March 8, 2000

Mr. Theodore A. Sullivan
Vice President Nuclear and Station Director
Entergy Nuclear Generation Company
Pilgrim Nuclear Power Station
600 Rocky Hill Road
Plymouth, MA 02360

SUBJECT: SITE-SPECIFIC WORKSHEETS FOR USE IN THE NUCLEAR REGULATORY COMMISSION'S SIGNIFICANCE DETERMINATION PROCESS (TAC NO. MA6544)

Dear Mr. Sullivan:

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 the Pilgrim Nuclear Power Station (Pilgrim) 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 before April 2000 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. In addition, 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 301-415-1445.

Sincerely,

/RA/

Alan B. Wang, Project Manager, Section 2
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-293

Enclosure: As Stated

cc w/encl: See next page

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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 301-415-1445.

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RISK-INFORMED INSPECTION NOTEBOOK FOR

PILGRIM NUCLEAR POWER STATION

UNIT 1

BWR-3, GE, WITH MARK I CONTAINMENT

Prepared by

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Prepared for

U. S. Nuclear Regulatory Commission
Office of Nuclear Regulatory Research
Division of Risk Analysis & Applications

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:

Mr. Jose G. Ibarra
U. S. Nuclear Regulatory Commission
RES/DSARE/REAHFB
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Rockville, MD 20852

ABSTRACT

This notebook contains summary information to support the Significance Determination Process (SDP) in risk-informed inspections for the Pilgrim Nuclear Power Station.

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

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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). 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) Medium LOCA, 4) Large LOCA, 5) Loss of Offsite Power (LOOP), and 6) 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. 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 Pilgrim Nuclear Power Station.

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.

2.1 Initiators and System Dependency Table for Pilgrim Nuclear Power Station, Unit 1

Affected System	Major Components	Support Systems	Initiating Event Scenarios
Safety Relief Valves (SRVs) / Automatic Depressurization System (ADS)	Relief valves, Air-operated valves	125 V-DC power, Instrument N_2	All but LLOCA
Power Conversion System (PCS)			All
Main Steam System	4 MSIVs, 4 Main Steam Lines, Turbine Bypass Valves		
Feedwater System	3 MD pumps, MOVs, 2 FW Reg. Valves	Offsite Power, 4160 V-AC (BOP), 120 V-AC (BOP), 125 VDC (BOP), TBCCW, Instrument Air	
Condensate System	3 MD pumps, Heat Exchangers, MOVs, Main Condenser	Offsite Power, 4160 V-AC (BOP), 480 V-AC (BOP), 120 V-AC (BOP), 125 VDC (BOP), TBCCW, Instrument Air, HVAC	
High Pressure Coolant Injection (HPCI)	1 MD pump, MOVs	B18 480 V-AC (Area Coolers), 125 V-DC, ECCS Start Logic, RBCCW-B, HVAC	All
Reactor Core Isolation Cooling (RCIC)	1ASD pump, MOVs	125 VDC, ECCS Start Logic	TRANS, SLOCA, LOOP
Control Rod Drive (CRD)	2 MD pumps, MOVs	4160 V-AC (BOP), 125 VDC, RBCCW Train B	Not Credited

Table 1 (Continued)

Affected System	Major Components	Support Systems	Initiating Event Scenarios
Core Standby Coolant Systems (CSCS): Residual Heat Removal (RHR) / Low Pressure Coolant Injection (LPCI)	2 Loops (2 MD RHR pumps and 1 HX per loop), MOVs	4160 V-AC, 480 V-AC, 125 V-DC, ECCS Start Logic, RBCCW, HVAC	All
Core Standby Coolant Systems (CSCS): Low Pressure Core Spray (LPCS)	2 MD pumps, MOVs	4160 V-AC, 480 V-AC, 125 V-DC, ECCS Start Logic, RBCCW, HVAC	All
Fire Protection System - RHR/Fire Water X-Tie	1 DD fire pump, Spool piece Manual valves, MOVs	4160 V-AC, 480 V-AC, Instrument Air, HVAC	LOOP
Fire Protection System - Containment Spray Mode	1 DD or MD fire pump, spool piece, Manual valves, MOVs	4160 V-AC, 480 V-AC, Instrument Air, HVAC	All
Recirculation Pump Trip (RPT)	Logic circuits, transmitters	480 V-AC, 125 V-DC	ATWS
Standby Liquid Control (SLC)	2 MD pumps, Explosive valves	480 V-AC, 125 V-DC	ATWS

Table 1 (Continued)

Affected System	Major Components	Support Systems	Initiating Event Scenarios
Containment Ventilation (CV)	2 DC MOVs rupture disc 8 in. line	480 V-AC, 125 V-DC, Nitrogen, Instrument Air, HVAC	All
RHR / Containment Spray (CS)	2 loops (2 MD RHR pumps, 1 HX per loop) MOVs	4160 V-AC, 480 V-AC, 125 V-DC, ECCS Start Logic, RBCCW, HVAC	All
RHR / Suppression Pool Cooling (SPC)	2 loops (2 MD RHR pumps, 1 HX per loop) MOVs	4160 V-AC, 480 V-AC, 125 VDC, ECCS Start Logic, RBCCW, HVAC	All
Electric Power System Onsite AC Power	Startup (SU) Xfmr 4160 V-AC buses, 480 V-AC buses 120/240 V-AC buses	Unit Aux. Xfmr, SU Xfmr, EDG A and B, 23 KV shutdown Xfmr, SBO DG, 24 VDC batteries, 24 VDC battery chargers	All
Electric Power System DC Power	250 VDC buses, 125 VDC buses	250 VDC batteries, 250 VDC battery chargers, 125 VDC batteries, 125 VDC battery chargers, HVAC	All
ECCS Start Logic	Logic circuits, Transmitters	120 V-AC, 125 VDC, 250 VDC	All
Salt Service Water (SSW)	2 loops (2 MD pumps per loop plus 1 MD swing pump) MOVs	480 V-AC, 120 V-AC	All

Table 1 (Continued)

Affected System	Major Components	Support Systems	Initiating Event Scenarios
Reactor Building Closed Cooling Water (RBCCW)	2 loops (3 MD pumps and 1 HX per loop) MOVs	480 V-AC, 120 V-AC, SSW	All
Turbine Building Closed Cooling Water (TBCCW)	1 loop (2 MD pumps, 2 Hxs) MOVs	480 V-AC, 120 V-AC, 125 VDC, SSW	TRANS, ATWS
Instrument Air (IA)/Nitrogen	3 Recip, 2 Screw MD compressors, dryers Purge Supply (N2) from PCAC system	480 V-AC, 120 V-AC, TBCCW	All
HVAC	MD fans, MO dampers, ducts	4160 V-AC, RBCCW, TBCCW	All

Notes:

1. Transient scenarios should be developed from those transient initiators that could have the greatest risk significance. For example, develop loss of DC bus transient scenarios for degraded 125 V-DC or AC power equipment, as well as, other transient initiators that may depend on equipment being supplied from degraded power sources. The choice of which transient scenarios to develop should generally be apparent from the specific given condition.

2. Information herein was developed from the Pilgrim Nuclear Generating Station IPE Response to Generic Letter 88-20, submitted to the NRC by letter dated September 30, 1992.

3. The Pilgrim Station electrical power system is not arranged in the conventional BWR division method.

Table 1 (Continued)

Table 1 (Continued)

4. The containment spray mode of the fire protection system is a manual alignment, requiring startup of one of the two fire protection pumps and manual alignment of flow to the wetwell airspace or drywell spray headers, further requiring installation of a spool piece and opening of some manual and motor-operated valves. For these reasons, the RHR in the containment spray mode is the preferred method of containment spray. (See PILG IPE page B.6-8).

5. Review of the results of the IPE analysis by Boston Edison Co. showed that core damage occurs prior to containment failure, and that the IPE analysis cannot take credit for successful core cooling after containment failure. The PILG IPE model was subsequently revised to show that core damage occurs prior to containment failure and the IPE was requantified to reflect this change in philosophy. (See PILG Response to RAI, December 28, 1995, Question 7d, page 17). Therefore, for the purposes of this SDP document, the Late Inventory (LI) functions have been removed from the event trees.

1.2 SDP WORKSHEETS

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

1. Transients
2. Small LOCA
3. Medium LOCA
4. Large LOCA
5. LOOP
6. Anticipated Transients Without Scram (ATWS)

Table 2.1 SDP Worksheet for Pilgrim Nuclear Power Station, Unit 1 —— Transients

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)	1/4 Steam lines, Condenser, 1 steam jet air ejector, 1 circulating water pump, 1/3 condensate pumps, 1/3 main feed pumps (Operator Action)
High Pressure Injection (HPI)	HPCI (1 ASD train) or RCIC (1 ASD train)
Depressurization (DEP)	3/4 safety relief valves (4 ADS SRVs and 2 safety valves) manually opened (high stress operator action)[1]
Low Pressure Injection (LPI)	1/4 RHR trains in LPCI Mode (1 multi-train system) or 1/2 LPCS trains (1 multi-train system) or 1/3 condensate pumps (operator action)
Containment Heat Removal (CHR)	PCS (operator action) or [1/4 RHR pumps in Shutdown Cooling (SDC) mode if Rx pressure <100 psig or in Suppression Pool Cooling (SPC) mode or in Containment Spray (CS) mode] (operator action) or 1/ 2 Fire pumps in Containment Spray mode (operator action)[2]
Containment Venting (CV)	CV through 1/1 8" bypass around Standby Gas Treatment System - (high stress operator action)
Late Inventory, Makeup (LI)	1/4 RHR trains in LPCI Mode (1 multi-train system) or 1/2 LPCS trains (1 multi-train system) or 1/3 condensate pumps (operator action)

Circle Affected Functions	**Recovery or Failed Train**	***Remaining* Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 TRANS - PCS - CHR - CV (4, 6)			
2 TRANS - PCS - HPI - LPI (8)			
3 TRANS - PCS - HPI - DEP (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.

Notes:

1. For depressurization, it is assumed that the operator will always inhibit automatic actuation of ADS, so that depressurization is a manual operator action. (See PILG IPE, page B.4-3).

2. The containment spray mode of the fire protection system is a manual alignment, requiring startup of one of the two fire protection pumps and manual alignment of flow to the wetwell airspace or drywell spray headers, further requiring installation of a spool piece and opening of some manual and motor-operated valves. For these reasons, the RHR in the containment spray mode is the preferred method of containment spray. (See PILG IPE page B.6-8).

3. Although credited in the original Pilgrim IPE of September 30, 1992, based on the revised IPE assumptions regarding core cooling following containment failure, Late Inventory Makeup (LI) no longer applies.

Table 2.2 SDP Worksheet for Pilgrim Nuclear Power Station, Unit 1 —— Small 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>**
Power Conversion System (PCS)	***N/A See NOTE 1*** *1/4 steam lines, Condenser, 1 steam jet air ejector, 1 circulating water pump, 1/3 condensate pumps, 1/3 main feed pumps (operator action)*[1]
Early Containment Control (EC)	Vapor suppression system - passive operation of suppression pool - 12/12 vacuum breakers remain closed (1 multi-train system)
High Pressure Injection (HPI)	HPCI (1 ASD train) or RCIC (1 ASD train)
Depressurization (DEP)	3/4 safety relief valves (4 ADS SRVs and 2 safety valves) manually opened (high stress operator action)[2]
Low Pressure Injection (LPI)	1/4 RHR trains in LPCI mode (1 multi-train system) or 1/2 LPCS trains (1 multi-train system) or 1/1 Fire Water Cross-tie (high stress operator action)[1]
Containment Heat Removal (CHR)	[1/4 RHR pumps in Shutdown Cooling (SDC) mode if Rx pressure <100 psig or in Suppression Pool Cooling (SPC) mode or in Containment Spray (CS) mode] (operator action) or 1/ 2 Fire pumps in Containment Spray mode (operator action)
Containment Venting (CV)	CV through 1/1 8" bypass around Standby Gas Treatment System - (high stress operator action)

<u>Circle Affected Functions</u>	**<u>Recovery or Failed Train</u>**	***<u>Remaining</u>*** **<u>Mitigation Capability Rating for Each Affected Sequence</u>**	**<u>Sequence Color</u>**
1 SLOCA - CHR - CV (3, 6)			
2 SLOCA - HPI - LPI (7)			
3 SLOCA - HPI -DEP (8)			

4 SLOCA - EC (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.			

Notes:

1. For a small LOCA, both the Feedwater and, following depressurization, the Condensate systems are assumed to be unavailable because a small LOCA in a feedwater line would eventually cause the inventory in the condenser hotwell to drain out the break, making condensate fail. Fire water can be injected into the RPV with the electric or diesel driven fire pumps through a quick-connect spool piece that must be installed to connect to the RHR injection lines. (See PILG IPE pages C.3-2, C.3-3 and B.5-7).

2. For depressurization, it is assumed that the operator will always inhibit automatic actuation of ADS, so that depressurization is a manual operator action. (See PILG IPE, page B.4-3).

3. Although credited in the original Pilgrim IPE of September 30, 1992, based on the revised IPE assumptions regarding core cooling following containment failure, Late Inventory Makeup (LI) no longer applies.

Table 2.3 SDP Worksheet for Pilgrim Nuclear Power Station, Unit 1 —— 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 (EI)	HPCI (1 ASD train)[1]
Early Containment Control (EC)	Vapor Suppression System - passive operation of suppression pool - 12/12 vacuum breakers remain closed (1 multi-train system)
Depressurization (DEP)	3/4 safety relief valves (4 ADS SRVs and 2 safety valves) manually opened (high stress operator action)[2]
Low Pressure Injection (LPI)	1/4 RHR trains in LPCI mode (1 multi-train system) or 1/2 LPCS trains (1 multi-train system)
Containment Heat Removal (CHR)	[1/4 RHR pumps in Shutdown Cooling (SDC) mode if Rx pressure <100 psig or in Suppression Pool Cooling (SPC) mode or in Containment Spray (CS) mode] (operator action) or 1/ 2 Fire pumps in Containment Spray mode (operator action)
Containment Venting (CV)	CV through 1/1 8" bypass around Standby Gas Treatment System - (high stress operator action)

Circle Affected Functions	**Recovery or Failed Train**	***Remaining* Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 MLOCA - CHR - CV (3, 7)			
2 MLOCA - LPI (4, 8)			
3 MLOCA - EI - DEP (9)			
4 MLOCA - EC (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. For a medium LOCA, both the Feedwater and, following depressurization, the Condensate systems are assumed to be unavailable because a medium LOCA in a feedwater line would eventually cause the inventory in the condenser hotwell to drain out the break, making condensate fail. (See PILG IPE pages C.3-6).

2. For depressurization, it is assumed that the operator will always inhibit automatic actuation of ADS, so that depressurization is a manual operator action. (See PILG IPE, page B.4-3).

3. Although credited in the original Pilgrim IPE of September 30, 1992, based on the revised IPE assumptions regarding core cooling following containment failure, Late Inventory Makeup (LI) no longer applies.

Table 2.4 SDP Worksheet for Pilgrim Nuclear Power Station, Unit 1 —— Large 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 (EI)	1/4 RHR train in LPCI mode (1 multi-train system) or 1/2 LPCS train (1 multi-train system)[1]
Early Containment Control (EC)	Vapor Suppression system- passive operation of suppression pool - 11/12 vacuum breakers remain closed (1 multi-train system)
Containment Heat Removal (CHR)	[1/4 RHR pumps in Shutdown Cooling (SDC) mode if Rx pressure <100 psig or in Suppression Pool Cooling (SPC) mode or in Containment Spray (CS) mode] (operator action) or 1/ 2 Fire pumps in Containment Spray mode (operator action)
Containment Venting (CV)	CV through 1/1 8" bypass around Standby Gas Treatment System - (high stress operator action)

Circle Affected Functions	**Recovery or Failed Train**	***Remaining* Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 LLOCA - CHR - CV (3)			
2 LLOCA - EI (4)			
3 LLOCA - EC (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.

<u>Notes</u>:

1. For a large LOCA, the Feedwater and Condensate systems are assumed to be unavailable because a small LOCA in a feedwater line would eventually cause the inventory in the condenser hotwell to drain out the break, making condensate fail. (See PILG IPE pages C.3-6).

2. Although credited in the original Pilgrim IPE of September 30, 1992, based on the revised IPE assumptions regarding core cooling following containment failure, Late Inventory Makeup (LI) no longer applies.

Table 2.5 SDP Worksheet for Pilgrim Nuclear Power Station, Unit 1 —— Loss of Offsite Power (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 Power (EAC)	1/2 EDGs (1 multi-train system) ***or 23 KV line (operator action)*** or 1/1 SBO diesel generator (operator action)[1]
Recovery of LOOP in 2 hrs (RLOOP2)	High stress operator action[1]
High Pressure Injection (HPI)	HPCI (1 ASD train) or RCIC (1 ASD train)[2]
Depressurization (DEP)	3/4 safety relief valves (4 ADS SRVs and 2 safety valves) manually opened (high stress operator action)[3]
Alternate Low Pressure Injection (ALTLPI)	1/1 Fire Water Cross-tie (high stress operator action)
Recovery of LOOP in 5 hrs (RLOOP 5)	Operator action[1]
Recovery of LOOP in 12 hrs (RLOOP12)	Operator action[1]
Containment Heat Removal (CHR)	PCS (operator action) or [1/4 RHR pumps in Shutdown Cooling (SDC) mode if Rx pressure <100 psig or in Suppression Pool Cooling (SPC) mode or in Containment Spray (CS) mode] (operator action) or 1/ 2 Fire pumps in Containment Spray mode (operator action)
Containment Venting (CV)	CV through 1/1 8" bypass around Standby Gas Treatment System - (high stress operator action)

Circle Affected Functions	**Recovery or Failed Train**	***Remaining*** **Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 LOOP - TRANS - CHR - CV (1,2,3,6, 8,11)			
2 LOOP - EAC - RLOOP12 (7, 12)			
3 LOOP - EAC - RLOOP2 - HPI - ALTLPI (13)			

4 LOOP - EAC - RLOOP2 - HPI - DEP (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. During a station blackout, key operator actions to recover offsite power or recover a diesel generator occur at 0-2 hours, 2-5 hours, 5-12 hours, and 12-15 hours.
   - At 0-2 hours, recovery of offsite power is assumed to be sufficient to allow for feedwater and main condenser restoration while recovery of a diesel generator or the 23 KV source allows for operation of most of the AC-powered systems and continued battery charging.
   - At 2-5 hours, recovery of offsite power is assumed to be sufficient to allow for initiation of a number of coolant inventory control systems.
   - At 5-12 hours, recovery of off-site power will still allow for successful heat removal before excessive impacts due to battery failures.
   - At 12-15 hours, recovery of off-site power will allow for successful heat removal before both the A and B batteries fail if load shedding has already been performed.
   - If no AC power is recovered within 15 hours, all DC batteries and core damage will occur. (See PILG IPE pages C.2-15 to C.2-18).

   Thus, the PILG IPE event trees for LOOP and SBO (Figs. C.2-1 through C.2-5), in particular Fig. C.2-3, contains headings associated primarily with operator actions such as recovering AC power or shedding loads from the batteries. The PILG Loss of DC operating procedures instruct the operators to begin shedding loads approximately 5 hours into a loss of AC transient. According to the SBO study (GE NEDC-31423, July 1987), if loads are not shed successfully, 125 VDC Battery A will deplete within 8 hours. The 12 hour time limit for the second time period was assumed to encompass this time and to correspond to AC power recovery time periods reported in the literature. Finally, with successful operator action to shed loads (which is the most likely pathway), both the A and B batteries will deplete within 15 hours. Thus 15 hours was chosen as the upper limit for the third AC power recovery time period. Failure to recover any source of AC power within 12 hours (no load shedding) or 15 hours (successful load shedding) is assumed to result in core damage. (See PILG C.2-9 and C.2-10).

**For the purposes of this SDP document, credit for battery load shedding has been neglected and credit for recovery of offsite power up to 12 hours only is assumed.

2. The HPCI system is designed to automatically transfer suction to the suppression pool on high suppression pool water level. However, suppression pool heatup may jeopardize HPCI system operation because the HPCI pump lube oil is cooled by the water being pumped. The BECo emergency operating procedures recommend bypassing the interlocks which switch HPCI pump suction to the suppression pool from the condensate storage tank. (See PILG IPE page B.3-2 and B.3.3).

3. For depressurization, it is assumed that the operator will always inhibit automatic actuation of ADS, so that depressurization is a manual operator action. (See PILG IPE, page B.4-3).

4. Although credited in the original Pilgrim IPE of September 30, 1992, based on the revised IPE assumptions regarding core cooling following containment failure, Late Inventory Makeup (LI) no longer applies.

Table 2.6 SDP Worksheet for Pilgrim Nuclear Power Station, Unit 1 —— ATWS

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:**
Overpressure Protection (OVERP)	With MSIV closure: *6/6* SRVs and safety valves must open (1 multi-train system); With no MSIV closure: *4/6* SRVs or safety valves must open (1 multi-train system)
Recirculation Pump Trip (RPT)	With MSIV closure: manual or automatic trip of 2/2 recirculation pumps (1 multi-train system) and 3/3 feedwater pumps trip (1 multi-train system); With no MSIV closure: manual or automatic trip of 1/ 2 recirculation pumps (1 multi-train system)[1]
Inhibit ADS (INH)	*High stress operator action*[2]
High Pressure Injection (HPI)	With MSIV closure: HPCI (1 ASD train) and RPV level control (high stress operator action); With no MSIV closure: 1/3 Feedwater pumps and 1/3 Condensate pumps and Operable Condenser and FW RPV level control (high stress operator action)
Reactivity Control (SLC)	1/ 2 SLC pumps (high stress operator action)[3]
Depressurization (DEP)	4/4 ADS SRVs manually opened (high stress operator action)[2]
Low Pressure Injection (LPI)	1/ 4 RHR trains in LPCI mode (1 multi-train system) or 1/ 2 LPCS trains (1 multi-train system)
Containment Heat Removal (CHR)	PCS (operator action) or [1/4 RHR pumps in Shutdown Cooling (SDC) mode if Rx pressure <100 psig or in Suppression Pool Cooling (SPC) mode or in Containment Spray (CS) mode] (operator action) or 1/ 2 Fire pumps in Containment Spray mode (operator action)

Circle Affected Functions	**Recovery or Failed Train**	***Remaining* Mitigation Capability Rating for Each Affected Sequence:**	**Sequence Color**
1 ATWS - CHR (2, 4)			
2 ATWS - HPI - SLC (5)			

3 ATWS - HPI - LPI (6)			
4 ATWS - HPI - DEP (7)			
5 ATWS - INH (8)			
6 ATWS - OVERP (9)			
7 ATWS - RPT (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.

<u>Notes</u>:

1. For ATWS sequences in which the main condenser IS NOT available (including MSIV closure):

   - Both recirculation pumps and all three feedwater pumps must trip to attain the necessary reduction in power.

   For ATWS sequences in which the main condenser IS available (such as No MSIV closure):

- One out of two recirculation pumps must trip to attain the necessary reduction in power. No feedwater pump trip is necessary. (See PILG IPE page C.5-16).

2. For depressurization, it is assumed that the operator will always inhibit automatic actuation of ADS, so that depressurization is a manual operator action. Operator inhibiting of ADS is not shown as a separate function on the PILG IPE ATWS success criteria or event trees. (See PILG IPE, page B.4-3, Table C.5-1 and Figs C.5-1 and C.5-2).

3. For events in which high pressure injection systems are in operation, calculations indicate that approximately 20 minutes would be available to initiate SLC if the reactor remains at high pressure, while 20 to 30 minutes would remain available if depressurization to low pressure systems occurs. (See PILG IPE page C.5-19).

General Notes

1. For a small LOCA, both the Feedwater and, following depressurization, the Condensate systems are assumed to be unavailable because a small LOCA in a feedwater line would eventually cause the inventory in the condenser hotwell to drain out the break, making condensate fail. Fire water can be injected into the RPV with the electric or diesel driven fire pumps through a quick-connect spool piece that must be installed to connect to the RHR injection lines. (See PILG IPE pages C.3-2, C.3-3 and B.5-7).

2. For a medium LOCA, both the Feedwater and, following depressurization, the Condensate systems are assumed to be unavailable because a medium LOCA in a feedwater line would eventually cause the inventory in the condenser hotwell to drain out the break, making condensate fail. (See PILG IPE pages C.3-6).

3. For a large LOCA, the Feedwater and Condensate systems are assumed to be unavailable because a small LOCA in a feedwater line would eventually cause the inventory in the condenser hotwell to drain out the break, making condensate fail. (See PILG IPE pages C.3-6).

4. During a station blackout, key operator actions to recover offsite power or recover a diesel generator occur at 0-2 hours, 2-5 hours, 5-12 hours, and 12-15 hours.
   - At 0-2 hours, recovery of offsite power is assumed to be sufficient to allow for feedwater and main condenser restoration while recovery of a diesel generator or the 23 KV source allows for operation of most of the AC-powered systems and continued battery charging.
   - At 2-5 hours, recovery of offsite power is assumed to be sufficient to allow for initiation of a number of coolant inventory control systems.
   - At 5-12 hours, recovery of off-site power will still allow for successful heat removal before excessive impacts due to battery failures.
   - At 12-15 hours, recovery of off-site power will allow for successful heat removal before both the A and B batteries fail if load shedding has already been performed.
   - If no AC power is recovered within 15 hours, all DC batteries and core damage will occur. (See PILG IPE pages C.2-15 to C.2-18).

5. The HPCI system is designed to automatically transfer suction to the suppression pool on high suppression pool water level. However, suppression pool heatup may jeopardize HPCI system operation because the HPCI pump lube oil is cooled by the water being pumped. The BECo emergency operating procedures recommend bypassing the interlocks which switch HPCI pump suction to the suppression pool from the condensate storage tank. (See PILG IPE page B.3-2 and B.3.3).

6. For ATWS sequences in which the main condenser IS NOT available (including MSIV closure):

   - Both recirculation pumps and all three feedwater pumps must trip to attain the necessary reduction in power.

For ATWS sequences in which the main condenser IS available (such as No MSIV closure):

- One out of two recirculation pumps must trip to attain the necessary reduction in power. No feedwater pump trip is necessary. (See PILG IPE page C.5-16).

7. For events in which high pressure injection systems are in operation, calculations indicate that approximately 20 minutes would be available to initiate SLC if the reactor remains at high pressure, while 20 to 30 minutes would remain available if depressurization to low pressure systems occurs. (See PILG IPE page C.5-19).

8. For depressurization, it is assumed that the operator will always inhibit automatic actuation of ADS, so that depressurization is a manual operator action. (See PILG IPE, page B.4-3).

9. The containment spray mode of the fire protection system is a manual alignment, requiring startup of one of the two fire protection pumps and manual alignment of flow to the wetwell airspace or drywell spray headers, further requiring installation of a spool piece and opening of some manual and motor-operated valves. For these reasons, the RHR in the containment spray mode is the preferred method of containment spray. (See PILG IPE page B.6-8).

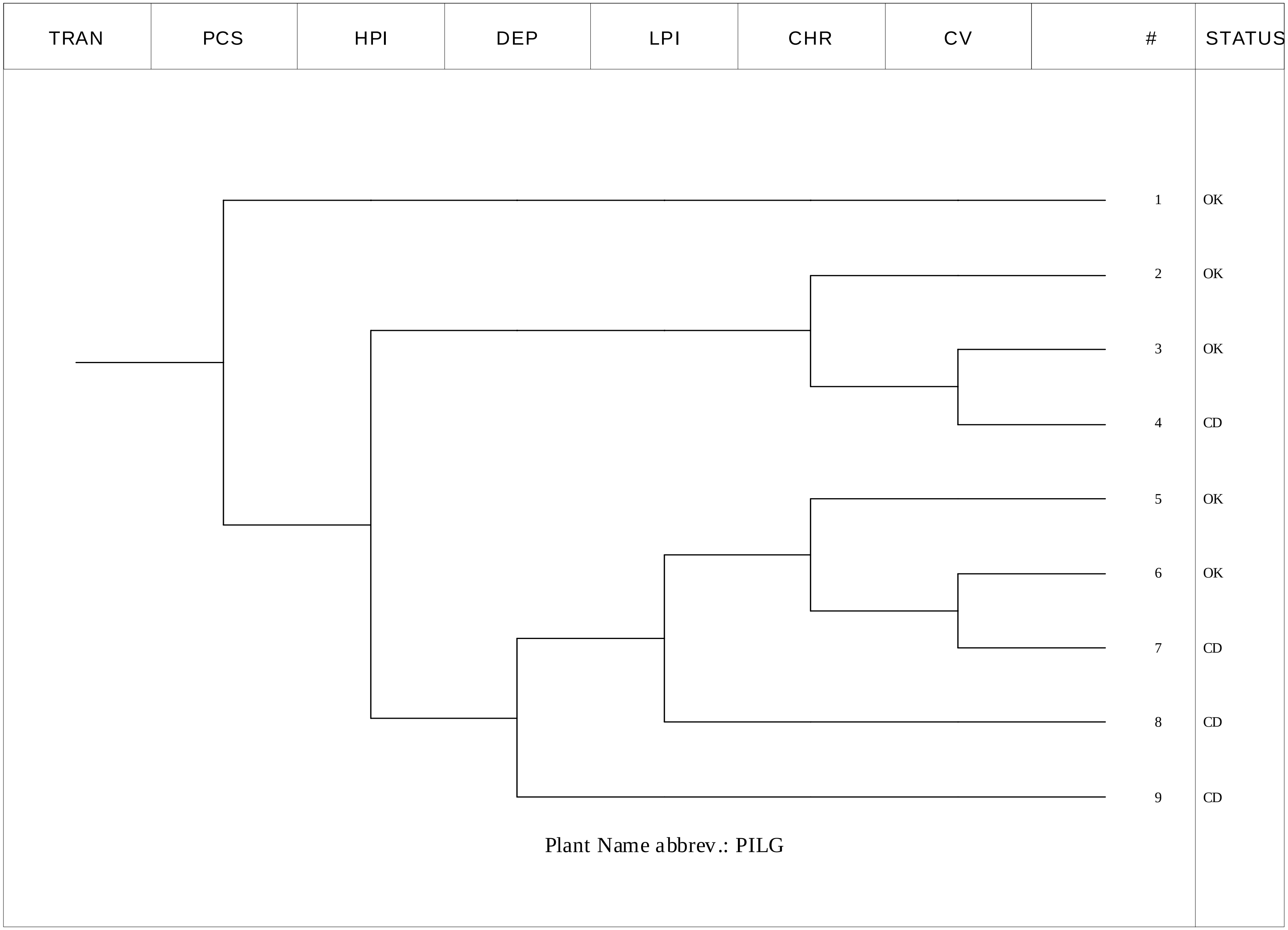
10. Subsequent review of the results of the IPE analysis by Boston Edison Co. showed that core damage occurs prior to containment failure, and that the IPE analysis cannot take credit for successful core cooling after containment failure. The PILG IPE model was subsequently revised to show that core damage occurs prior to containment failure and the IPE was requantified to reflect this change in philosophy. The original CDF was 5.8E-5 events/Rx year. The revised CDF is 2.84E-5 events/Rx year (despite the apparent increased conservatism with respect to core cooling following containment failure). (See PILG Response to RAI, December 28, 1995, Question 7d, page 17).

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. 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. Anticipated Transients Without Scram (ATWS)



Plant Name abbrev.: PILG

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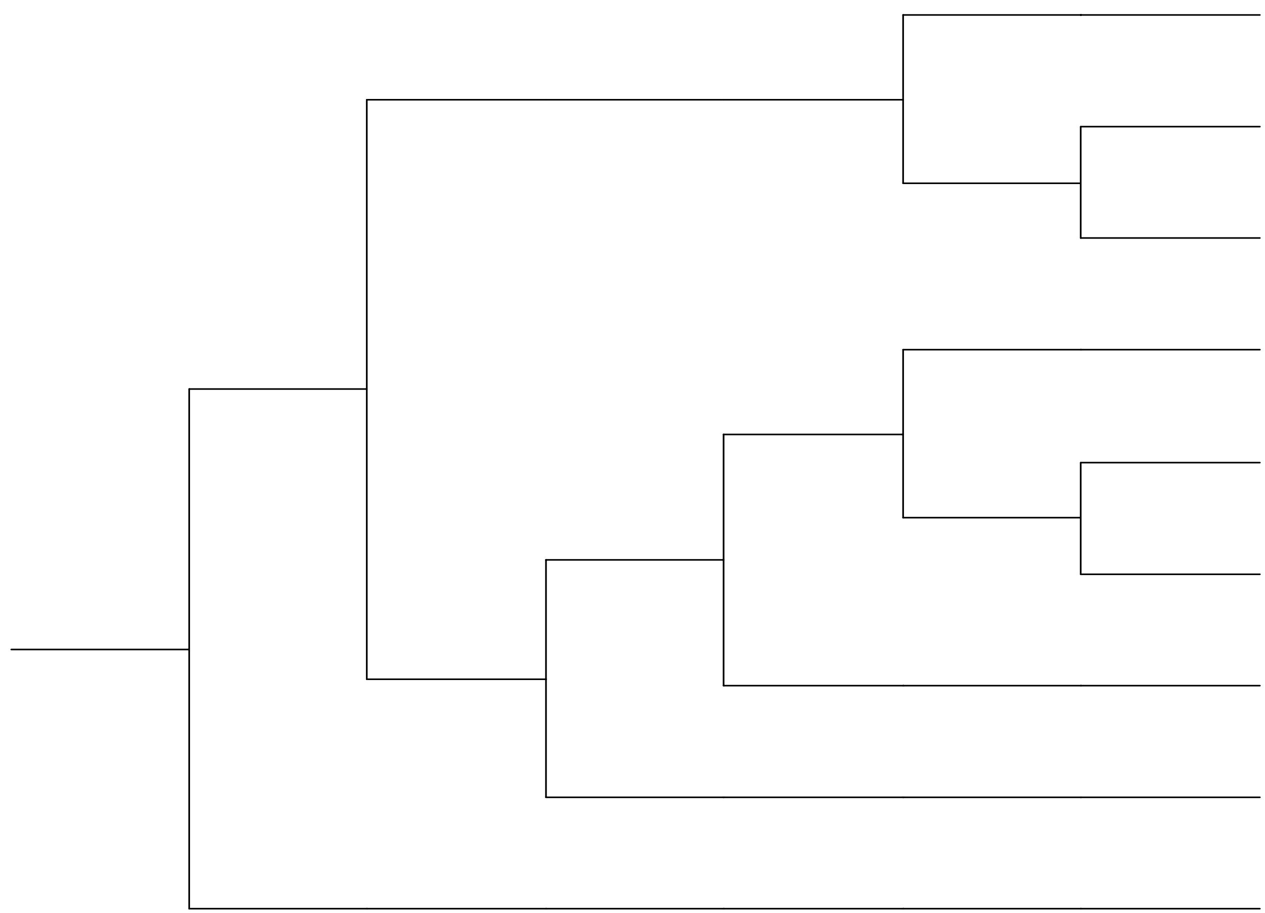

#	STATUS
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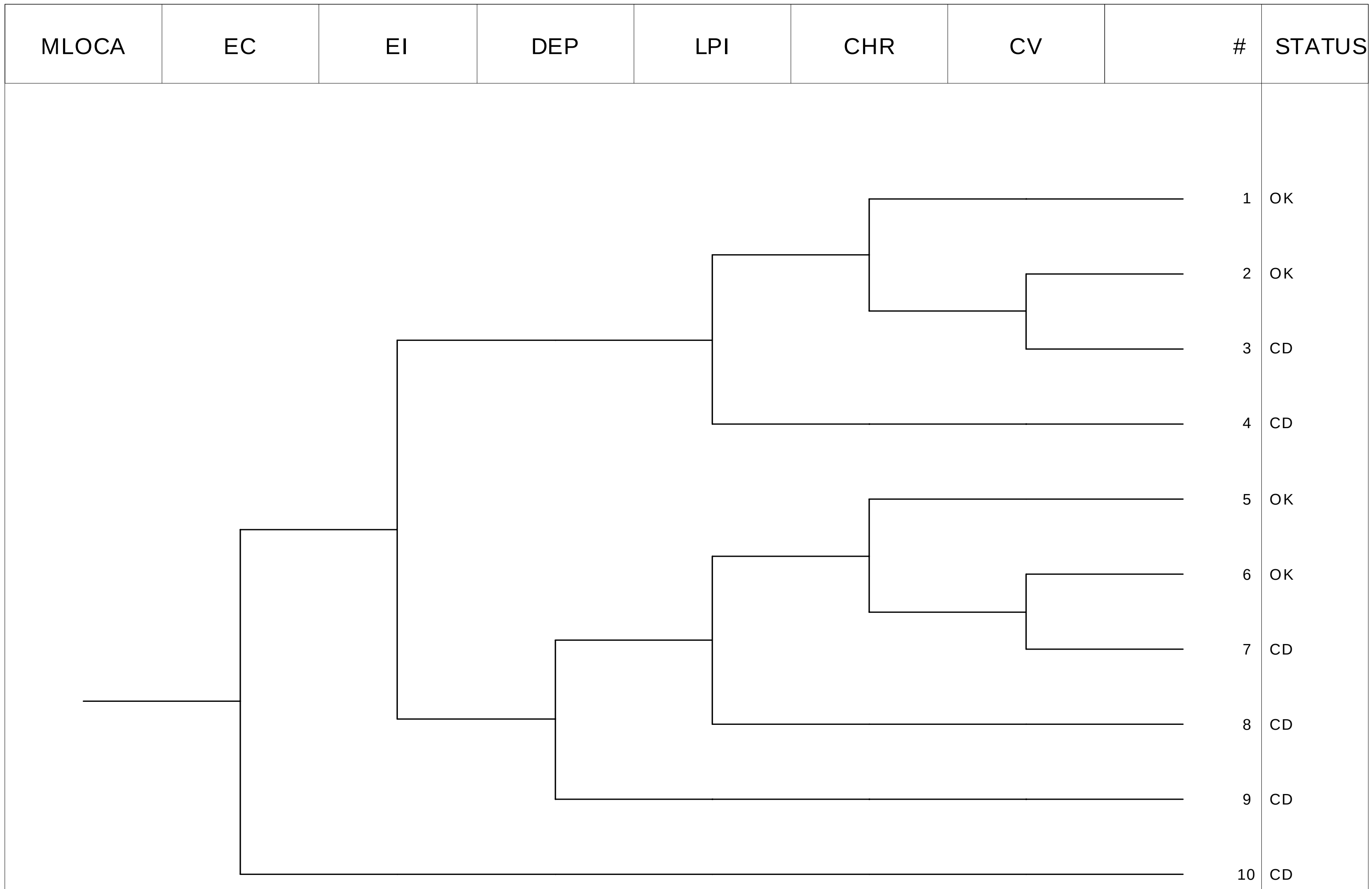
2	OK
3	CD
4	OK
5	OK
6	CD
7	CD
8	CD
9	CD

Plant Name abbrev.: PILG



Plant Name abbrev.: PILG

LLOCA	EC	EI	CHR	CV	#	STATUS
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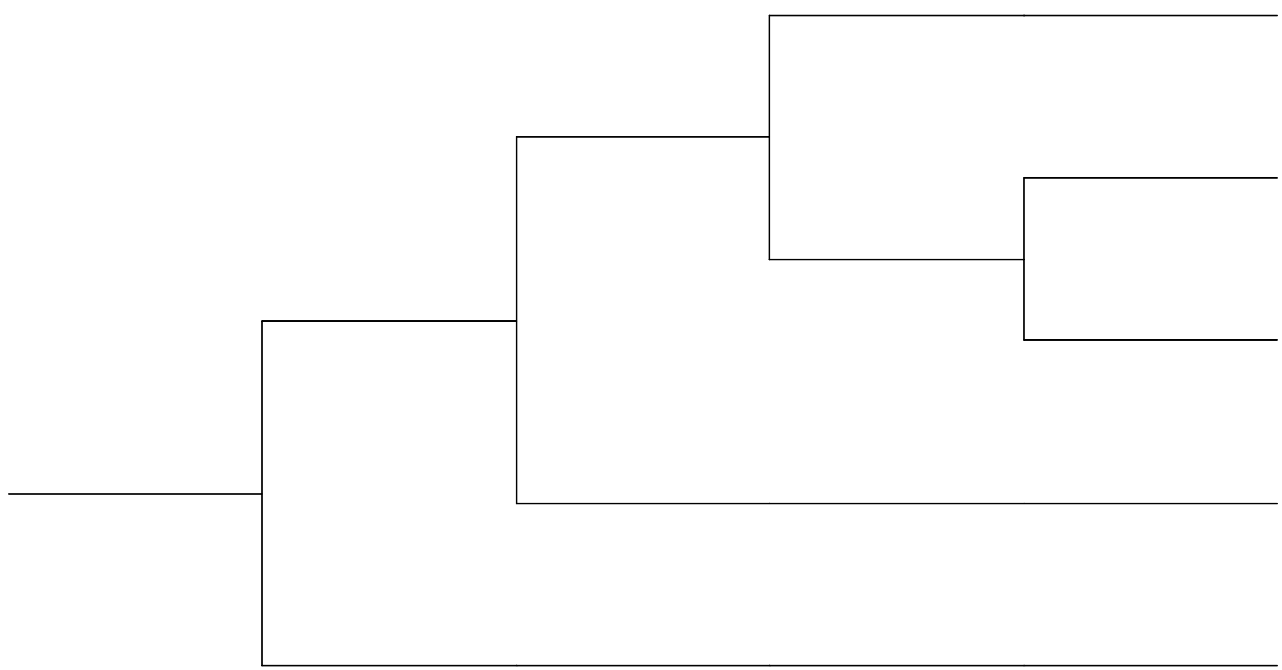
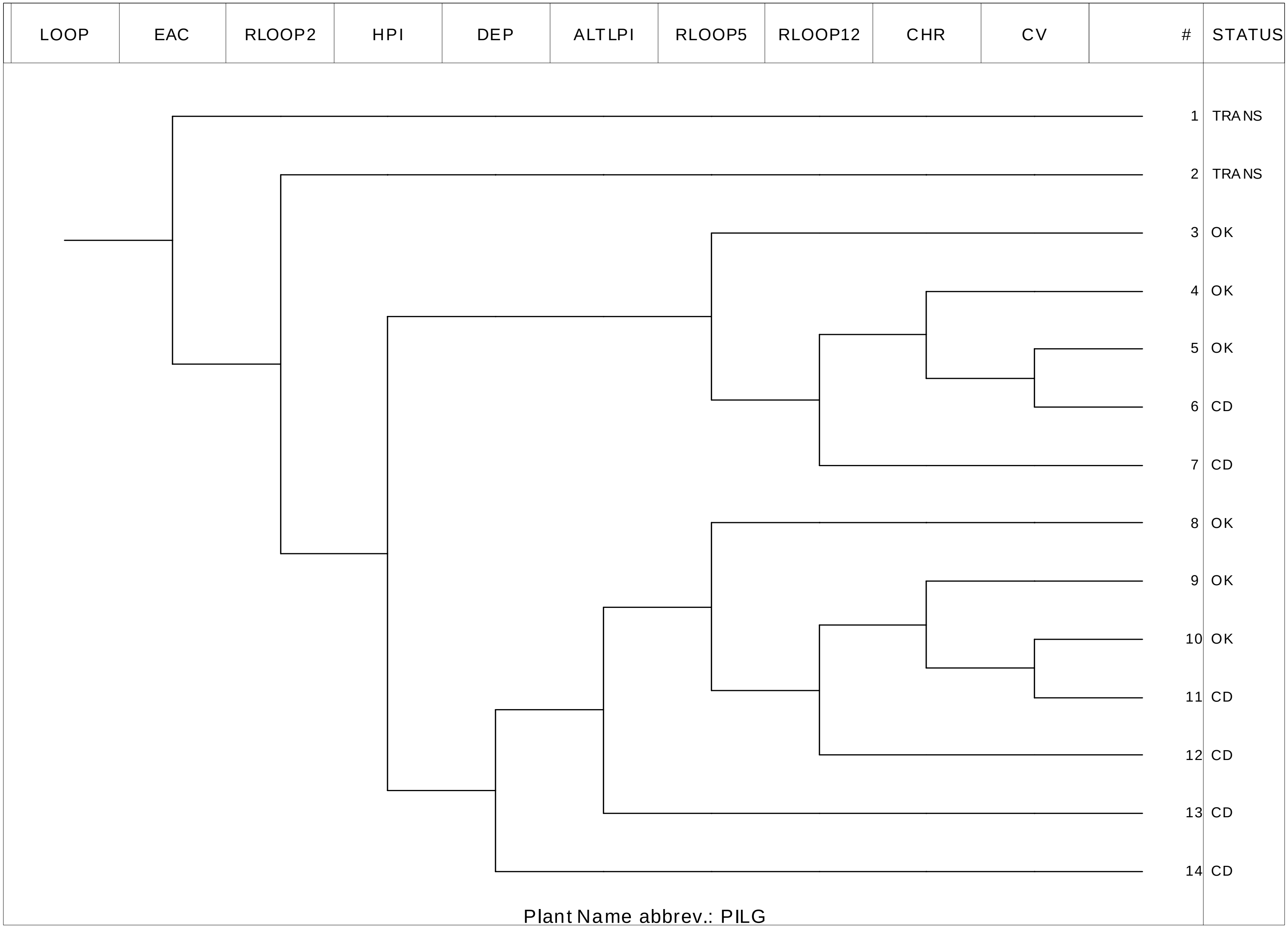
					2	OK
					3	CD
					4	CD
					5	CD



Plant Name abbrev.: PILG


LOOP
EAC
RLOOP2
HPI
DEP
ALTLPI
RLOOP5
RLOOP12
CHR
CV
#
STATUS
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2 TRANS
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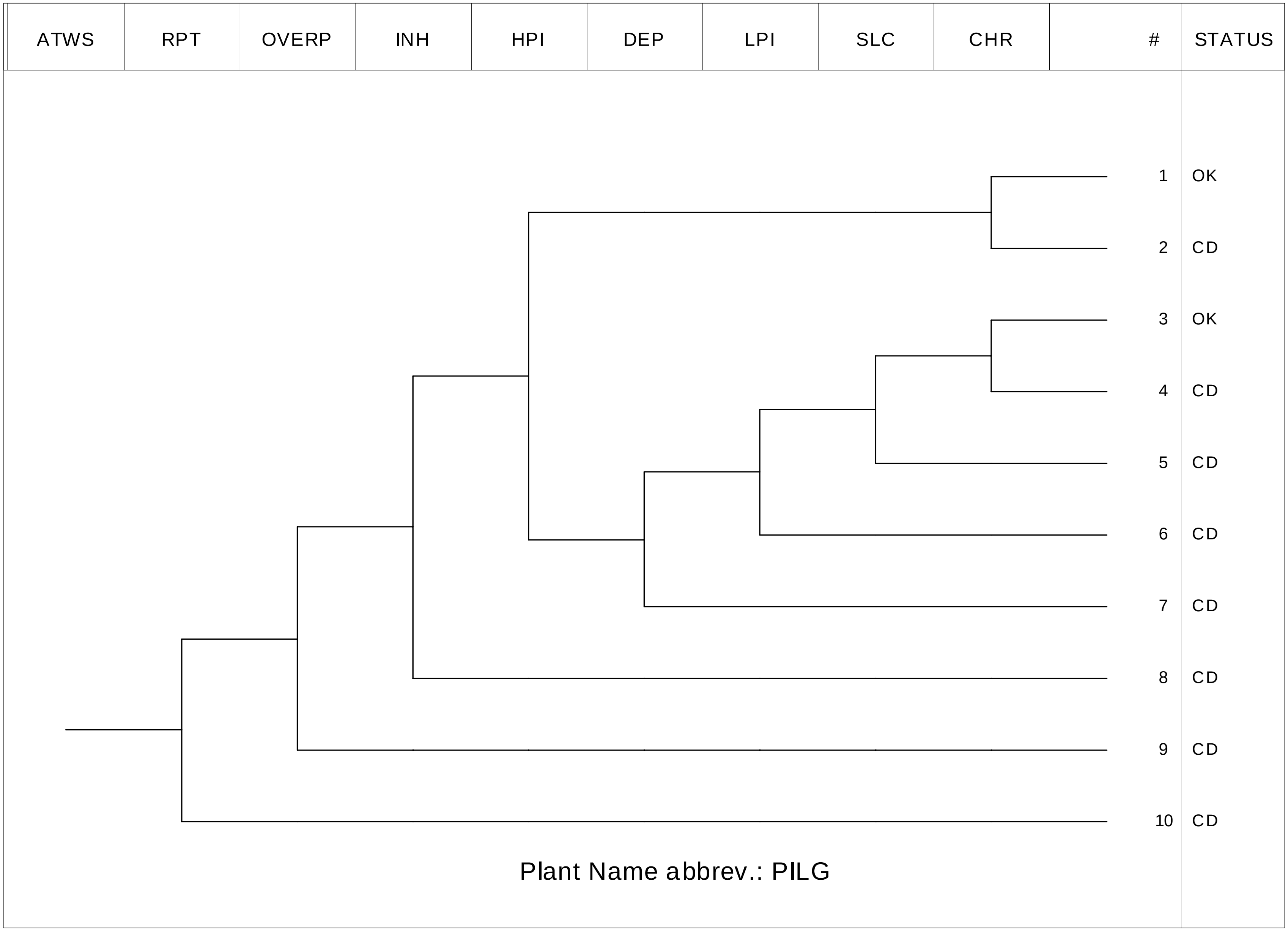
4 OK
5 OK
6 CD
7 CD
8 OK
9 OK
10 OK
11 CD
12 CD
13 CD
14 CD
Plant Name abbrev.: PILG


ATWS
RPT
OVERP
INH
HPI
DEP
LPI
SLC
CHR
#
STATUS
1 OK
2 CD
3 OK
4 CD
5 CD
6 CD
7 CD
8 CD
9 CD
10 CD
Plant Name abbrev.: PILG

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. Boston Edison Co., "Pilgrim Nuclear Power Station — Individual Plant Examination Report," September 30, 1992.

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