March 30, 2001

Mr. Michael Kansler
Sr. Vice President and Chief
Operating Officer
Entergy Nuclear Operations, Inc.
440 Hamilton Avenue
White Plains, NY 10601

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

Dear Mr. Knubel:

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 and is also publically available through the Nuclear Regulatory Commission (NRC) external website at http://www.nrc.gov/NRC/IM/index.html.

The 1999 Pilot Plant review effort clearly indicated that significant site-specific design and risk information was not captured in the Phase 2 worksheets forwarded to you last spring. Subsequently, a site visit was conducted by the NRC 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 attached 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 forwarded to the Chief, Probabilistic Safety Assessment Branch, NRR. We will continue to assess SDP accuracy and update the document based on continuing experience.

While the attached 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.

We will coordinate our efforts through your licensing or risk organizations as appropriate. If you have any questions, please contact me at 301-415-1494.

Sincerely,

/RA/ Richard J. Laufer for

George F. Wunder, Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-286

Enclosure: As stated

cc w/encl: See next page

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RISK-INFORMED INSPECTION NOTEBOOK FOR

INDIAN POINT NUCLEAR POWER PLANT

UNIT 3

PWR, WESTINGHOUSE, FOUR-LOOP PLANT WITH LARGE DRY CONTAINMENT

Prepared by

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Office of Nuclear Regulatory Research
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Enclosure

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.

U. S. Nuclear Regulatory Commission
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ABSTRACT

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

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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TABLES

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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 Power Operated Relief Valve (SORV),
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 Indian Point 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 Indian Point 3 Nuclear Power Plant

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)	**B**	**C**	**D**
III	1 per 10^{2} - 10^{3} yr	SGTR, Stuck open PORV/SRV (SORV), Small LOCA including RCP seal failures (SLOCA), MSLB (outside containment), Loss of Non-essential Service Water (LNSW), Loss of 125 V DC Bus 31 or 32 (LBDC), Loss of 480 V AC Bus 5A or 6A (LB5A or 6A)	**C**	**D**	**E**
IV	1 per 10^{3} - 10^{4} yr	Medium LOCA (MLOCA), LOOP and Loss of 480 V AC Bus 5A (LA5A), LOOP and Loss of 480 V AC Bus 6A (LA6A)	**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], ISLOCA, Common Cause Failure of 31 and 32 DC Power Panels (CCDC)	**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).

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 Indian Point 3 Nuclear Power Plant

Affected Systems	Major Components	Support Systems	Initiating Event
Accumulator System	Four accumulator tanks	None	MLOCA, LLOCA
Auxiliary Boiler Feedpump Building Ventilation System (AFV)	Two fans and dampers	120 V-AC	All except MLOCA, LLOCA
ATWS Mitigation System Actuation Circuitry (AMSAC)	Not in IPE	125 V-DC	ATWS
Auxiliary Feedwater System (AFW)	Two MDPs	480 V-AC, 118 V-AC(1), 125 V-DC, IAS(2), AFV	All except MLOCA, LLOCA
	One TDP	125 V-DC, IAS(2), AFV	TRANS, TPCS, SLOCA, SORV, LOOP, SGTR, MSLB, LNSW, LBDC, LB5A, LB6A, LA5A, LA6A
Chemical and Volume Control System (CVCS)	Three charging pumps	480 V-AC, 125 V-DC, CCW, City water(3), Ventilation(4)	ATWS, LNSW
	Two boric acid transfer pumps	480 V-AC, 118 V-AC, 125 V-DC	ATWS
Component Cooling Water (CCW) System	Three CCW pumps(5)	480 V-AC, 125 V-DC, SAS, non-essential SWS	All except LBDC, CCDC
	Four auxiliary CCW (ACCW) pumps	480 V-AC, (main) CCW(6)	TRANS, TPCS, SLOCA, SORV, MLOCA, LLOCA, LOOP, SGTR, MSLB, LB5A, LB6A, LA5A, LA6A
Condensate System (CDS)	Three pumps	6.9 kV, IAS, Turbine Hall Closed Cooling system (THCC)(4)	TRANS

Table 2 (Continued)

Affected Systems	Major Components	Support Systems	Initiating Event
Containment Air Recirculation Cooling and Filtration (CFC) System	Five fan cooler units	480 V-AC, 125 V-DC, SAS, essential SWS	LNSW
Containment Spray System (CSS)	Two pumps	480 V-AC, 125 V-DC, SAS	LNSW
Control Building Heating and Ventilation System (CBV)	Two fans	480 V-AC, 120 V-AC	All
Electric Power System	Offsite System: switchyard and three gas turbines	480 V-AC, 125 V-DC, 118 V-AC	All
	6.9 kV System	Offsite System, 480 V-AC, 125 V-DC, Turbine Hall Closed Cooling system (THCC)[4]	All
	480 V-AC System	6.9 kV, 120 V-AC, 125 V-DC, CBV	All
	125 V-DC System	480 V-AC, CBV. Cooling: Switchgear Room Ventilation and Battery Room Ventilation	All
	EDG System: three EDGs and one Appendix R EDG	480 V-AC, 125 V-DC, Essential SWS, SAS, DGV, Fuel oil transfer pumps	LOOP, HVAC
	118 V-AC System	480 V-AC, 125 V-DC, CBV	All
	120 V-AC System	Not in IPE	All
EDG Building Ventilation System (DGV)	Two fans and dampers per EDG room	480 V-AC, 125 V-DC	LOOP

Table 2 (Continued)

Affected Systems	Major Components	Support Systems	Initiating Event
High-Head Safety Injection (HHSI) System	Three pumps. Minimum shut off head is 1500 psig	480 V-AC, 125 V-DC, SAS, CCW, Ventilation(4), Heat Tracing System (HTS)	All except ATWS, LBDC, CCDC
Instrument Air (IAS) System	Three compressors	480 V-AC, 125 V-DC, non-essential SWS	All except MLOCA, LLOCA
Main Steam System (MSS)	Per SG: One atmospheric dump valve (ADV), five safety relief valves, one MSIV, one MSIV bypass valve, and one SG blowdown isolation valve, 3 Turbine Bypass Valves (TBVs)	118 V-AC, 125 V-DC, IAS, SAS	All except MLOCA, LLOCA, ATWS
Primary Pressure Relief (PPR) System	Two PORVs(7), three SRVs, and two pressurizer spray valves	480 V-AC, 118 V-AC, 125 V-DC, IAS	TRANS, TPCS, SLOCA, SORV, LOOP, SGTR, ATWS, MSLB, LNSW, LB5A, LB6A, LA5A, LA6A
Reactor Coolant Pumps (RCP)	Seals	1 / 3 Charging pumps to seal injection or 1 / 3 CCW pumps to thermal barrier heat exchanger	SLOCA
Recirculation System	Two recirculation pumps	480 V-AC, 118 V-AC, 125 V-DC, ACCW, SAS	TRANS, TPCS, SLOCA, SORV, MLOCA, LLOCA, LOOP, SGTR, MSLB, LB5A, LB6A, LA5A, LA6A
Residual Heat Removal (RHR) System	2 subsystems, each with one RHR pump and heat exchanger. Pumps can deliver water to RCS when RCS pressure of 450 psig has been reached	480 V-AC, 118 V-AC, 125 V-DC, CCW, SAS, Ventilation(4)	All except ATWS, LBDC, CCDC

Table 2 (Continued)

Affected Systems	Major Components	Support Systems	Initiating Event
Safeguards Actuation System (SAS)	Transmitters and relays	118 V-AC, 125 V-DC, CBV[8]	All
Essential Service Water System (ESWS)	Three pumps supplying one header[9]	480 V-AC, 118 V-AC, 125 V-DC	LOOP
Non-essential Service Water System (NSWS)	Three pumps supplying one header		LNSW

Notes:

1. 118 V-AC provides motive power to transmitters, valve controllers, turbine speed control valve, and pressure control valves.
2. IAS provides motive power to flow control valves.
3. City water can be aligned as alternate cooling to charging pumps after a Loss of Non-essential Service Water (LNSW) (loss of CCW).
4. The licensee does not include this system in its PRA model.
5. Usually, two pumps are in operation, with the third pump in standby. The standby pump will start automatically on low supply header pressure (IPE, page 3-375).
6. The suction of the ACCW pumps is the discharge of the (main) CCW pumps.
7. During normal power operation, the PORV block valves are open. During an SBO, power to the PORV block valves is unavailable. Therefore, stuck open PORVs cannot be isolated (IPE, page 3-62).
8. The IPE (Table D27) identifies an additional room cooling system as “CR Vent”.
9. Three back-up service water pumps, normally aligned to the essential header, are available.
10. Plant internal event CDF (including internal flooding) = 4.4E-5/year.

1.3 SDP WORKSHEETS

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

1. Transients with PCS Available (TRANS) (Reactor Trip)
2. Transients with Loss of PCS (TPCS)
3. Small LOCA (SLOCA)
4. Stuck-open PORV (SORV)
5. Medium LOCA (MLOCA)
6. Large LOCA (LLOCA)
7. Loss of Off-site Power (LOOP)
8. Steam Generator Tube Rupture (SGTR)
9. Anticipated Transients Without Scram (ATWS)
10. Main Steam Line Break (MSLB)
11. Loss of Non-essential Service Water (LNSW)
12. Loss of 125 V DC Bus 31 (LBDC)
13. Loss of 125 V DC Bus 32 (LBDC)
14. Loss of 480-V AC Bus 5A (LB5A)
15. Loss of 480 V AC Bus 6A (LB6A)
16. LOOP and Loss of 480 V AC Bus 5A (LA5A)
17. LOOP and Loss of 480 V AC Bus 6A (LA6A)
18. Interfacing Systems LOCA (ISLOCA)

Table 3.1 SDP Worksheet for Indian Point 3 Nuclear Power Plant — Transients with PCS Available (TRANS) (Reactor Trip)

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

Safety Functions Needed:	**Full Creditable Mitigation Capability for Each Safety Function:**
Secondary Heat Removal (AFW)	1/2 MDAFW trains (1 multi-train system) or 1/1 TDAFW train (1 ASD train) with 1/4 ADVs or 1/5 main steam safety relief valves (per SG)
Condensate System (CDS)	Operator depressurizes 1/4 steam generators and aligns 1/3 condensate pumps (operator action = 2) [1] with 1/4 ADVs or 1/5 main steam safety relief valves (per SG)
Early Inventory, High Pressure Injection (EIHP)	1/2 HHSI trains (1/3 pumps) (1 multi-train system)
Primary Heat Removal, Feed/Bleed (FB)	2/2 PORVs open for Feed/Bleed (operator action = 1) [2]
High Pressure Recirculation (HPR)	1/2 HHSI trains (1/3 pumps) with (1/2 RHR pumps or 1/2 recirculation pumps) and with operator switchover from injection to recirculation (operator action = 3)

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

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

Notes:

1. Licensee states that the HEP for “Align condensate system for secondary cooling” (OCOND), is 6.6E-3. We did not credit the MFW because the licensee estimated the human error probability (HEP) for “Re-establish main feedwater given failure of AFW” (OMFW) as 1.0.

2. Licensee states that the HEP for initiating primary cooling bleed and feed given operator failure to align condensate is 8E-2.

Table 3.2 SDP Worksheet for Indian Point 3 Nuclear Power Plant — Transients with Loss of PCS (TPCS)

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>:
Secondary Heat Removal (AFW)	1/2 MDAFW trains (1 multi-train system) or 1/1 TDAFW train (1 ASD train) with 1/4 ADVs or 1/5 main steam safety relief valves (per SG)
Early Inventory, High Pressure Injection (EIHP)	1/2 HHSI trains (1/3 pumps) (1 multi-train system)
Primary Heat Removal, Feed/Bleed (FB)	2/2 PORVs open for Feed/Bleed (operator action = 2)
High Pressure Recirculation (HPR)	1/2 HHSI trains (1/3 pumps) with (1/2 RHR pumps or 1/2 recirculation pumps) and with operator switchover from injection to recirculation (operator action = 3)

<u>**Circle Affected Functions**</u>	<u>**Recovery of Failed Train**</u>	<u>**Remaining Mitigation Capability Rating for Each Affected Sequence**</u>	<u>**Sequence Color**</u>
1 TPCS - AFW - HPR (3)			
2 TPCS - AFW - FB (4)			
3 TPCS - AFW - EIHP (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.

Table 3.3 SDP Worksheet for Indian Point 3 Nuclear Power Plant — 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:**
Early Inventory, HP Injection (EIHP)	1/2 HHSI trains (1/3 pumps) (1 multi-train system)
Secondary Heat Removal (AFW)	1/2 MDAFW trains (1 multi-train system) or 1 TDAFW train (1 ASD train) with 1/4 ADVs or 1/5 main steam safety relief valves (per SG)
RCS Depressurization (DEPR)	Operator depressurizes RCS by using 2 / 4 SG ADVs or 2/2 PORVs (operator action = 3)
Primary Bleed (FB)	2/2 PORVs open for Feed/Bleed (operator action = 2)
Low Pressure Injection (LPI)	1/2 RHR pumps (1 multi-train system)
High Pressure Recirculation (HPR)	1/2 HHSI trains (1/3 pumps) with (1/2 RHR pumps or 1/2 recirculation pumps) with operator switchover from injection to recirculation (operator action = 3)
Low Pressure Recirculation (LPR)	1/2 RHR pumps or 1/2 recirculation pumps with operator switchover from injection to recirculation (operator action = 3)

Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 SLOCA - LPR (2, 9)			
2 SLOCA - DEPR - HPR (4)			
3 SLOCA - AFW - HPR (6)			
4 SLOCA - AFW - FB (7)			

5 SLOCA - EIHP - LPI (10)			
6 SLOCA - EIHP - DEPR (11)			
7 SLOCA - EIHP - AFW (12)			

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 3.4 SDP Worksheet for Indian Point 3 Nuclear Power Plant — Stuck Open PORV (SORV) [(1)]

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

Safety Functions Needed:	**Full Creditable Mitigation Capability for Each Safety Function:**
Isolation of Small LOCA (BLK)	The closure of the block valve associated with stuck open PORV (operator action = 3)
Early Inventory, HP Injection (EIHP)	1/2 HHSI trains (1/3 pumps) (1 multi-train system)
Secondary Heat Removal (AFW)	1/2 MDAFW trains (1 multi-train system) or 1 TDAFW train (1 ASD train) with 1/4 ADVs or 1/5 main steam safety relief valves (per SG)
RCS Depressurization (DEPR)	Operator depressurizes RCS by using 2/4 SG ADVs or 2/2 PORVs (operator action = 3)
Primary Bleed (FB)	1/1 remaining PORV open for Feed/Bleed (operator action = 2)
Low Pressure Injection (LPI)	1/2 RHR pumps (1 multi-train system)
High Pressure Recirculation (HPR)	1/2 HHSI trains (1/3 pumps) with (1/2 RHR pumps or 1/2 recirculation pumps) and with operator switchover from injection to recirculation (operator action = 3)
Low Pressure Recirculation (LPR)	(1/2 RHR pumps or 1/2 recirculation pumps) with operator switchover from injection to recirculation (operator action = 3)

Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 SORV - BLK - LPR (2, 9)			
2 SORV - BLK - DEPR - HPR (4)			
3 SORV - BLK - AFW - HPR (6)			
4 SORV - BLK - AFW - FB (7)			

5 SORV - BLK - EIHP - LPI (10)			
6 SORV - BLK - EIHP - DEPR (11)			
7 SORV - BLK - EIHP - AFW (12)			

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 sequences of the SDP SORV worksheet are the same as those of the SDP small LOCA event tree with the addition of the failure of the safety function "Isolation of Small LOCA (BLK)" after the initiating event, SORV.

Table 3.5 SDP Worksheet for Indian Point 3 Nuclear Power Plant — Medium LOCA (MLOCA)(1)

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

Safety Functions Needed:	**Full Creditable Mitigation Capability for Each Safety Function:**
Early Inventory, Accumulators (EIAC)	2/3 remaining accumulators (1 multi-train system)
Early Inventory, HP Injection (EIHP)	1/2 HHSI trains (1/3 pumps) (1 multi-train system)
Low Pressure Recirculation (LPR)	(1/2 RHR pumps or 1/2 recirculation pumps) with operator switchover from injection to recirculation (operator action = 3)

Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 MLOCA - LPR (2)			
2 MLOCA - EIHP (3)			
3 MLOCA - EIAC (4)			

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

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

Notes:

1. For the sequences with successful EIHP, secondary heat removal is not required since the break size is large enough to remove all the decay heat (information provided by licensee, Note (4) of Notes to Table N3-1, page N-34). If EIHP fails, it is necessary to depressurize the primary to use the low pressure pumps. However, the licensee models such depressurization as “Depressurize RCS for low-head injection during intermediate LOCA and failure of HHSI” (ODEP), with a human error probability = 1.0 (information provided by licensee, Table 3.3.3.2, Post-Accident Human Actions Quantification Summary, 5/4/00). Accordingly, we did not give credit to this action, and a failure of EIHP leads to core damage.

Table 3.6 SDP Worksheet for Indian Point 3 Nuclear Power Plant — 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:**
Early Inventory, Accumulators (EIAC)	2/3 remaining accumulators (1 multi-train system)
Early Inventory, HP Injection (EIHP)	1/2 HHSI trains (2/3 pumps) (1 multi-train system)
Early Inventory, LP Injection (EILP)	1/2 RHR pumps (1 multi-train system)
Low Pressure Recirculation (LPR)	(1/2 RHR pumps or 1/2 recirculation pumps) with operator switchover from injection to recirculation (operator action = 3)
Hot Leg Recirculation (HLR)	Operator switches to hot leg recirculation at 14 hours (1 train) (1)

Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 LLOCA - HLR (2, 5)			
2 LLOCA - LPR (3, 6)			
3 LLOCA - EIHP - EILP (7)			
4 LLOCA - EIAC (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.

Note:

1. Operator action = 4 limited by hardware failure.

Table 3.7 SDP Worksheet for Indian Point 3 Nuclear Power Plant — Loss of Off-site 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 AC Power (EAC)	2/3 Emergency Diesel Generators (1 multi-train system) [1]
Turbine-driven AFW Pump (TDAFW)	1/1 TDP trains of AFW (1 ASD train) with 1/4 ADVs or 1/5 main steam safety relief valves (per SG)
Secondary Heat Removal (AFW)	1/2 MDAFW trains (1 multi-train system) or 1/1 TDAFW train (1 ASD train) with 1/4 ADVs or 1/5 main steam safety relief valves (per SG)
Recovery of AC Power in < 2 hrs (REC2)	SBO procedures implemented (operator action = 1) [2]
Recovery of AC Power in < 5 hrs (REC5)	SBO procedures implemented (operator action = 2) [2, 3]
Early Inventory, HP Injection (EIHP)	1/2 HHSI trains (1/3 pumps) (1 multi-train system)
Primary Heat Removal (FB)	2/2 PORVs open for Feed/Bleed (operator action = 2)
High Pressure Recirculation (HPR)	(1/2 RHR pumps or 1/2 recirculation pumps) with operator switchover from injection to recirculation (operator action = 3)

Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 LOOP - AFW - HPR (3)			
2 LOOP - AFW - FB (4)			
3 LOOP - AFW - EIHP (5)			
4 LOOP - EAC - HPR (7, 11) (AC recovered)			

5 LOOP - EAC - EIHP (8, 13) (AC recovered)			
6 LOOP - EAC - REC5 (9)			
7 LOOP - EAC - TDAFW - FB (12) (AC recovered)			
8 LOOP - EAC - TDAFW - REC2 (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.

Notes:

1. IPE's unavailability of the EDGs is 1.91E-2 (IPE, page 3-464).

2. IPE's discussion of SBO scenarios gives mitigating credit to the Appendix R diesel for recovery of AC power (IPE, page 3-63).

3. In an SBO situation, an RCP seal LOCA may occur, with subsequent core damage at about 5 hours.

Table 3.8 SDP Worksheet for Indian Point 3 Nuclear Power Plant — 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:**
Early Inventory, HP Injection (EIHP)	1/2 HHSI trains (1/3 pumps) (1 multi-train system)
Secondary Heat Removal (SHR)	1/2 MDPs of AFW (1 multi-train system) or 1/1 TDP of AFW (1 ASD Train) with 1/4 ADVs or 1/5 main steam safety relief valves (per SG)
Feed-and-Bleed (FB)	2/2 PORVs open for Feed/Bleed (operator action = 2)
Pressure Equalization (EQ)	Operator isolates the ruptured SG and depressurizes RCS using RCS pressurizer PORV (1/2) or pressurizer spray to less than setpoint of relief valves of SG (operator action = 2)
High Pressure Recirculation (HPR)	(1/2 RHR pumps or 1/2 recirculation pumps) with operator switchover from injection to recirculation (operator action = 3)
RCS Depressurization (DEP)	Operator depressurizes RCS using 2 / 2 PORVs (operator action = 2)
Low Pressure Injection (RHRINJ)	1/2 RHR pumps in LHSI mode (1 multi-train system)
Long-Term RCS Makeup Source (MKRWST)	Operator refills RWST with primary make-up water (operator action = 1) [1]

Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 SGTR - EQ - MKRWST (3, 7)			
2 SGTR - SHR - HPR (5)			
3 SGTR - SHR - FB (8)			
4 SGTR - EIHP - RHRINJ (10)			
5 SGTR - EIHP - DEP (11)			

6 SGTR - EIHP - EQ (12)			
7 SGTR - EIHP - SHR (13)			
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. Operator refills RWST is modeled by the licensee with the human action “Refill RWST for continued core cooling during SGTR” (event WRWST), with a human error probability (HEP) = 8.4E-2.

Table 3.9 SDP Worksheet for Indian Point 3 Nuclear Power Plant — 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)	AMSAC trips the turbine (1 train)
Primary Relief (SRV)	3/3 SRVs with 2/2 PORVs open (1 train)
Secondary Heat Removal (AFW)	2/2 MDPs of AFW (1 train) [1, 2]
Emergency Boration (EMBO)	Operator conducts emergency boration using 1/3 charging pumps with 1/2 boric acid transfer pumps (operator action = 2)

Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 ATWS - EMBO (2)			
2 ATWS - AFW (3)			
3 ATWS - SRV (4)			
4 ATWS - TTP (5)			

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

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

<u>Notes:</u>

1. The licensee (Attachment # 6, Section N3, Level I Functional Level Success Criteria, page N-34) stated that because peak RCS pressure is anticipated within two minutes of the initiating event, no credit was taken for use of the steam-turbine-driven AFW pump. The use of this pump for auxiliary feedwater system secondary cooling requires manual operation of the AFW system flow control valves. The licensee considers that no credit can be given to this pump due to the relatively short time available for the manual operation.

2. The success criteria for the steam relief from the steam generators was not found in the IPE.

Table 3.10 SDP Worksheet for Indian Point 3 Nuclear Power Plant — Main Steam Line Break (MSLB)

Estimated frequency (Table 1 row) __________ **Exposure time** __________ **Table 1 result (circle): A B C D E F G H**

Safety Functions Needed:	**Full Creditable Mitigation Capability for each Safety Function:**
Secondary Heat Removal (AFW)	1/2 MDAFW trains (1 multi-train system) or 1 / 1 TDAFW train (1 ASD train) with 1/4 ADVs or 1/5 main steam safety relief valves (per remaining SG) (1)
Isolation of Break (ISOL)	3/4 Main Steam Isolation Valves close (1 multi-train system) or AFW flow to the SG whose MSIV does not close is isolated (operator action = 2)
Early Inventory, HP Injection (EIHP)	1/2 HHSI trains (1/3 pumps) (1 multi-train system)
Primary Heat Removal, Feed/Bleed (FB)	2/2 pressurizer PORVs open for Feed/Bleed (operator action = 2)
Operator stops HP Injection (STOPSI)	Operator stops HHSI (operator action = 1) (2)
High Pressure Recirculation (HPR)	(1/2 RHR pumps or 1/2 recirculation pumps) with operator switchover from injection to recirculation (operator action = 3)

Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 MSLBOC - ISOL - STOPSI (3)(3)			
2 MSLBOC - AFW - HPR (6)			
3 MSLBOC - AFW - FB (7)			
4 MSLBOC - AFW -EIHP (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. AFW supply to the faulted steam generator must be isolated.

2. The operator terminates high-head safety injection to avoid potential reactor vessel pressurized thermal shock failure (IPE, page 3-79). IPE uses the event "Terminate SI during steamline break given failure to isolate feedwater flow" (RV), with a human error probability = 8.5E-2 (IPE, page 3-451).

3. When the break is not isolated (function ISOL) the RCS will be overcooled, and there is a potential for reactor vessel pressurized thermal shock failure. In this case, operators have to terminate both high- and low-head safety injection (information provided by licensee, Note (14) of Notes to Table N3-1, page N-35).

Table 3.11 SDP Worksheet for Indian Point 3 Nuclear Power Plant — Loss of Non-essential Service Water (LNSW)[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:**
Trip RCPs (RCPT)	Operators trip the RCPs (operator action = 3)
Alternate Cooling to Charging Pumps (CWCH)	Operators trip the RCPs and align city water to charging pumps within 30 minutes for RCP seal injection (operator action = 1) [2]
Secondary Heat Removal (AFW)	1/2 MDAFW trains (1 multi-train system) or 1/1 TDAFW train (1 ASD train) with 1/5 main steam safety relief valves [3]
Early Inventory, High Pressure Injection (EIHP)	1/2 HHSI trains (1/3 pumps) (1 multi-train system) [4]
Primary Heat Removal, Feed/Bleed (FB)	2/2 PORVs open for Feed/Bleed (operator action = 1) [5]
High Pressure Recirculation (HPRC)	1/2 HHSI trains (1/3 pumps) and with operator aligning city water to RHR pump 31 and conducting the switchover from injection to recirculation (operator action = 1) [6]
Containment Heat Removal (COHR)	3 / 5 fan coolers with 1 / 2 CSS pumps[7] (1 multi-train system)

Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 LNSW - AFW - COHR (3)			
2 LNSW - AFW - HPRC (4)			
3 LNSW - AFW - FB (5)			
4 LNSW - AFW - EIHP (6)			
5 LNSW - CWCH (7)			

<table>
<tr><td>6 LNSW - RCPT (8)</td><td></td><td></td><td></td></tr>
<tr><td colspan="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.</td></tr>
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Notes:

1. Loss of non-essential SWS results in a loss of CCW and PCS. The CCW supports the high-head safety injection pumps, the residual heat removal pumps, residual heat removal heat exchangers, and charging pumps. Loss of CCW requires a manual reactor scram due to loss of cooling to the RCPs. Since the charging pumps are unavailable, and the cooling to the thermal barrier of the RCP from CCW is also lost, normal cooling to the RCP seals is lost. The operators should also trip the RCPs within 10 minutes after the loss of CCW to prevent RCP seal degradation. If the operators trip the RCPs, RCP seal LOCA could still occur unless cooling is restored to the charging pumps to re-establish RCP seal cooling. The IPE therefore credits aligning city water to charging pumps within 30 minutes for RCP seal injection as a viable option to avoid RCP seal LOCA. The frequency of loss of non-essential SWS is 2.11E-3 / year, and the frequency of loss of CCW is 3.98E-4 / year; accordingly, a loss of CCW is bounded by the former loss.

2. The licensee estimated a human error probability (HEP) = 9.2E-2 for aligning city water to charging pumps within 30 minutes.

3. Non-essential Service Water supports IAS, which in turn provides motive power to flow control valves. We assume that these valves fail open.

4. The plant has shaft-driven circulating pumps that use water from the CCW to cool the SI pump seals. Credit to these pumps during a loss of CCW depends on the cause of this loss. If the CCW pumps failed, then the circulating pumps are credited; if there was a rupture in the CCW system, then these pumps are not credited.

5. The licensee estimated a human error probability (HEP) = 0.12 for initiating primary cooling bleed and feed given operator failure to align backup city water cooling to the charging pumps.

6. The IPE states (page 3-146) that city water cooling can be aligned to RHR pump 31 when on recirculation cooling. The licensee's human error probability (HEP) is a function of whether the operators aligned city water to charging pumps. If they were successful, the HEP for this action is 8.7E-2. If the operators failed to align city water to charging pumps, the licensee's HEP for this action is 0.75. Since the SDP worksheets do not give credit to actions with a HEP larger than 0.5, and the RCP seal LOCA (resulting from the failure to align city water to charging pumps) will require recirculation, we assumed in the SDP event tree that the failure to align city water to charging pumps results in an RCP seal LOCA that cannot be mitigated.

7. Since RHR heat exchangers do not have CCW, energy is removed using systems that remove containment heat.

Table 3.12 SDP Worksheet for Indian Point 3 Nuclear Power Plant — Loss of 125 V DC Bus 31 (LBDC)(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:**		
Secondary Heat Removal (AFW)	1/2 MDAFW trains (1 multi-train system) with 1/4 ADVs or 1/5 main steam safety relief valves		
Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 LBDC - AFW (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.			

Notes:

1. IP3 has four DC power panels that provide control power to both safety and non-safety loads. However, only a loss of DC power panel 31 or 32 leads to a reactor scram and the simultaneous degradation of safety systems. Loss of 125 V DC bus 31 results in loss of DC distribution panels 31, 31A and 33 and control power to 480 V AC safeguard bus 5A. Reactor trip SWGR 52/RTB and 52/BYA and subsequent resultant reactor scram. The loads of bus 31 are: EDG 33, CS pump 31, CFC units 31 and 33, SI pump 31, recirculation pump 31, charging pump 31, component cooling pump 31, SW pumps 31, 34, and 37, AFW pump 32, and PORV PCV-456. The fast bus transfer of the 6.9 kV buses will not occur, and the EDGs power the 480 VAC 2A and 6A buses. Feed/Bleed requires 2 / 2 PORVs to open, but one is unavailable, so it is not possible to provide decay heat removal using Feed/Bleed. Hence, failure of AFW leads to core damage. Loss of DC power panel 31 has a frequency = 2.6E-3 / year. Loss of a 125 V DC power panel has the potential for a RCP seal LOCA, but the IPE (page 3-123) considers its probability to be negligible.

Table 3.13 SDP Worksheet for Indian Point 3 Nuclear Power Plant — Loss of 125 V DC Bus 32 (LBDC)[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:**		
Secondary Heat Removal (AFW)	1/1 MDAFW trains (1 train) or 1/1 TDAFW train (1 ASD train) with 1/4 ADVs or 1/5 main steam safety relief valves		
Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 LBDC - AFW (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.			

Notes:

1. IP3 has four DC power panels that provide control power to both safety and non-safety loads. However, only a loss of DC power panel 31 or 32 leads to a reactor scram and the simultaneous degradation of safety systems. Loss of 125 V DC bus 32 results in loss of DC distribution panels 32, 32A and 34 and control power to 480 V AC safeguard bus 6A. Reactor trip SWGR 52/RTA and 52/BYB and subsequent resultant reactor scram. The loads of bus 32 are: EDG 32, CS pump 32, CFC unit 35, SI pump 33, recirculation pump 32, charging pump 33, component cooling pump 33, SW pumps 33, 36, and 39, AFW pump 33, RHR pump 32, and PORV PCV-455C. The fast bus transfer of the 6.9 kV buses will not occur, and the EDGs power the 480 VAC 2A and 5A buses. Feed/Bleed requires 2 / 2 PORVs to open, but one is unavailable, so it is not possible to provide decay heat removal using Feed/Bleed. Hence, failure of AFW leads to core damage. Loss of DC power panel 32 has a frequency = 2.60E-3 / year. Loss of a 125 V DC power panel has the potential for a RCP seal LOCA, but the IPE (page 3-123) considers its probability to be negligible.

Table 3.14 SDP Worksheet for Indian Point 3 Nuclear Power Plant — Loss of 480 V AC Bus 5A (LB5A)[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:**
Secondary Heat Removal (AFW)	1/2 MDAFW train (1 train) or 1/1 TDAFW train (1 ASD train) with 1/4 ADVs or 1/5 main steam safety relief valves
Early Inventory, High Pressure Injection (EIHP)	1/2 HHSI trains (1/2 pumps) (1 multi-train system)
Primary Heat Removal, Feed/Bleed (FB)	2/2 PORVs open for Feed/Bleed (operator action = 2)
High Pressure Recirculation (HPR)	1/2 HHSI trains (1/2 pumps) with (1/2 RHR pumps or 1/1 recirculation pumps) and with operator switchover from injection to recirculation (operator action = 3)

Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 LB5A - AFW - HPR (3)			
2 LB5A - AFW - FB (4)			
3 LB5A - AFW - EIHP (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. IP3 has four 480 V AC safeguard buses that supply power to safety-related equipment. Loss of bus 5A will result in plant shutdown and the simultaneous degradation of safety systems. The loads of bus 5A are: EDG 33 auxiliaries, CS pump 31, CFC units 31 and 33, SI pump 31, recirculation pump 31, charging pump 31, component cooling pump 31, auxiliary component cooling pumps 31 and 33 (MCC-36A), SW pumps 31, 34, and 37, boric acid transfer pump 31 (MCC-36A), hydrogen recombiner 32 (MCC-36A), battery charger 31, and standby battery charger 35. Battery chargers 31 and 35 (spare) are powered from bus 5A, and hence, DC bus 31 may also be lost (loss of DC bus 31 is addressed as a separate initiating event in its corresponding worksheet). Loss of 480 V AC bus 5A has a frequency = 2.60E-3 / year. The structure of the event tree is the same as that for Transients with Loss of PCS (TPCS). Loss of a 480 V AC safeguard bus has the potential for a RCP seal LOCA, but the IPE (page 3-123) considers its probability to be negligible.

Table 3.15 SDP Worksheet for Indian Point 3 Nuclear Power Plant — Loss of 480 V AC Bus 6A (LB6A)[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:**
Secondary Heat Removal (AFW)	1/1 MDAFW trains (1 train) or 1/1 TDAFW train (1 ASD train) with 1/4 ADVs or 1/5 main steam safety relief valves
Early Inventory, High Pressure Injection (EIHP)	1/2 HHSI trains (1/2 pumps) (1 multi-train system)
Primary Heat Removal, Feed/Bleed (FB)	2/2 PORVs open for Feed/Bleed (operator action = 2)
High Pressure Recirculation (HPR)	1/2 HHSI trains (1/2 pumps) with (1/1 RHR pumps or 1/1 recirculation pumps) and with operator switchover from injection to recirculation (operator action = 3)

Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 LB6A - AFW - HPR (3)			
2 LB6A - AFW - FB (4)			
3 LB6A - AFW - EIHP (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. IP3 has four 480 V AC safeguard buses that supply power to safety-related equipment. Loss of bus 6A will result in plant shutdown and the simultaneous degradation of safety systems. The loads of bus 6A are: EDG 32 auxiliaries, CS pump 32, CFC unit 35, SI pump 33, recirculation pump 32, charging pump 33, component cooling pump 33, AFW pump 33, RHR pump 32, auxiliary component cooling pumps 32 and 34 (MCC-36B), SW pumps 33, 36, and 39, boric acid transfer pump 32 (MCC-36B), hydrogen recombiner 32 (MCC-36B), motor-operated valve motive power from MCC-36B and MCC-37, and battery charger 32. Loss of 480 V AC bus 6A has a frequency = 2.60E-3 / year. The structure of the event tree is the same as that for Transients with Loss of PCS (TPCS). Loss of a 480 V AC safeguard bus has the potential for a RCP seal LOCA, but the IPE (page 3-123) considers its probability to be negligible.

Table 3.16 SDP Worksheet for Indian Point 3 Nuclear Power Plant — LOOP and Loss of 480 V AC Bus 5A (LA5A)(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:**
PORV Recloses (PORV)	2/2 Pressurizer PORVs reclose after opening during transient (2) (1 train)
Secondary Heat Removal (AFW)	1/2 MDAFW trains (1 multi-train system) or 1/1 TDAFW train (1 ASD train) with 1/4 ADVs or 1/5 main steam safety relief valves
Early Inventory, HP Injection (EIHP)	1/2 HHSI trains (1/2 pumps) (1 multi-train system)
Primary Heat Removal (FB)	1/1 remaining PORV open for Feed/Bleed (operator action = 2)
High Pressure Recirculation (HPR)	1/2 HHSI trains (1/2 pumps) with (1/2 RHR pumps or 1/1 recirculation pumps) and with operator switchover from injection to recirculation (operator action = 3)
RCS Depressurization (DEPR)	Operator depressurizes RCS by using 2/4 SG ADVs or 1/1 remaining PORV (operator action = 2)
Low Pressure Injection (LPI)	1/2 RHR pumps (1 multi-train system)
Low Pressure Recirculation (LPR)	(1/2 RHR pumps or 1/1 recirculation pumps) with operator switchover from injection to recirculation (operator action = 3)

Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 LA5A - PORV - LPR (3, 7)			
2 LA5A - PORV - DEPR - HPR (5)			
3 LA5A - PORV - EIHP - LPI (8)			

4 LA5A - PORV - EIHP - DEPR (9)			
5 LA5A - PORV - AFW - HPR (11)			
6 LA5A - PORV - AFW - FB (12)			
7 LA5A - PORV - AFW - EIHP (13)			
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. See the SDP Worksheet on Loss of 480 V AC bus 5A for details on the impact of this loss. The event tree for this loss has the initiating event LOAC.

2. The PORVs are powered from 125 VDC. The motor-operated block valves are powered from 480 VAC (IPE, page 3-356). Hence, if an 480 V AC bus is lost, there may not be motive power available to close the block valve of a stuck open PORV.

Table 3.17 SDP Worksheet for Indian Point 3 Nuclear Power Plant — LOOP and Loss of 480 V AC Bus 6A (LA6A)[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:**
PORV Recloses (PORV)	2/2 Pressurizer PORVs reclose after opening during transient [2] (1 train)
Secondary Heat Removal (AFW)	1/1 MDAFW trains (1 multi-train system) or 1/1 TDAFW train (1 ASD train) with 1/4 ADVs or 1/5 main steam safety relief valves
Early Inventory, HP Injection (EIHP)	1/2 HHSI trains (1/2 pumps) (1 multi-train system)
Primary Heat Removal (FB)	1/1 remaining PORV open for Feed/Bleed (operator action = 2)
High Pressure Recirculation (HPR)	(1/1 RHR pumps or 1/1 recirculation pumps) with operator switchover from injection to recirculation (operator action = 3)
RCS Depressurization (DEPR)	Operator depressurizes RCS by using 2/4 SG ADVs or 1/1 remaining PORV (operator action = 2)
Low Pressure Injection (LPI)	1/1 RHR pumps (1 multi-train system)
Low Pressure Recirculation (LPR)	(1/1 RHR pumps or 1/1 recirculation pumps) with operator switchover from injection to recirculation (operator action = 3)

Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**Sequence Color**
1 LA6A - PORV - LPR (3, 7)			
2 LA6A - PORV - DEPR - HPR (5)			
3 LA6A - PORV - EIHP - LPI (8)			

4 LA6A - PORV - EIHP - DEPR (9)			
5 LA6A - PORV - AFW - HPR (11)			
6 LA6A - PORV - AFW - FB (12)			
7 LA6A - PORV - AFW - EIHP (13)			
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. See the SDP Worksheet on Loss of 480 V AC bus 6A for details on the impact of this loss. The event tree for this loss has the initiating event LOAC.

2. The PORVs are powered from 125 VDC. The motor-operated block valves are powered from 480 VAC (IPE, page 3-356). Hence, if an 480 V AC bus is lost, there may not be motive power available to close the block valve of a stuck open PORV.

Table 3.18 SDP Worksheet for Indian Point 3 Nuclear Power Plant — Interfacing Systems LOCA (ISLOCA)(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:**
CVCS Normal Letdown Line 27	Relief valve CH-203 fails to open after the spurious closure of containment isolation valves CH-201 or CH-202, or valve PCV-135.
CVCS Excess Letdown Line 98	Spurious opening of normally closed valves CH-213A/B and HCV-123.
CCW (RHR Heat Exchanger) Lines 52A and 52	ISLOCA in line 52A upstream of valve AC-822B can occur upon failure of the tube sides of heat exchanger 32. An ISLOCA in line 52A downstream of valve AC-822B can occur upon failure of the tube sides of heat exchanger 32 subsequent to the failure of one of the following sets of valves: RCS cold leg loop 1 check valves SI-897A and SI-838A; RCS cold leg loop 2 check valves SI-897B and SI-838B; RCS cold leg loop 3 check valves SI-897C and SI-838C; or RCS cold leg loop 4 check valves SI-897D and SI-838D. There are similar possibilities of ISLOCA in line 52, which is connected to heat exchanger 31.
CCW Non-Regenerative Heat Exchanger Line 149	ISLOCA in line 149 upon tube failure of the non-regenerative heat exchanger.
CCW Excess Letdown Heat Exchanger Line 18	ISLOCA in line 18 upon tube failure of the excess letdown heat exchanger.
CCW Sample Heat Exchanger Line 166	CCW return line 166 from the pressurizer liquid and reactor coolant heat exchangers can be overpressured after heat exchanger tube failure. However, because the primary sampling lines have a diameter of 3/4 in, the expected break flow is within the capacity of the normally-operating charging pumps. Therefore, the core is unlikely to be uncovered.
SIS Line 56	An ISLOCA in line 56 can occur upon failure of one of the following sets of valves: RCS cold leg loop 1 check valves SI-897A, SI-857A, and SI-857G; RCS cold leg loop 2 check valves SI-897B, SI-857S, and SI-857T; RCS cold leg loop 3 check valves SI-897C, SI-857Q, and SI-857R; RCS cold leg loop 4 check valves SI-897D, SI-857U, and SI-857W; or RCS hot leg loop 3 check valves SI-857B, SI-857H, and normally-closed motor-operated valve SI-856B. An ISLOCA inside the SI pump room in the PAB at the 34 ft elevation is possible if check valve SI-849A or SI-852A also fails. Overpressure of line 56 and failure of check valve SI-858B diverts high-head injection flow away from the reactor and into the containment through line 16.

SIS Line 16	An ISLOCA in line 16 can occur upon failure of one of the following sets of valves: RCS cold leg loop 1 check valves SI-857E, and SI-857L; RCS cold leg loop 2 check valves SI-857D, and SI-857K; RCS cold leg loop 3 check valves SI-857F, and SI-857M; RCS cold leg loop 4 check valves SI-857C, and SI-857J; or RCS hot leg loop 3 check valves SI-857N, SI-857P, and motor-operated valve SI-856B. An ISLOCA inside the SI pump room in the PAB at the 34 ft elevation is possible if check valve SI-849A or SI-852A also fails. Overpressure of line 16 and failure of check valve SI-858A diverts high-head injection flow away from the reactor and into the containment through line 56.
SIS Line 60/29	An ISLOCA upstream of motor-operated valves (MOV) SI-888A/B in line 60 can occur upon failure of one of the following sets of valves: RCS cold leg loop 1 check valves SI-897A, and SI-838A; RCS cold leg loop 2 check valves SI-897B, and SI-838B; RCS cold leg loop 3 check valves SI-897C, and SI-838C; or RCS cold leg loop 4 check valves SI-897D, and SI-838D. The failure of any of these sets and inadvertent opening of CH-133 also pressurizes line 29. If MOV SI-888A or SI-888B is open, low pressure piping downstream of them and upstream of, or at, the safety injection pumps may fail. An ISLOCA can also occur upon rupture of line 29.
RHR Line 9	An ISLOCA in line 9 can occur upon failure of one of the following sets of valves: RCS cold leg loop 1 check valves SI-897A, and SI-838A; RCS cold leg loop 2 check valves SI-897B, and SI-838B; RCS cold leg loop 3 check valves SI-897C, and SI-838C; or RCS cold leg loop 4 check valves SI-897D, and SI-838D.
RHR Lines 10, 57 and 155	Overpressurization of RHR lines can occur if reactor coolant leaks past RHR shutdown cooling isolation valves AC-731, AC-730 and AC-732. If the first two valves become fully open, and lines 10, 57 or 155 fail, massive flooding in the PAB may result from the release of RCS and RWST inventories.
CCW (RHR) Lines 336 and 658	RCS overpressure of RHR line 10[2] results in RHR seal failure and pressurization of CCW return lines 336 and 658.
RHR Minimum Flow Lines 337, 3042 and 3043; RHR Lines 60, 89, 93, 94, 293, 355 and 358	An ISLOCA in these lines can occur upon failure of one of the following pairs of low-head injection isolation check valves: RCS cold leg loop 1 check valves SI-897A, and SI-838A; RCS cold leg loop 2 check valves SI-897B, and SI-838B; RCS cold leg loop 3 check valves SI-897C, and SI-838C; or RCS cold leg loop 4 check valves SI-897D, and SI-838D.

Circle Affected Functions	**Recovery of Failed Train**	**Remaining Mitigation Capability Rating for Each Affected Sequence**	**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.

<u>Notes</u>:

1. This worksheet is different from the other worksheets in that ISLOCA is typically an unmitigated initiating event in most PRAs. Therefore, the right side of the worksheet contains paths which may lead to an ISLOCA rather than mitigating systems to address an event in progress. As such, it is not intended to be referenced by the last column of Table 2, Initiators and System Dependency Table.

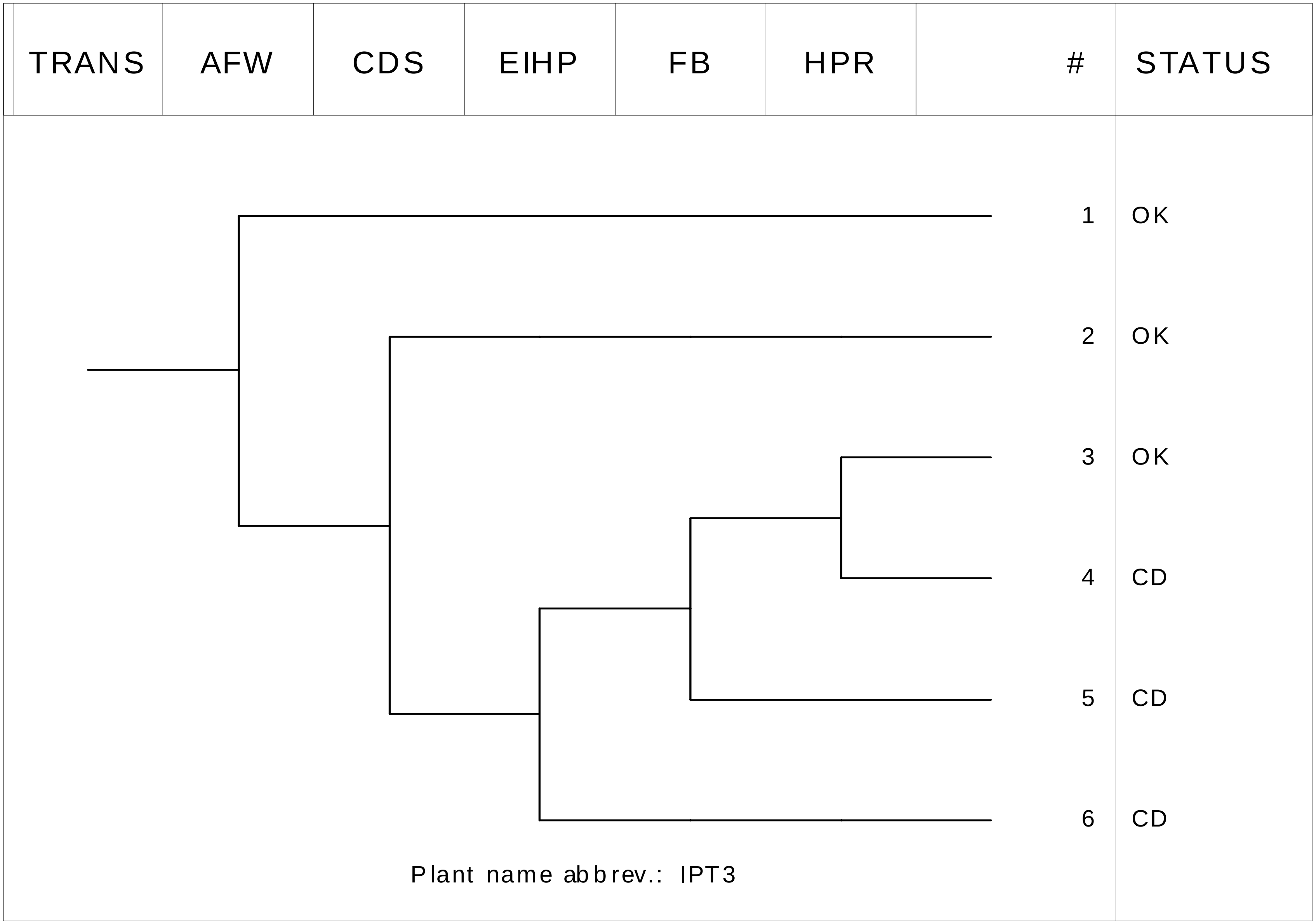
2. The IPE (page 3-185) does not state the cause of the overpressure of RHR line 10. The possible causes appear to be those mentioned in the row “RHR Lines 10, 57 and 155", above.

1.4 SDP EVENT TREES

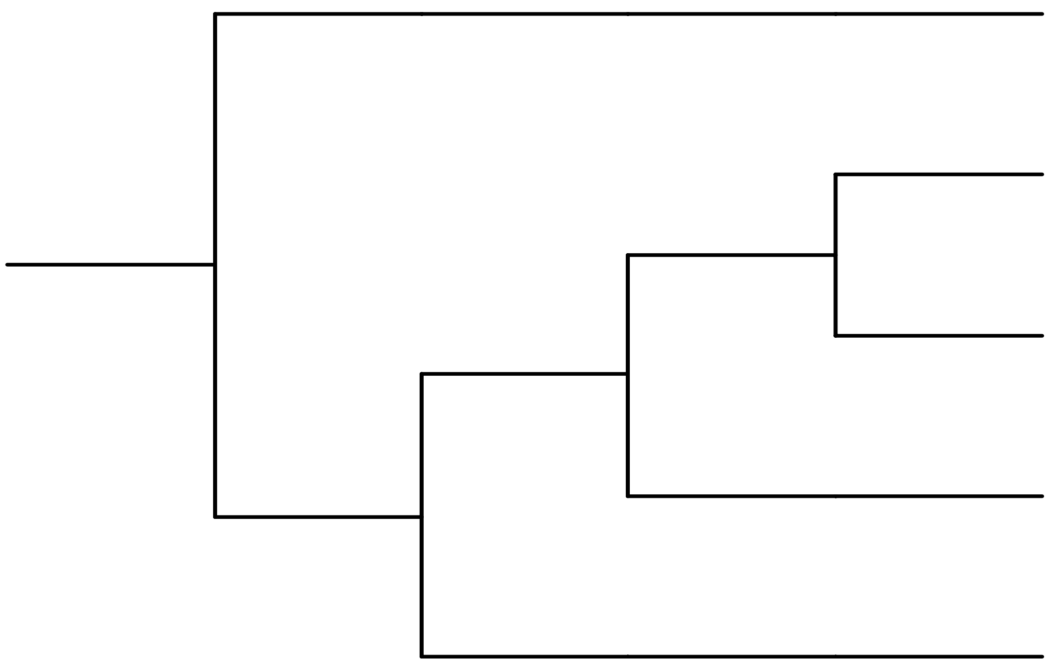
This section provides the simplified event trees called SDP event trees used to define the accident sequences identified in the SDP worksheets in the previous section. An event tree for the stuck-open PORV is not included since it is similar to the small LOCA event tree. The event tree headings are defined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

The following event trees are included:

1. Transients with PCS Available (TRANS) (Reactor Trip)
2. Transients with Loss of PCS (TPCS)
3. Small LOCA (SLOCA)
4. Medium LOCA (MLOCA)
5. Large LOCA (LLOCA)
6. Loss of Off-site Power (LOOP)
7. Steam Generator Tube Rupture (SGTR)
8. Anticipated Transients Without Scram (ATWS)
9. Main Steam Line Break (MSLB)
10. Loss of Non-essential Service Water (LNSW)
11. Loss of One 125 V DC Bus (LBDC)
12. LOOP and Loss of 480 V AC Bus (LOAC)


TRANS
AFW
CDS
EIHP
FB
HPR
#
STATUS
1 OK
2 OK
3 OK
4 CD
5 CD
6 CD
Plant name abbrev.: IPT3

TPCS	AFW	EIHP	FB	HPR	#	STATUS



#	STATUS
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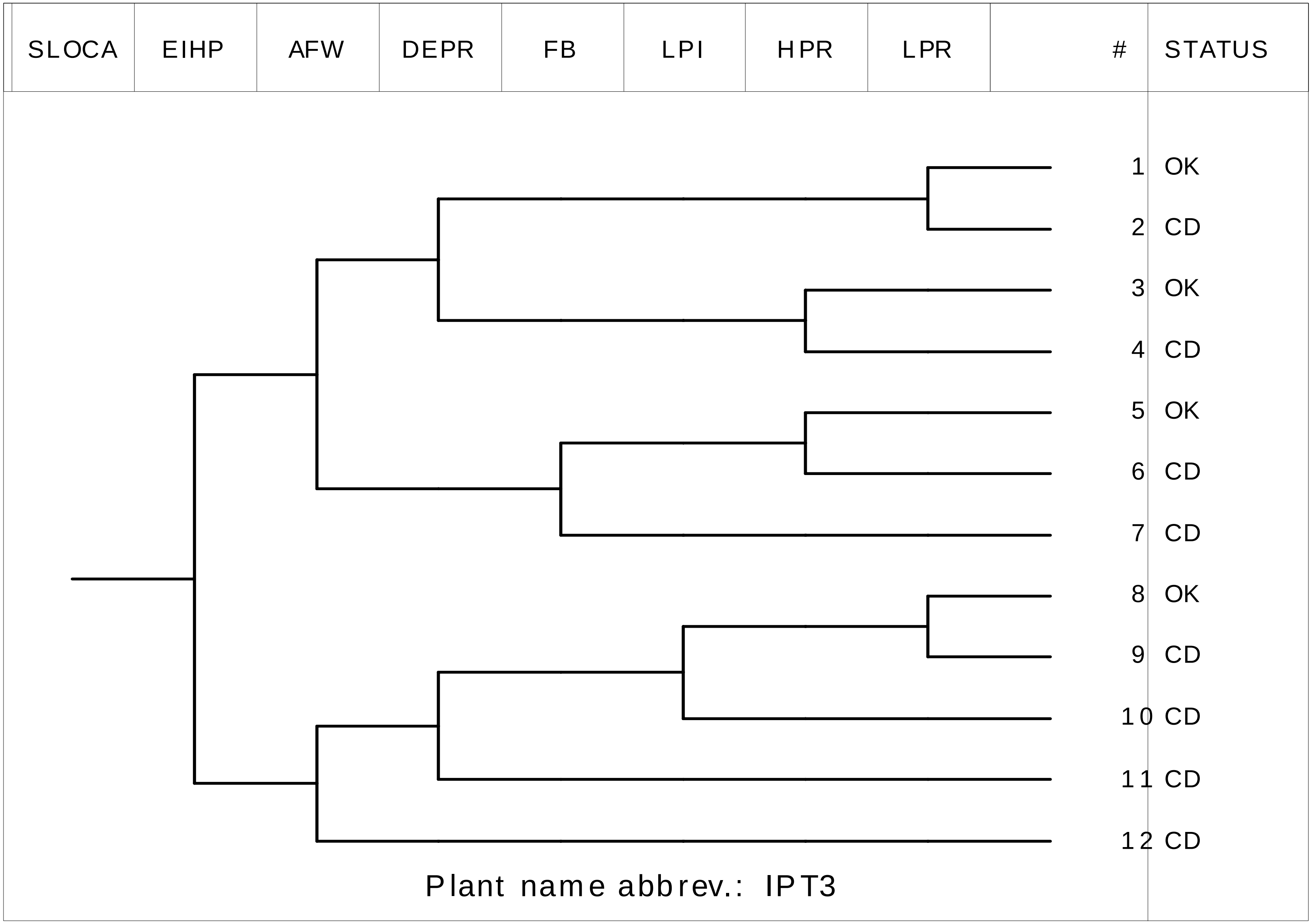
2	OK
3	CD
4	CD
5	CD

Plant name abbrev.: IPT3


SLOCA
EIHP
AFW
DEPR
FB
LPI
HPR
LPR
#
STATUS
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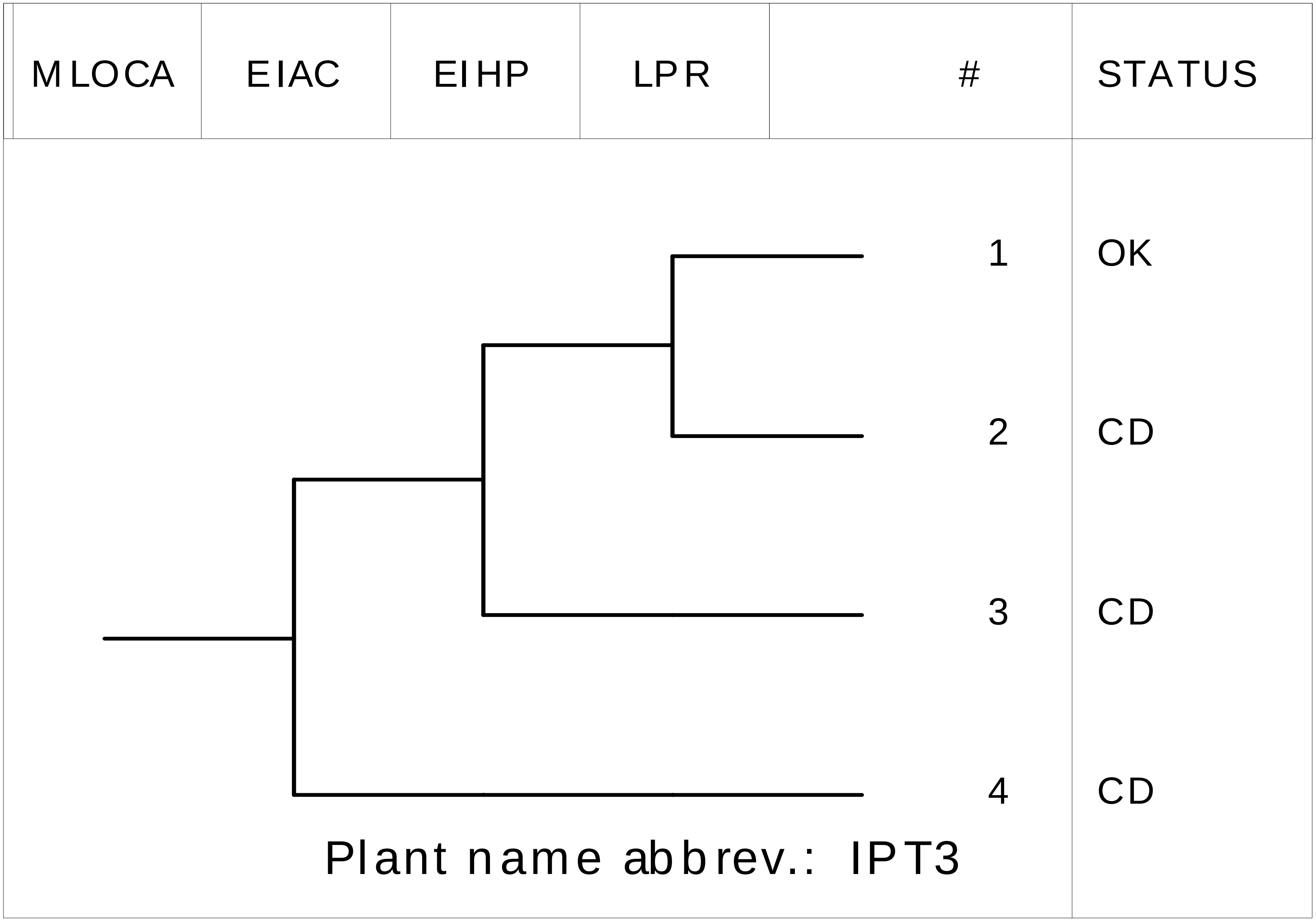
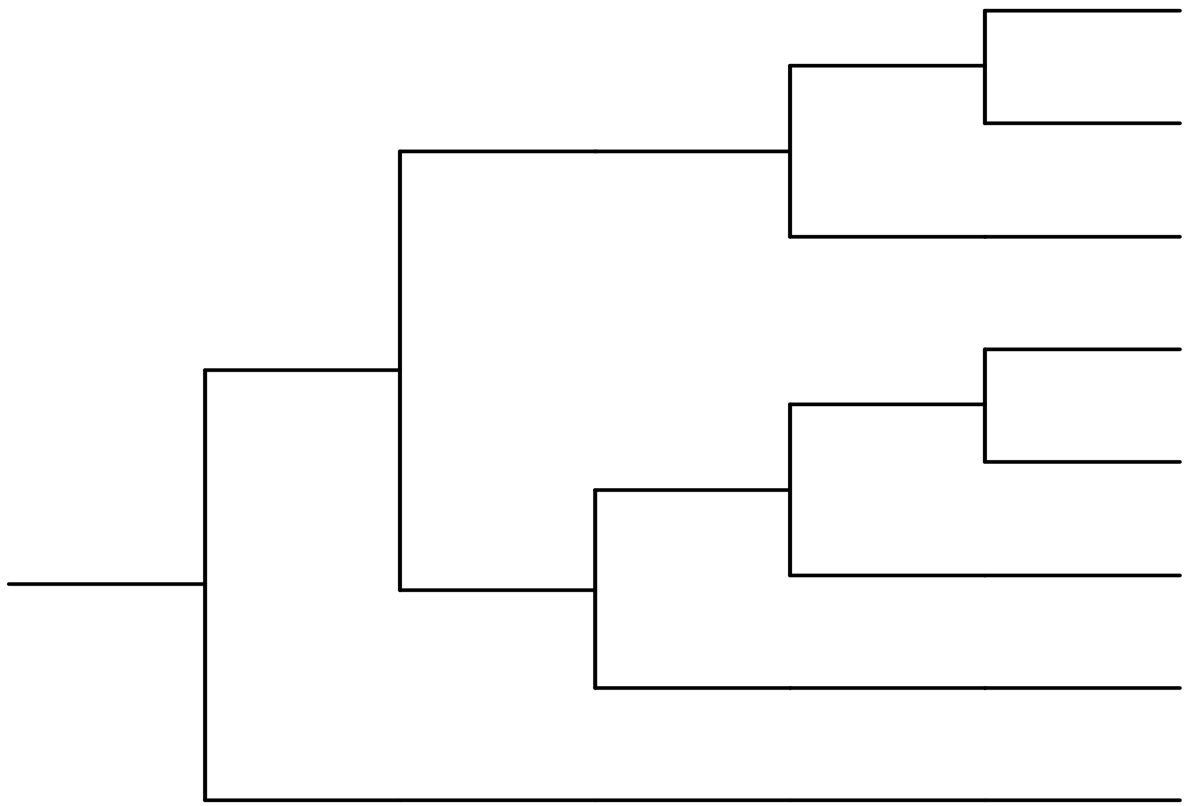
3 OK
4 CD
5 OK
6 CD
7 CD
8 OK
9 CD
10 CD
11 CD
12 CD
Plant name abbrev.: IPT3


MLOCA
EIAC
EIHP
LPR
#
STATUS
1
OK
2
CD
3
CD
4
CD
Plant name abbrev.: IPT3

LLOCA	EIAC	EIHP	EILP	LPR	HLR	#	STATUS



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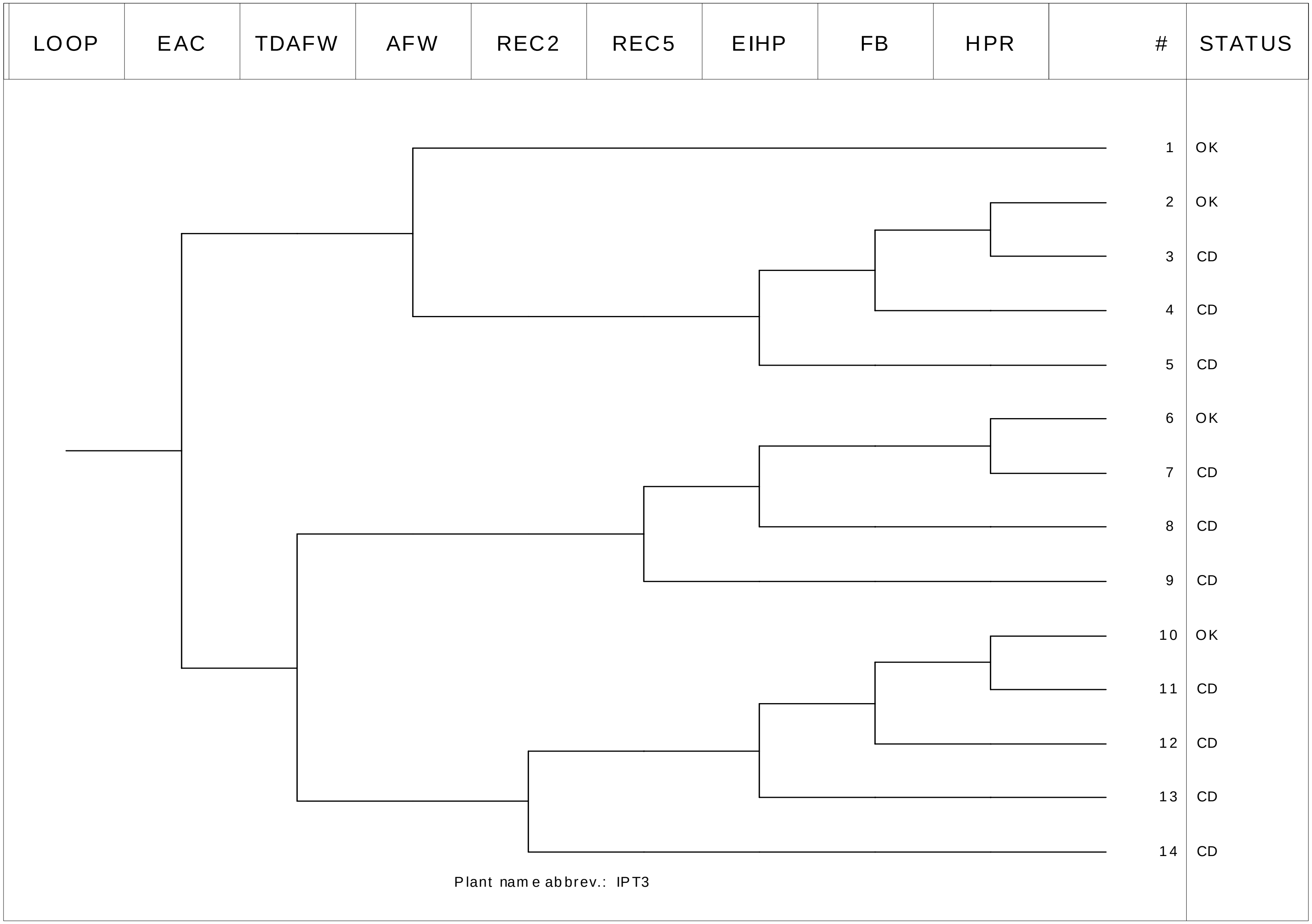
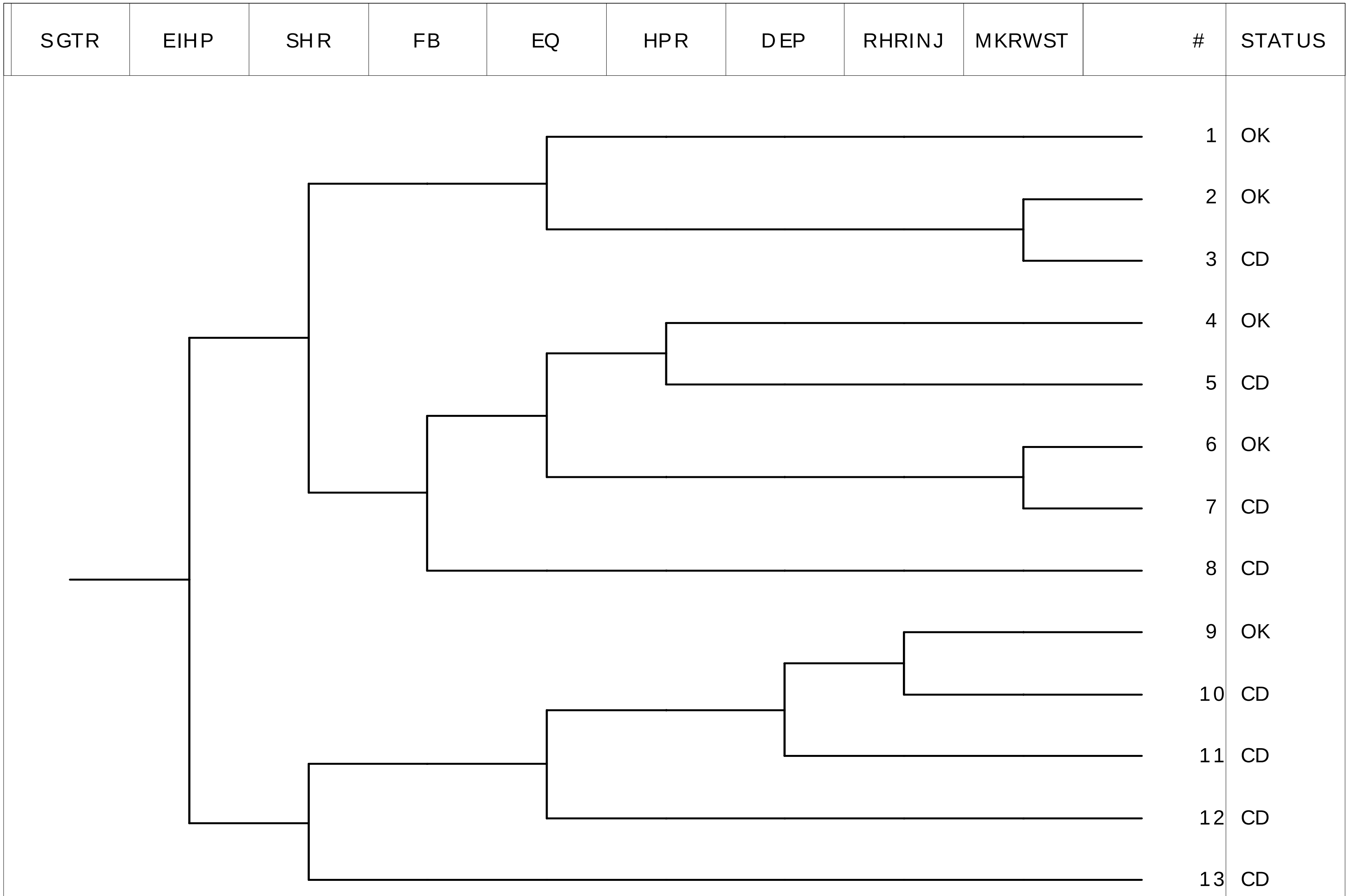
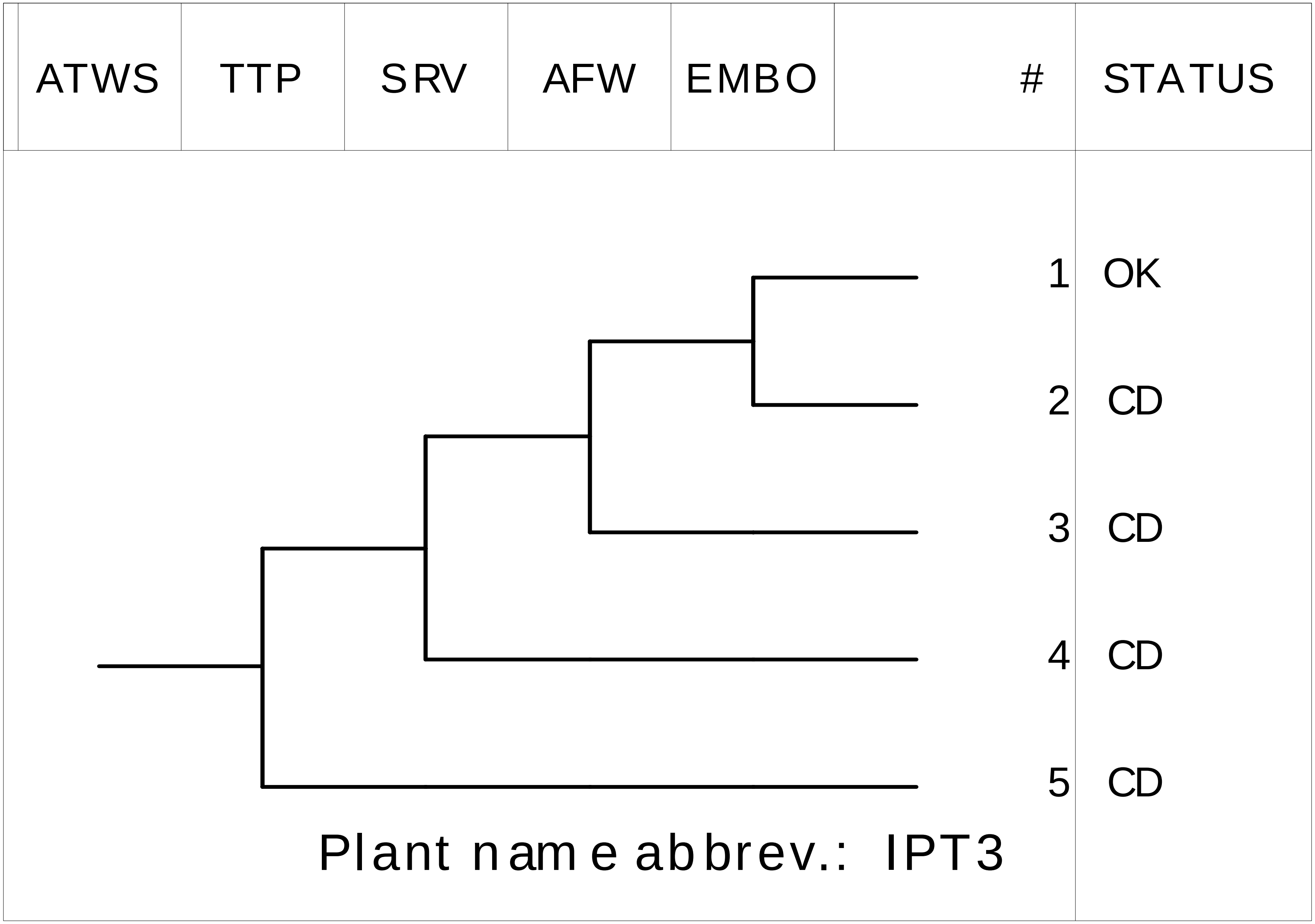
3	CD
4	OK
5	CD
6	CD
7	CD
8	CD

Plant name abbrev.: IPT3


LOOP
EAC
TDAFW
AFW
REC2
REC5
EIHP
FB
HPR
#
STATUS
1 OK
2 OK
3 CD
4 CD
5 CD
6 OK
7 CD
8 CD
9 CD
10 OK
11 CD
12 CD
13 CD
14 CD
Plant name abbrev.: IPT3



Plant name abbrev.: IPT3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TTP
SRV
AFW
EMBO
#
STATUS
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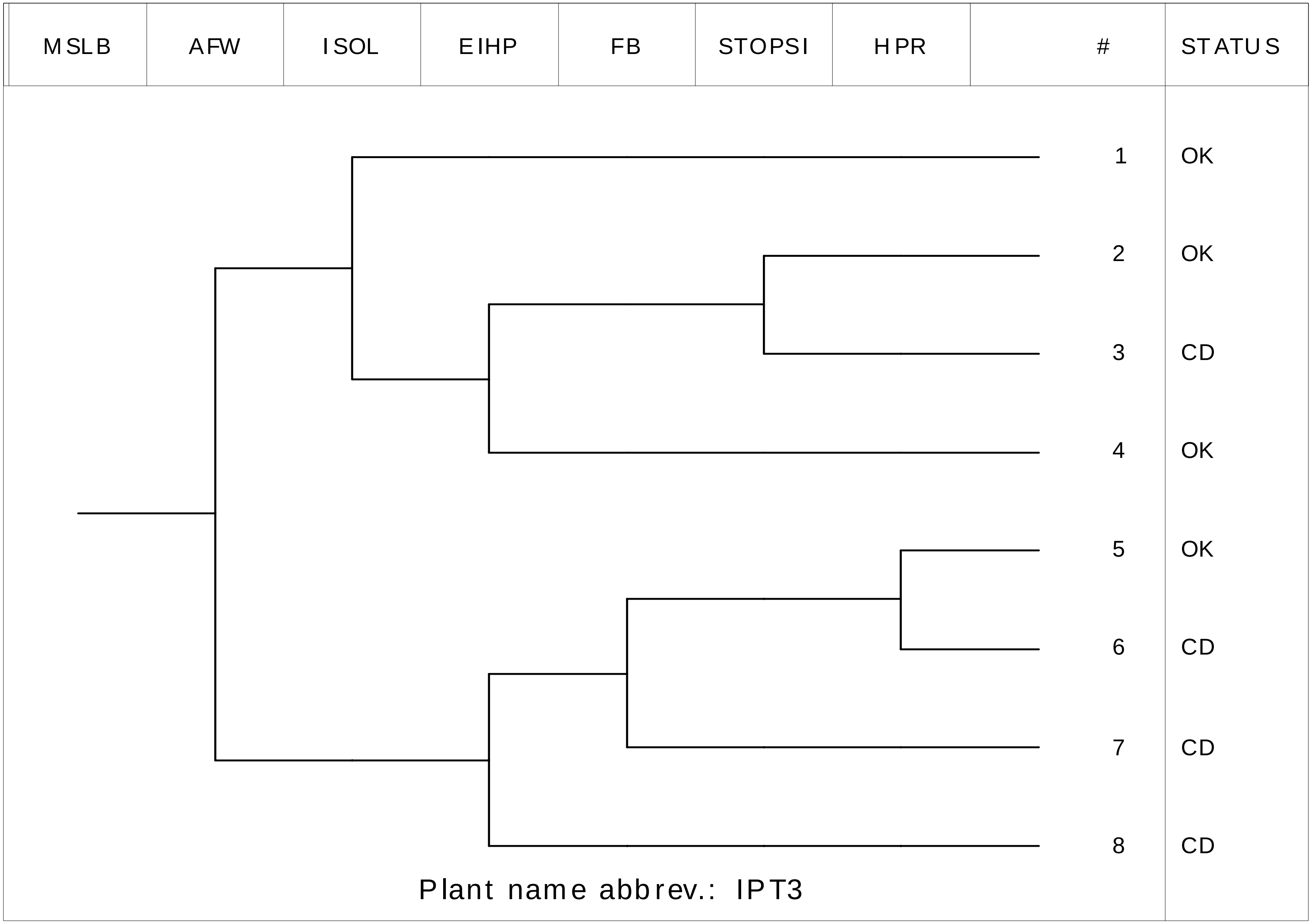
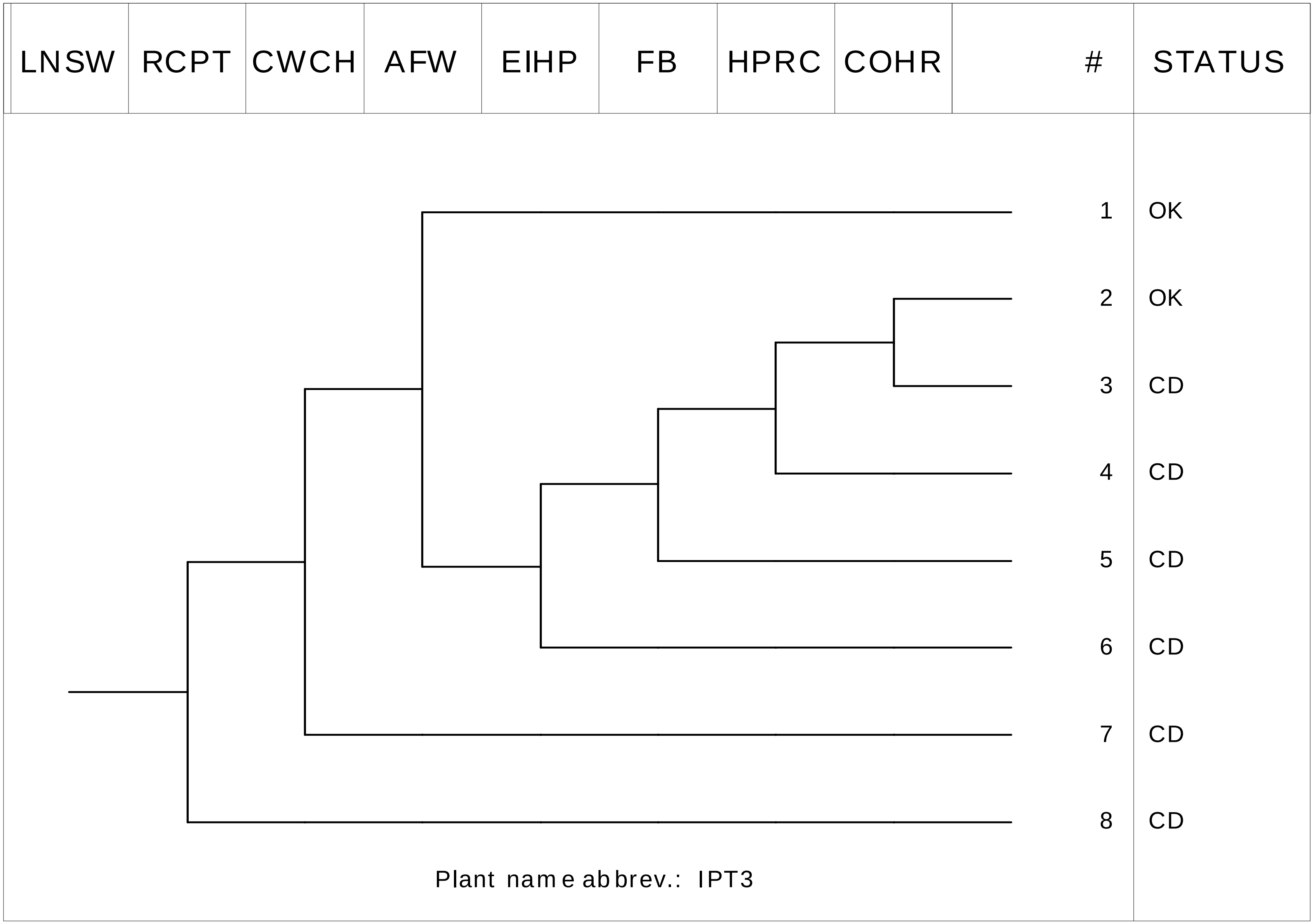
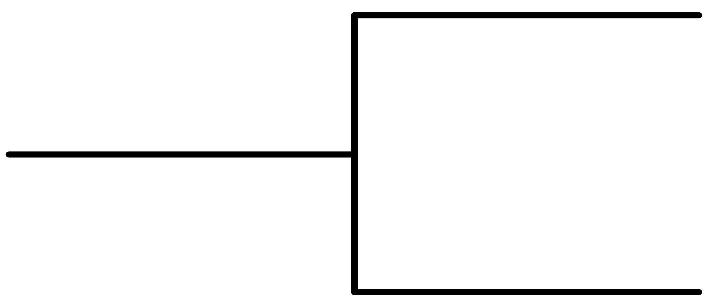
2 CD
3 CD
4 CD
5 CD
Plant name abbrev.: IPT3



Plant name abbrev.: IPT3


LNSW
RCPT
CWCH
AFW
EIHP
FB
HPRC
COHR
#
STATUS
1 OK
2 OK
3 CD
4 CD
5 CD
6 CD
7 CD
8 CD
Plant name abbrev.: IPT3

LBDC	AFW		#	STATUS
			1	OK
			2	CD



Plant name abbrev.: IPT3

LOAC	PORV	AFW	EIHP	FB	DEPR	HPR	LPI	LPR	#	STATUS
									1	OK
									2	OK
									3	CD
									4	OK
									5	CD
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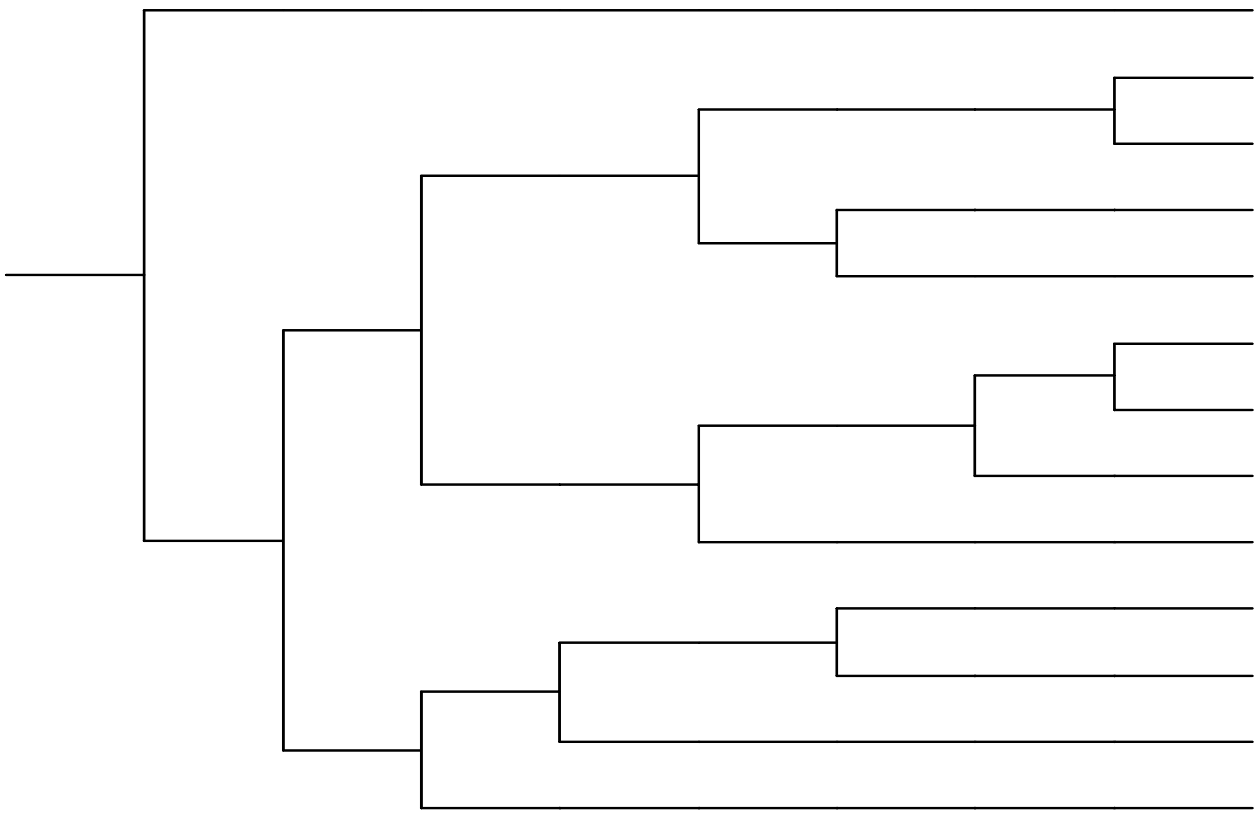
									7	CD
									8	CD
									9	CD
									10	OK
									11	CD
									12	CD
									13	CD



Plant name abbrev.: IPT3

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:

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

2) 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.

3) 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.

4) 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

NRC met with Indian Point 3 PRA personnel. The Licensee provided useful comments on the draft worksheets. The special initiators, plant response, and the initiator impacts were discussed. Other items included as a part of NRC review process were also discussed. Several questions were raised and written responses were received from the licensee subsequent to the meeting. The licensee responses were reviewed and incorporated into the SDP worksheet to the extent possible within the framework, scope, and limitations of the SDP worksheets. The licensee's comment and feed back have significantly contributed to the improvement of this document.

1) Licensee suggested to delete the Transient with Loss of PCS, but it was kept here because it is part of the standard SDP document.

2) Upon loss of IAS, the operators have to restore the system within minutes or manually scram the reactor. While the IAS supplies air throughout the plant, its loss does not adversely affect any safety system function because components required to support emergency operating conditions are designed to fail in a safe position or are provided with a back-up source of nitrogen. Hence, the loss of IAS has the same effect on the plant as a loss of main feedwater (PCS) (IPE, page 3-117).

3) Loss of Essential SWS affects EDG cooling and containment fan coolers. Accordingly, the structure of the event tree for a loss of Essential SWS is the same as Transients with PCS Available (TRANS). The frequency of this initiator is 1.47E-3 / year. Hence, it is bound by TRANS, and the corresponding worksheet was not developed.

4) Loss of HVAC in control building switchgear room has no immediate effect on the plant, but as the room heats up, electrical components in the 480 V AC and 125 V DC power supplies degrade, resulting in a reactor scram. Subsequently, the time during which safety-related systems are available is limited. Loss of HVAC in control building switchgear room fails multiple systems required for plant shutdown; in particular, this initiator eventually causes a loss of the 480 V AC buses and the subsequent failure of AFW motor-driven pumps.

   The licensee carried out an analysis, and concluded that high temperatures would impact the recirculation phase of injection. Operators can use the Appendix R EDG to power safe shutdown equipment (one dedicated train), and the plant's staff has a procedure to provide cooling to the switchgear room by opening doors. The licensee does not include the Loss of HVAC in control building switchgear room in its current PRA model.

5) The licensee requested to include internal flooding, but this is outside the current scope of the SDP documents.

6) The licensee suggested to remove the worksheets on "LOOP and Loss of One 480 V AC Bus". However, we kept them to be consistent with the SDP process.

7) The licensee suggested to modify the sequences or IE frequency of ATWS to include the fraction of cycle that the MTC is favorable and RCS overpressurization would not occur. However, such modification was not implemented because the SDP process assumes that the MTC is unfavorable, and this document uses a generic industry-averaged frequency.

8) Most of the licensee's comments on the SDP worksheets were incorporated. The remaining comments from the licensee are addressed mainly in Sections 1, "Information Supporting Significance Determination Process (SDP)", and 2.1, "Generic Guidelines and Assumptions (PWRs)" of this document.

REFERENCES

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

2. Power Authority of the State of New York, "Indian Point 3 Nuclear Power Plant – Individual Plant Examination Report," June 1994.

3. Indian Point Unit 3 Information with 5 attachments, no date.