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BRUNSWICK STEAM ELECTRIC PLANT UNIT 2
RISK-BASED INSPECTION GUIDE

Prepared by:

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Engineering Technology Division

November 1989

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BRUNSWICK STEAM ELECTRIC PLANT (BSEP)

1. INTRODUCTION

This document has been prepared to provide inspection guidance based on review of the Probabilistic Risk Assessment (PRA), (Reference 1). The guidance should be used to aid in the selection of areas to inspect and is not intended either to replace current NRC inspection guidance or to constitute an additional set of inspection requirements. The information contained herein is based almost entirely on the Brunswick PRA performed for the plant design as of October 1, 1986. Hence, recent system experience, failures, and modifications should be considered when reviewing these tables. Since plant modifications are normally an ongoing process it is recommended that relevant changes be catalogued so that this inspection guidance can be periodically revised as required.

2. DOMINANT ACCIDENT SEQUENCES

The Brunswick PRA has a number of different accident sequences that contribute significantly to overall core damage frequency (CDF), which is 2.1E-5/year for Unit 2. The sequences that dominate core damage frequency at Brunswick Unit 2 are grouped below by their initiating events.

- Anticipated Transients Without Scram (ATWS)
  (44% of core damage frequency)
- Station Blackout
  (38%)
- Failure of High Pressure Injection and ADS
  (13%)
- Failure of Long-Term Decay Heat Removal (4%)
- Other (1%)

Six sequences have frequencies greater than 1.0 E-6/year and contribute to 83% of total core damage frequency (CDF). Six more have frequencies greater than 1.0 E-7/year and contribute to 14% of total CDF. The remaining sixteen dominant sequences contribute to total CDF.

The sequences are grouped below by their initiating events or their common outcomes such as loss of long-term decay heat removal, etc.

2.1 Anticipated Transients Without Scram (ATWS) (43.5%)

In the BSEP PRA, ATWS events are divided into two major categories:

(1) Events where the MSIVs remain open and the turbine bypass is initially available.

(2) Events where the MSIVs are closed (isolation events). The success criteria assumed in the BSEP PRA for the two categories above are shown in Tables 1 and 2.

Of the 28 dominant accident sequences, all eight of the ATWS sequences are isolation events, except for one (ATWS without isolation and SLCS failure). The dominant ATWS sequences are as follows:

1. ATWS with isolation (MSIV closure) <u>and</u>:
   a) Failure of water level control at high pressure. (13.0%)
   b) Failure of water level control at low pressure. (11.7%)
   c) Failure of HPCI with ADS inhibited (5.9%)
   d) Failure of SLC (5.3%)
   e) Failure of HPCI, failure to inhibit ADS (although this is not actually a failure because ADS is desired in this case), and failure of water level control at low pressure. (4.4%)
   f) Failure to inhibit ADS and failure of water level control at low pressure.(0.4%)
   g) Failure to depressurize long-term to meet heat capacity thermal limit (HCTL). (~0.1%)

2. ATWS without isolation and failure of Standby Liquid Control (SLC) (2.7%)

During an ATWS it is necessary to provide protection of the core and containment until such time as subcriticality is achieved. The core requires coolant makeup to match the power level. Recirculation pump trip will reduce power to a level which can be accommodated by HPCI. Primary system boundary overpressure protection is provided by normal steam flow to the turbine or turbine bypass or steam discharge through the SRVs to the suppression pool.

In the event that the turbine bypass is unavailable, all of the heat being generated in the core will be directed to the suppression pool through the SRVs. For some ATWS events, normal suppression pool cooling will not be sufficient to control containment pressure and temperature within allowable limits. For those events, it is necessary to control water level at the top of the active fuel in order to limit the total heat production. It may also be necessary to depressurize the reactor long-term into the accident to meet the HCTLs, whether or not the HPCI system is operational.

For isolation transients, recirculation pump trip and SLCS activation must occur very early in the sequence to limit power to a level that can be accommodated by HPCI and also to limit suppression pool heatup. Suppression pool cooling by the RHR system has been determined by calculation not to be a significant factor in pool temperature control. For a 43 gpm SLCS capacity, water level control in the core is necessary to limit final suppression pool temperature. The operator must inhibit ADS and throttle HPCI and RCIC to accomplish this. If HPCI fails and the operator inhibits ADS, it is assumed to result in core damage.

Failure of the affected components is reflected in the accident sequences described above.

2.2 Station Blackout (37.3%)

Station blackout (no ac power) sequences, the next most sizable contributors to core damage frequency are initiated by loss of off-site power. Failure of both Diesel Generators 3 and 4 (DG3 and DG4) represents station blackout. If HPCI, RCIC, or ADS and LPCI (Unit 1 powered components) succeed, all reactor safety functions are provided for some time.

In BWR station blackout phenomenology, there are three issues which control accident sequence progression and timing. The first issue is battery depletion. With no ac power, the batteries are not being recharged. Depletion can be expected to occur between two and eight hours after loss of all ac power, depending on the load distribution. For BSEP Unit 2, depletion was assumed at 4 hours. It is then no longer possible to control HPCI, RCIC or LPCI because of lack of instrumentation in the reactor to measure water level.

HPCI and RCIC generally require room cooling for the turbine instrumentation and bearing cooling from the pumped fluid stream. LPCI generally requires both pump seal cooling and room cooling. Loss of cooling in the HPCI and RHR rooms or RCIC and LPCI rooms will not result in high enough room temperatures within 24 hours to result in turbine failure. Over heating of the HPCI turbine bearing will occur sometime after 6 hours when suppression pool temperature ranges from 200°F to 240°F. RCIC isolation will occur at about 6 to 8 hours because of high suppression pool backpressure. Early isolation of HPCI or RCIC on steam tunnel temperature at about 4 hours is expected to be overridden by the operators as per the emergency operating procedures. HPCI/RCIC failure will be delayed until 6 to 8 hours but battery depletion occurring at 4 hours takes precedence over these later events.

Finally, it is necessary to maintain the heat capacity thermal limits (HCTL). As station blackout continues, the suppression pool will continue to heat up reaching the upper HCTL at about 5 to 6 hours. The reactor must then be depressurized to maintain the limit, ultimately to a level so low that HPCI and RCIC become inoperable. There is then no further assurance that core damage will not occur. Critical timing for entering this steam cooling stage is about 8 to 10 hours from initiation of station blackout. However, again battery depletion would occur beforehand at about 4 hours.

During a station blackout, HPCI, RCIC and LPCI can operate until one of the three situations described above leads to their failure. In the case of a stuck open SRV, HPCI unavailability occurs at approximately two hours because of inadequate steam pressure.

The actual loss of off-site power sequences in the BSEP PRA, and their percent contribution to core damage frequency, are as follows:

Loss of off-site power <u>and</u>:

a) Failure of on-site ac power (DGs) dc battery failure at 4 hours, and failure to recover off-site power by 5 hours. (36.2%)
b) Failure of Unit 2 DGs 3 and 4, failure of HPCI and RCIC, and failure of LPCI (components powered by Unit 1 DGs). (0.5%)

c) Subsequent stuck open SRV, failure of Unit 2 DGs 3 and 4, success of LPCI (components powered by Unit 1 DGs), dc battery failure at 4 hours, and failure to recover off-site power within 5 hours. (0.3%)
d) Subsequent stuck open SRV, failure of Unit 2 DGS 3 and 4, failure of LPCI (components powered by Unit 1 DGs), and failure to recover off-site power by 2 hours. (0.2%)
e) Failure of Unit 2 DGs 3 and 4, failure of HPCI and RCIC (but success of LPCI), and failure to recover off-site power by 5 hours (0.1%)

2.3 Failure of High Pressure Injection (HPI) and Automatic Depressurization System (ADS) (12.6%)

Several accident sequences lead up to failure of high pressure injection, i.e. HPCI, RCIC and CRD, with subsequent failure of the ADS. These sequences include:

a) Transient with subsequent stuck open SRV, failure of HPCI, and failure of ADS. (10.2%)
b) Loss of off-site power, failure of HPCI, RCIC and CRD, and failure of ADS. (1.6%)
c) Failure of turbine bypass, with eventual MSIV closure, failure of HPCI, RCIC and CRD, and failure of ADS. (0.4%)
d) Turbine trip, subsequent loss of feedurater, failure of HPCI, RCIC and CRD, and failure of ADS. (0.2%)
e) Intermediate LOCA, failure of HPCI, and failure of ADS. (0.1%)
f) Small LOCA, failure of HPCI and RCIC, and failure of ADS. (0.1%)
g) Turbine trip, failure of turbine bypass, failure of HPCI, RCIC and CRD, and failure of ADS. (0.1%)

It should be noted that the adequacy of HPCI, RCIC or CRD to mitigate the transient is of course, dependent on the nature of the initiating event, e.g. RCIC is inadequate for intermediate LOCAs and is therefore its operability is irrelevant in that situation.

2.4 Failure of Long-Term Decay Heat Removal (3.7%)

The final major group of accident sequences is one which consists of transients which ultimately evolve into failure of the long-term decay heat removal function (LTDHR). LTDHR can be accomplished by RHR suppression pool cooling, suppression pool spray, shutdown cooling or alternate shutdown cooling. It can also be accomplished with the Condensate, Feedwater, CRD, LPCI or CS systems together with the Power Conversion System. These sequences include:

a) Failure of turbine bypass, with eventual MSIV closure, and failure of LTDHR. (2.7%)
b) Transient with stuck open SRV and loss of LTDHR. (0.5%)
c) Turbine trip, failure of turbine bypass, and failure of LTDHR. (0.3%)
d) Loss of dc bus 2A1, failure of turbine bypass, and loss of LTDHR. (0.1%)
e) Small LOCA and failure of LTDHR. (0.1%)

2.5 Other Dominant Accident Sequences (1.6%)

The remaining dominant accident sequences, which comprise somewhat over 1% of core damage frequency, are the following, all assumed to lead directly to core damage:

a) Reactor vessel rupture. (~1.4%)
b) Steamline break outside containment and failure of MSIV closure. (~0.1%)
c) Interfacing system LOCA. (~0.1%)

The reactor vessel rupture was assumed large enough such that no coolant injection systems can keep the core covered. The steamline LOCA with MSIV failure bypasses the containment as does the RHR interfacing system LOCA (the so-called "V sequence").

TABLE 1
ATWS WITH ISOLATION SUCCESS CRITERIA[1]

Reactor Subcriticality	Coolant Makeup[2]	Containment Protection[2]	Decay Heat Removal
Manual scram or RPT and SLCS	HPCI or 2/7 ADS and 1/2 CS or 2/7 ADS and 1/4 LPCI	Control water level throughout transient and SLCS and Depressurize at HCTL	Decay heat removal is not considered in the ATWS event trees. ATWS events which involve successful reactor subcriticality, coolant makeup, and containment protection but failure of long-term decay heat removal are negligible ($<1.0 \times 10^{-8}$/yr).

[1] Assumes a 43-gpm SLCS and no ARI
[2] The notation m/n denotes m out of n trains (or components) must be successful in order to fulfill the function in question.

TABLE 2
ATWS (WITHOUT ISOLATION) SUCCESS CRITERIA[1]

Reactor Subcriticality	Coolant Makeup[2]	Containment Protection[2]	Decay Heat Removal
Manual scram or RPT and SLCS	HPCI or Feedwater or 2/7 ADS and 1/2 CS or 2/7 ADS and 1/4 LPCI	SLCS and Depressurize at HCTL	Decay heat removal is not considered in the ATWS event trees. ATWS events which involve successful reactor subcriticality, coolant makeup, and containment protection but failure of long-term decay heat removal are negligible ($<1.0 \times 10^{-8}$/yr).

[1]Assumes a 43-gpm SLCS and no ARI

[2]The notation m/n denotes m out of n trains (or components) must be successful in order to fulfill the function in question.

3. SYSTEM PRIORITY LIST

The Brunswick core damage prevention systems have been ranked in Table 3 according to their importances in preventing core damage. Other plant systems not appearing in the list are generally of lesser importance than those included here.

There are two criteria that contribute to the risk significance of a system or component: the probability that it will fail and the amount that risk is increased when it is inoperable. In planning inspections, it is usually best to consider a combination of these two criteria so that the items most likely to cause significant risk increases are given the most attention. The "Inspection Importance Measure,"* which combines both criteria, has been used to rank the systems and components in this guide. However, some items with very low failure probabilities can cause very large increases in risk if they do become inoperable. Consequently, ranking systems solely on the basis of their risk contribution when inoperable** can result in a substantially different ordering of the plant's systems. When a system is known to be inoperable or is experiencing abnormally high failure rates, it is appropriate to consider the importance of that system's failure independently from the normal failure rate assumed for it in the PRA. Therefore, Table 3 also includes a second system list rank ordered by the importance of their failures.

*The Inspection Importance Measure is equivalent to the Fussell-Vesely Importance Measure for ranking purposes. Both measures combine the risk significance of a system's failure or unavailability with the probability that the system will fail or be unavailable.

**The Birnbaum Importance Measure considers only the risk significance of a system should it fail or be unavailable, regardless of the actual probability that it will fail or be unavailable.

TABLE 3
SYSTEM PRIORITY RANKING

By Contribution to Core Damage Frequency[1]	By Risk Significance of System Being Unavailable[2]
Emergency Diesel Generators High Pressure Coolant Injection	Reactor Protection Service Water
Automatic Depressurization Diesel Generator and Switchgear Cell Ventilation Standby Liquid Control Reactor Protection Service Water	Automatic Depressurization
Residual Heat Removal DC Power AC Power Battery Room Ventilation Reactor Core Isolation Cooling Reactor Water Cleanup[(a)]	Emergency Diesel Generators AC Power Battery Room Ventilation
Control Rod Drive Hydraulic	Diesel Generator and Switchgear Cell Ventilation DC Power Residual Heat Removal Standby Liquid Control High Pressure Coolant Injection
ECCS Actuation Screen Wash[(b)] Low Pressure Coolant Injection[(c)]	Reactor Water Cleanup[(a)] ECCS Actuation
Core Spray[(d)] Reactor Building Closed Cooling Water[(d)]	Control Rod Drive Hydraulic Reactor Core Isolation Cooling Screen Wash[(b)] Low Pressure Coolant Injection[(c)]
	Core Spray[(d)] Reactor Building Closed Cooling Water[(d)]

General Notes:

[1]The ranking in column 1 is appropriate to use for systems that are functioning normally. It is based on the Fussell–Vesely Importance Measure, which is the systems's contribution to the core damage frequency, assuming that the system is operating with normal reliability.

[2]The ranking in column 2 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 core damage frequency that results when the system is assumed to be inoperable.

[3]The containment system shown on these lists are ranked with respect to their contributions to core damage frequency, only. Their importance for accident consequence mitigation was not considered.

[4]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.

Specific Notes:

(a)Analyzed in PRA together with SLC.

(b)Included in PRA together with SWS.

(c)Included with RHR system.

(d)System not appearing among dominant accident sequence cutsets in the PRA. Therefore system inspection tables could not be developed for this system.

Because there is uncertainty in the data and modeling assumptions contained in the PRA, there is also uncertainty with respect to the risk significance of each system, and thus, uncertainty in their rank order in Table 3. Adjacent systems on the list should be considered to have approximately equal contributions to risk, except where they have been separated by dashed lines to indicate numerically significant differences in their importance measure values.

4. COMMON CAUSE OR DEPENDENT FAILURES

In the BSEP PRA, common cause, or dependent, failures are classified into several different categories. Common cause initiating event dependencies are divided into internal and external event classes.

Internal events include general transients such as loss of off-site power, failure of specific dc busses (special transients), LOCAs and interfacing system LOCAs leading directly to core damage. External events include physical interaction dependencies such as those resulting from fire, flood and seismic events. Human interaction dependencies, such as operator failure to initiate systems, etc. are discussed in Section 5 following.

It should be noted that loss of the Nuclear Service Water System was considered to be negligible as an initiating event.

From the results of the BSEP PRA, important dependent failures for the systems affected are as follows:

a) Two or more ADS SRVs fail to open because of o-ring leakage or other dependent failure mechanisms.
b) Diesel generators 3 and 4 fail to start or fail to run.
c) RHR system failures:
   i) LPCI Mode
      1) Loops A and B pumps (C002A,B,C and D) fail to start or run
      2) Loops A and B minimum flow MOVs (F007 A and B) fail to open
   ii) Suppression Pool Cooling (SPC) Mode
      1) Loops A and B heat exchanger bypass MOVs (F048 A and B) fail to open
      2) Loops A and B injection MOVs (F028 A and B) fail to open
      3) Loops A and B injection MOVs (F024 A and B, F027 A and B) fail to open
d) Standby Liquid Control (SLC) pumps A and B fail to start
e) Service Water System (SWS)
   i) 4 or more pumps fail to run
   ii) RHR heat exchanger MOVs (V105 and V101) fail to open

5. IMPORTANT HUMAN ERRORS (Including Recovery Actions)

Human errors can be very significant to overall plant risk. The BSEP PRA has identified several human errors as particularly important contributors to risk:

5.1 Pre-Accident Errors

These errors consist of failure to restore components to their proper position after testing, maintenance or calibration activities. The most important errors are:

a) Failure to restore I&C for valves, F007 A or B, RHR pump minimum flow recirculation to suppression pool isolation MOVs.
b) Miscalibration of flow switch N021 A or B controlling RHR minimum flow recirculation valves F007 A or B.

5.2 Operator Errors During an Accident

These errors pertain to actions identified in the operating procedures which involve manual operation or alignment from the Control Room of components which must be operated manually or have failed to operate automatically. The most important errors are:

1. CRD
   a) Failure to fully open CRD flow throttling valve F002A
   b) Failure to fully open CRD pressure regulating valve F003.
2. HPCI
   Operator fails to empty drainpot A. (If drain pot A fails to drain prior to starting the HPCI turbine, a slug of water could be forced through the turbine and out the exhaust line, which will cause water hammer damage and probable turbine trip on high exhaust pressure).
3. ATWS
   Failure to inhibit ADS and failure to control RPV water level at low pressure.
4. RHR
   Operator fails to correctly initiate suppression pool cooling through Loop A
5. SLC
   Operator fails to actuate the SLC system.

5.3 Post Accident Recovery Actions

This last category of human errors involves mitigating actions taken by the operators to recover from the effects of an accident. The most important of these are the following:

1. AC Power
   Failure to recover offsite power.
2. Diesel Generator Room Cooling
   Failure to open switchgear room doors after HVAC failure.

6. SYSTEM INSPECTION TABLES

Taken together, the systems ranked by their risk importance in the first column of Table 3 contribute 95% of the core damage frequency for Brunswick Unit 2. For each of those systems, inspection guidance is provided in the form of a failure mode table, an abbreviated walkdown checklist, and a simplified system diagram. Each of these is explained in detail below.

In using these tables, however, it is essential to remember that other systems and components are also important. If, through inattention, the failure probabilities of other systems were allowed to increase significantly, their contributions to risk might equal or exceed that of the systems in the following tables. Consequently, a balanced inspection program is essential to ensuring that the licensee is minimizing plant risk. The following tables allow an inspector to concentrate on systems and components that are most significant to risk. In so doing, however, cognizance of the status of systems performing other essential safety functions must be maintained.

APPENDIX A

Table A.X-1 – System Failure Modes

For each system X, a table A.X-1 of system failure modes is provided. The introduction to these tables provides a brief description of the system and the success criteria used for the system in the PRA. (Note that the PRA success criteria may be different from the success criteria contained in the FSAR.)

The entries in these tables are the dominant events (component failures, operator errors, etc.) contributing to system failure, provided in rank order according to their risk significance. Since most systems are designed with redundant trains, it will generally take more than one of these events to fail the entire system. No effort has been made to list all of the combinations of the events that are sufficient to produce system failure because that is usually apparent from the system description in the introduction. Where single events are sufficient to fail the entire system, that is noted in the brief discussion of the event. For certain events that are important primarily because of the circumstances of a particular accident sequence, that information is also noted.

Inspection focussed on the items in the table will address approximately 95% of the risk 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 inspector to draw from his experience, plant operating history, ASME Codes, NRC Bulletins and Information Notices, INPO SOERs, vendor notices and similar sources to determine how to actually conduct his inspections 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 and emergency operating procedures, check-off lists, training, etc.
TS — Technical specifications.
ISI — In-service inspection.

Table A.X-2 – Modified System Walkdown

As above, for each system X, a table A.X-2 is included which 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 should be observed when using the checklists, since they are based on certain versions of the licensee's system operating instructions. Valve numbers used are those identified in the licensee system checklists, or P&ID's.

Figure A.X – Simplified System Diagram

A simplified line diagram is provided for each system treated. These are intended to aid in visualizing the system configuration and the location of the components discussed in the two tables. The drawing is merely a simplified schematic of the actual P&ID's in effect at the time that the PRA was prepared. It is neither a complete representation of the P&ID's nor is it a controlled document. It was utilized in the preparation of the PRA and any significant differences between this drawing and actual plant conditions may affect the information provided in Tables A.X-1 and A.X-2 and should be reported to the appropriate NRC personnel.

APPENDIX B

Table B1 – Plant Operations Inspection Guidance

This table is a collection of all of the risk significant operator actions listed in the preceding system tables. It is provided as a cross reference for use in observing operator actions and training.

Table B2 – Surveillance and Calibration Inspection Guidance

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

Table B3 – Maintenance Inspection Guidance

This table is a collection of the risk significant components listed in the preceding system tables that are considered to be significantly influenced by maintenance activities. It is provided as a cross reference to assist the inspector in selecting risk important 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 returned to service.

APPENDIX C

Table C1 – Containment and Drywell Walkdown Table

Because they are normally inaccessible during operation, a separate walkdown checklist is provided for those components listed in the preceding system tables that are located inside the containment or drywell. This is intended for efficient inspection of those items when the opportunity arises.

APPENDIX D

Table D1 – Frontline-Support System Dependencies

In a matrix format, the dependencies of the frontline systems are correlated to their associated support systems. This illustrates the impact failures or outages of support systems have on the various frontline systems.

Table D2 – Support-Support System Dependencies

As in Table D1, in a matrix format, the dependencies of the support systems are correlated to other support systems which they interface with. This illustrates the impact failures or outages of support systems have on other support systems.

7. REFERENCES

1. Brunswick Steam Electric Plant Probabilistic Risk Assessment. Raleigh, North Carolina: Carolina Power and Light Company, April 1988.

2. B. Wooten and P. Lobner (Editor), "Nuclear Power Plant System Sourcebook-Brunswick 1 & 2, 50-325 and 50-324," Science Applications International Corp., Report No. SAIC 89/1011, January 1989.

APPENDIX A

Importance Basis and Failure Mode Identification Tables and Modified System Walkdown Tables

Emergency Diesel Generator (EDS) System

Table A.1-1. Importance Basis and Failure Mode Identification

CONDITIONS THAT CAN LEAD TO FAILURE

Mission Success Criteria

Four DG sets are located in the Diesel Generator Building (DGB) east of the plant. Unit 1 is generally served by DGs 1 and 2 through emergency buses E1 and E2 while Unit 2 is generally served by DGs 3 and 4 through emergency buses E3 and E4. Tie breakers allow interconnection of the emergency buses; however, they are left in the OPEN RACKED-OUT position to prevent paralleling of opposite division emergency power sources. Two of the Unit 2 RHR pumps and both LPCI injection valves are powered by Unit 1 emergency buses.

Each DG set consists of a General Electric generator driven by a Nordberg diesel engine, with a Woodward load-sensing type governor to maintain engine and generator speed. Auxiliary systems and components for each DG are also located in the DGB, except as noted. The DG systems for BSEP Units 1 and 2 are identical. The auxiliary systems for each DG are:

1. diesel engine air intake and exhaust system,
2. diesel engine fuel oil system,
3. diesel engine starting air system,
4. diesel engine cooling system,
5. diesel engine lube oil system
6. diesel generator jacket water system,
7. governor, and
8. generator.

The success criteria in the PRA for the DGs is to start on demand and provide electrical power for a six-hour mission time. DGs 3 and 4 can provide emergency ac power to buses E3 and E4. Only one of these DGs is required to provide emergency power. The six-hour mission time was determined based on considerations of off-site power recovery.

1. Diesel Generators 1,2,3 or 4 Fail to Start or Run

Under loss of offsite power conditions, failure of the diesels to start or run causes loss of all AC power (PT, MT)

It should be noted that an extended loss of offsite power incident occurred at BSEP beginning on the evening of June 17, 1989 in which the diesel generators were required to operate for a period of approximately 13 hours to provide power to Unit 2. This increases the importance of EDGs failing to run.

2. Diesel Generators 1,2,3 or 4 in Test or Maintenance

Similarly, unavailability of the diesels due to test or maintenance, combined with other failures, fails all AC power under loss of offsite power conditions (PT,MT,TS).

3. Diesel Generators 1,2,3 or 4 Generator Output or Output Breaker Failure

No power output from the diesel generators or failure of the output breakers fails all AC power under loss of offsite power conditions (PT,MT).

Note: Other significant DG related failure modes are identified for the DG Cell Ventilation, Battery Room, and Switchgear Cell Ventilation, AC Power and Service Water Systems. BNL has developed a proposed specific inspection guide for diesel generators. It is provided in Table A.1-3.

Emergency Diesel Generator (EDG) System

TABLE A.1-2 MODIFIED SYSTEM WALKDOWN

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
Diesel Gen. 1 Circuit Breaker Control Switch					—	Diesel Generator Bldg.– Gen. Control Panel DG. Cell 1	Neutral	
4160V System Circuit Breaker Control Switch					—	Diesel Generator Bldg.– Gen. Control Panel DG. Cell 1	Neutral	
Diesel Gen. 2 Circuit Breaker Control Switch					—	Diesel Generator Bldg.– Gen. Control Panel DG. Cell 2	Neutral	
4160V System Circuit Breaker Control Switch					—	Diesel Generator Bldg.– Gen. Control Panel DG. Cell 2	Neutral	
Diesel Gen. 3 Circuit Breaker Control Switch					—	Diesel Generator Bldg.– Gen. Control Panel DG. Cell 3	Neutral	
4160V System Circuit Breaker Control Switch					—	Diesel Generator Bldg.– Gen. Control Panel DG. Cell 3	Neutral	

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Emergency Diesel Generator (EDG) System

TABLE A.1-2 MODIFIED SYSTEM WALKDOWN (Cont'd)

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
Diesel Gen. 4 Circuit Breaker Control Switch					—	Diesel Generator Bldg.– Gen. Control Panel DG. Cell 4	Neutral	
4160V System Circuit Breaker Control Switch					—	Diesel Generator Bldg.– Gen. Control Panel DG. Cell 4	Neutral	
General Actuation Signal	1,2,3&4	D.G. Rooms	*					

*Check latest surveillance test to assure that any unsatisfactory items have been corrected.

TABLE A.1-2 (Cont'd)

REFERENCE DOCUMENTS

TITLE	I.D. NO.	REV	DATE
Systems Procedures:			
Diesel Generator Operating Procedure	OP-39	040	9/22/88
System Description:			
Brunswick PRA Volume I	Section M.3.4.10		
"Emergency Diesel Generators"			
P&ID's No.:			
Brunswick PRA Volume I	Fig. M.3.4-20		
"DG Simplified Diagram"			

Emergency Diesel Generator (EDG) System

Table A.1-3. Proposed Inspection Plan for Diesel Generators at Nuclear Plants

A. Objective

To review and evaluate Diesel Generator design operation, and maintenance at NPPs to ensure that the DGs will be available when needed to power safety systems.

B. Details

1. The inspection of the following items should focus on DG auxiliary systems as follows: Fuel Injection System, Turbocharger, Starting System, Speed/Load Control, Cooling Water, Lube Oil, Fuel Oil, Control and Monitoring Systems, and Generator.

2. Using the LER, 50.55e, and Part 21 systems computer printout and select 3 recent failures (within 2 years) for followup at the NPP. When at the plant select an additional 2 failures from the internal systems. Evaluate the licensee's response to these failures for proper failure analysis, corrective action, notification of vendor, Part 21 evaluation and documentation.

3. Maintenance: Refer to IE I.P.s 62700 and 62702, as they apply to DG maintenance. Additionally, does the NPP have, and have they implemented the DG vendors' maintenance recommendations (especially those recommendations unique to nuclear service DGs such as Colt's described in NSAC-79)? Are maintenance personnel specially trained on DGs? Is failure information fed back into maintenance program? Has the NPP implemented recommendations of various studies referenced in Section 4 above.

4. Design Change Control: Select two DG modifications and verify proper implementation. Utilizing information from DG vendor inspection on modifications recommended, verify that NPP is receiving all pertinent information in this area from the vendor. (Reference IE I.P. 37700).

5. Spare Parts and Procurement: Review how spare parts and services are purchased and parts stored, both from DG vendor and direct from subvendor. Verify adequate Part 21 and QA, particularly when vendors are only supplying commercial grade parts and services (e.g., Woodward Governor and Stewart and Stevenson). Verify ASME code specified where appropriate. Tour spare parts storage area. (Reference IE I.P. 38701B).

6. Training: Ensure appropriate DG specific training given to maintenance, operations, QA, and management personnel. Are there adequate documents to describe DG operation onsite (both main engine and auxiliary system)? (Reference IE I.P. 41700).

7. Observe DGs in operation. Ensure they run smoothly and are operated per procedure. Look for abnormal vibration and leaks (air, fuel oil, or lube oil). Check that readings are within specified limits. Are limits per DG vendor recommendations? Are recommendations clearly specified? Is air quality in DG room satisfactory without excessive dust? Are control cabinets properly gasketed? Are instruments calibrated? Is trending of operating data performed to detect degradation early?

8. Is NPP receiving all appropriate service information from vendor: design, maintenance, operational, etc? This is especially important for General Motors DG owners (verify they receive "Power Pointers" from GM).

9. Review site practices to limit DG cold fast starts.

10. Reliability records and calculations: Check logs, procedures, and calculations versus Reg. Guide 1.108 criteria.

11. Ensure that pertinent studies on DG performance have been reviewed and recommendations implemented as appropriate (e.g., NUREG/CR-0660 and NSAC-79).

12. Torquing: Ensure plant has adequate specifications for all torquing. Ensure it is documented and done with calibrated equipment. Observe re-torquing if in progress.

Source

J.C. Higgins and M. Subudhi, "A Review of Emergency Diesel Generator Performance at Nuclear Power Plants," NUREG/CR-4440, Brookhaven National Laboratory, November 1985.

References

1. NSAC-79, "A Limited Performance Review of Fairbanks Morse and General Motors Diesel Generators at Nuclear Plants," Nuclear Safety Analysis Center, Electric Power Research Institute, April 1984.

2. G. Boner and H. Hanners, "Enhancement of Onsite Emergency Diesel Generator Reliability," NUREG/CR-0660, University of Dayton, February 1979.

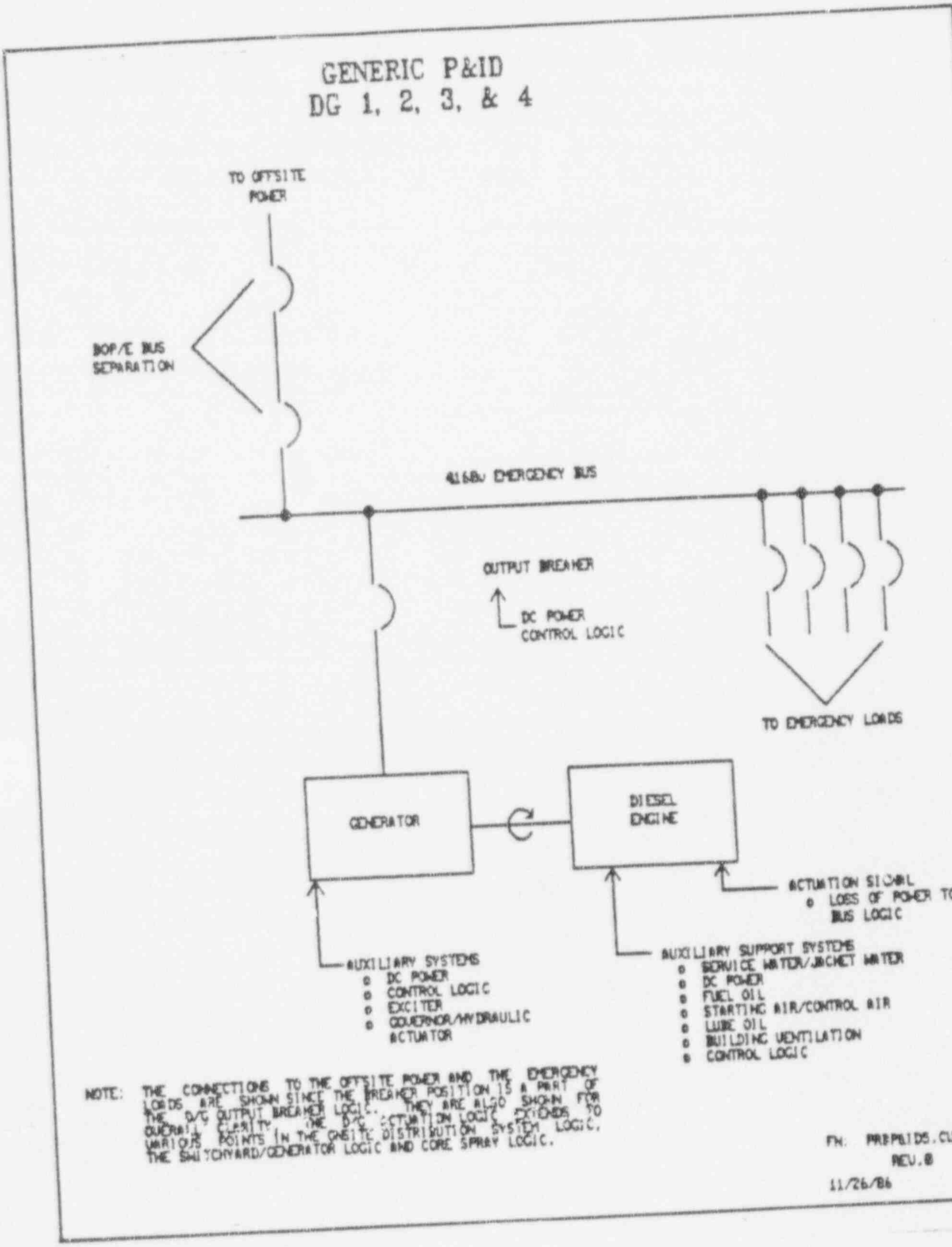

Fig. A.1-1 (PRA Fig. M.3.4-20). DG Simplified Diagram.
CAUTION: This is NOT a controlled document.

High Pressure Coolant Injection (HPCI) System

Table A.2-1 Importance Basis and Failure Mode Identification

CONDITIONS THAT CAN LEAD TO FAILURE

Mission Success Criteria

The High Pressure Coolant Injection (HPCI) System, an Engineered Safeguards System, serves to provide sufficient core cooling to prevent excessive fuel cladding temperatures in the event of a small line break of any unisolatable line directly associated with the nuclear boiler. It is designed to operate and maintain the reactor core covered when reactor pressures are high and the break area is small (up to five inches in diameter).

The success criterion for the HPCI system is the availability of the system to inject 4250GPM into the reactor vessel in accordance with technical specification requirements using the HPCI pump with flow through injection valve F006 for 24 hours.

HPCI consists of a 100% capacity single train with DC motor-operated valves and a steam turbine-driven pump. The normal water supply is from the condensate Storage Tank (CST). If the CST level decreases to a predetermined level, or the suppression pool level rises above a predetermined level, pump suction is automatically transferred to the suppression pool. Injection to the reactor vessel is into the A feedwater line (through valve F006) and is distributed through the feedwater spargers.

1. <u>HPCI Pump or Turbine Fails to Start or Run</u>

Since HPCI is a single train system, failure of the pump or turbine to start or run prevents HPCI flow (PT,MT).

2. <u>HPCI Pump or Turbine in Tests or Maintenance</u>

As above, unavailability of HPCI for test or maintenance prevents HPCI flow. (PT,MT).

3. <u>Failure of Level Switch LEN014</u>

Since LEN014 monitors the level in Drain Pot "A", failure of this switch can cause a water slug to the HPCI turbine. (PC,PT,MT).

4. <u>Operator Failure to Override Steam Tunnel High Temperature Trip</u>

During station blackout scenarios, early isolation of the HPCI turbine on high steam tunnel temperature will occur automatically unless overridden by the operators. Failure of the operators to override when required can lead to core damage (OP).

5. Normally Closed HPCI Pump Discharge Isolation MOV F006 Fails Closed

Failure of MOV F006 to open when required prevents HPCI flow to the feedwater injection line (PT,MT).

6. Delayed Actuation Signal to MOV F006 on Scram Coupled with Insufficient Flow Through the Mini-Flow Line Due to MOV F012 Failing to Open

Failure of F006 to open in a timely manner together with insufficient flow through the minimum flow line resulting from isolation MOV F012 failing to open will cause failure of the HPCI pump due to deadheading (PC,PT,MT).

7. Failure of Normally Closed Steam Supply Isolation Valve F001 to Open

Failure of MOV F001 to open when required prevents steam flow to the HPCI turbine (PT,MT).

8. Failure of Lube Oil Cooling

The HPCI pump discharge provides cooling water flow to the turbine lube oil cooler. Failure of this cooling system will cause HPCI turbine failure (PT,MT).

9. Failure of the Vacuum Breaker on Startup Causes Insufficient Steam Flow from the Turbine

Valves F075,76,77 and 79 form a vacuum breaker on the steam line to the suppression pool. If the breaker fails when HPCI initiates, condensing steam will cause a vacuum which will draw water from the suppression pool into the steam discharge line. This causes a trip on high exhaust line pressure and/or water hammer will rupture the steam discharge piping (PT,MT).

10. Insufficient Flow From HPCI Pump Discharge Line

Insufficient flow through the HPCI pump discharge line can be caused by failure of check valve F005 to open or by normally open MOV F007 failing closed (PT,MT).

11. Operator Fails to Empty Drain Pot "A"

If drain pot "A" fails to drain prior to starting the HPCI turbine, a slug of water could be forced through the turbine and out the exhaust line, thereby causing water hammer and HPCI turbine trip on high exhaust pressure (OP).

12. Failures in Pipe Segment Leading From Drain Pot "A" to Main Condenser

Failure of DC solenoid AOVs F028 and F029 to open when required could cause a slug of water from Drain Pot "A" to be forced through the turbine, thereby causing water hammer and HPCI turbine trip on high exhaust pressure as in 9 above (PT,MT).

High Pressure Coolant Injection (HPCI) System

TABLE A.2-2 MODIFIED SYSTEM WALKDOWN

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
HPCI T.D. Pump Control and Relay Logic					13	125V DC Dist. Panel 4A	On	
Level SW Drain Pot A	2-G-41 LSH-N014-1				14	CSS PNL 2A-RX EL 20' 120V AC	On	
MOV Injection Valve Control Switch (SS)	F006	Control Room Panel P601	Closed		Compartment B-17	MCC 2XDA EL 20'	On	
					4 (Heater)	120V AC Dist. Pn. 2DX EL	On	
MOV Control Switch (S3)	F001	Control Room Panel P601	Closed		Comp. B-21	MCC 2XDA EL 20'	On	
					8 (HTR)	120V AC Dist. Pnl. 2DXA El 17'	On	
MOV Control Switch	F012	Control Room Panel P601	Closed		Comp. B-16	MCC 2XDA EL 20'	On	
					Comp. B-24	120V AC Dist. Pnl. 2XDA EL 17'	On	
					11 (HTR)	120V AC Dist. Pnl. 2DXA EL 17	On	

High Pressure Coolant Injection (HPCI) System

TABLE A.2-2 MODIFIED SYSTEM WALKDOWN (Cont'd)

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
Lube Oil Cooling System		Reactor Bldg. EL 20'			Comp. B-11	MCC 2XDA EL 20	On	
Aux. Oil Pump					12 HTR	120V AC Dist. Pnl. 2DXA	On	
Local Control Switch	N/A	MCC 2XDA B-11	Stop					
Key Lock Switch	N/A	MCC 2XDA B-11	Norm					
Vacuum Breaker		Panel P601 Control Room				MCC 2XA EL 20'		
MOV	F075		Open		Comp. DE 2	MCC 2XA EL 20'	On	
					7 HTR	120V Dist. Panel HS3 EL 20'	On	
MOV	F079	Panel P601 Control Room	Open		Comp. DQO	MCC 2XB EL 20'	On	
Key Lock Local Sw.	F079	MCC 2XB	Norm		24 HTR		On	
ASSD Feed					DT 2	MCC 2XC	Off	

High Pressure Coolant Injection (HPCI) System

TABLE A.2-2 MODIFIED SYSTEM WALKDOWN (Cont'd)

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
MOV	F007	Panel P601 Control Room	Open		3	120V AC Dist. Panel 2DXA	On	
MOV	F028	Panel P601 Control Room	Open		AOV R1A-IV 203 & 103	HPCI Roof EL 4'	Open	
MOV	F029	Panel P601 Control Room	Open		R1A-IV 202 & 102	HPCI Roof EL 4'	Open	

TABLE A.2-2 (Cont'd)

REFERENCE DOCUMENTS

TITLE	I.D. NO.	REV	DATE
System Procedures:			
High Pressure Coolant Injection System	OP-19	060	8/11/88
System Description:			
Brunswick PRA Volume I	Section M.3.4.3		
"High Pressure Coolant Injection System"			
P&ID's No.:			
Brunswick PRA Volume 1	Fig. M.3.4-3		
"HPCI Simplified Diagram"	Sheets 1–3		

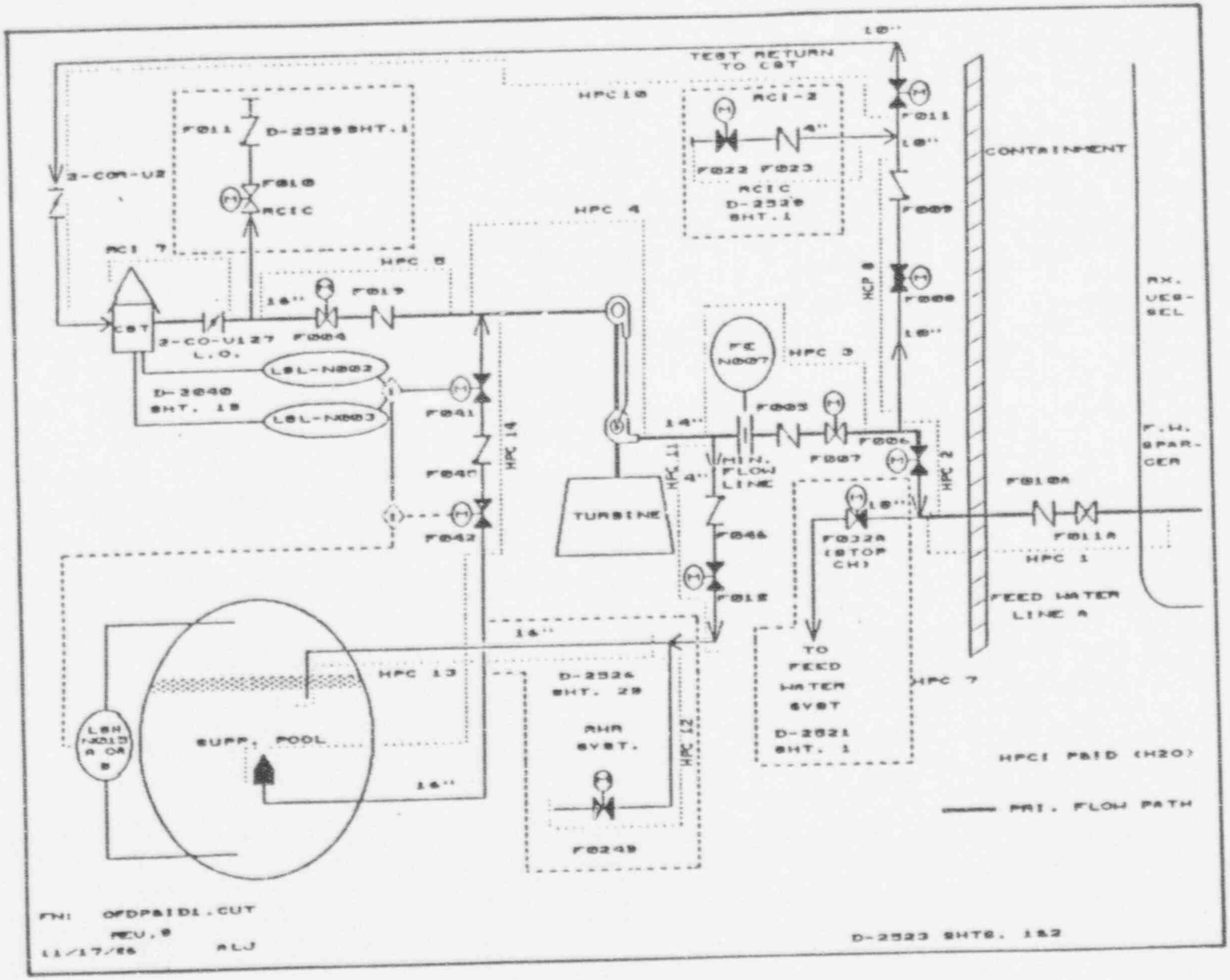

Fig. A.2-1 (PRA Fig. M.3.4-3). HPCI Simplified Diagram (Sheet 1 of 3).
CAUTION: This is NOT a controlled document.

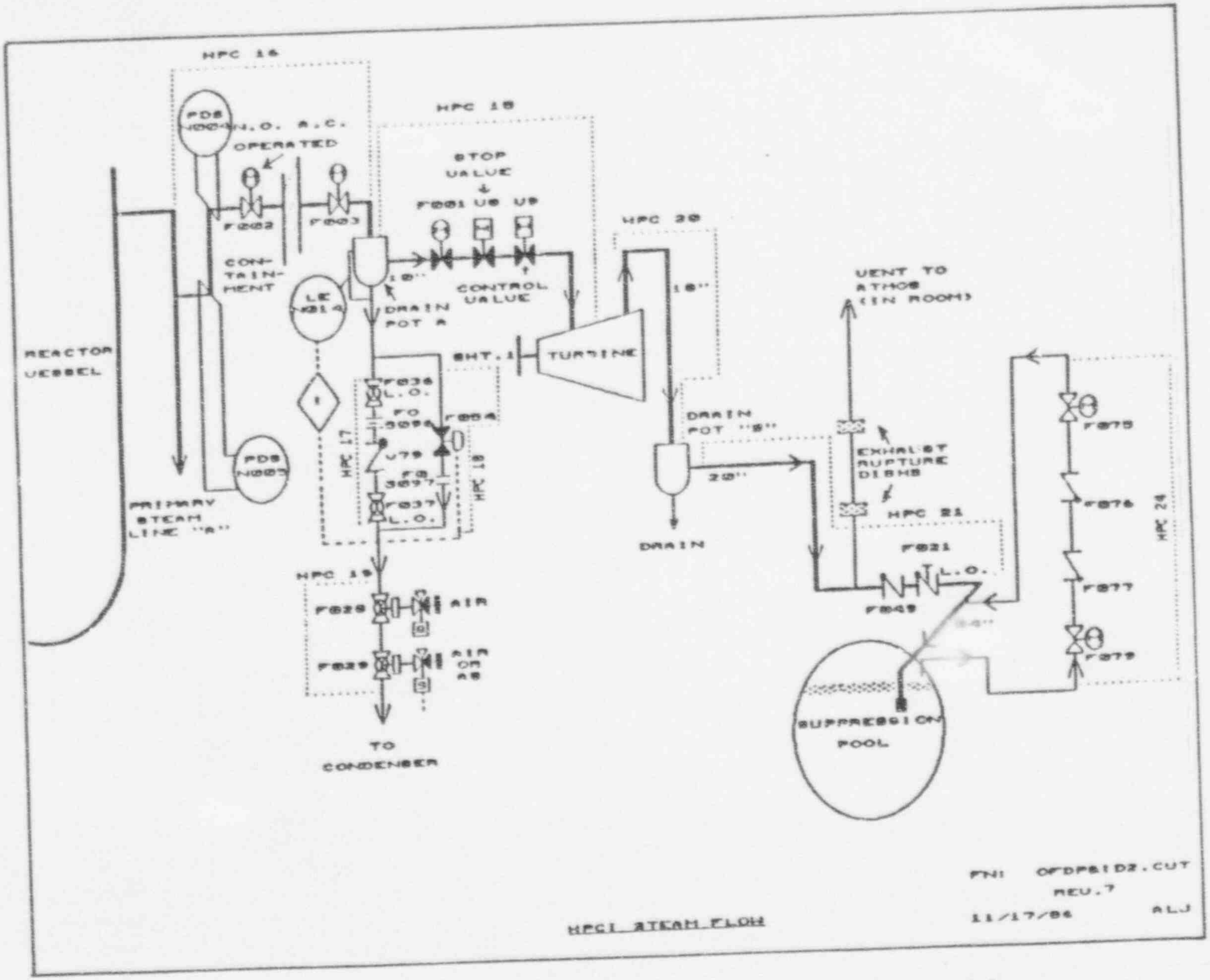

Fig. A.2-1 (PRA Fig. M.3.4-3). HPCI Simplified Diagram (Sheet 2 of 3).
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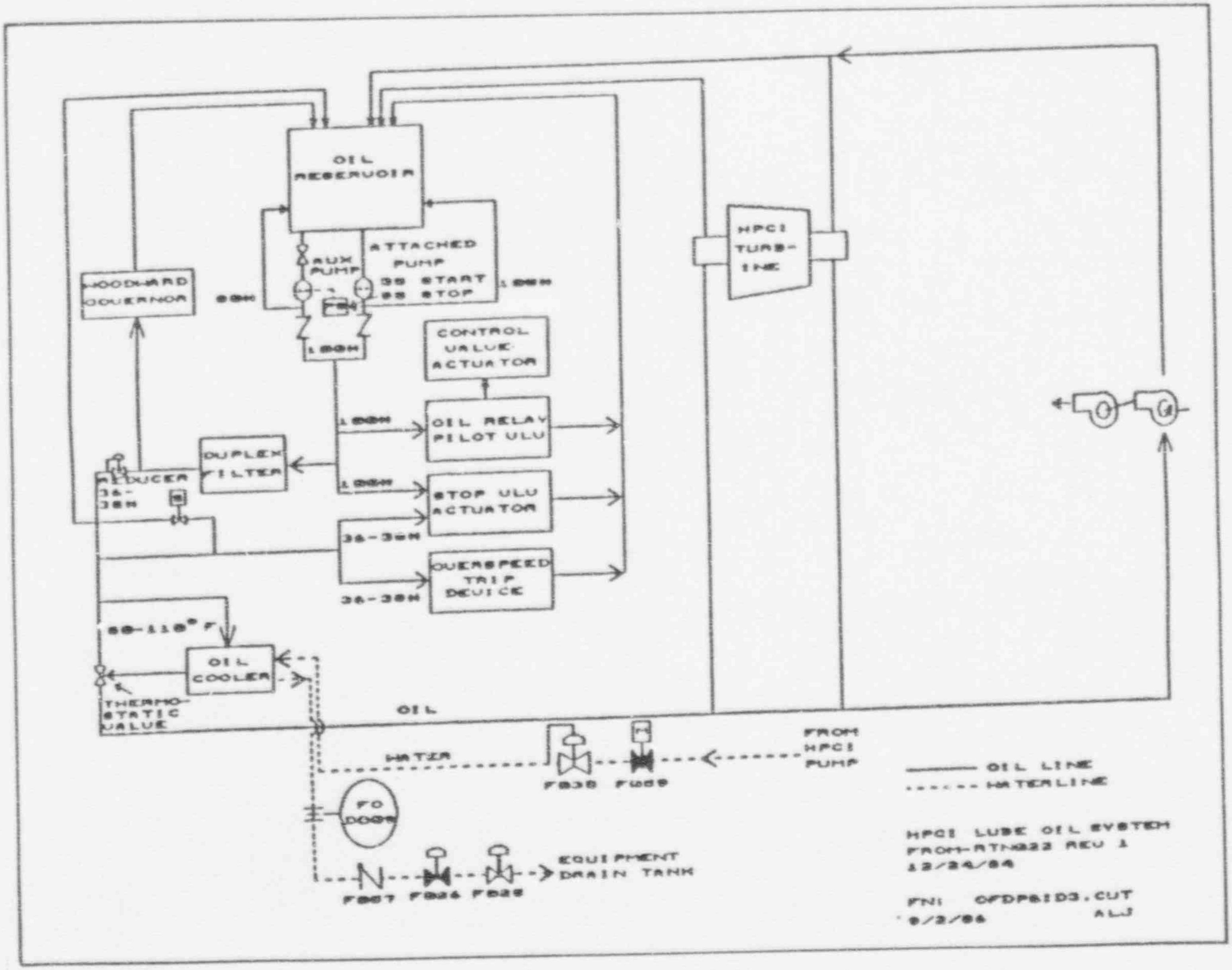

Fig. A.2-1 (PRA Fig. M.3.4-3). HPCI Simplified Diagram (Sheet 3 of 3).
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Automatic Depressurization System (ADS)

Table A.3-1 Importance Basis and Failure Mode Identification

CONDITIONS THAT CAN LEAD TO FAILURE

Mission Success Criteria

The Automatic Depressurization System (ADS) is an engineered safeguards system. As a Core Standby Cooling System (CSCS), it serves as a backup for HPCI. Should HPCI fail during loss-of-coolant accident (LOCA) conditions from a small line break, the ADS depressurizes the nuclear system so that the Low Pressure Coolant Injection (LPCI) or Core Spray Systems (CSS) can operate. ADS is designed to depressurize the reactor vessel in sufficient time to allow LPCI or CSS to provide core cooling to prevent excessive fuel cladding temperatures.

There are 11 safety-relief valves (SRVs) (B21-F031A through L, excluding I) associated with reactor pressure vessel overpressure protection. Seven of these 11 valves, B21-F013A, C, D, H, J, K, and L, are used in the ADS. When open, each valve discharges through a separate line to a point below the minimum water level of the suppression pool.

The success criterion for the ADS is two of seven ADS valves operating successfully to depressurize the RPV until the LPCI and/or CSS can be used.

The analysis given by the report Determination of the Minimum Number of Safety Relief Valves for Brunswick Boiling Water Reactor Depressurization (Nuclear Safety, CP&L) was considered. The analysis was done using the RELAP5 M0D2 computer code to determine if one SRV with CCS is sufficient to prevent core damage. The sequence analyzed was one in which a transient event is followed by a scram with loss of all high pressure systems that would normally be available for makeup water to the vessel (i.e., HPCI, RCIC, CRD, and feedwater). The analysis demonstrated that, with only one SRV open, pressure would be reduced to 350 psia within 16.67 minutes with no core damage problems.

Also analysis given by GE, document NEDC-30936-P, dated November 1985, BWR Owner's Group Technical Specification Improvement Methodology (With Demonstration For BWR ECCS Actuation Instrumentation) Part 1, was considered. The GE analysis showed two SRVs are required to depressurize the vessel below the shutoff head of the low pressure systems in time to provide adequate core cooling.

Therefore, using the conservative approach given by the GE document, it is assumed that two of the seven ADS valves must operate.

1. O-Ring Leakage or Other Dependent Failure Mechanisms Cause Two or More ADS SRVs Failing to Open

Dependent, or common cause, failure is most likely to cause ADS failure when required. Such failures can be caused by a O-ring leakage of the ADS SRVs (PT,MT).

Automatic Depressurization System

TABLE A.3-2 MODIFIED SYSTEM WALKDOWN

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
*D.C. Solenoid Valves				—	—	—	—	—
ADS Relay Logic A&B 125V D.C.	H12-P628				11	Control Bldg. Dist. Pnl. 4B. EL 49'	On	
Fluid Flow Detector Cabinet Emer. 125V AC	CB-X4-73				11	Control Bldg. Dist. Pnl. 32B EL 49'	On	
Relay Logic-A	E-11				3	Same Panel 4A	On	
Relay Logic-B	N/A				11	Same	On	
ASSD ADS Logic B Power Sup. Isol. Switch	Control Panel H12-P628	Electronic Equip. Room EL 49'	Normal		36	Cont. Bldg. EL 23' Dist. Pnl. 2AB	On	

*Note: Since "O" ring leakage is a common cause event that could affect all eleven relief valves, this walkthrough should assure that this matter is currently addressed in existing maintenance procedures.

TABLE A.3-2 (Cont'd)

REFERENCE DOCUMENTS

TITLE	I.D. NO.	REV	DATE
System Procedures:			
Automatic Depressurization	OP-20	003	3/28/88
System Description:			
Brunswick PRA Volume 1	Section M.3.4.6		
"Automatic Depressurization System"			
P&ID's No.:			
Brunswick PRA Volume 1	Fig. M.3.4-9		
"ADS Simplified Diagram"	Sheets 1 & 2		

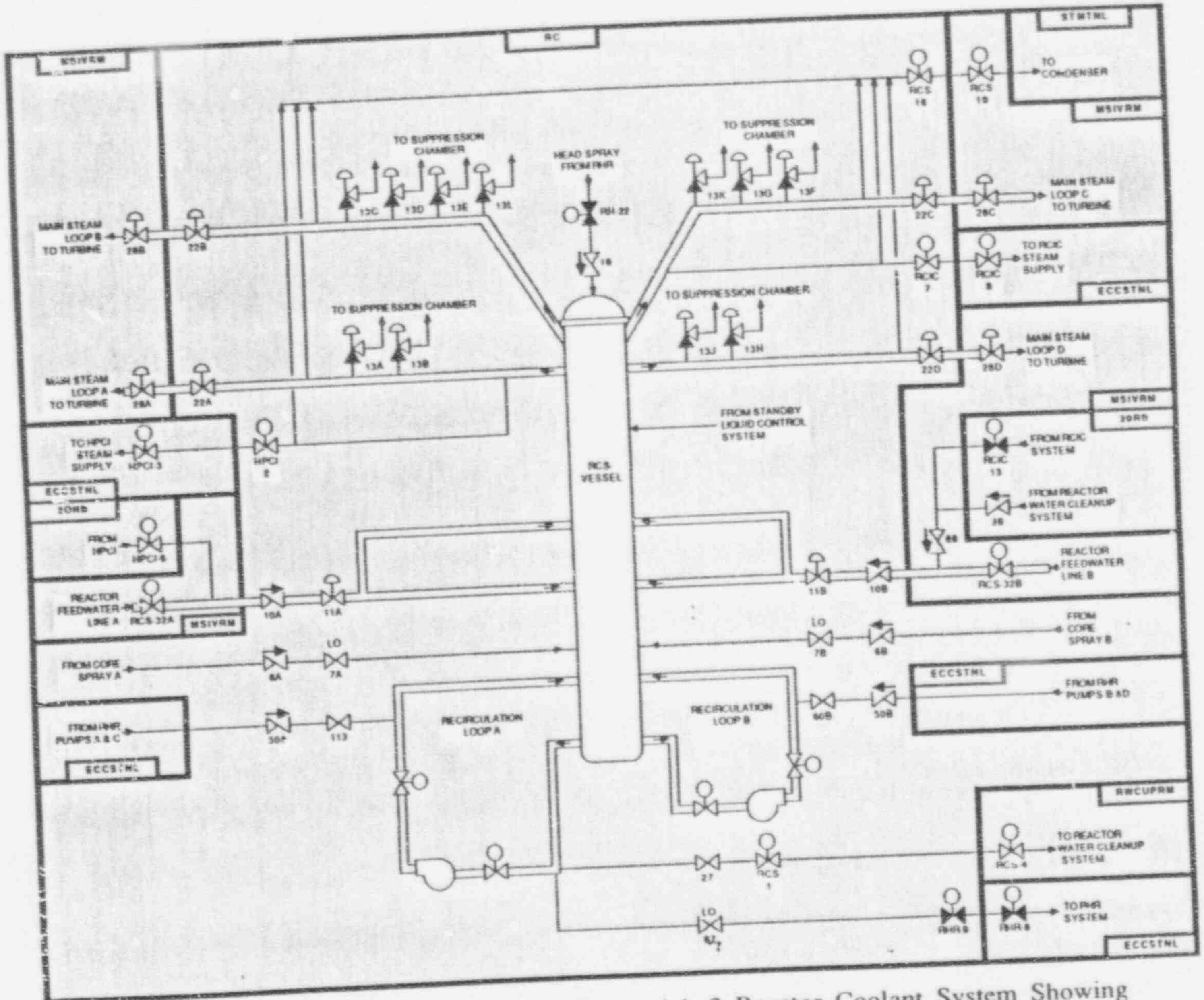

Fig. A.3-1 (SAIC 89/1011-Fig. 3.1-2). Brunswick 2 Reactor Coolant System Showing Automatic Depressurization System

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Diesel Generator and Switchgear Cell Fans

Table A.4-1 Importance Basis and Failure Mode Identification

CONDITIONS THAT CAN LEAD TO FAILURE

Mission Success Criteria

Diesel Generator and Switchgear Cell Fans

The supply air flow to the various cells is one of the major differences between the LOOP and other upset condition branches. For the LOOP events, supply flow requires three fans to be operational. Since only one fan is operating, this implies that the temperature sensing circuitry to automatically start the additional fans is functional. For all other upset conditions, only one fan is required.

The other condition required for successful cooling is that there is a flow path from the supply fan(s) into the cell and a flow path out of the cell. This requires that the supply dampers, exhaust dampers, exhaust fans, and control circuitry all function correctly. This requirement is considered to be very conservative.

The mission time is 24 hours.

Credit was taken in the PRA for the fourth supply fan for LOOP conditions, even though abnormal operating procedures do not address its use. Credit was also taken for SWGR cell cooling recovery by opening of doors. Because of the limited heat generation in these cells, opening the doors will create enough circulation to keep the cells from overheating.

1. Supply and Recirculation Damper Faults for Diesel Generator Cells 3 or 4

   Faults in the supply and recirculation dampers for DG Cells 3 or 4 will cause overheating of the cells (OP,MT).

2. Operator Failure to Open Switchgear Rooms After an HVAC Failure

   Opening of the Switchgear Room doors after a failure of the HVAC system can create enough circulation to keep the cells from overheating. Operator failure to take this action will cause the cells to overheat (OP).

3. Insufficient HVAC Flow Through Switchgear Rooms E3 and E4 Due to Faults in the Supply or Exhaust Dampers or Fans[1]

Faults in the supply or exhaust dampers or fans will cause insufficient HVAC flow through Switchgear Rooms E3 and E4. (OP, MT).

4. Exhaust Damper Faults for Diesel Generator Cells 3 or 4

Faults in the exhaust dampers for DG Cells 3 or 4 will cause overheating of the cells (OP,MT).

5. Power Not Available to Diesel Generator Cells 3 and 4 Exhaust Fans from MCC DGC and MCC DGD

As in 4 above, no power to the exhaust dampers for DG cells 3 and 4 will cause overheating of the cells (OP, MT).

6. Control Circuit Fails to Provide Actuation Signal to Diesel Generator Cells 3 and 4 Supply and Recirculation Dampers

Failure of the control circuit actuate the DG cell supply and recirculation dampers can cause overheating of the cells (PC, MT).

[1]Note: When the PRA was generated, the Switchgear Room HVAC System was dependent upon the Instrument Air System. Since that time, plant modifications have been made to eliminate this dependency. Therefore, faults in the supply or exhaust dampers or fans should be relatively less probable than the original results. (These faults are shown in the original order of probability.)

BRUNSWICK STEAM ELECTRIC PLANT—UNIT 2
RISK-BASED INSPECTION GUIDE

Diesel Generator and Switchgear Cell Fans

TABLE A.4-2 MODIFIED SYSTEM WALKDOWN

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
Supply Fan CSF-DG for DG Cell 3					Comp. No. DJ7	DG Bldg.–MCC-DGC-El. 23'-DG Cell No.3	Closed	
Supply Fan CSF-DG for DG Cell 3					DJ7 Normal/ Local Selector Switch	DG Bldg.–MCC-DGC-El. 23'-DG Cell No. 3	Normal	
Supply Fan CSF-DG for DG Cell 3					DJ7 Local Control Switch	DG Bldg.–MCC-DGC-El. 23'-DG Cell No. 3	Stop	
Exhaust Fan G-EF-DG for DG Cell 3					Comp. No. DJ6	DG Bldg.–MCC-DGC-El. 23'-DG Cell No. 3	Closed	
Exhaust Fan G-EF-DG for DG Cell 3					DJ6 Normal/ Local Selector Switch	DG Bldg.–MCC-DGC-El. 23'-DG Cell No. 3	Normal	
Exhaust Fan G-EF-DG for DG Cell 3					DJ6 Local Control Switch	DG Bldg.–MCC-DGC-El. 23'-DG Cell No. 3	Stop	
Supply Fan A-SF-DG					Comp. No. DR7	DG Bldg.–MCC-DGA-El. 23'-DG Cell 1	Closed	
					DR7 Normal/ Local Selector Switch	DG Bldg.–MCC-DGA-El. 23'-DG Cell 1	Normal	
					DR7 Local Control Switch	DG Bldg.–MCC-DGA-El. 23'-DG Cell 1	Stop	

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RISK-BASED INSPECTION GUIDE

Diesel Generator and Switchgear Cell Fans

TABLE A.4-2 MODIFIED SYSTEM WALKDOWN (Cont'd)

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
Supply Fan D-SF-DG					Comp. No. D68 Circuit Breaker	DG Bldg.– MCC-DGD-El. 23'-DG Cell No. 4	Closed	
					D68 Normal/ Local Selector Switch	DG Bldg.– MCC-DGD-El. 23'-DG Cell No. 4	Normal	
Supply Fan D-SF-DG					D68 Local Control Switch	DG Bldg.– MCC-DGD-El. 23'-DG Cell No. 4	Stop	
Exhaust Fan H-EF-DG DG Cell 4					Comp. No. D59	DG Bldg.– MCC-DGD-El. 23'-DG Cell No. 4	Closed	
Exhaust Fan H-EF-DG DG Cell 4					D59 Normal/ Local Selector Switch	DG Bldg.– MCC-DGD-El. 23'-DG Cell No. 4	Normal	
Exhaust Fan H-EF-DG DG Cell 4					D59 Local Control Switch	DG Bldg.– MCC-DGD-El. 23'-DG Cell No. 4	Stop	
Exhaust Fan D-EF-DG for 4160V SWGR Rm E-4					D51	DG Bldg.– MCC-DGD-El. 23'-DG Cell No. 4	Closed	

Diesel Generator and Switchgear Cell Fans

TABLE A.4-2 MODIFIED SYSTEM WALKDOWN (Cont'd)

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
Supply Fan B-SF-DG					Comp. No. DZ9	DG Bldg.–MCC-DGB-El. 23'-DG Cell No. 2	Closed	
					DZ9 Normal/ Local Selector Switch	DG Bldg.–MCC-DGB-El. 23'-DG Cell No. 2	Normal	
					DZ9 Local Control Switch	DG Bldg.–MCC-DGB-El. 23'-DG Cell No. 2	Stop	
DG Cell No. 1&3 Damper Control Power					6	Unit 2 Control Bldg. Panel 2-32A-El. 23'	Closed	
DG Cell No. 2&4 Damper Control Power					6	Unit 2 Control Bldg. Panel 2-32B-El. 50'	Closed	

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Diesel Generator and Switchgear Cell Fans

TABLE A.4-2 MODIFIED SYSTEM WALKDOWN (Cont'd)

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
Exhaust Fan C-EF-DG for 4160V SWGR E-3					Comp. No. D05	DG Bldg.–MCC-DGC-El. 23'-DG Cell No. 3*	Closed	
**Loss of Instrument Air to Supply and Recirculation Damper								

*As shown on Figure A.4-1.

**Loss of Instrument air supply can cause damper failure to open; check to be sure instrument air is available at supply damper.

TABLE A.4-2 (Cont'd)

REFERENCE DOCUMENTS

TITLE	I.D. NO.	REV	DATE
System Procedures:			
Diesel Generator Building Heating & Ventilation System	OP-37.4	012	4/21/88
System Description:			
Brunswick PRA Volume I "Heating, Ventilating and Air Conditioning Systems"	Section M.3.4.16		
P&ID's No.:			
Brunswick PRA Volume I "DG and SWGR Cell Cooler Detailed Diagram"	Fig. M.3.4-33		

Standby Liquid Control (SLC) System

Table A.5-1 Importance Basis and Failure Mode Identification

CONDITIONS THAT CAN LEAD TO FAILURE

Mission Success Criteria

The SLC system consists of a storage tank containing a neutron absorbing solution (sodiumpentaborate), two full capacity pumps, a test tank, a drain tank, and associated piping, valves, instrumentation and controls.

The system is placed in operation by manual operation of a control switch on the RTGB. Control switch operation starts the selected pump, causes two explosive type valves to open establishing a flow path from the storage tank to the reactor vessel, and isolates the Reactor Water Cleanup System (RWCU) from the reactor vessel. The solution enters the reactor vessel through a single injection line feeding the poison sparger in the lower plenum of the reactor vessel. Shutdown occurs within one to two hours as the solution is injected.

SLC operation is successful if sufficient sodium pentaborate solution from the storage tank is injected into the reactor vessel through both pump loops. Failure of the SLC occurs if one of the following occurs:

1. Faults in the injection line between the explosive (squib) valves and the reactor vessel.
2. Either pump loop fails.
3. The storage tank or piping to the SLC suction valves fails.
4. The RWCU isolation MOV fails to close.

1. Operator Fails to Actuate the SLC System

Since the SLCS can only be initiated manually, and given the operator reluctance to utilize the system because of the injection of sodium pentaborate into the reactor vessel, failure of the operator to initiate the system is the most likely failure mode (OP).

2. Normally Open Reactor Water Cleanup MOV G31-F004 Fails to Close

Failure of RWCU MOV F004 to close when required will cause flow diversion of the SLC pump discharge (PT,MT).

3. Faults in the SLC Injection Line from the Squib Valves to the Reactor Vessel

Faults in the pipe segment between the squib valves and the reactor vessel such as failure of check valves F006 and F007 to open, will prevent SLC flow to the vessel. This segment is subjected to periodic testing only during refueling outages. (PT,MT).

4. Plugging of Locked Open Manual Valve F001 at the SLC Tank

Plugging of valve F001, which is adjacent to the SLC tank, and is in the single line from the tank to the SLC pumps, prevents all SLC flow (PT, MT).

5. SLC Pumps A or B in Test or Maintenance[1]

Unavailability of the SLC pumps A or B due to test or maintenance prevents SLC flow (PT,MT,TS).

6. SLC Pumps A or B Fail to Start or Run[1]

SLC flow is prevented by failure of the SLC pumps A or B to start or run (PT, MT).

7. SLC Pumps Bypass Relief Valves F029A or F029B Fail to Close[1]

Failure of either SLC pump bypass relief valve F029A or F029B prevents SLC pump flow to the reactor vessel (PT,MT).

[1]Note: When the PRA was generated, it was assumed that enriched sodium pentaborate would be used, thereby requiring only one of two SLC pumps to operate. It has since been decided not to utilize the enriched solution, thereby requiring successful operation of both SLC pumps. This significantly increases the importance of pump unavailability.

BRUNSWICK STEAM ELECTRIC PLANT—UNIT 2
RISK-BASED INSPECTION GUIDE

Standby Liquid Control (SLC) System

TABLE A.5-2 MODIFIED SYSTEM WALKDOWN

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
RWCU Isolation MOV	G31-F004		Open			MCC ZXDB 125/250V DC	On	
SCL Stor. Tank Outlet Isol. Valve (Manual)	C41-F001	Reactor Bldg. SLC Control Station EL. 80'	Locked Open					
SLC Outboard Injection Valve (Manual)	C41-F005	Same	Locked Open					
SLC Inboard Injection Valve	C-41 F008	Inside Drywell EL. 38' AZ 206°	Locked Open					
SLC Pump Control Switch	C-41 CS-S1	Control Room Panel H12 P603	Locked Stop		11 12	120V AC Panel 2AB EL. 23' (Supply Bkr.) Same	On On	

BRUNSWICK STEAM ELECTRIC PLANT—UNIT 2
RISK-BASED INSPECTION GUIDE

Standby Liquid Control (SLC) System

TABLE A.5-2 MODIFIED SYSTEM WALKDOWN (Cont'd)

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
SLC Pump	2A	—	—	—	Comp. EG6	480V MCC 2XG EL. 80'	On	
					2 (HTR)	480V MCC 2XG EL. 80'	On	
SLC Pump	2B	—	—	—	Comp. EK2	480V MCC 2XH EL. 80'	On	
					2 (HTR)	480V MCC 2XH EL. 80'	On	

TABLE A.5-2 (Cont'd)

REFERENCE DOCUMENTS

TITLE	I.D. NO.	REV	DATE
System Procedures:			
Standby Liquid Control System	OP-05	026	9/23/88
System Description:			
Brunswick PRA Volume I	Section M.3.4.9		
"Standby Liquid Control System"			
P&ID's No.:			
Brunswick PRA Volume I	Fig. M.3.4-18		
"SLC Simplified Diagram"			

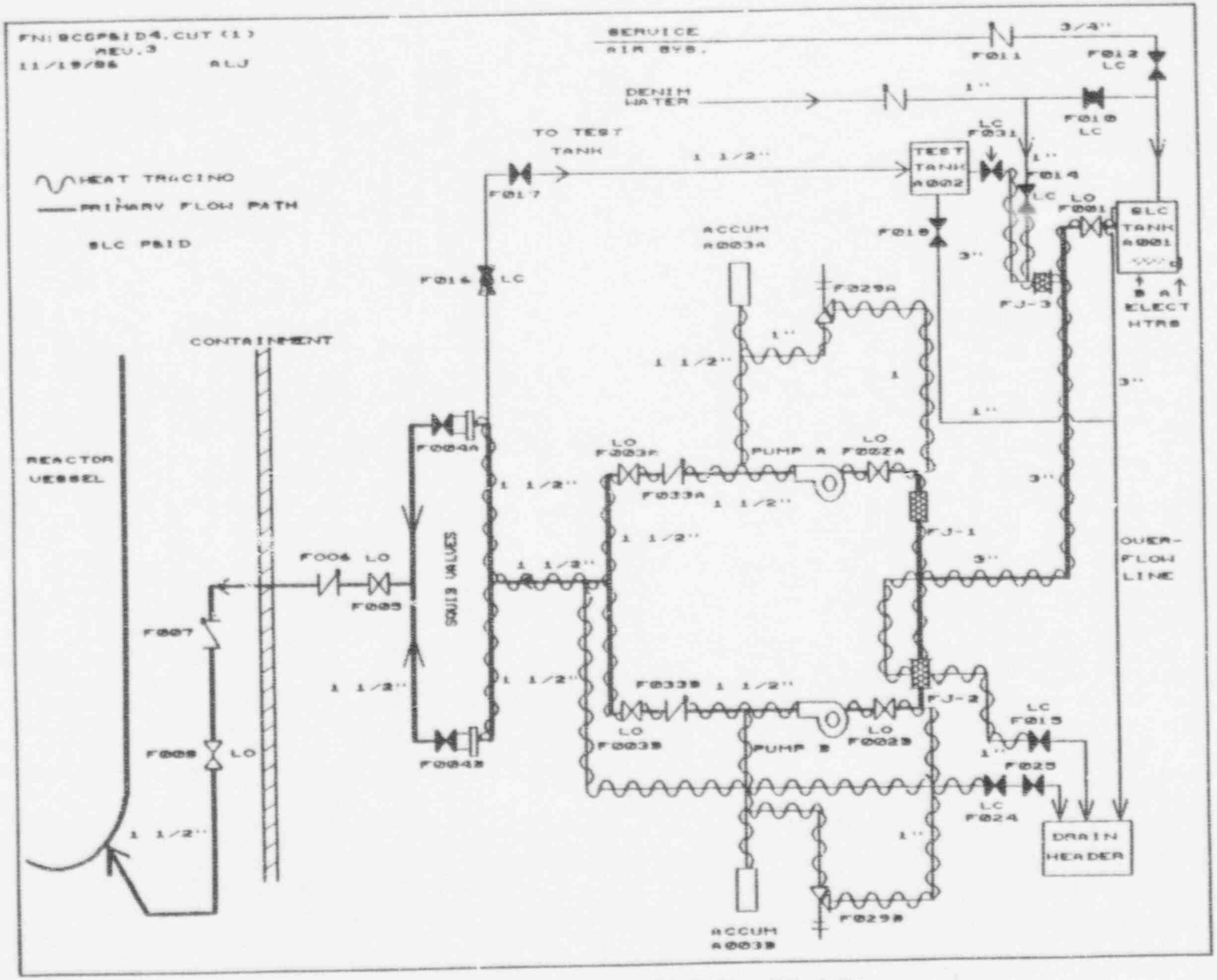

Fig. A.5-1 (PRA Fig. M.3.4-18). SLC Simplified Diagram.
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BRUNSWICK STEAM ELECTRIC PLANT—UNIT 2
RISK-BASED INSPECTION SYSTEM

Reactor Protection System (RPS)

Table A.6-1 Generalized Inspection Plan

Discussion

The Brunswick PRA does not model the Reactor Protection System (RPS) in any detail. RPS electrical failure and mechanical failure on demand were assigned values of 2.0E-5 and 1.0E-5, respectively, i.e., the system was simply treated as a data value. No dominant failure modes are determined. A generic inspection plan is adapted and discussed below.

System Description

The Reactor Protection System initiates protective action when the integrity of the nuclear fuel or the nuclear process barrier is threatened. Two types of protective action are available:

A. Reactor scram
B. Manual select rod insert (Unit 2 only)

The Reactor Protection System (RPS) contains two independent, fail-safe trip systems, each of which controls the continuity or discontinuity of electrical power to one of the two solenoid-operated scram pilot valves associated with each control rod.

The Alternate Rod Injection System provides a path for reactor shutdown which is diverse and independent from the Reactor Protection System. It is common to the normal shutdown components only in the scram inlet and discharge valves, the control rod drives, and the control rods themselves. The automatic signal to initiate the ARI function will come from high reactor vessel pressure or low reactor vessel water level.

ARI System logic uses existing recirculation pump trip initiation instrumentation. The high reactor vessel pressure setpoint will be such that a normal scram should have already been initiated at the time its setpoint is reached, and the low reactor vessel water level setpoint will be set lower than the reactor vessel low water level scram setpoint. The ARI System performs a function redundant to the backup scram system, although the ARI System has a different design basis. The ARI System is not redundant in itself.

Inspection Areas

1. Review and witness RPS function surveillance tests and preventive maintenance.
   - Include witness of partial manual scram test, single rod scram, tests of individual RPS channels, and RPS circuit breaker and motor generator set preventive maintenance.

- References include: NRC R.G. 1.22, "Periodic Testing of Protection System Actuation Function," for RPS and RRCS; R.G. 1.118, "Periodic Testing of Electric Power and Protection Systems," which endorses IEEE Std 338-1977, "Criteria for Periodic Testing of Nuclear Power Generating Station Safety Systems," for ARI only.
- Detailed guidance for review of LPRM and APRM calibration is contained in NRC Inspection Procedures 61703 and 61704.

2. Inspect sensing instrument racks for correct valve configuration, labelling, and separation.

3. Ensure no abnormal RPS alarms in the control room, and verify bypass conditions are properly logged and justified.

4. Check RPS panels for jumpers and lifted leads. Documentation of same with the appropriate review and approval is required.

5. Review post work testing of RPS maintenance tasks.

6. Review calibration records of RPS sensors and compare results to Brunswick technical specifications. Observe trends.

7. Review qualifications and training for technicians performing testing and/or maintenance on the system.

8. Review control rod drive mechanism maintenance inspection procedure and results. Ensure trending of detected wear is performed.

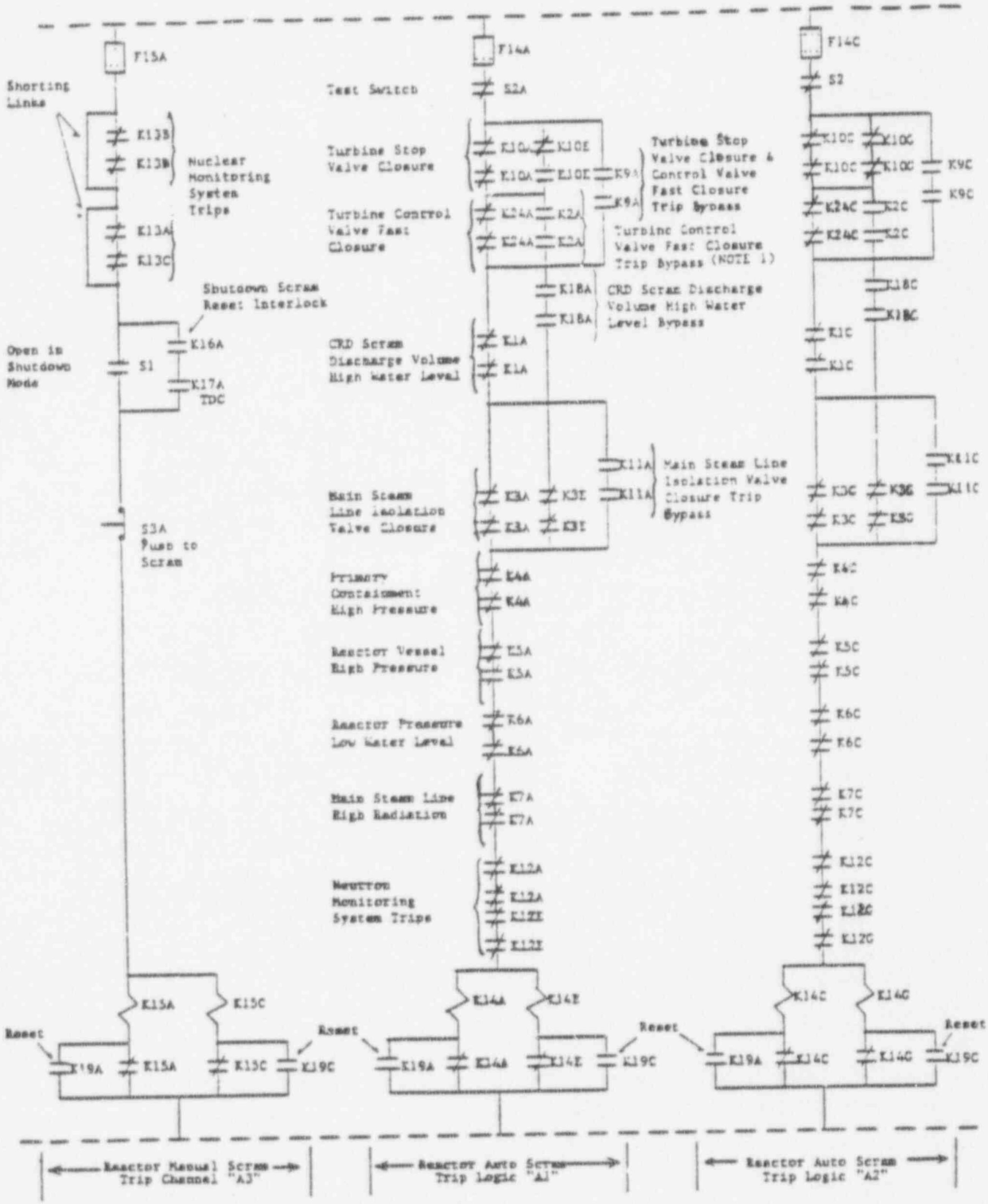
9. Review preventive maintenance practices for solenoid operated valves located in the instrument air header and at the HCU scram inlet and outlet valves.

10. Review surveillance and maintenance of ARI instruments.

TABLE A.6-1 (Cont'd)

REFERENCE DOCUMENTS

TITLE	I.D. NO.	REV	DATE
System Description:			
Unit O-Reactor Protection System Including	SD-03	014	8/29/88
Alternate Rod Injection System	Volume II		



NOTE 1: K2 contacts for Turbine Control Valve Fast Closure Trip Bypass are in Unit No. 2 circuitry only.

Contacts shown in normal condition.

Fig. A.6-1. Reactor Protection System—Schematic Diagram of Logics in One Trip System (SD-03, Rev. 14, Fig. 3-1).

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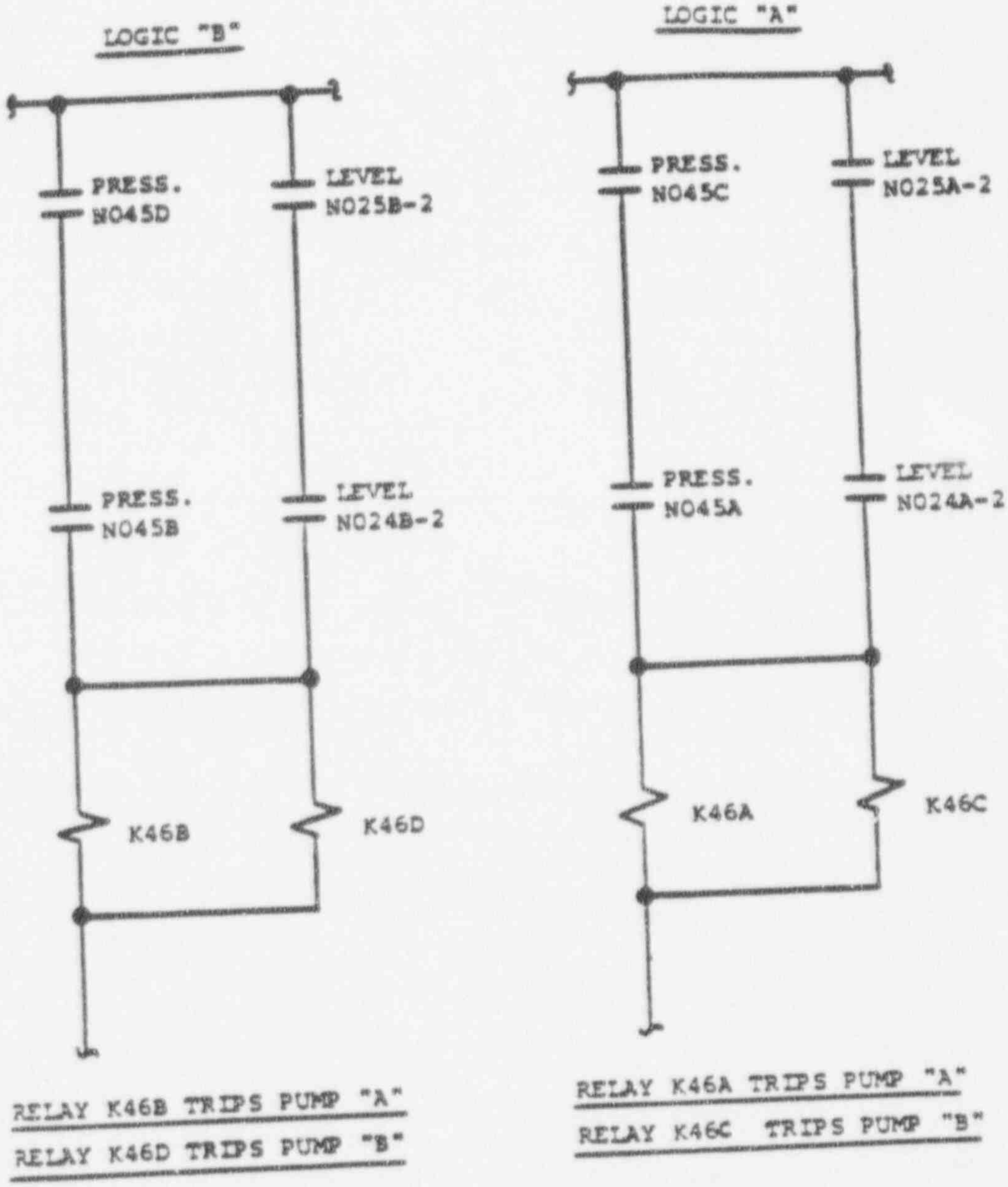

Fig. A.6-2. Recirculation Pump Trip (ATWS)/Alternate Rod Injection Logic (SD-03, Rev. 14, Fig. 3-11).

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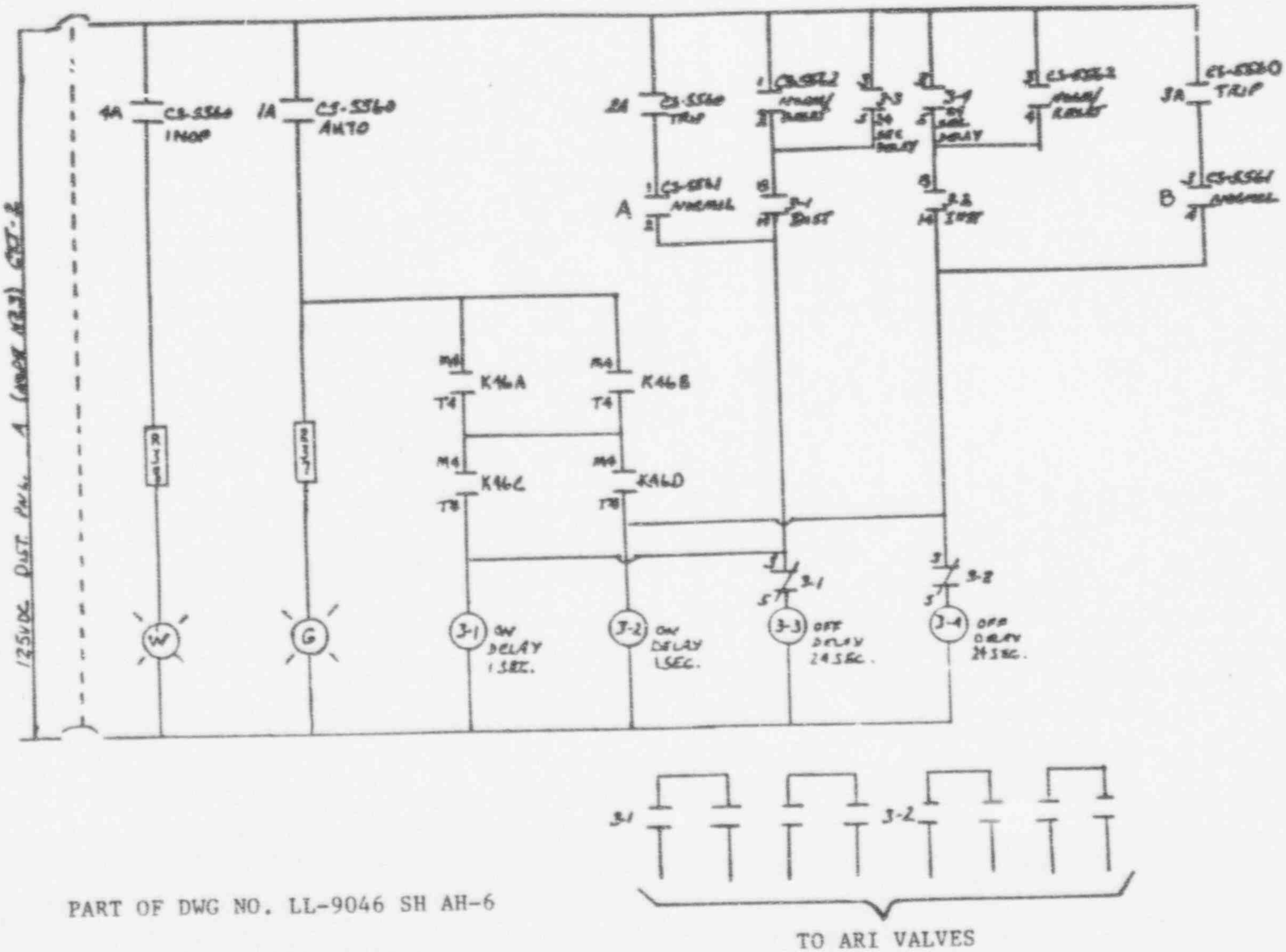

Fig. A.6-3. Alternate Rod Injection Logic (SD-03, Rev. 14, Fig. 3-12).
CAUTION: This is NOT a controlled document.

Service Water System (SWS)

Table A.7-1 Importance Basis and Failure Mode Identification

CONDITIONS THAT CAN LEAD TO FAILURE

Mission Success Criteria

The Service Water System (SWS) provides water from the Cape Fear River for cooling and lubrication of equipment in the Reactor Building, Diesel Generator Building, Turbine Building, and to the circulating water pumps. In the BSEP PRA, the SWS also covers the screen wash system which filters the intake water to the SWS pumps.

The Service Water System is subdivided into two major headers: one for nuclear or vital equipment in the Reactor Building and the other normally supplying conventional equipment in the Turbine Building. The two headers are normally operated independently, with each consisting of a group of service water pumps, parallel loads, and interconnecting headers. Cross-connect valves allow the conventional pumps to supply the nuclear equipment as conditions dictate.

Service water is provided from the Cape Fear River through four sets of motor-operated traveling screens into the SWS pump suction bay. This bay is common to both Units 1 and 2. Five vertical shaft pumps, located at the intake structure, take suction from the bay and discharge to the nuclear and conventional service water headers. Two pumps are aligned to the nuclear header. The remaining three pumps normally supply the conventional header but may also be aligned to the nuclear header.

In addition to the normal service water pumps, there are four additional pumps, the RHR Service Water Pumps. These are specifically used to increase the pressure of the RHR service water for the RHR heat exchangers to ensure that any leaks in the heat exchanger would not result in a radioactive release to the environment.

One operational screen is sufficient to supply to the service water intake bay to support operation of two SWS pumps per unit.

Two SWS pumps are necessary to supply sufficient water to the various loads. The FSAR (Section 9.2, page 9.2.1-5) states that, during the first 10 minutes of the design basis accident, only one pump is required but that a second pump is required later to support the RHR heat exchangers. Thus, it was assumed that two pumps would be the success criteria for satisfactory operation. This is a conservative value since some of the transients may require fewer than two pumps.

A bypass switch has been added in the Control Room to allow the operators to bypass the interlock requiring that one of two RHR SWS pumps must start in order for the discharge valves to open.

To ensure sufficient flow to the emergency equipment, it is assumed that the RBCCW heat exchanger must be isolated from the nuclear header and that the TBCCW heat exchangers must be isolated from the conventional header.

Due to the two-pump criteria and isolation of the TBCCW, the flow for the chlorination system was an assumed on-line diversion. Thus, flow through this path is already accounted for in the two-pump criteria.

A 24-hour mission time is assumed for the SWS.

1. Dependent, or Common Cause, Failure—4 or More SWS Pumps Fail to Run

Four or more SWS pumps failing to run will cause insufficient SWS flow (OP,PT,MT,TS)

2. Loop A or Loop B RHR Heat Exchanger is Unavailable Due to Test or Maintenance

The RHR heat exchangers 2A and 2B are cooled by separate RHR Service Water pumps. Unavailability of the pumps or heat exchangers due to test or maintenance prevents the suppression pool cooling and shutdown cooling modes of RHR operation (PT,MT,TS).

3. Loop A or Loop B RHR Heat Exchanger Fails Due to Plugging

Plugging of the RHR heat exchangers has occurred in the past at BSEP due to problems with the chlorination system (OP,PT,MT).

4. Diesel Generator 1,2,3 or 4 Heat Exchanger is Unavailable Due to Test or Maintenance

Unavailability of the DG 1,2,3 or 4 heat exchangers due to test or maintenance can cause loss of the diesels should they begin to operate under a loss of off-site power condition (PT,MT,TS).

5. Diesel Generator 1,2,3 or 4 Heat Exchanger is Unavailable Due to Plugging

Similar to the above, unavailability of the DG 1,2,3 or 4 heat exchangers due to plugging can cause a loss of the diesels should they begin to operate under a loss of off-site power condition (PT,MT).

6. Normally Closed RHR Heat Exchanger Suction header Isolation MOVs V105 and V101 Fail to Open

Failure of the normally closed RHR heat exchanger isolation MOVs V105 and V101 to open will prevent operation of the RHR heat exchangers (PT,MT).

7. <u>Insufficient Flow Through Screens 1A, 1B, 2A and 2B Due to Failure of Screen Drives or Other Faults</u>

Insufficient flow through screens 1A, 1B, 2A and 2B due to failure of the screen drives or other faults will cause loss of the Service Water pumps (OP, MT).

8. <u>Conventional Pump 2A, 2B or 2C Unavailable Due to Test or Maintenance</u>

In the BSEP PRA, it was assumed that any one Conventional SW Pump would be automatically started and aligned to the conventional header or that it could also be manually aligned to the nuclear header and then started. Therefore, unavailability of the Conventional Pump aligned to the nuclear header due to test or maintenance will prevent it from discharging to that header when required (OP,PT,MT,TS).

BRUNSWICK STEAM ELECTRIC PLANT—UNIT 2
RISK-BASED INSPECTION GUIDE

Service Water System (SWS)

TABLE A.7-2 MODIFIED SYSTEM WALKDOWN

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
RHR Heat Exchanger MOV Fails	V105	Control Room Panel 601	Closed		Comp. DM1	Reactor Bldg. 480V MCC 2XB-El. 20'-South	On	
					Comp. DM1 Normal Key Lock Switch	Reactor Bldg. 480V MCC 2XB-El. 20'-South	Normal	
					9 Motor Heater	Reactor Bldg. MCC 2XB-120V AC Dist. Pnl. HN9	On	
RHR Heat Exchanger MOV Fails	V101	Control Room Panel 601	Closed		Comp. DH5	Reactor Bldg. -480V MCC 2XA-El. 20'-North	On	
					23 Valve Motor Htr.	Reactor Bldg. MCC 2XA 120V AC Dist. Pnl. HS3	On	

Service Water System (SWS)

TABLE A.7-2 MODIFIED SYSTEM WALKDOWN (Cont'd)

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
Conventional Service Water Pump	2A				Comp. No. AJ4	DG Bldg. 4160V Emerg Bus E3-4KV Cell 3	Racked In	
					AJ4 Local Key Lock Switch	DG Bldg. 4160V Emerg Bus E3-4KV Cell 3	Normal	
					AJ4 Motor Htr.	DG Bldg. 4160V Emerg Bus E3-4KV Cell 3	On	
					AJ4 Elapsed Time Meter	DG Bldg. 4160V Emerg Bus E3-4KV Cell 3	On	
Conventional Service Water Pump	2B				Comp. No. AL2	DG Bldg. 4160V Emerg Bus E4-KV Cell 4	Racked In	
					AL2 Motor Htr.	DG Bldg. 4160V Emerg Bus E4-KV Cell 4	On	
					AL2 Elapsed Time Meter	DG Bldg. 4160V Emerg Bus E4-KV Cell 4	On	
Conventional Service Water Pump	2C				Comp. No. AF6	DG Bldg. 4160V Emerg Bus E1-4KV Cell 1	Racked In	
					AF6	DG Bldg. 4160V Emerg Bus E1-4KV Cell 1	On	
					AF6 Elapsed Time Meter	DG Bldg. 4160V Emerg Bus E1-4KV Cell 1	On	

Service Water System (SWS)

TABLE A.7-2 MODIFIED SYSTEM WALKDOWN (Cont'd)

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
Nuclear Service Water Pump	2A				Comp. No. AJ3	DG Bldg. 4160V Emerg Bus E3-4KV Cell 3	Racked In	
					AJ3 Motor Htr.	DG Bldg. 4160V Emerg Bus E3-4KV Cell 3	On	
					AJ3 Local Key Lock Switch	DG Bldg. 4160V Emerg Bus E3-4KV Cell 3	Normal	
					AJ3 Elapsed Time Meter	DG Bldg. 4160V Emerg Bus E3-4KV Cell 3	On	
Nuclear Service Water Pump	2B				Comp. No. AL1	DG Bldg. 4160V Emerg Bus E4-KV Cell 4	Racked In	
					AL1 Motor Htr.	DG Bldg. 4160V Emerg Bus E4-KV Cell 4	On	
					AL1 Local Key Lock Switch	DG Bldg. 4160V Emerg Bus E4-KV Cell 4	Normal	
					AL1 Elapsed Time Meter	DG Bldg. 4160V Emerg Bus E4-KV Cell 4	On	
*								

*Assume that screens 1A, 1B, 2A and 2B are operable.

TABLE A.7-2 (Cont'd)

REFERENCE DOCUMENTS

TITLE	I.D. NO.	REV	DATE
System Procedures:			
Service Water System	OP-43	057	3/9/88
System Description:			
Brunswick PRA Volume I	Section M.3.4.14		
"Service Water System"			
P&ID's No.:			
Brunswick PRA Volume I	Fig. M.3.4-27		
"SWS Simplified Diagram"	Sheets 1–5		

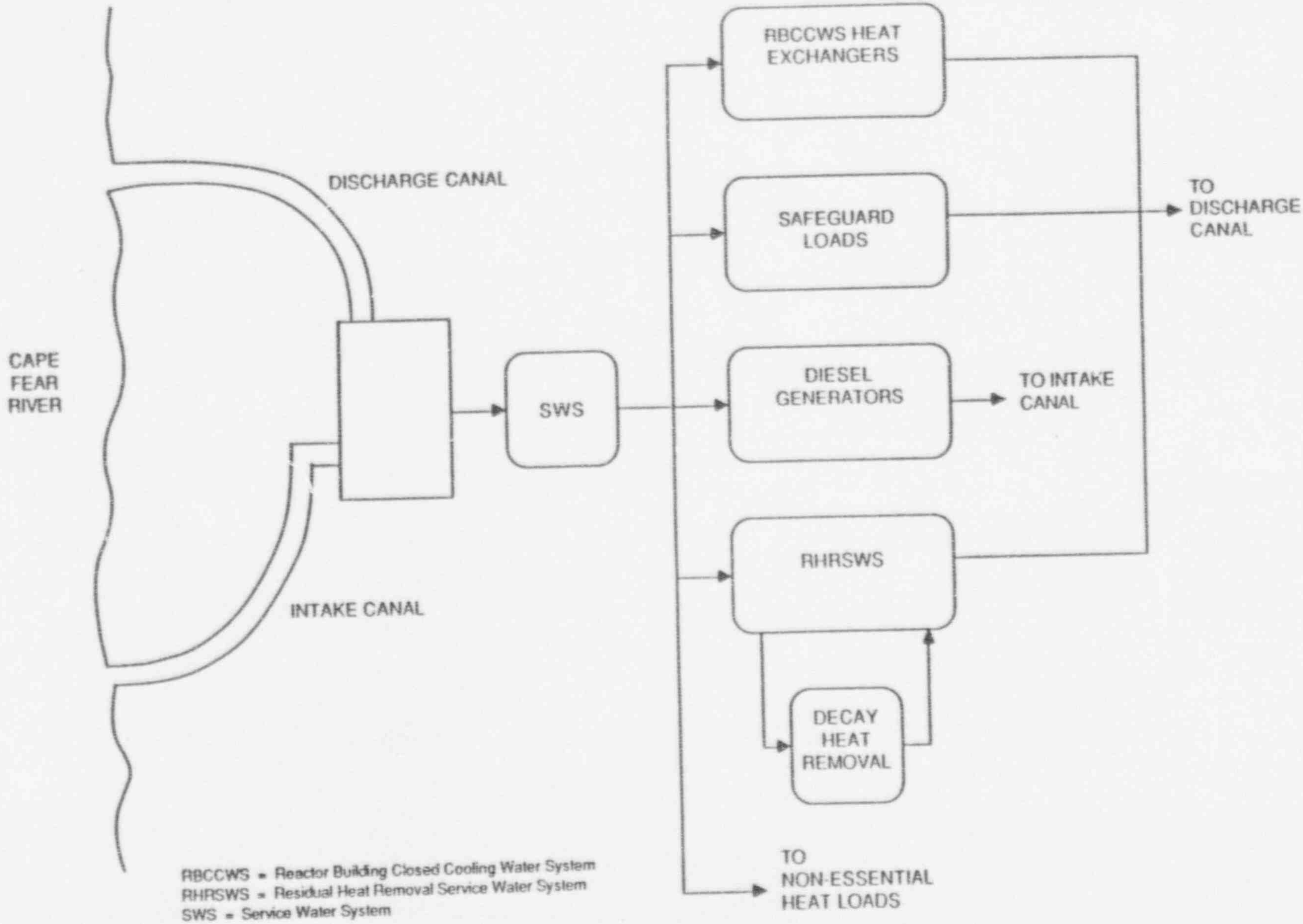

Fig. A.7-1 (SAIC 89/1011 Figure 3-1). Cooling Water Systems Functional Diagram for Brunswick 1 and 2

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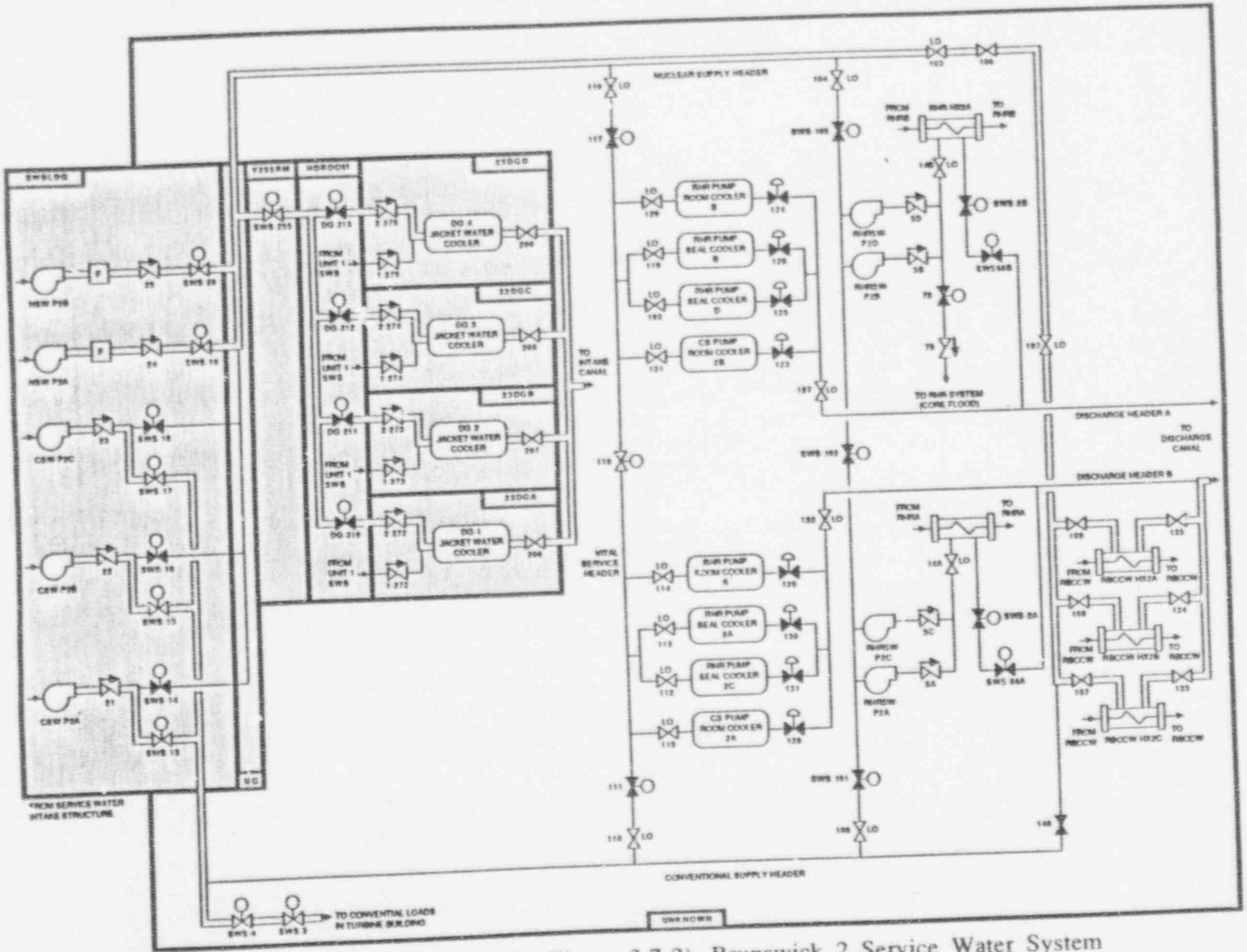

Fig. A.7-2 (SAIC 89/1011 Figure 3.7-2). Brunswick 2 Service Water System Showing Component Locations

CAUTION: This is NOT a controlled document.

Residual Heat Removal (RHR) System

Table A.8-1 Importance Basis and Failure Mode Identification

CONDITIONS THAT CAN LEAD TO FAILURE

Mission Success Criteria

The RHR system is a closed loop system of piping, valves, pumps, and heat exchangers designed to protect the reactor from overheating. The system is designed to perform three primary functions which comprise the following operating modes:

1. Low Pressure Coolant Injection (LPCI),
2. Containment cooling, and
3. Shutdown Cooling (SDC).

In addition to these three functions, the RHR system may be used to assist in fuel pool cooling and pumping water from the reactor or suppression pool to the Radwaste System. The containment cooling mode is subdivided into the Suppression Pool Cooling (SPC) mode and the Containment Spray (CNS) mode.

The system consists of two essentially complete and independent loops, identified as loop A and loop B, as shown in Figure M.3.4-12. Each loop is comprised of two pumps, piping, valves, a heat exchanger, and associated instrumentation and controls. Each loop is piped and valved flexibly to permit the system to carry out its multiple functions.

The source of pumpage can either be from the suppression pool or from reactor recirculation loop A suction header. The discharge point can be any of the following:

1. suppression pool via the RHR full flow test line through valves E11-F028A(B) and F024A(B),
2. suppression pool via the spray header through valves E11-F028A(B) and F024A(B),
3. drywell via containment spray header through valves E11-F016A(B) and F021A(B), or
4. reactor vessel via the reactor recirculation loop discharge header through valves E11-F017A(B) and F015A(B).

The RHR cross-tie valve (E11-F010) is maintained closed during normal operation.

LPCI Success/Failure Criteria. Success of the RHR in the LPCI mode is defined as at least one pump delivering rated flow to the reactor vessel with suction from the suppression pool. For large LOCAs where the break is in a recirculation loop, at least one pump must

discharge through the unbroken recirculation loop. For all other initiators the discharge of any one of the four RHR pumps to either recirculation loop is sufficient for success.

DHA Success/Failure Criteria. The DHA function is the removal of decay heat to the SWS system (and thence to the environment following a large or intermediate LOCA inside containment. It is identical to the LPCI mode except that MOV F048A (B) is closed. This arrangement of the RHRS is also expected to be used for transients and small LOCAs inside containment where high pressure injection systems have failed, reactor vessel has been depressurized, and LPCI has been actuated successfully. For a large or intermediate LOCA inside containment, the return path from the RPV to the suppression pool is through the break. For the transients, the return path is through the open ADS or SRV valves that were opened to depressurize the RPV. For DHA success, one pump must be able to inject cooled water into the RPV. Note that DHA is similar to alternate shutdown cooling except that it is not necessary to open one or more SRVs since the break or the valves used to depressurize the vessel provide the return path.

SPC Success/Failure Criteria. Success of the RHR in the SPC mode is considered to be success of one pump in one loop. Suction is from the suppression pool and discharge is to the suppression pool via either the test return line or the torus spray header. The SPC mode requires manual actuation by operators in the control room. If the RPV pressure has been lowered enough for automatic actuation of LPCI to occur, then it is assumed that the DHA operating mode applies for the long-term removal of decay heat. Therefore the SPC mode applies only to those cases following transients and small LOCAs inside containment where the HPCI or another high pressure system has been successful in supplying water to the core.

The SPC mode transfers heat from the containment to the SWS (and thus to the environment). Thus cooling via the heat exchangers is required.

It should be noted that for some ATWS events, the normal suppression pool cooling will not be sufficient to control temperature and pressure within allowable limits. For those events, it is necessary to control water level at the top of the active fuel to limit total heat production.

CNS Success/Failure Criteria. Success of the RHR in the CNS mode is considered to be the success of one pump in one loop. Suction is from the suppression pool and discharge must be to both the torus spray header and the drywell spray header. Except for the discharge path, the CNS mode is similar to the SPC mode.

SDC Success/Failure Criteria. In the SDC mode, suction is from the RPV via the suction portion of recirculation loop A through MOVs F008, F009, and F006A(B,C,D). Discharge is to the discharge portion of the appropriate recirculation loop through MOVs F015A(B) and F017A(B). The suppression pool suction valves F020A(B) and F005A(B,C,D) are closed for the loop operating in the SDC mode. The breakers for F008, F009, and F006A(B,C,D) are normally racked out, so these must be restored before the valve positions can be changed. The failure of either F008 or F009 to open are obvious single faults for both loops of the RHR in the SDC mode.

1. RHR Loop A or Loop B Unavailable Due to Test or Maintenance

Combined with a failure in the opposite loop, test or maintenance of either RHR loop will cause loss of LPCI and RHR functions (PT,MT).

2. RHR Heat Exchanger Bypass MOVs F048A and F048B Fail to Close When Required

Failure of RHR heat exchanger bypass MOVs F048A and F048 to close when required will prevent cooling of the suppression pool and the reactor vessel during the suppression pool cooling and shutdown cooling modes (PT,MT,OP).

3. RHR Mini-Flow Recirculation Line MOVs F007A and F007B Fail to Close When Required

Failure of the RHR mini-flow recirculation line MOVs F007A and F007B to close when required will cause insufficient flow from the LPCI/RHR pumps during the various modes of operation (PT,MT).

4. Instrumentation and Control for Mini-Flow Recirculation Line MOVs F007A and F007B Not Restored or PDIS N021A and N021B Miscalibrated

Failure to properly restore the I&C for MOVs F007A and F007B, or miscalibration of flow switch PDIS-N021A/B located in the discharge line of the two pumps per loop, will prevent proper automatic operation of the RHR pumps minimum flow bypass valves F007A and F007B. This can cause loss of the RHR pumps (PC,PT,MT).

5. Faults in Pressure Transmitters for MOVs F007A and F007B

As in 3 and 4 above, failure of the instrumentation controlling MOVs F007A and F007B, such as pressure transmitter faults, will prevent proper automatic operation of the RHR pumps minimum flow bypass and cause possible loss of the pumps (PC,PT,MT).

6. Suppression Pool Strainers S1 and S2 Fail Due to Plugging

Plugging of the suppression pool strainers S1 and S2 will prevent cooling of the pool by the RHR pumps (OP,PT,MT). Consequently, it will also prevent reactor core flooding and cooling as well.

7. RHR Injection Header to Suppression Pool Isolation MOVs F028A and F028B, F027A and F027B, or F024A and F024B and Fail to Open or Fail to Close When Required

Failure of the normally closed suppression pool injection header isolation MOVs F028A and F028B, F027A and F027B, or F024A and F024B, to open when required will prevent cooling and level control of the suppression pool. Failure of the valves to close when required will divert flow from the LPCI injection lines to the suppression pool (PT,MT).

8. Loops A and B RHR Pumps C002A,B,C and D Fail to Start or Run

Dependent, or common cause, failure of the four RHR pumps C002A,B,C, and D to start and run fails all RHR modes of operation (PT,MT). This is a significant risk contributor according to the PRA.

9. Common Cause Failure of RHR Heat Exchangers Due to Plugging

The BSEP has experienced plugging and rupture of an RHR heat exchanger several years past. If one exchanger is failed, the other is very likely to be fouled also. These events are caused by failure of the service water chlorination system leading in turn to the biological fouling (OP,PT,MT).

10. Operator Fails to Correctly Initiate Suppression Pool Cooling-Loops A and B

Suppression pool cooling is manually aligned when the pool upper temperature or level limits are reached. Failure of the operator to correctly initiate cooling will prevent this mode of operation (OP).

Residual Heat Removal (RHR) System

TABLE A.8-2 MODIFIED SYSTEM WALKDOWN

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
RHR Heat Exchanger MOV	F048A	Control Room Panel H12-P601	Open*		7 Motor Htr. Breaker	Reactor Bldg. -480V MCC 2XA (Rear) 120V AC Dist. Pnl. HL3 El. 20' North	On	
Control Switch (Valve)					Comp. No. DG2	Reactor Bldg. 480V MCC-2XA (Rear) El. 20' North	On	
RHR Heat Exchanger MOV	F048B	Control Room Panel H12-P601	Open		Comp No. DM8	Reactor Bldg. 480V MCC-2XB (Rear) El. 20' South	On	
					DM8 Normal/ Local Cont. Sw.	Reactor Bldg. 480V MCC-2XB (Rear) El. 20' South	Normal	
Control Switch (Valve)					18 Motor Htr. Circuit Breaker	Reactor Bldg. 480V MCC-2XB (Rear) 120V AC Dist. Pnl. HL4 El. 20' South	On	
RHR Min.-Flow Recirculation Line MOV	F007B	Control Room Panel H12-P601	Closed		Comp. No. DL3	Reactor Bldg. 480V MCC-2XB (Rear) El. 20' South	On	
					Comp. No. DL3 Local Key Lock Switch	Reactor Bldg. 480V MCC-2XB (Rear) El. 20' South	Normal	
Control Switch (Valve)					6 Motor Htr. Circuit Breaker	Reactor Bldg. 480V MCC 2XB (Rear) 120V AC Dist. Pnl. HL4 El. 20' South	On	

*Normal position depends on operating mode. Concern is "failure to close" from open position as shown in LPCI mode.

Residual Heat Removal (RHR) System

TABLE A.8-2 MODIFIED SYSTEM WALKDOWN (Cont'd)

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
RHR Min.-Flow Recircula-tion Line MOV	F007A	Control Room Panel H12-P601	Closed		16 Motor Htr. Breaker	Reactor Bldg. 480V MCC-2XA (Front) 120V AC Dist. Pnl. HS3 El. 20' North	On	
Control Sw. (Valve F007A)					Comp. No. DF1	480V MCC-2XA (Front) El. 20' North	On	
*Suppression Pool Strainer	S2		Clean					
*Suppression Pool Strainer	S1		Clean					

*Inspect strainer for possible clogging when suppression pool is drained.

Residual Heat Removal (RHR) System

TABLE A.8-2 MODIFIED SYSTEM WALKDOWN (Cont'd)

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
RHR Injection Header to Suppression Pool Isolation MOV	F028A	Control Room Panel H12-P601	Closed		Comp. No. DG0	Reactor Bldg. 480V MCC-2XA-2 (Front) El. 20' North	On	
Control Switch (Valve)	F028A	HPCI Roof/ NRHR Valve Location			6	Reactor Bldg. 480V MCC 2XA (Front) 120V AC Dist. Pnl. HS3 El. 20' North	On	
RHR Injection Header to Suppression Pool Isolation MCV	F028B	Control Room Panel H12-P601	Closed		Comp. No. DM5	480V MCC-2XB-2 (Front) El. 20' South	On	
Control Switch (Valve)	F028B				DM5 Local Control Switch	480V MCC 2XB-2 (Front) El. 20' South	Normal	
					16 Motor Htr. Breaker	Reactor Bldg. 480V MCC 2XB (Rear) 120V AC Dist. Pnl. HL4 El. 20' South	On	

Residual Heat Removal (RHR) System

TABLE A.8-2 MODIFIED SYSTEM WALKDOWN (Cont'd)

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
Suppression Pool Cooling Injection Line MOV	F024B	Control Room Panel H12-P601	Closed		Comp. No. DM2	480V MCC 2XB (Rear) El. 20' South	On	
Control Switch (Valve)	D				DM2 Norm./ Local Control Switch	480V MCC 2XB (Rear) El. 20' South	Normal	
					13 Motor Htr. Circuit Breaker	Reactor Bldg. 480V MCC 2XB (Rear) 120V AC Dist. Pnl. HL4 El. 20' South	On	
Suppression Pool Cooling Injection Line MOV	F024A	Control Room Panel H12-P601	Closed		Comp. No. DF7	480V MCC 2XA (Rear) El. 20' North	On	
Control Switch (Valve)		HPCI Roof NRHR-El.2'			3 Motor Htr. Circuit Breaker	480V MCC 2XA (Rear) 120V AC Dist. Pnl. HL3 El. 20' North	On	
Suppression Pool Cooling Injection Line MOV	F027B	Control Room Panel H12-P601	Closed		Comp. No. DM4	Reactor Bldg. 480V MCC 2XB (Rear) El. 20' South	On	
Control Switch (Valve)					15 Motor Htr. Circuit Breaker	Reactor Bldg. 480V MCC 2XB (Rear) 120V AC Dist. Pnl. HL4 El. 20' South	On	

Residual Heat Removal (RHR) System

TABLE A.8-2 MODIFIED SYSTEM WALKDOWN (Cont'd)

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
Suppression Pool Cooling Injection Line MOV Control Switch	F027A	Control Room Panel H12-P601	Closed		Comp. No. DF9	Reactor Bldg. 480V MCC 2XA (Rear) El. 20' North	On	
					4 Motor Htr. Circuit Breaker	Reactor Bldg. 480V MCC 2XA (Rear) 120V AC Dist. Pnl. HL3 El. 20' North	On	
Loop A RHR Pump	C002A				AJ1 Motor Bkr.	Loop A DG Bldg. 4160V Emerg. Bus E3 El. 45' Cell 3	Racked In	
					AJ1 Motor Htr. Circuit Breaker	Loop A DG Bldg. 4160V Emerg. Bus E3 El. 45' Cell 3	On	
					AJ1 Elapsed Time Circuit Breaker	Loop A DG Bldg. 4160V Emerg. Bus E3 El. 45' Cell 3	On	
Loop A RHR Pump	C002C				AF5 Motor Bkr.	Loop A DG Bldg. 4160V Emerg. Bus El. El. 45' Cell 1	Racked In	
					AF5 Motor Htr. Circuit Breaker	Loop A DG Bldg. 4160V Emerg. Bus El. El. 45' Cell 1	On	
					AF5 Elapsed Time Circuit Breaker	Loop A DG Bldg. 4160V Emerg. Bus El. El. 45' Cell 1	On	

BRUNSWICK STEAM ELECTRIC PLANT—UNIT 2
RISK-BASED INSPECTION GUIDE

Residual Heat Removal (RHR) System

TABLE A.8-2 MODIFIED SYSTEM WALKDOWN (Cont'd)

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
Loop B RHR Pump	C002B				AK3 Motor Circuit Breaker	Loop B DG Bldg. 4160 Emerg. Bus E4 El. 45' Cell 4	Racked In	
					AK3 Motor Htr. Circuit Breaker	Loop B DG Bldg. 4160 Emerg. Bus E4 El. 45' Cell 4	On	
					AK3 Elapsed Time Circuit Breaker	Loop B DG Bldg. 4160 Emerg. Bus E4 El. 45' Cell 4	On	
					AK3 Normal/ Local Control Switch	Loop B DG Bldg. 4160 Emerg. Bus E4 El 45' Cell 4	Normal	
Loop B RHR Pump	C002D				AG9 Motor Circuit Breaker	Loop B DG Bldg. 4160V Emerg. Bus E2 El. 45' Cell 2	Racked In	
					AG9 Motor Htr. Circuit Breaker	Loop B DG Bldg. 4160V Emerg. Bus E2 El. 45' Cell 2	On	
					AG9 Elapsed Time Circuit Breaker	Loop B DG Bldg. 4160V Emerg. Bus E2 El. 45' Cell 2	On	
					AG9 Normal/ Local Control Switch	Loop B DG Bldg. 4160V Emerg. Bus E2 El. 45' Cell 2	Normal	

Residual Heat Removal (RHR) System

TABLE A.8-2 MODIFIED SYSTEM WALKDOWN (Cont'd)

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
Low Pressure Instrument Isol. Valve	PDIS-N021B-4	Next to Instrument Rack H21-P021	Open					
High Pressure Instrument Isol. Valve	PDIS-N021B-3	Next to Instrument Rack H21-P021	Open					
High Pressure Instrument Test Valve	PDIS-N021B-10	Next to Instrument Rack H21-P021	Closed					
Low Pressure Instrument Test Valve	PDIS-N021B-11	Next to Instrument Rack H21-P021	Closed					
High Pressure Instrument Isolation Valve	PDIS-N021A-3	Next to Instrument Rack H12-P018	Open					
Low Pressure Instrument Isolation Valve	PDIS-N021A-4	Next to Instrument Rack H12-P018	Open					

Residual Heat Removal (RHR) System

TABLE A.8-2 MODIFIED SYSTEM WALKDOWN (Cont'd)

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
High Pressure Instrument Test Valve	PDIS-N021A-10	Next to Instrument Rack H12-P018	Closed					
Low Pressure Instrument Test Valve	PDIS-N021A-11	Next to Instrument Rack H12-P018	Closed					

Note: Pressure transmitters for MOV F007A and B should be visually inspected.

TABLE A.8-2 (Cont'd)

REFERENCE DOCUMENTS

TITLE	I.D. NO.	REV	DATE
System Procedures:			
Residual Heat Removal System	OP-17	078	9/27/88
System Description:			
Brunswick PRA Volume I	Section M.3.4.8		
"Residual Heat Removal System"			
P&ID's No.:			
Brunswick PRA Volume I	Figs. M.3.4-13 to		
"RHR (LPCI) Simplified Diagram"	M.3.4-16		
"RHR (SPC) Simplified Diagram"			
"RHR (CNS) Simplified Diagram"			
"RHR (SDC) Simplified Diagram"			

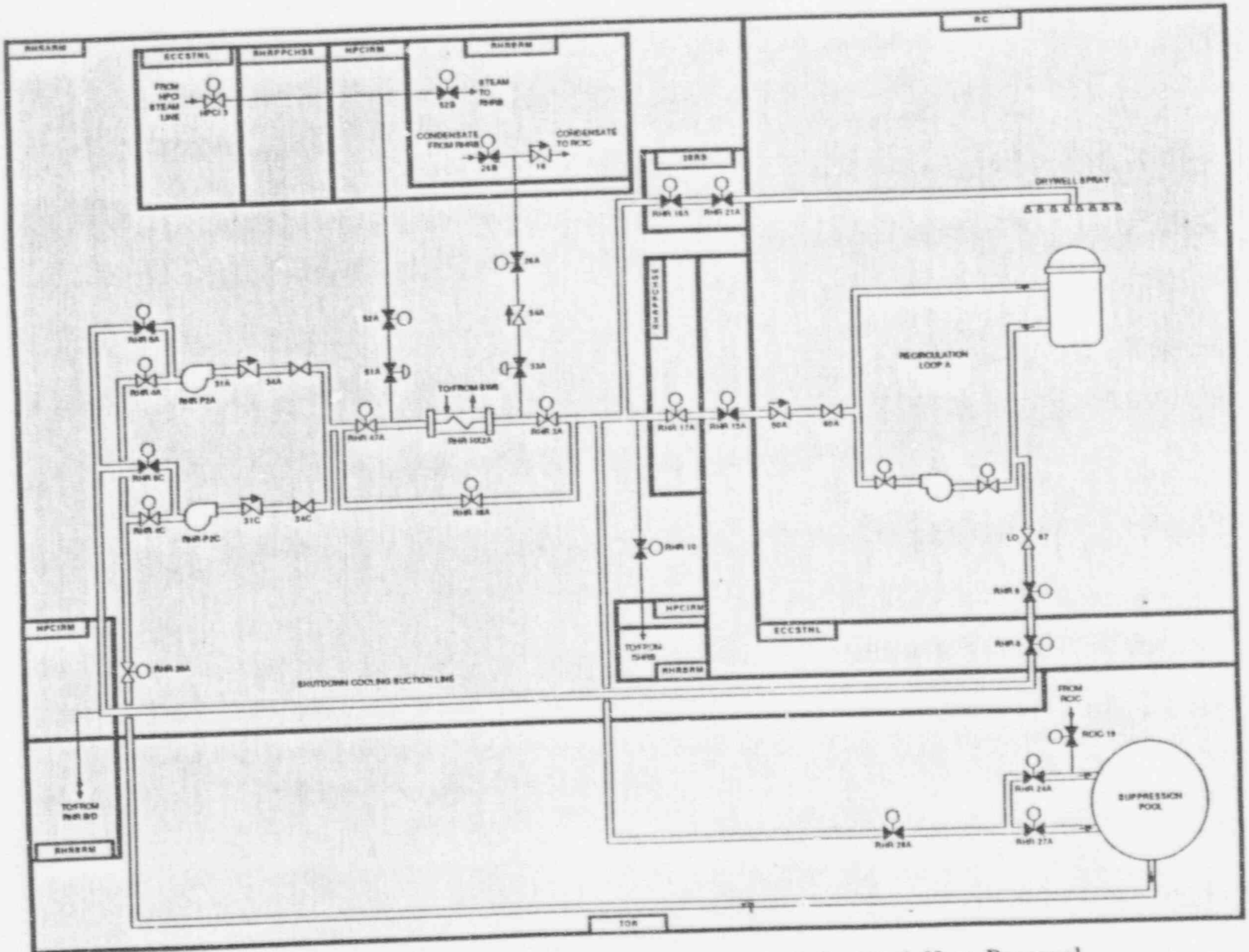

Fig. A.8-1 (SAIC 89/1011 Figure 3.3-6). Brunswick 2 Residual Heat Removal System, Loops A and C, Showing Component Locations
CAUTION: This is NOT a controlled document.

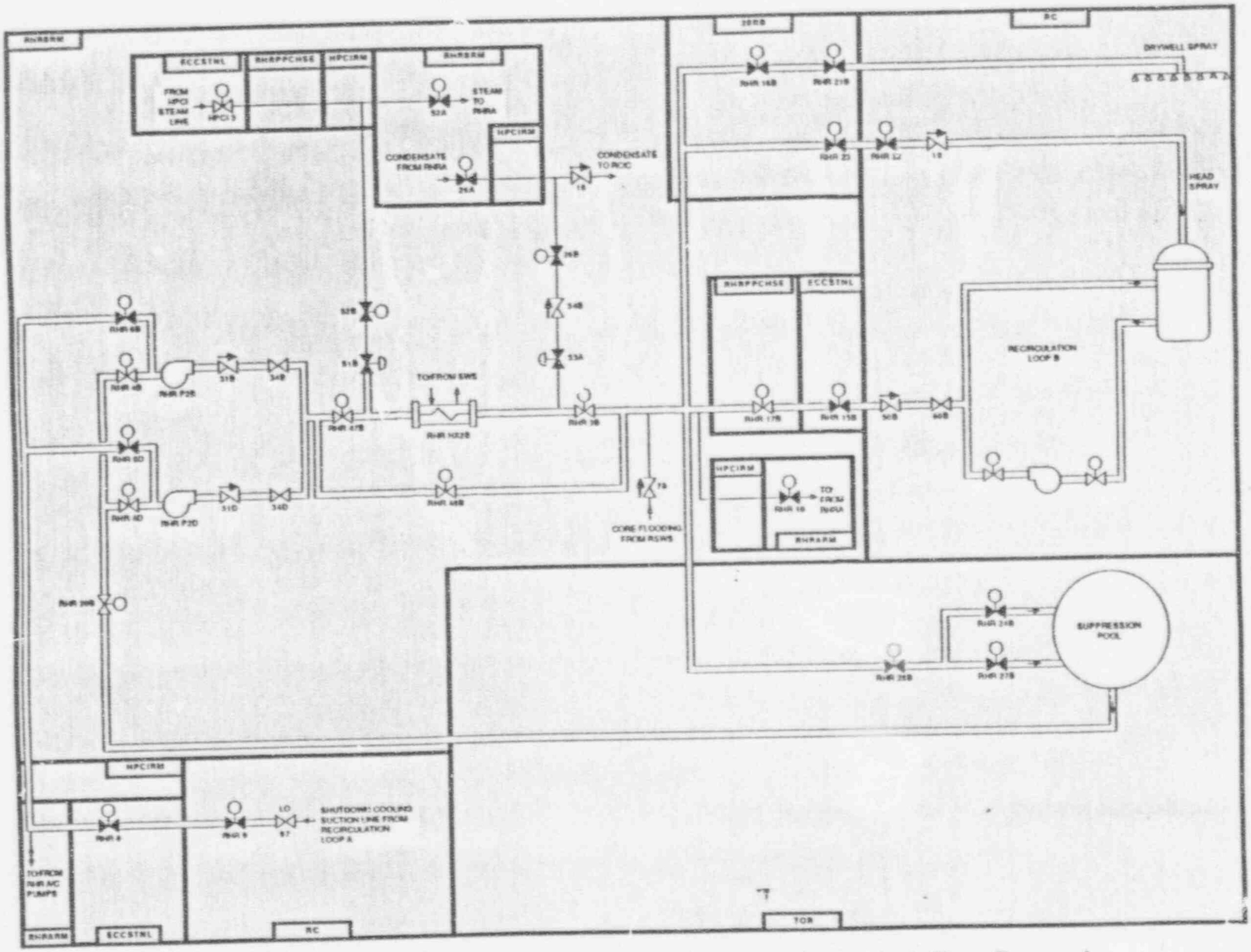

Fig. A.8-2 (SAIC 89/1011). Figure 3.3-8. Brunswick 2 Residual Heat Removal System, Loops B and D, Showing Component Locations
CAUTION: This is NOT a controlled document.

DC Power (DCP) System

Table A.9-1 Importance Basis and Failure Mode Identification

CONDITIONS THAT CAN LEAD TO FAILURE

Mission Success Criteria

The DCP consists of battery banks, battery chargers, circuit breakers, distribution switchboards, dc motor control centers (MCCs) and dc distribution panels. There are four sets of 125 V battery banks (2A-1, 2A-2, 2B-1, and 2B-2). Each battery bank has an associated battery charger to keep the battery bank fully charged. The battery chargers are fed from 480 Vac MCCs. Two battery banks and two battery chargers feed each of two distribution switchboards (2A and 2B). This enables each distribution switchboard to provide 125/250 Vdc power to the various Unit 2 dc MCCs and dc distribution panels. Switchboard 2A provides power to division 1 loads and switchboard 2B provides power to division 2 loads.

The DCP system is designed so that the loss of any single piece of equipment such as battery charger or distribution panel, etc., will not prevent the equipment required for emergency core cooling from operating. The DCP is configured so that the primary equipment, such as primary relays, is fed from one battery bus and the backup equipment is fed from the other battery bus. Some common dc loads are fed from both battery buses with a transfer switch provided to change to the alternate feed, should the primary fail.

The DCP system includes the circuit breakers that feed power from the 480 Vac MCCs to the battery chargers. However, as modeled in the PRA, the DCP system does not include the circuit breakers that feed power from the dc MCCs and the dc distribution panels to the various safety systems. The DCP is successful if the 125 Vdc MCCs and distribution panels that provide power to the safety-related loads remain energized. Failure can occur as a result of any of the following: a distribution switchboard failure, a feeder breaker failure,, or power unavailable from battery bank and associated charger. The mission time for the DCP was chosen to be 24 hours.

1. Fault in Battery Bank 2B-1 or in Output Breaker

   Failure of Battery Bank 2B-1 prevents power from being supplied to its essential loads (PC,MT,TS).

2. Fault in Battery Bank 2A-2 or in Output Breaker

   Similarly, failure of Battery Bank 2A-2 prevents power from being supplied to its essential loads (PC,MT,TS).

3. <u>Fault in Panel 4-A or in the Supply Breaker</u>

DC panel 4A supplies power to several reactor safeguards actuation loads. Failure of Panel 4A prevents power to the essential loads (PC,MT,TS).

4. <u>Fault in Panel 4B or in the Supply Breaker</u>

Similarly, DC Panel 4B supplies power to several reactor safeguards actuation loads. Failure of Panel 4B prevents power to the essential loads (PC,MT,TS).

5. <u>Fault in Distribution Switchboard 2A-P</u>

Failure of Distribution Switchboard 2A-P will prevent power from being supplied to the essential loads (PC,MT,TS).

6. <u>Fault in Distribution Switchboard 2B-N</u>

Similarly, failure of Distribution Switchboard 2B-N will prevent power from being supplied to the essential loads (PC,MT,TS).

7. <u>Fault in Battery Bank 2A-1 or in the Output Breaker</u>

Failure of Battery Bank 2A-1 prevents power from being supplied to its essential loads (PC,MT,TS).

8. <u>Fault in MCC 2XDA or in the Feeder from 2A-P or 2A-N</u>

A fault in MCC 2XDA or in the feeder from 2AP or 2A-N will prevent power from being supplied to its essential loads (PC,MT,TS).

DC Power (DCP) System

TABLE A.9-2 MODIFIED SYSTEM WALKDOWN

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
*Battery Bank Output Breaker	2E1				Compart. GM2	Contl. Bldg. 125/250V DC Switchboard 2B Div. II El. 23' Battery Room	Off**	
*Battery Bank Output Breaker	2A2				Compart. GK2	Contl. Bldg. 125/250V DC Switchboard 2A Div. I El. 23' Battery Room	Off**	
Panel Supply Breaker	4A				Compart. GI6	Contl. Bldg. 125/250V DC Switchboard 2AP Div. I El. 23' Battery Room	On	

*Examine battery bank and circuit breaker for temperature and ventilation conditions.
**During normal operation, this breaker will be in the "on" position.

DC Power (DCP) System

TABLE A.9-2 MODIFIED SYSTEM WALKDOWN (Cont'd)

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
*Battery Bank Output Breaker	2A1				Compt. GK1	Contl. Bldg. 125/250V DC Switchboard 2A Div. I El. 23' Battery Room	Off**	
DC-MCC	2XDA				Compt. GJ3	Contl. Bldg. 125/250V DC Switchboard 2A Div. I El. 23' Battery Room	On	
125 Volt Dist. Panel	4B	Battery Room	N/A	N/A	Compt. GK6	Contl. Bldg. El. 25' 125/250V DC Switchboard 2B Div. II	On	

*Examine battery bank and circuit breaker for temperature and ventilation conditions.
**During normal operation, the breaker will be in the "on" position.

TABLE A.9-2 (Cont'd)

REFERENCE DOCUMENTS

TITLE	I.D. NO.	REV	DATE
System Procedures:			
DC Electrical System Operating Procedure	OP-51	018	2/19/88
System Description:			
Brunswick PRA Volume I			
"DC Power System"			
P&ID's No.:			
Brunswick PRA Volume I	Fig. M.3.4-24		
"DCP Simplified Diagram"			

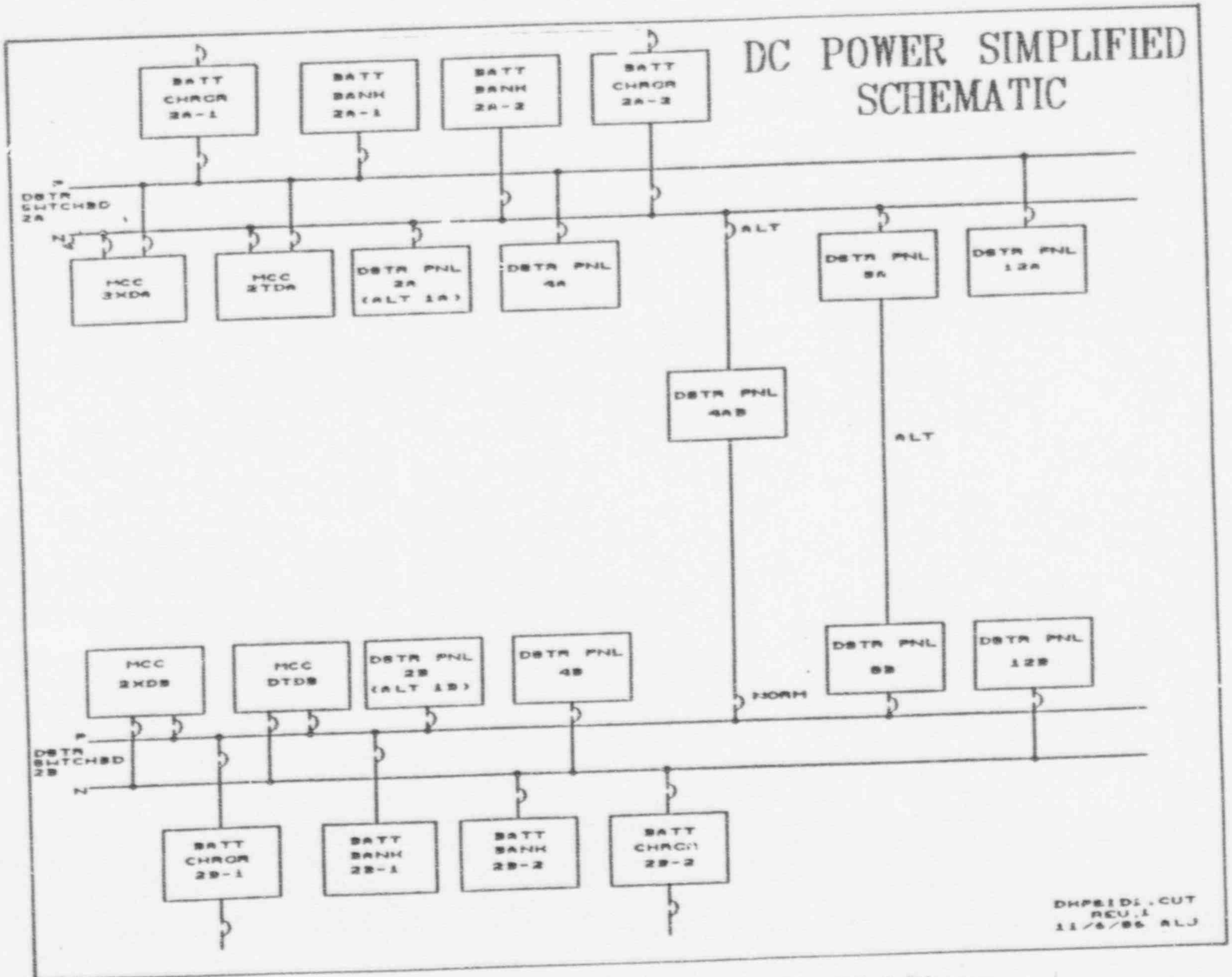

Fig. A.9-1 (PRA Fig. M.3.4-24). DC Power Simplified Diagram.
CAUTION: This is NOT a controlled document.

Table A.10-1 Importance Basis and Failure Mode Identification

CONDITIONS THAT CAN LEAD TO FAILURE

Mission Success Criteria

The ACP consists of transformers, circuit breakers, 4160 V auxiliary buses, 4160 V emergency buses, 480 V unit sub-stations, 480 V motor control centers, and 120-208 V power distribution panels. The ACP system does not include components in the switchyard, the CP&L power grid, the diesel generators, or the circuit breakers that feed power from the diesel generators to the emergency buses.

The auxiliary buses (1C, 1D, 2C, and 2D) receive ac power from either the startup auxiliary transformers (SAT) or the unit auxiliary transformers (UAT). The SAT is used during startup, shutdown, and abnormal plant operating conditions. It receives 230 kV power from the CP&L power grid and steps it down to 4160 V. The UAT is used during normal operation after the unit's main generator has been brought on line. It receives 24 kV power from the main generator and steps it down to 4160 V. Power transfer from the SAT to the UAT is performed by operator action, while power transfer from the UAT to the SAT is performed by either operator action or automatic fast transfer. Interlocks prevent power from being supplied from both the SAT and UAT simultaneously.

The emergency buses (E1, E2, E3, and E4) receive power from the auxiliary buses, except during loss of off-site power conditions. During loss of off-site power, the ac power is provided by the diesel generators. Each auxiliary bus feds an emergency bus with each feed containing two circuit breakers in series. The two circuit breakers work in tandem with one known as the "master" and the other known as the "slave". The master and slave breakers trip open as a result of automatic control circuitry actuation or operator action.

The emergency buses provide power directly to some safety-related equipment, such as RHR pumps, and to loads off of the 480 V unit substations. Power to Unit 2 RHR and RHR Service Water Pumps 2C and 2D is supplied from Unit 1 busses E1 and E2, respectively. Similarly, power to Unit 1 RHR and RHR Service Water Pumps 1A and 1B is supplied from Unit 2 busses E3 and E4, respectively. The LPCI injection valves are also supplied from the opposite unit busses. The unit substations consist of transformers that step down 4160 V power to 480 V power and 480 V switchgear bus sections. The unit substations, in turn, feed power to a multitude of 480 V motor control centers and to transformers that step down the 480 V power to 120-208 V power for use by power distribution panels. The power distribution panels feed 120-208 V power panels.

The ACP to the individual safety-related loads is successful if the 4160 V emergency buses, the 480 Vac MCCs, and the 120-208 Vac power panels that provide power to the safety-related loads remain energized. Failure can occur as a result of any one of the following: a bus fails, a transformer fails, a feeder breaker fails, a feeder breaker fails on ICC, insufficient HVAC provided to the Diesel Generator Building, diesel generator power not available, a MCC fails, or an ac power panel fails. The mission time for ACP was chosen to be 24 hours.

1. Failure to Recover Offsite Power Within One Half Hour

Combined with subsequent faults in the Emergency Diesel Generator System, failure to recover offsite power within one half hour can lead to loss of all AC power (OP,PC,MT,TS).

2. 480V Bus E7 Fault or 4160/408 Transformer Failure or Bus E7 Feeders Fail Open

480 V Bus E7 supplies several motor control centers and the battery chargers for DG3. Faults in the bus or in the 4160/408 transformer or of the feeders failing open causes unavailability of the connected loads (OP,PC,MT,TS).

3. 480 Bus E8 Fault or 4160/408 Transformer Failure or Bus E8 Feeders Fail Open

Similarly, 480V Bus E8 supplies several motor control centers and the battery chargers for DG4. Faults in the bus or in the 4160/408 transformer or of the feeders failing open causes unavailability of the connected loads (OP,PC,MT,TS).

4. 480 VAC MCC 2CA Fault or Feeder Fails Open

480 VAC MCC 2CA controls the supply and exhaust fans for Battery Room 2A and supplies the power to battery chargers 2A-1 and 2A-2. Faults in the MCC or feeder can cause insufficient HVAC flow through the battery room, thereby failing the battery, or preventing battery chargers 2A-1 and 2A-2 from functioning (OP, PC, MT).

5. 480 VAC MCC 2CB Fault or Feeder Fails Open

Similarly, 480 VAC MCC 2CB controls the supply and exhaust fans for Battery Room 2B and supplies the power to battery chargers 2B-1 and 2B-2. Faults in the MCC or feeder can cause insufficient HVAC flow through the battery room, thereby failing the battery, or preventing battery chargers 2B-1 and 2B-2 from functioning (OP, PC, MT).

6. Fault in 4160V E3 Bus

4160V Bus E3 supplies power from Diesel Generator 3 to 480V Bus E7 which in turn feeds several motor control centers and the battery chargers for DG3. Failure of this bus causes unavailability of the connected loads which include (OP,MT,TS):

a) Core Spray Pump 2A
b) RHR Pump 2A room cooling
c) RHR Pump 2A

d) Service Water System—
   Nuclear Header Pump 2A
e) Service Water System—
   Conventional Header Pump 2A

7. Fault in 4160V E4 Bus

4160V Bus E4 supplies power from Diesel Generator 4 to 480 Bus E8 which in turn also feeds several motor control centers and the battery chargers for DG4. Failure of this bus causes unavailability of the connected loads which include (OP,MT,TS):

a) Core Spray Pump 2B
b) RHR Pump 2B room cooling
c) RHR Pump 2B
d) Service Water System—
   Nuclear Header Pump 2B
e) Service Water System—
   Conventional Header Pump 2B

8. Power Fails for Battery Rooms 2A and 2B Duct Heaters

The duct heaters for Battery Rooms 2A and 2B are supplied from a common source. Failure of the duct heaters can cause failure of Batteries 2A and 2B due to overcooling (OP,MT).

9. Complete or Partial Loss of Off Site Power Caused by the Following:

a) Feeder to Bus E3 Fails Open Due to I&C Related or Other Faults
b) Startup Auxiliary Transformer (SAT) Fails
c) Feeder From SAT to Bus 2D Fails Open Due to I&C Related or Other Faults
d) 4160 VAC Bus 2D Fault

Assuming initially no power from the Unit Auxiliary Transformer (UAT), failure of the above can cause either a partial or complete loss of off-site power, forcing reliance upon Diesel Generator No. 3 (OP, PC, MT).

10. Complete or Partial Loss of Off-Site Power Caused by the Following:

a) Feeder to Bus E4 Fails Open Due to I&C Related or Other Faults
b) Startup Auxiliary Transformer (SAT) Fails
c) Feeder from SAT to Bus 2C Fails Open Due to I&C Related or Other Faults
d) 4160VAC Bus 2C Fault

Again assuming initially no power from the UAT, failure of the above can cause either a partial or complete loss of off-site power, forcing reliance upon Diesel Generator No. 4 (OP,PT,MT).

11. <u>480 VAC MCC 2XA Fault or Feeder Fails Open</u>

480 VAC MCC 2XA controls numerous motor-operated valves. Faults in the MCC or feeder can cause loss of power to those valves and components which include (OP, PC, MT):

a) MOV F027A—RHR suppression pool spray
b) MOV F024A—RHR suppression pool return
c) MOV F007A—RHR Pump 2A minimum flow bypass
d) MOVs F016A, F021A—RHR drywell spray
e) MOV F048A—RHR Heat Exchanger 2A bypass
f) MOVs F006A, F006C—RHR Pumps 2A, 2C shutdown cooling flowpath suction isolation valves
g) MOVs F008, F009—RHR outboard and inboard shutdown cooling isolation valves
h) MOVs F004A, F004C—RHR Pumps 2A, 2C suppression pool suction isolation valves
i) MOV F020A—RHR suppression pool isolation valve to pumps 2A and 2C suction lines
j) MOV V004—SWS conventional header isolation for Turbine Building Closed Cooling Water (TBCCW) System
k) MOV V106—SWS alignment valve for Reactor Building Closed Cooling Water (RBCCW) System
l) MOV V111—SWS alignment valve to Vital Service Header (to RHR pump seal coolers, RHR pump room coolers and Core Spray pump room coolers) from Conventional Header
m) MOV V118—SWS isolation valve for Vital Service Header from Conventional Header
n) MOV F002A—RHRSWS Pump 2A discharge
o) RHR Pump Room 2A Fan Cooler Unit (FCU)

12. <u>480 VAC MCC 2XB Fault or Feeder Fails Open</u>

480 VAC MCC 2XB, like its counterpart MCC 2XA, controls numerous motor-operated valves, almost all of which are the Train B counterparts to the Train A valves and components identified above. Faults in the MCC or feeder can cause loss of power to the valves and components which include (OP, PC, MT):

a) MOV F027B—RHR suppression pool spray
b) MOV F024B—RHR suppression pool return
c) MOV F007B—RHR pump 2B minimum flow bypass
d) MOVs F016B, F021B—RHR drywell spray
e) MOV F048B—RHR Heat Exchanger 2B bypass
f) MOVs F006B, F006D—RHR Pumps 2B, 2D shutdown cooling flowpath suction isolation valves
g) MOVs F004B, F004D—RHR Pumps 2B, 2D suppression pool suction isolation valves

h) MOV F020B—RHR suppression pool isolation valve to pumps 2B and 2D suction lines
i) MOV V105—SWS alignment valve for RHRSWS pumps from Nuclear Header
j) MOV V117—SWS alignment valve to Vital Service Header (to RHR pump seal coolers, RHR pump room coolers and Core Spray pump room coolers) from Conventional Header
k) MOV V102—RHRSWS crosstie valve between RHRSWS pumps 2A and 2C suction header and pumps 2B and 2D suction header
l) RHR Pump Room 2B Fan Cooler Unit (FCU)

13. <u>480 VAC MCC 2XA-2 Fault or Feeder Fails Open</u>

480 VAC MCC 2XA-2 controls RHR MOV F015A, the RHR return line to Recirculation Loop A isolation valve, which must open for the Low Pressure Coolant Injection (LPCI) and Shutdown Cooling (SDC) modes of operation, and MOV F028A, the RHR return line to the suppression pool isolation valve, which must open for the Suppression Pool Cooling (SPC) and Containment Spray (CNS) modes of operation. Faults in the MCC or feeder can prevent operation of these valves when required (OP, PC, MT).

14. <u>480 VAC MCC 2XB-2 Fault or Feeder Fails Open</u>

Similar to MCC 2XA-2 above, MCC 2XB-2 controls RHR MOV F015B and F028B, the Train B counterparts to MOV F015A and F028A discussed above. Faults in the MCC or feeder can prevent operation of these valves when required (OP, PC, MT).

15. <u>Power Not Available from Bus E1</u>

Unit 1 Bus E1 provides power directly to RHR and RHR Service Water Pumps 2C on Unit 2. Unit 1 Bus E1 is a Division I bus which can also be used to supply power to its companion Division I Bus E3 for Unit 2. Failure to provide power from Bus E1 when required fails Bus E3 of Unit 2 and Unit 2 RHR and RHRSW Pumps 2C. (OP, PT, MT, TS).

16. <u>Power Not Available from Bus E2</u>

Similarly, Unit 1 Bus E2 provides power directly to RHR and RHR Service Water Pumps 2D on Unit 2. Unit 1 Bus E2 is a Division II bus which also can be used to supply power to its companion Division II Bus E4 for Unit 2. Failure to provide power from Bus E2 when required fails Bus E4 of Unit 2 and Unit 2 RHR and RHRSW Pumps 2D. (OP, PT, MT, TS).

BRUNSWICK STEAM ELECTRIC PLANT—UNIT 2
RISK-BASED INSPECTION GUIDE

AC Power System*

TABLE A.10-2 MODIFIED SYSTEM WALKDOWN

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
Feed to 480V AC MCC 2CB					A09	Unit Sub-Station E-8 Div. II Diesel Bldg.	Closed	
Feed to 480V MCC 2CA					AY9	Unit Sub-Station E-7 in Div. I Diesel Bldg.	Closed	
Feed to 4160 Volt Bus E-4**					AJ9	Swgr. E-4	Closed	
					2ACB**	Swgr. 2C	Closed	
Feed to Bus 2C	Disc. No. AC6	Control Room Panel H12-XU5	On					

*The PRA projects most of the AC power failures as hardware failures or failures related to other systems (covered elsewhere). This table lists limited observations that can reduce the risks of AC power failure.

**Check latest surveillance records for satisfactory completion of any open items or concerns regarding the circuit breakers feeding this load.

BRUNSWICK STEAM ELECTRIC PLANT—UNIT 2
RISK-BASED INSPECTION GUIDE

AC Power System*

TABLE A.10-2 MODIFIED SYSTEM WALKDOWN (Cont'd)

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
Bkr. Feeding from SAT2 to Swgr. 2D**	Disconnect No. 2AD4	Control Room Panel H12-X45	On		2AD1	Div. Bop Swgr. 2D	Closed	
Feeder to Bus E-7 Breaker**					AZ1	Div. I Diesel Bldg.	Closed	
Feeder to Swgr. E-8**					AZ5	Div. II Diesel Bldg.	Closed	
Cir. Bkr. Feeding MCC 2XA**					AY 2	Sub-Station E-7 Div. I Diesel Bldg.	Closed	
Cir. Bkr. Feeding MCC 2XA2**					AT8	Sub-Station E-5 Div. I Diesel Bldg.	Closed	

**Check latest surveillance records for satisfactory completion of any open items or concerns regarding the circuit breakers feeding this load.

AC Power System*

TABLE A.10-2 MODIFIED SYSTEM WALKDOWN (Cont'd)

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
Cir. Bkr. Feeding MCC 2XB**					AO2	Sub-Station E-8 Div. II Diesel Bldg.	Closed	
4160V AC Bus E-2**					1ACB	Div. BOP Swgr. 1C	Closed	
					AG4	Div. II Swgr. E-2	Closed	
4160V AC Bus E-1**					1AD1	Div. BOP Swgr. 1D	Closed	
					AG6	Div. I Swgr. E-1	Closed	

**Check latest surveillance records for satisfactory completion of any open items or concerns regarding the circuit breakers feeding this load.

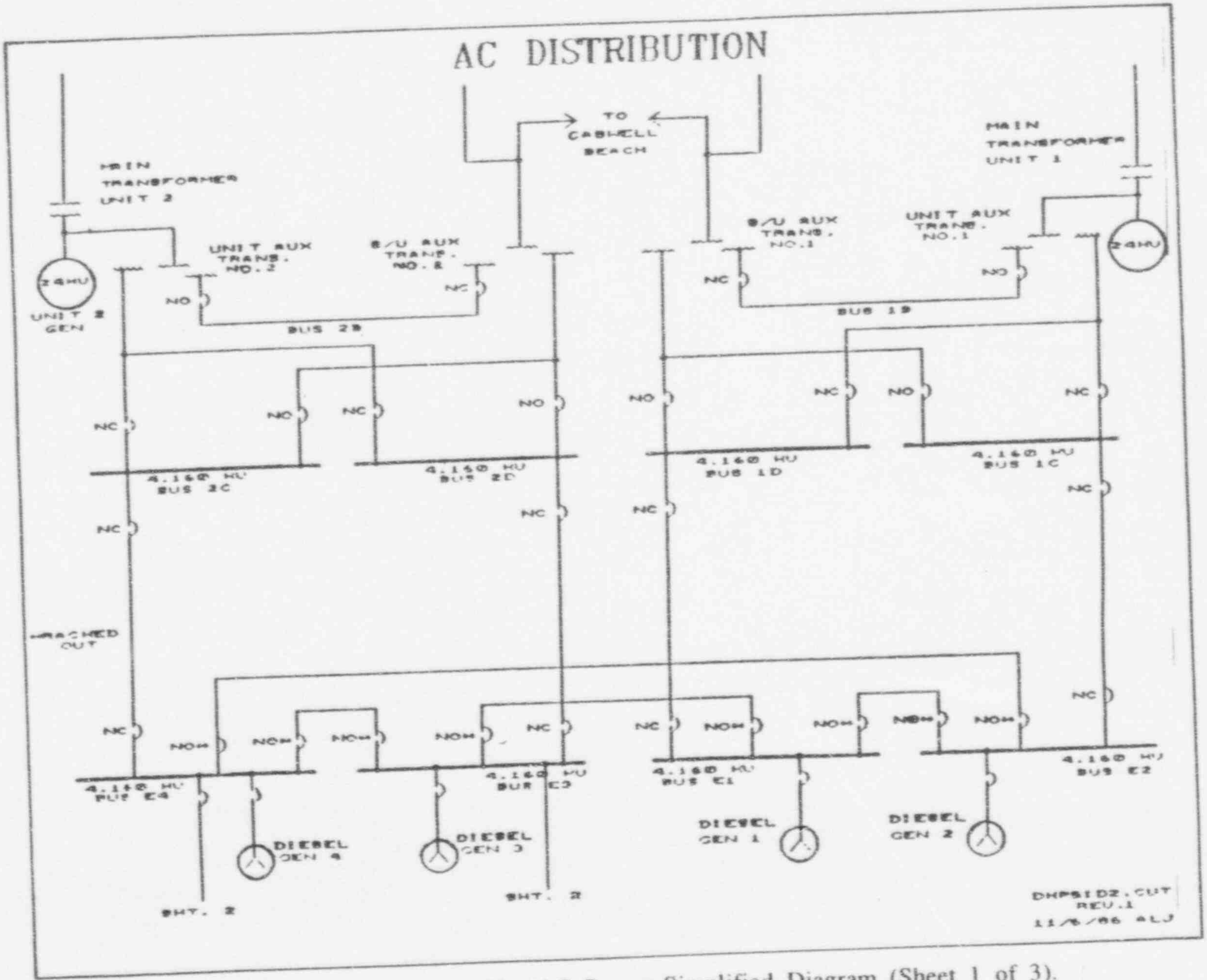

Fig. A.10-1 (PRA Fig. M.3.4-22). AC Power Simplified Diagram (Sheet 1 of 3).
CAUTION: This is NOT a controlled document.

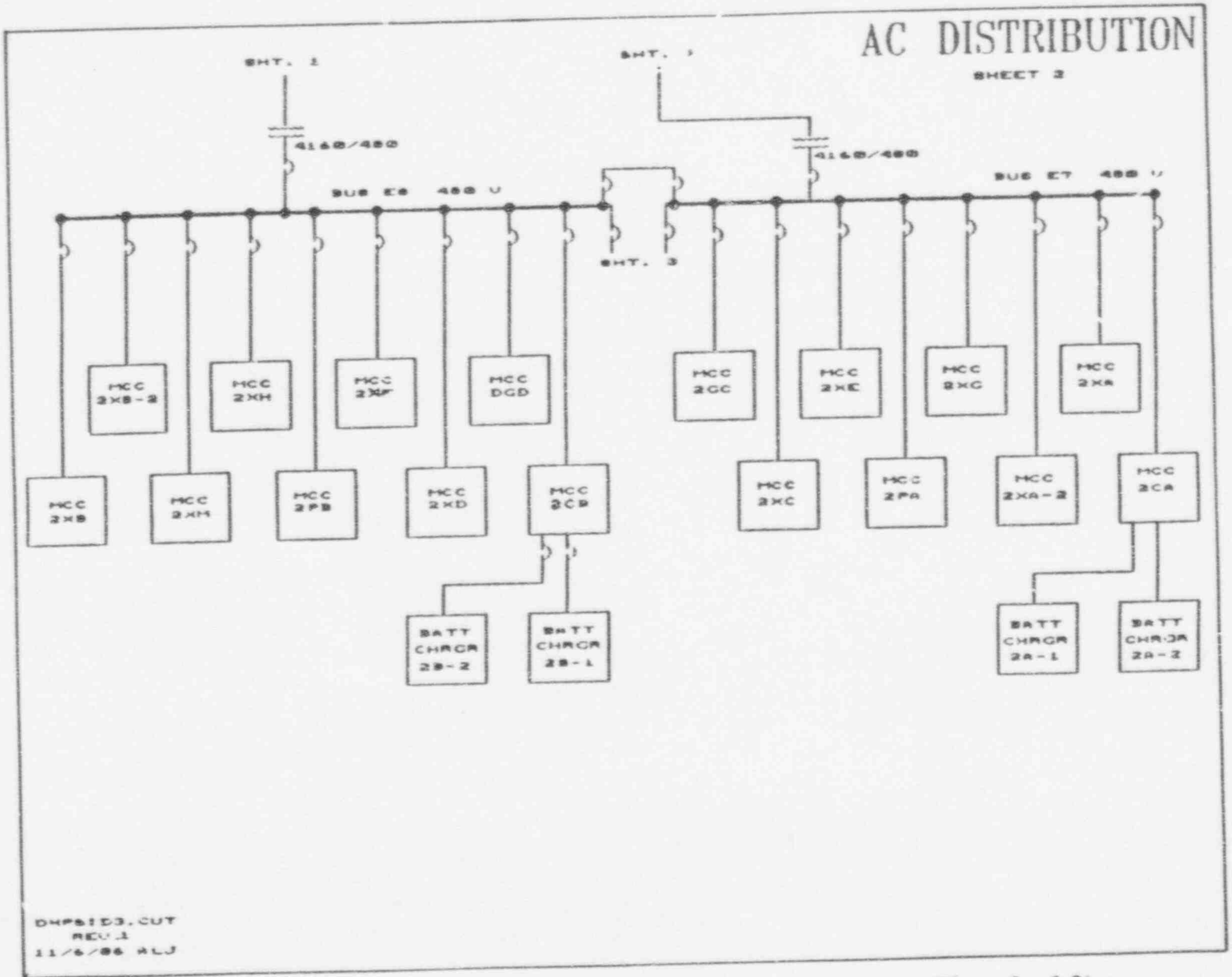

Fig. A.10-1 (PRA Fig. M.3.4-22). AC Power Simplified Diagram (Sheet 2 of 3).
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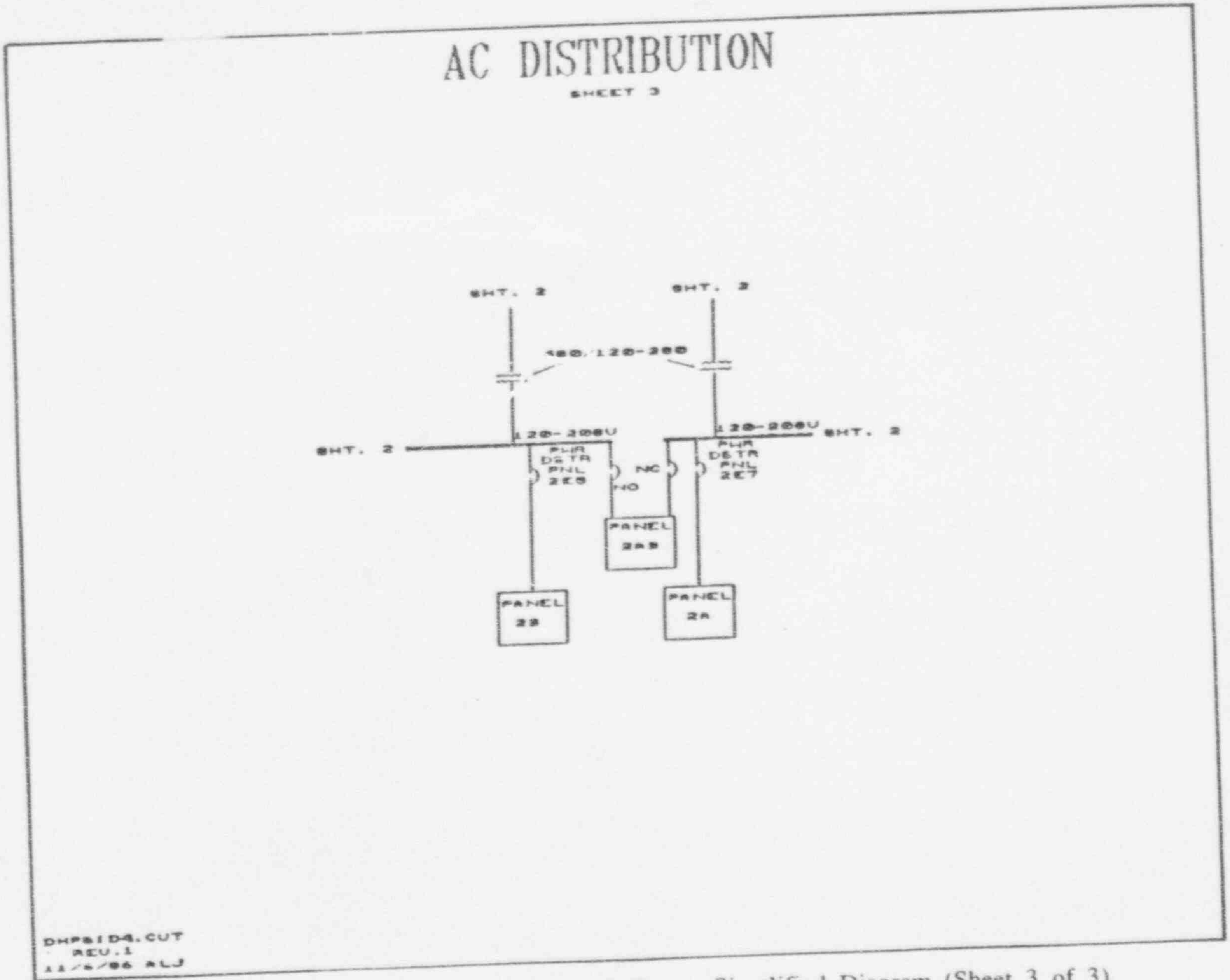

Fig. A.10-1 (PRA Fig. M.3.4-22). AC Power Simplified Diagram (Sheet 3 of 3).
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Battery Room Fans and Heaters

Table A.11-1 Importance Basis and Failure Mode Identification

CONDITIONS THAT CAN LEAD TO FAILURE

Mission Success Criteria

Battery Room Fans and Heaters

Successful operation of the battery room ventilation system for a 24 hour mission time requires that the supply and exhaust fans are operable and that there is a flow path from the supply fan to the battery room and out to the atmosphere. Failure of the system occurs if the intake filter is restricted, a fan is inoperable, or the dampers fail. The duct heater must be operational to provide sufficient heating. Thus, overcooling will result if the heater itself or the power to the heater fails.

Insufficient HVAC flow through the battery room can cause two types of faults:

1. Hydrogen buildup with the risk of explosion, and

2. Temperature increase with the risk of battery charge output failure.

HVAC analyses provided in the PRA indicate that the battery rooms will not heat up enough within 24 hours to fail the batteries. Hydrogen buildup during rapid charging without the HVAC system might be significant. However, the frequency of such an event combined with a reactor scram is assumed to be negligible in the PRA. Operator recovery action can be taken by opening the battery room doors.

1. Battery Rooms 2A and 2B Supply and Exhaust Fans Fail to Run

Failure of the battery room supply and exhaust fans to run can cause heat or hydrogen buildup in the battery rooms (OP,MT).

2. Battery Rooms 2A and 2B Supply and Exhaust Dampers Fail Closed Due to Faults in the Actuation Control for the Vortex Dampers

Faults in the actuation control for the vortex dampers will cause the battery room supply and exhaust dampers to fail closed, thereby causing inadequate battery room ventilation (PC, MT).

3. Failure of Thermostats for Battery Rooms 2A and 2B Duct Heaters Causing Overcooling

Failure of the thermostats for the duct heaters of Battery Rooms 2A and 2B can cause overcooling of the rooms and subsequently insufficient charge output from the batteries (PC,MT).

4. Insufficient Supply Flow to Battery Rooms 2A and 2B Due to Restriction of the Intake Plenum Air Filters

Restrictions in the intake plenum air filters can cause insufficient supply flow to the battery rooms (OP,MT).

Battery Room Fans and Heaters

TABLE A.11-2 MODIFIED SYSTEM WALKDOWN

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
Battery Room 2A Supply Fan	2C-SF-CB				Comp. No. PJ5	Control Bldg. Disconnect Switches Above Battery Room	On	
Battery Room 2B Supply Fan	2B-SF-CB				Comp. No. PJ6	Control Bldg. Disconnect Switches Above Battery Room	On	
Battery Room 2A Exhaust Fan	2C-EF-CB				Comp. No. PK0	Control Bldg. Disc. Switch EL-70	On	
Battery Room 2B Supply Fan Motor Heater Circuit Breaker	2B-SF-CB (C42)				Comp. No. 4	Control Bldg. 480V MCC 2CB 120V Distribution Panel	On	
Battery Room 2B Exhaust Fan Motor Heater Circuit Breaker (C43)	2B-SF-CB				5	Control Bldg. Disc. Sw. EL 70'	On	
					PJ9	Control Bldg. Disc. Sw. EL 70'	On	
Battery Room 2B Supply Fan	2B-SF-CB				Comp. No. C43	Unit 2 Control Bldg. 480V MCC 2CB EL 23'	On	

Battery Room Fans and Heaters

TABLE A.11-2 MODIFIED SYSTEM WALKDOWN (Cont'd)

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
Battery Room Exhaust Fan	2B-EF-CB				Comp. No. C42	Unit 2 Control Bldg. 480V MCC 2CB El. 23'	On	
Battery Room 2A Exhaust Fan Motor Heater Circuit Breaker (C20)	2C-EF-CB				12	Control Bldg. 480V MCC 2CA 120V Distribution Panel	On	
Battery Room 2A Supply Fan Motor Heater Circuit Breaker	2C-SF-CB				13	Control Bldg. 480V MCC 2CA 120V Distribution Panel	On	
Battery Room 2A Exhaust Fan	2C-EF-CB				Comp. No. C20	Unit 2 Control Bldg. 480V MCC 2CA El. 23'	On	
Battery Room 2A Supply Fan	2C-SF-CB				Comp. No. C21	Unit 2 Control Bldg. 480V MCC 2CA El. 23'	On	

Battery Room Fans and Heaters

TABLE A.11-2 MODIFIED SYSTEM WALKDOWN (Cont'd)

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
Battery Room 2A Damper SOV	926				21	Control Bldg. 120V Emergency Distribution Panel 2C-HY0 El. 23'	On	
Battery Room 2B Damper								
Battery Room 2A Supply & Exhaust Dampers*								
Battery Room 2B Supply & Exhaust Dampers*								
*Thermostat for Battery Room 2A								
*Thermostat for Battery Room 2B								

*Visually inspect thermostats and dampers for possible signs of failure.

TABLE A.11-2 (Cont'd)

REFERENCE DOCUMENTS

TITLE	I.D. NO.	REV	DATE
System Procedures:			
Control Building Ventilation System	OP-37	017	12/17/87
System Description:			
Brunswick PRA Volume I	Section M.3.4.16		
"Heating, Ventilating, and Air Conditioning			
Systems"			
P&ID's No.:			
Brunswick PRA Volume I	Fig. M.3.4-32		
"Battery Room Cooler and Heater Detailed			
Diagram (1 of 2) F-4080			
"Battery Room Cooler and Heater Detailed			
Diagram (2 of 2) F-03493			

Reactor Core Isolation Cooling (RCIC) System

Table A.12-1 Importance Basis and Failure Mode Identification

CONDITIONS THAT CAN LEAD TO FAILURE

Mission Success Criteria

The Reactor Core Isolation Cooling (RCIC) system can be operated manually or automatically to maintain the reactor vessel water level in the event of a reactor isolation accompanied by a loss of feedwater. RCIC is considered to be redundant to HPCI for transients and small LOCAs. For intermediate LOCAs and anticipated transients without scram (ATWSs), RCIC is not sufficient to replace HPCI.

The success criteria for the RCIC system is that the RCIC pump starts and continues to deliver rated flow of 400 GPM into the reactor vessel in accordance with the technical specification requirements for 24 hours. The flow rate is approximately equal to the reactor water boil-off rate 15 minutes after shutdown.

RCIC consists of a 100% capacity single train with dc motor-operated valves and a steam turbine-driven pump. The normal water supply is from the Condensate Storage Tank (CST). If the CST level decreases to a predetermined level, or the suppression pool level rises above a predetermined level, pump suction is automatically transferred to the suppression pool. The RCIC pump discharges into the B feedwater line (through valve F010) and subsequently into the feedwater spargers.

1. <u>RCIC Pump Fails to Start or Run</u>

   Failure of the RCIC pump to start or to run for its required mission time prevents RCIC flow (PT,MT,OP).

2. <u>RCIC Pump in Test or Maintenance</u>

   Testing or maintenance of the RCIC pump also prevents RCIC flow (PT,MT).

3. <u>Operator Failure to Override Steam Tunnel High Temperature Trip</u>

   During station blackout scenarios, early isolation of the RCIC turbine on high steam tunnel temperature will occur automatically unless overridden by the operators. Failure of the operators to override when required can lead to core damage (OP).

4. Faults in RCIC System Prevent RCIC Pump Discharge Flow

RCIC flow is prevented by the following (PT,MT):

a) Normally closed RCIC pump discharge isolation dc-powered MOV F013 fails to open when required
b) Normally closed RCIC turbine steam supply isolation dc-powered MOV F045 fails to open when required
c) Normally open RCIC pump discharge isolation dc-powered MOV F012 fails to remain open or check valve F014 fails to open
d) Normally open RCIC pump suction isolation dc-powered MOV F010 fails to remain open or check valve F011 fails to open

5. Failure of Lube Oil Cooling

The RCIC pump discharge provides cooling water flow to the turbine lube oil cooler. Failure of this cooling system will cause RCIC turbine failure (PT,MT).

6. Failure of the Vacuum Breaker on Startup Causes Insufficient Steam Flow from the Turbine

The RCIC turbine exhausts to the suppression pool via a submerged penetration. A vacuum breaker line, containing MOVs F062 and F066 and check valves F063 and F064, connects the turbine exhaust line to the suppression pool atmosphere allowing for vacuum relief of the turbine exhaust line. Failure of the vacuum relief can prevent exhaust steam flow from the RCIC turbine, thereby preventing RCIC flow (PT,MT).

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RISK-BASED INSPECTION GUIDE

Reactor Core Isolation Cooling (RCIC) System

TABLE A.12-2 MODIFIED SYSTEM WALKDOWN

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
Condensate Storage Tank Suction Valve Control Switch	F010	Control Room RTGB Pnl. H12-P601	Open		Comp. B-38	Reactor Bldg. 250V MCC 2XDB EL. 20'	On	
					B-38 Control Switch	Same	Normal	
					2 Htr.	120V AC Pnl. 2DXB EL. 17'	On	
Pump Discharge Valve Control Switch	F012	Same	Open		Comp. B 40	Reactor Bldg. 250V MCC 2XDB EL. 20'	On	
					Comp. B-40 Control Switch	Same	Normal	
					4 Htr.	120V AC Pnl. 2DXB EL. 17'	On	
RCIC Injection Valve Control Switch	F013	Same	Closed		Comp. B-41	2XDB EL. 20'	On	
					B-41 Control Switch	Same	Normal	
					5 Htr.	Panel 2DXB EL. 17'	On	
Vac. Pump Inboard Test Valve	F045*	Reactor Bldg. South RHR Pump Rm.	Closed		Comp. B-44	2XDB EL. 20'	On	
					B-44 Control Switch	Same	Normal	
					8 Htr.	2DXB EL. 17'	On	

*Note: The PRA identified the failure of this pump to start or to run its mission time as a significant failure. Examine pump for visual defects and assure that maintenance procedures are current.

...ctor Core Isolation Cooling (RCIC) System

TABLE A.12-2 MODIFIED SYSTEM WALKDOWN (Cont'd)

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
Turbine Exhaust Vac. Bkr. Valve	F062				Comp. DE4 Cir. Bkr.	MCC 2XA Reactor Bldg. EL. 20'	On	
					9 Htr.	Dist. Panel HS-3 Reactor Bldg. EL. 20'	On	
					DW2 Alt. Feed	MCC 2XD Reactor Bldg. EL. 20'	Off	
Turbine Exhaust Vac. Bkr. Valve Selector Switch	F066	MCC 2XB EL. 20' Reactor Bldg.	Normal		Comp. DL5	MCC 2XB Reactor Bldg. EL. 20'	On	
					8 Htr.	Dist. Panel HL4 Reactor Bldg. EL. 20'	On	

TABLE A.12-2 (Cont'd)

REFERENCE DOCUMENTS

TITLE	I.D. NO.	REV	DATE
System Procedures:			
Reactor Core Isolation Cooling System	OP-16	053	8/23/88
System Description:			
Brunswick PRA Volume I	Section M.3.4.4		
"Reactor Core Isolation Cooling System"			
P&ID's No.:			
Brunswick PRA	Fig. M.3.4-5		
"RCIC Simplified Diagram"	Sheets 1,2,3		

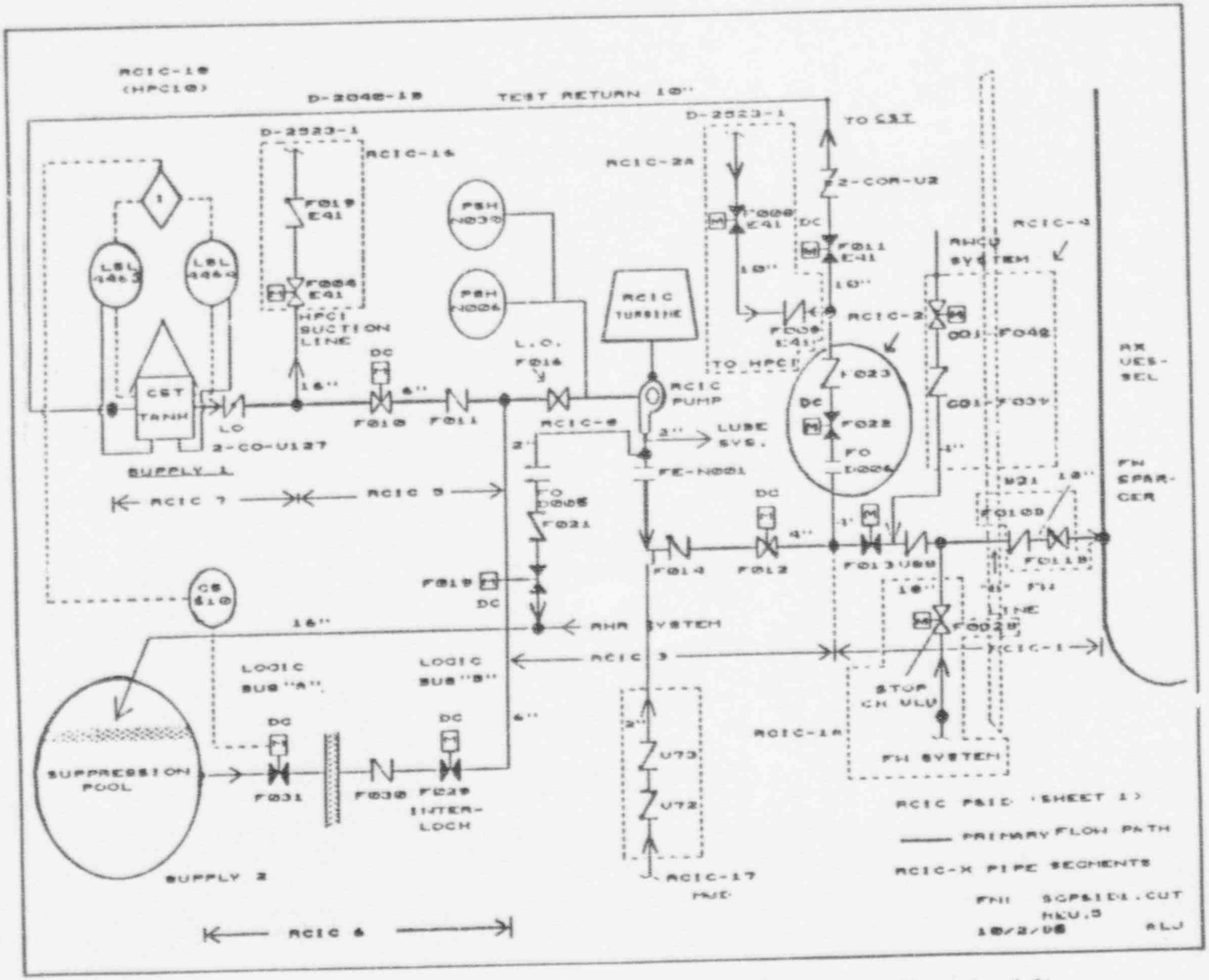

Fig. A.12-1 (PRA Fig. M.3.4-5). RCIC Simplified Diagram (Sheet 1 of 3).
CAUTION: This is NOT a controlled document.

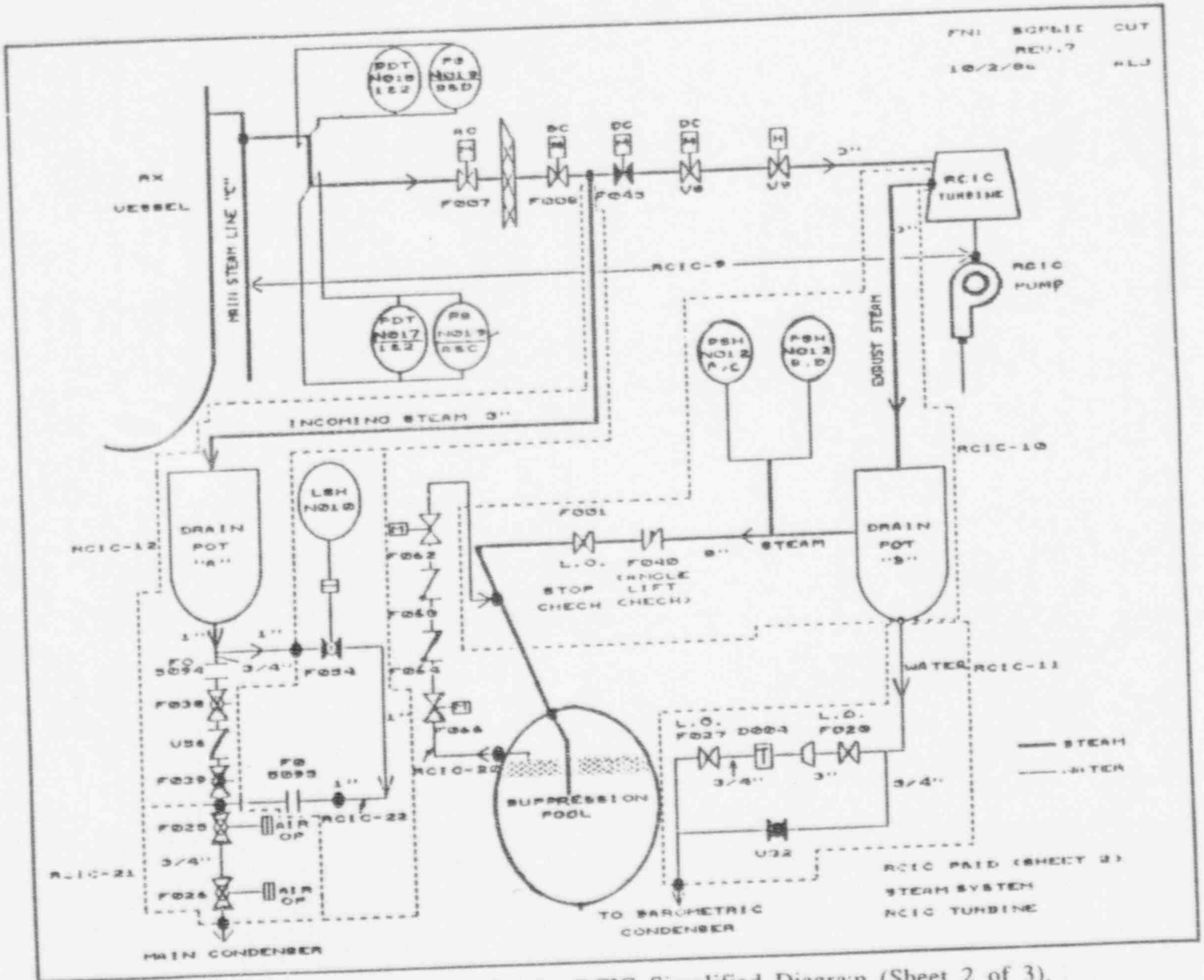

Fig. A.12-1 (PRA Fig. M.3.4-5). RCIC Simplified Diagram (Sheet 2 of 3).
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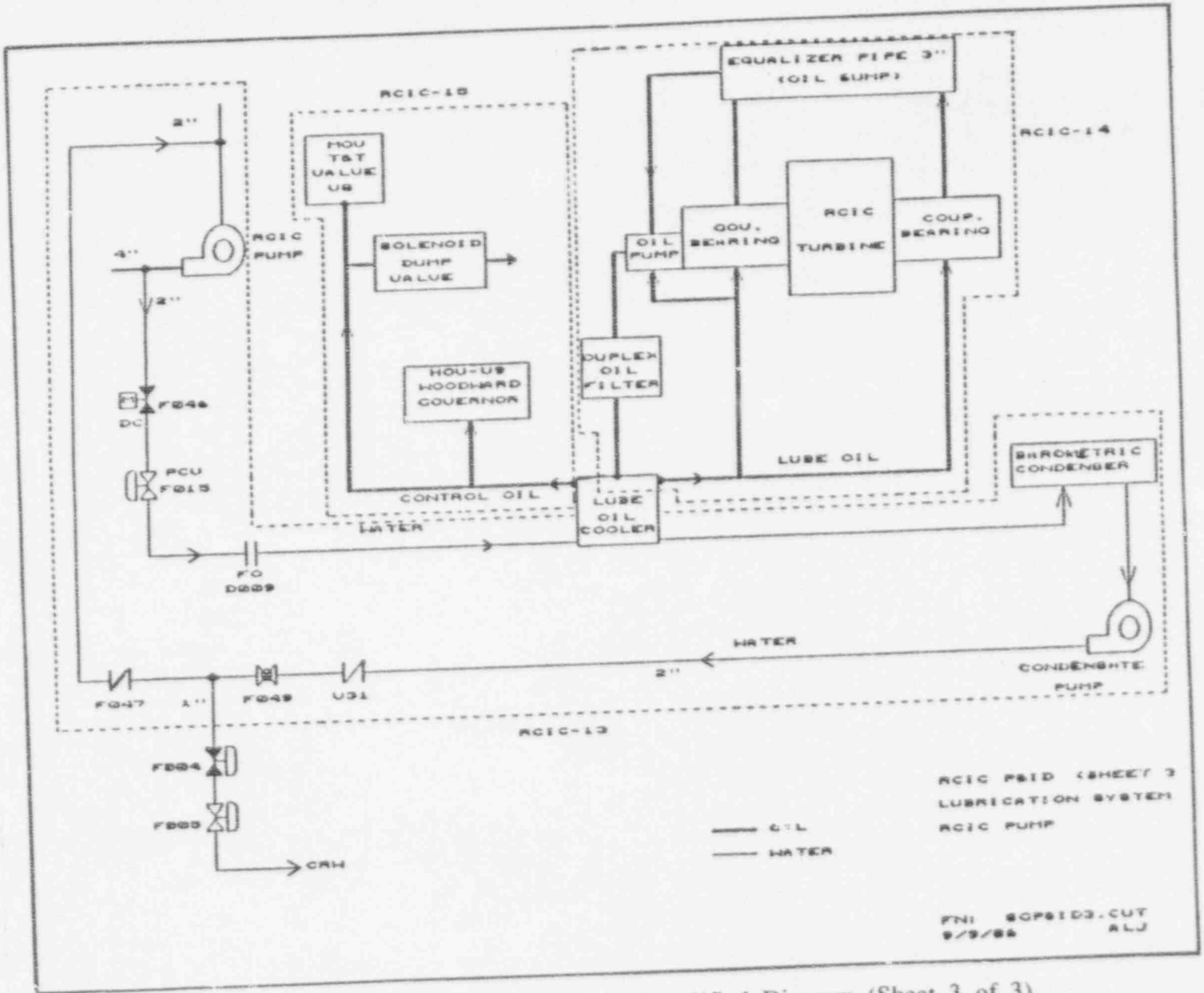

Fig. A.12-1 (PRA Fig. M.3.4-5). RCIC Simplified Diagram (Sheet 3 of 3).
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Control Rod Drive (CRD) Hydraulic System

Table A.13-1 Importance Basis and Failure Mode Identification

CONDITIONS THAT CAN LEAD TO FAILURE

Mission Success Criteria

During normal reactor operations the Control Rod Drive (CRD) system provides water from the Condensate Storage Tank (CST) to the CRD hydraulic control units (HCUs) and each individual control rod drive. High pressure water (1400 psig) is supplied by one of two pumps to drive individual rods into or out of the core and keep the HCU scram accumulators charged for rapid insertion of the rods in the event of a trip signal. In addition, CRD also provides approximately 0.44 gpm of cooling water flow to each drive mechanism. This flow enters the drive mechanism via the insert line (bottom of the drive piston) and flows past the piston seals into the reactor vessel.

To provide makeup water to the reactor vessel following a reactor trip, flow to the vessel can be increased by the plant operators to greater than 140 gpm. The flow path to the vessel in this condition is through the cooling water lines for each of the 137 CRD drives as described above. CRD is considered to be redundant to HPCI and RCIC for normal transients.

The success criterion for the CRD tree is the ability to inject water with both CRD pumps simultaneously through the scram valves and the cooling water header. This criterion was developed from reviews of independent studies by GE and CP&L which indicate that the CRD system is capable of providing sufficient makeup following a transient to maintain reactor vessel level. These studies concluded that level could be maintained with 135 gpm makeup (GE) or 140 gpm makeup (CP&L). Instrumentation currently available in the plant cannot measure system flow above 140 gpm. This appears to be at or near the upper limit of one pump. This, coupled with uncertainties in the analysis, resulted in the decision to define both pumps operating as the minimal success criterion.

1. <u>Operator Fails to Fully Open Valve F002A</u>

In the event of a transient, to properly use the CRD system for makeup to the reactor vessel, flow must be increased to the maximum amount of 140 gpm. Operator failure to fully open flow control valve F002A prevents adequate makeup to the vessel (OP).

2. <u>Operator Fails to Fully Open Valve F003</u>

As above, operator failure to open pressure regulating valve F003 prevents adequate makeup to the vessel (OP).

Control Rod Drive (CRD) Hydraulic System

TABLE A.13-2 MODIFIED SYSTEM WALKDOWN

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
Drive Water Pressure Control Valve	C-12 F003	Control Room RTGB Panel H12-P603	Open		Comp. EZ0	React. EL. 20 East 480V MCC 2XM Panel HR9	On	
					5 Htr.	React. EL. 20 East 480V MCC 2XM Panel HR9	On	
CRD Flow Control Valve	C-12 F002A	Reactor Bldg. Inst. Rack 1R-RB-2 EL. 20' East	Operable		10	Control Bldg. EL. 23' Panel 2 AB 120V AC	On	

TABLE A.13-2 (Cont'd)

REFERENCE DOCUMENTS

TITLE	I.D. NO.	REV	DATE
System Procedures:			
Control Rod Drive Hydraulic System	OP-08	013	8/11/88
System Description:			
Brunswick PRA Volume I	Section M.3.4.5		
"Control Rod Drive (CRD) Hydraulic"			
P&ID's No.:			
Brunswick PRA Volume I	Fig. M.3.4-7		
"CRD Simplified Diagram"			

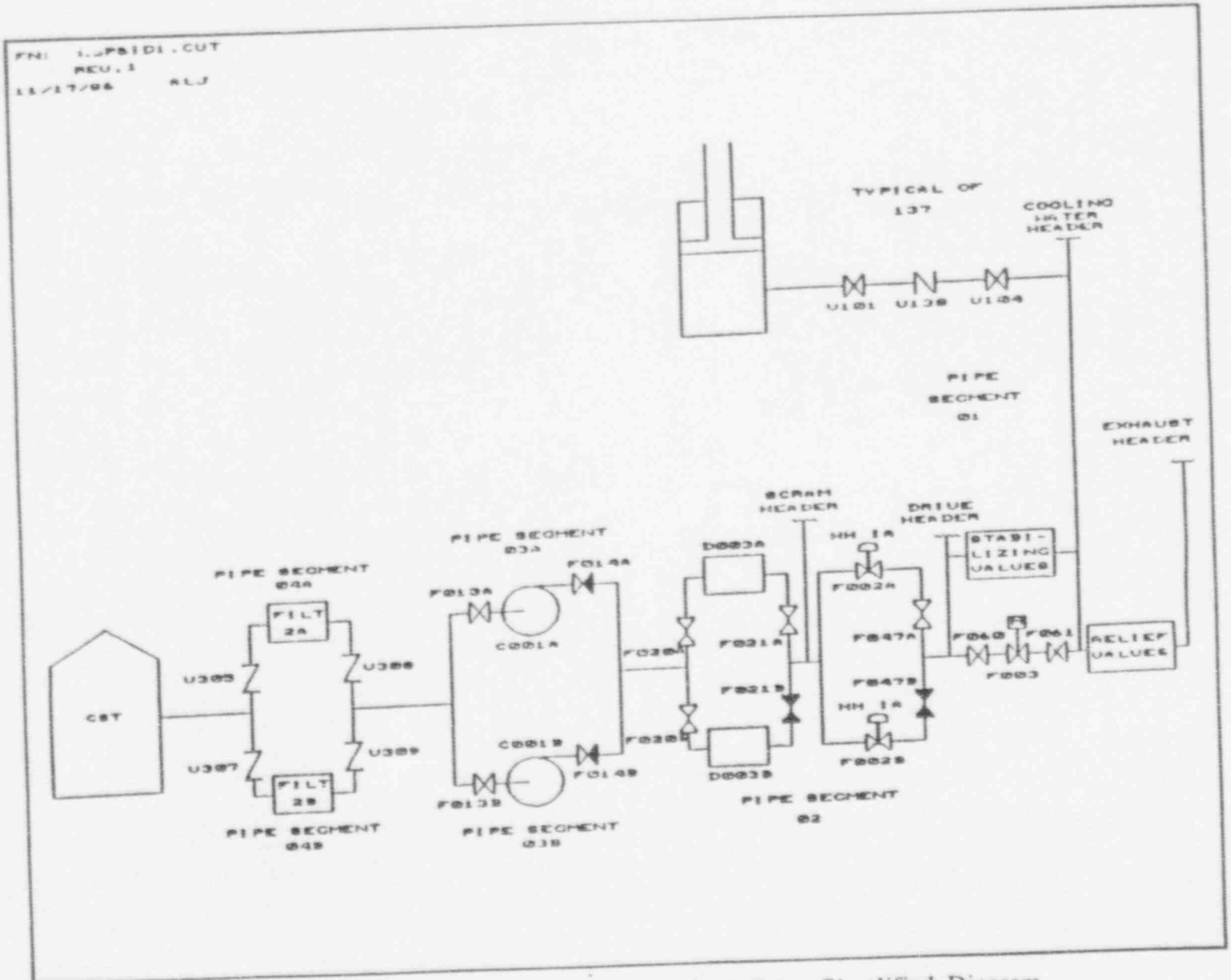

Fig. A.13-1 (PRA Fig. M.3.4-7). Control Rod Drive Simplified Diagram.
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Emergency Core Cooling System (ECCS) Actuation

Table A.14-1 Importance Basis and Failure Mode Identification

CONDITIONS THAT CAN LEAD TO FAILURE

Mission Success Criteria

The Emergency Core Cooling System (ECCS) actuation circuitry provides the necessary control signals to start each of the subsystems that comprise the ECCS. These subsystems are HPCI, ADS, LPCI, and CSS. The required actuations for the above systems occur automatically when a condition is sensed that may represent a potential for inadequacy of core cooling. This occurs regardless of the availability of off-site power.

The RCIC can also provide core cooling. The RCIC actuation circuitry is not properly a part of the ECCS; however, because of the functional similarity between the RCIC and HPCI systems, the RCIC actuation system is included along with the ECCS systems. The ECCS system actuation uses dc power from two different supply panels, Panel 4A and 4B. The subsystems that make up the ECCS are actuated by independent control circuits that provide pump start signals and valve control logic to ensure that each subsystem comes on-line when needed. Many control actions may be required to properly actuate one of these systems; not all are discussed below.

HPCI Actuation

The HPCI system is actuated by a low water level (LL2) signal or a high drywell pressure signal. Control signals are provided to both the turbine steam stop and the pump discharge valves.

The actuation control circuitry is powered from 125 Vdc and consists of a single-logic train that monitors reactor vessel level and containment atmospheric pressure. Four channels of each are provided and arranged so that one trip of the A or B channels and one trip of the C or D channels will cause a system actuation.

RCIC Actuation

The RCIC system is actuated by a reactor low water level (LL2) signal. Control signals are provided to both the turbine steam supply and the pump discharge valves.

The level signal causing actuation corresponds to a level of 118 inches relative to RPS instrument zero, and originates from the level indicating master trip units.

The actuation control circuitry is powered from 125 Vdc and consists of a single-logic train that monitors reactor vessel level. Four channels are provided and arranged so that one trip of the A or B channels and one trip of the C or D channels will cause a system actuation.

ADS Actuation

The ADS system is actuated by reactor vessel low water level (LL1 and LL3) signals at two distinct setpoints if low pressure core cooling is available.

After receipt of initiation signals, and after a delay provided by timers, the solenoid-operated air valves open, allowing nuclear system depressurization. The logic scheme for system actuation is a single trip system containing two logics, each logic of which can initiate automatic depressurization.

LPCI Actuation

The LPCI is actuated by low water level (LL3) in the reactor pressure vessel or high drywell pressure if the high drywell pressure is also accompanied by a low reactor vessel pressure signal.

The actuation control circuitry consists of two logic trains, each powered from a separate dc power supply. Successful actuation requires a signal from only one of the two independent, redundant logic trains. Each train monitors four independent channels of reactor vessel pressure, drywell pressure, and reactor vessel water level. Checks are also made of pump power supply availability. When one of the A or B channels and one of the C or D channels of level or pressure are tripped, a trip is successful for that particular segment. Actuation of the LPCI system depends on detecting a trip condition in the low reactor vessel water level network, or a trip in both the high drywell pressure and low reactor vessel pressure networks.

CSS Actuation

The CSS is actuated by low water level (LL3) in the reactor pressure vessel or high drywell pressure if the high drywell pressure is also accompanied by a low reactor vessel pressure signal.

The CSS logic scheme is comprised of one trip system per loop which actuates on receipt of sufficient low water signals or upon receipt of sufficient high drywell pressure signals and low reactor pressure signals. The same sensors actuate the trip systems for Loop A and Loop B using isolated relay contacts for isolation between trip systems.

ECCS Actuation Mission Success

ECCS actuation is successful if an actuation signal is provided when needed for each of the subsystems of the ECCS. Failure of the ECCS occurs if any component requiring an actuation fails to receive it. The following top level events were defined:

1. HPCI actuation to HPCI Turbine Steam Admission MOV F001 fails (HPC-ICC-FA-F001),
2. HPCI actuation to HPCI Pump Discharge MOV F006 fails (HPC-ICC-FAa-F006),
3. RCIC actuation to valve RCIC Pump Discharge MOV F013 fails (RCI-ICC-FA-G0001),
4. RCIC actuation to valve RCIC Turbine Steam Admission MOV F045 fails (RCI-ICC-FA-G0002),
5. Actuation signal to Recirculation Loop to RHR Pump Suction valve F015A fails (RHR-ICC-FA-G0001),
6. Actuation signal to Recirculation Loop to RHR Pump Suction valve F015B fails (RHR-ICC-FA-G0002),
7. LPCI actuation to RHR Pump C002A fails (RHR-ICC-FA-G0003),
8. LPCI actuation to RHR Pump C002B fails (RHR-ICC-FA-G0004),
9. LPCI actuation to RHR Pump C002C fails (RHR-ICC-FA-G0005),
10. LPCI actuation to RHR Pump C002D fails (RHR-ICC-FA-G0006),
11. Actuation signal to Core Spray Isolation MOV E21-F005A fails (CSS-ICC-FA-G0001),
12. Actuation signal to Core Spray Isolation MOV E21-F005B fails (CSS-ICC-FA-G0002),
13. CSS actuation of Core Spray Pump E21-C001A fails (CSS-ICC-FA-G0003), and
14. CSS actuation of Core Spray Pump E21-C001B fails (CSS-ICC-FA-G0004).

Note: Only the following failure modes contributed to the dominant accident sequences

1. <u>HPCI Actuation to HPCI Pump Discharge Isolation Valve F006 Fails Due to Failure of HPCI Turbine Stop Valve V8 or HPCI Turbine Steam Admission Valve F001 Permissives</u>

HPCI flow injection depends upon successful opening of valve F006. Failure of the V8 or F001 permissives prevents F006 from opening (PC,PT,MT).

2. <u>RCIC Actuation to RCIC Pump Discharge Isolation Valve F013 Fails Due to Failure of Permissives for Stop Valve or RCIC Turbine Steam Admission Valve F045 or Faults in K3 Relay</u>

RCIC flow injection depends upon successful opening of valve F013. Failure of the permissives for the RCIC turbine stop valve or valve F045, or faults in the K3 relay, prevent F013 from opening (PC,PT,MT).

Emergency Core Cooling System (ECCS) Actuation

TABLE A.14-2 MODIFIED SYSTEM WALKDOWN

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
HPCI Turbine Stop Valve	E41-V8	Control Room Panel P601	Closed					
HPCI Injection Valve	E41-F006	Control Room Panel P601	Closed		Comp. No. B17	Reactor Bldg. MCC 2XDA El. 20'	On	
					4 Motor Htr. Circuit Breaker	120V AC Dist. Pnl. 2DXA El. 17'	On	
Turbine Steam Supply Valve	E41-F001	Control Room Panel P601	Closed		Comp. No. B21	Reactor Bldg. MCC 2XDA El. 20'	On	
					8 Motor Htr. Circuit Breaker	Reactor Bldg. Miscellaneous 120V AC Dist. Pnl. 2DXA El. 17'	On	

Emergency Core Cooling System (ECCS) Actuation

TABLE A.14-2 MODIFIED SYSTEM WALKDOWN (Cont'd)

Description	ID No.	Location	Desired Position	Actual Position	Pow. Sup. Breaker #	Location	Required Position	Actual Position
Turbine Steam Supply Valve*	E51-F045	Control Room RTGB Panel H12-P601	Closed		Comp. No. B44 Normal/ Local Contl. Sw.	Reactor Bldg. 250V MCC 2XDB El. 20' South	Normal	
					B44 Circuit Breaker	Reactor Bldg. 250V MCC 2XDB El. 20' South	On	
					8 Motor Htr. Circuit Breaker	Reactor Bldg. Miscellaneous 120V AC Dist. Pnl. 2DXB El. 17' South	On	
RCIC Injection Valve	E51-F013	Control Room RTGB Panel H12-P601	Closed		Comp. No. B41 Normal/ Local Contl. Sw.	Reactor Bldg. 250V MCC 2XDB El. 20' South	Normal	
					B41 Circuit Breaker	Reactor Bldg. 250V MCC 2XDB El. 20' South	On	
					5 Motor Htr. Circuit Breaker	Reactor Bldg. Miscellaneous 120V AC Dist. Pnl. 2DXB El. 17' South	On	
RCIC Stop Valve Permissive*								
K-3 Relay*								

*Review latest surveillance tests on permissives logic for valves F045 and RCIC stop valve, and test of relay K-3.

TABLE A.14-2 (Cont'd)

REFERENCE DOCUMENTS

TITLE	I.D. NO.	REV	DATE
System Procedures:			
Reactor Core Isolation Cooling (RCIC)	OP-16	53	8/23/88
High Pressure Coolant Injection (HPCI)	OP-19	60	8/11/88
System Description:			
Brunswick PRA Volume I	Section M.3.13		
"Emergency Core Cooling System Actuation"			
P&ID's No.:			
Brunswick PRA Volume I	Fig. M.3.4-25		
"ECCS Actuation Diagram"	Sheets 1,2,3,4		

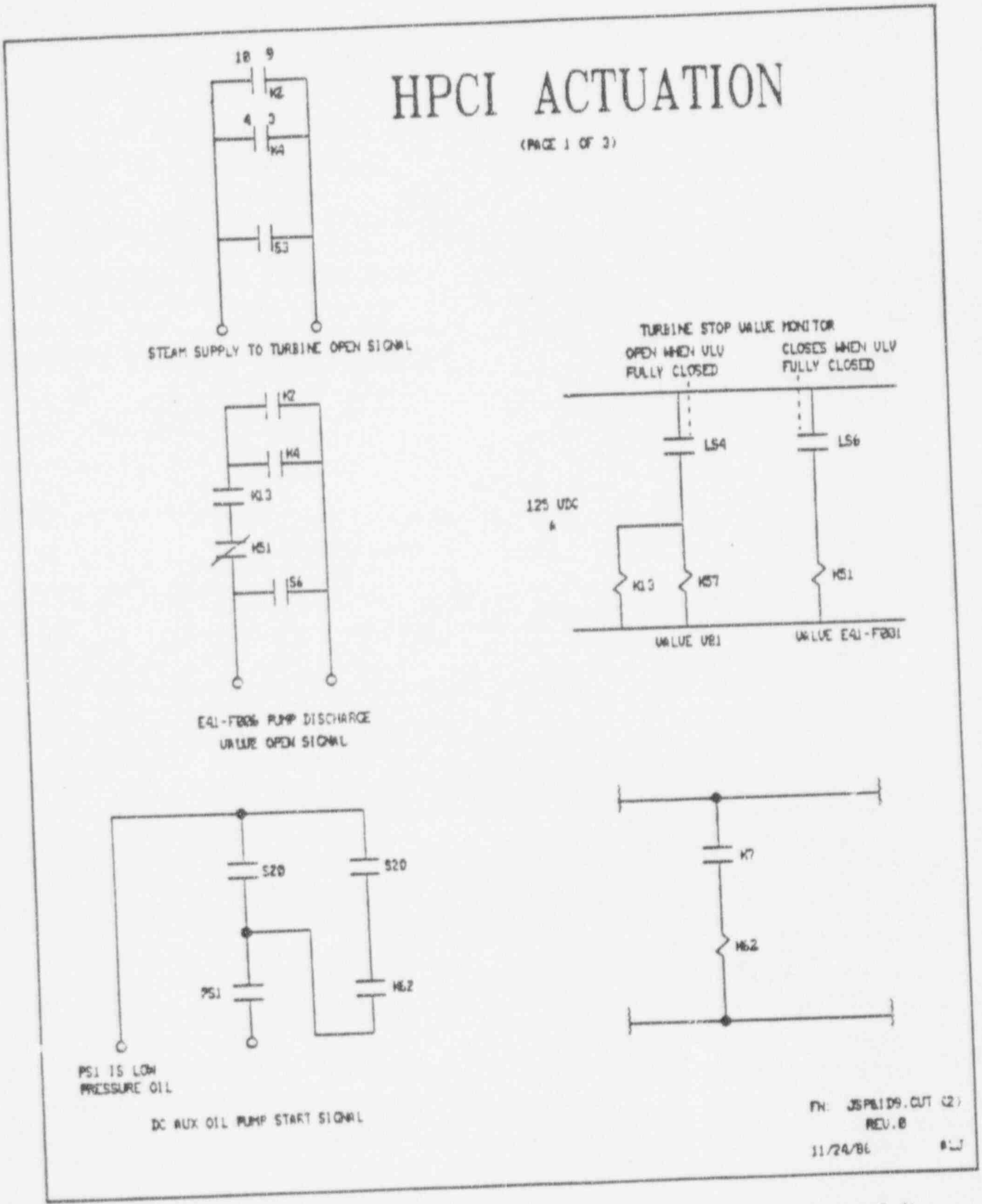

Fig. A.14-1 (PRA Fig. M.3.4-25). ECCS Actuation Diagram (Sheet 1 of 4).
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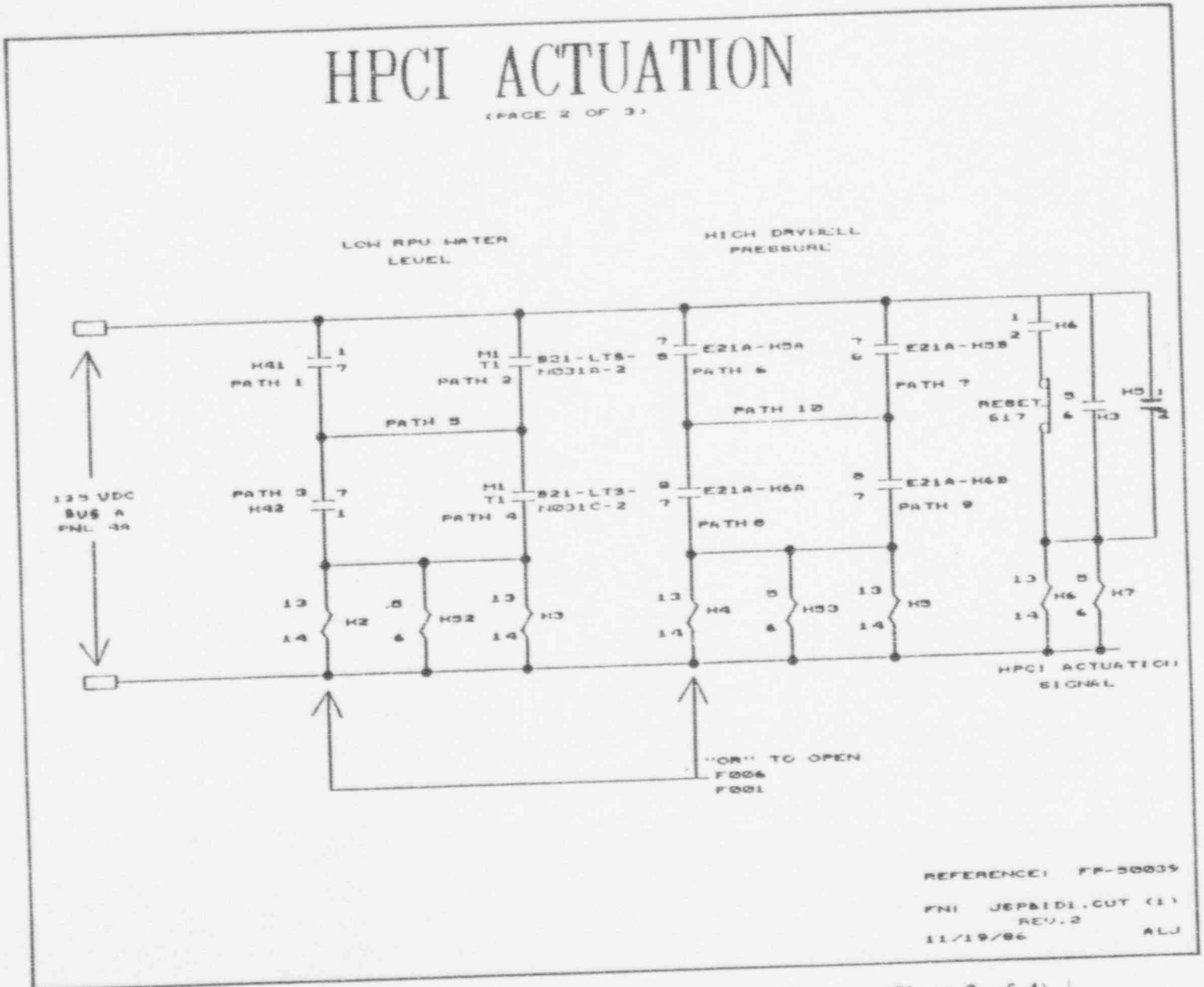

Fig. A.14-1 (PRA Fig. M.3.4-25). ECCS Actuation Diagram (Sheet 2 of 4).
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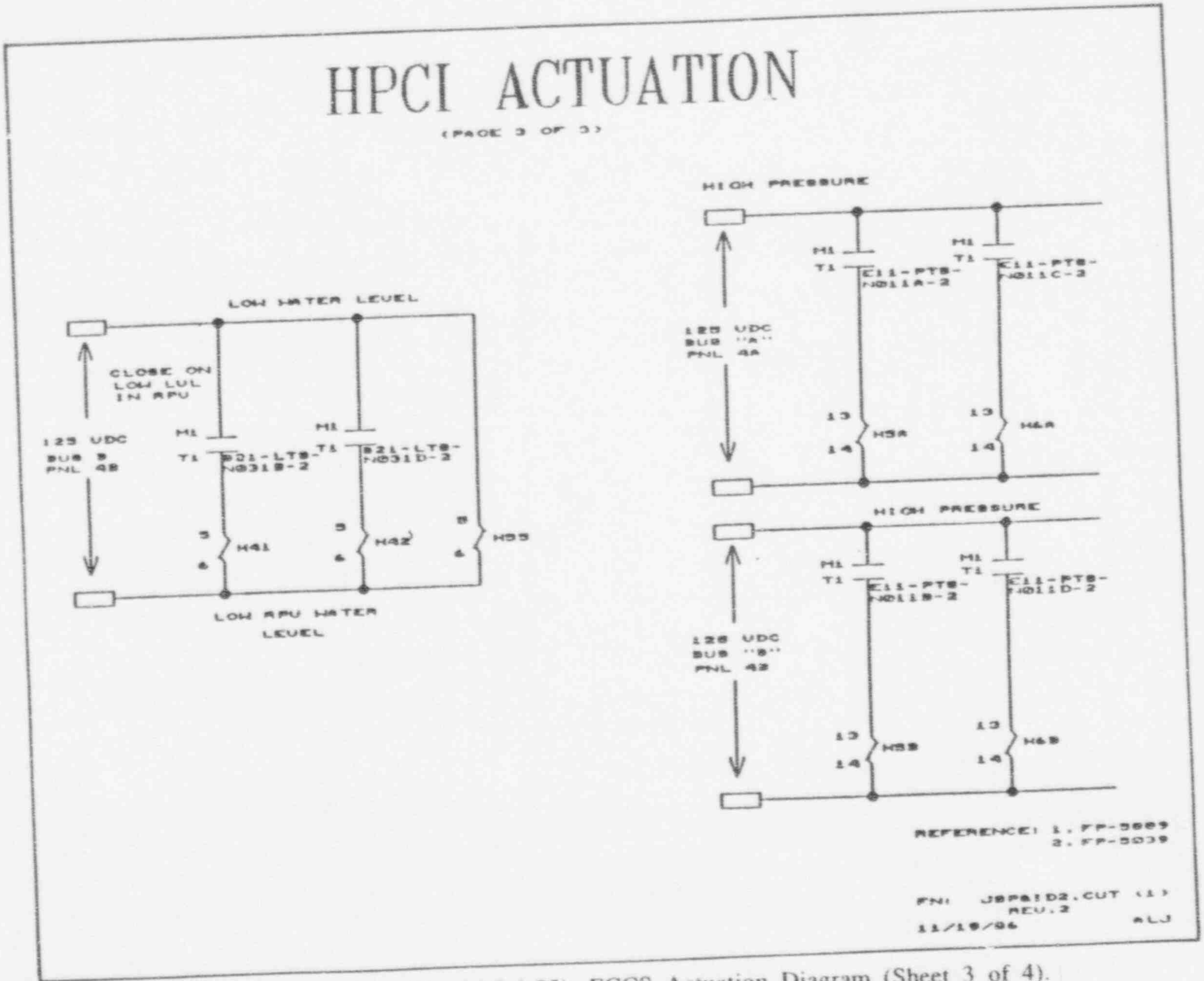

Fig. A.14-1 (PRA Fig. M.3.4-25). ECCS Actuation Diagram (Sheet 3 of 4).
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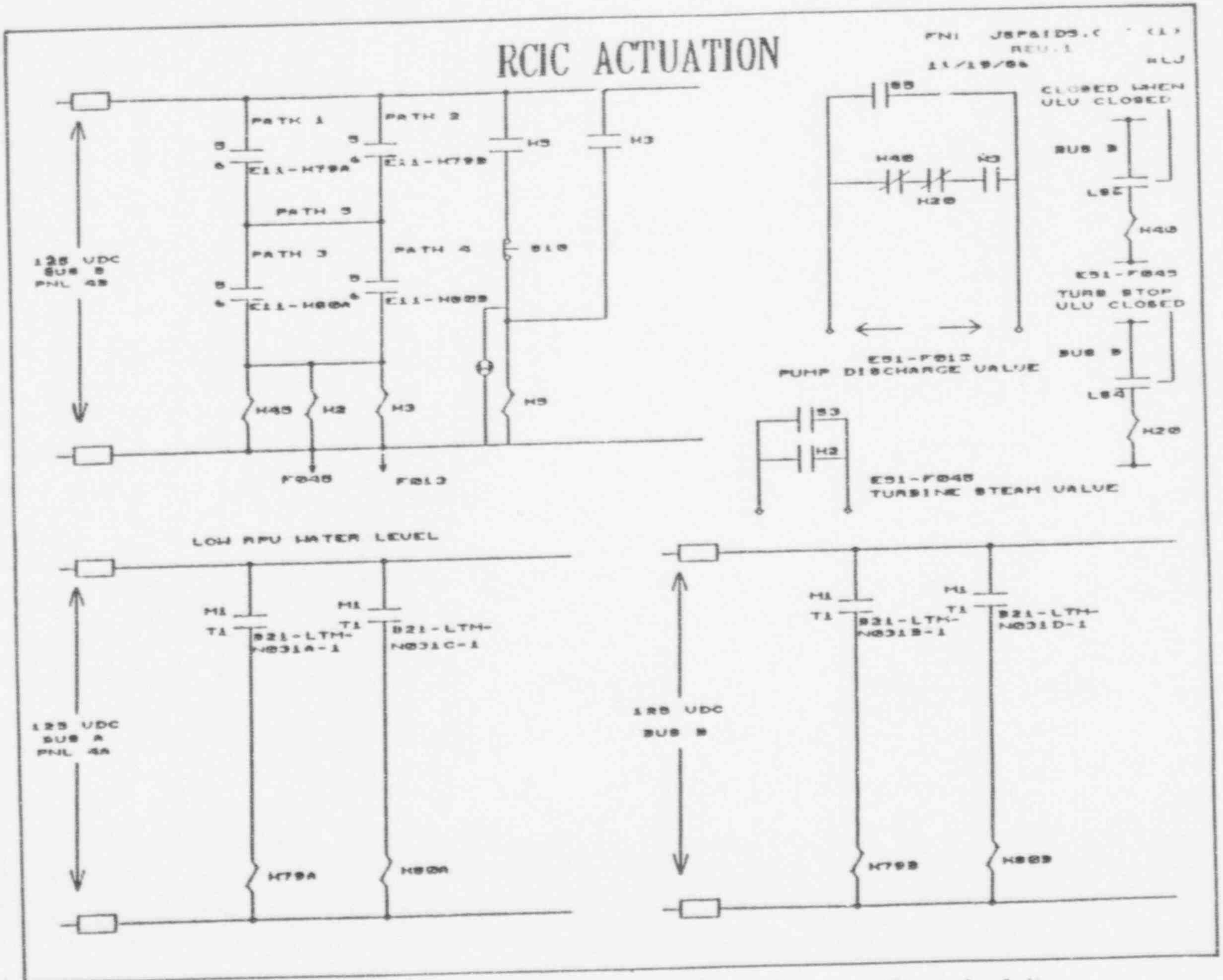

Fig. A.14-1 (PRA Fig. M.3.4-25). ECCS Actuation Diagram (Sheet 4 of 4).
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APPENDIX B

Plant Operations, Surveillance and Calibration, and Maintenance Inspection Tables

TABLE B.1 PLANT OPERATIONS INSPECTION GUIDE

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

Transient or System	Failure	Discussion
Anticipated Transient Without Scram (ATWS)	ATWS with Isolation (MSIV Closure) and Operator Failure to Control RPV Water Level at High or Low Pressure	Section 2, Par. 2.1
	ATWS With Isolation (MSIV Closure) and Operator Failure to Inhibit ADS (with HPCI Operational) and Failure to Control RPV Water Level at Low Pressure	Section 2, Par. 2.1
Reactor Core Isolation Cooling (RCIC)	Operator Fails to Override Steam Tunnel High Temperature	Table A.12-1, Item 3
High Pressure Coolant Injection (HPCI)	Operator Fails to Override Steam Tunnel High Temperature Trip	Table A.2-1, Item 4
	Operator Fails to Empty Drain Pot "A"	Table A.2-1, Item 10
Diesel Generator and Switchgear Cell Ventilation	Operator Fails to Open Switchgear Rooms After an HVAC Failure	Table A.4-1, Item 2
Standby Liquid Control (SLC) System	Operator Fails to Actuate the SLC System	Table A.5-1, Item 1
Residual Heat Removal (RHR) System	I&C for Mini-Flow Recirulation Line MOVs F007A and F007B Not Restored or PDIS N021A and N021BM	Table A.8-1, Item 4
	Operator Fails to Correctly Initiate Suppression Pool Cooling-Loops A and B	Table A.8-1, Item 10
Control Rod Drive (CRD) Hydraulic System	Operator Fails to Fully Open Valve F002A	Table A.13-1, Item 1
	Operator Fails to Fully Open Valve F003	Table A.13-1, Item 2

TABLE B.2 SURVEILLANCE AND CALIBRATION INSPECTION GUIDANCE

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

Transient or System	Failure	Discussion
Emergency Diesel Generators (EDG)	Diesel Generators 1, 2, 3 or 4 Fail to Start or Run	Table A.1-1, Item 1
	Diesel Generators 1, 2, 3 or 4 in Test or Maintenance	Table A.1-1, Item 2
	Diesel Generators 1, 2, 3 or 4 Generator Output or Output Breaker Failure	Table A.1-1, Item 3
High Pressure Coolant Injection (HPCI)	HPCI Turbine Fails to Start or Run	Table A.2-1, Item 1
	HPCI Pump or Turbine in Test or Maintenance	Table A.2-1, Item 2
	Failure of Level Switch	Table A.2-1, Item 3
	Normally Closed HPCI Pump Discharge Isolation MOV F006 Fails Closed	Table A.2-1, Item 4
	Delayed Actuation Signal to MOV F006 on Scram Coupled with Insufficient Flow Through the Mini-Flow Line Due to MOV F012 Failing to Open	Table A.2-1, Item 6
	Normally Closed Steam Supply Isolation MOV F001 Fails to Open	Table A.2-1, Item 7
	Failure of Lube Oil Cooling	Table A.2-1, Item 8
	Failure of the Vacuum Breaker on Startup Causes Insufficient Steam Flow from the Turbine	Table A.2-1, Item 9
	Insufficient Flow from HPCI Pump Discharge Line	Table A.2-1, Item 10
	Failures in Pipe Segment Leading from Drain Pot A to Barometric Condenser	Table A.2-1, Item 12

TABLE B.2 SURVEILLANCE AND CALIBRATION INSPECTION GUIDANCE (Cont'd)

Transient or System	Failure	Discussion
Automatic Depressurization System (ADS)	O-Ring Leakage or Other Dependent Failure Mechanisms Cause Two or More ADS SRVs Failing to Open	Table A.3-1, Item 1
Diesel Generator and Switchgear Cell Ventilation	Control Circuit Fails to Provide Actuation Signal to Diesel Generator Cells 1 and 2 Supply and Recirculation Dampers	Table A.4-1, Item 6
Standby Liquid Control (SLC)	Normally Open Reactor Water Cleanup MOV G31-F004 Fails to Close	Table A.5-1, Item 2
	Faults in the SLC Injection Line from the Squib Valves to the Reactor Vessel	Table A.5-1, Item 3
	Plugging of Locked Open Manual Valve F001 at the SLC Tank	Table A.5-1, Item 4
	SLC Pumps A and B in Test or Maintenance	Table A.5-1, Item 5
	SLC Pumps Fail to Start or Run	Table A.5-1, Item 6
	SLC Pumps Bypass Relief Valves F029A and F029B Fail to Close	Table A.5-1, Item 7
Service Water	Dependent, or Common Cause, Failure—4 or More SWS Pumps Fail to Run	Table A.7-1, Item 1
	Loop A or Loop B RHR Heat Exchanger is Unavailable Due to Test or Maintenance	Table A.7-1, Item 2
	Loop A or Loop B RHR Heat Exchanger Fails Due to Plugging	Table A.7-1, Item 3
	Diesel Generator 1, 2, 3 or 4 Hat Exchanger is Unavailable Due to Test or Maintenance	Table A.7-1, Item 4
	Diesel Generator 1, 2, 3 or 4 Heat Exchanger is Unavailable Due to Plugging	Table A.7-1, Item 5
	Normally Closed RHR Heat Exchanger Suction Header Isolation MOVs V105 and V101 Fail to Open	Table A.7-1, Item 6
	Conventional Pump 2C Unavailable Due to Test or Maintenance	Table A.7-1, Item 8

TABLE B.2 SURVEILLANCE AND CALIBRATION INSPECTION GUIDANCE (Cont'd)

Transient or System	Failure	Discussion
Residual Heat Removal (RHR)	RHR Loop A or Loop B Unavailable Due to Test or Maintenance	Table A.8-1, Item 1
	RHR Heat Exchanger Bypass MOVs F048A and F048B Fail to Close When Required	Table A.8-1, Item 2
	RHR Mini-Flow Recirculation Line MOVs F007A and F007B Fail to Close When Required	Table A.8-1, Item 3
	I&C for Mini-Flow Recirculation Line MOVs F007A and F007B Not Restored or PDIS N021A and N021B Miscalibrated	Table A.8-1, Item 4
	Faults in Pressure Transmitters for MOVs F007A and F007B	Table A.8-1, Item 5
	Suppression Pool Strainers S1 and S2 Fail Due to Plugging	Table A.8-1, Item 6
	RHR Injection Header to Suppression Pool Isolation MOVs F028A and F028B, F027A and F027B, or F024A and F024B Fail to Open or Fail to Close When Required	Table A.8-1, Item 7
	Loops A and B RHR Pumps C002A, B, C, D Fail to Start or Run	Table A.8-1, Item 8
	Common Cause Failure of RHR Heat Exchangers Due to Plugging	Table A.8-1, Item 9
DC Power	Fault in Battery Bank 2B-1 or in Output Breaker	Table A.9-1, Item 1
	Fault in Battery Bank 2A-2 or in Output Breaker	Table A.9-1, Item 2
	Fault in Panel 4A or in the Supply Breaker	Table A.9-1, Item 3
	Fault in Panel 4B or in the Supply Breaker	Table A.9-1, Item 4
	Fault in Distribution Switchboard 2A-P	Table A.9-1, Item 5
	Fault in Distribution Switchboard 2B-N	Table A.9-1, Item 6

TABLE B.2 SURVEILLANCE AND CALIBRATION INSPECTION GUIDANCE (Cont'd)

Transient or System	Failure	Discussion
	Fault in Battery Bank 2A-1 or in the Output Breaker	Table A.9-1, Item 7
	Fault in MCC 2XDA or in Feeder from 2A-P or 2A-N	Table A.9-1, Item 8
AC Power	Failure to Recover Offsite Power Within One Half Hour	Table A.10, Item 1
	480V Bus E7(E8) Fault or 4160/408 Transformer Failure or Bus E7(E8) Feeders Fail Open	Table A.10-1, Items 2–3
	480 VAC MCC 2CA(2CB) Fault or Feeder Fails Open	Table A.10-1, Items 4–5
	Complete or Partial Loss of Offsite Power Cause by: a) Feeder to Bus E3 Fails Open Due to I&C Related or Other Faults b) Startup Auxiliary Transformer Fails c) Feeder from SAT to Bus 2D Fails Open Due to I&C Related or Other Faults d) 4160 VAC Bus 2D Fault	Table A.10-1, Item 9
	Complete or Partial Loss of Offsite Power Caused by: a) Feeder to Bus E4 Fails Open Due to I&C Related or Other Faults b) Startup Auxiliary Transformer Fails c) Feeder from SAT to Bus 2C Fails Open Due to I&C Related or Other Faults d) 4160 VAC Bus 2C Fault	Table A.10-1, Item 10
	480 VAC MCC 2XA(2XB) Fault or Feeder Fails Open	Table A.10-1, Items 11–12
	480 VAC MCC 2XA-2(2XB-2) Fault or Feeder Fails Open	Table A.10-1, Items 13–14
	Power Not Available from Bus E1(E2)	Table A.10-1, Items 15–16

TABLE B.2 SURVEILLANCE AND CALIBRATION INSPECTION GUIDANCE (Cont'd)

Transient or System	Failure	Discussion
Battery Room Fans and Heaters	Battery Rooms 2A and 2B Supply and Exhaust Dampers Fail Closed Due to Faults in the Actuation Control for the Vortex Dampers	Table A.11-1, Item 2
	Failure of Thermostats for Battery Rooms 2A and 2B Duct Heaters Causing Overcooling	Table A.11-1, Item 3
Reactor Core Isolation Cooling (RCIC)	RCIC Pump Fails to Start or Run	Table A.12-1, Item 1
	RCIC Pump in Test or Maintenance	Table A.12-1, Item 2
	Faults in RCIC System Prevent RCIC Pump Discharge Flow	Table A.12-1, Item 4
	Failure of Lube Oil Cooling	Table A.12-1, Item 5
	Failure of Vacuum Breaker on Startup Causes Insufficient Steam Flow from the Turbine	Table A.12-1, Item 6
ECCS Actuation	HPCI Actuation to Valve F006 Fails Due to Failure of V8 or F001 Permissives	Table A.14-1, Item 1
	RCIC Actuation to Valve F013 Fails Due to Failure of Permissives for Stop Valve or F045 or Faults in K3 Relay	Table A.14-1, Item 2

TABLE B.3 MAINTENANCE INSPECTION GUIDANCE

Transient or System	Failure	Discussion
Emergency Diesel Generators (EDG)	Diesel Generators 1, 2, 3 or 4 Fail to Start or Run	Table A.1-1, Item 1
	Diesel Generators 1, 2, 3 or 4 in Test or Maintenance	Table A.1-1, Item 2
	Diesel Generators 1, 2, 3 or 4 Generator Output or Output Breaker Failure	Table A.1-1, Item 3
High Pressure Coolant Injection (HPCI)	HPCI Pump or Turbine Fails to Start or Run	Table A.2-1, Item 1
	HPCI Pump or Turbine in Test or Maintenance	Table A.2-1, Item 2
	Failure of Level Switch LEN014	Table A.2-1, Item 3
	Normally Closed HPCI Pump Discharge Isolation MOV F006 Fails Closed	Table A.2-1, Item 5
	Delayed Actuation Signal to MOV F006 on Scram Coupled with Insufficient Flow Through the Mini-Flow Due to MOV F012 Failing to Open	Table A.2-1, Item 6
	Normally Closed Steam Supply Isolation MOV F001 Fails to Open	Table A.2-1, Item 7
	Failure of Lube Oil Cooling	Table A.2-1, Item 8
	Failure of the Vacuum Breaker on Startup Causes Insufficient Steam Flow from the Turbine	Table A.2-1, Item 9
	Insuffiecient Flow from HPCI Pump Discharge Line	Table A.2-1, Item 10
	Failures in Pipe Segment Leading from Drain Pot "A" to Barometric Condenser	Table A.2-1, Item 12
Automatic Depressurization (ADS)	O-Ring Leakage or Other Dependent Failure Mechanisms Cause Two or More ADS SRVs Failing to Open	Table A.3-1, Item 1

TABLE B.3 MAINTENANCE INSPECTION GUIDANCE (Cont'd)

Transient or System	Failure	Discussion
Diesel Generator and Switchgear Cell Ventilation	Supply and Recirculation Damper Faults for Diesel Generator Cells 1, 2, 3 or 4	Table A.4-1, Item 1
	Insufficient HVAC Flow Through Switchgear Rooms E3 and E4 Due to Supply or Exhaust Damper Faults or Unavailability of the Instrument Air Supply	Table A.4-1, Item 3
	Exhaust Damper Faults for Diesel Generator Cells 1, 2, 3 or 4	Table A.4-1, Item 3
	Power Not Available to DG Cells 1 and 2 Exhaust Fans from MCC DGA and MCC DGB	Table A.4-1, Item 5
	Control Circuit Fails to Provide Actuation Signal to DG Cells 1 and 2 Supply and Recirculation Dampers	Table A.4-1, Item 6
Standby Liquid Control (SLC)	Normally Open Reactor Water Cleanup MOV G31-F004 Fails to Close	Table A.5-1, Item 2
	Faults in the SLC Injection Line from the Squib Valves to the Reactor Vessel	Table A.5-1, Item 3
	Plugging of Locked Open Manual Valve F001 at the SLC Tank	Table A.5-1, Item 4
	SLC Pumps A and B in Test or Maintenance	Table A.5-1, Item 5
	SLC Pumps A and B Fail to Start or Run	Table A.5-1, Item 6
	SLC Pumps Bypass Relief Valves F029A and F029B Fail to Close	Table A.5-1, Item 7
Service Water (SWS)	Dependent, or Common Cause, Failure—4 or More SWS Pumps Fail to Run	Table A.7-1, Item 1
	Loop A or Loop B RHR Heat Exchanger is Unavailable Due to Test or Maintenance	Table A.7-1, Item 2
	Loop A or Loop B RHR Heat Exchanger Fails Due to Plugging	Table A.7-1, Item 3

TABLE B.3 MAINTENANCE INSPECTION GUIDANCE (Cont'd)

Transient or System	Failure	Discussion
	DG 1, 2, 3 or 4 Hat Exchanger is Unavailable Due to Test or Maintenance	Table A.7-1, Item 4
	DG 1, 2, 3 or 4 Heat Exchanger is Unavailable Due to Plugging	Table A.7-1, Item 5
	Normally Closed RHR Heat Exchanger Suction Header Isolation MOVs V105 and V101 Fail to Open	Table A.7-1, Item 6
	Insufficient Flow Through Screens 1A, 1B, 2A and 2B Due to Failure of Screen Drives or Other Faults	Table A.7-1, Item 7
	Conventional Pump 2C Unavailable Due to Test or Maintenance	Table A.7-1, Item 8
Residual Heat Removal (RHR)	RHR Loop A or Loop B Unavailable Due to Test or Maintenance	Table A.8-1, Item 1
	RHR Heat Exchanger Bypass MOVs F048A and F048B Fail to Close When Required	Table A.8-1, Item 2
	RHR Mini-Flow Recirculation Line MOVs F007A and F007B Fail to Close When Required	Table A.8-1, Item 3
	I&C for Mini-Flow Recirculation Line MOVs F007A and F007B Not Restored or PDIS N021A and N021B Miscalibrated	Table A.8-1, Item 4
	Faults in Pressure Transmitters for MOVs F007A and F007B	Table A.8-1, Item 5
	Suppression Pool Strainers S1 and S2 Fail Due to Plugging	Table A.8-1, Item 6
	RHR Injection Header to Suppression Pool Isolation MOVs F028A and F028B, F027A and F027B, or F024A and F024B Fail to Open or Fail to Close When Required	Table A.8-1, Item 7
	Loop A and Loop B RHR Pumps C002A, B, C and D Fail to Start or Run	Table A.8-1, Item 8

TABLE B.3 MAINTENANCE INSPECTION GUIDANCE (Cont'd)

Transient or System	Failure	Discussion
	Common Cause Failure of RHR Heat Exchangers Due to Plugging	Table A.8-1, Item 9
DC Power (DCP)	Fault in Battery Bank 2B-1 (2A-2) or in Output Breaker	Table A.9-1, Items 1–2
	Fault in Panel 4A(4B) or in the Supply Breaker	Table A.9-1, Items 3–4
	Fault in Distribution Switchboard 2A-P (2B-N)	Table A.9-1, Items 5–6
	Fault in Battery Bank 2A-1 or in the Output Breaker	Table A.9-1, Item 7
	Fault in MCC 2XDA or in the Feeder from 2A-P or 2A-N	Table A.9-1, Item 8
AC Power (ACP)	Failure to Recover Offsite Power Within One Half Hour	Table A.9-1, Item 1
	480V Bus E7(E8) Fault or 4160/408 Transformer Failure or Bus E7(E8) Feeders Fail Open	Table A.9-1, Items 2–3
	480VAC MCC 2CA(2CB) Fault or Feeder Fails Open	Table A.9-1, Items 4–5
	Fault in 4160V E3(E4) Bus	Table A.9-1, Items 6–7
	Power Fails for Battery Rooms 2A and 2B Duct Heaters	Table A.9-1, Item 8
	Complete or Partial Loss of Offsite Power Caused by: a) Feeder to Bus E3(E4) Fails Open Due to I&C Related or Other Faults b) Startup Auxiliary Transformer Fails c) Feeder from SAT to Bus 2D(2C) Fails Open d) 4160 VAC Bus 2D(2C) Fault	Table A.9-1, Items 9–10
	480 VAC MCC 2XA(2XB) Fault or Feeder Fails Open	Table A.10-1, Items 11–12
	480 VAC MCC 2XA-2(2XB-2) Fault or Feeder Fails Open	Table A.10-1, Items 13–14
	Power Not Available from Bus E1(E2)	Table A.9-1, Items 15–16

TABLE B.3 MAINTENANCE INSPECTION GUIDANCE (Cont'd)

Transient or System	Failure	Discussion
Battery Rooms Fans and Heaters	Battery Rooms 2A and 2B Supply and Exhaust Fans Fail to Run	Table A.11-1, Item 1
	Battery Rooms 2A and 2B Supply and Exhaust Dampers Fail Closed Due to Faults in the Actuation Control for the Vortex Dampers	Table A.11-1, Item 2
	Failure of Thermostats for Battery Rooms 2A and 2B Duct Heaters Causing Overcooling	Table A.11-1, Item 3
	Insufficient Supply Flow to Battery Rooms 2A and 2B Due to Restriction of the Intake Plenum Air Filters	Table A.11-1, Item 4
Emergency Core Cooling System (ECCS) Actuation	HPCI Actuation to Valve F006 Fails Due to Failure of V8 or F001 Permissives	Table A.14-1, Item 1
	RCIC Actuation to Valve F013 Fails Due to Failure of Permissives for Stop Valve or F045 or Faults in K3 Relay	Table A.14-1, Item 2
Reactor Core Isolation Cooling (RCIC)	RCIC Pump Fails to Start or Run	Table A.12-1, Item 1
	RCIC Pump in Test or Maintenance	Table A.12-1, Item 2
	Faults in RCIC System Prevent RCIC Pump Discharge Flow	Table A.12-1, Item 4
	Failure of Lube Oil Cooling	Table A.12-1, Item 5
	Failure of the Vacuum Breaker on Startup Causes Insufficient Steam Flow from the Turbine	Table A.12-1, Item 6

APPENDIX C

Containment and Drywell Walkdown

BRUNSWICK STEAM ELECTRIC PLANT—UNIT 2
RISK-BASED INSPECTION GUIDE

TABLE C.1 CONTAINMENT AND DRYWELL WALKDOWN

Discussion

Since the drywell is generally inaccessible during normal plant operation, those components listed in the preceding tables which are located either within the drywell or otherwise in the containment are listed below:

TABLE C.1 CONTAINMENT AND DRYWELL WALKDOWN

Description	ID No.	Location	Desired Position	Actual Position
1. HPCI System: Steam Supply Line MOV (Ref. DWG F-2945)	F002	Containment El. 38'-1'-6" Above Grating AZ. 165°	Open	———
2. ADS Solenoid Valves F013A AZ 30° "A"PSL F013C AZ 80° "B"PSL F013D AZ 80° "B"PSL F013H AZ 330° "D"PSL F013J AZ 280° "D"PSL F013K AZ 265° "C"PSL F013L AZ 100° "B"PSL (Ref. DWG F-2945)		Containment El. 44' AZ. See Left	Operable	——— ——— ——— ——— ——— ——— ———
3. SLC System: Inboard Injection Valve (Ref. DWG F-2945)	F008	Containment El. 38'-2' Above Grating AZ. 190°	Locked Above Open	Grating
4. RHR/LPCI: Suppression Pool Strainers (Inspect only if accessible)		Suppression Pool	Clear	———
Shutdown Cooling Isolation Valve	F067	Containment El. 34'- Overhead 17' Grating AZ. 180°	Locked Open	

TABLE C.1 CONTAINMENT AND DRYWELL WALKDOWN (Cont'd)

Description	ID No.	Location	Desired Position	Actual Position
RHR Injection Line Valves (Ref. DWG F-2943)	F060A	Containment El. 31'- Overhead 17' Grating AZ. 90°	Open	
	F060B	Containment El. 31'- Overhead 17' Grating AZ. 270°	Open	
5. RCIC System: Steam Supply Line MOV (Ref. DWG F-2943	F007	Containment El. 31'- Overhead 17' Grating AZ. 185°	Open	

APPENDIX D

System Dependency Matrix

SYSTEM DEPENDENCY MATRIX

TABLE D.1 FRONTLINE–SUPPORT SYSTEM DEPENDENCIES

Frontline System	Support System: AC Power Division 1	AC Power Division 2	DC Power A	DC Power B	SWS	RBCCW	Instrument Air	HVAC
HPCI	Note A	Note A	X	Note A				Note B
RCIC	Note A	Note A	Note A	X				Note B
CRD	X	X	Note A	Note A		X		Note B
ADS			X	X			Note C	
CSS	X	X	X	X				Note B
RHR	X	X	X	X				Note B
SLC	X	X						
DGs			X	X	X		Note D	X

Note A — Components depend upon this system for certain types of operation but not for the safety function modeled in the PRA.

Note B — HVAC analyses (BSEP PRA Appendix A.2) indicate that room cooling is not required during a 24-hour mission time, except possibly for low frequency LOCAs or high energy line breaks.

Note C — Even if instrument air noninterruptible fails, the ADS valves have accumulators and backup nitrogen bottles. Therefore, failure of air to ADS was considered to be negligible.

Note D — The DGs have their own compressed air system.

SYSTEM DEPENDENCY MATRIX

TABLE D.2 SUPPORT–SUPPORT SYSTEM DEPENDENCIES

Support System	Support System: AC Power Division 1	AC Power Division 2	DC Power A	DC Power B	SWS	RBCCW	Instrument Air	HVAC
AC Power	X	X	X	X				X
DC Power	Note A	Note A	X	X				
SWS	X	X	Note B	Note B	X			
RBCCW	X	X			X	X		
HVAC	X	X					Note C	X

Note A — If ac power is lost, batteries can supply dc power to necessary systems for at least four hours.
Note B — Only for certain SWS applications.
Note C — The DG cell HVAC dampers fail closed upon loss of air. (These dampers, unlike other emergency HVAC dampers, are connected to interruptible air rather than noninterruptible air.) To overcome this design deficiency, the dampers are maintained open such that they remain open if air is lost.