March 8, 2000

Mr. Daniel G. Malone
Acting Director, Licensing
Palisades Plant
27780 Blue Star Memorial Highway
Covert, MI 49043

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

Dear Mr. Malone:

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

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

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

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

As a result of the recently concluded Pilot Plant review effort, the NRC has determined that site-specific risk data is needed in order to provide a repeatable determination of the significance of an issue. Therefore, the NRC has contracted with Brookhaven National Lab (BNL) to develop site-specific worksheets to be used in the SDP review. These enclosed worksheets were developed based on your Individual Plant Examination (IPE) submittals that were requested by Generic Letter 88-20. The NRC plans to use this site-specific information in evaluating the significance of issues identified at your facility when the revised reactor oversight process is implemented industry wide. It is recognized that the IPE utilized during this effort may not contain current information. Therefore, the NRC or its contractor will conduct a site visit 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 301-415-1320.

Sincerely,

/RA/

Darl S. Hood, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-255

Enclosure: As Stated

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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 301-415-3049

Sincerely,

/RA/

Darl S. Hood, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-255

Enclosure: As Stated

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RISK-INFORMED INSPECTION NOTEBOOK FOR

PALISADES NUCLEAR PLANT

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

Prepared by

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Prepared for

U. S. Nuclear Regulatory Commission
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NOTICE

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

Mr. Jose G. Ibarra
U. S. Nuclear Regulatory Commission
RES/DSARE/REAHFB
TWFN T10 E46
11545 Rockville Pike
Rockville, MD 20852

ABSTRACT

This notebook contains summary information to support the Significance Determination Process (SDP) in risk-informed inspections for the Palisades Nuclear Plant.

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

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

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1. INFORMATION SUPPORTING SIGNIFICANCE DETERMINATION PROCESS (SDP)

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

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

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

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

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

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

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

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

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

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

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

The three sections that follow include the initiators and dependency table, SDP worksheets, and the SDP event-trees for the Palisades Nuclear Plant.

1.1 INITIATORS AND SYSTEM DEPENDENCY

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

Table 1 (Continued)

Table 1 Initiators and System Dependency for Palisades Nuclear Power Plant[1]

Affected Systems	Major Components	Support Systems	Initia
Accumulators	4 Accumulators	No Primary Support System	LLOCA
AFW	2 MDPs	2.4 KV AC, 480V AC, IA[2], SIS, HVAC	Transient, SLOC ATWS
	1 TDP	DC, SIS, IA[1]	
CCW	3 Pumps in two trains with two HXs	2.4 KV AC, DC, SIS, SW, HVAC	Transient, SLOC LOOP, SGTR, A
Condensate / MFW	2 Condensate pumps 2 MFW pumps	IA, 4.16KV, 480VAC (MCC92)	Transient, SLOC
Containment Fan Coolers	4 Coolers (3 in service)	480VAC(MCC1), DC, SW	LLOCA
Containment Spray System	3MDPs through 2 Spray header and 2 HXs	2.4KVAC, DC, HVAC, CCW/SW	LLOCA, Transien LOOP
CHRG	3 Positive Displacement Charging pump train	480VAC(MCC1&2), DC, CCW,	Transient, SLOC SGTR, ATWS
DC Power System	Buses, and batteries	MCC1 for Battery chargers #1 & 4, and MCC2 for Battery chargers #2, &3	Transient, SLOC LOOP, SGTR, A
EDG	2 dedicated EDGs	DC, HVAC, SW	LOOP
ESS HVAC	Compressors and coolers	480VAC (MCC2 and MCC1), and SW	Transient, SLOC LOOP, SGTR, A
Instrument Air	3 Air compressors	480VAC, DC	Transient, SLOC ATWS
Main Steam	2 ADV and Turbine Bypass Valve per Steam Generator	DC, IA, 480VAC (Loss of Y10 will fail all four ADVs)	Transient, SLOC ATWS
PORV	2 PORVs with associated block valves normally closed	DC for PORVs and 480VAC for Block valves	Transient, SLOC ATWS
RCP	Seals (Not a major Concern for BJ pumps)	1 / 3 CCW pumps for seal cooling	LOOP, RCP sea
RHR/LHSI	2 RHR/LPSI pumps and heat exchangers	2.4 KVAC, DC, SIS, CCW, HVAC, and IA (for RHR only)	Transient, SLOC LOOP, SGTR
CCW	3 MDP through 2 HXs(1/3 pumps but 2/2 HXs are required)	2.4KVAC, SWS, DC, IA	

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Table 1 (Continued)

Affected Systems	Major Components	Support Systems	Initia
SW	3 Pumps	2.4KVAC, DC, SIS (Partial Dependency)	Transient, SLOC LOOP, SGTR, A

Notes:

1. Plant internal event CDF = 5.07 E-5/yr, and CDF for internal Flood 3.0E-7.
2. Loss of IA fails P-8C flow control valves, but P-8A&B flow control valves have N2 back up.
3. IPE does not consider RCP seal LOCA per test performed in AECL for Byron Jackson pumps of same design.

1.2 SDP WORKSHEETS

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

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

Table 2.1 SDP Worksheet for Palisades Nuclear Power Plant —— Transients

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>:**
Power Conversion System (PCS)	1/2 condensate pump with 1/2 ADVs and TBVs to decrease secondary si 500 psi (Operator Action)
Secondary Heat Removal (AFW)	1/2 MDPs (1 multi-train system) or 1/1 TDP (1 ASD Train)
Early Inv., High Pressure Injection (EIHP)	1/2 HPSI pump in injection mode for FB (1 multi-train system)
Primary Heat Removal, Feed/Bleed (FB)	1/2 PORV manually open (operator action)[(1)]
Containment Heat removal System (CHR)	1/3 MDP for containment spray through 1/2 SDC HX (1 multi-train system (1 multi-train system)
High Pressure Recirculation (HPR)	1/2 HPSI pump taking suction from the containment sump either directly Spray or LPSI Pump (1 multi-train system)
Make Up to CST (MUCST)	Make up water to CST from (1) demineralized water storage tank,(2)prim tank,(3)Service water, and (4) fire protection water (operator action)[(2)]

<u>Circle Affected Functions</u>	**<u>Recovery of Failed Train</u>**	***<u>Remaining Mitigation Capability Rating for Each Affe Sequence</u>***
1 TRANS - PCS - AFW - FB (17)		
2 TRANS - PCS - AFW - EIHP (16)		
3 TRANS - PCS - AFW - HPR (15)		
4 TRANS - PCS - AFW - CHR (14)		
5 TRANS - MUCST - FB (6,12)		
6 TRANS - MUCST - EIHP (5,11)		
7 TRANS - MUCST - HPR (4,10)		
8 TRANS - MUCST - CHR (3,9)		

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
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) Even though the original IPE did not consider the human error probability for FB, however as a response to NRC's question (date July 22,1994, page 104) they assigned a probability of 8.1E-5. Regardless of HEP value for Palisades, the SDP standard operator action is assigned.

(2) The MUCST is needed in those sequences where the operator does not do rapid depressurization to align SDC system for decay heat removal.

Table 2.2 SDP Worksheet for Palisades Nuclear Power Plant —— Small LOCA (<2")

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

<u>Safety Functions Needed</u>:	***Full*** **<u>Creditable Mitigation Capability for Each Safety Function</u>:**
Power Conversion System (PCS)	1/2 condensate pump with 1/2 ADVs and TBVs to decrease secondary sic psi (Operator Action)
Secondary Heat Removal (AFW)	1/2 MDPs (1 multi-train system) or 1/1 TDP (1 ASD Train)
Early Inventory, High Pressure Injection (EIHP)	1/2 HPSI pump in injection mode for FB (1 multi-train system)
Primary Heat Removal, Feed/Bleed (FB)	1/2 PORV manually open (operator action)[1]
Containment Heat removal System (CHR)	1/3 MDP for containment spray through 1/2 SDC HX (1 multi-train system) multi-train system)
High Pressure Recirculation (HPR)	1/2 HPSI pump taking suction from the containment sump either directly o Spray or LPSI Pump (1 multi-train system)
Make Up to CST (MUCST)	Make up water to CST from (1) demineralized water storage tank,(2)prima tank,(3)Service water, and (4) fire protection water (operator action)[2]

<u>Circle Affected Functions</u>	**<u>Recovery of Failed Train</u>**	***<u>Remaining Mitigation Capability Rating for Each Affec Sequence</u>***
1 SLOCA - HPR (2,6,10,14,19)		
2 SLOCA - EIHP (3,7,11,15,20)		
3 SLOCA - MUCST - FB (8,16)		
4 SLOCA - MUCST - HPR (5,13)		
4 SLOCA - PCS - AFW - CHR (18)		
5 SLOCA - PCS - AFW - FB(21)		

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) Even though the original IPE did not consider the human error probability for FB, however as a response to NRC's question (date July 22,1994, page 104) they assigned a probability of 8.1E-5. Regardless of HEP value for Palisades, the SDP standard operator action is assigned.

(2) The MUCST is needed in those sequences where the operator does not do rapid depressurization to align SDC system for decay heat removal.

Table 2.3 SDP Worksheet for Palisades Nuclear Power Plant —— Stuck Open PORV (SORV)[(1)]

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

Safety Functions Needed:	*Full* Creditable Mitigation Capability for Each Safety Function:
Isolation of Small LOCA (BLK)	Closure of Block valves if open. Note that the plant operate with normall (operator action)[(1)]
Early Inv., High Pressure Injection (EIHP)	1/2 HPSI pump in injection mode for FB (1 multi-train system)
Containment Heat removal System (CHR)	1/3 MDP for containment spray through 1/2 SDC HX (1 multi-train systen (1 multi-train system)
High Pressure Recirculation (HPR)	1/2 HPSI pump taking suction from the containment sump either directly Spray or LPSI Pump (1 multi-train system)

Circle Affected Functions	Recovery of Failed Train	*Remaining Mitigation Capability Rating for Each Affe Sequence*
1 SORV - BLK - HPR (3)		
2 SORV - BLK - EIHP (4)		
4 SORV - BLK - CHR (2)		

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

If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following c 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.

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Notes:

(1) This SDP worksheet may not be applicable since the block valves are normally closed. Stuck open PORV will act as a medium LOCA. This SDP sheet corresponds to the MLOCA event tree.

Table 2.4 SDP Worksheet for Palisades Nuclear Power Plant —— Medium LOCA (>2" and <18")

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

Safety Functions Needed:	***Full* Creditable Mitigation Capability for Each Safety Function:**
Early Inventory, HP Injection (EIHP)	1/2 HPSI Pump from SIRWST in Injection mode (1 multi-train system)(1)
Containment Heat Removal (CHR)	1/3 MDP for containment spray through 1/2 SDC HX (1 multi-train system) or 1 multi-train system)
High Pressure Recirculation (HPR)	1/2 HPSI pump taking suction from the containment sump either directly or thr LPSI Pump (1 multi-train system)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***
1 MLOCA - CHR (2)		
2 MLOCA - HPR (3)		
3 MLOCA - EIHP (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 t conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Note:

(1) The availability of secondary cooling will allow the use of 1/2 LPSI for injection however since HPSI is still needed for recirculation this success path is not credited.

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Table 2.5 SDP Worksheet for Palisades —— Large LOCA (>18")

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

Safety Functions Needed:	***Full* Creditable Mitigation Capability for Each Safety Function:**
Early Inventory Control (EIAC1)	2/4 SIT (1 multi-train system)
Early Inventory Alternative (EIAC2)	3/4 SIT (1 train)
Early Inventory, LP Injection (EILP)	1/2 LPSI pump from SIRWST in injection mode (1 multi-train system)
High Pressure Recirculation (HPR)	1/2 HPSI pump taking suction from the containment sump either directly o Spray or LPSI Pump (1 multi-train system)
Containment Press/Temp Control (CHR)	1/3 MDP for containment spray through 1/2 SDC HX (1 multi-train system) multi-train system)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affec Sequence***
1 LLOCA - CHR (2,6)		
2 LLOCA - HPR (3,7)		
3 LLOCA - EILP - EIHP (8)		
4 LLOCA - EIAC1 (4)		
5. LLOCA - EILP - EIAC2 (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 c 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.

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Table 2.6 SDP Worksheet for Palisades —— 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 AC Power (EAC)	1/2 EDG (1 multi-train system)
Turbine-driven AFW pump (TDAFW)	1/1 TDAFW (1 ASD Train)
Secondary Heat Removal (AFW)	1/2 MDPs (1 multi-train system) and 1/1 TDP (1 ASD Train)
Recovery of AC Power in < 2 hrs (REC2)	Restoring a source of AC in less than 2 hour (High stress operator/recov
Recovery of AC Power in < 8 hrs (REC8)[(1)]	Restoring a source of AC in less than 8 hour (operator action)
Make up to CST (MUCST)	Make up water to CST from (1) demineralized water storage tank,(2)prim tank,(3)Service water, and (4) fire protection water (operator action)
Early Inventory, HP Injection (EIHP)	1/2 HPSI pump in injection mode for FB (1 multi-train system)
Primary Heat Removal (FB)	1/2 PORV manually open (operator action)
Containment Heat Removal (CHR)	1/3 MDP for containment spray through 1/2 SDC HX (1 multi-train syster (1 multi-train system)
High Pressure Recirculation (HPR)	1/2 HPSI pump taking suction from the containment sump either directly Spray or LPSI Pump (1 multi-train system)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affe Sequence***
1 LOOP - AFW - FB (11)		
2 LOOP - MUCST - FB (6,17)		
3 LOOP - AFW - EIHP (10)		
4 LOOP - MUCST - EIHP (5,16)		
5 LOOP - AFW - HPR (9)		
6 LOOP - MUCST-HPR (4,15)		
7 LOOP - AFW - CHR (8)		

8 LOOP - MUCST - CHR (3,14)		
9 LOOP - EAC - TDAFW - FB (23) (AC restored in less than 2 hours)		
10 LOOP - EAC - TDAFW - EIHP (22) (AC restored in less than 2 hours)		
11 LOOP - EAC - TDAFW - HPR (21) (AC restored in less than 2 hours)		
12 LOOP - EAC - TDAFW - CHR (20) (AC restored in less than 2 hours)		
13 LOOP - EAC - TDAFW - REC2 (24)		
14 LOOP - EAC - REC8 (18)		

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) The emergency batteries would deplete after six hours at Palisades in SBO scenarios. Once battery depletes it is assumed that flow and level control of secondary cooling will be lost and core could not be recovered if a source of AC is not re-stored in eight hours.

Table 2.7 SDP Worksheet for Palisades Nuclear Power Plant —— SGTR

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

Safety Functions Needed:	***Full* Creditable Mitigation Capability for Each Safety Function:**
Secondary Heat Removal (AFW)	1/2 MDPs (1 multi-train system) and 1/1 TDP (1 ASD Train)
Early Inventory, HP Injection (EIHP)	1/2 HPSI pump in injection mode for FB (1 multi-train system)
Pressure Equalization (EQ)	Isolation of AFW to faulted SG, use of pressurizer spray to reduce primary pres of the SG safeties (high stress operator action)(1)
Feed-and-Bleed (FB)	1/2 PORV manually open (operator action)
High Pressure Recirculation (HPR)	1/2 HPSI pump taking suction from the containment sump either directly or thro LPSI Pump (1 multi-train system)
Containment Heat Removal (CHR)	1/3 MDP for containment spray through 1/2 SDC HX (1 multi-train system) or 1/ train system)
Shut down cooling (SDC)(2)	1/2 LPSI in RHR mode with CCW connected (operator action)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***
1 SGTR - EIHP (2,7,11)		
2 SGTR - EQ - MUCST - SDC (5)		
3 SGTR - EQ - MUCST - FB (6)		
4 SGTR - AFW - FB (10)		
5 SGTR - AFW - CHR (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 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) The human error associated with failure to equalize primary and secondary pressure is modeled in IPE under three separate events (SCF,IPC,SRV). The value of HEP can not therefore be easily obtained. One of the actions under IPC is about 3.6E-2. It is therefore assumed that High stress operator action is appropriate for this case.

(2) In the case of SGTR the operator will attempt to do rapid cooling to enter SDC operation prior to emptying the SIRWST. Therefore make up to SIRWST is not questioned in the event tree. However MUCST could still be needed for rapid cool down therefore it is questioned in the SDP event tree.

Table 2.8 SDP Worksheet for Palisades Nuclear Power Plant —— ATWS

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 Boration (CVCS)	1/3 Charging pump to inject borated water to vessel (operator action)
Turbine trip (TTP)	Automatic or manual trip (1 train system)(1)
Primary Relief (SRV)	3/3 SRVs to open to protect against early over-pressurization (1 train system)(2)
Primary Integrity (SRVR)	all 3/3 SRVs should re-close (1 train system)
Make up to CST (MUCST)	Make up water to CST from (1) demineralized water storage tank,(2)primary syster tank,(3)Service water, and (4) fire protection water (operator action)
Secondary Heat Removal (AFW)	1/2 MDAFW pump (1 train system), or 1/1TDAFW pump (1 ASD train)(3)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Seq***
1 ATWS - TTP (7)		
2 ATWS - SRV (6)		
3 ATWS - SRVR (5)		
4 ATWS - CVCS (4)		
5 ATWS - AFW (3)		
6 ATWS - MUCST(2)		

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

If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following c 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) In those cases where failure to scram is due to actuation and electrical fault, the automatic turbine trip function could be unavailable. Due to lack of detail, the function of turbine trip is treated as a one train system.

(2) Since the plant operates with block valves closed, PORVs can not be credited for over-pressure protection.

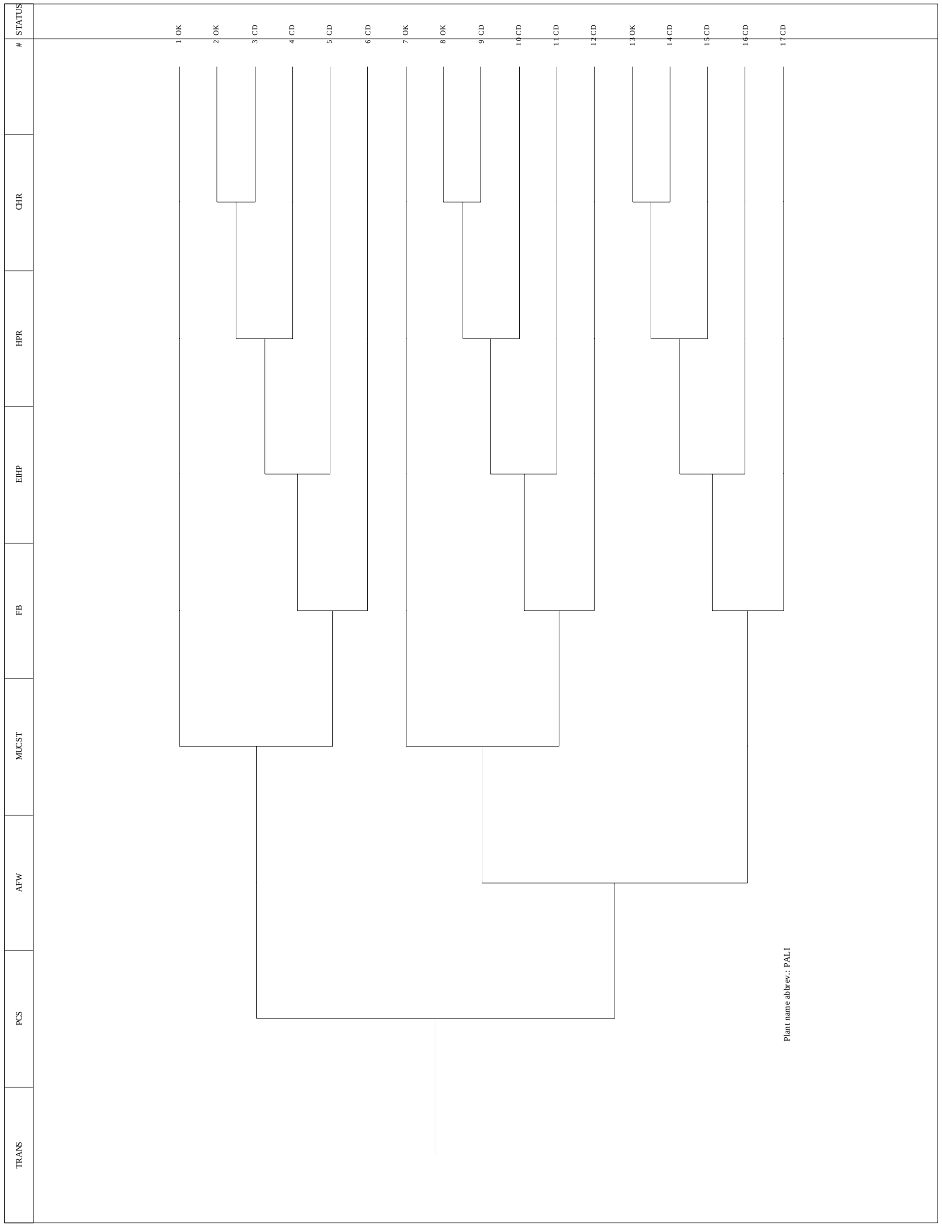
(3) The AFW flow should be manually increased above 165 gpm early in the scenario, therefore this function could have been assigned to operator action.

1.3 SDP Event Trees

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

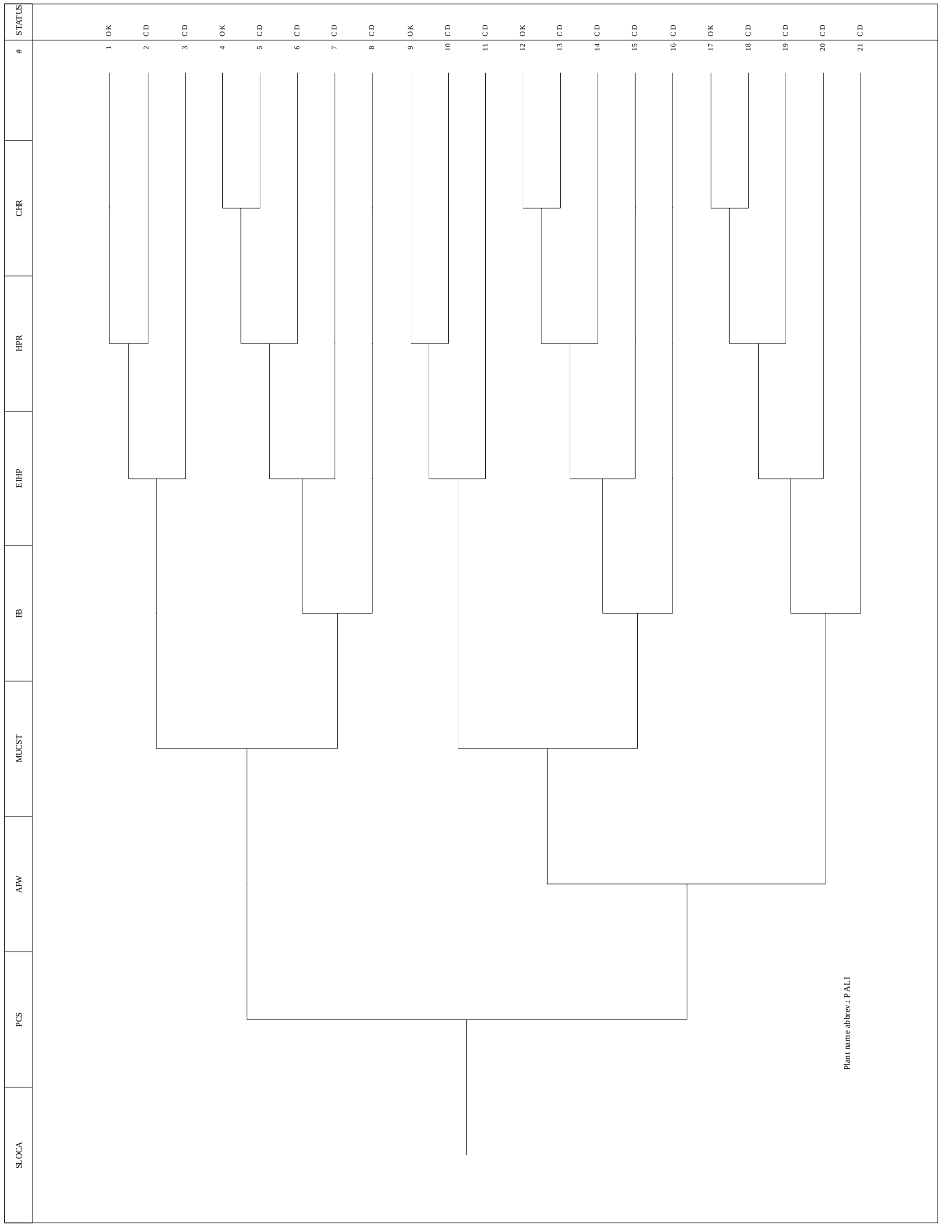
The following event trees are included:

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

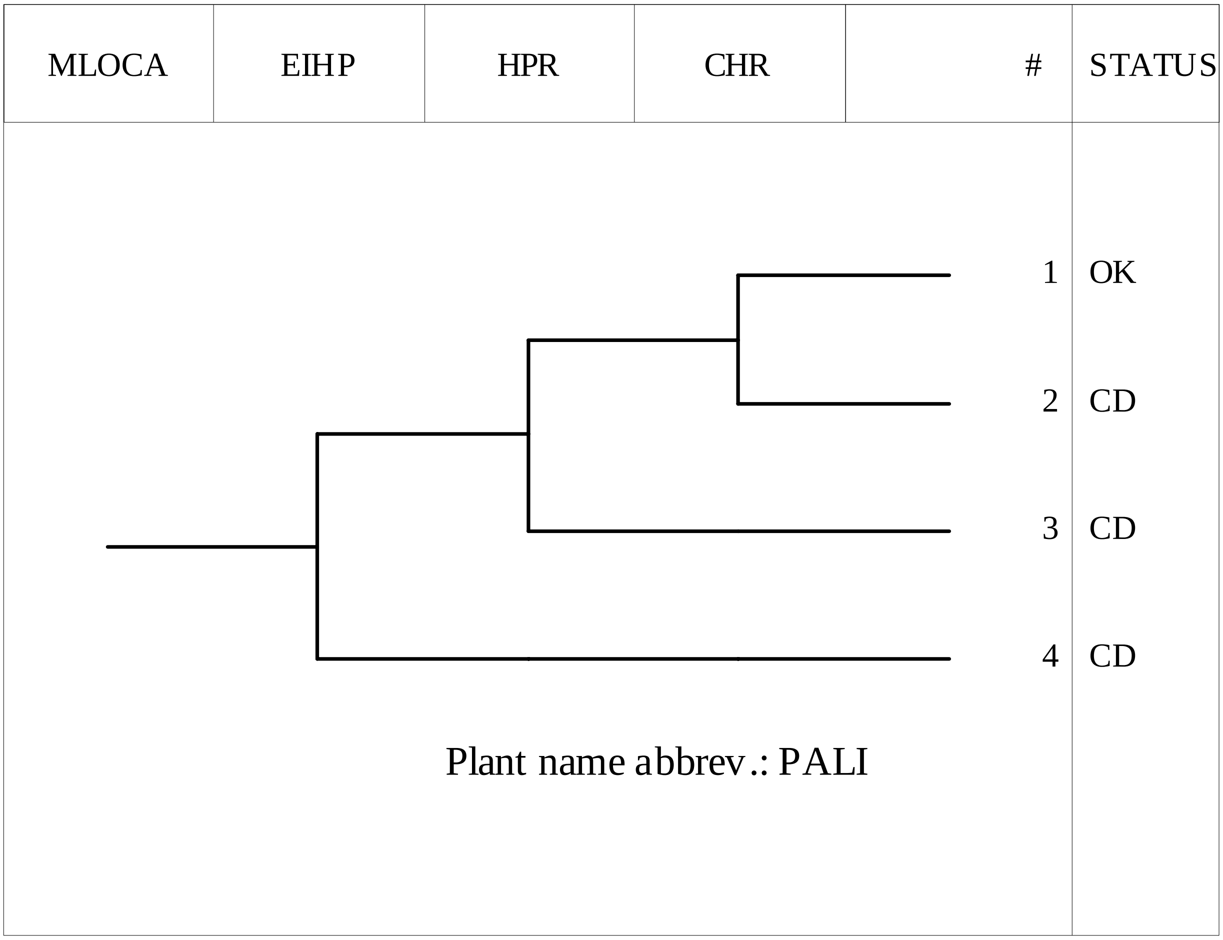


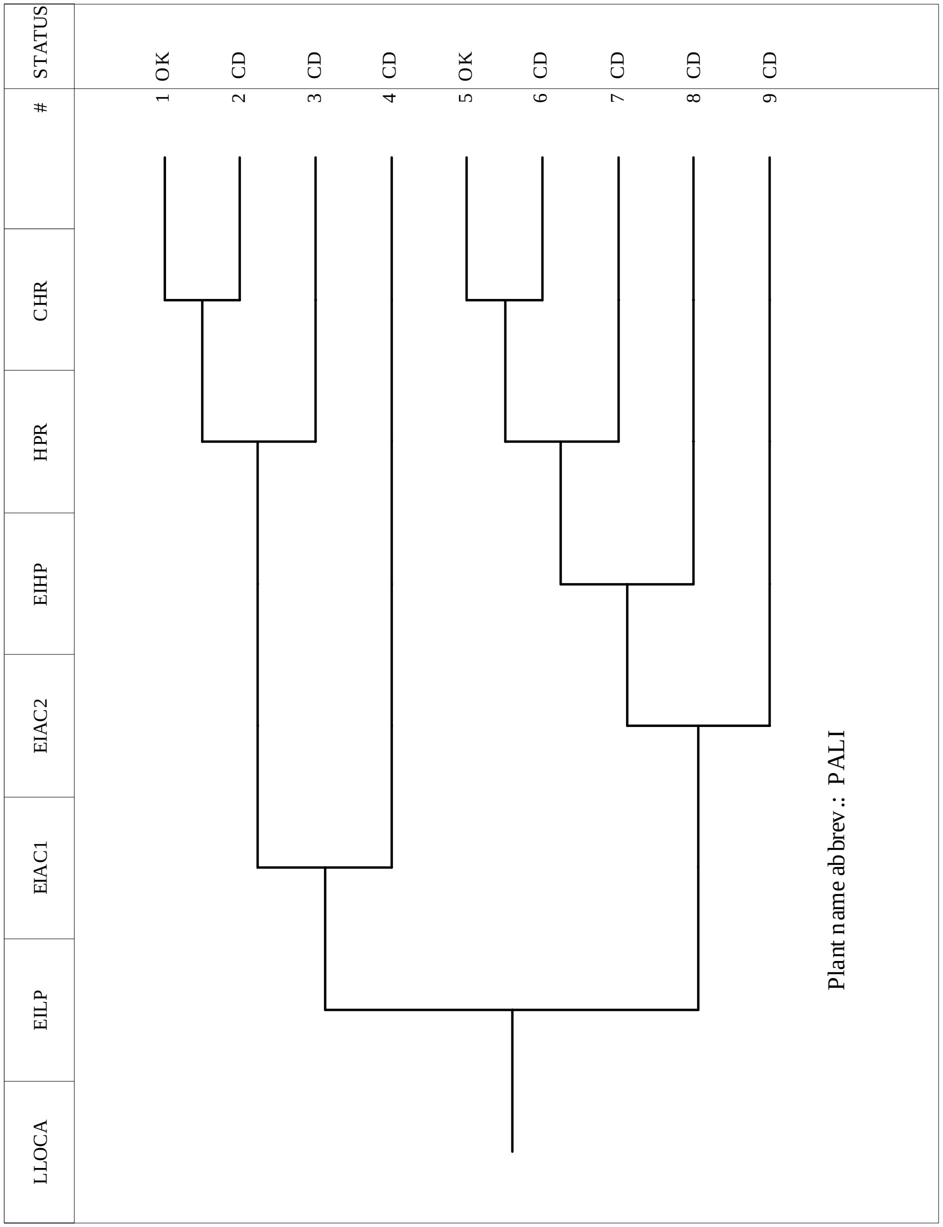
STATUS
#
CHR
HPR
EBHP
FB
MUCST
AFW
PCS
TRANS
1 OK
2 OK
3 CD
4 CD
5 CD
6 CD
7 OK
8 OK
9 CD
10 CD
11 CD
12 CD
13 OK
14 CD
15 CD
16 CD
17 CD
Plant name abbrev.: PALI

- 21 -

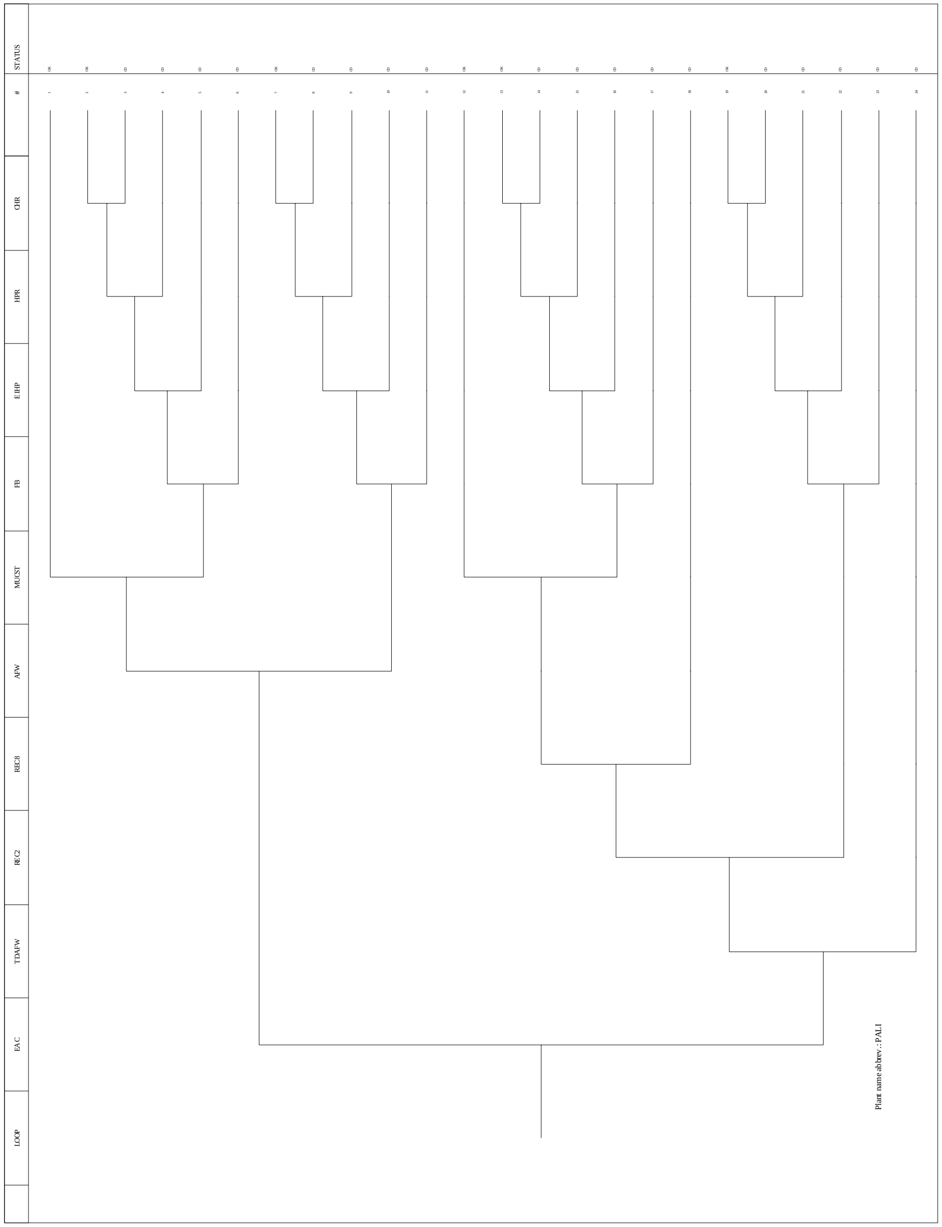

STATUS
#
CHR
HPR
EHP
FB
MUCST
AFW
PCS
SLOCA
1 OK
2 CD
3 CD
4 OK
5 CD
6 CD
7 CD
8 CD
9 OK
10 CD
11 CD
12 OK
13 CD
14 CD
15 CD
16 CD
17 OK
18 CD
19 CD
20 CD
21 CD
Plant name abbrev.: PALI

- 22 -


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

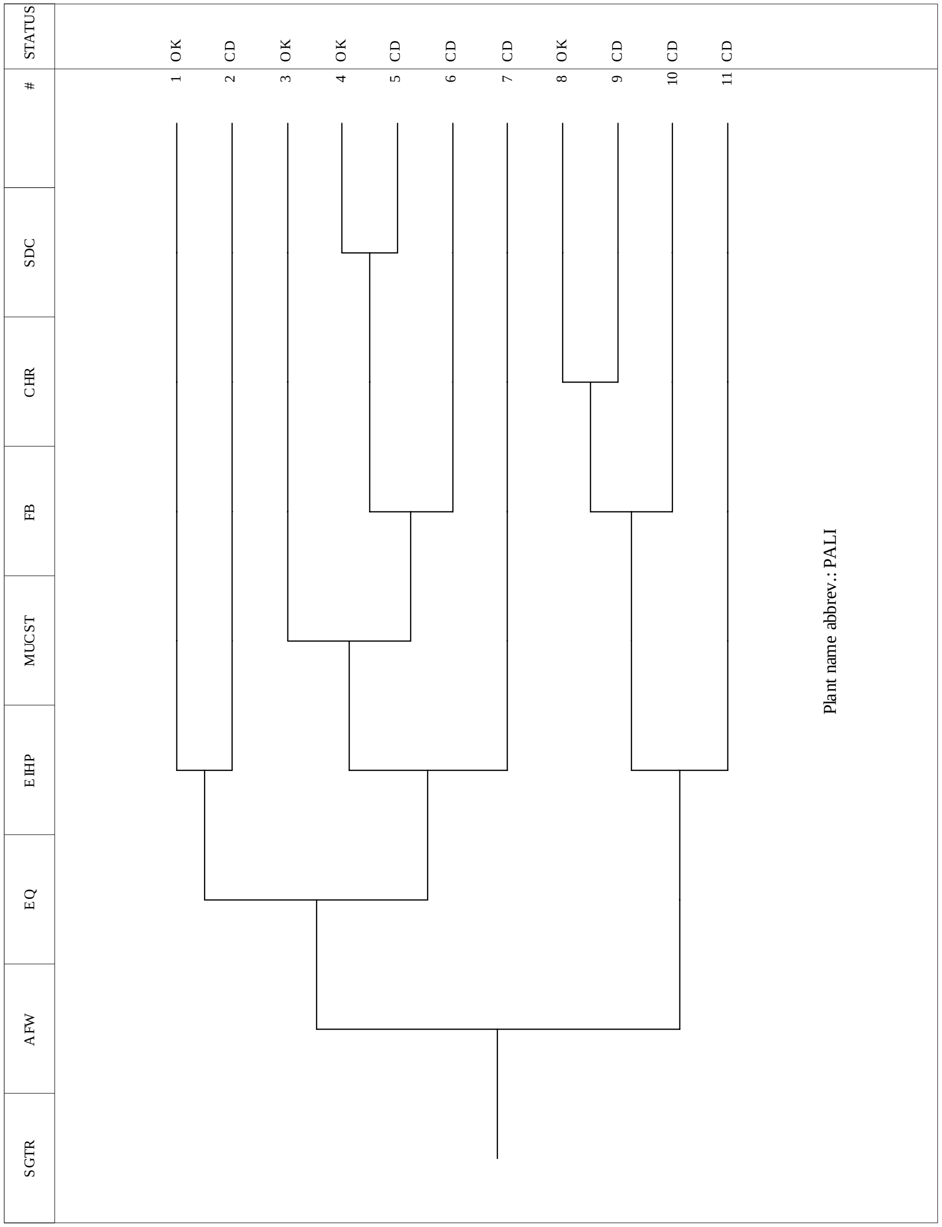

STATUS
#
CHR
HPR
EIHP
EIAC2
EIAC1
EILP
LLOCA
1 OK
2 CD
3 CD
4 CD
5 OK
6 CD
7 CD
8 CD
9 CD
Plant name abbrev.: PALI

- 24 -

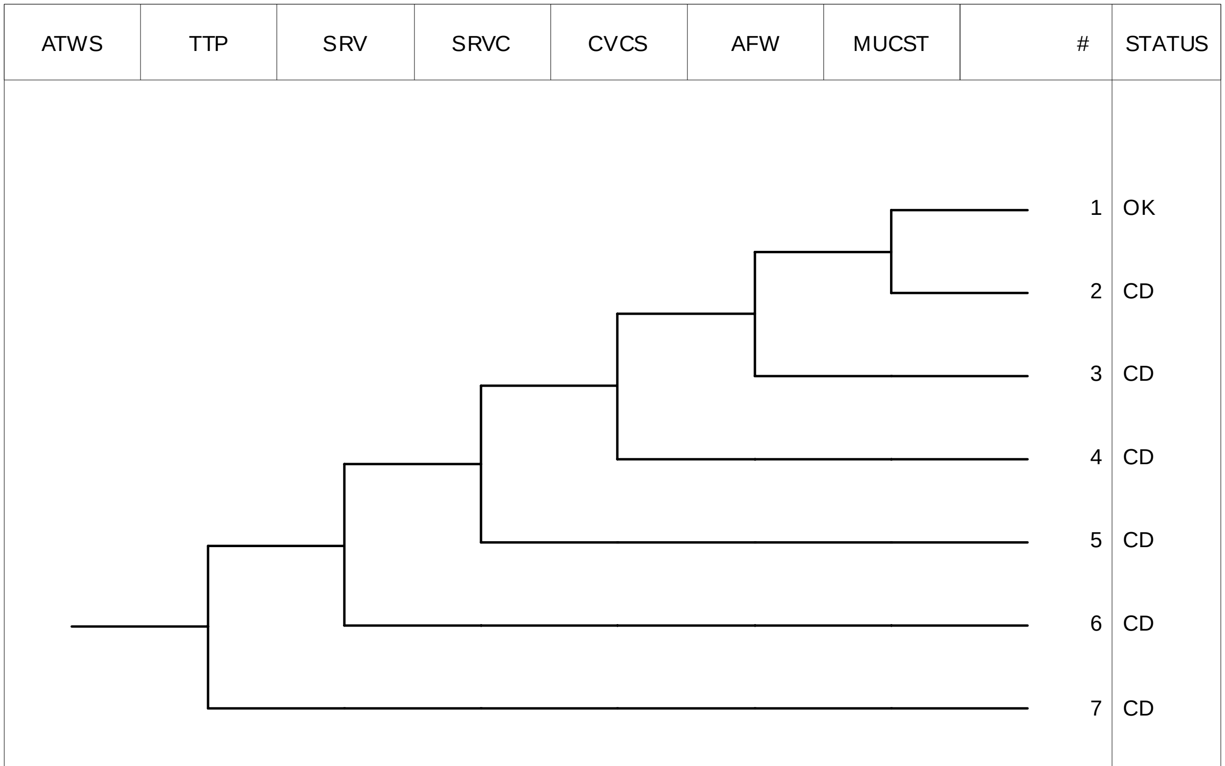


STATUS
#
CHR
HPR
EHP
FB
MUCST
APW
REC8
REC2
TDAFW
EAC
LOOP
Plant name abbrev.: PALI

- 25 -


STATUS
#
SDC
CHR
FB
MUCST
EIHP
EQ
AFW
SGTR
1 OK
2 CD
3 OK
4 OK
5 CD
6 CD
7 CD
8 OK
9 CD
10 CD
11 CD
Plant name abbrev.: PALI

- 26 -



Plant name abbrev.: PALI

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. Consumers Power Company, "Palisades Nuclear Plant Individual Plant Examination Report," dated January 1993.