April 23, 2002

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

Dear Mr. Venable:

Enclosed please find the Risk-Informed Inspection Notebook which incorporates the updated Significance Determination Process (SDP) Phase 2 Worksheets that inspectors will be using to characterize and risk-inform inspection findings. This document is one of the key implementation tools of the reactor safety SDP in the reactor oversight process.

Following initial implementation of the revised reactor oversight process, site visits were conducted by the Nuclear Regulatory Commission to verify and update plant equipment configuration data and to collect site-specific risk information from your staff. The enclosed document reflects the results of this visit.

The enclosed Phase 2 Worksheets have incorporated much of the information we obtained during our site visits. The staff encourages further licensee comments where it is identified that the Worksheets give inaccurately low significance determinations. Any comments should be provided to the Document Control Desk, with a copy to the Chief, Probabilistic Safety Assessment Branch, Nuclear Reactor Regulation. We will continue to assess SDP accuracy and update the document based on continuing experience.

While the enclosed Phase 2 Worksheets have been verified by our staff to include the site specific data we will continue to assess its accuracy throughout implementation and update the document based on comments by our inspectors and your staff.

Sincerely,

/RA/

N. Kalyanam, Project Manager, Section 1
Project Directorate IV
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No.: 50-382

Enclosures: As Stated

cc: See next page

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RISK-INFORMED INSPECTION NOTEBOOK FOR

WATERFORD NUCLEAR POWER PLANT

UNIT 3

PWR, COMBUSTION ENGINEERING, TWO-LOOP PLANT WITH LARGE DRY 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. All recommendations for improvement of this document should be forwarded to the Chief, Probabilistic Safety Assessment Branch, NRR, with a copy to the Chief, Inspection Program Branch, NRR.

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ABSTRACT

This notebook contains summary information to support the Significance Determination Process (SDP) in risk-informed inspections for the Waterford Nuclear Power Plant, Unit 3.

The information includes the following: Categories of Initiating Events Table, Initiators and System Dependency Table, SDP Worksheets, and SDP Event Trees. This information is used by the NRC's inspectors to identify the significance of their findings, i.e., in screening risk-significant findings, consistent with Phase 2 screening in SECY-99-007A. The Categories of Initiating Event Table is used to determine the likelihood rating for the applicable initiating events. The SDP worksheets are used to assess the remaining mitigation capability rating for the applicable initiating event likelihood ratings in identifying the significance of the inspector's findings. The Initiators and System Dependency Table and the SDP Event Trees (the simplified event trees developed in preparing the SDP worksheets) provide additional information supporting the use of SDP worksheets.

The information contained herein is based on the licensee's Individual Plant Examination (IPE) submittal, the updated Probabilistic Risk Assessment (PRA), and system information obtained from the licensee during site visits as part of the review of earlier versions of this notebook. Approaches used to maintain consistency within the SDP, specifically within similar plant types, resulted in sacrificing some plant-specific modeling approaches and details. Such generic considerations, along with changes made in response to plant-specific comments, are summarized.

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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. The first step in this is to identify the pertinent core damage scenarios that require further evaluation consistent with the specifics of the inspection findings. To aid in this process, this notebook provides the following information:

1. Estimated Likelihood Rating for Initiating Event Categories
2. Initiators and System Dependency Table
3. Significance Determination Process (SDP) Worksheets
4. SDP Event Trees.

Table 1, Categories of Initiating Events, is used to estimate the likelihood rating for different initiating events for a given degraded condition and the associated exposure time at the plant. This Table follows the format of Table 1 in SECY-99-007A. Initiating events are grouped in frequency bins that are one order of magnitude apart. The Table includes the initiating events that should be considered for the plant and for which SDP worksheets are provided. The following initiating events are categorized by industry-average frequency: transients (Reactor Trip) (TRANS); transients without power conversion system (TPCS); large, medium, and small loss of coolant accidents (LLOCA, MLOCA, and SLOCA); inadvertent or stuck open relief valve (IORV or SORV); main steam line break (MSLB), anticipated transients without scram (ATWS), and interfacing system LOCA (ISLOCA). The frequency of the remaining initiating events vary significantly from plant to plant, and accordingly, they are categorized by plant-specific frequency obtained from the licensee. They include loss of offsite power (LOOP) and special initiators caused by loss of support systems.

The Initiators and System Dependency Table shows the major dependencies between frontline- and support-systems, and identifies their involvement in different types of initiators. This table identifies the most risk-significant 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/success criteria 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 findings on the core-damage scenarios, SDP worksheets are provided. There are two sets of SDP worksheets; one for those initiators that can be mitigated by redundant trains of safety systems, and the other for those initiators that cannot be mitigated; however, their occurrence is prevented by several levels of redundant barriers.

The first set of SDP worksheets contain two parts. The first identifies the functions, the systems, or combinations thereof that have mitigating functions, the number of trains in each system, and the number of trains required (success criteria) for the initiator. It also characterizes the mitigation capability in terms of the available hardware (e.g., 1 train, 1 multi-train system) and the operator action involved. The second part of the SDP worksheet contains the core-damage accident sequences associated with each initiator; these sequences are based on SDP event trees. In the parenthesis next to each sequence, the corresponding event-tree branch number(s) representing the sequence is given. Multiple branch numbers indicate that the different accident sequences identified by the event tree have been merged into one through Boolean reduction. The SDP worksheets are developed for each of the initiating event categories, including the "Special Initiators", the exception being those which directly lead to a core damage (the inspections of these initiators are assessed differently; see SECY-99-007A). The special initiators are those that are caused by complete or partial loss of support systems. A special initiator typically leads to a reactor scram and degrades some frontline or support systems (e.g., Loss of CCW in PWRs).

In considering the special initiators, we defined a set of criteria for including them to maintain some consistency across the plants. These conditions are as follows:

1. The special initiator should degrade at least one of the mitigating safety functions thereby changing its mitigation capability in the worksheet. For example, when a safety function with two redundant trains, classified as a multi-train system, degrades to a one-train system, it is classified as 1 Train, due to the loss of one of the trains as a result of the special initiator.

2. The special initiators which degrade the mitigation capability of the systems/functions associated with the initiator from comparable transient sequences by two and higher orders of magnitude must be considered.

From the above considerations, the following classes of initiators are considered in this notebook:

1. Transients with power conversion system (PCS) available, called Transients (Reactor trip) (TRANS),
2. Transients without PCS available, called Transients w/o PCS (TPCS),
3. Small Loss of Coolant Accident (SLOCA),
4. Stuck-open Primary Safety Valve (SOSV),
5. Medium LOCA (MLOCA),
6. Large LOCA (LLOCA),
7. Steam Generator Tube Rupture (SGTR),
8. Anticipated Transients Without Scram (ATWS), and
9. Main Steam Line Break (MSLB).

Examples of special initiators included in the notebook are as follows:

1. Loss of Offsite Power (LOOP),
2. LOOP with failure of 1 Emergency AC bus or associated EDG (LEAC),
3. Loss of 1 DC Bus (LDC),

4. Loss of component cooling water (LCCW),
5. Loss of instrument air (LIA),
6. Loss of service water (LSW).

The worksheet for the LOOP includes LOOP with emergency AC power (EAC) available and LOOP without EAC, i.e., Station Blackout (SBO). LOOP with partial availability of EAC, i.e., LOOP with loss of a bus of EAC, is covered in a separate worksheet to avoid making the LOOP worksheet too large. In some plants, LOOP with failure of 1 EAC bus is a large contributor to the plant's core damage frequency (CDF).

The second set of SDP worksheets addresses those initiators that cannot be mitigated, i.e., can directly lead to core-damage. It currently includes the Interfacing System LOCA (ISLOCA) initiator. ISLOCAs are those initiators that could result in a loss of RCS inventory outside the containment, sometimes referred to as a "V" sequence. In PWRs, this event effectively bypasses the capability to utilize the containment sump recirculation once the RWST has emptied. Also, through bypassing the containment, the radiological consequences may be significant. In PWRs, this typically includes loss of RCS inventory through high- and low-pressure interfaces, such as RHR connections, RCP thermal barrier heat-exchanger, high-pressure injection piping if the design pressure (pump head) is much lower than RCS pressure, and, potentially, through excess letdown heat exchanger. RCS inventory loss through ISLOCA could vary significantly depending on the size of the leak path; some may be recoverable with minimal impact. The SDP worksheet for ISLOCA, therefore, identifies the major consequential leak paths, and the barriers that should fail, allowing the initiator to occur.

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. For special initiators whose event tree closely corresponds to another event tree (typically, the Transient (Reactor trip) or Transients w/o PCS event tree) with one or more functions eliminated or degraded, a separate event tree may not be drawn.

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/PRAs. The special initiators modeled for a plant is based on a review of the special initiators included in the plant IPE/PRA and the information provided by the licensee.

2. The event trees and sequences for each plant take into account the IPE/PRA models and event trees for all similar plants. For modeling the response to an initiating event, any major deviations in one plant from similar plants may be 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 developed 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/PRAs often are represented by a single tree. For example, some IPEs/PRAs 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 sometimes divided into two classes, the only difference between them being the need for reactor scram in the smaller break size. There may be some consolidation of transient event trees besides defining the special initiators following the criteria defined above.

5. Major actions by the operator during accident scenarios are credited using four categories of Human Error Probabilities (HEPs). They are termed operator action =1 (representing an error probability of 5E-2 to 0.5), operator action=2 (error probability of 5E-3 to 5E-2), operator action=3 (error probability of 5E-4 to 5E-3), and operator action=4 (error probability of 5E-5 to 5E-4). An human action is assigned to a category bin, based on a generic grouping of similar actions among a class of plants. This approach resulted in designation of some actions to a higher bin, even though the IPE/PRA HEP value may have been indicative of a lower category. In such cases, it is noted at the end of the worksheet. On the other hand, if the IPE/PRA HEP value suggests a higher category than that generically assumed, the HEP is assigned to a bin consistent with the IPE/PRA value in recognition of potential plant-specific design; a note is also given in these situations. Operator's actions belonging to category 4, i.e., operator action=4, may only be noted at the bottom of worksheet because, in those cases, equipment failures may have the dominating influence in determining the significance of the findings.

The four sections that follow include Categories for Initiating Events Table, Initiators and Dependency Table, SDP worksheets, and the SDP event trees for Waterford Nuclear Power Plant, Unit 3.

1.1 INITIATING EVENT LIKELIHOOD RATINGS

Table 1 presents the applicable initiating events for this plant and their estimated likelihood ratings corresponding to the exposure time for degraded conditions. The initiating events are grouped into rows based on their frequency. As mentioned earlier, loss of offsite power (LOOP) and special initiators are assigned to rows using the plant-specific frequency obtained from individual licensees. For other initiating events, industry-average values are used.

Table 1 Categories of Initiating Events for Waterford Nuclear Power Plant, Unit 3

Row	Approximate Frequency	Example Event Type	Estimated Likelihood Rating		
I	> 1 per 1-10 yr	Reactor Trip (TRANS), Loss of Power Conversion System (TPCS)	**A**	**B**	**C**
II	1 per 10-10^{2} yr	Loss of Offsite Power (LOOP), DCBUS, DCPDPAB	**B**	**C**	**D**
III	1 per 10^{2} - 10^{3} yr	Steam Generator Tube Rupture (SGTR), Stuck open SRV (SOSV), Small LOCA including RCP seal failures (SLOCA), MSLB/FLB (outside containment), Loss of Instrument Air (LIA)	**C**	**D**	**E**
IV	1 per 10^{3} - 10^{4} yr	Medium LOCA (MLOCA), Loss of SW system (LSSW), Loss of CCW (LCCW), LOOP with loss of one division of emergency AC (LOOP1EDG), DC3AB	**D**	**E**	**F**
V	1 per 10^{4} - 10^{5} yr	Large LOCA (LLOCA)	**E**	**F**	**G**
VI	less than 1 per 10^{5} yr	ATWS[1], Interfacing System LOCA (ISLOCA)	**F**	**G**	**H**
			> 30 days	**3-30 days**	**< 3 days**
			Exposure Time for Degraded Condition		

Note:

1. The SDP worksheets for ATWS core damage sequences assume that the ATWS is not recoverable by manual actuation of the reactor trip function. Thus, the ATWS frequency to be used by these worksheets must represent the ATWS condition that can only be mitigated by the systems shown in the worksheet (e.g., boration). Any inspection finding that represents a loss of capability for manual reactor trip for a postulated ATWS scenario should be evaluated by a risk analyst to consider the probability of a successful manual trip.

1.2 INITIATORS AND SYSTEM DEPENDENCY

Table 2 lists the systems included in the SDP worksheets, the major components in the systems, and the support system dependencies. The systems' involvements in different initiating events are noted in the last column.

Table 2 Initiators and System Dependency for Waterford Nuclear Power Plant, Unit 3[1]

Affected Systems	Major Components	Support Systems	Initiating Event
AC Power System	AC Power Distribution & AC Instrument Power, SUTs	DC, HVAC	All
EFW	2 MDPs, Condensate Storage Pool (CSP), AOVs	AC, DC, ESFAS, HVAC, IA[2]	All except MLOCA, and LLOCA
	1 TDP, CSP	ESFAS, DC, IA[2], Main Steam	
CCW	3 Pumps (1 swing pump) in two trains with one dry cooling tower and one CCW heat exchanger in each train, fans	AC, DC, ESFAS, ACCW, HVAC, IA	All
Auxiliary Component Cooling Water (ACCW)	2 Pumps and 2 Wet Cooling Towers, MOVs	AC, DC, ESFAS	All
Condensate / MFW	3 Condensate Pumps	AC, DC	TRANS, MSLB, LCCW, DCBUS, DCBUSAB, SLOCA, SOSV
	2 SGFPs, FWIVs	AC, DC, IA, Main Steam	
	1 Auxiliary Feedwater Pump (AFW), Condensate Storage Tank (CST)	AC, DC, IA	
Containment Cooling System (CCS)	4 Fan Coolers	AC (480V), DC, ESFAS, CCW	SLOCA, SOSV, MLOCA, LLOCA
Containment Spray System(CSS)	2 Trains, each with 1 pump and 1 SD heat exchanger	AC, DC, ESFAS, HVAC, IA, CCW	SLOCA, SOSV, MLOCA, LLOCA
HPSI	2 HPSI trains with a third swing train, 380 gpm at 1280 psi	AC, DC, ESFAS, CCW, HVAC	SLOCA, SOSV, MLOCA, LOOP, SGTR, ATWS, MSLB
Chemical Volume Control System (CVCS)	3 Pumps, 44 gpm at 2324 psi	AC, DC, IA, HVAC	SGTR, ATWS

Table 2 (Continued)

Affected Systems	Major Components	Support Systems	Initiating Event
DC Power System	Buses, Battery Chargers and Batteries	AC Distribution (without AC, battery capacity is 6 hrs.), HVAC	All
EDG	2 EDGs with 2 hr day tank (feed tank) each and a fuel transfer pump	DC, HVAC, CCW	LOOP, LOOP1EDG
HVAC (3)	Area fan coolers and 2 essential chilled water trains with a swing chiller	AC, CCW, ESFAS, DC	All
Instrument Air (IA)	2 Air Compressors	AC, DC, Turbine building cooling water	TIA
Main Steam	2 SGs, each with 1 ADV, 6 Safety Valves, 1 MSIV and 3 Turbine bypass valves	DC, IA, Vital AC	All
Pressurizer Pressure Relief	2 Safety valves open at 2500 psia	None	LOOP, LOOP1EDG, ATWS, SOSV
RCP	Seals	1 / 3 CCW pumps to thermal barrier heat exchanger, seal injection is not needed	SLOCA (RCP seal LOCA)
Safety Injection Tank (SIT)	4 SITs	none	LLOCA
LHSI	2 LPSI pumps	AC, DC, ESFAS, CCW, HVAC	LLOCA

Notes:

1. Plant internal event CDF (including internal floods) = 2.54 E-5/yr per Rev 2 PSA.

2. The safety systems requiring IA are also supported by Nitrogen backup. EFW therefore will not be adversely affected by a loss of IA.

(3) There is possibility for manual recovery of HVAC and backup alignment with ACCW system.

1.3 SDP WORKSHEETS

This section presents the SDP worksheets to be used in the Phase 2 evaluation of the inspection findings for the Waterford Nuclear Power Plant, Unit 3. The SDP worksheets are presented for the following initiating event categories:

1. Transients (Reactor Trip) (TRANS)
2. Transients without PCS (TPCS)
3. Small LOCA (SLOCA)
4. Stuck-open Safety Valve (SOSV)
5. Medium LOCA (MLOCA)
6. Large LOCA (LLOCA)
7. Loss of Offsite Power (LOOP)
8. Steam Generator Tube Rupture (SGTR)
9. Anticipated Transients without Scram (ATWS)
10. Main Stream Line Break (MSLB)
11. Loss of CCW (LCCW)
12. Loss of DC Bus A or B (DCBus)
13. Loss of DC Bus 3AB-DC-S (DC3AB)
14. Loss of DC PDP AB (DCPDPAB)
15. Loss of Instrument Air (TIA)
16. LOOP with 1 EDG Available (LOOP1EDG)
17. Interfacing System LOCA (ISLOCA)

Table 3.1 SDP Worksheet for Waterford Nuclear Power Plant, Unit 3 — Transients (Reactor Trip) (TRANS)

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 Main Feedwater pumps with 1/3 condensate pumps or 1/1 AFW pump or depressurization with 1/2 ADVs or 1/6 TBVs and feed with 1/3 condensate pumps (operator action = 3) [1]
Emergency Feedwater System (EFW)	1/2 MD EFW trains (1 multi-train system) or 1 TD EFW train (1 ASD train)

Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 TRANS - PCS - EFW (3)			

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

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

Note:

1. A HEP credit of 3 is given for combined PCS function per licensee's comment.

Table 3.2 SDP Worksheet for Waterford Nuclear Power Plant, Unit 3 — Transients without PCS (TPCS)

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

Safety Functions Needed:	**Full Creditable Mitigation Capability for Each Safety Function:**
Secondary Heat Removal (SHR)	1/2 MD EFW trains (1 multi-train system) or 1 TD EFW train(1 ASD train) or 1/1 AFW pump (operator action = 2) with steam relief through 1/2 ADV or 1/6 SSVs

Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 TPCS - SHR (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:

Table 3.3 SDP Worksheet for Waterford Nuclear Power Plant, Unit 3 — Small LOCA (SLOCA)

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:**
High Pressure Safety Injection (HPSI)	1/2 high pressure injection trains (3 pumps) inject to 2 intact cold legs from refueling water storage pool (RWSP) (1 multi-train system)
Main Feedwater (PCS)	1/2 MFW trains or 1 AFW train with steam relief through 1/2 ADV or 1/6 SSVs (operator action = 2) (1)
Emergency Feedwater System (EFW)	1/2 MD EFW trains (1 multi-train system) or 1 TD EFW train (1 ASD train)
High Pressure Recirculation (HPR)	1/2 HPSI trains (3 pumps) in recirculation mode (1 multi-train system) (2)
Containment Heat Removal (CHR)	1/2 fan cooler trains (2 fan cooling units per train) (1 multi-train system)or 1/2 trains of containment spray injection and recirculation (1 multi-train system)

Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 SLOCA - CHR (2,5)			
2 SLOCA - HPR (3,6)			
3 SLOCA - PCS - EFW (7)			
4 SLOCA - HPSI (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 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 credit of 2 for PCS is generically assigned based on CE plants.

2. Recirculation is actuated automatically. The operator has to manually close the RWSP discharge valves (as a back up to check valves).

Table 3.4 SDP Worksheet for Waterford Nuclear Power Plant, Unit 3 — Stuck Open Primary Safety Valve (SOSV) [1]

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

<u>Safety Functions Needed</u>:	**<u>Full Creditable Mitigation Capability for Each Safety Function</u>:**
Stuck Open PSV (SOSV)	1/2 PSV fail to reclose when demanded (1 train system).
High Pressure Safety Injection (HPSI)	1/2 high pressure injection trains (3 pumps) inject to 2 intact cold legs from refueling water storage pool (RWSP) (1 multi-train system)
Main Feedwater (PCS)	1/2 MFW trains or 1 AFW train (operator action = 2) [2]
Emergency Feedwater System (EFW)	1/2 MD EFW trains (1 multi-train system) or 1 TD EFW train (1 ASD train)
High Pressure Recirculation (HPR)	1/2 HPSI trains (3 pumps) in recirculation mode (1 multi-train system) [3]
Containment Heat Removal (CHR)	1/2 fan cooler trains (2 fan cooling units per train) (1 multi-train system) or 1/2 trains of containment spray injection and recirculation (1 multi-train system)

<u>Circle Affected Functions</u>	**<u>Recovery of Failed Train</u>**	**<u>Remaining Mitigation Capability Rating for Each Affected Sequence</u>**	**<u>Sequence Color</u>**
1 SOSV - HPSI (8)			
2 SOSV - PCS - EFW (7)			
3 SOSV - HPR (3,6)			
4 SOSV - CHR (2,5)			

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

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

Notes:

(1) The Waterford 3 plant does not have PORVs. The work sheet was developed to model the scenarios that started with a transient and a pressurizer safety valve failed to re-close.

(2) The human error probability used in the IPE is 3.0E-4 for operator failure to recognize the need for feedwater (Event OPER-2). A generic HEP credit of 2 is assigned.

(3) Recirculation is actuated automatically. The operator has to manually close the RWSP discharge valves (backup to check valves).

Table 3.5 SDP Worksheet for Waterford Nuclear Power Plant, Unit 3 — Medium LOCA (MLOCA)

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:**
High Pressure Safety Injection (HPSI)	1/2 high pressure injection trains (3 pumps) inject to 2 intact cold legs from refueling water storage pool (RWSP) (1 multi-train system)
High Pressure Recirculation (HPR)	1/2 HPSI trains (3 pumps) in recirculation mode (1 multi-train system) (1)
Containment Heat Removal (CHR)	1/2 fan cooler trains (2 fan cooling units per train) (1 multi-train system) or 1/2 trains of containment spray injection and recirculation (1multi-train system)

Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 MLOCA - CHR (2)			
2 MLOCA - HPR (3)			
3 MLOCA - HPSI (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) Recirculation is actuated automatically. The operator has to manually close the RWSP discharge valves (backup to check valves).

Table 3.6 SDP Worksheet for Waterford Nuclear Plant, Unit 3 — Large LOCA (LLOCA)

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:
Safety Injection Tank (SIT)	3/3 intact SITs inject into intact RCS legs (1 train system)
High Pressure Safety Injection (HPSI)	1/2 high pressure injection trains (3 pumps) inject to 2 intact cold legs from refueling water storage pool (RWSP) (1 multi-train system)
Low Pressure Safety Injection (LPSI)	1/2 LHSI trains inject from RWSP to 1 intact cold leg (1 multi-train system)
High Pressure Recirculation (HPR)	1/2 HPSI trains (3 pumps) in recirculation mode (1 multi-train system)
Containment Heat Removal (CHR)	1/2 fan cooler trains (2 fan cooling units per train) (1 multi-train system)or 1/2 trains of containment spray injection and recirculation (1 multi-train system)

Circle Affected Functions	Recovery of Failed Train	Remaining Mitigation Capability Rating for Each Affected Sequence	Sequence Color
1 LLOCA - CHR (2)			
2 LLOCA - HPR (3)			
3 LLOCA - LPSI (4)			
4 LLOCA - HPSI (5)			
5 LLOCA - SIT (6)			

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

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

Table 3.7 SDP Worksheet for Waterford Nuclear Power Plant, Unit 3 — Loss of Offsite Power (LOOP)

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

Safety Functions Needed:	**Full Creditable Mitigation Capability for Each Safety Function:**
Emergency Diesel Generator (EDG)	1/2 Emergency Diesel Generators (1 multi-train system)
Turbine-driven EFW pump (TDEFW)	1/1 TDP trains of EFW (1 ASD train)
Emergency Feedwater (EFW)	1/2 MDEFW trains (1 multi-train system) or 1 TDEFW train (1 ASD train)
Recovery of AC Power in < 1 hrs (REC1)	Recovery of offsite power (operator action = 1) [(1)]
Recovery of AC Power in < 6 hrs (REC8)	Recovery of an AC source (operator action = 2) [(2)]

Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 LOOP - EFW (2,6)			
2 LOOP - EDG - REC8 (4)			
3 LOOP - EDG - TDEFW - REC1 (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 PSA uses a probability of 0.458 for failure to recover in 50 minutes. (Event Z-LOOP-R0 on page 3.7-20)

(2) The PSA uses a probability of 2E-2 for failure to recover in 8 hours (Event Z-LOOP10). Note that RCP seal LOCA is not currently modeled in the worksheet. The probability of RCP seal LOCA in 4 hours for CE plants is about 1.E-3 which would result in core damage in 6 to 8 hours if offsite power is not recovered.

Table 3.8 SDP Worksheet for Waterford, Unit 3 — Steam Generator Tube Rupture (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:**
Pressure Equalization (EQ)	Operator isolates the ruptured SG and depressurizes RCS using 1/1 ADV or 1/6 SBVs or uses 1/3 charging pumps in 1/2 trains for RCS pressurizer spray and blowdown of the SGs to less than setpoint of relief valves (operator action = 3) (1)
Power Conversion System (PCS)	1/2 Main Feedwater trains with 1/3 condensate trains or 1/1 AFW to the unaffected SG pump (operator action = 2) (2)
Emergency Feedwater System (EFW)	1/2 MDPs of EFW (1 multi-train system) or 1/1 TDP of EFW (1 ASD Train) to the unaffected SG
RCS Inventory Makeup (RCSMU)	1/2 HPSI trains (3pumps) (1 multi-train system) or 2/3 charging pumps (1 train system)

Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 SGTR - EQ (6)			
2 SGTR - RCSMU (2,4)			
3 SGTR - PCS -EFW (5)			

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

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

Notes:

(1) The worksheet models the early and late equalization in one combined function consistent with the licensee's PSA. The worksheet assigns a credit of 3 for the overall HEP for EQ function. The PSA uses a HEP value of 1.4E-6 per licensee's comment.

(2) The HEP value based on the plant PSA is 1.2E-2. A HEP credit of 2 is therefore assigned.

Table 3.9 SDP Worksheet for Waterford Nuclear Power Plant, Unit 3 — Anticipated Transients without Scram (ATWS)

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

Safety Functions Needed:	**Full Creditable Mitigation Capability for Each Safety Function:**
Turbine Trip (TTP)	Manually trip the turbine (operator action = 2)
Emergency Feedwater (EFW)	1/2 MDEFW trains (1 multi-train system) or 1 TDEFW train (1 ASD train)
Primary Safety Valves Open (SRVO)	2/2 SVs open (1 train)
Emergency Boration (EB)	Operator conducts emergency boration using 1/3 charging pumps from RWSP or BAST (operator action = 3) (1)
Primary Safety Valves Reclose (SRVR)	2/2 SRVs reclose
High Pressure Safety Injection (HPSI)	1/2 high pressure injection trains (3 pumps) inject to 2 intact cold legs from refueling water storage pool (RWSP) (1 multi-train system)
High Pressure Recirculation (HPR)	1/2 HPSI trains (3 pumps) in recirculation mode (1 multi-train system)
Containment Heat Removal (CHR)	1/2 trains of fan coolers (1 multi-train system) or 1/2 trains of containment spray injection and recirculation (1 multi-train system)

Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 ATWS - TTP (9)			
2 ATWS - EFW (8)			
3 ATWS - SRVO (7)			
4 ATWS - EB (6)			

5 ATWS - SRVR - HPSI (5)			
6 ATWS - SRVR - HPR (4)			
7 ATWS - SRVR - CHR (3)			

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

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

Note:

(1) The IPE did not document the human error probability.

Table 3.10 SDP Worksheet for Waterford Nuclear Power Plant, Unit 3 — Main Steam Line Break (MSLB)

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

<u>Safety Functions Needed</u>:	**<u>Full Creditable Mitigation Capability for Each Safety Function</u>:**
Both Main Steam Isolation Valves (MSIV2)	Closure of both MSIVs (1 train)
One Main Steam Isolation Valves (MSIV1)	Closure of 1/1 remaining MSIV (1 train)
Isolation of Feed (FWI)	Closure of isolation valves feeding the affected SG (operator action = 2)
High Pressure Safety Injection (HPSI)	1/2 high pressure injection trains (3 pumps) (1 multi-train system)
Secondary Heat Removal (SHR)	1/2 MD EFW trains (1 multi-train system) or 1 TD EFW train (1 ASD train) or 1/1 AFW pump (operator action = 2)

<u>Circle Affected Functions</u>	**<u>Recovery of Failed Train</u>**	**<u>Remaining Mitigation Capability Rating for Each Affected Sequence</u>**	**<u>Sequence Color</u>**
1 MSLB - SHR (2,4)			
2 MSLB - MSIV2 - HPSI (5)			
3 MSLB - MSIV2 - FWI (6)			
4 MSLB - MSIV2 - MSIV1 (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.

Table 3.11 SDP Worksheet for Waterford Nuclear Power Plant, Unit 3 — Loss of CCW (LCCW) [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:**
Reactor Coolant Pump Trip (RCPSEAL)	Prevent a seal LOCA by tripping RCPs and closure of bleed off line (operator action = 3) [2]
Power Conversion System (PCS)	1/2 Main Feedwater trains with 1/3 condensate trains or 1/1 AFW pump or depressurization with 1/2 ADVs or 1/6 TBVs and feed with 1/3 condensate pumps (operator action = 3) [3]
Emergency Feedwater System (EFW)	1/2 MD EFW trains (1 multi-train system) or 1 TD EFW train (1 ASD train)

Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 LCCW - RCPSEAL (4)			
2 LCCW - PCS - EFW (3)			

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

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

Notes:

1. The frequency of loss of CCW is 5.0E03 per year per IPE. Loss of CCW leads to loss of cooling to RCP seals, high and low pressure safety injection pump oil coolers, and containment fan coolers. It requires a reactor trip and RCP trip.

2. The HEP for operator failure to trip the RCPs within 30 minutes of loss of seal cooling is 5.2E-5 (event OPER-6 on page 3.4-8). A combined HEP credit of 3 is generically given for the necessary operator actions to minimize the likelihood of RCP seal failures.

3. The human error probability used in the IPE is 3.0E-4 for operator failure to recognize the need for feedwater. (Event OPER-2 on page 3.4-8.)

Table 3.12 SDP Worksheet for Waterford Nuclear Power Plant, Unit 3 — Loss of DC Bus A or B (DCBus) (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 Main FW trains with 1/3 condensate trains or 1/1 AFW pump or depressurization with 1/2 ADVs and feed with 1/3 condensate pumps (operator action = 3) (2)		
Emergency Feedwater System (EFW)	1/1 MD EFW trains (1 train) or 1/1 TD EFW train (1 ASD train)		
Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 DCBUS - PCS - EFW (3)			

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

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

Notes:

1. This initiating event models either the loss of 125V DC Bus 3A-DC-S, or 3B-DC-S (referred to as TDC1 and TDC2 in the PSA). The frequency of loss of DC bus A or B is estimated to be about 5.0E-2 (2X2.5E-2) per year. It causes loss of DC to the reactor trip breakers, and 1 train of MDEFW, 1 train of safety injection, 1 train of containment spray, steam bypass valves, and 1 containment isolation train. The transient event tree is used for this initiating event with the impacts included in the worksheet.

2. The human error probability used in the IPE is 3.0E-4 for operator failure to recognize the need for feedwater. (Event OPER-2 on page 3.4-8.)

Table 3.13 SDP Worksheet for Waterford Nuclear Power Plant, Unit 3 — Loss of DC Bus 3AB-DC-S (DC3AB) (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)	Depressurization with 1/2 ADVs with 1/3 condensate pumps (operator action = 1) (2)		
Emergency Feedwater System (EFW)	1/2 MD EFW trains (1 multi-train system)		
Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 DC3AB - PCS - EFW (3)			
Identify any operator recovery actions that are credited to directly restore the degraded equipment or initiating event:			
If operator actions are required to credit placing mitigation equipment in service or for recovery actions, such credit should be given only if the following criteria are met: 1) sufficient time is available to implement these actions, 2) environmental conditions allow access where needed, 3) procedures exist, 4) training is conducted on the existing procedures under conditions similar to the scenario assumed, and 5) any equipment needed to complete these actions is available and ready for use.			

Notes:

1. This initiating event models the loss of 125 VDC Bus 3AB-DC-S (TDC3 in PSA). Since this bus provide control power to the MFW pumps, failure causes the loss of the ability to adjust to normal variations in feedwater flow demand. It is therefore assumed that the loss of this bus will result in reactor trip on high/low SG level. The frequency of loss of this bus is estimated to be about 3.94E-4 per year. It causes loss of TDEFW pump, main feedwater pumps, and AFW pump. The transient event tree is used for this initiating event with the impacts included in the worksheet.

2. The human error probability used in the IPE is 3.0E-4 for operator failure to recognize the need for feedwater (Event OPER-2 on page 3.4-8). A HEP credit of 1 is given in the worksheet accounting for unavailability of MFW/AFW pump.

Table 3.14 SDP Worksheet for Waterford Nuclear Power Plant, Unit 3 — Loss of DC PDP AB (DCPDPAB) (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)	Depressurization with 1/2 ADVs with 1/3 condensate pumps or use of 1/1 AFW pump (operator action = 2) (2)		
Emergency Feedwater System (EFW)	1/2 MD EFW trains (1 multi-train system) or 1/1 TDEFW pump (1 ASD train)		
Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 DCPDPAB - PCS - EFW (3)			

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

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

Notes:

1. The frequency of loss of DC power distribution panel PDP 3014 AB (TDC4 in PSA) is 2.57 E-2 per year. This panel provides power to the MFW pumps, MFW control valves, and steam bypass valves. This initiating event is modeled by the transient event tree with the impacted included in the worksheet.

2. Reduced credit due to unavailability of the MFW pumps.

Table 3.15 SDP Worksheet for Waterford Nuclear Power Plant, Unit 3 — Loss of Instrument Air (TIA) (1)

Estimated Frequency (Table 1 Row) __________	**Exposure Time** __________	**Table 1 Result (circle): A B C D E F G H**	
Safety Functions Needed: **Emergency Feedwater System (EFW)**	**Full Creditable Mitigation Capability for Each Safety Function:** 1/2 MD EFW trains (1 multi-train system) or 1 TD EFW train (1 ASD train)		
Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 TIA - EFW			
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 frequency of loss of instrument air is 0.081 per year. Loss of instrument air causes failure of MFW control valves, loss of steam supply to MFW turbine, and loss of air to steam bypass and main pressurizer spray valves. The EFW control valves, atmospheric dump valves, and various CCW valves have nitrogen accumulators. This initiating event basically causes a loss of PCS.

Table 3.16 SDP Worksheet for Waterford Nuclear Power Plant, Unit 3 — LOOP with 1 EDG available (LOOP1EDG) [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:**
Stuck Open Safety Valves (SOSV)	2/2 SVs re-close (1 train)
High Pressure Safety Injection (HPSI)	1/1 high pressure injection train (1 train) [2]
Emergency Feedwater System (EFW)	1/1 MD EFW train (1-train) or 1 TD EFW train (1 ASD train)
High Pressure Recirculation (HPR)	1/1 HPSI train in recirculation mode (1 train)
Containment Heat Removal (CHR)	1/1 fan coolers train or 1/1 trains of containment spray injection and recirculation (1 multi-train system)

Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 LOOP1EDG - EFW (2,6)			
2 LOOP1EDG - SOSV - CHR (4)			
3 LOOP1EDG - SOSV - HPR (5)			
4 LOOP1EDG -SOSV - HPSI (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 SDP supplements the LOOP event tree by modeling the condition in which only one EDG is available. The impacts are that the pressurizer safety valves may fail to re-close and only one MDEFW pump is unavailable.

2. The HEP for operator failure to start or align pump AB to inject is 9.8E-2 (event ZHFOPALNAB on page 3.4-8).

Table 3.17 SDP Worksheet for Waterford Nuclear Power Plant, Unit 3 — Interfacing System LOCA (ISLOCA) [(1)]

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

Initiating Pathways:

Shutdown Cooling Shutdown Isolation Valves
LPSI Cold Leg Discharge Check Valves
HPSI Cold Leg Discharge Check valves
HPSI Hot Leg Discharge Check Valves
Charging System Cold Leg Discharge Check Valves
Let Down Flow Control Valves
Check Valves in the suction line from the RWSP to the LPSI pumps
RCP Seal Cooling Heat Exchanger internal leakage

Mitigation Capability: Ensure Component Operability for Each Pathway:

Circle Affected Functions	Recovery of Failed Train	Remaining Mitigation Capability Rating for Each Affected Pathway	Sequence Color

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 interfacing system LOCA information was based on sections 3.1.2.9 page 3.1.34 in the original IPE. The IPE does nit provide the specific valve number and types for the ISLOCA paths. The RCP heat exchanger tube leakage is added for completeness.

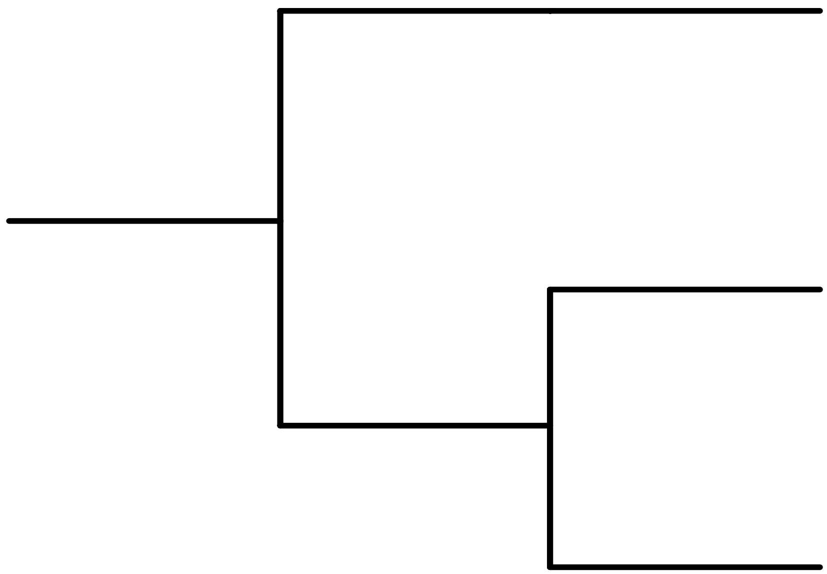
1.4 SDP EVENT TREES

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.

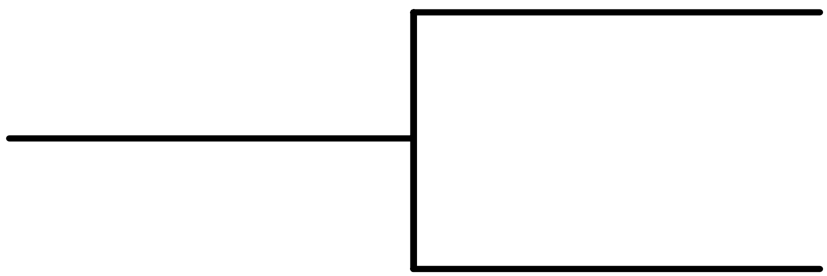
The following event trees are included:

(1) Transients (Reactor Trip) (TRANS)
(2) Transients without PCS (TPCS)
(3) Small LOCA (SLOCA)
(4) Stuck-open Safety Valve (SOSV)
(5) Medium LOCA (MLOCA)
(6) Large LOCA (LLOCA)
(7) Loss of Offsite Power (LOOP)
(8) Steam Generator Tube Rupture (SGTR)
(9) Anticipated Transients Without Scram (ATWS)
(10) Main Stream Line Break (MSLB)
(11) Loss of CCW (LCCW)
(12) LOOP with 1 EDG Available (LOOP1EDG)

TRANS	PCS	EFW	#	STATUS
			1	OK
			2	OK
			3	CD

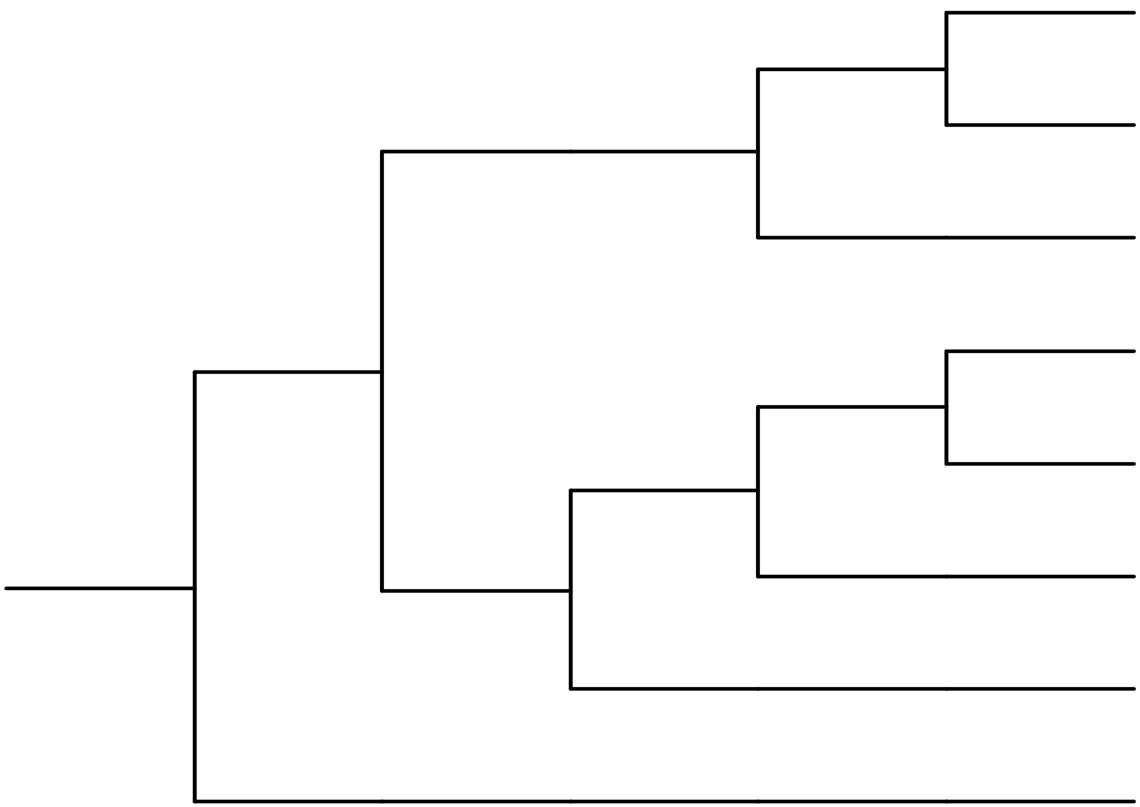


Plant Name Abbrev.: WTRF

TPCS	SHR	#	STATUS
		1	OK
		2	CD

Plant Name Abbrev.: WTRF

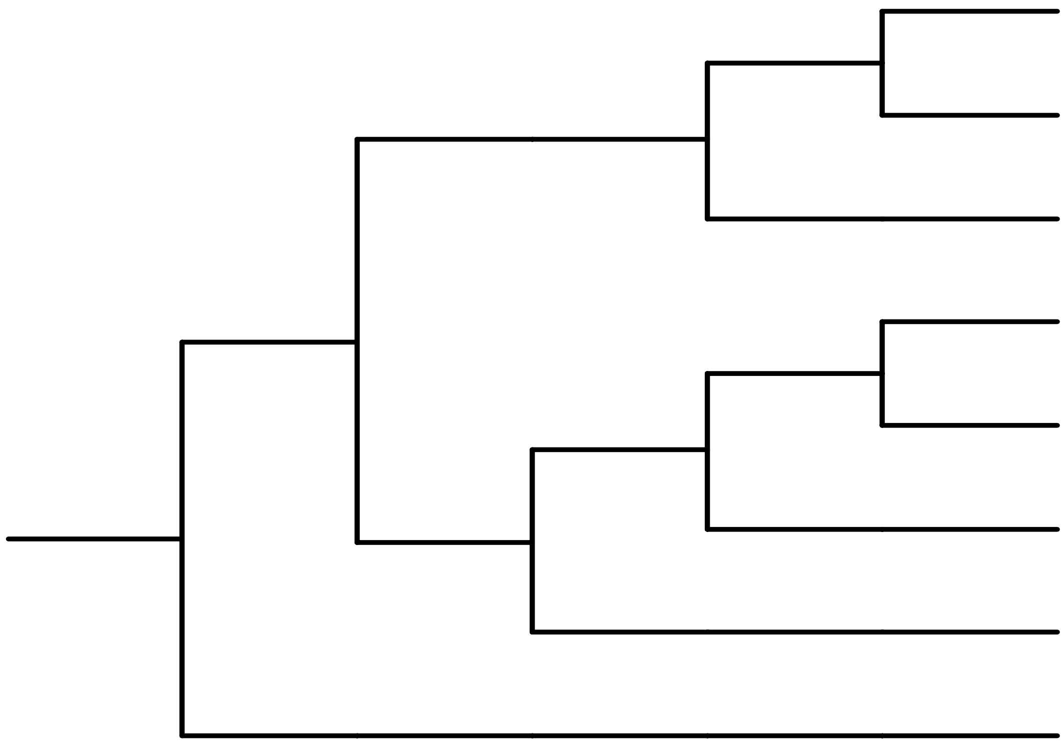
SLOCA	HPSI	PCS	EFW	HPR	CHR	#	STATUS



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

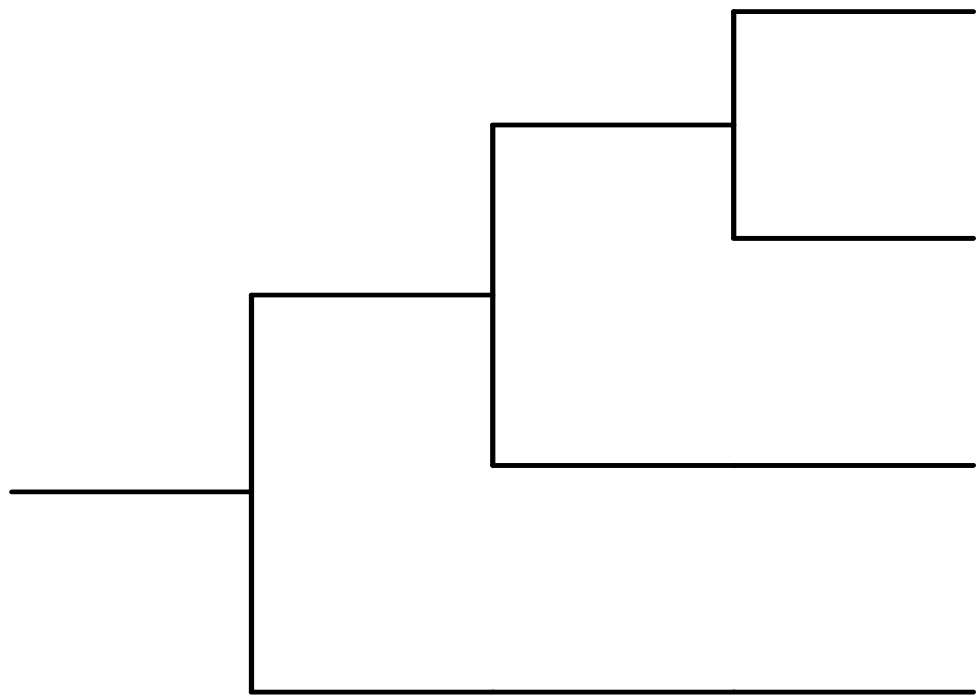
Plant Name Abbrev.: WTRF

SOSV	HPSI	PCS	EFW	HPR	CHR	#	STATUS
						1	OK
						2	CD
						3	CD
						4	OK
						5	CD
						6	CD
						7	CD
						8	CD



Plant Name Abbrev.: WTRF

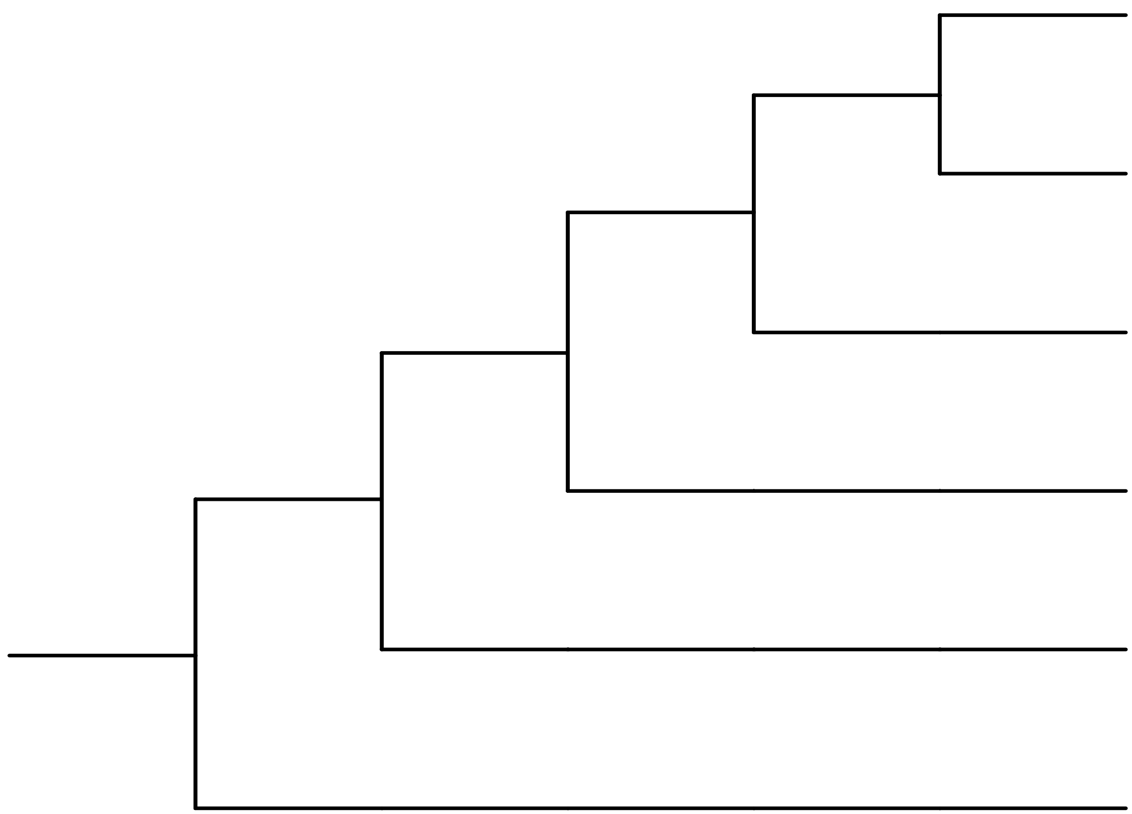
MLOCA	HPSI	HPR	CHR	#	STATUS



#	STATUS
1	OK
2	CD
3	CD
4	CD

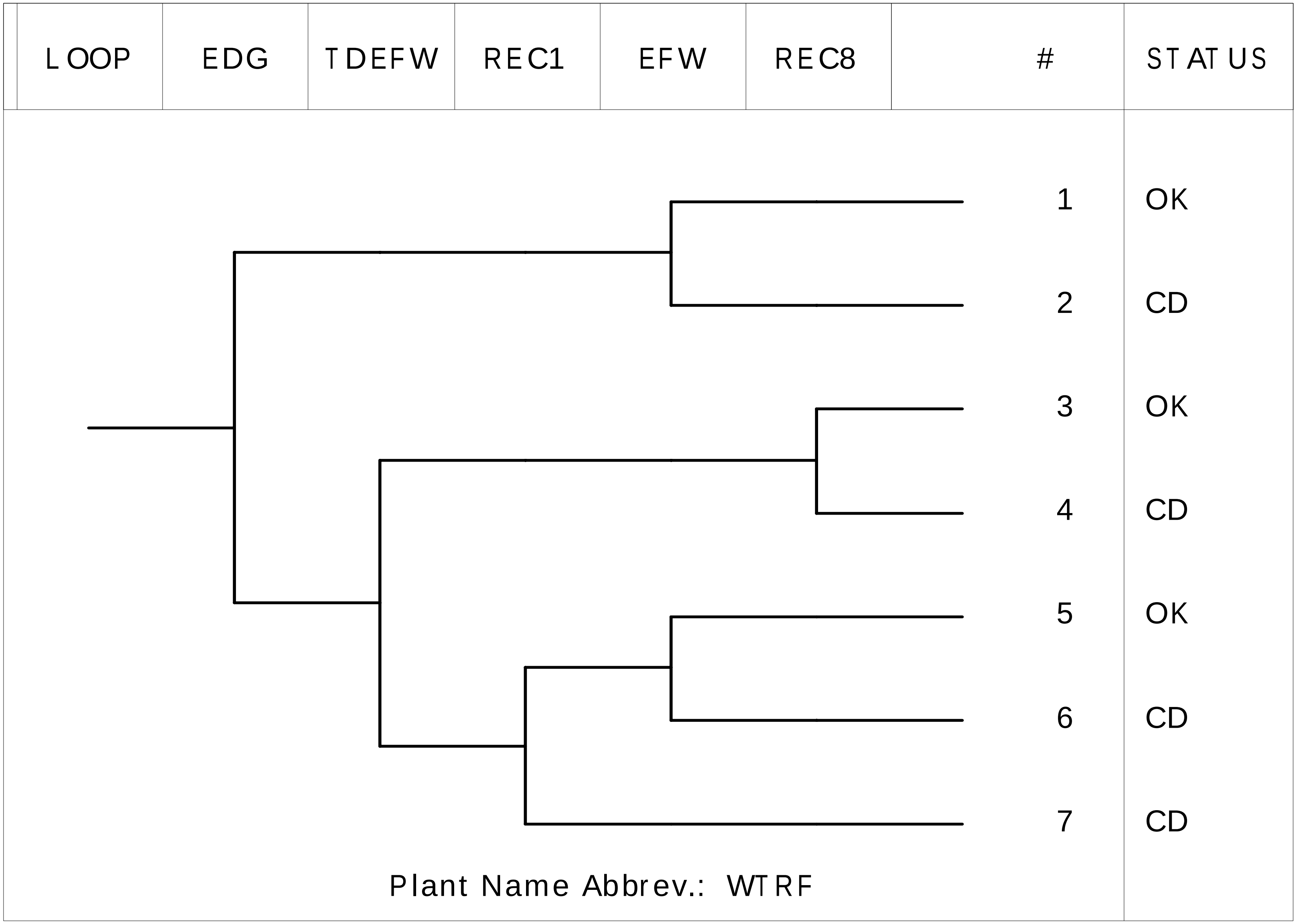
Plant Name Abbrev.: WTRF

LLOCA	SIT	HPSI	LPSI	HPR	CHR	#	STATUS

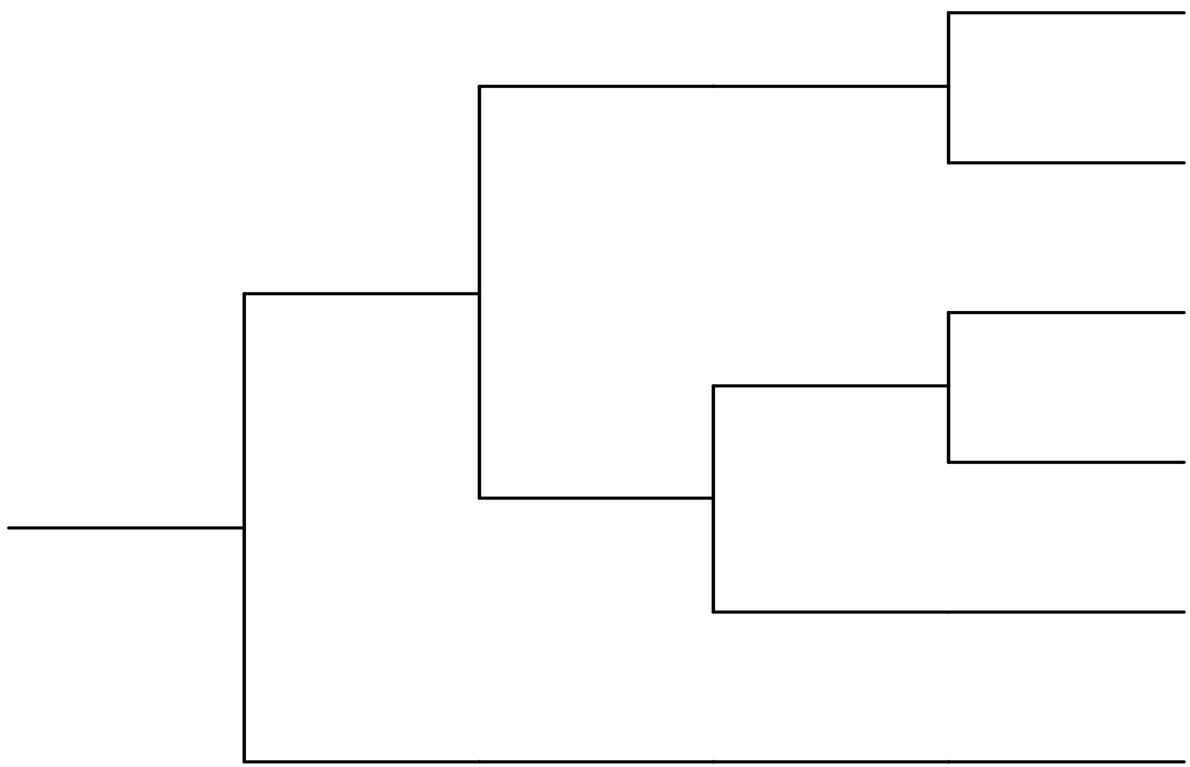


#	STATUS
1	OK
2	CD
3	CD
4	CD
5	CD
6	CD

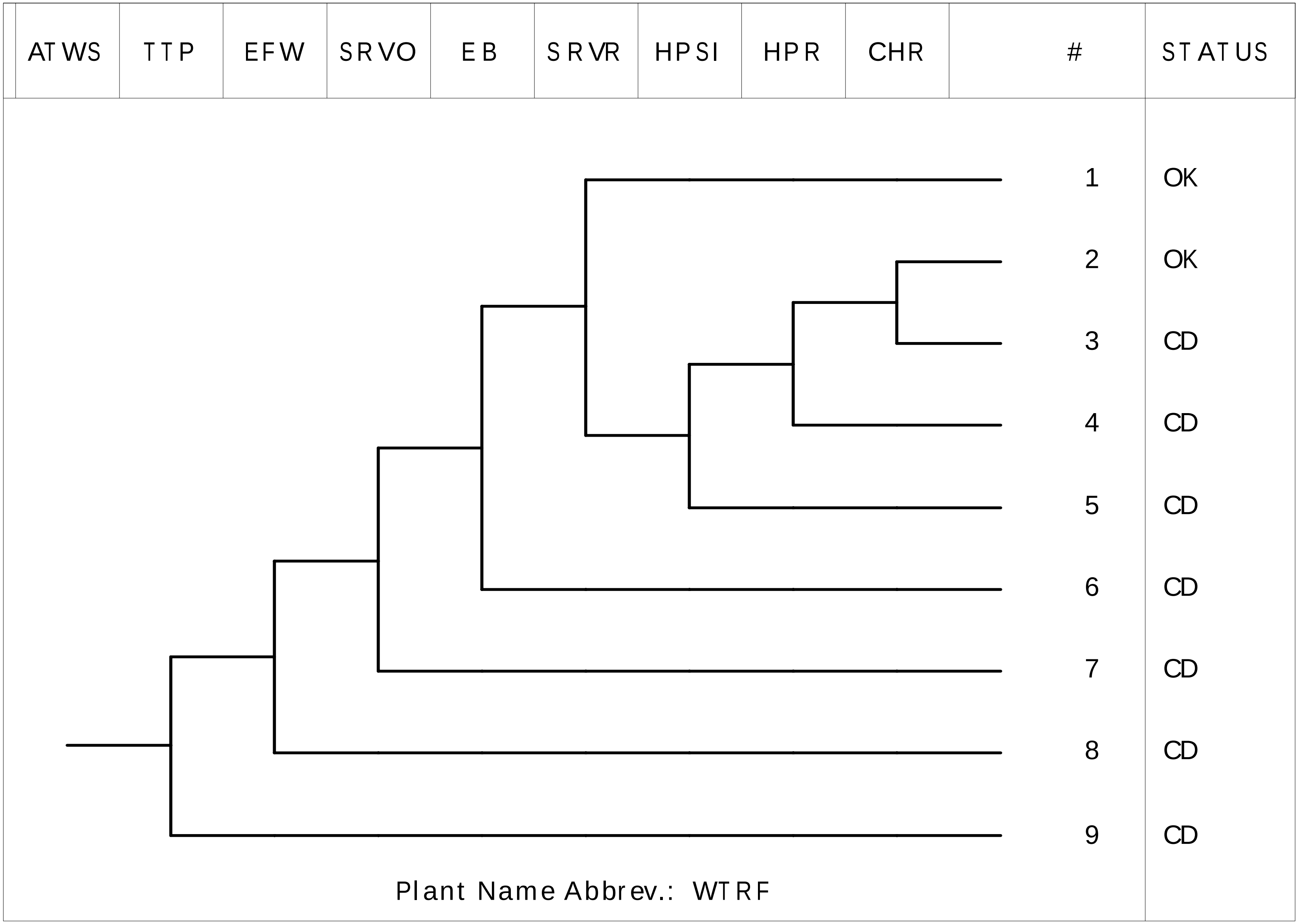
Plant Name Abbrev.: WTRF

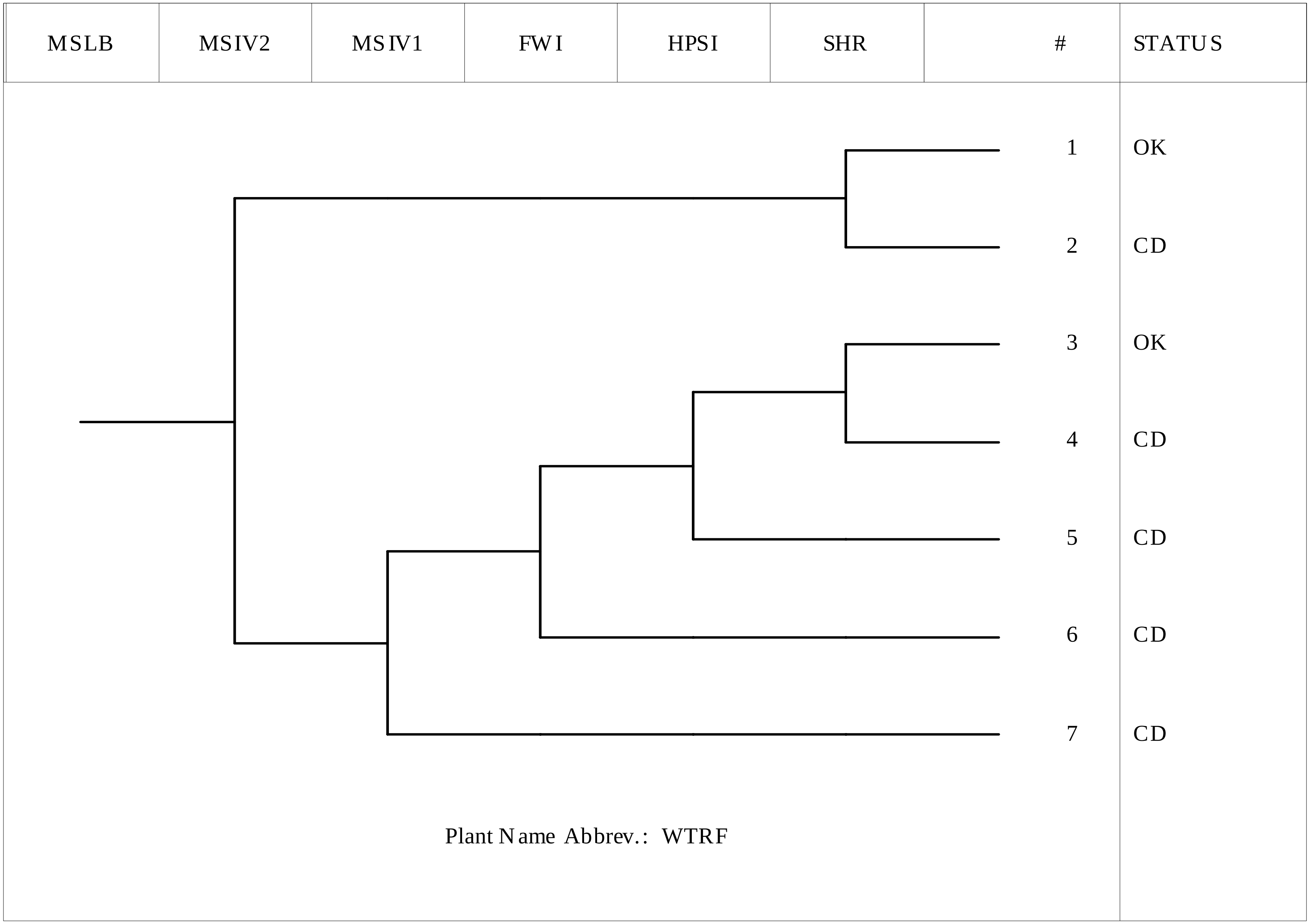

LOOP
EDG
TDEFW
REC1
EFW
REC8
#
STATUS
1 OK
2 CD
3 OK
4 CD
5 OK
6 CD
7 CD
Plant Name Abbrev.: WTRF

SGTR	EQ	PCS	EFW	RCSMU	#	STATUS
					1	OK
					2	CD
					3	OK
					4	CD
					5	CD
					6	CD



Plant Name Abbrev.: WTRF


ATWS
TTP
EFW
SRVO
EB
SRVR
HPSI
HPR
CHR
#
STATUS
1 OK
2 OK
3 CD
4 CD
5 CD
6 CD
7 CD
8 CD
9 CD
Plant Name Abbrev.: WTRF


MSLB
MSIV2
MSIV1
FWI
HPSI
SHR
#
STATUS
1 OK
2 CD
3 OK
4 CD
5 CD
6 CD
7 CD
Plant Name Abbrev.: WTRF

LCCW	RCPSEAL	PCS	EFW	#	STATUS
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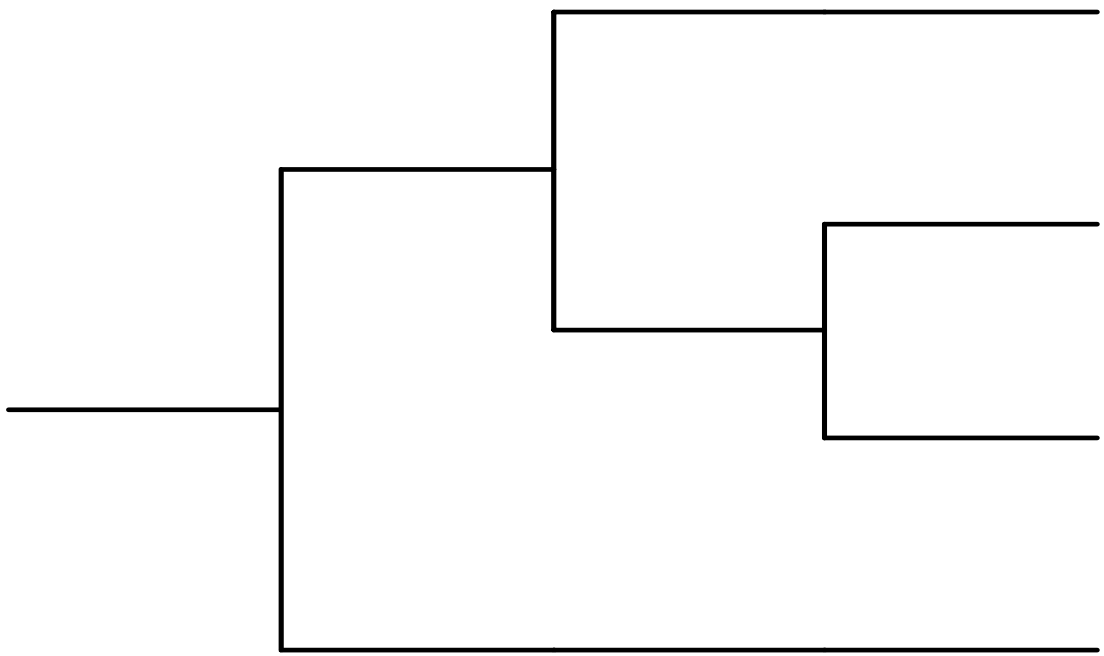
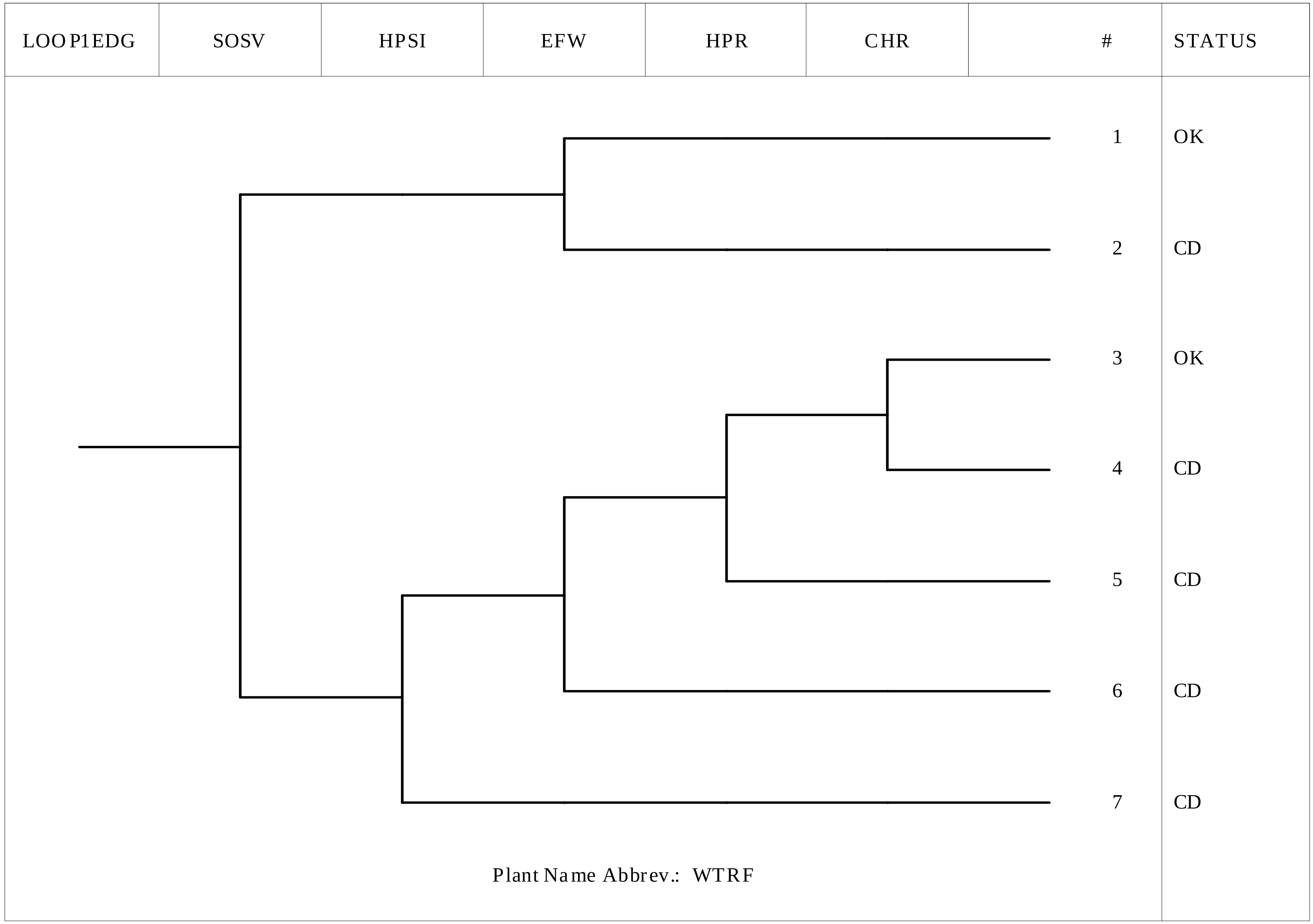
				2	OK
				3	CD
				4	CD



Plant Name Abbrev.: WTRF


LOOP1EDG
SOSV
HPSI
EFW
HPR
CHR
#
STATUS
1
OK
2
CD
3
OK
4
CD
5
CD
6
CD
7
CD
Plant Name Abbrev.: WTRF

2. RESOLUTION AND DISPOSITION OF COMMENTS

This section is composed of two subsections. Subsection 2.1 summarizes the generic assumptions that were used for developing the SDP worksheets for the PWR plants. These guidelines were based on the plant-specific comments provided by the licensee on the draft SDP worksheets and further examination of the applicability of those comments to similar plants. These assumptions which are used as guidelines for developing the SDP worksheets help the reader better understand the worksheets' scope and limitations. The generic guidelines and assumptions for PWRs are given here. Subsection 2.2 documents the plant-specific comments received on the draft version of the material included in this notebook and their resolution.

2.1 GENERIC GUIDELINES AND ASSUMPTIONS (PWRs)

The following generic guidelines and assumptions were used in developing the SDP worksheets for PWRs. These guidelines and assumptions were derived from a review of the licensee's comments, the resolutions of those comments, and the applicability to similar plants.

1. Assignment of plant-specific IEs into frequency rows:

Transient (Reactor trip) (TRANS), transients without PCS (TPCS), small, medium, and large LOCA (SLOCA, MLOCA, LLOCA), inadvertent or stuck-open PORV/SRV (SORV), main steam and feedwater line break (MSLB), anticipated transients without scram (ATWS), and interfacing system LOCAs (ISLOCA) are assigned into rows based on a consideration of the industry-average frequency. Plant-specific frequencies are considered for loss of offsite power (LOOP) and special initiators, and are assigned to the appropriate rows in Table 1.

2. Stuck open PORV/SRV as an IE in PWRs:

This event typically is not modeled in PRAs/IPEs as an initiating event. The failure of the PORVs/SRVs to re-close after opening is typically modeled within the transient event trees subsequent to the initiators. In addition, the intermittent failure or excessive leakage through PORVs as an initiator, albeit with much lower frequency, needed to be considered. To account for such failures and to keep the transient worksheets simple in the SDP, a separate worksheet for the SORV initiator was set up to explicitly model the contribution from such failures. This SDP worksheet, and the associated event tree, is similar to that of SLOCA. The frequency of PORV to re-close depends on the status of pressurizer. If the pressurizer is solid, then the frequency would be higher than the case in which the pressurizer level is maintained. Typically, this depends on early availability of secondary heat removal. However, the frequency for the SORV initiator is generically estimated for all PWR plants in Table 1.

3. Inclusion of special initiators:

The special initiators included in the worksheets are those applicable to this plant. A separate worksheet is included for each of them. The applicable special initiators are primarily based on the plant-specific IPEs/PRAs. In other words, the special initiators included are those modeled in the IPEs/PRAs unless shown to be negligible contributors. In some cases, a particular special initiator may be added for a plant even if it is not included in the IPE/PRA, if it is included in other plants of similar design, and is considered applicable for the plant. However, no attempt is made at this time to have a consistent set of special initiators across similarly designed plants. Except for the interfacing system LOCA (ISLOCA), if the occurrence of the special initiator results in a core damage, i.e., no mitigation capability exists for the initiating event, then a separate worksheet is not developed. For such cases, the inspection's focus is on the initiating event and the risk implication of the finding can be directly assessed. For ISLOCA, a separate worksheet is included noting the pathways that can lead to it.

4. Inclusion of systems under the support system column of the Initiators and System Dependency Table:

This Table shows the support systems for the support- and frontline systems. The intent is to include only the support systems, and not the systems supporting that support system, i.e., those systems whose failure will result in failure of the system being supported. Partial dependency, e.g., a backup system, is not included. If they are, this should be so noted. Sometimes, some subsystems on which inspection findings may be noted were included as a support system, e.g., the EDG fuel oil transfer pump as a support system for EDGs.

5. Coverage of system/components and functions included in the SDP worksheets:

The Initiators and System Dependency Table includes systems and components which are included in the SDP worksheets and those which can affect the performance of these systems and components. One-to-one matching of the event tree headings/functions to that included in the Table was not considered necessary.

6. Crediting of non-safety related equipment:

SDP worksheets credit or include safety-related equipment and also, non-safety related equipment, as used, in defining the accident sequences leading to core damage. In defining the success criteria for the functions needed, the components included are those covered under the Technical Specifications (TS) and the Maintenance Rule (MR). Credits for other components may have been removed in the SDP worksheets.

7. No credit for certain plant-specific mitigation capability:

The significance determination process (SDP) screens inspection findings for Phase 3 evaluations. Some conservative assumptions are made which result in not crediting some plant-specific features. Such assumptions are usually based on comparisons with plants of similar design, and they help to maintain consistency across the SDP worksheets for similar plant designs.

8. Crediting system trains with high unavailability:

Some system component/trains may have unavailability higher than 1E-2, but they are treated similarly to other trains with lower unavailability in the range of 1E-2. In this screening, this approach is considered adequate to keep the process simple. An exception is made for steam-driven components which are designated as Automatic Steam Driven (ASD) train with a credit of 1E-1.

9. Treating passive components (of high reliability) the same as active components:

Passive components, namely accumulators, are credited similarly to active components, even though they exhibit higher reliability. Considering the potential for common-cause failures, the reliability of a passive system is not expected to differ by more than an order of magnitude from active systems. Pipe failures were excluded, except as part of initiating events where the appropriate frequency is used. Accordingly, a separate designation for passive components was not considered necessary.

10. Crediting accumulators:

SDP worksheets assume the loss of the accumulator unit associated with the failed leg in LOCA scenarios. Accordingly, in defining the mitigation capability for the accumulators, the worksheets refer to the remaining accumulators. For example, in a plant with 4 accumulators with a success criteria of 1 out of 4, for large LOCA the mitigation capability is defined as 1/3 remaining accumulators (1 multi-train system), assuming the loss of the accumulator in the failed leg. For a plant with a success criteria of 2 out of 4 accumulators, the mitigation capability is defined as 2/3 remaining accumulators (1 multi-train system).

The inspection findings are then assessed as follows (using the example of the plant with 4 accumulators and success criteria of 2 out of 4):

4 Acc. Available	Credit=3
3 Acc. Available (1 Acc. is considered unavailable, based on inspection findings)	Credit=2
< 3 Acc. Available (2 or more Acc. are considered unavailable, Based on inspection findings)	Credit=0

11. Crediting operator actions:

The operator's actions modeled in the worksheets are categorized as follows: operator action=1 representing an error probability of 5E-2 to 0.5; operator action=2 representing an error probability of 5E-3 to 5E-2; operator action=3 representing an error probability of 5E-4 to 5E-3; and operator action=4 representing an error probability of 5E-5 to 5E-4. Actions with error probability > 0.5 are not credited. Thus, operator actions are associated with credits of 1, 2, 3, or 4. Since there is large variability in similar actions among different plants, a survey of the error probability across plants of similar design was used to categorize different operator actions. From this survey, similar actions across plants of similar design are assigned the same credit. If a plant uses a lower credit or recommends a lower credit for a particular action compared to our assessment of similar action based on plant survey, then the lower credit is assigned. An operator's action with a credit of 4, i.e., operator action=4, is noted at the bottom of the worksheet; the corresponding hardware failure, e.g., 1 multi-train system, is defined in the mitigating function.

12. Difference between plant-specific values and SDP designated credits for operator actions:

As noted, operator actions are assigned to a particular category based on a review of similar actions for plants with similar design. This results in some differences between plant-specific values and credit for the action in the worksheet. The plant-specific values are usually noted at the bottom of the worksheet.

13. Dependency among multiple operator actions:

IPEs or PRAs, in general, account for dependencies among the multiple operator actions that may be applicable. In the SDP screening approach, if multiple actions are involved in one function, then the credit for the function is designated as one operator action to the extent possible, considering the dependency involved.

14. Crediting the standby high-pressure pump:

The high-pressure injection system in some plants consists of three pumps with two of them auto-aligned and the third spare pump requiring manual action. The mitigating capability then is defined as : 1/2 HPI trains or use of a spare pump (1 multi-train system). Also, a footnote is added to reflect that the use of a spare pump could be given a credit of 1 (i.e.,1E-1) as a recovery action.

15. Emergency AC Power:

The full mitigating capability for emergency AC could include dedicated Emergency Diesel Generators (EDG), Swing EDG, SBO EDG, and finally, nearby fossil-power plants. The following guidelines are used in the SDP modeling of the Emergency AC power capability:

a) Describe the success criteria and the mitigation capability of dedicated EDGs.

b) Assign a mitigating capability of “operator action=1" for a swing EDG. The SDP worksheet assumes that the swing EDG is aligned to the other unit at the time of the LOOP (in a sense a dual unit LOOP is assumed). The operator, therefore, should trip, transfer, re-start, and load the swing EDG.

c) Assign a mitigating capability of “operator action=1" for an SBO EDG similar to the swing EDG. Note, some of the PWRs do not take credit for an SBO EDG for non-fire initiators. In these cases, credit is not given.

d) Do not credit the nearby power station as a backup to EDGs. The offsite power source from such a station could also be affected by the underlying cause for the LOOP. As an example, overhead cables connecting the station to the nuclear power plant also could have been damaged due to the bad weather which caused the LOOP. This level of detail should be left for a Phase 3 analysis.

16. Treatment of HPR and LPR:

The operation of both the HPR and LPR rely on the operation of the RHR pumps and the associated heat exchangers. Therefore, failure of LPR could imply failure of both HPR and LPR. A sequence which contains failure of both HPR and LPR as independent events will significantly underestimate the CDF contribution. To properly model this configuration within the SDP worksheets, the following procedure is used. Consider the successful depressurization and use of LPR as the preferred path. HPR is credited when depressurization has failed. In this manner, a sequence containing both HPR and LPR failures together is not generated.

17. SGTR event tree:

Event trees for SGTR vary from plant to plant depending on the size of primary-to-secondary leak, SG relief capacity, and the rate of rapid depressurization. However, there are several common functional steps that are addressed in the SDP worksheet: early isolation of the affected SG, initiation of primary cool-down and depressurization, and prevention of the SG overfill. These actions also include failure to maintain the secondary pressure below that of Main Steam safety valves which could occur either due to the failure of the relief valves to open or the operator's failure to follow the procedure. Failure to perform this task (sometimes referred to as early isolation and equalization) is assumed to cause continuous leakage of primary outside the containment. The success of this step implies the need for high-pressure makeup for a short period, followed by depressurization and cooldown for RHR entry (note, relief valves are assumed to re-close when primary pressure falls below that of the secondary). If the early makeup is not available or the operator fails to perform early isolation and equalization, rapid depressurization to RHR entry is usually assumed. This would typically require some kind of intermediate- or low-pressure makeup. Finally, depending on the size of the Refueling Water Storage Tank (RWST), sometimes it would be necessary to establish makeup to the RWST to allow sufficient time to enter the RHR mode.

18. ATWS scenarios:

The ATWS SDP worksheet assumes that these scenarios are not recoverable by operator actions, such as a manual trip. The failure of the scram system, therefore, is not recoverable, neither by the actuation of a back-up system nor through the actuation of manual scram. The initiator frequency, therefore, should only account for non-recoverable scrams, such as mechanical failure of the scram rods.

19. Recovery of losses of offsite power:

Recovery of losses of offsite power is assigned an operator-action category even though it is usually dominated by a recovery of offsite AC, independent of plant activities. Furthermore, the probability of recovery of offsite power in "X" hours (for example 4 hours) given it is not recovered earlier (for example, in the 1st hour) would be different from recovery in 4 hours with no condition. The SDP worksheet uses a simplified approach for treating recovery of AC by denoting it as an operator action=1 or 2 depending upon the HEP used in the IPE/PRA. A footnote highlighting the actual value used in the IPE/PRA is provided, when available.

20. RCP seal LOCA in a SBO:

The RCP seal LOCA in a SBO scenario is included in the LOOP worksheet. RCP seal LOCA resulting from loss of support functions is considered only if the loss of support function is a special initiator. The dependencies of RCP seal cooling are identified in Table 2.

21. RCP Seal LOCA for Westinghouse Plants during SBO Scenarios:

The modeling of the RCP seal failures upon loss of cooling and injection as occurs during SBO scenarios has been the subject of many studies (e.g., BNL Technical report W6211-08/99 and NUREG/CR-4906P). These studies are quite complex and assign probabilities of seal failure as a function of time (duration of SBO) and the associated leak rates. The leak rates, in turn, will determine what would be the safe period for recovery of the AC source and the use of SI pumps before core uncovery and damage. On the contrary, the SDP worksheets simplify the analysis of the RCP seal LOCA during the SBO scenarios using the following two assumptions: (1) The probability of catastrophic RCP seal failure is assumed to be 1 if the SBO lasts beyond two hours, and (2) Given a catastrophic seal LOCA, the available time prior to core damage for recovery of offsite power and establishing injection is about two hours. Therefore, in almost all cases, to prevent a core damage, a source of AC should be recovered within 4 hours in SBO scenarios.

22. Tripping the RCP on loss of CCW:

Upon loss of CCW, the motor cooling will be lost. The operation of RCPs without motor cooling could result in overheating and failure of bearings. Bearing failure, in turn, could cause the shaft to vibrate and thereby result in the potential for seal failure if the RCP is not tripped. In Westinghouse plants, the operator is instructed to trip the RCPs early in the scenario (from 2 to 10 minutes after detecting the loss of cooling). Failure to perform this action is conservatively assumed to result in seal failure and, potentially in a LOCA. This failure mechanism (occurrence of seal LOCA) due to failure to trip the RCPs upon loss of cooling is not considered likely in some plants, whereas it has been modeled explicitly in other plants. To ensure consistency, the trip of the RCP pumps are modeled in the SDP worksheets, and the operator failure to do this is assumed to result in a LOCA. In many cases, the failure to trip RCP following a loss of CCW results in core damage.

23. Hot leg/Cold leg switchover:

The hot leg to cold leg switchover during ECCS recirculation is typically done to avoid boron precipitation. This is typically part of the procedure for PWRs during medium and large LOCA scenarios. Some IPEs/PRAs do not consider the failure of this action as relevant to core damage. For plants needing the hot /cold switchover, it usually can only be accomplished with SI pumps and the ECCS recirculation also uses the SI pumps.

2.2 RESOLUTION OF PLANT-SPECIFIC COMMENTS

REFERENCES

1. NRC SECY-99-007A, Recommendations for Reactor Oversight Process Improvements (Follow-up to SECY-99-007), March 22, 1999.

2. Entergy Operations, Inc., "Waterford Nuclear Power Plant, Unit 3 Individual Plant Examination Submittal Report," June 1993.