March 14, 2000

Mr. Guy G. Campbell, Vice President - Nuclear
FirstEnergy Nuclear Operating Company
5501 North State Route 2
Oak Harbor, OH 43449-9760

SUBJECT: DAVIS-BESSE NUCLEAR POWER STATION - SITE-SPECIFIC WORKSHEETS FOR USE IN THE NUCLEAR REGULATORY COMMISSION'S SIGNIFICANCE DETERMINATION PROCESS

Dear Mr. Campbell:

The purpose of this letter is to provide you with one of the key implementation tools to be used by the Nuclear Regulatory Commission (NRC) in the revised reactor oversight process, which is currently expected to be implemented at the Davis-Besse Nuclear Power Station 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 301-415-1364.

Sincerely,

/RA/

Douglas V. Pickett, Senior Project Manager, Section 2
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-346

Enclosures: 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 301-415-1364.

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Douglas V. Pickett, Senior Project Manager, Section 2
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Mary E. O'Reilly
FirstEnergy
76 South Main Street
Akron, OH 44308

James L. Freels
Manager - Regulatory Affairs
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
5501 North State - Route 2
Oak Harbor, OH 43449-9760

Jay E. Silberg, Esq.
Shaw, Pittman, Potts
and Trowbridge
2300 N Street, NW.
Washington, DC 20037

Regional Administrator
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, IL 60523-4351

Michael A. Schoppman
Framatome Technologies Incorporated
1700 Rockville Pike, Suite 525
Rockville, MD 20852

Resident Inspector
U.S. Nuclear Regulatory Commission
5503 North State Route 2
Oak Harbor, OH 43449-9760

James H. Lash, Plant Manager
FirstEnergy Nuclear Operating Company
Davis-Besse Nuclear Power Station
5501 North State Route 2
Oak Harbor, OH 43449-9760

Harvey B. Brugger, Supervisor
Radiological Assistance Section
Bureau of Radiation Protection
Ohio Department of Health
P.O. Box 118
Columbus, OH 43266-0118

James R. Williams, Executive Director
Ohio Emergency Management Agency
2855 West Dublin Granville Road
Columbus, OH 43235-2206

Director
Ohio Department of Commerce
Division of Industrial Compliance
Bureau of Operations & Maintenance
6606 Tussing Road
P.O. Box 4009
Reynoldsburg, OH 43068-9009

Ohio Environmental Protection Agency
DERR--Compliance Unit
ATTN: Zack A. Clayton
P.O. Box 1049
Columbus, OH 43266-0149

State of Ohio
Public Utilities Commission
180 East Broad Street
Columbus, OH 43266-0573

Attorney General
Department of Attorney
30 East Broad Street
Columbus, OH 43216

President, Board of County
Commissioners of Ottawa County
Port Clinton, OH 43252

I RISK-INFORMED INSPECTION NOTEBOOK FOR

DAVIS-BESSE NUCLEAR POWER STATION

UNIT 1

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

Prepared by

Brookhaven National Laboratory
Department of Advanced Technology

Contributors

M. A. Azarm
J. Carbonaro
T. L. Chu
A. Fresco
J. Higgins
G. Martinez-Guridi
P. K. Samanta

NRC Technical Review Team

John Flack	**RES**
Morris Branch	**NRR**
Doug Coe	**NRR**
Gareth Parry	**NRR**
Peter Wilson	**NRR**
Jim Trapp	**Region I**
Michael Parker	**Region III**
William B. Jones	**Region IV**

Prepared for

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

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 Davis-Besse Nuclear 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.

CONTENTS

FIGURES

Page

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 Davis-Besse Nuclear 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 Initiators and System Dependency for Davis-Besse Nuclear Power Station, Unit 1 (1)

Affected Systems	Major Components	Support Systems	Initiating Event Scenarios (2)
AC Power System (AC)	AC Power Distribution and AC Instrument Power	DC	Transient, LPCS, SLOCA, SORV, MLOCA, LLOCA, LOOP, SGTR, ATWS, RCP seal LOCA
DC Power System	Buses, Battery Chargers and Batteries	AC, Room cooling	Transient, LPCS, SLOCA, SORV, MLOCA, LLOCA, LOOP, SGTR, ATWS, RCP seal LOCA
AFW	2 AFWTDPs, 1 AFWMDP	DC, AC, main steam, Steam Feedwater Rupture Control System (SFRCS)	Transient, LPCS, SLOCA, SORV, LOOP, SGTR, ATWS, RCP seal LOCA
HPI	2 Pumps each with 500 gpm at 1300 psig	AC, DC, SFAS, room cooling	Transient, LPCS, SLOCA, SORV, MLOCA, LOOP, SGTR, ATWS, RCP seal LOCA
LPI/DHR	2 Pumps and 2 Heat Exchangers	AC, DC, CCW, SFAS, IA, room cooling	Transient, LPCS, SLOCA, SORV, MLOCA, LLOCA, LOOP, SGTR, ATWS, RCP seal LOCA
Make Up Pumps	2 Pumps each with 150 gpm at 2500 psig	AC, DC, CCW, IA, room cooling	Transient, LPCS, SLOCA, SORV, LOOP, SGTR, ATWS, RCP seal LOCA
CS	2 pumps	AC, DC, SFAS, room cooling	Not needed
EDG	2 EDGs and 1 SBODG	AC, DC, CCW, room cooling	LOOP
Containment Air Coolers (CACs)	3 coolers with cooling coils, fans, and dampers	AC, SW, SFAS	Not needed

Table 1 (Continued)

Affected Systems	Major Components	Support Systems	Initiating Event Scenarios [2]
CCW	3 pumps in 2 loops	AC, DC, SW, SFAS	Transient, LPCS, SLOCA, SORV, MLOCA, LLOCA, LOOP, SGTR, ATWS, RCP seal LOCA
Service Water System (SW)	3 pumps in 2 trains	AC, DC, SFAS, room cooling	Transient, LPCS, SLOCA, SORV, MLOCA, LLOCA, LOOP, SGTR, ATWS, RCP seal LOCA
CFT (Core Flood Tanks)	2 Passive tank trains	NA	Not used.
Instrument Air /Sation Air (IA)	2 station air compressors and 1 emergency instrument air compressor	AC, DC, TPCW	Transient, LPCS, SLOCA, SORV, MLOCA, LLOCA, LOOP, SGTR, ATWS, RCP seal LOCA
Power Conversion System	2 TDMFW pumps and 3 condensate pumps	Offsite power, IA, TPCW, CW	Transient, SGTR, ATWS
AVVs (SG PORV), TBVs (turbine bypass valves), MSIVs	Per steam line: 1 AVV, 3 TBVs, and 1 MSIVs	AC, DC, IA	Transient, LPCS, SLOCA, SORV, MLOCA, LLOCA, LOOP, SGTR, ATWS, RCP seal LOCA
MSSV (main steam safety valves)	Per steam line: 9 MSSVs	None	Not needed
PORV	1 PORV with block valve	DC, AC (block valve)	Transient, LPCS, LOOP, ATWS
PSV	2 PSVs	None	ATWS
RCP	Seals	CCW for thermal barrier cooling, makeup pumps for seal injection	RCP seal LOCA
Room Cooling	Fans, cooling coils	SW, AC	Transient, LPCS, SLOCA, SORV, MLOCA, LLOCA, LOOP, SGTR, ATWS, RCP seal LOCA

Table 1 (Continued)

Affected Systems	Major Components	Support Systems	Initiating Event Scenarios [2]
Turbine Plant Cooling Water (TPCW)	No information	AC, DC, SW	Transient, LPCS, SLOCA, SORV, MLOCA, LLOCA, LOOP, SGTR, ATWS, RCP seal LOCA

Table 1 (Continued)

Notes:

1. Plant CDF is 6.6E-5 per year based on IPE submittal Dated 2/26/93.

2. The special initiating events have not been included in this table.

1.2 SDP WORKSHEETS

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

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

Table 2.1 SDP Worksheet for Davis-Besse Nuclear Power Station, Unit 1 —— Transients (Reactor Trip) (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 with 1 / 3 condensate trains (1 multi-train system)		
Secondary Heat Removal (AFW)	1 / 1 MDAFW trains (operator action) (2) or 1/2 TDAFW train (1 multi-train system)		
High Pressure Injection (EIHP)	1 / 2 Make up pump trains (1 multi-train system)		
Primary Heat Removal, Feed/Bleed (FB)	1 / 1 PORV (operator action) (3)		
High Pressure Recirculation (HPR)	1 / 2 makeup trains or 1 / 2 HPI trains taking suction from 1/2 LPI trains through LPI HX(4) (operator action)		
Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 TRANS - PCS - AFW - FB (6)			
2 TRANS - PCS - AFW -EIHP (5)			
3 TRANS - PCS - AFW - HPR (4)			

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

<u>Notes</u>:

(1) Davis-Besse IPE models stock open PORV or PSV and RCP seal LOCAs in the transient event tree.

(2) The HEP for operator failure to initiate MDAFW is 2.4E-3 (event BHEMDFPE on page 272)

(3) The human error for initiation of HPI cooling is 1.6E-2 (event UHAMUHPE on page 273). (The IPE also credits HPI cooling with 2 / 3 PORV and PSVs and 2/2 makeup pumps.)

(4) The HEP for operator failure to establish HPR is 4.8E-4. (Event XHAHPRTE on page 274.)

Table 2.2 SDP Worksheet for Davis-Besse Nuclear Power Station, Unit 1 —— Loss of PCS (LPCS)

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 / 1 MDAFW trains (operator action) (1) or 1/2 TDAFW train (1multi-train system)
High Pressure Injection (EIHP)	1 / 2 Make up pump trains (1 multi-train system)
Primary Heat Removal, Feed/Bleed (FB)	1 / 1 PORV (operator action) (2)
High Pressure Recirculation (HPR)	1 / 2 makeup trains or 1 / 2 HPI trains taking suction from 1/2 LPI trains through LPI HX(3) (operator action)

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

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

Notes:

(1) The HEP for operator failure to initiate MDAFW is 2.4E-3 (event BHEMDFPE on page 272)

(2) The human error for initiation of HPI cooling is 1.6E-2 (event UHAMUHPE on page 273). (The IPE also credits HPI cooling with 2 /3 PORV and PSVs and 2/2 makeup pumps.)

(3) The HEP for operator failure to establish HPR is 4.8E-4. (Event XHAHPRTE on page 274.)

Table 2.3 SDP Worksheet for Davis-Besse Nuclear Power Station, Unit 1 —— 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:**
Secondary Heat Removal (AFW)	1 / 1 MDAFW trains (operator action) (1) or 1/2 TDAFW train (1multi-train system)
Primary Heat Removal, Feed/Bleed (FB)	(operator action) (2)
High Pressure Injection (EIHP)	AFW successful: 1 / 2 HPI or 2/2 makeup (1 multi train system) AFW failed: 1 / 2 makeup pump trains (1 multi-train system)
High Pressure Recirculation (HPR)	1 / 2 makeup trains or 1 / 2 HPI trains taking suction from 1/2 LPI trains through LPI HX(3) (operator action)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 SLOCA - EIHP (3,6)			
2 SLOCA - HPR (2,5)			
3 SLOCA - AFW - FB (7)			

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

Notes:

(1) The HEP for operator failure to initiate MDAFW is 2.4E-3 (event BHEMDFPE on page 272)

(2) The human error for initiation of HPI cooling is 1.6E-2 (event UHAMUHPE on page 273). (The IPE also credits HPI cooling with 2 /3 PORV and PSVs and 2/2 makeup pumps.)

(3) The HEP for operator failure to establish HPR is 3.5E-3. (Event XHAHPRSE on page 274.) IPE also credits normal decay heat removal if AFW is successful.

Table 2.4 SDP Worksheet for Davis-Besse Nuclear Power Station, Unit 1 —— Stuck Open PORV (SORV)

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:**
Close the Block Valve (BLK)	(Operator action) [1]
Secondary Heat Removal (AFW)	1 / 1 MDAFW trains (operator action) [2] or 1/2 TDAFW train (1multi-train system)
Primary Heat Removal, Feed/Bleed (FB)	(Operator action) [3]
High Pressure Injection (EIHP)	AFW successful: 1 / 2 HPI or 2/2 makeup (1 multi train system) AFW failed: 1 / 2 makeup pump trains (1 multi-train system)
High Pressure Recirculation (HPR)	1 / 2 makeup trains or 1 / 2 HPI trains taking suction from 1/2 LPI trains through LPI HX[4] (operator action)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 SORV - BLK - EIHP (3,6)			
2 SORV - BLK - HPR (2,5)			
3 SORV - BLK - AFW - FB (7)			

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

<u>Notes</u>:

(1) The HEP for operator failure to close the block valve is 3.6E-3 (event RHA011CE on page 273).

(2) The HEP for operator failure to initiate MDAFW is 2.4E-3 (event BHEMDFPE on page 272)

(3) The human error for initiation of HPI cooling is 1.6E-2 (event UHAMUHPE on page 273). (The IPE also credits HPI cooling with 2 /3 PORV and PSVs and 2/2 makeup pumps.)

(4) The HEP for operator failure to establish HPR is 3.5E-3. (Event XHAHPRSE on page 274.) IPE also credits normal decay heat removal if AFW is successful.

Table 2.5 SDP Worksheet for Davis-Besse Nuclear Power Station, Unit 1 —— Medium 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 HPSI trains (1 multi-train systems)
Low Pressure Injection (EILP)	1/2 LPI train (1 multi-train system)
Low Pressure Recirculation (EILR)	1/2 LPI train taking suction from sump (operator action) (1)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 MLOCA - LPR (2)			
2 MLOCA - LPI (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 criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.

Note:

(1) The HEP for operator failure to initiate LPR is 4.4E-3 (event XHALPRME on page 274).

Table 2.6 SDP Worksheet for Davis-Besse Nuclear Power Station, Unit 1 —— Large LOCA

Estimated Frequency (Table 1 Row) _________	**Exposure Time** _________	**Table 1 Result (circle): A B C D E F G H**	
Safety Functions Needed:	***Full* Creditable Mitigation Capability for each Safety Function:**		
Low Pressure Injection (EILP) **Low Pressure Recirculation (EILR)**	1/2 LPI train (1 multi-train system) 1/2 LPI train taking suction from sump (operator action) (1)		
Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 LLOCA - EILP (3)			
2 LLOCA - EILR (2)			

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

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

Note:

(1) The HEP for operator failure to initiate LPR is 7.4E-3 (event XHALPRAE on page 274).

20

Table 2.7 SDP Worksheet for Davis-Besse Nuclear Power Station, Unit 1 —— LOOP [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:
Emergency AC Power (EAC)	1/2 EDGs or SBODG (1 multi-train system)
Turbine-driven EFW Pump (TDAFW)	Operation of 1/2 TDAFW pumps (1 multi-train system)
Recovery of AC Power in < 1 hrs (REC1)	SBO procedure and Recovery of an AC source in one hour (operator action)[2]
Recovery of AC Power in < 3 hrs (REC3)	SBO procedure and Recovery of an AC source in three hours (operator action) [3]
Early Inventory, HP Injection (EIHP)	1/2 makeup trains (1 multi-train system)
Primary Heat Removal (FB)	1/1 PORV and operator initiates HPI cooling(operator action) [4]
High Pressure Recirculation (HPR)	1/2 makeup pumps or 1/2 HPI trains taking suction from 1/2 LPI trains through LPI HX (operator action)[5]

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

4 LOOP - EAC - REC1 - HPR (4)			
5 LOOP - EAC - REC1 - FB (5)			
6 LOOP - EAC - TDAFW - EIHP (11)			
7. LOOP - EAC - TDAFW - FB (10)			
8. LOOP - EAC - TDAFW - HPR (9)			
Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event: If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.			

<u>Notes</u>:

(1) The IPE does not provide much information about LOOP event tree. It does not have a LOOP event tree. No discussion on battery capacity in a SBO is available. No recovery of offsite power model is discussed. No dominant sequences is associated with LOOP or SBO. This SDP worksheet and associated LOOP event tree are borrowed from ANO-1.

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

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

(4) The human error for initiation of HPI cooling is 1.6E-2 (event UHAMUHPE on page 273). (The IPE also credits HPI cooling with 2 / 3 PORV and PSVs and 2/2 makeup pumps.)

(5) The HEP for operator failure to establish HPR is 4.8E-4. (Event XHAHPRTE on page 274.)

Table 2.8 SDP Worksheet for Davis-Besse Nuclear Power Station, Unit 1 —— 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 Depressurization (DEPS)	Initial steaming the SGs to stop primary-to-secondary leakage, prevent high level trip of MFW, and facilitate isolation of faulted SGs (high stress operator action) (1)
Steam Generator Heat Removal (PCS/AFW)	DEPS successful: 1/2 MFW pumps with 1/3 condensate pumps or 1 / 1 MDAFW trains (operator action) (2) or 1/2 TDAFW train (1multi-train system) DEPS failed: 1 / 1 MDAFW trains (operator action) (2) or 1/2 TDAFW train (1multi-train system)
High Pressure Injection (EIHP)	1 / 2 Make up pump trains (1 multi-train system)
Isolate Faulted SG (ISO)	closure of MSIV, reclosure of primary relief valves (1 train)
Depressurize RCS to SDC (DEPP)	1/1 PORV (operator action) (3)
Shutdown Cooling (SDC)	1 / 2 RHR pumps in decay heat removal mode (operator action) (4)
High Pressure Recirculation (HPR)	1/2 makeup pumps or 1/2 HPI trains taking suction from 1/2 LPI trains through LPI HX (operator action)(5)

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 SGTR - ISO - SDC (3, 10)			
2 SGTR - ISO - DEPP - HPR (5, 12)			
3 SGTR - EIHP (6, 13)			

4 SGTR - AFW - PCS (7)			
5 SGTR - DEPS - AFW (14)			

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

Note:

(1) A few IPE operator actions are associated with this top event. The HEP for operator failure to depressurize steam generators to cooldown is 1.0E-4 (event CHASGDPE on page 272). The HEP for operator failure to close MSIV to isolate the faulted SG is 1.0E-4 (event IHAMSIVE on page 272). The HEP for operator failure to control AVVs manually to cool down is 1.2E-3 (event XHACLDNE on page 274).

(2) The HEP for operator failure to initiate MDAFW is 2.4E-3 (event BHEMDFPE on page 272)

(3) The HEP for operator failure to depressurize RCS to the SDC condition is 1.0E-4 (event CHASGDPE on page 272).

(4) The HEP for operator failure to establish SDC is 1.0E-4 (event DHADHRSE on page 272).

(5) The HEP for operator failure to establish HPR is 4.8E-4. (Event XHAHPRTE on page 274.)

Table 2.9 SDP Worksheet for Davis-Besse Nuclear Power Station, Unit 1 —— ATWS

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

Safety Functions Needed:	***Full* Creditable Mitigation Capability for Each Safety Function:**
Main Feed Water (MFW)	1 / 2 Feedwater trains with 1 / 3 condensate trains (1 multi-train system)
Secondary Heat Removal (AFW)	1/2 TDAFW train (1multi-train system)
Primary Relief (SRV)	2 / 2 SRVs and 1/1 PORV open (1 train)
Emergency Boration (EB)	Operator conducts emergency boration using 1 / 2 makeup pumps (operator action)[(1)]

Circle Affected Functions	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
1 ATWS - MFW - EB (3)			
2 ATWS - MFW - SRV (4)			
3 ATWS - MFW - AFW (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.

Note:

(1) The operator action related to initiation of emergency oration has a value of 5.1E-3 in IPE, (event KHABORAE on page 272).

Table 2.10 SDP Worksheet for Davis-Besse Nuclear Power Station, Unit 1 Special Initiators

Estimated Frequency (Table 1 Row) _________	**Exposure Time _________**	**Table 1 Result (circle): A B C D E F G H**	
Safety Functions Needed:	***Full* Creditable Mitigation Capability for Each Safety Function:**		
Circle Affected Functions Initiator: Loss of SW	**Recovery of Failed Train**	***Remaining Mitigation Capability Rating for Each Affected Sequence***	**Sequence Color**
Initiator: Loss of CCW			
Initiator: Loss of a DC bus			
Initiator: Loss of a train of SW			
Initiator: Loss of Instrument Air			
Initiator: Internal floods in CCW pump room, service water pump			

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

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

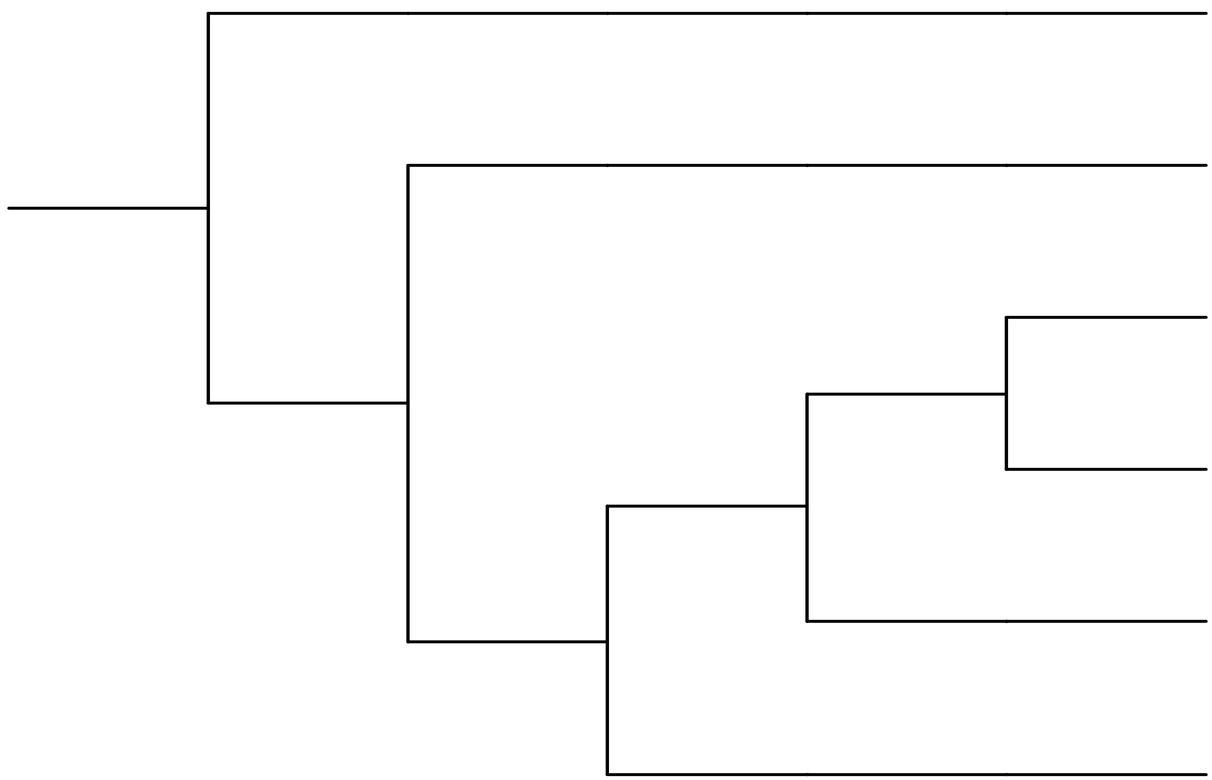
1.3 SDP EVENT TREES

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

The following event trees are included:

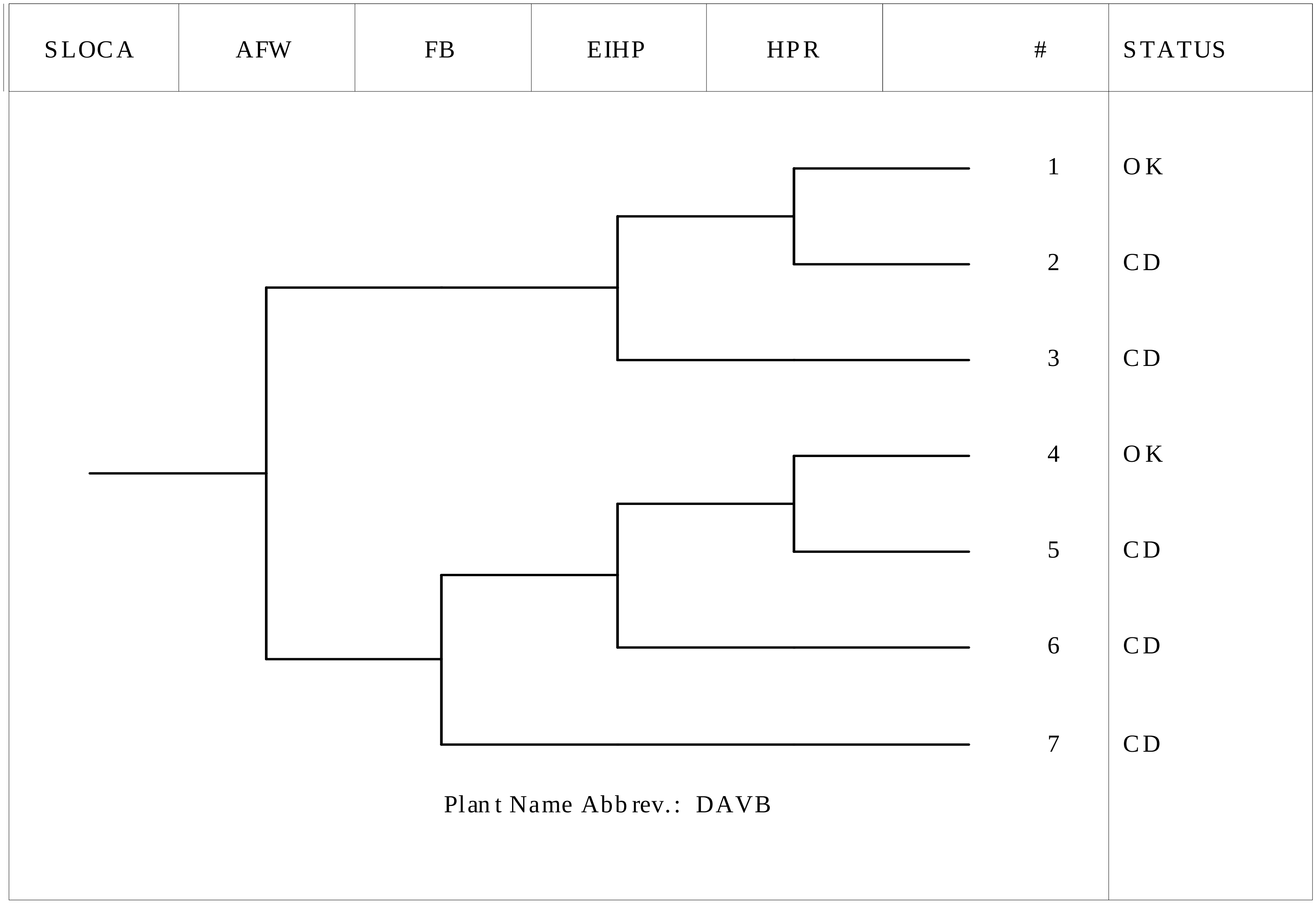
1. Transients
2. Small LOCA
3. Medium LOCA
4. Large LOCA
5. LOOP
6. Steam Generator Tube Rupture (SGTR)
7. Anticipated Transients Without Scram (ATWS)

TRANS	PCS	AFW	FB	EIHP	HPR	#	STATUS

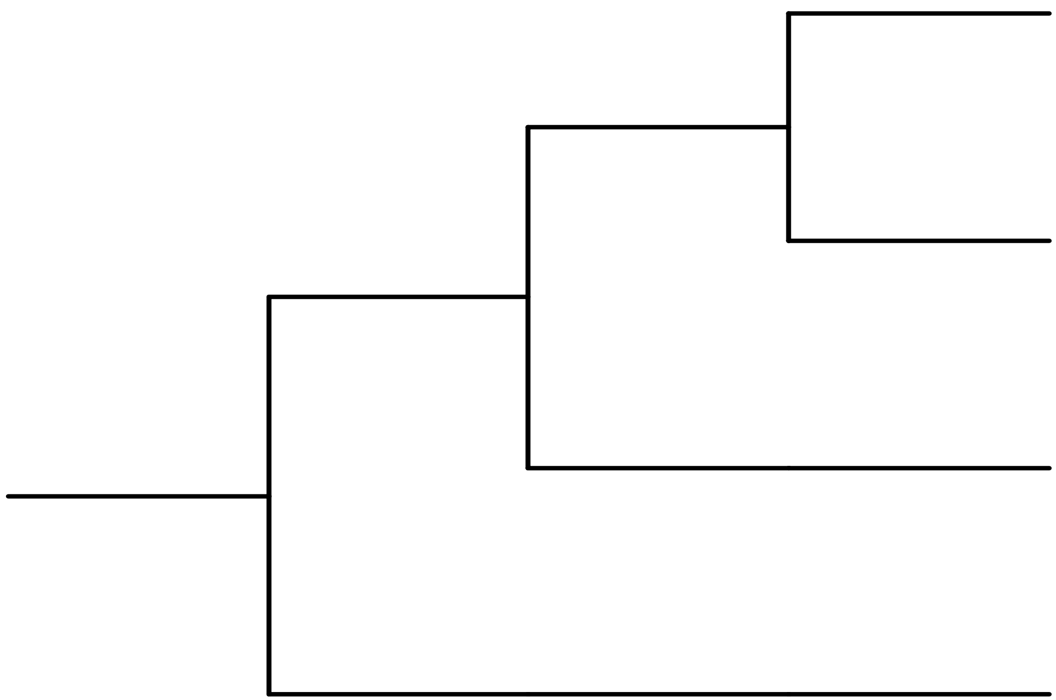


#	STATUS
1	OK
2	OK
3	OK
4	CD
5	CD
6	CD

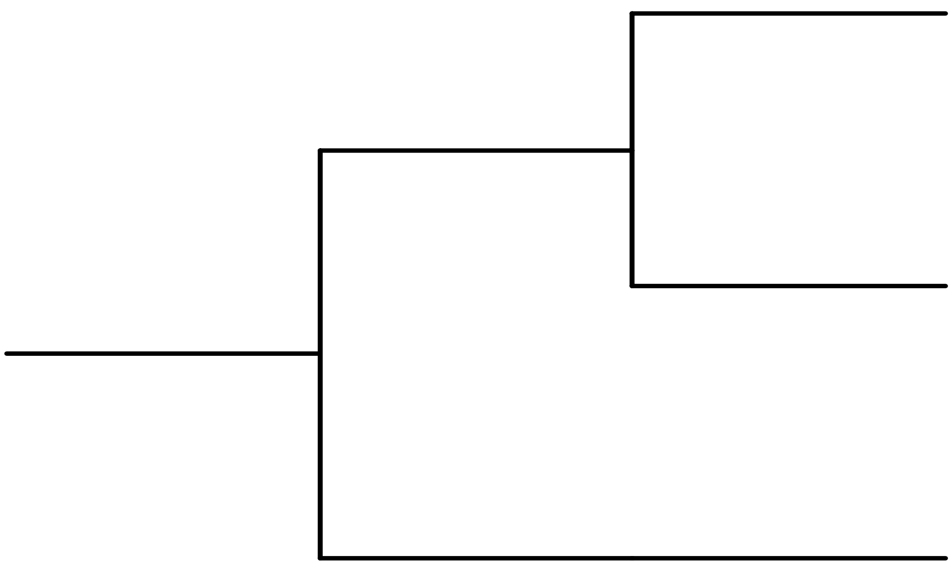
Plant Name Abbrev.: DAVB


SLOCA
AFW
FB
EIHP
HPR
#
STATUS
1 OK
2 CD
3 CD
4 OK
5 CD
6 CD
7 CD
Plant Name Abbrev.: DAVB

MLOCA	HPI	LPI	LPR	#	STATUS
				1	OK
				2	CD
				3	CD
				4	CD

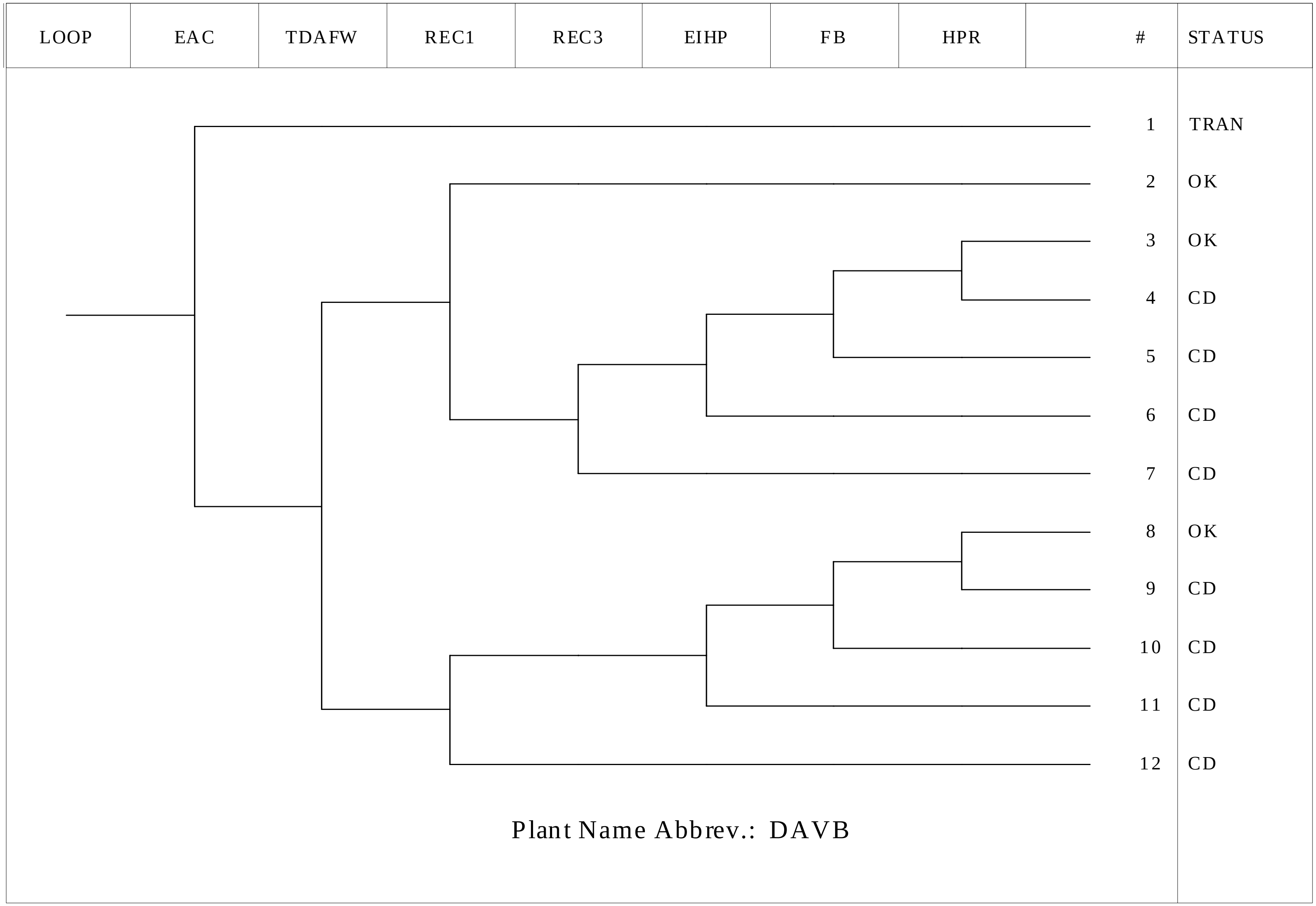


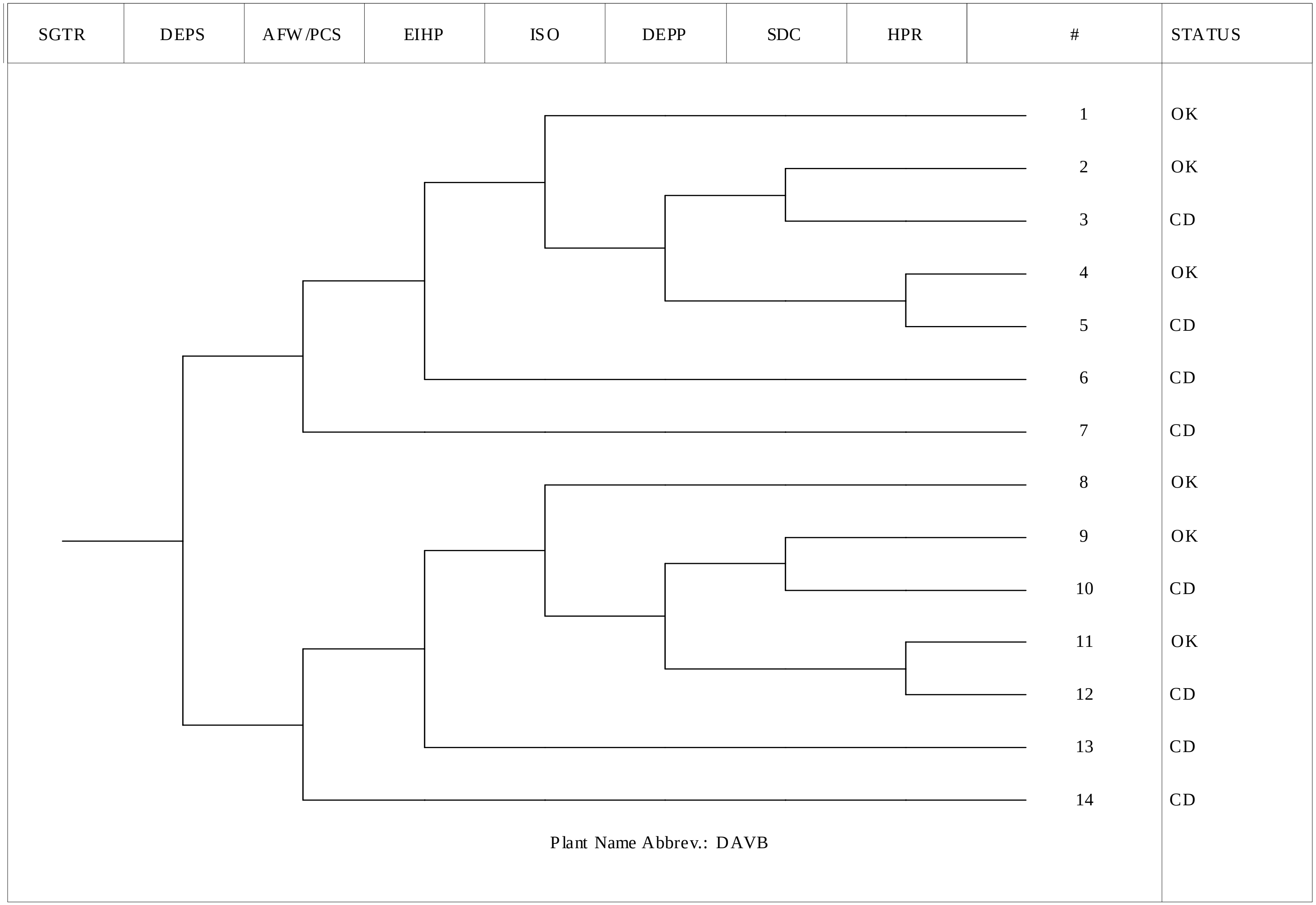
Plant Name Abbrev.: DAVB

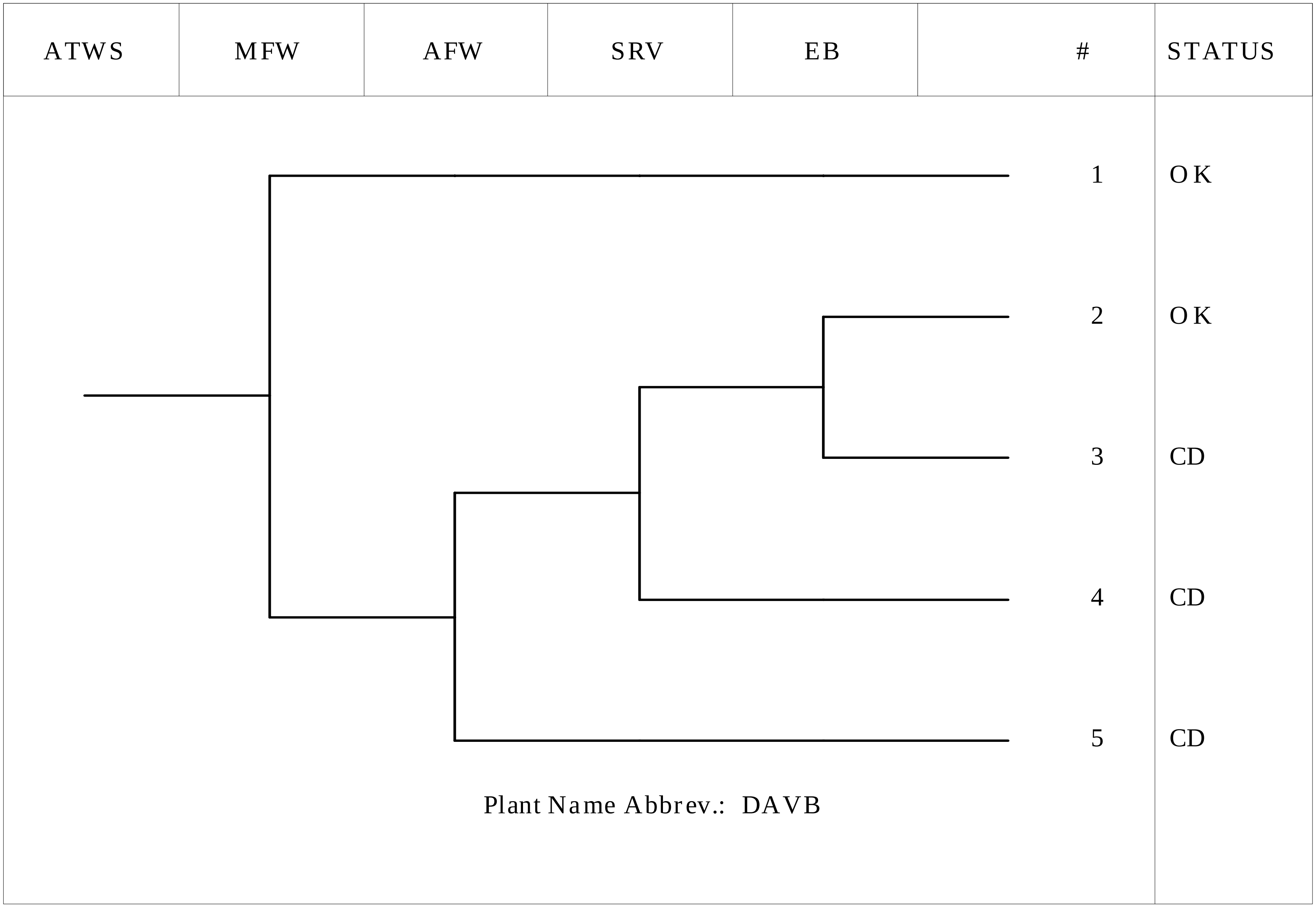


LLOCA	EILP	EILR	#	STATUS
			1	OK
			2	CD
			3	CD

Plant Name Abbrev.: DAVB


LOOP
EAC
TDAFW
REC1
REC3
EIHP
FB
HPR
#
STATUS
1 TRAN
2 OK
3 OK
4 CD
5 CD
6 CD
7 CD
8 OK
9 CD
10 CD
11 CD
12 CD
Plant Name Abbrev.: DAVB


SGTR
DEPS
AFW/PCS
EIHP
ISO
DEPP
SDC
HPR
#
STATUS
1 OK
2 OK
3 CD
4 OK
5 CD
6 CD
7 CD
8 OK
9 OK
10 CD
11 OK
12 CD
13 CD
14 CD
Plant Name Abbrev.: DAVB


ATWS
MFW
AFW
SRV
EB
#
STATUS
1
OK
2
OK
3
CD
4
CD
5
CD
Plant Name Abbrev.: DAVB

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. Centerior Energy, "Davis-Besse Nuclear Power Station, Unit 1, Individual Plant Examination Submittal Report," February 26, 1993.