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Office of the Nuclear Reactor Regulation

Subject: Submittal Supporting Accelerated ABWR Review Schedule - **ABWR SSAR CET Material**

Dear Chet:

Enclosed is a composite of the latest ABWR containment event tree material. For the most part, this material will replace the information currently contained in Section 19D.5 of the SSAR.

This material is annotated to indicate the source of the initial submittal. Double change bars indicate something not currently reviewd by the staff.

Please provide copies of this transmittal to Bob Palla and John Monninger.

Sincerely,

Jack Fox

Jack Fox
Advanced Reactor Programs

cc: Carol Buchholz (GE)
Norman Fletcher (DOE) w/o enclosure

See attached dist

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19D.5 ABWR CONTAINMENT EVENT TREES

19D.5.1 Overview

The accident sequence event trees described in Section 19D.4 model the event progression for the various accident initiators, and provide the classification and frequency of accident sequences. In these event trees, the sequences which are terminated safely without core damage are designated as "OK". The event sequences which are not successfully terminated could either directly lead to core damage or in some cases could lead to containment structural failure which in turn could lead to core damage. These event sequences are "binned" into various accident classes depending upon the expected event progression, timing and mode of containment structural failure and the amount of fission product release to the environment.

There are five basic classes (I through V), and a total of eleven classes including subclasses such as IA, IB, IC, etc. A Class IA event, for example, is a transient event with loss of high pressure water makeup systems followed by a failure to depressurize the reactor.

Generally, the event progressions for each of these classes of event are modeled in the containment event trees (CET). The CETs model recovery actions which could prevent core damage or arrest core damage if already initiated. Where recovery actions are unsuccessful, the CETs model core melt leading to reactor vessel rupture, containment structural failure and fission product release to the environment. The CET models are based on core-melt progression analysis discussed in Section 19E.2. The mode and location of containment structural failure is modeled based on a study of the containment capability discussed in Appendix 19F.

There is one CET for each of the accident classes. The end states of CETs are either states with insignificant or no release (i.e. core damage prevented or core melt arrested), or states with a release path to the environment resulting from the structural failure of the containment. Associated with each release path in each of the containment event trees, is a frequency of occurrence and a magnitude of fission product release. The frequencies are calculated by the CETs, and the fission product releases are evaluated using the fission product transport analysis discussed in Section 19E.2. The numerous release paths can be consolidated or "binned" into release categories by grouping them based on the expected amount of fission product release to the environment.

The consolidated release categories and the associated frequencies are used as input to the consequence analysis discussed in Section 19E.3.

19D.5.2 Accident Classes

In Section 19D.4 accident event trees are developed for each of the initiators. The end states of these accident event trees are "binned" (grouped) into five basic accident classes based on similarities in the subsequent core melt event progression and the containment response. The key factors that influence the definition of the accident classes are as follows:

(1) Type of initiating event (transient, LOCA, etc.).
(2) Relative times of core melt and containment structural failure.
(3) Whether suppression pool is bypassed.

The type of initiating event is significant because it determines the speed of the event progression. For instance, when no core cooling is available, core melt occurs faster for the LOCA event than for the transient event because of the faster depletion of the coolant inventory,.

The relative times of core melt and containment structural failure are important because if core melt occurs first, the time between core melt and containment failure is available for decay of radioactive material released in the accident. This time is also available for enabling the operator to recover failed water makeup systems in order to get water on top of the molten core or to regain suppression pool

cooling if it had been lost.

The significance of the containment bypass event is that following core melt the fission products are released to the environment without the beneficial effects of passing through the suppression pool.

Five basic accident classes, I through V, have been identified. A description of these five classes is provided below and is summarized in Table 19D.5-1.

Class I: Most Class I events are transients with failure of core cooling systems. In these cases, core melt starts about one hour after event initiation and the RPV fails about one hour later. Following RPV failure, a mixture of molten core material and other metals, called corium, leaves the RPV and comes in contact with the concrete on the drywell floor. The corium-concrete interaction produces steam and non-condensable gases (such as hydrogen, CO and CO_2) and the containment eventually fails by slow over-pressurization.

Event progressions for ATWS events with failure of core cooling systems are similar and are also considered as Class I events.

Class II: Most Class II events are transients with successful core cooling, but with failure of the containment heat removal systems. The suppression pool heats up and the containment pressure builds up slowly until the containment overpressure protection system (COPS) set point is reached in about 24 hours. This period is available to the operator to try to recover the failed systems. If the COPS fails to actuate, structural failure of the containment could occur affecting the core cooling function. Consideration of this possibility is included in the containment event trees.

Loss of core cooling leads to core melt and RPV failure. Class II core damage sequences are thus characterized by containment structural failure followed by core melt. At the time of core melt, the containment is in a failed state and the fission products are released to the atmosphere without the benefit of residence time in the containment.

Event progressions for LOCAs with successful core cooling and ATWS events with successful boron injection and successful core cooling but with failure of the containment heat removal system are similar to the event progression described for transient events and these events are also considered Class II events.

Class III: LOCAs with loss of core cooling are Class III events. As in the case of Class I events, Class III events are also characterized by core melt followed by containment failure. However, because of the loss-of-coolant accident, core uncovery, core melt and RPV failure occurs faster than for Class I events.

Class IV: Class IV events are ATWS events without boron injection but with core cooling available. Under these conditions the reactor continues to produce up to 20% power. The steam produced in the reactor is routed to the suppression pool through the safety relief valves. If this situation continues unmitigated, the containment is overpressurized leading to rupture disk opening or structural failure. As discussed under Class II events, structural failure could lead to loss of core cooling function. This in turn could result in core melt. In summary, Class IV core damage events are characterized by fast containment structural failure followed by core melt.

Following core melt, fission products are released to the environment without radioactive decay due to holdup within the containment.

Class V: Class V events are events in which the suppression pool is bypassed. There are two types of Class V events. In the first, the pool is bypassed at the beginning of the event. An example of this type of event is a LOCA outside the containment. If the break is not isolated and if core cooling is unavailable, core melt will result, and the fission products will be released directly to the atmosphere without going through the suppression pool. The second type of Class V event consists of accidents in which the suppression pool is bypassed during the course of the accident. An example of this is the drywell rupture following core melt and RPV failure.

19D.5.3 Accident Subclasses

19D.5.3.1 Class I Events

The accident Class I is further divided into four subclasses, IA through ID, as discussed below. A summary of the differences is provided in Table 19D.5-2.

Class IA events are characterized by high RPV pressure when the core melts. These are transient events followed by failure of high-pressure water makeup systems coupled with failure to depressurize the reactor (ADS failure, for example). The subsequent core melt event is called the high-pressure core melt. The core melt and RPV failure could result in ejection of molten corium at high pressure into the drywell, which could increase the potential for drywell failure. On the other hand, RPV failure would depressurize the reactor making the low-pressure systems available for flooding the molten core.

Class IB events are broken into three categories.

Class IB-1 events are station blackout events with RCIC failure. Neither core cooling nor containment heat removal is available in the beginning and the core melt starts. However, on site power is recovered in eight hours which increases the likelihood of a core melt arrest and recovery of containment heat removal system. If core melt is not arrested and containment heat removal is not recovered then the containment structure fails on over pressure after 20 hours. Core melt arrest is discussed in Subsection 19D.5.8.

Class IB-2 events are a special class of Station Blackout events. The RCIC is available for core cooling for about eight hours, after which it is assumed to be unavailable.

The suppression pool continues to heat up when RCIC is in operation. This impacts the time of containment structural failure and the time available for decay of fission products released during the accident.

Class IB-3 events are similar to Class IB-1 events except that on-site power is not recovered in eight hours. This leads to core melt and increased likelihood of containment structural failure.

Class IC events are ATWS events without boron injection coupled with loss of core cooling. Core melt occurs faster than it does for other Class I events.

Conservatively, ATWS events with successful boron injection but with loss of core cooling are also included in this subclass.

Class ID includes low-pressure core melt events. These are transients followed by loss of high pressure core cooling, successful reactor vessel depressurization, and loss of low pressure core cooling. Following core melt and RPV failure, the molten core falls on the drywell floor. Unlike the Class IA event, low-pressure systems are not readily available to flood the molten core.

19D.5.3.2 Class II Events

Past analyses have shown that, as long as the core is kept covered with water, the containment response (especially the time required for containment structural failure) is relatively independent of the type of initiating event. Therefore Class II events have not been divided into sub-classes.

19D.5.3.3 Class III Events

Theoretically, Class III events could be sub-divided like the Class I event with four classes - A, B, C and D. However, Sub-Classes B and C which would represent LOCA coincident with loss-of-offsite power and LOCA coincident with ATWS are events with negligible frequencies of occurrence and negligible contribution to risk. These are therefore grouped as part of Class IIIA events.

Class IIIA events are small or medium LOCAs with failure of high pressure coolant makeup systems followed by failure to depressurize the reactor. The low pressure coolant systems may be available but cannot inject water into the reactor because of the high reactor pressure. Core melt occurs with the reactor at high pressure. The core melt and subsequent RPV failure could result in ejection of molten corium at high pressure into the drywell, which could increase the potential for drywell failure. On the other hand, RPV failure would depressurize the reactor making the low-pressure systems avail-

able for flooding the molten core. A large LOCA is not a Class IIIA event because the break depressurizes the reactor.

Class IIID events are LOCAs (small, medium or large) followed by failure of both the high pressure and low pressure coolant makeup systems. The reactor vessel is depressurized by the large LOCA or by the depressurization function for the small and medium LOCA. Following core melt and RPV failure, the molten core falls on the drywell floor. Unlike Class IIIA events, low pressure coolant makeup systems are not readily available to flood the molten core.

19D.5.3.4 Class IV Events

Class IV events are low probability events characterized by fast containment overpressurization (and failure) and it is judged that further sub-classification of this event is not necessary.

19D.5.3.5 Class V Events

Theoretically, there could be pool bypass events associated with each of the four accident Classes I through IV. However, past PRAs have shown that frequencies of pool bypass events and their contribution to plant risk are low, and it is reasonable to group them all under one class without dividing them into subclasses.

19D.5.4 Equipment Recovery

Recovery of the following systems or functions has been modeled in the containment event trees:

Core Cooling
Containment Heat Removal
On-site Power (includes diesel generators)
Off-site Power

Equipment recovery is achieved through component repair. Typical repairs are fuse replacement, valve operator replacement, pump or motor replacement, etc.

System recovery probabilities are calculated using the exponential recovery formula:

$$P_f = \text{Exponential}(-T/MTTR)$$

where

P_f = Probability of failure to recover

T = Available repair time

$MTTR$ = Mean time to repair

A mean time to repair of 19 hours based on the WASH-1400 data, was assumed for the repair of most system components as long as the core and the RPV are intact.

For events involving loss of offsite power or station black out, the MTTR was based on recovery of on-site or off-site power.

For systems involving (multiple) redundant divisions of equipment, there is a potential for recovery of each of the failed divisions. For instance, if all three RHR loops failed, there is a potential that any one of them can be recovered (P_f), and the probability of failing to recover can be modeled as $P_f \times P_f \times P_f$. However, because of potential for common cause failure and limitations on the number of available operators etc, it was judged that the probability of failure to recover the failed function would be taken as half the value calculated for a single system (ie $0.5 \times P_f$) and not $P_f \times P_f \times P_f$.

In the accident fault and event trees, the successful operation of the RHR was treated very conservatively. If the system failed to operate as designed then the branch of the tree was designated as failure. However, in the ABWR the most common failures of the RHR result in degraded performance or delayed initiation. For these cases, no repairs of the system are required to allow successful core cooling.

As an example, the presence of a LOCA signal is deemed a failure of supression pool cooling because this signal causes flow to be directed to the vessel. However, in the ABWR the valves are normally aligned such that the heat exchanger removes heat from the containment even when the RHR is in LPFL mode. Therefore, containment heat removal is successful.

A similar failure mode used in the fault and

accident event trees was common mode failure of a flow transmitter which could result in the minimum flow bypass line remaining open. Were this to happen about 10% of the RHR flow would bypass the heat exchanger. However, because the decay heat falls to a fraction of the RHR capacity before containment structural failure occurs, the degraded system is still capable of adequate containment heat removal.

A third type of failure considered in the fault trees and accident trees was failure of the operator to initiate the system. Failure to initiate could occur either due to human error or a failure of the suppression pool temperature alarm. However, because there is a very long time to containment structural failure the probability of recovering from human error is very high. Furthermore, in the long term, high drywell pressure will provide a diverse signal indicating the need for containment heat removal.

In order to determine the effects of these types of conservatisms, the fault trees were reevaluated with the appropriate nodes deleted. It was found that a recovery factor of 0.5 was appropiate.

When the remaining failure modes were examined it was found that the vast majority involved failures in the pump or valve rooms. If the pumps did not run after core damage began then the radiation levels would be less than 10 Rem/hr. Although this value somewhat high, pump and valve rooms are still accesible. Therefore. the time available for RHR recovery is the time to containment structural failure. In order to represent the affects of radiation in the pump room the failure to recover heat removal was multiplied by 2.

The time available for repairing or recovering each system was determined by the time within which the system had to be operating to prevent the occurrence of failure (core melt, containment overpressure, etc.). The available repair times were obtained based on the core melt progression analysis discussed in Section 19E.2.

19D.5.5 Containment Capability

The ABWR containment design pressure is 45 psig. Past stress analyses performed for other PRAs have shown that the containments are capable of withstanding much higher pressure (typically 2-3 times the design pressure). A discussion of the ABWR containment capability is provided in Appendix 19F. The ultimate pressure capability of the ABWR containment is limited by that of the drywell head. The drywell pressure capability depends upon the temperature in the containment. At 500°F, the containment ultimate strength is evaluated to be 134 psig.

19D.5.6 Containment Structural Failure Modes And Locations

In recent years, many PRAs have focused on the issue of containment performance following a severe accident. Of special interest are events with early loss of containment structural integrity or suppression pool bypass, and events involving large releases of radioactivity. In the case of non-inerted contain-

ments, hydrogen generation and potential for subsequent hydrogen detonation are also of special interest. The ABWR containment is inerted.

19D.5.6.1 Containment Structural Failure Modes

In Appendix 19F it is concluded that when the containment is pressurized, the most likely mode of failure is the plastic yield of the drywell torispherical dome. Containment rupture which impairs the ability of the containment to provide structural support is not judged to be a credible mode of failure.

Containment leakage at pressures below the failure pressure is judged to be not significant (i.e. not sufficient to depressurize the containment). A certain amount of leakage is considered acceptable during plant operation since the risk associated with such leakage is judged to be acceptable. These acceptable limits are specified in the plant Technical Specifications. The amount of leakage increases as the containment is pressurized, but is not judged to be significant. (See Subsection 19E.2.3.4 for additional discussion.) However, if the temperature exceeds 500^{O}F, there is a potential for degradation of seals in the large operable penetrations such as the drywell head, equipment hatches and personnel airlocks. Even with this high temperature degradation of these seals, a conservative evaluation shows that leakage is expected to occur only when the containment pressure exceeds 52 psig (Appendix 19F).

Conservatively, in this PRA, containment structural integrity is judged to be breached if the containment pressure exceeds 134 psig, or the containment pressure exceeds 52 psig and the containment temperature exceeds 500^{O}F. In addition, under some conditions, exposure to very high temperatures could reduce the structural capability of the containment [see Subsection 19D.5.6.3(b)].

19D.5.6.2 Containment Failure Location & Probabilities

Based on a review of results of tests performed on sealed containment models, it is concluded in Appendix 19F that the most likely failure location is the drywell head. The ABWR containment event trees are developed such that the overpressurization events result in either drywell failure or opening of the COPS. The probability of drywell head failure is based on the uncertainty distributions for the head ultimate stength and the COPS as described in Subsection 19E.2.8.1.1. The sequences in which a combination of high temperature and pressure results in a containment leakage, the leakage is assumed to occur in the large drywell penetrations (e.g. equipment hatch) with a failure probability to 1.0.

19D.5.6.3 Failure Modes Explicitly Modeled in Containment Event Trees

The following containment failure modes are explicitly modeled in the CETs:

(1) Containment Overpressurization

Containment fails in the drywell when subjected to high pressure resulting from steam and non-condensable gases.

(2) Containment Leakage

Containment seals (such as the drywell head seal) fail when subjected to a combination of high temperature and pressure (500^{O}F and 52 psig).

(3) High Temperature Failure

When subjected to a very high temperature (e.g. greater than 700^{O}F) the drywell structural capacity is reduced due to reduction of material strength.

(4) Containment Failure at the Time of RPV Failure

Containment fails when the RPV fails due to factors such as vapor suppression failure, missile generation, etc.

19D.5.6.4 Failures Modes Not Explicitly Modeled in Containment Event Trees

(1) Steam Explosion

In-vessel and ex-vessel steam explosion leading to containment failure are not credible events as discussed in Subsection 19E.2.3.1 and Attachment 19EB. Therefore, they are not explicitly modeled in the CETs.

(2) Hydrogen Detonation

The ABWR containment is inerted during plant operation and therefore, failure modes

relating to hydrogen burning and detonation have been ruled out as having a negligible probability of occurrence. The risks associated with the small fraction of time (< 1%) of ABWR plant operation when the containment is not inerted is negligible, since these are associated only with the plant startup or shutdown process, and inerting can be restarted if an accident is initiated. There is a potential for hydrogen combustion in the reactor building, following the release of gases after the containment fails. Since the containment structural failure directly results in suppression pool bypass in the ABWR CETs, special modeling of hydrogen combustion was considered not necessary.

(3) RPV Rupture

RPV rupture, an initiating event which could potentially cause a structural failure of the ABWR containment, is judged to be a negligible contributor to risk.

(4) Basemat Penetration

Basemat penetration following core melt is not expected to result in the release of radioactive materials to the environment (Section 19E.2).

19D.5.7 Suppression Pool Bypass

19D.5.7.1 Introduction

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The magnitude of radioactive release to the environment for the severe accidents in which the suppression pool is bypassed is much higher than the severe accidents in which the release occurs through the suppression pool. Thus, suppression pool bypass paths are of special interest in BWR PRAs. This subsection discusses the various types of suppression pool bypass paths and describes how they are treated in the ABWR PRA. Some of these bypass paths are explicitly modeled in the CETs. Others have been studied separately in Subsection 19E.2.3.3 and found to be negligible contributors to ABWR plant risk. A summary of the various suppression pool bypass mechanisms and how they are treated in the ABWR PRA is provided in Table 19D.5-3.

19D.5.7.2 Ex-Containment LOCA

A LOCA can occur in the high pressure system piping which runs outside the containment. Rupture of high pressure piping in any of the following systems with a failure to isolate the break could result in this type of LOCA:

(1) Main steam lines
(2) Feedwater lines
(3) RCIC steam lines
(4) RWCU system lines

Another type of ex-containment LOCA is the interfacing systems LOCA where a low pressure system piping connected to a high pressure system is accidentally overpressurized resulting in the rupture of the low pressure piping. This, coupled with a failure to isolate, results in the interfacing systems LOCA. In ABWR, potential for interfacing LOCA has been essentially eliminated by increasing the design pressure of low pressure piping connected to higher pressure piping.

In each of the above cases, if the ex-containment LOCA is not mitigated, core melt could result and the radioactive material would be discharged to the environment directly through the break, bypassing the suppression pool.

This type of suppression pool bypass path has been studied as part of the suppression pool bypass study in Subsection 19E.2.3.3 and it was determined that this is not a significant contributor to ABWR plant risk. Therefore, this bypass path is not modeled in the CETs.

19D.5.7.3 Failure of Isolation Valves and Pipe Ruptures

There are other combinations of failure of isolation valves and rupture of pipes which could lead to suppression pool bypass. But unlike the ex-containment LOCA discussed above, these are by themselves not accident initiators but provide a path for bypassing the suppression pool for severe accidents which are initiated through other initiators. An example of this type of suppression pool bypass path is the one resulting from the rupture of RCIC piping, failure of RCIC discharge valve and operator inability to isolate the break.

This type of bypass paths is analyzed in detail in Subsection 19E.2.3.3 and it was determined that this is not a significant contributor to ABWR plant risk. Therefore, this bypass path is not modeled in the CETs.

19D.5.7.4 Failure of Drywell Vacuum Breaker

If the containment drywell vacuum breaker is in a failed state during a core melt event that leads to RPV failure, the suppression pool is bypassed. This bypass path has been studied as part of the suppression pool bypass study in Subsection 19E.2.3.3 and it was determined that could be a significant contributor to ABWR plant risk. Therefore, this bypass path is modeled in the CETs.

19D.5.7.5 Containment Structural Failure

The most likely structural failure of the containment occurs in the drywell. This failure mode bypasses the suppression pool and is modeled in the CETs. As discussed in Subsection 19D.5.6.3, two additional containment failure modes are modeled in the CETs. These also result in suppression pool bypass.

19D.5.7.6 Uncovery of Horizontal Vents

If after the RPV failure, the horizontal vents are uncovered due to low water level in the suppression pool, the pool will be bypassed. Calculations show that for all events other than ATWS, initial suppression pool inventory is sufficient to compensate for the evaporation loss for over 24 hours without uncovering the horizontal vents. It is assumed that suppression pool make up will be initiated within 24 hours (by using fire trucks if necessary) for each of these accidents. Therefore this type of suppression pool bypass is not explicitly modeled in the CETs.

19D.5.7.7 Low Probability Bypass Events

Some events such as RPV rupture, and in-vessel steam explosion, with extremely low probability of occurrence have the potential for causing suppression pool bypass. These are not specifically treated in the CETs because past PRAs have shown negligible contribution to risk attributable to these events. For references which provide additional details see Table 19D.5-3.

19D.5.8 Core Melt Arrest Success Criteria

19D.5.8.1 Introduction

After core melt has been initiated, the process can still be arrested if the core debris is cooled with sufficient water. The success criteria for arresting core melt is described in this subsection. The analytical basis for the success criteria is developed in Subsection 19E.2.1.4.3.

There are two ways to arrest core melt: 1) during the early stages of an accident core melt can be arrested prior to RPV failure, 2) if the RPV has been breached, core melt can be arrested prior to loss of structural integrity of the containment.

In each case, a means of getting water to the corium and a means of removing heat from the containment are required.

The core melt arrest success criteria is summarized in Table 19D.5-4.

19D.5.8.2 Core Melt Arrest Prior to RPV Failure

For arresting core melt within the RPV, one of the core cooling systems must be recovered within about an hour of the core melt initiation.

If the reactor is at high pressure, operation of one of the high pressure systems (HPCF B or C, RCIC, feedwater system) must be restored.

If the reactor is at low pressure, in addition to the high pressure system, operation of one of the low pressure systems (LPFL, condensate injection) may be recovered. Alternately, arrangement could be made to get the diesel driven fire water into the reactor.

CRD water supply is assumed to be not sufficient to arrest core melt.

For removing the heat from the containment, one of the RHR loops must be available.

19D.5.8.3 Core Melt Arrest Prior to Loss of Containment Structural Integrity

Following RPV failure, the molten core drops on the drywell floor. Core melt can be arrested by operating one of the two HPCF or any one of the low pressure systems (LPFL or condensate injection to the reactor vessel or diesel driven fire water system).

CRD water supply is assumed to be not sufficient for arresting core melt.

If none of the low pressure systems can be recovered in time to quench the corium on the lower drywell floor, the corium continues to heat up the lower drywell area. This melts the fuse at the ends of pipes in the passive flooder system resulting in the transfer of the suppression pool water to the lower drywell area. This passive flooder system is described in Subsection 9.5.12. The suppression pool water quenches the molten corium and the core melt process is arrested. This process is modeled in the CETs by the node P, representing "passive mitigation". It should be noted that even after passive flooder operation, the suppression pool water level stays high enough to cover the horizontal vents.

After the core melt is arrested, it is still necessary to remove heat from the containment. Containment heat can be removed by operation of one of the RHR systems. If the core melt is arrested but the RHR is not available, radioactivity is eventually released to the environment.

19D.5.9 Containment Release Categories

The amount of radioactive release to the environment depends upon a number of factors such as the timing of containment failure and the location of containment failure. Ideally, there is a specific radioactive release associated with each outcome of the containment event trees. However, evaluating the source terms for each event tree output is very time consuming. Therefore, the releases with similar characteristics are grouped ("binned") together to define release categories.

Detailed discussion of the binning process is provided in Subsection 19D.12.5.

19D.5.10 Containment Overpressure Protection

Subsection 6.2.5 describes a mitigation system called the overpressure protection subsystem of the atmospheric control system. This system protects the containment structural integrity and provides for controlled fission product release. If the containment exceeds its service level C limit, a rupture disk opens providing containment pressure relief. Since the system originates in the wetwell airspace, any fission product release will be scrubbed. The operation of COPS is indicated by "RD open" in the LCS node of the CETs.

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19D.5.11 Description of Containment Event Tree

19D.5.11.1 Subdivision of Accident Classes

Several of the accident subclasses were further subdivided to reduce the size of the individual trees. Each tree contains only sequences with high or low RPV pressure at the time of core damage, and the availability of RHR is known (Figure 19D.5-2). This subdivision was assigned to each accident subclass. Table 19D.5-5 summarizes this subdivision.

For event classes IB_1, IB_2, and IB_3, the status of successful depressurization is not known as a direct result of the accident event trees. The depressurization system in the ABWR is automatic and does not rely on AC power. The operability of the ADS system is discussed in Subsection 19E.2.1.2.2, where it is shown that there is adequate DC power and nitrogen supply to actuate the ADS system during a blackout event. As a backup to the automatic acuation, the emergency procedure guidelines (EPGs), contained in Appendix 18A, require the operator to manually initiate the ADS system when the water level reaches the top of active fuel. Since the EPGs are symptom based, there is no differentiation between station backout events and other events. Thus, there is no difference in the reliability of ADS for station backout events as compared to other transient events. Therefore, the probability of successful depressurization used in the development of Table 19D.5-5 is the same value as that used in the accident event trees.

As Table 19D.5-5 shows, the probability of several events is extremely low. Based on these very low frequencies, the IB1_1, IB2_1, IB3_1 and IV trees are not developed further. The IIIA_1 tree could have been similarly neglected. However, since the quantification is the exact analog of the IA_1 tree, it was developed.

19D.5.11.2 Level 2 Results

The logic diagram shown in Figure 19D.5-3 groups the set of Level 2 sequences into source term categories (STC) based on similar sequence characteristics judged to be important to the definition of the offsite source term and consequences. Five parameters are used to define the source term categories. This grouping resulted in the definition of 53 distinct source term categories. The characteristics of each source term category are determined by the branch attributes for the pathway through the diagram. The five grouping parameters are discussed below.

19D.5.11.2.1 Initiator Code (INITCODE)

This parameter groups the sequences based on the accident sequence type. The accident sequence type definition is described in Section 19E.2.2. Note that sequence types NSCL (class IC) and NSCH (class IE) were of such low probability that they were truncated prior to performing the Level 2 analysis and are not included in the grouping diagram. In addition, as a result of the low probability of Class IV ATWS sequences (sequence type NSRC), they were not evaluated in the Level 2 model although the Class IV frequency is shown in the logic diagram (STC 53).

19D.5.11.2.2 Core Melt Arrested In-Vessel (IV)

This grouping parameter groups sequences based on whether late in-vessel cooling is successful in preventing vessel failure.

19D.5.11.2.3 Mode of Release (REL MODE)

This parameter groups sequences based on the mode of any fission product release from the containment. The following important characteristics are considered.

19D.5.11.2.3.1 Normal Containment Leakage

Containment pressurization is terminated, so there is containment failure or COPS operation. These sequences have very small releases to the environment as a result of normal containment leakage.

19D.5.11.2.3.2 Rupture Disk

Operation of the COPS leads to nearly completerelease of the noble gases. Other fission product releases are negligible.

19D.5.11.2.3.3 Drywell Head Failure

Long-term steam and non-condensible gas production lead to over pressurization of the containment. The drywell head fails before the COPS opens.

19D.5.11 Description of Containment Event Tree

19D.5.11.1 Subdivision of Accident Classes

Several of the accident subclasses were further subdivided to reduce the size of the individual trees. Each tree contains only sequences with high or low RPV pressure at the time of core damage, and the availability of RHR is known (Figure 19D.5-2 and Table 19D.5-5). This subdivision was assigned to each accident subclass. Table 19D.5-5 summarizes this subdivision.

For event classes IB_1, IB_2, and IB_3, the status of successful depressurization is not known as a direct result of the accident event trees. The depressurization system in the ABWR is automatic and does not rely on AC power. The operability of the ADS system is discussed in Subsection 19E.2.1.2.2, where it is shown that there is adequate DC power and nitrogen supply to actuate the ADS system during a blackout event. As a backup to the automatic actuation, the emergency procedure guidelines (EPGs), contained in Appendix 18A, require the operator to manually initiate the ADS system when the water level reaches the top of active fuel. Since the EPGs are symptom based, there is no differentiation between station blackout events and other events. Thus, there is no difference in the reliability of ADS for station backout events as compared to other transient events. Therefore, the probability of successful depressurization used in the development of Table 19D.5-5 is the same value as that used in the accident event trees.

As Table 19D.5-5 shows, the probability of several events is extremely low. Based on these very low frequencies, the IB1_1, IB2_1, IB3_1 and IV trees are not developed further. The IIIA_1 tree could have been similarly neglected. However, since the quantification is the exact analog of the IA_1 tree, it was developed.

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The logic diagram shown in Figure 19D.5-3 groups the set of Level 2 sequences into source term categories (STC) based on similar sequence characteristics judged to be important to the definition of the offsite source term and consequences. Five parameters are used to define the source term categories. This grouping resulted in the definition of 53 distinct source term categories. The characteristics of each source term category are determined by the branch attributes for the pathway through the diagram. The five grouping parameters are discussed below.

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19D.5.11.2.3.1 Normal Containment Leakage

Containment pressurization is terminated, so there is containment failure or COPS operation. These sequences have very small releases to the environment as a result of normal containment leakage.

19D.5.11.2.3.2 Rupture Disk

Operation of the COPS leads to nearly complete release of the noble gases. Other fission product releases are negligible.

19D.5.11.2.3.3 Drywell Head Failure

Long-term steam and non-condensible gas production lead to over pressurization of the containment. The drywell head fails before the COPS opens.

19D.5.11.2.3.4 Penetration Over-temperature Failure

High temperatures lead to failure of the large penetration seals in the drywell.

19D.5.11.2.3.5 Early Containment Failure

Overpressure failure of the drywell head occurs at the time of RPV failure.

19D.5.11.2.4 Pool Bypass (POOL_BP)

This parameter groups sequences based on whether radionuclides released into the drywell gas space bypass the suppression pool for fission product scrubbing. All drywell containment failure modes result in eventual pool bypass and no branching is required. For sequences without containment failure, this parameter is irrelevant. Hence, branching under this heading is only significant for the COPS release mode.

19D.5.11.2.5 Drywell Spray (SPRAY)

Operation of the drywell sprays can be effective in mitigating the release of radionuclides. However, for sequences where vessel failure has not occurred and sequences where pool bypass has not occurred, operation of the sprays is not significant since suppression pool scrubbing will effectively mitigate the radionuclide releases. Therefore, branching is only considered for sequences with pool bypass. Note that for sequences with drywell penetration overtemperature failure, the drywell sprays are not operating and no branching is necessary.

19D.5.11.3 Containment Event Trees for Classes I and III

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Containment event trees were developed for each of the significant accident classes. These are shown in Figures 19D.5-4 to 19D.5-14.

In order to quantify the containment event trees for each class of accidents, decomposition event trees (DETs) were developed. The DETs depict the logic which was used to determine the branch point probabilities used for each node of the containment event trees. There is a one-to-one correspondence between the nodes on the CETs and the set of DETs. Most of the nodes on the trees are system related. These nodes are described here. A few nodes deal with phenomenological uncertainties. These nodes are described in detail in the uncertainty analysis of Section 19E.2.7.

Each branchpoint on the DET either assigns a probability to the event, or refers by rule to previous events on the CET or to the accident subclass. Assigned probabilities are shown on each branch. Branches which refer back to the CETs are termed sorting events. These are indicated on the tree by the symbol "<==".

19D.5.11.3.1 Operator Depressurizes Reactor (OP)

This DET, shown in Figure 19D.5-15, classifies the sequences based on the RPV pressure at the time of core damage. The only branch on the OP DET is the SUBCLASS event. Each subclass represents sequences which are either at full pressure or have been depressurized. Section 19D.5.11.1 discusses the subdivision of the accident subclass based on the RPV pressure.

Accident subclasses IB1_0, IB2_0, IB3_0 are depressurized based on the accident subclass division discussed in Subsection 19D.5.11.1. All ID and IIID were depressurized in the accident event trees contained in Section 19D.4. Sequences in the remaining accident subclasses have not been depressurized. Accident classes II and IV have not been subdivided into high- and low-pressure subclasses since this information was not necessary for the CET analysis.

This is a sorting type event quantified as 1 or 0 based on the accident subclass.

19D.5.11.3.2 Containment Heat Removal Available (CHR)

The second node of the CET, CHR, and its DET, shown in Figure 19D.5-16, check if containment heat removal was available at the beginning of the accident. Only accident subclasses IA_0 and IIIA_0 have RHR available at the time of core damage.

The CHR event is a sorting type event quantified as 1 or 0 based on the accident subclass.

19D.5.11.3.3 Core Melt Arrested in RPV (ARV)

The ARV node and DET, shown in Figure 19D.5-17, assess the probability that core damage is arrested in-vessel and vessel failure is prevented. In order to prevent RPV failure it is assumed that an in-vessel injection source must be recovered well before the time at which the vessel would otherwise fail. The success criteria for core melt arrest in-vessel are shown

in Table 19D.5-4. The probability of in-vessel core damage arrest varies for different subclasses because of differences in RPV pressure, availability of AC power, sequence timing, RHR availability and other factors.

19D.5.11.3.3.1 Accident Subclass (SUBCLASS)

The first event in the DET segregates the accident subclasses into groups with similar in-vessel recovery probabilities. This event is a sorting type event with an assigned probability of 0 or 1 based on the sequence subclass.

19D.5.11.3.3.2 Core Melt Arrested in RPV (ARV)

This event assigns a probability for in-vessel core melt arrest. Four cases were identified during the quantification for this event:

(1) Case 1 - Subclasses IA_0, IA_1, IIIA_0 and IIIA_1

These subclasses represent high RPV pressure sequences with failure of high-pressure injection. In order to arrest the core damage progression, recovery of a high-pressure injection system is required. As discussed in Subsection 19E.2.4.2, approximately 1 hour is available for system recovery. A recovery probability is calculated assuming a mean time to repair (MTTR) of 19 hours for the failed system:

CEB92-41p.15

Core melt arrest	0.05,
No core melt arrest	0.95.

(2) Case 2 - Subclass IB2_0

CEB92-41p.15 and CEB92-54#10

The subclass contains station blackout sequences with RCIC operation for eight hours and with operator depressurization of the RPV. If in-vessel injection is re-established within about two hours of loss of RCIC, core damage can be arrested and vessel failure can be prevented. If power is recovered, AC powered high- or low-pressure injection systems can cool the core and prevent vessel failure. The conditional probability of recovering power in this 2-hour period is obtained from the EPRI KAG (7/88 Draft) Table D2-2. A value of 0.6 is obtained by dividing the non-recovery probabilities for 8 and 10 hours. In addition, injection using the firewater addition system can also provide late in-vessel core cooling (Subsection 19E.2.2.3). The operator is expected to monitor the availability of DC power during the blackout, so there will be approximately 10 hours of warning time before use of the firewater addition system is necessary. The firewater system is assigned a failure probability of 0.01 based on operator error probability. This yields combined probabilities of:

Core melt arrest	0.994,
No core melt arrest	0.006.

(3) Case 3 - Subclasses ID and IIID

These subclasses contain low RPV pressure sequences with failure of all low-pressure injection. In order to arrest the core damage progression, recovery of a high- or low-pressure injection system or operation of the firewater addition system must occur within 1 hour (Subsection 19E.2.4.2). The probabilities are assigned based on failure of the operator to initiate injection:

CEB92-41p.16

Core melt arrest	0.9,
No core melt arrest	0.1.

(4) Case 4 - All Other Subclasses Excluding Class II

For all other subclasses, AC power is not available and the vessel is at high pressure; therefore, recovery of in-vessel injection in time to prevent vessel failure is not considered:

Core melt arrest	0.0,
No core melt arrest	1.0.

19D.5.11.3.4 Containment Intact at RPV Failure (CI)

This node indicates if the containment survives any energetic events which occur at vessel failure. Since fuel coolant interactions were ruled out as a significant contributor to containment failure (Subsections 19E.2.3.1 and 19E.2.6.7) direct containment heating is the only contributor to this node. The decomposition event tree is shown in Figure 19D.5-18 for completeness. However, the descriptions of the DET events and quantification is given in the uncertainty analysis for DCH in Section 19EA.2.

19D.5.11.3.5 Active Injection to the Lower Drywell (LDWI)

This node and its decomposition event tree, shown in Figure 19D.5-19, assess the probability that an active injection system to supply water to the lower drywell is available at, or soon after, RPV failure. High- or low-pressure in-vessel injection systems which deliver water to the vessel after vessel failure will result in water flowing from the vessel to the lower drywell. In addition, the AC independent firewater addition system can inject water to either the vessel or the upper drywell sprays.

19D.5.11.3.5.1 High-pressure Injection Recovered (HPI)

Recovery of a high-pressure injection system after vessel breach can supply water to the lower drywell. However, recovery of high-pressure injection following vessel breach has been conservatively neglected in this analysis.

19D.5.11.3.5.2 Accident Subclass (SUBCLASS)

This event separates the accident subclasses into groups with similar conditions for low-pressure in-vessel injection availability and firewater spray operation. This event is a sorting type event which has an assigned probability of 0 or 1 based on the sequence subclass.

19D.5.11.3.5.3 Low-pressure Injection Available after RPV Failure (LPI)

This event assesses the probability that low-pressure in-vessel injection will be available after RPV failure. There were two cases identified for the quantification of this event:

(1) Case 1 - Subclasses IA_0 and IIIA_0

These subclasses represent high RPV pressure sequences with failure of the high-pressure injection system and with the RHR system available. For this group of sequences, the probability of operation of low-pressure injection after RPV failure is very high:

Low-pressure Injection Available	0.999,
Low-pressure Inj. Not Available	0.001.

(2) Case 2 - All Other Subclasses

For all other subclasses the RHR system is not available at the onset of core damage, nor was it available for in-vessel recovery. Consequently, for this group of sequences the probability of operation of low-pressure injection after RPV failure was conservatively set to zero:

Low-pressure Injection Available	0.0,
Low-pressure Inj. Not Available	1.0.

19D.5.11.3.5.4 Firewater Injection to Drywell Sprays (FWS)

This event assesses the probability that the operators initiate the firewater injection system in the drywell spray mode following RPV failure. Injection via the sprays will cause the suppression pool level to increase and will eventually cause overflow into the lower drywell. Injection to the vessel provides immediate flooding of the lower drywell. Given the presence of the passive flooder (considered in the next node) there is virtually no sensitivity for the lower drywell to the use of the spray versus vessel injection mode. Therefore, in order to simplify later trees, injection is presumed to occur via the spray system and injection via the vessel is neglected.

The pumps of the AC independent firewater system are continuously charging. Furthermore, the onsite firewater system can be backed up by the use of fire trucks. Therefore, failure to inject is dominated by operator error. Two cases were identified for this event:

(1) Case 1 - All Short-term Core Melt Subclasses

For this case the operator has several hours to successfully initiate the firewater addition system. Although the firewater injection system was not operated quickly enough to arrest the core melt in the vessel, there was only a short time available for that action. Therefore, it is judged highly probable that the operator will properly operate the firewater system:

Firewater spray	0.99,
No Firewater spray	0.01.

(2) Case 2 - Long-term Core Melt Subclass, IB2_0

In this case somewhat more time had been available for the operator to prevent vessel breach

via the firewater addition system. However, there is substantially longer time for successful firewater injection in containment:

Firewater spray	0.95,
No Firewater spray	0.05.

19D.5.11.3.5.5 Active Injection to Lower Drywell (LDWI)

This event has no branching. It simply summarizes the branch decision taken in the previous branches in the DET. Thus, the summary branches are:

(1) In-vessel Injection (LPI provides lower drywell injection),

(2) Firewater Spray (The firewater system injects through the drywell sprays causing the suppression pool to overflow into the drywell),

(3) No DW Injection (Active injection systems do not supply water to the lower drywell).

19D.5.11.3.6 Passive Mitigation (P)

This node, shown in Figure 19D.5-20, assesses the probability that the passive flooder system operates to cover the debris in the lower drywell with suppression pool water after RPV failure. It is assumed for this node that no active injection systems have supplied water to the lower drywell, since this makes operation of the passive flooder unnecessary and the high temperatures necessary for flooder operation will not occur if active injection operates. Since the only requirement for operation of the passive flooder is the melting of the fusible valves near the drywell floor, operation of this system is considered to be extremely likely:

Passive Mitigation	0.999,
No Passive Mitigation	0.001.

19D.5.11.3.7 High-temperature Failure (HTF)

This section describes the decomposition event tree (Figure 19D.5-21) used to assess the probability that high temperature in the upper drywell will result in seal degradation and excessive leakage through the large movable penetrations in the upper drywell. The potential for seal degradation is presumed to exist if the temperature exceeds 500 F. Two situations could lead to this condition.

High-pressure melt ejection (HPME) may entrain significant quantities of core debris into the upper drywell in cases with high RPV pressure. In this situation, operation of the upper drywell sprays is required to assure that the upper drywell temperature remains below 500 F.

For sequences where HPME does not occur, high temperatures may result if the lower drywell is not flooded and the drywell sprays do not operate.

Thus, upper drywell overtemperature failures will be prevented in all cases by operation of the sprays in the upper drywell. Consequently, no branching under the HTF event heading in the CET is made if firewater injection was successful in drywell spray mode (Branch FW SPRAY in CET event LDWI) and the HTF DET is not evaluated for those sequence pathways.

19D.5.11.3.7.1 Accident Subclass (SUBCLASS)

The first event in the DET segregates the accident subclasses into two groups. For all accident subclasses except subclasses IA_0 and IIIA_0, operation of the drywell sprays has already been determined in CET event LDWI. For subclasses IA_0 and IIIA_0 with successful LPI after RPV failure, the question of drywell spray availability was not asked in CET event LDWI since successful lower drywell water addition was already known to exist via the RHR system.

This event is a sorting type event which has an assigned probability of 0 or 1 based on the sequence subclass.

19D.5.11.3.7.2 Operator Depressurizes Reactor (OP)

This event classifies the accident subclasses into high and low RPV pressure at RPV failure. For sequences with high RPV pressure, HPME at the time of RPV failure may result in debris entrainment into the upper drywell.

This event is a sorting type event which has an assigned probability of 0 or 1 based on the branch pathway followed under CET event heading OP.

19D.5.11.3.7.3 Mode of Active Injection to Lower Drywell (LDWI)

This event assesses whether the active injection systems have flooded the lower drywell soon after RPV failure. This event is a sorting type event which has an assigned probability of 0 or 1 based on the branch pathway followed under CET event heading LDWI.

19D.5.11.3.7.4 Drywell Sprays Operate (DW_SPRAY)

This event assesses whether the upper drywell sprays are available. Note that this event is only relevant for sequences in subclasses IA_0 and IIIA_0. For all other sequences entering this DET, failure of the drywell sprays has been previously determined in event LDWI.

(1) Case 1 - Subclasses IA_0 and IIIA_0

Sequences in these subclasses involve high RPV pressures with the RHR system available. Under these conditions it was determined that the operation of the upper drywell sprays would be extremely likely:

Drywell Spray	0.999,
No Drywell Spray	0.001.

(2) Case 2 - All Other Subclasses

No consideration of these cases is necessary since operability of the drywell sprays was determined previously.

19D.5.11.3.7.5 Water Supply to Lower Drywell (LDW)

This event assesses whether the lower drywell is flooded by the active injection systems or the passive flooder after RPV failure. For sequence pathways with success (IN_VESSEL INJ) under CET event heading LDWI, there will be water in the lower drywell and no branching is taken. For sequences with failure of active injection (NO DW INJECT), successful water addition to the lower drywell is determined by the success of the passive flooder (Branch PASSIVE MIT) in CET event P.

This event is a sorting type event which has an assigned probability of 0 or 1 based on the branch pathway followed under CET event heading P.

19D.5.11.3.7.6 High-temperature Failure (HTF)

This event assesses whether high-temperature failure of the large moveable penetrations in the upper drywell occurs. Four cases were identified in the quantification for this event:

(1) Case 1 - Sequences with Drywell Sprays Available

For these sequences drywell high-temperature failure will be prevented:

No high-temperature failure	1.0,
High-temperature failure	0.0.

(2) Case 2 - Low-pressure Sequences with Water Supply to the Lower Drywell

For these sequences, debris entrainment to the upper drywell does not occur. The debris in the lower drywell is submerged in a pool of water. Hence, high temperatures in the upper drywell will be prevented:

No high-temperature failure	1.0,
High-temperature failure	0.0.

(3) Case 3 - Sequences with no Drywell Sprays and no Water Supply to the Lower Drywell

For these sequences, the debris in the lower drywell is not submerged in a pool of water. Hence, core concrete attack in the lower drywell would generate high-temperature gasses which would convect into the upper drywell. Consequently, high temperatures in the upper drywell could be expected:

No high-temperature failure	0.0,
High-temperature failure	1.0.

(4) Case 4 - High-pressure Sequences with no Drywell Sprays and with Water Supply to the Lower Drywell

For these high-pressure sequences the debris in the lower drywell will be submerged in a pool of water. However, if HPME results in debris entrainment into the upper drywell then high temperatures in the upper drywell will occur since

drywell sprays are not available. This event is quantified based on the probability of HPME occurring (see Subsection 19EA.2.1.5 for a further discussion on the probability of HPME):

No high-temperature failure	0.2,
High-temperature failure	0.8.

19D.5.11.3.8 Core Debris Concrete Attack (CCI)

This node indicates if a substantial amount of core concrete attack occurs, and if so, whether the attack occurs in the presence of water. The decomposition event tree is shown in Figure 19D.5-22 for completeness; however, the descriptions of the DET events and quantification is given in the uncertainty analysis for Debris Coolability in Section 19EC.2.1.

19D.5.11.3.9 Pedestal Failure (PED)

This node indicates if the pedestal fails as a result of core concrete attack. If the pedestal fails it is assumed that tipping of the vessel will lead to tearing of the containment penetrations associated with the vessel, allowing fission product release. The decomposition event tree is shown in Figure 19D.5-23 for completeness; however, the descriptions of the DET events and quantification is given in the uncertainty analysis for Debris Coolability in Section 19EC.2.2.

19D.5.11.3.10 RHR Recovered Prior to Fission Product Release (RCH)

This section describes the decomposition event tree which assesses the probability that containment heat removal (primarily RHR) is recovered prior to the release of fission products through the containment overpressure protection system or as a result of drywell head failure (Figure 19D.5-24). The probability of RHR recovery varies for different accident subclasses: for sequences with core damage terminated in-vessel and for sequences with active injection to the lower drywell after RPV failure because of differences in the availability of AC power, sequence timing, and other factors.

19D.5.11.3.10.1 Accident Subclass (SUBCLASS)

The first event in the DET segregates the accident subclasses into groups with similar RHR recovery probabilities. This event is a sorting type event which has an assigned probability of 0 or 1 based on the sequence subclass.

19D.5.11.3.10.2 Core Melt Arrested in Vessel (ARV)

This event sorts the sequences into those with in-vessel core damage progression termination and those where RPV failure occurs. This event is a sorting type event which has an assigned probability of 0 or 1 based on the branch taken under CET event heading ARV.

19D.5.11.3.10.3 Active Injection to the Lower Drywell (L_DW_INJ)

This event classifies sequences into those with active injection into the lower drywell after RPV failure and those without active injection. This event is a sorting type event which has an assigned probability of 0 or 1 based on the branch taken under CET event heading LDWI.

19D.5.11.3.10.4 RHR Recovered Prior to Fission Product Release

Recovery of the RHR system is described in 19D.5.4. The RHR recovery probabilities for the different cases which were identified are summarized in Table 19D.5-6. The probabilities are based on knowledge about the use of active injection, and the time available for recovery. The impact pool bypass could have on the probability of RHR recovery is discussed in 19D.5.13. CE 693.41 p.28

19D.5.11.3.11 Pool Bypass (POOL_BP)

This node indicates if pool bypass occurs. The decomposition event tree is shown in Figure 19D.5-25 for completeness; however, the descriptions of the DET events and quantification is given in the uncertainty analysis for Pool Bypass in Subsection 19EE.2.

19D.5.11.3.12 Late Containment Status (LCS)

This section describes the decomposition event tree, shown in Figures 19D.5-26 and 19D.5-27, used to assess the containment status late in the accident sequence progression. Figure 19D.5-26 is used for cases where the RHR is known to be available. Figure 19D.5-27 is used for cases where RHR is not initially available. The two trees are otherwise identical.

19D.5.11.3.12.6 Rupture Disk Opens (RD)

This event estimates the probability that the rupture disk will open prior to drywell head failure given that the containment pressure exceeds the rupture disk setpoint. The probability that the rupture disk fails to open is dominated by the uncertainties in the drywell head ultimate pressure capability. Consequently, the probability of rupture disk opening is dependent on the drywell-to-wetwell pressure differential and whether or not the suppression pool is bypassed. The probabilities for failure of the rupture disk to open before drywell head failure are developed in Subsection 19E.2.8.1:

(1) Case 1 - Vapor Suppression OK and Firewater System Injects

For this set of sequences the drywell-to-wetwell pressure will be elevated as a result of the increase in the suppression pool depth. Consequently, there is an increased likelihood of drywell head failure prior to the Rupture Disk opening as compared to other sequences:

Rupture Disk Opens	0.95,
Rupture Disk Does not Open	0.05.

(2) Case 2 - Vapor Suppression OK and no Firewater Injection

For this set of sequences the drywell-to-wetwell pressure difference required to clear the horizontal vents is determined by the normal suppression pool level:

Rupture Disk Opens	0.98,
Rupture Disk Does not Open	0.02.

(3) Case 3 - Vapor Suppression Failed

For this set of sequences the drywell-to-wetwell pressures will be essentially the same. Consequently, there is a reduced likelihood of drywell head failure prior to rupture disk opening:

Rupture Disk Opens	0.99,
Rupture Disk Does not Open	0.01.

19D.5.11.3.12.7 Late Containment Status (LCS)

This event has no branching. It simply summarizes the decisions taken in previous branches. Three possible outcomes are considered: the containment is intact, the rupture disk opens, or the drywell head fails.

19D.5.11.4 Decomposition Event Trees for Class II

The containment event tree (CET) for Class II sequences is shown in Figure 19D.5-10. The supporting DETs are shown in Figures 19D.5-28 through 19D.5-30. This CET is substantially different from those for the Class I events. Class II consists of sequences with loss of containment heat removal (CHR) but with successful in-vessel injection. If CHR is not recovered within about 20 hours, the containment pressure will exceed the COPS rupture disk setpoint pressure. CEB92-41 p.15

The first event in the CET assesses the probability of recovery of the RHR system prior to COPS operation (or containment overpressure failure) given that RHR was not initially available (Figure 19D.5-28). If the RHR system is successfully recovered, containment pressure will decrease and the event will be terminated. This probability is estimated assuming a mean time to repair of 19 hours for the system.

The second event in the Class II CET assesses the probability that the COPS rupture disk opens prior to drywell head failure for sequences without recovery of RHR (Figure 19D.5-29). The probability of this event is consistent with the value described in Subsection 19D.5.11.3.12.6 with no pressure difference between the wetwell and drywell.

Given failure of the COPS rupture disk to open prior to drywell head failure, the third event in the CET assesses the probability that drywell head failure will result in loss of in-vessel injection and core damage (Figure 19D.5-30).

A discussion of the considerations and assumptions used to estimate these event probabilities is provided below.

19D.5.11.4.1 Loss of In-vessel Injection Given Venting with COPS

The COPS is designed to vent the wetwell gas space when the wetwell pressure exceeds 90 psig. As discussed below, high-suppression-pool temperatures or loss of NPSH will not threaten the ability of in-vessel injection systems to operate for an extended period of time after COPS initiation. In addition, random failures of the in-vessel injection systems during their mission time have been considered in the Level 1 analysis. Consequently, it was estimated that there was a negligible probability of failure of in-vessel injection given success in COPS operation.

19D.5.11.4.2 Loss of In-vessel Injection Given Containment Failure

The node CC models the probability that core cooling will be impacted following structural failure of the containment. The quantification of this node is described below.

For cases in which the core is successfully cooled but the containment is not, the containment will pressurize. If the rupture disk fails to open, the containment boundary will eventually be breached. But if core cooling is maintained, the offsite consequences of the breach will be negligible. If the containment boundary failure causes core cooling failure, the consequences would be more severe. Therefore, this potential was reviewed. The following general areas were reviewed and are briefly discussed below:

(1) drywell head failure,

(2) high temperatures in the suppression pool,

(3) high drywell temperatures.

The most likely containment failure location is the drywell head. Drywell head failure would pressurize the relatively small volume between the head and concrete shield plugs. This could levitate some of the plugs which would then fall, potentially causing equipment damage. There is no potential for plugs falling between the reactor vessel and drywell wall because the annular space is too small. The vessel vent could be damaged but the consequences would be no worse than a small LOCA. Although unlikely, plugs could fall through the vertical equipment hatch and damage electrical equipment and/or an RHR heat exchanger. It is extremely unlikely that more than one division of core cooling would be lost as a result.

High temperatures in the suppression pool would result in increased suction temperature for core cooling pumps. However, pump performance should not be impaired because the pumps are designed for water temperatures as high as 360°F. Further, condensate storage tank water and fire tank water temperatures would not be affected.

High drywell temperatures were considered for their potential effects on SRV performance, electrical equipment, and water level instrumentation. SRV performance should not be degraded because the expected temperature/time history is less severe than the LOCA condition for which the SRVs will be qualified. There is no electrical equipment in the drywell which is required to operate to establish or maintain core cooling. Effects on water level instrument accuracy should be small since the reference and variable legs experience the same elevation drop in the drywell.

After reviewing these potential causes of core cooling loss resulting from high-temperature conditions/containment failure, it was judged that the probability of core cooling loss ranged between 0.01 and 0.001. A value of 0.01 was used in the analyses for loss of conventional core cooling. In the class II sequences derived from the Level 1 PRA, firewater availability had not been considered. Firewater can be used as an additional source of water following containment failure. The firewater system is much less vulnerable to containment failure. The combined failure probability of conventional cooling and firewater is estimated to be 0.0001, but a value of 0.001 was used for conservatism.

19D.5.12 Discussion of Results

19D.5.12.1 Introduction

The results of the containment event tree analyses are discussed in this subsection. To recapitulate, the accident sequence event trees described in Section 19D.4 identified nine accident classes. The total accident sequence frequency was calculated to be 1.3E-6 per reactor-year. The CETs were used to identify the radioactive release categories and calculate the associated frequencies for use in the consequence analysis described in Section 19E.3. In addition the CETs helped answer the following questions:

a) What fraction of these accident sequences resulted in core damage?

b) In what fraction of these cases was core melt arrested?

c) What fraction of these accident sequences resulted in COPS actuation?

d) What fraction of these accident sequences resulted in containment failure?

e) Are the core damage and containment-related goals met?

The answers to these questions follow:

19D.5.12.2 Core Damage Frequency

The total internal event core damage frequency (CDF) calculated from the sum of all releases except OK on Table 19D.5-8 is 1.6E-7 per reactor-year. Thus ~12% of the total accident sequence frequency (1.3E-6 per reactor-year) results in core damage. In other words, 88% of the accident sequences do not result in core damage.

Classes I, III and IV result in core damage. However, only 1E-6 of all class II events (i.e., the frequency of STC #52 in Figure 19D.5-3 divided by the frequency of all Class II) results in core damage. This low CDF value is attributable to the recovery of failed systems (or AC power), the ability of the ABWR RHR pumps to pump saturated water without cavitating and the ability of core cooling systems to continue to inject water to the reactor following operation of the COPS system.

19D.5.12.3 Core Melt Arrest

In addition to the 94% of accident sequences which do not result in core damage, in another 11% (Release Node N frequency of 1.6E-7 per reactor-year) the core melt is arrested either in the RPV or in the containment, without ever breaching the containment structural integrity. This means that in virtually all of the accident sequences, either radioactive material remains in the reactor vessel or is contained within the containment boundaries and not released to the environment (except through normal containment leakage). This is attributable to equipment and power recovery prior to containment failure and to "passive mitigation," i.e., flooding of the molten core from the suppression pool water when passive flooder system actuates.

Considering only those accident sequences in which core melt starts, (i.e., exclude certain class II events where core melt was not initiated), then the core melt arrest constitutes approximately 86% of all such sequences. The frequency of core damage with significant fission product release, which includes all categories except NCL and OK, is 2.2E-8 per reactor year.

The containment design incorporates a containment overpressure protection system which is designed to ensure that any sequence which is not arrested in the containment will have low consequences. This system consists of a line originating in the wetwell which exhausts to the plant stack. If the containment pressure rises to a level where containment integrity could be challenged, a rupture disk opens relieving the containment pressure. If there is no suppression pool bypass, the containment does not reach the rupture disk setpoint for about 24 hours. This ensures a late release with low magnitude. The frequency of these events is 2.1E-8, or 13% of all core damage events. The frequency of all other events is only 9.8E-10. Thus, the upper bound for releases with the potential to be early or have high magnitude is 0.6%.

19D.5.12.4 Probability of Containment Structural Failure due to Loss of Heat Removal

One of the goals of the ABWR design is to assure that highly reliable heat removal systems be provided to reduce the probability of containment failure by loss of heat removal.

The frequency of containment structural failure resulting

from a loss of containment heat removal systems is evaluated to be 1.1E-9 per reactor-year. Core damage occurs in only 0.1% of these events. This low number demonstrates that the goal is met for the ABWR design. The ABWR features and other factors that contribute to this low value are:

(1) Three divisions of heat removal systems.

(2) Ability to re-establish the main condenser as a heat sink in certain accidents.

(3) Ability to remove heat using RWCU heat exchanger.

(4) Long times before containment pressure reaches a value which could threaten containment integrity, which enables recovery of power and failed heat removal systems.

(5) Presence of the containment overpressure protection system.

(6) Ability of the core cooling systems to continue to maintain the core cooling function following structural failure of the containment.

19D.5.12.5 Frequencies for Radioactive Release Categories

The important release characteristics for each of the severe accident sequences are summarized in Figure 19D.5-3. The first branch of the tree identifies the initiating event for each sequence. This information is used to specify the first four letters of the severe accident sequences used for the deterministic analyses performed in Section 19E.2.2. Later branches identify the potential impact of other important issues such as flooder operation and mode of fission product release. Table 19D.5-7 identifies the deterministic accident sequence associated with each of the end states in Figure 19D.5-3 with a frequency of at least 1E-11. Note that all sequences with an intact containment and no rupture disk opening are assigned to class NCL (Normal Containment Leakage). Sequences with a frequency of less than 1E-11 are neglected.

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The deterministic sequences are then binned according to the characteristics of the fission product release. Table 19E.3-6 indicates combination of the deterministic sequences into release bins. This combination was done by considering the timing and magnitude of the releases. Column P(i) of Table 19E.3-6 gives the probabilities associated with each of the consequence bins with frequency above 1E-10. These values are simply the result of summing all of the sequences in a given consequence bin.

STC#53 in Figure 19D.5-3 was binned with Case 9, the worst of the consequence bins. This is a very conservative assumption since the frequency associated with this sequence is the initiating event frequency for ATWS events. The assumption is made only because there is a negligible effect on the consequence analysis. If this assumption impacts the risk, a containment event tree should be developed for ATWS events.

19D.5.13 Sensitivity of Containment Performance Analysis to RHR Recovery Assumptions

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It was noted that late RHR recovery is considered prior to pool bypass in the CET. Hence, the effects of pool bypass on the probability of RHR recovery are not explicitly considered in the model.

The probabilities of late RHR (non-)recovery under various conditions are summarized in Table 19D.5-6. These probabilities assess whether RHR is recovered prior to operation of COPS or overpressure structural failure of the containment. The probability of RHR recovery will vary for different accident subclasses, for sequences with the core damage progression terminated in-vessel, and for sequences with active injection to the lower drywell after RPV failure because of differences in the availability of AC power, sequence timing and other factors.

19D.5.13.1 Minimum RHR Recovery Probability with Pool Bypass

MAAP analysis indicates that the time available for late recovery of RHR prior to COPS actuation would be 5 hours for sequences with pool bypass. Using the method described in Subsection 19D.5.4 the probability of failure to recover at least one division of RHR is 0.4.

This non-recovery probability is considered a bounding value which will be applied for sequences in all accident classes under all conditions for a conservative estimate of the impact of pool bypass and RHR recovery on COPS operation and containment structural failure. Note that late RHR

recovery is only an issue for those accident classes with RHR unavailable at core damage initiation. Accident classes IA_0 and IIIA_0 have RHR available and need not be considered. Furthermore, the impact of pool bypass on RHR recovery is not an issue for accident class II since these sequences are transients where steam discharge occurs directly to the suppression pool.

The corrected probability of RHR non-recovery can be approximated from the previously calculated probability (which neglected the effect of pool bypass) by adding an additional term to account for sequences which would otherwise have had successful recovery. The probability of these sequences is multiplied by the probability of bypass and the probability of recovery for sequences with pool bypass. The modified RHR non-recovery probabilities are shown on Table 19D.5-8 for comparison with the existing values.

19D.5.13.2 Impact on Sequences with In-vessel Core Damage Mitigation

For transient sequences with core melt arrest in-vessel steam discharges which occur will be directed into the suppression pool and the existence of a pool bypass pathway (open vacuum breaker) does not impact containment heatup and the time available to recover RHR. Only for LOCA sequences terminated in-vessel in accident classes IIIA_1 and IIID would pool bypass potentially impact containment pressurization rates and the RHR recovery probability. However, the probability of class IIIA_1 sequences with core melt arrest in-vessel is negligible.

The probability of class IIID sequences with core melt arrest in-vessel is 1.89E-10. The impact of adjusting the RHR recovery probability for this sequence is given in Table 19D.5-9.

19D.5.13.3 Sequences with RPV Failure

An inspection of the CETs shows that almost all of the RPV failure sequences have drywell sprays available. For these sequences pool bypass will not effect containment pressurization since the steam will be quenched by the spray. Therefore, only sequences with no drywell spray availability need to be considered further. The impact of pool bypass on the probability of COPS operation and containment structural failure for these sequences is summarized below.

Frequency of sequences without in-vessel core damage arrest or sprays (and without early containment failure or drywell overtemperature failure) may be calculated from the CETs by multiplying the accident class frequency times the branch point probabilities for all branches up to and including HTF for the pathway following "NO DW INJECT". Most of the classes have negligible impact on the risk (i.e. the frequency of sequence pathway is less than 1E-11). Only class IB1 and ID must be considered further. The impact of pool bypass for these events is then estimated by replacing the original RHR non-recovery probabilities with the values taken from Table 19D.5-8. The results are shown in Table 19D.5-9.

19D.5.13.4 Conclusions

The release category frequencies were modified to account for the impact of suppression pool bypass on the time available for recovery of containment heat removal. The overall frequency of COPS operation is 2.1E-8 and that for late drywell head structural failure is 5.3E-10. The impact of pool bypass on the late RHR recovery probability is to increase the frequency of COPS operation by 4.8E-12 (an increase of 0.02%) and of drywell head failure by 7.7E-14 (an increase of .02%). Thus, consideration of pool bypass in the calculation of RHR recovery has no impact on risk.

Table 19D.5-1

DESCRIPTION OF ACCIDENT EVENT CLASSES

Event	Boron Injected?	Core Cooling Available?	Containment Heat Removal Available?	Relative Time of Core Melt and Containment Structural Failure	Accident Class
Transient	Not Applicable (N/A)	No	Yes	Core Melts First	I
Transient	N/A	No	No	Core Melts First	I
Transient	N/A	Yes	Yes	Successful Mitigation	Plant OK
Transient	N/A	Yes	No	Containment Fails First	II
LOCA	N/A	No	Yes	Core Melts First	III
LOCA	N/A	No	No	Core Melts First	III
LOCA	N/A	Yes	Yes	Successful Mitigation	Plant OK
LOCA	N/A	Yes	No	Containment Fails First	II
ATWS	Yes	No	Yes	Core Melts First	I
ATWS	Yes	No	No	Core Melts First	I
ATWS	Yes	Yes	Yes	Successful Mitigation	Plant OK
ATWS	Yes	Yes	No	Containment Fails First	II
ATWS	No	No	Yes	Core Melts First	I
ATWS	No	No	No	Core Melts First	I

Table 19D.5-1 (Continued)

DESCRIPTION OF ACCIDENT EVENT CLASSES

Event	Boron Injected?	Core Cooling Available?	Containment Heat Removal Available?	Relative Time of Core Melt and Containment Structural Failure	Accident Class
ATWS	No	Yes	Yes	Containment Fails First	IV
ATWS	No	Yes	No	Containment Fails First	IV
Containment Bypass	N/A	No	Yes or No	Containment (i.e. suppression pool) is bypassed. Therefore, no suppression pool scrubbing of radioactive releases. Relative time of core melt and containment failure immaterial.	V
Containment Bypass	N/A	Yes	Yes or No	Successful mitigation.	Plant OK

Table 19D.5-2

DESCRIPTION OF ACCIDENT EVENT I SUB-CLASSES

Event	Boron Injected?	Reactor Pressure	Core Cooling Available?	Accident Class	Comments
All transients except certain station black-out (SBO) events	Not Applicable (N/A)	High	No	IA	Because of high reactor pressure, there is a potential for containment structural failure shortly after core melt.
Station blackout events	N/A	High	No	IB-1	No core cooling or containment heat removal at the beginning because of absence of on-site and off-site power and RCIC failure. However, on-site power recovered in eight hours increasing the likelihood of recovery of core cooling and containment heat removal system.
Station black-out events	N/A	High	RCIC Available for the first eight hours	IB-2	Sequence with core decay heat reduced due to RCIC operation. Also suppression pool heats up prior to core melt shortening the time to containment structural failure.
Station black-out events	N/A	High	No	1B-3	No core cooling or containment heat removal.
ATWS	Yes or No	High or Low	No	IC	
All transients	N/A	Low	No	ID	

Table 19D.5-3

TREATMENT OF SUPPRESSION POOL BYPASS MECHANISMS IN THE PRA

Suppression Pool Bypass Mechanism	How Treated in the PRA	Reference Section/ Subsection
1. Ex-Containment LOCA	Containment bypass study	19E.2.3.3
- High Pressure Systems		
- Interfacing Systems LOCA		
2. Failure of Isolation Valves, Pipe Rupture	Containment bypass study	19E.2.3.3
3. Normal Containment Leaks (containment temperature $<500^{O}$F or pressure <52 psig)	Modeled in CETs	19E.2.4.3
4. Containment Leaks (due to high containment temperature ($>500^{O}$F) and high pressure (52 psig))	Modeled in CETs	
5. Containment structural failure due overpressure (>100 psig)	Modeled in CETs	
6. High Temperature Failure of the containment ($>700^{O}$F)	Modeled in CETs	
7. Uncovery of Horizontal Vent	Not expected to occur in the first 24 hours and therefore not modeled in CETs	19D.5.7.4
8. Low Probability Events	Not modeled in CETs	
- RPV Rupture		
- In-Vessel Steam Explosion		19E.2.1.3.1
- Ex-Vessel Steam Explosion		19E.2.1.3.1
- Basemat Penetration Following Core Melt		19E.2.1.3.6
9. Vacuum breaker leakage of failure	Modeled in CETs	

Table 19D.5-4

SUCCESS CRITERIA FOR CORE MELT ARREST

CASE	REACTOR AT HIGH PRESSURE	REACTOR AT LOW PRESSURE
CORE MELT ARREST IN RPV	Core Cooling 1 of 2 HPCF or RCIC or FW Containment Heat Removal 1 of 3 RHR	Core Cooling 1 of 2 HPCF or 1 of 3 LPFL or Condensate Injection or Fire System Water Injection Containment Heat Removal 1 of 3 RHR
RPV FAILS BUT CORE MELT ARRESTED PRIOR TO FISSION PRODUCT RELEASE	Not Applicable	Core Cooling 1 of 2 HPCF or 1 of 3 LPFL or Condensate Injection or Fire System Water Injection or Passive Flooder System Containment Heat Removal 1 of 3 RHR

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Table 1

Table 19D.5-5
DIVISION OF ACCIDENT SUBCLASSES

Accident Class	Subclass	Subdivided Subclass	RPV Pressure	RHR	Frequency
I	IA	IA_0	High	Yes	4.21 E-8
		IA_1	High	No	4.25 E-10
	IB1	IB1_0	Low	No	2.57 E-8
		IB1_1	High	No	2.57 E-12
	IB2	IB2_0	Low	No	1.62 E-8
		IB2_1	High	No	1.62E-12
	IB3	IB3_0	Low	No	8.86 E-10
		IB3_1	High	No	8.86 E-14
	ID	ID	Low	No	6.95 E-8
II	II	II	n/a	n/a	1.10E-6
III	IIIA	IIIA_0	High	Yes	3.83 E-10
		IIIA_1	High	No	3.87 E-12
	IIID	IIID	Low	No	2.10 E-10
IV	IV	IV	n/a	n/a	1.66 E-10

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Table 2

Table 19D.5-6

CONDITIONAL PROBABILITIES FOR RECOVERY OF CONTAINMENT HEAT REMOVAL

Accident Subclass	Core Melt Arrest in RPV	Active Injection Lower DW	RHR Recovery Probability
IA_1, IIIA_1	CM ARREST		0.95
	NO ARREST	INJECT	0.99
	NO ARREST	NO INJECT	0.9
IB1_0, IB1_1	CM ARREST		1.0
	NO ARREST	INJECT	0.99
	NO ARREST	NO INJECT	0.9
IB2_0	CM ARREST		0.95
	NO ARREST	INJECT	0.9
	NO ARREST	NO INJECT	0.9
IB3_0, IB3_1	CM ARREST		1.0
	NO ARREST	INJECT	0.95
	NO ARREST	NO INJECT	0.9
ID, IIID	CM ARREST		0.8
	NO ARREST	INJECT	0.8
	NO ARREST	NO INJECT	0.8

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Table 19D.5-7

BINNING OF CONTAINMENT EVENT TREE RESULTS

Stc #	Deterministic Bin	Consequence Bin	
1	NCL	NCL	
4	NCL	NCL	
5	LCHPPFP	Case 7	
6	LCHPFSR	Case 1	
8	LCHPPBR	Case 8	See Notes
10	LCHPPBD	Case 7	See Notes
12	LCHP00E	Case 8	
13	NCL	NCL	
14	LCLPFSR	Case 1	See Notes
15	LCLPFSR	Case 1	See Notes
16	NCL	NCL	
18	LCLPFSR	Case 1	See Notes
19	LCLPFSD	Case 7	
21	LCLPFSD	Case 7	
25	NCL	NCL	
26	LCLPFSR	Case 1	See Notes
28	NCL	NCL	
30	SBRCPFR	Case 1	
37	NCL	NCL	
38	LBLCFSR	Case 1	See Notes
40	NCL	NCL	

Notes:

Sequences 8 and 10: Releases taken from suppression pool bypass study in Attachment 19EE.

Sequence 14, 26 and 38: Sequence is arrested in vessel indicating high probability of the use of the firewater addition system.

Sequence 15: This sequence is binned with those which have releases through the rupture disk since any fission products which are released from the vessel will be scrubbed through the suppression pool.

Sequence 30: No credit taken for firewater system since a long time was available to prevent core damage but the operator failed to do so.

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Table 1

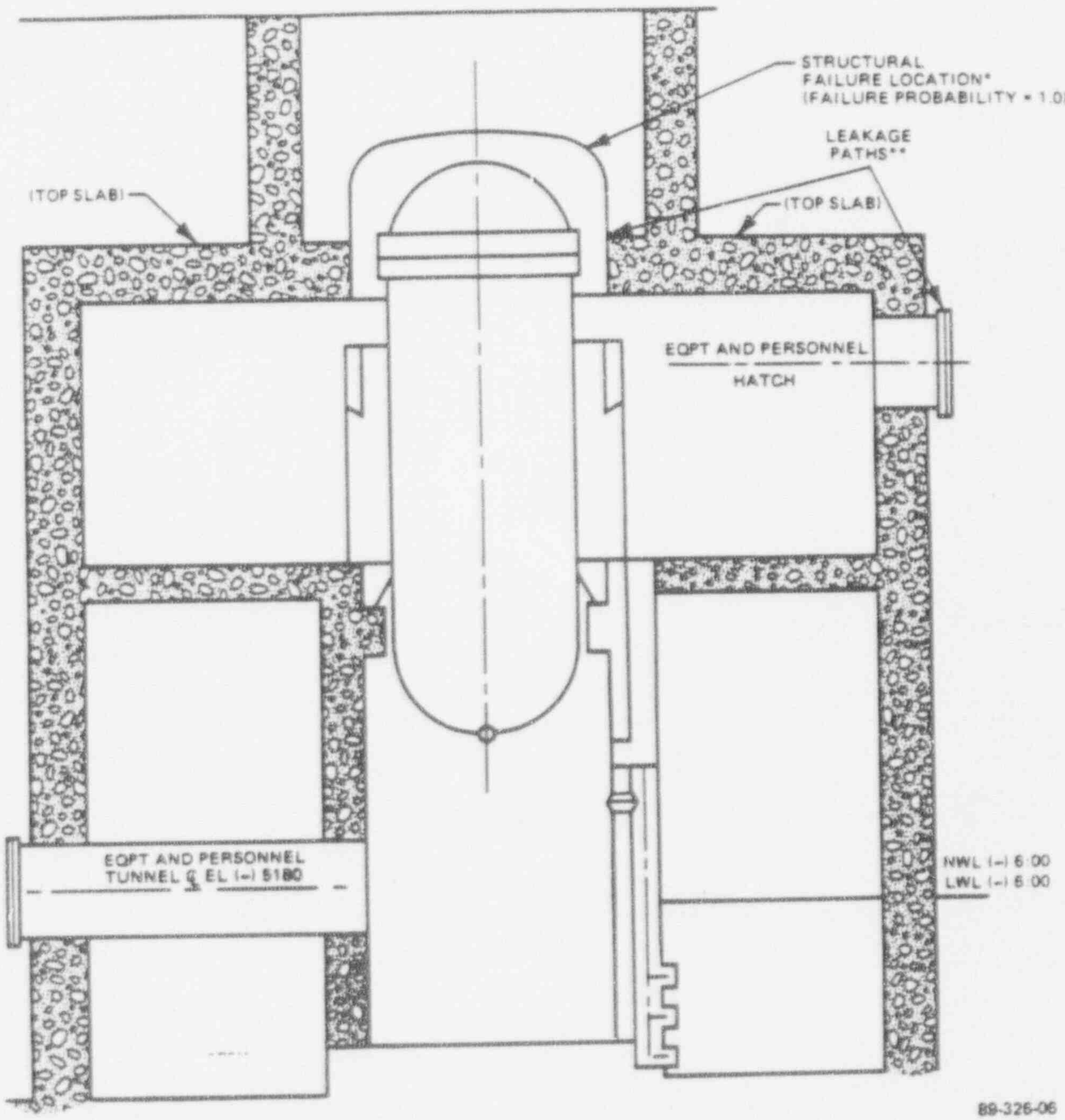
Table 19D.5-8
IMPACT OF SUPPRESSION POOL BYPASS ON CONDITIONAL PROBABILITIES FOR RECOVERY OF CONTAINMENT HEAT REMOVAL

Accident Subclass	Core Melt Arrest in RPV	Active Injection Lower DW	Existing RHR Non-Recovery Probability ($P_{f\text{-}RHR}$)	Modified RHR Non-Recovery Probability ($P_{f\text{-}RHR\text{-}PB}$)
IA_1, IIIA_1	CM ARREST		0.05	0.06
	NO ARREST	INJECT	0.01	0.02
	NO ARREST	NO INJECT	0.1	0.11
IB1_0	CM ARREST		0.0	0.01
	NO ARREST	INJECT	0.01	0.02
	NO ARREST	NO INJECT	0.1	0.11
IB2_0	CM ARREST		0.05	0.06
	NO ARREST	INJECT	0.1	0.11
	NO ARREST	NO INJECT	0.1	0.11
IB3_0	CM ARREST		0.0	0.01
	NO ARREST	INJECT	0.05	0.06
	NO ARREST	NO INJECT	0.1	0.11
ID, IIID	CM ARREST		0.2	0.21
	NO ARREST	INJECT	0.2	0.21
	NO ARREST	NO INJECT	0.2	0.21

Table 19D.5-9

IMPACT OF MODIFIED VALUES FOR RECOVERY OF CONTAINMENT HEAT REMOVAL ON CONTAINMENT PERFORMANCE

	Existing Results	Modified Results
Class IIID sequences with in-vessel recovery		
No Cont Leak	1.51E-10	1.49E-10
RD Open	3.74E-11	3.93E-11
DW Head Fail	3.78E-13	3.97E-13
Class IB1 sequences without in-vessel recovery or drywell sprays		
No Cont Leak	2.11E-10	2.08E-10
RD Open	4.43E-11	4.66E-11
DW Head Fail	8.94E-13	9.40E-13
Class ID sequences without in-vessel recovery or drywell sprays		
No Cont Leak	5.07E-11	5.01E-11
RD Open	1.82E-11	1.88E-11
DW Head Fail	3.67E-13	3.79E-13



* STRUCTURAL FAILURE OCCURS AT 100 psig WHEN TEMPERATURE IS 500°F

** LEAKAGE THROUGH DRYWELL HEAD, EQUIPMENT HATCHES, PERSONAL AIR LOCKS OCCUR WHEN TEMPERATURE EXCEEDS 500°F AND PRESSURE EXCEEDS 52 psig.

Figure 19D.5–1 ABWR CONTAINMENT FAILURE LOCATION AND PROBABILITIES

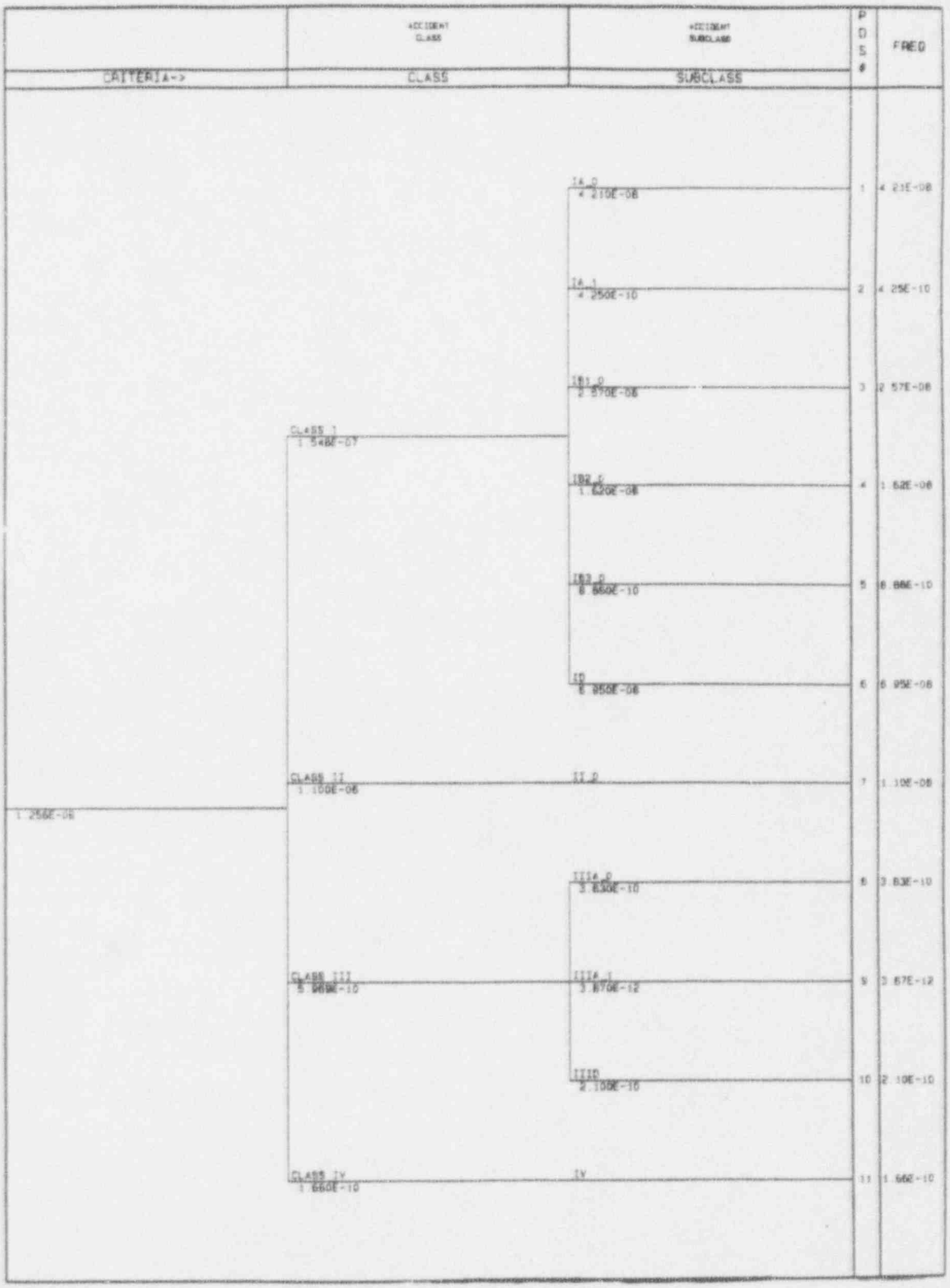


Figure 19D.5-2
ACCIDENT CLASS STRUCTURE

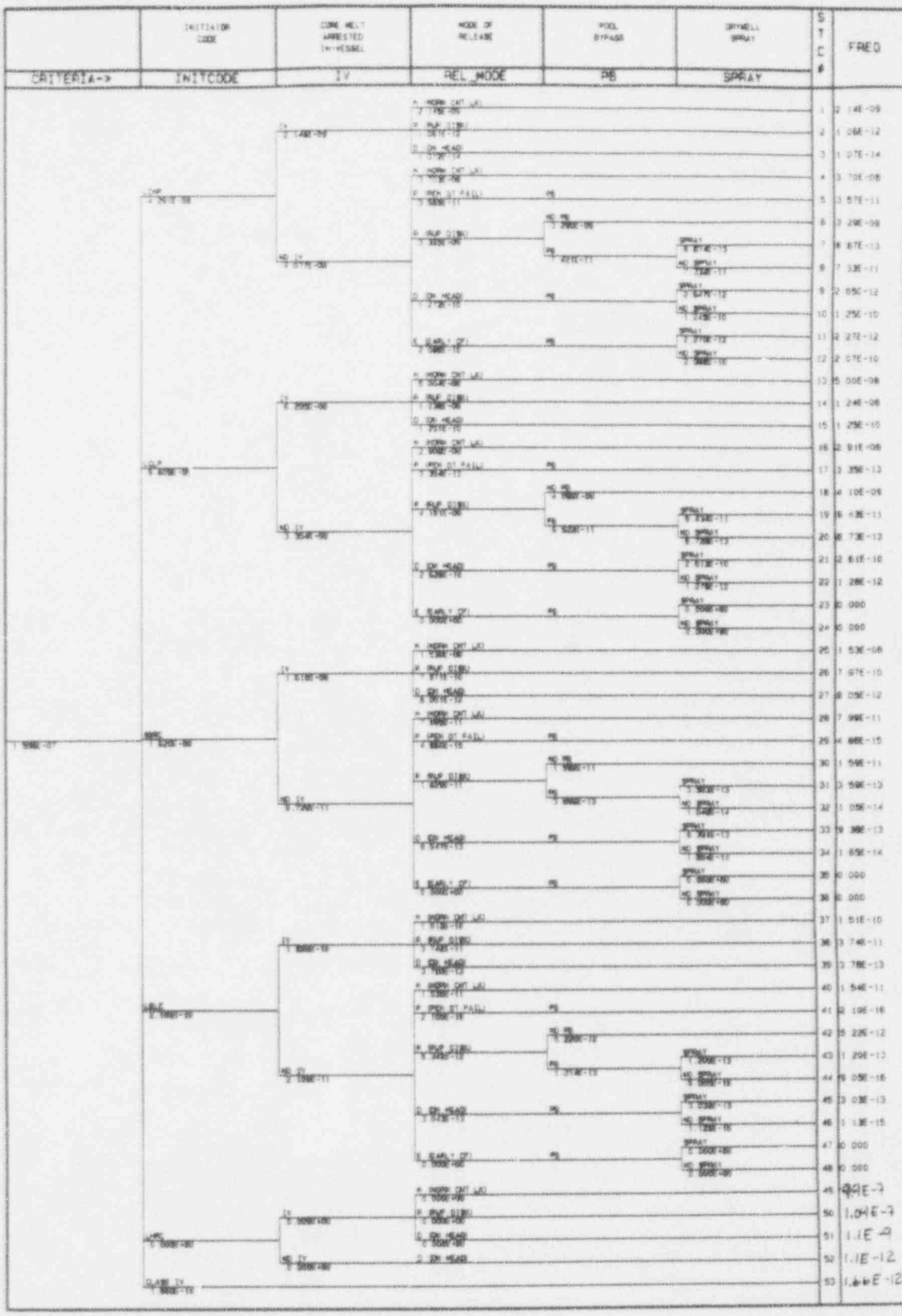


Figure 19D.5-3

SOURCE TERM CATEGORY GROUPING

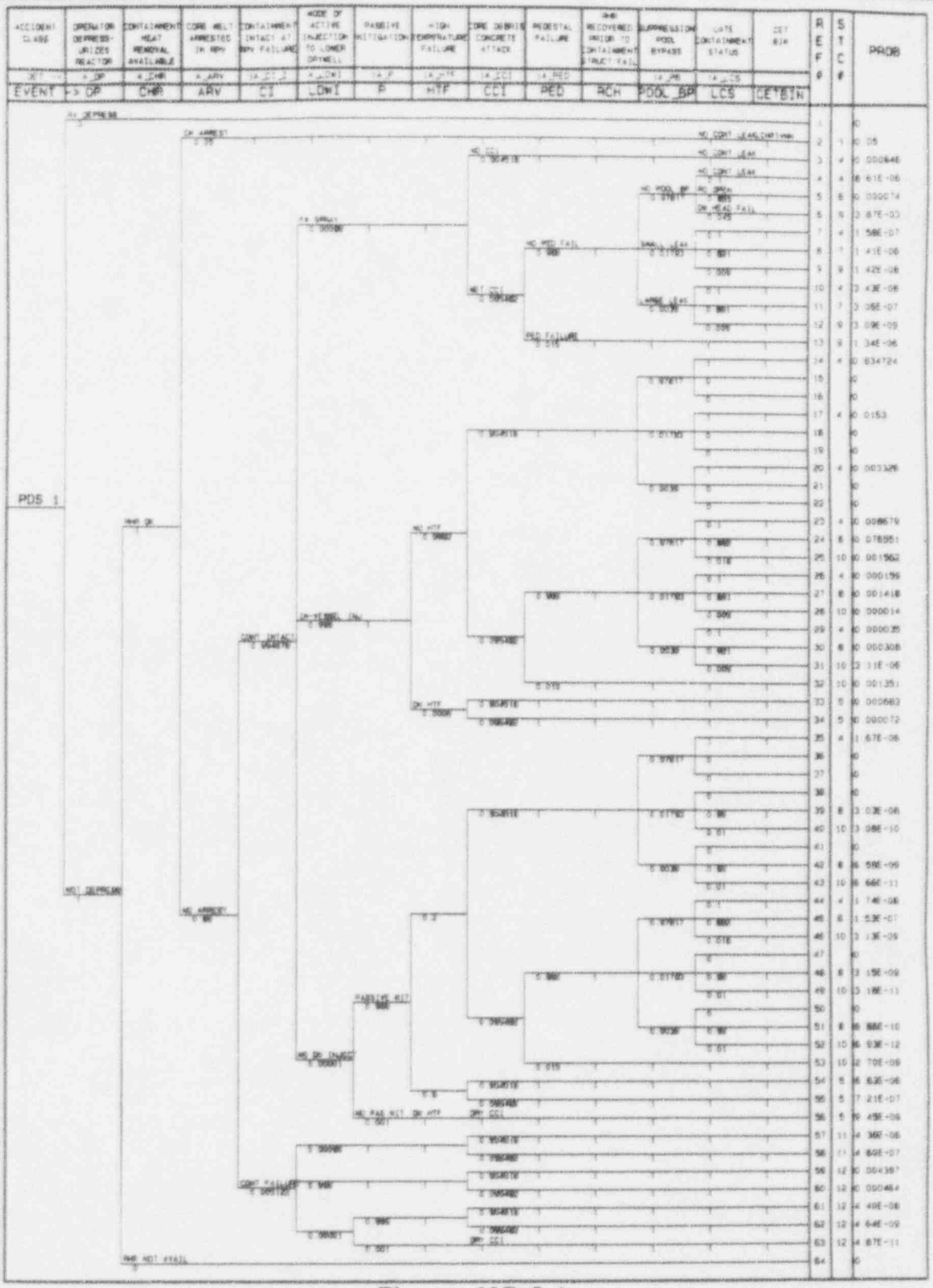


Figure 19D.5-4

PDS 1 - CONTAINMENT EVENT EVALUATION CET FOR CLASS IA SEQUENCES

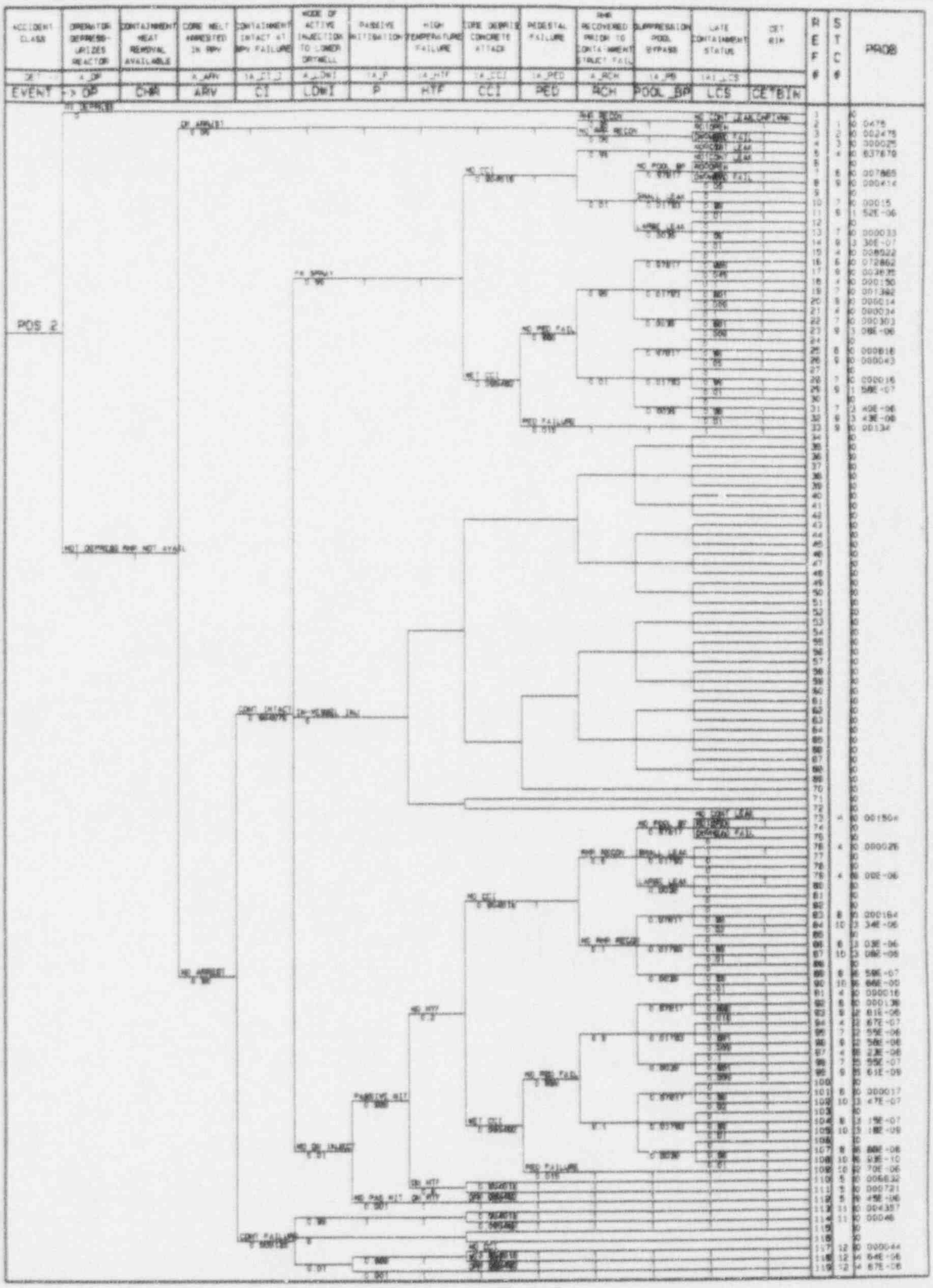


Figure 19D.5-5

PDS 2 - CONTAINMENT EVENT EVALUATION CET FOR CLASS IA_1 SEQUENCES

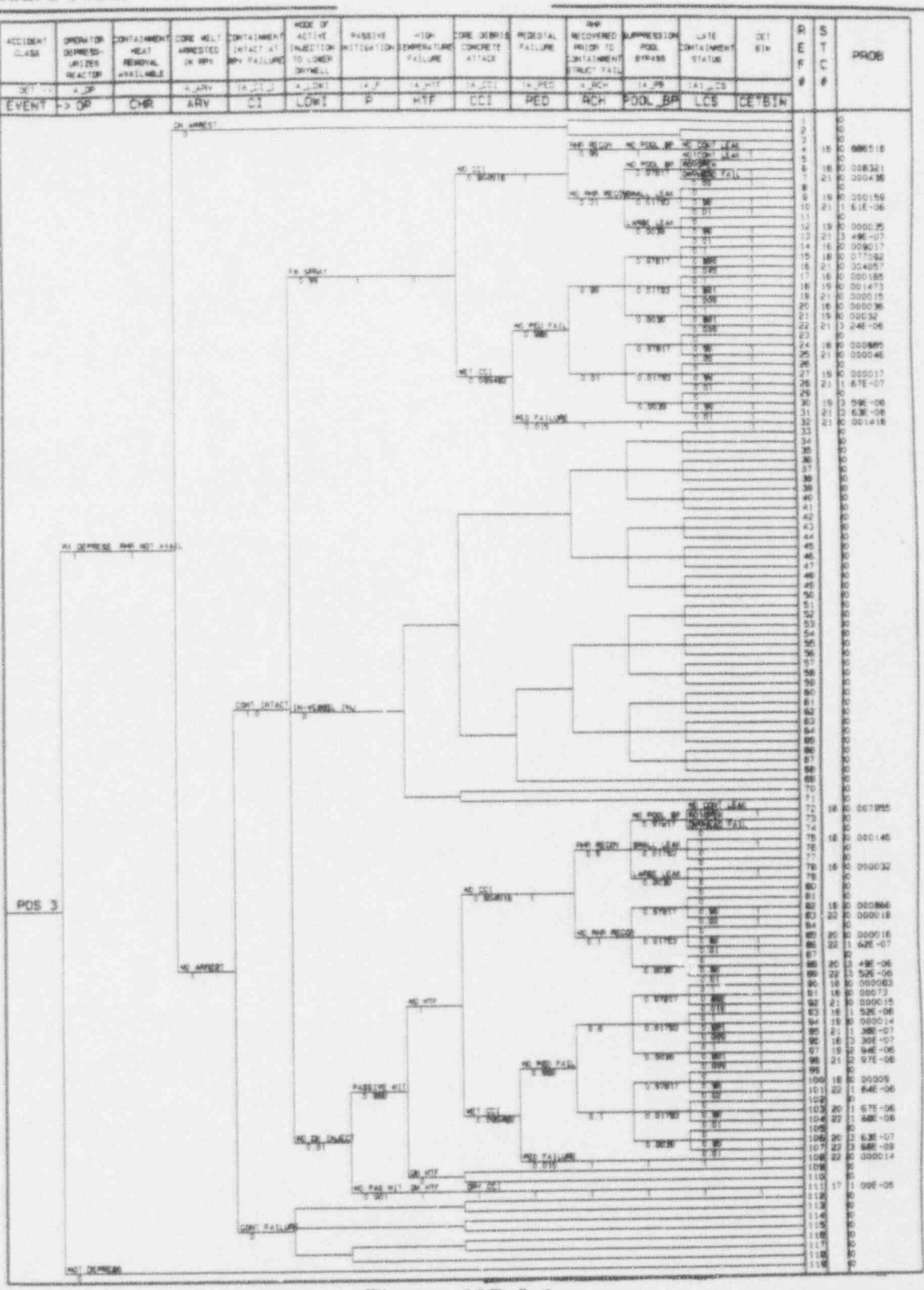


Figure 19D.5-6

PDS 3 - CONTAINMENT EVENT EVALUATION CET FOR CLASS IB1 SEQUENCES

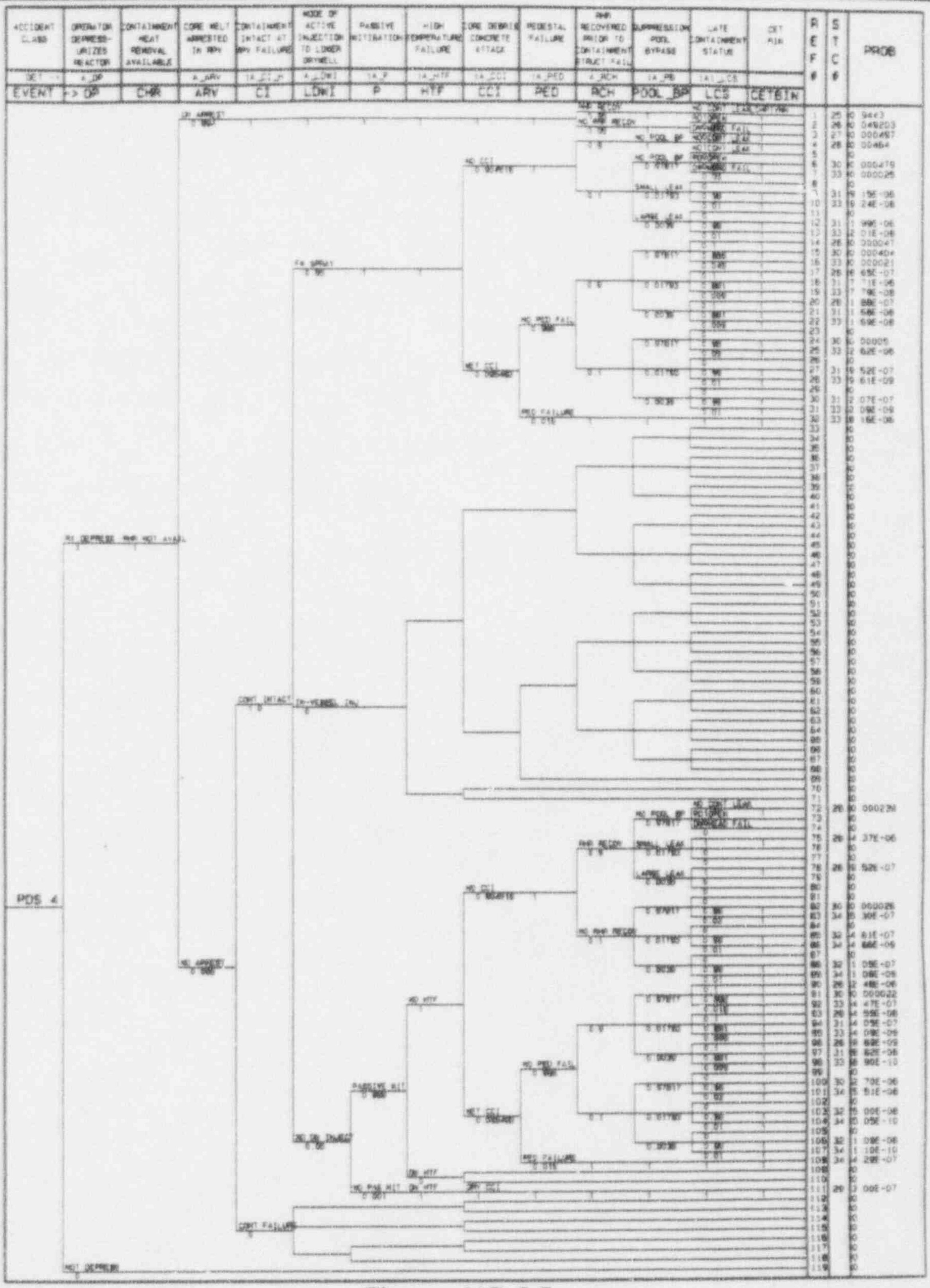


Figure 19D.5-7

PDS 4 - CONTAINMENT EVENT EVALUATION CET FOR CLASS IB2 SEQUENCES

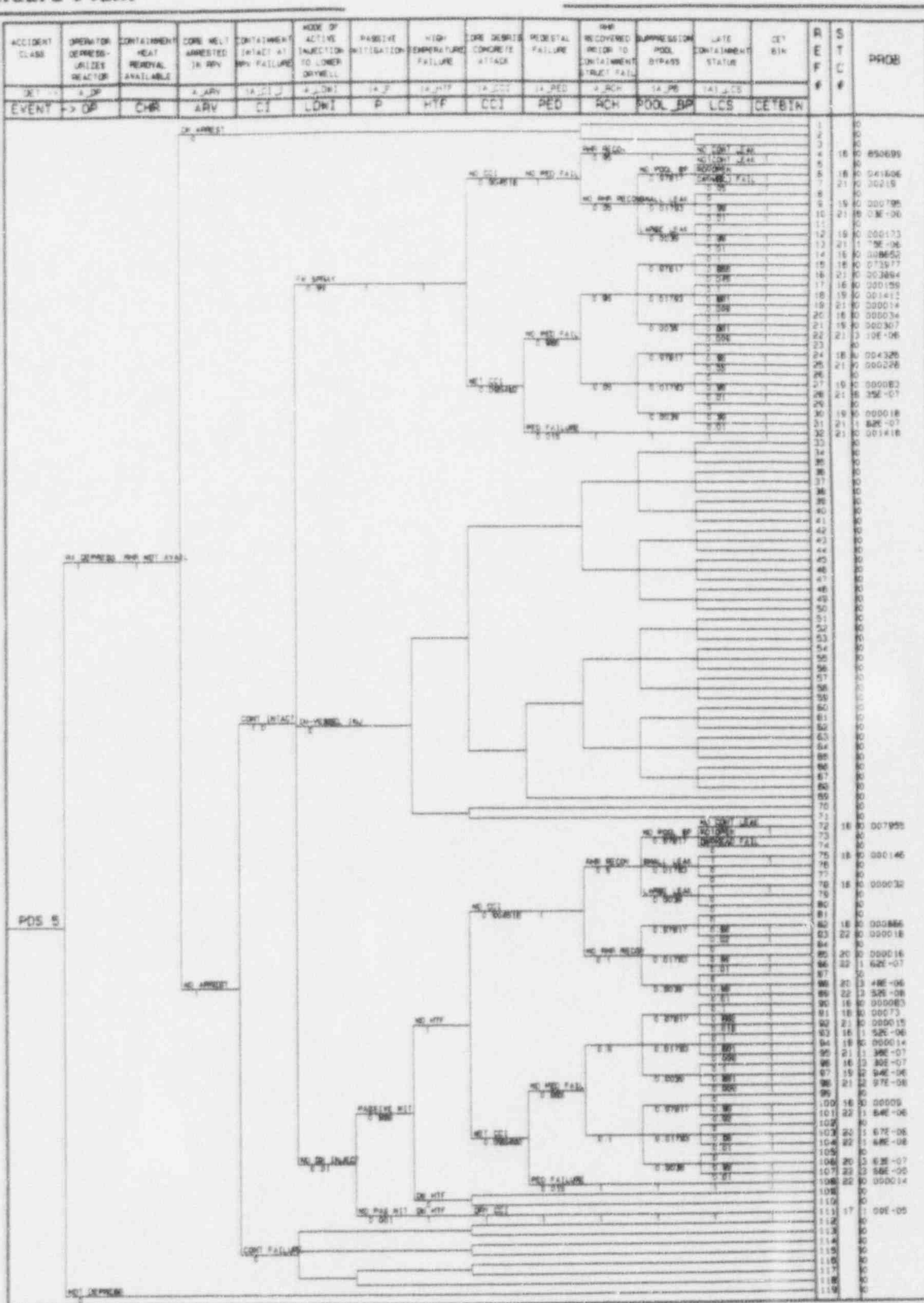


Figure 19D.5-8

PDS 5 - CONTAINMENT EVENT EVALUATION CET FOR CLASS IB3 SEQUENCES

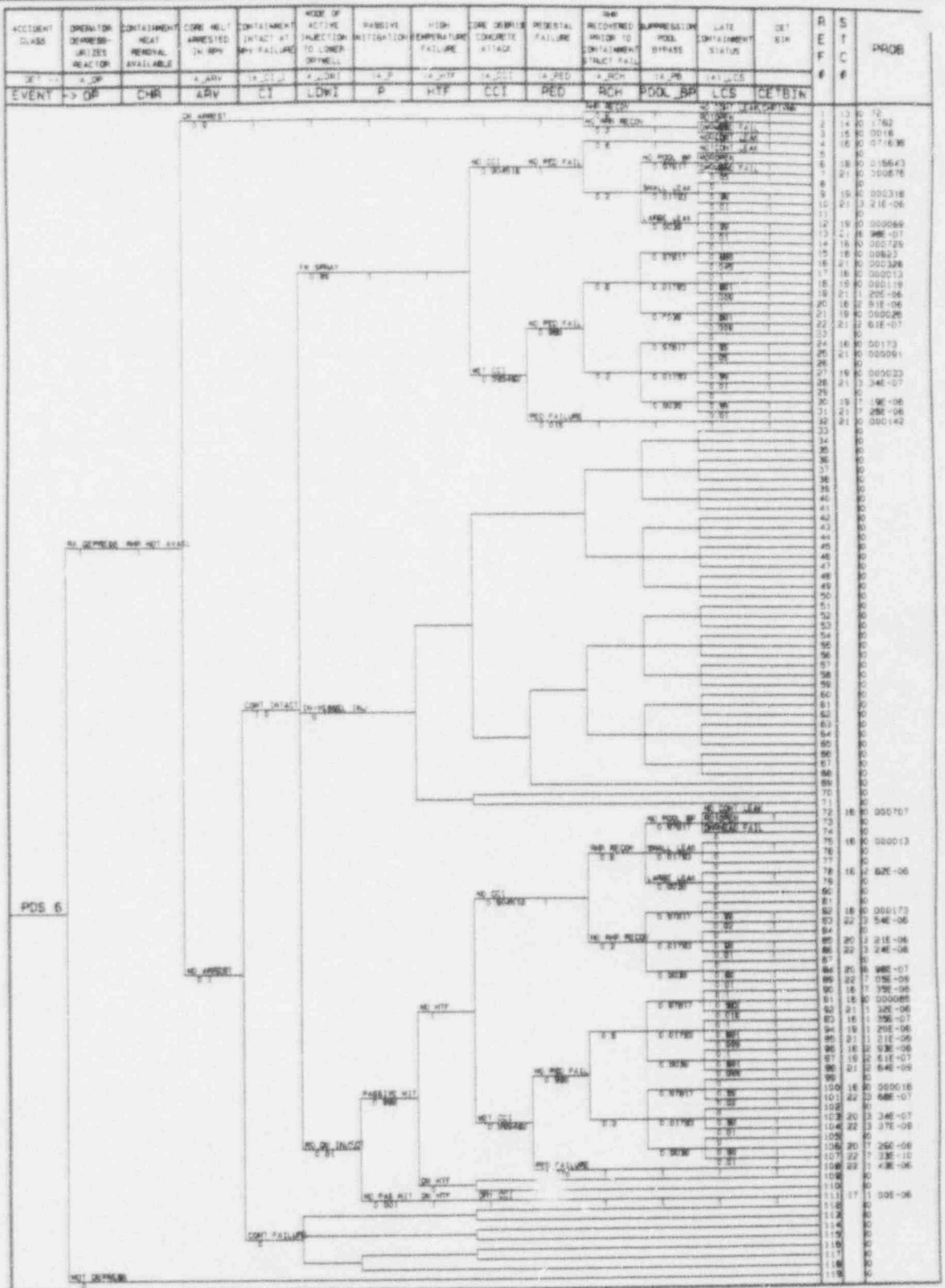


Figure 19D.5-9

PDS 6 - CONTAINMENT EVENT EVALUATION CET FOR CLASS ID SEQUENCES

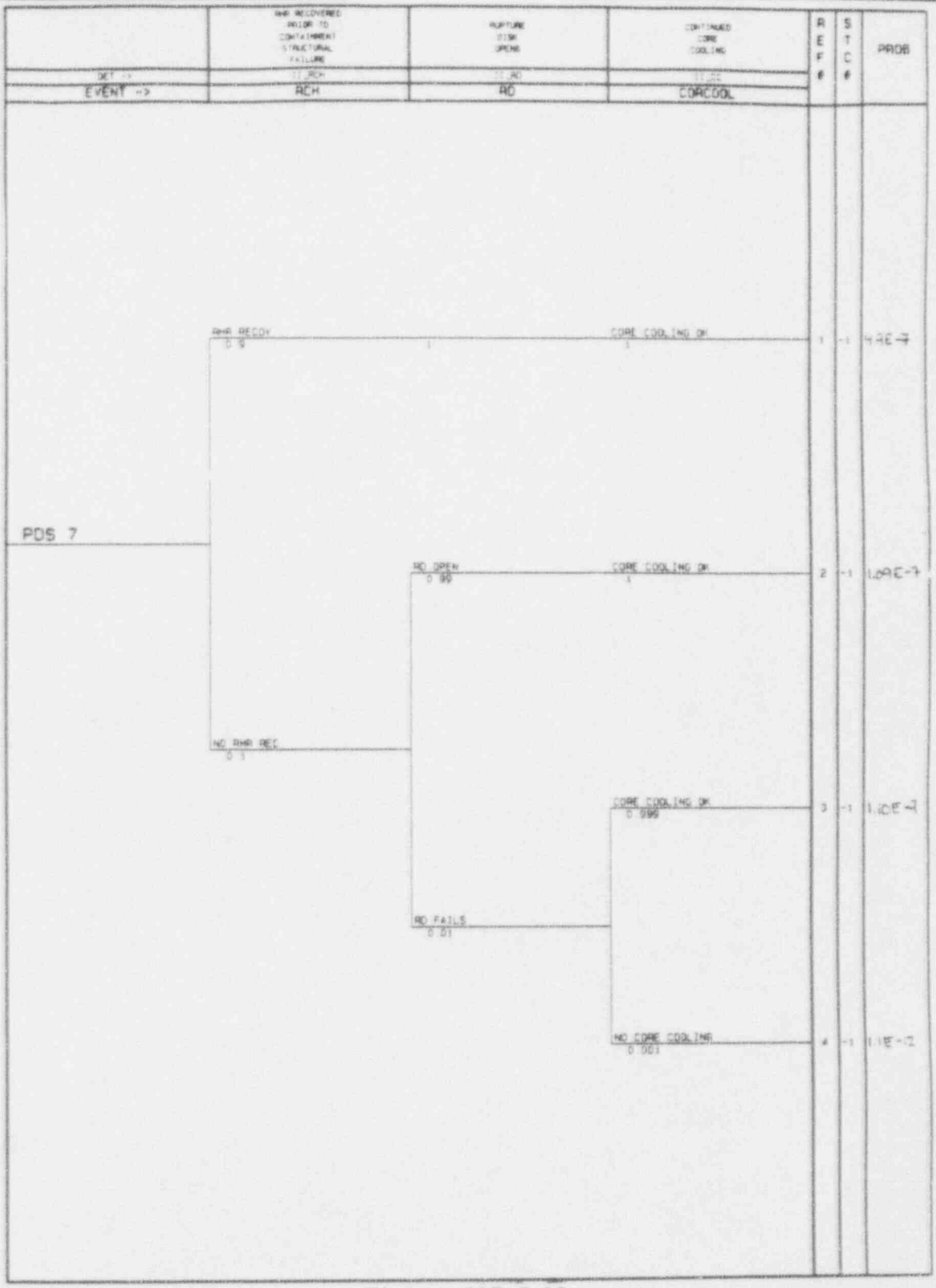


Figure 19D.5-10

PDS 7 - CONTAINMENT EVENT EVALUATION CET FOR CLASS II SEQUENCES

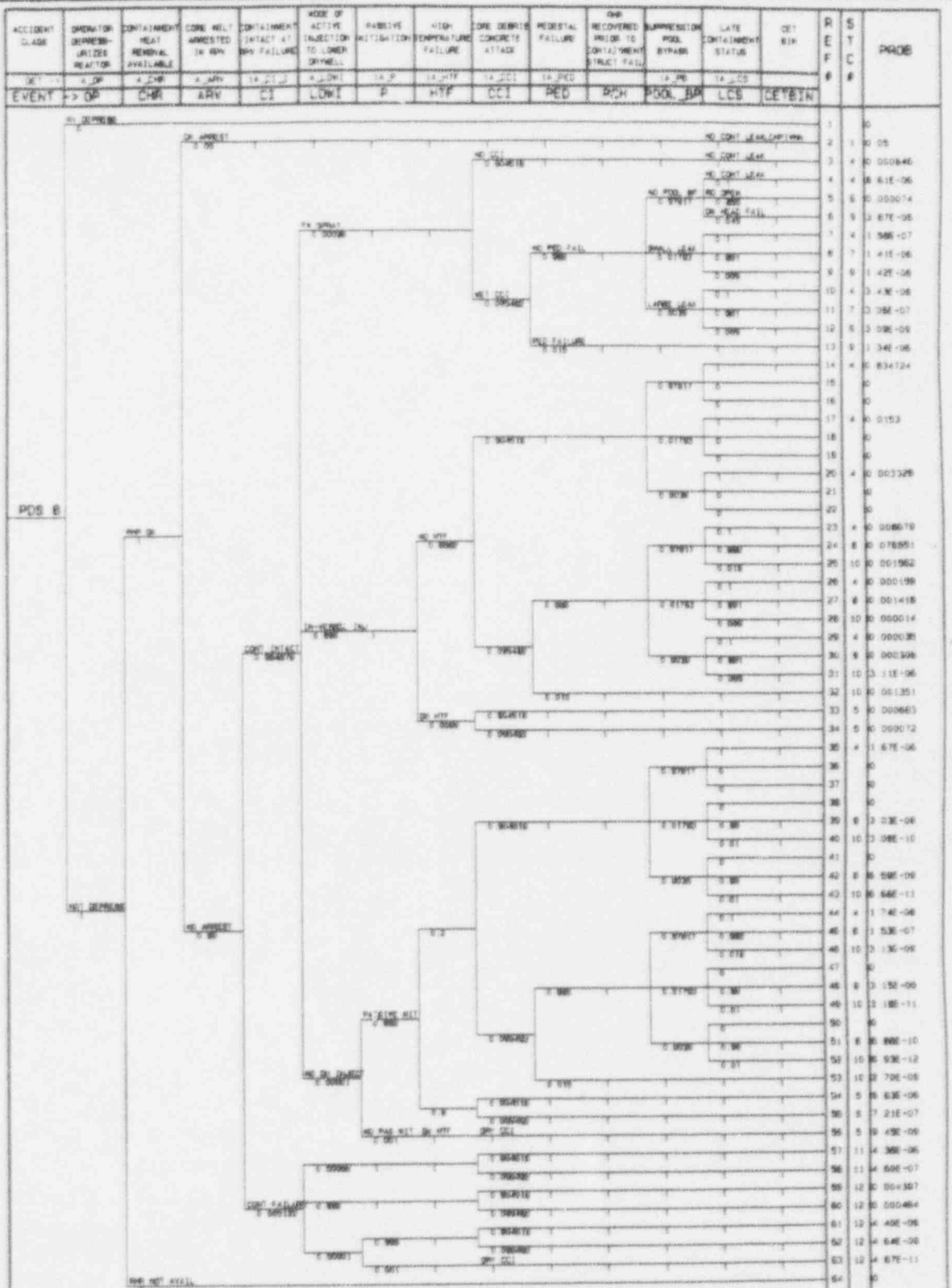


Figure 19D.5-11

PDS 8 - CONTAINMENT EVENT EVALUATION CET FOR CLASS IIIA SEQUENCES

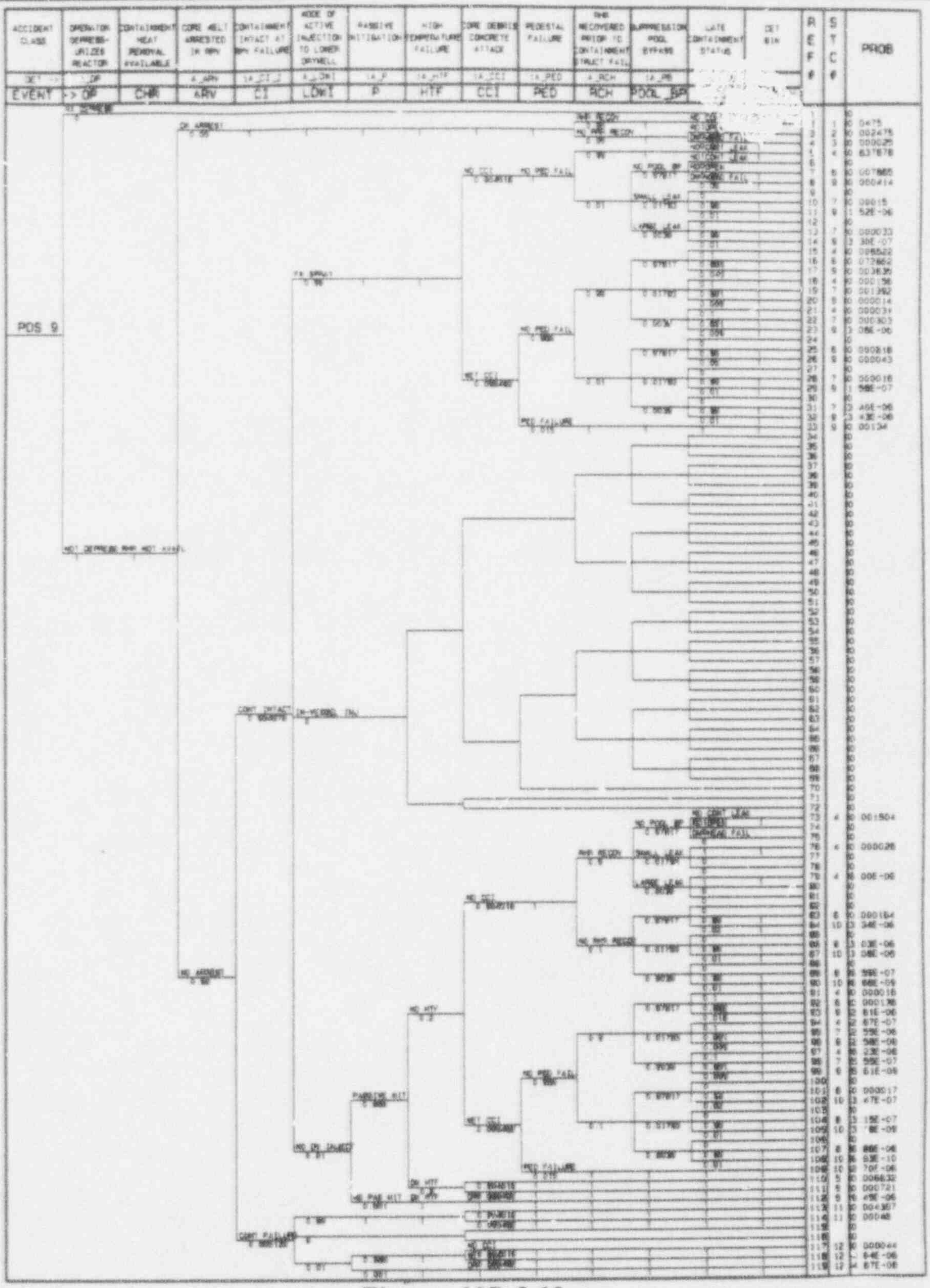


Figure 19D.5-12

PDS 9 - CONTAINMENT EVENT EVALUATION CET FOR CLASS IIIA_1 SEQUENCES

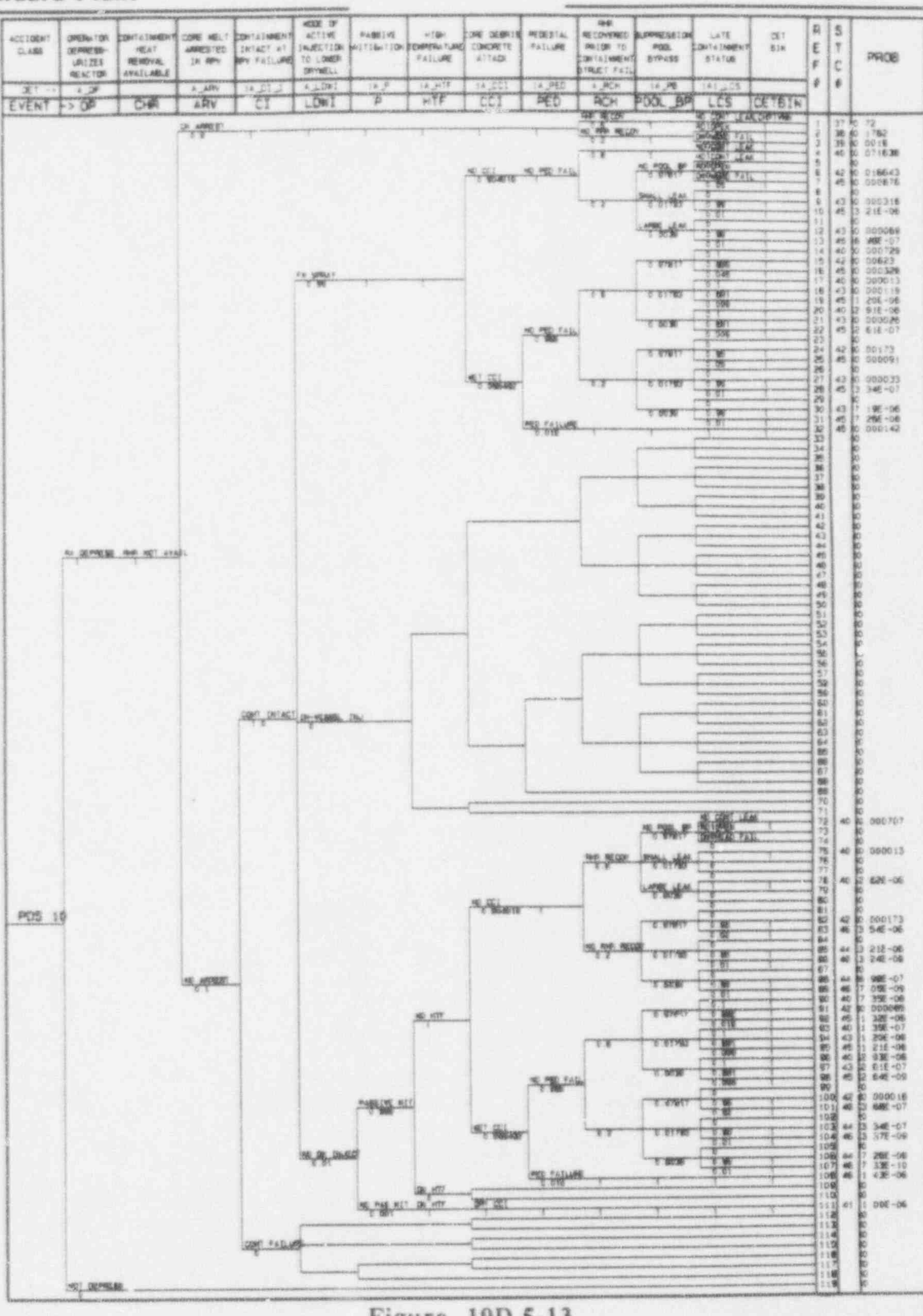


Figure 19D.5-13

PDS 10 - CONTAINMENT EVENT EVALUATION CET FOR CLASS IIID SEQUENCES

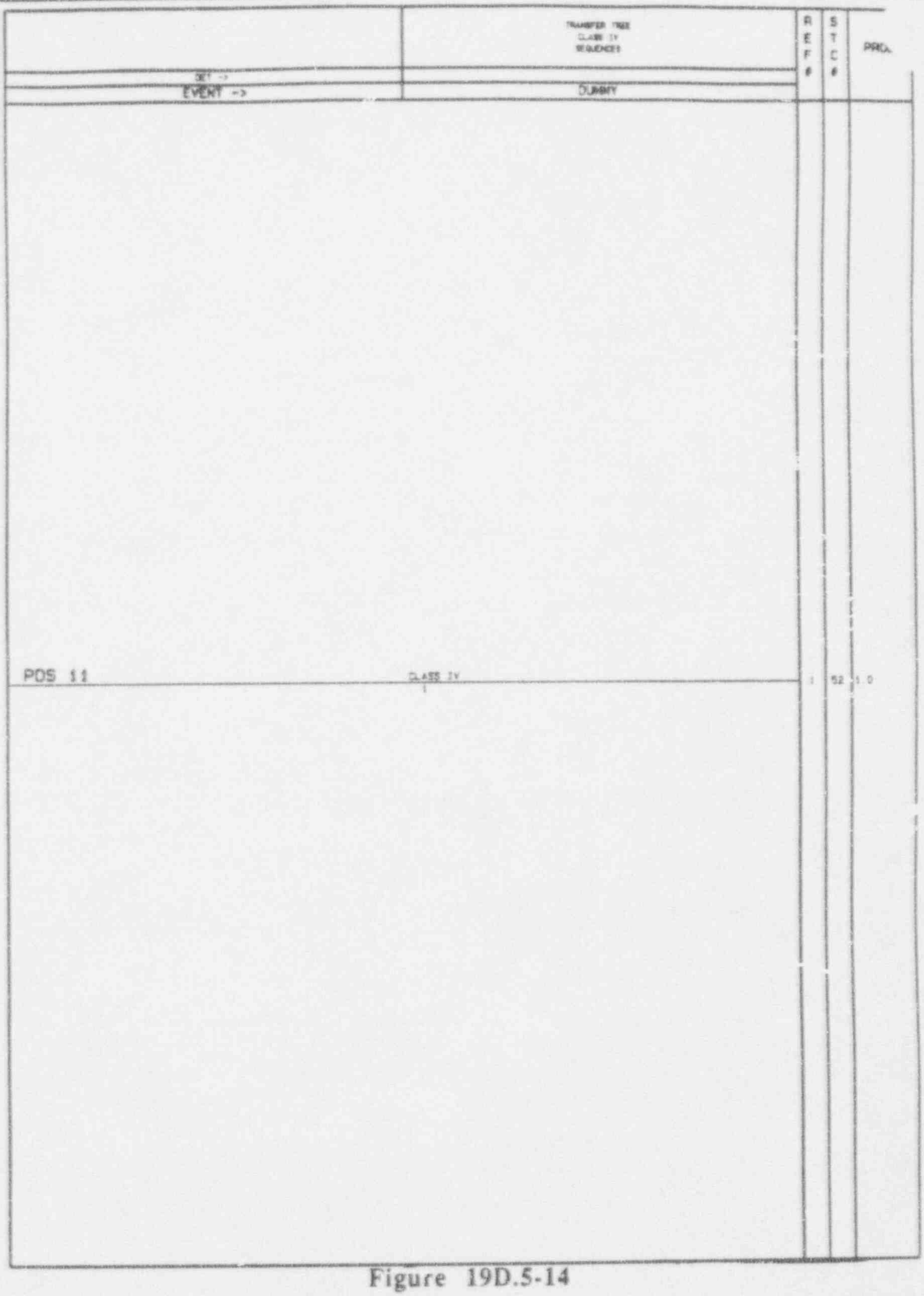


Figure 19D.5-14

PDS - 11 -CONTAINMENT EVENT EVALUATION CET FOR CLASS IV SEQUENCES

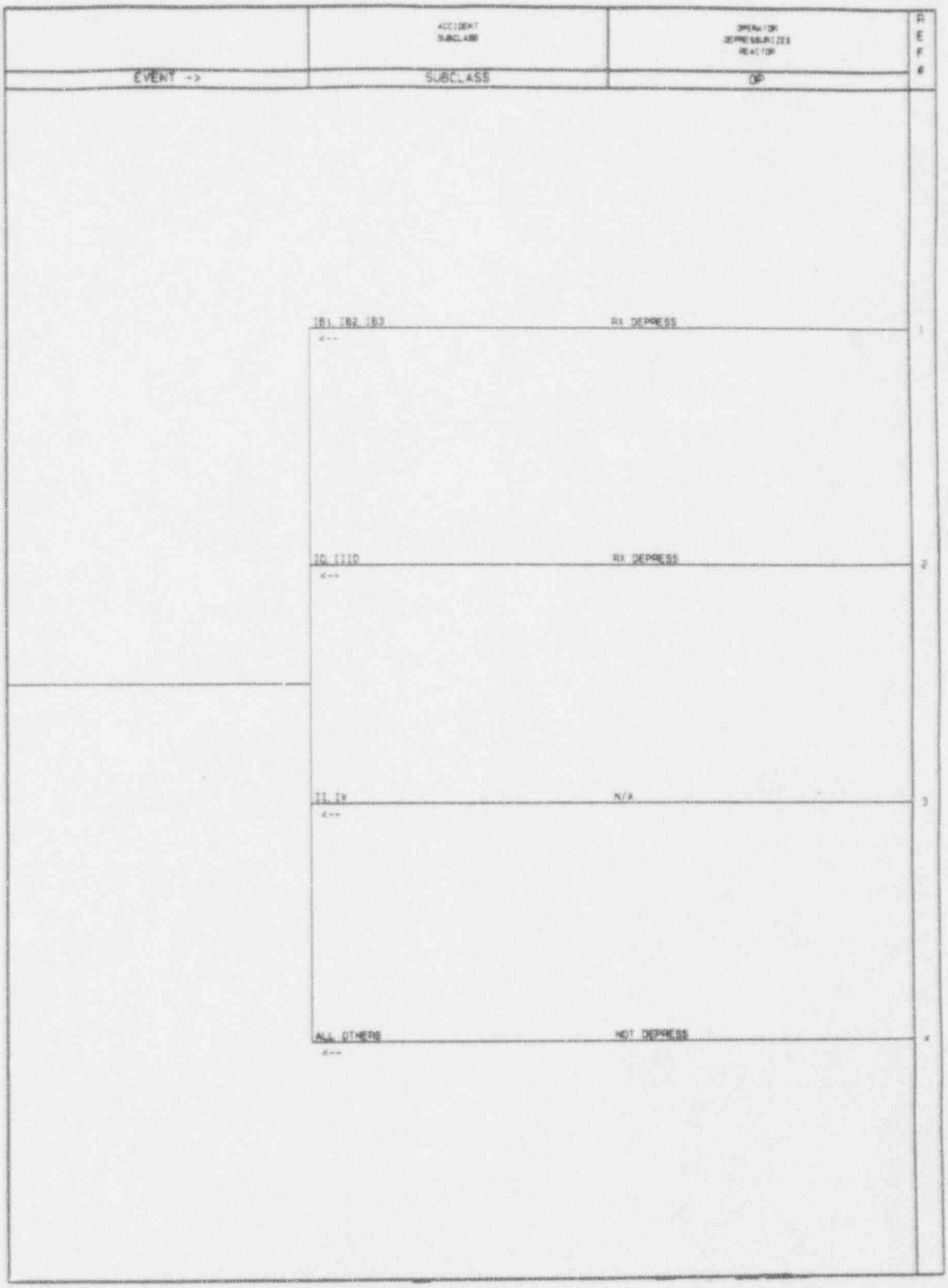


Figure 19D.5-15

CONTAINMENT EVENT EVALUATION DET FOR OPERATOR DEPRESSURIZES REACTOR

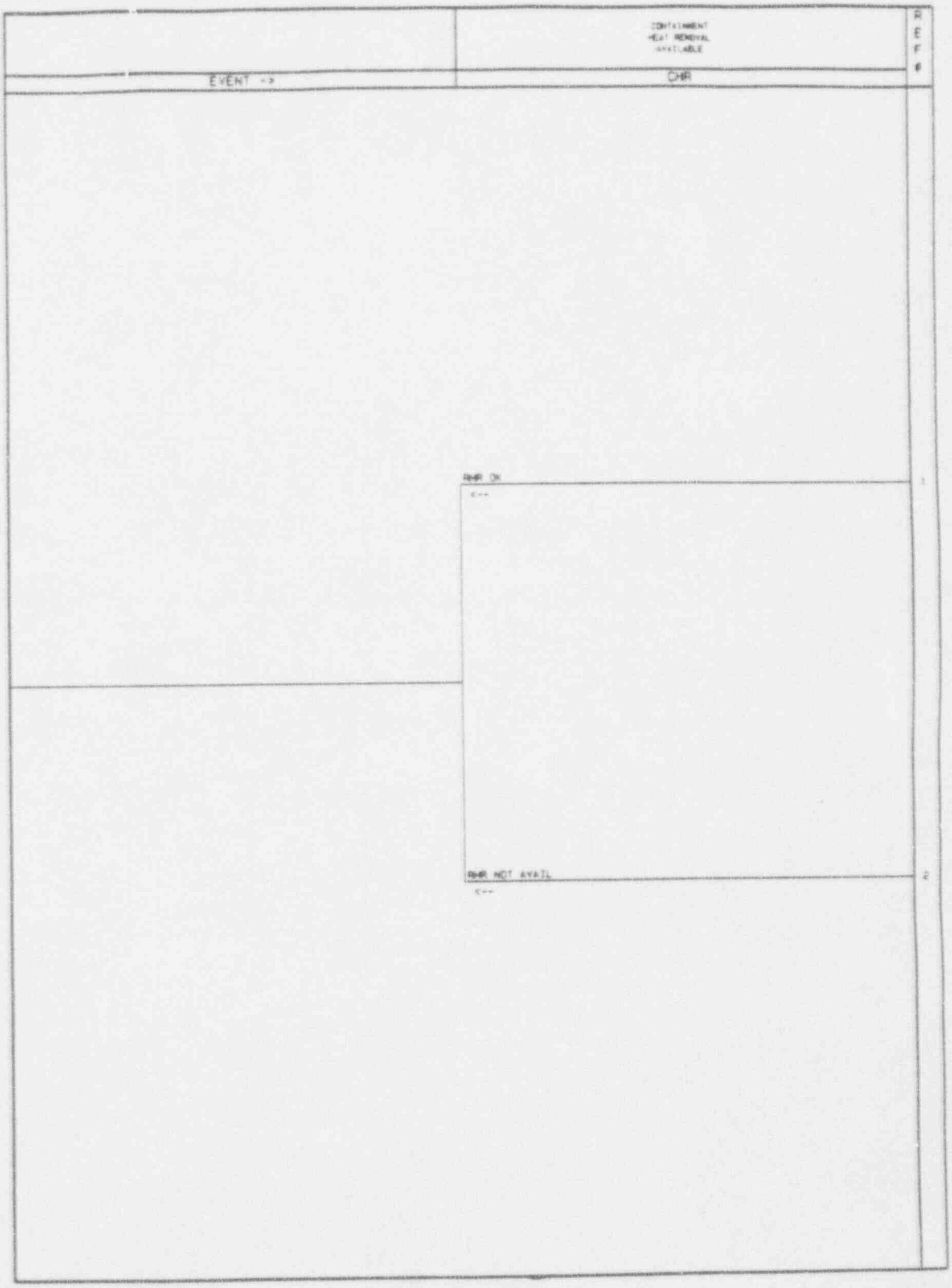


Figure 19D.5-16

CONTAINMENT EVENT EVALUATION DET FOR CONTAINMENT HEAT REMOVAL AVAILABLE

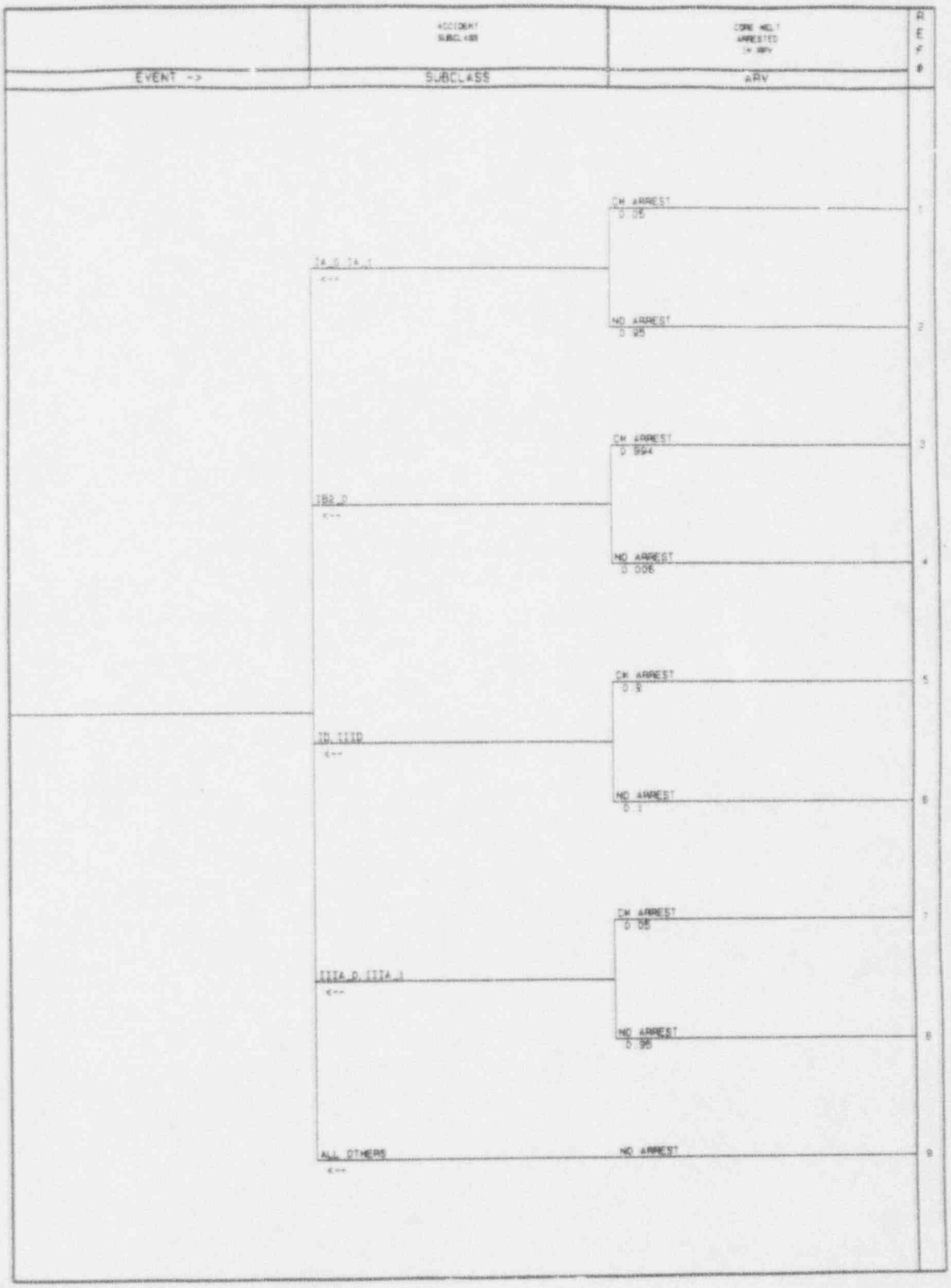


Figure 19D.5-17

CONTAINMENT EVENT EVALUATION DET FOR CORE MELT ARRESTED IN RPV

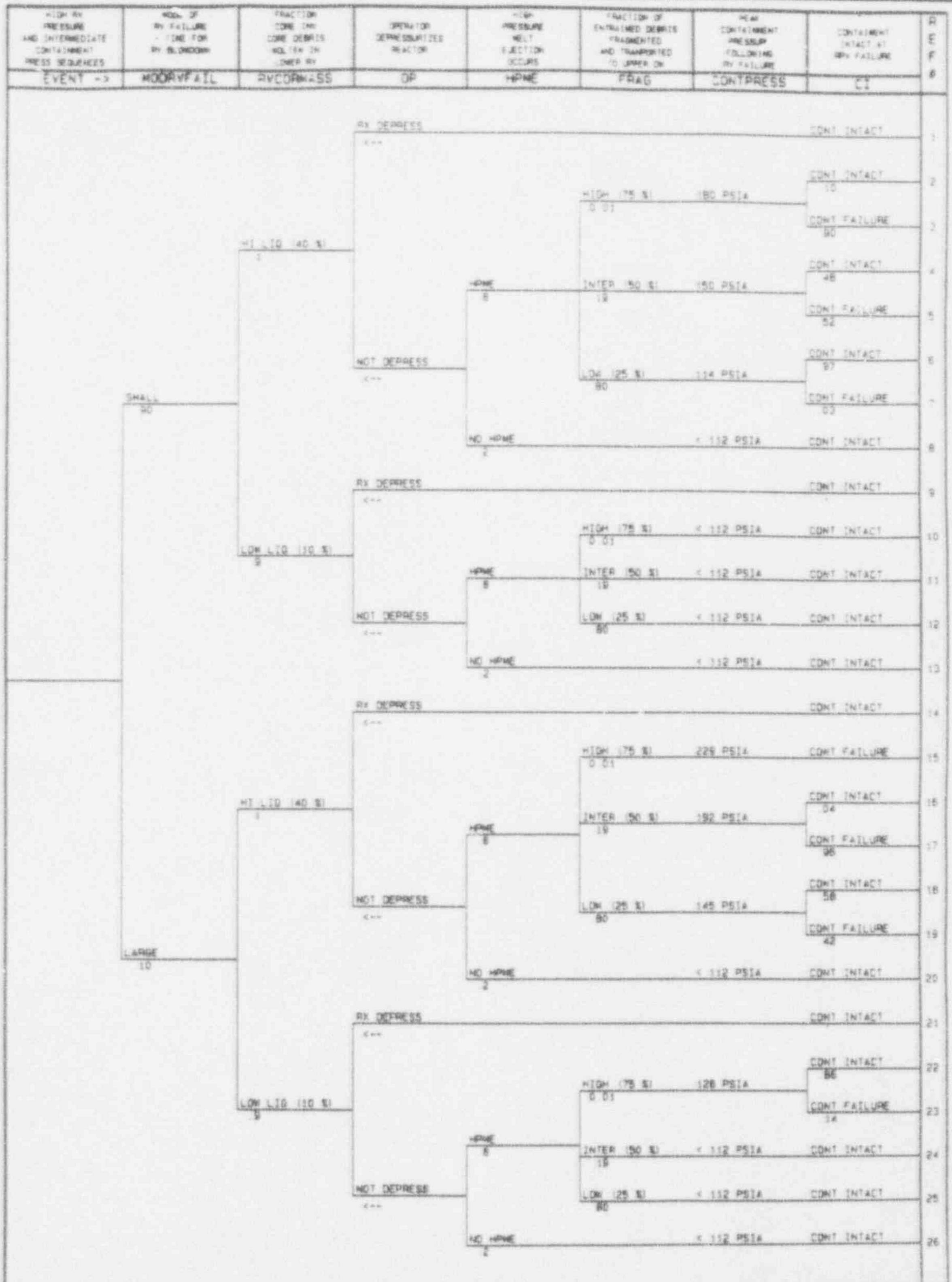


Figure 19D.5-18

CONTAINMENT EVENT EVALUATION DET FOR PROBABILITY OF EARLY CONTAINMENT FAILURE HIGH RV PRESS AND LOW CONT PRESS SEQUENCES

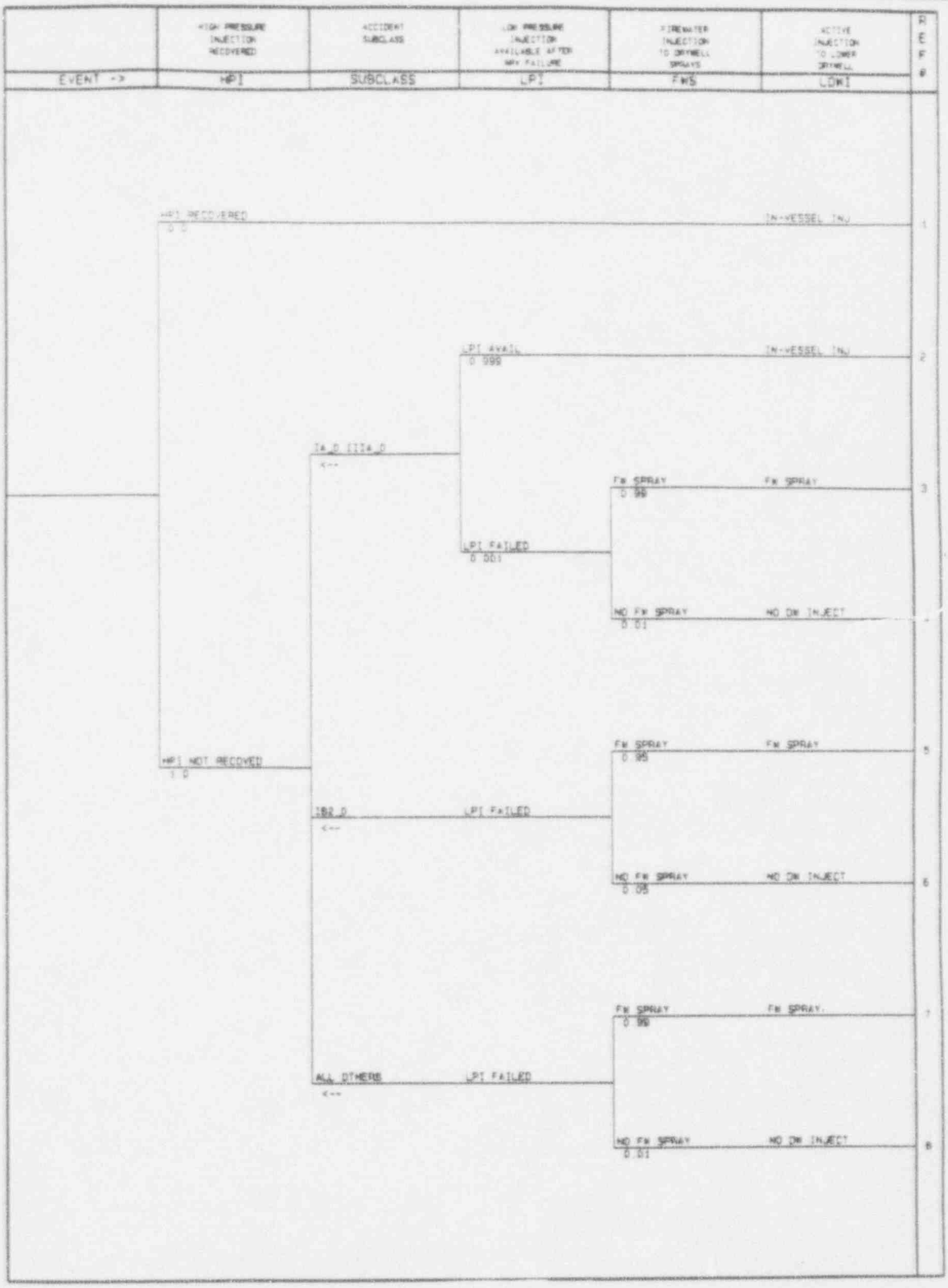


Figure 19D.5-19

CONTAINMENT EVENT EVALUATION DET FOR ACTIVE INJECTION TO LOWER DRYWELL

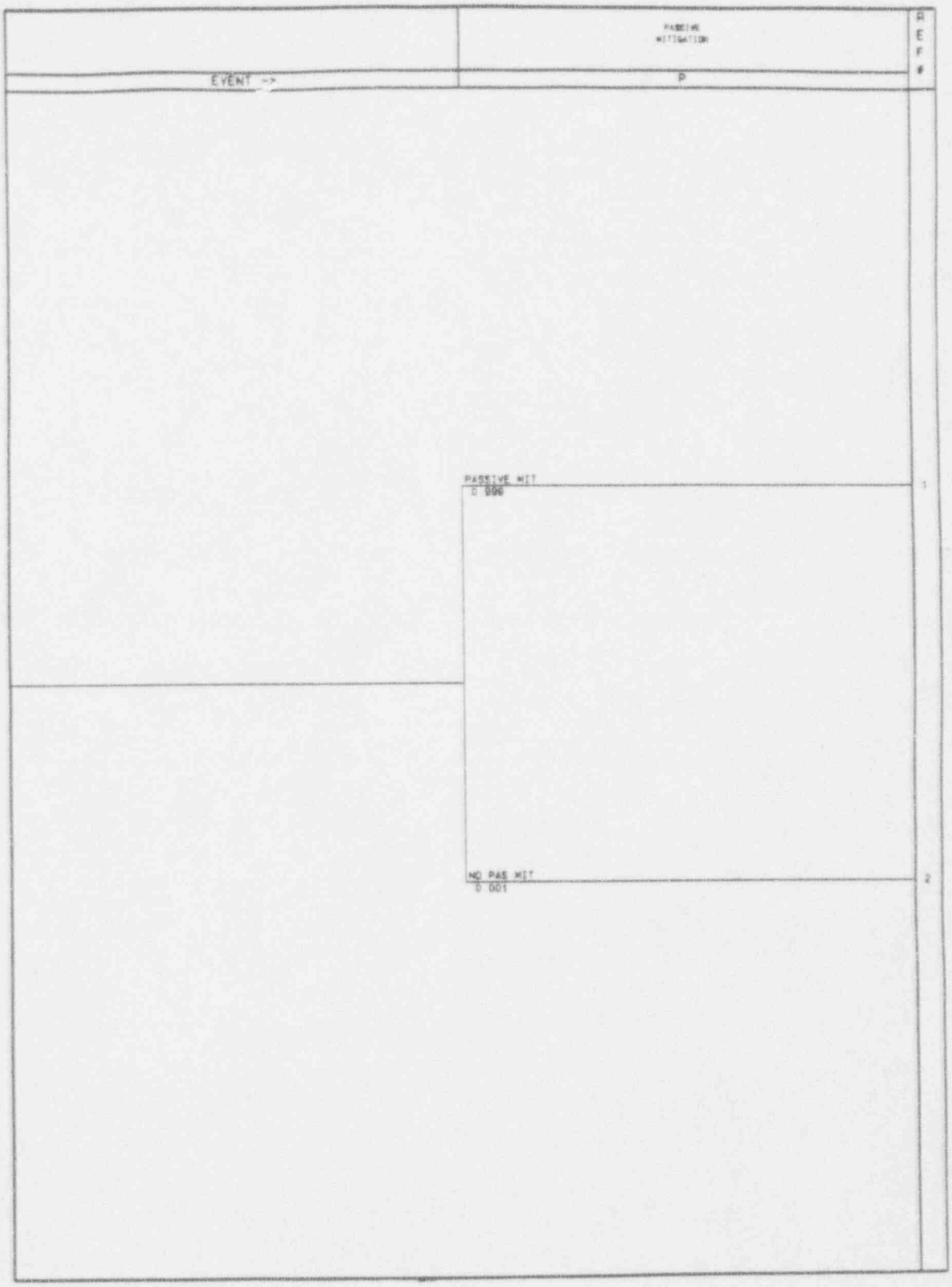


Figure 19D.5-20

CONTAINMENT EVENT EVALUATION DET FOR PASSIVE MITIGATION

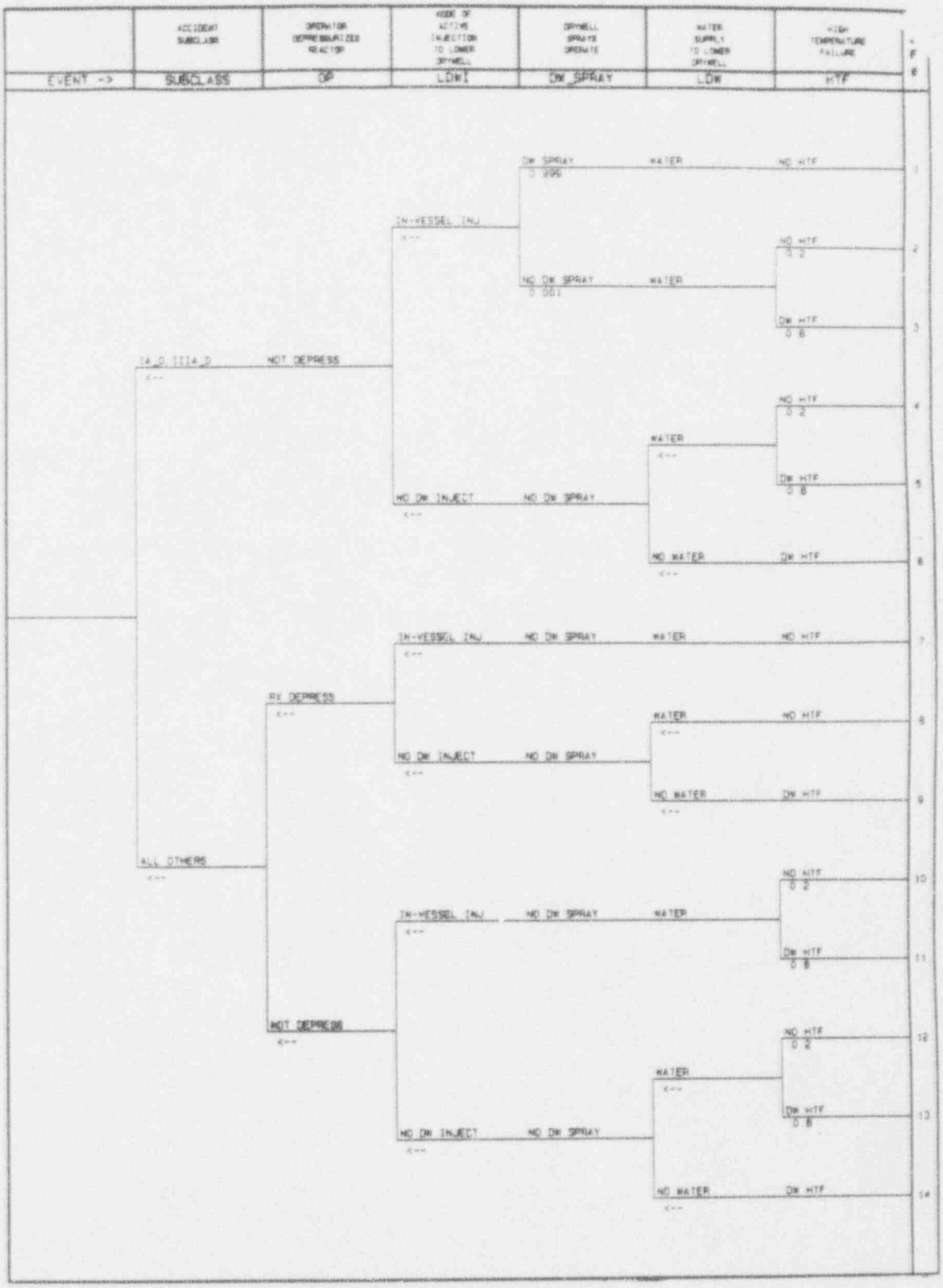


Figure 19D.5-21

CONTAINMENT EVENT EVALUATION DET FOR HIGH-TEMPERATURE FAILURE

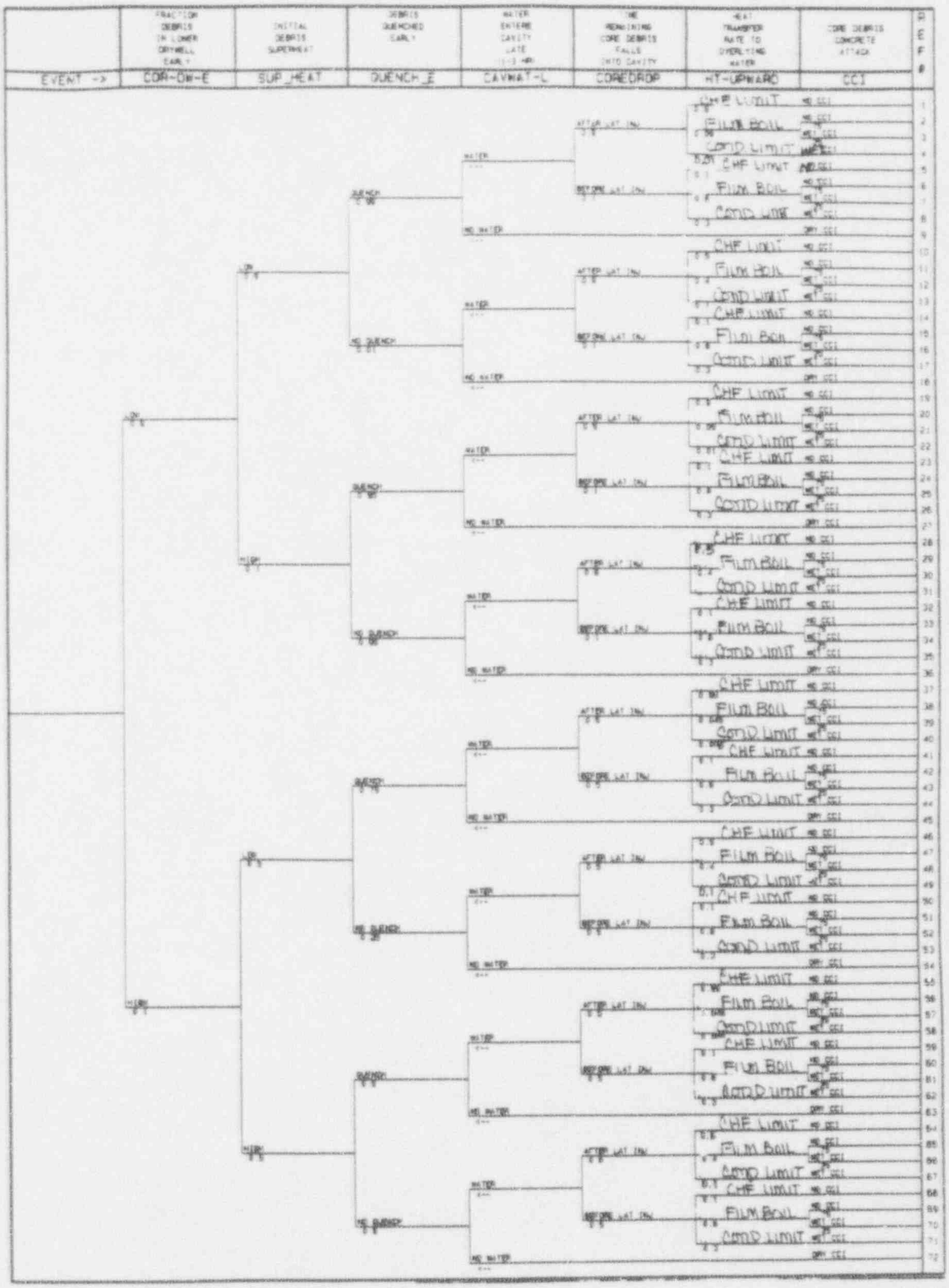


Figure 19D.5-22
CORE DEBRIS CONCRETE ATTACK DET

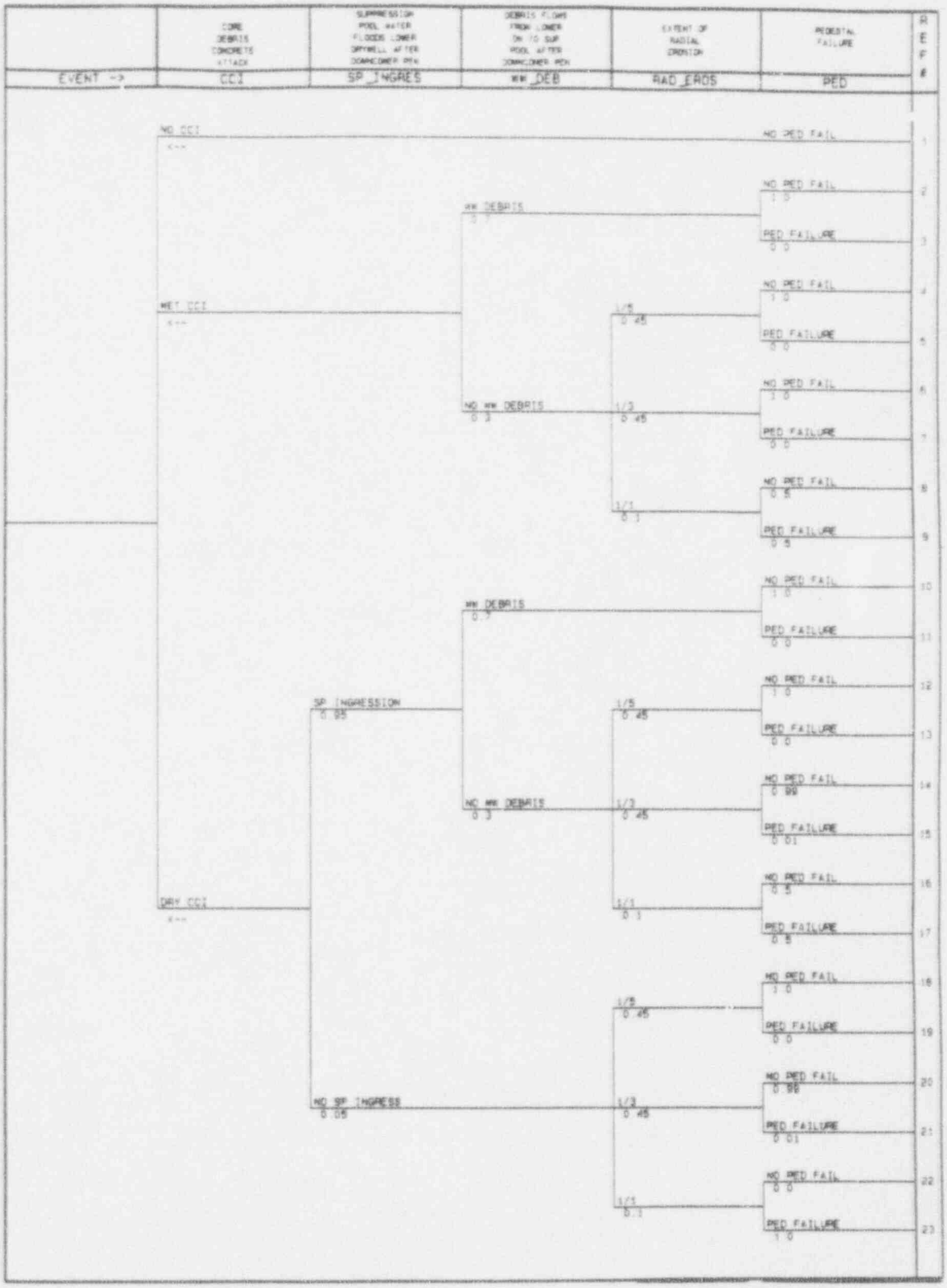


Figure 19D.5-23
CONTAINMENT EVENT EVALUATION DET FOR PEDESTAL FAILURE

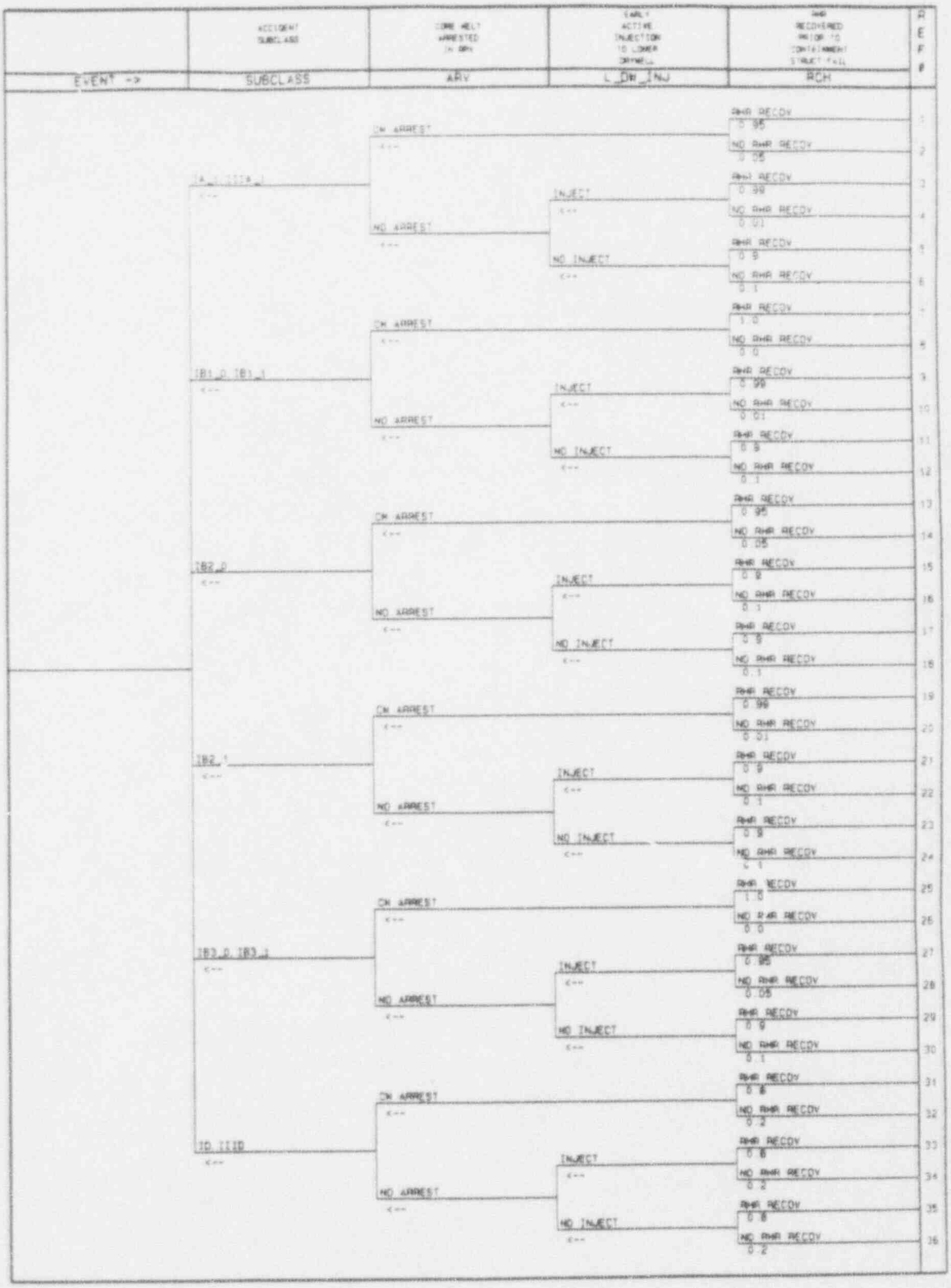


Figure 19D.5-24

CONTAINMENT EVENT EVALUATION DET FOR RHR RECOVERY PRIOR TO CONTAINMENT STRUCT FAILURE

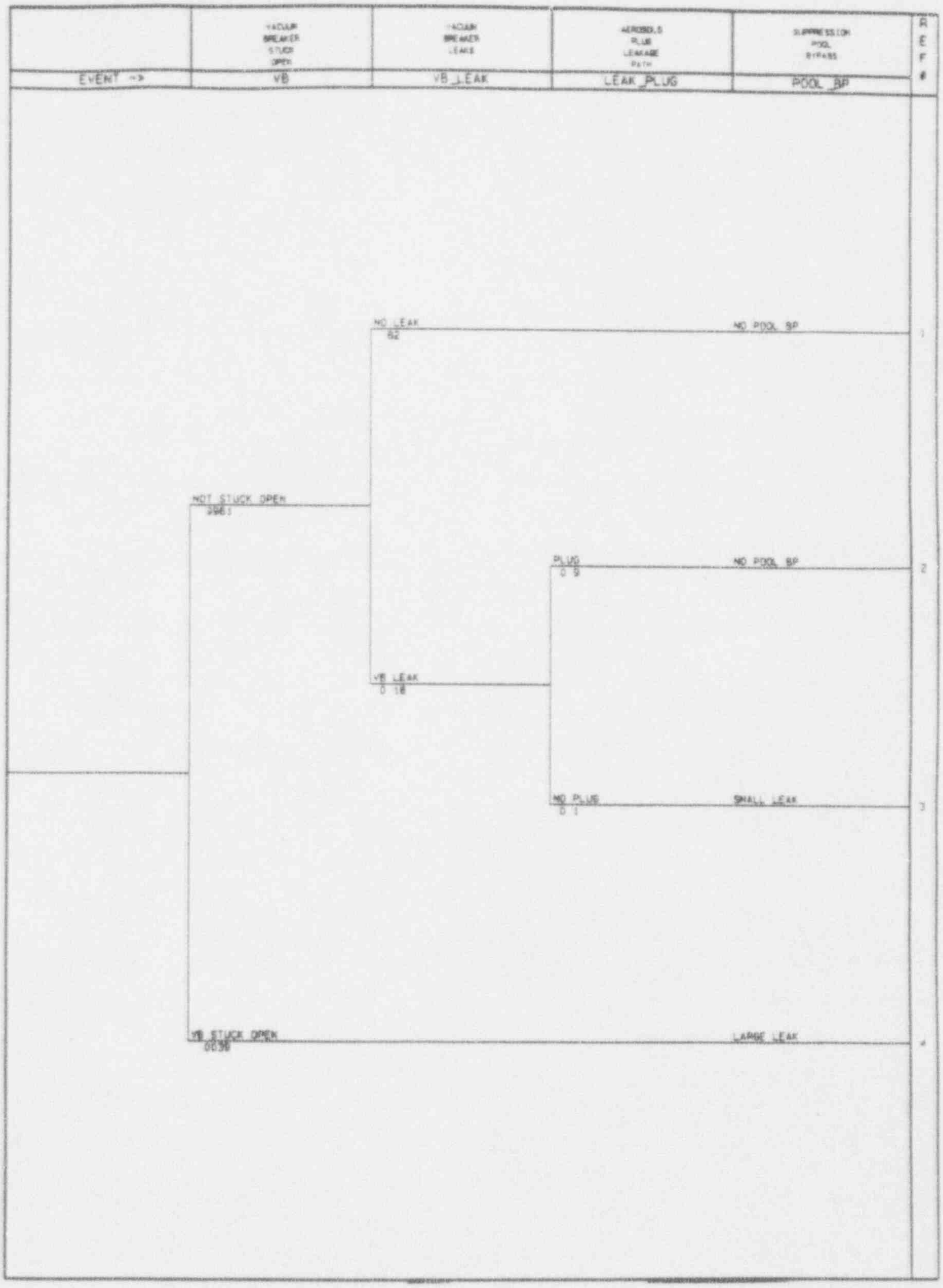


Figure 19D.5-25

CONTAINMENT EVENT EVALUATION DET FOR SUPPRESSION POOL BYPASS

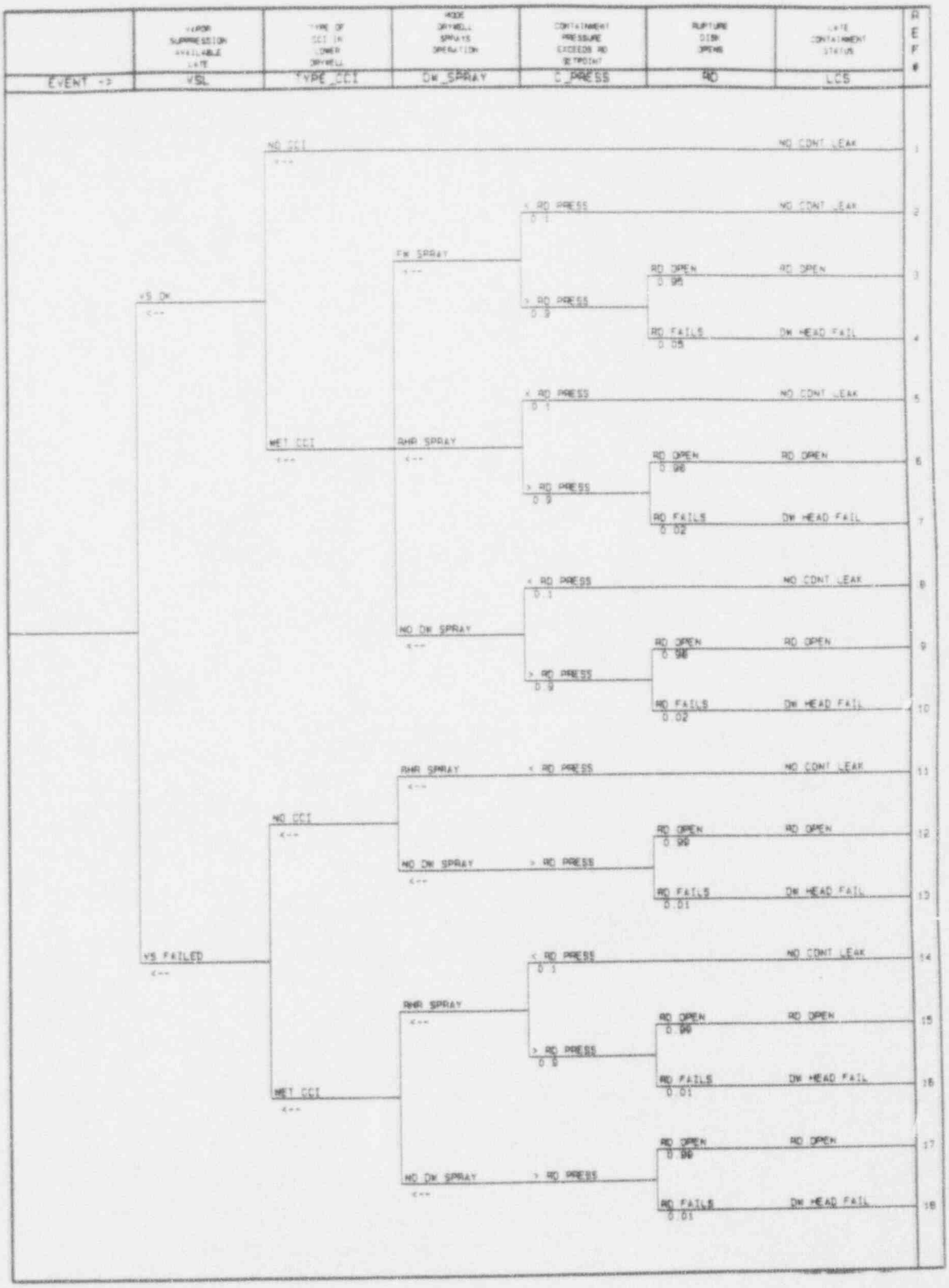


Figure 19D.5-26

CONTAINMENT EVENT EVALUATION DET FOR LATE CONTAINMENT STATUS FOR SEQUENCES WITH RHR AVAILABLE AT CORE DAMAGE

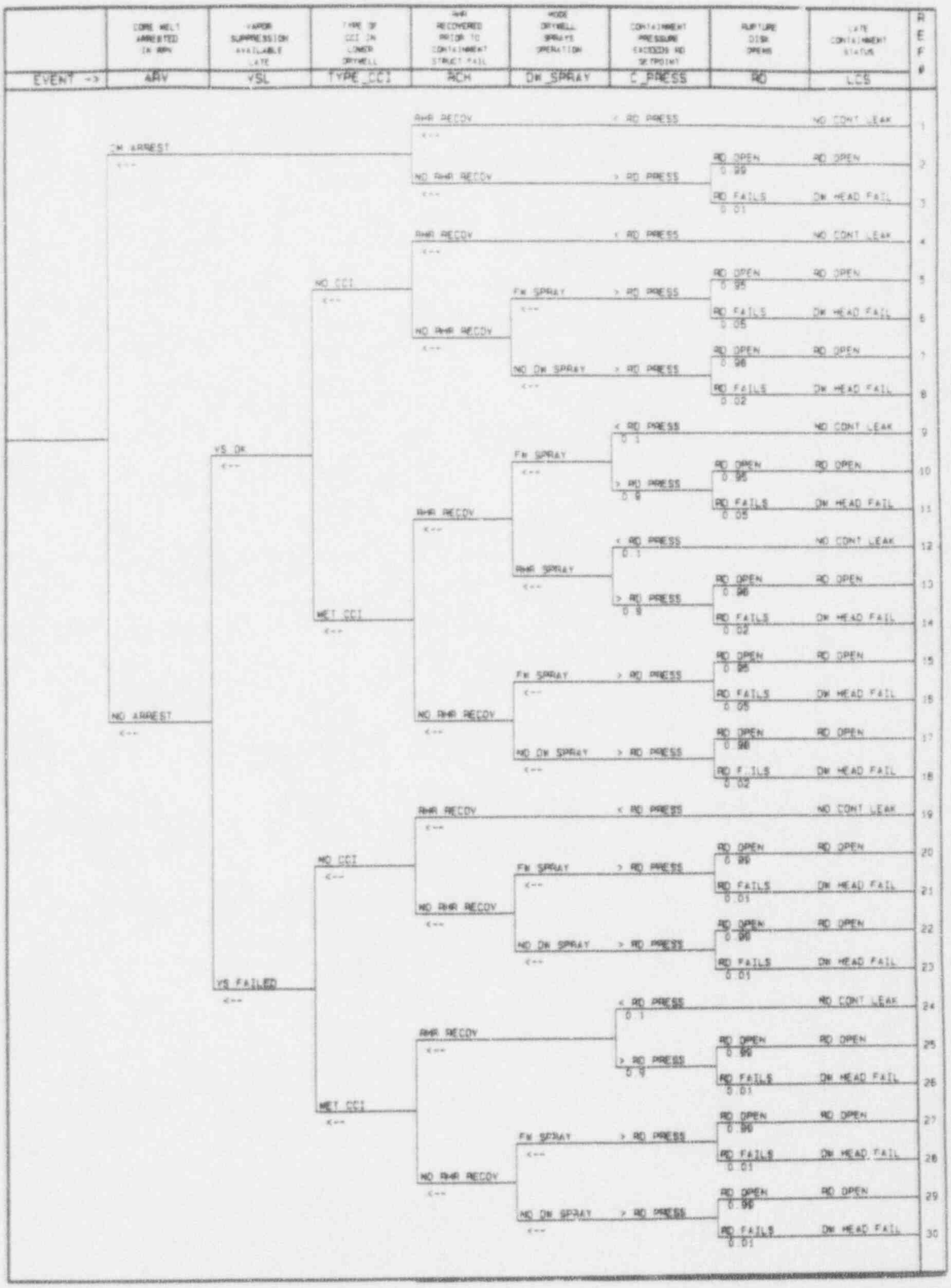


Figure 19D.5-27

CONTAINMENT EVENT EVALUATION DET FOR LATE CONTAINMENT STATUS FOR SEQUENCES WITH RHR NOT AVAILABLE AT CORE DAMAGE

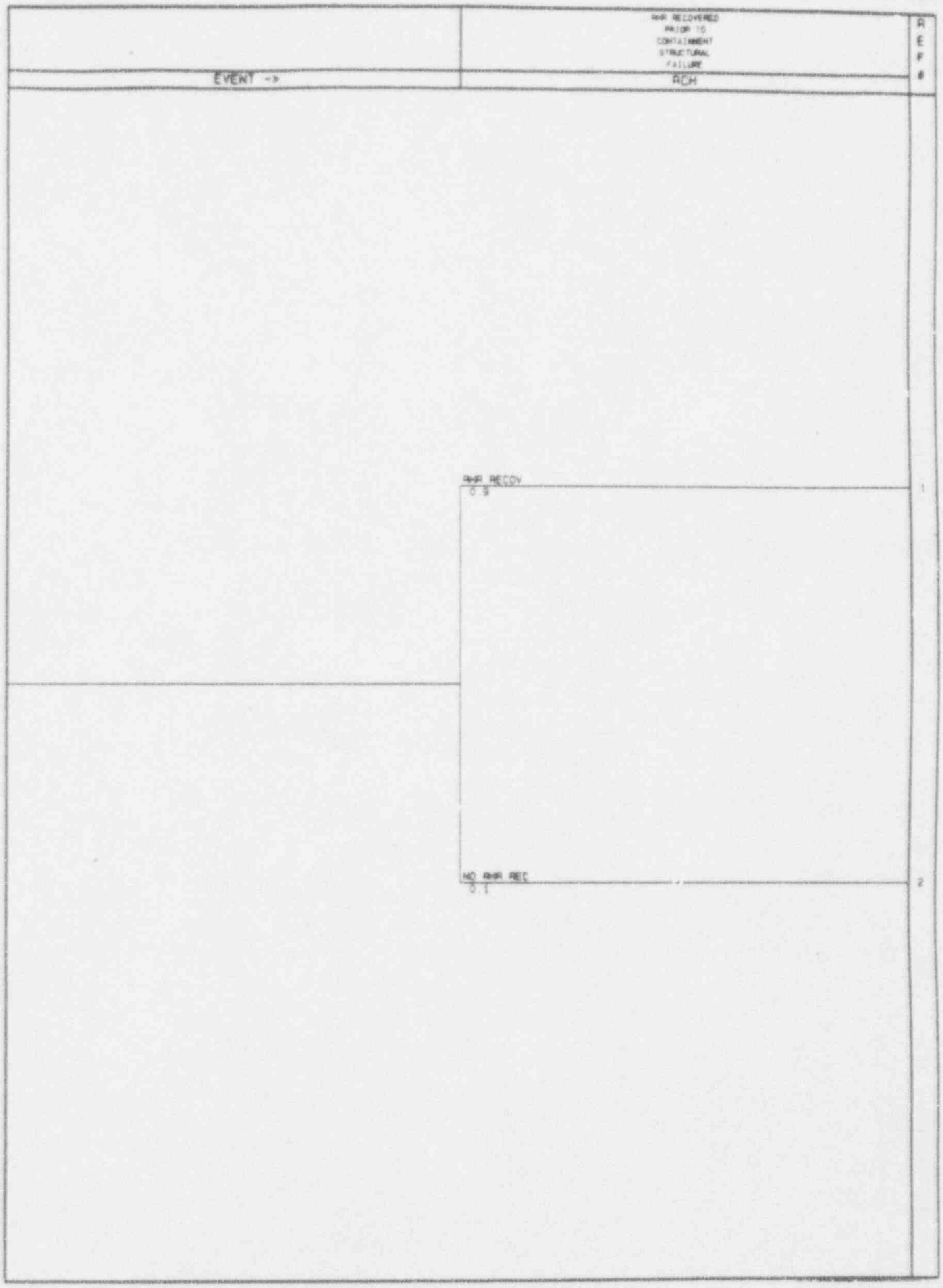


Figure 19D.5-28

CONTAINMENT EVENT EVALUATION DET FOR RHR RECOVERY PRIOR TO RUPTURE DISK SETPOINT PRESSURE (CLASS II)

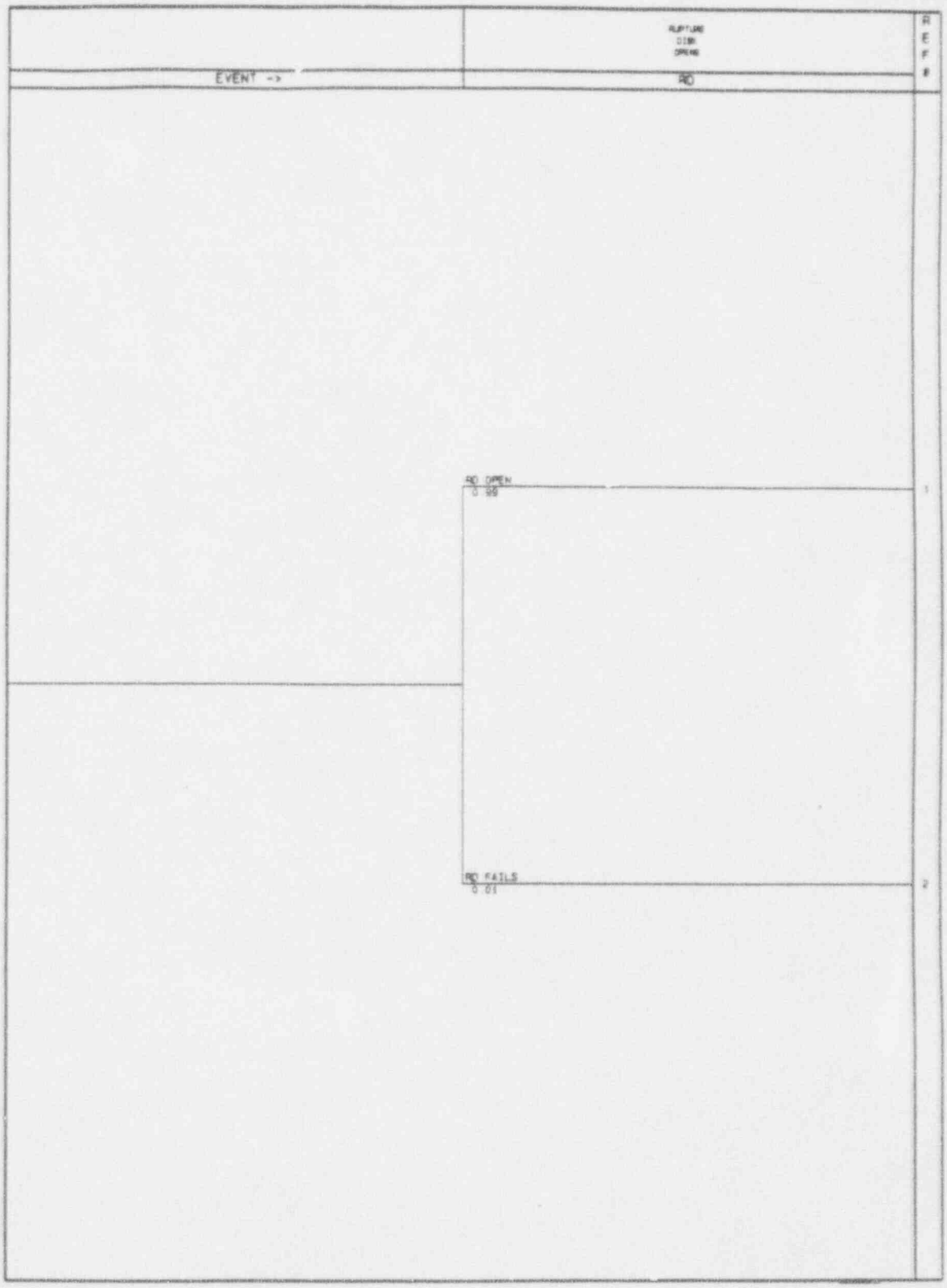


Figure 19D.5-29

CONTAINMENT EVENT EVALUATION DET FOR RUPTURE DISK OPENS (CLASS II)

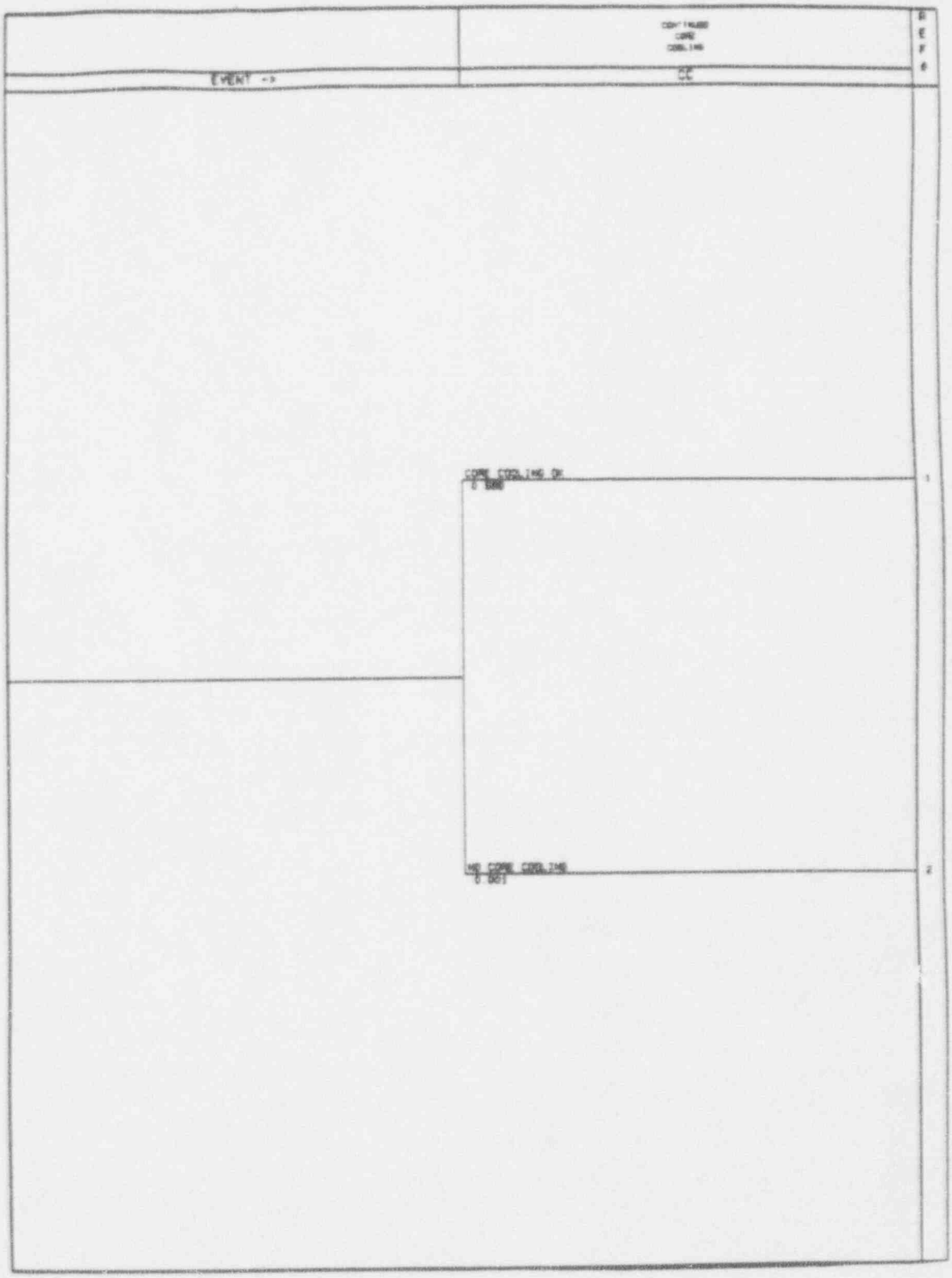


Figure 19D.5-30

CONTAINMENT EVENT EVALUATION DET FOR CORE COOLING RECOVERY (CLASS II)