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INFORMAL REPORT

PILOT PRA APPLICATIONS PROGRAM FOR INSPECTION AT INDIAN POINT 2

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EG&G Idaho

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ABSTRACT

This report contains information on the development of an application of probabilistic risk assessment (PRA) to the inspection of nuclear power plants. The current inspection program was analyzed to determine the appropriate method for interjecting PRA information. Once the method was determined, a trial application was performed for the Indian Point Unit 2 inspection effort. Recommendations for further development and application are given.

SUMMARY

The Probabilistic Risk Assessment (PRA) Application Program for Inspection at Indian Point Unit 2 was performed for Region I of the Office of Inspection and Enforcement (IE) of the Nuclear Regulatory Commission (NRC) by EG&G Idaho, Inc. at the Idaho National Engineering Laboratory (INEL). The purpose of the program was to develop and test a method for applying PRA techniques and results to the ongoing inspection effort for operating nuclear power plants. Using the PRA to help identify the risk significant events and equipment at a given site will assist IE inspectors in focusing resources on the portions of a facility that contribute most to public risk.

The current inspection approach was analyzed to determine the best method for interjecting PRA information into the process. Official IE documentation and interviews with inspection personnel at the region and at the Indian Point Station were used in this analysis.

Various potential methods for affecting the inspection program included an entire rebuild of the operating phase inspection framework, modification of the frequencies or priorities of the inspection modules within the framework, and rewriting the modules which give specific direction to the inspection staff. The selected method retains the current framework and directs specific component or system selection for inspection at the sites which had PRAs performed. A comparison of the proposed program, which includes the PRA enhancements, with the current program is given.

The method which was developed was applied at the Indian Point Station. Trial inspections were performed using the information derived from the PRA. A full application for Indian Point Unit 2 was performed and is presented here.

Recommendations were developed for the application of the methodology at other sites with completed PRAs and for development of a methodology for sites without PRAs.

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CONTENTS

ABSTRACT ii

SUMMARY iii

ACKNOWLEDGMENTS iv

PURPOSE AND SCOPE 1

THE CURRENT INSPECTION PROGRAM 2

APPLYING PRA INFORMATION TO INSPECTION 4

PRA Duplication 4

Importance Analysis 6

Event Tree Element Characterization 8

IE Module Characterization 9

Sequence and System Importance 10

Module Ranking 10

SUGGESTED PRA BASED PROGRAM ENHANCEMENTS 19

COMPARISON OF CURRENT AND SUGGESTED PROGRAMS 21

RECOMMENDATIONS 22

REFERENCES 24

APPENDIX A--SUGGESTED INSPECTION GUIDANCE FOR INDIAN POINT UNIT 2 25

TABLES

1. Event Tree Element Characterization 11

2. System Importance Calculation Results 14

3. Modules Addressing PRA Related Systems or Functions 15

4. Modules Directly Affecting Important Systems or Functions at IP2 16

5. Indian Point 2 Most Important Sequences 17

PILOT PRA APPLICATIONS PROGRAM FOR INSPECTION AT INDIAN POINT UNIT 2

PURPOSE AND SCOPE

The work described in this report was performed for the Nuclear Regulatory Commission (NRC) Region I under FIN A6553. The basic task of this project is to provide the NRC inspection effort with a PRA based method to improve the allocation of resources. For the initial attempt of this task, the Indian Point nuclear power plants were selected as prototypical for the approach, with the Indian Point Probabilistic Safety Study[1] (IPPSS) used as the base document. The INEL and Brookhaven National Laboratory were selected to devise a method to apply to units 2 and 3. This report documents the program for unit 2.

As the project progressed, it was decided that a coordinated effort between the laboratories was desired, and that a common methodology should be developed and applied to both plants. The scope of Task 1 of the project was to summarize the current prioritization scheme, to identify the sources of guidance used by the inspectors, to formulate a PRA-based prioritization scheme that would also include the current input variables, to identify sections of the IE manual for potential PRA-based review, and to determine ways to apply PRA methodology to programmatic IE manual modules.

Task 2 involved the performance of several inspections to test the methods and insights developed in Task 1. Task 3 involved modifying the methodology described in Task 1 to include insights gained in the performance of the inspections of Task 2, providing a prioritization scheme for the Indian Point plants, and providing detailed modifications and suggestions for improvement of the IE program described in the IE Manual Chapter 2515. Task 4 will involve application of the suggested methodology for programmatic modules. This report covers tasks 1 to 3.

The basic philosophy of the current IE program is that the individual closest to the plant will be the one who can best select the areas of the plant that require the closest scrutiny. The resident inspector is instructed in the manual to spend twenty percent of his inspection hours on self-directed inspection, which gives him the ability to investigate areas of the plant which may not otherwise be scheduled for inspection in the near future. The inspector could, however, spend more time in areas where his interest or background dictates. This self-directed time also allows the inspector to followup on events or areas that appear to need a closer look without a requirement for a specific inspection procedure. Also, on a quarterly basis, the resident inspector submits to the region a formal request for inspections to be performed by the region-based specialist inspectors. These requests are based on chapter 2515 requirements or perceived need. It is the resident inspector's responsibility to request inspections to meet the 2515 interval requirements, and to monitor progress on other requirements from NRC directives and reactive event followup. There are three areas for which the resident does not have input in regards to schedule. These three areas are Emergency Preparedness, Health Physics, and Security, which are handled entirely through the region.

At the Region office, the resident inspector requests are all compiled by the Division of Project and Resident Programs (DPRP) for prioritization across the region. Here various additional considerations are applied to the process. Included are plant status (since the region must inspect construction, near term operating license, and operating plants), SALP results, population profiles, and the application of resources to NRC directives (instructions, bulletins, and orders) and to reactive events. Finally, the requests go to the Division of Engineering and Technical Programs (DETP), where assignments are made for Regional Technical Specialist Inspectors from the sections in that division (e.g., Health Physics Specialists from the Facilities Radiation Protection Section). Region I is currently upgrading their regional prioritization program. This upgrade may render some of this section's information obsolete.

event tree, and each sequence of each event tree be entered into a data file. All of the failure values for each event in each power configuration are also entered. After this the damage states, (states leading to core damage), and their associated probabilities leading out to the consequence of concern are provided. Importance calculations are performed and a printout is obtained. The printout includes a listing of every sequence, the event importances to each damage state, and the total event importances to plant and public risk. The Indian Point PRA is of the large event tree, small fault tree type with each of the 41 plant event tree elements being evaluated based on eight specific electric power system availability states. Because of this, each of the 15 event trees had to be considered by the SQUIMP computer code eight times, once for each power state. Also, most of the plant event trees had sequences which led to Anticipated Transient Without Scram (ATWS) scenarios, for which another large event tree had been developed.

One of the techniques applied in the PRA was a detailed study of external events. Each event was studied individually in the PRA, and special event trees were not developed. For this reason, only the important sequences from the external event initiators were included in our analysis. Also, due to the lack of formal event trees, the external event sequences were hard-wired into the importance calculator code. Their inclusion was necessary for establishing the correct total risk for normalizing some of the importance measures.

The plant event tree sequences were each placed into one of 25 plant damage states or were determined to lead to successful core cooling. The 25 damage states each had specific containment and site response characteristics with varying levels of exposure, injury, and fatality probabilities. The containment and site characteristics used for evaluation of risk in our study was the occurrence of one fatality.

Fussell-Vesely Importance (FV) - The Fussell-Vesely importance[4] represents the fractional portion of the risk which involves failure of the event. It is the inspection importance normalized to total risk, and therefore exhibits the same characteristics for ranking as the Inspection importance.

Risk Achievement Worth, Interval (RAWI) - This measure, and the next three, were developed by W. E. Vesely[5]. The Risk Achievement Worth represents the increase in risk if the event were assumed to be failed. It is calculated as the Birnbaum importance for the event minus the Inspection importance for that event, since the Birnbaum measure represents the risk with event failure for sequences containing the event and the Inspection measure represents the initial risk for those same sequences.

Risk Achievement Worth, Ratio (RAWR) - This measure is similar to the RAWI, except that the ratio is used. Note that the total risk was included in both terms of the above calculation for RAWI, and therefore was cancelled out of the result. In the RAWR calculation, this does not occur since the sum of the total risk and the RAWI is divided by the total risk.

Risk Reduction Worth, Interval (RRWI) - This measure represents the decrease in risk if the event is made perfectly reliable. The interval is the risk contribution of all sequences not containing the event failure subtracted from the total risk. Since this is numerically equivalent to the Inspection importance, it was not calculated.

Risk Reduction Worth, Ratio (RRWR) - This measure is similar to the RRWI, except that the ratio is used. It is the ratio of total risk to the risk from sequences that do not involve failure of the event. It is calculated as total risk divided by the difference of total risk and Inspection importance.

M^{Sj} - This measure was developed and used by Brookhaven in their ranking of systems at Limerick Generating Station in a startup phase

evaluated to establish the failure rate of these events. This system characterization was necessary because the event tree elements tended to be functional rather than systemic. The second characterization involved the failure modes which dominated the failures of the systems. By combining the results of the two characterizations, the event tree elements could then be associated with specific failure modes, and then eventually associated with the various modules.

Table 1 shows the results of the element characterization. Each entry gives the event designation that was used in IPPSS (or one which was added in order to include non-event-tree sequences such as the 'V' sequence and external events), the description of the event, the system(s) involved, and the Inspection and Birnbaum measures for those failure modes. The total risk is not a simple sum of the element inspection importance measures, since several events were involved in each sequence, and each sequence then contributes to the importance of several events. For that reason, for hand calculations for system importances, each contributing sequence had to be compared between events, insuring that, for each failure sequence, double counting had not occurred. The results of these system importance calculations are given in Table 2.

IE Module Characterization

The next step in applying PRA importances to the inspection effort was to determine the effects that the modules could have on failures. The module review initially included all of the IE modules, not just the ones included in the operating phase inspection program. The modules were first reviewed to determine if the modules addressed failure modes or events involving systems which were PRA related. The list of modules from the operating phase which were associated with PRA related systems or functions is given in Table 3. After this, the modules were again reviewed to determine which were system based and which were programmatic, since the initial attempt at prioritization was to address system based modules only. Table 4 shows the modules which were retained as system based and were included in the final PRA based inspection guidance.

TABLE 1. EVENT TREE ELEMENT CHARACTERIZATION

Event Designation	Event Event Description	System(s)[a] Involved	Inspection Importance (INCS)	Birnbaum Importance (BBCS)
B	Buses 2A,3A,6A Available	EP	1.7E-12	2.7E-08
C	Buses 2A,3A,5A Available	EP	1.5E-11	7.0E-07
D	Buses 5A,6A Available	EP	1.3E-12	9.3E-09
E	Buses 2A,3A Available	EP	6.4E-10	1.2E-03
F	Bus 6A Available	EP	5.1E-10	1.3E-05
G	Bus 5A Available	EP	1.6E-11	2.3E-05
H	No AC Power Available	EP	3.6E-08	1.2E-03
A8	Seismic Events Leading to Release Category 8A	RWST, PZR	1.5E-11	3.1E-05
CB	Wind-induced Control Building Failures	--	2.2E-09	6.1E-05
CF	Fan Coolers (SGTR)	CF	2.2E-15	6.7E-12
CF1	Fan Coolers	CF	3.7E-08	3.8E-08
CF2	Fan Coolers	CF	5.8E-10	6.8E-10
CS	Containment Spray	CS	3.7E-08	4.0E-08
FZ1	CCW Pump Room Fire Zone	FP	1.0E-11	3.0E-05
FZ1A	Electrical Tunnel, PAB Side, Fire Zone	FP	1.9E-10	9.4E-05
FZ10	DG Room Fire Zone	FP	2.6E-11	3.0E-05
FZ11	Cable Spreading Room (Control Building) Fire Zone	FP	4.9E-11	3.0E-05
FZ14	Switchgear Room Fire Zone	FP	1.5E-10	6.1E-05
FZ15	Control Room Fire Zone	FP	1.5E-11	3.0E-05
FZ23	AFWS Pump Room Fire Zone	FP	2.9E-11	3.0E-05
FZ32A	Electrical Tunnel, Control Building Side, Fire Zone	FP	1.2E-10	9.4E-05
FZ74A	Electrical Penetration Area Fire Zone	FP	3.0E-12	3.0E-05
HH1	High Head Injection	SIS	1.8E-08	7.7E-08
HH2	High Head Injection, Given Reactor Trip	SIS	6.6E-10	6.5E-07
HH3	High Head Injection and Boron Injection	SIS	2.2E-13	1.7E-09
K2	Reactor Trip	RCS	2.2E-13	1.1E-08
K3	Reactor Trip	RCS, RPS	4.9E-11	2.4E-06
K5	Turbine Runback	TPS	0.0E-00	0.0E-00
L1	AFW Actuation and Secondary Cooling	AFW	6.7E-10	4.9E-06
L2	AFW Actuation and Secondary Cooling - ATWS	AFW	3.8E-11	4.4E-09
L3	AFW Actuation and Secondary Cooling	AFW	1.0E-11	5.3E-08
LP	Cold Leg Injection Path "V" Sequence	SIS, RHR	0.0E-00	0.0E-00

TABLE 1. (continued)

Event Designation	Event Event Description	System(s)[a] Involved	Inspection Importance (INCS)	Birnbaum Importance (BBCS)
RS	Recirculation Spray	RHR,CS	0.0E-00	0.0E-00
RW2	Seismic Events Leading to Release Category 2RW	--	4.4E-10	6.1E-05
S	Reactor Trip and Safety Injection Signals	SAS,RPS	2.2E-13	8.3E-05
SA1	Safety Injection Actuation Signal	SAS	4.2E-12	6.8E-07
SA2	Safety Injection Actuation Signal and High Head Pumps	SAS	4.2E-12	6.8E-07
SL1	No Secondary Side Leakage to Atmosphere Given Operator (OP41,42) Success	PCS	8.3E-11	8.3E-07
SL2	No Secondary Side Leakage to Atmosphere Given Operator (OP41,42) Failure	PCS	2.1E-10	4.1E-08
SO	Safety Injection Operable	SIS	3.4E-15	3.4E-13
TK	Refueling Water Storage Tank	RWST	1.7E-14	7.3E-07
TT1	Turbine Trip	PCS,TPS	1.3E-13	4.7E-08
TT2	Turbine Trip/MSIV Closure	PCS,TPS	8.1E-13	4.5E-09
Z1	Seismic Events Leading to Release Category Z-1	--	1.9E-09	3.0E-01

a. For system descriptions see Table 2.

TABLE 3. MODULES ADDRESSING PRA RELATED SYSTEMS OR FUNCTIONS

MODULE NUMBER	MODULE TITLE
37700	Design, Design Changes, Modifications
38701	Procurement Program
41701	Requalification Training
56700	Calibration
61701	Surveillance (Complex)
61705	Incore/Excore Detector Calibration
61706	Core Thermal Power Evaluation
61719	Surveillance - Containment Integrated Leak Rate Test
61720	Surveillance - Containment Leak Rate Testing (Type B&C)
61725	Surveillance Testing and Calibration Control Program
61726	Monthly Surveillance Observation
62700	Maintenance
62702	Maintenance Program
62703	Monthly Maintenance Observation
64704	Fire Protection/Prevention Program Implementation
71707	Operational Safety Verification
71710	ESF System Walkdown
71711	Plant Startup from Refueling

TABLE 5. INDIAN POINT 2 MOST IMPORTANT SEQUENCES

Initiator	Faulted Events	Systems Involved	Public Health Importance
V	A	RHR	.701
ET1	H	EP	.120
ET2	H	EP	.120
ET11B41	F	EP	.003
ET3	A R2	RHR	.003
ET2	E	EP	.002
ET1	A R1	EP, RHR	.002
ET2	A R1	EP, RHR	.002
ET1	E	EP	.002
ET4	A OP41 SL2	PZR, PCS	.001
ET11A	H L1	EP, AFW	.001
ET12A	H L1	EP, AFW	.001
ET7	H L1	EP, AFW	.001
ET1	A LP1	ACC, RHR	.001
FIRE	A FZ1A	FP	.001
FIRE	A FZ14	FP	.001
FIRE	A FZ32A	FP	.001
ET4	A SL1	PCS	.001

Notes:

Initiator	Description
V	Interfacing System Loss of Coolant Accident
ET1	Large Loss of Coolant Accident
ET2	Medium Loss of Coolant Accident
ET3	Small Loss of Coolant Accident
ET4	Steam Generator Tube Rupture
ET7	Soss of Main Feedwater
ET11A	Turbine Trip
ET11B41	Loss of Offsite Power
ET12A	Spurious Safety Injection

SUGGESTED PRA BASED PROGRAM ENHANCEMENTS

The previous section discussed briefly several options which were available for risk directed inspection enhancement. The module reviews which were performed revealed an overall program which provides for inspection of all aspects of the nuclear facility, with a flexible framework which allows customizing the program implementation to address the many varied plants. It was decided that the best point at which to insert risk based inspection guidance was not in module selection or frequency, but in the direction that the inspector's efforts should take once a module is selected. After a small set of minimum requirements, module selection should be more a function of plant operator performance or other external factors rather than plant risk characteristics.

The information that the PRA can provide to focus inspections should therefore not be placed in a new module and should not necessarily change the frequency at which a modular inspection is performed. But it should direct inspections of the functional areas to the systems and components for which each area is an important contributor to risk. Thus, from plant to plant the IE Manual remains the same, but the plant specific information on important plant systems, components, and functions is the only variable.

Appendix A gives the guidance that is developed for Indian Point 2. It includes, for each important system, a description of the risk significant components or functions, a listing of the IE Manual modules which should address the failures of concern, and a modified checkoff list which can be used to check that important equipment in standby systems are at least visually ready to operate. The appendix was developed using the methodology described in the previous sections, with the component and function identification coming from the fault tree analyses performed for the PRA.

The guidance provided in the appendix would be used differently by persons in different inspection roles. The regional specialist inspector would use the guidance for each plant to familiarize himself with the risk

COMPARISON OF CURRENT AND SUGGESTED PROGRAMS

As mentioned in a previous section, the decision was made to inject PRA results into the inspection program in additional information instead of modifying the current structure of the IE Manual. For this reason the only difference in the current and the suggested programs for inspection is the addition of a guide to the risk significant systems, components, and functions at plants for which such guides have been developed. At the present time, only Indian Point Units 1 and 2 have such guides. The methodology for developing these guides is usable for developing guides at any plant that already has a completed level 3 PRA. For plants with level 1 PRAs, which only present the risk of core melt and not public risk, the methodology could be used, but with reduced effectiveness. In the Recommendations section which follows, suggestions will be made regarding the potential for applying a similar methodology to plants for which PRAs have not been performed.

The use of the guides produced in this project will probably follow the current philosophy of the IE Manual, which is to allow considerable flexibility to the inspector in the performance of his duties. Thus, a strict requirement for using the guides on a frequent basis is not suggested. Instead, the inspector would use the guide to make his current efforts more efficient in addressing plant risk.

in these plants are two more Westinghouse reactors. Once these inspection guides have been developed, a generic list of systems could be generated. Qualitative system studies, with a generous amount of quantitative insight, could then provide the list of components with which to complete the guidance. This would provide PRA-like input to the inspection effort with a minimal expense, especially when considering the cost of full PRAs. Generic guidance could be provided in this manner for most Westinghouse and GE reactors. The question of availability of an adequate number of studies on CE and B&W plants precludes determination of feasibility of generic studies at these plants at this time.

An area which is also still at question is the use of plant risk characteristics for comparison of plants in each region. The current scheduling of inspection by regional specialist inspectors does not include relative risk by quantification. It does include risk as identified by population density and operator performance. Due to the variance in the age and analysts of PRAs, and therefore a question regarding the appropriateness of comparison, we would suggest that the current method of regional prioritization is adequate. It follows naturally that a nationwide prioritization of resources by risk would also be inappropriate.

APPENDIX A

INDIAN POINT UNIT 2
PRA IDENTIFIED INSPECTION GUIDANCE

GUIDANCE FOR PROBABILISTIC RISK BASED INSPECTION

This appendix provides guidance to inspectors based on the results of the Indian Point Probabilistic Safety Study. The guidance permits the focusing of the inspector's resources on those systems that make up at least 95% of the risk and components that are significant contributors to the failure of those systems.

The probabilistic risk assessment (PRA) is an analytical technique for integrating diverse aspects of design and operation in order to assess the risk of a particular nuclear power plant and to develop an information base for analyzing plant-specific and generic issues. An assessment of the plant-specific risk provides both a measure of potential accident risk to the public and insights into the adequacy of plant design and operation. This assessment is achieved by identifying those sequences of potential events that contribute to risk and by establishing which features of the plant contribute most to the frequency of such sequences. These plant features may be influenced by hardware failures, tests and maintenances during operations, or human errors resulting from tests, maintenance, or operational activities. Thus, a probabilistic analysis reveals the features of a plant that merit close attention and provides a focus for improving safety. Information developed in the assessment will be used to make decisions about the allocation of resources by directing attention to the factors and failure modes that contribute to plant risk, and therefore should be involved in an effort to reduce risk.

The failure mode identification table for each system, Table XX-1 describes the way in which each component failure contributes to risk. By understanding the type of failure that is significant, the inspector can focus on particular subcomponents, operations, tests, etc. For example, a premature transfer of a power supply will not usually be a harmful event, while the failure to transfer usually is. The inspector would want to review or observe tests that cause transfer, and calibrations of relays that ensure proper dropout voltage or frequency. This table should be useful to all inspectors, regardless of experience levels or familiarity with the plant.

PRA IDENTIFIED IMPORTANT SYSTEM TABLES

1.0 Residual Heat Removal System (Recirculation Components)

2.0 Service Water System

3.0 Component Cooling Water System

4.0 Containment Fan Cooling System

5.0 Electric Power System

6.0 Containment Spray System

7.0 Safety Injection System

8.0 Accumulator System

9.0 Pressurizer Relief System

10.0 Auxiliary Feedwater System

TABLE 1-1. (continued)

Conditions That Lead to Failure

5. Hot Leg Injection Valves 856 B, F Fail to Open

Hot leg recirculation is initiated manually at a time near the end of the 24 hour cooldown period. Failure of these valves prevents continued cooldown. Scheduled cycling of these valves, and proper power availability, as shown by checkoff list usage, will enhance proper operation.

6. Spray Header Inlet Valves 889 A, B Fail to Open

Containment cooling recirculation is required when the RWST finally empties after the injection and spray phases. Failure of both valves is required. Scheduled cycling of these valves, and proper power availability, as shown by checkoff list usage, will enhance proper operation. The Containment Fan Cooling System provides a backup for this function.

7. RHR Recirculation Suction Valves 730, 731 Rupture/Fail to Remain Closed

Failure of these valves bypasses containment, and initiates a LOCA. Under normal operation, the valves are closed, with power deenergized. Position and integrity are very significant. The proper performance of leak checks and checkoff list usage will minimize such failures. These records should be reviewed.

8. Cold Leg Injection Inlet Valves 746, 747 Rupture/Fail to Remain Closed

Failure of these valves, along with rupture of the injection check valves, bypasses containment and initiates a LOCA. The valves are normally closed, but have power available, and open on SI. This 'V' sequence is only 1% as significant as the hot leg valves 730, 731. The required position of these valves should be verified, and leakage checks of the cold leg check valves should be observed or reviewed.

9. Cold Leg Injection Check Valves 838 A, B, C, D, 897 A, B, C, D Rupture

These valves operate with 746 or 747 as initiator. Performance testing for backleakage or failure to close should be reviewed or observed.

TABLE 1-3. MODIFIED RESIDUAL HEAT REMOVAL SYSTEM (RECIRCULATION COMPONENTS) WALKDOWN

Component	Required Position	Actual Position
Power Supplies		
MCC26A		
746 RHR Heat Exchanger 22 Outlet Stop Valve Disc. Sw. 2H	Closed	_____
RHR Heat Exchanger 21 Supply Shut-Off Valve 822A Disc. Sw. 1M	Closed	_____
MCC26B		
747 RHR Heat Exchanger 21 Outlet Valve Disc. Sw. 2H	Closed	_____
RHR Heat Exchanger 22 Supply Shut-Off Valve 822B Disc. Sw. 1M	Closed	_____
480V Switchgear 21		
Residual Heat Removal Pump 21 (Bus3A) Breaker 5C	Racked in Control fuse in	_____ _____
Residual Heat Removal Pump 22 (Bus6A) Breaker 11A	Racked in Control fuse in	_____ _____
Lineup Following RHR System Shutdown		
Inside Containment		
889A RHR Spray Header Stop	Closed	_____
889B RHR Spray Header Stop	Closed	_____
730 RHR Loop Inlet Stop	Closed	_____
731 RHR Loop Inlet Stop	Closed	_____
746 RHR HX 22 Outlet Stop	Closed	_____
747 RHR HX 21 Outlet Stop	Closed	_____
MCC26A		
731 RHR Hot Leg Inlet Stop Disc. Sw. 2M	Open Locked	_____ _____
889A RHR Spray Header Stop Disc. Sw. 4MR	Closed	_____

TABLE 1-3. (continued)

Component	Required Position	Actual Position
RHR Heat Exchanger 21 CCW Outlet Stop Valve 822B	Valve Closed Switch Neutral	_____ _____
Panel SBF-1 (CCR)		
822A RHR Heat Exchanger Comp. Cool. Outlet Valve	Closed	_____
822B RHR Heat Exchanger Comp. Cool. Outlet Valve	Closed	_____

References: Indian Point Station Unit 2 Checkoff List 4.1, Rev. 1, January 1985.
Indian Point Station Unit 2 Checkoff List 4.2, Rev.0, October 1984.

TABLE 2-1. (continued)

Conditions That Lead to Failure
5. Pump Discharge Check Valve Fails Open With the failure of a pump to start or run (item 1), the associated check valve must close and prevent backleakage from the discharge header to the intake structure. Such backleakage would fail the system. The only defense available is backflow checks during monthly pump testing and surveillance.
6. Pump Train Unavailable Due to Maintenance This failure affects the nonessential header only, since all three pumps on the essential header must be operable, with an eight hour LCO. The events here are both scheduled and corrective maintenances during operation. The maintenance performance should be reviewed to ascertain whether less maintenance down time could be achieved.

TABLE 2-3. MODIFIED SERVICE WATER SYSTEM WALKDOWN

NOTE: Perform Sections for 21, 22, and 23 if the essential header is aligned to service water pumps 21, 22 and 23. Perform Sections for 24, 25, and 26 if the essential header is aligned to service water pumps 24, 25 and 26. All other sections apply in either case.

Component	Required Position	Actual Position
CENTRAL CONTROL ROOM		
Panel SHF		
Essential Header Selector Switch	21, 22, 23	_____
	or	
	24, 25, 26	_____
Panel SHF		
Service Water Pump 21 Indicating Light	AUTO	_____
	ON	_____
	OFF	_____
Service Water Pump 22 Indicating Light	AUTO	_____
	ON	_____
	OFF	_____
Service Water Pump 23 Indicating Light	AUTO	_____
	ON	_____
	OFF	_____
Service Water Pump 24 Indicating Light	AUTO	_____
	ON	_____
	OFF	_____
Service Water Pump 25 Indicating Light	AUTO	_____
	ON	_____
	OFF	_____
Service Water Pump 26 Indicating Light	AUTO	_____
	ON	_____
	OFF	_____
Panel SBF-1		
Essential Header 21, 22, 23		
SWN-1111 Conventional Supply from Header 4 Indicating Light	OPEN	_____

TABLE 2-3. (continued)

Component	Required Position	Actual Position
PRIMARY AUXILIARY BUILDING		
Common Fan Cooling Unit (FCU) Supply		
Essential Header 21, 22, 23		
SWN-39 Supply from Header 1	OPEN	_____
SWN-38 Supply from Header 4	CLOSED	_____
Essential Header 24, 25, 26		
SWN-39 Supply from Header 1	CLOSED	_____
SWN-38 Supply from Header 4	OPEN	_____
Component Cooling Heat Exchangers		
Header Supplies		
Essential Header 21, 22, 23		
SWN-32 Supply from Header 1	CLOSED	_____
SWN-31 Supply from Header 4	OPEN	_____
Essential Header 24, 25, 26		
SWN-32 Supply from Header 1	OPEN	_____
SWN-31 Supply from Header 4	CLOSED	_____
Service Water Pump 21		
SWN-599 Strainer Blowdown Manual Valve	OPEN	_____
SWN-2 Pump Outlet Stop	OPEN	_____
SWN-506 Test Header Stop	CLOSED	_____
SWN-600 Strainer Blowdown Manual Valve	OPEN	_____
MOV-SWN-622 Pump Strainer Outlet	OPEN	_____

TABLE 2-3. (continued)

Component	Required Position	Actual Position
Service Water Pump 26		
SWN-589 Strainer Blowdown Manual Valve	OPEN	_____
SWN-2-5 Pump Outlet Stop	OPEN	_____
SWN-501 Test Header Stop	CLOSED	_____
SWN-590 Strainer Blowdown Manual Valve	OPEN	_____
MOV-SWN-617 Pump Strainer Outlet	CLOSED	_____
MCC-21 (15 ft. El. Dock)		
Breaker 1 DL-Feed for Zurn Strainers MO Backwash Valves And SW Pit Heaters	ENERGIZED	_____
SW PIT PANELS		
Zurn Strainer PNL 21		
Panel Switch	ON	_____
Motor Start Button Indicating Light	ON	_____
Back Wash Switch	AUTO	_____
Zurn Strainer PNL 22		
Panel Switch	ON	_____
Motor Start Button Indicating Light	ON	_____
Back Wash Switch	AUTO	_____
Zurn Strainer PNL 23		
Panel Switch	ON	_____

TABLE 2-3. (continued)

Component	Required Position	Actual Position
Essential Header 21, 22, 23		
SWN-6 Oil Coolers Supply from Header 4	CLOSED	_____
SWN-7 Oil Coolers Supply from Header 1	OPEN	_____
SWN-1111 Conventional Supply from Header 4	OPEN	_____
SWN-1112 Conventional Supply from Header 1	CLOSED	_____
Essential Header 24, 25, 26		
SWN-6 Oil Coolers Supply from Header 4	OPEN	_____
SWN-7 Oil Coolers Supply from Header 1	CLOSED	_____
SWN-1111 Conventional Supply from Header 4	CLOSED	_____
SWN-1112 Conventional Supply from Header 1	OPEN	_____
Heat Exchanger 21		
Essential Header 21, 22, 23		
SWN-70 Supply from Header 1	OPEN	_____
SWN-27 Supply from Header 4	CLOSED	_____
Essential Header 24, 25, 26		
SWN-70 Supply from Header 1	CLOSED	_____
SWN-27 Supply from Header 4	OPEN	_____

TABLE 2-3. (continued)

Component	Required Position	Actual Position
Essential Header 24, 25, 26		
SWN-83-3 Supply from Header 1	CLOSED	_____
SWN-83-4 Supply from Header 4	OPEN	_____
Diesel Generators		
Essential Header 21, 22, 23		
SWN-30 Supply from Header 1	OPEN	_____
SWN-29 Supply from Header 4	CLOSED	_____
Essential Header 24, 25, 26		
SWN-30 Supply from Header 1	CLOSED	_____
SWN-29 Supply from Header 4	OPEN	_____
Power Supplies and Local Control Panels		
480V SWGR 21 Bus 2A		
Service Water Pump 22 Breaker	RACKED IN	_____
Breaker Control Fuses	IN	_____
Service Water Pump 25 Breaker	RACKED IN	_____
Breaker Control Fuses	IN	_____
480V SWGR 22 Bus 3A		
Service Water Pump 22 Breaker	RACKED IN	_____
Breaker Control Fuses	IN	_____
Service Water Pump 25 Breaker	RACKED IN	_____
Breaker Control Fuses	IN	_____

TABLE 3-1. COMPONENT COOLING WATER SYSTEM FAILURE MODE IDENTIFICATION

Conditions That Lead to Failure

1. Manual Valves 734A or 734B Fail to Remain Open.

Failure of either Valve 734A or Valve 734B to remain open, resulting in loss of cooling for the SI and RHR pumps, causes a total loss of injection capability and therefore renders the remaining portions of the CCW supported mitigation systems ineffective. The failures of interest involve mainly hardware failures of the valve during challenges to the safety systems. These failures include disk to stem separation, vibrating or drifting closed, or plugging. Observation and review of system operation in normal and off-normal system line-ups, including containment isolation conditions, and observation and review of maintenance for these valves would minimize failures when challenged. Failures under normal operation would cause alarm of low RHR pump flow and would cause a shutdown for repair, but would not cause secondary equipment failures.

2. Manual Valves SWN-31 or SWN-32 Fail to Remain Open.

These valves are the service water supply valves from the two service water headers to the CCW heat exchangers. Only one of them is open at a time, with both CCW heat exchangers being fed from one header. Due to the normal operation of the CCW system, this failure would be detected by heat exchanger high temperatures under normal plant operation. As above, variation in system operating conditions and component maintenance should be observed or reviewed to minimize failure.

3. Component Cooling Water Pumps 21, 22, 23 Fail to Start/Fail to Run.

With a Safety Injection initiation, the CCW pumps are stripped from the 480V switchgear buses. With offsite power available, the pumps are restarted by the safeguards sequence signal. With loss of offsite power, the pumps are restarted during recirculation switchover. Failures to start include control circuits, with the low pressure auto-start included. Activities which help minimize pump failures include observation of surveillance, maintenance, calibration, and lineup.

TABLE 3-3. MODIFIED COMPONENT COOLING WATER SYSTEM WALKDOWN

Component	Required Position	Actual Position
COMPONENT COOLING HEAT EXCHANGERS 21 AND 22		
763 CCW Pumps Outlet PI-600 Stop	Open	_____
COMPONENT COOLING PUMPS		
Component Cooling Pump 21		
760C Inlet Stop	Open	_____
A-71 Cooling Supply to Pump Glands	Open	_____
1858E Drain Stop	Closed	_____
	Capped	_____
1858F Drain Stop	Closed	_____
	Capped	_____
A-1 Vent Stop	Closed	_____
	Capped	_____
762C Outlet Stop	Open	_____
Component Cooling Pump 22		
760B Inlet Stop	Open	_____
A-72 Vent Stop	Closed	_____
	Capped	_____
1858C Drain Stop	Closed	_____
	Capped	_____
1858D Drain Stop	Closed	_____
	Capped	_____

TABLE 3-3. (continued)

Component	Required Position	Actual Position
Component Cooling Breaker Pump 22 (Bus 2A)	Racked In	_____
	Control Fuse In	_____
480V Switchgear 22		
Component Cooling Breaker Pump 23 (Bus 3A)	Racked In	_____
	Control Fuse In	_____
Panel SG		
Component Cooling Pump 21	Auto	_____
Indicating Light	On	_____
Component Cooling Pump 22	Auto	_____
Indicating Light	On	_____
Component Cooling Pump 23	Auto	_____
Indicating Light	On	_____
Panel S8F.1		
Component Cooling Pump 21 Indicating Light	On	_____
Component Cooling Pump 22 Indicating Light	On	_____
Component Cooling Pump 23 Indicating Light	On	_____

Reference: Indian Point Station Unit 2 Checkoff List 4.1, Rev. 1, January 1985.

TABLE 4-2. I&E MODULES FOR CONTAINMENT FAN COOLING SYSTEM INSPECTION

Module	Title	Components	Failure Mode[a]
61701	Surveillance (Complex)	TCV 1104, 1105 FCU 21, 22, 23, 24, 25	1 2
61726	Monthly Surveillance Observation	FCU 21, 22, 23, 24, 25	2
62700	Maintenance	FCU 21, 22, 23, 24, 25	2,3
62703	Monthly Maintenance Observation	FCU 21, 22, 23, 24, 25	2,3
71707	Operational Safety Verification	FCU 21, 22, 23, 24, 25	2
71710	ESF System Walkdown	FCU 21, 22, 23, 24, 25 TCV 1104, 1105	2 1

a. See Table 4-1 for failure mode identification.

TABLE 4-3. (continued)

Component	Required Position	Actual Position
Safeguards Supervisory Panel (SB-2) (cont.)		
Containment Recirc. Fan 23	Auto	_____
Indicating Light	Run	_____
Containment Recirc. Fan 24	Auto	_____
Indicating Light	Run	_____
Containment Recirc. Fan 25	Auto	_____
Indicating Light	Run	_____
FCU 21 Valves 1293-1293A (FCV-21-1)	Closed	_____
Indicating Light	Green	_____
FCU 21 Valves 1294-1204A (FCV-21-2)	Closed	_____
Indicating Light	Green	_____
FCU 21 Valves 1295-1295A (FCV-21-3)	Open	_____
Indicating Light	Red	_____
FCU 22 Valves 1296-1296A (FCV-22-4)	Closed	_____
Indicating Light	Green	_____
FCU 22 Valves 1297-1297A (FCV-22-5)	Closed	_____
Indicating Light	Green	_____
FCU 22 Valves 1298-1298A (FCV-22-6)	Open	_____
Indicating Light	Red	_____
FCU 23 Valves 1299-1299A (FCV-23-7)	Closed	_____
Indicating Light	Green	_____
FCU 23 Valves 1300-1300A (FCV-23-8)	Closed	_____
Indicating Light	Green	_____
FCU 23 Valves 1301-1301A (FCV-23-9)	Open	_____
Indicating Light	Red	_____
FCU 24 Valves 1302-1302A (FCV-24-10)	Closed	_____
Indicating Light	Green	_____
FCU 24 Valves 1303-1303A (FCV-24-11)	Closed	_____
Indicating Light	Green	_____
FCU 24 Valves 1304-1304A (FCV-24-12)	Open	_____
Indicating Light	Red	_____

TABLE 5-1. ELECTRIC POWER SYSTEM FAILURE MODE IDENTIFICATION

Conditions That Lead to Failure

1. Diesel Generators Maintenance

With outside power unavailable, the dominant failures are caused by diesel unavailability from maintenance. This maintenance includes both scheduled and unscheduled maintenance. This failure causes loss of ESF loads from the respective failed bus. The performance of maintenance should be reviewed to ensure that efficient scheduling is done, and that repairs are performed correctly, minimizing down time.

2. Diesel Generators Failure to Start

The failure of a diesel to start is usually a result of the failure of a diesel subsystem or a combination of subsystems being at the edges of allowed tolerances. Failure to start causes loss of ESF loads from the bus. Proper maintenance, surveillance, and lineup checks will minimize failures to start. Reviews of maintenance records and performance tests will enhance availability of the DG's.

3. Diesel Generators Improper Post-Maintenance Lineup

The failure to properly align the diesel for automatic starting also results in a loss of ESF loads from the buses. The most critical lineup failure involves the local diesel generator engine control switch. Observing operator lineup following post-maintenance testing or performing the diesel generator checkoff list will reduce the magnitude of this problem.

4. 6.9KV Bus Transfers Failure to Automatically Transfer

On a loss of the Unit Auxiliary Transformer, the 6.9KV buses 2 and 3 should transfer to the Station Auxiliary Transformer through buses 5 and 6. Failure to transfer causes a loss of ESF loads on buses 2A and 3A. The calibration and testing of the relays and automatic transfers, and checks on the lineup for automatic operation of these transfers, should be reviewed.

TABLE 5-2. I&E MODULES FOR ELECTRIC POWER SYSTEM INSPECTION

Module	Title	Components	Failure Mode[a]
56700	Calibration	6.9KV Bus Transfers	4
		480V Bus Undervoltage Relays	6
61701	Surveillance (Complex)	Diesel Generators 21, 22, 23	2,3
		480V Bus Undervoltage Relays	6
		6.9KV Bus Transfers	4
61726	Monthly Surveillance Observation	Diesel Generators 21, 22, 23	2,3
62700	Maintenance	Diesel Generators 21, 22, 23	1,2
62703	Monthly Maintenance Observation	Diesel Generators 21, 22, 23	1,2
71707	Operational Safety Verification	Diesel Generators 21, 22, 23	3
		6.9KV Bus Transfers	4
		MCC 24, 27, 29 Supply Breakers	5
71710	ESF System Walkdown	Diesel Generators 21, 22, 23	3
		6.9KV Bus Transfers	4
		MCC 24, 27, 29 Supply Breakers	5

a. See Table 5-1 for failure mode identification.

TABLE 5-3. (continued)

Component	Required Position	Actual Position
MCC 24		
Diesel Generator 22 Fuel Oil Pump (and other auxiliaries) Disc. Switch 6DL	Closed	_____
Panel SH (CCR)		
Circuit Breaker EG-1	Auto/Open	_____
Control Power (Normal)	Light On	_____
Circuit Breaker EG-2A	Auto/Open	_____
Control Power (Normal)	Light On	_____
Circuit Breaker EG-2B	Auto/Open	_____
Control Power (Normal)	Light On	_____
Circuit Breaker EG-3	Auto/Open	_____
Control Power (Normal)	Light On	_____

Diesel Generators

Due to the integrated nature of the diesel generator failure to start failure mode, the entire diesel generator checkoff list should be performed.

Reference: Indian Point Station Unit 2 Checkoff List 27.5, Rev.0, January 1985.

TABLE 6-2. I&E MODULES FOR CONTAINMENT SPRAY SYSTEM INSPECTION

Module	Title	Components	Failure Mode[a]
61701	Surveillance (Complex)	Pumps 21 and 22	1
		Valves 866 A-D	2
		Valves 867 A, B	5
61726	Monthly Surveillance Observation	Pumps 21 and 22	1
		Valves 866 A-D	2
62700	Maintenance	Pumps 21 and 22	4
		Valves 866 A-D	2
62703	Monthly Maintenance Observation	Pumps 21 and 22	4
		Valves 866 A-D	2
		Valves 869 A, B	3
		Valves 867 A, B	5
71707	Operational Safety Verification	Pumps 21 and 22	1
		Valves 869 A, B	3
		Valves 866 A-D	2
71710	ESF System Walkdown	Pumps 21 and 22	1
		Valves 869 A, B	3
		Valves 866 A-D	2

a. See Table 6-1 for failure mode identification.

TABLE 6-3. (continued)

Component	Required Position	Actual Position
Valve Line-Up		
869A Spray Header 21 Stop	Open	_____
	Locked	_____
869B Spray Header 22 Stop	Open	_____
	Locked	_____
MCC 26AA		
869A Spray Header 21 Stop Breaker B	Open	_____
	Locked	_____
MCC 26BB		
869B Spray Header 22 Stop Breaker L1	Open	_____
	Locked	_____
SAFEGUARDS PANEL - SB-1		
Spray Pump 21 Disch. Stop Vlvs. 866A - 866B (Both Closed)	Auto	_____
Spray Pump 22 Disch. Stop Vlvs. 866C - 866D (Both Closed)	Auto	_____
Cont. Spray Pump No. 21 480V Bus 5A	Auto	_____
Cont. Spray Pump No. 22 480V Bus 6A	Auto	_____

Reference: Indian Point Station Unit 2 Checkoff List 10.2, Rev. 0, October 1984.

TABLE 7-2. I&E MODULES FOR SAFETY INJECTION SYSTEM INSPECTION

Module	Title	Components	Failure Mode[a]
61701	Surveillance (Complex)	MV 846 MOV 1810 CV 847 SI Pumps	1-3
61726	Monthly Surveillance Observation	MOV 1810 SI Pumps	1, 3,
62700	Maintenance	MV 846 MOV 1810 CV 847 SI Pumps	1-3
62703	Monthly Maintenance Observation	MV 846 MOV 1810 CV 847 SI Pumps	1-3
71707	Operational Safety Verification	MV 846 MOV 1810 SI Pumps	1, 3,
71710	ESF System Walkdown	MV 846 MOV 1810 SI Pumps	1, 3,

a. See Table 7-1 for failure mode identification

TABLE 8-1. ACCUMULATOR SYSTEM FAILURE MODE IDENTIFICATION

Conditions That Lead to Failure
1. Check Valves 895 A, B, C, D, 897 A, B, C, D Fail to Open The two check valves on each injection leg must open or accumulator system failure results. Their surveillance and maintenance records should be examined to determine that the surveillance and maintenance have been performed properly, when required, by qualified personnel.
2. MOVs 894 A,B,C,D Fail to Remain Open Failure of any one of the three MOVs on good legs constitutes system failure given a large LOCA. These valves are deenergized open. Their surveillance and maintenance records should be examined to determine that the maintenance and surveillance have been performed properly, when required, by qualified personnel.

TABLE 8-3. MODIFIED ACCUMULATOR SYSTEM WALKDOWN

CAUTION: Accumulator discharge valves are not to be opened until RCS pressure approaches 1000 psig.

	Component	Required Position	Actual Position
MCC 26A			
894A	Accumulator 21 Outlet Stop	Open	_____
	Disc Sw 3MR	Locked	_____
894C	Accumulator 23 Outlet Stop	Open	_____
	Disc Sw 2M	Locked	_____
MCC 26B			
894B	Accumulator 22 Outlet Stop	Open	_____
	Disc Sw 3MR	Locked	_____
894D	Accumulator 24 Outlet Stop	Open	_____
	Disc Sw 2DR	Locked	_____
VALVE LINEUP			
894A	Accumulator 21 Outlet Stop Valve	Open	_____
		Locked	_____
894B	Accumulator 22 Outlet Stop Valve	Open	_____
		Locked	_____
894C	Accumulator 23 Outlet Stop Valve	Open	_____
		Locked	_____
894D	Accumulator 23 Outlet Stop Valve	Open	_____
		Locked	_____

Reference: Indian Point Station Unit 2 Checkoff List 10.1, Rev. 2, May 1985.

TABLE 9-2. I&E MODULES FOR PRESSURIZER RELIEF SYSTEM INSPECTION

Module	Title	Components	Failure Mode[a]
56700	Calibration	PORVs 455C, 456 SRVs 464, 466, 468	1
61701	Surveillance (Complex)	PORVs 455C, 456 SRVs 464, 466, 468	1
61726	Monthly Surveillance Observation	PORVs 455C, 456 SRVs 464, 466, 468	1
62700	Maintenance	PORVs 455C, 456 SRV's 464, 466, 468	1
62703	Monthly Maintenance Observation	PORVs 455C, 456 SRV's 464, 466, 468	1
71707	Operational Safety Verification	Block Valves 535, 536	2
71710	ESF System Walkdown	Block Valves 535, 536	2

a. See Table 9-1 for failure mode identification.

TABLE 10-1. AUXILIARY FEEDWATER SYSTEM FAILURE MODE IDENTIFICATION

Conditions That Lead to Failure

1. Pumps 21, 22, 23 Fail to Start/Run

Under degraded power conditions, failures of the pumps to start and then run for the required period of time constitutes the dominant failures. For loss of one bus, the total system failure involves failure of the remaining motor driven train and failure of the turbine train, or failure of the water supplies. Pump surveillance and lineups should be observed or reviewed to minimize this failure.

2. Pumps 21, 22, 23 Unavailable Due to Maintenance

Under degraded power conditions, pump maintenance is also a significant contributor to risk. With no AC power available, the turbine pump maintenance is a single failure event. For these reasons pump maintenance should be streamlined and minimized. Observation or review of maintenance procedures or activities will address this problem.

3. Operator Fails to Feed with Turbine Pump 22

When the auxiliary feedwater pumps are started Pump 22 comes up to a minimum speed. The operator is then required to raise the governor speed setting in order to use the turbine driven pump for feeding. This failure, combined with Pump 21 or 22 unavailability or loss of all AC power, is significant. Operator awareness of this, and attention to AFW pump status should be evaluated.

4. PCV 1139 Fails to Open

This valve is the air operated steam supply valve for turbine driven Pump 22. On loss of all AC power this failure would fail all feedwater. On loss of one AC bus, only one pump would be left for feeding the steam generators. Surveillance of this valve should be observed or reviewed to enhance its availability.

5. PCV 406 A, B, C, or D Fails to Open

With loss of one AC bus and unavailability of the turbine driven pump, failure of one of these motor driven pump discharge valves would fail steam generator cooling. Observation or review of valve function during surveillance will minimize unavailability when needed.

TABLE 10-2. I&E MODULES FOR AUXILIARY FEEDWATER SYSTEM INSPECTION

Module	Title	Components	Failure Mode[a]
61701	Surveillance (complex)	Pumps 21, 22, 23	1
		PCV 1139	4
		PCV 406 A, B, C, D	5
		BFD 36, 38, 41, 43	9
		BFD 48-1,-3, -5, -7	9
61726	Monthly Surveillance Observation	CT64	6
		LCV 1158	7
		PCV 1139	4
		FCV 406 A, B, C, D	5
		Pumps 21, 22, 23	1
		BFD 36, 38, 41, 43	9
		BFD 48-1, -3, -5, -7	9
62700	Maintenance	Pump 21, 22, 23	2
62703	Monthly Maintenance Observation	Pumps 21, 22, 23	2
71707	Operational Safety Verification	PCV 1187,1188, 1189	8
		BFD 36, 38, 41, 43	9
		BFD 48-1, -3, -5, -7	9
		FCV 405 A, B, C, D	3
		Pump 22	3
		Pumps 21, 22, 23	1
71710	ESF System Walkdown	CT6, 64	6
		CT 49	10
		BFD 36, 38, 41, 43	9
		BFD 48-1, -3, -5, -7	1
		Pumps 21, 22, 23	1

a. See Table 10-1 for failure mode identification.

TABLE 10-3. (continued)

Component	Required Position	Actual Position
Auxiliary Boiler Feed Pump 23 Discharge to Steam Generator 23		
BFD-62-2 FCV-406C Inlet Stop	Open	_____
BFD-41 FCV-406C Outlet Stop	Open	_____
Auxiliary Boiler Feed Pump 23 Discharge to Steam Generator 24		
BFD-62-3 FCV-406D Inlet Stop	Open	_____
BFD-43 FCV-406D Outlet Stop	Open	_____
Auxiliary Boiler Feed Pump Breakers		
Pump 21 (Bus 3A) Breaker	Racked In	_____
Control Fuses	In	_____
Pump 23 (Bus 6A) Breaker	Racked In	_____
Control Fuses	In	_____
Local Control Panels		
Auxiliary Boiler Feed Pump 21 Transfer	Remote	_____
Switch Indicating Light	Off	_____
Auxiliary Boiler Feed Pump 22 Transfer	Auto	_____
Switch Indicating Light	Off	_____
Auxiliary Boiler Feed Pump 23 Transfer	Auto	_____
Switch Indicating Light	Off	_____
Condensate and Boiler Feedwater Supervisory Panel (SC Panel)		
Auxiliary Feedwater Pump 21	Auto	_____
Indicating Light	Off	_____
Auxiliary Feedwater Pump 22	Auto	_____
Indicating Light	Off	_____
Local Control Switch Indicating Light	Amber Light Off	_____

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This report contains information on the development of an application of probabilistic risk assessment (PRA) to the inspection of nuclear power plants. The current inspection program was analyzed to determine the appropriate method for interjecting PRA information. Once the method was determined, a trial application was performed for the Indian Point Unit 2 inspection effort. Recommendations for further development and application are given.

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