note: STA recovery could have been added at the end of branches: A_1, C_3, C_4 and D_3, as failure of these actions would result in reaching the STA red path. This was not credited, however, for simplification of the tree.

Appendix D.2
Loss of all Unit 1 Power and Control due to Cable Vault Fire
Failure to Crosstie Unit 2 AFW & CVCS

This analysis calculates the human error probability that the operators will not successfully crosstie the Unit 1 auxiliary feedwater (AFW) system and chemical and volume control system (CVCS) to Unit 2 following a loss of all Unit 1 power and control.

Background and Assumptions

This analysis addresses the critical human actions necessary to prevent core damage, following a total loss of Unit 1 safety systems. This loss is caused by a fire in one of the cable vaults, where the automatic suppression systems fail. This fire causes evacuation of the control room, and therefore, use of the emergency remote shutdown (ERS) procedure series, 1-OHP 4025. Upon evacuation of the control room, the ERS crew would gather at the hot shutdown panel in the Unit 2 control room, even when the hot shutdown panel was not operational. This location is the command post for the ERS actions. The shift supervisor or assistant shift supervisor would follow through the main ERS shutdown procedure (1-OHP 4025.001.001, Reference 28a), and instruct operators to go out into the plant and complete certain tasks, when dictated by the main ERS procedure. The actions necessary to complete these tasks are often contained in other sections of the ERS series. The command post will instruct operators to complete a step or task, and report back via radio. There will not be anyone checking these remote operator actions.

The actions found to be critical to prevent core melt are listed in the *Task Analysis* section. Although the STA is still expected to maintain an overview of events in this ERS scenario, this is conservatively not credited.

This analysis is based on the ASEP Nominal HRA for Post-Accident Tasks (Chapter 8 of Reference 49), and Reference 50 was used to model recovery actions and dependence. Insights and class handouts from the Process Safety Institute's Human Reliability Analysis class were also used for this analysis (Reference 51).

The basic assumptions used in this analysis are listed below:

1) The cable vault fire forces evacuation of the control room, as it is assumed to result in a loss of indication and control in the control room, as well as a significant amount of smoke.

2) Both motor driven trains of Unit 2 AFW are assumed to be available.

The Analysis

This analysis was performed by reviewing the event, success criteria and corresponding procedures, talking with training and operations personnel (see References 53a-f), performing a task analysis, performing a timing analysis, and then developing and quantifying an HRA event tree. The task analysis identified the critical actions and the recovery actions, as described below. The HRA event tree is included as Figure D-2. Table D-5 includes the following information for each failure limb: person performing action, estimated HEP and source, corresponding procedure step (from task analysis) and an explanation of the action. Table D-6 is the quantification of the event tree, which resulted in a failure 0-probability of .11.

Task Analysis

As soon as the operators are forced from the control room, they will enter the main Emergency Remote Shutdown (ERS) procedure (01-OHP 4025.001.001, Reference 28a). The shift supervisor will go through this procedure, and send the operators out into the plant to perform the required tasks. The operator actions considered critical to prevent core melt, as well as recovery actions, are listed below. Many of these tasks require the use of other sections of the 4025 ERS series, as dictated by the main ERS procedure (References 28a - 28f).

Critical actions include aligning the backup power to the six local shutdown indication (LSI) panels, and establishing the crosstie to Unit 2 AFW and CVCS. The critical actions and recovery actions are listed below, and the corresponding sections of the 4025 ERS series are included. The critical actions are steps: 1-6(LS-1-1), 3(LS-6-1), 4 (LS-2-2), 29e(001.001) and 1c(LS-6-2). The recovery actions are steps: 4(LS-2-1), and 29i and 36a(001.001).

STEP	ACTION/EXPECTED RESPONSE	RESPONSE NOT OBTAINED

From LS-1-1 (Reference 28a):

__ 1. Align 1-LSI-1 For Operation:

a. Place the following 1-LSI-1 LOCAL/REMOTE switches in LOCAL:

- 1-BLI-110, #11 SG Wide Range Level
- 1-BLI-140, #14 SG Wide Range Level

__ 2. Align 1-LSI-5 For Operation:

b. Align the following power supply switches:

1) U-1 (Normal Power) - OFF

2) U-2 (Backup Power) - UNIT 2

__ 3. Align 1-LSI-2 For Operation:

a. Place the following 1-LSI-2 LOCAL/REMOTE switches in LOCAL:

- 1-BLI-120, #12 SG Wide Range Level
- 1-BLI-130, #13 SG Wide Range Level

__ 4. Align 1-LSI-6 For Operation:

b. Align the following power supply switches:

1) U-1 (Normal Power) - OFF

2) U-2 (Backup Power) - UNIT 2

__ 5. Align 1-LSI-3 For Operation:

a. Place the following 1-LSI-3 LOCAL/REMOTE switches in LOCAL:

- 1-QFI-200, Charging Pumps Discharge Flow
- 1-QFI-301, Letdown Hx Outlet Flow
- 1-NLI-151, PRZ Cold Cal Level
- 1-NPS-122, RCS Wide Range Pressure

__ 6. Align 1-LSI-4 For Operation:

b. Align the following power supply switches:

1) U-1 (Normal Power) - OFF

2) U-2 (Backup Power) - UNIT 2

From LS-6-1 (Reference 28e):

__ 3. SLOWLY OPEN 2-CS-536, CVCS Charging Pumps Discharge Crosstie Header Unit 2 Shutoff Valve

From LS-2-2 (Reference 28d):

__ 4. Open 2-FW-129, 2E Motor Driven Auxiliary Feedwater Pump Discharge to Unit 1 Crosstie Shutoff Valve

From 001.001 (Reference 28a):

__ 29. Align U2 MDAFPs For Cross-Tie Operation:

e. Start 2E MDAFP

From LS-2-1 (Reference 28c):

__ 4. Open 1-FW-129, 1E Motor Driven Auxiliary Feedwater Pump Discharge to Unit 2 Crosstie Shutoff Valve

From 001.001 (Reference 28a):

__ 29. Align U2 MDAFPs For Cross-Tie Operation:

i. Start 2W MDAFP

__ 36. Initiate CVCS Cross-tie Operations:

a. Verify complete 01-OHP 4025.LS-6, RCS MAKE-UP, SEAL INJECTION, AND BORATION WITH CVCS CROSS-TIE, LS-6-1, SEAL INJECTION FROM CVCS CROSS-TIE

From LS-6-2 (Reference 28f):

__ 1. Initiate CVCS Crosstie Operation:

c. SLOWLY OPEN 1-CS-535, CVCS Charging Pumps Discharge Crosstie Header to Unit 1 RCP Seal Injection Emergency Flow Control Valve, to obtain 25 gpm flow indication on 12-QFI-201

Timing Analysis

Diagnosis error is considered negligible for this scenario, as the smoke and loss of control in the control room will cause definite evacuation from the control room and entry into the Emergency Remote Shutdown Procedure (Reference 28a). An explicit timing analysis, therefore, is not warranted. A brief timing study is included.

From reactor trip, it takes the operators about 30 minutes to isolate the RCS and steam generators and crosstie AFW and CVCS (Table 12.3-1 of Reference 29). Following a station blackout with no AFW, core uncovery is expected to begin at about two hours (Reference 56). There is plenty of time, therefore, for the operators to perform the critical actions, and for recovery of errors.

Table D-4
Explanation of Terms and Values in Figure D-2

Failure Limb & (Person*)	Estimated HEP* and Source (Reference)	Procedure Step (Procedure Reference #)	Explanation
A1 (crew)	.001 ϵ {51}	n/a	Operators fail to diagnose need to evacuate control room and use Emergency Remote Shutdown procedure, even though smoke is filling control room and all control room indication and control is gone.
B1 (OP)	6 * .001 6ϵ {51}	1-6 {28b}	Operator fails to align backup power to each of the 6 LSI panels. These are needed for indication for emergency remote shutdown. If a needed panel is dead, they will try to connect power to it.
C1 (OP)	.05 T8-5 #4 {49}	3 {28e}	Operator fails to open the first Unit 2 CVCS cross-tie valve, 2-CS-536.
C2 (OP)	.53 T8-5 #4 {49} with HD T20-21 #4b {50}	36a {28a}	Operator fails to verify complete LS-6-1 (opened 2-CS-536).
D1 (OP)	.05 T8-5 #4 {49}	4 {28d}	Operator fails to open the 1W/2E AFW cross-tie valve (2-FW-129)
D2 (OP)	.53 T8-5 #4 {49} with HD T20-21 #4b {50}	4 {28c}	Operator fails to open the 1E/2W AFW cross-tie valve (1-FW-129)
D3 (OP)	.05 T8-5 #4 {49}	29i {28a}	Operator fails to start the 2W MDAFP.
E1 (OP)	.05 T8-5 #4 {49}	29e {28a}	Operator fails to start the 2E MDAFP.
G1 (OP)	.05 T8-5 #4 {49}	1c {28f}	Operator fails to open the second Unit 2 CVCS cross-tie valve, 1-CS-535.
G2 (OP)	.53 T8-5 #4 {49} with HD T20-21 #4b {50}	{28a}	Operator fails to later throttle open 1-CS-535. (There are many steps in the procedure that would lead the operator to open valve 1-CS-535, if they had failed to in G1.)

* Key

crew - entire control room crew
OP - operator, licensed or non-licensed
HD - high dependence

Table D-5
Calculation of Total Failure Probability

$F_1 = A_1 = .001$

$F_2 = a_1B_1 = .006$

$F_3 = a_1b_1C_1C_2 = .026$

$F_4 = a_1b_1c_1D_1D_2 + a_1b_1C_1c_2D_1D_2 + a_1b_1c_1d_1E_1D_2 = a_1b_1D_2 [c_1D_1 + C_1c_2D_1 + c_1d_1E_1] = .049$

$F_5 = a_1b_1d_2D_3 [c_1D_1 + C_1c_2D_1 + c_1d_1E_1] = .002$

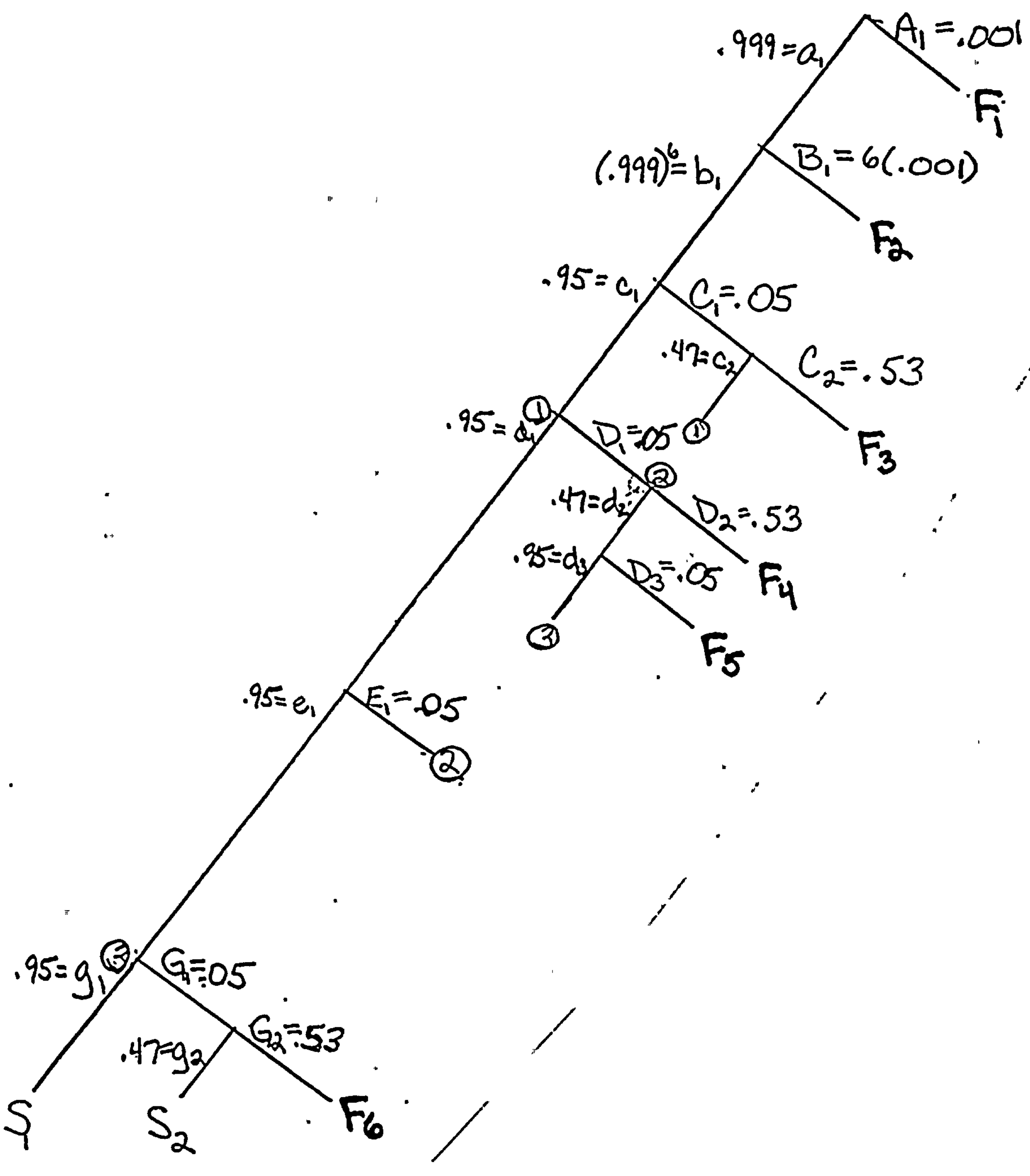
$F_6 = a_1b_1c_1d_1e_1G_1G_2 + a_1b_1d_2d_3G_1G_2 [c_1D_1 + C_1c_2D_1 + c_1d_1E_1] = .024$

$F_T = F_1 + F_2 + F_3 + F_4 + F_5 + F_6$

$F_T = .001 + .006 + .026 + .049 + .002 + .024$

$F_T = 0.11$

Figure D-2
Loss of Unit 1 Power and Control Due to Fire Event Tree



APPENDIX E
CALCULATION OF ESTIMATED CORE DAMAGE FREQUENCIES FOR ZONES WITH INITIATING EVENTS OTHER THAN TRA

In this appendix, the core damage frequency is estimated for each zone with an initiating event concern (other than TRA). When more than one initiating event was credible for a zone, the most limiting event was used for the calculation. The core damage frequency values were estimated using the following equation:

$$CDF_{new} = [IEF_{new}/IEF_{old}]*[CDF_{old}]*[FIREF_{zone}]$$

where:

CDF_{new} = estimated core damage frequency, to be determined

CDF_{old} = initiating event's original contribution to core damage frequency, Revision 0 of IPE (from Table 3.4-1, "Accident Event Summary", Reference 57)

IEF_{new} = initiating event frequency, based on equipment in zone (from Table 10)

IEF_{old} = old initiating event frequency, Revision 0 of IPE (from Table 3.4-1, "Accident Event Summary", Reference 57)

$FIREF_{zone}$ = Fire initiation frequency for each zone (from Tables 4 through 9)

Core damage frequencies are estimated in this appendix for 21 zones: 6N, 13, 15, 16, 29A, 29B, 29E, 29G, 40A, 40B, 42A, 42C, 42D, 43, 44N, 44S, 51, 52, 79, 112 and 114.

Zone 6N

LOSE:

- W train of CCW (lose MCC 1-AZV-A)
- D/G 1AB (lose both of its fuel oil transfer pumps)
- MCC's for both ESW strainers

NOTES:

Also lose all AFW (lose pumps and other equipment), E CCP lube oil pump & other various components. MCC 1-AZV-A is found in the following fault trees: CCWW, CCWWL, HPI, CSR, HP5, CF and HPR.

SCREEN: not screened out

Can show that this will not be screened out by just considering the loss of 1 train of CCW:

IEF(CCW) = 0.01 (Table 10, CCW(A))

FIREF(Zone 6N) = 1.0E-03

$CDF_{new} = [IEF_{new}/IEF_{old}]*[CDF_{old}]*[FIREF_{zone}]$

$CDF_{new} = [0.01/8.71E\text{-}04]*[1.38E\text{-}05]*[1.0E\text{-}03]$

$CDF_{new} = 1.58E\text{-}07$

Actual value would be even greater than this.

Zone 13

LOSE:

- D/G 1CD (lose both fuel oil transfer pumps)

NOTES:

Lose both fuel oil transfer pumps for D/G 2CD, but this is not relevant.

Cable 1-9655R for D/G 1AB that runs through this zone is for testing only. Its loss only matters if the diesel is in testing at the time of the fire, as it may be incapable of transferring its supply to its required loads. The probability of this diesel being in testing (STP.027) will be added to its failure probability below. Nothing else is in zone.

SCREEN: screened out

IEF(SBO) = [Chance of one D/G failing to start & run + Probability it is in Testing (STP.027, from line 485 of SIMON.DAT, Rev. 0 of PRA)] * [Probability of a LOSP (0.04/365)]

IEF(SBO) = [1.9E-02 + 6.0E-03] * [1.1E-04] = 2.75E-06

FIREF(Zone 13) = 1.0E-03

CDF_{new} = [2.75E-06/1.40E-05] * [1.13E-06] * [1.0E-03]

CDF_{new} = 2.22E-10

Zone 15

LOSE:

- All ESW for Unit 1 (PP-7E, PP-7W & WMO-707)
- E CCW train (PP-10E)
- D/G 1CD (D/G, both fuel oil transfer pumps, WMO-725)

NOTES:

Also lose both RHR pumps, E CCP and EMDAFP.

SCREEN: not screened out

When all ESW for Unit 1 is lost, all CCW is also lost. A total loss of CCW is used as the initiating event example.

IEF(CCW) = 1.0

FIREF(Zones 15) = 2.2E-02

CDF_{new} = [0.01/8.71E-04]*[1.38E-05]*[2.2E-02]

CDF_{new} = 3.49E-04

Zone 16

LOSE:

- Entire W ESW Header (PP-7W & WMO-705)
- W CCW train (PP-10W)
- D/G 1AB (D/G, fuel oil transfer pumps & WMO-721)

NOTES:

Also lose W RHR Pump, W MDAFP and W CCP.

SCREEN: not screened out

Can show that this will not be screened out by just considering the loss of 1 train of CCW (this gives a higher result than the loss of one header of ESW):

IEF(CCW) = 0.01 (Table 10, CCW(A))

FIREF(Zone 16) = 2.2E-02

CDF_{new} = [0.01/8.71E-04]*[1.38E-05]*[2.2E-02]

CDF_{new} = 3.49E-06

Actual value would be even greater than this.

Zone 29A

LOSE:

- 1E ESW train (PP-7E, WMO-701 & OME-34E)

NOTES:

Nothing else is in zone.

SCREEN: screened out

IEF(ESW) = 4.5E-05 (Table 10, ESW(A))

FIREF(Zone 29A) = 1.0E-03

CDF_{new} = [4.5E-05/3.7E-05] * [6.04E-07] * [1.0E-03]

CDF_{new} = 7.29E-10

Zone 29B

LOSE:

- Both U1 trains of ESW (lose PP-7W, WMO-701, WMO-702, OME-34E & OME-34W)

NOTES:

Nothing else is in zone.

SCREEN: not screened out

IEF(ESW) = 6.6E-03 (Table 10, ESW(E))

FIREF(Zone 29B) = 1.0E-03

CDF_{new} = [6.6E-03/3.73E-05] * [6.04E-07] * [1.0E-03]

CDF_{new} = 1.07E-07

Zone 29E

LOSE:

- both U1 trains of ESW (lose MCC PS-D, MCC PS-A, ESWSE, ESWSW, WMO-701, WMO-702)

NOTES:

Nothing else is in zone

SCREEN: not screened out

IEF(ESW) = 6.6E-03 (Table 10, ESW(E))

FIREF(Zone 29E) = 1.0E-03

CDF_{new} = [6.6E-03/3.73E-05] * [6.04E-07] * [1.0E-03]

CDF_{new} = 1.07E-07

Zone 29G

LOSE:

- all 4 trains of ESW (1-PP-7E, 2-PP-7E, 1-PP-7W, 2-PP-7W, 1-ESWSE, 2-ESWSE, 1-ESWSW, 2-ESWSW, 1-WMO-701, 2-WMO-703, 1-WMO-702, 2-WMO-704)
- both D/G's (1AB & 1CD)

NOTES:

Nothing else is in zone.

SCREEN: not screened out

When all ESW is lost, all CCW is also lost. A total loss of CCW is used as the initiating event example.

IEF(CCW) = 1.0

FIREF(Zone 29G) = 1.0E-03

CDF_{new} = [1.0/8.71E-04] * [1.38E-05] * [1.0E-03]

CDF_{new} = 1.58E-05

Zone 40A

LOSE:

- W CCW train (PP-10W)
- W ESW train (PP-7W)
- D/G 1AB
- Train A 250VDC (both battery chargers and transfer cabinet).

NOTES:

Also lose 600V busses 11A and 11B, WMDAFP, W RHR pump & W CCP.

SCREEN: not screened out'

Can show this will not be screened out by just considering the 1 train of CCW:

IEF(CCW) = 1.0E-02 (Table 10, CCW(A))

FIREF(Zone 40A) = 2.9E-03

CDF_{new} = [1.0E-02/8.71E-04] * [1.38E-05] * [2.9E-03]

CDF_{new} = 4.59E-07

Actual value would be even greater than this.

Zone 40B

LOSE:

- E CCW train (PP-10E)
- E ESW train (PP-7E)
- D/G 1CD

NOTES:

Also lose 600V busses 11C and 11D, EMDAFP, E RHR pump & E CCP.

SCREEN: not screened out

Can show this will not be screened out by just considering the loss of 1 train of CCW:

IEF(CCW) = 1.0E-02 (Table 10, CCW(A))

FIREF(Zone 40B) = 2.9E-03

CDF_{new} = [1.0E-02/8.71E-04] * [1.38E-05] * [2.9E-03]

CDF_{new} = 4.59E-07

Actual value would be even greater than this.

Zone 42A

LOSE:

- Entire W ESW Header (PP-7W & WMO-705)
- W CCW train (PP-10W)
- D/G 1AB (D/G, fuel oil transfer pumps & WMO-721)

NOTES:

Also lose W RHR Pump, 600V busses 11A & 11C, W MDAFP, W CCP, and several MCC's which affect various fault trees.

SCREEN: not screened out

Can show that this will not be screened out by just considering the loss of 1 train of CCW (this gives a higher result than the loss of one header of ESW):

IEF(CCW) = 0.01 (Table 10, CCW(A))

FIREF(Zone 42A) = 7.1E-03

CDF_{new} = [0.01/8.71E-04]*[1.38E-05]*[7.1E-03]

CDF_{new} = 1.12E-06

Actual value would be even greater than this.

Zone 42C

LOSE:

- W CCW train (PP-10W)
- W ESW train (PP-7W)
- D/G 1AB (D/G)
- Train B 250VDC (transfer cabinet)

NOTES:

Also lose W RHR Pump, W MDAFP, W CCP, 600V busses 11A & 11C and 120VAC distribution panels.

SCREEN: not screened out

Can show that this will not be screened out by just considering the loss of 1 train of CCW:

IEF(CCW) = 0.01 (Table 10, CCW(A))

FIREF(Zone 42C) = 1.0E-03

CDF_{new} = [0.01/8.71E-04]*[1.38E-05]*[1.0E-03]

CDF_{new} = 1.58E-07

Actual value would be even greater than this.

Zone 42D

LOSE:

- D/G 1AB (D/G)
- Train B 250VDC (battery, transfer cabinet, distribution cabinet)

NOTES:

Nothing else is in zone.

SCREEN: not screened out

IEF(250VDC) = 1.0 (Table 10, 250VDC)

FIREF(Zone 42D) = 3.2E-03

CDF_{new} = [1.0/1.16E-02] * [6.04E-07] * [3.2E-03]

CDF_{new} = 1.68E-07

Zone 43

LOSE:

- W CCW train (MCC AM-A, provides power to CMO-420)

NOTES:

The only thing in this zone is MCC 1-AM-A. MCC 1-AM-A is found in the following fault trees: CCWW, CCWWL, LPR, HPR, CCWL and AFS. Although a cable for MCC 1-AM-D runs through this zone, MCC 1-AM-D is not lost because this cable (1-8546G) is a spare abandoned cable.

SCREEN: not screened out

Can show this zone will not be screened out by just considering the one train of CCW:

IEF(CCW) = 1.0E-02 (Table 10, CCW(A))

FIREF(Zone 43) = 1.0E-03

CDF_{new} = [1.0E-02/8.71E-04] * [1.38E-05] * [1.0E-03]

CDF_{new} = 1.58E-07

Actual value would be even greater than this.

Zone 44N

LOSE:

- All U1 CCW (PP-10E, PP-10W, CMO-410, CMO-420, CMO-419, CMO-429, HE-15E, HE-15W, WMO-731, WMO-733, WMO-735, WMO-737, MCC 1-AM-A and MCC 1-AZV-A)
- E ESW train (PP-7E)
- One D/G fuel oil transfer pump (1-1AB1)

NOTES:

Also lose: all U1 AFW (lose all three pumps, 1-ABN and various valves), all U1 CVCS (both lube oil pumps and various valves) and various MS and RHR valves. MCC 1-AM-A is found in the following fault trees: CCWW, CCWWL, LPR, HPR, CCWL and AFS. MCC 1-AZV-A is found in the following fault trees: CCWW, CCWWL, HPI, HP5, HPR, CSR and CF. Many U2 valves and pumps are in zone, however, the only Unit 2 ESW components that are affected are the ESW supply and discharge valves to a Unit 2 CCW heat exchanger.

SCREEN: not screened out

Can show this zone will not be screened out by just considering the loss of CCW:

IEF(CCW) = 1.0

FIREF(Zone 44N) = 1.4E-03

CDF_{new} = [1.0/8.71E-04] * [1.38E-05] * [1.4E-03]

CDF_{new} = 2.22E-05

Actual value would be even greater than this.

Zone 44S

LOSE:

- All U1 CCW (PP-10E, PP-10W, CMO-420, WMO-737)

NOTES:

Also in zone: CMO-411 and CMO-413 (modelled in CF only) and IMO-255 (modelled in HPI, HP5 and HPR). Many U2 CCW cables and components are in this zone, as well as many other U2 cables (MS, CVCS, AFW, RHR, D/G's and electric power). The only Unit 2 ESW components affected, however, are the ESW supply and discharge valves to the U2 CCW heat exchangers.

SCREEN: not screened out

Can show this zone will not be screened out by just considering the loss of CCW:

IEF(CCW) = 1.0

FIREF(Zone 44S) = 2.4E-03

CDF_{new} = [1.0/8.71E-04] * [1.38E-05] * [2.4E-03]

CDF_{new} = 3.80E-05

Zone 51

LOSE:

- All U1 CCW (CMO-420 (W train discharge valve), WMO-731 and WMO-733 (ESW cooling to E CCW train))

NOTES:

Various CVCS valves are also in zone (ICM-250, IMO-910, QMO-200, QMO-201, QMO-451). Nothing else is in zone.

SCREEN: not screened out

Can show this zone will not be screened out by just considering the loss of CCW:

IEF(CCW) = 1.0

FIREF(Zone 51) = 1.1E-03

CDF_{new} = [1.0/8.71E-04] * [1.38E-05] * [1.1E-03]

CDF_{new} = 1.74E-05

Actual value would be even greater than this.

Zone 52

LOSE:

- All U1 CCW (CMO-420 (W train discharge valve), WMO-731 and WMO-733 (ESW cooling to E CCW train))

NOTES:

Also lose various MS valves (U1 & U2), AFW valves (U1 & U2), TDAFP and CVCS valves (U1 & U2). There are no Unit 2 ESW components in this zone. 250VCD distribution cabinets 1-ABN and 1-DCN are also in zone. 1-DCN only takes out the N-train, and 1-ABN affects AFW, as it is the control power to the TDAFP (found in: AF1, AFT & AFS). MCC 1-AM-A and MCC 1-AM-D are in this zone. MCC 1-AM-A is found in the following fault trees: CCWW, CCWWL, LPR, HPR, CCWL and AFS, and MCC 1-AM-D is found in: HPI, CSR, LPR, DCN, HP5, CF and HPR. MCC 2-AM-A and 2-AM-D are also in this zone.

SCREEN: not screened out

Can show this zone will not be screened out by just considering the loss of CCW:

IEF(CCW) = 1.0

FIREF(Zone 52) = 2.2E-03

CDF_{new} = [1.0/8.71E-04] * [1.38E-05] * [2.2E-03]

CDF_{new} = 3.49E-05

Actual value would be even greater than this.

Zone 79

LOSE:

- All U1 CCW (PP-10E, PP-10W)
- Both ESW Headers (PP-7E, PP-7W and cross-tie valves WMO-705 and WMO-707)
- Both D/G's (DGAB, DGCD, all 4 fuel oil transfer pumps, 2 of 4 ESW supply valves (WMO-721 and WMO-725))

NOTES:

Also lose: all CVCS (PP-50E & PP-50W), all RHR (PP-35E & PP-35W) and W MDAFP. Although no Unit 2 ESW cables or components are in this zone, Unit 2 ESW is unavailable since cables for crosstie valves 1-WMO-705 and 1-WMO-707 are in zone.

SCREEN: not screened out

Can show this zone will not be screened out by just considering the total loss of CCW:

IEF(CCW) = 1.0

FIREF(Zone 79) = 6.0E-03

CDF_{new} = [1.0/8.71E-04] * [1.38E-05] * [6.0E-03]

CDF_{new} = 9.48E-05

Actual value would be even greater than this.

Zone 112

LOSE:

- Both ESW headers (PP-7E, PP-7W and cross-tie valves WMO-705 and WMO-707)

NOTES:

There is nothing else in this zone. Although no Unit 2 ESW cables or components are in this zone, Unit 2 ESW is unavailable since cables for crosstie valves 1-WMO-705 and 1-WMO-707 are in zone.

SCREEN: not screened out

When all ESW is lost, all CCW is also lost. A total loss of CCW is used as the initiating event example.

IEF(CCW) = 1.0

FIREF(Zone 112) = 1.0E-03

CDF_{new} = [1.0/8.71E-04] * [1.38E-05] * [1.0E-03]

CDF_{new} = 1.58E-05

Zone 114

LOSE:

- 2 of 4 ESW supply valves to EDG's (WMO-721 & WMO-725) and a fuel oil transfer pump (1AB1)

NOTES:

The only other components in this zone are LSI components, which are not relevant.

SCREEN: screened out

IEF(SBO) = 5.2E-08 (Table 10, SBO(B))

FIREF(Zone 114) = 1.0E-03

CDF_{new} = [5.2E-08/1.40E-05] * [1.13E-06] * [1.0E-03]

CDF_{new} = 4.20E-12

APPENDIX F
COMPBRN RUN

**(Note: This appendix contains computer output.
It was not included to reduce the volume of the submittal.)**

APPENDIX G
CALCULATION OF INITIATING EVENT FREQUENCIES

**(Note: This appendix contains computer output.
It was not included to reduce the volume of the submittal.)**

APPENDIX G
CALCULATION OF INITIATING EVENT FREQUENCIES

This appendix documents the calculation of initiating event frequencies upon the loss of components or trains of component cooling water (CCW), essential service water (ESW) and diesel generators. The initiating event frequencies impacted by such a loss are loss of CCW, loss of ESW and station blackout (SBO).

The initiating event frequencies were found by making the necessary changes to the .SIM files, and then quantifying the fault trees. This method is consistent with that used in the IPE. The .SIM files and a summary of the output files are included in this appendix, as listed below:

Initiating Event	Components/Trains Lost	Initiating Event Frequency	Table #: .SIM File	Table #: .OUT File
Loss of CCW	1 CCW operating train	1.0E-02	G.1.1	G.1.2
	1 CCW standby train	2.3E-04	G.1.3	G.1.4
Loss of ESW	1 ESW operating train	4.5E-05	G.2.1	G.2.2
	1 ESW standby train	1.2E-05	G.2.3	G.2.4
	Both ESW operating trains	6.6E-03	G.2.5	G.2.6
	Both ESW standby trains	3.4E-04	G.2.7	G.2.8
	Both U1 ESW trains	6.6E-03	G.2.9	G.2.10
	Both U2 ESW trains	3.4E-04	G.2.11	G.2.12
	Both ESW trains/header aligned to U1 loads	6.9E-03	G.2.13	G.2.14
	Both ESW trains/header aligned to U2 loads	2.4E-05	G.2.15	G.2.16
Loss of 250 VDC Power	1 250 VDC train	1.0	—	—
SBO	1 diesel generator	2.2E-06	G.3.1A G.3.1B	G.3.2
	2 of 4 ESW supply valves to diesel generators	5.2E-08	G.3.3A G.3.3B	G.3.4

Note: These results are conservative. Some double counting was left in the quantification. For example, for the CCW standby train case, the standby pump was failed. Even so, a dominant cutset remained where that same pump is in test and maintenance.