May 22, 2000

Mr. Robert P. Powers, Senior Vice President
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

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

Dear Mr. Powers:

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 Donald C. Cook Nuclear plant following the restart of Unit 2, scheduled for late spring of 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. 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 301-415-1345.

Sincerely,

/RA/

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-315 and 50-316

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 301-415-1345.

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/RA/

John F. Stang, Senior Project Manager, Section 1
Project Directorate III
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cc:

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, IL 60532-4351

Attorney General
Department of Attorney General
525 West Ottawa Street
Lansing, MI 48913

Township Supervisor
Lake Township Hall
P.O. Box 818
Bridgman, MI 49106

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
7700 Red Arrow Highway
Stevensville, MI 49127

David W. Jenkins, Esquire
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

Mayor, City of Bridgman
P.O. Box 366
Bridgman, MI 49106

Special Assistant to the Governor
Room 1 - State Capitol
Lansing, MI 48909

Drinking Water and Radiological
  Protection Division
Michigan Department of
  Environmental Quality
3423 N. Martin Luther King Jr Blvd
P.O. Box 30630, CPH Mailroom
Lansing, MI 48909-8130

Robert C. Godley
Director, Regulatory Affairs
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

David A. Lochbaum
Union of Concerned Scientists
1616 P Street NW, Suite 310
Washington, DC 20036-1495

A. Christopher Bakken, Site Vice President
Indiana Michigan Power Company
Nuclear Generation Group
One Cook Place
Bridgman, MI 49106

Michael W. Rencheck
Vice President, Nuclear Engineering
Indiana Michigan Power Company
Nuclear Generation Group
500 Circle Drive
Buchanan, MI 49107

RISK-INFORMED INSPECTION NOTEBOOK FOR

DONALD C. COOK NUCLEAR PLANT

UNITS 1 AND 2

PWR, WESTINGHOUSE, FOUR-LOOP PLANT WITH ICE CONDENSER CONTAINMENT

Prepared by

Brookhaven National Laboratory
Department of Advanced Technology

Contributors

M. A. Azarm
J. Carbonaro
T. L. Chu
A. Fresco
J. Higgins
G. Martinez-Guridi
P. K. Samanta

NRC Technical Review Team

John Flack	**RES**
Morris Branch	**NRR**
Doug Coe	**NRR**
Gareth Parry	**NRR**
Peter Wilson	**NRR**
Jim Trapp	**Region I**
Michael Parker	**Region III**
William B. Jones	**Region IV**

Prepared for

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

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:

Mr. Jose G. Ibarra
U. S. Nuclear Regulatory Commission
RES/DSARE/REAHFB
TWFN T10 E46
11545 Rockville Pike
Rockville, MD 20852

ABSTRACT

This notebook contains summary information to support the Significance Determination Process (SDP) in risk-informed inspections for the Donald C. Cook Nuclear 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.

CONTENTS

FIGURES

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 Donald C. Cook Nuclear 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 Donald C. Cook Nuclear Plant, Units 1 and 2

Affected Systems	Major Components	Support Systems	Initiating Event
AC Power System	4160 V-AC	250 V-DC	Transient, SLOCA, SORV, MLOCA, LLOCA, LOOP, SGTR, MSLB, ATWS, RCP seal LOCA
	600 V-AC	4160 V-AC, 250 V-DC	
	120 V-AC	600 V-AC, 250 V-DC	
Accumulators	4 Accumulators per unit	None	LLOCA
AFW	2 MDPs	4160 V-AC, 600 V-AC, 250 V-DC, 120 V-AC, CA, Signals, ESW (1)	Transient, SLOCA, SORV, LOOP, SGTR, MSLB, ATWS
	1 TDP		
CCW	2 CCW pumps (2) and 2 Heat exchangers.	4160 V-AC, 600 V-AC, 250 V-DC, CA, Signals	Transient, SLOCA, SORV, MLOCA, LLOCA, LOOP, SGTR, MSLB, RCP seal LOCA
Condensate / MFW	3 50% Condensate pumps and 3 50% Condensate booster pumps	600 V-AC, 250 V-DC, CA	Transient
	2 50% Turbine-driven variable speed pumps	600 V-AC, CA, Signals	
Containment Fans	2 Air Recirculation Fans	600 V-AC, CCW, Signals	LLOCA
Containment Spray System	2 100% Capacity trains each with a spray pump and a Heat exchanger	4160 V-AC, 600 V-AC, 250 V-DC, ESW, Signals	LLOCA
DC Power System	250 V-DC	600 V-AC	Transient, SLOCA, SORV, MLOCA, LLOCA, LOOP, SGTR, MSLB, ATWS, RCP seal LOCA
EDG	2 EDGs per unit	600 V-AC, 250 V-DC, ESW, Signals	LOOP

Table 1 (Continued)

Affected Systems	Major Components	Support Systems	Initiating Event
Signals		120 V-AC	Transient, SLOCA, SORV, MLOCA, LLOCA, LOOP, SGTR, ATWS, MSLB. RCP seal LOCA
HPI/HPR	2 Charging (CC) and 2 safety injection (SI) pumps	4160 V-AC, 600 V-AC, 250 V-DC, CCW, Signals	Transient, SLOCA, SORV, MLOCA, LLOCA, LOOP, SGTR, MSLB, ATWS, RCP seal LOCA
Control Air (CA)	2 Plant Air Compressors (PACs) for both units. (3)	600 V-AC, 250 V-DC, NESW	Transient, SLOCA, SORV, LOOP, SGTR, MSLB, ATWS
Main Steam	1 SG stop valve, 5 Safety valves and 1 PORV per SG; 9 Steam dump valves	250 V-DC, Signals	SGTR, MSLB
SG PORVs	1 PORV per SG	120 V-AC, CA	SLOCA, SORV, SGR, MSLB
Nonessential Service Water System (NESW)	4 NESW Supply pumps shared by 2 units; 2 Pumps are located in each unit	600 V-AC, 250 V-DC, CA	Transients, SLOCA, SORV, LOOP, SGTR, MSLB, ATWS
Pressurizer Pressure Relief	3 Safety valves and 3 PORVs with associated block valves	600 V-AC, 250 V-DC, 120 V-AC, CA	Transient, SLOCA, LOOP, SGTR, MSLB, ATWS
RCP	Seals	1 / 2 Charging pumps to seal injection or 1 / 2 CCW pumps to thermal barrier heat exchanger	LOOP, RCP seal LOCA
LPI/LPR	2 RHR Pumps and 2 Heat exchangers	4160 V-AC, 600 V-AC, 250 V-DC, CCW, Signals	Transient, SLOCA, SORV, MLOCA, LLOCA, LOOP, SGTR, MSLB
Essential Service Water Syatem (ESW)	4 ESW Pumps shared by both units(4)	4160 V-AC, 600 V-AC, 250 V-DC, CA, Signals	Transient, SLOCA, SORV, MLOCA, LLOCA, LOOP, SGTR, MSLB, ATWS, RCP seal LOCA

Table 1 (Continued)

Table 1 (Continued)

Notes:

(1) The primary source of water is the unit's 500,000 gallon CST. An emergency backup is available from the opposite unit's CST through normally CST cross-tie valve, 12-CRV-51. If both unit's CST are unavailable, then the system may be aligned to take suction from ESW.

(2) An installed spare maintenance pump is available as a replacement for either unit's CCW system. The two headers are normally cross-tied through the miscellaneous service header, but after an accident occurs, the headers are se parated by the operator and function as independent system.

(3) Additionally, each unit has a control air compressor (CAS) which is capable of supplying that unit's control air (instrument air) needs.

(4) The ESW headers are arranged in such a way that a rupture in either header will not jeopardize the safety function of the system. Two pumps are sufficient to supply all service water requirements for unit operation, shutdown, refueling, or post-accident operation, including a LOCA in one unit and a simultaneous hot shutdown in the other. However, a third pump is normally started under the shutdown and refueling operations.

(5) As stated in the submittal, the Donald C. Cook Nuclear Plant IPE analyzed the design and operation of Unit 1. Unit 2 was examined and no differences from Unit 1 were identified which would have impacted the IPE results. The IPE, therefore, applies to either unit.

(6) Plant internal event CDF (including internal floods) = 6.26 E-5 per year

1.2 SDP WORKSHEETS

This section presents the SDP worksheets to be used in the Phase 2 evaluation of the inspection findings for the Donald C. Cook Nuclear 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
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)
10. Special Initiators

Table 2.1 SDP Worksheet for Donald C. Cook 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 to 2/4 SGs or AFW from the opposite unit (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 CC and 1/ 2 SI pumps (1 multi-train system)
Primary Heat Removal, Feed/Bleed (FB)	2/ 3 PORVs and block valves open for Feed/Bleed (operator action) (1)
High Pressure Recirculation (HPR)	1 / 2 CC and 1/ 2 SI 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.

<u>Note</u>:

(1) The human error probability (HEP) assessed in the IPE (page 3-156) for establishing bleed and feed cooling is 3.4E-05.

Table 2.2 SDP Worksheet for Donald C. Cook 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

<u>Safety Functions Needed</u>:	***<u>Full</u>* <u>Creditable Mitigation Capability for Each Safety Function</u>:**
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 CC and 1/ 2 SI pumps (1 multi-train system)
Primary Heat Removal, Feed/Bleed (FB)	2/ 3 PORVs and block valves open for Feed/Bleed (operator action) (1)
High Pressure Recirculation (HPR)	1 / 2 CC and 1/ 2 SI pumps with 1 / 2 RHR pumps (Requires operator action for switchover = 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 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.

<u>Note</u>:

(1) The human error probability (HEP) assessed in the IPE (page 3-156) for establishing bleed and feed cooling is 3.4E-05.

Table 2.3 SDP Worksheet for Donald C. Cook 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 CC and 1/ 2 SI pumps (1 multi-train system).
Primary Cooldown (RCSCOOL)	RCS cool down using AFW and 2/ 4 SG PORVs (operator action)
Primary Bleed (FB)	2/ 3 PORVs and block valves open for Feed/Bleed (operator action) [1]
High Pressure Recirculation (HPR)	1/ 2 CC and 1/ 2 SI 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 - HPR (2, 4,7)			
2 SLOCA - RCSCOOL - FB (5)			
3 SLOCA - AFW - FB (8)			
4 SLOCA - EIHP (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.

Note:

(1) The human error probability (HEP) assessed in the IPE (page 3-156) for establishing bleed and feed cooling is 3.4E-05.

Table 2.4 SDP Worksheet for Donald C. Cook 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 CC and 1/ 2 SI 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 (RCSCOOL)	Operator cools down RCS using AFW and steam dump (operator action)
Primary Bleed (FB)	1 / 2 remaining PORVs open for Feed/Bleed (operator action [1]
High Pressure Recirculation (HPR)	1 / 2 CC and SI pumps with 1 / 2 RHR pumps (operator action)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 SORV - BLK - HPR (2,4,7)			
2 SORV - BLK - RCSCOOL - FB (5)			
3 SORV - BLK - AFW - F&B (8)			
4 SORV - BLK - EIHP (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.

<u>Note</u>:

(1) The human error probability (HEP) assessed in the IPE (page 3-156) for establishing bleed and feed cooling is 3.4E-05

Table 2.5 SDP Worksheet for Donald C. Cook 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, Accumulators (EIAC)	3 of 3 accumulators to the intact cold legs (1 multi-train system)
Early Inventory, HP Injection (EIHP)	1/ 2 CC and 1/ 2 SI pumps to 1of 3 intact loops (1 multi-train system).
RCS Cooldown (RCSCOOL) (1)	Operator cools down RCS using AFW and 2/ 4 SG PORVs (operator actions)
RCS Depressurization (RCSDEP) (2)	Operator depressurizes RCS using 2/4 SG PORVs and 2/ 3 pressurizer PORVs (high stress operator action)
Low Pressure Injection (LPI)	1/ 2 RHR pumps injecting to 1/3 cold legs (1 multi-train system)
High Pressure Recirculation (HPR)	1/ 2 RHR trains supplying 1/ 2 CC and 1/ 2 SI pumps (operator action)
Low Pressure Recirculation (LPR)	1/ 2 RHR pump trains switched from RWST 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 - RCSCOOL (3)			
3 MLOCA - EIHP - LPR (5)			
4 MLOCA - EIHP - LPI (6)			
5 MLOCA - EIHP - RCSDEP (7)			

6 MLOCA - EIAC (8)			

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) RCS cool down should be initiated within 2 hours following the accident. IPE estimates an error probability of 6.23E-4 (pg. 3-25)

(2) RCS depressurization should be initiated within 20 minutes of accident initiation. IPE estimates and error probability of 1.51E-1. It is, accordingly, designated as a high stress operator action.

Table 2.6 SDP Worksheet for Donald C. Cook 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

Safety Functions Needed:	***Full* Creditable Mitigation Capability for Each Safety Function:**
Early Inventory, Accumulators (EIAC)	3 of 3 accumulators to intact cold legs (1 Train)
Containment Pressure / Temperature Control (CNT)	1 / 2 CS trains (1 multi-train system) or 1/ 2 air recirculation fan (1 multi-train system)
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)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 LLOCA - LPR (2)			
2 LLOCA - CNT (3)			
3 LLOCA - EILP (4)			
4 LLOCA - EIAC (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) In the IPE, the human error probability associated with switch over to LPR is estimated as 5.31E-5

Table 2.7 SDP Worksheet for Donald C. Cook 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)
Auxiliary Feed Water Pumps (AFW)	1/ 2 MDAFW pumps(1 multi-train system) or 1/1 TDAFW pump(1ASD train)
Turbine-driven AFW pump (TDAFW)	1 / 1 TDP trains of AFW (1 train)
Motor-driven AFW Pumps (MDAFW)	1 / 2 MDAFW trains (1 multi-train system)
Recovery of AC Power in < 2 hrs (REC2)	SBO procedures implemented (operator action under high stress) (1)
Recovery of AC Power in < 5 hrs (REC5)	SBO procedures implemented (operator action)
Early Inventory, HP Injection (EIHP)	1 / 2 SI and 1/ 2 CC pumps (1 multi-train system)
Primary Heat Removal (FB)	Operator uses RCS pressurizer 2/3 PORVs and block valves (operator action)
High Pressure Recirculation (HPR)	1 / 2 RHR pumps supplying 1/ 2 SI and 1/ 2 CC pumps (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, 9, 14, 16) (AC recovered)			
5 LOOP - EAC - MDAFW - FB (10, 15) (AC recovered)			

6 LOOP - EAC - MDAFW - EIHP (11, 18) (AC recovered)			
7 LOOP - EAC - REC5 (12)			
8 LOOP - EAC - TDAFW - REC2 (19) (AC recovered)			
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) Specific times for recovery of offsite power could not be identified from the IPE. The battery life is estimated at 4 hrs. and the CST depletion time is estimated at 6 hrs. The recovery of LOOP at 2 hrs. is considered based on an estimate of SG dry out in 1 hr. and subsequent core damage in another hour. The recovery of LOOP at 5 hrs. is considered based on battery depletion at 4 hrs. and subsequent core damage at 5hrs. The IPE provides an operator error estimate of failure to recover offsite power in 1-8 hrs. at 3.05E-2.

Table 2.8 SDP Worksheet for Donald C. Cook 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:**
Auxiliary Feedwater System (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 / 4 SI or CC pumps (1 multi-train system)
SG Insolation (SGI)	Operator closes MSIV on the faulted SG (operator action) (1)
Pressure Equalization (EQ)	Operator depressurizes RCS using 2/4 SG PORVs (operator action) (2)
Feed-and-Bleed (FB)	Operator uses RCS pressurizer PORV and block valves (2/3) (operator action under high stress)
High Pressure Recirculation (HPR)	1 / 2 RHR pump trains supplying 1 / 2 CC and 1/ 2 SI pumps (operator action)
Late Depressurization (LDEP)	RCS pressure reduced using 2/ 4 SG PORVs and 1/3 pressurizer PORVs prior to draining RWST (operator action) (3)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 SGTR - EQ - LDEP (3, 5)			
2 SGTR - EIHP - EQ (7)			
3 SGTR - EIHP - SGI (8)			
4 SGTR - AFW - HPR (10)			
5 SGTR - AFW - FB (11)			

6 SGTR - AFW - EIHP (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) Operator action to isolate the faulted SG is assigned an error probability of 5.23E-6 (IPE, pg., 3-155)

(2) Operator action to cool down and depressurize RCS and terminate safety injection before ruptures SG fills is assigned an erroroprobability of 1.05E-4

(3) Operator action to provide long term cool down and RCS depressurization is assigned an error probability of 2.1E-4.

Table 2.9 SDP Worksheet for Donald C. Cook Nuclear Plant, Units 1 and 2 —— 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:**
Long Term Shutdown (LTS)	Operator conducts emergency boration using 1 / 2 charging pumps with 1 / 2 boric acid transfer pumps (operator action)
AMSAC	AMSAC trips turbine and generates signal to start AFW (1 Train)
Primary Relief (PPR)	3 / 3 SRVs with 3/3 PORVs open (1 train)
Secondary Heat Removal (AFW)	2 / 2 MDPs of AFW (1train) 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 - LTS (2)			
2 ATWS - PPR (3)			
3 ATWS - AFW (4)			
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.

Notes:

(1) The human error probability associated with LTS is estimated as 1.35E-3 (IPE, pg., 3-157)

Table 2.10 SDP Worksheet for Donald C. Cook Nuclear Plant, Units 1 and 2 —— MSLB

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**:
ECCS Injection (CHG)	1 / 2 CC pumps to 1of 4 loops with boration from BIT (1 multi-train system)
Main steam isolation (ISOL)	3 of 4 MSIVs shut (1 train)
Auxiliary Feed Water (AFW)	2/ 2 MDPs (1 train) or 1/ 1 TDP (1 train)
Feed and Bleed (FB)	Operator opens 2/3 PORVs and 1/ 2 SI and 1/ 2 CC pumps (operator action)
High Pressure Recirculation (HPR)	1/ 2 RHR trains supplying 1/ 2 SI pumps and 1/ 2CC 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 (5)			
4 MSLB - CHG (6)			

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

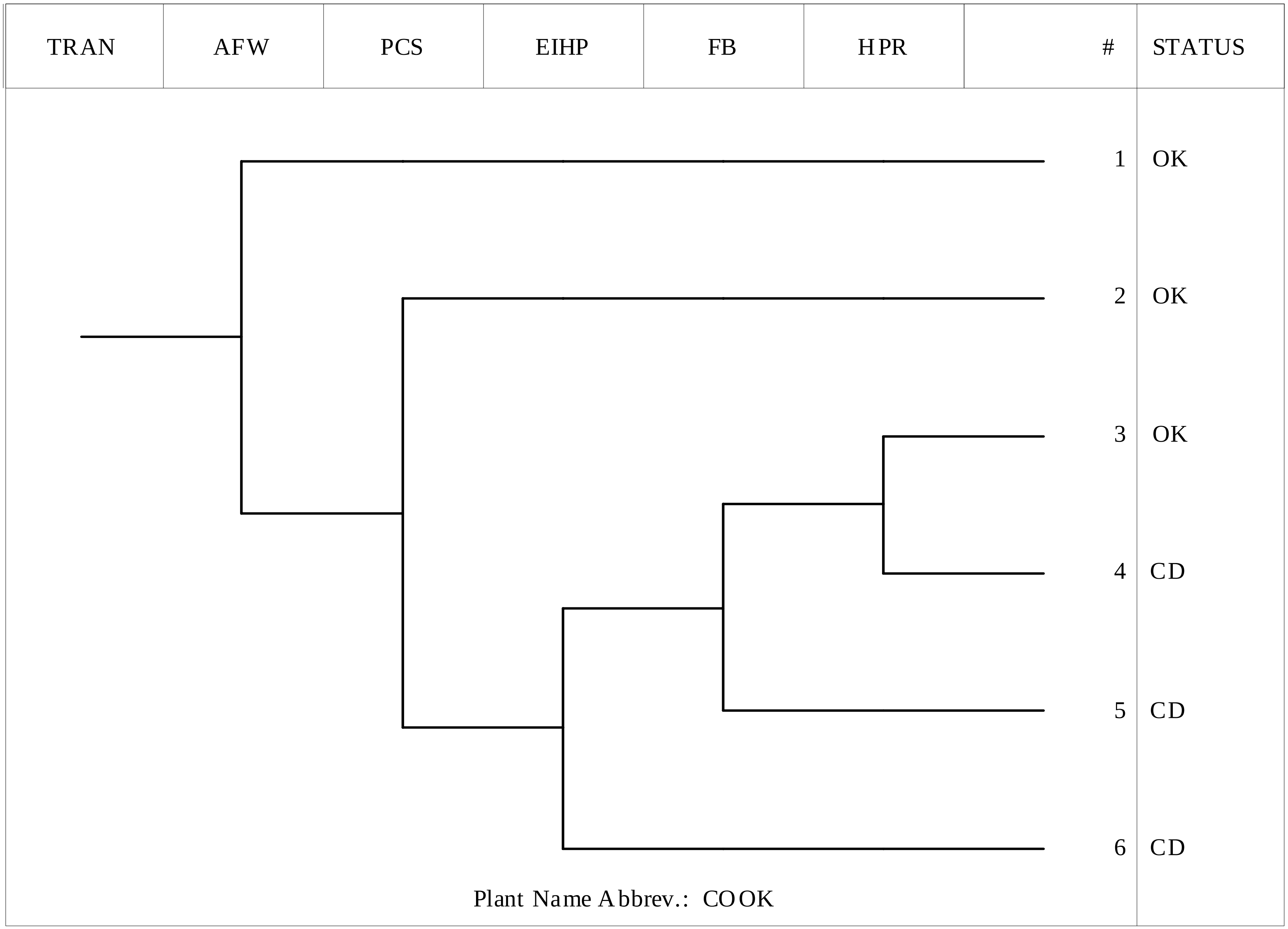
If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures 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.

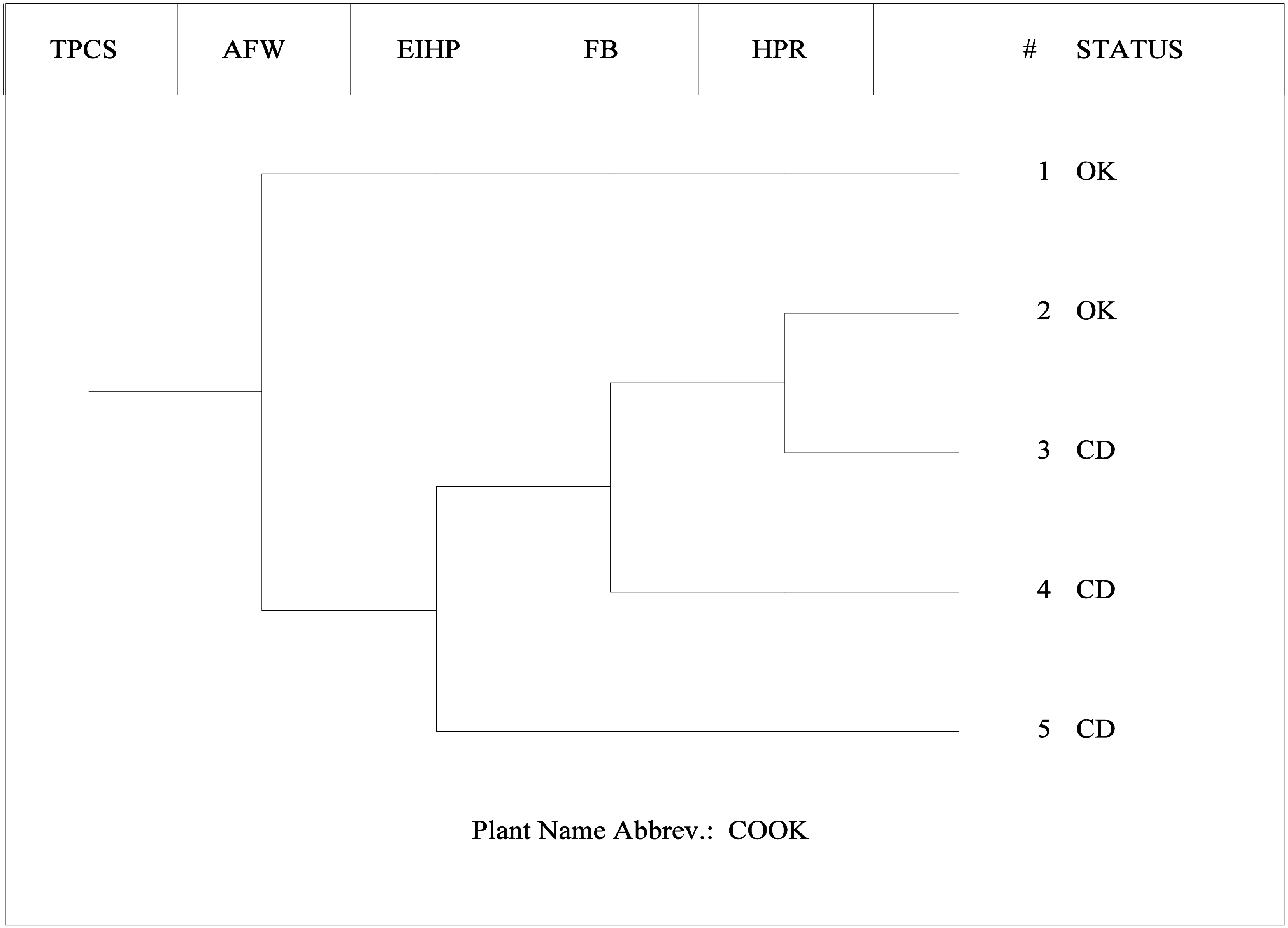
1.3 SDP EVENT TREES

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.

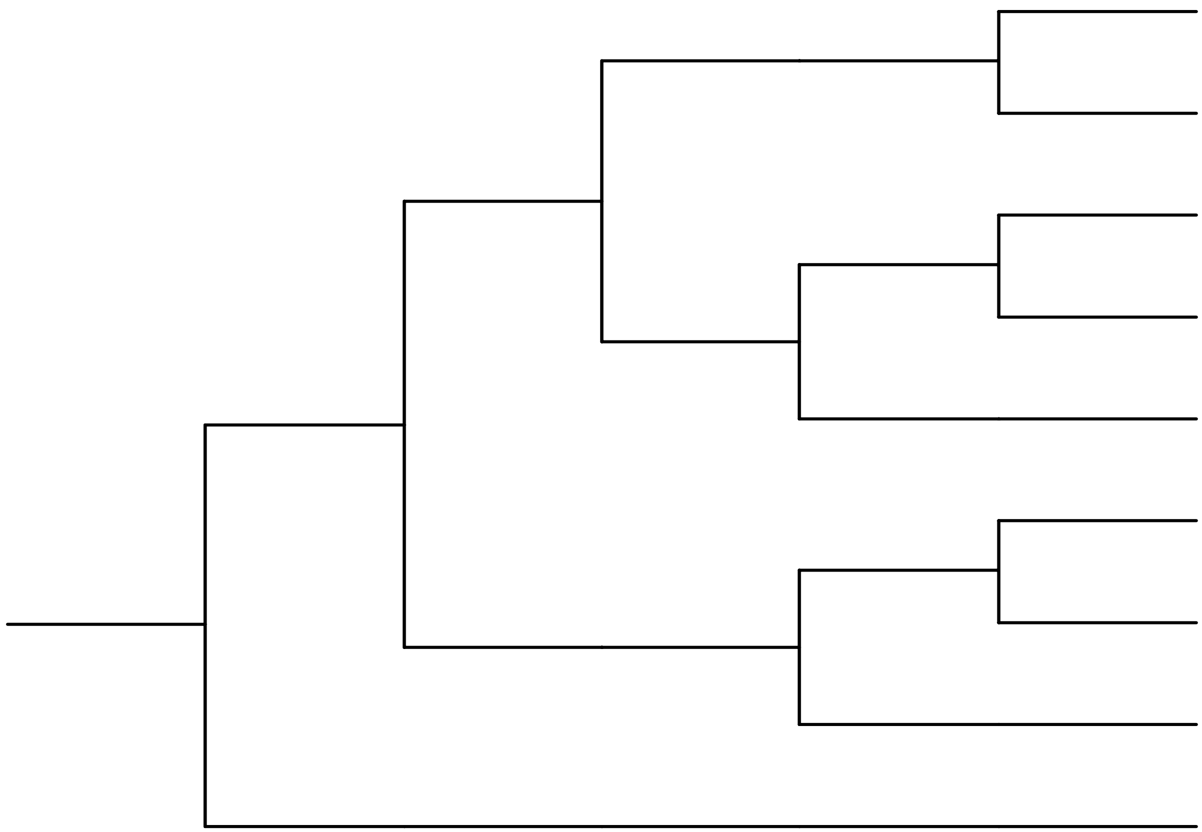
The following event trees are included:

1. Transients
2. Small LOCA
3. Medium LOCA
4. Large LOCA
5. LOOP
6. Steam Generator Tube Rupture (SGTR)
7. Anticipated Transients Without Scram (ATWS)


TRAN
AFW
PCS
EIHP
FB
HPR
#
STATUS
1 OK
2 OK
3 OK
4 CD
5 CD
6 CD
Plant Name Abbrev.: COOK

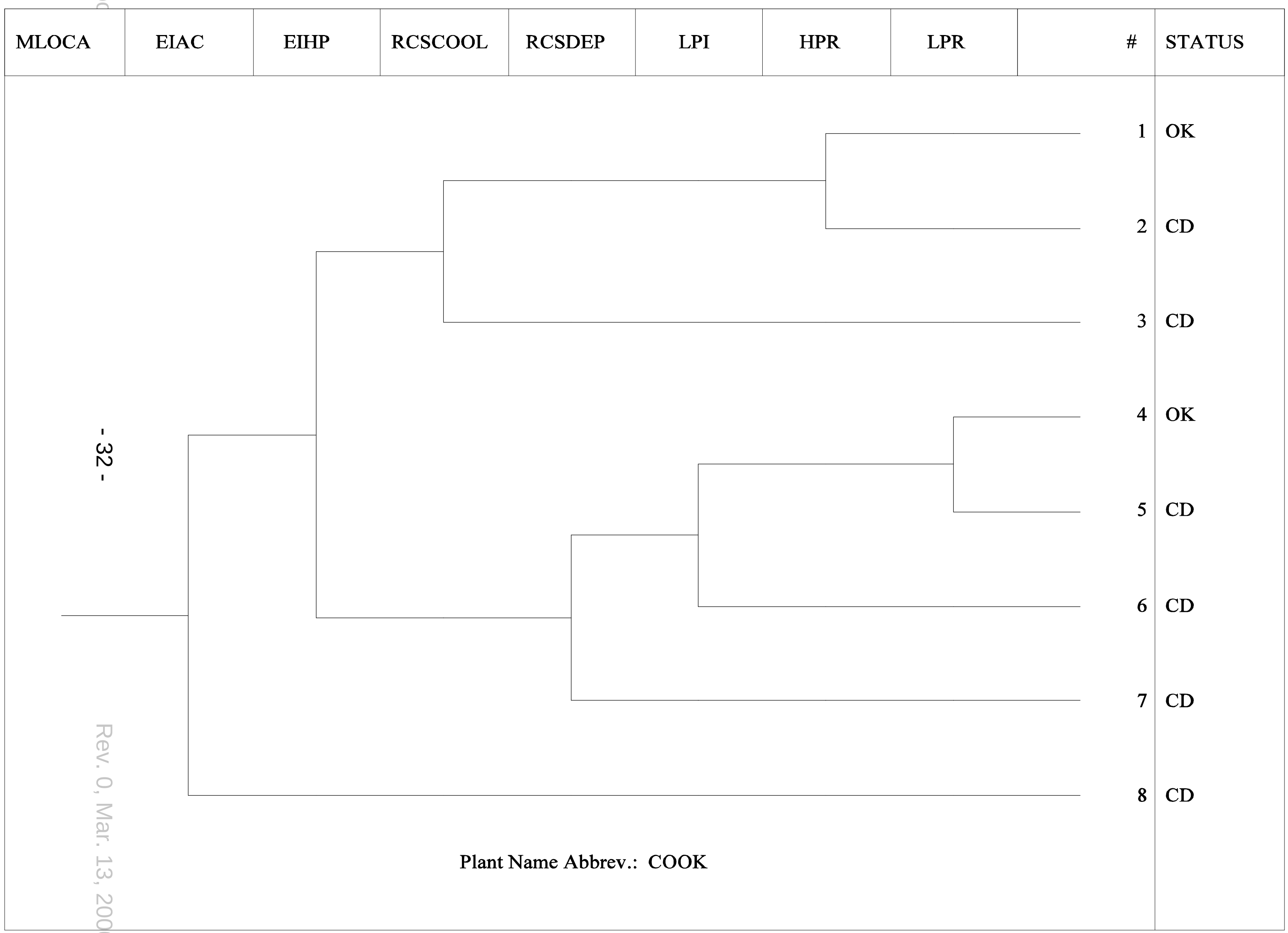

TPCS
AFW
EIHP
FB
HPR
#
STATUS
1 OK
2 OK
3 CD
4 CD
5 CD
Plant Name Abbrev.: COOK

SLOCA	EIHP	AFW	RCSCOOL	FB	HPR	#	STATUS

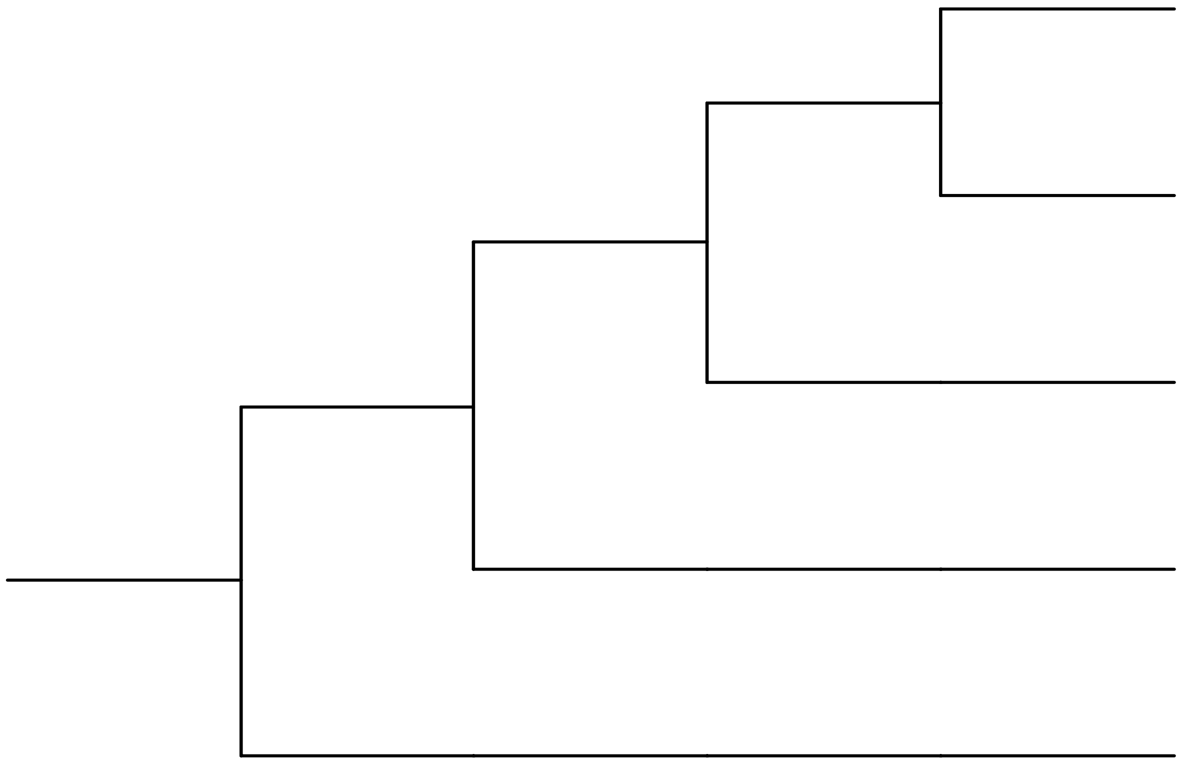


#	STATUS
1	OK
2	CD
3	OK
4	CD
5	CD
6	OK
7	CD
8	CD
9	CD

Plant Name Abbrev.: COOK

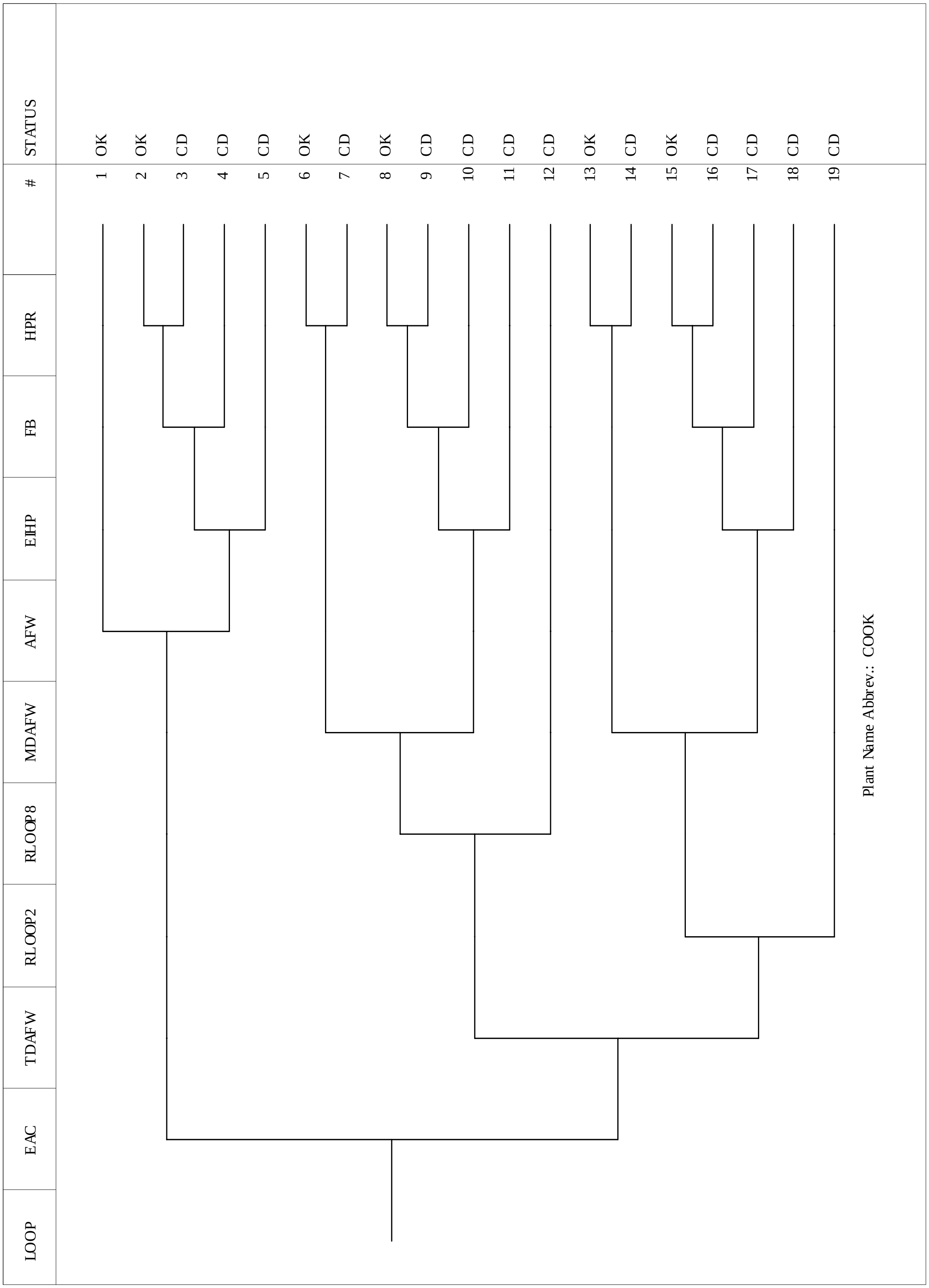

MLOCA
EIAC
EIHP
RCSCOOL
RCSDEP
LPI
HPR
LPR
#
STATUS
1 OK
2 CD
3 CD
4 OK
5 CD
6 CD
7 CD
8 CD
Plant Name Abbrev.: COOK

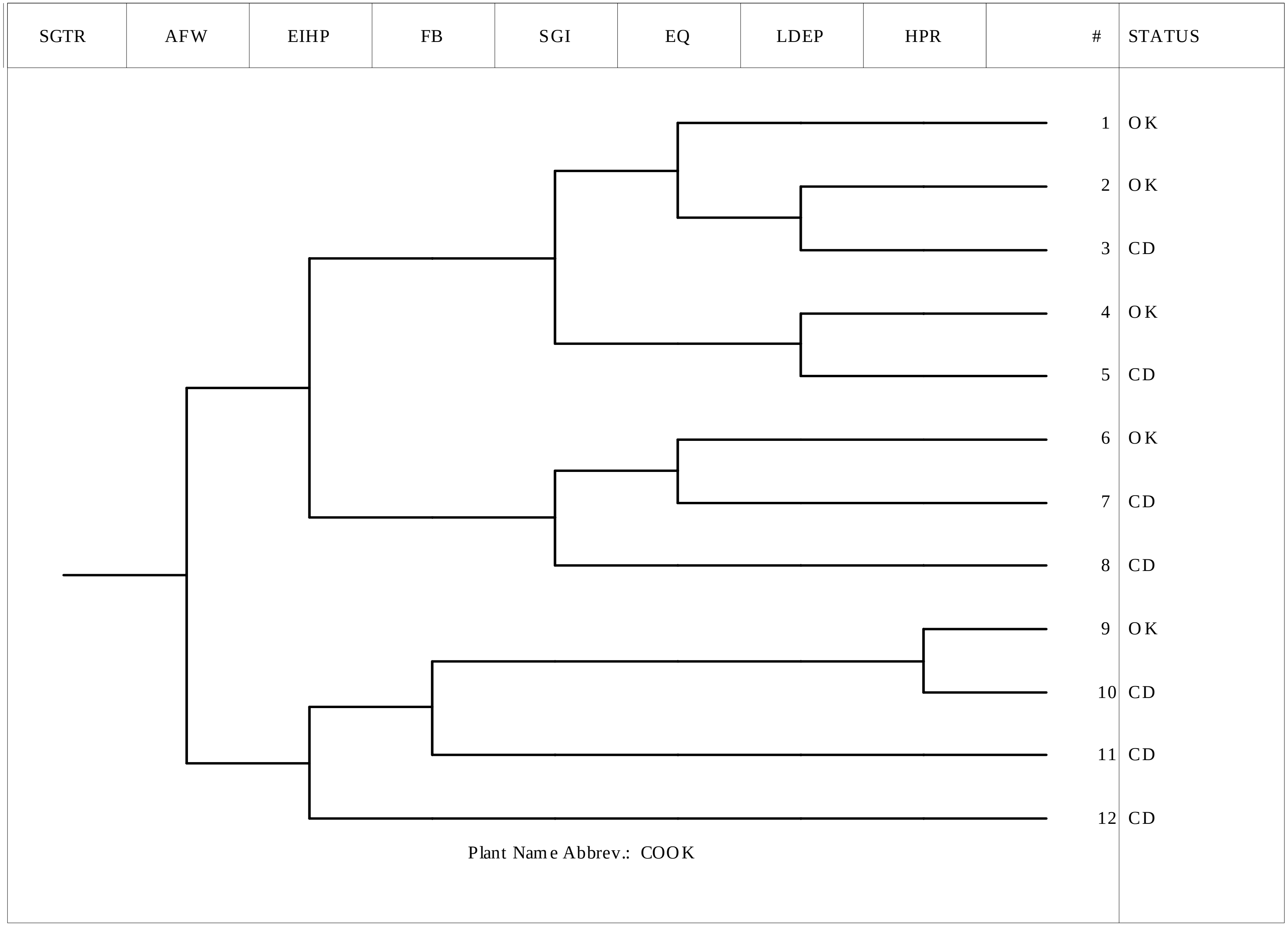
LLOCA	ACC	EILP	CNT	LPR	#	STATUS



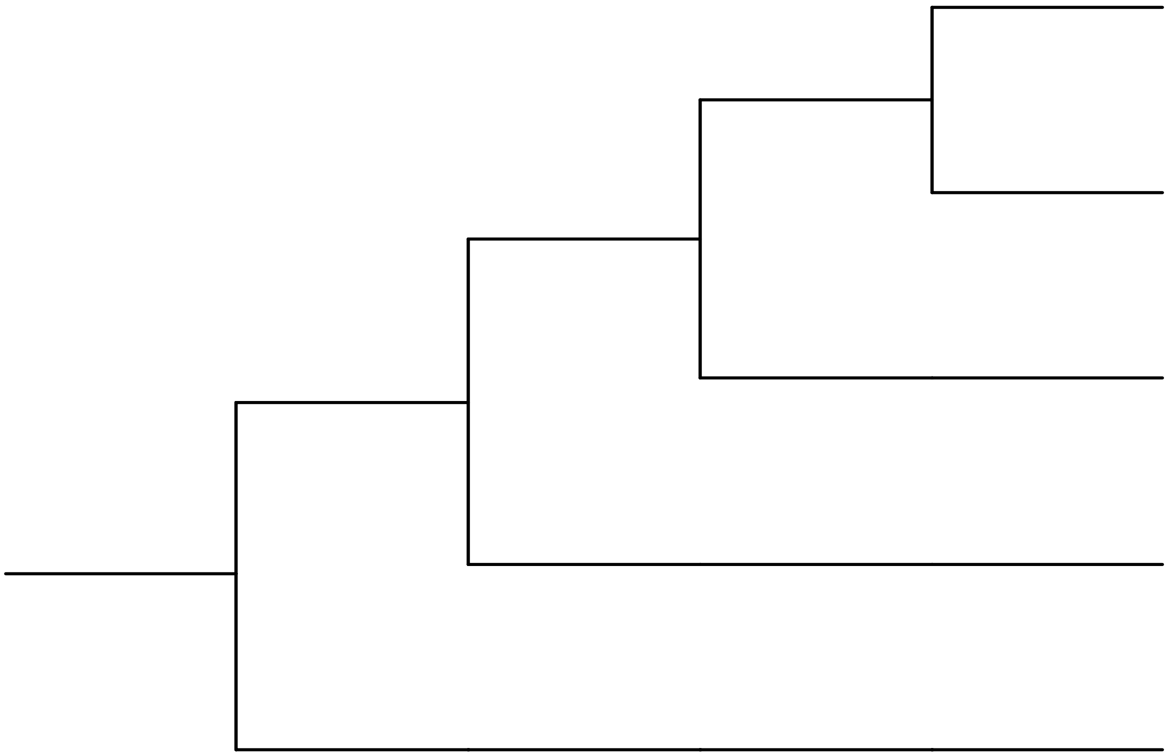
#	STATUS
1	OK
2	CD
3	CD
4	CD
5	CD

Plant Name Abbrev.: COOK


STATUS
#
HPR
FB
EIHP
AFW
MDAFW
RLOOP8
RLOOP2
TDAFW
EAC
LOOP
1 OK
2 OK
3 CD
4 CD
5 CD
6 OK
7 CD
8 OK
9 CD
10 CD
11 CD
12 CD
13 OK
14 CD
15 OK
16 CD
17 CD
18 CD
19 CD
Plant Name Abbrev.: COOK

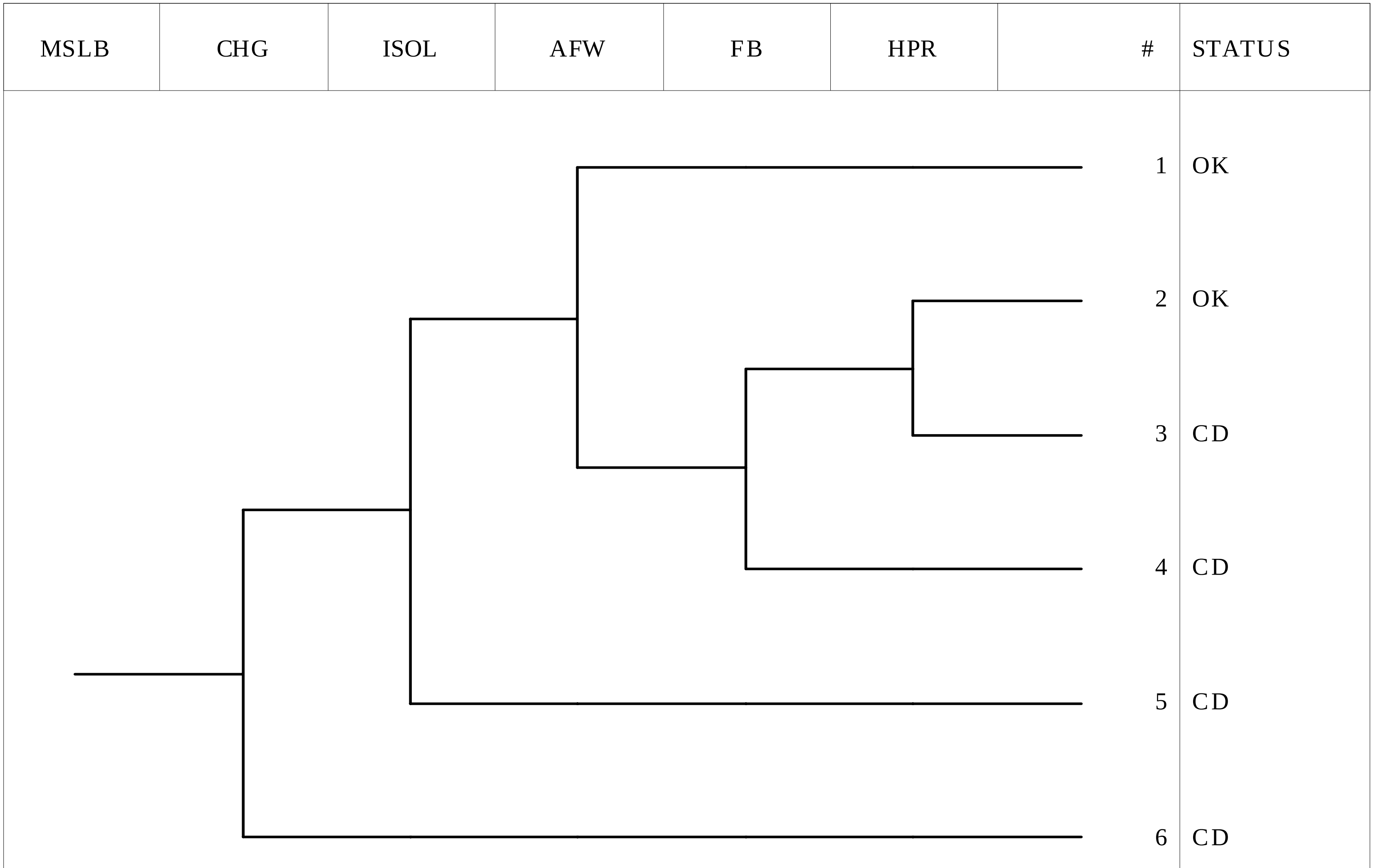

SGTR
AFW
EIHP
FB
SGI
EQ
LDEP
HPR
#
STATUS
1 OK
2 OK
3 CD
4 OK
5 CD
6 OK
7 CD
8 CD
9 OK
10 CD
11 CD
12 CD
Plant Name Abbrev.: COOK

ATWS	AMSAC	AFW	PPR	LTS	#	STATUS



#	STATUS
1	OK
2	CD
3	CD
4	CD
5	CD

Plant Name Abbrev.: COOK



Plant Name Abbrev.: COOK

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. American Electric Power Service Corporation, "Donald C. Cook Nuclear Plant, Units 1 and 2 Individual Plant Examination Submittal Report," April 1992.