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C.1 INITIATING EVENT IDENTIFICATION AND CLASSIFICATION

The primary purpose of this section is to present the initiating events that were identified and selected for quantification in the Diablo Canyon risk model. Another purpose is to document the completeness in coverage of accident sequences to the extent that such completeness is influenced by initiating event selection. This section contains the results of applying the methods described in Appendix A to obtain a set of initiating events uniquely appropriate to Diablo Canyon. Following a summary of these initiating events, which is presented in Section C.1.1, the process of identifying candidate initiating events using three separate methods is discussed in Section C.1.2. In Section C.1.3, three additional approaches are used to check the completeness of the candidate events, and, in Section C.1.4, the final selection and grouping of initiating events are documented.

C.1.1 INITIATING EVENT SUMMARY

The 45 initiating event categories that were selected for quantification in the Diablo Canyon risk model are listed in Table C.1-1. The term "category" is used in the sense that many specific events can be identified for each category by breaking the category down by cause, failure mode, degree of severity, etc. The distinctions made by specifying different categories are those necessary to account for the influence of the initiating events on the development or unfolding of the accident sequences in the event sequence model and to isolate key factors of importance in quantifying accident sequence frequencies and damage levels. For this reason, discrete seismic hazard intensity levels and distinct fire locations and magnitudes are counted as separate initiating event categories in the tabulation below.

The 45 initiating event categories in Table C.1-1 are grouped to help show the coverage of the major classes of initiating events as indicated below.

Major Class	Number of Initiating Event Categories
Loss of Coolant Inventory	7
General Transients	14
Common Cause Initiating Events	
• Support System Faults	6
• External Events and Spatial Interactions (including fires)	24
Total	51

Of the 18 categories set aside for external events and internal spatial interactions, the following breakdown is given:

External Events and Spatial Interactions Class	Number of Initiating Event Categories
Seismic Events	6
Fires	14
Flooding/Jets/Sprays	3
Other	1
TOTAL	24

Each seismic event was quantified for point estimate results at different, discrete levels of earthquake ground acceleration and covered locations throughout the plant and varying degrees of damage. The particular ground acceleration levels were selected on the basis of the seismicity and fragility curves, as discussed more fully in Appendix F. The different fire cases cover 24 different fire locations and different degrees of damage done by the fire in 7 of these locations. The three flooding cases include events in eight different locations. The remaining event is a large vessel impacting the intake structure. A detailed breakdown of these events is given in Appendix F.

In terms of the number of initiating events quantified in a risk model, the set presented in Table C.1-1 for Diablo Canyon represents the results of an exhaustive search by the team in order to be as complete as possible. Such a comprehensive treatment stems not from anything peculiar to Diablo Canyon, but rather reflects the intention of the risk analysis team to respond in a constructive way to peer review comments on previously published PRA reports. These comments point toward the need for enhanced completeness and improved treatment of the dependencies between initiating events and the required mitigating systems. The case for adequacy of completeness in coverage of initiating events in this project is made in the following sections.

C.1.2 INITIATING EVENT IDENTIFICATION

The different methods used to actually identify candidate initiating events for Diablo Canyon include the following:

- Master Logic Diagram
- Heat Balance Fault Tree
- Failure Modes and Effects Analysis

The master logic diagram method accounted for the identification of most initiating event categories that were finally selected for quantification. The heat balance fault tree method resulted in a finer structure for defining initiating event categories and enhanced completeness. Failure modes and effects analysis (FMEA) was used here to systematically identify support system failure modes that result in common cause initiating events. The FMEAs are not only used to generate additional initiating event categories but to also subdivide the original set to facilitate the treatment of dependence in event tree quantification.

C.1.2.1 Master Logic Diagram Method

The master logic diagram of Figure C.1-1 traces the thought process that follows from the question: "How can a significant release of radioactivity to the environment occur?" Level I in the diagram represents such a release. Level II then says that if such a release occurs, it must originate either in a damaged core or in a noncore source of radioactivity, such as the spent fuel storage pool or the gaseous, liquid, and solid waste facilities. Past experience and analysis (Reference C.1-1) have clearly shown that releases from the core are by far the major source of risk at a nuclear power plant; therefore, the remainder of the master logic diagram emphasizes the core branch. It is generally recognized that sources of radioactivity at the plant (other than the reactor core) and the possible mechanisms for their release provide a negligible risk of public health impact (Reference C.1-2). Noncore releases were not analyzed further in this project.

Level III then expresses the fact that a release from the core to the environment can occur only if the core is damaged and there is a coincident failure of the containment function.

The containment function is provided first by the fuel cladding, second by the reactor coolant system (RCS) pressure boundary, and finally, in most cases, by the containment structure itself. The exceptions are cases in which failure of the RCS pressure boundary results in a release pathway that bypasses the containment structure; e.g., steam generator tube rupture. Conditional failure of the containment function is evaluated on a scenario-by-scenario basis. It is dependent on the initiating event as well as the system response to the initiating event.

The initiators for core damage are evaluated as follows. When the reactor plant is operating in a stable manner, the thermal energy generated in the core is in equilibrium with that removed by the reactor coolant system. For core damage to occur, the fuel temperature must increase. This is addressed in level IV of the diagram. A fuel temperature increase can only occur if the heat generation rate exceeds the heat removal rate; i.e., by a decrease in core cooling or by excessive core power. This departure from thermal equilibrium must occur or heatup is not possible; thus, the diagram is complete at level IV.

The excess core power branch bypasses levels V and VI, proceeding directly to the level VII initiating event category, core power increase. The events of interest are those that are initiated by some

form of reactivity increase leading to increased reactor power coupled with a failure of reactor shutdown for sufficient time to threaten RCS integrity or to cause core damage.

The branch labeled "Decrease in Core Cooling" at level IV includes all core undercooling events. Level V identifies the two paths leading to a decrease in core cooling: (1) structural breaches of the primary coolant boundary or (2) inadequate core heat removal with an intact reactor coolant system (RCS). In level VI, the latter branch is further divided into those events that directly cause a reduction in core cooling, leading to an initial core temperature increase, and those events that indirectly lead to a decrease in core heat removal. They initiate sequences that can eventually lead to undercooling situations if required operator action or actuation of standby systems fail.

The events representing primary coolant boundary failure can be further divided into the six LOCA initiating event categories shown on level VII to reflect differences in the response of the plant and associated success criteria.

1. Excessive LOCA. Beyond the capability of the emergency core cooling system (ECCS) to maintain inventory control; i.e., for this study, assumed to be anything more severe than the design basis LOCA event.

2. Large LOCA. Within the capability of the ECCS. Large enough and located appropriately to rapidly depressurize the reactor coolant system so that successful operation of the accumulators and the low pressure residual heat removal (RHR) system, first in the injection mode and, then, in the recirculation mode, is needed to prevent core damage.

3. Medium LOCA*. An intermediate size LOCA (2 to 6 inches in diameter) that gradually depressurizes the reactor coolant system sufficiently that secondary heat removal is not required. Two of the four centrifugal charging and safety injection pumps are required, and one of two RHR pumps are required for injection. RHR is also required for recirculation cooling.

4. Small LOCA*. A LOCA sufficiently small (i.e., less than 2 inches in diameter) that RCS heat removal via the steam generators or RHR shutdown cooling is required in addition to one of the four contrifugal charging and safety injection pumps for makeup. Small LOCAs are assumed to be large enough to exceed the makeup capacity of normal charging. There are two subcategories of small LOCA:

   a. Isolable. The small LOCA results from a leaking PORV.

   b. Nonisolable. The small LOCA is the result of a break or leak other than from a PORV.

*The categorization of LOCA events is based on analysis results of generic Westinghouse plants.

5. LOCA Outside the Containment (interfacing systems LOCA). A LOCA that originates at an interface between the RCS and low pressure systems and may cause failure or degraded performance in the ECCS (e.g., recirculation from the containment sump may not be available) and a release path bypassing the containment.

6. Steam Generator Tube Rupture. A rupture or break in one or more of the steam generator tubes.

The differences in depressurization rates among the various sized LOCAs are necessary to distinguish the different capabilities of various subsystems of the emergency core cooling systems, as described in Section C.4.

Any primary coolant boundary failure that results in a leak can be assigned to one of the above categories based on leak path or size and response of the charging and letdown systems. The particular method of the above categorization has been chosen to capture differences in event sequence development for the respective categories.

The direct initiating events for insufficient heat removal must change the reactor coolant system pressure, temperature, or flow rate. One group of these initiators, the LOCAs, has already been cataloged under the more severe "reactor coolant boundary failure" category and need not be repeated here. All the other direct initiators cause a reduction in primary system flow or secondary cooling. Basically, all are expected to progress in similar fashion, so a single general decrease in heat removal event tree should suffice. Separate initiating event groups are provided to account for important differences in reactor coolant system pressure response under ATWS conditions (complete loss of feedwater and turbine trip) and for dependent failures due to specific initiators (the remaining three). Three of the direct initiators are common cause initiators, MLD-10, MLD-11, and MLD-12, and although the same event tree structure may be appropriate, quantification will be different for each.

Note that during normal controlled shutdown, the plant is near equilibrium, shutdown proceeds at a controlled rate, and standby systems are started before they are needed. If such systems fail, most of the normal systems are available to maintain operation, the allowed response and recovery systems are much greater and the number of safety functions that must be performed to provide sufficient core cooling is reduced. Therefore, normally controlled shutdown and startup are not considered to be initiating events because they are insignificant risk contributors.

The indirect initiating events include all events not yet enumerated that disrupt the stable operation of the plant and that require standby systems to terminate transient effects. Two rapid cooldown events are included as indirect initiating events: steam release inside and outside containment. If the increased steaming rate is sustained and the reactor does not trip, the sequence can cause a core power increase. If the reactor trips and standby cooling systems fail, a heatup condition (insufficient core heat removal) can develop. Spurious safety injection can lead to reactor trip, increase in RCS pressure, and, if standby cooling systems fail, to a heatup condition as well.

A general indirect initiator is included to cover all other events that would be expected to proceed via reactor trip. Finally, five common cause initiators, MLD-18, MLD-19, MLD-20, MLD-21, and MLD-22, have been identified for this indirect category.

It is possible that the group, "insufficient core heat removal initiators," can also lead to reactor coolant boundary failure. Such failures will be developed along the accident sequences; therefore, this connection is not shown on the master logic diagram (MLD). The diagram is thus complete at level VI. The specific decomposition to level VII represents a judgment of the number of unique scenario models (logic and data) needed to quantify the risk. At this level, the completeness of the categorization process is afforded by the inclusion of two general categories: one for direct initiators and one for indirect initiators, which are intended to account for all initiating events not otherwise identified at this level. The 24 categories identified at level VII and numbered MLD-1 through MLD-24 represent the first major step of the process of categorizing initiating events that is continued in the following sections.

C.1.2.2 Heat Balance Fault Tree Method

As described in Appendix A, the heat balance fault tree (HBFT) method attacks the initiating event issue from a different direction. The top event for the heat balance fault tree method is "initiating event occurs," which is substantially different than "potential offsite release." The fault tree logic development that ensues is based on the concept that any initiating event must involve an upset or imbalance in the thermal equilibrium that otherwise exists in the reactor core and its heat removal systems. By noting the heat transport paths associated with maintaining this equilibrium, which involves transferring the heat generated in the reactor fuel rods by the processes of fission and radioactive decay, the logical development of the fault tree can proceed to the point of identifying initiating events. However, it should be noted that, due to the detailed nature of this analysis, a portion of the events identified by the HBFT method either will not cause a plant trip or can be grouped with other events. Table C.1-2 provides an analysis for grouping or eliminating of the preliminary initiating events identified by the HBFT method.

The principal heat transport paths illustrated in Figure C.1-2 basically involve the transfer of thermal energy from the reactor core, to the reactor coolant system, to the secondary coolant system via the steam generators, and finally, directly to the environment and to the turbine generator where it is converted to electrical energy and delivered to the electrical grid. There are additional paths that represent heat losses and paths to directly bypass thermal energy from intermediate points in the principal pathways to the environment and, in some cases, via the containment heat removal systems. These alternative paths were considered only insofar as they impact the heat balance along the principal heat transport path. For example, initiators involving the steam dump valves or PORVs are of interest, but initiators occurring in the containment heat removal systems are not. Another important element of the thermal equilibrium process is the heat removal by the various

component cooling systems, which are special cases of heat losses but necessary to maintain equilibrium; i.e., steady state conditions. These steady state conditions are not restricted to reactor power operation states, but include any hot or cold shutdown condition in which thermal equilibrium is maintained. Of course, consideration of shutdown conditions is not unique to this approach. In addition to steady state initial conditions, the process of identifying initiating events must also concern itself with the possibility of an initiating event occurring during an otherwise routine startup or shutdown of the plant. For the purpose of the heat balance fault tree analysis, these near steady state conditions are regarded as steady states to make the analysis as comprehensive as possible.

The heat balance fault tree is presented in Figure C.1-3. The logic development is based on the heat balance diagram of Figure C.1-2, which indicates that any perturbations in the steady state conditions of the plant associated with an initiating event would have to represent an imbalance: (1) in energy transfer between the core and reactor coolant system, (2) between the reactor coolant and secondary coolant system (SCS), or (3) between the SCS and the plant output. Note that the fault tree is developed for an instant in time; therefore, each event is developed to identify the source of the perturbation. Over an interval of time, a given initiating event may result in the occurrence of all three of the above imbalances; however, in this analysis, we are only interested in identifying how the transient was initiated, since the event sequence analysis described in Section C.4 covers the transient plant response as it pertains to accident sequence definition. The consideration of departures from equilibrium at an instant in time makes it necessary to consider each initiator only once.

At level 3, it is shown that the source of each imbalance can originate on either side of the energy transfer points identified in level 2. This creates seven categories because the third energy transfer point distributes energy to two output points: the electrical output and the thermal waste heat release to the environment. At level 4, it is shown that each change identified at level 3 can start out as an increase or a decrease in energy generation or transfer. This approach leads to the identification of some categories that, on the surface, would appear to be insignificant or redundant in terms of potential for accident sequence initiation because they appear to enhance plant ability to maintain basic safety functions; i.e., control core reactivity, provide core and RCS heat removal, etc. Such categories include: decrease in core heat generation, increase in RCS heat removal from the core, increase in secondary cooling system (SCS) heat removal from the RCS, etc. These categories are included in the analysis because of the desire to examine any plant transient condition that has the potential to shut down the normally operating systems and therefore challenge the standby systems as a means of maintaining basic safety functions. Stated another way, even if a transient starts out in a favorable direction with respect to maintaining basic safety functions, if certain control and protection systems fail to respond properly, the transient could become unfavorable with respect to these safety functions. Since the likelihood of these failures is considered in the event tree quantification, it is

appropriate to identify these types of seemingly favorable events. Such events were normally classified as indirect initiators in the MLD analysis described in the preceding section.

The transfer symbols at the bottom of the fault tree in Figure C.1-3a correspond with the names for the 14 initiating event categories, each of which is developed down an additional 2 levels in the continuation of the fault tree in Figures C.1-3b through C.1-3h. At the sixth level, the initiating event categories are specified at a level of detail normally adequate for constructing and quantifying event trees. As described in Section C.1.4, after examining each of the initiating events individually, some of the preliminary initiating events are screened out and others are grouped for the purpose of constructing and quantifying event trees. Since we are only concerned with qualitatively identifying a complete set of categories in this analysis, in cases where a particular initiating event would simultaneously cause the occurrence of two or more conditions identified at level 4, the event appears only once. Since the fault tree has only "OR" gates and each event appears only once, a complete list of initiating event categories simply consists of a list of the entries at the bottom level of the tree. As with any valid fault tree, completeness can be assured at each level except the last. In consideration of this, the independent checks made in Section C.1.3 demonstrate the adequacy of the list with respect to completeness.

A cross-reference between the list of preliminary initiating event categories identified using the master logic diagram and heat balance fault tree methods can be made upon examination of Table C.1-3. This table shows the relationship between these categories at two different levels of categorization for the HBFT method by noting the letter prefix and its correspondence with the level 4 categories in the HBFT (Figure C.1-3). In general, it can be seen that the level of classification has been carried down to a finer level in the HBFT categories, especially in regard to breaking up the two general categories in the MLD, "general loss of heat removal (MLD-7)" and "general indirect initiators (MLD-17)," into 21 and 25 separate categories, respectively. This distinction is not fundamental since both approaches can be developed to the same level. However, there are several exceptions to this observation worth noting. First, the specification of loss of coolant inventory categories under which both approaches include the inadvertent opening of a PORV was identical, perhaps reflecting the emphasis this class of initiators has received in past PRAs and other safety studies of light water reactors. This factor transcends either the MLD or HBFT methods. Second, loss of instrument AC power, loss of instrument air, loss of ventilation and air conditioning, and noncore releases were identified in the MLD approach, but not in the HBFT. With regard to loss of instrument AC power, loss of instrument air, and loss of ventilation and air conditioning event, the HBFT simply was not carried down to a fine enough level of detail to resolve the support systems failures per se. However, a more detailed analysis on dependent failures related to the three events is carried out by the failure modes and effects analysis method. The noncore releases category was not included in the HBFT because of the focus on departures from reactor core thermal equilibrium. The release of noncore sources of

radioactivity was not analyzed further because of the extremely low potential to impact public health and safety for those sources.

A final important conclusion from the comparison of preliminary initiating event categories in Table C.1-3 is enhanced coverage of support system failures that were identified using the MLD method as a result of a special search for common cause initiating events. A common cause initiating event is an event that causes an initiating event and failure or degradation of one or more systems that are called upon to respond to that initiating event. At the level of detail found optimum for the HBFT, the detailed consideration of support system failures was not found to be practical. In fact, the same observation can be made about the MLD with the implication that the differences obtained in the area probably reflect the different analysts that performed the evaluations.

The conclusion that was drawn upon completion of the HBFT was that its approach to categorization was better defined and more complete than the MLD categories for accident sequences involving the reactor core. The only fundamental differences are the different top events chosen for the respective trees. The MLD approach does have the advantage of organizing the events at the lowest levels with respect to the generation of appropriate event tree structures. Both are subject to the usual strengths and weaknesses of fault tree analysis. The principal advantage of having applied both approaches is the value of having analysts attack the same problem from two related although somewhat different perspectives. It was further concluded that additional analyses were needed to search for common cause initiating events to ensure that they are adequately resolved for proper treatment of dependent failures. In principle, although the fault tree approach can also be applied to tackle this problem, it was found in practice that the size of the fault trees already experienced with the HBFT and MLD approaches could not be increased without resorting to computer analysis. The identification of common cause initiating events concerns itself with two broad classes of events: support system failures and external events. These additional analyses performed to supplement the fault tree approaches are described below.

C.1.2.3 External Common Cause Initiating Events

An important class of common cause initiating events are due to "external events" that include events external to the plant and events such as fires, which can occur inside the plant but external to the plant processes. External events analysis is such a major task in the development of a risk model that an entire section has been reserved for its documentation (Appendix F). As an integral part of this analysis, physical interactions are identified that cause one or more initiating events and possible additional damage to one or more plant systems. The analysis of these interactions is specific to the types of external events, which include:

- Seismic Events
- Fires and Explosions
- Internal and External Flooding

- Aircraft Crash
- Wind and Wind-Generated Missiles
- Turbine Missile
- Hazardous Chemical Releases
- Transportation Accidents

A list of common cause initiating events identified in Appendix F for the above sources of physical interaction and selected for quantification via the plant model is presented in Table C.1-1.

C.1.2.4 Common Cause Initiating Events Due to Support System Failures: Failure Modes and Effects Analysis Method

As seen in the previous sections, a number of approaches can be conceived for identifying potential contributors to the list of initiating events. Each approach offers its own unique perspective that helps build confidence in the fact that the list of initiating events is complete with regard to the significant (or dominant) risk contributors. The master logic diagram and the heat balance fault tree methods are examples of top-down, or deductive approaches that work from an endpoint down to finer levels of categorization and that reveal the causes of the originally defined effect. Another perspective can be achieved from a bottom-up approach; that is, investigating the possible malfunctioning of various systems and subsystems within a plant and identifying the effects of those malfunctions on plant operation. This approach provides a different means of identifying potential initiating events and augments the search for initiating events performed in the systems analysis task in Appendices D and E.

The analysis presented below is of the bottom-up type of approach and is known as failure modes and effects analysis. This analysis focuses on the following seven Diablo Canyon support systems and subsystems known to contain potential common cause initiating events.

1. Electric Power System (EPS)

2. Auxiliary Salt Water (ASW) System

3. Ventilation and Air Conditioning (VAC) Systems

4. Instrument Air System (IAS)

5. Component Cooling Water (CCW) System

6. Service Cooling Water (SCW) System

7. Solid State Protection System (SSPS)/Engineered Safety Features Actuation System (ESFAS)

A simple, specialized failure modes and effects analysis was performed for the first six support systems and is presented in Table C.1-4. This provides another approach to looking for initiators in support systems. Support system initiators were identified using the MLD method, but the HBFT was not developed sufficiently to include them. Failure modes are listed for each system (focusing on the major subsystems in the case of the EPS). For each failure mode, the frontline or support systems whose functions are adversely affected by the failure are listed followed by the associated initiating event category. It should be pointed out that the purpose of this analysis is to identify an initiating event after a postulated support system failure and is not to investigate the extent of plant systems affected. The analysis is suspended either after identification of the first initiating event or confirmation that no plant trip will occur after the postulated failure. An FMEA was not performed for the SSPS and ESFAS systems because several initiators, which have already been identified via the HBFT and MLD approaches, are judged to be the most important in this category. This conclusion is supported in the systems analysis task as described in Appendix D. The specific initiator categories associated with these systems are:

- Inadvertent Safety Injection
- Reactor Trip

The greatest area of interest concerns losing all buses, loops, or trains of a single support system (or subsystem) because of the more severe effects that such failure conditions have on other support and frontline systems and on their associated initiating event categories. Although losing single bus, loop, or train was expected to have less severe effects, such losses were included in the FMEA tables to enhance completeness and to account for their relatively greater frequency of occurrence. Many of the failure modes investigated for the support systems in Table C.1-4 were found to result in minimum impact to plant normal operation. For example, failure of one auxiliary salt water pump or of one component cooling water pump would generate a low pressure signal at the pump discharge header to actuate the standby unit and the required cooling water flow will not be interrupted. Failures of one 125V DC vital bus or both trains of auxiliary salt water or component cooling water are identified as potentially significant initiators. Examination of the FMEA results identify two additional initiating event categories; they are loss of the 480V switchgear ventilation and loss of instrument air. The loss of the instrument air initiator is grouped together with the loss of main feedwater category. Although a loss of instrument air would impact other equipment, no credit is currently given (i.e., a conservative assumption) for the quantification. Thus, modeled impacts are the same as if these were just a loss of main feedwater.

Rather than model all three initiating events representing loss of one nonvital DC bus, it was decided to conservatively model loss of any one of the three as loss of DC bus 12 associated with electric power train G. Failure of DC bus 12 is judged to be more limiting because (1) it causes the failure of two vital instrument buses and the backup power supply from the regulated transformer via 480V bus 1G, (2) it provides control power to one of the two Unit 1 auxiliary saltwater

pumps, (3) its failure causes the feedwater regulator valves to close, (4) its failure precludes one of the two turbine trip circuits, (5) its failure prevents operation of the turbine-driven auxiliary feedwater pump without manual intervention, and (6) it provides the control power for one of the two spray pumps and RHR pumps. These impacts are judged to be more limiting than loss of either of the other two. A number of other partial support system failures are found to lead to plant trip, but these were judged to be unimportant on a case-by-case basis. Thus, this analysis serves to confirm the previously developed list of initiating events for Diablo Canyon, thereby increasing confidence in its completeness.

As a final note on the process of identification of initiating events, the systems analysis task provides an important input. This is accomplished in a fashion similar to the FMEA in which each systems analyst analyzes and documents the system failure modes that cause initiating events. Please refer to Appendix D for details.

C.1.3 CHECKS ON INITIATING EVENT COMPLETENESS

To ensure completeness in the listing of Diablo Canyon initiating events, a review was made of the initiating events identified in the EPRI NP-2230 report (Reference C.1-3), the WASH-1400 report (Reference C.1-4), the Diablo Canyon Final Safety Analysis Report (FSAR), the PRA Procedures Guide (Reference C.1-5), the Indian Point Probabilistic Safety Study (IPPSS) (Reference C.1-6), and the Seabrook Station Probabilistic Safety Assessment (SSPSA) (Reference C.1-7).

Table C.1-5 lists the initiating events extracted from the above sources belonging to the various major categories. The events listed within a major category from these sources correspond to those identified for Diablo Canyon by the HBFT method given in column 2 of Table C.1-5. It can be seen that there is not an exact matching between the candidate Diablo Canyon initiating events and those from the other sources. This, however, is not unexpected since the approach and criteria adopted in the initiating event selection and categorization can be very plant specific or it can also be very generic, encompassing all commercial pressurized water reactor (PWR) plant designs; e.g., the WASH-1400 report.

In the EPRI NP-2230 report, only anticipated transients were compiled; hence, no LOCA events were listed under the corresponding column in the table. On the other hand, the PRA Procedures Guide does not list many important transient events. A reference to the EPRI NP-801 report (Reference C.1-8) was made in this guide with regard to these events. The Diablo Canyon FSAR examined many important initiating events associated with design basis accidents and their analyses (see table under column, Diablo Canyon FSAR). Most of these events were prescribed from generic light water reactor (LWR) licensing ground rules.

After gaining an understanding of the differences in the initiating event lists in Table C.1-5, it was concluded that the Diablo Canyon categories identified in the previous sections are as complete a set as can be reasonably expected. None of the identified differences point to a need

to add any new initiating event categories to the list although there is the need to properly resolve the common cause initiating events, as described in the previous section.

C.1.4 FINAL SELECTION AND GROUPING OF INITIATING EVENTS

Having described the many approaches taken to ensure adequate completeness in the process of identifying candidate initiating events for Diablo Canyon, it is necessary to perform additional grouping and screening to facilitate development of the event sequence model. The objective here is to select a sufficient set of initiating events for detailed event sequence modeling and quantification. It is only necessary and practical to analyze those initiating events that make appreciable contributions to risk. Given knowledge of the approximate frequency of the initiating events and the relative impact of the events on plant systems and structures, it is possible and desirable to group and screen initiating events to simplify the quantification of risk without introducing significant errors in the risk estimate. As a hypothetical example, if a turbine missile that results in only the loss of one diesel generator occurs at a frequency of say, 1×10^{-7} per reactor year and loss of offsite power and failure of both diesels due to other causes is known to occur at say, 1×10^{-4} per reactor year, the turbine missile scenario need not be analyzed in great detail to come to the conclusion that its contribution to risk is relatively insignificant.

The process of screening and grouping initiating events is an iterative process that interacts with the development of the event sequence model. The analysis of all the candidate initiating events identified by the HBFT method is presented in Table C.1-2. This table describes how the preliminary initiating event categories were grouped and why some were screened out from the final quantification. The final list for quantification is presented in Table C.1-6 with cross-references to MLD, HBFT, and FMEA.

Examination of Table C.1-6 and related figures shows that final initiating event categories 1 through 14 are identified by both the master logic diagram method and the heat balance fault tree method. However, categories 9, 10, 14b, 14c, 14d, and 14f are identified only in the general event categories of the MLD, while the HBFT method provides a finer definition for each of these initiating events categories. For the balance of final initiating events, it is shown that categories 15 through 21 are derived from both the MLD method and the FMEA method.

As noted earlier, the group selected for detailed modeling represents a very comprehensive set of initiating events compared with usual practice in a PRA project. The event sequence model for each of these initiating events is described in Sections C.4 and F.1.

C.1.5 REFERENCES

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C.1-4. U.S. Nuclear Regulatory Commission, "Reactor Safety Study: An Assessment of Accident Risks in U.S. Nuclear Power Plants," WASH-1400 (NUREG-75/014), October 1975.

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C.1-6. "Indian Point Probabilistic Safety Study," Consolidated Edison Company of New York, Inc., and the Power Authority of the State of New York, March 1982.

C.1-7. Pickard, Lowe and Garrick, Inc., "Seabrook Station Probabilistic Safety Assessment," prepared for the Public Service Company of New Hampshire and the Yankee Atomic Electric Company, PLG-0300, December 1983.

C.1-8. Electric Power Research Institute, "Frequency of Anticipated Transient, A Reappraisal," EPRI NP-801, ATWS, July 1978.

TABLE C.1-1. INITIATING EVENT CATEGORIES SELECTED FOR QUANTIFICATION OF THE DIABLO STATION RISK MODEL

Group	Initiating Event Categories Selected for Separate Quantification	Code Designator
Loss of Coolant Inventory	1. Excessive LOCA	ELOCA
	2. Large LOCA	LLOCA
	3. Medium LOCA	MLOCA
	4. Small LOCA, Nonisolable	SLOCN
	5. Small LOCA, Isolable	SLOCI
	6. Interfacing Systems LOCA	
	6a. At RHR Pump Suction	VS
	6b. At RHR Pump Discharge	VD
	7. Steam Generator Tube Rupture	SGTR
Transients	8. Reactor Trip	RT
	9. Turbine Trip	TT
	10. Loss of Condenser Vacuum	LCV
	11. Closure of All MSIVs	AMSIV
	12. Steam Line Break Inside Containment	SLBI
	13. Steam Line Break Outside Containment	SLBO
	14. Inadvertent Safety Injection	ISI
	15. Main Steam Relief Valve Opening	MSRV
	16. Total Main Feedwater Loss (includes feedwater line break)	TLMFW
	17. Partial Main Feedwater Loss	PLMFW
	18. Excessive Feedwater	EXFW
	19. Closure of One Main Steam Isolation Valve (MSIV)	IMSIV
	20. Core Power Excursion	CPEXC
	21. Loss of Primary Flow	LOPF
Common Cause Initiating Events		
Support System Faults	22. Loss of Offsite Power	LOSP
	23. Loss of One DC Bus	L1DC
	24. Total Loss of Auxiliary Salt Water	LOSW
	25. Total Loss of Component Cooling Water	LPCC
	26. Loss of 480V Switchgear Ventilation	LOSWV
	27. Loss of Control Room Ventilation	LOCV
Seismic Events	28. 0.2g to 1.25g	EQ0.7
	29. 1.25g to 1.75g	EQ1.5
	30. 1.75g to 2.0g	EQ1.9
	31. 2.0g to 2.5g	EQ2.25
	32. 2.5g to 3.0g	EQ2.75
	33. 3.0g to 4.0g	EQ3.5

TABLE C.1-1 (continued)

Group	Initiating Event Categories Selected for Separate Quantification	Code Designator
Fire and Smoke	34. Loss of Both Motor-Driven AFW Pumps	FS1
	35. Loss of All Charging Pumps and MSIV Closure	FS2
	36. Loss of Component Cooling	FS3
	37. Loss of Control Ventilation	FS4
	38. Loss of Auxiliary Saltwater	FS5
	39. Loss of 4-kV Buses HF and HG	FS6
	40. Loss of 4-kV Buses HG and HH	FS7
	41. Loss of 4-kV Buses HF, HG, and HH	FS8
	42. Control Room Fire at Vertical Board VB-1	FS9
	43. Control Room Fire at Vertical Board VB-2	FS10
	44. Control Room Fire at the Interface of Vertical Boards VB-2 and VB-3	FS11
	45. Control Room Fire at Vertical Board VB-4	FS12
	46. Cable Spreading Room Fire One	FS13
	47. Cable Spreading Room Fire Two	FS14
Flood, Jets, and Sprays (pipe breaks)	48. Loss of All Auxiliary Feedwater	FS15
	49. Loss of Both Motor-Driven AFW Pumps	FS16
	50. Loss of Auxiliary Saltwater	FS17
Other (vessel impact)	51. Loss of Auxiliary Saltwater	EX1

TABLE C.1-2. QUALITATIVE ANALYSIS AND GROUPING OF INITIATING EVENT CATEGORIES IDENTIFIED BY THE HBFT METHOD

Major Category	Initiating Event Category	Final Category Assignment*	Basis for Screening from Final Quantification
CI, Increase in Core Heat Generation	1a Control Rod Ejection	20	
	1b Control Rod Disassembly	20	
	1c Uncontrolled Rod/Bank Withdrawal	20	
	3a CVCS Malfunction Boron Dilution	N/A	Slowly moving transient. Frequency and level of impact bounded by group 14e.
	3b Boron Precipitation	N/A	
	4a Improper Fuel Loading	N/A	
	4b Burnable Poison and Fuel Geometry Change	N/A	
CD, Decrease in Core Heat Generation	1a Reactor Trip	8	
	1b Control Rod Drop	N/A	
	1c Manual Reactor Trip	8	Frequency and level of impact bounded by group 7.
	1d Plant Control System Malfunctions	N/A	
	3a CVCS Malfunction Boron Concentration	N/A	
CRI, Increase in RCS Heat Removal from Core	1a Startup of Idle RCS Pump	20	
	1b Increase in RCS Pump Speed	N/A	Impact is inconsequential, precluded by use of fixed speed pumps.
	3a CVCS and Function; Increase Charging over Letdown	N/A	Frequency and impact bounded by group 13a.
	3b Spurious Safety Injection	14	
	4a Increase in Pressurizer Heater Output	N/A	

*N/A indicates category not analyzed further for reason indicated. Categories are numbered for reference to Table C.1-1.

TABLE C.1-2 (continued)

Major Category	Initiating Event Category	Final Category Assignment*	Basis for Screening from Final Quantification
CRD, Decrease in Heat Removal from Core	1a RC Pump(s) Trip	21	
	1b RC Pump Shaft Seizure	21	
	1c RCP Shaft Break	21	
	1d Reduction in RCP Pump Speed	21	
	1e RC Loop Flow Blockage	21	
	1f Core Blockage and Boron Precipitation	N/A	Considered only in long-term response to large LOCA.
	3a CVCS Decrease in Charging over Letdown	N/A	Frequency and impact bounded by group 4.
	4a Decrease in Pressurizer Heaters	N/A	
	4b Pressurizer Spray Actuation	N/A	
	5a Small LOCA Inside Containment; Nonisolable	4;8	
	5b Medium LOCA Inside Containment	3	
	5c Large LOCA Inside Containment	2	
	5d Isolable Small LOCA	5	
	5e Interfacing Systems LOCA	6	
	5f Steam Generator Tube Rupture	7	
	5g Excessive LOCA	1	
RSI, Increase in RCS Heat Transfer to SCS	4a Freeing of Steam Generator Tube Blockage	N/A	Mild transient that would not directly challenge any system.
RSD, Decrease in RCS Heat Transfer to SCS	4a Steam Generator Tube Blockage	N/A	Mild transient that would not directly challenge any system.

*N/A indicates category not analyzed further for reason indicated. Categories are numbered for reference to Table C.1-1.

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TABLE C.1-2 (continued)

Major Category	Initiating Event Category	Final Category Assignment*	Basis for Screening from Final Quantification
SRI, Increase in SCS Heat Removal from RCS	1a Opening of Feedwater Heater Bypass Valve	20	
	1b Loss of Steam to FWH	20	
	2a Increased Feed Pump Speed	18	
	2b Start of Idle Feed Pump	18	
	2c Spurious AFW Actuation	18	
	2d Start of Idle Heater Drain Pump	18	
	3a Main Steam Relief Valve Opens	15	
	3b MSSV Opens	N/A	Frequency and impact bounded by group 13b.
	3c TCV Opens	N/A	
	3d TBV Opens	N/A	
	3e Steam Line Break - Inside Containment	12	
	3f Steam Line Break - Outside Containment	13	
	4a Freeing of Steam Generator Crud Blockage	N/A	Mild transient that would not challenge any system.
SRD, Decrease in SCS Heat Removal from RCS	1a Freeing of Feedwater Heater Steam Side Blockage	N/A	Mild transient that would not challenge any system.
	2a Reduction Feedwater Pump Speed	17	
	2b Feedwater Pumps(s) Trip	16	
	2c Condensate Pump(s) Trip	16	
	2d Feedwater Heater Drain Pump(s) Trip	16	
	2e Feedwater Pipe Leak or Rupture	16	
	2f Feedwater Isolation Valve Closure	16	
	2h Closure of Feedwater Regulator Valve	16	
	3a Closure of One or More MSIVs	11;19	
	3b Closure of TCVs or Closure of TSVs	9	
	4a Steam Generator and SCS Flow Blockage	17	

*N/A indicates category not analyzed further for reason indicated. Categories are numbered for reference to Table C.1-1.

TABLE C.1-2 (continued)

Major Category	Initiating Event Category	Final Category Assignment*	Basis for Screening from Final Quantification
SPI, Increase in Energy Transfer to Plant Output, MLD-17	2b Enhancement of Condenser Vacuum	N/A	Mild transient that would not challenge any system.
SPD, Decrease in Energy Transfer in Plant Output MLD-7	2a Reduction of Condenser Vacuum	N/A	Mild transient that would not challenge any system.
	2b Loss of Condenser Vacuum	10	
EI, Increase in Electrical Output MLD-17	1a Increase in Electricity Demand	N/A	Any unfavorable outcome would lead to turbine-generator trip; analyzed in group 8.
	1b Grid Instabilities	N/A	
	1c Generator Load/Limiter Fault	N/A	
ED, Decrease in Electrical Output MLD-14a	1a Decrease in Electricity Demand	N/A	Frequency and impact bounded by group 8.
	1b Loss of Generator Load	9	
	1c Generator Failure	9	
	1d Grid Instabilities	N/A	Also analyzed in turbine missile analysis.
	2a Key Breaker Fails Open	9	
	3a Generator Trip	9	
	4a Turbine Trip	9	
	4b Turbine Blade/Rotor Failure	9	
TI, Increase in Thermal Output MLD-17	2a Startup of Idle CW Pump	N/A	
	2b Increase in CW Pump Speed	N/A	
	2c Freeing of CW Tunnel Blockage	N/A	Mild transient that does not challenge any system.
	3a Natural Fluctuations in Ocean Temperature	N/A	

*N/A indicates category not analyzed further for reason indicated. Categories are numbered for reference to Table C.1-1.

TABLE C.1-2 (continued)

Major Category	Initiating Event Category	Final Category Assignment*	Basis for Screening from Final Quantification
TI, Increase in Thermal Output MLD-17 (continued)	4a Freeing of Plugged or Blocked Condenser Tubes	N/A	
TD, Decrease in Thermal Output MLD-7, MLD-8	2a Trip of One or More CW Pumps	10	
	2b Decrease in CW Pump Speed	10	
	2c Debris in CW Pump	10	
	2d Leak or Rupture of CW Intake/ Condenser Expansion Joint	N/A	Mild transient that does not challenge any system.
	4a Plugging and Blockage of Condenser Tubes	N/A	
	3a Natural Fluctuation in Ocean Temperature	N/A	
	3b Demusseling Operations	N/A	
Support System Faults	Electric Power System Faults	22;23	
	Auxiliary Salt Water System (MLD-11)	24	
	Component Cooling Water System Faults (MLD-12)	25	
	Service Cooling Water Faults	14;15	
	Instrument Air Faults	16	
	480V Switchgear Ventilation Faults	26	
	Control Room Ventilation Faults	27	

*N/A indicates category not analyzed further for reason indicated. Categories are numbered for reference to Table C.1-1.

TABLE C.1-3. CROSS-REFERENCE TABLE FOR PRELIMINARY INITIATING EVENT CATEGORIES IDENTIFIED USING THE MLD AND HBFT METHODS

Master Logic Diagram Categories	Corresponding Heat Balance Fault Tree Categories	
MLD-1 Excess LOCA	CRD-5g	Excessive LOCA
MLD-2 Large LOCA	CRD-5c	Large LOCA
MLD-3 Medium LOCA	CRD-5b	Medium LOCA
MLD-4 Small LOCA	CRD-5a	Small LOCA, Nonisolable
	CRD-5d	Small LOCA, Isolable
MLD-5 LOCA Outside Containment	CRD-5e	Interfacing Systems LOCA
MLD-6 Steam Generator Tube Rupture	CRD-5f	Steam Generator Tube Rupture
MLD-7 General Loss of Heat Removal	CRD-1a	Reactor Coolant Pump (RCP) Trip
	CRD-1b	RCP Shaft Seizure
	CRD-1c	RCP Shaft Break
	CRD-1d	Reduction in RCP Speed
	CRD-1e	RCS Loop Blockage
	CRD-1f	Core Blockage and Boron Precipitation
	CRD-3a	CVCS Decrease in Charging Pump over Letdown
	CRD-4a	Decrease in Pressurizer Heaters
	CRD-4b	Pressurizer Spray Actuation
	RSD-4a	Steam Generator Tube Blockage
	SRD-1a	Freeing of Feedwater Heater Steam Side Blockage
	SRD-2a	Reduction in Feedwater Pump Speed
	SRD-4a	Steam Generator SCS Flow Blockage
	SPD-2a	Reduction in Condenser Vacuum
	SPD-2b	Loss of Condenser Vacuum
	TD-2b	Decrease in Circulating Water Pump Speed

TABLE C.1-3 (continued)

Master Logic Diagram Categories	Corresponding Heat Balance Fault Tree Categories	
MLD-7 General Loss of Heat Removal (continued)	SRD-3a	Natural Fluctuations in Ocean Temperature
	SRD-4a	Plugging/Blockage of Condenser Tubes
	TD-2a	Trip of Circulating Water Pumps
	TD-2c	Circulating Water Intake Blockage
	TD-2d	Circulating Water Intake Rupture
MLD-8 Complete Loss of Feedwater	SRD-2b	Feedwater Pumps Trip
	SRD-2c	Condensate Pumps Trip
	SRD-2d	Feedwater Heater Drain Pumps Trip
	SRD-2e	Feedwater Pipe Rupture
	SRD-2f	Feedwater Isolation Valve Closure
	SRD-2h	Closure of Feedwater Regulator Valve
	SRD-3a	Closure of All MSIVs
MLD-9 Turbine Trip	ED-1b	Loss of Generator Load
	ED-1c	Generator Failure
	ED-1d	Grid Instabilities
	ED-2a	Generator Breaker Failure
	ED-3a	Generator Trip
	ED-4a	Turbine Trip
	ED-4b	Turbine Blade/Rotor Failure
	SRD-3b	Closure of TCVs or TSVs
MLD-10 Loss of Load and Offsite Power	*	
MLD-18 Loss of One DC Bus	*	
MLD-19 Loss of One AC Bus	*, **	

*Not developed to sufficient detail by the HBFT method. FMEA provides an analysis on the plant systems affected.
**FMEA shows that one AC bus failure is not an initiating event; i.e., it does not interrupt power generation.

TABLE C.1-3 (continued)

Master Logic Diagram Categories	Corresponding Heat Balance Fault Tree Categories	
MLD-11 Complete Loss of Auxiliary Salt Water	*	
MLD-12 Complete Loss Of CCW	*	
MLD-13 Reactor Trip	CD-1a	Reactor Trip
	CD-1b	Control Rod Drop
	CD-1c	Manual Reactor Trip
	CD-1d	Plant Control System Faults
	CD-3a	CVCS Boron Increase
MLD-14 Loss of Steam Inside Containment	SRI-3e	Steam Line Break Inside Containment
MLD-15 Loss of Steam Outside Containment	SRI-3f	Steam Line Break Outside Containment
	SRI-3a	Opening of Atmospheric Relief Valve
	SRI-3b	Opening of MSSV
	SRI-3c	TCV Opens
	SRI-3d	TBV Opens
MLD-16 Spurious Safety Injection	CRI-3b	Spurious Safety Injection
MLD-17 General Indirect Initiator	CRI-1b	Increase in RCP Speed
	CRI-3a	CVCS Increase Charging Over Letdown
	CRI-4a	Increase in Pressurizer Heaters
	CRD-4a	Decrease in Pressurizer Heaters
	CRD-4b	Pressurizer Spray Actuation

*Not developed to sufficient detail by the HBFT method. FMEA provides an analysis on the plant systems affected.

TABLE C.1-3 (continued)

Master Logic Diagram Categories	Corresponding Heat Balance Fault Tree Categories	
MLD-17 General Indirect Initiator (continued)	RSI-4a	Freeing of Steam Generator Tube Blockage
	SRI-2a	Increase in Feed Pump Speed
	SRI-2b	Startup of Idle Feed Pump
	SRI-2c	Spurious AFW Actuation
	SRI-2d	Startup of Idle Feedwater Heater Drain Pump
	SRI-3c	Opening of TCV
	SRI-3d	Opening of TBV
	SRI-4a	Freeing of Steam Generator Blockage
	SPI-2b	Enhancement of Condenser Vacuum
	EI-1a	Increase in Electricity Demand
	EI-1b	Grid Instabilities
	EI-1c	Generator Load/Limiter Fault
	ED-1a	Decrease in Electricity Demand
	TI-2a	Start of Idle Circulating Water Pump
	TI-2b	Increase in Circulating Water Pump Speed
	TI-2c	Freeing of Circulating Water Intake Blockage
	TI-2d	Natural Fluctuations in Ocean Temperature
	TI-4a	Freeing of Plugged Condenser Tubes
	TD-3a	Natural Fluctuations in Ocean Temperature
	TD-3b	Demusseling Operations
	TD-4a	Blockage of Condenser Tubes
MLD-20 Loss of Instrument AC Power	*, **	
MLD-21 Loss of Instrument Air	*	

*Not separately identified by the HBFT method. FMEA provides an analysis on the plant systems affected.
**FMEA shows that failure of more than one instrument channel is a low frequency event and is not included as a separate initiating event.

TABLE C.1-3 (continued)

Master Logic Diagram Categories	Corresponding Heat Balance Fault Tree Categories	
MDL-22 Loss of Ventilation and Air Conditioning (VAC) Systems	*	
MLD-23 Core Power Increase	CI-1a	Control Rod Ejection
	CI-1b	Control Rod Disassembly
	CI-1c	Uncontrolled Rod/Bank Withdrawal
	CI-3a	CVCS Boron Dilution
	CI-3b	Boron Precipitation
	CI-4a	Improper Fuel Loading
	CI-4b	Burnable Poison and Fuel Geometry Change
	SRI-1a	Feedwater Heater Bypass Valve Opens
	SRI-1b	Loss of Steam to Feedwater Heater
	CRI-1a	Start Idle RCP
MLD-24 Noncore Release	Does Not Cause Plant Heat Imbalance**	

*Not separately identified by the HBFT method. FMEA provides an analysis on the plant systems affected.
**As discussed in Section C.1.2.1, noncore release imposes a negligible risk of public health impact and is not analyzed in this study.

TABLE C.1-4. FAILURE MODES AND EFFECTS ANALYSIS OF SUPPORT SYSTEM INDUCED COMMON CAUSE INITIATING EVENTS*

Support System	Failure Mode	Systems Affected	IE Category
1. Electric Power	Lose Offsite Grid	Reactor Coolant Pumps**	CRD-1a
		Main Feedwater Pumps**	SRD-2b
		Control Rod Drive**	CD-1a
		Turbine**	ED-4a
		Instrument Air**	See Section 4, This Table
	Lose All Three 4.16-kV Vital Buses	Safety Injection Pumps	TOPS†
		Charging Pumps	MLD-12
		Residual Heat Removal	TOPS
		Component Cooling Water	MLD-12
		Auxiliary Salt Water	MLD-11
		Pressurizer PORV Block Valves	None
		Containment Spray	TOPS
		Auxiliary Feedwater	TOPS
		Diesel Generators	TOPS
		VAC Systems	See Section 3, This Table
		480V Buses	See Systems under Loss of All 480V Vital Buses
		Containment Isolation††	None
	Lose All 480V Vital Buses	VAC System	See Section 3, This Table
		Containment Fan Cooler	TOPS
		Safety System Valves	None

*This analysis serves only to identify if an initiating event occurs after a postulated support system failure and is not to investigate the extent of all plant systems affected, therefore the FMEA search is suspended after identification of the first initiating event.
**If the plant successfully ramps back to house load, then this equipment function will not be affected.
†TOPS means technical specification-required orderly plant shutdown, therefore these failures are not considered to be initiating events.
††Loss of power causes motor-operated valves to fail as is.

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TABLE C.1-4 (continued)

Support System	Failure Mode	Systems Affected	IE Category
1. Electric Power (continued)	Lose All 120V Instrument AC Power	Solid State Protection System	SSPS*
	Lose All Three 125V DC Vital Buses	All Vital AC Bus Breakers	None**
		Reactor Protection System	CO-1a
		Diesel Generators	TOPS
		Instrument Air	†
		Pressurizer PORVs	None
		Auxiliary Feedwater	TOPS
		Charging Pumps	MLD-12
		Safety Injection Pumps	TOPS
		Residual Heat Removal	TOPS
		Containment Spray	TOPS
		Reactor Coolant Pumps	None††
		Main Feedwater Pumps	MLD-8††
		Auxiliary Salt Water**	None
		Component Cooling Water**	None
	Lose One 4.16-kV Vital Bus	Injection Pump	TOPS
		Charging Pump	TOPS
		Residual Heat Removal	TOPS
		Component Cooling Water	TOPS
		Auxiliary Salt Water	TOPS
		Pressurizer PORV Block Valve	None
		Containment Spray	TOPS

*Plant will be tripped by loss of RCPs if more than one instrument AC channel failure occurs. Random failure of more than one passive system is an extremely low frequency event and therefore is not included as a separate initiating event. However, multiple failure of instrument channels due to external causes (e.g., earthquake and 480V switchgear ventilation) are addressed.
**Breakers will be unable to open or close. Energized equipment will continue operating, but standby equipment will not be able to start from control room.
†Failure of 125V DC bus 13 leads to isolation of air supply to loads inside the containment.
††Loss of breaker control power for the equipment. Failure of 125V DC buses 11 or 12 will lead to closure of the feedwater regulator valves.

TABLE C.1-4 (continued)

Support System	Failure Mode	Systems Affected	IE Category
1. Electric Power (continued)		Auxiliary Feedwater	TOPS
		Diesel Generator	TOPS
		VAC Systems	None
		480V Bus	See Systems under Loss of One 480V Bus
	Lose One 480V Vital Bus	VAC System	None
		Containment Fan Cooler	None
		Safety System Valves	None
	Lose One 120V Instrument AC	Solid State Protection System	None
	Lose One 125V DC Vital Bus	Vital AC Buses	None
		Diesel Generators	TOPS
		Instrument Air	*
		Pressurizer PORV	None
		Auxiliary Feedwater	TOPS
		Charging Pump	TOPS
		Safety Injection Pump	TOPS
		Residual Heat Removal	TOPS
		Containment Spray	TOPS
		Reactor Coolant Pump	None**
		Main Feedwater Pump	**
		Auxiliary Salt Water	None
		Component Cooling Water	None
2. Auxiliary Salt Water	Lose Both Trains	Component Cooling Water	MLD-12
	Lose One Train	Component Cooling Water	None

*Failure of 125V DC bus 13 leads to isolation of air supply to loads inside the containment.
**Loss of breaker control power for the equipment. Failure of 125V DC bus 11 or 12 will lead to closure of the feedwater regulator valves.

TABLE C.1-4 (continued)

Support System	Failure Mode	Systems Affected	IE Category
3. Ventilation and Air Conditioning System	Lose Auxiliary Building Ventilation System	Component Cooling Water*	MLD-12
		Charging Pumps*	MLD-12
		Residual Heat Removal*	TOPS
		Safety Injection Pumps*	TOPS
		Containment Spray Pumps*	TOPS
	Lose Fuel Handling Building Ventilation System	Auxiliary Feedwater*	TOPS
	Lose Auxiliary Salt Water Pump Room Ventilation System	Auxiliary Salt Water Pump	MLD-11
	Lose 4-kV Switchgear Rooms Ventilation Systems	4.16-kV Vital Buses*	See Systems under Loss of All 4.16-kV Vital Buses
	Lose 480V Switchgear Room Ventilation System	480V Vital Buses	See Systems under Loss of All 480V Vital Buses
		125V DC Buses	See Systems under Loss of All 125V DC Vital Buses
		120V AC Inverters	See Systems under Loss of All 120V Instrument AC Buses**
	Lose Control Room Ventilation System	Solid State Protection System	MLD-22†

*Long-term effects of the indicated equipment were found not to impact the indicated equipment because the resulting room heatup was very slow.
**Loss of 480V switchgear ventilation will overheat the power supplies for the solid state protection system; this is a new initiating event category identified by support system FMEA.
†New initiating event category identified by the support system FMEA.

TABLE C.1-4 (continued)

Support System	Failure Mode	Systems Affected	IE Category
4. Instrument Air	Lose Common Headers	Main Feedwater System	MFS*
		MSIVs	**
		Pressurizer PORVs†	None
		Main Steam Dumps (10%, 35%, and 40%)	None
5. Component Cooling Water	Lose All Headers	Residual Heat Removal	MLD-12
		Safety Injection Pumps	MLD-12
		Charging Pumps	MLD-12
		Containment Fan Coolers	MLD-12
		Reactor Coolant Pump	MLD-12
	Lose One Header	Residual Heat Removal	TOPS
		Safety Injection Pump	TOPS
		Charging Pumps	TOPS
		Containment Fan Coolers	TOPS
		Reactor Coolant Pump	TOPS
6. Service Cooling Water	Lose Service Cooling Water Supply from Common Header	Main Feedwater	SCW††
		Condensate	SCW††
		Turbine Generator	SCW††
		Instrument Air	See Section 4, This Table

*Loss of instrument air leads to the closure of feedwater regulator valves. New initiating event category identified by support system FMEA.
**Loss of instrument air causes a number of air-operated valves and dampers in the following systems to transfer to the safe position: auxiliary saltwater component cooling water, control room ventilation, residual heat removal, containment fan coolers, and containment isolation.
†After the loss of air supply to the MSIVs, the accumulators will only maintain the MSIVs open for a brief period of time.
††New initiating event category identified by support system FMEA. This is grouped with MLD-8, "Complete Loss of Feedwater."

TABLE C.1-5. CROSS-REFERENCE OF DIABLO CANYON STATION HBFT INITIATING EVENTS WITH OTHER PUBLISHED LISTS OF INITIATING EVENTS

Heat Balance Fault Tree							
Major Category	Initiating Event Category	EPRI NP-2230[a]	WASH-1400[b]	Diablo Canyon[c] FSAR	PRA Procedures[d] Guide	Indian Point Probabilistic[e] Safety Study	Seabrook Station[f] Probabilistic Safety Assessment
CI, Increase in Core Heat Generation	1a. Control Rod Ejection 1b. Control Rod Disassembly 1c. Uncontrolled Rod/Bank Withdrawal 3a. CVCS Malfunction--Boron Dilution 3b. Boron Precipitation 4a. Improper Fuel Loading 4b. Burnable Poison/Fuel Geometry Change	(2) Uncontrolled Rod Withdrawal (11) CVCS Malfunction--Boron Dilution	(13) Uncontrolled Rod Withdrawal At Full Power At Startup (13) Boron Dilution by Malfunction in CVCS	• Uncontrolled Rod Cluster Control Assembly Bank Withdrawal at Power (Section 15.2.2) Uncontrolled Boron Dilution (Section 15.2.4) • Single Rod Cluster Control Assembly Withdrawal at Full Power (Section 15.3.5) • Rupture of a Control Rod Drive Mechanism Housing (rod cluster control assembly ejection) (Section 15.4.6) • Inadvertent Loading of a Fuel Assembly into an Improper Position (Section 15.3.3)	• Excessive Control Rod Group Withdrawal • Excessive Control Rod Withdrawal • Control Rod Ejection (due to CRD weld failure) • Inadvertent Deboration	12. Core Power Increase 12.1 Uncontrolled Rod Withdrawal 12.2 Boron Dilution - Chemical Volume Control System Malfunction 12.3 Core Inlet Temperature Drop 12.4 Other Positive Reactivity Addition	15. Core Power Excursion
CD, Decrease in Core Heat Generation	1a. Reactor Trip 1b. Control Rod Drop 1c. Manual Reactor Trip 1d. Plant Control System Malfunction	(6). High or Low Pressure (8) High Pressurizer Pressure (12) Pressurizer, Temperature,	(14) Control Rod Drop	• Rod Cluster Control Assembly Misoperation (Section 15.2.3)	• Control Rod Drop • Control Rod Group Drop • Inadvertent Boration	9.1 Reactor Trip 9.1.1 Control Rod Drive Mechanism Problems and/or Rod Drop 9.1.2 High or Low Pressurizer Pressure 9.1.3 High Pressurizer Level	

a. The EPRI NP-2230 report does not include LOCA events in the initiating event list. Only anticipated transients were considered.
b. Unless otherwise stated, the events were taken from Table 4-9 of Appendix I of the WASH-1400 report (NUREG-75/014).
c. Specific FSAR sections cited in parentheses.
d. Events listed were taken from Table 3-6 of the PRA Procedure Guide (NUREG/CR-2300). Other transient events in the report were referred to EPRI NP-801.
e. Events were taken from Table 1.1.1-1 of Section 1 (Plant Analysis) of the IPPSS.
f. Events listed were taken from Table 5.2-7 of the SSPSA.

TABLE C.1-5 (continued)

Heat Balance Fault Tree							
Major Category	Initiating Event Category	EPRI NP-2230[a]	WASH-1400[b]	Diablo Canyon[c] FSAR	PRA Procedures[d] Guide	Indian Point Probabilistic[e] Safety Study	Seabrook Station[f] Probabilistic Safety Assessment
CD (continued)	3a. CVCS Malfunction - Increased Boration	(36) Pressurizer Spray Failure (39) Automatic Trip - No Transient (40) Manual Trip Due to False Signals (38) Spurious Trips - Cause Unknown				9.1.4 Spurious Automatic Trip--No Transient Condition 9.1.5 Automatic/Manual Trip - Operator Error 9.1.6 Manual Trip Due to False Signal 9.1.8 Spurious Trip--Cause Unknown 9.1.9 Primary System Pressure, Temperature, Power Imbalance 9.1.10 Other Reactor Trips 9.2 Reactor Trip Due to Loss of Component Cooling Water 9.3 Reactor Trip Due to Loss of DC or AC Power (Note: loss of DC or AC power was assessed to be precursors to a decrease in RCS heat removalfrom core; i.e., category CRD)	7. Reactor Trip
CRI, Increase in RCS Heat Removal from Core	1a. Startup of Idle RCS Pump 1b. Increase of RCS Pump Speed 3a. CVCS Malfunction--Increased Charging Over Over Letdown 3b. Spurious Safety Injection 4a. Increase in Pressurizer Heater Output	(13) Startup of Inactive Coolant Pump (9) Inadvertent Safety Injection Signal	(2) Startup of Inactive Coolant Loop (2) Spurious Signals from SICS	• Startup of an Inactive Reactor Coolant Loop (Section 15.2.6) • Spurious Operation of the Safety Injection System at Power (Section 15.2.14)	• Pressurizer Heater Fails On	9.1.7 Spurious Safety Injection	20. Inadvertent Safety Injection

a. The EPRI NP-2230 report does not include LOCA events in the initiating event list. Only anticipated transients were considered.
b. Unless otherwise stated, the events were taken from Table 4-9 of Appendix I of the WASH-1400 report (NUREG-75/014).
c. Specific FSAR sections cited in parentheses.
d. Events listed were taken from Table 3-6 of the PRA Procedure Guide (NUREG/CR-2300). Other transient events in the report were referred to EPRI NP-801.
e. Events were taken from Table 1.1.1-1 of Section 1 (Plant Analysis) of the IPPSS.
f. Events listed were taken from Table 5.2-7 of the SSPSA.

TABLE C.1-5 (continued)

Heat Balance Fault Tree							
Major Category	Initiating Event Category	EPRI NP-2230[a]	WASH-1400[b]	Diablo Canyon[c] FSAR	PRA Procedures[d] Guide	Indian Point Probabilistic[e] Safety Study	Seabrook Station[f] Probabilistic Safety Assessment
CRD, Decrease in RCS Heat Removal From Core	1a. RC Pump(s) Trip 1b. RC Pump Shaft Seizure 1c. RCP Shaft Break 1d. Reduction in RCP Pump Speed 1e. RC Loop Flow Blockage 1f. Core Blockage/Boron Precipitation 3a. CVCS Decrease in Charging Over Letdown 4a. Decrease in Pressurizer Heaters 4b. Pressurizer Spray Actuation 5a. Small LOCA Inside the Containment 5b. Medium LOCA Inside the Containment 5c. Large LOCA Inside the Containment	(1) Loss of RCS Flow (one loop) (14) Total Loss of RCS Flow (26) Steam Generator Leakage (37) Loss of Power to Necessary Plant Systems (31) Loss of Component Cooling (32) Loss of Service Water System	(18) Loss of RCS Coolant Flow (main RCS circulating pump malfunctions) • Small LOCA - S2 (1/2 to 2 inches) (Section I-4.1.3) • Small LOCA - S1 (2 to 6 inches (Section I-4.1.2) • Large LOCA (> 6 inches) (Section I-4.1.1) • Reactor Vessel Rupture (Section I-4.1.4) • Steam Generator Ruptures (Section I-4.1.5) (17) Accidental Opening of Pressurizer Safety or Relief Valves	• Partial Loss of Forced Reactor Coolant Flow (Section 15.2.5) • Complete Loss of Forced Reactor Coolant Flow (Section 15.3.4) • Single Reactor Coolant Pump Locked Rotor (Section 15.4.4) • Loss of Reactor Coolant from Small Ruptured Pipes or from Cracks in Large Pipes that Actuate Emergency Core Cooling System (Section 15.3.1) • Major Reactor Coolant System Pipe Ruptures (LOCA) (Section 15.4.1)	• RCP Trip (due to low flow indication - real or spurious, etc.) • RCP Shaft Seizure/Break • Core Flow Blockage • Pressurizer Spray Fails On • Small RCS Pipe Breaks • RCS Seal Failure • CRDM Seal Leakage • Inadvertent PSV Opening	1. Large Loss of Coolant Accident (blowdown in range of 2 to to 6-inch pipe rupture) 1.1 Pipe Failures 1.2 Valve Failures 1.3 Vessel Failures 1.4 Other Large LOCAs 2. Medium Loss of Coolant Accident (blowdown less than 2" - 6" inch pipe rupture) 2.1 Pipe Failures 2.2 Pressurizer Safety and 2.3 Other Valve Failures 2.4 Other Medium LOCAs 3. Small Loss of Coolant Accident (blowdown less than 2-inch pipe rupture) 3.1 Pipe Failure 3.2 Pressurizer Relief Valve or Safety Valve Failure 3.3 Other Valve Failures 3.4 Control Rod Drive Mechanism 3.5 Reactor Coolant Pump Seal Failure (four or less) 3.6 Other Small LOCAs 4. Leakage to Secondary Coolant 4.1 Single Steam Generator Tube Rupture 4.2 Other Steam Generator Leaks	16. Loss of Primary Flow 2. Large LOCA 3. Medium LOCA 4. Small LOCA

a. The EPRI NP-2230 report does not include LOCA events in the initiating event list. Only anticipated transients were considered.
b. Unless otherwise stated, the events were taken from Table 4-9 of Appendix I of the WASH-1400 report (NUREG-75/014).
c. Specific FSAR sections cited in parentheses.
d. Events listed were taken from Table 3-6 of the PRA Procedure Guide (NUREG/CR-2300). Other transient events in the report were referred to EPRI NP-801.
e. Events were taken from Table 1.1.1-1 of Section 1 (Plant Analysis) of the IPPSS.
f. Events listed were taken from Table 5.2-7 of the SSPSA.

TABLE C.1-5 (continued)

Heat Balance Fault Tree		EPRI NP-2230[a]	WASH-1400[b]	Diablo Canyon[c] FSAR	PRA Procedures[d] Guide	Indian Point Probabilistic[e] Safety Study	Seabrook Station[f] Probabilistic Safety Assessment
Major Category	Initiating Event Category						
CRD (continued)	5d. Isolable LOCA Inside the Containment 5e. Interfacing Systems LOCA 5f. Steam Generator Tube Rupture 5g. Excessive LOCA		• RCS Ruptures into Interfacing Systems (Section I-4.1.6) (9) Loss of AC Power Incoming from Offsite Network • Abrupt Seizure of All Main RCS Recirculation Pumps	• Steam Generator Tube Rupture (Section 15.4.3) • Loss of Offsite Power to the Station Auxiliaries (station blackout) (Section 15.2.9)	• Medium RCS Pipe Breaks • Large RCS Pipe Breaks • Reactor Vessel Rupture • Inadvertent PORV Opening • Letdown or Sample Line Break • Steam Generator Tube Leak	5. Loss of Reactor Coolant Flow 5.1 Loss of Reactor Coolant Flow in One Loop 5.2 Loss of Reactor Coolant Flow in All Loops 5.3 Other Losses of Reactor Coolant Flow	5. Interfacing Systems LOCA 6. Steam Generator Tube Rupture 1. Excessive LOCA
RSI, Increase in RCS Heat Transfer to SCS	4a. Freeing of Steam Generator Tube Blockage						
RSD, Decrease in RCS Heat Transfer to SCS	4a. Steam Generator Tube Blockage			• Accidental Depressurization of the Reactor Coolant System (Section 15.2.11)			

a. The EPRI NP-2230 report does not include LOCA events in the initiating event list. Only anticipated transients were considered.
b. Unless otherwise stated, the events were taken from Table 4-9 of Appendix I of the WASH-1400 report (NUREG-75/014).
c. Specific FSAR sections cited in parentheses.
d. Events listed were taken from Table 3-6 of the PRA Procedure Guide (NUREG/CR-2300). Other transient events in the report were referred to EPRI NP-801.
e. Events were taken from Table 1.1.1-1 of Section 1 (Plant Analysis) of the IPPSS.
f. Events listed were taken from Table 5.2-7 of the SSPSA.

TABLE C.1-5 (continued)

Heat Balance Fault Tree		EPRI NP-2230[a]	WASH-1400[b]	Diablo Canyon[c] FSAR	PRA Procedures[d] Guide	Indian Point Probabilistic[e] Safety Study	Seabrook Station[f] Probabilistic Safety Assessment
Major Category	Initiating Event Category						
SRI, Increase in SCS Heat Removal from RCS	1a. Feedwater Heater Bypass Valve Opens 1b. Loss of Steam to Feedwater Heater 2a. Increase in Feed Pump Speed 2b. Start of Idle Feed Pump 2c. Spurious EFW Actuation 2d. Start of Idle Heater Drain Pump 3a. Atmospheric Relief Valve Opens 3b. MSSV Opens 3c. TCV Opens 3d. TBV Opens 3e. Steam Line Break Inside Containment 3f. Steam Line Break Outside Containment 4g. Freeing of Steam Generator Crud Blockage	(19) Increase in Feedwater Flow (one loop) (20) Increase in Feedwater Flow (all loops) (21) Feedwater Flow Instability--Operator Error (excessive feedwater flow occurs) (22) Feedwater Flow Instability--Miscellaneous Mechanical Causes (excessive feedwater flow results) (29) Sudden Opening of Steam Relief Valves	(11) Increase in Main Feedwater Flow; Malfunctions in Feedwater Flow Control (15) Inadvertent Opening of Steam Generator Power-Operated Relief Valves (~10% sudden load demand) (12) Malfunctions of Control Resulting in Inadvertent Opening of All Turbine Steam Bypass Valves (~40% sudden load demand) • Rupture of Lines in Main Steam System	• Excessive Heat Removal Due to Feedwater System Malfunction (Section 15.2.10) • Accidental Depressurization of the Reactor Coolant System (Section 15.2.12) • Excessive Load Increase Incident (Section 15.2.11) • Accidental Depressurization of the Main Steam System (Section 15.2.13) • Minor Secondary System Pipe Breaks (Section 15.3.2) • Major Secondary System Pipe Rupture (Section 15.4.2)	• Turbine Control Valve Open • Inadvertent Opening of Turbine Bypass Valve (TBV) • Inadvertent Opening of All TBVs	10. Steam Release Inside Containment 10.1 Steam Pipe Rupture Inside Containment 10.2 Feedwater Pipe Rupture Inside Containment 10.3 Steam Relief Valve or Safety Valves Open Inadvertently (included with inside containment group for functional reasons--leak upstream of MSIVS) 10.4 Other Steam Releases Inside Containment 11. Steam Release (demand) Outside Containment 11.1 Steam Pipe Rupture Outside Containment 11.2 Throttle-Valve Opening/Electrohydraulic Control Problems 11.3 Steam Dump Valves Failing Open 11.4 Other Steam Releases Outside Containment 8.1.2 Increase in Feedwater Flow in One or More Steam Generators	11. Excessive Feedwater Flow 19. MSRV Opening 17. Steam Line Break - Inside Containment 18. Steam Line Break - Outside Containment

a. The EPRI NP-2230 report does not include LOCA events in the initiating event list. Only anticipated transients were considered.
b. Unless otherwise stated, the events were taken from Table 4-9 of Appendix I of the WASH-1400 report (NUREG-75/014).
c. Specific FSAR sections cited in parentheses.
d. Events listed were taken from Table 3-6 of the PRA Procedure Guide (NUREG/CR-2300). Other transient events in the report were referred to EPRI NP-801.
e. Events were taken from Table 1.1.1-1 of Section 1 (Plant Analysis) of the IPPSS.
f. Events listed were taken from Table 5.2-7 of the SSPSA.

TABLE C.1-5 (continued)

Heat Balance Fault Tree		EPRI NP-2230[a]	WASH-1400[b]	Diablo Canyon[c] FSAR	PRA Procedures[d] Guide	Indian Point Probabilistic[e] Safety Study	Seabrook Station[f] Probabilistic Safety Assessment
Major Category	Initiating Event Category						
SRD, Decrease in SCS Heat Removal from RCS	1a. Freeing of Feedwater Heater Steam Side Blockage 2a. Reduction Feedwater Pump Speed 2b. Feedwater Pump(s) Trip 2c. Condensate Pump(s) Trip 2d. Feedwater Heater Drain Pump(s) Trip 2e. Feedwater Pipe Leak or Rupture 2f. Feedwater Isolation Valve Closure 3a. Closure of One or More MSIVs 3b. Closure of TCVs 3c. Closure of TSVs 4a. Steam Generator/ SCS Flow Blockage	(15) Loss or Reduction in Feedwater Flow (one loop) (16) Total Loss of Feedwater Flow (all loops) (17) Full or Partial Closure of MSIV (one loop) (18) Closure of All MSIVs (21) Feedwater Flow Instability - Operator Error (insufficient feedwater flow occurs)	(8) Loss of Condensate Pumps (5) Loss of Main Station Generator with Failure to Relay Auxiliary Loads (e.g., main feedwater pumps, condensate pumps) to AC Power Incoming from Offsite Network (4) Inadvertent Closure of Main Steam Line Isolation Valves	• Loss of Normal Feedwater (Section 15.2.8) • Loss of External Electrical Load and/or Turbine Trip (Section 15.2.7)	(d)	6. Loss of Feedwater Flow 6.1 Feedwater Pipe Rupture Outside Containment 6.2 Loss/Reduction of Feedwater Flow in One Steam Generator 6.3 Loss of Feedwater Flow in All Steam Generators 6.4 Feedwater Flow Instability Operator Error 6.5 Feedwater Flow Instability Mechanical Causes 6.6 Loss of One Condensate Pump 6.7 Loss of All Condensate Pumps 6.8 Condenser Leakage 6.9 Other Secondary Leakage 7. Partial Loss of Steam Flow 7.1 Full Closure of MSIV 7.2 Partial Closure of MSIV 7.3 Other Losses of Steam Flow	10. Partial Feedwater Flow Loss 9. Total Main Feedwater Loss 8. Turbine Trip 13. Closure of One MSIV 14. Closure of All MSIVs

a. The EPRI NP-2230 report does not include LOCA events in the initiating event list. Only anticipated transients were considered.
b. Unless otherwise stated, the events were taken from Table 4-9 of Appendix I of the WASH-1400 report (NUREG-75/014).
c. Specific FSAR sections cited in parentheses.
d. Events listed were taken from Table 3-6 of the PRA Procedure Guide (NUREG/CR-2300). Other transient events in the report were referred to EPRI NP-801.
e. Events were taken from Table 1.1.1-1 of Section 1 (Plant Analysis) of the IPPSS.
f. Events listed were taken from Table 5.2-7 of the SSPSA.

TABLE C.1-5 (continued)

Heat Balance Fault Tree							
Major Category	Initiating Event Category	EPRI NP-2230[a]	WASH-1400[b]	Diablo Canyon[c] FSAR	PRA Procedures[d] Guide	Indian Point Probabilistic[e] Safety Study	Seabrook Station[f] Probabilistic Safety Assessment
SRD		(22) Feedwater Flow Instability - Operator Error (insufficient feedwater flow results) (23) Loss of Condensate Pumps (one loop) (24) Loss of Condensate Pumps (all loops) (28) Miscellaneous Leakage in Secondary System (other than condenser leakage)				8.1.1 Closure of All MSIVs	
SPI, Increase in Energy Transfer to Plant Output	2b. Enhancement of Condenser Vacuum						
SPD, Decrease in Energy Transfer to Plant Output	2a. Reduction of Condenser Vacuum 2b. Loss of Condenser Vacuum	(25) Loss of Condenser Vacuum (27) Condenser Leakage	(3) Loss of Condenser Vacuum		(d)	8.1.3 Loss of Condenser Vacuum	12. Loss of Condenser Vacuum

a. The EPRI NP-2230 report does not include LOCA events in the initiating event list. Only anticipated transients were considered.
b. Unless otherwise stated, the events were taken from Table 4-9 of Appendix I of the WASH-1400 report (NUREG-75/014).
c. Specific FSAR sections cited in parentheses.
d. Events listed were taken from Table 3-6 of the PRA Procedure Guide (NUREG/CR-2300). Other transient events in the report were referred to EPRI NP-801.
e. Events were taken from Table 1.1.1-1 of Section 1 (Plant Analysis) of the IPPSS.
f. Events listed were taken from Table 5.2-7 of the SSPSA.

TABLE C.1-5 (continued)

Heat Balance Fault Tree							
Major Category	Initiating Event Category	EPRI NP-2230[a]	WASH-1400[b]	Diablo Canyon[c] FSAR	PRA Procedures[d] Guide	Indian Point Probabilistic[e] Safety Study	Seabrook Station[f] Probabilistic Safety Assessment
EI, Increase in Electrical Output	1a. Increase in Electricity Demand 1b. Grid Instability 1c. Generator Load/Limiter Fault						
ED, Decrease in Electrical Output	1a. Decrease in Electricity Demand 1b. Loss of Generator Load 1c. Generator Failure 1d. Grid Instabilities 2a. Key Breaker Fails Open 3a. Generator Trip 4a. Turbine Trip 4b. Turbine Blade/Rotor Failure	(33) Turbine Trip, Throttle Valve Closure, EHC Problems (34) Generator Trip or Generator Caused Faults	(1) Turbine Trip	Turbine Trip (Section 15.2.7)	(d)	8. Turbine Trip 8.1 Turbine Trip (general) 8.1.5 Throttle-Valve Closure/Electrohydraulic Control 8.1.6 Generator Trip or Generator-Caused Faults 8.1.7 Turbine Trip Due to Overspeed 8.1.8 Other Turbine Trips 8.2 Turbine Trip Due to Loss of Offsite Power 8.3 Turbine Trip Due to Loss of Service Water (Note: loss of offsite power and loss of service water were assessed to be precursors to a decrease in RCS heat removal from core; i.e., category CRD).	8. Turbine Trip
TI, Increase in Thermal Output	2a. Startup of Idle Circulatin Water Pump 2b. Increase in Circulating Water Pump Speed						

a. The EPRI NP-2230 report does not include LOCA events in the initiating event list. Only anticipated transients were considered.
b. Unless otherwise stated, the events were taken from Table 4-9 of Appendix I of the WASH-1400 report (NUREG-75/014).
c. Specific FSAR sections cited in parentheses.
d. Events listed were taken from Table 3-6 of the PRA Procedure Guide (NUREG/CR-2300). Other transient events in the report were referred to EPRI NP-801.
e. Events were taken from Table 1.1.1-1 of Section 1 (Plant Analysis) of the IPPSS.
f. Events listed were taken from Table 5.2-7 of the SSPSA.

TABLE C.1-5 (continued)

Heat Balance Fault Tree		EPRI NP-2230[a]	WASH-1400[b]	Diablo Canyon[c] FSAR	PRA Procedures[d] Guide	Indian Point Probabilistic[e] Safety Study	Seabrook Station[f] Probabilistic Safety Assessment
Major Category	Initiating Event Category						
TI (continued)	2c. Freeing of Circulating Water Tunnel Blockage 3a. Natural Fluctuations in Ocean Temperature 4a. Freeing of Plugged/ Blocked Condenser Tubes						
TD, Decrease in Thermal Output	2a. Trip of One or More Circulating Water Pumps 2b. Decrease in Circulating Water Pump Speed 2c. Debris Blockage of Circulating Water Intake 2d. Leak or of Rupture Circulating Water Intake/ Condenser Expansion Joint 3a. Natural Fluctuations in Ocean Temperature 3b. Demusseling Operations 4a. Plugging/ Blockage of Condenser Tubes	(30) Loss of Circulating Water	(6) Loss of Main Circulating Water Pumps for Condenser Cooling		(d)	8.1.4 Loss of Circulating Water	

a. The EPRI NP-2230 report does not include LOCA events in the initiating event list. Only anticipated transients were considered.
b. Unless otherwise stated, the events were taken from Table 4-9 of Appendix I of the WASH-1400 report (NUREG-75/014).
c. Specific FSAR sections cited in parentheses.
d. Events listed were taken from Table 3-6 of the PRA Procedure Guide (NUREG/CR-2300). Other transient events in the report were referred to EPRI NP-801.
e. Events were taken from Table 1.1.1-1 of Section 1 (Plant Analysis) of the IPPSS.
f. Events listed were taken from Table 5.2-7 of the SSPSA.

TABLE C.1-5 (continued)

Heat Balance Fault Tree		EPRI NP-2230[a]	WASH-1400[b]	Diablo Canyon[c] FSAR	PRA Procedures[d] Guide	Indian Point Probabilistic[e] Safety Study	Seabrook Station[f] Probabilistic Safety Assessment
Major Category	Initiating Event Category						
Support System Faults (these are broken down more finely in Table 5.2-4)	Electric Power System Faults Auxiliary Salt Water System Faults Component Cooling Water System Faults Service Cooling Water System Faults Instrument Air Faults						

a. The EPRI NP-2230 report does not include LOCA events in the initiating event list. Only anticipated transients were considered.
b. Unless otherwise stated, the events were taken from Table 4-9 of Appendix I of the WASH-1400 report (NUREG-75/014).
c. Specific FSAR sections cited in parentheses.
d. Events listed were taken from Table 3-6 of the PRA Procedure Guide (NUREG/CR-2300). Other transient events in the report were referred to EPRI NP-801.
e. Events were taken from Table 1.1.1-1 of Section 1 (Plant Analysis) of the IPPSS.
f. Events listed were taken from Table 5.2-7 of the SSPSA.

TABLE C.1-6. FINAL INITIATING EVENT CATEGORIES WITH CROSS REFERENCE TO HEAT BALANCE FAULT TREE, MASTER LOGIC DIAGRAM CATEGORIES, AND FMEA

Group	Final Initiating Event Categories for Risk Quantification	Heat Balance Fault Tree Categories	Master Logic Diagram Categories	Identified by FMEA
LOCAs				
	1 - Excessive LOCA	CRD-5g	MLD-1	--
	2 - Large LOCA	CRD-5c	MLD-2	--
	3 - Medium LOCA	CRD-5b	MLD-3	--
	4 - Small LOCA - Nonisolable	CRD-5a	MLD-4	--
	5 - Small LOCA - Isolable	CRD-5d	MLD-4	--
	6 - Interfacing Systems LOCA	CRD-5e	MLD-5	--
	7 - Steam Generator Tube Rupture	CRD-5f	MLD-6	--
Transients				
	8 - Reactor Trip	CD-1a, CD-1c	MLD-13	Yes
	9 - Turbine Trip	SRD-3b, ED-1c, ED-2a, ED-3a, ED-4a, ED-1b, ED-4b	MLD-9	Yes
	10 - Loss of Condenser Vacuum	SPD-2b, TD-2a, TD-2b, TD-2c	MLD-7	--
	11 - Closure of All MSIVs	SRD-3a	MLD-7	--
	12 - Steam Line Break - Inside Containment	SRI-3e	MLD-14	--
	13 - Steam Line Break - Outside Containment	SRI-3f	MLD-15	--
	14 - Inadvertent Safety Injection	CRI-3b	MLD-16	--
	15 - MSRV Opening	SRI-3a	MLD-15	--

TABLE C.1-6 (continued)

Group	Final Initiating Event Categories for Risk Quantification	Heat Balance Fault Tree Categories	Master Logic Diagram Categories	Identified by FMEA
	16 - Total Main Feedwater Loss	SRD-2b, SRD-2c, SRD-2d, SRD-2e, SRD-2f, SRD-2h	MLD-8	Yes
	17 - Partial Feedwater Flow	SRD-2a, SRD-4a	MLD-7	--
	18 - Excessive Feedwater Flow	SRI-2a, SRI-2b, SRI-2c SRI-2d	MLD-17	--
	19 - Closure of One MSIV	SRD-3a	MLD-7	--
	20 - Core Power Excursion	CI-1a, CI-1b, CI-1c, SRI-1a, SRI-1b, CRI-1a	MLD-23	--
	21 - Loss of Primary Flow	CRD-1a, CRD-1b, CRD-1c, CRD-1d, CRD-1e	MLD-7	Yes
Common Cause Initiating Events				
Support System Faults	22 - Total Loss of Offsite Power		MLD-10	Yes
	23 - Total Loss of One DC Bus		MLD-18	Yes
	24 - Total Loss of Auxiliary Salt Water		MLD-11	Yes
	25 - Total Loss of Component Cooling Water		MLD-12	Yes
	26 - Loss of 480V Switchgear Ventilation		MLD-22	Yes
	27 - Loss of Control Room Ventilation		MLD-22	Yes
Seismic, Fires, Flooding, Turbine Missiles, Tornado Missiles, Aircraft Crash, and Other Events*				

*These initiators are investigated in detail separately. See Appendix F, "External Events and Implant Hazards."

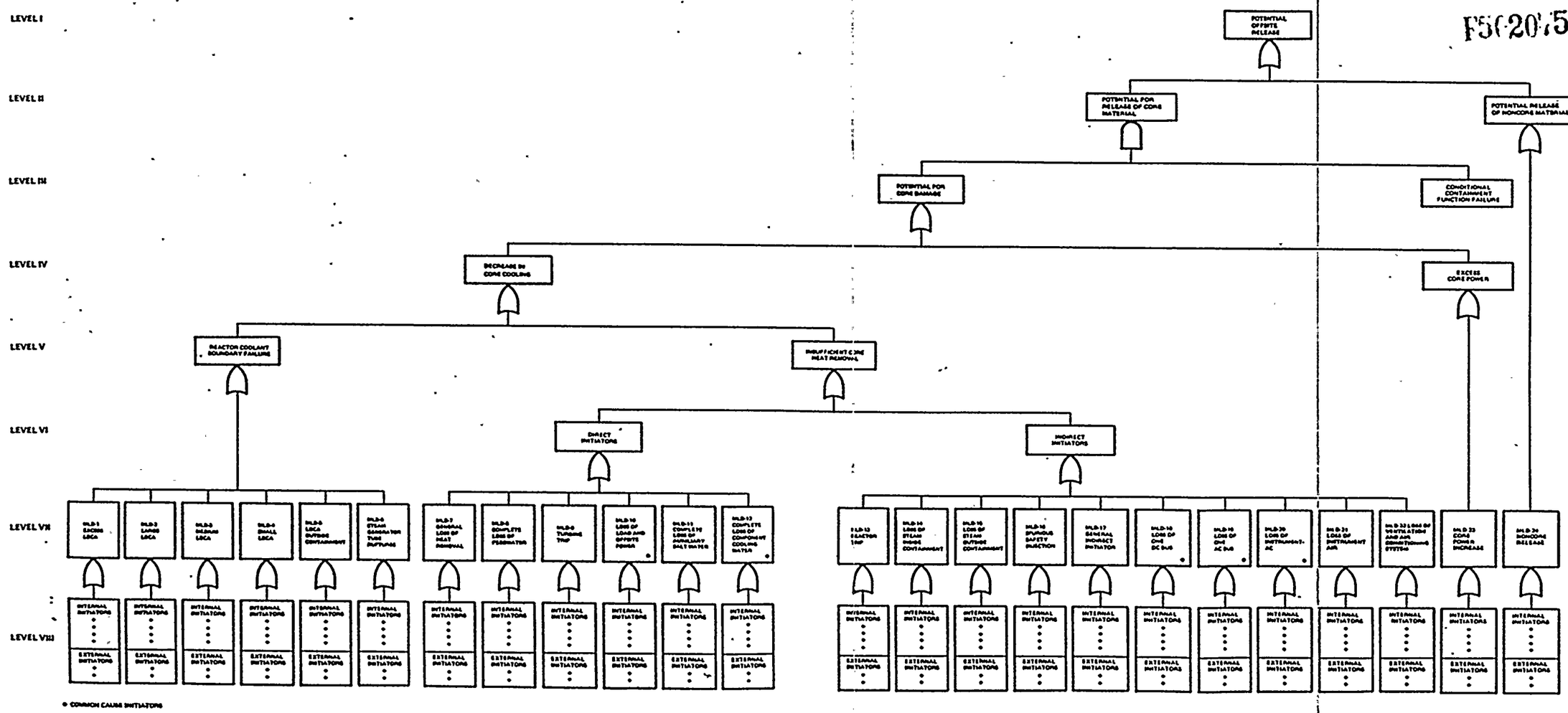

FIGURE C.1-1. MASTER LOGIC DIAGRAM

F5(20)5

TI APERTURE CARD

Also Available On Aperture Card

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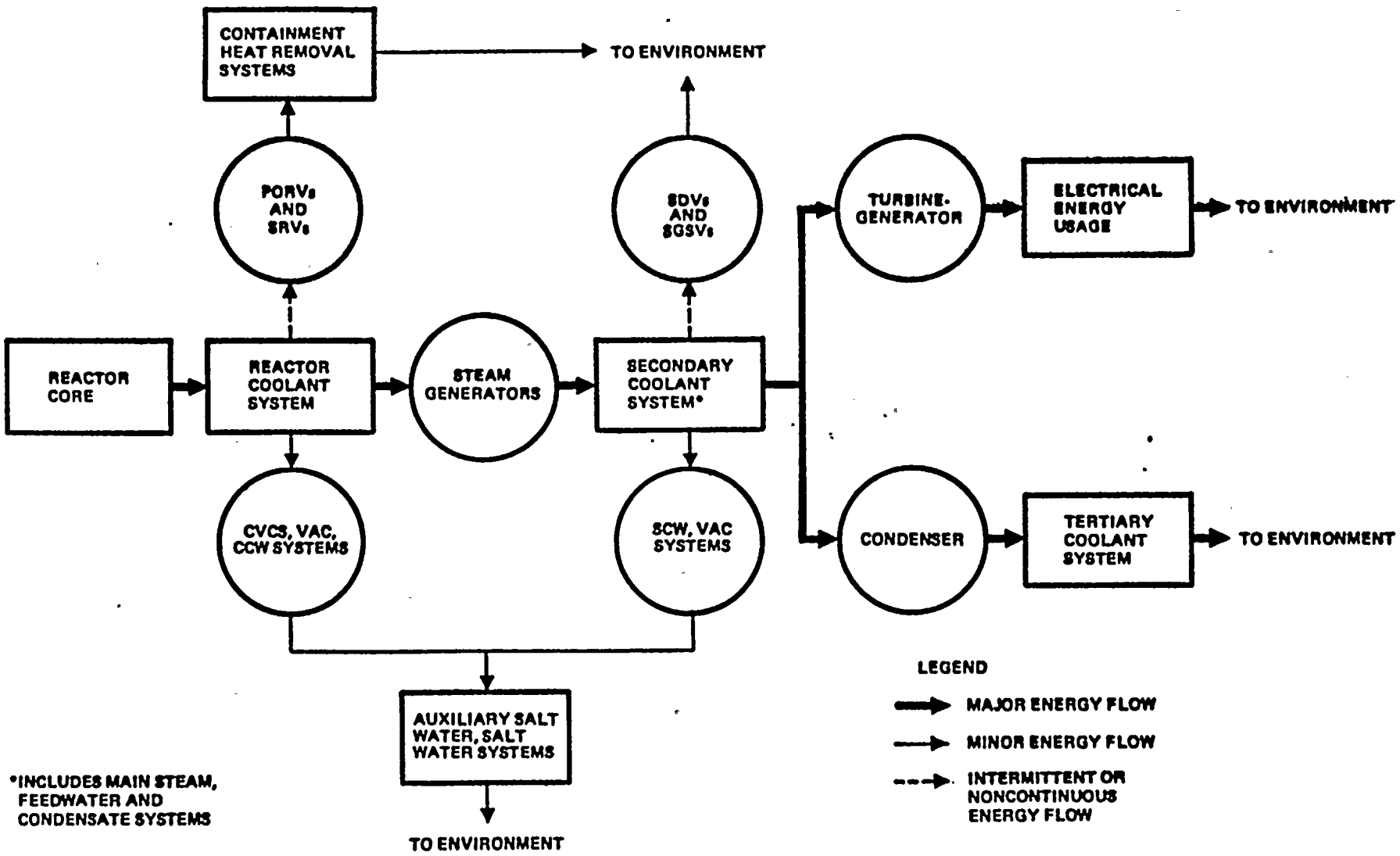


FIGURE C.1-2. PRINCIPAL HEAT TRANSPORT PATHS THAT DIRECTLY INFLUENCE REACTOR CORE THERMAL EQUILIBRIUM

F5(20/5

F5C2075

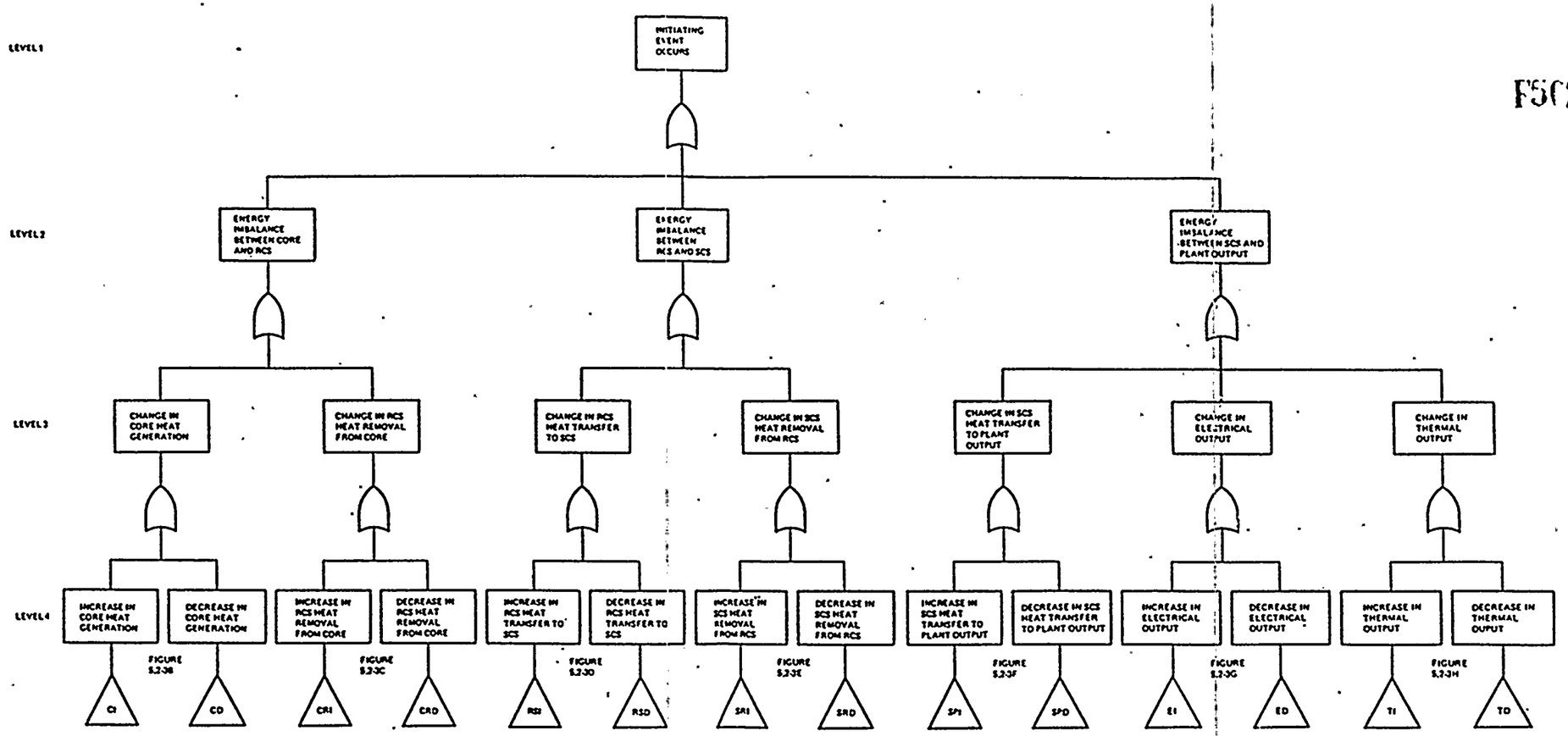
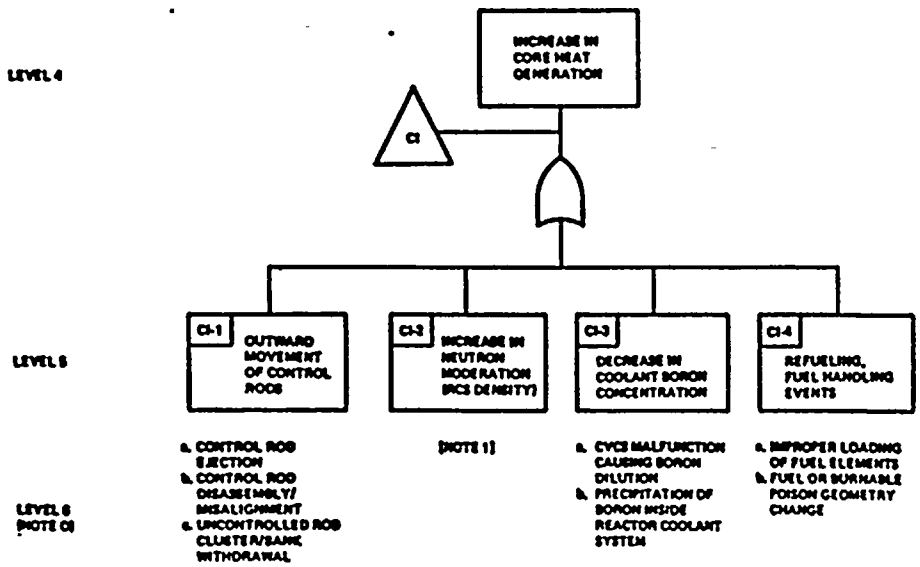
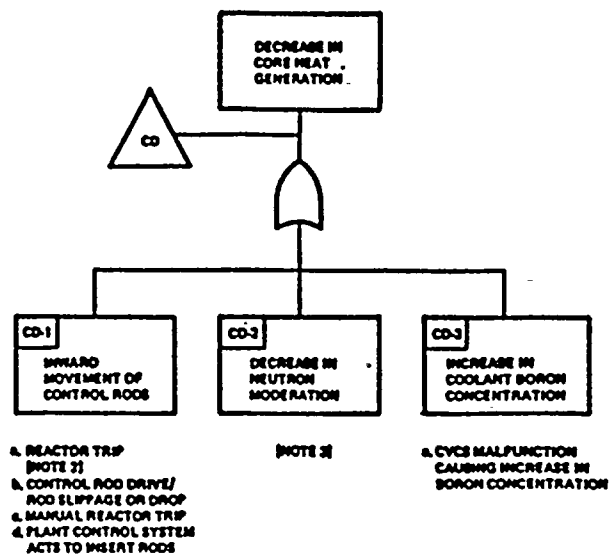


FIGURE C.1-3a. HEAT BALANCE FAULT TREE DEVELOPED TO LEVEL INDICATING CHANGES IN PRINCIPAL ENERGY TRANSFER PROCESSES

8803160246-0.2





NOTES:

A. COMPLETENESS CANNOT BE ASSURED AT LEVEL 6: ONLY THOSE EVENTS IDENTIFIED AND JUDGED SIGNIFICANT ARE LISTED.

1. NO PRIMARY INITIATORS IDENTIFIED. THIS CAN ACCOUNT AS A SECONDARY EFFECT OF PRIMARY.

   INITIATORS LISTED UNDER CAI INCREASE IN RCS HEAT REMOVAL FROM CORE.

2. INCLUDES SPURIOUS REACTOR TRIPS AS WELL AS THOSE THAT OCCUR AS A SECONDARY RESULT OF OTHER CAUSES THAT RESULT IN REACTOR TRIP INPUT PARAMETERS EXCEEDING THEIR TRIP SETPOINTS.

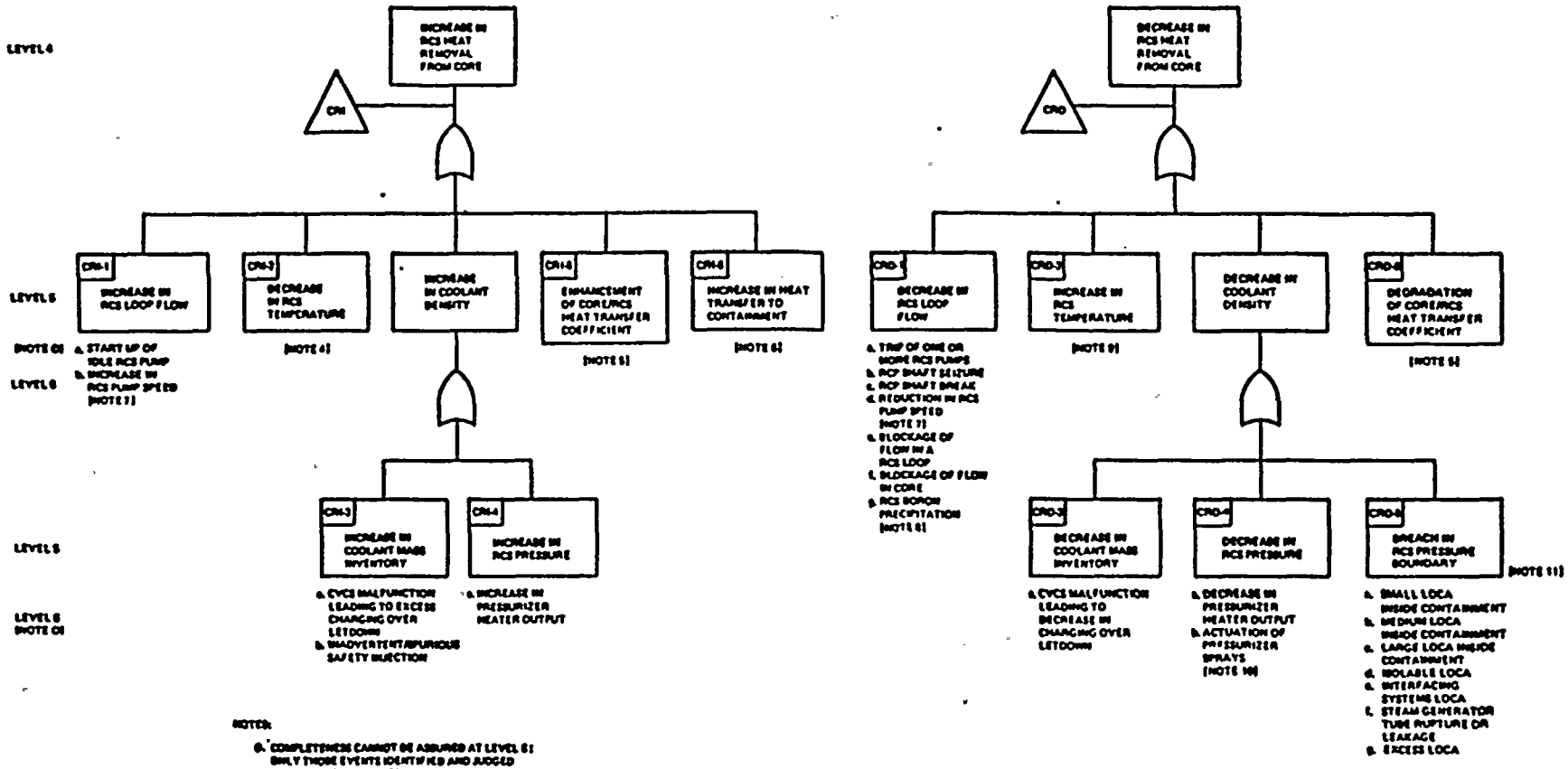
3. ALTHOUGH THIS CAN OCCUR AS A SECONDARY EFFECT OF OTHER PRIMARY INITIATORS IDENTIFIED ELSEWHERE IN THIS FAULT TREE,

   SUCH AS ANY INITIATORS IDENTIFIED UNDER CAO DECREASE IN

   SECONDARY HEAT REMOVAL) OR OTHERS THAT CAUSE INCREASE IN RCS TEMPERATURE OR DECREASE IN RCS PRESSURE. THERE ARE NO PRIMARY INITIATORS IN THIS CATEGORY.

FIGURE C.1-3b. HEAT BALANCE FAULT SUBTREES FOR INCREASE AND DECREASE IN CORE HEAT GENERATION

F502075



NOTES:

1. COMPLETENESS CANNOT BE ASSURED AT LEVEL 6! ONLY THOSE EVENTS IDENTIFIED AND JUDGED SIGNIFICANT ARE LISTED.
4. ALTHOUGH DECREASE IN RCS TEMPERATURE CAN OCCUR AS A SECONDARY EFFECT OF OTHER PRIMARY INITIATOR GROUPS FOUND IN THIS FAULT TREE, SUCH AS THOSE IDENTIFIED UNDER CRI, INCREASE IN RCS HEAT REMOVAL FROM RCS, NO PRIMARY INITIATORS WERE IDENTIFIED FOR THIS CATEGORY.
5. THE ONLY INITIATORS IDENTIFIED IN THESE CATEGORIES, e.g. CHANGES IN NUCLEATE BOILING AND REMOVAL OF FLOW BLOCKAGE, WOULD ONLY OCCUR DURING TRANSIENT CONDITIONS CAUSED BY SOME OTHER INITIATOR SUCH AS LARGE LOCA.
6. ALL INITIATORS THAT WERE IDENTIFIED IN THIS CATEGORY ARE CLASSIFIED AS PRIMARY INITIATORS UNDER CRO, DECREASE IN RCS HEAT REMOVAL FROM CORE.
7. SINCE PUMPS ARE FIXED SPEED, SPEED FLUCTUATIONS WOULD HAVE TO BE CAUSED BY ELECTRICAL PROBLEMS, SUCH AS FREQUENCY/VOLTAGE FLUCTUATIONS.
8. COULD OCCUR DURING SHUTDOWN AT LOW TEMPERATURE, HOWEVER PROCESS IS SELF-LIMITING → RCS TEMPERATURE INCREASE WOULD INCREASE BORON SOLUBILITY AND REDUCE BLOCKAGE.
9. CAN OCCUR AS A SECONDARY EFFECT OF OTHER PRIMARY INITIATORS SUCH AS CRO, DECREASE IN RCS HEAT REMOVAL FROM RCS, BUT NO PRIMARY INITIATORS WERE IDENTIFIED.
10. IN ADDITION TO THESE PRIMARY INITIATORS, SECONDARY INITIATORS OF RCS PRESSURE DECREASE ARE FOUND UNDER CRI, INCREASE IN RCS HEAT REMOVAL FROM RCS.
11. DIFFERENT SIZES AND LOCATIONS BROKEN OUT TO DISTINGUISH BETWEEN DIFFERENT MITIGATING SYSTEMS EFFECTS AND REQUIREMENTS.

FIGURE C.1-3c. HEAT BALANCE FAULT SUBTREES FOR INCREASE AND DECREASE IN RCS HEAT REMOVAL FROM CORE

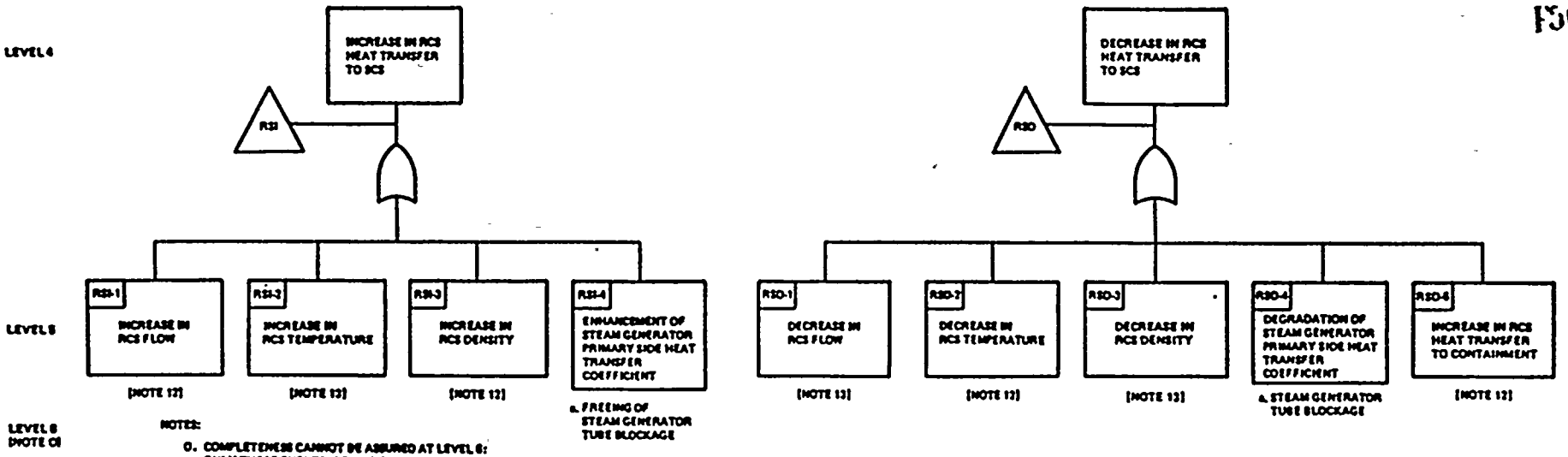


FIGURE C.1-3d. HEAT BALANCE FAULT SUBTREES FOR INCREASE AND DECREASE IN RCS HEAT TRANSFER TO SCS

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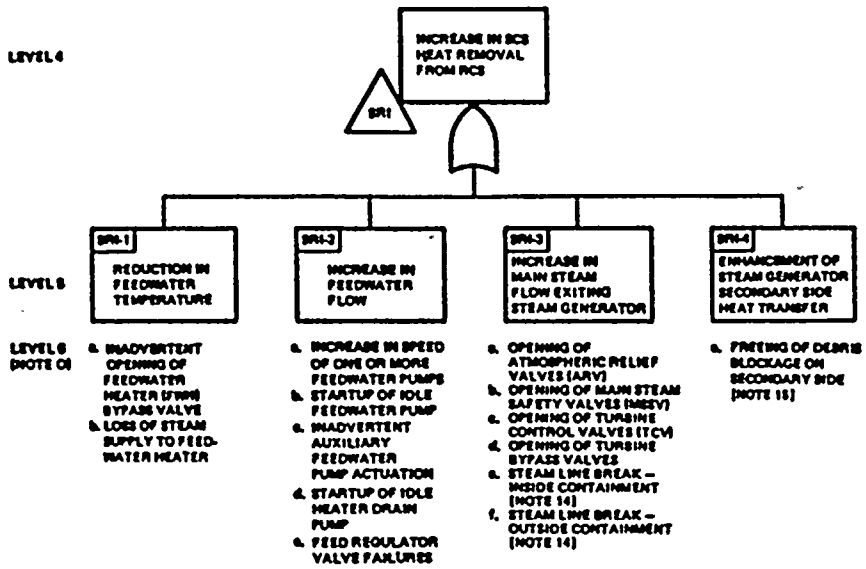
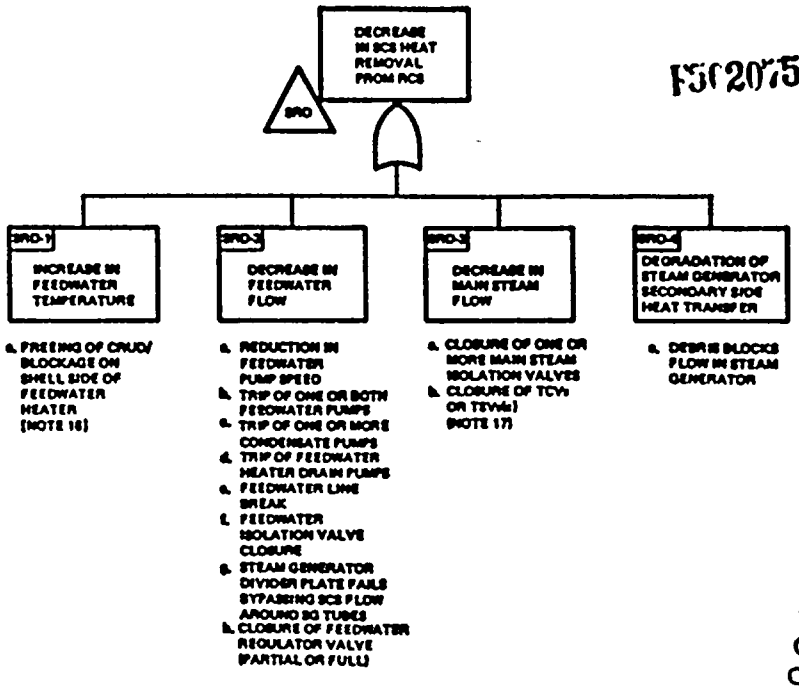


NOTES:

O COMPLETENESS CANNOT BE ASSURED AT LEVEL 6; ONLY THOSE EVENTS IDENTIFIED AND JUDGED SIGNIFICANT ARE LISTED

14. CAN ALSO BE POSTULATED AS A SECONDARY EFFECT OF OTHER PRIMARY INITIATORS SUCH AS THOSE LISTED UNDER EI AND SPI.

15. OTHER INITIATORS CAN BE POSTULATED, SUCH AS ENHANCEMENT OF NUCLEATE BOILING, WHICH WOULD BE SECONDARY TO OTHER PRIMARY INITIATORS IDENTIFIED IN THIS FAULT TREE, e.g. INITIATORS UNDER PSI, INCREASE HEAT TRANSFER TO SCS.



16. CAN ALSO OCCUR AS A SECONDARY EFFECT OF OTHER PRIMARY INITIATORS SUCH AS THOSE UNDER TO, DECREASE IN THERMAL OUTPUT: e.g. REDUCTION IN CIRCULATING WATER FLOW TO CONDENSER.

17. CAN ALSO OCCUR AS A SECONDARY EFFECT OF OTHER PRIMARY INITIATORS SUCH AS THOSE UNDER EO AND SPO.

FIGURE C.1-3a. HEAT BALANCE FAULT SUBTREES FOR INCREASE AND DECREASE IN SCS HEAT REMOVAL FROM RCS

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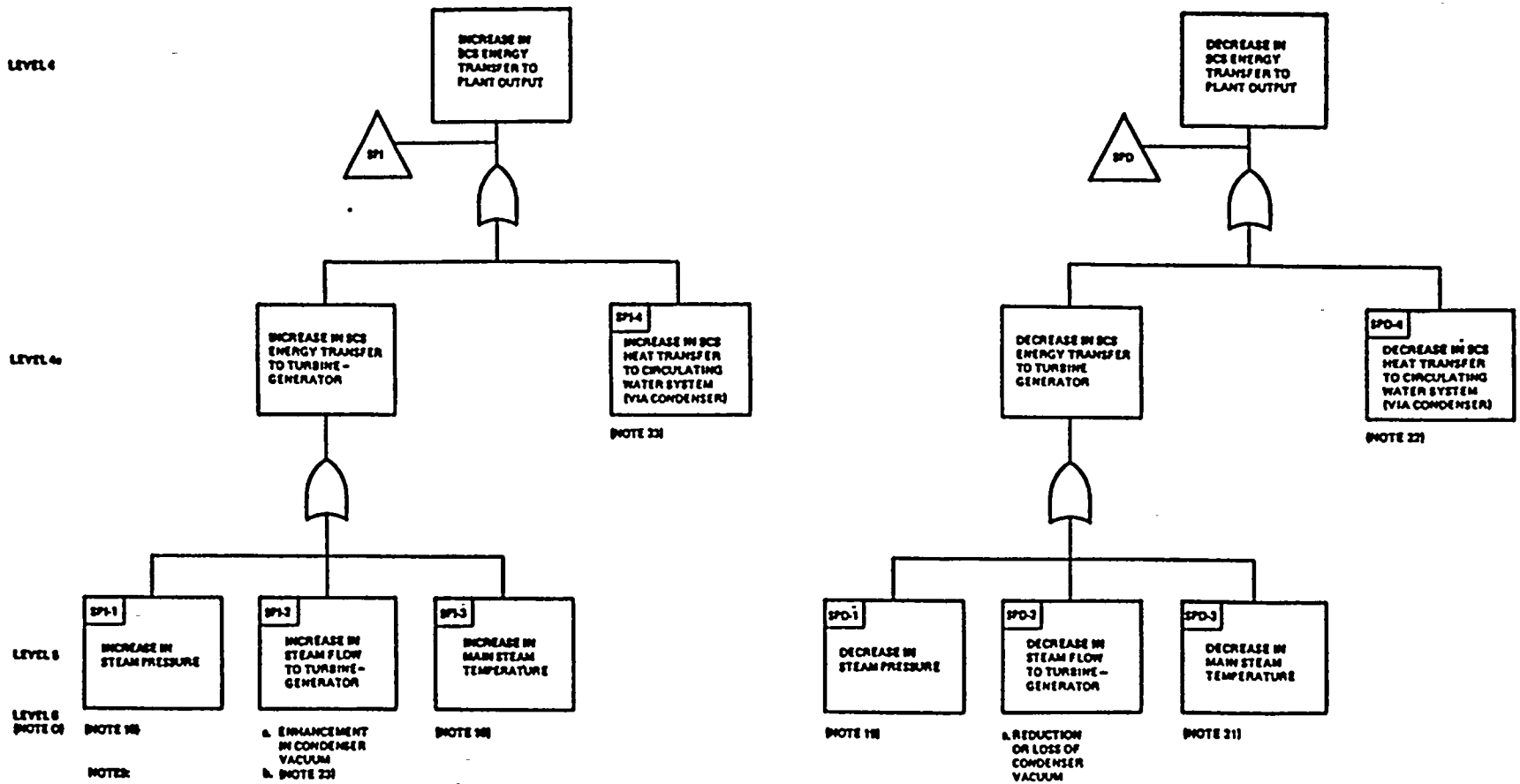


NOTES:

C. COMPLETENESS CANNOT BE ASSURED AT LEVEL 6: ONLY THOSE EVENTS IDENTIFIED AND JUDGED SIGNIFICANT ARE LISTED.

18. CAN ONLY OCCUR AS A SECONDARY EFFECT OF PRIMARY INITIATORS IDENTIFIED ELSEWHERE ON THIS FAULT TREE, SUCH AS THOSE LISTED UNDER △ .1. , INCREASE IN RCS HEAT TRANSFER TO SCS.

19. PRIMARY INITIATORS SUCH AS OPENING OF ARVs OR TCVs ALREADY CLASSIFIED UNDER SRI, INCREASE IN SCS HEAT REMOVAL FROM RCS.

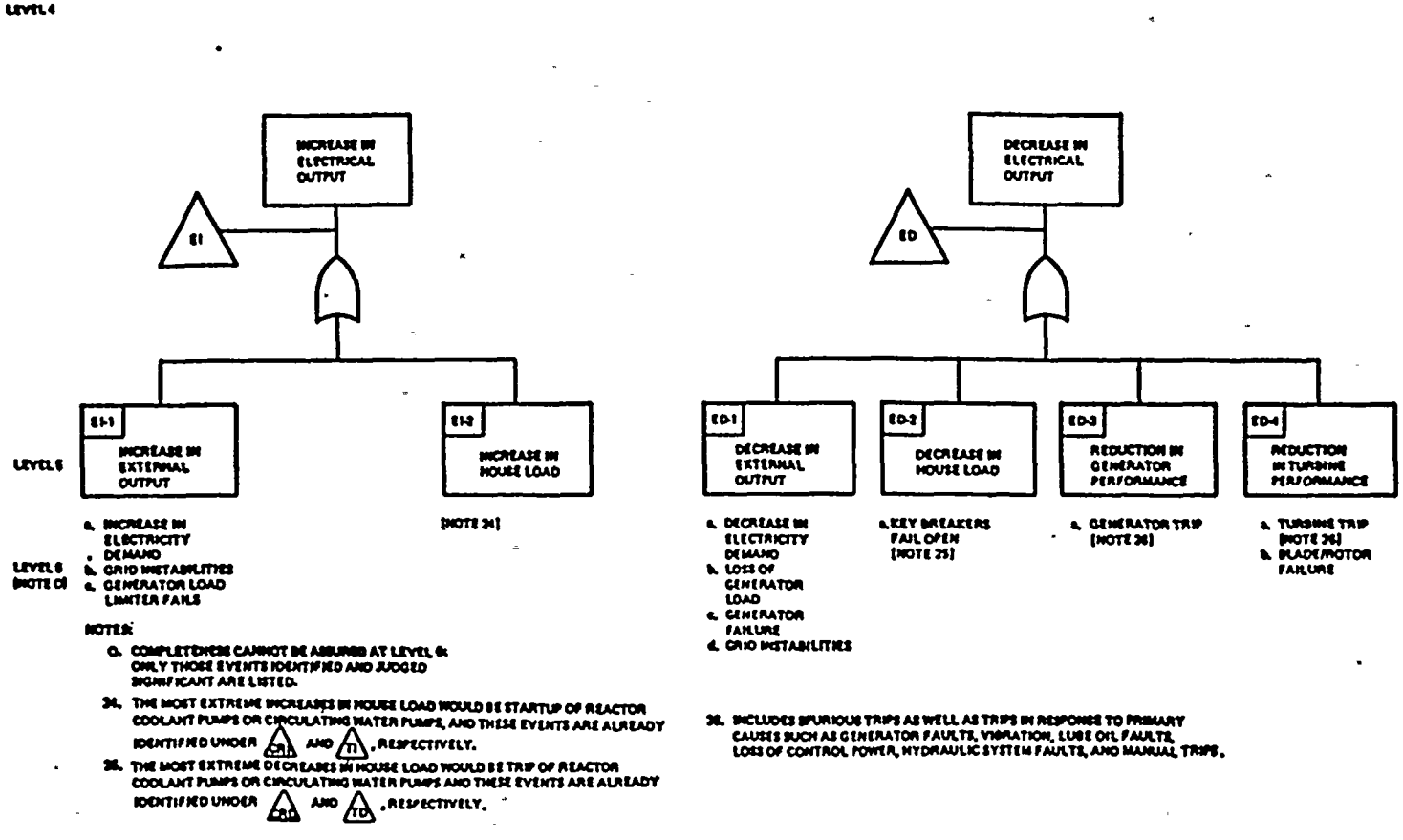
20. PRIMARY INITIATORS SUCH AS CLOSURE OF TCVs, TSVs, MSIVs, STEAM LINE BREAK, OPENING OF TBVs ALREADY CLASSIFIED UNDER SRD-3, DECREASE IN MAINSTEAM FLOW AND SRI-3, INCREASE IN MAINSTEAM FLOW EXITING STEAM GENERATOR

21. CAN ONLY OCCUR AS A SECONDARY RESULT FROM PRIMARY INITIATORS SUCH AS THOSE LISTED UNDER △ .1. , INCREASE OF RCS HEAT TRANSFER TO SCS

22. CAN ONLY OCCUR AS A SECONDARY RESULT OF PRIMARY INITIATORS SUCH AS THOSE LISTED UNDER DECREASE IN SCS ENERGY TRANSFER TO TURBINE GENERATOR

23. OPENING OF TCVs AND TBVs ALREADY IDENTIFIED UNDER SRI-3, INCREASE IN MAINSTEAM FLOW EXITING STEAM GENERATOR

FIGURE C.1-3f. HEAT BALANCE SUBTREES FOR INCREASE AND DECREASE IN SCS ENERGY TRANSFER TO PLANT OUTPUT



NOTES:

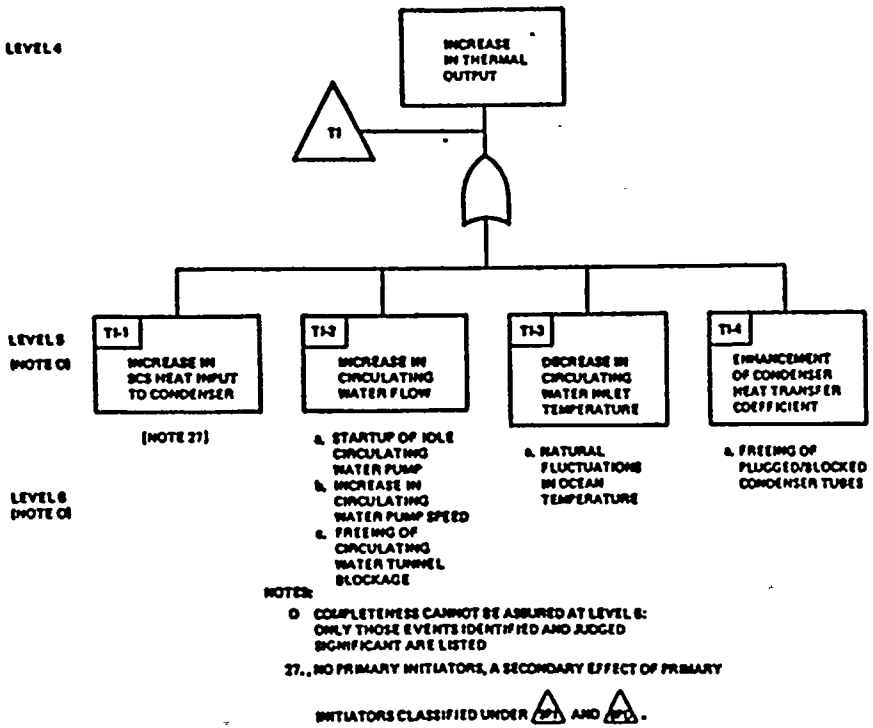
C. COMPLETENESS CANNOT BE ASSURED AT LEVEL 6; ONLY THOSE EVENTS IDENTIFIED AND JUDGED SIGNIFICANT ARE LISTED.

24. THE MOST EXTREME INCREASES IN HOUSE LOAD WOULD BE STARTUP OF REACTOR COOLANT PUMPS OR CIRCULATING WATER PUMPS, AND THESE EVENTS ARE ALREADY IDENTIFIED UNDER CRI AND TI, RESPECTIVELY.

25. THE MOST EXTREME DECREASES IN HOUSE LOAD WOULD BE TRIP OF REACTOR COOLANT PUMPS OR CIRCULATING WATER PUMPS AND THESE EVENTS ARE ALREADY IDENTIFIED UNDER CRD AND TD, RESPECTIVELY.

26. INCLUDES SPURIOUS TRIPS AS WELL AS TRIPS IN RESPONSE TO PRIMARY CAUSES SUCH AS GENERATOR FAULTS, VIBRATION, LUBE OIL FAULTS, LOSS OF CONTROL POWER, HYDRAULIC SYSTEM FAULTS, AND MANUAL TRIPS.

FIGURE C.1-3g. HEAT BALANCE FAULT SUBTREES FOR INCREASE AND DECREASE IN PLANT ELECTRICAL OUTPUT



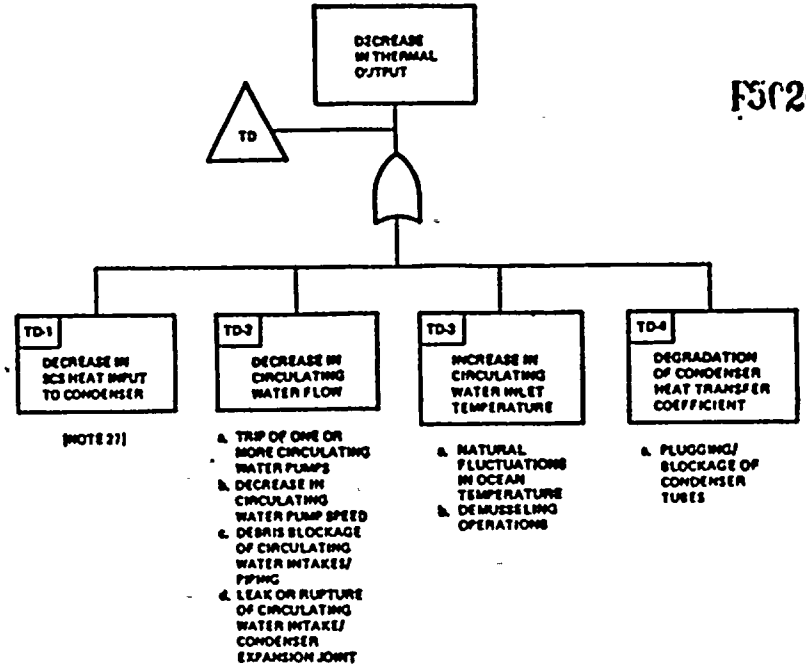


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FIGURE C.1-3h. HEAT BALANCE FAULT SUBTREES FOR INCREASE AND DECREASE IN THERMAL OUTPUT

C.3 PLANT DAMAGE STATES

C.3.1 INTRODUCTION AND OVERVIEW

This appendix defines the plant damage states, which are used to categorize the plant model scenarios. These plant damage states are the discrete end states of the plant model for sequences resulting in core damage. They also would represent the initiating events for a follow-on containment response analysis that may be considered in the future, and they are defined from this perspective. The objective is to define the plant damage states so that, within a plant damage state, the sequence-to-sequence variability in the containment response is small compared to the uncertainties in the outcome of the top events on the containment event tree.

In this manner, the issue of sequence variability is separated from the issue of containment response uncertainty, and the analysis of split fractions on the containment event tree can be limited to the assessment of containment response uncertainty.

Since a containment response analysis is not being performed at this time, a containment event tree is not being developed. It is therefore necessary to judge the level of differentiation needed in the definition of the plant damage states. To provide adequate confidence that a redefinition of plant damage states will not be required, should a containment response analysis be performed in the future, a larger number of discrete plant damage states is provided than what might actually be used. Since PLG has performed a containment response analysis for

several large dry PWRs, there is a high degree of confidence that these plant damage states would be appropriate for a Diablo Canyon containment response analysis. The next section will review the Diablo Canyon containment design to identify important design features for a containment response analysis and to determine whether any of these are substantially different from other large dry PWRs.

C.3.2 EVALUATION OF DIABLO CANYON CONTAINMENT DESIGN FEATURES

C.3.2.1 Containment Design Review Approach

A containment walkdown was performed to provide a visual inspection of the main containment design features, which are important for the PRA. The containment features are important for two reasons: (1) the definition of the plant damage states and (2) the assessment of the physical progression of accident sequences leading to releases of radionuclides from the containment. Since the current project plan for the Diablo Canyon PRA does not include a containment response analysis, the near-term focus is on the definition of plant damage states. The containment walkdown and the documentation provided in this section will form the basis for determining the plant damage states.

The plant damage states constitute the interface between the plant model and the containment model in a PRA. Therefore, the plant damage states represent, on one hand, the end points of the plant model and, on the other hand, the initial conditions for the containment response model. Since they represent the containment response initial conditions, they

must capture those aspects of a plant model scenario that are of key importance in determining the accident progression, the containment response, and the release of radionuclides from the containment. Capturing these key features for the containment response requires an evaluation of the following design features of the plant:

1. The containment heat removal systems and the containment fission product removal systems.

2. The reactor cavity size and configuration.

3. An evaluation of the access paths for water to enter the reactor cavity and the containment conditions under which water can enter the reactor cavity.

4. The behavior of core debris at the time of vessel melt-through during an accident sequence and an evaluation of the geometric configuration for debris dispersal pathways out of the reactor cavity.

5. An evaluation of the containment configuration for mixing the containment atmosphere when no forced convection and mixing are provided by the fan coolers.

6. An identification of potential containment bypass modes and the size of these containment bypass leak paths after containment isolation, including both bypasses with direct communication between the containment atmosphere and the environment and bypasses with direct

communication between the reactor coolant system and the environment. Any filtration or fission product scrubbing in these leak paths is also of importance.

7. The configuration of the containment ECCS sump is important for assessing potential obstructions of the recirculation flow path after core debris has been released into the containment during an accident sequence.

8. The RHR system configuration in the auxiliary building is examined because of the importance of the so-called V-sequence, which involves the postulated failure of the isolation valves between the high pressure and low pressure portions of the reactor coolant system and RHR piping.

The relevant containment feature about each of these eight items, observed during the containment walkdown, will be discussed and evaluated in the following sections.

C.3.2.2 Containment Heat Removal Systems and Containment Fission Product Removal Systems

The Diablo Canyon containment includes five containment fan cooler units and two containment spray injection and recirculation trains. In principle, either of these systems can perform both containment heat removal and the containment fission product scrubbing function following a core melt accident. The containment spray system in the injection mode both cools and removes fission products from the containment atmosphere.

In the recirculation mode of operation, the cooling function is provided through the RHR heat exchangers, while fission product scrubbing continues by the spray droplets. Also in the recirculation mode, the discharge from the RHR heat exchangers is split. Most of the flow is directed to the reactor coolant system, and the balance of the flow is diverted to the containment spray headers. Effective fission product removal in the recirculation mode will therefore require that an appropriate fraction of the recirculation flow will be diverted to the spray headers.

The fan cooler units consist of fans, cooling coils, and filters. The fan coolers have two operating modes. During normal operation, air inlet dampers direct air into a plenum from which the air flow passes over the cooling coils to the fans that distribute the air to the lower portions of the containment. Under accident conditions, a switchover to the accident flow mode occurs. The normal flow inlet dampers close, and the accident dampers open. In the accident flow mode, the air is drawn through the far end of the fan cooler housing. The air is directed, in succession, through the moisture separator section, through the HEPA filter section, through the accident dampers, and through the cooling coil section into the fans. Substantial fission product scrubbing can occur, either in the HEPA filters or by moisture condensation on the cooling coils. However, if flow is in the accident mode, the air is first drawn through the HEPA filters. The possibility of HEPA filter

plugging due to suspended aerosols in the containment atmosphere would be addressed for each plant damage state in the containment response analysis. A containment response analysis is beyond the currently defined scope for the DCPRA. Plugging of the HEPA filters would progressively reduce the air flow through the fan cooler units and would lead to a condition in which neither the containment heat removal function nor the containment fission product scrubbing function could be performed by the fan coolers. Operator action to switch the fan coolers to the normal operating flow path would bypass the plugged HEPA filters and provide continued operation of the fan cooler units while the fission product scrubbing function would be performed by steam condensation on the cooling coils. However, this would require that the fan cooler be operable at low speed but with the normal flow path. Therefore, under conditions of high aerosol content in the containment atmosphere, the availability of fan coolers may depend on operator actions. The time available for these operator actions would depend on how long the fan coolers can survive without damage in a zero-flow operating mode.

C.3.2.3 Reactor Cavity Configuration

The configuration of the reactor cavity at the Diablo Canyon plant is similar in concept to those at the Zion plant, the Indian Point units, and Seabrook Station. It features a deep cavity in the center of the containment below the reactor vessel, with an instrument tunnel through which the in-core instrument tubes lead up to the seal table. The floor

of the reactor cavity is 27.5 feet below the floor of the containment. The floor area of the reactor cavity is approximately 500 square feet, which is sufficient to cool the core debris after it penetrates through the reactor vessel as long as the debris is covered by water. This is supported by the findings in previous PRAs. Therefore, continued concrete attack would not be expected under such conditions. The reactor cavity sump is located underneath the reactor vessel. Debris would collect in the sump and displace any water that would be in it. The ability to cool the debris in the sump region would need to be assessed.

There are five potential pathways by which water could enter the reactor cavity or by which debris or hydrogen gas could be released from the reactor cavity. These five pathways are:

1. The fire door that separates the reactor cavity access room from the main containmemt floor area. This fire door is normally closed. It represents an opening of approximately 20 square feet.

2. The ventilation dampers in the reactor cavity access room. The position of these dampers under accident conditions has not been identified. These dampers represent a combined potential opening of 20 square feet.

3. The manhole access, with a steel cover plate, located between the biological shield and the reactor cavity access room measures approximately 24 square feet in cross section. It is assumed that the steel cover for this manhole access hatch is normally in place.

4. The concrete access hatch to the reactor coolant drain tank area. This opening measures approximately 30 to 40 square feet. It is normally closed off by three concrete blocks, which are 14 to 15 inches thick.

5. The clearance between the reactor vessel thermal insulation and the biological shield was observed to be about 3 inches at the bottom end of the thermal insulation. It was not possible to verify that this clearance extends all the way up through the floor of the equipment lay-down space where it would open into the main containment volume. However, it is expected that this clearance, or an equivalent area clearance, between the reactor cavity and the main containment volume around the reactor vessel does exist for ventilation purposes.

The clearance around the reactor vessel is the only flow path between the reactor cavity and the main containment volume, which is normally unobstructed. All other potential pathways would require the forceful removal of a barrier, such as a damper, ventilation ducting, a door, or an access hatch cover.

C.3.2.4 Water Access to the Reactor Cavity

An important consideration in accident sequences is how accessible water is to the reactor cavity to cover and cool any core debris that may collect in the reactor cavity. Based on the foregoing discussion, water that may collect on the containment floor would not have direct access to

the reactor cavity except from leakage through fire door gaps and the possible gaps in the access hatch covers.

Two separate conditions are considered. Dry containment conditions result if, during an accident sequence, the refueling water storage tank is not injected. The release of all of the primary coolant system water inventory under the containment floor will yield a water depth of less than 1 foot. A portion of this water would exist as steam in the containment atmosphere, and a portion would collect in other areas, such as the containment sumps. The net depth of water on the containment floor would be about 6 inches. There is a 6-1/2-inch curb in the reactor cavity access room and around the manhole access. Therefore, it is questionable whether any water would enter the reactor cavity under dry accident conditions, so it is assumed the reactor cavity would remain dry.

Wet containment conditions result from accident sequences in which the refueling water storage tank is injected into the containment during the accident sequence. The resulting depth of water on the containment floor would be approximately 4 to 6 feet, and as much water would enter the reactor cavity as the leakage past the fire door and the access hatches permits. However, the hydrostatic water pressure would tend to reinforce the seal at the fire door and at the access hatch cover plates. Therefore, without further investigation, access of water to the reactor cavity under wet accident conditions cannot be guaranteed, at least not before vessel melt-through occurs.

For accident sequences in which the containment spray system is operating in the injection and/or recirculation mode, there is one additional pathway for water to reach the reactor cavity. Spray water collecting on the equipment laydown floor in the refueling bay can drain from that bay by three different pathways:

1. Over the edge of the refueling bay into the refueling transfer canal.

2. Through the drains in the equipment laydown floor leading to the containment sump.

3. By draining around the reactor vessel through the gap between the biological shield and the thermal insulation leading into the reactor cavity if this drain path can be verified to exist. For certain accident sequences, this may be the only water access pathway to the reactor cavity, and an evaluation would be required to determine whether sufficient water can be supplied to the reactor cavity through this pathway.

Water that collects in the reactor cavity cannot drain out of the reactor cavity. However, the sump pumps, which are located at Elevation 91' in the reactor cavity access room, can pump water out of the reactor cavity sump via the eductors. Therefore, it will be necessary to determine the conditions under which the sump pumps can operate, whether the operator has indications that would lead him to shut off the sump pumps under certain conditions, and the flow rate at which the sump pumps can pump

water out of the reactor cavity. These questions must be resolved to determine whether water that reaches the reactor cavity would remain in the reactor cavity.

C.3.2.5 Debris Behavior and Disposition at Vessel Melt-Through

Two conditions are distinguished about debris behavior at vessel melt-through. For accident sequences involving large and medium size loss of coolant accidents, the pressure in the reactor vessel at the time of the vessel melt-through will be in equilibrium with the containment atmosphere pressure. At vessel melt-through, the debris will be discharged into the reactor cavity by draining out of the reactor vessel under gravity forces. Without water in the reactor cavity, concrete attack will begin, and the gas evolution rate and gas composition will be determined by the concrete aggregate composition. No dynamic pressure transients in the reactor cavity would be anticipated; therefore, the leakage pathway around the reactor vessel may remain the only pathway for gases to escape from the reactor cavity. If the reactor cavity is filled with water, the interaction between the debris and the water will expel the water from the reactor cavity and create a pressure transient that could be expected to fail the manhole access hatch cover, the door to the reactor cavity access room, and the ventilation damper and ductwork in the reactor cavity access room.

The second condition that is distinguished at the time of vessel melt-through involves transient and small loss of coolant accident

sequences. These sequences are characterized by an elevated pressure in the reactor coolant system at the time of vessel melt-through on the order of 1,000 psia or higher. At vessel melt-through, the debris is forcibly ejected from the reactor vessel by the pressure in the reactor coolant system. The dynamic effects of this pressurized blowdown and the dynamic effect of the gas ejection, which follows the debris ejection, could be expected to cause a pressure transient in the reactor cavity that also fails the manhole access cover plate, the door to the reactor cavity access room, and the ventilation damper and ductwork to the reactor cavity access room even if no water is present in the reactor cavity. A portion of the core debris will be dispersed from the reactor cavity to the main containment floor area by this mechanism. If the refueling water storage tank has been injected, water will flood the reactor cavity after the blowdown, and it will cool any debris remaining in the reactor cavity. Since the reactor cavity sump pump is located in the reactor cavity access room and not on the reactor cavity floor, the sump pump may or may not continue to operate after such a dispersal transient. This question may require further investigation if a containment response analysis is performed.

C.3.2.6 Containment Mixing

The upper compartment of the Diablo Canyon containment is open and well mixed even if the fan coolers are not operating. The floors and ceilings of the lower compartments have a limited amount of grating in different locations. Particularly, inside the crane wall, the amount of grating that was found to communicate from level to level was not large. The

steam generator and reactor coolant pump compartments communicate to the upper containment compartment through a 2 to 4-inch-wide gap around the steam generators at Elevation 140'. The reactor cavity communicates to the upper containment compartment through a narrow gap between the reactor vessel insulation and the biological shield wall. Other communication paths may exist after a dynamic reactor vessel melt-through event. Our overall assessment is that the upper compartment is well mixed, but that the lower compartments might behave as separate communicating volumes.

C.3.2.7 Containment Bypass

The plant damage states will require a classification of conditions under which the containment isolation function is bypassed. Of interest are isolation failures and penetrations that either directly communicate with the containment atmosphere, such as the air purge lines, or that communicate directly with the reactor coolant system, such as the seal return lines for the reactor coolant pumps or the RHR penetrations. The penetrations of interest are identified as part of the normal plant analysis process and do not require inspection at the site. Only the so-called "V-sequence," which involves containment bypass through the RHR penetration, has been explicitly addressed during the site walk because of the importance of configuration details for this sequence. The relevant details are discussed in Section C.3.2.9 (RHR System Configuration).

C.3.2.8 Containment Sump Configuration

The containment sump configuration was inspected to determine whether it is likely that a sump recirculation flow path can be maintained after vessel melt-through when a substantial portion of the core debris is located on the containment floor. The containment sump includes a 4-inch curb on the inside that would prevent debris particles that are large enough to settle on the floor from reaching the sump pump. The recirculation suction pipe is mounted vertically in the floor of the containment sump, without a curb and with only a small, raised pipe section. Therefore, debris that would spill over the 4-inch curb could reach the RHR pump through the RHR suction line. However, it is unlikely that an accumulation of debris around the ECCS containment sump in excess of 4 inches would occur. Furthermore, the steel grating, which prevents floating debris, such as pipe insulation material, from reaching the reactor cavity, is expected to remain in place and to be unaffected by any debris relocation transient impacts. Therefore, the recirculation cooling system is not expected to be adversely affected by the events occurring at vessel melt-through and the associated debris release into the containment.

C.3.2.9 RHR System Configuration

Analyses under the IDCOR program indicate that, for the so-called "V-sequence" in which the interfacing valves between the high pressure reactor coolant system and the low pressure RHR system are postulated to fail, the low pressure piping system can be expected to remain intact.

Leakage at the RHR pump seals is considered to be a more likely failure mode and release path. In addition, relief valves for the low pressure system and their discharge location must be considered as potential leak paths. The relief valve piping can be traced on the system drawings, whereas the RHR pump configuration was inspected during the site visit.

The RHR pumps are located at Elevation 58' in the auxiliary building. There is only one elevation lower than the RHR pump floor; namely, the pipe tunnel at floor Elevation 54'. The RHR pumps are located on the floor, with the pump seals located at an elevation less than 27 inches above the floor. Water accumulating in the RHR pump room will fill the the RHR pump room sump, which goes down to Elevation 54'. If the sump pumps are operating and are not stopped by the operator, water is not likely to accumulate on the RHR pump room floor. If the RHR sump pumps are not operating, water will accumulate up to a level of 37 inches above the floor, then begin to drain through a drainpipe from the RHR pump room into the pipe tunnel where water will begin to accumulate at Elevation 54'. There are 6-inch drains in the floor of the pipe tunnel, which probably reduce to a 4-inch drain pipe. It can be assumed that the RHR pump seals will be at least 10 inches under water if the RHR sump pumps are not operating.

Any radioactivity not retained in the RHR pump seal leakage path or in the water on the RHR pump floor will be airborne in the RHR pump room.

From there, several possible atmospheric leakage paths to the environment exist, as in the following:

1. Radioactivity can be transported through the auxiliary building ventilation system if it is operating. This ventilation system may be shut down and isolated, either by the operator or by fusible-link fire dampers. The existence of fire dampers for the RHR pump room has not been verified.

2. Radioactivity can be transported through the overflow drainpipe into the pipe tunnel. A loop seal may exist in this drain pipe. The leakage pathway from the pipe tunnel to the environment would need to be established, but is presumed to exist.

3. Radioactivity may be transported from the RHR pump room to the RHR heat exchanger room above via the ladder access and damper connection. From the RHR heat exchanger room, radioactivity can be transported to the charging pump room. For train 1, the door between the charging pump room and the RHR heat exchanger room is a solid fire door, whereas for train 2, the door is a 2-inch grated steel door. From the charging pump room, radioactivity can diffuse to the remainder of the auxiliary building.

4. If the pressure transient in the RHR pump room fails the ventilation register and ductwork or one of the two sets of access doors to the RHR pump room, then radioactivity can also be transported through these failed components into the remainder of the auxiliary building.

Except for the forced ventilation pathway, the pathways for atmospheric transport of radionuclides from the RHR pump room to the environment is a torturous path within the auxiliary building. The rooms for the ECCS components are located in cubicles with narrow passage keyways, or they are isolated by fire doors. A closer inspection of the auxiliary building and these potential leak paths to the environment would be required to model the auxiliary building.

C.3.2.10 Conclusions

The information collected during the site visit, together with the documentation of the containment safety systems in the Diablo Canyon FSAR and the arrangement drawings for the containment and auxiliary building, will form an adequate basis to define plant damage states. A separate site visit is not expected to be required for this purpose. Some minor questions may require resolution as part of the containment response analysis before the survivability of the fan coolers can be determined, such as the question about the iodine removal units (Section C.3.2.2) and the drain path around the reactor vessel to the reactor cavity (Section C.3.2.4).

C.3.3 PLANT DAMAGE STATES

The plant damage states represent the second set of pinch points or groups of accident scenarios. This set of pinch points includes the initiating events, coupled with subsequent plant system failures that lead to some significant core degradation and release of fission products

from the fuel. The containment event tree is the logic framework for developing the accident scenarios from plant damage states to release categories, which is the third set of pinch points.

The grouping of scenarios into plant damage states proceeds from the premise that the broad spectrum of many thousands of plant failure scenarios can be discretized into a manageable number of representative categories. Within each category, a single assessment of the core and containment response will represent the response for all the individual scenarios in the category. Therefore, each plant damage state collects all those sequences for which the core melt progression, the release of fission products from the fuel, the containment environment, the source term mitigation, and the containment states are similar.

As the "initiating events" for the containment event tree, the plant damage states need to convey that information that (1) is of importance to evaluate the containment response and (2) is determined by the events preceding core uncovery. The information important to the containment response can be grouped into two classes: (1) information relating to the physical conditions at the time of core melt and vessel melt-through and (2) information relating to the availability of containment systems.

Past experience in the evaluation of PWR containment response to core melt sequences indicates that two physical parameters are of the highest significance:

- The pressure inside the reactor pressure vessel at the time when the debris melts through the vessel is an important parameter. A high primary system pressure up to the time of vessel failure can:

  - Create conditions that suppress an in-vessel steam explosion.

  - Eject an instrument penetration from the vessel bottom after the penetration weld has melted from contact with molten fuel.

  - Forceably eject molten fuel through the instrument penetration hole. The limited rate of fuel ejection can minimize the extent of molten fuel interaction with water on the reactor cavity floor.

  - Entrain molten debris out of the reactor cavity in the high pressure gas jet that follows the ejection of molten fuel.

  - Rapidly release a significant amount of accumulated hydrogen into the containment at vessel failure.

- The presence of a substantial depth of water in the reactor cavity underneath the pressure vessel and the ability to naturally reflux water condensing from the containment atmosphere into the reactor

cavity are important considerations about the containment response because the interaction with hot core debris, when vessel melt-through occurs, can:

- Enhance the dispersion of core debris from the reactor cavity into other containment volumes.

- Fragment and quench the remaining debris in the reactor cavity and cool the debris by continuous refluxing of water into the cavity.

- Cause the containment pressure to increase by vaporization of steam and heatup of the containment atmosphere.

- Enhance the release of fission products from the fuel due to potential oxidation of fine particulates.

Alternatively, in a dry cavity without water, the following processes can take place:

- Molten core debris that remains in the reactor cavity will attack concrete and penetrate into the basemat if the debris is more than a few inches deep. If a substantial fraction of the core debris remains in the cavity, basemat melt-through can eventually occur.

- The decomposition of concrete and water from the concrete can add steam, noncondensible gases, and flammable gases to the containment environment.

- Gas sparging through the molten debris can transport additional fission products from the debris into the containment atmosphere.

In addition, the time after reactor shutdown at which core melt starts is considered important; i.e., the distinction between early and late core melt. This parameter can affect the decay heat level that drives the core melt processes, the radionuclide inventory available for release, and the warning time for emergency measures. The time of core melt does not directly influence the containment failure modes, and the impact on the release timing and magnitude would be accounted for in the uncertainty distributions for these release parameters. Therefore, it is not necessary to establish separate plant damage states to distinguish sequences with early versus late core melt times.

The availability of containment safety systems to perform the following functions is of key importance in the evaluation of containment response:

- The state of the containment (intact, failed, unisolated, or bypassed) at the time core damage starts; i.e., when the containment event tree is entered. This distinction not only includes containment isolation failures and interfacing systems loss of

coolant accidents, but is also of importance for external events that can potentially cause containment failure prior to core damage, such as earthquakes, storms, or external missiles.

- The availability of containment engineered safety features for containment atmosphere cooling and for active fission product scrubbing from the atmosphere, such as containment sprays, containment spray recirculation using the RHR heat exchangers, and fan coolers.

- The availability of active filtration and other mechanisms for fission product removal from the containment atmosphere or in the leakage path if the containment is failed at the time of core damage initiation, such as containment sprays operating with the containment failed.

The five parameters discussed above are used to define plant damage states as the combinations of conditions in the containment at the time of core melt. These plant damage states are shown in Table C.3-1 as a matrix of 5 physical conditions in the containment (numerals 1 through 5) and 14 combinations of containment safety function availability (letters A through N), for a total of 70 potential plant states.

Containment physical conditions are labeled numerically from 1 through 5, whereas the containment safety function availability states are labeled A through N. Each plant damage state is identified by a number-letter combination; e.g., for plant damage state 4D in Table C.3-1, the RCS pressure is high at vessel melt-through, and the RWST is being injected

after vessel melt-through. Furthermore, the containment is intact and isolated when core damage starts, but neither active fission product removal systems nor active containment heat removal systems are available.

Each plant damage state decision parameter represented in Table C.3-1 is discussed in the following paragraphs. Decision criteria for each plant damage state parameter are required to distinguish quantitatively among plant damage states. The decision criteria have been derived on the basis of the Diablo Canyon design configuration and associated design criteria, physical processes assessments, and experience from other PRAs.

- Reactor Coolant System Pressure at Vessel Melt-Through. A pressure level of 300 psia in the reactor vessel at the time of vessel melt-through has been selected as the level at which high pressure effects are considered. Above this pressure level, debris sweepout from the cavity has the potential for removing a significant portion of the debris. However, the disposition of the swept-out debris is not clear other than it is accumulated in the reactor cavity access room. The disposition of swept-out debris would be determined as part of the analysis of accident progression in a containment response analysis.

  The exact measure level at which debris sweepout is effective does not have to be determined very accurately because analysis has shown that the RCS pressure in PWRs at the time of vessel melt-through tends to be either high or low. For accident sequences involving large or medium size loss of coolant accidents, the RCS pressure

would be nearly in equilibrium with the containment atmospheric pressure; i.e., well below the 300-psia value. To these sequences, plant damage state numerals 1 and 2 would apply. For small loss of coolant accidents and for transient accident sequences, on the other hand, the RCS pressure at vessel melt-through is typically in the range of 1,000 psia to 2,250 psia; i.e., well above the 300-psia value. To these sequences, plant damage state numerals 3, 4, and 5 would apply.

- RWST Injection (water in reactor cavity). Access of water to the reactor cavity was discussed in Section C.3.2.4. It was concluded that water flooding the reactor cavity will not occur unless the RWST has been injected into the containment. Even if the RWST is injected, flooding of the reactor cavity is uncertain prior to vessel melt-through for the reasons discussed in Section C.3.2.4. Flooding of the reactor cavity following vessel melt-through would be likely for high pressure accident sequences. For low pressure sequences, cavity flooding is not ensured because the dynamic effects at vessel breach may not be sufficient to create a water access path. Uncertainties about water access to the reactor cavity would be resolved as part of an accident progression and containment response analysis. The plant damage states will distinguish RWST injection possibilities in such a manner that water access to the cavity can be correctly modeled in a containment response analysis.

Three modes of RWST injection are distinguished; namely, injection before vessel melt-through, injection after vessel melt-through, and

no RWST injection. These are important to distinguish if water access to the cavity is possible before vessel melt-through. If water access to the cavity before vessel melt-through is not possible, a distinction between RWST injection before and after vessel melt-through would not be necessary.

RWST injection before vessel melt-through requires that either the RWST is injected in an injection core cooling mode or, if there is no vessel injection, that the containment spray systems are actuated well before vessel melt-through so that the majority of the RWST would be injected before vessel melt-through. In this case, water is available to flood the cavity before vessel melt-through; however, depending on the resolution of the water access issue, the cavity may remain dry until vessel melt-through. If water can gain access into the cavity, the cavity would be flooded to the top and debris would discharge into the water. This would cause a dynamic water relocation out of the cavity, followed by reflooding of the cavity, and the containment atmosphere source term would be reduced by fission products retention in the water.

RWST injection after vessel melt-through is still possible for high pressure accident sequences even when the high pressure injection mode is not available and containment spray injection has not been activated. At vessel melt-through, the RCS pressure drops to be in equilibrium with the containment pressure, and, at this time, the low pressure injection system may start to deliver flow to the pressure vessel. The injected water would drain through the hole in the

vessel bottom directly into the reactor cavity to quickly cover any debris that has accumulated on the cavity floor. One low pressure injection train would deliver adequate flow for this purpose. For low pressure accident sequences, this RWST injection mode is not possible because, to fail RWST injection before vessel melt-through, the low pressure injection systems must also have failed.

No RWST injection is taken into account if RWST injection does not start before or at vessel melt-through because, by this time, automatic start signals should have been generated for all systems that can inject RWST water. Long-term recovery of systems that can inject water into the containment is possible, and, if performed carefully, even late injection of water can significantly delay the time of late overpressure failure and reduce the radionuclide source term. In a Level 2 or Level 3 PRA, this would be considered as a recovery action for the containment model.

The two RCS pressure states and the three RWST injection states are combined into five states labeled by numerals 1 to 5, as shown on the left-hand side of Table C.3-1. States 1 and 2 are the low RCS pressure states, and states 3, 4, and 5 are the high RCS pressure states. The RWST is injected prior to vessel melt-through for states 2 and 5. For states 1 and 3, there is no RWST injection and the high pressure state with RWST injection after vessel melt-through is designated state 4. The success criteria for these five states are defined in Table C.3-2.

The containment system states are made up from three considerations; namely, the status of the containment boundary, the status of the containment heat removal systems, and the status of the fission product scrubbing systems. These three containment features are combined into 14 containment system states, which are labeled A to N on Table C.3-1. The success criteria for these 14 states are defined in Table C.3-2.

Three conditions of the containment boundary at the time of core melt are considered; namely, (1) intact and isolated, (2) not isolated, and (3) bypassed. Since the plant damage states only collect accident sequences that proceed to core damage, it is necessary to consider all pathways for radionuclide release to the environment. Such pathways can either proceed via the containment atmosphere through an opening or unisolated penetration that communicates with the containment atmosphere or from the RCS via a primary coolant fluid penetration or by failure of the pressure boundary between the primary coolant system and the secondary coolant system.

1. Containment Intact and Isolated. This state represents the normal status of the containment during an accident condition. All penetrations are isolated and leaktight within their leakage specification. The leak rate from the containment structure, as a whole, is within the technical specification limit. Containment heat removal is required to prevent continuous containment pressurization following vessel melt-through. Four substates are distinguished for an isolated containment, depending on the status of the containment heat removal systems and the fission product removal systems. The

resulting four containment system states are designated A through D on Table C.3-1.

(A) Both the containment heat removal function and the fission product removal function are available. Either operation of the fan coolers or of containment spray recirculation cooling would satisfy this state.

(B) Only containment heat removal is available, but not fission product removal. Recirculation cooling of the containment sump water without containment recirculation spray and without fan coolers operating would represent this state.

(C) Containment heat removal is not available, but fission product removal is available. Containment spray recirculation cooling without containment sump cooling and without fan coolers would satisfy this state. Since the RHR pumps provide the flow for both the sump recirculation mode and the spray recirculation mode with the spray flow fraction regulated by the spray valve, this state is extremely unlikely at Diablo Canyon. It would require that the RHR pumps operate in the recirculation mode with the discharge path for sump recirculation blocked downstream of the spray line branchoff.

(D) Neither containment heat removal nor fission product removal is available.

Two unisolated containment states are considered, depending on the size of the unisolated leak path. All unisolated containment conditions represent leakge paths in excess of the containment design basis leak rate that leak from the containment atmosphere to the environment. A small leak exists if the leak rate is less than what would occur through a 3-inch diameter orifice. With this leak area, the pressure in the containment would still increase for some time after vessel melt-through in the absence of containment heat removal. The release duration associated with a 3-inch diameter hole would be an extended release over a period of 10 to 20 hours in which a relatively small fraction of the inventory would be released before evacuation is complete. An unisolated containment with a large leak area is one with a penetration that has a diameter greater than 3 inches in diameter that can be open, at least temporarily, during normal operation and remains open. The limiting penetration is normally the online atmospheric purge line that can be opened for a limiting number of hours, as specified in the technical specifications.

2. Unisolated Containment with a Small Leak Path (less than 3-inch diameter orifice equivalent). For this isolation failure category, the same four subcategories apply as for the isolated containment case because containment heat removal is still required to prevent the continued pressurization of the containment. The four resulting containment states are labeled E through H on Table C.3-1. The success criteria for the containment heat removal and fission product

removal functions in these four substates are the same as for substates A through D, as is shown in Table C.3-2.

3. Unisolated Containment with Large Leak Path (more than 3-inch diameter orifice equivalent). For leak areas more than 3 inches in diameter, the containment does not continue to pressurize following vessel melt-through even if containment heat removal is not available and the containment depressurizes to atmospheric pressure. Therefore, the only distinction necessary is the availability of fission product removal. However, in this case, the fan coolers are not considered effective in removing fission products because the containment residence time for the volatile and gaseous fission products is short and it is necessary to scrub the fission products in their leakage path. Only the containment spray system operating continuously in both the injection and the recirculation mode is considered effective. Any additional fission product scrubbing feature directly in the leakage path can also be considered if it remains effective in removing and retaining fission products. The two substates (with and without fission product scrubbing) are labeled I and J in Table C.3-1, and their success criteria are defined in Table C.3-2.

Containment bypass conditions are characterized by a leakage path that bypasses the containment atmosphere. In this release path, radionuclides pass directly from the reactor coolant system to the environment outside the containment through an unisolated piping penetration that directly connects to the RCS. Two such bypass conditions are distinguished;

namely, unisolated steam generator tube ruptures and interfacing systems LOCAs (V-sequence). Other cases may be identified, such as an unisolated RCS pump seal return line following a pump seal LOCA. These would be grouped into the most appropriate plant damage state for the conditions explicitly modeled. For such accident sequences, the release frequently passes into the auxiliary building, such as for a V-sequence release at the RHR pump seals. The radionuclide retention in the structures that form the release path would be modeled in determining the source terms in a Level 2 or Level 3 PRA.

- Containment Bypass - Unisolated Steam Generator Tube Rupture. This bypass condition represents a leakage path directly from the RCS to the environment via the secondary side steam piping and the steam generator relief valves. This type of accident sequence has been gaining relative importance for risk, particularly in large dry PWRs, because the risk asssociated with isolated containment conditions appears to be significantly less than that predicted in WASH-1400. Prior to vessel melt-through, the accident behavior and release is determined by the primary system, whereas, after vessel melt-through, the pressure in the containment building determines the driving force for a continued release. Therefore, a distinction is made, depending on the availability of containment heat removal, that controls the containment pressure as long as the leak area is small. The two substates (with and without containment heat removal) are labeled K and L in Table C.3-1, and their success criteria are identified in Table C.3-2.

- Containment Bypass--V-Sequence. This bypass condition represents a leakage path through the auxiliary building. The retention of radionuclides for this leak path is very plant-specific, depending on the auxiliary building layout. This type of accident sequence has been analyzed in all PRAs, but only recently have modeling advances been made that justify and require that this accident category be treated as an integral part of the plant damage state concept. Prior to vessel melt-through, the accident behavior and the release into the auxiliary building are determined by the primary system, whereas, after vessel melt-through, the pressure in the containment building determines the driving force for a continued release. Therefore, a distinction is made, depending on the availability of containment heat removal, that controls the containment pressure as long as the leak area is small. The two substates (with and without containment heat removal) are labeled M and N in Table C.3-1, and their success criteria are identified in Table C.3-2.

The plant damage state matrix is shown in Table C.3-1. Every plant damage state indicated by a number-letter combination is explicitly modeled as an accident sequence end state in the plant model. Because of the combination of success criteria, the frequency for some of the plant damage states may be very low. When it was possible to anticipate such states, they were asssigned to corresponding, but more conservative, plant damage states. In this case, the conservative substitute plant damage state is indicated in parentheses. Some of the plant damage state combinations on Table C.3-1 are not possible because the success criteria

for the numeral condition and the letter condition are mutually exclusive. Such combinations are marked by an "x," and they are not considered in the plant model.

Plant damage states 1B, 1C, 1F, 1G, 1I, 3B, 3C, 3F, 3G, and 3I are not possible because, for numeral states 1 and 3, there is no RWST injection, whereas, for the corresponding letter states, the RWST must have been injected to meet the success criteria. The remaining plant damage states, with letter designations B and C, are expected to have a very low frequency compared to either the A-states or the D-states. For the B-states, the recirculation spray lineup must fail, although the operators establish sump recirculation cooling and it operates normally. This is considered much less likely than the unavailability of recirculation cooling. For the C-states, spray recirculation must operate while sump cooling is not available. Since the most likely cause for failure of sump cooling is due to the unavailability of the sump suction valves of RHR pumps, it is more likely that both recirculation spray and the sump cooling function are unavailable. For this reason, the remaining B and C-states have been assigned to the corresponding D-states, which are of a higher consequence and are expected to have a higher frequency. The same reasoning applies to the remaining plant damage states in the F and G-states, and they have also been conservatively assigned to the corresponding H-states.

The remaining plant damage states shown in Table C.3-1 are retained for the plant model, although some are expected to show very low frequency because their relative frequencies cannot be identified convincingly at

this time. The success criteria for the numeral conditions and for the letter conditions are listed in Table C.3-2. When the success criteria are not stated in terms of specific equipment or train requirements, the success criteria are intended to be consistent with those used for the same function in the plant model analysis.

C.3.4 ACCIDENT SEQUENCES REPRESENTING PLANT DAMAGE STATES

Note: This section will define two accident sequences for each significant plant damage state, assessed on the basis of plant damage state frequency and severity. The two accident sequences represent a best estimate and a conservative representation of each plant damage state. The definition of these accident sequences provides input to a Level 2 analysis only and is not used or needed for the completion of a Level 1 PRA. For a concurrent Level 2 PRA, this would be done on the basis of preliminary results, which, in case of the Diablo Canyon PRA, would be the results from the Phase II analysis. Since, for the Diablo Canyon PRA, a concurrent Level 2 analysis is not planned or scheduled, the definition of accident sequences representing the plant damage states for the accident progression, containment response, and source term analysis has been postponed until the results of Phase III are available so that the selection of sequences can be based on the final results and not on the preliminary results. Should a follow-on Level 2 analysis be performed, these sequences will be much more meaningful and will represent full continuity to the Level 1 PRA. This section will be completed on the completion schedule for Phase III.

TABLE C.3-1. PLANT DAMAGE STATES - DCPRA

RCS Pressure at Vessel Melt-Through	RWST Injection or Equivalent			Containment Isolation and Bypass Status													
				Containment Isolated, Not Bypassed				Containment Not Isolated						Containment Bypassed			
								Leak < 3-Inch Diameter				Leak > 3-Inch Diameter		Unisolated SGTR		V-Sequence	
	Before VMT	After VMT	None	CHR and FPR	CHR	FPR	None	CHR and FPR	CHR	FPR	None	FPR	No FPR	CHR	No CHR	CHR	No CHR
				A	B	C	D	E	F	G	H	I	J	K	L	M	N
Low < 300 psia	No	No	Yes 1	1A	X	X	1D	1E	X	X	1H	X	1J	1K	1L	1M	1N
	Yes	--	-- 2	2A	(2D)	(2D)	2D	2E	2H	(2H)	2H	(2J)	2J	2K	2L	2M	2N
High > 300 psia	No	No	Yes 3	3A	X	X	3D	3E	X	X	3H	X	3J	3K	3L	3M	3N
	No	Yes	-- 4	4A	(4D)	(4D)	4D	4E	4H	(4H)	4H	(4J)	4J	4K	4L	4M	4N
	Yes	--	-- 5	5A	(5D)	(5D)	5D	5E	5H	(5H)	5H	(5J)	5J	5K	5L	5M	5N
Success Criteria				FC or Spray Recirculation Cooling	Sump Cooling without Spray. No FC.	Spray Recirculation without Sump Cooling. No FC.	None	As "A"	As "B"	As "C"	As "D"	Spray Injection and Recirculation	None	FC or Spray Recirculation Cooling	None	FC or Spray Recirculation Cooling	None

NOTES:

1. VMT = Vessel Melt-Through
2. CHR = Containment Heat Removal
3. FPR = Fission Product Removal
4. X = Plant Damage State Not Possible or Not Modeled
5. ( ) = Not Modeled; Conservatively Assigned to the Plant Damage State Indicated in Parenthesis
6. FC = Containment Fan Coolers

TABLE C.3-2. PLANT DAMAGE STATE SYSTEM SUCCESS CRITERIA

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State	Success Criteria
1	Large or medium LOCA; no injection.
2	Large or medium LOCA, with injection.
3	Small LOCA or general transient, no high pressure injection, and no low pressure injection.
4	Small LOCA or general transient, no high pressure injection, and with low pressure injection.
5	Small LOCA or general transient, with high pressure injection.
	Containment Isolated and Intact
A	Two of five fan coolers or one of two spray recirculation trains with one of two RHR recirculation cooling trains.
B	One of two RHR recirculation cooling trains for sump cooling. No fan coolers.
C	One of two spray recirculation trains without sump cooling. No fan coolers.
D	None.
	Containment Not Isolated; < 3-Inch Leak
E	Two of five fan coolers or one of two spray recirculation trains with one of two RHR recirculation cooling trains.
F	One of two RHR recirculation cooling trains for sump cooling. No fan coolers.
G	One of two spray recirculation trains without sump cooling. No fan coolers.
H	None.
	Containment Not Isolated; > 3-Inch Leak
I	Spray injection and recirculation.
J	None.
	Containment Bypassed
K	Unisolated steam generator tube rupture with two of five fan coolers or one of two spray recirculation cooling trains available.
L	Unisolated steam generator tube rupture, no fan coolers, and no spray recirculation cooling.
M	Interfacing systems LOCA (V-sequence) with two of five fan coolers or one of two spray recirculation cooling trains available.
N	Interfacing systems LOCA (V-sequence), no fan coolers, and no spray recirculation cooling.

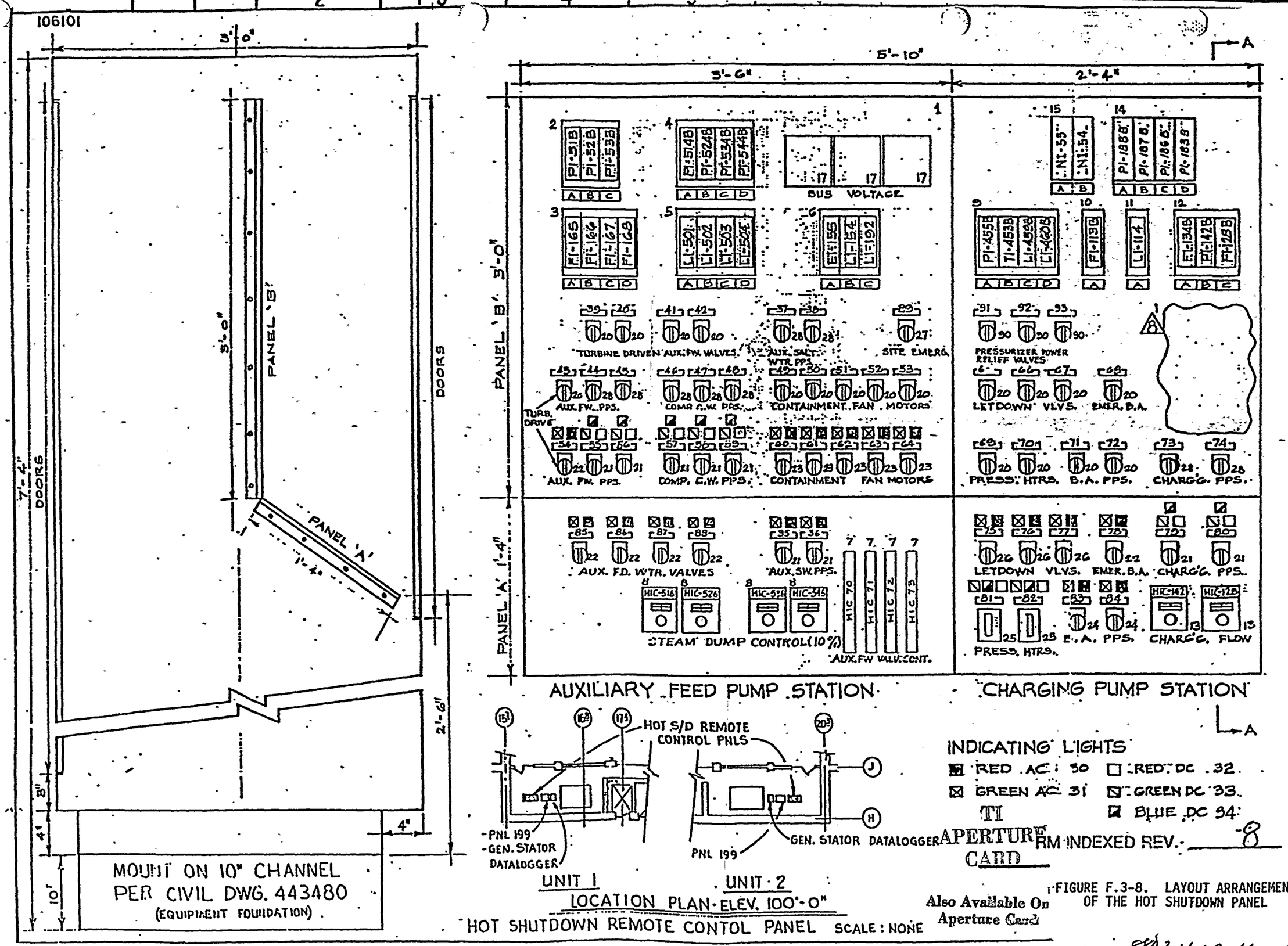

FIGURE F.3-8. LAYOUT ARRANGEMENT OF THE HOT SHUTDOWN PANEL

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F502129

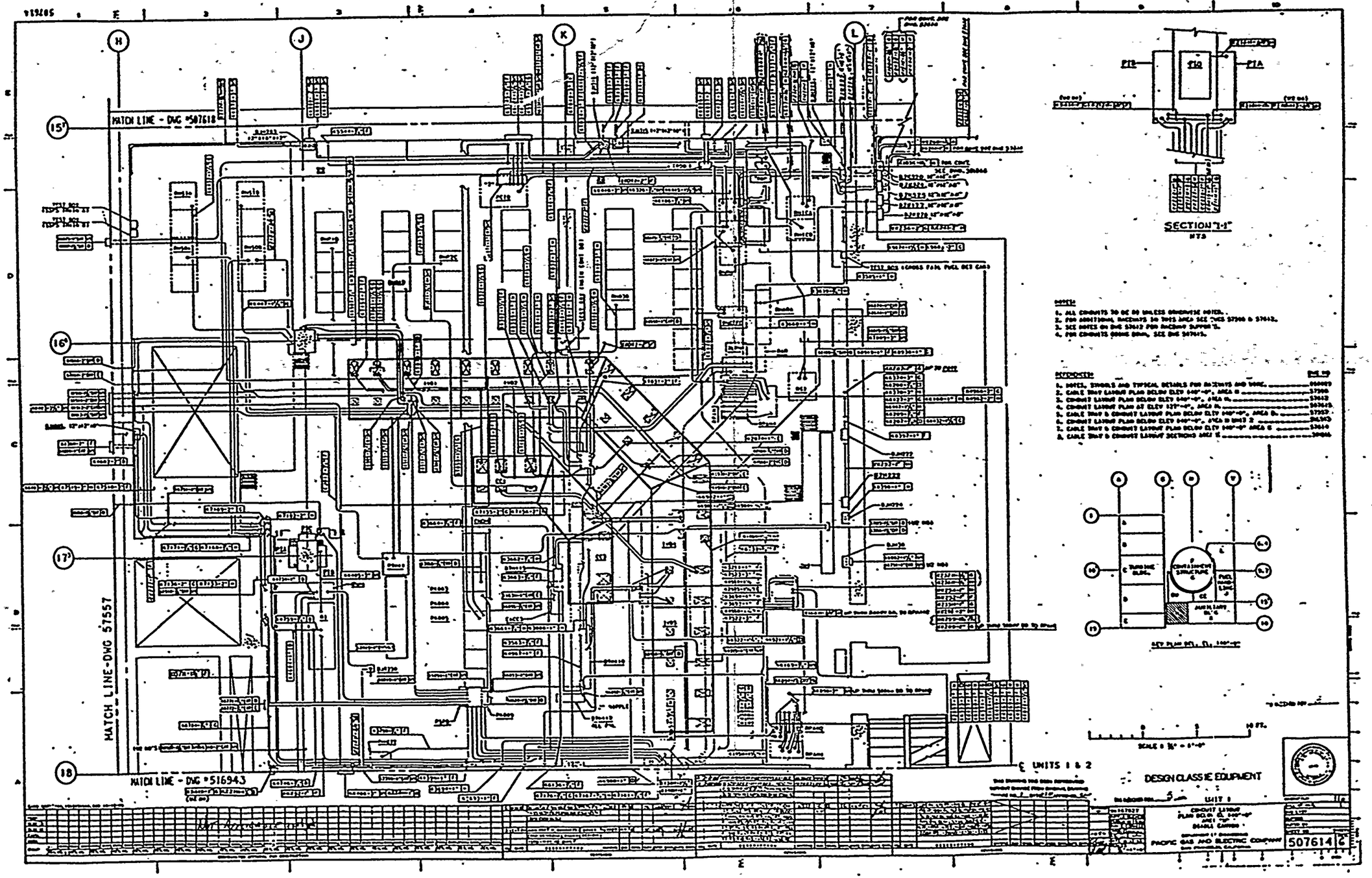

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FIGURE F.3-9. CONDUIT LAYOUT OF THE CABLE SPREADING ROOM

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