Mr. C. Randy Hutchinson January 24, 2000
Vice President, Operations ANO
Entergy Operations, Inc.
1448 S. R. 333
Russellville, AR 72801

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

Dear Mr. Hutchinson:

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 Arkansas Nuclear One, Units 1 and 2 (ANO-1 & 2) 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 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 301-415-1320.

Sincerely,

/RA/

M. Christopher Nolan, Project Manager, Section 1
Project Directorate IV & Decommissioning
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-313 and 50-368

Enclosures: As Stated

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

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RISK-INFORMED INSPECTION NOTEBOOK FOR

ARKANSAS NUCLEAR ONE

UNIT 1

PWR, BABCOCK & WILCOX, TWO-LOOP PLANT WITH LARGE DRY CONTAINMENT

Prepared by

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Prepared for

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Office of Nuclear Regulatory Research
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NOTICE

This notebook was developed for the NRC's inspection teams to support risk-informed inspections. The activities involved in these inspections are discussed in “Reactor Oversight Process Improvement,” SECY-99-007A, March 1999. The user of this notebook is assumed to be an inspector with an extensive understanding of plant-specific design features and operation. Therefore, the notebook is not a stand-alone document, and may not be suitable for use by non-specialists. This notebook will be periodically updated with new or replacement pages incorporating additional information on this plant. Technical errors in, and recommended updates to, this document should be brought to the attention of the following person:

Mr. Jose G. Ibarra
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ABSTRACT

This notebook contains summary information to support the Significance Determination Process (SDP) in risk-informed inspections for the Arkansas Nuclear One, Unit 1.

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

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

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1. INFORMATION SUPPORTING SIGNIFICANCE DETERMINATION PROCESS (SDP)

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

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

The initiator and system dependency table shows the major dependencies between front-line- and support-systems, and identifies their involvement in different types of initiators. The information in this table identifies the most risk-significant front-line- and support-systems; it is not an exhaustive nor comprehensive compilation of the dependency matrix as known in Probabilistic Risk Assessments (PRAs). 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 Arkansas Nuclear One, Unit 1.

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 Arkansas Nuclear One, Unit 1

Affected Systems	Major Components	Support Systems	Initiating Event Scenarios
EFWS	EFWTDP EFWMDP	125 VDC,4.16 KV, 480VAC 15v-DC Red (for EFIC)	Transient, LOOP, MSLB/MFLB (outside Cont.), SGTR, SLOCA, ATWS
HPI	Pumps,Valves I&C including DC for 4.16 KV Breakers	4.16 KV, 125 VDC (Train A uses red bus, train B uses green),SW (Train I for A train, Train II for B train)	Transient, LOOP, MSLB/MFLB (outside Cont.), SGTR, SLOCA, ATWS
HPI (Recirc.)	Pumps,Valves I&C including DC for 4.16 KV breakers	4.16 KV and 480 VAC (A:red, B:green), 125 VDC (A:red, B: green),SW (A: Train I, B:Train II), DHR heat exchanger, and LPI pumps	Transient, LOOP, MSLB/MFLB (outside Cont.), SGTR, SLOCA, ATWS
LPI/DHR (Recirc.)	Pumps, Valves, I&C including DC for 4.16 KV breakers	4.16 KV (A: red, B: green), 125 VDC (A:red, B: Green), SW (A: Train I, B: Train II, not needed for injection)	Transient, LOOP, MSLB/MFLB (outside Cont.), SGTR, SLOCA, ATWS, MLOCA, and LLOCA
CS (Recirc.): Reactor Bldg. (RB) Spray	Pumps, Heat Exch., Valves	4.16KV (A:red, B:green), 125 VDC (A: red, B: green), SW (A: Train I, B: Train II), DHR heat Exch., and LPI pumps	Transient, LOOP, MSLB/MFLB (outside Cont.), SGTR, SLOCA, ATWS, MLOCA, and LLOCA
EDG	Cooling (unit-1 only), HVAC, Start system, Fuel System	Service Water (Red: Train I, Green: Train II), 125 VDC	LOOP
RB Fan Cooling	Compressors, HXs, Valves	4.16 KV (train A: red, Train B: green), 125 VDC (Train A: red, Train B: green), SW (train A: SW I, Train B: SW II)	Transient, LOOP, MSLB/MFLB (outside Cont.), SGTR, SLOCA, ATWS, MLOCA, and LLOCA
ICW (Intermediate Cooling Water)	Pumps, Valves, Heat Exch.	AC from green bus, SW I for train A, SW II for train B	Transient, LOOP, MSLB/MFLB (outside Cont.), SGTR, SLOCA, ATWS, MLOCA, and LLOCA, ATWS

Table 1 (Continued)

Affected Systems	Major Components	Support Systems	Initiating Event Scenarios
Service Water System	Pumps and Valves	4.16KV/480 VAC (red and green for trains I and II), 125VDC (red and green for trains I&II), ICW (for trains I&II only)	Transient, LOOP, MSLB/MFLB (outside Cont.), SGTR, SLOCA, ATWS, MLOCA, and LLOCA, ATWS
CFT (Core Flood Tanks)	2 Passive tank trains	NA	MLOCA and LLOCA
ADVs (SG PORV)	Valves (MOVs)	480 VAC (Red and Green)	Transient, LOOP, MSLB/MFLB (outside Cont.), SGTR, SLOCA, ATWS, MLOCA, and LLOCA, ATWS
ERV	Valve	125 VDC Red Division (fail closed on loss of DC), and 480 VAC green for the block valve	Transient, LOOP, MSLB/MFLB (outside Cont.), SGTR, SLOCA, ATWS, MLOCA, and LLOCA, ATWS
RCP	Seals	ICW to RCP cooling or HPI seal injection to RCP seals or operator trips RCPs within 30 minutes of loss of seal cooling and injection	Transients. LOOP, SLOCA from RCP.

Plant CDF is 4.7E-5 per year based on IPE submittal Dated 4/29/93.

1.2 SDP WORKSHEETS

This section presents the SDP worksheets to be used in the Phase 2 evaluation of the inspection findings for the Arkansas Nuclear One, Unit 1. The SDP worksheets are presented for the following initiating event categories:

1. Transients
2. Small LOCA
3. Stuck-open PORV (Not applicable to this plant)
4. Medium LOCA
5. Large LOCA
6. LOOP
7. Steam Generator Tube Rupture (SGTR)
8. Anticipated Transients Without Scram (ATWS)
9. Main Steam Line Break (MSLB)

Table 2.1 SDP Worksheet for Arkansas Nuclear One, Unit 1 —— Transients

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>:**
Power Conversion System (PCS)	1 / 2 Feedwater[1] trains with 1 / 3 condensate trains (Operator action)
Secondary Heat Removal (EFW)	1 / 1 MDEFW trains (1 train) or 1 TDEFW train (1 ASD train)
High Pressure Injection (EIHP)	1 / 3 HPI trains from BWST (1 multi-train system)
Primary Heat Removal, Feed/Bleed (FB)[2]	1 / 1 ERVs or 1 / 2 SRVs open for Feed/Bleed and initiate HPI cooling (operator action)
High Pressure Recirculation (HPR)	1 / 2 HPI trains[3] taking suction from 1/2 LPI trains through LPI HX[4] (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 TRANS - PCS - EFW - FB (6)			
2 TRANS - PCS - EFW -EIHP (5)			
3 TRANS - PCS - EFW - HPR (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) ANO1-1 also could rely on a non safety AFW pump to be used for FW purpose after scram which requires operator action for alignment.

(2) The human error for initiation of HPI for FB is 1.2E-2 in IPE therefore it is assigned as an operator action.

(3) The third HPI train could be realigned in the recirculation mode manually if any of other two is not available.

(4) The IPE also credits 2/4 RBFC if the LPI HX is not available.

Table 2.2 SDP Worksheet for Arkansas Nuclear One, Unit 1 —— Small 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)[2]	1/3 HPI trains (1 multi-train systems)
Power Conversion System (PCS)[3]	1/2 feedwater trains and 1/3 Condensate pump (operator action)
Secondary Heat Removal (EFW)	1 / 1 MDEFW trains (1 train) or 1 TDEFW train (1 ASD train)
Primary Bleed (FB)	1 / 1 ERVs or 1 / 2 SRVs open for Feed/Bleed and initiate HPI cooling (High stress operator action)
High Pressure Recirc (HPR)	1 / 2 HPI trains taking suction from 1/2 LPI trains through LPI HX (operator action)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 SLOCA - EIHP (3,6,9)			
2 SLOCA - HPR (2,5,8)			
3 SLOCA - EFW -PCS- FB [4] (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) The IPE defines one category for small LOCA with break sizes greater than 3/8 of inch to 4.3 inches. Small LOCA model in SDP is developed for more demanding portion of break sizes from 3/8 of inches to 1.9 inches where secondary cooling is required. For breaks less than one inch single phase cooling using natural circulation would take place. For breaks greater than one inch up to 1.9 inches the natural circulation would be interrupted, vessel would void, and secondary cooling would be by the boiler condenser mode (reflux cooling). Also included in this category of small LOCA is the LOCAs through RCP seals. Important to note that loss of SW would result in loss of both HPI and ICW and if the operator fails to trip the RCPs in 30 minutes, small LOCA would occur.

(2) For FB, the HPI interruption and restoration is considered as a high stress operator action in the SDP sheet, however included under F&B function.

(3) ANO1-1 also could rely on a non safety AFW pump to be used for FW purpose after scram which requires operator action for alignment

(4) For break sizes greater than 1.9" up to 4.3 inches the flow through the break is sufficient for decay heat removal and the operation of Secondary heat removal of F&B is not required. That is the third sequence in SDP may not be considered core damage for these break sizes.

Table 2.3 SDP Worksheet for Arkansas Nuclear One, Unit 1 —— Medium LOCA[(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*** **Creditable Mitigation Capability for Each Safety Function**</u>:
Early Inventory, HP Injection (EIHP)	1/3 HPSI trains (1 multi-train systems)
Early Inventory Control (EIAC)	1/2 CFTs (1 multi-train system)
Low Pressure Injection (EILP)	1/2 LPI train (1 multi-train system)
Containment Heat Removal (CNT)	1/2 LPI HX. Cooling sump water (operator action) or 2/4 RBFC (1 multi-train system)
Low Pressure Recirculation (EILR)	1/2 LPI train taking suction from sump (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 MLOCA - EIHP-EIAC (9)			
2 MLOCA - EILP (4,8)			
3. MLOCA - EILR (2,6)			
4. MLOCA - CNT(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.

Note:

(1) Medium LOCA is defined for break sizes greater than 4.3 inches up to 10 inches.

Table 2.4 SDP Worksheet for Arkansas Nuclear One, Unit 1 —— Large 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>:**
Early Inventory Control (EIAC)	1/2 CFTs (1 multi-train system)
Containment Press/Temp Control (CNT)	1/2 LPI HX. Cooling sump water (operator action) or 2/4 RBFC (1 multi-train system)
Early Inventory, LP Injection (EILP)	1/ 2 LPI pump trains (1 multi-train system).
Low Pressure Recirc (EILR)	1 / 2 LPI pump trains taking suction from containment sump (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 LLOCA - EILP (4)			
2 LLOCA - EILR (2)			
3. LLOCA - CNT (3)			
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.

Table 2.5 SDP Worksheet for Arkansas Nuclear One, Unit 1 —— LOOP

Estimated Fequency (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 EDGs (1 multi-train system)(1)
Turbine-driven EFW pump (TDEFW)	Operation of TDEFW pump (1 ASD train)(2)
Recovery of AC Power in < 1 hrs (REC1)	SBO procedure and Recovery of an AC source in one hour (3) (high stress operator action)
Recovery of AC Power in < 3 hrs (REC3)	SBO procedure and Recovery of an AC source in three hours (4) (operator action)
Early Inventory, HP Injection (EIHP)	1/3 HPI trains (1 multi-train system)
Primary Heat Removal (FB)	1/1 ERVs or 1/2 SRVs open for Feed/Bleed and initiate HPI cooling (operator action)
High Pressure Recirculation (HPR)	1/2 HPI trains taking suction from 1/2 LPI trains through LPI HX (operator action)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 LOOP - EAC - REC3 (7) (failure to recover AC in 3 hours)			
2 LOOP - EAC - TDEFW-REC1 (12)			
3 LOOP - EAC - REC1 - EIHP (6)			
4 LOOP - EAC - REC1 - HPR (4)			

5 LOOP - EAC - REC1 - FB (5)			
6 LOOP - EAC - TDAFW - EIHP (11)			
7. LOOP - EAC - TDAFW - FB (10)			
8. LOOP - EAC - TDAFW - HPR (9)			

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

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

<u>Notes</u>:

(1) EDG day tank has sufficient fuel for one hour and requires fuel transfer pump to operate. Failure of room cooling assumes to fail the EDGs.

(2) The batteries are assumed to be depleted in two hours.

(3) Core damage is assumed to occur in one hour if no secondary heat removal.

(4) Core damage is assumed in one hour after battery depletion.

Table 2.6 SDP Worksheet for Arkansas Nuclear One, Unit 1 —— SGTR

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>:**
Early Inventory HP injection (EIHP)	**1/3 HPSI trains (1 multi-train system)**
Power Conversion System (PCS)	**1/2 feedwater trains and 1/3 Condensate pump: extended operation requires throttling of flow to allow depressurization (high stress operator action)**
Secondary Heat Removal (EFW)	1/1 MDEFW (1 train) or 1/1 TDEFW (1 ASD train)
Feed and Bleed (FB)	1 / 1 ERVs or 1 / 2 SRVs open for Feed/Bleed and initiate HPI cooling (High stress operator action)
High Pressure Recirculation (HPR)	1 / 3 HPI trains taking suction from 1/2 LPI trains through LPI HX[4] (operator action)
Shutdown Cooling (SDC)	1/2 LPI in DHR cooling mode (operator action)
Primary/Secondary Pressure Equalization and Cool down (ISO/EQ)	Pressure equalization below SG safety setpoints ;assumes secondary cooling available for rapid cool down (High stress 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 SGTR - EIHP (2,5,8,11)			
2 SGTR - ISO/EQ - SDC (4,7)			
3 SGTR - PCS - EFW - FB[1] (10)			
4 SGTR - PCS - EFW - HPR[1] (9)			

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

Note:

(1) Failure of PCS and EFW implies failure of ISO/EQ. Therefore in sequences (3) and (4) failure to equalize should be considered with the probability of one.

Table 2.7 SDP Worksheet for Arkansas Nuclear One, Unit 1 —— 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:**
Turbine trip (TTP)	Operator trips the turbine (operator action)
Emergency Boration (HPI)	Operator conducts emergency boration using 1 / 3 HPI pump (operator action)(1)
Secondary Heat Removal (EFW)	1 / 1 EFWMDP (1 train) or 1 / 1 EFWTDP (1 ASD Train) with 5 out of 8 SRVs on the intact SG
Long Term Cooling (LTC)	1/2 LPI in DHR mode or 1/3 HPI in Recirculation mode (1 multi-train system)
Primary Relief (SRV)	2 / 2 SRVs with 1/1 ERV open (1 train)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 ATWS - SRV (5)			
2 ATWS - EFW (4)			
3 ATWS - HPI (3)			
4 ATWS - TTP (6)			
5 ATWS - LTC (2)			

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:

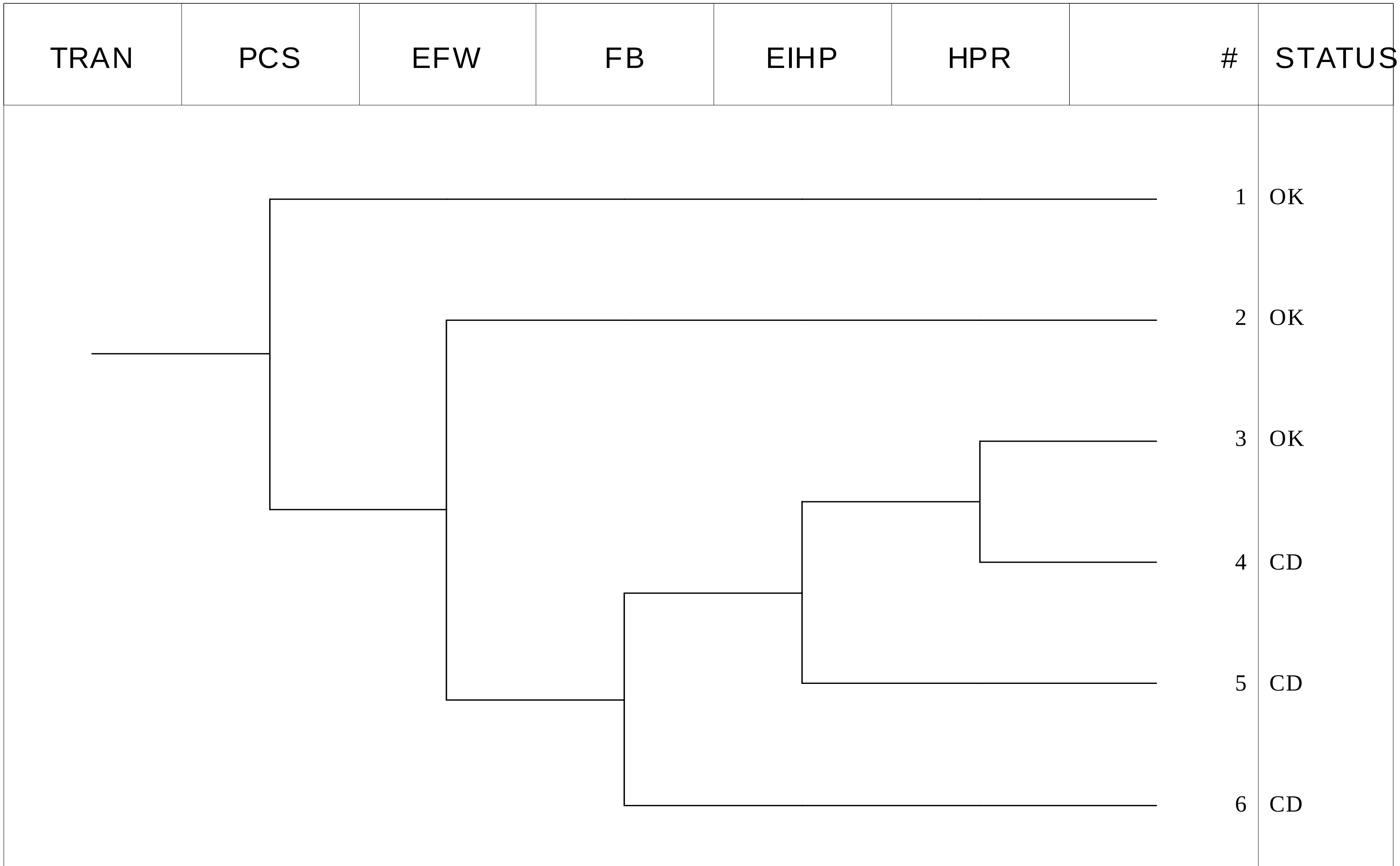
1. The operator action related to initiation of emergency oration has a value of 4.5E-3 in IPE, therefore is assigned as an operator action.

1.3 SDP Event Trees

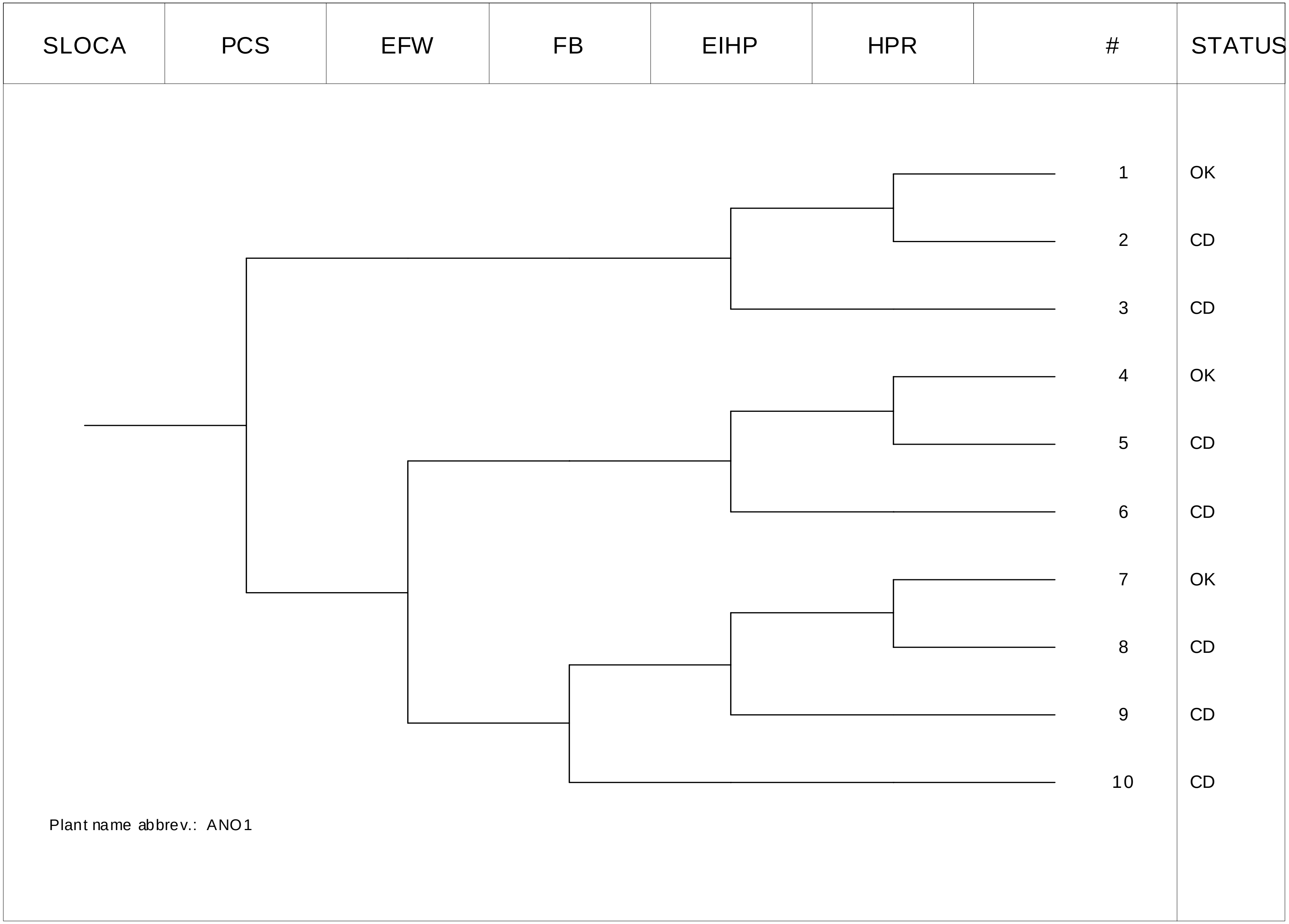
This section provides the simplified event trees called SDP event trees used to define the accident sequences identified in the SDP worksheets in the previous section. An event tree for the stuck-open PORV is not included since it is similar to the small LOCA event tree. The event tree headings are defined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

The following event trees are included:

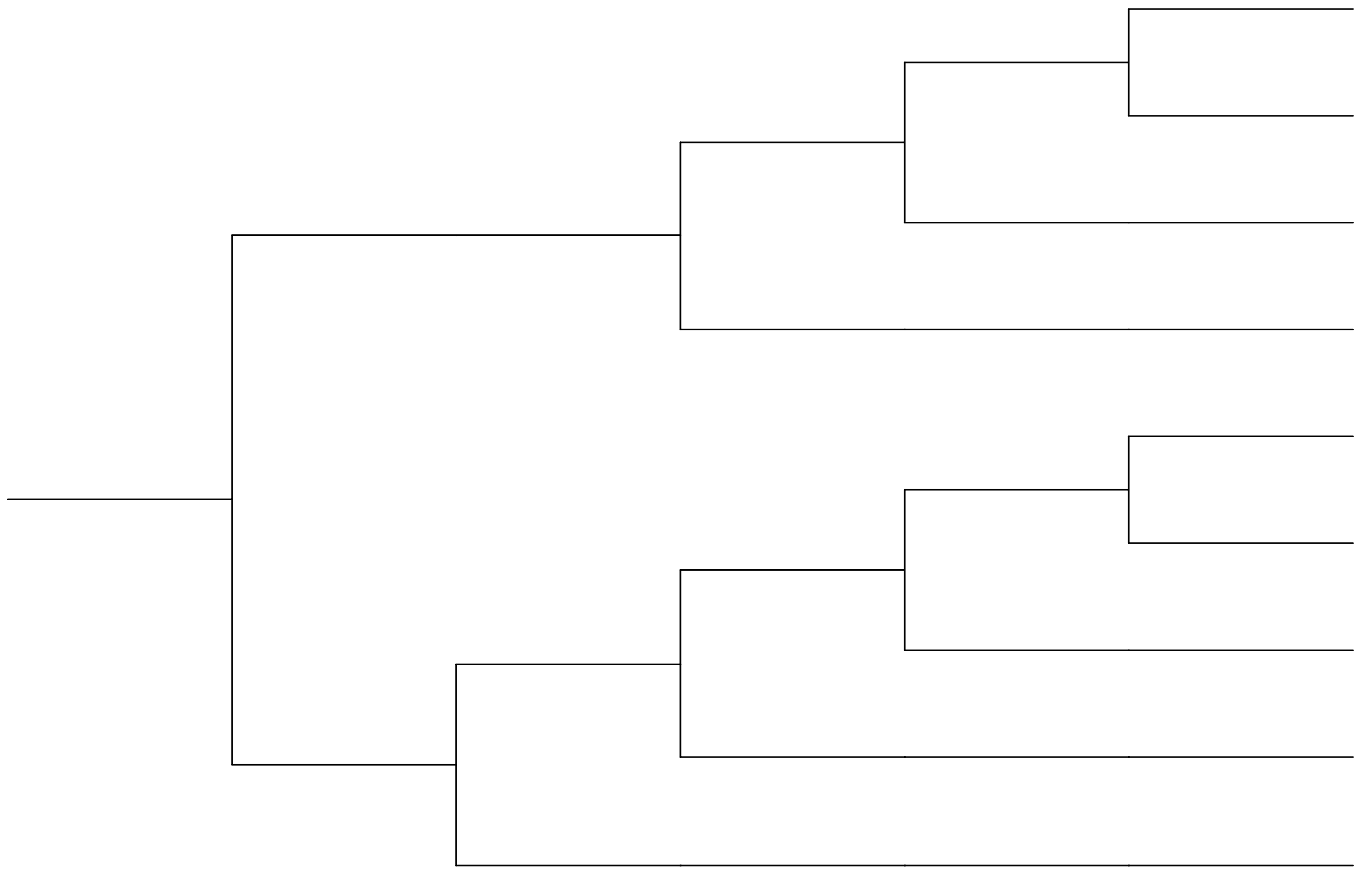
1. Transients
2. Small LOCA
3. Medium LOCA
4. Large LOCA
5. LOOP
6. Steam Generator Tube Rupture (SGTR)
7. Anticipated Transients Without Scram (ATWS)



Plant name abbrev.: ANO1

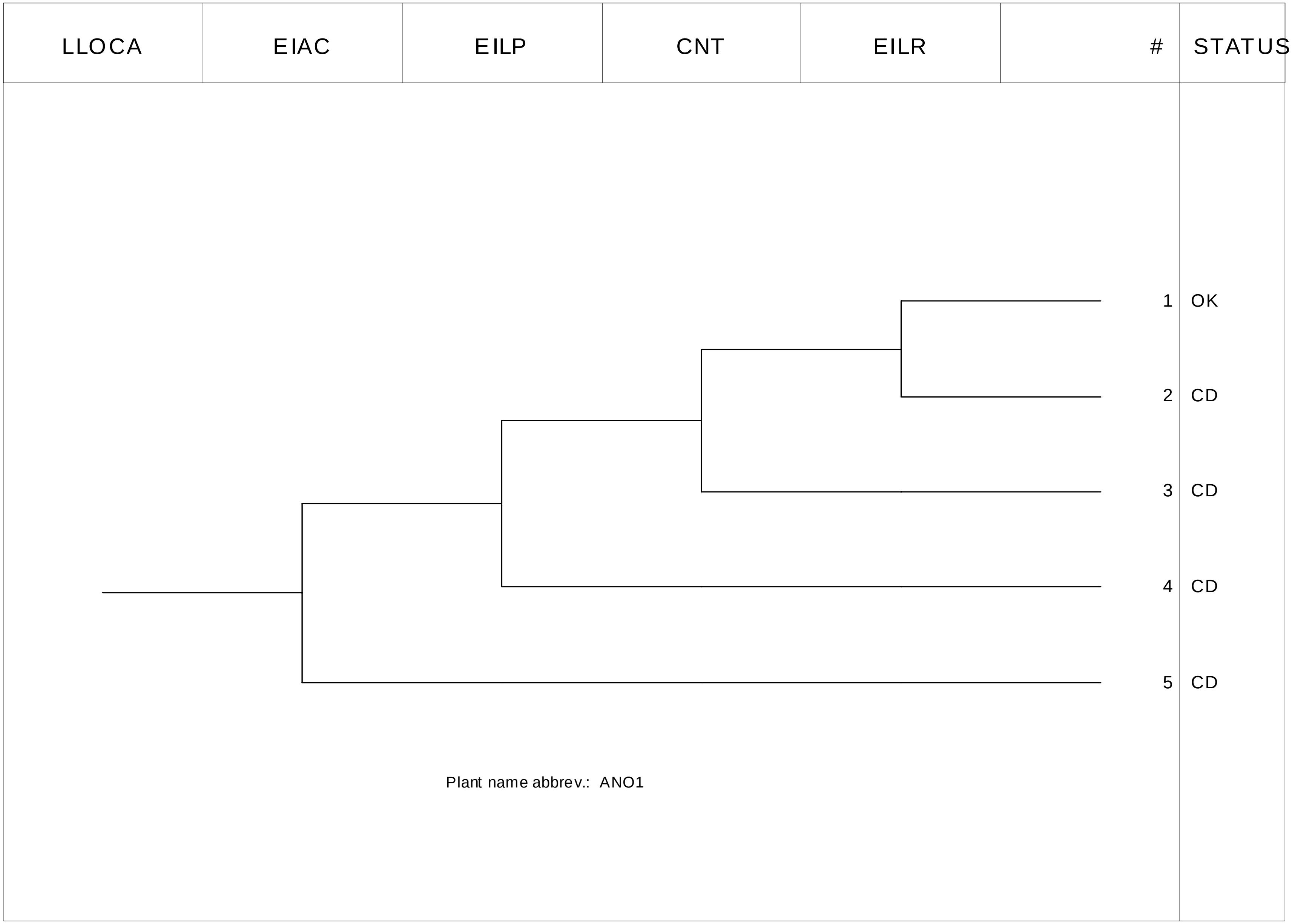

SLOCA
PCS
EFW
FB
EIHP
HPR
#
STATUS
1 OK
2 CD
3 CD
4 OK
5 CD
6 CD
7 OK
8 CD
9 CD
10 CD
Plant name abbrev.: ANO1

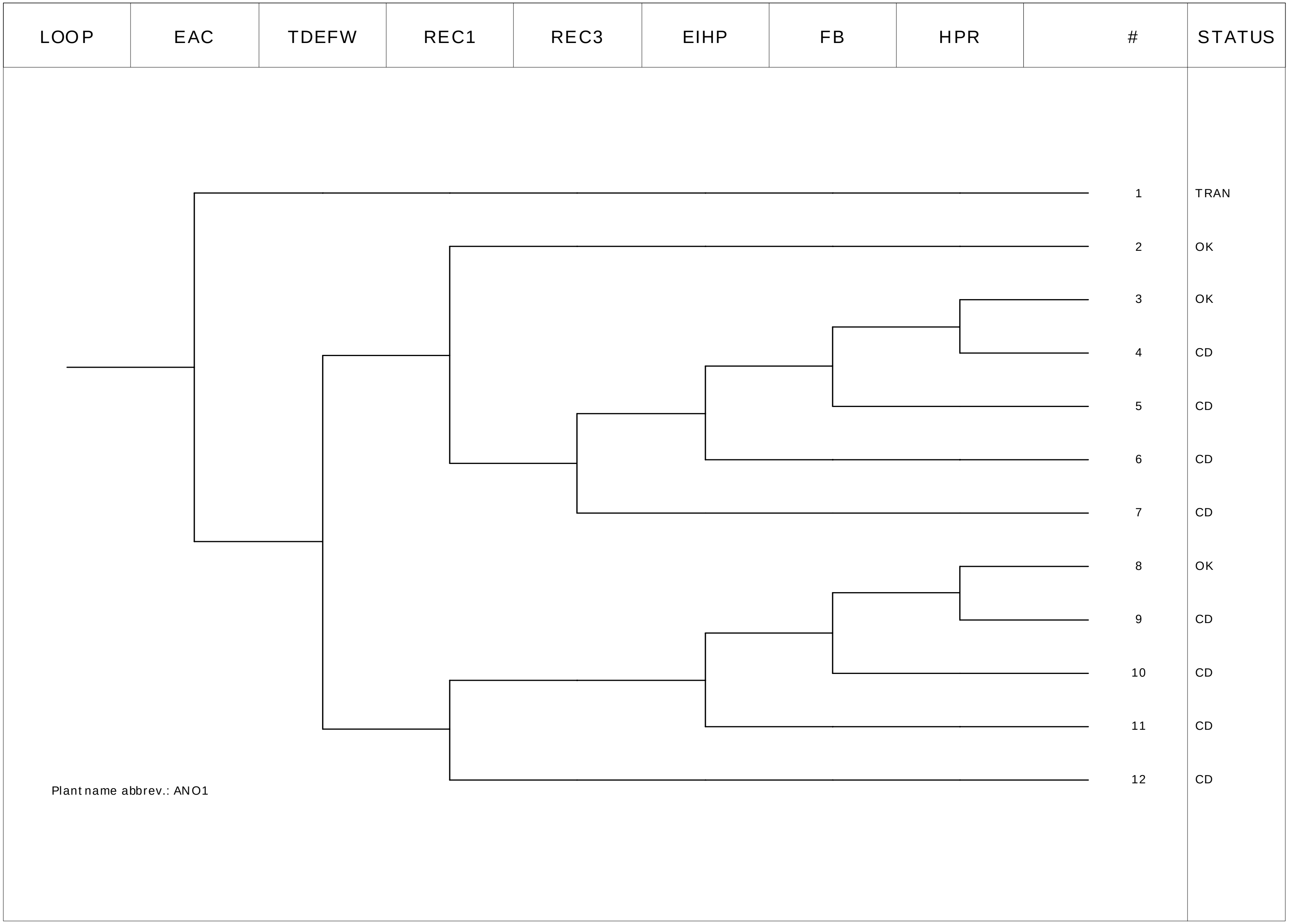
MLOCA	EIHP	EIAC	EILP	CNT	EILR	#	STATUS

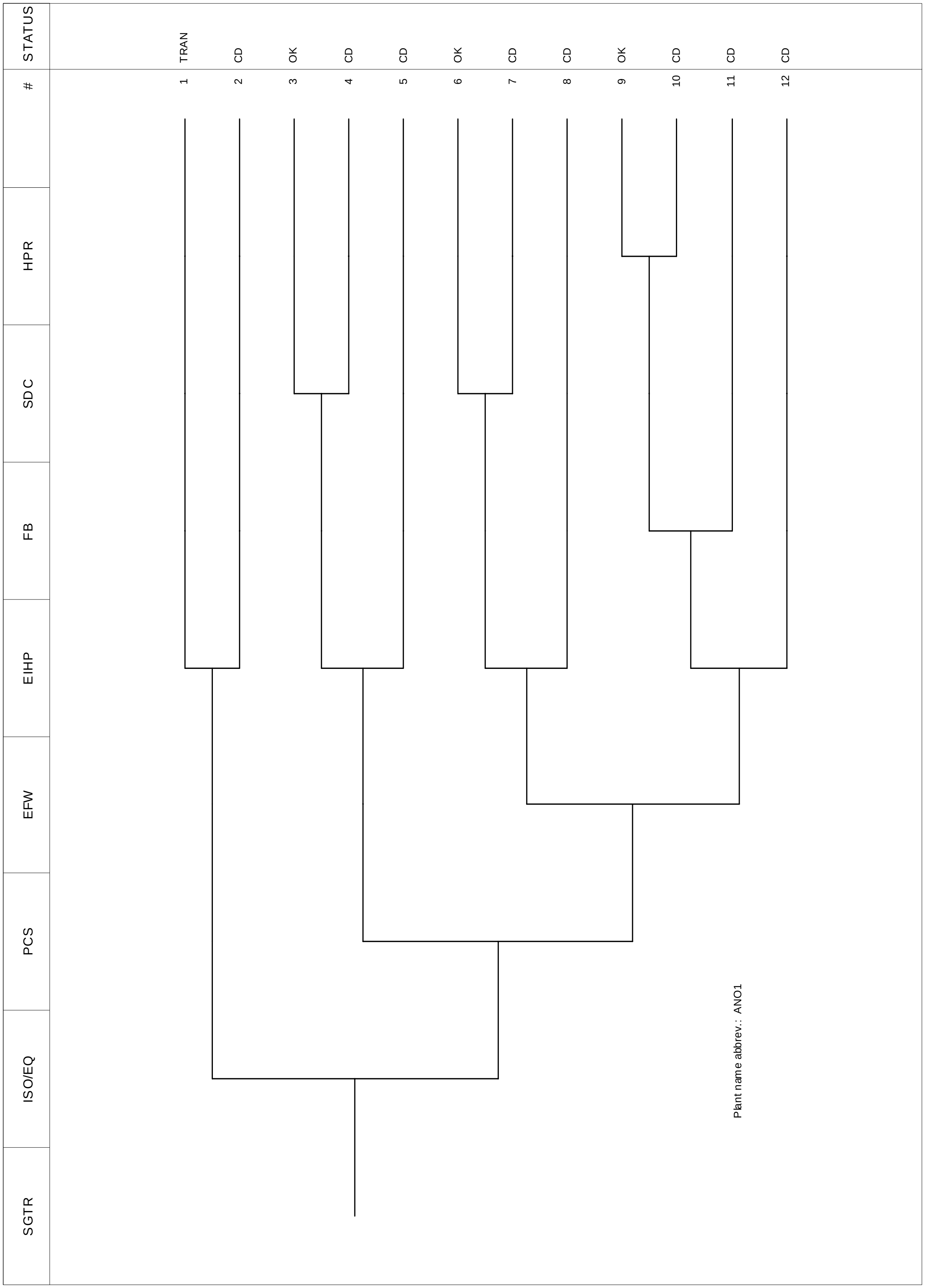


#	STATUS
1	OK
2	CD
3	CD
4	CD
5	OK
6	CD
7	CD
8	CD
9	CD

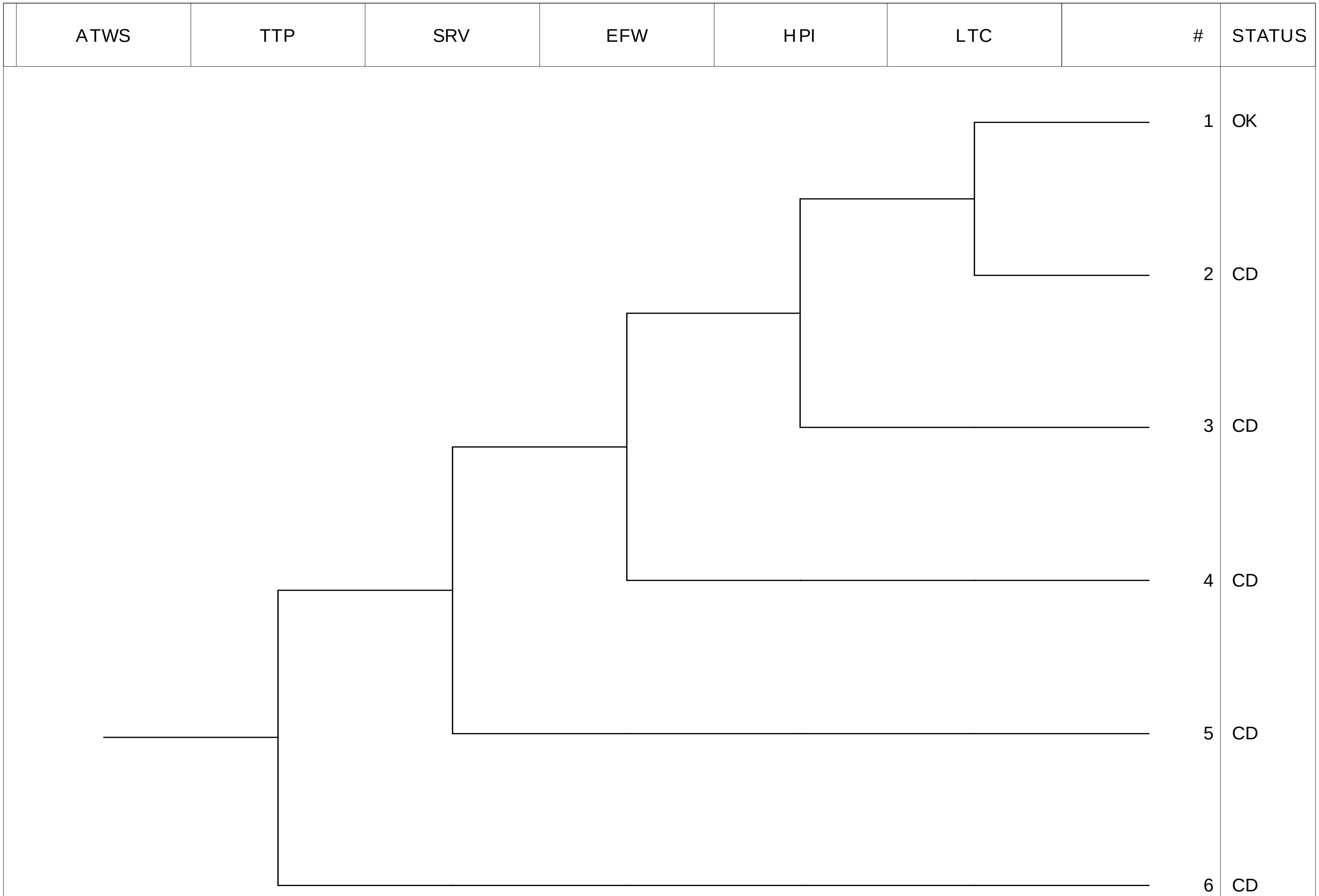
Plant name abbrev.: ANO1


LLOCA
EIAC
EILP
CNT
EILR
#
STATUS
1 OK
2 CD
3 CD
4 CD
5 CD
Plant name abbrev.: ANO1


LOOP
EAC
TDEFW
REC1
REC3
EIHP
FB
HPR
#
STATUS
1 TRAN
2 OK
3 OK
4 CD
5 CD
6 CD
7 CD
8 OK
9 CD
10 CD
11 CD
12 CD
Plant name abbrev.: ANO1



SGTR
ISO/EQ
PCS
EFW
EIHP
FB
SDC
HPR
#
STATUS
1 TRAN
2 CD
3 OK
4 CD
5 CD
6 OK
7 CD
8 CD
9 OK
10 CD
11 CD
12 CD
Plant name abbrev.: ANO1



Plant name abbrev.: ANO1

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., "Arkansas Nuclear One, Unit 1 Individual Plant Examination," April 29, 1993.

RISK-INFORMED INSPECTION NOTEBOOK FOR

ARKANSAS NUCLEAR ONE

UNIT 2

PWR, C-E, TWO-LOOP PLANT WITH LARGE DRY CONTAINMENT

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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:

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ABSTRACT

This notebook contains summary information to support the Significance Determination Process (SDP) in risk-informed inspections for the Arkansas Nuclear One, Unit 2.

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

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

CONTENTS

FIGURES

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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 Arkansas Nuclear One, Unit 2.

1.1 INITIATORS AND SYSTEM DEPENDENCY

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

Table 1 Initiators and System Dependency for Arkansas Nuclear One, Unit 1

Affected Systems	Major Components	Support Systems	Initiating Event
Safety Injection Tank (SIT)	Four Passive SITs (P<610 psig)	NA	LLOCA
EFWS	EFWTDP EFWMDP	125 V-DC Red and Green 4.16 KV bus Red and green 480 VAC Room Cooling &SW loops I&II	Transient, Loop, MSLB/MFLB (outside Cont.), SGTR, SLOCA, ATWS
HPI	Pumps,Valves I&C including DC for 4.16 KV Breakers	4.16 KV (A red, B green), 125 VDC (A red, B green),SW (A: Train I, B: Train II)	Transient, Loop, MSLB/MFLB (outside Cont.), SGTR, SLOCA, ATWS
HPI (Recirc.)	Pumps,Valves I&C including DC for 4.16 KV breakers	4.16 KV (A: red, B: green), 125 VDC (A: red, B: green),Room cooling ,SW (A: Train I, B: Train II), RAS	Transient, Loop, MSLB/MFLB (outside Cont.), SGTR, SLOCA, ATWS
LPI/SDC	Pumps, Valves, I&C including DC for 4.16 KV breakers	4.16 KV (A: red, B: green),480 VAC, 125 VDC (A: red, B: Green), SW (A: Train I, B: Train II), Room Cooling, Inst. Air for SDC flow&temp. Control.	Transient, Loop, MSLB/MFLB (outside Cont.), SGTR, SLOCA, ATWS, MLOCA, and LLOCA
CCS (Containment Spray System)	Pumps, Heat Exch., Valves	4.16KV (A: red, B: green), 480 V-AC (MCC-2B52, MCC-2B62, MCC-2B51, and MCC-2B61), 125 VDC (A: red, B: green), SW (A: Train I, B: Train II)	Transient, Loop, MSLB/MFLB (outside Cont.), SGTR, SLOCA, ATWS, MLOCA, and LLOCA

Table 1 (Continued)

Affected Systems	Major Components	Support Systems	Initiating Event
EDG	Cooling (unit-1 only), HVAC, Start system, Fuel System	Service Water (Red for Train I, Green for Train II), 125 VDC (red AC/DC, Green AC/DC), Room Cooling	LOOP
CCW	Pumps, Vlaves, Heat Exch.	AC from Non ESF and Red bus, SW I, SW II, Instrument air for control valves (fail safe)	Transient, Loop, MSLB/MFLB (outside Cont.), SGTR, SLOCA, ATWS, MLOCA, and LLOCA, ATWS
Service Water System (SWS)	Pumps and Valves	4.16KV/480 VAC (red and green for loops I and II), 125VDC (red and green for loops I&II)	Transient, Loop, MSLB/MFLB (outside Cont.), SGTR, SLOCA, ATWS, MLOCA, and LLOCA, ATWS
ADVs (SG PORV)	Valves (MOVs)	Instrument air system, 125 VDC	Transient, Loop, MSLB/MFLB (outside Cont.), SGTR, SLOCA, ATWS, MLOCA, and LLOCA, ATWS
Primary Safety Relief Valves (PSV)	Spring Loaded Relief Valves (P>2500 psig)	None	ATWS
LTOP and high point vent valves	Valves (MOVs)	Combination of 125 VDC, and 480 VAC, however it appears that DC by itself is sufficient for manual valve operation.	Transient, Loop, MSLB/MFLB (outside Cont.), SGTR, SLOCA, ATWS, MLOCA, and LLOCA, ATWS
RCP	Seals	Operator trips RCPs within 30 minutes of loss of seal cooling, and loss of CCW to seal cooling	Transients, LOOP, SLOCA from RCP.

Notes:

(1) The information is based on IPE submittal of 8/22/92 with the CDF of 3.4E-5.

(2) The plant is equipped with two Primary safety valves , failure of primary safety valve to re-close is equivalent to a medium LOCA.

(3) There are about four different paths for primary pressure relief that the operator could rely on. The IPE indicates that the ECCS vent valves (two paths) each is equivalent to the capacity of about 8 typical CE PORV (pp3.5-7). Since IPE does not explicitly identify the success criteria, we will use 1/4 relief paths for success criteria.

1.2 SDP WORKSHEETS

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

1. Transients
2. Small LOCA
3. Stuck-open PORV (Not applicable to this plant)
4. Medium LOCA
5. Large LOCA
6. LOOP
7. Steam Generator Tube Rupture (SGTR)
8. Anticipated Transients Without Scram (ATWS)
9. Main Steam Line Break (MSLB)

Table 2.1 SDP Worksheet for Arkansas Nuclear One, Unit 2 —— 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 / 2 Feedwater trains with 1 / 4 condensate trains (Operator action)
Secondary Heat Removal (EFW)	1 / 2 MDEFW trains (1 train) or 1 TDEFW train (1 ASD train)
High Pressure Injection (EIHP)	1 / 3 HPI Pumps from RWT (1 multi-train system)
Primary Heat Removal, Feed/Bleed (FB)	1/4 LTOP or ECCS vent path to open for Feed/Bleed and initiate HPI cooling (Operator action)
Long Term Cooling (SDC)[(1)]	1/2 LPI pump and the associated Heat Exhange. In SDC mode (Operator action)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 TRANS - PCS - EFW - FB (6)			
2 TRANS - PCS - EFW -EIHP (5)			
3 TRANS - PCS - EFW - SDC (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.

<u>Notes</u>:

(1) In those transients that RCS integrity is lost,(e.g. failure of a PSV to re-close) then the recirculation of HPR and the operation of Containment spray would be needed. These sequences are treated under LOCAs.

Table 2.2 SDP Worksheet for Arkansas Nuclear One, Unit 2 —— Small 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)	1/3 HPSI trains (1 multi-train systems)
Power Conversion System (PCS)	1/2 feedwater trains and 1/4 Condensate pump (operator action)
Secondary Heat Removal (EFW)	1 / 1 MDEFW trains (1 train) or 1 TDEFW train (1 ASD train)
Primary Bleed (F&B)	1/4 LTOP or ECCS vent path to open for Feed/Bleed and initiate HPI cooling (operator action)
Containment Spray in Recirc. (CSR)	1/2 CS including the associated SDC HX.[2] Auto-aligned by RAS (multi-train system)
High Pressure Recirc (HPR)	1 / 3 HPI pumps taking suction from containment sump and auto-aligned by RAS (1 multi-train system)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 SLOCA - EIHP (3,6,10)			
2 SLOCA-EFW-PCS-CSR (9)			
3 SLOCA - HPR (2,5,8)			
4 SLOCA - EFW -PCS- FB (11)			

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

Notes:

(1) The IPE defines one category for small LOCA with break sizes greater than 0.3 inch to 1.9 inches.

(2) If the SDC heat exchanger is not available and the there is no secondary heat removal, then IPE is crediting 2/2 CFC as a means for removing decay heat. However it still requires the operation of CS pumps.

Table 2.3 SDP Worksheet for Arkansas Nuclear One, Unit 2 —— 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)	1/3 HPSI trains (1 multi-train systems)
Containment Spray Recirculation	1/2 CS including the associated SDC HX.(2) Auto-aligned by RAS (multi-train system)
High Pressure Recirculation (HPR)	1/3 HPSI pumps taking suction from sump and auto-aligned by RAS (1 multi-train system) (1 multi-train system)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 MLOCA - EIHP (4)			
2 MLOCA - HPR (2)			
3. MLOCA - CSR (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) Medium LOCA is defined for break sizes greater than 1.9 inches up to 4.3 inches.

(2) If the SDC heat exchanger is not available and the there is no secondary heat removal, then IPE is crediting 2/2 CFC as a means for removing decay heat. However it still requires the operation of CS pumps.

Table 2.4 SDP Worksheet for Arkansas Nuclear One, Unit 2 —— Large 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 Control (EIAC)	3/4 SITs (1 train system)
High Pressure Injection (EIHP)	1/3 HPI pumps taking suction from RWT (1 multi-train system)
Containment Spray Recirculation (CSR)	1/2 CS trains including the associated SDC HX.(2) Auto-aligned by RAS (1 multi-train system)
Early Inventory, LP Injection (EILP)	1/ 2 LPSI pumps (1 multi-train system).
High Pressure Recirc (HPR)	1 / 3 HPI pump trains taking suction from containment sump and auto-aligned by RAS (1 multi-train system)

Circle Affected Functions	Recovery of Failed Train	*Remaining Mitigation Capability Rating for Each Affected Sequence*	Sequence Color
1 LLOCA - EILP (4)			
2 LLOCA - EIAC (5)			
3 LLOCA - EIHP (6)			
4 LLOCA - CSR (3)			

5 LLOCA - HPR (2)				
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. Large LOCA is for break sizes greater than 4.3 inches.

2. If the SDC heat exchanger is not available and the there is no secondary heat removal, then IPE is crediting 2/2 CFC as a means for removing decay heat. However it still requires the operation of CS pumps.

Table 2.5 SDP Worksheet for Arkansas Nuclear One, Unit 2 —— LOOP

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>:**
Emergency AC Power (EAC)	1/2 EDGs (1 multi-train system)[(1)]
Turbine-driven EFW pump (TDEFW)	Operation of TDEFW pump (1 ASD train)[(2)]
Recovery of AC Power in < 1 hrs (REC1)	SBO procedure and Recovery of an AC source in one hour [(3)] (high Stress operator action)
Recovery of AC Power in < 10 hrs (REC10)	SBO procedure and Recovery of an AC source in ten hours [(4)] (operator action)
Early Inventory, HP Injection (EIHP)	1/3 HPI trains (1 multi-train system)
Secondary Heat Removal (EFW)	1 / 1 MDEFW trains (1 train) or 1 TDEFW train (1 ASD train)
Primary Heat Removal (FB)	1/4 LTOP or ECCS vent path to open for Feed/Bleed and initiate HPI cooling (operator action)
Long Term Cooling (SDC)	1/2 LPI pump and the associated Heat Exh. In SDC mode (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 LOOP - EAC - REC10 (7) (AC can not be restored in 10 hours)			
2 LOOP - EAC - TDEFW - REC1 (8)			
3 LOOP - EAC - REC1 - EIHP (6) (AC recovered prior to core uncovery)[(5)]			

4 LOOP - EAC - REC1 - FB (5) (AC recovered prior to core uncovery)			
5 LOOP - EAC - REC1 - SDC (4)			
6 LOOP - EFW - EIHP (1,2)[6] (AC recovered in less than one hour)			
7 LOOP - EFW - FB (1,2) (AC recovered in less than 1 hour)			
8 LOOP - EFW - SDC (1,2) (AC recovered in less than 1 hour)			

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) EDG day tank has sufficient fuel for two and half hour. Failure of room cooling assumes to fail the EDGs.

(2) The batteries are assumed to be depleted in eight hours.

(3) Core damage is assumed to occur in one hour if no secondary heat removal. IPE uses 55 minutes as maximum delay for initiation of once through cooling in T2 transients.

(4) Core damage is assumed in two hour after battery depletion, that is REC10 refers to recovery of offsite power after battery depletes but prior to core uncovery. There is also possibility that the battery depletes in less than eight hours with some probability. In these cases REC10 would be recovery of AC source two hours after battery depletion.

(5) A major contributor to sequences 3 and 4 is recovery of AC source in less than 10 hours but after the battery is depleted (e.g. after 8 hours).

(6) Sequences 6,7, and 8 basically are transient type sequences with loss of decay heat removal. If Ac power is either available or become available in less than one hour the plant response would be similar to transients.

Table 2.6 SDP Worksheet for Arkansas Nuclear One, Unit 2 —— SGTR

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>:**
Early Inventory HP injection (EIHP)	1/3 HPSI trains (1 multi-train system)
Power Conversion System (PCS)	1/2 feedwater trains and 1/4 Condensate pump: extended operation requires throttling of flow to allow depressurization (high stress operator action)
Secondary Heat Removal (EFW)	1/1 MDEFW (1 train) or 1/1 TDEFW (1 diverse ASD train)
Feed and Bleed (FB)	1/4 LTOP or ECCS vent path to open for Feed/Bleed and initiate HPI cooling (operator action)
High Pressure Recirculation (HPR)	1 / 3 HPI trains taking suction from containment sump (1 multi-train system)
Containment Spray Recirculation (CSR)	1/2 CS trains including the associated SDC HX.(2) Auto-aligned by RAS (1 multi-train system)
Shutdown Cooling (SDC)	1/2 LPI in DHR cooling mode (operator action)
Primary/Secondary Pressure Equalization and Cool Down (ISO/EQ)	Pressure equalization below SG safety setpoints ;assumes secondary cooling available for rapid cool down (High stress 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 SGTR - EIHP (2,5,8,3)			
2 SGTR - ISO/EQ - SDC (4,7)			
3 SGTR - ISO/EQ - PCS - EFW - FB(1) (2)			

4. SGTR -ISO/EQ - PCS - EFW - CSR (11)			
5 SGTR -ISO/EQ - PCS - EFW - HPR[(1)] (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.			

<u>Notes</u>:

(1) Failure of PCS and EFW implies failure of ISO/EQ. Therefore in sequences (3) , (4) , and (5)failure to equalize should be considered with the probability of one.

Table 2.7 SDP Worksheet for Arkansas Nuclear One, Unit 2 —— ATWS

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

Safety Functions Needed:	**Full Creditable Mitigation Capability for Each Safety Function:**
Turbine trip (TTP)	Operator trips the turbine (operator action)
High Pressure Injection (EIHP)	Operator conducts emergency boration using 1 / 3 HPI[(1)](1 multi-train system).
Emergency Boration (EB)	Operator conducts emergency oration using 1/3charging pumps from BAM (operator action)
Secondary Heat Removal (EFW)	1 / 1 EFWMDP and 1 / 1 EFWTDP (1 Train)
Shut Down Cooling (SDC)	1/2 LPI in DHR mode or 1/3 HPI in Recirculation mode (1 multi-train system)
Primary Relief Re-close (SRVR)	2/2 SRV re-close (1 train)
Primary Relief Open (SRVO)	2 / 2 SRVs open (1 train)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 ATWS - SRV (10)			
2 ATWS - EFW (4,9)			
3 ATWS - EB (3)			
4 ATWS - SRVR - EIHP (8)			

5 ATWS - TTP (11)			
6 ATWS - SRVR - CSR (7)			
7 ATWS-SRVR-HPR (6)			
8 ATWS - SDC (2)			

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

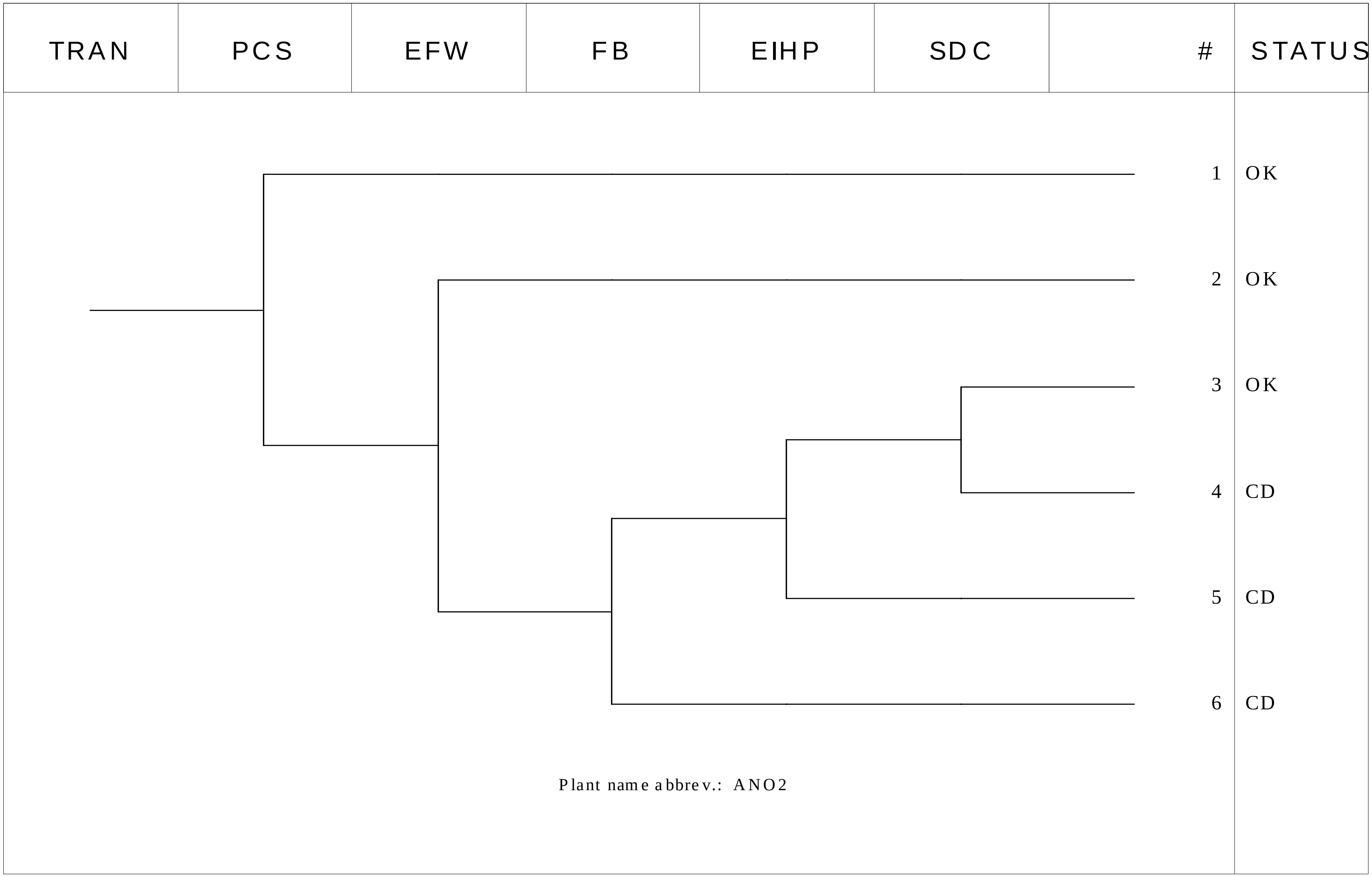
If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures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

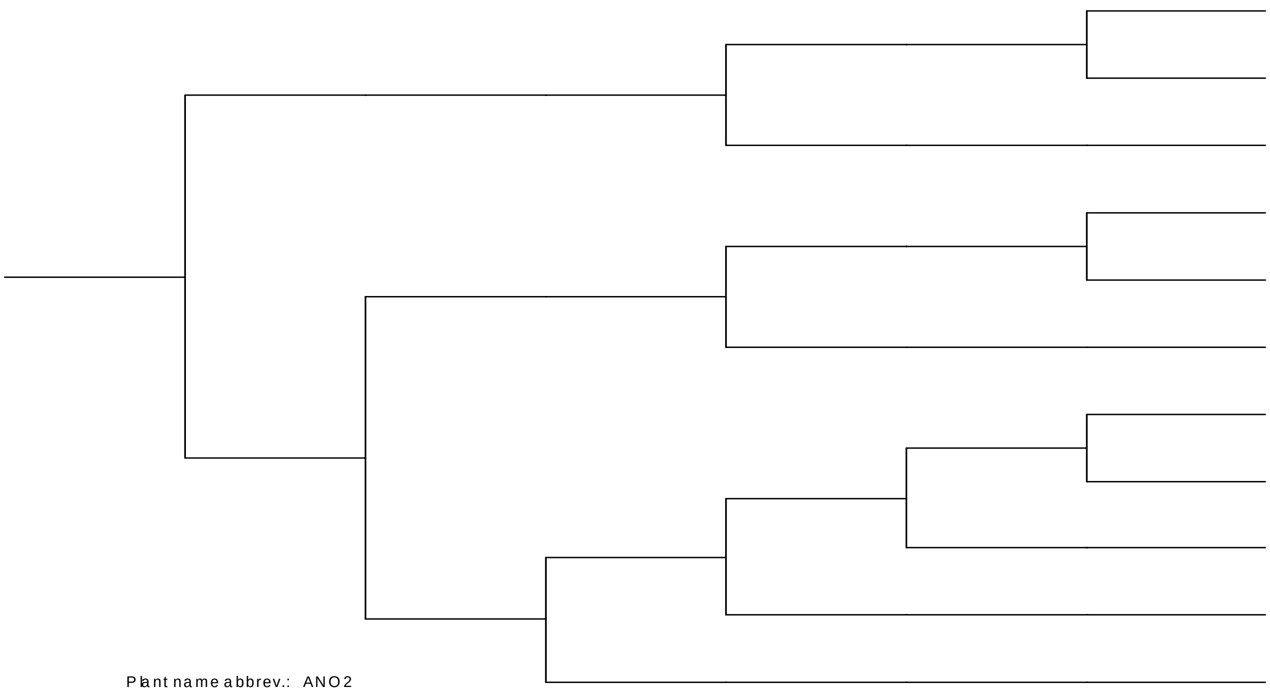
1.3 SDP Event Trees

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

The following event trees are included:

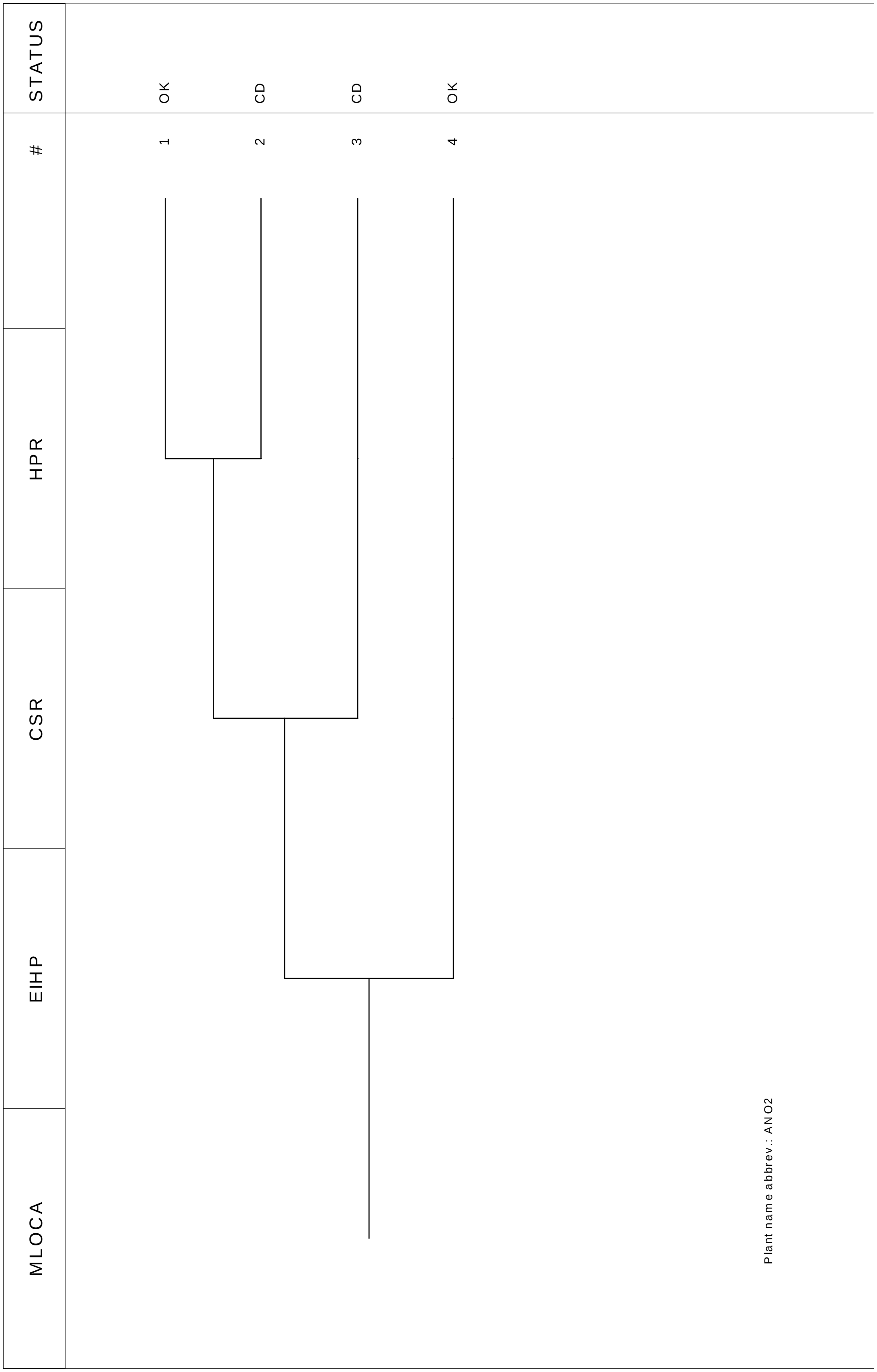
1. Transients
2. Small LOCA
3. Medium LOCA
4. Large LOCA
5. LOOP
6. Steam Generator Tube Rupture (SGTR)
7. Anticipated Transients Without Scram (ATWS)


TRAN
PCS
EFW
FB
EIHP
SDC
#
STATUS
1 OK
2 OK
3 OK
4 CD
5 CD
6 CD
Plant name abbrev.: ANO2

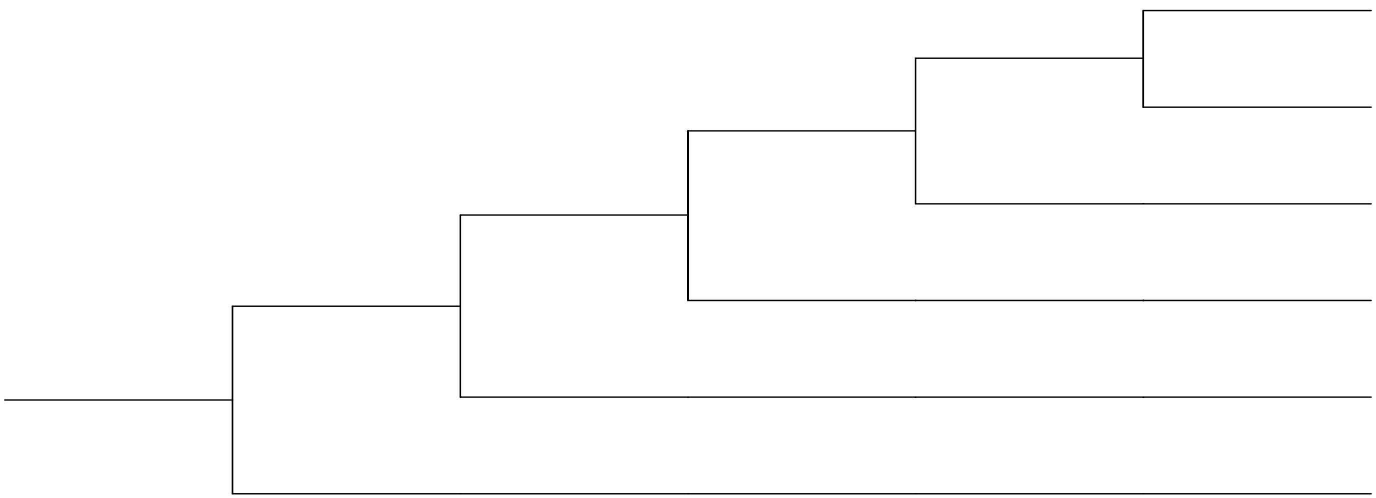


SLOCA	PCS	EFW	FB	EIHP	CSR	HPR	#	STATUS
							1	OK
							2	CD
							3	CD
							4	OK
							5	CD
							6	CD
							7	OK
							8	CD
							9	OK
							10	CD
							11	CD

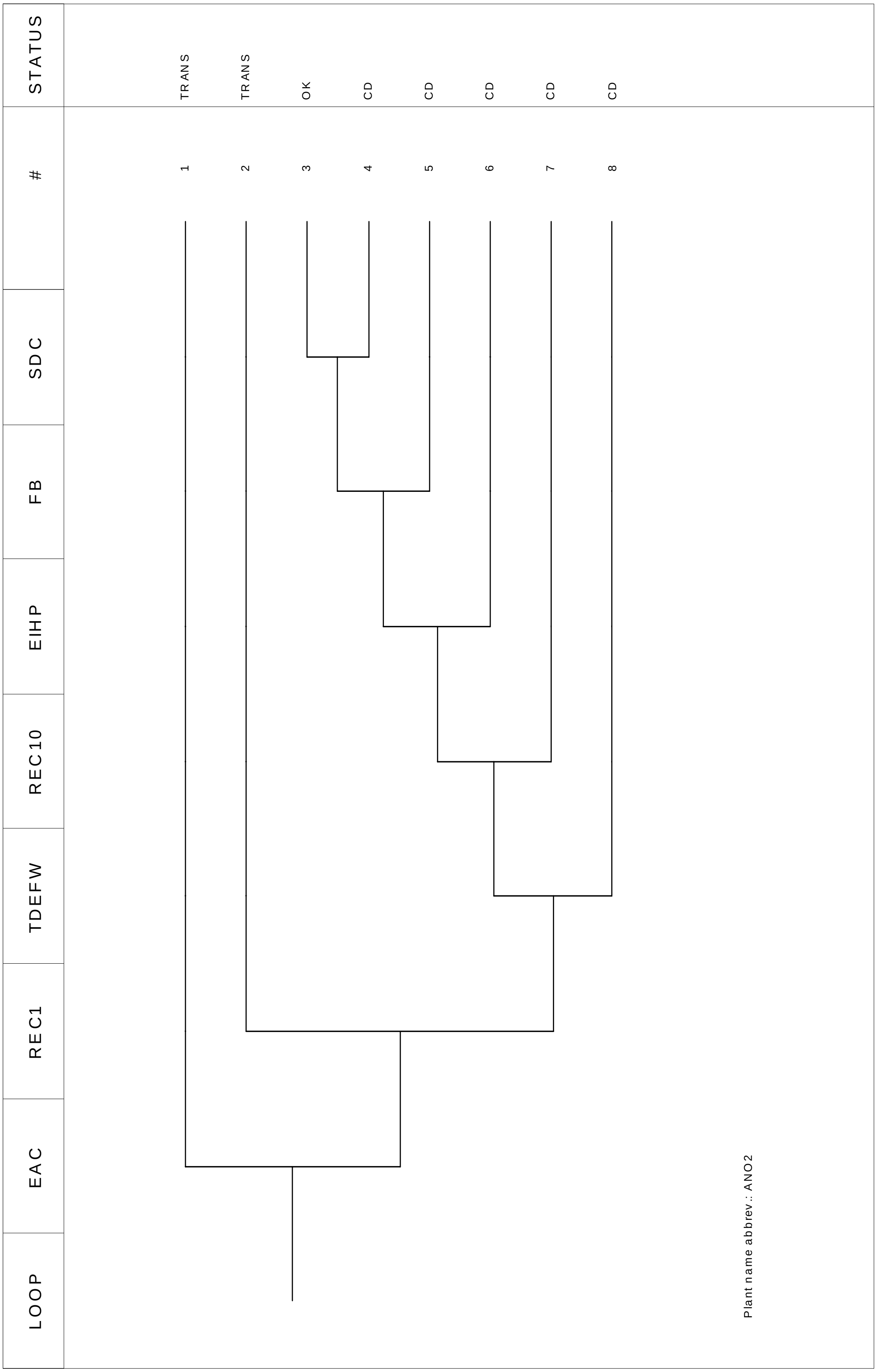
Plant name abbrev.: ANO2


MLOCA
EIHP
CSR
HPR
#
STATUS
1 OK
2 CD
3 CD
4 OK
Plant name abbrev.: ANO2

LLOCA	EIHP	EIAC	EILP	CSR	HPR	#	STATUS
						1	OK
						2	CD
						3	CD
						4	CD
						5	CD
						6	CD



Plant name abbrev.: ANO2


LOOP
EAC
REC1
TDEFW
REC10
EIHP
FB
SDC
#
STATUS
1 TRANS
2 TRANS
3 OK
4 CD
5 CD
6 CD
7 CD
8 CD
Plant name abbrev.: ANO2

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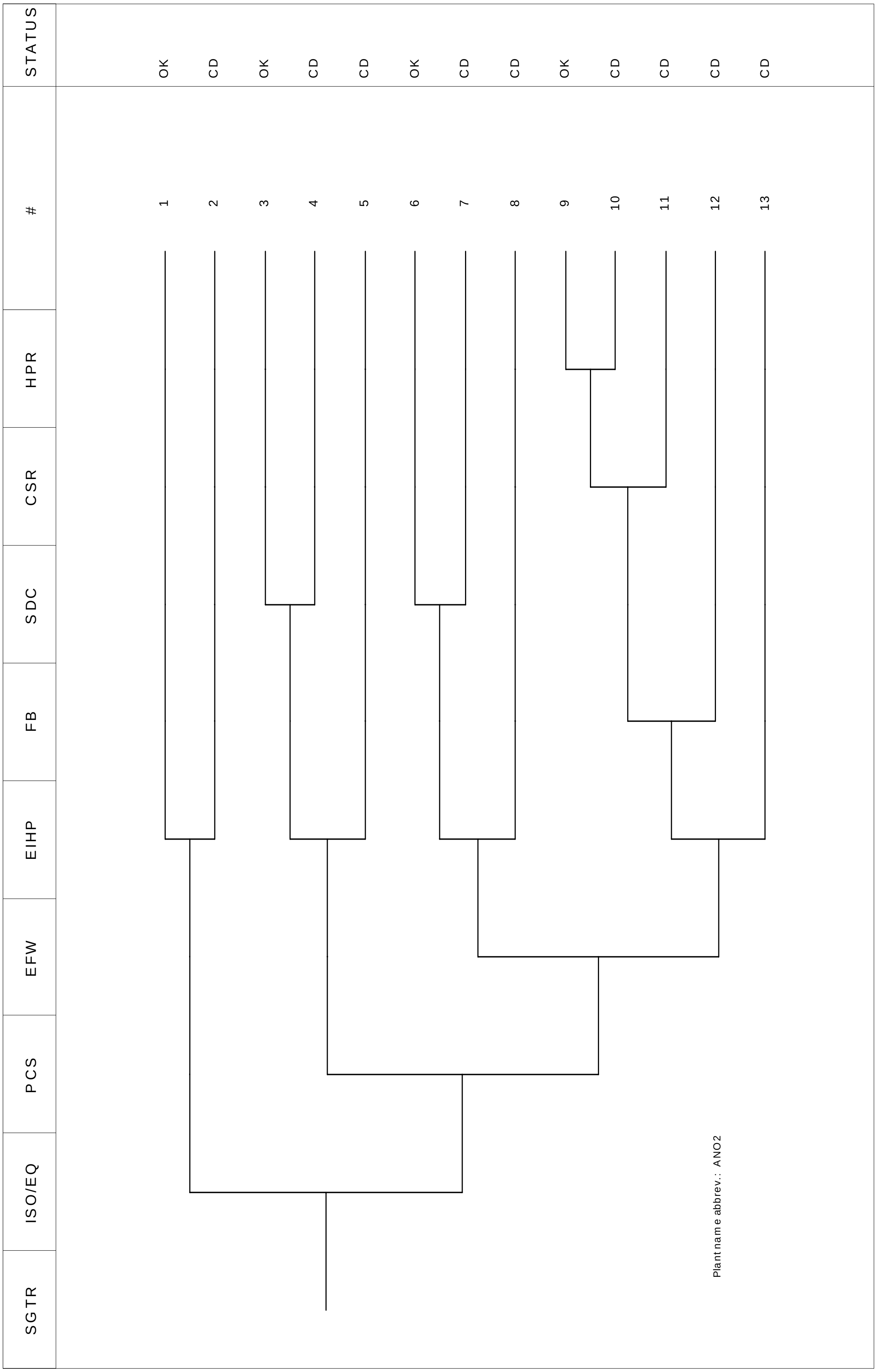
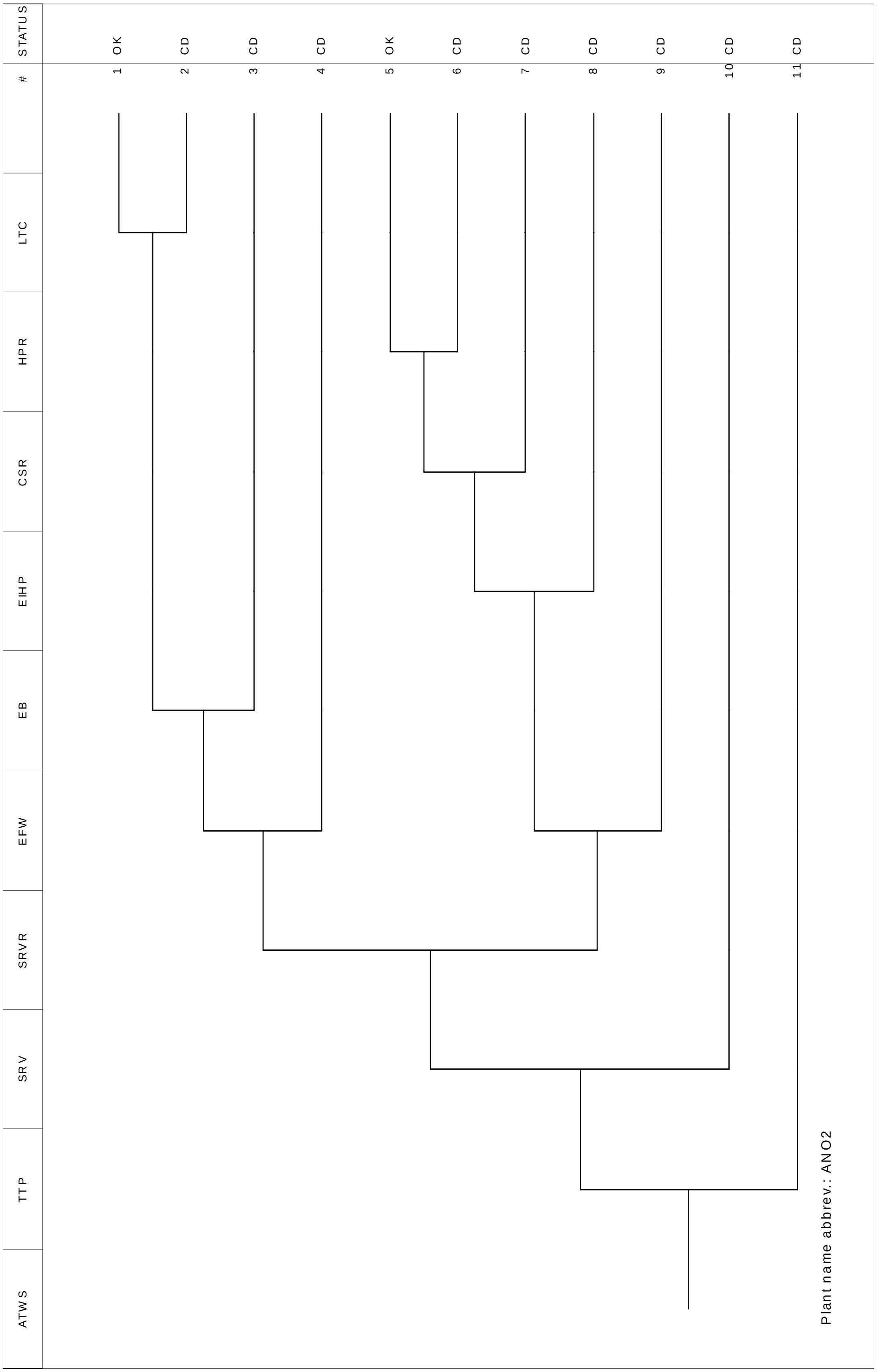

SGTR
ISO/EQ
PCS
EFW
EIHP
FB
SDC
CSR
HPR
#
STATUS
1 OK
2 CD
3 OK
4 CD
5 CD
6 OK
7 CD
8 CD
9 OK
10 CD
11 CD
12 CD
13 CD
Plant name abbrev.: ANO2


STATUS
#
LTC
HPR
CSR
EIHP
EB
EFW
SRVR
SRV
TTP
ATWS
1 OK
2 CD
3 CD
4 CD
5 OK
6 CD
7 CD
8 CD
9 CD
10 CD
11 CD
Plant name abbrev.: ANO2

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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. Entergy Operations, Inc., "Arkansas Nuclear One, Unit 2 Individual Plant Examination," August 28, 1992.