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

March 15, 2000

Mr. Charles H. Cruse
Vice President - Nuclear Energy
Baltimore Gas and Electric Company
Calvert Cliffs Nuclear Power Plant
1650 Calvert Cliffs Parkway
Lusby, MD 20657-4702

SUBJECT: SITE-SPECIFIC WORKSHEETS FOR USE IN THE NUCLEAR REGULATORY COMMISSION'S SIGNIFICANCE DETERMINATION PROCESS - CALVERT CLIFFS NUCLEAR POWER PLANT, UNIT NOS. 1 AND 2 (TAC NO. MA6544)

Dear Mr. Cruse:

The purpose of this letter is to provide you with one of the key implementation tools to be used by the Nuclear Regulatory Commission (NRC) in the revised reactor oversight process, which is currently expected to be implemented at Calvert Cliffs Nuclear Power Plant, Unit Nos. 1 and 2 in late May 2000. Included in the enclosed Risk-Informed Inspection Notebooks are the Significance Determination Process (SDP) worksheets that inspectors will be using to risk-characterize inspection findings. The SDP is discussed in more detail below.

On January 8, 1999, the NRC staff described to the Commission plans and recommendations to improve the reactor oversight process in SECY-99-007, "Recommendations for Reactor Oversight Process Improvements." SECY-99-007 is available on the NRC's web site at www.nrc.gov/NRC/COMMISSION/SECYS/index.html. The new process, developed with stakeholder involvement, is designed around a risk-informed framework, which is intended to focus both the NRC's and licensee's attention and resources on those issues of more risk significance.

The performance assessment portion of the new process involves the use of both licensee-submitted performance indicator data and inspection findings that have been appropriately categorized based on their risk significance. In order to properly categorize an inspection finding, the NRC has developed the SDP. This process was described to the Commission in SECY-99-007A, "Recommendations for Reactor Oversight Process Improvements (Follow-up to SECY-99-007)," dated March 22, 1999, also available at the same NRC web site noted above.

The SDP for power operations involves evaluating an inspection finding's impact on the plant's capability to limit the frequency of initiating events; ensure the availability, reliability, and capability of mitigating systems; and ensure the integrity of the fuel cladding, reactor coolant system, and containment barriers. As described in SECY-99-007A, the SDP involves the use of three tables: Table 1 is the estimated likelihood for initiating event occurrence during the degraded period, Table 2 describes how the significance is determined based on remaining mitigation system capabilities, and Table 3 provides the bases for the failure probabilities associated with the remaining mitigation equipment and strategies.

Template NRR-106

As a result of the recently concluded Pilot Plant review effort, the NRC has determined that site-specific risk data is needed in order to provide a repeatable determination of the significance of an issue. Therefore, the NRC has contracted with Brookhaven National Lab (BNL) to develop site-specific worksheets to be used in the SDP review. These enclosed worksheets were developed based on your Individual Plant Examination (IPE) submittals that were requested by Generic Letter 88-20. The NRC plans to use this site-specific information in evaluating the significance of issues identified at your facility when the revised reactor oversight process is implemented industry wide. It is recognized that the IPE utilized during this effort may not contain current information. Therefore, the NRC or its contractor will conduct a site visit to discuss with your staff any changes that may be appropriate. Specific dates for the site visit have not been determined, but will be communicated to you in the near future. All site visits should be accomplished by June 2000. The NRC is not requesting a written response or comments on the enclosed worksheets developed by BNL.

We will coordinate our efforts through your licensing or risk organizations as appropriate. If you have any questions, please contact me at 301-415-3473.

Sincerely,

/RA/

Alexander W. Dromerick, Sr. Project Manager, Section I
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-317 and 50-318

Enclosure: As Stated

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As a result of the recently concluded Pilot Plant review effort, the NRC has determined that site-specific risk data is needed in order to provide a repeatable determination of the significance of an issue. Therefore, the NRC has contracted with Brookhaven National Lab (BNL) to develop site-specific worksheets to be used in the SDP review. These enclosed worksheets were developed based on your Individual Plant Examination (IPE) submittals that were requested by Generic Letter 88-20. The NRC plans to use this site-specific information in evaluating the significance of issues identified at your facility when the revised reactor oversight process is implemented industry wide. It is recognized that the IPE utilized during this effort may not contain current information. Therefore, the NRC or its contractor will conduct a site visit to discuss with your staff any changes that may be appropriate. Specific dates for the site visit have not been determined, but will be communicated to you in the near future. All site visits should be accomplished by June 2000. The NRC is not requesting a written response or comments on the enclosed worksheets developed by BNL.

We will coordinate our efforts through your licensing or risk organizations as appropriate. If you have any questions, please contact me at 301-415-3473.

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RISK-INFORMED INSPECTION NOTEBOOK FOR

CALVERT CLIFFS NUCLEAR POWER PLANT

UNITS 1 AND 2

PWR, C-E, TWO-LOOP PLANT WITH LARGE DRY CONTAINMENT

Prepared by

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Enclosure

NOTICE

This notebook was developed for the NRC's inspection teams to support risk-informed inspections. The activities involved in these inspections are discussed in "Reactor Oversight Process Improvement," SECY-99-007A, March 1999. The user of this notebook is assumed to be an inspector with an extensive understanding of plant-specific design features and operation. Therefore, the notebook is not a stand-alone document, and may not be suitable for use by non-specialists. This notebook will be periodically updated with new or replacement pages incorporating additional information on this plant. Technical errors in, and recommended updates to, this document should be brought to the attention of the following person:

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ABSTRACT

This notebook contains summary information to support the Significance Determination Process (SDP) in risk-informed inspections for the Calvert Cliffs Nuclear Power Plant, Units 1 and 2.

SDP worksheets support the significance determination process in risk-informed inspections and are intended to be used by the NRC's inspectors in identifying the significance of their findings, i.e., in screening risk-significant findings, consistent with Phase-2 screening in SECY-99-007A. To support the SDP, additional information is given in an Initiators and System Dependency table, and as simplified event-trees, called SDP event-trees, developed in preparing the SDP worksheets.

The information contained herein is based on the licensee's IPE submittal. The information is revised based on IPE updates or other licensee or review comments providing updated information and/or additional details.

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TABLES

1. INFORMATION SUPPORTING SIGNIFICANCE DETERMINATION PROCESS (SDP)

SECY-99-007A (NRC, March 1999) describes the process for making a Phase-2 evaluation of the inspection findings. In Phase 2, the first step is to identify the pertinent core damage scenarios that require further evaluation based on the specifics of the inspection findings. To aid in this process, this notebook provides the following information:

1. Initiator and System Dependency Table
2. Significance Determination Process (SDP) Worksheets
3. SDP Event Trees

The initiator and system dependency table shows the major dependencies between front-line- and support-systems, and identifies their involvement in different types of initiators. The information in this table identifies the most risk-significant front-line- and support-systems; it is not an exhaustive nor comprehensive compilation of the dependency matrix as known in Probabilistic Risk Assessments (PRAs). For pressurized water reactors (PWRs), the support systems for Reactor Coolant Pump (RCP) seals are explicitly denoted to assure that the inspection findings on them are properly accounted for. This table is used to identify the SDP worksheets to be evaluated, corresponding to the inspection's findings on systems and components.

To evaluate the impact of the inspection's finding on the core-damage scenarios, the SDP worksheets are developed and provided. They contain two parts. The first part identifies the functions, the systems, or combinations thereof that can perform mitigating functions, the number of trains in each system, and the number of trains required (success criteria) for each class of initiators. The second part of the SDP worksheet contains the core-damage accident sequences associated with each initiator class; these sequences are based on SDP event trees. In the parenthesis next to each of the sequence the corresponding event tree branch number(s) representing the sequence is included. Multiple branch numbers indicate that the different accident sequences identified by the event tree are merged into one through the boolean reduction. The classes of initiators that are considered in this notebook are 1) Transients, 2) Small Loss of Coolant Accident (LOCA), 3) Stuck-open Power Operated Relief Valve (PORV), 4) Medium LOCA, 5) Large LOCA, 6) Loss of Offsite Power (LOOP), 7) Steam Generator Tube Rupture (SGTR), and 8) Anticipated Transients Without Scram (ATWS). Main Steam Line Break (MSLB) events are included separately if they are treated as such in the licensee's Individual Plant Examination (IPE) submittal.

Following the SDP worksheets, the SDP event trees corresponding to each of the worksheets are presented. The SDP event trees are simplified event trees developed to define the accident sequences identified in the SDP worksheets.

The following items were considered in establishing the SDP event trees and the core-damage sequences in the SDP worksheets:

1. Event trees and sequences were developed such that the worksheet contains all the major accident sequences identified by the plant-specific IPEs. In cases where a plant-specific feature introduced a sequence that is not fully captured by our existing set of initiators and event trees, then a separate worksheet is included.

2. The event trees and sequences for each plant took into account the IPE models and event trees for all similar plants. Any major deviations in one plant from similar plants typically are noted at the end of the worksheet.

3. The event trees and the sequences were designed to capture core-damage scenarios, without including containment-failure probabilities and consequences. Therefore, branches of event trees that are only for the purpose of a Level II PRA analysis are not considered. The resulting sequences are merged using Boolean logic.

4. The simplified event-trees focus on classes of initiators, as defined above. In so doing, many separate event trees in the IPEs often are represented by a single tree. For example, some IPEs define four classes of LOCAs rather than the three classes considered here. The sizes of LOCAs for which high-pressure injection is not required are some times divided into two classes, the only difference between them being the need for reactor scram in the smaller break size. Some IPEs also may define several classes of transients, depending on the initiator's impact on the systems. Such differentiations generally are not considered in the SDP worksheets unless they could not be accounted for by the Initiator and System Dependency table.

5. Major operator actions during accident scenarios are assigned as high stress operator action or an operator action using simple, standard criteria among a class of plants. This approach resulted in the designation of some actions as high-stress operator actions, even though the PRA may have assumed a (routine) operator action; hence, they have been assigned an error probability less than 5E-2 in the IPE. In such cases, a note is given at the end of the worksheet.

The three sections that follow include the initiators and dependency table, SDP worksheets, and the SDP event-trees for the Calvert Cliffs Nuclear Power Plant, Units 1 and 2.

1.1 INITIATORS AND SYSTEM DEPENDENCY

Table 1 provides the list of the systems included in the SDP worksheets, the major components in the systems, and the support system dependencies. The system involvements in different initiating events are noted in the last column.

Table 1 Initiators and System Dependency for Calvert Cliffs Nuclear Power Plant, Units 1 and 2

Affected Systems	Major Components	Support Systems	Initiating Event
Safety Injection Tank (SIT)	Four Passive SITs (P<200 psig)	NA	LLOCA
AFWS	Two AFWTDP	125 V-DC, IA, Main Steam, Condensate, Service Water (SRW), Room cooling	Transient, Loop, MSLB/MFLB (outside Cont.), SGTR, SLOCA, ATWS
	One AFWMDP but has a cross tie to the MDP of the other unit	4.16 kV bus 11, 480 V-AC, IA	
HPI	Three high Pressure Pumps (1257 psi), two of which normally aligned to take suction from RWST. The Switch over automatic through RAS.	4.16 kV, 480 V-AC, 125 V-DC, SRW, Room cooling, CCW (Seal cooler)	Transient, Loop, MSLB/MFLB (outside Cont.), SGTR, SLOCA, ATWS
LPI / SDC	Two Pumps with separate suction from RWST discharging to a common header	4.16 kV bus 11, 480 V-AC, 125 V-DC, CCW, Room Cooling	Transient, Loop, MSLB/MFLB (outside Cont.), SGTR, SLOCA, ATWS, MLOCA, and LLOCA
CCS (Containment Spray System)	Two CS Pumps, Two spray headers, and a circular spray ring, Valves	480 V-AC bus 11 and 14, 120 V-AC, 125 V-DC, CCW, RHR/LPI Hx	Transient, Loop, MSLB/MFLB (outside Cont.), SGTR, SLOCA, ATWS, MLOCA, and LLOCA
EDG	3 EDGs for both units	125 V-DC, 120 V-AC, HVAC, SRW	LOOP
CCW	Three MDPs in two trains with two HXs cooled by Salt Water (SW), a head tank, an additive tank and the valves.	125 V-DC, 120 V-AC, 480 V-AC, 4 kV, SW	Transient, Loop, MSLB/MFLB (outside Cont.), SGTR, SLOCA, ATWS, MLOCA, and LLOCA, ATWS

Table 1 (Continued)

Affected Systems	Major Components	Support Systems	Initiating Event
Service Water System (SRW)	Two cross connected trains, each with one Pump and one heat exchanger. A third pump could also supply the two train if needed.	4.16 kV, 480 V-AC, 125 V-DC, 120 V-AC	Transient, Loop, MSLB/MFLB (outside Cont.), SGTR, SLOCA, ATWS, MLOCA, and LLOCA, ATWS
Salt Water System (SW)	Two trains, each with a saltwater pump, a CCW Hx, a SRW Hx, and ECCS pump room air cooler. A third pump could be aligned to each train if needed.	4.16 kV, 480 V-AC, 125 V-DC, 120 V-AC	Transient, Loop, MSLB/MFLB (Outside Containment), SGTR, SLOCA, ATWS, MLOCA, and LLOCA, ATWS
Main Steam Line (MSL)	2 MSIVs, eight safety valves (MSSVs), two ADVs, and four TBVs with total capacity of 40%	Instrument air system, 125 V-DC, and 120 V-AC (for all valves except MSSVs)	Transient, Loop, MSLB/MFLB (outside Cont.), SGTR, SLOCA, ATWS, MLOCA, and LLOCA, ATWS
Primary Safety Relief Valves (PSV)	Two Spring Loaded Safety Relief Valves (P>2500 psig)	None	ATWS
PORVs / Block Valves	Two reverse-seated PORVs (2400 psi) and Two Block Valves	480 V-AC for both PORV and Block but from opposite bus 125 V-DC is require to actuate the relay for the valve operation	Transient, Loop, MSLB/MFLB (outside Cont.), SGTR, SLOCA, ATWS, MLOCA, and LLOCA, ATWS
RCP	BJ Seals (4 stage including vapor seal)	1/3 CCW pump for seal cooling. Operator trips RCPs within 45 minutes of loss of seal cooling, and loss of CCW to seal cooling	Transients, LOOP, SLOCA from RCP.
CVCS	3 Charging Pumps and 2 boric acid pumps	480 V-AC, 125 V-DC, Compressed air from salt water system (SWAC)	SGTR, SLOCA, ATWS, MLOCA, LLOCA, and ATWS

Table 1 (Continued)

Affected Systems	Major Components	Support Systems	Initiating Event
Compressed Air (SWAC)	Upon SIAS the normal IA is isolated and the two SW system air Compressor will start. Connection to AFW valves will be manual.	4 kV, and 480 V-AC	SGTR, SLOCA, ATWS, MLOCA, LLOCA, and ATWS, LOOP

Notes:

(1) The information is based on IPE submittal of 12/30/93 with the CDF of 2.4E-4.

(2) Failure of primary safety valve and PORV to re-close is considered to be equivalent to a very small LOCA (>0.005 sq ft or approximately D>1")

1.2 SDP WORKSHEETS

This section presents the SDP worksheets to be used in the Phase 2 evaluation of the inspection findings for the Calvert Cliffs Nuclear Power Plant, Units 1 and 2. The SDP worksheets are presented for the following initiating event categories:

1. Transients
2. Small LOCA
3. Stuck-open PORV (Not applicable to this plant)
4. Medium LOCA
5. Large LOCA
6. LOOP
7. Steam Generator Tube Rupture (SGTR)
8. Anticipated Transients Without Scram (ATWS)
9. Main Steam Line Break (MSLB)

Table 2.1 SDP Worksheet for Calvert Cliffs Nuclear Power Plant, Units 1 & 2 —— Transients (Reactor Trip)

Estimated Frequency (Table 1 Row) ________ Exposure Time ________ Table 1 Result (circle): A B C D E F G H

<u>Safety Functions Needed</u>:	<u>*Full* Creditable Mitigation Capability for Each Safety Function</u>:
Power Conversion System (PCS)	1 / 3 condensate and condensate booster pump with operation of 1/4 TBVs (operator action)
Secondary Heat Removal (AFW)	1 / 2 TDAFW trains (2 ASD train) or 1 MDAFW train (1 train) or cross connect to other unit MDAFW (high stress operator action) with steam discharge through 1/2 ADV, or 1/4 TBV, or 1/8 MSSVs
High Pressure Injection (EIHP)	1 / 2 HPI Pumps from RWT (1 multi-train system)[(1)]
Primary Heat Removal, Feed/Bleed (FB)	2/2 PORV to open for Feed/Bleed and initiate HPI cooling (High stress operator action)[(2)]
High Pressure Recirculation (HPR)	1/2 HPI pump to operate in recirculation modes after the RAS initiation(1 multi train system)
Containment Heat Removal (CNT)	2/4 Containment air cooler (1 multi-train system), or 1/2 containment spray with alignment of SDC Hx (operator action)

<u>Circle Affected Functions</u>	<u>Recovery of Failed Train</u>	<u>*Remaining Mitigation Capability Rating for Each Affected Sequence*</u>	<u>Sequence Color</u>
1 TRANS - PCS - AFW - FB (7)			
2 TRANS - PCS - AFW -EIHP (6)			
3 TRANS - PCS - AFW - HPR (5)			
4 TRANS - PCS -AFW - CNT (4)			

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event: If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Notes:

(1) The third HPI pump could be manually aligned for OTCC during transients as a recovery action.

(2) The Once-Through-Core-Cooling (OTCC) would require of opening of 1/8 MSSVs once all feed to SG is lost. If the RCPs are running The OTCC should be done in less than 10 minutes, if the RCPs are tripped then about 40 minutes would be available. The driving factor is the water level and the initial inventory in the SGs. A high stress operator action is assigned to OTCC initiation to include the SG inventory issue discussed here.

Table 2.2 SDP Worksheet for Calvert Cliffs Nuclear Power Plant, Units 1 & 2 — Transients with Loss of PCS (TPCS)

Estimated Frequency (Table 1 Row) ________ Exposure Time ________ Table 1 Result (circle): A B C D E F G H

Safety Functions Needed:	*Full* Creditable Mitigation Capability for Each Safety Function:
Power Conversion System (PCS)	1 / 3 condensate and condensate booster pump with operation of 1/4 TBVs (operator action)
Secondary Heat Removal (AFW)	1 / 2 TDAFW trains (2 ASD train) or 1 MDAFW train (1 train) or cross connect to other unit MDAFW (high stress operator action) with steam discharge through 1/2 ADV, or 1/4 TBV, or 1/8 MSSVs
High Pressure Injection (EIHP)	1 / 2 HPI Pumps from RWT (1 multi-train system)(1)
Primary Heat Removal, Feed/Bleed (FB)	2/2 PORV to open for Feed/Bleed and initiate HPI cooling (High stress operator action)(2)
High Pressure Recirculation (HPR)	1/2 HPI pump to operate in recirculation modes after the RAS initiation(1 multi train system)
Containment Heat Removal (CNT)	2/4 Containment air cooler (1 multi-train system), or 1/2 containment spray with alignment of SDC Hx (operator action)

Circle Affected Functions	Recovery of Failed Train	*Remaining Mitigation Capability Rating for Each Affected Sequence*	Sequence Color
1 TPCS - AFW - FB (7)			
2 TPCS - AFW -EIHP (6)			
3 TPCS - AFW - HPR (5)			
4 TPCS -AFW - CNT (4)			

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:
If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Notes:

(1) The third HPI pump could be manually aligned for OTCC during transients as a recovery action.

(2) The Once-Through-Core-Cooling (OTCC) would require of opening of 1/8 MSSVs once all feed to SG is lost. If the RCPs are running The OTCC should be done in less than 10 minutes, if the RCPs are tripped then about 40 minutes would be available. The driving factor is the water level and the initial inventory in the SGs. A high stress operator action is assigned to OTCC initiation to include the SG inventory issue discussed here.

Table 2.3 SDP Worksheet for Calvert Cliffs Nuclear Power Plant, Units 1 & 2 —— Small LOCA (1)

Estimated Frequency (Table 1 Row) _________ Exposure Time _________ Table 1 Result (circle): A B C D E F G H

<u>Safety Functions Needed</u>:	<u>*Full* Creditable Mitigation Capability for Each Safety Function</u>:
Early Inventory, HP Injection (EIHP)	1 / 3 HPI Pumps from RWT (1 multi-train system)
Power Conversion System (PCS)	1 / 3 condensate and condensate booster pump with operation of 1/4 TBVs (Recovery action)
Secondary Heat Removal (AFW)	1 / 2 TDAFW trains (2 ASD train) or 1 MDAFW train (1 train) or cross connect to other unit MDAFW (high stress operator action) with steam discharge through 1/2 ADV, or 1/4 TBV, or 1/8 MSSVs
Primary Bleed (FB)	2/2 PORV to open for Feed/Bleed and initiate HPI cooling (operator action)(2)
High Pressure Recirculation (HPR)	1/2 HPI pump to operate in recirculation modes after the RAS initiation(1 multi train system)
Containment Heat Removal (CNT)	2/4 Containment air cooler (1 multi-train system), or 1/2 containment spray with alignment of SDC Hx (operator action)

<u>Circle Affected Functions</u>	<u>Recovery of Failed Train</u>	<u>*Remaining Mitigation Capability Rating for Each Affected Sequence*</u>	<u>Sequence Color</u>
1 SLOCA - EIHP (9)			
2 SLOCA-AFW-PCS-FB (8)			
3 SLOCA - HPR (2,4,7)			
4 SLOCA - AFW - PCS - CNT (11)			

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event: If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Notes:

(1) The IPE defines one category for small LOCA with break sizes greater than 0.95 to 1.9 inches.

(2) The Once-Through-Core-Cooling (OTCC) would require of opening of 1/8 MSSVs once all feed to SG is lost. The RCPs are assumed tripped and the FB is assigned as the operator action.

Table 2.4 SDP Worksheet for Calvert Cliffs Nuclear Power Plant, Units 1 & 2 —— Stuck Open PORV (SORV) [1]

Estimated Frequency (Table 1 Row) ________ Exposure Time ________ Table 1 Result (circle): A B C D E F G H

Safety Functions Needed:	***Full* Creditable Mitigation Capability for Each Safety Function:**
Blocking Leakage (BLK)	Closure of the associated Block Valve (Operator action)
Early Inventory, HP Injection (EIHP)	1 / 3 HPI Pumps from RWT (1 multi-train system)
Power Conversion System (PCS)	1 / 3 condensate and condensate booster pump with operation of 1/4 TBVs (Recovery action)
Secondary Heat Removal (AFW)	1 / 2 TDAFW trains (2 ASD train) or 1 MDAFW train (1 train) or cross connect to other unit MDAFW (high stress operator action) with steam discharge through 1/2 ADV, or 1/4 TBV, or 1/8 MSSVs
Primary Bleed (FB)	2/2 PORV to open for Feed/Bleed and initiate HPI cooling (operator action)[2]
High Pressure Recirculation (HPR)	1/2 HPI pump to operate in recirculation modes after the RAS initiation(1 multi train system)
Containment Heat Removal (CNT)	2/4 Containment air cooler (1 multi-train system), or 1/2 containment spray with alignment of SDC Hx (operator action)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 SORV - BLK - EIHP (9)			
2 SORV - BLK-AFW-PCS-FB (8)			
3 SORV - BLK - HPR (2,4,7)			
4 SORV - BLK - AFW - PCS - CNT (11)			

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:
If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

<u>Notes</u>:

(1) The IPE defines one Stuck open PORV similar to small LOCA with break sizes greater than 0.95 to 1.9 inches.

(2) The Once-Through-Core-Cooling (OTCC) would require of opening of 1/8 MSSVs once all feed to SG is lost. The RCPs are assumed tripped and the FB is assigned as the operator action.

Table 2.5 SDP Worksheet for Calvert Cliffs Nuclear Power Plant, Units 1 & 2 —— Medium LOCA (1)

Estimated Frequency (Table 1 Row) __________ Exposure Time __________ Table 1 Result (circle): A B C D E F G H

<u>Safety Functions Needed</u>:	***<u>Full</u>* <u>Creditable Mitigation Capability for Each Safety Function</u>:**
Early Inventory, HP Injection (EIHP)	1/3 HPSI trains (1 multi-train systems)
High Pressure Recirculation (HPR)	1/3 HPSI pumps taking suction from sump and auto-aligned by RAS (1 multi-train system) (1 multi-train system)
Containment Heat Removal (CNT)	2/4 Containment air cooler (1 multi-train system), or 1/2 containment spray with alignment of SDC Hx (operator action)

<u>Circle Affected Functions</u>	**<u>Recovery of Failed Train</u>**	***<u>Remaining Mitigation Capability Rating for Each Affected Sequence</u>***	**<u>Sequence Color</u>**
1 MLOCA - EIHP (4)			
2 MLOCA - HPR (2)			
3. MLOCA - CNT (3)			

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:

If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Notes:

(1) Medium LOCA is defined for break sizes greater than 1.9 inches up to 4.3 inches. In this LOCA category decay heat will be removed by the flow through the break.

Table 2.6 SDP Worksheet for Calvert Cliffs Nuclear Power Plant, Units 1 & 2 —— Large LOCA[1]

Estimated Frequency (Table 1 Row) ________	Exposure Time ________	Table 1 Result (circle): A B C D E F G H	

Safety Functions Needed:	***Full* Creditable Mitigation Capability for Each Safety Function:**
Early Inventory Control (EIAC)	3/4 SITs (1 train system)[2]
High Pressure Injection (EIHP)	1/2 HPI pumps taking suction from RWT (1 multi-train system)
Early Inventory, LP Injection (EILP)	1/ 2 LPSI pumps (1 multi-train system)[2]
High Pressure Recirculation (HPR)	2/2 HPI pump trains taking suction from containment sump and auto-aligned by RAS including the long term core flush function (operator action)
Containment Heat Removal (CNT)	2/4 Containment air cooler (1 multi-train system), or 1/2 containment spray with alignment of SDC Hx (operator action)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 LLOCA - EILP[2] (4)			
2 LLOCA - EIAC (5)			
3 LLOCA - EIHP (6)			
4 LLOCA - CNT (3)			
5 LLOCA - HPR (2)			

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event: If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Notes:

(1) Large LOCA is for break sizes greater than 4.3 inches.

(2) For Large LOCAs the availability of 4/4 SITs or 3/4 SITs and 1/2 LPSI would be sufficient for early inventory control. The SDP sheet conservatively assumes that the function EILP is always needed for these sizes of break.

Table 2.7 SDP Worksheet for Calvert Cliffs Nuclear Power Plant, Units 1 & 2 —— LOOP

Estimated Frequency (Table 1 Row) ________ Exposure Time ________ Table 1 Result (circle): A B C D E F G H

Safety Functions Needed:	***Full* Creditable Mitigation Capability for Each Safety Function:**
Emergency AC Power (EAC)	1/2 EDGs (1 multi-train system)[1]
Turbine-driven AFW pump (TDAFW)	Operation of 1/2 TDAFW pump (2 ASD train)[2]
Recovery of AC Power in < 1 hrs (REC1)	SBO procedure and Recovery of an AC source in one hour [3] (high Stress operator action)
Recovery of AC Power in < 3 hrs (REC3)	SBO procedure and Recovery of an AC source in 3 hours [4] (operator action)
Early Inventory, HP Injection (EIHP)	1/2 HPI trains (1 multi-train system)
Secondary Heat Removal (AFW)	1 / 1 MDAFW trains (1 train) or 1 TDAFW train (1 ASD train)
Primary Heat Removal (FB)	2/2 PORVs open for Feed/Bleed and initiate HPI cooling (operator action)
Long Term Cooling (SDC)	1/2 LPI pump and the associated Heat Exchanger. In SDC mode (operator action)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 LOOP - EAC - REC3 (7) (AC can not be restored in 3 hours)			
2 LOOP - EAC - TDAFW - REC1 (8)			
3 LOOP - EAC - REC1 - EIHP (6) (AC recovered prior to core uncovery)[5]			
4 LOOP - EAC - REC1 - FB (5) (AC recovered prior to core uncovery)			

5 LOOP - EAC - REC1 - SDC (4)			
6 LOOP - AFW - EIHP (1,2)[6] (AC recovered in less than one hour)			
7 LOOP - AFW - FB (1,2) (AC recovered in less than 1 hour)			
8 LOOP - AFW - SDC (1,2) (AC recovered in less than 1 hour)			

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:

If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Notes:

General Note: The IPE information is very sparse and in light of proposed recommendation for increasing the number of EDGs as a result of the IPE, it is strongly recommended that these sheets to be revisited and updated by newer information.

(1) There is three EDGs for both units. One recommendation from IPE was to add a safety EDG and an SBO EDG to the plant for a total of five EDGs. This would results in two EDGs per unit plus one SBO EDG for both units. This modification is not reflected in the current SDP sheets.

(2) The turbine driven pump is assumed to operate for 12 hours before the need for HVAC. The batteries are assumed to be depleted in two hours. There is a potential for CCF in both TDAFW pumps.

(3) Core damage is assumed to occur in one hour if no secondary heat removal. IPE uses 60 minutes as maximum delay for initiation of once through cooling in T2 transients.

(4) Core damage is assumed in one hour after battery depletion, that is REC3 refers to recovery of offsite power after battery depletes but prior to core uncovery.

(5) A major contributor to sequences 3 and 4 is recovery of AC source in less than 3 hours but after the battery is depleted (e.g. after 2 hours).

(6) Sequences 6,7, and 8 basically are transient type sequences with loss of decay heat removal. If AC power is either available or become available in less than one hour the plant response would be similar to transients.

Table 2.8 SDP Worksheet for Calvert Cliffs Nuclear Power Plant, Units 1 & 2 —— SGTR

Estimated Frequency (Table 1 Row) _________ Exposure Time _________ Table 1 Result (circle): A B C D E F G H

Safety Functions Needed:	***Full* Creditable Mitigation Capability for Each Safety Function:**
Early Inventory HP injection (EIHP)	1/3 HPSI trains (1 multi-train system)
Secondary Heat Removal (AFW)	1 / 2 TDAFW trains (2 ASD train) or 1 MDAFW train (1 train) or cross connect to other unit MDAFW (high stress operator action) with steam discharge through 1/2 ADV, or 1/4 TBV, or 1/8 MSSVs
Primary/Secondary Pressure Equalization and Cool Down (ISO/EQ)	Pressure equalization below SG safety setpoints ;assumes secondary cooling available for rapid cool down (High stress operator action)
Late Equalization and Isolation (LISO/LEQ)	Primary depressurization and equalization with subsequent isolation of the affected SG (operator action- 4.75 hours after SGTR)
Shutdown Cooling (SDC)	1/2 LPI in DHR cooling mode or make up to RWST for continued operation of EIHP (operator action)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 SGTR - AFW (6)			
2 SGTR - ISO/EQ - EIHP (5)			
3 SGTR - ISO/EQ - LISO/LEQ (4)			
4. SGTR -ISO/EQ - SDC (3)			

Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event: If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Note:

General Note: IPE assumes conservatively that OTCC will not provide adequate decay heat removal during SGTR (see pp 3.1.2-14).

Table 2.9 SDP Worksheet for Calvert Cliffs Nuclear Power Plant, Units 1 & 2 —— ATWS

Estimated Frequency (Table 1 Row) ________	Exposure Time ________	Table 1 Result (circle): A B C D E F G H	
<u>Safety Functions Needed</u>:	**<u>Full Creditable Mitigation Capability for Each Safety Function</u>:**		
Turbine Trip (TTP)	RPS automatically actuates turbine trip even in ATWS (1 train)		
<u>Circle Affected Functions</u>	**<u>Recovery of Failed Train</u>**	***<u>Remaining Mitigation Capability Rating for Each Affected Sequence</u>***	**<u>Sequence Color</u>**
Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:			
If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.			

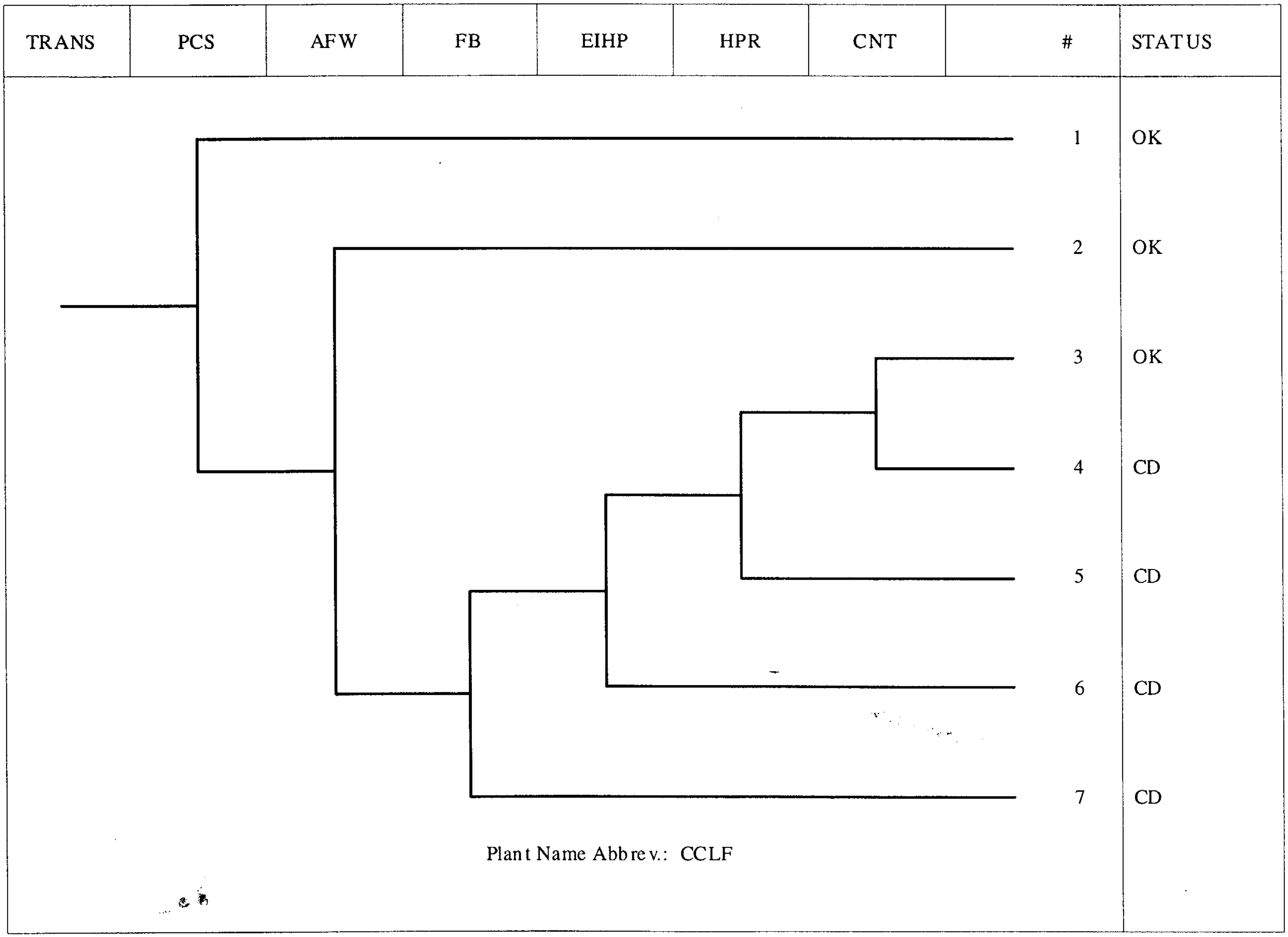
<u>Genera Note</u>: ATWS is assumed to directly lead to core damage in the IPE, p 3.1.2-6.

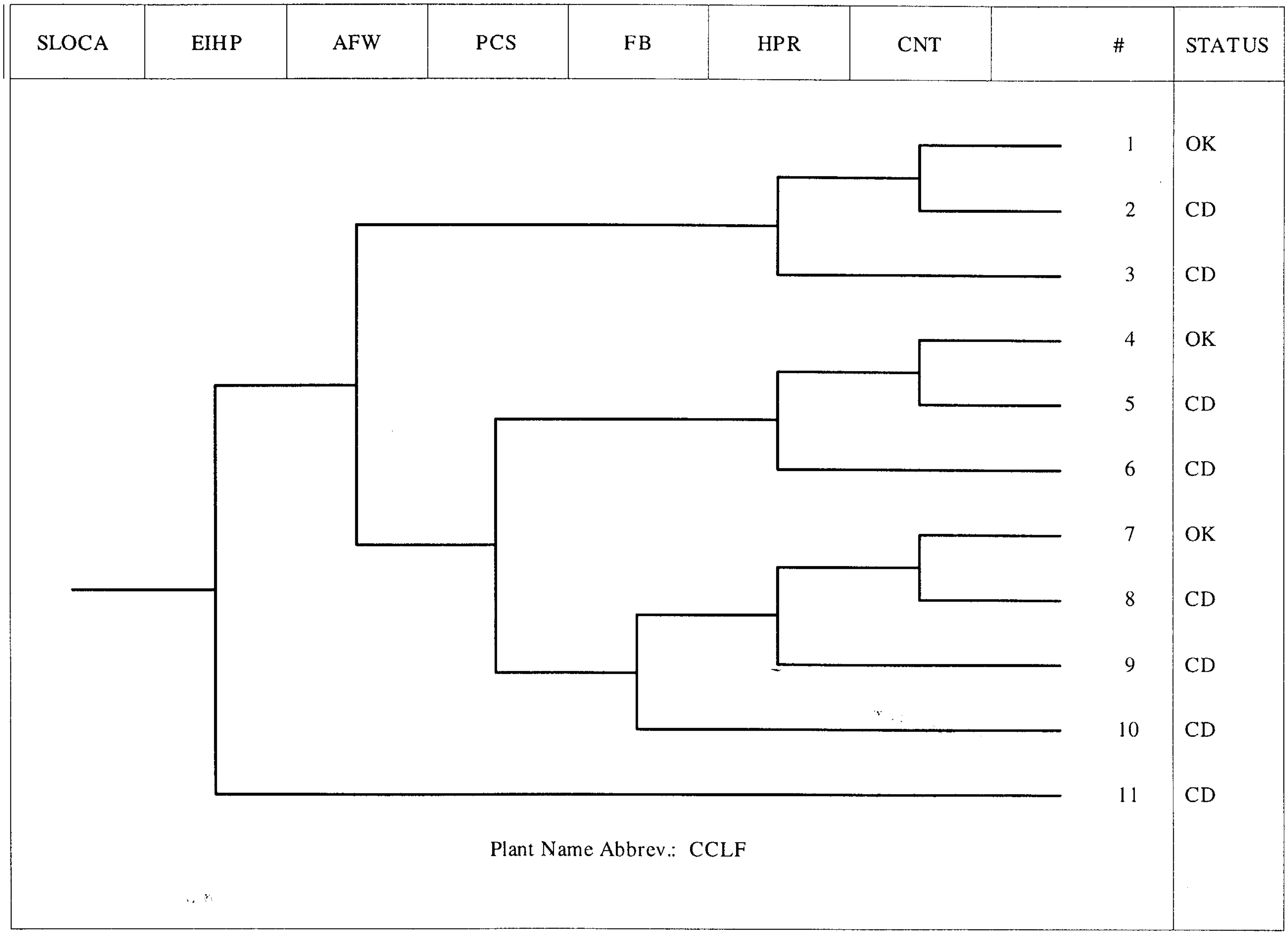
1.3 SDP EVENT TREES

This section provides the simplified event trees called SDP event trees used to define the accident sequences identified in the SDP worksheets in the previous section. An event tree for the stuck-open PORV is not included since it is similar to the small LOCA event tree. The event tree headings are defined in the corresponding SDP worksheets.

The following event trees are included:

1. Transients
2. Small LOCA
3. Medium LOCA
4. Large LOCA
5. LOOP
6. Steam Generator Tube Rupture (SGTR)


TRANS
PCS
AFW
FB
EIHP
HPR
CNT
#
STATUS
1
OK
2
OK
3
OK
4
CD
5
CD
6
CD
7
CD
Plant Name Abbrev.: CCLF


SLOCA
EIHP
AFW
PCS
FB
HPR
CNT
#
STATUS
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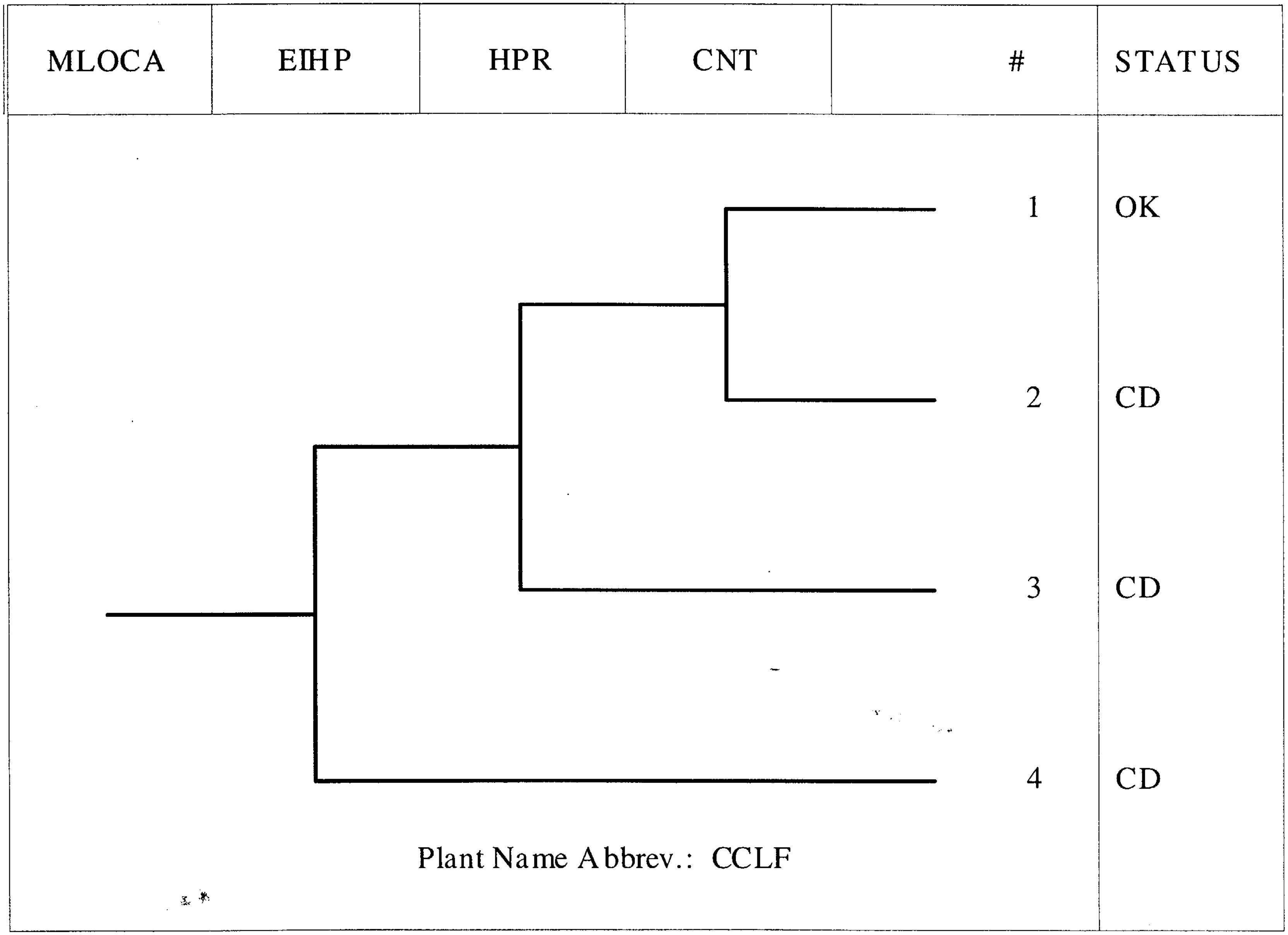
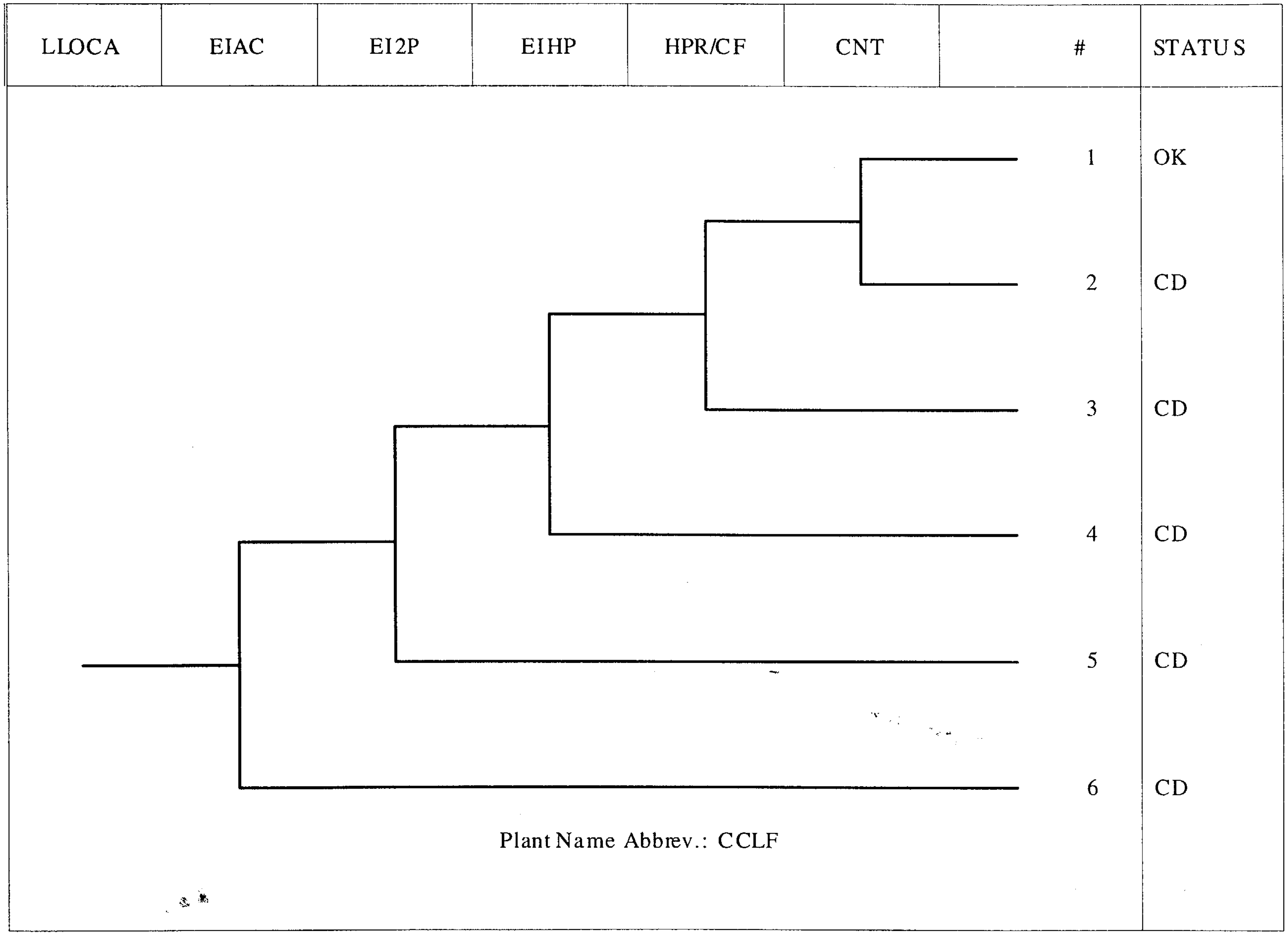
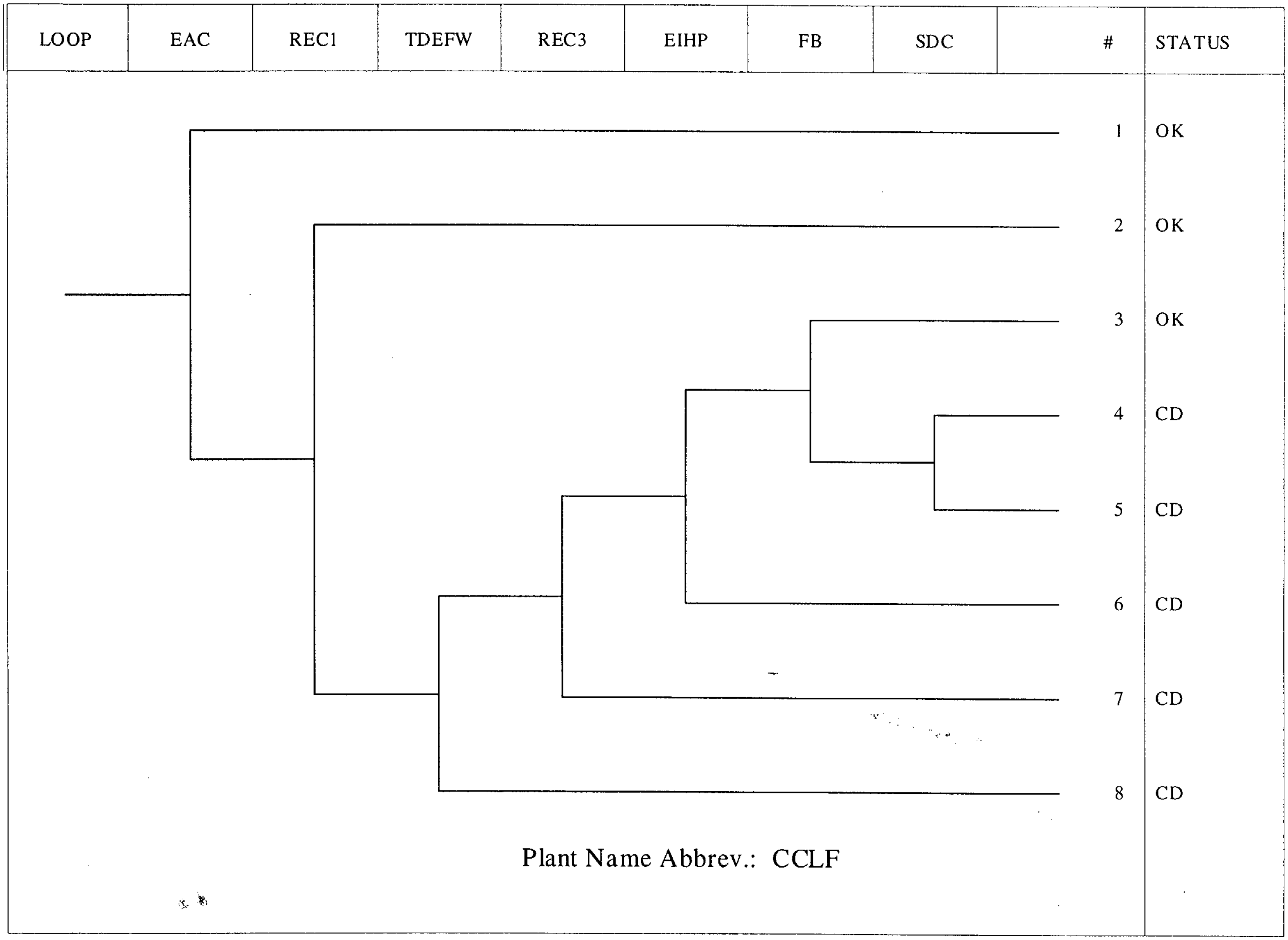
2 CD
3 CD
4 OK
5 CD
6 CD
7 OK
8 CD
9 CD
10 CD
11 CD
Plant Name Abbrev.: CCLF

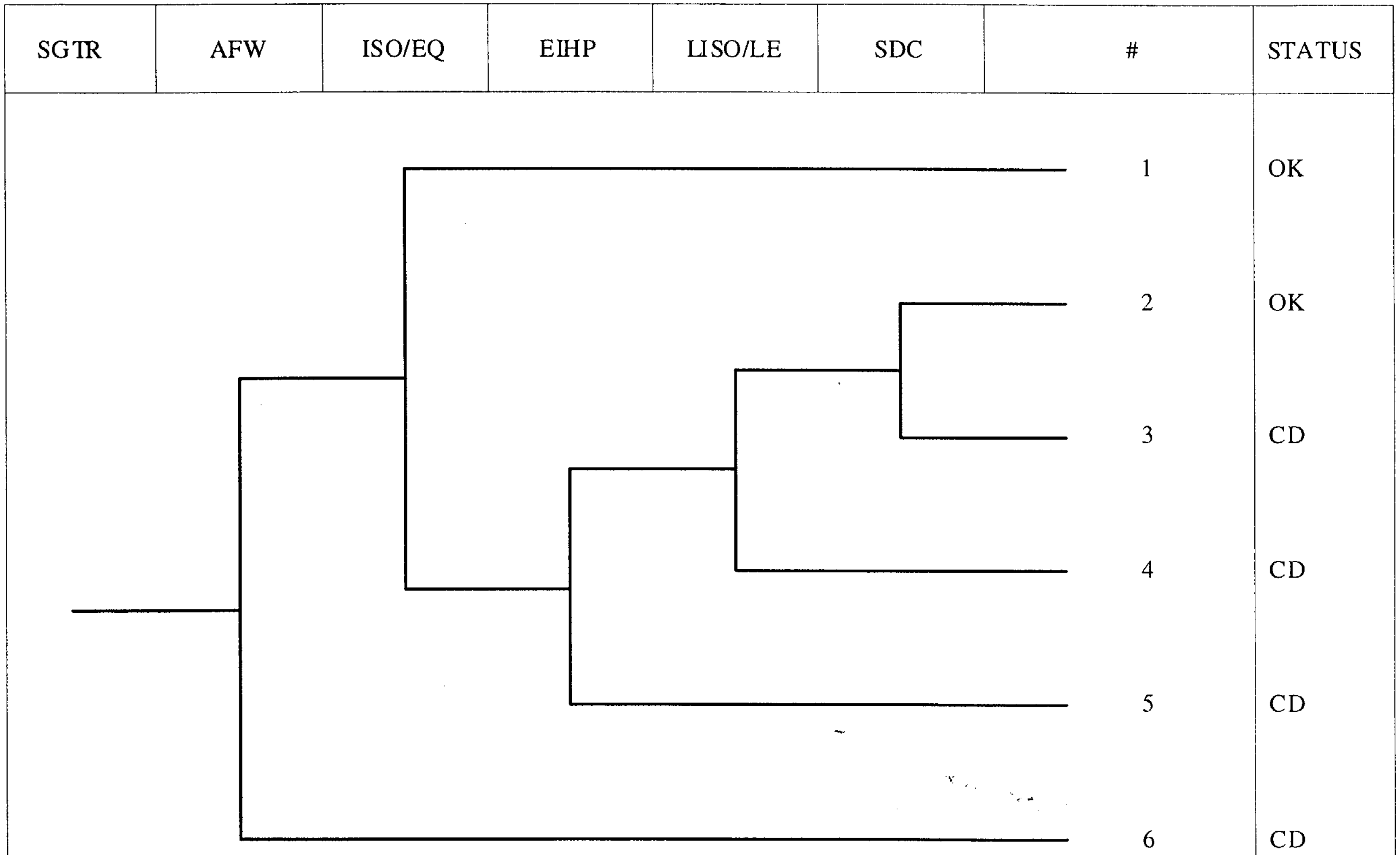

MLOCA
EIHP
HPR
CNT
#
STATUS
1
OK
2
CD
3
CD
4
CD
Plant Name Abbrev.: CCLF



LLOCA
EIAC
EI2P
EIHP
HPR/CF
CNT
#
STATUS
1
OK
2
CD
3
CD
4
CD
5
CD
6
CD
Plant Name Abbrev.: CCLF



Plant Name Abbrev.: CCLF



Plant Name Abbrev.: CCLF

2. RESOLUTION AND DISPOSITION OF COMMENTS

This section documents the comments received on the material included in this report and their resolution. This section is blank until comments are received and are addressed.

REFERENCES

1. NRC SECY-99-007A, Recommendations for Reactor Oversight Process Improvements (Follow-up to SECY-99-007), March 22, 1999.

2. Baltimore Gas and Electric Company, "Calvert Cliffs Nuclear Power Plant, Units 1 and 2 Individual Plant Examination Submittal Report," December 1993.