February 14, 2000

Mr. Michael F. Hammer
Site General Manager
Monticello Nuclear Generating Plant
2807 West County Road 75
Monticello, MN 55362-9637

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

Dear Mr. Hammer:

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 Monticello Nuclear Generating Plant in April 2000. Included in the enclosed draft 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 301-415-2296.

Sincerely,

/RA/

Carl F. Lyon, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-263

Enclosure: Draft Risk-Informed Inspection Notebook

cc w/o encl: See next page

cc w/encl: Plant Manager
Monticello Nuclear Generating Plant
ATTN: Site Licensing
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2807 West Country Road 75
Monticello, MN 55362-9637

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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 301-415-2296.

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RISK-INFORMED INSPECTION NOTEBOOK FOR

MONTICELLO NUCLEAR GENERATING PLANT

BWR-3, GE, WITH MARK I CONTAINMENT

Prepared by

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Prepared for

U. S. Nuclear Regulatory Commission
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January 2000

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
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* **ABSTRACT**

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

1 Initiators and System Dependency Table for Monticello

Affected System	Major Components	Support Systems	Initiating Event Scenarios
Safety Relief Valves (SRVs) / Automatic Depressurization System (ADS) - {Reactor Vessel Pressure Relief System}	Five Relief Valves, Three Air-operated Valves	120 V-AC power, 125 V-DC power, Instrument Air/Nitrogen, ECCS Initiation Logic	All but LLOCA (see Note 4)
Power Conversion System (PCS)			TRANS, SLOCA, MLOCA, ATWS
Main Steam System	Four MSIVs, Four Main Steam Lines, Turbine Bypass Valves	MSIV Remain Open: FW, HPCI, RCIC, CRD, Condensate, LPCI, Core Spray, RHRSW, Condensate SW (Keep-Fill), Main Condenser, 120 V-AC, 125 V-DC, Service Water, Instrument Air/Nitrogen, Steam Tunnel Cooling	
Feedwater (FW) System	Two MD Pumps, MOVs, Two FW Reg. Valves, Startup Reg. Valve	Offsite Power, Condensate, Open MSIV, Main Condenser, 4160 V-AC (BOP), 480 V-AC (BOP), 120 V-AC (BOP), 125 V-DC (BOP), RBCCW, Instrument Air	
Condensate System	MD Pumps, Heat Exchangers, MOVs, Main Condenser	Offsite Power, MSIV Open, Main Condenser, 4160 V-AC (BOP), 480 V-AC (BOP), 125 V-DC (BOP), Instrument Air, HVAC; for Main Condenser, also: Service Water, Steam Jet Air Ejector, Mechanical Vacuum Pump, Circulating Water Pumps, Steam Seal System, Bypass Valve Control, H2 Recombiner	
High Pressure Coolant Injection (HPCI)	One ASD Pump, MOVs	SRVs/ADS (to remain closed), Torus Cooling, Torus Drywell Sprays, Containment Venting, **(HVAC 480 V-AC)**, 120 V-AC, 250 V-DC, 125 V-DC, Service Water, ESW, Instrument Air, **[HPCI Room Cooling (HVAC)]**, ECCS Initiation Logic (SEE NOTE 5)	TRANS, SLOCA, MLOCA, LOOP, ATWS

Table 1 (Continued)

Affected System	Major Components	Support Systems	Initiating Event Scenarios
Reactor Core Isolation Cooling (RCIC)	One ASD Pump, MOVs	SRVs/ADS (to remain closed), Torus Cooling, Torus Drywell Sprays, Containment Venting, **[HVAC 480 V-AC]**, 120 V-AC, 250 V-DC, 125 V-DC, Service Water, ESW, **Instrument Air**, **RCIC Room Cooling (HVAC)**, ECCS Initiation Logic	TRANS, SLOCA, LOOP, ATWS
Control Rod Drive (CRD)	Two MD Pumps, MOVs	Condensate, Containment Venting, 4160 V-AC (BOP), 125 V-DC, Service Water, RBCCW, Instrument Air	TRANS
Residual Heat Removal (RHR) / Low Pressure Coolant Injection (LPCI)	Two Loops (two MD RHR pumps and one Heat Exchanger per loop), MOVs	MSIV Open, Main Condenser, Torus Cooling, Torus/Drywell Sprays, 4160 V-AC, 480 V-AC, 125 V-DC, RHR Service Water (RHRSW), **Service Water, RBCCW**, **ESW, Condensate SW (Keep-Fill), Instrument Air, RHR Room Cooling (HVAC)**, ECCS Initiation Logic	All
Low Pressure Core Spray (LPCS)	Two MD Pumps, MOVs	MSIV Open, Main Condenser, Torus Cooling, Torus/Drywell Sprays, 4160 V-AC, 480 V-AC, 125 V-DC, RHR Service Water, **Service Water, RBCCW, ESW, Condensate SW (Keep-Fill), RHR/CS Room Cooling (HVAC),** ECCS Initiation Logic	All
Residual Heat Removal Service Water (RHRSW)	Two Loops (2 MD Pumps per loop), MOVs	LPCI, MSIV Open, Main Condenser, Torus Cooling, Torus/Drywell Sprays, 4160 V-AC, 120 V-AC, 125 V-DC, **Service Water,** Instrument Air	All
Condensate Service Water (COND SW) "Keep-Fill"	Two MD Pumps, AOVs	CRD, SRV/ADS, LPCI, Core Spray, MSIV Open, Main Condenser, 480 V-AC, Instrument Air	Not credited in IPE PRA
Torus Cooling	Two Loops (two MD RHR Pumps and one Heat Exchanger per loop), MOVs	RHRSW, Torus/Drywell Sprays, 4160 V-AC, 480 V-AC, 125 V-DC	TRANS, SLOCA, MLOCA, LLOCA, ATWS
Torus/Drywell Sprays	RHR Pumps, SPC MOVs	Same as RHR System	Not credited in Level 1 PRA

Table 1 (Continued)

Affected System	Major Components	Support Systems	Initiating Event Scenarios
Drywell Coolers	Fans, Dampers	480 V-AC, 120 V-AC, Service Water, RBCCW	Not credited in Level 1 PRA
Containment Venting - Torus / Drywell Venting (See Note 6)	Two 18" Vent and Purge Lines, Duct work	120 V-AC, Service Water, Instrument Air	TRANS, SLOCA, MLOCA, LLOCA, LOOP
Onsite AC Power 4160 V-AC 480 V-AC 120 V-AC	Two EDGs, 4160 V-AC Buses, 480 V-AC Load Centers	EDG fuel oil, Lube Oil, EDG-ESW, ECCS Initiation Logic	LOOP
DC Power 250 V-DC 125 V-DC	Two Divisions 250 V-DC, Two Divisions 125 V-DC	250 V-DC Batteries, 125 V-DC Batteries	All
Service Water	Three MD pumps, AOVs	AC Power, DC Power, Instrument Air	LOOP
Reactor Building Closed Cooling Water (RBCCW)	MD pumps, Three Heat Exchangers	AC Power, DC Power	See Note 7
Emergency Service Water (ESW)	Two MD Pumps, Manual Valves	AC Power, DC Power	See Note 8
EDG Emergency Service Water (EDG-ESW)	Two MD Pumps, Manual Valves, AOVs, Spool Piece Cross-tie	AC Power, DC Power	LOOP
Instrument Air and Nitrogen	Three MD Compressors, Air Supply Header, Liquid Nitrogen Tank, Control Valves	AC Power, DC Power, Service Water	All
Room Cooling	Fans, Cooling Units	AC Power, DC Power, Service Water	See Note 9
Emergency Core Cooling System (ECCS) Initiation Logic	Instrumentation, Transmitters, Logic Circuits, Control Circuits	AC Power, DC Power	All

Table 1 (Continued)

Affected System	Major Components	Support Systems	Initiating Event Scenarios
Recirculation Pump Trip (RPT)	Logic Circuits, Transmitters	Alternate Rod Injection (ARI), 250 V-DC, 125 V-DC	ATWS
Standby Liquid Control (SLC)	Two MD Pumps, Explosive Valves	480 V-AC	ATWS

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 Monticello Nuclear Generating Plant IPE Response to Generic Letter 88-20, submitted to the NRC by letter dated February 27, 1992.

(3) The original baseline IPE core damage frequency (CDF) from internal events is 1.9E-5 events/Rx year, excluding external flooding. With internal flooding, there is an increase of 0.7E-5 events/Rx year to a total of 2.6E-5 events/Rx year. (See MONT IPE Fig. 1.4-1).

(4) Because the emergency operating procedures direct manual control of reactor depressurization, inhibiting ADS and manually initiating SRVs is assumed for any event in which loss of high pressure injection leading to low reactor level occurs. The ability of the SRVs to open on reactor pressure above the SRV setpoints is not affected by operator actions to control individual valves manually. (See MONT IPE page 3.1-11).

(5) Although the systems identified in bold provide direct support to the particular affected system, for various reasons such as timing of the potential impact, the MONT IPE assumes that failure of these systems has no impact on the operation of the affected system. (See MONT IPE Tables 3.2-3, 3.2-4 and 3.2-5 and associated footnotes).

(6) Containment venting is a method of last resort which is initiated by actuating smaller 2" containment atmospheric system valves and progressively opening larger penetrations until containment pressure can be maintained below 56 psig, the design pressure. Venting is into the reactor building through ductwork which would burst open. EOPs instruct maintaining pressure below 56 psig as opposed to depressurizing the containment by venting. Probability of failure of containment venting is 5.0E-3. (See MONT IPE page 3.4-40).

(7) RBCCW provides support to RHR and CS pump shaft seals. No impact is assumed during 24 hour mission time in MONT IPE. (See MONT IPE page 3.2-33).

Table 1 (Continued)

(8) ESW provides support to RHR and CS pump room coolers. No impact is assumed on pump operation during 24 hour mission time in MONT IPE. (See MONT IPE page 3.2-32).

(9) MONT IPE assumes that Room Cooling is important only for steam tunnel cooling. A loss of steam tunnel cooling results in significant heatup and MSIV isolation. (See MONT IPE page 3.2-16).

1.2 SDP WORKSHEETS

This section presents the SDP worksheets to be used in the Phase 2 evaluation of the inspection findings for the Monticello Nuclear Generating Plant. 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 Monticello —— 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, one steam jet air ejector, one steam seal system, one circulating water pump, 1/2 condensate pumps, 1/2 main feed pumps (Operator Action)		
High Pressure Injection (HPI)	HPCI (one ASD train) or RCIC (one ASD train) or 1/ 2 CRD pumps (operator action)		
Depressurization (DEP)	2/8 safety relief valves (3 ADS SRVs and 5 SRVs) manually opened (high stress operator action)		
Low Pressure Injection (LPI)	1/4 RHR trains in LPCI Mode (one multi-train system) or 1/2 CS pumps (one multi-train system) or 1/2 condensate pumps (operator action) or [1/ 4 RHR SW pumps (two Loops - two pumps per Loop) and LPCI path](operator action) or 2/2 CRD pumps (operator action)		
Containment Heat Removal (CHR)	[1/4 RHR pumps in Suppression Pool Cooling (SPC) mode] (operator action) or 1/1 18" Containment Vent Line (operator action)		
Circle Affected Functions	**Recovery of Failed Train**	***Remaining* Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 TRANS - PCS - CHR (3, 5)			
2 TRANS - PCS - HPI - LPI (6)			
3 TRANS - PCS - HPI - DEP (7)			

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 RHR, HPCI, and RCIC minimum flow valves fail open on air loss. This will result in a 10% flow diversion to the suppression pool. A 10% flow diversion is assumed to have no impact on these systems. (See MONT IPE page 3.2-33).

(2) The MONT IPE assumes that Control Rod Drive makeup is adequate only is other high pressure systems have been capable of maintaining inventory early in the event. Accident sequences involving use of CRD for makeup include stuck open safety valves (in which HPCI is assumed to trip following depressurization of the reactor) and loss of decay heat removal sequences (in which HPCI and RCIC are assumed to be inadequate late in the event due to high suppression pool temperatures). CRD has a significant effect on the time available for operator action to depressurize the reactor and initiate low pressure injection should other high pressure systems be unavailable. (MONT IPE 3.2.1.8, page 3.2-6).

The inspector should use his or her own discretion, based on the operating history, condition, operator training and other factors, of the CRD system as to whether or not to allow any credit for use of the CRD pumps as an injection source.

Table 2.2 SDP Worksheet for Monticello —— 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:**
Power Conversion System (PCS)	1/4 steam lines, Condenser, one steam jet air ejector, one steam seal system, one circulating water pump, 1/2 condensate pumps, 1/2 main feed pumps (operator action)
Early Containment Control (EC)	Vapor suppression system - passive operation of suppression pool - 6/8 vacuum breakers remain closed (one multi-train system)
High Pressure Injection (HPI)	HPCI (one ASD train) or RCIC (one ASD train)
Depressurization (DEP)	2/8 safety relief valves manually opened (high stress operator action)[1]
Low Pressure Injection (LPI)	1/4 RHR trains in LPCI Mode (one multi-train system) or 1/2 CS trains (one multi-train system) or 1/2 condensate pumps (operator action) or [1/ 4 RHR SW pumps (two Loops - two pumps per Loop) and LPCI path] (operator action)
Containment Heat Removal (CHR)	[1/4 RHR pumps in Suppression Pool Cooling (SPC) mode] (operator action) or 1/1 18" Containment Vent Line (operator action)

Circle Affected Functions	**Recovery or Failed Train**	***Remaining* Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 SLOCA - PCS - CHR (3, 5)			
2 SLOCA - PCS - HPI - LPI (6)			
3 SLOCA - PCS - HPI - DEP (7)			

4 SLOCA - PCS - EC (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) The MONT IPE assumes that Control Rod Drive makeup is adequate only is other high pressure systems have been capable of maintaining inventory early in the event. Accident sequences involving use of CRD for makeup include stuck open safety valves (in which HPCI is assumed to trip following depressurization of the reactor) and loss of decay heat removal sequences (in which HPCI and RCIC are assumed to be inadequate late in the event due to high suppression pool temperatures). CRD has a significant effect on the time available for operator action to depressurize the reactor and initiate low pressure injection should other high pressure systems be unavailable. (MONT IPE 3.2.1.8, page 3.2-6).

The inspector should use his or her own discretion, based on the operating history, condition, operator training and other factors, of the CRD system as to whether or not to allow any credit for use of the CRD pumps as an injection source.

Table 2.3 SDP Worksheet for Monticello —— 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 Containment Control (EC)	Vapor Suppression System - passive operation of suppression pool - 6/8 vacuum breakers remain closed (one multi-train system)
High Pressure Injection (HPI)	HPCI (one ASD train) or [1/4 steam lines, Condenser, one steam jet air ejector, one steam seal system, one circulating water pump, 1/2 condensate pumps, 1/2 main feed pumps] (operator action)[1]
Depressurization (DEP)	2/8 safety relief valves manually opened (high stress operator action)
Low Pressure Injection (LPI)	1/4 RHR trains in LPCI mode (one multi-train system) or 1/2 CS pumps (one multi-train system) or [1/ 4 RHR SW pumps (two Loops - two pumps per Loop) and LPCI path] (operator action)
Containment Heat Removal (CHR)	[1/4 RHR pumps in Suppression Pool Cooling (SPC) mode] (operator action) or 1/1 18" Containment Vent Line (operator action)

Circle Affected Functions	**Recovery or Failed Train**	***Remaining* Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 MLOCA - CHR (2, 5)			
2 MLOCA - LPI (3, 6)			
3 MLOCA - HPI - DEP (7)			
4 MLOCA - EC (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.

<u>Note</u>:

(1) For a medium LOCA, the reactor will eventually depressurize due to the break. Therefore, successful operation of a low pressure pump will eventually be required. (See MONT IPE Table 3.1-2, page 3.1-27).

(2) The MONT IPE assumes that Control Rod Drive makeup is adequate only is other high pressure systems have been capable of maintaining inventory early in the event. Accident sequences involving use of CRD for makeup include stuck open safety valves (in which HPCI is assumed to trip following depressurization of the reactor) and loss of decay heat removal sequences (in which HPCI and RCIC are assumed to be inadequate late in the event due to high suppression pool temperatures). CRD has a significant effect on the time available for operator action to depressurize the reactor and initiate low pressure injection should other high pressure systems be unavailable. (MONT IPE 3.2.1.8, page 3.2-6).

The inspector should use his or her own discretion, based on the operating history, condition, operator training and other factors, of the CRD system as to whether or not to allow any credit for use of the CRD pumps as an injection source.

Table 2.4 SDP Worksheet for Monticello —— 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 Containment Control (EC)	Vapor Suppression system- passive operation of suppression pool - 6/8 vacuum breakers remain closed (one multi-train system)
Early Inventory (EI)	1/4 RHR trains in LPCI mode (one multi-train system) or 1/2 CS trains (one multi-train system) or [1/ 4 RHR SW pumps (two Loops - two pumps per Loop) and LPCI path] (operator action)
Containment Heat Removal (CHR)	[1/4 RHR pumps in Suppression Pool Cooling (SPC) mode] (operator action) or 1/1 18" Containment Vent Line (operator action)

Circle Affected Functions	**Recovery or Failed Train**	***Remaining* Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 LLOCA - CHR (2)			
2 LLOCA - LPI (3)			
3 LLOCA - EC (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 2.5 SDP Worksheet for Monticello —— 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 (one multi-train system) [1]
High Pressure Injection (HPI)	HPCI (one ASD train) or RCIC (one ASD train)[2]
Depressurization (DEP)	2/8 safety relief valves (3 ADS SRVs and 5 safety valves) manually opened (high stress operator action)[3,4, 5]
Recovery of LOOP in 2 hrs (RLOOP2)	High stress operator action[1]
Recovery of LOOP in 4 hrs (RLOOP4)	Operator action[1]
Recovery of LOOP in 6 hrs (RLOOP 6)	Operator action[1]

Circle Affected Functions	**Recovery or Failed Train**	***Remaining* Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 LOOP - CHR (1,2,4,6)			
2 LOOP - EAC - RLOOP6 (3)			
3 LOOP - HPI - LPI (4)			
4 LOOP - EAC - HPI - RLOOP4 (5)			

5 LOOP - HPI - DEP (6)			
6 LOOP - EAC - HPI - DEP -RLOOP2 (7)			

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) Sequences for which offsite power is not recovered but onsite power is available transfer from the LOOP event trees into one of two other event trees that are similar to the MSIV closure frontline system event tree. These sequences include any in which at least one train of emergency AC power is available and vessel water level is successfully maintained by either the HPCI/RCI systems or by low pressure systems following reactor depressurization. System availabilities applied to these event trees differs depending on whether one or two diesel generators are available. (See MONT IPE page 3.1-17).

(2) If core damage during LOOP is a result of random failure of HPCI and RCIC, core damage is assumed to occur at about 25 minutes after the initiating event with an intact containment. This is the same timing as for Class 1A Accidents: Loss of High Pressure Injection and Failure of Depressurization. (See MONT IPE page 3.4-8).

(3) If the DC power system batteries become unavailable, it is assumed that the HPCI and RCIC pumps become unavailable. No credit is taken for battery replacement with other charged batteries. The batteries are assumed to last 4 hours. They may be able to last longer with actions such as load shedding, but no credit was taken in the PRA for load shedding. (See MONT IPE page 3.4-8).

(4) If the reactor is at pressure when battery depletion occurs, core damage is assumed 2 hours after failure of high pressure injection systems. This is longer than the 30 minutes assumed for core damage during other transients due to the lower decay heat load. (See MONT IPE page 3.4-8).

(5) Because the SRV pneumatic supply valves are supplied by AC power, it is assumed that vessel depressurization is possible with accumulators only for a brief period of time during a SBO (1 hour). Core melt is assumed to occur at high reactor pressure at 6 hours with an intact containment where HPCI or RCIC are successful. (See MONT IPE page 3.4-8).

(6) In the response to the NRC's request for additional information, the licensee indicated that the MONT IPE LOOP event trees were developed to be flexible, so they could be used if the plant were modified to improve the capabilities of reactor depressurization and coolant makeup during a station blackout. The Rev. 0 IPE analysis assumes that failure always occurs for functions Q (HPI), X (DEP), and V (LPI) during phase III of a SBO. Branches of the LOOP event trees that indicate a transfer to Phase IV are not used and the event, Phase IV, is not developed. (See Feb. 15, 1993 RAI, F.E. 17, page 19).

Table 2.6 SDP Worksheet for Monticello —— 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)	7/8 SRVs and safety valves must open (one multi-train system)[1]
Recirculation Pump Trip (RPT)	Manual or automatic trip of 1/ 2 recirculation pumps (one multi-train system)[1]
Inhibit ADS (INH)	High stress operator action[2]
High Pressure Injection (HPI)	[HPCI (one ASD train) or RCIC (one ASD train)] and RPV level control (high stress operator action)
Reactivity Control (SLC)	1/ 2 SLC pumps (high stress operator action)[3, 4]
Depressurization (DEP)	7/8 ADS SRVs manually opened (high stress operator action)
Low Pressure Injection (LPI)	1/ 4 RHR trains in LPCI mode (one multi-train system)
Containment Heat Removal (CHR)	[1/4 RHR pumps in Suppression Pool Cooling (SPC) mode] (operator action) or 1/1 18" Containment Vent Line (operator action)

Circle Affected Functions	**Recovery or Failed Train**	***Remaining* Mitigation Capability Rating for Each Affected Sequence:**	**Sequence Color**
1 ATWS - CHR (2,5)			
2 ATWS - SLC (3,7)			
3 ATWS - HPI - LPI (6)			
4 ATWS - HPI - DEP (8)			

5 ATWS - INH (9)			
5 ATWS - RPT (10)			
6 ATWS - OVERP (11)			
Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event: If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.			

Notes:

(1) In the event of failure to scram from 100% power, the ultimate capacity of the containment in the drywell is assumed to be reached within one hour. Containment failure is assumed to fail all reactor building injection systems resulting in core damage. Containment venting or RHR system operation is assumed to be inadequate for containment heat removal if the reactor is not shut down. (See MONT IPE pages 3.4-23 and 24).

(2) The MONT IPE discussion for ATWS does not directly indicate that the operators inhibit ADS during an ATWS. However, based on other discussions that Class 1A - Loss of High Pressure Injection and Failure to Depressurize accidents can be attributed to failure of operator action to depressurize the reactor, that the EOPs instruct ADS inhibit to allow time for recovery of high pressure systems, permit low volume high pressure systems to recover level slowly, and to permit the maximum time possible to assuring low pressure systems are aligned and operating, that the benefits of ADS inhibit and the importance of depressurization have been recommended for reinforcement in operator training, that the operators have also been trained on the impact of feedwater system recovery on reducing the risk of this damage class,

and that ADS inhibit is modeled in the ATWS scenarios for similar BWRs by other licensees, for the purposes of this SDP document, failure of ADS inhibit has been assumed to lead to core damage during an ATWS sequence at MONT. (See MONT IPE page 6-4).

(3) The probability of failure to initiate SLC without the main condenser available is assumed to be 4.0E-2. With the main condenser available, the probability is assumed to 5.0E-3. (See MONT IPE Table 3.3-3, page 3.3-23).

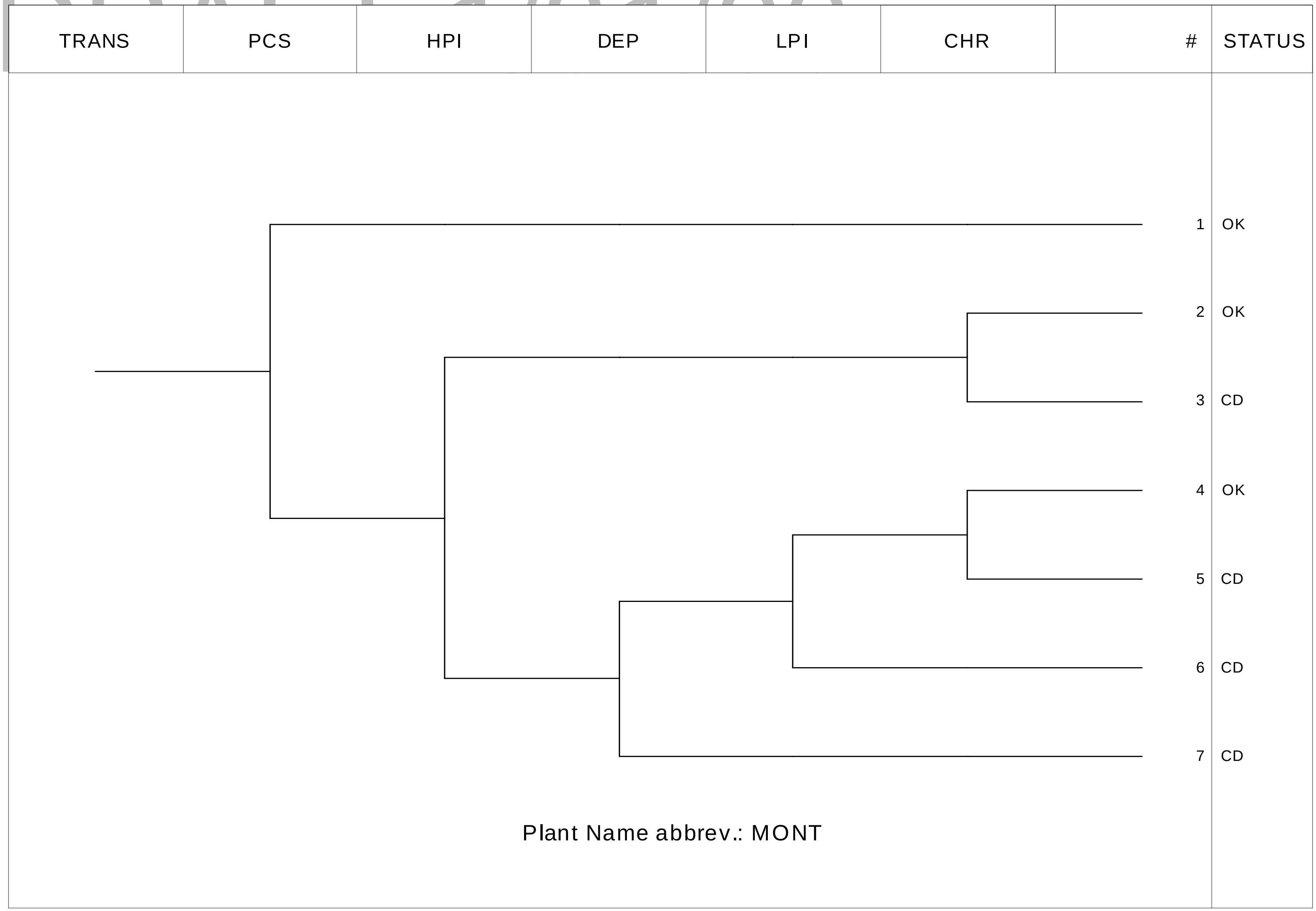
(4) Performing level/power control and directing all steam to the main condenser allows for alternate boron injection in the event that both SLC pumps fail. (See MONT IPE Table 3.1-3, page 3.1-28).

1.3 SDP Event Trees

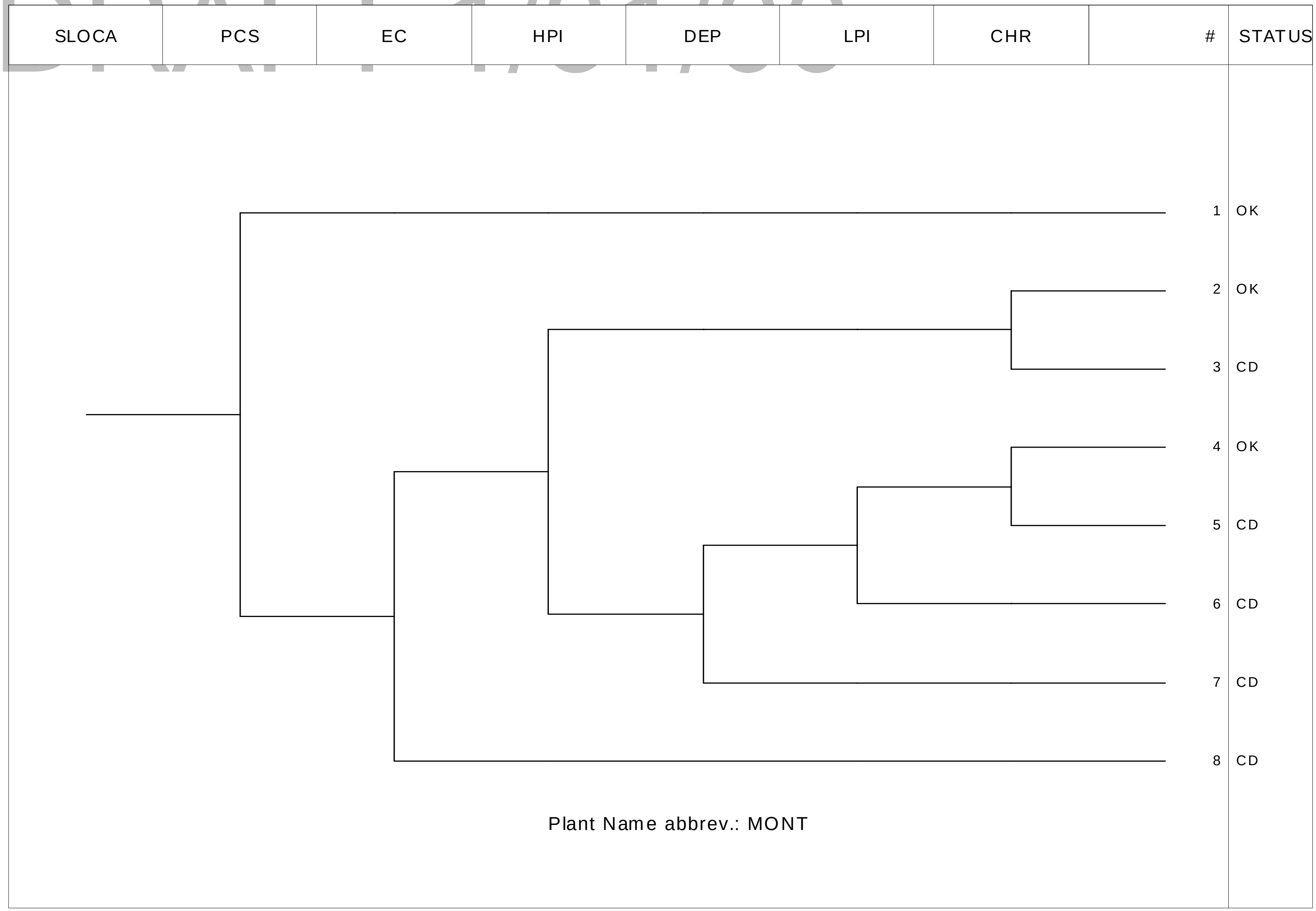
This section provides the simplified event trees, called SDP event trees, used to define the accident sequences identified in the SDP worksheets in the previous section. The event tree headings are defined in the corresponding SDP worksheets.

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.: MONT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PCS
EC
HPI
DEP
LPI
CHR
#
STATUS
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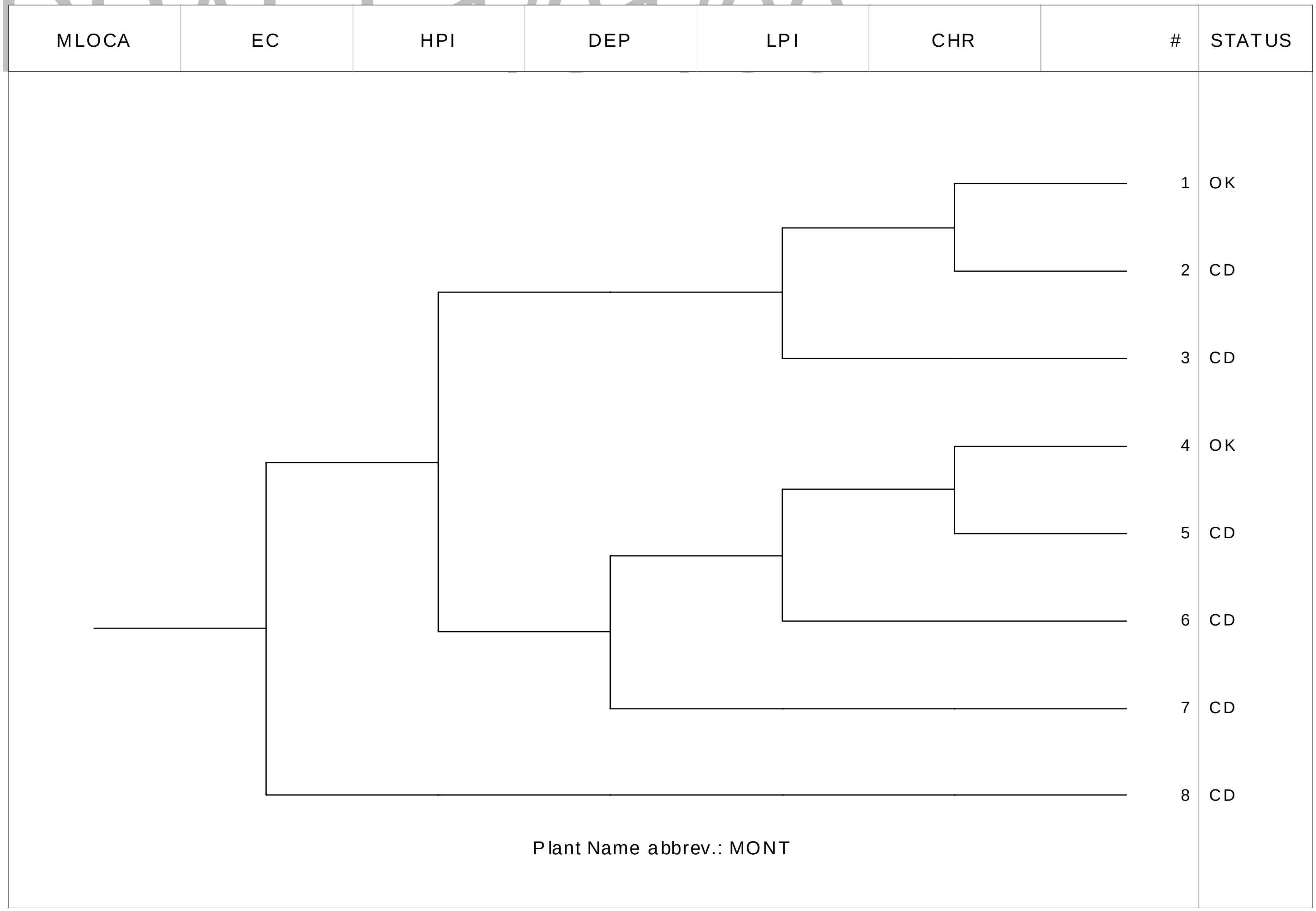
2 OK
3 CD
4 OK
5 CD
6 CD
7 CD
8 CD
Plant Name abbrev.: MONT


MLOCA
EC
HPI
DEP
LPI
CHR
#
STATUS
1 OK
2 CD
3 CD
4 OK
5 CD
6 CD
7 CD
8 CD
Plant Name abbrev.: MONT

LLOCA	EC	LPI	CHR	#	STATUS
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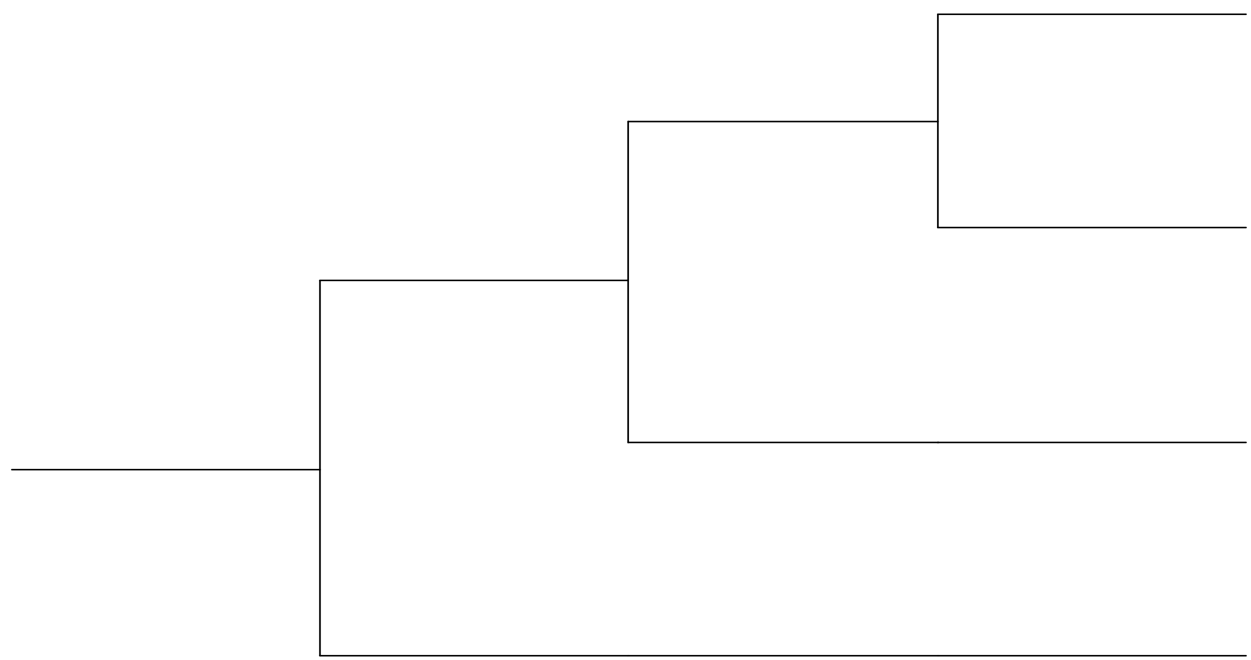
				2	CD
				3	CD
				4	CD



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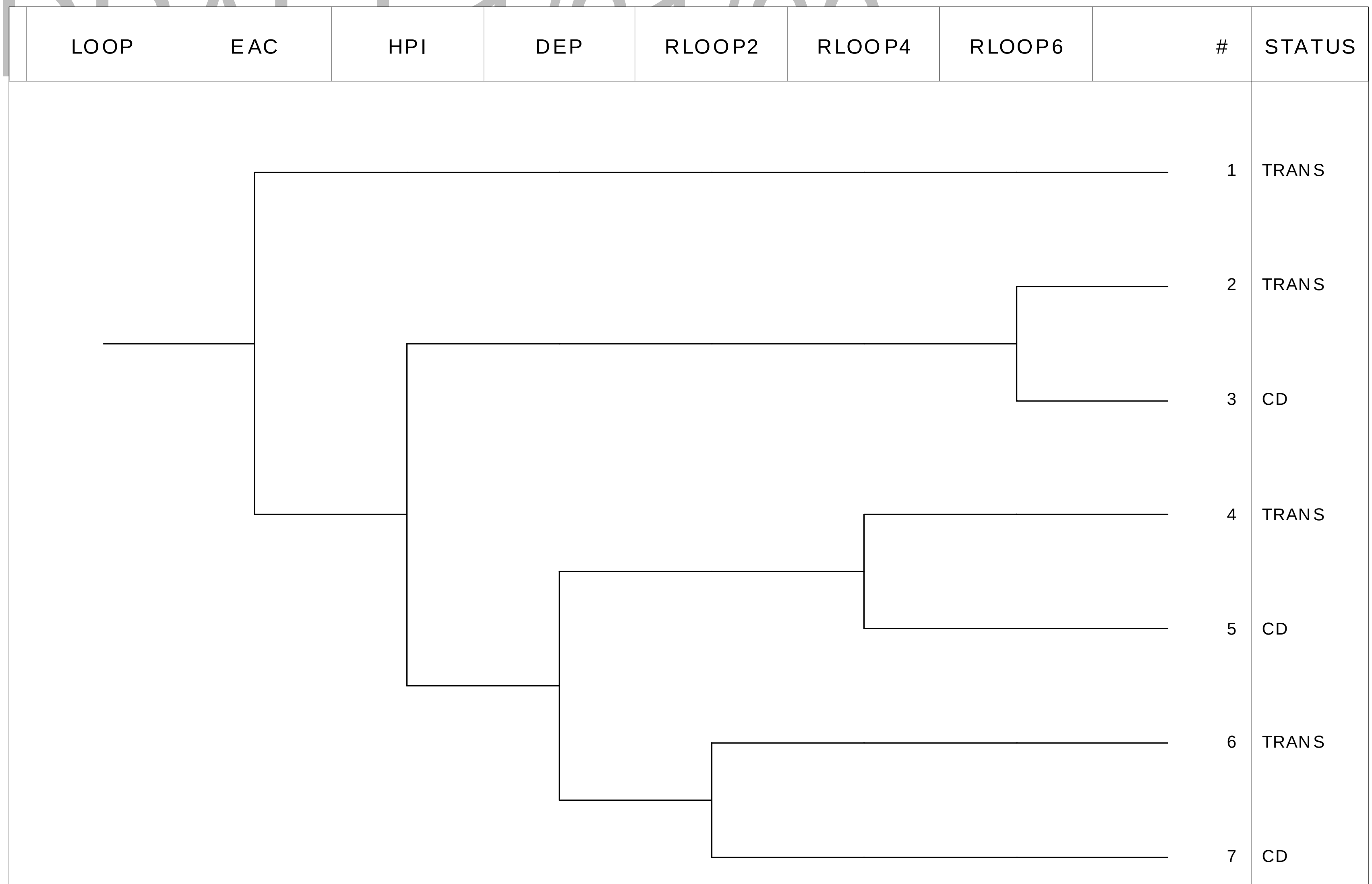

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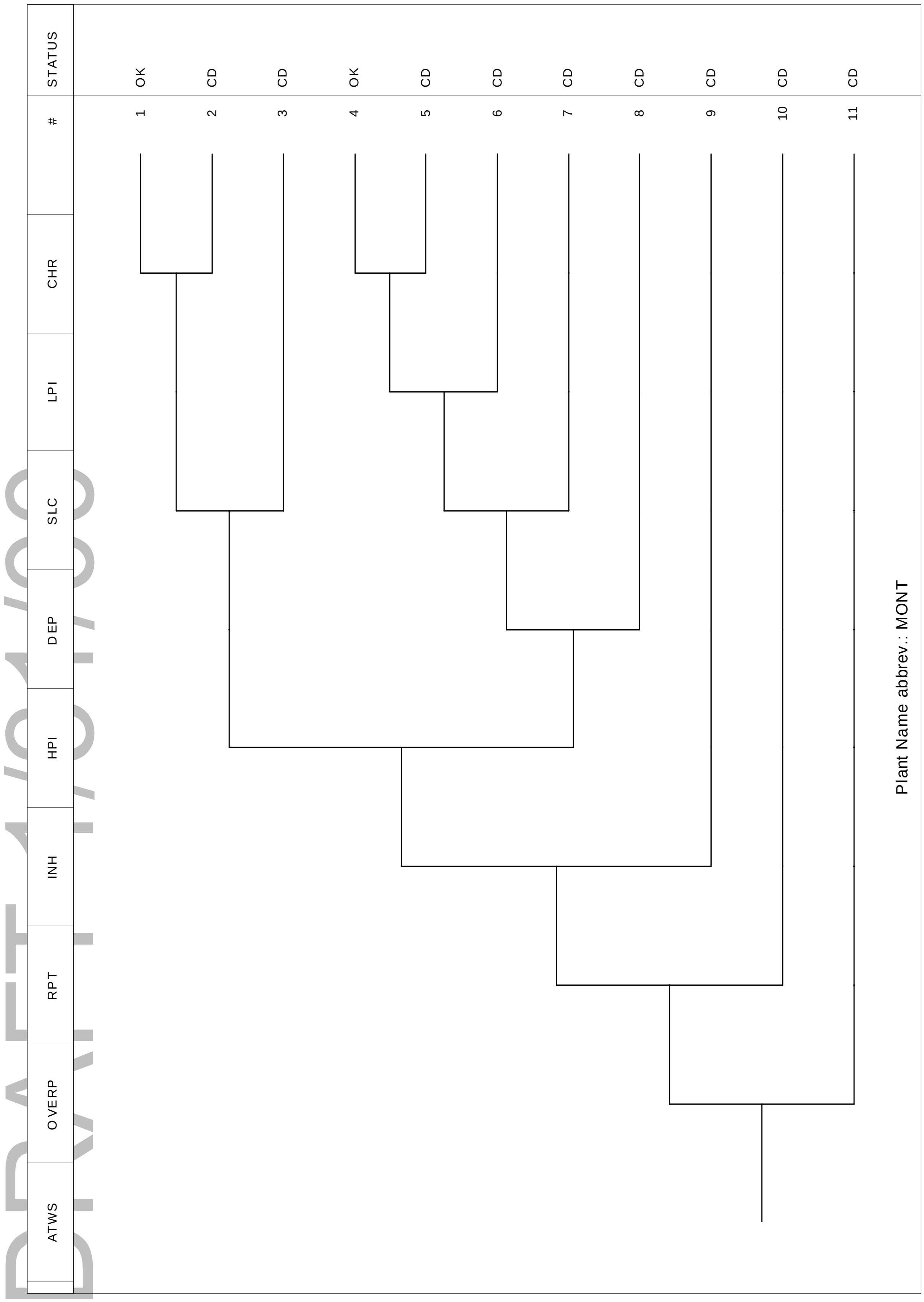
ATWS
OVERP
RPT
INH
HPI
DEP
SLC
LPI
CHR
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STATUS
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2 CD
3 CD
4 OK
5 CD
6 CD
7 CD
8 CD
9 CD
10 CD
11 CD
Plant Name abbrev.: MONT

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. Northern States Power Company, "Monticello Nuclear Generating Plant Individual Plant Examination Report," February 1992.