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

March 28, 2000

Dr. Robert C. Mecredy
Vice President, Nuclear Operations
Rochester Gas and Electric Corporation
89 East Avenue
Rochester, NY 14649

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

Dear Dr. Mecredy:

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 R. E. Ginna Nuclear Power Plant in April 2000. Included in the enclosed Risk-Informed Inspection Notebooks are the Significance Determination Process (SDP) worksheets that inspectors will be using to risk-characterize inspection findings. The SDP is discussed in more detail below.

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

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

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

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

We will coordinate our efforts through your licensing or risk organizations as appropriate. If you have any questions, please contact me at 301-415-1320.

Sincerely,

Guy S. Vissing, Sr. Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-244

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

We will coordinate our efforts through your licensing or risk organizations as appropriate. If you have any questions, please contact me at 301-415-1441.

Sincerely,

/RA original signed by H. Pastis for/

Guy S. Vissing, Sr. Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-244

Enclosure: As stated

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R.E. Ginna Nuclear Power Plant

Ho K. Nieh, Jr., Sr. Resident Inspector
R.E. Ginna Plant
U.S. Nuclear Regulatory Commission
1503 Lake Road
Ontario, NY 14519

Regional Administrator, Region I
U.S. Nuclear Regulatory Commission
475 Allendale Road
King of Prussia, PA 19406

Mr. F. William Valentino, President
New York State Energy, Research,
and Development Authority
Corporate Plaza West
286 Washington Avenue Extension
Albany, NY 12203-6399

Charles Donaldson, Esquire
Assistant Attorney General
New York Department of Law
120 Broadway
New York, NY 10271

Nicholas S. Reynolds
Winston & Strawn
1400 S Street N.W.
Washington, DC 20005-3502

Ms. Thelma Wideman, Director
Wayne County Emergency Management
Office
Wayne County Emergency Operations Center
7336 Route 31
Lyons, NY 14489

Ms. Mary Louise Meisenzahl
Administrator, Monroe County
Office of Emergency Preparedness
111 West Falls Road, Room 11
Rochester, NY 14620

Mr. Paul Eddy
New York State Department of
Public Service
3 Empire State Plaza, 10th Floor
Albany, NY 12223

RISK-INFORMED INSPECTION NOTEBOOK FOR

R. E. GINNA NUCLEAR POWER PLANT

PWR, WESTINGHOUSE, TWO-LOOP PLANT WITH LARGE DRY 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 R. E. Ginna Nuclear Power Plant

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

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

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TABLES

1. INFORMATION SUPPORTING SIGNIFICANCE DETERMINATION PROCESS (SDP)

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

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

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

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

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

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

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

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

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

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

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

The three sections that follow include the initiators and dependency table, SDP worksheets, and the SDP event-trees for the R. E. Ginna Nuclear Power Plant.

1.1 INITIATORS AND SYSTEM DEPENDENCY

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

Table 1 Initiators and System Dependency for R. E. Ginna Nuclear Power Plant

Affected Systems	Major Components	Support Systems	Initiating Event
AC Power System	AC Power Distribution and AC Instrument Power	DC	Transient, SLOCA, MLOCA, LLOCA, LOOP, SGTR, ATWS, RCP seal LOCA
AFW	Two MDPs	4160 V-AC, SW[1], ESFAS, HVAC	Transient, SLOCA, LOOP, SGTR, ATWS
	One TDP	DC, SW, ESFAS, Main Steam	
Standby AFW	Two MDPs	4160 V-AC, SW, HVAC	
CCW	Two pumps and two Heat Exchangers	480 V-AC, DC, ESFAS, SW	Transient, SLOCA, MLOCA, LLOCA, LOOP, SGTR, ATWS, RCP seal LOCA
Condensate / MFW	Two Condensate pumps Two MFW pumps	4.16 kV AC, DC, SW, IA[2]	Transient
Containment Spray (CS) System[3]	Two CS pumps	480 V-AC, DC, ESFAS, CCW,	Not used
Chemical and Volume Control System (CVCS)-Charging Pumps	Three Charging pumps[4]	480 V-AC, IA	
HPSI	Three SI pumps (2 trains)	4160 V-AC, 480 V-AC, 125 V-DC, SW, CCW, ESFAS	Transient, SLOCA, MLOCA, LOOP, SGTR, ATWS, RCP seal LOCA
DC Power	Batteries, battery chargers and Distribution Panels	480 V-AC, HVAC	Transient, SLOCA, MLOCA, LLOCA, LOOP, SGTR, ATWS, RCP seal LOCA
EDG	Two EDGs	DC, SW, ESFAS, HVAC, IA	LOOP
ESFAS	Two digital actuation trains and four analog instrumentation channels	120 V-AC instrument bus, DC	Transient, SLOCA, MLOCA, LLOCA, LOOP, SGTR, ATWS, RCP seal LOCA

Table 1 (Continued)

Affected Systems	Major Components	Support Systems	Initiating Event
Instrument Air (IA)	Three Air Compressors plus one service air compressor	480 V-AC, DC, SW	Transient, SLOCA, LOOP, SGTR, ATWS
Instrument Bus System	Instrument Buses A, B, C, and D	Inverter A, B for Buses A and C; 480V AC for Buses B and D	
Main Steam	Four Code safety valves and 1 ARV for each SG	DC, IA	SGTR
Pressurizer Pressure Relief	Two Safety valves and two PORVs with associated block valves	120 V-AC, DC, IA (PORVs)	Transient, SLOCA, LOOP, SGTR, ATWS
RCP	Seals	1 / 3 Charging pumps to seal injection or 1 / 2 CCW pumps to thermal barrier heat exchanger	LOOP, RCP seal LOCA
RHR	Two RHR pumps and heat exchangers	4.16 kV AC, 480 V-AC, 125 V-DC, RPS, CCW[5]	Transient, SLOCA, MLOCA, LLOCA, LOOP, SGTR
SW	Four pumps	4160 V-AC, 480 V-AC, 125 V-DC, IA	Transient, SLOCA, MLOCA, LLOCA, LOOP, SGTR, ATWS, RCP seal LOCA

Table 1 (Continued)

Notes:

(1) SW system provides a backup water supply to the AFW system.

(2) Feed water regulating valves are air-operated.

(3) Ventilation support of the CS pumps is not required during the post-accident conditions (IPE page 6-35)

(4) 2/3 Charging pumps are normally operating.

(5) CCW is supplied to the RHR pump seal heat exchangers and to the RHR heat exchangers. CCW supply to either set of heat exchangers is not required for the injection mode; however, CCW supply to both set of heat exchangers is required for recirculation and RHR operation.

(6) The plant internal event CDF is 5.02E-5/yr.

1.2 SDP WORKSHEETS

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

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

Table 2.1 SDP Worksheet for R. E. Ginna Nuclear Power Plant —— 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:**
Secondary Heat Removal (AFW)	1 / 2 MDAFW trains (1 multi-train system) or 1/1 TDAFW train (1 ASD train)
Standby AFW or Main Feed Water (SAFW/MFW)	Operator aligns 1/ 2 SAFW pumps or 1/ 2 MFW pumps (Operator action)(1)
Early Inventory, High Pressure Injection (EIHP)	1 / 2 HPSI pump trains (3 pumps) (1 multi-train system)
Primary Heat Removal, Feed/Bleed (FB)	1 / 2 PORVs open for Feed/Bleed (high stress operator action)(2)
High Pressure Recirculation (HPR)(3)	1 / 2 HPSI pump trains with 1 / 2 RHR pumps and 1/ 2 RHR HXs(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 - AFW - SAFW/MFW - HPR (4)			
2 TRANS - AFW - SAFW/MFW - FB (5)			
3 TRANS - AFW - SAFW/MFW - EIHP (6)			

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

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

Notes:

(1) Operator failure to align SAFW pump and operator failure to reestablish MFW are respectively assigned probabilities of 1.0E-1, and 1.2E-2.

(2) Operator failure to implement feed and bleed is assigned a probability of 5.3E-2 for single actions and 4.1E-01 for multiple actions. Accordingly, it is defined as a high stress operator action here.

(3) The success criteria in Page 4-42 of the IPE notes the need for high pressure recirculation in case of feed/bleed. The transient event tree does not consider this function. We have included it here considering the success criteria and based on our observation for other plants of similar design.

Table 2.2 SDP Worksheet for R. E. Ginna Nuclear Power Plant —— Transients (w/o PCS)

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

Safety Functions Needed:	***Full* Creditable Mitigation Capability for Each Safety Function:**
Secondary Heat Removal (AFW)	1 / 2 MDAFW trains (one multi-train system) or 1/1 TDAFW train (one ASD train)
Standby AFW (SAFW)	Operator aligns 1/ 2 SAFW pumps (Operator action)[1]
Early Inventory, High Pressure Injection (EIHP)	1 / 2 HPSI pump trains (one multi-train system)
Primary Heat Removal, Feed/Bleed (FB)	1 / 2 PORVs open for Feed/Bleed (operator action)[2]
High Pressure Recirculation (HPR)[3]	1 / 2 HPSI pumps with 1 / 2 RHR pumps (Requires operator action for switchover = operator action)

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

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

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

Notes:

(1) The operator failure to align SAFW is assigned a probability of 5E-3.

(2) Operator failure to implement feed and bleed is assigned a probability of 5.3E-2 for single actions and 4.1E-01 for multiple actions. Accordingly, it is defined as a high stress operator action here.

(3) The success criteria in Page 4-42 of the IPE notes the need for high pressure recirculation in case of feed/bleed. The transient event tree does not consider this function. We have included it here considering the success criteria and based on our observation of other plants of similar design.

Table 2.3 SDP Worksheet for R. E. Ginna Nuclear Power Plant —— Small LOCA

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

Safety Functions Needed:	***Full* Creditable Mitigation Capability for Each Safety Function:**
Early Inventory, HP Injection (EIHP)	1 / 2 HPSI pumps (one multi-train system).
Secondary Heat Removal (AFW)	1 / 2 MDAFW trains (one multi-train system) or one TDAFW train (one ASD train)
Standby AFW or MFW (SAFW/MFW)	Operator aligns 1/ 2 SAFW pumps or 1/ 2 MFW pumps (Operator action)
RCS Cooldown/ Depressurization (RCSDEP)	Operator depressurizes RCS using 2/2 PORV and SG (operator action)
Accumulators (ACC)	1/ 2 Accumulators (one multi-train system)
Low Pressure Injection (LPI)	1 / 2 RHR pumps (one multi-train system)
High Pressure Recirculation (HPR)	1 / 2 HPSI pumps with 1 / 2 RHR pumps (Requires operator action for switchover = operator action)
Low Pressure Recirculation (RHR)	1/ 2 RHR pumps taking suction from sump and discharging to vessel (operator action)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 SLOCA - RHR (3, 6, 10, 15)			
2 SLOCA - HPR - RHR (3, 6, 10, 15)			
3 SLOCA - HPR - RCSDEP (4,7)			
4 SLOCA -- HPR - AFW - SAFW/MFW (8)			
5 SLOCA - EIHP - LPI (11)			

6 SLOCA - EIHP - ACC (12, 17)			
7 SLOCA - EIHP - RCSDEP (13, 18)			
8 SLOCA - EIHP - AFW - SAFW/MFW(19)			
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) Since HPR and RHR both depend on RHR pumps, Sequence 1 above considers the hardware failure of RHR pump trains. Sequence 2 is kept to include the human failures relating to LPR and HPR functions.

Table 2.4 SDP Worksheet for R. E. Ginna Nuclear Power Plant —— Stuck Open PORV (SORV)

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

<u>Safety Functions Needed</u>:	***Full* Creditable Mitigation Capability for Each Safety Function**:
Early Inventory, HP Injection (EIHP)	1 / 2 HPSI pumps (one 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 (one multi-train system) or 1/1 TDAFW train (one ASD train)
Standby AFW or MFW (SAFW/MFW)	Operator aligns 1/ 2 SAFW pumps or 1/ 2 MFW pumps (Operator action)
RCS Cooldown/ Depressurization (RCSDEP)	Operator depressurizes RCS using 1/ 2 PORV and SG (operator action)
Accumulators (ACC)	1/ 2 Accumulators (one multi-train system)
Low Pressure Injection (LPI)	1 / 2 RHR pumps (one multi-train system)
High Pressure Recirculation (HPR)	1 / 2 HPSI pumps with 1 / 2 RHR pumps (Requires operator action for switchover = operator action)
Low Pressure Recirculation (RHR)	1/ 2 RHR pumps taking suction from sump and discharging to vessel (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 SORV - BLK - RHR (3, 6, 10, 15))			
2 SORV - BLK - HPR - RHR (3, 6, 10, 15))			
3 SORV - BLK - RCSDEP (4, 7)			
4 SORV - BLK - AFW - SAFW/MFW (8)			

5 SORV - BLK - EIHP - LPI (11)			
6 SORV - BLK - EIHP - ACC (12, 17)			
7 SORV - BLK - EIHP - RCSDEP (13, 18)			
8 SORV - BLK - EIHP - AFW - SAFW/MFW (19)			

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) Since HPR and RHR both depend on RHR pumps, Sequence 1 above considers the hardware failure of RHR trains. Sequence 2 is kept to include the human failures relating to LPR and RHR functions.

Table 2.5 SDP Worksheet for R. E. Ginna Nuclear Power Plant —— Medium LOCA [1]

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

Safety Functions Needed:	***Full* Creditable Mitigation Capability for Each Safety Function:**
Early Inventory, HP Injection (EIHP)	1/ 2 HPSI pumps (one multi-train system).
Low Pressure Injection (EILP)	1/ 2 RHR pumps (one multi-train system)
Low Pressure Recirculation (LPR)	1 / 2 RHR pump trains with operator switchover from injection to recirculation (operator action)
Containment Heat Removal (CHR) [2]	1/ 2 CS pumps or 1/4 CFCUs (2 multi-train systems)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 MLOCA - CHR (2)			
2 MLOCA - LPR (3)			
3 MLOCA - EILP (4)			
4 MLOCA - EIHP (5)			

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

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

Notes:

(1) The RCS pressure control is assured by the break size (between 2" to 5", and multiple PORV or SRV LOCA), and no further requirements of RCS pressure control function is considered.

(2) Maintaining containment cooling using either the containment spray system or the containment recirculation fan coolers are considered necessary since the failure of these systems could lead to a containment failure directly affecting SI and RHR pumps. Other plants of similar design, Point Beach and Kewaunee, do not have this assumption.

Table 2.6 SDP Worksheet for R. E. Ginna Nuclear Power Plant —— 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)	1 / 1 Accumulator to the intact loop (1 train)
Early Inventory, LP Injection (EILP)	1 / 2 RHR pump trains (1 multi-train system)
Low Pressure Recirculation (LPR)	1 / 2 RHR trains; Operator switchover from injection to recirculation (operator action)
Containment Heat Removal (CHR) [1]	1/ 2 CS pumps or 1/4 CFCUs (2 multi-train systems)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 LLOCA - CHR (2)			
2 LLOCA - LPR (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.

Notes:

(1) Maintaining containment cooling using either the containment spray system or the containment recirculation fan coolers are considered necessary since the failure of these systems could lead to a containment failure directly affecting SI and RHR pumps. Other plants of similar design, Point Beach and Kewaunee, do not have this assumption.

Table 2.7 SDP Worksheet for R. E. Ginna Nuclear Plant —— 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 (one multi-train system)(1)
Secondary Heat Removal (AFW)	1/ 2 MD AFW trains (1 multi-train system) or 1/1 TDAFW train (1 ASD train)
Turbine-driven AFW pump (TDAFW)	1 / 1 TDP trains of AFW (one train)
Secondary Heat Removal (MDAFW)	1 / 2 MDAFW trains (one multi-train system)
Standby AFW or MFW (SAFW/MFW)	Operator aligns 1/ 2 SAFW pumps or 1/ 2 MFW pumps (Operator action)
Recovery of AC Power in < 1 hrs (REC1)	SBO procedures implemented (operator action under high stress)(1)
Recovery of AC Power in < 10 hrs (REC10)	SBO procedures implemented (operator action)(1)
RCS Depressurization (RCSDEP)	Operator depressurizes using 2/2 PORVs and SG ARVs (Operator action)
High Pressure Injection (EIHP)	1/ 2 HPSI pumps (1 multi-train system)
Primary Heat Removal (FB)	Operator opens 1 / 2 PORVs (operator action)
High Pressure Recirculation (HPR)	1 / 2 HPSI pumps with 1/ 2 RHR pumps and with operator action for switchover (operator action)
Low Pressure Recirculation (LPR)	Operator aligns 1/ 2 RHR Pumps (Operator action)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 LOOP - AFW - SAFW/MFW - HPR (1)			
2 LOOP - AFW - SAFW/MFW - FB (1)			
3 LOOP - AFW - SAFW/MFW - EIHP (1)			
4 LOOP - EAC - RHR (3, 7, 10) (AC recovered)			

5 LOOP - EAC - RCSDEP (4, 8, 11) (AC recovered)			
6 LOOP - EAC - REC10 (5)			
7 LOOP - EAC - TDAFW - MDAFW - SAFW/MFW (12)			
8 LOOP - EAC - TDAFW - REC1 (13)			
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 Ginna SBO timing evaluations, it is assumed that offsite power must be restored within 1 hour if the TDAFW pump fails at the time of the SBO event. The operator error probability to restore offsite power within 1 hour is approximately estimated as 0.36.

(2) Ginna estimates that the batteries would have sufficient capability to run for 6 hours or more given the current loading and battery capacity (i.e., 8 hour batteries at constant loading). Consequently, it is assumed that if offsite power is not available within 6 hours, the TDAFW pump will fail due to battery depletion. With 1 AFW pump available for the first 6 hours, the SGs do not dry out until 8.5 hours with fuel damage occurring at 10.8 hours. So, it is assumed that offsite power must be restored within 10 hours (or 4 hours after TDAFW failure due to battery depletion) in order to prevent core damage. The operator error probability for restoring offsite power within 10 hours is estimated at 2.7E-2.

Table 2.8 SDP Worksheet for R. E. Ginna Nuclear Power Plant —— SGTR

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

Safety Functions Needed:	***Full* Creditable Mitigation Capability for Each Safety Function:**
Secondary Heat Removal (AFW) [1]	1 / 1 MDPs of AFW (one multi-train system) or 1 / 1 TDP of AFW (one ASD Train)
Early Inventory, HP Injection (EIHP)	1 / 2 HPSI pumps (one multi-train system)
Standby AFW or Main Feedwater (SAFW/MFW)	Operator aligns 1/ 2 SAFW pumps or 1/ 2 MFW pumps (Operator action)
SG Isolation (SGI)	Operator isolates the ruptured SG (Operator action)[2]
Pressure Equalization (EQ)	Operator depressurizes RCS using 1 / 1 SG ARV (on each SG fed by AFW) or RCS pressurizer PORV (1 / 2) to less than setpoint of relief valves of SG (operator action under high stress)[3]
RCS Depressurization (RCSDEP)	Operator initiated rapid cooldown when SI is not available or SG is not isolated (Operator action)
Residual Heat Removal (RHR)	Operator aligns 1/ 2 RHR pumps (Operator action)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 SGTR - EQ - RHR (3, 6)			
2 SGTR - AFW - SAFW/MFW (7,14,19)			
3 SGTR - EIHP - RHR (9, 12)			
4 SGTR - EIHP - RCSDEP (10, 13)			
5 SGTR - SGI - RHR (16, 18)			

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) Ginna Station PRA assumes that operators must initiate rapid cooldown to RHR conditions within 45 minutes if SI is not available. The operator error probability is estimated at 3.7E-2.

(2) Ginna Station PRA assumes that isolation of the ruptured SG must occur within 45 minutes or rapid cool down to RHR conditions is initiated. The operator error probability for this action is estimated at 7.2E-3.

Table 2.9 SDP Worksheet for R. E. Ginna Nuclear Plant —— MSLB [1]

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

Safety Functions Needed:	***Full* Creditable Mitigation Capability for Each Safety Function:**
Early Inventory, HP Injection (EIHP)	1 / 2 HPSI pumps (one multi-train system)
Secondary Heat Removal (AFW)	1 / 1 MDPs of AFW[2] (one train) or 1 / 1 TDP of AFW (one ASD Train)
Standby AFW or MFW (SAFW/MFW)	Operator aligns 1/ 2 SAFW pumps or 1/ 2 MFW pumps (Operator action)
Main Steam Isolation (ISOL)	Operator isolates the feedwater line break (operator action)
Feed and Bleed (FB)	Operator action with 1/ 2 PORV (Operator action)
High Pressure Recirculation (HPR)	1/ 2 HPSI pumps taking suction from RHR pumps (Operator action)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 MSLB -AFW - SAFW/MFW - HPR (4,12)			
2 MSLB - AFW - SAFW/MFW - FB (5, 13)			
3 MSLB - EIHP - AFW - SAFW/MFW (8)			
4 MSLB - EIHP - ISOL (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.

Table 2.10 SDP Worksheet for R. E. Ginna Nuclear Plant —— 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:**
Emergency Boration (HPI)	Operator conducts emergency boration using 2/3 charging pumps with 2/ 2 boric acid transfer pumps (operator action)
AMSAC (AMSAC)	AMSAC trips the turbine (one multi-train system)
Primary Relief (PR)	2/2 SRVs with 2/ 2 PORVs open (one train)
Secondary Heat Removal (AFW)	2/ 3 AFW pumps to at least 1/ 2 SG (one multi-train system)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 ATWS - PR (3)			
2 ATWS - AFW (4)			
3 ATWS - HPI (5)			
4 ATWS - ATWS (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.

Note:

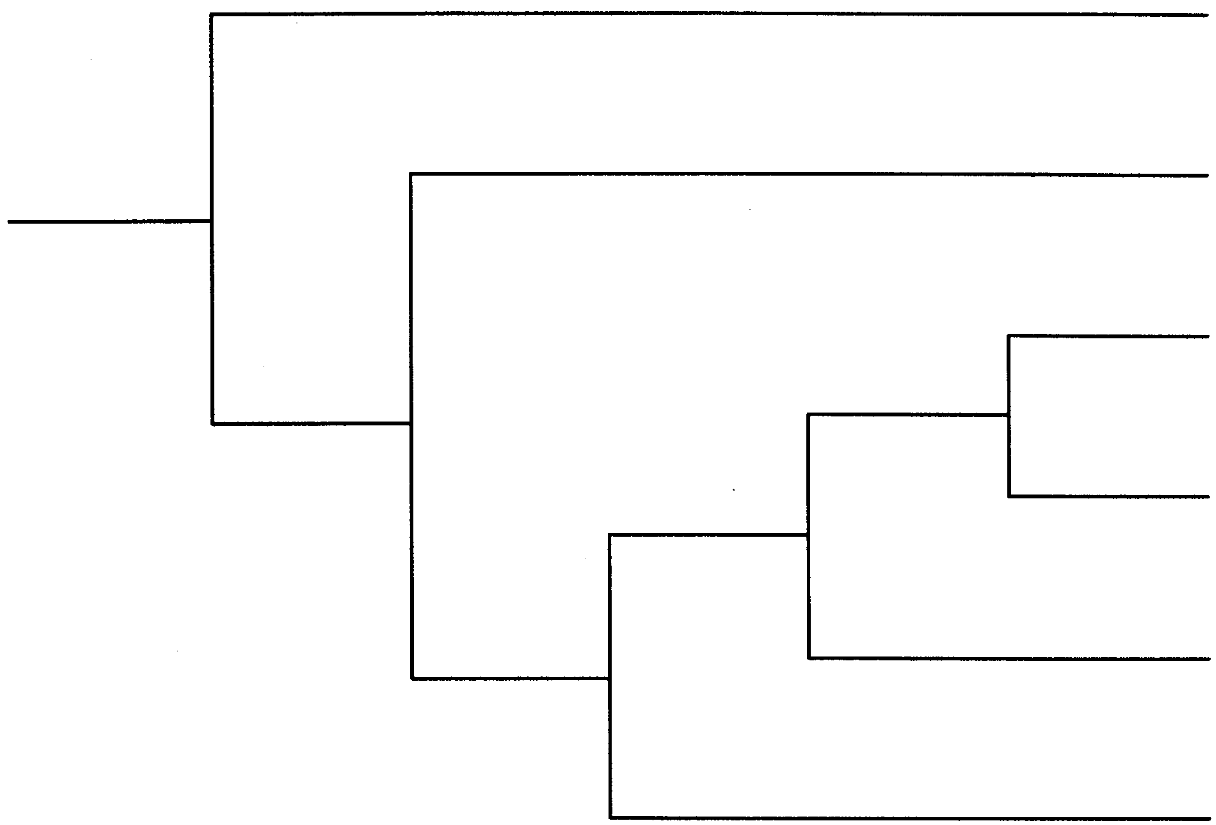
(1) The ATWS model here assumes that MFW is not available.

1.3 SDP EVENT TREES

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

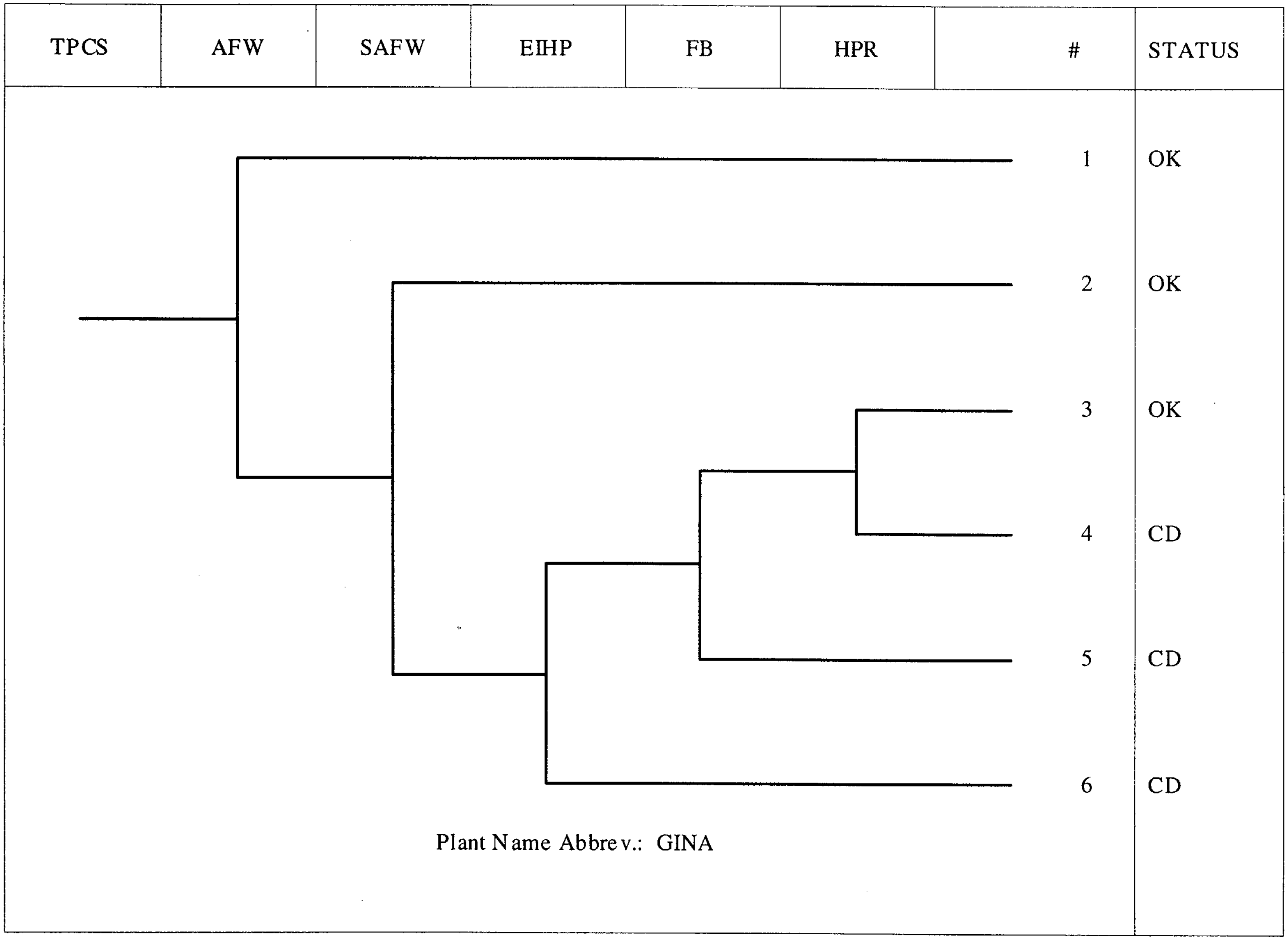
The following event trees are included:

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

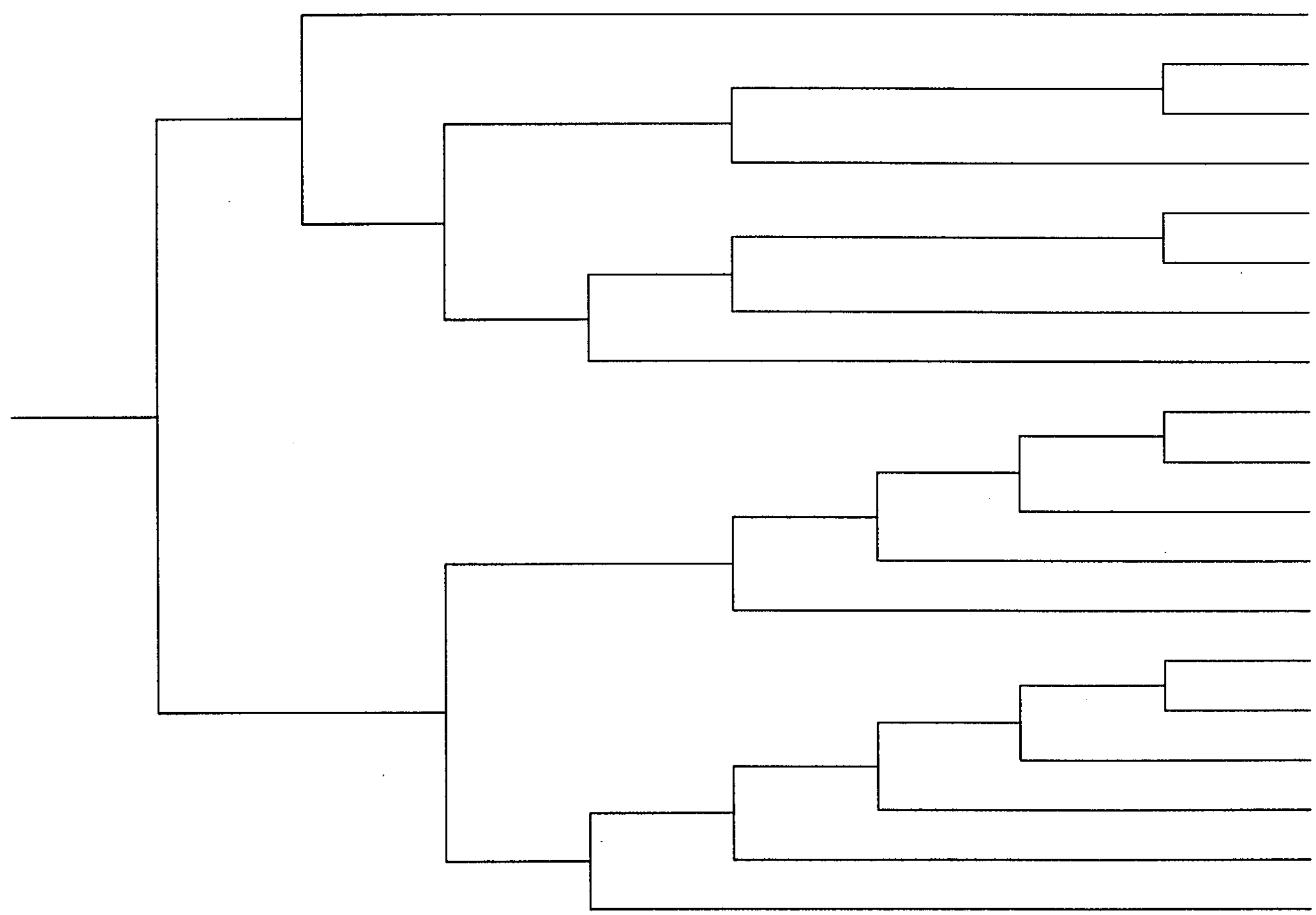


TRANS	AFW	SAFW/MF	EIHP	FB	HPR	#	STATUS
						1	OK
						2	OK
						3	OK
						4	CD
						5	CD
						6	CD

Plant Name Abbrev.: GINA


TPCS
AFW
SAFW
EIHP
FB
HPR
#
STATUS
1
OK
2
OK
3
OK
4
CD
5
CD
6
CD
Plant Name Abbrev.: GINA

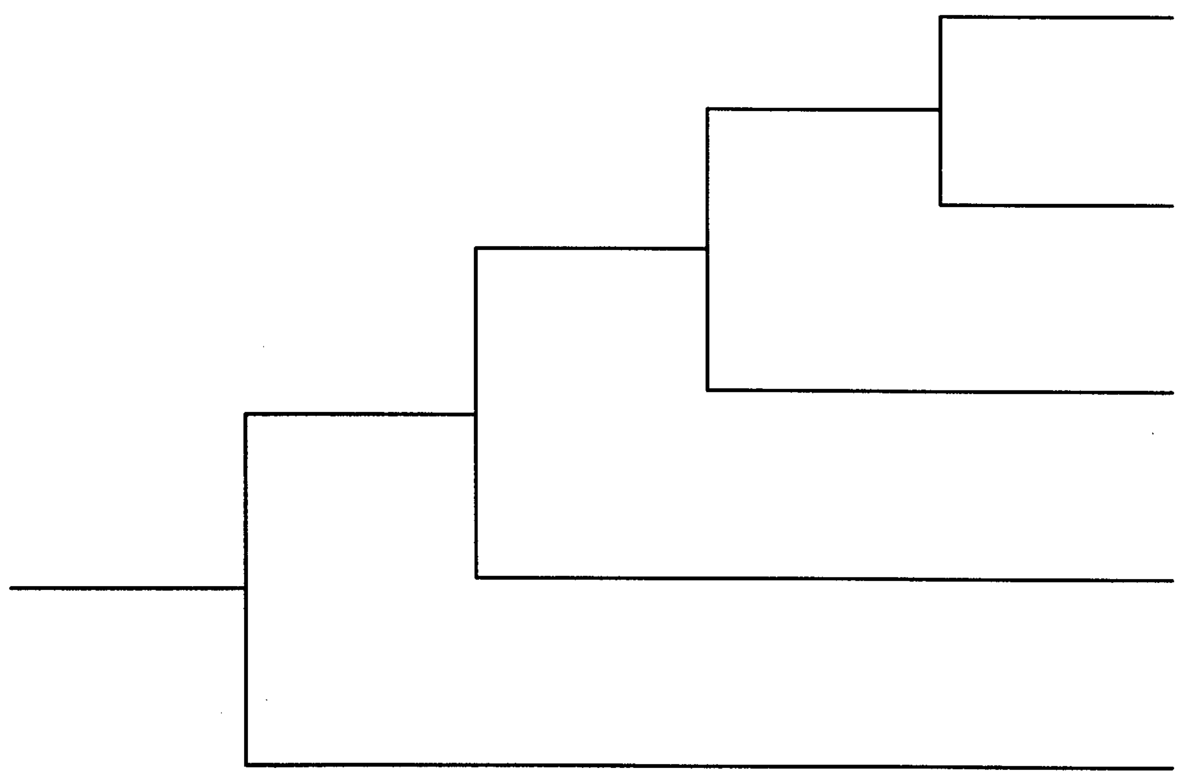
SLOCA	EIHP	HPR	AFW	SAFW/MF	RCSDEP	ACC	LPI	RHR	#	STATUS



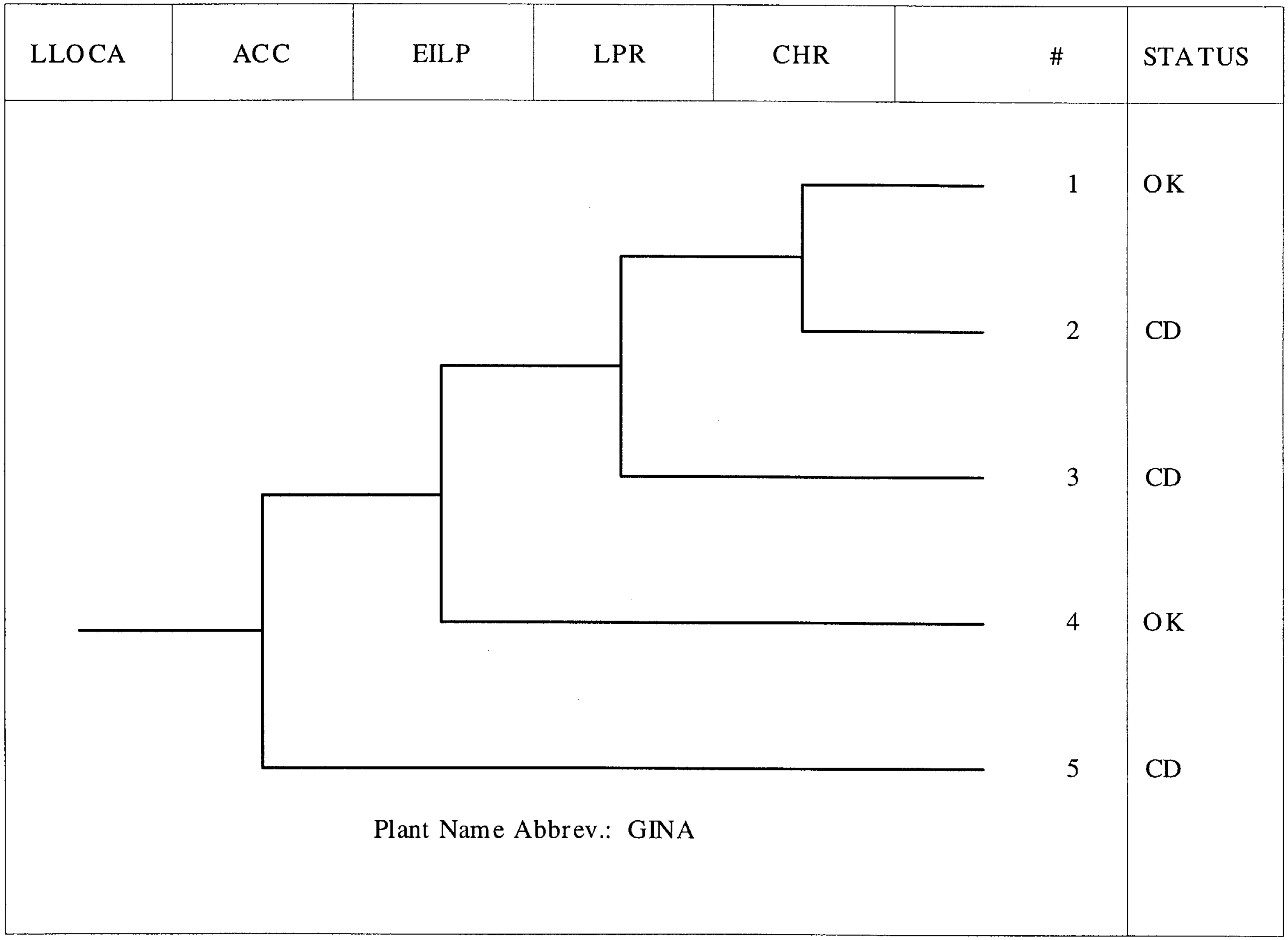
#	STATUS
1	OK
2	OK
3	CD
4	CD
5	OK
6	CD
7	CD
8	CD
9	OK
10	CD
11	CD
12	CD
13	CD
14	OK
15	CD
16	CD
17	CD
18	CD
19	CD

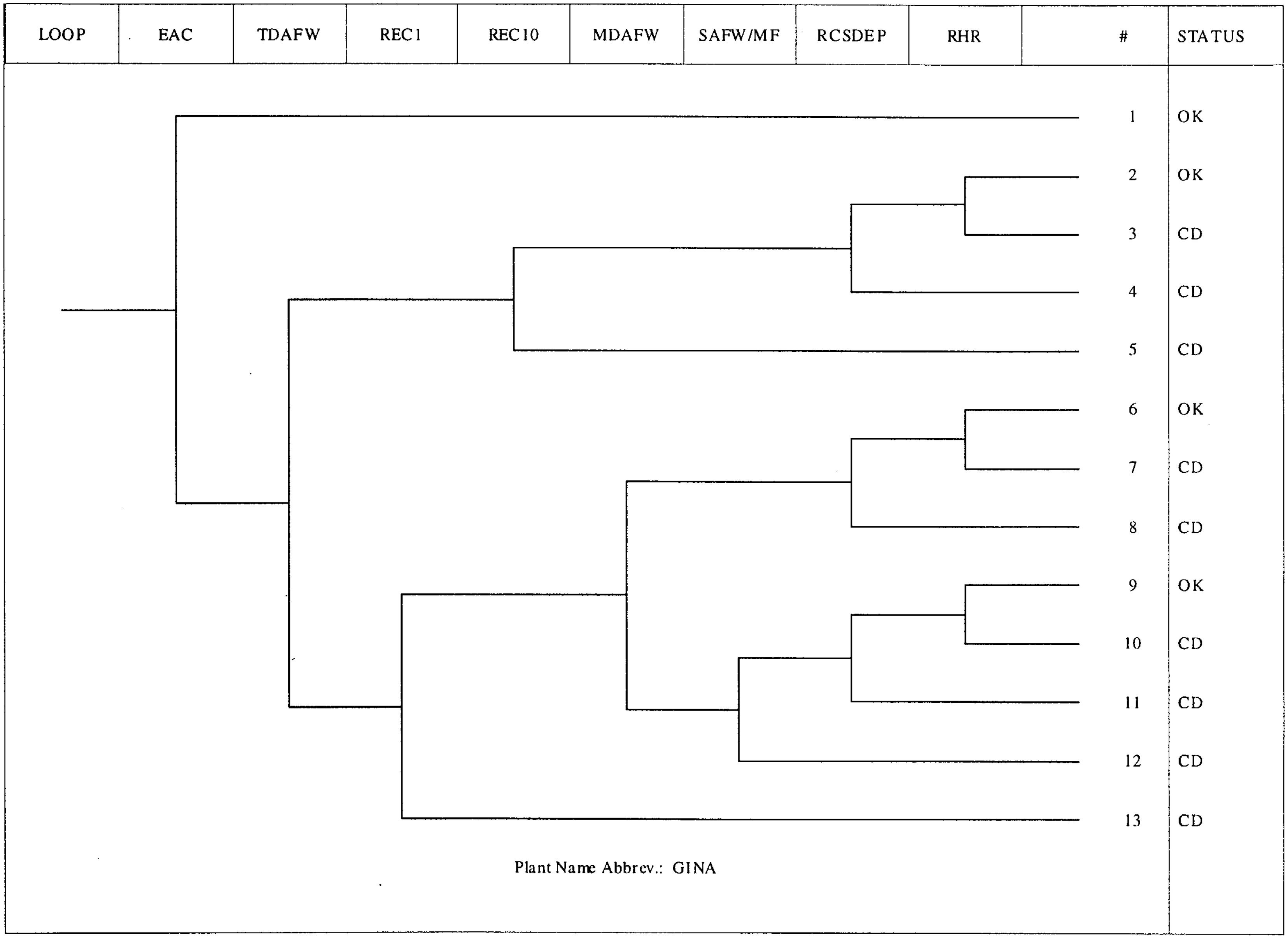
Plant Name Abbrev.: GINA

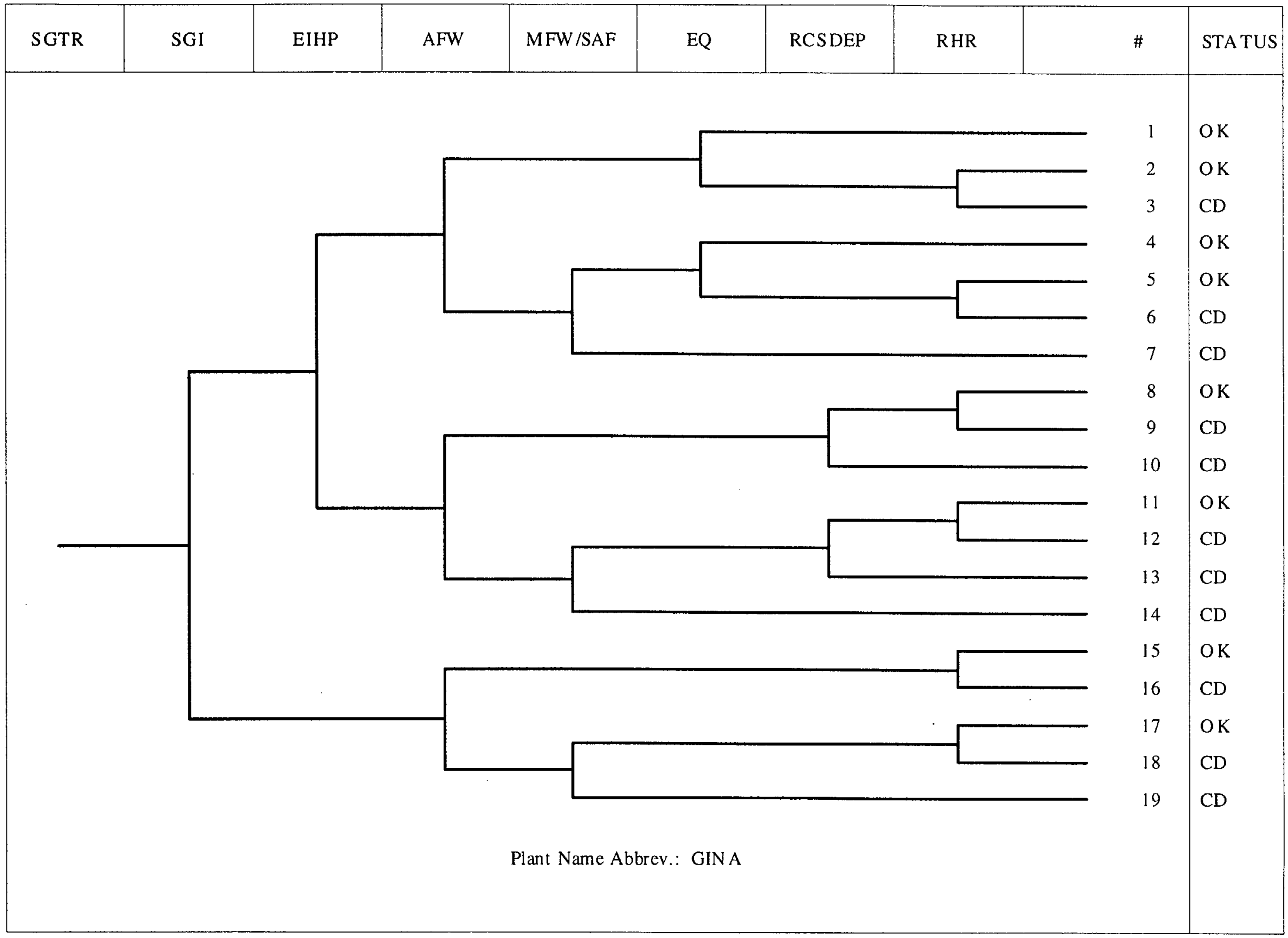
MLOCA	EIHP	EILP	LPR	CHR	#	STATUS
					1	OK
					2	CD
					3	CD
					4	OK
					5	CD



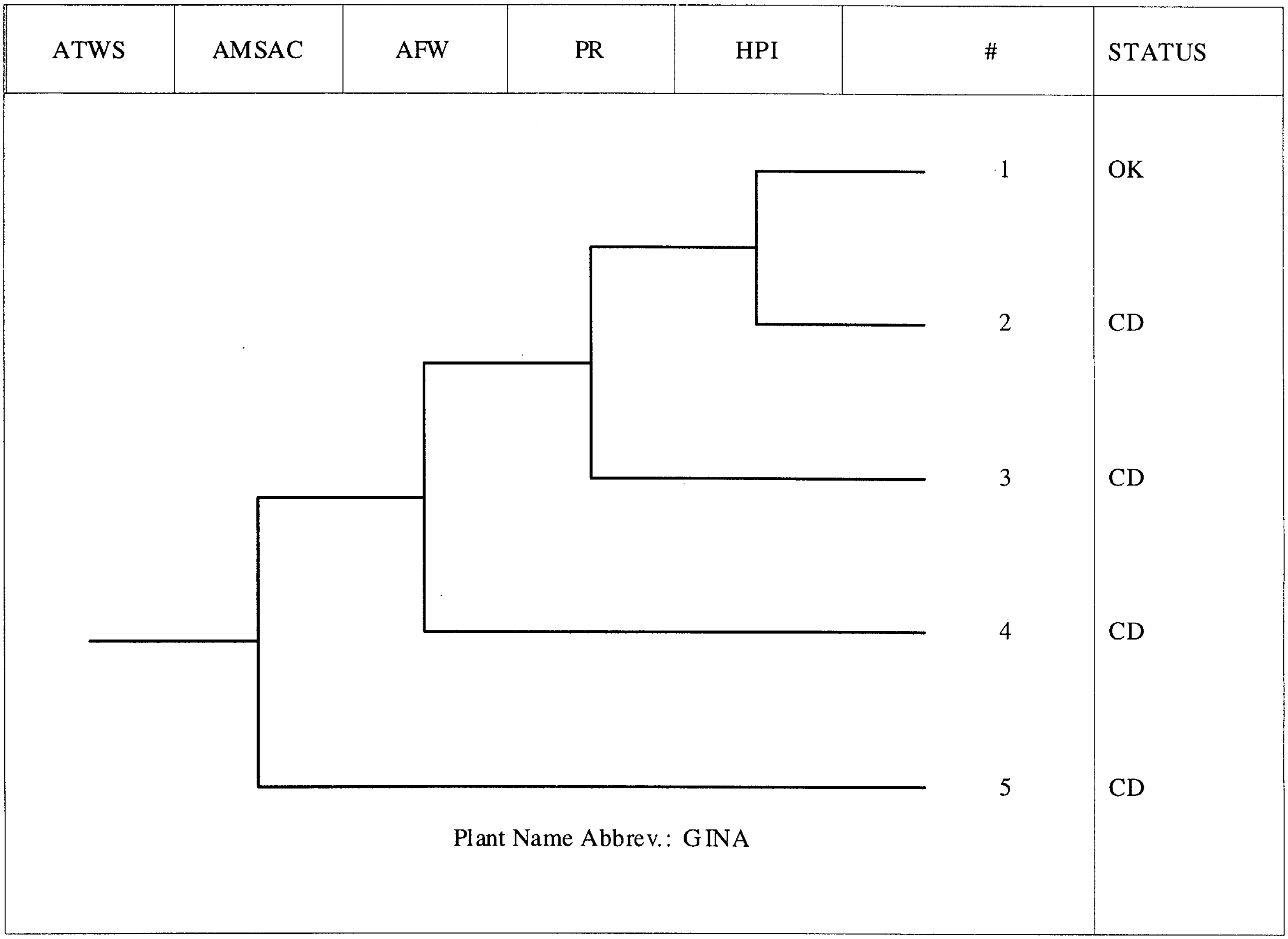
Plant Name Abbrev.: GINA

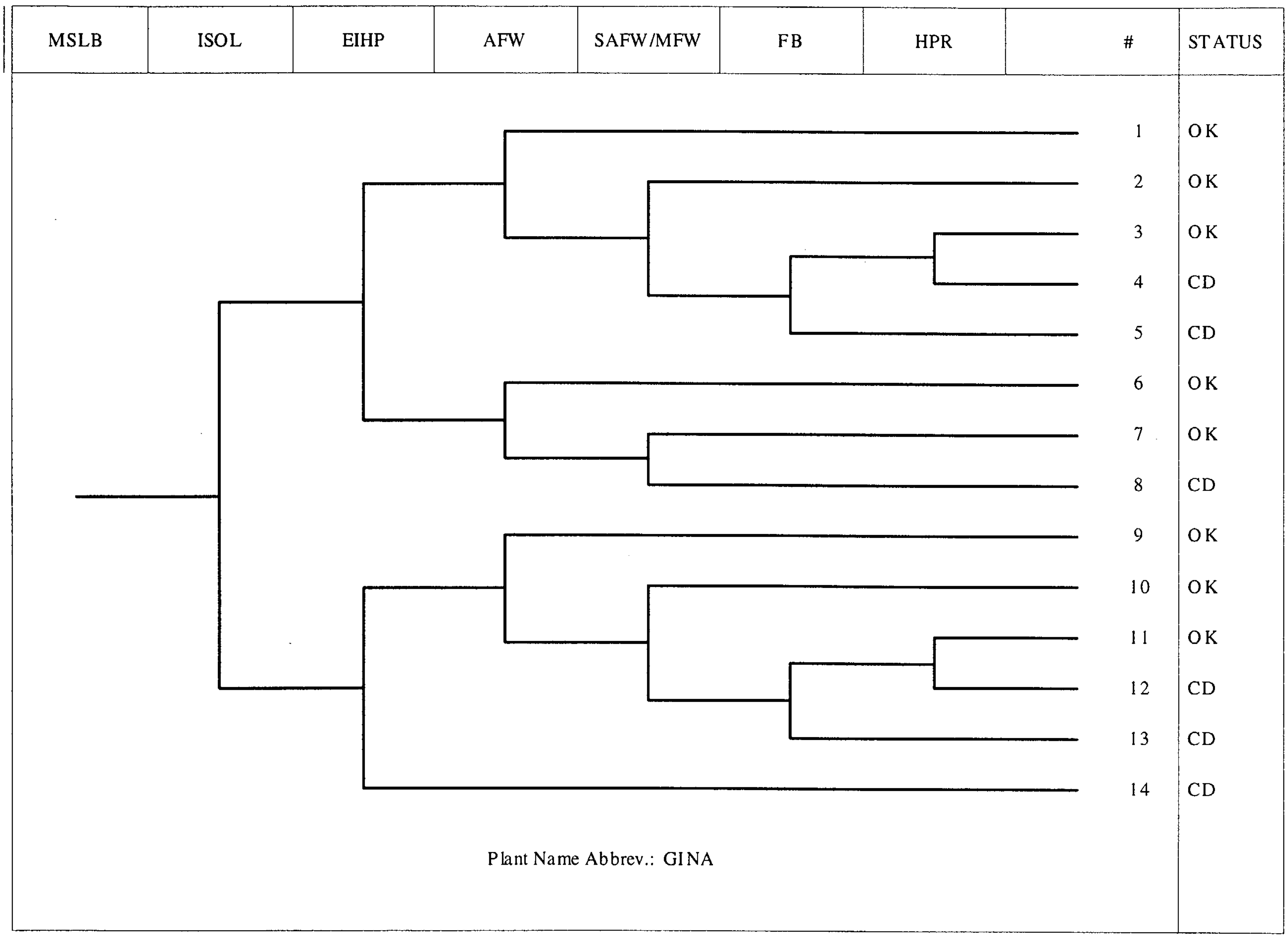

LLOCA
ACC
EILP
LPR
CHR
#
STATUS
1 OK
2 CD
3 CD
4 OK
5 CD
Plant Name Abbrev.: GINA


LOOP
EAC
TDAFW
REC1
REC10
MDAFW
SAFW/MF
RCSDEP
RHR
#
STATUS
1 OK
2 OK
3 CD
4 CD
5 CD
6 OK
7 CD
8 CD
9 OK
10 CD
11 CD
12 CD
13 CD
Plant Name Abbrev.: GINA



Plant Name Abbrev.: GINA


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



Plant Name Abbrev.: GINA

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. Rochester Gas and Electric Corporation, "R. E. Ginna Nuclear Power Plant Individual Plant Examination Submittal Report," Revision 1, January 1997.