March 16, 2000

Mr. G. R. Peterson
Site Vice President
Catawba Nuclear Station
Duke Energy Corporation
4800 Concord Road
York, South Carolina 29745-9635

SUBJECT: CATAWBA NUCLEAR STATION, UNITS 1 AND 2 RE: SITE-SPECIFIC WORKSHEETS FOR USE IN THE NUCLEAR REGULATORY COMMISSION'S SIGNIFICANCE DETERMINATION PROCESS (TAC NO. MA6544)

Dear Mr. Peterson:

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 Catawba Nuclear Station, Units 1 and 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. This information was sent electronically to Ms. Martha Purser of your staff on March 14, 2000. The SDP is discussed in more detail below.

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

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

The SDP for power operations involves evaluating an inspection finding's impact on the plant's capability to limit the frequency of initiating events; ensure the availability, reliability, and capability of mitigating systems; and ensure the integrity of the fuel cladding, reactor coolant system, and containment barriers. As described in SECY-99-007A, the SDP involves the use of three tables:

Table 1 is the estimated likelihood for initiating event occurrence during the degraded period, Table 2 describes how the significance is determined based on remaining mitigation system capabilities, and Table 3 provides the bases for the failure probabilities associated with the remaining mitigation equipment and strategies.

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

We will coordinate our efforts through your licensing or risk organizations as appropriate. If you have any questions, please contact me at 301-415-3025.

Sincerely,

/RA/

Chandu P. Patel, Project Manager, Section 1
Project Directorate II
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-413 and 50-414

Enclosure: As stated

cc w/encl: See next page

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Chandu P. Patel, Project Manager, Section 1
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RISK-INFORMED INSPECTION NOTEBOOK FOR

CATAWBA NUCLEAR STATION

UNITS 1 AND 2

PWR, WESTINGHOUSE, FOUR-LOOP PLANT WITH ICE CONDENSER CONTAINMENT

Prepared by

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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 McGuire Nuclear Station.

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

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

CONTENTS

Page

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 Catawba Nuclear Station.

1.1 INITIATORS AND SYSTEM DEPENDENCY

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

Table 1 (Continued)

Table 1 Initiators and System Dependency for Catawba Units 1 and 2 [1]

Affected Systems	Major Components	Support Systems	Initiating Event
Accumulators	Four Accumulators	600 V-AC (Cold leg injection)	
Essential Auxiliary Power System	AC Power Distribution & AC Instrument Power	125 V-DC 1 EVDA for train A, and 1EVDD for train B (4 kV AC and lower) 1 DCA and 1 DCB for 6.9 kV HVAC two trains for each switchgear room [2]	Transient, SLOCA, SORV, MLOCA, LLOCA, LOOP, SGTR, ATWS, RCP seal LOCA
AFW (CA)	Two MDPs	4 kV AC,125 V-DC, ESFAS, RN, IA (VI:Control Valves Open), Water supply (UST, Condenser, CA CST, and RN)	Transient, SLOCA, SORV, LOOP, SGTR, ATWS
	One TDP	ESFAS, IA (Control Valves Open), Main Steam System, Water supply (UST, Condenser, CA CST, and RN)	
CCW (KC)	Four headers each with two pumps and one heat exchanger	4 kV AC, 600 V-AC, 125V-DC, RN (Dependent but fail safe are VI, and ESFAS)	Transient, SLOCA, MLOCA, LLOCA, LOOP, SGTR, ATWS, RCP seal LOCA
Condensate / MFW	3 50% Condensate pumps 2 50% Turbine-driven FPs	6900 V-AC,600 V-AC,125 V-DC, IA(VI), Recirculated cooling water (KR), Condenser circulating water (RC)	Transient
Containment Spray System (NS)	Two Trains, each with one pump, one heat exchanger and one header	4160 V-AC, 600 V-AC, 120 V-AC, 125 V-DC, RN, ESFAS, HVAC, FWST (RWST)	LLOCA
CVCS (NV)	One PDP (32 GPM) Two CCP(150GPM@2670PSIG)	4160 V-AC, 600 V-AC (also for PDP), 125 V-DC, RN, FWST	Transient, SLOCA, SORV, MLOCA, LOOP, SGTR, ATWS, RCP seal LOCA

Table 1 (Continued)

Affected Systems	Major Components	Support Systems	Initiating Event
DC Power System	Four Divisions of panel boards and distribution centers, Buses, battery chargers and batteries	600 V-AC Dist. (without AC, battery capacity is 3 hrs.)	Transient, SLOCA, SORV, MLOCA, LLOCA, LOOP, SGTR, ATWS, RCP seal LOCA
EDG	Two EDGs per unit	125 V-DC, RN, VG(DG starting air), KD (DG cooling water), LD (lube oil system), and VD (HVAC)	LOOP
ESFAS	Dual Train Control System	120 V-AC, HVAC (some actuation are fail safe)	Transient, SLOCA, SORV, MLOCA, LLOCA, LOOP, SGTR, ATWS, RCP seal LOCA
HVAC: Control area (VC) and aux. Bldg. HVAC including control room, and Engineering Safeguard pump room (VA)	Two Train of chilled water (YC) with cross connection, each with a compression tank, a chilled water pump, and a chiller.	RN, 4160 V-AC, 600 V-AC, 120 V-AC	Transient, SLOCA, SORV, MLOCA, LLOCA, LOOP, SGTR, ATWS, RCP seal LOCA
Instrument Air (VI)	Six Air compressors three operating and three backup per unit	600 V-AC,120 V-AC, Recirculated cooling water (KR)	Transient, SLOCA, SORV, LOOP, SGTR, ATWS
Main Steam (SM)	Per SG: One Secondary PORV and the associated block valve, fove safety relief valves. One MSIV and eight condenser steam dump valves with total of 40% capacity	VI, 125 V-DC for MSIVs and secondary PORVs, 600 V-AC for block valves	SGTR
Pressurizer Pressure Relief (NC)	Three Safety valves and three PORVs with associated block valves, and two air operated pressurizer spray valves	125 V-DC, and VI with backup N2 for PORVs VI and 125 V-DC for prizer spray valves 600 V-AC for block valves	Transient, SLOCA, SORV, LOOP, SGTR, ATWS

Table 1 (Continued)

Affected Systems	Major Components	Support Systems	Initiating Event
RCP	Seals	1 / 2 Charging pumps (NV Trains) to seal injection or 1 / 2 KC trains (two pumps) to thermal barrier heat exchanger of all four pumps 1/1 SSF pump for seal injection as a recovery action	LOOP, RCP seal LOCA
RHR/LPSI (ND)	Two trains each with a pump, and a heat exchanger (with ability to cross connect trains)	4160 V-AC, 600 V-AC, 120 V-AC, 125 V-DC, Component cooling water (KC), HVAC (i.e., RN)	Transient, SLOCA, SORV, MLOCA, LLOCA, LOOP, SGTR
Nuclear Service Water (RN)	Two Pumps (100%) in two train	4160 V-AC, 600 V-AC, 125 V-DC	Transient, SLOCA, SORV, MLOCA, LLOCA, LOOP, SGTR, ATWS, RCP seal LOCA
HPIS (NI)	Two Pumps in two trains with shutoff at 1500 psi. In recirculation manual alignment to ND pumps	4160 V-AC, 600 V-AC, 125 V-DC, RN, FWST	Transient, SLOCA, SORV, MLOCA, LOOP, SGTR, ATWS, RCP seal LOCA
Standby Shutdown Facility (SSF)	One Diesel generator and two makeup pumps with associated support	NA	Transient, LOOP, RCP seal LOCA

1. Plant internal event CDF = 4.0E-5/yr, Seismic 1.4E-5/yr, and Tornado 1.9E-5/yr, fire and flood contributions are negligible.

2. Loss of switchgear room HVAC is shown not to be a risk concern for the internal events.

1.2 SDP WORKSHEETS

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

1. Transients (Reactor Trip)
2. Transients with Loss of PCS (TPCS)
3. Small LOCA
4. Stuck-open PORV
5. Medium LOCA
6. Large LOCA
7. LOOP
8. Steam Generator Tube Rupture (SGTR)
9. Anticipated Transients Without Scram (ATWS)
10. Main Steam and Feed Line Break (MSLB/FLB)
11. Loss of Nuclear SW/CCW (TRN)
12. Loss of 4160 Essential Bus A (T4A)
13. Loss of 125 V-DC 1EDF Bus (T1ED)
14. Transients with Loss of Instrument Air (TIA)

Table 2.1 SDP Worksheet for Catawba Nuclear Station, Units 1 and 2 —— Transients (Reactor Trip)

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 trains to 2/4 SG (High stress operator action)(1)
Secondary Heat Removal (AFW)	(1/2 MDAFW trains (1 multi-train system) or 1/1 TDAFW train (1 ASD train)) to 2/4 SG
Early Inventory, High Pressure Injection (EIHP)	(1/2 NV trains (1 multi-train system) or 2/2 NI trains (1 train)) to 3/4 loops
Primary Heat Removal, Feed/Bleed (FB)	2/3 PORVs open for Feed and Bleed (Operator action)(2)
High Pressure Recirculation (HPR)	(1/2 NV trains or 1/2 NI trains) taking suction from 1/2 LPSI (RHR) trains (operator action)(3)

<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 - AFW - HPR (4)			
2 TRANS - PCS - AFW - EIHP (5)			
3 TRANS - PCS - AFW - FB (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.

<u>Notes</u>:

(1) The HEP value in the IPE is 2.0E-1.

(2) The human error probability (HEP) for FB in IPE is 1.0E-2.

(3) The human error probability for switch over to recirculation is 3.2E-3.

Table 2.2 SDP Worksheet for Catawba Nuclear Station, Units 1 and 2 —— Transients with Loss of PCS [(1)] (TPCS)

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

Safety Functions Needed:	***Full* Creditable Mitigation Capability for Each Safety Function**:
Secondary Heat Removal (AFW)	(1/2 MDAFW trains (1 multi-train system) or 1 TDAFW train (1 ASD train)) to 2/4 SG
Early Inventory, High Pressure Injection (EIHP)	1/2 NV trains (1 multi-train system) or 2/2 NI trains (1 train) to 3/4 loops
Primary Heat Removal, Feed/Bleed (FB)	2/3 PORVs open for Feed and Bleed (Operator action)[(2)]
High Pressure Recirculation (HPR)	(1/2 NV trains or 1/2 NI trains) taking suction from 1/2 LPSI (RHR) trains (operator action)[(3)]

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 TPCS - AFW - HPR (4)			
2 TPCS - AFW - EIHP (5)			
3 TPCS - AFW - FB (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 frequency of transients with unrecoverable loss of PCS is estimated around 0.1 per year based on IPE (taking into account 0.2 for recovery of PCS).

(2) The human error probability (HEP) for FB in IPE is 1.0E-2.

(3) The human error probability for switch over to recirculation is 3.2E-3.

Table 2.3 SDP Worksheet for Catawba Nuclear Station, Units 1 and 2 —— Small LOCA

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

Safety Functions Needed:	***Full* Creditable Mitigation Capability for Each Safety Function:**
Early Inventory, HP Injection (EIHP)	(1/2 NV trains (1 multi-train system) or ½ NI trains (1 multi-train system)) injecting into 2 cold legs
Secondary Heat Removal (AFW)	(1/2 MDAFW trains (1 multi-train system) or 1/1 TDAFW train (1 ASD train)) to 2 intact SGs
Primary Bleed (FB)	2/3 PORVs open for Feed and Bleed (Operator action)(1)
High Pressure Recirculation (HPR)	(1/2 NV trains or 1/2 NI trains) taking suction from 1/2 LPSI (RHR) trains (operator action)(2)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 SLOCA - HPR (2,4)			
2 SLOCA - AFW - FB (5)			
3 SLOCA - EIHP (6)			

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

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

Notes:

(1) The human error probability (HEP) for FB in IPE is 1.0E-2.

(2) The human error probability for switch over to recirculation is 3.2E-3.

Table 2.4 SDP Worksheet for Catawba Nuclear Station, Units 1 and 2 —— Stuck Open PORV (SORV)[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:**
Operator Closes the Block Valve (BLK)	Closure of the associated block valve (Operator action)[2]
Early Inventory, HP Injection (EIHP)	(1/2 NV trains (1 multi-train system) or 1/2 NI trains (1 multi-train system)) injecting into 2 cold legs
Secondary Heat Removal (AFW)	(1/2 MDAFW trains (1 multi-train system) or 1/1 TDAFW train (1 ASD train)) to 2 intact SGs
Primary Bleed (F&B)	1/2 remaining PORVs open for Feed and Bleed (Operator action)[3]
High Pressure Recirculation (HPR)	(1/2 NV trains or 1/2 NI trains) taking suction from 1/2 LPSI (RHR) trains (operator action)[4]

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 SORV - BLK - HPR (2,4)			
2 SORV - BLK - AFW - FB (5)			
3 SORV - BLK - EIHP (6)			

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

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

<u>Notes</u>:

(1) The stuck open PORV accounts for both PORVs inadvertently opens (or major leaks) during operation and failure of PORV to re-seat after a transient, e.g, loss of feed water transients. In some scenarios when the pressurizer goes solid or those scenarios with loss of secondary heat removal and failure of one or more PORVs, there is a possibility of SRV demand and its subsequent failure to re-close. The stuck open SRV is equivalent to Medium LOCA and it is not treated here.

(2) The HEP value for operator to close the block valve is estimated as 1.0E-3 in the IPE.

(3) The human error probability (HEP) for FB in IPE is 1.0E-2.

(4) The human error probability for switch over to recirculation is 3.2E-3.

Table 2.5 SDP Worksheet for Catawba Nuclear Station, Units 1 and 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)	(1/2 NV trains (1 multi-train system) and 1/2 NI trains (1 multi-train system)) injecting into 2/4 cold leg
High Pressure Recirculation (HPR)	(1/2 NV trains or 1/2 NI trains) taking suction from 1/2 LPSI (RHR) trains (operator action)[1]

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 MLOCA - HPR (2)			
2 MLOCA - EIHP (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) MLOCA is defined as equivalent break sizes from 2.0 to 5.0 inches.

(2) The human error probability for switch over to recirculation is 3.2E-3.

Table 2.6 SDP Worksheet for Catawba Nuclear Station, Units 1 and 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, Accumulators (EIAC)	3/4 Accumulators (1 train)
Low Pressure Injection (EILP)	1/2 LPSI (RHR) trains with suction from FWST (1 multi-train system)
Low Pressure Recirculation (LPR)	1/2 LPSI (RHR) trains with suction automatically transferred to containment sump (1multi-train system but potential for single failure[1])

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 LLOCA - LPR (2)			
2 LLOCA - EILP (3)			
3 LLOCA - EIAC (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) LLOCA is defoend as break sizes greater than 5.0 inches.

(2) The miscalibration of level monitor in the FWST could cause premature transfer and damage of the LPSI pumps.

Table 2.7 SDP Worksheet for Catawba Nuclear Station, Units 1 and 2 —— LOOP

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

Safety Functions Needed:	***Full* Creditable Mitigation Capability for Each Safety Function:**
Emergency AC Power (EAC)	1/2 EDG (1 multi-train system) or recovery of offsite power in less than 0.5 hours[(1)] (Recovery Action)
PORV Fail to Close (SORV)	3/3 PORV re-close after opening (1 train system) or operator closesthe block valve associated with the failed PORV (Operator action)
Turbine-driven AFW Pump (TDAFW)	1/1 TDAFW train (1 ASD train)
Safe Shutdown Facility (SSF)	One Diesel generator and the makeup pumps with associated support injecting to pump seals (operator action under high stress)[(2)]
Secondary Heat Removal (AFW2)	1/2 MDAFW trains to 2/4 SGs (1 multi-train system)
Recovery of AC Power in < 2 hrs (REC2)	Recovery of AC source in less than 2 hours [(3)] (operator action under high stress)
Recovery of AC Power in < 4 hrs (REC4)	Recovery of AC source in less than 4 hours [(4)] (operator action)
Early Inventory, HP Injection (EIHP)	(1/2 NV trains (1 multi-train system) or 2/2 NI trains (1 train)) to 3/4 loops
Feed and Bleed (FB)	2/3 PORVs open for Feed and Bleed (Operator action)[(5)]
High Pressure Recirculation (HPR)	(1/2 NV trains or 1/2 NI trains) taking suction from 1/2 LPSI (RHR) trains (operator action)[(6)]

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 LOOP - TDAFW - AFW2 - HPR (4,14)			
2 LOOP - TDAFW - AFW2 - FB (5,15)			
3 LOOP - TDAFW - AFW2 - EIHP (6,16)			

4 LOOP - SORV - HPR (8,11,29) (AC recovered only for 29)			
5 LOOP - SORV - EIHP (9,12) (AC recovered only for 30)			
6 LOOP - EAC - REC4 (18)			
7 LOOP - EAC - SSF - HPR (20) (AC recovered in 2 hours)			
8 LOOP - EAC - SSF - EIHP (21) (AC recovered in 2 hours)			
9 LOOP - EAC - SSF - REC2 (22)			
10 LOOP - EAC - TDAFW - HPR (24) (AC recovered in 2 hours)			
11 LOOP - EAC - TDAFW - FB (25) (AC recovered in 2 hours)			
12 LOOP - EAC - TDAFW - EIHP (26) (AC recovered in 2 hours)			
13 LOOP - EAC - TDAFW - REC2 (27)			

14 LOOP - EAC - SORV - REC2 (31)			
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 assumes conservatively that significant RCP seal leakage without injection and cooling could start in about 15 minutes and SG dry out in about 20 to 30 minutes. We therefore assume if the AC is recovered in less than 30 minutes, neither seal cooling nor SG dry out has occurred.

(2) The IPE uses a human error probability of about 5E-2 which is considered here as high stress operator action.

(3) In SBO scenarios; it is assumed that it takes approximately 2 hours after the RCP seal LOCA or total loss of secondary cooling for the core damage to occur.

(4) The batteries are expected to deplete in about 3 hours in SBO scenarios, and it is conservatively assumed that core damage would occur in 4 hours.

(5) The human error probability (HEP) for FB in IPE is 1.0E-2.

(6) The human error probability for switch over to recirculation is 3.2E-3.

Table 2.8 SDP Worksheet for Catawba Nuclear Station, Units 1 and 2 —— SGTR

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

Safety Functions Needed:	***Full* Creditable Mitigation Capability for Each Safety Function:**
Secondary Heat Removal (AFW)	(1/2 MDAFW trains (1 multi-train system) or 1 TDAFW train (1 ASD train)) to 2/4 SG
Early Inventory, HP Injection (EIHP)	1/2 NV trains (1 multi-train system) or 1/2 NI trains (1 train) to 2/4 loops
Pressure Equalization (EQ)	Rapid cool down through intact secondary SGs using 2/3 SG PORVs or 3/3 open MSIVs (operator action)
Primary depress-Pzr Spray (PDS)	Pzr. Normal or Aux. Sprays and throttling of injection flow (high stress operator action)(1)
Primary depress- PORVs (PDP)	1/3 PORVs open (high stress operator action)(2)
Feed-and-Bleed (FB)	2/3 PORVs open for Feed and Bleed (Operator action)(3)
High Pressure Recirculation (HPR)	(1/2 NV trains or 1/2 NI trains) taking suction from 1/2 LPSI (RHR) trains (operator action)(4)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 SGTR - EQ - PDS - PDP (4)			
2 SGTR - EQ - EIHP (5)			
3 SGTR - AFW - HPR (7)			
4 SGTR - AFW - FB (8)			
5 SGTR - AFW - EIHP (9)			

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

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

Notes:

General Note: The function "equalization" refers to rapid cool down using the intact SG such that the primary pressure equalized with secondary pressure before the SG PORVs on the damaged SG lifts. This is typically a high stress operator action. The licensee calls this function RCSCOOL and the associated HEP values indicate normal operator action.

(1) IPE value for operator error probability for PDS is 0.1 .

(2) The human error probability (HEP) assessed in the IPE for PDP is 0.1 .

(3) The HEP for FB assessed in IPE is 1.0E-2. Note that failure of PORV to re-seat and the failure of operator to close block valve is not included in the event tree since the likelihood is low and the sequences are covered under SORV.

(4) The human error probability for switch over to recirculation is 3.2E-3.

Table 2.9 SDP Worksheet for Catawba Nuclear Station, Units 1 and 2 —— 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>:**
Emergency Boration (HPI)	1/2 NV trains injecting into 3 or more loops (operator action)(1)
Turbine trip (TTP)	2/2 AMSAC channels (1 train)
Primary Relief (SRV)	2/3 PORVs (1 train system) or 2/3 SRVs (1 train system)(2)
Secondary Heat Removal (AFW)	2/2 MDAFW trains (1 train)

<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 - SRV (2)			
2 ATWS - AFW (3)			
3 ATWS - HPI (4)			
4 ATWS - TTP (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) IPE assigns an operator failure probability of 0.01.

(2) The IPE and the licensee submitted work sheets do not discuss the opening of PORV and SRVs. The success criteria are assumed based on previous studies and need to be verified. The favorable MTC (typically 95% of the time) is assumed.

Table 2.10 SDP Worksheet for Catawba Nuclear Station, Units 1 and 2 —— MSLB/FLB

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:**
Break Isolation (ISO)	Isolation of break depending on location[1] (operator action)
Power Conversion System (PCS)	1/2 feedwater trains to 2/3 SG (High stress operator action)[2]
Secondary Heat Removal (AFW)	1/2 MDAFW trains (1 multi-train system) or 1 TDAFW train (1 ASD train) to 2/3 SG
Early Inv., High Pressure Injection (EIHP)	1/2 NV trains (1 multi-train system) or 2/2 NI trains (1 train) to 3/4 loops
Primary Heat Removal, Feed/Bleed (FB)	2/3 PORVs open for Feed and Bleed (Operator action)[3]
High Pressure Recirculation (HPR)	1/2 NV trains or 1/2 NI trains taking suction from 1/2 LPSI (RHR) trains (operator action)[4]

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 MSLB/FLB - PCS - AFW - HPR (4)			
2 MSLB/FLB - PCS - AFW - EIHP (5)			
3 MSLB/FLB - PCS - AFW - FB (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) SLB inside containment will cause actuation of containment spray, and the affected SG can not be isolated. SLB outside containment and FLB inside or outside containment is assumed not to actuate containment spray, and they typically can be isolated. However consistent with IPE, the affected SG is not credited. The event tree is similar to transient and has not been developed.

(2) The HEP value in the IPE is 2.0E-1.

(3) The human error probability (HEP) for FB in IPE is 1.0E-2.

(4) The human error probability for switch over to recirculation is 3.2E-3.

Table 2.11 SDP Worksheet for Catawba Nuclear Station, Units 1 and 2 —— Loss of Nuclear SW/CCW (TRN) (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:**
Safe Shutdown Facility (SSF)	One Diesel generator and the makeup pumps with associated support injecting to pump seals (high stress operator action)(2)
Secondary Heat Removal (AFW)	1/2 TDAFW train (1 ASD train) to 2/3 SG

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 TRN - SSF			
2 TRN - AFW			

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 loss of RN is estimated to be 4E-3 per year. Loss of CCW (KC) result in similar sequences as loss of RN and has a frequency of 6.4E-4. Therefore the total estimated frequency for this sheet is about 1E-3 .It results in reactor trip due to overheating of RCP motors. The TDAFW and the SSF systems will be available.

(2) The IPE uses a human error probability of about 5E-2 for alignment of SSF which is considered here as high stress operator action.

Table 2.12 SDP Worksheet for Catawba Nuclear Station, Units 1 and 2 — Loss of 4160 Essential Bus A (T4A)(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:**
Power Conversion System (PCS)	1/2 feedwater trains to 2/4 SG (High stress operator action)(2)
Secondary Heat Removal (AFW)	1/1 MDAFW trains (1 train system) or 1/2 TDAFW train (1 ASD train) to 2/4 SG
Early Inventory, High Pressure Injection (EIHP)	1/1 NV trains (1 train system) to 3/4 loops
Primary Heat Removal, Feed/Bleed (FB)	2/3 PORVs open for Feed and Bleed (Operator action)(3)
High Pressure Recirculation (HPR)	(1/1 NV trains or 1/1 NI trains) taking suction from 1/1 LPSI (RHR) trains (operator action)(4)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 T4A - PCS - AFW - HPR (4)			
2 T4A - PCS - AFW - EIHP (5)			
3 T4A - PCS - AFW - FB (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) This special initiator is loss of 4 KV essential bus with a frequency of 3.8E-3 per year. The events are enveloped by loss of bus A. This special initiator will cause the loss of one train (operating train) of all system. It would also behave like transients with reduced mitigating capability.

(2) The HEP value in the IPE is 2.0E-1.

(3) The human error probability (HEP) for FB in IPE is 1.0E-2.

(4) The human error probability for switch over to recirculation is 3.2E-3.

Table 2.13 SDP Worksheet for Catawba Nuclear Station, Units 1 and 2 — Loss of 125 V-DC 1EDF Bus (TIED) (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>:**		
Secondary Heat Removal (AFW)	1/2 MDAFW trains (1 train system) or 1/1 TDAFW train (1 ASD train) to 3/4 SG		
<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 T1ED - AFW			
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) This special initiator is loss of vital I&C which is bounded by loss of bus 1EDF with a frequency of 5E-2 per year. This special initiator will cause the loss of Feed water and auto-actuation of one train (non-operating train) of all system. It would also impact the operation of 2/3 PORVs and therefore make Feed and Bleed Unavailable however, SRVs could open in response of primary over-pressurization but long term cooling is not viable. SRVs could fail to re-seat which in this case establishes a path for Feed and Bleed but it is not credited. The AFW pumps will be available but may require some manual actuation. Transient with loss of PCS and feed and bleed could be used for this initiator. This initiator did not make it to dominant sequences in IPE . Therefore, this worksheet may need to be reviewed and verified by Licensee.

Table 2.14 SDP Worksheet for Catawba Nuclear Station, Units 1 and 2 — Transients with Loss of Instrument Air [1] (TIA)

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

Safety Functions Needed:	***Full* Creditable Mitigation Capability for Each Safety Function:**
Secondary Heat Removal (AFW)	1/2 MDAFW trains or 1 TDAFW train to 2/4 SG (operator action)[2]
Early Inventory, High Pressure Injection (EIHP)	1/2 NV trains (1 multi-train system) or 2/2 NI trains (1 train) to 3/4 loops
Primary Heat Removal, Feed/Bleed (FB)	2/3 PORVs open for Feed and Bleed (Operator action)[3]
High Pressure Recirculation (HPR)	(1/2 NV trains or 1/2 NI trains) taking suction from 1/2 LPSI (RHR) trains (operator action)[4]

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 TRANS - AFW - HPR (4)			
2 TRANS - AFW - EIHP (5)			
3 TRANS - AFW - FB (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 frequency of loss of instrument air is estimated to be 0.1 per year. It causes loss of feed water and MSIV closure. The IPE claims both the primary and the secondary PORVs have enough nitrogen backup (2.2-5). The nitrogen backup for secondary POTVs is not addressed in IPE.

(2) The AFW is assumed to be an operator action since the SG-PORVs are not available.The AFW valves will be fully open in loss of IA which could cause SG overfill. The operator action is for controlling flow to avoid SG overfill in this condition.

(3) The human error probability (HEP) for FB in IPE is 1.0E-2.

(4) The human error probability for switch over to recirculation is 3.2E-3.

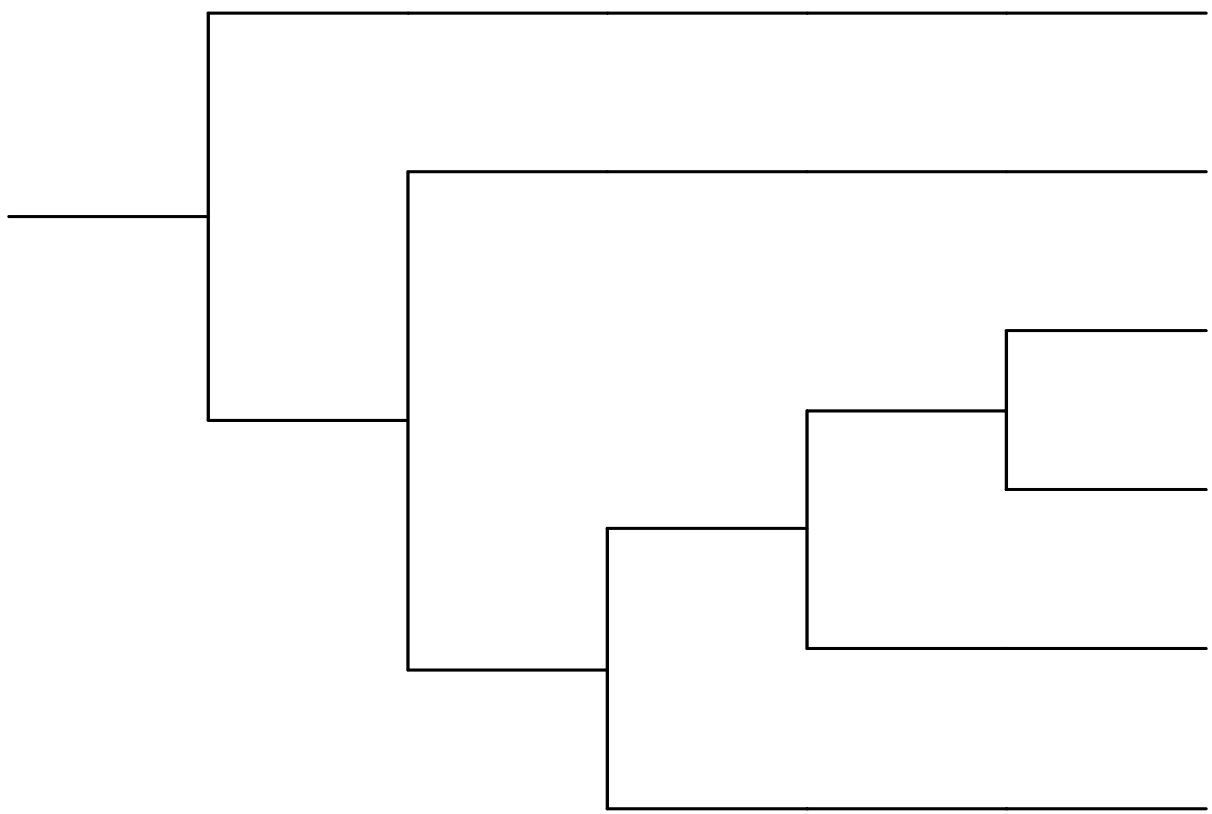
1.3 SDP EVENT TREES

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:

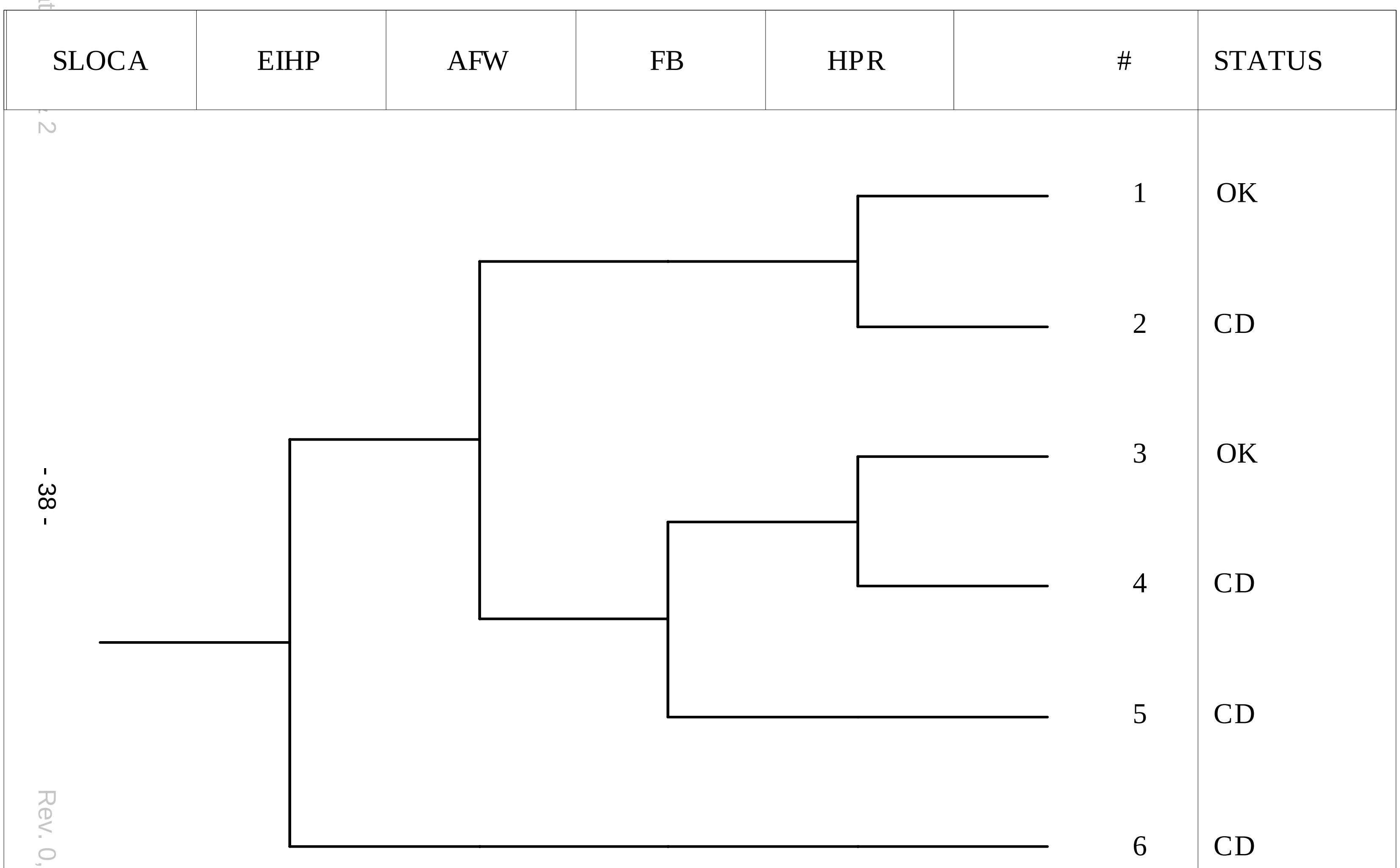
1. Transients
2. Small LOCA
3. Medium LOCA
4. Large LOCA
5. LOOP
6. Steam Generator Tube Rupture (SGTR)
7. Anticipated Transients Without Scram (ATWS)

TRANS	PCS	AFW	FB	EIHP	HPR	#	STATUS

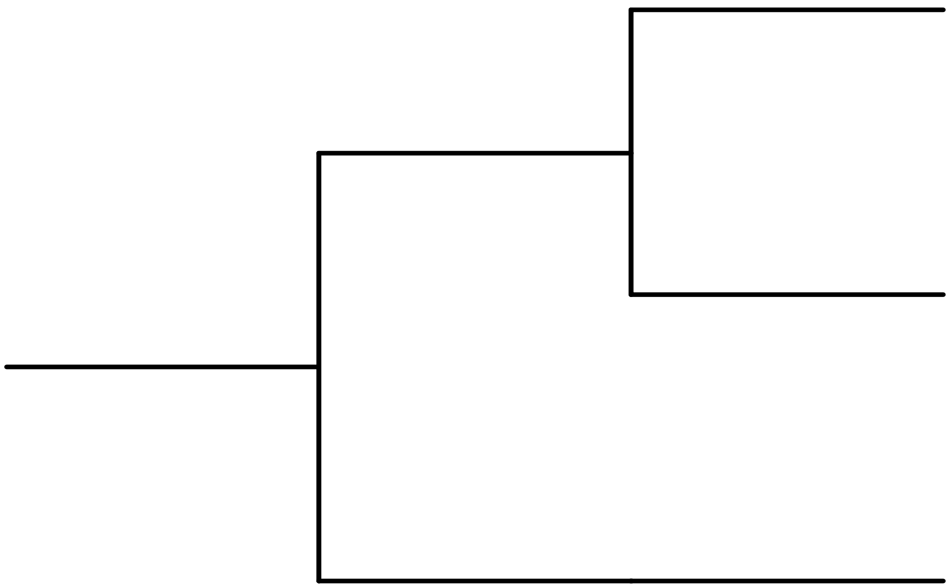


#	STATUS
1	OK
2	OK
3	OK
4	CD
5	CD
6	CD

Plant Name Abbrev.: CATA



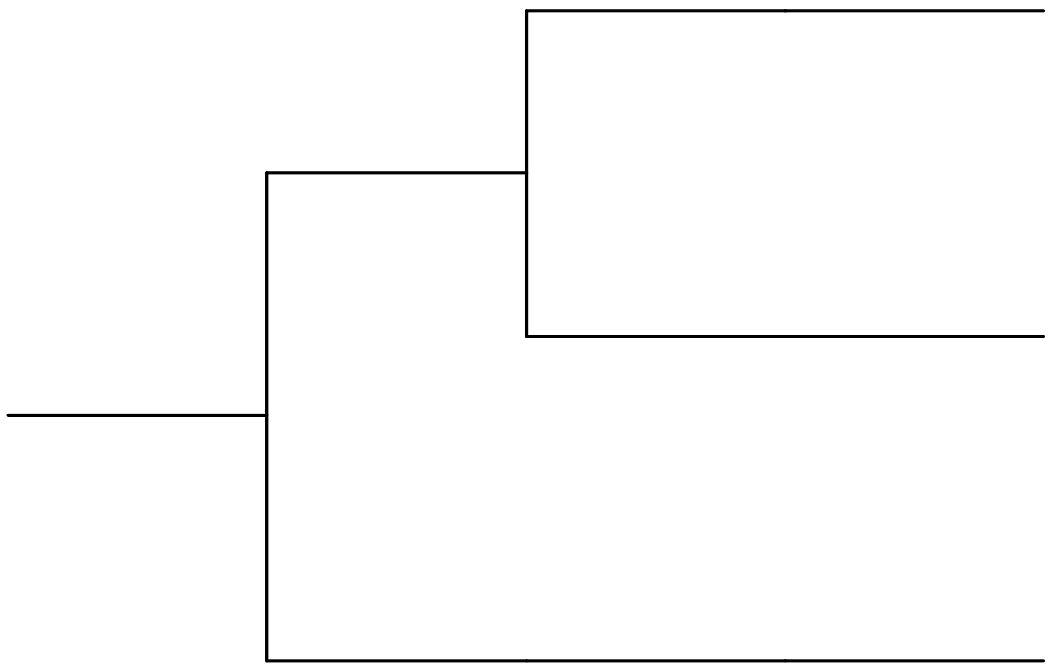
Plant Name Abbrev.: CATA



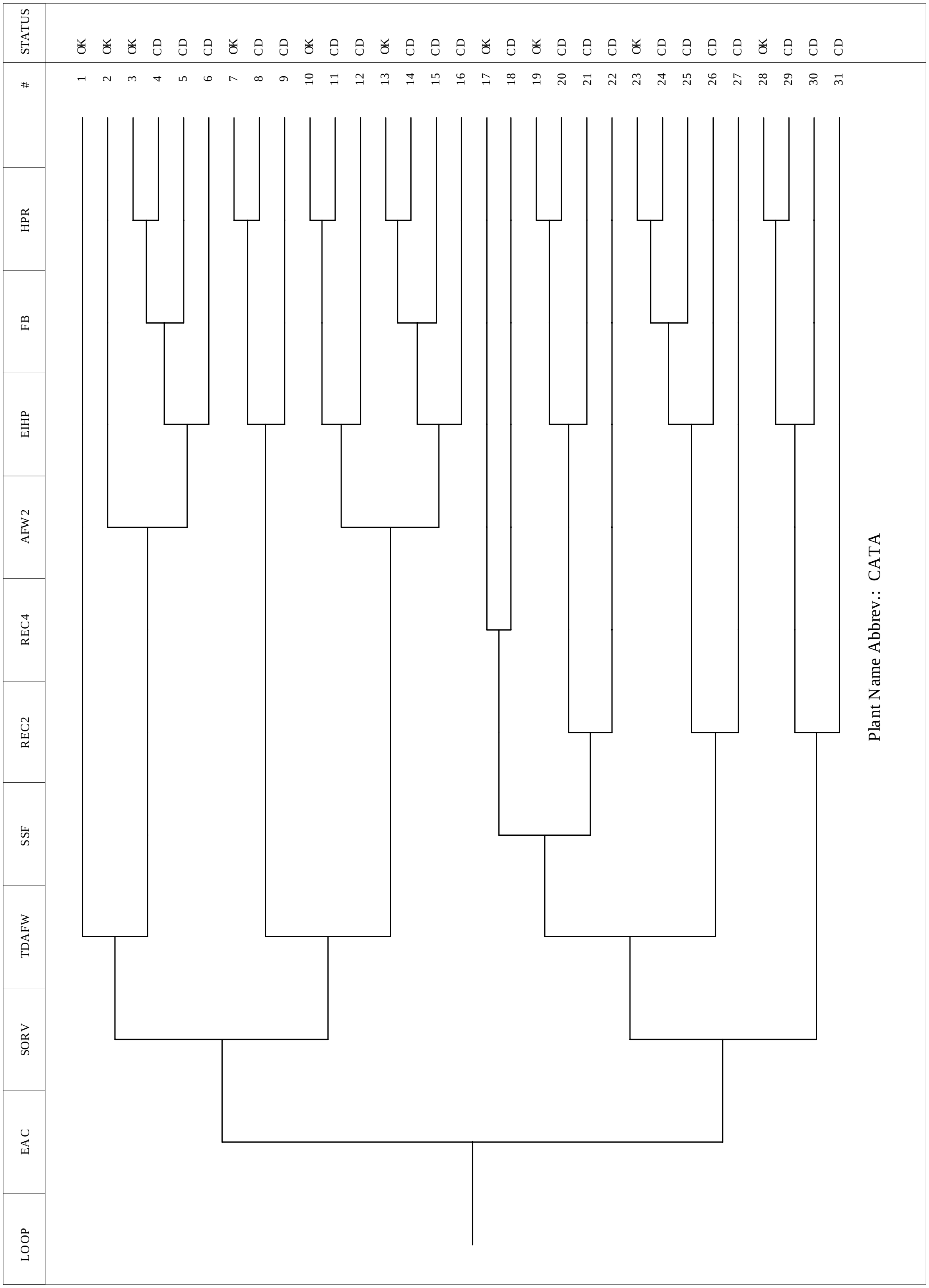
MLOCA	EIHP	HPR	#	STATUS
			1	OK
			2	CD
			3	CD

Plant Name Abbrev.: CATA

LLOCA	EIAC	EILP	LPR	#	STATUS
				1	OK
				2	CD
				3	CD



Plant Name Abbrev.: CATA


STATUS
#
HPR
FB
EIHP
AFW2
REC4
REC2
SSF
TDAFW
SORV
EAC
LOOP
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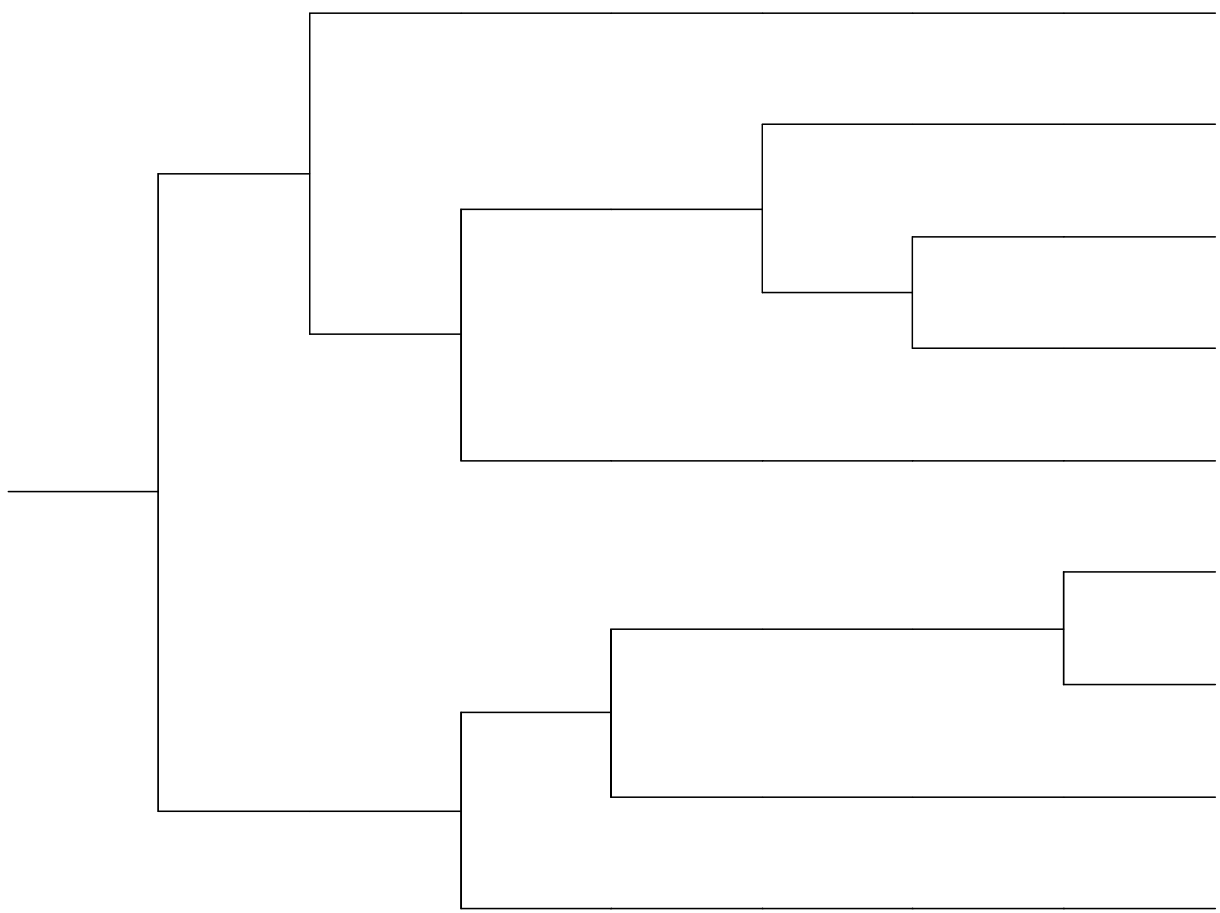
3 OK
4 CD
5 CD
6 CD
7 OK
8 CD
9 CD
10 OK
11 CD
12 CD
13 OK
14 CD
15 CD
16 CD
17 OK
18 CD
19 OK
20 CD
21 CD
22 CD
23 OK
24 CD
25 CD
26 CD
27 CD
28 OK
29 CD
30 CD
31 CD
Plant Name Abbrev.: CATA

SGTR	AFW	EQ	EIHP	FB	PDS	PDP	HPR	#	STATUS



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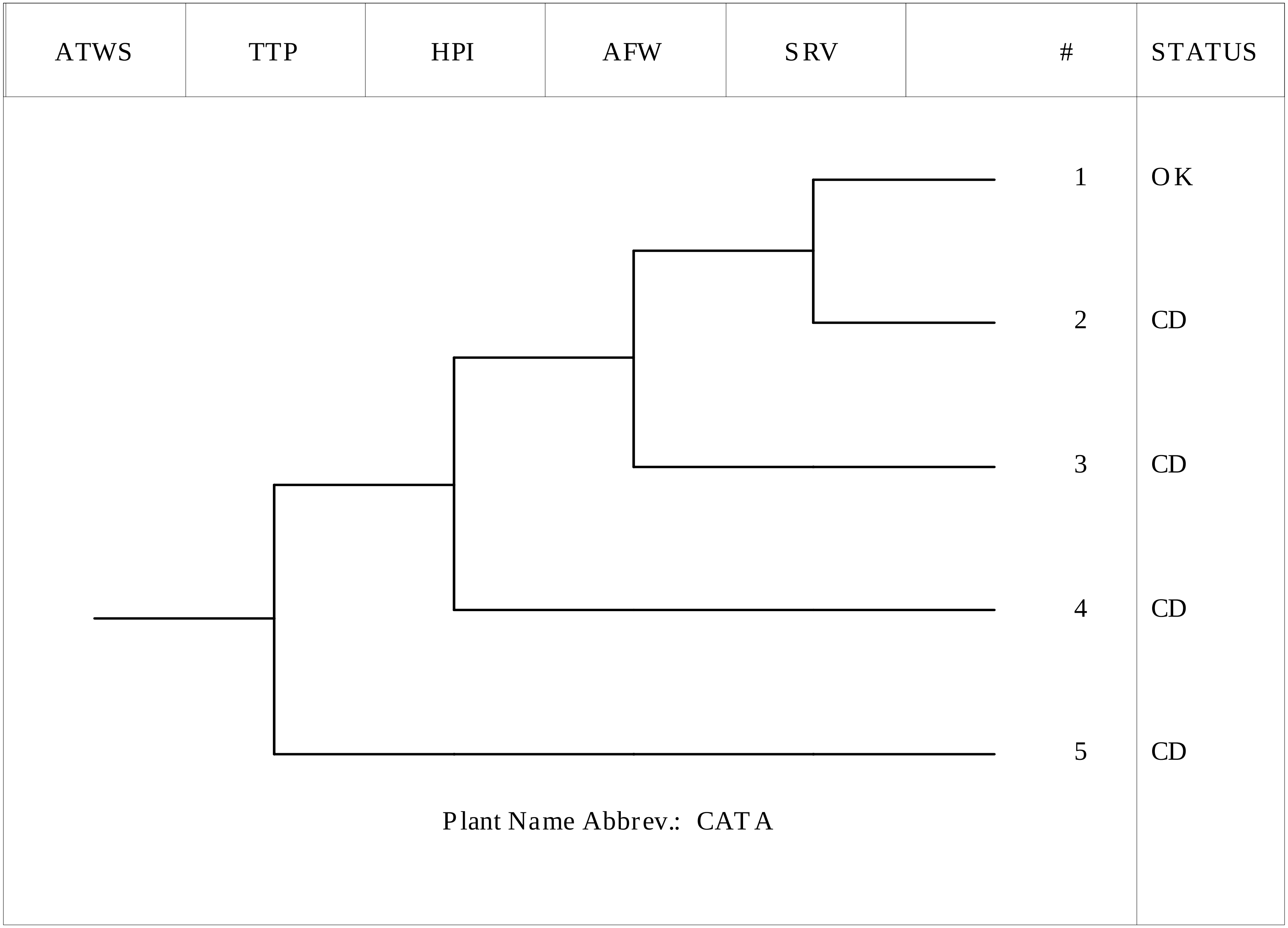
2	OK
3	OK
4	CD
5	CD
6	OK
7	CD
8	CD
9	CD

Plant Name Abbrev.: CATA


ATWS
TTP
HPI
AFW
SRV
#
STATUS
1
OK
2
CD
3
CD
4
CD
5
CD
Plant Name Abbrev.: CATA

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. Duke Power Company, "Catawba Nuclear Station, Units 1 & 2 Individual Plant Examination Report," September 1992.