February 9, 1990

Docket No. 50-289

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Mr. Henry D. Hukill,
Vice President and Director - TMI-1
GPU Nuclear Corporation
P. O. Box 480
Middletown, Pennsylvania 17057

Dear Mr. Hukill:

SUBJECT: RISK-BASED INSPECTION GUIDE FOR THREE MILE ISLAND, UNIT 1

The NRC has a program for producing plant-specific inspection guidance for resident inspectors using insights from probabilistic risk assessments (PRAs). The Risk-Based Inspection Guide (RIG) for a particular plant contains information about its dominant accident sequences, the relative importance of the various plant systems that contribute to these sequences, and the more risk-significant failure modes of the important systems.

The purpose of the RIGs is to help focus NRC's inspection activities on the most risk-significant aspects of each plant so that our limited inspection efforts have the most benefit to public health and safety. For example, the RIG directs increased attention to a limited number of plant components and it provides guidance for selecting activities to observe and for events to follow up. Plant-specific vulnerabilities are highlighted. In addition, inspection checklists are provided for the most risk-significant failure modes.

The enclosed draft RIG for Three Mile Island Unit was produced for NRC under a contract with Battelle's Pacific Northwest Laboratories. Your PRA was used as the basis for this RIG, with some modification to remove the sensitivity to the control room HVAC system.

At this time, we are providing the draft RIG to the Resident Inspector at TMI for his review and comment. We would also be pleased to receive any technical comments or suggestions that you care to make on the draft. Your staff should feel free to contact our Project Engineer, Steve Long, at any time to discuss the draft or the project. He can be reached at (301) 492-3162.

When comments have been received, we will make any necessary corrections and issue the completed RIG for use at TMI. We would like to receive any comments you may have within about 60 days.

Sincerely,

/S/

Ronald W. Hernan, Senior Project Manager
Project Directorate I-4
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosure:
As stated

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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cc w/enclosure:
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DRAFT

RISK BASED INSPECTION GUIDE FOR THREE MILE ISLAND NUCLEAR STATION UNIT 1

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Prepared for
Division of Radiation Protection and
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Office of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Washington, DC 20555
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Pacific Northwest Laboratory
Richland, Washington 99352

ABSTRACT

The level one probabilistic risk assessment (PRA) for Three Mile Island Nuclear Station Unit 1 (TMI-1) has been analyzed to identify plant systems and components important to minimizing public risk, as measured by system contributions to the annual probability of core damage, and to identify the primary failure modes of these components. This report presents a series of tables, organized by system and prioritized by risk importance, which identify components associated with over 95% of the inspectable risk due to plant operation. The systems addressed, in descending order of importance, are: the Decay Heat Removal, High Pressure Injection, Decay Heat Cooling Water, AC Power, Nuclear Services Cooling Water, Main Steam, Emergency Feedwater, Reactor Coolant System Pressure Control, Intermediate Closed Cooling Water, Instrument Air, DC Power, and Engineered Safeguards Actuation Systems. This ranking is based on the Fussell-Vesely Importance Measure of risk importance, i.e., the fraction of the total annual probability of core damage which involves failures of the system of interest.

Though not involved in the prevention of core damage and thus not ranked, containment protection systems are of fundamental importance in preventing and minimizing public risk due to a release of radionuclides, should core damage occur. Therefore, containment protection systems are included in this report, consisting of: the Reactor Building Isolation, Reactor Building Spray, and Reactor Building Emergency Cooling Systems.

SUMMARY

The Probabilistic Risk Assessment (PRA) Applications Program for inspection at Three Mile Island Nuclear Station Unit 1 (TMI-1) was performed for the United States Nuclear Regulatory Commission (USNRC) at Pacific Northwest Laboratory (PNL). This program applies a previously developed methodology to identify and present information which is useful for the planning and performance of nuclear power plant inspections.

The level I PRA for TMI-1 has been analyzed to identify plant systems and components important to minimizing public risk, as measured by system contributions to the annual probability of core damage for the plant. This information is provided as a series of tables, organized by system and prioritized by risk importance, that identify the components associated with over 95% of the annual probability of core damage for TMI-1 operation.

In Appendix A, tables are presented for twelve systems, which are ordered by system risk importance, as measured by the fraction of the total core damage annual probability associated with failures of each system. In addition, three reactor building systems are presented to address containment protection capabilities. Two tables are presented for each system. The first table presents the component failure modes identified in the PRA as important for each system. The second table provides a modified system walkdown check-off list identifying the proper line-up of each important component during normal operation. A simplified flow diagram is also provided for each system.

The tables were developed by first ordering the plant systems according to system risk importance. To accomplished this, the fraction of the total core damage annual probability which involved failures of components from each system was calculated (i.e., Fussel-Vesely Importance Measure). Systems were then selected from the ordered list until more than 95% of the annual probability of core damage was addressed. For each selected system, component failure modes were ranked according to their importance to system failure. Components were selected for inclusion in the tables 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 and check-off list of normal operational state for all components associated with over 95% of the annual probability of core damage for TMI-1 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 the inspectors to concentrate their efforts on systems important to the prevention of core damage. 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 many tools to be used by experienced inspectors.

ACKNOWLEDGEMENTS

This analysis was performed under sponsorship by the United States Nuclear Regulatory Commission (NRC). We wish to thank the NRC Lead Engineer, Mr. Steven Long, for his insights. We also wish to thank our colleagues at Brookhaven National Laboratory (BNL) for providing guidance in formatting this report.

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1.0 INTRODUCTION

This inspection guide has been prepared for the United States Nuclear Regulatory Commission (USNRC) as part of an extensive program to develop information based on Probabilistic Risk Assessments (PRAs) for use in the planning and performance of nuclear power plant inspections. This particular inspection guide has been prepared to provide inspection guidance for the Three Mile Island Nuclear Station Unit 1 (TMI-1). This inspection guide should be used to aid in the selection of areas to inspect and is not intended either to replace current USNRC inspection guidance or to constitute an additional set of inspection requirements. The information contained herein is based almost entirely on the TMI-1 level I PRA (GPUN, 1987), which was performed by Pickard, Lowe and Garrick, Inc. (PLG) for General Public Utilities Nuclear Corporation (GPUN). The TMI-1 PRA is considered to be a level I PRA since it models plant system responses to internal and external initiating events to determine the annual probability of core damage, but it does not model containment responses, radionuclide releases and transport, and radiation doses to the public. Only the internal initiating events analyses have been considered for this inspection guide. Since it has been a number of years since the PRA was completed and plant modifications are normally an ongoing process, it is recommended that recent system experiences, failures, and modifications be considered when reviewing these tables and relevant system changes be catalogued so that this inspection guidance can be periodically revised as required.

Information from the TMI-1 PRA internal initiating events and component unavailabilities analyses was used to identify the plant systems and components important to minimizing the annual probability of core damage, and to identify the failure modes for these components. The body of this report consists of a series of tables, organized by system and prioritized by system importance, which identify the components that dominate the plant core damage annual probability.

Containment protection systems have also been included. These systems, though not involved in the prevention of core damage, are of fundamental importance in preventing and minimizing public risk due to a release of radionuclides, should core damage occur. The containment protection systems included are the Reactor Building Spray System (Appendix A14), Reactor Building Emergency Cooling System (Appendix A15), and Reactor Building Isolation System (Appendix A13). This inspection guide identifies the failure modes for the components that were analyzed and found to be important for these systems in the TMI-1 PRA.

2.0 DOMINANT ACCIDENT SEQUENCES

A review of the TMI-1 PRA has identified one overly conservative assumption which causes the system risk importances calculated using the PRA results to be misleading. Specifically, it is assumed in the PRA that a loss of the Control Building Ventilation System on a hot summer day will result in the heatup of a single room in the control building to 104°F within a couple of hours, resulting in the complete failure of all of the vital equipment within the entire building. This is not consistent with actual plant experience involving loss of control room ventilation events. It is also inconsistent with USNRC staff and industry conclusions related to station blackout, where operator actions such as opening cabinets to allow natural circulation cooling of electrical panels to maintain operability has been accepted (USNRC, 1988).

The PRA for Crystal River Unit 3 (Florida Power Corporation, 1987), which has a similar Control Building Ventilation System, considered loss of room cooling and concluded that it was not a significant contributor to the probability of core damage. The Crystal River analysis assumed a slower heatup rate and did not assume complete, immediate, failure of all components upon reaching their temperature limit of environmental qualification. In addition, tests in September of 1987 at TMI-1 indicated a longer time before the hottest rooms reached 104°F, which may take as long as 24 hours. This longer time period allows more time for operator actions to mitigate the problem and also means it is less likely that the outside air temperature will remain high enough, long enough to be of concern. The assumption regarding the Control Building Ventilation System was recognized in the TMI-1 PRA as being overly conservative and it was stated that in subsequent revisions this portion of the analysis would be reviewed, based on the results of the tests, and revised. These revisions to the PRA have not yet been performed.

On the basis of this information it is our assessment that a more realistic analysis of TMI-1 would result in the conclusion that the Control Building Ventilation System is a negligible contributor to the probability of core damage. Consequently, the contribution from the failure of the Control Building Ventilation System has been eliminated from this inspection guidance.

The importance of this adjustment to the PRA results is evident in the fact that approximately 45% of the core damage probability in the TMI-1 PRA from internal initiating events resulted from events initiated by failure of the Control Building Ventilation System. This inspection guide, therefore, is based on a core damage probability of 2.43×10^{-4} per year, instead of the value of 4.43×10^{-4} per year published in the PRA. With the exception of this modification, the PRA results were basically used as reported.

The TMI-1 PRA identifies a number of different accident sequences that contribute significantly to the probability of core damage. Each of these dominant accident sequences is composed of several similar but distinct combinations of failures. These sequences have been sorted into four initiator groups, and the contribution to core damage from these groups is shown in Figure 2.1. These four groups are dominated by four loss-of-support system scenarios, two transient scenarios, three very small loss of coolant accident

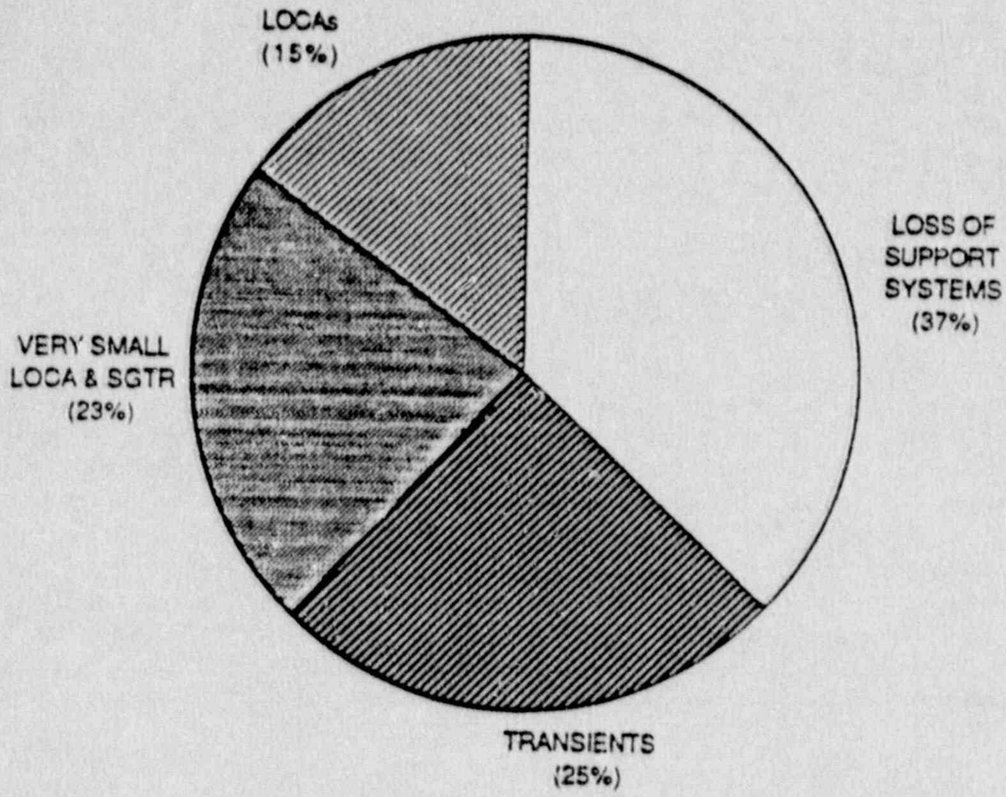

Figure 2.1 Internal Initiating Event Group Contributions to the Annual Probability of Core Damage

(LOCA) and steam generator tube rupture (SGTR) scenarios, and two LOCA scenarios. The contribution to the probability of core damage from the internal initiating events of significance is provided in Table 2.1 Each of the four initiator groups is discussed below.

2.1 LOSS OF SUPPORT SYSTEM SCENARIOS (37%)

There are four scenarios that dominate the contribution to core damage for sequences initiated by the loss of a support system. Each of these scenarios is discussed below.

2.1.1 Loss of Instrument Air

A loss of instrument air causes the reactor coolant pump (RCP) seal injection air-operated valve MU-V20 to fail closed. This valve is in a common header to all four RCP seals and will thus stop seal injection flow to all four RCP seals. Failure of RCP thermal barrier cooling from the Intermediate Closed Cooling Water (ICCW) System results in no cooling to the RCP seals. If this occurs, the RCP seals will overheat, degrade, and eventually fail, leaking Reactor Coolant System (RCS) inventory (i.e., a RCP seal LOCA) unless the operator manually reopens the RCP seal injection air-operated valve MU-V20. This is a particularly important failure for cases in which a loss of offsite power is coincident with the loss of instrument air, since the ICCW system may be locked out by an Engineered Safeguards Actuation System (ESAS) signal in this situation and is thus unavailable for thermal barrier cooling. If an RCP seal LOCA occurs and primary makeup flow from the High Pressure Injection (HPI) System fails, RCS inventory will gradually be lost out the RCP seal and eventually result in core damage.

2.1.2 Loss of River Water

River water fails due to a clogged intake screen causing the turbine to trip. The river water failure causes the failure of the Nuclear Services Cooling Water (NSCW) System and the Decay Heat Cooling Water (DHCW) System, which support numerous safety systems including the High Pressure Injection (HPI) System and the Decay Heat Removal (DHR) System. This results in only the Emergency Feedwater (EFW) System being available for core heat removal and no long-term decay heat removal capability. Recovery of river water is thus essential to shutting down the reactor safely, and the success or failure of the EFW System determines the amount of time available for this recovery action. Failure to recover river water results in core damage without injection.

Table 2.1 Contribution to the TMI-1 Annual Probability of Core Damage*

INITIATING EVENT		PERCENTAGE OF CORE DAMAGE PROBABILITY
Loss of Support System Scenarios		37
Loss of Offsite Power	12	
Loss of Instrument Air	8	
Loss of River Water to the Pump House	7	
Loss of ATA Power	5	
Loss of One Train of DC Power	4	
Transient Scenarios		25
Reactor Trip	9	
Excessive Main Feedwater	7	
Turbine Trip	5	
Very Small LOCA and SGTR Scenarios		23
SGTR	16	
Very Small LOCA	7	
LOCA Scenarios		15
Medium LOCA	8	
Large LOCA	4	
Small LOCA	3	

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* The contribution from a loss of control building ventilation is assumed to be negligible and has been eliminated. These percentage contributions consider only internal initiating events and have been recalculated to reflect the elimination of the control building ventilation initiator.

2.1.3 Station Blackout

A station blackout occurs (i.e., loss of offsite power and both trains of emergency AC power fail), resulting in only the turbine-driven pump train of the EFW System being available to remove the heat from the core (HPI and DHR are both AC power dependent). The ensuing failure of the EFW turbine-driven pump train greatly reduces the time available to recover AC power before core damage commences. Failure to recover AC power quickly enough also results in an RCP seal LOCA, which will allow the primary water to leak out of the RCS. Therefore, an RCP seal LOCA occurs, followed by eventual core damage without injection or core heat removal if AC power is not recovered.

2.1.4 Loss of One Train of DC Power

A loss of one train of DC power occurs, resulting in only one train of safety systems being available and also causing the HPI operating pump to stop. Failure of this pump to restart or to be restarted by the operator fails high-pressure injection. A failure of the available train of the DHR system or its cooling water support system (i.e., the DHCW System) will fail high-pressure recirculation. Recirculation flow is achieved by manually switching the DHR pump suction from the nearly-empty borated water storage tank (BWST) to the reactor building sump. The HPI System is then realigned to take suction from the DHR pumps in the recirculation mode. Recirculation failure may be due to several reasons, including: operator failure to recognize the event before depletion of the BWST, failure of the low-level alarm on the BWST to alert the operator of near-empty conditions, or equipment failure in the lines that take suction from the sump. An independent failure of the EFW System fails the remaining means of core heat removal and core damage occurs.

2.2 TRANSIENT SCENARIOS (25%)

There are two scenarios that dominate the contribution to core damage for sequences initiated by transient events. Each of these scenarios is discussed below.

2.2.1 Excessive Main Feedwater

Excessive amounts of main feedwater are fed to the steam generators. This results in the RCS cooling down and depressurizing far enough to cause a 1600 psig signal to be sent by the ESAS to start the HPI System and closes the HPI minimum-flow recirculation line to the makeup tank. The operator fails to reestablish HPI minimum-flow recirculation after the HPI flow has been throttled, thus causing the HPI pumps to fail. An independent failure of RCP seal cooling from the ICCW System leads to neither injection nor cooling of the RCP seals and thus these seals degrade and begin to leak. Core damage commences since there is no makeup for the inventory being lost through the seals (i.e., RCP seal LOCA scenario without injection).

2.2.2 Reactor Trip

A reactor trip occurs with a continued small RCS inventory loss (e.g., via a stuck-open power-operated relief valve (PORV)). Failure of both trains of DHCW fails DHR, which causes an inability to perform recirculation. Failing to recover DHCW prior to the depletion of the BWST will result in core damage.

2.3 VERY SMALL LOCA AND SGTR SCENARIOS (23%)

There are three scenarios that dominate the contribution to core damage for sequences initiated by very small LOCA or SGTR events. Each of these scenarios is discussed below.

2.3.1 Very Small LOCA or SGTR and DHR Failure

A very small LOCA or a SGTR occurs followed by the independent failure of the DHR System. The DHR System failure can be caused by failures within this system and/or failures within its cooling water support system, DHCW. This fails recirculation and long-term heat removal capabilities. Core damage commences upon depletion of the BWST.

2.3.2 SGTR and One Train of Electric Power Fails

A SGTR occurs and one train of electric power fails. This is followed by the failure to cooldown the RCS and stop the RCS inventory loss to the secondary side of the steam generator. Since the lost RCS inventory is on the secondary side, it does not reach the reactor building sump and thus fails recirculation makeup to the RCS. Core damage begins when the BWST is depleted.

2.3.3 Very Small LOCA and Recirculation Failure

A very small LOCA occurs and either hardware failures and/or operator errors fails the establishment of high-pressure recirculation from the reactor building sump before the BWST is depleted. This results in the failure of recirculation makeup to the RCS and eventually core damage occurs.

2.4 LOCA SCENARIOS (15%)

There are two scenarios that dominate the contribution to core damage for sequences initiated by large, medium, or small LOCAs. Each of these scenarios is discussed below.

2.4.1 Medium or Large LOCA

A medium or large LOCA is followed by the failure of recirculation flow from the reactor building sump due to either hardware failure or operator error. Core damage commences when injection fails due to the depletion of the BWST inventory.

2.4.2 Small LOCA

A small LOCA occurs and both trains of DHCW fail due to independent or common causes. Without DHCW, the DHR pumps fail and core damage results since decay heat is not removed from the core and core inventory is lost out the break.

3.0 SYSTEM PRIORITY LIST

The TMI-1 plant systems have been ranked in Table 3.1 according to their importance in preventing core damage caused by internal initiating events. Two different rankings are provided for use under two types of circumstances. Under normal conditions, the left-hand column should be used. For degraded or inoperable systems, the right-hand column should be used.

The two system prioritization lists have been included in Table 3.1 because they provide different types of risk insights that are useful in the inspection process. The left-hand column indicates the system's contribution to the probability of core damage as provided by the Fussell-Vesely Importance Measure, given that the system is operating with the reliability found in the PRA. Generally, when planning an inspection without knowledge of specific system problems, those systems that contribute most to the probability of core damage should be given priority attention in order to most efficiently minimize risks.

However, when one or more systems exhibit unusually higher failure rates or unusual types of failures from those presented in the PRA, then the failure probabilities determined in the PRA may not be appropriate for those systems. While the system or systems are experiencing these operational problems, the affected system's contribution to the probability of core damage will probably be greater than is indicated by the left-hand column. The increase in the probability of core damage when the system is inoperable is indicated by the right-hand column, which is based on the Birnbaum Importance Measure. The right-hand column can be used to estimate how much more important these systems have become when they are having problems. Affected systems with high rankings in the right-hand column should be considered to have become much more important than indicated by their rank in the left-hand column, while systems with low rankings in the right-hand column would have a smaller increase in rank than indicated in the left-hand column. Therefore, the right-hand column is the appropriate choice for estimating the risk significance of inspection findings that indicate a system is inoperable or degraded. The inspection team should take these factors into account whenever possible.

Systems that are adjacent in the table should be considered to have approximately equal contributions to the probability of core damage because of the uncertainties in the PRA. Where the difference between importance measures of adjacent systems is significant, they have been separated by a dashed line.

As discussed in Section 2.0, the Control Building Ventilation System has been eliminated from this analysis and has been assumed to be a negligible contributor to the probability of core damage at TMI-1. Other plant systems not appearing in these lists are generally of lesser importance than those that are included in the table.

Table 3.1 <u>System Priority Ranking</u>[1,4,5]

By Contribution to the Probability of Core Damage[2]	By Risk Significance of the System Being Unavailable[3]
Decay Heat Removal	AC Power
High Pressure Injection	Decay Heat Removal
Decay Heat Cooling Water	Intermediate Closed Cooling Water
------------------------------	------------------------------
AC Power	Engineered Safeguards Actuation
Nuclear Services Cooling Water	High Pressure Injection
Main Steam	Reactor Protection
Emergency Feedwater	Nuclear Services Cooling Water
RCS Pressure Control	Emergency Feedwater
Intermediate Closed Cooling Water	DC Power
------------------------------	------------------------------
Instrument Air	RCS Pressure Control
DC Power	Main Steam
Engineered Safeguards Actuation	Decay Heat Cooling Water
	Instrument Air

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1 The Control Building Ventilation System is not included in this table, based on the assessment that its contribution to the probability of core damage would be negligible if a more realistic analysis of the system were conducted in the TMI-1 PRA.

2 The ranking in the left-hand column is appropriate to use for systems that are functioning normally. It is based on the Fussell-Vesely Importance Measure, which is the system's contribution to the probability of core damage, assuming that the system is operating with normal reliability.

3 The ranking in the right-hand column is appropriate to use for determining the significance of known system degradation or inoperability. It is based on the Birnbaum Importance Measure, which indicates the increase in the probability of core damage that results when the system is assumed to be inoperable.

4 Containment protection systems are not shown since they do not effect the probability of core damage. However, these containment protection systems are included in the Appendices.

5 The dashed lines represent significant differences between importances of systems that are adjacent in the lists. Systems not separated by dashed lines should be assumed to have importances approximately equivalent to each other, within the precision of the PRA quantification.

4.0 COMMON CAUSE FAILURES

The failure of multiple items from some common cause can be very significant to the probability of core damage. Many simultaneous failures of redundant system trains are the direct result of an initiating event (e.g., a loss of instrument air initiating event may result in the failure of a number of air-operated valves). This is especially the case when external events (e.g., earthquakes, fires, floods, etc.) are considered. In addition to the effects of initiating events on systems and components, the TMI-1 PRA has identified a few common cause failures that are particularly important. The particularly important common cause failures involve the following components:

1. DHR pumps DH-P1A and DH-P1B
2. DHR cross-tie to HPI, valves DH-V7A and DH-V7B
3. Reactor building sump isolation valves DH-V6A and DH-V6B
4. Decay Heat Closed Cooling Water (DHCCW) pumps DC-P1A and DC-P1B
5. Decay Heat River Water (DHRW) pump discharge valves DR-V1A and DR-V1B
6. DHRW pumps DR-P1A and DR-P1B
7. ESAS time delay relays
8. Nuclear Services River Water (NSRW) pumps NR-P1A, NR-P1B, and NR-P1C
9. Nuclear Services Closed Cooling Water (NSCCW) pumps NS-P1A, NS-P1B, and NS-P1C

These failures appear in cutsets contributing to approximately 10% of the probability of core damage from internal initiating events.

These and other common cause failures, not considered to be as important as the events above but still considered to be significant contributors, are identified in Table 4.1 in descending order of importance and in the failure mode tables in the Appendices.

Table 4.1 Common Cause Failure Contributors

DHR pumps DH-P1A and DH-P1B
DHR motor-operated cross-tie valves DH-V7A and DH-V7B
Reactor building sump isolation valves DH-V6A and DH-V6B
DHCCW pumps DC-P1A and DC-P1B
DHRW motor-operated pump discharge valves DR-V1A and DR-V1B
DHRW pumps DR-P1A and DR-P1B
ESAS time delay relays
NSRW pumps NR-P1A and NR-P1B, and NR-P1C
NSCCW pumps NS-P1A, NS-P1B, and NS-P1C
ESAS relays (other than time delay or actuation relays)
DHR injection valves DH-V4A and DH-V4B
DHR heat exchangers DH-HX1A and DH-HX1B
HPI injection valves MU-V16A, MU-V16B, MU-V16C, and MU-V16D
BWST isolation valves MU-V14A and MU-V14B
ESAS actuation relays
Reactor trip breakers
Diesel generator sets EP-DG1A and EP-DG1B
ESAS bistables
HPI pumps MU-P1A, MU-P1B, and MU-P1C

5.0 IMPORTANT HUMAN ERRORS

Human errors can be very significant to overall plant risk. The TMI-1 PRA has identified several human errors as particularly important contributors to the probability of core damage. The most important human errors have been categorized as either pre-accident or post-accident errors. A discussion of the important human errors in each category follows.

Other human errors, less important but still significant, are also identified in Appendix B, Table B-1, Plant Operations Inspection Guidance.

5.1 PRE-ACCIDENT ERRORS

There was only one particularly important pre-accident human error found in the TMI-1 PRA. This error consisted of the miscalibration of the ESAS sensors and appears in cutsets contributing approximately 1% to the probability of core damage from internal initiating events. A couple of slightly less significant contributors found in the PRA are:

1. Miscalibration of Reactor Protection System sensors
2. Operator fails to return the DHR System to its normal alignment after test or maintenance activities.

5.2 POST-ACCIDENT ERRORS

There were numerous post-accident human errors found to be particularly important in the TMI-1 PRA. These post-accident human errors have been further broken down into three categories: operator restoration and recovery actions, manual actions to actuate systems, and manual backup actions to automatic actuations.

5.2.1 Operator Restoration and Recovery Actions

1. The operator fails to recover at least one train of onsite AC power or offsite power either before the station batteries are depleted (for station blackout conditions) or to recover systems which require AC power.
2. The operator fails to clear the river water screen before the plant trips (this is included as part of the loss of river water initiating event)
3. The operator fails to recover the river water system (the time available for this recovery is dependent on the availability of the EFW turbine-driven pump)

4. The operator fails to trip the main feedwater pumps to avoid an overcooling event

5. The operator fails to locate and isolate a leaking NSCW heat exchanger before it causes the failure of the NSCW system.

5.2.2 Manual Actions to Actuate Systems

1. The operator fails to initiate switchover to recirculation flow from the reactor building sump following the BWST low-level alarm (time available depends on the rate of RCS inventory loss: 1 minute for a large LOCA and 10 minutes for a medium LOCA)

2. The operator fails to establish HPI minimum-flow recirculation after throttling HPI

3. The operator fails to manually actuate a 4 psig reactor building pressure signal under conditions in which this signal would not be automatically received (e.g., during reactor building purging)

4. The operator fails to start HPI pumps

5. The operator fails to open the DHR dropline valves

6. The operator fails to initiate cooldown

7. The operator fails to manually load the instrument air compressors onto the emergency AC power source and start the compressors after a loss of offsite power.

5.2.3 Manual Backup Actions to Automatic Actuations

1. The operator fails to locally control EFW flow after a loss of offsite power, with no instrument air and/or only one train of emergency AC power available

2. The operator fails to throttle HPI flow, to avoid opening the PORV

3. The operator fails to bypass the instrument air dryer transfer valve, after the valve fails closed

4. The operator fails to replenish the 2-hour air bottles used to control EFW air-operated valves, after a loss of instrument air

5. The operator fails to reopen the RCP seal injection air-operated valve MU-V20, after a loss of instrument air

6. The operator incorrectly or inadvertently throttles the HPI flow, failing injection

7. The operator fails to cross-connect HPI pump MU-P1C for injection.

6.0 SYSTEM INSPECTION TABLES

Taken together, the systems ranked by their importance in Table 3.1 contribute to over 95% of the probability of core damage from internally initiated events for TMI-1. For each of these systems, inspection guidance is provided in Appendix A in the form of a failure mode table, an abbreviated walkdown checklist, and a simplified system diagram. The simplified system diagrams are either from the TMI-PRA or are from the USNRC Nuclear Power Plant System Sourcebook for TMI-1 (USNRC, 1989).

The information in these tables allows an inspector to quickly identify the components most important to the probability of core damage. In particular, the system walkdown tables can be used to rapidly review the line-up of important system components on a routine basis. These tables can also be used when selecting systems for the performance of more detailed inspection activities.

In using these tables, however, it is essential to remember that other systems and components are also important. If, through inattention, the likelihood of other systems failing was allowed to increase significantly, their contribution to the probability of core damage might equal or exceed that of the systems in the following tables. Consequently, a balanced inspection program is essential to assure that the safety objectives of plant inspections are met by allowing an inspector to concentrate on systems that are most significant to the probability of core damage. In so doing, however, cognizance of the status of systems performing other essential safety functions must be maintained.

7.0 REFERENCES

Three Mile Island Nuclear Station Unit 1 Probabilistic Risk Assessment, PLG-0525, prepared for General Public Utilities Nuclear Corporation by Pickard, Lowe and Garrick, Inc., November 1987.

Memorandum, John W. Craig, Chief of Plant Systems Branch, Division of Engineering and Systems Technology, USNRC, to Ashok Thadani, Assistant Director for Systems, Division of Engineering and Systems Technology, USNRC, Subject: "Observations regarding insights from recently submitted plant-specific PRAs", July 1988.

Crystal River Unit 3 Probabilistic Risk Assessment, Florida Power Corporation and Science Applications International Corporation, July 1987.

Nuclear Power Plant System Sourcebook for TMI-1, prepared for the United States Nuclear Regulatory Commission by Science Applications International Corporation, September 1989.

APPENDIX A SYSTEM INSPECTION TABLES

This Appendix contains information from the TMI-1 PRA for each of the systems listed in Table 3.1 and for the Reactor Building Isolation, Sprays, and Emergency Cooling Systems. The systems are addressed in decreasing order of risk importance: Decay Heat Removal (A1), High Pressure Injection (A2), Decay Heat Cooling Water (A3), AC Power (A4), Nuclear Services Cooling Water (A5), Main Steam (A6), Emergency Feedwater (A7), Reactor Coolant System Pressure Control (A8), Intermediate Closed Cooling Water (A9), Instrument Air (A10), DC Power (A11), and Engineering Safeguards Actuation (A12). In addition, containment protection systems are addressed, consisting of: Reactor Building Isolation (A13), Reactor Building Spray (A14), and Reactor Building Emergency Cooling (A15). For each system, a separate table or figure presents: system failure modes, a modified system walkdown table, and a simplified system diagram. Each of these is discussed below.

A.1 SYSTEM FAILURE MODE TABLES

The system failure mode table provides a brief description of the system and its success criteria as used in the PRA. This may differ from the success criteria contained in the FSAR and may not completely reflect the present operation of the system. The table then provides the dominant events (i.e., component failures or operator errors) contributing to system failure, listed in order of decreasing importance, along with a brief discussion. Since most systems are designed with redundant trains, it will generally take more than one of these events to fail the entire system. Where single events are sufficient to fail the entire system, it is noted in the brief discussion of the event. Otherwise, no effort has been made to list all of the possible combinations of events that are sufficient to produce system failure. For certain events that are important primarily because of the circumstances of a particular accident sequence, that information is also noted.

Inspections focussed on the items in the table will address approximately 95% of the probability of failure for that system. Because PRAs do not contain the detail necessary to attribute the listed failures to the most probable specific root causes, it is necessary for the inspectors to draw from their own experiences, plant operating history, ASME codes, NRC Bulletins and Information Notices, INPO SOERs, vendor notices and similar sources to determine how to actually conduct an inspection of the listed items. Where appropriate, codes have been included following each event description to indicate which licensee programs/activities provide inspectable aspects of the risk. These codes are as follows:

PC - Periodic calibration activities, procedures, and training

PT - Periodic testing activities, procedures, and training

MT - Preventive or unscheduled maintenance activities, procedures, and training

OP - Normal or emergency operating procedures, check-off lists, training, etc.

TS - Technical specifications

ISI - In-service inspection

A.2 MODIFIED SYSTEM WALKDOWN TABLES

This table provides an abbreviated version of the licensee's system checklist, where available, but includes only those items which are related to the dominant failure modes. It is generally much less than the normal checklist. It can be used to rapidly review the line-up of important system components on a routine basis. Caution, however, should be observed when using the modified system walkdown tables, since they are based on certain versions of the licensee's system operating instructions. Valve numbers used are those identified in the PRA and/or simplified system diagrams.

A.3 SIMPLIFIED SYSTEM DIAGRAMS

A simplified one-line diagram is provided for each system discussed. These diagrams are intended to aid in visualizing the system configuration and the location of the components discussed in the associated two tables. Since these diagrams are neither complete nor controlled, they should not be used in place of up-to-date system diagrams during inspection activities.

Decay Heat Removal (DHR) System

Table A1-1 Importance Basis and Failure Mode Identification

CONDITIONS THAT CAN LEAD TO FAILURE

The Decay Heat Removal (DHR) System, also designated as the Low Pressure Injection (LPI) System, is designed to perform both normal and emergency functions during various modes of operation. The DHR System is comprised of two pumps, separated into two trains with cross-connection capabilities. The most frequently used function is long-term (decay) heat removal after a shutdown or reactor trip. The emergency functions are low pressure injection and low pressure recirculation. In the LPI mode, the system provides two flow paths for injecting borated water from the BWST directly into the reactor vessel after a LOCA. In the recirculation mode, the system also provides two flow paths for recirculating the RCS inventory spilled through the break during a LOCA from the reactor building sump back to the reactor vessel. The recirculation mode can also be coupled with the HPI System pumps to perform high-pressure recirculation. In this mode of operation the HPI pumps take suction from the discharge of the LPI pumps.

The success of long-term heat removal is especially sensitive to the performance of this system and the associated DHCW System, which provides cooling water to the DHR System (see Table A3-1). For long-term heat removal success, one DHR pump and its associated cooler is required to function. The DHR pump and cooler success depends on the cooling water provided by the DHCW System and, since there are no cross-connection capabilities in this system, failure of a single component in a train of DHCW will fail the associated DHR pump and cooler.

For emergency functions, system success is defined in the TMI-1 PRA as one of the two DHR pumps injecting borated water directly into the reactor vessel during both injection and recirculation modes. In addition, for high-pressure recirculation success at least one DHR pump must provide flow to at least one HPI pump.

1. Operator Fails Switchover to Recirculation Mode

   Upon reaching a low level in the BWST, the operator has 1 minute for a large LOCA and 10 minutes for a medium LOCA to manually switch the DHR pump suction from the BWST to the reactor building sumps. Failure to promptly switchover will result in the depletion of the BWST and the ensuing failure of the DHR pumps, thus failing long-term heat removal and recirculation capabilities. (OP)

2. Common Cause Failure of the DHR Pumps DH-P1A and DH-P1B

This fails the DHR system and thus fails long-term heat removal and low-pressure injection. (MT, PT)

3. Piggy-Back Strainer Unavailable due to Maintenance

Maintenance on a piggy-back strainer requires the isolation and thus unavailability of the affected train of DHR. (MT, TS)

4. DHR Pump DH-P1A or DH-P1B Fails

This fails one train of DHR. (MT, PT)

5. DHR Pump DH-P1A or DH-P1B Unavailable due to Maintenance

This fails one train of DHR. (MT, TS)

6. DHR Sump Isolation Valve DH-V6A or DH-V6B Fails to Open

The failure of DHR sump isolation valve DH-V6A or DH-V6B to open fails the corresponding train of low-pressure recirculation. (MT, PT)

7. DHR Pump DH-P1A or DH-P1B Unavailable due to Operability Testing

This fails one train of DHR. (PT, TS)

8. Common Cause Failure of Piggy-Back Cross-Tie Valves DH-V7A and DH-V7B To Open

Failure of cross-tie motor-operated valves DH-V7A and DH-V7B to open defeats the capability to supply flow to the corrsponding HPI pump trains in the recirculation mode. (MT, PT)

9. Common Cause Failure of DHR Sump Isolation Valves DH-V6A and DH-V6B to Open

The failure of both DHR sump isolation valves to open results in the failure of both high-pressure and low-pressure recirculation. (MT, PT)

10. DHR Piggy-Back Cross-Tie Valve DH-V7A or DH-V7B Fails to Open

Failure of either cross-tie motor-operated valve to open fails the supply of water to a train of HPI in the recirculation mode. (MT, PT)

11. DHR Injection Valve DH-V4A or DH-V4B Unavailable due to Maintenance

The unavailability of an injection valves results in the failure of low-pressure injection paths. (MT, TS)

12. DHR Cooler DH-C1A or DH-C1B Unavailable due to Maintenance

The unavailability of a cooler results in the failure of one train of DHR. (MT, TS)

13. DHR Cooler DH-C1A or DH-C1B Fails

The failure of either DHR cooler fails the corresponding train of DHR. (MT, PT)

14. Operator Fails to Return Alignment to Normal After Test or Maintenance

The main failure mode is the operator failing to return the system to an injection alignment after maintenance or testing, which results in the failure of either one or both trains of DHR. (OP, MT, PT, TS)

15. Operator Fails to Open Dropline Isolation Valves DH-V1, DH-V2, and/or DH-V3

This failure causes the failure of long-term heat removal. The operator either fails to open all three motor-operated dropline isolation valves DH-V1, DH-V2, and DH-V3 remotely from the control room or, if valves fail to open, the operator fails to manually open them. To manually open valves DH-V1 and DH-V2 the operator must enter containment. (OP)

16. Dropline Isolation Valve DH-V1, DH-V2, or DH-V3 Unavailable due to Maintenance

This fails the capability of long-term heat removal. (MT, TS)

17. Common Cause Failure of DHR Injection Valves DH-V4A and DH-4B to Open

Failure of both motor-operated valves to open fails the corresponding trains of low-pressure injection. (MT, PT)

18. Common Cause Failure of DHR Heat Exchangers DH-C1A and DH-C1B

This fails both trains of DHR long-term heat removal capability. (MT, PT)

19. Reactor Building Sump Clogs

This fails the ability to perform both low-pressure and high-pressure recirculation. (MT, PT)

20. Motor-Operated Dropline Valves DH-V1, DH-V2, or DH-V3 Fail to Open

This failure causes the failure of long-term heat removal. (MT, PT)

21. DHR Sump Isolation Valve DH-V6A or DH-V6B Unavailable due to Maintenance

This fails recirculation capabilities from one of the trains of the reactor building sump. (MT, TS)

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Decay Heat Removal (DHR) System

Table A1-2 Modified System Walkdown*

COMPONENT NUMBER	COMPONENT NAME	LOCATION	REQUIRED POSITION	ACTUAL POSITION
RB Sump	Reactor Building Sump	281 RB		______
DH-V6A	Sump Isolation Valve	DH VT A	Closed	______
DH-V6B	Sump Isolation Valve	DH VT B	Closed	______
DH-P1A	DHR Pump	DH VT A	Standby	______
DH-P1B	DHR Pump	DH VT B	Standby	______
	A Piggy-Back Strainer	281 AB		______
	B Piggy-Back Strainer	281 AB		______
DH-V7A	Piggy-Back Cross-Tie Valve	281 AB	Closed	______
DH-V7B	Piggy-Back Cross-Tie Valve	281 AB	Closed	______
DH-V4A	DHR Injection Valve	281 AB	Closed	______
DH-V4B	DHR Injection Valve	281 AB	Closed	______
DH-C1A	DHR Cooler	DH VT A		______
DH-C1B	DHR Cooler	DH VT B		______
DH-V20A	Pump Discharge Test Valve	281 AB	Locked Close	______
DH-V20B	Pump Discharge Test Valve	281 AB	Locked Close	______
DH-V21	Pump Flow Test Valve	281 AB	Locked Close	______
DH-V1	Dropline Isolation Valve	309 RB	Closed	______
DH-V2	Dropline Isolation Valve	281 RB	Closed	______
DH-V3	Dropline Isolation Valve	281 AB	Closed	______

* DH VT A and DH VT B are located on 261 elevation of the Auxiliary Building.

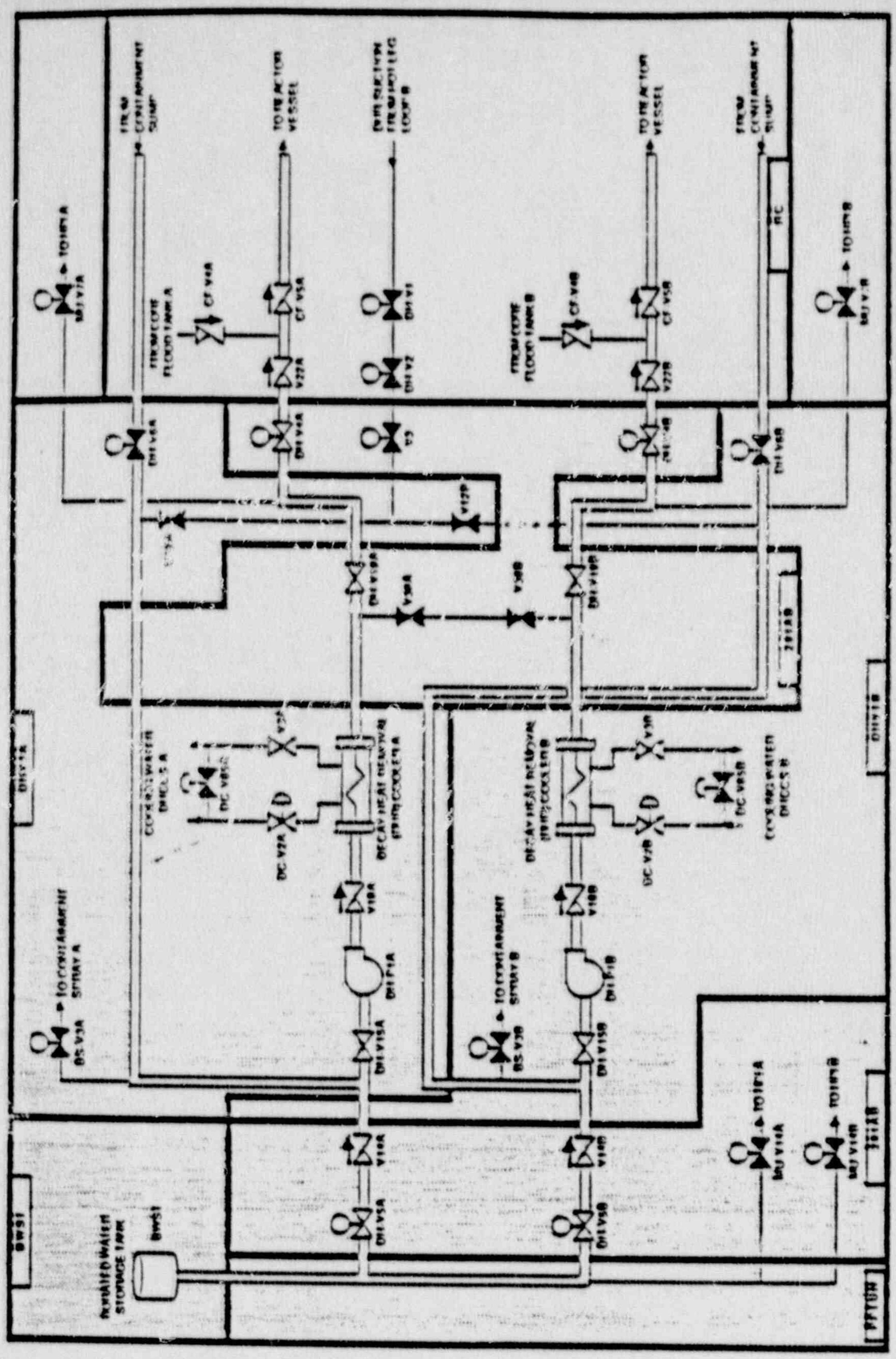

FIGURE A1.1 Simplified System Drawing of Decay Heat Removal

High Pressure Injection (HPI) System

Table A2-1 Importance Basis and Failure Mode Identification

CONDITIONS THAT CAN LEAD TO FAILURE

The High Pressure Injection (HPI) System, also designated as the Makeup and Purification System, is designed to perform both normal and emergency functions. In the TMI-1 PRA, it is analyzed for two important functions: 1) supplying high pressure injection water to the RCS in the case of a small LOCA which remains at a high RCS pressure, to prevent core uncovery and 2) supplying seal water to the RCP seals. This latter function works in tandem with the ICCW System, which provides RCP thermal barrier cooling, to prevent the RCP seals from overheating, degrading, and eventually leaking (i.e., RCP seal LOCA). It should be noted that the ICCW System is not an engineered safeguards system, and thus may be unavailable in many sequences. This tends to increase the importance of the HPI System RCP seal injection function.

The system is comprised of three pumps, separated into two trains. One of the pumps, MU-P1B, can be powered and actuated from either one of the trains. Under normal conditions, one pump performs the makeup and purification functions and the other two pumps are in engineered safeguards standby of different trains. The operating pump suction is aligned to the RCS, via the makeup tank, and the discharge is aligned to the normal RCS makeup and RCP seal injection paths. During accident conditions, the HPI pumps in standby are automatically started by the ESAS and the pump suctions are automatically aligned to the BWST. In this mode of operation borated water is injected into the RCS, via all four RCS cold legs, and RCP seal injection is maintained. If a low water level is detected in the BWST, a switchover to the recirculation mode of operation is achieved by manually realigning the pump suctions to the reactor building sumps, via the discharge of the DHR pumps.

System success is defined in the TMI-1 PRA as one of the three HPI pumps injecting borated water to the RCS cold legs and providing seal water to all four RCP seals during both injection and recirculation modes. In addition, the operator may eventually have to throttle the HPI flow to prevent the opening of a pressurizer PORV and/or pressurizer safety valves (PSVs).

1. <u>Operator Fails to Establish Minimum-Flow Recirculation</u>

   After an ESAS isolation signal automatically closes the minimum-flow recirculation motor-operated valves MU-V36 and MU-V37, these valves must be manually reopened when the HPI flow is throttled. This failure results in the failure of the HPI pumps. (OP)

2. Operator Fails to Throttle HPI Flow Using Makeup Valve MU-V217 and/or Injection Valves MU-V16A, MU-V16B, MU-V16C, and MU-V16D.

   This failure results in the HPI System overfilling the RCS such that pressurizer PORV and/or PSVs are opened to relieve the water and remain open until the HPI flow is eventually throttled. Even if the HPI flow is throttled after these valves have been opened there is the potential for these valves to fail to reseat. These conditions result in a transient-induced LOCA. (OP)

3. Operator Fails to Reopen the RCP Seal Injection Air-Operated Valve MU-V20

   Air-operated valve MU-V20 must be manually reopened after a loss of instrument air. This valve is in a common header to all four RCP seals and will thus stop seal injection flow to all four RCP seals. This is a particularly important failure for cases in which a loss of offsite power causes the loss of instrument air, since the ICCW system may be locked out by an ESAS signal in this situation and is thus unavailable for thermal barrier cooling. This situation may result in an RCP seal LOCA. (OP)

4. Operator Fails to Start HPI Pumps MU-P1A, MU-P1B, and/or MU-P1C for HPI Cooling

   This failure results in the failure of HPI. (OP)

5. Makeup Flow Path Valves MU-V217 and/or MU-V16A, MU-V16B, MU-V16C, and MU-V16D Fail to Throttle

   Failure of the normal makeup motor-operated valve MU-V217 and/or injection flow path motor-operated valves MU-V16A, MU-V16B, MU-V16C, and MU-V16D to be throttled results in an inability to throttle the HPI flow to prevent the filling of the RCS and subsequent opening of the pressurizer PORV and/or PSVs. In addition, if a loss of offsite power occurs within 1 hour after valve MU-V217 is opened, the valve will remain open and remote throttling of HPI flow is guaranteed to fail. The assumption was made in the TMI-1 PRA that the normal makeup valve MU-17 did not have an adverse impact on the system if it did not throttle closed. (MT, PT)

6. HPI Pump MU-P1A, MU-P1B, or MU-P1C Unavailable Due to Maintenance

   Maintenance on either one or two HPI pumps fails the corresponding train of HPI. (MT, TS)

7. HPI Pump MU-P1A, MU-P1B, or MU-P1C Fails to Start and Run

   Failure of either one or two HPI pumps fails the corresponding train of HPI. (MT, PT)

8. Common Cause Failure of Injection Flow Path Valves MU-V16A, MU-V16B, MU-V16C, and MU-V16D to Open

   Injection flow path motor-operated valve MU-V16A, MU-V16B, MU-V16C, and MU-V16D failure combinations result in the failure to inject borated water into the RCS cold legs. (MT, PT)

9. <u>Common Cause Failure of the BWST Isolation Valves MU-V14A and MU-V14B to Open</u>

Failure of motor-operated isolation valves MU-V14A and MU-V14B to open, fails the corresponding train of HPI pumps. (MT, PT)

10. <u>Independent Failures of Both BWST Isolation Valves MU-14A and MU-14B to Open</u>

This fails the BWST suction path for the HPI pumps. (MT, PT)

11. <u>Operator Incorrectly or Inadvertently Throttles HPI Flow</u>

This fails injection to the RCP cold legs. (OP)

12. <u>Minimum-Flow Recirculation Valve MU-V36 or MU-V37 Fails to Open</u>

Failure of either MU-V36 or MU-V37 to open results in the failure of the HPI pumps. (MT, PT)

13. <u>Failure of 1 of 7 Check Valves in an Injection Path to Open</u>

Each HPI flow path contains seven check valves in series. Failure of any one of these check valves fails its HPI path. (MT, PT)

14. <u>Common Cause Failure of the HPI Pumps MU-P1A, MU-P1B, and MU-P1C</u>

This fails high-pressure injection. (MT, PT)

15. <u>Operator Fails to Cross-Connect HPI Pump MU-P1C for Injection by Opening Valves MU-V76A and MU-V76B</u>

This failure results in the failure to use HPI Pump MU-P1C by aligning it to another injection path, when its injection path is failed or when the other HPI pumps fail to operate. (OP)

16. <u>Failure of Valves MU-V20, MU-V76A, or MU-V76B to Open</u>

This failure is most important after a loss of instrument air causes valve MU-V20 to fail closed. The operator must open these valves to continue RCP seal injection. (MT, PT)

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High Pressure Injection (HPI) System

Table A2-2 Modified System Walkdown*

COMPONENT NUMBER	COMPONENT NAME	LOCATION	REQUIRED POSITION	ACTUAL POSITION
MU-V36	Mini-Flow Recirc. Valve	281 AB	Open	______
MU-V37	Mini-Flow Recirc. Valve	281 AB	Open	______
MU-V17	Normal Makeup Flow Valve	281 AB	Open	______
MU-V217	Makeup Flow Valve	281 AB	Closed	______
MU-V16A	HPI Injection Valve	281 AB	Closed	______
MU-V16B	HPI Injection Valve	281 AB	Closed	______
MU-V16C	HPI Injection Valve	305 AB	Closed	______
MU-V16D	HPI Injection Valve	305 AB	Closed	______
MU-V20	RCP Seal Injection Valve	305 AB	Open	______
MU-P1A	HPI Pump	HPI RM A		______
MU-P1B	HPI Pump	HPI RM B		______
MU-P1C	HPI Pump	HPI RM C		______
MU-V14A	BWST Isolation Valve	281 AB	Closed	______
MU-V14B	BWST isolation Valve	281 AB	Closed	______
MU-V73A	Pump Discharge Check Valve	HPI RM A		______
MU-V73B	Pump Discharge Check Valve	HPI RM B		______
MU-V73C	Pump Discharge Check Valve	HPI RM C		______
MU-V107A	HPI Injection Check Valve	RB	Closed	______
MU-V107B	HPI Injection Check Valve	RB	Closed	______
MU-V107C	HPI Injection Check Valve	RB	Closed	______
MU-V107D	HPI Injection Check Valve	RB	Closed	______

MU-V95	HPI Injection Check Valve	RB	Closed	________
MU-V94	HPI Injection Check Valve	RB	Open	________
MU-86B	HPI Injection Check Valve	RB	Closed	________
MU-86A	HPI Injection Check Valve	RB	Closed	________
MU-220	HPI Injection Check Valve	RB	Closed	________
MU-219	HPI Injection Check Valve	RB	Open	________
MU-V77A	Cross-Connect Valve	281 AB	Lock Open	________
MU-V77B	Cross-Connect Valve	281 AB	Lock Open	________
MU-V76A	Cross-Connect Valve	281 AB	Lock Closed	______
MU-V76B	Cross-Connect Valve	281 AB	Lock Closed	______

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* The normal operation of this system has one HPI pump running and aligned to the normal charging flow path and RCP seal injection path, with the remaining pumps in standby.

HPI RM A, HPI RM B, and HPI RM C are located on 281 elevation of the Auxiliary Building.

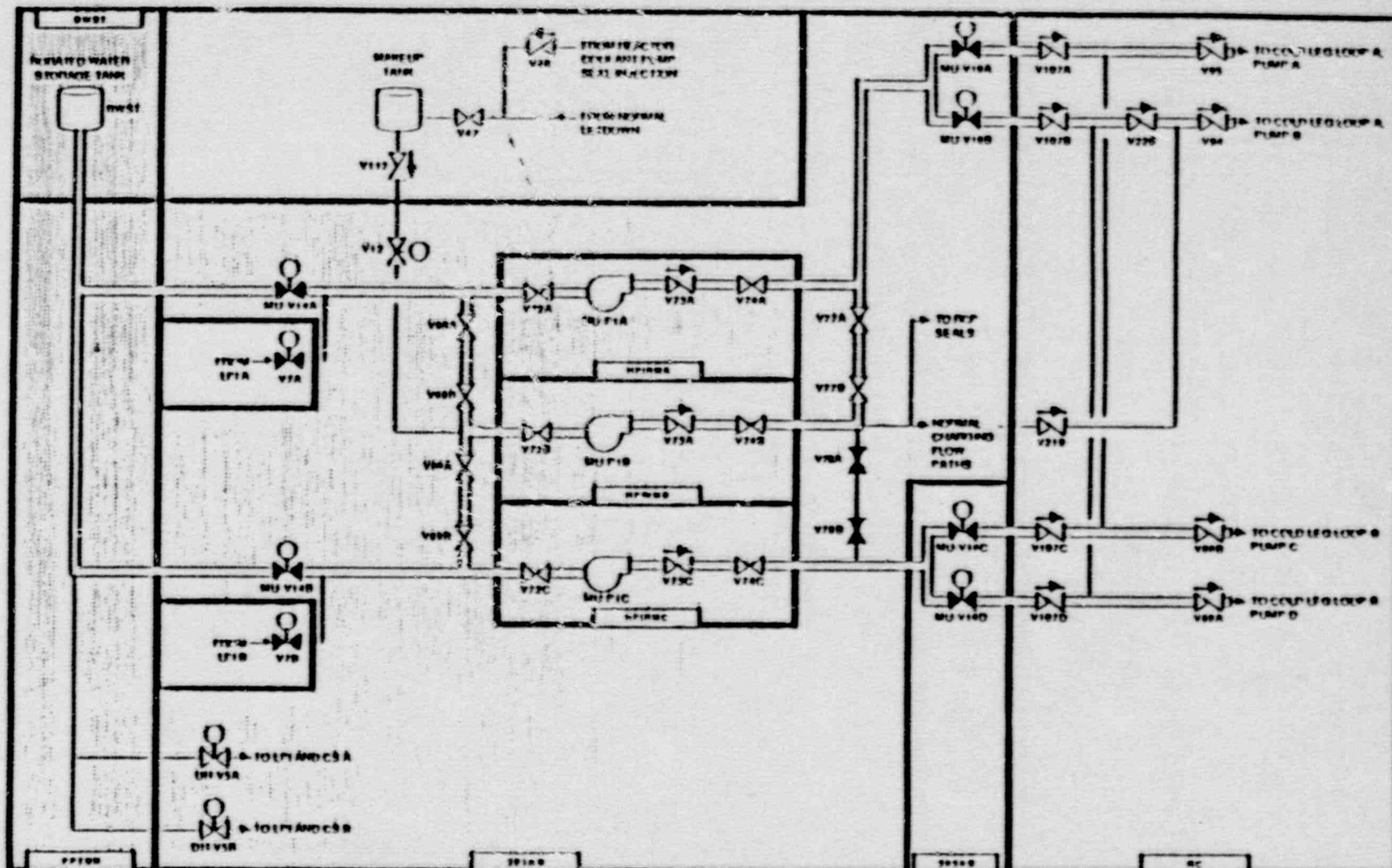

FIGURE A2.1 Simplified System Drawing of High Pressure Injection.

Decay Heat Cooling Water (DHCW) System

Table A3-1 Importance Basis and Failure Mode Identification

CONDITIONS THAT CAN LEAD TO FAILURE

The Decay Heat Cooling Water (DHCW) System consists of the Decay Heat Closed Cooling Water (DHCCW) System and the Decay Heat River Water (DHRW) System. The DHCCW System is comprised of two pumps and associated service coolers, separated into two independent trains with no cross-connection capabilities. Each train of DHCCW provides cooling water to its respective DHR cooler, DHR pump, reactor building spray pump, and HPI pumps 1A or 1C. The DHRW System is comprised of two pumps, separated into two independent trains with no cross-connection capabilities, and supports its respective train of DHCCW by cooling the service coolers. The DHRW System completes the heat transfer path to the ultimate heat sink (i.e., the Susquehanna River).

Since the DHCW System is solely a support system, its success criteria are dependent upon the success criteria of the frontline systems it serves. Based on the frontline system success criteria, DHCW System success is defined as one of the two trains of DHCCW providing cooling water to its respective loads and the associated train of DHRW providing cooling water to the DHCW service coolers.

1. <u>DHRW Pump DR-P1A or DR-P1B Unavailable due to Maintenance</u>

   This fails one train of DHCW. (MT, TS)

2. <u>Failure of DHRW Pump DR-P1A or DR-P1B to Start and Run</u>

   This fails one train of DHCW. (MT, PT)

3. <u>Failure of DHRW Pump Discharge Valve DR-V1A or DR-V1B to Open</u>

   This fails one train of DHCW. (MT, PT)

4. <u>Failure of DHCCW Pump DC-P1A or DC-P1B to Start and Run</u>

   This fails one train of DHCW. (MT, PT)

5. <u>DHCCW Pump DC-P1A or DC-P1B Unavailable due to Maintenance</u>

   This fails one train of DHCW. (MT, TS)

6. <u>DHRW Strainer Unavailable due to Maintenance</u>

This fails one train of DHCW. (MT, TS)

7. <u>Decay Heat Service Cooler DC-HX1A or DC-HX1B Unavailable due to Maintenance</u>

This fails one train of DHCW. (MT, TS)

8. <u>Common Cause Failure of the DHCCW Pumps DC-P1A and DC-P1B</u>

This fails the DHCW System, which results in the failure of the DHR System and thus fails long-term heat removal and low-pressure injection capabilities. (MT, PT)

9. <u>Common Cause Failure of the DHRW Pump Discharge Valves DR-V1A and DR-V1B to Open</u>

This fails the DHCW System, which results in the failure of the DHR System and thus fails long-term heat removal and low-pressure injection capabilities. (MT, PT)

10. <u>Common Cause Failure of the DHRW Pumps DR-P1A and DR-P1B</u>

This fails the DHCW System, which results in the failure of the DHR System and thus fails long-term heat removal and low-pressure injection capabilities. (MT, PT)

Decay Heat Cooling Water (DHCW) System

Table A3-2 Modified System Walkdown*

COMPONENT NUMBER	COMPONENT NAME	LOCATION	REQUIRED POSITION	ACTUAL POSITION
DR-P1A	DHRW Pump	SCREEN HS	Standby	________
DR-P1B	DHRW Pump	SCREEN HS	Standby	________
DR-V1A	DHRW Pump Discharge Valve	SCREEN HS	Closed	________
DR-V1B	DHRW Pump Discharge Valve	SCREEN HS	Closed	________
DC-P1A	DHCCW Pump	305 AB	Standby	________
DC-P1B	DHCCW Pump	305 AB	Standby	________
DR-S1A	DHRW Strainer	SCREEN HS		________
DR-S1B	DHRW Strainer	SCREEN HS		________
DC-HX1A	Decay Heat Service Cooler	HX VT	Standby	________
DC-HX1B	Decay Heat Service Cooler	HX VT	Standby	________

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* HX VT is located on 271 elevation of the Auxiliary Building.

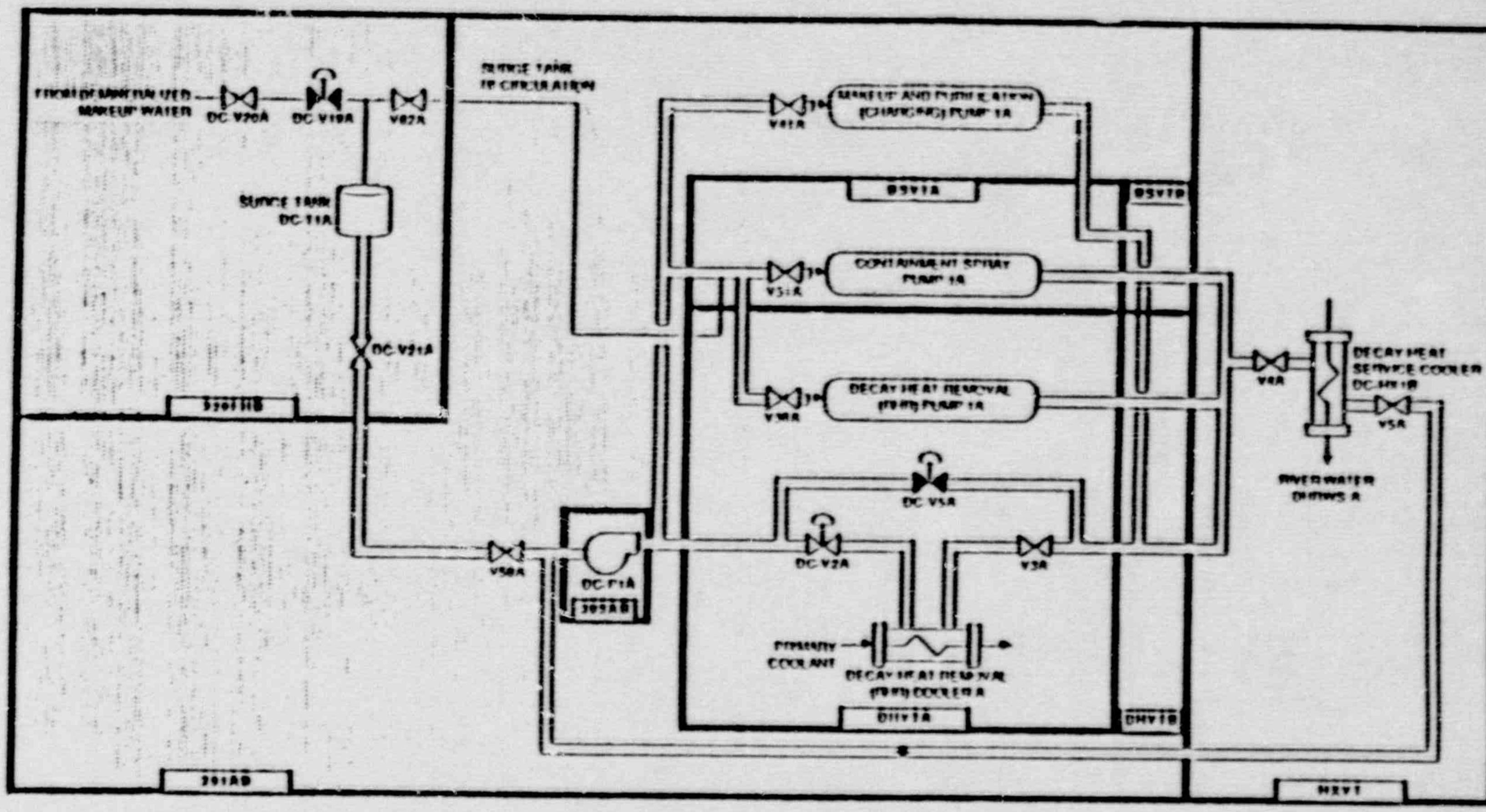

FIGURE A3.1 Simplified System Drawing of Decay Heat Closed Cooling Water Loop A.

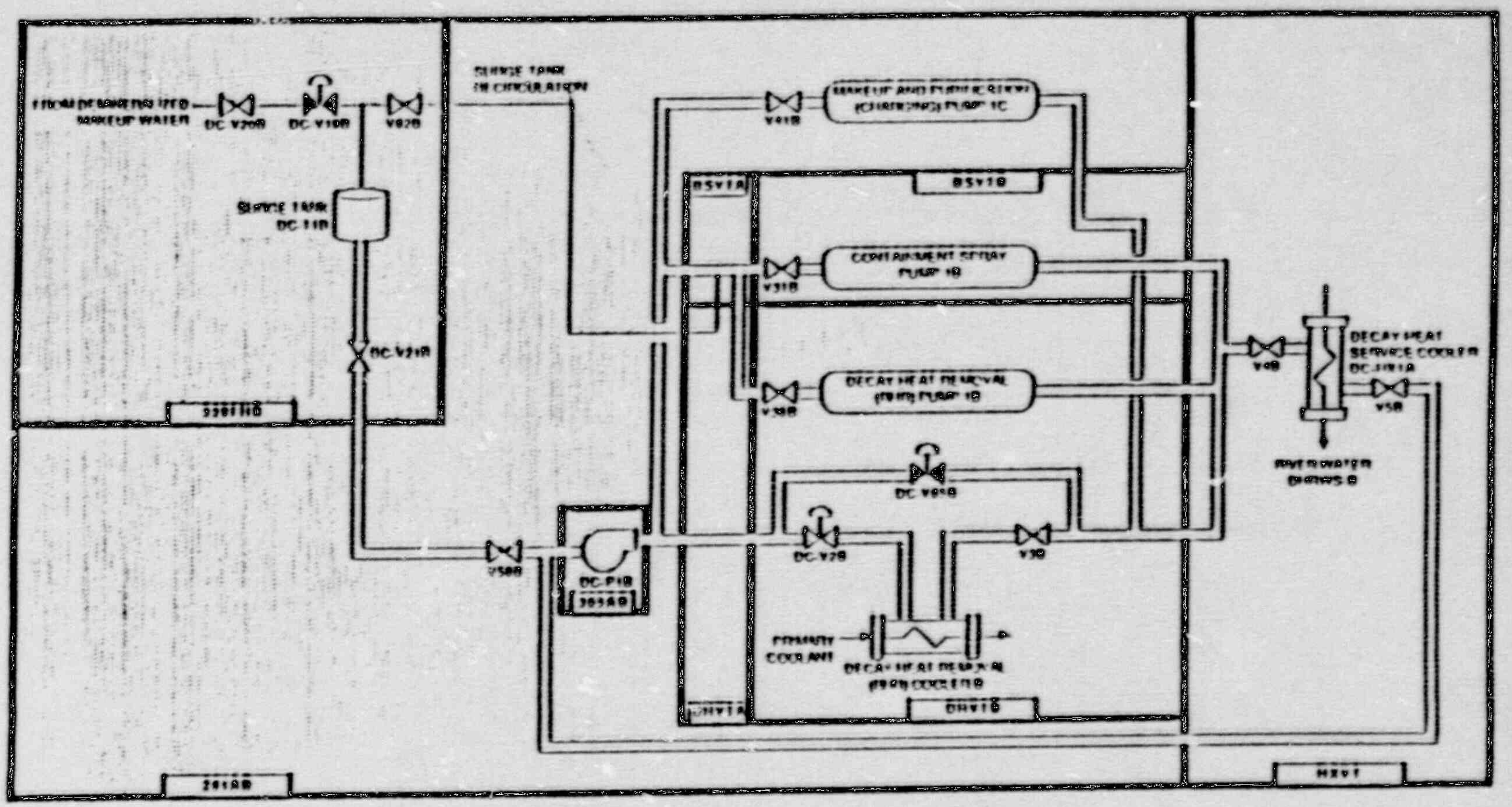

FIGURE A3.2 Simplified System Drawing of Decay Heat Closed Cooling Water Loop B.

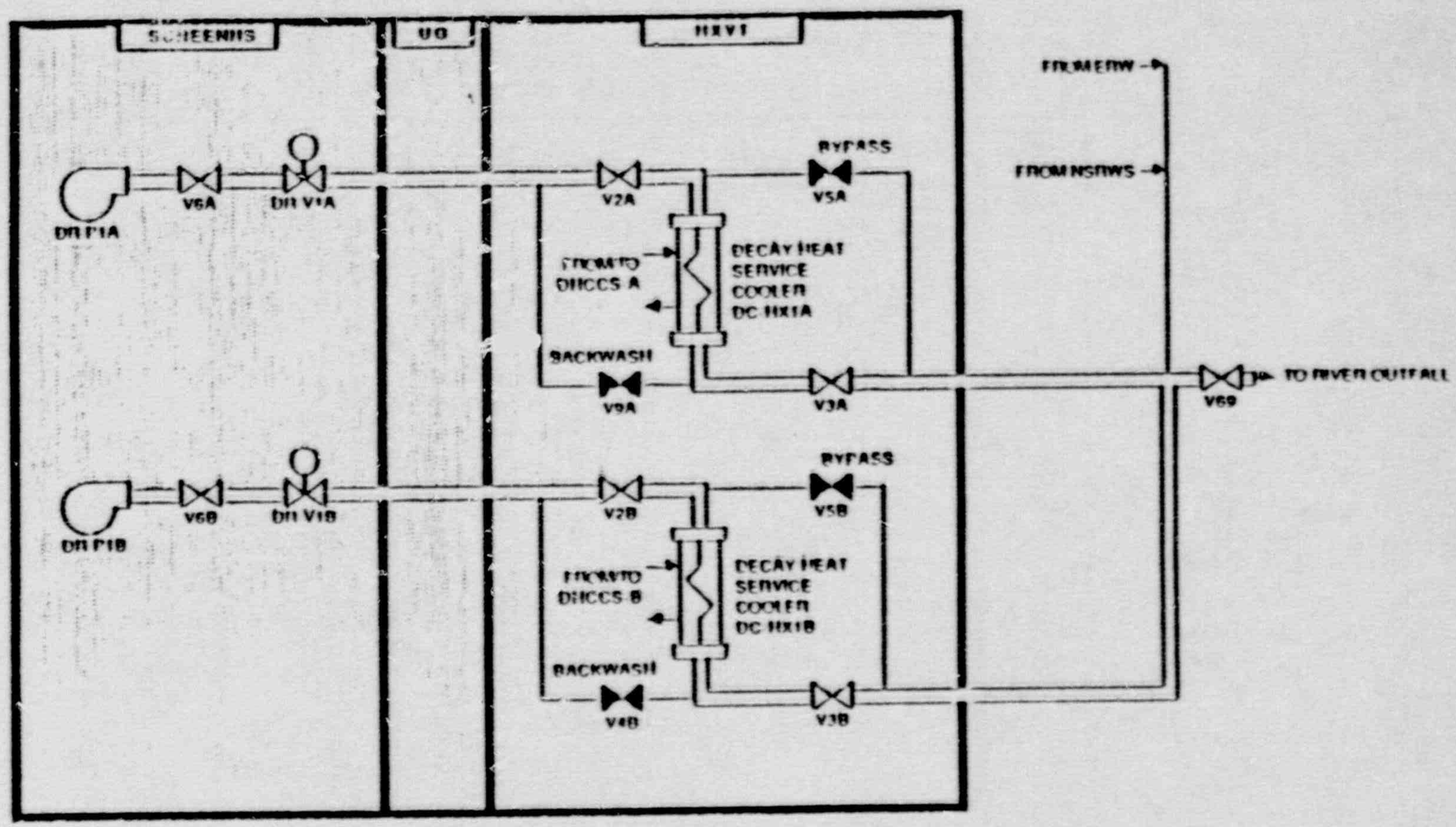

FIGURE A3.3 Simplified System Drawing of Decay Heat River Water.

AC Power System

Table A4-1 Importance Basis and Failure Mode Identification

CONDITIONS THAT CAN LEAD TO FAILURE

The AC Power System supplies AC electrical power to various systems vital for normal operation and/or response to accidents. Among these vital systems are those which shutdown the reactor, remove decay and sensible heat from the reactor core and the containment building, and limit the release of radioactive material from the containment. The AC Power System is required for the operation of motor-operated pumps, fans, valves, and instrumentation and controls in these systems.

During normal operation, AC power is supplied by the offsite network to the 230 kV electrical substation, which in turn supplies power through one of two auxiliary transformers to the onsite electrical distribution system and on to the plant loads. On a loss of this preferred power source, AC power is supplied to vital plant loads by an automatic transfer to the onsite emergency AC power source. This onsite emergency AC power source is composed of two independent diesel generators, which supply power to the vital loads through independent trains of the onsite electrical distribution system, consisting of 4160 V switchgear, 480 V load centers and motor control centers, 120 V instrumentation panels, and associated transformers and circuit breakers.

Since the AC Power System is solely a support system, its success criteria are dependent upon the success criteria of the frontline and support systems it serves. Successful operation of a train of AC power is defined as AC power available to components and/or systems from either the offsite preferred power source or the emergency onsite power source.

1. Failure of the Offsite Grid

   This failure can occur as an initiating event (i.e., loss of offsite power) or can occur at some time after a plant trip has occurred (i.e., blackout following trip). In either case, a demand is placed on the diesel generators to start and provide AC power to the vital components until offsite power can be recovered. (MT, PT, TS)

2. Operator Fails to Recover AC Power

   This failure mode appears in the TMI-1 PRA as the failed operator recovery action to restore AC power from either onsite sources or from offsite source before either the station batteries are depleted (for station blackout conditions) or to recover systems supported by AC power. (OP)

3. <u>Diesel Generator EP-DG1A or EP-DG1B Unavailable due to Maintenance</u>

While a diesel generator is undergoing maintenance, its train of onsite emergency power is assumed to be unavailable and thus essentially failed if needed. (MT, TS)

4. <u>Failure of Diesel Generator EP-DG1A or EP-DG1B to Start and Run</u>

The failure of a diesel generator fails its train of onsite emergency power. (MT, PT)

5. <u>Circuit Breaker Transfers Open</u>

This failure interrupts the AC power being supplied to an electrical bus and can thus fail the motor-operated equipment and electrical buses being supplied by this bus. This failure is more significant at the higher electrical bus levels (e.g., 4160 V switchgear) since its effects can cascade through lower electrical buses. (MT, PT)

6. <u>Failure of an Electrical Bus</u>

This failure results in a failure to supply AC power to the motor-operated equipment and electrical buses being supplied by the failed bus. This failure is more significant at the higher electrical bus levels (e.g., 4160 V switchgear) since its effects can cascade through lower electrical buses. In particular, if the 4160 V switchgear is the failed electrical bus then this entire train of AC power will be failed since this is the initial bus fed by both the preferred offsite power source and the onsite diesel generator. (MT, PT)

7. <u>Failure of a Transformer</u>

This failure interrupts the AC power being supplied to an electrical bus and can thus fail the motor-operated equipment and electrical buses being supplied by through this transformer. This failure can occur at the auxiliary transformer level, resulting in a loss of the preferred offsite source of power, or can occur at lower levels (e.g., 4160 V to 480 V transformer), resulting in AC power loss to motor-operated equipment and buses supplied through the failed transformer. (MT, PT)

8. <u>Common Cause Failure of Diesel Generators EP-DG1A and EP-DG1B to Start and Run</u>

The failure of both diesel generators results in a complete loss of the onsite emergency power source and if coupled with a loss of offsite power can result in station blackout conditions. Under these conditions, numerous systems will be inoperable and only those components supplied by the DC Power System (using the station batteries) will be functional. This situation can be maintained for a few hours, but eventually the station batteries will be depleted. Therefore, recovery of the offsite and/or onsite emergency power sources is essential to safely shutting down the plant under these circumstances. (MT, PT)

THREE MILE ISLAND NUCLEAR STATION UNIT 1
RISK-BASED INSPECTION GUIDE

AC Power System

Table A4-2 Modified System Walkdown*

COMPONENT NUMBER	COMPONENT NAME	LOCATION	REQUIRED POSITION	ACTUAL POSITION
	Offsite Grid	SWITCHYARD	Energized	________
1A	230/4.16 kV Aux. Transformer		Energized	________
1B	230/4.16 kV Aux Transformer		Energized	________
1SA-E2	1A to 1E-ESSB Circuit Breaker	355 CB	Closed	________
1SB-E2	1B to 1E-ESSB Circuit Breaker	355 CB	Open	________
1SA-D2	1A to 1D-ESSA Circuit Breaker	355 CB	Open	________
1SB-D2	1B to 1D-ESSA Circuit Breaker	355 CB	Closed	________
EG-Y1A	Diesel Generator	305 DG	Standby	________
EG-Y1B	Diesel Generator	305 DG	Standby	________
G1-02	DG 1A Circuit Breaker		Open	________
G11-02	DG 1B Circuit Breaker		Open	________
1D-ESSA	4160 V Electrical Bus	338 CB	Energized	________
1E-ESSB	4160 V Electrical Bus	338 CB	Energized	________
1P-ESSA	4160/480 Transformer		Energized	________
1R-ESSA	4160/480 Transformer		Energized	________
1S-ESSB	4160/480 Transformer		Energized	________
1T-ESSB	4160/480 Transformer		Energized	________
1N-ESSA	4160/480 Transformer		Energized	________
1P-ESSA	480 V Electrical Bus	322 CB	Energized	________
1R-ESSA	480 V Electrical Bus	SCREEN HS	Energized	________
1N-ESSA	480 V Electrical Bus	322 TB	Energized	________

1S-ESSB	480 V Electrical Bus	322 CB	Energized	________
1T-ESSB	480 V Electrical Bus	SCREEN HS	Energized	________
1A-ES	480 V Motor Control Center	322 CB	Energized	________
1A-ESV	480 V Motor Control Center	322 CB	Energized	________
1A-ESSH	480 V Motor Control Center	SCREEN HS	Energized	________
1C-ESV	480 V Motor Control Center	281 AB	Energized	________
1B-ES	480 V Motor Control Center	322 CB	Energized	________
1B-ESV	480 V Motor Control Center	322 CB	Energized	________
1B-ESSH	480 V Motor Control Center	SCREEN HS	Energized	________
VBA	120 V Distribution Panel	INV RM A	Energized	________
VBB	120 V Distribution Panel	INV RM B	Energized	________
VBC	120 V Distribution Panel	INV RM A	Energized	________
VBD	120 V Distribution Panel	INV RM B	Energized	________
INV-1A	Inverter	INV RM A	Energized	________
INV-1B	Inverter	INV RM B	Energized	________
INV-1C	Inverter	INV RM A	Energized	________
INV-1D	Inverter	INV RM B	Energized	________
INV-1E	Inverter	INV RM A	Energized	________
Varies	Other Circuit Breakers	Varies	Varies	________

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* Due to the integrated nature of the emergency diesel generator start and run failure modes, the lineup of all automatic diesel generator support functions (e.g., fuel oil, starting air, cooling water, etc.) should be verified.

INV RM A and INV RM B are located on 322 elevation of the Control Building.

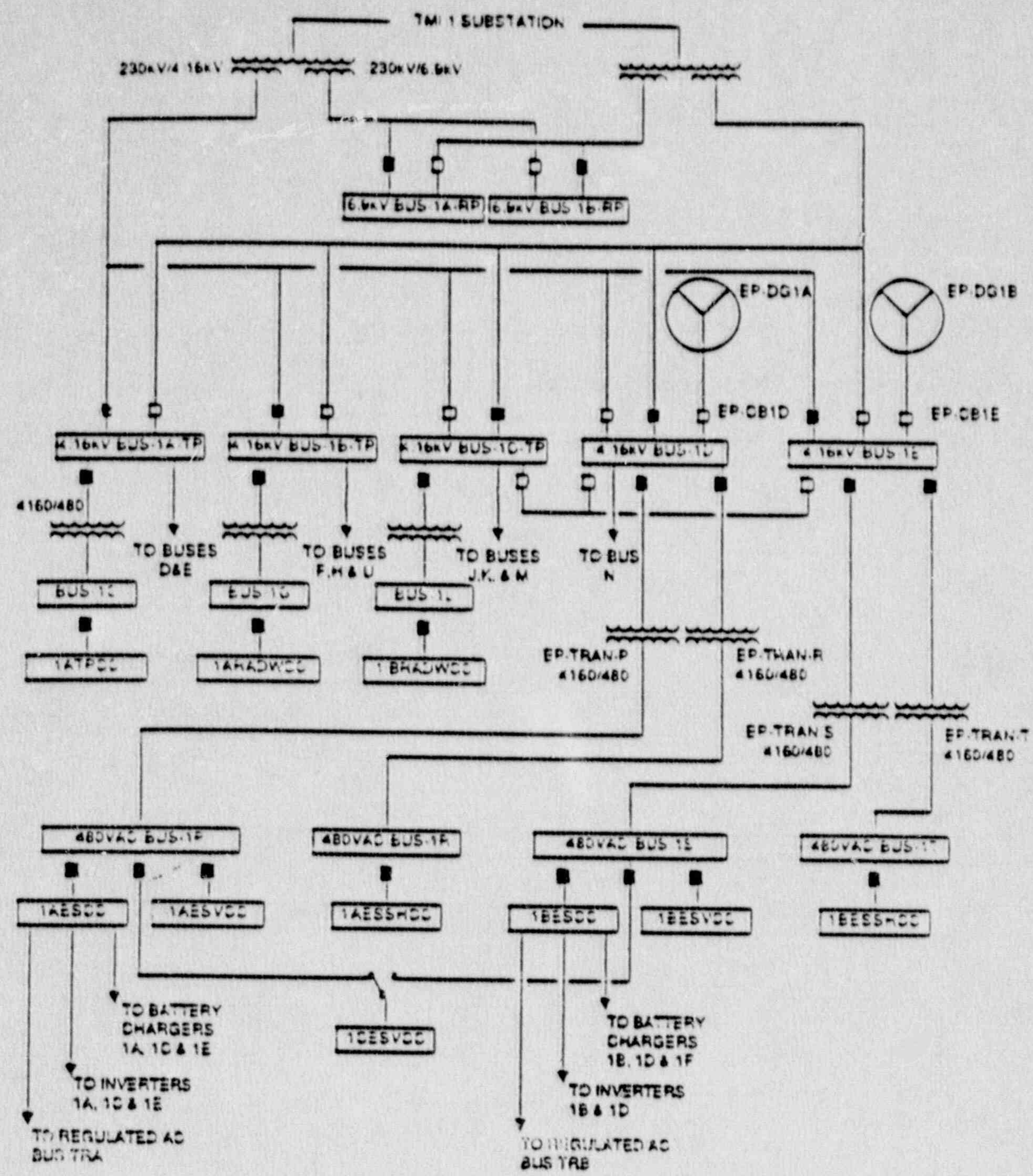

FIGURE A4.1 Simplified System Drawing of 6900V to 480V AC Power.

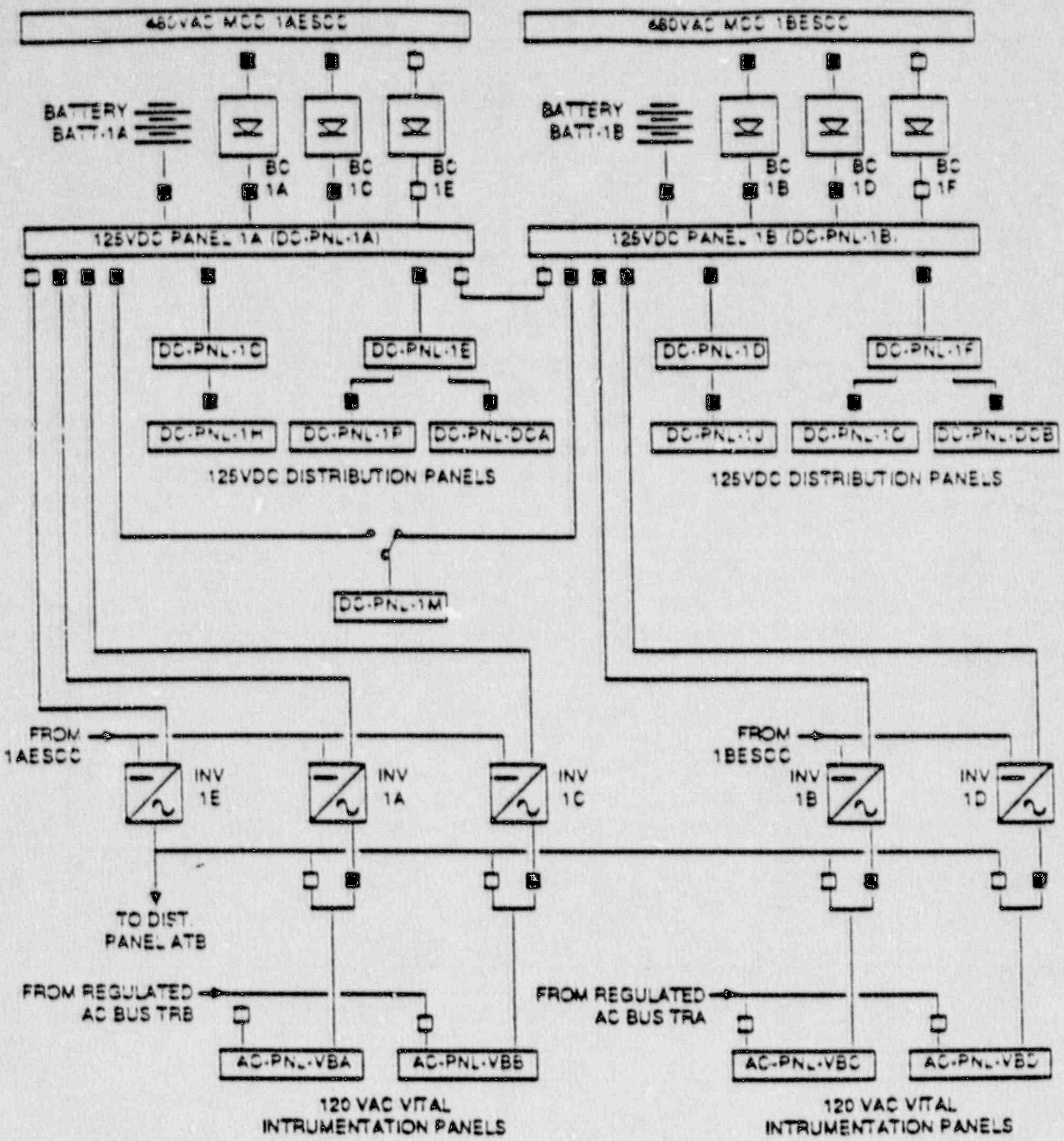

FIGURE A4.2 Simplified System Drawing of 480V to 120V AC Power.

Nuclear Services Cooling Water (NSCW) System

Table A5-1 Importance Basis and Failure Mode Identification

CONDITIONS THAT CAN LEAD TO FAILURE

The Nuclear Services Cooling Water (NSCW) System is separated into the Nuclear Services River Water (NSRW) System and the Nuclear Services Closed Cooling Water (NSCCW) System. The NSCCW System is comprised of three pumps and four NSCW coolers and provides cooling water to numerous systems, including the HPI pumps, the reactor building fans, and the RCP motors. The NSRW System is also comprised of three pumps and supports the NSCCW by providing the cooling water for each of the NSCW coolers. The NSRW System completes the heat transfer path to the ultimate heat sink (i.e., the Susquehanna River). Two NSCCW pumps and two NSRW pumps are normally operating with three of the four NSCW coolers valved in for operation. Upon receiving signals from the ESAS, the pumps in standby also start. Since there are three pumps in each system and only two trains of power, it should be noted that two pumps in each system receive power from the same train of power.

Since the NSCW System is solely a support system, its success criteria are dependent upon the success criteria of the frontline systems it serves. Based on the frontline system success criteria, NSCW System success is defined as at least one of the three NSCCW pumps providing cooling water to the supported systems and at least one of the three NSRW pumps providing cooling water to at least two of the four NSCW coolers. The TMI-1 PRA also considered this success criterion in determining the potential for the NSCW System to cause a plant trip and thus be an initiating event.

1. Operator Fails to Clear the River Water Screen

   This failure appears in the TMI-1 PRA as a restoration action by the operator to clear the river water screen to avoid tripping the plant. It is included as part of the loss of river water initiating event. (OP)

2. Failure to Recover River Water

   This failure mode appears in the TMI-1 PRA as a failed recovery action by the operator. (OP)

3. Failure of NSRW Pump NR-P1A, NR-P1B, or NR-P1C or Failure of NSCCW Pump NS-P1A, NS-P1B, or NS-P1C

   Single or double failures reduce the redundancy in the affected system. (MT, PT)

4. Rupture or Leakage of One of the NSCCW Supported Heat Exchangers or Rupture or Leakage of NSCW Heat Exchanger NS-HX1A, NS-HX1B, NS-HX1C, NS-HX1D

   This failure involves the leakage from a heat exchanger at a high enough rate that the normal makeup pump DW-P1 cannot keep up. This failure mode is important for the system in providing cooling to other systems and also as an initiating event. (MT, PT)

5. Operator Fails to Isolate Leaking Heat Exchanger (Cooler)

   This failure involves the leakage from a heat exchanger, as mentioned in failure mode 4 above. A relatively high probability for the operator failing to locate and isolate the leak in time to prevent system failure was assigned in the TMI-1 PRA. This is because the heat exchangers are located in two different buildings and the warning time provided by the surge tank low-level alarm may be only a few minutes. (OP)

6. Failure of NSRW Common Header Isolation Valve NR-V3 or NR-V5 to Remain Open

   Since all three NSRW pump trains are headered into a common line, failure of an isolation valve to remain open will fail the flow to the NSCW cooler. This failure mode is important as an initiating event due to the longer time involved (i.e., 1 year mission time for initiating events versus 24 hours for accident response) and thus a greater potential for failure during the time period. (MT, PT)

7. Common Cause Failure of NSRW Pumps NR-P1A, NR-P1B, and NR-P1C or Common Cause Failure of NSCCW Pumps NS-P1A, NS-P1B, and NS-P1C

   Either common cause failure results in the failur[illegible] NSCW System. (MT, PT)

8. Failure of NSCCW Common Header Piping

   Since all three NSCCW pump trains are headered into a common line, failure of this line can fail the system's ability to providing cooling to the supported systems. This failure mode is only important when the NSCW System is analyzed as an initiating event. This is due to the longer mission time involved, which increases the probability that this failure mode will occur during its mission time. (MT, PT)

9. NSRW Pump NR-P1A, NR-P1B, or NR-P1C or NSCCW Pump NS-P1A, NS-P1B, or NS-P1C Unavailable due to Maintenance

   This fails one of the pumps thus reducing the redundancy of the system. (MT, TS)

10. Failure of the NSCCW Surge Tank NS-T1

    Failure of the surge tank will fail the system's ability to providing cooling to the supported systems. This failure mode is only important when the NSCW System is analyzed as an initiating event. This is due to the longer mission time involved, which increases the probability that this failure mode will occur. (MT, PT)

11. Common Cause Failure of the NSCCW Heat Exchangers (Coolers)

The plugging of most, if not all, NSCCW supported heat exchangers will fail the ability of the NSCW System to cool these systems. (MT, PT)

Nuclear Services Cooling Water (NSCW) System

Table A5-2 Modified System Walkdown*

COMPONENT NUMBER	COMPONENT NAME	LOCATION	REQUIRED POSITION	ACTUAL POSITION
	NSRW Screen	SCREEN HS		______
NR-P1A	NSRW Pump	SCREEN HS		______
NR-P1B	NSRW Pump	SCREEN HS		______
NR-P1C	NSRW Pump	SCREEN HS		______
NS-P1A	NSCCW Pump	305 AB		______
NS-P1B	NSCCW Pump	305 AB		______
NS-P1C	NSCCW Pump	305 AB		______
Varies	NSCW Supported Heat Exchanger	Varies		______
NS-HX1A	NSCW Heat Exchanger	HX VT		______
NS-HX1B	NSCW Heat Exchanger	HX VT		______
NS-HX1C	NSCW Heat Exchanger	HX VT		______
NS-HX1D	NSCW Heat Exchanger	HX VT		______
DW-P1	Normal NSCW Makeup Pump			______
NR-V3	NSRW Isolation Valve	SCREEN HS	Open	______
NR-V5	NSRW Isolation Valve	HX VT	Open	______
	NSCCW Common Header Pipe	305 AB		______
NS-T1	NSCCW Surge Tank	348 FHB		______

* The normal operation of this system has two NSCCW pumps and two NSRW pumps running, with the remaining pumps in standby. Three NSCW heat exchangers are also typically valved in for operation.

HX VT is located on 271 elevation of the Auxiliary Building.

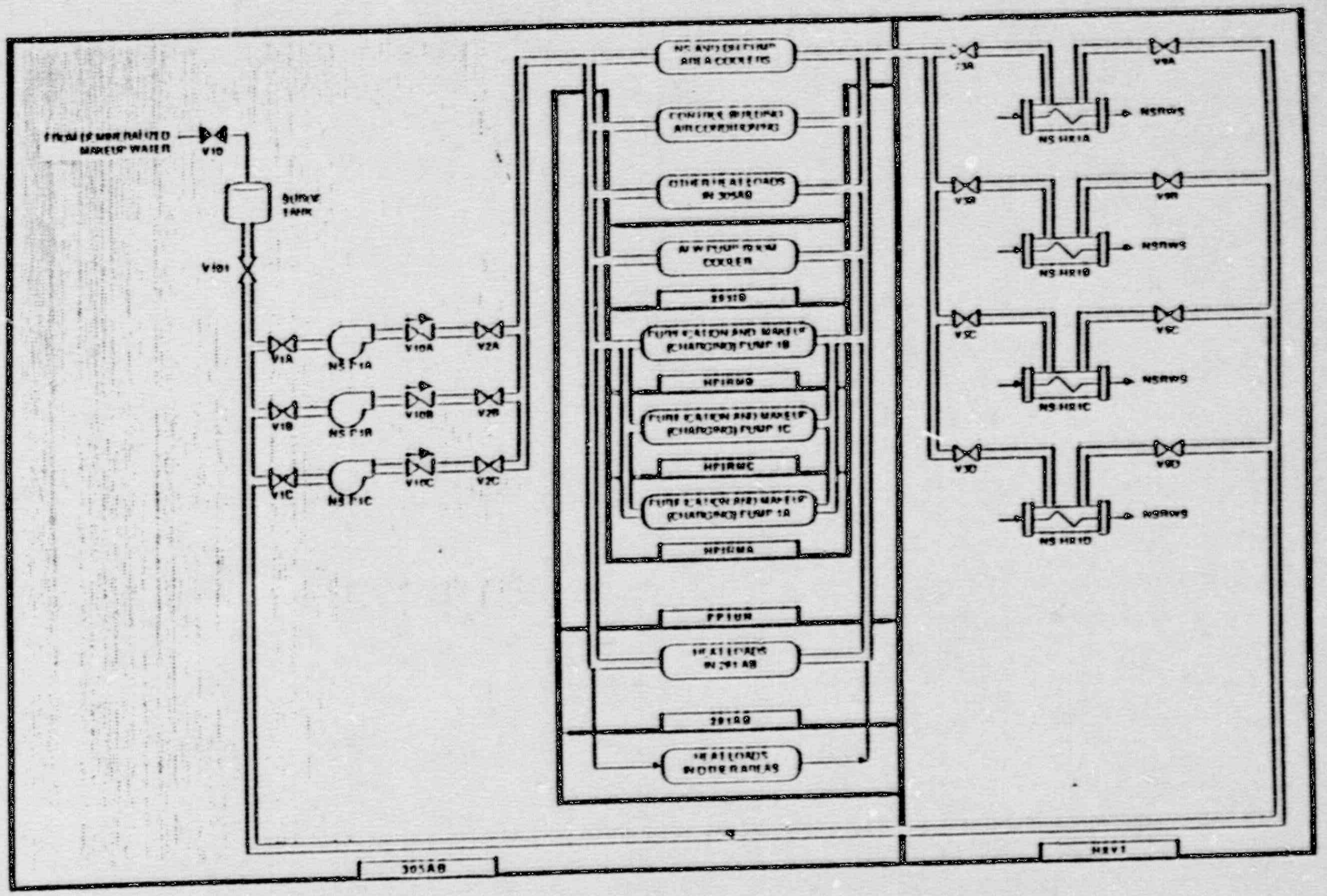

FIGURE A5.1 Simplified System Drawing of Nuclear Services Closed Cooling Water.

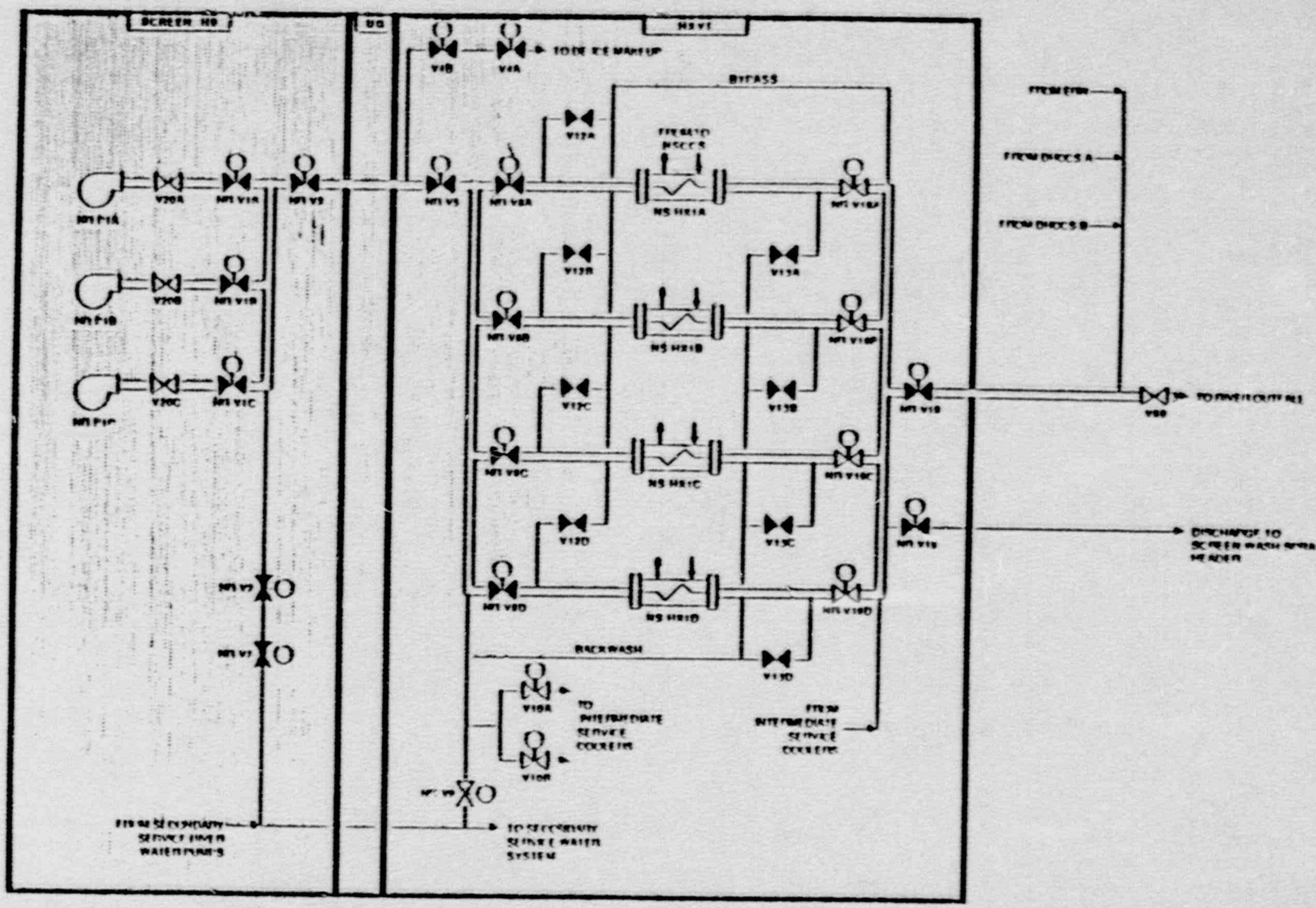

FIGURE A5.2 Simplified System Drawing of Nuclear Services River Water.

THREE MILE ISLAND NUCLEAR STATION, UNIT 1
RISK-BASED INSPECTION GUIDE

Main Steam System

Table A6-1 Importance Basis and Failure Mode Identification

CONDITIONS THAT CAN LEAD TO FAILURE

The Main Steam System delivers steam from the steam generators to the high-pressure turbine and the main feedwater turbines during startup, power operation, and when shutting down. Under station blackout, loss of both main feedwater pumps, or loss of all four reactor coolant pump conditions, the Main Steam System delivers steam to the EFW System pump turbine. In the event of a loss of partial load, this system dissipates energy in the RCS through steam relief via the turbine bypass valves (TBVs) to the condenser and via the main steam safety valves and atmospheric dump valves (ADVs) to the environment. This system can also be used to cooldown the RCS to establish the conditions necessary for entry into long-term decay heat removal. At low loads the TBVs and ADVs may need to be modulated to prevent excessive cooling of the RCS.

The Main Steam System consists of two main steam lines from each steam generator to the high-pressure turbines. Each main line is furnished with a main steam isolation stop valve, four or five main steam safety valves, and branch lines that supply steam to the main feedwater and emergency feedwater turbines. Also located on branch lines off the main steam lines are the TBVs (three per steam generator) and the ADVs (one per steam generator).

For the operation of this system to be deemed successful, the valves must operate when demanded and reseat after the demand is removed. Because of the large redundancy of available valves for secondary steam relief (i.e., there are nine main steam safety valves per steam generator of which only one or two are required to open), the probability of steam relief failure is very small. The TMI-1 PRA determined that there were two system failures of importance. The first results from the failure of valves to reseat after opening in response to an initiating event and thus causes an overcooling of the RCS and/or an unisolated path from the steam generator to the environment. The other system failure of importance was the failure of the ADVs to operate to cooldown and depressurize the RCS for entry into long-term decay heat removal. This failure mode is closely related to failures of the RCS-Pressure Control System (see Table A7)

1. <u>Failure of One of Nine Main Steam Safety Valves on Either Steam Generator to Reseat</u>

   This failure results in an unisolated path to the environment and potential for overcooling the RCS. (MT, PT)

2. <u>Failure of Turbine Bypass Valve MS-V3A, MS-V3B, MS-V3C, MS-V3D, MS-V3E, or MS-V3F or Atmospheric Dump Valve MS-V4A or MS-V4B to Close</u>

This failure causes isolation failure and may result in RCS overcooling conditions. (MT, PT)

3. <u>Failure of Atmospheric Dump Valve MS-V4A or MS-V4B to Open and/or Modulate</u>

The failure of ADVs to open and/or modulate results in an inability to bring the RCS pressure down to a level at which entry into long-term decay heat removal can be accomplished. (MT, PT)

4. <u>Failure of Emergency Feedwater Pump Turbine Valve MS-V10A, MS-V10B, MS-V13A, or MS-V13B to Close</u>

This failure results in an isolation failure for those conditions requiring such isolation. This failure is particularly important under conditions of a loss of instrument air coincident with a need for isolation, since valves MS-V13A and MS-V13B are air-operated and fail open on a loss of instrument air. (MT, PT)

5. <u>Intermittent Operation Failure of ADVs MS-V4A or MS-V4B and TBVs MS-V3A, MS-V3B, MS-V3C, MS-V3D, MS-V3E, or MS-V3F</u>

This results in an isolation failure. (MT, PT)

THREE MILE ISLAND NUCLEAR STATION UNIT 1
RISK-BASED INSPECTION GUIDE

Main Steam System

Table A6-2 Modified System Walkdown*

COMPONENT NUMBER	COMPONENT NAME	LOCATION	REQUIRED POSITION	ACTUAL POSITION
MS-V17A	Main Steam Safety Valve	322 IB	Closed	________
MS-V18A	Main Steam Safety Valve	322 IB	Closed	________
MS-V19A	Main Steam Safety Valve	322 IB	Closed	________
MS-V20A	Main Steam Safety Valve	322 IB	Closed	________
MS-V21A	Main Steam Safety Valve	322 IB	Closed	________
MS-V17B	Main Steam Safety Valve	322 IB	Closed	________
MS-V18B	Main Steam Safety Valve	322 IB	Closed	________
MS-V19B	Main Steam Safety Valve	322 IB	Closed	________
MS-V20B	Main Steam Safety Valve	322 IB	Closed	________
MS-V17C	Main Steam Safety Valve	322 IB	Closed	________
MS-V18C	Main Steam Safety Valve	322 IB	Closed	________
MS-V19C	Main Steam Safety Valve	322 IB	Closed	________
MS-V20C	Main Steam Safety Valve	322 IB	Closed	________
MS-V21C	Main Steam Safety Valve	322 IB	Closed	________
MS-V17D	Main Steam Safety Valve	322 IB	Closed	________
MS-V18D	Main Steam Safety Valve	322 IB	Closed	________
MS-V19D	Main Steam Safety Valve	322 IB	Closed	________
MS-V20D	Main Steam Safety Valve	322 IB	Closed	________
MS-V3A	Turbine Bypass Valve	322 TB	Closed	________
MS-V3B	Turbine Bypass Valve	322 TB	Closed	________
MS-V3C	Turbine Bypass Valve	322 TB	Closed	________

MS-V3D	Turbine Bypass Valve	322 TB	Closed	________
MS-V3E	Turbine Bypass Valve	322 TB	Closed	________
MS-V3F	Turbine Bypass Valve	322 TB	Closed	________
MS-V4A	Atmospheric Dump Valve	295 IB	Closed	________
MS-V4B	Atmospheric Dump Valve	295 IB	Closed	________
MS-V10A	EFW Pump Turbine Valve	295 IB	Closed	________
MS-V10B	EFW Pump Turbine Valve	295 IB	Closed	________
MS-V13A	EFW Pump Turbine Valve	295 IB	Closed	________
MS-V13B	EFW Pump Turbine Valve	295 IB	Closed	________

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* Walkdown is ineffective against main steam safety valve failures to reseat.

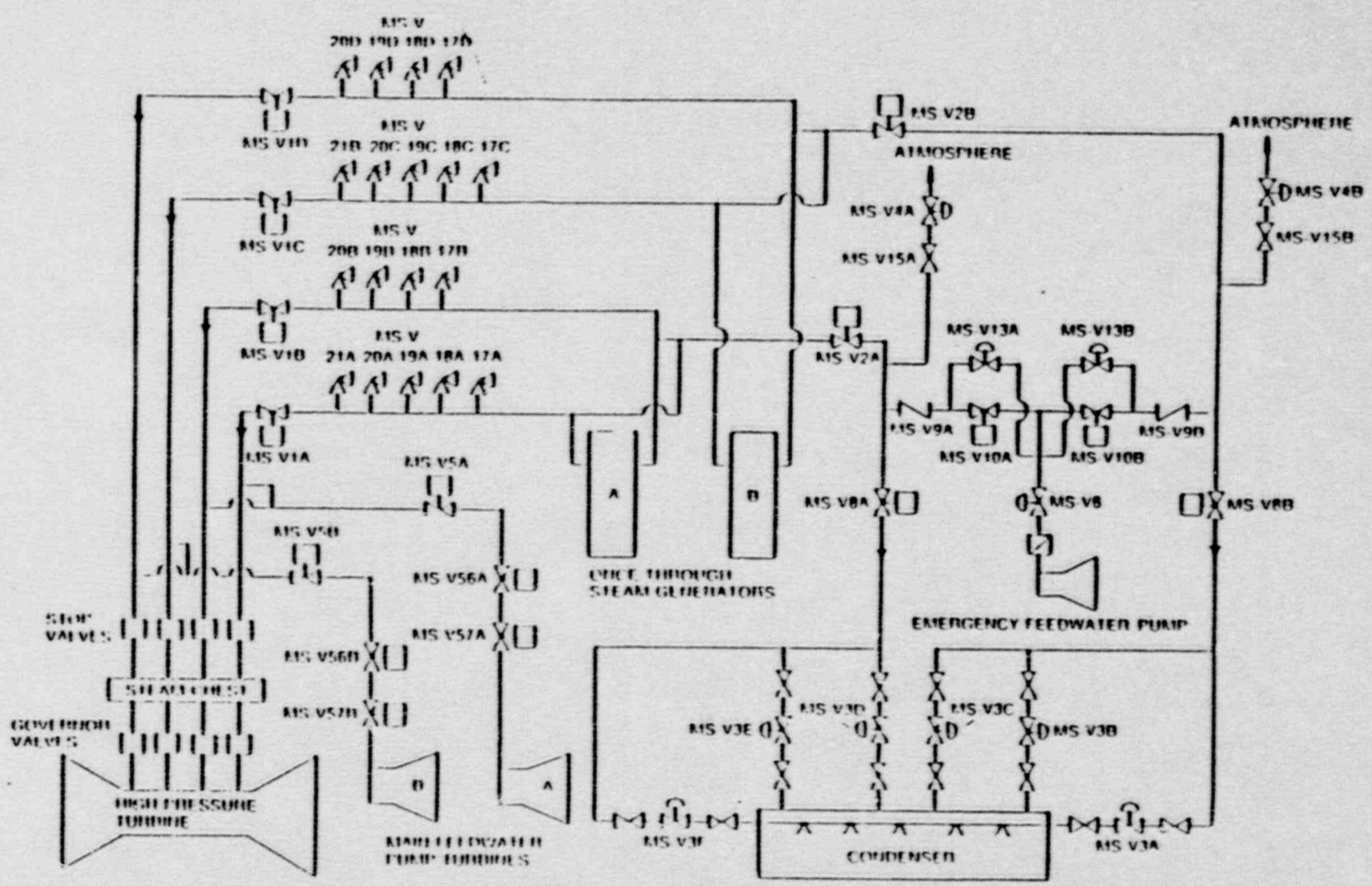

FIGURE A6.1 Simplified System Drawing of Main Steam.

CONDITIONS THAT CAN LEAD TO FAILURE

The Emergency Feedwater (EFW) System is used to supply feedwater to the steam generators whenever the Main Feedwater System is unavailable and the RCS pressure is too high to permit heat removal by the DHR System. The system consists of two motor-driven pump trains and one steam turbine-driven pump train, which take suction from the two condensate storage tanks and can be aligned to take suction from the Emergency River Water System. The EFW turbine is supplied with steam from both steam generators.

Operational success of the EFW System is defined as the supply of the appropriate amount of feedwater to the steam generators to remove decay heat. This is achieved by EFW flow from at least one of the three pumps to at least one of the two steam generators. It is also necessary to control the EFW flow to prevent overcooling the RCS by overfeeding the steam generators for the first few hours following an initiating event.

1. Operator Fails to Locally Control EFW Flow Through Control Valves EF-V30A, EF-V30B, EF-V30C, and EF-V30D

   This failure occurs after a loss of the normal compressed air supply (as a result of an initiating event) and depletion of the two-hour backup train of air bottles causes the EFW flow control valves to fail closed. Failure of the operator to manually open these valves results in insufficient feedwater being supplied to the steam generators. (OP)

2. Operator Fails to Replenish the 2-Hour Air Bottles

   This failure is similar to failure 1 above. In this scenario, the normal compressed air supply is lost and the 2-hour backup train of air bottles supply the EFW flow control valves. However, the operator fails to replenish the 2-hour air bottles and, therefore, after depleting the air bottles the flow control valves fail closed and stop the feedwater supply to the steam generators. (OP)

3. Failure of Turbine-Driven Pump EF-P1 to Start and Run

   This failure results in the failure to provide feedwater to the steam generators independent of electrical power, since the only other pumps available are both motor-operated pumps. The occurrence of a station blackout coincident with this failure results in no secondary side heat removal capability. (MT, PT)

4. <u>Failure of the 2-Hour Train of Air Bottles</u>

This failure yields similar results to the operator erros addressed in items 1 and 2 above. It is important for sequences involving the loss of the normal compressed air supply, which then require the backup air bottles to control the EFW flow to the steam generators. Failure of the 2-hour train of air bottles causes the EFW flow control valves to fail closed, requiring local operator action to throttle open these valves. This failure can be caused by the rupture of one of the seven air bottles on the train, inadvertent isolation of the air bottles, or inadvertent venting of the air bottles while in standby. The venting failure results from the inadvertent position transfer of switching valves IA-V1625A and IA-V1625B (position changes which cannot be detected except during the annual operability testing of the air system). (MT, PT)

5. <u>Failure of Motor-Driven Pump EF-P2A or EF-P2B to Start and Run</u>

This failure results in one train of EFW not being able to supply feedwater to the steam generators. (MT, PT)

6. <u>Motor-Driven Pump EF-P2A or EF-P2B Unavailable due to Maintenance</u>

This failure results in one train of EFW not being able to supply feedwater to the steam generators. (MT, TS)

7. <u>Control Valves EF-V30A, EF-V30B, EF-V30C, or EF-V30D and Block Valves EF-V52, EF-V53, EF-V54, EF-V55 Unavailable due to Testing</u>

Channel actuation and control circuitry testing may fail one train of EFW. (PT, TS)

THREE MILE ISLAND NUCLEAR STATION UNIT 1
RISK-BASED INSPECTION GUIDE

Emergency Feedwater (EFW) System

Table A7-2 Modified System Walkdown

COMPONENT NUMBER	COMPONENT NAME	LOCATION	REQUIRED POSITION	ACTUAL POSITION
EF-V30A	EFW Flow Control Valve	295 IB	Open	______
EF-V30B	EFW Flow Control Valve	295 IB	Open	______
EF-V30C	EFW Flow Control Valve	295 IB	Open	______
EF-V30D	EFW Flow Control Valve	295 IB	Open	______
EF-P1	Turbine-Driven Pump	295 IB	Standby	______
	7 Train A 2-Hour Air Bottles	305 DG	Standby	______
	7 Train B 2-Hour Air Bottles	305 DG	Standby	______
IA-V1625A	2-Hour Air Switching Valve	305 DG	Open(Bottle)	______
IA-V1626A	Normal Air Switching Valve	305 DG	Open(to IA)	______
IA-V1625B	2-Hour Air Switching Valve	305 DG	Open(Bottle)	______
IA-V1626B	Normal Air Switching Valve	305 DG	Open(to IA)	______
EF-P2A	Motor-Driven Pump	295 IB	Standby	______
EF-P2B	Motor-Driven Pump	295 IB	Standby	______
EF-V52	Block Valve	295 IB	Open	______
EF-V53	Block Valve	295 IB	Open	______
EF-V54	Block Valve	295 IB	Open	______
EF-V55	Block Valve	295 IB	Open	______

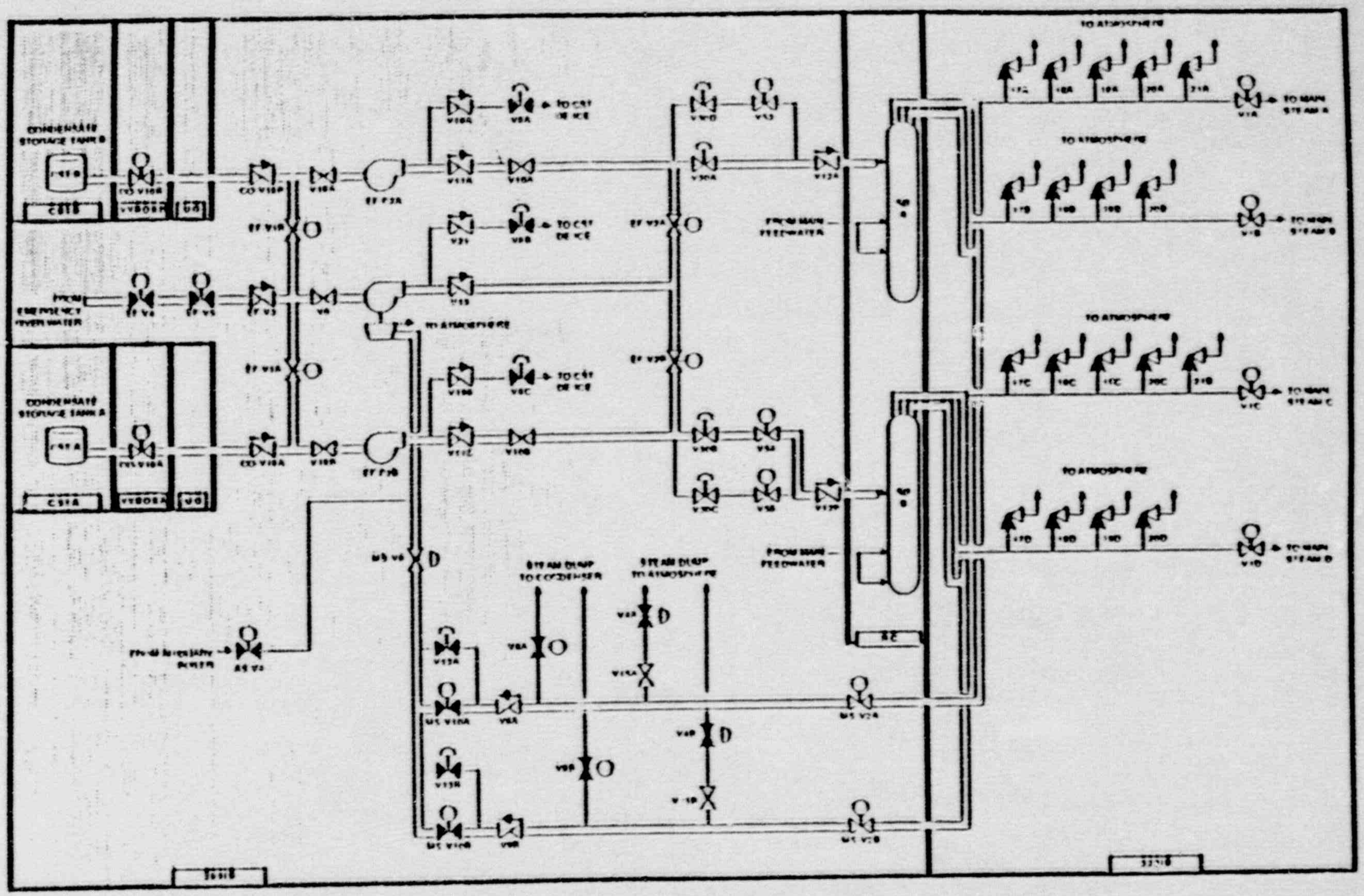

FIGURE A7.1 Simplified System Drawing of Emergency Feedwater.

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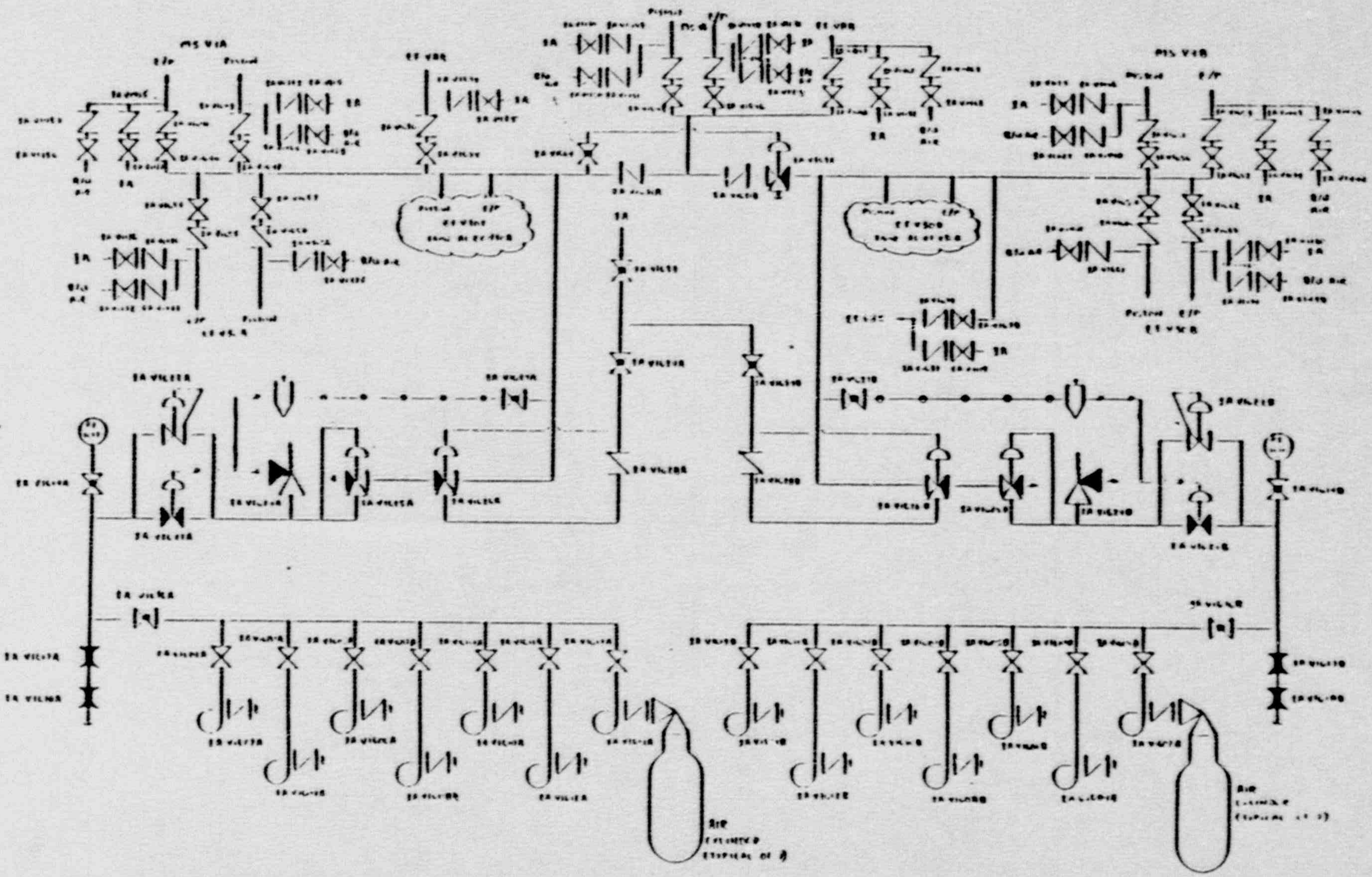

FIGURE A7.2 Simplified System Drawing of Emergency Feedwater 2-Hour Air Supply.

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Reactor Coolant System - Pressure Control System

Table A8-1 Importance Basis and Failure Mode Identification

CONDITIONS THAT CAN LEAD TO FAILURE

The 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. In addition, this system is used as a relief path for HPI cooling (i.e., feed and bleed) and can be used to cooldown and depressurize the RCS to establish the conditions necessary for entry into long-term decay heat removal.

1. <u>Failure of Pressurizer Safety Valve RC-RV1A or RC-RV1B to Reseat</u>

   This failure can occur either after relieving the pressure induced by a transient event or after passing water caused by the HPI pumps filling the RCS with water and continuing makeup. These conditions result in a transient-induced LOCA scenario. (MT, PT)

2. <u>PORV RC-RV2 Unavailable due to Maintenance</u>

   This failure results in an inability of the pressurizer PORV to relieve the increased pressure induced by a transient event. It also fails rapid cooldown and depressurization capabilities. (MT, TS)

3. <u>Operator Fails to Initiate Cooldown</u>

   The operator fails to initiate cooldown using either the Main Steam System ADVs and pressurizer sprays or a PORV. (OP)

Reactor Coolant System - Pressure Control System

Table A8-2 Modified System Walkdown

COMPONENT NUMBER	COMPONENT NAME	LOCATION	REQUIRED POSITION	ACTUAL POSITION
RC-RV1A	Pressurizer Safety Valve	RB	Closed	________
RC-RV1B	Pressurizer Safety Valve	RB	Closed	________
RC-RV2	Pressurizer PORV	RB	Closed	________
RC-V2	PORV Block Valve	RB	Open	________
RC-V1	Pressurizer Spray Valve	RB	Closed	________
RC-V3	Pressurizer Spray Block Valve	RB	Open	________
RC-V31	Pressurizer Spray Valve	RB	Open	________

* Walkdown is ineffective against failures of the pressurizer safety valves and PORV to reseat.

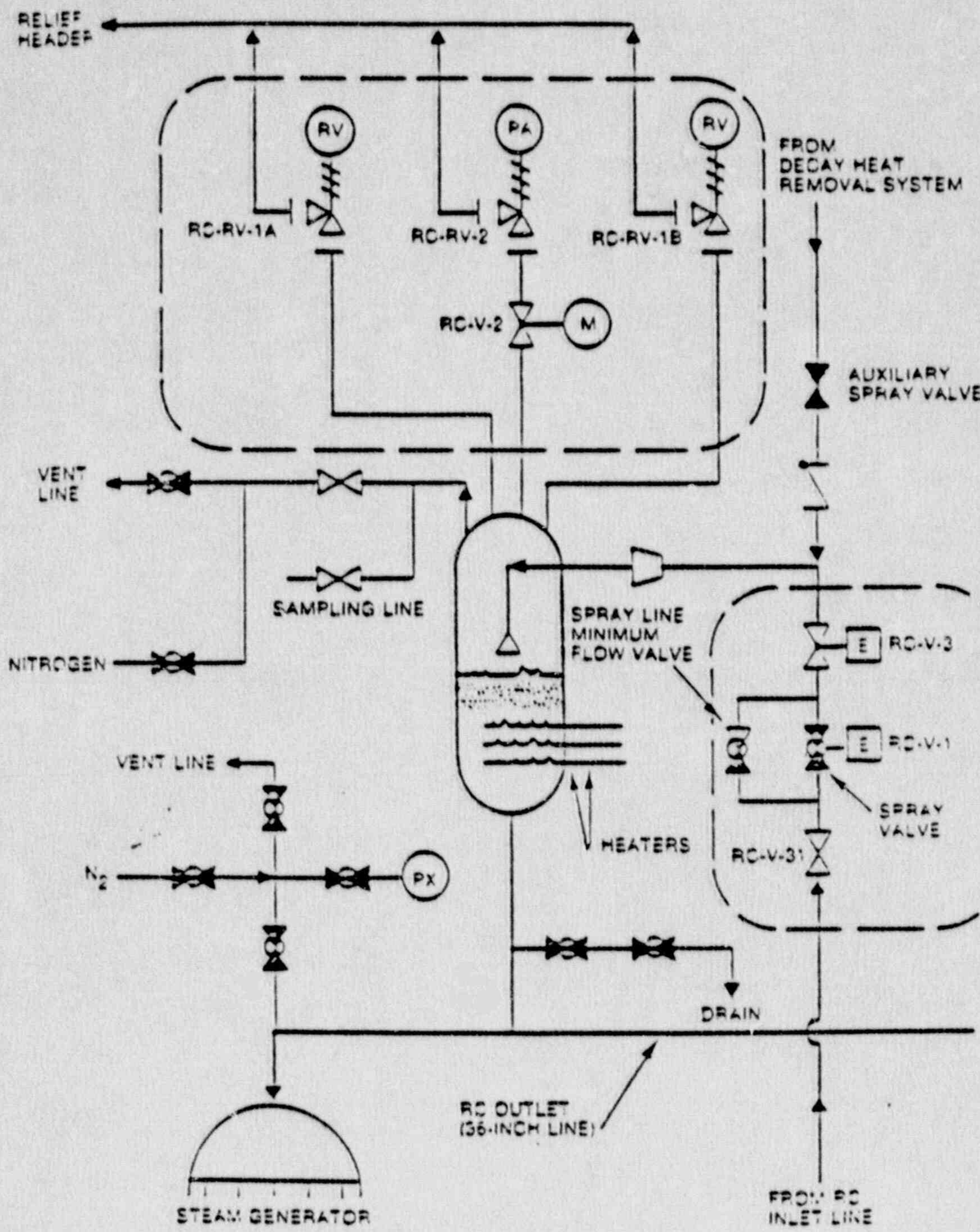

FIGURE A8.1 Simplified System Drawing of Reactor Coolant System Pressure Control.

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Intermediate Closed Cooling Water (ICCW) System

Table A9-1 Importance Basis and Failure Mode Identification

CONDITIONS THAT CAN LEAD TO FAILURE

Though the Intermediate Closed Cooling Water (ICCW) System is not an engineered safeguards system, it is essential for the normal operation of TMI-1, since it provides cooling to the RCP thermal barriers. When this system is operated the RCP seals are prevented from becoming overheated and leaking, even if the HPI System fails to provide RCP seal injection.

Typically, one of the two pumps is operating while the other pump is in standby. However, on a loss of offsite power the operating pump will stop and must be manually started. Since this system is not designated as a safety system it does not have any mandatory surveillance requirements or technical specifications.

Success is defined in the TMI-1 PRA as one of two ICCW pumps and heat exchangers providing sufficient cooling flow to all four RCP thermal barriers.

1. Failure of ICCW Air-Operated Valves IC-V3 or IC-V4 to Remain Open

   The failure of either ICCW air-operated valve IC-V3 or IC-V4 to remain open will fail the cooling to the RCP thermal barriers. If a loss of offsite power occurs, these valves will fail closed due to the ensuing loss of air to the valves and thus failure of RCP thermal barrier cooling would be guaranteed. (MT, PT)

2. Failure due to Reverse Leakage of the Pump Discharge Check Valves IC-V13A or IC-13B

   The failure of discharge check valves on the idle ICCW pump, either IC-V13A or IC-V13B, due to reverse leakage will fail the flow from the ICCW operating pump by diverting the flow through the leaking check valve of the idle pump and not provide sufficient flow to the RCP thermal barriers. (MT, PT)

3. Failure of the ICCW Heat Exchangers IC-C1A and IC-C1B

   Failure of the ICCW heat exchangers due to plugging or from other causes results in a failure to cool the RCP thermal barriers and/or failure in the return line. (MT, PT)

4. ICCW Pumps IC-P1A or IC-P1B Unavailable due to Maintenance

This fails one of the ICCW pumps. (MT)

5. Failure of ICCW Pump IC-P1A or IC-P1B to Start and Run

This fails one of the ICCW pumps and is especially important after a loss of offsite power since the operating pump will stop and need to be restarted manually. (MT, PT)

6. Failure of Pump Discharge Check Valve IC-V13A or IC-13B to Open

This failure results in the failure of the flow from the ICCW pumps. (MT, PT)

7. Failure of Motor-Operated Valves IC-V2, IC-V79A, IC-V79B, IC-V79C, or IC-V79D to Open

The failure of any one of five motor-operated valves, especially motor-operated valve IC-V2, which is in the common return header to the pumps, will fail RCP thermal barrier cooling. (MT, PT)

8. Failure of RCP Cooler RC-C1A, RC-C1B, RC-C1C, or RC-C1D or RCP Cooler Discharge Valve IC-V79A, IC-V79B, IC-V79C, or IC-V79D

This fails the ICCW cooling to at least one of the RCPs. (MT, PT)

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Intermediate Closed Cooling Water (ICCW) System

Table A9-2 Modified System Walkdown*

COMPONENT NUMBER	COMPONENT NAME	LOCATION	REQUIRED POSITION	ACTUAL POSITION
IC-V3	ICCW Isolation Valve	281 AB	Open	________
IC-V4	ICCW Isolation Valve	305 AB	Open	________
IC-V13A	Pump Discharge Check Valve	AB		________
IC-V13B	Pump Discharge Check Valve	AB		________
IC-C1A	ICCW Heat Exchanger	271 AB		________
IC-C1B	ICCW Heat Exchanger	271 AB		________
IC-P1A	ICCW Pump	305 AB		________
IC-P1B	ICCW Pump	305 AB		________
IC-V2	ICCW Isolation Valve	281 RB	Open	________
IC-V20	RCDT Cooler Valve	281 RB	Closed	________
RC-C1A	RCP Cooler A	RB	Valved In	________
RC-C1B	RCP Cooler B	RB	Valved In	________
RC-C1C	RCP Cooler C	RB	Valved In	________
RC-C1D	RCP Cooler D	RB	Valved In	________
IC-V79A	RCP Cooler Discharge Valve	305 RB	Open	________
IC-V79B	RCP Cooler Discharge Valve	305 RB	Open	________
IC-V79C	RCP Cooler Discharge Valve	305 RB	Open	________
IC-V79D	RCP Cooler Discharge Valve	305 RB	Open	________

* The normal operation of this system has one ICCW pump running, with the remaining pump in standby. Similarly, one heat exchanger is also typically valved in for operation.

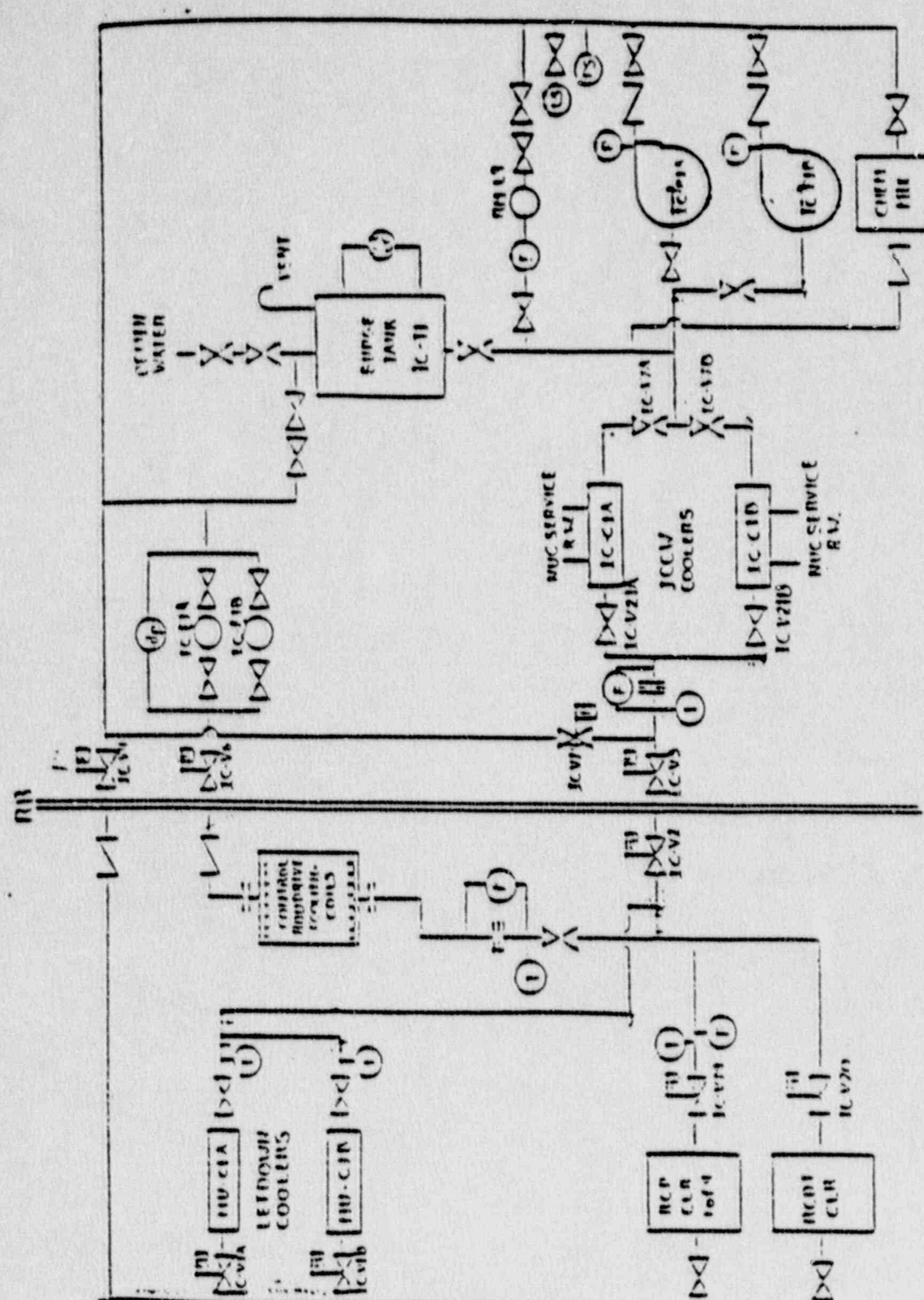

FIGURE A9.1 Simplified System Drawing of Intermediate Closed Cooling Water.

Instrument Air (IA) System

Table A10-1 Importance Basis and Failure Mode Identification

CONDITIONS THAT CAN LEAD TO FAILURE

The Instrument Air (IA) System normally supplies 100 psig air throughout TMI-1 on a demand basis. The demands for the system modelled in the TMI-1 PRA included motive power required for the air-operated valves in the HPI System seal injection path, ICCW System, Main Feedwater System, and EFW System. The TMI-1 PRA also analyzed the potential for failure of the IA System causing a plant trip (i.e., loss of IA initiating event). Although the system is not designated safety-related, without instrument air available, TMI-1 would be required to shutdown.

The IA System consists primarily of two nonlubricated air compressors, each discharging through a separate aftercooler and air receiver. In addition, an automatic, heat reactivated, air dryer in combination with a prefilter and afterfilter removes dirt particles and moisture from the air prior to its distribution to loads. Typically only one of the two compressors is operating and the other is in standby. However, on low pressure the standby IA compressor will automatically be started.

Backup to the IA System is provided by two lubricated compressors in the Service Air System. Air automatically flows from this system through an oil removal filter to the IA System if air pressure continues to drop below the set point that initiated the IA compressor that was in standby. However, the Service Air System cannot be loaded onto the diesel generator emergency AC power buses and thus is unavailable in cases of a loss of offsite power.

1. <u>Failure of the Dryer Transfer Valve IA-V1813</u>

   This failure blocks the air flow to the system loads, unless operator action is taken to bypass this valve. This failure mode is dominant as part of the IA initiating event and for cases in which all support systems are available. (MT, PT)

2. <u>Operator Fails to Bypass Dryer Transfer Valve Using Bypass Valve IA-V19</u>

   In conjunction with failure mode 1 above, this failure will result in a loss of IA to all system loads. This failure mode is dominant as part of the IA initiating event and for cases in which all support systems are available. (OP)

3. <u>Both Prefilter Trains Fail</u>

The failure of both prefilter trains fails the IA System. This failure mode is dominant as part of the IA initiating event and for cases in which all support systems are available. (MT, PT)

4. <u>Both Afterfilter Trains Fail</u>

As with failure mode 3 above, the failure of both afterfilter trains fails the IA System. This failure mode is dominant as part of the IA initiating event and for cases in which all support systems are available. (MT, PT)

5. <u>Operator Fails to Start Instrument Air Compressor IA-P1A or IA-P1B</u>

This failure mode dominates conditions in which offsite power is lost, which fails the Service Air System and requires the IA compressors to be manually loaded onto the diesel generator emergency AC buses and restarted. Operator failure to restart the IA compressors results in the failure of air to the system loads. (OP)

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Instrument Air (IA) System

Table A10-2 Modified System Walkdown*

COMPONENT NUMBER	COMPONENT NAME	LOCATION	REQUIRED POSITION	ACTUAL POSITION
IA-Q1	Air Dryer	295 IB	Valved In	________
IA-V1813	Dryer Transfer Air Valve	IB	Open	________
IA-V19	Air Dryer Bypass Valve	IB	Closed	________
IA-F2A	Prefilter	295 IB		________
IA-V15A	Prefilter Isolation Valve	IB		________
IA-V17A	Prefilter Isolation Valve	IB		________
IA-F2B	Prefilter	295 IB		________
IA-V15B	Prefilter Isolation Valve	IB		________
IA-V17B	Prefilter Isolation Valve	IB		________
IA-F3A	Afterfilter	295 IB		________
IA-23A	Afterfilter Isolation Valve	IB		________
IA-24A	Afterfilter Isolation Valve	IB		________
IA-F3B	Afterfilter	295 IB		________
IA-23B	Afterfilter Isolation Valve	IB		________
IA-V24B	Afterfilter Isolation Valve	IB		________
IA-P1A	Instrument Air Compressor	295 IB		________
IA-P1B	Instrument Air Compressor	295 IB		________

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* The normal operation of this system has one compressor running and the remaining compressor in standby. In addition, only one prefilter and one afterfilter is valved in while the other filters are in standby.

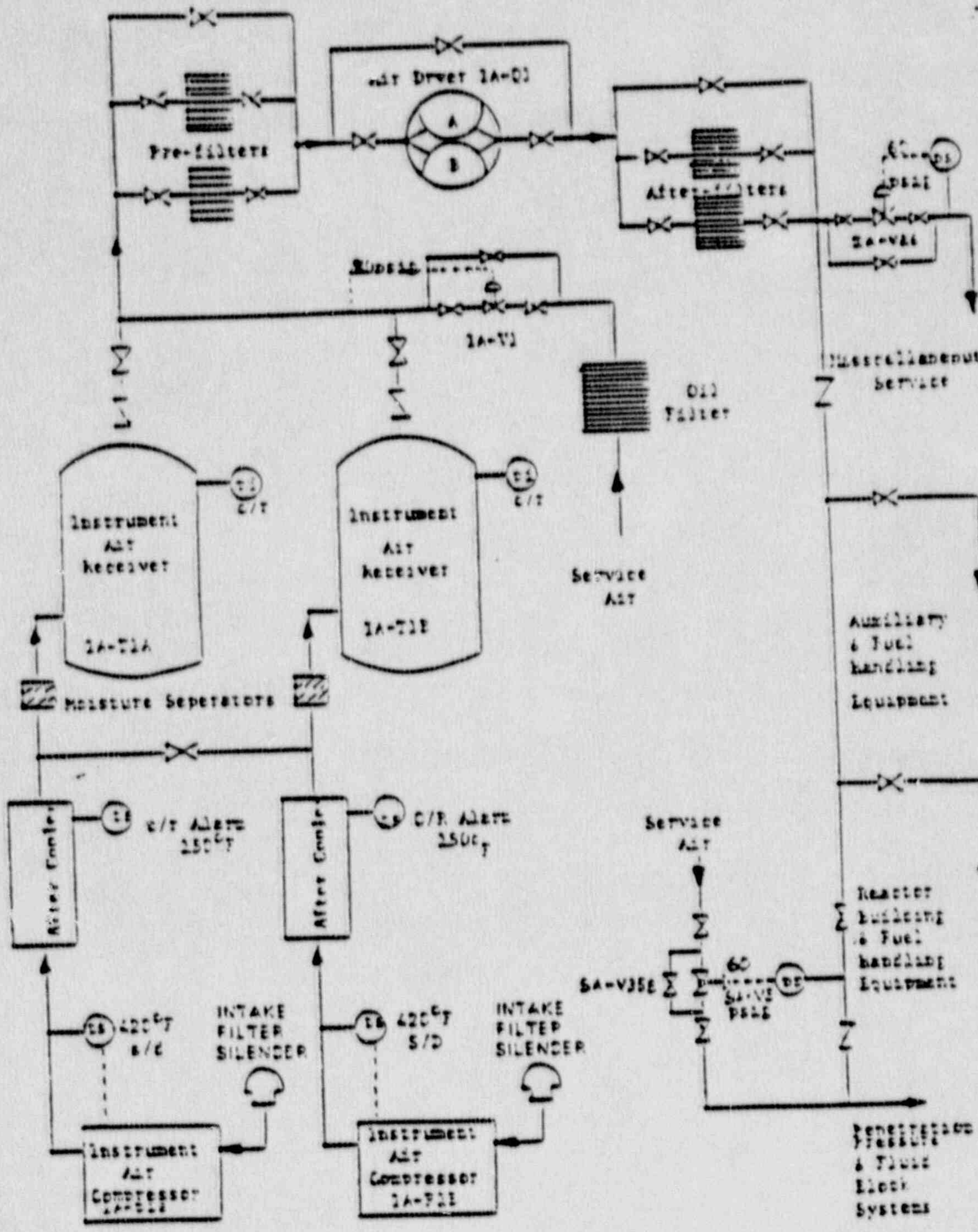

FIGURE A10.1 Simplified System Drawing of Instrument Air.

DC Power System

Table A11-1 Importance Basis and Failure Mode Identification

CONDITIONS THAT CAN LEAD TO FAILURE

The DC Power System provides continuous power for control, instrumentation, reactor protection, and engineered safeguards actuation. As part of its function, the DC Power System provides control power for the diesel generators, which are part of the emergency onsite AC power source, and power for the control valves in the EFW System. It also supplies power through inverters to the vital 120 V buses.

The DC Power System is composed of two separate trains each consisting of two 125 V DC station batteries, three battery chargers, and numerous distribution panels. Normally power is supplied through the battery chargers, with the batteries in float maintaining a full charge. Upon a loss of AC power, the entire DC load is drawn from the batteries. The use of both station batteries in a train is sufficient to power its essential loads for two hours.

This system is solely a support system and its success criteria are dependent upon the systems it supports. The failure modes described below are for failure of one train of the DC Power System. For a train of the DC Power System to succeed, at least two battery chargers or both station batteries must provide DC power to the necessary loads.

1. Failure of Battery Charger DC-BC1A, DC-BC1C, or DC-BC1E on Train 1A or Failure of Battery Charger DC-BC1B, DC-BC1D, or DC-BC1F on Train 1B

   This failure reduces the redundancy of a train of DC power, by requiring the remaining battery chargers or both station batteries to function. (MT, PT)

2. Failure of DC Batteries on Train 1A or Train 1B on Demand or During Operation

   The failure of Train 1A batteries BATT-1A or BATT-1C or Train 1B batteries BATT-1B or BATT-1D results in the loss of a train of DC power unless the power supplied through the battery chargers is provided. (MT, PT)

3. Failure of a Fuse

   This is the failure of one of fifteen fuses and results in the failure of a train of DC power to a particular distribution panel load. (MT, PT)

4. <u>Train 1A Battery or Train 1B Battery Unavailable due to Maintenance</u>

Maintenance on a battery eliminates the use of batteries as a backup to the normal supply through the battery chargers. (MT, TS)

5. <u>Failure of an Electrical DC Bus</u>

This fails the DC power to one of the distribution panel loads. (MT, PT)

6. <u>Circuit Breaker Transfers Open</u>

This failure disconnects one of the distribution panels from the DC power source, thus failing its loads. (MT, PT)

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DC Power System

Table A11-2 Modified System Walkdown*

COMPONENT NUMBER	COMPONENT NAME	LOCATION	REQUIRED POSITION	ACTUAL POSITION
BC-1A	Battery Charger	322 CB	Energized	________
BC-1B	Battery Charger	322 CB	Energized	________
BC-1C	Battery Charger	322 CB	Energized	________
BC-1D	Battery Charger	322 CB	Energized	________
BC-1E	Battery Charger	322 CB	Energized	________
BC-1F	Battery Charger	322 CB	Energized	________
BATT-1A	DC Battery	INV RM A	In Float	________
BATT-1C	DC Battery	INV RM A	In Float	________
BATT-1B	DC Battery	INV RM B	In Float	________
BATT-1D	DC Battery	INV RM B	In Float	________
Varies	Fuse	Varies	Varies	________
DC-PNL-1A	125 V DC Panel	INV RM A	Energized	________
DC-PNL-1C	125 V DC Dist. Panel	INV RM A	Energized	________
DC-PNL-1H	125 V DC Dist. Panel	INV RM A	Energized	________
DC-PNL-1E	125 V DC Dist. Panel	INV RM A	Energized	________
DC-PNL-1P	125 V DC Dist. Panel	DG CR A	Energized	________
DC-PNL-DCA	125 V DC Dist. Panel	SWITCHYARD	Energized	________
DC-PNL-1M	125 V DC Dist. Panel	322 CB	Energized	________
DC-PNL-1B	125 V DC Panel	INV RM B	Energized	________
DC-PNL-1D	125 V DC Dist. Panel	INV RM B	Energized	________
DC-PNL-1J	125 V DC Dist. Panel	INV RM B	Energized	________

DC-PNL-1F	125 V DC Dist. Panel	INV RM B	Energized	________
DC-PNL-1Q	125 V DC Dist. Panel	DG CR B	Energized	________
DC-PNL-DCB	125 V DC Dist. Panel	SWITCHYARD	Energized	________
Varies	Circuit Breakers	Varies	Varies	________

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* INV RM A and INV RM B are located on 322 elevation of the Control Building.

DG CR A and DG CR B are located on 305 elevation of the Diesel Generator Building.

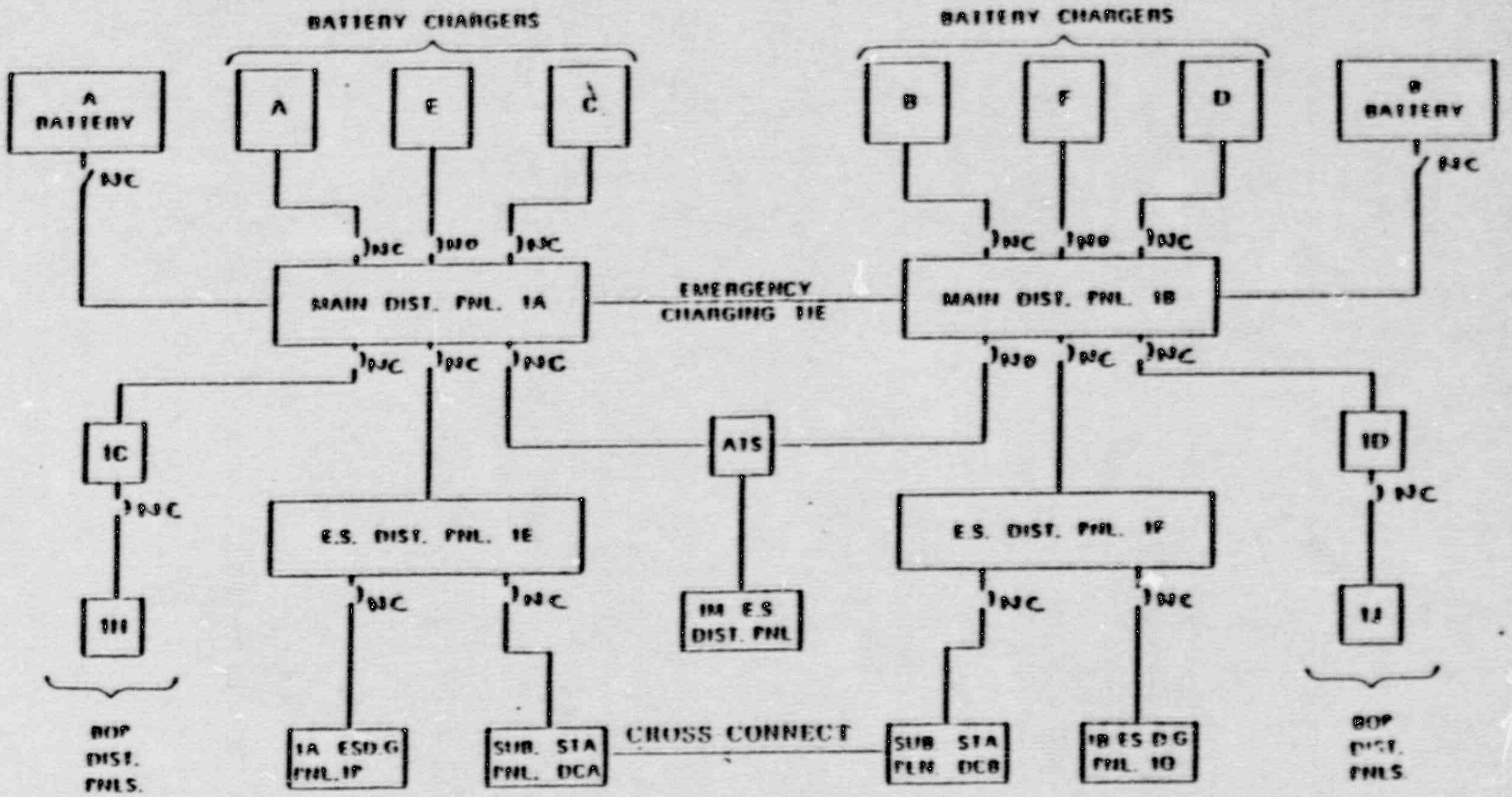

FIGURE A11.1 Simplified System Drawing of 125V DC Power.

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Engineered Safeguards Actuation System (ESAS)

Table A12-1 Importance Basis and Failure Mode Identification

CONDITIONS THAT CAN LEAD TO FAILURE

The Engineered Safeguards Actuation System (ESAS) monitors parameters to detect the loss of integrity in the RCS pressure boundary. A pressure of 1600 psig in the RCS and/or 4 psig in the reactor building will initiate the ESAS to send a signal to actuate the operation of the HPI, DHR, Reactor Building Isolation, Reactor Building Emergency Cooling, and Reactor Building Spray Systems. In addition, the signal is used to start the emergency diesel generators and to control loading sequencing.

Each actuation parameter is measured by three sensors, with the output signal of each sensor monitored by a bistable that has two output relays (one for each of two channels). Since this system supports the actuation of numerous safety systems, its success criteria are dependent upon the success criteria of the supported systems. The failure modes described below are for failure of a signal to be sent to actuate a system when required.

1. <u>Operator Fails to Manually Actuate 4 psig Reactor Building Pressure Signal</u>

   This fails the actuation signals for the situation in which automatic actuation is prevented due to the reactor building purge valves being initially open, preventing the building pressure from actually reaching 4 psig. The manual actuation circuitry bypasses the two-out-of-three logic for channel actuation. (OP)

2. <u>Multiple ESAS Sensors Miscalibrated</u>

   Miscalibration of multiple sensors results in signals not being sent when required, potentially failing the entire ESAS. (MT, PT, PC, OP)

3. <u>Common Cause Failure of Time Delay Relays to Operate on Demand</u>

   This failure results in the failure of the ESAS. (MT, PT)

4. <u>Common Cause Failure of Relays (Other Than Time Delay or Actuation) to Operate on Demand</u>

   This failure results in the failure of the ESAS. (MT, PT)

5. Hardware Failure of Sensor

Failure of a sensor may result in a reduction in the redundancy of the system by not actuating the system when required. (MT, PT)

6. Common Cause Failure of Actuation Relays to Operate on Demand

This failure may result in the failure of the ESAS. (MT, PT)

7. Common Cause Failure of Bistables to Operate on Demand

This failure may result in the failure of the ESAS. (MT, PT)

8. Failure of Actuation Relays to Operate on Demand

This failure results in a reduction of the redundancy of the system from two-out-of-three to a need for two-out-of-two channels to function for system operation. (MT, PT)

9. Failure of Relays (Other Than Time Delay or Actuation) to Operate on Demand

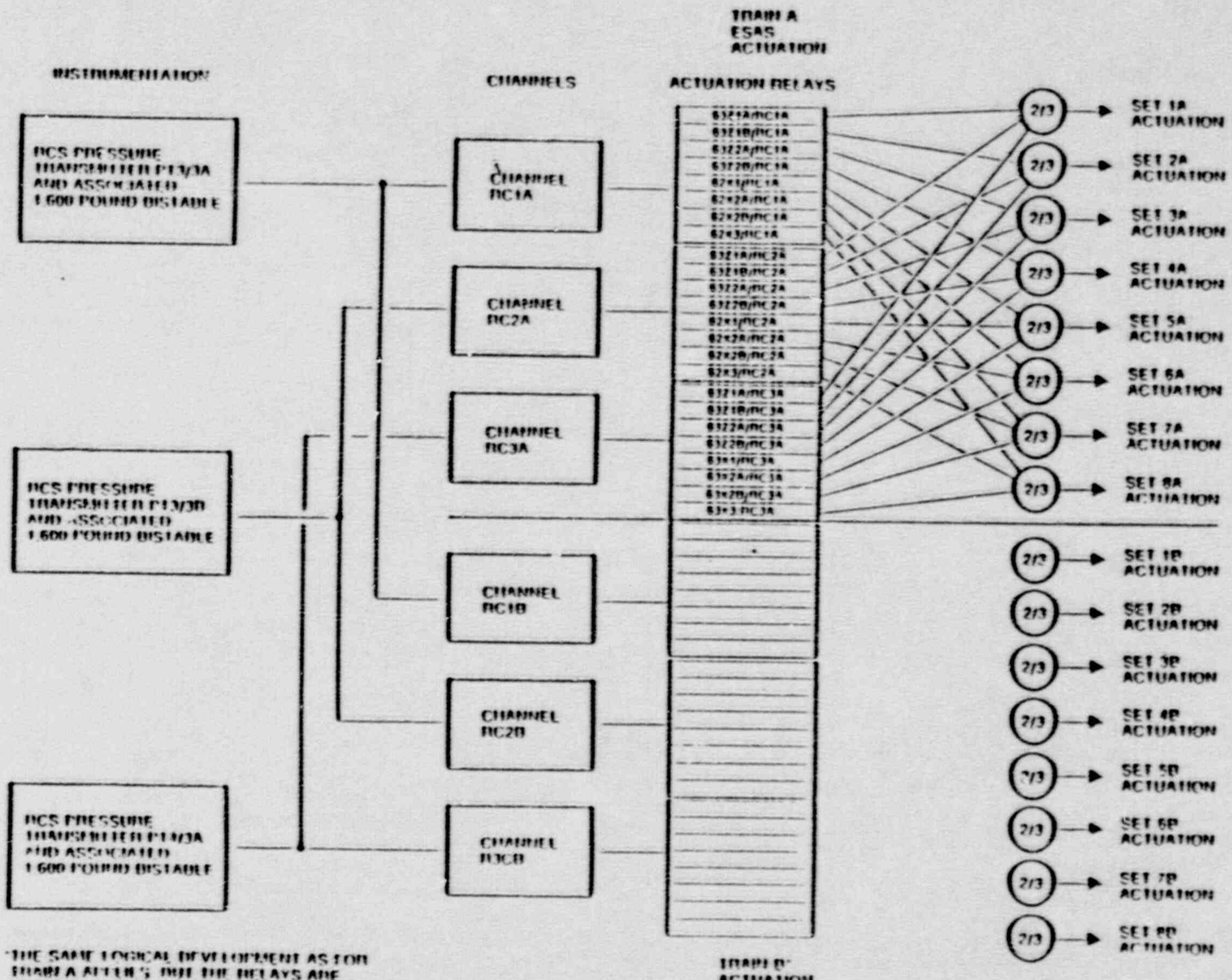
This failure results in a reduction of the redundancy of the system from two-out-of-three to a need for two-out-of-two channels to function for system operation. (MT, PT)

Engineered Safeguards Actuation System (ESAS)

Table A12-2 Modified System Walkdown*

COMPONENT NUMBER	COMPONENT NAME	LOCATION	REQUIRED POSITION	ACTUAL POSITION
Varies	Time Delay Relays	338 CB	Varies	______
Varies	Sensors	Varies	Varies	______
Varies	Bistables	Varies	Varies	______
Varies	Actuation Relays	338 CB	Varies	______
Varies	Other Relays	338 CB	Varies	______

* Walkdown is ineffective against risk significant ESAS failures.



*THE SAME LOGICAL DEVELOPMENT AS FOR TRAIN A APPLIES, BUT THE RELAYS ARE PHYSICALLY DIFFERENT AND THE REACTOR COOLANT DESIGNATION ENDS WITH A B VERSUS AN A

FIGURE A12.1 Simplified System Drawing of Engineered Safeguards Actuation - 1600 psig Reactor Coolant System Pressure Signal.

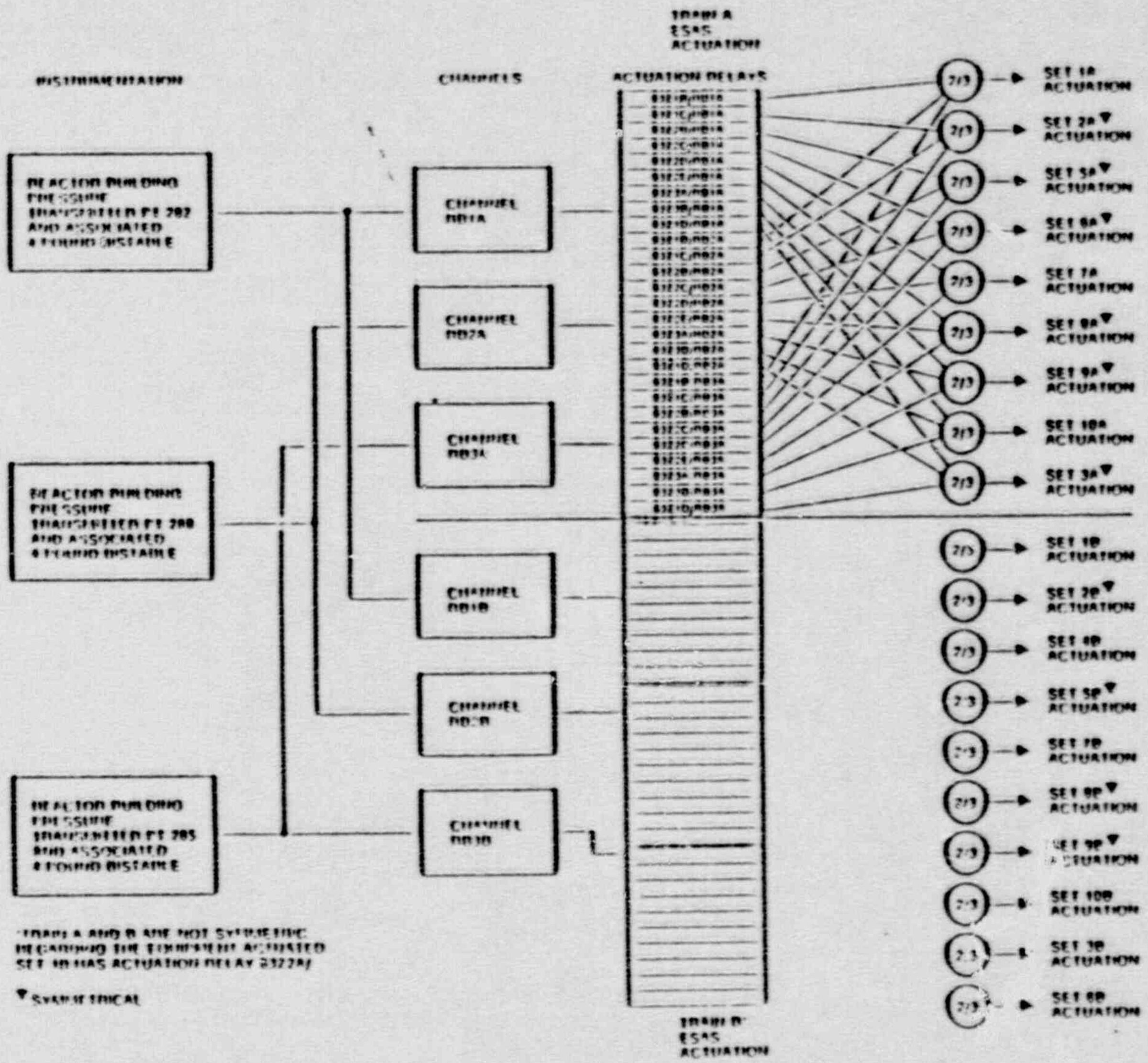


FIGURE A12.2 Simplified System Drawing of Engineered Safeguards Actuation - 4 psig Reactor Building Pressure Signal.

Reactor Building Isolation System

Table A13-1 Importance Basis and Failure Mode Identification

CONDITIONS THAT CAN LEAD TO FAILURE

The Reactor Building Isolation System is important for minimizing radionuclide releases if a core damage accident should occur. This system closes containment penetrations that are not required for operation of the engineered safeguards systems to prevent the leakage of radioactive materials to the environment from within the reactor building or RCS. Isolation signals are sent to the appropriate valves by the ESAS.

System success is defined as the closure of containment penetrations after having receiving an isolation signal from the ESAS, for 1 week after the initiating event. This long mission time was chosen in the TMI-1 PRA since the system is called upon after other systems have failed and possible core damage and the subsequent release of radioactive material to the reactor building atmosphere may have occurred.

1. Reactor Building Purge in Progress

   A frequency of 100 hours per month was estimated for reactor building purging, based on ALARA practices and the estimated number of activities, and duration of these activities, requiring reactor building entry per month. This is accomplished by opening the 48 inch purge valves AH-V1A, AH-V1B, AH-V1C, and AH-V1D. Isolation is failed when the reactor building is being purged. (MT, PT, OP)

2. Sump Draining in Progress

   A frequency of 1 hour per week was estimated for draining the reactor building sump, based on the amount of normal leakage and condensation in the reactor building. Isolation is failed while the reactor building sump is being drained and no signal from the ESAS is received. (MT, PT, OP)

Reactor Building Isolation System

Table A13-2 Modified System Walkdown

COMPONENT NUMBER	COMPONENT NAME	LOCATION	REQUIRED POSITION	ACTUAL POSITION
AH-V1A	RB Purge Isolation Valve	IB	Closed	________
AH-V1B	RB Purge Isolation Valve	RB	Closed	________
AH-V1C	RB Purge Isolation Valve	RB	Closed	________
AH-V1D	RB Purge Isolation Valve	IB	Closed	________
WDL-V534	RB Sump Drain Valve	AB	Closed	________
WDL-V535	RB Sump Drain Valve	AB	Closed	________

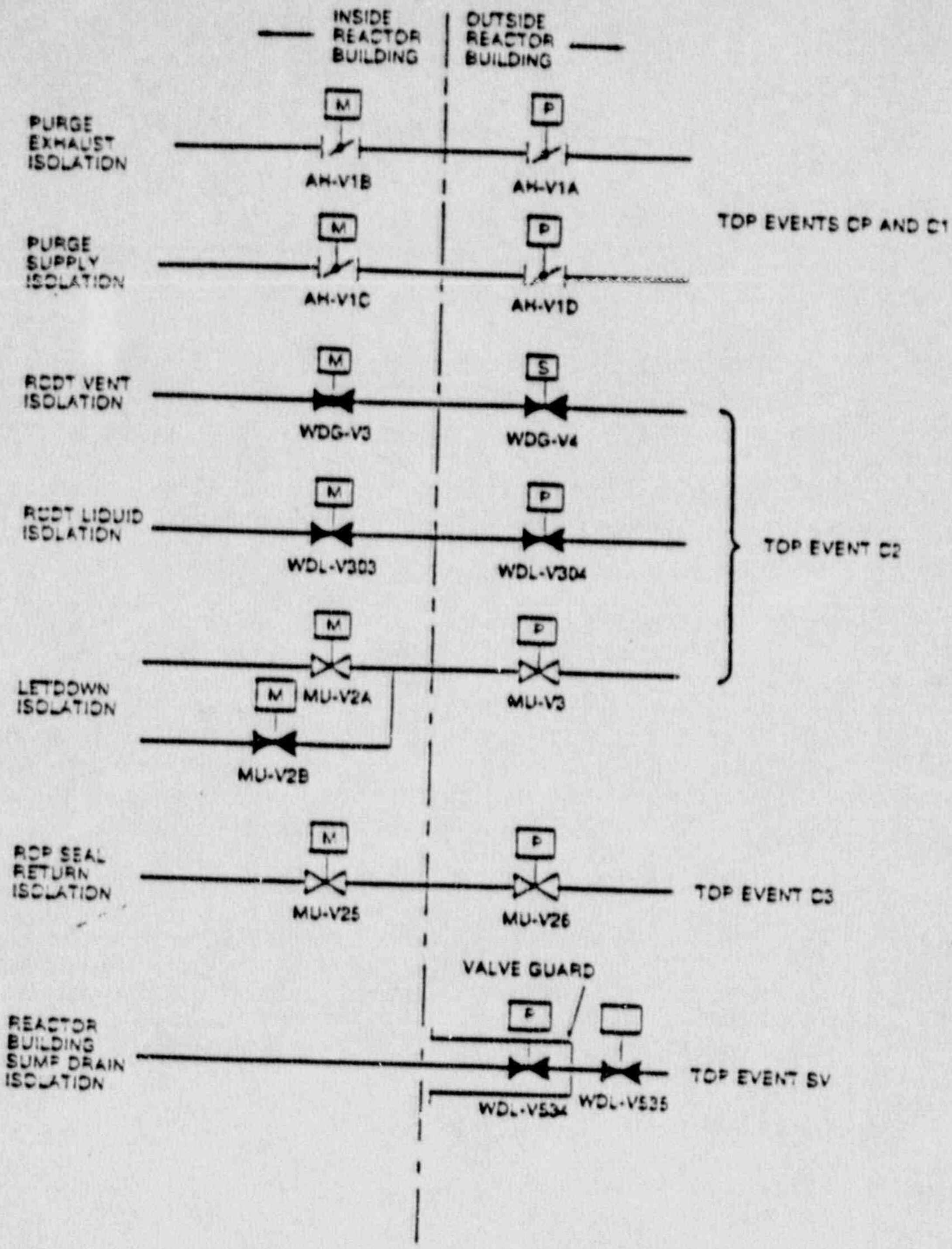


FIGURE A13.1 Simplified System Drawing of Reactor Building Isolation.

Reactor Building Spray System

Table A14-1 Importance Basis and Failure Mode Identification

CONDITIONS THAT CAN LEAD TO FAILURE

The Reactor Building Spray System is important for minimizing radionuclide releases if a core damage accident should occur. This system is designed to furnish reactor building atmosphere cooling in conjunction with the heat removal systems to limit post-accident reactor building pressure and to remove the airborne fission products from the reactor building atmosphere, thus reducing the inventory of airborne fission products available for leakage to the environment if the reactor building should not be isolated or should fail due to overstress.

The Reactor Building Spray System is separated into two trains, each consisting of a pump, a set of spray headers, and the associated piping, valves, instrumentation, and controls. Each pump initially takes suction from the BWST through an interface with the DHR System, eventually switching over to the reactor building sump when the BWST is depleted. The spray is injected into the reactor building atmosphere through a set of spray headers and nozzles in each train. System success is defined as the operation of at least one train of reactor building sprays injecting into reactor building atmosphere.

1. <u>Operator Fails to Manually Actuate Spray System</u>

   Under conditions in which automatic actuation is failed, the operator must manually actuate the system. This condition is most likely to result from situations when the containment purge line is initially open and remains open. The TMI-1 PRA assumed a high value for the operator failing to initiate the sprays since under these conditions the operator has already failed to manually actuate the ESAS. (OP)

2. <u>Valves BS-V1A and BS-V3A or BS-V1B and BS-V3B Unavailable due to Maintenance</u>

   Maintenance reduces the redundancy of the system such that any single failure in the opposite train will fail this system. (MT, TS)

3. <u>Spray Pump BS-P1A or BS-P1B Unavailable due to Maintenance</u>

   Maintenance reduces the redundancy of the system such that any single failure in the opposite train will fail this system. (MT, TS)

4. <u>Failure of Pump Motor-Operated Suction Valve BS-V3A or BS-V3B to Open</u>

   This failure fails one train of sprays. (MT, PT)

5. <u>Failure of Pump Motor-Operated Discharge Valve BS-V1A or BS-V1B to Open</u>

This failure fails one train of sprays. (MT, PT)

6. <u>Failure of Spray Pump BS-P1A or BS-P1B to Start and Run</u>

This failure fails one train of sprays. (MT, PT)

Reactor Building Spray System

Table A14-2 Modified System Walkdown*

COMPONENT NUMBER	COMPONENT NAME	LOCATION	REQUIRED POSITION	ACTUAL POSITION
BS-P1A	Spray Pump	261 AB	Standby	________
BS-P1B	Spray Pump	261 AB	Standby	________
BS-V3A	Pump Suction Valve	DH VT A	Closed	________
BS-V3B	Pump Suction Valve	DH VT B	Closed	________
BS-V1A	Pump Discharge Valve	AB	Closed	________
BS-V1B	Pump Discharge Valve	AB	Closed	________

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* DH VT A and DH VT B are located on 261 elevation of the Auxiliary Building.

Valves BS-V1A and BS-V1B are located in the Auxiliary Building basement on the mezzanine above their respective vault area.

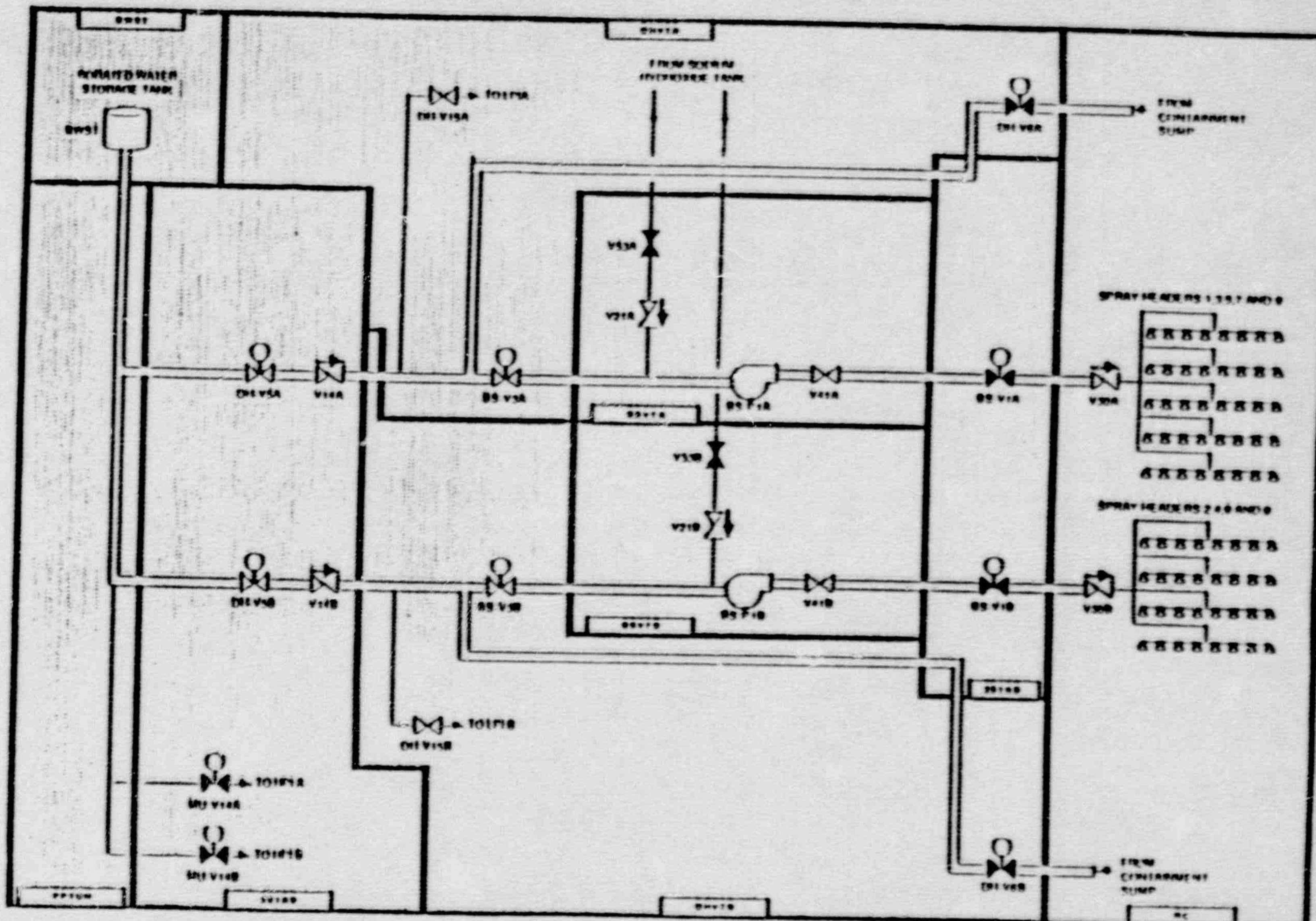

FIGURE A14.1 Simplified System Drawing of Reactor Building Spray.

THREE MILE ISLAND NUCLEAR STATION, UNIT 1
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Reactor Building Emergency Cooling System

Table A15-1 Importance Basis and Failure Mode Identification

CONDITIONS THAT CAN LEAD TO FAILURE

The Reactor Building Emergency Cooling System is important for minimizing radionuclide releases if a core damage accident should occur. This system removes heat from the reactor building atmosphere to limit the stress on the structure. Reactor building air recirculation and cooling units work in conjunction with the Reactor Building Spray System to remove the decay heat released to the reactor building atmosphere during the post-accident period and thereby to maintain the reactor building pressure below its design limit of 55 psig.

The system is comprised of two trains of emergency cooling water pumps supplying three trains of air handling units, with each unit consisting of an emergency cooling coil, a normal cooling coil, and a two-speed fan. Under normal conditions, typically two cooling units are operating, with the normal cooling coils receiving cooling water from the Industrial Cooler System. For emergency cooling, the Industrial Cooler System and normal cooling coils are isolated, the emergency cooling coils are aligned, both emergency cooling water pumps are started, and all three cooling units are started, operating at a reduced speed, with the heat rejected to river water. The back-pressure regulating valve on the emergency cooling coil discharge line maintains emergency system pressure above maximum reactor building design pressure and prevents leakage out of the building through a damaged system.

Upon receipt of an isolation signal from the ESAS, the Reactor Building Emergency Cooling System is automatically switched to the emergency mode by energizing all three recirculating air handling units, starting the emergency cooling pumps, opening the emergency cooling coil isolation valve on the outlet side of the coil, and closing the normal cooling coil isolation valve. Success of this system is defined as the operation of at least one emergency cooling water pump and at least two emergency cooling units throughout the mission time.

1. <u>Pressure Regulating Valve RR-V6 Fails to Open</u>

   Failure of this valve to open automatically results in the failure of the cooling flow from the cooling units to the mechanical draft cooling towers. This valve is located in a common header, thus failing the flow from both emergency cooling water pumps through all three cooling units. This valve is only tested during refueling and may have failed in its standby position (i.e., closed) and be undetectable until the system is required to function. (MT, PT)

2. Operator Fails to Open Bypass Valve RR-V5

In connection with failure mode 1 above, the operator can bypass the failed pressure regulating valve RR-V6 by remotely opening valve RR-V5 from the control room. However, the TMI-1 PRA assumed the operator would not be able to respond in time to protect the emergency cooling water pumps. Additional instrumentation was expected to be incorporated into the system during following refueling outages, which may aid the operator in bypassing the failed valve and thus reduce the significance of this failure mode and failure mode 1 above. (OP)

3. Cooling Unit AH-E1A, AH-E1B, or AH-E1C Unavailable due to Maintenance

Maintenance on a cooling unit reduces the redundancy of the system from two-out-of-three to two-out-of two. Any single failure in the remaining operating trains will fail this system. (MT, TS)

4. Emergency Cooling Water Pump Train RR-P1A or RR-P1B Unavailable due to Maintenance

Maintenance on a pump train will reduce the redundancy of the system such that any single failure in the other pump train will fail the system. (MT, TS)

5. Cooling Units AH-E1A, AH-E1B, and/or AH-E1C Fail

Failure of any two cooling units will result in the failure of this system. A cooling unit can be failed either by the failure of a cooling coil inlet or outlet valve to open or by the failure of the fan to start and run. (MT, PT)

6. Common Cause Failure of Pump Motor-Operated Discharge Valves RR-V1A and RR-V1B

Failure of the emergency cooling water pump motor-operated discharge valves to open fails both trains of cooling water and thus fails the system. (MT, PT)

7. Common Cause Failure of Emergency Cooling Water Pumps RR-P1A and RR-P1B

Failure of both emergency cooling water pumps fail the system. (MT, PT)

8. Common Cause Failure of Cooling Units AH-F1A, AH-F1B, and AH-F1C

Failure of any two cooling units results in the failure of the system. (MT, PT)

9. Emergency Cooling Water Pump Train RR-P1A or RR-P1B Fails to Start or Run

This failure results in only one train of cooling water being available to the cooling units. Any single failure in the remaining pump train will result in the failure of the system. (MT, PT)

10. <u>Common Cause Failure of Fan Cooling Unit Motor-Operated Discharge Valves RR-V4A, RR-V4B, RR-V4C, and RR-V4D</u>

Failure of the valves failing any two trains of cooling units will fail the system. (MT, PT)

Reactor Building Emergency Cooling System

Table A15-2 Modified System Walkdown*

COMPONENT NUMBER	COMPONENT NAME	LOCATION	REQUIRED POSITION	ACTUAL POSITION
RR-V6	Pressure Regulating Valve	295 IB	Auto(Closed)	______
RR-V5	Bypass Valve	295 IB	Closed	______
AH-E1A	Cooling Unit	281 RB		______
AH-E1B	Cooling Unit	281 RB		______
AH-E1C	Cooling Unit	281 RB		______
RR-P1A	Emergency Cooling Water Pump	SCREEN HS	Standby	______
RR-P1B	Emergency Cooling Water Pump	SCREEN HS	Standby	______
RR-V1A	Pump Discharge Valve	SCREEN HS	Closed	______
RR-V1B	Pump Discharge Valve	SCREEN HS	Closed	______
RR-V4A	Cooling Unit Discharge Valve	295 IB	Closed	______
RR-V4B	Cooling Unit Discharge Valve	295 IB	Closed	______
RR-V4C	Cooling Unit Discharge Valve	295 IB	Closed	______
RR-V4D	Cooling Unit Discharge Valve	295 IB	Closed	______

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* The normal operation of this system has two cooling units operating and the remaining cooling unit in standby. The normal cooling coils receive cooling water from the Industrial Cooler System.

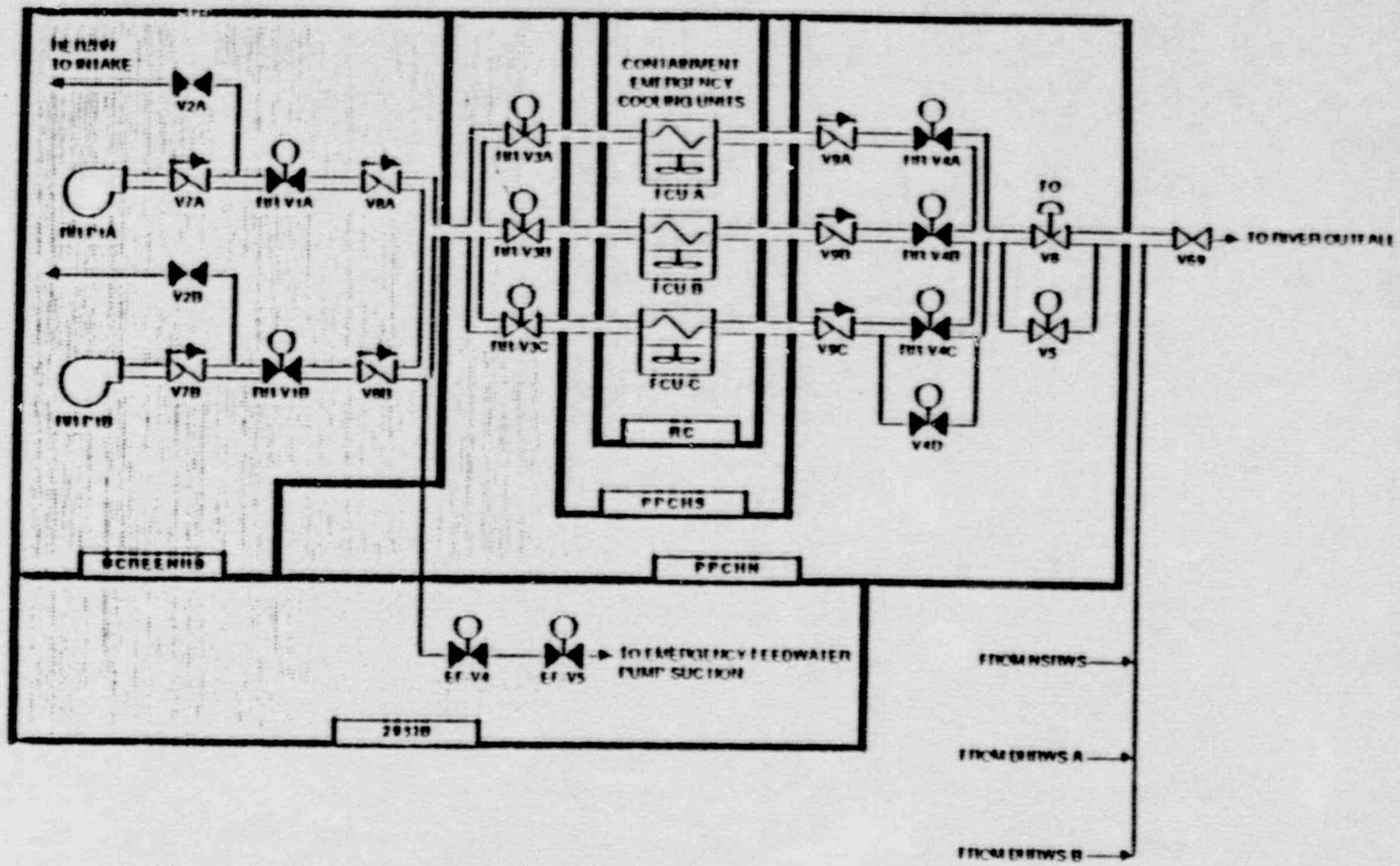

FIGURE A15.1 Simplified System Drawing of Reactor Building Emergency Cooling.

APPENDIX B TABLES OF PLANT OPERATIONS, SURVEILLANCE AND CALIBRATION, AND MAINTENANCE INSPECTION GUIDANCE

This appendix provides tables based on sorting information from the Appendix A failure mode tables prepared for all systems determined to be important in the TMI-1 PRA. Each table is discussed below.

B.1 TABLE B1 - PLANT OPERATIONS INSPECTION GUIDANCE

This table is a collection of all of the significant operator actions listed in the Appendix A failure mode tables. It is provided as a cross reference for use in observing operator actions and training.

B.2 TABLE B2 - SURVEILLANCE AND CALIBRATION INSPECTION GUIDANCE

This table is a collection of all of the significant components listed in the Appendix A failure mode tables that are considered to be influenced by surveillance and calibration activities. It is provided as a cross reference to assist in selecting activities for observation during inspections of the licensee's surveillance and calibration programs.

B.3 TABLE B3 - MAINTENANCE INSPECTION GUIDANCE

This table is a collection of the significant components listed in the Appendix A failure mode tables that are considered to be influenced by maintenance activities. It is provided as a cross reference to assist the inspector in selecting activities for observation during inspections of the licensee's maintenance program. Important factors include the frequency and duration of maintenance as well as errors that degrade the component or render it inoperable when it is return to service.

TABLE B1 - PLANT OPERATIONS INSPECTION GUIDANCE

Recognizing that normal system line-up is important for any given standby safety system, the following human errors are identified in the TMI-1 PRA as important.

SYSTEM	ACTION FAILED	DISCUSSION
Decay Heat Removal (DHR)	Switchover to Recirculation	Table A1-1, Item 1
	Return Alignment to Normal After Test or Maintenance	Table A1-1, Item 14
	Open Dropline Isolation Valves	Table A1-1, Item 15
High Pressure Injection (HPI)	Establish Minimum-Flow Recirculation	Table A2-1, Item 1
	Throttle HPI Flow	Table A2-1, Item 2
	Reopen the RCP Seal Injection Valve MU-V20	Table A2-1, Item 3
	Start HPI Pumps for HPI Cooling	Table A2-1, Item 4
	Maintain HPI Flow (Operator Incorrectly or Inadvertently Throttles HPI Flow)	Table A2-1, Item 11
	Cross-Connect HPI Pump MU-P1C for Injection	Table A2-1, Item 15
AC Power	Recover one train of AC power	Table A4-1, Item 2
Nuclear Services Cooling Water (NSCW)	Clear the river water screen	Table A5-1, Item 1
	Recover the river water system	Table A5-1, Item 2
	Isolate Leaking Heat Exchanger	Table A5-1, Item 5
Emergency Feedwater (EFW)	Locally Control EFW Flow Through Control Valves	Table A7-1, Item 1
	Replenish the 2-Hour Air Bottles	Table A7-1, Item 2
Reactor Coolant System Pressure Control	Initiate Cooldown	Table A8-1, Item 3

Instrument Air (IA)	Bypass Dryer Transfer Valve	Table A10-1, Item 2
	Start Instrument Air Compressor	Table A10-1, Item 5
Engineered Safeguards Actuation (ESA)	Manually Actuate 4 psig Reactor Building Pressure Signal	Table A12-1, Item 1
	ESAS Sensors Calibration	Table A12-1, Item 2
Reactor Building Isolation	Reactor Building Purge in Progress	Table A13-1, Item 1
	Sump Draining in Progress	Table A13-1, Item 2
Reactor Building Spray	Manually Actuate Spray System	Table A14-1, Item 1
Reactor Building Emergency Cooling	Open Bypass Valve RR-V5	Table A15-1, Item 2

TABLE B2 - SURVEILLANCE AND CALIBRATION INSPECTION GUIDANCE

The listed components are the significant components for which surveillance and/or calibration should minimize failure.

SYSTEM	FAILURE MODE	DISCUSSION
Decay Heat Removal (DHR)	Common Cause Failure of the DHR Pumps	Table A1-1, Item 2
	Piggy-Back Strainer in Maintenance	Table A1-1, Item 3
	DHR Pump Train Fails	Table A1-1, Item 4
	DHR Pump in Maintenance	Table A1-1, Item 5
	DHR Sump Isolation Valve Fails to Open	Table A1-1, Item 6
	DHR Pump Unavailable due to Operability Testing	Table A1-1, Item 7
	Common Cause Failure of Cross-Tie Valves To Open	Table A1-1, Item 8
	Common Cause Failure of DHR Sump Isolation Valves to Open	Table A1-1, Item 9
	DHR Cross-Tie Valve Fails to Open	Table A1-1, Item 10
	DHR Injection Valves in Maintenance	Table A1-1, Item 11
	DHR Cooler in Maintenance	Table A1-1, Item 12
	DHR Cooler Fails	Table A1-1, Item 13
	Dropline Isolation Valve in Maintenance	Table A1-1, Item 16
	Common Cause Failure of DHR Injection Valves to Open	Table A1-1, Item 17

	Common Cause Failure of DHR Heat Exchangers	Table A1-1, Item 18
	Reactor Building Sump Clogs	Table A1-1, Item 19
	Motor-Operated Dropline Valves DH-V1, DH-V2, or DH-V3 Fail to Open	Table A1-1, Item 20
	DHR Recirculation Train Isolation Valves in Maintenance	Table A1-1, Item 21
High Pressure Injection (HPI)	Makeup Flow Path Valves Fail to Throttle	Table A2-1, Item 5
	HPI Pump in Maintenance	Table A2-1, Item 6
	HPI Pump Fails to Start and Run	Table A2-1, Item 7
	Common Cause Failure of Injection Flow Path Valves to Open	Table A2-1, Item 8
	Common Cause Failure of the BWST Isolation Valves to Open	Table A2-1, Item 9
	Failure of Both BWST Isolation Valves to Open	Table A2-1, Item 10
	Minimum-Flow Recirculation Valve Fails to Open	Table A2-1, Item 12
	Failure of 1 of 7 Check Valves to Open	Table A2-1, Item 13
	Common Cause Failure of the HPI Pumps	Table A2-1, Item 14
	Failure of Valves MU-V20, MU-V76A, or MU-V76B to Open	Table A2-1, Item 16
Decay Heat Cooling Water (DHCW)	DHRW Pump in Maintenance	Table A3-1, Item 1
	Failure of DHRW Pump to Start and Run	Table A3-1, Item 2
	Failure of DHRW Pump Discharge Valve to Open	Table A3-1, Item 3
	Failure of DHCCW Pump to Start and Run	Table A3-1, Item 4
	DHCCW Pump in Maintenance	Table A3-1, Item 5

	DHRW Strainer in Maintenance	Table A3-1, Item 6
	Decay Heat Service Cooler in Maintenance	Table A3-1, Item 7
	Common Cause Failure of the DHCCW Pumps	Table A3-1, Item 8
	Common Cause Failure of the DHRW Pump Discharge Valves to Open	Table A3-1, Item 9
	Common Cause Failure of the DHRW Pumps	Table A3-1, Item 10
AC Power	Failure of the Offsite Grid	Table A4-1, Item 1
	Diesel Generator in Maintenance	Table A4-1, Item 3
	Failure of Diesel Generator to Start and Run	Table A4-1, Item 4
	Circuit Breaker Transfers Open	Table A4-1, Item 5
	Failure of an Electrical Bus	Table A4-1, Item 6
	Failure of a Transformer	Table A4-1, Item 7
	Common Cause Failure of the Diesel Generators to Start and Run	Table A4-1, Item 8
Nuclear Services Cooling Water (NSCW)	Failure of a NSRW Pump	Table A5-1, Item 3
	Failure of a NSCCW Pump	Table A5-1, Item 3
	Rupture or Leakage of NSCCW Heat Exchanger (Cooler)	Table A5-1, Item 4
	Rupture or Leakage of Nuclear Services Heat Exchanger (Cooler)	Table A5-1, Item 4
	Failure of NSRW Common Header Motor-Operated Isolation Valve to Remain Open	Table A5-1, Item 6
	Common Cause Failure of All Three NSRW Pumps	Table A5-1, Item 7
	Common Cause Failure of All Three NSCCW Pumps	Table A5-1, Item 7

	Failure of NSCCW Pipe Less Than 3 Inches in Diameter	Table A5-1, Item 8
	NSRW Pump in Maintenance	Table A5-1, Item 9
	NSCCW Pump in Maintenance	Table A5-1, Item 9
	Failure of the NSCCW Surge Tank	Table A5-1, Item 10
	Common Cause Failure of the NSCCW Heat Exchangers (Coolers)	Table A5-1, Item 11
Main Steam	Failure of the Main Steam Safety Valves to Reseat	Table A6-1, Item 1
	Failure of the Turbine Bypass Valves and Atmospheric Dump Valves to Close	Table A6-1, Item 2
	Failure of the Atmospheric Dump Valves to Control RCS Pressure	Table A6-1, Item 3
	Failure of the Emergency Feedwater Pump Turbine Valve to Close	Table A6-1, Item 4
	Intermittent Operation Failure of the ADVs and TBVs	Table A6-1, Item 5
Emergency Feedwater (EFW)	Failure of the Turbine-Driven Pump to Start and Run	Table A7-1, Item 3
	Failure of the 2-Hour Train of Air Bottles	Table A7-1, Item 4
	Failure of a Motor-Driven Pump to Start and Run	Table A7-1, Item 5
	Motor-Driven Pump in Maintenance	Table A7-1, Item 6
	Control Valves and Block Valves Unavailable due to Testing	Table A7-1, Item 7
Reactor Coolant System Pressure Control	Failure of Pressurizer Safety Valves to Reseat	Table A8-1, Item 1
	PORV Train in Maintenance	Table A8-1, Item 2
Intermediate Closed Cooling Water (ICCW)	Failure of ICCW Air-Operated Valves to Remain Open	Table A9-1, Item 1

	Failure due to Reverse Leakage of Pump Discharge Check Valves	Table A9-1, Item 2
	Failure of the ICCW Heat Exchangers	Table A9-1, Item 3
	ICCW Pump in Maintenance	Table A9-1, Item 4
	Failure of an ICCW Pump to Start and Run	Table A9-1, Item 5
	Failure of Pump Discharge Check Valves to Open	Table A9-1, Item 6
	Failure of Motor-Operated Valves to Open	Table A9-1, Item 7
	Failure of RCP Cooler and Valving	Table A9-1, Item 8
Instrument Air (IA)	Failure of the Dryer Transfer Valve	Table A10-1, Item 1
	Both Prefilter Trains Fail	Table A10-1, Item 3
	Both Afterfilter Trains Fail	Table A10-1, Item 4
DC Power	Failure of a Battery Charger	Table A11-1, Item 1
	Failure of DC Batteries on Demand or During Operation	Table A11-1, Item 2
	Failure of a Fuse	Table A11-1, Item 3
	Battery in Maintenance	Table A11-1, Item 4
	Failure of an Electrical DC Bus	Table A11-1, Item 5
	Circuit Breaker Transfers Open	Table A11-1, Item 6
Engineered Safeguards Actuation (ESA)	ESAS Sensors Miscalibrated	Table A12-1, Item 2
	Common Cause Failure of Time Delay Relays to Operate on Demand	Table A12-1, Item 3
	Common Cause Failure of Relays (Other Than Time Delay or Actuation) to Operate on Demand	Table A12-1, Item 4
	Hardware Failure of Sensor	Table A12-1, Item 5

	Common Cause Failure of Actuation Relays to Operate on Demand	Table A12-1, Item 6
	Common Cause Failure of Bistables to Operate on Demand	Table A12-1, Item 7
	Failure of Actuation Relays to Operate on Demand	Table A12-1, Item 8
	Failure of Relays (Other Than Time Delay or Actuation) to Operate on Demand	Table A12-1, Item 9
Reactor Building Isolation	Reactor Building Purge in Progress	Table A13-1, Item 1
	Sump Draining in Progress	Table A13-1, Item 2
Reactor Building Spray	Spray Valves in Maintenance	Table A14-1, Item 2
	Spray Pump in Maintenance	Table A14-1, Item 3
	Failure of Pump Train Motor-Operated Suction Valve to Open	Table A14-1, Item 4
	Failure of Pump Train Motor-Operated Discharge Valve to Open	Table A14-1, Item 5
	Failure of Spray Pump to Start and Run	Table A14-1, Item 6
Reactor Building Emergency Cooling	Pressure Regulating Valve RR-V6 Fails to Open	Table A15-1, Item 1
	Cooling Unit in Maintenance	Table A15-1, Item 3
	Emergency Cooling Water Pump Train in Maintenance	Table A15-1, Item 4
	Cooling Units Fail	Table A15-1, Item 5
	Common Cause Failure of Pump Motor-Operated Discharge Valves	Table A15-1, Item 6
	Common Cause Failure of Emergency Cooling Water Pumps	Table A15-1, Item 7
	Common Cause Failure of Cooling Units	Table A15-1, Item 8
	Pump Train Fails to Start or Run	Table A15-1, Item 9

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TABLE B3 - MAINTENANCE INSPECTION GUIDANCE

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

SYSTEM	FAILURE MODE	DISCUSSION
Decay Heat Removal (DHR)	Common Cause Failure of the DHR Pumps	Table A1-1, Item 2
	Piggy-Back Strainer in Maintenance	Table A1-1, Item 3
	DHR Pump Train Fails	Table A1-1, Item 4
	DHR Pump in Maintenance	Table A1-1, Item 5
	DHR Sump Isolation Valve Fails to Open	Table A1-1, Item 6
	DHR Pump Unavailable due to Operability Testing	Table A1-1, Item 7
	Common Cause Failure of Cross-Tie Valves To Open	Table A1-1, Item 8
	Common Cause Failure of DHR Sump Isolation Valves to Open	Table A1-1, Item 9
	DHR Cross-Tie Valve Fails to Open	Table A1-1, Item 10
	DHR Injection Valves in Maintenance	Table A1-1, Item 11
	DHR Cooler in Maintenance	Table A1-1, Item 12
	DHR Cooler Fails	Table A1-1, Item 13
	Dropline Isolation Valve in Maintenance	Table A1-1, Item 16
	Common Cause Failure of DHR Injection Valves to Open	Table A1-1, Item 17

	Common Cause Failure of DHR Heat Exchangers	Table A1-1, Item 18
	Reactor Building Sump Clogs	Table A1-1, Item 19
	Motor-Operated Dropline Valves DH-V1, DH-V2, or DH-V3 Fail to Open	Table A1-1, Item 20
	DHR Recirculation Train Isolation Valves in Maintenance	Table A1-1, Item 21
High Pressure Injection (HPI)	Makeup Flow Path Valves Fail to Throttle	Table A2-1, Item 5
	HPI Pump in Maintenance	Table A2-1, Item 6
	HPI Pump Fails to Start and Run	Table A2-1, Item 7
	Common Cause Failure of Injection Flow Path Valves to Open	Table A2-1, Item 8
	Common Cause Failure of the BWST Isolation Valves to Open	Table A2-1, Item 9
	Failure of Both BWST Isolation Valves to Open	Table A2-1, Item 10
	Minimum-Flow Recirculation Valve Fails to Open	Table A2-1, Item 12
	Failure of 1 of 7 Check Valves to Open	Table A2-1, Item 13
	Common Cause Failure of the HPI Pumps	Table A2-1, Item 14
	Failure of Valves MU-V20, MU-V76A, or MU-V76B to Open	Table A2-1, Item 16
Decay Heat Cooling Water (DHCW)	DHRW Pump in Maintenance	Table A3-1, Item 1
	Failure of DHRW Pump to Start and Run	Table A3-1, Item 2
	Failure of DHRW Pump Discharge Valve to Open	Table A3-1, Item 3
	Failure of DHCCW Pump to Start and Run	Table A3-1, Item 4
	DHCCW Pump in Maintenance	Table A3-1, Item 5

	DHRW Strainer in Maintenance	Table A3-1, Item 6
	Decay Heat Service Cooler in Maintenance	Table A3-1, Item 7
	Common Cause Failure of the DHCCW Pumps	Table A3-1, Item 8
	Common Cause Failure of the DHRW Pump Discharge Valves to Open	Table A3-1, Item 9
	Common Cause Failure of the DHRW Pumps	Table A3-1, Item 10
AC Power	Failure of the Offsite Grid	Table A4-1, Item 1
	Diesel Generator in Maintenance	Table A4-1, Item 3
	Failure of Diesel Generator to Start and Run	Table A4-1, Item 4
	Circuit Breaker Transfers Open	Table A4-1, Item 5
	Failure of an Electrical Bus	Table A4-1, Item 6
	Failure of a Transformer	Table A4-1, Item 7
	Common Cause Failure of the Diesel Generators to Start and Run	Table A4-1, Item 8
Nuclear Services Cooling Water (NSCW)	Failure of a NSRW Pump	Table A5-1, Item 3
	Failure of a NSCCW Pump	Table A5-1, Item 3
	Rupture or Leakage of NSCCW Heat Exchanger (Cooler)	Table A5-1, Item 4
	Rupture or Leakage of Nuclear Services Heat Exchanger (Cooler)	Table A5-1, Item 4
	Failure of NSRW Common Header Motor-Operated Isolation Valve to Remain Open	Table A5-1, Item 6
	Common Cause Failure of All Three NSRW Pumps	Table A5-1, Item 7
	Common Cause Failure of All Three NSCCW Pumps	Table A5-1, Item 7

	Failure of NSCCW Pipe Less Than 3 Inches in Diameter	Table A5-1, Item 8
	NSRW Pump in Maintenance	Table A5-1, Item 9
	NSCCW Pump in Maintenance	Table A5-1, Item 9
	Failure of the NSCCW Surge Tank	Table A5-1, Item 10
	Common Cause Failure of the NSCCW Heat Exchangers (Coolers)	Table A5-1, Item 11
Main Steam	Failure of the Main Steam Safety Valves to Reseat	Table A6-1, Item 1
	Failure of the Turbine Bypass Valves and Atmospheric Dump Valves to Close	Table A6-1, Item 2
	Failure of the Atmospheric Dump Valves to Control RCS Pressure	Table A6-1, Item 3
	Failure of the Emergency Feedwater Pump Turbine Valve to Close	Table A6-1, Item 4
	Intermittent Operation Failure of the ADVs and TBVs	Table A6-1, Item 5
Emergency Feedwater (EFW)	Failure of the Turbine-Driven Pump to Start and Run	Table A7-1, Item 3
	Failure of the 2-Hour Train of Air Bottles	Table A7-1, Item 4
	Failure of a Motor-Driven Pump to Start and Run	Table A7-1, Item 5
	Motor-Driven Pump in Maintenance	Table A7-1, Item 6
	Control Valves and Block Valves Unavailable due to Testing	Table A7-1, Item 7
Reactor Coolant System Pressure Control	Failure of Pressurizer Safety Valves to Reseat	Table A8-1, Item 1
	PORV Train in Maintenance	Table A8-1, Item 2
Intermediate Closed Cooling Water (ICCW)	Failure of ICCW Air-Operated Valves to Remain Open	Table A9-1, Item 1

	Failure due to Reverse Leakage of Pump Discharge Check Valves	Table A9-1, Item 2
	Failure of the ICCW Heat Exchangers	Table A9-1, Item 3
	ICCW Pump in Maintenance	Table A9-1, Item 4
	Failure of an ICCW Pump to Start and Run	Table A9-1, Item 5
	Failure of Pump Discharge Check Valves to Open	Table A9-1, Item 6
	Failure of Motor-Operated Valves to Open	Table A9-1, Item 7
	Failure of RCP Cooler and Valving	Table A9-1, Item 8
Instrument Air (IA)	Failure of the Dryer Transfer Valve	Table A10-1, Item 1
	Both Prefilter Trains Fail	Table A10-1, Item 3
	Both Afterfilter Trains Fail	Table A10-1, Item 4
DC Power	Failure of a Battery Charger	Table A11-1, Item 1
	Failure of DC Batteries on Demand or During Operation	Table A11-1, Item 2
	Failure of a Fuse	Table A11-1, Item 3
	Battery in Maintenance	Table A11-1, Item 4
	Failure of an Electrical DC Bus	Table A11-1, Item 5
	Circuit Breaker Transfers Open	Table A11-1, Item 6
Engineered Safeguards Actuation (ESA)	ESAS Sensors Miscalibrated	Table A12-1, Item 2
	Common Cause Failure of Time Delay Relays to Operate on Demand	Table A12-1, Item 3
	Common Cause Failure of Relays (Other Than Time Delay or Actuation) to Operate on Demand	Table A12-1, Item 4
	Hardware Failure of Sensor	Table A12-1, Item 5

	Common Cause Failure of Actuation Relays to Operate on Demand	Table A12-1, Item 6
	Common Cause Failure of Bistables to Operate on Demand	Table A12-1, Item 7
	Failure of Actuation Relays to Operate on Demand	Table A12-1, Item 8
	Failure of Relays (Other Than Time Delay or Actuation) to Operate on Demand	Table A12-1, Item 9
Reactor Building Isolation	Reactor Building Purge in Progress	Table A13-1, Item 1
	Sump Draining in Progress	Table A13-1, Item 2
Reactor Building Spray	Spray Valves in Maintenance	Table A14-1, Item 2
	Spray Pump in Maintenance	Table A14-1, Item 3
	Failure of Pump Train Motor-Operated Suction Valve to Open	Table A14-1, Item 4
	Failure of Pump Train Motor-Operated Discharge Valve to Open	Table A14-1, Item 5
	Failure of Spray Pump to Start and Run	Table A14-1, Item 6
Reactor Building Emergency Cooling	Pressure Regulating Valve RR-V6 Fails to Open	Table A15-1, Item 1
	Cooling Unit in Maintenance	Table A15-1, Item 3
	Emergency Cooling Water Pump Train in Maintenance	Table A15-1, Item 4
	Cooling Units Fail	Table A15-1, Item 5
	Common Cause Failure of Pump Motor-Operated Discharge Valves	Table A15-1, Item 6
	Common Cause Failure of Emergency Cooling Water Pumps	Table A15-1, Item 7
	Common Cause Failure of Cooling Units	Table A15-1, Item 8
	Pump Train Fails to Start or Run	Table A15-1, Item 9

APPENDIX C REA'TOR BUILDING WALKDOWN

Because they are normally inaccessible during operation, a separate walkdown checklist is provided for those components listed in the Appendix A failure mode tables that are located inside the reactor building. This is intended for efficient inspection of those items when the opportunity arises.

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Reactor Building Walkdown

Table C-1 Modified Reactor Building Walkdown

COMPONENT NUMBER	COMPONENT NAME	LOCATION	REQUIRED POSITION	ACTUAL POSITION
RB Sump	Reactor Building Sump	RB		______
DH-V1	Dropline Isolation Valve	309 RB	Closed	______
DH-V2	Dropline Isolation Valve	281 RB	Closed	______
MU-V107A	HPI Injection Check Valve	RB	Closed	______
MU-V107B	HPI Injection Check Valve	RB	Closed	______
MU-V107C	HPI Injection Check Valve	RB	Closed	______
MU-V107D	HPI Injection Check Valve	RB	Closed	______
MU-V95	HPI Injection Check Valve	RB	Closed	______
MU-V94	HPI Injection Check Valve	RB	Open	______
MU-86B	HPI Injection Check Valve	RB	Closed	______
MU-86A	HPI Injection Check Valve	RB	Closed	______
MU-220	HPI Injection Check Valve	RB	Closed	______
MU-219	HPI Injection Check Valve	RB	Open	______
RC-RV1A	Pressurizer Safety Valve	RB	Closed	______
RC-RV1B	Pressurizer Safety Valve	RB	Closed	______
RC-RV2	Pressurizer PORV	RB	Closed	______
RC-V2	PORV Block Valve	RB	Open	______
RC-V1	Pressurizer Spray Valve	RB	Closed	______
RC-V3	Pressurizer Spray Block Valve	RB	Open	______
RC-V31	Pressurizer Spray Valve	RB	Open	______
IC-V2	ICCW Isolation Valve	RB	Open	______

IC-V20	RCDT Cooler Valve	281 RB	Closed	________
RC-C1A	RCP Cooler A	RB	Valved In	________
RC-C1B	RCP Cooler B	RB	Valved In	________
RC-C1C	RCP Cooler C	RB	Valved In	________
RC-C1D	RCP Cooler D	RB	Valved In	________
IC-V79A	RCP Cooler Discharge Valve	305 RB	Open	________
IC-V79B	RCP Cooler Discharge Valve	305 RB	Open	________
IC-V79C	RCP Cooler Discharge Valve	305 RB	Open	________
IC-V79D	RCP Cooler Discharge Valve	305 RB	Open	________
AH-E1A	Cooling Unit	281 RB		________
AH-E1B	Cooling Unit	281 RB		________
AH-E1C	Cooling Unit	281 RB		________

APPENDIX D SUPPORT SYSTEM DEPENDENCY MATRIX

This appendix provides a dependency matrix (Table D-1) to help delineate the relationship between frontline systems and their supporting systems. There is a separate dependency matrix (Table D-2) included to delineate the interrelations between support systems and other support systems. These tables address only direct dependencies, not indirect dependencies. As an example, the RCPs depend directly on ICCW, which in turn depends directly on IA. Both of these direct dependencies are shown in the appropriate matrix. However, the indirect dependency of the RCPs on IA (via ICCW) is not shown.

In addition to the dependencies shown by the matrices, there are two frontline system interdependencies worth noting. One frontline system interdependency is between the HPI System and the DHR System during the recirculation mode of operation. HPI success depends on the successful operation of DHR because the recirculation mode suction source for the HPI pumps is from the discharge of the DHR pumps, which are manually aligned to take suction from the reactor building sumps. The other frontline system interdependency is between the RCPs and the HPI System. The HPI System provides seal injection cooling to the RCPs. This function works in conjunction with the ICCW System cooling of the RCP thermal barriers to maintain cooling of RCPs and thus avert the potential for an RCP seal LOCA.

Table D-1 Frontline System Dependencies on Support Systems

Support System / Front Line System	AC Power	DC Power	ESAS	Instrument Air	NSCW	DHCW	ICCW	Main Steam
DHR	X	X	X			X		
HPI	X	X	X	X	X	X		
EFW	X	X	X	X				X
RCS Pressure Control	X	X						
Reactor Building Isolation	X	X	X	X				
Reactor Building Sprays	X	X	X			X		
Reactor Building Cooling	X	X	X	X	X			
RCPs	X				X		X	

Table D-2 Support System Interdependencies

	AC Power	DC Power	ESAS	Instrument Air	NSCW	DHCW	ICCW	Main Steam
AC Power	---	X	X					
DC Power	X	---						
ESAS		X	---					
Instrument Air	X		X	---				
NSCW	X	X	X		---			
DHCW	X	X	X		X	---		
ICCW	X	X		X	X		---	
Main Steam	X	X		X				---