File Center

Template = NRR-056

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

February 10, 2000

Mr. William A. Eaton
Vice President, Operations GGNS
Entergy Operations, Inc.
P. O. Box 756
Port Gibson, MS 39150

SUBJECT: GRAND GULF NUCLEAR STATION, UNIT 1 RE: SITE-SPECIFIC WORKSHEETS FOR USE IN THE NUCLEAR REGULATORY COMMISSION'S SIGNIFICANCE DETERMINATION PROCESS (TAC NO. MA6544)

Dear Mr. Eaton:

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 Grand Gulf Nuclear Station (GGNS) in April 2000. Included in the enclosed Risk-Informed Inspection Notebook 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.

DF01

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) submittal that was 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 301-415-2623.

Sincerely,
/RA/
S. Patrick Sekerak, Project Manager, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-416

Enclosures: Risk-Informed Inspection Notebook

cc: See next page

TEMPLATE = NRR-056

DISTRIBUTION
PUBLIC
File Center
OGC
ACRS
K.Brockman, RIV
J.Kilcrease,RIV
J.Hurley, RIV
PDIV-1 Reading
S.Richards (cover ltr only)
W.Dean
D.Coe

To receive a copy of this document, indicate "C" in the box									
OFFICE	PDIV-1/PM	C	PDIV-1/LA						PDIV-1/SC C
NAME	P.Sekerak		D.Johnson dlj						R.Gramm RG
DATE	2 / 9 /00		02/ 10 /00		/ /00		/ /00		2/ 10 /00

DOCUMENT NAME: G:\PDIV-1\GrandGulf\LtrMA6544.wpd
OFFICIAL RECORD COPY

cc:

Executive Vice President
& Chief Operating Officer
Entergy Operations, Inc.
P. O. Box 31995
Jackson, MS 39286-1995

Wise, Carter, Child & Caraway
P. O. Box 651
Jackson, MS 39205

Winston & Strawn
1400 L Street, N.W. - 12th Floor
Washington, DC 20005-3502

Director
Division of Solid Waste Management
Mississippi Department of Natural
Resources
P. O. Box 10385
Jackson, MS 39209

President
Claiborne County Board of Supervisors
P. O. Box 339
Port Gibson, MS 39150

Regional Administrator, Region IV
U.S. Nuclear Regulatory Commission
611 Ryan Plaza Drive, Suite 1000
Arlington, TX 76011

Senior Resident Inspector
U. S. Nuclear Regulatory Commission
P. O. Box 399
Port Gibson, MS 39150

General Manager, GGNS
Entergy Operations, Inc.
P. O. Box 756
Port Gibson, MS 39150

Attorney General
Department of Justice
State of Louisiana
P. O. Box 94005
Baton Rouge, LA 70804-9005

State Health Officer
State Board of Health
P. O. Box 1700
Jackson, MS 39205

Office of the Governor
State of Mississippi
Jackson, MS 39201

Attorney General
Asst. Attorney General
State of Mississippi
P. O. Box 22947
Jackson, MS 39225

Vice President, Operations Support
Entergy Operations, Inc.
P.O. Box 31995
Jackson, MS 39286-1995

Director, Nuclear Safety Assurance
Entergy Operations, Inc.
P.O. Box 756
Port Gibson, MS 39150

David A. Lochbaum
Nuclear Safety Engineer
Union of Concerned Scientists
1616 P Street N.W., Suite 310
Washington, DC 20036-1495

Steve Floyd
Nuclear Energy Institute
1776 I Street N.W., Suite 400
Washington, DC 20006

RISK-INFORMED INSPECTION NOTEBOOK FOR

GRAND GULF NUCLEAR STATION

UNIT 1

BWR-6, GE, WITH MARK III CONTAINMENT

Prepared by

Brookhaven National Laboratory
Department of Advanced Technology

Contributors

M. A. Azarm
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

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
M/S 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 Grand Gulf Unit I Nuclear Station.

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

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

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). 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 Grand Gulf Unit I Nuclear Station.

1.1 INITIATORS AND SYSTEM DEPENDENCY

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

Table 1 Initiators and System Dependency Table for Grand Gulf Unit I

Affected System	Major Components	Support Systems	Initiating Event Scenarios
Reactor Vessel Depressurization System ADS/SRVs	Self-actuating and power-operated safety relief valves	ESF 25 VDC; Instrument Air System - backup by accumulators	Transient[1], LOOP, SLOCA, MLOCA, ATWS
Control Rod Drive Hydraulic System (CRDH)	Two CRD MD pumps and piping	ESF AC, BOP AC, ESF DC, CCW, Instrument Air (required for enhanced flow mode only 2/2 pumps)	Transient[1], LOOP, ATWS
Standby Liquid Control (SLC)	Two SLC MD pumps and explosive valves	ESF 480 V-AC	ATWS
RHR/ LPCI Residual Heat Removal/ (Low Pressure Coolant Injection) Trains A, B, and C	Three RHR MD pumps A, B, and C, MOVs	Engineered Safety Features (ESF) 4160 V-AC and 480 V-AC and 125 V-DC Div I to LPCI Train A, Div II to Trains B and C, SSW Train A to LPCI Train A, SSW Train B to LPCI Trains B and C	Transient, SLOCA, MLOCA, LLOCA, LOOP, ATWS
RHR/SPC (Suppression Pool Cooling) Trains A and B	Two MD Pumps A and B, MOVs	ESF 4160 V-AC and 480 V-AC, ESF 125 V-DC, SSW	Transient, SLOCA, MLOCA, LLOCA, LOOP, ATWS
RHR/CS (Containment Spray) Trains A and B	Two MD Pumps A and B, MOVs	ESF 4160 VAC and 480V AC, ESF 125 V DC, SSW	Transient, SLOCA, MLOCA, LLOCA, LOOP
RHR/SDC (Shutdown Cooling) Trains A and B	Two MD Pumps A and B, MOVs	4160V AC and 480V AC, ESF 125 VDC, Standby Service Water (SSW), (RHR Pump Room HVAC not required)	Transient[1], LOOP, ATWS
Low Pressure Core Spray (LPCS)	one MD Pump, MOVs	ESF AC Div I, ESF DC Div I, ECCS Pump Room HVAC (pump can operate approximately 10 to 12 hours without HVAC)	Transient[1], LOOP, ATWS, SLOCA, MLOCA, LLOCA

Table 1 (Continued)

Affected System	Major Components	Support Systems	Initiating Event Scenarios
High Pressure Core Spray (HPCS)	One MD pump and dedicated diesel generator	ESF AC Div III, ESF DC Div III, ECCS Room HVAC	Transient[1], LOOP, ATWS, SLOCA, MLOCA
Suppression Pool Makeup (SPMU)	Two trains of two MOVs per train	ESF AC, ESF DC	SLOCA, MLOCA, LLOCA, ATWS
Reactor Core Isolation Cooling (RCIC)	One TD pump, MOVs	ESF DC Div I, ESF DC Div II (for redundant actuation logic and Level 8 protection instrumentation), Steam Tunnel HVAC (RCIC pump will operate for 30 minutes after steam leak detection signal is initiated. No isolation occurs during SBO due to loss of power to timer)	Transient,[1] SLOCA, LOOP, ATWS
DC Power System ESF DC BOP DC	Divs. 1, 2, and 3 DC Power buses, BOP DC Power buses, batteries	<u>ESF Power Distribution</u> See individual safety systems <u>BOP Power Distribution</u>	All
Power Conversion System (PCS)	35% capacity turbine bypass, 4 main steam lines, two MSIVs per line, two TD feedwater pumps, three MD condensate pumps, three MD condensate booster pumps	500 KV (offsite power), 120 V-AC (non class 1E), 250 VDC (for main turbine oil pumps), 480 V-AC (non class 1E), Turbine Building Cooling Water (TBCW), Circulating Water, Instrument Air, Steam Tunnel HVAC	Transient[1], SLOCA , ATWS
Containment Venting	4 MOVs	ESF AC, ESF DC, Instrument Air	Transient, SLOCA, MLOCA, LLOCA, LOOP, ATWS
Standby Service Water (SSW) A, B, and C	Three MD pumps, headers, MOVs	ESF AC and ESF DC Div I to SSW A, Div II to SSW B, Div III to SSW C	All

Table 1 (Continued)

Affected System	Major Components	Support Systems	Initiating Event Scenarios
SSW/ RHR X-TIE	One train of MOVs	ESF AC Div II, SSW Train B	Transient[1], SLOCA, MLOCA, LLOCA, LOOP, ATWS
Component Cooling Water (CCW)	Three MD pumps, headers, MOVs	ESF AC Div III (Train B CCW pump), BOP AC, ESF DC Div II, BOP DC (DC required to start pumps - pumps normally operating - DC not modeled), SSW B (alternate source of cooling water under certain conditions), PSW	All
Turbine Building Cooling Water (TBCW)	Three MD pumps, headers, MOVs	BOP AC, BOP DC (DC required to start standby pump), PSW	Transient, SLOCA, ATWS
Plant Service Water (PSW)	8 MD pumps	ESF AC, BOP AC, ESF DC, BOP DC (DC required to start pumps - pumps normally operating - DC not modeled), Instrument Air	All
Instrument Air/Service Air (IA/SW)	2 MD centrifugal compressors, dryers	Offsite power, ESF AC Power, BOP AC Power, ESF DC Power, BOP DC Power, TPCCW	Transient[1], SLOCA, MLOCA, LLOCA, LOOP, ATWS
Fire Water Injection	One MD pump, two DD pumps	ESF AC Div I, Div II , BOP AC, Instrument Air	Transient,[1] SLOCA, MLOCA, LOOP, ATWS

Table 1 (Continued)

Affected System	Major Components	Support Systems	Initiating Event Scenarios
AC Power System	Three EDG, busses, two ESF transformers, 8 BOP transformers	ESF AC Power System (EDGs 1, 2, and 3) ESF 125V DC, SSW, EDG HVAC (DG X-tie: 1 and 2): ESF AC Div 3 to DGX 1, ESF AC Div 3 to DGX 2, ESF DC Div 3 to DGX 1, ESF DC Div 3 to DGX 2 BOP AC Power System Component Cooling Water (CCW), Turbine Building Cooling Water (TBCW), Plant Service Water (PSW), Chilled Water, Circulating Water, Instrument Air, Steam Tunnel HVAC	All
Chilled Water (CW)		BOP AC, BOP DC (DC required to start standby pump), PSW, Instrument Air	Transient
DG Rooms HVAC	One Fan per room, inlet and outlet dampers	ESF AC Div I, II, and III	LOOP
SSW Pump House HVAC (Trains A and B)	One Fan per room, inlet and outlet dampers	ESF AC Div I and II to Train A, Div II to Train B	All
ECCS Pump Rooms HVAC	Fan coil units with SSW cooling water	ESF AC Divs I, II, and III, SSW Trains A, B, and C	All
Steam Tunnel HVAC	Fans, dampers	BOP AC, CW	Transient, SLOCA, 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 125v 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. The above information is based upon the GGNS Response to Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities" submitted to the NRC by letter dated December 23, 1992.

3. The overall core damage frequency for internal events and flooding is 1.72E-5 per reactor-year based on the December 23, 1992 IPE submittal.

1.2 SDP WORKSHEETS

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

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

Table 2.1 SDP Worksheet for Grand Gulf Unit I —— 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)	1/3 Feedwater pumps if Rx pressure > 400 psi and 1/3 Condensate Injection if Rx Pressure ≤400 psi and 1/ 4 MS lines with Turbine Bypass (TB) valves open with operable condenser (operator action)
Early Inventory Control High Pressure Injection (EIHP)	HPCS (1 train) or RCIC (1 ASD train)
Early Inventory Control CRD Pumps (EICRD)	2/ 2 CRD pumps (operator action) [1]
Depressurization (DEP)	4/20 SRVs manually opened (high stress operator action) [2]
Low Press Injection (LPI)	1/3 Condensate pumps if Rx Pressure ≤ 400 psi and 1/ 4 MS lines with Turbine Bypass (TB) valves open with operable condenser} (operator action) or 1/3 RHR trains in LPCI Mode (1 multi-train system) or 1 / 1 LPCS pumps (1 diverse train) or 1/1 Standby Service Water (SSW) cross-tie (operator action)
Residual Heat Removal Suppression Pool Cooling (RHR-SPC)	1/ 2 RHR pumps and corresponding 1/ 2 RHR heat exchangers in suppression pool cooling (SPC) mode (operator action)
Late Inventory CRD (LICRD)	1/2 CRD pumps (operator action) [1]
Late Depressurization (LDEP)	{4/20 SRVs manually opened and [(1/3 condensate pumps if Rx Pressure ≤ 400 psi and 1/ 4 MS lines with Turbine Bypass (TB) valves open with operable condenser) or 1/1 SSW cross-tie or 1/3 Firewater pumps if Rx pressure <150 psi]} (operator action)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining* Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 Trans - PCS- RHRSPC - LICRD - LDEP (5)			
2 Trans - PCS - EIHP - LPI (8)			

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

Notes:

1. According to GGNS IPE Tables 3.1-6 to 3.1-8, operation of both control rod drive pumps (CRD), i.e., at maximized flow, is only successful when the vessel is at high pressure. One CRD pump is sufficient if CRD is only used in the long term, i.e. when coolant makeup has been provided for a long time. The event trees show enhanced CRD (2 pump operation) as a high pressure injection source only if depressurization fails.

(2) Worksheets consider Depressurization (DEP) using SRVs as a high-stress operator action;
GGNS IPE Table 3.3.-7 assigns a failure probability for failure to depressurize (DEP) of 3.0E-04, ≤ 3.5 mins.

Table 2.2 SDP Worksheet for Grand Gulf Unit I —— 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)	{1/3 Feedwater pumps if Rx pressure > 400 psi and 1/3 Condensate Injection if Rx Pressure ≤ 400 psi and 1/ 4 MS lines with Turbine Bypass (TB) valves open with operable condenser} (operator action)
Early Containment Control (EC)	{Vapor suppression system (VSS) passive operation of suppression pool and vacuum breakers and 1/ 2 Suppression Pool Makeup} (operator action)
Early Inventory Control High Pressure (EIHP)	HPCS (1 train) or RCIC (1 ASD train)
Depressurization (DEP)	4/20 SRVs manually opened (high stress operator action)[1]
Low Pressure Injection (LPI)	(1/3 Condensate Injection if Rx Pressure ≤ 400 psi and 1/ 4 MS lines with Turbine Bypass (TB) valves open with operable condenser} (operator action) or 1/3 RHR trains in LPCI Mode (1 multi-train system) or 1 / 1 LPCS pumps (1 diverse train)
Containment Heat Removal (CHR)	{1/ 2 suppression pool makeup (SPMU) lines with 2 MOVs in each line to open and 1/2 RHR pumps in SPC or CSC modes} (operator action)
Late Depressurization (LDEP)	4/20 SRVs manually opened (high stress operator action)
Late Inventory Makeup (LI)	{1/3 Condensate Injection if Rx Pressure ≤ 400 psi and 1/ 4 MS lines with Turbine Bypass (TB) valves open with operable condenser} (operator action) or 1/1 Standby Service Water (SSW) cross-tie (operator action) or 1/3 Firewater pumps (operator action)[1]

Circle Affected Functions	**Recovery or Failed Train**	***Remaining* Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 SLOCA - PCS - CHR - LI (4)			
2 SLOCA -PCS -CHR - LDEP (5)			

3 SLOCA - PCS - EI - LPI - LI (8)			
4 SLOCA - PCS - EI- DEP (9)			
5 SLOCA - EC - LI (11)			
6 SLOCA - EC - DEP (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. Depressurization (DEP) using SRVs is considered a high-stress operator action; GGNS IPE assigns a faillure probability for failure to depressurize (DEP) of 3.0E-04, $\leq$ 3.5 mins.
2. Dependency of EC on PCS is not modeled. Failure of EC includes failure due to PCS dependency.

Table 2.3 SDP Worksheet for Grand Gulf Unit I —— 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 (VSS) passive operation of suppression pool and vacuum breakers and 1/2 Suppression Pool Makeup} (1 multi-train system)
Early Inventory Control (EI)	HPCS (1 train)
Depressurization (DEP)	3/20 SRVs (High Stress Operator Action)[1]
Low Press Injection (LPI)	1/3 RHR trains in LPCI mode (1 multi-train system) or 1 / 1 LPCS train (1 diverse train)
Containment Heat Removal (CHR)	1/2 RHR trains in SPC or CSC mode (operator action)
Late Depressurization (LDEP)	3/20 SRVs (High Stress Operator Action)
Late Inventory, Makeup (LI)	1/1 SSW cross-tie to RHR injection (operator action)[2]

Circle Affected Functions	**Recovery or Failed Train**	***Remaining* Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 MLOCA - CHR - LI (3)			
2 MLOCA - CHR - LDEP (4)			
3 MLOCA - EI - LPI - LI (7)			
4 MLOCA - EI - DEP (8)			

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

Notes:

1. Depressurization (DEP) using SRVs is considered a high-stress operator action; GGNS IPE assigns a faillure probability for failure to depressurize (DEP) of 3.0E-04, $\leq$ 3.5 mins.

(2) In late inventory (LI) operator actions using condensate pumps or firewater pumps are not considered in the GGNS event trees.

Table 2.4 SDP Worksheet for Grand Gulf Unit I —— 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 (VSS) passive operation of suppression pool and vacuum breakers and 1/2 Suppression Pool Makeup} (1 multi-train system)
Early Inventory Control (EI)	HPCS (1 train) or 1/3 RHR trains in LPCI mode (1 multi-train system) or 1/ 1 LPCS train (1 diverse train)[1]
Containment Heat Removal (CHR)	1/2 RHR trains in SPC or CSC mode (operator action)
Containment Venting (CV)	4/4 Containment Venting Valves open -(high stress operator action)[2]
Late Inventory, Makeup (LI)	1/1 SSW cross-tie to RHR injection (Operator Action)

Circle Affected Functions	Recovery or Failed Train	*Remaining* Mitigation Capability Rating for Each Affected Sequence	Sequence Color
1 LLOCA - CHR - LI (3, 5)			
2 LLOCA - EI (6, 9)			
3 LLOCA - EC - LI (8)			

4 LLOCA - CHR - CV (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.			

Notes:

1. Condensate is not credited in this sequence since makeup to the hotwell may not be sufficient.

2. Containment venting is considered high-stress operator action;
   GGNS IPE Table 3.3-7 assigns a failure probability of 2.4E-3 with an error factor of 10 for failure to bypass containment isolation within 45-60 minutes.

Table 2.5 SDP Worksheet for Grand Gulf Unit I —— (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 Div 1 or Div 2 DGs (EAC1&2)	1/ 2 EDGs (1 multi-train system)
Recovery of Offsite Power within 1 Hour (REC1)	High stress operator action[1, 3, 4]
High Pressure Core Spray Pump and Div 3 DG (HPCS)	HPCS pump and motor (1 train) and 1/1 Div 3 DG (1 train)
Reactor Core Isolation Cooling (RCIC)	RCIC (1ASD train)[2]
Recovery of Offsite Power within 8 Hours (REC8)	Operator action[1, 3]
HPCS DG Div 3 X-Tie to Div 1 or Div 2 (DIV 3 X-TIE)	Cross-tie Div 3 DG to Div 1 or Div 2 DG (operator action)[5]
Containment Heat Removal (CHR)	1/2 RHR pumps in SPC or CSC or SDC (operator action)[5]

Circle Affected Functions	**Recovery or Failed Train**	***Remaining* Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 LOOP - EAC1&2 - REC1 - REC8 - CHR (5)			
2 LOOP - EAC1&2 - REC1 - RE8 - DIV3XTIE (6)			
3 LOOP - EAC1&2 - REC 1 - HPCS - REC8 (8)			
4 LOOP - EAC1&2 - REC1 - HPCS - RCIC (9)			

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

Notes:

(1) In the GGNS IPE, Loss of Offsite Power Event Tree, Fig. 3.1.2.3-1, no credit is taken for restoring offsite power. Instead, credit is taken for cross-tying the Div. 3 diesel generator, which is normally dedicated for the HPCS pump and motor, to either Div. 1 or Div. 2 emergency buses. However, important human errors in Tables 3.3-8 and 3.3-9 of the GGNS IPE include:
   b) Failure to recover diesel from maintenance in 1 hour: 9.0E-1
   c) Failure to recover diesel from hardware failures in 1 hour: 9.0E-1
   d) Failure to recover offsite power in 4 hours: 6.4E-2
   e) Failure to recover offsite power in 10 hours: 2.0E-2

(2) Successful operation of the RCIC system is based on the assumption that the operator bypasses the high temperature isolation signal that originates from the Steam Tunnel upon loss of HVAC in the tunnel. (See GGNS IPE page 3.4-4)

(3) The failure to recovery offsite power within 1 hour is based on the time to core damage when there is no reactor pressure vessel injection. The failure to recover offsite power within 8 hours is based on the 6 hours running time for the HPCS and RCIC systems using the condensate storage tank (CST) as the suction source after which time pump suction must be transferred to the suppression pool which is then at a high temperature. The 2 hour additional time to reach 8 hours takes into consideration the reduced reactor decay heat and also the time to core damage at that point if no reactor pressure vessel injection occurs after 6 hours. (See GGNS IPE pages 3.4-3 and 3.4-8).

(4) The GGNS IPE also considers station blackout (SBO) as occurring 3 hours after loss of offsite power (LOOP) if there are failures of the HVAC systems in the EDG and standby service water (SSW) pump rooms. In such cases, core damage is expected to occur after 4 hours. These sequences are conservatively accounted for by the 1 hour recovery time to restore offsite power. (See GGNS IPE page 3.4-3).

(5) The HPCS Div 3 emergency diesel generator, when cross-tied to Div 1 or Div 2 AC power, is credited with providing Div 1 or Div 2 standby service water and Div 1 or Div 2 RHR suppression pool cooling mode. (See GGNS IPE page 3.1-15).

Table 2.6 SDP Worksheet for Grand Gulf Unit I —— 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)	4/20 SRVs (1 multi-train system)
Recirculation Pump Trip (RPT)	Manual or automatic trip of recirculation pumps (1 multi-train system)
Inhibit ADS and HPCS (INH)	Operator inhibits ADS and HPCS (high stress operator action)[1a, 2b]
High Pressure Injection (HPI)	1/3 Feedwater pumps if Rx pressure > 400 psi and 1/ 4 MS lines with Turbine Bypass (TB) valves open with operable condenser and controls RPV level with FW level control (High stress operator action) or {MSIV closure and RCIC)(operator action)
Reactivity Control (SLC)	1 / 2 SLC pumps and valves (high stress operator action)[2a]
Depressurization (DEP)	4/20 SRVs manually opened (high stress operator action)[1b]
Low Pressure Injection (LPI)	1/3 Condensate Pumps (operator action) or 1/3 RHR pumps in LPCI mode (1 multi-train system) or 1/1 LPCS pump (1 diverse train) or 1 / 2 CRD pumps (operator action) or 1/1 SSW cross-tie (operator action)
Containment Heat Removal (CHR)	1/2 RHR pumps in SPC (operator action) or 4/4 Containment venting valves open (high stress operator action)[3]

Circle Affected Functions	**Recovery or Failed Train**	***Remaining* Mitigation Capability Rating for Each Affected Sequence:**	**Sequence Color**
1 ATWS - OVERP (10)			
2 ATWS - SLC (7)			

3 ATWS - RPT (9)			
4 ATWS - HPI - LPI (5)			
5 ATWS - HPI - DEP (6)			
6 ATWS - INH (8)			
7 ATWS - CHR (2, 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.

Notes:

(1) For ATWS:

a. During an ATWS event, although HPCS starts automatically together with RCIC at RPV Level 2, HPCS is secured to prevent relatively cold unborated water from being sprayed directly on the core.

b. The GGNS IPE description for ATWS does not indicate how many SRVs are required to open for depressurization.

(2) Human event probabilities (HEPs) under 1E-2 in GGNS IPE:

(a) SLC during ATWS - 7.0E-4, - 5 mins.

(b) INH - Failure to inhibit ADS and HPCS during ATWS - 1.0E-5, - 2 mins.

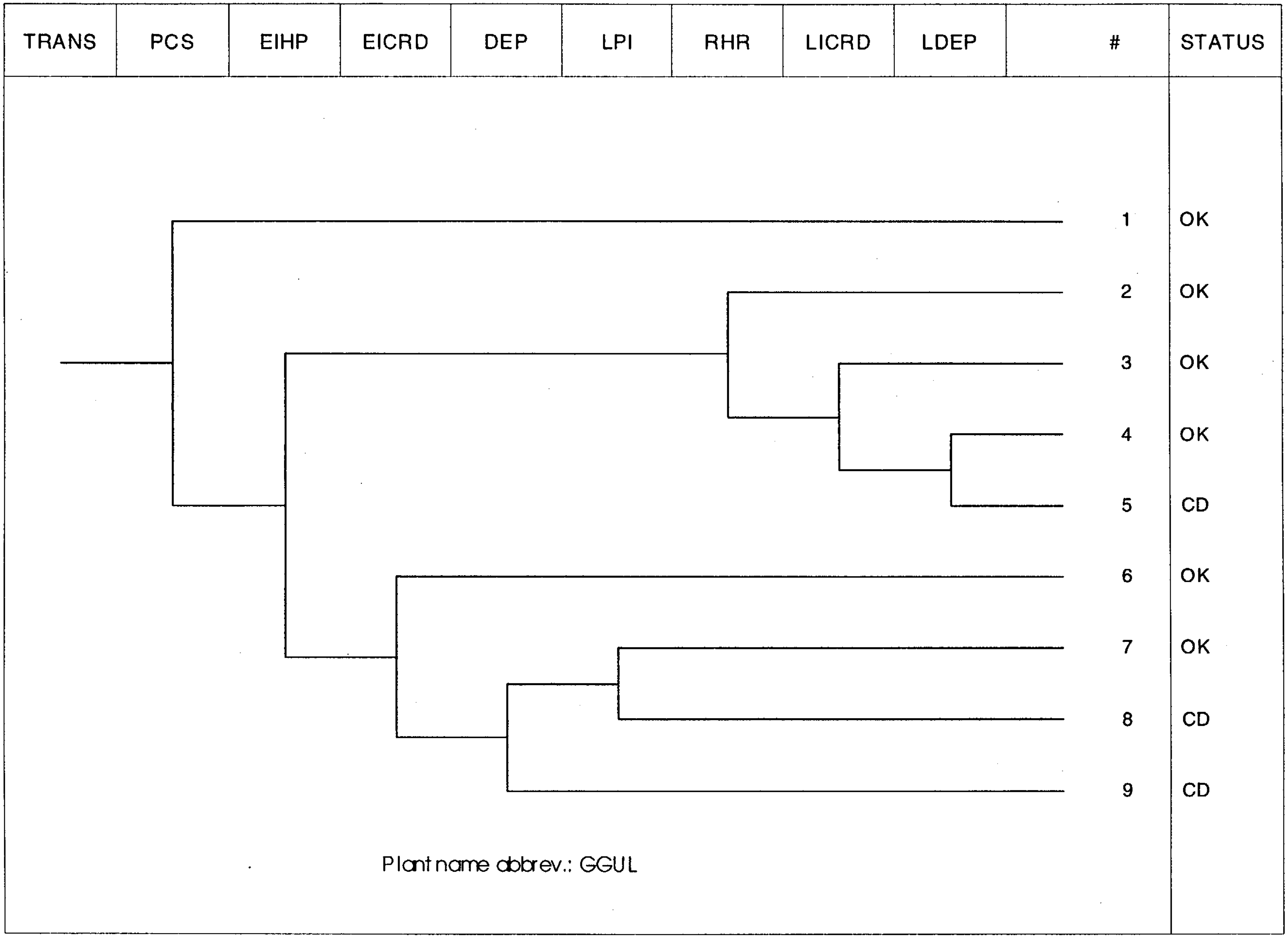
(3) The worksheets consider containment venting to be a high-stress operator action. GGNS IPE Table 3.3-7 assigns a failure probability of 2.4E-3 with an error factor of 10 for failure to bypass containment isolation within 45-60 minutes.

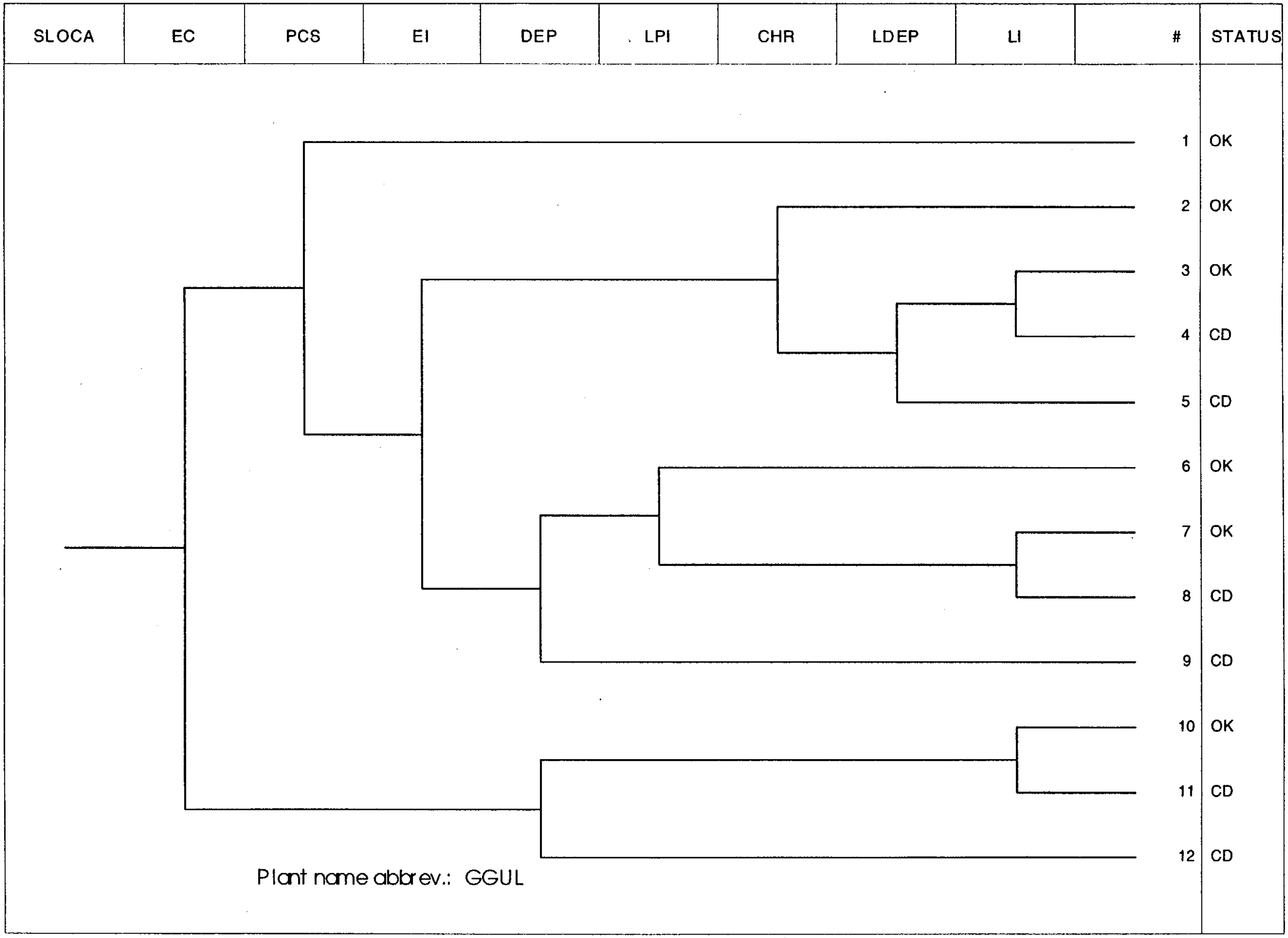
1.3 SDP Event Trees

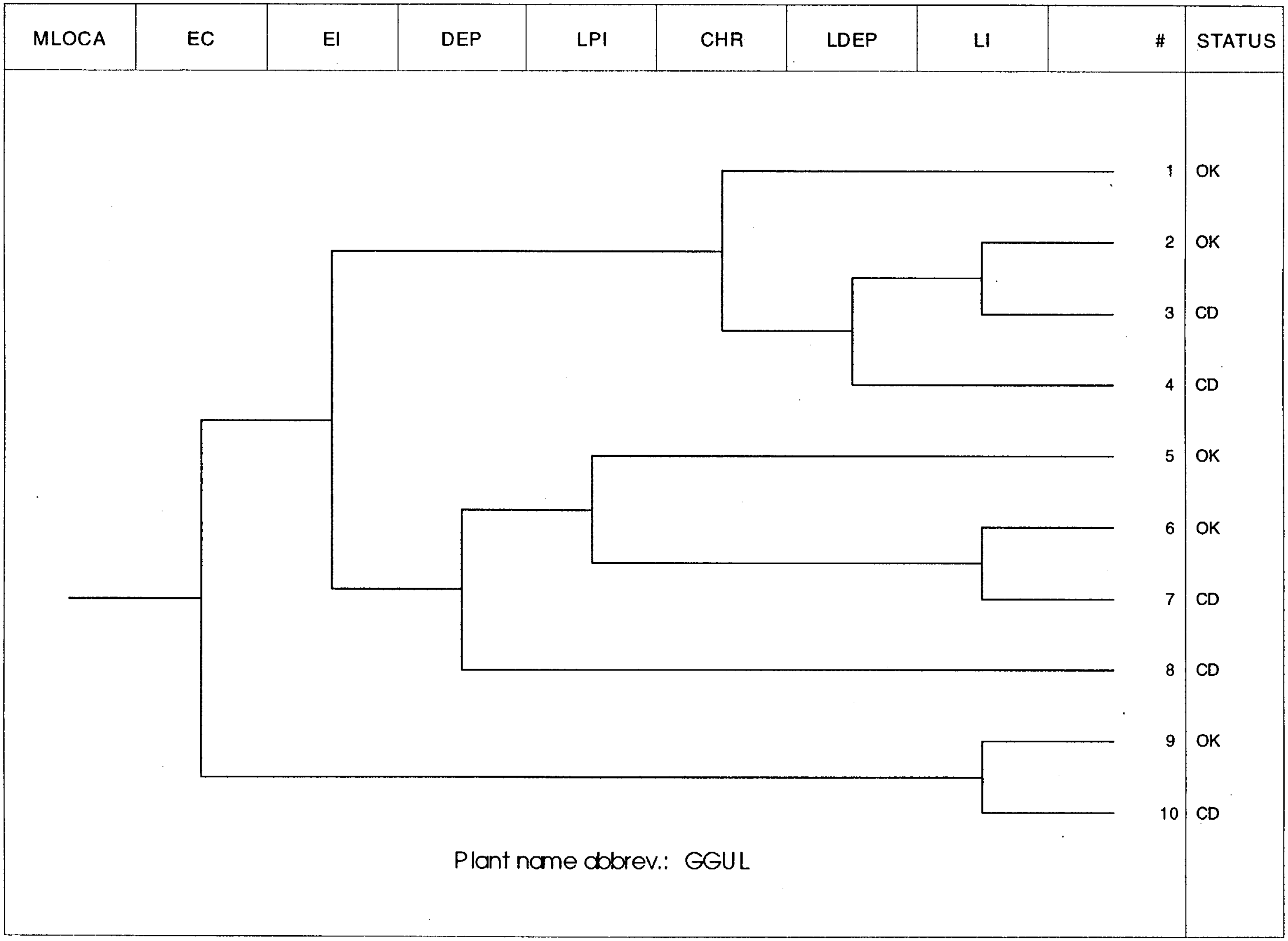
This section provides the simplified event trees, called SDP event trees, used to define the accident sequences identified in the SDP worksheets in the previous section. The event tree headings are defined in the corresponding SDP worksheets.

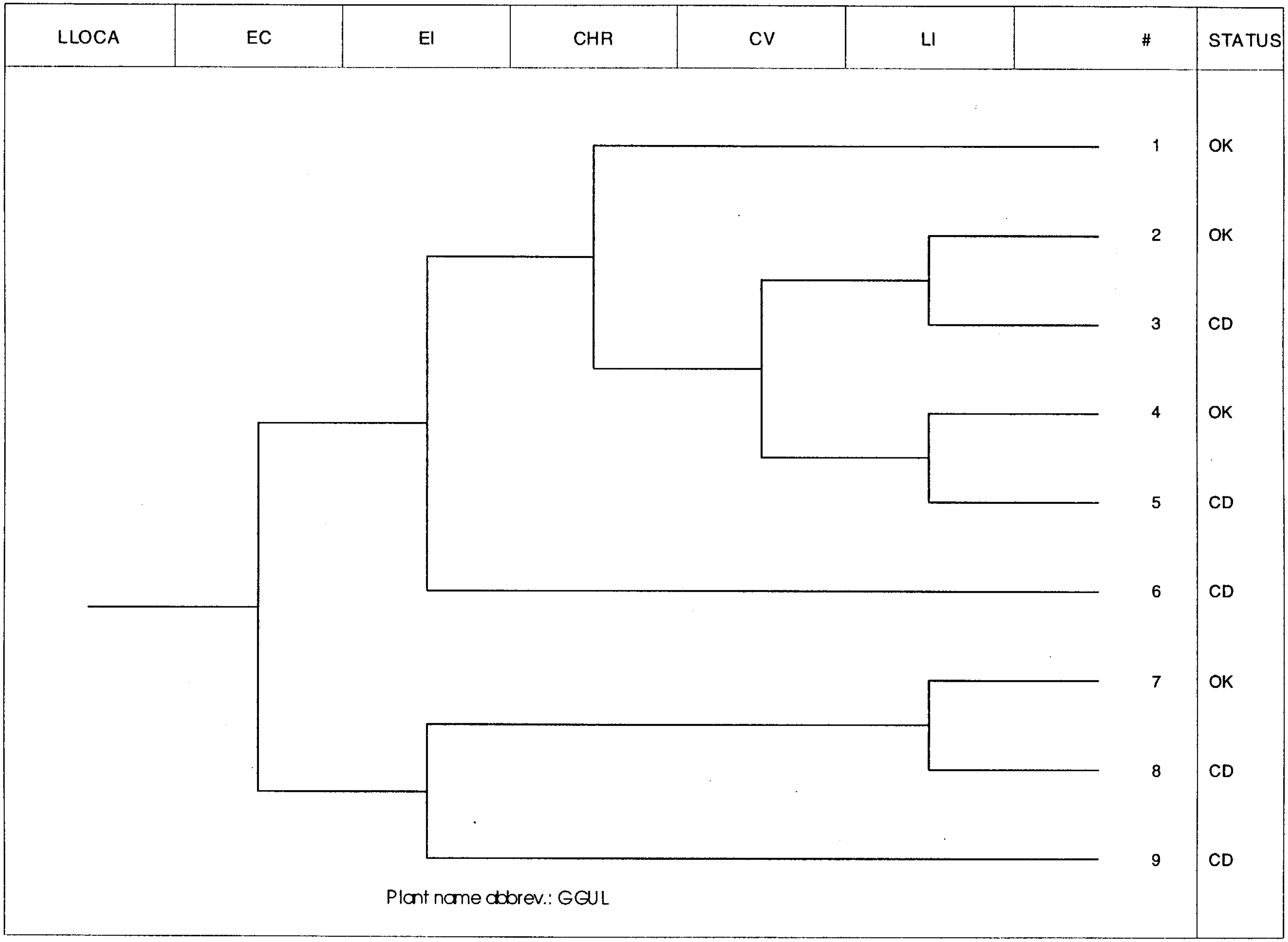
The following event trees are included:

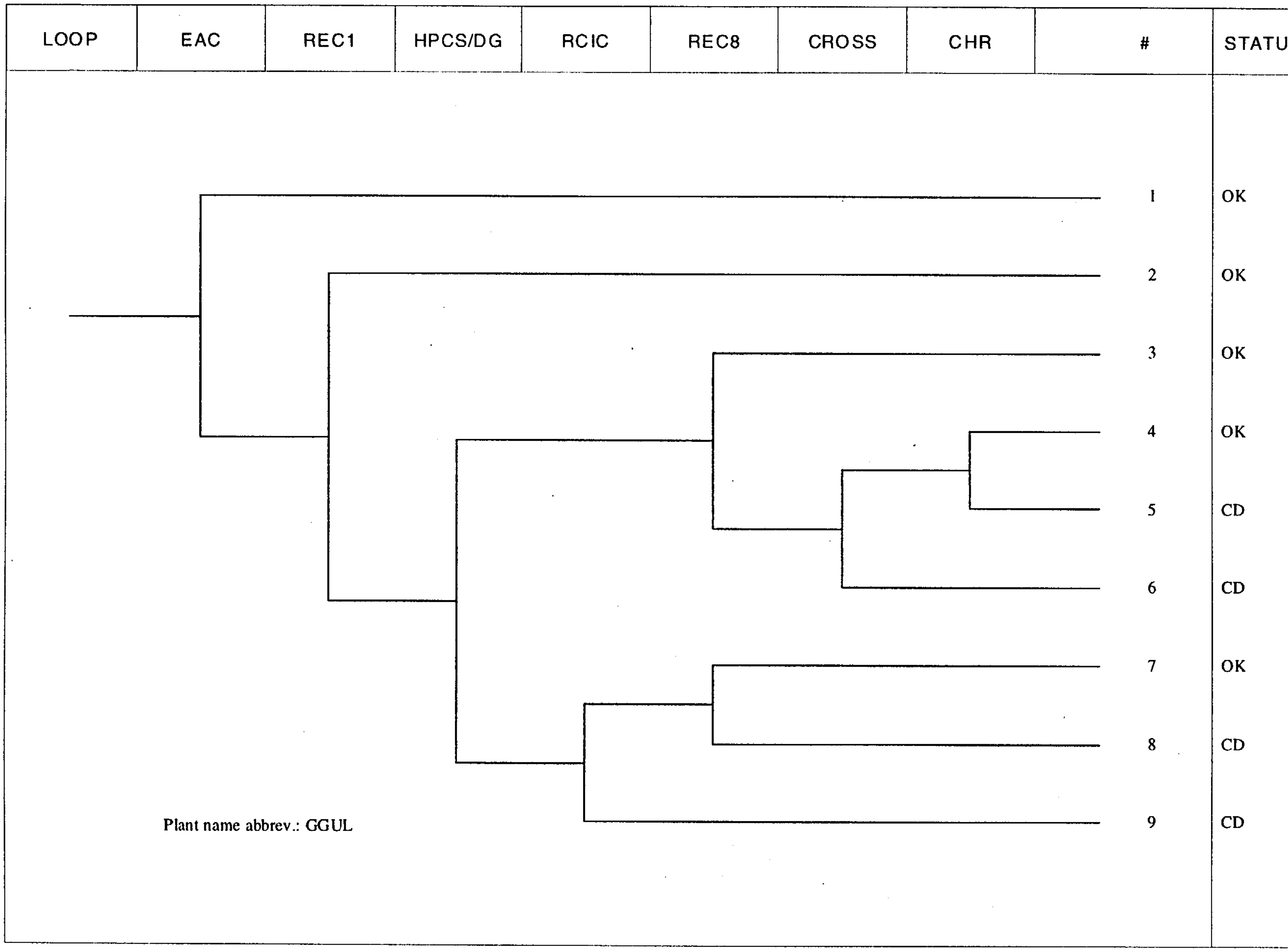
1. Transients
2. Small LOCA
3. Medium LOCA
4. Large LOCA
5. LOOP
6. Anticipated Transients Without Scram (ATWS)

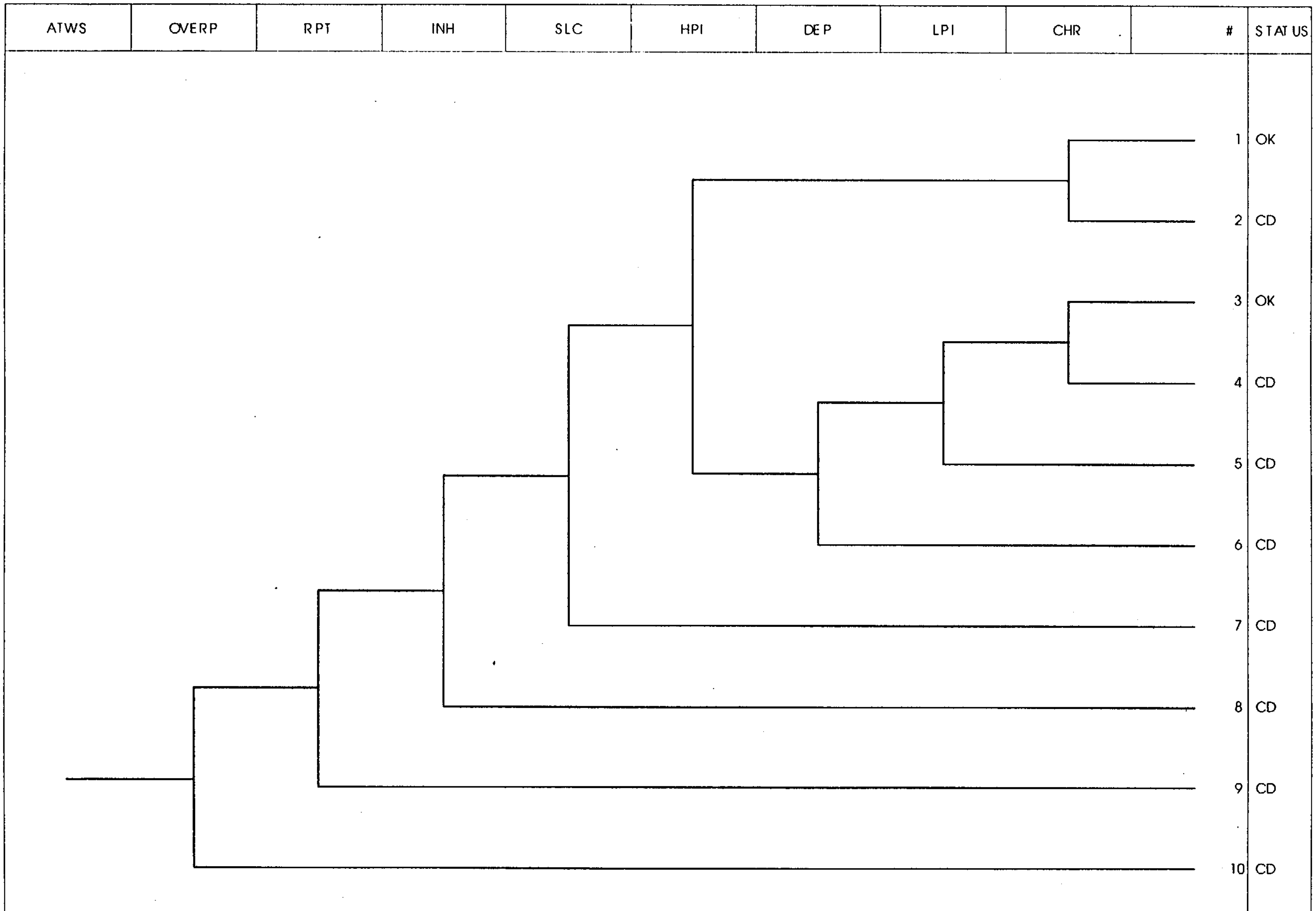

TRANS
PCS
EIHP
EICRD
DEP
LPI
RHR
LICRD
LDEP
#
STATUS
1 OK
2 OK
3 OK
4 OK
5 CD
6 OK
7 OK
8 CD
9 CD
Plant name abbrev.: GGUL


SLOCA
EC
PCS
EI
DEP
LPI
CHR
LDEP
LI
#
STATUS
1 OK
2 OK
3 OK
4 CD
5 CD
6 OK
7 OK
8 CD
9 CD
10 OK
11 CD
12 CD
Plant name abbrev.: GGUL


MLOCA
EC
EI
DEP
LPI
CHR
LDEP
LI
#
STATUS
1 OK
2 OK
3 CD
4 CD
5 OK
6 OK
7 CD
8 CD
9 OK
10 CD
Plant name abbrev.: GGUL


LLOCA
EC
EI
CHR
CV
LI
#
STATUS
1 OK
2 OK
3 CD
4 OK
5 CD
6 CD
7 OK
8 CD
9 CD
Plant name abbrev.: GGUL


LOOP
EAC
REC1
HPCS/DG
RCIC
REC8
CROSS
CHR
#
STATUS
1 OK
2 OK
3 OK
4 OK
5 CD
6 CD
7 OK
8 CD
9 CD
Plant name abbrev.: GGUL



Plant name abbrev.: GGUL

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. Entergy Operations, Inc., "Grand Gulf Unit 1 – Individual Plant Examination Report," dated February 1993.