Mr. Charles M. Dugger
Vice President Operations
Entergy Operations, Inc.
17265 River Road
Killona, LA 70066-0751

February 8, 2000

SUBJECT: WATERFORD STEAM ELECTRIC STATION, UNIT 3, SITE-SPECIFIC WORKSHEETS FOR USE IN THE NUCLEAR REGULATORY COMMISSION'S SIGNIFICANCE DETERMINATION PROCESS (TAC NO. MA6544)

Dear Mr. Dugger:

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 Waterford Steam Electric Station, Unit 3, (Waterford 3) 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 also 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 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 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 in the near future to discuss with your staff any changes that may be appropriate. We are not requesting written comments on the NRC's work product.

We will coordinate our efforts through your licensing or risk organizations as appropriate. If you have any questions, please contact me at 301-415-1480.

Sincerely,

/RA/

N. Kalyanam, Project Manager, Section 1
Project Directorate IV and Decommissioning
Division of Licensing Project management
Office of Nuclear Reactor Regulation

Docket No. 50-382

Enclosure: Risk-Informed Inspection Notebook

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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 301-415-1480.

Sincerely,

/RA/

N. Kalyanam, Project Manager, Section 1
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Division of Licensing Project management
Office of Nuclear Reactor Regulation

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RISK-INFORMED INSPECTION NOTEBOOK FOR

WATERFORD 3 NUCLEAR POWER PLANT

PWR, COMBUSTION ENGINEERING, TWO-LOOP PLANT WITH LARGE DRY CONTAINMENT

Prepared by

Brookhaven National Laboratory
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Prepared for

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

May 1999

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 Waterford 3 Nuclear 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.

CONTENTS

FIGURES

TABLES

1. INFORMATION SUPPORTING SIGNIFICANCE DETERMINATION PROCESS (SDP)

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

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

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

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

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

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

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

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

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

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

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

The three sections that follow include the initiators and dependency table, SDP worksheets, and the SDP event-trees for the Waterford 3 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 Waterford 3

Affected Systems	Major Components	Support Systems	Initiating Event
AC Power System	AC Power Distribution & AC Instrument Power	DC, HVAC	Transient, SLOCA, MLOCA, LLOCA, LOOP, SGTR, ATWS, RCP seal LOCA
EFW	2 MDPs, condensate storage pool (CSP)	AC, DC, ESFAS, HVAC	Transient, SLOCA, LOOP, SGTR, ATWS, RCP seal LOCA
	1 TDP, CSP	ESFAS, DC, main steam	
CCW	3 Pumps in two trains with one dry cooling tower and one CCW heat exchanger in each train	AC, DC, ESFAS, ACCW, HVAC, IA	Transient, SLOCA, MLOCA, LLOCA, LOOP, SGTR, ATWS, RCP seal LOCA
Auxiliary Component Cooling Water (ACCW)	2 Pumps and 2 wet cooling towers	AC,DC	Transient, SLOCA, MLOCA, LLOCA, LOOP, SGTR, ATWS, RCP seal LOCA
Condensate / MFW	3 Condensate pumps	AC	Transient, SLOCA
	2 SGFPs	AC, DC, IA, main steam	
	1 Auxiliary feedwater pump (AFW), condensate storage tank (CST)	AC, DC, IA	
Containment Cooling System (CCS)	4 Fan coolers	AC, ESFAS, CCW	SLOCA, MLOCA, LLOCA, RCP seal LOCA
Containment Spray System(CSS)	2 Trains, each with 1 pump	AC, DC, ESFAS, HVAC, IA, CCW	SLOCA, MLOCA, LLOCA, RCP seal LOCA

Affected Systems	Major Components	Support Systems	Initiating Event
HPSI	2 HPSI trains with a third swing train	AC, DC, ESFAS, CCW, HVAC	Transient, SLOCA, MLOCA, LOOP, SGTR, ATWS, RCP seal LOCA
Charging Pumps (CHG)	3 Pumps	AC, DC, IA, HVAC	SGTR, ATWS
DC Power System	Buses, battery chargers and batteries	AC Dist. (without AC, battery capacity is 4 hrs.)	Transient, SLOCA, MLOCA, LLOCA, LOOP, SGTR, ATWS, RCP seal LOCA
EDG	2 EDGs:	DC, HVAC, CCW	LOOP
HVAC	Area fan coolers and 3 essential service chilled water trains	AC, CCW, ESFAS?, DC	Transient, SLOCA, MLOCA, LLOCA, LOOP, SGTR, ATWS, RCP seal LOCA
Instrument Air (IA)	2 Air compressors	AC, DC, turbine building cooling water	Transient, SLOCA, MLOCA, LLOCA, LOOP, SGTR, ATWS, RCP seal LOCA
Main Steam	2 SGs, each with1 ARV, 6 safety valves, 1 MSIV and 3 turbine bypass valves	DC, IA, Vital AC	Transient, SLOCA, MLOCA, LLOCA, LOOP, SGTR, ATWS, RCP seal LOCA
Pressurizer Pressure Relief	2 Safety valves open at 2500 psia	none	Transient, LOOP, ATWS
RCP	Seals	1 / 3 CCW pumps to thermal barrier heat exchanger	RCP seal LOCA
Safety Injection Tank (SIT)	4 SITs	none	LLOCA
LHSI	2 LPSI pumps	AC, DC, ESFAS, CCW, HVAC	LLOCA

Notes:

(1) Plant internal event CDF (including internal floods) = 1.67 E-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 Waterford 3 Nuclear Plant. The SDP worksheets are presented for the following initiating event categories:

1. Transients
2. Small LOCA
3. Stuck Open PSV
4. Medium LOCA
5. Large LOCA
6. LOOP
7. Steam Generator Tube Rupture (SGTR)
8. Anticipated Transients Without Scram (ATWS)
9. Special Initiators

Table 2.1 SDP Worksheet for Waterford 3 Nuclear Plant   Transients

<table>
<tr><td colspan="4">Estimated Frequency (Table 1 Row) _________   Exposure Time _________   Table 1 Result (circle): A B C D E F G H</td></tr>
<tr><td><u>Safety Functions Needed</u>:</td><td colspan="3"><u>Full Creditable Mitigation Capability for Each Safety Function</u>:</td></tr>
<tr><td>Power Conversion System (PCS)</td><td colspan="3">1 / 2 Main Feedwater trains with 1 / 3 condensate trains or 1/1 AFW pump or depressurization with 1 / 2 ADVs or 1/6 TBVs and feed with 1/3 condensate pumps (operator action) (1)</td></tr>
<tr><td>Emergency Feedwater System (EFW)</td><td colspan="3">1 / 2 MD EFW trains (1 multi-train system) or 1 TD EFW train (1 ASD train)</td></tr>
<tr><td><u>Circle Affected Functions</u></td><td><u>Recovery of Failed Train</u></td><td><u>Remaining Mitigation Capability Rating for Each Affected Sequence</u></td><td><u>Sequence Color</u></td></tr>
<tr><td>1 TRANS - PCS - EFW (3)</td><td></td><td></td><td></td></tr>
<tr><td colspan="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.</td></tr>
</table>

<u>Notes:</u>

(1) The human error probability used in the IPE is 3.0E-4 for operator failure to recognize the need for feedwater. (Event OPER-2 on page 3.4-8.)

Table 2.2 SDP Worksheet for Waterford 3 Nuclear 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:**		
High Head Safety Injection (HPSI)	1 /3 high pressure injection pumps inject to 2 intact cold legs from refueling water storage pool (RWSP) (1 multi-train system)		
Main Feedwater (MFW)	1 / 2 MFW trains or 1AFW train (operator action) [(1)]		
Emergency Feedwater System(EFW)	1 / 2 MD EFW trains (1 multi-train system) or 1 TD EFW train (1 ASD train)		
High Pressure Recirculation (HPR)	1 / 3 HPSI pumps in recirculation mode (operator action) [(2)]		
Containment Heat Removal (CHR)	1/4 trains of fan coolers or 1 /2 trains of containment spray injection and recirculation (2 multi-train systems)		
Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 SLOCA - CHR (2,5)			
2 SLOCA - HPR (3,6)			
3 SLOCA - MFW - EFW (7)			
4 SLOCA - HPSI (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 human error probability used in the IPE is 3.0E-4 for operator failure to recognize the need for feedwater. (Event OPER-2 on page 3.4-8.)

(2) The IPE did not document the human error probability. Recirculation is actuated automatically. The operator has to manually close the RWSP discharge valves.

Table 2.3 SDP Worksheet for Waterford 3 Nuclear Plant **Stuck Open PORV (SORV)**[1]

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

<u>Safety Functions Needed</u>:	***<u>Full</u>* <u>Creditable Mitigation Capability for Each Safety Function</u>:**
Stuck Open PSV (SOSV)	1 / 2 PSV fail to reclose when demended (1 train system).
Isolation of Small LOCA (BLK)	Failure to re-close the PSV (Probability of 1)
High Head Safety Injection (HPSI)	1 /3 high pressure injection pumps inject to 2 intact cold legs from refueling water storage pool (RWSP) (1 multi-train system)
Main Feedwater (MFW)	1 / 2 MFW trains or 1AFW train (operator action) [2]
Emergency Feedwater System(EFW)	1 / 2 MD EFW trains (1 multi-train system) or 1 TD EFW train (1 ASD train)
High Pressure Recirculation (HPR)	1 / 3 HPSI pumps in recirculation mode (operator action) [3]
Containment Heat Removal (CHR)	1/4 trains of fan coolers or 1 /2 trains of containment spray injection and recirculation (2 multi-train systems)

<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 SOSV - BLK - HPSI (9)			
2 SOSV - BLK - MFW - EFW (8)			
3 SOSV - BLK - HPR (4,7)			

4 SOSV - BLK - CHR (3,6)			
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) The Waterfrod 3 plant does not have PORVs. The work sheet was developed to model the scenarios that started with a transient and a pressurizer safety valve failed to re-close.
(2) The human error probability used in the IPE is 3.0E-4 for operator failure to recognize the need for feedwater. (Event OPER-2 on page 3.4-8.)
(3) The IPE did not document this human error probability. Recirculation is actuated automatically. The operator has to manually close the RWSP discharge valves.

Table 2.4 SDP Worksheet for Waterford 3 Nuclear Plant Medium LOCA

Estimated Frequency (Table 1 Row) __________	Exposure Time __________	Table 1 Result (circle): A B C D E F G H	
<u>Safety Functions Needed</u>:	***<u>Full</u>* <u>Creditable Mitigation Capability for Each Safety Function</u>:**		
High Head Safety Injection (HPSI)	1/3 high pressure injection pumps inject to 2 intact cold legs from refueling water storage pool (RWSP) (1 multi-train system)		
High Pressure Recirculation (HPR)	1/3 HPSI pumps in recirculation mode (operator action) [(1)]		
Containment Heat Removal (CHR)	1/4 trains of fan coolers (multi-train systems) or 1 /2 trains of containment spray injection and recirculation (multi-train systems)		
<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 MLOCA - CHR (2)			
2 MLOCA - HPR (3)			
3 MLOCA - HPSI (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) The IPE did not document the human error probability. Recirculation is actuated automatically. The operator has to manually close the RWSP discharge valves.

Table 2.5 SDP Worksheet for Waterford Nuclear 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:**
Safety Injection Tank(SIT)	3/3 intact SITs inject into intact RCS legs (1 train system/high reliability)
High Head Safety Injection (HPSI)	1/3 high pressure injection pumps inject to 2 intact cold legs from refueling water storage pool (RWSP) (1 multi-train system)
Low Pressure Safety Injection(LPSI)	1 / 2 LHSI trains inject from RWSP to 1 intact cold leg (1 multi-train system)
High Pressure Recirculation (HPR)	1 / 3 HPSI pumps in recirculation mode (operator action) (1)
Containment Heat Removal (CHR)	1/4 trains of fan coolers (multi-train systems) or 1 /2 trains of containment spray injection and recirculation (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 - HPR (3)			
3 LLOCA - LPSI (4)			
4 LLOCA - HPSI (5)			

5 LLOCA - SIT (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) The IPE did not document the human error probability. Recirculation is actuated automatically. The operator has to manually close the RWSP discharge valves.

Table 2.6 SDP Worksheet for Waterford 3 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 Diesel Generator (EDG)	1 / 2 Emergency Diesel Generators (1 multi-train system)		
Turbine-driven EFW pump (TDEFW)	1 / 1 TDP trains of EFW (1 ASD train)		
Emergency Feedwater (EFW)	1 / 2 MDEFW trains (1 multi-train system) or 1 TDEFW train (1 ASD train)		
Recovery of AC Power in < 1 hrs (REC1)	(operator action under high stress) (1)		
Recovery of AC Power in < 6 hrs (REC6)	(operator action) (2)		
Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 LOOP - EFW (2,6)			
2 LOOP - EAC - REC6 (4)			
3 LOOP - EAC - TDEFW - REC1(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 IPE uses a probability of 0.286 for failure to recover in 50 minutes. (Event Z-LOOP-R0 on page 3.7-20)
(2) The IPE uses a probability of 0.186 for failure to recover in 6 hours. (Event Z-LOOP6)

Table 2.7 SDP Worksheet for Waterford 3 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:**
Pressure Equalization (EQ)	Operator isolates the ruptured SG and depressurizes RCS using 1 / 1 ADV or 1/6 SBVs and RCS pressurizer spray or blowdown the ruptured SG to less than setpoint of relief valves (high stress operator action) (1)
Power Conversion System (PCS)	1 / 2 Main Feedwater trains with 1 / 3 condensate trains or 1/1 AFW to the unaffected SG pump (operator action) (2)
Emergency Feedwater System (EFW)	1 / 2 MDPs of EFW (1 multi-train system) or 1 / 1 TDP of EFW (1 ASD Train) to the unaffected SG
RCS Inventory Makeup(RCSMU)	1 / 3 high head injection pumps (1 multi-train system) or 2/3 charging pumps (1 multi-train system)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 SGTR - EQ (6)			
2 SGTR - RCSMU (2,4)			
3 SGTR - PCS -EFW (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 IPE did not document the human error probability.
(2) The IPE did not document the human error probability. Recirculation is actuated automatically. The operator has to manually close the RWSP discharge valves.

Table 2.8 SDP Worksheet for Waterford 3 Nuclear Plant ATWS

Estimated Frequency (Table 1 Row) __________	Exposure Time __________	Table 1 Result (circle): A B C D E F G H	
<u>Safety Functions Needed</u>:	**<u>Full Creditable Mitigation Capability for Each Safety Function</u>:**		
Turbine Trip (TTP)	Manually trip the turbine (operator action)		
Emergency Feedwater (EFW)	1 / 2 MDEFW trains (1 multi-train system) or 1 TDEFW train (1 ASD train)		
Primary Safety Valves Open (SRVO)	2/2 SVs open (1 train)		
Emergency Boration (EB)	Operator conducts emergency boration using 1 / 3 charging pumps from RWSP or BAST (operator action) [1]		
Primary Safety Valves Reclose (SRVR)	2/2 SRVs reclose		
High Head Safety Injection (HPSI)	1/3 high pressure injection pumps inject to 2 intact cold legs from refueling water storage pool (RWSP) (1 multi-train system)		
High Pressure Recirculation (HPR)	1/3 HPSI pumps in recirculation mode (operator action) [2]		
Containment Heat Removal (CHR)	1/4 trains of fan coolers (multi-train systems) or 1 /2 trains of containment spray injection and recirculation (multi-train systems)		
<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 ATWS - TTP (9)			
2 ATWS - EFW (8)			

3 ATWS - SRVO (7)			
4 ATWS - RC (6)			
5 ATWS - SRVR - HPSI (5)			
6 ATWS - SRVR - HPR (4)			
7 ATWS - SRVR - CHR (3)			

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 IPE did not document the human error probability.
(2) The IPE did not document the human error probability. Recirculation is actuated automatically. The operator has to manually close the RWSP discharge valves.

Table 2.9 SDP Worksheet for Farley Nuclear Plant — Special Initiators

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:**		
Interfacing System LOCAs	none		
Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
Initiator: Internal failure of the RHR suction valves [(1)]			
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 frequency of interfacing LOCA is 4.86E-7 per year. It is assumed to lead to core damage.

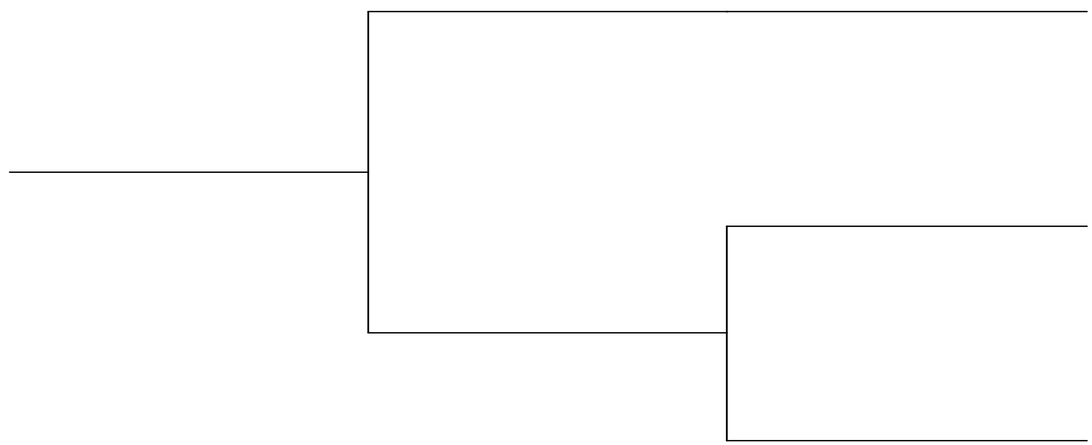
1.3 SDP Event Trees

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

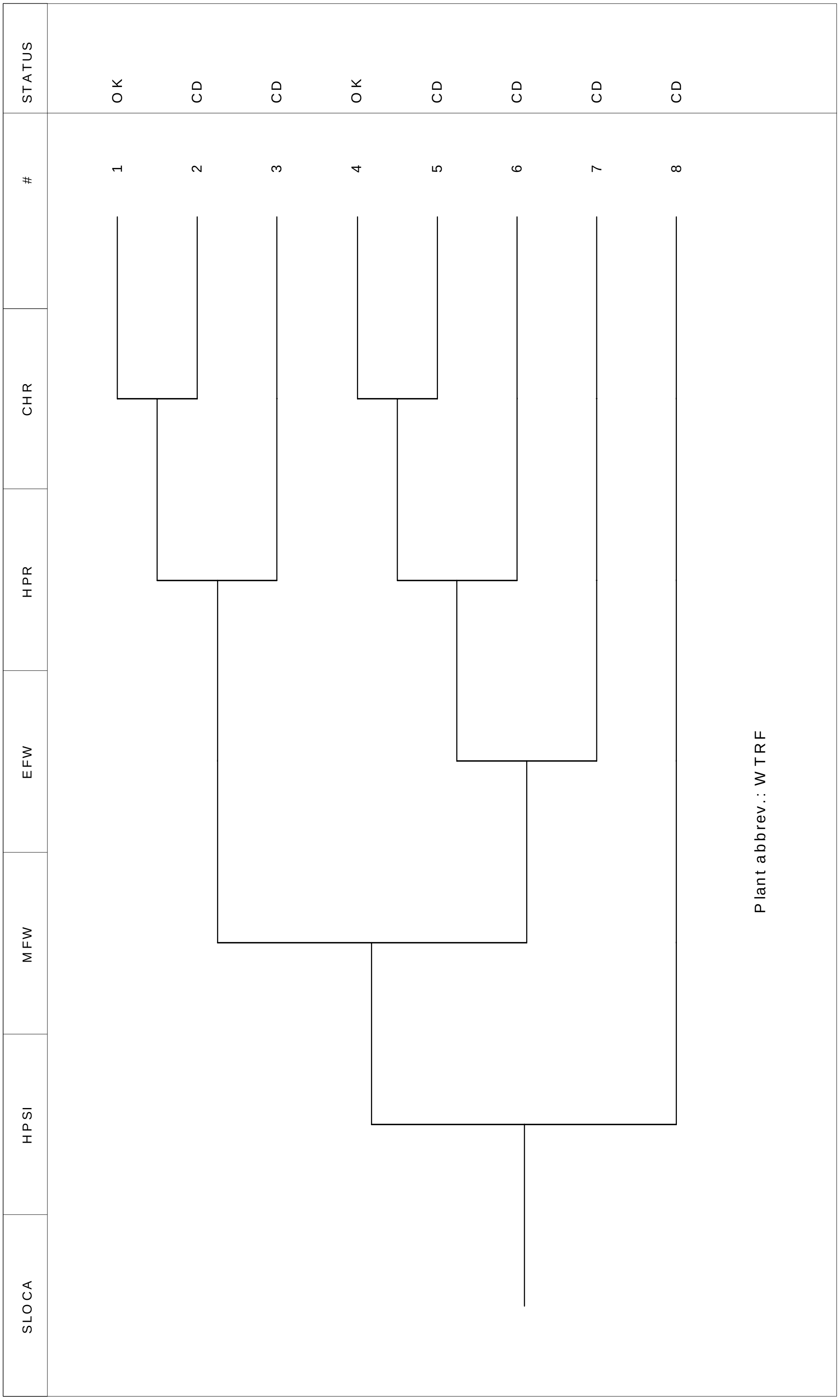
The following event trees are included:

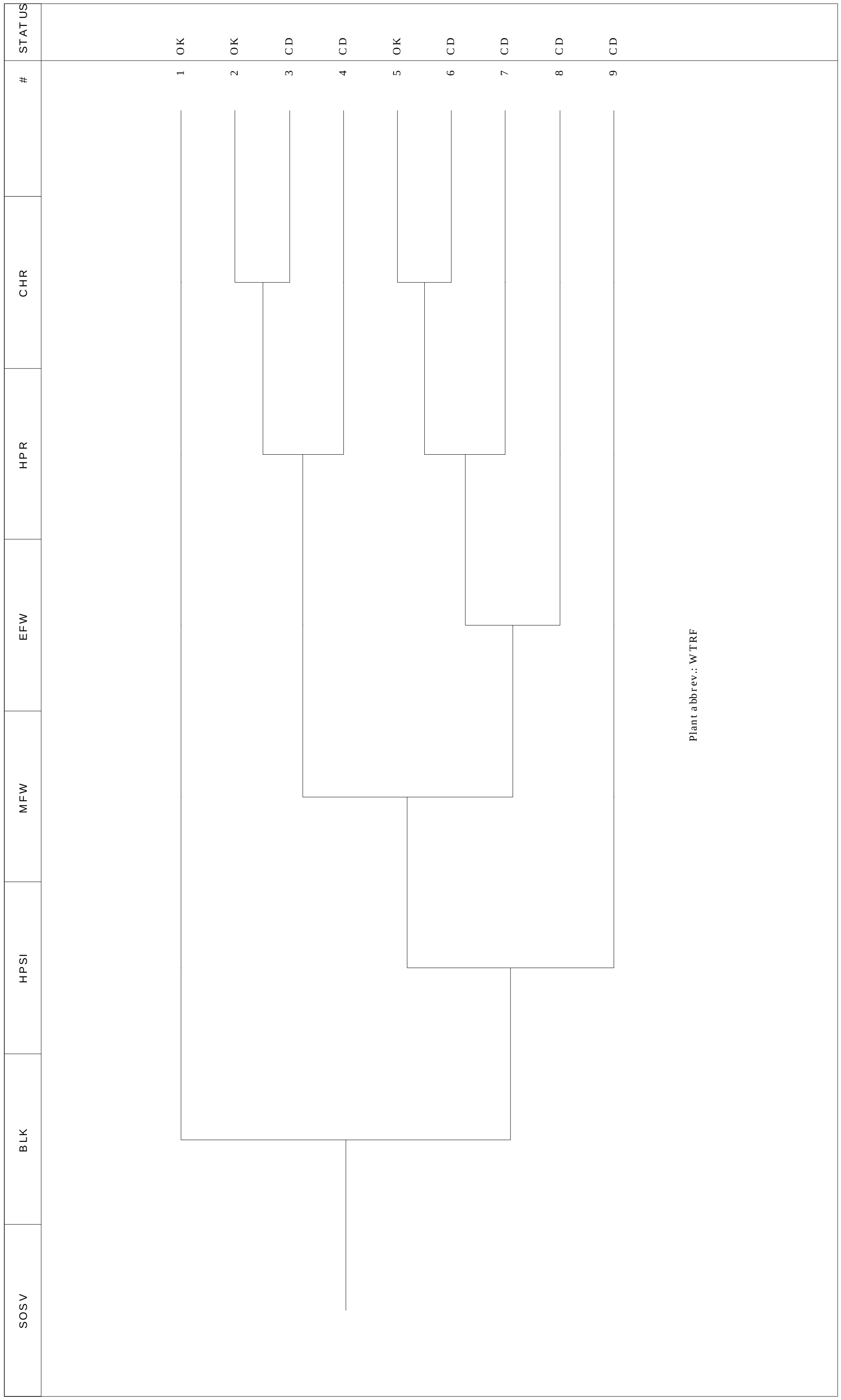
1. Transients
2. Small LOCA
3. Medium LOCA
4. Large LOCA
5. LOOP
6. Steam Generator Tube Rupture (SGTR)
7. Anticipated Transients Without Scram (ATWS)

TRANS	PCS	EFW	#	STATUS
			1	OK
			2	OK
			3	CD



Plant abbrev.: WTRF


SLOCA
HPSI
MFW
EFW
HPR
CHR
#
STATUS
1 OK
2 CD
3 CD
4 OK
5 CD
6 CD
7 CD
8 CD
Plant abbrev.: WTRF


STATUS
#
OK
OK
CD
CD
OK
CD
CD
CD
CD
1
2
3
4
5
6
7
8
9
CHR
HPR
EFW
MFW
HPSI
BLK
SOSV
Plant abbrev.: WTRF

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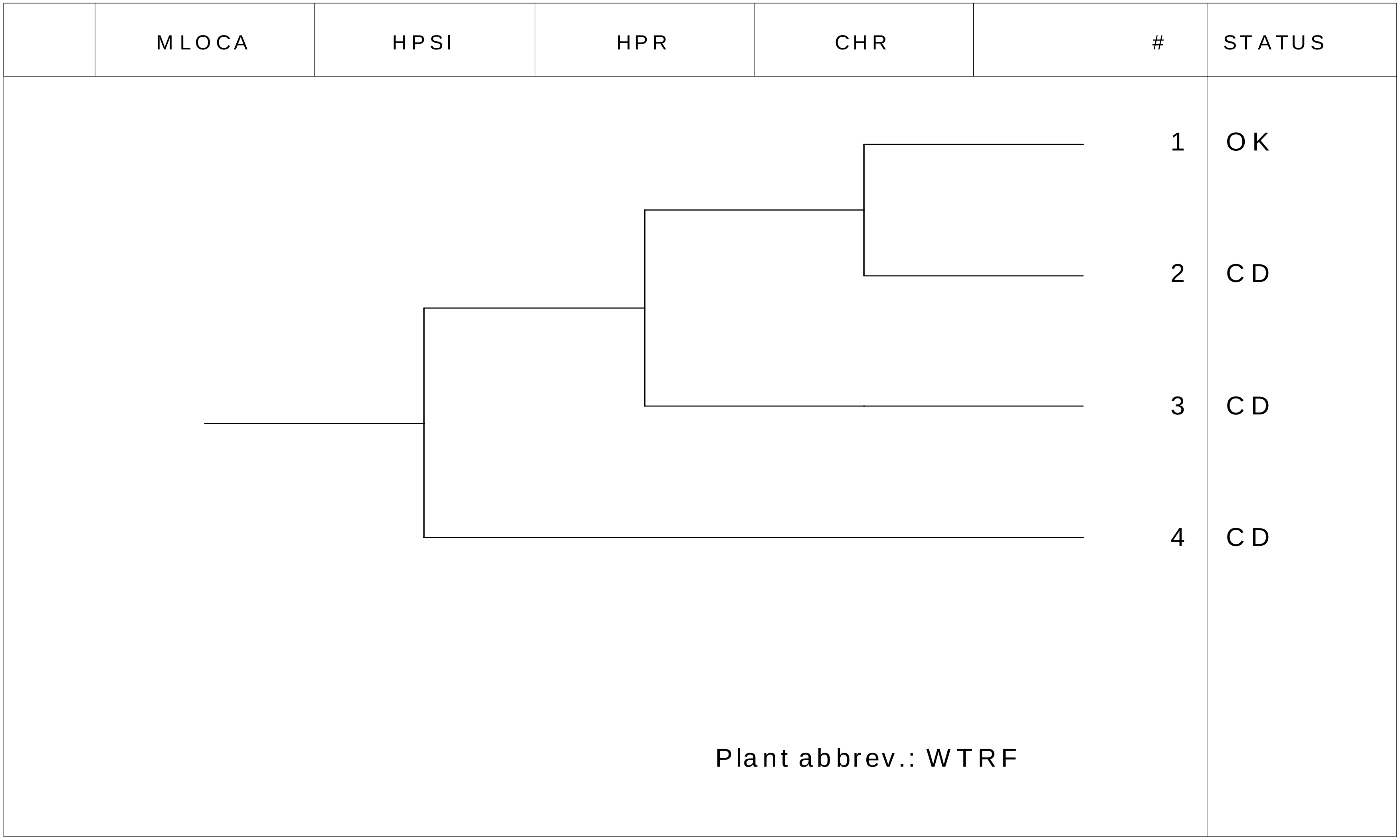
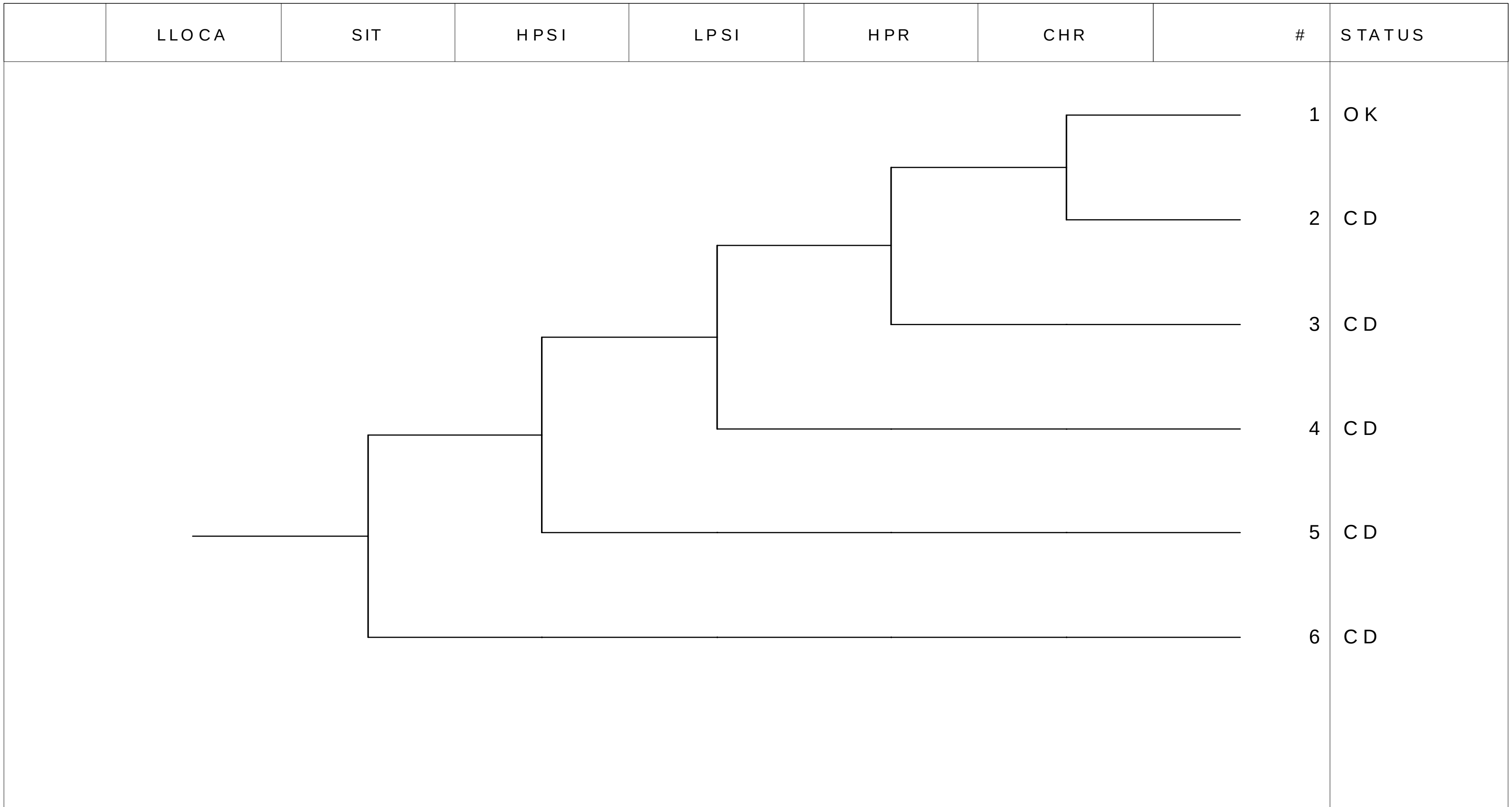
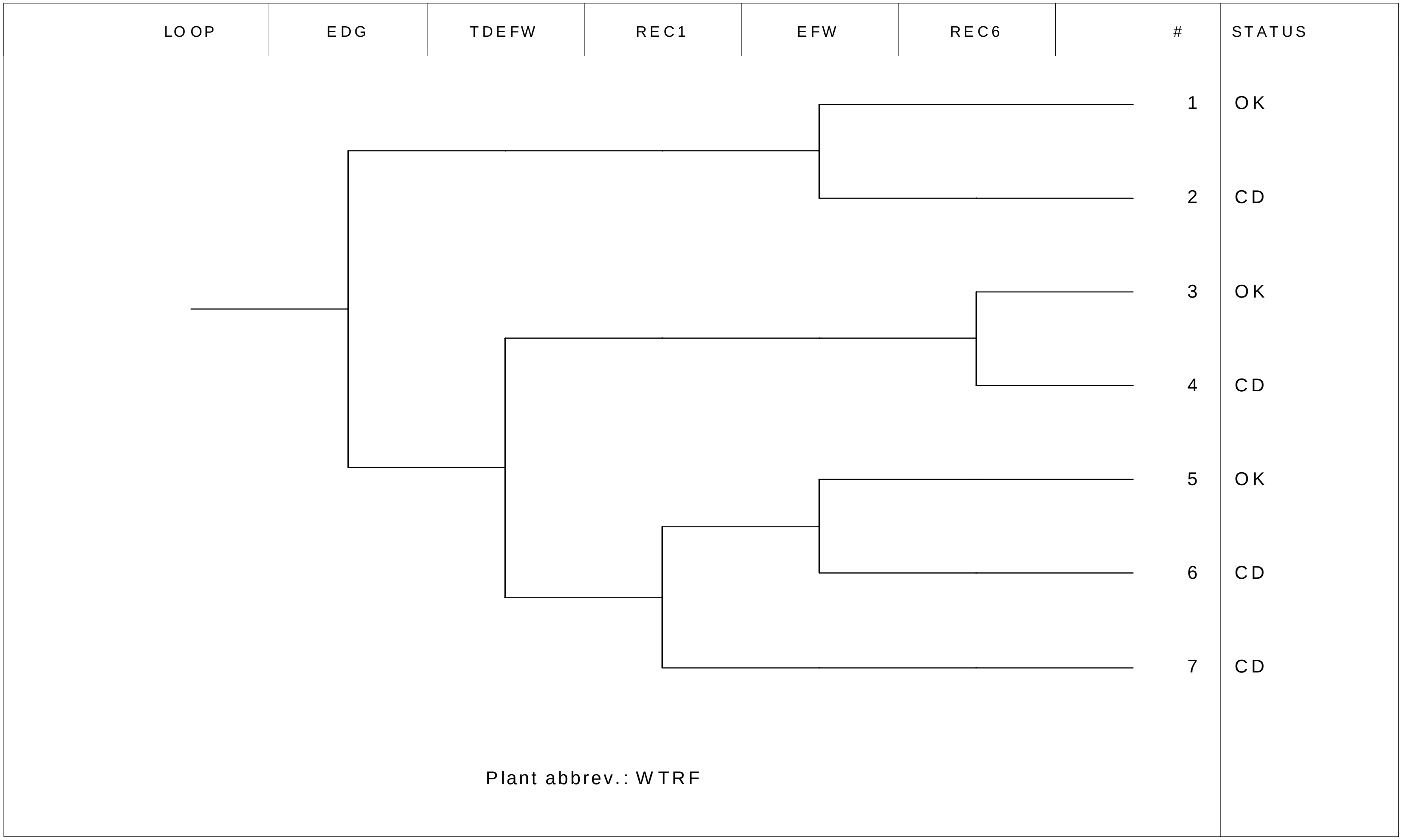
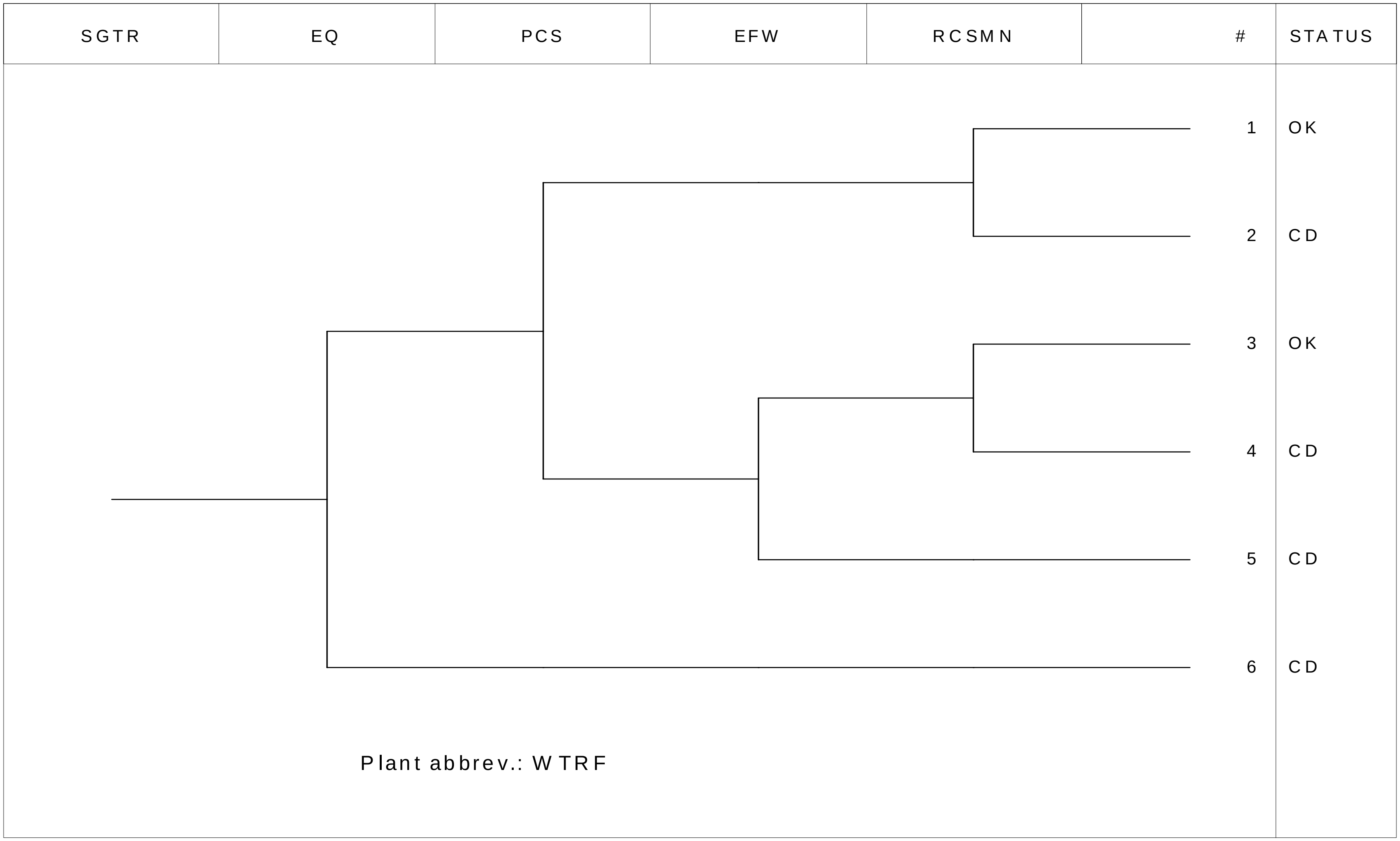
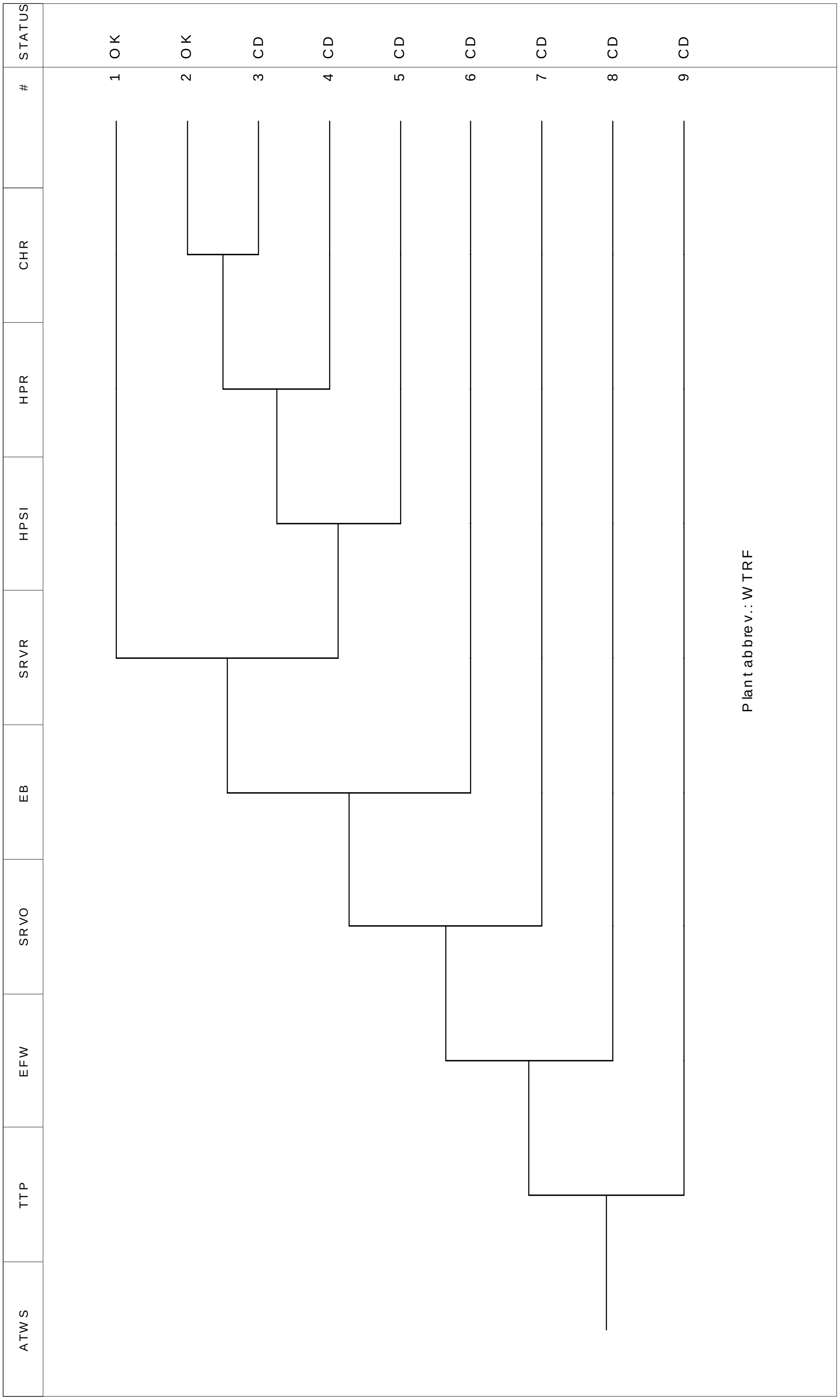

MLOCA
HPSI
HPR
CHR
#
STATUS
1 OK
2 CD
3 CD
4 CD
Plant abbrev.: WTRF



Plant abbrev.: WTRF


LOOP
EDG
TDEFW
REC1
EFW
REC6
#
STATUS
1 OK
2 CD
3 OK
4 CD
5 OK
6 CD
7 CD
Plant abbrev.: WTRF


SGTR
EQ
PCS
EFW
RCSMN
#
STATUS
1
OK
2
CD
3
OK
4
CD
5
CD
6
CD
Plant abbrev.: WTRF


STATUS
#
CHR
HPR
HPSI
SRVR
EB
SRVO
EFW
TTP
ATWS
1 OK
2 OK
3 CD
4 CD
5 CD
6 CD
7 CD
8 CD
9 CD
Plant abbrev.: WTRF

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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. Waterford Steam Electric Station, Unit 3,IPE submittal,