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PRA Applications Program for Inspection at Arkansas Nuclear One Unit 1

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Pacific Northwest Laboratory
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ABSTRACT

The level one PRA for ANO-1 has been analyzed to identify plant systems and components important to minimizing public risk, as measured by system contributions to plant core melt frequency, and to identify the primary failure modes of these components. This information has been tabulated, and correlated with inspection modules from the NRC Inspection and Enforcement Manual. The report presents a series of tables, organized by system and prioritized by risk importance, which identify components associated with 98% of the inspectable risk due to plant operation. The systems addressed, in descending order of risk importance, are: DC Power, High Pressure Injection, Low Pressure Injection, Service Water, Reactor Protection, Emergency Feedwater, Vital AC Power, Safety Relief Valves, Main Feedwater, and Emergency Feedwater Initiation and Control. This ranking is based on the Fussel-Vesely measure of risk importance, i.e., the fraction of the total core melt frequency which involves failures of the system of interest.

SUMMARY

The PRA Applications Program for inspection at ANO-1 was performed for the NRC at Pacific Northwest Laboratory. This program applies a previously developed methodology to identify and present information which is useful for the planning and performance of powerplant inspections.

The level one PRA for ANO-1 (Kolb et al. 1982) has been analyzed to identify plant systems and components important to minimizing public risk, as measured by system contributions to plant core melt frequency. This information has been tabulated and correlated with inspection modules from the NRC Inspection and Enforcement Manual (NRC 1984) which are used by inspectors in the planning and performance of inspections. The body of this report consists of a series of tables, organized by system and prioritized by risk importance, which identify components associated with 98% of the core melt probability resulting from plant operation.

Following a section describing important accident initiators and sequences identified in the PRA, tabulations are presented for ten systems. These system tables are ordered by system risk importance, as measured by the fraction of the total core melt probability associated with failures of each system. Three tables are presented for each system. The first table presents the failure modes identified in the PRA for each important system component. The second table correlates each component with the IE inspection modules most related to ensuring component reliability. The third table provides a modified system check off list identifying the proper line-up of each component during normal operation.

The tabulations were developed by the following analysis procedure. First, the plant systems were ordered according to system risk importance. To accomplish this, the dominant cut sets representing more than 98% of the core melt probability were listed, and the fraction of the total core melt probability which involved failures of components from each system was calculated [this is the Fussel-Vesely Importance measure (Henley 1981)]. Systems were then selected from the ordered list until more than 98% of the core melt probability was accounted for. Second, for each selected system, the fault tree from the PRA was reanalyzed to rank system components according to their importance to system failure. For each system, components were selected for inclusion in the tabulations until more than 95% of the system failure probability had been addressed.

The tables thus present, in decreasing order of system importance, the failure modes, applicable inspection modules, and a check off list of normal operational state for all components associated with 98% of the core melt probability associated with plant operation. This information allows an inspector to readily identify important systems and components when developing an inspection, plan, and when walking down systems in the plant.

The information presented in this document allows an inspector to concentrate his efforts on systems important to the prevention of core melt.

However, it is essential that inspections not focus exclusively on these systems. Other systems which perform essential safety functions, but are absent from the tables because of high reliability and redundancy, must also be addressed to ensure that their importance is not increased by allowing their reliability to decrease. A balanced inspection program is essential. This information represents but one of the many tools to be used by experienced inspectors.

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ACKNOWLEDGMENTS

Thanks are extended to Dave Campbell and Randy Kirchner of JBF Associates, Inc., for information which they provided concerning the effects of plant modifications made since the PRA for ANO-1 was performed. This information was developed during their work on a plant-specific version of the interactive PRA analysis code PRISIM. They used the ANO-1 PRA as a starting point, but modified the analysis somewhat as a result of extensive on-site studies. They provided us with modified system fault trees and revised availability data developed in their work, which we used in our analysis to ensure that our results are as up to date as possible.

Thanks are also extended to our Project Manager from NRC Region 1, Bernie Hillman, for inviting us to join the ongoing program to which this analysis contributes. We also wish to thank our colleagues at Brookhaven National Laboratory and at the Idaho National Engineering Laboratory for many discussions. In particular, we thank Ron Wright of INEL for providing us with a version of the IRRAS fault tree analysis code specially adapted for use on an IBM/PC.

1.0 INTRODUCTION

This work was performed for the U.S. Nuclear Regulatory Commission (NRC) as part of an extensive program to develop information based on probabilistic risk analyses (PRAs) for use in the planning and performance of nuclear powerplant inspections. Due to the broad scope of this program, project work has been divided among three national laboratories, each of which concentrates upon a particular reactor type. Thus, Brookhaven National Laboratory analyzes plants powered by boiling water reactors (BWRs), and at Idaho National Engineering Laboratory analyzes pressurized water reactor plants (PWRs) built by Westinghouse. Pacific Northwest Laboratory (PNL) analyzes PWRs from both Babcock and Wilcox and Combustion Engineering, due to the smaller number of plants from these vendors.

In this particular project, information from the Arkansas Nuclear One Unit 1 (ANO-1) PRA (Kolb et al. 1982) has been used to identify plant systems and components important to minimizing the probability of core melt, and to identify failure modes for these components. This information has been tabulated and correlated with inspection modules from the NRC Inspection and Enforcement (IE) Manual (USNRC 1984) which are used by inspectors in the planning and performance of inspections. The body of this report consists of a series of tables, organized by system and prioritized by system importance, which identify components associated with 98% of the plant core melt probability.

Previous studies in this program (Hinton and Wright 1986, Higgins 1986) have addressed how PRA-based information may be best incorporated into inspection planning, performance and evaluation. The conclusion of this previous work was that the existing IE Manual provides a logical and effective framework for inspection planning. This manual contains an extensive sequence of inspection procedures, or modules, addressing functional areas such as calibration, surveillance, maintenance, ESF system walkdown, etc. It also contains a methodology for selecting inspection modules for performance, plus guidance on the frequency at which modules should be performed. It was concluded that this manual should be retained as the general framework for inspection planning. PRA-based information, which is necessarily plant specific, should be provided for each plant. This information should then be used in the inspection planning process to help focus on areas where public risk is most sensitive to performance degradation.

The NRC program is, therefore, directed towards the preparation of a series of plant-specific appendices to the IE Manual which contain plant-specific information of a common type and safety significance. These appendices are structured according to a common format. Each appendix begins with a description of accident initiators and sequences important at the plant. This is followed by a listing of plant systems associated with 98% of the plant core melt probability, which is ordered according to the importance of each system to plant damage. For each system addressed, the components associated with 95% of the probability of system failure are identified and ranked according to importance. Three tables are presented for each system.

The first identifies the failure modes by which each component contributes to plant damage. The second correlates each component with the IE inspection modules most related to ensuring component reliability. The third provides a modified system check-off list identifying the proper line-up of each component during normal operation. The body of this report presents the plant-specific appendix developed for the ANO-1 plant. It follows the format described above.

PRAs have been performed for less than one quarter of the nation's nuclear plants. Consequently, a significant aspect of the NRC program addresses the development of generic insights which may be utilized to guide inspection planning for plants without a PRA. As plant-specific appendices are developed the information is reviewed to identify dominant generic contributors to risk including: initiating events, accident sequences, important systems and components, component failure modes, significant human errors, and common cause failures.

The compilation of generic insights resulting from the analysis of PRAs indicates systems and components which may have risk importance at other plants. For application to a specific site, plant-specific information must be used to evaluate the relevance and applicability of the generic insights. For instance, important functions may be performed by different systems at different plants, or, systems may be either more vulnerable (single failure dependencies) or less vulnerable (redundancies) at different plants. PNL has performed an analysis of the Rancho Seco plant (no PRA), using the results of PRAs for the ANO-1 and Oconee-3 plants, plus a detailed comparison of system designs at the three plants (Gore and Huenefeld 1987). EG&G and Brookhaven are performing similar studies using generic insights and plant-specific information to address plants for which PRAs have been performed (Higgins et al. 1987). Future comparison of results from those studies with results obtained from analyzing the plant-specific PRAs will provide an indication of how effective this approach is in identifying important systems and components.

As was noted above, this document reports the results of a detailed analysis of the PRA performed for the ANO-1 plant. It was not necessary to utilize generic insights in the performance of this analysis. Rather, the results of this study will contribute to the database of generic information to be utilized in the analyses of plants which lack PRAs. The analysis approach used in this study is discussed in the following Section 2.0. The results of the analysis are presented in Sections 3.0 and 4.0, according to the above-described format for plant-specific appendices to the IE Manual.

2.0 ANALYSIS OF THE ANO-1 PRA

The analysis required three major steps to produce the tables presented in Section 4. The first step was the calculation of risk importance for each system from information in the PRA. This was used to select systems to be analyzed for component importances. The second step was the re-analysis of system fault trees from the PRA to calculate component importances. The third step was the correlation of components and their dominant failure modes with inspection modules relevant to maintaining component reliability. These steps are discussed below.

2.1 CALCULATION OF SYSTEM IMPORTANCES

The selection of systems for detailed fault tree analysis required that they be ranked according to an appropriate measure of risk. The ANO-1 PRA is a level 1 PRA. Core melt probability is addressed in detail, with only a limited analysis of subsequent containment failure mechanisms, and radionuclide releases to the public. Consequently, for this study core melt frequency is used as the risk measure used to rank system importance.

The Fussel-Vesely (F-V) Importance measure (Henley 1981) applied to core melt frequency was selected as the specific risk measure used to rank systems and components. The F-V Importance is the fraction of the total risk (core melt frequency) which results from failures involving the system or component of interest. Thus, high values of F-V Importance identify systems which are the greatest contributors to risk. In addition, the increase in risk due to a given percentage increase in system failure probability is also highest for systems with highest F-V Importance values. Thus, this measure identifies not only the systems which are the greatest contributors to risk, but also those for which risk is most sensitive to performance degradation. It is therefore the logical measure to use for ranking system importance for inspection attention, to ensure that safety performance is maintained.

Appendix C of the ANO-1 PRA presents a detailed listing of initiating events and cut set elements, and associated unavailabilities (both with and without recovery factors). The dominant cut sets presented in the body of the PRA were selected from this list. This listing of about 500 cut sets was analyzed to determine system importances. Each element of each cut set was analyzed to determine what system was responsible for the root cause failure represented by the cut set element. Core melt frequencies associated with each cut set were input to a spread sheet data file (recovery factors from the PRA were included). This file was then manipulated into system-based sub-files, each of which included data only from cut sets involving failures of components in a given system. The F-V Importance of each system was then calculated by summing the sub-files and dividing by the total from all of the cut sets.

One significant deviation from the results of the PRA was made in the system importance analysis. This deviation resulted from changes made to the

plant since performance of the PRA. At the time the PRA was performed, the loss of one vital AC or DC bus would trip the plant and fail the power conversion system. Consequently, loss of any of these buses was an initiating event, which could lead to core melt via appropriate cut sets listed in the PRA. Plant modifications made since the PRA was performed removed the dependence of normal plant operations on the vital bus availability, such that loss of a vital bus no longer results in a trip of the reactor. Consequently, vital bus loss should no longer be included as an initiating event in the PRA. Since our objective was to develop information that is as up to date as possible, it was decided to eliminate from the analysis all cut sets resulting from vital AC or DC bus loss initiators (initiators T(A3), T(D01) and T(D02) in the PRA).

The obvious effect of eliminating the electrical-transient initiated cut sets (about 100 cut sets) from the system importance calculations was to reduce the importance of the vital AC and DC power systems. However, in the PRA analysis the electrical upsets were also associated with a loss of the Main Feedwater System (MFW). Many of the eliminated cut sets involved subsequent loss of Emergency Feedwater (EFW) or failure of the Emergency Feedwater Control System (EFC), followed by the opening of Safety Relief Valves (SRVs) and their failure to reseat. This resulted in the calculation of lower importance values for the EFW, EFC and SRV systems than would have been calculated using the unmodified results of the PRA. The positions of these systems in the risk-prioritized ranking have therefore been lowered; however these systems have not been displaced from the list by this modification.

The systems tables in Section 3 are presented in order of calculated system importance. DC Power has the highest F-V Importance, primarily due to the high probability of a single cut set representing common mode failure of both station batteries following a loss of offsite power. High Pressure Injection (HPI) and Low Pressure Injection (LPI) follow, with all three of these systems having an Importance exceeding 20%.

The Service Water System (SWS), Reactor Protection System (RPS), EFW System and Vital AC Power Systems follow with importance values between 8% and 12%. They are followed by the Safety Relief Valves, the Power Conversion System (PCS) and the EFC Systems, all of which have Importance values of 5% or less.

2.2 CALCULATION OF COMPONENT IMPORTANCES

Construction of the tables presented in Section 4 of this report required the identification of components associated with at least 95% of the system failure probability for each of the systems selected for anlaysis. This required a reanalysis of the fault trees presented in Appendix B of the PRA document to identify the components most important to system failure. It was not possible to extract information with this degree of detail from the cut sets published in the PRA because, in general, the cut set elements were not basic events. Instead, many contained "module" elements, which combined the effects of several possible failures causing the final result (i.e., failure

of a pump, or of its suction or discharge valves located in a single run of piping, any of which would prevent flow through the line).

For systems selected for analysis, the system fault trees were reanalyzed using the Integrated Reliability and Risk Analysis (IRRAS) computer code (Russel et al. 1987) run on an IBM-PC. For all but six systems, the fault tree analyzed was that published in this PRA. However, modified fault trees were used for the HPI, LPI, SWS, EFW, AC, and EFIC systems in order to incorporate the effects of plant modifications made since the PRA was completed. These fault trees had been modified during work to develop and apply the PRISIM interactive PRA analysis code (Kirchner et al. 1986) at the ANO-1 plant. They were provided by the authors of PRISIM through the courtesy of J.B.F. Associates, Inc. Fault tree gates and component reliability data were input to the code and processed with an integrated fault tree analysis package. IRRAS identified the dominant minimal cut sets, and quantified the fault trees by ordering cut sets by probability. IRRAS also calculated the F-V Importance of both cut sets and of system component failures. The calculated importance of the component failures was then used to select components for inclusion in the tables. For all systems analyzed, components comprising more than 95% of the total component importance were selected for tabulation.

Considerable care was required in checking the input information supplied to IRRAS for analysis, because the code version available to us lacked error-diagnostic capability to check that the input produced a coherent fault tree. Thus, an input error could effectively remove a segment of a tree from analysis. In addition to careful checking of input against the reference fault tree, calculated system and module event failure probabilities were subsequently compared against values published in the PRA. Final values calculated agreed well with published values for all systems.

2.3 PREPARATION OF TABLES

For each system, the components selected for inclusion in the tables were grouped according to type for discussion of failure modes (e.g., pump suction and discharge MOVs in parallel trains). For many components, cut set elements indicated more than one failure mode (e.g., failure to operate, operator failure to initiate, inappropriate change of position). These failure modes were grouped and discussed for each component type in the system failure mode identification tables.

The characteristics of each component were assessed to determine what types of inspection would be most appropriate for ensuring component reliability. This information was then used to prepare a table for each system correlating each of the relevant IE inspection modules with components which should be addressed when the module is applied to the system. This table also contained a cross correlation to the failure modes which would be minimized by the given type of inspection. For instance, pump failure to start and run is addressed in modules for Surveillance, Operational Safety Verification, and ESF System

Walkdown. It is also addressed through the Maintenance module, in terms of minimizing unavailability due to maintenance scheduling and work.

For each system, an abbreviated system walkdown table was prepared addressing only the selected system components. This table identifies the normal operating state or position of each component determined to be risk significant from the PRA. It was compiled using information from the PRA, and also from plant system descriptions, operator training information, and plant drawings. In many cases it was possible to correlate and verify this information using system lineup tables from plant operating procedures. In general, these tables are considerably shorter than lineup tables in procedures. They therefore allow an inspector with limited available system walkdown time to concentrate on risk-significant components, without concern that he may be overlooking something important.

2.4 CONCLUSIONS AND RECOMMENDATIONS

In this project, we have identified the systems and components most important to public risk during operation of the ANO-1 powerplant. They are identified in Tables 4.1 through 4.10. Systems are addressed in the order of decreasing importance, as determined by the fraction of the total core melt frequency which involves the failure of each system. This information has been developed from the PRA analysis of the ANO-1 plant (Kolb et al. 1982). An attempt has been made to incorporate the effects of plant changes made since performance of the PRA. Subsequent plant changes may require that these results be further updated.

The DC Power, HPI and LPI Systems are the most important systems for minimization of core damage. Systems of intermediate importance include the Service Water System, RPS, EFW and Vital AC Power Systems. Lower importance systems include the Safety Relief Valves, Power Conversion (i.e., MFW) System and the Emergency Feedwater Control System.

The information in these tables allows an inspector to identify quickly the components most important to public risk--a combination of failure probability and of the consequences of the failure. This information allows him to direct his attention to these components preferentially. In particular, by using the system walkdown tables he can rapidly review the line up of important system components on a routine basis. He may also use these tables when selecting systems for the performance or more detailed inspection activities.

In using these tables, however, it is essential to remember that other systems are also important. If, through inattention, the failure probabilities of other systems were allowed to increase significantly, their risk significance might exceed that of systems in the tables. Consequently, a balanced inspection program is essential to minimizing plant risk. The tables allow an inspector to concentrate on systems of highest risk importance. In so doing, however, he must maintain cognizance of the status of systems performing other essential safety functions, and ensure that their reliability is maintained.

3.0 IMPORTANT ACCIDENT INITIATORS AND SEQUENCES

Two basic types of accident initiators are addressed in the PRA: Loss of Coolant Accidents (LOCAs), and transients. However, subsequent event sequences leading to core melt are not distinct, because each of the transient types addressed has the potential for inducing LOCA events. Table 3.1 identifies the event types, and presents the core melt frequency per year of operation estimated in the PRA document for events of each type.

TABLE 3.1. Initiating Event Categories

Initiating Events	Mean annual core-melt frequency
very small (0.4" to 1.2" dia.)	1.0E-5
small (1.2" to 1.7" dia)	1.2E-6
medium (1.7" to 4.0" dia)	1.6E-6
Transients	
Loss of offsite power	1.1E-5
Power conversion system upset (MFW)	2.0E-6
Trip; all front line systems OK	3.8E-6

Table 3.1 presents only the initiating events which the PRA analysis found to cause a core melt frequency greater than 10^{-7} per year. Other initiators were addressed in the PRA (e.g., large LOCA, loss of Service Water). As was discussed in Section 2.0, transients initiated by loss of vital AC and DC buses were addressed in the PRA, but have not been included in this analysis. Once again, the reason for this is that plant modifications made since the PRA was performed have removed loss of these vital buses from the category of initiating events (because they would not now result in reactor trip). For the unmodified plant, core melt frequencies for transients initiated by the vital bus loss events were similar in magnitude to the frequencies presented in Table 3.1 for other transient initiating events.

The following discussion presents the various types of event sequences identified in the PRA as most likely to lead to core melt following occurrence of these initiating events. In several cases, there is more than one type of sequence which may lead to core melt following a given initiating event.

3.1 VERY SMALL LOCA (0.4" to 1.2" dia.)

1. The very small LOCA is followed by a loss of all HPI, resulting in core uncovery and melting. The dominant HPI failure mode is operator failure to initiate the system. Other HPI system failure modes involve local faults of the HPI, LPI (pump suction) and SWS systems.

2. The very small LOCA is followed by loss of EFW, and then by loss of HPI during operation in the recirculation mode. EFW system failure modes are local system failures or failures of the EFIC control system. The dominant HPI system failure mode is due to loss of cooling by SWS. Others are due to local faults in the LPI (pump suction), AC Power, and DC Power systems.

3.2 SMALL LOCA (1.2" to 1.7" dia.)

1. The small LOCA is followed by loss of HPI during operation in the recirculation mode. The dominant HPI system failure mode is failure of the operator to switch HPI pump suction from the BWST to the RB sump through the LPI system when the BWST is depleted. Other HPI failure modes involve local faults of the LPI (pump suction), SWS and AC Power systems.

3.3 MEDIUM LOCA (1.7" to 4.0" dia.)

1. The medium LOCA is followed by loss of HPI during operation in the injection or recirculation mode. The dominant HPI failure mode is failure of the operator to switch HPI pump suction from the BWST to the RB sump through the LPI system when the BWST is depleted. Other HPI failure modes involve local faults of the LPI (pump suction), SWS, Pump Room Cooling and AC Power systems.

3.4 LOSS OF OFFSITE POWER

1. Loss of offsite power fails MFW. It is followed by loss of EFW, the opening and failure to reseat of one or more safety relief valves (transient induced LOCA), and the failure of one or more train of HPI. EFW failure modes are local faults in the system or failure of the EFIC control system. Failure modes for one train of HPI include local faults in the DC Power system (dominant), the AC Power system, or the SWS system.

2. Loss of offsite power fails MFW. It is followed by loss of EFW and loss of HPI. The dominant failure mode is common mode failure of both station batteries (DC Power) which fails both EFW and HPI. Important failure modes for failure of both trains of HPI include failure of both emergency diesel generators (AC Power) or local

faults in both trains of DC power. Local faults in one train of HPI or of LPI (pump suction) or of SWS (pump cooling), in combination with faults in the opposite train of AC or DC power are also important modes for total failure of HPI. EFW failure modes are local faults in the system or failure of the EFIC control system.

3.5 POWER CONVERSION SYSTEM UPSET (MFW LOSS)

1. MFW loss is followed by the opening and failure to reseat of one or more safety relief valves (transient-induced LOCA). EFW may or may not fail, depending on the transient, and HPI fails. If EFW fails, failure of one train of HPI leads to core melt. HPI failure modes include operator failure to switch suction to the RB sump through the LPI system on BWST depletion, and local faults in the HPI, LPI (pump suction), SWS and AC power systems. EFW failure modes are local faults in the system or failure of the EFIC control system.

2. MFW loss is followed by failure of the EFW and HPI systems. EFW and HPI failure modes are as above.

3.6 REACTOR TRIP WITH ALL FRONT LINE SYSTEMS AVAILABLE

1. Reactor trip is followed by the independent failure of MFW, the opening and failure to reseat of at least one SRV (transient induced LOCA), and failure of at least one train of HPI. EFW may or may not fail, depending on the transient. EFW failure modes are local faults in the system or failure of th EFIC control system. HPI failure modes include local faults in DC Power, AC Power, SWS and HPI system components, and operator failure to switch suction to the RB sump through the LPI system after BWST depletion.

2. Reactor fails to trip on receipt of a valid trip signal, and HPI fails preventing reactor shutdown by borated water injection or makeup of primary coolant losses. RPS system failure modes are double circuit breaker failures. The dominant failure mode of the HPI system is operator failure to actuate it.

4.0 SYSTEM INSPECTION PLANS

Tables are presented for each of the systems selected in the analysis which identify important system failure modes, IE modules applicable to the inspection of system components, and the required position of each important component during normal system operation (i.e., system walkdown checklist). The systems are presented in decreasing order of risk importance, and together comprise more than 98% of the public risk associated with plant operation.

4.1 DC POWER SYSTEM

TABLE 4.1A. DC POWER SYSTEM FAILURE MODE IDENTIFICATION

The DC system at ANO-1 provides continuous power for control, instrumentation, reactor protection systems and engineered safeguards actuation systems, and emergency safeguard actuation control systems. As part of its function, the DC power system provides control power for the diesel generators for the emergency AC electrical system. In addition, it powers the control valves in the emergency feedwater system and provides control power to the Emergency AC circuit breakers for the 4160 V switchgear, and 480 V load centers. It also powers the inverters supplying the 120 V vital AC system.

Conditions that Lead to Failure

1. Failure of Battery Chargers D03, D04, or D05

   Three battery chargers (D03, D04, and D05) are supplied with two (D03 and D04) serving as normal supplies to the DC buses with the associated battery floating on the bus. The third battery charger (D05) serves as a standby battery charger to either bus. Combined unavailability of battery chargers and battery sets can prevent DC power from being supplied to the DC buses. Periodic testing, maintenance, and surveillance in accordance with Technical Specifications requirements will help maintain availability, as will attention to minimizing prolonged maintenance activities. Operator training and awareness of Emergency Operating Procedures will enhance the probability of recovery.

2. Failure of Distribution Panels D11 and D21

   Two distribution panels (D11 and D21) are provided, supplying the DC instrumentation and control power for the plant. Failures of these panels can prevent electrical power from being supplied to the respective loads. Periodic testing, and maintenance should be observed and reviewed, and appropriate breaker lineups maintained.

3. Failure of DC Buses D01 and D02

   Two DC buses (D01 and D02) are provided for vital instrumentation, distribution panels, emergency lighting and motors. Failures of these buses would usually be associated with subcomponent failures in the control circuit, or improper breaker positions for automatic operation. Maintenance, surveillance, and system lineup should be observed and/or records of these activities should be reviewed.

TABLE 4.1A. (continued)

4. 120 Volt Vital AC Distribution Panels RS1, RS2, RS3, and RS4 Unavailable

The DC system powers four redundant 120 V vital AC distribution panels through inverters to supply power for nuclear instrumentation, reactor protection systems, engineered safeguard actuation systems and other vital loads. Failure of these panels can fail these vital loads. Maintenance and surveillance should maintain availability, and operator training and awareness will enhance the probability of recovery.

5. Failure of 480 Volt Motor Control Centers (MCC) B51 and B61

Failure of the MCC causes loss of power to the battery. In combination with insufficient power from the batteries, this may cause loss of power to the DC buses. Periodic testing and maintenance should be performed to maintain availability.

6. Failure of Static Inverter Y11, Y13, Y22, and Y24

These inverters convert DC power to AC power for 120 V distribution panels. Failures of these inverters may result from electronic component failures or DC bus failures. Observation of maintenance, surveillance, or review of the records of these functions should be performed.

7. Battery Sets D06 or D07 Unavailable

A battery set being unavailable either due to component outage or scheduled or unscheduled maintenance combined with failure of offsite power sources and emergency diesel generator failures can prevent electrical power from being supplied to the buses. Periodic testing of battery voltage and specific gravity, according to Technical Specifications, and proper maintenance should maintain the probability of failure at a low value.

8. Normally Closed Circuit Breakers Fail Open

Failure of normally closed circuit breakers in the open position leads to loss of power to associated buses or distribution panels. Periodic maintenance, and verification of system lineup should be observed or reviewed.

TABLE 4.1B. IE MODULES FOR DC POWER SYSTEM INSPECTION

Module	Title	Components	Failure Mode[a]
41700	Training	Battery Chargers D03 D04, D05	1
		Panels RS1, RS2, RS3 RS4	4
61701	Surveillance (Complex)	DC Buses D01, D02	3
		Battery Sets D06, D07	7
61725	Surveillance Testing and Calibration Program	Battery Chargers D03, D04, D05	1
		Panels RS1, RS2, RS3 RS4	4
		Battery Sets D06, D07	7
61726	Monthly Surveillance Observation	Battery Chargers D03, D04, D05	1
		DC Buses D01, D02	3
		Panels RS1, RS2, RS3, RS4	4
		Inverters Y11, Y13, Y22, Y24	6
		Battery Sets D06, D07	7
		Transformer Circuit Breakers	8
62700	Maintenance	Battery Chargers D03, D04, D05	1
		Distribution Panels D11, D21	2
		DC Buses D01, D02	3
		Panels RS1, RS2, RS3, RS4	4
		Motor Control Centers B51, B61	5
		Inverters Y11, Y13, Y22, Y24	6
		Battery Sets D06, D07	7
		Transformer Circuit Breakers	8

TABLE 4.1B. (continued)

Module	Title	Components	Failure Mode[a]
71707	Operational Safety Verification	Battery Chargers D03, D04, D05	1
		Distribution Panels D11, D21	2
		DC Buses D01, D02	3
		Panels RS1, RS2, RS3, RS4	4
		Motor Control Centers B51, B61	5
		Inverters Y11, Y13, Y22, Y24	6
		Battery Sets D06, D07	7
		Transformer Circuit Breakers	8
71710	ESF System Walkdown	Battery Chargers D03, D04, D05	1
		Distribution Panels D11, D21	2
		DC Buses D01, D02	3
		Panels RS1, RS2, RS3, RS4	4
		Motor Control Centers B51, B61	5
		Inverters Y11, Y13, Y22, Y24	6
		Battery Sets D06, D07	7
		Transformer Circuit Breakers	8

(a) See Table 4.1A for failure identification

TABLE 4.1C. DC POWER SYSTEM WALKDOWN

Component Number	Component Name	Location	Required[a] Position	Actual Position
D01 (112)	Panel D11 Feed from D01 or D02	D01	Position to feed from D01	_____
D02 (212)	Panel D21 Feed from D02 or D01	D02	Position to feed from D02	_____
122A	Supply from Battery Charger D03 Breaker	D01	Closed	_____
122B	Supply from Battery Charger D05 Breaker	D01	Closed	_____
123	Panel RA1 Breaker	D01	Closed	_____
124	Emergency Supply to Panel D-21	D01	Closed	_____
152A	DC Supply to Inverter Y-11	D01	Closed	_____
152B	DC Supply to Inverter Y-13	D01	Closed	_____
222A	Supply from Battery Charger D04 Breaker	D02	Closed	_____
223	Panel RA2 Breaker	D02	Closed	_____
224	Emergency Supply to Panel D-11	D02	Closed	_____
242A	DC Supply to Inverter Y-22	D02	Closed	_____
242B	DC Supply to Inverter Y-24	D02	Closed	_____
5141A	Inverter Transfer Switch Y-11	B51	Closed	_____
5141B	Inverter Y-11	B51	Closed	_____
5145A	Inverter Transfer Switch Y-13	B51	Closed	_____
5145B	Inverter Y-13	B51	Closed	_____
5193A	Battery Charger D03	B51	Closed	_____
5622B	Battery Charger D05	B56	Closed	_____
6121A	Inverter Transfer Switch Y-22	B61	Closed	_____
6121B	Inverter Y-22	B61	Closed	_____

TABLE 4.1C. (Continued)

Component Number	Component Name	Location	Required[a] Position	Actual Position
6143A	Battery Charger D04	B61	Closed	_____
6145A	Inverter Transfer Switch Y-24	B61	Closed	_____
6145B	Inverter Y-24	B61	Closed	_____

(a) Due to the integrated nature of the DC power system failure mode, power available to the buses, distribution panels, and batteries should be verified.

4.2 HIGH PRESSURE INJECTION SYSTEM

TABLE 4.2A. HIGH PRESSURE INJECTION SYSTEM FAILURE MODE IDENTIFICATION

The High Pressure Injection (HPI) system is designed to perform both normal and emergency functions in several modes of operation. Under normal conditions, the HPI system is known as the Makeup and Purification (MU) system. In this mode the suction source for the pump is the reactor coolant system via the makeup tank. During an accident, the pump suction is realigned to the Borated Water Storage Tank (BWST) and borated water is injected into the reactor vessel via the Reactor Coolant System (RCS) cold legs. This prevents uncovering of the core for small reactor coolant piping leaks where high system pressure can be maintained, and delays uncovering of the core for intermediate-sized leaks. In addition, HPI provides makeup to the reactor coolant system during depressurization following a reactor shutdown due to a transient initiating event and thus prevents core uncovery resulting from coolant volume shrinkage.

Conditions that Lead to Failure

1. Operating Pump P36A, B, C Fails to Run or Standby Pumps Fail to Start and Run

   Failure of any combination of two out of three pumps can prevent sufficient cooling water flow to the RCS cold legs. The failure causes are random hardware or electrical failures of these three pumps or failure of the pump room cooling units. Testing, maintenance, and surveillance of the pumps which are not in use according to Technical Specifications should maintain the availability of these pumps.

2. Failure of Pump Suction Manual Valves MU-18A, B and C

   These are manual valves in the HPI system suction lines. They must be locked open during normal operation or emergency conditions. The important failure causes are random hardware failures. Maintenance of these valves should be reviewed or observed, and valves should be returned to standby position and verified.

3. Failures of Pump Discharge Manual Valves MU-20A, B, or C, and Check Valves MU-19A, B or C

   The required position for these valves is "open" during normal operation or emergency condition. The dominant failure cause is random hardware failure. A contributing cause is manual valve closure by operators during maintenance or testing. Maintenance and testing of these valves should be observed or reviewed, and operator awareness of the importance of restoration of proper lineup ascertained.

TABLE 4.2A. (continued)

4. BWST Discharge Valves CV-1407 and CV-1408 Fail to Open on Demand

These are the motor-operated discharge valves for the BWST. They must be open following a LOCA. The dominant failure cause is random hardware failure. A contributing cause is operator failure to open these valves if actuation fails. Power availability, operator awareness, surveillance, and maintenance of these valves should be reviewed and observed to maintain reliability.

5. Motor-Operated Discharge Valves CV-1219, 1220, 1227, and 1228 Fail to Open on Demand

The flow of HPI system to the RCS cold legs is controlled by means of motor-operated valves CV-1219, 1220, 1227, and 1228. Failure of these valves in "closed" position will prevent sufficient coolant injection to the cold legs. The important cause is multiple hardware failures. Operator failure to actuate valves manually also affects probability of recovery from these failures. Power availability, operator awareness, surveillance, and maintenance of these valves should be reviewed and observed to maintain reliability.

6. Check Valves MU-1211, 1213, 1214, and 1215 Fail Closed

These are discharge header check valves for HPI. Failure of these valves in closed position will prevent HPI flow to the designated cold leg. Testing and maintenance of these valves according to Technical Specifications should maintain the availability of these valves.

7. Operator Fails to Initiate HPI and Establish Feed-and-Bleed for Small Loss-of-Coolant Accident (LOCA)

During an accident, the HPI pump suction is realigned to the BWST and borated water is injected into the reactor vessel via the RCS cold legs. This HPI prevents uncovering of the core for small LOCA where high system pressure can be maintained. The dominant failure is operator failure to initiate the HPI and establish feed-and-bleed. Operator awareness of criteria for HPI initiation and adherence to emergency procedures is important.

8. Manual Valves BW-2 and BW-3 Fail Closed

These failures result primarily from operator failure to restore valves to the open position after testing, and failure to discover the error. The valves are manual suction header valves. They should be locked open. These operator errors are addressed by proper post-test surveillance, which should be reviewed and observed.

TABLE 4.2B. IE MODULES FOR HIGH PRESSURE INJECTION SYSTEM INSPECTION

Module	Title	Components	Failure Mode[a]
41700	Training	Valves CV-1407, 1408	4
		Valves CV-1219, 1220 1227, 1228	5
		Emergency Procedures	7
		Valves BW-2, 3	8
61701	Surveillance (Complex)	HPI Pumps 36A,B,C	1
61725	Surveillance Testing and Calibration Program	HPI Pumps 36A,B,C	1
		Valves CV-1407, 1408	4
		Valves CV-1219, 1220 1227, 1228	5
		Valves MU-1211, 1213 1214, 1215	6
61726	Monthly Surveillance Observation	HPI Pumps 36A,B,C	1
		Valves CV-1407, 1408	4
		Valves CV-1219, 1220 1227, 1228	5
		Valves BW-2, 3	8
62700	Maintenance	HPI Pumps 36A,B,C	1
		Valves MU-18A,B,C,	2
		Valves MU-20A,B,C, MU-19A,B,C	3
		Valves CV-1407, 1408	4
		Valves CV-1219, 1220 1227, 1228	5
		Valves MU-1211, 1213, 1214, 1215	6
		Valves BW-2, 3	8

TABLE 4.2B. (continued)

Module	Title	Components	Failure Mode[a]
71707	Operational Safety Verification	HPI Pumps 36A,B,C	1
		Valves MU-18A,B,C,	2
		Valves MU-20A,B,C	3
		Valves CV-1407, 1408	4
		Valves CV-1219, 1220 1227, 1228	5
		Valves BW-2, 3	8
71710	ESF System Walkdown	HPI Pumps 36A,B,C	1
		Valves MU-18A,B,C,	2
		Valves MU-20A,B,C,	3
		Valves CV-1407, 1408	4
		Valves CV-1219, 1220 1227, 1228	5
		Valves BW-2, 3	8

(a) See Table 4.2A for failure identification

TABLE 4.2C. MODIFIED HIGH PRESSURE INJECTION SYSTEM WALKDOWN

Component Number	Component Name	Location	Required Position	Actual Position
	Electrical			
306	HPI Pump 36A Breaker	A3	Racked In	_____
307	HPI Pump 36B Breaker	A3	Racked In	_____
406	HPI Pump 36C Breaker	A4	Racked In	_____
407	HPI Pump 36B Breaker	A4	Racked In	_____
5151	CV-1219 HPI Loop A Isolation Valve Breaker	B51	Closed	_____
5152	CV-1220 HPI Loop A Isolation Valve Breaker	B51	Closed	_____
6151	CV-1227 HPI Loop B Isolation Valve Breaker	B61	Closed	_____
6152	CV-1228 HPI Loop B Isolation Valve Breaker	B61	Closed	_____
5164	CV-1407 BWST Outlet Valve Breaker	B51	Closed	_____
6164	CV-1408 BWST Outlet Valve Breaker	B61	Closed	_____
	Valves			
CV-1219	HPI Loop A Isolation Valve	UNPPR	Closed	_____
CV-1220	HPI Loop A Isolation Valve	UNPPR	Closed	_____
CV-1227	HPI Loop B Isolation Valve	UNPPR	Closed	_____
CV-1228	HPI Loop B Isolation Valve	UNPPR	Closed	_____
CV-1407	BWST Outlet Valve	El.354	Closed	_____
CV-1408	BWST Outlet Valve	El.354	Closed	_____
MV-18A	Pump P36A Suction Valve	P36A Rm	Open	_____
MV-18B	Pump P36B Suction Valve	P36B Rm	Open	_____
MV-18C	Pump P36C Suction Valve	P36C Rm	Open	_____
MV-20A	Pump P36A Discharge Valve	P36A Rm	Open	_____
MV-20B	Pump P36B Discharge Valve	P36B Rm	Open	_____
MV-20C	Pump P36C Discharge Valve	P36C Rm	Open	_____
BW-2	BWST Supply to Pump 36C Suction Valve	P36A Rm	Open	_____
BW-3	BWST Supply to Pump 36A Suction Valve	P36A Rm	Open	_____

4.3 LOW-PRESSURE INJECTION SYSTEM

TABLE 4.3A. LOW-PRESSURE INJECTION SYSTEM FAILURE MODE IDENTIFICATION

The Low-Pressure Injection (LPI) system is designed to perform both normal and emergency functions in several modes of operation. Under normal conditions, the most frequently used function is Decay Heat Removal (DHR) after a shutdown. The system is also used to supply water for auxiliary spray to the pressurizer, to maintain the proper reactor-coolant temperatures for refueling, and to provide a means for filling and draining the fuel-transfer canal. The emergency functions are LPI and Low Pressure Recirculation (LPR). In the LPI mode, the system provides two flow paths for injecting borated water from the Borated-Water Storage Tank (BWST) into the reactor vessel after a Loss-of-Coolant Accident (LOCA). In the LPR mode, it also provides two flow paths for recirculating the reactor coolant spilled in a LOCA from the reactor-building emergency sump back to the reactor vessel. The LPR mode can also be coupled with high-pressure pumps to provide high-pressure recirculation.

Conditions That Lead to Failure

1. LPI Pumps P34A and P34B or Heat Exchangers E35A and E35B Unavailable due to Maintenance or Testing

   This is the dominant failure mode for the low head recirculation operation. Both scheduled and unscheduled maintenance and testing are included. Maintenance or testing activity, and training should be reviewed or observed to minimize this unavailability, by enhancing the timeless and correctness of these activities.

2. LPI Pumps P34A and P34B Fail to Start or Run

   Failure of pumps P34A and P34B will prevent water flow from being provided to the reactor vessel. The important failure causes are random hardware or electrical circuit failures or failure of the pump room cooling units, and human errors in following procedures to recover from failures. Training, operator awareness, surveillance and maintenance of these pumps should be reviewed or observed to maintain reliability.

3. Manual Valves DH-3A, DH-3B, BW-8A and BW-8B or Check Valves DH-2A, DH-2B, BW-4A and BW-4B Fail Closed

   Failure closed of these valves in the suction and discharge lines will prevent water flow from being provided to the reactor vessel. The dominant failure cause is random hardware failure. A contributing cause is operator failure to open these valves after test. Maintenance and surveillance of these valves should be reviewed or observed to maintain reliability, and restoration of proper post-test lineup should be verified.

TABLE 4.3A (continued)

4. Pump or Heat Exchanger Failure due to Insufficient Cooling by Service Water System Failure

This failure mode can lead to failure of either or both LPI trains. The important failure cause is random hardware failures of the Service Water System (SWS) Valves CV-3840 and 3841 which supply cooling to pumps P34A and B, and Valves CV-3821 and 3822 which supply cooling to heat exchangers E35A and B. Surveillance and maintenance of these SWS valves should be reviewed or observed.

5. Motor-Operated Valves CV-1405, CV-1406, CV-1407, CV-1408, CV-1400 or 1401 Fail to Open on Demand

These valves are the LPI pump suction and discharge lines including lines from both the BWST and the RB sump. They must open following automatic actuation signals. The important failure causes are random hardware or electrical failures. Surveillance and maintenance of the valves according to Technical Specifications should help maintain availability.

6. Pneumatic Valves CV-1428 and 1429 Unavailable due to Maintenance

This includes both scheduled and unscheduled maintenance. The performance of maintenance should be reviewed to ensure that efficient scheduling is done, and that repairs are performed correctly, minimizing downtime.

7. Coupled Human Error-Failure to Close Valves DH-8A and B After Test

These include failures to realign or close a valve at the end of a test, and failure to discover and correct the error. The valves are manual valves allowing recirculation flow to the BWST. They should be locked closed after testing. These errors are addressed by proper test performance and post-test surveillance, which should be reviewed or observed.

8. Check Valves DH-13A, DH-13B, DH-14A, or DH-14B Fail Closed

Failure of these check valves in the closed position will prevent water flow to the reactor vessel. The dominant failure cause is random hardware failure. Maintenance of these valves should be reviewed or observed to maintain reliability.

TABLE 4.3B. IE MODULES FOR LOW-PRESSURE INJECTION SYSTEM INSPECTION

Module	Title	Components	Failure Mode[a]
41700	Training	Pumps P34A,B	1,2
		Heat Exchangers E35A,B	1
		Valves DH-8A,B	7
61701	Surveillance (Complex)	Pumps P34A,B and Heat Exchangers E35A,B	1
61725	Surveillance Testing and Calibration Program	Pumps P34A,B	1,2
		Heat Exchangers E35A,B	1
61726	Monthly Surveillance Observation	Pumps P34A,B	1,2
		Heat Exchangers E35A,B	1
		Valves DH-8A,B, 3A,B	3,7
		SWS Valves CV-3840,3841, 3821, 3822	4
		Valves CV-1405,1406 1407,1408,1400, 1401	5
62700	Maintenance	Pumps P34A,B	1,2
		Heat Exchangers E35A,B	1
		Valves DH-4A,B,8A,B 2A,B,3A,B	3,7
		SWS Valves CV-3840,3841, 3821, 3822	4
		Valves CV-1405,1406 1407,1408,1400, 1401	5
		Valves CV-1428,1429	6
		Valves DH-13A,B,14A,B	8
71707	Operational Safety Verification	Pumps P34A,B	1,2
		Heat Exchangers E35A,B	1
		Valves DH-4A,B,8A,B 2A,B,3A,B	3,7
		SWS Valves CV-3840,3841, 3821, 3822	4
		Valves CV-1405,1406 1407,1408,1400, 1401	5

TABLE 4.3B. (continued)

Module	Title	Components	Failure Mode[a]
71710	ESF System Walkdown	Pumps P34A,B	1,2
		Heat Exchangers E35A,B	1
		Valves DH-8A,B, 3A,B	3,7
		SWS Valves CV-3840,3841, 3821, 3822	4
		Valves CV-1405,1406 1407,1408,1400,1401	5

(a) See Table 4.3A for failure identification

TABLE 4.3C. MODIFIED LOW-PRESSURE INJECTION SYSTEM WALKDOWN

Component Number	Component Name	Location	Required Position	Actual Position
	Electrical			
305	LPI Pump 34A Breaker	A3	Racked in	_____
405	LPI Pump 34B Breaker	A4	Racked in	_____
51112	CV-1405 LPI Pump 34A Sump Suction Valve Breaker	B51	Closed	_____
6166	CV-1406 LPI Pump 34B Sump Suction Valve Breaker	B61	Closed	_____
5164	CV-1407 Outlet Valve Breaker	B51	Closed	_____
6164	CV-1408 Outlet Valve Breaker	B61	Closed	_____
6161	CV-1400 LPI Line "B" Isolation Valve Breaker	B61	Closed	_____
51114	CV-1401 LPI Line "A" Isolation Valve Breaker	B51	Closed	_____
5182	CV-3822 LPI Cooler "A" SW Supply Valve	B51	Closed	_____
6183	CV-3821 LPI Cooler "B" SW Supply Valve	B61	Closed	_____
	Valves			
BW-8A	LPI Pump 34A Suction Valve	P34A Rm	Open	_____
BW-8B	LPI Pump 34B Suction Valve	P34B Rm	Open	_____
DH-3A	P34A Discharge Valve	P34A Rm	Open	_____
DH-3B	P34B Discharge Valve	P34B Rm	Open	_____

TABLE 4.3C. (continued)

Component Number	Component Name	Location	Required Position	Actual Position
CV-3840	P34A Cooler SW Inlet Valve	P34A Rm	Closed	_____
CV-3841	P34B Cooler SW Inlet Valve	P34B Rm	Closed	_____
CV-1405	LPI Pump 34A Sump Suction Valve	P34A Rm	Closed	_____
CV-1406	LPI Pump 34B Sump Suction Valve	P34B Rm	Closed	_____
CV-1407	BWST Discnarge Valve	El.354	Closed	_____
CV-1408	BWST Discharge Valve	El.354	Closed	_____
CV-1400	LPI Loop "B" Isolation Valve	UNPPR	Closed	_____
CV-1401	LPI Loop "A" Isolation Valve	UNPPR	Closed	_____
CV-3821	DH Cooler "B" SW Isolation Valve	El.317	Closed	_____
CV-3822	DH Cooler "A" SW Isolation Valve	El.335	Closed	_____
CV-1428	DH Cooler "A" Outlet Valve	P34A Rm	Open	_____
CV-1429	DH Cooler "B" Outlet Valve	P34B Rm	Open	_____

4.4 SERVICE WATER SYSTEM

TABLE 4.4A. SERVICE WATER SYSTEM FAILURE MODE IDENTIFICATION

The Service Water System (SWS) supplies cooling water for many emergency and non-emergency needs throughout the plant. The SWS consists of two redundant loops with three pumps and associated valves and piping. The normal cooling is supplied from Lake Dardanelle through the intake structure; however, an emergency pond is available in case of loss of flow from the lake. The SWS is normally discharged back to the lake via the circulating water discharge flume. If the lake source is lost, the SWS would be discharged back to the emergency pond. The emergency pond also serves as a heat sink for normal plant shutdown of either Unit 1 or Unit 2.

Conditions That Lead to Failure

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1. Motor-Operated Valves CV-3820 and CV-3643 Fail to Close on Demand

   These are the Intermediate Auxiliary Cooling isolation valves for the service water system. They must be closed following engineered safeguard actuation signals. The important failure cause is random hardware and electrical failures. Surveillance and maintenance of these valves should be observed or reviewed.

2. Failures of Manual Valves SW-2A, 2B, 2C, or Check Valves SW-1A, 1B, 1C

   Operation of two of the three service water pumps is required to supply the designated nuclear headers during normal and emergency conditions. Failure of these valves in the closed position will prevent service water flow to the designated headers. Maintenance of these valves should be reviewed or observed to ensure their operability and proper lineup.

3. Operating Pumps 4A,B,C Fail to Run or Standby Pump Fails to Start and Run

   Failure of pumps may prevent sufficient service water flow from being provided to the essential header. Testing, maintenance, and surveillance of the pumps which are not in use according to the Technical Specifications should maintain reliability.

TABLE 4.4A. (Continued)

4. Crosstie Motor-Operated Valves CV-3640, 3642 or CV-3644, 3646 Fail to Close on Demand

   In the event of an Engineered Safeguard (ES) actuation with a simultaneous loss of offsite power, the crossover valves between the two operating SWS pumps will close. Failure to close these valves which supply service water to the Intermediate Cooling Water (ICW) or Auxiliary Cooling Water (ACW) systems may overload and fail the SWS. Proper maintenance and testing of these valves according to Technical Specifications should help prevent failures.

5. Motor-Operated Valves CV-3641 and CV-3645 Fail to Remain Open.

   These motor-operated valves must remain open to allow flow from the pumps to the designated headers. The dominant falure cause is random hardware failure. Valve maintenance, surveillance and the availability of electrical power should be reviewed or observed to maintain reliability.

TABLE 4.4B. IE MODULES FOR SERVICE WATER SYSTEM INSPECTION

Module	Title	Components	Failure Mode[a]
41700	Training	Valves CV-3640,3642 3644,3646	4
61701	Surveillance (Complex)	Pumps 4A,B,C	3
61725	Surveillance Testing Calibration Program	Valves CV-3820,3643	1
		Pumps 4A,B,C	3
		Valves CV-3640,3642, 3644,3646	4
		Valves CV-3641,3645	5
61726	Monthly Surveillance Observation	Valves CV-3820,3643	1
		Pumps 4A,B,C	3
		Valves CV-3641,3645	5
62700	Maintenance	Valves CV-3820,3643	1
		Valves SW-2A,B,C, 1A,B,C	2
		Pumps 4A,B,C	3
		Valves CV-3640,3642, 3644,3646	4
		Valves CV-3641,3645	5
71707	Operational Safety Verification	Valves CV-3820,3643	1
		Valves SW-2A,B,C	2
		Pumps 4A,B,C	3
		Valves CV-3641,3645	5
71710	ESF System Walkdown	Valves CV-3820,3643	1
		Valves SW-2A,B,C	2
		Pumps 4A,B,C	3
		Valves CV-3540,3642, 3644,3646	4
		Valves CV-3641,3645	5

(a) See Table 4.4A for failure identification

TABLE 4.4C. SERVICE WATER SYSTEM WALKDOWN

Component Number	Component Name	Location	Required Position	Actual Position
	Electrical			
5181	CV-3820 SW to ICW Valve Breaker	B51	Closed	_____
5653	CV-3643 SW to ACW Valve Breaker	B56	Closed	_____
302	Pump 4A Breaker	A3	Racked In	_____
303	Pump 4B Breaker	A4	Racked In	_____
403	Pump 4B Breaker	A4	Racked In	_____
402	Pump 4C Breaker	A4	Racked In	_____
6223	CV-3640 Loop 2 Crossover Breaker	B62	Closed	_____
6224	CV-4642 Loop 2 Crossover Breaker	B62	Closed	_____
5223	CV-3644 Loop 1 Crossover Breaker	B52	Closed	_____
5224	CV-3646 Loop 1 Crossover Breaker	B52	Closed	_____
6184	CV-3641 SW to Loop II Discharge Valve Breaker	B61	Closed	_____
51121	CV-3645 SW to Loop I Discharge Valve Breaker	B51	Closed	_____
	Valves			
CV-3820	Intermediate Cooling Motor Operated Valve	El.335	Open	_____
CV-3643	Auxiliary Cooling Motor Operated Valve	Intk	Open	_____
CV-3640	Loop 2 Crossover Motor Operated Valve	Intk	Open	_____
CV-3642	Loop 2 Crossover Motor Operated Valve	Intk	Open	_____
CV-3640	Loop 1 Crossover Motor Operated Valve	Intk	Open	_____

TABLE 4.4C. (Continued)

Component Number	Component Name	Location	Required Position	Actual Position
CV-3646	Loop 1 Crossover Motor Operated Valve	Intk	Open	_____
CV-3641	Loop 2 Discharge Motor Operated Valve	Intk	Open	_____
CV-3645	Loop 1 Discharge Motor Operated Valve	Intk	Open	_____
SW-2A	Pump P4A Discharge Manual Valve	Intk	Open	_____
SW-2B	Pump P4B Discharge Manual Valve	Intk	Open	_____
SW-2C	Pump P4C Discharge Manual Valve	Intk	Open	_____

4.5 REACTOR PROTECTION SYSTEM

TABLE 4.5A. REACTOR PROTECTION SYSTEM FAILURE MODE IDENTIFICATION

The Reactor Protection System (RPS) consists of redundant sensors, relays, logic, and other equipment necessary to monitor selected nuclear steam supply system conditions and to effect a reliable and rapid reactor shutdown (reactor trip) if any, or combination of monitored conditions reach specified safety system settings. Successful RPS operation protects the nuclear fuel from cladding damage and helps prevent reactor coolant system overpressure by limiting energy input.

Conditions That Lead to Failure

1. Reactor Trip Breakers A or B Fail Closed

   Reactor trip breakers A and B supply main and alternate AC power to the RPS system. Failure of either breaker to trip will provide system power unless interrupted by other breaker trips in the DC power circuit. The cause is hardware failure of the trip breakers. Maintenance and surveillance of these breakers should be observed or reviewed to minimize these failures. Operator training and awareness of Emergency Operating Procedures will enhance the probability of recovery.

2. DC Reactor Trip Breakers C1, C2 or D1, D2 Fail Closed

   Reactor trip breakers C and D supply main and alternate DC power to the safety rod hold circuit. Failure of any one of these breakers will result in two of the four safety rods failing to insert, and failure of both of C1 and C2 or D1 will result in all safety rods failing to insert. The cause is hardware failure of the trip breakers. Maintenance and surveillance should be observed or reviewed, and operator awareness of procedures will maintain reliability and enhance the probability of recovery.

3. Wiring Fault in Reactor Protection Channel

   A hardware-related wiring fault in the RPS channel may prevent the trip breakers actuation. Maintenance of these channels should be observed or reviewed to maintain reliability.

4. Trip Relays Fail Closed

   The required position for these relays is "open" once the trip signals have been generated from the process parameters. The important failure causes are the result of the relay coil failure to de-energize and random hardware failure. Surveillance and maintenance of these relays should be reviewed and observed.

TABLE 4.5B. IE MODULES FOR REACTOR PROTECTION SYSTEM INSPECTION

Module	Title	Components	Failure Mode[a]
41700	Training	Trip Breakers A,B	1
		Trip Breakers C,D	2
61701	Surveillance (Complex)	Trip Breakers A,B	1
		Trip Breakers C,D	2
		Trip Relays	4
61725	Surveillance Testing Calibration Program	Trip Breakers A,B	1
		Trip Breakers C,D	2
		Trip Relays	4
61726	Monthly Surveillance Observation	Trip Breakers A,B	1
		Trip Breakers C,D	2
		Trip Relays	4
62700	Maintenance	Trip Breakers A,B	1
		Trip Breakers C,D	2
		Channels Wiring	3
		Trip Relays	4
71707	Operational Safety Verification	Trip Breakers A,B	1
		Trip Breakers C,D	2
		Trip Relays	4

(a) See Table 4.5A for failure identification

TABLE 4.5C. MODIFIED REACTOR PROTECTION SYSTEM WALKDOWN

Component Number	Component Name	Location	Required Position	Actual Position
Walkdown is ineffective against risk significant RPS failures.				

4.6 EMERGENCY FEEDWATER SYSTEM

TABLE 4.6A. EMERGENCY FEEDWATER SYSTEM FAILURE MODE IDENTIFICATION

The purpose of the ANO-1 Emergency Feedwater system (EFW) is to backup the Main Feedwater system (MFW) in removing post-shutdown decay heat from the reactor coolant system via the steam generators. During normal shutdowns the MFW is throttled down to a level capable of removing decay heat and the EFW is not utilized. However, if the plant shutdown is caused by a loss of the MFW or the reactor coolant pumps, or if the MFW is lost subsequent to the plant shutdown, then the EFW is put into operation.

Conditions That Lead to Failure

1. Turbine-Driven Emergency Feedwater Pump P7A Fails to Start or Run

   This is the primary contributor to secondary system failure to provide cooling to the steam generators. Dominant system failure modes are turbine-driven emergency feedwater pump failure to start or run due to random hardware or control system faults, or operator failure to start the pump, or loss of room cooling, ventilation system, or instrument air. Observation or review of surveillance, maintenance, and lineup of this pump will maintain availability. Training in Emergency Operating Procedures and system malfunction response will enhance recovery when it is possible.

2. Motor-Driven Pump P7B Fails to Start or Run

   Failure of motor-driven pump P7B to start or run when required, or to be repaired while under maintenance will prevent cooling water flow from being supplied to the steam generator. Dominant system failure modes are pump failure to start or run due to random hardware or control system faults, or operator failure to start the pump, or loss of room cooling or ventilation system. As with the turbine-driven pump, maintenance and surveillance should be reviewed, lineup checked, and operator awareness of response procedures for malfunctions verified.

TABLE 4.6A. (Continued)

3. Motor-Operated Valves CV-2613, and CV-2663 Fail to Open or CV-2667, and 2617 Fail Closed

Steam supply for EFW pump P7A turbine is obtained from both steam generators via valves 2667, and 2617. Downstream of these valves, the pipes join to form a common supply to the pump turbine through parallel valves CV-2613 or CV-2663. Failure of valves CV-2613 and CV-2663 to open or failure of 2667, or 2617 to remain open will fail turbine-driven pump P7A and prevent cooling water to flow to the steam generators. The important failure cause is random hardware or control system failures. Proper system lineup, operator training and awareness of Emergency and abnormal Operating Procedures will enhance recovery. Surveillance and maintenance should be reviewed or observed to maintain reliability.

4. Failure of Motor-Operated Valves CV-2800, 2802, 2803, and 2806

These are pump suction valves for the EFW system. A common control switch for each pair causes the valves to assume opposite positions; that is, if one valve, e.g., CV-2806 is open, the other valve CV-2802 is closed and vice versa. Failure of these valves in the improper position will prevent EFW flow to the designated steam generator. The dominant failure cause is human error failure to manually realign suction for the EFW pumps. Operator awareness of criteria for switchover and adherence to emergency procedures is important.

TABLE 4.6B. MODIFIED EMERGENCY FEEDWATER SYSTEM INSPECTION

Module	Title	Components	Failure Mode[a]
41700	Training	TD Pump P7A	1
		MD Pump P7B	2
		Valves CV-2613,2663, 2667, 2617	3
		Valves CV-2800,2802, 2803,2806	4
61701	Surveillance (Complex)	TD Pump P7A	1
		MD Pump P7B	2
61725	Surveillance Testing and Calibration Program	TD Pump P7A	1
		MD Pump P7B	2
		Valves CV-2613,2663, 2667, 2617	3
61726	Monthly Surveillance Observation	TD Pump P7A	1
		MD Pump P7B	2
		Valves CV2613,2663, 2667,2617	3
62700	Maintenance	TD Pump P7A	1
		MD Pump P7B	2
		Valves CV2613,2663, 2667,2617	3
		Valves CV-2800,2802 2803,2806	4
71707	Operational Safety Verification	TD Pump P7A	1
		MD Pump P7B	2
		Valves CV-2613,2663 2667,2617	3
		Valves CV-2800,2802 2803,2806	4
71710	ESF System Walkdown	TD Pump P7A	1
		MD Pump P7B	2
		Valves CV-2613,2663 2667,2617	3
		Valves CV-2800,2802 2803,2806	4

(a) See Table 4.6A for failure identification

TABLE 4.6C. MODIFIED EMERGENCY FEEDWATER SYSTEM WALKDOWN

Component Number	Component Name	Location	Required Position	Actual Position
	Electrical			
311	Motor-Driven EFW Pump Breaker	A3	Racked in	_____
5173	CV-2800 Motor-Driven Pump 7B CST Suction Valve Breaker	B51	Closed	_____
5193	CV-2803 Motor-Driven Pump 7B SW Suction Valve Breaker	B51	Closed	_____
6175	CV-2802 Turbine-Driven Pump 7A CST Suction Valve Breaker	B61	Closed	_____
6181	CV-2806 Turbine-Driven Pump 7A SW Suction Valve Breaker	B61	Closed	_____
6241	CV-2617 SG B Steam Supply to P7A Valve Breaker	B62	Closed	_____
5241	CV-2667 SG A Steam Supply to P7A Valve Breaker	B52	Closed	_____
2512	CV-2613 P7A Steam Supply Valve Breaker	D25	Closed	_____
1512	CV-2663 P7A Steam Supply Valve Breaker	D15	Closed	_____
	Valves			
CV-2800	Motor-Driven Pump 7B CST Suction Valve	Pump Rm	Open	_____
CV-2803	Motor-Driven Pump 7B SW Suction Valve	Pump Rm	Closed	_____
CV-2802	Turbine-Driven Pump 7A CST Suction Valve	Pump Rm	Open	_____
CV-2806	Turbine-Driven Pump 7A SW Suction Valve	Pump Rm	Closed	_____
CV-2667	SG A Steam Supply to P7A Valve	Penthouse	Open	_____
CV-2617	SG B Steam Supply to P7A Valve	Penthouse	Open	_____

TABLE 4.5C. (Continued)

Component Number	Component Name	Location	Required Position	Actual Position
CV-2613	P7A Steam Supply Valve	Penthouse	Closed	_____
CV-2663	P7A Steam Supply Valve	Penthouse	Closed	_____

4.7 CLASS 1E AC POWER SYSTEM

TABLE 4.7A. CLASS 1E AC POWER SYSTEM FAILURE MODE IDENTIFICATION

The purpose of the Class 1E AC electrical power system is to provide electrical power to components in systems which are deemed vital to mitigate the consequences of loss-of-coolant accidents and transients. These vital systems include those which shut the reactor down, remove the decay and sensible heat of the coolant and building, and limit the release of radioactive material from the reactor building. In addition, the Class 1E AC electrical power system supplies power to the DC power system via three battery chargers.

Conditions that Lead to Failure

1. Failure of 480 Volt Load Centers B5 or B6

   Two load centers, B5 and B6, are provided. They serve engineered safeguard or essential loads and motor control centers. Each has a 4160/480 volt transformer between it and its energy source. Failure of these load centers would usually be associated with subcomponent failures in the control circuit, transformers, or improper lineup for automatic operation. Observation of maintenance, surveillance, and system lineup, or a review of the records of these functions should be performed.

2. Failure of 4160 Volt Distribution Panel A3 or A4

   The normal sources of power of the class 1E AC are through the 4160 volt distribution panels, A3 and A4. Loss of these panels will prevent electrical power from being supplied to the corresponding safeguards components. The unavailability could be caused by panel failure, circuit breaker failure, or maintenance outage. Periodic testing, and proper maintenance should minimize unavailability due to these causes.

3. Failure of Motor Control Center (MCC) B56

   The motor control center, B56, supplies electrical power to emergency lighting, standby battery charger, turbine generator emergency bearing and pump, and the electrical system room chillers and coolers. The unavailability could be caused by control circuit or circuit breaker failures. Proper surveillance and maintenance of the motor control center and the protective devices should be reviewed and observed.

TABLE 4.7A. (Continued)

4. Motor Control Center B51, B52, B61, or B62 Unavailable

These motor control centers being unavailable either due to component failure or scheduled or unscheduled maintenance or testing, combined with failure to restore the motor control centers to service, prevents the electrical power from being supplied to the respective loads. Periodic testing, proper maintenance, and surveillance should be observed and reviewed.

5. Failure of Emergency Diesel Generators DG1, DG2

Failure of DG1 or DG2 either to start or to run when required, or to be repaired while under maintenance will prevent electrical power from being supplied to the corresponding safeguards component buses. When combined with the loss of offsite power, a total loss of power source can result. Periodic maintenance, surveillance in accordance with Technical Specifications, calibration activities, and lineup check will enhance the availability. Operator training and awareness of Emergency Operating Procedures will enhance the probability of recovery.

TABLE 4.7B. IE MODULES FOR CLASS 1E AC POWER SYSTEM INSPECTION

Module	Title	Components	Failure Mode[a]
41700	Training	DG1,DG2	5
61701	Surveillance (Complex)	DG1,DG2	5
61725	Surveillance Testing and Calibration Program	Load Centers B5,6	1
		MCC B56	3
		MCC B51,52,61,62	4
		DG1,DG2	5
61726	Monthly Surveillance Observation	Load Centers B5,6	1
		MCC B56	3
		MCC B51,52,61,62	4
		DG1,DG2	5
62700	Maintenance	Load Centers B5,6	1
		Switchgear A3,4	2
		MCC B56	3
		MCC B51,52,61,62	4
		DG1,DG2	5
71707	Operational Safety Verification	Load Centers B5,6	1
		Switchgear A3,4	2
		MCC B56	3
		MCC B51,52,61,62	4
		DG1,DG2	5
71710	ESF System Walkdown	Load Centers B5,6	1
		Switchgear A3,4	2
		MCC B56	3
		MCC B51,52,61,62	4
		DG1,DG2	5

(a) See Table 4.7A for failure identification

TABLE 4.7C. CLASS 1E AC POWER SYSTEM WALKDOWN

Component Number	Component Name	Location	Required Position	Actual Position
521	MCC B-51 Supply Breaker	B5	Closed	_____
522	B-5 Supply to B-56	B5	Closed	_____
532	MCC B-52 Supply Breaker	B5	Closed	_____
621	MCC B-61 Supply	B6	Closed	_____
614	MCC B-62 Supply Breaker	B6	Closed	_____
622	B-6 Supply to B-56	B6	Closed	_____
512	B-5 Supply Breaker	B5	Closed	_____
612	B-6 Supply Breaker	B6	Closed	_____
308	DG-1 Output Breaker	A3	Open	_____
309	A1 to A3 Tie Breaker	A3	Closed	_____
310	A3 to A4 Tie Breaker	A3	Open	_____
408	DG-2 Output Breaker	A4	Open	_____
409	A2 to A4 Supply Breaker	A4	Closed	_____
410	A4 to A3 Tie Breaker	A4	Open	_____
301	Transformer X-5 Supply Breaker	A3	Closed	_____
401	Transformer X-6 Supply Breaker	A4	Closed	_____
513	B-5 to B-6 Tie Breaker	B5	Open	_____
613	B-6 to B-5 Tie Breaker	B6	Open	_____
5143B	Battery charger D03 Breaker	B51	Closed	_____
DG1	Diesel Generator 1		Note a	_____
DG2	Diesel Generator 2		Note a	_____

a. Due to the integrated nature of the diesel generator failure to start or to run failure modes, the lineup of all automatic diesel support functions (service water, fuel oil, starting air, etc.) should be checked.

4.8 SAFETY RELIEF VALVE SYSTEM

TABLE 4.8A. SAFETY RELIEF VALVE SYSTEM FAILURE MODE IDENTIFICATION

The safety relief valves or primary pressure control system is part of the reactor-coolant system (RCS). During normal operation the pressurizer establishes and maintains the RCS pressure within prescribed limits and provides a steam surge chamber and a water reserve to accommodate changes in the density of the reactor coolant. Under abnormal conditions, the relief valves on the pressurizer are the means of external pressure relief for the RCS.

Conditions that Lead to Failure

1. Pressurizer Relief Valves RC-1001, RC-1002 Fail to Close After Steam Relief

   The required position for these valves is "closed" once the pressurizer pressure has decreased below the relief valve set point. The failure mode is random hardware failures of these valves. Surveillance and maintenance of these valves, including setpoint testing and adjustment, should be reviewed or observed.

TABLE 4.8B. IE MODULES FOR SAFETY RELIEF VALVE SYSTEM INSPECTION

Module	Title	Components	Failure Mode[a]
41700	Training	Pressurizer Relief Valves	1
61701	Surveillance (Complex)	Pressurizer Relief Valves	1
61725	Surveillance Testing and Calibration Program	Pressurizer Relief Valves	1
61726	Monthly Surveillance Observation	Pressurizer Relief Valves	1
62700	Maintenance	Pressurizer Relief Valves	1
71707	Operational Safety Verification	Pressurizer Relief Valves	1

(a) See Table 4.8A for failure identification.

TABLE 4.8C. MODIFIED SAFETY RELIEF VALVE SYSTEM WALKDOWN

Componen* Number	Component Name	Location	Required Position	Actual Position
Walkdown is ineffective against failure of the SRVs to reseat.				

4.9 POWER CONVERSION SYSTEM

TABLE 4.9A. POWER CONVERSION SYSTEM FAILURE MODE IDENTIFICATION

Power Conversion System (PCS) at ANO-1 is designed to provide feedwater to the secondary side of the steam generators which, in turn, transfers energy to the turbine generator system. Following a reactor trip, the PCS is also capable of delivering feedwater to the steam generators at a reduced rate to provide for decay heat removal. This is accomplished by throttling the PCS feedwater flow to a level commensurate with decay heat and allowing this water to boil off to the condenser or atmosphere.

Conditions That Lead to Failure

1. Human Error-System Operation Inhibited, or Failure to Control the Startup Feedwater Valves CV-2623 or CV-2673

   This is the primary contributor to system failure to provide cooling to the steam generators. The failure cause is that, after reactor trip, the operator may assume control of the startup feedwater valves CV-2623 and CV-2673 and inadvertently cut off flow. A similar failure mode, however, can lead to excessive feedwater flow and hence a feedwater-pump trip on high steam-generator level. Operator training and awareness of Abnormal and Emergency Operating Procedures for system operation should be verified.

2. Failure of Steam Driven Main Feedwater Pumps

   Failure of steam-driven feedwater pumps P1A and P1B contribute significantly to the failure of steam generator cooling. The important failure cause following a reactor trip, with one feedwater pump automatically tripped after the reactor trip, is loss of the second feedwater pump due to loss of cooling or hardware failures. These failures can totally interrupt main feedwater supply despite its initial availability during an accident. Operator training, awareness of Emergency Operating Procedures, and proper maintenance and surveillance should be reviewed and observed.

3. Failure of Motor-Driven Condensate Pumps P2A, P2B and P2C

   Failure of these pumps will prevent sufficient cooling water flow to the main feedwater lines. The important failure cause is hardware failure. Proper maintenance and surveillance of these pumps should maintain their reliability.

TABLE 4.9B. IE MODULES FOR POWER CONVERSION SYSTEM INSPECTION

Module	Title	Components	Failure Mode[a]
41700	Training	Control Feedwater Valves CV-2623, 2673	1
		Steam-Driven Feedwater Pumps P1A, P1B	2
61701	Surveillance (Complex)	Control Feedwater Valves CV-2623, 2673	1
		Steam-Driven Feedwater Pumps P1A, P1B	2
61725	Surveillance Testing and Calibration Program	Control Feedwater Valves CV-2623, 2673	1
		Steam-Driven Feedwater Pumps P1A, P1B	2
		Motor-Driven Condensate Pumps P2A, P2B, P2C	3
62700	Maintenance	Control Feedwater Valves CV-2623, 2673	1
		Steam-Driven Feedwater Pumps P1A, P1B	2
		Motor-Driven Condensate Pumps P2A, P2B, P2C	3
71707	Operational Safety Verification	Control Feedwater Valves CV-2623, 2673	1
		Steam-Driven Feedwater Pumps P1A, P1B	2
		Motor-Driven Condensate Pumps P2A, P2B, P2C	3

(a) See Table 4.9A for failure identification

TABLE 4.9C. POWER CONVERSION SYSTEM WALKDOWN

Component Number	Component Name	Location	Required[a] Position	Actual Position
	Electrical			
105	Condensate Pump 2A Breaker	A1	Racked in	_____
205	Condensate Pump 2B Breaker	A2	Racked in	_____
106	Condensate Pump 2B Breaker	A1	Racked in	_____
	Valves			
CV-2623	Control Valve for Startup, Train 1		Open	_____
CV-2673	Control Valve for Startup, Train 2		Open	_____

4.10 EMERGENCY FEEDWATER INITIATION AND CONTROL SYSTEM

TABLE 4.10A. EMERGENCY FEEDWATER INITIATION AND CONTROL SYSTEM FAILURE MODE IDENTIFICATION

The Emergency Feedwater Initiation and Control (EFIC) System comprises three different logic systems: the initiation system, vector system, and control system. The EFIC functions which pertain to prevention of steam generator overfill and to isolation of a steam generator if low steam generator pressure occurs. The system also provides for manual initiation or shutdown of the emergency feedwater system, and for manual assumption of manual control of emergency feedwater after it has been automatically initiated.

Conditions That Lead to Failure

1. Operator Fails to Manually Initiate EFIC System Given Failure of Logics or Cables

   This is the dominant failure for the EFIC system. It involves operator failure to properly interpret the plant status and properly initiate the EFIC system, given the system failure. Operator training, awareness of proper procedures, and familiarity with associated controls should be assessed.

2. Failure of Actuation Logic Elements 1,2,3AB, 1,2,3AC, 1,2,3BD, or 1,2,3CD, or Relays 4AB, AC, BD or CD

   Failure of actuation logic elements or relays will prevent outputs of the initiation logic from being provided to the emergency feedwater system. The important failure causes are random electrical or hardware failures of the actuation logic element cables or their associated relays. Calibration activities, surveillance, and maintenance should be reviewed or observed to maintain reliability.

3. Failure of Control Logic Elements

   Simultaneous failure of four control logic elements, ABC1, ABC2, BCD1, and IP7B to function, given the presence of an actuation signal, will cause the EFIC system failure. The important causes are hardware failures of the control logic elements or their associated cables. Calibration, maintenance, and surveillance of these control logic elements should reduce the probability of failure.

TABLE 4.10B. IE MODULES FOR EMERGENCY FEEDWATER INITIATION AND CONTROL SYSTEM INSPECTION

Module	Title	Components	Failure Mode[a]
41700	Training	Manual Initiation Criteria	1
61701	Surveillance (Complex)	Cables, Actuation Logic Elements 1,2,3AB, 1,2,3AC, 1,2,3BD, 1,2,3CD	2
		Control Logic Elements ABC1, ABC2, BCD1, IP7B	3
61725	Surveillance Testing and Calibration Program	Cables, Actuation Logic Elements 1,2,3AB, 1,2,3AC, 1,2,3BD, 1,2,3CD Relays 4AB, AC, BD, CD	2
		Cables, Control Logic Elements ABC1, BCD1, IP7B	3
61726	Monthly Surveillance Observation	Cables, Actuation Logic Elements 1,2,3AB, 1,2,3AC, 1,2,3BD, 1,2,3CD Relays 4AB, AC, BD, CD	2
		Cables, Control Logic Elements ABC1, BCD1, IP7B	3
62700	Maintenance	Cables, Actuation Logic Elements 1,2,3AB, 1,2,3AC, 1,2,3BD, 1,2,3CD Relays 4AB, AC, BD, CD	2
		Cables, Control Logic Elements ABC1, BCD1, IP7B	3

TABLE 4.10B. (continued)

Module	Title	Components	Failure Mode[a]
71707	Operational Safety Verification	Cables, Actuation Logic Elements 1,2,3AB, 1,2,3AC, 1,2,3BD, 1,2,3CD Relays 4AB, AC, BD, CD	2
		Cables, Control Logic Elements ABC1, BCD1, IP7B	3

(a) See Table 4.10A for failure identification

TABLE 4.10C. MODIFIED EMERGENCY FEEDWATER INITIATION AND CONTROL SYSTEM WALKDOWN

Component Number	Component Name	Location	Required Position	Actual Position

Walkdown is ineffective against risk significant EFIC system failures. However, power availability, proper switch positioning, etc., should be verified to maintain the system reliability.

TABLE 11. PLANT OPERATIONS INSPECTION GUIDANCE

Recognizing that the normal system lineup is important for any given standby safety system, the following human errors are specially identified in the PRA as important to risk.

System	Failure	Discussion
High Pressure Injection	Post-maintenance/Testing Lineup Failure	Table 4.2A, Items 3,8
	Switchover/Recovery Failure	Table 4.2A, Items 4,5
	Feed and Bleed Control Failure	Table 4.2A, Item 7
Low-Pressure Injection	Switchover/Recovery Failure	Table 4.3A, Item 2
	Post-Test/Maintenance Lineup Failure	Table 4.3A, Items 3,7
Emergency Feedwater	Switchover/Recovery Failure	Table 4.6A, Items 1,2,4
Power Conversion System	FW Valves Lineup Failure	Table 4.9A, Item 1
	Switchover/Recovery Failure	Table 4.9A, Item 2
Emergency Feedwater Initiation and Control	Switchover/Recovery Failure	Table 4.10A, Item 1

TABLE 12. SURVEILLANCE INSPECTION GUIDANCE

The listed components are the risk significant components for which proper surveillance should minimize failure.

System	Component	Discussion
DC Power	Battery Chargers D03, D04, D05	Table 4.1A, Item 1
	DC Buses D01, D02	Table 4.1A, Item 3
	Panels RS1, RS2, RS3, RS4	Table 4.1A, Item 4
	Inverters Y11, Y13, Y22, Y24	Table 4.1A, Item 6
	Battery Sets D06, D07	Table 4.1A, Item 7
	Transformer Circuit Breakers	Table 4.1A, Item 8
High Pressure Injection	HPI Pumps 36A,B,C	Table 4.2A, Item 1
	Valves CV-1407, 1408	Table 4.2A, Item 4
	Valves CV-1219, 1220, 1227, 1228	Table 4.2A, Item 5
	Valves BW-2, 3	Table 4.2A, Item 8
Low Pressure Injection	Pumps P34A,B	Table 4.3A, Items 1,2
	Valves DH-4A,B,8A,B, 2A,B, 3A,B	Table 4.3A, Items 3,7
	SWS Valves CV-3840,3841, 3821, 3822	Table 4.3A, Item 4
	Valves CV-1405,1406, 1407, 1408, 1400, 1401	Table 4.3A, Item 5
Service Water	Valves CV-3820,3643	Table 4.4A, Item 1
	Pumps 4A,B,C	Table 4.4A, Item 3
	Valves CV-3640,3642, 3644, 3646	Table 4.4A, Item 4
	Valves CV-3641,3645	Table 4.4A, Item 5
Reactor Protection	Trip Breakers A,B	Table 4.5A, Item 1
	Trip Breakers C,D	Table 4.5A, Item 2
	Trip Relays	Table 4.5A, Item 4
Emergency Feedwater	TD Pump P7A	Table 4.6A, Item 1
	MD Pump P7B	Table 4.6A, Item 2
	Valves CV2613-2663,2667,2617	Table 4.6A, Item 3
AC Power	Load Centers B5,6	Table 4.7A, Item 1
	MCC B56	Table 4.7A, Item 3
	MCC B51,52,61,62	Table 4.7A, Item 4
	DG1,DG2	Table 4.7A, Item 5
Safety Relief Valve	Pressurizer Relief Valves	Table 4.8A, Item 1
Power Conversion	Control Feedwater Valves CV-2623, 2673	Table 4.9A, Item 1
	Steam-Driven Feedwater Pumps P1A, P1B	Table 4.9A, Item 2
	Motor-Driven Condensate Pumps P2A, P2B, P2C	Table 4.9A, Item 3

TABLE 12. (continued)

System	Component	Discussion
Emergency Feedwater	Actuation Logic Elements	Table 10.A, Item 2
Initiation and Control	Control Logic Elements	Table 10.A, Item 3

TABLE 13. MAINTENANCE INSPECTION GUIDANCE

The components listed here are significant to risk because of unavailability for maintenance or testing. The dominant contributors are usually frequency of maintenance and duration of maintenance, with some contribution due to improperly performed maintenance.

System	Component	Discussion
DC Power	Battery Chargers D03,D04, D05	Table 4.1A, Item 1
	Distribution Panels D11, D21	Table 4.1A, Item 2
	DC Buses D01, D02	Table 4.1A, Item 3
	Panels RS1, RS2, RS3, RS4	Table 4.1A, Item 4
	Motor Control Centers B51, B61	Table 4.1A, Item 5
	Inverters Y11, Y13, Y22, Y24	Table 4.1A, Item 6
	Battery Sets D06, D07	Table 4.1A, Item 7
	Transformer Circuit Breakers	Table 4.1A, Item 8
High Pressure Injection	HPI Pumps 36A,B,C	Table 4.2A, Item 1
	Valves MU-18A,B,C,	Table 4.2A, Item 2
	Valves MU-20A,B,C, MU-19A,B,C	Table 4.2A, Item 3
	Valves CV-1407, 1408	Table 4.2A, Item 4
	Valves CV-1219, 1220, 1227, 1228	Table 4.2A, Item 5
	Valves MU-1211, 1213, 1214, 1215	Table 4.2A, Item 6
	Valves BW-2, 3	Table 4.2A, Item 8
Low Pressure Injection	Pumps P34A,B	Table 4.3A, Items 1,2
	Heat Exchangers E35A,B	Table 4.3A, Item 1
	Valves DH-4A,B, 8A,B, 2A,B, 3A,B	Table 4.3A, Items 3,7
	SWS Valves CV-3840,3841, 3821, 3822	Table 4.3A, Item 4
	Valves CV-1405,1406, 1407, 1408 1400, 1401	Table 4.3A, Item 5
	Valves CV-1428,1429	Table 4.3A, Item 6
	Valves DH-13A,B,14A,B	Table 4.3A, Item 8
Service Water	Valves CV-3820,3643	Table 4.4A, Item 1
	Valves SW-2A,B,C, 1A,B,C	Table 4.4A, Item 2
	Pumps 4A,B,C	Table 4.4A, Item 3
	Valves CV-3640,3642, 3644, 3646	Table 4.4A, Item 4
	Valves CV-3641,3645	Table 4.4A, Item 5
Reactor Protection	Trip Breakers A,B	Table 4.5A, Item 1
	Trip Breakers C,D	Table 4.5A, Item 2
	Channels Wiring	Table 4.5A, Item 3
	Trip Relays	Table 4.5A, Item 4

TABLE 13. (Continued)

System	Component	Discussion
Emergency Feedwater	TD Pump P7A	Table 4.6A, Item 1
	MD Pump P7B	Table 4.6A, Item 2
	Valves CV-2613,2663,2667,2617	Table 4.6A, Item 3
	Valves CV-2800,2802, 2803, 2806	Table 4.6A, Item 4
AC Power	Load Centers B5,6	Table 4.7A, Item 1
	Switchgear A3,4	Table 4.7A, Item 2
	MCC B56	Table 4.7A, Item 3
	MCC B51,52,61,62	Table 4.7A, Item 4
	DG1,DG2	Table 4.7A, Item 5
Safety Relief Valve	Pressurizer Relief Valves	Table 4.8A, Item 1
Power Conversion	Control Feedwater Valves CV-2623,2673	Table 4.9A, Item 1
	Steam-Driven Feedwater Pumps P1A, P1B	Table 4.9A, Item 2
	Motor-Driven Condensate Pumps P2A, P2B, P2C	Table 4.9A, Item 3
Emergency Feedwater	Actuation Logic Elements	Table 10.A, Item 2
Initiation and Control	Control Logic Elements	Table 10.A, Item 3

TABLE 14. QUALITY ASSURANCE/ADMINISTRATIVE CONTROL INSPECTION GUIDANCE

The failures listed here are the ones which the QA/Administrative staff can affect. For example, QA should ensure that both regular and post-maintenance surveillance actually test for failure mode of concern for significant equipment. Also, in the case of equipment unavailabilities, administrative control should work to minimize the plant risk.

System	Component	Discussion
DC Power	Battery Chargers D03, D04, D05	Table 4.1A, Item 1
	Distribution Panels D11, D21	Table 4.1A, Item 2
	DC Buses D01, D02	Table 4.1A, Item 3
	Panels RS1, RS2, RS3, RS4	Table 4.1A, Item 4
	Motor Control Centers B51, B61	Table 4.1A, Item 5
	Inverters Y11, Y13, Y22, Y24	Table 4.1A, Item 6
	Battery Sets D06, D07	Table 4.1A, Item 7
	Transformer Circuit Breakers	Table 4.1A, Item 8
High Pressure Injection	HPI Pumps 36A,B,C	Table 4.2A, Item 1
	Valves MU-18A,B,C,	Table 4.2A, Item 2
	Valves MU-20A,B,C, MU-19A,B,C	Table 4.2A, Item 3
	Valves CV-1407, 1408	Table 4.2A, Item 4
	Valves CV-1219, 1220, 1227, 1228	Table 4.2A, Item 5
	Valves MU-1211, 1213, 1214, 1215	Table 4.2A, Item 6
	Valves BW-2, 3	Table 4.2A, Item 8
Low Pressure Injection	Pumps P34A,B	Table 4.3A, Items 1,2
	Heat Exchangers E35A,B	Table 4.3A, Item 1
	Valves DH-4A,B,8A,B, 2A,B, 3A,B	Table 4.3A, Items 3,7
	SWS Valves CV-3840,3841, 3821, 3822	Table 4.3A, Item 4
	Valves CV-1405,1406, 1407, 1408, 1400,1401	Table 4.3A, Item 5
	Valves DH-13A,B,14A,B	Table 4.3A, Item 8
Service Water	Valves CV-3820,3643	Table 4.4A, Item 1
	Valves SW-2A,B,C, 1A,B,C	Table 4.4A, Item 2
	Pumps 4A,B,C	Table 4.4A, Item 3
	Valves CV-3540,3642, 3644, 3646	Table 4.4A, Item 4
	Valves CV-3641,3645	Table 4.4A, Item 5
Reactor Protection	Trip Breakers A,B	Table 4.5A, Item 1
	Trip Breakers C,D	Table 4.5A, Item 2
	Trip Relays	Table 4.5A, Item 4
Emergency Feedwater	TD Pump P7A	Table 4.6A, Item 1
	MD Pump P7B	Table 4.6A, Item 2
	Valves CV-2613, 2663, 2667, 2617	Table 4.6A, Item 3
	Valves CV-2800,2802, 2803, 2806	Table 4.6A, Item 4

TABLE 14. (continued)

System	Component	Discussion
AC Power	Load Centers B5,6	Table 4.7A, Item 1
	Switchgear A3,4	Table 4.7A, Item 2
	MCC B56	Table 4.7A, Item 3
	MCC B51,52,61,62	Table 4.7A, Item 4
	DG1,DG2	Table 4.7A, Item 5
Safety Relief Valve	Pressurizer Relief Valves	Table 4.8A, Item 1
Power Conversion	Control Feedwater Valves CV-2623, 2673	Table 4.9A, Item 1
	Steam-Driven Feedwater Pumps P1A, P1B	Table 4.9A, Item 2
	Motor-Driven Condensate Pumps P2A, P2B, P2C	Table 4.9A, Item 3
Emergency Feedwater	Actuation Logic Elements	Table 10.A, Item 2
Initiation and Control	Control Logic Elements	Table 10.A, Item 3

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The ANO-1 PRA has been analyzed to identify plant systems and components important to minimizing public risk, and to identify and primary failure modes of these components. This information has been tabulated, and correlated with inspection modules from the NRC Inspection and Enforcement Manual. The report presents a series of tables, organized by system and prioritizied by public risk, which identify components associated with 98% of the inspectable risk due to plant operation. The systems addressed, in descending order of risk importance, are: DC Power, High Pressure Injection, Low Pressure Injection, Service Water, Reactor Protection, Emergency Feedwater, Emergency AC Power, Safety Relief Valves, Power Conversion, and Emergency Feedwater Initiation and Control. This ranking is based on the Fussel-Vesely measure of ranking importance applied to core melt probability, i.e., the fraction of the core melt probability which involves failures of the system of interest.

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