March 11, 2000

Mr. Mike Reandeau
Director - Licensing
Clinton Power Station
P.O. Box 678
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Clinton, IL 61727

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

Dear Mr. Reandeau:

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

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

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

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

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.

Contact me if you have any questions.

Sincerely,

/RA/

Jon B. Hopkins, Senior Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-461

Enclosure: As stated

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RISK-INFORMED INSPECTION NOTEBOOK FOR

CLINTON POWER STATION

UNIT 1

BWR-6, GE, WITH MARK III 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 Clinton Power Station, 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). This table is used to identify the SDP worksheets to be evaluated, corresponding to the inspection's findings on systems and components.

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

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

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

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

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

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

4. The simplified event-trees focus on classes of initiators, as defined above. In so doing, many separate event trees in the IPEs often are represented by a single tree. For example, some IPEs define four classes of LOCAs rather than the three classes considered here. Such differentiations generally are not considered in the SDP worksheets unless they could not be accounted for by the Initiator and System Dependency table.

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

The three sections that follow include the initiators and dependency table, SDP worksheets, and the SDP event-trees for the Clinton Power Station, 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 (Continued)

Table 1 Initiators and System Dependency Table for Clinton Power Station, Unit 1

Affected Systems	Major Components	Support Systems	Initia
Power Conversion System / Feedwater Delivery	Two TD & one MD Feedwater Pumps, Four MD Condensate Booster Pumps, Four MD Condensate Pumps, Feedwater Control (Condensate Storage Tank)	BOP Auxiliary AC Power, BOP DC Power, Plant Service Water (WS), Turbine Oil (TO), Turbine Building Closed Cooling Water (WT)	Trans MLOC
High Pressure Core Spray (HPCS)	MD Pump with Dedicated Standby Diesel Generator	Offsite Power, Div III Aux. AC Power, Div III DC Power, Div IV DC Power, Div III SDSW (SX), PSSW (WS), Nuclear System Protection System (NSPS)	Trans MLOC
Low Pressure Core Spray (LPCS)	MD Pump	Div I Aux. AC Power, Div I DC Power, Div I SDSW (SX), PSSW (WS), TBCCW (WT), Instrument Air / Service Air (IA/SA)	Trans MLOC
Main Steam (MS)	Four Main Steam lines, 16 SRVs, 35% Capacity TBP, Main Condenser, Condenser Air Removal, Circulating Water	BOP AC Power, BOP DC Power, IA/SA	Trans
Automatic Depressurization System (ADS)	Seven (of above 16) Safety Relief Valves (SRVs), Air Accumulators Back-up Air Bottles	Divs I and II AC and DC Power, IA/SA, NSPS	Trans MLOC ATWS
Residual Heat Removal (RHR) / Low Pressure Coolant Injection (LPCI)	Three MD pumps Trains A, B, C	Div I AC and DC Power (Train A), Div II AC and DC Power (Trains B and C), Div I SX (Train A), Div II SX (Trains B and C), PSWS (WS), NSPS	Trans MLOC
RHR/ Suppression Pool Cooling (SPC)	Two MD pumps Trains A and B, MOVs	Same as LPCI Trains A and B	Trans
RHR/ Containment Spray (CS)	Two MD pumps Trains A and B, MOVs	Same as LPCI Trains A and B	*N/A fo*
Reactor Core Isolation Cooling (RCIC)	One TD pump, MOVs	Divs I and II AC Aux Power, Div I DC Power, Div I SX, PSWS (WS), IA/SA, NSPS	Trans
Emergency Core Cooling System (ECCS) / Alternate Rod Insertion Initiation (ARII)	Initiation logic for HPCS, LPCS, RHR, RCIC, and EDGs	Divs I, II, III, IV DC Power	Trans MLOC ATWS
Standby Liquid Control (SLC)	Two MD pumps, explosive valves, neutron absorber storage tank	Divs I and II AC Aux Power	ATWS

Table 1 (Continued)

Affected Systems	Major Components	Support Systems	Initia
Fire Protection System (as RPV injection source)	Three DD pumps, Cross-tie to WS system	Div II AC Aux Power, Div II SX, Div II WS	Trans
Control Rod Drive (as RPV injection source)	Two MD pumps (140 GPM total capacity)	BOP Aux AC Power, WS, WT, IA/SA	Trans
Hydrogen Igniters (HI)	Two divisions of 115 (total) glow plug type igniters	Divs I and II Aux AC Power	*N/A fo*
Containment Venting (CV)	One flow path thru RHR to spent fuel pool One flow path thru FC to spent fuel pool One flow path thru CCP to atmosphere	Divs I and II Aux AC Power, BOP Aux AC Power, IA/SA, Fuel Pool Cooling and Cleanup (FC), Continuous Containment Purge(CCP or VR)	*N/A fo*
Containment Isolation (CI)	Initiation logic to close containment isolation valves	Divs I, II, III, IV DC Power	Trans MLOC
Auxiliary AC Power (Divisional and BOP AC power)	Two offsite sources, switchyard, Three diesel generators, diesel oil, diesel ventilation	Offsite power, unit auxiliary transformers (UAT), reserve auxiliary transformer (RAT), emergency reserve auxiliary transformer (ERAT), Divs I, II, and III emergency diesel generators (EDGs), diesel oil, diesel ventilation HVAC, SX, Divs I, II, and III DC power	Trans MLOC ATWS
Direct Current Power	Inverters, Nuclear System Protection System (NSPS) Power Supplies, Switchgear Heat Removal	Six 125 V-DC batteries and chargers, motor control centers, auxiliaries, DC buses, eight 120 V Aux AC power buses, inverters, power supplies, inverter cooling systems	Trans MLOC ATWS
Instrument Air (IA) (excluding ADS air supplies)	Three centrifugal compressors, three service air dryers, IA ring headers	Compressor lube oil cooling water (Component Cooling - CC)	Trans MLOC
Shutdown Service Water (SX) System (includes Emergency Core Cooling System (ECCS) Heat Removal)	Three MD pumps, strainers, three independent subsystem headers, three SX heat exchangers	Divs I, II, and III Aux AC Power, Divs I, II, and III DC Power	Trans MLOC
Miscellaneous Support:			
Plant Service Water System (WS)	Three MD pumps, two strainers, two WS seal water pumps	BOP Aux AC Power, BOP DC Power	Trans MLOC
Turbine Building Closed Cooling Water System (WT)	Two MD pumps, two heat exchangers	BOP Aux AC Power, BOP DC Power, WS	Trans MLOC

Table 1 (Continued)

Affected Systems	Major Components	Support Systems	Initia
Component Cooling Water System (CC)	Three MD pumps, two heat exchangers	BOP Aux AC Power, BOP DC Power, WS	Trans MLOC

Notes:

(1) Transient scenarios should be developed from those transient initiators that could have the greatest risk significance. For example, develop loss of DC bus transient scenarios for degraded 125 V DC or AC power equipment, as well as other transient initiators that may depend on equipment being supplied from degraded power sources. The choice of which transient scenarios to develop should generally be apparent from the specific given condition.

(2) The above information is based upon the Clinton Power Station Response to Generic Letter 88-20, "Individual Plant Examination for Severe Accident Vulnerabilities" submitted to the NRC by letter dated September 23, 1992.

(3) The overall core damage frequency for internal events and flooding is 2.6E-5 events per reactor-year based on the September 23, 1992 IPE submittal.

(4) According to the CLNT IPE, pages 3-28 and 3-29: "Loss of the steam (vapor) suppression system, i.e., bypassing the suppression pool, is postulated to occur only after drywell temperature reaches 700°F because of potential penetration failure. This temperature occurs only after core damage. Loss of steam suppression could also be postulated to occur either by bypassing the suppression pool or by a loss of pool inventory. Bypass of the drywell at lower temperatures is not considered feasible because two vacuum breakers in series which are used to vent into the drywell would have to fail. Loss of suppression pool inventory, such that the weir vents become uncovered, is only expected to occur if containment pressure reaches 93.75 psig. Failure of emergency core cooling system (ECCS) suction piping which penetrates below the suppression pool water is not considered credible because this piping is exposed to low pressure conditions and is seismically qualified. The treatment of suppression pool capability is consistent with the assumption made for Grand Gulf in NUREG/CR-4500.

Upper pool dump is not required for maintaining net positive suction head for the ECCS pumps in the event of various LOCAs. A conservative calculation was performed to determine the minimum suppression pool inventory following a LOCA. This calculation assumed that the drywell volume to the top of the weir wall was completely filled with water from the suppression pool following a LOCA. Additionally, the suppression pool inventory was assumed to be further reduced by ECCS system operation to restore reactor vessel inventory. This calculation proved that suppression pool inventory is sufficient to provide adequate NPSH for all ECCS pumps and maintain adequate weir vent coverage."

(5) There are nine (9) automatic depressurization system (ADS) safety relief valves (SRVs) and seven (7) power operated SRVs. Upon a low reactor water level signal, these seven SRVs would be isolated from their normal sources of instrument air (IA) and since these SRVs are not connected to a backup air supply and their accumulators are not large enough to supply air for the entire mission time, *they were not included in the IPE PRA model for automatically initiated depressurization.* (See CLNT IPE, pages 3-18, 3-78 and 3-79).

CLNT RAI response to Question 13a dated November 22, 1995: "ADS was the only means of depressurization modeled. Depressurizing through the bypass valves using the pressure regulator or bypass valve jack was not modeled in the CLNT PRA although these actions are allowed in the emergency operating procedures....Note: alternate depressurization methods have been included in PRA updates subsequent to IPE submittal, but failure to initiate ADS is still very important...."

Table 1 (Continued)

The initial screening value for the operator failing to manually initiate ADS was 2.8E-3. The final value was 5.0E-4. (See CLNT IPE, Table 3.3-5, page 3-199 and RAI response to Question 31a dated November 22, 1995).

(6) According to CLNT IPE pages 3-221 and 3-28: "Analysis has shown that the ECCS pumps can take suction from the suppression pool even under saturation conditions. Low pressure core spray (LPCS), high pressure core spray (HPCS), and residual heat removal (RHR) pumps (in the low pressure coolant injection, LPCI, mode) do not lose suction after loss of containment heat removal or depressurization following containment venting or containment failure unless the failure is in the suppression pool. If the suppression pool were at saturation conditions, analysis (USAR 6.3.1.1.3) shows that sufficient net positive suction head remains available."

Also, CLNT Response to RAI Question 15a, dated November 22, 1995: "CLNT has a Mark III containment with a large suppression pool and net free air volume. For sequences in which the reactor is shut down and core damage is averted (ECCS systems running), the containment would not fail during the 24 hour mission time of the IPE, even if containment heat removal systems were unavailable. The decay heat energy added to the containment is insufficient to cause enough of the suppression pool inventory to boil to exceed the ultimate strength capacity of the containment....For non-ATWS sequences in which core damage occurs, there may be sufficient hydrogen generated to cause containment failure due to hydrogen burns. However these sequences already involved core damage (ECCS and other injection systems failed) and containment failure so the sequence determination has already been made."

(7) If the SRVs are used to depressurize the reactor, or if the RCIC system is operating, then at least one loop of the RHR system is aligned in the suppression pool cooling mode to remove heat from the containment. (If there is a large break LOCA and pressure is increasing inside containment, the RHR system can be aligned to the containment spray (CS) mode. Only the suppression pool cooling mode of RHR is modeled in the CLNT Level 1 PRA and only as a support for successful RCIC operation. The shutdown cooling (SDC) mode of RHR is not included in the model because it is not needed to prevent core damage during the 24 hour mission time of the IPE. The CS mode of RHR is modeled in the containment analysis (Level 2) because its primary function is to maintain containment integrity. Success for containment heat removal is successful operation of one train of RHR in the suppression pool cooling mode. (CLNT IPE, pages 3-21 and 3-22).

(8) Because the ECCS systems would not be affected by containment performance during the mission time of the Level 1 analysis, no credit for containment venting was required in the Level 1 analysis. (CLNT RAI response to Question 15b dated November 22, 1995).

1.2 SDP WORKSHEETS

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

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

Table 2.1 SDP Worksheet for Clinton Power Station, Unit 1 —— Transients

Estimated Frequency (Table 1 Row) __________ Exposure Time __________ Table 1 Result (circle): A B

Safety Functions Needed:	***Full* Creditable Mitigation Capability for Each Safety Function:**
Power Conversion System (PCS)	{[1/ 2 TD or 1/1 MD Feedwater pumps] and 1/ 4 condensate pumps and 1/ 4 pumps and 1/ 4 main steam lines with 35% capacity turbine bypass (TB) anc (Operator Action)
Early Inventory High Pressure Injection (EIHP)	HPCS pump (1 train) or RCIC pump (1 ASD train)
Early Inventory Control Rod Drive Pumps (EICRD)	1/ 2 CRD pumps (1 train) after successful operation of HPCS or RCIC [1]
Depressurization (DEP)	1/16 safety relief valves (SRVs) (auto ADS is inhibited) (High stress operato
Low Pressure Injection (LPI)	[1/3 RHR pumps in LPCI mode](1 multi-train system) or [1/1 LPCS pumps] (condensate (CD) pumps and 1/ 4 condensate booster (CB) pumps if Rx pres condensate booster (CB) pumps if Rx pressure <250 psi]} (operator action) diesel-driven) at Rx pressure <= 73 psi] [2] (Operator Action)
Residual Heat Removal Suppression Pool Cooling (RHRSPC)	{[1/ 2 RHR pumps in suppression pool cooling (SPC) mode for RCIC operati

Circle Affected Functions	**Recovery of Failed Train**	***Remaining* Mitigation Capability Rating for Each Affecte Sequence**
1 Trans - PCS - RHRSPC (4)		
2 Trans - PCS - EIHP - EICRD - LPI (6)		
3 Trans - PCS - EIHP - EICRD - DEP (7)		

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

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

Notes:

(1) After a reactor scram at rated pressure, if the operators took no action to increase control rod drive (CRD) flow, CRD flow would increase automatically to 140 GPM. If the second CRD pump were to be used, it would have to be manually initiated and the total flow rate would be only 150 gpm. This is due to high pressure drop in the

lines. The Clinton IPE PRA model assumed only the 140 gpm flow rate of one CRD pump after 1 hour. (See CLNT IPE pages 3-30, 3-32, and 3-80).

(2) Use of the fire protection system requires that a check valve between WS and FP systems would need to be disassembled. The fire pumps require several hours to align before injection into the reactor pressure vessel can begin. As an injection source, the fire pumps are not modeled as a front-line system but are used as a recovery upon delayed failure of other systems. (CLNT IPE pages 3-20, 3-32 and 3-89).

Table 2.2 SDP Worksheet for Clinton Power Station, Unit 1 —— Small LOCA

Estimated Frequency (Table 1 Row) __________ Exposure Time __________ Table 1 Result (circle): A B

Safety Functions Needed:	***Full* Creditable Mitigation Capability for Each Safety Function:**
Power Conversion System (PCS)	{[1/ 2 TD or 1/1 MD Feedwater pumps] and 1/ 4 condensate pumps ar booster pumps and 1/ 4 main steam lines with 35% capacity turbine by operable condenser} (Operator Action)
Early Inventory High Pressure Core Spray (EIHPCS)	HPCS pump (1 train)
Early Inventory High Pressure RCIC (EIRCIC)	RCIC pump (1 ASD train)
Depressurization (DEP)	1/16 safety relief valves (SRVs) (auto ADS is inhibited) (High stress op
Low Pressure Injection (LPI)	1/3 RHR pumps in LPCI mode (1 multi-train system) or 1/1 LPCS pum condensate (CD) pumps and 1/ 4 condensate booster (CB) pumps if R or [1/ 4 condensate booster (CB) pumps if Rx pressure <250 psi] (Ope
Residual Heat Removal Suppression Pool Cooling (RHRSPC)	{[1/ 2 RHR pumps in suppression pool cooling (SPC) mode for RCIC o action)

Circle Affected Functions	**Recovery or Failed Train**	***Remaining* Mitigation Capability Rating for Each Af Sequence**
1 SLOCA - PCS - EIHPCS - RHRSPC (4)		
2 SLOCA -PCS - EIHPCS - EIRCIC - LPI (6)		
3 SLOCA - PCS - EIHPCS - EIRCIC - DEP (7)		

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

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

Notes:

(1) A small break LOCA does not depressurize the reactor to the point at which low pressure injection systems can provide makeup. High pressure injection systems initially provide makeup. If feedwater (FW) fails, then RCIC provides makeup with suppression pool cooling in operation. If RCIC fails, then HPCS provides makeup. If

HPCS fails, then the reactor must be manually depressurized before low pressure injection systems can provide makeup. (See CLNT IPE page 3-25).

The worksheets consider depressurization (DEP) using SRVs to be a high-stress operator action. The initial screening value for the operator failing to manually initiate ADS was 2.8E-3. The final value was 5.0E-4. (See CLNT IPE, Table 3.3-5, page 3-199 and RAI response to Question 31a dated November 22, 1995).

Table 2.3 SDP Worksheet for Clinton Power Station, Unit 1 —— Medium LOCA

Estimated Frequency (Table 1 Row) __________ Exposure Time __________ Table 1 Result (circle): A B

Safety Functions Needed:

Full **Creditable Mitigation Capability for Each Safety Function:**

Safety Function	Full Creditable Mitigation Capability
Early Inventory (EIHP)	HPCS (1 train)
Depressurization (DEP)	1/16 safety relief valves (SRVs) (auto ADS is inhibited) (High stress operator action)
Low Pressure Injection (LPI)	1/3 RHR pumps in LPCI mode (1 multi-train system) or 1/1 LPCS pumps (1 train) or pumps and 1/ 4 condensate booster (CB) pumps if Rx pressure < 725 psi] or [1/ 4 co (CB) pumps if Rx pressure <250 psi] (Operator Action)

Circle Affected Functions	**Recovery or Failed Train**	***Remaining*** **Mitigation Capability Rating for Each Affected Sequ**
1 MLOCA - EIHP - LPI (3)		
2 MLOCA - EIHP - DEP (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 time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on th conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Notes:

(1) A medium break LOCA does not depressurize the reactor to the point at which low pressure injection systems can provide makeup. The reactor must be manually depressurized for the low pressure injection systems to succeed. RCIC does not have sufficient capacity to maintain coverage of the core. Feedwater is not available because makeup to the condenser may be insufficient. MLOCA is similar to SLOCA except that FW, RCIC, and suppression pool cooling (following RCIC operation) are not included. *[The worksheets assume that suppression pool cooling would be required for containment heat removal (CHR) following DEP but that failure of CHR would not lead to core damage.]* (See CLNT IPE, pages 3-25 and 3-26)

Table 2.4 SDP Worksheet for Clinton Power Station, Unit 1 —— Large LOCA

Estimated Frequency (Table 1 Row) ________ Exposure Time __________ Table 1 Result (circle): A B C

Safety Functions Needed:	***Full*** **Creditable Mitigation Capability for Each Safety Function:**
Early Inventory (EI)	HPCS (1 train) or {1/3 RHR pumps (A, B, or C) in LPCI mode (1 multi-train system) or 1 train) or [1/ 4 condensate (CD) pumps and 1/ 4 condensate booster (CB) pumps if Rx p [1/ 4 condensate booster (CB) pumps if Rx pressure <250 psi] (Operator Action)
Late Inventory Makeup (LI)	{1/3 RHR pumps (A, B, or C) in LPCI mode or 1/1 LPCS pumps}(2 diverse systems) or pumps and 1/ 4 condensate booster pumps if Rx pressure < 300 psi} (Operator Action) diesel-driven) at Rx pressure <= 73 psi (high stress operator action)

Circle Affected Functions	**Recovery or Failed Train**	***Remaining*** **Mitigation Capability Rating for Each Affected Sequen**
1 LLOCA - EI - LI (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 c time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on th conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Notes:

(1) If there is a large break LOCA and pressure is increasing inside containment, the RHR system can be aligned to the containment spray (CS) mode. Only the suppression pool cooling mode of RHR is modeled in the CLNT Level 1 PRA and only as a support for successful RCIC operation. RCIC operation is not applicable to LLOCA. The shutdown cooling (SDC) mode of RHR is not included in the model because it is not needed to prevent core damage during the 24 hour mission time of the IPE. The CS mode of RHR is modeled in the containment analysis (Level 2) because its primary function is to maintain containment integrity. Success for containment heat removal is successful operation of one train of RHR in the suppression pool cooling mode. (CLNT IPE, pages 3-21 and 3-22).

(2) A large break LOCA depressurizes the reactor to a point at which low pressure injection systems can provide makeup. HPCS can also provide makeup. The LLOCA is similar to an MLOCA except that manual depressurization of the reactor is not required. (See CLNT IPE pages 3-25 and 3-26).

Table 2.5 SDP Worksheet for Clinton Power Station, Unit 1 —— (LOOP)

Estimated Frequency (Table 1 Row) __________ Exposure Time __________ Table 1 Result (circle): A B

Safety Functions Needed:	*Full* Creditable Mitigation Capability for Each Safety Function:
Emergency Power Div 1 or Div 2 DGs (EAC)	1/ 2 EDGs (1 multi-train system)[1]
Recovery of Offsite Power within 30 minutes (0.5 Hours) (REC0.5)	High stress operator action[1]
High Pressure Core Spray (HPCS)	HPCS pump (1 train)[1, 2, 4]
Reactor Core Isolation Cooling (RCIC)	RCIC pump (1 ASD train)[1, 2, 4]
Recovery of Offsite Power within 4 hours (REC4)	operator action[1, 3, 4]
Residual Heat Removal Suppression Pool Cooling (RHRSPC)	1/ 2 RHR pumps in SPC mode (operator action)

Circle Affected Functions	Recovery or Failed Train	*Remaining* Mitigation Capability Rating for Each Af Sequence
1 LOOP - RHRSPC (1, 6)		
2 LOOP - HPCS - RCIC - EICRD - LPI (1)		
3 LOOP - HPCS - RCIC - DEP (1)		
4 LOOP - EAC -REC4 (4, 7)		
5 LOOP - EAC - HPCS - RCIC (8)		

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

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

Notes:

(1) In the CLNT IPE, accident class 1B sequences are characterized by a loss of off-site and on-site AC power and a loss of coolant inventory makeup. In addition to those events initiated by a loss of off-site power

(LOOP), other initiating events combined with a subsequent random LOOP are included in this class. If a random failure of HPCS and RCIC occurs early in the event, then level would reach top of active fuel in approximately 25 minutes after the initiating event. If AC power to the battery chargers is not available, then the batteries will eventually be depleted. The batteries are assumed to be available for four (4) hours if load shedding is performed by the operators in one hour. If load shedding is not performed, then the batteries are assumed to fail in one (1) hour. Depletion of the batteries causes failure of RCIC. If RCIC is initially available, then after the batteries are depleted and if HPCS is not available, the level would reach top of active fuel in 1.25 hours. (See CLNT IPE, pages 3-217 to 3-219).

CLNT RAI Response to Question 7b dated November 22, 1995: (Submit basis that injection systems can operate up to 4 hours without HVAC): “Safety-related components located in the ECCS or RCIC rooms including the pumps have been environmentally qualified for severe temperature conditions including a High Energy Line Break (HELB). A heatup analysis was performed using the methodology contained in NUMARC 87-00 for the Low Pressure Core Spray Room. Similar results are expected for the other ECCS rooms. Because the room temperature rise for the first three hours was substantially below the HELB equipment qualification temperature envelope and only slightly above for the period from three to six hours, the assumption was made that the pumps would remain operable for four hours after room cooling was lost.

(2) *For this SDP worksheet, given that the CRD pumps can be used during a Transient, it is assumed that the control rod drive (CRD) pumps can be manually loaded onto the emergency buses to respond as a high pressure injection source during a loss of offsite power.*

(3) Low pressure injection systems are not available unless the off-site power or Division 1 or Division 2 diesel generators are recovered, because air supplies for opening the SRVs to depressurize the reactor vessel would be depleted. (See CLNT IPE, page 3-219).

(4) The diesel driven fire pumps could be used as a low pressure injection source. However, since it takes several hours to align for reactor injection, it is not modeled for this event. (See CLNT IPE, page 3-219).

(5) Important post accident operator actions (human error probabilities) from CLNT IPE Tables 3.3-5 (Final Values) are as follows:

   (1) Operator Fails to Recover Offsite Power Within 0.5 Hours of Loss - 4.2E-1 (from Dominant Cutsets - Attachment to CLNT RAI Response of November 22, 1995, Sequence TLU1U3)

   (2) Operator fails to place a feedpump back in service - 5.0E-4

   (3) Load shedding per CPS 4200.01 not successful (shedding DC battery loads) - 3.0E-2

Table 2.6 SDP Worksheet for Clinton Power Station, Unit 1 —— ATWS

Estimated Frequency (Table 1 Row) __________ Exposure Time __________ Table 1 Result (circle): A B

<u>Safety Functions Needed</u>:	***<u>Full</u>*** **<u>Creditable Mitigation Capability for Each Safety Function:</u>**
Overpressure Protection (OVERP)	4/16 SRVs (1 multi-train system)[1]
Recirculation Pump Trip (RPT)	Manual or automatic trip of recirculation pumps (1 multi-train system)
Reactivity Control (SLC)	1 / 2 SLC pumps and valves (high stress operator action)[2]
Inhibit ADS (INH)	Operator inhibits ADS (high stress operator action)[2]
Main Condenser Heat Sink (CDSR)	{[1/ 2 TD or 1/1 MD Feedwater pumps] and 1/ 4 condensate pumps and 1/ 4 con pumps and 1/ 4 main steam lines with 35% capacity turbine bypass (TB) and op (Operator Action)[3]

<u>Circle Affected Functions</u>	**<u>Recovery or Failed Train</u>**	***<u>Remaining</u>*** **<u>Mitigation Capability Rating for Each Affected Se</u>**
1 ATWS - CDSR (2)		
2 ATWS - INH (3)		
3 ATWS - SLC (4)		
4 ATWS - RPT (5)		
5 ATWS - OVERP (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 time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on t 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) For ATWS, at least four SRVs must function to allow depressurization. (See CLNT IPE page 3-18).

(2) Important post accident operator actions (human error probabilities) from CLNT IPE Tables 3.3-5 (Final Values) are as follows:

a. Operator fails to manually initiate ADS - 5.0E-4
b. Operator fails to initiate SLC A & B - 4.0E-4

(3) According to CLNT IPE pages 3-221 and 3-28: “Analysis has shown that the ECCS pumps can take suction from the suppression pool even under saturation conditions. Low pressure core spray (LPCS), high pressure core spray (HPCS), and residual heat removal (RHR) pumps (in the low pressure coolant injection, LPCI, mode) do not lose suction after loss of containment heat removal or depressurization following containment venting or containment failure unless the failure is in the suppression pool. If the suppression pool were at saturation conditions, analysis (USAR 6.3.1.1.3) shows that sufficient net positive suction head remains available.”

Also, CLNT Response to RAI Question 15a, dated November 22, 1995: “CLNT has a Mark III containment with a large suppression pool and net free air volume. For sequences in which the reactor is shut down and core damage is averted (ECCS systems running), the containment would not fail during the 24 hour mission time of the IPE, even if containment heat removal systems were unavailable. The decay heat energy added to the containment is insufficient to cause enough of the suppression pool inventory to boil to exceed the ultimate strength capacity of the containment. *For the ATWS sequences in which the reactor can not be shut down, core damage is assumed to occur if the main condenser is lost as a heat sink. Therefore, ECCS could not prevent core damage, whether available or not.”*

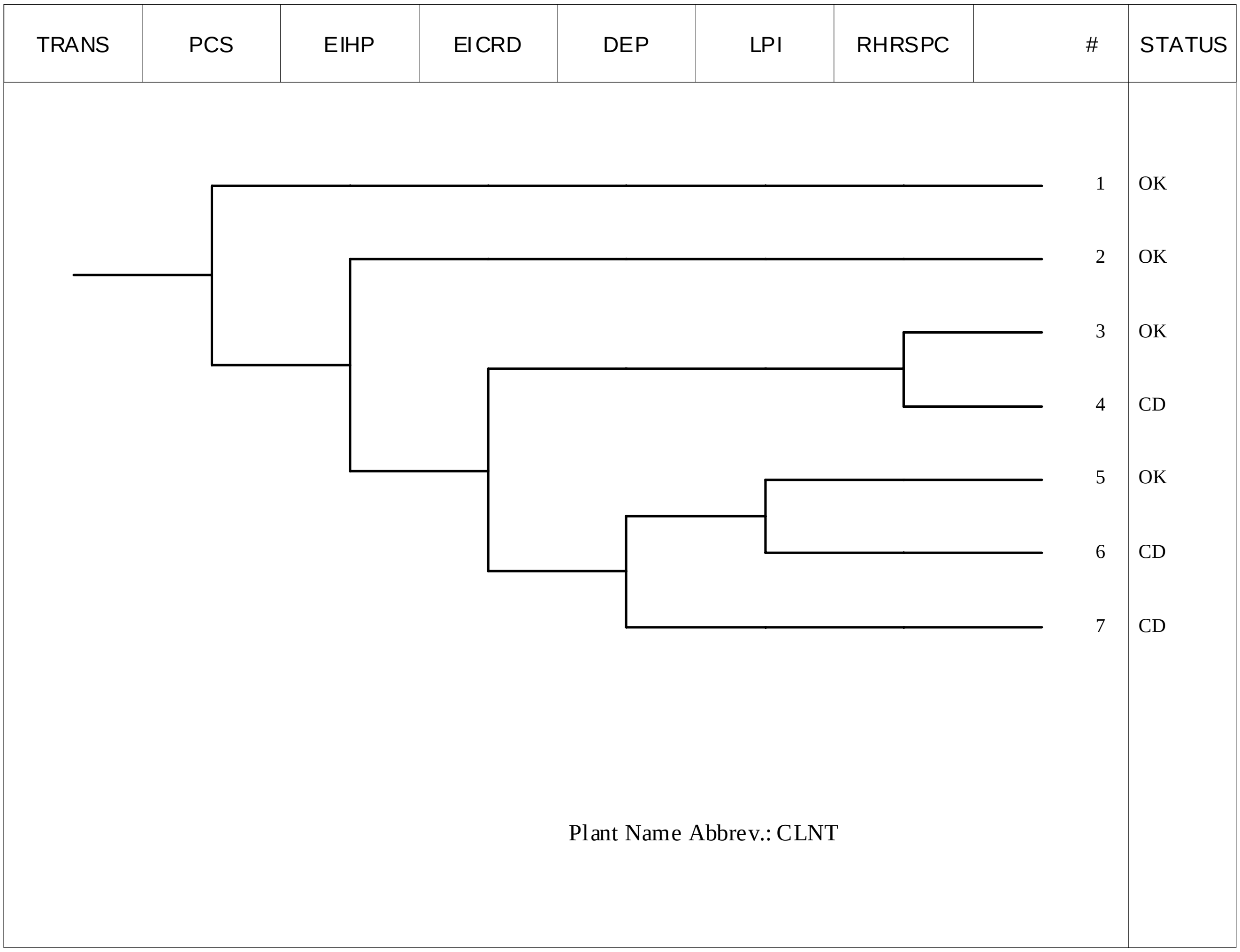
(4) Additional functions [High Pressure Injection (HPI), Depressurization (DEP), Low Pressure Injection (LPI), Residual Heat Removal Suppression Pool Cooling (RHRSPC)] were retained in the CLNT ATWS event trees for potential impact on containment response in the Level 2 PRA analysis only. (See CLNT IPE Page 3-28). These functions are not included in the SDP worksheets for ATWS.

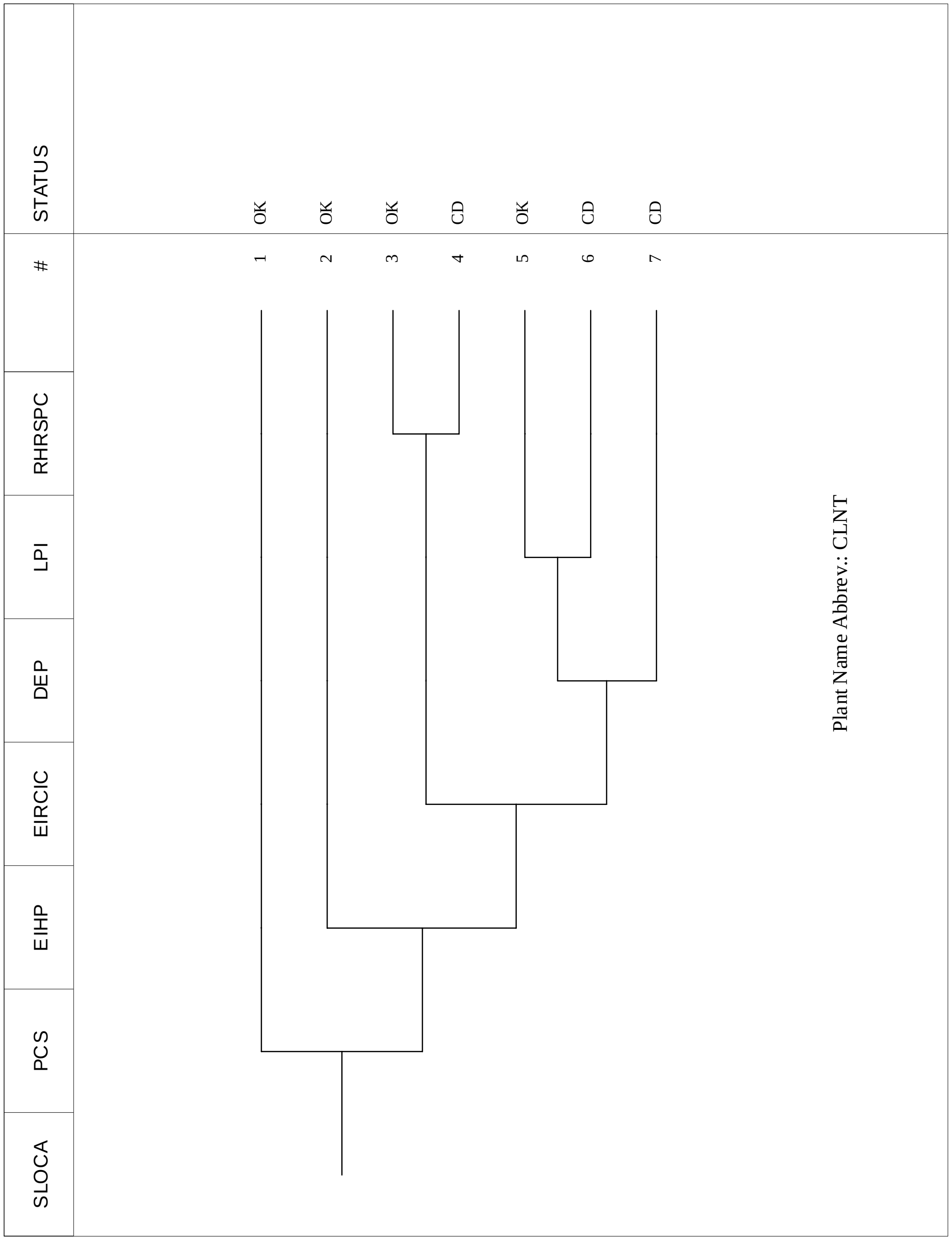
1.3 SDP Event Trees

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

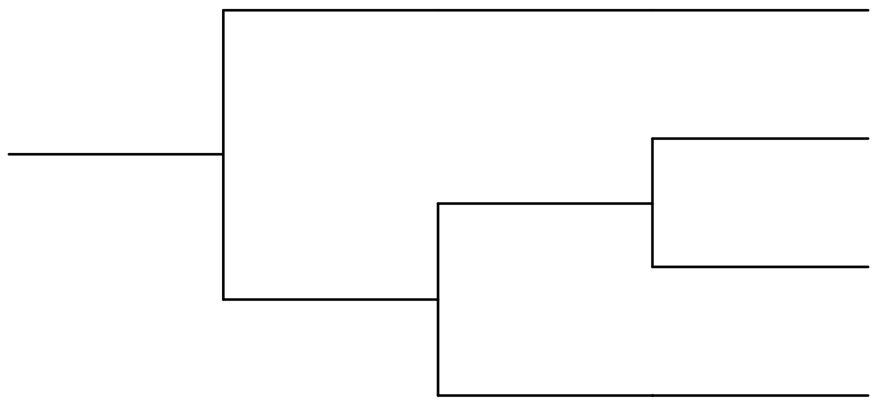
The following event trees are included:

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


TRANS
PCS
EIHP
EICRD
DEP
LPI
RHRSPC
#
STATUS
1 OK
2 OK
3 OK
4 CD
5 OK
6 CD
7 CD
Plant Name Abbrev.: CLNT

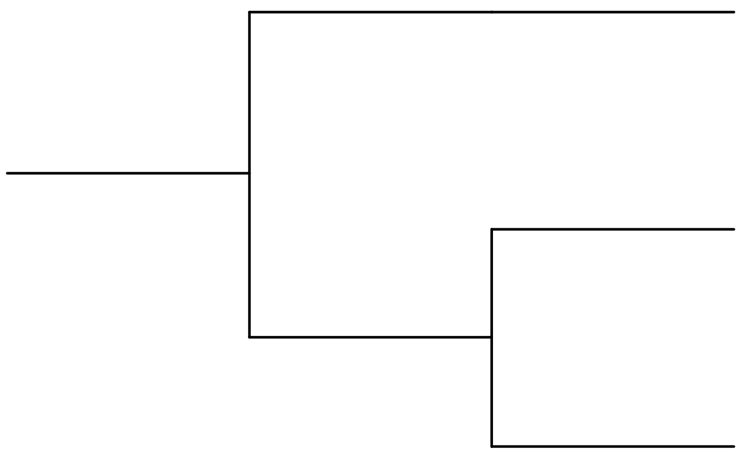

SLOCA
PCS
EIHP
EIRCIC
DEP
LPI
RHRSPC
#
STATUS
1 OK
2 OK
3 OK
4 CD
5 OK
6 CD
7 CD
Plant Name Abbrev.: CLNT

	MLOCA	EIHP	DEP	LPI	#	STATUS
					1	OK
					2	OK
					3	OK
					4	CD

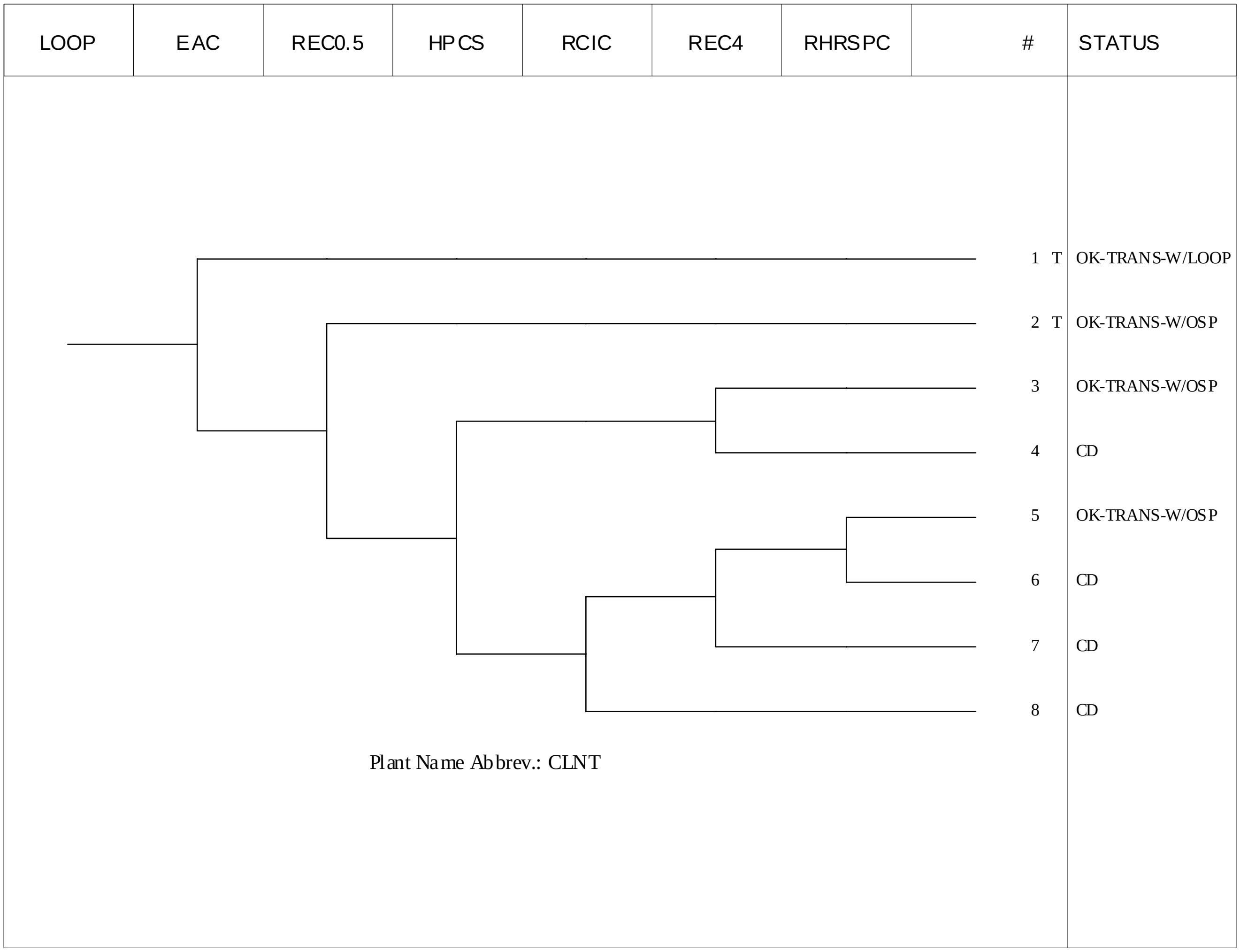


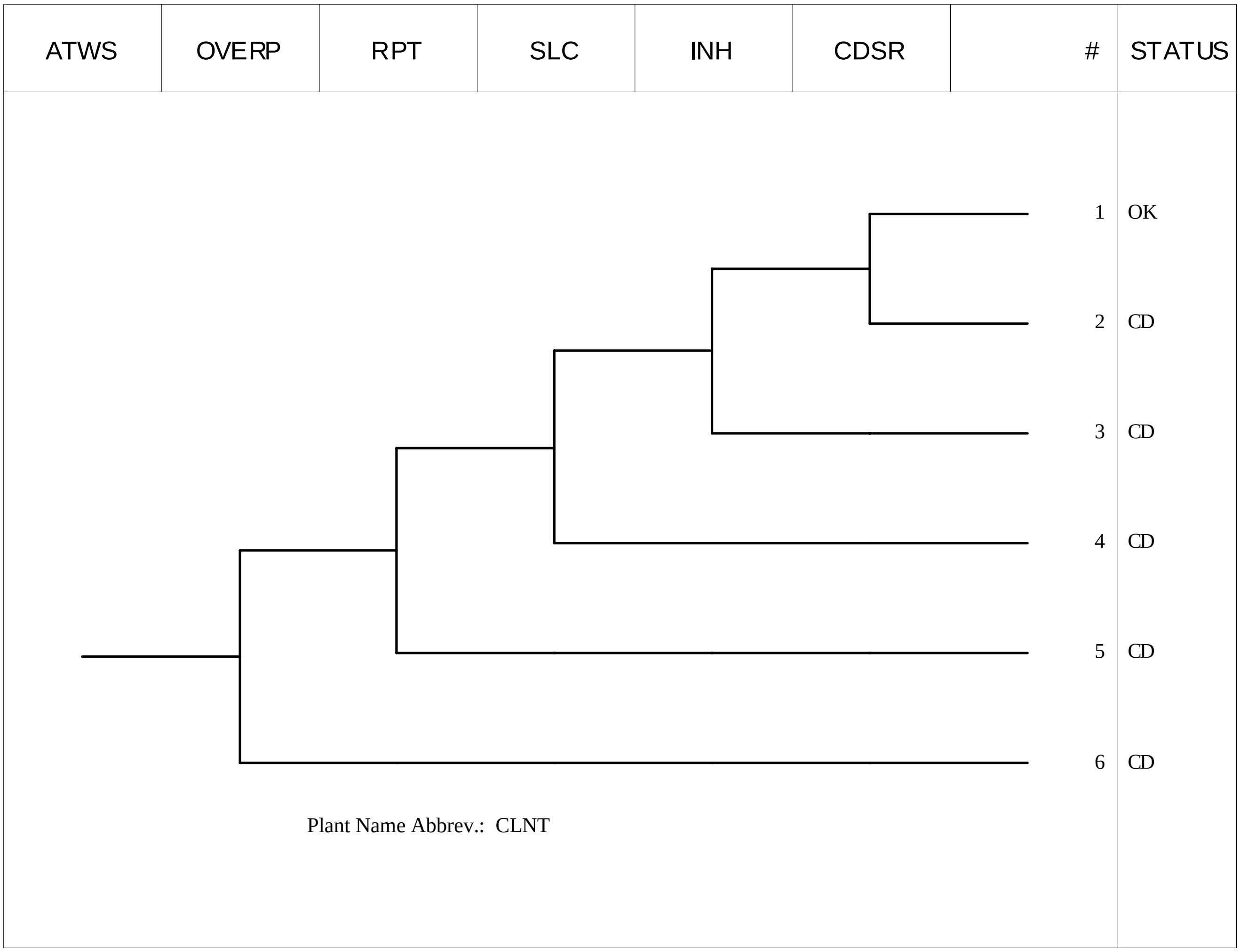
Plant Name Abbrev.: CLNT

	LLOCA	EI	LI	#	STATUS
				1	OK
				2	OK
				3	CD



Plant Name Abbrev.: CLNT


LOOP
EAC
REC0.5
HPCS
RCIC
REC4
RHRSPC
#
STATUS
1 T
OK-TRANS-W/LOOP
2 T
OK-TRANS-W/OSP
3
OK-TRANS-W/OSP
4
CD
5
OK-TRANS-W/OSP
6
CD
7
CD
8
CD
Plant Name Abbrev.: CLNT


ATWS
OVERP
RPT
SLC
INH
CDSR
#
STATUS
1 OK
2 CD
3 CD
4 CD
5 CD
6 CD
Plant Name Abbrev.: CLNT

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. Illinois Power Company, "Clinton Power Station, Unit 1 – Individual Plant Examination Report," September 23, 1992.