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Risk-Based Inspection Guide for Crystal River Unit 3 Nuclear Power Plant

Prepared by
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Pacific Northwest Laboratory
Operated by
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ABSTRACT

The Level 1 probabilistic risk assessment (PRA) for Crystal River Unit 3 (CR-3) has been analyzed to identify plant systems and components important to minimizing public risk, as measured by system contributions to plant core damage frequency, and to identify the primary failure modes of these components. The report presents a series of tables, organized by system and prioritized by risk importance, which identify components associated with 98% of the inspectable risk due to plant operation. The systems addressed, in descending order of risk importance are: Low Pressure Injection, AC Power, Service Water, Demineralized Water, High Pressure Injection, DC Power, Emergency Feedwater, Reactor Coolant Pressure Control, and Power Conversion. This ranking is based on the Fussell-Vesely measure of risk importance, i.e., the fraction of the total core damage frequency which involves failures of the system of interest.

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SUMMARY

The Risk-Based Inspection Guide for Crystal River Unit 3 (CR-3) was compiled for the U.S. Nuclear Regulatory Commission (NRC) at Pacific Northwest Laboratory (PNL). It is based upon a previously developed methodology for identification and presentation of information which is useful for the planning and performance of powerplant inspections.

The Level 1 probabilistic risk assessment (PRA) for CR-3 (Averett et al. 1987) has been analyzed to identify plant systems and components important to minimizing public risk, as measured by system contributions to plant core melt frequency. The body of this report consists of a series of tables, organized by system and prioritized by risk importance, which identify components associated with 98% of the core damage probability resulting from plant operation.

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

The tabulations were developed by the following analysis procedure. Plant systems were ordered according to system risk importance. To accomplish this, the dominant cut sets representing more than 98% of the core melt probability were listed, and the fraction of the total core damage probability which involved failures of components from each system was calculated (this is the Fussell-Vesely Importance measure). Systems were then selected from the ordered list until more than 98% of the core melt probability was accounted for. Within systems, components were then ranked by similar Fussell-Vesely Importance calculations using the dominant cut set elements.

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

The information presented in this document allows an inspector to concentrate his efforts on systems important to the prevention of core damage. However, it is essential that inspections not focus exclusively on these systems. Other systems which perform essential safety functions, but are absent from the tables because of high reliability and redundancy, must also be addressed to ensure that their importance is not increased by allowing their

reliability to decrease. A balanced inspection program is essential. This information represents but one of the many tools to be used by experienced inspectors.

ACKNOWLEDGMENTS

Thanks are extended to M. W. Averett of Florida Power Corporation, Project Manager of the Crystal River Unit 3 PRA, for information which he provided concerning the accident sequence and fault tree analysis results and many discussions during the performance of this analysis. This analysis was performed under sponsorship by the U.S. NRC, Technical Leader Dr. Steven Long, whom we wish to thank for his insights.

1.0 INTRODUCTION

Th document has been prepared to provide inspection guidance based on PNL's review of the Crystal River Unit 3 Probabilistic Risk Assessment (PRA) prepared jointly by Florida Power Corporation and Science Applications International Corporation (Averett et al. 1987). 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 derived from a revised listing of dominant cutsets produced by Florida Power Corporation during 1990 (Averett and Miskiewicz 1990). It therefore contains more current information than the reference document. Nevertheless, 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.**

2.0 DOMINANT ACCIDENT SEQUENCES

The Crystal River PRA identifies a number of different accident sequences that contribute significantly to overall core damage frequency (CDF). Based on the revised listing of cutsets supplied by Florida Power Corporation (Averett and Miskiewicz 1990), the total core damage frequency is 1.4E-5/year. The sequences that dominate this core damage frequency are identified below by their initiating events and the percentage of the total CDF which they represent.

- Steam Generator Tube Rupture (40%)
- Small Break LOCA (26%)
- Loss of Offsite Power (21%)
- Large Break LOCA (11%)
- Other Initiators (2%)

These and other accident sequences which contribute significantly to the CDF are described in more detail in the following subsections. These descriptions are based on the information provided in Reference 2.

2.1 STEAM GENERATOR TUBE RUPTURE

The steam generator tube rupture (SGTR) is a small break LOCA, but a LOCA which immediately bypasses containment, greatly increasing the probability of a release of radioactivity to the environment. Operator response to an SGTR must replenish the reactor coolant system (RCS) inventory being lost through the ruptured tubes and must depressurize the RCS and the secondary systems to bring the plant to a stable decay heat removal condition. One complication is that RCS inventory lost through the ruptured tubes does not collect in the containment sump, so the operator cannot initiate recirculation mode core cooling when the borated water storage tank (BWST) is depleted. Instead, action needs to be taken to stop the loss of RCS inventory through the ruptured tubes and/or to refill the BWST.

The SGTR core damage accident sequences fall into two classes: 1) an SGTR followed by failure of high pressure injection (HPI) system and 2) an SGTR with successful high pressure injection, but subsequent failure to depressurize and begin long term decay heat removal and/or failure to refill the BWST. The second class dominates the SGTR contribution to core damage risk.

2.2 SMALL BREAK LOCA

Following a small break LOCA, the immediate concern is to replace RCS inventory being lost out the break. Two major classes of core damage sequences are important: 1) a small break LOCA followed by early failure of high pressure injection and 2) a small break LOCA with successful high pressure injection, followed by failure of high pressure recirculation mode cooling of the core. The second class of sequences dominates the small break LOCA contribution to total CDF, because both the HPI and low pressure injection (LPI) systems must function correctly for successful high pressure recirculation cooling.

Because LPI is a two train system (as opposed to the three train HPI system), high pressure recirculation failures are dominated by common cause or independent failures of two different LPI components, each failing one of the two LPI trains.

2.3 LOSS OF OFFSITE POWER

Loss of Offsite Power (LOOP), as defined by the CR-3 PRA, is the loss of the 230 kV switchyard. A CR-3 LOOP leaves the emergency diesel generators (EDGs) as the only power source for the 4.16 kV Engineered Safeguards Buses 3A and 3B. The LOOP core damage sequences are all station blackout sequences. Most involve loss of offsite power, failure of both EDGs, and failure to recover offsite power in time to prevent core uncovery. For most such sequences, core heat removal through the steam generators (SGs) succeeds, using feedwater supplied by the steam-turbine-driven emergency feedwater pump. If offsite power is not restored within about four hours, then the loss of DC power due to battery depletion is assumed to cause loss of the ability to control SG feedwater level. Loss of level control is assumed to lead to either a dryout of the SG or overfill of the SG with water carryover into the emergency feedwater turbine steam lines.

2.4 LARGE BREAK LOCA

In a large break LOCA, the RCS depressurizes rapidly, allowing reflooding (temporarily) of the core by injection from the core flood tanks. The LPI system must operate successfully in injection mode, injecting borated water from the BWST. When the BWST is depleted, the operators must switch the source of LPI pump suction to the containment sump, initiating core cooling in recirculation mode (recirculating water from the sump, through the Decay Heat Removal system heat exchangers, to the RCS, and back out the break to the sump).

Core damage risk subsequent to a large break LOCA is dominated by failure of the operator to successfully switchover from low pressure injection to low pressure recirculation upon depletion of the BWST. Other important sequences involve separate, independent or common cause, failures of LPI components disabling both LPI trains.

2.5 OTHER INITIATORS

Core damage accident sequences beginning with other initiators constitute less than 2% of the total core damage risk and are not discussed in detail in this report.

2.6 IMPORTANT HUMAN ERRORS (Including Recovery Actions)

Human errors can be very significant to overall plant risk. Examination of the dominant cutsets from the Crystal River 3 PRA identified several human errors as particularly important contributors to risk.

2.6.1 Pre-Accident Errors

1. Miscalibration of the level transmitters of the EFW storage tank can prevent operators from recognizing loss of tank inventory, and thereby fail this source of EFW.

2. Misalignment closed of the surge tank outlet valve in either train of the Decay Heat Closed Cycle Cooling System (valves DCV-19 or DCV-20) may disable that service water train of heat removal. Although this is a low probability event, it could lead to loss of that train of HPI and LPI.

3. Latent errors during testing or maintenance could disable the pump in either train of the Decay Heat Seawater System (RWP-3A or RWP-3B), preventing cooling of the associated DHCCC heat exchanger. These pumps could also be disabled if the flush water valves were left unavailable. As above, these are low probability events, but they may result in disabling HPI and LPI pumps, and the associated LPI coolers.

4. Mispositioning the control of the standby HPI pump, preventing automatic start on demand, could result in system failure when the alternate HPI pump is unavailable due to maintenance, and the running pump is failed by a transient.

2.6.2 Post-Accident Errors

1. Failure of the crew to restore offsite power following a station blackout event has a high risk importance, since loss of both the motor-driven EFW pump and HPI results. The turbine-driven EFW pump fails two hours after the B diesel generator fails, due to depletion of the train B batteries, loss of emergency feedwater monitoring and control, and subsequent steam generator overfill or dryout. Core damage commences 50 minutes after loss of all HPI and EFW. Important event sequences include failure of both diesels on demand, resulting in core damage after 2 hours 50 minutes, and coincident failures on demand of the A diesel and the B battery, resulting in core damage after just 50 minutes.

2. Failure of the operators to refill the Borated Water Storage Tank following a steam generator tube rupture can lead to loss of HPI after BWST depletion, since no water accumulates in the reactor building sump for recirculation during this event.

3. Failure of the operators to provide makeup from the reactor coolant Bleed Tanks during an event where HPI is lost due to blockage of BWST suction or BWST failure could result in core uncovery and melting.

Other human errors are identified in the system failure mode tables.

2.7 COMMON CAUSE FAILURES

Based on the results of Crystal River 3 PRA (revised), the following common cause failures are identified to be important:

- Common Cause Failure of Service Water System

  Failure of the Service Water system due to common cause failures will prevent cooling from being provided to key front line equipment (e.g., makeup pumps). The important failure is the common mode failure of the standby Raw Water pumps 3A and 3B or Service Water pumps 1A and 1B to start and run under the emergency conditions.

- Common Cause Failure of DC Batteries

  The station batteries at Crystal River are also susceptible to common cause failure. The important common mode failures are either failure of the battery ventilation system or miscalibration of the battery chargers.

- Common Cause Failure of Makeup Pumps

  The important failure modes are pump failures to start or to run as required. Hardware failure is the dominant failure mode.

- Common Cause Failure of Emergency Feedwater Pumps

  Operation of the Emergency Feedwater system requires a number of important support systems, e.g., ac power, dc power, Emergency Feedwater Instrumentation and Control (EFIC), etc. These support systems are also used for other functions and support other systems. These dependencies make the pumps susceptible to common cause failure, despite the fact that one is turbine-driven and one is motor-driven. The important failure modes are pump failures to start or to run as required.

Other common cause failures, not considered to be as important as those identified above, are addressed in the system failure mode tables.

3.0 SYSTEM PRIORITY LIST

The Crystal River plant systems have been ranked in Table 3.1 according to their importance in preventing core damage. Two different rankings are provided for use under two types of circumstances. Under normal conditions, the left-hand column should be used. For degraded or inoperable systems, the right-hand column should be used, as discussed below. Plant systems not appearing on these lists are generally of lesser importance than those included.

TABLE 3.1. SYSTEM PRIORITY RANKING

By Contribution to Core Damage Frequency(a)	By Risk Significance of the System Being Unavailable(b)
Low Pressure Injection	
AC Power	AC Power
Human Operators	
----------(c)	----------
Service Water	
Demineralized Water	DC Power
High Pressure Injection	Low Pressure Injection
Chemical Addition	High Pressure Injection
----------	----------
DC Power	
RCS Pressure Control	Service Water
Power Conversion	Human Operators

(a) 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 system's contribution to the core damage frequency, assuming that the system is operating with normal reliability.

(b) 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.

(c) 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 PRA quantification. Systems which appear in Column 1, but not in Column 2, have Birnbaum Importances an order of magnitude or more lower than the six systems appearing in Column 2.

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

However, when one or more systems exhibit unusually high failure rates or unusual types of failures, then the probabilities assumed in the PRA are not really appropriate for the failures of those systems. While their problems persist, the affected systems contribute more to the risk of core damage than is indicated by the left-hand column. The increase in the core damage frequency when the system is inoperable is indicated by the right-hand column, based on the Birnbaum Importance Measure. The right-hand column can be used to estimate how much more important these systems have become when they are having problems. (Affected systems with high rankings in the right-hand column should be considered to have become much more important than indicated by their rank in the left-hand column, while systems with lower rankings in the right-hand column would have smaller increases above the rank indicated in the left-hand column.) Similarly the right-hand column is the appropriate choice for estimating the risk significance of inspection findings that indicate a system is inoperable or degraded.

Adjacent systems on the list should be considered to have approximately equal contributions to risk because of the uncertainties in the PRA. Where the difference between importance measures of adjacent systems is significant, they have been separated by the dashed lines.

4.0 SYSTEM INSPECTION TABLES

Tables are presented for each of the risk-important systems selected in the analysis. These tables identify important system failure modes, and the required position of each important component during normal system operation (i.e., system walkdown checklist). The systems are presented in decreasing order of Fussell-Vesely risk importance, and together comprise more than 98% of the risk associated with plant operation. To provide useful information for the inspector, simplified system drawings from the Crystal River PRA are reproduced at the end of each section.

4.1 LOW-PRESSURE INJECTION SYSTEM

The purposes of the Low Pressure Injection (LPI) system are to remove decay heat during normal shutdown and as an engineered safeguards system during plant transients and accidents. The LPI system consists of two redundant pump trains in parallel. The pumps can be aligned to receive suction directly from the primary coolant system through the decay heat removal drop line, from the borated water storage tank (BWST), or from the reactor building sump. Each pump discharges through a separate LPI line to the reactor vessel via one of the two core flood valves. Although cross over lines exist on both the suction and discharge sides of the pumps, the normal system configuration is to have both lines isolated.

During normal plant shutdown, the LPI system provides continual circulation of primary coolant water through the decay heat removal (RHR) heat exchangers to remove fission product decay heat and to maintain acceptable core temperatures. During plant transients, the LPI system provides emergency core cooling injection from the BWST to the primary system during the early stages of a large-break loss-of-coolant accident (LOCA) and recirculates water between the primary system and the reactor building sump to provide long-term core cooling. The LPI system can also provide suction from the reactor building sump to the high pressure injection pumps if required for long-term high pressure recirculation. The minimum functional requirement is one operating LPI pump and one intact flow train.

The LPI System is also known as the Decay Heat Removal (DHR) System. It is a standby system and it does not normally operate unless the plant is shut down. During shutdown, the system provides closed loop circulation of primary coolant water through one DHR pump and one DHR heat exchanger. DHR mode can only be actuated when the primary system pressure has been reduced to below 200 psi, since a primary system pressure interlock precludes opening of motor operated valves in the DHR drop line at pressures higher than 200 psi.

During an accident, the DHR pumps start automatically upon receipt of a high pressure injection actuation signal, but the LPI line isolation valves do

not open. The DHR pumps operate on minimum flow recirculation until the primary system pressure falls to the low pressure injection actuation signal setpoint, at which time the injection valves open and the pumps begin supplying ECC water to the primary system from the BWST. When the BWST level drops to the low level alarm point, the operator must manually transfer suction from the BWST to the reactor building sump. During this recirculation phase, discharge flow from the LPI pumps can be directed either directly to the primary system or to the suction of the HPI pumps.

The LPI system requires the following support functions provided by other systems:

- AC power for the DHR Pumps is provided by 4160 V ES Buses 3A and 3B.
- Cooling for DHR Pumps and DHR Heat Exchangers is provided by Decay Heat Closed Cycle Cooling.
- AC power for motor operated valves is provided by 480 V ES MCCs 3A1, 3A3, 3B1, and 3B3.
- DC power for AC circuit breakers is provided by 125 V DPDP-5A and DPDP-5B.
- Automatic actuation is provided by the Engineered Safeguards Actuation System.

TABLE 4.1A. LPI SYSTEM FAILURE MODE IDENTIFICATION

Conditions that Lead to Failure

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1. Failure of DHR Drop Line Valves DHV-3, DHV-4, OR DHV-41

This is the principal mode of failure for the decay heat removal function. During normal plant shutdown, failure of any one of the DHR drop line valves would prevent water from being removed from the RCS, cooled, and recirculated to the reactor core. During emergency operation, failure of the DHR dropline, in combination with the reactor building sump being unavailable (Item 7 below), would prevent water from being provided for long-term post-accident core cooling. The primary cause of failure of the drop line is the failure of any one of the motor-operated valves DHV-3, DHV-4, or DHV-41 to open on demand, due to random hardware failure or electrical failure. Another potential cause is valve control circuitry failure (i.e., failure of the RCS pressure interlock on DHV-3 and DHV-4). Power availability, operator training, awareness, and maintenance and testing of these valves should be reviewed or observed to maintain reliability.

2. Failure of the "Piggy-back" line from LPI discharge to HPI suction

Failure of both DHV-11 and DHV-12 causes failure of high pressure injection in "piggy-back" recirculation mode. Failure of one of the valves is recoverable by opening DHV-7 and DHV-8 (on the cross-connect between the two LPI discharge lines) to allow the either LPI pump to feed the remaining piggy-back line. In the event of concurrent make-up pump failures it might also be necessary to open valves on the cross-connecting header on the suction side of the make-up (i.e., HPI) pumps. To prevent failure of both valves, inspection should focus on maintenance of the valves and the associated controls and electrical supply. Training and procedures should address recovery actions for failures of one train of LPI or one train of HPI.

3. Failure of Valves DHV-42 and DHV-43 between Containment Sump and LPI Pump Suction

Failure of motor-operated valves DHV-42 and DHV-43 to open on demand will prevent water from being recycled from the containment sump back into the RCS. The important failure causes are random hardware or electrical failures. Power availability, maintenance and surveillance of these valves should be reviewed or observed to maintain reliability.

4. Failure of Decay Heat Pumps

Failure of the Low Pressure Injection Pumps DHP-1A and DHP-1B will prevent RCS inventory make-up under some circumstances (i.e., during the recirculation cooling phase of both high pressure and low pressure accident sequences). The dominant failure modes are failure of these pumps to start and run. When one of the pumps is in maintenance, the DHR system will be unable to provide water to the RCS or to the HPI pump suction, if concurrent failures disable the other LPI train and the cross-over headers. The important failure mechanisms are random hardware or electrical failures and human errors in following procedures to recover from failures. Operator training, awareness, surveillance and maintenance, including post-test surveillance of these pumps, should be reviewed or observed to maintain reliability.

5. Failure of BWST Valves DHV-34 and DHV-35

Failure to provide borated water to the reactor pressure vessel following a transient or accident may be caused either by valve failures or low level in the BWST. Failure of the motor-operated valve DHV-34 or check valve DHV-33 on train A combined with a failure of the motor-operated valve DHV-35 or check valve DHV-36 on train B results in borated water being unavailable for injection into the RCS. The dominant cause is failure of the valves DHV-34 and DHV-35 to open on demand. Operator awareness, maintenance and testing of the valves, as well as checking BWST level according to Technical Specifications should improve system reliability.

6. Failure of Valves in Decay Heat Removal Pump Discharge Lines

Failure of the motor-operated discharge valve DHV-5 on train A combined with the failure of discharge valve DHV-6 on train B will prevent water flow from being provided to the reactor vessel from the DHR pumps. The dominant mode is that DHV-5 and DHV-6 fail to open on demand. An additional failure cause is that the motor-operated valves transfer closed. Power availability, maintenance, and surveillance of these valves should be reviewed or observed to maintain reliability.

7. Reactor Building Sump Unavailable

Failure of both the motor-operated valve WDV-3 and the air-operated valve WDV-4 to close on demand may result in unavailability of the reactor building sump. This is one of the principle failure modes for the low pressure recirculation function of the LPI system. Another mode of the failure is for the sump screens to be plugged. The important failure causes are random hardware or electrical failures. Maintenance of these valves should be reviewed or observed to maintain reliability. Verification and review of check-off lists and of the emergency operating procedures requirements to close these valves should improve sump availability.

8. BWST Vacuum Breaker Fails

The failure of the BWST vacuum breakers, DHV-69 and DHV-70, to open when required can lead to failure of high pressure injection or low pressure injection due to loss of net positive suction head for the injection pumps, as a vacuum is drawn on the BWST by the injection pumps. The consequences of this failure could be failure of one or more injection pumps (if they continue to pump under cavitation conditions) or structural failure of the BWST itself. Common cause failures due to vendor or maintenance commonalities are the most important for these components. Inspectors should review maintenance records, procedures, and scheduling.

TABLE 4.1B. MODIFIED LPI SYSTEM WALKDOWN

Component Number	Component Name	Location	Required Position	Actual Position
	Electrical Components			
	Breaker on ES Bus 3A for Decay Heat Removal Pump DHP-1A		Racked In/ Closed	________
	Breaker on ES Bus 3B for Decay Heat Removal Pump DHP-1B		Racked In/ Closed	________
	Breakers for MOV DHV-3		Closed	________
	Breakers for MOV DHV-4		Closed	________
	Breakers for MOV DHV-5		Closed	________
	Breakers for MOV DHV-6		Closed	________
	Breakers for MOV DHV-34		Closed	________
	Breakers for MOV DHV-35		Closed	________
	Breakers for MOV DHV-41		Closed	________
	Breakers for MOV DHV-42		Closed	________
	Breakers for MOV DHV-43		Closed	________
	Breakers for MOV WDV-3		Closed	________
	Valves			
DHV-3	MO Decay Heat Discharge Valve		Closed	________
DHV-4	MO Decay Heat Discharge Valve		Closed	________
DHV-5	MO Decay Heat Suction Valve		Closed	________
DHV-6	MO Decay Heat Suction Valve		Closed	________
DHV-34	MO BWST Discharge Valve		Closed	________
DHV-35	MO BWST Discharge Valve		Closed	________
DHV-41	MO Decay Heat Discharge Valve		Closed	________
DHV 42	MO Containment Sump Discharge Valve		Closed	________

TABLE 4.1B. (contd)

Component Number	Component Name	Location	Required Position	Actual Position
DHV-43	MO Containment Sump Discharge Valve		Closed	
DHV-69	BWST Vacuum Breaker		Closed	
DHV-70	BWST Vacuum Breaker		Closed	
WDV-3	MO RB Sump Valve		Open	
WDV-4	Air-Operated RB Sump Valve		Open	

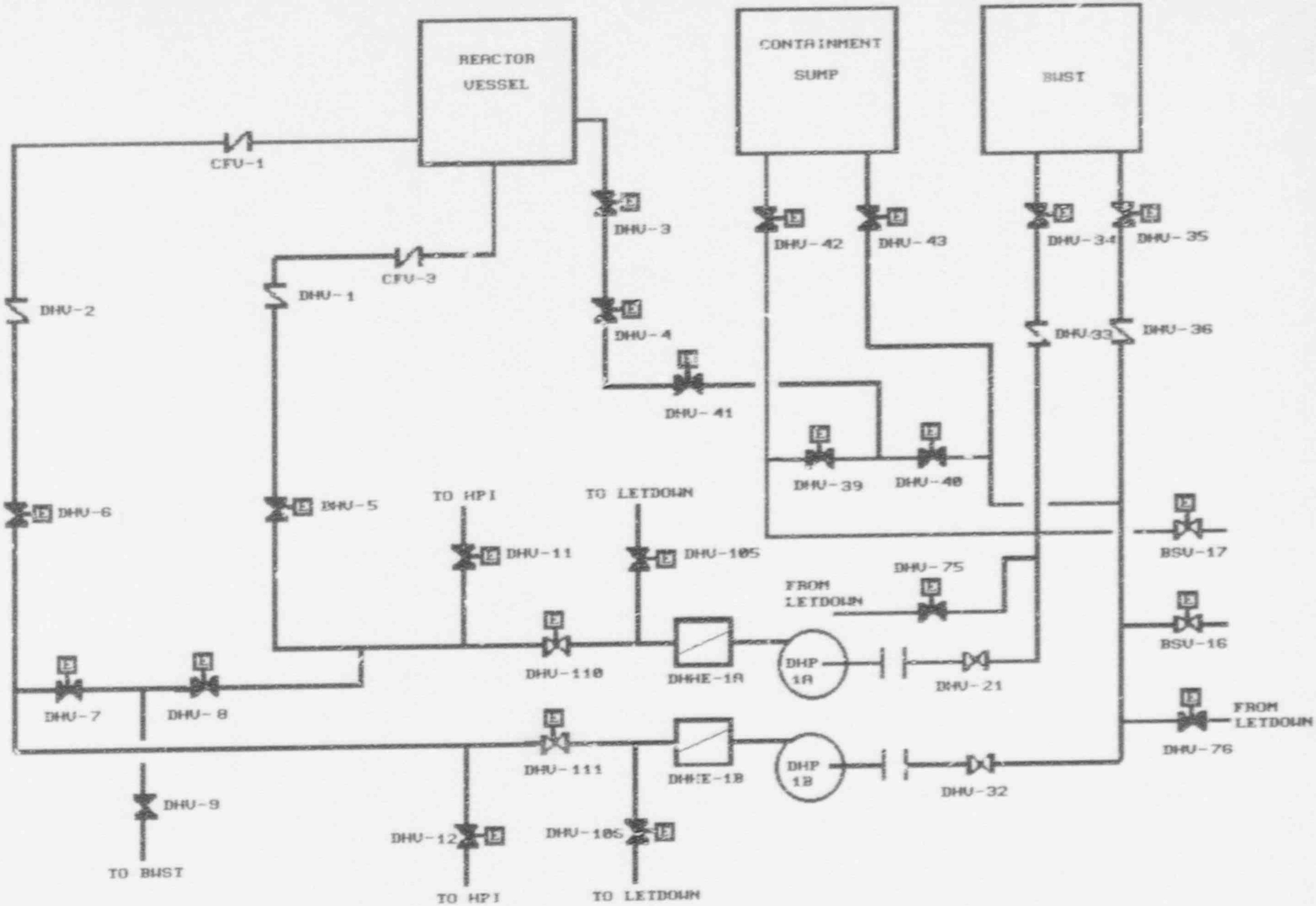

FIGURE 4.1.
SIMPLIFIED SYSTEM DRAWING OF DHR/LPI SYSTEM (REF. 1)

4.2 AC POWER SYSTEM

The AC Power System at Crystal River 3 provides AC power at various voltages to all other plant systems. AC electrical power is the motive force for the majority of the auxiliary and safeguards system pumps, motor-operated valves, and instrumentation.

The AC power system is divided into six separate trains. These are the 6900 V Bus 3A, 6900 V Bus 3B, 4160 V Unit Bus 3A, 4160 V Unit Bus 3B, 4160 V Engineered Safeguards Bus 3A and 4160 V Engineered Safeguards Bus 3B. Power to the 6900 V buses (3A and 3B), which power the reactor coolant pumps, is normally supplied by the Crystal River 3 main generator, when the plant is operating. All other buses are normally supplied by the Unit 3 Startup Transformer, which is connected to the 230 kV switchyard. Since the Unit 3 generator supplies power to the 500 kV switchyard, all other electrically operated equipment in the plant (powered from the 230 kV switchyard) is isolated from the effects of a main generator trip.

The two Engineered Safeguards buses are backed up by diesel generators which automatically start on either low bus voltage or on an Engineered Safeguards Actuation Signal. The Engineered Safeguards buses can also be powered from the Crystal River Units 1 and 2 startup transformers. The two Unit Buses have no backup power supply.

In the event of a unit trip, power to the 6900 V bus is automatically transferred from the main generator to the Unit 3 startup transformer, so that all electrical equipment is powered from the 230 kV switchyard. If the 230 kV switchyard is lost, all normal AC power to all plant equipment is lost, with only the equipment that is powered by the ES buses having an available backup power supply. As soon as low voltage is detected at one of the 4160 V ES buses, its associated diesel generator starts automatically and is automatically connected to the bus when it has reached the proper operating voltage.

When the bus voltage is returned to normal, the equipment that was shed from the bus during the undervoltage transient can be sequenced back onto the bus either automatically or manually.

The AC Power system requires the following support functions provided by other systems:

- DC power to circuit breakers is provided by DC power distribution panels DPDP-3A, DPDP-3B, DPDP-5A and DPDP-5B
- DC power to the diesel generators is provided by DC power distribution panels DPDP-6A and DPDP-6B.

TABLE 4.2A. AC POWER SYSTEM FAILURE MODE IDENTIFICATION

Conditions that Lead to Failure

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1. Emergency Diesel Generators 3A and/or 3B Fail to Operate or are in Maintenance

The emergency diesel generators provide essential backup power to the Emergency Safeguards buses. Failure of these diesel generators (DGs) to operate following a loss of normal offsite power can result in a loss of all AC power. The emergency diesel generators have the highest risk importance of all plant components. The dominant failure mode is failure of a DG to start and to run. A secondary failure mode is unavailability of a DG due to maintenance activities. Periodic maintenance and surveillance in accordance with the Technical Specifications, and proper system lineup checks will enhance availability. In addition, maintenance activities should be reviewed and observed to help ensure that efficient scheduling is being done (including staggered maintenance and surveillance, to minimize the probability of maintenance-related common-cause failure of both EDGs), and that repairs are performed correctly, minimizing DG downtime. Operator training and awareness of Emergency Operating Procedures could also enhance the probability of recovery.

2. Feeder Breaker 3209 or 3210 Fails to Operate

Failure of these feeder breakers can result in loss of electrical power to one or both of the Engineered Safeguards buses. Hardware or electrical component failures are the dominant failure mechanisms. Review and observation of the periodic maintenance and surveillance procedures along with verification of proper breaker position should help ensure breaker reliability.

3. 4.16 kV Engineered Safeguards Bus 3A or 3B in Maintenance or Fails to Operate

Failure of these Engineered Safeguards (ES) buses will lead to a loss of all AC power to one or both trains of Engineered Safeguards equipment. The principal cause of failure is unavailability of the bus due to maintenance. Failure of these ES buses could also be the result of subcomponent failures in the control circuitry, transformers, or improper lineup for automatic operation. The performance of maintenance should be reviewed to help ensure that efficient scheduling is done, and that repairs are done correctly, minimizing ES bus downtime. In addition, procedures for periodic surveillance, maintenance, and proper system lineup should be reviewed or observed to maintain the bus reliability.

4. Failure of Unit 3 Startup Transformer Switchovers

Following a trip of the main generator, power to the 6900 V buses is automatically transferred to the Unit 3 Startup Transformer, so all electrical equipment will be powered from the 230 kV switchyard. If the 230 kV switchyard is lost, normal AC power to plant equipment is lost. Failure of the Unit 3 Startup Transformer in conjunction with failure of the Unit 1 and 2 Startup Transformers, which serve as backup power sources, could also result in a loss of normal AC power to vital systems. The dominant failure mode is that the transformer transfer switch fails to close. Contributing failure modes are operator failure to switchover or failure to lineup to the 230 kV switchyard. Observation and review of the periodic maintenance and testing should maintain availability. Operator training for awareness of potential system malfunctions and selection of appropriate responses will enhance recovery probability.

5. 4.16 kV ES Bus 3A (or 3B) Feeder Breaker 3205 (or 3206) Fails to Operate

Failure of these ES bus feeder breakers can result in loss of electrical power to their respective buses, even though the EDGs are functioning. Hardware or electrical component failures are the dominant failure mechanisms. Review and observation of the periodic maintenance and surveillance procedures, along with verification of proper breaker position should help ensure breaker reliability.

TABLE 4.2B. MODIFIED AC POWER SYSTEM WALKDOWN

Component Number	Component Name	Location	Required Position	Actual Position
3209	ES Bus 3A Feeder Breaker		Open	______
3210	ES Bus 3B Feeder Breaker		Open	______
3103	500 kV to 230 kV Switchover Breaker		Open	______
3104	500 kV to 230 kV Switchover Breaker		Open	______
3205	ES Bus 3A Feeder Breaker		Closed	______
3206	ES Bus 3B Feeder Breaker		Closed	______
3211	ES Bus 3A Switchover to Unit 1 & 2 S.U. Transformer Breaker		Open	______
3212	ES Bus 3B Switchover to Unit 1 & 2 S.U. Transformer Breaker		Open	______
DG-3A	Diesel Generator 3A		(a)	______
DG-3B	Diesel Generator 3B		(a)	______

(a) Due to the integrated nature of the diesel generator failure-to-start or failure-to-run failure modes, the lineup of all automatic diesel generator support functions (service water, fuel oil, starting air, etc.) should be verified.
(b) These are required positions during normal, not emergency, operations.

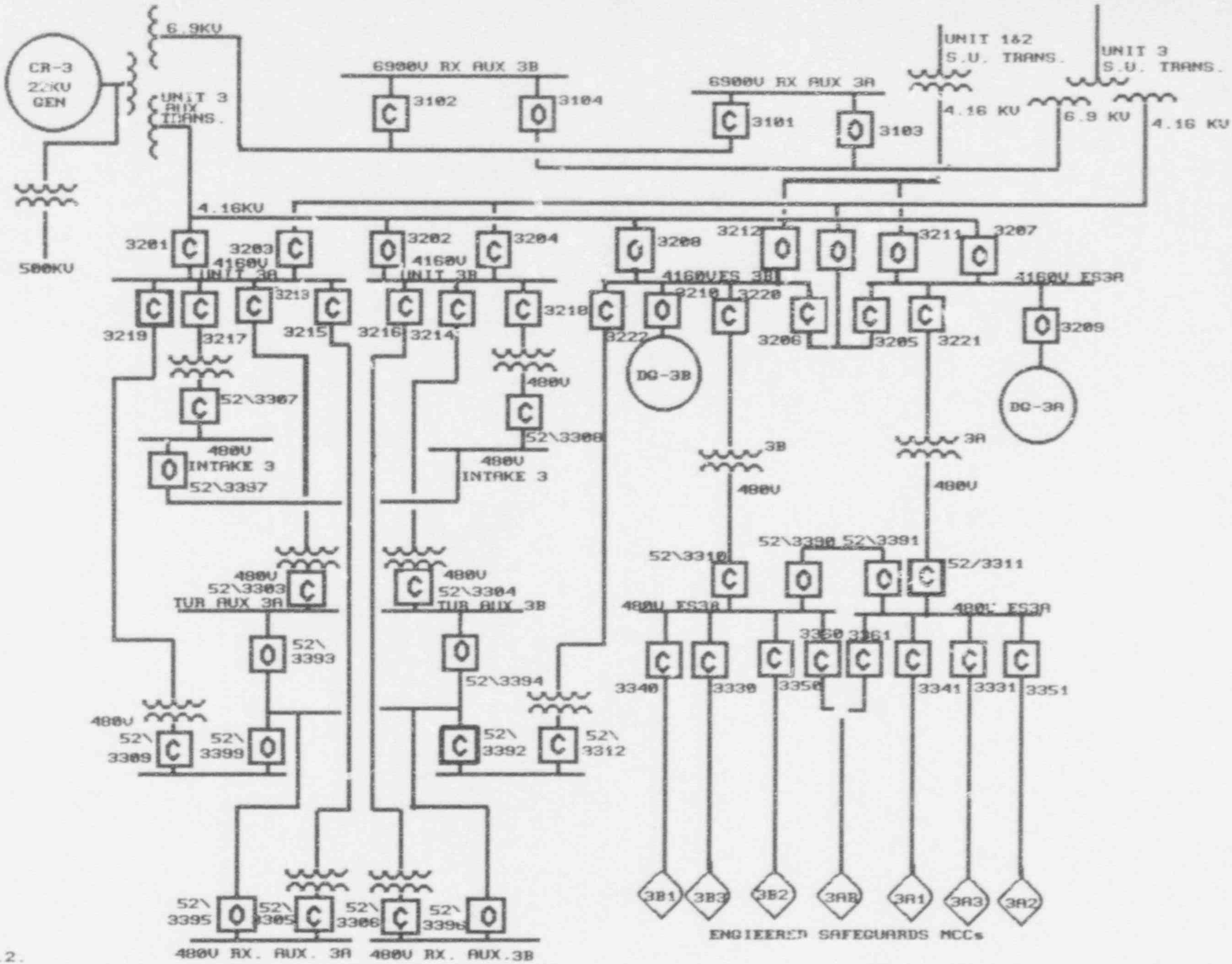


FIGURE 4.2.
SIMPLIFIED SYSTEEM DRAWING OF AC POWER (REF. 1)

4.3 SERVICE WATER SYSTEMS

The Nuclear Services Cooling Water systems at Crystal River 3 include the Nuclear Services Closed Cycle Cooling (Service Water) System, the Decay Heat Closed Cycle Cooling System, and the Nuclear Services and Decay Heat Sea Water (Raw Water) System. The Nuclear Services Closed Cycle Cooling (NSCCC) system provides cooling to the letdown coolers, reactor coolant drain tanks, control rod drives, reactor coolant pumps, spent fuel pool coolers, spent fuel pumps' air handling units, the control complex water chillers, sample coolers, seal return coolers, waste evaporator packages, reactor coolant evaporator packages, waste gas compressors, the motor-driven emergency feedwater pump, the NSCCC pump motors, the raw water pump motors, and Make-up Pumps A and B under normal conditions. The NSCCC system also provides backup cooling to Make-up Pump C.

The NSCCC system consists of three service water pumps in parallel, one normal duty and two emergency pumps. The two emergency pumps are unusual in that each consists of one pump motor and two half-sized pumps connected by a common shaft. Each of these pumps' capacity is roughly 50% greater than that of the normal service water pump to accommodate the extra loading created by the reactor building fan assemblies in an emergency. Each of the emergency pumps is capable of handling 100% of the emergency cooling loads. Any three of the four service water heat exchangers will provide the necessary cooling during any situation. Under normal conditions the NSCCC system has one service water pump, SWP-1C, operating with SWP-1A and SWP-1B in standby. In the event of ESAS signal, both the SWP-1A and SWP-1B pump drivers start automatically, with SWP-1C being shut off 15 seconds later.

The Decay Heat Closed Cycle Cooling (DHCCC) system provides primary cooling to the Decay Heat Removal heat exchangers, the Decay Heat Removal pump motors, Make-up Pump C, the Reactor Building Spray pump motors, the Decay Heat Service Sea Water pump motors, and the DHCCC air handling units. The DHCCC system also provides manual backup cooling to Make-up Pump A. The DHCCC system consists of two completely separate trains. Each train has its own pump, surge tank, and heat exchanger and can handle the emergency heat loads generated in any situation.

The Raw Water system, also called the Nuclear Services and Decay Heat Seawater System (NSSW and DHSW, respectively), provides the ultimate heat sink for the equipment cooled by the NSCCC and DHCCC systems. The NSSW System consists of three pumps in parallel, one normal duty and two emergency pumps. As is the case with the NSCCC system, both of the emergency pumps start on an ESAS signal, tripping the normal duty pump after 15 seconds. Each of the emergency pumps is capable of handling 100% of the emergency cooling loads.

The Decay Heat portion of the Seawater system (DHSW) provides the ultimate heat sink for the decay heat removal and other DHCCC cooling loads. The Decay Heat Seawater system consists of two trains, separate except for a cross-connect just prior to the discharge canal. Each train consists of a pump, heat exchanger, and associated valves. The Nuclear Services and Decay Heat Seawater Systems share a common discharge header.

The NSCCC system requires the following support functions provided by other systems:

- AC power for the pump driver for pumps SWP-1A and 2A is provided by the 4160 V ES Bus 3A. AC power for the pump driver for pumps SWP-1B and 2B is provided by the 4160 V ES Bus 3B. AC power for pump SWP-1C is provided by the 4160 V Unit Bus 3B.

- Emergency actuation of pumps SWP-1A and 1B is provided by the Engineered Safeguards Actuation System.

The DHCCC system requires the following support functions provided by other systems:

- AC power for pump DCP-1A and air handling unit AHHE-30A is provided by the 480 V ES Bus 3A. AC power for pump DCP-1B and air handling unit AHHE-30B is provided by the 480 V ES Bus 3B.

- DC power for AC circuit breaker for the pumps is provided by DPDP-5A and DPDP-5B.

- Automatic actuation is provided by the Engineered Safeguards Actuation System.

The NSSW and DHWS systems require the following support functions provided by other systems.

- AC power for pump RWP-1 is provided by the 4160 V Unit Bus 3A. AC power for pumps RWP-2A and RWP-3A is provided by the 4160 V ES Bus 3A. AC Power for pumps RWP-2B and RWP-3B is provided by the 4160 V ES Bus 3B. AC power for pump DOE-2A is provided by ES MCC 3A1. AC power for pump DOE-2B is provided by ES MCC 3B1.

TABLE 4.3A. SERVICE WATER SYSTEMS FAILURE MODE IDENTIFICATION

Conditions that Lead to Failure

---

1. <u>RWP-3A and RWP-3B Flush Water Valves Fail Closed</u>

Failure of these Raw Water Pump Flush Water valves to remain open may prevent sufficient flow from being provided to the heat exchangers. The dominant failure cause is random hardware failure. Periodic surveillance, testing and maintenance of these valves should help ensure reliable operation.

2. Failure of RWP Pumps 3A, 3B

The dominant failure mode for pumps RWP-3A or RWP-3B is failure to start or run. This may be caused by random hardware failure of the pumps or by latent human error. A secondary contributor to pump failure is physical plugging of the seawater side of the Heat Exchangers, DCHE-1A or DCHE-1B. Review and observation of periodic maintenance and testing procedures, including post-maintenance surveillance of these pumps or heat exchangers should help maintain system reliability. Particular attention should be paid to any maintenance-, vendor-, or environmental-related potential causes of common cause failure of the two pumps.

3. Failure of DHCCC Pumps 1A, 1B

The DHCCC system provides primary cooling to several of the plant's essential heat removal components. Except for normal decay heat removal conditions (during shutdown), the DHCCC system is activated only upon receipt of an ESAS signal, whereupon pumps 1A and 1B start automatically. Failure of the DHCCC pumps could prevent sufficient water flow from being provided to cool these components. The dominant failure mode is failure of the DHCCC pumps to start or run. A contributing failure is misalignment of the pump suction valves DCV-19 or DCV-20. Observation and review of surveillance, maintenance, and lineup of these pumps or valves should maintain system reliability.

TABLE 4.3B. MODIFIED SERVICE WATER SYSTEMS WALKDOWN

Component Number	Component Name	Location	Required Position	Actual Position
	Electrical Components			
	DCP-1A, Pump Breaker		Racked in/ Closed	______
	DCP-1B, Pump Breaker		Racked in/ Closed	______
	RWP-3A, Pump Breaker		Racked in/ Closed	______
	RWP-3B, Pump Breaker		Racked in/ Closed	______
	Valves			
DCV-19	Surge Tank Discharge Valve		Open	______
DCV-20	Surge Tank Discharge Valve		Open	______
	RWP-3A Flush Water Valve		Open	______
	RWP-3B Flush Water Valve		Open	______

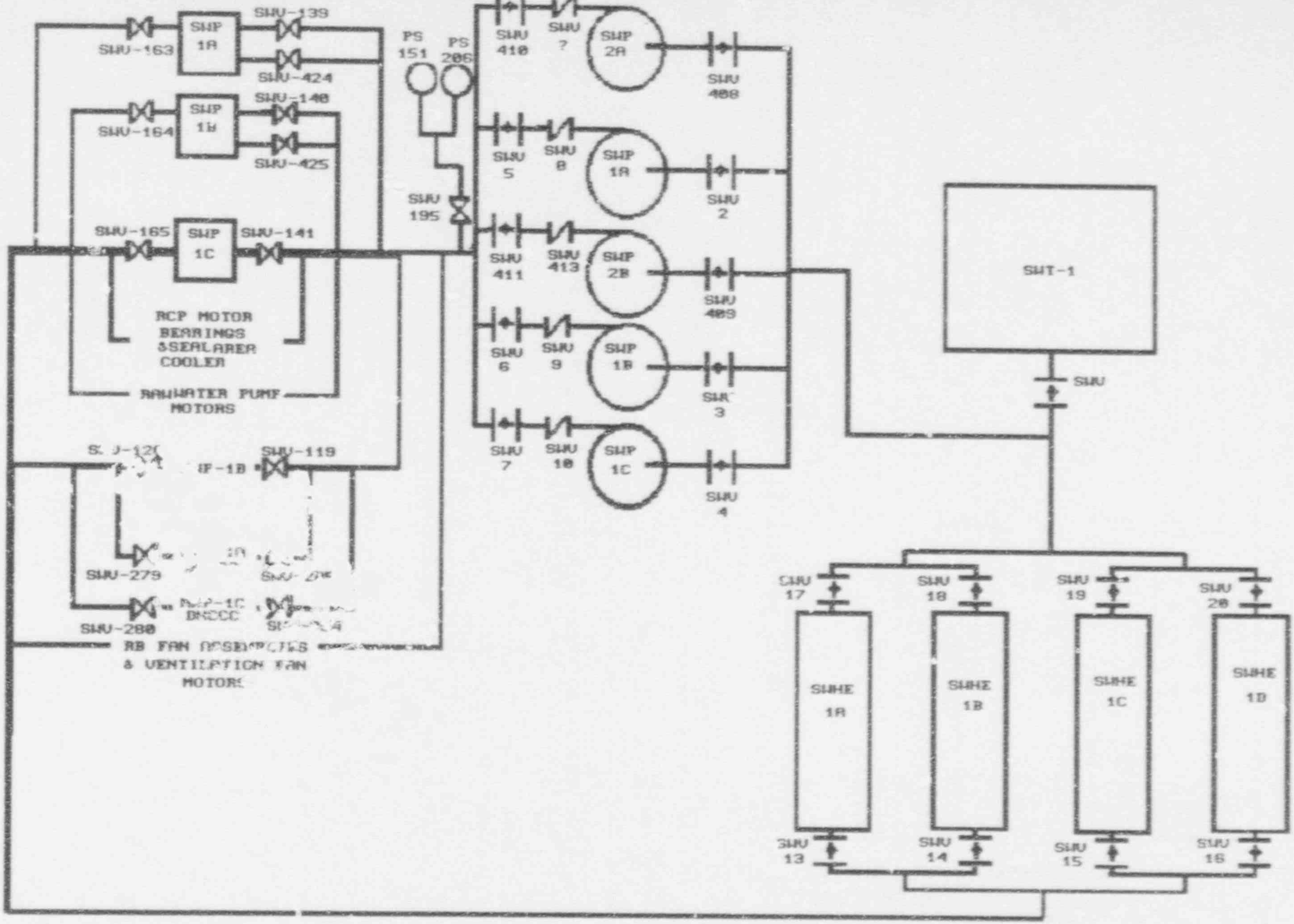
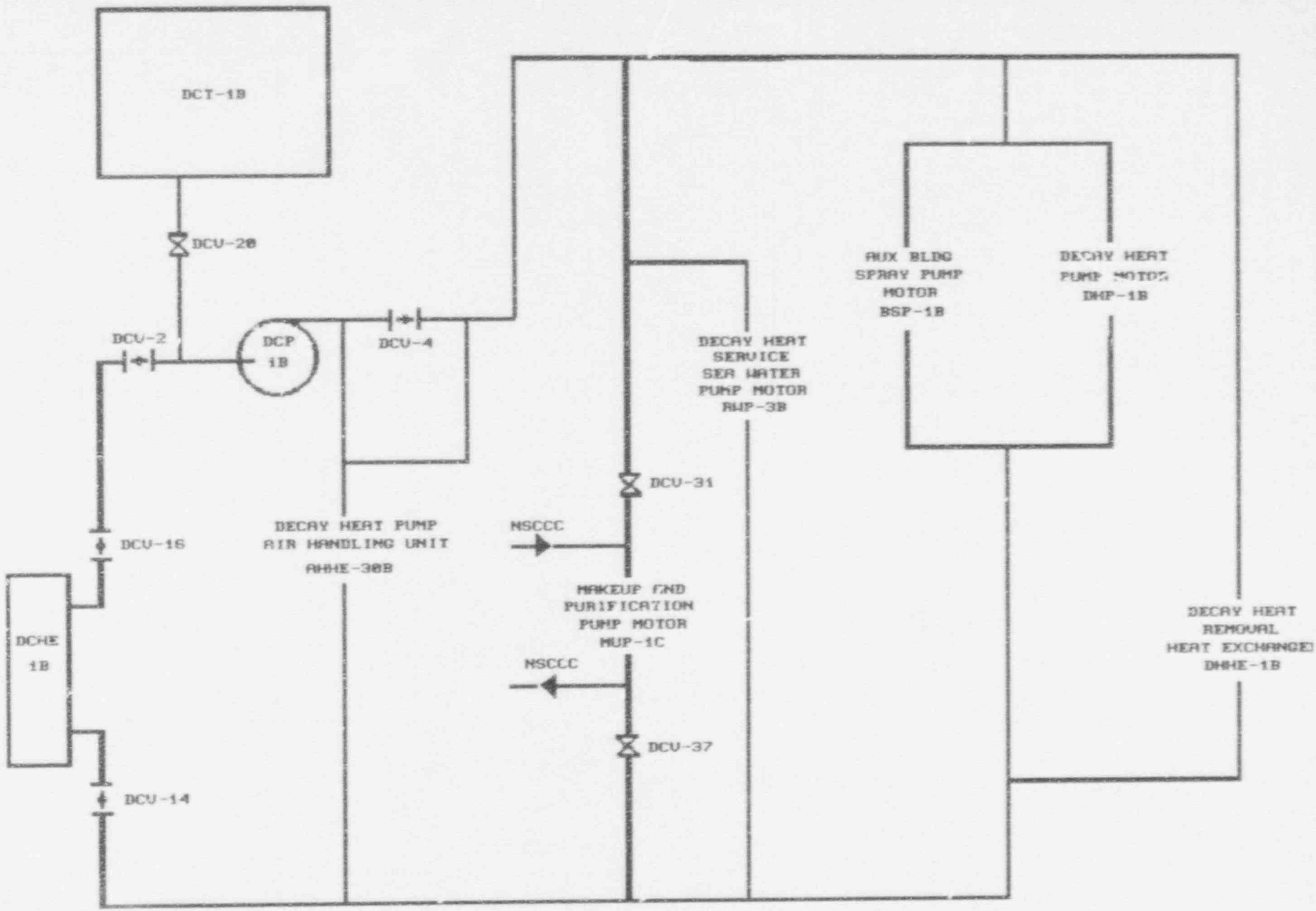


FIGURE 4.3A.
SIMPLIFIED SYSTEM DRAWING OF SWS (NSCCC) (REF. 1)



FIGRUE 4.3B. SIMPLIFIED SYSTEM DRAWING OF SWS (D/CCC) (REF. 1)

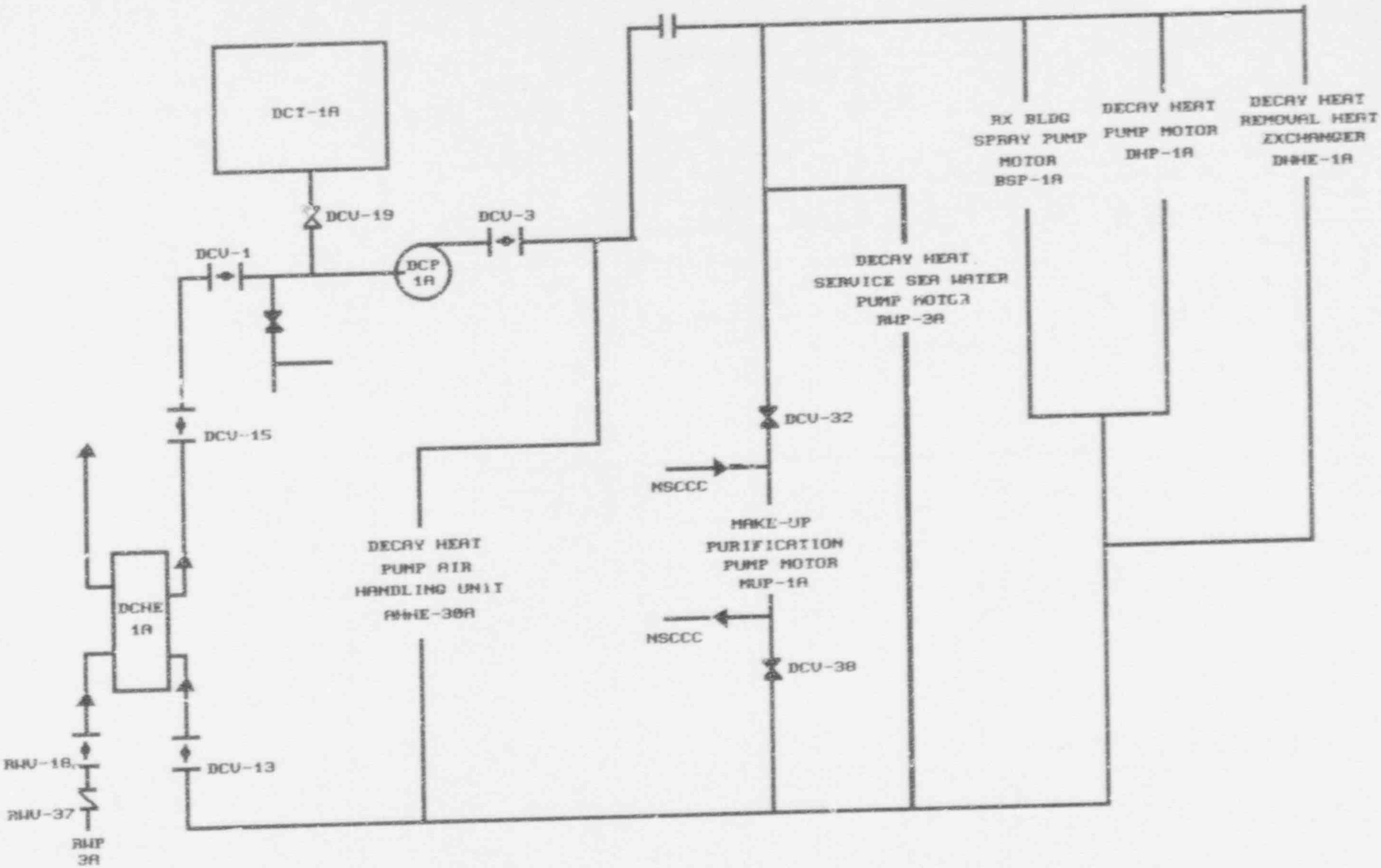
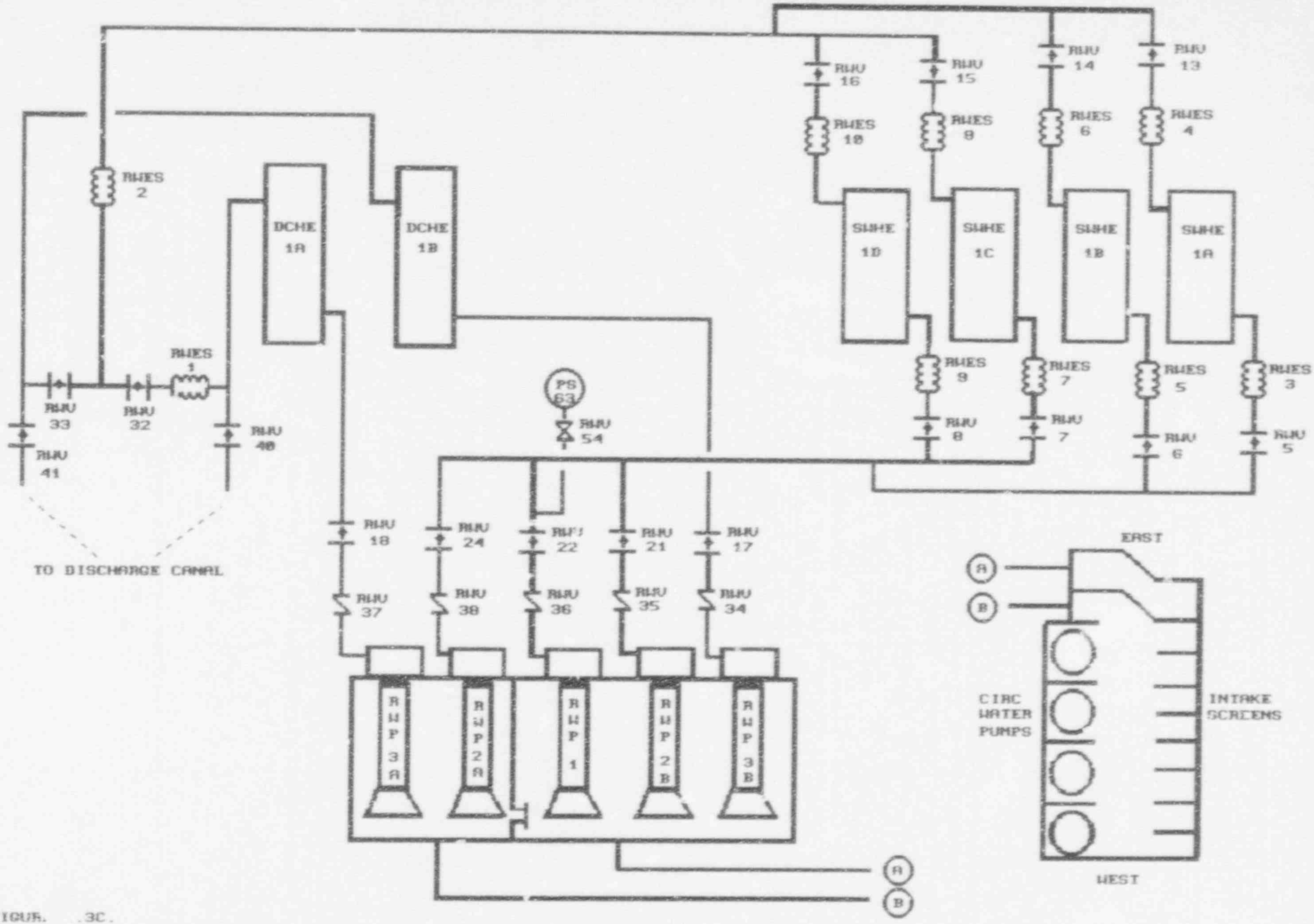


FIGURE 4.38. (CONTD)



FIGUR. .3C.
SIMPLIFIED SYSTEM DRAWING OF SWS (NSSW AND DHSW) (REF. 1)

4.4 DEMINERALIZED WATER SYSTEM

The Demineralized Water system provides demineralized water to other plant systems and for RCS makeup, after boration by the Chemical Addition system. The risk-significant Demineralized Water system failures all occur during steam generator tube rupture (SGTR) sequences. After an SGTR, the operators face a complex task of simultaneously managing the temperature and pressure of the RCS and the secondary system and providing borated makeup water to replace the RCS inventory lost to the secondary through the ruptured steam generator tubes. This management process can include, for some sequences, the need to refill the BWST, since the lost inventory is not being collected in the containment sump and it will not be possible to switchover to recirculation mode core cooling.

It is anticipated that presently contemplated changes in the emergency operating procedures for SGTR accident sequences will reduce or eliminate the risk-importance of Demineralized Water system components. Emphasis will be placed on isolation of the faulted steam generator (SG) combined with an orderly cooldown of the RCS and secondary system using the unfaulted SG. These changes will reduce the need to refill the BWST with borated demineralized water.

4.5 HIGH-PRESSURE INJECTION SYSTEM

The High-Pressure Injection (HPI) system at Crystal River 3 serves as both a support system during normal operation and as an engineered safeguard system during plant transients and accidents. The Crystal River 3 HPI system is also known as the makeup and purification (MUP) system. It consists of three makeup pumps in parallel. Makeup pump 1B is normally running to provide makeup flow and seal injection flow through separate lines to the primary system. Normal suction is obtained from the letdown system via the makeup tank. During an accident, suction is obtained either directly from the BWST or from the containment sump via the LPI system ("piggy-back" mode), and flow is provided to the primary system through four high pressure injection lines which are separate from the normal makeup and seal injection lines.

During normal operation, the HPI system provides normal makeup water to the primary coolant loop and seal injection water to the reactor coolant pumps. During plant transients, the HPI system provides emergency core cooling (ECC) injection from the BWST to the primary system during the early stage of a loss-of-coolant accident (LOCA) and receives suction from the LPI pumps for long-term recirculation of cooling water between the primary system and the reactor building sump. The HPI system can also be used to provide primary system feed for feed-and-bleed cooling following a loss of both main and emergency feedwater system. The "success criteria" for emergency operation of the HPI system are is one operating HPI pump and one intact flow train.

Unless it is down for maintenance, Makeup Pump 1B is normally running and Pumps 1A and 1C are in standby. In the event of a high pressure injection actuation signal from the ESAS, pumps 1A and 1C start automatically. The injection valves open automatically. The normally closed valves MUV-62 and MUV-73 also open to provide two flow paths from the BWST to the makeup pumps. The system then provides ECC water to the primary system until the BWST is drained to its low level alarm point. When the BWST low level alarm sounds, the operator must manually transfer suction to the containment sump for long-term cooling via the DHR pumps in piggy-back mode by 1) opening the valves DHV-11 and DHV-12 between the HPI suction and the LPI pump discharges, 2) opening the valves DHV-42 and DHV-43 between the containment sump and the LPI pump suction, and 3) starting one or both of the DHR (LPI) pumps.

Support functions provided to the HPI system by other systems include provision of AC and DC electric power, and automatic initiation by the Engineered Safeguard Actuation System.

TABLE 4.5A. HPI SYSTEM FAILURE MODE IDENTIFICATION

Conditions that Lead to Failure

---

1. <u>Failure of the "Piggy-back" Valves DHV-11, DHV-12</u>

Fail [illegible] valves DHV-11 and DHV-12 to operate will prevent "piggy back" mode [illegible] where water from the containment sump is provided to the HPI pum[illegible] pumps. The dominant failure mode is that valves DHV-11 and DHV[illegible] open on demand. The important failure causes are random hardware [illegible]al failures. Training and operator awareness of emergency operati[illegible] [illegible]ures will enhance valve availability. Maintenance of these valves should be reviewed or observed.

2. <u>Makeup Pumps Unavailable due to Failure or Maintenance</u>

Under normal operation, makeup pump 1B is normally running and pumps 1A and 1C are in standby. In the event of ESAS signal, pumps 1A and 1C start automatically. Failures of pumps 1A and 1C to start and run or maintenance unavailability of these pumps in combination with failure of pump 1B to continue running can prevent delivery of HPI flow to the RCS. The failure causes are random hardware or electrical failures of the pump or common cause failure. Maintenance or testing activities and training should be reviewed to ensure that efficient scheduling is done, that repairs are performed correctly, and that systems are properly returned to service following maintenance and testing. Because of the number of potential cross-connects operator training may enhance the probability of recovery from these failures.

3. <u>Failure of BWST Suction Valves MUV-73 and MUV-58 to Open on Demand</u>

Following an ESAS actuation, failure of the valves MUV-73 and MUV-58 to open on demand may prevent water from the BWST from being provided by makeup pumps. Hardware or electrical failures are the dominant failure mechanisms. Maintenance and testing of these valves according to Technical Specifications should maintain valve availability. Verification and review of the emergency operating procedures and check-off lists should minimize the probability of failure.

TABLE 4.5B. MODIFIED HPI SYSTEM WALKDOWN

Component Number	Component Name	Location	Required Position	Actual Position
	Electrical			
	Breaker for Makeup Pump MUP-1A		Racked In/ Closed	
	Breaker for Makeup Pump MUP-1B		Racked In/ Closed	
	Breaker for Makeup Pump MUP-1C		Racked In/ Closed	
	Breakers for MOV DHV-11		Closed	
	Breakers for MOV DHV-12		Closed	
	Breakers for MOV MUV-58		Open	
	Breakers for MOV MUV-73		Closed	
	Valves			
DHV-11	Piggy-back MO Valve		Closed	
DHV-12	Piggy-back MO Valve		Closed	
MUV-58	BWST Suction MO Valve		Open	
MUV-73	BWST Suction MO Valve		Closed	

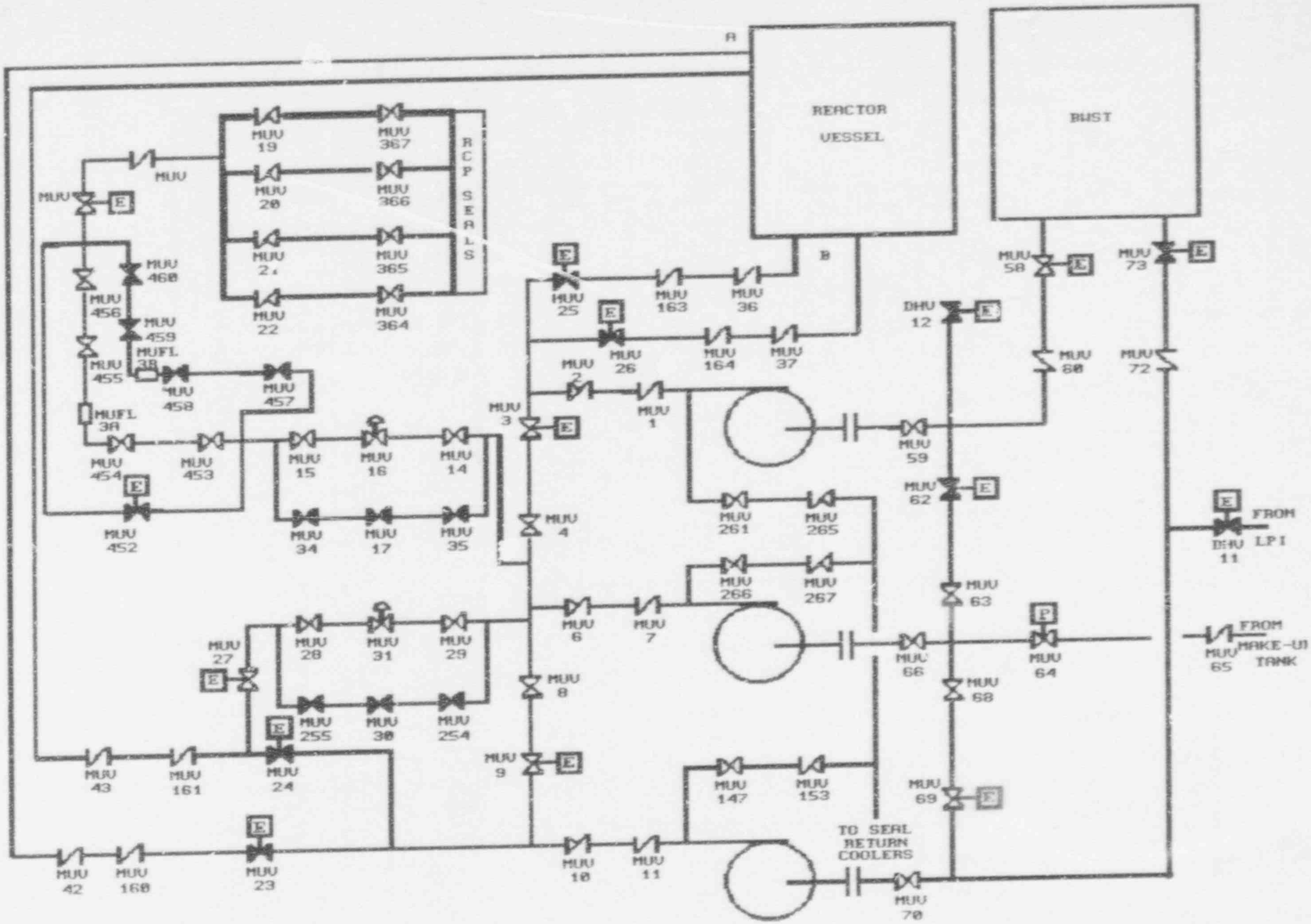


FIGURE 4.5.
SIMPLIFIED SYSTEM DRAWING OF MAKEUP/HPI SYSTEM (REF. 1)

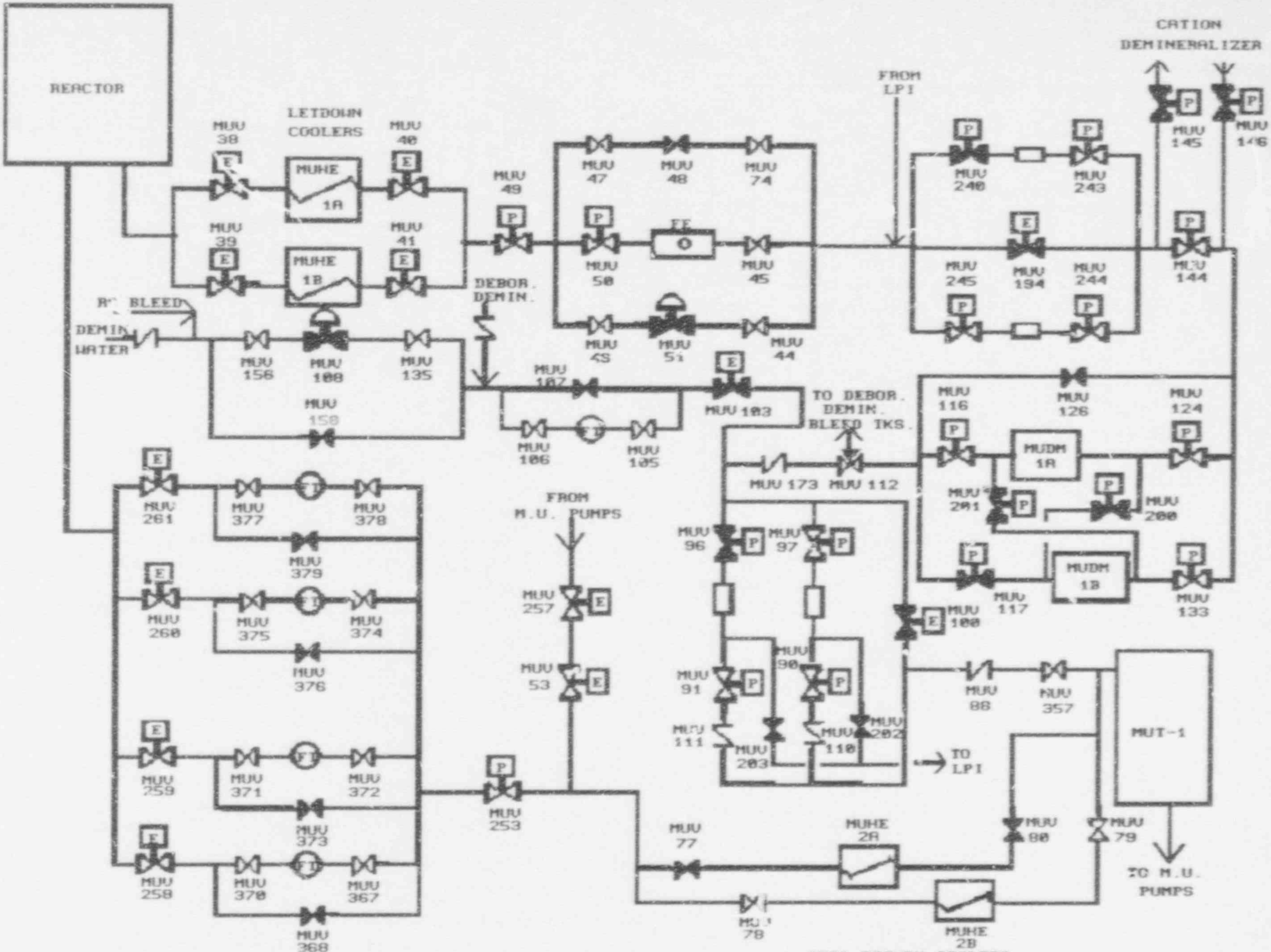


FIGURE 4.5. (CONT.)

4.6 DC POWER SYSTEM

The Crystal River 3 DC Power system provides an uninterruptible source of DC power at 250, 125 and 24 V to plant electrical equipment, such as DC-powered valves, DC-powered AC circuit breakers, and DC control logic. The DC Power system also serves as the primary source of power to the 120 V instrument buses in the AC Power system.

The DC Power system at Crystal River 3 is divided into two separate and redundant trains. Each train has its own battery bank consisting of two 125 V cell groups in series. Charging power to the batteries is supplied by six AC battery chargers (two normally operating AC battery chargers and an installed spare for each train). The capacity of the two normally operating chargers is sufficient to maintain battery charge while supplying all normal DC loads on a given train. The DC Power distribution system consists of a main distribution panel for each train. All other distribution panels receive power from the main distribution panel.

The DC Power system normally operates with the four operating battery chargers providing power for all plant loads. If one of the normally operating chargers is out of service, the installed spare charger can be placed in operation by manual operator action. If an operating charger fails or power is lost to the chargers, the battery bank automatically takes over the load for that train until the charger(s) can be restored.

The DC Power system requires the following support functions provided by other systems:

- AC power for the battery chargers is provided by 480 V ES MCC's 3A3 and 3B2.

TABLE 4.6A. DC POWER SYSTEM FAILURE MODE IDENTIFICATION

Conditions that Lead to Failure

---

1. <u>Failure (including common cause) of Battery 3A and/or 3B</u>

Failure of either of these batteries results in a loss of power from the battery to its respective 125 V DC bus. This failure in combination with other failures can result in a loss of all power at the affected 125 V DC bus. Local faults of the battery itself are the dominant failure mechanisms. The secondary contribution is common cause failure due to latent human error. Periodic testing of the battery voltage and specific gravity, in accordance with the Technical Specifications, as well as proper battery maintenance, should be reviewed and monitored. Operator training and awareness of system malfunction should improve probability of recovery.

2. <u>Failures of Battery Chargers 3A, 3B, 3C, 3D</u>

These battery chargers provide charging power to the DC batteries as well as powering the main distribution panel. Failure of these battery chargers, in combination with failure of the standby chargers and failure of battery sets, can prevent DC power from being supplied to the DC buses. Periodic maintenance, testing and surveillance in accordance with the Technical Specifications requirements will help maintain battery charger reliability. Operator training and awareness of Emergency Operating Procedures will enhance the probability of successful recovery.

3. <u>Failure of DC Distribution Panels DPDP-1A and/or DPDP-1B</u>

These distribution panels supply the DC instrumentation and control systems power to the plant. Failure of these distribution panels can prevent electrical power from being supplied to their respective loads. Local faults of the distribution panels are the dominant failure mechanisms. Periodic testing and maintenance should be observed and reviewed, and appropriate breaker lineups should be verified.

4.6 DC POWER SYSTEM

The Crystal River 3 DC Power system provides an uninterruptible source of DC power at 250, 125 and 24 V to plant electrical equipment, such as DC-powered valves, DC-powered AC circuit breakers, and DC control logic. The DC Power system also serves as the primary source of power to the 120 V instrument buses in the AC Power system.

The DC Power system at Crystal River 3 is divided into two separate and redundant trains. Each train has its own battery bank consisting of two 125 V cell groups in series. Charging power to the batteries is supplied by six AC battery chargers (two normally operating AC battery chargers and an installed spare for each train). The capacity of the two normally operating chargers is sufficient to maintain battery charge while supplying all normal DC loads on a given train. The DC Power distribution system consists of a main distribution panel for each train. All other distribution panels receive power from the main distribution panel.

The DC Power system normally operates with the four operating battery chargers providing power for all plant loads. If one of the normally operating chargers is out of service, the installed spare charger can be placed in operation by manual operator action. If an operating charger fails or power is lost to the chargers, the battery bank automatically takes over the load for that train until the charger(s) can be restored.

The DC Power system requires the following support functions provided by other systems:

- AC power for the battery chargers is provided by 480 V ES MCC's 3A3 and 3B[illegible].

TABLE 4.6A. DC POWER SYSTEM FAILURE MODE IDENTIFICATION

Conditions that Lead to Failure

---

1. Failure (including common cause) of Battery 3A and/or 3B

Failure of either of these batteries results in a loss of power from the battery to its respective 125 V DC bus. This failure in combination with other failures can result in a loss of all power at the affected 125 V DC bus. Local faults of the battery itself are the dominant failure mechanisms. The secondary contribution is common cause failure due to latent human error. Periodic testing of the battery voltage and specific gravity, in accordance with the Technical Specifications, as well as proper battery maintenance, should be reviewed and monitored. Operator training and awareness of system malfunction should improve probability of recovery.

2. Failures of Battery Chargers 3A, 3B, 3C, 3D

These battery chargers provide charging power to the DC batteries as well as powering the main distribution panel. Failure of these battery chargers, in combination with failure of the standby chargers and failure of battery sets, can prevent DC power from being supplied to the DC buses. Periodic maintenance, testing and surveillance in accordance with the Technical Specifications requirements will help maintain battery charger reliability. Operator training and awareness of Emergency Operating Procedures will enhance the probability of successful recovery.

3. Failure of DC Distribution Panels DPDP-1A and/or DPDP-1B

These distribution panels supply the DC instrumentation and control systems power to the plant. Failure of these distribution panels can prevent electrical power from being supplied to their respective loads. Local faults of the distribution panels are the dominant failure mechanisms. Periodic testing and maintenance should be observed and reviewed, and appropriate breaker lineups should be verified.

TABLE 4.6B. MODIFIED DC POWER SYSTEM WALKDOWN

Component Number	Component Name	Location	Required Position	Actual Position
	Battery Charger 3A Supply Breaker		Closed	________
	Battery Charger 3B Supply Breaker		Closed	________
	Battery Charger 3C Supply Breaker		Closed	________
	Battery Charger 3D Supply Breaker		Closed	________
DPDP-1A	DC Panel DPDP-1A Breaker		Closed	________
DPDP-1B	DC Panel DPDP-1B Breaker		Closed	________
3A	Battery 3A		(a)	________
3B	Battery 3B		(a)	________

(a) Surveillance and testing of the battery should be performed in accordance with the plant Technical Specifications and approved procedures.

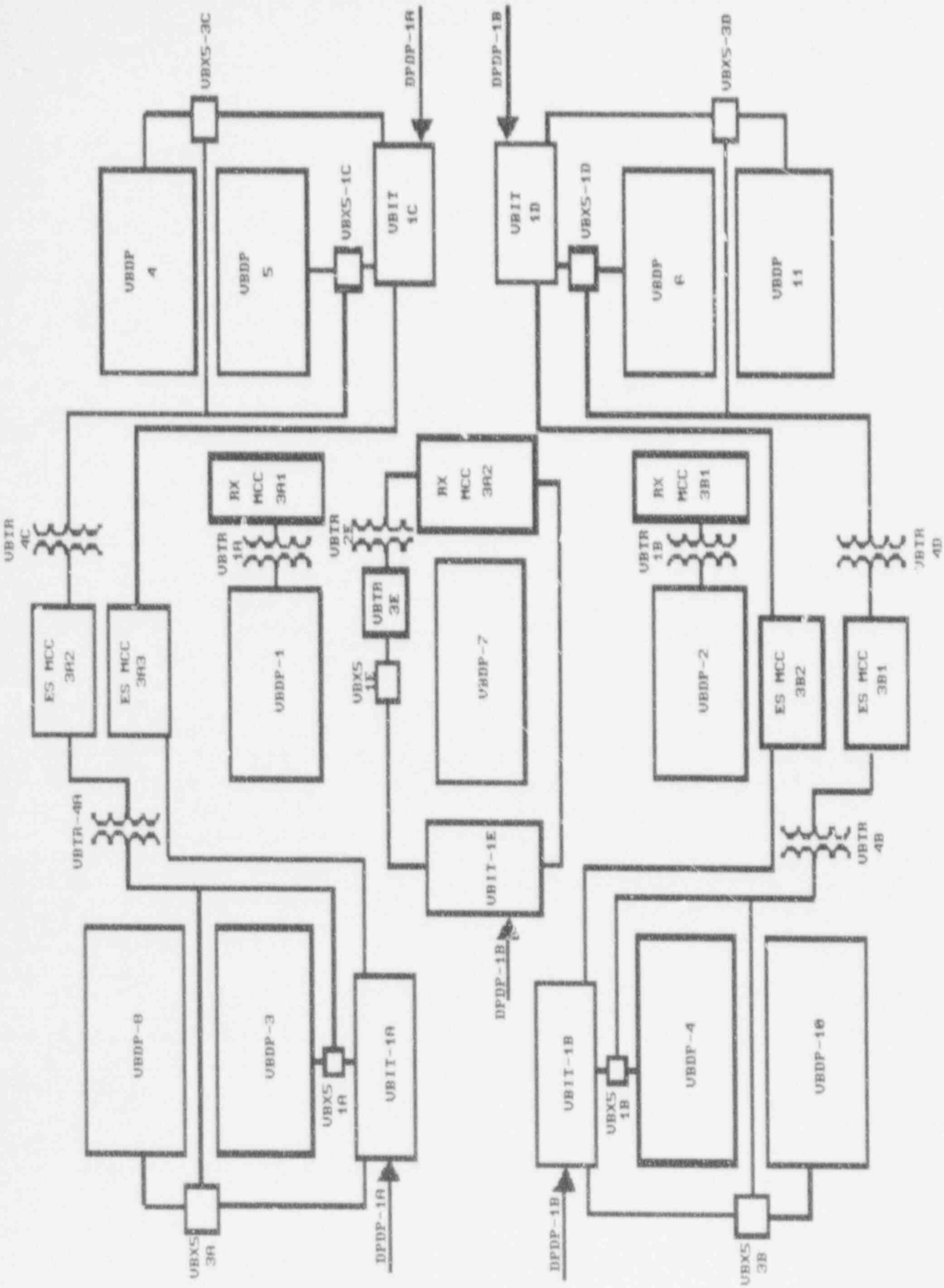


FIGURE 4.6. SIMPLIFIED SYSTEM DRAWING OF DC POWER (REF. 1)

4.7 EMERGENCY FEEDWATER SYSTEM

The emergency feedwater (EFW) system at Crystal River 3 is a standby system and is used to back up the main feedwater (MFW) system in removing post shutdown heat from the reactor coolant system via the steam generators when MFW is lost. During normal shutdowns the main feedwater flow is throttled down to a level capable of removing decay heat and the EFW system is not used. If the plant shutdown is caused by an interruption of the main feedwater flow, the EFW system is automatically put into operation. Also, if main feedwater is lost subsequent to a reactor trip, EFW will be automatically initiated.

The EFW system model also includes the emergency feedwater initiation and control system (EFIC). The EFIC serves several functions that include: automatic initiation of the emergency feedwater system pumps and valves; control of the emergency feedwater flow rate; regulation of secondary side pressure during emergency feedwater system operation; and isolation of the main steam lines on low steam generator secondary side pressure. The EFIC system consists of four redundant instrument and actuation channels.

The EFW system consists of two trains, each capable of supplying emergency feedwater to either or both steam generators. One train contains a motor driven pump, and the second train contains a turbine driven pump. The turbine driven pump is powered by steam from either or both steam generators. The motor driven pump is cooled with water from the nuclear services closed cycle cooling system and the turbine driven pump is self cooled. There are three sources of emergency feedwater: a dedicated EFW storage tank, the condensate storage tank and the condenser hotwell.

There are two EFW injection lines. Each provides emergency feedwater to a spray header in one of the steam generators. Each injection line may receive EFW from either pump train. Flow is controlled through solenoid valves in each of the lines.

The EFW is normally aligned to the dedicated EFW tank, and all block and flow control valves are normally open. The EFW pumps may be started manually or automatically with the EFIC system. The dedicated EFW tank can provide decay heat removal and cooldown for a minimum of twelve hours. Switching to an alternate EFW supply or refilling the tank requires operator action.

The EFW system requires the following support functions provided by other systems:

- AC power for EFW pump EFP-1 is provided from ES Bus 3A.
- cooling water for EFP-1 is provided by the nuclear services closed cycle cooling system.
- electric power for the EFIC channels is provided from 120 V AC Panels VBDP-7, VBDP-8, VBDP-9 and VBDP-10.

- DC power for the solenoid (control) valves is provided from 125 V DPDP-5A and DPDP-5B.

- DC power for the block valves is provided from 125 V DPDP-8C and DPDP-8D.

- DC power for the pump suction valves to the condenser hotwell is provided from 125 V DPDP-3A and DPDP-3B.

- DC power for the steam admission valves to EFP-2 is provided from 125 V DPDP-5B.

Other interfaces, e.g. power supplies to other EFW valves, do not appear in the EFW system fault tree and are therefore not listed.

TABLE 4.7A. EMERGENCY FEEDWATER SYSTEM FAILURE MODE IDENTIFICATION

Conditions that Lead to Failure

---

1. Emergency Feedwater Pumps EFP-1, EFP-2 Fail to Operate

Failure of the motor-driven pump EFP-1 and turbine-driven EFP-2 to operate will prevent water flow from being provided by the EFW system. The important failure causes are the pump hardware, electrical or steam supply failures. Another cause is common cause due to latent human errors. Training, operator awareness, and surveillance of these pumps should be reviewed or observed to maintain reliability. If one pump is unavailable due to maintenance and the other pump fails to start and run, the total loss of the EFW system may result. The performance of maintenance and testing and training for these activities should be reviewed to ensure that scheduling is efficient, and that repairs are performed correctly. Operator understanding of Emergency Operating Procedures involving the EFW System should also be reviewed.

2. Operator Fails to Switch EFW Suction Source

The EFW system has three sources of emergency feedwater with the emergency feedwater tank being used for the first twelve hours. Operator action is required to switch to an alternate EFW supply. The dominant failure causes are that the operator simply fails to switch to an alternate suction source or the two level transmitters fail to respond. Operator awareness of criteria for switchover and adherence to emergency procedures is important. Maintenance and testing activities should be reviewed or observed to determine that the transmitters are operational and properly calibrated.

3. Failure of Control Valves EFV-55, EFV-56, EFV-57, and EFV-58

Failure of these control valves in the closed position will prevent flow of EFW to the steam generators. The dominant failure modes are the loss of signal from EFIC system or valve hardware failures. Testing and maintenance of these valves should be reviewed or observed to maintain reliability. Operator understanding of and training on the Emergency Operating Procedures controlling recovery from these failures should also be reviewed.

TABLE 4.7B. MODIFIED EFW SYSTEM WALKDOWN

Component Number	Component Name	Location	Required Position	Actual Position
	Electrical			
	EFP-1 Pump Breaker		Racked In/ Closed	________
	EFP-2 Pump Breaker		Racked In/ Closed	________
	Valves			
EFV-55	EFP-2 Discharge Valve		Open	________
EFV-56	EFP-2 Discharge Valve		Open	________
EFV-57	EFP-1 Discharge Valve		Open	________
EFV-58	EFP-1 Discharge Valve		Open	________

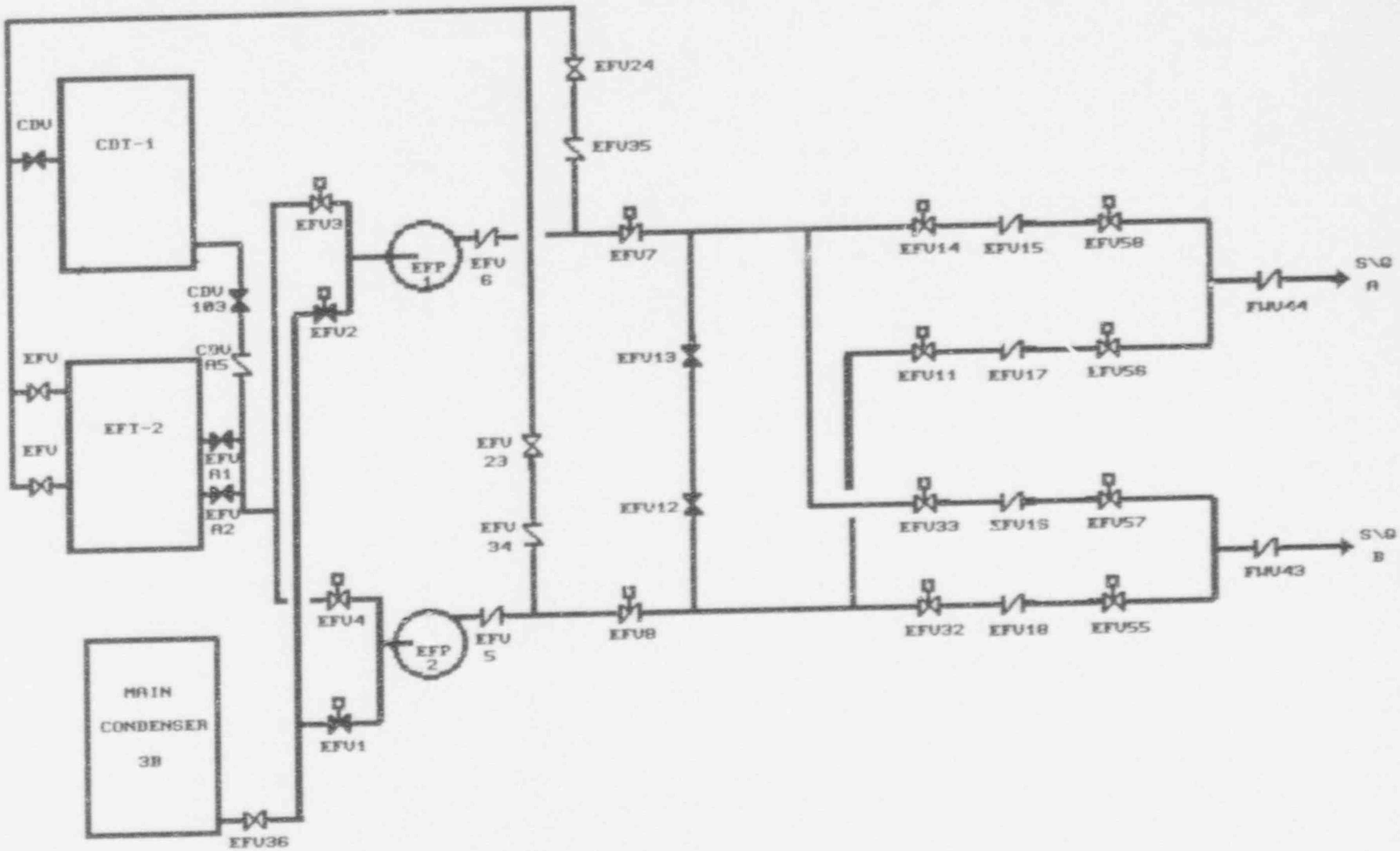


FIGURE 4.7. SIMPLIFIED DRAWING OF EFW SYSTEM (REF. 1)

4.8 REACTOR COOLANT PRESSURE CONTROL SYSTEM

This system maintains control over RCS pressure by manipulating the makeup and letdown flows and by controlling the pressurizer heaters and the pressurizer spray valves. In addition, protection against over-pressurization transients is provided by the pilot-operated relief valve and the passive (i.e., spring loaded) safety relief valves.

TABLE 4.8A. REACTOR COOLANT PRESSURE CONTROL SYSTEM FAILURE MODE IDENTIFICATION

Conditions that Lead to Failure

---

1. Failure of a Pilot-Operated Relief Valve to Reclose

Occurrence of this failure during a plant transient that requires actuation of the pilot-operated relief valve (PORV) initiates a small-break LOCA, which then challenges other plant safety systems. As a standby component, the PORV is difficult to fully test; PORVs have been subject to vendor- and maintenance-related common cause failures. Maintenance procedures, schedules, records, and plant experience with PORVs should be reviewed.

4.9 POWER CONVERSION SYSTEM (PCS)

The power conversion system (PCS) at Crystal River 3 transforms thermal energy from the reactor coolant system through the steam generators into electrical energy. The functions of interest are: providing main feedwater to the steam generators following a reactor trip and relieving steam from the steam generators to the condenser via the turbine bypass valves. The plant systems of interest are therefore: the feedwater and condensate systems, the main steam system, and the integrated control system.

The feedwater and condensate systems consist of two main trains of pumps and heaters, supplying water from the main condenser to each steam generator. The two trains are crosstied at several locations by means of common headers. Two motor-driven condensate pumps provide condensate from the condenser hotwells to the de-aerator tank. Level in the tank is maintained by cycling condensate to the condensate storage tank, as required. The motor-driven feedwater booster pumps take suction from the de-aerator tank and supply flow to the main feedwater pumps. The main feedwater pumps are turbine-driven. During normal operation each pump supplies flow to one steam generator.

Feedwater flow during normal operation and following reactor trip is controlled by the integrated control system. This system processes plant signals and provides control to the feedwater system, the main steam system and the reactor control system. The major components of the integrated control system are the unit load demand subsystem, integrated master control subsystem, feedwater control subsystem and reactor control subsystem.

During normal operation, the main feedwater system provides controlled feedwater flow to each steam generator via the main feedwater valves FWV-29 and FWV-30. The integrated control monitors reactor power, steam generator level and other parameters to match feedwater flow with demand.

Following a reactor trip, the demand for feedwater flow is sharply reduced. The main feedwater system and the integrated control system are designed to run back feedwater flow to meet this reduced demand. The main feedwater inlet valves close and flow is provided via the startup line and valves FWV-39F and FWV-40F. Normally one train of feedwater and condensate flow stop shortly after a reactor trip and the feedwater crosstie valve FWV-28 opens so that one main feedwater pump provides flow to both steam generators. During this time, the integrated control system regulates the steam generator pressure by relieving steam to the condenser through the turbine bypass valves.

The power conversion system includes several plant systems. These systems as a group are self contained except for their requirements for electric power. Interfaces with the electric power system are:

- 480 V ES MCC-3B1 provides power to FWV-28
- 480 V ES MCC-3A1 provides power to FWV-29 and FWV-31

- 480 V ES MCC-3B1 provides power to FWV-32 and FWV-28
- 480 V ES MCC-3A provides power to FWV-14
- 480 V ES MCC-3A provides power to FWV-14
- 480 V ES MCC-3B provides power to FWV-15
- TB MCC-3A provides power to FWV-26, FWV-25, FWV-8, FWV-1, and to the lube
- oil pump and turbine turning gear for main feedwater pump 2A
- 4 kv buses 3A and 3B provide power to the feedwater booster pumps, the condensate pumps the circulating water pumps and the secondary services closed cycle cooling pumps

TABLE 4.9A. POWER CONVERSION SYSTEM FAILURE MODE IDENTIFICATION

Conditions that Lead to Failure

---

1. OTSG A, B Level Control Faults

This is the primary contributor to secondary system failure to provide cooling to the steam generators. The primary failure causes of the OTSG level control are the failures of a controller system and their associated integrated control system. Other causes may include human errors following system maintenance or testing. Training, observation surveillance and maintenance of these control systems should be reviewed or observed to maintain reliability.

2. Startup Valves to Either Steam Generator Fail Closed

Following a reactor trip, the demand for feedwater flow is sharply reduced. The main feedwater inlet valves close and flow is provided via the startup lines (valves FWV-39F and FWV-40F). Failure of these valves in the closed position would prevent cooling water being provided to the steam generators. The dominant failure mode is random valve hardware failures. Testing, surveillance and maintenance of these valves should be reviewed or observed.

3. Failure of Control of De-aerator Level

Failure of flow control elements CDP-1A or CDP-1B to respond or failure of the controllers CDP-1A or CDP-1B may result in the loss of main feedwater. The important failure causes are random hardware or control circuit failures. The contributing failure cause is the human errors following testing or

maintenance. The periodic testing, surveillance and maintenance of these control systems should be reviewed or observed to maintain maximum availability.

4. Failure of the Turbine Bypass or Atmospheric Dump Valves

These are hardware failures. The failure modes for the turbine bypass valves are hardware failures or ICS pressure control faults. The failure modes for the atmospheric dump valves are demand failures, or hardware failures. These failures can be minimized by reviewing or observing valve and control system, surveillance, testing, valve calibration, and checking system lineup for standby operation.

TABLE 4.9B. MODIFIED POWER CONVERSION SYSTEM WALKDOWN

Component Number	Component Name	Location	Required Position	Actual Position
	Air			
FWV-39F	Startup Valve		On	________
FWV-40F	Startup Valve		On	________
	Electrical			
CDP-1A	Control Element CDP-1A Breaker		Closed	________
CDP-1B	Control Element CDP-1B Breaker		Closed	________
	Valves			
	Turbine Bypass Valves		Closed	________
	Atmospheric Dump Valves		Closed	________
FWV-39F	Startup Valve			________
FWV-40F	Startup Valve		Open	________

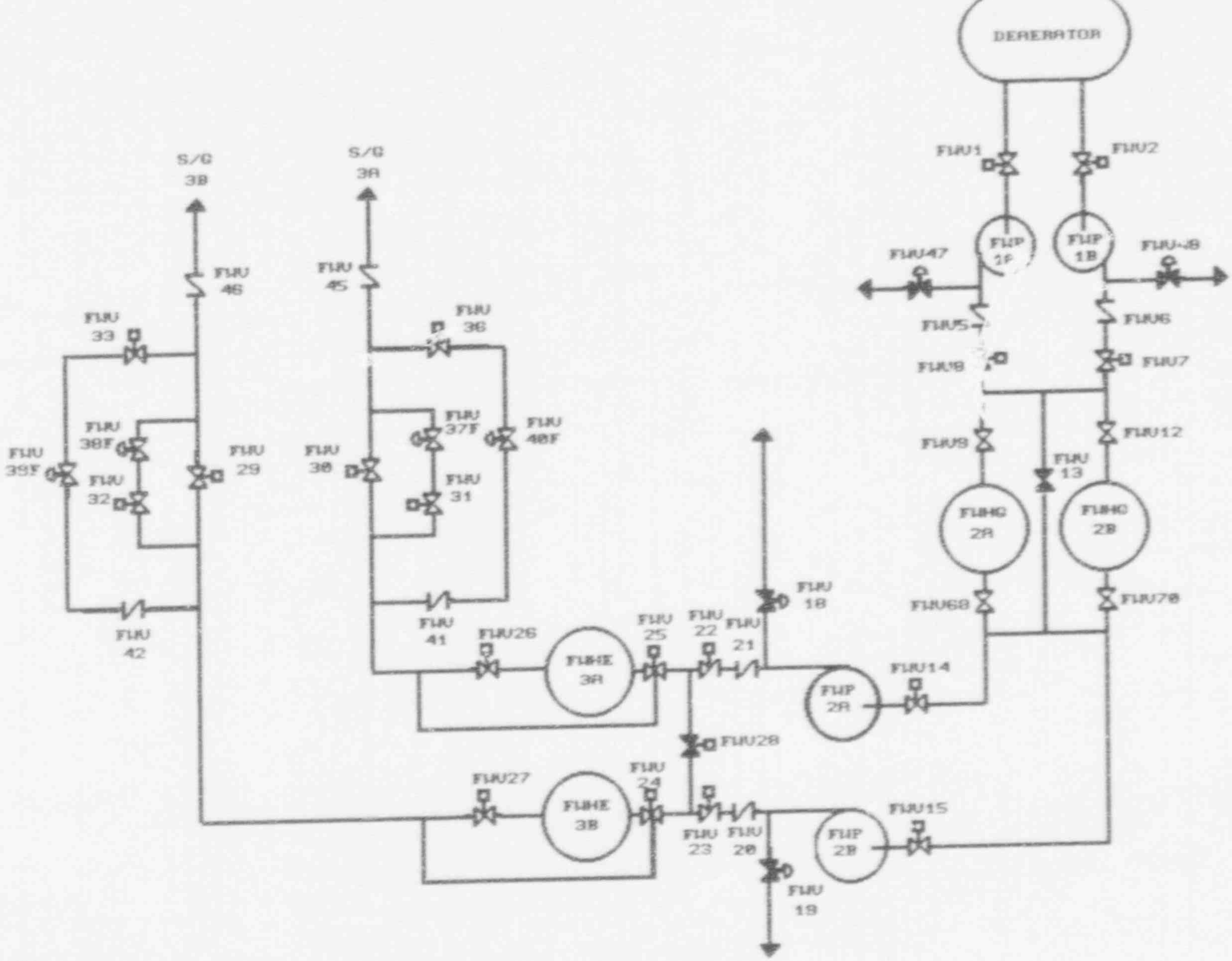


FIGURE 4.9A.

SIMPLIFIED SYSTEM DRAWING OF MAIN FEEDWATER SYSTEM (REF. 1)

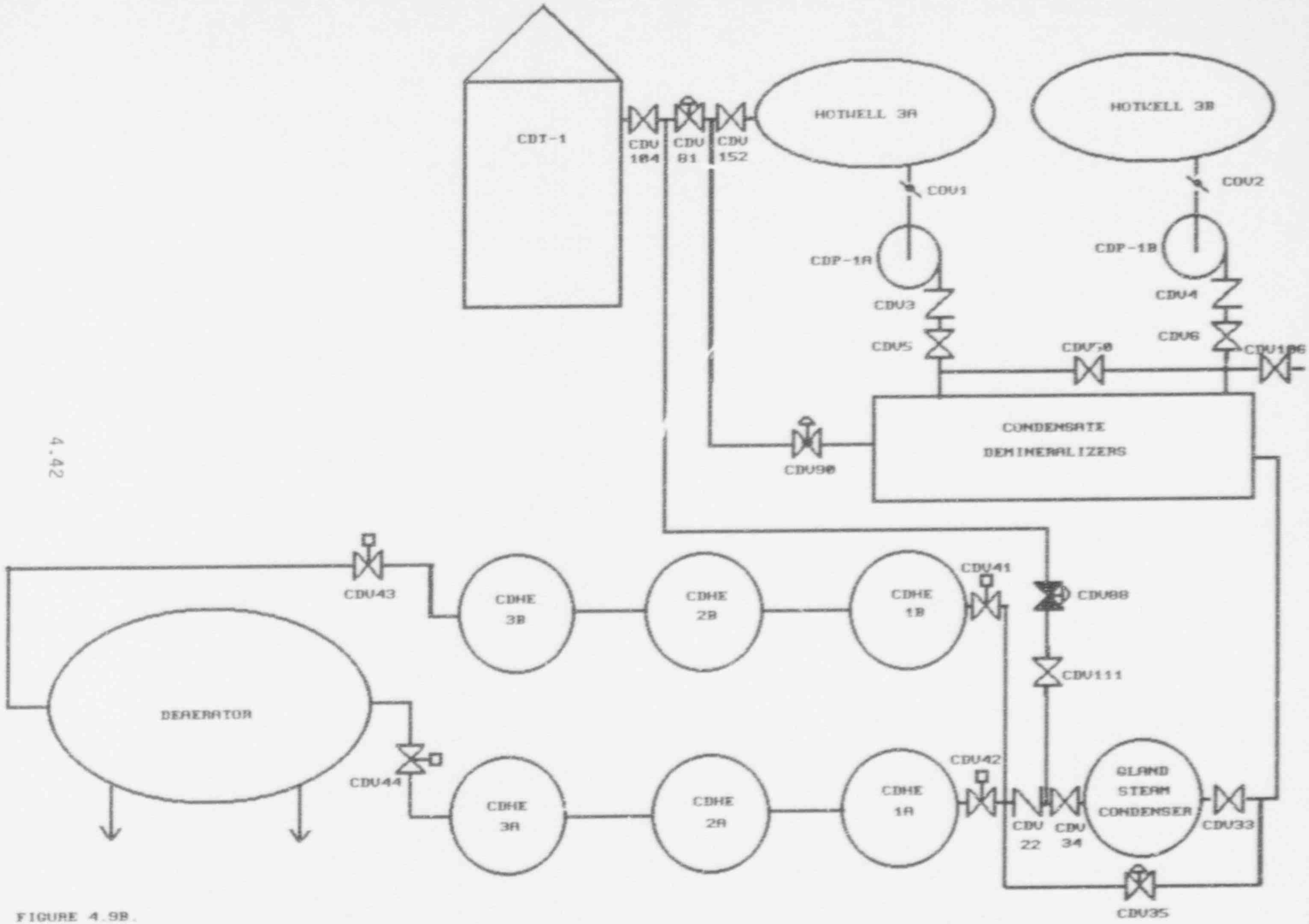


FIGURE 4.9B.
SIMPLIFIED SYSTEM DRAWING OF CONDENSATE SYSTEM (REF. 1)

TABLE 4.10. PLANT OPERATIONS INSPECTION GUIDANCE

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

System	Failure	Discussion
AC Power	Switchover/Recovery Failure	Table 4.2A, Item 5
	Switchover/Recovery Failure	Table 4.2A, Item 1
High-Pressure Injection	Switchover/Recovery Failure	Table 4.5A, Item 3
	Improper Alignment/Recovery Failure	Table 4.5A, Item 2
DC Power	Improper Alignment/Recovery Failure	Table 4.6A, Item 2
	Improper Alignment/Recovery Failure	Table 4.6A, Item 1
Low-Pressure Injection	Improper Alignment/Recovery	Table 4.1A, Item 5
	Switchover/Recovery Failure	Table 4.1A, Item 4
	Improper Alignment/Recovery Failure	Table 4.1A, Item 1
	Improper Alignment/Recovery Failure	Table 4.1A, Item 7
Emergency Feedwater	Improper Alignment/Recovery Failure	Table 4.7A, Item 1
	Switchover/Recovery Failure	Table 4.7A, Item 2
Power Conversion	Improper Alignment/Recovery Failure	Table 4.9A, Item 1

TABLE 4.11. SURVEILLANCE INSPECTION GUIDANCE

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

System	Component	Discussion
AC Power	4.16 kV E.S. Buses 3A, 3B	Table 4.2A, Item 3
	Unit 3 S.U. Transformer Switchover	Table 4.2A, Item 4
	Diesel Generators 3A, 3B	Table 4.2A, Item 1
	4.16 kV E.S. Buses 3A, 3B Feeder Breakers	Table 4.2A, Item 2
Service Water	DHCCC Pumps 1A, 1B	Table 4.3A, Item 3
	RWP-3A, -3B	Table 4.3A, Item 2
	RWP-3A, -3B Flush Water Valves	Table 4.3A, Item 1
High-Pressure Injection	Makeup Pumps	Table 4.5A, Item 2
	BWST Suction Valves	Table 4.5A, Item 3
DC Power	Battery Chargers	Table 4.6A, Item 2
	DC Distribution Panel	Table 4.6A, Item 3
	Batteries	Table 4.6A, Item 1
Low-Pressure Injection	Decay Heat Discharge Valves	Table 4.1A, Item 6
	Containment Sump Valves	Table 4.1A, Item 3
	BWST Valves	Table 4.1A, Item 5
	Decay Heat Pumps	Table 4.1A, Item 4
	DHR Drop Line	Table 4.1A, Item 1
Emergency Feedwater	EFP-1, EFP-2	Table 4.7A, Item 1
	Control Valves	Table 4.7A, Item 3
Power Conversion	OTSG Level Control	Table 4.9A, Item 1
	Startup Valves	Table 4.9A, Item 2
	De-aerator Level Control	Table 4.9A, Item 3
	Turbine Bypass or ADVs	Table 4.9A, Item 4

TABLE 4.12. MAINTENANCE INSPECTION GUIDANCE

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

System	Component	Discussion
AC Power	4.16 kV E.S. Buses 3A, 3B	Table 4.2A, Item 3
	Unit B Startup Transformer Switch	Table 4.2A, Item 4
	Diesel Generators 3A, 3B	Table 4.2A, Item 1
	4.16 kV E.S. Buses 3A, 3B Feeder Breakers	Table 4.2A, Item 2
Service Water	DHCCC Pumps	Table 4.3A, Item 3
	RWP Pumps	Table 4.3A, Item 2
	RWPs Flush Water Valves	Table 4.3A, Item 1
High-Pressure Injection	Makeup Pumps	Table 4.5A, Item 2
	BWST Suction Valves	Table 4.5A, Item 3
DC Power	Battery Chargers	Table 4.6A, Item 2
	DC Distribution Panel	Table 4.6A, Item 3
	Batteries	Table 4.6A, Item 1
Low-Pressure Injection	Decay Heat Discharge Lines	Table 4.1A, Item 6
	Recirculation Valves	Table 4.1A, Item 2
	BWST Valves	Table 4.1A, Item 5
	Decay Heat Pumps	Table 4.1A, Item 4
	DHR Drop Line	Table 4.1A, Item 1
	Containment Sump	Table 4.1A, Item 7
	Piggy-back Line Valves	Table 4.1A, Item 2
	BWST Vacuum Breakers	Table 4.1A, Item 8
Emergency Feedwater	EFPs 1 and 2	Table 4.7A, Item 1
	Control Valves	Table 4.7A, Item 3
Power Conversion	OTSG Level Control	Table 4.9A, Item 1
	Startup Valves	Table 4.9A, Item 2
	De-aerator Level Control	Table 4.9A, Item 3
	Turbine Bypass or ADVs	Table 4.9A, Item 4

TABLE 4.13. QUALITY ASSURANCE/ADMINISTRATIVE CONTROL INSPECTION GUIDANCE

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

System	Component	Discussion
AC Power	4.16 kV E.S. Buses 3A, 3B	Table 4.2A, Item 3
	Unit 3 S.U. Transformer Switchover	Table 4.2A, Item 4
	Diesel Generators 3A, 3B	Table 4.2A, Item 1
	4.16 kV E.S. Buses Feeder Breakers	Table 4.2A, Item 2
Service Water	DHCCC Pumps	Table 4.3A, Item 3
	RWP Pumps	Table 4.3A, Item 2
High-Pressure Injection	Line Suction Valves	Table 4.5A, Item 1
	Makeup Pumps	Table 4.5A, Item 2
	BWST Suction Valves	Table 4.5A, Item 3
DC Power	Battery Chargers	Table 4.6A, Item 2
	DC Distribution Panel	Table 4.6A, Item 3
	Batteries	Table 4.6A, Item 1
Low-Pressure Injection	Decay Heat Discharge Valves	Table 4.1A, Item 6
	Containment Sump Valves	Table 4.1A, Item 3
	BWST Valves	Table 4.1A, Item 5
	Decay Heat Pumps	Table 4.1A, Item 4
	DHR Drop Line	Table 4.1A, Item 1
	Containment Sump	Table 4.1A, Item 7
Emergency Feedwater	EFP-1, -2	Table 4.7A, Item 1
	Control Valves	Table 4.7A, Item 3
Power Conversion	OTSG Level Control	Table 4.9A, Item 1
	Startup Valves	Table 4.9A, Item 2
	De-aerator Control	Table 4.9A, Item 3
	Turbine Bypass or ADVs	Table 4.9A, Item 4

5.0 CONTAINMENT PROTECTION SYSTEMS AT CR-3

In the event of a core melt accident, the public risk due to radiation release is minimized by the containment building. The analysis in this report has not addressed public risk, except through the probability of core melt, because the PRA which was analyzed is a "level 1" analysis and includes only a cursory analysis of release quantities and their effects.

If the containment functions as designed, the public risk resulting from a core melt will be small (e.g., TMI-2 accident), compared to the risk when containment fails with gross releases of radioactivity to the environment. During severe accidents, the containment is protected by two systems--the Reactor Building Spray System (RBS) and the Reactor Building Fan Assemblies (RBFA). They limit the temperature and pressure of steam and air in the containment, and reduce the airborne radioactivity by entraining it in water spray.

In the analysis of the Oconee-3 level 3 PRA (Gore, Vo, and Harris 1987), where systems were prioritized on the basis of public risk, the most risk-important systems were found to be the containment spray and the containment air cooling (e.g., RBS and RBFA) systems. This is because event sequences leading to significant radioactivity releases almost always involved failure of one or both of these systems, which then led to failure of the containment.

In this section, the components of the containment spray and air cooling systems which were found to be important in the Oconee PRA, and their dominant failure modes are identified. It is reasonable to expect that these components and failure modes are important at Crystal River 3 also. In each case, the modes identified contributed to 95% or more of the failure probability of the system. The importance of these systems and components to public risk should be kept in mind during inspection planning at Crystal River 3.

5.1 REACTOR BUILDING SPRAY SYSTEM

Conditions that Lead to Failure

---

1. Human Error - System Operation Inhibited or Failure to Restore Valves or Pump Switchgear after Testing

   Operator failure to restore correct system lineup for automatic pump start and flow to spray nozzles is the most important failure in the Oconee PRA.

2. Spray Pump Failure to Start or Run

Pump hardware or control circuit failures are important at Oconee, as are human errors in the associated procedures for surveillance or maintenance.

3. Failure of Motor-Operated Discharge Valve to Open

(Crystal River 3 Valves BSV-3 and BSV-4). The dominant failure mode at Oconee is hardware failure, with human failure to manually actuate these valves when necessary being a contributing mode.

4. Pump Trains Unavailable Due to Maintenance and Testing

Both scheduled and unscheduled activities are included. Minimization of this time and conformance to Technical Specifications requirements are important at Oconee.

5. Pump Suction Valves Fail to Open or Check Valves Stick Closed

(Crystal River 3 Valves BSV-16 and BSV-17). The dominant Oconee failure modes are human error, electrical failure, or hardware failures. Lineup for standby operation and proper surveillance and maintenance are important.

5.2 REACTOR BUILDING FAN ASSEMBLIES

Conditions that Lead to Failure

1. Operating Fans Fail to Run and Non-Operating Fan Fails to Start and Run

Fan failure due to hardware failure is the dominant system failure mode at Oconee.

2. Operating Fans Fail to Run and Non-Operating Fan in Maintenance

At Oconee, system failure due to fan maintenance unavailability in combination with hardware failures is a significant system failure mode.

3. Motor-Operated Damper to Common Duct Header Fails to Open

Damper misoperation is a significant failure mode at Oconee.

4. Dropout Plates Fail to Drop

Failure of fusible dropout plates to drop and open ductwork bypasses in a post-LOCA environment is a significant Oconee failure mode.

5. Start Switches Improperly Positioned

Human error in positioning control switches, preventing proper automatic system operation, is also an important failure mode at Oconee.

6.0 REFERENCES

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Averett, M. W., and D. N. Miskiewicz. 1990. Crystal River 3 Probabilistic Risk Assessment Summary Document, Florida Power Corporation, Saint Petersburg, Florida.

Gore, B. F., T. V. Vo, and M. S. Harris. 1987. PRA Applications Program for Inspection at Oconee Unit 3, NUREG/CR-5006, prepared by Pacific Northwest Laboratory for the U.S. Nuclear Regulatory Commission, Washington, D.C.

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10. SUPPLEMENTARY NOTES

11. ABSTRACT (200 words or less)

The Level 1 probabilistic risk assessment (PRA) for Crystal River Unit 3 (CR-3) has been analyzed to identify plant systems and components important to minimizing public risk, as measured by system contributions to plant core damage frequency, and to identify the primary failure modes of these components. The report presents a series of tables, organized by system and prioritized by risk importance, which identify components associated with 98% of the inspectable risk due to plant operation. The systems addressed, in descending order of risk importance are: Low Pressure Injection, AC Power, Service Water, Demineralized Water, High Pressure Injection, DC Power, Emergency Feedwater, Reactor Coolant Pressure Control, and Power Conversion. This ranking is based on the Fussell-Vesely measure of risk importance, i.e., the fraction of the total core damage frequency which involves failures of the system of interest.

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