March 10, 2000

Mr. Michael B. Sellman
Senior Vice President and
Chief Nuclear Officer
Wisconsin Electric Power Company
231 West Michigan Street
Milwaukee, WI 53201

SUBJECT: POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2 - SITE-SPECIFIC WORKSHEETS FOR USE IN THE NUCLEAR REGULATORY COMMISSION'S SIGNIFICANCE DETERMINATION PROCESS (TAC NO. MA6544)

Dear Mr. Sellman:

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 Point Beach Nuclear Plant, Units 1 and 2 in April 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.

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 301-415-1355.

Sincerely,

/RA/

Beth A. Wetzel, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-266 and 50-301

Enclosures: As Stated

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We will coordinate our efforts through your licensing or risk organizations as appropriate. If you have any questions, please contact me at 301-415-1355.

Sincerely,

/RA/

Beth A. Wetzel, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

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RISK-INFORMED INSPECTION NOTEBOOK FOR

POINT BEACH NUCLEAR PLANT

UNITS 1 AND 2

PWR, WESTINGHOUSE, TWO-LOOP PLANT WITH LARGE DRY CONTAINMENT

Prepared by

Brookhaven National Laboratory
Department of Advanced Technology

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Prepared for

U. S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
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November 1999

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 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:

Mr. Jose G. Ibarra
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ABSTRACT

This notebook contains summary information to support the Significance Determination Process (SDP) in risk-informed inspections for the Point Beach Nuclear Plant

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 Point Beach Nuclear Plant.

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 Point Beach Units 1 & 2

Affected Systems	Major Components	Support Systems	Initiating Event
AC Power System	AC Power Distribution & AC Instrument Power	DC, HVAC	Transient, SLOCA, SORV, MLOCA, LLOCA, LOOP, SGTR, MSLB, ATWS, RCP seal LOCA
AFW	Two MDPs	480 V-AC, DC, SW[1], ESFAS, Fire Water[2]	Transient, SLOCA, SORV, LOOP, SGTR, MSLB, ATWS
	One TDP	DC, SW[1], ESFAS, FW	
CCW[3]	Four pumps and four Heat Exchangers; Two pumps for each unit , one dedicated and two common Heat Exchangers	480 V-AC, DC, ESFAS, SW	Transient, SLOCA, SORV, MLOCA, LLOCA, LOOP, SGTR, MSLB, ATWS, RCP seal LOCA
Condensate / MFW	Two Condensate pumps Two MFW pumps	4.16 kV AC, DC, SW, IA	Transient
Containment Air Recirculation System (CARS)	Four Air Fan Cooling Units	480 V-AC, DC, ESFAS, SW	Not used
Containment Spray System	Two CS pumps	480 V-AC, DC, ESFAS, CCW[4]	Not used
HPSI	Two SI pumps	480 AC, DC, IA[5]	Transient, SLOCA, SORV, MLOCA, LOOP, SGTR, MSLB, ATWS, RCP seal LOCA
CVCS/Charging	Three pumps	480 V-AC, DC, IA[5]	RCP Seal LOCA, ATWS
DC Power	Buses, battery chargers and batteries	Battery Chargers[6]	Transient, SLOCA, SORV, MLOCA, LLOCA, LOOP, SGTR, MSLB, ATWS, RCP seal LOCA

Table 1 (Continued)

Affected Systems	Major Components	Support Systems	Initiating Event
EDG	Two EDGs shared by both units	DC, SW[2], Fuel Oil	LOOP
ESFAS		120 V-AC,DC	Transient, SLOCA, SORV, MLOCA, LLOCA, LOOP, SGTR, MSLB, ATWS, RCP seal LOCA
Instrument Air	Four Air Compressors	480 V-AC, DC, SW	Transient, SLOCA, SORV, LOOP, SGTR, ATWS
Main Steam	Four Code safety valves for each SG and two ADVs, one per SG	DC, IA	MSLB, SGTR
Pressurizer Pressure Relief	Three Safety valves and two PORVs with associated block valves	120 V-AC, DC, IA (PORVs)	Transient, SLOCA, SORV, LOOP, SGTR, MSLB, ATWS
RCP	Seals	1 / 3 Charging pumps to seal injection or 1 / 2 CCW pumps to thermal barrier heat exchanger	LOOP, RCP seal LOCA
RHR/LHSI	Two RHR/LPSI pumps and heat exchanger	4.16 kV AC, DC, ESFAS, CCW,	Transient, SLOCA, SORV, MLOCA, LLOCA, LOOP, SGTR, MSLB
SW	Six pumps supplying both units	480 V-AC, DC, ESFAS	Transient, SLOCA, SORV, MLOCA, LLOCA, LOOP, SGTR, MSLB, ATWS, RCP seal LOCA

Notes:

(1) SW is required after 4 hours at which time it is needed for bearing cooling and AFW pump suction water supply.

(2) Fire Water is an alternate supply of water for EDG cooling and for CST refill. Hose can be used to supply cooling to DGs. Also, CSTs can be refilled after they have been depleted (4 hours) using a hose.

Table 1 (Continued)

(3) CCW systems are unit specific but can be manually cross-connected.

(4) Required for ECCS recirculation only for pump seal coolers and RHR heat exchangers.

(5) Air required for pump speed control on feed and bleed cooling. Air not required for RCP seal cooling.

(6) Battery Charger required after one hour on the station batteries.

(7) Plant internal event CDF is 1.15E-04/yr.

1.2 SDP WORKSHEETS

This section presents the SDP worksheets to be used in the Phase 2 evaluation of the inspection findings for the Point Beach Nuclear Plant. The SDP worksheets are presented for the following initiating event categories:

1. Transients (Reactor trip)
2. Transients without PCS
3. Small LOCA
4. Stuck-open PORV
5. Medium LOCA
6. Large LOCA
7. LOOP
8. Steam Generator Tube Rupture (SGTR)
9. Anticipated Transients Without Scram (ATWS)
10. Main Steam Line Break (MSLB)

Table 2.1 SDP Worksheet for Point Beach Nuclear Plant, Units 1 and 2 —— Transients (Reactor trip)

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 / 2 Main Feedwater trains with 1 / 2 condensate trains (operator action)
Secondary Heat Removal (AFW)	1 / 2 MDAFW trains (1 multi-train system) or 1 TDAFW train (1 ASD train)
Early Inventory, High Pressure Injection (EIHP)	1 / 2 HPSI pumps (1 multi-train system)
Primary Heat Removal, Feed/Bleed (FB)	1 / 2 PORVs and block valves open for Feed/Bleed (operator action)(1)
High Pressure Recirculation (HPR)	1 / 2 HPSI pumps with 1 / 2 RHR pumps (Requires operator action for switchover = operator action)

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

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 human error probability (HEP) assessed in the IPE for establishing bleed and feed is approximately 2.0E-2.

Table 2.2 SDP Worksheet for Point Beach Nuclear Plant, Units 1 and 2 —— Transients w/o PCS

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:**
Secondary Heat Removal (AFW)	1 / 2 MDAFW trains (1 multi-train system) or 1 TDAFW train (1 ASD train)
Early Inventory, High Pressure Injection (EIHP)	1 / 2 HPSI pumps (1 multi-train system)
Primary Heat Removal, Feed/Bleed (FB)	1 / 2 PORVs and block valves open for Feed/Bleed (operator action)(1)
High Pressure Recirculation (HPR)	1 / 2 HPSI pumps with 1 / 2 RHR pumps (Requires operator action for switchover = operator action)

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

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 human error probability (HEP) assessed in the IPE for establishing bleed and feed is approximately 2.0E-2.

Table 2.3 SDP Worksheet for Point Beach Nuclear Plant, Units 1 and 2 —— Small LOCA

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 / 2 HPSI pumps (1 multi-train system).
Secondary Heat Removal (AFW)	1 / 2 MDAFW trains (1 multi-train system) or 1 TDAFW train (1 ASD train)
RCS Cooldown / Depressurization (RCSDEP)	Operator depressurizes RCS using pressurizer spray or 1 / 2 PORVs and atmospheric steam dump valves (operator action)
Primary Bleed (FB)	1 / 2 PORVs and block valves open for Feed/Bleed (operator action)(1)
Accumulators (ACC)	1/ 1 Accumulators (1 Train)
Low Pressure Injection (LPI)	1 / 2 RHR pumps (1 multi-train system)
Low Pressure Recirculation (LPR)	1/ 2 RHR pumps taling suction from sump (operator action)
High Pressure Recirculation (HPR)	1 / 2 HPSI pumps with 1 / 2 RHR pumps (Requires operator action for switchover = operator action)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 SLOCA - RCSDEP(2) - HPR (3)			
2 SLOCA - AFW - HPR (5)			
3 SLOCA - AFW - FB (6)			
4 SLOCA - EIHP - LPR (8)			

5 SLOCA - EIHP - LPI (9)			
6 SLOCA - EIHP - ACC (10)			
7 SLOCA - EIHP - RCSDEP[2] (11)			
8 SLOCA - EIHP - AFW (12)			

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 human error probability (HEP) assessed in the IPE for establishing bleed and feed cooling is approximately 2.0E-2.

(2) In sequence 1, the accident is under control and the RCSDEP involves cool down and depressurization to establish RHR entry conditions. In Sequence 7, RCPDEP is accomplished through secondary depressurization by dumping steam to the condenser if available or by dumping steam through the atmospheric dump valves.

Table 2.4 SDP Worksheet for Point Beach Nuclear Plant, Units 1 and 2 —— Stuck Open PORV (SORV)

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 / 2 HPSI pumps (1 multi-train system).
Isolation of Small LOCA (BLK)	The closure of the block valve associated with stuck open PORV (recovery action)
Secondary Heat Removal (AFW)	1 / 2 MDAFW trains (1 multi-train system) or 1 TDAFW train (1 ASD train)
RCS Cooldown / Depressurization (RCSDEP)	Operator depressurizes RCS using pressurizer sprays and 1 / 2 PORVs and block valves or atmospheric dump valves (operator action)
Primary Bleed (FB)	Operator action through stuck-open PORV (operator action under high stress)(1)
Accumulators (ACC)	1/ 1 Accumulators (1 Train)
Low Pressure Injection (LPI)	1 / 2 RHR pumps (1 multi-train system)
High Pressure Recirculation (HPR)	1 / 2 HPSI pumps with 1 / 2 RHR pumps (Requires operator action for switchover = operator action)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 SORV - BLK - RCSDEP(2) - LPR (3)			
2 SORV - BLK - AFW - HPR (5)			
3 SORV - BLK - AFW - FB (6)			
4 SORV - BLK - EIHP - LPR (8)			

5 SORV - BLK - EIHP - LPI (9)			
6 SORV - BLK - EIHP - ACC (10)			
7 SORV - BLK - EIHP - RCSDEP(2) (11)			
8 SORV - BLK - EIHP - AFW (12)			

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 human error probability (HEP) assessed in the IPE for establishing bleed and feed cooling is approximately 2.0E-2.

(2) In sequence 1, the accident is under control and the RCSDEP involves cooldown and depressurization to establish RHR entry conditions. In Sequence 7, RCPDEP is accomplished through secondary depressurization by dumping steam to the condenser if available or by dumping steam through the atmospheric dump valves.

Table 2.5 SDP Worksheet for Point Beach Nuclear Plant, Units 1 and 2 —— Medium LOCA

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/ 2 HPSI pumps (1 multi-train system).
Auxiliary Feedwater (AFW)	1/ 2 MDAFW pumps (1 multi-train system) or 1/1 TDAFW pumps (1 ASD train)
RCS Depressurization (DEP)	Operator depressurizes using 1/ 2 atmospheric dump valves (Operator action)
Accumulator (ACC)	1/1 ACC injection to 1intact loop (1 train)
Low Pressure Injection (LPI)	1/ 2 RHR pumps (1 multi-train system)
High Pressure Recirculation (HPR)	1/ 2 HPSI pumps taking suction from 1/ 2 RHR pumps (Operator action)
Low Pressure Recirculation (LPR)	1 / 2 RHR pump trains with operator switchover from injection to recirculation (operator action)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 MLOCA - HPR (2)			
2 MLOCA - ACC (3,7)			
3 MLOCA - EIHP - LPR (5)			
4 MLOCA - EIHP - LPI (6)			
5 MLOCA - EIHP - DEP (8)			

6 MLOCA - EIHP - AFW (9)			
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.			

Table 2.6 SDP Worksheet for Point Beach Nuclear Plant, Units 1 and 2 —— Large LOCA

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, Accumulators (EIAC)	1/1 Accumulator to the intact loop (1 Train)
Early Inventory, LP Injection (EILP)	1/2 RHR pump trains (1 multi-train system)
Low Pressure Recirculation (LPR)	1/2 RHR trains; Operator switchover from injection to recirculation (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 LLOCA - LPR (2)			
2 LLOCA - EILP (3)			
3 LLOCA - EIAC (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.

Table 2.7 SDP Worksheet for Point Beach Nuclear Plant, Units 1 and 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 Emergency Diesel Generators (1 multi-train system)or a Gas Turbine (1 diverse system)
Turbine-driven AFW Pump (TDAFW)	1 / 1 TDP trains of AFW (1 train)(1)
Secondary Heat Removal (AFW)	1 / 2 MDAFW trains (1 multi-train system) or 1 TDAFW train (1 ASD train)
Recovery of AC Power in < 2 hrs (REC2)	SBO procedures implemented (operator action under high stress)(1)
Recovery of AC Power in < 7 hrs (REC7)	SBO procedures implemented (operator action)(2)
Early Inventory, HP Injection (EIHP)	1 / 2 HPSI pumps (1 multi-train system)
Primary Heat Removal (FB)	Operator uses RCS pressurizer 1 / 2 PORVs and block valves (operator action)
High Pressure Recirculation (HPR)	1 / 2 HPSI pumps with 1/ 2 RHR pumps and with operator action for switchover (operator action)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 LOOP - AFW - HPR (3)			
2 LOOP - AFW - FB (4)			
3 LOOP - AFW - EIHP (5)			
4 LOOP - EAC - HPR (7, 11) (AC recovered)			

5 LOOP - EAC - EIHP (8, 13) (AC recovered)			
6 LOOP - EAC - REC7 (9)			
7 LOOP - EAC - TDAFW - FB (12) (AC recovered)			
8 LOOP - EAC - TDAFW - REC2 (14)			

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) For the functions “Recovery of AC Power in < 2 hrs (REC2)” and “Recovery of AC Power in < 5 hrs (REC5)” the corresponding estimated probabilities in the IPE are 2.8E-1 and 4.0E-2.

(2) In an SBO situation, an RCP seal LOCA may occur, with subsequent core damage at about 7 hours.

Table 2.8 SDP Worksheet for Point Beach Nuclear Plant, Units 1 and 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:
Secondary Heat Removal (AFW)	1 / 2 MDPs of AFW (1 multi-train system) or 1 / 1 TDP of AFW (1 ASD Train)
Early Inventory, HP Injection (EIHP)	1 / 2 HPSI pumps (1 multi-train system)
Main Feedwater (MFW)	Operator establishes feedwater from the condensate pump(1) (Operator action)
SG Isolation (SGI)	Operator isolates the ruptured SG (Operator action)(2)
Pressure Equalization (EQ)	Operator depressurizes RCS using 1 / 1 SG ARV (on each SG fed by AFW) or RCS pressurizer PORV (1 / 2) to less than setpoint of relief valves of SG (operator action under high stress)(3)
Decay Heat Removal (DHR)	Cool down and depressurize primary and align 1/ 2 RHR pumps (Operator action)

Circle Affected Functions	Recovery of Failed Train	*Remaining Mitigation Capability Rating for Each Affected Sequence*	Sequence Color
1 SGTR - EQ - DHR (3,8,13)			
2 SGTR - SGI - DHR (5,10)			
3 SGTR - AFW - SGI (14)			
4 SGTR - AFW - EIHP (15)			
5 SGTR - AFW - MFW (16)			

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) Point Beach SGTR analysis credits the recovery of main feedwater if auxiliary feedwater fails, but does not credit the use of feed and bleed if all feedwater fail.

(2) Failure to identify and isolate a ruptured SG is assigned an error probability of 4.5E-3. Failure to isolate ruptured SG and stop TDAFW flow is assigned an error probability of 8.5E-03 in the IPE.

(3) Failure to cooldown and depressurize for SGTR is assigned a failure probability of 7.7E-03.

Table 2.9 SDP Worksheet for Point Beach Nuclear Plant, Units 1 and 2 —— MSLB[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, HP Injection (EIHP)	1 / 2 HPSI pumps (1 multi-train system)
Secondary Heat Removal (AFW)	1 / 2 MDPs of AFW (1 multi-train system) or 1 / 1 TDP of AFW (1 ASD Train)
Main Steam Isolation (ISOL)	Automatic signal for MSIV closure and operator verification (1 train)
Feed and Bleed (FB)	Operator action with 1/ 2 PORV (Operator action)
High Pressure Recirculation (HPR)	1/ 2 HPSI pumps taking suction from RHR pumps (Operator action)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 MSLB -AFW - HPR (3)			
2 MSLB - AFW - FB (4)			
3 MSLB - ISOL - HPR (6)			
4 MSLB - ISOL - FB (7)			
5 MSLB - EIHP - AFW (9)			
6 MSLB - EIHP - ISOL (10)			

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) PBNS models intermediate size isolable steamline break.

Table 2.10 SDP Worksheet for Point Beach Nuclear Plant, Units 1 and 2 —— ATWS(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:**		
Emergency Boration (HPI)	Operator conducts emergency boration using 1 / 2 charging pumps with 1 / 2 boric acid transfer pumps (operator action)		
AMSAC (AMSAC)	AMSAC trips the turbine (1 multi-train system)		
Primary Relief (PR)	3 / 3 SRVs with 2 / 2 PORVs open (1 train)		
Secondary Heat Removal (AFW)	2 / 2 MDPs of AFW (1 multi-train system) or 1 / 1 TDP of AFW (1 ASD Train)		
Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 ATWS - PR (3)			
2 ATWS - AFW (4)			
3 ATWS - HPI (2)			
4 ATWS - AMSAC (5)			

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:

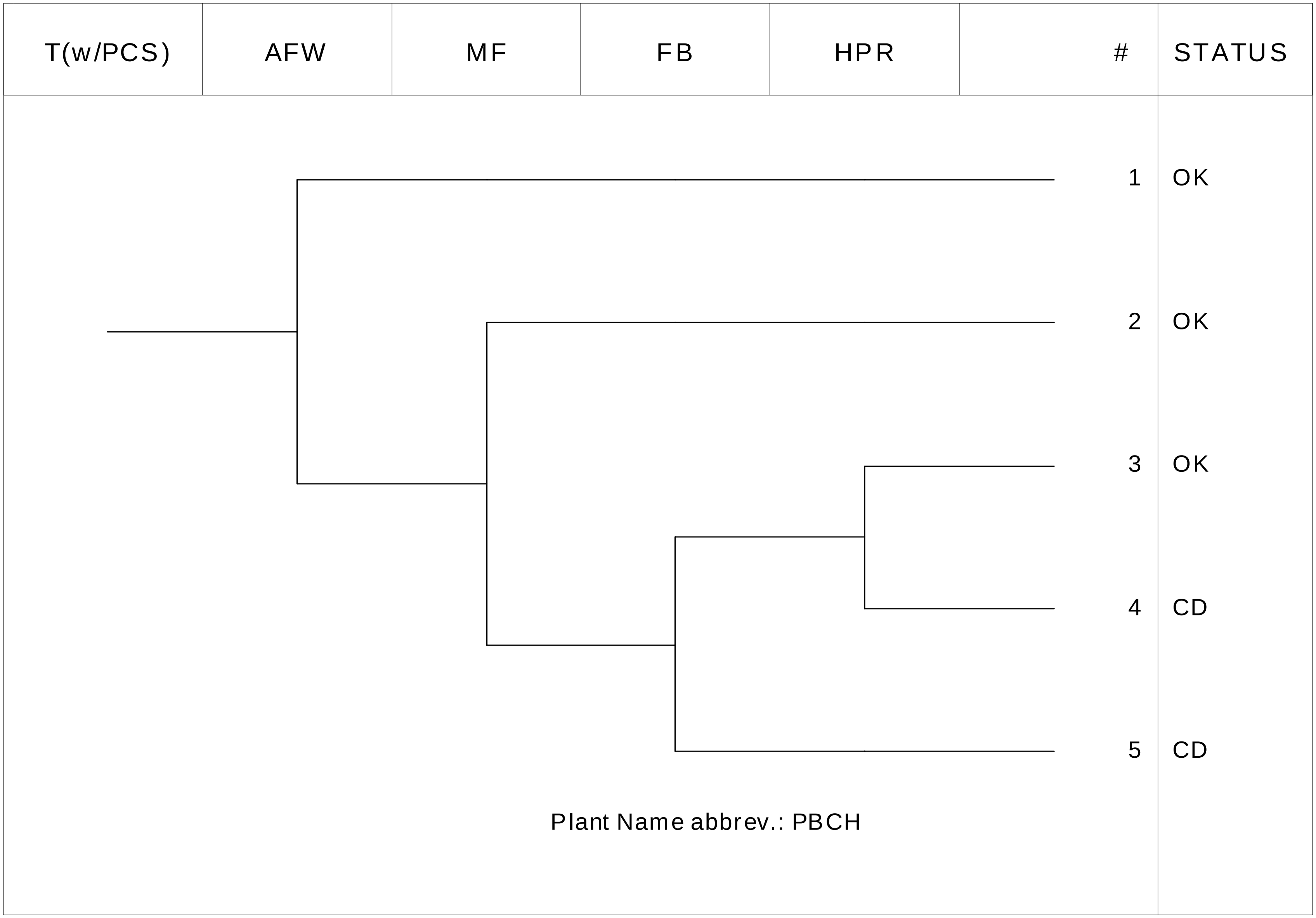
(1) PBNS models the most severe ATWS events where the power level is greater than 40% with main feed water unavailable.

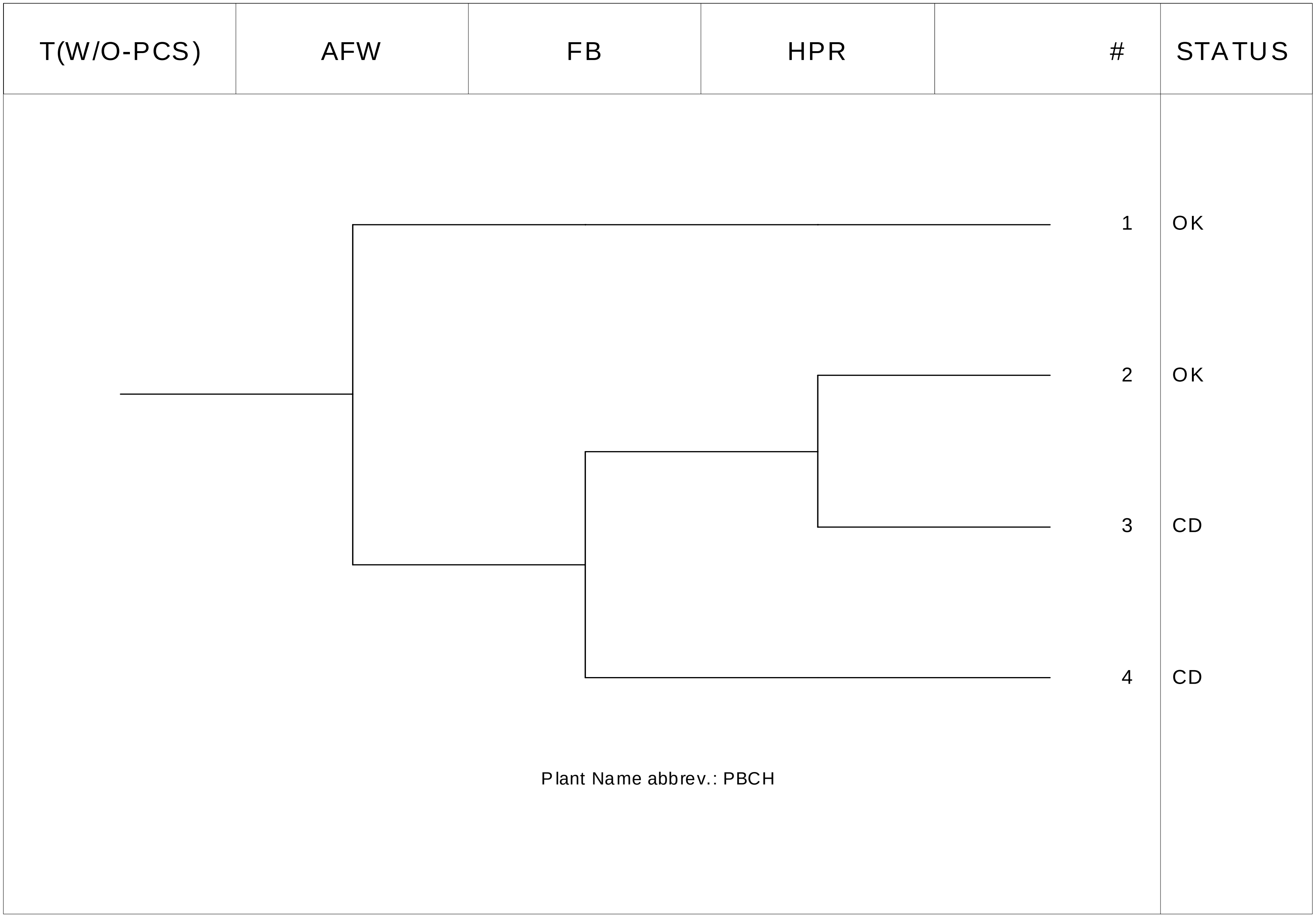
1.3 SDP Event Trees

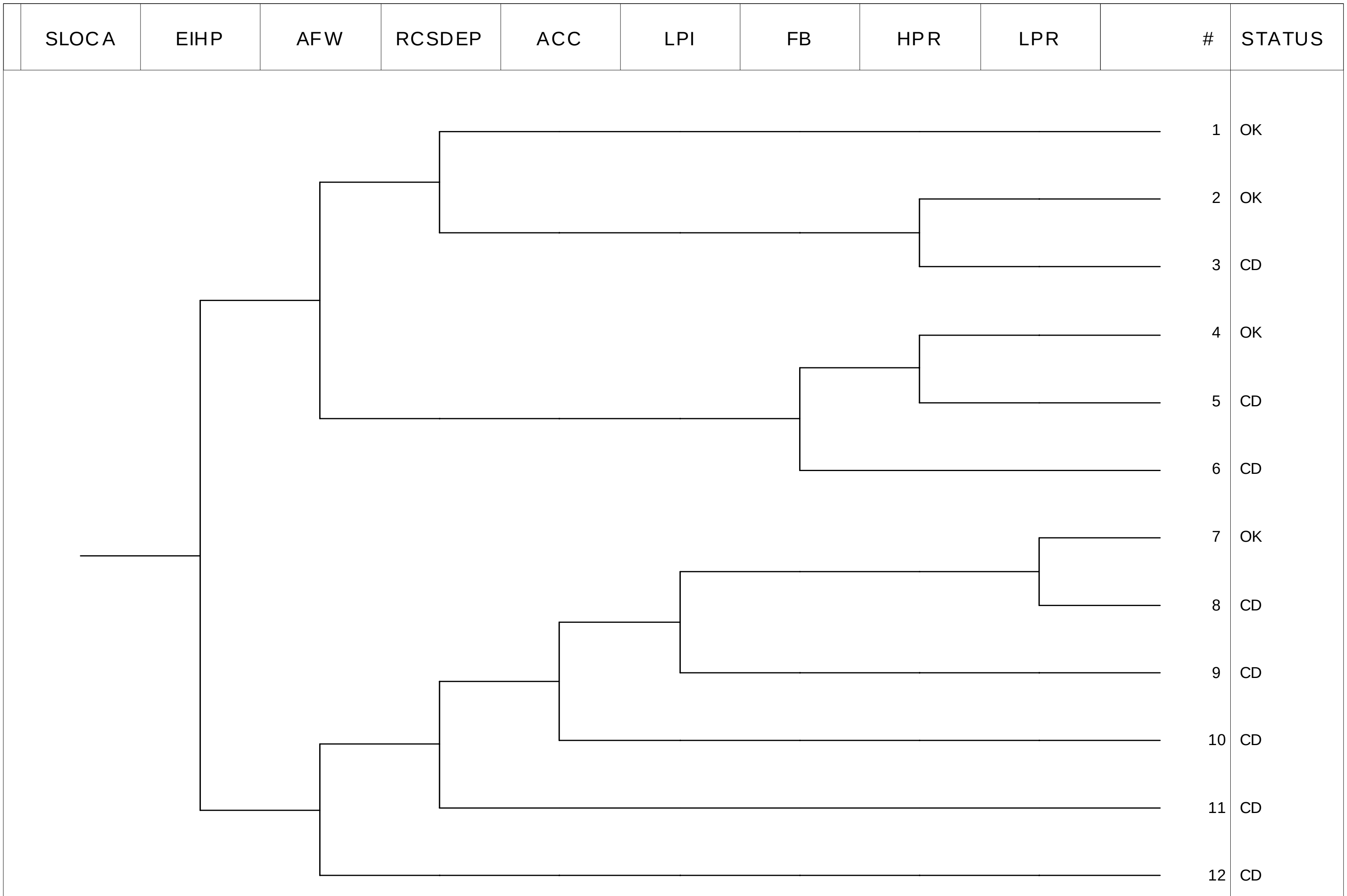
This section provides the simplified event trees called SDP event trees used to define the accident sequences identified in the SDP worksheets in the previous section. An event tree for the stuck-open PORV is not included since it is similar to the small LOCA event tree. The event tree headings are defined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

The following event trees are included:

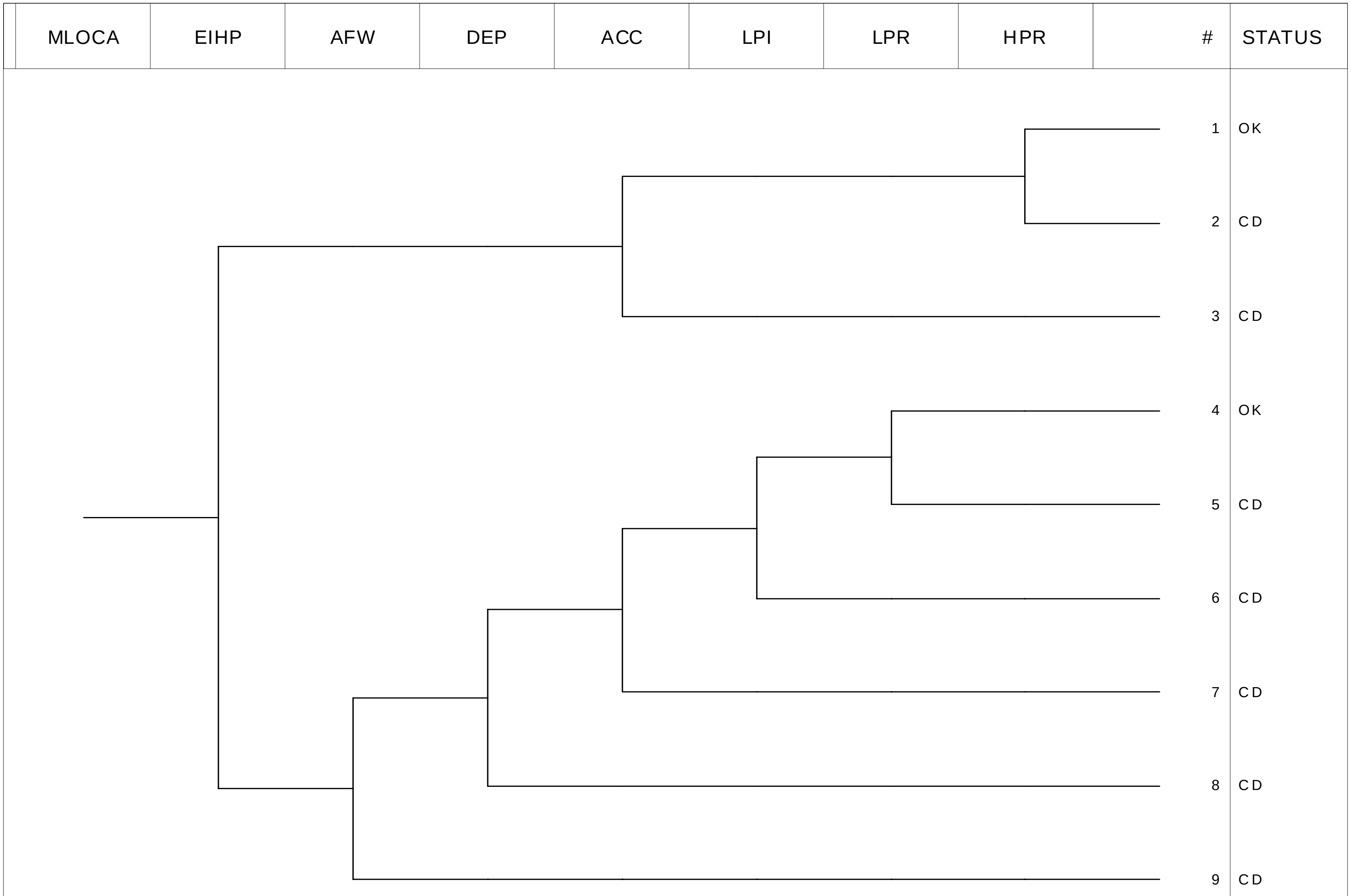
1. Transients w/ PCS
2. Transients w/o PCS
3. Small LOCA
4. Medium LOCA
5. Large LOCA
6. LOOP
7. Steam Generator Tube Rupture (SGTR)
8. Anticipated Transients Without Scram (ATWS)
9. Main Line Steam Break (MSLB)


T(w/PCS)
AFW
MF
FB
HPR
#
STATUS
1 OK
2 OK
3 OK
4 CD
5 CD
Plant Name abbrev.: PBCH


T(W/O-PCS)
AFW
FB
HPR
#
STATUS
1
OK
2
OK
3
CD
4
CD
Plant Name abbrev.: PBCH

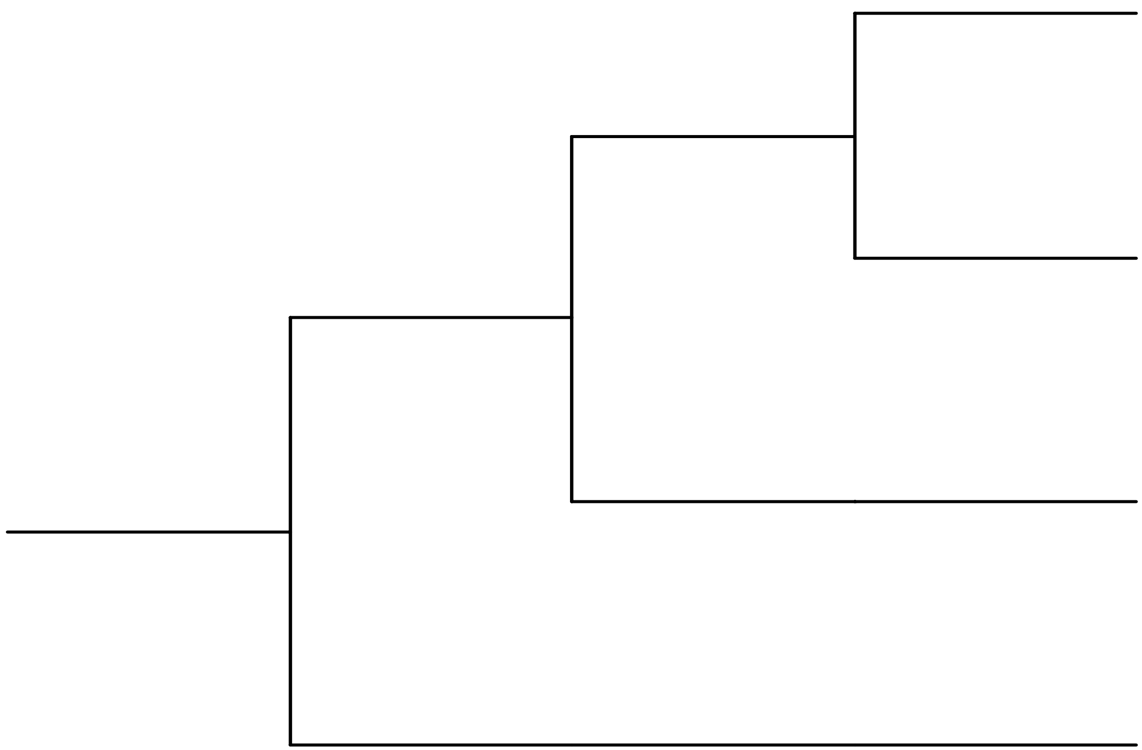


Plant Name abbrev.: PBCH

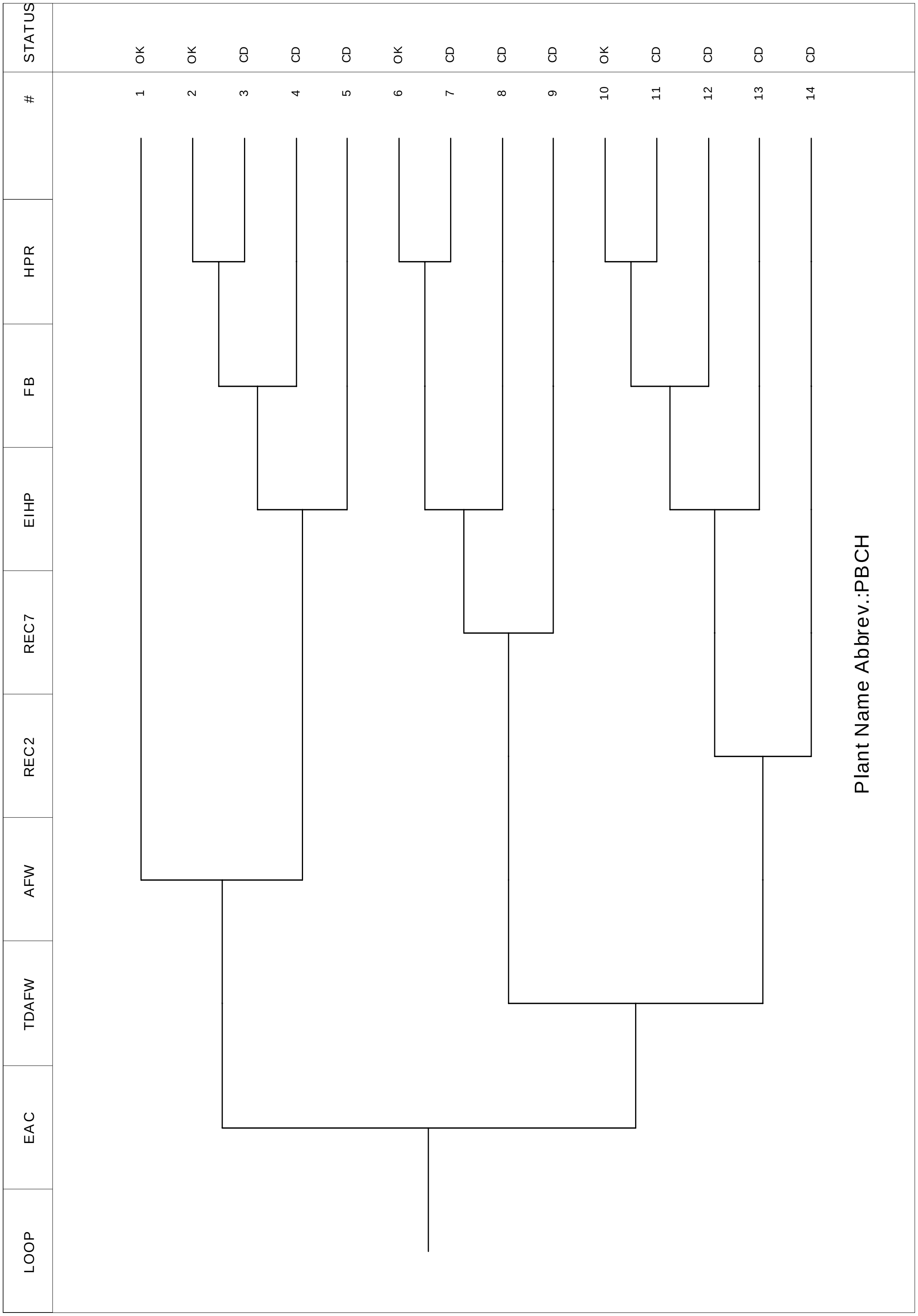


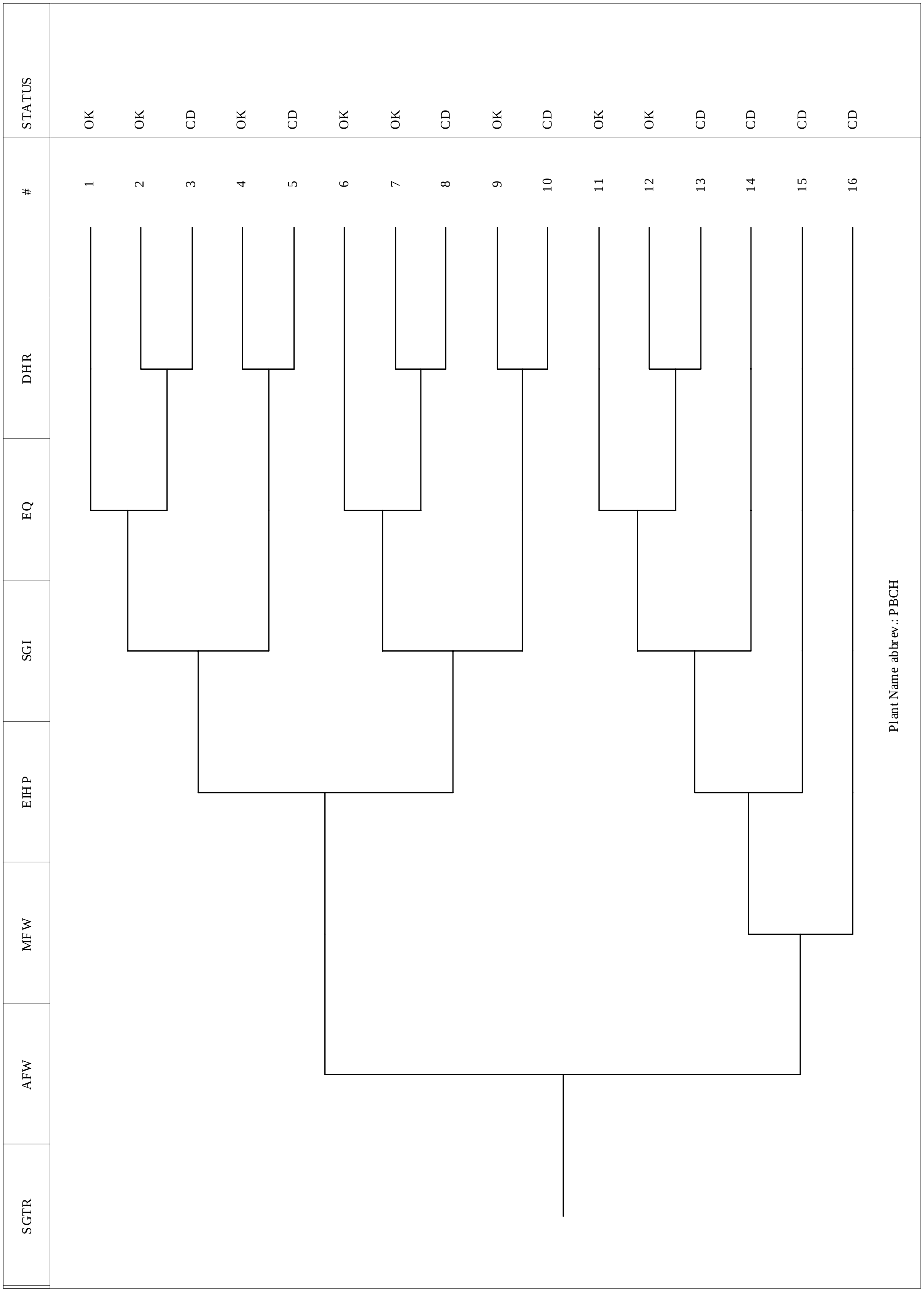
Plant Name abbrev.: PBCH

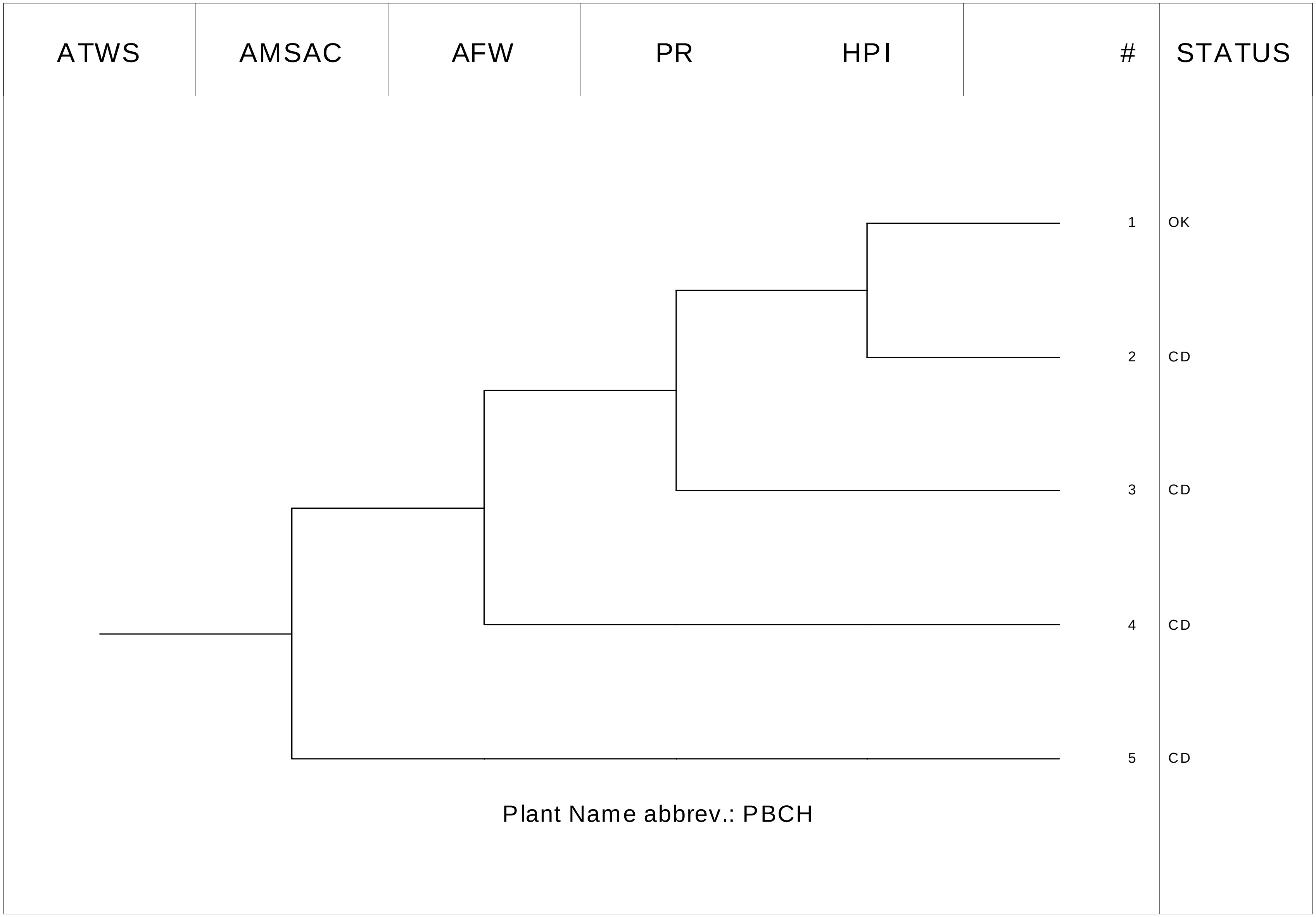
LLOCA	EIAC	EILP	LPR	#	STATUS
				1	OK
				2	CD
				3	CD
				4	CD

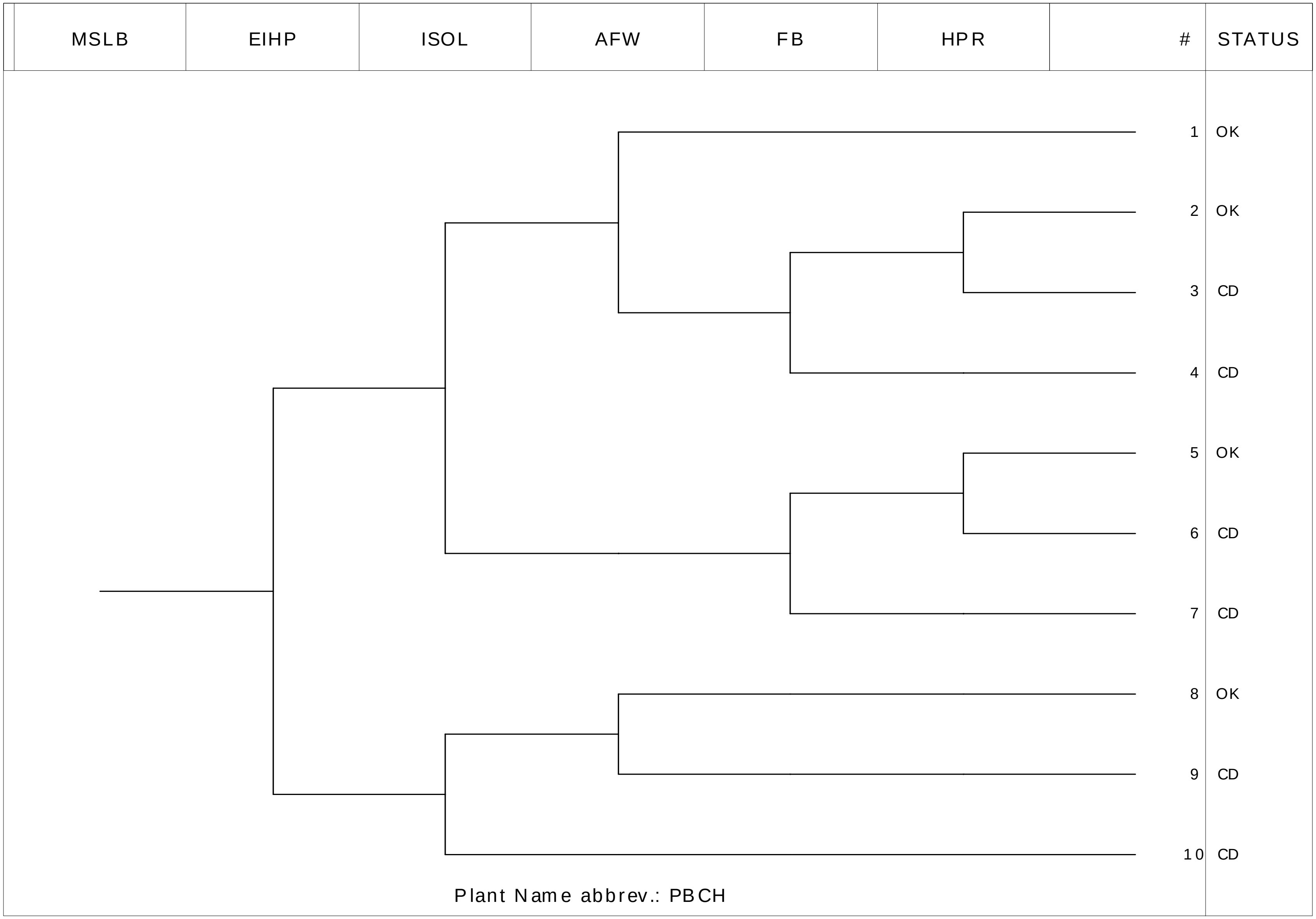


Plant Name abbrev.: PBCH


STATUS
#
HPR
FB
EIHP
REC7
REC2
AFW
TDAFW
EAC
LOOP
1 OK
2 OK
3 CD
4 CD
5 CD
6 OK
7 CD
8 CD
9 CD
10 OK
11 CD
12 CD
13 CD
14 CD
Plant Name Abbrev.:PBCH


SGTR
AFW
MFW
EIHP
SGI
EQ
DHR
#
STATUS
1
2
3
4
5
6
7
8
9
10
11
12
13
14
15
16
OK
OK
CD
OK
CD
OK
OK
CD
OK
CD
OK
OK
CD
CD
CD
CD
Plant Name abbrev.: PBCH


ATWS
AMSAC
AFW
PR
HPI
#
STATUS
1 OK
2 CD
3 CD
4 CD
5 CD
Plant Name abbrev.: PBCH


MSLB
EIHP
ISOL
AFW
FB
HPR
#
STATUS
1 OK
2 OK
3 CD
4 CD
5 OK
6 CD
7 CD
8 OK
9 CD
10 CD
Plant Name abbrev.: PBCH

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. Wisconsin Electric Power Company, "Point Beach Nuclear Plant, Units 1 and 2 — Individual Plant Examination Report," June 30, 1993.