

CHAPTER 7: INSTRUMENTATION AND CONTROLS

7.1 INTRODUCTION7.1.1 Identification and Classification of Safety-Related and Power Generation Systems7.1.1.1 General

Depending on their function, instrumentation and control systems may be classified as either power generation systems or safety systems. In some cases, portions of a system may have a safety function while other portions of the same system may be classified as power generation. A complete description of the reasoning behind this system of classification can be found in Subsection 1.2.1.

The systems presented in this chapter have been classified under safety design-basis systems, power generation design-basis systems, reactor protection systems (RPS), engineered safety feature (ESF) systems (containment isolation, emergency core cooling system, etc.), safe shutdown systems, other safety and power generation systems, and control systems. Figure 7.1-1 lists the Fermi 2 safety-related instrumentation, control, and supporting systems. Instrumentation and control systems identical to those of nuclear power plants of similar design that have recently received construction permits or operating licenses are identified in Table 7.1-1.

7.1.1.2 Identification of Individual Systems

The RPS instrumentation and control initiates an automatic reactor shutdown (scram) if monitored system variables exceed preestablished limits. This action prevents fuel damage, limits system pressure, and thus restricts the release of radioactive material.

The primary containment and reactor vessel isolation control system (CRVICS) initiates closure of various automatic isolation valves in response to a limiting value of a system variable. The closure of isolation valves enables containment of radioactive materials either inside the reactor pressure vessel (RPV) or inside the primary containment. The system responds to various indications of pipe breaks or radioactive material release.

The emergency core cooling system (ECCS) instrumentation and control provides initiation and control of specific core cooling systems such as the high-pressure coolant injection (HPCI) system, the automatic depressurization system (ADS), the core spray system, and the low-pressure coolant injection (LPCI) system.

The neutron monitoring system (NMS) instrumentation and control uses in-core neutron detectors to monitor core neutron flux. The NMS provides signals to the RPS to shut down the reactor when an overpower condition is detected. High average neutron flux is used as the overpower indicator during power operation. Intermediate range detectors are used as overpower indicators during startup and shutdown. The NMS also provides power level indication during planned normal operation.

The refueling interlocks instrumentation and control serves as a backup to procedural core reactivity control during refueling operations.

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The reactor manual control system (RMCS) instrumentation and control allows the operator to manipulate control rods and to determine their positions. Various interlocks are provided in the control circuitry to prevent multiple operator errors or equipment malfunctions from requiring the action of the RPS.

The RPV instrumentation monitors and transmits information concerning key RPV operating variables.

The recirculation flow control system (RFCS) instrumentation and control controls the reactor recirculation pumps and motor-generator sets to vary the coolant flow rate through the core. This system permits either manual or automatic control. The recirculation pump trip (RPT) function of the RFCS is designed to mitigate the effects of an anticipated transient without scram (ATWS) event.

The feedwater system instrumentation and control regulates the feedwater system flow rate so that proper RPV water level is maintained. The feedwater control system uses RPV water level, main steam flow, and feedwater flow signals to regulate feedwater flow. The system is arranged to permit single-element (level only), three-element (level, steam flow, feed flow), or manual operation.

Pressure-regulator and turbine-generator instrumentation and control work together to allow proper generator and reactor response to load-demand changes. The pressure regulator acts to keep nuclear system pressure essentially constant, so that pressure-induced core reactivity changes are controlled. To maintain constant pressure, the pressure regulator adjusts the turbine control valves or turbine bypass valves. The turbine-generator controls regulate turbine speed during startup. If the generator electrical load is lost, the turbine-generator speed-load controls initiate rapid closure of the turbine control valves (coincident with fast opening of the bypass valves) to prevent excessive turbine overspeed.

The process radiation monitor system (PRMS) instrumentation and control for process liquid and gas lines provides control of radioactive material released from the Fermi site. The main steam line radiation monitors detect gross release of fission products from the fuel and provide a trip signal resulting in reactor scram and MSIV isolation.

The area radiation monitor system (ARMS) instrumentation provides gamma-sensitive detectors throughout the plant. Outputs are recorded on multipoint recorders.

Reactor core isolation cooling (RCIC) system instrumentation and control causes the addition of makeup water to the RPV in the event that the reactor feedwater supply system is lost during plant operation.

Standby liquid control system (SLCS) instrumentation and control provides for manual initiation of a reactivity control system redundant to manual control rod movement which can shut the reactor down from rated power to the cold condition if withdrawn control rods cannot be inserted to achieve reactor shutdown. In addition, SLCS instrumentation and control provides for manual initiation of a pH control system following a LOCA in the event of fuel failure.

Reactor water cleanup (RWCU) system instrumentation and control provides for manual initiation of system equipment to maintain high water purity and reduce concentrations of fission products in the reactor water.

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The leak detection system (LDS) instrumentation and control uses various temperature, pressure, and flow sensors to detect, annunciate, and isolate (in certain cases) water and steam leaks in selected reactor systems.

The residual heat removal (RHR) system instrumentation and control provides for manual initiation of cooling to remove the decay and sensible heat from the RPV so that the reactor can be refueled and serviced.

Radwaste system instrumentation and control supports manual processing and disposing of the radioactive process wastes generated during power operation.

The emergency diesel generator (EDG) instrumentation and controls automatically provide ac power to those devices necessary to effect a safe shutdown with subsequent reactor decay heat removal should normal offsite power not be available.

The alternate rod insertion (ARI) function of the control rod drive (CRD) system is designed to mitigate the potential consequences of an ATWS. The ARI equipment is redundant and diverse to the RPS and has its own detection and actuation logic.

The various instrumentation and control system designers and fabricators are identified in Table 7.1-2.

Emergency support facilities, which include an onsite technical support center (TSC), an onsite operational support center (OSC), an onsite emergency operations facility (EOF), an alternate (offsite) EOF, and the Integrated Plant Computer System (IPCS) for data handling and computational capabilities are provided to support operations in the event of an emergency.

7.1.1.3 Classification

7.1.1.3.1 Safety-Related Systems

Safety systems are those systems whose actions are necessary to protect the integrity of radioactive material barriers and/or prevent the release of radioactive material. These systems may be components, groups of components, or groups of systems. A complete list of these systems is shown in Figure 7.1-1.

7.1.1.3.2 Power Generation Systems

Power generation systems are systems whose actions are not required to protect the integrity of radioactive material barriers and/or prevent the release of radioactive material. The instrumentation and control portions of these systems may, by their actions, prevent the plant from exceeding preset limits that would cause action of the safety systems. A complete list of these systems is shown in Figure 7.1-1.

7.1.1.3.3 General Functional Requirements

Power generation systems and safety systems may have both a safety design basis and a power generation design basis, depending on their function. The safety design basis states in functional terms the unique design requirements that establish limits for the operation of the system. The general functional requirements portion of the safety design basis is those

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requirements that have been determined to be sufficient to ensure the adequacy and reliability of the system from a safety viewpoint. Many of these requirements have been introduced into various codes, criteria, and regulatory requirements.

7.1.1.3.4 Specific Regulatory Requirements

All systems have been examined with respect to specific regulatory requirements applicable to instrumentation and control. These regulatory requirements consist of all applicable codes including 10 CFR 50, Appendix A, General Design Criteria; 10 CFR 50, Appendix B, Quality Assurance Criteria; and regulatory guides.

As a result of this examination, it has been determined that two IEEE standards are applicable to the instrumentation and control associated with every safety-related system: IEEE 344-1971 and IEEE 323-1971. Compliance with the requirements of IEEE 323-1971 and IEEE 344-1971 for GE-supplied systems is discussed in NEDO-10698 and NEDO-10678, respectively, and Sections 3.11 and 3.10 of the UFSAR.

Fermi 2 complies with IEEE 336-1971, except as modified by the Edison Quality Assurance (QA) procedures.

The specific regulatory requirements applicable to each system's instrumentation and control are specified in appropriate subsections. The four most important safety systems have been reduced to the subsystem level and the applicable regulatory requirements are specified. This information is contained in Figures 7.1-2 through 7.1-5.

7.1.2 Identification of Safety and Power Generation Criteria

Design bases and criteria for instrumentation and control equipment design are based on the need to have the system perform its intended function while meeting requirements of applicable general design criteria, regulatory guides, and industry standards.

The plant instrumentation and control systems are listed by functional classification and regulatory classification in Figure 7.1-1.

Nominal instrument setpoints and ranges are shown in Chapter 7. Final instrument setpoints are provided in the Technical Specifications.

7.1.2.1 Design Bases

IEEE 279-1971 defines the design requirements with respect to the design bases of safety-related systems. Using the IEEE 279-1971 format, the following fulfills these requirements:

- a. The generating station conditions that require protective action are
 1. Excessive radioactive releases to the atmosphere
 2. Excessive nuclear system stress
 3. Excessive containment stress.
- b. The generating station variables that require monitoring to provide protective actions are listed in Tables 7.2-2, 7.2-3, 7.3-5 through 7.3-8, and 7.3-10

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- c. The minimum number of sensors and locations required to monitor safety-related variables is shown in Tables 7.2-2, 7.2-3, 7.3-5 through 7.3-8, and 7.3-10
- d. Conservative operational limits for each safety-related variable are discussed in the Technical Specifications
- e. The margin between operational limits and the level of determining the onset of unsafe conditions is discussed in the Technical Specifications
- f. Levels requiring protective action are discussed in the Technical Specifications
- g. Range of energy supply and environmental conditions of safety systems is shown in Section 8.3 and Tables 3.11-1 through 3.11-4, respectively
- h. Malfunctions, accidents, and other unusual events that could cause damage to safety systems are discussed in Subsections 7.2.2.2.1 and 7.3.1.3
- i. Minimum performance requirements are shown in Tables 7.2-2, 7.2-3, 7.3-5, 7.3-6, 7.3-8, and 7.3-10.

7.1.2.1.1 Reactor Protection System

7.1.2.1.1.1 Safety Design Bases

General Functional Requirements

The RPS is designed to meet the following functional requirements:

- a. The RPS initiates a reactor scram with precision and reliability to prevent or limit fuel damage following abnormal operational transients
- b. The RPS initiates a scram with precision and reliability to prevent damage to the nuclear system process barrier as a result of excessive internal pressure: that is, to prevent nuclear system pressure from exceeding the limit allowed by applicable industry codes
- c. To limit the uncontrolled release of radioactive materials from the fuel or nuclear system process barrier, the RPS precisely and reliably initiates a reactor scram upon gross failure of either of these barriers
- d. To detect conditions that threaten the fuel or nuclear system process barriers, RPS inputs are derived from variables that are true direct measures of operational conditions
- e. The RPS responds correctly to the sensed variables over the expected range of magnitudes and rates of change
- f. An adequate number of sensors are provided for monitoring essential variables that have spatial dependence
- g. The following bases ensure that the RPS is designed with sufficient reliability
 - 1. If failure of a control or regulating system causes a plant condition that requires a reactor scram but also prevents action by necessary RPS

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channels, the remaining portions of the RPS meet the requirements of Items a., b., and c. above

2. Loss of one power supply neither causes nor prevents a reactor scram
 3. Once initiated, a RPS action goes to completion. Return to normal operation requires deliberate operator action
 4. There is sufficient electrical and physical separation between redundant instrumentation and control equipment monitoring the same variable to prevent environmental factors, electrical transients, or physical events from impairing the ability of the system to respond correctly
 5. Earthquake ground motions, as amplified by building and supporting structures, do not impair the ability of the RPS to initiate a reactor scram. See also Section 3.10
 6. No single failure within the RPS prevents proper RPS action when required to satisfy the safety design bases Items a., b., and c. above
 7. Any one intentional bypass, maintenance operation, calibration operation, or test to verify operational availability does not impair the ability of the RPS to respond correctly
 8. The system is designed for a high probability that when the required number of sensors for any monitored variable exceeds the scram setpoint, the event results in an automatic scram and does not impair the ability of the system to respond correctly as other monitored variables exceed their scram trip points.
 9. The operation of the Hydrogen Water Chemistry System is prevented from affecting RPS operation by the use of contact-to-coil separation.
- h. The following bases reduce the probability that RPS operational reliability and precision will be degraded by operator error:
1. Access to trip settings, component calibration controls, test points, and other terminal points are under the control of plant operations supervisory personnel
 2. Manual bypass of instrumentation and control equipment components is under the control of the main control room operator. If the ability to trip some essential part of the system has been bypassed, this fact is continuously indicated in the main control room.
- i. The RPS and ESF equipment is physically identified as safety equipment in the plant as follows:

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1. Equipment associated with the RPS, primary containment isolation system, and ESF equipment is identified so that two facts are apparent: first, that the equipment is part of the RPS, primary containment isolation system, or an ESF system; and second, that the equipment is associated with a particular grouping (or division) of enforced segregation
2. Panels and racks associated with these systems are labeled with marker plates that are conspicuous by means of color, shape, or color of engraving fill. The information on the marker plate includes both system and division identification
3. Junction and/or pull boxes enclosing wiring for the RPS and an ESF system have identification similar to and compatible with the panels and racks described above
4. Wiring and cables outside cabinets and panels are suitably color-coded to identify the division. Identification tags or markers for wiring conduits are conspicuously different from other similar tags and markers and shall include both system and division identity
5. Those trays or conduits that carry RPS or ESF system wiring are to be identified with conspicuous tags at entrance and exit points of each room through which they pass.

Specific Regulatory Requirements

The RPS is designed to meet the following functional requirements:

- a. Industry Standards - In addition to the previous functional design requirements, the RPS complies with the requirements of IEEE 279-1971. A point-by-point comparison of IEEE 279-1968 is contained in Topical Report NEDO-10139. Section 7.2.2.2.2 of the UFSAR lists those topics where IEEE 279-1971 differs from IEEE-279-1968 and shows conformance to those differences. IEEE 323-1971, IEEE 338-1971, IEEE 379-1972, and IEEE 344-1971 also apply to the RPS
- b. General Design Criteria of (GDC) 10 CFR 50 - GDC 13,20-24, and 29 of 10 CFR 50, Appendix A, have also been implemented in the design of the RPS
- c. Regulatory Guides - Regulatory Guide 1.22, Periodic Testing of Protection System Actuation Function, applies with respect to periodic testing, and Regulatory Guide 1.53, Trial-Use Guide for the Application of the Single-Failure Criterion to Nuclear Power Generating Station Protection Systems, applies with respect to single-failure criteria.

7.1.2.1.1.2 Power Generation Design Basis

The RPS has no power generation objective. The setpoints, power sources, and instrumentation and control are arranged in such a manner as to preclude spurious scrams.

7.1.2.1.2 Containment and Reactor Vessel Isolation Control System

Safety Design Bases - General Functional Requirements

The following functional design bases are implemented in the containment and reactor vessel isolation control system (CRVICS):

- a. The time required to close the main steam isolation valves (MSIVs) is short in order to minimize the loss of coolant from a main steam line break
- b. The time required to close the MSIVs is not so short that inadvertent isolation of steam lines causes a more severe transient than the transient resulting from closure of the turbine stop valves coincident with failure of the turbine bypass system. This ensures that the MSIV closure speed is compatible with the ability of the RPS to protect the fuel and nuclear system process barrier
- c. To ensure the timely isolation of main steam lines, at least one of the isolation valves in each of the main steam lines does not rely on continuity of any variety of electrical power to achieve closure
- d. To provide the operator with means redundant to the automatic isolation functions to take action in the event of a failure of the nuclear system process barrier, it is possible for the main control room operator to manually initiate isolation of the RPV
- e. To limit the release of radioactive materials to the environs, the containment, drywell, and reactor vessel isolation control system, with precision and reliability, initiates timely isolation of penetrations through the containment and drywell structure whenever the values of monitored variables exceed preselected operational limits
- f. To provide assurance that important variables are monitored with precision, an adequate number of sensors are provided (Table 7.3-9)
- g. To provide assurance that conditions indicating a failure of the nuclear system process barrier are detected with sufficient timeliness and precision, primary CRVICS inputs are derived, to the extent feasible and practical, from variables that are direct measures of operational conditions
- h. The steam resulting from a design-basis LOCA flows to the pressure suppression pool to limit pressure in the containment
- i. The power supplies for the containment, drywell, and reactor vessel isolation control system are arranged so that loss of one supply cannot prevent automatic isolation when required
- j. The system is designed so that, once initiated, automatic isolation action goes to completion. Return to normal operation after isolation action requires deliberate operator action
- k. Earthquake ground motions do not impair the ability of the containment, drywell, and reactor vessel isolation control system to initiate automatic isolation

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- l. Any one failure, maintenance operation, calibration operation, or test to verify operational availability does not impair the functional ability of the isolation control system to respond correctly to essential monitored variables, assuming no other active failure occurs
- m. The system is designed for a high probability that, should any essential monitored variable exceed the isolation setpoint, the event results in automatic isolation and does not impair the ability of the system to respond correctly as other monitored variables exceed their trip points
- n. There is sufficient electrical and physical wiring and piping separation between trip channels monitoring the same essential variables to prevent environmental factors, electrical faults, and physical events from impairing the ability of the system to respond correctly, in accordance with Paragraph 4.6 of IEEE 279-1971.

Safety Design Bases - Specific Regulatory Requirements

The requirements of IEEE 279-1971 and IEEE 338-1971 are met by the CRVICS. See Section 3.10 for IEEE 344-1971 and Section 3.11 for IEEE 323-1971 conformance discussions.

7.1.2.1.3 Emergency Core Cooling System

Safety Design Bases - General Functional Requirements

The ECCS instrumentation and control is designed to meet the following functional safety design bases:

- a. They automatically initiate and control the ECCS to prevent fuel cladding temperatures from reaching the NRC interim acceptance criterion
- b. They respond to a need for emergency core cooling, regardless of the physical location of the malfunction or break that causes the need
- c. The following safety design bases are specified to limit dependence on operator judgment in times of stress:
 1. The ECCS responds automatically so that no action is required of plant operators within 10 minutes after a LOCA
 2. The performance of the ECCS is indicated by main control room instrumentation.
- d. Facilities for manual control of the ECCS are provided in the main control room.

Safety Design Bases - Specific Regulatory Requirements

The ECCS instrumentation and control is designed to meet the following specific regulatory requirements:

- a. The instrumentation and control meets the requirements of IEEE 279-1971. The following safety design bases are specified to ensure reliability:

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1. No single malfunction, maintenance, calibration, or test procedure prevents function of the ECCS, assuming no other active or passive failure occurs
 2. No protective device automatically interrupts performance or availability of the ECCS unless continued operation would cause complete failure. Such protective devices indicate abnormal conditions for operator decision and action.
- b. The instrumentation and control meets the requirements of IEEE 338-1971
 - c. The instrumentation and control meets the requirements of IEEE 323-1971 as discussed in Section 3.11
 - d. The instrumentation and control meets the requirements of IEEE 344-1971 as discussed in Section 3.10
 - e. The requirements of GDC 13, 35, 36, and 37 of 10 CFR 50, Appendix A, are met
 - f. The requirements of Regulatory Guide 1.22 are met.

7.1.2.1.4 Neutron Monitoring System

7.1.2.1.4.1 Source Range Monitor System

The source range monitor (SRM) system meets the following power generation design bases:

- a. Neutrons generated by irradiated fuel and neutron detectors together provide a signal-to-noise ratio of at least 2:1 and a count rate of at least 3 counts per second with all control rods fully inserted prior to initial power operation

The minimum count rate may be reduced to ≥ 0.7 CPS provided the signal-to-noise ratio is ≥ 20 and is not applicable during certain refueling operations covered by Technical Specification 3.3.1.2 when the minimum count rate may not be able to be maintained.
- b. The SRM system is able to
 1. Indicate during the worst possible startup rod withdrawal conditions a measurable increase in output signal from at least one detecting channel before the reactor period is less than 20 sec
 2. Indicate substantial increases in output signals with the maximum permitted number of SRM system channels out of service during normal reactor startup operations
 3. Have channels on scale when the intermediate range monitor (IRM) system first indicates neutron flux during a reactor startup

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4. Provide a measure of the time rate of change of the neutron flux (reactor period) for operational convenience
5. Generate interlock signals to block control rod withdrawal if the count rate exceeds a preset value or falls below a preset limit if the IRMs are not above the second range or if certain electronic failures occur.

7.1.2.1.4.2 Intermediate Range Monitor System

Safety Design Basis

The IRM system generates a trip signal that can be used to prevent fuel damage caused by abnormal operational transients that occur while operating in the intermediate power range. The independence and redundancy incorporated in the design of the IRM system are consistent with the safety design bases of the RPS. The IRM system is designed in accordance with the same federal codes, regulatory guides, and IEEE standards applied to the RPS.

Power Generation Design Bases

The IRM system generates a trip signal to block rod withdrawal if the IRM system reading exceeds a preset value or if the IRM system is not operating properly. The IRM system has overlapping neutron flux indications relative to the SRM system and power range monitoring subsystems.

7.1.2.1.4.3 Local Power Range Monitor System

Power Generation Design Bases

The local power range monitor (LPRM) system meets the power generation design bases and supplies the following:

- a. Signals to the average power range monitor (APRM) system proportional to the local neutron flux at various locations within the reactor core
- b. Signals to the rod block monitor (RBM) system to indicate changes in local relative neutron flux during the movement of control rods
- c. Signals to alarm high or low local neutron flux
- d. Signals proportional to the local neutron flux to operator display assemblies to be used for operator evaluation of power distribution, local heat flux, minimum critical power ratio, and fuel burnup rate
- e. A sufficient number of LPRM signals to support the APRM safety design bases.

7.1.2.1.4.4 Average Power Range Monitor System

Safety Design Basis

During the worst permitted input LPRM system bypass conditions, the APRM system generates a trip signal in response to average neutron flux increases resulting from abnormal operational transients in time to prevent fuel damage. Each APRM also includes an OPRM

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Upscale Function that generates a trip signal upon detection of thermal hydraulic induced power oscillations. The APRM system is designed in accordance with the requirements of the safety design bases of the RPS.

Power Generation Design Bases

The APRM system provides

- a. A continuous indication of average reactor power from a few percent to 125 percent rated reactor power to the operator in the main control room
- b. A continuous indication of average reactor power from a few percent to 125 percent rated reactor power to the Integrated Plant Computer System (IPCS)
- c. Interlock signals for blocking further rod withdrawal to avoid an unnecessary scram actuation
- d. A reference power level for the RBM system

7.1.2.1.4.5 Rod Block Monitor System

The power generation design bases for the RBM system meet the following power generation design bases:

- a. Prevent local fuel damage that may result from a single rod withdrawal error
- b. Provide a signal used by the operator to evaluate the change in the local relative power level during control rod movement
- c. Prevent any single short or open of any single input to the RBM system from affecting any other inputs to the RBM system
- d. Meet GDC 24 of 10 CFR 50, Appendix A.

7.1.2.1.4.6 Traversing In-Core Probe System

The traversing in-core probe (TIP) system meets the following power generation design bases:

- a. Provides a signal proportional to the axial neutron flux distribution at selected small axial intervals over the regions of the core where LPRM system detector assemblies are located. This signal is of high precision to allow reliable calibration of LPRM system gains
- b. Provides accurate indication of the position of the flux measurement which allows pointwise or continuous measurement of the axial neutron flux distribution.

7.1.2.1.5 Refueling Interlocks

Refueling interlocks meet the following safety design bases:

- a. During fuel movements in or over the reactor core, all control rods are in their fully inserted positions

- b. No more than one control rod can be withdrawn from its fully inserted position at any time when the reactor is in the refuel mode.

7.1.2.1.6 Reactor Manual Control System

Power Generation Design Basis

The RMCS provides the reactor operator with the means for controlling the power level and power distribution in the core. This is done by control rod positioning capability, which depends on electrical circuitry and switches. Position and power indicators provide surveillance of actions taken and the results of these actions.

Classification

This system is a power generation system, not essential for safety, and is classified in Chapter 3.

7.1.2.1.7 Reactor Vessel Power Generation Instrumentation

The power generation design bases for the RPV instrumentation consist of maintaining proper operating conditions. To maintain proper operating conditions, the RPV instrumentation is designed to provide the operator with sufficient indication of RPV temperature, reactor core flow rate, RPV water level, RPV pressure, and nuclear system leakage. These instruments augment existing information such that the operator can start up, operate, shut down, and service the reactor efficiently. Because the RPV instrumentation used for RPS, ESF, safe shutdown systems, and certain control systems is described and evaluated in other portions of this document, only those instruments not required for safety systems are described (Subsection 7.6.1.2).

7.1.2.1.8 Recirculation Flow Control System Safety Design Bases

The RFCS functions so that no abnormal operational transient resulting from a malfunction in the RFCS can result in damaging the fuel or exceeding nuclear system pressure limits.

Power Generation Design Bases

The RFCS is designed to allow manual recirculation flow adjustment, thereby enabling manual control of reactor power level.

7.1.2.1.9 Feedwater Control System

The feedwater control system meets the power generation design bases by regulating the feedwater flow to maintain adequate water level in the RPV according to the requirements of the steam separators, and to prevent uncovering of the reactor core over the entire power range of the reactor.

7.1.2.1.10 Pressure Regulator and Turbine-Generator Control

One of the main features of direct cycle BWRs is the direct passage of the nuclear steam supply system (NSSS) generated steam through the turbine. In this system the turbine is slaved to the reactor, in that all the steam generated by the reactor is normally accepted by

the turbine. The operation of the reactor demands that the pressure regulator concept be applied to maintain a constant turbine inlet pressure with load-following ability handled by variation of the reactor recirculation flow or control rod position.

The turbine pressure regulator, in maintaining constant stop valve pressure, operates the steam bypass system such that up to 23.5 percent of nuclear boiler rated flow can be bypassed when operating below the maximum steam flow limit as well as during the startup and shutdown phases.

The pressure regulator and turbine-generator control system accomplishes the following control functions:

- a. Controls turbine speed and turbine acceleration
- b. Operates the steam bypass system to keep reactor pressure within limits, and avoids large power transients
- c. Adjusts (manually) 52-in. manifold pressure to nullify a 30 psi drop over a reactor flow of 0 to 100 percent.

7.1.2.1.11 Process Radiation Monitor System

The process radiation monitor system is discussed in Section 11.4.

7.1.2.1.12 Area Radiation Monitor System

The area radiation monitor system is discussed in Section 12.1.

7.1.2.1.13 Offsite Environs Radiological Monitoring Programs

This material is discussed in the Offsite Dose Calculation Manual (ODCM).

7.1.2.1.14 Rad-Chem Radiation Monitoring Instruments

This material is discussed in Section 12.3.

7.1.2.1.15 Plant Computer Systems

7.1.2.1.15.1 Integrated Plant Computer System (IPCS)

The IPCS is a non-safety related computer system that combines various functions of legacy computer systems that it replaced. The IPCS provides the capability of monitoring, recording and displaying plant parameters via strategically located display devices.

The IPCS meets the following power generation design bases:

- a. The IPCS is designed for use with, and has capacity for, the Fermi 2 plant alone.
- b. The Scan, Log and Alarm (SLA) function provides continuous monitoring of plant parameters through on-line data acquisition equipment. Plant parameters are alarmed and logged based on pre-determined setpoints.

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- c. The IPCS supplies information to the operator via a man-machine interface (MMI) consisting of video displays and printers mounted within the operating panels.
- d. Data archival of plant parameters is provided on both a short term (at process scan rates) and on a long term basis (at a reduced scan rate).
- e. The Nuclear Steam Supply System (NSSS) function processes the heat balance data related to core operation into a condensed and usable form that assists in operating the core within prescribed limits.

Reactor heat balance analysis is accomplished with both periodic and on-demand programs.

The results from these calculations are displayed through alarms and on-demand and periodic computer printouts.

- f. The Balance of Plant (BOP) function provides extended features beyond the NSSS function to other plant systems. The on-line data values required for monitoring BOP systems are obtained from BOP system sensors shared with other systems and from sensors installed specifically to provide input data for the computer. The IPCS is able to perform certain BOP calculations to aid with equipment operation and equipment operation documentation.

The on-line data-gathering and computation ability of the IPCS allows the display of on-line equipment performance indicators. These indicators provide a condensed summary of BOP equipment operational status.

- g. The Emergency Response function is designed to gather data from selected plant parameters and data systems for use in the Safety Parameter Display System (SPDS) function and Emergency Response Data System (ERDS) function.

The SPDS function calculates and displays the value and status of the primary variables of the following systems:

1. Core Cooling
2. Fuel integrity
3. Reactivity
4. Reactor coolant system integrity
5. Containment integrity
6. Radioactivity effluent to the environment

The design basis of the SPDS function is to display to operating personnel a minimum set of parameters that define the status of the plant as necessary to assess plant safety status.

The ERDS function provides the NRC with SPDS data through a dedicated datalink.

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- h. The Meteorological (MET) function is designed to provide calculations using various meteorological parameters obtained from the Meteorological Data Acquisition System (MDAS). These calculations are used to support the requirements of Regulatory Guide 1.23 "Onsite Meteorological Programs".
- i. The Transient Recording and Analysis (TRA) function is designed to provide high-speed recording of select plant parameters that are of significant importance during plant transients. The analysis portion of the function provides statistical data reduction capabilities to aid operating personnel in understanding the event.

The IPCS interfaces with a wide variety of external systems through specialized data links for providing or obtaining process parameters. These systems include:

- a. 3D-Monicores Computer System (3DM)
- b. Power Range Neutron Monitor System (PRNM)
- c. Rod Worth Minimizer System (RWM)
- d. Traversing Incore Probe (TIP) System
- e. Radiological Dose Assessment Application (Raddose V)
- f. Meteorological Data Acquisition System (MDAS)

7.1.2.1.15.2 3D-Monicores Computer System (3DM)

The 3DM computer is designed to determine periodically the three-dimensional power density distribution for the reactor core, and to provide printed logs that permit accurate assessment of core thermal performance.

The 3DM computer provides nearly continuous monitoring of the core margins to operating limits and appropriate alarms based on established core operating limits. This aids the operator in ensuring that the core is operating within acceptable limits at all times, especially during periods of power level changes.

7.1.2.1.16 Standby Gas Treatment System

Safety Design Basis - General Functional Requirements

The standby gas treatment system (SGTS) instrumentation and control meets the following safety design bases:

- a. The instrumentation and control initiates the SGTS to provide filtration of air released from the reactor building following a fuel-handling accident or LOCA
- b. The instrumentation and control limits the possibility of exfiltration from the reactor building to outdoors by maintaining negative pressure in the reactor building area
- c. The SGTS responds automatically so that no initiating action is required of plant operators following a LOCA or fuel-handling accident
- d. The responses of the SGTS are indicated on the main control panel

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- e. Facilities for the manual control of the SGTS are provided in the main control room
- f. No single failure, maintenance, calibration, or test prevents operation of the SGTS
- g. Any installed means of manually interrupting the availability of the SGTS is under the control of the operator or other supervisory personnel
- h. Loss of interruptible instrument air and/or offsite electric power does not affect the normal function of the SGTS
- i. The physical events accompanying a LOCA or fuel-handling accident could not prevent correct functioning of the instrumentation and controls
- j. Seismic motions resulting from earthquake ground motion of the design-basis earthquake, missile, wind, and flood do not impair the operation of the instrumentation and control.

Safety Design Basis - Specific Regulatory Requirements

The requirements of IEEE 279-1971 and IEEE 344-1971 are met by the SGTS instrumentation and control. Additionally, GDC 13, 20-24, and 29 of 10 CFR 50, Appendix A, and Regulatory Guide 1.22 have been implemented in the design of this control system.

7.1.2.1.17 Control Center Atmospheric Control System

The control center atmospheric control is provided by the control center heating, ventilation, and air conditioning (HVAC) system, described in Subsection 9.4.1. The instrumentation and control for this system meets the following design bases.

Safety design bases include all the bases described under power generation design bases and the following:

- a. The system controls are interlocked with the RMS to isolate the main control room and automatically route the outside makeup air for the control center HVAC system through the emergency and recirculation filter trains so that main control room habitability is maintained
- b. The system operates in conjunction with ionization detection systems to annunciate in the main control room on detection of combustion products in the main control room ceiling space
- c. The system has the capability to purge rooms manually with 100 percent outside air, or to route the outside air and recirculation air mixture of the control center HVAC system manually through odor, smoke, and halogen-removing filters that are normally bypassed
- d. No single failure, maintenance, calibration, or test operation prevents the functioning of the control center HVAC instrumentation and control. A single active failure in the Halon fire protection system will cause closure of smoke/Halon dampers to the relay room, cable spreading room or computer room. Manual actions are required to reopen these dampers to reestablish airflow.

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- e. Any installed means of manually interrupting the availability of the control center HVAC system is under the control of the operator or other supervisory personnel
- f. Loss of offsite electric power does not affect the normal functioning of instrumentation and controls
- g. The physical events accompanying a LOCA or fuel-handling accident do not prevent correct functioning of the instrumentation and controls
- h. Seismic motions resulting from earthquake ground motion, missile, wind, and flood do not impair the operation of the instrumentation and controls
- i. The requirements of IEEE 279-1971, IEEE 323-1971, IEEE 338-1971, and IEEE 344-1971 are met by the control center HVAC system instrumentation and control. Additionally, GDC 13, 19, 20-24, and 29 of 10 CFR 50, Appendix A, and Regulatory Guide 1.22 have been implemented in the design of this control system.
- j. The system has the following controls, interlocks, and overrides:
 - 1. The system can be manually selected to any of the four modes (i.e., normal, purge, chlorine or recirculation).
 - 2. The system will transfer to the purge or recirculation mode automatically upon receipt of the appropriate signals.
 - 3. The automatic purge mode will override the normal mode.
 - 4. The manual chlorine mode will override all modes except the automatic recirculation mode.
 - 5. The automatic recirculation mode will override all modes.

Power Generation Design Bases

The power generation design bases are

- a. To control the temperature and humidity in the control center for operator comfort and electronic equipment stability. A small net positive pressure is maintained with respect to the outdoors and other areas of the plant on a year-round basis
- b. To indicate temperature and status of operating equipment, such as supply and return air fans and the refrigeration unit, in the main control room
- c. To annunciate on the control panel any operating transients that require operator's attention. This includes high temperature, loss of airflow from supply and return air fans, loss of refrigeration unit, high pressure drop across the supply air filters, and low positive pressure differential between the control center atmosphere and outdoors
- d. To provide capability in the main control room to manually control and operate various components of the control center HVAC system

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- e. To provide a means to test instrumentation and controls and operation of redundant equipment to ensure availability at all times.

7.1.2.1.18 Emergency Equipment Cooling Water System

Safety Design Bases

General Functional Requirements - The instrumentation and control of the emergency equipment cooling water (EECW) system is designed to initiate and maintain operation of the EECW system automatically when normal operation of the reactor building closed cooling water (RBCCW) system is impaired (as indicated by a low differential header pressure), high drywell pressure is experienced, or upon loss of offsite ac power. The controls are provided

- a. To open or close appropriate motor-controlled valves to retain essential cooling circuits and isolate those that are not required to be in service in an emergency
- b. To start operation of the pumps of both loops to establish flow of the emergency equipment service water (EESW) system (the latter is used to remove heat from the EECW system heat exchangers)
- c. To regulate the temperature of the EECW within the required range at the outlet of the EECW system heat exchanger
- d. To maintain the demin level in the EECW makeup tank within the required range during normal plant operation.

Manual controls for initiating operation of the EECW system and its return to the standby state are also provided. The EECW makeup tanks are supplied with demineralized water during normal plant operation. The EECW system makeup tank is supplied via a crosstie line and a makeup pump from the EESW system to provide an alternate makeup supply for each division when the normal makeup supply to the tank is lost during and after the design basis accident. After EECW start, the EECW makeup tanks are replenished and pressurized by makeup pumps utilizing EESW water. The makeup pumps automatically start on makeup tank low pressure or low level, if the makeup tank isolation valve is open and normal makeup pump suction pressure is achieved. Instrumentation and controls are provided to automatically maintain EECW makeup tank pressure, and provide a source of safety-related water (EESW) during EECW system operation.

Specific Regulatory Requirements - The protection system functions contained in functions a. and b. above are required to comply with IEEE 279-1971, IEEE 308-1971, IEEE 323-1971, IEEE 336-1971, IEEE 338-1971 and IEEE 344-1971; GDC 18, GDC 20-24, and GDC 29 of 10 CFR 50, Appendix A and Appendix B; and Regulatory Guide 1.22. EECW system monitoring and control functions c. and d. are required to comply with the requirements of GDC 13.

Power Generation Design Basis

EECW may be manually initiated with the nonessential loads subsequently restored to facilitate RBCCW heat exchanger cleaning, to enhance drywell cooling during high lake water (GSW) temperature, for testing, or to provide RHR Reservoir freeze protection during extreme cold weather. A Loss of RBCCW while EECW is operating in this mode will not

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reinitiate EECW or re-isolate the nonessential loads. This action is not required, however, since this is not a condition requiring protective action as described in Section 7.1.2.1. The demineralized water level in the makeup tank is automatically maintained above a specified minimum amount during normal plant operation. Automatic makeup from EESW is provided for the condition when the normal demineralized water makeup supply is not available and the makeup tank is connected to the EECW loop (i.e., when the P4400F602A(B) valve is open).

7.1.2.1.19 Emergency Core Cooling System Auxiliary Systems

The ECCS auxiliary systems support operation of ECCS equipment. Instrumentation required for operation of this ECCS equipment, therefore, meets the redundancy and separation requirements of the ECCS equipment. The ECCS auxiliary systems consist of

- a. Cooling water (EECW) system described in Subsection 7.1.2.1.18
- b. Essential electric power systems described in Subsection 7.1.2.1.25
- c. Area coolers for rooms and areas containing ECCS equipment (Section 9.4)
- d. Leak detection in ECCS equipment rooms and areas, as described in Subsection 7.1.2.1.26.

Safety Design Basis

The EECW system is designed to be available for essential equipment as outlined in Subsection 7.1.2.1.18. The electric power available for the EECW system is also described in Subsection 7.1.2.1.25. Either the area coolers are designed for operation during emergency conditions, or the ECCS equipment is designed so that loss of the coolers does not jeopardize operation of the ECCS. Leak detection instrumentation monitors primarily for leaks of reactor water or steam. Leak detection instrumentation that automatically isolates ECCS equipment meets the redundancy/ separation requirements for those ECCSs. Subsection 7.1.2.1.26 describes the leak detection instrumentation design bases in more detail.

The EECW system is designed for the maximum expected heat load of ECCS emergency equipment that is used to provide equipment cooling and ventilation space cooling for the HPCI, RCIC, RHR, and core spray systems.

Power Generation Design Bases

The EECW and electrical power sources for the ECCS equipment are maintained in readiness so that they are available when needed. This includes maintaining a minimum level of condensate in the makeup tank. The room air ventilation system is also designed to filter and/or reroute air from rooms where airborne radiation may be present. The LDS initiates an alarm in the main control room in sufficient time for operating personnel to correct or isolate the leak. In some cases the LDS automatically isolates the leaking system.

7.1.2.1.20 Reactor Core Isolation Cooling System

Safety Design Basis - General Functional Requirements

The RCIC system is designed to meet the following general functional requirements:

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- a. The system is capable of maintaining sufficient coolant in the RPV in case of a loss of main feedwater flow
- b. Provisions are made for automatic and remote manual operation of the system
- c. To provide a high degree of assurance that the system operates when necessary, the power supply for the system is from immediately available energy sources of high reliability
- d. To provide a high degree of assurance that the system operates when necessary, provision is made for periodic testing during reactor operation.

Safety Design Basis - Specific Regulatory Requirements

The RCIC system is considered a safety system rather than an ECCS because it is required for safe shutdown. The system is designed to meet the requirements, with exceptions as described in Subsection 7.4.2.2.2, of the federal codes, regulatory guides, and IEEE standards applied to the ESF systems.

7.1.2.1.21 Standby Liquid Control System

Safety Design Bases

General Functional Requirements - The major components of the SLCS consist of a storage tank, two positive displacement pumps, two explosive valves, and two check valves between the explosive valves and the reactor, as shown in Figure 7.4-3. The flow path is from the storage tank through the pumps, explosive valves, and check valves, and into the reactor to the bottom of the core plate. This system is capable of shutting the reactor down from full power to cold shutdown and maintaining the reactor in a subcritical state at atmospheric temperature and pressure conditions by pumping sodium pentaborate, a neutron absorber, into the reactor.

The sodium pentaborate also increases suppression pool pH to prevent iodine re-evolution following a LOCA in the event of fuel failure.

Dual components and dual circuits are used in portions of the system; however, this manually operated system is subject to single failure. Monitoring and testability have been provided for the components and circuits that are deemed most likely to fail. Redundant power sources supply power to this system.

The SLCS electrical components necessary for the injection of boron have been classified as QA Level 1M to indicate that they were not originally intended, procured, designed, or classified as safety related, but will be maintained and tested as a safety-related system.

Specific Regulatory Requirements - General Design Criterion 26 of 10 CFR 50, Appendix A, which requires the provision of an independent method of reactivity control, applies.

Power Generation Design Basis

The system is designed to shut the reactor down from full power to cold atmospheric conditions with sufficient margin to maintain the reactor subcritical at the cold condition in the event that manual control rod movement cannot be accomplished with the RMCS.

7.1.2.1.22 Primary Containment Monitoring System

The primary containment monitoring system consists of four instrumentation subsystems that collectively monitor the primary containment atmosphere for hydrogen concentration, oxygen concentration, gaseous radiation level, and temperature and pressure, and that monitor the pressure suppression pool water for temperature and level. The radiation monitor is principally provided to enhance the capability for detecting reactor water or steam leaks. The radiation monitor activates an alarm upon detecting radiation at or above a predetermined level. Monitoring of the other parameters is provided to secure information on transients resulting from a LOCA. The monitored parameters are indicated and recorded in the main control room.

The primary containment atmosphere monitoring system also serves to provide information on monitored parameters in the course of plant operations when conditions are normal. The four primary containment monitor subsystems are designated

- a. Primary containment radiation monitor and the hydrogen/ oxygen monitor subsystem
- b. Primary containment temperature monitor subsystem
- c. Primary containment pressure monitor subsystem
- d. Pressure suppression pool water level indicator subsystem.

The design bases and regulatory requirements for each of these subsystems are individually defined below.

7.1.2.1.22.1 Primary Containment Radiation Monitor and Hydrogen/ Oxygen Monitor Subsystems

Safety Design Bases

General Functional Requirements - The primary containment radiation monitor is designed to meet the following safety design bases:

- a. Provide continuous radiation monitoring of the primary containment atmosphere during power operation, startup and hot shutdown of the reactor.
- b. Provide particulate and halogen filters in the atmospheric sample flow line to collect integrated samples of these substances, on separate filters, for purposes of radiation analysis
- c. Provides a high-radiation alarm with fully adjustable setpoints in the main control room.
- d. Provide a diverse reactor coolant pressure boundary leak detection method using noble gas activity.

Specific Regulatory Requirements - The instrumentation and control of the primary containment radiation monitor subsystem is designed to conform to General Design Criterion 30 of 10 CFR 50, Appendix A and Regulatory Guide 1.45. The hydrogen/oxygen monitor subsystem is designed to meet Regulatory Guides 1.7 and 1.97, Category 3 and 2 requirements, respectively.

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Power Generation Design Bases

The primary containment radiation monitor and hydrogen/oxygen monitor subsystems are designed to meet the following power generation design bases:

- a. Provide indication in the main control room of the noble gas radioactivity and hydrogen/oxygen content of the primary containment atmosphere during normal operation

The oxygen monitors provide verification of the status of the inerted atmosphere of containment and oxygen levels in the containment atmosphere following a significant beyond-design-basis accident for combustible gas control and accident management, including emergency planning.

The hydrogen monitors provide diagnosis of the course of significant beyond-design-basis accidents for accident management, including emergency planning.

- b. Provide means for obtaining radioactivity analysis of particulate and halogen content in the primary containment atmosphere
- c. Provide an instrument failure (offscale low) alarm.
- d. Provide high hydrogen and high oxygen alarms with fully adjustable setpoints in the main control room.

7.1.2.1.22.2 Primary Containment Temperature Monitor Subsystem

Safety Design Bases

General Functional Requirements - The primary containment temperature monitor subsystem is designed to meet the following safety design bases:

- a. Provide continuous monitoring of the drywell atmosphere temperature with a distributed arrangement of temperature sensors to secure representative temperature data in the drywell region
- b. Provide continuous monitoring of drywell cap atmospheric temperature with a sensor suitably located to secure representative temperature information in the cap region
- c. Provide continuous measurement of drywell wall temperature with an arrangement of sensors distributed to obtain a representative determination of the wall temperature conditions in the drywell region
- d. Provide continuous measurement of atmospheric temperature in the pressure suppression chamber
- e. Provide continuous measurement of water temperature in the pressure suppression chamber.

Safety Design Bases

Specific Regulatory Requirements - The primary containment temperature monitor subsystem is designed to conform to GDC 13 of 10 CFR 50, Appendix A, and QA Criteria of 10 CFR 50, Appendix B, for nuclear power plants.

7.1.2.1.22.3 Primary Containment Pressure Monitor Subsystem

Safety Design Bases

General Functional Requirements - The primary containment pressure monitor subsystem is designed to meet the following safety design bases:

- a. Provide continuous measurement of drywell atmospheric pressure
- b. Provide continuous measurement of pressure suppression chamber atmospheric pressure.

Specific Regulatory Requirements - The primary containment pressure monitor subsystem is designed to meet GDC 13 of 10 CFR 50, Regulatory Guide 1.97, Appendix A, and QA Criteria of 10 CFR 50, Appendix B, for nuclear power plants.

Power Generation Design Basis

The primary containment pressure monitor subsystem provides a chart recorder in the main control room to continuously record and display the primary containment pressure monitored by this subsystem.

7.1.2.1.22.4 Pressure Suppression Pool Water Level Indicator Subsystem

Safety Design Bases

General Functional Requirements - The pressure suppression pool water level indicator subsystem is designed to provide measurement of water level in the pressure suppression chamber over the maximum practical range.

Specific Regulatory Requirements - The pressure suppression pool water level indicator system is designed to meet GDC 13 of 10 CFR 50, Regulatory Guide 1.97, Appendix A, and QA Criteria of 10 CFR 50, Appendix B, for nuclear power plants.

Power Generation Design Basis

The pressure suppression pool water level indicator subsystem is designed to provide a display in the main control room that indicates the water level in the pressure suppression chamber.

7.1.2.1.23 Radwaste Control System

7.1.2.1.23.1 Liquid Radwaste System

The safety design bases ensure that the liquid radwaste system instrumentation and control is designed to provide information to the liquid radwaste process operator. This information is needed to limit releases of radioactivity to the environment. Further discussion can be found in Subsection 7.7.1.6 and in Sections 11.2 and 11.4.

7.1.2.1.23.2 Gaseous Radwaste System

The safety design bases ensure that the gaseous radwaste system instrumentation and control system is designed to monitor and control the gaseous processing systems (offgas system and 2-minute holdup pipe system). It also detects, indicates, and alarms improper or abnormal conditions in the gaseous radwaste systems in time for corrective action. Further discussion can be found in Subsection 7.7.1.5 and in Sections 11.3 and 11.4.

7.1.2.1.24 Reactor Water Cleanup System

The purpose of the RWCU system is to provide continuous processing of the reactor water so that the purity is maintained within specified limits. The system also provides the means for removal of reactor water. For example, to maintain reactor water level during startup, it is necessary to dump water due to swell.

7.1.2.1.25 Power Systems

The power systems considered in this subsection include those electrical power sources used in, or associated with, shutting down the reactor and limiting the release of radioactive material following a design-basis event. The power systems include the standby ac system (Subsection 8.3.1), the plant dc system (Subsection 8.3.2), instrument ac power (Subsection 8.3.1), RPS power supplies (Subsection 8.3.1), and special power supplies for individual systems.

Electrical power supplies are available onsite as required to provide the electrical energy requirements of the ESF and safe- shutdown systems during all safety design-basis events and as long thereafter as required to satisfy safety requirements.

The design of the standby ac power system complies with accepted industrial standards for nuclear power plants and is compatible with the ESF system equipment design and arrangement.

Safety Design Bases

General Functional Requirements - The power systems are designed to meet the following general functional requirements:

a. Standby ac Power System

1. General - The primary requirement of the standby ac power system is to maintain a high degree of reliability and timely availability of power sources for the ESF and safe-shutdown systems. This power is required to be made available promptly, within approximately 10 sec, and automatically on either a failure of preferred power sources at any time, or on a LOCA signal

Before the diesel generator is connected to a bus, all offsite source breakers and bus load breakers, with the exception of certain selected breakers, are signaled to trip. The generator is then sequentially loaded to prevent overload or excessive voltage drop. Shutdown of the diesel

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generator is manual only, except for specific automatic trips to prevent equipment destruction

2. Load Assignment - Assignment of loads to emergency safety system buses is such that failure of a single standby power source does not prevent a safe shutdown of the reactor under conditions of a design-basis accident (DBA) concurrent with a design-basis event. "Design-basis event" is used here in the same sense as defined in IEEE 308-1971, i.e., any or all of a set of postulated environmental events for which the plant and ESF systems have been designed

Automatic starting is required of all loads that may be required within 10 minutes after a LOCA. Automatically started loads may be stopped manually and other loads started manually as required by plant conditions.

Automatically connected loads include the emergency core cooling pumps and valves, safety-related instrument power supply transformers, containment isolation valves (ac only), drywell cooling equipment, emergency lighting, standby gas treatment and control center heating, ventilation, and air-conditioning, main control room habitability, EECW system, ECCS room coolers, auxiliary building heating and ventilation, EDG auxiliary equipment, RHR complex ventilation equipment, and reactor building sumps

Manually connectable loads are defined in Table 8.3-3.

- b. Plant dc Power System - The primary requirement of the plant dc power system is to maintain highly reliable and continuously available sources of dc power for the control of a minimum complement of the ECCS and the ac power system equipment during operating conditions and during a DBA concurrent with a design-basis event

Voltage variations are maintained within the demonstrated operating limits of each connected device with appropriate allowances for voltage drop in the cabling. Control battery terminal voltage range on a 130-V dc system is discussed in section 8.3.2.2.4

The dc power sources for redundant ESF equipment must be arranged so as not to compromise the required independence or reduce redundancy below an acceptable level during a design-basis event (i.e., loss of one battery shall not disable any ESF function)

The EDGs are equipped with sufficient protective devices to prevent destruction of the unit, e.g., overspeed trip, low oil pressure trip, generator differential relays, and crankcase overpressure. Other protective devices are used for protection when in test mode, but such devices alarm only when the unit is required to perform the designed safety function

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- c. Instrument ac Power - Power for process instrumentation associated with redundant ESF systems is to be provided by a standby source from the same division as the pump motors and ac valve motors for each system. This power is not classified as essential to nuclear safety, but is to be made available automatically when the bus to which it is connected is energized

The instrumentation power for the HPCI and RCIC systems is to be from a separate inverter that feeds from the same station battery that powers HPCI and RCIC controls, respectively.

- d. Other Power Supplies

1. Reactor Protection System - Power supplies for the RPS are required to have sufficient stored energy to ride through switching transients within the switchyard or auxiliary power system. The safe failure characteristic of the RPS on loss of power exempts the RPS power supplies from being classified essential. However, redundancy is provided to avoid unnecessary plant shutdown on interruption of power to one RPS bus
2. Process Radiation Monitoring System - Certain aspects of the PRMS require 120-V ac power for purposes of recording and/or control. This power is provided from an instrument bus or an inverter power supply as appropriate.

Safety Design Bases

Specific Regulatory Requirements - The standby ac power and dc power systems are essential to safe shutdown of the reactor and/or for emergency core cooling, and therefore comply with all applicable AEC and IEEE standards for design, qualification, and testing. These include IEEE 279-1971, IEEE 308-1971, IEEE 323-1971, IEEE 338-1971, IEEE 344-1971; GDC 1 through 5, 12, 18 and 19 of 10 CFR 50, Appendix A, and Regulatory Guides 1.6 and 1.9.

7.1.2.1.26 Leak Detection System

7.1.2.1.26.1 Reactor Coolant Pressure Boundary Leakage Detection

Safety Design Bases

General Functional Requirements - The safety design basis for the LDS for setting leakage rate limits is that signals are provided to permit isolation of abnormal leakage before the results of this leakage become unacceptable.

The unacceptable results are a threat of significant compromise to the nuclear system process barrier and a leakage rate in excess of the coolant makeup capability to the reactor vessel.

Specific Regulatory Requirements - The part of leak detection that is related to isolation circuits is designed to meet requirements of the ESF systems and to conform to those federal codes, regulatory guides, and IEEE standards which apply to ESF systems.

Power Generation Design Basis

A means is provided to detect abnormal leakage from the nuclear system process barrier.

7.1.2.1.26.2 Emergency Core Cooling System Suction Line Detection

The ECCS suction line LDS is designed to provide information that would allow the manual closing of the valve in the broken line before the net positive suction head (NPSH) is lost to the redundant ECCS.

7.1.2.1.27 Reactor Shutdown Cooling System

Safety Design Bases

The instrumentation and control for the reactor shutdown cooling mode of the RHR system is designed to meet the following functional design bases:

- a. Instrumentation and manual control are provided to enable the system to remove the residual heat (decay heat and sensible heat) from the RPV during normal shutdown
- b. All facilities for manual control of the shutdown cooling system are provided in the main control room
- c. Response of the shutdown cooling system is indicated by main control room instrumentation.

Power Generation Design Bases

The instrumentation and control for the reactor shutdown cooling system is designed to meet the following power generation design bases:

- a. Provide cooling for the reactor during the shutdown operation when the vessel pressure is below the design pressure of the shutdown piping system
- b. Cool the reactor water to a temperature which is practical for refueling and servicing operation.

7.1.2.1.28 Plant Cooling Systems

Two closed cooling water systems are used at Fermi 2 for removal of heat from equipment and space coolers. These are the RBCCW system and the turbine building closed cooling water (TBCCW) system, both described in Section 9.2. The EECW system, an ESF described in Subsection 7.3.4.2, forms an integral part of the RBCCW system.

7.1.2.1.28.1 Reactor Building Closed Cooling Water System Power Generation Design Bases

The instrumentation and control of the RBCCW system is designed in accordance with the following functional requirements:

- a. It maintains the required flow of cooling water in the system and its two divisions during normal conditions and postulated abnormal conditions of the plant

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- b. On loss of offsite power, high drywell pressure, or on drop of differential pressure across the supply and return headers of either division beyond the preset limit, the EECW system will automatically isolate areas of the RBCCW system not essential for emergency cooling and to take over supplying the coolant flow that is required. A loss of RBCCW while EECW is operating for RBCCW heat exchanger cleaning, enhanced drywell cooling, testing, or RHR reservoir freeze protection will not reinitiate EECW or reisolate the nonessential loads. This action is not required, however, since this is not a condition requiring protective action as described in Section 7.1.2.1.
- c. Restoration of the system to normal operation is by manual control.

7.1.2.1.28.2 Turbine Building Closed Cooling Water System Power Generation Design Bases

The instrumentation and control of the TBCCW system is designed in accordance with the following functional requirements.

- a. It maintains the required flow of cooling water in this system during normal conditions of plant operation
- b. It automatically becomes deactivated on loss of offsite power
- c. Restoration of the system to normal operation after gain of power is by manual control.

7.1.2.1.29 Fuel Pool Cooling and Cleanup System

The fuel pool cooling and cleanup system (FPCCS) instrumentation and control is not required for power generation. Its function is to provide annunciation and control so that the FPCCS can maintain the spent fuel and equipment storage pools and the reactor water well below a desired temperature and at a degree of clarity necessary to refuel and service the reactor.

7.1.2.1.30 Post-LOCA Combustible Gas Control System

The NRC amended 10 CFR 50.44, "Standards for combustible gas control system in light-water-cooled power reactors" on October 16, 2003 to eliminate the requirements for hydrogen recombiners. The hydrogen recombiner Technical Specification requirements were subsequently removed by License Amendment 159, dated March 15, 2004. The wording in this UFSAR section associated with these changes will remain unaltered until after the hydrogen recombiner system has been abandoned in place or removed from the plant.

The instrumentation and control for the post-LOCA combustible gas control system (CGCS) is designed to meet the following functional safety design bases:

- a. Provides for manual actuation of the system by placing the selector switch in the main control room in the READY position, and then manually initiating the system from the relay room cabinet, energizing the blower, starting water flow to the gas cooler, and regulating electric power to the heater elements

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- b. On the basis of signals from redundant thermocouples, maintains the proper temperature of the gas leaving the heater section by regulating electric power to the heaters
- c. When the gas temperature reaches the preset operating setpoint (about 1200°F), further throttle open (manually) the gas inlet valves as required, to maximize the flow of gas from the containment atmospheres (drywell or suppression chamber as selected by the reactor operator)
- d. Manually adjusts and regulates the flow mix between the gas being recirculated from the recombiner outlet and the flow from the containment atmosphere, in order to maintain a predetermined combustible gas concentration in the gas mixture entering the heater unit and recombiner chamber
- e. Provides automatic regulation of the heater power to maintain the proper operating temperature in the recombiner chamber
- f. Provides heater overtemperature protection and such other alarms, instruments, and signals that permit the reactor operator to readily determine the operational status and performance of the CGCS
- g. Meets applicable regulatory requirements of IEEE 279- 1971, Regulatory Guide 1.7, and GDC 41 of Appendix A of 10 CFR 50.

7.1.2.1.31 Control Air System

The instrumentation and control of the control air system is designed in accordance with the following functional requirements.

- a. The control air system maintains the required quantity and quality of control air to both interruptible and noninterruptible control air users
- b. On loss of control air pressure below a preset limit, the control air compressors are automatically started. If pressure continues to decay further to a lower preset value, the two divisions of the control air system are automatically isolated from all interruptible control air users and the station air system so that each control air compressor is supplying only its own essential division
- c. It provides for manual actuation of the system from the main control room for testing of the system or for manual initiation of the system.

7.1.2.1.32 Alternate Rod Insertion

The safety design bases are as follows:

- a. The sensors, transmitters, trip units, and associated logic for the ARI are Class 1E, redundant to the reactor protection system, and environmentally and seismically qualified to IEEE 323-1974 and IEEE 344-1975
- b. The ARI sensors monitor reactor pressure and water level and trip the reactor if these variables reach their respective trip setpoints. The trip is accomplished by energizing the ARI valves, thereby venting the air supply holding the scram valves shut.

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7.1.2.2 Independence of Redundant Safety-Related Systems

The criteria for the separation of safety-related mechanical and electrical equipment are discussed in Section 3.12 and Subsection 8.3.1. The independence of redundant safety-related systems satisfies the applicable requirements of IEEE 279-1971. The requirements of 10 CFR 50, Appendix B, are met as described in Chapter 17.

7.1.2.3 Physical Identification of Safety-Related Equipment

Equipment associated with the RPS, the ESF, the safe shutdown systems, and the auxiliary electrical equipment associated with these systems are identified so that it is apparent that

- a. The equipment is part of the RPS, ESF, or safe shutdown system
- b. The equipment item is associated with a particular grouping (or division) of enforced segregation.

The identification consists of marking panels and equipment racks with marker plates that are conspicuously different in color than those for other panels or racks. These markers include identification of the proper division of the equipment within the system.

The equipment identification number and the applicable segregation code, both numerical and color code, are applied to each piece of safety-related equipment either before or during that equipment's installation.

7.1.2.4 Conformance To IEEE-317

Qualification of the penetration assemblies and their associated electrical services is provided by compliance with IEEE 317-1972. Power cables are provided with reliable decoupling devices at their load centers to ensure fault interruption prior to any penetration damage. All cables having safety-related functions are separated from their redundant counterparts in different penetration assemblies.

7.1.2.5 Conformance To IEEE-323

IEEE 323-1971 applies to equipment purchased before November 15, 1974, and IEEE 323-1974 applies to equipment purchased on or after November 15, 1974.

Written procedures and responsibilities are developed for the design and qualification of all Class 1 electric equipment. This includes preparation of specifications, qualification procedures, and documentation for Class 1 equipment. Qualification testing or analysis is accomplished prior to release of the engineering design for production. Standards manuals are maintained containing specifications, practices, and procedures for implementing qualification requirements; and an auditable file of qualification documents is available for review.

7.1.2.6 Conformance To IEEE-336

The implementation of the Quality Assurance Procedures for construction activities ensures compliance with the requirements of IEEE 336-1971.

7.1.2.7 Conformance To IEEE-338 and Regulatory Guide 1.22

For a more detailed description of conformance, for all safety-related systems, see Subsections 7.2.2, 7.3.2, 7.4.2, and 7.6.2.

7.1.3 Protection System Inservice Testability

This section is provided to describe the analog transmitter/trip unit (AT/TU) system. The AT/TU system is a plant protection system testability feature generically applied to the reactor protection (trip) system, ESF systems, and the RCIC system.

The AT/TU system provides highly accurate continuous monitoring of process parameters, excellent setpoint stability, and convenient on-line testability.

For additional testability discussions, refer to Topical Report NEDO-21617-A, dated December 1978, Analog Transmitter/Trip Unit System for Engineered Safeguard Sensor Inputs (Reference 1).

7.1.3.1 General Description

The AT/TU system uses analog instrument channels to monitor important plant variables (e.g., reactor water level, reactor pressure, drywell pressure, and process flow). The analog transmitter converts the process variable sensed to a 4- to 20-mA linear signal. The signal is transmitted to electronic trip units located on the fourth floor of the auxiliary building. The trip units compare the transmitted signal with a fixed reference signal (setpoint). When the transmitted signal increases above or decreases below the setpoint, the trip unit activates an associated relay. The relay provides either open or closed contacts on activation.

The trip units consist of master trip assemblies, slave trip assemblies, and calibration units. The master trip unit is a plug-in printed circuit assembly designed to accept a 4- to 20-mA signal from a remote transmitter. The trip unit contains the circuitry necessary to condition the transmitter current, compare with the setpoint, provide trip output, and provide analog output signals. An alarm is generated by an inoperative or out-of-service trip unit. The master trip unit also contains a panel meter that displays transmitter current and is scaled in the units of the process variable being measured by the transmitter wired to the master trip unit. A switch position selection internal to the master trip unit allows for selection of either high trip point or low trip point. This allows the testing of trip circuitry for a particular channel with the trip circuitry either energized or deenergized during normal operation.

The slave trip unit is used in conjunction with a master trip unit when different setpoints from a common transmitter are desired. The slave trip unit receives its input signal from the analog output of a master trip unit. There is no direct connection to any 4- to 20-mA transmitter. No analog output signals are generated by the slave unit. Calibration of the slave unit is accomplished by commanding the master trip unit, which drives the slave unit under test into the calibration mode, and then performing the normal calibration procedure.

The calibration unit furnishes the means by which an in-place calibration check of the master and slave trip units can be performed. The calibration unit contains a stable current source and a transient current source. The stable current source is used to verify the calibration

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point of any given channel. The transient current source is used to provide step current input into a selected trip unit so that the response time of that channel can be determined.

During calibration, the trip action is displayed on the removable display assembly. The accuracy of the analog output of the master trip unit may also be checked during the calibration procedure with an external meter or recorder.

7.1.3.2 Analysis

For a discussion of conformance with regulatory guides and IEEE standards, see Reference 1.

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7.1 INTRODUCTION

REFERENCES

1. General Electric Company, Analog Transmitter/Trip Unit System for Engineered Safeguard Sensor Inputs, NEDO-21617-A, December 1978.

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TABLE 7.1-1 LICENSED REACTOR SYSTEMS FUNCTIONALLY IDENTICAL TO FFERMI 2*

<u>System</u>	<u>Plants With Construction Permit or Operating License</u>
Reactor protection system	Hatch 1, Duane Arnold
Primary containment and RPV isolation control system	Hatch 1, Duane Arnold
Emergency core cooling system	Hatch 1, Duane Arnold
Neutron monitoring system	Hatch 1, Duane Arnold
Refueling interlocks	Hatch 1, Duane Arnold, Dresden 2 and 3
Reactor core isolation cooling system	Hatch 1, Duane Arnold
Standby liquid control system	Dresden 2 and 3
Reactor water cleanup system	Hatch 1, Duane Arnold
Shutdown cooling system	Hatch 1, Duane Arnold

* This table was a true comparison with the listed Nuclear Power Plants' systems at the time NRC issued NUREG-0798, "Safety Evaluation Report Related to the Operation of Enrico Fermi Atomic Power Plant, Unit No. 2.," July 1981. Refer to pages 7-1 and 7-2 in the SER for the NRC acknowledgement.

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TABLE 7.1-2 INSTRUMENTATION AND CONTROL SYSTEMS DESIGNERS AND FABRICATORS

	<u>System</u>	<u>Designer</u>	<u>Fabricator</u>
1.	Reactor protection system	GE	GE
2.	Containment and RPV isolation control system	Edison/GE	GE
3.	Emergency core cooling system:	GE	GE
	a. HPCI c. ADS		
	b. LPCI d. CS system		
4.	Neutron monitoring system	GE	GE
5.	Refueling interlocks	GE	GE
6.	RPV power generator instrumentation	GE	GE
7.	Recirculation flow control system	GE	GE
8.	Feedwater control system	GE	GE
9.	Pressure regulator and turbine generator control system	GE/GEC	GEC
10.	Process radiation monitoring system	Edison/GE	GE/Gulf GA Eberline
a.	Process liquid radiation monitoring system	GE/Gulf GA	GE/Gulf GA
	(1) Radwaste bldg. effluent radiation monitor	Edison/GE	GE
	(2) General service water effluent radiation monitor	GE	GE
	(3) Circulating water reservoir decant radiation monitor	Gulf GA	Gulf GA
	(4) RBCCW system radiation monitor	GE	GE
	(5) EECW system radiation monitor	Gulf GA	Gulf GA
	(6) RHRSW radiation monitor	Gulf GA	Gulf GA

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TABLE 7.1-2 INSTRUMENTATION AND CONTROL SYSTEMS DESIGNERS AND FABRICATORS

	<u>System</u>	<u>Designer</u>	<u>Fabricator</u>
b.	Main steam line radiation monitor system	GE	GE
c.	Offgas system radiation monitors		
(1)	2-minute holdup pipe radiation monitor	Gulf GA	Gulf GA
(2)	Offgas radiation monitor	GE	GE
d.	Reactor bldg. exhaust plenum radiation monitor	Edison	Eberline
e.	Reactor bldg. ventilation exhaust radiation monitor	Edison	Gulf GA
f.	Fuel pool ventilation exhaust radiation monitor	GE	GE
g.	Standby gas treatment system exhaust radiation monitor	Edison	Eberline
h.	Control Center makeup air radiation monitor	Edison	Gulf GA
i.	Radwaste bldg. ventilation exhaust radiation monitor	Edison	Eberline
j.	Deleted		
k.	Turbine bldg. ventilation exhaust radiation monitor	Edison	Eberline
11.	Area radiation monitoring system	Edison/GE	GE
12.	Site environs radiation monitoring system	Edison	Refer to Chapter 11
13.	Health physics and laboratory analysis radiation monitoring system	Edison	Refer to Chapter 12
14.	Integrated plant computer system	DS&S	Various

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TABLE 7.1-2 INSTRUMENTATION AND CONTROL SYSTEMS DESIGNERS AND FABRICATORS

	<u>System</u>	<u>Designer</u>	<u>Fabricator</u>
15.	Standby gas treatment system	CVI Inc.	CVI Inc.
16.	Control center HVAC control system	Edison	Various
17.	Emergency equipment cooling water control system	Edison	Various
18.	Emergency core cooling system, auxiliary systems, control systems	Edison	Various
19.	Reactor core isolation cooling system	GE	GE
20.	Standby liquid control system	GE	GE
21.	Primary containment monitor system		
a.	Primary containment radiation monitor and hydrogen/oxygen monitoring system	Edison	Exo-Sensors/ GA
b.	Primary containment temperature monitor system	Edison	Various
c.	Primary containment pressure monitor system	Edison	Various
d.	Pressure suppression pool water level indicator system	Edison	Various
22.	Radwaste control system	GE	GE
a.	Liquid radwaste system	NUS	NUS/Edison
b.	Gaseous radwaste system	Edison/ Kraftwerk Union	Various
23.	Reactor water cleanup system	GE	GE
24.	Power systems		
a.	Standby ac	Edison	Various
b.	Plant dc	Edison	Various
c.	Instrumentation ac	Edison	Various

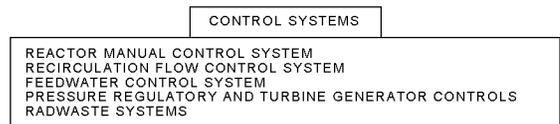
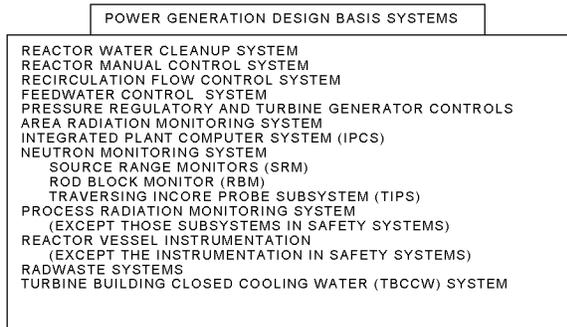
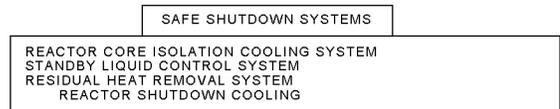
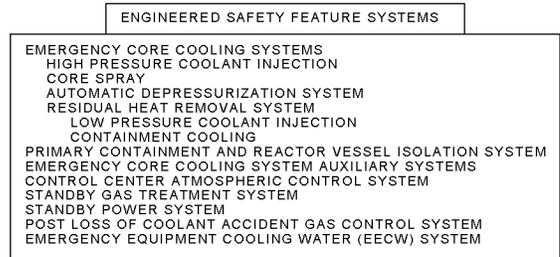
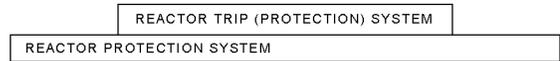
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TABLE 7.1-2 INSTRUMENTATION AND CONTROL SYSTEMS DESIGNERS AND FABRICATORS

	<u>System</u>	<u>Designer</u>	<u>Fabricator</u>
25.	Leak detection system		
	Reactor coolant pressure boundary leakage detection	GE	GE
26.	Residual heat removal shutdown cooling control system	GE	GE
27.	Plant cooling system		
a.	Reactor building closed cooling water system	Edison	Edison/ Erector
b.	Turbine building closed cooling water system	Edison	Edison/ Erector
28.	Fuel pool cooling and cleanup system	GEGE	
29.	Reactor/Auxiliary building HVAC and pressure control	A. H. Smith Associates	Various
30.	Post-LOCA combustible gas control system	AI	AI
31.	Remote shutdown system	Edison	Reliance
32.	Turbine-generator overspeed trip	GEC/Edison/ GE (set points)	GEC/Edison
33.	Vital buses/load-shedding instrumentation and control	Edison	Various
34.	Plant emergency communication	Edison	GAI/Tronics
35.	Supplemental cooling chilled water system	Edison	Various

FUNCTIONAL CLASSIFICATION

REGULATORY CLASSIFICATION



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FIGURE 7.1-1
PLANT INSTRUMENTATION AND CONTROL
SYSTEMS CLASSIFICATION

	SCRAM DISCHARGE VOLUME	MSL ISOLATION VALVE CLOSURE	TURBINE STOP VALVE CLOSURE	TURBINE CONTROL VALVE FAST CLOSURE	REACTOR LOW WATER LEVEL	MSL HIGH RADIATION	NEUTRON MONITORING SYSTEM IRM	NEUTRON MONITORING SYSTEM APRM	DRYWELL HIGH PRESSURE	REACTOR HIGH PRESSURE	MANUAL SWITCH INPUTS	BYPASS INPUTS	TRIP LOGIC TRIP ACTUATOR OUTPUTS
IEEE-279	X	X	X	X	X	X	X	X	X	X	X	X	X
IEEE-308													
IEEE-317													
IEEE-323	X	X	X	X	X	X	X	X	X	X	X	X	X
IEEE-344	X	X			X	X	X	X	X	X	X	X	X
IEEE-336	X	X	X	X	X	X	X	X	X	X	X	X	X
IEEE-338	X	X	X	X	X	X	X	X	X	X	X	X	X
IEEE-334													
R.G. 1.6													
R.G. 1.9													
R.G. 1.11		X			X		X	X	X	X			
R.G. 1.12													
R.G. 1.21													
R.G. 1.22	X	X	X	X	X	X	X	X	X	X	X	X	X
R.G. 1.30	X	X	X	X	X	X	X	X	X	X	X	X	X
GDC 10								X					
GDC 12								X					
GDC 13	X	X	X	X	X	X	X	X	X				
GDC 17													
GDC 18													
GDC 19											X		
GDC 20	X	X	X	X	X	X	X	X	X	X		X	X
GDC 21	X	X	X	X	X	X	X	X	X	X	X	X	X
GDC 22	X	X	X	X	X	X	X	X	X	X	X	X	X
GDC 23	X	X	X	X	X	X	X	X	X	X	X	X	X
GDC 24	X	X	X	X	X	X	X	X	X	X	X	X	X
GDC 25													

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FIGURE 7.1-2

REACTOR PROTECTION SYSTEM MATRIX

	REACTOR LOW WATER LEVEL	MSL HIGH RADIATION	MSL HIGH FLOW	MSL SPACE HIGH TEMPERATURE	MSL SPACE HIGH DIFF. TEMPERATURE	REACTOR LOW PRESSURE	DRYWELL HIGH PRESSURE	PROCESS VENTILATION RADIATION MON	REACTOR CU LOOP HIGH FLOW	REACTOR CU LOOP HIGH SPACE TEMPERATURE	REACTOR CU LOOP HIGH SPACE DIFF TEMPERATURE	MANUAL SWITCH INPUTS	BYPASS INPUTS	TRIP LOGIC TRIP ACTUATOR OUTPUTS
IEEE-279	X	X	X	X	X	X	X	X	X	X	X	X	X	X
IEEE-308														
IEEE-317														
IEEE-323	X	X	X	X	X	X	X	X	X	X	X	X	X	X
IEEE-344	X	X	X	X	X	X	X	X	X	X	X	X	X	X
IEEE-336	X	X	X	X	X	X	X	X	X	X	X	X	X	X
IEEE-338	X	X	X	X	X	X	X	X	X	X	X	X	X	X
IEEE-334														
R.G. 1.6														
R.G. 1.9														
R.G. 1.11	X					X	X							
R.G. 1.12														
R.G. 1.21														
R.G. 1.22	X	X	X	X	X	X	X	X	X	X	X	X		X
R.G. 1.30	X	X	X	X	X	X	X	X	X	X	X	X	X	X
GDC 13	X	X	X	X	X	X	X	X	X	X	X			
GDC 17														
GDC 18														
GDC 19												X		
GDC 20	X	X	X	X	X	X	X	X	X	X	X		X	X
GDC 21	X	X	X	X	X	X	X	X	X	X	X	X	X	X
GDC 22	X	X	X	X	X	X	X	X	X	X	X	X	X	X
GDC 23	X	X	X	X	X	X	X	X	X	X	X	X	X	X
GDC 24	X	X	X	X	X	X	X	X	X	X	X	X	X	X
GDC 25														

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FIGURE 7.1-3

CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM MATRIX

	REACTOR LOW WATER LEVEL	DRYWELL HIGH PRESSURE	LPCI FLOW SUFFICIENT	LPCI BATTERY VOLTAGE	CSA/RHRA/RHRC BATTERY VOLTAGE	CSA/RHRA/RHRC EMERG BUS VOLTAGE	CSA/RHRA/RHRC FLOW SUFFICIENT	CSA/RHRA/RHRC INJECTION VALVE ΔP	CSB/RHRB/RHRD BATTERY VOLTAGE	CSB/RHRB/RHRD BUS VOLTAGE	CSB/RHRB/RHRD FLOW SUFFICIENT	CSB/RHRB/RHRD INJECTION VALVE ΔP	MANUAL SWITCH	BYPASS INPUTS	TRIP LOGIC TRIP ACTUATOR OUTPUTS
IEEE-279	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN
IEEE-308			XN		XN	XN	XN			XN	XN				
IEEE-317															
IEEE-323	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
IEEE-344	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
IEEE-336	XC	X	X	X	X	X	X	X	X	X	X	X	X	X	X
IEEE-338	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
IEEE-384															
IEEE-387			X				XN				XN				
R.G. 1.6			XN		XN	XN	XN			XN	XN				
R.G. 1.9			X				X				X				
R.G. 1.11	X	X													
R.G. 1.12															
R.G. 1.21															
R.G. 1.22	X	X											X		X
R.G. 1.30	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
GDC 13	X	X	X	X	X	X	X	X	X	X	X	X			
GDC 17			XN		XN	XN	XN			XN	XN				
GDC 18			X		X	X	X			X	X				
GDC 19			X										X		
GDC 20	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN		XN	XN
GDC 21	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN
GDC 22	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN
GDC 23	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN
GDC 24	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X
GDC 25															
GDC 29	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN
GDC 35	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN
GDC 37	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN

x = Applicable
xn = Applicable on network basis

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FIGURE 7.1-4

LOW PRESSURE ECCS – LPCI, CS A, CS B, RHR A, RHR B, RHR C, RHR D NETWORK

	REACTOR LOW WATER LEVEL	PRIMARY CONTAINMENT HIGH PRESSURE	HPCI FLOW SUFFICIENT	HPCI BATTERY VOLTAGE	ADS A BATTERY VOLTAGE	ADS A A-C INTLK. PERMISSIVE	ADS A TIMER	ADS B BATTERY VOLTAGE	ADS B A-C INTLK. PERMISSIVE	ADS B TIMER	MANUAL SWITCH INPUTS	BYPASS INPUTS	TRIP LOGIC - TRIP ACTUATOR OUTPUTS
IEEE-279	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN
IEEE-308				XN	XN			XN					
IEEE-317													
IEEE-323	X	X	X	X	X	X	X	X	X	X	X	X	X
IEEE-336													
IEEE-338	X	X	X	X	X	X	X	X	X	X	X	X	X
IEEE-344	X	X	X	X	X	X	X	X	X	X	X	X	X
IEEE-334													
IEEE-387													
R.G. 1.6				XN	XN			XN					
R.G. 1.9													
R.G. 1.11	X	X											
R.G. 1.12													
R.G. 1.21													
R.G. 1.22	X	X									X		X
GDC 13	X	X	X	X	X	X		X	X				
GDC 17				XN	XN			XN					
GDC 18				X									
GDC 19											X		
GDC 20	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN		XN	XN
GDC 21	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN
GDC 22	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN
GDC 23	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN
GDC 24	X	X	X	X	X	X	X	X	X	X	X	X	X
GDC 29	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN
GDC 35	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN
GDC 37	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN	XN

X = APPLICABLE
XN = APPLICABLE ON NETWORK BASIS

<p>Fermi 2 UPDATED FINAL SAFETY ANALYSIS REPORT</p>
<p>FIGURE 7.1-5 HIGH PRESSURE ECCS - HPCI AND ADS NETWORK</p>

7.2 REACTOR PROTECTION SYSTEM

7.2.1 Description

7.2.1.1 Reactor Protection System Instrumentation and Control System Description

7.2.1.1.1 System Identification

7.2.1.1.1.1 Identification

The reactor protection system (RPS) includes the motor-generator power supplies, sensors, relays, bypass circuitry, and switches that cause rapid insertion of control rods (scram) to shut down the reactor. It also includes outputs to the Integrated Plant Computer System (IPCS) and annunciators, although these latter two systems are not part of the RPS. Trip functions are summarized in Figure 7.2-1.

A completely redundant capability, the alternate rod insertion function of the control rod drive (CRD) system is provided to mitigate anticipated transient without scram (ATWS) events (see Subsection 7.6.1.18).

7.2.1.1.1.2 Classification

The RPS is classified as Safety Class 2, Category I, and Quality Group B.

7.2.1.1.1.3 Reference Design

The Fermi 2 RPS is similar, except for system size, to the Edwin I. Hatch, Unit 1 RPS. There are no differences other than those instrument panel locations within the plant and manual scram logic arrangement.

7.2.1.1.2 Power Sources

The RPS receives power from two high-inertia ac motor-generator sets (Figure 7.2-2). A flywheel provides high inertia sufficient to maintain voltage and frequency within 5 percent of rated values for at least 1 sec following a total loss of power to the drive motor.

Alternate power is available to reach the RPS buses. The 120-V ac supply bus A is available to RPS bus A, and the 120-V ac alternate supply bus B is available to RPS bus B.

The RPS power supplies have been modified to prevent the inadvertent application of out-of-tolerance voltage or frequency power to the RPS relay trip logic. The electrical protection assembly consists of a GE type TFJ-175A circuit breaker with an under-voltage release controlled by a protection logic circuit card. The protection logic disconnects the RPS logic from the RPS power supply whenever voltage or frequency exceeds normal tolerances.

The protection is redundant and includes each alternate power supply, as shown in Figure 7.2-2. The electrical protection assemblies (EPA) are packaged in enclosures that are mounted seismically on the outside wall of each RPS motor-generator set cubicle. Two assemblies are connected in electrical series between each source of RPS power and the

respective RPS distribution panel. Controls for testing and operation are provided on each assembly along with status indication for the particular trip parameters. Following a trip, the breaker must be reset locally.

The EPA's are qualified to meet IEEE 344-1975 and IEEE 323-1974.

The EPA trip setpoints are within ± 10 percent of nominal ac voltage and -5 percent of the nominal frequency of 60 Hz.

Each protection logic has an independent time delay adjustable from 0.3 to 3.6 sec to prevent spurious trips and the resulting scrams.

7.2.1.1.3 Equipment Design

7.2.1.1.3.1 Initiating Circuits

Neutron monitoring system (NMS) instrumentation is described in Section 7.6. Figure 7.2-3 clarifies the relationship between NMS channels, NMS logics, and the RPS logics. The NMS channels are part of the NMS. The NMS logics are part of the RPS. As shown in Figure 7.2-4, there are four NMS logics associated with each trip system of the RPS. Each RPS logic receives inputs from two NMS logics. Each NMS logic receives signals from one intermediate range monitor (IRM) channel and one average power range monitor (APRM) voter channel. The position of the mode switch determines which input signals effect the output signal from the logic. The NMS logics are arranged so that failure of any one logic cannot prevent the initiation of a high neutron flux scram. The RPS logic is a "one-out-of-two-taken-twice" system as discussed in Subsection 7.2.1.1.3.2.

Reactor pressure is measured at two locations. A pipe from each location is routed through the primary containment and terminates in the reactor building. Two panel-mounted pressure transmitters monitor the pressure in each pipe. Cables from these transmitters are routed to the main control room. One pair of the transmitters is physically separated from the other pair. Each transmitter provides a high-pressure signal to one channel. The transmitters are arranged so that two transmitters provide an input to trip system A and two transmitters provide an input to trip system B, as shown in Figure 7.2-5. The physical separation and the signal arrangement ensure that no single physical event can prevent a scram caused by nuclear system high pressure.

Reactor pressure vessel (RPV) low-water-level signals are initiated from differential pressure transmitters that sense the difference between the pressure due to a constant reference column of water and the pressure due to the actual water level in the vessel. A reference leg backfill system provides a continuous flow of water from the CRD charging header to the reactor water level reference legs. This flow prevents accumulation of non-condensable gases in the reference leg, and the associated erroneous high water level indication which could result from degassing in the reference leg upon system depressurization. The transmitters are arranged on two sets of taps in the same way as the nuclear system high pressure transmitters (Figure 7.2-5).

Two instrument lines attached to taps on the RPV, one above and one below the water level, are required for the differential pressure measurement for each transmitter. The two pairs of lines terminate outside the primary containment and inside the reactor building. They are

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physically separated from each other and tap off the RPV at widely separated points. Other systems sense pressure and level from these same pipes. The physical separation and signal arrangement ensure that no single physical event can prevent a scram due to RPV low water level.

Turbine stop valve closure inputs to the RPS come from valve stem position switches mounted on the four turbine stop valves. To provide the earliest positive indication of closure, each of the double-pole, double-throw switches opens before the valve is more than 10 percent closed. Either of the two channels associated with one stop valve can signal valve closure, as shown in Figure 7.2-6. The logic is arranged so that closure of three or more stop valves initiates a scram, when the reactor is operating above 29.5 percent of rated power.

Turbine control valve fast closure inputs to the RPS come directly from contacts of the relays that effect control valve fast closure. Operation of any two of these relays will initiate control valve fast closure. Fast closure of one control valve in each RPS logic will initiate a scram whenever the reactor is operating above 29.5 percent of rated power.

Position switches mounted on the eight main steam isolation valves (MSIVs) signal MSIV closure to the RPS. To provide the earliest positive indication of closure, each of the double-pole, double-throw switches is arranged to open before the valve is more than 10 percent closed. Either of the two channels associated with one isolation valve can signal valve closure. To facilitate the description of the logic arrangement, the position-sensing channels for each valve are identified and assigned to RPS logics as follows:

<u>Valve Identification</u>	<u>Position-Sensing Channels</u>	<u>Trip Channel Relays</u>	<u>Assignments</u>
Main steam line A, inboard valve	F022A (1) and (2)	A, B	A1, B1
Main steam line A, outboard valve	F028A (1) and (2)	A, B	A1, B1
Main steam line B, inboard valve	F022B (1) and (2)	E, D	A1, B2
Main steam line B, outboard valve	F028B (1) and (2)	E, D	A1, B2
Main steam line C, inboard valve	F022C (1) and (2)	C, F	A2, B1
Main steam line C, outboard valve	F028C (1) and (2)	C, F	A2, B1
Main steam line D, inboard valve	F022D (1) and (2)	G, H	A2, B2
Main steam line D, outboard valve	F028D (1) and (2)	G, H	A2, B2

Thus, each logic receives signals from the valves associated with two steam lines as shown in Figure 7.2-7. The arrangement of signals within each logic requires closing of at least one

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valve in each of the steam lines associated with that logic to cause a trip of that logic. For example, closure of the inboard valve of steam line A and the outboard valve of steam line C causes a trip of logic B1. This in turn causes trip system B to trip. No scram occurs because no trips occur in trip system A. In no case does closure of two valves or isolation of two steam lines cause a scram due to valve closure. Closure of one valve in any three steam lines causes a scram.

Wiring for the position-sensing channels from one position switch is physically separated in the same way that wiring to duplicate sensors on a common process tap is separated. The wiring for position-sensing channels feeding the different trip logics of one trip system is also separated.

The MSIV closure scram function is effective only if the reactor mode switch is in RUN.

The effects of the logic arrangement and separation provided for the MSIV closure scram are as follows:

- a. Closure of one valve for test purposes with one steam line already isolated will not cause a scram resulting from valve closure
- b. Automatic scram will occur on isolation of any three steam lines
- c. No single failure can prevent an automatic scram required for fuel protection due to MSIV closure.

Four nonindicating level switches (one for each channel) provide scram discharge volume (SDV) high-water-level inputs to the four RPS channels. An additional level-indicating switch (trip unit), with transmitter, in each channel is redundant to the level switch in that channel. This arrangement provides diversity to ensure that no single event could prevent a scram caused by SDV high water level. With the scram setting listed in Table 7.2-1 and in the Technical Specifications, a scram is initiated when sufficient capacity remains in the SDV to accommodate a scram. Both the amount of water discharged and the volume of air trapped above the free surface during a scram have been considered in the selection of the trip setting.

Drywell pressure is monitored by four pressure transmitters as described in Subsection 7.3.2.2.8.f. The transmitters are physically separated and electrically connected to the RPS so that no single failure can prevent a scram caused by primary containment high pressure.

Main steam line radiation is monitored by four radiation monitors, which are discussed and evaluated in Section 11.4. Each monitor provides a trip signal to one channel when high gamma radiation is detected in the vicinity of the main steam lines (Figure 7.2-5).

Main condenser low vacuum trip will be effected indirectly through main steam line isolation. A main condenser vacuum of approximately 7 PSIA will cause steam line isolation valve closure, which in turn causes reactor trip.

Four turbine first-stage pressure transmitters are provided to initiate the automatic bypass of the turbine control valve fast closure and turbine stop valve closure scrams when the first-stage pressure is below some preset fraction of rated pressure corresponding to 29.5 percent of rated power. The transmitters are arranged so that no single failure can prevent a turbine stop valve closure scram or turbine control valve fast closure scram.

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Channel and logic relays are fast-response, high-reliability relays. Power relays for interrupting the scram pilot valve solenoids are magnetic contactors. The system response time, from the opening of a sensor contact up to and including the opening of the trip actuator contacts, is less than 50 msec. The time requirements for control rod movement are discussed in Subsection 4.5.2.

Sensing elements have enclosures to withstand conditions resulting from a steam or water line break long enough to perform satisfactorily. Environmental specifications for the instruments of the RPS are given in Table 3.11-1.

To gain access to those calibration and trip setting controls located outside the main control room, operations personnel must remove a cover plate, access plug, or sealing device before any trip setting can be adjusted.

Wiring for the RPS, outside of the enclosures in the main control room, is run in rigid metallic conduits used for no other wiring. The wires from duplicate sensors on a common process tap are run in separate conduits. Wires from sensors of different variables in the same RPS logic can be run in the same conduit.

The scram pilot valve solenoids are powered from eight actuator logic circuits, four circuits from trip system A and four from trip system B. The four circuits associated with any one trip system are run in separate conduits.

Electrical panels, junction boxes, and components of the RPS are prominently identified by nameplates. Circuits entering junction boxes or pull boxes are conspicuously marked inside the boxes. Wiring and cabling outside cabinets and panels are identified by color, tag, or other conspicuous means.

7.2.1.1.3.2 Logic

The basic arrangement of the RPS actuators and actuator logic is illustrated in Figure 7.2-8. The system is arranged as two separately powered trip systems. Each trip system has two automatic trip logics, as shown in Figure 7.2-9. Each logic used for automatic trip receives input signals from at least one channel for each monitored variable. At least four channels for each monitored variable are required, one for each of its four automatic trip logics.

Each automatic trip logic provides two inputs into each of the actuator logics of one trip system, as shown in Figure 7.2-8. Thus, either of the two automatic trip logics associated with one trip system can produce a trip-system trip. The logic is a "one-out-of-two" arrangement. To produce a scram, the actuator logics of both trip systems must be tripped. The overall logic of the RPS is termed "one-out-of-two taken twice."

7.2.1.1.3.3 Scram Bypasses

A number of manual and automatic scram bypasses are provided. These account for the varying protection requirements that depend on reactor conditions. They also allow for instrument service during reactor operations. All manual bypass switches are in the main control room under the direct control of the main control room operator. The bypass status of trip system components is continuously indicated in the main control room.

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To properly reset the RPS at plant shutdown and during initial plant startup, a bypass is required for the MSIV closure scram trip. This bypass has been designed to be in effect when the mode switch is in the SHUTDOWN, REFUEL, or STARTUP position.

Hence, the bypass is necessary to provide for proper RPS reset action whenever the MSIVs are closed during very low power operation.

In the terms of the power generation design bases, the actual pressure scram setpoint is established from considerations of reducing reactor overpressure in the event of isolation at high power levels.

Since the high-pressure scram and reactor relief valves provide protection against overpressure, there would be no safety problem if the reactor were held at normal operating pressure and at a low power level with the MSIVs closed.

The scram initiated by placing the mode switch in SHUTDOWN is automatically bypassed after a short time delay. The bypass allows the CRD hydraulic system valve lineup to be restored to normal. An annunciator in the main control room indicates the bypassed condition. The turbine control valve fast closure scram and turbine stop valve closure scram are automatically bypassed if the turbine first-stage pressure is less than 29.5 percent of rated power. Closure of these valves from a low initial power level does not threaten the integrity of any radioactive material release barrier.

Turbine and generator trip bypass is effected by four pressure switches associated with the turbine first stage. Any one channel in a bypass state produces a main control room annunciation.

Bypasses for the NMS channels are described in Subsection 7.6.1.13.

The scram discharge high water level trip bypass is controlled by the manual operation of two keylocked switches, a bypass switch, and the mode switch. The mode switch must be in either the SHUTDOWN or the REFUEL position. Four bypass channels emanate from the four banks of the RPS mode switch and are each connected into the RPS logic. This bypass allows the operator to reset the RPS scram relays so that the system is restored to operation while the operator drains the scram discharge volume. In addition, actuating the bypass initiates a control rod block. Resetting the trip actuators opens the scram discharge volume vent and drain valves. An annunciator in the main control room indicates the bypass condition.

The RPS reset switch is used to momentarily bypass the seal-in contacts of the final actuators of the reactor shutdown systems. These seal-in contacts are located downstream from the protection channel outputs. The reset is effected in conjunction with auxiliary relays. If a single channel is tripped, the reset is accomplished immediately upon operation of the reset switch. On the other hand, if a reactor scram situation is present, manual reset is prohibited for a 10-sec period to permit the control rods to achieve their fully inserted position.

7.2.1.1.3.4 Interlocks

The scram discharge volume high-water-level trip bypass signal interlocks with the reactor manual control system (RMCS) to initiate a rod block. The interlock is performed using isolating relay contacts so that no failure in the control system can prevent a scram.

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The RPV low water level, primary containment high pressure, and turbine stop valve position signals are shared with the primary containment and reactor vessel isolation-control system (CRVICS). The sensors feed sensor relays in the RPS. Contacts from these relays interlock to the primary containment and reactor vessel isolation system.

7.2.1.1.3.5 Redundancy and Diversity

The RPS is divided into two divisions. Each division duplicates the function of the other to the extent that either may perform the required function regardless of the state of operation or failure of the other.

Functional diversity is provided by monitoring dependent RPV variables. Pressure, water level, and neutron flux are all interdependent and are separate inputs to the system. Also, MSIV closure, turbine stop valve closure, and turbine control valve fast closure are anticipatory of an RPV high pressure and are separate inputs to the system.

7.2.1.1.3.6 Actuated Devices

The actuator logic opens when a trip signal is received, and then deenergizes the scram valve pilot solenoids. There are two pilot solenoids per control rod. Both solenoids must deenergize to open the inlet and outlet scram valves to allow drive water to scram a control rod. One solenoid receives its signal from trip system A and the other from trip system B. The failure of one control rod to scram will not prevent a complete shutdown.

The individual control rods and their controls are not part of the RPS. Further information on the scram valves and control rods is contained in Subsection 4.5.2.

7.2.1.1.3.7 Separation

Four sensor channels monitor these various process variables listed in Subsection 7.2.1.1.3.1. Separation criteria for the sensors are given in Section 3.12. The sensor devices are separated in such a way that no single failure can prevent a scram. All protection system wiring outside the control system cabinets is run in rigid metal conduit. Six physically separated cabinet bays are provided for the four scram logics. Where two RPS channels of the same trip system enter the same bay they are separated by barriers.

The mode switch, scram discharge volume high-water-level trip bypass switch, scram reset switch, and manual scram switch are all mounted on one control panel. Each device is mounted in a can and has a sufficient number of barrier devices to maintain adequate separation. Conduit is provided from the cans to the logic cabinets.

The outputs from the logic cabinets to the scram valves are run in four conduits for trip system A and four conduits for trip system B. The four conduits match the four scram groups shown in Figure 7.2-2. The groups are selected so that the failure of one group to scram will not prevent a reactor shutdown.

7.2.1.1.3.8 Testability

The RPS can be tested during reactor operation by six separate tests. The first of these is the manual actuator test. By depressing the manual scram button for one trip channel, the

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manual actuators are deenergized, opening contacts in the actuator logics. After the first trip channel is reset, the remaining three manual trip channels are tested sequentially in a similar manner. The total test verifies the ability to deenergize all eight groups of scram pilot valve solenoids by using the manual scram pushbutton switches. In addition to main control room and sequence recorder indications, scram group indicator lights verify that the actuator contacts have opened.

The second test is the automatic actuator test. It is accomplished by operating the keylocked test switches one at a time for each automatic logic. The switch deenergizes the actuators for that logic and causes the associated actuator contacts to open. The test verifies the ability of each logic to deenergize the actuator logics associated with the parent trip system. In addition to annunciator and sequence recorder indications, the actuator and contact action can be verified by observing the physical position of these devices.

The third test includes calibration of the NMS by means of simulated inputs from calibration logic. Subsection 7.6.1.13 describes the calibration procedure.

The fourth test is the single rod scram test, which verifies capability of each rod to scram. It is accomplished by operating a toggle switch on the RPS test cabinet in the control center particular CRD. Timing traces can be made for each rod scrambled. Prior to the test, a physics review must be conducted to ensure that the rod pattern during scram testing will not create a rod of excessive reactivity worth.

The fifth test involves applying a test signal to each RPS channel in turn and observing that a logic trip results. The test signals can be applied to the process type sensing instruments (pressure and differential pressure) through calibration taps.

The sixth test involves applying a test signal to each RPS trip channel associated with the CRD Scram Discharge Volume High Water Level, Drywell High Pressure, Reactor High Pressure, Reactor Low Water Level, NMS Trip and Main Steam Line High Radiation and observing a trip relay contact closure using the RPS Test Box (RTB). The RTB lamp connected across the contacts of the trip relay during the functional test maintains circuit continuity and keeps the RPS Scram contactors energized while monitoring the status of the trip relay contacts (open/closed).

RPS response times are verified on a channel basis during preoperational testing and can be verified thereafter by similar tests with exception to the sensors. The neutron flux and radiation sensors, the primary sensor response time is included in the measurement of overall channel response time. This measured response time is added to an allowance for instrument line delay, as appropriate, for each application. This approach is consistent with the definition of response time, which is the maximum allowable time from when the variable being measured just exceeds the trip setpoint to the deenergizing of the control rod scram solenoids. The applicable test criterion is that the adjusted test-based value must not exceed the value used for the safety analysis.

During preoperational testing, and subsequently on a surveillance basis, the sensor response time was measured using a hydraulic ramp-test method similar to that described in Electric Power Research Institute Report No. NP-267, Sensor Response Time Verification. To the results of this measurement is added the delay for instrument line length as appropriate for

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each application. Also, the noise analysis method can be used for the sensor response verification.

The periodic response time testing for the reactor vessel steam dome pressure-high and the reactor vessel low water level-L3 have been eliminated. The BWROG Report NEDO-32291A and Supplement 1 provide the required analyses as briefly described in 7.2.1.1.3.8.1.

The response time of the trip comparators and trip delays is determined using the transient current source test method described in NEDO 21617-A, Analog Transmitters/Trip Unit System for Engineered Safeguard Sensor Trip Inputs. This test is performed as part of the preoperational test.

The balance of the RPS channel logic response time is tested using accepted methods that are documented in existing preoperational test procedures.

The reactor protection system instrumentation response times are shown in Technical Requirements Manual Volume I Table 3.3.1.1-1, which is referenced in UFSAR Table 7.2-4. Response time testing is required by the Technical Specifications. Technical Specification Table 3.3.1-2 was deleted from the Technical Specifications and added to the UFSAR as Table 7.2-4 (TRM Table 3.3.1.1-1) in agreement with NRC Generic Letter 93-08 and Amendment Number 100 to the Technical Specifications. The response times information of UFSAR Table 7.2-4 was then relocated to the Technical Requirements Manual Volume I.

7.2.1.1.3.8.1 Elimination of Response Time Testing

The elimination of selected response time testing requirements are supported by the analyses performed by the Boiling Water Reactor Owner's Group (BWROG) report. The BWROG report demonstrated that other periodic tests required by Technical Specifications (TS), such as channel calibrations, channel checks, channel functional tests, and logic system functional tests provide adequate assurance that instrument response times are within acceptable limits. The evaluation is documented in NEDO-32291A and Supplement 1, "System Analyses for Elimination of Selected Response Time Testing Requirements." The analyses assert that the response time tests are of little safety significance and result in unnecessary personnel radiation exposure, reduced availability of systems during plant shutdown, increased potential for inadvertent actuations of safety systems, and a significant burden to utility resources.

The basis for eliminating response time testing is consistent with Regulatory Guide 1.118 (Revision 2) which endorses IEEE 338-1977 which states:

"Response time testing of all safety equipment, per se, is not required if, in lieu of response time testing, the response time of safety system equipment is verified by functional testing, calibration checks or other tests, or both. This is acceptable if it can be demonstrated that changes in response time beyond acceptable limits are accompanied by changes in performance characteristics which are detectable during routine periodic tests."

NEDO-32291A and Supplement 1 identify the potential failure modes of components in the affected instrumentation loops which could potentially impact the instrument loop response time. In addition, industry operating experience was reviewed to identify failures that affect response times and how they were detected. The failure modes identified were then

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evaluated to determine if the effect on response time would be detected by other testing requirements contained in TS. The results of this analysis demonstrate that other TS testing requirements (channel calibrations, channel checks, channel functional tests, and logic system functional tests) are sufficient to identify failure modes or degradations in instrument response times and assure operation of the analyzed instrument loops within acceptable limits. Furthermore, there were no failure modes identified that can be detected by response time testing that cannot also be detected by other TS-required tests.

A BWROG survey has concluded that instrument response time delays of 5 seconds can be reasonably detected by instrument technicians. A safety evaluation has confirmed that a 5-second increase in the response time of individual specific functions has a very low safety significance. This realistic bases evaluation showed that significant margin exists in the licensing analysis.

Within the trip function, redundancy exists in most safety trip functions (e.g., neutron flux, water level, drywell pressure). Also for most of these instruments, the response times are insignificant compared to the system actuation times.

NEDO-32291A and Supplement 1 are applicable to Fermi 2 and the affected components are evaluated in NEDO-32291A and Supplement 1. The vendors do not require periodic response time testing for these components. Fermi is in compliance with the guidelines of Supplement 1 to NRC Bulletin 90-01.

The recommendations from EPRI NP-7243 “Investigation of Response Time Testing Requirements” are:

1. The response time testing is required after replacing or refurbishing the transmitter (e.g., sensor cell, or variable damping) prior to returning the transmitters to service.
2. The transmitters that utilize capillary tubes are not used in any application that requires response time testing.

Furthermore, the technicians are in direct communication to verify that the response of the transmitter to the step input change or fast ramp is prompt, and in all cases less than five seconds. During this excursion, the transmitter/instrument loop is monitored for sluggishness or erratic operation that would be indicative of degraded transmitter/instrument loop performance.

The sensor response time may be assumed to be the design sensor response time. This will allow Fermi 2 to use manufacturer response time data and eliminate the requirement for a separate measurement of the sensor response time. Prior to return to service of a new transmitter or following refurbishment of a transmitter (e.g., sensor cell or variable damping components), a hydraulic response time test will be performed to determine an initial sensor-specific response time value.

7.2.1.1.4 Environmental Considerations

Electrical modules for the RPS are located in the primary containment, in the reactor building, and in the turbine building. The environmental conditions for these areas are shown in Tables 3.11-1 and 3.11-4.

Cabling for the RPS will be run in conduit or in an enclosed ferromagnetic cable tray. Separation will be in accordance with Section 3.12 and Subsection 8.3.1.

7.2.1.1.5 Operational Considerations

7.2.1.1.5.1 Normal

During normal operation, all sensor and trip contacts essential to safety are closed; channels, logics, and actuators are energized. In contrast, however, trip contact bypass channels consist of normally open contact networks that close to bypass.

7.2.1.1.5.2 Scram Functions

The following paragraphs discuss the functional considerations for the variables or conditions monitored by the RPS. Table 7.2-1 lists the preliminary specifications for instruments that provide signals for the system. Figure 7.2-1 summarizes the locations from which the RPS may receive a signal that causes a scram.

There are two pilot scram valves and two scram valves for each control rod, arranged as shown in Figure 7.2-2. Each pilot scram valve is solenoid operated, with the solenoids normally energized. The pilot scram valves control the air supply to the scram valves for each control rod. When either pilot scram valve is energized, air pressure holds the scram valves closed. The scram valves control the supply and discharge paths for CRD water. As shown in Figure 7.2-2, one of the scram pilot valves for each control rod is controlled by actuator logics A, and the other valve is controlled by actuator logics B. There are two dc solenoid-operated backup scram valves that provide a second means of controlling the air supply to the scram valves for all control rods. The dc solenoid for each backup scram valve is normally deenergized. The backup scram valves are energized (initiate scram) when trip systems A and B are both tripped.

The functional arrangement of sensors and channels that constitute a single logic is shown in Figure 7.2-2. A simplified logic schematic is included in Figure 7.2-9. When a channel sensor contact opens, its sensor relay deenergizes, causing contacts in the logic to open. The opening of contacts in the logic deenergizes its actuators. When deenergized, the actuators open contacts in all of the actuator logics for that trip system. This action results in deenergizing the scram pilot valve solenoids associated with that trip system (one scram pilot valve solenoid for each control rod). However, the other scram pilot valve solenoid for each rod must also be deenergized before the rods can be scrambled.

If a trip also occurs in any of the logics of the other trip system, the remaining scram pilot valve solenoid for each rod is deenergized. This permits the air to vent from the scram valves and allows CRD drive water to act on the CRD piston. Thus, all control rods are scrambled. The water displaced by the movement of each rod piston is vented into a scram discharge volume. When the solenoid for each backup scram valve is energized, the backup scram valves vent the air supply for the scram valve. This action initiates insertion of any errant control rods regardless of the action of the scram pilot valves (Figure 7.2-2).

A scram can be initiated manually. There are two sets of manual scram pushbuttons located on the surface of the main operating panel. The first set associated with logics A1 and B1 is

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located directly above the control rod pushbutton matrix on the "A" surface of the reactor control panel as shown on Figure 7.5-1. A second set of pushbuttons associated with logics A2 and B2 is located on the "B" surface of the reactor control panel as shown in Figure 7.5-1. These pushbuttons are approximately 21 in. apart and 12 in. from the first set of the manual scram pushbuttons. Each of the four manual scram pushbuttons is individually canned and the control wiring is run in conduit within the control panel. To cause a manual scram, at least one button in each trip system must be depressed.

The manual scram pushbuttons in the first set are close enough to permit one hand motion to initiate the scram. By operating the manual scram button for one logic at a time and then resetting that logic, each actuator logic can be tested for manual scram capability. The reactor operator also can scram the reactor by interrupting power to the reactor protection system or by placing the mode switch in its shutdown position.

To restore the RPS to normal operation following any single trip system trip or scram, the actuators must be reset manually. The actuators can be reset only after a 10-sec delay, and only if the conditions that caused the scram have been cleared. The actuators are reset by operating switches in the main control room. Figure 7.2-2 shows the functional arrangement of reset contacts for trip system A.

When an RPS sensor trips, it lights a printed red annunciator window, common to all the channels for that variable, which indicates the out-of-limit variable. This window is located on the reactor control panel in the main control room. Each trip system lights a red annunciator window which indicates which trip system has tripped. An RPS channel trip also sounds a buzzer or horn that can be silenced by the operator. The annunciator window lights latch in until the initiating contact is reset. Reset is not possible until the condition causing the trip has been cleared. A sequence-of-events recorder identifies each tripped channel; however, the physical position of the RPS relays may also be used to identify the individual sensor that tripped in a group of sensors monitoring the same variable. The location of alarm windows permits the operator to quickly identify the cause of RPS trips and to evaluate the threat to the fuel or nuclear system process barrier.

All RPS trip events are recorded on a sequence-of-events recorder that includes nuclear steam supply system (NSSS) inputs. This record permits analysis of operational transient events that occur too rapidly for operator recognition.

The sequence-of-events recorder provides the time and alarm type of each event and can resolve the order of occurrence down to 1 msec. A lesser time difference causes the events to be treated as simultaneous.

Use of the events recorder is not required for plant safety. The printout of trips is particularly useful in routinely verifying the correct operation of pressure, level, and valve position switches as trip points are passed during startup, shutdown, and maintenance operations.

Reactor protection system inputs to annunciators, recorders, and the computer are arranged so that no malfunction of the annunciating, recording, or computing equipment can functionally disable the RPS. Direct signals from RPS sensors are not used as inputs to annunciating or data-logging equipment. Relay contact isolation is provided between the primary signal and the information output.

7.2.1.1.5.3 Operation Information

Indicators

Indicators are installed in the manual scram switches to indicate a trip system manual trip. Scram group indicators extinguish when an actuator logic opens. Process indicators for all RPS trip variables are available in the main control room.

Annunciators

Each RPS input is provided to the annunciator system through isolated relay contacts. Manual and automatic trip system trips also signal the annunciator system.

7.2.1.1.5.4 Setpoints

Nominal values for trip system setpoints are summarized in Table 7.2-1.

In response to the NRC letter from J. F. Stolz to W. H. Jens dated April 12, 1977, that defined specific requirements for instrument trip setpoint values, Edison has instituted a formal program with the cooperation of GE to develop the required technical data. The referenced setpoint data are presently included in the Technical Specifications.

Neutron Monitoring System Trip

To protect the fuel against high heat generation rates, neutron flux is monitored and used to initiate a reactor scram. The NMS setpoints and their bases are discussed in Subsection 7.6.1.13.

Nuclear System High Pressure

High pressure within the nuclear system threatens to rupture the nuclear system process barrier. A nuclear system pressure increase during reactor operation compresses the steam voids and results in a positive reactivity insertion. This causes increased core heat generation that could lead to fuel failure and system overpressurization. A scram counteracts a pressure increase by quickly reducing core fission heat generation. The nuclear system high-pressure scram setting is chosen slightly above the RPV maximum normal operating pressure to permit normal operation without spurious scram, yet provides a wide margin to the maximum allowable nuclear system pressure. The location of the pressure measurement, as compared to the location of highest nuclear system pressure during transients, has also been considered in the selection of the high-pressure scram setting. The nuclear system high-pressure scram setting also protects the core from exceeding thermal-hydraulic limits due to pressure increases during events that occur when the reactor is operating below rated power and flow.

Reactor Vessel Low Water Level

Low water level in the RPV indicates that the fuel is in danger of being inadequately cooled. Decreasing the water level while the reactor is operating at power decreases the reactor coolant inlet subcooling. The effect is the same as raising feedwater temperature. Should water level decrease too far, fuel damage could result. A reactor scram protects the fuel by reducing the fission heat generation within the core. The RPV low-water-level scram setting has been selected to prevent fuel damage following abnormal operational transients. These transients are caused by either single equipment malfunctions or single operator errors, and

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result in a decreasing RPV water level. The scram setting is far enough below normal operational levels to avoid spurious scrams. The setting is high enough above the top of the active fuel to ensure that enough water is available to account for evaporation loss and displacement of coolant following the most severe abnormal operational transient involving a level decrease. The selected scram setting was used in developing thermal-hydraulic limits. The limits set operational limits on the thermal power level for various coolant flow rates.

Turbine Stop Valve Closure

Closure of the turbine stop valve with the reactor at power can result in a significant addition of positive reactivity to the core as the nuclear system pressure rise causes steam voids to collapse. The turbine stop valve closure scram initiates a scram earlier than does either the NMS or nuclear system high pressure. It provides a satisfactory margin below core thermal-hydraulic limits for this category of abnormal operational transients. The scram counteracts the addition of positive reactivity resulting from increasing pressure by inserting negative reactivity with control rods. Although the nuclear system high-pressure scram, in conjunction with the pressure relief system, is adequate to preclude overpressurizing the nuclear system, the turbine stop valve closure scram provides additional margin to the nuclear system pressure limit. The turbine stop valve closure scram setting provides the earliest positive indication of valve closure.

Turbine Control Valve Fast Closure

With the reactor and turbine generator at power, fast closure of the turbine control valves can result in a significant addition of positive reactivity to the core as nuclear system pressure rises. The turbine control valve fast closure scram initiates a scram earlier than either the NMS or nuclear system high pressure. It provides a satisfactory margin to core thermal-hydraulic limits for this category of abnormal operational transients. The scram counteracts the addition of positive reactivity resulting from increasing pressure by inserting negative reactivity with control rods. Although the nuclear system high-pressure scram, in conjunction with the pressure relief system, is adequate to preclude overpressurizing the nuclear system, the turbine control valve fast closure scram provides additional margin to the nuclear system pressure limit. The turbine control valve fast closure scram setting is selected to provide timely indication of control valve fast closure.

Main Steam Line Isolation

The MSIV closure scram protects the reactor on loss of the heat sink. The MSIV closure initiates scram earlier than the NMS or nuclear system high pressure. Automatic closure of the MSIVs is initiated when conditions indicate a steam line break. The main steam line isolation scram setting is selected to give the earliest positive indication of isolation valve closure. The logic allows functional testing of main steam line trip channels with one steam line isolated.

Scram Discharge Volume High Water Level

Water displaced by the CRD pistons during a scram goes to the scram discharge volume. If the scram discharge volume fills with the water so that insufficient capacity remains for the water displaced during a scram, control rod movement would be hindered during a scram. To prevent this situation, the reactor is scrammed when the water level in the discharge

volume is filling up, yet is low enough to ensure that the remaining capacity in the volume can accommodate a scram.

Primary Containment High Pressure

High pressure inside the primary containment may indicate a break in the nuclear system process barrier. It is prudent to scram the reactor in such a situation, to minimize the possibility of fuel damage and to reduce energy transfer from the core to the coolant. The drywell high-pressure scram setting is selected to be as low as possible without inducing spurious scrams.

Main Steam Line High Radiation

High radiation in the vicinity of the main steam lines may indicate a gross fuel failure in the core. When high radiation is detected near the steam line, a scram is initiated to limit the release of fission products from the fuel. This condition also signals the primary CRVICS to initiate containment of the released fission products. The high radiation trip setting is selected high enough above background radiation levels to avoid spurious scrams, yet low enough to promptly detect a gross release of fission products from the fuel. More information on the trip setting is available in Subsection 11.4.3.8.2.3.

Manual Scram

Pushbuttons are located in the main control room to enable the operator to shut down the reactor by initiating a scram.

Mode Switch in SHUTDOWN

When the mode switch is in SHUTDOWN, the reactor is to be shut down with all control rods inserted. This scram is not considered a protective function because it is not required to protect the fuel or nuclear system process barrier, and it bears no relationship to minimizing the release of radioactive material from any barrier. The scram signal is removed after a short delay, permitting a scram reset that restores the normal valve lineup in the CRD hydraulic system.

7.2.1.1.5.5 Mode Switch

A conveniently located, multiposition, keylock mode switch is provided to select the necessary scram functions for various plant conditions. The mode switch selects the appropriate sensors for scram functions and provides appropriate bypasses. The switch also interlocks such functions as control rod blocks and refueling equipment restrictions, which are not considered here as part of the RPS. The switch is designed to provide separation between the two trip systems. The mode switch positions and their related scram functions are

- a. SHUTDOWN - Initiates a reactor scram; bypasses main steam line isolation scram
- b. REFUEL - Selects NMS scram for low neutron flux level operation; bypasses main steam line isolation scram

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- c. STARTUP - Selects NMS scram for low neutron flux level operation; bypasses main steam line isolation scram
- d. RUN - Selects NMS scram for power range operation.

7.2.1.2 Design-Basis Information

The design-basis information required by Section 3 of IEEE 279-1971 is provided in Subsection 7.1.2.1.1.

7.2.2 Analysis

7.2.2.1 General

Presented below are analyses to demonstrate how the various general functional requirements and the specific regulatory requirements listed under the RPS design bases described in Subsection 7.1.2.1.1.1 are satisfied. Considerations of loss of instrument air and loss of cooling water to vital equipment are discussed in Chapter 15.

7.2.2.2 Reactor Protection System

7.2.2.2.1 Conformance With General Functional Requirements

The RPS is designed to provide timely protection against the onset and consequences of conditions that threaten the integrity of the fuel barrier and the nuclear system process barrier. Chapter 15 identifies and evaluates events that jeopardize the fuel barrier and nuclear system process barrier. The methods of assessing barrier damage and radioactive material releases, along with the methods by which abnormal events are sought and identified, are presented in that chapter.

Design procedure has been to select tentative scram trip setting such that spurious scrams and operating inconvenience are avoided. It is then verified by analysis that the reactor fuel and nuclear system process barriers are protected. In all cases, the specific scram trip point selected is a value that prevents damage to the fuel or nuclear system process barriers, taking into consideration previous operating experience.

The scrams initiated by NMS variables, nuclear system high pressure, turbine stop valve closure, turbine control valve fast closure, and RPV low water level, prevent fuel damage following abnormal operational transients. Specifically, these scram functions initiate a scram in time to prevent the core from exceeding the thermal-hydraulic safety limit during abnormal operational transients. Chapter 15 identifies and evaluates the threats to fuel integrity posed by abnormal operational events. In no case does the core exceed the thermal-hydraulic safety limit.

The scram initiated by nuclear system high pressure, in conjunction with the pressure relief system, is sufficient to prevent damage to the nuclear system process barrier as a result of internal pressure. For turbine-generator trips, the stop valve closure scram and turbine control valve fast closure scram provide a greater margin to the nuclear system pressure safety limit than does the high pressure scram. Chapter 15 identifies and evaluates accidents

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and abnormal operational events that result in nuclear system pressure increases. In no case does pressure exceed the nuclear system safety limit.

The scrams initiated by the main steam line MSIV closure, and RPV low water level satisfactorily limit the radiological consequences of gross failure of the fuel or nuclear system process barriers. Chapter 15 evaluates gross failures of the fuel and nuclear system process barriers. In no case does the release of radioactive material to the environs result in exposures that exceed the guideline values of applicable published regulations.

Neutron flux is the only essential variable of significant spatial dependence that provides inputs to the RPS. The basis for the number and locations of neutron flux detectors is discussed in Subsection 7.6.1.13. The other requirements are fulfilled through the combination of logic arrangement, channel redundancy, wiring scheme, physical isolation, power supply redundancy, and component environmental capabilities.

The RPS uses "one-out-of-two-taken-twice" logic. Theoretically, its reliability is slightly higher than a "two-out-of-three" system and slightly lower than a "one-out-of-two" system. The differences can be neglected in a practical sense, however, because they are slight. The dual trip system is advantageous because it can be thoroughly tested during reactor operation without causing a scram. This capability for a thorough testing program significantly increases reliability.

The use of a different channel for each logic input allows the system to sustain any channel failure without preventing other sensors that monitor the same variable from initiating a scram. Any maintenance operation, calibration operation, or test results in only a single trip system trip. This leaves at least two channels per monitored variable capable of initiating a scram. The resistance to spurious scrams contributes to plant safety because reduced cycling of the reactor through its operating modes decreases the probability of error or failure.

When an essential monitored variable exceeds its scram trip point, it is sensed by at least two independent sensors in each trip system. Only one channel must trip in each trip system to initiate a scram. Thus, the arrangement of two channels per trip system ensures that a scram will occur as a monitored variable exceeds its scram setting.

Each control rod is controlled as an individual unit. A failure of the controls for one rod would not affect other rods. The backup scram valves provide a second method of venting the air pressure from the scram valves, even if either scram pilot valve solenoid for any control rod fails to deenergize when a scram is required.

Sensors, channels, and logics of the RPS are not used for control of process systems. Therefore, failure in the instrumentation and control of process systems cannot induce failure of any portion of the protection system.

Failure of either RPS motor-generator set would result, at worst, in a single trip system trip. Alternative power is available to the RPS buses. A complete, sustained loss of electrical power to both buses would result in a scram, delayed by the motor-generator set flywheel inertia.

The environment in which the instruments and equipment of the RPS must operate was considered in setting the environmental specifications given in Tables 3.11-1, 3.11-3, and 3.11-4. The specifications for the instruments located in the reactor or turbine buildings are based on the worst expected ambient conditions.

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Design of the system to comply with safety class requirements and the fail-safe characteristics of the system ensure safe shutdown of the reactor during earthquake ground motion. The system fails in a direction that causes a reactor scram only when subjected to extremes of vibration and shock.

To ensure that the RPS remains functional, the number of operable channels for the essential monitored variables is maintained at or above the minimum given in Tables 7.2-2 and 7.2-3. The minimum applies to any untripped trip system; a tripped trip system may have any number of inoperative channels. Because reactor protection requirements vary with the mode in which the reactor operates, the tables show different functional requirements for the RUN and STARTUP modes. These are the only modes in which more than one control rod can be withdrawn from the fully inserted position.

In case of a LOCA, reactor shutdown occurs immediately following the accident, as one or more process variables exceed their specified setpoint. Operation verification that shutdown has occurred may be made by observing one or more of the following indications:

- a. Control rod status lamps indicating each rod fully inserted
- b. Control rod scram pilot valve status lamps indicating open valves
- c. Neutron monitoring power range channels and recorders downscale
- d. Annunciators for RPS variables and trip logic in the tripped state
- e. Sequence-of-events recorder log of trips
- f. IPCS control rod position log.

7.2.2.2.2 Conformance To Specific Regulatory Requirements

7.2.2.2.2.1 Industry Standards

IEEE 279-1971

IEEE 279-1971 is satisfied as follows (except for manual scram, which is addressed below):

NEDO-10139, "Compliance of Protection Systems to Industry Criteria: General Electric BWR Nuclear Steam Supply System," demonstrates compliance of the RPS with IEEE 279-1968. The following paragraphs address the differences between IEEE 279-1968 and IEEE 279-1971 standards:

- a. Paragraph 4.7 - Control and Protection System Interaction. The RPS interlocks to control systems only through isolation devices such that no failure or combination of failures in the control system will have any effect on the RPS
- b. Paragraph 4.22 - Identification of Protection System. Each system cabinet is marked with the words "Reactor Protection System" and the particular redundant portion is listed on a distinctively colored marker plate. Cabling outside the cabinets is identified by color coding (as discussed in Subsection 8.3.1).

Exact design comparisons with the testability requirements of IEEE 279-1971 4.9, 4.10, and 4.11 are given in NEDO-10139:

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a.	Scram discharge volume	Pages 2-26, 2-27
b.	Main steam line isolation valve	Page 2-39
c.	Turbine stop valve	Pages 2-56, 2-57
d.	Turbine control valve	Pages 2-69, 2-70
e.	Reactor water level	Page 2-99
f.	Main steam line radiation	Pages 2-112, 2-113
g.	Neutron monitoring system	Page 2-125 (Plus NEDC-32410P-A, Pages 4-5, 4-6)
h.	Drywell pressure	Page 2-138
i.	Reactor pressure	Page 2-146
j.	Mode switch	Pages 2-164, 2-165
k.	Discharge volume bypass	Pages 2-170, 2-171
l.	Main steam line valve bypass	Page 2-178
m.	Turbine trip bypass	Page 2-186

The RPS manual scram function satisfies IEEE 279-1971 as follows:

a. Paragraph 4.2 - Single Failure Criterion

RPS manual controls comply with the single failure criterion. Four manual scram pushbuttons are arranged into two groups on one main control room Bench Board and the switches are provided with physical and electrical separation.

b. Paragraph 4.3 - Quality of Components and Modules

The RPS manual switches are selected to be of high quality and reliability.

c. Paragraph 4.4 - Equipment Qualification

Manual switches and trip logic components are certified by the vendor that they perform in accordance with the requirements listed on the purchase specification as well as in the intended application. This certification, in

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conjunction with the existing field experience with these components in this application, serves to qualify these components.

- d. Paragraph 4.5 - Channel Integrity
The manual switches and components are specified to operate under normal and abnormal conditions of environment, energy supply, malfunctions, and accidents.
- e. Paragraph 4.6 - Channel Independence
The manual scram pushbutton is a channel component. The trip channels are physically separated and electrically isolated to comply with this design requirement.
- f. Paragraph 4.7 - Control and Protection System Interaction
The manual scram pushbutton has no control interaction.
- g. Paragraph 4.8 - Derivation of System Inputs
Not applicable.
- h. Paragraph 4.9 - Capability for Sensor Checks
Not applicable.
- i. Paragraph 4.10 - Capability for Test and Calibration
A manual scram switch permits each individual trip logic, trip actuator, and trip actuator logic to be tested on a periodic basis.
- j. Paragraph 4.11 - Channel Bypass or Removal from Operation
Since actuation of one manual scram pushbutton places its RPS trip system in a tripped condition, it is in compliance with this design requirement.
- k. Paragraph 4.12 - Operating Bypasses
Not applicable.
- l. Paragraph 4.13 - Indication of Bypasses
Not applicable.
- m. Paragraph 4.14 - Access to Means for Bypassing
Not applicable.
- n. Paragraph 4.15 - Multiple Set Points
Not applicable.
- o. Paragraph 4.16 - Completion of Protective Action Once It Is Initiated
Once the manual scram push buttons are depressed, it is only necessary to maintain them in that condition until the scram contactors have de-energized and open their seal-in contacts. At this point, the trip actuator logic proceeds to initiate reactor scram regardless of the state of the manual scram push buttons.
- p. Paragraph 4.17 - Manual Actuation

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Four manual scram pushbutton controls are provided on one main control room Bench Board to permit manual initiation of reactor scram at the system level. The four manual scram pushbuttons (one in each of the four RPS trip logics) comply with this design requirement. The logic for the manual scram is one-out-of-two twice. No single failure in the manual or automatic portions of the RPS can prevent either a manual or automatic scram.

- q. Paragraph 4.18 - Access to Set Point Adjustments, Calibration, and Test Points
Not applicable.
- r. Paragraph 4.19 - Identification of Protective Actions
When any manual scram pushbutton is depressed, a control room annunciation is initiated and an IPCS alarm record is produced to identify the tripped RPS trip logic.
- s. Paragraph 4.20 - Information Readout
The manual scram function complies with this requirement.
- t. Paragraph 4.21 - System Repair
The manual scram function complies with this requirement.

The RPS is fail-safe and its power supplies are thus unnecessary for scram. A total loss of power causes a scram. A loss of one power source causes a trip system trip. IEEE 308-1971 does not apply to the RPS.

IEEE 323-1971

"General Guide for Qualifying Class I Electric Equipment" is satisfied by complete qualification testing and certification of all essential components. Records covering all essential components are maintained. For a complete summary of how the RPS complies with IEEE 323-1971, refer to NEDO-10698. See also Section 3.11.

IEEE 336-1971

"Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During Construction of Nuclear Power Generating Stations" is satisfied except as modified by the Edison Quality Assurance Procedures.

IEEE 338-1971

"Periodic Testing of Protection Systems" is complied with by being able to test the RPS from sensors to final actuators at any time during plant operation. The test must be performed in overlapping portions.

IEEE 344-1971

Conformance to IEEE 344-1971 is described in Section 3.10.

IEEE 379-1972

"Trial-Use Guide for the Application of the Single-Failure Criterion to Nuclear Power Generating Station Protection Systems" is judged to be satisfied by the RPS design criteria described in NEDO-10139.

7.2.2.2.2.2 Conformance To Regulatory Guides and 10 CFR 50

The RPS is designed so that it may be tested during plant operation from sensor device to final actuator device in compliance with Regulatory Guide 1.22. The test must be performed in overlapping portions so that an actual reactor scram does not occur as a result of the testing.

The RPS is judged to comply with Regulatory Guide 1.53 since all of the additional provisions of Regulatory Guide 1.53 as applied to IEEE 379 are met or exceeded by the actual design.

10 CFR 50, Appendix B

"Quality Assurance Criteria for Nuclear Power Plants." A Quality Assurance program has been established that includes quality control at the component vendor, at the nuclear steam supplier, at various stages of construction, and during installation at the nuclear power plant site. System design is continually checked for conformance to the applicable industry criteria. Periodic testing ensures that the system is available and adequate to perform its intended purpose. Quality assurance records are maintained by the nuclear steam supplier and Edison. For a complete description of the Quality Assurance Program, see Chapter 17.

General Design Criteria of 10 CFR 50, Appendix A

- a. Criterion 13 - Each RPS input is monitored and annunciated
- b. Criterion 19 - Instrumentation and control is provided in the main control room. The reactor can also be shut down from outside the main control room by opening breakers
- c. Criterion 20 - The RPS constantly monitors the appropriate plant variables to maintain the fuel barrier and primary coolant pressure boundary. It automatically initiates a scram when the variables exceed the established setpoints
- d. Criterion 21 - The RPS is designed with four independent and separated output channels. No single failure or operator action can prevent a scram. The system can be tested during plant operation to ensure its availability
- e. Criterion 22 - The redundant portions of the RPS are separated such that no single failure or credible natural disaster can prevent a scram. Functional diversity is used by measuring flux, pressure, and level (all dependent variables) in the reactor vessel
- f. Criterion 23 - The RPS is fail-safe. A loss of electrical power or air supply will not prevent a scram. Postulated adverse environments will not prevent a scram
- g. Criterion 24 - The RPS has no control function
- h. Criterion 29 - The RPS is highly reliable so that it is able to scram in the event of anticipated operational occurrences.

7.2.2.2.2.3 Instrument Ranges and Setpoints

The design criteria used in selecting instrument span and trip setpoints for safety-related applications consider the following factors:

- a. The selection of instrument range is based on knowledge of the expected variation of the process variable being monitored. In all cases, the range selected is greater than the expected variable excursions
- b. The accuracy of each trip setpoint is better than or equal to the accuracy assumed in the accident analysis performed for the Fermi 2 plant design
- c. Trip setpoints are normally located in the portion of the instrument range of greatest accuracy. In all cases, the setpoint is located in the portion of the instrument's range that is consistent with the required accuracy
- d. All of the safety-related trip setpoints are chosen to allow for the normal expected instrument setpoint drift without exceeding associated Technical Specifications
- e. All setpoints are verified on a prescribed schedule as outlined in the Technical Specifications.

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TABLE 7.2-1 REACTOR PROTECTION SYSTEM INSTRUMENTATION SPECIFICATIONS

<u>Scram Function</u>	<u>Instrument</u>	<u>Trip Setting^a</u>
Nuclear system high pressure	Pressure transmitter	1080 psig
Primary containment high pressure	Pressure transmitter	1.68 psig
RPV low water level	Level transmitter	173.4 in. above top of active fuel
Scram discharge volume high water level	Level switch/ transmitter	50 gal
Turbine stop valve closure	Position switch	Before 10 percent valve closure
Turbine control valve fast closure	Valve fast closure initiation logic	Start of control valve fast closure
Main steam line isolation valve closure	Position switch	Before 10 percent valve closure
Neutron monitoring system scram	Neutron detector	(IRM) 120/125 divisions of FS APRM) Refer to UFSAR Table 7.6-9 for APRM system trip setpoints.
Main steam line high radiation	Gamma ion chamber	3.0 x full power background

^a Nominal values given for information. See Technical Specifications for actual operational settings.

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TABLE 7.2-2 CHANNELS REQUIRED FOR FUNCTIONAL PERFORMANCE OF REACTOR PROTECTION SYSTEM: STARTUP MODE

This table shows the normal and minimum number of channels required for the functional performance of the RPS in the STARTUP mode. The "Normal" column lists the normal number of channels per trip system. The "Minimum" column lists the minimum number of channels per untripped trip system required to maintain functional performance.

<u>Channel Description</u>	<u>Normal</u>	<u>Minimum</u> ^{a,b}
Neutron monitoring system (APRM) ^c	2	2
Neutron monitoring system (IRM)	2	2
Nuclear system high pressure	2	2
Primary containment high pressure	2	2
RPV low water level	2	2
Scram discharge volume high water level	2	2
Manual scram	2	2
Each main steam line isolation valve position	0 (bypassed)	0

^a During testing of sensors, the channel should be tripped when the initial state of the sensor is not essential to the test.

^b Nominal values given for information. See Technical Specifications for operational requirements.

^c Number of channels refers to final two-out-of-four voter channels for APRM. See Technical Specifications for more specific requirements related to APRM channels.

TABLE 7.2-3 CHANNELS REQUIRED FOR FUNCTIONAL PERFORMANCE OF REACTOR PROTECTION SYSTEM: RUN MODE

This table shows the normal and minimum number of channels required for the functional performance of the RPS in the RUN Mode. The "Normal" column lists the normal number of channels per trip system. The "Minimum" column lists the minimum number of channels per untripped trip system required to maintain functional performance.

<u>Channel Description</u>	<u>Normal</u>	<u>Minimum^{a,b}</u>
Neutron monitoring system (APRM) ^c	2	2
Nuclear system high pressure	2	2
Primary containment high pressure	2	2
RPV low water level	2	2
Scram discharge volume high water level	2	2
Manual scram	2	2
Each main steam line isolation valve position	4	4
Each turbine stop valve position	4	4
Turbine control valve fast closure	2	2
Turbine first-stage pressure (bypass channel)	2	2

^a During testing of sensors, a channel may be placed in an inoperable status for up to 6 hr for required surveillance without placing the trip system in the tripped condition, provided that at least one operable channel in the same trip system is monitoring that parameter.

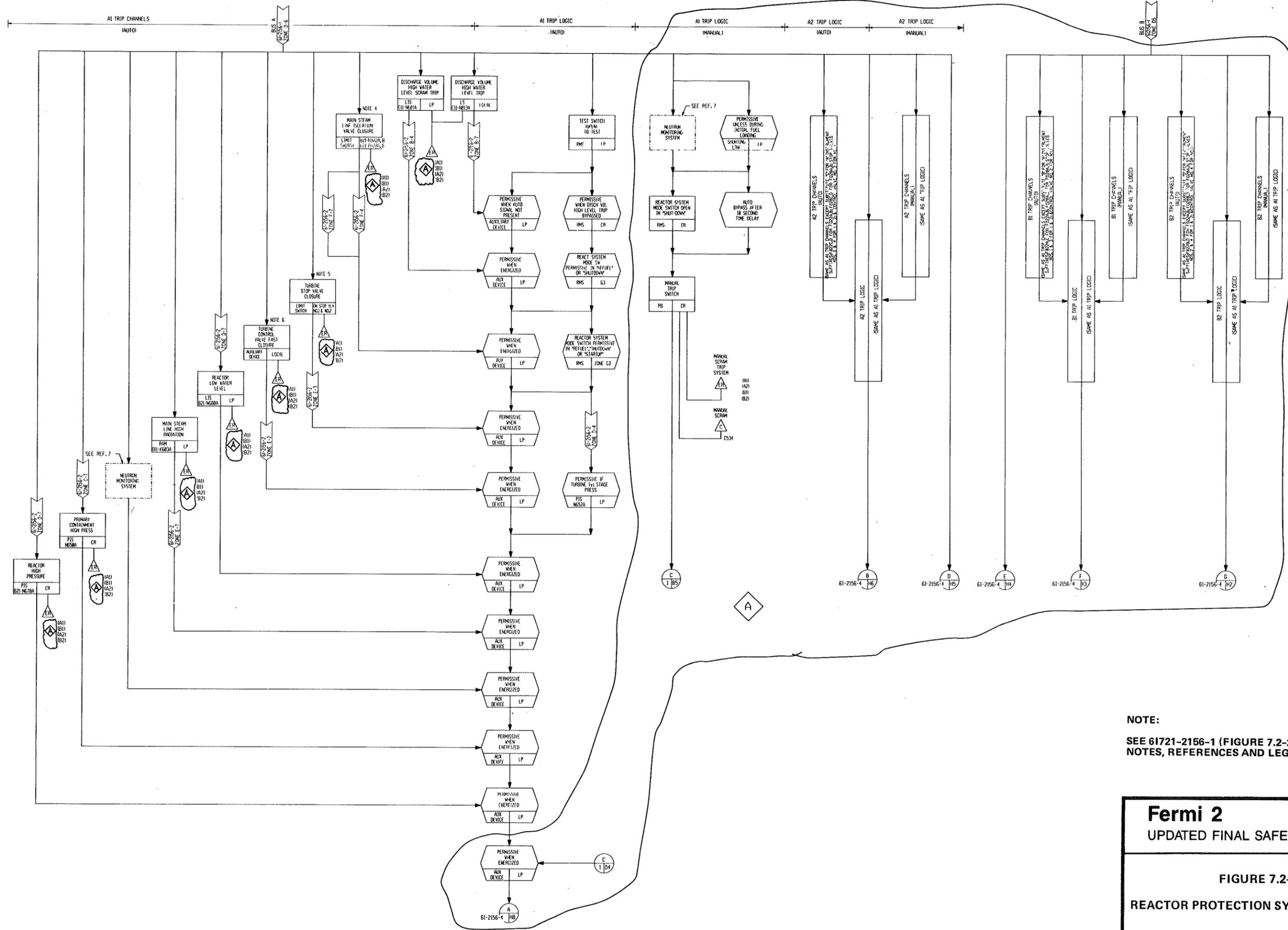
^b Nominal values given for information. See Technical Specifications for operational requirements.

^c Number of channels refers to final two-out-of-four voter channels for APRM including the OPRM function. OPRM/APRM functions are independently voted in the two-out-of-four voters. See Technical Specifications for more specific requirements related to APRM channels.

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TABLE 7.2-4 (TRM TABLE 3.3.1.1-1) REACTOR PROTECTION SYSTEM RESPONSE
TIMES

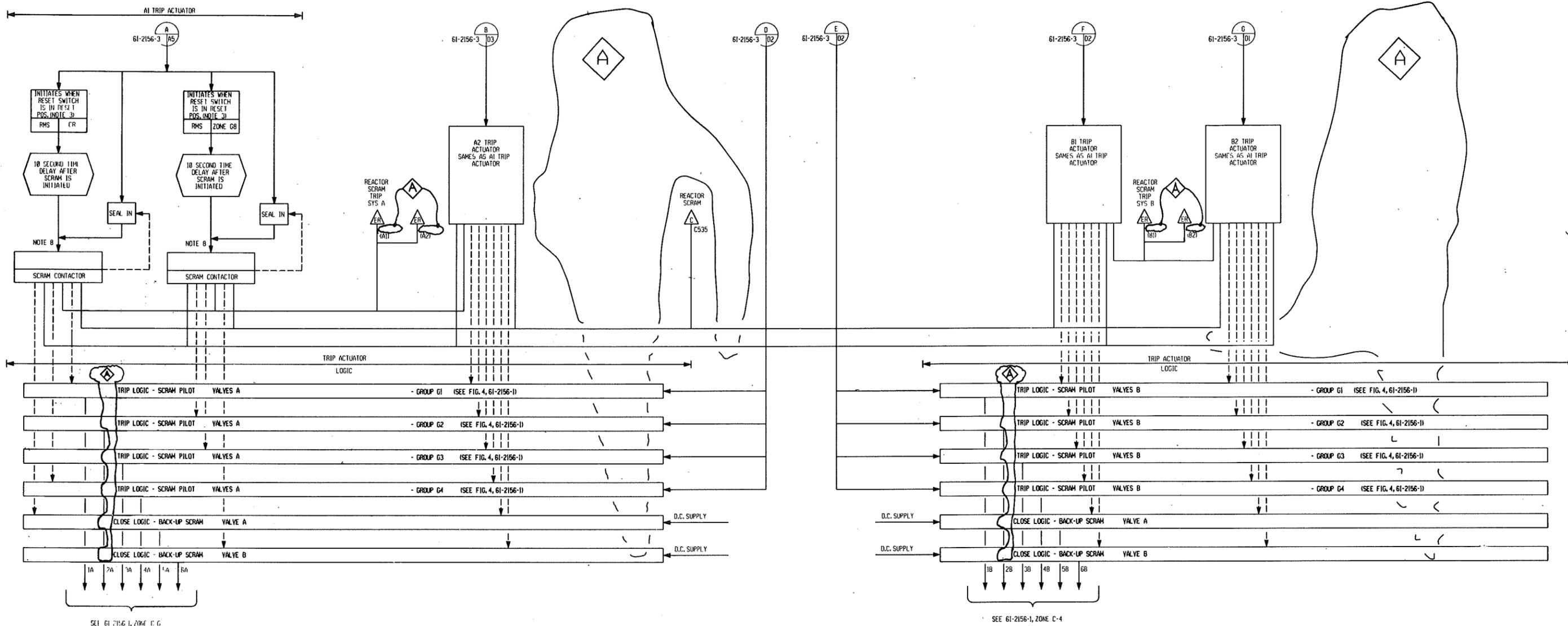
The Reactor Protection System Response Times are listed in Technical Requirements Manual (TRM) Volume I Table 3.3.1.1-1. TRM Volume I is incorporated by reference into the UFSAR.



NOTE:
 SEE 61721-2156-1 (FIGURE 7.2-2, SHEET 1) FOR
 NOTES, REFERENCES AND LEGEND.

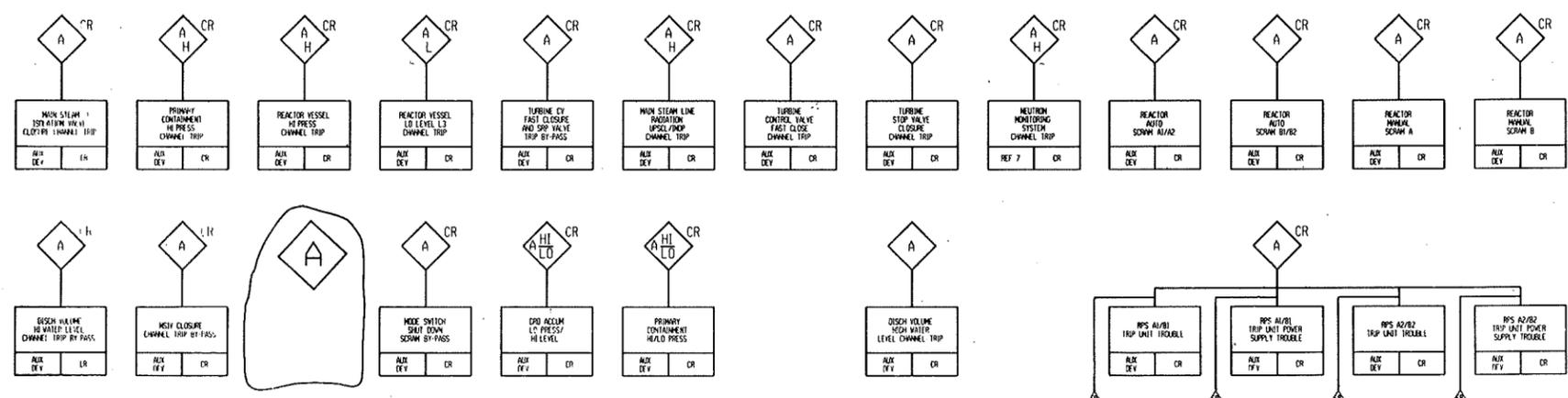
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FIGURE 7.2-1, SHEET 1
 REACTOR PROTECTION SYSTEM SCRAM FUNCTIONS



NOTE:
SEE 61721-2156-1 (FIGURE 7.2-2, SHEET 1) FOR NOTES, REFERENCES AND LEGEND.

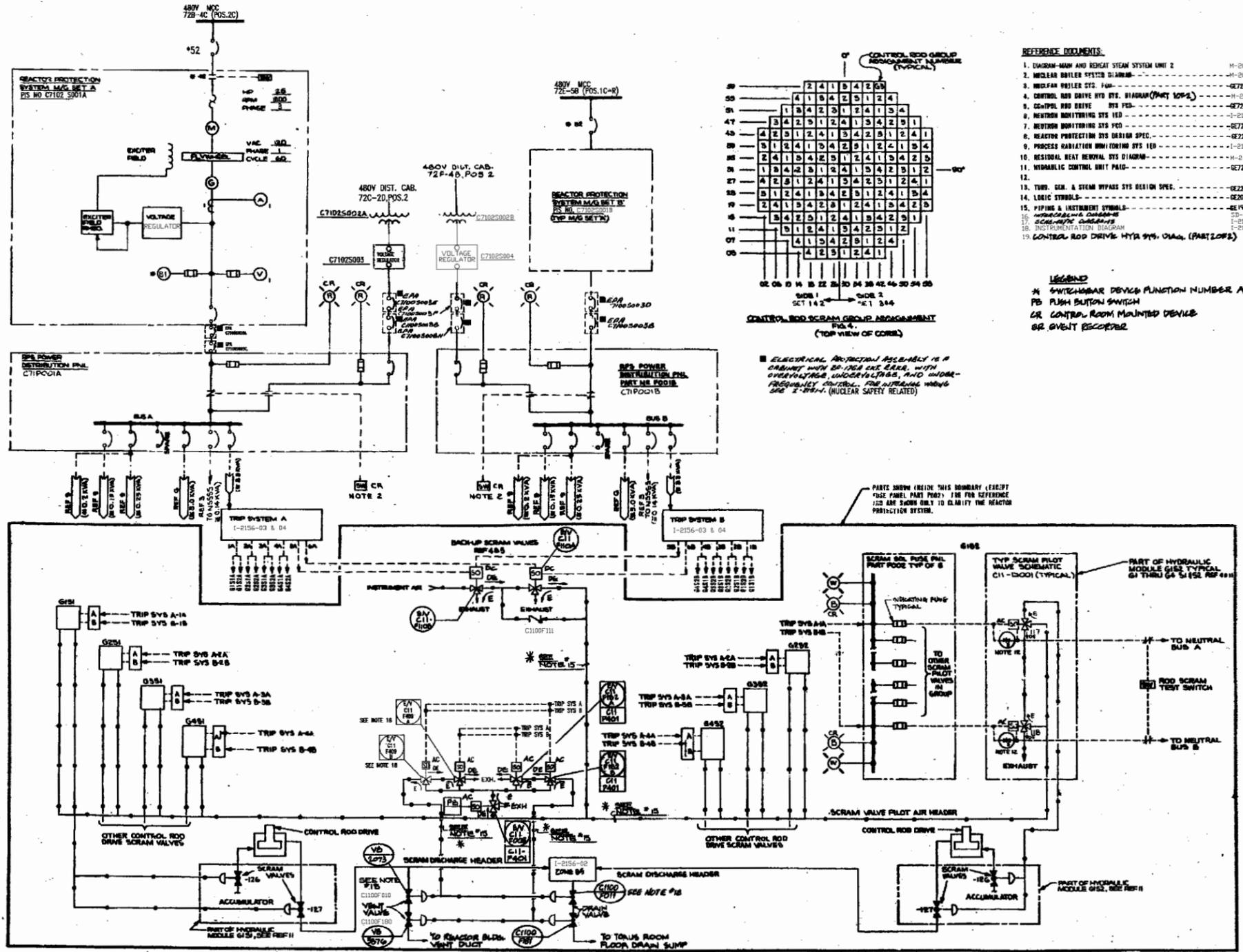
THIS DRAWING SUPERCEDES EDISON'S SITE MASTER (VENDOR DRAWING 732E181 SHT 2 REV 13) OF EDISON FILE NO. R1-109



CONTROL ROOM ANNUNCIATORS

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FIGURE 7.2-1, SHEET 2
REACTOR PROTECTION SYSTEM SCRAM FUNCTIONS



REFERENCE DOCUMENTS:

1. DIAGRAM-MAIN AND REHEAT STEAM SYSTEM UNIT 2	M-2002
2. NUCLEAR BOILER SYSTEM DIAGRAM	M-2009, 2090
3. REACTOR PROTECTION SYSTEM (PART 1001)	(R1-152)
4. CONTROL ROD DRIVE SYS DIAGRAM (PART 1001)	(R1-152)
5. CONTROL ROD DRIVE SYS DIAGRAM (PART 1001)	(R1-152)
6. NEUTRON MONITORING SYS 1E0	(R1-150)
7. NEUTRON MONITORING SYS 1E0	(R1-150)
8. REACTOR PROTECTION SYS DESIGN SPEC.	(R1-105)
9. PROCESS ORBITATION MONITORING SYS 1E0	(R1-105)
10. RESONANT BEAT REMOVAL SYS DIAGRAM	(R1-105)
11. HYDRAULIC CONTROL UNIT PAIG	(R1-105)
12.	
13. TURB. GEN. & STEAM BYPASS SYS DESIGN SPEC.	(R1-155)
14. LOGIC SYMBOLS	(R1-20)
15. PIPING & INSTRUMENT SYMBOLS	(R1-20)
16. INSTRUMENTATION DIAGRAM	(R1-20)
17. INSTRUMENTATION DIAGRAM	(R1-20)
18. INSTRUMENTATION DIAGRAM	(R1-20)
19. CONTROL ROD DRIVE HYDRAULIC SYS. DIAG. (PART 1001)	(R1-152)

LEGEND

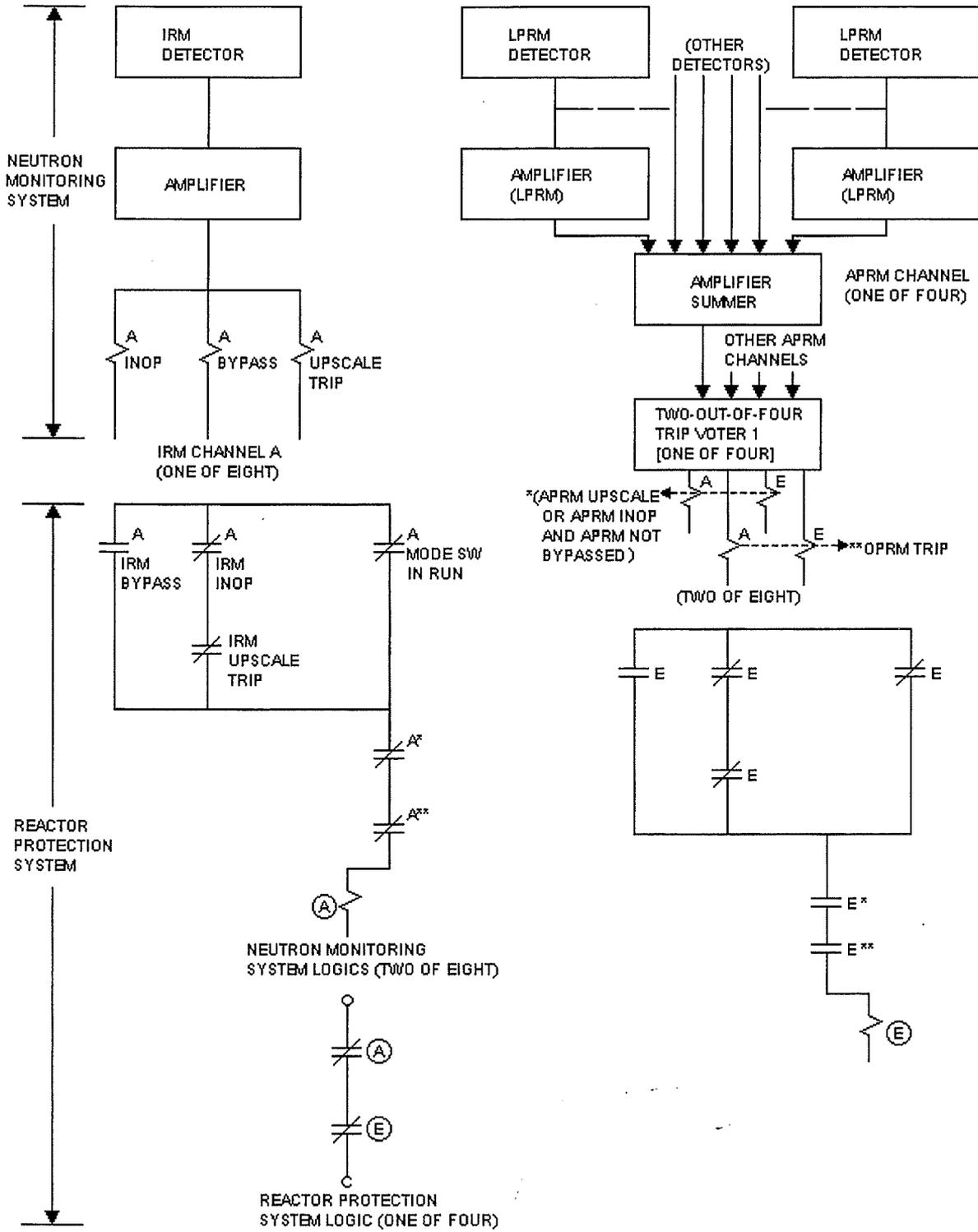
* SWITCHABLE DEVICE FUNCTION NUMBER AND SPEC. C 312
 PB PUSH BUTTON SWITCH
 CR CONTROL ROOM MOUNTED DEVICE
 SR EVENT RECORDER

- NOTES:**
- DEVICES USED IN TRIP SYSTEM "A" ARE IDENTIFIED BY LETTERS A, C, E, G, ETC. THOSE USED IN TRIP SYSTEM "B" ARE B, D, F, H, ETC.
 - SYSTEM SHALL BE ARRANGED SO THAT THE BUS CANNOT BE ENERGIZED FROM THE M/G SET AND ALTERNATE SOURCE SIMULTANEOUSLY.
 - ONE RESET SWITCH SHALL BE USED FOR BOTH TRIP SYSTEMS A AND B AND SO ARRANGED THAT ONE ACTUATION OF THE SWITCH WILL RESET ALL SCRAM PILOT VALVES (A AND B) IN GROUP 1 AND GROUP 4 AS WELL AS BACKUP SCRAM VALVE A. A DIFFERENT ACTUATION OF THE SWITCH WILL RESET ALL SCRAM PILOT VALVES (A AND B) IN GROUP 2 AND GROUP 3 AS WELL AS BACKUP SCRAM VALVE B. ACTUATION OF THE SWITCH TO BOTH GROUP 1-4 AND GROUP 2-3 POSITIONS SHALL BE REQUIRED BEFORE THE CRD DRAIN AND VENT VALVES RESET. THE RESET SWITCH SHALL BE SO CONSTRUCTED THAT RPS CHANNELS A2 AND B2 ARE PHYSICALLY SEPARATED FROM CHANNELS A1 AND B1.
 - MAIN STEAM LINE ISOLATION VALVE CLOSURE SHALL BE ARRANGED SO THAT ANY ONE STEAMLINE MAY BE ISOLATED BY FULL CLOSURE OF ITS ISOLATION VALVES AND THE ISOLATION VALVE FOR ANY ONE OTHER STEAM LINE CAN BE CLOSED MORE THAN 1000 WITHOUT CAUSING A RPS SCRAM.
 - LOGIC FOR THE "TURBINE STOP VALVE CLOSURE" TRIP SHALL BE ARRANGED SO THAT CLOSURE OF 3 OUT OF 4 STOP VALVES WILL CAUSE A RPS SCRAM. PROVISION SHALL BE MADE TO ALLOW CLOSURE OF ONE STOP VALVE (FOR TEST PURPOSES) WITHOUT CAUSING A TRIP OF EITHER TRIP SYSTEM A OR B.
 - TRIP CHANNELS FOR THE "TURBINE CONTROL VALVE FAST CLOSURE" TRIP SHALL BE DERIVED FROM THOSE EVENTS CAUSING FAST CLOSURE OF THE CONTROL VALVES.
 - NOTE DELETED
 - THIS SIGNAL, WHEN ABSENT, DE-ENERGIZES THE SCRAM CONTACTOR.
 - EACH MAIN STEAM LINE RADIATION MONITOR MONITORS ALL FOUR MAIN STEAM LINES.
 - UNLESS OTHERWISE SHOWN, ALL INSTRUMENT PIS NUMBERS ARE PREFIXED C71. ALL VALVE AND EQUIPMENT PIS NUMBERS ARE PREFIXED C7100.
 - FOR LOCATION AND IDENTIFICATION OF INSTRUMENTS SEE CECD.
 - SELENIUM THYRECTOR SUPPRESSORS (OR EQUIVALENT) SHALL BE USED TO SUPPRESS ELECTRICAL ARCING OF SCRAM SOLENOIDS.
 - THE DISCHARGE VOLUME HIGH LEVEL BYPASS SWITCH SHALL BE SO CONSTRUCTED THAT RPS CHANNELS A1 AND B1 ARE PHYSICALLY SEPARATED FROM CHANNELS A2 AND B2.
 - SHEET 2 OF THE GENERAL ELECTRIC DWG. 732E181 WAS REPRODUCED AS 1-2156-03 & -04
 - THIS DRAWING DOES NOT SHOW ALL DEVICES IN THE SCRAM VALVE PILOT AIR HEADER. SEE REF. 4 AND 19. THIS DRAWING SHOWS DEVICES CONTROLLED BY THE REACTOR PROTECTION SYSTEM.
 - FOR PROCESS COMPUTER INPUT POINTS SEE DWG 1-2155-14.
 - FOR INDICATION OF SCRAM DISCHARGE VENT & DRAIN VALVES SEE NOTE 33 OF REFERENCE 4.
 - ADJUSTABLE TIME DELAY (0-30 SEC) FUNCTIONS TO ENERGIZE VALVES C11F409A & C11F409B FOR OPENING VALVES C1100F00 & C1100F01.

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FIGURE 7.2-2 SHEET 1
 REACTOR PROTECTION SYSTEM
 INSTRUMENT DIAGRAM

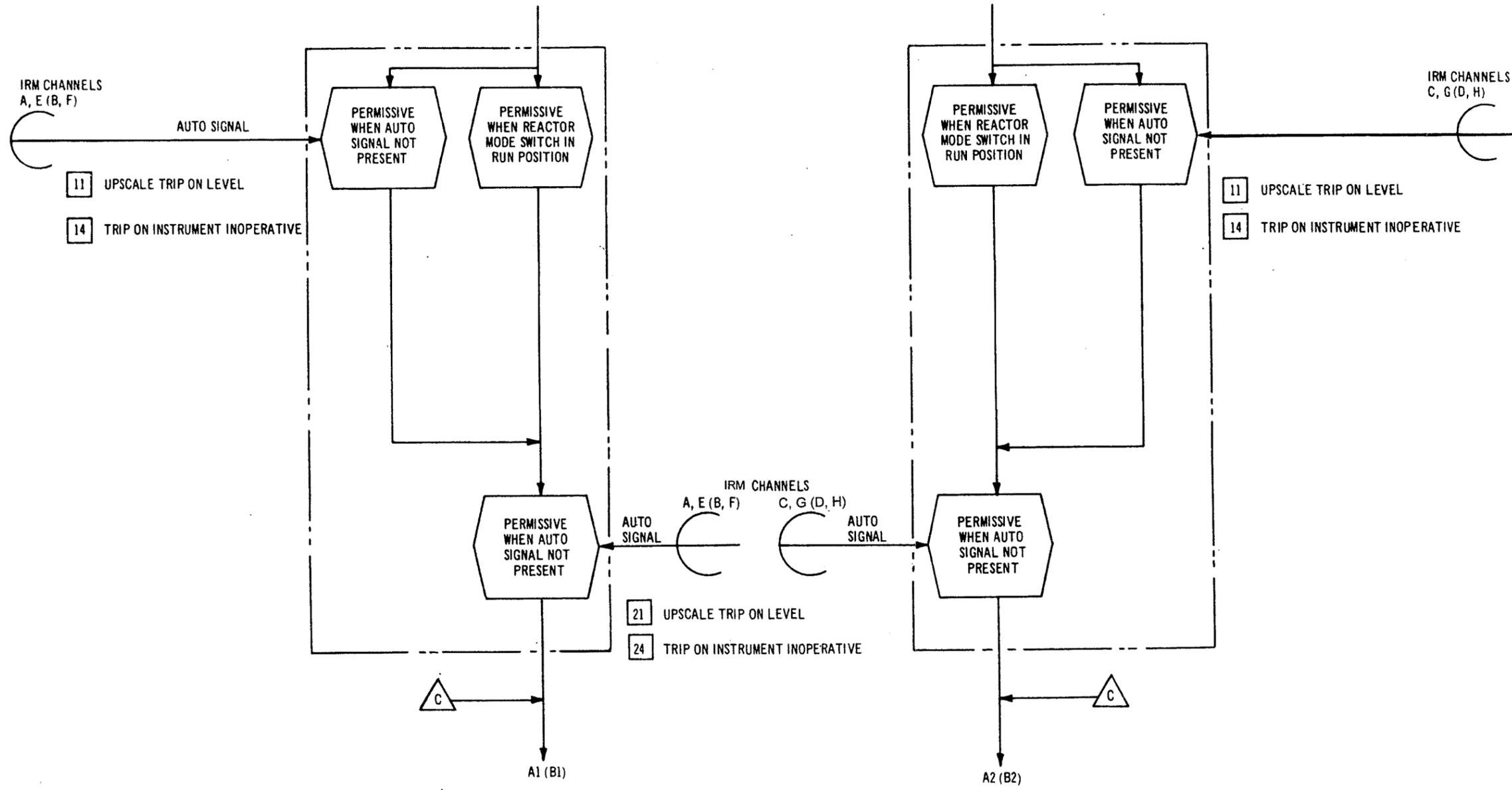
NEUTRON MONITORING SYSTEM TRIP CHANNELS



* (APRM UPSCALE OR APRM INOP) AND (APRM NOT BYPASSED)
 ** (OPRM TRIP)
 FROM 2-OUT-OF-4 TRIP VOTERS

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FIGURE 7.2-3
 RELATIONSHIP BETWEEN NEUTRON MONITORING SYSTEM AND REACTOR TRIP SYSTEM

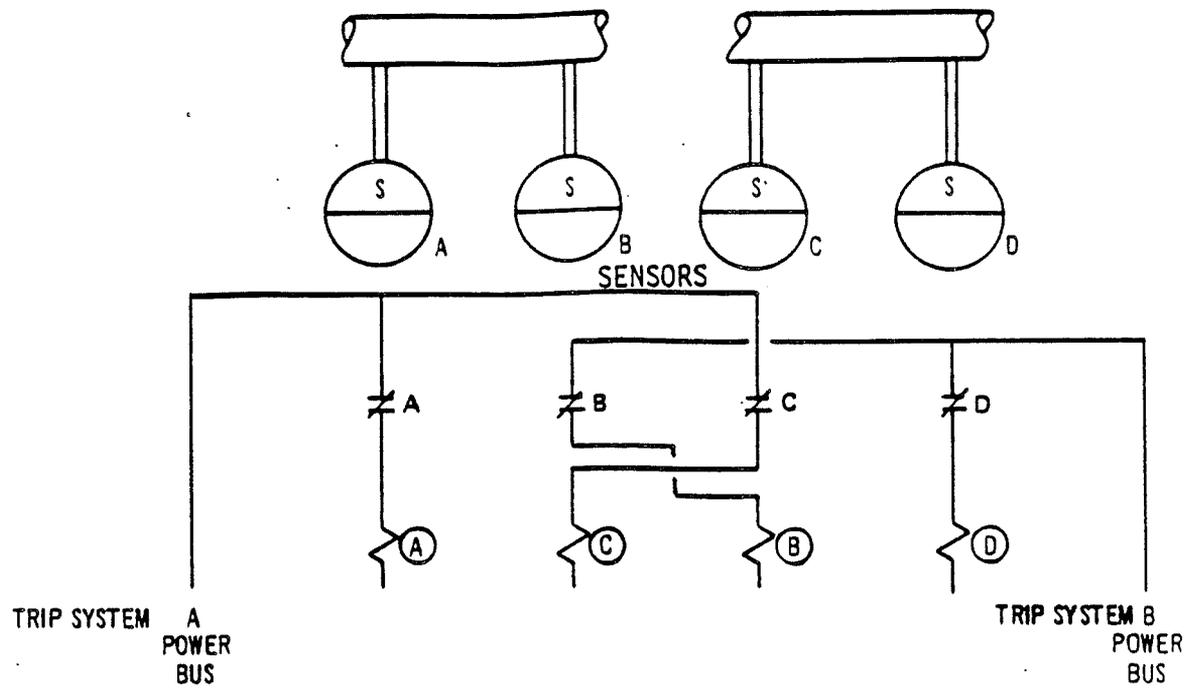


ONE OF TWO NEUTRON MONITORING SYSTEM LOGICS ASSOCIATED WITH REACTOR PROTECTION SYSTEM LOGIC A1

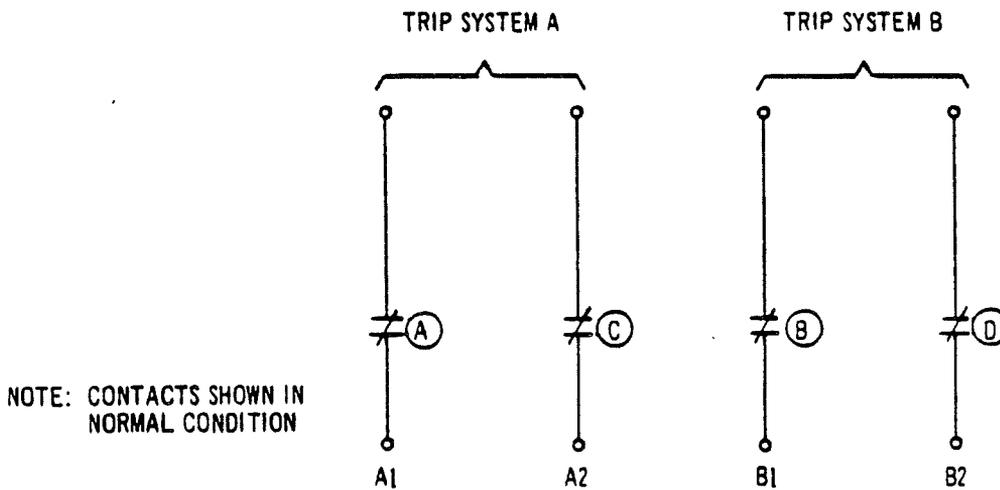
ONE OF TWO NEUTRON MONITORING SYSTEM LOGICS ASSOCIATED WITH REACTOR PROTECTION SYSTEM LOGIC A2

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FIGURE 7.2-4
 NEUTRON MONITORING SYSTEM LOGICS



CHANNELS



REACTOR PROTECTION SYSTEM LOGICS

TYPICAL LOGIC & CHANNEL CONFIGURATION FOR:
 SCRAM DISCHARGE VOLUME HIGH WATER LEVEL
 TURBINE CONTROL VALVE FAST CLOSURE
 REACTOR VESSEL LOW WATER LEVEL

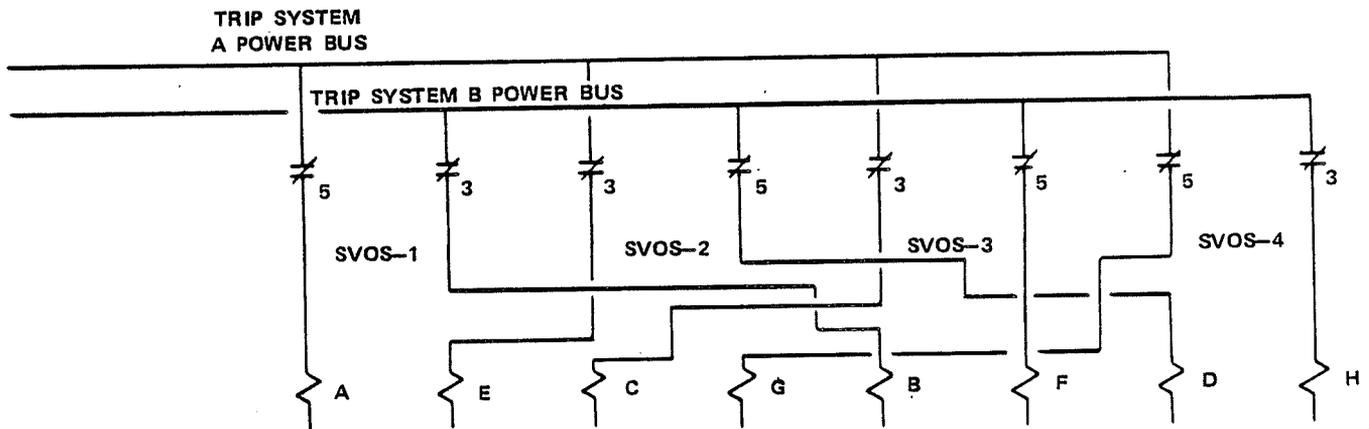
MAIN STEAM LINE HIGH RADIATION
 PRIMARY CONTAINMENT HIGH PRESSURE
 NUCLEAR SYSTEM HIGH PRESSURE

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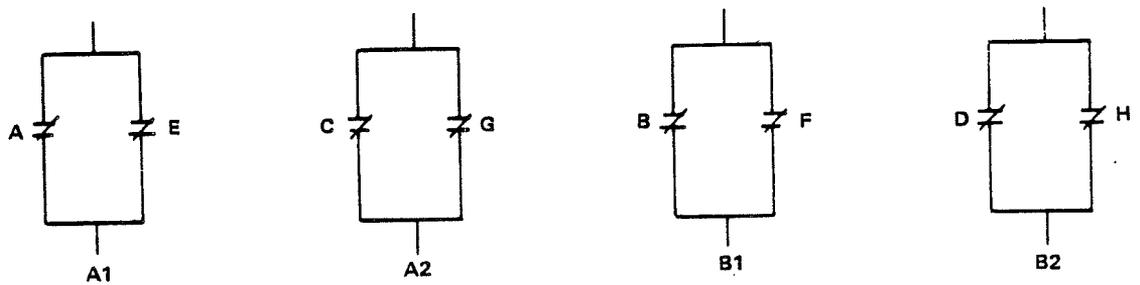
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FIGURE 7.2-5

TYPICAL ARRANGEMENT OF CHANNELS AND LOGICS



TURBINE STOP VALVE CLOSURE CHANNELS



REACTOR PROTECTION SYSTEM LOGICS

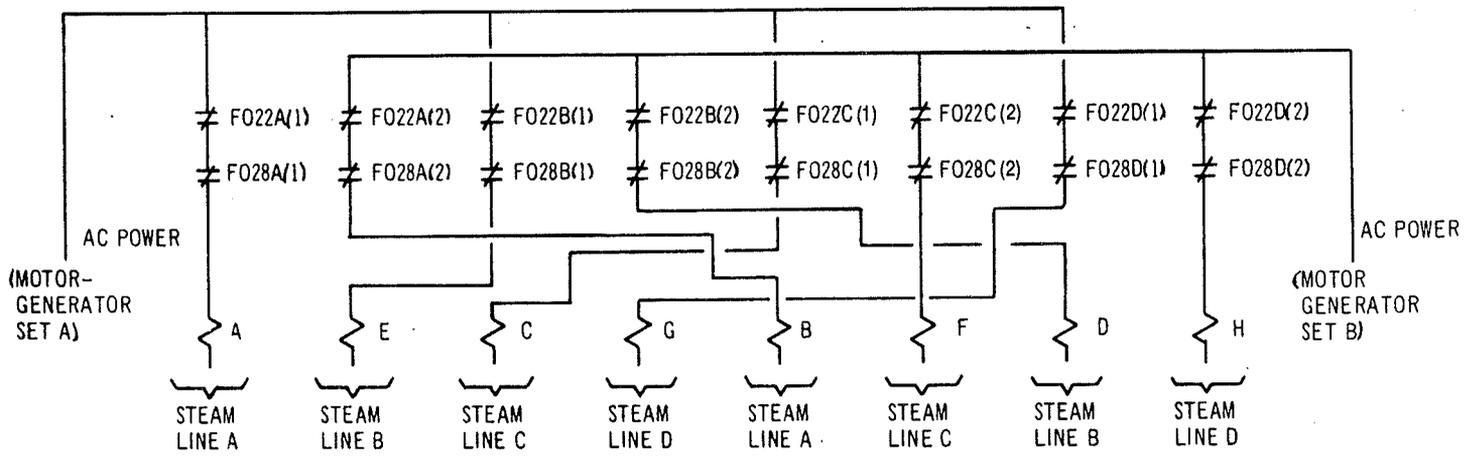
NOTE: CONTACTS SHOWN IN NORMAL CONDITION

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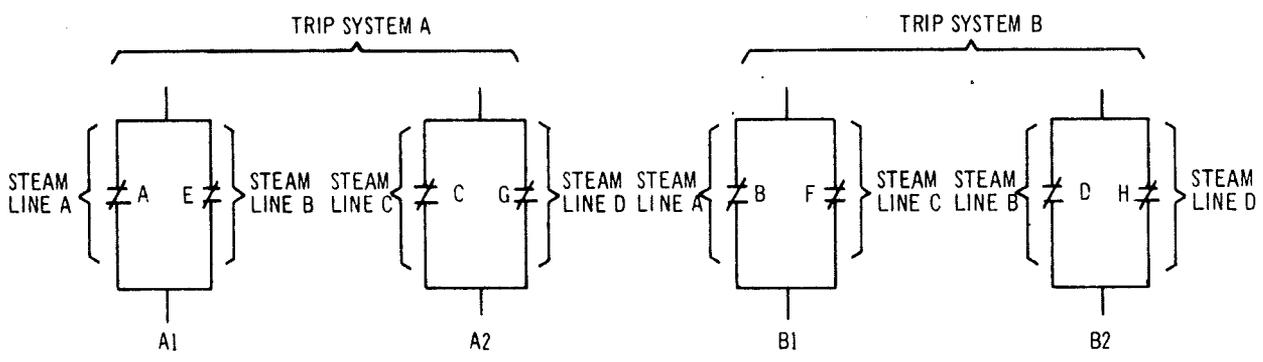
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FIGURE 7.2-6

TYPICAL CONFIGURATION FOR TURBINE STOP
CLOSURE REACTOR TRIP



MAIN STEAM LINE ISOLATION CHANNELS
 (SWITCH CONTACTS SHOWN IN POSITIONS WHEN ISOLATION VALVES LESS THAN 10% CLOSED)



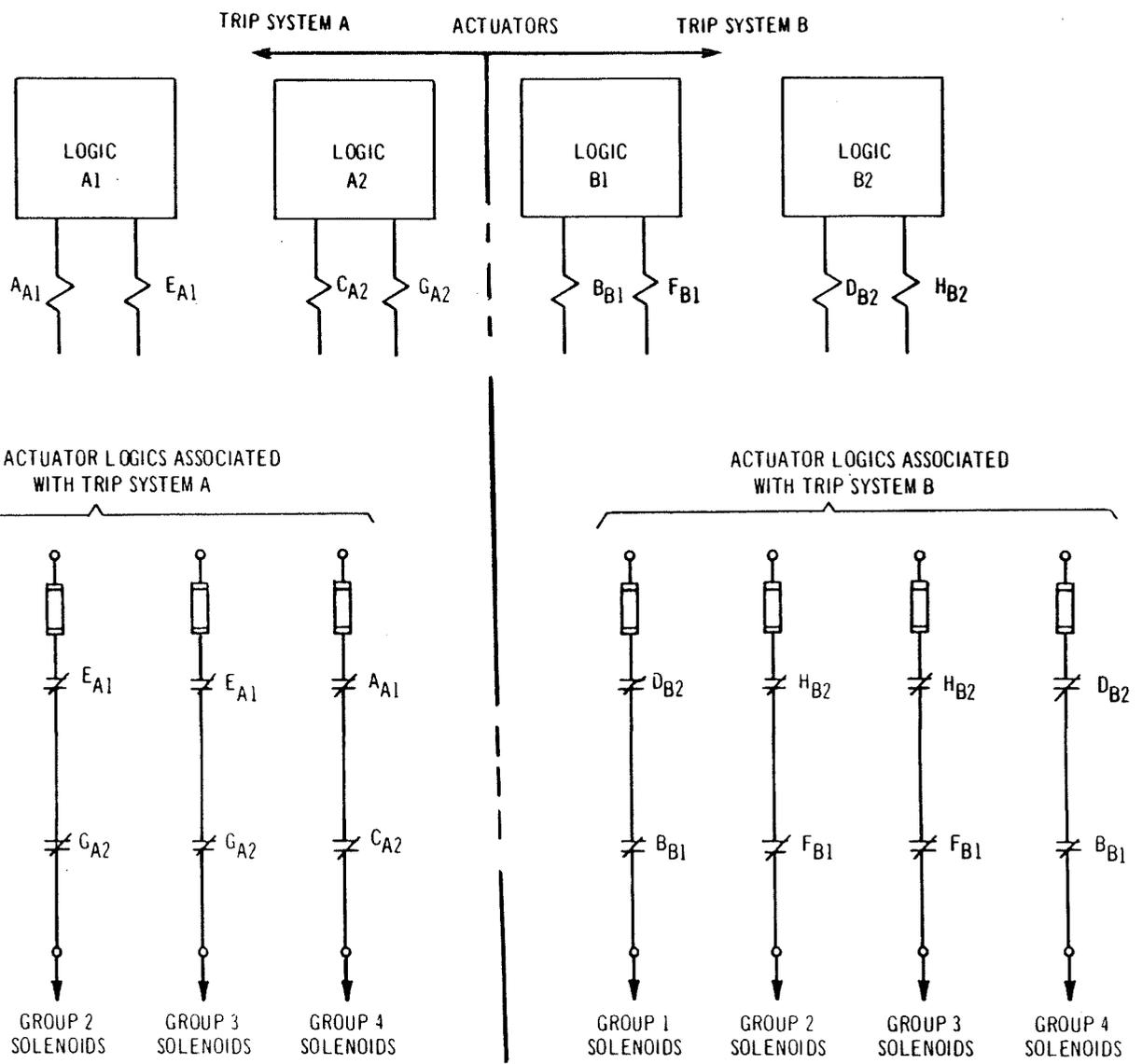
REACTOR PROTECTION SYSTEM LOGICS
 (CONTACTS SHOWN IN NORMAL CONDITION)

KEY: F022A - STEAM LINE A, INBOARD VALVE
 F028A - STEAM LINE A, OUTBOARD VALVE
 F022B - STEAM LINE B, INBOARD VALVE
 F028B - STEAM LINE B, OUTBOARD VALVE

F022C - STEAM LINE C, INBOARD VALVE
 F028C - STEAM LINE C, OUTBOARD VALVE
 F022D - STEAM LINE D, INBOARD VALVE
 F028D - STEAM LINE D, OUTBOARD VALVE

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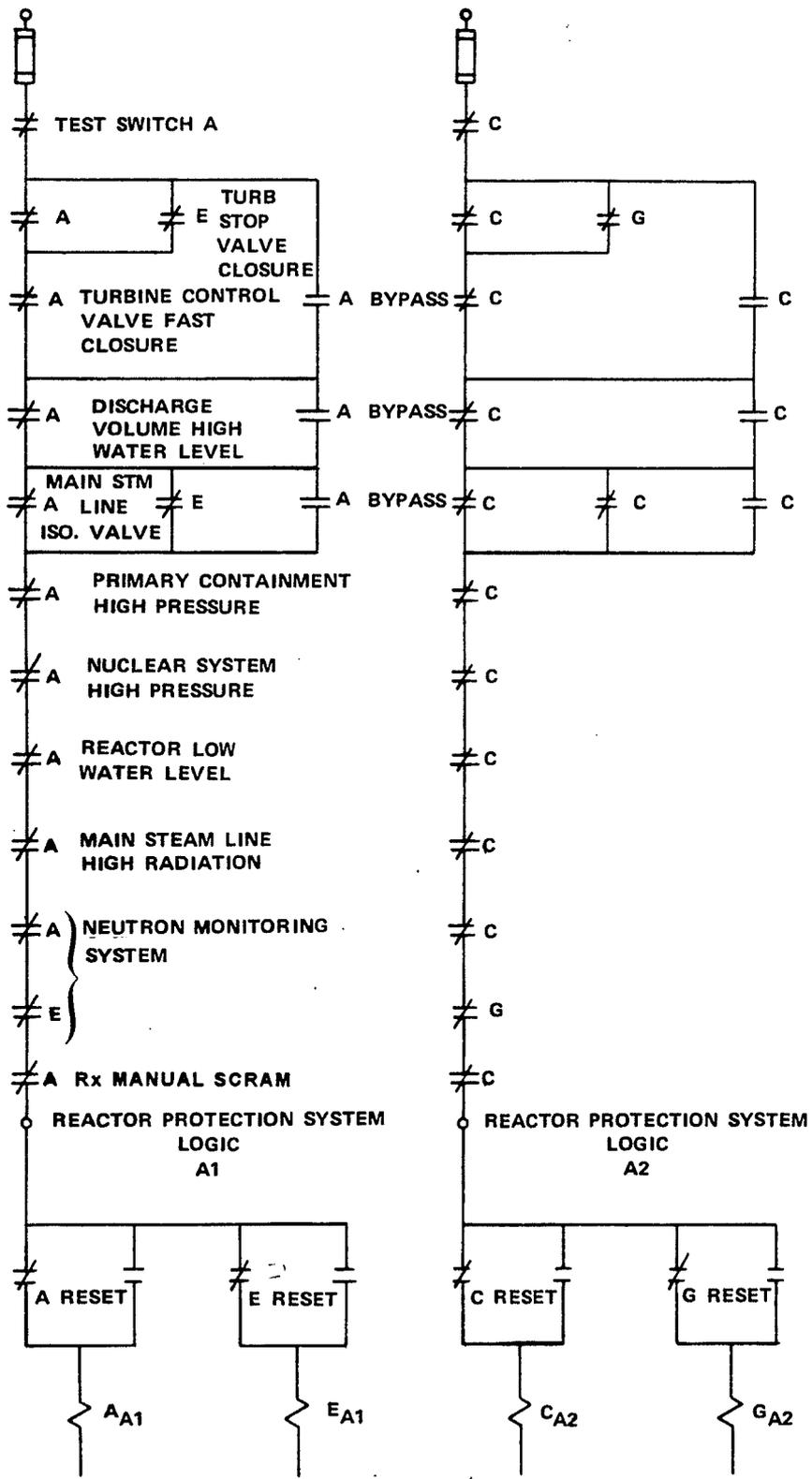
FIGURE 7.2-7
 TYPICAL CONFIGURATION FOR MAIN STEAM LINE
 ISOLATION REACTOR TRIP



NOTE: CONTACTS SHOWN IN NORMAL CONDITION

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FIGURE 7.2-8
 ACTUATORS AND ACTUATOR LOGIC



NOTE: CONTACTS SHOWN IN NORMAL OPERATING STATE

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FIGURE 7.2-9
 LOGICS IN ONE TRIP SYSTEM

7.3 ENGINEERED SAFETY FEATURE SYSTEMS

Included in this section are descriptions and analyses of the instrumentation and controls for the following engineered safety feature (ESF) systems:

- a. Emergency core cooling system (ECCS)
- b. Primary containment and reactor vessel isolation control system
- c. Emergency core cooling auxiliary system
- d. Emergency equipment cooling water system
- e. Main control room atmospheric control system
- f. Standby gas treatment system
- g. Standby power system
- h. Post-LOCA combustible gas control system.

The format of this section departs from the Regulatory Guide 1.70, Revision 2, Standard Format Guide in that the description and analysis are grouped together under each system heading rather than by descriptions and by analyses.

The main steam isolation valve leakage control system is discussed in Subsection 6.2.6.

7.3.1 Emergency Core Cooling System

7.3.1.1 Design-Basis Information

The design-basis information for the ECCS, required by Section 3 of IEEE 279-1971, is provided in Subsection 7.1.2.1.3.

7.3.1.2 System Description

The ECCS includes the following subsystems:

- a. High-pressure coolant injection system (HPCI)
- b. Automatic depressurization system (ADS)
- c. Core spray system
- d. Low-pressure coolant injection (LPCI) mode of the residual heat removal (RHR) system.

The purpose of ECCS instrumentation and control is to initiate appropriate responses from the ECCS to ensure that the fuel is adequately cooled in the event of a design-basis LOCA. The cooling provided by the system restricts the release of radioactive materials from the fuel by preventing or limiting the extent of fuel damage following situations in which reactor coolant is lost.

The equipment involved in the control of these systems includes automatic injection valves, steam turbine pump controls, electric pump controls, relief valve controls, and the switches, contacts, and relays that make up sensory logic channels. Testable check valves and certain

automatic isolation valves are not included in this description since they are pertinent to the containment and reactor vessel isolation control system (CRVICS).

Power Sources

The instrumentation and control of the ECCS is powered by the 130-V dc and 120-V ac systems, and by the standby power system when required. The redundancy and separation of these power supply systems are consistent with the redundancy and separation of the ECCS instrumentation and control. Both of these power supply systems are described in detail in Chapter 8.

7.3.1.2.1 High Pressure Coolant Injection System Instrumentation and Control

When actuated, the HPCI system pumps water from either the condensate storage tank or the suppression chamber to the reactor pressure vessel (RPV) via the "A" feedwater pipeline. The HPCI system includes one turbine-driven pump, one dc motor-driven auxiliary oil pump, one barometric condenser dc condensate pump, one barometric condenser dc vacuum pump, other auxiliaries, automatic valves, control devices for this equipment, sensors, trip channels, and logic circuitry. The arrangement of equipment and control devices is shown in Figure 7.3-1.

Pressure and level transmitters used in the HPCI system are located on racks in the reactor building. The only operating component for the HPCI system that is located inside the primary containment is one of the two isolation valves in the HPCI turbine steam supply isolation valves. The rest of the HPCI system instrumentation and control components are located outside the primary containment. Cables connect the sensors to control circuitry in the main control room. The system is arranged to allow a full-flow functional test of the system during normal reactor power operation. The system will automatically return from the full-flow test mode to accident response operation.

The controls automatically initiate the HPCI system on receipt of either an RPV low water level signal or a primary containment high-pressure signal, and bring the system to its design flow rate, given in Section 6.3, within 60 sec. The controls then function to provide design makeup water flow to the RPV until the water level in the RPV reaches an upper limit. At this time the HPCI system shuts down until further need is indicated. The HPCI system would automatically restart on low water level and operate indefinitely without manual intervention. The controls are arranged to allow manual startup, operation, and shutdown from the main control room.

7.3.1.2.1.1 Initiating Circuits

The RPV low water level is monitored by four level transmitters that sense the differences between the pressure of a constant reference column of water and the pressure due to the actual height of water in the vessel. Two pipelines, attached to taps above and below the normal water level on the RPV, are required for the level transmitters. The lines are physically separated from each other and tap off the RPV at widely separated points. These same lines are also used for pressure and water level instruments for other systems.

A backfill system is installed on each level instrument reference leg. The system provides a metered flow of water from the control rod drive system to each leg. The flow is low enough

to not affect the performance of the instrumentation. The backfill is designed to prevent the accumulation of dissolved noncondensable gases in the reference legs.

The level transmitters and primary containment high-pressure transmitters for the HPCI are arranged in pairs, with the transmitter contacts in a "one-out- of-two taken twice" electrical arrangement. This arrangement ensures that no single event can prevent HPCI initiation from RPV low water level or drywell high pressure. Cables from the level transmitters lead to the trip unit racks located in the fourth floor of the reactor building for logic and sequencing action.

The primary containment high-pressure initiation signal for the HPCI system uses output from the same trip unit that serves the RHR and core spray systems, as described in Subsection 7.3.1.2.3.

The HPCI system turbine is functionally controlled as shown in Figure 7.3-2. A speed governor limits the turbine speed to its maximum operating level. A control governor receives a pump flow signal and adjusts the turbine steam control valve so that design HPCI pump discharge flow rate is obtained. Manual control of the governor is possible in the test mode, but the governor automatically returns to automatic control on receipt of a HPCI initiation signal.

Figure 7.3-2 shows the various modes of turbine control. The flow signal used for automatic control of the turbine is derived from a differential pressure measurement across a flow element in the HPCI pump discharge line. The governor controls the pressure applied to the hydraulic operator of the turbine control valve, which in turn controls the steam flow to the turbine. Hydraulic pressure is supplied for both the turbine control valve and the turbine stop valve by the dc-powered oil pump during startup, and then by the shaft-driven hydraulic oil pump when the turbine reaches a certain speed.

On receipt of an initiation signal, the auxiliary oil pump starts, providing hydraulic pressure for the turbine stop valve and turbine control valve hydraulic operator. Since there is no flow at first in the HPCI system, the flow signal runs the control governor to the high speed stop. As hydraulic oil pressure is developed, the turbine stop valve and the turbine control valve open simultaneously, and the turbine accelerates toward the speed setting of either the control governor or the speed governor, whichever is lower. As HPCI flow increases, the flow signal adjusts the control governor setting so that rated flow is maintained. The turbine is automatically or manually shut down by tripping the turbine stop valve closed if any of the following signals are detected:

- a. Turbine overspeed (automatic)
- b. High turbine exhaust pressure (automatic)
- c. Low pump suction pressure (automatic)
- d. RPV high water level (automatic)
- e. HPCI isolation signal (automatic)
- f. Manual pushbutton
- g. HPCI steam supply pressure low (automatic).

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Turbine overspeed indicates a malfunction of the turbine control mechanism. High turbine exhaust pressure indicates a condition that threatens the physical integrity of the exhaust line. Low pump suction pressure warns that cavitation and lack of cooling can cause damage to the pump, which could place it out of service. A turbine trip is initiated for these conditions so that if the causes of the abnormal conditions can be found and corrected, the system can be quickly restored to service. The trip settings are selected far enough above or below normal values so that a spurious turbine trip is unlikely, but not too close to values that could cause damage before the turbine is shut down.

Turbine overspeed is detected by a standard turbine overspeed detection device. Two pressure transmitters are used to detect high turbine exhaust pressure; either transmitter can initiate turbine shutdown. One pressure switch is used to detect low HPCI pump suction pressure.

High water level in the RPV indicates that the HPCI system has performed satisfactorily in providing makeup water to the RPV. Further increase in level could result in HPCI turbine damage caused by gross carryover of moisture. The RPV high-water-level setting, which trips the turbine, is near the top of the steam separators and is sufficient to prevent gross moisture carryover to the turbine. Two level transmitters that sense differential pressure are arranged so that both transmitters are required to trip simultaneously to initiate a turbine shutdown.

UFSAR Section 7.5.1.4.2.1 describes that the wide-range water level system is uncompensated for variation in reactor water density and is calibrated to be most accurate at operational pressure and temperature conditions. At low reactor coolant temperatures and pressures, the higher water density causes the wide-range instruments to read higher than both the narrow-range instruments and the actual water level. Below approximately 550 psig, this phenomenon results in a wide-range level above the RPV high water level setting (Level 8) when the actual water level is normal (Reference 5). With wide-range level above Level 8, HPCI automatic initiation on the primary containment high-pressure signal and HPCI manual initiation are both inhibited by the Level 8 trip signal. HPCI automatic initiation on RPV low water level (Level 2) remains available since the Level 8 signal is automatically reset by the occurrence of a Level 2 actuation signal. For accidents occurring below 600 psig for which HPCI may be effective, analysis (Reference 5) has shown that HPCI automatic initiation at Level 2 is sufficient to perform the intended safety function and that the analyses of record from normal reactor pressure are bounding. Amendment 206 (Reference 6) revised the Technical Specifications to indicate that the HPCI functions of automatic initiation on primary containment high-pressure and manual initiation are not required to be operable below a reactor pressure of 550 psig.

The control scheme for the turbine auxiliary oil pump is shown in Figure 7.3-2. The controls are arranged for automatic manual control. On receipt of an HPCI initiation signal, the auxiliary oil pump starts and provides hydraulic pressure to open the turbine stop valve and the turbine control valve. As the turbine gains speed, the shaft-driven oil pump begins to supply hydraulic pressure. Should the shaft-driven oil pump malfunction, causing oil pressure to drop, the auxiliary oil pump restarts.

Operation of the barometric condenser components, which consist of the barometric condenser condensate pump (dc), the barometric condenser vacuum pump (dc), and the

barometric condenser water level instrumentation, prevents outleakage from the turbine shaft seals. Operation of this equipment is automatic, as shown in Figure 7.3-2, and failure does not prevent the HPCI system from providing water to the RPV.

7.3.1.2.1.2 Logic and Sequencing

The RPV low water level and primary containment (drywell) high pressure are the two functions that can automatically start the HPCI system, as indicated in Figure 7.3-2 Sheet 1. The RPV low water level is an indication that reactor coolant is being lost and that the fuel is in danger of being overheated. Primary containment high pressure is an indication that a breach of the nuclear system process barrier has occurred inside the drywell.

The logic scheme used for the initiating functions is a "one-out-of-two taken twice" arrangement for both RPV low water level and high drywell pressure. Either one can initiate HPCI. The logic is powered from reliable dc buses. Level transmitters and drywell pressure transmitters are shared with core spray initiation.

Instrument settings for the HPCI system instrumentation and control are listed in Table 7.3-1. The RPV low water level (L2) setting for HPCI initiation is selected high enough above the active fuel to start the HPCI in time to prevent fuel clad failure and to prevent an unacceptable fraction of the core from reaching the temperature at which fuel fragmentation occurs (Section 6.3). The water level setting is far enough below normal levels that spurious HPCI system startups are avoided. The primary containment high-pressure setting is selected to be as low as possible without inducing spurious HPCI system startup.

To prevent the turbine pump from being damaged by overheating at reduced HPCI pump discharge flow, a pump discharge bypass is provided to route the water discharged from the pump to the suppression chamber. The bypass is controlled by an automatic, dc motor-operated valve whose control scheme is shown in Figure 7.3-2. At high HPCI flow, the valve is closed; at low flow, the valve is opened. Flow switches that measure the pressure difference across a flow element in the HPCI pump discharge line provide the signals used for flow indication.

The HPCI initially uses the condensate storage tank as the source of coolant to provide high-grade water to the reactor. A single failure of the condensate low-level switches or suppression pool high-level switches could cause a switchover of HPCI source water from the condensate storage tank to the suppression pool. A premature switchover has no adverse safety impact. The transfer to the suppression pool feature is to ensure an adequate long-term quantity of coolant or to control the pool level. The long-term source of water for the HPCI system is the suppression pool; thus a failure causes switchover to the desired suction source.

7.3.1.2.1.3 Bypasses and Interlocks

To prevent the HPCI steam supply line from filling up with water and cooling, a condensate drain pot, steam line drain, and appropriate valves are provided in a drain line arrangement just upstream of the turbine supply valve. The control scheme is shown in Figure 7.3-2. The controls position valves so that during normal operation, steam line drainage is routed to the main condenser. On receipt of an HPCI initiation signal, the drainage path is isolated. The water level in the steam line drain condensate pot is controlled by a level switch and a pilot air-operated solenoid valve that energizes to allow condensate to flow out of the pot.

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During test operation, the HPCI pump discharge is routed to the condensate storage tank. Two valves, a dc motor operated valve (E4150F008) and an air operated valve (E41F011), are installed in the pump discharge to the condensate storage tank line. The piping arrangement is shown in Figure 7.3-1. The control scheme for the two valves is shown in Figure 7.3-2. On receipt of an HPCI system initiation signal, the two valves close and remain closed. The valves are interlocked to close if either of the suppression chamber suction valves are fully open. Valve E41F011 functions as a throttle valve while operating in the test mode. It is a fail-close globe valve with flow over the seat, capable of fast closure. It is credited for closure against pump shut off head. Valve E4150F008 is a slower motor operated valve which provides redundant isolation of the test line. If manual transfer from the test mode to vessel injection is desired, operator action is needed to close E41F011 prior to opening the HPCI injection valve. Numerous indications pertinent to the operation and condition of the HPCI system are available to the main control room operator. Figures 7.3-1 and 7.3-2 show the various indications provided.

7.3.1.2.1.4 Redundancy and Diversity

The HPCI system is actuated either by RPV low water level or by primary containment high pressure. Both of these conditions could result from a LOCA. The redundancy of the HPCI system initiating circuits is consistent with the design of the HPCI system. A single failure does not prevent activation.

7.3.1.2.1.5 Actuated Devices

All automatic valves in the HPCI system are equipped with remote manual test capability so that the entire system can be operated from the main control room. Motor-operated valves are provided with appropriate limit switches to turn off the motors when the fully open or fully closed positions are reached. Valves that are automatically closed upon isolation signals are equipped with manual reset devices so that they cannot be reopened without operator action. All essential components of the HPCI system controls operate independently of offsite ac power.

To ensure that the HPCI system can be brought to the design flow rate within 60 sec from the receipt of the initiation signal, the following operating times for essential HPCI system valves are provided by the valve operation mechanisms.

- a. HPCI turbine steam supply valve - 50 sec
- b. HPCI pump discharge valves - 40/50 sec
- c. HPCI pump minimum flow bypass valve - 22.5 sec.

The operating time is the time required for the valve to travel from the fully closed to the fully open position, or vice versa. A HPCI steam supply line inboard isolation valve and the bypass valve around the HPCI outboard isolation valve are provided; they are intended to isolate the HPCI steam line in the event of a break in that line. A normally closed dc motor-operated isolation valve is located in the turbine steam supply line just upstream of the turbine stop valve. The piping and logic scheme for this valve is shown in Figures 7.3-1 and 7.3-2. On receipt of a HPCI system initiation signal, this valve opens and remains open until closed by operator action from the main control room.

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Two isolation valves are provided in the steam supply line to the turbine. The valve inside the drywell is a Division I ac-powered valve that is normally open. The valve outside the drywell is a Division II dc-powered valve and is normally closed. A small bypass valve provides a warmup path around the closed valve to keep the turbine steam line free of water. The HPCI steam supply outboard isolation valve is signaled open on a HPCI initiation, and the HPCI steam supply inboard, and outboard isolation valves, and the bypass valve around the HPCI steam supply outboard isolation valve close automatically on a HPCI system isolation. The isolation signal takes precedence over the initiation signal. The control diagram is shown in Figure 7.3-2.

The primary element instrumentation for HPCI system isolation consists of the following:

- a. Inside valve E41-F002
 1. Ambient temperature sensor - emergency area cooler high temperature. Isolation started as soon as activated
 2. Differential pressure transmitter - HPCI steam line high flow; a time delay has been added to this isolation to prevent spurious trips that could result from pressure spikes associated with pump startup
 3. Pressure transmitters - HPCI turbine exhaust diaphragm high pressure
 4. Pressure transmitters - HPCI steam supply pressure low.
- b. Outside valve E41-F003 - Instrumentation similar to that described for the inside valve causes the outside valve to isolate if the conditions warrant isolation. Both valves can be individually actuated by manual pushbutton switches

Three pump suction valves are provided in the HPCI system. One valve allows pump suction from the condensate storage tank while the other two series valves allow water to be taken from the suppression chamber. The condensate storage tank is the preferred source. All three valves are operated by dc motors. The control arrangement is shown in Figure 7.3-2

On receipt of a HPCI system initiation signal, the condensate storage tank suction valve receives an open signal. If the water level in the condensate storage tank falls below a preselected level, the suppression chamber suction valves automatically open and the condensate storage tank suction valve automatically closes. Two level transmitters detect the condensate storage tank low-water-level condition. Either transmitter causes the suppression chamber suction valves to open and the condensate storage tank suction valve to close. The suppression chamber suction valves also automatically open if a high water level is detected in the chamber. Two level transmitters monitor the water level. Either transmitter can initiate opening of the suppression chamber suction valves. If open, the suppression chamber suction valves automatically close on receipt of the signals that initiate HPCI steam line isolation

Two dc motor-operated HPCI pump discharge valves in the pump discharge pipeline are provided. The control schemes for these two valves are shown in

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Figure 7.3-2. Both valves are arranged to open on receipt of either one of the HPCI system initiation signals. One of the pump discharge valves closes automatically on receipt of a turbine trip signal. The other valve remains open after HPCI system initiation until closed by the operator in the main control room.

7.3.1.2.1.6 Separation

General

Separation within the ECCS is such that no single failure can prevent core cooling. Instrumentation and control equipment and wiring are segregated into separate divisions designated Divisions I and II. Separate requirements are also maintained for the control and motive power for the ECCS. System separation is as follows:

<u>Division I</u>	<u>Division II</u>
Core spray pump A and pump C	Core spray pump B and pump D
Automatic depressurization	HPCI
RHR A and C	RHR B and D

Systems shown opposite each other are redundant. In addition, should HPCI fail to reduce RPV pressure through coolant makeup injection, the ADS will depressurize the RPV to allow LPCI and Core Spray to provide adequate core cooling. Control logic for all Division I systems is the 260/130-V dc Division I battery and for Division II systems is the 260/130-V dc Division II battery.

Specific

The HPCI system is a Division II system except for the HPCI main pump test line isolation valve E41F011, in which its motive force is fed from the interruptible air supply (note: the solenoid valves for E41F011 remain in Division II logic), and inside isolation valve E4150-F002, which is Division I ac powered. The E4150-F002 valve is controlled by logic operated from Division I 260/130-V dc battery so that no single failure can prevent the automatic closure of at least one valve of the pair of isolation valves. To maintain the required separation, HPCI system logic relays, instruments, and manual controls are mounted so that separation from Division I is maintained. Logic relays, instruments, and manual controls for outboard steam line isolation valve E41-F003 and bypass valve E41-F600 are separated from Division I equipment.

7.3.1.2.1.7 Testability

Instrumentation and control of the ECCS is designed to be completely testable during reactor operation. Specific test schedules for this and subsequent systems in this section (Section 7.3) are given in the Technical Specifications. Systems providing core cooling water are arranged with bypass valves so that pumps may be operated at design flow.

Instrumentation and control is designed to establish that the following functions are met:

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- a. Each instrument channel functions independently of all others
- b. Sensing devices respond to process variables and provide channel trips at correct values
- c. Sensors and associated instrument channels respond to both steady-state and transient changes in the process variable within specified accuracy and time limitations, and provide channel trips at correct values even when affected by process variations that may extend grossly beyond the expected trip setpoint
- d. Paralleled circuit elements can perform their intended functions independently
- e. Series circuit elements are free from shorts that can abrogate their function
- f. Redundant instrument or logic channels are free from interconnecting shorts that could violate independence if a single malfunction should occur
- g. No element of the system is omitted from the test if it could impair system operability in any way. (If the test is done in parts, then the parts must overlap sufficiently to ensure operability of the entire system)
- h. Each monitoring alarm or indication function is operable.

The emergency core cooling system response times are shown in Technical Requirements Manual Volume I Table 3.3.5.1-1, which is referenced in UFSAR Table 7.3-11. Response time testing is required by the Technical Specifications. Technical Specification Table 3.3.3-3 was deleted from the Technical Specifications and added to the UFSAR as Table 7.3-11 (TRM Table 3.3.5.1-1) in agreement with NRC Generic Letter 93-08 and Amendment Number 100 to the Technical Specifications. The response times information of the UFSAR Table 7.3-11 was then relocated to the Technical Requirements Manual Volume I.

The periodic response time testing for the ECCS instrument channels has been eliminated. The BWROG Report NEDO-32291A provides the required analyses as briefly described in 7.2.1.1.3.8.1.

Specific

The HPCI system is provided with test jacks in each logic. The low reactor level or high drywell pressure "one-out-of-two taken twice" circuit can be tested completely by actuating only one instrument channel at a time. Insertion of the test plug at the logic relay panel actuates an annunciator in the main control room, indicating that the HPCI system is in test status.

7.3.1.2.1.8 Environmental Considerations

The control mechanism for the inboard isolation valve on the HPCI system turbine steam line is the only HPCI system control component located inside the primary containment that must remain functional in the environment resulting from a LOCA. The environmental capabilities of this valve are discussed in Subsection 7.3.2.2.9. The HPCI system instrumentation and control equipment located outside the primary containment is selected in consideration of the normal and accident environments in which it must operate. These conditions are listed in Table 3.11-3.

7.3.1.2.1.9 Operational Considerations

The HPCI system is not required for normal operations. Under the abnormal or accident conditions when it is required, initiation and control are provided automatically for at least 10 minutes. The automatic depressurization system (ADS) can also depressurize the reactor vessel to a point when the low pressure ECCS systems can be initiated to inject water into RPV. With the incorporation of the high drywell pressure signal bypass timer into the ADS automatic initiation logic, as discussed in Subsection 7.3.1.2.2.2, no operator actions are required to actuate ADS after any LOCA. When the bypass timer times out, the high drywell pressure initiation permissive signal is bypassed, and the ADS will be automatically initiated based on reactor vessel low water level signal alone. This is true even when the main steam line isolation valves (MSIV) line breaks outside the drywell and the break becomes isolated due to MSIV closure.

7.3.1.2.2 Automatic Depressurization System Instrumentation and Control

Automatically controlled relief valves are installed on the main steam lines inside the primary containment. The valves are dual purpose in that they relieve pressure either by normal mechanical action or by automatic action of an electric-pneumatic control system. Actuation is initiated on receipt of a signal indicating high drywell pressure, low RPV water level, and core spray and/or RHR pumps running. A time delay allows the operator to delay actuation if the HPCI system is in operation. The relief by normal mechanical action is intended to prevent overpressurization of the nuclear steam supply system (NSSS). If the HPCI system is not available during a small-break LOCA, the depressurization by automatic action of the ADS is intended to reduce NSSS pressure so that the core spray system or LPCI system can inject water into the RPV.

The automatic instrumentation and control equipment for the relief valves is described in this subsection. The instrumentation and control for one of the relief valves is discussed. Other relief valves equipped for automatic depressurization are identical.

The control system consists of pressure and water level sensors arranged in trip systems that control a solenoid-operated pilot air valve. The solenoid-operated pilot valve controls the pneumatic pressure applied to a bellows-actuator that operates the relief valve directly. An accumulator is included with the control equipment to store pneumatic energy for relief valve operation. The accumulator is sized to provide pneumatic pressure for five actuations of the pilot valve during interruptions if the pneumatic supply to the accumulator is switched from the normal to the emergency backup supply source.

Cables from the sensors lead to the trip unit racks located in the reactor building, where the logic arrangements are formed in cabinets. The electrical control circuitry is powered by dc power from the plant batteries. The power supplies for the redundant control circuits are selected and arranged to maintain tripping ability in the event of an electrical power circuit failure. Electrical elements in the control system energize to open the relief valve.

7.3.1.2.2.1 Initiating Circuits

The pressure and level transmitters used to initiate one ADS logic are separated from those used to initiate the redundant logic on the same ADS valve. Reactor pressure vessel low

water level is detected by six transmitters that measure differential pressure. Primary containment high pressure is detected by four pressure transmitters. The primary containment high pressure signals are arranged to seal into the control circuitry. These signals must be manually reset to clear.

A timer is used in each ADS logic. The time delay setting before actuation of the ADS is long enough that the HPCI system has time to operate, yet not so long that the LPCI and core spray systems are unable to adequately cool the fuel if the HPCI system cannot. An alarm in the main control room is activated when either of the timers is timing. Resetting the ADS initiating signals recycles the timer. A display of the time remaining before the ADS actuates is available to the operator in the main control room.

7.3.1.2.2.2 Logic and Sequencing

The two initiating signals used for ADS are RPV low water level (level 1) and drywell high pressure. Simultaneous occurrences of RPV low water level and drywell high pressure conditions initiate a nominal 105 second time delay. After that time delay, ADS safety relief valves will operate if a sufficient number of low pressure ECCS pumps (RHR and/or core spray) are available for adequate core cooling. RPV low water level (level 1) signal also initiates a bypass timer which is set for a nominal 7 minutes. This time delay is provided to bypass the drywell high pressure permissive. If for some reason the drywell high pressure is not detected, the RPV low water level signal alone will actuate the ADS safety relief valves. The 7 minute bypass time delay, plus the original 105 second time delay and the permissive from appropriate ECCS pump discharge pressure will provide for ADS actuation. The instrument trip settings are given in Table 7.3-2.

Figure 7.3-6 shows the logic for ADS actuation, with the High Drywell Pressure Bypass Timer started on level 1. A nominal bypass timer time delay setpoint of 7 minutes was established for Fermi 2 from the unique analyses performed by General Electric, which is consistent with the analysis presented in NEDO-24708A (Figure Group 3.5.2.1-33). The results of these analyses demonstrate that adequate core cooling is ensured for isolation events, even with the ADS blowdown delayed after level 1 for an analytical time of 10 minutes. A subsequent confirmatory analysis established the adequacy of the bypass timer setpoint including increases in reactor rated thermal power to 3486 MWth (refer to Subsection 6.3.2.2 for analytical values).

Starting the bypass timer at level 1 allows the operator enough time to control the system manually and still ensure automatic depressurization in time to prevent excessive fuel heat-up, even under the worst-case conditions described above.

Primary containment high pressure indicates a breach in the nuclear system process barrier inside the drywell. For each logic train, a permissive signal indicating LPCI or core spray pump discharge pressure is also required. Discharge pressure on either of the two LPCI pumps or two of the core spray pumps (one discharge pressure sensor per pump) in the same division is sufficient to give the permissive signal. This signal prevents initiation of the ADS until the low-pressure ECCS is operating.

After receipt of the initiation signals and after a delay provided by timers, each of the solenoid pilot air valves is energized. This allows pneumatic pressure from the accumulator

to act on the air cylinder operator. The air cylinder operator holds the relief valve open. Lights in the main control room indicate when the SRV is opened.

Manual reset switches are provided for the ADS initiation signal and primary containment high-pressure signals. By resetting these signals manually, the delay times are recycled. The operator can use the reset pushbuttons to delay or prevent automatic opening of the relief valves if such delay or prevention is prudent.

A manual inhibit switch is also provided for each ADS trip system. These switches allow the operator to inhibit ADS operation without repeatedly pressing the reset pushbuttons. Operation of the manual inhibit switch will activate a white indicating light and an annunciator to alert the operator of the inhibit action. Enabling the inhibit function will not terminate an ADS logic actuation after the 105 second time delay has elapsed. At this point, only the reset pushbutton can be used to affect the ADS operation. Refer to Subsection 6.3.2.17 for criteria in using the reset pushbutton switch.

Control switches are available in the main control room for each SRV associated with the ADS. The OPEN position is for manual SRV operation.

Two divisional ADS logic systems are provided: ADS "A,C" logic and ADS "B,D" logic (Figure 7.3-4). Division I sensors for low reactor water level and high drywell pressure initiate ADS "A,C" logic, Division II sensors initiate ADS "B,D" logic. Either ADS "A,C" logic or "B,D" logic actuates the solenoid pilot valve on each ADS valve.

The RPV low water level initiation setting for the ADS is selected to depressurize the RPV in time to allow adequate cooling of the fuel by the LPCI system or core spray system following a LOCA in which the HPCI system fails to perform its function adequately. The primary containment high-pressure setting is selected as low as possible without inducing spurious initiation of the ADS. This provides timely depressurization of the RPV if the HPCI system fails to start or fails after it successfully starts following a LOCA. Since the ADS is a backup for HPCI, different drywell pressure-sensing transmitters are used for ADS and HPCI.

The low-pressure pump discharge pressure setting used as a permissive for depressurization is selected to ensure that at least one of the four LPCI pumps or one of the core spray loops has received electrical power, has started, and is capable of delivering water into the RPV. The setting is high enough to ensure that the pump will deliver at near rated flow without being so low as to provide an erroneous signal indicating that the pump is actually running.

7.3.1.2.2.3 Bypasses and Interlocks

It is possible for the operator to manually delay the depressurizing action by depressing the timer reset pushbutton. The operator may also interrupt the depressurization at any time by the same action.

A manual switch is also provided to allow the operator to inhibit ADS operation (prior to its automatic initiation) instead of successively pressing the reset pushbuttons to reset the ADS timer.

The operator would make these decisions based on an assessment of other plant conditions.

7.3.1.2.2.4 Redundancy and Diversity

The ADS is initiated by a combination of high drywell pressure and low RPV water level. The initiating circuits for each of these parameters are redundant, as verified by the circuit description in this section.

7.3.1.2.2.5 Actuated Devices

All relief valves in the ADS are equipped with remote manual switches so that the ADS valves can be manually as well as automatically operated. The valves also relieve pressure by built-in mechanical action.

7.3.1.2.2.6 Separation

General

Refer to Subsection 7.3.1.2.1.6.

Specific

The ADS is a Division I system, but also makes use of Division II power and pneumatic supply. The "A,C" sensing and control logic is connected to the Section 1 half of the 260/130-V dc Division I battery. The "B,D" control logic is fed from the Section 2 half of the 260/130-V dc Division I battery, with automatic transfer to the Section 1 half. The "B,D" sensing and interposing relay circuitry (to the "B,D" control logic) is fed from the Division II battery.

Each valve is normally fed from the Section 1 half of the Division I battery, but each has a power monitor to automatically transfer to the Section 2 half of Division I battery on a power failure.

7.3.1.2.2.7 Testability

Refer to Subsection 7.3.1.2.1.7.

Specific

The ADS has two trip systems; either one can initiate automatic depressurization. Each trip system has two trip logics, both of which must trip to initiate depressurization. Four test jacks are provided, one in each trip logic. To prevent spurious actuation of the ADS during testing, only one trip logic is actuated at a time. An alarm is provided if a test plug is inserted on both trip logics. Operation of the test plug switch along with actuation of the ADS reactor level interlock and the ac interlock (RHR or core spray pumps running) closes one of the two series relay contacts in the valve-solenoid circuit. This causes a light to turn on, indicating proper trip logic operation. When the test is performed, continuity of the solenoid circuit is verified. Testing of the other trip logic and trip system is accomplished in a similar manner. Annunciation is provided in the main control room whenever a test plug is inserted to indicate ADS in test status.

7.3.1.2.2.8 Environmental Considerations

The signal cables, solenoid valves, pressure switches for indication, and SRV operators are the only instrumentation and control equipment for the ADS located inside the primary containment. They remain functional in the environment resulting from a LOCA. These items operate in the most severe environment resulting from a design-basis LOCA (Section 3.11). Gamma and neutron radiation is also considered in the selection of these items. Equipment located outside the drywell also operates in its normal and accident environments.

7.3.1.2.2.9 Operational Considerations

The instrumentation and control of the ADS is not required for normal plant operations. When automatic depressurization is required, it is initiated automatically by the circuits described in this section. No operator action is required for at least 10 minutes following initiation of the system.

A temperature element is installed on the SRV discharge piping several feet from the valve body. The temperature element is connected to a multipoint recorder in the main control room so that a means of detecting SRV leakage during plant operation is provided. When the temperature in any SRV discharge line exceeds a preset value, an alarm is sounded in the main control room. The alarm setting is enough above normal rated power temperatures to avoid spurious alarms, yet low enough to give early indication of SRV leakage.

7.3.1.2.3 Core Spray System Instrumentation and Control

The core spray system consists of two independent spray loops, as illustrated in Figure 7.3-7. The core spray system is capable of supplying sufficient cooling water to the RPV to adequately cool the core following a design-basis LOCA. The two spray loops are physically and electrically separated so that no single physical event makes both loops inoperable. Each loop includes two ac pumps, appropriate valves, and the piping to route water from the suppression chamber to the RPV.

The instrumentation and control for the core spray system includes the sensors, relays wiring, and valve-operating mechanisms used to start, operate, and test the system. Except for the testable check valve in each spray loop, which is inside the primary containment, the sensors and valve closing mechanisms for the core spray system are located in the reactor building.

Cables from the sensors are routed to the trip unit racks located in the auxiliary building, where the control circuitry is assembled in electrical panels. Each core spray pump is powered from a different ac bus which is capable of receiving standby power. The power supply for automatic valves in each loop is the same as that used for the core spray pump in that loop. Control power for each of the core spray loops comes from separate dc buses.

The electrical equipment for one core spray loop is located in a separate cabinet from that used for the electrical equipment for the other loop.

7.3.1.2.3.1 Initiating Circuits

Primary containment pressure is monitored by four pressure transmitters mounted on instrument racks outside the drywell, but inside the reactor building. Cables are routed from

the transmitters to the relay logic cabinets. Each drywell high-pressure trip channel provides an input into the trip logic shown in Figure 7.3-8. Pipes that terminate in the reactor building allow the transmitters to communicate with the drywell interior.

Four drywell pressure transmitters are electrically connected to a "one-out-of-two taken twice" circuit as well as four water-level transmitters to both loops, so that no single event can prevent the initiation of the core spray system due to primary containment high pressure. Contacts from the primary containment high-pressure signal relays are also used in the HPCI, LPCI/RHR, and core spray systems.

Contacts from the RPV low water level (Level 1), initiation signal relays are used in the ADS, core spray, LPCI, and primary CRVICS systems.

7.3.1.2.3.2 Logic and Sequencing

The control scheme for the core spray system is illustrated in Figure 7.3-8. Trip settings are given in Table 7.3-3. The overall operation of the system following the receipt of an initiating signal is as follows:

- a. Test bypass valves are closed and interlocked to prevent opening
- b. If normal ac power is available, the core spray pumps in both spray loops start 5 sec after receiving the initiation signal
- c. If normal ac power is not available, the core spray pumps in both spray loops start 5 sec after standby power becomes available to that particular pump
- d. When the RPV pressure drops to a preselected value, valves open in the pump discharge lines, allowing water to be sprayed over the core.

The RPV low water level indicates that the core is in danger of being overheated due to loss of coolant. Drywell high pressure indicates that a breach of the nuclear system process barrier has occurred inside the drywell. The considerations used in establishing the RPV low water level and primary containment high-pressure settings and the instruments that provide the initiating signals are the same as those used for the HPCI system.

To prevent pump overheating at reduced core spray pump flow, a pump discharge bypass is provided from each loop. The bypass routes the discharge from both pumps in a loop back to the suppression chamber. The bypass is controlled by an automatic motor-operated valve whose control scheme is shown in Figure 7.3-8. At core spray high loop flow, the bypass valve is closed; at low flow, the bypass valve is opened. A flow switch measures the flow in each of the two loops.

7.3.1.2.3.3 Bypasses and Interlocks

During test operation, each core spray loop discharge can be routed to the suppression pool. Motor-operated valves are installed in the test lines. On receipt of a core spray initiation signal, the bypass valve closes and remains closed. The piping arrangement is shown in Figure 7.3-7; the control scheme for the two valves is shown in Figure 7.3-8.

7.3.1.2.3.4 Redundancy and Diversity

The core spray system is completely redundant with two independent spray loops. Initiation of the system is described in Subsection 7.3.1.2.3.1.

7.3.1.2.3.5 Actuated Devices

The control arrangements for the core spray pumps are shown in Figure 7.3-8. The circuitry provides for the detection of power available so that all pumps are automatically started. Each of the four pumps can be controlled by a main control room remote switch, or by the automatic control system. A pressure transducer on the discharge line from each core spray pump provides a signal in the main control room to indicate the successful startup of a pump. If a core spray initiation signal is received when normal ac power is not available, all core spray pumps start 5 sec after restoration of the particular bus voltage from which the pump motor receives power, and avoids overloading the source of standby power. The core spray pump motors are provided with overload protection. Overload relays are applied to maintain power as long as possible without immediate damage to the motors or emergency power system. Valve motors are protected by overload alarms and trips.

Flow measuring instrumentation is provided in each of the two core spray loop discharge lines. The instrumentation provides flow indication in the main control room.

Except where specified otherwise, the remainder of this description of the core spray system refers to one spray loop. The second core spray loop is identical. The control arrangements for the various automatic valves in the core spray system are indicated in Figure 7.3-8.

Each of the valves is equipped with limit switches to turn off the valve motor when the valve reaches the limits of movement. Appropriate interlocks prevent the incorrect positioning of the valves by manual action after the system has been automatically actuated. All motor-operated valves are equipped with limit switches that provide main control room indication of valve position. Each automatic valve can be operated from the main control room.

On receipt of an initiation signal, the test bypass valve is interlocked shut. Having received the initiation signal, the core spray pump discharge valves are automatically opened when NSSS pressure drops to a preselected value. The setting is selected low enough so that the low-pressure portions of the core spray system are not overpressurized, yet high enough to open the valves in time to provide adequate cooling for the fuel. Four pressure transmitters are used to monitor nuclear system pressure. The transmitters can initiate opening of the discharge valves on a "one-out-of-two taken twice" basis. The signal received on automatic core spray initiation overrides all other signals. The full-stroke operating times of the motor-operated pump discharge valves are selected to be rapid enough to ensure proper delivery of water to the RPV in a design-basis accident (DBA). The full stroke operation times are as follows:

- a. Test bypass valve - 108 sec
- b. Pump suction valve - 80 sec
- c. Outboard pump discharge isolation valves - 13 sec
- d. Inboard pump discharge isolation valves - 12 sec

7.3.1.2.3.6 Separation

General

Refer to Subsection 7.3.1.2.1.6.

Specific

The core spray system consists of independent Division I and II systems. Pumps A and C are in Division I and pumps B and D are in Division II. Two separate logics located in separate panels are used. Logic for the "A" loop is operated by the 260/130-V dc Division I battery and logic for the "B" loop is operated by the 260/130-V dc Division II battery.

7.3.1.2.3.7 Testability

General

Refer to Subsections 7.1.3.1 and 7.3.1.2.1.7.

Specific

The core spray system is provided with a test jack in both "A" and "B" logics. The low reactor level or high drywell pressure "one-out-of-two taken twice" circuit can be completely tested by only actuating one instrument channel at a time. Insertion of the test plug at either logic relay panel actuates an annunciator in the main control room, which indicates that the core spray system is in test status.

7.3.1.2.3.8 Environmental Considerations

The only control components pertinent to core spray system operation that are located inside the primary containment are those controlling the testable check valve on each of the two injection lines. Other equipment, located outside the drywell, is selected in consideration of the normal and accident environments in which it must operate (Table 3.11-3).

7.3.1.2.3.9 Operational Considerations

The core spray system is not required for normal operations. When it is required for accident conditions, it is initiated automatically by the circuitry described in this section. No operator action is required for at least 10 minutes following initiation. After this time, manual operation may be assumed by the operator.

Core spray system pressure between the two pump discharge valves is monitored by a pressure switch to permit detection of leakage from the nuclear system into the core spray system outside the primary containment. A detection system is provided to continuously confirm the integrity of the core spray piping between the inside of the RPV and the core shroud. A differential pressure switch measures the pressure difference between the bottom of the core and the inside of the core spray sparger pipe just outside the RPV. If the core spray sparger piping is sound, this pressure difference will be the pressure drop across the core. If integrity is lost, this pressure drop will include the core pressure drop and the steam separator pressure drop. An increase in the normal pressure drop initiates an alarm in the

main control room. Pressure in the core spray pump suction line is monitored by a locally mounted pressure indicator to permit determination of suction head and pump performance.

7.3.1.2.4 Low-Pressure Coolant Injection Instrumentation and Control

Low-pressure coolant injection is an operating mode of the RHR system. The LPCI system is designed to provide water to the RPV following the design-basis LOCA.

Figure 5.5-13 shows the entire RHR system, including the equipment used for LPCI operation.

The instrumentation for LPCI system operation controls other valves in the RHR system. This ensures that the water pumped from the suppression chamber by the main system pumps is routed directly to the reactor. These interlocking features are described in this subsection.

Operation of the LPCI system uses four pumps and two loops, although only three out of four pumps are needed for LPCI cooling. Each loop injects into the reactor through the recirculation pump loop. Figure 5.5-13 shows the location of instruments, control equipment, and LPCI system components. Except for the LPCI system testable check valves, the components pertinent to LPCI system operation are located outside the primary containment.

Power for the LPCI system pumps is supplied from ac buses that can receive standby ac power. Each pump is powered from a separate bus. Motive power for the automatic valves comes from one of the lines that powers the pumps for that loop. Control power for the LPCI components comes from the dc buses.

The LPCI is arranged for automatic and remote manual operation from the main control room. Manual operation allows the operator to act independently of the automatic controls in the event of a LOCA.

7.3.1.2.4.1 Initiating Circuits

The two automatic initiation functions provided for the LPCI systems are RPV low water level and primary containment (drywell) high pressure. Either of these functions initiates the LPCI system.

The low level initiation signal for the LPCI system is a "one-out-of-two taken twice" circuit arrangement using relay contacts from the core spray system. It is used in conjunction with the primary containment high-pressure initiation signal. The high-pressure initiation signal uses pressure transmitters such as those described for the core spray system in Subsection 7.3.1.2.3. A discussion of the LPCI mode loop selection logic is provided in Subsection 6.3.2. Additional information can be found in Subsection 7.3.1.2.4.10.

7.3.1.2.4.2 Logic and Sequencing

The overall LPCI system operating sequence following the receipt of an initiation signal is as follows:

- a. All four main system pumps start with no delay, taking suction from the suppression chamber. The valves in the suction paths from the suppression chamber are kept open so that no automatic action is required to line up suction,

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except when the system is lined up in shutdown cooling. For the loop in shutdown cooling, the suction path requires manual action to realign, and the operating pump(s) need to be reset after trip.

- b. Valves used for other RHR operating modes (containment spray, RHR, etc.) are automatically positioned so that the water pumped from the suppression chamber is routed correctly
- c. When nuclear system pressure has dropped to a value at which the LPCI system pumps are capable of injecting water into the vessel, the LPCI system injection valves automatically open. If a shutdown cooling isolation has occurred, then the logic needs to be manually reset to permit the LPCI system injection valves, F015A/B to open.
- d. The LPCI loops then deliver water to the RPV until vessel water level is adequate to provide core cooling. Cooling water level is ensured since the pump is sealed in. The LPCI cannot be canceled for 5 minutes.

In the descriptions of the LPCI system instrumentation and control that follow, Figure 5.5-13 can be used to determine the physical location of sensors. Figure 7.3-9 can be used to determine the functional use of each sensor in the control circuitry for LPCI system components. Instrument characteristics and settings are given in Table 7.3-4. Actuation logic is shown in Figures 7.3-9 Sheets 1 and 2.

Additional information that provides a more detailed description of the differential pressure sensors used in the LPCI loop selection logic and additional clarification of the loop selection logic can be found in Subsection 7.3.1.2.4.10.

7.3.1.2.4.3 Bypasses and Interlocks

When an RHR loop is operating in the SDC mode, the loop is designed to isolate automatically on low reactor water level (i.e., Level 3) or high reactor pressure. If the system isolates on decreasing level before LPCI initiates (since Level 3 is higher than Level 1), the common SDC suction (E11F008 and 9) valves and the LPCI injection valves (E11F015A and B) in both divisions close, and pumps in the loop that is operating in the SDC mode trip on loss of suction path. The LPCI loop pumps that are lined up in standby mode are not affected. Under these circumstances, if LPCI injection is necessary, operator action would be necessary to align RHR to the LPCI mode. The LPCI injection valves' logic would have to be reset for both the loop in SDC and the loop in LPCI standby mode using the divisional push buttons in the control room. In the loop that was in SDC, the pump control logic would have to be reset before the pump could be started. The torus suction valves for the loop that was in SDC would have to be opened. All of these actions would be performed at the control room panels for the associated RHR loops.

As shown on Sheet 2 of Figure 7.3-9, there are three time-delay interlocks in the loop selection logic:

- a. A 0.5-sec delay to determine if either recirculation loop is shut down (in which case, the other loop is also shut down)

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- b. A 2.0-sec delay to allow momentum effects to settle and system parameters to stabilize
- c. A 0.5-sec delay while loop selection logic is being cycled.

Once the specific recirculation loop is selected for injection and the reactor pressure is below the RHR overpressure interlock setpoint, the RHR outboard and inboard valve circuits for that loop receive an OPEN permissive and a CLOSE block. The signal to the outboard valve is locked in for 5 minutes; this time is considered sufficient for the system to reflood the core to at least two-thirds of its height. Expiration of the 5-minute lock-in period does not initiate valve closure, but does give the operator the facility to throttle the flow.

The other loop, the loop not selected for LPCI injection, receives a CLOSE signal for 10 minutes when the loop selection is made. If the LPCI initiation signal remains, there is no capability in the logic to manually bypass the 10- and 5-minute delays in the loop selection logic. Once the loop is selected, the operator cannot change loops for 10 minutes.

To protect the main system pumps from overheating at low flow rates, a minimum flow bypass line is provided that routes water from the pump discharge to the suppression chamber. A motor-operated valve controls the flow in each bypass line. The minimum-flow bypass valve automatically opens on sensing low flow in the discharge line, and automatically closes when flow is above the low flow setting. Figure 5.5-13 shows the location of the flow sensors. The OPEN circuit contains a 15-sec delay permissive; this prevents loss of reactor vessel inventory to the suppression pool during shutdown cooling mode initiation.

The valves that divert water for containment cooling (F016, F021, F024, F027, F028) are signaled closed on receipt of an LPCI system initiation signal. These valves cannot be opened by manual action unless two conditions exist: the accident initiation signal indicating the need for containment cooling is present; and the RPV water level inside the core shroud is above the level equivalent to two-thirds the core height, which indicates that the pumps are not needed for the LPCI function.

Two differential-pressure transmitters are used to monitor water level inside the core shroud. Each is separately piped to the RPV. A keylock switch in the main control room allows manual override of the two-thirds core height and accident initiation signal permissives for the containment cooling valves.

The RHR heat exchanger bypass valve, F048, receives an OPEN and block CLOSE permissive from the LPCI initiation signal so maximum flow is available for injection. After 3 minutes, this permissive is blocked and the operator can manually close, throttle, or leave the valve in the open position.

7.3.1.2.4.4 Redundancy and Diversity

The LPCI system is redundant in that two separate loops are provided with pumps A and C feeding into loop A, and pumps B and D feeding into loop B. Loops A and B are tied together by means of a cross-header with a locked-open valve in the header. Initiation of the system is described in Subsection 7.3.1.2.4.

7.3.1.2.4.5 Actuated Devices

The functional control arrangement for the LPCI system pumps is shown in Figure 7.3-9. If ac power is available, all four LPCI system pumps start with no delay. Otherwise, they start as soon as the emergency power is available. The operator can manually control the pumps from the main control room. This permits him to use the pumps for other purposes such as containment cooling.

Two pressure-indicating transmitters are installed in each pump discharge line to verify that pumps are operating following an initiation signal. The pressure signal is used in the ADS to verify availability of low-pressure core cooling. The pressure transmitters are located upstream of the pump discharge check valves to prevent the operating pump discharge pressure from concealing a pump failure.

The main system pump motors are provided with overload protection. The overload relays maintain power on the motor as long as possible without harming the motor or jeopardizing the emergency power system.

All automatic valves used in the LPCI function are equipped with remote-manual test capability. The entire system can be operated from the main control room. Motor-operated valves have limit switches to turn off the motors when the fully open positions are reached. Torque switches are also provided to control valve motor forces when valves are closing. Thermal overload devices are used to trip motor-operated valves. Valves that have vessel and containment isolation requirements are described in Subsection 7.3.2.

The LPCI system pump suction valves from the suppression pool are normally open. To reposition the valves, a keylock switch must be turned in the main control room. On receipt of an LPCI initiation signal, certain reactor shutdown cooling system valves and the RHR test line valves are signaled to close, although they are normally closed, to ensure that the LPCI system pump discharge is correctly routed. Included in this set of valves are the valves that, if not closed, would permit the main system pumps to take suction from the reactor recirculation loops, a lineup used during normal shutdown cooling system operation.

A timer similar to that used in the LPCI system pump control circuitry cancels the LPCI open signal to the heat exchanger bypass valves after a 3-minute delay, which is time enough to permit satisfactory start of the LPCI system. The signal cancellation allows the operator to control the flow through the heat exchangers for other postaccident purposes. Canceling the open signal does not cause the bypass valves to close.

7.3.1.2.4.6 Separation

General

Refer to Subsection 7.3.1.2.1.6.

Specific

The LPCI system is a Division I and II system. Pumps A and C are in Division I, and pumps B and D are in Division II. Two separate logics located in separate panels are used. Logic for loop A (pumps A and C) is operated by the 260/130-V dc Division I battery and logic for loop B is operated by the 260/130-V dc Division II battery.

7.3.1.2.4.7 Testability

General

Refer to Subsection 7.3.1.2.1.7.

Specific

The LPCI system is provided with test jacks in each logic. The low reactor level or high drywell pressure "one-out-of-two taken twice" circuit can be completely tested by actuating only one instrument channel at a time. The other test jacks are used in the logic to facilitate testing as required. Insertion of the test plug in any jack actuates an annunciator in the main control room, indicating that LPCI is in test status.

7.3.1.2.4.8 Environmental Considerations

The only control components pertinent to LPCI system operation that are located inside the primary containment are those controlling the testable check valves on the injection lines. Other equipment, located outside the drywell, is selected in consideration of the normal and accident environments in which it must operate, as described in Table 3.11-3.

7.3.1.2.4.9 Operational Considerations

The LPCI system is a mode of the RHR system. The pumps, valves, piping, and other equipment used for the LPCI system are used for other modes of the RHR system. The LPCI mode is not required for normal operation.

7.3.1.2.4.10 Low-Pressure Coolant Injection Loop Selection Logic

Because the LPCI system injects water into the reactor through the discharge piping of one of the recirculation loops, it is necessary to make certain that the water is not injected into a broken recirculation loop. To satisfy this requirement, a break-detection system is provided to select the recirculation loop that is broken. This system then provides a signal that causes the LPCI water to be injected through the unbroken loop.

The location of the break in the recirculation system is determined by comparing the pressure of the two recirculation loops. The broken loop will indicate a lower pressure than the unbroken loop. The loop with the higher pressure is then used for LPCI injection or, if both pressures are the same, loop B is selected for injection. A diagram showing the relative physical location of the loop selection differential measurement can be found in Figure 5.5-2.

This logic system for break detection or loop selection is shown in Figure 7.3-9 and the details follow:

- a. The entire LPCI system is activated by either high drywell pressure or reactor low water level. Each of these signals is of the one-out-of-two-twice type
- b. The recirculation pump differential switches set up the network logic in the optimum arrangement depending on whether one pump or two pumps are operating. If only one pump is operating, the pressure difference due to the pump flow tends to mask the pressure difference due to the break. To avoid

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this, the loop selection time is delayed (0.5 sec) to determine if either recirculation pump is shut down and to allow proper selection of the unbroken loop. If only one pump is operating, the pump is tripped by the logic circuit

- c. The reactor-vessel-pressure permissive delays the one- pump-operating side of the logic network until the reactor pressure has dropped to less than about 900 psig. The delay is added to provide time for the recirculation pump coastdown
- d. After satisfaction of the reactor-pressure permissive mentioned above or if both recirculation pumps have indicated ΔP greater than the setpoint, the logic network is delayed about 2 sec to allow momentum effects to settle and system parameters to stabilize
- e. Finally, the loop selection is made. If loop A pressure is greater than that of loop B, then loop B is broken and injection will occur in loop A. If the pressure at loop A is not greater than that at loop B, the 0.5-sec timer will run out, causing loop B to be selected. The 0.5-sec time delay allows the loop selection logic to function. The ΔP is measured from each recirculation loop riser pipe to the corresponding riser pipe on the other recirculation loop. The taps are located as close to the reactor vessel as possible. This arrangement provides a one-out-of-two-twice logic

Loop selection differential pressure trip comparator set-points are adjusted to a value that gives the earliest valid indication of a break. The differential pressure comparator output contact is closed when the pressure in the recirculation pump A riser is approximately 1.0 psi higher than the pressure in the B recirculation pump riser

- f. Once the specific recirculation loop is selected for injection and the reactor pressure is below about 500 psig, the RHR outboard and inboard valve circuits for that loop receive an "open" permissive and a "close" block.

Because of the design of the logic circuitry, all cases except when a loop B break is detected cause injection through loop B. The interconnecting line between both RHR loop discharge lines permits total injection to either recirculation loop. In the accident mode, the core is flooded to an adequate height and the level is maintained by the LPCI operating alone with three of four pumps operating. The design basis requires 30,000 gpm (three of four pumps). Injecting into both loops simultaneously would produce some loss of inventory due to a postulated break in one of the loops.

A complete description of the sensors and trip units is the main subject of NEDO-21617, Analog Transmitter/Trip Unit System for Engineered Safeguard Sensor Trip Inputs.

7.3.1.3 Analysis of the Emergency Core Cooling System

7.3.1.3.1 Conformance To General Functional Requirements

In Chapters 6 and 15, the individual and combined capabilities of the ECCS are evaluated. Consideration of failure in plant instrument air and loss of cooling water to vital equipment is presented in Chapter 15. The safety design bases mentioned below are given in Subsection

7.1.2.1.3. The control equipment characteristics and trip settings are described in Subsection 7.3.1.2 and were considered in the analysis of ECCS performance. For the entire range of nuclear process system break sizes, the cooling systems are effective both in preventing fuel-cladding failure and in preventing more than a small fraction of the reactor core from reaching the temperature at which a gross release of fission products can occur. This conclusion is valid even with significant failures in individual cooling systems because of the overlapping capabilities of the ECCS. The instrumentation and control for the ECCS satisfies the requirements of safety design basis Item a.

Safety design basis Item b. requires that instrumentation for the ECCS respond to the potential inadequacy of core cooling regardless of the location of a breach in the nuclear system process barrier. The RPV low water level initiating function, which alone can actuate HPCI, LPCI, and core spray, meets this safety design basis because a breach in the nuclear system process barrier inside or outside the primary containment is sensed by the low water level trip channels. Because of the isolation responses of the CRVICS to a breach of the nuclear system outside the primary containment, the use of the RPV low-water signal is satisfactory as the only emergency cooling system initiating function that is completely independent of breach location.

The other major initiating function, primary containment high pressure, is provided as a diverse backup to water level to ensure isolation of all NSSS breaches inside the primary containment. This second initiating function is independent of the physical location of the breach within the drywell. The method used to initiate the ADS, which employs RPV low water level and primary containment high pressure, requires that the nuclear system breach be inside the drywell because of the required primary containment high pressure signal. For breaks outside primary containment, or for breaks inside primary containment which do not result in primary containment high pressure, the primary containment high pressure permissive is bypassed after a time delay following a low reactor water signal. This control arrangement is satisfactory in view of the automatic isolation of the RPV by the CRVICS for breaches outside the primary containment, and because the ADS is required only if the HPCI fails. This meets Safety design basis Item b.

An evaluation of ECCS controls shows that no operator action is required to initiate the correct responses of the ECCS.

The alarms and indications provided to the operator in the main control room allow interpretation of any situation requiring ECCS operations, and verify the response of each system. Manual controls are illustrated on functional control diagrams. The main control room operator can manually initiate every essential operation of the ECCS.

The degree to which safety is dependent on operator judgment and response has been appropriately limited by the design of the ECCS control equipment. Therefore, safety design bases Items c.1., c.2., and d. of Subsection 7.1.2.1.3 are satisfied.

The redundancy provided in the design of the control equipment for the ECCS is consistent with the redundancy of the cooling systems themselves. The arrangement of the initiating signals for the ECCS which come from common sensors is the same as that provided by the dual trip system arrangement of the RPS. No failure of a single initiating trip channel can prevent the start of the cooling systems.

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The numbers of control components provided in the design for individual cooling system components are consistent with the need for the controlled equipment. An evaluation of the control schemes for each ECCS component shows that no single control failure can prevent the combined cooling systems from providing the core with adequate cooling.

In performing this evaluation, the redundancy of components and cooling systems was considered. The functional control diagrams provided with the descriptions of cooling systems controls were used in assessing the functional effects of instrumentation failures. In the course of the evaluation, protection devices that can interrupt the planned operation of cooling system components were investigated for the results of their normal protective action as well as the effect of maloperation on core cooling effectiveness. The only protection devices that can act to interrupt planned ECCS operation are those that must act to prevent complete failure of the component or system. Examples of such devices are the HPCI turbine overspeed trip, HPCI steam line break isolation trip, pump trips on low suction pressure, and automatically controlled minimum flow bypass valves for pumps. In every case the action of a protective device cannot prevent other redundant cooling systems from providing adequate cooling to the core.

The minimum number of trip channels and sensors, as given in Tables 7.3-5 through 7.3-8, is sufficient to ensure correct functional performance of the ECCS. In determining the minimum number of trip channels needed to ensure functional performance, the use and redundancy of sensors in control circuitry and the redundancy of the controlled equipment in any individual cooling system were considered.

Where no redundancy of trip channels is available in the controls of a cooling system component required to function if the system is to operate, functional performance is not possible unless the trip channels are operable. Where two or more sensors of a monitored variable are arranged in parallel in control circuitry, inoperability of one parallel branch does not compromise performance of the system.

It should be noted that the various degrees of redundancy in control circuitry for the components of the ECCS reflect considerations for the integrated performance of the systems. The tables referenced in this subsection consider only the functional performance of each individual cooling system. To determine the proper state in which an inoperable sensor or trip channel should be placed, the functional effect of the channel and the proper action of the controlled equipment in a LOCA are considered. The condition given in the tables for inoperable sensors provides assurance that the essential functions of each individual ECCS are not degraded in a LOCA situation.

Because the control arrangement used for the ADS is designed to avoid spurious actuation, the information in Table 7.3-6 is worthy of special consideration. The relief valves are controlled by two trip systems, either one of which can initiate automatic depressurization. Each trip system has two trip logics, both of which must trip to initiate depressurization. Table 7.3-6 shows the minimum number of functional trip channels necessary for automatic depressurization.

The conditions indicated by Table 7.3-6 result in both trip systems always remaining capable of initiating automatic depressurization. If an inoperable sensor is in the tripped state or if a synthetic trip signal is inserted in the control circuitry, automatic depressurization can be initiated when the other initiation signals are received. The prohibition against

simultaneously inoperative RPV low water level and primary containment high pressure trip channels in any one trip logic is necessary to prevent situations where a trip logic is continuously in the tripped condition. If the trip logics containing the timers are affected, the planned delay in automatic depressurization is eliminated. The trip channel conditions indicated in Table 7.3-6 avoid these undesirable situations.

The LPCI system logic arrangement for the injection valves and recirculation loop valves warrants special consideration in the evaluation of conditions affecting LPCI system performance. The LPCI system sensing circuit for break detection and valve selection is arranged so that failure of a single device or circuit to function on demand does not prevent correct selection of a loop for injection. The system is effective in providing the proper amount of coolant flow into the undamaged recirculation loop under all combinations of recirculation loop pumping conditions, break sizes, and break location.

The conditions represented by Tables 7.3-5 through 7.3-8 are the result of a functional analysis of each individual ECCS. Because of the redundancy in methods of supplying cooling water to the fuel in LOCA situation, and because it is the cooling of the fuel that must be ensured in such a situation, the minimum trip channel conditions in these tables are in excess of those required operationally to ensure core cooling capability. Operational requirements for the ECCS will be determined from the reliability aspects of the integrated performances of the systems when the specific characteristics of core cooling system components are known.

The locations of controls where operation of ECCS components can be adjusted or interrupted have been surveyed. Controls are located in areas under the surveillance of operations personnel.

The environmental capabilities of instrumentation for the ECCS are discussed in the descriptions of the individual systems. Components located inside the primary containment that are essential to ECCS performance are designed to operate in the environment resulting from a LOCA.

7.3.1.3.2 Conformance To Specific Regulatory Requirements

Conformance to Regulatory Guide 1.22 is discussed in Subsections 7.3.1.2.1 through 7.3.1.2.4. Conformance to the requirements of General Design Criteria (GDC) 13, 35, 36, and 37 of 10 CFR 50, Appendix A, is discussed in Subsections 7.3.1.2.1 through 7.3.1.2.4. The requirements of 10 CFR 50, Appendix B, are met as described in Chapter 17.

7.3.1.3.3 Conformance To IEEE 279-1971

The provisions of the HPCI, ADS, core spray, and LPCI systems design that fulfill the general requirements of IEEE 279-1971 are given, for the most part, in the GE Topical Report, Compliance of Protection Systems to Industry Criteria; General Electric BWR Nuclear Steam Supply System, NEDO-10139, Subsections 3.4.3, 3.5.2, 3.2.2, and 3.3.2, respectively.

The HPCI, ADS, core spray, and LPCI interlock no control systems; therefore, no failure or combination of failures in the control systems can have any effect on the HPCI, ADS, core spray, or LPCI system.

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The ECCS equipment cabinets are identified by means of colored nameplates, in conformance with the 1971 identification requirements. Controls for each subsystem are grouped in one area of the control panel. Relays are located in separated panels for each division and subsystem.

7.3.1.3.4 Industry Standard IEEE 338-1971

The ECCS conforms to IEEE 338-1971.

7.3.1.3.5 Industry Standard IEEE 323-1971

Conformance to IEEE 323-1971 is described in NEDO-10698. See also Section 3.11.

7.3.1.3.6 Industry Standard IEEE 344-1971

Conformance to IEEE 344-1971 is described in NEDO-10678. See also Section 3.10.

7.3.2 Containment and Reactor Vessel Isolation Control System

7.3.2.1 Design-Basis Information

The design-basis information for the CRVICS, as required by Section 3 of IEEE 279-1971, is provided in Subsection 7.1.2.1.2 and is supplemented by the following:

- a. To limit the uncontrolled release of radioactive materials to the environs, the CRVICS shall initiate, with precision and reliability, timely isolation of penetrations through the primary containment structure, which could otherwise allow the uncontrolled release of radioactive materials whenever the values of monitored variables exceed preselected operational limits
- b. To provide assurance that important variables are monitored with a precision sufficient to fulfill safety design basis Item a., the CRVICS shall respond correctly to the sensed variables over the expected range of magnitudes and rates of change
- c. To provide assurance that important variables are monitored with a precision sufficient to fulfill safety design basis Item a., an adequate number of spatially independent sensors are provided for monitoring essential variables that have spatial dependence
- d. To provide assurance that conditions indicative of a gross failure of the nuclear system process barrier are detected with sufficient timeliness and precision to fulfill safety design basis Item a., CRVICS inputs shall be derived, to the extent feasible and practical, from variables that are true, direct measures of operational conditions
- e. The time required for closure of the main steam isolation valves (MSIVs) shall be short, so that the release of radioactive material and the loss of coolant as a result of a breach of a steam line outside the primary containment are minimal

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- f. The time required for closure of the MSIVs shall not be so short that inadvertent isolation of steam lines causes a more severe transient than that resulting from closure of the turbine stop valves coincident with failure of the turbine bypass system. This basis ensures that the MSIV closure speed is compatible with the ability of the reactor protection system (RPS) to protect the fuel and nuclear system process barrier
- g. To provide assurance that closure of Class A and Class B automatic isolation valves is initiated (Subsection 7.3.2.2.1) when required, with sufficient reliability to fulfill safety design basis Item a., the following safety design bases are specified for the systems controlling Class A and Class B automatic isolation valves:
 - 1. Any one failure, maintenance operation, calibration operation, or test to verify operational availability shall not impair the functional ability of the isolation control system to respond correctly to essential monitored variables, assuming no other single active failure
 - 2. The system shall be designed for a high probability that when any essential monitored variable exceeds the isolation setpoint, the event shall either result in automatic isolation or shall not impair the ability of the system to respond correctly as other monitored variables exceed their trip points
 - 3. Where a plant condition that requires isolation can be brought on by a failure or malfunction of a control or regulating system, and the same failure or malfunction prevents action by one or more isolation control system channels designed to provide protection against the unsafe condition, the remaining portions of the isolation control system shall meet the requirements of safety design bases Items a., b., c., and g.1.
 - 4. The power supplies for the CRVICS shall be arranged so that loss of one supply cannot prevent automatic isolation when required
 - 5. The system shall be designed so that, once initiated, automatic isolation action goes to completion. Return to normal operation after isolation action requires deliberate operator action
 - 6. There shall be sufficient electrical and physical separation between trip channels monitoring the same essential variable to prevent environmental factors, electrical faults, and physical events from impairing the ability of the system to respond correctly
 - 7. Earthquake ground motions shall not impair the ability of the CRVICS to initiate automatic isolation.
- h. To ensure that the timely isolation of main steam lines is accomplished, when required, with extraordinary reliability, at least one of the isolation valves in

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each of the steam lines does not rely on continuity of any variety of electrical power for the motive force to achieve closure

- i. To reduce the probability that the operational reliability and precision of the CRVICS are degraded by operator error, the following safety design bases are specified for Class A and Class B automatic isolation valves:
 1. Access to all trip settings, component calibration controls, test points, and other terminal points for equipment associated with essential monitored variables shall be under the physical control and supervision of the main control room operator
 2. The means for bypassing trip channels, trip logics, or system components is under the control of the main control room operator. If the ability to trip some essential part of the system has been bypassed, this fact shall be continuously indicated in the main control room.
- j. To provide the operator with a means independent of the automatic isolation functions to take action in the event of a failure of the nuclear system process barrier, it shall be possible for the main control room operator to manually initiate isolation of the primary containment and RPV
- k. The following bases are specified to provide the operator with the means to assess the condition of the CRVICS and to identify conditions indicative of a gross failure of the nuclear system process barrier
 1. The CRVICS is designed to provide the operator with information pertinent to the status of the system
 2. Means are provided for prompt identification of instrument channel and trip system responses.
- l. It shall be possible to check the operational availability of each trip channel and trip logic during reactor operation.

7.3.2.2 System Description

7.3.2.2.1 Identification

Class A isolation valves are in lines that communicate directly with the RPV and penetrate the primary containment. These lines generally have two isolation valves in series, one inside the primary containment and the other outside the primary containment.

Class B isolation valves are in lines that do not communicate directly with the RPV, but penetrate the primary containment free space. These lines have two isolation valves in series, both of which are outside the primary containment.

Class C isolation valves are in lines that penetrate the primary containment, but do not communicate directly with the RPV, the primary containment free space, or the environs (closed systems). These lines require one isolation valve located outside the primary

containment. The CRVICS includes the sensors, trip channels, switches, and remotely activated valve-closing mechanisms associated with the valves, which, when closed, effect isolation of the primary containment and/or the RPV.

It should be noted that the control systems for the Class A and Class B isolation valves, which close by automatic action pursuant to the safety design bases, are the main subjects of this section.

However, Class C remotely operated isolation valves are included because they add to the operator's ability to effect manual isolation. Testable check valves are also included because they provide the operator with the ability to ensure that the check valve disk can respond to reverse flow.

7.3.2.2.2 Power Supply

The two power supplies for the trip systems and trip logics are fed from the same two electrical buses that supply the RPS trip systems. Each of the buses has its own motor-generator set. Either bus can receive alternative power from a bus that can be energized by standby power. The buses cannot be simultaneously supplied from the same power source.

Isolation valves receive electrical power from buses that are reliable, in that power would be available from standby power sources except those isolation valves that are powered from the RPS. These valves automatically isolate on loss of offsite power. Power for the operation of the two valves in a line comes from different divisional sources. The MSIVs use ac, dc, and air or nitrogen pressure in the control scheme.

7.3.2.2.3 Physical Arrangement

Table 6.2-2 lists the lines that penetrate the primary containment and indicates the types and locations of the isolation valve(s) installed in each line. Lines that penetrate the primary containment and are in direct communication with the RPV generally have two Class A isolation valves, one inside the primary containment and one which is outside the primary containment. Lines that penetrate the primary containment and that communicate with the primary containment free space, but which do not communicate directly with the RPV, generally have two Class B isolation valves located outside the primary containment.

Class A and Class B automatic isolation valves are considered essential for protection against the gross release of radioactive material in the event of a breach in the nuclear system process barrier. Process lines that penetrate the primary containment but do not communicate directly with the RPV, the primary containment free space, or the environs, have at least one Class C isolation valve located outside the primary containment. This Class C valve may close either by process action (reverse flow) or by remote manual operation.

Table 6.2-2 presents information about all piping penetrations in the primary containment. Only the controls for the automatic isolation valves are discussed in this subsection. The valves that are the subject of this text are specially identified in the detailed descriptions that follow.

Power cables are run in conduits from appropriate electrical sources to the motor or solenoid involved in the operation of each isolation valve. Valve position switches are mounted on

the valve for which position is to be indicated. Switches are enclosed in cases to protect them from environmental conditions.

The control arrangement for the MSIVs includes pneumatic piping and an accumulator for each valve. Pressure and water-level sensors are mounted on instrument racks in either the reactor building or the turbine building. Cables from each sensor are routed in conduits and cable trays to the trip unit racks located in the reactor building. All signals transmitted to the main control room are electrical; no pipe from the nuclear system or the primary containment penetrates the main control room. Pipes used to transmit level information from the RPV to sensing instruments terminate inside the secondary containment (reactor building). The sensor cables and power supply cables are routed to cabinets in the control center where the logic arrangements of the system are formed.

To ensure continued protection against the uncontrolled release of radioactive material during and after earthquake ground motions, the control system required for automatic closure of Class A and Class B valves is seismic designed as Category I equipment as described in Subsection 7.1.2.1.2. This meets safety design basis Item g.7.

7.3.2.2.4 Logic

The basic logic arrangement is one in which an automatic isolation valve is controlled by redundant trip systems. In cases where many isolation valves close on the same signal, two trip systems control the entire group. Where just one or two valves must close in response to a special signal, two trip systems may be formed from the instruments provided to sense the special condition. Valves that respond to the signals from common trip systems are identified in the detailed description of isolation functions.

Each trip system has two trip logics, each of which receives input signals from at least one trip channel for each monitored variable. Thus, two trip channels are required for each essential monitored variable to provide independent inputs to the trip logics of one trip system. A total of four trip channels for each essential monitored variable is required for the trip logics of each trip system. The trip actuators associated with one trip logic provide inputs into the trip actuator logics for either one or two isolation trip systems. The two automatic trip logics associated with each trip system can produce a redundant isolation valve closure. For main steam line isolation valves only, both trip systems are used to actuate closure of inboard and outboard isolation valves. The logic is "one-out-of-two taken twice" arrangement for each variable.

The basic logic arrangement described above does not apply to Class C isolation valves and testable check valves. Exceptions to the basic logic arrangement are made in several instances for certain Class A and Class B isolation valves. The reasons for this are explained in Subsection 7.3.2.1.

7.3.2.2.5 Operation

During normal operation of the isolation control system, when isolation is not required, sensor and trip contacts essential to safety are closed; trip channels, trip logics, and trip actuators are normally energized. Whenever a trip channel sensor contact opens, its auxiliary relay deenergizes, causing contacts in the trip logic to open. The opening of contacts in the trip logic deenergizes its trip actuators. When deenergized, the trip actuators open contacts in

all the trip actuator logics for that trip system. If a trip then occurs in any of the trip logics of the other trip system, the trip actuator logics for the trip system are deenergized. With both trip systems tripped, appropriate contacts open or close in valve-control circuitry to actuate associated valve-closing mechanisms. Automatic isolation valves that are normally closed also receive the isolation signal.

The control system for each Class A isolation valve is designed to provide closure of the valve in time to prevent uncovering of the fuel as a result of a break in the pipeline which the valve isolates. The control systems for Class A and Class B isolation valves are designed to provide closure of the valves with sufficient rapidity to restrict the release of radioactive material to the environs below the guideline values of published regulations.

All automatic Class A and Class B valves and remotely operable Class C valves can be closed by manipulating switches in the main control room, thus providing the operator with a means independent of the automatic isolation functions to take action in the event of a failure of the nuclear system process barrier. This meets safety design basis Item j.

Once isolation is initiated, the valve continues to close, even if the condition that caused isolation is restored to normal. The operator must manually operate switches in the main control room to reopen a valve that has been automatically closed. Unless manual override features are provided in the manual control circuitry, the operator cannot reopen the valve until the conditions that initiated isolation have cleared. This is the equivalent of a manual reset and meets safety design basis Item g.5.

A trip of an isolation control system trip channel is annunciated in the main control room so that the operator is immediately informed of the condition. The response of isolation valves is indicated by "open-closed" lights. All motor-operated Class A and Class B isolation valves have a set of "open-closed" lights. The lights for each valve are located on the main control room panel at the manual control switches that control the valve. A second set of valve group displays that indicate status of the eighteen (18) containment isolation valve groups and individual valve status is available on the Integrated Plant Computer System (IPCS) as part of SPDS.

Inputs to annunciators, indicators, and the computer are arranged so that no malfunction of the annunciating, indicating, or computing equipment can functionally disable the system. Signals directly from the isolation control system sensors are not used as inputs to annunciating or data-logging equipment. Isolation is provided between the primary signal and the information output. The arrangement of indications pertinent to the status and response of the CRVICS satisfies safety design bases Items k.1. and k.2.

7.3.2.2.6 Isolation Valve Closing Devices

Table 6.2-2 itemizes the type of closing device provided for each isolation valve intended for use in automatic or remote manual isolation of the primary containment or RPV. In order that automatic Class A valves be fully closed in time to prevent the RPV water level from falling below the top of the active fuel as a result of a break of the line the valve isolates, the valve-closing mechanisms are designed to give the maximum closing times specified in the Technical Requirements Manual. In many cases a standard closing rate of 12 in./minute is adequate to meet isolation requirements. Using the standard rate, a 12-in. valve is closed in 60 sec. Because of the relatively long time required for fission products to reach the

containment atmosphere following a break in the nuclear system process barrier inside the primary containment, a 1-minute closure time is adequate for the automatic closing devices on most Class B isolation valves.

Motor-operators for Class A and Class B isolation valves are selected with capabilities suitable to the physical and environmental requirements of service. The required valve-closing rates were considered in selecting motor-operators. Appropriate torque and limit switches are used to ensure proper valve seating. Handwheels, which are automatically disengaged from the motor-operator when the motor is energized, are provided for local-manual operation.

Direct solenoid-operated isolation valves and solenoid air-pilot valves are chosen with electrical and mechanical characteristics that make them suitable for their services. Appropriate water-tight or weather-tight housing is used to ensure proper operation under accident conditions.

The pneumatic actuator used for testable check valves is designed to allow for the opening of the valve at near zero psi differential pressure across the valve. The actuator cannot close the valve against forward flow, or prevent the closing of the valve against reverse flow. Thus, the check valve will neither hinder forward fluid flow nor fail to stop reverse flow regardless of the condition of the actuator.

The MSIVs are spring-closing, pneumatic, piston-operated valves designed to close on loss of pneumatic pressure to the valve operator. This is a fail-safe design. The control arrangement is shown in Figure 7.3-10. Closure time for the valves is adjustable between 3 and 10 sec. Each valve is piloted by two, three-way, packless, direct-acting, solenoid-operated pilot valves, one of which is powered by ac and the other by dc. An accumulator is located close to each isolation valve to provide pneumatic pressure for valve closing in the event of failure of the normal gas supply system.

The valve pilot system and the pneumatic lines, as shown in Figure 7.3-11, are arranged so that when one or both solenoid-operated pilot valves are energized, normal gas supply provides pneumatic pressure to the gas-operated pilot valve to direct gas pressure to the main valve operator. This overcomes the closing force exerted by the spring and keeps the main valve open. When both pilots are deenergized, as would be the result when both trip systems trip, or when the manual switch is placed in the closed position, the path through which gas pressure acts is switched so that the opposite side of the valve operator is pressurized. This assists the spring in closing the valve. In the event of gas-supply failure, the loss of gas pressure causes the gas-operated pilot valve to move by spring force to the position resulting in main valve closure. Main valve closure is then effected by means of the gas stored in the accumulator and by the spring.

Gas pressure, acting alone, and the force exerted by the spring, acting alone, are each capable of independently closing the valve. The isolation valves inside the primary containment (inboard) are designed to close under either pneumatic pressure or spring force with the vented side of the piston operator at the containment peak accident pressure. The outboard valve is exactly the same design, although it will be subjected only to atmospheric pressures.

The accumulator volume was chosen to provide enough pressure to close the valve when the pneumatic supply to the accumulator has failed. The supply line to the accumulator is large

enough to make up pressure to the accumulator at a rate faster than the rate that the valve operation bleeds pressure from the accumulator during valve opening or closing.

A separate, single, solenoid-operated pilot valve with an independent test switch is included to allow manual testing of each isolation valve from the main control room. The testing arrangement is designed to give a slow closure of the isolation valve being tested so that rapid changes in steam flow and NSSS pressure are avoided. Slow closure of a valve during testing requires 50 to 60 sec.

7.3.2.2.7 Isolation Functions and Settings

The isolation trip settings of the CRVICS are listed in Table 7.3-9. The functions that initiate automatic isolation are itemized in Table 6.2-2 in terms of the lines that penetrate the primary containment. Table 6.2-2 includes all lines of concern for isolation purposes. Although this section is concerned with the electrical control systems that initiate isolation to prevent direct release of radioactive material from the primary containment or nuclear system process barrier, the additional information given in Table 6.2-2 can be used to assess the overall (electrical and mechanical) isolation effectiveness of each system having lines that penetrate the primary containment.

Isolation functions and trip settings used for the electrical control of isolation valves in fulfillment of the previously-stated safety design bases are discussed in the following subsection. The role each isolation function plays in initiating isolation of barrier valves or groups of valves is illustrated in the functional control diagrams of Figures 7.3-2, 7.3-12, 7.3-13, 7.3-14, and 7.4-1.

7.3.2.2.7.1 Reactor Vessel Low Water Level

A low-water level in the RPV could indicate that reactor coolant is being lost through a breach in the nuclear system process barrier and that the core is in danger of becoming overheated as the reactor coolant inventory diminishes. Reactor vessel low-water level initiates closure of various Class A valves and Class B valves. The closure of Class A valves is intended to either isolate a breach in any of the lines in which valves are closed, or to conserve reactor coolant by closing off process lines. The closure of Class B valves is intended to prevent the escape of radioactive materials from the primary containment through process lines that are in communication with the primary containment free space.

Three RPV low water-level isolation trip settings are used to completely isolate the RPV and the primary containment. The level signals are defined as follows and are shown in Figure 7.3-12:

- a. Level 3 (L3) is the highest of the three and also initiates the level scram and isolates the RHR system
- b. Level 2 (L2) is the initiation level for the reactor core isolation cooling (RCIC) and HPCI systems and is selected to be less than the volume resulting from a void collapse occurring in the event of a scram from full power. Level 2 also closes certain containment isolation valves

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- c. Level 1 (L1) is selected far enough above the top of the active fuel and is selected based on the time required for the RHR and core spray systems to function in the event of a large break. Level 1 also isolates the MSIVs.

Isolation of the following lines occurs when the level reaches L3, which is the highest or most conservative level (Table 6.2-2, signal C):

- a. Drywell equipment drain sump discharge
- b. Drywell floor drain sump discharge
- c. RHR shutdown cooling
- d. Traversing in-core probe (TIP) system withdrawal.

The second level (L2) isolates the majority of the nuclear pressure boundary lines and the primary and secondary containment paths. This is also the level that starts the HPCI and RCIC systems, and it has been selected to be lower than the level change resulting from a void collapse following a scram from full power. Specifically, isolation of the following lines is initiated on Level 2 (Table 6.2-2, signal B):

- a. Reactor sample lines
- b. Reactor water cleanup
- c. Drywell air and nitrogen inlet
- d. Suppression chamber exhaust
- e. Suppression chamber air and nitrogen inlet
- f. Drywell exhaust
- g. Drywell pressure control
- h. Suppression chamber pressure control
- i. Purge to standby gas treatment
- j. Control center heating, ventilation, and air conditioning (HVAC)
- k. Reactor building ventilation
- l. Recirculation pump seal purge
- m. Torus water management
- n. Primary containment radiation monitoring.

The final isolation level is Level 1 (L1). This level setting provides automatic isolation for the following lines, which penetrate the primary containment, if they are open (Table 6.2-2, signal A):

- a. RHR containment spray
- b. RHR test line
- c. Core spray test line
- d. Suppression chamber spray

- e. Main steam
- f. Main steam line drains.

7.3.2.2.7.2 Main Steam Line High Radiation

High radiation in the vicinity of the main steam lines could indicate a gross release of fission products from the fuel. High radiation near the main steam lines initiates isolation of the following pipelines (Table 6.2-2, signal D):

- a. All main steam lines
- b. Main steam line drain
- c. Reactor water sample line.

The high-radiation trip setting is selected high enough above background radiation levels so that spurious isolation is avoided, yet low enough to promptly detect a gross release of fission products from the fuel. Further information regarding high-radiation setpoint is available in Section 11.4.

7.3.2.2.7.3 Main Steam Line Space High Temperature

High temperature in the space in which the main steam lines are located outside the primary containment could indicate a breach in a main steam line. The automatic closure of various Class A valves prevents both the excessive loss of reactor coolant and the release of significant amounts of radioactive material from the nuclear system process barrier. When high temperatures occur in the main steam line space, all four main steam lines and the main steam line drain are isolated.

The main steam line space high-temperature trip is set far enough above the temperature expected during operations at rated power so that spurious isolation is avoided, yet low enough to provide early indication of a steam line break.

7.3.2.2.7.4 Main Steam Line High Flow

Main steam line high flow could indicate a break in a main steam line. The automatic closure of various Class A valves prevents the excessive loss of reactor coolant and the release of significant amounts of radioactive material from the nuclear system process barrier. On detection of the main steam line high flow, all four main steam lines and the main steam line drain are isolated.

The main steam line high-flow-trip setting was selected high enough to permit the isolation of one main steam line for the test at rated power without causing an automatic isolation of the rest of the steam lines, yet low enough to permit early detection of a steam line break.

7.3.2.2.7.5 Low Steam Pressure at Turbine Inlet

Low steam pressure at the turbine inlet while the reactor is operating could indicate a malfunction of the nuclear system pressure regulator, at which time the turbine control valves or turbine bypass valves open fully. This action causes rapid depressurization of the nuclear system. From part-load operating conditions, the rate of decrease of nuclear system

saturation temperature could exceed the allowable rate of change of vessel temperature. A rapid depressurization of the RPV while the reactor is near full power could result in undesirable differential pressure across the channels around some fuel bundles of sufficient magnitude to cause mechanical deformation of channel walls. Such depressurizations, without preventive action, could require thorough vessel analysis or core inspection prior to returning the reactor to power operation. To avoid the time-consuming requirements following a rapid depressurization, the steam pressure at the turbine inlet is monitored. On falling below a preselected value with the reactor in the RUN mode, isolation of all four main steam lines and the main steam drain line is initiated.

The low-steam-pressure isolation setting was selected far enough below normal turbine inlet pressures so that spurious isolation is avoided, yet high enough to provide timely detection of a pressure regulator malfunction. Although this isolation function is not required to satisfy any of the safety design bases for this system, this discussion is included here to make the listing of isolation functions complete.

7.3.2.2.7.6 Primary Containment (Drywell) High Pressure

High pressure in the drywell could indicate a breach of the nuclear system process barrier inside the drywell. The automatic closure of various Class B valves prevents the release of significant amounts of radioactive material from the primary containment. On detection of a high drywell pressure, the following pipelines are isolated:

- a. Drywell equipment drain discharge
- b. Drywell floor drain discharge
- c. TIP tubes
- d. Drywell air and nitrogen inlet
- e. Suppression chamber exhaust valves
- f. Suppression chamber air and nitrogen inlet
- g. Drywell exhaust
- h. Drywell pressure control
- i. Suppression chamber pressure control
- j. Purge to standby gas treatment
- k. Control center HVAC recirculation mode
- l. Reactor building ventilation system isolation
- m. Torus water management
- n. Primary containment radiation monitoring.
- o. Reactor recirculation pumps seal purge supply lines
- p. EECW Division 1 and 2 drywell cooling supply lines (Note: isolation signal from ECCS logic, not from RPS logic)

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The primary containment high pressure isolation setting was selected to be as low as possible without inducing spurious isolation trips.

7.3.2.2.7.7 Reactor Core Isolation Cooling Turbine Steam Line Space High Temperature

High temperature in the vicinity of the RCIC turbine could indicate a break in the RCIC steam line. The automatic closure of certain Class A valves prevents the excessive loss of radioactive material from the nuclear system process barrier. When high temperature occurs in the RCIC area, the RCIC turbine steam line is isolated. The high-temperature isolation setting was selected far enough above anticipated normal RCIC system operational levels so that spurious operation is avoided, yet low enough to provide timely detection of an RCIC turbine steam line break.

7.3.2.2.7.8 Reactor Core Isolation Cooling Turbine High Steam Flow

Reactor core isolation cooling turbine high steam flow could indicate a break in the RCIC turbine steam line. The automatic closure of certain Class A valves prevents the excessive loss of reactor coolant and the release of significant amounts of radioactive materials from the nuclear system process barrier. The RCIC turbine high-steam-flow trip setting was selected high enough to avoid spurious isolation, yet low enough to provide timely detection of a RCIC turbine steam line break. An electrical time-delay circuit prevents spurious isolations on the turbine startup transient.

The logic arrangement used for this function is shown in Figure 7.4-1, and is an exception to the usual logic requirements since high steam flow is the second method of detecting a RCIC turbine steam line break.

7.3.2.2.7.9 Reactor Core Isolation Cooling Turbine Steam Line Low Pressure

The RCIC turbine steam line low pressure is used to automatically close the two isolation valves in the RCIC turbine steam line so that steam and radioactive gases do not escape from the RCIC turbine shaft seals into the reactor building after steam pressure has decreased to such a low value that the turbine cannot be operated. The isolation setpoint is chosen at a pressure below that at which the RCIC turbine can operate effectively.

7.3.2.2.7.10 High-Pressure Coolant Injection Turbine Steam Line Space High Temperature

High temperature in the vicinity of the HPCI turbine could indicate a break in the HPCI turbine steam line. The automatic closure of certain Class A valves prevents the excessive loss of reactor coolant and the release of significant amounts of radioactive material from the nuclear system process barrier. When high temperature occurs in the HPCI turbine area, the HPCI turbine steam supply line is isolated. The high-temperature isolation setting was selected far enough above anticipated normal HPCI system operational levels so that spurious isolation is avoided, yet low enough to provide timely detection of an HPCI turbine steam line break.

7.3.2.2.7.11 High-Pressure Coolant Injection Turbine High Steam Flow

The HPCI turbine high steam flow could indicate a break in the HPCI turbine steam line. The automatic closure of certain Class A valves prevents the excessive loss of reactor coolant and the release of significant amounts of radioactive materials from the nuclear system process barrier.

On detection of HPCI turbine high steam flow, the HPCI turbine steam line is isolated. The high-steam-flow trip setting was selected high enough to avoid spurious isolation, yet low enough to provide timely detection of an HPCI turbine steam line break. An electrical time-delay circuit prevents spurious isolations on the turbine startup transient.

The logic arrangement used for this function is shown in Figure 7.3-2, and is an exception to the usual logic requirement since high steam flow is the second method of detecting an HPCI turbine steam line break.

7.3.2.2.7.12 High-Pressure Coolant Injection Turbine Steam Line Low Pressure

The HPCI turbine steam line low pressure is used to automatically close the two isolation valves in the HPCI turbine steam line so that steam and radioactive gases do not escape from the HPCI turbine shaft seals into the reactor building after steam pressure has decreased to such a low value that the turbine cannot be operated. The isolation setpoint is chosen at a pressure below that at which the HPCI turbine can operate.

7.3.2.2.7.13 Reactor Building Ventilation Exhaust High Radiation

High radiation prior to the reactor building ventilation exhaust fans could indicate a breach of the nuclear system process barrier inside the primary containment, which would result in increased airborne radioactivity levels in the primary containment exhaust to the secondary containment. The automatic closure of certain Class B valves acts to close off release routes for radioactive material from the primary containment into the secondary containment (reactor building). Reactor building ventilation exhaust high radiation initiates isolation of the following pipelines:

- a. Drywell air and nitrogen inlet
- b. Suppression chamber exhaust
- c. Suppression chamber air and nitrogen inlet
- d. Drywell exhaust
- e. Drywell pressure control
- f. Suppression chamber pressure control
- g. Purge to standby gas treatment
- h. Reactor building supply and exhaust (vent)
- i. Control center normal air intake and exhaust.

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The high-radiation trip setting selected is far enough above background radiation levels to avoid spurious isolation, yet low enough to provide timely detection of nuclear system process barrier leaks inside the primary containment. Because the primary containment high pressure isolation function and the RPV low-water-level isolation function are adequate in effecting appropriate isolation of the above pipelines for gross breaks, the reactor building ventilation exhaust high-radiation isolation function is provided as a third redundant method of detecting breaks in the nuclear system process barrier significant enough to require automatic isolation.

7.3.2.2.8 Instrumentation

Sensors providing inputs to the CRVICS are not used for the automatic control of the process system. Thus, the functional controls of the protection and process systems are separated. Trip channels are physically and electrically separated to reduce the probability that a single physical event could prevent isolation. Trip channels for one monitored variable that are grouped near each other provide inputs to different isolation trip systems. The sensors are used functionally in the isolation control system, as illustrated in Figures 7.3-2, 7.3-9, 7.3-12, 7.3-13, and 7.4-1. Table 7.3-9 lists instrument characteristics. The sensors are described in the following paragraphs:

- a. Reactor vessel low-water-level signals are initiated from eight level transmitters (differential pressure transmitters) that sense the difference between the pressure due to the constant reference column of water and the pressure due to the actual water level in the vessel.

A backfill system is installed on each level instrument reference leg. The system provides a metered flow of water from the control rod drive system to each leg. The flow is low enough to not affect the performance of the instrumentation. The backfill is designed to prevent the accumulation of dissolved noncondensable gases in the reference legs.

Four narrow-range transmitters provide an input to individual trip units that provide an isolation signal when the reactor water level drops to the first (highest) water level (L3) trip settings. Each of the four wide-range transmitters provides signals to two trip units. One trip unit provides the isolation signal for the second low water level (L2) trip setting. The other trip unit provides the isolation signal for the third and lowest (L1) trip setting.

Logic channel trips are arranged in one-out-of-two-twice logic. Channels A or C and B or D are required to initiate isolation for both inboard and outboard MSIVs. Two of the four transmitters for each trip level are connected to one pair of taps (A and B). The other two transmitters (C and D) are connected to taps that are 180° around the RPV from the first pair. This physical separation ensures that no single physical event can prevent isolation if it were required. Cables from the transmitters are routed to the trip units in the auxiliary building.

- b. Main steam line radiation is monitored by four radiation monitors, which are described in Subsection 11.4.3.8.2.3

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- c. High temperature in the vicinity of the main steam lines is detected by 16 resistance temperature detectors (RTDs) located along the main steam lines between the drywell wall and the reactor building steam tunnel pressure relief doors. Eight additional RTDs sense high temperature in the turbine building steam tunnel. The detectors are located or shielded so that they are sensitive to air temperature and not to the radiated heat from hot equipment. The main steam line space temperature detection system is designed to detect leaks of from 1 percent to 10 percent of rated steam flow
- d. High flow in each main steam line is sensed by four differential pressure transmitters that sense the pressure difference across the flow restrictor in that line. The logic is arranged as two trip systems with two trip logics in each system. Any two trip logics can trip the isolation valve. Each trip logic receives an input from a high-steam-flow trip channel for each steam line
- e. Main steam line low pressure is sensed by four pressure transmitters that sense pressure downstream of the outboard MSIVs. The sensing point is located at the header that connects the four steam lines upstream of the turbine stop valves. The logic is arranged as two trip systems with two trip logics per system. Any two trip logics associated with each trip system can trip the isolation valves
- f. Primary containment pressure is monitored by four pressure transmitters that are mounted on instrument racks outside the drywell. Pipes that terminate in the reactor building connect the transmitters to the drywell interior. Cables are routed from the transmitter to the main control room. The transmitters are grouped in pairs, physically separated, and electrically connected to the isolation control system so that no single event prevents isolation due to primary containment high pressure
- g. High temperature in the vicinity of the RCIC turbine is sensed by two ambient and two differential temperature measurements. Only the ambient temperature sensors can initiate RCIC isolation
- h. High flow in the RCIC turbine steam line is sensed by two differential pressure transmitters, which monitor the differential pressure across a mechanical flow element installed in the RCIC turbine steam supply pipeline. The tripping of either trip channel initiates isolation of the RCIC turbine steam line. This is an exception to the usual sensor requirement
- i. Low pressure in the RCIC turbine steam line is sensed by four pressure transmitters from the RCIC turbine steam line upstream of the isolation valves. The transmitters are arranged as two trip systems, either of which must trip to initiate isolation of the RCIC turbine steam line. Each trip system receives inputs from two pressure transmitters, both of which must trip to trip the system
- j. High temperature in the vicinity of the HPCI turbine is sensed by two ambient and two differential temperature measurements. Only the ambient temperature sensors can initiate isolation

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- k. High flow in the HPCI turbine steam line is sensed by two differential pressure transmitters which monitor the differential pressure across a mechanical flow element installed in the HPCI turbine steam line. The tripping of either transmitter initiates isolation of the HPCI turbine steam line
- l. Low pressure in the HPCI turbine steam line is sensed by four pressure transmitters from the HPCI turbine steam line upstream of the isolation valves. The transmitters are arranged as two trip systems, either of which can initiate isolation of the HPCI turbine steam line. Each trip system receives inputs from two pressure transmitters, both of which must trip to trip the trip system
- m. Reactor building ventilation exhaust radiation is monitored by two independent redundant monitors. Each monitoring trip channel provides the isolation function as described in Subsections 7.3.2.2.7.13 and 11.4.3.8.2.4. The primary containment high pressure isolation function and the RPV low water level isolation function are adequate in effecting the isolation of the pipelines that could release radioactivity due to breach of the nuclear system process barrier inside the primary containment. The reactor building ventilation exhaust radiation is provided as a third redundant method of detecting breaks in the nuclear system process barrier (significant enough to require automatic isolation).

In addition to the above, the fuel pool ventilation exhaust radiation monitoring system is provided to detect a high radiation level in the ductwork that could be due to fission gases from a refueling accident. Four fuel pool ventilation exhaust detectors in a redundant "one out of two" logic provide the isolation function as described in Subsection 11.4.3.8.2.11.

- n. High temperature in the spaces occupied by the reactor shutdown cooling system piping outside the primary containment is sensed by temperature switches that activate alarms only, indicating possible pipe breaks. Automatic isolation on high temperature is not required since the RPV low-water-level isolation function is adequate in preventing the release of significant amounts of radioactive material in the event that either of these two systems suffers a breach.

Sensor trip channel and trip logic relays are high reliability relays equal to type-HFA relays made by GE. Table 7.3-10 lists the minimum numbers of trip channels needed to ensure that the isolation control system retains its functional capabilities.

7.3.2.2.9 Environmental Capabilities

Special consideration has been given to isolation requirements during a LOCA inside the drywell. Components of the CRVICS that are located inside the primary containment and that must operate during a LOCA are the cables, control mechanism, and valve operators or isolation valves inside the drywell. These isolation components are required to be functional in a LOCA environment.

Electrical cables for isolation valves in the same lines are routed separately. Motor-operators for valves inside the primary containment are of the totally enclosed type; those outside the

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primary containment have weatherproof type enclosures. Solenoid valves, whether used for direct valve isolation or as an air or gas pilot, are provided with watertight enclosures. All cables and operators are capable of operation in the most unfavorable ambient conditions anticipated for design-basis accident (DBA) conditions. Temperature, pressure, humidity, and radiation are considered in the selection of equipment for the system. Cables used in high radiation areas have radiation-resistant insulation. Shielded cables are used whenever necessary to eliminate interference from magnetic fields. Electrical cables are selected with insulation designed for this service.

Closing mechanisms and valve operators are considered satisfactory for use in the isolation control system only after completion of environmental testing under LOCA conditions or after submission of evidence from the manufacturer describing the results of suitable prior tests.

Verification that the isolation equipment has been designed, built, and installed in conformance to the specified criteria is accomplished through the following series of tests in the vendor's shop or after installation at the plant before startup, during startup, and thereafter, where appropriate, during the service life of the equipment:

- a. Material qualification tests
- b. Weld qualification tests
- c. Metallurgical tests
- d. Hydrostatic tests
- e. Leakage tests
- f. Closing time tests
- g. Preoperational tests
- h. Startup tests
- i. Periodic tests
- j. Verification of type of materials used for insulation
- k. Environmental testing of electrical equipment under simulated accident conditions.

Control is also exercised through review of equipment design during bid review and by approval of vendor's drawings during the fabrication stage. Purchase specifications require extensive control of materials and of the fabrication procedure.

7.3.2.3 Analysis

The CRVICS is described in Subsection 7.3.2.2. The safety design bases and specific regulatory requirements of this system are stated in Subsection 7.3.2.1. This analysis shows compliance with these requirements.

7.3.2.3.1 Safety Evaluation Analysis

The CRVICS, in conjunction with other safety systems, is designed to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and nuclear system process barriers. It is the objective of Chapter 15 to identify and evaluate postulated events resulting in gross failure of the fuel barrier and the nuclear system process barrier. The consequences of such gross failures are described and evaluated in that section.

Design procedure has been to select tentative isolation trip settings that are far enough above or below normal operating levels that spurious isolation and operating inconvenience are avoided. It is then verified by analysis that the release of radioactive material following postulated gross failures of the fuel and nuclear system process barrier is kept within acceptable bounds. Trip-setting selection is based on operating experience and constrained by the safety design and the safety analyses.

Chapter 15 shows that the actions initiated by the CRVICS, in conjunction with other safety systems, are sufficient to prevent releases of radioactive material from exceeding the guide values of published regulations. Because the actions of the system are effective in restricting the uncontrolled release of radioactive materials under accident situations, the CRVICS meets the precision and timeliness requirements of safety design basis Item a.

The CRVICS meets the precision and timeliness requirements of safety design basis Item a. using instruments with the characteristics described in Table 7.3-9. Therefore, it is concluded that safety design basis Item b. is met.

Temperatures in the spaces occupied by various steam lines outside the primary containment are the only essential variables of significant spatial dependence that provide inputs to the CRVICS. The large number of temperature sensors and their dispersed arrangement near the steam lines requiring this type of break protection provide assurance that a significant break will be detected rapidly and accurately. The number of sensors provided for steam line break detection satisfies safety design basis Item c.

Because the CRVICS meets the timeliness and precision requirements of safety design basis Item a. by monitoring variables that are true, direct measures of operational conditions, it is concluded that safety design basis Item d. is satisfied.

Subsection 15.6.4 evaluates a gross breach in a main steam line outside the primary containment during operation at rated power. The evaluation shows that the main steam lines are automatically isolated in time to prevent both a release of radioactive material in excess of the guideline values of published regulations, and to prevent the loss of coolant from being great enough to allow uncovering of the core. The time required for automatic closure of the MSIVs meets the requirements of safety design basis Item e. The shortest closure time of which the MSIVs are capable is 3 sec. The transient resulting from a simultaneous closure of all MSIVs in 3 sec during reactor operation at rated power is considerably less severe than the transient resulting from inadvertent closure of the turbine stop valves (which occurs in a small fraction of 1 sec) coincident with failure of the turbine bypass system. This conclusion is substantiated in Subsection 15.2.3. This meets safety design basis Item f.

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The safety design bases Items g., h., and i. must be fulfilled for the CRVICS to meet the design reliability requirements of safety design basis Item a. It has already been shown that safety design bases Items g.5. and g.7. have been met. The remainder of the reliability requirement is met by a combination of logic arrangement, sensor redundancy, wiring scheme, physical isolation, power supply arrangement, and environmental capabilities. These subjects are discussed in the following paragraphs.

Because essential variables are monitored by four trip channels arranged for physical and electrical independence, and because a dual trip system arrangement is used to initiate closure of automatic isolation valves, no single failure, maintenance operation, calibration operation, or test can prevent the system from initiating valve closure. An analysis of the isolation control system shows that the system does not fail to respond to essential variables as a result of single electrical failures such as short circuits, ground, and open circuits. A single trip system trip is the result of these failures. Isolation is initiated on a trip of the remaining trip system. For some of the exceptions to the usual logic arrangement, a single failure could result in inadvertent isolation of a pipeline. With respect to the release of radioactive material from the nuclear system process barrier, such inadvertent valve closures are in the safe direction and do not pose any safety problems. This meets safety design basis Item g.1.

The redundancy of trip channels for all essential variables provides a high probability that whenever an essential variable exceeds the isolation setting, the system initiates isolation. In the unlikely event that all trip channels for one essential variable in one trip system fail in such a way that a system trip does not occur, the system could still respond properly as other monitored variables exceed their isolation settings. This meets safety design basis Item g.2. The sensors, circuitry, and logic channels used in the CRVICS are not used in the control of any process system. Thus, malfunction and failures in the controls of process systems have no direct effect on the isolation control system. This meets safety design basis Item g.3.

The various power supplies used for the isolation control system logic circuitry and for valve operation provide assurance that the required isolation can be effected in spite of power failures. If ac power for valves inside the primary containment is lost, dc power is available for operation of valves outside the primary containment. The main steam isolation valve control arrangement is resistant to both ac and dc power failures. Because both solenoid-operated pilot valves must be deenergized, loss of a single power supply neither causes inadvertent isolation nor prevents isolation if required.

The logic circuitry for each channel is powered from the separate sources available from the RPS buses. A loss of power here results in a single trip system trip. In no case does a loss of a single power supply prevent isolation. This meets safety design basis Item g.4.

The isolation control system can operate under the most unfavorable environmental condition associated with normal operation. The discussion of the effects of rapid nuclear system depressurization on level measurement given in Subsection 7.2.1.1.3.1 is equally applicable to the RPV low-water-level transmitters used in the CRVICS. The temperature, pressure, differential pressure, and level transmitters, cables, and valve-closing mechanisms used were selected with ratings that make them suitable for use in the environment in which they must operate.

The special considerations made for the environmental conditions resulting from a LOCA inside the drywell are adequate to ensure operability of essential isolation components located inside the drywell.

The wall of the primary containment effectively separates adverse environmental conditions that might otherwise affect both isolation valves in a line. The location of isolation valves on either side of the wall decouples the effects of environmental factors with respect to the ability to isolate any given line. The previously discussed electrical isolation of control circuitry prevents failures in one part of the control system from propagating to another part. Electrical transients have no significant effect on the functioning of the isolation control system. Therefore, it is concluded that safety design basis Item g.6. is satisfied.

The design of the MSIVs meets the requirement of safety design basis Item h.1. in that the motive force for closing each MSIV is derived from both a source of pneumatic or gas pressure, and the energy is stored in a spring. Either energy source is capable of independently closing the valve. None of the valves rely on continuity of any sort of electrical power to achieve closure in response to essential safety signals. Total loss of the power used to control the valves would result in closure. This meets safety design basis Item h.2.

Calibration and test controls for pressure and level transmitters are located on the transmitters themselves. These transmitters are located in the turbine building and reactor building. To gain access to the setting controls on each transmitter, a cover plate, access plug, or sealing device must be removed by operations personnel before any adjustment in trip settings can be effected. The location of calibration and test controls in areas under the control of supervision or of the main control room operator reduces the probability that operational reliability will be degraded by operator error. This meets safety design basis Item i.1.

Because no manual bypasses (except for those used during a BDBEE) are provided in the isolation control system, safety design basis Item i.2. is met. Because safety design bases Items g., h., and i. have been met, it can be concluded that the CRVICS satisfies the reliability requirement of item safety design basis a. That the system satisfied safety design bases Items j., k.1., and k.2. was shown in the description of the system. The following subsection, covering inspection and testing of the system, demonstrates that safety design basis Item 1. is satisfied.

7.3.2.3.2 Inspection and Testing

All parts of the primary containment isolation control system are testable during reactor operation. Isolation valves can be tested to ensure that they are capable of closing by operating manual switches in the main control room and observing the position lights and any associated process effects. Testable check valves are arranged to verify that the valve disk is free to open and close. The trip channel and trip system responses can be functionally tested by applying test signals to each trip channel and observing the trip system response.

Functional testing and calibration schedules developed using available failure rate data, reliability analyses, and operating experience are presented in the Technical Specifications. The schedules represent an optimization of CRVICS reliability by considering, on one hand, the failure probabilities of individual components, and, on the other hand, the reliability

effects during individual component testing on the portions of the system not undergoing tests.

The isolation actuation system instrumentation response times are shown in Technical Requirements Manual Volume I Table 3.3.6.1-1, which is referenced in UFSAR Table 7.3-12. Response time testing is required by the Technical Specifications. Technical Specification Table 3.3.2-3 was deleted from the Technical Specifications and added to the UFSAR as Table 7.3-12 (TRM Table 3.3.6.1-1) in agreement with NRC Generic Letter 93-08 and Amendment Number 100 to the Technical Specifications. The response times information of UFSAR Table 7.3-12 was then relocated to the Technical Requirements Manual Volume I.

The response time testing for the trip functions associated with the diesel start and sequencing of loads is eliminated in agreement with NRC Generic Letter 93-05 and Amendment Number 99 to the Technical Specifications.

The periodic sensors response time testing for the reactor vessel low water level-level 1 and the main steam line flow-high has been eliminated. The BWROG Report NEDO-32291A provides the required analyses as briefly described in 7.2.1.1.3.8.1.

7.3.2.3.3 Specific Regulatory Requirements Conformance

7.3.2.3.3.1 IEEE 279-1971

Conformance to IEEE 279-1971 is demonstrated in Topical Report NEDO-10139, Paragraph 4.2. The 21 subparagraphs of 4.2 cover the 21 subparagraphs of IEEE 279-1968. The following discussion is addressed to IEEE 279-1971, subparagraphs 4.7, 4.17, and 4.22, which are different from those in IEEE 279-1968:

- a. Paragraph 4.7.1: Classification of Equipment - There is no control function in the system. It is strictly a protection system
- b. Paragraph 4.7.2: Isolation Devices - Since there is no control function, no isolation devices are required
- c. Paragraph 4.7.3: Single Random Failure - No single random failure of a control system can prevent proper action of the isolation system channel designed to protect against the condition
- d. Paragraph 4.7.4 - Analysis of 4.7.3 applies directly
- e. Paragraph 4.17: Manual Initiation - Manual initiation controls are provided and separated in such a manner as to prevent a single failure from inhibiting an isolation.

The separation of devices is maintained in both the manual and automatic portions of the system so that no single failure in either the manual or automatic portions can prevent an isolation by either manual or automatic means. There are no areas of the system that are common to manual and automatic functions

- f. Paragraph 4.2: Identification - Panels and racks that house isolation system equipment are identified with a distinctive color marker plate listing the system

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name and the designation of the particular redundant portion of the system. Instrument cables are identified in accordance with IEEE 279-1971.

In addition, NEDO-10139 demonstrated conformance with IEEE 279-1971, Paragraph 4.14, by the position that there are no manual controls that bypass the containment isolation control function. This conformance with IEEE 279-1971 is maintained but is modified to note the use of keylock switches added to allow operation of the Hardened Containment Vent System in support of the plant's response to a BDBEE in accordance with NRC Order EA-13-109.

7.3.2.3.3.2 Industry Standard IEEE 323-1971

Compliance with this standard is discussed in Section 3.11 and NEDO-10698.

7.3.2.3.3.3 IEEE 338-1971

The system is testable during reactor operation. The tests that may be performed will cover the sensors through the final actuators, demonstrate independence of channels, and bare any credible failures while not negating any isolation.

7.3.2.3.3.4 Industry Standard IEEE 344-1971

Compliance with this standard is discussed in Section 3.10 and NEDO-10678.

7.3.2.3.3.5 Regulatory Guide 1.22

Regulatory Guide 1.22 requires periodic testing of protection system actuation functions. The MSIVs and associated logic and sensor devices may be tested from the sensor device to one of the two solenoids required for valve closure. The valve may be exercised closed with either a slow-acting test solenoid or the normal closing solenoid to verify that there are no obstructions to the valve stem at full power. A reduction in power is necessary before performing a valve closure. All the isolation valves, other than the MSIVs, may be tested from sensor to actuator during plant operation. The test may cause isolation of the process lines involved, but their isolation is tolerable.

7.3.2.3.3.6 10 CFR 50, Appendix A

General Design Criterion 13

The integrity of the reactor core and the reactor coolant pressure boundary (RCPB) is ensured by monitoring the appropriate plant variables and closing various isolation valves, as detailed in the various description sections.

7.3.2.3.3.7 10 CFR 50, Appendix B

The guidelines of 10 CFR 50, Appendix B, are met as described in Chapter 17.

7.3.3 Emergency Core Cooling System Auxiliary Systems Instrumentation and Control

7.3.3.1 Design-Basis Information

The design-basis information for the instrumentation and control of the reactor building closed cooling water (RBCCW) system, as required by Section 3 of IEEE 279-1971, is provided in Subsection 7.1.2.1.28.

7.3.3.2 System Description

7.3.3.2.1 Cooling System for Reactor Auxiliaries

The RBCCW system and its backup, the emergency equipment cooling water (EECW) system, provide cooling water for the ECCS auxiliary equipment and area air coolers, as described in Subsection 9.2.2. A discussion of the EECW system is contained in Subsection 7.3.4. A description of the RBCCW system, including instrumentation and control, is presented in Subsection 7.6.1.14.

7.3.3.2.2 Control Circuits

The RBCCW system operates and supplies services during normal operation of the plant. It continues to operate during an accident unless interrupted by some abnormal condition. The system is shut down by the head tank low-level signal using "one-out-of-two taken twice" logic. Each individual RBCCW pump (outside of the RBCCW supplemental cooling loops) shuts down on low pump suction, also using "one-out-of-two taken twice" logic.

In the event that the RBCCW system cannot maintain adequate flow to the EECW loops, the EECW system is automatically started by low differential pressure between the supply and return headers. Logic is "one-out-of-two." A loss of offsite power directly initiates the EECW system to anticipate the loss of power to the RBCCW system. The EECW system is also auto-initiated on high drywell pressure.

7.3.3.3 Analysis

Description of the analysis of the EECW system is found in Subsection 7.3.4.3.

7.3.4 Emergency Equipment Cooling Water System

7.3.4.1 Design-Basis Information

The design-basis information for the instrumentation and control of the EECW system, as required by Section 3 of IEEE 279-1971, is provided in Subsection 7.1.2.1.18.

7.3.4.2 System Description

The EECW system ensures cooling water to remove heat from emergency equipment on loss of offsite power, on high drywell pressure, or failure of the RBCCW system. Since this system is described in Subsection 9.2.2, the following discussion provides additional information on the EECW instrumentation and control. The EECW system is shown in Figures 9.2-3 and 9.2-4.

7.3.4.2.1 Power Sources

Instruments and controls for the EECW system receive electrical power from the redundant 120-V, 60-Hz instrument power systems described in Subsection 8.3.1 and from Division I and Division II 130 V dc Class IE batteries as described in Subsection 8.3.2.1.2. Those instruments and controls requiring pneumatic power receive plant instrument air as described in Subsection 9.3.1.

7.3.4.2.2 Equipment Design

Each of the two redundant and separate EECW loops has electrically and physically separate controls and instruments.

7.3.4.2.3 Initiation and Control Circuits

The EECW system can be started manually from the main control room. Low RBCCW flow to the EECW loops causes loss of EECW header differential pressure, which automatically isolates the RBCCW and starts the EECW loops. If EECW has already been initiated for the purposes of RBCCW heat exchanger cleaning, enhanced drywell cooling, testing, or RHR reservoir freeze protection, the location of the differential pressure sensors inside the EECW system envelope will not sense a RBCCW low flow condition and, therefore, will not cause EECW to automatically reinitiate and reisolate the nonessential loads. This action is not required since this is not a condition requiring protective action as described in Section 7.1.2.1. The EECW system is also automatically initiated by a loss of offsite power or on high drywell pressure. Automatic start of EECW makeup pump is achieved if makeup tank has low pressure or level and makeup tank isolation valve is open, and normal pump suction pressure.

7.3.4.2.4 Logic

The EECW system logic is shown in Figure 7.3-15. These logic schemes are identical for each of the two redundant EECW loops. Level switches are used to alarm when there is insufficient water inventory. If offsite power is available, the EECW pumps start nominally 1.5 seconds after receipt of the initiation signal. If offsite power is unavailable, the EECW pumps are sequenced on the EDG buses by the automatic load sequencer. See Table 8.3-5.

7.3.4.2.5 Testability

Control and logic circuitry can be tested by placing that loop in operation from the main control room. If an auto-initiate signal is received during a test, the manual signal is automatically overridden by the auto-initiate signal, and both loops will be placed into operation as required.

7.3.4.3 Analysis

7.3.4.3.1 Conformance To Specific Regulatory Requirements

The specific requirements of IEEE 279-1971, to which attention has been directed in the design of the EECW system, are itemized below by paragraph number as they appear in IEEE 279-1971.

- a. Paragraph 4.1: Automatic Initiation - This requirement is met by incorporating capability in the design for automatic startup of the EECW system on loss of offsite power, on high drywell pressure, or on occurrence of low pressure across the supply and return headers of either cooling loop
- b. Paragraph 4.2: Single Failure - The single-failure criterion is met by having an independently controlled EECW loop for each of the two RBCCW divisions
- c. Paragraph 4.3: Quality Assurance - This requirement is met as described in Chapter 17
- d. Paragraph 4.4: Equipment Qualification - This requirement is met as described in Chapter 3
- e. Paragraph 4.5: Channel Integrity - This requirement is met by supplying electrical power to the EECW system from buses backed up by diesel generators. The routings of power, signal, and control circuits take separate paths. The EECW system is designed to withstand seismic accelerations
- f. Paragraph 4.6: Channel Independence - This requirement is met by the independent instrumentation and controls provided in the EECW system and separate power feeds that are used
- g. Paragraph 4.7: Control Interaction - The requirement of this criterion is met by the complete independence of controls of the two divisions of the EECW system
- h. Paragraph 4.8: Direct Inputs - This requirement is met by the provision of separate and independent instrumentation to supply signal inputs for control of the two loops of the EECW system
- i. Paragraph 4.9: Sensor Checks - This requirement is met by introducing a test signal (in one channel at a time) sufficient to verify that a logic trip is achievable when the parameter deviates beyond the setpoint. Correct response of each sensor is verified by observing that its output indicates a deviation of the parameter beyond the setpoint value
- j. Paragraph 4.10: Testability - This requirement is satisfied by the automatic override of a manual control command if an emergency condition arises during testing
- k. Paragraph 4.11: Channel Bypass - This requirement is met by the cooling adequacy of one loop, allowing one loop to be tested during operation without loss of protection

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- l. Paragraph 4.12: Operation of Bypasses - This requirement is met by the automatic override of manual control, which is provided to automatically initiate operation of the system if an emergency arises during a test
- m. Paragraph 4.13: Bypass Indication - This requirement is met by the display provisions in the main control room, which indicate the operational or nonoperational state of the EECW system
- n. Paragraph 4.14: Bypass Access - This requirement is met by the administrative control that is imposed on use of the operational controls of the EECW system
- o. Paragraph 4.15: Multiple Setpoints - This requirement is not applicable to the EECW system
- p. Paragraph 4.16: Action Completion - This requirement is satisfied by the functional characteristics of the automatic controls of the EECW system
- q. Paragraph 4.17: Manual Access - This requirement is satisfied by the manual control provisions incorporated in the EECW system controls
- r. Paragraph 4.18: Setpoint Access - This requirement is satisfied by the administrative control that is imposed on use of the setpoint adjustments
- s. Paragraph 4.19: Identification - This requirement is satisfied by the indicating lamps and sequential recorders that the design incorporates to indicate the state of the EECW system and its valves and pumps
- t. Paragraph 4.20: Information Readout - This requirement is satisfied by the readout instruments provided to display temperature, pressure, and flow parameters in the EECW system

A sequential recorder is provided to register initiation of water pump operation and tripout of the pump motor circuit breaker
- u. Paragraph 4.21: System Repair - This requirement is met by the readily identifiable modular design of the instrumentation and control components
- v. Paragraph 4.22: Identification - This requirement is met by using appropriate tags and color schemes to enable easy identification of circuits and components that are part of the EECW safeguard system

The following additional IEEE criteria are met by the provisions outlined in the sections or chapters of this UFSAR that are indicated:

- a. IEEE 323-(): Section 3.11 (IEEE 323-1971 for equipment purchased before November 15, 1974; IEEE 323-1974 for equipment purchased on or after November 15, 1974)
- b. IEEE 336-1971:Chapter 17
- c. IEEE 338-1971:Subsection 7.3.1.3.4
- d. IEEE 344-1971:Section 3.10.

The requirements of Regulatory Guide 1.22 are met on the basis of the manual test and control provisions that the EECW system design incorporates.

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Evaluation of the EECW system against criteria of 10 CFR 50, Appendix A and Appendix B, is as follows:

- a. Criterion 13 - This criterion is met by using qualified differential pressure sensors and operating them in a "one-out-of-two" logic arrangement
- b. Criterion 20 - This criterion is met by providing the automatic mode of startup as stated in Subsection 7.1.2.1.18
- c. Criterion 21 - The EECW system provides assurance that, through its standby redundancy, each loop has sufficient reliability to fulfill the single-failure criterion. No single component failure, maintenance operation, calibration operation, or test to verify operational availability impairs the ability of the system to perform its intended safety function. There is sufficient electrical and physical separation between channels and between trip logic circuits monitoring the same variable to prevent environmental factors, electrical transients, and physical events from impairing the ability to respond correctly

The EECW system includes design features that permit inservice testing. This enhances the functional reliability of the system by enabling early detection of malfunctioning components in the course of routine tests
- d. Criterion 22 - Physical separation, separate power feeds, and separate controls are provided for the two cooling loops. This ensures that the EECW system of each loop, providing necessary cooling capacity, is available for the required safety function. Details of separation criteria and independence are contained in Section 3.12 and Subsection 9.2.2
- e. Criterion 23 - Since the two loops are independent, failure of one loop will not affect operation of the other
- f. Criterion 24 - Since no signals required for control of the reactor are used for control of the EECW system, this criterion is satisfied
- g. Criterion 29 - High functional reliability of the EECW system is achieved through the combination of sensor redundancy, control logic arrangement, functional and physical separation of loops, operating power independence, fail-safe design, and inservice testability. These requirements are discussed in detail in Criteria 21 through 24.

An extremely high probability of correct system response to anticipated operational occurrences is maintained by a thorough program of inservice testing and surveillance. Active components can be tested or removed from service for maintenance during reactor operation without compromising protective control functions, even in the event of a subsequent single failure. Components important to safety are tested during normal reactor operation. Functional testing and calibration schedules are developed using available failure rate data, reliability analyses, and operating experience. These schedules represent an optimization of system reliability by considering the failure probabilities of individual components, and also the reliability effects during individual component testing on the portion of the system not undergoing test. The capability for inservice testing ensures the high functional

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reliability of the system should a monitored parameter exceed the corrective action setpoint.

The guidelines of 10 CFR 50, Appendix B, are met as described in Chapter 17.

7.3.5 Control Center Atmospheric Control System Instrumentation and Control

7.3.5.1 Design-Basis Information

The design-basis information for the instrumentation and control of the control center atmospheric control system, as required by Section 3 of IEEE 279-1971, is provided in Subsection 7.1.2.1.

7.3.5.2 System Description

The instrumentation and control for the control center HVAC system functions to ensure the habitability of the control center under all plant operating conditions is described in Section 6.4 and Subsections 9.4.1 and 12.2.2.1.

7.3.5.2.1 Power Sources

Each redundant control center HVAC system is comprised of a supply air fan, return air fan, electric heating coil, and a refrigeration unit. Power supply for these components of each control center HVAC system is from separate essential ac buses that can receive standby ac power. Control power for isolation dampers, instrumentation, and controls comes from the bus that powers the corresponding equipment train.

7.3.5.2.2 Initiating Circuits, Logic, and Sequencing

Various components of each redundant control center HVAC system are initiated as follows:

- a. The supply and return air fans are initiated manually by control switches at the main control board
- b. The refrigeration unit is provided with a manual/ automatic selector switch on the main control board. While in automatic mode, the refrigeration unit is initiated by the demand signal from the thermostat in the chilled water piping
- c. A subsystem of the process radiation monitoring system (PRMS) monitors radiation levels in the main control room air intake. High radiation indication by detectors in the main control room air intake duct downstream of the filter train activates an alarm within the control center. High-high level automatically places the control center HVAC system in full recirculation mode
- d. If combustion products are detected in the control center by the smoke (ionization) detectors, one of the following manual actions is initiated via hand switches in the main control room at the discretion of the operator:
 1. The outside air intake and exhaust air damper can be fully opened and the recirculation air damper fully closed to purge the control center air (smoke purge mode)

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2. Route outside air and recirculation air mixture (approximately 7 percent of the total mixture) from the control center HVAC system through normally bypassed odor and smoke-removing filters.
 - e. If the control center HVAC system is operating in the normal mode and one of the automatic gaseous suppression systems is initiated automatically by the fire detection system, the smoke purge mode is automatically initiated. However, the smoke purge mode is overridden if the recirculation mode is signaled to start.
 - f. If an automatic isolation occurs due to detection of a potential breach of the primary reactor pressure boundary, as indicated by low reactor water level, high drywell pressure, high radiation level as monitored by the fuel pool ventilation exhaust, or the reactor building ventilation exhaust, the emergency makeup outside air is automatically provided to pressurize the main control room.

7.3.5.2.3 Bypasses and Interlocks

All of the isolation dampers in each control center HVAC system equipment are interlocked with the operation of corresponding supply air and return air fans. Operation of any of these fans opens all the corresponding isolation dampers. The supply-air and return-air fans are operated manually by hand switches.

To prevent short-cycling and a possible freeze-up of the evaporator, the refrigeration machine start is interlocked with the operation of the supply-air fan and corresponding return-air fans, condenser cooling water, and chilled water pump. The operation of the refrigeration machine is further interlocked with safety protection cutout; i.e., low-pressure and high-pressure cutout in refrigerant circuit, and oil failure switch in the compressor lubrication circuit. To guard against overheating, the electric heating coil is interlocked with supply-air fan operation and a thermal cutout switch. Low temperature of the chilled-water line is alarmed.

Zone mixing dampers are controlled by thermostats in each zone. The operation of the refrigeration machine is controlled by a thermostat in the chilled-water return pipe. The electric heating coil is controlled by a thermostat in the hot deck of the air handling unit.

All of the isolation dampers in the outside air intakes and the emergency-makeup-air filter train are appropriately interlocked to serve the required function. The electric heating coil for humidity control in the emergency-makeup-air filter train is interlocked with the emergency-makeup-air fans.

7.3.5.2.4 Redundancy and Diversity

Instrumentation and control equipment for each control center HVAC system is completely independent of one another.

7.3.5.2.5 Actuated Devices

The normal and emergency operation of each control center HVAC system involves the following actuated devices:

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- a. Supply-air fan
- b. Return-air fan
- c. Electric heating coil
- d. Refrigeration unit
- e. Emergency-makeup-air electric heating coil
- f. Emergency-makeup-air fan
- g. Corresponding isolation and control dampers
- h. Chilled water pump.

7.3.5.2.6 Separation

The channels and logic circuits are physically and electrically separated to preclude the possibility that a single event would prevent operation of the control center HVAC system. Electrical cables for instrumentation and control on each control center HVAC system are routed separately.

7.3.5.2.7 Testability

Control and logic circuitry used in the controls for the control center HVAC system can be individually checked by applying test or calibration signals to the sensors and observing trip or control responses. Operation of each redundant HVAC system is periodically rotated to permit on-line checking and testing of performance of the complete system. The automatic control circuitry for the emergency equipment is designed to restore its normal function in response to initiation signals.

7.3.5.2.8 Environmental Considerations

Temperature, pressure, humidity, and radiation dosage are considered in selection of various equipment, instrumentation, and controls for the control center HVAC system. These are described in Section 3.11 and Subsection 9.4.1.

7.3.5.2.9 Operational Considerations

The control center HVAC system is required during normal and abnormal plant operating conditions. The automatic circuitry is designed to start the emergency equipment if the signal for its initiation is received, as described in this section.

7.3.5.3 Analysis

Conformance To General Functional Requirements

The control center HVAC system instrumentation and controls are designed to ensure the habitability of the main control room during and after all the normal and abnormal plant operating conditions. Certain components of the system are required during normal and abnormal plant operating conditions only. The controls for the system provide warning to the operator of any abnormal operating transients in the system, and automatically initiate

action that provides protection against the consequences of the release of radioactive material to outdoor environs following any accident.

Chapter 15 identifies and evaluates postulated events that can result in release of fission products due to an accident. The consequences of such an accident are described and evaluated.

Because essential variables are monitored by channels arranged for physical and electrical independence, no single failure, maintenance operation, calibration operation, or test can prevent the system from performing its function. A single active failure in the Halon fire protection system will cause closure of smoke/Halon dampers to the relay room, cable spreading room or computer room. Manual actions are required to reopen these dampers to reestablish airflow.

The sensor circuitry and logic used in the control center HVAC system are not used in the control of any process system. Thus, malfunction and failures in the controls of the process systems have no direct effect on the control center HVAC system.

The power supplies used for the system logic circuitry and controls provide assurance that the required performance cannot be affected by a loss of offsite electric power or loss of instrument air. In no case does the loss of a single power supply prevent function of the control center HVAC system.

Portions of the system required to operate during and following the design basis accident to provide acceptable environments within the control center have been qualified both environmentally and seismically.

Inputs to annunciators and indicators are arranged so that no malfunction of the annunciating and indicating device can functionally disable the system. Direct signals from the control center HVAC system control system sensors are not used as inputs to annunciating or data-logging equipment.

All controls for interrupting any part of the system operation are located in the main control room. All controls and instrumentation essential to the operation of the control center HVAC system meet the IEEE 279-1971 criteria.

7.3.6 Standby Gas Treatment System Instrumentation and Control

7.3.6.1 Design-Basis Information

The design basis information for the instrumentation and control of the standby gas treatment system (SGTS), as required by Section 3 of IEEE 279-1971, is provided in Subsection 7.1.2.1.16.

7.3.6.2 System Description

The instrumentation and control of the SGTS are used to maintain, when necessary, a preset constant flow that will maintain the reactor building at a negative pressure and preclude leakage of radioactive particulates and gases directly to the outdoors. The SGTS is designed to reduce radioactive particulates and gaseous concentration in the exhaust air from the reactor building before exhausting to the outdoors.

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The SGTS is described in detail in Subsection 6.2.3 and is shown schematically in Figure 6.2-20.

7.3.6.2.1 Power Sources

Each SGTS exhaust equipment train has an exhaust fan, a standby cooling fan, an electric heating coil, and associated air- operated isolation valves that require power. Power supply for various components of each SGTS equipment train and the instrument air compressor is from separate essential ac buses that can receive standby ac power. Motive power for isolation valves and the controls comes from the bus that powers the corresponding equipment train, except for the isolation valves in the reactor building ventilation system supply and exhaust duct headers. These valves are operated by air cylinders with instrument air being controlled by solenoid valves for each isolation valve. If either control air or electric control power were lost, the isolation valves would be closed by springs mounted on the valves.

7.3.6.2.2 Initiating Circuits

The system is automatically started in response to any one of the following signals:

- a. High drywell pressure (Subsection 7.3.2.2.7.6)
- b. Low reactor water level (Subsection 7.3.2.2.7.1)
- c. High radiation in fuel pool ventilation exhaust (Subsection 11.4.3.8.2.11)
- d. High radiation in the reactor building ventilation exhaust (Subsection 11.4.3.8.2.4)
- e. Manual activation from the main control room
- f. Downscale trip due to loss of offsite power to radiation monitors located in the Reactor Building and fuel pool ventilation exhaust system.

7.3.6.2.3 Logic and Sequencing

The following actions take place simultaneously on receipt of an initiation signal:

- a. Closure trip of reactor building isolation valves
- b. Trip of reactor building ventilation system
- c. Opening of SGTS isolation valves
- d. Startup of both SGTS equipment trains and annunciation of an alarm on the main control panel.

When both trains are automatically started, the audible and visual alarm on the main control panel warns the operator to shut down one of the trains. Individual hand switches located on the main control panel for each of the equipment trains permit manual operation.

7.3.6.2.4 Bypasses and Interlocks

All of the air-operated isolation valves pertinent to a SGTS equipment train are interlocked through a relay circuit with the operation of the SGTS unit.

The SGTS cooling fan is interlocked so as not to operate when the SGTS exhaust fan is in operation.

To protect against overheating, the electric heating coil for relative humidity control of the charcoal filters is interlocked with the SGTS exhaust fan operation, and high temperature is indicated by an alarm in the main control room.

Airflow through the SGTS is controlled automatically with a vortex damper on the exhaust fan valve, and flow is recorded on the main control panel. Low flow initiates an alarm to alert the operator to start the redundant SGTS equipment train.

To prevent fire in the charcoal bed, a source of CO₂ will purge air from the charcoal filters if the bed temperature exceeds 310°F. A pressure switch in the discharge line of the CO₂ unit will annunciate an alarm on the main control panel after the purge process begins.

On receipt of an initiation signal, the reactor building ventilation isolation valves close and remain closed unless a manual reset switch is activated.

7.3.6.2.5 Redundancy and Diversity

Each SGTS unit is automatically initiated by independent control systems.

7.3.6.2.6 Actuated Device

Initiation of the SGTS includes starting of the SGTS exhaust fan, energizing electric heating for preheating air, deenergizing charcoal bed heaters, and opening valves on the inlet and outlet sides of the SGTS equipment train.

7.3.6.2.7 Separation

The channels and logic circuits are physically and electrically separated to preclude the possibility that a single event would prevent operation of the SGTS. Electrical cables for instrumentation and control on each SGTS equipment train are routed separately.

7.3.6.2.8 Testability

Control and logic circuitry used in the controls for the SGTS can be individually checked by applying test or calibration signals to the sensors and observing trip or control responses. Operation of the isolation valves and fans from manual switches verifies the ability of breaker and damper mechanisms to operate.

7.3.6.2.9 Environmental Considerations

Temperature, pressure, humidity, and radiation dosage are considered in the selection of the various equipment, instrumentation, and controls for the SGTS described in Section 3.11 and Subsection 6.2.3.

7.3.6.2.10 Operational Considerations

The SGTS is available, if required, during normal plant operating conditions when any division is being tested. The other division is available for operation should it be needed.

7.3.6.3 Analysis

Conformance To General Functional Requirements

The SGTS control system is designed to initiate action that provides timely protection against the consequences of the release of radioactive materials inside the secondary containment following any accident. Chapter 15 identifies and evaluates postulated events that can result in release of fission products due to an accident. The consequences of such an accident are described and evaluated.

Because essential variables are monitored by channels arranged for physical and electrical independence, and because a dual trip system arrangement is used to initiate the SGTS, no single failure, maintenance operation, calibration operation, or test can prevent the system from operating when required. The sensor circuitry and logics used in the SGTS control system are not used in the control of any process system. Thus, malfunction and failures in the controls of process systems have no direct effect on the SGTS control system.

The various motive power supplies used for the SGTS logic circuitry and controls provide assurance that the required initiation can be effected in spite of loss of electric power or loss of instrument air. In no case does a loss of single power supply prevent initiation of the SGTS when required. Required instruments, isolation valve closing mechanisms, and cables of the SGTS can operate under the environmental conditions associated with postaccident operation. Active components of SGTS instrumentation and control can be tested and calibrated during plant operation.

All sensors and associated equipment are designed to meet Category I requirements, and are protected from fire, explosion, missiles, lightning, wind, and flood to preclude functional degradation of the system performance.

Inputs to annunciators and indicators are arranged so that no malfunction of the annunciating and indicating device can functionally disable the system. Direct signals from the SGTS control system sensors are not used as inputs to annunciating or data-logging equipment. Isolation is provided between primary signal and the information output.

7.3.7 Standby Power System

7.3.7.1 Design-Basis Information

The design-basis information for the instrumentation and control of the standby power system, as required by Section 3 of IEEE 279-1971, is provided in Subsection 7.1.2.1.

7.3.7.2 System Description

Four emergency diesel generators (EDGs) provide the power necessary for the ECCS during a loss of system power. A detailed explanation of this system and its corresponding instrumentation and control can be found in Subsection 8.3.1.

A battery system of redundant 130-V dc power-control batteries provides the dc power required during any ECCS function. Full-size battery chargers normally carry the load. However, the capability of the batteries is enough to handle the load for a sufficient amount of time should power to the chargers be lost. A detailed explanation of the battery system can be found in Subsection 8.3.2.

7.3.7.3 Analysis

Conformance To General Functional Requirements

The standby power systems are designed to provide electrical power availability to the ECCS and other safety-related systems at maximum reliability should normal offsite power not be available. Each supply has an independent redundant counterpart, thereby ensuring against single failure (IEEE 279-1971, Paragraph 4.2). The equipment, cabling interconnections, and circuit breakers are all qualified to Class 1E standards and housed in Category I structures.

An explanation of the EDGs and their conformance is found in Subsection 8.3.1. An explanation of the battery systems and their conformance is found in Subsection 8.3.2.

7.3.8 Post-LOCA Combustible Gas Control System

The NRC amended 10 CFR 50.44, "Standards for combustible gas control system in light-water-cooled power reactors" on October 16, 2003 to eliminate the requirements for hydrogen recombiners. The hydrogen recombiner Technical Specification requirements were subsequently removed by License Amendment 159, dated March 15, 2004. The wording in this UFSAR section associated with these changes will remain unaltered until after the hydrogen recombiner system has been abandoned in place or removed from the plant.

7.3.8.1 Design-Basis Information

The design bases for the post-LOCA CGCS are described in Subsection 6.2.5, and functional requirements for the instrumentation and control system are given in Subsection 6.2.5 and 7.1.2.1.30. Gas concentrations inside the primary containment are monitored by the Hydrogen/Oxygen analyzer subsystem of the primary containment monitor system. The primary containment monitor system is further described in Subsection 7.6.1.12.

7.3.8.1.1 Redundancy

The instrumentation and control for the post-LOCA combustible gas control system (CGCS) conforms to the redundancy requirements of IEEE 279-1971. The post-LOCA CGCS includes two complete and independent sets of recombiner controls and power supplies. In general, the components in each are not further duplicated for additional redundancy, except that more heater capacity is provided than required and all thermocouples are duplicated.

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7.3.8.1.2 Separation

Separation is such that no single failure can prevent the operation of the CGCS. Instrumentation and Control equipment is segregated into separate divisions designated Division I and II. The CGCS conforms to the separation requirements of IEEE 279-1971.

7.3.8.1.3 Reliability and Testability

Refer to subsection 7.3.1.2.1.7.

7.3.8.2 System Description

7.3.8.2.1 General

The purpose of the Fermi 2 post-LOCA CGCS is to control combustible gas concentrations in the containment atmosphere to ensure that the concentration is maintained below stated limits, in conformance with the requirements of Regulatory Guide 1.7. Oxygen is the minority constituent in an inerted containment and is the controlled parameter.

The Control Console contains the instrumentation, annunciators, switches and lights to facilitate remote operation of the recombiner system. The control console contains the following equipment which controls and monitors the operation of the recombiner:

- a. Operate switch
- b. Inlet valve flow controller
- c. Recirc valve flow controller
- c. Reaction chamber temperature controller
- e. Annunciators and status lights
- f. Motor operated valve control switches.

The system is initiated by actuating the operate switch, which starts the Blower, opens the cooling water valve and starts the heaters. Automatic temperature control and temperature limits have been preset and temperature rises to the point where a controlled exothermic chemical recombination of Hydrogen and Oxygen occurs. The power to the heaters is controlled to cycle the heaters to maintain reaction chamber temperature at 1300°F. Periodically, as the Hydrogen and Oxygen concentrations decrease, the inlet flow would be manually increased and recirc flow manually decreased, until the recirc valve is fully closed.

The system continues to operate automatically as the Hydrogen and Oxygen gases are depleted, with the heater power gradually increasing to maintain the reaction chamber temperature at its control setpoint. The control room is provided with status lights for:

- a. Trickle heat power off
- b. System in ready mode
- c. System in start up mode
- d. System in operate mode.

7.3.8.2.2 Subsystems

7.3.8.2.2.1 Electrical Heater and Control System

The electric heater in each recombiner consists of 36 U-shaped, horizontally mounted elements. Each heater is independently replaceable from its cold end without interfering with the process piping. These heaters radiate their heat to a pipe coil containing the process gas to be heated.

Power to the heater bank is regulated by a 90-kW, 480-V three-phase silicon-controlled rectifier controller. This SCR unit is controlled by one of two solid-state temperature controllers acting through an "auctioneer" circuit. The Temperature controllers are proportional mode controllers with an adjustable control zone. The temperature controllers operate from type K thermocouples, in Inconel sheaths, exposed directly to the process gas. Control thermocouples are located at the heater outlet and at a location in the reaction chamber where the H₂/O₂ reaction is normally essentially complete. If either of the temperature signals varies from the preselected setpoint, the signal controls the SCR through the auctioneer circuit and thereby regulates the heater power.

Heater overtemperature protection is provided by a temperature switch and a contactor capable of interrupting power to the SCR controller.

7.3.8.2.2.2 Gas Cooler Unit

A direct-contact water-spray gas cooler is provided to cool the gases leaving the reaction chamber from about 1300°F to 210°F (maximum) prior to return to the pressure suppression chamber. The cooling water is turned on or off by a motorized valve actuated by the main start switch. Indication is provided from the valve-open limit switch.

7.3.8.2.2.3 Recirculation Flow Control System

The recirculation flow control system (RFCS) includes two flow-meters, two motor-operated valves, and a centrifugal blower. The purpose of these components is to move controlled amounts of process gas through the system, and control the recirculation of controlled amounts of the reacted effluent back to the inlet to act as a diluent. The inlet and bypass valves are positioned remotely to modulate flow based on the use of curves or tables.

The flow control valves are 3-in. pipe size and can regulate flow rates between zero and 150 scfm.

7.3.8.2.2.4 Trickle Heat System

To ensure that the heated components of the system remain dry over the long periods of standby, a low-power heater is provided to maintain the temperature of these components between 100°F and 175°F. A controller located in the power cabinet will be used to regulate the temperature, with display provided in the main control room to indicate that the heater is off.

7.3.8.2.3 System Interlocks and Shutdown circuits

The system logic and shutdown circuits are shown in Figure 5 of Reference 4. A summary is provided below:

- a. The Blower motor starts when:
 1. Both the control room and relay room Off-Ready switches are in "Ready"
 2. Blower interface relay is energized
 3. System permissives are within following limits:
 - (a) Blower discharge temperature
 - (b) Inlet pressure
 - (c) Reaction chamber shell temperature
 - (d) Return gas temperature
 - (e) Heater wall temperature.
- b. The Main heater contactor closes when:
 1. Water valve is open
 2. Blower flow is above preset limit
 3. System permissives are within following limits:
 - (a) 2/3 through heater temperature
 - (b) Heater outlet gas temperature.
- c. The Main heater SCR modulates the power output when:
 1. Heater contactor is closed
 2. The Reaction chamber temperature controller provides an output signal. This occurs when reaction chamber temperature is outside of the setpoint control zone (nominal $\pm 50^{\circ}\text{F}$)
- d. High temperature signals will shutdown the heaters and blowers.
- e. A loss of cooling water causes a rise in gas temperatures which automatically shuts off heater power.
- f. A loss of power to the heater and blower shuts down the system.

Further details of system operation and instrumentation and control circuits are contained in Reference 4.

7.3.8.3 Power Supply

The two power sources for the system are supplied by 480-V ESF buses. These two buses are supplied from 4160-V buses, which in an emergency are supplied by onsite EDGs. Each recombiner unit requires 120 kW. The power panel distributes 480 V ac and 120 V ac for power and control.

7.3.8.4 Physical Location and Environmental Conditions

The recombiners are located in an area outside containment in the reactor building. The power control center is located in the auxiliary building. The control package is located in the relay room and will be accessible to personnel after the postulated LOCA. The environmental conditions for these areas are discussed in Section 3.11.

7.3.8.5 Analysis

The redundant post-LOCA CGCS is provided to ensure that, under conservatively assumed conditions, the concentration of combustible gases in the containment atmosphere does not exceed the volume percent limits of Regulatory Guide 1.7. An analysis of the system, described in Subsection 6.2.5, confirms that the system accomplishes the required safety function, if needed, with adequate margin and redundancy to ensure that containment integrity is not endangered. Adequate redundancy of the instrumentation and control system ensures that a single failure of any active component does not prevent the system from performing its required safety function.

7.3.9 Residual Heat Removal Service Water System Instrumentation and Control

7.3.9.1 Design-Basis Information

The design-basis information for the instrumentation and control of the residual heat removal service water (RHRSW) system, as required in Section 3 of IEEE 279-1971, is provided in Subsection 7.1.2.1.27.

7.3.9.2 System Description

The RHRSW system provides cooling water to remove heat from the RHR system. The RHRSW system includes a closed-cycle supply of water, pumps, and mechanical draft cooling towers to reject the heat to the environment. The system will operate with or without a loss of offsite power. The RHRSW system is described in Subsection 9.2.5 and the system diagram is provided in Figure 9.2-6. The following discussion provides additional information on the RHRSW instrumentation and control.

7.3.9.2.1 Power Sources

Instrumentation and controls for the RHRSW system receive electrical power from the redundant 120-V, 60-Hz instrument power systems described in Subsection 8.3.1. Part of the control logic is direct current, powered by the Class 1E direct current system described in

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Subsection 8.3.2. The pressure control valves requiring pneumatic power receive plant instrument air as described in Subsection 9.3.1.

7.3.9.2.2 Equipment Design

Each of the two separate, redundant RHRSW loops has electrically and physically separate controls and instruments.

7.3.9.2.3 Initiation and Control Circuits

The RHRSW pumps, valves, and cooling tower fans are all initiated manually from the main control room.

7.3.9.2.4 Logic

The RHRSW system is a manually initiated system; therefore, there is no automatic initiation logic. The RHRSW pumps automatically trip if they are operating and a LOCA signal is received, as indicated in Figure 7.3-9, Sheet 1. This trip is provided to allow the automatic loading of other emergency safety feature equipment on the emergency diesel generators if a loss of offsite power occurs. The interlock can be bypassed by a keylock switch so that the pumps can be started if there is a long-term LOCA signal present. The cooling tower fan motors automatically load-shed if a loss of offsite power occurs. The motors must be manually reset from the main control room before they will restart.

7.3.9.2.5 Testability

Each RHRSW loop can be tested from the control room by starting the RHRSW pumps and/or cooling tower fans. If an accident signal is received during a test, the system pumps trip off and remain off until the operator manually restarts.

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7.3 ENGINEERED SAFETY FEATURE SYSTEMS

REFERENCES

1. U.S. Nuclear Regulatory Commission, NRC Action Plan Developed as a Result of the TMI-2 Accident, NUREG-0660, May 1980; Revision 1, August 1980.
2. U.S. Nuclear Regulatory Commission, Clarification of TMI Action Plan Requirements, NUREG-0737, October 1980.
3. Letter from W. H. Jens, Detroit Edison, to B. J. Youngblood, NRC, Subject: Modifications of ADS Logic (NUREG-0737, Item II.K.3.18), dated July 31, 1984 (EF2-66712).
4. J.O. Henrie and S.A. Itow, Thermal Hydrogen Recombiner System for Mark I and II Boiling Water Reactors, Report AI-77-55, Atomics International Division of North American Rockwell, September 12, 1977.
5. 004N0874, Revision 0, "HPCI/RCIC Unavailability – Wide Range Level Off-Calibration Level 8 Trip at Low Pressure – for the Enrico Fermi 2 Nuclear Power Plant," dated February 2017.
6. Letter from the USNRC to DTE Electric Company, "Fermi 2 – Issuance of Amendment to Revise High Pressure Coolant Injection System and Reactor Core Isolation Cooling System Actuation Instrumentation Technical Specifications (CAC No. MF9330)," dated April 14, 2017.

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TABLE 7.3-1 HIGH PRESSURE COOLANT INJECTION SYSTEM INSTRUMENT SPECIFICATIONS

HPCI Function	Instrument	Trip Settings ^a	Instrument Range
RPV high water level turbine trip	Level transmitter	Level 8 ^b	10 to 220 in. ^c
Turbine exhaust diaphragm high pressure	Pressure transmitter	10 psig	0 to 50 psig
Turbine exhaust high pressure	Pressure switch	140 psig	0 to 200 psig
HPCI system pump low suction pressure	Pressure switch	15 in. Hg V ac	30 in Hg to 0.5 psig
HPCI system pump high suction pressure	Pressure switch	70 psig	2 to 75 psig
RPV low water level	Level transmitter	Level 2 ^b	10 to 220 in. ^c
Primary containment (drywell) high pressure ^d	Pressure transmitter	2 psig	0 to 5 psig
HPCI system steam supply low pressure	Pressure transmitter	100 psig	0 to 200 psig
Condensate storage lank low level	Level transmitter	45,000 gal	-10 in./0/+10 in. H ₂ O
Turbine overspeed	Centrifugal device	122 percent of turbine rated speed	

^a Nominal values are given for information. See Technical Specifications for actual operational settings.

^b Shown in Figure 7.3-12.

^c Zero is at the top of the active fuel.

^d Incident detection circuitry instrumentation.

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TABLE 7.3-2 AUTOMATIC DEPRESSURIZATION SYSTEM INSTRUMENT TRIP SETTINGS

<u>System Function</u>	<u>Instrument</u>	<u>Trip Settings^a</u>	<u>Instrument Range</u>
Reactor vessel low water level (permissive) ^d	Level transmitter	Level 3 ^b	160 to 220 in. ^c
Reactor vessel low water level (permissive) ^d	Level transmitter	Level 1 ^b	10 to 220 in. ^c
Primary containment (drywell) high pressure ^d	Pressure transmitter	1.68 psig	0 to 5 psig
Primary containment (drywell) high pressure bypass time delay ^d	Timer	7 min	1 to 30 min
Automatic depressurization time delay ^d	Timer	105 sec	10 to 300 sec
LPCI pump discharge pressure ^d	Pressure transmitter	118.5 psig	0 to 500 psig
Core spray pump discharge pressure ^d	Pressure transmitter	143.5 psig	0 to 500 psig

^a Nominal values are given for information. See Technical Specifications for actual operational settings.

^b Shown in Figure 7.3-12.

^c Zero is at the top of the active fuel.

^d Incident detection circuitry instrumentation.

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TABLE 7.3-3 CORE SPRAY SYSTEM INSTRUMENT SPECIFICATIONS

<u>Core Spray Function</u>	<u>Instrument</u>	<u>Trip Settings^a</u>	<u>Instrument Range</u>
RPV low water level ^d	Level Transmitter	Level 1 ^b	10 to 220 in. ^c
Primary containment high pressure ^d	Pressure transmitter	1.68 psig	0 to 5 psig
RPV low pressure	Pressure transmitter	469 psig ^e Decreasing	0 to 1200 psig
Core spray sparger high differential pressure	Differential pressure switch	0.2 psid	-7 to +2 psid
Pump discharge flow	Flow indicator	--	0 to 10,000 gpm
Pump suction pressure	Pressure indicator	--	-30 in. Hg to 30 psig
Pump discharge pressure	Pressure transmitter	143.5 psig	0 to 50 psig

^a Nominal values are given for information. See Technical Specifications for actual operational settings.

^b Shown in Figure 7.3-12.

^c Zero is at the top of the active fuel.

^d Incident detection circuitry instrumentation.

^e Approximate setting.

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TABLE 7.3-4 LOW PRESSURE COOLANT INJECTION INSTRUMENT SPECIFICATIONS

LPCI FunctionF	Instrument	Trip Settings ^a	Instrument Range
RPV low water level (LPCI loop selection) ^d	Level transmitter	Level 2 ^b	10 to 220 in. ^c
RPV water level (LPCI pump start signal) ^d	Level transmitter	Level 1 ^b	10 to 220 in. ^c
Primary containment (drywell) high pressure(LPCI initiation) ^d	Pressure transmitter	1.68 psig	0 to 5 psig
RPV low water level (inside shroud)	Level transmitter	Level 0 ^b	-150 to +50 in.
Recirculation loop break detection	Differential pressure transmitter	0.63 psid ^e Trip on upscale	0 to 2 psid
LPCI break detection circuit	Timer	1/2 sec	0.15 to 3 sec
LPCI break detection circuit	Timer	2 sec	0.15 to 3 sec
LPCI reactor vessel low pressure	Pressure transmitter	925 psig	0 to 1500 psig
LPCI valve initiation signal cancellation	Timer	10 minutes	3 to 30 minutes
LPCI pump low flow	Flow switch	1500 gpm	0 to 20 in. WC
RPV pressure permissive (loop selection)	Pressure transmitter	469.5 psig	0 to 1200 psig
Recirculation pumps differential pressure transmitter	Differential pressure transmitter	1.63 psid Trip on downscale	0 to 5 psid

^aNominal values are given for information. See Technical Specifications for actual operational settings.

^bShown in Figure 7.3-12.

^cZero is at the top of the active fuel.

^dIncident detection circuitry instrumentation.

^eRepeatability of ±0.5 percent on trip point. Return from overrange of 200 psi to 0 psi in 100 msec maximum.

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TABLE 7.3-5 HIGH PRESSURE COOLANT INJECTION SYSTEM: MINIMUM NUMBERS OF TRIP CHANNELS REQUIRED FOR FUNCTIONAL PERFORMANCE

Component Affected	Trip Channel	Instrument Type	Number of Trip Channels Provided	Minimum Number of Trip Channels Required To Maintain Functional Performance ^a
HPCI system initiation	RPV low water level	Level transmitter	4	2 per untripped trip system
HPCI system initiation	Primary containment high pressure	Pressure transmitter	4	2 per untripped trip system
HPCI system turbine	HPCI system pump discharge flow	Flow indicator controller	1	1
HPCI system turbine	RPV high water level	Level transmitter	2	1 per untripped sytem
HPCI system turbine	Turbine exhaust diaphragm high pressure	Pressure transmitter	2	1 ^b
HPCI system turbine	HPCI system pump low suction pressure	Pressure switch	1	1 ^b
Minimum flow bypass valve	HPCI system pump flow	Flow switch	1	1
HPCIS steam supply valve and suppression chamber suction valve	HPCI system steam supply low pressure	Pressure transmitter	4	2 per untripped trip system
Suppression chamber suction valve	Condensate storage tank low level and suppression pool high level	Level transmitter	4 ^c	2

^a Nominal values are given for information. See Technical Specifications for operational requirements.

^b An inoperable trip channel should be placed in the untripped state.

^c Two each: condensate storage low, suppression pool high.

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TABLE 7.3-6 AUTOMATIC DEPRESSURIZATION SYSTEM: MINIMUM NUMBERS OF TRIP CHANNELS REQUIRED FOR FUNCTIONAL PERFORMANCE

Initiating Function	Instrument Type	Number of Trip Channels Provided	Minimum Number of Trip Channels Required To Maintain Functional Performance ^{a,b}
RPV low water level (Level 1)	Level transmitter	2 per trip system	2 per untripped trip system
RPV low water level (Level 3)	Level transmitter	1 per trip system	1 per untripped trip system
Primary containment high pressure	Pressure transmitter	2 per trip system	2 per untripped trip system
Time delay (ADS timer)	Timer	1 per trip system	2 per untripped trip system
Time delay (ADS drywell high pressure bypass timer)	Timer	2 per trip system	2 per untripped trip system
ac interlock (RHR or core spray pump running)	Pressure transmitter	1 per pump	1 per pump

^a One trip logic of each trip system must be fully operable. Both an RPV low water level trip channel and a primary containment high-pressure trip channel should not be inoperable in any one trip logic.

^b Nominal values are given for information. See Technical Specifications for operational requirements.

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**TABLE 7.3-7 CORE SPRAY SYSTEM: MINIMUM NUMBERS OF TRIP CHANNELS
REQUIRED FOR FUNCTIONAL PERFORMANCE**

Component Affected	Trip Channel	Instrument Type	Number of Trip Channels Provided	Minimum Number of Trip Channels Required to Maintain Functional Performance ^a
Core spray system	RPV low water level	Level transmitter	4	2 per untripped trip system
Core spray system	Primary containment high pressure	Pressure transmitter	4	2 per untripped trip system
Core spray discharge valve	RPV low pressure	Pressure transmitter	4	2 per untripped trip system
Core spray sparger leak detection	Core pressure differential	Differential pressure switch	1 per sparger (alarm only)	1 per sparger (alarm only)

^a Nominal values are given for information. See Technical Specifications for operational requirements.

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TABLE 7.3-8 LOW-PRESSURE COOLANT INJECTION: MINIMUM NUMBERS OF TRIP CHANNELS REQUIRED FOR FUNCTIONAL PERFORMANCE

Component Affected	Trip Channel	Instrument Type	Number of Trip Channels Provided	Minimum Number of Trip Channels Required To Maintain Functional Performance ^a
LPCI initiation	RPV low water level	Level transmitter	4	2 per untripped trip system
LPCI initiation	Primary containment high pressure	Pressure transmitter	4	2 per untripped trip system
Containment spray valves	RPV low water level inside shroud	Level transmitter	1	1 ^b
Minimum flow bypass valves	LPCI pumps discharge low flow	Flow switch	1, 2 ^c (one per loop)	1, 2 ^c
LPCI injection valves and recirculation loop valves	Recirculation loop break	Differential pressure transmitter	4	2
LPCI injection valves	RPV low pressure	Pressure transmitter	4	2
Reactor recirculation pumps	RPV low water level	Level transmitter	4	2
Containment cooling valves	Primary containment (drywell) high pressure	Pressure transmitter	4	2

^a Nominal values are given for information. See Technical Specifications for operational requirements.

^b An inoperable sensor should be placed in the untripped state.

^c One channel to open, two channels to close.

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TABLE 7.3-9 PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM INSTRUMENTATION SPECIFICATIONS

Isolation Function	Sensor	Instrument Range	Trip Setting ^a
RPV low water level (L3)	Differential pressure transmitter	160 to 220 in.	L3 ^b
Reactor vessel low water level (L2)	Differential pressure transmitter	10 to 220 in.	L2 ^b
Reactor vessel low water level (L1)	Differential pressure transmitter	10 to 220 in.	L1 ^b
Main steam line high radiation	Radiation monitor	0 to 10 ⁶ mR/hr	3.0 x full power background
Main steam tunnel high temperature	Temperature sensor	50 to 350°F	140°F
Main steam line high flow	Differential pressure transmitter	0 to 150 psi	102 psid
Main steam line low pressure	Pressure transmitter	0 to 1200 psig	756 psig
Primary containment high pressure	Pressure transmitter	0 to 5 psig	1.68 psig
RCIC turbine area high temperature	Temperature sensor	50 to 350°F	154°F
RCIC turbine steam line high flow	Differential pressure transmitter	-300 to +300 in H ₂ O	+109 in. H ₂ O -109 in. H ₂ O
RCIC turbine steam line low pressure	Pressure transmitter	0 to 200 psig	62 psig
HPCI turbine area high temperature	Temperature sensor	50 to 350°F	154°F
HPCI turbine steam line high flow	Differential pressure transmitter	-500 to +500 in. H ₂ O	+425 in. H ₂ O -425 in. H ₂ O
HPCI turbine steam line low pressure	Pressure transmitter	0 to 200 psi	110 psig
Fuel pool ventilation exhaust high radiation	Radiation monitor	0.01 to 100 mR/hr	10 mR/hr
Reactor water cleanup system space high temperature	Temperature sensor	50 to 350°F	175°F

^a Nominal values are given for information. See Technical Specifications and/or Technical Requirements Manual, Vol I for actual operational limits.

^b See Figure 7.3-12 Sheet 3.

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TABLE 7.3-10 PRIMARY CONTAINMENT AND REACTOR VESSEL ISOLATION CONTROL SYSTEM: MINIMUM NUMBERS OF TRIP CHANNELS REQUIRED FOR FUNCTIONAL PERFORMANCE^a

Trip Channel Description	Normal Number of Trip Channels Per Trip System	Minimum Number of Trip Channels Required Per Untripped Trip System To Maintain Functional Performance ^b
RPV low water level (first setting) (level 3)	2	2
RPV low water level (second setting) (level 2)	2	2
RPV low water level (third setting) (level 1)	2	2
Main steam line high radiation	2	2
Main steam line space high temperature	4	4
Main steam line high flow	2/line	2/line
Main steam line low pressure	2	2
Primary containment high pressure	2	2
RCIC steam line space high temperature	1	1
RCIC steam line high flow	1	1
RCIC steam line low pressure	2	2
HPCI steam line space high temperature	1	1
HPCI steam line high flow	1	1
HPCI steam line low pressure	2	2
Fuel pool ventilation exhaust high radiation	2	2

^aThese data are derived from Technical Specifications.

^bNominal values are given for references only. See Technical Specifications for operational limits.

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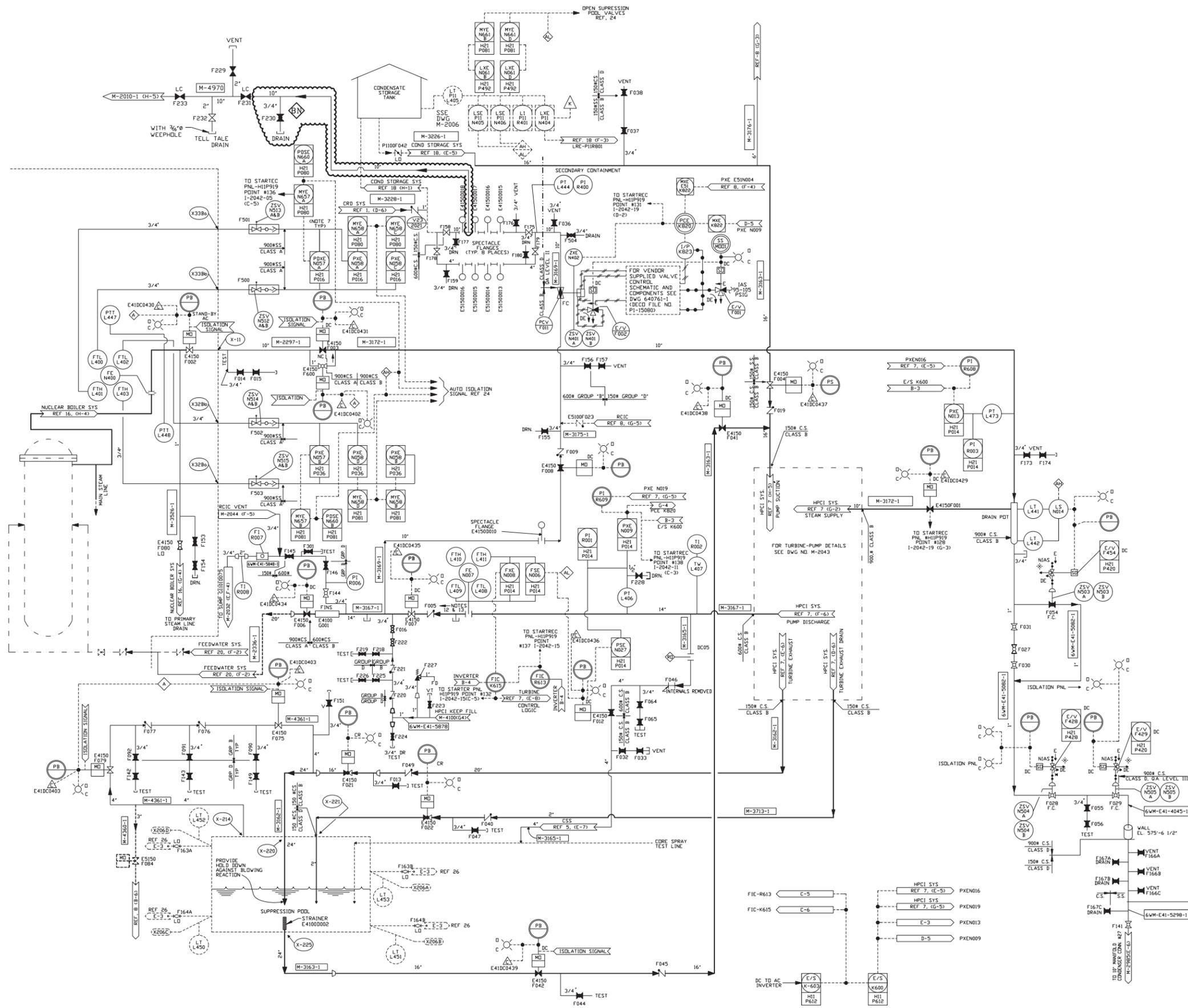
TABLE 7.3-11 (TRM – 3.3.5.1-1) EMERGENCY CORE COOLING SYSTEM
RESPONSE TIMES

The Emergency Core Cooling System Response Times are listed in Technical Requirements Manual (TRM) Volume I Table 3.3.5.1-1. TRM Volume I is incorporated by reference into the UFSAR.

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TABLE 7.3-12 (TRM TABLE 3.3.6.1-1) ISOLATION ACTUATION SYSTEM
INSTRUMENTATION RESPONSE TIME

The Isolation Actuation System Instrumentation Response Times are listed in Technical Requirements Manual (TRM) Volume I Table 3.3.6.1-1. TRM Volume I is incorporated by reference into the UFSAR.



REF NO	TITLE	DECDNO	GENO
1	DIAGRAM-CONTROL ROD DRIVE HYDRAULIC SYSTEM (CRD)	M-2081	729E612AB
2	DIAGRAM-STAND-BY LIQUID CONTROL SYSTEM (SLC)	M-2082	729E601AB
3	DIAGRAM-RESIDUAL HEAT REMOVAL SYSTEM-DIVISION I (RHRI)	M-2084	729E602AB
4	DIAGRAM-RESIDUAL HEAT REMOVAL SYSTEM-DIVISION II (RHRI)	M-2083	729E602AB
5	DIAGRAM-CORE SPRAY SYSTEM (CSS)	M-2034	731E991
6	DIAGRAM-HIGH PRESSURE COOLANT INJECTION SYSTEM (HPCI)	M-2035	729E600AB
7	DIAGRAM-HIGH PRESSURE COOLANT INJECTION SYSTEM BAROMETRIC CONDENSER (HPCIC)	M-2043	729E600AB
8	DIAGRAM-REACTOR CORE ISOLATION COOLING SYSTEM (RCIC)	M-2044	729E604AB
9	DIAGRAM-REACTOR CORE ISOLATION COOLING SYSTEM BAROMETRIC CONDENSER (RCIC)	M-2045	729E604AB
10	DIAGRAM-REACTOR WATER CLEAN-UP SYSTEM (RWCU)	M-2046	731E948AB
11	DIAGRAM-REACTOR WATER CLEAN-UP FILTER/DEMINERALIZER (RWCU)	M-2047	731E980AB
12	DIAGRAM-FUEL POND COOLING SYSTEM (FPC)	M-2048	729E609A
13	DIAGRAM-FUEL POND COOLING SYSTEM FILTER/DEMINERALIZER (FPC)	M-2049	7321844B
14	DIAGRAM-REACTOR BLDG CLOSED COOLING & ENERGY EQUIP COOLING WATER SYSTEM	M-2027	
15	DIAGRAM-DEMINERALIZED SERVICE WATER RISER SYSTEM	M-2678	
16	DIAGRAM-NUCLEAR BOILER SYSTEM	M-2089	729E616AB
17	DIAGRAM-REACTOR WATER CLEAN-UP SYSTEM PHASE SEPARATORS (RWCU)	M-2088	729E614A
18	DIAGRAM-CONDENSATE STORAGE SYSTEM	M-2006	
19	DIAGRAM-STATION AIR SYSTEM	M-2085	
20	DIAGRAM-FEEDWATER SYSTEM	M-2023	
21	DIAGRAM-REACTOR RECIRCULATION SYSTEM (RRC)	M-2833	729E603AB
22	DIAGRAM-RESIDUAL HEAT REMOVAL SERVICE WATER SYSTEM	M-2012	
23	DIAGRAM-NUCLEAR BOILER SYSTEM	M-2090	729E616AB
24	DIAGRAM-HIGH PRESSURE COOLANT INJECTION (HPCI)		729E627AB
25	LEGEND OF SYMBOLS & INST IDENT FOR PLANT SYSTEM DIAGRAMS	M-2001	
26	PRIMARY CONT MONITOR SYS	I-2679-01	
27	PIPING AND INSTRUMENT SYMBOLS	R1-25	197A567

- NOTES:**
- THIS DIAGRAM REPLACES GENERAL ELECTRIC DIAGRAM #729E600AB SH 1 REV 3.
 - SPECIFIC SYSTEM DESIGN REQUIREMENTS ARE GIVEN IN THE HIGH PRESSURE COOLANT INJECTION SYSTEM DESIGN SPECIFICATION NUMBER 3071-504.
 - THE PLANT IDENTIFICATION NUMBER FOR THE HPCI SYSTEM IS E4100.
 - FOR ASSEMBLY DWG NO. SEE KEY PLAN M-2320.
 - FOR GENERAL ARRANGEMENT SEE A-2000.
 - TURBINE SUCTION LINE IS MONITORED FOR LEAKAGE.
 - EXCESS FLOW CHECK VALVES TYP F500 HAVE LOCAL RESET AND BOTH LOCAL AND CONTROL ROOM POSITION INDICATION.
 - TORUS PENETRATIONS X206A-D ARE ALSO USED BY THE PRIMARY CONTAINMENT MONITORING SYSTEM. SEPARATE EXCESS FLOW CHECK VALVES ARE PROVIDED FOR EACH SYSTEM.
 - ALL HPCI PIPING AND EQUIPMENT SHOWN ON THIS DIAGRAM IS GA LEVEL 1 UNLESS NOTED OTHERWISE.
 - FOR VENT, DRAIN AND TEST CONNECTIONS WITH THE DOUBLE VALVE ARRANGEMENT, AFTER THE NUCLEAR VALVE (FIRST VALVE) THE CLASSIFICATION IS GROUP B, SEISMIC II, GA II (EXCEPT FOR VENT WITH VALVES E4100F032 & E4100F033).
 - VALVE E4100F005 HAS THE HARDWARE CAPABILITY TO OPERATE AS AN AIR-OPERATED, EXERCISABLE CHECK VALVE. NO AIR SUPPLY LINES ARE ROUTED TO THE VALVE. THE EXERCISABLE FUNCTION IS NOT OPERABLE.
 - VALVE E4100F005 IS AN ASME CLASS 1 VALVE INSTALLED IN A CLASS 2 APPLICATION.

TRANSMITTER NUMBER	DIV	POWER SUPPLY	TRIP UNIT
PXE-N009		K600	
PXE-N013		K600	
PXE-N016		K600	
PXE-N019		K600	
PXE-N055A.C I		B21K609A, K609C	N655A.C
PXE-N055B.D II		B21K609B, K609D	N655B.D
oPXE-N057A I		B21K609A, K609C	N657A.N660A
oPXE-N057B II		B21K609B, K609D	N657B.N660B
PXE-N058A.C I		B21K609A, K609C	N658A.C
PXE-N058B.D II		B21K609B, K609D	N658B.D
LXE-N061B.D II		B21K609B, K609D	N661B.D
LXE-N062B.D II		B21K609B, K609D	N662B.D

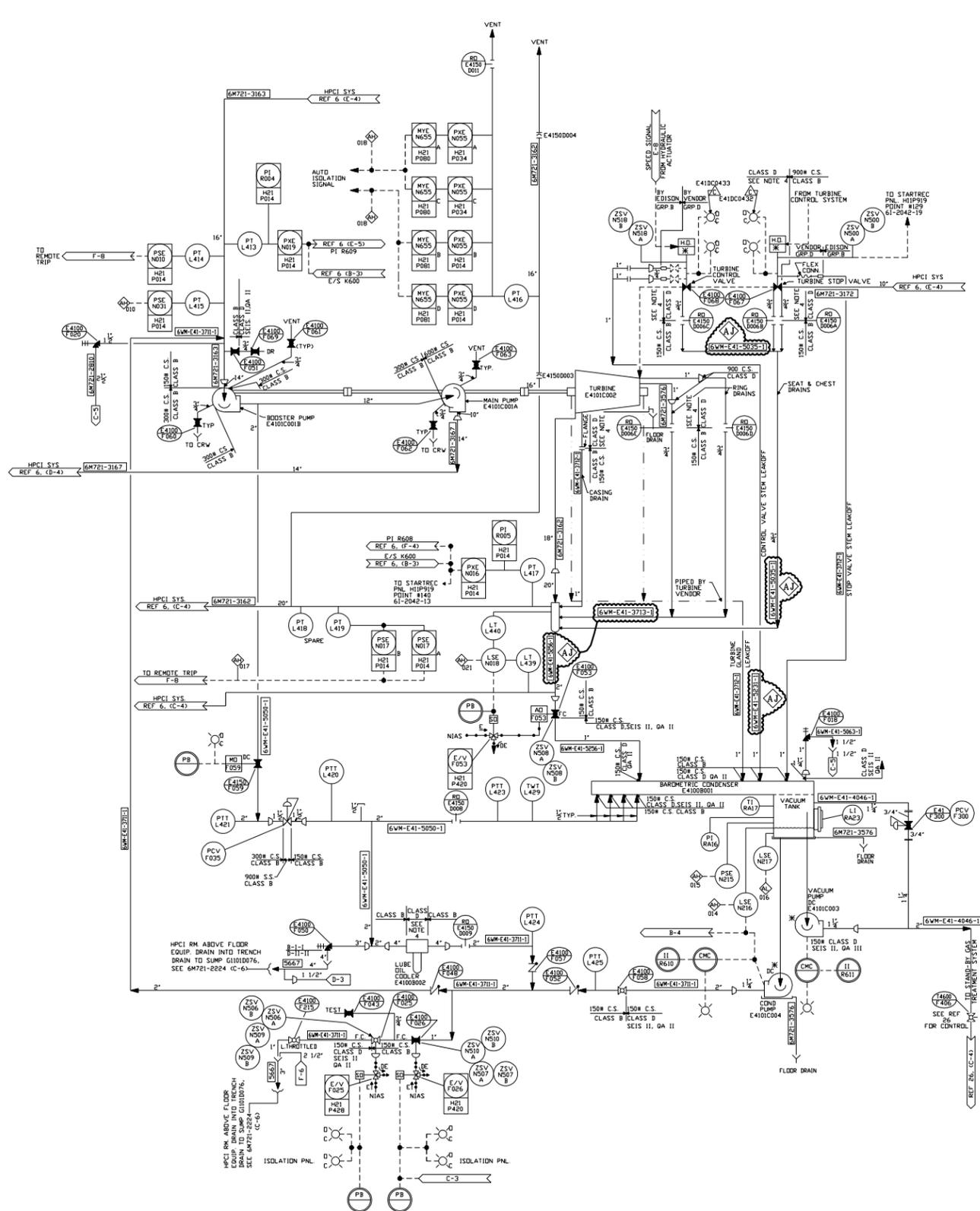
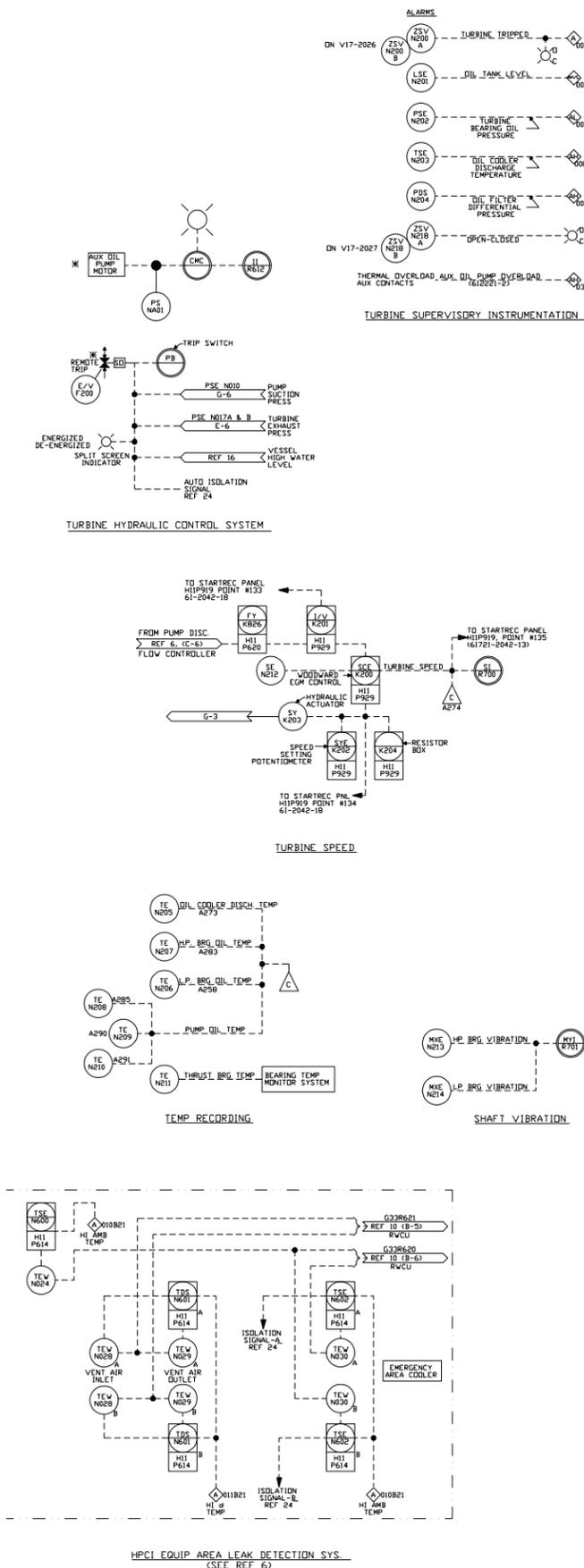
- LEGEND**
- M- DWG NO. IDENTIFIES PIPING ISOMETRIC FOR FABRICATION AND ERECTION.
- INSTRUMENT & CONTROL SYSTEM NOTES:**
- UNLESS OTHERWISE SHOWN ALL INSTRUMENT NO.'S ON THIS DIAGRAM ARE FOR SYSTEM E41.
 - NOTE REMOVED.
 - NOTE REMOVED.

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UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 7.3-1, SHEET 1

HIGH PRESSURE COOLANT INJECTION SYSTEM P&ID



DIAGRAMS

REF. NO.	TITLE	DECD. NO.	GEND.
1	DIAGRAM-CONTROL ROD DRIVE HYDRAULIC SYSTEM (CRD)	6M721-2081	729E612AB
2	DIAGRAM-STAND-BY LIQUID CONTROL SYSTEM (SLC)	6M721-2082	729E601AB
3	DIAGRAM-RESIDUAL HEAT REMOVAL SYSTEM-DIVISION I (RHRI)	6M721-2084	729E602AB
4	DIAGRAM-RESIDUAL HEAT REMOVAL SYSTEM-DIVISION II (RHRI)	6M721-2033	729E602AB
5	DIAGRAM-CORE SPRAY SYSTEM (CSS)	6M721-2034	731E901
6	DIAGRAM-HIGH PRESSURE COOLANT INJECTION SYSTEM (HPCI)	6M721-2035	729E600AD
7	DIAGRAM-HIGH PRESSURE COOLANT INJECTION SYSTEM BAROMETRIC CONDENSER (HPCI)	6M721-2043	729E600AB
8	DIAGRAM-REACTOR CORE ISOLATION COOLING SYSTEM (RCIC)	6M721-2044	729E604AB
9	DIAGRAM-REACTOR CORE ISOLATION COOLING SYSTEM BAROMETRIC CONDENSER (RCIC)	6M721-2045	729E604AB
10	DIAGRAM-REACTOR WATER CLEAN-UP SYSTEM (RWCU)	6M721-2046	731E348AB
11	DIAGRAM-REACTOR WATER CLEAN-UP FILTER/DEMINERALIZER (RWCU)	6M721-2047	731E300AB
12	DIAGRAM-FUEL POOL COOLING SYSTEM (FPC)	6M721-2048	729E699AA
13	DIAGRAM-FUEL POOL COOLING SYSTEM FILTER/DEMINERALIZER (FPC)	6M721-2049	731E104AB
14	DIAGRAM-REACTOR BLDG CLOSED COOLING & DRYING EQUIP. COOLING WATER SYSTEM	6M721-2027	
15	DIAGRAM-DEMINERALIZED SERVICE WATER RISER SYSTEM	6M721-2670	
16	DIAGRAM-NUCLEAR BOILER SYSTEM	6M721-2089	729E616AB
17	DIAGRAM-REACTOR WATER CLEAN-UP SYSTEM PHASE SEPARATORS (RWCU)	6M721-2088	729E614AA
18	DIAGRAM-CONDENSATE STORAGE SYSTEM	6M721-2086	
19	DIAGRAM-STATION AIR SYSTEM	6M721-2085	
20	DIAGRAM-FEED WATER SYSTEM	6M721-2023	
21	DIAGRAM-REACTOR RECIRCULATION SYSTEM (RR)	6M721-2833	729E603AB
22	DIAGRAM-RESIDUAL HEAT REMOVAL SERVICE WATER SYSTEM	6M721N-2012	
23	DIAGRAM-NUCLEAR BOILER SYSTEM	6M721-2090	729E616AB
24	DIAGRAM-HIGH PRESSURE COOLANT INJECTION (HPCI)	6M721-2042-182	729E627AB
25	LEGEND OF SYMBOLS & INSTR. IDENT. FOR PLANT SYSTEM DIAGRAMS	6M721-2001	
26	DIAGRAM-STANDBY GAS TREATMENT	7M721-2709	
27	INDEX FOR STARTREC POINTS	6M721-2042-182	
28	PIPING AND INSTRUMENT SYMBOLS	RI-25	1978567

- NOTES:
- CHEMICAL CLEANING CONNECTIONS, VALVE, ETC. REQUIRED ARE TO BE PROVIDED AS NECESSARY.
 - PIPING HIGH POINT VENTS & LOW POINT DRAINS TO BE ADDED AS NECESSARY.
 - PIPING SHOWN IN PHANTOM TO BE ARRANGED AND SUPPLIED BY TURBINE VENDOR.
 - THIS NON-ISOLABLE PORTION OF PIPING SHALL NOT BE HYDROSTATICALLY TESTED AND SHALL BE CLASSIFIED AS CLASS D PIPING. THE TURBINE AND ASSOCIATED PIPING AND VALVES ARE NON-ASME COMPONENTS—PER ASME SECTION III, 1971 CODE, ARTICLE NA-1130. HOWEVER, THIS PIPING IS SEISMIC I, GA LEVEL I, AND MEETS THE CODE REQUIREMENTS FOR CLASS B PIPING EXCEPT FOR THE "N" STAMP AND HYDRO-TESTING.
 - ALL HPCI EQUIPMENT AND PIPING COMPONENTS ARE GA LEVEL I UNLESS NOTED OTHERWISE.
 - THIS DIAGRAM REPLACES GENERAL ELECTRIC DIAGRAM #729E600AB SH. 2 REV.3
 - SPECIFIC SYSTEM DESIGN REQUIREMENTS ARE GIVEN IN THE HIGH PRESSURE COOLANT INJECTION SYSTEM SPECIFICATION NUMBER 3071-504.
 - THE PLANT IDENTIFICATION NUMBER FOR THE HPCI SYSTEM IS E4100.
 - FOR ASSEMBLY DWG. NOS. SEE KEY PLAN 6M721-2320.
 - FOR GENERAL ARRANGEMENT SEE 7A721-2000.
 - SEE 6M721-2035 FOR TABLE ON TRANSMITTER, POWER SUPPLY & TRIP UNIT.

LEGEND

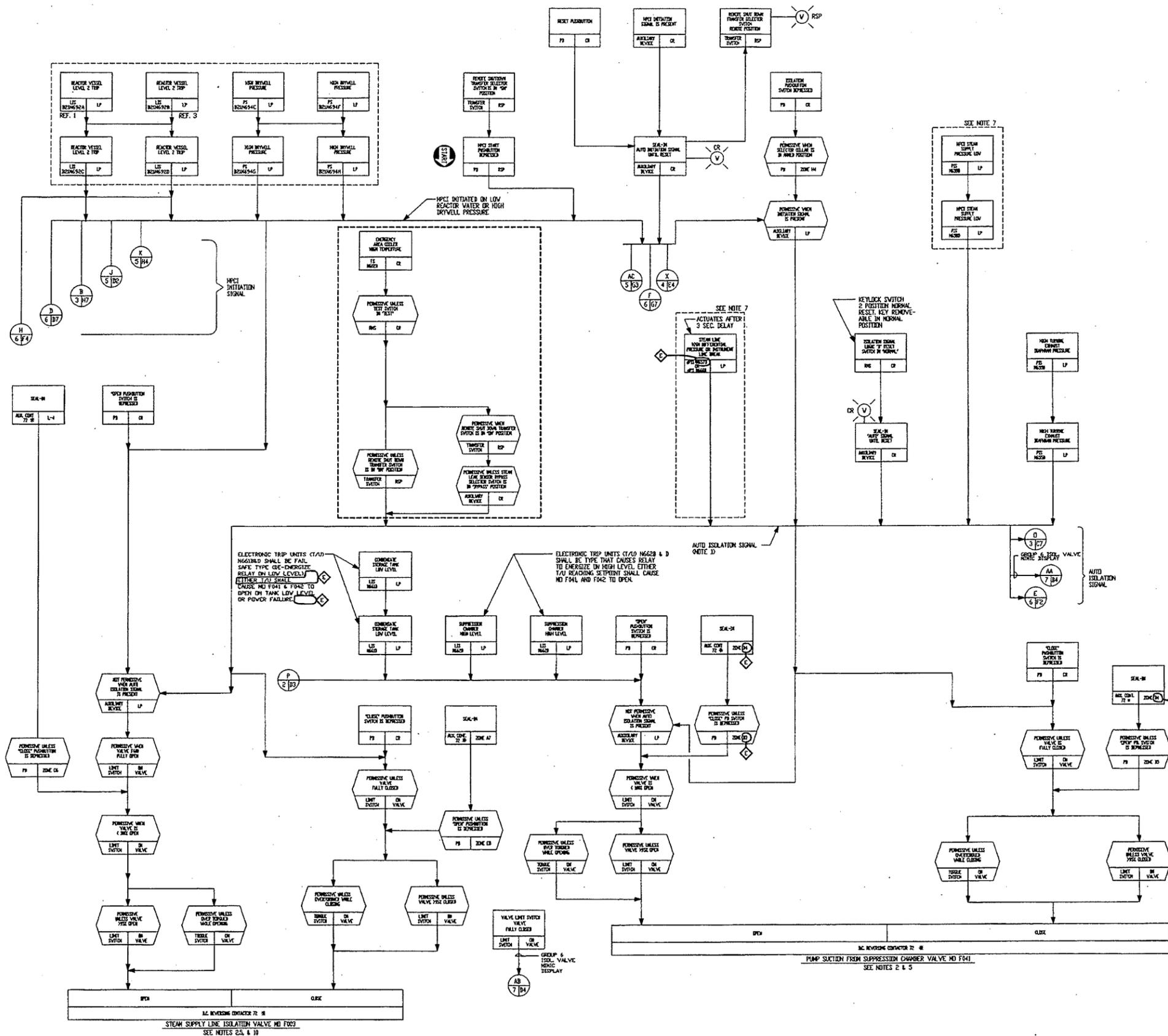
— DWG. NO.

1. 6M721- IDENTIFIES PIPING ISOMETRIC FOR FABRICATION & ERECTION.

- INSTRUMENT & CONTROL SYSTEM
- NOTES:
- UNLESS OTHERWISE SHOWN, ALL INSTRUMENT NOS. ON THIS DIAGRAM ARE FOR SYSTEM E41.
 - NOTE REMOVED
 - NOTE REMOVED
 - AN * INDICATES DEVICE OR INSTRUMENT IS SUPPLIED AS PART OF TURBINE E410C002.

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FIGURE 7.3-1, SHEET 2
HIGH PRESSURE COOLANT INJECTION SYSTEM P&ID



- NOTES:**
1. THE HPCI SYSTEM IS ARRANGED FOR TEST OF PUMP AT FULL FLOW & ALL VALVES FOR OPEN AND CLOSE CAPACITY AT ANY TIME EXCEPT WHEN INITIATION SIGNAL OR AUTO ISOLATION SIGNAL IS ACTIVATED. IN EVENT THE INITIATION SIGNAL OCCURS WHILE TEST IS UNDERWAY THE SYSTEM AUTOMATICALLY RETURNS TO START-UP MODE.
 2. ALL POWER FOR OPERATION OF DC VALVE MOTORS SHALL ORIGINATE FROM A PLANT DC BUS. POWER FOR AC MOTOR OPERATED VALVES SHALL ORIGINATE FROM ANY EMERGENCY AC BUS.
 3. UNLESS OTHERWISE SHOWN ALL INSTRUMENT PIS NUMBERS ARE PREFIXED E410. ALL VALVE AND EQUIPMENT PIS NUMBERS ARE PREFIXED E4100. E4150 (AS APPLICABLE).
 4. DELETED
 5. AUXILIARY RELAYS & DEVICES NOT SHOWN IN LOGIC DIAGRAMS EXCEPT WHERE REQUIRED TO CLARIFY FUNCTION.
 6. DELETED
 7. ISOLATION SIGNAL SWITCHES SHALL BE OF THE TYPE THAT CLOSE CONTACTS FOR THE SPECIFIED ISOLATION EVENT, WHERE AUXILIARY RELAYS ARE USED IN THE ISOLATION CHANNELS, THEY SHALL BE POWERED FROM THE STATION BATTERIES G02740.
 8. FOR LOCATION AND IDENTIFICATION OF INSTRUMENTS SEE (E001).
 9. DELETED
 10. AN ALTERNATE SOURCE OF EMERGENCY AC POWER SHALL BE PROVIDED TO THE ISOLATION VALVE NO F102.
 11. THE HPCI SYSTEM SHALL BE DESIGNED IN ACCORDANCE WITH THE GENERAL CRITERIA FOR NUCLEAR POWER PLANT PROTECTION SYSTEM IEEE 279.
 12. FOR CONFIGURATION OF PANEL MOUNTED OPERATORS AND INDICATORS SEE DECO DWG. M-2290 SH 2, REF 15, PURCHASE SPECIFICATION DATA SHEET SECTION 2.A.
 13. ALL MOTORS SHALL BE PROTECTED BY THERMAL OVERLOAD TRIPS AND BREAKER SHORT CIRCUIT CURRENT PROTECTIVE TRIPS.

REFERENCE DOCUMENTS

1. NUCLEAR BOILER P&ID.....	M-2889, M-2930, M-3538
2. NUCLEAR BOILER SYSTEM LOGIC DIAGRAM.....	I-2090
3. RHR SYSTEM P&ID.....	M-2883, M-2884
4. RHR SYSTEM LOGIC DIAGRAM.....	I-2200
5. CORE SPRAY SYSTEM LOGIC DIAGRAM.....	I-2210
6. HPCI SYSTEM P&ID.....	M-2825, M-2843
7. HPCI SYSTEM LOGIC DIAGRAM.....	I-2220
8. HPCI TURBINE CONT. SYSTEM.....	TERRY TURBINE MANUAL - VWR4-2 SERIES
9. DELETED	
10. LOGIC SYMBOLS.....	209447-56 R1-20
11. REACTOR VAPOR CLEANUP SYSTEM P&ID.....	M-2846, M-2847
12. LEAK DETECTION SYSTEM DESIGN SPECIFICATION.....	20A441, R1-138
13. RCIC SYSTEM P&ID.....	M-2844, M-2845
14. ELECTRICAL EQUIPMENT SEPARATION FOR SAFEGUARD SYSTEM.....	20A3777, R1-1410
15. GENERAL REQUIREMENTS FOR INSTRUMENTATION & ELECTRICAL EQUIPMENT.....	20A1768, R1-446
16. HPCI SCHEMATIC DIAGRAMS.....	I-2225-01 THRU 15
17. LOGIC DIAGRAM-GROUP 6-HPCI ISOL. VALVE HMDIC DISPLAY.....	I-2220-07
18. LOGIC DIAGRAM-GROUP 7-HPCI ISOL. VALVE HMDIC DISPLAY.....	I-2220-08

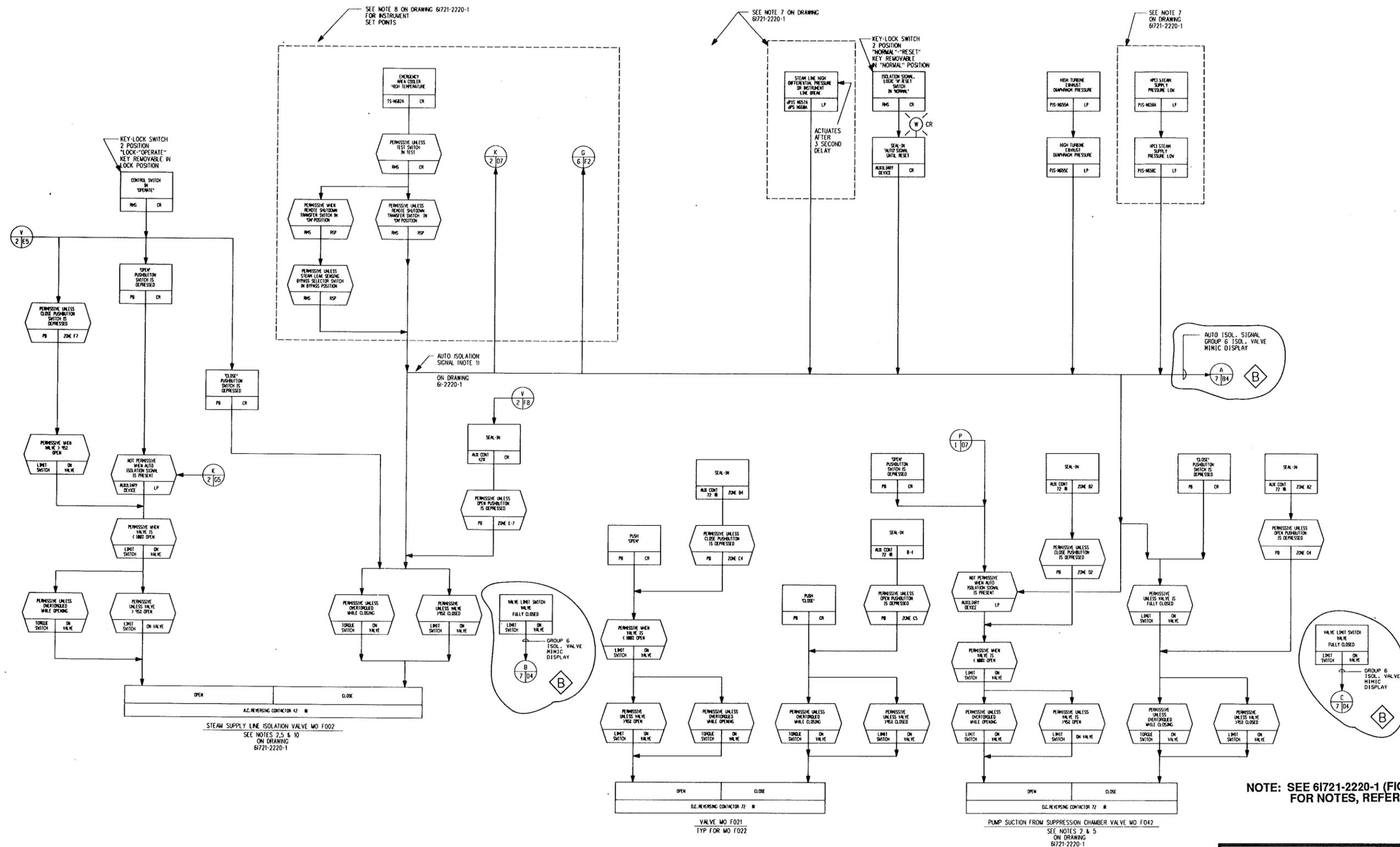
- LEGEND**
- * = SWITCHGEAR SERVICE FUNCTION IN ACCORDANCE WITH SPEC. C37.2
 - ⊗ = FURNISHED WITH TURBINE (SEE REF. DRAWING 8)
 - ⊕ = COORDINATED MANUAL CONTROL SWITCH
 - PB = PUSHBUTTON SWITCH
 - AM = AMMETER
 - RSP = REMOTE SHUTDOWN PANEL
 - RMS = REMOTE MANUAL SWITCH
 - CR = CONTROL ROOM OR RELAY ROOM
 - LP = LOCAL
 - X = LETTER DESIGNATION
 - X 100 = SHEET NO. OR REFERENCE DRAWING NO.
 - MATCH CIRCLE

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FIGURE 7.3-2, SHEET 1

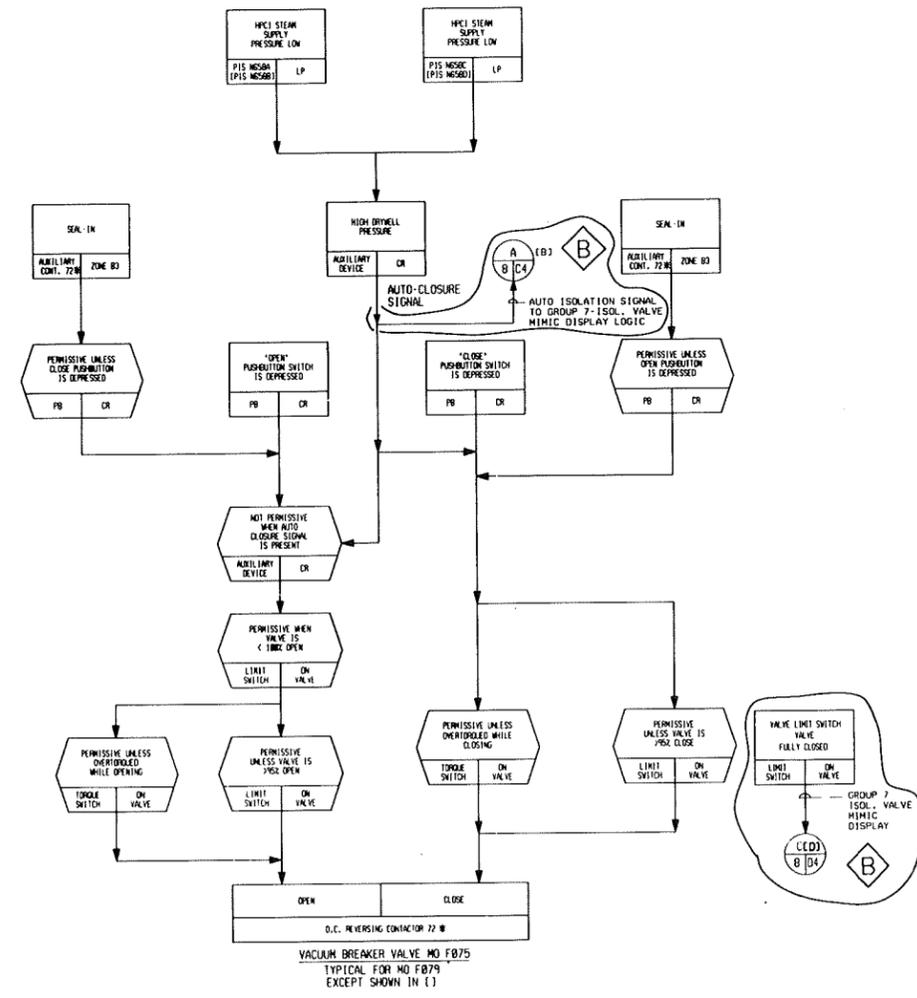
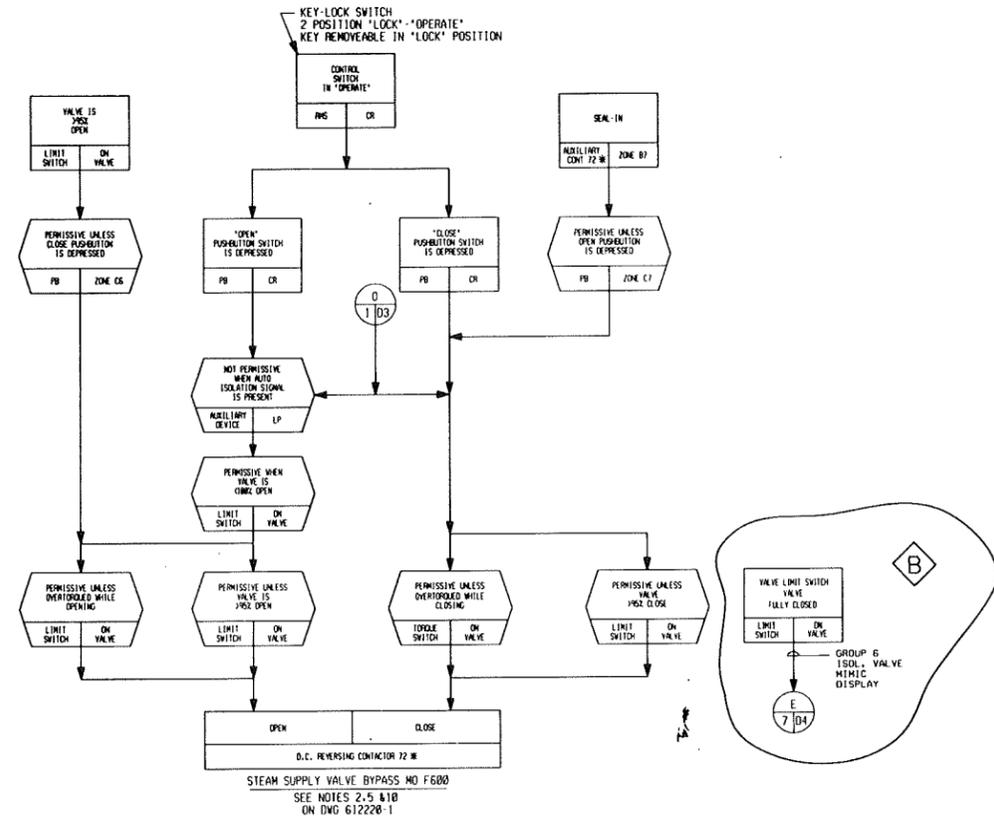
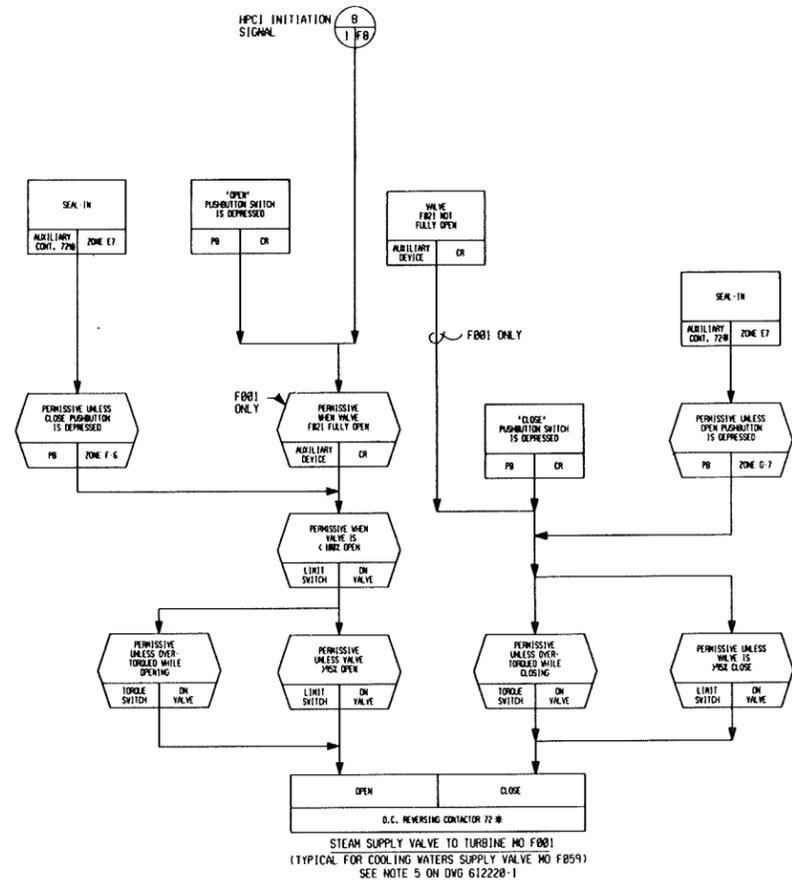
HIGH PRESSURE COOLANT INJECTION SYSTEM FCD



NOTE: SEE 61721-2220-1 (FIGURE 7.3-2, SHEET 1) FOR NOTES, REFERENCES AND LEGEND.

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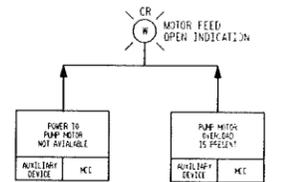
FIGURE 7.3-2, SHEET 2
 HIGH PRESSURE COOLANT INJECTION SYSTEM
 FCD



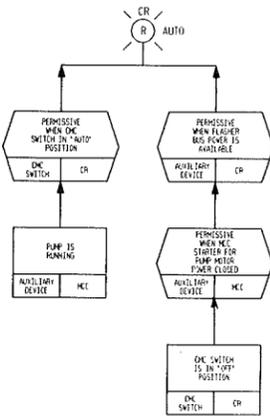
NOTE: SEE 61721-2220-1 (FIGURE 7.3-2, SHEET 1) FOR NOTES, REFERENCES AND LEGEND.

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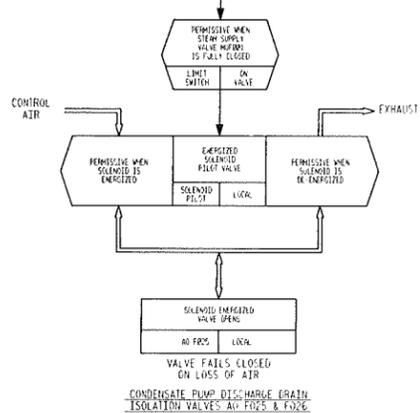
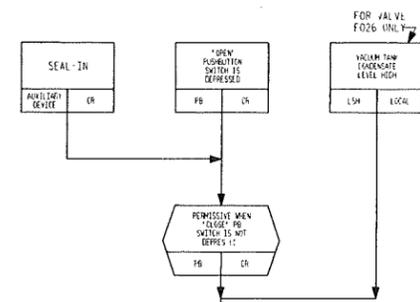
FIGURE 7.3-2, SHEET 3
HIGH PRESSURE COOLANT INJECTION SYSTEM FCD



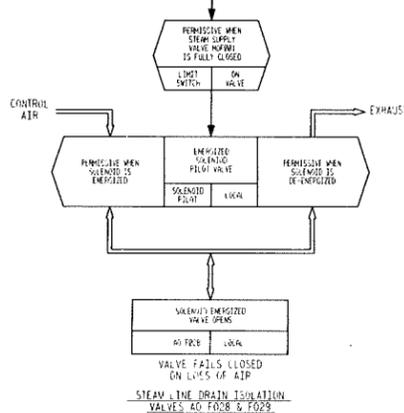
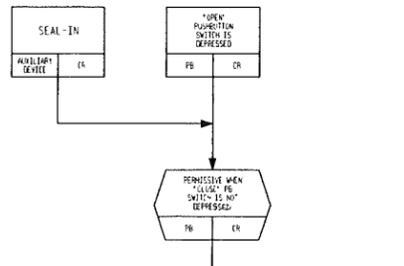
MOTOR FEED OPEN INDICATION
TYPICAL FOR HPCEI OIL PUMP E4101-C005,
BAROMETRIC CONDENSER VACUUM PUMP
E4101-C003 & CONDENSATE PUMP E4101-C004.



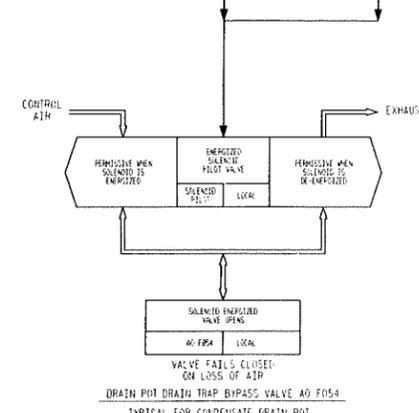
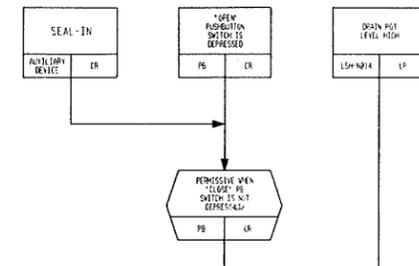
AUTOFLASHER LIGHT
FOR HPCEI OIL PUMP E4101-C005,
BAROMETRIC CONDENSER VACUUM
PUMP E4101-C003 & CONDENSATE
PUMP E4101-C004.



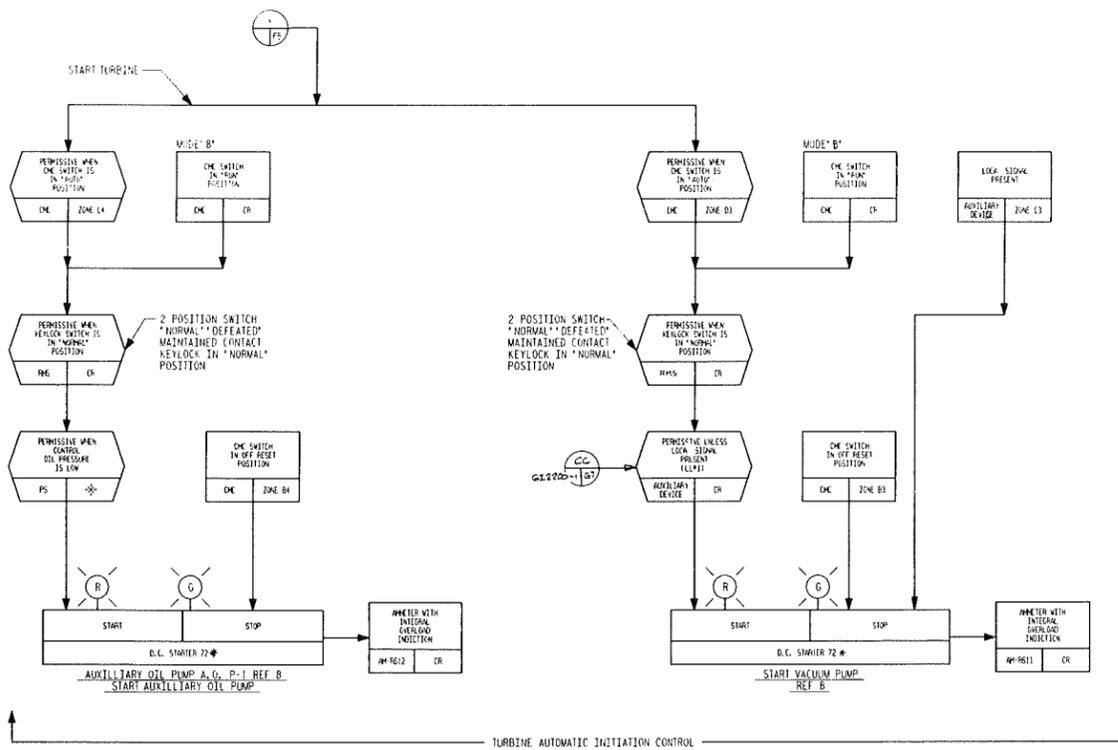
CONDENSATE PUMP DISCHARGE DRAIN
ISOLATION VALVES: AO F025 & F026



STEAM LINE DRAIN ISOLATION
VALVES: AO F028 & F029



DRAIN POT DRAIN TRAP BYPASS VALVE AO F054
TYPICAL FOR CONDENSATE (RAIN POT)
VALVE AO F053



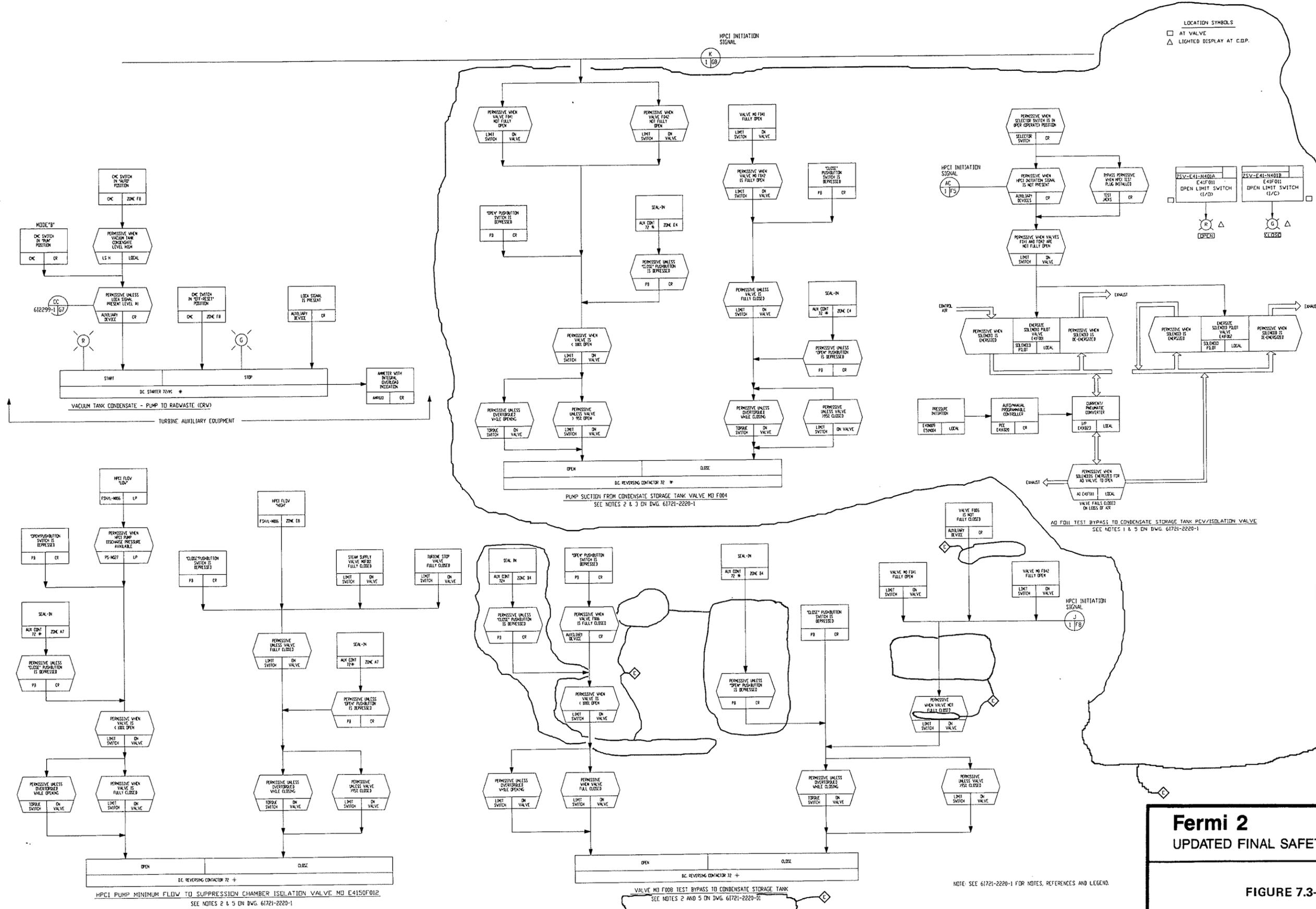
NOTE: SEE 61721-2220-1 (FIGURE 7.3-2, SHEET 1)
FOR NOTES, REFERENCES AND LEGEND.

THIS FIGURE IS NOT UPDATED PERIODICALLY

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FIGURE 7.3-2, SHEET 4

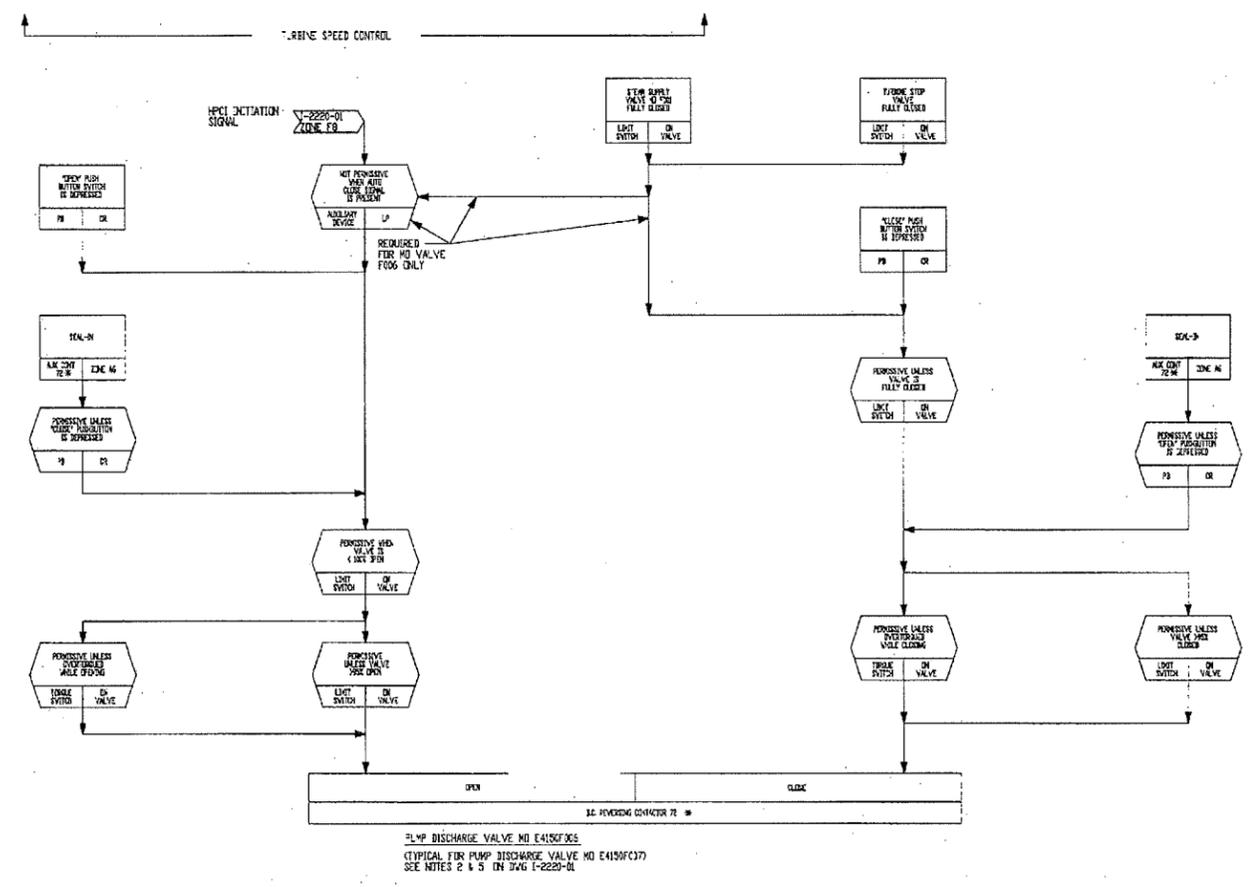
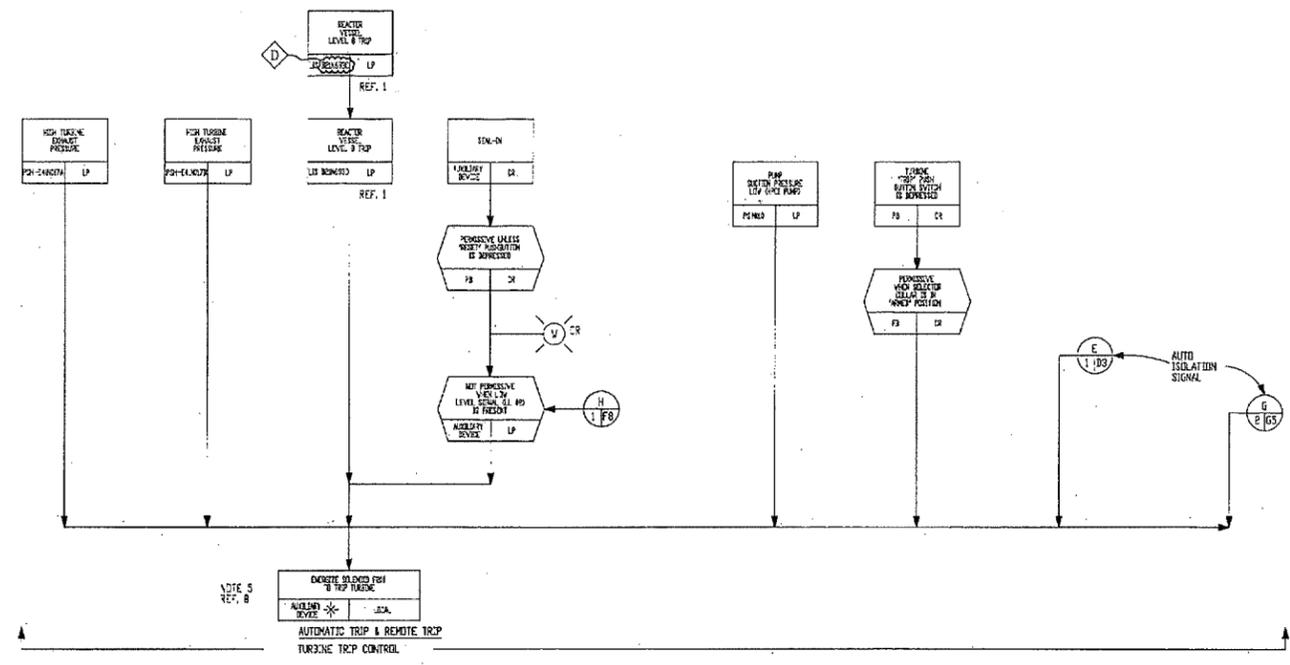
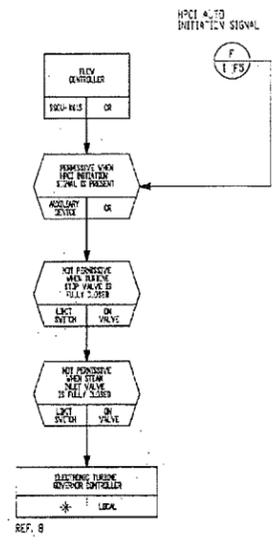
HIGH PRESSURE COOLANT INJECTION SYSTEM FCD



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FIGURE 7.3-2, SHEET 5

HIGH PRESSURE COOLANT INJECTION SYSTEM FCD



NOTE: SEE 61721-2220-1 (FIGURE 7.3.2, SHEET 1) FOR NOTES, REFERENCES AND LEGEND.

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FIGURE 7.3-2, SHEET 6

HIGH PRESSURE COOLANT INJECTION SYSTEM FCD

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FIGURE 7.3-3 HAS BEEN DELETED

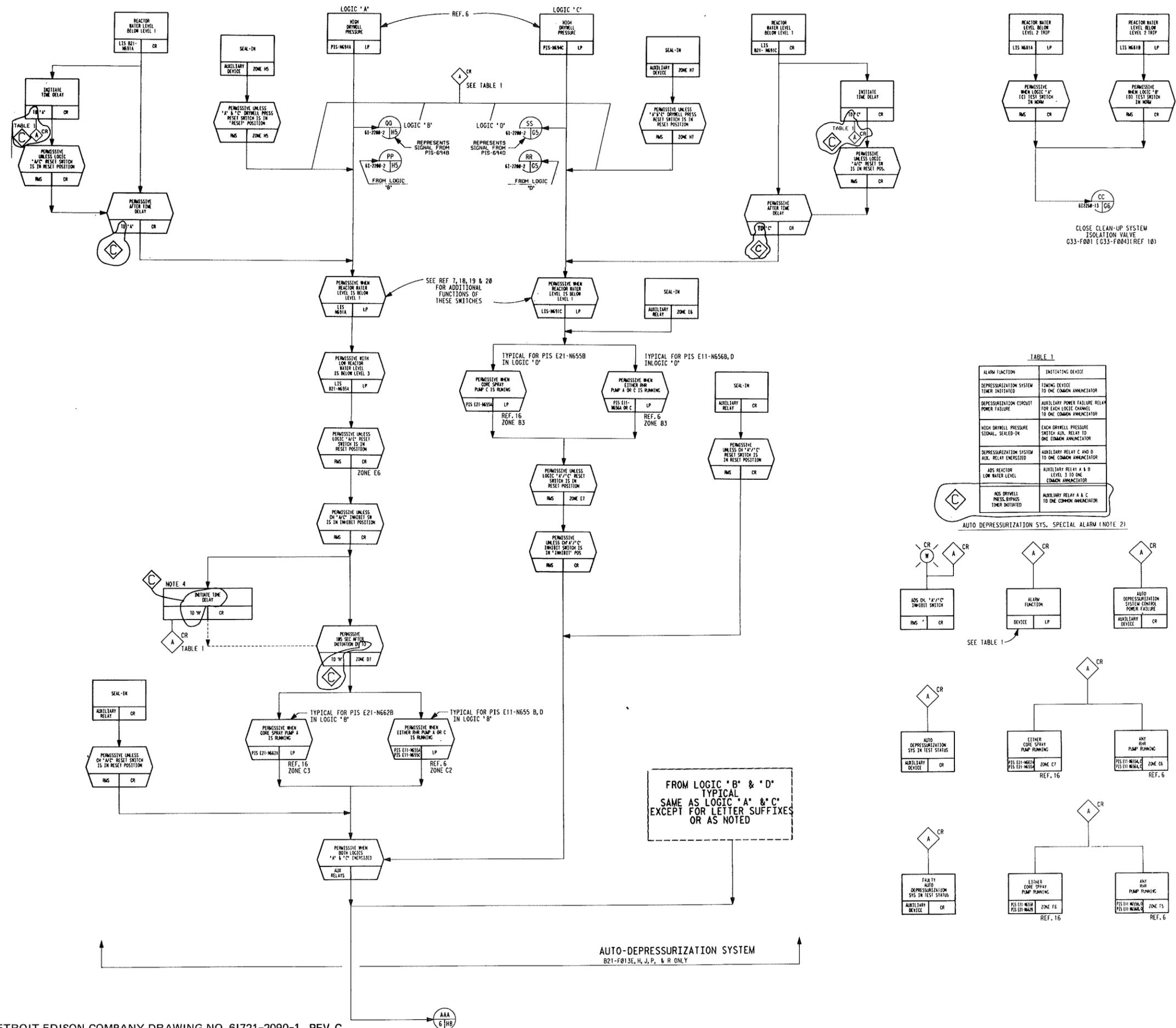
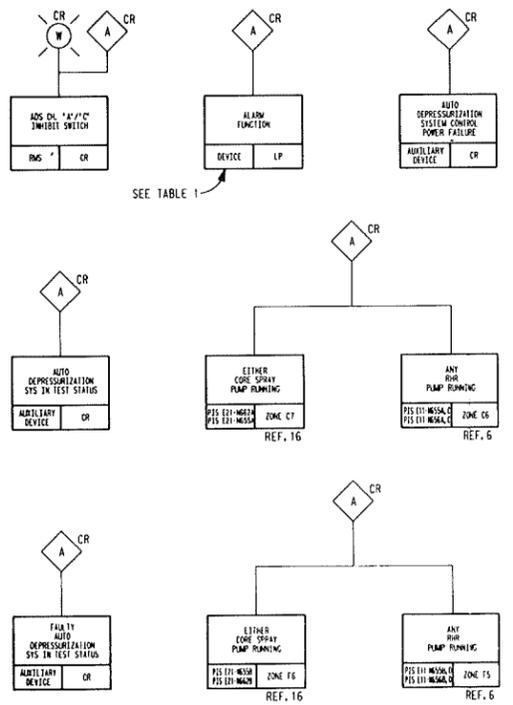


TABLE 1

ALARM FUNCTION	INITIATING DEVICE
DEPRESSURIZATION SYSTEM TIMER INITIATED	TIMING DEVICE TO ONE COMMON ANNUNCIATOR
DEPRESSURIZATION CIRCUIT POWER FAILURE	AUXILIARY POWER FAILURE RELAY FOR EACH LOGIC CHANNEL TO ONE COMMON ANNUNCIATOR
HIGH DRYWELL PRESSURE SIGNAL, SEALED-IN	EACH DRYWELL PRESSURE SWITCH AUX. RELAY TO ONE COMMON ANNUNCIATOR
DEPRESSURIZATION SYSTEM AUX. RELAY ENERGIZED	AUXILIARY RELAY C AND D TO ONE COMMON ANNUNCIATOR
ADS REACTOR LOW WATER LEVEL	AUXILIARY RELAY A & B LEVEL 3 TO ONE COMMON ANNUNCIATOR
ADS DRYWELL PRESS. BIPHAS TIMER INITIATED	AUXILIARY RELAY A & C TO ONE COMMON ANNUNCIATOR



- NOTES
- WHEN TEST SOLENOID PILOT IS ENERGIZED, THE MAIN STEAM ISOLATION VALVE OPERATOR IS SLOWLY EXHAUSTED (60 SECOND CLOSURE TIME) WHILE VALVE IS CLOSED BY ACTION OF THE VALVE SPRING WITHOUT AID OF AIR PRESSURE.
 - THE ALARMS AND VALVE INDICATING LIGHTS SHOWN ON THE LOGIC DIAGRAM ARE SYSTEM REQUIREMENTS IN ADDITION TO THOSE SHOWN ON THE SYSTEM P&ID. ADDITIONAL INFORMATION ON ALARMS, VALVE POSITION INDICATING LIGHTS AND PROCESS INSTRUMENTATION NOT SHOWN ON THIS LOGIC DIAGRAM MAY BE OBTAINED FROM REFERENCE #1. AUXILIARY RELAY & DEVICES ARE NOT SHOWN ON LOGIC DIAGRAM EXCEPT WHERE REQUIRED TO CLARIFY FUNCTION.
 - MAIN STEAM LINE ISOLATION LOGIC A, B, C, D AND AC SOLENOID LOGIC SHALL BE POWERED FROM REACTOR PROTECTION SYSTEM W-C SETS. RC SOLENOID LOGIC SHALL BE POWER FROM STATION BATTERIES.
 - TIMING DEVICES SHALL BE A TYPE THAT RESETS ON LOSS OF POWER OR LOSS OF INITIATING SIGNAL.
 - ALL EQUIPMENT AND INSTRUMENTS ARE PREFIXED BY THE SYSTEM NUMBER 'B21' UNLESS OTHERWISE NOTED.
 - FOR LOCATION AND IDENTIFICATION OF INSTRUMENTS SEE INSTRUMENT DATA SHEET LISTED IN MASTER PARTS LIST (MPL) FOR EACH INSTRUMENT.
 -
 - THE NUCLEAR BOILER SYSTEM SHALL BE DESIGNED IN ACCORDANCE WITH REFERENCE #17 AND WITH *PROPOSED CRITERIA FOR NUCLEAR POWER PLANT PROTECTION SYSTEMS IEEE 279* AS APPLICABLE TO THE CONTROL CIRCUITRY.
 - ISOLATION LOGIC SHALL BE 'FAILSAFE' IE., LOGIC SHALL BE DESIGNED TO INITIATE ISOLATION FUNCTIONS WHEN DE-ENERGIZED.
 - FOR CONFIGURATION OF PANEL MOUNTED OPERATORS AND INDICATORS SEE DRAWING 6M-2298 SHEET 2, REFERENCE #21, AND PURCHASE SPECIFICATION DATA SHEET SECTION 2, 4.
 - THE DEVICES IN THIS AREA AS WELL AS OTHER DEVICES IN THIS LOGIC ARE ESSENTIAL AND MUST MEET THE REQUIREMENTS OF IEEE-279.
 - FOR TRIP SETTINGS SEE NUCLEAR BOILER SYSTEM DESIGN SPECIFICATION DATA SHEET REFERENCE #26.
 - TYPICAL FOR EACH VALVE. FOR VALVE NUMBERS AND DEVICE SUFFIX LETTERS SEE REFERENCE #1.
 - ADDITIONAL NUCLEAR BOILER SYSTEM LOGIC IS SHOWN ON REFERENCES #4, #7, #18, #19 & #20.
 -

REFERENCE DOCUMENTS

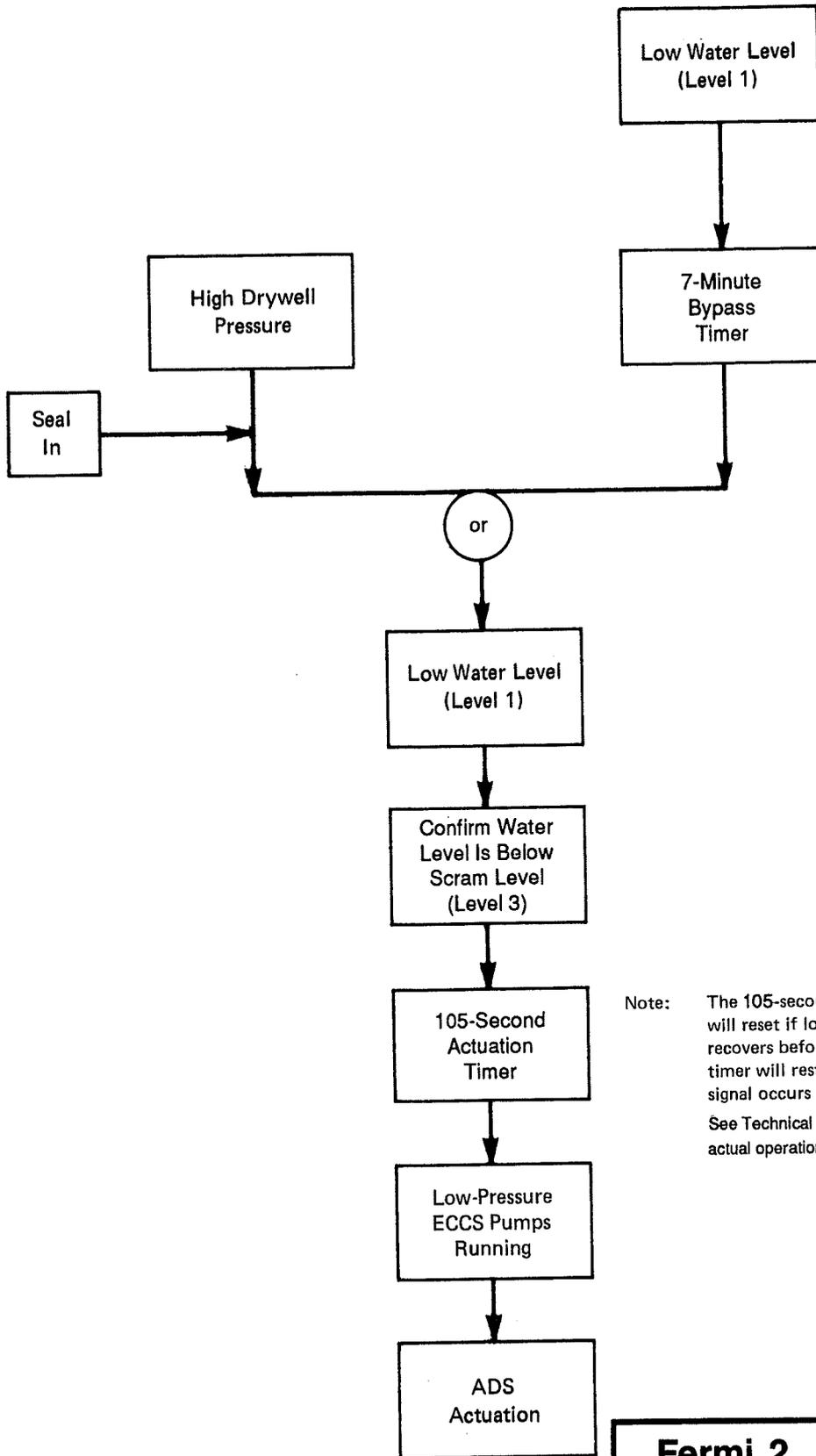
- NUCLEAR BOILER SYSTEM P&ID.....6M-2089, 6M-2090, 6M-5538
- REACTOR RECIRCULATION SYSTEM LOGIC DIAGRAM.....6I-2100
- NEUTRON MONITORING SYSTEM LOGIC DIAGRAM.....6I-2140
- REACTOR PROTECTION SYSTEM LOGIC DIAGRAM.....6I-2156-1, 2
- PROCESS RADIATION SYSTEM IED.....6I-2181-1 THRU 6
- RHR SYSTEM P&ID.....6M-2083, 6M-2084
- RHR SYSTEM LOGIC DIAGRAM.....6I-2200
-
- REACTOR WATER CLEANUP SYSTEM P&ID.....6M2046, 6M2047
- REACTOR WATER CLEANUP SYSTEM LOGIC DIAGRAM.....6I-2260
- NUCLEAR BOILER SYSTEM DESIGN SPEC. DATA SHEET.....22A2919AB; (R1-207)
- NUCLEAR BOILER SYSTEM DESIGN SPEC.....22A2919; (R1-200)
- LOGIC SYMBOLS.....209A4756; (R1-20)
- PIPING & INSTRUMENT SYMBOLS.....197R567; (R1-2507)
- REACTOR RECIRCULATION SYSTEM P&ID.....6M-2033, 6I-2106-1, 2
- CORE SPRAY P&ID.....6M-2034
- ELECTRICAL EQUIPMENT SEPARATION FOR SAFEGUARD SYS.....22A3777; (R1-1410)
- CORE SPRAY LOGIC DIAGRAM.....6I-2210
- HPCI LOGIC DIAGRAM.....6I-2220
- RCIC LOGIC DIAGRAM.....6I-2230
- GENERAL REQUIREMENTS FOR INSTR & ELECT EQUIP.....22A1769; (R1-446, 483)
- BWR PLANT REQUIREMENT.....22A1300BC; (R1-3231)
- STEAM LEAK DETECTION SYSTEM.....22A1441; (R1-138)
-22A1441AE; (R1-139)
-
- STANDBY LIQUID CONTROL SYSTEM LOGIC DIAGRAM.....6I2130
- NUCLEAR BOILER SYSTEM DESIGN SPEC DATA SHEET.....22A2919AB; (R1-207)
- NUCLEAR BOILER SYSTEM DESIGN SPEC.....22A2919; (R1-200)
- LOGIC DIAGRAM-GROUP 1 ISOL. VALVE MIMIC DISPLAY...41721-2090-8
- LOGIC DIAGRAM-GROUP 4 ISOL. VALVE MIMIC DISPLAY...41721-2200-13
- LOGIC DIAGRAM-GROUP 13 ISOL. VALVE MIMIC DISPLAY...41721-2160-72
- LOGIC DIAGRAM-GROUP 17 ISOL. VALVE MIMIC DISPLAY...41721-2100-14
- LOGIC DIAGRAM-GROUP 2 ISOL. VALVE MIMIC DISPLAY...41721-2100-13

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FIGURE 7.3-4
 AUTOMATIC DEPRESSURIZATION SYSTEM FCD

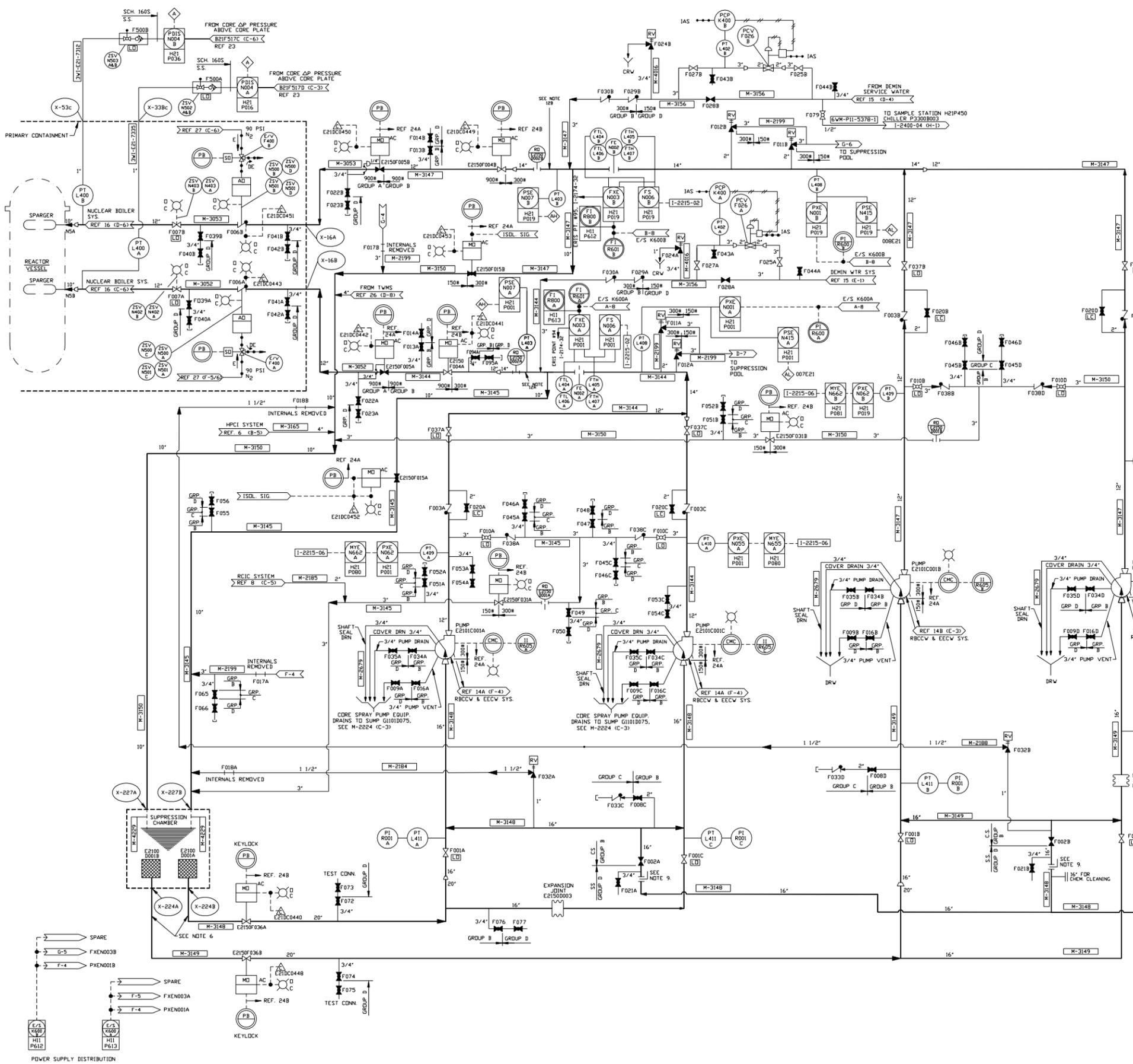
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FIGURE 7.3-5 INTENTIONALLY DELETED



Note: The 105-second actuation timer will reset if low-water-level trip recovers before it times out. The timer will restart if the low-level signal occurs again.
See Technical Specifications for actual operational settings.

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FIGURE 7.3-6 AUTOMATIC DEPRESSURIZATION SYSTEM DELAYED BYPASS OF DRYWELL PRESSURE TRIP WITH BYPASS TIMER STARTED AT LOW WATER LEVEL (LEVEL 1)



DIAGRAMS			
REF. NO.	TITLE	DECD. NO.	G.E. NO.
1	DIAGRAM - CONTROL ROD DRIVE HYDRAULIC SYSTEM (SLS)	M-2081	729E612AB
2	DIAGRAM - STAND-BY LIQUID CONTROL SYSTEM (SLS)	M-2082	729E610AB
3	DIAGRAM - RESIDUAL HEAT REMOVAL SYSTEM DIVISION I (RHRI)	M-2084	729E602AB
4	DIAGRAM - RESIDUAL HEAT REMOVAL SYSTEM DIVISION II (RHRI)	M-2083	729E602AB
5	DIAGRAM - CORE SPRAY SYSTEM (CSS)	M-2034	731E981
6	DIAGRAM HIGH PRESSURE COOLANT INJECTION SYSTEM (HPCI)	M-2025	729E600AB
7	DIAGRAM - HIGH PRESSURE COOLANT INJECTION SYSTEM BAROMETRIC CONDENSER (HPCI)	M-2043	729E600AB
8	DIAGRAM - REACTOR CORE ISOLATION COOLING SYSTEM (RCIC)	M-2044	729E604AB
9	DIAGRAM - REACTOR CORE ISOLATION COOLING SYSTEM BAROMETRIC CONDENSER (RCIC)	M-2045	729E604AB
10	DIAGRAM - REACTOR WATER CLEAN-UP SYSTEM (RWCU)	M-2046	731E348AB
11	DIAGRAM - REACTOR WATER CLEAN-UP FILTER/DEMINERALIZER (RWCU)	M-2047	731E380AB
12	DIAGRAM - FUEL POOL COOLING SYSTEM (FPC)	M-2048	729E609A
13	DIAGRAM - FUEL POOL COOLING SYSTEM FILTER/DEMINERALIZER (FPC)	M-2049	7321848AB
14A	DIAGRAM - REACTOR BLDG CLOSED COOLING & EMERG. EQUIP. COOLING WATER SYSTEM DIV. I	M-5544	
14B	DIAGRAM - REACTOR BLDG CLOSED COOLING & EMERG. EQUIP. COOLING WATER SYSTEM DIV. II	M-5557	
15	DIAGRAM - DEMINERALIZED SERVICE WATER RISER SYSTEM	M-2678	
16	DIAGRAM - NUCLEAR BOILER SYSTEM	M-2089	729E616AB
17	DIAGRAM - REACTOR WATER CLEAN-UP SYSTEM PHASE SEPARATORS (RWCU)	M-2088	729E614AB
18	DIAGRAM - CONDENSATE STORAGE SYSTEM	M-2006	
19	DIAGRAM - STATION AIR SYSTEM	M-2085	
20	DIAGRAM - FEED WATER SYSTEM	M-2023	
21	DIAGRAM - REACTOR RECIRCULATION SYSTEM (RRCS)	M-2033	729E603AB
22			
23	DIAGRAM - NUCLEAR BOILER SYSTEM	M-2090	SEE REF. 16 729E616AB
24A	LOGIC DIAGRAM - CORE SPRAY SYSTEM	I-2210-01	729E613AB
24B	LOGIC DIAGRAM - CORE SPRAY SYSTEM	I-2210-02	729E613AB
25	LEGEND OF SYMBOLS & INSTR. IDENT. FOR PLANT SYSTEM DIAGRAMS	M-2001	
26	DIAGRAM-TORUS WATER MANAGEMENT SYSTEM	M-4100	
27	DIAGRAM-PRIMARY CONTAINMENT PNEUMATIC SUPPLY	M-5007	

- NOTES:**
- THIS DIAGRAM REPLACES THE GENERAL ELECTRIC DIAGRAM-731E981, DECD. FILE #RI-100.
 - SPECIFIC SYSTEM DESIGN REQUIREMENTS ARE GIVEN IN THE CORE SPRAY SYSTEM DESIGN SPECIFICATION # 3071-502.
 - THE PLANT IDENTIFICATION NUMBER FOR THE CORE SPRAY SYSTEM IS E2100.
 - DELETED
 - DELETED
 - SUCTION LINES BETWEEN PENETRATIONS X-224A & B AND PUMPS C001A THRU D ARE LEAKAGE MONITORED FOR PIPE BREAK DETECTION.
 - DELETED
 - INSTRUMENT LINE EXCESS FLOW CHECK VALVES ARE SOLENOID RESET BY A LOCAL PUSH BUTTON OPEN CLOSED INDICATION IS IN THE CONTROL ROOM AND AT THE LOCAL PANEL.
 - START-UP STRAINER TO BE INSTALLED PRIOR TO SYSTEM FLUSHING STRAINER MUST BE REMOVED PRIOR TO PLANT START-UP.
 - UNLESS OTHERWISE NOTED, ALL PIPING AND VALVES SHOWN ON THIS DIAGRAM ARE 90 LEVEL, L, CLASS B.
 - DESIGNATES VALVE LOCKED POSITION EITHER LOCKED OPEN/LOCKED CLOSED AS INDICATED BY LD OR LC.
 - THESE LOCATIONS ARE REQUIRED TO HAVE A UT EXAMINATION PERFORMED FOLLOWING DRAINING AND FILL AND VENT TO VERIFY THAT NO VOID IS PRESENT THE FOLLOWING INSTRUMENTS APPLY. SEE THE P&ID EVENT FOR THE SPECIFIC AREAS TO BE EXAMINED. A UPSTREAM OF E2150002A-P&ID EVENT 2906 B UPSTREAM OF E2150002B-P&ID EVENT 2907

- LEGEND**
- IDENTIFIES PIPING ISOMETRIC FOR FABRICATION & ERECTION.
- INSTRUMENT & CONTROL SYSTEM**
- NOTES:
UNLESS OTHERWISE SHOWN:
ALL INSTRUMENT PIS NUMBERS ARE PREFIXED E21
ALL VALVE AND EQUIPMENT PIS NUMBERS ARE PREFIXED E2100
- REFERENCE:
1. LEGEND OF SYMBOLS & INSTR. IDENT FOR PLANT SYS DIA: DECD M-2001

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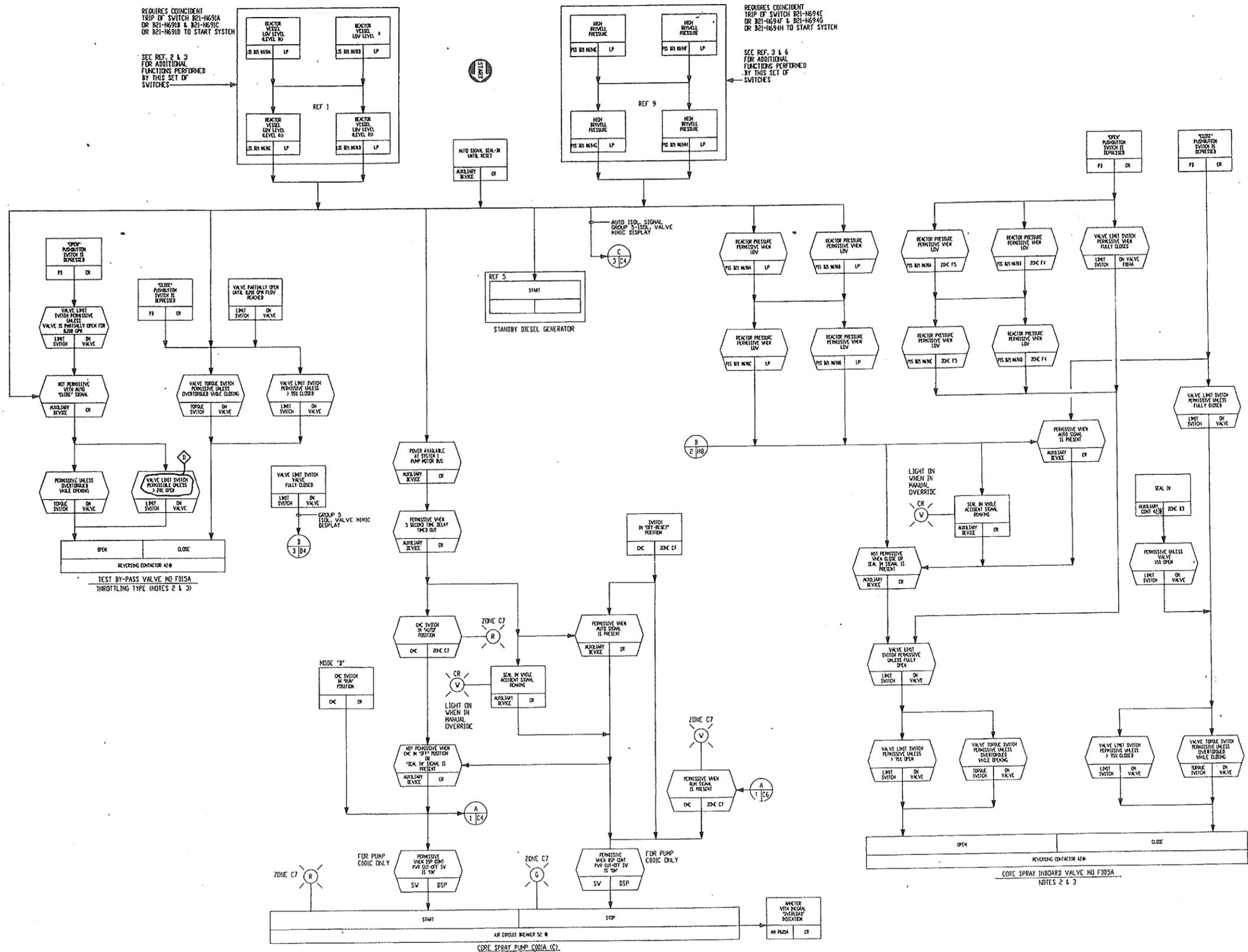
FIGURE 7.3-7
CORE SPRAY SYSTEM P&ID

REQUIRES COINCIDENT TRIP OF SWITCH B21-H691A OR B21-H691B & B21-H691C OR B21-H691D TO START SYSTEM

SEE REF. 2 & 3 FOR ADDITIONAL FUNCTIONS PERFORMED BY THIS SET OF SWITCHES

REQUIRES COINCIDENT TRIP OF SWITCH B21-H694E OR B21-H694F & B21-H694G OR B21-H694H TO START SYSTEM

SEE REF. 3 & 6 FOR ADDITIONAL FUNCTIONS PERFORMED BY THIS SET OF SWITCHES



NOTES:

- THE CONTROL SYSTEM AS DRAWN SHOWS SYSTEM I OPERATING SEQUENCE AFTER LOW WATER LEVEL OR HIGH DRYWELL PRESSURE. SIGNAL IS AS FOLLOWS:
 - CONDITION A WITH PLANT ON NORMAL AUXILIARY POWER OR STANDBY DIESEL GENERATOR
 - ACTIVE PUMPS: COOL A & C SYSTEM I STARTS; COOL B & D SYSTEM II STARTS
 - VALVES:
 - F04A SYSTEM I - OPENS AFTER REACTOR LOW PRESS. PERM.
 - F04B SYSTEM II - OPENS AFTER REACTOR LOW PRESS. PERM.
 - F05A SYSTEM I - OPENS AFTER REACTOR LOW PRESS. PERM.
 - F05B SYSTEM II - OPENS AFTER REACTOR LOW PRESS. PERM.
 - F015A SYSTEM I - CLOSING IF OPEN - NO DELAY
 - F015B SYSTEM II - CLOSING IF OPEN - NO DELAY
 - F031A SYSTEM I - CLOSING AFTER FLOW ESTABLISHED
 - F031B SYSTEM II - CLOSING AFTER FLOW ESTABLISHED
- SYSTEM II CIRCUITS ARE IDENTICAL TO CIRCUITS FOR SYSTEM I EXCEPT COMPONENT PARTS ARE IDENTIFIED AS FOLLOWS:

SYSTEM I	SYSTEM II
PUMP	COOL A & C
MIN FLOW BYPASS VALVES	COOL B & D
INBOARD ISOL. VALVES	F031 A
TESTABLE CHECK VALVES	F031 B
PUMP SUCTION VALVES	F031 C
TEST BYPASS VALVE	F031 D
- PUMP MOTOR CIRCUITS SHALL BE PROVIDED WITH OVERLOAD PROTECTION. OVERLOAD RELAYS SHALL MAINTAIN POWER AS LONG AS POSSIBLE WITHOUT CAUSING DAMAGE TO THE MOTOR OR THE ENERGIZED POWER SYSTEM. VALVE MOTORS SHALL BE PROVIDED WITH OVERLOAD ALARMS AND SHORT CIRCUIT PROTECTIVE TRIPS.
- MOTIVE POWER FOR SYSTEM I PUMPS SHALL ORIGINATE FROM A DIFFERENT EMERGENCY AC BUS THAN POWER FOR SYSTEM II PUMPS. POWER FOR VALVES IN EACH SYSTEM SHALL ORIGINATE FROM THE SAME BUS SUPPLYING PUMP POWER. CONTROL POWER FOR PUMPS & VALVES OF SYSTEM I SHALL BE FROM A DIFFERENT SOURCE THAN CONTROL POWER FOR SYSTEM II.
- N = SWITCHGEAR DEVICE FUNCTION NUMBERS ANSI SPEC. C37.2
- ALL EQUIPMENT AND INSTRUMENTS ARE PREFIXED BY SYSTEM NO.(21) UNLESS OTHERWISE NOTED.
- FOR LOCATION SETPOINT AND IDENTIFICATION OF INSTRUMENTS SEE "CENTRAL COMPONENT DATA" BASIS.
- SEE REFERENCE P & I'S FOR ADDITIONAL INFORMATION ON ALARMS, VALVE INDICATING LIGHTS AND PROCESS INSTRUMENTATION.
- THE CORE SPRAY SYSTEM SHALL BE DESIGNED IN ACCORDANCE WITH IEEE 279 AS APPLICABLE TO THE CONTROL CIRCUITRY.
- THE CORE SPRAY SYSTEM SHALL BE DESIGNED IN ACCORDANCE WITH REF. 10.
- FOR CONFIGURATION OF PANEL MOUNTED OPERATORS AND INDICATORS SEE DECD DIV 6H-2290 & REF. 11.

REFERENCE DOCUMENTS:

- NUCLEAR BOILER SYSTEM P&ID..... 642089/62990465538
- NUCLEAR BOILER SYSTEM LOGIC DIAGRAM..... 612690
- RHR SYSTEM LOGIC DIAGRAM..... 612200
- CORE SPRAY SYSTEM P&ID..... 642634
- CORE SPRAY SYSTEM DESIGN SPECIFICATION..... 22A1478 R4-47
- HPIC SYSTEM LOGIC DIAGRAM..... 612220
- RCIC SYSTEM LOGIC DIAGRAM..... 209A4756R1-20
- LOGIC SYMBOLS..... 642083/642084
- RHR SYSTEM P&ID.....
- ELECTRICAL EQUIPMENT SEPARATION FOR SAFEGUARD SYS.22A2777R1-1410
- GENERAL REQUIREMENTS FOR INSTR & ELECT EQUIP..... 21A1768R1-446
- CORE SPRAY SYSTEM SCHEMATICS 61721-2215-1 THRU 6 61721-2211-1 THRU 9
- LOGIC DIAG GROUP 5 ISOL. VALVE MINIC DISPLAY..... 41721-2210-3

LEGEND

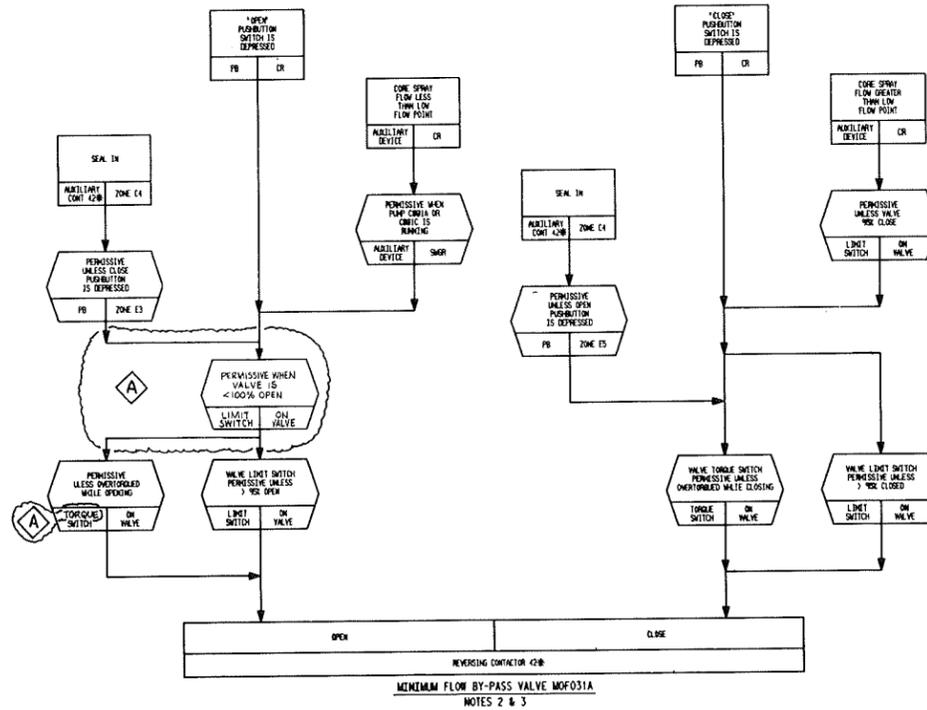
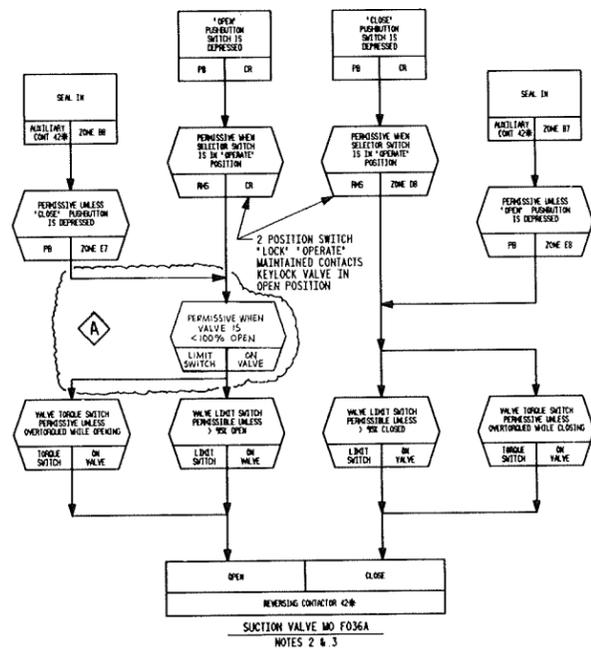
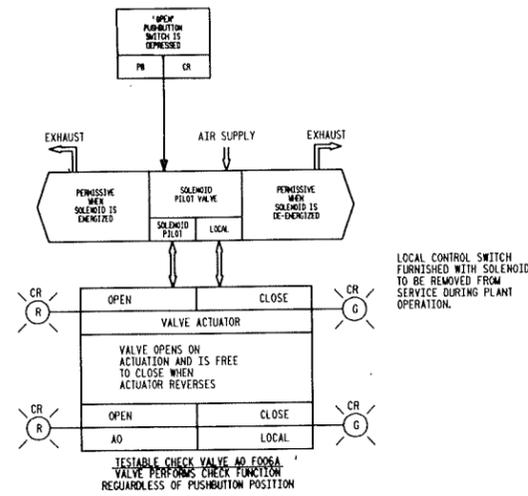
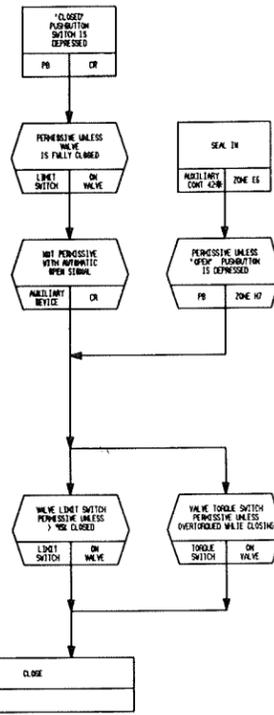
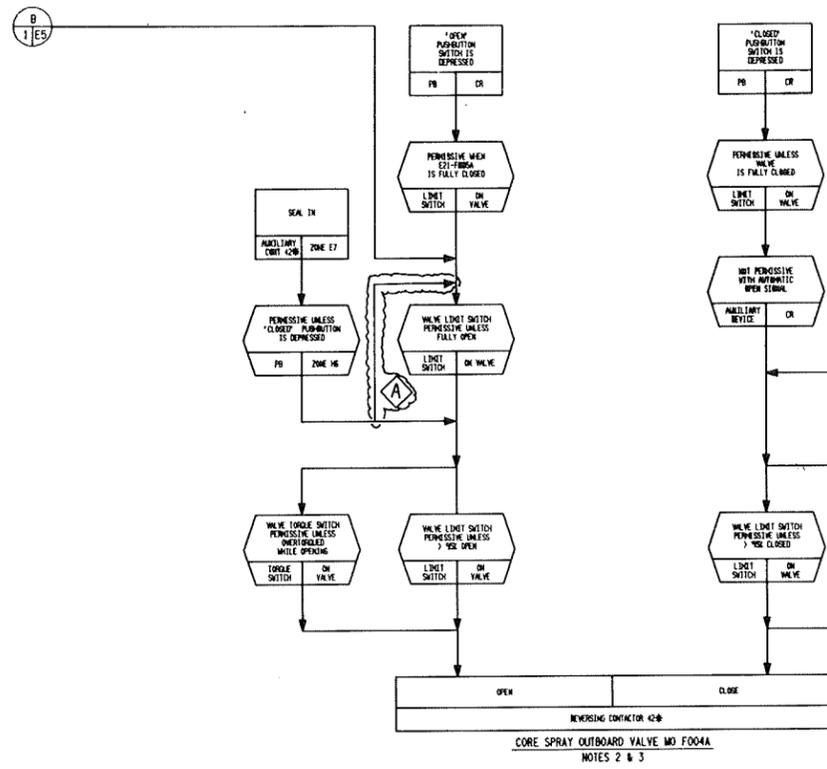
- CMC - COORDINATED MANUAL CONTROL SWITCH
- PB - PUSHBUTTON SWITCH
- AM - AMMETER
- CR - CONTROL ROOM & RELAY ROOM
- LP - LOCAL PANEL
- AO - AIR OPERATED
- X - LETTER DESIGNATION
- X/XX - ZONE DESIGNATION
- SHEET NO. OR REFERENCE DRAWING NO.

NOTES

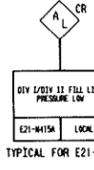
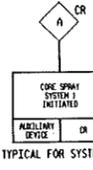
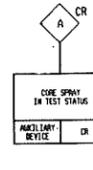
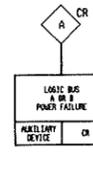
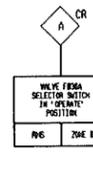
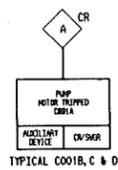
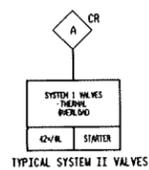
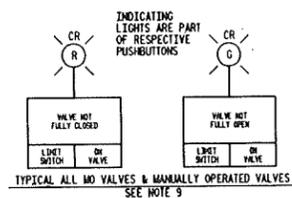
- THIS DRAWING SUPERSEDES EDISON'S SITE MASTER CRYSTAL DRAWING 229E612AB, REV.53 OF EDISON FILE NO.64-07.

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 UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 7.3-8, SHEET 1
 CORE SPRAY SYSTEM FCD

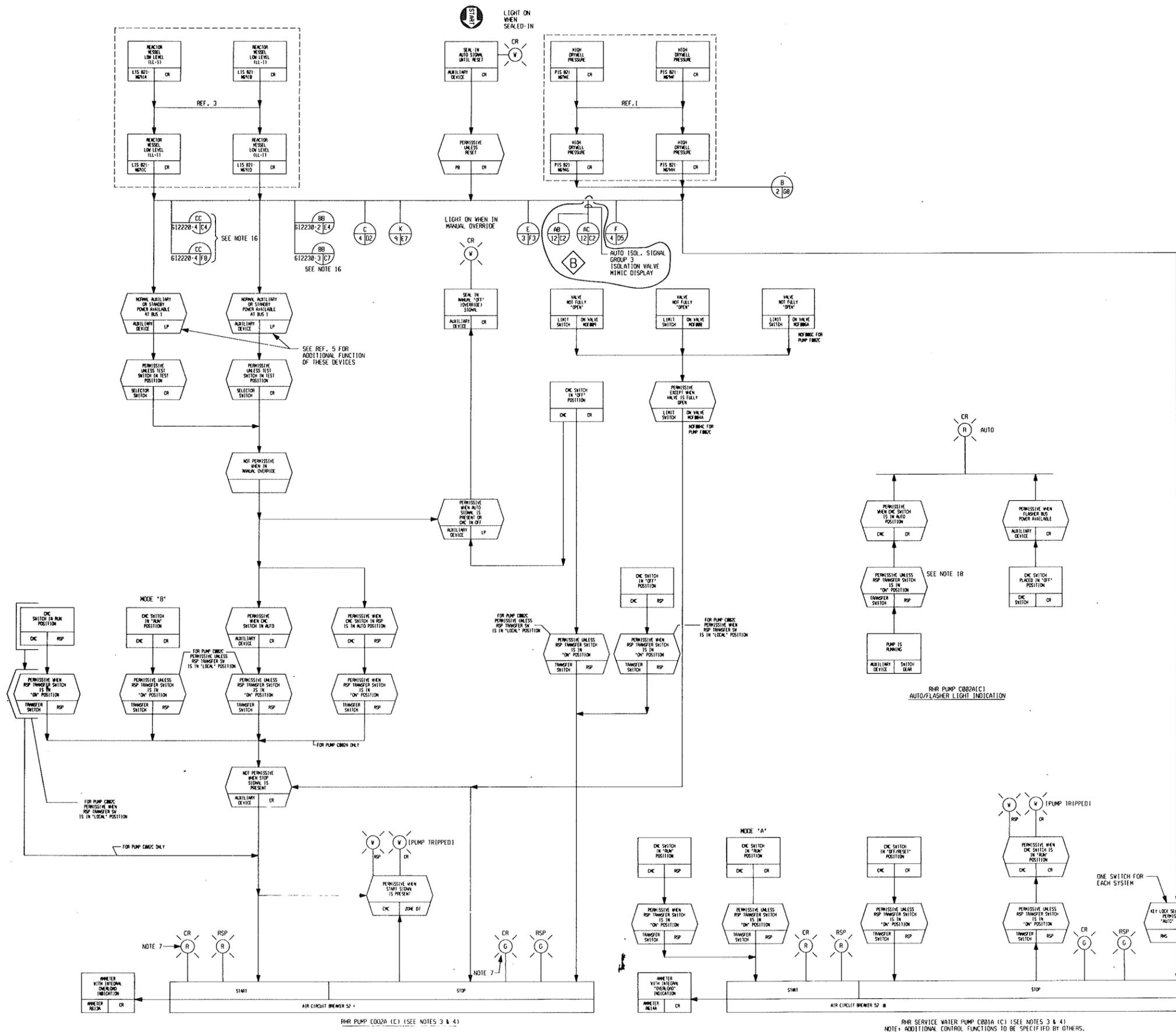


NOTE: SEE 61721-2210-1 (FIGURE 7.3-8, SHEET 1) FOR NOTES, REFERENCES AND LEGEND.



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FIGURE 7.3-8, SHEET 2
CORE SPRAY SYSTEM FCD



- NOTES**
- THE CONTROL SYSTEM AS DRAWN SHOWS SYSTEM I, THE OPERATING SEQUENCE, AFTER LOW WATER LEVEL SIGNAL OR HIGH DRYWELL PRESSURE IS AS FOLLOWS:
 CONDITION A PLANT ON NORMAL/AUXILIARY POWER & STANDBY DIESEL
 PUMP-C002A STARTS NO DELAY
 PUMP-C002B STARTS NO DELAY
 PUMP-C002C STARTS NO DELAY
 PUMP-C002D STARTS NO DELAY
 VALVES:
 F017A OPENS AFTER REACTOR LOW PRESSURE PERMISSIVE
 F017B OPENS AFTER REACTOR LOW PRESSURE PERMISSIVE
 CONT COOLING F016A
 F021A
 F028A CLOSE, IF OPEN, (NORMALLY MAINTAINED CLOSED)
 F024A
 F027A
 HEAT EXCHANGE VALVE F008A-CLOSES (IF OPEN)
 SERVICE WATER PUMPS C001A, B, C, D, STOP IF RUNNING
 - MANUAL CONTROL SYSTEM FOR THROTTLING TYPE MOTOR OPERATED VALVES SHALL BE DESIGNED TO ALLOW VALVES TO BE STOPPED AT ANY DESIRED POSITION.
 - SYSTEM I CIRCUITS & EQUIPMENT ARE SHOWN, SYSTEM II CIRCUITS & EQUIPMENT ARE IDENTICAL EXCEPT (UNLESS OTHERWISE NOTED) CORRESPONDING EQUIPMENT NUMBER, SUFFIXES ARE AS FOLLOWS:
 SYS I - A, C
 SYS II - B, D
 - PUMP MOTORS SHALL BE PROTECTED WITH OVERLOAD TRIPS, OVERLOAD RELAYS TO BE APPLIED SO AS TO MAINTAIN POWER ON MOTOR AS LONG AS POSSIBLE WITHOUT IMMEDIATE DAMAGE TO MOTORS OR HARM TO EMERGENCY POWER SYSTEM, VALVE MOTORS SHALL BE PROVIDED WITH OVERLOAD ALARMS, IN ADDITION, VALVE MOTORS ARE TO BE PROVIDED WITH SHORT CIRCUIT CURRENT PROTECTIVE TRIPS.
 - AUXILIARY RELAYS & DEVICES ARE NOT SHOWN ON THE FUNCTIONAL CONTROL DIAGRAM EXCEPT WHERE NEEDED TO CLARIFY THE FUNCTIONAL REQUIREMENTS.
 - MOTIVE POWER FOR SYSTEM I PUMPS SHALL ORIGINATE FROM A DIFFERENT EMERGENCY AC BUS THAN THE PUMPS OF SYSTEM II, MOTIVE POWER FOR INJECTION VALVES IN BOTH SYSTEMS SHALL ORIGINATE FROM A COMMON BUS WHICH IS AUTOMATICALLY CONNECTED TO ALTERNATE EMERGENCY BUS SOURCES, POWER FOR THE OTHER VALVES IN EACH SYSTEM SHALL ORIGINATE FROM THE SAME BUS SUPPLYING PUMP POWER, LOGIC CONTROL POWER FOR SYSTEM I SHALL BE FROM A DIFFERENT SOURCE THAN SYSTEM II.
 - STATUS LIGHTS SHALL BE AS FOLLOWS:
 VALVES: GREEN- ON FOR CLOSED POSITION
 RED- ON FOR OPEN POSITION
 BOTH ON FOR INTERMEDIATE POSITION
 PUMPS: RED- ON FOR PUMP RUNNING
 GREEN- ON FOR PUMP STOP
 - ALL AUXILIARY TIMING DEVICES SHALL BE ADJUSTABLE FROM 0 TO FULL SCALE, FULL SCALE SHALL BE AT LEAST 1-1/2 TIMES GREATER THAN SPECIFIED TIME SETTING.
 - ALL EQUIPMENT & INSTRUMENTS ARE PREFIXED BY E11 UNLESS OTHERWISE NOTED.
 - THE FOUR ELECTRONIC PRESSURE SWITCHES ARE ARRANGED FOR ONE OUT OF TWO TWICE LOGIC.
 - FOR LOCATION, IDENTIFICATION AND SET POINTS OF INSTRUMENTS SEE MASTER INSTRUMENT LIST.
 - THE SENSING CIRCUIT FOR BREAK DETECTION AND VALVE SELECTION IS TO BE ARRANGED SO THAT FAILURE OF A SINGLE DEVICE OR CIRCUIT TO FUNCTION ON DEMAND WILL NOT PREVENT CORRECT SELECTION OF LOOP FOR INJECTION.
 - THE RESIDUAL HEAT REMOVAL SYSTEM SHALL BE DESIGNED IN ACCORDANCE WITH "PROPOSED CRITERIA FOR NUCLEAR POWER PLANT PROTECTION SYSTEM (SEE 27)" AS APPLICABLE TO THE CONTROL CIRCUITRY.
 - THE RESIDUAL HEAT REMOVAL SYSTEM SHALL BE DESIGNED IN ACCORDANCE WITH REFERENCE 10.
 - FOR CONFIGURATION OF PANEL MOUNTED OPERATORS AND INDICATORS SEE DWG 6M-2290 SHEET 2 REFERENCE 11, PURCHASE SPECIFICATION DATA SECTION 2.4
 - SIGNAL IS PROVIDED FROM DIV 1 FOR RCIC SYSTEM AND DIV 2 FOR HPCI SYSTEM.
 - EDISON BYPASS OPENING AND CLOSING TORQUE SWITCHES OVER SPECIFIC RANGES OF MOV POSITIONS, EDISON CAN CONTROL, THE FOLLOWING FROM THE REMOTE SHUTDOWN PANEL (RSP/F008A, F009A, F009B, F015A, F017A, F024A, F028A, F040A, F068A, E1102-C002A & C, E1151-C001A & C.

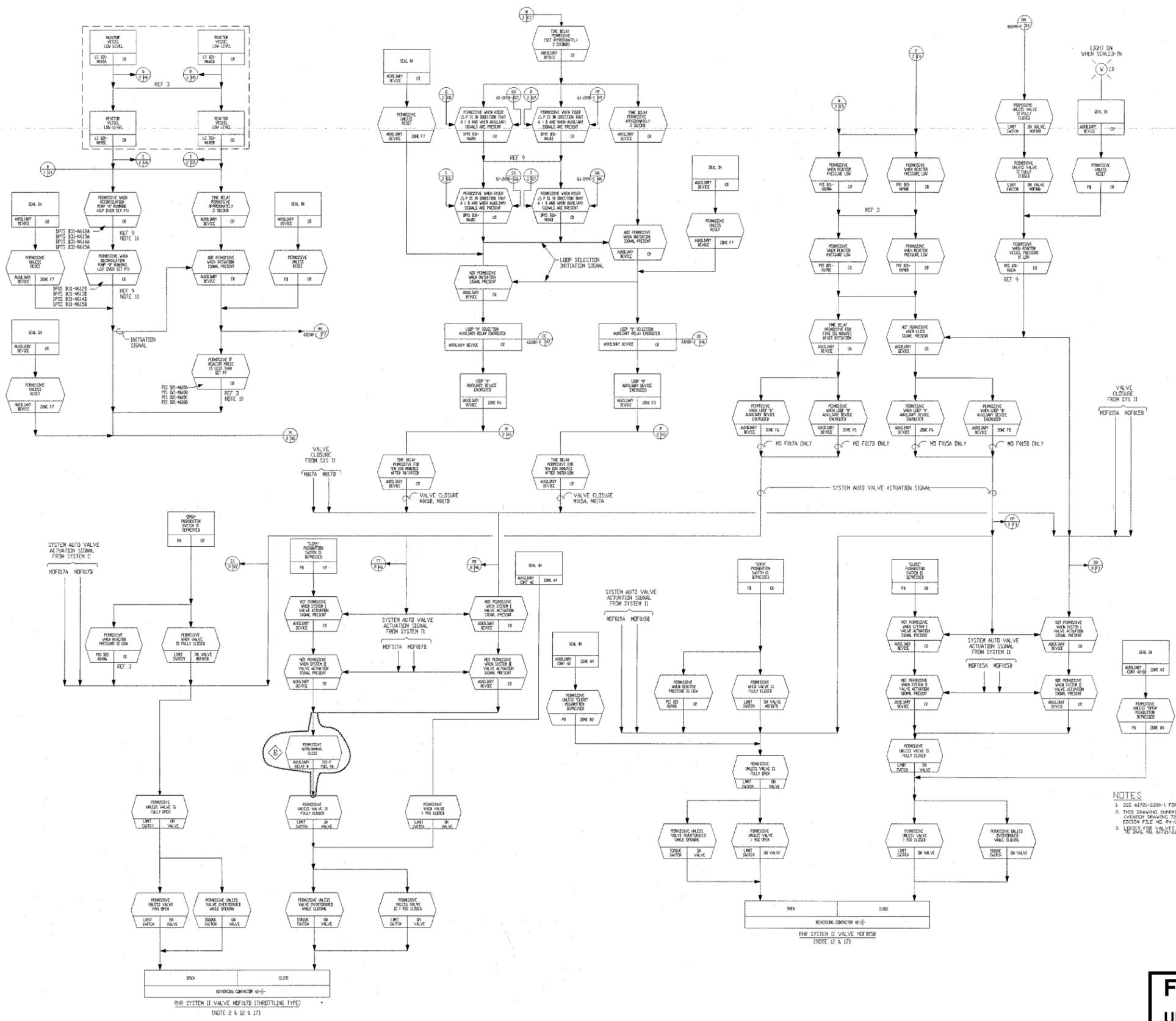
REFERENCE DOCUMENTS:

1. RESIDUAL HEAT REMOVAL SYSTEM P & ID	6M-2083, 6M-2084
2. LOGIC SYMBOLS	209A47561 RI-20
3. NUCLEAR BOILER SYSTEMS P & ID	6M2089, 6M2090, 6M5538
4. NUCLEAR BOILER SYSTEMS LOGIC DIAGRAM	612200
5. CORE SPRAY SYSTEM LOGIC DIAGRAM	612210
6. RECIRC FLOW CONTROL SYSTEM LOGIC DIAGRAM	612200
7. HPCI SYSTEM LOGIC DIAGRAM	612200
8. RCIC SYSTEM LOGIC DIAGRAM	612200
9. REACTOR RECIRC. SYSTEM PAID	6M2833
10. ELECTRICAL EQUIPMENT SEPARATION FOR SAFEGUARD SYS	2243777/RI-1418
11. GENERAL REQUIREMENTS FOR INSTR & ELEC EQUIP	2161760/RI-446
12. RESIDUAL HEAT REMOVAL SYSTEMS SCHEMATICS	612205-1 THRU 32, 34, 37 & 38A, 70 THROUGH 81, 8182201-33, 34, 35, 36
13. RHR SERVICE WATER SCHEMATICS	41721-2200-12
14. LOGIC DIAGRAM-GROUP 3- RHR ISOLATION VALVE MIMIC DISPLAY	41721-2200-12
15. LOGIC DIAGRAM-GROUP 4- RHR ISOLATION VALVE MIMIC DISPLAY	41721-2200-13

- LEGEND:**
- CMC COORDINATED MANUAL CONTROL SWITCH
 - PB PUSHBUTTON SWITCH
 - AM AMMETER
 - # SWITCHGEAR DEVICE FUNCTION NO. ANSI SPEC. C37.2
 - RSP REMOTE SHUTDOWN PANEL
 - LP LOCAL CONTROL ROOM/RELAY ROOM
 - CR REMOTE MANUAL SWITCH

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FIGURE 7.3-9, SHEET 1
 RESIDUAL HEAT REMOVAL SYSTEM FCD

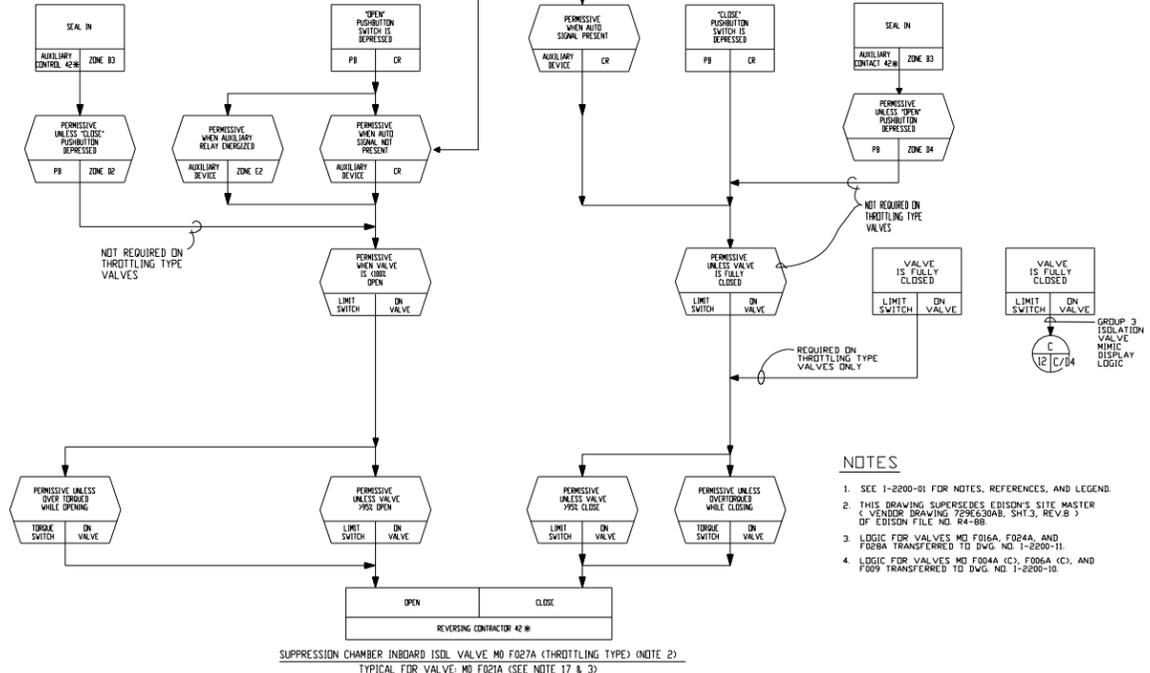
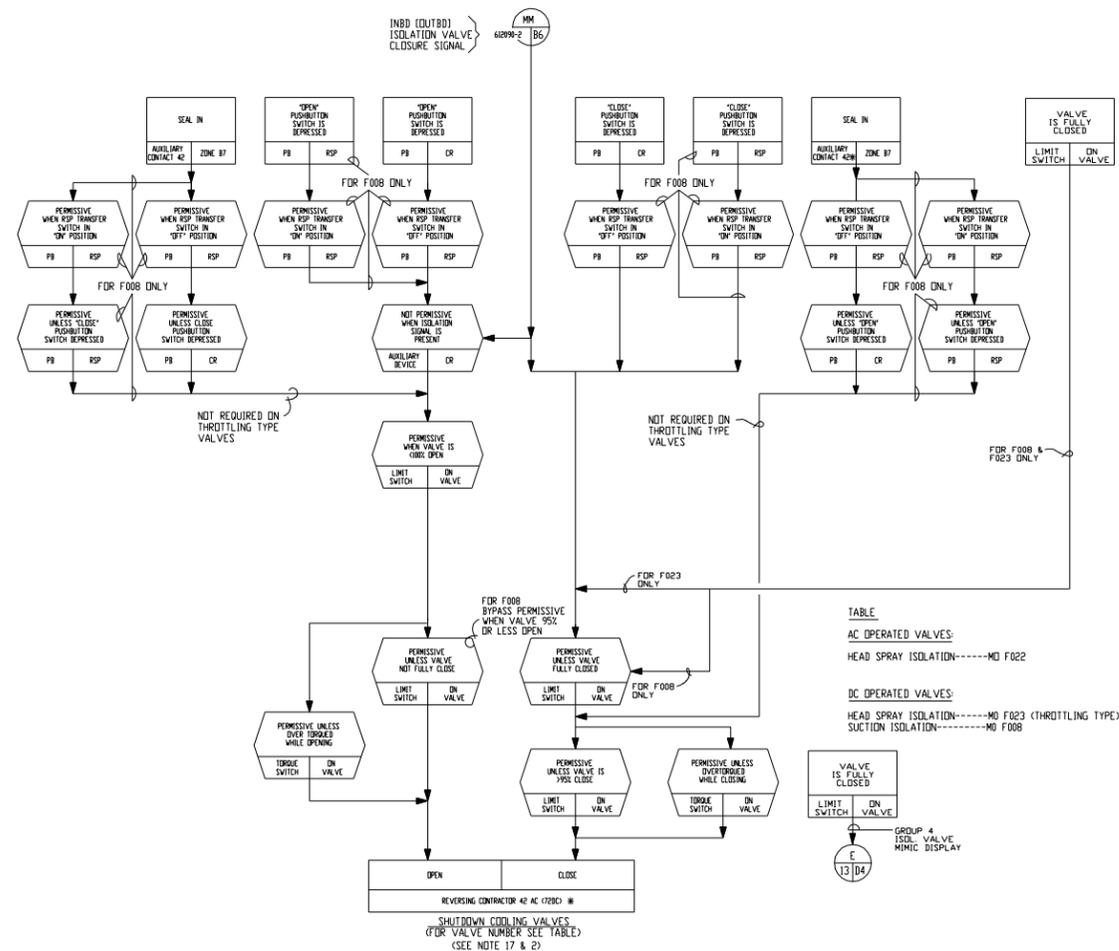
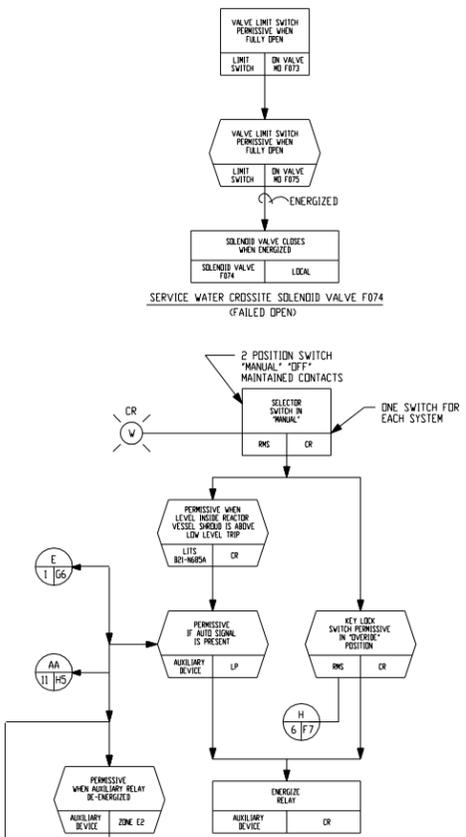
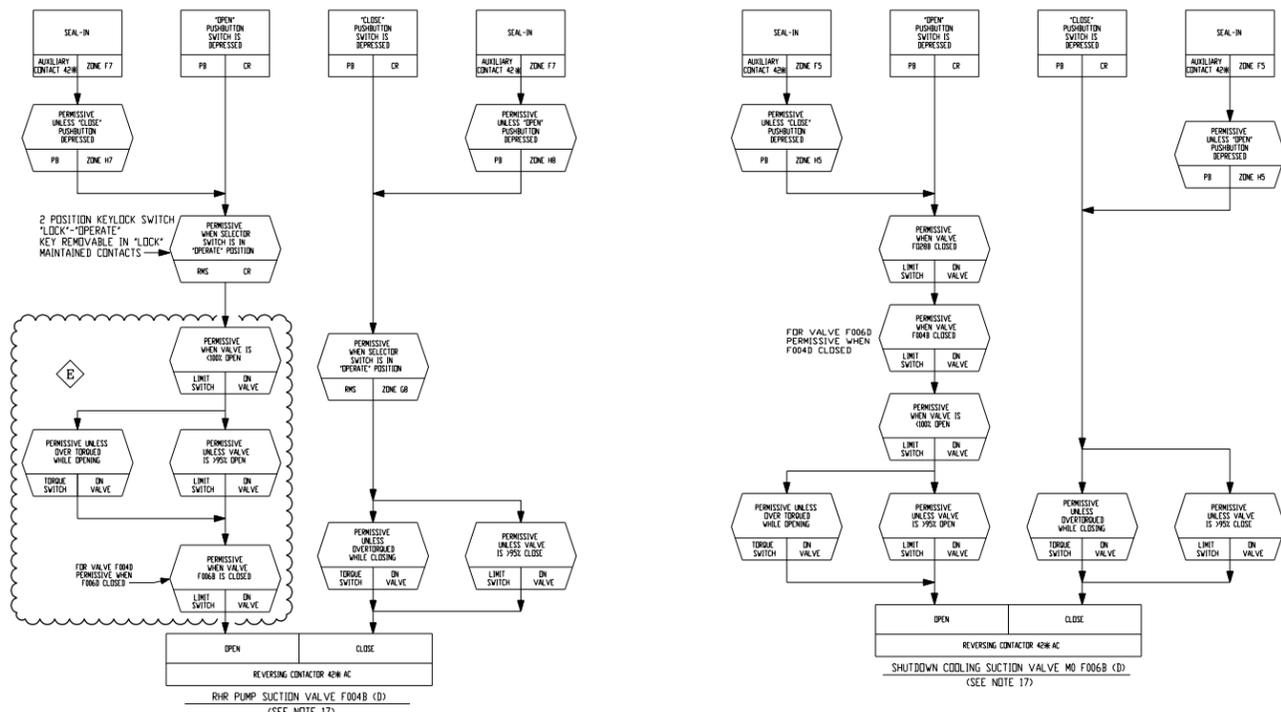


NOTES
 1. SEE 44721-2200-1 FOR NOTES, REFERENCES, AND LEGEND.
 2. THIS DRAWING SUPERSEDES EDISON'S SITE MASTER SYSTEM DRAWING, PROJECT NO. 5412, REV. 7) OF EDISON FILE NO. P4-08.
 3. LIMITS FOR VALVES NO. F075A AND F077A TRANSFERRED TO SHEET 2 OF 2.

FOR SYSTEM II VALVE MEF078 (SHOOTING TYPE)
 DATE 2 R 12 & 17

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FIGURE 7.3-9, SHEET 2
 RESIDUAL HEAT REMOVAL SYSTEM FCD

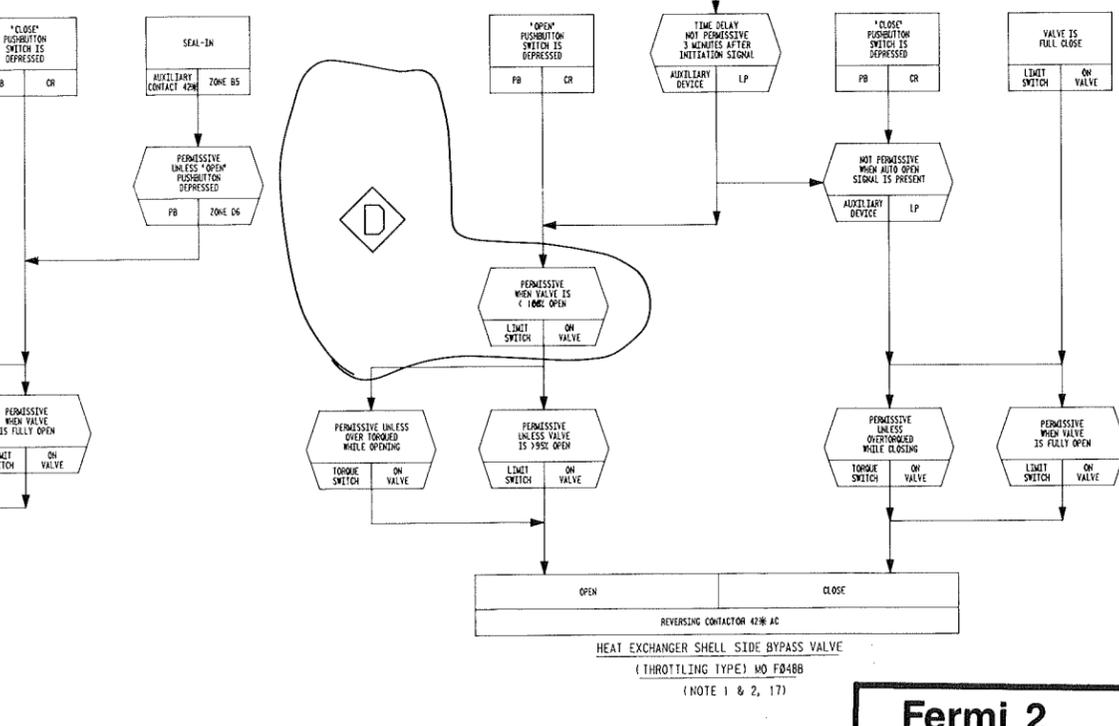
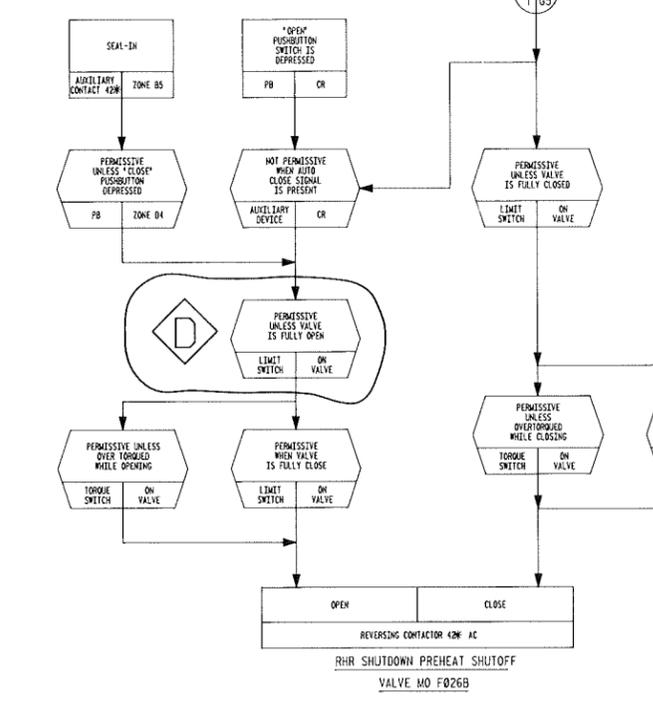
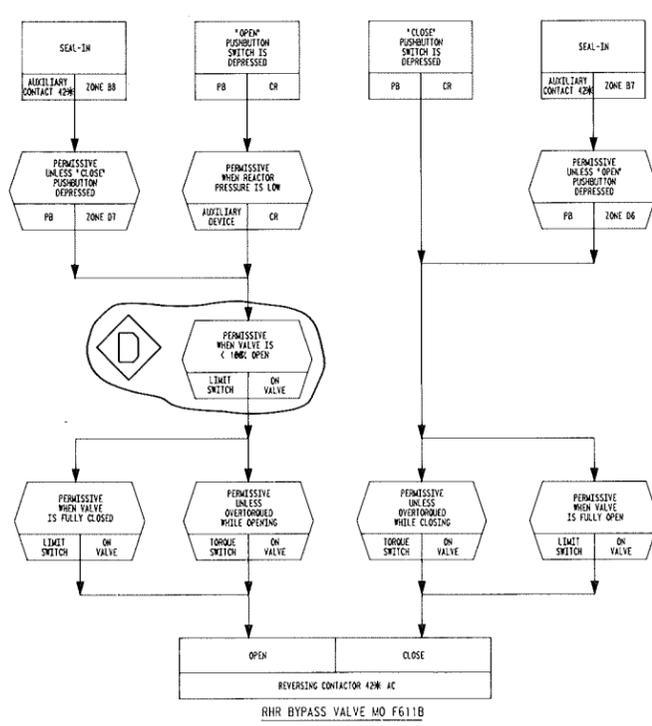
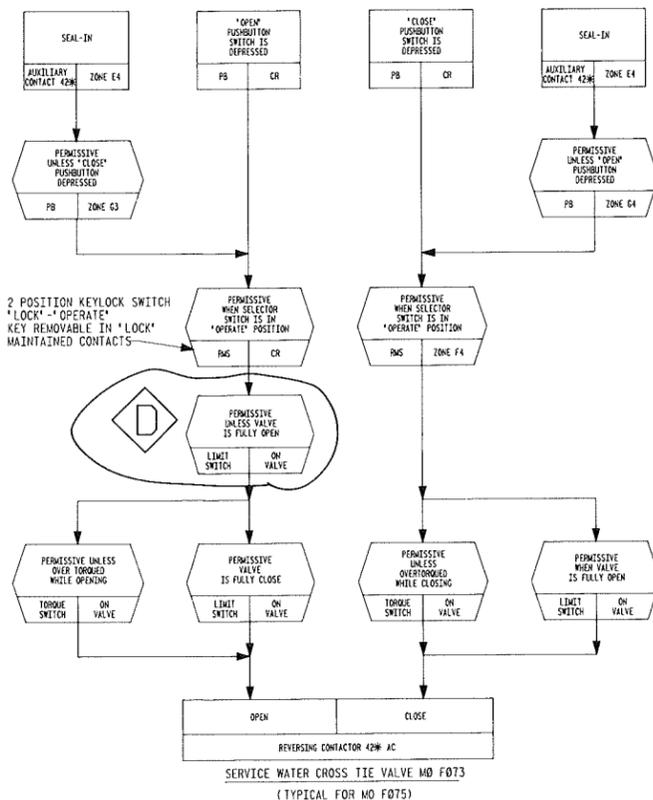
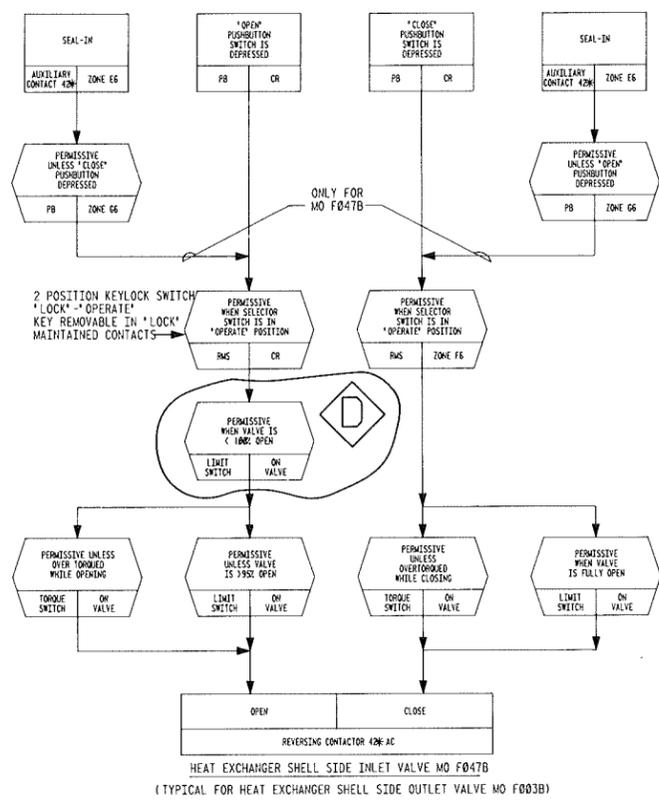


- NOTES
- SEE 1-2200-01 FOR NOTES, REFERENCES, AND LEGEND
 - THIS DRAWING SUPERSEDES EDISON'S SITE MASTER (FERRIS) DRAWING 7256-2048, SHEET 3, REV. B OF EDISON FILE NO. R4-88
 - LOGIC FOR VALVES M0 F016A, F024A, AND F028A TRANSFERRED TO DWG. NO. 1-2200-11
 - LOGIC FOR VALVES M0 F004A (C), F006A (C), AND F009 TRANSFERRED TO DWG. NO. 1-2200-10

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 UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 7.3-9, SHEET 3

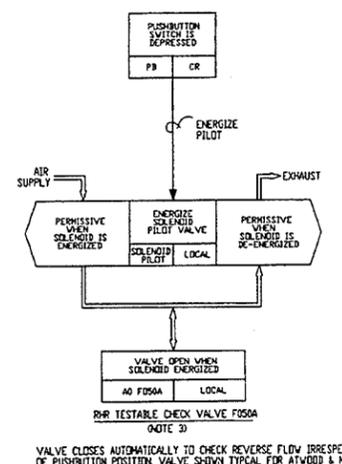
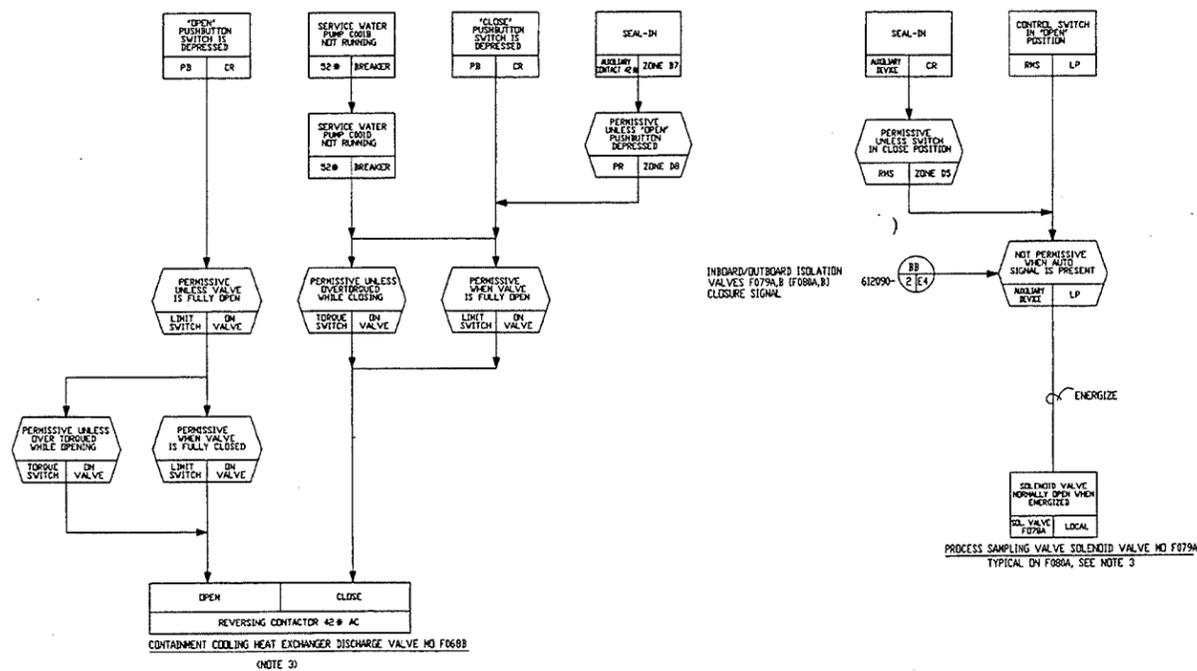
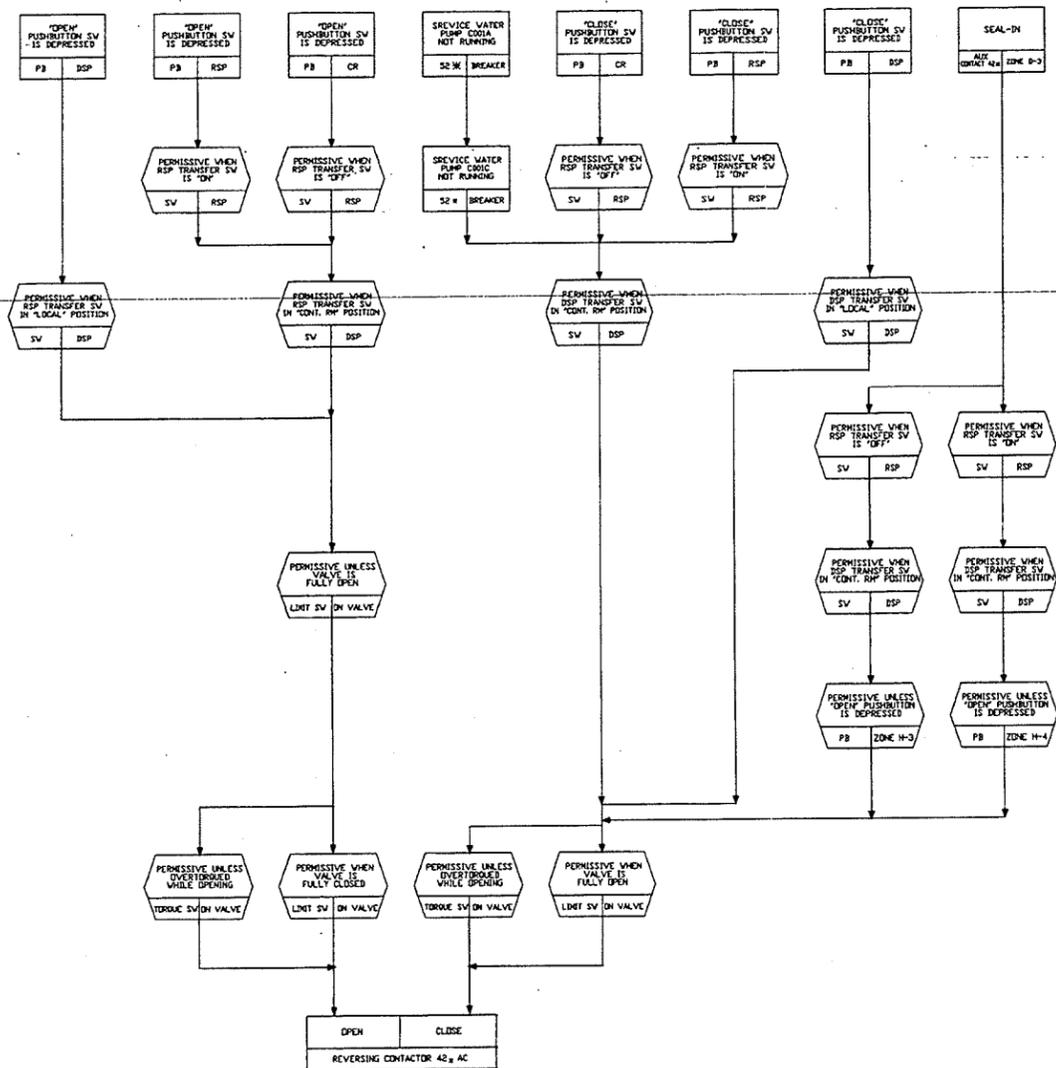
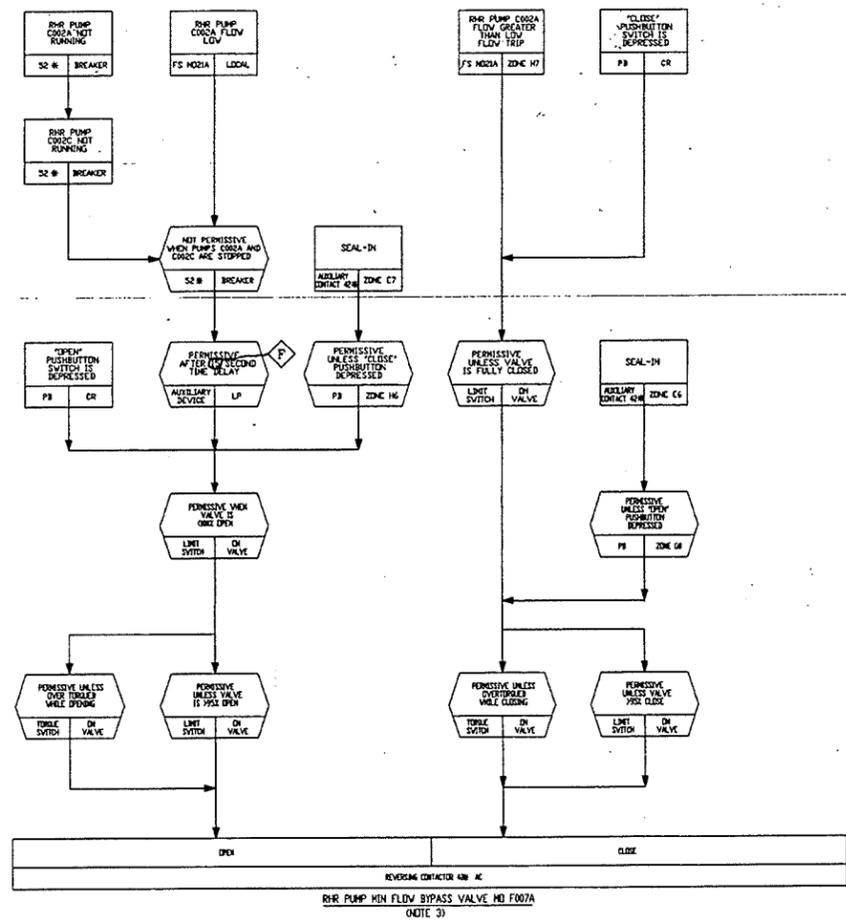
RESIDUAL HEAT REMOVAL SYSTEM FCD



- NOTES
- SEE 61721-2200-1 (FIGURE 7.3-9, SHEET 1) FOR NOTES, REFERENCES, AND LEGEND.
 - THIS DRAWING SUPERSEDES EDISON'S SITE MASTER (VENDOR DRAWING 729E630AB, SHT. 4) OF EDISON FILE NO. R4-88.
 - LOGICS FOR VALVES MO F003A, F010, F047A, F048A, AND F611A TRANSFERRED TO DWG. NO. 61721-2200-9 (FIGURE 7.3-9, SHEET 8).

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 UPDATED FINAL SAFETY ANALYSIS REPORT

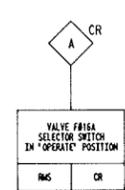
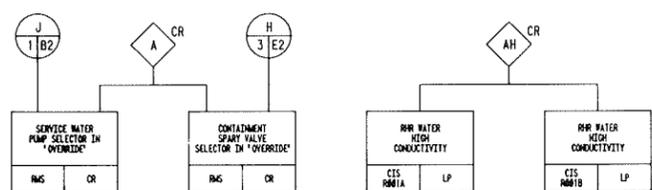
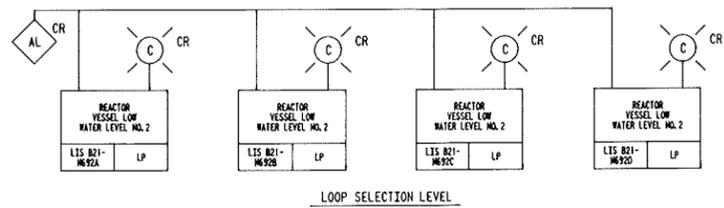
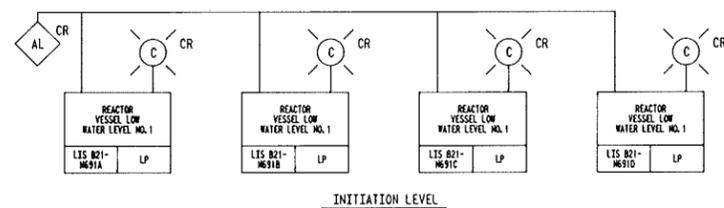
FIGURE 7.3-9, SHEET 4
 RESIDUAL HEAT REMOVAL SYSTEM FCD



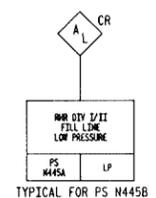
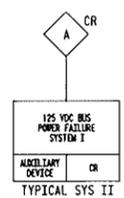
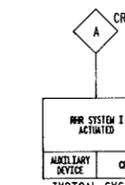
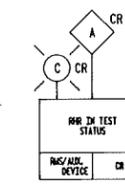
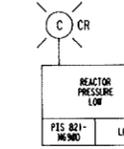
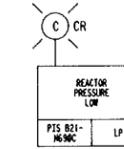
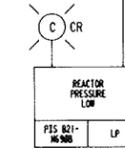
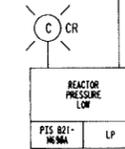
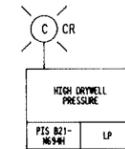
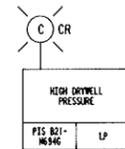
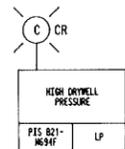
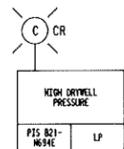
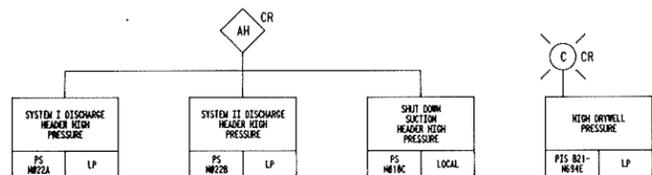
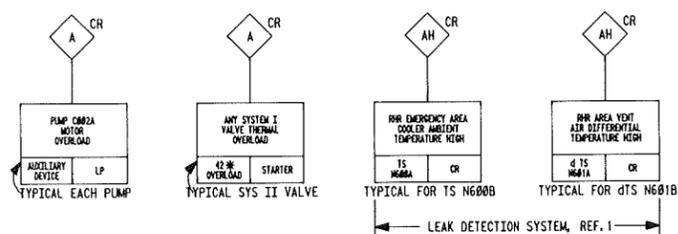
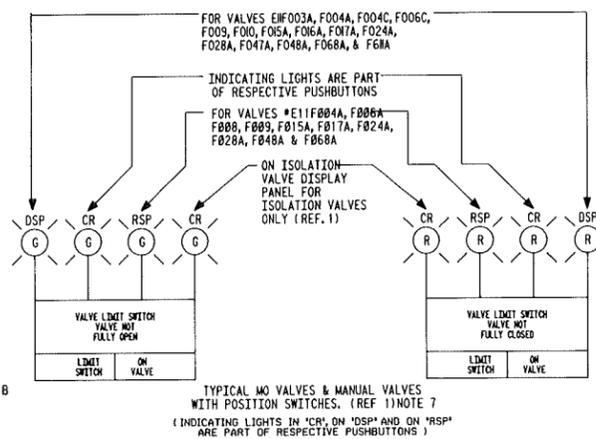
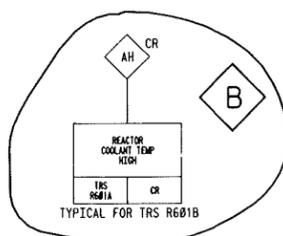
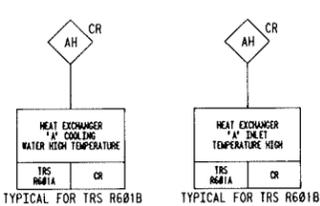
NOTES
 1. SEE I-2200-1 FOR NOTES, REFERENCES, AND LEGEND.
 2. THIS DRAWING SUPERSEDES EDISON'S SITE MASTER.
 3. VENDOR DRAWING 72503048 INT 5 REV 0 3 OF EDISON FILE NO. R4-88.

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FIGURE 7.3-9, SHEET 5
 RESIDUAL HEAT REMOVAL SYSTEM FCD



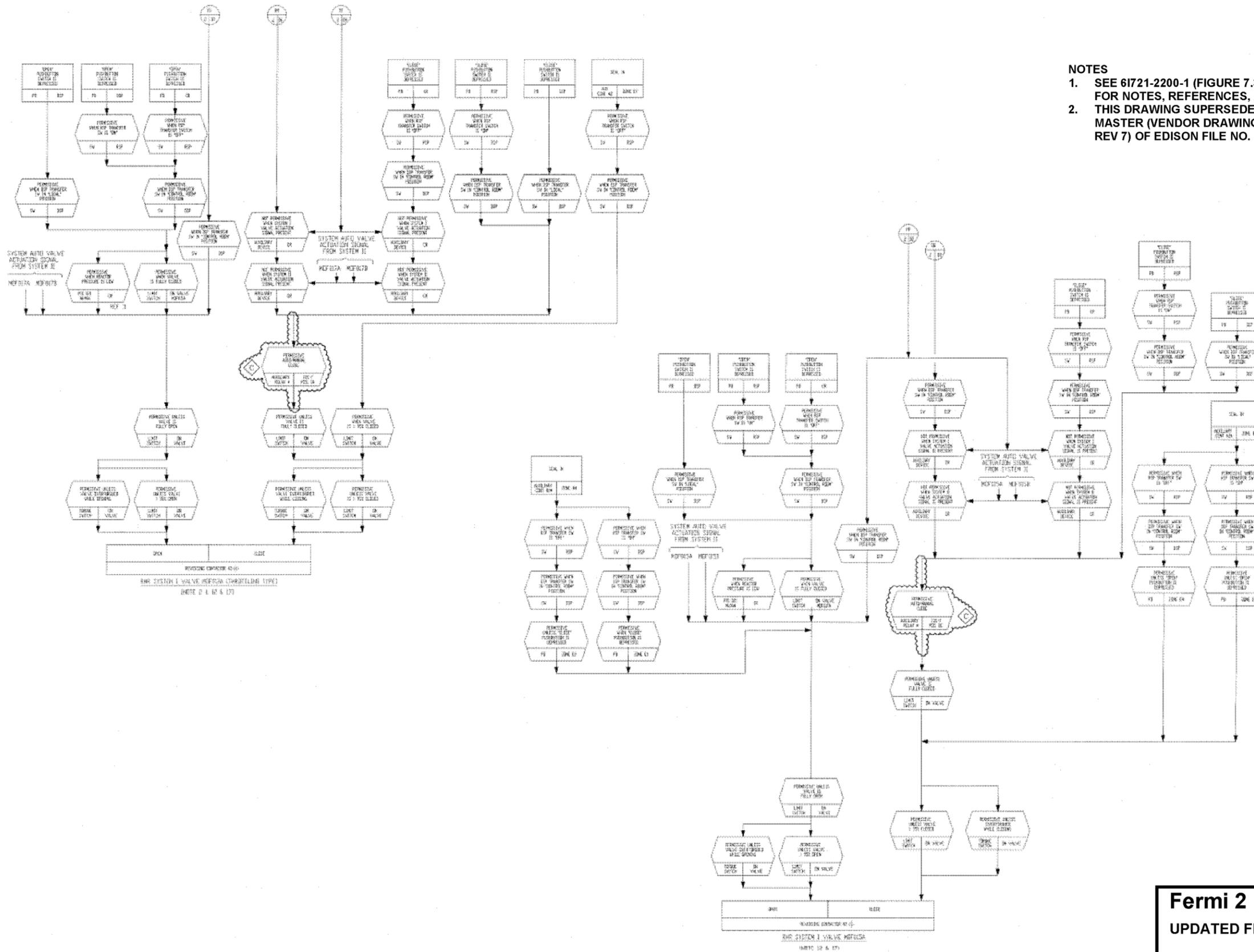
TYP FOR F028A
F010, F003A, F047A
F004AC, F073, F075
F016B, F028B, F004B, D,
F047B, F003B & F060B



- NOTES
- SEE 61721-2200-1 (FIGURE 7.3-9, SHEET 1) FOR NOTES, REFERENCES, AND LEGEND.
 - THIS DRAWING SUPERSEDES EDISON'S SITE MASTER (VENDOR DRAWING 729E630AB, SHT. 6, REV. 11) OF EDISON FILE R4-88.

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UPDATED FINAL SAFETY ANALYSIS REPORT

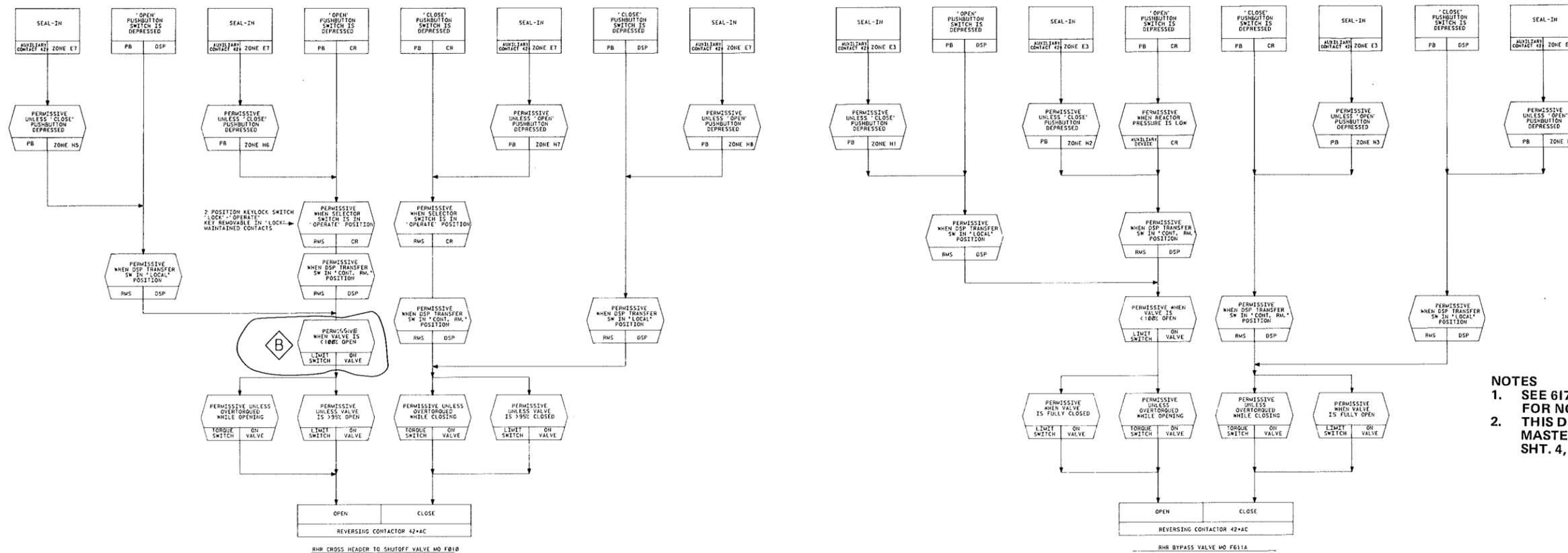
FIGURE 7.3-9, SHEET 6
RESIDUAL HEAT REMOVAL SYSTEM FCD



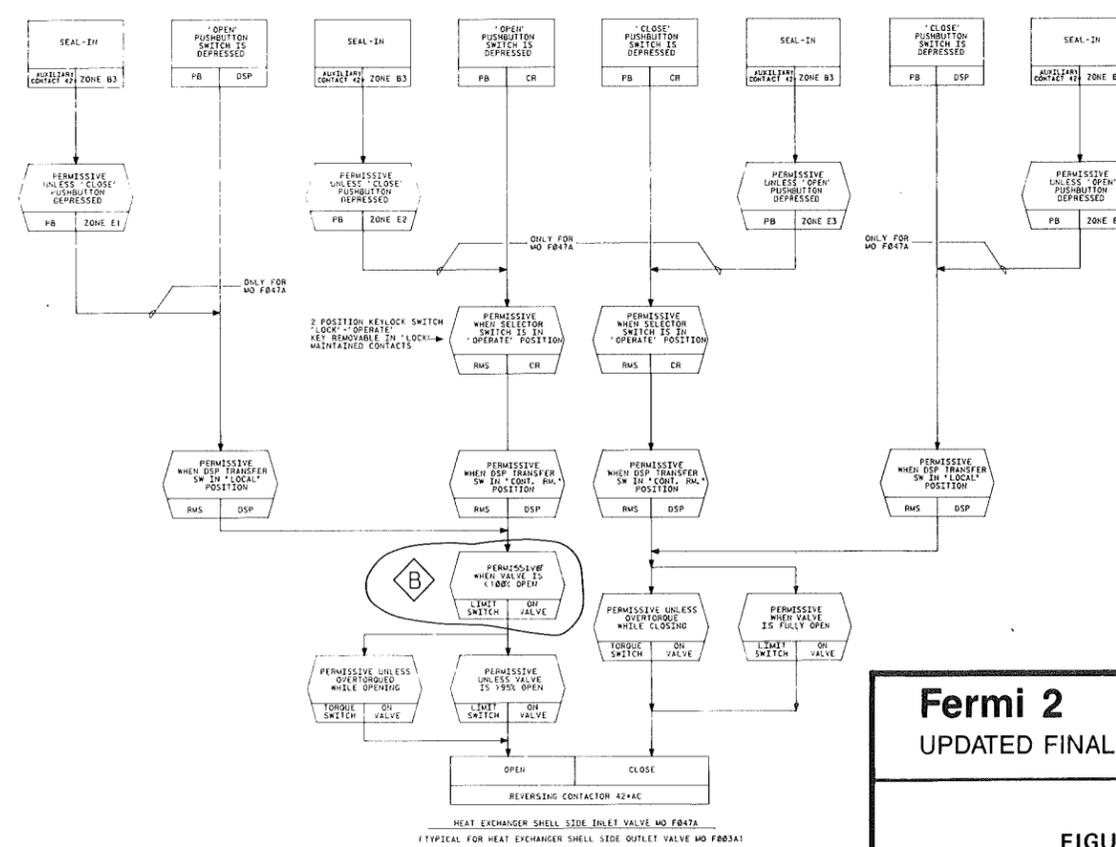
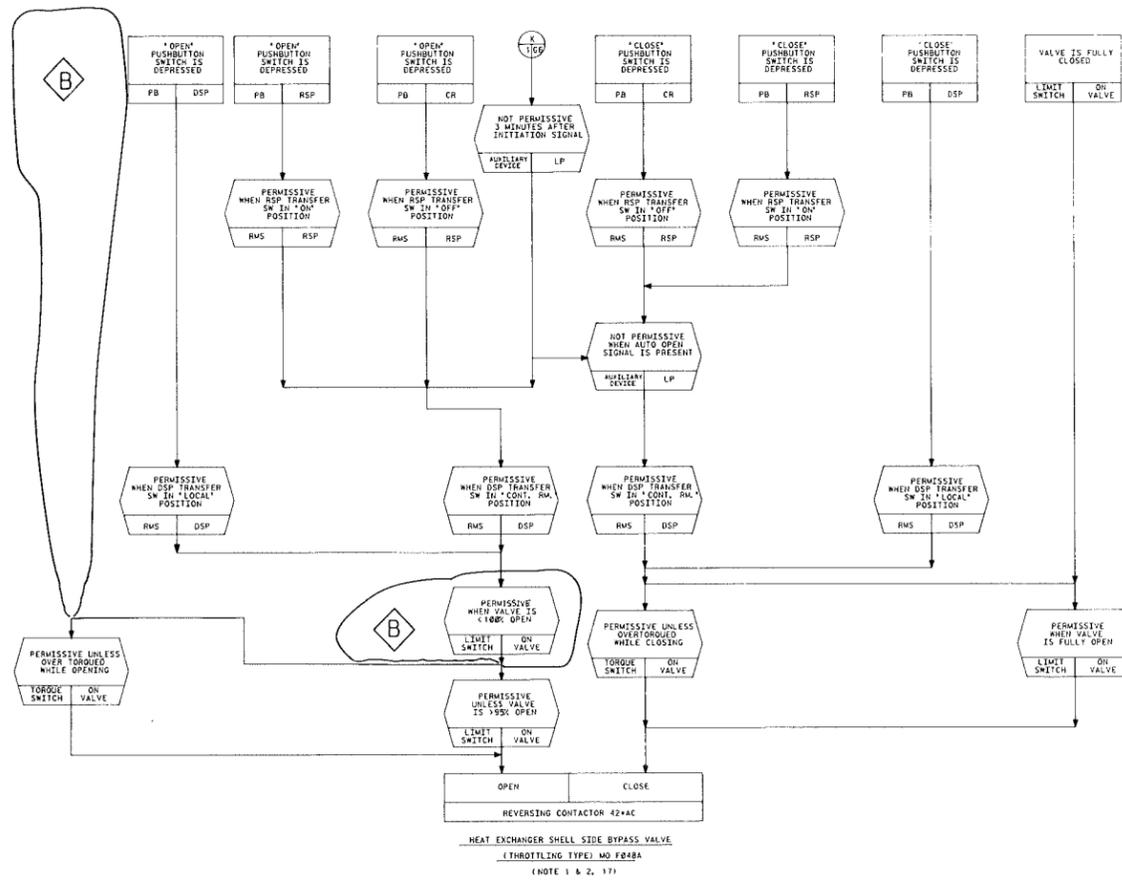
- NOTES
1. SEE 61721-2200-1 (FIGURE 7.3-9, SHEET 1) FOR NOTES, REFERENCES, AND LEGEND
 2. THIS DRAWING SUPERSEDES EDISON'S SITE MASTER (VENDOR DRAWING 729E630AB, SHT. 2 REV 7) OF EDISON FILE NO. R4-88.

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FIGURE 7.3-9, SHEET 7
RESIDUAL HEAT REMOVAL SYSTEM FCD

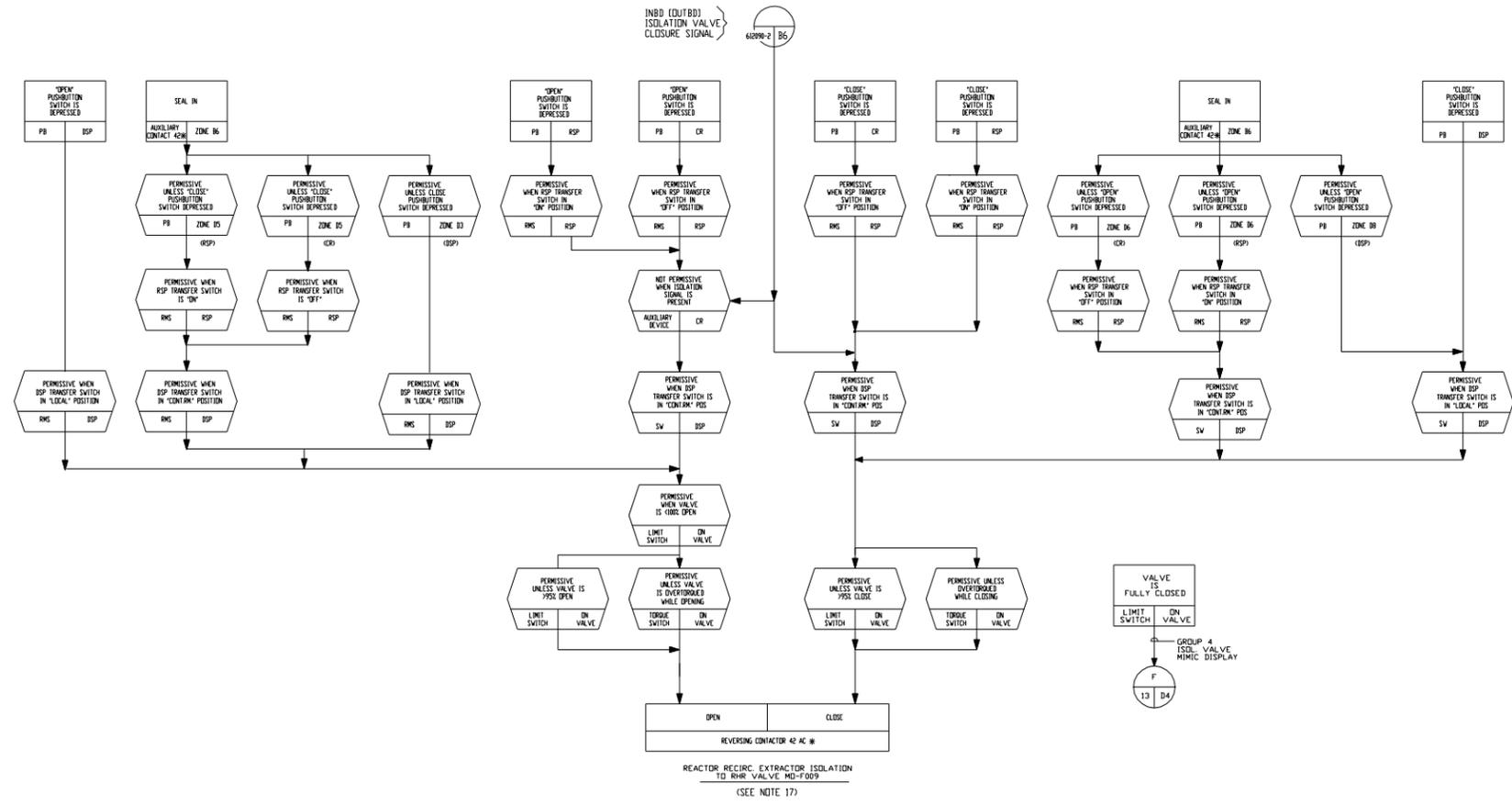
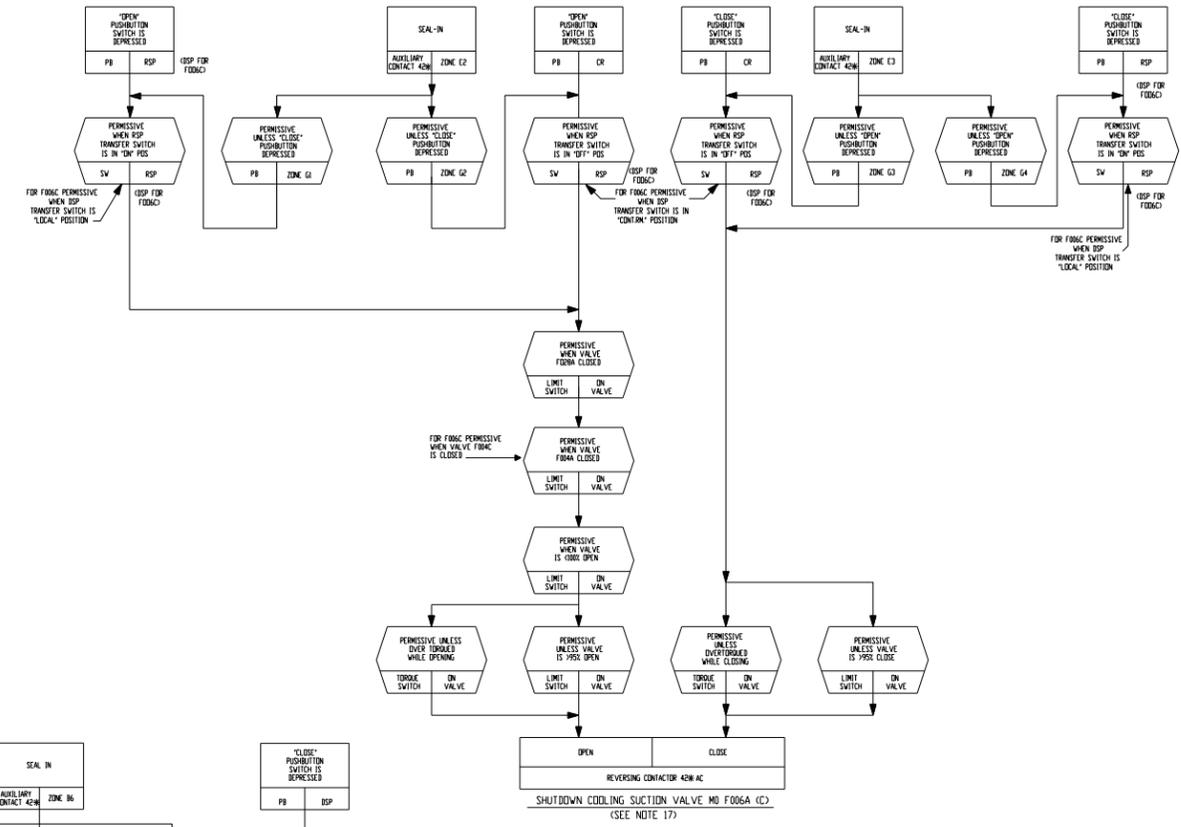
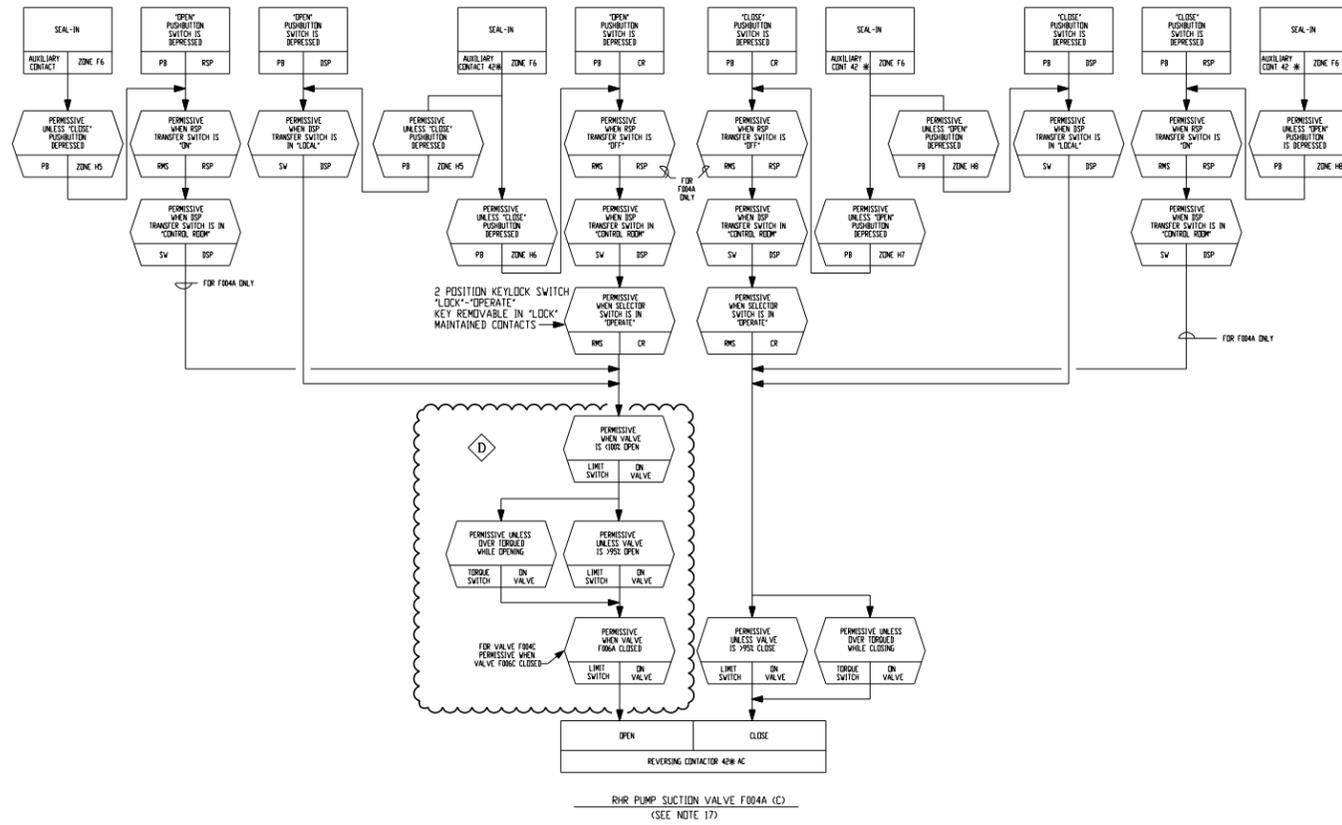


- NOTES
- SEE 61721-2200-1 (FIGURE 7.3-9, SHEET 1) FOR NOTES, REFERENCES, AND LEGEND.
 - THIS DRAWING SUPERSEDES EDISON'S SITE MASTER (VENDOR DRAWING 729E630AB, SHT. 4, REV. 8) OF EDISON FILE NO. R4-88.



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FIGURE 7.3-9, SHEET 8
 RESIDUAL HEAT REMOVAL SYSTEM FCD

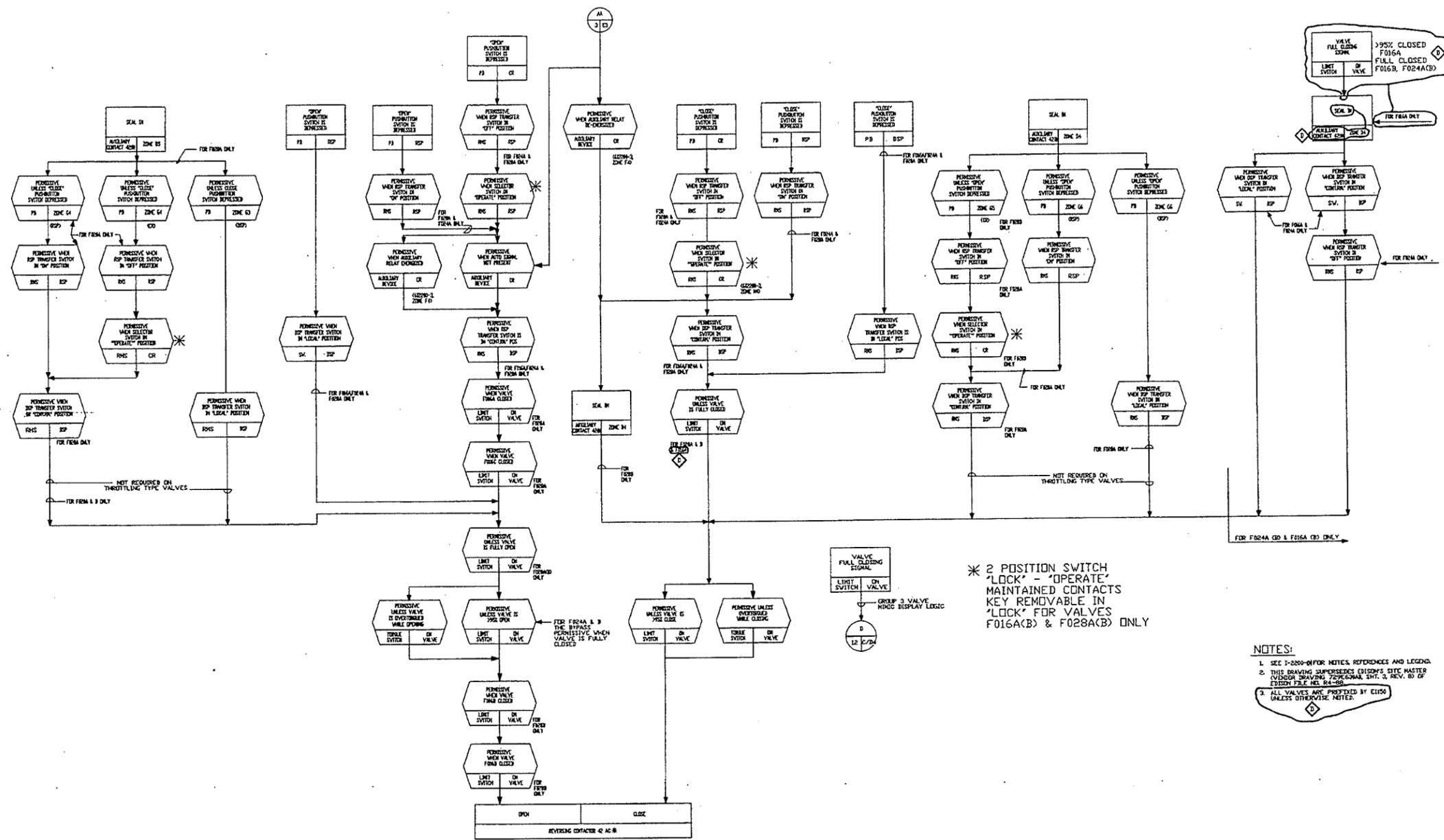


NOTES
 1. SEE I-2200-01 FOR NOTES, REFERENCES, AND LEGEND.
 2. THIS DRAWING SUPERSEDES EDISON'S SITE MASTER CVENTOR DRAWING 7996330AB, SHT. 3. REV. BY GP EDISON FILE NO. 84-88

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FIGURE 7.3-9, SHEET 9

RESIDUAL HEAT REMOVAL SYSTEM FCD

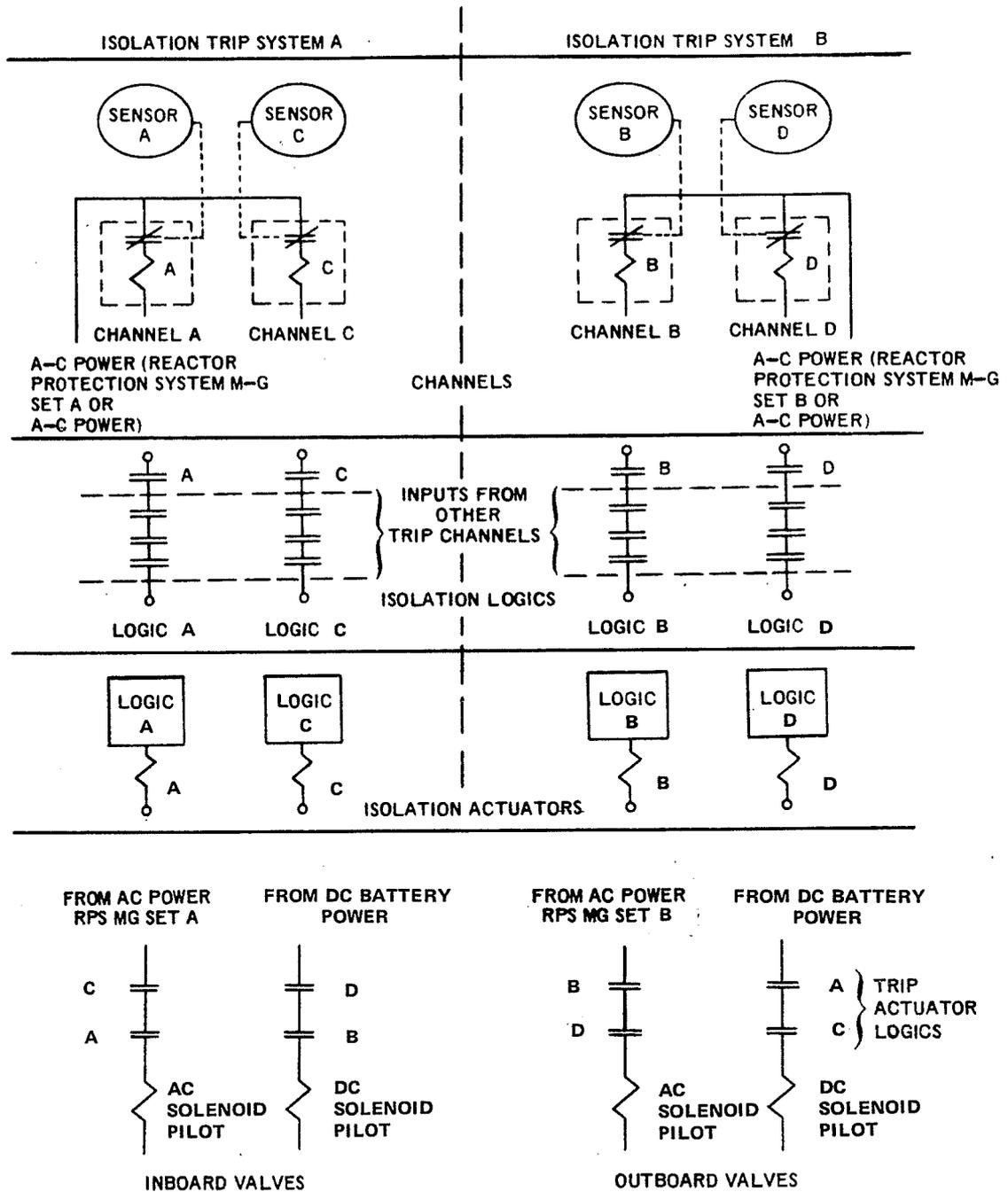


SUPPRESSION CHAMBER SPRAY OUTBOARD ISOL. VALVE, M0-F028A(B)
 TYPICAL FOR THROTTLING TYPE VALVES, M0-F024A(B) & -F016A(B)
 UNLESS OTHERWISE NOTED
 (NOTES 2 & 3)

- NOTES:
1. SEE I-2000-01 FOR NOTES, REFERENCES AND LEGEND.
 2. THIS DRAWING SUPERSEDES DISCOM'S SITE MASTER VALVE DRAWING TYPICAL INT. 3, REV. 03 OF DESIGN FILE NO. 04-285.
 3. ALL VALVES ARE PROVIDED BY C1150 UNLESS OTHERWISE NOTED.

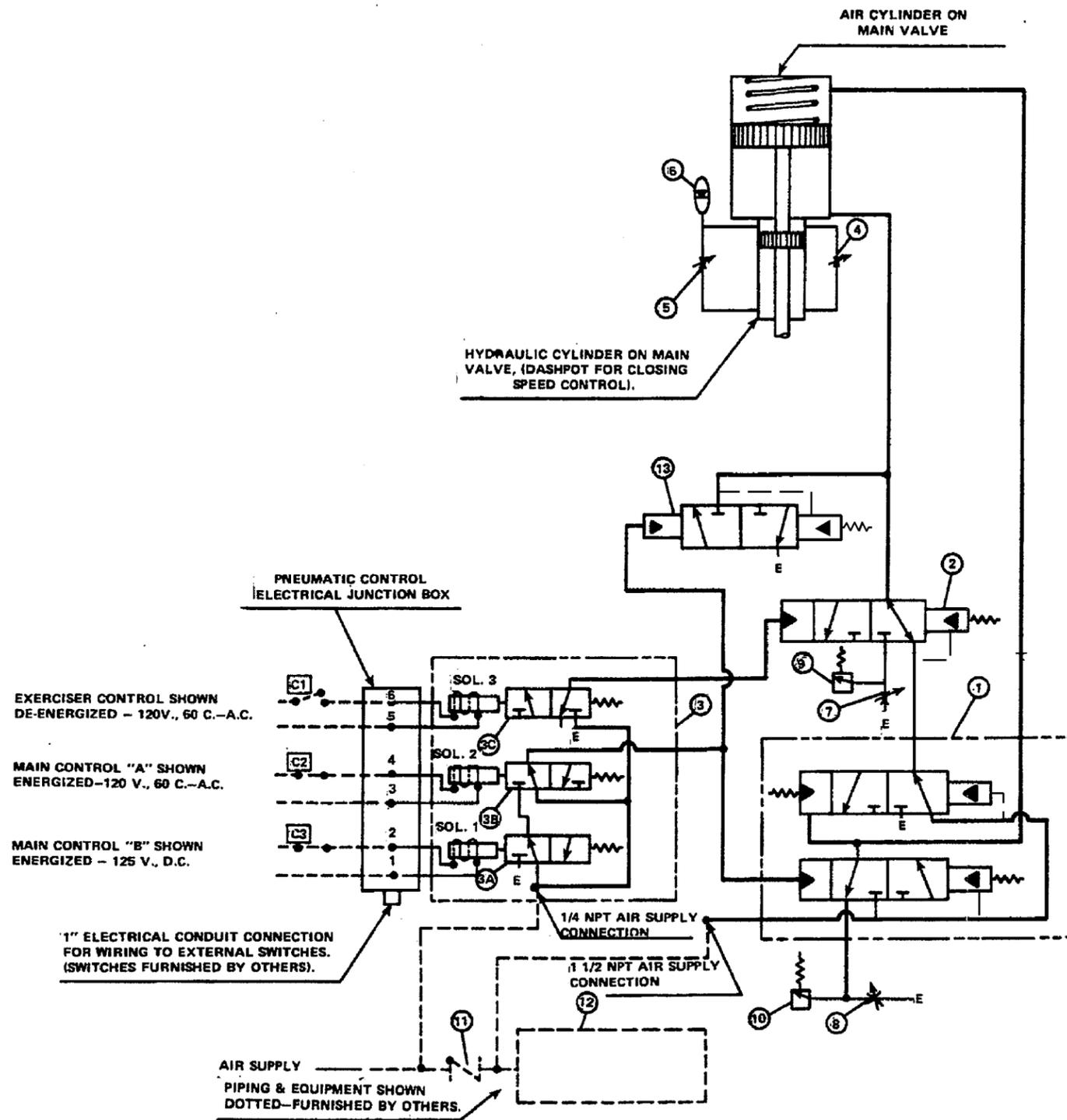
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FIGURE 7.3-9, SHEET 10
 RESIDUAL HEAT REMOVAL SYSTEM FCD



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FIGURE 7.3-10
 TYPICAL ISOLATION CONTROL SYSTEM FOR MAIN
 STEAM LINE ISOLATION VALVES



SCHEMATIC CONTROL DIAGRAM

LEGEND

- ① 1 1/4" FOUR-WAY, 5 PORTED, POPPET TYPE, AIR PILOT OPERATED VALVE (MAIN CONTROL VALVE).
- ② 1 1/4" THREE-WAY AIR PILOT OPERATED VALVE (EXERCISER CONTROL VALVE).
- ③ 1/4" SOLENOID PILOT VALVE ASSEMBLY, CONSISTING OF:
 - ③A 1/4" THREE-WAY D.C. SOLENOID OPERATED VALVE (MAIN PILOT CONTROL VALVE).
 - ③B 1/4" THREE-WAY A.C. SOLENOID OPERATED VALVE (MAIN PILOT CONTROL VALVE).
 - ③C 1/4" THREE-WAY A.C. SOLENOID OPERATED VALVE (EXERCISER CONTROL VALVE).
- ④ 3/4" NEEDLE VALVE (FOR FINE ADJUSTMENT OF CLOSURE SPEED).
- ⑤ 3/4" NEEDLE VALVE (FOR COARSE ADJUSTMENT OF CLOSURE SPEED).
- ⑥ ACCUMULATOR (TO COMPENSATE FOR THERMAL EXPANSION OF OIL).
- ⑦ 1/2" AIR METERING VALVE (FOR ADJUSTMENT OF EXERCISER CLOSURE SPEED).
- ⑧ 3/4" AIR METERING VALVE (FOR ADJUSTMENT OF OPENING SPEED).
- ⑨ 3/4" RELIEF VALVE.
- ⑩ 1" RELIEF VALVE.
- ⑪ 1 1/2" CHECK VALVE (FURNISHED BY OTHERS).
- ⑫ AIR STORAGE TANK (FURNISHED BY OTHERS).
- ⑬ 1/2" TWO-WAY, POPPET TYPE, AIR PILOT OPERATED VALVE.

OPERATION

TO OPEN VALVE - CLOSE SWITCH C2 AND/OR SWITCH C3.

TO CLOSE VALVE - OPEN SWITCHES C2 AND C3.

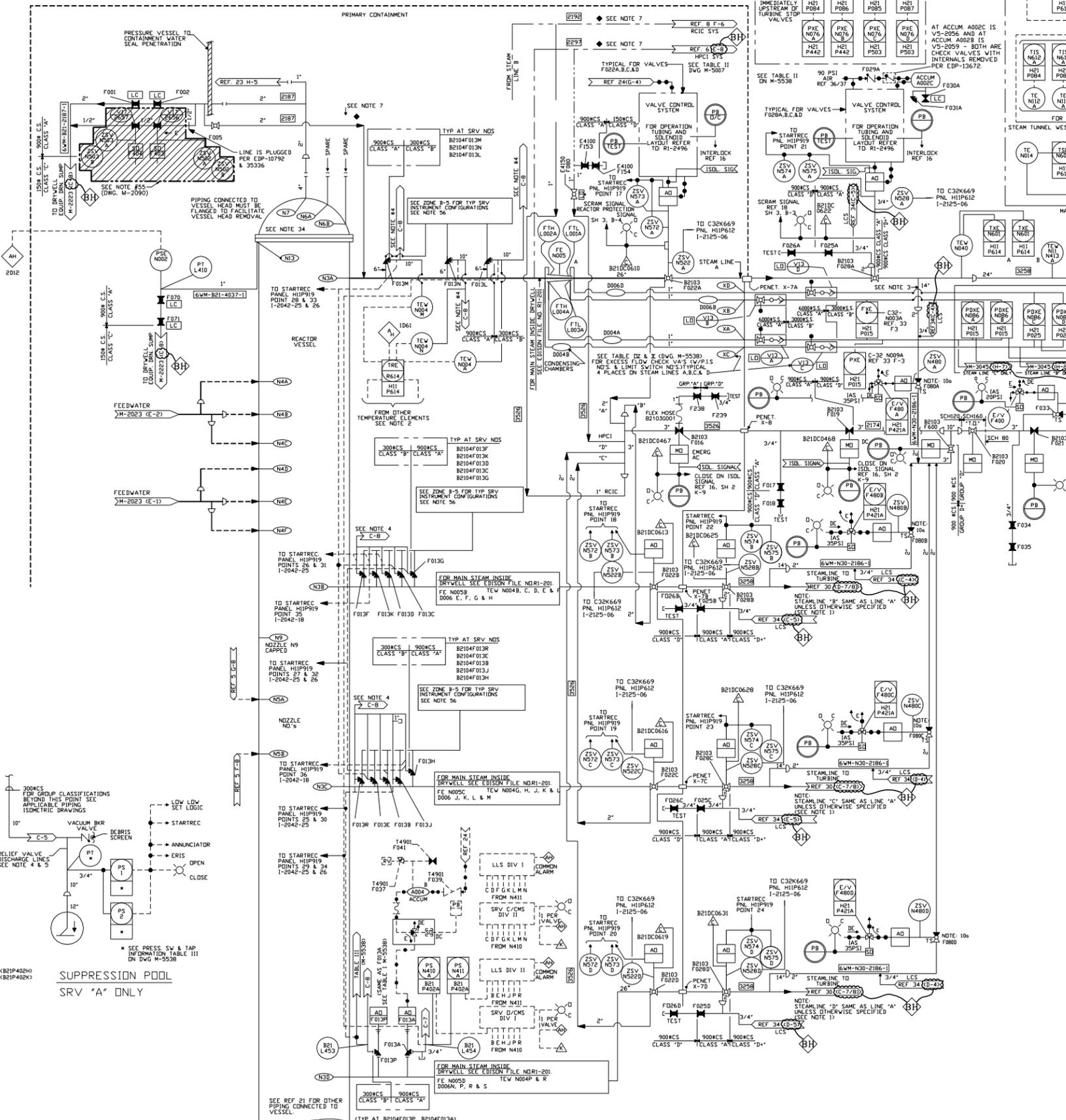
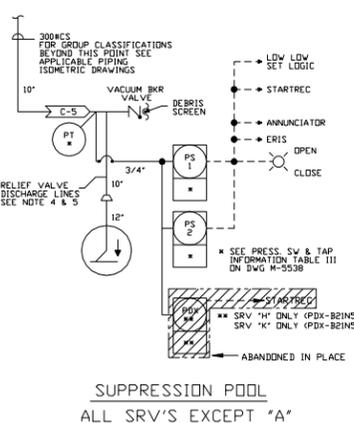
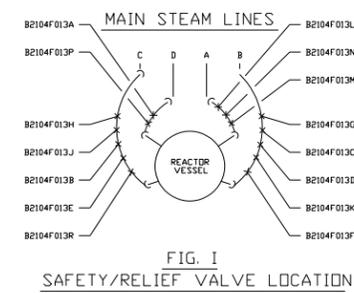
TO EXERCISE VALVE CLOSED - CLOSE SWITCH C1 (SWITCHES C2 AND C3 CLOSED).

TO EXERCISE VALVE OPEN - OPEN SWITCH C1 (SWITCHES C2 AND C3 CLOSED).

TO OPEN AFTER EXERCISING - RETURN SWITCH C1 TO NORMAL OPEN POSITION.

REF NO	TITLE	DECD NO	GE NO
1	DIAGRAM CONTROL ROD DRIVE HYDRAULIC SYSTEM (CRD)	M-2081	729E612AB
2	DIAGRAM TANK-BY LOGIC CONTROL SYSTEM (SLC)	M-2082	729E601AB
3	DIAGRAM RESIDUAL HEAT REMOVAL SYSTEM-DIVISION I (GRD)	M-2084	729E602AB
4	DIAGRAM RESIDUAL HEAT REMOVAL SYSTEM-DIVISION II (GRD)	M-2083	729E602AB
5	DIAGRAM CORE SPRAY SYSTEM (CSS)	M-2034	
6	DIAGRAM HIGH PRESSURE COOLANT INJECTION SYSTEM (HIC)	M-2035	729E600AB
7	DIAGRAM SYSTEM BARRIERS (SBD)	M-2043	729E600AB
8	DIAGRAM REACTOR CORE ISOLATION COOLING SYSTEM (RCS)	M-2044	729E604AB
9	DIAGRAM REACTOR CORE ISOLATION COOLING SYSTEM (RCS)	M-2045	729E604AB
10	DIAGRAM REACTOR WATER CLEAN-UP FILTER (RWC)	M-2046	731E348AB
11	DIAGRAM REACTOR WATER CLEAN-UP FILTER (RWC)	M-2047	731E348AB
12	DIAGRAM FUEL POOL COOLING SYSTEM (MFC)	M-2048	729E609AA
13	DIAGRAM REACTOR WATER CLEAN-UP SYSTEM (RWC)	M-2049	731A44AB
14	DIAGRAM REACTOR BLDG CLOSED COOLING & DECONTAMINATION SYSTEM	M-2027A	
15	DIAGRAM DEMINERALIZED SERVICE WATER SYSTEM	M-2678	
16	DIAGRAM NUCLEAR BOILER SYSTEM	M-2089	729E616AB
17	DIAGRAM REACTOR WATER CLEAN-UP SHAFT STATION (RWC)	M-2088	729E614AB
18	DIAGRAM CONDENSATE STORAGE SYSTEM	M-2006	
19	DIAGRAM STATION AIR SYSTEM	M-2085	
20	DIAGRAM FEEDWATER SYSTEM	M-2023	
21	DIAGRAM REACTOR RECIRCULATING SYSTEM	M-2833	729E603AB
22	DIAGRAM RESIDUAL HEAT REMOVAL SERVICE WATER SYSTEM	M-2012	
23	DIAGRAM NUCLEAR BOILER SYSTEM	M-2090	729E616AB
24	DIAGRAM PRIMARY CONTAINMENT PNEUMATIC SUPPLY SYSTEM	M-5007	
25	NUCLEAR BOILER FCD (LOGIC)		729E628AB
26	REACTOR PROTECTION SYSTEM DIAGRAM	I-2156-1	
27	RHR SYSTEM FCD		
28	CORE SPRAY SYSTEM FCD		
29	REACTOR RECIRC. SYSTEM FCD		
30	MAIN AND REHEAT STEAM SYSTEMS	M-2002	
31	RPS CONDUIT INSTALLATION	I-2866-1	
32	CONTROL AIR DIAGRAM	M-2015	
33	INST. DIAGRAM FEEDWATER CONT.	I-2126-1	732E153
34	MSIV LEAKAGE CONTROL SYSTEM	M-3045	
35	INSTRUMENT INFORMATION TABLES	M-5538	
36	INTERRUPTIBLE AND NON-INTERRUPTIBLE CONTROL AIR	M-4615	
37	INTERRUPTIBLE CONTROL AIR HEADER & BRANCH ROUTING	I-2453-6	

* M-2027 SUPERCEDED BY DWG'S M-5357, M-5358 & M-5444



- NOTES**
- STEAM LINES ENCLOSED IN BOXES SHALL HAVE PART NUMBERS CORRESPONDING TO ITS RESPECTIVE LINE NUMBER UNLESS OTHERWISE NOTED.
EXAMPLE: XXXB IS ON LINE "B"
XXXXC IS ON LINE "C"
 - ALL RELIEF AND SAFETY VALVE DISCHARGE THERMOCOUPLES SHALL BE CONNECTED TO TEMPERATURE RECORDER R614. SEE SRV REF. INFO TABLE I ON DWG M-5538.
 - BRAIN LINES SHALL BE INSTALLED AT LOW POINT OF STEAM LINES OTHER THAN REHEAT LINES. BRAIN LINES SHALL BE INSTALLED AS REQUIRED.
 - A SEPARATE LINE IS TO BE RUN INTO THE SUPPRESSION CHAMBER THROUGH THE VENT PIPING FOR EACH INDIVIDUAL RELIEF VALVE. LINE IS TO BE ROUTED TO PASS RELIEF VALVE CAPACITY WITHOUT EXCEEDING PRESSURE SPECIFIED ON PROCESS DIAGRAM M-5538.
 - PROVIDE HOLD-DOWN AGAINST BLOWING REACTION (INSIDE SUPPRESSION POOL).
 - DESIGN PRESSURE AND TEMPERATURE TO BE ESTABLISHED BY PIPING DESIGNER BASED ON FEED PUMP SHUT-OFF PRESSURE AND SYSTEM ARRANGEMENT. FOI TO BE DESIGNATED FOR MOST SEVERE UPSTREAM AND DOWNSTREAM SERVICE.
 - LINE INDICATED BY * ARE SINGLE LINES AND ARE CONNECTED ONLY WHERE ACTUALLY SHOWN.
 - INSTRUMENTS, INSTRUMENT PIPING AND VALVING MUST COMPLY WITH THE REQUIREMENTS OF PROCESS INSTRUMENT PIPING AND TUBING INSTALL. SPECIFICATION 2823732.
 - DESIGNATES VALVE LOCKED POSITION EITHER LOCKED OPEN/ LOCKED CLOSED AS INDICATED BY LD OR LC.
 - ALL MOTOR AND SOLENOID OPERATED VALVES ARE NORMAL AC UNLESS OTHERWISE NOTED. *V17-2100 IS DC OPERATED.
 - VALVES BE100708A-D FAIL OPEN (FO) ON LOSS OF AIR AND FAIL CLOSE (FC) ON LOSS OF ELECTRICAL POWER.
 - ALL VENTS, DRAINS AND RELIEF VALVE DISCHARGE TO CRW OR DRW.
 - FOR LOCATION AND IDENTIFICATION OF INSTRUMENTS SEE DECD.
 - AN EXPANSION LEG SHALL BE PROVIDED IN INSTRUMENT SENSING LINE BETWEEN POT (PART 0007) AND THE WATER TIGHT PENETRATION SEAL BOTTOM OF REACTOR (PART 0007). THE EXPANSION LEG AND PIPING INSTALLATION SHALL BE DESIGNED TO ALLOW FOR MAXIMUM CHANGE OF VESSEL LENGTH WITH TEMPERATURE TO AVOID OVERSTRESSING THE PIPING, OR THE SEAL OR DAMAGE TO THE INSULATION AROUND THE VESSEL.
- NOTES CONTINUED ON DRAWING M-2090---

- REFERENCES:**
- ISOMETRIC DRAWING M-3259-1
 - INSTRUMENT INFORMATION TABLES, SEE DRAWING M-5538.
 - MOTOR CONT. LOGIC DIAG. I-2070-08

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FIGURE 7.3-12, SHEET 1
NUCLEAR STEAM SUPPLY SYSTEM
P&ID

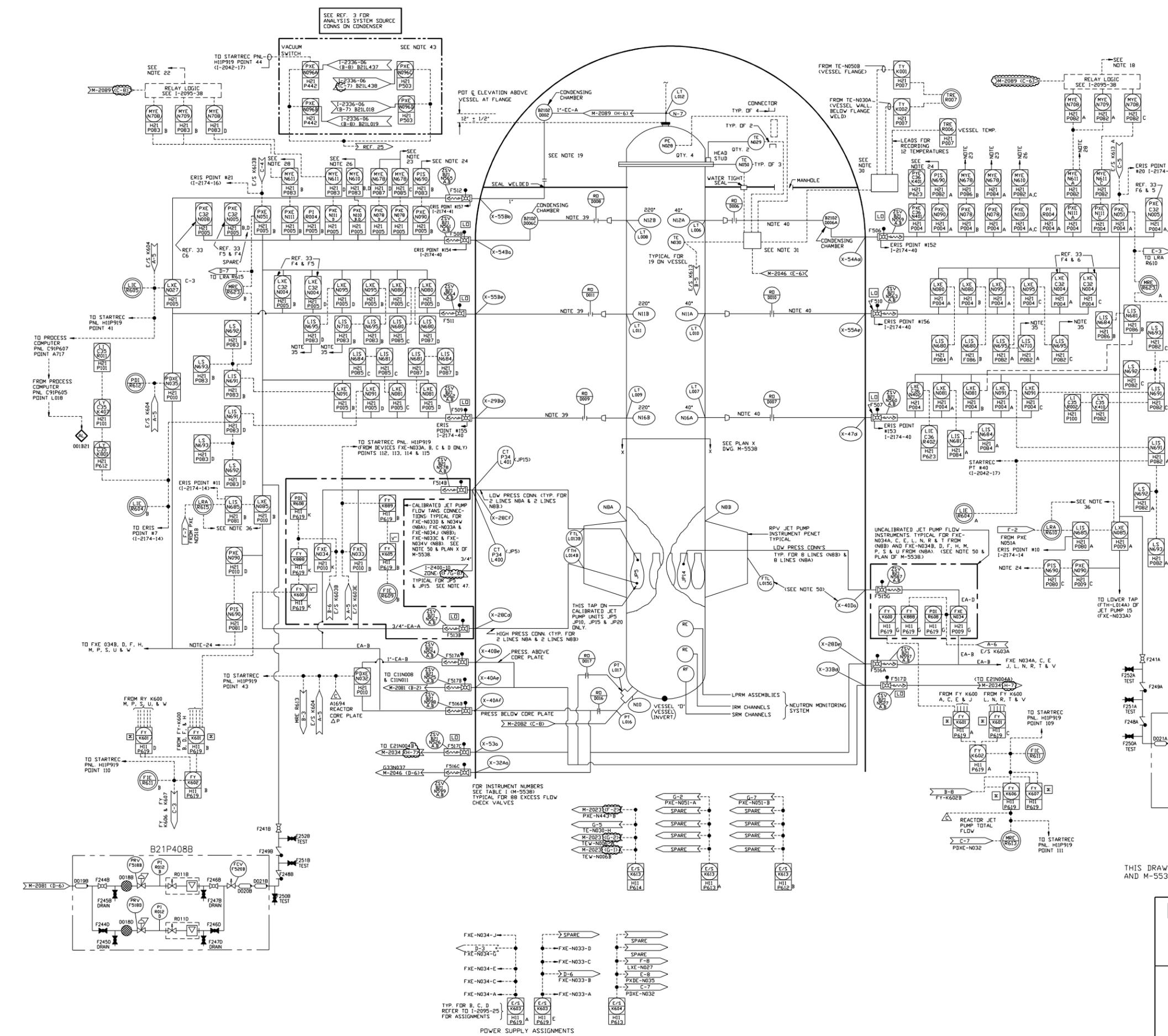
NOTES: (CONTINUED FROM DWG. M-2089)

- 14. LINES TO DIFFERENTIAL PRESSURE TRANSMITTERS SHOULD BE SHORT AS PRACTICAL.
- 15. PIPING HIGH VENT AND LOW POINT DRAINS ARE TO BE ADDED ON ALL SUCH HIGH POINTS OR LOW POINTS NOT SERVED BY EQUIPMENT VENTS AND DRAINS.
- 16. CHEMICAL CLEANING CONNECTIONS, VALVES, ETC. ARE TO BE PROVIDED AS NECESSARY.
- 17. UNLESS OTHERWISE SHOWN ALL INSTRUMENT PIS NUMBERS ARE PREFIXED B21 AND ALL VALVE & EQUIPMENT PIS NUMBERS ARE PREFIXED B2100.
- 18. LOW LOW SET LOGIC-OPEN SRV-F0134.
- 19. PROVISION FOR INSTRUMENT LINE ISOLATION IS BY EXCESS FLOW CHECK VALVES. AN ORIFICE (1/4" MIN) IS TO BE PROVIDED WITHIN THE PRIMARY CONTAINMENT IN EACH INSTRUMENT LINE WHICH CONNECTS TO THE REACTOR COOLANT PRESSURE BOUNDARY. POSITION INDICATION IS PROVIDED IN THE CONTROL ROOM.
- 20. DELETED
- 21. ALARMS ASSOCIATED WITH THE SYSTEMS INITIATED BY THE REACTOR PROTECTION SYSTEM OR SAFEGUARD SYSTEM LEVEL AND PRESSURE INSTRUMENTATION IS SHOWN ON THE PAID FOR THE PARTICULAR SYSTEM.
- 22. LOW LOW SET LOGIC-OPEN SRV-F013G.
- 23. SCRAM (REF. 26) AND INITIATE PRIMARY CONTAINMENT ISOLATION EXCEPT MAIN STEAM LINE.
- 24. CORE SPRAY AND RHR SYSTEM VALVE OPENING PERMISSIVE (REF. 27 & 28).
- 25. MAIN STEAM LINE ISOLATION VALVE CLOSURE INTERLOCK (REF. 26) & CONTRIBUTE TO LOW CONDENSER VACUUM BY-PASS (REF. 25).
- 26. RHR INTERLOCK (PCI MODE) (REF. 27).
- 27. ALTERNATE TAP SET ON FEEDWATER FLOW ELEMENT.
- 28. REACTOR PUMP (REF. 29).
- 29. DELETED
- 30. T/C TYPE JUNCTION BOX LOCALLY MOUNTED. EACH T/C JUNCTION BOX TO HAVE OWN SET OF TERMINALS.
- 31. WATER TIGHT JUNCTION BOX TO BE LOCATED INSIDE DRYWELL.
- 32. SUMMER K606 & K607 INPUTS SHALL BE INTERLOCKED WITH REACTOR PUMP & VALVES TO ADD INPUTS WHEN BOTH PUMPS ARE RUNNING & THEIR DISCHARGE VALVES ARE OPEN OR SUBTRACT ONE INPUT WHEN THE CORRESPONDING PUMP IS STOPPED OR ITS DISCHARGE VALVE IS CLOSED.
- 33. DELETED
- 34. M272LS 6A & 6B MAY BE INTERCHANGED.
- 35. RPV LEVEL EIGHT TRIP OF REACTOR FEED PUMP TURBINE MAIN TURBINE GENERATOR & STAND BY FEED WATER SYSTEM.
- 36. INSTRUMENTS READ FULL SCALE WHEN JET PUMPS ARE IN OPERATION.
- 37. DELETED
- 38. MAIN BENCH BOARD LIGHTS OR LIGHT DISPLAYS ARE TO BE PER DECD PANEL DWGS. H11P601, P602 & P603.
- 39. ROUTE INSTRUMENT LINES TO DIVISION II SIDE DRYWELL PENETRATIONS.
- 40. ROUTE INSTRUMENT LINES TO DIVISION I SIDE DRYWELL PENETRATIONS.
- 41. RECIRCULATION LINES TO HOTWELL TO COMPLY WITH REF. 20 WATER QUANTITY SECTION 7.
- 42. SAMPLE PROBE(S) AND FEEDWATER SAMPLE STATION TO COMPLY WITH REF. 20 WATER SAMPLING, SECTION 7.
- 43. LOW CONDENSER VACUUM SWITCHES CONNECTED THROUGH SEPARATE CALIBRATION VALVES TO OPPOSITE SIDES OF THE CONDENSER ABOVE THE HIGH CONDENSATE LEVEL. THE VACUUM SWITCHES MUST BE ACCESSIBLE DURING PLANT OPERATION.
- 44. DELETED
- 45. SEE PROPER SYSTEM FCD OR IED FOR FUNCTION OF THE WATER LEVEL TRIP.
- 46. AVAILABLE CORROSION DATA INDICATES THAT FOR BARE CARBON STEEL THE SRV DISCHARGE PIPING BELOW SUPPRESSION POOL WATER LEVEL SHOULD HAVE A CORROSION ALLOWANCE OF 0.250 INCHES. FOR A 40 YEAR LIFE SINCE THIS PIPING WILL CORRODE FROM BOTH SIDES. THE IN-SERVICE INSPECTION PROGRAM SHOULD INCLUDE OCCASIONAL SURVEILLANCE OF THIS PIPING PARTICULARLY IF A SMALLER CORROSION ALLOWANCE IS USED.
- 47. TYPICAL FOR TWO JET PUMPS 180° APART. POST ACCIDENT SAMPLE TAPS SHALL BE LOCATED DOWNSTREAM FROM THE 1/4" RESTRICTED DRIFICE. THE POST ACCIDENT SAMPLING SHALL BE LEAKTIGHT.
- 48. FOR LLS VALVES. SEE TABLE I (M-5538) FOR THE ASSOCIATED ACCUMULATOR.
- 49. B2109A,B,C,D,E,F,G, & H AND B2109A4,B,C,D,E,F,G, & H ARE SHOWN ON REF. 3 & 4.
- 50. SEE TABLE IX ON M-5538 FOR CORRELATION OF JET PUMP NOS. RPV NOZZLE NO., FLOW TAP NO., PENET NO. & EXCESS FLOW CHECK VLV. NO.
- 51. DELETED
- 52. THESE LINES HAVE BEEN DUPLICATED ON DIAGRAM M-2985 AND SHOULD BE USED IN CONJUNCTION WITH THIS DIAGRAM.
- 53. REFERENCE TABLE I ON M-5538 FOR SAFETY RELIEF VALVE INFORMATION, TABLE I IN VENDOR MANUAL VHR-46 FOR SRV CROSS REFERENCE MATRIX AND CSD DATA BASE. FOR CURRENTLY INSTALLED SRV PIS NUMBER ALTERNATE I.D. AND MANUFACTURER BODY/PILOT ASSY. & SERIAL NUMBERS.
- 54. □ DESIGNATES ISOMETRIC DRAWING NUMBER (PREIX 6M721-) OR VALVE LOCKED POSITION EITHER LOCKED OPEN, LOCKED CLOSED AS INDICATED BY 'LO' OR 'LC'.
- 55. VALVES B2103F, 403 & F444. ASSOCIATED CONNECTIONS, WIRES AND CABLES HAVE BEEN DE-ENERGIZED, SPARED, AND ABANDONED IN PLACE.
- 56. SEE TABLES I & III ON M-5538 FOR APPROPRIATE INSTRUMENT NUMBERS, ASSOCIATED COMPONENTS AND MSRV DISCHARGE PIPING ISOMETRIC DRAWINGS.

THIS DRAWING TO BE USED IN CONJUNCTION WITH M-2089 AND M-5538

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FIGURE 7.3-12 SHEET 2
NUCLEAR STEAM SUPPLY SYSTEM
P&ID



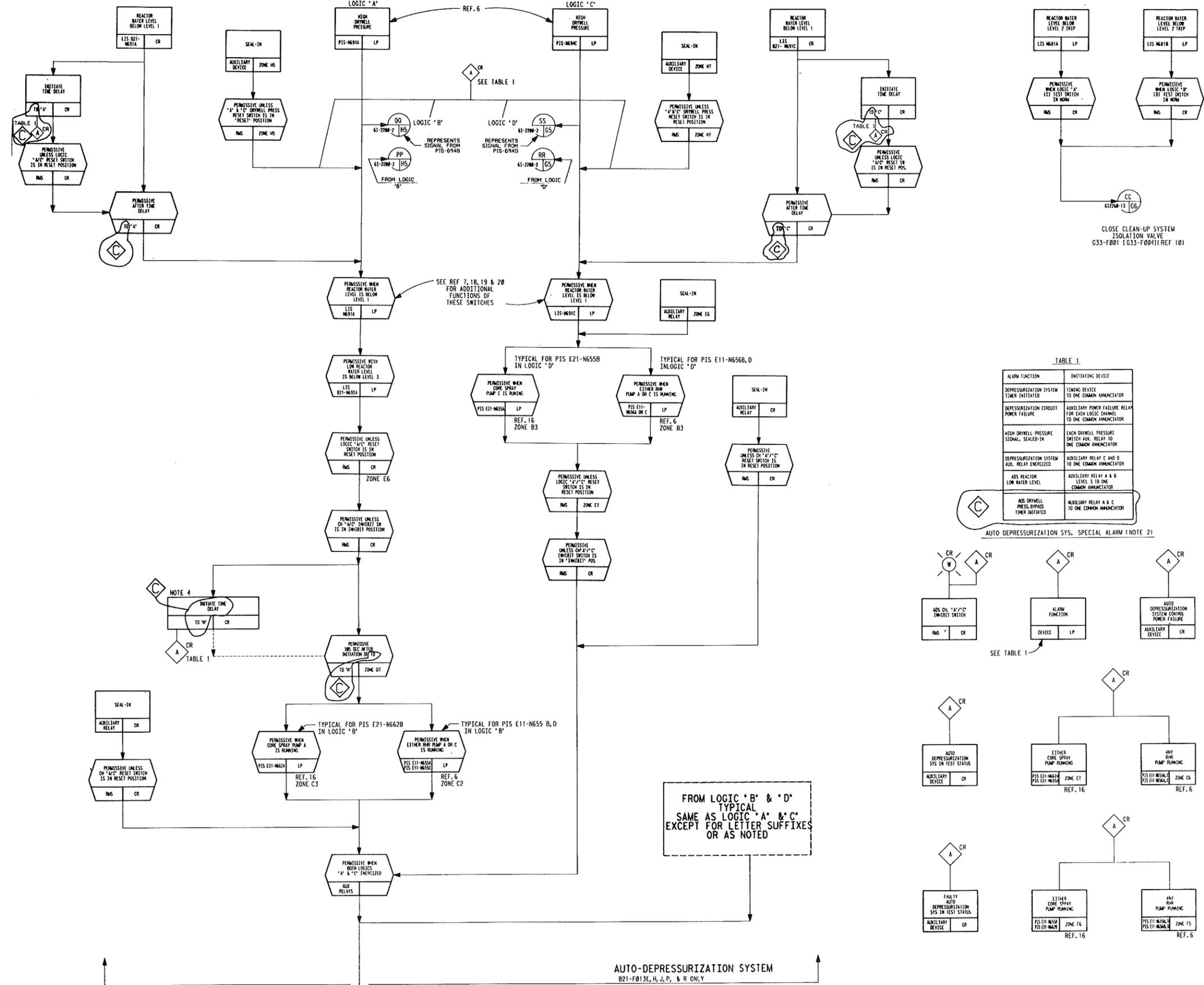
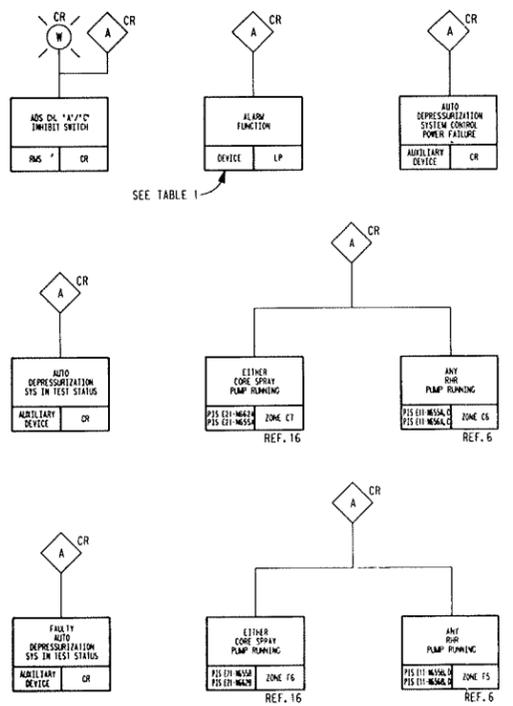


TABLE 1

ALARM FUNCTION	INITIATING DEVICE
DEPRESSURIZATION SYSTEM TIMER INITIATED	TIMING DEVICE TO ONE COMMON ANNUNCIATOR
DEPRESSURIZATION CIRCUIT POWER FAILURE	AUXILIARY POWER FAILURE RELAY FOR EACH LOGIC CHANNEL TO ONE COMMON ANNUNCIATOR
HIGH DRYWELL PRESSURE SIGNAL, SEALED-IN	EACH DRYWELL PRESSURE SWITCH AUX. RELAY TO ONE COMMON ANNUNCIATOR
DEPRESSURIZATION SYSTEM AUX. RELAY ENERGIZED	AUXILIARY RELAY C AND D TO ONE COMMON ANNUNCIATOR
ADS REACTOR LOW WATER LEVEL	AUXILIARY RELAY A & B LEVEL 3 TO ONE COMMON ANNUNCIATOR
ADS DRYWELL PRESS. DIFFERS TIMER INITIATED	AUXILIARY RELAY A & C TO ONE COMMON ANNUNCIATOR

AUTO DEPRESSURIZATION SYS. SPECIAL ALARM NOTE 21



NOTES

1. WHEN TEST SOLENOID PILOT IS ENERGIZED, THE MAIN STEAM ISOLATION VALVE OPERATOR IS SLOWLY EXHAUSTED (60 SECOND CLOSURE TIME) WHILE VALVE IS CLOSED BY ACTION OF THE VALVE SPRING WITHOUT AID OF AIR PRESSURE.
2. THE ALARMS AND VALVE INDICATING LIGHTS SHOWN ON THE LOGIC DIAGRAM ARE SYSTEM REQUIREMENTS IN ADDITION TO THOSE SHOWN ON THE SYSTEM P&ID. ADDITIONAL INFORMATION ON ALARMS, VALVE POSITION INDICATING LIGHTS AND PROCESS INSTRUMENTATION NOT SHOWN ON THIS LOGIC DIAGRAM MAY BE OBTAINED FROM REFERENCE #1. AUXILIARY RELAY & DEVICES ARE NOT SHOWN ON LOGIC DIAGRAM EXCEPT WHERE REQUIRED TO CLARIFY FUNCTION.
3. MAIN STEAM LINE ISOLATION LOGIC A, B, C, D AND AC SOLENOID LOGIC SHALL BE POWERED FROM REACTOR PROTECTION SYSTEM W-C SETS, DC SOLENOID LOGIC SHALL BE POWER FROM STATION BATTERIES.
4. TIMING DEVICES SHALL BE A TYPE THAT RESETS ON LOSS OF POWER OR LOSS OF INITIATING SIGNAL.
5. ALL EQUIPMENT AND INSTRUMENTS ARE PREFIXED BY THE SYSTEM NUMBER "B21" UNLESS OTHERWISE NOTED.
6. FOR LOCATION AND IDENTIFICATION OF INSTRUMENTS SEE INSTRUMENT DATA SHEET LISTED IN MASTER PARTS LIST (MPL) FOR EACH INSTRUMENT.
- 7.
8. THE NUCLEAR BOILER SYSTEM SHALL BE DESIGNED IN ACCORDANCE WITH REFERENCE #17 AND WITH "PROPOSED CRITERIA FOR NUCLEAR POWER PLANT PROTECTION SYSTEMS IEEE 279" AS APPLICABLE TO THE CONTROL CIRCUITRY.
9. ISOLATION LOGIC SHALL BE "FAILSAFE" I.E., LOGIC SHALL BE DESIGNED TO INITIATE ISOLATION FUNCTIONS WHEN DE-ENERGIZED.
10. FOR CONFIGURATION OF PANEL MOUNTED OPERATORS AND INDICATORS SEE DRAWING 6M-2290 SHEET 2, REFERENCE #21, AND PURCHASE SPECIFICATION DATA SHEET SECTION 2.4.
11. THE DEVICES IN THIS AREA AS WELL AS OTHER DEVICES IN THIS LOGIC ARE ESSENTIAL AND MUST MEET THE REQUIREMENTS OF IEEE-279.
12. FOR TRIP SETTINGS SEE NUCLEAR BOILER SYSTEM DESIGN SPECIFICATION DATA SHEET REFERENCE #26.
13. TYPICAL FOR EACH VALVE, FOR VALVE NUMBERS AND DEVICE SUFFIX LETTERS SEE REFERENCE #1.
14. ADDITIONAL NUCLEAR BOILER SYSTEM LOGIC IS SHOWN ON REFERENCES #4, #7, #18, #19 & #20.
- 15.

REFERENCE DOCUMENTS

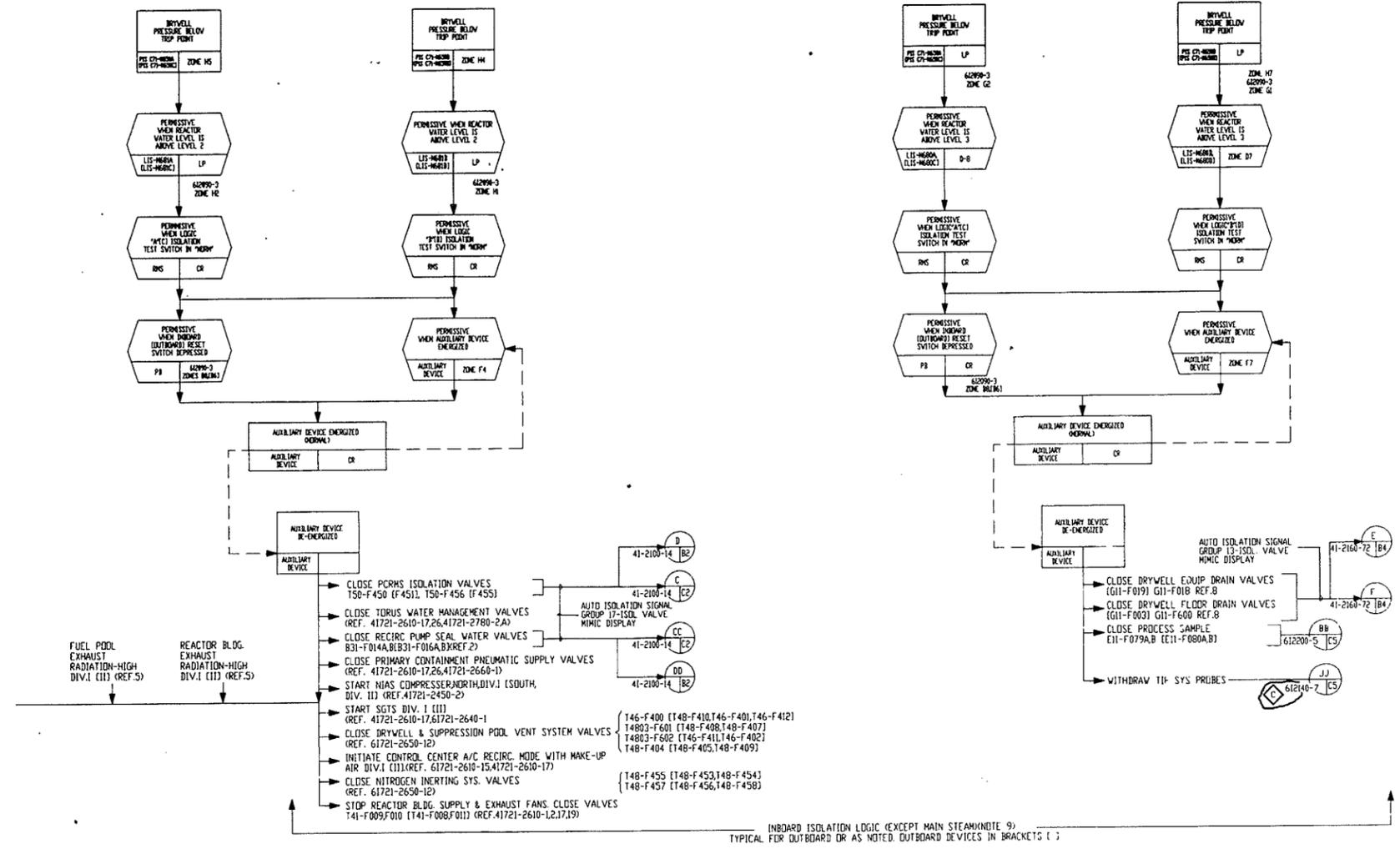
1. NUCLEAR BOILER SYSTEM P&ID.....6M-2089, 6M-2090, 6M-5538
2. REACTOR RECIRCULATION SYSTEM LOGIC DIAGRAM.....6I-2100
3. NEUTRON MONITORING SYSTEM LOGIC DIAGRAM.....6I-2140
4. REACTOR PROTECTION SYSTEM IED.....6I-2156-1, 2
5. PROCESS RADIATION SYSTEM IED.....6I-2181-1 THRU 6
6. RHR SYSTEM P&ID.....6M-2083, 6M-2084
7. RHR SYSTEM LOGIC DIAGRAM.....6I-2200
- 8.
- 9.
10. REACTOR WATER CLEANUP SYSTEM P&ID.....6M2046, 6M2047
11. REACTOR WATER CLEANUP SYSTEM LOGIC DIAGRAM.....6I-2260
12. NBS SCHEMATIC CONTROL DIAGRAM.....6I-2095-1 THRU 38
13. LOGIC SYMBOLS.....20944756; (RI-20)
14. PIPING & INSTRUMENT SYMBOLS.....1970567; (RI-2507)
15. REACTOR RECIRCULATION SYSTEM P&ID.....6M-2033, 6I-2106-1, 2
16. CORE SPRAY P&ID.....6M-2034
17. ELECTRICAL EQUIPMENT SEPARATION FOR SAFEGUARD SYS..22A3777; (RI-1410)
18. CORE SPRAY LOGIC DIAGRAM.....6I-2210
19. HPCI LOGIC DIAGRAM.....6I-2220
20. RCIC LOGIC DIAGRAM.....6I-2230
21. GENERAL REQUIREMENTS FOR INSTR & ELECT EQUIP.....22A1768; (RI-445, 483)
22. BWR PLANT REQUIREMENT.....22A13080C; (RI-3231)
23. STEAM LEAK DETECTION SYSTEM.....22A1441; (RI-138)
24. 22A1441AE; (RI-139)
25. STANDBY LIQUID CONTROL SYSTEM LOGIC DIAGRAM.....6I2130
26. NUCLEAR BOILER SYSTEM DESIGN SPEC DATA SHEET.....22A2919AB; (RI-207)
27. NUCLEAR BOILER SYSTEM DESIGN SPEC.....22A2919; (RI-200)
28. LOGIC DIAGRAM-GROUP 1 ISOL. VALVE MIMIC DISPLAY...4I721-2090-8
29. LOGIC DIAGRAM-GROUP 4 ISOL. VALVE MIMIC DISPLAY...4I721-2200-13
30. LOGIC DIAGRAM-GROUP 13 ISOL. VALVE MIMIC DISPLAY...4I721-2160-72
31. LOGIC DIAGRAM-GROUP 17 ISOL. VALVE MIMIC DISPLAY...4I721-2100-14
32. LOGIC DIAGRAM-GROUP 2 ISOL. VALVE MIMIC DISPLAY...4I721-2100-13

LEGEND:

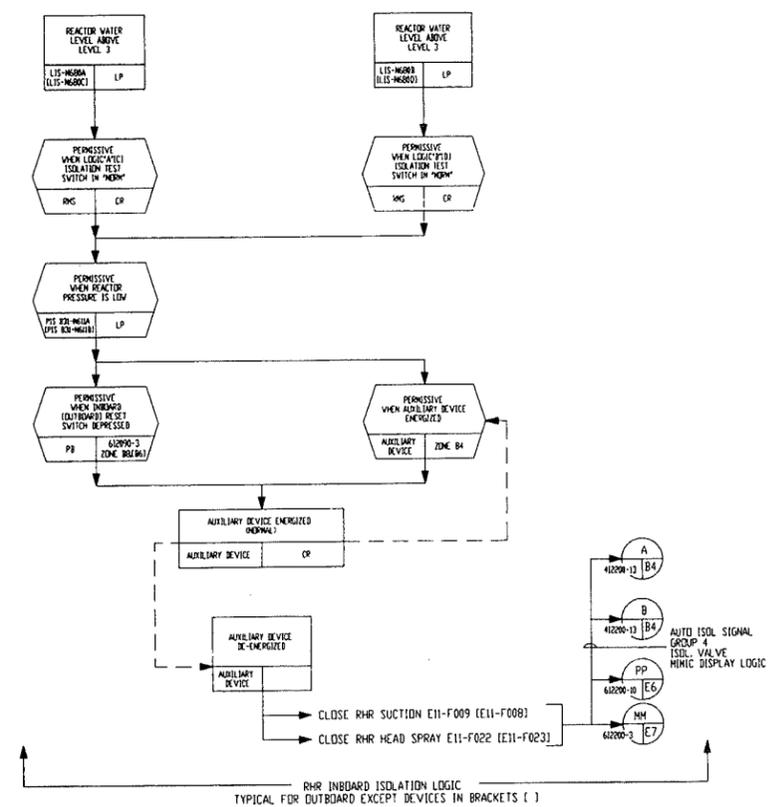
- * SWITCHGEAR DEVICE FUNCTION NUMBER ANSI SPEC C37.2
- * FURNISHED AS PART OF VALVE OR DEVICE ASSEMBLY
- PB = PUSHBUTTON SWITCH
- ADS = AUTO DEPRESSURIZATION SYSTEM
- CR = CONTROL ROOM
- LP = LOCAL
- TD = TIME DELAY

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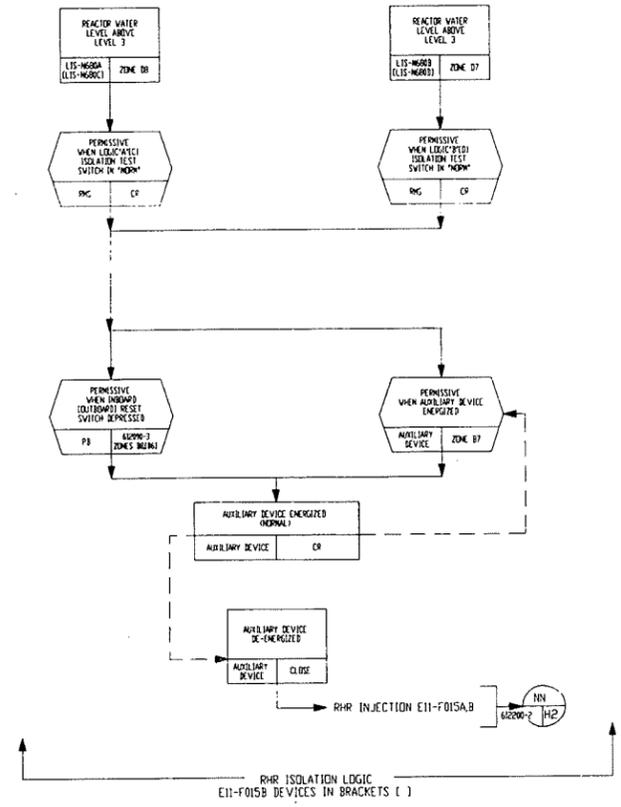
FIGURE 7.3-13, SHEET 1
 NUCLEAR BOILER SYSTEM FCD



INBOARD ISOLATION LOGIC (EXCEPT MAIN STEAMNOTE 9)
TYPICAL FOR OUTBOARD OR AS NOTED. OUTBOARD DEVICES IN BRACKETS ()



RHR INBOARD ISOLATION LOGIC
TYPICAL FOR OUTBOARD EXCEPT DEVICES IN BRACKETS ()

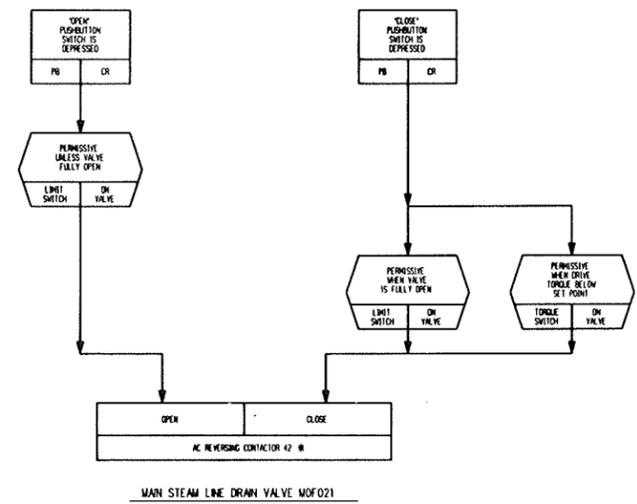
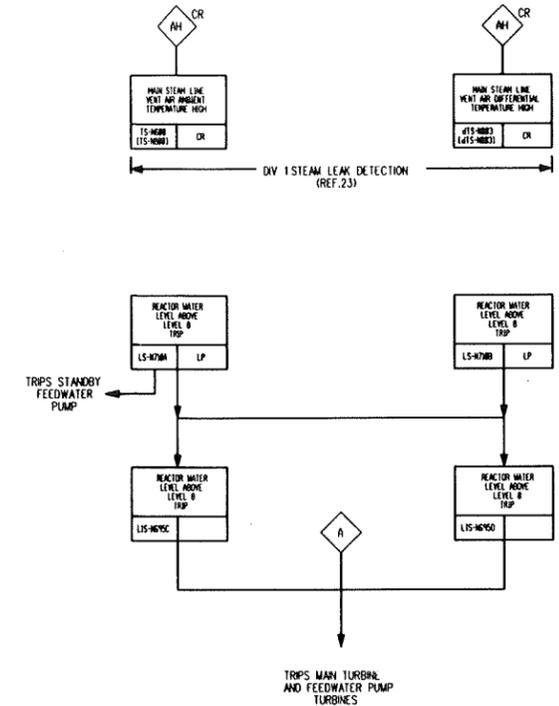
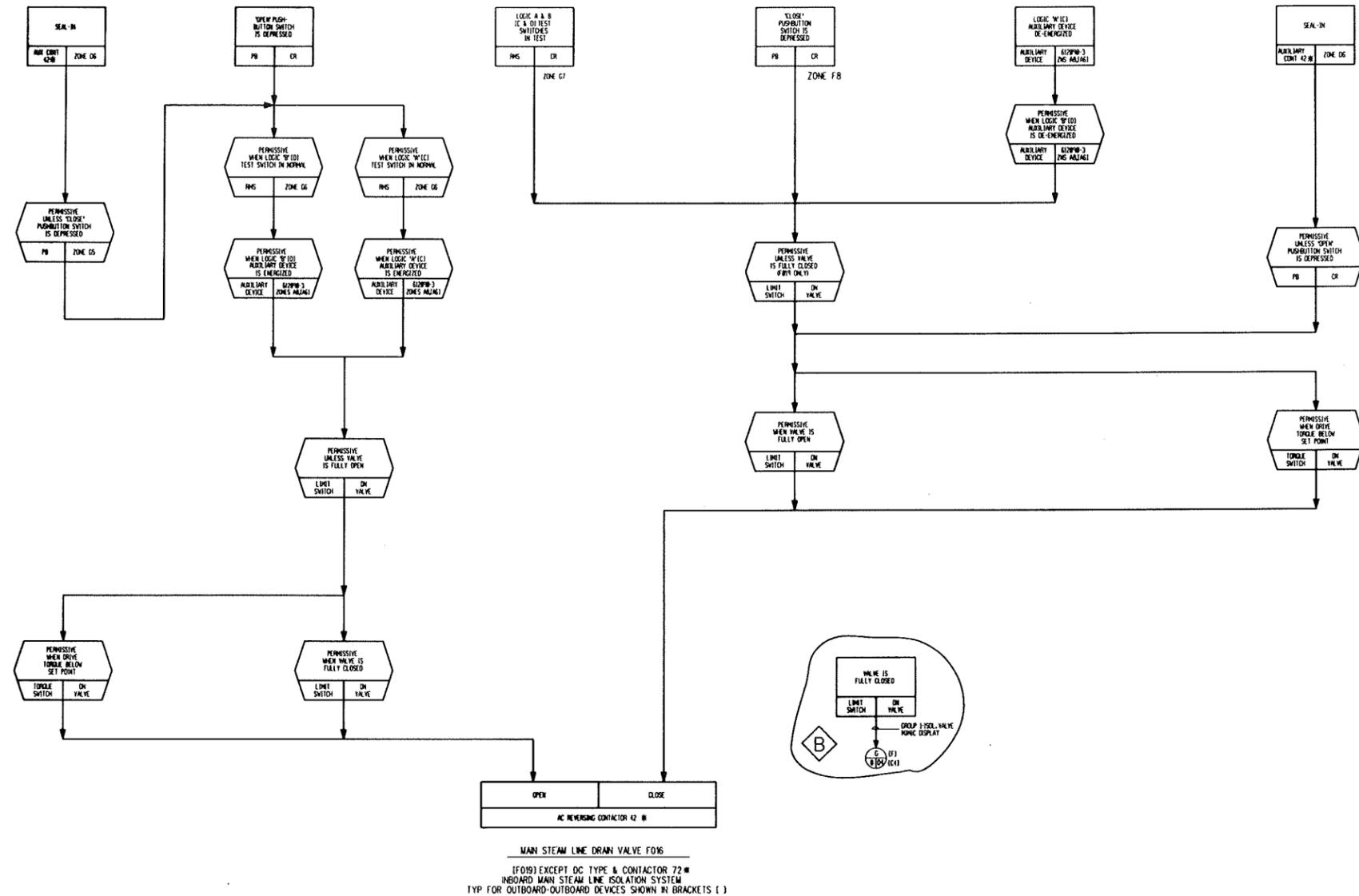


RHR ISOLATION LOGIC
E11-F015B DEVICES IN BRACKETS ()

NOTES
SEE 61721-2090-1 FOR NOTES, REFERENCES, AND LEGEND.

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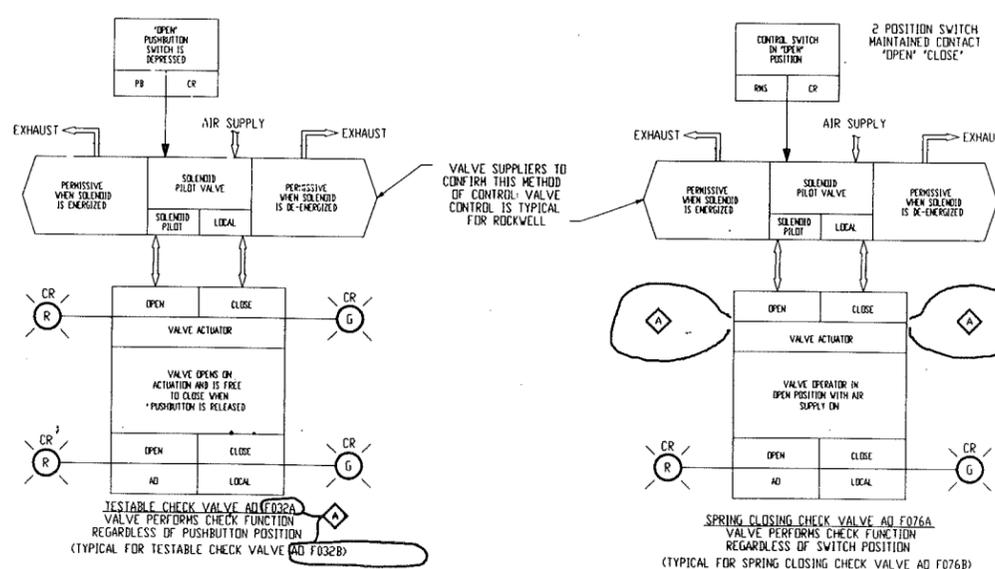
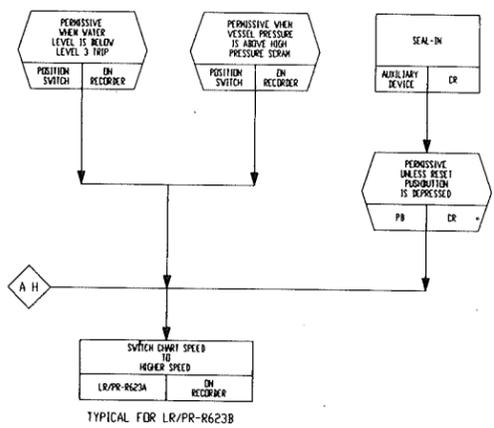
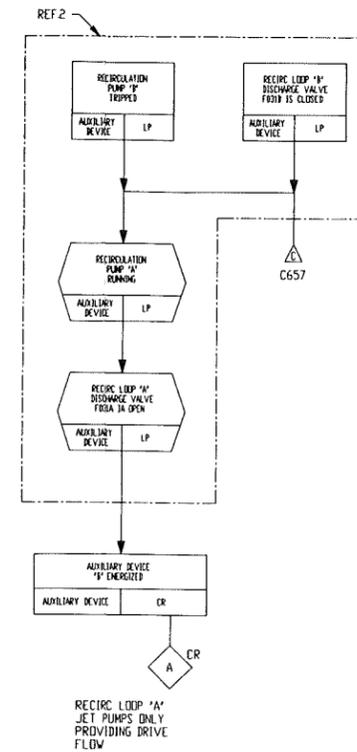
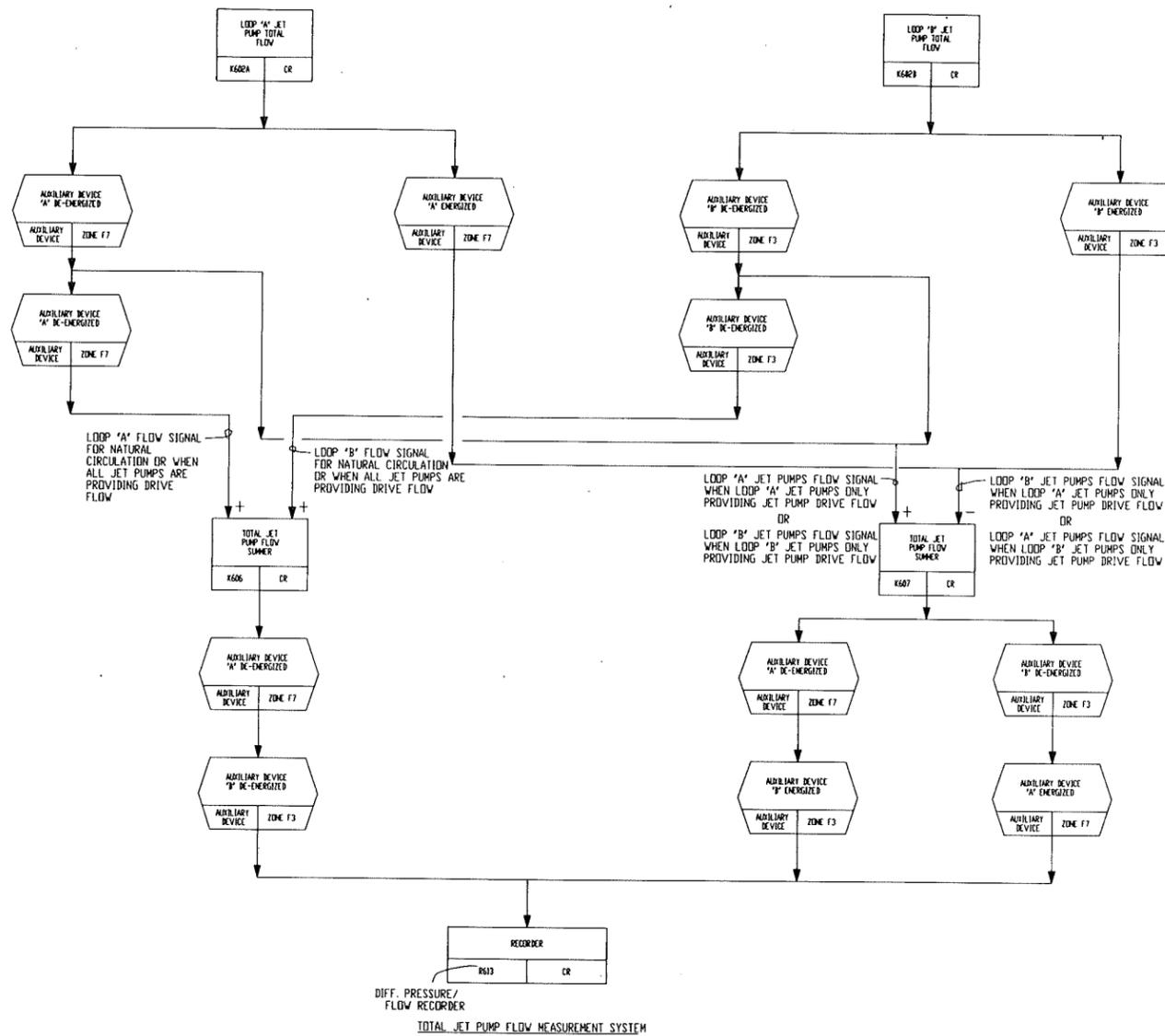
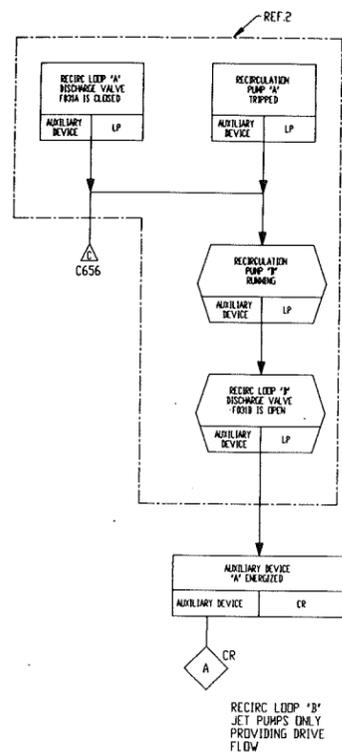
FIGURE 7.3-13, SHEET 2
NUCLEAR BOILER SYSTEM FCD



NOTES
 1. SEE 61721-2090-1 (FIGURE 7.3-13, SHEET 1)
 FOR NOTES, REFERENCES, AND LEGEND.

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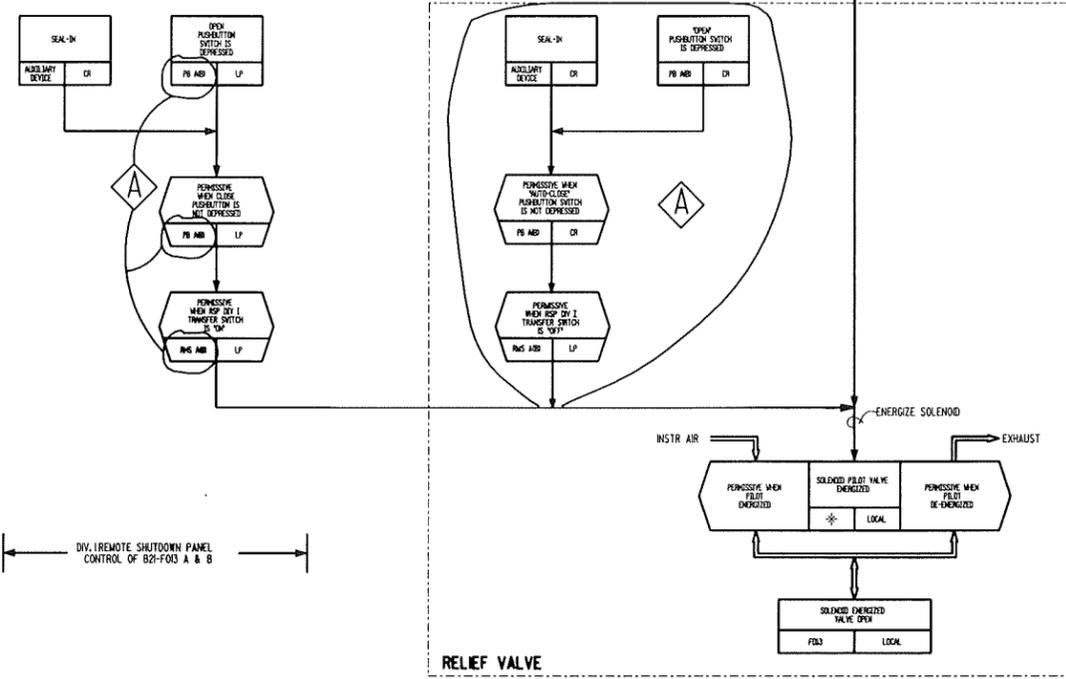
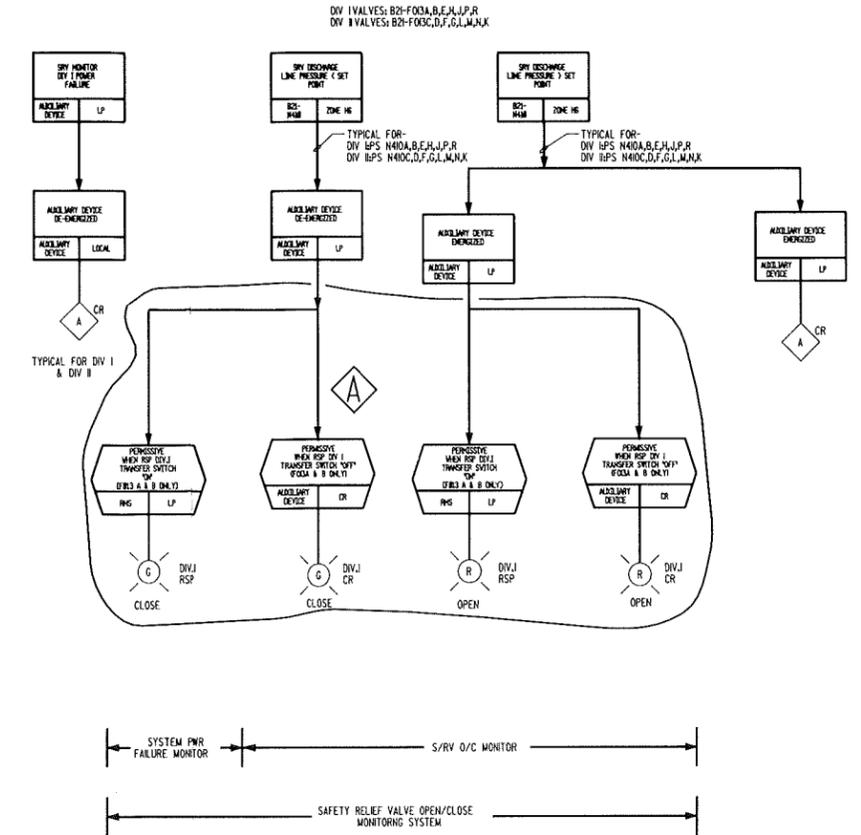
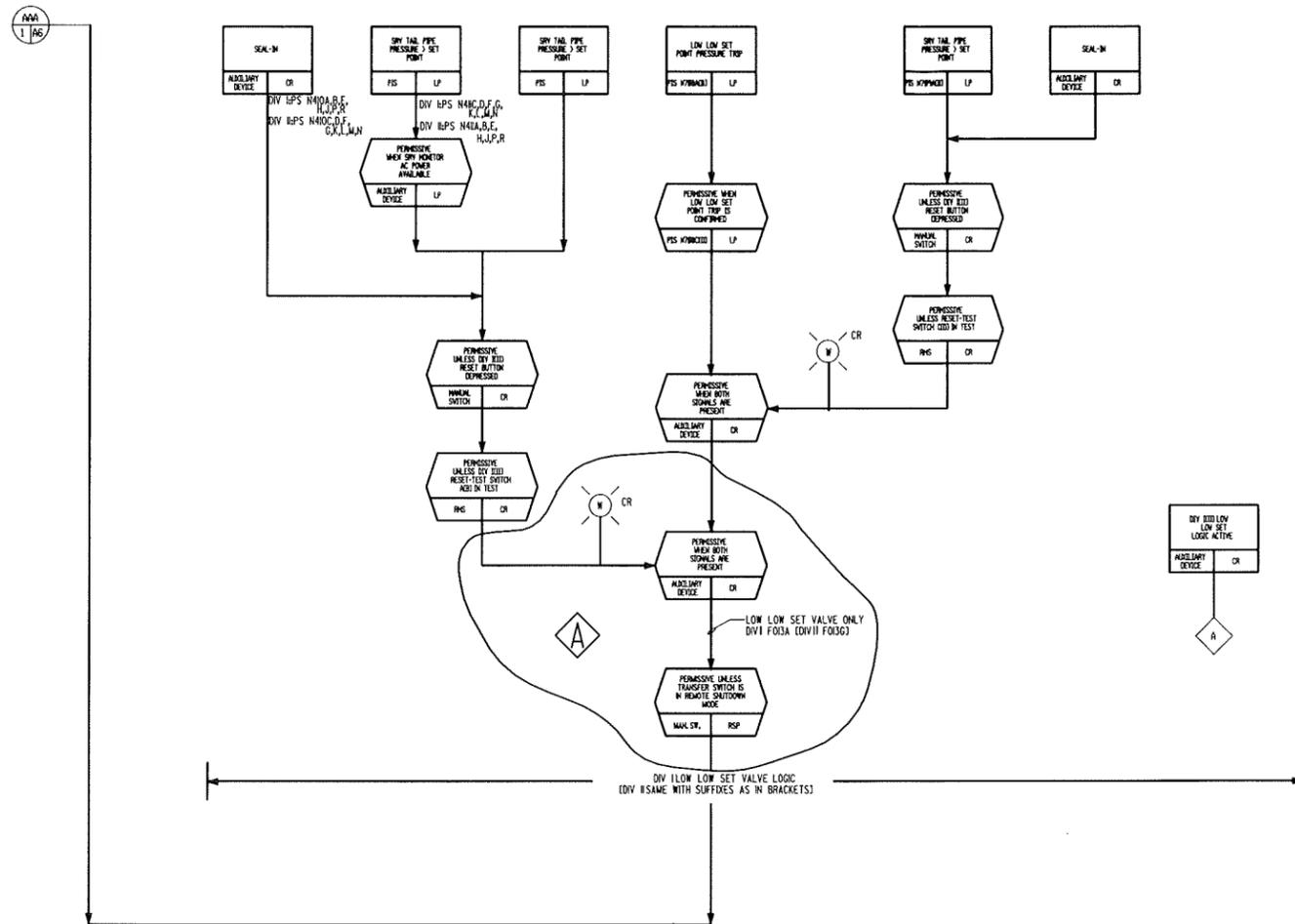
FIGURE 7.3-13, SHEET 4
 NUCLEAR BOILER SYSTEM FCD



NOTES
 1. SEE 61721-2090-1 (FIGURE 7.3-13, SHEET 1) FOR NOTES, REFERENCES, AND LEGEND.

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FIGURE 7.3-13, SHEET 5
 NUCLEAR BOILER SYSTEM FCD



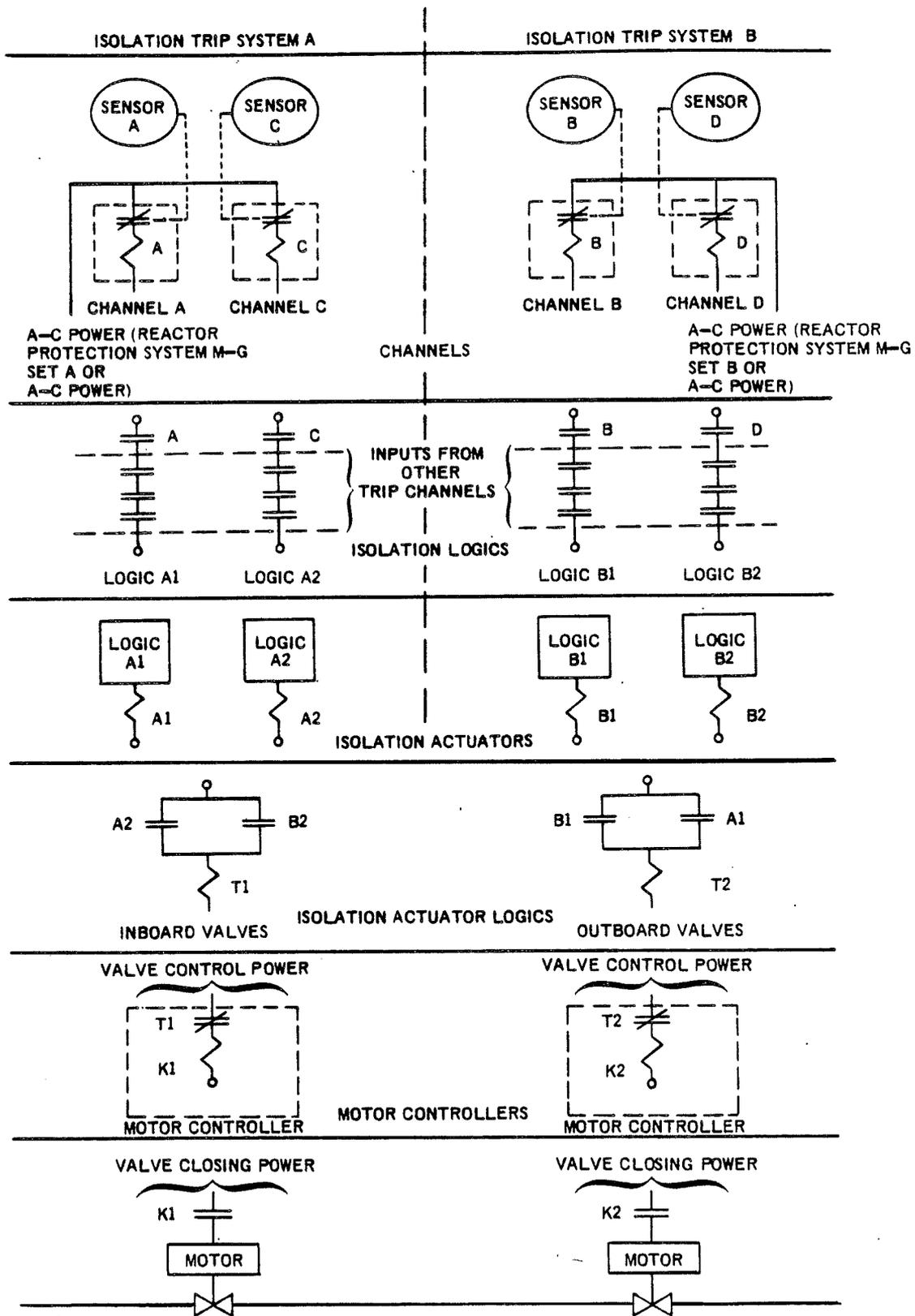
- NOTES
- SEE 61721-2090-1 (FIGURE 7.3-13, SHEET 1) FOR NOTES, REFERENCES, AND LEGEND.
 - THIS DRAWING SUPERSEDES EDISON'S SITE MASTER (VENDOR DRAWING 729E68AB, SHT. 5, SHT. 15) EDISON FILE NO. R1-152.

THIS FIGURE IS NOT UPDATED PERIODICALLY

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FIGURE 7.3-13, SHEET 6
 NUCLEAR BOILER SYSTEM FCD

TYPICAL FOR ALL RELIEF VALVES
 DIV I: B2-F00A,B,E,H,J,P & R
 DIV II: B2-F00C,D,F,G,I,M & N



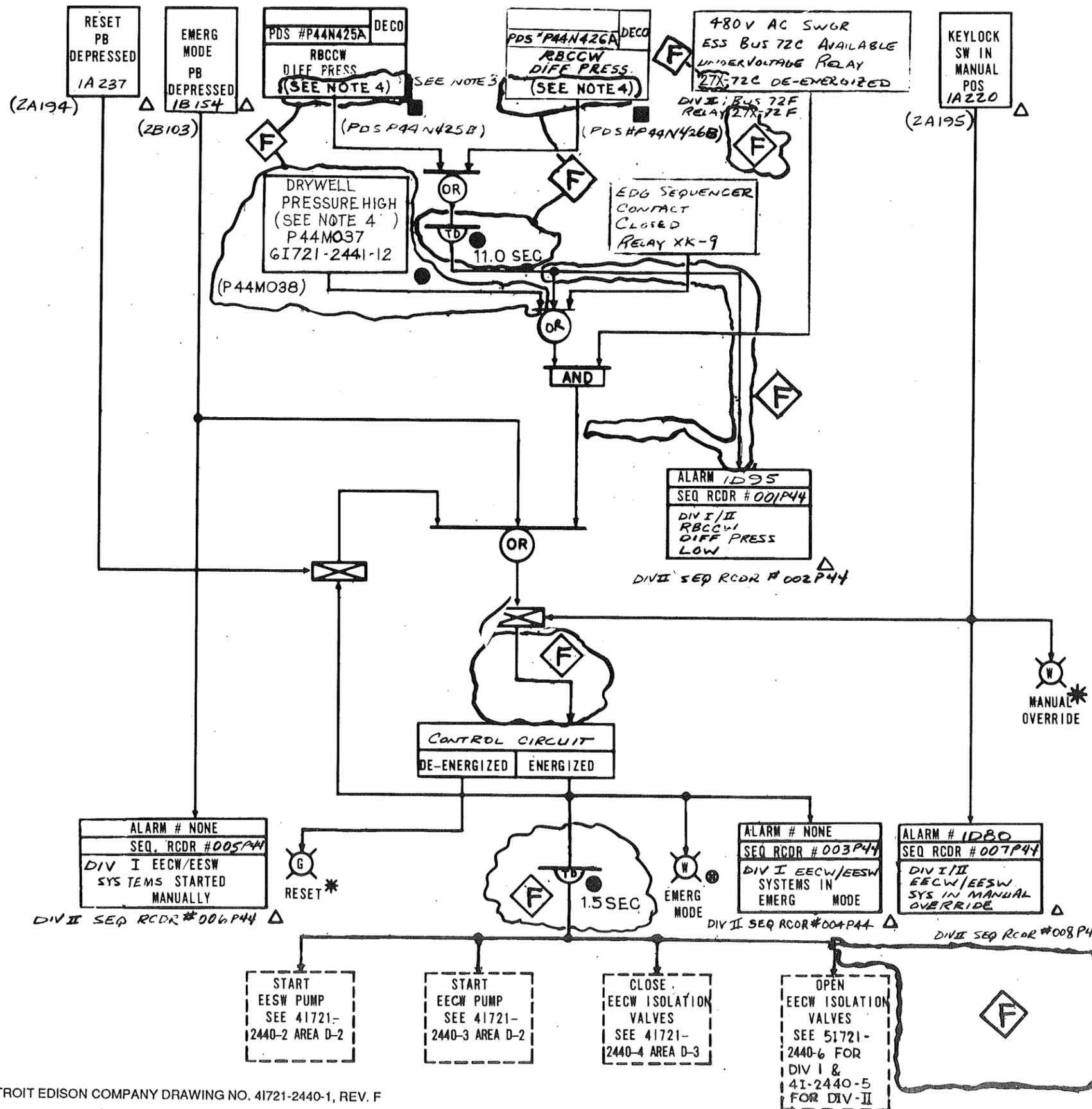
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FIGURE 7.3-14

TYPICAL ISOLATION CONTROL SYSTEM USING MOTOR-OPERATED VALVES

EECW AND EESW SYSTEMS DIVISION I
 DIVISION II IS SIMILAR (SEE NOTE 3)



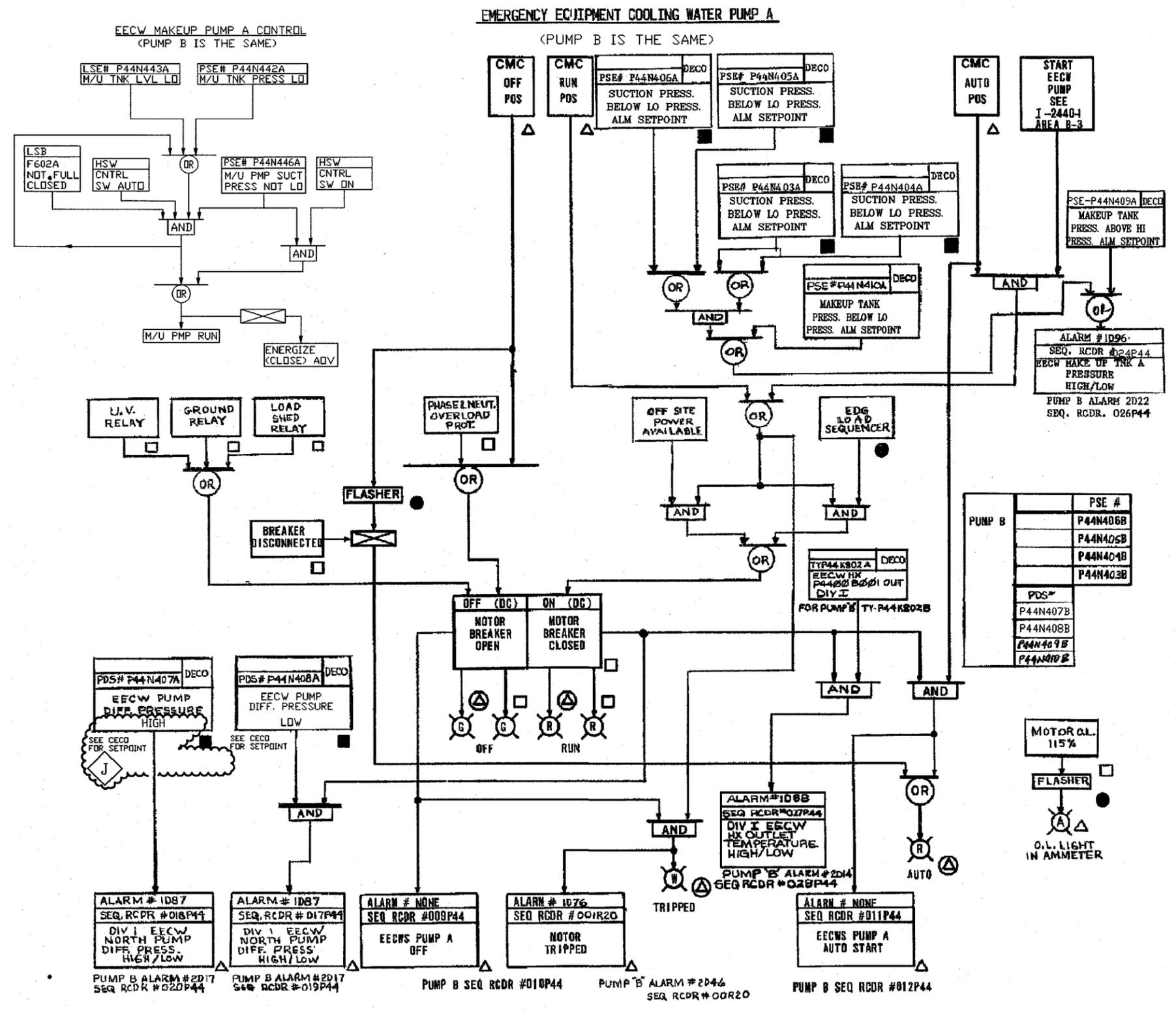
- LOCATION SYMBOLS**
- △ IN CONTROL ROOM
 - ⊗ IN PUSHBUTTON
 - IN RELAY ROOM
 - AT CONTROLLED EQUIPMENT
 - * IN SPLIT SCREEN INDICATOR

- NOTES**
1. EMERG PB IS COVERED TO PREVENT ACCIDENTAL INITIATION OF EECW AND EESW SYSTEMS.
 2. SEE SPECIFIC LOGIC DIAGRAMS FOR EQUIPMENT CONTROLLED IN EACH DIVISION.
 3. DIVISION I DEVICE IDENTIFICATION APPEARS INSIDE THE CONTROL FUNCTION BOX. DIVISION II DEVICE IDENTIFICATION APPEARS UNDER THE CONTROL FUNCTION BOX IN PARENTHESIS.
 4. SEE CECO FOR SPECIFIC SETPOINT VALUE.

- REFERENCE DWGS & DOCUMENTS**
- 6M721-5444 P&ID - EECW/RBCCW DIV I
 - 6M721-5357 " " " DIV II
 - 6M721-5358 " - RBCCW
 - 51721-2440-6 LOGIC DIAG - EECWS ISOLATION VALVES DIV I
 - 41721-2440-10 LOGIC DIAG - EECWS ISOLATION VALVES
 - 41721-2070-1 LOGIC SYMBOLS & LEGEND
 - 41721-2440-2 LOGIC DIAG - EESWS PUMPS A&B
 - 41721-2440-3 LOGIC DIAG - EECWS PUMPS A&B
 - 41721-2440-4 LOGIC DIAG - EECWS ISOLATION VALVES
 - 1721-2440-5,8 LOGIC DIAG - EECWS ISOLATION VALVES DIV II
 - 61721-2441 SCHM DIAG - EECW & EESW SYSTEMS -12, -13
 - 6M-2290 SH 2 STANDARD PANEL CONFIGURATIONS
 - DECO DESIGN INSTRUCTION #29 EECW SYSTEM
 - DECO MASTER INSTRUMENT LIST
 - 61721-2003 ARRANGEMENT OF EQUIP ON PNL -1-2 H11-P601
 - 61721-2005 ARRANGEMENT OF EQUIP ON PNL -1-2 H11-P602
 - 61721-2714 EDG LOADING SEQUENCE LISTS -35, 36

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FIGURE 7.3-15, SHEET 1
 EMERGENCY EQUIPMENT COOLING
 WATER LOGIC DIAGRAM



EQUIPMENT 480 VAC, 100 HP					
PUMP	MOTOR	FEDER	PNL ITEM NO.		
DIV I	DIV II	CODE NO.	BUS NO.	POS	CMC SWITCH
A		P4400C001A	72C	3D	1B152
B		P4400C001B	72F	3B	2B83

- LOCATION SYMBOLS**
- △ IN CONTROL ROOM
 - ⊖ IN CMC SWITCH
 - IN RELAY ROOM
 - AT CONTROLLED EQUIPMENT
 - AT SWITCHGEAR

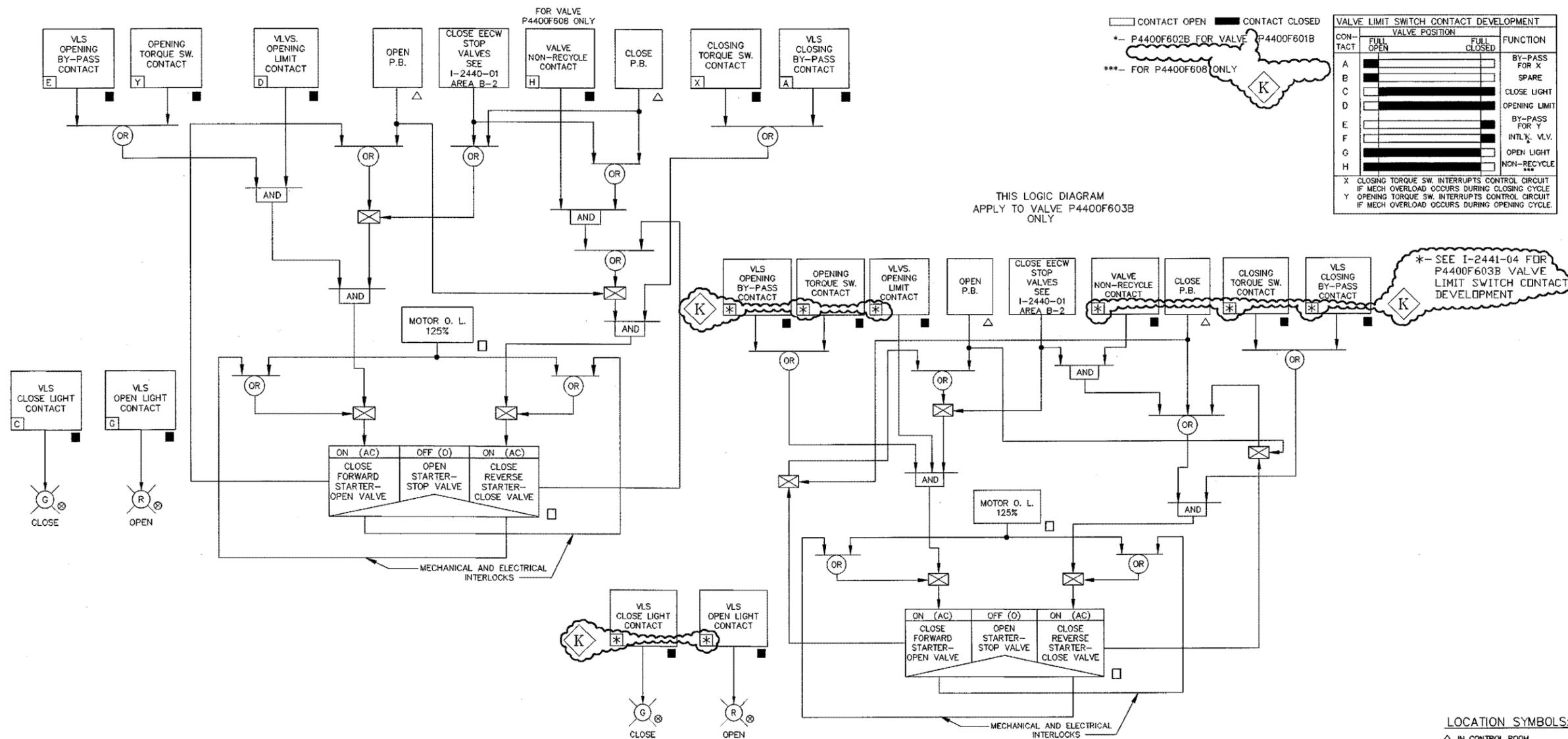
NOTE
 REFER TO DECO STANDARD DWG 6M-2290, SHT 2, MODE B, FOR EXPLANATION OF MOTOR CONTROL SWITCH AND ASSOCIATED FUNCTIONS

- REFERENCE DWGS & DOCUMENTS**
- I-2070-01 LOGIC SYMBOLS & LEGEND
 - I-2441-01, 02 SCHM DIAG-EECW PUMPS A & B
 - SD-2510-01 ONE LINE 480 VAC SWITCHGEAR
 - M-5444, 5357 PIPING DIAG. RBCCW & EECW SYSTEMS
 - M-2001 LEGEND OF SYMBOLS & INSTR IDENT FOR PLANT SYST DIAGRAMS
 - I-2003-01, 02, 03 ARRANGEMENT OF EQUIP ON PNL H11P601
 - I-2005-01, 02 ARRANGEMENT OF EQUIP ON PNL H11P602
 - I-2440-01 LOGIC-EECW & EESW SYST A/M CTRL
 - CECD CENTRAL COMPONENT DATABASE
 - 6M-2290, SHT 2, STANDARD PANEL CONFIGURATIONS

PUMP B	PSE #
	P44N406B
	P44N405B
	P44N404B
	P44N403B
	PDS*
	P44N407B
	P44N408B
	P44N409B
	P44N410B

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FIGURE 7.3-15, SHEET 2
 EMERGENCY EQUIPMENT COOLING WATER LOGIC DIAGRAM



**	VALVE			FEEDER	PANEL ITEM NO.	LIMIT SW. CONTACTS		SCHEMATIC DIAGRAM
	IDENT.	DIV.	PIS NO.			MCC NO.	POS.	
12	RBCCW TO REAC BLDG EQUIP SUMP HX A INLET	I	P4400F605A	72C-3A	7B	1A227	1A228	SPARE SPARE I-2441-08
8	EECW RETURN TO RBCCW	II	P4400F601B	72F-4A	5B-R	2A217	2A218	SPARE P4400F602B I-2441-04
9	RBCCW TO EECW	II	P4400F603B	72F-4A	5C-R	2A215	2A216	USED P4400F602B I-2441-04
5	RBCCW TO REAC BLDG EQUIP SUMP HX B INLET	II	P4400F605B	72F-4A	4A-R	2A221	2A222	SPARE SPARE I-2441-07
14	RBCCW TO CTRL ROD DRIVE PUMPS INLET	II	P4400F604	72F-4A	5D-R	2A227	2A228	SPARE SPARE I-2441-07
4	RBCCW TO DRYWELL EQUIP SUMP HX INLET	II	P4400F608	72F-4A	5B	2A229	2A230	USED SPARE I-2441-05
15	RBCCW TO BATT RM SPACE COOLER	I	P4400F613	72B-2A	4A	8A57	8A58	SPARE SPARE I-2441-18
16	RBCCW TO DRYWELL PENET COOLER	I	P4400F614	72C-3A	1A	1A256	1A257	SPARE SPARE I-2441-18

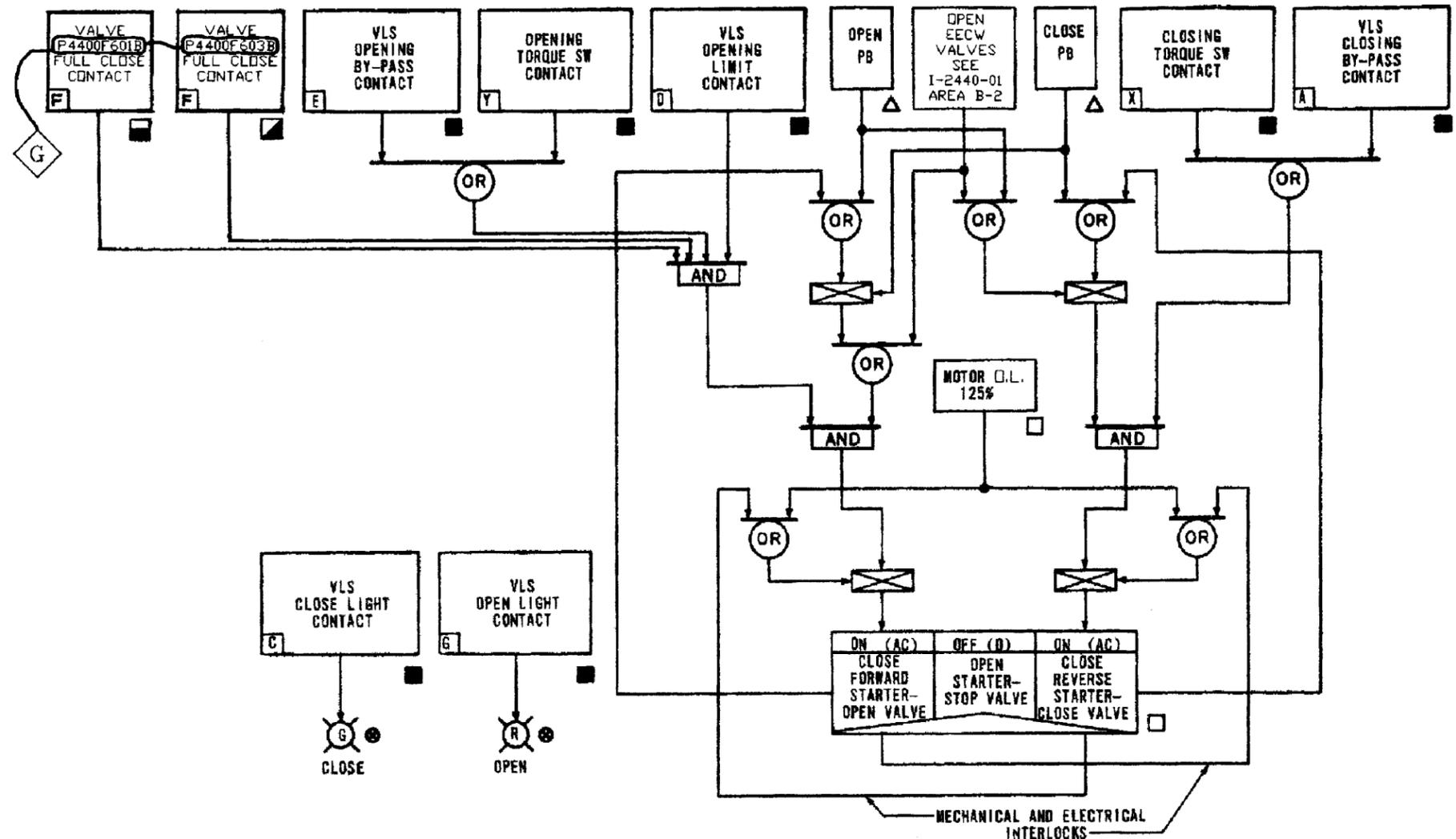
** - VALVE NUMBER FROM TABLE ON SYSTEM DIAGRAM M-5444.

REFERENCE DWG'S & DOCUMENTS

- I-2070-01 LOGIC SYMBOLS AND LEGEND
- I-2440-01 LOGIC DIAG-EECW & EESW SYSTEMS
- I-2441-04 SCHM DIAG - EECWS VALVES - AUTO CLOSE
- SD-2512-18 480 VAC MCC 72C-3A FRONTAL ELEV.
- SD-2512-19B 480 VAC MCC 72F-4A FRONTAL ELEV.
- M-2001 LEGEND OF SYMBOLS & INSTR IDENT FOR PLANT SYSTEM DIAGRAMS
- I-2003-01-02-03 ARRANGEMENT OF EQUIP. ON PANEL H11P601
- I-2005-01-02 ARRANGEMENT OF EQUIP. ON PANEL H11P602
- 6M2290, SHT. 2 STANDARD PANEL CONFIGURATIONS
- SD-2512-20 480 VAC MCC 72B-2A FRONTAL ELEV.
- M-5357 PIPING DIAGRAM - EECW DIVISION II
- M-5444 PIPING DIAGRAM - EECW DIVISION I
- I-2440-05 LOGIC DIAGRAM EECW DIV. 2, AUTO OPEN
- I-2440-06 LOGIC DIAGRAM EECW DIV. 1, AUTO OPEN

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FIGURE 7.3-15, SHEET 3
 EMERGENCY EQUIPMENT COOLING
 WATER LOGIC DIAGRAM



VALVE LIMIT SWITCH CONTACT DEVELOPMENT			
CONTACT	VALVE POSITION		FUNCTION
	FULL OPEN	FULL CLOSED	
A	█	█	BY-PASS FOR X
B	█	█	SPARE
C	█	█	CLOSE LIGHT
D	█	█	OPENING LIMIT
E	█	█	BY-PASS FOR Y
F	█	█	SPARE
G	█	█	OPEN LIGHT
R	█	█	INTERLOCK W/ MAKEUP PUMP

X CLOSING TORQUE SWITCH INTERRUPTS CONTROL CIRCUIT IF MECH OVERLOAD OCCURS DURING CLOSING CYCLE
Y OPENING TORQUE SWITCH INTERRUPTS CONTROL CIRCUIT IF MECH OVERLOAD OCCURS DURING OPENING CYCLE.

CONTACT OPEN
 CONTACT CLOSED

LOCATION SYMBOLS

- ▲ IN CONTROL ROOM
- IN PUSHBUTTON
- AT CONTROLLED EQUIPMENT
- AT SWITCHGEAR
- ▣ AT VALVE P4400F601B LIMIT SWITCH
- ▤ AT VALVE P4400F603B LIMIT SWITCH

NOTE

REFER TO DECO STANDARD DWG 6M-2290, SHT 2, DETAIL 2 FOR EXPLANATION OF CONTROL PUSHBUTTONS.

**	VALVE				FEEDER		PANEL ITEM NO	
	IDENT	DIV	CODE NO.	TAG NO.	WCC	POS	OPEN PB	CLOSE PB
7	EECW MU TK B OUTLET	II	P4400F602B	VB-2374	72F-4A	5A	2A219	2A220

** VALVE NO. FROM TABLE ON SYST DTAG - M-5357

REFERENCE DWGS & DOCUMENTS

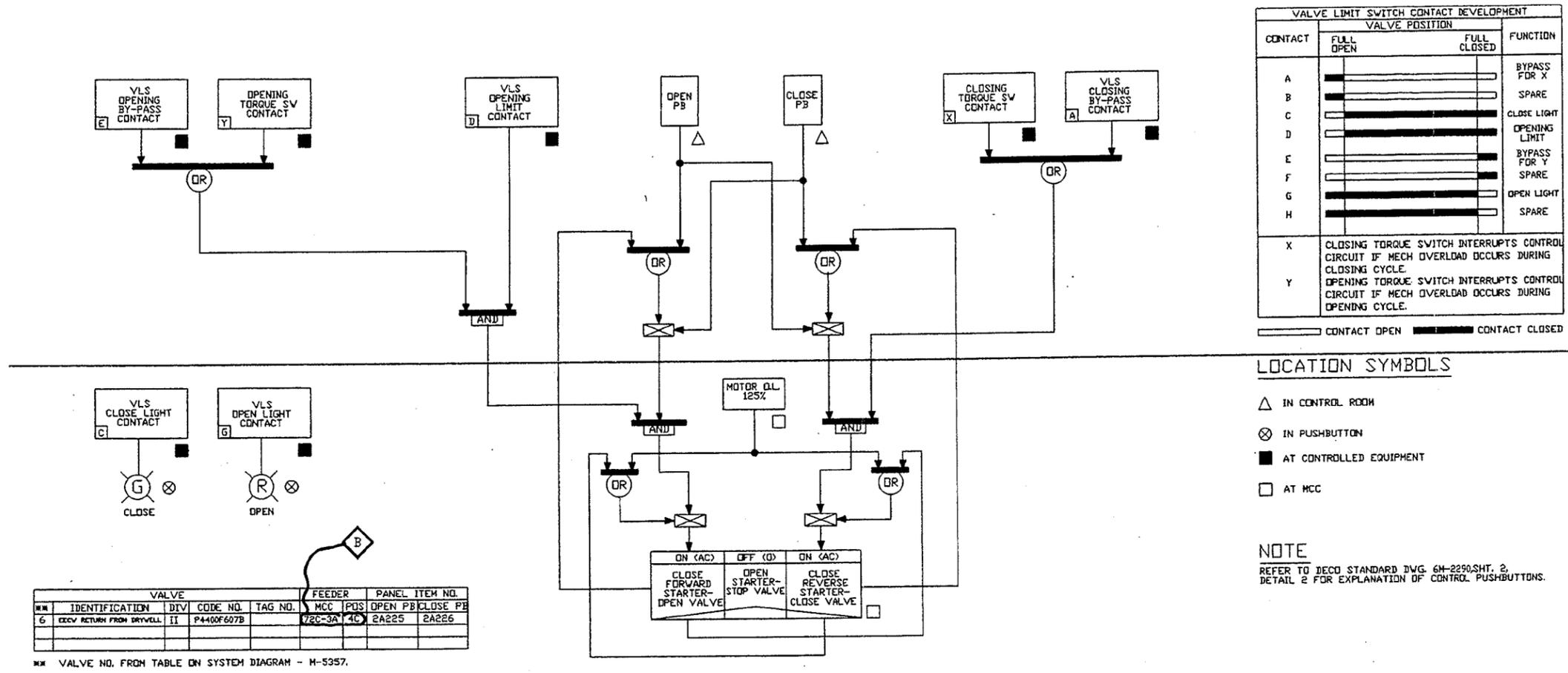
- I-2070-01 LOGIC SYMBOLS AND LEGEND
- I-2440-01 LOGIC DIAG. - EECW & EESW SYSTEMS
- I-2441-06 SCHM DIAG. - EECWS VALVES - AUTO OPEN
- SD-2512-19A 480VAC MCC 72F-4A FRONTAL ELEV. (FRONT)
- SD-2512-19B 480VAC MCC 72F-4A FRONTAL ELEV. (REAR)
- M-5357 PIPING DIAG. - EECW SYSTEM (DIV. II)
- M-5358 PIPING DIAG. - RBCCW SYSTEM
- M-5444 PIPING DIAG. - EECW SYSTEM (DIV. I)
- M-2001 LEGEND OF SYMBLS & INSTRUMENT IDENTIFICATION FOR PLANT SYSTEM DIAGRAMS
- I-2005-01, 02 ARRANGEMENT OF EQUIPMENT ON PANEL H11P602
- I-2440-06 EECW SYSTEM VALVES - AUTO OPEN DIVISION I LOGIC DIAGRAMS
- I-2441-26 SCHEM DIAG. EECW MAKEUP PUMP P4400C002B



DECO DESIGN INSTRUCTION #29 EECW SYSTEM
6M-2290, SHT. 2, STANDARD PANEL CONFIGURATIONS

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FIGURE 7.3-15, SHEET 4
**EMERGENCY EQUIPMENT COOLING
WATER LOGIC DIAGRAM**



VALVE LIMIT SWITCH CONTACT DEVELOPMENT			
CONTACT	VALVE POSITION		FUNCTION
	FULL OPEN	FULL CLOSED	
A	█	█	BYPASS FOR X
B	█	█	SPARE
C	█	█	CLOSE LIGHT
D	█	█	OPENING LIMIT
E	█	█	BYPASS FOR Y
F	█	█	SPARE
G	█	█	OPEN LIGHT
H	█	█	SPARE
X	CLOSING TORQUE SWITCH INTERRUPTS CONTROL CIRCUIT IF MECH OVERLOAD OCCURS DURING CLOSING CYCLE.		
Y	OPENING TORQUE SWITCH INTERRUPTS CONTROL CIRCUIT IF MECH OVERLOAD OCCURS DURING OPENING CYCLE.		

— CONTACT OPEN █ CONTACT CLOSED

LOCATION SYMBOLS

- △ IN CONTROL ROOM
- ⊗ IN PUSHBUTTON
- AT CONTROLLED EQUIPMENT
- AT MCC

NOTE
REFER TO DECO STANDARD DWG. 6M-2290, SHT. 2, DETAIL 2 FOR EXPLANATION OF CONTROL PUSHBUTTONS.

VALVE		FEEDER		PANEL ITEM NO.				
NUM	IDENTIFICATION	DIV	CODE NO.	TAG NO.	MCC	POS	OPEN PB	CLOSE PB
6	EECV RETURN FROM DRYWELL	II	P4400F607B		72C-3A	4C	2A225	2A226

NUM VALVE NO. FROM TABLE ON SYSTEM DIAGRAM - M-5357.

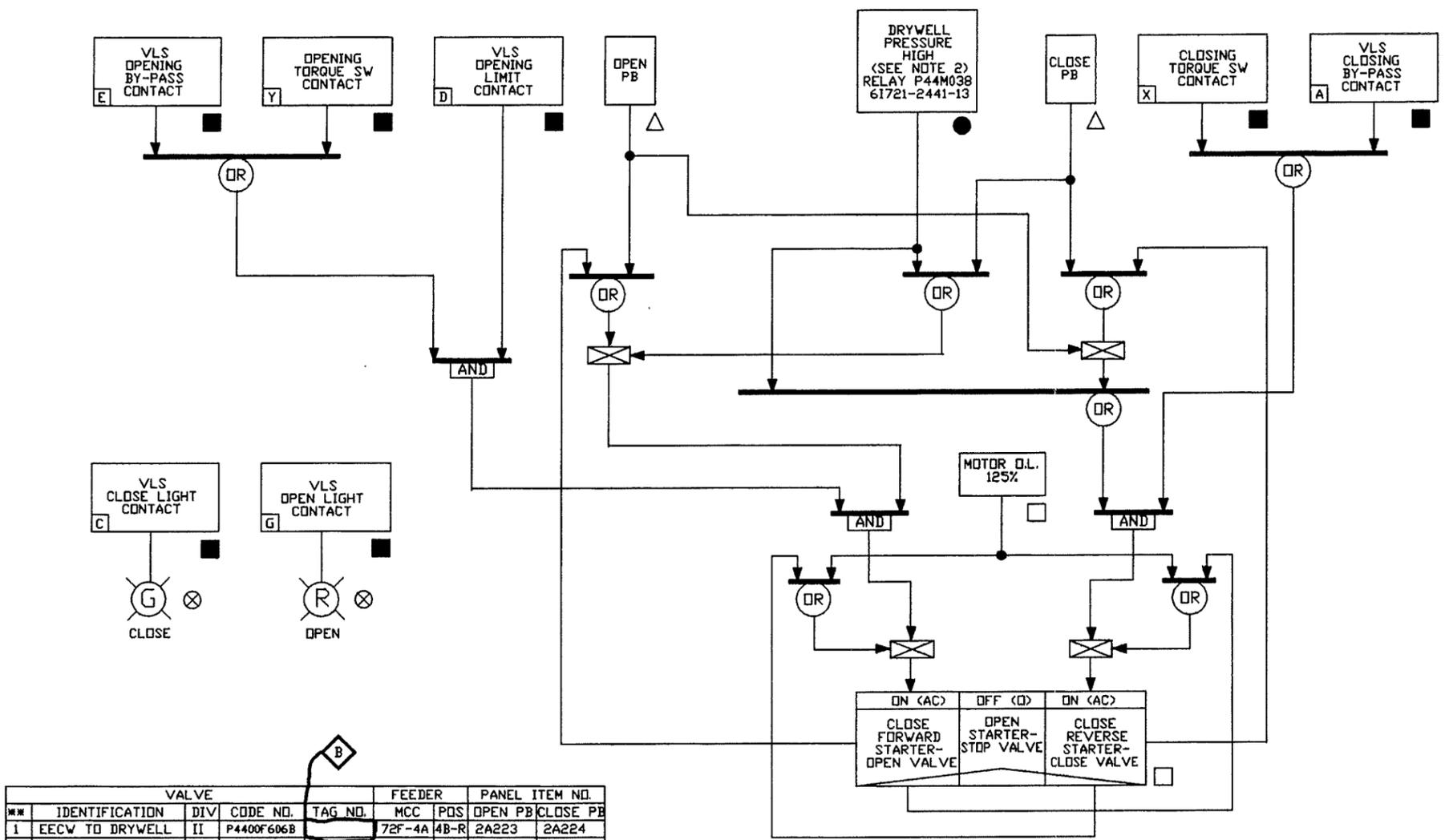
REFERENCE DWGS. & DOCUMENTS

- I-2070-1 LOGIC SYMBOLS AND LEGEND
- I-2440-1 LOGIC DIAG-EECV & EESV SYSTEMS
- I-2441-8 SCHM DIAG-EECVS VALVES - AUTO OPEN
- SD-2512-18 480 VAC MCC (72C-3A) FRONTAL ELEV (FRT)
- M-5357, PIPING DIAGRAM-RBCCV & EECV SYSTEMS
- M-2001, LEGEND OF SYMBOLS & INSTR IDENTIFICATION FOR PLANT SYSTEM DIAGRAMS
- I-2005-1 -2, ARRANGEMENT OF EQUIP ON PANEL H11-P602
- DECO DESIGN INSTRUCTION #29, EECV SYSTEM
- 6M-2290, SHT. 2, STANDARD PANEL CONFIGURATIONS
- I-2440-6, EMERG. EQPT. COOLING WATER SYSTEM VALVES - AUTO OPEN, DIV. 2 LOGIC DIAGRAMS

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FIGURE 7.3-15, SHEET 5
EMERGENCY EQUIPMENT COOLING
WATER LOGIC DIAGRAM



VALVE		FEEDER		PANEL ITEM NO.				
**	IDENTIFICATION	DIV	CODE NO.	TAG NO.	MCC	POS	OPEN PB	CLOSE PB
1	EECW TO DRYWELL	II	P4400F606B		72F-4A	4B-R	2A223	2A224

** VALVE NO. FROM TABLE ON SYSTEM DIAGRAM - 6M721-5357.

VALVE LIMIT SWITCH CONTACT DEVELOPMENT			
CONTACT	VALVE POSITION		FUNCTION
	FULL OPEN	FULL CLOSED	
A	█	█	BYPASS FOR X
B	█	█	SPARE
C	█	█	CLOSE LIGHT
D	█	█	OPENING LIMIT
E	█	█	BYPASS FOR Y
F	█	█	SPARE
G	█	█	OPEN LIGHT
H	█	█	SPARE
X	CLOSING TORQUE SWITCH INTERRUPTS CONTROL CIRCUIT IF MECH OVERLOAD OCCURS DURING CLOSING CYCLE.		
Y	OPENING TORQUE SWITCH INTERRUPTS CONTROL CIRCUIT IF MECH OVERLOAD OCCURS DURING OPENING CYCLE.		

CONTACT OPEN (white bar) CONTACT CLOSED (black bar)

LOCATION SYMBOLS

- △ IN CONTROL ROOM
- ⊗ IN PUSHBUTTON
- AT CONTROLLED EQUIPMENT
- AT MCC
- IN RELAY ROOM

NOTE

1. REFER TO DECD STANDARD DWG. 6M721-2290, SHT. 2, DETAIL 2 FOR EXPLANATION OF CONTROL PUSHBUTTONS.
2. SEE CECD FOR SPECIFIC SETPOINT VALUE

REFERENCE DWGS. & DOCUMENTS

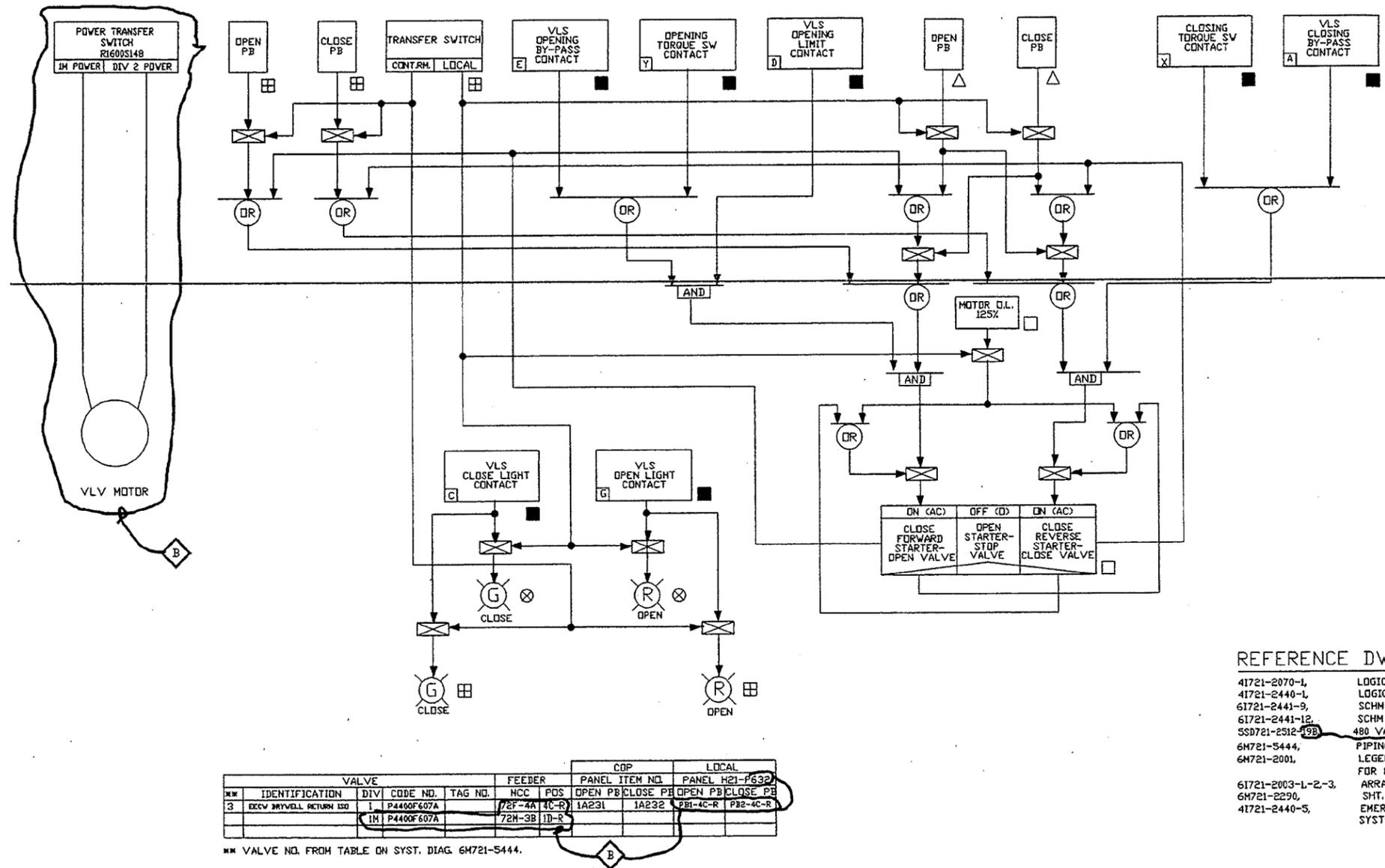
- 41721-2070-1, LOGIC SYMBOLS AND LEGEND
- 41721-2440-1, LOGIC DIAG-EECW & EESW SYSTEMS
- 61721-2215-2, ELEMENTARY DIAG-CORE SPRAY SYSTEM
- 61721-2441-6, SCHM DIAG-EECW VALVES-AUTO OPEN
- 61721-2441-13, SCHM DIAG-EECW & EESW 'B' SYSTEM
- 5SD721-2512-19A, 480 VAC MCC 72F-4A FRONTAL ELEV (FRT)
- 5SD721-2512-19B, 480 VAC MCC 72F-4A
- 6M721-5357, PIPING DIAGRAM-RBCCW & EECW SYSTEMS
- 6M721-2001, LEGEND OF SYMBOLS & INSTR IDENTIFICATION FOR PLANT SYSTEM DIAGRAMS FRONTAL ELEV (REAR)
- 61721-2005-1, -2, ARRANGEMENT OF EQUIP ON PANEL H11-P602
- DECD DESIGN INSTRUCTION #29, EECW SYSTEM
- 6M721-2290, SHT. 2, STANDARD PANEL CONFIGURATIONS
- 51721-2440-6, EMERG. EQPT. COOLING WATER SYSTEM VALVES - AUTO OPEN, DIV. I LOGIC DIAGRAMS

EECW SYSTEM VALVE - AUTO CLOSE DIVISION II

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FIGURE 7.3-15, SHEET 6

EMERGENCY EQUIPMENT COOLING
 WATER LOGIC DIAGRAM



VALVE LIMIT SWITCH CONTACT DEVELOPMENT			
CONTACT	VALVE POSITION		FUNCTION
	FULL OPEN	FULL CLOSED	
A	—	—	BYPASS FOR X
B	—	—	SPARE
C	—	—	CLOSE LIGHT
D	—	—	OPENING LIMIT
E	—	—	BYPASS FOR Y
F	—	—	SPARE
G	—	—	OPEN LIGHT
H	—	—	SPARE
X	CLOSING TORQUE SWITCH INTERRUPTS CONTROL CIRCUIT IF MECH OVERLOAD OCCURS DURING CLOSING CYCLE.		
Y	OPENING TORQUE SWITCH INTERRUPTS CONTROL CIRCUIT IF MECH OVERLOAD OCCURS DURING OPENING CYCLE.		

LOCATION SYMBOLS

- △ IN CONTROL ROOM
- ⊗ IN PUSHBUTTON
- AT CONTROLLED EQUIPMENT
- AT MCC
- ⊞ IN DEDICATED SHUTDOWN PANEL H21-P625

NOTE
REFER TO DECC STANDARD DWG. 6H721-2290, SHT. 2, DETAIL 2 FOR EXPLANATION OF CONTROL PUSHBUTTONS.

REFERENCE DWGS. & DOCUMENTS

- 41721-2070-1, LOGIC SYMBOLS AND LEGEND
- 41721-2440-1, LOGIC DIAG-ECCV & EESW SYSTEMS
- 61721-2441-9, SCHM DIAG-ECCV VALVES-
- 61721-2441-12, SCHM DIAG-ECCV & EESW 'A' SYSTEM
- 550721-2512-19B, 480 VAC MCC (72F-48) FRONTAL ELEV (REAR)
- 6H721-5444, PIPING DIAGRAM-RBCCW & ECCV SYSTEMS
- 6H721-2001, LEGEND OF SYMBOLS & INSTR IDENTIFICATION FOR PLANT SYSTEM DIAGRAMS
- 61721-2003-1-2-3, ARRANGEMENT OF EQUIP ON PANEL H11-P601
- 6H721-2290, SHT. 2, STANDARD PANEL CONFIGURATIONS
- 41721-2440-5, EMERG EQPT. COOLING WATER SYSTEM VALVE- AUTO OPEN DIV. II LOGIC DIAGRAM

VALVE NO.	VALVE			FEEDER		COP		LOCAL	
	IDENTIFICATION	DIV	CODE NO.	TAG NO.	NCC	POS	PANEL ITEM NO.	PANEL H21-P632	PANEL H21-P625
3	ECCV BRIVELL RETURN ISO	1	P4400F607A	72F-4A 4C-R	1A231	1A232	PBI-4C-R	P82-4C-R	
			1M P4400F607A	72H-3B 1D-R					

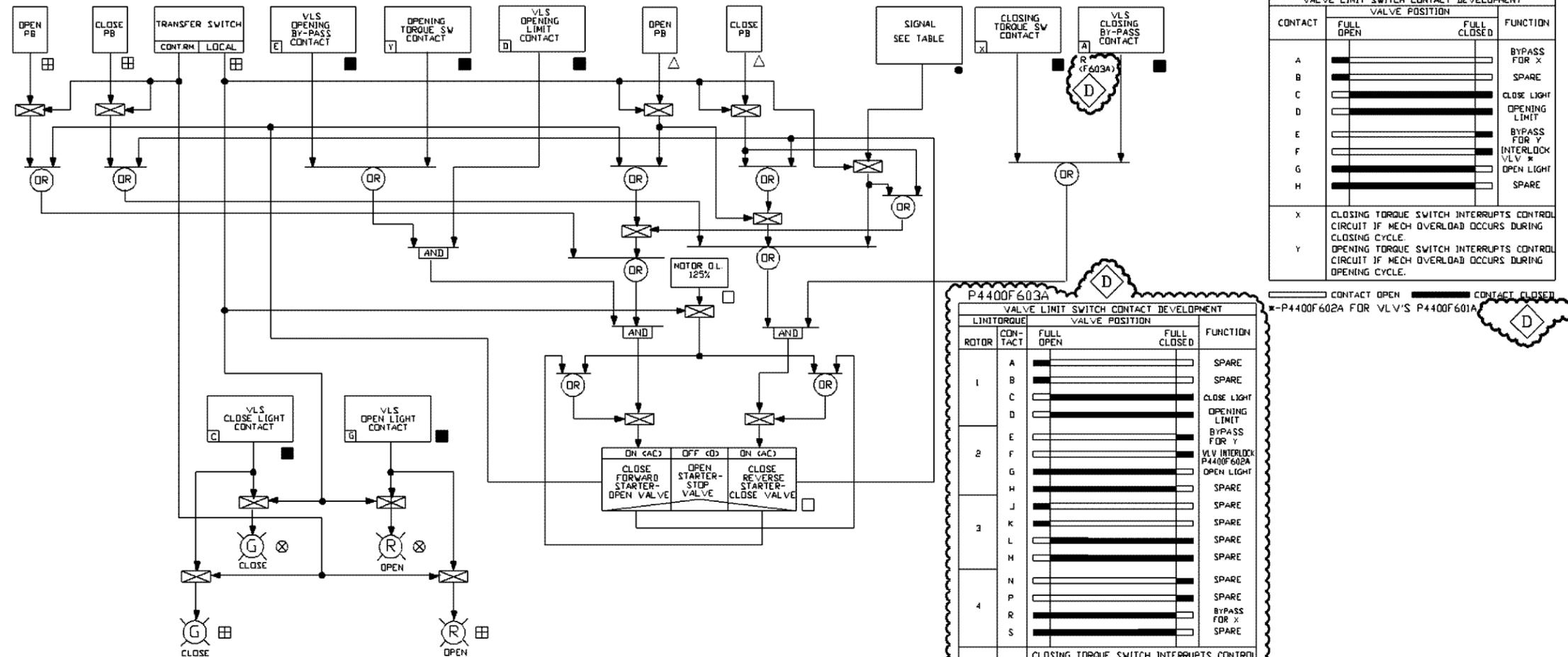
*** VALVE NO. FROM TABLE ON SYST. DIAG 6H721-5444.

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FIGURE 7.3-15, SHEET 7

EMERGENCY EQUIPMENT COOLING
WATER LOGIC DIAGRAM



VALVE LIMIT SWITCH CONTACT DEVELOPMENT			
CONTACT	VALVE POSITION		FUNCTION
	FULL OPEN	FULL CLOSED	
A	—	—	BYPASS FOR X
B	—	—	SPARE
C	—	—	CLOSE LIGHT
D	—	—	OPENING LIMIT
E	—	—	BYPASS FOR Y
F	—	—	INTERLOCK VLV *
G	—	—	OPEN LIGHT
H	—	—	SPARE
X	CLOSING TORQUE SWITCH INTERRUPTS CONTROL CIRCUIT IF MECH OVERLOAD OCCURS DURING CLOSING CYCLE.		
Y	OPENING TORQUE SWITCH INTERRUPTS CONTROL CIRCUIT IF MECH OVERLOAD OCCURS DURING OPENING CYCLE.		

VALVE LIMIT SWITCH CONTACT DEVELOPMENT				
LIMITORQUE ROTOR	CON-TACT	VALVE POSITION		FUNCTION
		FULL OPEN	FULL CLOSED	
1	A	—	—	SPARE
	B	—	—	SPARE
	C	—	—	CLOSE LIGHT
	D	—	—	OPENING LIMIT
2	E	—	—	BYPASS FOR Y
	F	—	—	VLV INTERLOCK P4400F602A
	G	—	—	OPEN LIGHT
	H	—	—	SPARE
3	J	—	—	SPARE
	K	—	—	SPARE
	L	—	—	SPARE
	M	—	—	SPARE
4	N	—	—	SPARE
	P	—	—	SPARE
	R	—	—	BYPASS FOR X
	S	—	—	SPARE
TSC	X	CLOSING TORQUE SWITCH INTERRUPTS CONTROL CIRCUIT IF MECH OVERLOAD OCCURS DURING CLOSING CYCLE.		
TSD	Y	OPENING TORQUE SWITCH INTERRUPTS CONTROL CIRCUIT IF MECH OVERLOAD OCCURS DURING OPENING CYCLE.		

VALVE				CDP		LOCAL		LIMIT SW CONTACTS		SCHEMATIC DIAGRAM	SIGNAL DESCRIPTION	
** IDENTIFICATION	DIV	CODE NO.	TAG NO.	MCC	POS	OPEN PB	CLOSE PB	OPEN PB	CLOSE PB			'H' CLOSING LIMIT
11 EECW DRYWELL SUPPLY ISD	I	P4400F606A		72C-3A	4D	1A229	1A230	PBI-4D	PB2-4D	SPARE	P4400F602A	1-2441-09 DRYWELL PRESSURE HIGH (RELAY P44M037)
10 RBCCW TO EECW	I	P4400F601A	V8-2323	72C-3A	7C	1A223	1A224	PBI-7C	PB2-7C	SPARE	P4400F602A	1-2441-03 SYSTEM AUTO CONTROL (RELAY P44M808A)
2 RBCCW TO EECW	I	P4400F603A	V8-2324	72C-3A	7D	1A221	1A222	PBI-7D	PB2-7D			1-2441-03 SYSTEM AUTO CONTROL (RELAY P44M808A)

** VALVE NO. FROM TABLE ON SYST. DIAG. M-5444.

- LOCATION SYMBOLS**
- △ IN CONTROL ROOM
 - ⊗ IN PUSHBUTTON
 - AT CONTROLLED EQUIPMENT
 - AT MCC
 - ▣ IN DEDICATED SHUTDOWN PANEL H21P625
 - IN RELAY ROOM

NOTE

- REFER TO RECD STANDARD DWG. H-2290, SHT. 2, DETAIL 2 FOR EXPLANATION OF CONTROL PUSHBUTTONS.
- SEE CECD FOR SPECIFIC SETPOINT VALUE.

REFERENCE DWGS. & DOCUMENTS

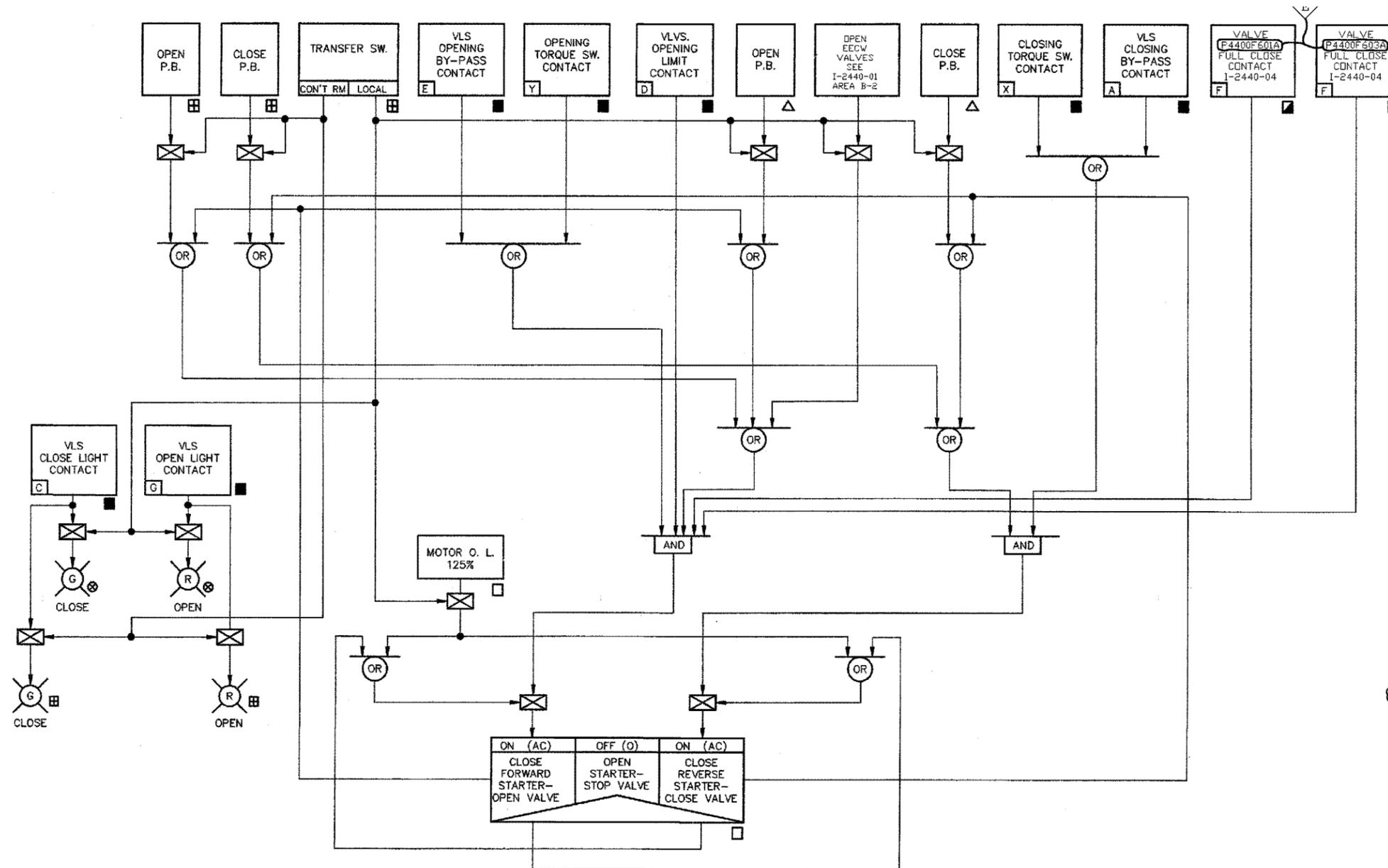
- 1-2070-01, LOGIC SYMBOLS AND LEGEND
- 1-2440-01, LOGIC DIAG-EECW & EESW SYSTEMS
- 1-2441-09, SCHM DIAG-EECW'S VALVES
- 1-2441-12, SCHM DIAG-EECW & EESW 'A' SYSTEM
- SD-2512-18, 480 VAC MCC 72C-3A FRONTAL ELEV.
- N-5444, PIPING DIAGRAM-RBCCW & EECW SYSTEMS
- N-2001, LEGEND OF SYMBOLS & INSTR IDENTIFICATION FOR PLANT SYSTEM DIAGRAMS
- I-2003-01,-02,-03, ARRANGEMENT OF EQUIP ON PANEL H1JP601
- N-2290, SHT. 2, STANDARD PANEL CONFIGURATIONS
- I-2440-05, EMERG. EQPT. COOLING WATER SYSTEM VALVES AUTO OPEN DIV. 11 LOGIC
- I-2441-03, SCH. DIAG. EECW RETURN TO RBCCW TO EECW STOP VALVE

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FIGURE 7.3-15, SHEET 8

EMERGENCY EQUIPMENT COOLING
 WATER LOGIC DIAGRAM



VALVE LIMIT SWITCH CONTACT DEVELOPMENT			
CON-TACT	VALVE POSITION		FUNCTION
	FULL OPEN	FULL CLOSED	
A	█	█	BY-PASS FOR X
B	█	█	SPARE
C	█	█	CLOSE LIGHT
D	█	█	OPENING LIMIT
E	█	█	BY-PASS FOR Y
F	█	█	SPARE
G	█	█	OPEN LIGHT
H	█	█	INTERLOCK VALVE MAKEUP PUMP

X CLOSING TORQUE SW. INTERRUPTS CONTROL CIRCUIT IF MECH OVERLOAD OCCURS DURING CLOSING CYCLE.
 Y OPENING TORQUE SW. INTERRUPTS CONTROL CIRCUIT IF MECH OVERLOAD OCCURS DURING OPENING CYCLE.

□ CONTACT OPEN █ CONTACT CLOSED

- LOCATION SYMBOLS:**
- △ IN CONTROL ROOM
 - ⊗ IN PUSHBUTTON
 - AT CONTROLLED EQUIPMENT
 - AT M.C.C.
 - ⊞ IN DEDICATED SHUTDOWN PANEL H21P625
 - ▣ AT VALVE P4400F601A LIMIT SWITCH
 - ▤ AT VALVE P4400F603A LIMIT SWITCH

REFERENCE DWG'S & DOCUMENTS

- I-2070-01 LOGIS SYMBOLS AND LEGEND
- I-2440-01 LOGIC DIAG. - EECW & EESW SYSTEMS
- I-2441-21 SCHM DIAG. EECWS VALVES - AUTO OPEN
- M-5358 PIPING DIAG. - RBCCW SYSTEM
- SD-2512-18 480VAC MCC 72C-3A FRONTAL ELEV.
- M-5444 PIPING DIAG. - EECW SYSTEM DIV. I
- M-2001 LEGEND OF SYMBOLS & INSTRUMENT IDENTIFICATION FOR PLANT SYSTEM DIAGRAMS
- I-2003-01, 02, 03 ARRANGEMENT OF EQUIP. ON PANEL H11P601
- 6M-2229, SHT. 2 STANDARD PANEL CONFIGURATION
- I-2440-05 EMERGENCY EQUIPMENT COOLING WATER SYSTEM VALVE - AUTO OPEN DIV. II LOGIC DIAGRAM
- I-2441-25 SCHEMATIC DIAGRAM - EECW MAKEUP PUMP P4400C002A

NOTES:

1. REFER TO EDISON STANDARD DWG 6M-2290, SH. 2 DETAIL 2 FOR EXPLANATION OF CONTROL PUSHBUTTONS.

VALVE				FEEDER			COP		LOCAL	
** IDENT	DIV	CODE NO.	TAG NO.	MCC	POS	OPEN PB	CLOSE PB	OPEN PB	CLOSE PB	
13 EECW M. U. TANK A OUTLET	I	P4400F602A	V8-2407	72C-3A	7A	1A225	1A226	PB1-7A	PB2-7A	

** VALVE NO. FROM TABLE ON SYST. DIAG. M-5444

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FIGURE 7.3-15, SHEET 9

**EMERGENCY EQUIPMENT COOLING
 WATER LOGIC DIAGRAM**

7.4 SAFE-SHUTDOWN SYSTEMS

7.4.1 Description

7.4.1.1 Reactor Core Isolation Cooling System Instrumentation and Control

7.4.1.1.1 System Identification

7.4.1.1.1.1 Function

The reactor core isolation cooling (RCIC) system provides core cooling during reactor shutdown by pumping makeup water into the reactor pressure vessel (RPV) in case of a loss of flow from the main feedwater system. It is activated in time to preclude conditions that lead to inadequate core cooling.

7.4.1.1.1.2 Classification

Electrical modules for the RCIC system are classified as Safety Class 2 and Category I.

7.4.1.1.2 Power Sources

The RCIC pump is turbine driven and the RCIC trip system is powered by the Division I 260/130-V dc battery.

7.4.1.1.3 Equipment Design

When actuated, the RCIC system pumps water from either the condensate storage tank or the suppression chamber to the RPV via the feedwater lines. The RCIC system includes one turbine-driven pump, one barometric condenser dc vacuum pump, one vacuum dc condensate pump, automatic valves, control devices for this equipment, sensors, and logic circuitry. The arrangement of equipment and control devices is shown in Figure 5.5-7.

Pressure and level transmitters used in the RCIC system are located on racks in the reactor building. The only operating component of the RCIC system that is located inside the primary containment is one of the two RCIC system turbine steam supply isolation valves.

The rest of the RCIC system instrumentation and control components are located outside the primary containment. Cables connect the sensors to control circuitry in the main control room. The system is designed to allow a full flow functional test of the system during normal reactor power operation. The system will automatically return to normal system operation if called upon to do so during the test.

7.4.1.1.3.1 Initiating Circuits

Reactor pressure vessel low water level is monitored by four level transmitters which sense the difference between the pressure of a constant reference leg of water and the pressure resulting from the actual height of water in the vessel. Two pipelines, attached to taps above and below the water level on the RPV, are required for each of the two reference legs used

with the RCIC. The lines are physically separated from each other and tap off the RPV at widely separated points. Two pairs of differential-pressure sensing lines from the two reference legs terminate outside the primary containment and inside the reactor building.

A backfill system is installed on each level instrument reference leg. The system provides a metered flow of water from the control rod drive system to each leg. The flow is low enough to not affect the performance of the instrumentation. The backfill is designed to prevent the accumulation of dissolved noncondensable gases in the reference legs.

The RCIC system is initiated only by low water level. The RCIC initiation circuit is arranged in a "one-out-of-two taken twice" logic.

The RCIC system is automatically initiated after the receipt of an RPV low water level signal, and produces the design flow rate within 50 sec. The controls then function to provide a flow of makeup water to the RPV until the amount of water delivered to the RPV is adequate to restore vessel level. At this time, the RCIC system automatically shuts down by closing the turbine steam supply valve and the steam warmup bypass valve, if it is still open. The system will automatically reinitiate if the water level returns to the low-level trip point. The controls are arranged to allow remote manual startup, operation, and shutdown.

The RCIC turbine is functionally controlled as shown in Figure 7.4-1. Minimizing initial peak speed of the turbine is accomplished by use of a warmup bypass valve. A speed governor limits the turbine speed to its maximum operating level. A control governor receives a RCIC system flow signal and adjusts the turbine steam control valve so that design pump discharge flow rate is obtained. Manual control of the governor is possible in the test mode; however, the governor automatically returns to automatic control on receipt of an RCIC system initiation signal. The flow signal used for automatic control of the turbine is derived from a differential pressure measurement across a flow element in the RCIC pump discharge line. The governor controls the position of the hydraulic operator on the turbine control valve, which in turn controls the steam flow to the turbine. Hydraulic pressure is supplied by the shaft-driven hydraulic oil pump.

The turbine is automatically shut down by tripping the turbine trip and throttle valve closed if any of the following conditions are detected

- a. Turbine overspeed
- b. High turbine exhaust pressure
- c. An RCIC isolation signal from Logic A or B
- d. Low pump suction pressure
- e. Manual trip.

Turbine overspeed indicates a malfunction of the turbine control mechanism. High turbine exhaust pressure indicates a condition that threatens the physical integrity of the exhaust line. Low pump suction pressure warns that cavitation and lack of cooling can cause damage to the pump which could place it out of service. A turbine trip is initiated for these conditions so that the system can be quickly restored to service if the causes of the abnormal conditions can be found and corrected. The trip settings are selected far enough from normal values so that a spurious turbine trip is unlikely, but not so far that damage occurs before the turbine is shut down. Turbine overspeed is detected by a standard turbine overspeed mechanical device.

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Two pressure switches are used to detect high turbine exhaust pressure; either switch can initiate turbine shutdown. One pressure switch is used to detect low RCIC system pump suction pressure.

High water level (Level 8) in the RPV indicates that the RCIC system has performed satisfactorily in providing makeup water to the RPV. Further increase in level could result in RCIC system turbine damage caused by gross carryover of moisture. The RPV high water level setting that closes the RCIC turbine steam supply valve is near the top of the steam separators and is sufficient to prevent gross moisture carryover to the turbine. Two level transmitters that sense differential pressure are arranged so that both transmitters are required to trip in order to halt RCIC operation.

UFSAR Section 7.5.1.4.2.1 describes that the wide-range water level system is uncompensated for variation in reactor water density and is calibrated to be most accurate at operational pressure and temperature conditions. At low reactor coolant temperatures and pressures, the higher water density causes the wide-range instruments to read higher than both the narrow-range instruments and the actual water level. Below approximately 550 psig, this phenomenon results in a wide-range level above the RPV high water level setting (Level 8) when the actual water level is normal. With wide-range level above Level 8, RCIC manual initiation is inhibited by the Level 8 trip signal. RCIC automatic initiation on RPV low water level (Level 2) remains available since the Level 8 signal is automatically reset by the occurrence of a Level 2 actuation signal. For accidents occurring below 600 psig for which RCIC may be effective, analysis has shown that RCIC automatic initiation at Level 2 is sufficient to perform the intended safety function and that the analyses of record from normal reactor pressure are bounding. Amendment 206 revised the Technical Specifications to indicate that the RCIC function of manual initiation is not required to be operable below a reactor pressure of 550 psig.

7.4.1.1.3.2 Logic and Sequencing

Reactor pressure vessel low water level automatically starts the RCIC system, as indicated in Figure 7.4-1.

The RCIC trip is powered by the Division I 260/130-V dc battery.

Instrument settings for the RCIC system instrumentation and control are listed in Table 7.4-1. The water level setting is far enough below normal levels that spurious RCIC system startups are avoided.

To prevent the turbine pump from being damaged by overheating at reduced RCIC pump discharge flow, a pump discharge bypass is provided to route the water discharged from the pump back to the suppression pool. The bypass is controlled by an automatic, dc motor-operated valve whose control scheme is shown in Figure 7.4-2. At RCIC high flow, the valve is closed; at low flow, the valve is opened. A flow switch that measures the pressure difference across a flow element in the RCIC pump discharge line provides the signals.

7.4.1.1.3.3 Bypasses and Interlocks

The RCIC steam supply line is maintained hot to prevent build-up of condensate by utilizing a condensate drain pot, steam line drain, and appropriate valves in a drain line arrangement just upstream of the turbine supply valve. The water level in the steam line drain condensate pot is controlled by a level switch and a solenoid piloted air operator, which energizes to allow condensate to bypass a manually controlled globe valve during periods of high condensate such as warming the steam line. The control scheme is shown in Figure 7.4-1. The controls position valves so that during normal operation, steam line drainage is routed to the main condenser. On receipt of a RCIC initiation signal and subsequent opening of RCIC turbine inlet valve E5150F045, the drainage path is isolated.

During test operation, the RCIC pump discharge is routed to the condensate storage tank. A dc motor-operated valve is installed in the pump-discharge-to-condensate-storage-tank line. The piping arrangement is shown in Figure 5.5-7. The control scheme for the valves is shown in Figure 7.4-1. On receipt of a RCIC system initiation signal, the valve closes and remains closed. The valve is interlocked closed if either of the suppression chamber suction valves is not fully opened. Numerous indications pertinent to the operation and condition of the RCIC system are available to the main control room operator. Figure 7.4-1 shows the various indications provided.

Keylock switches have been added to inboard and outboard steam isolation valve control circuitry, as shown in Figure 7.4-1, to ensure deliberate operator action to manually close these valves. Additionally, a control room annunciator alarms when the F007 and F008 valves are not in the fully open position. This prevents damage from water hammer caused by inadvertent valve reopening. Should either or both of these valves be closed, the outboard isolation valve can be slowly reopened to allow any moisture in the line to drain. Then line pressure across the inboard isolation valve is equalized, and the downstream line is warmed by slowly opening the inboard isolation valve.

7.4.1.1.3.4 Redundancy and Diversity

Four reactor water level sensors in a "one-out-of-two taken twice" circuit supply the signal which results from a loss-of-water inventory condition.

7.4.1.1.3.5 Actuated Devices

All automatic valves in the RCIC are equipped with remote manual test capability so that the entire system can be operated from the main control room. Motor-operated valves are provided with appropriate limit switches to turn off the motors when the fully open or fully closed positions are reached. Logic circuitry that controls valves which are automatically closed on isolation or turbine trip signals is equipped with manual reset devices so that the valves cannot be reopened without operator action. All required components of the RCIC controls operate independently of ac power.

To ensure that the RCIC system can be brought to design flow rate within 50 sec from the receipt of the initiation signal, the following maximum operating times for essential RCIC valves are provided by the valve operation mechanisms:

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- a. RCIC turbine steam supply valve - 45 sec
- b. RCIC steam warmup bypass valve - 10 sec
- c. RCIC pump discharge injection valves - 30 sec
- d. RCIC pump minimum flow bypass valve - 25 sec

The operating time is the time required for the valve to travel from the fully closed to the fully open position, or vice versa. The two RCIC steam supply line isolation valves are normally open. They are intended to isolate the RCIC steam line in the event of a break in that line. A normally closed dc motor-operated isolation valve and a normally closed dc motor-operated warmup bypass valve are located in the turbine steam supply line just upstream of the turbine stop valve. The control schemes for these valves are shown in Figure 7.4-1. On receipt of a RCIC initiation signal, the valves open. The turbine steam supply valve remains open until closed by operator action or a level 8 trip. The warmup bypass valve remains open for 25 seconds then auto closes if a level 2 signal is not present.

Two normally open isolation valves are provided in the steam supply line to the turbine. The valve inside the drywell is controlled by an ac motor. The valve outside the drywell is controlled by a dc motor. The control diagram is shown in Figure 7.4-1. The valves automatically close on receipt of an RCIC isolation signal.

The instrumentation for RCIC isolation consists of the following:

- a. Inside valve
 1. Ambient temperature sensor - emergency area cooler high temperature. Isolation is initiated immediately
 2. Differential pressure transmitter - RCIC steam line high flow or instrument line break. A time delay has been installed to prevent inadvertent system isolation due to pressure spikes associated with pump startup. The delay device setpoints (approximately 3 sec), along with the surveillance intervals, are included in the Technical Specifications
 3. Two pressure transmitters - RCIC turbine exhaust diaphragm high pressure. Both transmitters must activate to isolate
 4. Two pressure transmitters - RCIC steam supply pressure low. Both transmitters must activate to isolate.

- b. Outside valve

A similar set of instrumentation causes the outside valve to isolate with the addition of manual isolation if the low-level initiation signal is present.

Three pump suction valves are provided in the RCIC system. One valve lines up pump suction from the condensate storage tank, the other two from the suppression chamber. The condensate storage tank is the preferred source. All three valves are operated by dc motors. The control arrangement is shown in Figure 7.4-1. On receipt of an RCIC initiation signal, the condensate storage tank suction valve automatically opens. On receipt of a condensate

storage tank low level signal, RCIC suction is automatically switched to the suppression pool. This is further discussed in Subsection 7.4.1.1.3.8.

Two dc motor-operated RCIC pump discharge valves are provided in the pump discharge line. The control schemes for these two valves are shown in Figure 7.4-1. Both valves are arranged to open on receipt of the RCIC initiation signal. One of the pump discharge valves closes automatically if a turbine trip occurs. The other valve remains open after RCIC initiation until closed by operator action in the main control room.

7.4.1.1.3.6 Separation

As in the emergency core cooling system (ECCS), the RCIC system is separated into divisions designated I and II (Subsection 7.3.1.2.1.6). The RCIC is a Division I system, but the inside steam line valve is in Division II: therefore, part of the RCIC logic is treated as Division II. The inside valve is an ac powered valve. The rest of the valves are dc-powered valves. Division I logic is powered by a 260/130-V dc Division I battery, and the Division II logic is powered by a 260/130-V dc Division II battery.

7.4.1.1.3.7 Testability

The RCIC may be tested to design flow during normal plant operation. Water is drawn from the condensate storage tank and discharged through a full flow test return line to the condensate storage tank. The discharge valve from the pump to the feedwater line remains closed during the test, and reactor operation remains undisturbed. Design of the control system is such that the RCIC system returns to the operating mode from the full flow test if system initiation is required.

7.4.1.1.3.8 Automatic Switchover of RCIC System Suction

In the original design, the switchover from the condensate storage tank to the suppression chamber as a source of water was to be manually controlled. However, as a result of discussions with the NRC, this was changed to automatically controlled. Automatic RCIC suction transfer occurs when the trip logic is deenergized. The trip logic is developed within the Division II HPCI system from redundant analog CST level transmitters and trip units. The HPCI level trip units provide redundant signals to the Division I RCIC suction transfer circuitry through auxiliary relay contacts. Either of these redundant low level signals will automatically open RCIC valves F029 and F031 (refer to Figures 5.5-7, 7.4-1, and 7.4-2). The RCIC suction transfer then uses the full-open position limit switches on F029 and F031 to initiate the closure of F010, the valve in the pump suction connection to the condensate storage tank. Panel status information is provided for the operator in the form of valve position indication.

The condensate storage tank level instrumentation is designed to meet the position of the NRC's Instrumentation and Controls System Branch with respect to freeze protection.

A single source connection penetrates the tank. This source connection is common to both the analog transmitters that monitor tank level for the purpose of transferring the RCIC/HPCI pump suction and the transmitter associated with the continuous wide-range tank level indication provided in the main control room. This equipment is contained within a large

insulated steel cabinet (H21-P492) welded directly to the exterior of the condensate storage tank about 3 ft above ground level. The environment within the cabinet is maintained above freezing by a radiant strip heater and a local-control thermostat. A temperature-sensing device that is independent of the strip heater and its associated control thermostat is also located within the cabinet. This sensor produces a visual and audible alarm in the main control room whenever the temperature in the transmitter cabinet falls below 40°F. The cabinet temperature control and the low-temperature alarm are electrically independent and powered from completely independent and diverse power sources. A failure of either would not affect the ability of the other to perform its function. To guarantee the continued performance of the environmental control and monitoring systems, Edison will perform a yearly functional surveillance of the systems prior to the advent of freezing weather.

Edison has based its justification of the nonseismic location of the transmitters used in the suction transfer system primarily on the degree of conservatism in instrumentation seismic design. The level transmitters used in this transfer application were seismically qualified as described in the licensing topical report NEDO-21617.

Fermi site ground response spectra applicable to a transmitter mounting on the tank located at grade level would fall well below the values used for qualification of the transmitters in the reference document. As a result, the transmitters are expected to operate properly during and after a seismic event. As an added degree of conservatism, a failure of the tank which results in a loss of inventory and/or loss of the current signal from either transmitter will cause trip units (E41-N661 B and D) and associated trip relays to transfer the RCIC and HPCI suction valves to the suppression pool. These trip units and relays are located on the fourth floor of the reactor building in panel H21-P081. These devices and cabinet are located within the seismically qualified portion of the plant and meet the environmental and seismic qualification requirements for Class 1E electrical equipment.

All of the equipment that accomplishes the automatic suction valve transfer on low condensate tank level is classified as Quality Level 1. The transmitters were purchased as qualified instruments along with the balance of the transfer system and are included with the trip units and relays in the Technical Specifications because the surveillance requirement includes the entire measurement loop.

7.4.1.1.4 Environmental Considerations

The only RCIC control component located inside the primary containment that must remain functional in the environment resulting from a LOCA is the control mechanism for the inside isolation valve. The environmental capabilities of this valve are discussed in Subsection 7.3.2.2.9. The RCIC instrumentation and control equipment located outside the primary containment is selected in consideration of the normal and accident environments in which it must operate. Refer to Subsection 7.4.1.1.3.8 for information on the environmental considerations for the HPCI/RCIC instruments on the condensate storage tank.

Level sensing instrumentation used as inputs to the RCIC logic from residual heat removal (RHR) is discussed in Subsection 7.3.1.2.4.

7.4.1.1.5 Operational Considerations

7.4.1.1.5.1 General Information

Core cooling is required in the event that the reactor becomes isolated from the main condensers during normal operation by a closure of the main steam isolation valves (MSIVs). Cooling is necessary because of the core fission product decay heat. Steam is vented through the pressure safety/relief valves to the suppression pool. The RCIC system maintains reactor water level by providing the makeup water. Initiation and control are automatic.

The provisions taken in accordance with General Design Criterion (GDC) 19 of 10 CFR 50, Appendix A, to provide the required equipment outside the main control room for hot and cold shutdown, are described in Subsection 7.5.1.5.1.

7.4.1.1.5.2 Setpoints

A list of setpoints for the RCIC system can be found in Table 7.4-1.

7.4.1.2 Standby Liquid Control System Instrumentation and Control

7.4.1.2.1 System Identification

7.4.1.2.1.1 Function

The instrumentation and control system for the standby liquid control system (SLCS) is designed to inject water-soluble neutron-absorber solution well above saturation temperature.

7.4.1.2.1.2 Classification

The SLCS is a backup method of manually shutting down the reactor to cold subcritical independently from the control rod drive system. Thus, the system is considered a control system and not a safety system. The standby liquid control process equipment, instrumentation, and control essential for injection of the neutron-absorber solution into the reactor are designed to withstand Category I earthquake loads. Nonprocess equipment and instrumentation and control are designed as a nonseismic system. The SLCS has been reclassified to identify that it was not originally intended, procured, designed, or classified as safety related, but it will be maintained and tested as a safety-related system after completion of its preoperational tests.

7.4.1.2.2 Power Sources

The power supply to explosive valve F004A and injection pump C001A is from automatically restored MCC 72B-4C. The power supply to explosive valve F004B and injection pump C001B is from automatically restored MCC 72E-5B. The location of these pumps and valves is shown in Figure 7.4-3. The power supply to the tank heaters and heater controls can also be connected to an engineered safety feature (ESF) bus. The 120-V ac power supply to the main control room benchboard indicator lights is powered from an inductive BOP MPU, and the level and pressure transmitters are powered from restorable instrument MPU 1.

7.4.1.2.3 Equipment Design

7.4.1.2.3.1 Initiating Circuits

The standby liquid control is initiated in the main control room by turning a keylocking switch to either system A or system B. The key is removable in the center OFF position. When either system is initiated, both explosive valves (F004A and F004B) are fired, and the selected pump C001A or C001B is started. Should the selected pump fail to start, the key switch may be turned to the alternate pump.

7.4.1.2.3.2 Logic and Sequencing

When the SLCS is initiated, both the explosive valves fire and the pump that has been selected for injection starts.

7.4.1.2.3.3 Bypasses and Interlocks

There are no bypasses. When the SLCS is initiated to inject soluble neutron absorber into the reactor, the outboard isolation valve of the reactor water cleanup (RWCU) is automatically closed.

7.4.1.2.3.4 Redundancy and Diversity

The redundancy exists in duplicated pumps, explosive valves, and power supply as outlined in Subsection 7.4.1.2.2.

7.4.1.2.3.5 Actuated Devices

When the SLCS is initiated to inject soluble neutron absorber into the reactor, one of the two injection pumps and each of the two explosive valves are actuated.

7.4.1.2.3.6 Testability

The instrumentation and control system of the SLCS is tested when the system test is performed as outlined in Subsection 4.5.2.4.4.

7.4.1.2.4 Environmental Considerations

The environmental considerations for the instrumentation and control portions of the SLCS are the same as for the active mechanical components of the system. This is discussed in Section 3.11 and Subsection 4.5.2.4.3.

7.4.1.2.5 Operational Considerations

7.4.1.2.5.1 General Information

The control scheme for the SLCS can be found in Figure 7.4-3. The standby liquid control is manually initiated in the main control room by inserting the proper key into the keylocking

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switch and turning it to either system A or system B. The time it takes to complete the injection is between 50 and 125 minutes. When the injection is completed, the system is manually turned off by returning the keylocking switch to the OFF position.

7.4.1.2.5.2 Operator Information

The SLCS indicators are as follows:

- a. The system pressure is indicated with an indicator that has a range of 0-1800 psig in the main control room
- b. The storage tank level is indicated with an indicator that has a range of near empty to near full, calibrated to read in inches of liquid storage in the main control room
- c. The continuity of the explosive valve dual primer ignition circuit is monitored by measuring a trickle current through the primers. If either of the dual primer or the primer ignition circuit becomes open-circuited, the continuity meter reads downscale
- d. Indicator lights in the main control room show if either pump is running, stopped, or tripped
- e. Indicator lights in the main control room show whether or not the explosive valve firing circuitry has continuity
- f. Indicator lights in the main control room show if service valve F008 is open or closed, as shown in Figure 7.4-3
- g. Indicator lights in the main control room show if the F006 check valve disk is open or closed
- h. Indicator lights on the local panel show if the manually controlled high-power storage tank heater is on or off
- i. Indicator lights on the local panel for the low-power storage tank heater have been de-energized and abandoned in place.

The SLCS main control room annunciators annunciate when

- a. There is a loss of continuity of either explosive valve primers
- b. The standby liquid storage temperature becomes too hot or too cold
- c. The standby liquid tank level is too high or too low.

7.4.1.2.5.3 Setpoints

The SLCS has setpoints for the various instruments as follows:

- a. The loss of continuity meter is set to activate the annunciator just below trickle current that is observed when the primers of the explosive valves are new
- b. The high and low standby liquid temperature switch is set to activate the annunciator at temperatures of approximately 110°F and 48°F, respectively

- c. The high and low standby liquid storage tank level switch is set to activate the annunciator when the volume is approximately 2975 gal net and 2618 gal net of the storage tank capacity, respectively

7.4.1.3 Reactor Shutdown Cooling System Instrumentation and Control

7.4.1.3.1 System Identification

The shutdown cooling mode is a function of the RHR system and is placed in operation during a normal shutdown and cooldown.

7.4.1.3.2 Power Sources

The power sources for the reactor shutdown cooling system instrumentation and control are as described in the ECCS discussion in Subsection 7.3.1.2.

7.4.1.3.3 Equipment Design

The reactor water is cooled by taking suction from one of the recirculation loops as shown in Figure 5.5-13. During the shutdown cooling mode, only one RHR system heat exchanger is required. This allows the remaining RHR system division to be held in standby for use in either the low-pressure coolant injection (LPCI) mode or containment cooling mode. One RHR division's valve alignment is shifted from the standby mode lineup (suction from the torus) needed for LPCI and containment cooling to the shutdown mode lineup (suction from reactor recirculation loop-B) after the reactor is depressurized. One RHR heat exchanger removes enough decay heat, even with declining reactor water approach temperature, so that the proper cooldown rate may be achieved.

If it is necessary to discharge a complete core load of reactor fuel to the spent fuel pool, the cooling capacity of the fuel pool cooling and cleanup system (FPCCS) heat exchangers may be exceeded. A means is provided for making a physical connection between the spent fuel pool and the RHR system. The RHR heat exchangers have greater cooling capacity than the FPCCS heat exchangers, and can maintain the spent fuel pool within its design temperature until the decay heat load is within the capacity of the FPCCS.

7.4.1.3.3.1 Initiating Circuits

The reactor shutdown cooling system is initiated only by manual action. The system cannot be actuated unless certain requirements, described in the following subsections, are met.

7.4.1.3.3.2 Bypasses and Interlocks

To prevent opening the shutdown cooling valves except under proper conditions, interlocks are provided as shown in Table 7.4-2.

The two RHR pumps used for shutdown cooling are interlocked to trip the pumps if the shutdown cooling valves and suction valves from the suppression pool are not properly positioned.

7.4.1.3.3.3 Actuating Devices

All motor-operated valves in the shutdown cooling system are equipped with remote manual switches in the main control room.

7.4.1.3.3.4 Testability

The shutdown cooling system pumps of the RHR system may be tested to full capacity during normal plant operation.

7.4.1.3.4 Environmental Considerations

The only shutdown cooling control component located inside the drywell that must remain functional in the environment is the control mechanism for the (inboard) isolation shutdown cooling suction valve. The environmental capabilities of this valve are discussed in Subsection 7.3.2.2.9. The instrumentation and control equipment located outside the drywell is selected in consideration of the normal and accident environments in which it must operate.

7.4.1.3.5 Operational Considerations

All controls for the shutdown cooling system are located in the main control room. Operator information is provided as described in the RHR discussion of the LPCI mode in Subsection 7.3.1.2.4.

The provisions taken in accordance with GDC 19 of 10 CFR 50, Appendix A, to provide the required equipment outside the main control room for hot and cold shutdown, are described in Subsection 7.5.1.5.1.

7.4.2 Analysis

7.4.2.1 General

Presented below are analyses that show how the safe shutdown systems satisfy their design bases listed in Section 7.1.

7.4.2.2 Reactor Core Isolation Cooling System Instrumentation and Control

7.4.2.2.1 Conformance To General Functional Requirements

For events other than pipe breaks, the RCIC system has a makeup capacity sufficient to prevent the RPV water level from decreasing to the level where the core is uncovered without using the ECCS.

To ensure to a high degree that the RCIC system operates when necessary and in time to provide adequate core cooling, the power supply for the system is taken from reliable sources that are immediately available. Evaluation of instrumentation configuration for the RCIC system shows that no failure of a single initiating sensor either prevents the starting or causes false starting of the system.

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A design flow functional test of the RCIC system can be performed during plant operation by taking suction from the demineralized water in the condensate storage tank and discharging through the full flow test return line back to the condensate storage tank.

During the test, the discharge valve to the feed line remains closed and reactor operation is undisturbed. Control system design provides automatic return from the full flow test mode to the operating mode if system initiation is required during testing.

7.4.2.2.2 Conformance To Specific Regulatory Requirements of IEEE 279-1971

7.4.2.2.2.1 Single-Failure Criterion (IEEE 279-1971, Paragraph 4.2)

The RCIC system, by itself, is not required to meet the single- failure criterion. The control logic circuits for the RCIC system initiation and control are housed in a single relay cabinet, and the power supply for the control logic and other RCIC equipment is from a single dc power source.

The RCIC initiation sensors and wiring up to the RCIC relay logic cabinet do, however, meet the single-failure criterion. Physical separation of instrument lines is provided so that no single instrument rack destruction or single instrument line (pipe) failure can prevent RCIC initiation. Wiring separation between divisions also provides tolerance to single wireway destruction (including shorts, opens, and grounds) in the accident detection portion of the control logic. The single-failure criterion is not applied to the logic relay cabinet or to other equipment required to function for RCIC operation.

7.4.2.2.2.2 Quality Components (IEEE 279-1971, Paragraph 4.3)

This requirement is described in NEDO-10139, which applies equally to the core spray and RCIC systems.

7.4.2.2.2.3 Equipment Qualification (IEEE 279-1971, Paragraph 4.4)

Environmental

No components of the RCIC control system are required to operate in the drywell environment. The RCIC steam line isolation valve located inside the drywell is a normally open valve and is required to operate only to isolate the primary containment.

Other process sensor equipment for RCIC initiation is located in the reactor building and is capable of accurate operation in ambient temperature conditions that result from abnormal conditions.

Panels and relay cabinets are located in typical power station control room and/or auxiliary relay room environments. Therefore, environmental testing of components mounted in these enclosures is not warranted.

There are no components in the RCIC control system that have not demonstrated their reliable operability in previous applications in nuclear power plant protection systems or in extensive industrial use.

7.4.2.2.2.4 Channel Integrity (IEEE 279-1971, Paragraph 4.5)

The RCIC system instrument initiation channels meet the single- failure criterion as discussed in Subsection 7.4.2.2.2.1 above, and thus satisfy the channel integrity objective of this paragraph.

By definition (IEEE 279-1971, Paragraph 2.2), a channel loses its identity where single-action signals are combined. Therefore, since instrument channels are combined into a single trip system, this paragraph of IEEE 279-1971 does not strictly apply for the RCIC control system.

7.4.2.2.2.5 Channel Independence (IEEE 279-1971, Paragraph 4.6)

Channel independence for initiation sensors is provided by electrical and mechanical separation. The A and C sensors for RPV level, for instance, are located on one local instrument panel identified as Division I equipment, and the B and D sensors are located on a second instrument panel widely separated from the first and identified as Division II equipment.

The A and C sensors have a common pair of process taps that are widely separated from the corresponding taps for sensors B and D. Disabling of one or both sensors in one location does not disable the control for RCIC initiation.

7.4.2.2.2.6 Control and Protection Interaction (IEEE 279-1971, Paragraph 4.7)

The RCIC system is strictly an off-on system, and no signal whose failure could cause need of RCIC can also prevent RCIC from starting. Annunciator circuits using contacts of sensor relays and logic relays cannot impair the operability of the RCIC system control because of the electrical separation between controls. A short between the annunciator wiring and the RCIC control wiring could result in a single ground on the dc control circuit without affecting circuit operability.

7.4.2.2.2.7 Derivation of System Inputs (IEEE 279-1971, Paragraph 4.8)

The input that starts the RCIC system is a direct measure of the variable that indicates need for core cooling; e.g., RPV low water level.

7.4.2.2.2.8 Capability for Sensor Checks (IEEE 279-1971, Paragraph 4.9)

All sensors are of the pressure-sensing type and are installed with calibration taps and instrument valves so that testing during normal plant operation or during shutdown is permitted.

The reactor low-pressure transmitters can be easily checked for operability during plant operation by observing the analog output of respective transmitters. The RPV level transmitters are also checked for operability in a similar fashion. Refer to Subsection 7.1.3.1.

7.4.2.2.2.9 Capability for Test and Calibration (IEEE 279-1971, Paragraph 4.10)

The RCIC control system is capable of being completely tested under normal plant operation to verify that each element of the system, active or passive, is capable of performing its intended function. Sensors can be exercised by applying test pressures.

The RCIC system can be manually started in the test mode by opening steam supply valves to the RCIC turbine to pump water from the condensate storage tank through the test return valves back to the condensate storage tank, while the reactor is at pressure.

Motor-operated valves can be exercised by the appropriate control relays and starters, and all indications and annunciations can be observed as the system is tested.

7.4.2.2.2.10 Channel Bypass or Removal From Operation (IEEE 279-1971, Paragraph 4.11)

Calibration of a sensor that introduces a single instrument channel trip will not cause a protective function without the coincident trip of a second channel. There are no instrument channel bypasses in the RCIC system. Removal of a sensor from operation during calibration does not prevent the redundant instrument channel from functioning.

7.4.2.2.2.11 Operating Bypasses (IEEE 279-1971, Paragraph 4.12)

The RCIC system design contains no operating bypasses.

7.4.2.2.2.12 Indication of Bypasses (IEEE 279-1971, Paragraph 4.13)

Indication of bypasses provided is as discussed in Subsection 7.4.2.2.2.11.

7.4.2.2.2.13 Access To Means for Bypassing (IEEE 279-1971, Paragraph 4.14)

Access to motor control centers and instrument valves is controlled. Access to other means of bypassing is located in the main control room and is therefore under the administrative control of the operators.

7.4.2.2.2.14 Multiple Setpoint (IEEE 279-1971, Paragraph 4.15)

This is not applicable because all setpoints are fixed.

7.4.2.2.2.15 Completion of Protective Action Once It Is Initiated (IEEE 279-1971, Paragraph 4.16)

The final control elements for the RCIC system are essentially bi-stable, i.e., motor-operated valves stay open or closed once they have reached their desired position, even though their starter may drop out (which they do when the limit switch is reached). In the case of pump starters, the automatic initiation signal is electrically sealed in.

Thus, once protection action is initiated (i.e., flow established), it must go to completion or continue until terminated by deliberate operator action or automatically stopped on high vessel water level or system malfunction trip signals.

7.4.2.2.2.16 Manual Actuation (IEEE 279-1971, Paragraph 4.17)

Each piece of RCIC actuation equipment required to operate (pumps and valves) is capable of manual initiation electrically from the control panel in the main control room. Failure of logic circuitry to initiate the RCIC system will not affect the manual control of equipment.

However, failures of active components or control circuit failure which produces a turbine trip may disable the manual actuation of the RCIC system. Failures of this type are continuously monitored by alarms.

7.4.2.2.2.17 Access To Setpoint Adjustment (IEEE 279-1971 Paragraph 4.18)

Setpoint adjustments for the RCIC system sensors are integral with the sensors on the local instrument racks and cannot be changed without the use of tools to remove covers over these adjustments. Control relay cabinets are capable of being locked to prevent unauthorized actuation. Because of these restrictions, compliance with this requirement of IEEE 279-1971 is considered complete.

7.4.2.2.2.18 Identification of Protective Actions (IEEE 279-1971, Paragraph 4.19)

Protective actions are directly indicated and identified by annunciator operation or action of the sensor relay, which has an identification tag and a clear glass window front that permits convenient visible verification of the relay position. This combination of annunciation and visible relay actuation is considered to fulfill the requirements of this criterion.

7.4.2.2.2.19 Information Readout (IEEE 279-1971, Paragraph 4.20)

The RCIC control system is designed to provide the operator with accurate and timely information pertinent to its status. It does not introduce signals into other systems that could cause anomalous indications confusing to the operator. Periodic testing is the means provided for verifying the operability of the RCIC components and, by proper selection of test periods, to be compatible with the historically established reliability of the components tested, complete and timely indications are made available. Sufficient information is provided on a continuous basis so that the operator can have a high degree of confidence that the RCIC function is available and/or operating properly.

In addition to the annunciator alarms shown on the functional control diagram in Figure 7.4-1, the following alarms are provided:

- a. Failure of control power to the RCIC system
- b. Valve overload alarm.

In addition to the annunciators, the other indications on the main control room panel are

- a. Valve position lights
- b. Pump monitor lights
- c. Pump suction/discharge pressure indicator
- d. RCIC pump flow indicator

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- e. Turbine exhaust line pressure indicator
- f. Turbine steam inlet pressure indicator
- g. Turbine speed indicator
- h. Turbine supervisory indicators.
- i. Barometric condenser vacuum pump current
- j. Barometric condenser condensate pump current

7.4.2.2.2.20 System Repair (IEEE 279-1971, Paragraph 4.21)

The RCIC control system is designed to permit repair or replacement of components.

Recognition and location of a failed component will be accomplished during periodic testing. The simplicity of the logic will make the detection and location relatively easy, and components are mounted in such a way that they can be conveniently replaced in a short time. Sensors that are connected to the instrument piping cannot be changed so readily but are connected with separable screwed or bolted fittings.

7.4.2.2.2.21 Identification (IEEE 279-1971, Paragraph 4.22)

The RCIC system is identified uniquely as a Division I system. All controls and instruments are located in one area of the main control room panel and are clearly identified by nameplates.

Relays are located in one panel for RCIC use only. Relays and panels are identified by nameplates.

7.4.2.2.3 Conformance To Specific Regulatory Requirements

The RCIC system conforms to the following regulatory requirements:

- a. IEEE 323-1971 - This is discussed in Section 3.11 and GE Topical Report NEDO-10698
- b. IEEE 338-1971 - Only paragraphs of IEEE 338-1971 that apply to the design of the RCIC system will be covered
Capability for Sensor Checks (IEEE 338-1971, 2.1) is discussed in Subsection 7.4.2.2.2.8. Capability for Test and Calibration (IEEE 338-1971, 2.2) is discussed in Subsection 7.4.2.2.2.9
- c. IEEE 344-1971 - This is discussed in Topical Report NEDO-10678
- d. 10 CFR 50, Appendix A requirements -
 - 1. Criterion 13 - Subsections 7.4.1.1.3.1, 7.4.1.1.3.2, 7.4.1.1.3.3
 - 2. Criterion 37 - Subsection 7.4.1.1.3.7.
- e. 10 CFR 50, Appendix B requirements - The requirements of 10 CFR 50, Appendix B, are met as described in Chapter 17
- f. Regulatory Guide 1.22 - Subsections 7.4.2.2.2.8 and 7.4.2.2.2.9.

7.4.2.3 Standby Liquid Control System Instrumentation and Control

7.4.2.3.1 Conformance To General Functional Requirements

Redundant positive displacement pumps, explosive valves, and control circuits for these components have been provided as described in Subsection 7.4.1.2. This constitutes all the active equipment required for injection of the sodium pentaborate solution. Continuity relays provide monitoring on the explosive valves, and indicator lights provide indication on the main reactor control panel of system status as described in Subsection 7.4.1.2.5.2.

Testability is described in Subsection 7.4.1.2.3.6. Redundant power sources are described in Subsection 7.4.1.2.2.

7.4.2.3.2 Conformance To Specific Regulatory Requirements

Qualification of Class 1E electrical equipment in accordance with IEEE 323-1971 and seismic design of Class 1E electrical equipment in accordance with IEEE 344-1971 are covered in Topical Reports NEDO-10698 and NEDO-10678, respectively, and Sections 3.11 and 3.10.

The requirements of 10 CFR 50, Appendix B, are described in Chapter 17.

7.4.2.4 Reactor Shutdown Cooling System Instrumentation and Control

7.4.2.4.1 Conformance To General Functional Requirements

The design of the reactor shutdown cooling system instrumentation and controls meets all the functional requirements of Subsection 7.1.2.1.27 as follows:

7.4.2.4.1.1 Valves

Manual controls and position indicators are provided in the main control room. Interlocks are provided to prevent opening of the valves if shutdown conditions are not met. Interlocks are also provided to close the valves if an isolation signal is present or if high reactor pressure exists.

Redundant sensors (N111A and B) are provided for the RHR shutdown cooling pressure interlocks. These sensing loops meet or exceed the EICSB-3 Branch Technical Position. The interlocks are designed as part of the testability option and, therefore, formal diversity of the sensors and trip units has not been provided. Formal test procedures are used to verify operability of the interlocks. It is Edison's position that the accuracy, reliability, testing, and inherent on-line status monitoring of the analog transmitter/trip unit design obviate the need for diverse instruments.

7.4.2.4.1.2 Instrumentation

Shutdown flow indicator is provided. The RHR cooling water and service water temperatures are provided. Head spray flow indication is no longer provided. A permanent modification has removed the head spray piping, disabling this flow path.

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7.4.2.4.1.3 Annunciation

The following annunciators are provided:

- a. Division I/Division II RHR valves thermal overload
- b. RHR heat exchanger cooling water discharge temperature high
- c. RHR system overpressure
- d. RHR pump motor tripped.

7.4.2.4.1.4 Pumps

Manual controls and stop and start indicators are provided in the main control room. Interlocks are provided to trip the pumps if the shutdown cooling valves are not properly set up.

7.4.2.4.2 Conformance To Specific Regulatory Requirements

Conformances to regulatory requirements are the same as those specified for the ESF systems.

Consideration of failure of plant instrument air and loss of cooling water to safe-shutdown equipment is given in Chapter 15. These systems are not specifically designed for consideration of plant load rejection or turbine trip, but the plant is designed to handle those situations and shut down safely.

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TABLE 7.4-1 REACTOR CORE ISOLATION COOLING INSTRUMENT SPECIFICATION

RCIC Function	Instrument	Trip Settings ^a	Range
RPV high water level turbine trip	Level transmitter	Level 8 ^b	10 to 220 in. ^c
Turbine exhaust diaphragm high pressure	Pressure transmitter	10 psig	0 to 30 psi
RCIC system pump low suction pressure	Pressure switch	Low - 20 in. Hg vacuum ^d	30 in. Hg vacuum to 10 psig
RCIC system pump high suction pressure	Pressure switch	High - 70 psig	0.5 to 80 psig
RPV low water level ^e	Level transmitter	Level 2 ^b	10 to 220 in. ^c
RCIC system steam supply low pressure	Pressure transmitter	50 psig	0 to 200 psig
Turbine overspeed	Centrifugal device	122.3 percent of rated speed	

^a Nominal values are given for information. See the Technical Specifications for operational limits.

^b Figure 7.3-12.

^c Zero is at the top of the active fuel.

^d Approximate setting.

^e Incident detection circuitry instrumentation.

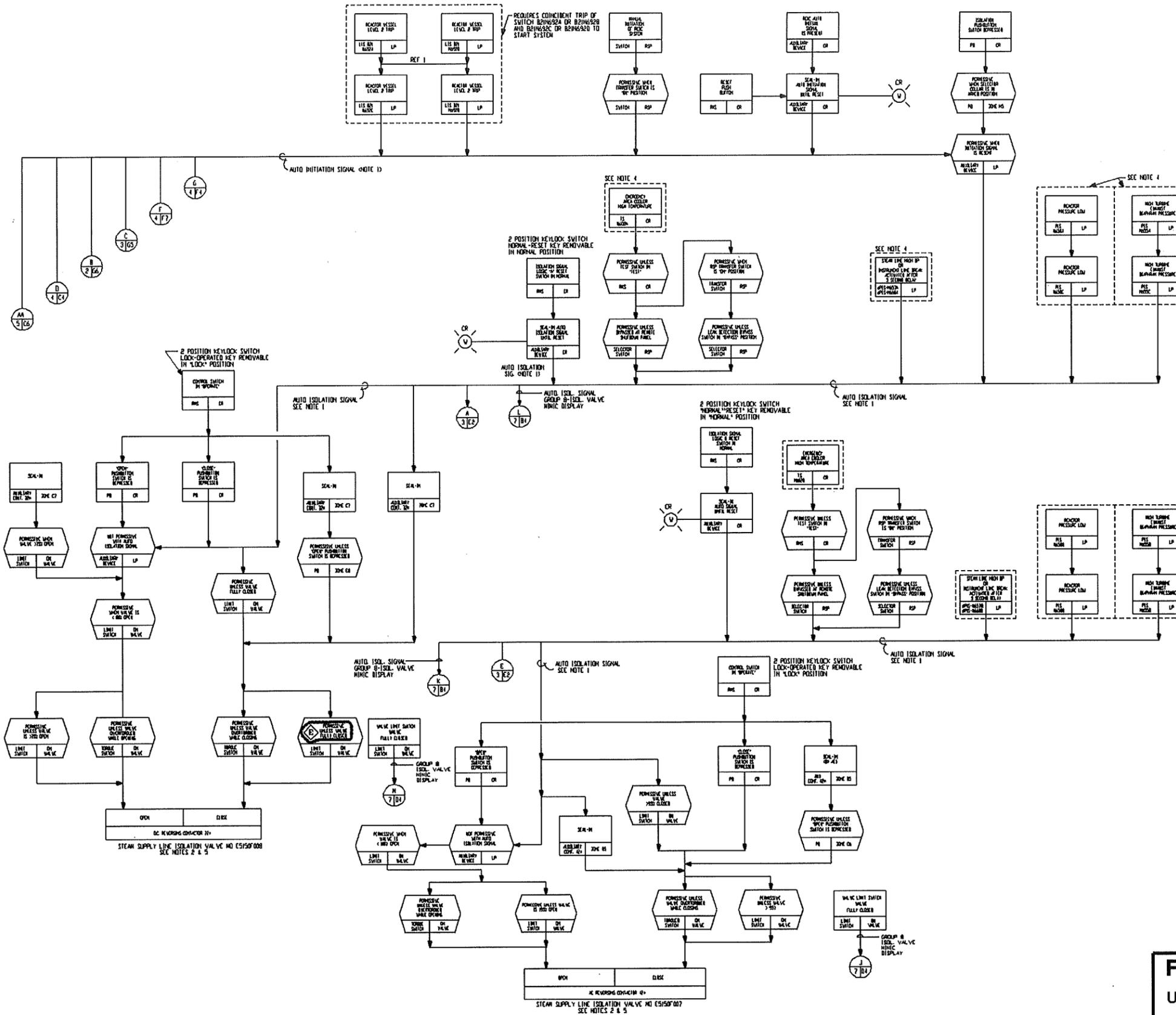
TABLE 7.4-2 REACTOR SHUTDOWN COOLING BYPASSES AND INTERLOCKS

Valve Function Manual Open	Reactor Pressure Exceeds Shutdown	Isolation Valve Closure Signal
Inboard suction isolation	Cannot open	Cannot open
Outboard suction isolation	Cannot open	Cannot open
Reactor injection	Can open	Cannot open
Head spray ^c	Cannot open	Cannot open
Valve function Automatic ^a close or Manual ^b close		
Inboard suction isolation	Closes A and M	Closes A and M
Reactor injection	Closes A and M	Closes A and M
Head spray ^c	Closes A and M	Closes A and M

^a Automatic is abbreviated as “A.”

^b Manual is abbreviated as “M.”

^c Head spray piping attached to reactor pressure vessel (RPV) is removed. The remaining pipe in drywell is blanked off.



- NOTES:**
1. THE RWC SYSTEM IS ARRANGED FOR TEST OF PUMP AT FULL FLOW & ALL VALVES FOR OPEN & CLOSE CAPABILITY AT ANY TIME EXCEPT WHEN INITIATION SIGNAL OR AUTO ISOLATION SIGNAL OCCURS WHILE TEST IS UNDERWAY. THE SYSTEM AUTOMATICALLY RETURNS TO STARTUP MODE.
 2. ALL POWER FOR OPERATION OF DC VALVE MOTORS SHALL ORIGINATE FROM A PLANT DC BUS. POWER FOR AC OPERATED SHALL ORIGINATE FROM AN EMERGENCY AC BUS.
 3. UNLESS OTHERWISE SHOWN ALL DISTRIBUTION PIS INCL. ARE PREFIXED E51. ALL VALVE AND EQUIPMENT PIS INCL. ARE PREFIXED E5100.
 4. ISOLATION SIGNAL SWITCHES SHALL BE OF THE TYPE THAT CLOSE CONTACTS FOR THE SPECIFIED ISOLATION EVENT. WHERE AUXILIARY RELAYS ARE USED IN THE ISOLATION CHANNELS THEY SHALL BE POWERED FROM THE STATION BATTERY SO THAT THE LOSS OF AC POWER DOES NOT CAUSE AN ISOLATION TRIP.
 5. AUXILIARY RELAYS & DEVICES NOT SHOWN ON FUNCTIONAL CONTROL DIAGRAMS EXCEPT WHERE REQUIRED TO CLARIFY FUNCTION.
 6. DELETED.
 7. FURNISHED WITH TURBINE.
 8. DELETED.
 9. THE RWC SYSTEM SHALL BE DESIGNED IN ACCORDANCE WITH PROPOSED CRITERIA FOR NUCLEAR POWER PLANT PROTECTION SYSTEM IEEE-299 INsofar AS PRACTICABLE.
 10. FOR LOCATION, IDENTIFICATION AND SET POINTS OF INSTRUMENTS SEE THE MASTER INSTRUMENT LIST.
 11. FOR CONFIGURATION OF PANEL MOUNTED OPERATORS AND INDICATORS SEE BWS-01-2230 SH. REF. 12 PURCHASE SPECIFICATION DATA SHEET SECT. 2.1.
 12. ALL MOTORS SHALL BE PROTECTED BY THERMAL OVERLOAD TRIPS AND BREAKER SHORT CIRCUIT CURRENT PROTECTIVE TRIPS.

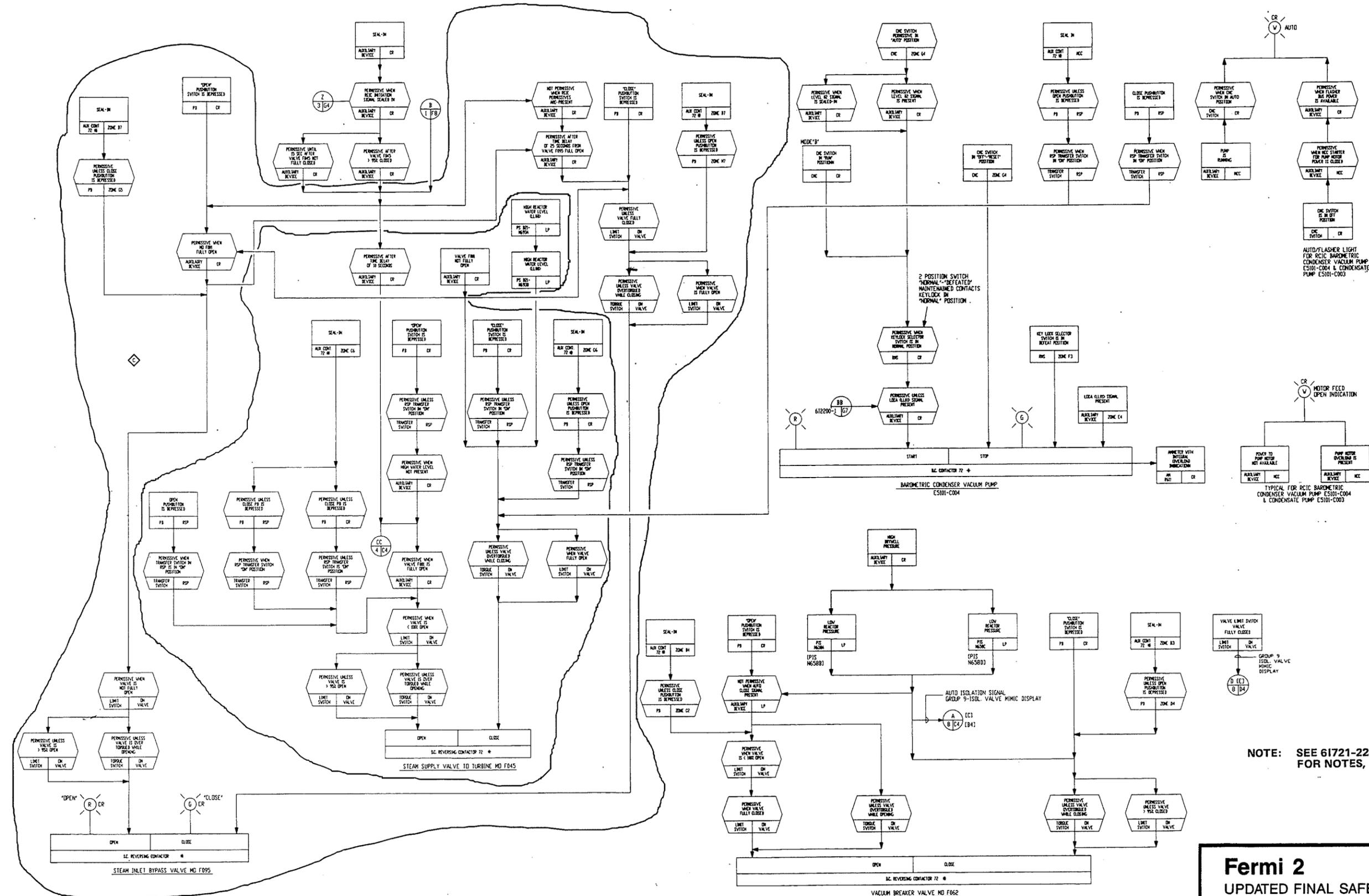
- REFERENCE DOCUMENTS:**
1. NUCLEAR BOILER PAID-----N-2095, N-2090, N-2538
 2. NUCLEAR BOILER MISC. SYSTEM LOGIC DIAGRAM-----I-2090-01
 3. RWC SYSTEM LOGIC DIAGRAM-----I-2230-01
 4. CORE SPRAY SYSTEM LOGIC DIAGRAM-----I-2201-01
 5. RWC SYSTEM LOGIC DIAGRAM-----I-2230-01
 6. RWC SYSTEM PAID-----N-2014, N-2015
 7. LOGIC SYMBOLS-----NSA1756; RI-20
 8. TURBINE CONT. SYS. & ELEC. WIRING-----TERRY TURBINE VENDOR MANUAL MESI-01-C-002-J1-002
 9. REACTOR WATER CLEANER SYS. PAID-----N-2046, N-2047
 10. LEAK DETECTION SYSTEM DESIGN SPACE-----224141; RI-138
 11. ELEC. EQUIP. SEPARATION FOR SAFEGUARD SYS.-----224377; RI-140
 12. GENERAL REQUIREMENTS FOR INSTRUMENTATION & ELEC. EQUIP.-----214768; RI-146, 148
 13. RWC SCHEMATICS-----I-2231-01 THRU 10 I-2232-01 THRU 11
 14. LOGIC DIAGRAM-GROUP 9-RWC ISOLATION VALVE MIMIC DISPLAY-----I-2230-07
 15. LOGIC DIAGRAM-GROUP 9-RWC ISOLATION VALVE MIMIC DISPLAY-----I-2230-08

- LEGEND:**
- CMC COORDINATED MANUAL CONTROL SWITCH
 - PS PUSHBUTTON SWITCH
 - AM AMMETER
 - SK SWITCHGEAR DEVICE FUNCTION NUMBERS
 - ANSI SPEC. C17.2
 - RP REMOTE SWITCH ROOM PANEL
 - RMS REMOTE MANUAL SWITCH
 - LP LOCAL PANEL
 - CR CONTROL ROOM & RELAY ROOM

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FIGURE 7.4-1, SHEET 1

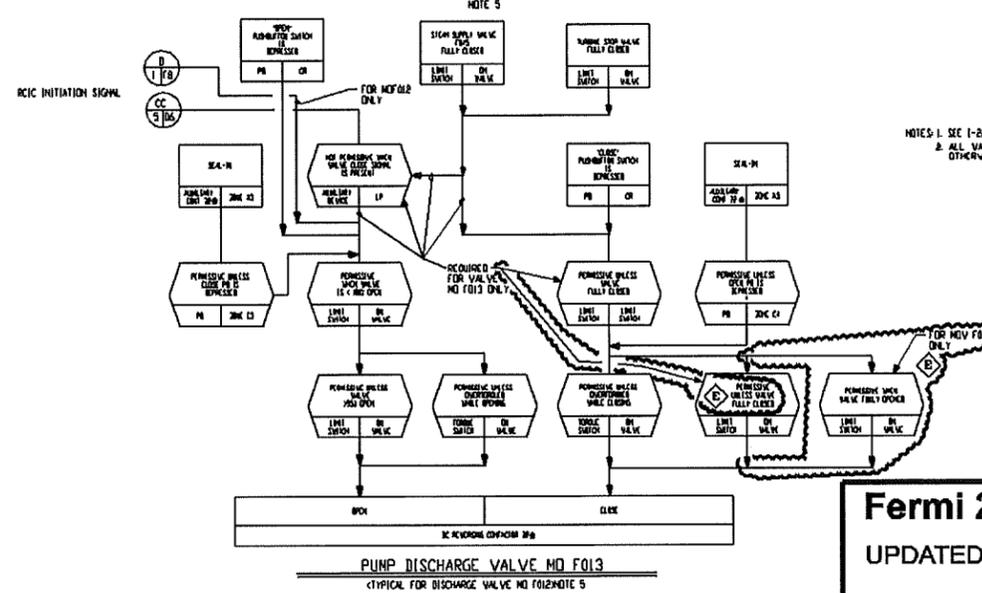
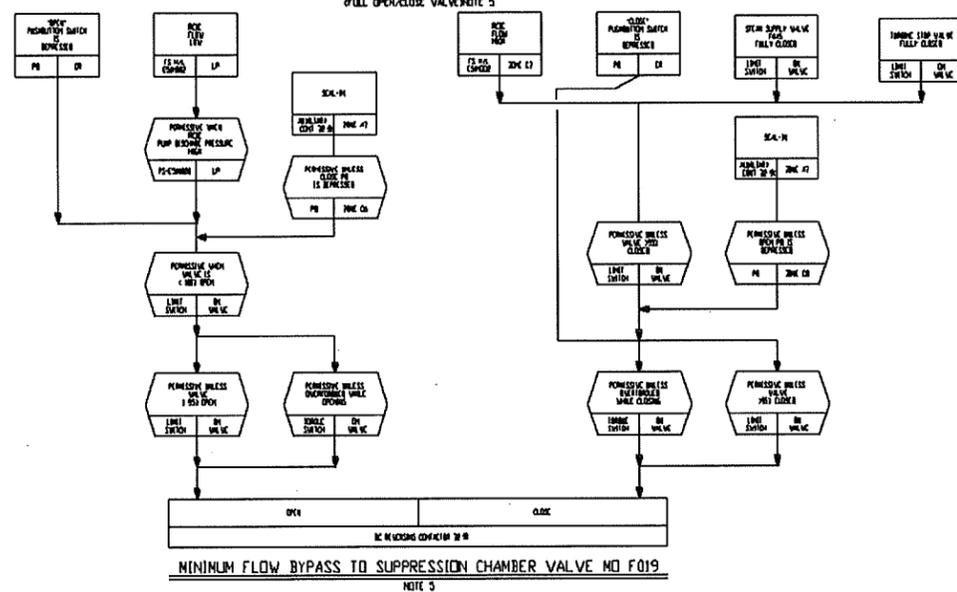
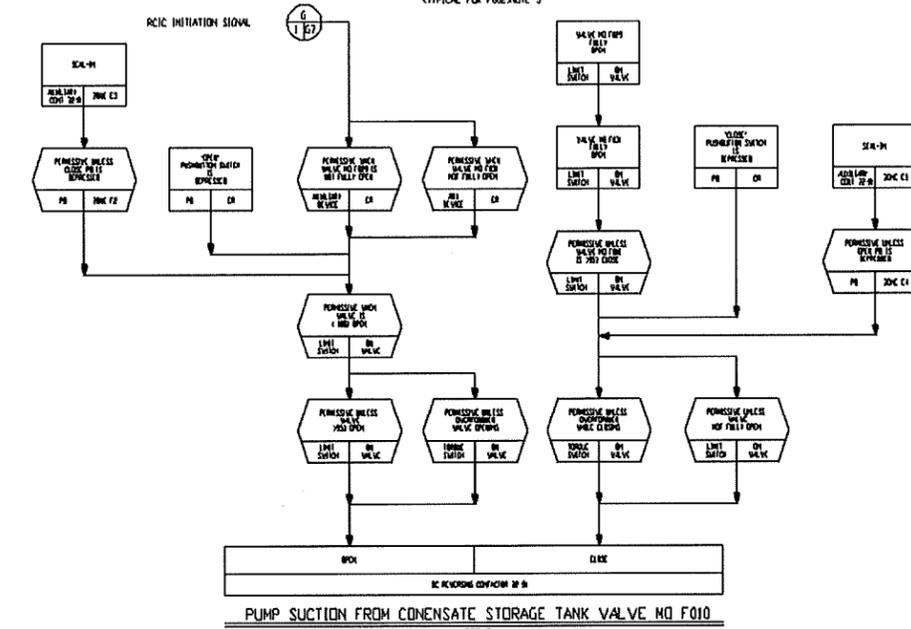
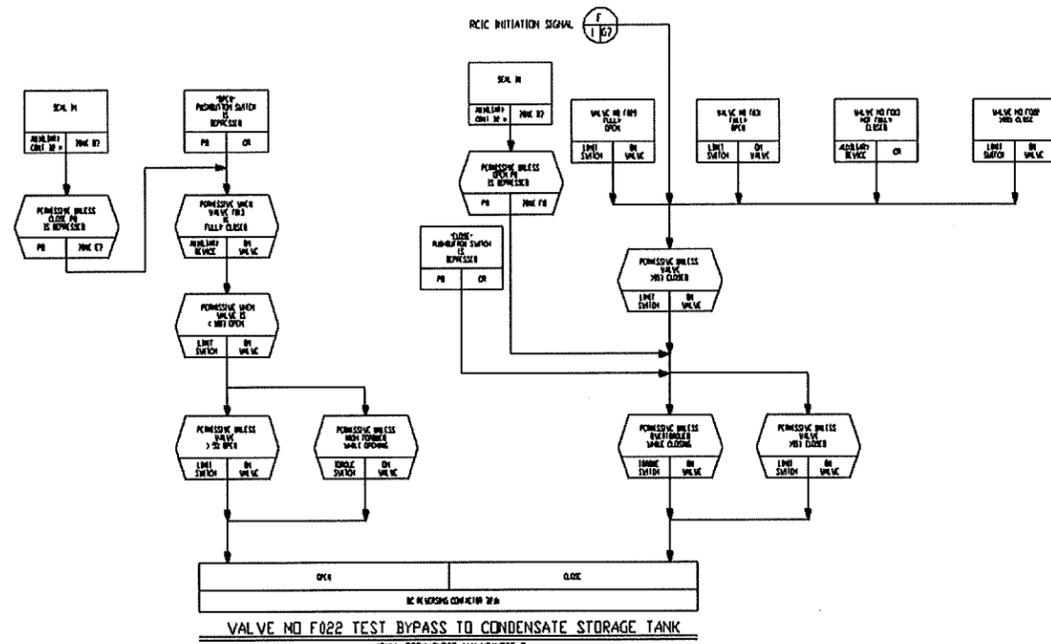
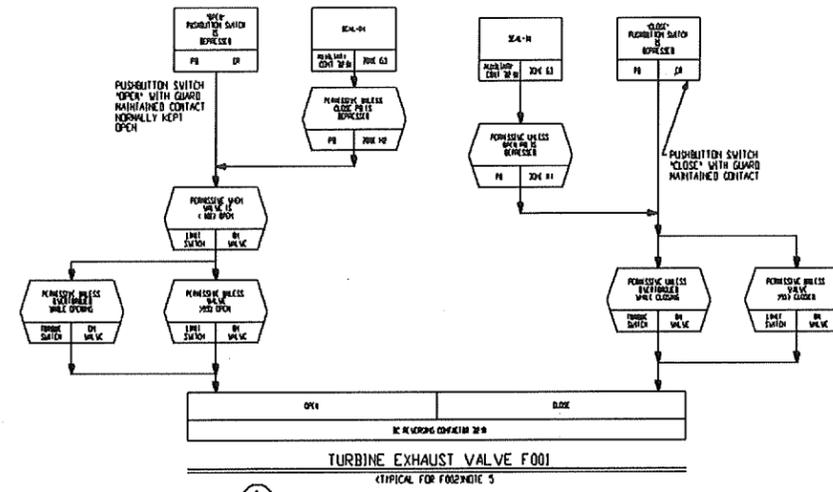
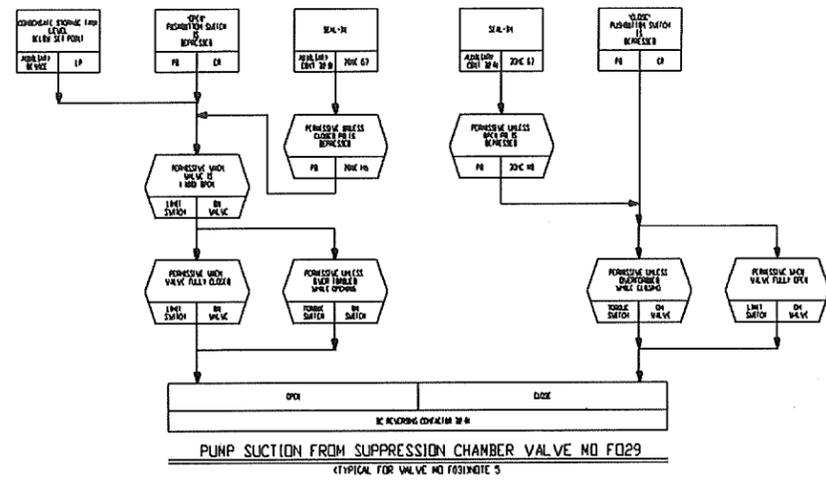
REACTOR CORE ISOLATION COOLING SYSTEM FCI



NOTE: SEE 61721-2230-1 (FIGURE 7.4-1, SHEET 1) FOR NOTES, REFERENCES AND LEGEND.

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FIGURE 7.4-1, SHEET 2
 REACTOR CORE ISOLATION COOLING SYSTEM FCI



NOTES: 1. SEE 1-2230-01 FOR NOTES, REFERENCES AND LEGEND.
2. ALL VALVE NUMBERS ARE PREFIXED CS/SA UNLESS OTHERWISE SHOWN

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FIGURE 7.4-1, SHEET 4
REACTOR CORE ISOLATION COOLING SYSTEM FCD

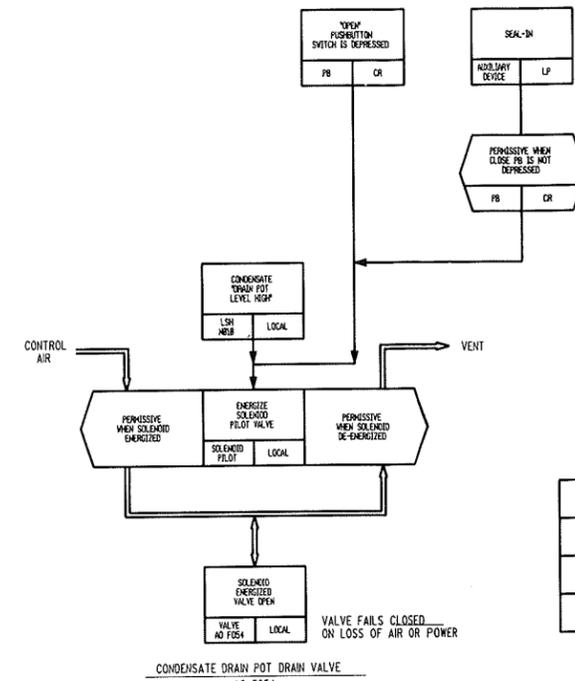
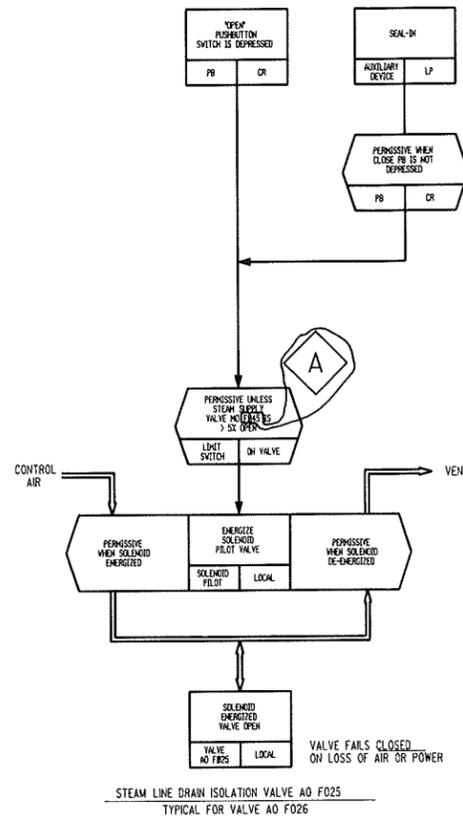
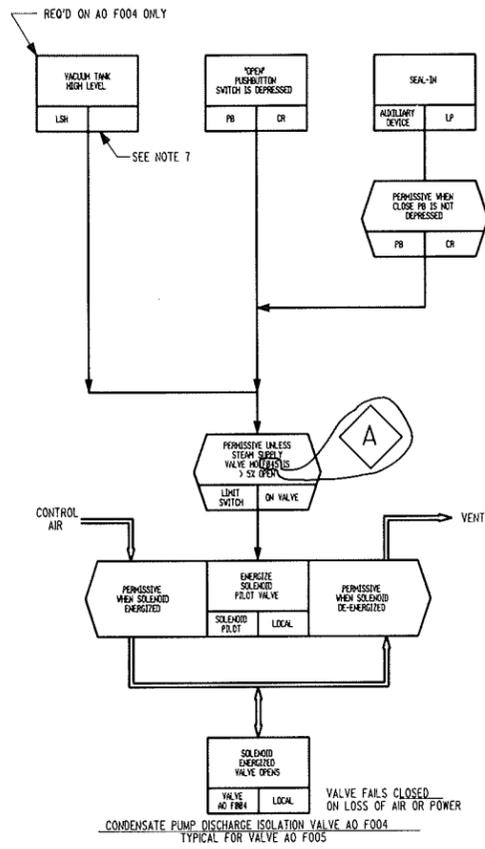


TABLE 1

ALARM FUNCTION	INITIATING DEVICE	DEVICE LOCATION	TYPE OF ALARM
TURBINE BRG OIL PRESSURE LOW	P.S.	SEE NOTE 7	LOW
VACUUM TANK PRESSURE HIGH	PSH	SEE NOTE 7	HIGH
RCIC FLOW LOW	FSL NO02	LP	LOW

TURBINE CONDITIONAL SUPERVISORY ALARMS

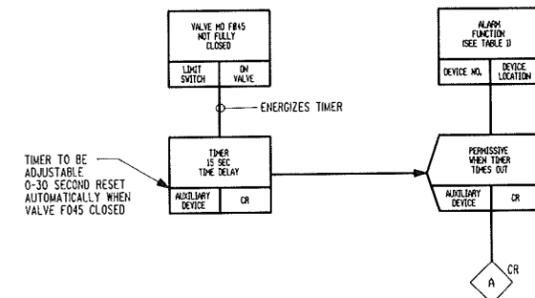
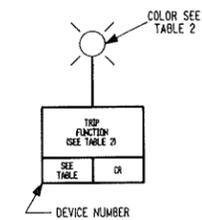
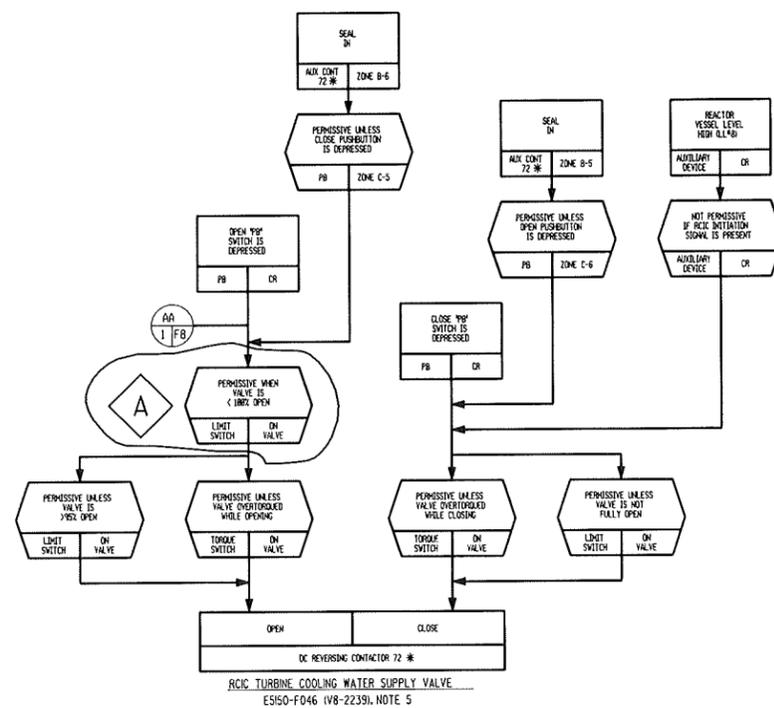


TABLE 2

ACTU. DEVICE	TRIP FUNCTION	COLOR	PNL LOCATION
REF. B	TRIP THROTTLE VALVE OPEN	RED	CR
	TRIP THROTTLE VALVE CLOSED	GREEN	CR
	GOVERNOR VALVE OPEN	RED	CR
	GOVERNOR VALVE CLOSED	GREEN	CR

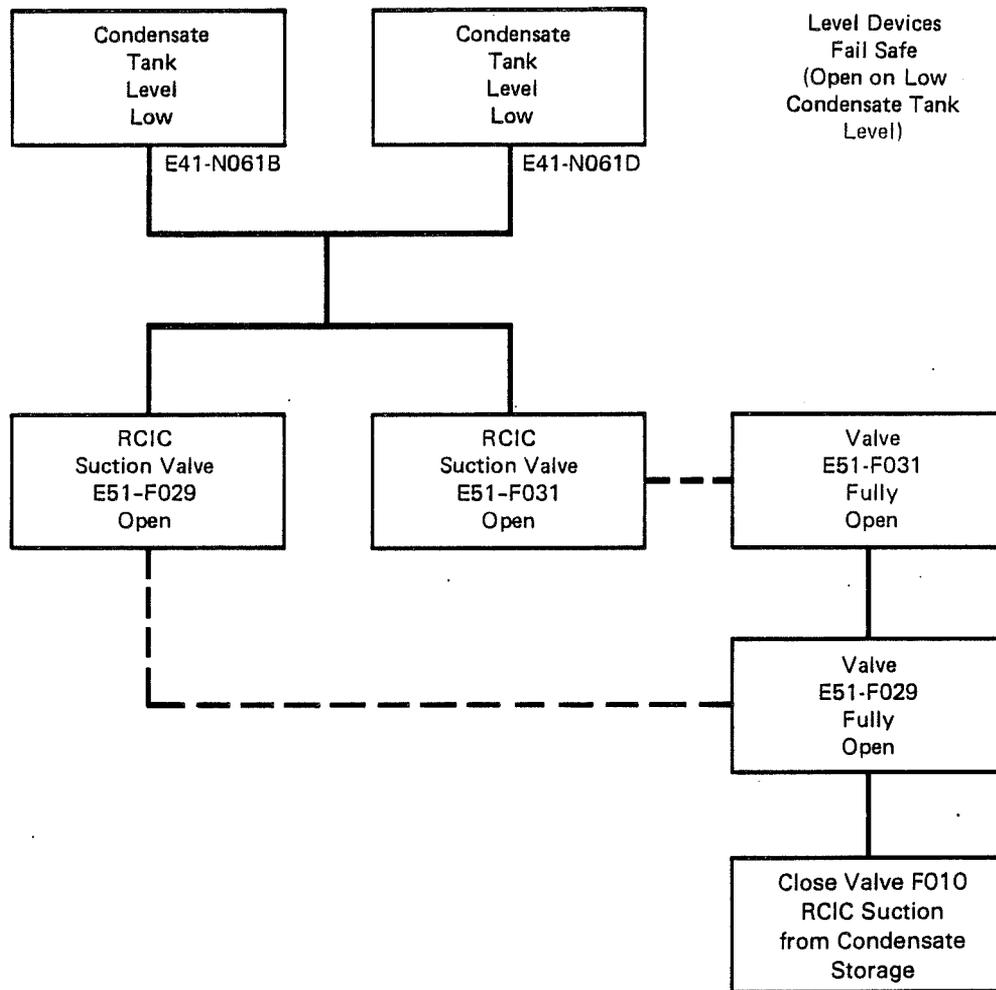
TURBINE INDICATING LIGHTS



NOTE: SEE 61721-2230-1 (FIGURE 7.4-1, SHEET 1) FOR NOTES, REFERENCES AND LEGEND.

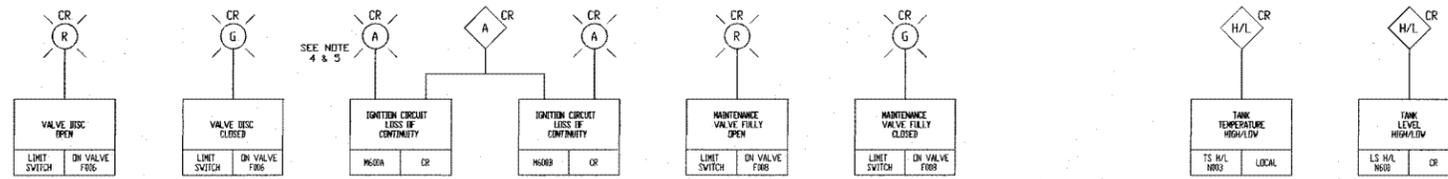
Fermi 2
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FIGURE 7.4-1, SHEET 5
REACTOR CORE ISOLATION COOLING SYSTEM FCD

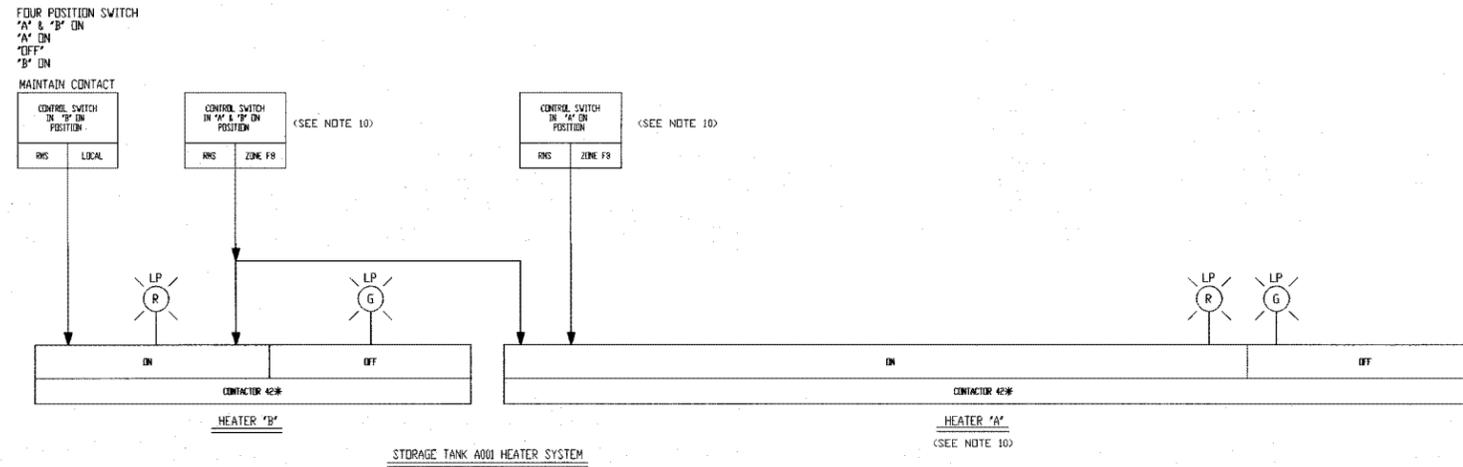


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FIGURE 7.4-2
 SIMPLIFIED LOGIC DIAGRAM OF REACTOR CORE ISOLATION COOLING AUTOMATIC SUCTION SWITCHOVER



INDICATOR LIGHTS & ALARMS



STORAGE TANK A001 HEATER SYSTEM

NOTES:

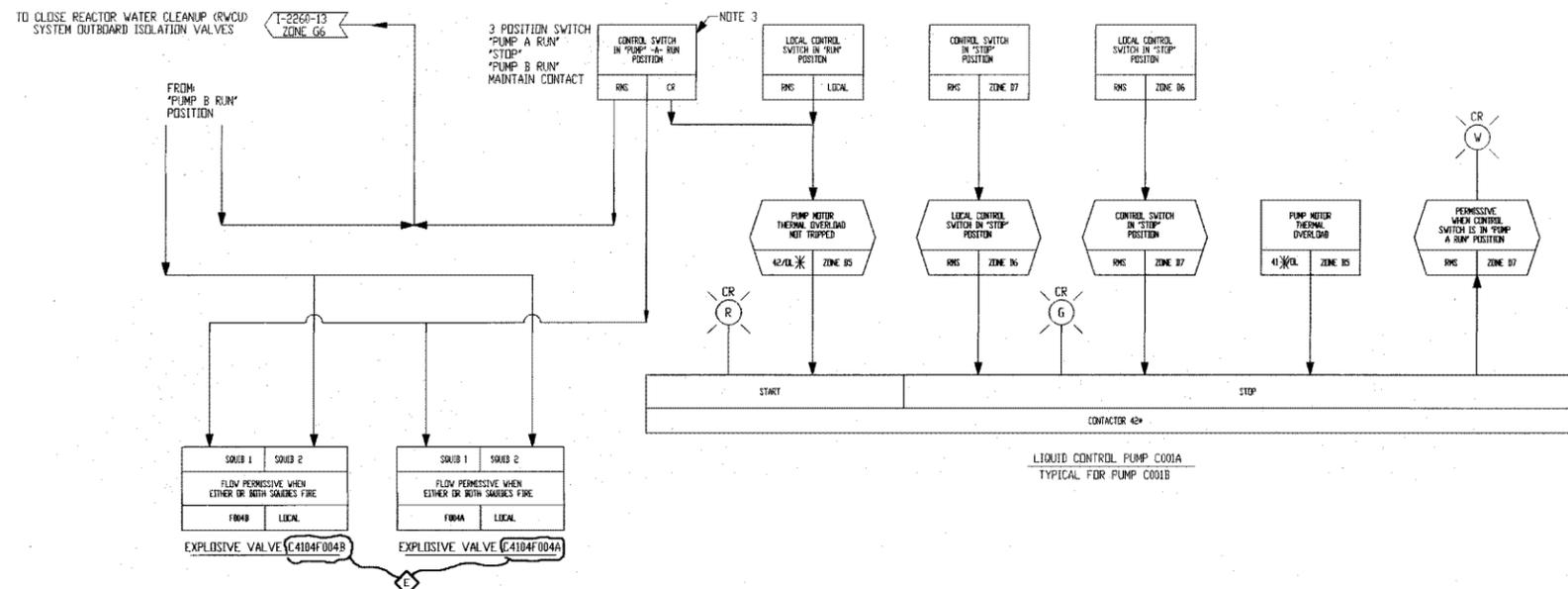
1. REMOVED
2. REMOVED
3. RMS CONTROL SWITCH IS KEYLOCK IN 'STOP' POSITION. MAINTAIN CONTACT IN 'RUN' POSITIONS. SWITCH BYPASSES CONTINUITY MONITOR IN 'RUN' POSITIONS TO FIRE VALVE SQUIBS OF BOTH VALVES (C4104F004A & C4104F004B) E
4. CONTINUITY INDICATING LIGHT IS ON WHEN IGNITION CIRCUIT HAS CONTINUITY THROUGH SQUIB CIRCUITS. ANNUNCIATOR TRIPS AND LIGHT GOES OUT UPON LOSS OF CONTINUITY IN IGNITION CIRCUIT.
5. CONTINUITY MONITOR CIRCUIT SHALL BE CURRENT LIMITED AND WIRED TO PREVENT ACCIDENTAL FIRING OF EXPLOSIVE VALVES.
6. REMOVED
7. AUXILIARY RELAYS AND DEVICES NOT SHOWN ON THE LOGIC DIAGRAM.
8. REMOVED
9. REMOVED
10. HEATER 'A' (C4101S001) IS ABANDONED IN PLACE PER TSR-30753.

LEGEND

* = SWITCHGEAR DEVICE FUNCTION NUMBER ANSI SPEC. C37.2

REFERENCE DOCUMENTS

- | | |
|---|----------------|
| 1. STANDBY LIQUID CONTROL SYSTEM P&ID..... | M-2082 |
| 2. REACTOR WATER CLEANUP SYSTEM LOGIC DIAGRAM.... | I-2260-13 |
| 3. LOGIC SYMBOLS..... | 209A4756;R1-20 |
| 4. STANDBY LIQUID CONT SYSTEM DESIGN SPEC..... | 22A1482;R1-167 |



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FIGURE 7.4-3
STANDBY LIQUID CONTROL SYSTEM FCD

7.5 SAFETY-RELATED AND POWER GENERATION DISPLAY INSTRUMENTATION

7.5.1 Description

7.5.1.1 General

A description of the instrumentation that provides information to the operator to enable him to perform required safety functions is provided in this subsection.

A Human Factors Engineering design review program was established to maintain control room and remote shutdown panel instrumentation in conformance with the general human factors conventions adapted from NRC Human Factors Criteria (NUREG-0700 "Guidelines for Control Room Design Reviews") as well as plant specific conventions. This program was originally described in section 5.0 of Supplement 2 to the DCRDR "Summary Report for the Fermi 2 Control Room."

7.5.1.2 Normal Operation

The normal plant process variable indicators and recorders are described in Section 7.6 and are shown on the piping and instrumentation diagrams for the various nuclear steam supply systems (NSSSs). Information channel ranges and indicators are selected on the basis of giving the operator the necessary information, during expected operational perturbations, to perform all the normal plant maneuvers and to be able to track all the process variables pertinent to safety. Description of the control rod position indicating system is given in Subsection 7.7.1.1.5.

7.5.1.3 Abnormal Transient Occurrences

The ranges of indicators and recorders provided are capable of covering the extremes of process variables and providing necessary information to enable the operator to perform required safety functions.

7.5.1.4 Accident Conditions

Information readouts are provided to accommodate events up to and including a LOCA. These readouts are designed from the standpoint of operator action, information, and event tracking requirements, providing assurance that requirements for all other credible events or incidents will be covered.

7.5.1.4.1 Initial Accident Event

The design basis of all engineered safety feature (ESF) systems to mitigate accident event conditions takes into consideration that no operator action or assistance is necessary for the first 10 minutes of the event. This requirement makes it mandatory that all protective action necessary in the first 10 minutes be automatic. Therefore, although continuous tracking of process variables is available, no operator action based on them is required or recommended.

7.5.1.4.2 Postaccident Tracking

After 10 minutes, operator action is optional, based on the information available. Within 30 minutes, however, containment cooling must be initiated.

The process instrumentation described in the following subsections provides information to the operator for his use in monitoring reactor conditions after a LOCA.

This instrumentation was designed to conform to the requirements of Regulatory Guide 1.97 [formerly, Branch Technical Position (BTP) ICSB-23]. A formal type-test seismic qualification based on IEEE Standard 344-1971 was obtained for the strip-chart recorders used in the systems.

Details of the tests performed are available in the General Electric Seismic Summary Report for panel H11-P602, dated December 1977.

7.5.1.4.2.1 Reactor Water Level

Appropriate vessel water level instrumentation described below is operable during and after postulated design-basis accidents.

The emergency core cooling system (ECCS) equipment and reactor protection and containment isolation system initiation is automatic. In addition, 11 level indicators are located in the control room. These instrument designations, their ranges, and control room location are presented in Table 7.5-1. Their ranges vary to cover the active fuel to the top of the reactor vessel so that the required range of the reactor vessel water level is monitored.

Two wide-range water level signals are transmitted from two nuclear boiler system independent differential pressure transmitters and are recorded on two, multi-point recorders in the main control room. One point records the wide-range level and the other point records the reactor pressure on each of the two recorders. The differential pressure transmitters have one side connected to a condensing-type chamber reference leg and the other side connected directly to a vessel nozzle for the variable leg. The water level system is uncompensated for variation in reactor water density and is calibrated to be most accurate at operational pressure and temperature conditions. The range of the recorded level is from the top of the feedwater control range (just above the high level turbine trip point) down to a point near the top of the active fuel. The power sources for the two channels are inverter-fed from the two divisional batteries. Both pressure and level recorders are equipped to automatically switch from a low (normal) speed to high speed when signal levels reach preset values as shown in Figure 7.3-12.

Two fuel-zone water level signals are transmitted from two nuclear boiler system independent differential-pressure transmitters. Signals go to water level recorders (programmable), one in each division. The differential-pressure transmitters have one side connected to a condensing-type chamber reference leg and the other side connected directly to the bottom tap of a calibrated jet pump for the variable leg. The water level system is uncompensated for variation in reactor water density and is calibrated to be most accurate at saturated atmospheric conditions. The programmable recorders perform mathematical conversion of the fuel-zone water level measurements to readings which account for the difference between calibration and off-calibration conditions expected during the accident.

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The recorders use reactor vessel pressure signals to calculate the signal conversions. The level range is from near the bottom of the active fuel to over the top of the active fuel, as shown in Figure 7.3-12. The ranges of the wide-range level and the fuel-zone level overlap. Power sources are as stated in the previous paragraph. The feedwater control system has other reactor water level recorders/indicators in the main control room.

Fermi 2 is a BWR/4 design that uses only nonheated reference columns that are maintained full of condensate by the condensing chambers for all of the level measurements. A backfill system is installed on each level instrument reference leg. The system provides a metered flow of water from the control rod drive system to each leg. The flow is low enough to not affect the performance of the instrumentation. The backfill is designed to prevent the accumulation of dissolved noncondensable gases in the reference legs. Yarway reference columns are not used. Reactor vessel taps are divided into two separate divisions to maintain spatial diversity. Fermi 2 has three upper-steam-space taps (one on the head vent line), similar to the BWR/3 design. There are two intermediate (water zone) taps that sense the water variable leg inside the vessel annulus approximately 13 ft above the top of the active fuel. There are two lower taps that sense the water variable leg inside the vessel annulus outside the core shroud at approximately 10 in. above the top of the active fuel. Additionally, two lower taps that sense the water variable leg inside the jet pump above the pump diffuser tap similar to the BWR/3 are provided.

The power for the feedwater system level instrumentation is supplied from a vital instrument bus. Power is supplied to the reactor protection system (RPS) trip system level instruments from the RPS motor-generator sets. Power for the balance of the level instrumentation that is part of the ECCS is supplied by safety-grade inverter power supplies powered by the appropriate divisional battery.

With respect to drywell sensing line routing, Fermi 2 meets the requirements outlined in Figure 2.3.2.2-8 of NEDO 24708A, and, therefore, the level instruments are relatively independent of drywell temperature changes.

All of the level sensors are located on spatially separated divisional safety-grade instrument racks located in the reactor building approximately 15 ft from the drywell wall. All the level instrument channel response times are well within the design criteria.

In response to NRC requests for additional information on water level indication errors, Edison has provided information to further demonstrate the adequacy of the Fermi 2 water-level instrument design in response to high drywell temperatures that may lead to reference-leg flashing. In the unlikely case that flashing occurred, the expected error would be about 4 in. of indicated level. Even if the entire reference-leg portion in the drywell boiled off, the hypothetical error would not seriously impact adversely either manual or automatic actions to safely mitigate the worst-case transient identified.

7.5.1.4.2.2 Reactor Pressure

Two reactor pressure signals are transmitted from two independent pressure transmitters and are recorded on two, multi-point recorders in the main control room (same recorders described in Subsection 7.5.1.4.2.1 above). One point records pressure; the other records the wide-range level. The range of recorded pressure is from 0 to 1500 psig. Additionally, fuel-zone water level recorders (programmable) use reactor pressure signals to correct water level

measurements for off-calibration pressure. The feedwater control system has other pressure signals recorded in the main control room. This range is sufficient to include the safety limit pressure.

7.5.1.4.2.3 Shutdown, Isolation, and Core Cooling Indication

The following information furnished to the main control room operator permits him to assess reactor shutdown, isolation, and availability of emergency core cooling following the postulated accidents.

- a. Reactor shutdown occurs as one or more process variables exceed their specified setpoint. Operator verification that shutdown has occurred may be made by observing one or more of the following indications:
 1. Control rod status lamps indicating each rod fully inserted. Power source is a battery-powered inverter supply (Figure 7.5-1)
 2. Control rod scram valve status lamps indicating open valves. Power source is a battery-powered inverter supply (Figure 7.5-1)
 3. Neutron-monitoring power-range channels and recorders downscale. Power sources are the RPS motor-generator sets (Figure 7.5-1) and battery-powered inverters
 4. Annunciators for RPS variables and trip logic in the tripped state. Power source is dc from station battery (Figure 7.5-1)
 5. Logging of control rod positions on the Integrated Plant Computer System (IPCS). Power source is computer power supply from uninterruptible power supply (UPS) A and B
 6. Events recorder logging of trips. Power source is from station battery.
- b. Reactor isolation occurs after the accident as various environmental and process variables exceed their setpoints. The operator may verify reactor isolation by observing one or more of the following indications:
 1. Isolation valve position lamps indicating valve closure. Power source is the same as for the associated motor operator
 2. Main steam line flow indication downscale. Power source is a battery-powered inverter supply (Figure 7.5-1)
 3. Annunciators for the primary RPS variables and trip logic in the tripped state. Power source is dc from station battery
 4. Events recorder logging of trips.
- c. Operation of emergency core cooling following the LOCA may be verified by observing the following instrumentation:

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1. Annunciators for high pressure coolant injection (HPCI), core spray, residual heat removal (RHR), and automatic depressurization system (ADS) sensor initiation logic trips. Power source is dc from a station battery
 2. HPCI pump discharge pressure and flow indicators (Figure 7.3-1). Power for these instruments is derived from a battery-operated inverter supply
 3. ADS valve position status (Figure 7.3-4). Position-indicator power is derived from battery powered inverters
 4. Divisional core spray discharge pressure indicators, loop flow indicators, and pump motor ammeters (Figures 7.3-7 and 7.3-8). Power for the respective instruments is derived from the appropriate ESF bus
 5. Divisional low pressure coolant injection (LPCI) (RHR) pump discharge header pressure indicators, loop flow indicators, loop flow recorders, and pump motor ammeters (Figure 7.3-9). Power for these instruments is derived from the associated ESF bus
 6. Divisional RHR service water loop flow indicators and outlet temperature recorders (Figure 7.3-9). Power for these instruments is derived from the associated ESF bus
 7. Injection valve position status. Power source is the same as for the valve motor
 8. Events recorder logging of trips in the emergency core cooling network. Power source is the 130-V dc power system
 9. Relief valve discharge pipe temperature monitors. Power source is 120-V ac supplied from a 120-V ac instrument bus.
- d. Conditions of significant timed interlocks that restrict the flexibility of the ESF systems are indicated by the following devices:
1. An indicating timer, which displays the amount of time remaining until manual ADS activation is permissive. This device is powered from the ADS logic supply
 2. Four timers, which indicate that the required time has elapsed and the RHR (LPCI) "failed" loop injection valves can be reopened. These devices are powered from their respective divisional logic supply
 3. Two timers, which indicate that the required time has elapsed and the RHR (LPCI) injection valves can be closed manually. These devices are powered from their respective divisional logic supply.

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- e. The following indicating devices are provided to permit rapid assessment of the standby power system status and to enhance the manual reconnection of loads:
 1. Each of the emergency diesel generator (EDG) automatic load sequencer systems provides visual indication of the exact state of the automatically sequenced loads. These devices are powered by the sequencer system power supply. Specific details of this system are described in Subsection 8.3.1.1.7.
 2. A digital meter that displays remaining generator capacity for each EDG is provided. Power is supplied to these instruments from the respective EDG bus. This system is described more fully in Subsection 8.3.1.1.11.

7.5.1.4.2.4 Primary Containment Indication

The following systems provide the control room operator with necessary information regarding the full range of environmental conditions possible within the primary containment following an accident or incident.

- a. Drywell temperature in various locations within the drywell volume is recorded on redundant strip-chart recorders (Subsection 7.6.1.12.2). The power supply for these recorders is derived from the respective ESF bus
- b. Drywell pressure is monitored by both narrow-range and wide-range pressure transmitters and recorded on redundant strip-chart recorders (Subsection 7.6.1.12.3). The power supply for these devices is derived from battery-powered inverters
- c. Torus temperature is monitored by thermocouples located in both the torus air space and the suppression pool water. Continuous strip-chart recording of these temperatures is provided on redundant recorders (Subsection 7.6.1.12.2). The recorders are supplied with power from the appropriate ESF bus
- d. Torus pressure is monitored by both narrow-range and wide-range pressure transmitters and recorded on redundant strip-chart recorders (Subsection 7.6.1.12.3). The power supply for these devices is derived from battery-powered inverters
- e. Suppression pool water level is continuously recorded on redundant recorders (Subsection 7.6.1.12.4). Power for this instrumentation is supplied by the battery-powered inverters
- f. Radiation level within the drywell is monitored and recorded (Section 11.4). The power for this instrumentation is derived from a 120-V ac instrument bus.

7.5.1.4.2.5 Bypassed and Inoperable Status Indication for Nuclear Safety Systems

In addition to administrative procedures for determining and indicating bypassed or inoperable status of systems, channel bypassed and inoperable status indication is provided for the RPS and ESF systems in accordance with the requirements of IEEE 279-1971, Section 4.13. As stated in Sections 7.1, 7.2 and 7.3, the RPS and ESF systems comply with

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IEEE 279-1971, as supported by NEDO-10139, which has been incorporated into the UFSAR by reference. Refer to Sections 7.2 and 7.3 for descriptions of control room indication related to specific RPS and ESF functions.

7.5.1.4.3 Safety Parameter Display System (SPDS)

SPDS is a function of the Integrated Plant Computer System (IPCS) that provides a specific selection of emergency response information. SPDS uses data from selected plant data systems and processes the data for display on the IPCS. SPDS information can be displayed on any IPCS terminal, which includes those specifically located in the control room, the technical support center (TSC), and the emergency operations facility (EOF). The SPDS display in the control room is provided to assist the operators in assessing the safety status of the plant following an accident. The IPCS and SPDS are described in subsections 7.6.1.9.1 and 7.6.1.9.1.2.5.1 respectively.

7.5.1.5 Special Condition: Loss of Habitability of Main Control Room

7.5.1.5.1 Criteria

It is necessary to be able to carry out the reactor shutdown functions from outside the main control room and to bring the reactor to cold condition in an orderly fashion in compliance with General Design Criterion 19 of 10 CFR 50, Appendix A. This requirement applies when the main control room becomes uninhabitable for any reason and is accomplished using the remote shutdown panel discussed in this section.

Appendix R to 10 CFR 50 requires that the plant be safely shut down remote from the control room in the event of a fire in the main control room, relay room, cable spreading room, and other areas containing equipment or cabling of both divisions required for safe shutdown. This capability is provided by the alternative shutdown system described in Subsection 7.5.2.5.

7.5.1.5.2 Remote Shutdown Panel

The remote shutdown panel is located in the Division I switchgear room on the second floor of the reactor building. At this location it cannot be damaged by failure of any other equipment. The remote shutdown system panel is designed to comply with the requirements of Quality Assurance Level I, Seismic Category I. The following systems have instrumentation and controls on the remote panel, as shown in Figures 7.5-2 and 7.5-3.

- a. Reactor core isolation cooling (RCIC) system
- b. RHR system
- c. Recirculation flow control system
- d. Nuclear boiler system
- e. Control rod drive (CRD) system
- f. Residual heat removal service water (RHRSW) system
- g. Primary containment monitoring system.

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7.5.1.5.3 Conditions Assumed To Exist As the Main Control Room Becomes Inaccessible

- a. The plant is operating normally at, or less than, design power
- b. Loss of offsite ac power is considered unlikely, but credible. For shutdown outside the control room coincidental with the loss of offsite ac power, the instrumentation and controls of the alternative shutdown system, e.g., dedicated shutdown panel and its associated procedures, will be used as described in UFSAR Section 7.5.2.5.2.
- c. No LOCA or transients shall be assumed; therefore, complete control of ESF systems from outside the main control room will not be required
- d. Plant personnel evacuate the main control room
- e. The main control room continues to be inaccessible during the entire shutdown procedure
- f. The event that causes the main control room to become inaccessible is assumed to be such that the operator can manually scram the reactor before leaving the main control room. As a backup, the operator can manually scram the reactor, and if necessary, close the MSIVs from outside the Main Control Room.
- g. The main turbine pressure regulators may be controlling reactor pressure via the bypass valves; however, in the interest of simplicity and safety, it is assumed that this function is lost. Therefore, main steam line isolation is assumed to occur, and reactor pressure is relieved through the relief valves to the suppression pool. The feedwater control system is also assumed to be unavailable due to reactor isolation
- h. Reactor water is made up by the RCIC system
- i. The dc services are supplied from at least one plant dc power system for each essential system or equipment item in the remote shutdown system.

7.5.1.5.4 Description

The system provides remote control for the reactor systems needed to carry out the shutdown function from outside the main control room and to bring the reactor to cold condition in an orderly fashion. This system also provides a variation to the normal system used in the main control room, permitting the shutdown of the reactor when feedwater is unavailable and the normal heat sinks (turbine and condenser) are lost.

Automatic activation of relief valves and the RCIC system brings the reactor to a hot shutdown condition. During this phase of shutdown, the suppression pool is cooled by operating the RHR system in the suppression pool cooling mode. Reactor pressure is controlled and core decay and sensible heat are rejected to the suppression pool by relieving steam pressure through the relief valves. Reactor water inventory is maintained by the RCIC system.

Manual operation of the relief valves cools the reactor and reduces its pressure at a controlled rate until reactor pressure becomes so low that the RCIC system discontinues operation. This

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condition is reached at 50 to 100 psig reactor pressure. The RHR system is then operated in the shutdown cooling mode wherein the RHR system heat exchanger is connected directly into the reactor water circuit to bring the reactor to the cold low-pressure condition.

7.5.1.5.5 Procedure for Reactor Shutdown From Outside the Main Control Room

- a. If evacuation becomes necessary, the operator will scram the reactor by the manual scram switches or reactor mode switch at the main control room panel as he leaves the main control room
- b. The main turbine pressure regulator will, under normal conditions, control the reactor pressure while rejecting heat (steam) through the turbine bypass valves. The feedwater control system will control water level
- c. As a backup, the operator can manually scram the reactor and close the MSIVs from outside the Main Control Room
- d. The remainder of the procedure as described assumes that the automatic pressure regulator is not available from time zero and the main steam isolation valves (MSIVs) are closed, but the actual procedure may be written to utilize any plant equipment that is available as long as this worse case condition is provided for
- e. Key-controlled transfer switches at the remote panel are operated to transfer control to the remote shutdown panel
- f. Relief valves open automatically and cycle to control reactor pressure. Reactor level starts to drop at a rate depending on prior power level and elapsed time from scram
- g. The operator starts the RCIC system manually before Level 2 (Figure 7.3-12) is reached and monitors the water level thereafter. The water level will continue to fall
- h. One relief valve opens and closes automatically, by the Low - Low Set Function
- i. Reactor level reaches its lowest point at about 80 in. above top of active fuel if the RCIC system was initiated at low level. Level starts to rise as a result of RCIC system flow. Pressure relief is through one relief valve in automatic intermittent operation
- j. Water level is returned to normal by operation of the RCIC system
- k. One relief valve is still in automatic intermittent operation. The RCIC system turbine automatically shuts down when Level 8 is reached. It starts automatically when the level drops to the initiation level
- l. Reduction of reactor pressure is started by manually actuating one relief valve. While activating relief valves, the operator observes the reactor level and suppression pool temperature

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- m. Relief valves are closed before level drops below Level 3. The reactor cooldown rate will be controlled to not exceed 100°F per hour. Reactor level varies between Level 3 and Level 8
- n. The RHR system with one pump, one heat exchanger, and the RHRSW system is used to cool the suppression pool
- o. The operator activates two relief valves to maintain reduction of pressure to 250 psig while observing pool temperature, which is not to exceed 140°F unless the reactor pressure decreases to less than 250 psig
- p. Reactor pressure is reduced to 100 psig, allowing the suppression pool temperature to reach 170°F if necessary
- q. The RHR system is placed in the shutdown cooling mode. The RHR operation continues until the reactor is in the cold/low-pressure condition
- r. Normal reactor water level is maintained after being placed in the shutdown cooling mode.

7.5.2 Analysis

7.5.2.1 General

The safety-related and power-generation display instrumentation provides adequate information to allow operators to make correct decisions as bases for manual control actions permitted under normal, abnormal transient, and accident conditions.

Information instrumentation having no direct input to ESF systems, except through the operator as a link, is considered to be outside the scope of existing IEEE Standards. However, insofar as practical, instruments are selected from those types qualified under IEEE 279-1971 and IEEE 323-1971. Redundancy and independence or diversity are provided in all of the information systems used for the basis of operator-controlled safeguards action.

This instrumentation is designed to operate during normal operation, accident, and postaccident environmental conditions. The design criteria that the instrumentation must meet are discussed more fully in Subsection 7.1.2. The specific design qualifications of the instruments referenced in this section are tabulated in Table 7.5-2.

7.5.2.2 Normal Operation

Subsection 7.5.1.2 describes the basis for selecting ranges for instrumentation and, inasmuch as monitoring requirements for abnormal transient or accident conditions exceed those for normal operation, the normal ranges are covered adequately.

7.5.2.3 Abnormal Occurrences

These occurrences will result in conditions lesser in consequence than those defined to be accident conditions in Subsection 7.5.2.4. Proper accident tracking, therefore, qualifies abnormal occurrence tracking.

7.5.2.4 Accident Conditions

The LOCA is the most extreme operational event. Information readouts are designed to accommodate this event from the standpoint of operator action, information, and event tracking requirements, and therefore cover all other design-basis events or incidents requirements.

7.5.2.4.1 Initial Accident Event

The design bases of all ESF systems to mitigate accident event conditions take into consideration that no operator action or assistance is required or recommended for the first 10 minutes of the event. This requirement makes it mandatory that all protective action necessary in the first 10 minutes be automatic. Therefore, although continuous tracking of variables is available, no operator action based upon them is intended.

7.5.2.4.2 Postaccident Tracking

After 10 minutes, operator action is optional. The following process instrumentation provides information to the operator after a design-basis accident (DBA) for his use in monitoring reactor conditions within the primary containment.

7.5.2.4.2.1 Reactor Water Level

Vessel water-level instrumentation described in Subsection 7.5.1.4.2.1 is redundant, electrically independent, and is operable following all credible accident events. Subsection 15.6.2 discusses the postulated instrument line break scenarios. This instrumentation complies with independence and redundancy requirements of IEEE 279-1971.

7.5.2.4.2.2 Reactor Pressure

Pressure instrumentation described in Subsection 7.5.1.4.2.2 is redundant, electrically independent, and is operable following all credible accident events. This instrumentation complies with the independence and redundancy requirements of IEEE 279-1971.

7.5.2.4.2.3 Shutdown, Isolation, and Core Cooling Indication

This information instrumentation will have no direct input to the ESF systems and is considered to be outside the scope of existing 36IEEE Standards. However, insofar as practical, instruments will be selected from those types that are qualified under IEEE-279 and IEEE-323. Redundancy and independence or diversity will be provided in all systems which are used for operator control and ESF status information.

7.5.2.4.2.4 Primary Containment Indication

Primary containment instrumentation described in Subsection 7.5.1.4.2.4 is designed to be redundant, electrically independent, and remain operable following all credible accident events. The ranges have been selected to cover the design conditions of the containment.

7.5.2.4.3 Safety Parameter Display System

See Subsection 7.6.1.9.1.2.5.1 for a discussion of the design analysis of the SPDS.

7.5.2.5 Special Condition: Post-Fire Reactor Shutdown From Outside the Main Control Room

Analysis of reactor shutdown from outside the main control room is included in Subsection 7.5.1.5 for non-fire scenarios requiring control room evacuation. This section discusses the alternative shutdown system, which includes the dedicated shutdown panel, used for post-fire shutdown for fires in the main control room and selected other areas.

7.5.2.5.1 Design Bases

The alternative shutdown system was designed and installed to meet the requirements of 10 CFR 50, Appendix R, Section III, paragraphs G and L. The alternative shutdown system was designed to provide safe-shutdown capability separate and remote from the control center complex (control room, relay room, and cable spreading room, fire zones 03AB, 07AB and 09AB) and other selected auxiliary building fire zones 08AB, 11AB and 13AB when a fire in the complex or these zones is assumed to significantly damage the equipment/cabling in these zones. In the context of the discussion of alternative shutdown design basis, these six fire zones are referred to as the dedicated shutdown areas of concern. The above fire zones are described in UFSAR 9A.4. These zones are not the Fire Detection Zone numbers. UFSAR Figure 9A-1 provides a descriptive table cross-referencing the UFSAR Fire Zones and the Fire Detection Zones used in the abnormal operating procedure.

The objectives of the alternative shutdown system are to

- a. Achieve and maintain subcritical reactivity conditions in the reactor
- b. Maintain reactor coolant inventory
- c. Achieve and maintain hot shutdown
- d. Achieve cold shutdown conditions within 72 hr
- e. Maintain cold shutdown conditions thereafter.

The reactor is shut down and maintained subcritical by control rod insertion. The portions of the CRD system necessary for reactor scram are designed to fail safely (actuate) if subjected to a fire. The core is kept covered by establishing standby feedwater flow to make up for loss of reactor vessel water inventory. Hot shutdown is achieved and maintained by establishing primary containment cooling and torus cooling. The primary containment fan and cooling unit operation (Subsection 9.4.5) and the torus cooling mode of the system (Subsection 5.5.7) are established prior to exceeding established drywell temperature and suppression pool water temperature design limits, respectively. Cold shutdown is achieved by the shutdown cooling mode of the RHR system.

The alternative shutdown system provides a dedicated shutdown panel located in the radwaste building, second floor*, from which an operator can monitor the reactor and keep the reactor core covered with water. The system design uses appropriate systems already

installed, with installation of the panel and necessary control and transfer switches to make it functional.

NOTE: * In past correspondence with the NRC, the dedicated shutdown panel has also been called the 3L panel because it satisfies Paragraph III.L, of Appendix R to 10 CFR 50.

7.5.2.5.2 System Description

The alternative shutdown system consists of one of the four combustion turbine generators (CTGs), the standby feedwater (SBFW) system, a dedicated shutdown control panel and associated instrumentation, a Distributed Control System (DCS), and Division I portions of the following systems: RHR, RHRSW, emergency equipment cooling water (EECW), and emergency equipment service water (EESW). The dedicated shutdown panel is supplemented by local manual operator actions to achieve hot or cold shutdown.

The four CTGs (Subsection 8.2.1.2) are oil-fired turbine generators located onsite, remote from the fire areas of concern. The CTG 11-1 is used to provide emergency power when a fire occurs in the fire areas of concern, or on loss of offsite power should the EDGs be unavailable. CTG 11-1 has black start capability. The CTG starting diesel is located in an enclosed heated compartment and is equipped with a float tank, which provides an initial supply of warm fuel oil, to ensure its operability. Diesel fuel is maintained in the CTG Fuel Oil Tank with a fuel level maintained by plant procedures to ensure nominal fuel availability for 72 hours of operation for a single CTG unit at 10 MW load.

If CTG 11-1 is not available, either CTG 11-2, CTG 11-3 or CTG 11-4 (with AC starting motors) can be established on a standby basis as the black start power source for alternative shutdown power source using a standby starting diesel generator. Cold weather equipment preparation is addressed within the System Operating Procedure during the cold weather season.

The CTGs' control, instrumentation, and power cabling is located, isolated, and/or routed independent from the fire areas of concern, except for the CTG supervisory control circuit, which has transfer and lockout features that ensure that it is isolated from control room CTG circuitry.

Control of the breakers in the Fermi 120-kV switchyard and control of the CTGs is via a supervisory system. The essential elements of the system consist of a Distributed Control System panel (with local I/O) located at the 120-kV switchyard/SBO DG area, fiber optic communication lines, and actuating devices.

The Distributed Control System (DCS) functions as follows: field devices interface with the local DCS I/O panels and processors for 120-kV switchyard & CTGs process the signals and transmit the data via redundant fiber optic lines to a 3L Remote I/O panel in the Dedicated Shutdown Panel area in Fermi 2. The 3L Remote I/O panel interfaces with the Dedicated Shutdown panel for monitoring and control functions for the CTGs and 120-kV switchyard equipment. A block diagram of the CTG DCS control system is presented in Figure 7.5-4.

A dedicated shutdown system transfer pushbutton is provided in the control room, which, when activated, communicates with the DCS to initiate a start signal to CTG 11-1, arm the 120-kV switchyard undervoltage scheme, and inhibit control signals for the 120-kV

Switchyard and CTGs from the Fermi 2 control room. At the same time, 120-kV Switchyard and CTG 11-1 control is transferred to the Dedicated Shutdown Panel (H21-P623). If the control room is abandoned before the transfer pushbutton is activated, transfer can be accomplished at the Dedicated Shutdown Panel H21-P623 and the CTG 11-1 can be started manually after transfer. The possible fire induced spurious equipment actuations before transfer occurs are recoverable at the H21-P623 panel. The undervoltage scheme isolates 120-kV switchyard buses from offsite supplies and aligns breakers to provide power for Dedicated Shutdown from CTG 11-1.

The CTG power is supplied via peaker bus 1-2B through breaker A6 to the 4160-V Class 1E bus or via the main 120-kV bus through transformer SS64 (see Figure 8.3-1). The 4160-V bus provides power for the SBFW pumps; the Division I RHR, RHRSW, EECW, and EESW pumps and associated powered equipment through downstream electrical buses (Figure 7.5-5).

The SBFW system, Figure 7.5-6, described in Subsection 10.4.8, provides an alternative makeup water source for the reactor vessel. After transfer, the SBFW system is manually controlled and operated from the dedicated shutdown panel to maintain level above the top of the core. Control and transfer switches necessary for operating associated feedwater system valves and breakers are installed on the dedicated shutdown panel. Also, SBFW system flow is indicated on the dedicated shutdown panel. Power for the feedwater pump motors is from the CTGs or offsite power via the 4160-V electrical bus.

If the CTG is operating in parallel on the grid at the time offsite power is lost, the CTG output breaker is assumed to trip. However, the CTG turbine will not trip. The plant operator must take steps to isolate the grid, reclose the CTG output breaker and line up the two SBFW buses. If the CTG was not in operation, it could be started from the main control room before abandonment, but in the worst case, it would be started from the dedicated shutdown panel.

The RHR and RHRSW systems, described in Subsections 5.5.7 and 9.2.5, provide cooling capability for the reactor and torus water. The RHR system functional modes (1) torus cooling (Figure 7.5-7) and (2) shutdown cooling (Figure 7.5-8) modes, are described in Subsections 5.5.7.3.1 and 5.5.7.3.2.

The RHRSW system (Subsection 9.2.5) provides the heat sink for the reactor core by providing the cooling medium for the RHR heat exchanger. The EECW system functions as described in Subsection 9.2.2.1 to cool equipment required for reactor shutdown. The EECW is cooled by the EESW described in Subsection 9.2.5; a simplified flow diagram is shown in Figure 7.5-9.

The dedicated shutdown control panel is a local operation station, remote from the fire areas of concern, with instrumentation and control switches and transfer switches necessary for operating the SBFW shutdown system required to keep the reactor core covered with water. Instrumentation, control switches, and transfer switches on the panel are listed in Table 7.5-3.

Hot and cold shutdown can be achieved from the dedicated shutdown panel with manual operator action required locally in the reactor/auxiliary building and RHR complex. Local operation includes controlling equipment at local panels, switchgear, MCCs, distribution

panels, and valves. Figures 7.5-7 and 7.5-8 show the flow paths involved for both hot and cold shutdown.

Auxiliary systems required to support the alternative shutdown system are listed below and are described in the sections identified.

The SBFW system requires no auxiliary support system. Both pump and motor have a forced-flow lube-oil system that is driven off the pump shaft. The lube-oil system is cooled by a portion of the pump discharge flow routed to an oil cooler. Motor windings are designed to take a 74°F rise in temperature over a continuous rating of 111°F, which results in a maximum temperature of 185°F in approximately 60 hr. Once in shutdown cooling, the SBFW system will be turned off; inventory makeup will no longer be required. Shutdown cooling will be started in less than 11.75 hr. Space cooler, heat exchanger, and pump/motor cooling for the other systems is supplied either by EECW, EESW, or RHRSW as specified in Subsections 9.2.2 and 9.2.5.

Auxiliary support systems (i.e., heating, ventilation, and air conditioning [HVAC], or other fluid systems) are not required for the EECW system, EESW system, or the Division I switchgear room. This is because of the small heat loads generated (under the scenario very little electric equipment is energized) and because the EECW and EESW pumps require no external cooling or seal water. The dedicated shutdown panel area in the second floor of the radwaste building is provided with a local area cooler as described in Section 9.4.3. The function of the Dedicated Shutdown Air Conditioning Unit is to provide cooling, if needed, to maintain habitability at the Dedicated Shutdown Panel location for the duration of a post-fire shutdown requiring the use of the panel. This cooler is manually restored to the 72M Bus which is powered by dedicated shutdown power sources, including CTG 11-1. Once started using a switch near the Dedicated Shutdown Panel, area temperature is automatically controlled by local thermostat.

The RHR pump requires the support of a room cooler and a pump bearing cooler. Both the bearing cooler and pump room cooler are supplied by the EECW system. The room cooler also requires operation of a fan unit.

Drywell cooling is accomplished by establishing EECW flow to the drywell cooling units and operation of their associated fan units.

If normal communications links are not available, communications between the operators at locations in the plant and the dedicated shutdown panel operator are via hand-held portable radios that operate by either radio-to-radio, or radio-to-portable, repeater-to-radio communication links. Communication between the Dedicated CTG operator and the Main Control Room or local Dedicated Shutdown Panel operator is achieved using the local telephone system.

Eight-hour emergency lighting for safe-shutdown capability is provided for all local operations and for access/egress routes to and from local safe-shutdown areas. Use of local emergency lighting for the CTG area is addressed within the System Operating Procedure.

To implement the alternative shutdown concept, it must be ensured that cabling and required devices are not in, or do not pass through, a fire zone for which the concept is being relied upon, or that an adequate level of protection is provided. To achieve this objective, transfer switches have been installed that completely isolate any cabling that passes through the fire

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zones of concern from their associated actuating devices. New cable that is required for instrumentation and CTG supervisory control is routed to ensure it does not pass through the fire areas of concern.

Interfaces between Class 1E and non-Class 1E components meet the electrical separation criteria for Section 8.3 and appropriate IEEE criteria for environmental and seismic qualification.

Table 7.5-5 lists the 4160-V switchgear and motor control center (MCC) positions that have the above-described transfer function.

7.5.2.5.3 Procedure

An Abnormal Operating Procedure provides procedural guidance to achieve and maintain safe shutdown in the event that a fire in any of the dedicated shutdown areas of concern warrants post-fire shutdown from outside the main control room using the dedicated shutdown panel. The procedure provides direction regarding conditions upon which the procedure should be entered, actions to be taken in the control room before it is abandoned, as well as immediate and longer-term manual actions at the dedicated shutdown panel and other plant locations. These actions, in the aggregate, assure that the plant is put in a known and analyzed configuration to support the performance of the design basis functions described in Section 7.5.2.5.1 consistent with the safe shutdown analysis described in Appendix 9A.

When the standby diesel generator is utilized, the system operating procedure addresses starting and maintaining the standby diesel generator to power the CTG 11-2, CTG 11-3, or CTG 11-4 480 volt starting motors and auxiliaries. An additional dedicated CTG operator is required at the CTG area whenever the standby diesel generator is used to provide power for the Dedicated Shutdown System.

7.5.2.5.4 Safety Evaluation

Post-fire shutdown outside the control room using the alternative shutdown system and dedicated shutdown panel has been analyzed as described in Section 9A.3. This analysis includes circuit faults including open circuits, shorts to ground and hot shorts that could directly affect safe shutdown systems as well as common shutdown functions, e.g., loss of RPV inventory or spurious SRV operation. The safe shutdown analysis demonstrates that the core will remain covered with the Standby Feedwater System delivering flow to the RPV within 29 minutes. The time studies for establishing SBFW flow include allowance for the power supply starting times, breaker and valve operating times and operator transit times. In addition, the analysis demonstrates that design limits for suppression pool and drywell cooling established within approximately 3 and 4.5 hours, respectively, following reactor scram from full power.

Therefore, the alternative shutdown system (including the instrumentation and controls located on the dedicated shutdown panel), in conjunction with proceduralized manual actions taken at the panel and at other plant locations, provides the capability required to achieve and maintain safe shutdown following a fire that requires shutdown from outside the control room.

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A black start test of an alternate CTG demonstrated that use of an alternate CTG supported the performance goal of the alternative Dedicated Shutdown system which is Standby Feedwater system delivering flow to the RPV within 29 minutes.

7.5.2.5.5 Tests and Inspections

Except for where equipment directly interfaces with essential Class 1E components, the alternative shutdown system is considered a BOP system. Quality assurance requirements for the Class 1E portion of the alternative shutdown system will be the same as for the Class 1E equipment it is interfacing with. Quality assurance requirements applied to those portions of the system not interfacing directly with a Class 1E system will be appropriate for the use of that portion of that system. Periodic testing is described in Section 9A.6.

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TABLE 7.5-1 CONTROL ROOM LEVEL INDICATION

Instrument Name	Type and No.	Scale (in.)	Control Room Panel
Postaccident pressure/level recorder A	MRE-R623A	+10 to +220	601
Postaccident pressure/level recorder B	MRE-R623B	+10 to +220	602
Core level recorder	LR-R615		602
Core level recorder	LR-R610		601
Flood-up level indicator	LIE-R605	+160 to +560	603
Wide-range level indicator A	LI-R604A	+10 to +220	601
Wide-range level indicator B	LI-R604B	+10 to +220	602
Narrow-range level indicator A	LI-R606A	+160 to +220	603
Narrow-range level indicator B	LI-R606B	+160 to +220	603
Narrow-range level indicator C	LI-R606C	+160 to +220	603
Narrow-range level recorder A/B	LR-R614	+160 to +220	603

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TABLE 7.5-2 SAFETY-RELATED AND POWER GENERATION DISPLAY INSTRUMENTATION

	Reactor Water Level Transmitter <u>B21-N091A-D & B21-N085A-B^a</u>	Reactor Pressure Transmitter <u>B21-N051A-B</u>	Pressure/ Level Recorder <u>B21-R623A-B</u>	Neutron Monitoring Power Range Recorders <u>C51-R603A-B</u>
Design classes QA level/seismic category ⁱ	I/I	I/I	I/I	1M/II/I
Power supply	120-V ac Div. I/II inverter bus	120-V ac Div. I/II inverter bus	120-V ac Div. I/II inverter bus	120-V ac BOP inverter bus
Number of channels	6	2	2	4 ^b
Alarm setpoint(s) ^c	173 in. decreasing for N091 ^d – 42 in. for N085	NA ^e	Alarm on Auto-switchover to high chart speed	NA
Control logic	ECCS level logic	NA	NA	NA
Instrument range	N091 ^d – 10 to 220 in. N085 ^d - -150/0/50 in.	0-1500 psig	Compatible with inputs	0-125 percent
Instrument accuracy ^{f,j}	N091 - ±0.25 percent ^f N085 - ±0.25 percent	±0.25 percent	±0.5 percent	±0.5 percent
	<u>Annunciators RPS Variables^l</u>	<u>Integrated Plant Computer</u>	<u>Sequence-of-Events Recorder^l</u>	<u>Main Steam Flow Recorder C32-R607^g</u>
Design classes QA level/seismic category ⁱ	NQ/II/I	NQ/II/I	NQ/II/I	NQ/II/I
Power supply	130-V dc BOP battery inverter supply	120 V ac from UPS A and B	130-V dc BOP battery inverter supply	120-V ac BOP inverter bus
Number of channels	2 per variable	1 per variable	2 per variable	1
Alarm setpoint(s)	NA	NA	NA	NA
Control logic	Open circuit to alarm	NA	Open circuit to alarm	NA
Instrument range	NA	NA	NA	0-17 x 10 ⁶ lb/hr
Instrument accuracy ^{f,j}	NA	NA	NA	±0.5 percent

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TABLE 7.5-2 SAFETY-RELATED AND POWER GENERATION DISPLAY INSTRUMENTATION

	HPCI Discharge Pressure Transmitter <u>E41-N009</u>	HPCI Discharge Pressure Indicator <u>E41-R609</u>	HPCI Discharge Flow Transmitter <u>E41-N008</u>	HPCI Discharge Flow Indicator <u>E41-R613</u>	ADS Valve Position Indicator Lamp <u>s</u>
Design classes QA level/seismic category ¹	I/I	1M/II/I	I/I	I/I	I/I
Power supply	Inverter from Div. II Battery	Inverter from Div. II Battery	Inverter from Div. II Battery	Inverter from Div. II Battery	Div. I battery
Number of channels	1	1	1	1	1 per valve
Alarm setpoint(s)	NA	NA	NA	NA	NA
Control logic	NA	NA	NA	NA	NA
Instrument range	0-1500 psig	0-1500 psig	0-8000 gpm	0-8000 gpm	NA
Instrument accuracy ^{f,j}	±0.4percent	±0.5 percent	±0.25 percent	±.2 percent	NA
	Core Spray Discharge Pressure Transmitter <u>E21-N001A-B</u>	Core Spray Discharge Pressure Indicator <u>E21-R600A-B</u>	Core Spray Discharge Flow Transmitter <u>E21-N003A-B</u>	Core Spray Discharge Flow Indicator <u>E21-R601A-B</u>	
Design classes QA level/seismic category ¹	I/I	1M/II/I	I/I	1M/II/I	
Power supply	120-V ac Div. I/II	120-V ac Div. I/II	120-V ac Div. I/II	120-V ac Div. I/II	
Number of channels	2	2	2	2	
Alarm setpoint(s)	NA	NA	NA	NA	
Control logic	NA	NA	NA	NA	
Instrument range	0-600 psig	0-600 psig	0-10,000 gpm*	0-10,000 gpm*	
Instrument accuracy ^{f,j}	±0.4 percent	±0.5 percent	±0.25 percent	±1 percent	
			*9150 to 10,000 gpm on the scale not used		
	Core Spray Pump Motor Current <u>E11-N056A-D</u>	RHR (LPCI Mode) Pump Discharge Header Pressure Transmitter <u>E11-N056A-D</u>	RHR (LPCI Mode) Pump Discharge Header Pressure Indicator <u>E11-R803/R804</u>	RHR (LPCI Mode) Flow Transmitter <u>E11-N015A-B</u>	
Design classes QA level/seismic category ¹	NQ/II/I	I/I	1M/II/I	I/I	
Power Supply	Current transformer	120-V ac. Div. I/II inst. bus	120-V ac. Div. I/II inst. bus	120-V ac. Div. I/II inst. bus	
Number of channels	4 (1 per motor)	4 (1 per motor)	4 (1 per motor)	4 (1 per motor)	
Alarm setpoint(s) ^c	125 percent	NA	NA	NA	
Control logic	NA	NA	NA	NA	
Instrument range	0-200 percent	0-500 psig	0-500 psig	0-774.8" W.C.	
Instrument accuracy ^{f,j}	±2 percent	±0.25 percent	±0.5 percent	±0.25 percent	

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**TABLE 7.5-2 SAFETY-RELATED AND POWER GENERATION DISPLAY
INSTRUMENTATION**

	RHR (LPCI Mode) Flow Recorder <u>E11-R608A-B</u>	RHR Pump Motor Current <u>Motor Current</u>	RHR Service Water Flow Transmitter <u>E11-N007A-B</u>	RHR Service Water Flow Indicator <u>E11-R602A-B</u>	RHR Service Water Thermocouples <u>E11-N005A-B</u>
Design classes QA level/seismic category ¹	1M/II/I	NQ/II/I	I/I	1M/II/I	NQ/II/I
Power Supply	120-V ac Div. I/II inst. bus	Current transformer	120-V ac Div. I/II inst. bus	120-V ac Div. I/II inst. bus	NA
Number of channels	2	4 (1 per motor)	2	2	2
Alarm setpoint(s) ^c	NA	289 amps	NA	NA	NA
Control logic	NA	NA	NA	NA	NA
Instrument range	0-28,000 gpm	0-500 amps	0-10,000 gpm	0-10,000 gpm	0-400 °F
Instrument accuracy ^{fj}	±0.5 percent	±2 percent	±0.4 percent	±1 percent	NA
	RHR Service Water Outlet Temperature Recorder <u>E11-R601A-B</u>	Relief Valve Discharge Thermocouples <u>B21-N004A-H, J-N, P, R</u>	Relief Valve Discharge Temperature Recorder <u>B21-R614</u>	<u>Indicating Timer on ADS</u>	
Design classes QA level/seismic category ¹	NQ/II/I	NQ/II/I	1M/II/I	NQ/II/I	
Power supply	120-V ac Div. I/II inst. Bus	NA	120-V ac BOP inst. Bus	120-V ac Div. I battery	
Number of channels	2	15	15 (one recorder)	1	
Alarm setpoint(s) ^c	175 °F	NA	220 °F	NA	
Control logic	NA	NA	NA	Starts on energization of ADS timers	
Instrument range	0-400 °F	0-600 °F	0-600 °F	105-0 sec (count down)	
Instrument accuracy ^{fj}	±0.5 percent	Per ANSI C96.1	±0.2 percent	±1 sec.	

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TABLE 7.5-2 SAFETY-RELATED AND POWER GENERATION DISPLAY INSTRUMENTATION

	<u>Automatic Load Sequencer</u>	<u>Digital "Remaining Capacity" Meter</u>	<u>Drywell Thermocouples T50-N409B, - N412A</u>	<u>Drywell and Torus Temperature Recorder T50-R800A-B</u>	<u>Pressure Transmitter (Narrow Range) T50-N401A-B</u>
Design classes QA level/seismic category ⁱ	I/I	NQ/II/I	I/I	I/I	I/I
Power supply	260-130-V dc Div. I&II battery	120-V ac inst. Buses Div. I&II	NA	120-V ac Div I: Instrument Bus Div II: Inverter Bus	Div. I: battery inverter on Div. I Div. II: battery inverter on Div. II
Number of channels	4	4	2	6 (2 – Drywell) (4 – Torus) page 7.5-33	2
Alarm setpoint(s) ^c	NA	NA	NA	NA	NA
Control logic	NA	NA	NA	NA	NA
Instrument range	NA	3000-0 kW	See recorder	0-400 °F	-5 to +5 psig
Instrument accuracy ^{f,j}	NA	±1 percent	Standard TC wire	±0.3 percent	±0.25 percent
	<u>Drywell Pressure Transmitter (Wide Range) T50-N415A-B</u>		<u>Drywell Narrow Range, Wide Range and Torus Wide Range, Narrow Range Pressure Recorder T50-R802A-B</u>		
Design classes QA level/seismic category ⁱ	I/I		I/I		
Power supply	CH A battery inverter on Div. I CH B battery inverter, Div. II		Div. I - battery inverter on Div. I Div. II – battery inverter on Div. II		
Number of channels	2		2		
Alarm setpoint(s) ^c	NA		NA		
Control logic	NA		NA		
Instrument range ^k	0-250 psig		-5 to +5 psig, 0 to 250 psig, 0-80 psig, -5 to +15 psig		
Instrument accuracy ^{f,j}	±0.25 percent		±0.25 percent		

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TABLE 7.5-2 SAFETY-RELATED AND POWER GENERATION DISPLAY INSTRUMENTATION

	<u>Torus Thermocouples T50-N402A, -N403B, -N405B, &N404A</u>	<u>Containment Radiation D11-K816 A-B</u>	<u>Torus Pressure Transmitters Wide-Range T50-N414A-B</u>	<u>Torus Pressure Transmitters Narrow- Range T50-N499 A-B</u>
Design classes QA level/seismic category ⁱ	I/I	I/I	I/I	I/I
Power supply	NA	CH A-120-V ac inst. bus, Div. I CH B-120-V ac inst. bus, Div. II	CH A-battery – inverter on Div. I CH B-battery – inverter on Div. II	CH A-battery – inverter on Div. I CH B-battery – inverter on Div. II
Number of channels	4	2	2	2
Alarm setpoint(s) ^c	NA	NA	NA	NA
Control logic	NA	NA	NA	NA
Instrument range	NA See recorder	10 ⁰ to 10 ⁸ rad/hr	NA 0 to 80 psig	NA -5 to +15 psig
Instrument accuracy ^{fj}	Standard TC wire	--	±0.25 percent	±0.25 percent
	<u>Suppression Pool Water Level Transmitter T50-N406 A-B</u>	<u>Suppression Pool Water Level Recorder T50-R804 A-B</u>	<u>Drywell Radiation Instrument T50-N003</u>	<u>Drywell Radiation Recorder T50-R809</u>
Design classes QA level/seismic category ⁱ	I/I	I/I	IM/II/I	1M/II/I
Power supply	Div. I-Battery inverter on Div. I Div. II-Battery inverter on Div. II	120-V ac inst. buses Div. 1 & II	120-V ac inst. buses BOP	120-V ac inst. buses BOP
Number of channels	2	2	1	1
Alarm setpoint(s) ^c	NA	NA	To be established after background is measured	To be established after background is measured
Control logic	NA	NA	NA	NA
Instrument range	-144 to +56 in.	-144 to +56 in.	Variable over suitable range	Compatible with radiation instrument recorder output
Instrument accuracy ^{fj}	±0.25 percent	±0.25 percent	2 percent	±0.5 percent

^a Wide range shutdown indication. Reads full scale when jet pumps are operating.

^b Two recorders display two channels each.

^c Nominal values are given for information. See Technical Specifications for operational limits.

^d Measured from top of active fuel.

^e NA = Not Applicable.

^f Accuracy specified in percent of full scale unless otherwise noted.

^g Recorder shared with feedwater flow signal.

^h May be obtained from analyzers or recorders or combination of the two depending on make selected.

ⁱ The instrument seismic category and QA level information provided in the UFSAR tables may have been upgraded to meet the Pressure Boundary Integrity (PBI) or other requirements. The instrument seismic category and QA level information is available in the Fermi 2 Central Component Database.

^j The instrument accuracy information provided in the UFSAR tables is a bounding value. For actual value see the Fermi 2 Central Component Database.

^k Deleted

^l The Visual Annunciator System (C9700) combines the annunciator and sequence-of-events recorder function using redundant hardware and application software.

TABLE 7.5-3 DEDICATED SHUTDOWN PANEL INSTRUMENTATION AND CONTROLS

Instrumentation

- Reactor pressure
- Reactor level
- Condensate storage tank level
- Torus temperature
- Torus level
- Primary containment temperature (drywell)
- Standby feedwater flow
- Bus voltage monitor for buses 101, 102, 1, 1-2, 64
- Combustion turbine generator - voltage, frequency, watts, VARs
- Undervoltage trip armed
- Supervisory control transferred

Controls

120 - kV breaker control

- 1) breaker GM
- 2) breaker GK
- 3) breaker GH
- 4) breaker GD

CTG control

- 1) raise/lower voltage
- 2) raise/lower governor
- 3) power block control

13.8- kV breaker control

- 1) breaker A2
- 2) breaker A6
- 3) breaker A7
- 4) breaker B6

Standby feedwater system

- 1) SBFW pump A (4160-V breaker V2)
- 2) SBFW pump B (4160-V breaker W4)
- 3) SBFW discharge isolation valve (N2103 F001)
- 4) SBFW low flow discharge valve (N2103 F003)
- 5) SBFW high flow discharge valve (N2103 F002)

13.2 kV breaker control

- 1) breaker A
- 2) breaker B
- 3) breaker C
- 4) breaker D

Safety/relief valve B21-F013G

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TABLE 7.5-3 DEDICATED SHUTDOWN PANEL INSTRUMENTATION AND CONTROLS

4160-V breaker control

- 1) breaker V1
- 2) breaker V2
- 3) breaker V3
- 4) breaker W4
- 5) breaker W5

Transfer switches

- 1) EF2 Supervisory Control
- 2) EF2 System Controls

TABLE 7.5-4 HAS BEEN INTENTIONALLY DELETED

TABLE 7.5-5 4160-V SWITCHGEAR AND MOTOR CONTROL CENTERS WITH TRANSFER AND LOCAL CONTROL CAPABILITY

Breaker V1, V2, V3, W4, W5, C5, C6, C8, C11, C9, C10
(no local control for C9 and C10, just transfer)

480-V MCC 72B-2A:

Valve

P4400F616 (EECW inboard containment isolation)

480-V MCC 72B-3A:

Valve

E1150F028A (RHR torus return)

E1150F024A (RHR torus return)

E1150F004A (RHR torus pump suction)

E1150F611A (RHR valve F017 bypass)

Fan

T4700C001

480-V MCC 72C-F:

Valve

B3105F031A (reactor recirculation pump discharge isolation)

E1150F010 (RHR Division II cross tie)

E1150F015A (RHR shutdown cooling return to vessel)

E1150F017A (RHR shutdown cooling return to vessel)

480-V MCC 72C-3A

Valve

E1150F009 (RHR shutdown cooling inboard suction)

P4400F606A (EECW supply - outboard containment isolation)

E1150F003A (RHR heat exchanger outlet)

E1150F004C (RHR torus pump suction)

TABLE 7.5-5 4160-V SWITCHGEAR AND MOTOR CONTROL CENTERS WITH TRANSFER AND LOCAL CONTROL CAPABILITY

E1150F047A (RHR heat exchanger inlet)
E1150F068A (RHR service water throttle)
E1150F048A (RHR heat exchanger bypass)
P4400F601A (EECW return to RBCCW)
P4400F602A (EECW make-up tank outlet)
P4400F603A (RBCCW supply to EECW)
E1150F006C (RHR pump suction isolation)
E1150F016A (RHR drywell spray line)
E41-F400 (torus water level isolation valve)
T50-F412A (torus water level isolation valve)

Fan

T4700C002 (containment cooling fan)
T4100B018 (RHR room cooler fan)

MCC 2PC 1

Valve

N2103F001 (SBFW discharge isolation)
N2103F002 (SBFW high flow discharge throttle valve)
N2103F003 (SBFW low flow discharge throttle valve)

MCC 72F-4A (with alternate supply from 72M-3B)

Valve

P4400F607A (EECW return - outboard containment isolation)

MCC 72M-3B, Compt 5BR

Battery Charger

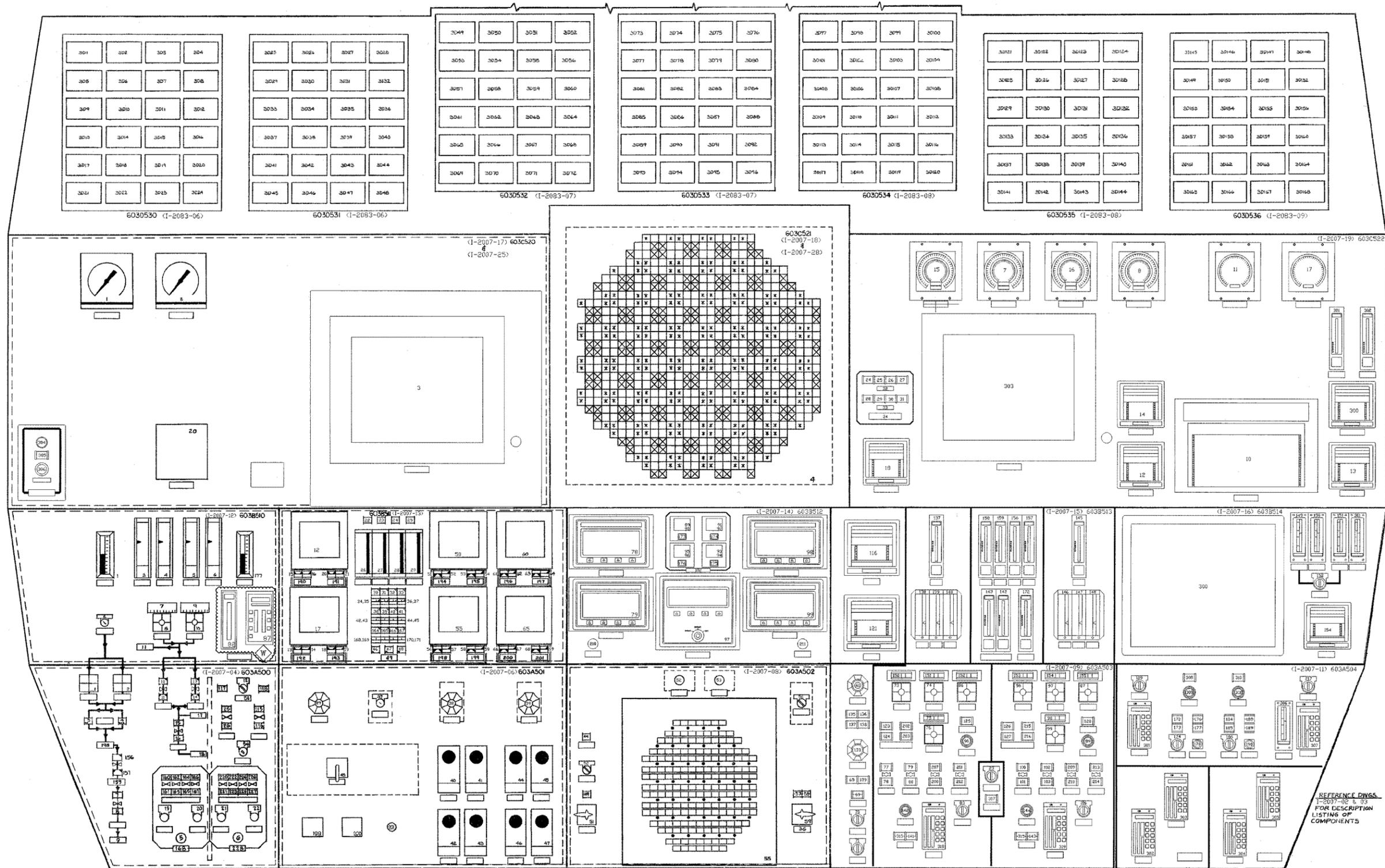
R3200S022A (BOP Battery Charger 2C-1, DC power for SBFW valves)

TABLE 7.5-5 4160-V SWITCHGEAR AND MOTOR CONTROL CENTERS WITH
TRANSFER AND LOCAL CONTROL CAPABILITY

MCC 72S-2A, Compt 5C

Battery Charger

R3200S022C (BOP Battery Charger 2C1-2, DC power for SBFW valves)



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FIGURE 7.5-1, SHEET 1
 REACTOR CONTROL BOARD

PANEL ITEM NO.	VALVE NO.	INSTRUM OR FIS NO.	DESCRIPTION	ENGRAVING	P & ID DWG NO.	SYSTEM REF NO.	I & C DWG NO.
3A1		L4103 C001A	CNC INDICATOR	STANDBY LIQUID PUMP I-A	M-2082	C81	I-2131-01
3A2		C4103 C001B	CNC INDICATOR	STAND BY LIQUID PUMP I-B			
3A3	VRA-2008	C41 F004A	BACK LIGHTED DISPLAY (AMB) CIRC 'A' COMFY				
3A4	VRA-2008	C41 F004B	BACK LIGHTED DISPLAY (AMB) CIRC 'B' COMFY	SAUIB	M-2082	C41	I-2131-01
3A5		B3100 M040	RESET PUSHBUTTON	ATWS - DIV I		B31	I-2105-18&-19
3A6		B3100 M041	RESET PUSHBUTTON	ATWS - DIV II		B31	I-2105-18&-19
3A7	VRA-2013	C4100 F008	BACK LIGHTED DISPLAY OPEN (RED)		M-2082	C41	I-2135-01
3A8	VRA-2013	C4100 F008	BACK LIGHTED DISPLAY CLOSED (GREEN)	MANUAL INJECTION RELY F008	M-2082	C41	I-2135-01
3A9			MARKER PLATE	TO REACTOR			
3A10		C11 F002A	BACK LIGHTED DISPLAY OPEN (RED)		M-2081	C11	I-2115-20
3A11	-/-	-/-	BACK LIGHTED DISPLAY CLOSED (GREEN)	PC VALVE F008A			-/-
3A12			2 POSITION SWITCH A & B		SEE 3A41		I-2115-11
3A13		C11 F002B	BACK LIGHTED DISPLAY OPEN (RED)				I-2115-20
3A14	-/-	-/-	BACK LIGHTED DISPLAY CLOSED (GREEN)	PC VALVE F008B			-/-
3A15	V08-2187	C1152 F003	BACK LIGHTED PB OPEN (RED)				I-2115-03
3A16	-/-	-/-	BACK LIGHTED PB CLOSE (GREEN)	DRICLS WTR PRESS CONT CU-F003	M-2081	C11	-/-
3A17			MARKER PLATE	DRIVE WATER			
3A18			MARKER PLATE	COOLING WATER			
3A19		B3100 M031A	ARMED SEL. S.W.	ATWS - DIV I CHANNEL A		B31	I-2105-18
3A20		B3100 M031C		ATWS - DIV I CHANNEL C			I-2105-19
3A21		B3100 M031B		ATWS - DIV II CHANNEL B			I-2105-18
3A22		B3100 M031D	ARMED SEL. S.W.	ATWS - DIV II CHANNEL D		B31	I-2105-19
3A23		B3100 M059	SELECTOR SWITCH	SEE I-2975-116		B31	I-2105-05
3A24			MARKER PLATE	STABILIZER VALVE SELECT			
3A25	V08-2086	C1100 F011A/B	BACK LIGHTED DISPLAY OPEN (RED)		M-2081	C11	I-2155-11
3A26	-/-	-/-	BACK LIGHTED DISPLAY CLOSED (GREEN)	SCRAM DISCH VOLUME DRAIN CH-F011/F15		C11	-/-
3A27			5 POSITION JOYSTICK SELECTOR SWITCH			C51	I-2145-52
3A28			N/A				
3A29			N/A				
3A30			N/A				
3A31							
3A32							
3A33			N/A				
3A34			N/A				
3A35			N/A				
3A36			MARKER PLATE	ROD MOVEMENT CONTROL			
3A37			KEYLOCK SW. NORMAL BYPASS	SCRAM DISCH VOL HI H2O LEVEL BYPASS		C71	I-2155-06

PANEL ITEM NO.	VALVE NO.	INSTRUM OR FIS NO.	DESCRIPTION	ENGRAVING	P & ID DWG NO.	SYSTEM REF NO.	I & C DWG NO.
3A38			5 POSITION JOYSTICK SELECTOR SWITCH	IRM BY-PASS A		C51	I-2145-55
3A39			5 POSITION JOYSTICK SELECTOR SWITCH	IRM BY-PASS B			I-2145-56
3A40			IRM RANGE SWITCH	IRM A			I-2145-52, 55 & 59
3A41				IRM C			-/-
3A42				IRM H			I-2145-52, 55 & 59
3A43				IRM F			-/-
3A44				IRM D			-/-
3A45				IRM B			-/-
3A46				IRM E			I-2145-52, 55 & 59
3A47			IRM RANGE SWITCH	IRM G		C51	-/-
3A48			1 POS. KEYLOCK SW. 5-11. SCRAM DISCH VOL HI H2O LEVEL BYPASS	REAC MODE		C71	I-2155-03
3A49			BACK LIGHTED DISPLAY (WHITE)	REFUEL MODE ONE ROD PERMISSIVE		C11	I-2115-12
3A50			2 POSITION SWITCH OFF-ON	ROD SELECT POWER		C11	I-2115-11
3A51			3 POS. 50M SW. OVERRIDE, CFP EMERGENCY ROD IN	ROD OUT NOTCH OVERRIDE		C11	-/-
3A52			PUSHBUTTON SW. MOMENT. ACT. LIGHTED RED	MANUAL SCRAM TRIP A1		C71	I-2155-08
3A53			PUSHBUTTON SW. MOMENT. ACTION LIGHTED RED	MANUAL SCRAM TRIP B1		C71	I-2155-09
3A54		C11 F008	KEYLOCK SEL SW. VOLUME TEST	SCRAM DISCH VOLUME ISOLATION	M-2081	C11	-/-
3A55			G E INSERT ROD SELECT			C11	I-2115-07
3A56			KEYLOCK SELCT SW. GROUP 1 & 2 NORM. GROUP 3 & 4	SCRAM RESET		C71	I-215-07 2145-02
3A57			BACK LIGHTED SPLIT SCREEN DISPLAY	TOP ROD OUT SETTLE (AMB) * BOT ROD IN (GRN) *	SEE 3A38	C11	I-2115-12
3A58			BACK LIGHTED SPLIT SCREEN DISPLAY	TOP ROD OUT PERM (W) * BOT ROD OUT (RED) *	SEE 3A38	C11	-/-
3A59			3 POS 50M SW. IN, OFF, OUT NOTCH			C11	I-2115-11
3A60							
3A61		C5100 M003	5 POSITION JOYSTICK F.D. SWITCH	APRM BYPASS		C51	
3A62							
3A63							
3A64							
3A65			BACK LIGHTED DISPLAY WHITE	RBM A BYPASSED		C51	
3A66							
3A67							
3A68							
3A69			BACK LIGHTED DISPLAY (RED) SEL. BLOCK TEST (WH)	TIMING MALF SEL. BLOCK		C11	I-2115-12
3A70			3 POSITION SWITCH RESET-TEST	TIMER MALFUNCTION		C11	I-2115-13
3A71			3 POSITION SWITCH RESET-TEST	ROD DRIFT ALARM		C11	I-2115-16
3A72							
3A73			CNC SWITCH	SEE I-2975-116		B31	I-2101-06
3A74			CNC SWITCH	SEE I-2975-116			I-2101-07

PANEL ITEM NO.	VALVE NO.	INSTRUM OR FIS NO.	DESCRIPTION	ENGRAVING	P & ID DWG NO.	SYSTEM REF NO.	I & C DWG NO.
3A75		B31 R628A	AMMETER				I-2101-01
3A76			CNC SWITCH	SEE I-2975-116			-/-
3A77	V8-2001	B3105 F028A	BACK LIGHTED PB OPEN (RED)		M-2833		I-2101-03
3A78	-/-	-/-	BACK LIGHTED PB CLOSE (GREEN)	SEE I-2975-116	-/-		-/-
3A79	V8-2003	B3105 F031A	BACK LIGHTED PB OPEN (RED)		M-2833		I-2101-05
3A80	-/-	-/-	BACK LIGHTED PB CLOSE (GREEN)	SEE I-2975-116	-/-	B31	-/-
3A81							
3A82			N/A				
3A83			3 POSITION SWITCH TRIP-NORM-RESET			B31	I-2105-03
3A84							
3A85							
3A86			CNC SWITCH	SEE I-2975-116		B31	I-2101-12
3A87			CNC SWITCH	SEE I-2975-116		B31	-/-
3A88			N/A				
3A89			PUSHBUTTON SWITCH RESET	SEE I-2975-116		B31	I-2105-04
3A90			PUSHBUTTON SWITCH RESET	SEE I-2975-116		-/-	I-2105-03
3A91			N/A				
3A92			N/A				
3A93			MARKER PLATE	RBM "A"			
3A94			MARKER PLATE	RBM "B"			
3A95			N/A				

NOTES:

1. INFORMATION IN ENGRAVING COLUMN NOTED WITH AN ASTERISK (*) INDICATES LENS ENGRAVING ONLY
2. INFORMATION IN DESCRIPTION COLUMN NOTED "N/A" INDICATES ITEM NUMBER IS NOT APPLICABLE TO THIS PANEL

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FIGURE 7.5-1, SHEET 2
 REACTOR CONTROL BOARD

PANEL ITEM NO.	VALVE NO.	INSTRUM OR PIS. NO.	DESCRIPTION	ENGRAVING	P & ID DWG	SYSTEM REF NO.	I & C DWG NO.
3A196			ONC SWITCH	SEE I-2975-116		B31	I-2101-08
3A197			ONC SWITCH	SEE I-2975-116			I-2101-09
3A198		B31 R628B	AMMETER				I-2101-02
3A199			ONC SWITCH	SEE I-2975-116			-11-
3A100	VB-2002	B3105 F023B	BACK LIGHTED PB OPEN (RED)		M-2833		I-2101-03
3A101	-11-	-11-	BACK LIGHTED PB CLOSE (GREEN)	SEE I-2975-116			-11-
3A102	VB 2004	B3105 F031B	BACK LIGHTED PB OPEN (RED)				I-2101-04
3A103	-11-	-11-	BACK LIGHTED PB CLOSE (GREEN)	SEE I-2975-116	M-2833	B31	-11-
3A104							
3A105			N/A				
3A106			3 POSITION SWITCH TRIP-NORM-RESET	SEE I-2975-116		B31	I-2105-04
3A107							
3A108							
3A109			3 POSITION SEL SW 3 ELEMENT	SEE I-2975-116		C32	I-2125-05
3A110							
3A111			N/A				
3A112							
3A113							
3A114			N/A				
3A115	VB-2073	C1100 F010/F180	BACK LIGHTED DISPLAY OPEN (RED)		M-2081	C11	I-2155-11
3A116	-11-	-11-	BACK LIGHTED DISPLAY CLOSED (GREEN)	SCRAM DISCHARGE VOLUME VENT C11-F000/F180			-11-
3A117		C11 F007A	BACK LIGHTED DISPLAY (WHITE)	"A" VALVES *	SEE		I-2115-12
3A118		C11 F007B	BACK LIGHTED DISPLAY (WHITE)	"B" VALVES *	3A24 M-2081	C11	-11-
3A119							
3A120							
3A121							
3A122							
3A123			BACK LIGHTED DISPLAY (RED) CLOSED	SEE I-2975-116		B31	I-2105-16
3A124			BACK LIGHTED DISPLAY (WHITE) TRIPPED	SEE I-2975-116			-11-
3A125			BACK LIGHTED DISPLAY POWER (RED) AVAILABLE	SEE I-2975-116			I-2105-03
3A126			BACK LIGHTED DISPLAY (RED) CLOSED	SEE I-2975-116			I-2105-16
3A127			BACK LIGHTED DISPLAY (WHITE) TRIPPED	SEE I-2975-116			-11-
3A128			BACK LIGHTED DISPLAY POWER (RED) AVAIL	SEE I-2975-116		B31	I-2104-04
3A129							
3A130							
3A131							
3A132							

PANEL ITEM NO.	VALVE NO.	INSTRUM OR PIS. NO.	DESCRIPTION	ENGRAVING	P & ID DWG	SYSTEM REF NO.	I & C DWG NO.
3A133		C5100 M001	5 POSITION JOYSTICK SEL. SWITCH	RBM BYPASS			C51
3A134							
3A135			BACK LIGHTED DISPLAY WHITE	APRM 1 BYPASSED			C51
3A136			BACK LIGHTED DISPLAY WHITE	APRM 2 BYPASSED			C51
3A137			BACK LIGHTED DISPLAY WHITE	APRM 3 BYPASSED			C51
3A138			BACK LIGHTED DISPLAY WHITE	APRM 4 BYPASSED			C51
3A139			BACK LIGHTED DISPLAY WHITE	RBM B BYPASSED			C51
3A140							
3A141			BACK LIGHTED SPLIT SCREEN DISPLAY (AMB)	TOP: LIMITER 2 BOT: LIMITER 3		B31	I-2105-05
3A142			PUSHBUTTON SWITCH RESET	SEE I-2975-116			
3A143			BACK LIGHTED SPLIT SCREEN DISPLAY (AMB)	TOP: LIMITER 2 BOT: LIMITER 3			
3A144			PUSHBUTTON SWITCH RESET	SEE I-2975-116		B31	I-2105-05
3A145							
3A146							
3A147							
3A148							
3A149			BACK LIGHTED DISPLAY (WHITE)	NOTCH OVRDR *	SEE 3A51	C11	I-2115-12
3A150		B31 R630A	AMMETER FOR 3A73			B31	I-2101-06
3A151		B31 R631A	FOR 3A74				I-2101-07
3A152		B31 R632A	FOR 3A86				I-2101-12
3A153		B31 R630B	FOR 3A96				I-2101-08
3A154		B31 R631B	FOR 3A97				I-2101-09
3A155		B31 R632B	AMMETER FOR 3A87			B31	I-2101-12
3A156	YR4-2036	C4100 F006	BACK LIGHTED DISPLAY (RED)	VALVE DISC OPEN *	M-2082	C11	I-2135-01
3A157	-11-	-11-	BACK LIGHTED DISPLAY (GRN)	VALVE DISC CLOSED *	SEE 3A159	-11-	-11-
3A158			N/A				
3A159			MARKER PLATE	CHECK VALVE C41-F006			
3A160		C11 F160A	BACK LIGHTED DISPLAY OPEN (RED)		M-2081	C11	I-2105-17
3A161	-11-	-11-	BACK LIGHTED DISPLAY CLOSED (GRN)	C11-F160A			
3A162		C11 F162A	BACK LIGHTED DISPLAY OPEN (RED)				
3A163	-11-	-11-	BACK LIGHTED DISPLAY CLOSED (GRN)	C11-F162A	SEE 3A168		
3A164		C11 F162C	BACK LIGHTED DISPLAY OPEN (RED)				
3A165	-11-	-11-	BACK LIGHTED DISPLAY CLOSED (GRN)	C11-F162C			
3A166		C11 F163A	BACK LIGHTED DISPLAY OPEN (RED)				
3A167	-11-	-11-	BACK LIGHTED DISPLAY CLOSED (GRN)	C11-F163A	M-2081	C11	I-2105-17

PANEL ITEM NO.	VALVE NO.	INSTRUM OR PIS. NO.	DESCRIPTION	ENGRAVING	P & ID DWG	SYSTEM REF NO.	I & C DWG NO.
3A168			MARKER PLATE	ATWS ARI/RPT DW I			
3A169			N/A				
3A170							
3A171			N/A				
3A172	V12-2002	B2100 F076A	BACK LIGHTED DISPLAY OPEN (RED)		M-2023	B21	I-2095-21
3A173	-11-	-11-	BACK LIGHTED DISPLAY CLOSED (GRN)				-11-
3A174	-11-	-11-	2 POS SEL SW CLOSE OPEN	SEE I-2975-116		-11-	-11-
3A175							
3A176	V12-2004	B2100 F032A	BACK LIGHTED SPLIT SCREEN DISPLAY (RED)	TOP: ACT DISCH OPEN * BOT: DISCH OPEN *	M-2023	B21	I-2095-21
3A177	-11-	-11-	BACK LIGHTED SPLIT SCREEN DISPLAY (GRN)	TOP: ACT DISCH OPEN * BOT: DISCH OPEN *		-11-	-11-
3A178	-11-	-11-	TEST PUSHBUTTON	SEE I-2975-116		-11-	-11-
3A179							
3A180							
3A181							
3A182							
3A183							
3A184	V12-2001	B2100 F076B	BACK LIGHTED DISPLAY OPEN (RED)		M-2023	B21	I-2095-21
3A185	-11-	-11-	BACK LIGHTED DISPLAY CLOSED (GRN)			-11-	-11-
3A186	-11-	-11-	2 POS SELECTOR SW CLOSE OPEN	SEE I-2975-116		-11-	-11-
3A187							
3A188	V12-2003	B2100 F032B	BACK LIGHTED SPLIT SCREEN DISPLAY (RED)	TOP: ACT DISCH OPEN * BOT: DISCH OPEN *	M-2023	B21	I-2095-21
3A189	-11-	-11-	BACK LIGHTED SPLIT SCREEN DISPLAY (GRN)	TOP: ACT DISCH OPEN * BOT: DISCH OPEN *		-11-	-11-
3A190	-11-	-11-	TEST PUSHBUTTON	SEE I-2975-116		-11-	-11-

NOTES:

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2. INFORMATION IN DESCRIPTION COLUMN NOTED "N/A" INDICATES ITEM NUMBER IS NOT APPLICABLE TO THIS PANEL

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FIGURE 7.5-1, SHEET 3
REACTOR CONTROL BOARD

PANEL ITEM NO.	VALVE NO.	INSTRUM OR PIS NO.	DESCRIPTION	ENGRAVING	P&ID DWG NO.	SYSTEM REF NO.	I&C DWG NO.
3A191							
3A192							
3A193							
3A194							
3A195							
3A196							
3A197							
3A198			MARKER PLATE	STAND-BY LIQUID			
3A199			COVER PLATE				
3A200			COVER PLATE				
3A201			SNAP BUTTON PLUG				
3A202			BACKLIGHTED DISPLAY (RED)	SEE I-2975-116		B31	I-2105-16
3A203			BACKLIGHTED DISPLAY (W.H.) TRIPPED	SEE I-2975-116			
381		C41 R601	LEVEL INDICATOR	STAND-BY LIQUID TANK LEVEL		C41	I-2135-01
382			N POS KEYLOCK SW PUMP A STOP RUN	STAND-BY LIQ. CONTROL SYS INITIATION		C41	I-2131-01
383		C11 R604	FLOW INDICATOR	DRIVE WATER FLOW		C11	I-2115-18
384		C11 R602	DIFFERENTIAL PRESSURE INDICATOR	DRIVE H ₂ O TO REACTOR DIFF PRESS			
385		C11 R605	FLOW INDICATOR	COOLING WATER FLOW			
386		C11 R603	DIFFERENTIAL PRESSURE INDICATOR	COOLING WATER H ₂ O TO REACTOR DIFF PRESS			I-2115-18
387		C11 R609A	AMMETER				I-2111-01
388			CNC SWITCH	CONTROL ROD DRIVE PUMP A			-11-
389		C11 R609B	AMMETER				I-2111-02
3810			CNC SWITCH	CONTROL ROD DRIVE PUMP B		C11	-11-
3811			MARKER PLATE	CHARGING WATER			
3812		C51 R602A	RECORDER 2 PEN		SEE 38190 & 38191	C51	I-2145-51
3813			BACK LIGHTED SPLIT SCREEN DISPLAY	REFER TO DETAIL G I-2007-03	SEE		I-2145-52
3814				REFER TO DETAIL D I-2007-03	38192		
3815				REFER TO DETAIL G I-2007-03	SEE		
3816			BACK LIGHTED SPLIT SCREEN DISPLAY	REFER TO DETAIL D I-2007-03	38190		I-2145-52
3817		C51 R602B	RECORDER 2 PEN		SEE 38192 & 38193		I-2145-51
3818			BACK LIGHTED SPLIT SCREEN DISPLAY	REFER TO DETAIL G I-2007-03	SEE		I-2145-52
3819				REFER TO DETAIL D I-2007-03	38193		
3820				REFER TO DETAIL G I-2007-03	SEE		
3821			BACK LIGHTED SPLIT SCREEN DISPLAY	REFER TO DETAIL D I-2007-03	38191		
3822			BACK LIGHTED DISPLAY GREEN	PERIOD *			

PANEL ITEM NO.	VALVE NO.	INSTRUM OR PIS NO.	DESCRIPTION	ENGRAVING	P&ID DWG NO.	SYSTEM REF NO.	I&C DWG NO.
3823							
3824							
3825			BACK LIGHTED DISPLAY GREEN	PERIOD *			I-2145-52
3826		C51 R601A	PERIOD INDICATOR	SRM PERIOD CH A			I-2145-51
3827		C51 R601B		SRM PERIOD CH B			
3828		C51 R601C		SRM PERIOD CH C			
3829		C51 R601D	PERIOD INDICATOR	SRM PERIOD CH D			I-2145-51
3830			BACK LIGHTED DISPLAY GREEN	I-2007-03 REFER TO DETAIL A			I-2145-52
3831							
3832							
3833			BACK LIGHTED DISPLAY GREEN				I-2145-52
3834			BACK LIGHTED PB WITH 3-WAY SPLIT SCREEN(W)				I-2145-61
3835							I-2145-62
3836							I-2145-61
3837			BACK LIGHTED PB WITH 3-WAY SPLIT SCREEN(W)				I-2145-62
3838			BACK LIGHTED DISPLAY GREEN		SEE		I-2145-55
3839					38199		-11-
3840							I-2145-56
3841			BACK LIGHTED DISPLAY GREEN				-11-
3842			BACK LIGHTED PB WITH 3-WAY SPLIT SCREEN(W)				I-2145-62
3843							-11-
3844							I-2145-63
3845			BACK LIGHTED PB WITH 3-WAY SPLIT SCREEN(W)	I-2007-03 REFER TO DETAIL 'A'			-11-
3846			BACKLIGHTED PB WHITE	DRIVE IN *			I-2145-61
3847			BACKLIGHTED PB GREEN	POWER ON *			-11-
3848			BACKLIGHTED PB WHITE	DRIVE OUT *		C51	-11-
3849			MARKER PLATE	START UP RANGE DETR DRIVE CONTROL			
3850		C51 R607A	RECORDER 2 PEN		SEE 38194 & 38195	C51	I-2145-53
3851			BACK LIGHTED SPLIT SCREEN DISPLAY	REFER TO DETAIL C I-2007-03	SEE		I-2145-55
3852				REFER TO DETAIL D I-2007-03	38194		
3853				REFER TO DETAIL C I-2007-03	SEE		
3854			BACK LIGHTED SPLIT SCREEN DISPLAY	REFER TO DETAIL D I-2007-03	38195		I-2145-55
3855		C51 R607D	RECORDER 2 PEN		SEE 38198 & 38199		I-2145-54
3856			BACK LIGHTED SPLIT SCREEN DISPLAY	REFER TO DETAIL C I-2007-03	SEE		I-2145-56
3857			BACK LIGHTED SPLIT SCREEN DISPLAY	REFER TO DETAIL D I-2007-03	38198		
3858			BACK LIGHTED SPLIT SCREEN DISPLAY	REFER TO DETAIL C I-2007-03	SEE	C51	
3859			BACK LIGHTED SPLIT SCREEN DISPLAY	REFER TO DETAIL D I-2007-03	38199	C51	I-2145-56

PANEL ITEM NO.	VALVE NO.	INSTRUM OR PIS NO.	DESCRIPTION	ENGRAVING	P&ID DWG	SYSTEM REF NO.	I&C DWG NO.
3860		C51 R607B	RECORDER 2 PEN		SEE 38196 & 38197		I-2145-54
3861			BACK LIGHTED SPLIT SCREEN DISPLAY	REFER TO DETAIL C I-2007-03	SEE		I-2145-56
3862				REFER TO DETAIL D I-2007-03	38196		
3863				REFER TO DETAIL C I-2007-03	SEE		
3864			BACK LIGHTED SPLIT SCREEN DISPLAY	REFER TO DETAIL D I-2007-03	38197		I-2145-56
3865		C51 R607C	RECORDER 2 PEN		SEE 38200 & 38201		I-2145-53
3866			BACK LIGHTED SPLIT SCREEN DISPLAY	REFER TO DETAIL C I-2007-03	SEE		I-2145-55
3867				REFER TO DETAIL D I-2007-03	38200		
3868				REFER TO DETAIL C I-2007-03	SEE		
3869			BACK LIGHTED SPLIT SCREEN DISPLAY	REFER TO DETAIL D I-2007-03	38201	C51	I-2145-55
3870							
3871							
3872							
3873							
3874							
3875							
3876							
3877							
3878		C51 R809A	APRM 1 & 3 DISPLAY			C51	
3879		C51 R809C	RBM 'A' DISPLAY			C51	
3880							
3881							

NOTES:

1. INFORMATION IN ENGRAVING COLUMN NOTED WITH AN ASTERISK (*) INDICATES LENS ENGRAVING ONLY
2. INFORMATION IN DESCRIPTION COLUMN NOTED "N/A" INDICATES ITEM NUMBER IS NOT APPLICABLE TO THIS PANEL

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FIGURE 7.5-1, SHEET 4
REACTOR CONTROL BOARD

PANEL ITEM NO.	VALVE NO.	INSTRUM OR FIS NO.	DESCRIPTION	ENGRAVING	P&ID DWG. NO.	SYSTEM REF. NO.	I&C DWG. NO.
3882		C11 R800	FLOW INDICATOR		SEE 3887	C11	I-2115-18
3883							
3884							
3885							
3886							
3887		X C11K612	CONTROLLER	CONTROL ROD DRIVE HYDRAULIC SYSTEM FLOW		C11	I-2115-18
3888							
3889			DIGITAL DISPLAY	TENS		C11	I-2115-10
3890			UNITS	TENS			I-2115-10
3891			UNITS	TENS			
3892			UNITS	TENS	SEE 38102		
3893			UNITS	TENS			
3894			UNITS	TENS			
3895			DIGITAL DISPLAY	UNITS			I-2115-10
3896		C11 J600	ROD WORTH MINIMIZER DISPLAY	ROD WORTH MINIMIZER		C11	I-2115-21
3898		C51 R809B	APRM 2 & 4 DISPLAY			C51	
3899		C51 R809D	RBM 'B' DISPLAY			C51	
38100							
38101							
38102			MARKER PLATE	SELECT ROD GROUP 4-ROD POSITION DISPLAY			
38103							
38104							
38105							
38106							
38107			N/A				
38108							
38109							
38110							
38111							
38112							
38113							
38114							
38115							
38116		C51 R603A	RECORDER	2 PEN		C51	
38117							
38118							

PANEL ITEM NO.	VALVE NO.	INSTRUM OR FIS NO.	DESCRIPTION	ENGRAVING	P&ID DWG. NO.	SYSTEM REF. NO.	I&C DWG. NO.
38119							
38120							
38121		C51 R603B	RECORDER	2 PEN		C51	
38122							
38123							
38124							
38125							
38126							
38127							
38128							
38129							
38130							
38131							
38132							
38133							
38134							
38135							
38136							
38137		B31 R617	FLOW INDICATOR	SEE I-2975-120		B31	I-2145-42
38138		B31 R623A	VOLTAGE INDICATOR	SEE I-2975-120		B31	I-2101-10
38139		B31 R627A	AMMETER	SEE I-2975-120		B31	I-2101-10
38140		B31 R624A	WATTMETER	SEE I-2975-120		B31	-11-
38141				N/A			
38142		B21 R611B	FLOW INDICATOR	SEE I-2975-120		B21	I-2095-25
38143		B21 R611A	FLOW INDICATOR	SEE I-2975-120		B21	-11-
38144							
38145		B31 R613	FLOW INDICATOR	SEE I-2975-120		B31	I-2145-42
38146		B31 R623B	VOLTAGE INDICATOR	SEE I-2975-120			I-2101-11
38147		B31 R627B	AMMETER	SEE I-2975-120			-11-
38148		B31 R624B	WATTMETER	SEE I-2975-120		B31	-11-
38149		C32 R606A	LEVEL INDICATOR	SEE I-2975-120		C32	I-2125-05
38150		C32 R606C	LEVEL INDICATOR	SEE I-2975-120			I-2125-03
38151		C32 R606B	LEVEL INDICATOR	SEE I-2975-120			I-2125-05
38152			2 POSITION SWITCH	A LEVEL B LEVEL	SEE I-2975-120	C32	I-2125-03
38153							
38154		C32 R607	RECORDER	2 PT	SEE I-2975-120	C32	I-2125-04
38155				N/A			

PANEL ITEM NO.	VALVE NO.	INSTRUM OR FIS NO.	DESCRIPTION	ENGRAVING	P&ID DWG. NO.	SYSTEM REF. NO.	I&C DWG. NO.
38156		B21 R609B	FLOW INDICATOR	SEE I-2975-120		B21	I-2095-25
38157		B21 R609D	FLOW INDICATOR	SEE I-2975-120			
38158		B21 R607A	FLOW INDICATOR	SEE I-2975-120			
38159		B21 R609C	FLOW INDICATOR	SEE I-2975-120		B21	I-2095-25
38160							
38161							
38162							
38163							
38164			BACK LIGHTED DISPLAY GREEN	I-2007-03 REFER TO DETAIL A			I-2145-56
38165							-11-
38166							I-2145-55
38167			BACK LIGHTED DISPLAY GREEN				-11-
38168		B21 R612	BACK LIGHTED PB WITH 3-WAY SPLIT SCREEN(W)				I-2145-63
38169							-11-
38170							I-2145-62
38171			BACK LIGHTED PB WITH 3-WAY SPLIT SCREEN(W)	I-2007-03 REFER TO DETAIL A		C51	I-2145-63
38172		B21 R612	DIFF. PRESS. INDICATOR	SEE I-2975-120		B21	I-2095-25
38173			BACK LIGHTED DISPLAY (WHITE)	SELECTED ROD		C11	I-2115-24
38174							-11-
38175							-11-
38176			BACK LIGHTED DISPLAY (WHITE)	SELECTED ROD		C11	I-2115-10
38177		B21 R605	LEVEL INDICATOR	REACTOR VESSEL FLOOD UP LEVEL		B21	I-2095-24

NOTES:

1. INFORMATION IN ENGRAVING COLUMN NOTED WITH AN ASTERISK (*) INDICATES LENS ENGRAVING ONLY
2. INFORMATION IN DESCRIPTION COLUMN NOTED "N/A" INDICATES ITEM NUMBER IS NOT APPLICABLE TO THIS PANEL

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FIGURE 7.5-1, SHEET 5
REACTOR CONTROL BOARD

PANEL ITEM NO.	VALVE NO.	INSTRUM. OR PIS NO.	DESCRIPTION	ENGRAVING	P&ID DWG NO.	SYSTEM REF. NO.	ISC DWG NO.
3B178							
3B179							
3B180							
3B181							
3B182							
3B183							
3B184							
3B185							
3B186							
3B187							
3B188							
3B189							
3B190			MARKER PLATE	SRM 'A' BLUE PEN			
3B191				SRM 'B' RED PEN			
3B192				SRM 'C' BLUE PEN			
3B193				SRM 'D' RED PEN			
3B194				IRM 'A' BLUE PEN			
3B195				IRM 'C' RED PEN			
3B196				IRM 'D' BLUE PEN			
3B197				IRM 'B' RED PEN			
3B198				IRM 'H' BLUE PEN			
3B199				IRM 'F' RED PEN			
3B200				IRM 'E' BLUE PEN			
3B201			MARKER PLATE	IRM 'G' RED PEN			
3B210			PUSHBUTTON (RED)	MANUAL SCRAM TRIP A2		C71	2155-08A
3B211			PUSHBUTTON (RED)	MANUAL SCRAM TRIP B2			2155-09A
3B212			RECORDER COVER RESTRAINT				2007-15
3C24			BACKLIGHTEED DISPLAY GROUP (POWER ON/OFF)			C71	2155-10
3C25							
3C26							
3C27							
3C28							
3C29							
3C30							
3C31			BACKLIGHTEED DISPLAY GROUP (POWER ON/OFF)			C71	2155-10
3C32			MARKER PLATE	SEE I-2975-125			
3C33			MARKER PLATE	SEE I-2975-125			
3C34			MARKER PLATE	SEE I-2975-125			
3C35							
3C36							
3C37							
3C38			FLUSH MOUNT HI-COM HAND SET			H40	

PANEL ITEM NO.	VALVE NO.	INSTRUM. OR PIS NO.	DESCRIPTION	ENGRAVING	P&ID DWG NO.	SYSTEM REF. NO.	ISC DWG NO.
3A207	VB-3710	B31-F014A	BACKLIGHTEED PB OPEN (RED)		GM-2833	B31	2105-13
3A208	--	--	BACKLIGHTEED PB CLOSE (GRN)	SEE I-2975-116			--
3A209	VB-3596	B31-F014B	BACKLIGHTEED PB OPEN (RED)				2105-14
3A210	--	--	BACKLIGHTEED PB CLOSE (GRN)	SEE I-2975-116			--
3A211	VB-3767	B31-F016A	BACKLIGHTEED PB OPEN (RED)				2105-13
3A212	--	--	BACKLIGHTEED PB CLOSE (GRN)	SEE I-2975-116			--
3A213	VB-3768	B31-F016B	BACKLIGHTEED PB OPEN (RED)				2105-14
3A214	--	--	BACKLIGHTEED PB CLOSE (GRN)	SEE I-2975-116	GM-2833	B31	--
3A215			BACKLIGHTEED DISPLAY CLOSED (RED)	SEE I-2975-116		B31	2105-16
3A216			BACKLIGHTEED DISPLAY TRIPPED (RM)	SEE I-2975-116		--	--
3A217			START RUN 2 POSITION SW	SEE I-2975-116		C32	2125-04
3A218			N/A				
3A220		C11-F160B	BACKLIGHTEED DISPLAY OPEN (RED)		GM-2081	C11	2106-17
3A221	--	--	BACKLIGHTEED DISPLAY CLOSED (GRN)	C11-F160B			
3A222		C11-F162B	BACKLIGHTEED DISPLAY OPEN (RED)				
3A223	--	--	BACKLIGHTEED DISPLAY CLOSED (GRN)	C11-F162B			
3A224		C11-F162D	BACKLIGHTEED DISPLAY OPEN (RED)	SEE 3A228			
3A225	--	--	BACKLIGHTEED DISPLAY CLOSED (GRN)	C11-F162D			
3A226		C11-F163B	BACKLIGHTEED DISPLAY OPEN (RED)				
3A227	--	--	BACKLIGHTEED DISPLAY CLOSED (GRN)	C11-F163B			
3A228			MARKER PLATE	ATWS ARI/RPT DIV. II			
3C1		C41R600	PRESSURE INDICATOR 1 PT	STANDBY LIQUID PUMPS A&B DISCHARGE PRESS.		C81	2135-01
3C2		C11R601	PRESSURE INDICATOR 1 PT	CONTROL ROD DRIVE CHARGE WATER PRESSURE		C11	2115-18
3C3		C96R800B	DISPLAY MONITOR			C96	
3C4			ROD DISPLAY INSERT			C11	2115-29
3C5							
3C6							
3C7		B31R629A	PRESSURE INDICATOR 2 PT	SEE I-2975-125		B31	2095-24
3C8		B31R629B	PRESSURE INDICATOR 2 PT	SEE I-2975-125		B31	--
3C9			N/A				
3C10		B31R601	TEMPERATURE RECORDER 12 POINTS	SEE I-2975-125		B31	2105-01
3C11		C32R605A	PRESSURE INDICATOR 1 PT	SEE I-2975-125		C32	2125-03
3C12		B21R613	RECORDER 2 PT.	SEE I-2975-125		B21	2095-25
3C13		C32R609	RECORDER 2 PT.	SEE I-2975-125		C32	2125-03

PANEL ITEM NO.	VALVE NO.	INSTRUM. OR PIS NO.	DESCRIPTION	ENGRAVING	P&ID DWG NO.	SYSTEM REF. NO.	ISC DWG NO.
3C14		B31R614	RECORDER 2 PT.	SEE I-2975-125		B31	2145-30
3C15		B31R633A	PRESSURE INDICATOR 2 PT	SEE I-2975-125		B31	2095-24
3C16		B31R633B	PRESSURE INDICATOR 2 PT	SEE I-2975-125		B31	--
3C17		C32R605B	PRESSURE INDICATOR 2 PT	SEE I-2975-125		C32	2125-03
3C18		B31R650	TEMPERATURE RECORDER 2 PT.	SEE I-2975-125		B31	2095-24
3C19			N/A				
3C20		B21R007	RECORDER 2-PEN	REACTOR VESSEL TEMPERATURE RED-FLANGE BLUE-GRN		B21	2095-23
3C21							
3C22			N/A				
3C23							
6030830			ANNUNCIATOR LIGHT BOX 24 WINDOWS	SEE REF DWG NO. I-2083-06			
6030831				SEE REF DWG NO. I-2083-06			
6030832				SEE REF DWG NO. I-2083-07			
6030833				SEE REF DWG NO. I-2083-07			
6030834				SEE REF DWG NO. I-2083-08			
6030835				SEE REF DWG NO. I-2083-08			
6030836			ANNUNCIATOR LIGHT BOX 24 WINDOWS	SEE REF DWG NO. I-2083-09			

321A		COMMUNICATION JACK			H40
322A		COMMUNICATION JACK			H40

321		PUSHBUTTON	ANNUNCIATOR ACKNOWLEDGE	▲	
322		PUSHBUTTON	ANNUNCIATOR SILENCE	▲	
323		PUSHBUTTON	ANNUNCIATOR ACKNOWLEDGE	▲	
324		PUSHBUTTON	ANNUNCIATOR SILENCE	▲	

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FIGURE 7.5-1, SHEET 6
REACTOR CONTROL BOARD

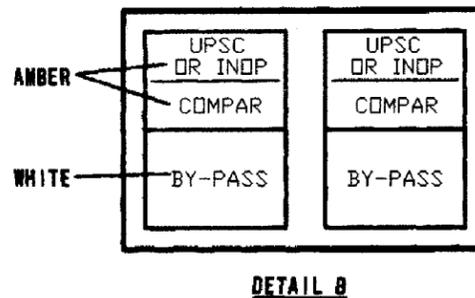
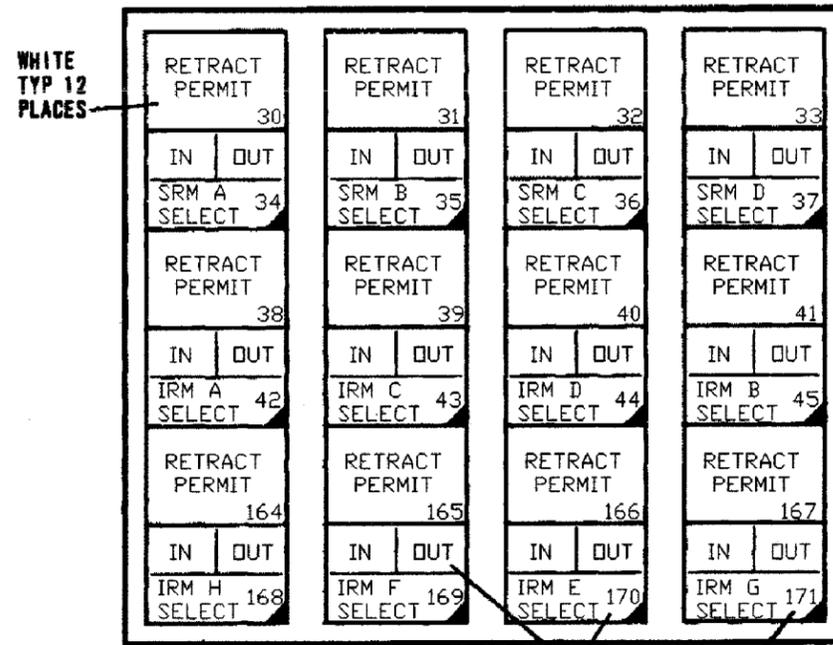
NOTES:

1. INFORMATION IN DESCRIPTION COLUMN NOTED "N/A" INDICATES ITEM NUMBER IS NOT APPLICABLE TO THIS PANEL.
2. PANEL ITEMS, MARKED WITH A ▲, LOCATIONS ARE SHOWN ON I-2007-21.

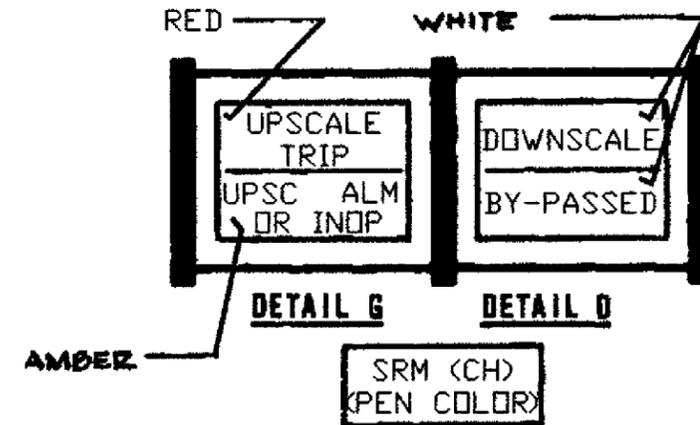
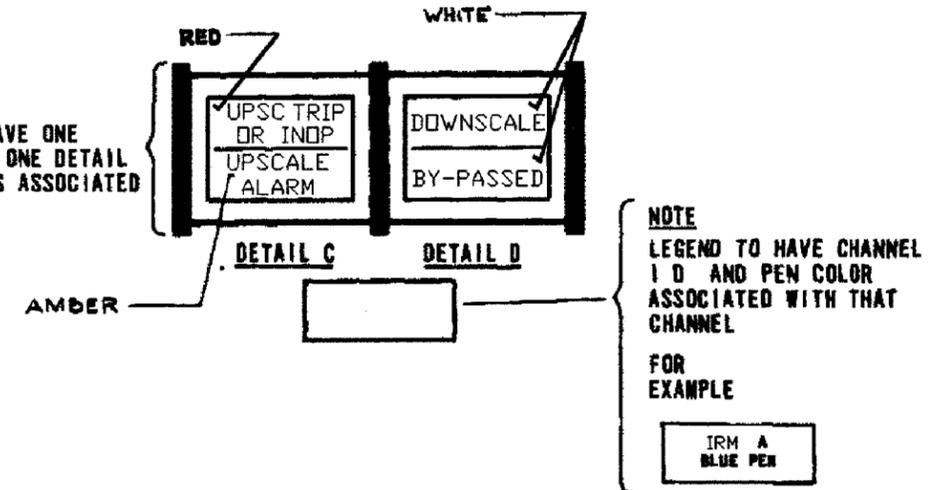
PANEL ITEM NO.	VALVE NO.	PIS NO.	DESCRIPTION	ENGRAVING	P&ID DWG NO.	SYSTEM	I&C DWG NO.
3A300							
3A301		C32R618	CONTROL STATION	SEE I-2975-116		C32	
3A302		C32R616A	CONTROL STATION	SEE I-2975-116		C32	
3A303		N21R658A	CONTROL STATION	SEE I-2975-116		C32	
3A304		C32R616B	CONTROL STATION	SEE I-2975-116		C32	
3A305		N21R658B	CONTROL STATION	SEE I-2975-116		C32	
3A306		N21R605	INDICATOR	SEE I-2975-116		C32	
3A307		C32R620	CONTROL STATION	SEE I-2975-116		C32	
3A308			BACKLIGHTED DISPLAY	POST SCRAM		C32	
3A309			PUSHBUTTON	RESET		C32	
3A310			BACKLIGHTED DISPLAY	POST SCRAM		C32	
3A311			PUSHBUTTON	RESET		C32	
3A312							
3A313			BACKLIGHTED SPLIT SCREEN DISPLAY	LIMITER 1 LIMITER 4		B31	
3A314							
3A315			BACKLIGHTED SPLIT SCREEN DISPLAY	LIMITER 1 LIMITER 4		B31	
3A316							
3A317			BACKLIGHTED DISPLAY	RUNBACK		B31	
3A318		B31R621A	CONTROL STATION	SEE I-2975-116		B31	
3A319		B31R621B	CONTROL STATION	SEE I-2975-116		B31	

PANEL ITEM NO.	VALVE NO.	PIS NO.	DESCRIPTION	ENGRAVING	P&ID DWG NO.	SYSTEM	I&C DWG NO.
3B300			FLAT PANEL DISPLAY			C32	
3B301		C32R606D	LEVEL INDICATOR	SEE I-2975-120		C32	

PANEL ITEM NO.	VALVE NO.	PIS NO.	DESCRIPTION	ENGRAVING	P&ID DWG NO.	SYSTEM	I&C DWG NO.
3C300		C32R614	RECORDER	SEE I-2975-125		C32	
3C301		C32R604A	FLOW INDICATOR	SEE I-2975-125		C32	
3C302		C32R604B	FLOW INDICATOR	SEE I-2975-125		C32	
3C303		C96R800C	DISPLAY MONITOR			C96	
3C304			PIEZO ALARM			C97	
3C305			BACKLIGHTED DISPLAY	TOTAL SYSTEM FAILURE		C97	
3C306			PUSHBUTTON			C97	



NOTE
EACH IRM CHANNEL TO HAVE ONE
DETAIL 'C' & ONE DETAIL
'D' UNDER ITS ASSOCIATED
RECORDER



NOTES:

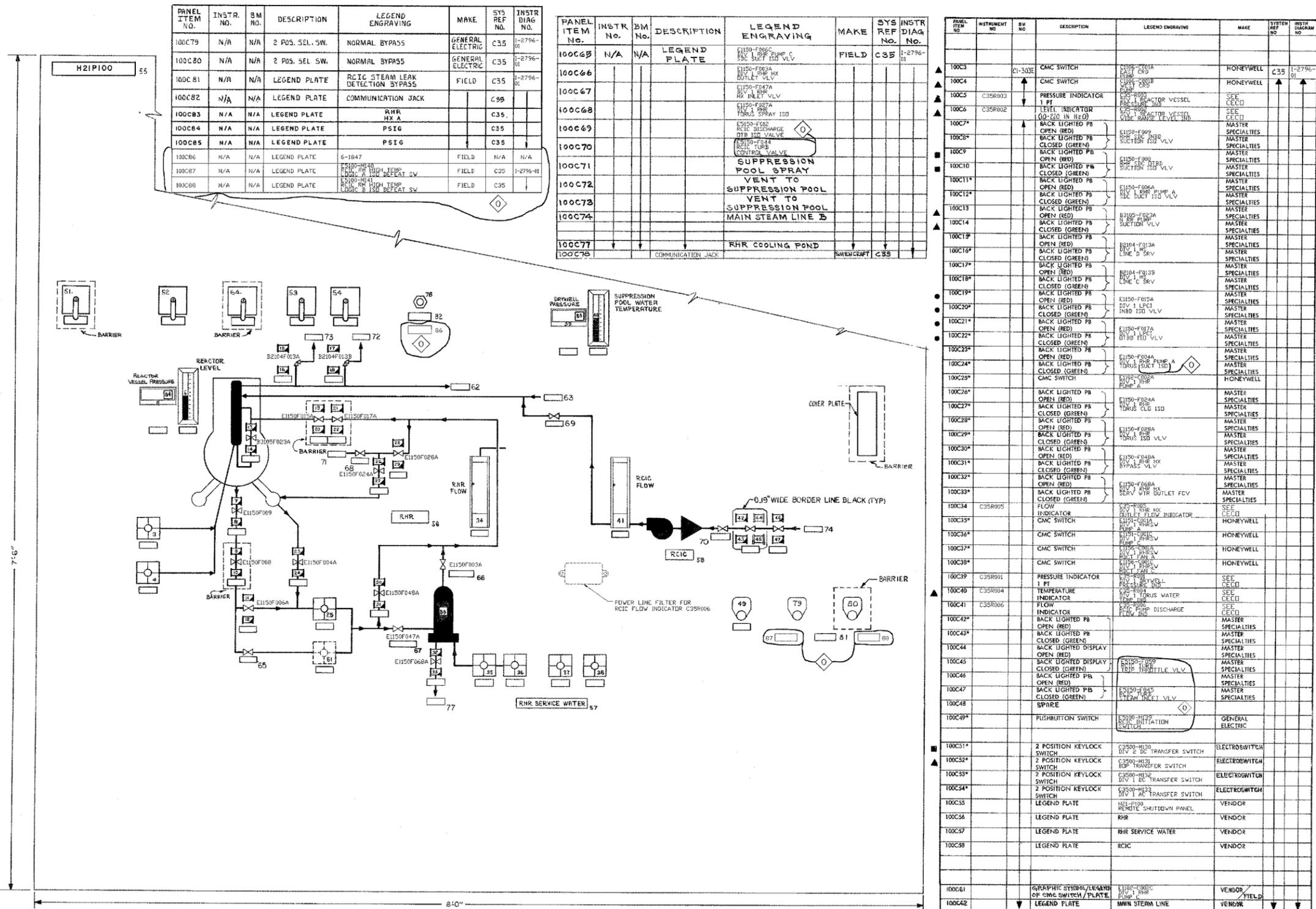
1. INFORMATION IN DESCRIPTION COLUMN NOTED "N/A" INDICATES ITEM NUMBER IS NOT APPLICABLE TO THIS PANEL.
2. PANEL ITEMS, MARKED WITH A TRIANGLE ▲, LOCATIONS ARE SHOWN ON I-2007-21.

REFERENCE DRAWINGS

- SEE I-2007-02 FOR CONTINUATION OF EQUIPMENT LIST
- SEE DRAWING I-2007-01 FOR PANEL LAYOUT

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FIGURE 7.5-1, SHEET 7
REACTOR CONTROL BOARD



PANEL ITEM NO.	INSTR. NO.	BM NO.	DESCRIPTION	LEGEND ENGRAVING	MAKE	SYS REF NO.	INSTR DIAG NO.
100C79	N/A	N/A	2 POS. 5EL. SW.	NORMAL BYPASS	GENERAL ELECTRIC	C35	I-2796-01
100C80	N/A	N/A	2 POS. 5EL. SW.	NORMAL BYPASS	GENERAL ELECTRIC	C35	I-2796-01
100C81	N/A	N/A	LEGEND PLATE	RCIC STEAM LEAK DETECTION BYPASS	FIELD	C35	I-2796-01
100C82	N/A	N/A	LEGEND PLATE	COMMUNICATION JACK	C98		
100C83	N/A	N/A	LEGEND PLATE	RHR HX A	C35		
100C84	N/A	N/A	LEGEND PLATE	PSIG	C35		
100C85	N/A	N/A	LEGEND PLATE	PSIG	C35		
100C86	N/A	N/A	LEGEND PLATE	6-1847	FIELD	N/A	N/A
100C87	N/A	N/A	LEGEND PLATE	RHR HIGH TEMP 100C 2 1SD DEFEAT SV	FIELD	C35	I-2796-01
100C88	N/A	N/A	LEGEND PLATE	RHR HIGH TEMP 100C 2 1SD DEFEAT SV	FIELD	C35	I-2796-01

PANEL ITEM NO.	INSTR. NO.	BM NO.	DESCRIPTION	LEGEND ENGRAVING	MAKE	SYS REF NO.	INSTR DIAG NO.
100C65	N/A	N/A	LEGEND PLATE	E1150-F066C RHR PUMP C SUC SUPT ISD VLV	FIELD	C35	I-2796-01
100C66				E1150-F062A DIV 1 RHR HX HOTLEY VLV			
100C67				E1150-F047A RHR HX HX INLET VLV			
100C68				E1150-F027A RHR HX TORUS SPRAY ISD			
100C69				E1150-F012 RCIC DISCHARGE DTB ISD VALVE			
100C70				E1150-F011 RCIC TURB DISCHARGE VALVE			
100C71				SUPPRESSION POOL SPRAY VENT TO SUPPRESSION POOL			
100C72				SUPPRESSION POOL VENT TO SUPPRESSION POOL			
100C73				MAIN STEAM LINE			
100C74							
100C77				RHR COOLING POND	SWEMCRAFT	C35	
100C78				COMMUNICATION JACK			

PANEL ITEM NO.	INSTRUMENT NO.	BM NO.	DESCRIPTION	LEGEND ENGRAVING	MAKE	SYS REF NO.	INSTR DIAG NO.
100C3			CMC SWITCH	E1150-F001A RHR HX	HONEYWELL	C35	I-2796-01
100C4			CMC SWITCH	E1150-F001B RHR HX	HONEYWELL		
100C5	C35R003		PRESSURE INDICATOR	E1150-F003 RHR REACTOR VESSEL	SEE CECD		
100C6	C35R002		LEVEL INDICATOR	E1150-F004 RHR REACTOR VESSEL	SEE CECD		
100C7*			BACK LIGHTED PB	E1150-F009 RHR HX INSD	MASTER SPECIALTIES		
100C8*			BACK LIGHTED PB	E1150-F009 RHR HX INSD	MASTER SPECIALTIES		
100C9			BACK LIGHTED PB	E1150-F008 RHR HX INSD	MASTER SPECIALTIES		
100C10			BACK LIGHTED PB	E1150-F008 RHR HX INSD	MASTER SPECIALTIES		
100C11*			BACK LIGHTED PB	E1150-F006A RHR HX INSD	MASTER SPECIALTIES		
100C12*			BACK LIGHTED PB	E1150-F006A RHR HX INSD	MASTER SPECIALTIES		
100C13			BACK LIGHTED PB	E1150-F006A RHR HX INSD	MASTER SPECIALTIES		
100C14			BACK LIGHTED PB	E1150-F006A RHR HX INSD	MASTER SPECIALTIES		
100C15*			BACK LIGHTED PB	E1150-F006A RHR HX INSD	MASTER SPECIALTIES		
100C16*			BACK LIGHTED PB	E1150-F013A RHR HX INSD	MASTER SPECIALTIES		
100C17*			BACK LIGHTED PB	E1150-F013A RHR HX INSD	MASTER SPECIALTIES		
100C18*			BACK LIGHTED PB	E1150-F013B RHR HX INSD	MASTER SPECIALTIES		
100C19*			BACK LIGHTED PB	E1150-F013A RHR HX INSD	MASTER SPECIALTIES		
100C20*			BACK LIGHTED PB	E1150-F013A RHR HX INSD	MASTER SPECIALTIES		
100C21*			BACK LIGHTED PB	E1150-F013A RHR HX INSD	MASTER SPECIALTIES		
100C22*			BACK LIGHTED PB	E1150-F013A RHR HX INSD	MASTER SPECIALTIES		
100C23*			BACK LIGHTED PB	E1150-F004A RHR HX INSD	MASTER SPECIALTIES		
100C24*			BACK LIGHTED PB	E1150-F004A RHR HX INSD	MASTER SPECIALTIES		
100C25*			CMC SWITCH	E1150-F004A RHR HX INSD	HONEYWELL		
100C26*			BACK LIGHTED PB	E1150-F024A RHR HX INSD	MASTER SPECIALTIES		
100C27*			BACK LIGHTED PB	E1150-F024A RHR HX INSD	MASTER SPECIALTIES		
100C28*			BACK LIGHTED PB	E1150-F024A RHR HX INSD	MASTER SPECIALTIES		
100C29*			BACK LIGHTED PB	E1150-F024A RHR HX INSD	MASTER SPECIALTIES		
100C30*			BACK LIGHTED PB	E1150-F024A RHR HX INSD	MASTER SPECIALTIES		
100C31*			BACK LIGHTED PB	E1150-F048A RHR HX INSD	MASTER SPECIALTIES		
100C32*			BACK LIGHTED PB	E1150-F048A RHR HX INSD	MASTER SPECIALTIES		
100C33*			BACK LIGHTED PB	E1150-F048A RHR HX INSD	MASTER SPECIALTIES		
100C34	C35R005		FLOW INDICATOR	E1150-F008 RHR HX INSD	SEE CECD		
100C35*			CMC SWITCH	E1150-F008 RHR HX INSD	HONEYWELL		
100C36*			CMC SWITCH	E1150-F008 RHR HX INSD	HONEYWELL		
100C37*			CMC SWITCH	E1150-F008 RHR HX INSD	HONEYWELL		
100C38*			CMC SWITCH	E1150-F008 RHR HX INSD	HONEYWELL		
100C39	C35R001		PRESSURE INDICATOR	E1150-F008 RHR HX INSD	SEE CECD		
100C40	C35R004		TEMPERATURE INDICATOR	E1150-F008 RHR HX INSD	SEE CECD		
100C41	C35R006		FLOW INDICATOR	E1150-F008 RHR HX INSD	SEE CECD		
100C42*			BACK LIGHTED PB	E1150-F008 RHR HX INSD	MASTER SPECIALTIES		
100C43*			BACK LIGHTED PB	E1150-F008 RHR HX INSD	MASTER SPECIALTIES		
100C44			BACK LIGHTED DISPLAY	E1150-F008 RHR HX INSD	MASTER SPECIALTIES		
100C45			BACK LIGHTED DISPLAY	E1150-F008 RHR HX INSD	MASTER SPECIALTIES		
100C46			BACK LIGHTED PB	E1150-F008 RHR HX INSD	MASTER SPECIALTIES		
100C47			BACK LIGHTED PB	E1150-F008 RHR HX INSD	MASTER SPECIALTIES		
100C48			SPARE				
100C49*			PUSHBUTTON SWITCH	E1150-F008 RHR HX INSD	GENERAL ELECTRIC		
100C51*			2 POSITION KEYLOCK SWITCH	C3500-M130 DIV 2 DC TRANSFER SWITCH	ELECTROSWITCH		
100C52*			2 POSITION KEYLOCK SWITCH	C3500-M130 DIV 2 DC TRANSFER SWITCH	ELECTROSWITCH		
100C53*			2 POSITION KEYLOCK SWITCH	C3500-M130 DIV 2 DC TRANSFER SWITCH	ELECTROSWITCH		
100C54*			2 POSITION KEYLOCK SWITCH	C3500-M130 DIV 2 DC TRANSFER SWITCH	ELECTROSWITCH		
100C55			LEGEND PLATE	H2I-1109 REMOTE SHUTDOWN PANEL	VENDOR		
100C56			LEGEND PLATE	RHR	VENDOR		
100C57			LEGEND PLATE	RHR SERVICE WATER	VENDOR		
100C58			LEGEND PLATE	RCIC	VENDOR		
100C61			GRAPHIC SYMBOL/LEGEND OF CMC SWITCHES/PLATE	E1150-F008 RHR HX INSD	VENDOR	FIELD	
100C62			LEGEND PLATE	MAIN STEAM LINE	VENDOR		
100C65	CI-303E		LEGEND PLATE	FEEDWATER LINE	VENDOR	C35	I-2796-01
100C64	CI-303E		2 POSITION KEYLOCK SWITCH	C3500-M130 DIV 2 DC TRANSFER SWITCH	ELECTROSWITCH	C35	I-2796-01

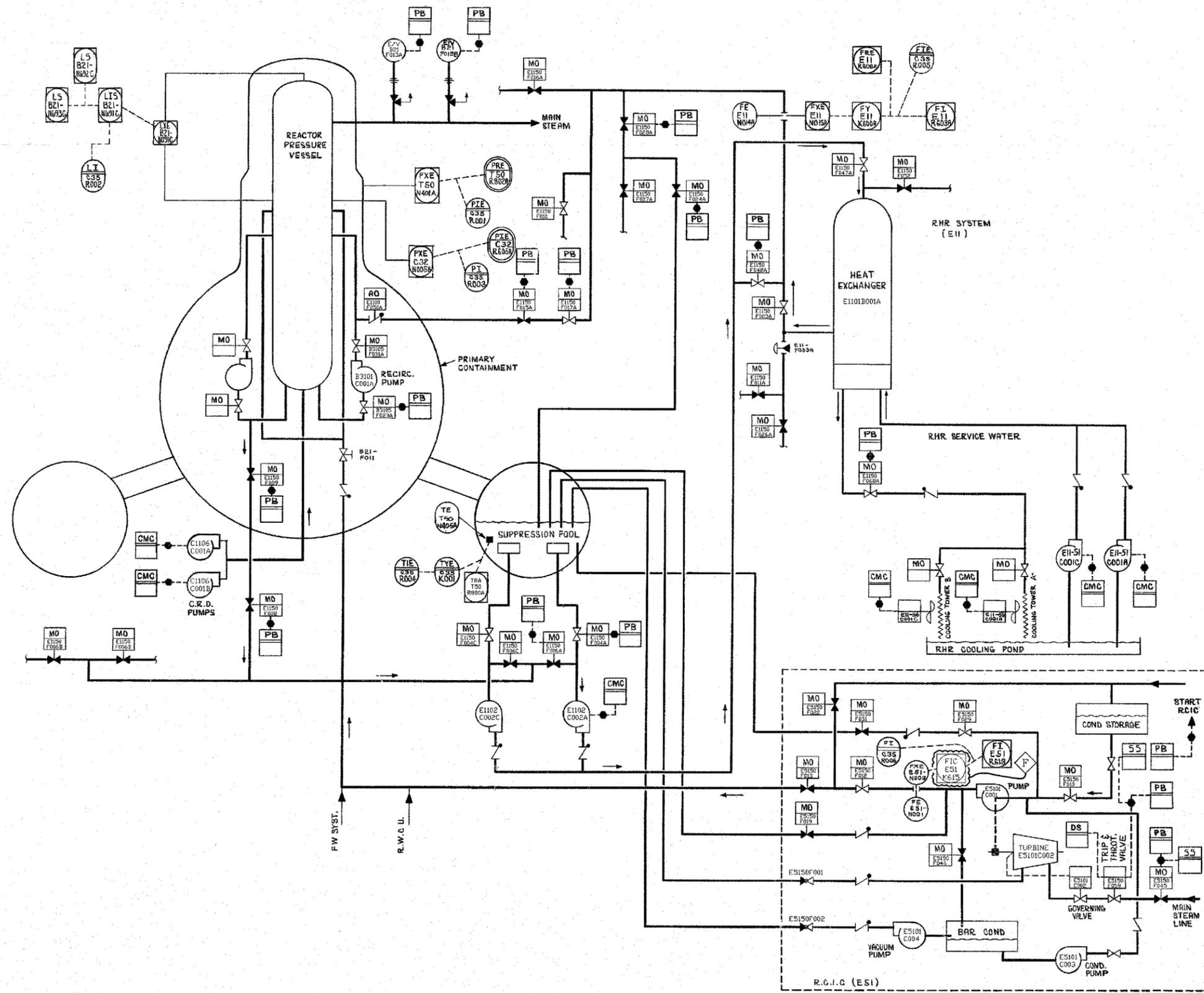
NOTES
ALL PANEL ITEMS MARKED WITH AN ASTERISK * ARE DESIGNATED CLASS I.E.
▲ DESIGNATES BOP ITEMS
■ DESIGNATES DAVI ITEMS
● DESIGNATES SHAW BUS ITEMS

SYMBOL LEGEND
 BACK LIGHTED PB
 BACK LIGHTED DISPLAY
 BACK LIGHTED SPLIT SCREEN DISPLAY
 LEGEND PLATE

REFERENCE DOCUMENTS
SYSTEM DIAGRAM I-2796-01
FOR PANEL LOCATION SEE A-2001
(REACTOR BLDG. 2ND FLOOR - SWITCH GEAR RM)
EMBEDMENT DETAILS - E-2832-05
FOR APPLICABLE CODES, REFER TO BM CI-303E

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FIGURE 7.5-2
**INSTRUMENTATION AND CONTROL ON
 REMOTE SHUTDOWN PANEL**



REFERENCE DRAWINGS

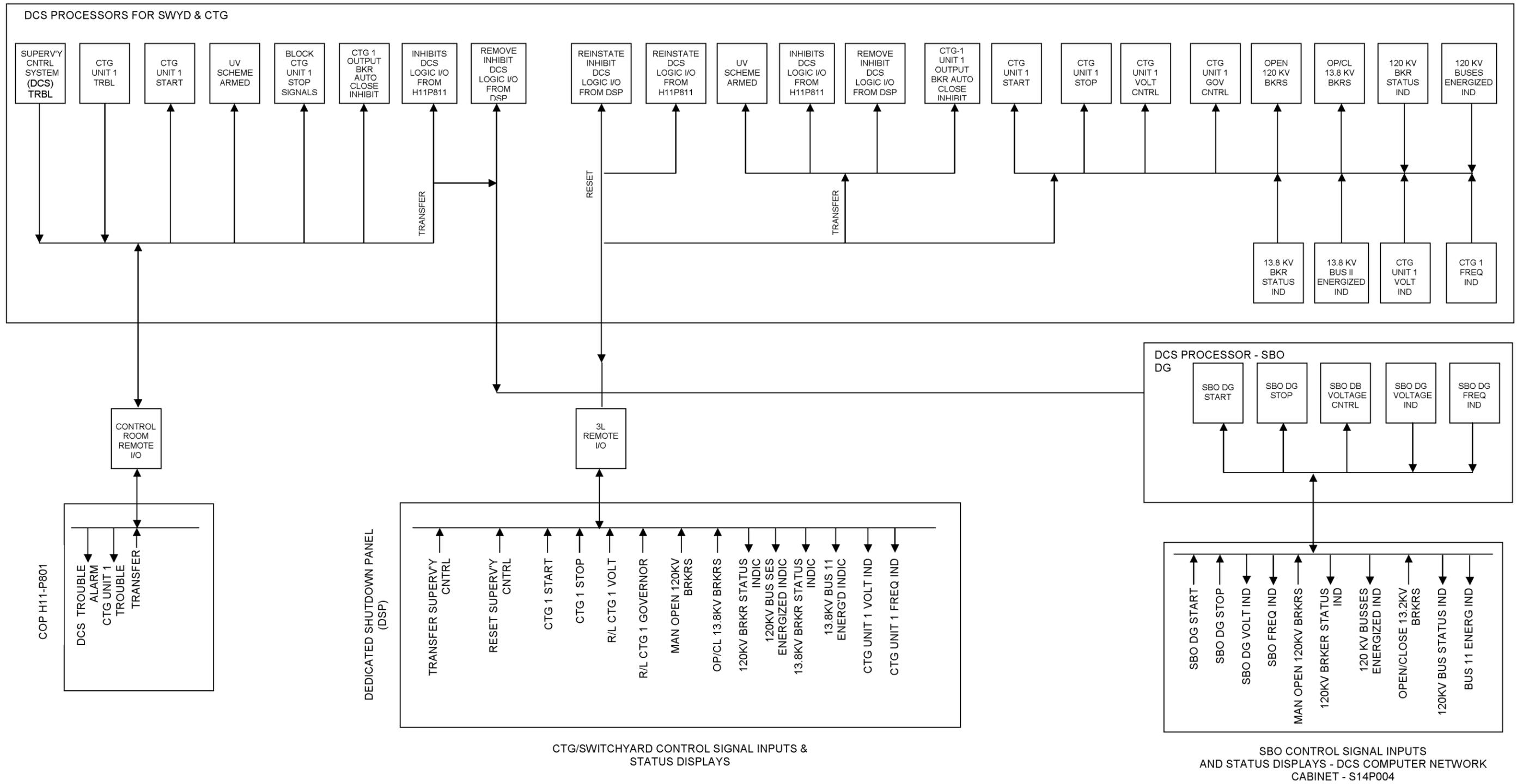
REF. No.	SYST. No.	DESCRIPTION	DWG. No.
1	C11	CONTROL ROD DRIVE HYDRAULIC SYSTEM	6M 721-2081
2	E11	RHR DIV. I SYSTEM DIAGRAM	6M 721-2084
3	E11	RHR DIV. II SYSTEM DIAGRAM	6M 721-2083
4	E41	HPCI SYSTEM DIAGRAM	6M 721-2035
5	E41	HPCI BARROMETRIC CONDENSER	6M 721-2043
6	E51	RCIC SYSTEM DIAGRAM	6M 721-2044
7	E51	" " " "	6M 721-2045
8	B21	NUCLEAR BOILER SYSTEM	6M 721-2069
9	B21	" " " "	6M 721-2090
10	W21	FEEDWATER SYSTEM	6M 721-2028
11	B31	REACTOR RECIRCULATION SYSTEM	6M 721-2033
12	P11	CONDENSATE STORAGE SYSTEM	6M 721-2006
13	T50	PRIMARY CONTAINMENT MONITORING SYST.	6T 721-2679-1
14	E11	RHR SERVICE WATER SYSTEM	6M 721-2052
15	-	LEGEND OF SYSTEM AND INST. IDENT.	6M 721-2001

- NOTES:**
- SYSTEM NUMBER IS C35
 - SYMBOLS AS FOLLOWS DENOTE EQUIPMENT MOUNTED ON REMOTE SHUTDOWN CONTROL PANEL:

 - SYMBOL ● DENOTES TRANSFER RELAY CONTACTS
 - REFER TO DI-117 REMOTE SHUTDOWN SYSTEM.
 - FOR QA LEVELS, REFER TO VARIOUS SYSTEM DIAGRAMS AS INDICATED UNDER REFERENCE DRAWINGS
 - FOR DETAILS OF REMOTE SHUTDOWN PANEL, REFER TO BM: CI-303E
 - FOR PANEL ARRANGEMENT AND DESCRIPTIONS OF EQUIPMENT, SEE 6T721-2791-1
 - FOR PANEL WIRING SEE 6T721-2793-1&2
 - FOR ANALOG CHANNEL SCHEMATIC DIAGRAMS SEE 6T721-2795-1
 - THE PURPOSE OF THIS DIAGRAM IS TO SHOW THE INTERFACE OF THE REMOTE SHUTDOWN PANEL WITH THE EXISTING PLANT SYSTEMS AND IS NOT INTENDED AS A GOVERNING DOCUMENT OVER THOSE SYSTEMS. REFER TO THE ABOVE TABLE OF REFERENCE DWGS. FOR THE GOVERNING DOCUMENTS OF THE INDIVIDUAL SYSTEMS.

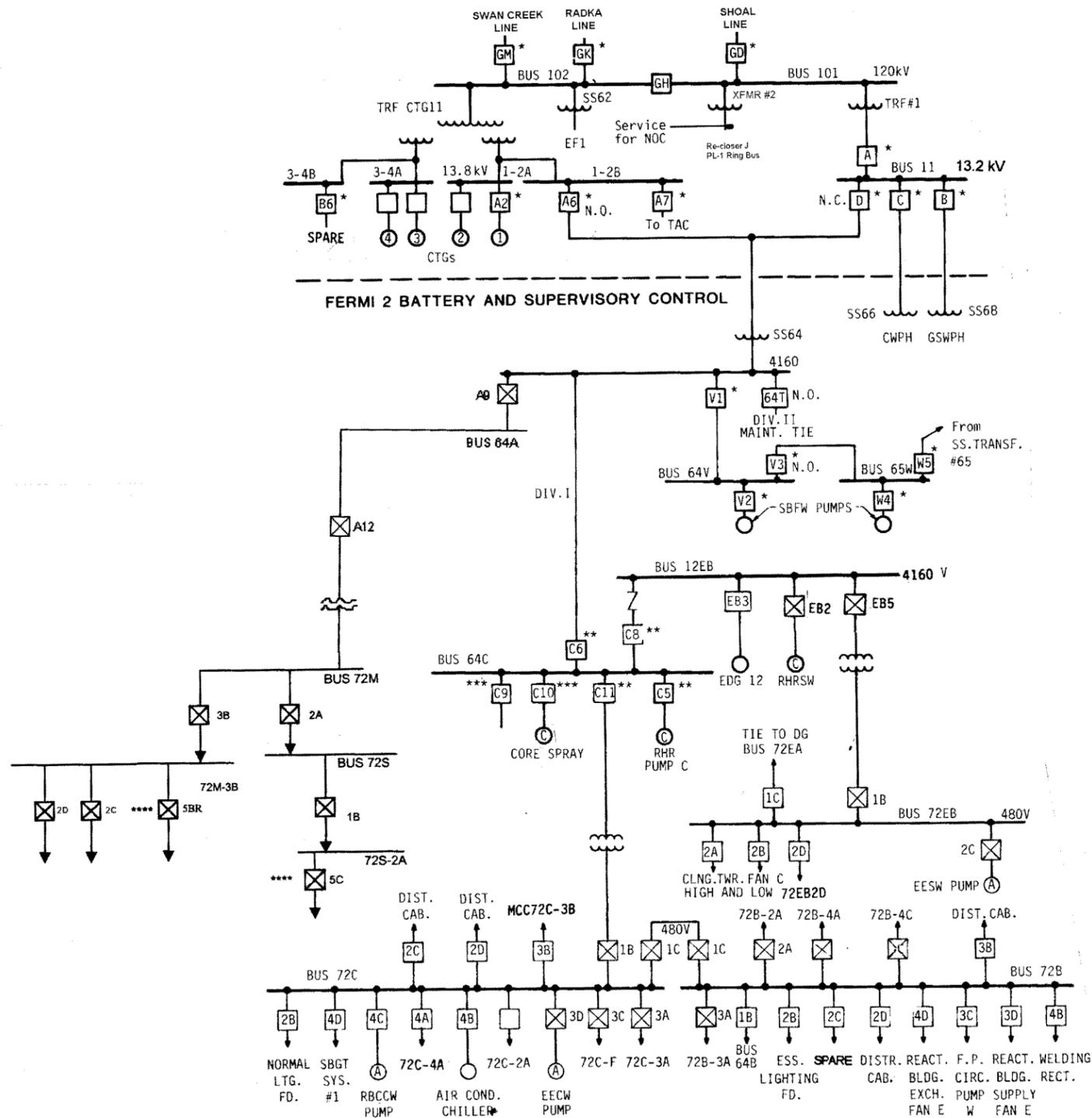
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FIGURE 7.5-3
REMOTE SHUTDOWN SYSTEM



NOTE: DCS PROCESSORS FOR SWYD, CTG AND SBO DG ARE LINKED FOR DATA TRANSFER

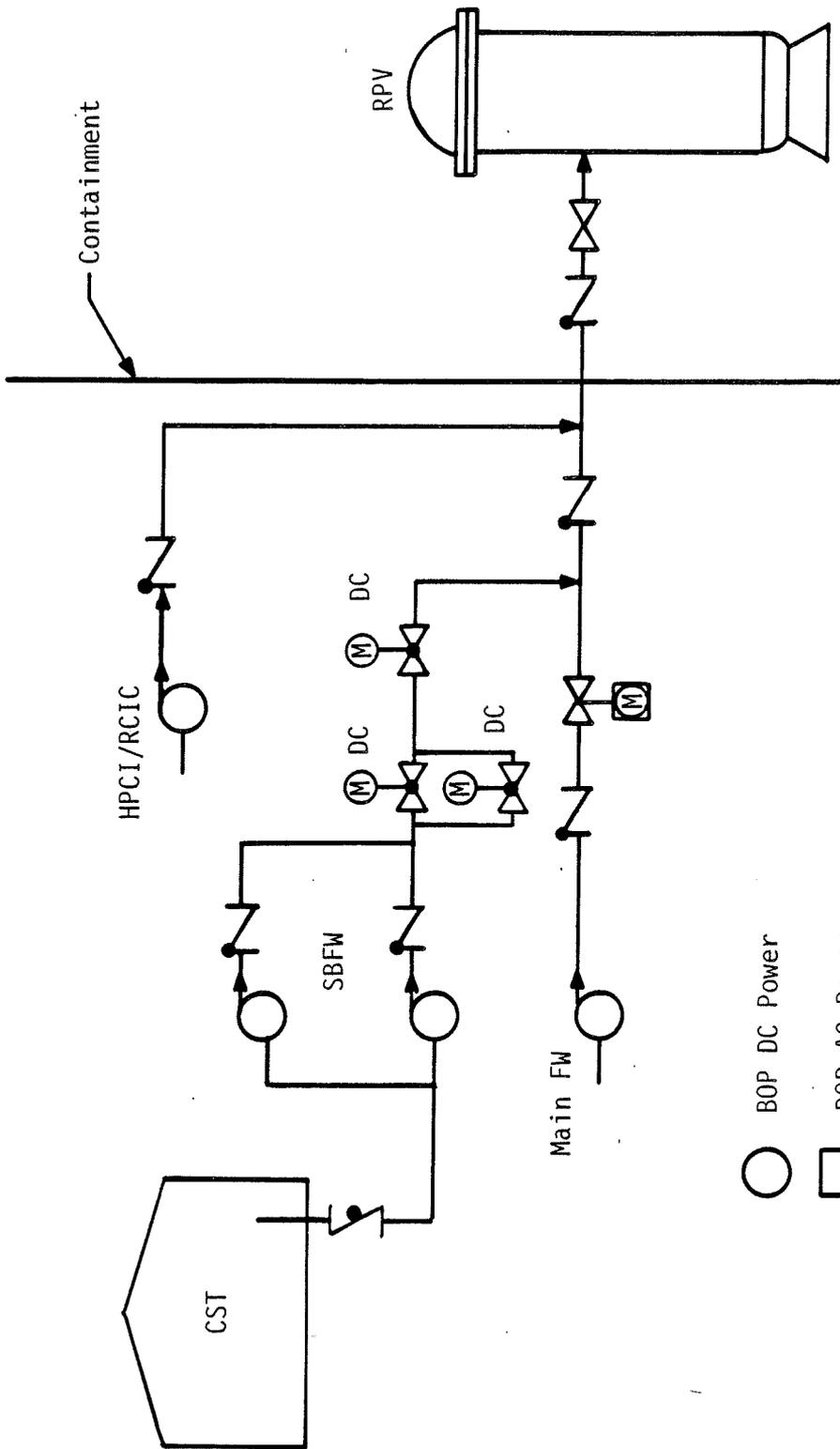
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 FIGURE 7.5-4
 SUPERVISORY AND TELEMETERING CONTROL BLOCK DIAGRAM



- * Transfer with control and indication at dedicated shutdown panel.
- ** Transfer with local control and indication in DIV. 1 SWGR room.
- *** Transfer and control in DIV. 1 SWGR room (can take local control to disable only).
- **** Transfer with local control at MCC compartment
- ☒ Breaker locally closed if open.

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FIGURE 7.5-5
 ONE-LINE DIAGRAM - ALTERNATE SHUTDOWN SYSTEM

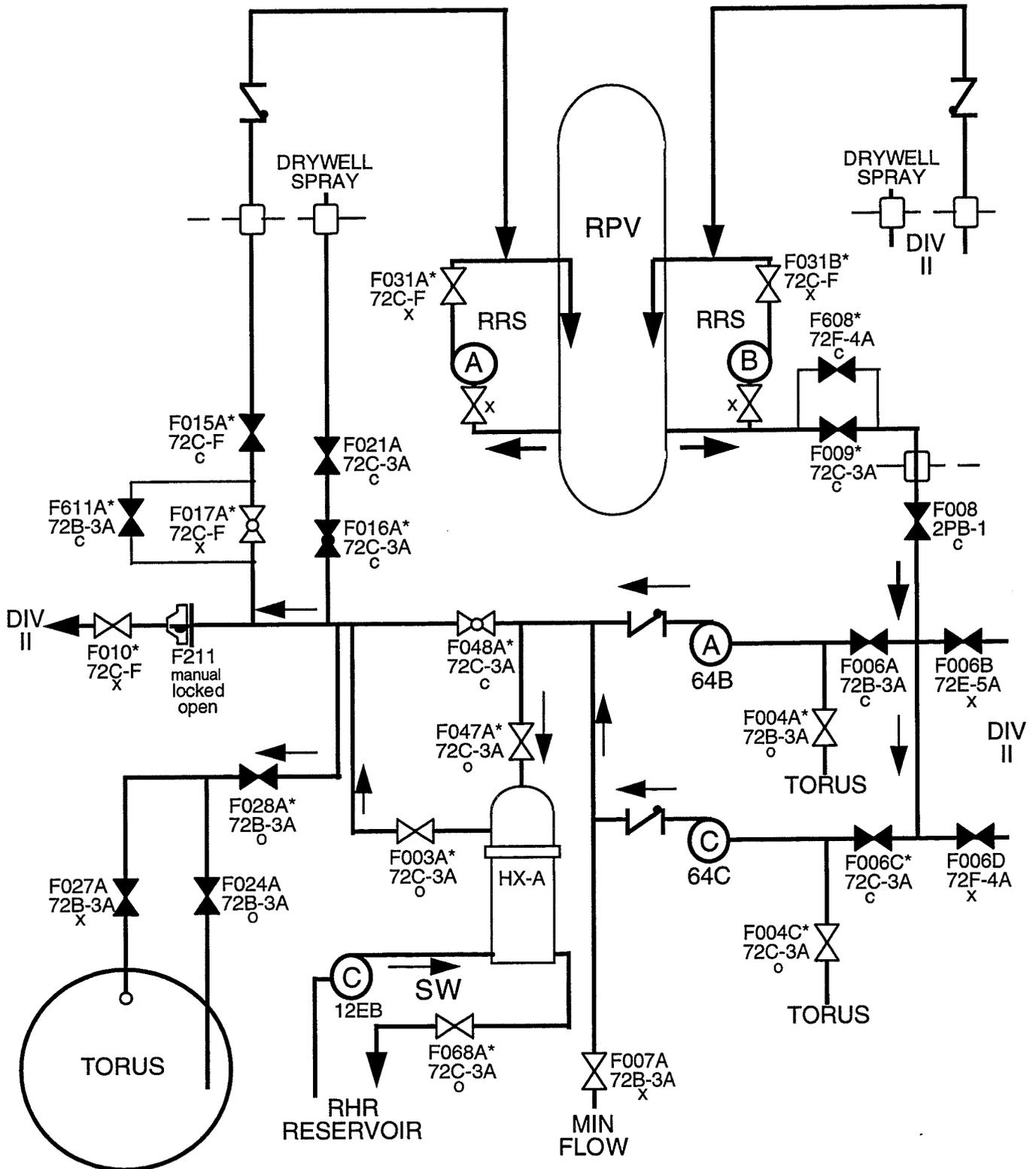


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FIGURE 7.5-6

STANDBY FEEDWATER SYSTEM
SIMPLIFIED DIAGRAM



TORUS COOLING

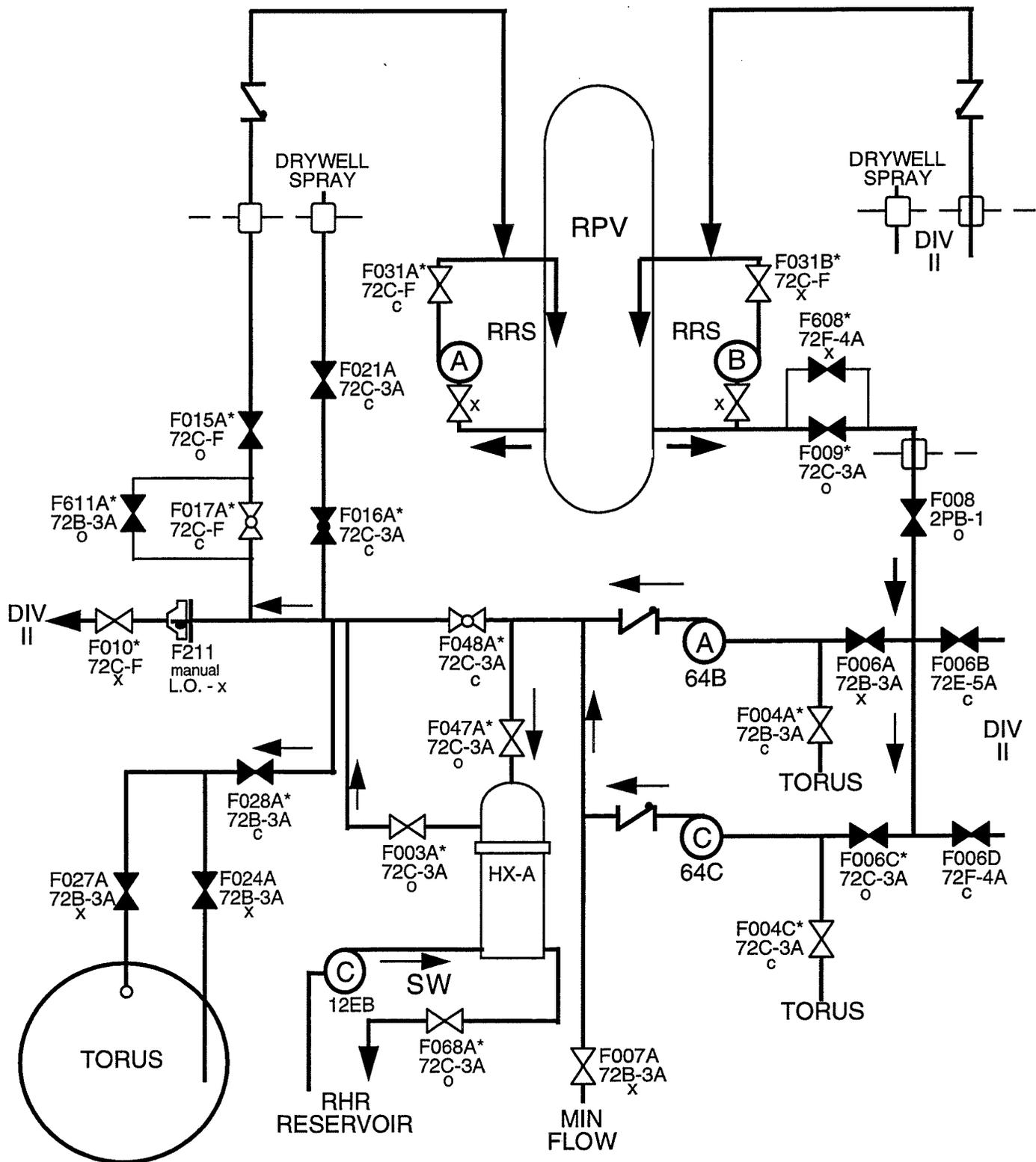
1. Valves shown before loss of offsite power
2. Valve position required for torus cooling lineup:
 - o - open
 - c - closed
 - x - either open or closed
3. Count: 8 valves need control outside control center
 - * - Valve has transfer capability with control and indication (local control capability with appropriate indication at the motor control center isolable from the control center)

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FIGURE 7.5-7

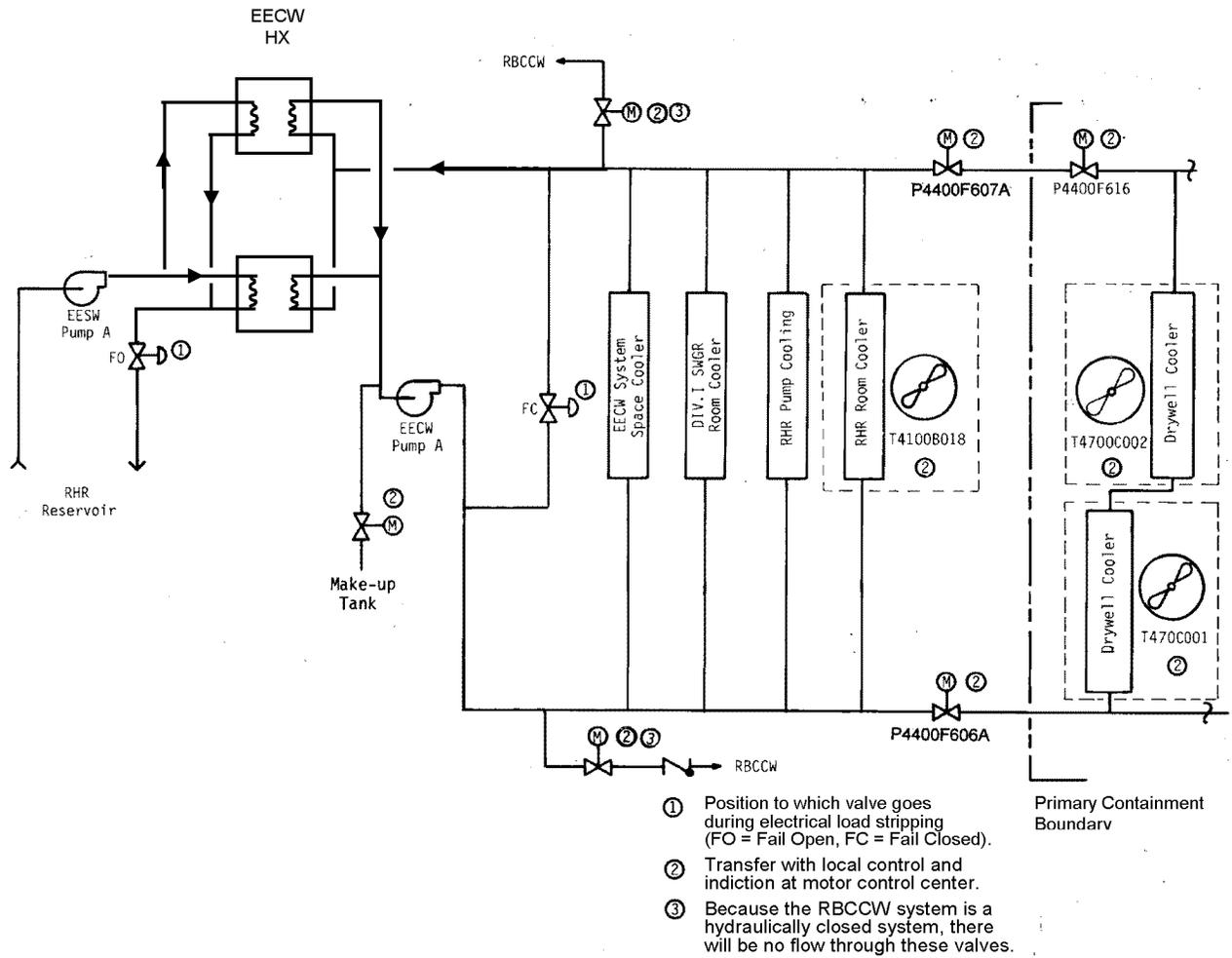
RESIDUAL HEAT REMOVAL SYSTEM
TORUS COOLING MODE



SHUTDOWN COOLING

1. Valves shown before loss of offsite power unless previously aligned for torus cooling
2. Valve position required for shutdown cooling lineup:
 o - open
 c - closed
 x - either open or closed
3. Count: 11 valves need attention
 * - Valve has transfer capability with control and indication

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FIGURE 7.5-8 RESIDUAL HEAT REMOVAL SYSTEM SHUTDOWN COOLING MODE



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FIGURE 7.5-9
 EMERGENCY EQUIPMENT COOLING WATER SYSTEM
 SIMPLIFIED DIAGRAM

7.6 OTHER SYSTEMS REQUIRED FOR SAFETY AND POWER GENERATION

7.6.1 Description

7.6.1.1 Refueling Interlocks System

7.6.1.1.1 System Identification

The purpose of the refueling interlocks system is to restrict the movement of control rods and the operation of refueling equipment to reinforce operational procedures that prevent making the reactor critical during refueling operations.

7.6.1.1.2 Power Sources

Both channels are powered by the control rod drive (CRD) system power supply. A failure of this power supply will prevent any rod motion.

7.6.1.1.3 Equipment Design

7.6.1.1.3.1 Circuit Description

The refueling interlocks circuitry senses the condition of the refueling equipment and the control rods. Depending on the sensed condition, interlocks are actuated to prevent the movement of the refueling equipment or withdrawal of control rods (rod block). Dual channel Circuitry is provided to sense the following conditions:

- a. All rods inserted
- b. Refueling platform positioned near or near over the core
- c. Refueling platform hoists fuel-loaded (fuel grapple, frame-mounted hoist, trolley-mounted hoist)
- d. Fuel grapple not at full-up position.

The indicated conditions are combined in logic circuits to satisfy all restrictions on refueling equipment operations and control rod movement (Figure 7.7-1). A two-channel dc circuit indicates that all rods are in. The rod-in condition for each rod is established by the closure of a magnetically operated reed switch in the rod position indicator probe. The rod-in switch must be closed for each rod position indicator probe. The rod-in switch must be closed for each rod before the "all-rods-in" signal is generated. Both channels must register the "all-rods-in" signal for the refueling interlock circuitry to indicate the "all-rods-in" condition.

During refueling operations, no more than one control rod is permitted to be withdrawn. This restriction is enforced by a redundant logic circuit that uses the "all-rods-in" signal and a rod selection signal to prevent the selection of a second rod for movement with any other rod not fully inserted. The simultaneous selection of two control rods is prevented by the interconnection arrangement of the select pushbuttons. With the mode switch in the REFUEL position, the circuitry prevents the withdrawal of more than one control rod and the movement of the loaded refueling platform over the core with any control rod withdrawn.

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Operation of refueling equipment is prevented by interrupting the power supply to the equipment. The refueling platform is provided with two mechanical switches attached to the platform, which are tripped open by a long, stationary ramp mounted adjacent to the platform rail. The switches open before the platform or any of its hoists are physically located over the reactor pressure vessel (RPV) to indicate the approach of the platform toward its position over the core.

The hoists on the refueling platform and the service platform are provided with switches that open when the hoists are fuel loaded.

This circuitry indicates when fuel is loaded on any hoist.

7.6.1.1.3.2 Bypasses and Interlocks

NOTE: Service platform and hoist equipment are permanently removed. However, bypass for the load interlock system will remain in place in order to provide original function.

A bypass for the service platform hoist load interlock is provided. When the service platform is no longer needed, its power plug is removed. This deenergizes the power supply to the hoist. The platform can then be moved away from the core. Deenergizing the hoist power supply opens the hoist load switches and gives a false indication that the hoist is loaded. This indication prevents control rod withdrawal with the mode switch in the STARTUP or REFUEL position. A bypass plug allows control rod movement in this situation. The bypass plug is physically arranged to prevent the connection of the service platform power plug unless the bypass plug is removed. The rod block interlocks and refueling platform interlocks provide two independent levels of interlock action. The interlocks that restrict operation of the platform hoist and grapple provide a third level of interlock action since they would be required only after a failure of a rod block and refueling platform interlock. It is pertinent to note that the strict procedural control exercised during refueling operations may be considered a fourth level of backup.

7.6.1.1.3.3 Redundancy and Diversity

The refueling interlocks are designed such that a single interlock failure will not cause an accident. These refueling interlocks are provided for use during planned refueling operations. Criticality is prevented during the insertion of fuel, provided control rods in the vicinity of the vacant fuel space are fully inserted during the fuel insertion. The refueling interlock system accomplishes this by:

- a. Preventing operation of the fuel-loaded refueling equipment over the core whenever any control rod is withdrawn
- b. Preventing control rod withdrawal whenever fuel-loading equipment is over the core
- c. Preventing withdrawal of more than one control rod when the mode switch is in the REFUEL position.

The refueling interlocks have been carefully designed using redundancy of sensors and circuitry, to provide a high level of reliability and assurance that the stated design bases will

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be met. Each of the individual refueling interlocks discussed above need not meet the single-failure criterion of IEEE 279-1971 because the four essentially independent levels of protection provide assurance that the design basis is met. For any of the "situations" listed in Table 7.6-1, a single interlock failure will not cause an accident, result in potential physical damage to fuel, or result in radiation exposure to personnel during fuel-handling operations.

7.6.1.1.3.4 Testability

Complete functional testing of all refueling interlocks before refueling outages positively indicates that the interlocks operate in the situations for which they were designed. The interlocks are subjected to valid operational tests by loading each hoist with a suitable test weight, positioning the refueling platform, and withdrawing control rods.

7.6.1.1.4 Environmental Considerations

The refueling equipment is subject to conditions during normal operation that are less severe than those listed in Table 3.11-1. The refueling interlocks are not required to operate under the conditions listed in Table 3.11-1.

7.6.1.1.5 Operational Considerations

The refueling interlocks system is required only during refueling operations.

In the refueling mode, the main control room operator has an indicator light for "refuel mode one rod permissive" whenever all control rods are fully inserted. He can compare this indication with control rod position data from the computer as well as control rod in/out status display. Furthermore, whenever a control rod withdrawal block situation occurs, the operator receives annunciation and computer logging of the rod block. He can compare these outputs with the status of the variable providing the rod block condition.

Both channels of the control rod withdrawal interlocks must agree that permissive conditions exist in order to move control rods; otherwise, a control rod withdrawal block is placed into effect. Failure of one channel may initiate a rod withdrawal block, but does not prevent application of a valid control rod withdrawal block from the remaining operable channel.

In terms of refueling platform interlocks, the platform operator has indicators for the platform x-y position and z position of the fuel grapple. Pushbuttons and rotary control switches are provided for local control of the platform and its hoists. The platform operator can immediately determine whether the platform and hoists are responding to his local instructions. In conjunction with the main control room operator, the local operator can verify proper operation of each of the three categories of interlocks listed previously.

7.6.1.2 Reactor Pressure Vessel Instrumentation

Figure 7.3-12 shows the instrument numbers, arrangements of the sensors, and sensing equipment used to monitor the RPV conditions. Because the RPV sensors used for the reactor protection system (RPS), engineered safety feature (ESF) systems, and control systems have been described and evaluated in other portions of this document, only the sensors that are not required for those systems are described in this subsection.

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7.6.1.2.1 System Identification

The purpose of the RPV instrumentation is to monitor the key RPV operating parameters during plant operation.

7.6.1.2.1.1 Function

These instruments and systems are used to provide the operator with information during normal plant operation, startup, and shutdown. They are monitoring devices only and provide no active power control or safety function.

7.6.1.2.1.2 Classification

The systems and instruments discussed in this subsection are designed to operate under normal and peak operating conditions of system pressures and ambient pressures and temperatures. However, no special industry classifications are imposed on these instruments.

7.6.1.2.2 Equipment Design

The instrument sensing lines to the various pressure and level sensors slope downward from the vessel to the instrument rack at a nominal 1/8 in./ft (including allowance for piping sag).

7.6.1.2.3 Circuit Description

Basic design information for this system is given in Table 7.6-2.

7.6.1.2.3.1 Reactor Pressure Vessel Temperature

The RPV temperature is determined on the basis of reactor coolant temperature. Temperatures needed for operation and for compliance with the Technical Specifications operating limits are obtained from one of several sources, depending on the operating condition. During normal operation, either reactor pressure and/or the inlet temperature of the coolant in the recirculation loops can be used to determine the vessel temperature. Below the operating span of the resistance temperature detectors in the recirculation loop, the vessel pressure is used for determining the temperature. Below 212°F the vessel coolant temperature, and thus the vessel temperature, is reasonably well shown by the reactor water cleanup (RWCU) system inlet temperature. These three sources of input are most conveniently available from the Integrated Plant Computer System (IPCS). During normal operation, vessel thermal transients are limited via operational constraints on parameters other than temperature.

Reactor pressure vessel thermocouples are provided as a means of observing vessel metal surface temperature behavior in response to changes in vessel coolant temperature during startup and during power-operation testing. Indications based on the thermocouples are not used for controlling the rate of heating or cooling or limiting the vessel thermal stresses.

7.6.1.2.3.2 Reactor Pressure Vessel Water Level

The reactor vessel water level instrumentation systems are discussed in other sections as follows:

- a. Reactor water level instrumentation that initiates reactor scram is discussed in Subsection 7.2.1.1.3.1
- b. Reactor water level is maintained by the feedwater control system (Subsection 7.7.1.3).

The reactor water level system that pertains to this section is used to monitor, in the main control room, the reactor water level during the shutdown condition when the reactor system is flooded for maintenance and head removal. The water level design is the condensate chamber reference leg type that is not compensated for change in density. The vessel condition that provides accurate water level information is 0 psig pressure and ambient temperature. The range of the instrument is from the bottom of the feedwater control operating range to a level over the top of the RPV head. Figure 7.3-12 shows specific values at which alarms and safety actions are initiated.

7.6.1.2.3.3 Reactor Core Hydraulics

Figure 7.3-12 shows the flow instruments, differential pressure instruments, and recorders provided so that the core coolant flow rates and the hydraulic performance of RPV internals can be determined.

The differential pressure between the throat of each jet pump and of the core inlet plenum is measured and indicated in the relay room. Four jet pumps, two associated with each recirculation loop, are specially calibrated. They are provided with pressure taps in the diffuser sections. The differential pressure measured between the diffuser tap and the throat tap allows precise flow calibration using the jet pump prototype test performance data for each of the calibrated jet pumps. The flow rates through the remaining jet pumps are calculated from the flows shown by the four calibrated jet pumps. The flow rates through the jet pumps associated with each recirculation loop are summed to provide main control room indication of the core flow rate associated with each recirculation loop (Figure 7.3-12). Total flows for both loops are summed and recorded in the main control room to indicate the total flow through the core. During the operation of a single recirculation loop, total core flow indication is derived by subtracting the reverse flow signal from the forward flow signal of the active jet pumps. This function is provided automatically any time a single recirculation pump is operating.

A differential pressure transmitter indicates core plate pressure drop by measuring the pressure difference between the core inlet plenum and the space just above the core support assembly. The instrument sensing line used to determine the pressure in the core inlet plenum is the same line used for the injection of the standby liquid from the standby liquid control system (SLCS). An instrument sensing line is provided for measuring pressure above the core support assembly. The differential pressure across the core plate is indicated and recorded in the main control room.

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A differential pressure transmitter and control room indicator indicate the jet pump developed head by measuring the difference between the jet pump suction pressure (reactor annulus – between vessel wall and core shroud) and the jet pump discharge pressure (pressure below the core plate).

This instrumentation permits the determination of total core flow in two ways. The first method is the readout of the summed flow measurements from all the jet pumps as described in the preceding paragraphs. The second method involves establishing a correlation between drive loop flow rate and core flow rate with reactor power as a parameter. The correlation can then be used to convert the flow in the recirculation pump loops to core flow rate. This correlation is of a temporary nature because it changes with a fixed core arrangement over a period of time as a result of crud buildup on the fuel. The main control room flow rate readouts of the specially calibrated jet pumps can be used to cross check the flow rate readouts of all the other jet pumps. A discrepancy in the cross-checks is reason enough to check local flow indications.

7.6.1.2.3.4 Reactor Pressure Vessel Pressure

Pressure indicators and transmitters detect RPV internal pressure from the same instrument lines used for measuring RPV water level.

The following list shows the subsections in which the RPV pressure-measuring instruments are discussed:

- a. Pressure transmitters for initiating scram or for bypassing main steam isolation valve (MSIV) closure are discussed in Subsection 7.2.1.1.3
- b. Pressure transmitters used for high-pressure coolant injection (HPCI), core spray, low-pressure coolant injection (LPCI), and the automatic depressurization system (ADS) are discussed in Subsection 7.3.1.2
- c. Pressure transmitters and recorders used for feedwater control are discussed in Subsection 7.7.1.4
- d. Pressure transmitters used for wide range pressure recordings are discussed in Subsection 7.5.1.4.2.

7.6.1.2.3.5 Reactor Pressure Vessel Head Seal Leak Detection

The pressure between the inner and outer head-seal rings is sensed by a pressure switch. If the inner seal fails, the pressure at the pressure switch is the vessel pressure and the pressure switch trips, sounding an annunciator in the main control room. The plant continues to operate with the outer seal as a backup and the inner seal can be repaired at the next outage when the head is removed. If both the inner and outer head seals fail, the leak is detected by an increase in drywell temperature and pressure. This system is part of the leak detection system (LDS), which is described in Subsection 7.6.1.8.

7.6.1.2.3.6 Safety/Relief Valve Seat Leak Detection

Thermocouples are located near the discharge of the safety/relief valve seat. The temperature signal goes to a multipoint recorder with an alarm. The alarm will be activated by any

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temperature in excess of a set temperature, signaling that one of the safety/ relief valve seats has started to leak. This system is part of the LDS (Subsection 7.6.1.8).

7.6.1.2.3.7 Other Instruments

- a. The steam temperature is measured at the steam manifold and is recorded in the main control room
- b. The feedwater temperature is measured and transmitted to the main control room.

7.6.1.2.4 Testability

Pressure, differential pressure, water level, and flow instruments are located outside the drywell and are piped so that calibration and test signals can be applied during reactor operation.

7.6.1.2.5 Environmental Considerations

There are no special environmental considerations for the instruments described in this subsection.

7.6.1.2.6 Operational Considerations

7.6.1.2.6.1 Normal

The RPV instrumentation discussed in this subsection is designed to augment the existing information from the ESF such that the operator can start up, operate at power, shut down, and service the RPV in an efficient manner. None of this instrumentation is required to initiate any ESF.

7.6.1.2.6.2 Operator Information

The following information is available to the operator:

- a. Selected RPV thermocouples are recorded on a multipoint recorder at a local rack
- b. The shutdown flooding water level is indicated in the main control room
- c. The flow for each of the four calibrated jet pumps is indicated in the main control room
- d. The differential pressure for all the jet pumps (calibrated and uncalibrated) is indicated in the main control room and relay room
- e. The recirculation core flow that is generated by each recirculation loop is indicated in the main control room
- f. The total core flow is recorded by one pen of a two-pen recorder in the main control room. The other pen records the core plate differential pressure
- g. The jet pump developed head is indicated in the main control room

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- h. The reactor head seal LDS activates an annunciator when the reactor head inner seal fails
- i. The discharge temperatures of all the safety/relief valves (SRVs) are shown on a multipoint recorder in the relay room. Any temperature in excess of setpoint turns on an annunciator indicating that an SRV seat has started to leak.

7.6.1.2.7 Setpoints

The annunciator alarm setpoints for the reactor head seal leak detection, SRV seat leak detection, and feedwater corrosion product monitor are set so the sensitivity to the variable being measured provides adequate information.

Figure 7.3-12 includes a chart showing the relative indicated water levels at which various automatic alarms and safety actions are initiated. Specific level values are shown in Figure 7.3-12. Each of the listed actions is described and evaluated in the subsection of this report where the system involved is described. The following list tells where various level measuring components and their setpoints are discussed:

- a. Level transmitters for initiating scram are discussed in Subsection 7.2.1.1
- b. Level transmitters for initiating primary containment or vessel isolation are discussed in Subsection 7.3.2.2.8
- c. Level transmitters used for initiating HPCI, LPCI, core spray, and ADS, and the level transmitters to shut down the HPCI pump drive turbine, are discussed in Subsection 7.3.1.2
- d. Level transmitters to initiate reactor core isolation cooling (RCIC) and the level transmitters to shut down the RCIC pump drive turbine are discussed in Subsection 7.4.1.1
- e. Level trips to initiate various alarms and trip the main turbine and the turbine-driven feed pumps are discussed in Subsection 7.7.1.3 and 7.7.1.4.

7.6.1.3 Process Radiation Monitor System

The Process Radiation Monitor system is described in Section 11.4.

7.6.1.4 Area Radiation Monitor System

The Area Radiation Monitor system is described in Subsection 12.1.4.

7.6.1.5 Offsite Environs Radiation Monitor Systems

These systems are described in the Offsite Dose Calculation Manual (ODCM).

7.6.1.6 Rad-Chem Radiation Monitoring Instruments

These systems are described in Section 12.3.

7.6.1.7 Reactor Water Cleanup System Instrumentation and Control

7.6.1.7.1 System Identification

The purpose of the RWCU system instrumentation and control is to provide protection for the system equipment from overheating and overpressurization and to provide the operator with information concerning the effectiveness of operation of the system.

This system is not safety related, and all instrumentation components in the system used only for RWCU operation are nonessential. The instrumentation is a standard industrial type for which performance has been proven by years of service throughout the industry.

7.6.1.7.2 Power Sources

The RWCU instrumentation is fed from the 120-V ac instrumentation bus. No backup power source is necessary since the RWCU system is not a safety-related system. The RWCU instrumentation is arranged in groups or circuits, and each such circuit is protected by a suitable fuse. Thus, a short-circuit within the system will have only a local effect that can be corrected easily without interrupting reactor operation.

7.6.1.7.3 Equipment Design

7.6.1.7.3.1 Circuit Description

The RWCU system is described in Subsection 5.5.8. This subsection describes the circuitry used to protect the resin and the filter-demineralizer. These circuits are shown in Figure 5.5-19 and the operating logic is shown in Figure 7.6-1.

To prevent resins from entering the reactor recirculation system in the event of a filter-demineralizer resin support failure, a strainer is installed on the outlet of each filter-demineralizer unit. Each strainer is provided with a local alarm energized by high differential pressure. A bypass line is provided around the filter-demineralizer units for bypassing the units when necessary.

Relief valves and instrumentation are provided to protect the equipment against overpressurization and the resins against overheating. The system is automatically isolated when signaled by any of the following occurrences:

- a. High temperature downstream of the nonregenerative heat exchanger - to protect the ion exchange resins from deterioration due to high temperature (Table 7.6-2)
- b. Reactor vessel low water level - to protect the core in case of a possible break in the RWCU system piping and equipment (Subsection 7.3.2.2.7.1)
- c. SLCS actuation - to prevent removal of the boron by the RWCU system filter-demineralizers
- d. Cleanup system equipment area high ambient temperatures - part of the plant LDS
- e. High temperature increase across the system's ventilation ducts - part of the plant LDS

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- f. High change in system inlet flow in comparison to the system outlet flow - part of the plant LDS.

In the event of low flow or loss of flow in the system, flow is maintained through each filter-demineralizer by its own holding pump. Sample points are provided upstream of the RWCU system and downstream of each filter-demineralizer unit for continuous indication and recording of system conductivity. High conductivity is annunciated in the main control room. The influent sample point is also used as the normal source of reactor coolant samples. Samples analysis also indicates the effectiveness of the filter-demineralizer units.

7.6.1.7.3.2 Testability

Because the RWCU system is usually in service during plant operation, satisfactory performance is demonstrated without the need for any special inspection or testing.

7.6.1.7.4 Environmental Considerations

The RWCU system is not required for safety purposes, nor is it required to operate after the design-basis accident (DBA). The RWCU system is required to operate in the normal plant environment for power generation purposes only.

The RWCU control instrumentation located in the RWCU equipment area is subject to the environment described in Table 3.11-3.

7.6.1.7.5 Operational Considerations

The RWCU system instrumentation and control is not required for safe operation of the plant. It provides a means of monitoring parameters of the system and protecting the system.

7.6.1.8 Leak Detection System

7.6.1.8.1 System Identification

This subsection discusses the instrumentation and controls associated with the LDS. The system itself is discussed in Subsection 5.2.7.

The LDS serves to detect leakage from the nuclear boiler pressure boundary and auxiliary and ESF systems. It also generates isolation signals to systems that are leaking in excess of determined limits.

7.6.1.8.2 Power Sources

Power source separation is applicable to leak detection channels that are associated with the isolation valve system. Two power sources are used to comply with separation criteria so that redundant channels receive power from separate sources. Power is provided by dc/ac inverter A and dc/ac inverter B.

Inboard and outboard isolation valves in the same line are on separate power sources.

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7.6.1.8.3 Systems and Components Provided With Leak Detection Systems

The following systems and components include leak detection instrumentation and control:

- a. Main steam lines
- b. Reactor core isolation cooling
- c. High-pressure coolant injection
- d. Recirculation pumps
- e. Residual heat removal
- f. Reactor water cleanup
- g. Safety/relief automatic depressurization system valves
- h. Reactor vessel head seal
- i. Emergency core cooling system suction lines.

7.6.1.8.4 System Design

The LDS detects leaks by use of the following techniques:

- a. Sensing excess flow in process piping systems
- b. Sensing pressure and temperature changes in the primary containment
- c. Monitoring temperatures in areas containing equipment and piping systems (Figure 7.6-2)
- d. Monitoring activity of the drain sumps.

Detected leaks are annunciated in the main control room and, in certain cases, isolated from the nuclear steam supply system (NSSS) pressure boundary.

Leaks as small as 5 gpm are detected by either temperature and pressure changes or drain sump activities. Leaks greater than 5 gpm are also detected by changes in reactor water level and by change of flow in process lines.

Temperature detectors are located or shielded such that they are sensitive to air temperature only, and not to heat radiated from the equipment. Temperature sensors have individual alarm setpoints adjustable over a range of flow rates corresponding to leakage of up to 35 gpm.

Reactor coolant leakage of 5 gpm actuates an alarm in the main control room.

Specific information concerning the LDS is given in Table 7.6-2.

7.6.1.8.5 Leak Detection Within the Primary Containment

7.6.1.8.5.1 General

Leaks within the primary containment are detected by the following methods (Figure 7.6-3):

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- a. Monitoring pressure and temperature in the primary containment
- b. Monitoring equipment drain and floor drain sump pump activity
- c. Monitoring the drywell floor drain sump level
- d. Monitoring the cooling water differential temperature of the closed cooling water system
- e. Monitoring reactor water level.

In addition, a second method of leak detection uses recognition of increased containment atmosphere radioactivity as indicative of a system leak.

7.6.1.8.5.2 Pressure Measurement in Primary Containment

The primary containment is pressurized and maintained at a slightly positive pressure during reactor operation. The normal operating pressure is about 0.5 psig. The pressure may fluctuate as a result of barometric pressure changes and outleakages, but a pressure rise above the operating level indicates a process system leak.

Drywell pressure is monitored in the main control room as part of the primary containment monitoring system (Subsection 7.6.1.12). High drywell pressure activates an alarm in the main control room and initiates automatic response of the RPS and ESF systems (Sections 7.2 and 7.3).

7.6.1.8.5.3 Temperature Measurement in Primary Containment

Drywell atmosphere temperature is maintained at approximately 135°F during reactor operation by heat exchangers of the drywell cooling system. An abnormal temperature rise significantly above 135°F indicates a high-energy process leak. A temperature rise will be detected by monitoring:

- a. Drywell temperature at various elevations
- b. Differential water temperature of closed cooling water system.

7.6.1.8.5.4 Primary Containment Sump Activity Monitoring

Equipment drain and floor drain sumps are provided with "fill-up" and "pump-out" rate measurements. Excessive rates are annunciated in the main control room.

The equipment drain sump collects only identified leakage and is equipped with high/low-level switches that control the sump drain. A sump filling/pump frequency in excess of the normal rate or excessive pumping time activates an annunciator in the main control room (Figure 7.6-4). Normal leakage (filling/pumping frequency) is to be determined during operational testing. The equipment drain sump receives drainage from pump seal leakoff, RPV head flange vent drain, and valve packing leakoff.

The floor drain sump is provided with the normal level switches for control of the pumps similar to the equipment drain sump. Additionally, an analog level transmitter is installed in the floor drain sump to provide a very sensitive level change measurement. A continuous analog display of sump level is derived from the transmitter and is located in the control

center. An operator alarm is activated whenever the level measurement detects a sump inleakage greater than 1 gpm. This level monitor is designed and installed to remain functional following a seismic event and thereby meet the requirements of Regulatory Guide 1.45.

The floor drain sump collects unidentified leakage. This leakage is collected from CRDs, valve flanges, floor drains, the closed cooling water system, drywell cooling unit drains, and other potential sources not already identified. Leakage from the closed cooling water system is detected by decreased levels in the system surge tank.

The unidentified-leakage rate is that portion of the total leakage rate received in the drywell floor drain sump. A leakage rate of 150 gpm has been calculated to be the liquid leakage from a crack large enough to propagate rapidly. An allowance for reasonable leakage that does not compromise barrier integrity and is not identifiable is made for normal plant operation.

The unidentified-leakage rate limit is established at 5 gpm, which is far enough below the 150-gpm leakage rate to allow time for corrective action to be taken before the process barrier is significantly compromised. Normal background leakage will be determined during operational testing.

7.6.1.8.5.5 Reactor Vessel Head Seal Leak Detection

The RPV head is provided with double seals with a pressure switch sensing the pressure between the seals. High pressure (Table 7.6-2) is indicative of leakage past the inner seal and activates an annunciator in the main control room. The RPV head seal leak detection is also described in Subsection 7.6.1.2.3.5.

7.6.1.8.5.6 Recirculation Pump Seal Leak Detection

There are two recirculation pump LDSs, one for each of the pumps in the recirculation loop. Each LDS monitors the flow rate (leakage) past its associated pump's shaft by measuring the pressure within the seal cavity. There are two monitored seal cavities per pump.

The recirculation pump LDS consists of two types of monitoring circuits (Figure 7.6-5). The first of these monitors the pressure levels within the seal cavities, presenting the plant operator with a visual display of the pressure in each cavity. The second type of circuit monitors the rate of liquid flow from the seal cavities.

The pressure levels within seal cavity number 1 and seal cavity number 2 are measured with identical instrumentation (Table 7.6-2).

All condensate flowing past the recirculation pump seal packings and into the seal cavities is collected and sent by one of two drain systems to the drywell equipment sump for disposal. The first system drains the major portion of the condensate collected within the number 2 seal cavity. The condensate flow rate through the drain system is measured (high/low) by a flow switch. The point at which the microswitch closes can be adjusted so that switch actuation occurs only above or below certain flow rates (Table 7.6-2). Excessively high or low flow rates through this drain system activate the "Pump Seal Staging Flow" annunciator in the main control room.

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7.6.1.8.5.7 Safety/Relief Valve Automatic Depressurization System Leak Detection

A temperature element (sensor) is used to detect leakage past each relief or safety valve. These temperatures are recorded on a multipoint recorder in the relay room. Normally, all relief and safety valves are in the shut-tight condition and remain at about the same temperature.

Steam passage through the valve elevates the sensed temperature at the exhaust, causing an "abnormal" temperature reading on the recorder. Microswitch contacts on the recorder close on high temperature (Table 7.6-2) to activate the "SRV Open" annunciator in the main control room.

7.6.1.8.6 Reactor Building Sump Activity Monitoring

Instrumentation for monitoring equipment drain sump and floor drain sump activities is the same in design as that described for drywell sump monitoring in Subsection 7.6.1.8.5.4.

7.6.1.8.7 Main Steam Line Leak Detection System

7.6.1.8.7.1 System Function

The main steam lines are continuously monitored for leaks by the main steam line LDS. Steam line leaks will cause changes in at least one of the following monitored operating parameters: sensed temperature, flow rate, or low water level in the RPV. If a leak is detected, the LDS responds by triggering an annunciator in the main control room and, depending upon the activating parameter, initiates steam line isolation action.

7.6.1.8.7.2 Physical Description

The main steam line LDS resistance temperature detectors (RTDs) are located throughout the main steam line tunnel, positioned such that they are screened from direct thermal radiation and yet are still able to respond to the temperature of the ambient air. The RTDs are used to trip the MSIVs closed.

The flow-rate monitoring components of the main steam line LDS consist of a set of four differential pressure transmitters and an associated flow element for each main steam line. The outputs of the differential pressure transmitters are connected to components of the nuclear steam supply shutoff system that give a coincidence signal for main steam line isolation at a flow of approximately 130 percent.

Reactor water level and main steam line tunnel area temperature are monitored by circuits associated with the containment and reactor vessel isolation system to indicate the presence of a steam leak. The coverage of this discussion extends only to the sensing instrumentation and not to circuit arrangement or response. Such information may be found in the description of the primary containment and reactor vessel isolation control system.

Under conditions of normal reactor operation at constant power, reactor water level should remain fairly constant since the rate of steam mass flow leaving the reactor is matched by the feedwater mass flow rate into the RPV. However, given a condition of continued steam

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leakage from the closed system, the condensate reservoir level and the reactor water level decrease.

Reactor water level is monitored by level transmitters of the containment and reactor vessel isolation control system in addition to the normal complement of process-monitoring instruments. Reactor water level falling below the predetermined minimum allowable level results in switch actuation and subsequent containment and reactor vessel isolation control system responses.

7.6.1.8.8 Reactor Water Cleanup System Leak Detection

Leakage in the high temperature process flow of the RWCU system external to the primary containment is detected by temperature-sensing elements. Temperature sensors are located in the inlet and outlet ventilation ducts to measure the temperature difference. Local ambient-temperature sensors are located in all compartments containing equipment for these systems. Alarms in the main control room annunciate a temperature rise corresponding to excessive leakage. In addition to annunciation, a high cleanup-room temperature rise actuates automatic isolation of the RWCU system.

In addition to the temperature-detection method, leakage is detected by means of a flow comparison between RWCU system inlet and outlet. If the inlet flow exceeds outlet flow by approximately 55 gpm, as governed by the Technical Specifications, an alarm is actuated and the RWCU system is isolated automatically.

7.6.1.8.9 Residual Heat Removal System Leak Detection

The residual heat removal (RHR) leak detection components are divided into two groups, one sensitive to RHR system leaks external to the primary containment, and the other sensitive to system leaks internal to the primary containment. Leak detection instruments of the first group use devices that are sensitive to temperature and that monitor area ambient and differential temperatures. The second group of instruments monitors the pressure level within the drywell. Additionally, liquid leakage from system components contained within the drywell is collected and the rate of accumulation measured. The ambient and differential temperature monitoring circuits consist of thermocouples, switch point modules, and meters.

The thermocouples are mounted in their individual holders which, in turn, are mounted in the RHR equipment area such that they are sensitive primarily to the air temperature. The switch-point modules and meters are mounted on the leak detection panel in the relay room. A high ambient temperature lights the point module alarm indicator on the leak detection panel and activates the high ambient temperature alarm.

7.6.1.8.10 Reactor Core Isolation Cooling and High-Pressure Coolant Injection Systems

Leaks in the RCIC or HPCI systems are detected by differential pressure transmitters and by local temperature detectors that are functionally the same as those described for main steam line leak detection (Subsection 7.6.1.8.7).

Downstream of the differential pressure elements, gross leaks in the system are detected by a set of two differential pressure transmitters sensing differential pressure across an orifice plate. Flow in excess of specified limits isolates the system and activates an alarm in the

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main control room. A 3-sec time delay has been installed to prevent inadvertent system isolation due to pressure spikes. Gross leaks upstream of the differential pressure elements may be detected by a set of four pressure transmitters. The primary function of these transmitters is to detect low reactor pressure and to provide HPCI or RCIC turbine isolation signal.

The turbine exhaust vent lines of the HPCI system and the RCIC system are monitored for pressure by means of four pressure transmitters. A high-pressure signal isolates the system and activates an alarm in the main control room. Temperature sensors are located in the inlet and outlet of the ventilation duct of the equipment area and in the inlet to emergency coolers for measuring temperature-difference rise and room ambient temperature in the event of steam leakage. High temperature and high temperature difference are annunciated in the main control room. A high area temperature will automatically isolate the respective system.

The power required to operate the logics associated with the RCIC and HPCI LDSs is continuously monitored. Loss of power is identified by the "RCIC LOGIC POWER FAILURE" or "HPCI LOGIC POWER FAILURE" annunciators in the main control room.

7.6.1.8.11 Leak Detection in the Emergency Core Cooling System Piping Routing Area Adjacent to Suppression Pool

Temperature elements are located in the inlet and outlet of the ventilation ducts of the suppression pool area. High temperature and high temperature differences are annunciated in the main control room.

7.6.1.8.12 Emergency Core Cooling System Suction Lines Leak Detection

The purpose of this LDS is to provide information that would allow the closing of the valve in a broken emergency core cooling system (ECCS) line before net positive suction head (NPSH) is lost to the redundant system.

A sump-level alarm contact notifies the operator that a significant leak exists in the torus area. This signal allows the operator to terminate the loss of torus water.

7.6.1.8.13 Feedwater Leak Detection

A separate feedwater LDS is not provided. Leaks from the feedwater lines will be detected by one or a combination of the following methods:

- a. Primary containment sumps high flow rate
- b. Differential water temperature of closed cooling water system
- c. Primary containment high pressure
- d. Primary containment high temperatures
- e. Reactor building sump high flow rate.

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7.6.1.8.14 Testability

The proper operation of the sensors and the logic associated with the LDS are verified for proper operation during the LDS preoperational test and during inspection tests that are provided for the various components as they apply during plant operation.

Each temperature switch, both ambient- and differential-type, is connected to dual thermocouple elements. Each temperature switch can be checked for operation by observing the ambient temperature or differential, and then turning the trip-point adjustment and verifying that the switch operates at the proper temperature. Each temperature switch contains a trip light that lights when temperature exceeds the setpoint. The setpoint is reset manually to its required value by adjusting the setpoint on the meter in the main control room. In addition, keylock test switches are provided so that logic can be tested without sending an isolation signal to the system involved. Thus, complete system check can be confirmed by checking activation of the isolation relay associated with each switch.

The containment drain monitor system can be tested by supplying makeup water to the sump at a sufficient flow rate to bring the water level above the sump high-level pump-actuation point in less than predetermined time.

The RWCU differential-flow leak detection is tested by inputting a mA signal to simulate a high differential flow. Alarm and indicator lights monitor the status of the trip circuit.

7.6.1.8.15 Environmental Considerations

The sensors, wiring, other equipment, and electronics associated with the isolation valve logic are designed to withstand the conditions that follow a LOCA.

7.6.1.9 Plant Computer Systems

7.6.1.9.1 Integrated Plant Computer System (IPCS)

7.6.1.9.1.1 System Description

The IPCS is a computer system that combines various functions of the legacy computer systems that it replaced. The IPCS provides the capability of monitoring, recording and displaying plant parameters via strategically located display devices. The IPCS is designed to be highly reliable and provide current information for selected plant variables. All real-time data displays will update the current field conditions in a timely manner.

The IPCS is not required for safe operation of the plant. Hardwired instrumentation and control allows the operator to safely operate the plant in all modes in the absence of the IPCS. There is no safety objective for the IPCS.

The IPCS consists of several computing nodes interconnected through a local area network (LAN) configuration. These computing nodes work in conjunction with each other to form a single cohesive system.

The IPCS computing nodes have self-checking provisions. It performs diagnostic checks to determine the functionality of certain portions of the system hardware and software. It also

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performs internal software checks to verify that input signals and selected program computations are within specific limits or reasonable bounds.

The IPCS consists of two redundant computers, operating in parallel, simultaneously monitoring the same field signals. Either of the computers can be designated as the main computer (Master), and the other will be designated as the backup computer (Slave). Under normal operating conditions, both the Master computer and Slave computer perform all the key functions; however, only the Master computer is allowed to output data and calculation results. In the event the Master computer fails, the Slave computer will become the Master computer, and automatically assume control of all functions. The operating personnel will be informed of the fail-over. If the Slave computer fails while the Master computer is operating normally, the operating personnel will be informed and the Master computer will continue to function normally.

The operating personnel use display consoles to enter information into the IPCS, and to request various functions. Diagnostic messages on the display consoles, together with printer outputs and annunciator outputs permit the IPCS to communicate plant and system status to the operating personnel.

The IPCS has the capability for on-line storage and retrieval of historical data, which are to be used for time-history displays and other data analysis functions. These analysis functions include, but are not limited to, determining the plant steady-state operating conditions prior to an initiating event and evaluating the transient conditions producing the event and the post-event equipment performance.

The IPCS has on-line storage for pre-event and post-event data. The declaration of an event is signaled to the system from the display consoles or automatically for a scram. Additional post-event data can be stored at the operator's request. The historical data storage and recovery are performed by the IPCS without interrupting the other functions of the system, such as data acquisition and console display. The IPCS has the capability of transferring the historical data onto magnetic media.

The IPCS consolidates the following functions into a homogenous computer system. Separate legacy computers provided these functions during the initial licensing of the plant:

- a. Scan, Log and Alarm (SLA)
- b. Man-Machine Interface (MMI)
- c. Nuclear Steam Supply System (NSSS) Function
- d. Balance of Plant (BOP) Function
- e. Emergency Response Functions
 1. Safety Parameter Display System (SPDS) Function
 2. Emergency Response Data System (ERDS) Function
- f. Meteorological (MET) Function
- g. Transient Recording and Analysis (TRA) Function
- h. Data Archival Function
- i. Special Functions

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In addition to the function consolidation listed above, the IPCS provides external interfaces with the digital processor/computer equipment associated with each of the following systems, which are further described in Section 7.6.1.9.1.3:

GE 3D-Monicores Computer System (3DM) Interface

GE Multi-Vendor DAS (MVD) Interface (PRNM and RWM Interface)

GE Traversing Incore Probe (TIP) Interface

Eberline SS-1 Radiation Monitor System

Radiological Dose Assessment Application Interface

Meteorological Data Acquisition System (MDAS) Interface

Visual Annunciator System (VAS) Interface

7.6.1.9.1.2 System Functions

7.6.1.9.1.2.1 Scan, Log and Alarm (SLA) Functions

The SLA function gathers data from selected plant data systems, provides the signal conditioning for conversion to engineering units, provides out-of-scale checking of each data point, and keeps a live database of all the current values of the data points.

The IPCS has the capability to alarm the main control room annunciator system in the event of abnormal IPCS operation.

7.6.1.9.1.2.2 Man-Machine Interface (MMI)

The IPCS generates displays and data summaries for use in the control room, technical support center (TSC), and emergency operations facility (EOF). The IPCS also retains a history of each data point. The current value or the historical values of a data point are accessible from the display consoles.

The MMI, the display consoles, and the form of the display on these consoles are designed considering human factors engineering. The system has predesigned displays that are called onto the screen by the operator.

7.6.1.9.1.2.3 Nuclear Steam Supply System (NSSS) Function

The NSSS functions provided are based on Fermi 2 requirements and General Electric recommendations for BWR heat balance and interface support to core monitoring software 3DM.

MMI screens and reports are provided in support of the NSSS performance calculations.

7.6.1.9.1.2.4 Balance of Plant (BOP) Function

The BOP function provides calculations that are based on Fermi 2 requirements, General Electric recommendations, industry recognized practices, and an analysis of the Fermi 2 BOP cycle arrangement and operation.

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Accumulation calculations are provided for the determination of BOP related accumulated data. These accumulated data values are primarily associated with the plant electrical generation data and require the determination of both daily and monthly accumulated totals.

7.6.1.9.1.2.5 Emergency Response Functions

The Emergency Response Functions include the Safety Parameter Display System (SPDS) and the Emergency Response Data System (ERDS). When the plant was originally licensed, various capabilities to support the Emergency Plan were implemented on a dedicated computer system called the Emergency Response Information System (ERIS). These capabilities have been incorporated into the IPCS and are referred to as the Emergency Response Function.

7.6.1.9.1.2.5.1 Safety Parameter Display System (SPDS) Function

Description

An IPCS display with continuous SPDS status indication is provided in the control room, the TSC, and the EOF.

The SPDS function of the IPCS was added to the Fermi 2 design to aid operating personnel in assessing the safety status of the plant. The SPDS function display is accessible and visible to operating personnel and is distinguishable from other displays. The SPDS function display does not inhibit physical or visual access to operator interfaces with other systems located in the control room.

The SPDS design provides for the validation of parameters associated with the function. Operating personnel are alerted to any unsuccessful validation.

Interfaces between the SPDS function and safety-related systems are through isolation means. Interfaces between the SPDS function and non-safety-related systems are designed to ensure the integrity of the SPDS function.

The SPDS functional design includes the consideration of the following human engineering criteria:

- a. Presenting information in directly usable form
- b. Designing displays for quick identification of unsafe conditions
- c. Easy selection of the display required
- d. Minimizing reflection and glare.

The SPDS function presents the value or status of the primary variables of the following safety parameters:

- a. Core cooling
- b. Fuel integrity
- c. Reactivity
- d. Reactor coolant system integrity
- e. Containment integrity

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f. Radioactivity effluent to the environment.

A primary variable is defined as the monitored variable that provides the most direct indication needed to assess the status or value of a safety parameter. Secondary variables are those monitored variables that provide additional information about the safety parameters.

The primary variable associated with each safety parameter is shown in Table 7.6-3 and is discussed below:

- a. Core cooling. The primary method to assess adequate cooling in BWRs is by a direct measurement of the reactor water level. Natural circulation capability is an inherent BWR feature. There are no traps that might block the natural circulation. Steam and noncondensibles rise to the top during normal operation and during accident conditions. As long as there is adequate water level, there is assurance of adequate core cooling
- b. Core and fuel integrity. When the containment is isolated, the presence of fuel damage is determined by taking a sample of reactor coolant and performing a spectral analysis of the sample. During normal operation, the presence of fuel damage is determined by offgas radiation readings
- c. Reactivity. The neutron instrument is the primary variable for determining this parameter
- d. Reactor coolant system integrity. This parameter is assessed by monitoring reactor pressure, drywell pressure, drywell sump collection rate, and RPV isolation
- e. Containment integrity. This parameter is assessed by measuring drywell and torus pressure, containment isolation, combustible gas level, torus temperature, torus level, and drywell temperature
- f. Radioactivity effluent to environment. This parameter is assessed by monitoring the radioactivity at planned plant release points.

The parameters associated with the SPDS displays are listed in Table 7.6-4.

Emergency procedure guidelines (EPGs) have been developed by the BWR Owners Group; they are symptom based and designed to improve the operator's ability to mitigate the consequences of a broad range of initiating events and subsequent multiple operator errors. The Fermi 2 Emergency Operating Procedures (EOPs) are based on the EPGs. The EOPs identify entry conditions and contain parameter versus parameter limit curves. The emergency response function includes the EOP limit curves and parameter information that is supportive of determining entry conditions.

The SPDS function displays are comprised of an overview display, critical safety function displays (generally a bar and/or trend), and the EOP limit curves.

Design Analysis

The graphics provided to the operator by the SPDS function are one of the man-machine interfaces to the IPCS. The IPCS acquires both digital and analog inputs from field sensors and computer data links with monitoring and control systems throughout the plant.

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Existing signal loops of monitoring and control systems were tapped to provide inputs for the SPDS function. The isolation requirement for analog safety-related circuits was provided by using a qualified modulator isolator for each circuit and a demodulator card in the data acquisition system. Dry contact inputs were provided for digital inputs. Engineering was completed in accordance with applicable design criteria to ensure that the SPDS function cannot adversely affect safety-related systems.

Signals to the IPCS for the SPDS function are processed and validated to prevent misleading the operator. Redundant input signals are used for selected parameters and comparison limits are performed for validation. Additional information processing is performed for analog, digital, and derived parameters and includes the following:

- a. Sensor range limit checks
- b. Conversion to engineering units
- c. Validation routine processing
- d. On-line diagnostics for transmission
- e. Time tagging of data.

The SPDS function incorporates human factors engineering guidance. The operator's interface with the displays and keyboard have been designed to provide easily accessed and readily understood displays. The BWR Owners Group Control Room Improvements Committee developed the initial Graphic Display System (GDS) in a program which had extensive human factors evaluation. The program included development and dynamic screening of the GDS and later a simulator evaluation of the displays by operators. The Fermi 2 SPDS function includes many features of the GDS, and has incorporated most of the recommendations from the findings of the simulator evaluation. Some of the human factors criteria that were considered in the SPDS function are listed in the Description subsection above.

The SPDS design requirements with regard to parameter selection, isolation, signal validation, and human factors engineering have been analyzed. The critical safety function based, and EOP-related, selected parameters are sufficient to assess the safety status of the identified functions for a wide range of events, which include symptoms of severe accidents.

7.6.1.9.1.2.5.2 Emergency Response Data System (ERDS) Function

The Fermi 2 ERDS function provides selected plant parameters from the IPCS computer to the NRC. The ERDS function was developed to provide the NRC accurate and timely data on four types of plant parameters, namely:

- a. Core and coolant system conditions
- b. Conditions inside the containment
- c. Radioactivity release rates
- d. Data from the plant's meteorological tower.

The ERDS function is for use during emergencies to transmit information to the NRC Operations Center. The ERDS function datalink will operate in conjunction with the

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Emergency Notification System (ENS), and would be supplemented with voice transmission of essential data not available from the ERDS function.

7.6.1.9.1.2.6 Meteorological (MET) Function

Section 2.3.3.2 describes the Meteorological Data Acquisition System (MDAS) and its interface with the IPCS hardware that replaced the legacy MDAS computer system. The dual-processor IPCS hardware and associated peripherals support the Regulatory Guide 1.23 meteorological function requirements of the original system and provide a platform for performing calculations, providing meteorological data for display at various plant locations, and archiving meteorological data.

The following are examples of calculations (based on data obtained from MDAS) done as part of the IPCS meteorological function:

Sigma Phi (measure of wind stability based on the variability of the vertical component of wind direction and horizontal wind speed)

Pasquill Stability Class

Lake Breeze Status.

In addition, the IPCS MET function uses a Best Value algorithm to determine which value to archive from either the primary or secondary MDAS instrument train data.

7.6.1.9.1.2.7 Transient Recording and Analysis (TRA) Function

The TRA function replaces the legacy General Electric GETARS computer system functionality. The TRA has the capability to process analog and digital points at a scan rate equal to or faster than 100 samples per second, continuously, without degradation of performance of any other IPCS function. The TRA supports auto archiving of data based on data triggers. The ability to utilize both digital states and analog alarm setpoints is provided.

The TRA function plot and report resolution is 10 milliseconds. The TRA function plot and report function provides summary statistics such as mean, minimum, maximum, and standard deviation.

7.6.1.9.1.2.8 Long Term Data Archive (LTA) Function

A computer that is separate from the Master/Slave computers performs the Long Term Data Archive (LTA) function of the IPCS. The LTA computer communicates with the Master computer to receive current plant process data, both analog and digital, at a predefined interval. The LTA computer contains specialized software and a separate I/O database.

Data for points that have been deleted from the IPCS computer database, are retained on the LTA computer, for later retrieval.

7.6.1.9.1.3 External Interfaces

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7.6.1.9.1.3.1 GE 3D-Monicores Computer System (3DM) Interface

The 3DM operation requires periodic transmission of live and static plant data to and from the IPCS. All communications with the 3DM are initiated and monitored by the IPCS.

Live and static plant data transmitted from the IPCS to 3DM includes the following plant inputs and status (represented as individual points):

- a. Heat balance input points
- b. LPRM and APRM data
- c. Control rod data
- d. TIP data (static)

Live plant data transmitted from the 3DM to IPCS includes the following:

- a. RWM messages

7.6.1.9.1.3.2 GE Multi-Vendor DAS (MVD) Interface

The MVD is a data acquisition system interface to the General Electric Power Range Neutron Monitor (PRNM) and the Rod Worth Minimizer (RWM) NUMAC systems. The IPCS MVD interface provides the necessary analog and digital points acquired by the PRNM and RWM in support of real-time plant monitoring and application functions. All data received by the MVD from the PRNM and RWM is date and time stamped and is accessible by a two-way communication protocol interface.

The IPCS RWM acquired data, through the MVD interface, consists of the following signals:

- a. Control rod positions and status (including substitute)
- b. Control rod movement messages
- c. RBM digital points
- d. RWM messages

The IPCS PRNM data consists of the following signals

- a. LPRM flux and associated digital points
- b. APRM flux and associated digital points
- c. RBM flux and associated digital points
- d. Recirculation flow
- e. Oscillation Power Range Monitor Units (OPRM)

7.6.1.9.1.3.3 GE Traversing Incore Probe (TIP) Interface

The IPCS interfaces with the GE TIP system to collect the analog and digital data generated during TIP operation. The IPCS can collect this data simultaneously from all five TIP machines. The IPCS MMI capabilities support viewing TIP traces and TIP operational status in real time.

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The IPCS accumulates the TIP data collected for all TIP machines and combines it with other plant data obtained from the real-time database and archives, writing such data into the TIP files. TIP data files are created and saved for use by the 3D Monicore system.

7.6.1.9.1.3.4 Eberline SS-1 Radiation Monitor System (SS1) Interface

The SS1 radiation monitor has replaced the legacy Eberline CT2B radiation monitor. The SS1 interface acquires current value/status and historical value/status of various radiation monitoring points and processes them for storage in the real-time and archive database, as appropriate.

The IPCS performs all necessary calculations in support of the SPDS function and Dose Assessment interface, including a complex, best channel selection algorithm.

7.6.1.9.1.3.5 Radiological Dose Assessment Application Interface

The Radiological Dose Assessment program calculates the off-site radiological doses based on meteorological and radiological data available in the IPCS. The Earth Tech Raddose V software has been selected to replace the legacy ERIS dose assessment program.

The Raddose V implementation method runs on IPCS MMI in the control room, EOF, and TSC in two separate modes:

- a. Utilizing manually input data
- b. Using selected partial (or total) meteorological and radiological data automatically acquired from the IPCS.

Data and control files, necessary for sharing information between multiple dose assessment nodes, reside in multiple locations within the IPCS.

7.6.1.9.1.4 Power Sources

The IPCS has a reliable AC UPS power source. In the event of a complete loss of offsite power, data will be retained by the IPCS during the outage for display once power is restored. Non-essential peripheral devices are supplied from a reliable AC source with an automatic throw-over switch between normal and standby sources.

7.6.1.9.1.5 Environmental Considerations

All the IPCS equipment is designed for continuous duty up to 105°F and 90 percent relative humidity, except meteorological instrument building equipment, which is designed for 95 percent relative humidity.

7.6.1.9.1.6 Human Factors Engineering

Industry-accepted human factors considerations are followed in designing the man-machine interface. These considerations include, but are not limited to, the following:

- a. Simplicity of entering commands
- b. Feedback recognition of operator commands

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- c. Operator input error prevention and error detection
- d. Flexibility of display access
- e. Flexibility of data entry

7.6.1.9.2 3D-Monicores Computer System (3DM)

7.6.1.9.2.1 System Description

The objectives of the 3DM are to provide a quick and accurate calculation of core thermal performance and to facilitate data reduction, accounting, and logging functions.

The 3DM is not required for safe operation of the plant. Hardwired instrumentation and control allows the operator to safely operate the plant in all modes in the absence of the 3DM. There is no safety objective for the 3DM.

The 3DM consists of two redundant computers. Either of the computers can be designated as the main computer (Normal), and the other will be designated as the backup computer (Standby). Under normal operating conditions, the Normal computer performs all the functions and outputs data and calculation results. The Normal computer also updates the Standby computer at predefined intervals. In the event the Normal computer fails, the Standby computer will become the Normal computer after manual intervention by operating personnel, and assume control of all functions. If the Standby computer fails while the Normal computer is operating normally, the Normal computer will continue to function normally. Either computer can be placed in Failover mode for off-line activities without affecting the operating computer.

3DM receives all plant process data from the IPCS via a high speed datalink. The IPCS gathers, formats and transmits the plant process data at predefined intervals.

The key 3DM features and capabilities are:

- a. Adaptation of 3D diffusion theory solution to measured TIP and LPRM data
- b. Use of full or partial TIP measurements of LPRM calibration
- c. PANACEA 11 diffusion theory based substitute for non-functional TIPs and LPRMs
- d. Calculating core performance parameter distributions and supplying gain corrections for each of the 172 LPRMs based on TIP data measurements
- e. Calculating core margins based on LPRM and thermo-hydraulic readings
- f. Providing displays and printouts of core thermal margins and core performance parameters automatically and on demand
- g. Providing predictive capability to study and evaluate potential impact of operational changes
- h. Providing a digital interface between the Power Range Neutron Monitor (PRNM) system and the IPCS. The interface is provided by a GE Multi-Vendor

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Data Acquisition System (MVD) component as described in Section 7.6.1.9.1.3.2. The MVD also transmits calibration data to the PRNM from 3DM

- i. Receiving and displaying RWM messages.

7.6.1.9.2.2 Operational Considerations

The local power density of every 6-in segment for every fuel assembly is calculated, using plant inputs of pressure, temperature, flow, LPRM levels, control rod positions, and the calculated fuel exposure. Total core thermal power is calculated from a reactor heat balance. Iterative computational methods are used to establish a compatible relationship between the core coolant flow and core power distribution. The results are subsequently interpreted as local power at specified axial segments for each fuel bundle in the core.

The core power distribution calculation sequence may be completed periodically or on demand. The computer has the capability to automatically print a periodic log for record purposes.

Flux level and position data from the plant are read into the IPCS, processed and formatted, and subsequently transmitted to 3DM. 3DM evaluates the data and determines gain adjustment factors by which the LPRM amplifier gains can be altered to compensate for exposure-induced sensitivity loss. The LPRM amplifier gains are not physically altered except immediately prior to calibration of the affected LPRM using the traversing in-core probe (TIP) system. The gain adjustment factor computations help to indicate to the operator when such a calibration procedure is necessary.

Using the power distribution data, a distribution of fuel-exposure increments from the time of the previous power distribution calculation is determined and is used to update the distribution of cumulative fuel exposure. Each fuel bundle is identified by batch and location, and its exposure is stored for each of the axial segments used in the power distribution calculation.

Exposure increments are determined periodically for each section of each control rod. The corresponding percent boron depletions are periodically updated. The exposure increment of each LPRM is determined periodically and is used to update both the cumulative ion chamber exposures and the correction factors for exposure-dependent LPRM sensitivity loss.

3DM provides on-line capability to determine monthly isotopic composition for each fuel bundle in the core. This evaluation consists of computing the weight of one isotope of neptunium, three of uranium, and five of plutonium, as well as the total uranium and total plutonium content. The isotopic composition is calculated for each segment of each fuel bundle and summed accordingly by bundles and batches. The method of analysis consists of relating the computed fuel exposure and average void fraction for the fuel to computer-stored isotopic characteristics applicable to the specific fuel type.

All functions and reports can be executed on demand by the operating personnel.

7.6.1.9.2.3 Power Sources

The 3DM has a reliable AC source with an automatic throw-over switch between normal and standby sources.

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7.6.1.9.2.4 Environmental Considerations

All the 3DM equipment is designed for continuous duty up to 105°F and 90 percent relative humidity.

7.6.1.10 (Deleted)

7.6.1.11 Sequence of Events Recorder

The sequence of events recorder function provides the basic alarm detection management, reporting and real-time display of all sequence of events input signals to the annunciator system. The inputs include NSSS and balance-of-plant (BOP) data.

The input signals are independent of the IPCS and are electrically isolated from it. An outage of the computer would not affect operation or reliability of the sequence of events recorder.

The sequence of events recorder displays the time and alarm type of each event and can resolve the time of occurrence with a resolution approaching 1 msec.

7.6.1.12 Primary Containment Monitor System

The primary containment monitor system consists of the following five monitor subsystems:

- a. Primary containment radiation monitor
- b. Primary containment temperature monitor
- c. Primary containment pressure monitor
- d. Pressure suppression pool water level indicator
- e. Hydrogen/oxygen monitor.

Division II AOVs T5000F420B and T5000F421B are designed to be reopened in the event of an extended AC power failure, using DC solenoid valves T50F459B and T50F468B respectively, as show in Figure 7.6-11.

The primary containment radiation monitor subsystem supplements the LDS. The primary containment radiation monitor is not designed to operate following the DBA. The radiation monitor designed to monitor post-DBA containment radiation is the containment area high range monitor discussed in Section 11.4. The other four primary containment monitor subsystems, namely, the primary containment temperature monitor, primary containment pressure monitor, the pressure suppression pool water-level indicator, and the hydrogen/oxygen monitor, are required to operate after a LOCA, and are designed to meet the redundancy and separation requirements listed in Subsection 7.6.2.12. Under normal plant operation, however, these subsystems provide display of the monitored parameters for additional information on the operating conditions of the plant.

Descriptions of the five subsystems that compose the primary containment monitor system are presented in Subsections 7.6.1.12.1 through 7.6.1.12.4.

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7.6.1.12.1 Primary Containment Radiation and Hydrogen/Oxygen Monitor Subsystem

7.6.1.12.1.1 System Identification

The primary containment radiation monitor subsystem is incorporated for monitoring the radioactivity of the atmosphere within the primary containment to provide additional information related to primary coolant leak detection. This provision improves the total drywell leak-detection diversity and enhances the sensitivity of leak detection beyond that which is available with the drywell sump system. An alarm and annunciator are actuated when the radiation level reaches a predetermined setpoint level. This subsystem has no control function (Table 7.6-2).

The primary containment radiation monitor uses a beta scintillation detector viewing a sample flow of primary containment atmosphere as it passes through a gaseous detector chamber. The sample is drawn from the primary containment, through the filters, past the gaseous sample detector, and returned to the containment by a sample pump. The piping and valve arrangement for sample flow are illustrated in Figure 7.6-11. Remotely controlled valves are used to select either all the drywell atmosphere sampling points, selected locations within the drywell, or the suppression chamber atmosphere for radiation monitoring.

The main sample flow loop supplies both the primary containment radiation monitor subsystem and the hydrogen/oxygen monitor subsystem in parallel. The hydrogen/oxygen monitor subsystem operates continuously to provide indication in the main control room of the concentration of hydrogen/oxygen in the containment. Levels of hydrogen and oxygen in excess of preset limits are alarmed in the main control room (Table 7.6-2).

The total time lag from intake of drywell atmosphere sample loop manifold to the monitoring instrument sampling point is designed to be less than 5 minutes. Within the primary containment radiation monitor are particulate and halogen filters to collect integrated samples for subsequent analysis.

Associated with each beta scintillation detector is a logarithmic count rate circuit, power supply unit, and meter readout. A recorder is provided in the main control room for display of radiation level. A flowmeter is provided in the sample line, with local display of flow rate, and means for actuation of the alarm and annunciator associated with the primary containment radiation monitor on loss of sample flow.

7.6.1.12.1.2 Classification

The primary containment hydrogen/oxygen monitor subsystem is seismically and environmentally qualified to meet the requirements of Regulatory Guide 1.97, Rev 2, Category 3 and 2, respectively. The radiation monitor has not been qualified environmentally or seismically.

7.6.1.12.1.3 Supporting Systems

Electrical Power

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The electrical power required for operation of the primary containment radiation and hydrogen/oxygen monitor subsystems is supplied from the 480-V ESF motor control centers (MCCs) and the 120-V ac instrument bus as described in Subsection 8.3.1.

Pneumatic Power

The pneumatic power required for operation of valves in the sample lines will be supplied by an uninterruptible air system for the primary containment monitoring system isolation valves and an interruptible air system for the primary containment radiation monitoring system isolation valves as described in Subsection 9.3.1.

7.6.1.12.1.4 Equipment Design

Initiating Circuits

Control of the primary containment radiation and hydrogen/oxygen monitor subsystems for normal operation, test, and calibration is manual. The hydrogen/oxygen monitor subsystem is normally operated continuously from plant startup to shutdown.

Logic

The primary containment radiation monitor subsystem incorporates trip logic circuits for alarm and annunciator operation. A low mode alarm trip is provided to indicate instrument failure on loss of normal background reading, and a high mode trip to indicate a radiation level exceeding a predetermined normal background level. The hydrogen/oxygen monitor has alarms as defined in Table 7.6-2.

Actuated Devices

The primary containment radiation monitor and hydrogen/oxygen monitor subsystems have no control function. Devices actuated by the primary containment radiation monitor are an alarm and an annunciator on high radiation and on loss of background signal, the latter being indicative of instrument failure.

Testability

The primary containment radiation monitor and the hydrogen/oxygen monitor subsystems are fully testable during normal plant operation.

7.6.1.12.1.5 Environmental Considerations

The primary containment hydrogen/oxygen monitor subsystem is designed to operate reliably under normal and postulated abnormal environmental conditions in the equipment area. The local environmental conditions are defined in Table 3.11-1.

The oxygen monitors provide verification of the status of the inerted atmosphere of containment and oxygen levels in the containment atmosphere following a significant beyond design-basis accident for combustible gas control and accident management, including emergency planning.

The hydrogen monitors provide diagnosis of the course of significant beyond-design-basis accidents for accident management, including emergency planning.

7.6.1.12.1.6 Operational Considerations

Normal

During power operation, startup, or hot shutdown of the reactor, the primary containment radiation monitor subsystem is in continuous operation to detect and alarm a high level of radiation in the monitored atmosphere. The hydrogen/oxygen monitor subsystem is normally operated continuously from plant startup to shutdown.

Safety Function

On occurrence of radiation above the alarm setpoint level, the abnormal condition will be alarmed and annunciated in the main control room. High levels of hydrogen and/or oxygen are alarmed in the main control room.

Operator Information

The primary containment radiation monitor subsystem has provisions in the main control room for indicating the radiation count rate per unit volume of sample, for checking the setpoint at which the high radiation alarm is actuated, and for alarming low sample flow rate. A chart recorder is provided to record the monitored radiation levels. By the actuation of an alarm and annunciator, the operator is informed of high radiation level, instrument failure as evidenced by loss of normal background reading, or loss of sample flow. A number of functional alarms are provided in the hydrogen/oxygen monitors to ensure proper system operation.

Setpoints

The setpoint for actuation of the alarm and annunciator on high radiation will be established after the normal background is determined. The setpoint is adjustable over the instrument range. The alarm setpoints for the hydrogen/oxygen monitors can be found in Table 7.6-2.

7.6.1.12.2 Primary Containment Temperature Monitoring Subsystem

7.6.1.12.2.1 System Identification

The primary containment temperature monitor subsystem uses thermocouple detectors to measure drywell atmosphere temperature, pressure suppression pool chamber atmospheric temperature, and pressure suppression pool water temperature. To achieve representative temperature measurements in the primary containment, 16 sensors in the drywell, four sensors for suppression pool atmosphere, and four sensors for suppression pool water are used. The monitored temperatures are continuously recorded on two stripchart recorders. The primary containment temperature monitor subsystem has no control function; its purpose is that of data acquisition. Because of the importance of securing these temperature data on postulated accident (LOCA) conditions and other abnormal plant conditions, two redundant temperature monitors are provided.

In parallel with the primary containment temperature monitor subsystem (PCTMS), and in response to additional design requirements, two additional temperature monitoring systems have been installed to monitor the drywell air temperature and the suppression pool

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temperature, during normal mode of plant operation and transient condition, as discussed below.

The additional drywell air temperature monitoring subsystem uses 28 thermocouples (which are independent of the PCTMS) installed at six elevations. The temperature information is recorded in the main control room and is used by the operators to compute the volumetric average temperature for determination of the Technical Specifications operating limit. The additional suppression pool temperature monitoring subsystem uses a recorder and eight thermocouples (which are all independent of the PCTMS). These eight thermocouples in the torus are used by operators to compute the suppression pool water bulk average temperature for determination of the Technical Specification operating limit. The thermocouples are placed so that each thermocouple monitors the discharge of two SRVs. The recorder is located in the main control room and will alarm on bulk average water temperature $\geq 95.0^{\circ}\text{F}$ or detection of an open T/C. The technical specifications allow the maximum average temperature of the suppression pool to be 95°F during operational conditions 1 or 2 unless the thermal power is less than or equal to 1%, or a test is being performed which adds heat to the suppression chamber. To alert an operator of this limit, an alarm will be set at approximately 95.0°F (increasing).

The additional thermocouples in the drywell and torus and the associated recorders which are used during normal mode of operation to monitor the drywell volumetric average and suppression pool bulk average temperature are classified as quality assurance (QA) level 1M and seismic category II/I. This classification means that these components are maintained like a QA level 1 component, and they will maintain their structural and mounting integrity during a seismic event.

The torus water temperature recorder and Division I drywell air temperature recorder are powered from a non-class IE distribution cabinet that is fed by a bus which is automatically restored by the emergency diesel generators (EDG) on a loss-of-offsite power. Thus, in the event of loss-of-offsite power, these recorders can be powered from the EDGs by closing a breaker. The Division II drywell air temperature recorder is powered from a class IE power bus.

7.6.1.12.2.2 Classification

The primary containment temperature monitor subsystem is not a fully qualified system. The set of 24 primary containment monitoring thermocouples have been installed seismically. Two thermocouples in the drywell, two in the suppression pool air space, and two in the suppression pool water have also been qualified environmentally for postulated accident conditions.

7.6.1.12.2.3 Power Sources

Operating power for the two identical temperature monitors of the primary containment temperature monitor subsystem is supplied from separate 120-V ac instrument buses to prevent total loss of monitoring capability on interruption of an instrument bus.

7.6.1.12.2.4 Equipment Design

Initiating Circuits

Both monitors are in service when the reactor is operating in order to provide necessary backup monitoring.

Redundancy

Two identical monitors are provided in the primary containment temperature monitor subsystem.

Separation

Instrument control and power feed circuits of the two monitors comprising this subsystem are separate. Routing of thermocouple and power circuits by separate paths and penetrations precludes total loss of temperature monitoring capability by a single destructive event.

Testability

To facilitate test and calibration, the temperature detectors are removable from their working locations. The recorders are tested and checked for calibration by the standard technique of using a millivolt box after disconnection of the thermocouple circuits.

7.6.1.12.2.5 Environmental Considerations

The thermocouples, associated thermocouple lead circuits, and recording instrumentation are of a design that provides reliable operation under normal and postulated abnormal environmental conditions. The environmental conditions of specific plant areas are defined in Tables 3.11-1, 3.11-3, and 3.11-4.

7.6.1.12.2.6 Operational Considerations

Normal

The primary containment temperature monitor subsystem operates continuously during operation of the reactor, and secures recordings of the monitored temperatures. Should a LOCA occur, obtained recordings provide essential information about the temperatures monitored by the primary containment temperature monitor subsystem.

Operator Information

Monitored temperatures are displayed on the recorder chart in the main control room, making this information directly available to the operations personnel when the plant is in operation.

7.6.1.12.3 Primary Containment Pressure Monitor Subsystem

7.6.1.12.3.1 System Identification

The primary containment pressure monitor subsystem monitors the atmospheric pressure of the drywell and the pressure suppression chamber and records on redundant chart recorders in the main control room. The instrumentation for the drywell uses two pressure ranges, -5 to +5 psig and 0 to 250 psig. The instrumentation for the pressure suppression chamber uses

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two ranges also, -5 to +15 psig and 0 to 80 psig. The low range of the pressure monitoring instrumentation enables detection of a change in drywell and or pressure suppression chamber pressure resulting from a primary containment leak and containment sprays, and provides for sensitive monitoring during normal operation of the plant and during shutdown and LOCA conditions. In order to provide continued monitoring of the drywell pressure during an extended loss of AC power, T5000F420B can be reopened using the DC solenoid valve T50F459B, as shown in Figure 7.6.11. The low range pressure monitoring instrumentation may also provide a means of detecting degradation of the containment pressure boundary. The high pressure ranges provide the capability to measure a pressure transient arising from a LOCA. Pressure is displayed on a multipoint recorder. Two complete and independent pressure monitors comprise the primary containment pressure monitor system. This subsystem has no control function; its purpose is that of data acquisition and advisory information.

Pressure transmitters used to initiate ECCSs or to provide inputs to the RPS are described in Subsection 7.3.1 and Section 7.2.

7.6.1.12.3.2 Classification

The primary containment pressure monitor subsystem is designed and installed as Category I. The pressure transmitters incorporated in this subsystem are also environmentally qualified for postulated accident conditions.

7.6.1.12.3.3 Power Sources

The electrical power for the powered equipment of the two pressure monitors in this subsystem is supplied from separate battery-powered inverters.

7.6.1.12.3.4 Equipment Design

Initiating Circuits

Control of operation of the pressure detector equipment is manual. Operation of this monitoring subsystem is continuous when the reactor is in operation.

Redundancy

Two identical monitors are provided in the primary containment pressure monitor subsystem.

Separation

Electrical circuits of the two identical pressure monitors comprising this subsystem are routed separately to minimize vulnerability to total impairment of the monitor subsystem by a single destructive event.

Testability

To facilitate periodic checks of operation of this subsystem, provisions are incorporated to allow for in-place testing of the detectors and convenient removal for testing when the reactor is shut down.

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7.6.1.12.3.5 Environmental Considerations

The equipment of the primary containment pressure monitor subsystem is designed to operate reliably under the normal and postulated abnormal conditions of the equipment areas. The environmental conditions of these areas are defined in Table 3.11-1.

7.6.1.12.3.6 Operational Considerations

Normal

The primary containment pressure monitor subsystem operates continuously during operation of the reactor, securing recordings of the pressures within the primary containment.

Safety Function

Should a LOCA occur, secured recordings provide information sought about transients in the pressures monitored by the primary containment pressure monitor subsystem.

Operator Information

Monitored pressures are displayed on the recorder chart in the main control room as they are being recorded, making this information directly available while the plant is in normal operation. The pressure may fluctuate for a variety of reasons including changes in barometric pressure. A pressure rise above normal could indicate a process system leak as described in Subsection 7.6.1.8.5.2. The low range pressure monitoring instrumentation may also provide a means of detecting degradation of the containment boundary depending on the magnitude of the degradation. Unexplained changes in pressure during normal operation would therefore result in an investigation to determine the cause using the Corrective Action program as appropriate. If the drywell pressure increases above the trip value, the chart recorder increases chart speed to obtain better transient resolution. Peak pressure is also obtained by a peak-pressure indicator.

7.6.1.12.4 Pressure Suppression Pool Water Level Indicator Subsystem

7.6.1.12.4.1 System Identification

The pressure suppression pool water-level indicator subsystem continuously monitors and records on a chart recorder in the main control room the water level in the pressure suppression chamber. The principal function of this monitor subsystem is to obtain data on water level in the pressure suppression chamber on occurrence of a LOCA. The subsystem also serves to indicate and record the water level in the course of normal operation of the plant and during shutdown and LOCA condition. In order to provide continued monitoring of the suppression pool level during an extended loss of AC power, T5000F421B can be reopened using the DC solenoid valve T50F468B, as shown in Figure 7.6.11. This is a supplementary function because the pressure suppression chamber water level is maintained, and the level indication is necessarily provided in the main control room as part of the torus water management system.

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7.6.1.12.4.2 Classification

The pressure suppression pool water level indicator subsystem is designed and installed as Category I. The transmitters are qualified to meet the environmental conditions of postulated accidents.

7.6.1.12.4.3 Power Sources

Operating electrical power for the pressure suppression pool water level indicator subsystem is supplied from separate battery-powered inverters.

7.6.1.12.4.4 Equipment Design

Initiating Circuits

Control of operation of the pressure suppression pool water level indicator subsystem is manual. Operation of the subsystem is continuous during reactor operation and when the reactor is shut down.

Redundancy

Two identical level-indicator instrumentation provisions comprise the pressure suppression pool water level indicator system.

Separation

The two independent level-indicator provisions comprising this subsystem have signal and power lines routed separately to minimize vulnerability to impairment of both subsystems by a single destructive event.

Testability

The pressure suppression pool water level indicator subsystem incorporates means to allow for complete testing of the subsystem during periods when the reactor is shut down.

7.6.1.12.4.5 Environmental Considerations

The equipment of the pressure suppression pool water level indicator subsystem is designed to operate reliably under the normal and postulated abnormal conditions to which the equipment would be exposed. The environmental conditions of the equipment areas are defined in Tables 3.11-1, 3.11-3, and 3.11-4.

7.6.1.12.4.6 Operational Considerations

Normal

The pressure suppression pool water level indicator subsystem is in continuous operation during operation of the reactor as well as during periods of shutdown, unless the equipment is taken out of service for test or maintenance purposes.

Safety Function

Should a LOCA occur, the recordings obtained provide essential information about the pressure suppression chamber water level during the abnormal conditions.

Operator Information

The water level in the pressure suppression pool is continuously indicated as well as recorded on a chart in the main control room.

7.6.1.13 Neutron Monitoring System Instrumentation and Control

The neutron monitoring system (NMS) consists of six major subsystems which are

- a. Source range monitor
- b. Intermediate range monitor
- c. Local power range monitor
- d. Average power range monitor
- e. Rod block monitor
- f. Traversing in-core probe.

7.6.1.13.1 System Identification

The purpose of this system is to monitor neutron flux levels of the core over the range from shutdown to full power, and to provide signals to the RPS (Section 7.2). It also provides information for operation and control of the reactor. Basic system information is given in Table 7.6-2.

Certain portions of the intermediate range monitor (IRM) and average power range monitor (APRM) systems provide a safety function, and portions of the rod block monitor (RBM) have been designed to meet IEEE 279-1971. All other portions of the NMS have no safety function.

7.6.1.13.2 Power Sources

The power supplies for each system are discussed in the individual circuit description.

7.6.1.13.3 Source Range Monitor System

7.6.1.13.3.1 Equipment Design

Circuit Description

The source range monitor (SRM) provides neutron flux information during reactor startup and low-flux-level operations. There are four SRM channels. Each includes one detector that can be physically positioned in the core from the main control room (Figures 7.6-12 through 7.6-14). The detectors are inserted into the core for a reactor startup. They can be withdrawn if the indicated count rate is between preset limits or if the IRM is on the third range or above.

The power for the monitors is supplied from the two separate +24 V dc buses. Two monitors are powered from each bus. The detector drives are powered by a 208-V ac three-phase bus.

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Each detector assembly consists of a miniature fission chamber and a low-loss, insulated transmission cable.

The sensitivity of the detector is 1.2×10^{-3} cps/nv nominal. The detector cable is connected underneath the RPV to shielded coaxial cable. This shielded cable carries the pulses to a pulse current preamplifier located outside the primary containment.

The detector and cable are located inside the RPV in a dry tube sealed against reactor vessel pressure. A remote-controlled detector drive system moves the detector along the dry tube.

Vertical positioning of the chamber is possible from above the centerline of the active length of fuel to approximately 2-1/2 ft below the reactor fuel region, as shown in Figure 7.6-13.

When a detector arrives at a travel endpoint, detector motion is automatically stopped. The SRM/IRM drive control logic is presented in Sheet 6 of Figure 7.6-16. The electronics for the SRMs, their trips, and their bypasses are located in one cabinet. Source-range signal-conditioning equipment is designed so that it can be used for open-core experiments.

A charge-sensitive preamplifier provides amplification and impedance matching for the signal conditioning electronics (Figure 7.6-17). The signal conditioning equipment converts the current pulses to analog dc voltages that correspond to the logarithm of the count rate. The equipment also derives the period. The output is displayed on front panel meters and is provided to remote meters and recorders.

The logarithmic count rate meter displays the rate of occurrence of the input current pulses. The period meter displays the time in seconds for the count rate to change by a factor of 2.72. In addition, the equipment contains integral test and calibration circuits, trip circuits, power supplies, and selector circuits.

The trip outputs of the SRM operate in the fail-safe mode. Loss of power to the SRM causes the associated outputs to become tripped (Figure 7.6-16, Sheet 2).

The SRM provides signals indicating SRM upscale, downscale, inoperative, and incorrect detector position to the RMCS to block rod withdrawal under certain conditions. Any SRM channel can initiate a rod block. These rod blocking functions are discussed in Subsection 7.7.1.1.3.5. Appropriate lights and annunciators are also actuated to indicate the existence of these conditions (Table 7.6-5). One in one group of four SRM channels can be bypassed at any one time by the operation of a switch on the operator's console.

Testability

Each SRM channel is tested and calibrated. Inspection and testing are performed as required on the SRM detector drive mechanism. The mechanism can be checked for full insertion and retraction capability. The various combinations of SRM trips can be introduced to ensure the operability of the rod blocking functions.

7.6.1.13.3.2 Environmental Considerations

The wiring, cables, and connectors located within the drywell are designed for the environmental conditions identified in NEDO 31558A per References 7 and 8 in Appendix A.

7.6.1.13.4 Intermediate Range Monitor System

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7.6.1.13.4.1 Equipment Design

Circuit Description

The IRM monitors neutron flux from the upper portion of the SRM range to the lower portion of the power range monitoring subsystems. The IRM system has eight IRM channels, each of which includes one detector that can be positioned in the core by remote control. The detectors are inserted into the core for a reactor startup and are withdrawn after the reactor mode selector switch is turned to the RUN position and the LPRM is operative.

Power Supply

Power is supplied separately from the two 24-V dc sources. The supplies are split according to their use so that loss of a power supply results in the loss of only one trip system of the RPS.

Physical Arrangement

Each detector assembly consists of a miniature fission chamber attached to a low-loss, insulated transmission cable. When coupled to the signal conditioning equipment, the detector produces a reading of approximately 30 percent on the most sensitive range with a neutron flux of 10^8 nv. The detector cable is connected underneath the RPV to a shielded cable that carries the pulses generated in the fission chamber through the primary containment to the preamplifier.

The detector and cable are located in the drywell. They are movable in the same manner as the SRM detectors and use the same type of mechanical arrangement (Reference 1) and power supply.

Signal Conditioning

A voltage amplifier unit located outside the primary containment serves as a preamplifier. This unit converts the current pulses to voltage pulses, modifies the voltage signal, and provides impedance matching. The preamplifier output signal is coupled by a cable to the IRM signal conditioning electronics as shown in Figure 7.6-18.

Each IRM channel receives its input signal from the preamplifier and operates on it with various combinations of preamplification gain and amplifier attenuation ratios. The amplification and attenuation ratios of the IRM and preamplifier are selected by an operator's console-mounted range switch that provides 10 ranges of increasing attenuation acting on the signal from the fission chamber (the first six ranges are called low range and the last four ranges are called high range). As the neutron flux of the reactor core increases from 1×10^8 nv to 1.5×10^{13} nv, the signal from the fission chamber is attenuated to keep the input signal to the inverter in the same range. The output signal, which is proportional to neutron flux at the detector, is amplified and supplied to a locally mounted meter. Outputs are also provided for a remote meter and recorder.

Trip Functions

The IRMs are arranged in the core as shown in Figure 7.6-12 and are divided into two groups of IRM channels. Each group is associated with one of the two trip systems of the RPS. Two IRM channels and their trip auxiliary are installed in each bay of a four-bay cabinet.

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Full-length side covers isolate the cabinet bays. The arrangement of IRM channels allows one IRM channel in each group to be bypassed without compromising intermediate range neutron monitoring.

Each IRM channel includes four trip circuits as standard equipment. One trip circuit is used as an instrument trouble trip. It operates on when the high voltage drops below a preset level, when one of the modules is not plugged in, or when the OPERATE-CALIBRATE switch is not in the OPERATE position. Each of the other trip circuits can be specified to trip when preset downscale or upscale levels are reached.

The trip functions actuated by the IRM trips are indicated in Table 7.6-6. The reactor mode switch determines whether IRM trips are effective in initiating a rod block or a reactor scram (Figure 7.6-16, Sheet 1). Subsection 7.7.1.1 describes the IRM rod block trips. With the reactor mode switch in REFUEL or STARTUP, an IRM upscale or inoperative trip signal actuates a NMS trip of the RPS. Only one of the IRM channels must trip to initiate an NMS trip of the associated trip system of the RPS.

Testability

Each IRM channel is tested and calibrated using the appropriate Checkout and Initial Operations, Preoperational, or Surveillance Procedures. The IRM detector drive mechanisms and the IRM rod blocking functions are checked in the same manner as for the SRM channels. Each IRM channel can be checked to ensure that the IRM high flux scram function is operable.

7.6.1.13.4.2 Environmental Considerations

The wiring, cables, and connectors located in the primary containment are designed for the environmental conditions identified in NEDO 31558A per References 7 and 8 in Appendix A.

7.6.1.13.5 Local Power Range Monitor System

7.6.1.13.5.1 Equipment Design

Description

The LPRM consists of fission chamber detectors, signal conditioning equipment, and trip functions. The LPRM also provides outputs to the APRM, RBM, and Integrated Plant Computer System (IPCS).

Power Supply

Detector polarizing voltage for the LPRMs is supplied by eight pairs of redundant-dc power supplies, adjustable from 75 to 200 V dc. The 75-200 V dc power supplies can supply up to 3 milliamperes for each LPRM detector which ensures that the chambers can be operated in the saturated region at the maximum specified neutron fluxes. Each dc power supply pair powers approximately one-eighth of the LPRMs. Power for the dc power supplies comes redundantly from the two 120 V ac Reactor Protection System buses via intermediate dc power supplies. These intermediate dc supplies also provide power for the LPRM amplifiers.

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Physical Arrangement

The LPRMs include 43 LPRM detector strings having detectors located at different axial heights in the core. Each string contains four fission chambers. These assemblies are distributed to monitor four horizontal planes throughout the core. Figure 7.6-12 shows the LPRM detector radial layout scheme that provides a detector assembly at every fourth intersection of the water channels around the fuel bundles. Thus, every location has either an actual detector assembly or a symmetrically equivalent assembly in some other quadrant.

The detector assemblies (see Figure 7.6-19) are inserted in the core in spaces between the fuel assemblies. They are inserted through thimbles mounted permanently at the bottom of the core lattice and penetrate the bottom of the RPV. These thimbles are welded to the RPV at the penetration point. They extend down into the access area below the RPV where they terminate in a flange. The flange mates to the mounting flange on the in-core detector assembly. The detector assemblies are locked at the top end to the top fuel guide by means of a spring-loaded plunger. Special water-sealing caps are placed over the connection end of the assembly and over the penetration at the bottom of the vessel during installation or removal of an assembly. This prevents loss of reactor coolant water on removal of an assembly, and also prevents the connection end of the assembly from being immersed in the water during installation or removal.

Each LPRM detector assembly contains four miniature fission chambers with an associated solid sheath cable. The chambers are vertically spaced in the LPRM detector assemblies in a way that gives adequate axial coverage of the core, complementing the radial coverage given by the horizontal arrangement of the LPRM detector assemblies. Each fission chamber produces a current that is coupled with the LPRM signal conditioning equipment to provide the desired scale indications.

Each miniature chamber consists of two concentric cylinders that act as electrodes. The inner cylinder (the collector) is mounted on insulators and is separated from the outer cylinder by a small gap. The gas between the electrodes is ionized by the charged particles produced as a result of neutron fissioning of the uranium-coated outer electrode. The chamber is operated at a polarizing potential of approximately 100 V dc. The negative ions produced in the gas are accelerated to the collector by the potential difference maintained between the electrodes. In a given neutron flux, all the ions produced in the ion chamber can be collected if the polarizing voltage is high enough. When this situation exists, the ion chamber is considered to be saturated. Output current is then independent of operating voltage and is reasonably linear over the design operating range (Reference 1).

Each assembly also contains a calibration tube for a traversing in-core probe (TIP). The enclosing tube around the entire assembly contains holes that allow circulation of the reactor coolant water to cool the fission chambers. Numerous tests have been performed on the chamber assemblies, including tests of linearity, gamma sensitivity, and cable effects (Reference 1). These tests and experience in operating reactors provide confidence in the ability of the LPRM system to monitor neutron flux to the design accuracy throughout design lifetime.

Signal Conditioning

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The current signals from the LPRM detectors are transmitted to the LPRM amplifiers in the main control room. The current signal from a chamber is transmitted directly to its amplifier through shielded cable. The amplifier is a linear current amplifier whose voltage output is proportional to the current input and therefore proportional to the magnitude of the neutron flux. The amplifier output is "read" by the digital processing electronics. The digital electronics applies hardware gain corrections, performs filtering and applies the LPRM gain factors. The digital electronics provides suitable output signals for the computer recorders, annunciators, etc. The LPRM amplifiers also isolate the detector signals from the rest of the processing so that individual faults in one LPRM signal path will not affect other LPRM signals.

The LPRM signals are indicated on the reactor console. When a central control rod is selected for movement, the LPRM values associated with the nearest 16 LPRM detectors are displayed on console readouts. Each of the four axially spaced LPRM detector signals from each of the four LPRM assemblies are displayed. The operator can readily obtain readings of all the LPRM amplifiers by selecting the control rods in the correct order or by selecting summary LPRM screens on digital operator displays. Subsection 7.7.1.1 describes in greater detail the indications on the reactor console.

Trip Functions

The trip functions for the LPRM provide trip signals to activate displays and annunciators. The outputs for the trip functions are designed to go to the "tripped" state on loss of power to the processing electronics. Table 7.6-7 indicates the trips.

The trip levels can be adjusted to within ± 0.1 of full-scale deflection, and the expected error is approximately ± 1 percent of full scale.

Testability

LPRM channels are calibrated using data from previous full power runs and TIP data and are tested.

7.6.1.13.5.2 Environmental Considerations

Each individual chamber of the assembly is a moisture-proof, pressure-sealed unit. The chambers are designed for the environmental conditions identified in NEDO 31558A per References 7 and 8 in Appendix A.

7.6.1.13.6 Average Power Range Monitor System

7.6.1.13.6.1 Equipment Design

Description

The APRM system has four APRM channels, each of which uses input signals from a number of LPRM channels. Each of the four APRM channels provide inputs to four two-out-of-four voter channels. Two of the voter channels are associated with each of the trip systems of the Reactor Protection System. All four APRM channels are associated with both of the Reactor Protection System trip systems in that they provide inputs to each of the four voter channels.

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Each APRM also includes an Oscillation Power Range Monitor (OPRM) Upscale Function which monitors small groups of LPRM signals to detect thermal-hydraulic instabilities. The OPRM Upscale function is enabled in the intended region on the plant power/flow map. The OPRM Upscale Function receives input signals from the LPRMs within the reactor core, which are combined into cells for evaluation by the OPRM algorithms. An OPRM Upscale trip function is generated from an APRM channel when the period based detection algorithm (basis for the safety analysis) in that channel detects oscillatory changes in the neutron flux, indicated by the combined signals of the LPRM detectors in a cell, with the period confirmations and relative cell amplitude exceeding specific setpoints. One or more cells in a channel exceeding the trip conditions will result in a channel trip. An OPRM Upscale trip is also issued from any channel if either the growth rate or amplitude based algorithms detect growing oscillatory changes in the neutron flux for one or more cells in that channel.

Power Supply

The APRM channels receive power redundantly from the 120-V ac supplied RPS buses. Each APRM two-out-of-four voter channel receives power from the same 120 V ac power as the Reactor Protection System trip system with which it is associated.

Signal Conditioning

The APRM channel uses digital electronic equipment which averages the output signals from a selected set of LPRMs, generates trip outputs via the two-out-of-four voter channels, and provides signals to readout equipment. Each APRM channel can average the output signals from up to 43 LPRM channels. Assignment of LPRM channels to an APRM is shown in Figure 7.6-20. The letters at the detector locations in Figure 7.6-20 refer to the axial positions of the detectors in the LPRM detector assembly. Position A is the bottom position, position B and C are above position A, and position D is the topmost LPRM detector position. The pattern provides LPRM signals from all four core axial LPRM detector positions throughout the core. Some LPRM detectors may be bypassed, but the averaging logic automatically corrects for these by removing them from the average. The APRM value calculated from the LPRM inputs is adjusted by a digitally entered factor to allow calibration of the APRM.

Each APRM channel calculates a recirculation flow signal which is used to determine the APRM's flow-biased rod block and scram setpoints. Each signal is determined by summing the flow signals from the two recirculation loops (Figure 5.5-2). These signals are sensed from two flow elements, one in each recirculation loop. The differential pressure from each flow element is routed to four differential pressure transducers (eight total). The signals from two differential pressure transducers, one from each flow element, are routed to two inputs to the APRM digital electronics. Table 7.6-8 indicates the flow function trips.

The APRM Channel Check surveillance will include a step to confirm that an APRM self-test is still running. It will also include a step to confirm that the RBM self-test is still running since the RBM hardware performs the recirculation flow comparison checks, however, the alarm is bypassed when the reactor mode switch is not in the "RUN" position. A surveillance (channel check) finding that the self-test is not operating in both RBMs (means the recirculation flow comparison function is not available) will not automatically result in any APRM/OPRM channel being declared inoperable, but will result in an increased rate of "flow comparison" manual surveillances. The flow comparison surveillance will be

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performed at nominally hourly frequencies to correspond to the self-test frequencies assumed in the unavailability analysis for the PRNM system.

All APRM channels are powered redundantly, via intermediate low voltage dc power supplies, from both the "A" and "B" Reactor Protection System 120 V ac power buses. The LPRM signal processing equipment is powered by the same sources as their associated APRM channels.

Trip Function

The digital electronics for the APRM channel provides trip signals directly to the Reactor Manual Control System and via the APRM two-out-of-four voter channels to the Reactor Protection System (RPS). Any two unbypassed APRM channels, via the APRM two-out-of-four voter channels, can initiate an RPS trip in both RPS trip systems. Any one unbypassed APRM can initiate a rod block. Table 7.6-9 lists the APRM trip functions. A simplified circuit arrangement is shown in Figure 7.6-21.

The APRM simulated thermal power upscale rod block and scram trip setpoints are varied as a function of reactor recirculation flow. The slope of the upscale rod block and scram trip response curves is set to track the required trip setpoints with recirculation flow changes. Subsection 7.7.1.1.3.5 discusses the thermal power monitor and the APRM in greater detail.

At least two unbypassed APRM channels must be in the upscale or inoperative trip state to cause an RPS trip output from the APRM two-out-of-four voter channels. In that condition, all four voter channels will provide an RPS trip output, two to each RPS trip system. If only one unbypassed APRM channel is providing a trip output, each of the four APRM two-out-of-four voter channels will have a half-trip, but no trip signals will be sent to the RPS. The trips from one APRM can be bypassed by operator action in the control room. Trip outputs to the RPS are transmitted by removing voltage to a relay coil, so loss of power results in actuating the RPS trips. A simplified APRM/RPS interface circuit arrangement is shown in Figures 7.2-3a and 7.2-3b.

In the startup mode of operation, the APRM "fixed" upscale trip setpoint is set down to a low level. This trip function is provided in addition to the existing IRM upscale trip in the startup mode. The trip settings are listed in Table 7.6-9.

The trip functions are performed by digital comparisons in APRM electronics. The APRM flux value is developed by averaging the LPRM signals and then adjusting the average to be APRM power. The APRM power is processed through a first order filter with a six second time constant to calculate simulated thermal power. These calculations are all performed by the digital processor and result in a digital representation of APRM and simulated thermal power. For each RPS trip and rod block alarm, the APRM power or simulated thermal power, as applicable (see Table 7.5-4), is digitally compared to the setpoint (which was previously entered and stored). If the power value exceeds the setpoint, the applicable trip is issued.

Testability

The APRM channels are calibrated using data from previous fullpower runs and are tested by procedures in the applicable instruction manual. Each APRM channel can be tested individually for the operability of the APRM scram and rod blocking functions by introducing test signals.

7.6.1.13.6.2 Environmental Considerations

All APRM equipment is installed and operated in the main control room environment as described in Table 3.11-4.

7.6.1.13.7 Rod Block Monitor System

7.6.1.13.7.1 Equipment Design

Circuit Description

The RBM has two channels; each channel uses input signals from a number of LPRM channels. A trip signal from either RBM channel initiates a rod block. One RBM channel can be bypassed without loss of subsystem function. The minimum number of LPRM inputs required for each RBM channel to prevent an instrument inoperative alarm is four when using four LPRM assemblies, three when using three LPRM assemblies, and two when using two LPRM assemblies (Figures 7.6-22 and 7.6-22(a)).

Power Supply

The RBM power is supplied by two pairs of redundant dc power supplies, one pair for each RBM. Each dc supply in a pair is supplied by one of the two 120-V ac RPS buses, one RBM per bus.

Signal Conditioning

The RBM signal is generated by averaging a set of LPRM signals. The LPRM signals used depends on the control rod selected. Upon selection of a rod for withdrawal or insertion, the conditioned signals from the LPRMs around the rod will be automatically selected by the two RBM channels. (Figure 7.6-22 shows examples of the four possible LPRM/selected rod assignment combinations.) For a typical non-edge rod, each RBM channel averages LPRM inputs from two of the four B-position and D-position detectors, and all four of the C-position detectors. A-position LPRM detectors are not included in the RBM averages, but are displayed to the operator. When a rod near, but not at, the edge of the core is selected, where there are fewer than four, but at least two, LPRM strings around the rod, the number of detectors used by the RBM channels is either six or four depending on how many LPRM strings are available. If a detector has been bypassed in the LPRM system, that detector is automatically deleted from the RBM processing and the averaging logic is adjusted to average only the remaining detectors.

After selection of a control rod, each RBM channel calculates the average of the related LPRM detectors and calculates a gain factor that will adjust the average to 100. Thereafter, until another rod is selected, the gain factor is applied to the LPRM average to obtain the RBM signal value. The RBM signal value is compared to RBM trip setpoints.

When a peripheral rod is selected, or if the APRM value from the RBM's associated APRM is below the automatic bypass level (approximately 30% power), the RBM function is automatically bypassed, the rod block outputs are set to "permissive", and the RBM average is set to zero.

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Each RBM channel receives the total recirculation flow and status from all four APRMs using high-speed fiber optic communication links to provide circuit isolation. The RBM channel provides a trouble alarm (flow compare) when the difference between max and min values for total recirculation flow exceeds a user defined setpoint (typically 10%), however, the alarm is bypassed when the reactor mode switch is not in the "RUN" position.

A surveillance (channel check) finding that the self-test is not operating in both RBMs (means the recirculation flow comparison function is not available) will not automatically result in any APRM/OPRM channel being declared inoperable, but will result in an increased rate of "flow comparison" manual surveillances. The flow comparison surveillance will be performed at nominally hourly frequencies to correspond to the self-test frequencies assumed in the unavailability analysis for the PRNM system.

Trip Function

The RBM supplies a trip signal to the Reactor Manual Control System to inhibit control rod withdrawal. The trip is set whenever the RBM signal value exceeds the RBM setpoint. There are three different setpoints, each a percentage above the RBM initial value of 100. Figure 7.6-22(b) illustrates the trip setpoints. The particular setpoint that is applied is selected based on the simulated thermal power value from the RBM's associated APRM channel (an alternate APRM channel is assigned and is automatically used for inputs if the primary APRM channel is bypassed or inoperative). Higher APRM simulated thermal power values select a lower setpoint. That is, at higher power levels, the percentage increase in the RBM value allowed is less than at lower power levels.

Below 30 percent power, fuel damage cannot occur for any single rod withdrawal; hence, the RBM system is automatically bypassed. The low trip setpoint (LTSP) is enforced between 30 percent and 65 percent power, the intermediate trip setpoint (ITSP) is enforced between 65 percent and 85 percent power, and the high trip setpoint (HTSP) is enforced between 85 percent and 100 percent power. The core power input used to automatically select the applicable RBM trip is provided by the APRM. The RBM system is automatically bypassed if the control rod has one or more adjacent fuel bundles comprising the outer boundary of the core. The operator can bypass one of the two RBMs at any time. Either RBM can inhibit control rod withdrawal (Figure 7.6-16, Sheet 1). Table 7.6-10 indicates the trips.

Isolation Separation and Redundancy - The RBM channels A and B are redundant, separate, and isolated. The only exception is the sharing of LPRM C level detectors by both RBM channels. The impact on the availability of the RBM system due to the sharing of the C level detectors is small (Reference 9) and the benefits of the improved signal response far outweigh any perceived loss in signal redundancy; some other salient features are:

- a. Redundant, separate, isolated rod selection information (including isolated contacts for each rod selection push button) provided directly to each RBM channel.
- b. Independent, isolated RBM level readouts and status displays from the RBM channels.
- c. Independent, isolated rod block signals from RBM to the RMCS circuitry.

Testability

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The RBM channels are tested and calibrated. The RBMs are functionally tested by introducing test signals into the RBM's channels.

7.6.1.13.7.2 Environmental Considerations

See the description for the APRM.

7.6.1.13.8 Traversing In-Core Probe System

7.6.1.13.8.1 Equipment Design

Circuit Description

The TIP system includes five TIP machines. Each TIP machine includes the following components:

- a. One TIP
- b. One drive mechanism
- c. One indexing mechanism
- d. Up to 10 in-core guide tubes
- e. One chamber shield
- f. One guide tube valve.

The subsystem allows calibration of LPRM signals by correlating TIP signals to LPRM signals as the TIP is positioned in various radial and axial locations in the core. The guide tubes inside the reactor are divided into groups. Each group has its own associated TIP machine.

Physical Arrangement

A TIP drive mechanism uses a fission chamber attached to a flexible drive cable. The cable is driven from outside the drywell by a gearbox assembly. The flexible cable is contained by guide tubes that penetrate the reactor core. The guide tubes are a part of the LPRM detector assembly. The indexing mechanism allows the use of a single detector in any one of ten different tube paths. The tenth tube is used for TIP cross-calibration with the other TIP machines.

The control system provides both manual and semiautomatic operation. Electronics on the TIP panel amplifies and displays the TIP signal. Core position versus neutron flux is recorded on an X-Y recorder in the relay room and is provided to the computer. A block diagram of the drive system is shown in Figure 7.6-23.

The heart of each TIP machine is the probe (Figure 7.6-24). It consists of a detector (fission chamber) and the associated signal drive cable. The body of the fission chamber is made of titanium with a neutron-sensitive inner coating of uranium-235. The chamber can operate in a neutron flux level of greater than 10^{14} nv. The saturation voltage is approximately 150 V dc (Reference 1).

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The signal current from the detector is transmitted from the TIP to amplifiers and readout equipment by means of signal cable, which is an integral part of the mechanical drive cable. The cable drive mechanism contains the drive motor, the cable takeup reel, an analog probe position indicator for the recorder, and a mechanical counter. The mechanical counter provides digital pulses to the control unit for positioning the TIP at specific locations along the guide tube.

The drive mechanism inserts and withdraws the TIP and its cable from the reactor and provides detector position indication signals. The drive mechanism consists of a motor and drive gearbox that drives the cable in the manner of a rack and pinion. A two-speed motor provides a high speed for insertion and withdrawal (approximately 60 fpm) and a low speed for scanning the reactor core (approximately 7.5 fpm). A takeup reel is included in the cable drive mechanism to coil the drive cable as it is withdrawn from the reactor. This reel makes it possible to connect the TIP and its cable to the amplifier through a connector rather than slip rings. This reduces possible noise and maintenance problems.

The analog position indicator and the mechanical counter (digital) are also driven directly from the output shaft of the cable drive motor. The analog position signal and a flux amplifier output are used to plot neutron flux versus TIP position. The TIP position signal is also available to the Integrated Plant Computer System (IPCS). The digital counter is used to position the TIP in the guide tube with a linear position accuracy of ± 1 in. The digital counter can control TIP positions at the top of the core for initiation of scan and at the bottom of the core for changing to fast withdrawal speed.

A circular transfer machine with 10 indexing points functions as an indexing mechanism. One of the 10 locations is available for access to the guide tube common to all the TIP machines. Indexing to a particular tube location is accomplished manually at the control panel by means of a position selector switch that energizes the electrically actuated rotating mechanism.

The tube transfer mechanism is part of the indexing mechanism and consists of a fixed circular plate containing 10 holes on the reactor side that mate to a rotating single-hole plate. The rotating plate aligns and mechanically locks with each fixed-hole position in succession. The indexing mechanism is actuated by a motor-operated rotating drive. Electrical interlocks prevent the indexing mechanism from changing positions until the probe cable has been completely retracted beyond the transfer point. Additional electrical interlocks prevent the cable drive motor from moving the cable until the probe cable has been completely retracted beyond the transfer point. Additional electrical interlocks prevent the cable drive motor from moving the cable until the transfer mechanisms have indexed to the preselected guide tube location (Figure 7.6-16, Sheet 7).

A valve system is provided with a valve on each guide tube entering the primary containment. These valves are closed except when the TIP system is in operation. A ball valve and a cable shearing valve are mounted in the guide tubing just outside the primary containment. They prevent the loss of containment integrity. A guide tube ball valve is opened manually prior to TIP insertion. The shear valve is used only if containment isolation is required when the TIP is beyond the ball valve and when power to the TIP system fails. The shear valve, which is controlled by a manually operated keylock switch, can cut the

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cable and close off the guide tube. The shear valves are actuated by detonation squibs. The continuity of the squib circuits is monitored by indicator lights in the main control room.

A guide tube ball valve is normally deenergized and is in the closed position. When the ball valve is manually opened, it actuates a set of contacts which gives a signal light indication at the TIP control panel (Figure 7.6-16).

Signal Conditioning

The readout instruments and electrical controls for the TIP machines are mounted in a cabinet in the relay room. Because there are several groups of guide tubes, each with an associated TIP machine, there are also several groups of readout equipment controls mounted in the cabinet. Each set of readout equipment consists of a dc amplifier and a dc power supply for the TIP polarizing voltage. A common X-Y recorder records the flux variations of each scan. An X-Y output is provided for the operator through the man-machine interface of the Integrated Plant Computer System (IPCS).

Testability

The TIP system equipment is tested and calibrated using heat balance data and by use of the common channel.

7.6.1.13.8.2 Environmental Considerations

The equipment and cabling located in the primary containment are designed for the environmental conditions identified in NEDO 31558A per References 7 and 8 in Appendix A.

7.6.1.13.9 Thermal Power Monitor (TPM)

The thermal power monitor (TPM) typically involves the flow-weighted APRM scram in conjunction with a 6-sec time constant circuit.

The APRM has two output signals. The APRM neutron flux signal is representative of the core average neutron flux. The APRM-simulated thermal power (STP) signal represents the fuel surface heat flux. This signal is obtained by passing the neutron flux signal through a nonadjustable, 6-sec first order ("RC") filter to represent the fuel dynamics. A scram signal occurs when

- a. The APRM neutron flux signal exceeds a setpoint that is independent of the recirculation flow rate or
- b. The APRM STP signal exceeds a setpoint that is dependent on the recirculation flow rate.

If the time constant, which affects scram initiation by the TPM, is less than the effective time constant for the fuel for this type of transient, the TPM should provide a conservative measure of the time variation in surface heat flux. However, if the time constant is appreciably larger than that for the fuel, the fixed APRM trip without a time constant would provide the scram protection. The resulting maximum critical power ratio (MCPR) would then be less than that predicted for the TPM scram, which has a lower setpoint.

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A General Electric analysis reported in the Supplemental Reload Licensing Report indicates that with a 6-sec time constant, the TPM scram occurs before the high-neutron-flux scram, because of its lower setpoint of 117 percent nuclear boiler rated (NBR). Therefore, it was appropriate to take credit for TPM scram for the loss of feedwater heater event. Assuming the APRM neutron-flux scram occurs first, the surface heat flux will be below the 117 percent setpoint, and the result will be even less severe.

The TPM used in Fermi 2 is safety grade and is designed to be single-failure proof.

7.6.1.14 Plant Cooling Water Systems Instrumentation and Control

7.6.1.14.1 System Identification

7.6.1.14.1.1 Reactor Building Closed Cooling Water System

The reactor building closed cooling water system (RBCCW) contains three 50 percent-capacity water pumps, two 50 percent-capacity heat exchangers, a makeup tank, a bypass valve for regulating the differential pressure across the supply and return water headers, motorized isolation valves, and a service water supply for discharge of heat from the heat exchangers. During normal operation (with or without RBCCW supplemental cooling in operation), two heat exchangers and two pumps are in service, and one pump is retained in standby. Automatic controls are provided in the RBCCWS to maintain within their operational range the demin water level in the makeup tank, the water temperature at the outlet of the heat exchanger, and the differential pressure between the supply and return water headers.

When the GSW temperature is 60°F or greater, operators have the option of placing the RBCCW supplemental cooling loops in service. Each loop is furnished with two 100 percent-capacity pumps and a plate and frame heat exchanger. Each RBCCW-SC loop takes suction from the RBCCW return header downstream of the RBCCW/EECW system interface, passes this water through a plate and frame heat exchanger in that loop to cool the RBCCW supplemental cooling water with chilled water from the supplemental cooling chilled water (SCCW) system, and discharges the cooled water to the RBCCW supply header just upstream of the RBCCW/EECW interface. When the RBCCW supplemental cooling pumps are in operation, each RBCCW supplemental cooling loop alone provides RBCCW flow to its respective division of EECW; thus the RBCCW supplemental cooling loops operate in parallel with the two 50 percent-capacity RBCCW pumps that service the nonessential loads outside of the EECW loops. Further details of the RBCCW supplemental cooling loops are described under Section 9.2.2.2.

7.6.1.14.1.2 Turbine Building Closed Cooling Water Systems

The TBCCWS contains three 50 percent-capacity water pumps, two 100 percent-capacity heat exchangers, a makeup tank, a bypass valve for regulating the differential pressure across the supply and return headers, and a service water supply for discharge of heat from the heat exchangers. One pump and one heat exchanger are retained on standby. Automatic controls are provided to maintain condensate level in the makeup tank and the water temperature at

the outlet of the heat exchanger outlet to the supply header, and to regulate the differential pressure across the supply and return headers.

7.6.1.14.1.3 Classification

The RBCCWS is classified by regions, as indicated in Subsection 9.2.2. In the regions of the RBCCWS pumps, heat exchangers, makeup tank, and service water system, (including the RBCCW supplemental cooling loops), the classification is Quality Group D. In the regions of the drywell and the emergency equipment cooling water system (EECWS) components, the classification is Category I, Quality Group B and C, respectively. Elsewhere the classification is Quality Group D. The turbine building closed cooling water system (TBCCWS) and the supplemental cooling chilled water (SCCW) system are classified Quality Group D.

7.6.1.14.2 Supporting Systems

7.6.1.14.2.1 Electrical Power

The electrical power for the instrumentation and control of the RBCCWS and TBCCWS is from the 120-V ac circuit which is stepped down from the 480 bus supplying power to the pumps. Power for operating the coils of the pump circuit breakers and associated condition-indicating lights is 130 V dc. A detailed description of the ac and dc power system is contained in Subsection 8.3.1.

7.6.1.14.2.2 Pneumatic Power

In the RBCCWS and TBCCWS, pneumatic power is used for operation of the makeup tank level controller, heat exchanger outlet temperature controller, and water header differential pressure controller. In the RBCCW supplemental cooling loop, pneumatic power is used for operation of the temperature control valves. The pneumatic power is supplied by the air system described in Subsection 9.3.1.

7.6.1.14.2.3 Service Water

Cooling water for the RBCCWS and TBCCWS heat exchangers is taken from the general service water (GSW) system of the plant. Details of the GSW system are contained in Subsection 9.2.1.

GSW also provides the source of condenser cooling water for the supplemental cooling chilled water auxiliary support system to the RBCCW supplemental cooling loops. Details of the SCCW system are contained in Subsection 9.2.9.

7.6.1.14.3 Equipment Design

7.6.1.14.3.1 Initiating Circuits

Normal initiation of pump operation and shutdown in the RBCCWS and TBCCWS is by manual control. Selection of one or two pumps to be operational is made by the operator in

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response to bypass-valve position indications that are activated by two position switches, one closing at the 10 percent valve open position and the other at the 85 percent open position. Auxiliary controllers within the RBCCWS and TBCCWS function automatically to regulate the demin water level in the makeup tank, temperature at the process outlet of the heat exchanger, and the differential pressure across the supply and return headers.

Separate controls automatically regulate the bypass of RBCCW flow around the RBCCW supplemental cooling plate-and-frame heat exchangers to regulate the temperature of the water supplied by the RBCCW supplemental cooling loops when they are in operation. Manual operation of the temperature control valve is described in section 9.2.2.2. When RBCCW supplemental cooling is not in service, the RBCCW differential pressure controller functions to maintain EECW differential header pressure. With RBCCW supplemental cooling in operation, EECW differential header pressure is maintained by operation of the RBCCW supplemental cooling pumps themselves. In this mode of operation, the RBCCW differential pressure controller does not function to maintain EECW header differential pressure; however, it functions to allow two RBCCW pump operation outside of the RBCCW supplemental cooling loops.

7.6.1.14.3.2 Logic and Sequencing

The control logic diagrams for the RBCCWS and TBCCWS pumps are illustrated in Figures 7.6-25(1) (excluding RBCCW Supplemental Cooling) and 7.6-26, respectively. Operation of the pumps is prevented if the level in the associated makeup tank or the suction pressure at the pump inlet is at or below the lower limit of the established control range (Table 7.6-2). The control logic diagram for the RBCCW supplemental cooling loops is illustrated in Figure 7.6-25(2). RBCCW supplemental cooling pump operation is prevented on the same RBCCW makeup tank low level level signal. The RBCCW supplemental cooling pumps are not equipped with low pump suction trips. Equipment protection is provided instead with low flow trips. An EECW start signal initiates closure of the RBCCW/EECW system isolation valves; a low flow condition in the RBCCW supplemental cooling loops results, which trips the RBCCW supplemental cooling pumps. The low flow trip is not required to assure the operation of the EECW during or after receipt of the initiation signal.

Four sensors for each of the two parameters (demin water level and suction pressure) supply inputs to a "one-out-of-two taken twice" logic arrangement to inhibit pump operation if these parameters are below normal. Activation of the logic for low suction pressure trips only the pump at which the low pressure is sensed.

Activation of the logic for low demin water level in the makeup tank trips all the pumps in that system.

7.6.1.14.3.3 Bypasses and Interlocks

Loss of offsite power, high drywell pressure, or drop in differential pressure across the supply and return headers of the RBCCWS results in isolation of coolant circuits that are not essential in an emergency and in automatic initiation of EECWS operation.

Restoration of RBCCWS operation after power becomes available is by manual control. On loss of offsite power, the TBCCWS is deactivated. Initiation of system operation after power becomes available is by manual control.

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7.6.1.14.3.4 Redundancy and Diversity

The RBCCWS contains two divisions of flow, one of which is a redundant division. Under normal conditions both divisions are operational. On loss of offsite power, high drywell pressure, or drop in differential pressure across the supply and return headers, the EECWS automatically takes over the function of supplying cooling water to vital equipment served by the two divisions (Section 7.3).

7.6.1.14.3.5 Testability

The controls of the RBCCWS and TBCCWS are fully testable during normal operation of the plant as well as during shutdown periods.

7.6.1.14.4 Environmental Considerations

The instrumentation and control of the RBCCWS and TBCCWS is designed to function with reliability under the environmental conditions that would be encountered under normal or postulated abnormal conditions. These conditions are defined in Tables 3.11-3 and 3.11-4.

7.6.1.14.5 Operational Considerations

7.6.1.14.5.1 Normal

Under normal plant conditions, the RBCCWS and TBCCWS are operated with one or two pumps, initiated manually, to meet the cooling needs as they arise. In the event of a malfunction of a pump or heat exchanger, operation of the standby unit is manually initiated, and the malfunctioning unit is deactivated by manual action or by automatic trip. When the GSW supply temperature exceeds 60°F, the operators have the option of initiating the RBCCW supplemental cooling loop(s) to provide additional RBCCW cooling capacity.

7.6.1.14.5.2 Operator Information

Attention of the operator is secured when the need arises for an increase or decrease of pumping capacity in the RBCCWS or TBCCWS by control room annunciators which indicate the near-open or near-closed positions of the bypass valve. Readout instruments are provided locally to display the demin water level and gas pressure in the makeup tanks and the pumps discharge pressure. Control room annunciators also alert operators to abnormal demin water level or gas pressure in the makeup tanks. Readout instruments provided in the main control room display process outlet temperature of the heat exchangers and supply/return header pressure. With the RBCCW supplemental cooling loops in operation, the supply temperature to the EECW loops is indicated by the EECW divisional supply temperatures to the drywell. High and low demin water level, low gas pressure in the makeup tanks, and drop in differential pressure or in the supply and return headers of either EECW division are alarmed. Recorders register the initiation, shutdown, or tripout of the pumps in each system.

7.6.1.15 Fuel Pool Cooling and Cleanup System Instrumentation and Control

7.6.1.15.1 System Identification

7.6.1.15.1.1 Function

The purpose of the fuel pool cooling and cleanup system (FPCCS) instrumentation and control is to provide protection for the system from overheating and to provide the operator with information concerning the effectiveness of operation of the system.

7.6.1.15.1.2 Instrumentation Classification

The FPCCS is not a safety-related system. Therefore, the instrumentation is classified as nonessential. The instrumentation is a standard industrial type for which performance has been proven by years of service throughout the industry.

7.6.1.15.2 Power Sources

The FPCCS instrumentation is fed from the plant instrumentation bus. No backup power source is necessary since the FPCCS is not a safety-related system. The system wiring is protected against short circuit by appropriate fuses. Thus, a short circuit within the FPCCS wiring has only a local effect, which can be corrected without shutting down the FPCCS.

7.6.1.15.3 Equipment Design

The equipment for the FPCCS system is comprised of circulating pumps, heat exchangers, filters, surge tanks, and required piping, valves, instrumentation and controls. The spent fuel pool water is continually circulated in a closed loop except when the FPCCS is used to drain the reactor well and dryer-separator pit. The FPCCS functions and description are explained in Section 9.1.3.

The operating configurations for the FPCCS are obtained by means of manually operated valves and weir gates. The system operation is monitored by instrumentation that provides the operator with a means of evaluating system performance and also provides alarms in the event of a malfunction.

Irradiated components and spent fuel require cooling in addition to the shielding provided by the spent fuel pool. The cooling is provided on a continuous basis during normal operation of the FPCCS. The instrumentation of the FPCCS measures conductivity, temperature, system flow rate, level, and leakage. Indications of the system performance are furnished to the plant operators.

7.6.1.15.3.1 Conductivity

The ionic concentration in the water leaving each demineralizer unit is monitored by a conductivity-measuring system consisting of a conductivity cell, indicating transmitter, and a recorder.

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7.6.1.15.3.2 Pump Discharge Pressure

The discharge pressure of each FPCCS pump is monitored by its pressure-indicating switch. If the pump discharge pressure falls below the switch setpoint, a contact set of the switch actuates a local indicator lamp. Another flow switch contact set opens in the actuating path of the alarm annunciator "Fuel Pool Cooling Trouble."

7.6.1.15.3.3 Temperature

The temperature of the liquid in the piping system associated with the FPCCS pumps, heat exchangers, and valves is monitored by individual temperature elements, and the observed temperatures are recorded by a multipoint recorder. Temperature elements monitor the operation of the FPCCS. In each instance, the temperature levels observed by the temperature monitoring circuits are recorded on a multipoint recorder in the main control room.

7.6.1.15.3.4 Leakage

The leakage rates past the refueling bellows and the gate seal or drywell-to-reactor-well seal are monitored by circuit arrangements consisting of a flow switch and alarm annunciator (Figure 9.1-23). Liquid leakage past the refueling bellows is caught and routed past the flow switch before being piped to the drywell equipment drain sump. When the leakage liquid flow rate through the flow switch exceeds the high flow setpoint, flow switch contact sets energize an indicator lamp and initiate the alarm annunciator, "Fuel Pool Cooling Trouble."

7.6.1.15.3.5 Level

The water levels in both the spent fuel pool and the skimmer surge tank are monitored by alarm annunciator circuits. The spent fuel pool water level is monitored by a level sensor switch assembly that trips if the water level rises above the high level setpoint. On the high-level trip, the alarm annunciator activates and a local indicator lamp lights. Lowered water level automatically resets the level-sensing alarm circuitry.

The skimmer surge tank water level is monitored for both high and low water level. A high-water condition trips a level switch, which activates the alarm annunciator and lights a white indicator lamp. Subsequent lowering of the water level below the high trip setpoint automatically resets the alarm circuitry. Excessively low water levels in the skimmer surge tank cause a trip in the low water level sensor level switch. An alarm annunciator actuates and a white indicator lamp lights on the trip. The alarm circuit automatically resets with normal water level restored in the tank.

All annunciator alarms are located in the main control room panel with their attendant instrumentation located at local panels. The sensors are located at their respective monitoring positions. Specific alarm points are listed in Table 7.6-11.

To comply with NRC Order EA 12-051, spent fuel pool levels are monitored by Primary and Backup instrument channels. Each channel consists of a seismically installed level probe in the spent fuel pool, a signal processor with battery backup, and a remote level indicator. Primary and backup indicators are located in the main control room and reactor building

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second floor which are capable of supporting the following spent fuel pool actual water levels:

Level 1: Level that is adequate to support operation of the normal fuel pool cooling system, i.e. the surface of the water is maintained at Elevation 683'6" by scuppers that act as skimmers and wave suppressors.

Level 2: Level that is adequate to provide substantial radiation shielding for a person standing on the spent fuel pool operating deck, i.e. 18' above the top of the fuel in the storage racks (el. 679' 1/8")

Level 3: Level where fuel remains covered and actions to implement make-up water addition should no longer be deferred, i.e. <12 in. above the top of the fuel in the storage racks (el. 661' 1/8")

7.6.1.15.4 Testability

Because the FPCCS is usually in service during plant operation, satisfactory performance is demonstrated without the need for any special inspection or testing beyond that specified in the manufacturer's instructions.

7.6.1.15.5 Environmental Considerations

The FPCCS is not required for safety purposes, nor is it required to operate after the DBA. The FPCCS is required to operate in normal plant environment only.

7.6.1.15.6 Operational Considerations

The FPCCS instrumentation and control is not required for safe operation of the plant. It provides a means of monitoring parameters of the system and protecting the system.

7.6.1.16 Deleted

7.6.1.17 Control Air System

7.6.1.17.1 System Function

The station air system normally provides control air for operation and control of various plant systems that are safety related as well as those that are nonsafety related. If the control air system pressure drops to 85 psig, indicating an abnormal loss of air pressure, the control air compressors will automatically start for the purpose of supply to its associated division. If pressure drops to 75 psig, the control air system will be isolated from the station air system and also the nonsafety related plant systems. See Subsection 9.3.1 for a more detailed discussion of the control air system.

7.6.1.17.2 Classification

The noninterruptible portion of the control air system, including the compressors, filters, dryers, afterfilters, and receivers, is classified as Category I. The interruptible portion of the

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control air system, including its filter, dryer, afterfilter, and receiver, is classified as nonseismic.

7.6.1.17.3 Supporting Systems

7.6.1.17.3.1 Electrical Power

The electrical power required for operation of the control air system is supplied from the 480-V ac bus as described in Subsection 8.3.1.

7.6.1.17.3.2 Service Water

The cooling water required for operation of the control air system is supplied from the RBCCW or EECW system as described in Subsection 9.2.2.

7.6.1.17.4 Equipment Design

7.6.1.17.4.1 Initiating Circuits

Initiation of the noninterruptible control air system compressors occurs automatically on detection of low control air header pressure (85 psig), loss of offsite power, or a level 2 LOCA signal. In addition, isolation of the noninterruptible control air system from the station air and interruptible control air systems occurs automatically on detection of low noninterruptible control air header pressure (75 psig) or a loss of offsite power. Normally, the interties between the station air system and control air systems are open and the noninterruptible control air compressors are in auto standby.

7.6.1.17.4.2 Logic and Sequencing

Pressure sensors are provided to detect low control air system pressure. Activation of the logic for low control air header pressure causes isolation of the control air system and startup of the control air compressor.

7.6.1.17.4.3 Bypasses and Interlocks

A drop in control air header pressure results in isolation to prevent the use of noninterruptible control air by nonessential control air users. If offsite power is available, the control air compressors start prior to system isolation. On loss of offsite power, the control air compressors are started by the automatic load sequencer when diesel generator power becomes available. See Subsection 8.3.1.1.7 for a more detailed discussion of the automatic load sequencer.

7.6.1.17.4.4 Redundancy and Diversity

The noninterruptible control air system consists of two divisions for redundancy. Under normal operating conditions both divisions are supplied from the station air system. On loss of noninterruptible control air pressure, the control air compressors automatically take over the function of supplying control air to the vital equipment served by the two divisions of the

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control system. The interruptible control air system is supplied separately from the station air system. A normally closed tie from this system can be opened to provide air supply (if available) to the Division II noninterruptible control air system in the event of loss of its normal and control air compressor supply.

7.6.1.17.4.5 Testability

The controls for the control air system are fully testable during normal plant operation as well as during shutdown periods.

7.6.1.17.5 Environmental Considerations

The instrumentation and control of the control air system is designed to function with reliability under the environmental conditions that would be encountered under normal or postulated accident conditions. These conditions are defined in Table 3.11-4.

7.6.1.17.6 Operational Considerations

7.6.1.17.6.1 Normal

Under normal operating conditions, station air is the supply to both control air systems through their respective dryers. Under these conditions, the noninterruptible control compressors are normally in standby. In the event of low control air header pressure, the noninterruptible control air system divisions are isolated from the interruptible air users. The control air compressors will start automatically prior to this isolation.

7.6.1.17.6.2 Operator Information

Readout instruments are provided in the main control room to display and record the Division I and II control air pressures.

Recorders register the automatic initiation of the control air system compressors.

7.6.1.18 Alternate Rod Insertion

7.6.1.18.1 Equipment Identification

The alternate rod insertion (ARI) components of the CRD system are designed to mitigate the potential consequences of an anticipated transient without scram (ATWS) event. The ARI components are redundant to the RPS.

7.6.1.18.2 Equipment Design

7.6.1.18.2.1 Initiating Circuits

There are three initiating signals used for the ARI logics, namely:

- a. Reactor dome high pressure

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- b. Reactor low water level 2
- c. Manual initiation in the main control room

Any one of the above signals can initiate the divisional ARI logics as shown in Figure 7.7-3, Sheet 4. Additional immediate response to the initiation signals includes the recirculation pump motor generator field breaker trip (see Subsection 7.7.1.2.3.1).

7.6.1.18.2.2 Logic

Two divisional ARI logic systems are provided: Division I, consisting of logic channels A and C, and Division II for logic channels B and D. The signal to insert the control rods is generated in two separate divisions on two-out-of-two logic channels in a given division.

The ARI logic receives reactor dome pressure and water level signals from the nuclear boiler system. The logic causes automatic energization of the ARI solenoid valves when either the reactor high-pressure trip set point or low-water level 2 set point is reached. The ARI logic can also be initiated manually from the main control room. Each ARI logic channel is provided with a disarmed/armed pushbutton switch. Both pushbutton switches in a given division must be depressed to energize the ARI logic and initiate control rod insertion. The ARI initiation signals are designed to seal in the initiation logic to ensure completion of the ARI function until it is reset manually. A reset pushbutton per division is provided in the main control room to clear the ARI logic. A timer is used in each of the ARI logic channels to inhibit the reset function for approximately 30 seconds after the initiation signal is received. A 30-second time delay is selected to ensure completion of the ARI function before the logic can be reset.

The initiation of the two separate ARI logics results in the energization of eight Class 1E dc solenoid valves (four per division). Two of these, F160A and B, vent the scram air supply line just downstream of the F110A and B backup scram valves. (Refer to Figure 7.6-36). These ARI valves also act to block the supply of air to the scram header. Check valves F161A and B provide an air-flow path around the F160 valves in the event one or more of them fails. Four additional ARI valves, F162A, B, C, and D, vent the A and B scram header to the atmosphere. As the header depressurizes, the scram valves at each hydraulic control unit will spring open scrambling the rods. Two ARI valves, F163A and B, vent the scram air header to the scram discharge volume drain and vent valves, closing these valves and isolating the scram discharge volume. All eight ARI valves are normally deenergized.

7.6.1.18.2.3 Annunciation and Indication

The manual initiation pushbutton switch in the main control room activates an annunciator window whenever it is placed in armed position. A separate annunciator window is activated upon initiation of the ARI logic circuits. The open and close position of the ARI solenoid valves are also indicated in the main control room.

7.6.1.18.2.4 Testability

Four separate ARI initiation logic channels are provided to permit maintenance, repair, test, or calibration of all circuit devices (at power) up to but not including the final trip devices

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(ARI solenoid valves). Each ARI logic channel is provided with a test jack and indicating lights to verify logic activation in any given division.

7.6.1.19 Safety/Relief Valves

7.6.1.19.1 System Identification

The nuclear pressure relief system is designed to prevent over-pressurization of the nuclear system that could lead to the failure of the reactor coolant pressure boundary.

7.6.1.19.2 Safety/Relief Valve Equipment Design

Safety/relief valves (SRVs) are dual-functioning types: automatic self-actuating and solenoid operated. The valves are self-actuated when reactor pressure exceeds spring set pressures that are adjustable in range. The SRVs are divided into three spring-set-pressure groups. The first group consists of five valves set to open when vessel pressure exceeds 1135 psig, the second group consists of five valves set to open when vessel pressure exceeds 1145 psig, and the third group consists of five valves set to open when vessel pressure exceeds 1155 psig. The solenoid-operated air pilot valves permit remote manual or automatic opening. The pilot valve controls the pneumatic pressure applied to an air cylinder operator that controls valve opening and closing. Each valve associated with ADS has an accumulator to store pneumatic energy with sufficient capacity for several relief valve operations. The valves are capable of remote manual opening at any pressure above 100 psig and staying open, once opened, until pressure decreases to 50 psig. Five of the SRVs are used for ADS (Subsection 7.3.1.2.2). Two SRVs are used for low-low setpoint relief (Subsection 7.6.1.19.9).

7.6.1.19.3 Initiating Circuits

Reactor pressure exceeding the setpoint actuates the SRV. The SRV can also be manually actuated (by remote manual switch) or automatically by the ADS and low-low setpoint relief logic.

7.6.1.19.4 Logic and Sequencing

No automatic logic is involved in the overpressure safety function of the SRVs. (See Subsection 7.6.1.19.9 for low-low setpoint relief logic and Subsection 7.3.1.2.2 for the ADS logic.)

7.6.1.19.5 Bypasses and Interlocks

Bypasses are not used in the normal SRV function. An arming circuit is used as an interlock to prevent the low-low setpoint valves from prematurely actuating during normal plant operation. The interlock is required because the reopening setpoint of the low valve is near the normal reactor operating range.

7.6.1.19.6 Redundancy and Diversity

Seven of the SRVs and their respective monitoring system pressure switches are powered by Division I. The other eight SRVs and their respective monitoring system pressure switches are powered by Division II. The SRVs are designed to meet ASME Boiler and Pressure Vessel Code Section III, and therefore diversity is not provided.

7.6.1.19.7 Actuated Devices

Relief valves are actuated by the following two means:

- a. Self-actuation by reactor pressure exceeding the spring set pressure setpoint
- b. Solenoid pilot operation by remote manual control or automatically by ADS (Subsection 7.3.1.2.2) or low-low set relief logic.

7.6.1.19.8 Separation

Logic circuitry, controls, and instrumentation are designed to maintain physical and electrical separation between Division 1 and Division II.

7.6.1.19.9 Low-Low Setpoint Relief Logic

Two of the 15 SRVs are provided with lower opening and closing setpoints that override the normal setpoints following initial opening of one or more SRVs using the normal setpoint. Logic for this low-low setpoint consists of reactor pressure transmitters that are enabled (armed) by a separate reactor high-pressure (scram) signal and a signal that one or more SRVs are open. The two low-low set SRVs have slightly different opening and closing setpoints, thus ensuring that only one SRV at a time will reopen on increasing pressure after initial SRV actuation and closure. This arrangement serves to damp reactor pressure surges.

The low-low set logic automatically seals itself into control of the two selected valves and actuates an annunciator in the control room. This logic remains sealed in until manually reset by the operator.

Since the two valves will already have opened when reactor pressure exceeded the original (normal) overpressure safety setpoint, the low-low set logic acts to hold the valves open past their normal reclose points until the pressure decreases to a predetermined "low-low" setpoint. Thus, the valves remain open longer than the other SRVs.

The low-low set logic is designed with redundancy and single-failure criteria; that is, no single electrical failure will (1) prevent any low-low set function from operating, and (2) cause inadvertent seal-in of low-low set logic.

The two valves associated with low-low set are arranged in two independent secondary setpoint groups or ranges (low and high). The low- and high-pressure groups consist of one valve each, having both reopen and reclose setpoints independently and uniquely adjustable. These are set considerably lower than their normal SRV setpoints.

Each SRV valve has its own set of two tailpipe pressure switches. These pressure switches are arranged in two divisions for each low-low set valve so that opening of a single SRV will

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result in arming of both divisions of low-low set logic. The single-failure criterion is thus met for this function.

The operability of the low-low function is dependent on the operability of the instrumentation channels providing inputs to the low-low set logic. Besides the reactor steam dome pressure-high and low-low set pressure setpoint signals, each division of the low-low set logic normally receives at least five SRV pressure switch inputs from one group of SRVs with the same pressure setpoint. The low-low set logic is capable of performing its function (i.e., preventing multiple actuations of the SRVs) even if both pressure switches associated with one SRV tailpipe become inoperable. The loss of SRV position indication in this case will not challenge the assumptions of the safety analyses for a stuck-open SRV event (see Section 15.1.4).

7.6.1.19.10 Low-Low Setpoint Relief Logic Testability

The SRV system has two low-low setpoint logics, one in Division I and one in Division II. Either one can perform the low-low set function. Each valve has its own set of pressure switches. The sensors are arranged in two separate channels per each division and two-out-of-two logic is used to open the valves. Thus, the sensors and logic of each channel can be tested separately without actually actuating the valves. Indicator lights are provided to facilitate logic testing.

7.6.1.19.11 Environmental Considerations

The solenoid valves and their cables, pressure switches for indication, and the SRV operators are the only SRV controls located inside the drywell. All equipment will meet applicable environmental requirements.

7.6.1.19.12 Operator Information

A temperature element is installed on the SRV discharge piping several feet from the valve body. The temperature element is connected to a multipoint recorder in the control center to provide a means of detecting SRV leakage during plant operation. When the temperature in any SRV discharge piping exceeds a preset value, an alarm is sounded in the control room. The alarm setting is far enough above normal (rated power) drywell ambient temperatures to avoid spurious alarms, yet low enough to give early indication of significant SRV leakage.

Valve actuation is monitored by the SRV open/closed monitoring system (SRVOCMS) pressure switches connected to the SRV discharge line. An open SRV pressurizes the discharge line, which actuates the pressure switch that provides the input to the SRV monitor circuit. The monitor circuit provides inputs to SRV annunciators in the control room, to the Integrated Plant Computer System (IPCS), to the open-close indicators in the control room, and to the low-low setpoint relief logic. The SRVOCMS setpoint is selected so the pressure switch will actuate when the SRV opens in the expected operating range but will not respond to a leaking SRV. The expected operating range of the SRVOCMS is from 200 psig to the SRV safety function actuation point. The SRVOCMS uses Class 1E power and has a power supply monitor with annunciation upon loss of power. If a pressure switch becomes inoperable, SRV position indication relies on monitoring the SRV tail-pipe temperature recorder in the relay room as a backup means for determination of an open SRV.

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7.6.1.20 Rod Worth Minimizer (RWM)

7.6.1.20.1 System Identification

7.6.1.20.1.1 Function

The objective of RWM is to provide backup to the operator for control rod pattern control in reactor startup and for control rod manipulation during low power operations. The nuclear measurement analysis and control RWM function is described in Reference 7.

7.6.1.20.1.2 Classification

The RWM is used for power generation only.

7.6.1.20.2 Power Source

The RWM receives its power from the 120V AC uninterruptible power supply.

7.6.1.20.3 Description

The RWM microcomputer system is a stand-alone microcomputer-based system with an RWM operator display and a continuous operating self-test feature that enforces adherence to established startup, shutdown, and low power level control rod procedures. The RWM microprocessor prevents the operator from establishing control rod patterns that are not consistent with prestored RWM sequences by initiating appropriate rod withdrawal block and rod insert block signals to the reactor manual control system rod block circuitry. The RWM sequences stored in the microprocessor memory are based on control rod worth at acceptable levels as determined by the design basis rod drop accident analyses.

7.6.1.20.3.1 RWM Inputs

Sequence

Up to four sequences are simultaneously stored for sequence control operation.

The operator is permitted to switch between sequences when all rods are in or when above the low power alarm point (LPAP).

The operator is permitted to switch between sequences at any power level, when both sequences conform to the present rod configuration within a single insert or withdraw error, not exceeding two notches.

Sequence selection is accomplished under keylock control with insert and withdraw blocks applied.

Bypass/Operate/Test

A keylock switch is provided for selection of operate or alternately bypass during sequence control operation.

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During reactor shutdown, the test mode provides a single rod permissive function and a shutdown margin test facility.

Control Rod Selected

The input is a binary coded identification of the control rod selected by the operator.

Control Rod Position

The input is a binary coded identification of all rod positions.

Control Rod Drive Selected and Driving

The RWM uses the rod selected and driving input to identify the envelope of rod motion for the selected rod.

Control Rod Bypass

A maximum of eight control rods can be bypassed under keylock control.

Reactor Power Level

Feedwater system signals are used to implement two digital inputs to permit automatic bypass of the RWM function.

The low power set point (LPSP) identifies the power level at which the RWM is automatically bypassed on reactor startup and automatically initiated on reactor shutdown.

The low power alarm point (LPAP) identifies the approach to the LPSP on reactor shutdown.

Select Insert and Select Withdraw

The select insert and select withdraw inputs identify the direction of intended rod motion to permit termination of insert or withdraw motion at the respective insert or withdraw limit.

Insert Bus and Withdraw Bus

The reactor manual control system insert and withdraw bus is monitored to permit timing of rod drive motion.

7.6.1.20.3.2 RWM Outputs

Isolated contact outputs provide RWM Block and Annunciator functions.

RWM insert block and withdraw block are applied for each rod selection to inhibit rod motions which would result in insert or withdraw error.

RWM rod drive block and settle functions are used to terminate continuous rod motion at the respective insert or withdraw limit or if the RWM senses Multiple Rod Motion (See Section 7.6.1.20.3.4).

RWM annunciation draws operator attention to the RWM message log which identifies the reason for the action taken. RWM Annunciation is not systematically applied with insert or withdraw block since these are routinely applied to limit and inhibit rod motion.

7.6.1.20.3.3 RWM Indications

The RWM operator display panel provides indications of operating status including

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- a. Selected Rod
Identification of the coordinate of the selected rod along with
Position of the selected rod
Select error status
Insert block status
Withdraw block status
- b. Insert Error
Identification of control rod coordinate and rod position for up to three insert error rods. Insert error is corrected as the next rod motion.
- c. Withdraw Error
Identification of control rod coordinate and position identification for one withdraw error. Withdraw error is corrected as the next rod motion.
- d. Latched Step
Identification of the current RWM sequence step number.
- e. Selected Sequence
Identification of the selected sequence.
- f. Power Level
Identification of power level is identified as "Below LPSP", "Transition Region", "Above LPAP", or "Unknown". When power level is indicated to be below LPSP and above LPAP, the power level is identified as "Unknown" and the RWM defaults to below LPSP operation.

7.6.1.20.3.4 Additional Functions

In addition to enforcing adherence to established control rod sequences, the RWM performs additional functions. These additional functions, with the exception of the multiple rod motion (MRM) rod drive block and display, are “utilities” which are used to record and display rod position/time data.

- a. Single Rod Scram Timing
Single rod scram timing is selected by the operator to record rod position/time data during single rod scram testing.
- b. Full Core Scram Timing
Full core scram timing is automatically initiated by a reactor scram signal and records the rod position/time data during a reactor scram.
- c. Rod Drive Timing
Rod drive timing is selected by the operator to record rod position/time data for the rod being driven/tested.

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d. Single Rod Scram Data Display

Single rod scram data display is selected by the operator during single rod scram testing to display the actual scram time for the rod under test, the average Technical Specification scram time, and margin of the scram time of the tested rod to the Technical Specification time. The display is available at the RWM operators display and RWM computer.

e. Shutdown Verification Display

The Shutdown Verification display is automatically initiated by a reactor scram signal and immediately displays if all rods are full-in, if all rods are inserted to or beyond the shutdown margin limit, and how many rods are not full-in. The shutdown verification screen is displayed at the RWM operators display.

f. Multiple Rod Motion (MRM) Rod Drive Block and Display

The multiple rod motion (MRM) rod drive block and display are initiated when the RWM senses that a rod (or rods) other than the selected rod is moving. MRM is defined as a movement of an unselected rod (or rods) that has resulted from a failure in the reactor manual control system (RMCS) when a valid rod is selected and being moved by the operator. The purpose of the MRM rod drive block is to terminate and limit rod motion of both the selected and any unselected rods to one notch, if an MRM were to occur. The MRM firmware will actuate the existing RWM rod drive block and settle relays to the RMCS and automatically initiate the MRM display at the RWM operator display if the RWM senses that an unselected rod (or rods) is moving. If the RWM is bypassed (keylock switch on the operator display), the MRM screen is automatically selected, but the MRM rod drive block is inhibited by the RWM bypass relay.

7.6.1.20.4 Environmental Considerations

The RWM is not used for credit in the safety analysis, nor is it required to operate during or after any design basis accidents. The RWM is employed to operate in the normal plant environment for power operation.

7.6.1.20.5 Operational Considerations

The RWM function does not interface with normal reactor operation, and in the event of its failure does not cause new rod patterns. The RWM function may be bypassed and its rod block function disabled only by specific procedural control initiated by the operator, in accordance with the Technical Specifications.

With the RWM inoperable, a second licensed operator or other technically qualified member of the unit technical staff who is present at the reactor control console verifies the control rod movement compliance with the prescribed control rod pattern. The requirements for a rod motion verifier and the specified actions expected of the operator and verifier are proceduralized including:

Procedural guidance for control of Rod Pull Sheets to ensure correct pullsheets are used

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Explicit instructions to the operator and verifier are contained in a Rod Pull Cover Sheet
Each operator and verifier reviews the cover sheet prior to pulling rods.

Explicit instructions are included to the verifier as to how and where to verify proper rod selection and positioning.

7.6.2 Analysis

7.6.2.1 Refueling Interlock System Instrumentation and Control

7.6.2.1.1 Conformance To General Functional Requirements

a. Safety Evaluation

The refueling interlocks, in combination with core nuclear design and refueling procedures, limit the probability of an inadvertent criticality. The nuclear characteristics of the core ensure that the reactor is subcritical even when the highest-worth control rod is fully withdrawn. Also, refueling procedures are written to avoid situations in which inadvertent criticality is possible. The combination of refueling interlocks for control rods and the refueling platform provides redundant methods of preventing inadvertent criticality even after procedural violations. The interlocks on hoists provide yet another method of avoiding inadvertent criticality.

Table 7.6-1 illustrates the effectiveness of the refueling interlocks. This table considers various operational situations involving rod movement, hoist load conditions, refueling platform movement and position, and mode switch manipulation. The initial conditions in situations 4 and 5 appear to contradict the action of refueling interlocks, because the initial conditions indicate that more than one control rod is withdrawn, yet the mode switch is in REFUEL. Such initial conditions are possible if the rods are withdrawn when the mode switch is in STARTUP and then turned to REFUEL. In all cases, correct operation of the refueling interlock prevents either the operation of loaded refueling equipment over the core when any control rod is withdrawn or the withdrawal of any control rod when fuel-loaded refueling equipment is operating over the core. In addition, when the mode switch is in REFUEL, only one rod can be withdrawn; selection of a second rod initiates a rod block.

7.6.2.1.2 Conformance To Specific Regulatory Requirements

No specific regulatory requirements apply to refueling interlocks. The refueling interlocks are designed to be normally energized (fail-safe).

IEEE standards do not apply because the refueling interlocks are not required for any postulated DBA or for safe shutdown. Furthermore, the interlocks are required only for the refueling mode of plant operation.

The requirements of 10 CFR 50, Appendix B, are met in the manner set forth in Chapter 17.

7.6.2.2 Reactor Pressure Vessel Instrumentation and Control

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7.6.2.2.1 Conformance To General Functional Requirements

The RPV instrumentation and systems are designed to augment the existing information from the ESF systems such that the operator can start up, operate at power, shut down, and service the reactor systems in an efficient manner. None of this instrumentation is required to initiate an RPS or ESF system.

7.6.2.2.2 Conformance To Specific Regulatory Requirements

There are no specific regulatory requirements imposed on the RPV instruments and subsystems discussed in Subsection 7.6.1.2 because of the reasons stated in Subsection 7.6.2.2.1 above.

The requirements of 10 CFR 50, Appendix B, are met in the manner set forth in Chapter 17.

7.6.2.3 Process Radiation Monitor Systems

The process radiation monitor systems are described in Section 11.4.

7.6.2.4 Area Radiation Monitor System Instrumentation and Control

See Subsection 12.1.4.

7.6.2.5 Offsite Environs Radiation Monitor Systems

See the Offsite Dose Calculation Manual (ODCM).

7.6.2.6 Rad-Chem Radiation Monitoring Instruments

See Section 12.3.

7.6.2.7 Reactor Water Cleanup System Instrumentation and Control

7.6.2.7.1 General

The RWCU system is not a safety-related system. Therefore, the instrumentation supplied is for the plant equipment protection only.

7.6.2.7.2 Conformance To General Functional Requirements

The RWCU system is protected against overpressurization by relief valves. The ion exchange resin is protected from high temperature by temperature switches upstream of the filter-demineralizer unit. One switch activates an alarm when the water temperature reaches 130°F. A second switch provides a signal at 140°F to close a motor-operated valve in the suction line to the RWCU pumps, which subsequently trips the pumps on low discharge flow.

Three motor-operated isolation valves close automatically on a reactor low water level signal. The outermost isolation valves G33F004 and G33F220 also close automatically when

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the standby liquid control system is activated. The isolation valves provide a pump trip when the valves close.

A high differential pressure across the filter-demineralizer or its discharge strainer automatically closes the unit's outlet valve after sounding an alarm. The holding pump starts whenever there is low flow through a filter-demineralizer. The precoat pump does not start when the level in the precoat tank is low.

Sampling stations are provided to obtain reactor water samples from the entrance and exit of both filter-demineralizers.

The system instrumentation and control for flow, pressure, temperature, and conductivity is recorded and/or indicated on a panel in the main control room. Instrumentation and control for backwashing and precoating the filter-demineralizers is on a local panel in the reactor building. Alarms are sounded in the main control room to alert the operator to abnormal conditions.

The RWCU system is controlled by the operator from the main control room.

A list of the RWCU system annunciators is given in Table 7.6-14.

7.6.2.7.3 Conformance To Specific Regulatory Requirements

Since the RWCU system is not a safety-related system, no specific regulatory requirement is applicable.

7.6.2.8 Leak Detection System Instrumentation and Control

7.6.2.8.1 General

The part of LDS instrumentation that is related to the system isolation circuitry is designed to meet requirements of the ESF system.

7.6.2.8.2 Conformance To General Functional Requirements

There are at least two different methods of detecting abnormal leakage from each system within the nuclear system process barrier and in each area as shown in Table 5.2-11. The instrumentation is designed so that it may be set to provide alarms at established leakage rate limits and isolate the affected system, if necessary. The alarm points are determined analytically based on design data and on measurements of appropriate parameters made during startup and preoperational tests. This satisfies the power generation design basis and safety design basis.

The unidentified leakage rate limit is based, with an adequate margin for contingencies, on the crack size large enough to propagate rapidly. The established limit is sufficiently low so that even if the entire unidentified leakage rate were coming from a single crack in the nuclear system process barrier, corrective action could be taken before the integrity of the barrier is threatened with significant compromise.

The limit on total leakage rate is established so that in the absence of normal ac power and feedwater, and without using the ECCSs, the leakage loss from the nuclear system could be

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replaced. The limit on total leakage also allows a reasonable margin below the discharge capability of either the floor drain or equipment drain sump pumps. Thus, the established total leakage rate limit allows sufficient time for corrective action to be taken before either the nuclear system coolant makeup or the drywell sump removal capabilities are exceeded.

7.6.2.8.3 Conformance To Specific Regulatory Requirements

Compliance With Regulatory Guide 1.22

The portion of the LDS that provides outputs to the system isolation logic is designed so that complete periodic testing of the isolation system actuation function is provided. This is accomplished by tripping the LDS one channel at a time from the leak detection panel in the main control room. An indicator lamp is provided to show that the particular channel is tripped.

Compliance With General Design Criteria 13 and 19-24 of 10 CFR 50

The leak detection sensors and associated electronics are designed to monitor the reactor coolant leakage over all expected ranges required for the safety of the plant. Automatic initiation of the system isolation action, reliability, testability, independence, and separation have been factored into leak detection design as required for isolation systems.

Compliance With IEEE 279-1971

Compliance of the LDS with IEEE 279-1971 is included in the IEEE 279-1971 compliance discussion of the CRVICS (Subsection 7.3.2.3.3.1) for which this system provides logic trip signals.

Compliance With IEEE 323-1971 - Leak detection compliance is shown in Topical Report NEDO-10698. See also Section 3.11.

Compliance With IEEE 338-1971 - Leak detection compliance with IEEE 338-1971 is shown. All active components of the LDS associated with the isolation signal can be tested during plant operation.

Compliance With IEEE 344-1971 - Leak detection compliance with IEEE 344-1971 is shown in Topical Report NEDO-10678. See also Section 3.10.

7.6.2.9 Integrated Plant Computer System (IPCS)

The IPCS, exclusive of the meteorological and emergency response functions, is designed to provide the operator with certain information as defined in the equipment description in Subsection 7.6.1.9. The system augments existing information from other systems such that the operator can start up, operate at power, and shut down in an efficient manner. There are no specific regulatory requirements associated with this portion of the IPCS capabilities.

See Subsection 7.6.1.9.1.2.5.1 for a discussion of the design analysis of the IPCS Safety Parameter Display System (SPDS) function. NRC guidance on safety parameter systems is contained in Supplement 1 to NUREG-0737.

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The IPCS, inclusive of all its functions, is not required to initiate any ESF or safety-related system.

7.6.2.10 (Deleted)

7.6.2.11 Sequence Recorder

The sequence recorder is designed to provide the necessary data and information systems to permit diagnosing the causes of unscheduled reactor shutdowns and determine the proper functioning of safety-related equipment. The requirements of generic letter 83-28 are met for post-trip review. The power source is reliable and non-interruptible. The system meets the requirements to record, recall, and display data and information to permit post-trip review.

7.6.2.12 Primary Containment Monitor Systems

7.6.2.12.1 Primary Containment Radiation and Hydrogen/Oxygen Monitor System

Conformance To Specific Regulatory Requirements

The primary containment radiation monitor subsystem is designed to monitor the primary containment for determination of radiation level during reactor operation or shutdown periods. The rate of flow of drywell atmosphere sample is sufficiently high to provide readings representative of the radiation level in the drywell in less than 5 minutes. Filters are provided in the sample supply line to the primary containment radiation monitor to collect particulates and halogens, on separate filters, for analysis. Monitored radiation level above a predetermined level is alarmed and annunciated in the main control room.

The diversity requirement of Regulatory Guide 1.45 is met by the noble gas activity monitor alarm; activity (cpm) is not required to be correlated with leak rate (gpm) per an exemption granted in NUREG 0798, Section 5.2.5.

The requirements of General Design Criterion (GDC) 30 of 10 CFR 50, Appendix A, are met in that the primary containment radiation monitor system provides means, as required, for monitoring the reactor primary containment atmosphere radioactivity.

In addition, the primary containment is monitored for hydrogen/oxygen with indication and high alarms in the main control room in compliance with Regulatory Guide 1.7 and Regulatory Guide 1.97, Rev 2, Category 3 and 2 requirements, respectively.

The design of the primary containment radiation monitor system incorporates provisions for indicating activity level of noble gases, and collecting particulates and halogens on filter papers for laboratory analysis. Also provided are trip logic provisions for actuating an alarm and an annunciator to inform operations personnel of low-scale conditions that would be indicative of instrument failure.

7.6.2.12.2 Primary Containment Temperature Monitor Subsystem

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7.6.2.12.2.1 Conformance To General Functional Requirements

The primary containment temperature monitor subsystem is designed to fulfill the safety and power generation design bases and industry standards that are stated under Subsection 7.1.2.1.22.

7.6.2.12.2.2 Conformance To Specific Regulatory Requirements

The primary containment temperature monitor subsystem is designed to monitor continuously the temperature of the drywell atmosphere, drywell walls, drywell cap atmosphere, pressure suppression chamber atmosphere, and pressure suppression chamber water pool.

The requirements of GDC 13 of 10 CFR 50, Appendix A, are met in that the primary containment temperature monitor system provides instrumentation to obtain temperature measurements in the designated areas during normal operation, as well as postulated abnormal conditions of a LOCA.

The requirements of 10 CFR 50, Appendix B, are met in the manner set forth in Chapter 17.

The design of the primary containment temperature monitor subsystem includes multipoint recorders in the main control room, on which the temperatures monitored by the subsystem are continuously recorded and displayed.

7.6.2.12.3 Primary Containment Pressure Monitor Subsystem

7.6.2.12.3.1 Conformance To General Functional Requirements

The primary containment pressure monitor subsystem is designed to fulfill the safety and power generation design bases and industry standards stated in Subsection 7.1.2.1.22.

7.6.2.12.3.2 Conformance To Specific Regulatory Requirements

The primary containment pressure monitor system is designed to continuously monitor atmospheric pressure in the drywell and in the pressure suppression chamber.

The requirements of GDC 13 of 10 CFR 50, Appendix A, are met in that the primary containment pressure monitor subsystem has instrumentation that is provided, as required, for measurement and recording of pressure during normal operation, as well as postulated abnormal conditions of a LOCA.

The requirements of 10 CFR 50, Appendix B, are met in the manner set forth in Chapter 17.

The design of the primary containment pressure monitor subsystem includes chart recorders in the main control room, on which the pressure monitored by the subsystem is continuously recorded and displayed.

7.6.2.12.4 Pressure Suppression Pool Water Level Indicator Subsystem

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7.6.2.12.4.1 Conformance To General Functional Requirements

This system is designed to fulfill the safety and power generation design bases and industry standards stated in Subsection 7.1.2.1.22.

7.6.2.12.4.2 Conformance To Specific Regulatory Requirements

The design of the pressure suppression pool water level indicator subsystem provides continuous monitoring of the water level in the pressure suppression chamber.

The requirements of GDC 13 of 10 CFR 50, Appendix A, are met in that the pressure suppression pool water level indicator system constitutes instrumentation that is provided, as required, to monitor the pool water level during normal operation as well as postulated abnormal conditions of a LOCA.

The requirements of 10 CFR 50, Appendix B, are met in the manner set forth in Chapter 17.

The pressure suppression pool water level indicator subsystem design includes chart recorders in the main control room on which the monitored level of the water pool is continuously recorded and displayed.

7.6.2.13 Neutron Monitoring System Instrumentation and Control

7.6.2.13.1 Source Range Monitor System

7.6.2.13.1.1 Conformance To General Functional Requirements

The arrangement of the source range monitors (SRM) in the reactor is shown in Figure 7.6-12. This arrangement and irradiated fuel produce at least three counts per second in the SRM using the sensitivity noted in Subsection 7.6.1.13.3. If the discriminator setting is adjusted to produce the specified sensitivity, the signal-to-noise count ratio is well above the 2:1 design basis for cold startup.

If the multiplication of one section of the core increases to put that section of the reactor on a 20-sec period, the nearest SRM chamber shows an increase in count rate. In general, at least one detector indicates the change in multiplications.

Normal startup procedures ensure that withdrawal of control rods is distributed about the core to prevent excessive multiplication in any one section of the core. Hence, each SRM chamber can respond in some degree during the initial rod withdrawal.

Examination of the sensitivity of the SRM detectors and their operating ranges of 10^6 counts per second indicates that the IRM is on scale before the SRM reaches full scale (Figure 7.6-15). Further overlap is provided by partial retraction of the SRM chambers. Such retraction is possible only if the indicated SRM count rate remains above the rod block trip level (approximately 100 counts per second), or if the IRM has been set to the third or any less sensitive (higher) IRM range.

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7.6.2.13.1.2 Conformance To Specific Regulatory Requirements

There are no specific regulatory requirements of the SRM system.

7.6.2.13.2 Intermediate Range Monitor System

7.6.2.13.2.1 Conformance To General Functional Requirements

Subsection 7.2.1.1 evaluates the arrangement of redundant input signals to the RPS. The NMS trip input to the RPS and the trip channels used in actuating a NMS trip are of equivalent independence and redundancy to other RPS inputs.

The number and locations of the IRM detectors have been analytically and experimentally determined to provide sufficient intermediate range flux level information under the worst permitted bypass or detector failure conditions. To verify this, a range of rod withdrawal accidents has been analyzed. The most severe case assumes that the reactor is barely subcritical. One-fourth of the control rods plus one more rod have been removed in the normal operating sequence (Figure 7.6-37). The error or malfunction is removal of the control rod adjacent to the last rod withdrawn. This rod has been chosen to maximize the distance to the second nearest detector for each trip system. It is assumed that the nearest detector in each RPS trip system is bypassed.

A scram signal is initiated when one IRM detector in each RPS trip system reaches its scram trip level. The neutron flux versus distance resulting from this withdrawal is shown in Figure 7.6-38. Note that the second nearest detector in trip system B is farther away than the second nearest detector in trip system A. The ratio of the neutron flux at this point to the peak flux is 1:4100. This detector reaches its high scram trip setting of 95 percent of full scale at a local flux approximately 3.3×10^8 nv. At that time the peak flux in the core is 1.35×10^{12} nv or 2.7 percent rated average flux. The core average power is 0.07 percent when scram occurs. For this scram point to be valid, the IRM must be on the correct range. To ensure that each IRM is on the correct range, a rod block is initiated any time the IRM is both downscale and not on the most sensitive (lowest) scale. A rod block is initiated if the IRM detectors are not fully inserted in the core unless the reactor mode switch is in the RUN position.

The IRM scram trips and the IRM rod block trips are automatically bypassed when the reactor mode switch is in the RUN position.

The IRM detectors and electronics have been tested under operating conditions and verified to have the operational characteristics described. They provide the level of precision and reliability required by the RPS safety design bases.

The IRM is the primary source of information as the reactor approaches the power range. Its linear steps (approximately a half decade) and the rod blocking features on both high flux level and low flux level require that all the IRM'S are on the correct range as core reactivity is increased by rod withdrawal. The SRM overlaps the IRM. The sensitivity of the IRM is such that the IRM is on scale on the least sensitive (highest) range with approximately 15 percent reactor power.

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7.6.2.13.2.2 Conformance To Specific Regulatory Requirements

Compliance With Regulatory Guide 1.22

The portion of the IRM system that provides outputs to the RPS is designed to provide complete periodic testing of protection system actuation function as desired. This provision is accomplished by initiating an output trip on one IRM channel at any given time, which will result in tripping one of the two RPS trip systems. Details are provided in Topical Report NEDO-10139, Subsection 2.2.8 (Reference 8).

Operator indication of IRM bypass is provided by indicator lamps as described in NEDO-10139, Subsection 2.2.8.13 (Reference 8).

Compliance With General Design Criteria 13 and 22-24 of 10 CFR 50

The IRM detectors and associated electronics are designed to monitor the in-core flux over all expected ranges required for safety of the plant.

Automatic initiation of RPS action, reliability, stability, independence, and separation has been factored into the IRM design as required for protection systems.

Compliance With IEEE 279-1971 - The IRM design is shown to comply with the design requirements of IEEE 279-1971 in Subsection 2.2.8 of Reference 8.

Compliance With IEEE 323-1971 - IRM compliance is shown in Topical Report NEDO-10698. See also FSAR Section 3.11.

Compliance with IEEE 338-1971 - IRM compliance with IEEE 338-1971 is shown in Subsections 2.2.8.9 and 2.2.8.10 of Reference 8.

Compliance With IEEE 344-1971 - IRM compliance is shown in Topical Report NEDO-10678. See also FSAR Section 3.10.

Compliance with 10 CFR 50, Appendix B - The requirements of 10 CFR 50, Appendix B, are met in the manner set forth in Chapter 17.

7.6.2.13.3 Local Power Range Monitor System

7.6.2.13.3.1 Conformance To General Functional Requirements

The LPRM provides detailed information about neutron flux throughout the reactor core. The number of LPRM assemblies and their distribution is determined by extensive calculational and experimental procedures. The division of the LPRM into various groups for ac power supply allows operation with one ac power supply failed or in service without limiting reactor operation.

Individual failed chambers can be bypassed. Neutron flux information for a failed chamber location can be interpolated from nearby chambers. Also, a substitute reading for a failed chamber can be derived from an octant-symmetric chamber, or an actual flux indication can be obtained by inserting a TIP to the failed chamber position. Each output is electrically isolated so that an event (grounding the signal or applying a stray voltage) on the reception end does not destroy the validity of the LPRM signal. Tests and experience attest to the

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ability of the detector to respond proportionately to the local neutron flux changes (Reference 1).

7.6.2.13.3.2 Conformance To Specific Regulatory Requirements

There are no specific regulatory requirements of the LPRM subsystem. Because they form inputs to the APRM system, however, a minimum number of LPRMs must be operable for each APRM as defined in the APRM safety design basis.

7.6.2.13.4 Average Power Range Monitor System

7.6.2.13.4.1 Conformance To General Functional Requirements

Each APRM derives its signal from LPRM information. The assignment, power separation, cabinet separation, and LPRM signal isolation are in accord with the safety design bases of the RPS. There are four APRM channels with the Reactor Protection System trip outputs from each routed to each of four APRM two-out-of-four voter channels. Two voter channels are associated with each Reactor Protection System trip system. This configuration allows one APRM channel to be bypassed plus one failure while still meeting the Reactor Protection System safety design basis.

Above a plant power level defined by Technical Specifications, the APRM power (and simulated thermal power) is adjusted periodically based on heat balance to match true reactor power. This adjustment is made regularly at a rate sufficient to compensate for LPRM burnup and the related change in APRM values. However, coolant flow changes, control rod movements, and failed or bypassed LPRM inputs can also affect the relationship between APRM measured flux and true reactor power. These predictable APRM variations are included in the analysis performed to determine minimum number of LPRM inputs required to be operable in order for the APRM channel to be operable. The analysis is performed considering worst case combinations of failed LPRM inputs, at rated conditions by assuming both continuous withdrawal of the maximum worth control rod and reduction of recirculation flow to 40% of rated flow. The minimum number of LPRM inputs for an APRM is determined such that the average of the remaining operable LPRM inputs still allows the APRM to track power excursions within the acceptance criteria assumed in plant safety analysis. If the number of operable LPRMs is less than the required minimum, the APRM channel is declared inoperable.

The flow-referenced APRM scram setpoint is adequate to prevent fuel damage during an abnormal operational transient, as demonstrated in Chapter 15.

The APRM also includes an OPRM Upscale function to provide compliance with GDCs 10 and 12, thereby providing protection from exceeding the fuel MCPR safety limit due to anticipated thermal hydraulic induced power oscillations. The OPRM utilizes three algorithms for detecting thermal-hydraulic instability related neutron flux oscillations: the period based detection, the amplitude based, and the growth rate based algorithms. All three are implemented in the OPRM Upscale function, but the safety analysis takes credit for the period based algorithm. The remaining algorithms provide defense in depth and additional protection against unanticipated oscillations. The OPRM Upscale function receives input signals from the LPRMs within the reactor core, which are combined into cells for evaluation

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by the OPRM algorithms. The OPRM Upscale function is enabled in the intended region on the plant power/flow map. The plant power level and recirculation drive flow conditions are defined by Technical Specifications.

7.6.2.13.4.2 Conformance To Specific Regulatory Requirements

a. Compliance With Regulatory Guide 1.22

The portion of the APRM subsystem that provides outputs to the RPS is designed to provide complete periodic testing of protection system actuation functions as desired. This provision is accomplished by initiating an output trip of one APRM channel at any given time, which will result in tripping one of the two RPS trip systems. Details are provided in Subsection 2.2.8 of Reference 8.

Compliance With General Design Criteria 10 and 12

The OPRM Upscale Function provides compliance with GDC 10 and GDC 12 by providing a hardware/software system that detects and acts to suppress thermal-hydraulic instabilities, thereby providing protection from exceeding the fuel MCPR safety limit due to thermal hydraulic induced power oscillations.

Compliance With General Design Criteria 13 and 20-24 of 10 CFR 50

The APRM detection and associated electronics are designed to monitor the in-core flux over all expected ranges required for safety of the plant.

Automatic initiation of protection system action, reliability, testability, independence, and separation has been factored into the APRM design as required for protection systems.

The requirements of 10 CFR 50, Appendix B, are met in the manner set forth in Chapter 17.

Compliance With IEEE 279-1971 - The APRM design is shown to comply with the design requirements of IEEE 279-1971 in Subsection 2.2.8 of Reference 8 and Subsection 4.4.1.1 of Reference 10.

Compliance With IEEE 323-1971 - APRM compliance is shown in Topical Reports NEDO-10698 and NEDC 32410P-A. See also FSAR Section 3.11.

Compliance With IEEE 338-1971 - APRM compliance with IEEE 338-1971 is shown in Subsections 2.2.8.9 and 2.2.8.10 of Reference 8.

Compliance with IEEE 344-1971 - APRM compliance is shown in Topical Reports NEDO-10678 and NEDC 32410P-A. See also FSAR Section 3.10.

7.6.2.13.5 Rod Block Monitor System

7.6.2.13.5.1 Conformance To General Functional Requirements

Motion of a control rod causes the LPRMs adjacent to the control rod to respond strongly to the change in power in the region of the rod in motion. Figure 7.6-41 illustrates the calculated responses of the two RBMs to the withdrawal of a selected control rod. The RBM setpoints may be selected such that the rod withdrawal error (RWE) is not the limiting transient. Figure 7.6-42 shows the relationship between MCPR and the RBM setpoint. It

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also shows that for an example operating limit MCPR (OLMCPR) requirement of 1.28, there is a 0.08 margin with 108 percent RBM setpoint (0.03 for 111 percent RBM setpoint). These margins are more than adequate to protect against any RWE events. The RBM setpoints conservatively assume a probability of 15 percent that any given LPRM has failed. The RBM setpoints are also valid for peripheral cells with less than four LPRM strings (the RBM cells near the core periphery may possess fewer than four control rods and have one, two, or three LPRM strings). In some peripheral cases, the responses are actually improved because the missing strings are the weaker signal inputs in a standard RBM cell.

7.6.2.13.5.2 Conformance To Specific Regulatory Requirements

Compliance With General Design Criterion 24 of 10 CFR 50, Appendix A

The RBM provides an interlocking function in the control rod withdrawal portion of the CRD RMCS. This design is separated from the protective functions in the plant to ensure their independence.

The RBM is designed to prevent control rod withdrawal error, given an imposed single failure within the RBM. One of the two RBM channels is sufficient to provide an appropriate control rod withdrawal block.

Compliance with 10 CFR 50, Appendix B

The requirements of 10 CFR 50, Appendix B, are met in the manner set forth in Chapter 17.

7.6.2.13.6 Traversing In-Core Probe Subsystem

7.6.2.13.6.1 Conformance To General Functional Requirements

An adequate number of TIP machines is supplied to ensure that each LPRM assembly can be probed by a TIP and that one LPRM assembly (the central one) can be probed by every TIP to allow intercalibration. Typical TIPs have been tested to prove linearity (Reference 1). The system has been field tested in an operating reactor to ensure reproducibility for repetitive measurements. The mechanical equipment has undergone life testing under simulated operating conditions to ensure that all specifications can be met. The system design allows semiautomatic operation for LPRM calibration and Integrated Plant Computer System (IPCS) TIP processing function use. The TIP machines can be operated manually to allow pointwise flux mapping.

7.6.2.13.6.2 Conformance To Specific Regulatory Requirements

There are no specific regulatory requirements of the TIP subsystem.

7.6.2.14 Plant Cooling Water Systems Instrumentation and Control

Conformance To General Functional Requirements

The instrumentation and control of the RBCCWS and TBCCWS is designed to permit reliable operation and testing for each instrument loop or subsystem. Controls for the essential portion of the RBCCWS are described in Subsection 7.3.4

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The nonessential portions of the RBCCWS and the TBCCWS are designed to shut down upon loss of offsite ac power. These systems can be restarted manually from the main control room. These systems are designed for manual startup, shutdown, and testing. Automatic controls are provided for maintaining condensate level in the makeup tank, gas pressure in the makeup tank, heat exchanger outlet temperature, and differential pressure across the supply and return headers. Indications, alarms, and/or warning lights for these variables are provided in the main control room. Deviations from normal conditions are thereby brought to the attention of the main control room operator who subsequently can take the appropriate action.

7.6.2.15 Fuel Pool Cleanup System Instrumentation and Control

7.6.2.15.1 General

The FPCCS is not a safety-related system. Therefore, the instrumentation supplied is for the plant equipment protection and for operator information about the system.

7.6.2.15.2 Conformance To General Functional Requirements

The FPCCS is monitored for conductivity, temperature, pool level, flow rate, and leakage.

The conductivity measurement provides the operator with information required to ensure that impurities in the water are limited to acceptable levels.

The low flow (pump discharge pressure) and temperature monitoring provide the operator with information required to ensure that the desired temperature is not exceeded and that filtering is maintained. Pool level and leakage monitoring provide the operator with information assuring the maintenance of adequate shielding and cooling.

Interface

The FPCCS is an independent system during normal operations. Evaporative losses in the system are replaced by the condensate storage system. If the heat load should become excessive, the shutdown cooling portion of the RHR system is operated in parallel with the FPCCS to remove the excess heat load.

7.6.2.15.3 Conformance To Specific Regulatory Requirements

System analysis shows that none of the regulatory requirements are applicable to the FPCCS.

7.6.2.17 Control Air System

Conformance To General Functional Requirements

The instrumentation and control of the control air system is designed to permit reliable operation and testing of each divisional loop of the control air system. The control air system is designed to fulfill the safety and power generation design bases stated in Subsection 7.1.2.1.31.

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7.6.2.18 Alternate Rod Insertion

Conformance To General Functional Requirements

The sensors, transmitters, trip units, associated logic, and ARI valves are Class 1E, redundant to and diverse from the reactor protection system, are seismically and environmentally qualified to meet IEEE 323-1974 and IEEE 344-1975, and are supplied with Class 1E dc power.

The ARI equipment is physically separated into two redundant divisions. Either division will be automatically energized to actuate and scram the reactor upon receipt of high reactor pressure or vessel low-water-level 2 signals. The ARI logic may also be initiated manually from the main control room. (See Subsection 7.6.1.18.2 for further details).

7.6.2.19 Safety/Relief Valves Analysis

7.6.2.19.1 Conformance To General Functional Requirements

The SRVs furnished meet requirements of the ASME Boiler and Pressure Vessel Code Section III, Article 9. The valves are operable in two modes: self-actuated or power-actuated solenoid pressure relieving mode. The automatic mode is independent of the power-actuated mode. Failure of the power-actuated mode does not affect the self-actuated mode.

7.6.2.19.2 Conformance To Specific Regulatory Requirements

Compliance With Regulatory Guide 1.22

The logic channels up to the SRV solenoid operators are designed to enhance periodic testing.

Compliance With IEEE 279-1971

A demonstration of the single-failure withstand capability of the generic low-low set design was presented in the BWR Owners Group letter to the NRC, D. B. Vassallo, dated November 19, 1982, titled "Low-Low Set Logic/Lowered MSIV for Mark I Plants."

Compliance With IEEE 323-1974

System components are environmentally qualified as described in Section 3.11.

Compliance With IEEE 344-1975

System components are seismically qualified as described in Section 3.10.

7.6.2.20 Rod Worth Minimizer System

7.6.2.20.1 Conformance to General Functional Requirements

The RWM protects against the existence of a rod worth which could result in the damage to the reactor coolant pressure boundary in the unlikely event of a control rod drop accident.

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7.6.2.20.2 Conformance to Specific Regulatory Requirements

There are no specific regulatory requirements for the RWM. The Fermi 2 RWM has been designed to enforce operator adherence to the predetermined sequence of control rod motions during operation at low power levels.

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7.6 ALL OTER INSTRUMENTATION SYSTEMS REQUIRED FOR SAFETY

REFERENCES

1. W. R. Morgan, In-Core Neutron Monitoring System for General Electric Boiling Water Reactors, General Electric Co., APED-5706, Nov. 1968, revised April 1969.
2. General Electric Co., "Licensing Summary, The Nuclear Measurement Analysis and Control Rod Worth Minimizer (NUMACRWM) Enhanced RPCS Application," NEDO-31146, June 1986.
3. General Electric Co., "Compliance of Protection Systems to Industry Criteria: General Electric BWR Nuclear Steam Supply System," NEDO-10139, June 1970.
4. GE Nuclear Energy, Maximum Extended Operating Domain Analysis for Detroit Edison Company Enrico Fermi Energy Center Unit 2, NEDC-31843P, July 1990
5. GE Nuclear Energy, Licensing Topical Report, "Nuclear Measurement Analysis and Control Power Range Neutron Monitor (NUMAC-PRNM) Retrofit Plus Option III Stability Trip Function," NEDC-32410P-A, Volumes 1 and 2, October 1995, Including Supplement 1, November 1997.

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TABLE 7.6-1 REFUELING INTERLOCK EFFECTIVENESS

<u>Situation</u>	<u>Refueling Platform Position</u>	<u>Refueling TMH^a</u>	<u>Platform FMH^b</u>	<u>Hoists FG^c</u>	<u>Service Platform Hoist</u>	<u>Control Rods</u>	<u>Mode Switch</u>	<u>Attempt</u>	<u>Result</u>	
1.	Not near core	UL ^d	UL	UL	UL	All rods in	Refuel	Move refueling platform over core	No restrictions	
2.	Not near core	UL	UL	UL	UL	All rods in	Refuel	Withdraw rods	Cannot withdraw more than one rod	
3.	Not near core	UL	UL	UL	UL	One rod withdrawn	Refuel	Move refueling platform over core	No restrictions	
4.	Not near core	Any hoist loaded or FG not fully up			UL	One or more rods withdrawn	Refuel	Move refueling platform over core	Platform stopped before over core	
5.	Not near core	UL	UL	UL	UL	More than one rod withdrawn	Refuel	Move refueling platform over core	Platform stopped before over core	
6.	Over core	UL	UL	UL	UL	All rods in	Refuel	Withdraw rods	Cannot withdraw more than one rod	
7.	Over core	Any hoist loaded or FG not fully up					All rods in	Refuel	Withdraw rods	Rod block
8.	Not near core	UL	UL	UL	L ^e	All rods in	Refuel	Withdraw rods	Rod block	
9.	Not near core	UL	UL	UL	L	All rods in	Refuel	Operate service platform hoist	No restrictions	
10.	Not near core	UL	UL	UL	L	One rod withdrawn	Refuel	Operate service platform hoist	Hoist operation prevented	
11.	Not near core	UL	UL	UL	UL	All rods in	Startup	Move refueling platform over core	Platform stopped before over core	
12.	Not near core	UL	UL	UL	L	All rods in	Startup	Operate service platform hoist	No restrictions	
13.	Not near core	UL	UL	UL	L	One rod withdrawn	Startup	Operate service platform hoist	Hoist operation prevented	
14.	Not near core	UL	UL	UL	L	All rods in	Startup	Withdraw rods	Rod block	

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TABLE 7.6-1 REFUELING INTERLOCK EFFECTIVENESS

<u>Situation</u>	<u>Refueling Platform Position</u>	<u>Refueling TMH^a</u>	<u>Platform FMH^b</u>	<u>Hoists FG^c</u>	<u>Service Platform Hoist</u>	<u>Control Rods</u>	<u>Mode Switch</u>	<u>Attempt</u>	<u>Result</u>
15.	Not near core	UL	UL	UL	UL	All rods in	Startup	Withdraw rods	No restrictions
16.	Over core	UL	UL	UL	UL	All rods in	Startup	Withdraw rods	Rod block
17.	Any	UL	Any condition	UL	Any condition	Any condition reactor not at power	Startup	Turn mode switch to run	Scram

^a THM – trolley mounted hoist.
^b FMH – frame mounted hoist.
^c FG – fuel grapple.
^d UL – unloaded
^e L – fuel loaded

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TABLE 7.6-2 GENERAL INSTRUMENTATION INFORMATION

	<u>Reactor Pressure Vessel Temperature</u>	<u>Reactor Pressure Vessel Level Wide Range During Shutdown</u>	<u>Reactor Core Hydraulics Flow</u>	<u>Reactor Core Hydraulics Differential Pressure</u>
Design classes quality/seismic category ^b	III/NA ^a	III/NA	III/NA	III/NA
Power supply	120-V ac inst. bus	120-V ac inst. bus	120-V ac inst. bus	120-V ac inst. bus
No. of channels	12	1	20	1
Alarm setpoint(s) ^c	NA	NA	NA	NA
Control logic	NA	NA	NA	NA
Instrument range	0 to 600°	160 to 560 in. H ₂ O	0 to 80 x 10 ⁶ pph	0-30 psid
Instrument accuracy ^c	±6 °F	±0.2 percent	±2 percent	±2 percent

^a NA = Not Applicable

^b The instrument seismic category and QA level information provided in the UFSAR tables may have been upgraded to meet the Pressure Boundary Integrity (PBI) or other requirements. The instrument seismic category and QA level information is available in the Fermi 2 Central Component Database.

^c The instrument accuracy information provided in the UFSAR tables is a bounding value.

	<u>Reactor Pressure Vessel Pressure</u>	<u>Reactor Pressure Vessel Steam Temperature</u>	<u>Reactor Pressure Vessel Feedwater Temperature</u>	<u>RBCCW System Radiation Monitoring Subsystem</u>
Design classes quality/seismic category ^c	I/I	III/NA ^a	III/NA	III/NA
Power supply	120-V ac invert. bus	120-V ac inst. bus	120-V ac inst. bus	24-V dc and 120-V ac inst. bus
No. of channels	2	2	6; 4 computer input 1-2 flow correctors	1
Alarm setpoint(s) ^b	NA	NA	NA	(b)
Control logic	NA	NA	NA for computer input 1/1 temp. correction mass flow meter	NA
Instrument range	0 to 1500 psig	400 to 550 °F	300 to 450 °F	10 ⁻¹ to 10 ⁶ cps gamma
Instrument accuracy ^d sensitivity ^e	±30 psig	±0.3 °F	±0.35 °F	1 x 10 ⁻⁴ µCi/ml estimated

^a NA = Not Applicable.

^b Variable to be set periodically in the field.

^c The instrument seismic category and the QA level information provided in the UFSAR tables may have been upgraded to meet the Pressure Boundary Integrity (PBI) or other requirements. The instrument seismic category and QA level information is available in the Fermi 2 Central Component Database.

^d The instrument accuracy information provided in the UFSAR tables is a bounding value.

^e Instrument sensitivity is provided to be consistent with other radiation monitors.

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TABLE 7.6-2 GENERAL INSTRUMENTATION INFORMATION

	<u>EECW Radiation Monitoring Subsystem</u>	<u>RHR System Water Radiation Monitoring Subsystem</u>	<u>General Service Water Radiation Monitoring Subsystem</u>	<u>Radwaste Effluent Radiation Monitoring Subsystem</u>
Design classes quality/seismic category ^d	III/NA ^a	III/NA	III/NA	III/NA
Power supply	120-V ac inst. buses	120-V ac inst. buses	24-V dc and 120-V ac inst. bus	24-V dc and 120-V ac inst. bus
No. of channels	2	2	1	1
Alarm setpoint(s)	(^b)	(^b)	(^b)	(^c)
Control logic	NA	NA	NA	1/1
Instrument range	10 ¹ to 10 ⁷ cpm gamma	10 ¹ to 10 ⁷ cpm gamma	10 ¹ to 10 ⁶ cps gamma	10 ¹ to 10 ⁶ cps gamma
Instrument sensitivity	5 cpm Cs-137	8 cpm Cs-137	5 x 10 ⁻⁹ μCi/cm ³ estimated	1 x 10 ⁻⁴ μCi/ml estimated

^a NA = Not Applicable.

^b Variable; to be set periodically in the field. REV 10 11/00

^c Alarm setpoints to be field determined such that discharge concentration in decant line is less than 10 CFR 20 Table II, Column 2 limits.

^d The instrument seismic category and QA level information provided in the UFSAR tables may have been upgraded to meet the Pressure Boundary Integrity (PBI) or other requirements. The instrument seismic category and QA level information is available in the Fermi 2 Central Component Database.

	<u>Circulating Water Decant Line, Radiation Monitoring Subsystem</u>	<u>Main Steam Line Radiation Monitoring Subsystem</u>	<u>Off-Gas Radiation Monitoring Subsystem</u>	<u>Reactor Building Exhaust Plenum Radiation Monitoring Subsystem</u>
Design classes quality/seismic category ^c	III/NA ^a	I/I	III/NA	III/NA
Power supply	120-V ac inst. buses	120-V ac RPS buses 120-V ac inst. buses	RPS buses A & B 24-V dc bus A	120-V ac inst. bus
No. of channels	1	4	3 (2 log, 1 linear)	1
Alarm setpoint(s)	(^b)	^b 3 x background	(^b)	(^b)
Control logic	NA	½ twice	NA	NA
Instrument range	10 ⁻¹ to 10 ⁷ cpm gamma	10 ⁰ to 10 ⁶ mR/h	10 ⁰ to 10 ⁶ mR/h	See Table 11.4-1
Instrument sensitivity	8 cpm Cs-137	3.7 x 10 ⁻¹⁰ amp/R/h	3 x 10 ⁻¹⁰ amp/R/hr	80 cpm/mR/hr

^a NA = Not Applicable.

^b Variable; to be set periodically in the field.

^c The instrument seismic category and QA level information provided in the UFSAR tables may have been upgraded to meet the Pressure Boundary Integrity (PBI) or other requirements. The instrument seismic category and QA level information is available in the Fermi 2 Central Component Database.

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TABLE 7.6-2 GENERAL INSTRUMENTATION INFORMATION

	<u>Offgas Vent Pipe Radiation Monitoring Subsystem (Installed Spare)</u>	<u>Fuel Pool Ventilation Exhaust Radiation Monitoring Subsystem</u>	<u>Turbine Building Ventilation Exhaust Radiation Monitoring Subsystem</u>
Design classes quality/seismic category ^d	III/NA ^a	I/I	III/NA
Power supply	24-V dc bus A&B RPS buses A&B 120 Vac inst. bus	24-V dc bus A&B RPS buses A&B 120 V ac inst. bus	120-V ac inst. buses
No. of channels	2	4	1
Alarm setpoint(s)	(^b)	(^c)	(^b)
Control logic	1/6	1/4	1/1
Instrument range	10 ⁻¹ to 10 ⁶ cps	10 ⁻² to 10 ² mR/h (G-M)	See Table 11.4-2
Instrument sensitivity	Not specified	0.01 mR/hr	80 cpm/mR/hr

^a NA = Not Applicable.

^b Variable; to be set periodically in the field.

^c Refer to Technical Specifications

^d The instrument seismic category and QA level information provided in the UFSAR tables may have been upgraded to meet the Pressure Boundary Integrity (PBI) or other requirements. The instrument seismic category and QA level information is available in the Fermi 2 Central Component Database.

	<u>Radwaste Building Ventilation Exhaust Radiation Monitoring Subsystem</u>	<u>Reactor Building Ventilation Exhaust Radiation Monitoring Subsystem</u>	<u>Control Center Makeup Air Manifold Radiation Monitoring Subsystem</u>	<u>Standby Gas Treatment System, Vent Exhaust Radiation Monitoring Subsystem</u>
Design classes quality/seismic category ^c	III/NA ^a	III/NA	III/NA	III/NA
Power supply	120-V ac inst. buses	120-V ac inst. buses	120-V ac inst. buses	120-V ac inst. buses
No. of channels	1	2	2	2, 1 per vent
Alarm setpoint(s)	(^b)	(^b)	(^b)	(^b)
Control logic	1/1	1/2	1/2	NA
Instrument range	See Table 11.4-2	10 ¹ to 10 ⁷ cpm (Beta)	10 ¹ to 10 ⁷ cpm (Beta)	See Table 11.4-2
Instrument sensitivity	80 cpm/mR/hr	8 cpm Xe-133	8 cpm Xe-133	80 cpm/mR/hr

^a NA = Not Applicable.

^b Variable; to be set periodically in the field.

^c The instrument seismic category and QA level information provided in the UFSAR tables may have been upgraded to meet the Pressure Boundary Integrity (PBI) or other requirements. The instrument seismic category and QA level information is available in the Fermi 2 Central Component Database.

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TABLE 7.6-2 GENERAL INSTRUMENTATION INFORMATION

	<u>Two-Minute Holdup Pipe Radiation Monitoring Subsystem</u>	<u>Area Radiation Monitoring System</u>	<u>Reactor Water Cleanup Non-Regenerative Heat Exchanger Downstream Temperature</u>	<u>Drywell Leak Detection System Pressure</u>
Design classes quality/seismic category ^d	III/NA ^a	III/NA	III/NA	I/I
Power supply	120-V ac inst. buses	120-V inst. local power bus	120-V ac RPS bus	120-V ac invert. and/or inst. buses
No. of channels	2	48	1	8 4 drywell 4 suppr. Pool
Alarm setpoint(s) ^c	(^b)	Varies with location	130°	NA
Control logic	1/2	NA	1/1	NA
Instrument range	10 ¹ to 10 ⁷ cpm gamma	Varies 10 ⁻² to 10 ² up to 10 ² to 10 ⁶ mR/h	75° to 205°	0 to 250 psig -5 to +5 paig 0 to 80 psig -5 to +15 psig
Instrument accuracy ^e	10 cpm (sensitivity)	±20 percent	±3 °F	±0.25 percent span

^a NA = Not Applicable.

^b Variable; to be set periodically in the field.

^c Nominal value or refer to technical specification for setpoint information (as applicable).

^d The instrument seismic category and QA level information provided in the UFSAR tables may have been upgraded to meet the Pressure Boundary Integrity (PBI) or other requirements. The instrument seismic category and QA level information is available in the Fermi 2 Central Component Database.

^e The instrument accuracy information provided in the UFSAR tables is a bounding value.

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TABLE 7.6-2 GENERAL INSTRUMENTATION INFORMATION

	Control Center Emergency Air South Inlet Radiation Monitor	Control Center Emergency Air North Inlet Radiation Monitor
Design classes quality/seismic category ^c	I/I	I/I
Power supply	120-V ac inst buses	120-V ac inst buses
No. of channels	2	2
Alarm setpoint(s)	(b)	(b)
Control logic	1/2	1/2
Instrument range	10 ⁻¹ to 10 ⁷ cpm (Beta)	10 ⁻¹ to 10 ⁷ cpm (Beta)
Instrument sensitivity	8 cpm/pci/cm ³	8 cpm/pci/cm ³

^a NA = Not Applicable.

^b Variable; to be set periodically in the field.

^c The instrument seismic category and QA level information provided in the UFSAR tables may have been upgraded to meet the Pressure Boundary Integrity (PBI) or other requirements. The instrument seismic category and QA level information is available in the Fermi 2 Central Component Database.

	Drywell Leak Detection System Area Temperature	Drywell Leak Detection System Closed Cooling Water Differential Temperature from Atmospheric Coolers	Drywell sump Level Monitors Leak Detection System Sump Pumpout Rate	Drywell Sump Level Monitors Leak Detection System Sump Fill Rate
Design classes Quality/seismic category ^d	III/I ^a	III/NA ^b	III/NA	III/NA
Power supply	120-V ac inst. bus	120-V ac inst. bus	120-V ac inst. bus	120-V ac inst. bus
No. of channels	29	2	2	2
Alarm setpoint(s) ^c	4 to 115, 135, 145, 180 °F	35 °F Diff	5.1 min. 6.8 min	80.4 min. (Floor drain) 20.1 min. (Equip. drain)
Control logic	NA	NA	NA	NA
Instrument range	0 to 360 °F	95 to 150 °F	NA	NA
Instrument accuracy ^e	±1.6 °F	±0.1 percent	±9 sec ±45 sec.	±45 sec. (Floor drain) ±45 sec. (Equip. drain)

^a Seismic installation.

^b NA = Not Applicable.

^c Nominal value or refer to technical specification for setpoint information (as applicable).

^d The instrument seismic category and QA level information provided in the UFSAR tables may have been upgraded to meet the Pressure Boundary Integrity (PBI) or other requirements. The instrument seismic category and QA level information is available in the Fermi 2 Central Component Database.

^e The instrument accuracy information provided in the UFSAR tables is a bounding value.

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TABLE 7.6-2 GENERAL INSTRUMENTATION INFORMATION

	<u>Reactor Pressure Vessel Head Seal Leak Detection (Inter-seal Pressure)</u>	<u>Recirculation Pump Leak Detection System Seal Cavity Pressure</u>	<u>Recirculation Pump Detection System Seal Leakage Rate</u>	<u>Safety Relief Valve Leak Detection System Discharge Pipe Temperature</u>
Design classes quality/seismic category ^c	III/NA ^a	III/NA	III/NA	III/NA
Power supply	120-V ac inst. bus	120-V ac inst. bus	120-V ac inst. bus	120-V ac inst. bus
No. of channels	1	1 per cavity to 2 cavity per pump	1 per pump	15
Alarm setpoint(s) ^b	600 psig	NA	0.1 gpm	220 °F
Control logic	NA	NA	NA	NA
Instrument range	0 to 1500 psig	0 to 1250 psig	0 to 0.55 gpm(A) 0 to 1.25 gpm(B)	0 to 600°
Instrument accuracy ^d	±30 psig	±2 percent	±2percent	±6 °F

^a NA = Not Applicable. ^b Nominal value or refer to technical specification for setpoint information (as applicable).

^c The instrument seismic category and QA level information provided in the UFSAR tables may have been upgraded to meet the Pressure Boundary Integrity (PBI) or other requirements. The instrument seismic category and QA level information is available in the Fermi 2 Central Component Database.

^d The instrument accuracy information provided in the UFSAR tables is a bounding value.

	<u>Main Steam Line Leak Detection Area Temperature</u>	<u>Main Steam Line Leak Detection System Flow</u>	<u>Main Steam Line Tunnel Detection System Differential Temperature</u>	<u>Main Steam Line Tunnel Leak Detection System Temperature</u>
Design classes quality/seismic category ^d	III/NA ^a	I/I	III/NA	I/I
Power supply	120-V ac invt. bus	120-V ac RPS buses	120-V ac invt. bus	120-V ac RPS bus
No. of channels	2	16	2	16
Alarm setpoint(s) ^c	160 °F increasing	NA ^b	70 °F increasing	NA ^{a,b}
Control logic	NA	1/4 isolates monitored steam line	NA	1/4 isolates monitored steam line
Instrument range	50° to 350°	0 to 150 psid	0 to 150° ΔT (50 to 350 °F)	50 to 350 °
Instrument accuracy ^e	±6 °F	psi	±3 °F	±2 °F

^a NA = Not Applicable.

^b See Technical Specifications for trip setpoint.

^c Nominal value or refer to technical specification for setpoint information (as applicable).

^d The instrument seismic category and QA level information provided in the UFSAR tables may have been upgraded to meet the Pressure Boundary Integrity (PBI) or other requirements. The instrument seismic category and QA level information is available in the Fermi 2 Central Component Database.

^e The instrument accuracy information provided in the UFSAR tables is a bounding value.

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TABLE 7.6-2 GENERAL INSTRUMENTATION INFORMATION

	<u>RWCU System Leak Detection System Area Temperature Monitor</u>	<u>RWCU System Leak Detection System High Flow Rate</u>	<u>RWCU Differential Temperature Trip</u>	<u>RHR Leak Detection System Area Temperature</u>
Design classes quality/seismic category ^d	I/I	I/I	I/I	III/NA ^a
Power supply	120-V ac invt. bus	120-V ac inst. bus	120-V ac invt. bus	120-V ac invt. bus
No. of channels	12	2	4	2
Alarm setpoint(s) ^c	175 °F ^b	NA ^{a,b}	NA ^{a,b}	148 °F
Control logic	1/5 per valve	1 per valve	1/5 per valve	NA
Instrument range	50 to 350 °F	0 to 400 gpm	0 to 150° ΔT	50 to 350°
Instrument accuracy ^e	±6 °F	±2.5 percent	±1 percent span	±6 percent

^a NA = Not Applicable.

^b See Technical Specifications for trip setpoint.

^c Nominal value or refer to technical specification or Technical Requirements Manual for setpoint information (as applicable).

^d The instrument seismic category and QA level information provided in the UFSAR tables may have been upgraded to meet the Pressure Boundary Integrity (PBI) or other requirements. The instrument seismic category and QA level information is available in the Fermi 2 Central Component Database.

^e The instrument accuracy information provided in the UFSAR tables is a bounding value.

	<u>RHR Leak Detection System Area Differential Temperature</u>	<u>RCIC Leak Detection System Steam Line Low Pressure</u>	<u>RCIC Leak Detection System Area Temperature</u>	<u>RCIC Leak Detection System Vent Differential Temperature</u>
Design classes quality/seismic category ^d	III/NA ^a	I/I	I/I	I/I
Power supply	120-V ac invt. bus	120-V ac invt. bus	120-V ac invt. bus	120-V ac invt. bus
No. of channels	2	4	3	2
Alarm setpoint(s) ^c	50 °F ΔT	NA ^b	NA ^b	50 °F ΔT
Control logic	NA	2/2 (Redundant)	1/2	NA
Instrument range	0 to 150 ° ΔT	0 to 200 psig	50 to 350°	0 to 150 ° ΔT
Instrument accuracy ^e	±3 °F	±0.25 percent	±6 °F	±3 °F

^a NA = Not Applicable.

^b See Technical Specifications for trip setpoint.

^c Nominal value or refer to technical specification for setpoint information (as applicable).

^d The instrument seismic category and QA level information provided in the UFSAR tables may have been upgraded to meet the Pressure Boundary Integrity (PBI) or other requirements. The instrument seismic category and QA level information is available in the Fermi 2 Central Component Database.

^e The instrument accuracy information provided in the UFSAR tables is a bounding value.

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TABLE 7.6-2 GENERAL INSTRUMENTATION INFORMATION

	<u>RCIC Leak Detection Steam Flow Rate (°P)</u>	<u>HPCI Leak Detection System Steam Line Low Pressure</u>	<u>HPCI Leak Detection System Area Temperature</u>	<u>HPCI Leak Detection System Area Differential Temperature</u>
Design classes quality/seismic category ^d	I/I	I/I	I/I	I/I
Power supply	120-V ac invt. bus	120-V ac inst. bus 120-V ac invt. bus	120-V ac invt. bus	120-V ac invt. bus
No. of channels	2	4	3	2
Alarm setpoint(s)	NA ^{a,b}	NA ^{a,b}	NA ^{a,b}	50 °F ΔT ^c
Control logic	1/2	2/2 redundant	1/2	NA
Instrument range	±300 inch H ₂ O	0 to 200 psig	50° to 350°	0 to 150° ΔT
Instrument accuracy ^e	±0.25 percent	±0.25 percent	±6 °F	±3 °F

^a NA = Not Applicable.

^b See Technical Specifications for trip setpoint.

^c Nominal value or refer to technical specification for setpoint information (as applicable).

^d The instrument seismic category and QA level information provided in the UFSAR tables may have been upgraded to meet the Pressure Boundary Integrity (PBI) or other requirements. The instrument seismic category and QA level information is available in the Fermi 2 Central Component Database.

^e The instrument accuracy information provided in the UFSAR tables is a bounding value.

	<u>HPCI Leak Detection System Steam Flow (Differential Pressure)</u>	<u>Suppression Pool Leak Detection System Area Temperature</u>	<u>Suppression Pool Leak Detection System Area Differential Temperature</u>
Design classes quality/seismic category ^d	I/I ^a	I/I	I/I
Power supply	120-V invt. bus	120-V ac invt. bus	120-V ac inst. bus 120-V ac invt. bus
No. of channels	2	4	4
Alarm setpoint(s) ^c	NA ^b	90 °F > ambient	50 °F °T
Control logic	1/2	NA	NA
Instrument range	±500 in. H ₂ O	50° to 350 °F	0 to 150° °T
Instrument accuracy ^e	±0.25 percent	±6 °F	±3 °F

^a NA = Not Applicable.

^b See Technical Specifications for trip setpoint.

^c Nominal value or refer to technical specification for setpoint information (as applicable).

^d The instrument seismic category and QA level information provided in the UFSAR tables may have been upgraded to meet the Pressure Boundary Integrity (PBI) or other requirements. The instrument seismic category and QA level information is available in the Fermi 2 Central Component Database.

^e The instrument accuracy information provided in the UFSAR tables is a bounding value.

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TABLE 7.6-2 GENERAL INSTRUMENTATION INFORMATION

	<u>ECCS Suction Lines Leak Detection System Sump Level Fill Rate</u>	<u>Sequence of Events Recorder, Nuclear Steam Supply / Balance-of- Plant</u>	<u>Primary Containment Radiation Monitor</u>
Design classes quality/seismic category ^b	III/II/I	III/NA ^a	III/NA
Power supply	130-V dc inst. bus	130-V dc BOP battery inverter supply	120-V ac inst. bus
No. of channels	1	2560 inputs	1 each for gas and particulates (particulates is installed spare)
Alarm setpoint(s)	Field set during startup	NA	
Control logic	Sump fill rate	NA	NA
Instrument range	Timer (0 to 30 min)	NA	
Instrument accuracy ^c	±2 percent	Records inputs with a 1 msec resolution	

^a NA = Not Applicable.

^b The instrument seismic category and QA level information provided in the UFSAR tables may have been upgraded to meet the Pressure Boundary Integrity (PBI) or other requirements. The instrument seismic category and QA level information is available in the Fermi 2 Central Component Database.

^c The instrument accuracy information provided in the UFSAR tables is a bounding value.

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TABLE 7.6-2 GENERAL INSTRUMENTATION INFORMATION

	<u>Primary Containment Hydrogen Analyzer^f</u>	<u>Primary Containment Oxygen Analyzer^g</u>	<u>Suppression Pool Water Level</u>	<u>Neutron Monitor System, Source Range Monitor</u>	<u>Neutron Monitor System, Intermediate Range Monitors</u>
Design classes quality/seismic category ^c	I/I	I/I	I/I	III/NA ^a	I/I
Power supply	120-V ac inst. bus	120-V ac inst. bus	120-V ac invert. Bus 120-V ac invert	±24-V dc buses	±24-V dc buses
No. of channels	2	2	2	4	8
Alarm setpoint(s) ^b	High H ₂ 1.0 percent 3.5 percent	High O ₂ 3.5 percent 4.5 percent	NA	3 c/s down 10 ⁵ c/s up	
Control logic	NA	NA	NA	1/4 for rod block	1/8 trips RPS; 1 channel isolatable
Instrument range ^e	0 to 30 percent H ₂	0 to 10 percent O ₂ 0 to 30 percent O ₂	+56 to -144 in. referenced to normal H ₂ O level	1x10 ³ to 1x10 ⁹ nv	10 ⁸ to 1.5x10 ¹³ nv
Instrument accuracy ^d	±2.0 percent full scale	±3.0 percent full scale	±0.025 percent	±10 percent linear 1.2 x 10 ⁻³ cps/nv (nominal sensitivity)	±15 percent

^a NA = Not Applicable.

^b Nominal value or refer to technical specification for setpoint information (as applicable).

^c The instrument seismic category and QA level information provided in the UFSAR tables may have been upgraded to meet the Pressure Boundary Integrity (PBI) or other requirements. The instrument seismic category and QA level information is available in the Fermi 2 Central Component Database.

^d The instrument accuracy information provided in the UFSAR tables is a bounding value.

^e The oxygen analyzer instrument range of 0 to 10% oxygen is provided to meet Regulatory Guide 1.97 Rev 2 requirements and 0 to 30% oxygen is provided for information only.

^f The hydrogen analyzer is required to meet Regulatory Guide 1.97, Rev 2 Category 3 requirements.

^g The oxygen analyzer is required to meet Regulatory Guide 1.97, Rev 2 Category 2 requirements.

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TABLE 7.6-2 GENERAL INSTRUMENTATION INFORMATION

	<u>Neutron Monitor System Local Power Range Monitor</u>	<u>Neutron Monitor System Average Power Range Monitor</u>	<u>Neutron Monitor System Transversing In-Core Probe</u>	<u>Neutron Monitor System Rod Block Monitor</u>
Design classes quality/seismic category ^b	I/I	I/I	III/NA ^a	1M/II/I
Power supply	RPS buses	120-V ac RPS buses	120-V ac local power	120-V ac RPS buses
No. of channels	172	4	5	2
Setpoint(s)	See Table 7.6-7	See Table 7.6-9	NA	See Table 7.6-10
Control logic	Loss of power causes APRM to trip RPS	See Table 7.6-9	NA ^a	1/2
Instrument range	to 10 ¹⁴ nv	0 to 125 percent full power	2.8 x 10 ¹² to 2.8 x 10 ¹⁴ nv	0 to 125 percent power/flow
Instrument accuracy ^c	±1 percent full scale	±1 percent full scale	Position ±1 in flux ±1.0 percent full scale	±1.5 percent

^a NA = Not Applicable.

^b The instrument seismic category and QA level information provided in the UFSAR tables may have been upgraded to meet the Pressure Boundary Integrity (PBI) or other requirements. The instrument seismic category and QA level information is available in the Fermi 2 Central Component Database.

^c The instrument accuracy information provided in the UFSAR tables is a bounding value.

	<u>RBCCW Makeup Tank Level Trip</u>	<u>RBCCW Low Suction Pressure Trip</u>	<u>TBCCW Makeup Tank Condensate Level Trip</u>	<u>TBCCW Low Suction Pressure Trip</u>
Design classes quality/seismic category ^c	III/NA ^a	III/NA	III/NA ^a	III/NA
Power supply	120-V ac from power to pump	120-V ac from power to pumps	120-V ac from power to pumps	120-V ac from power to pumps
No. of channels	4	4	4	4
Alarm setpoint(s) ^b	6 in. decreasing	< 6 psig	6 in decreasing	< 7 psig
Control logic	1/2 twice	1/2 twice	1/2 twice	1/2 twice
Instrument range	0 to 48 in. WCD	30 HG to 20 psig	0 to 48 in. WCD	0 to 20 psig
Instrument accuracy ^d	±1/2 percent	±1/2 percent	±1/2 percent	±0.25 percent

^a NA = Not Applicable.

^b Nominal value or refer to technical specification for setpoint information (as applicable).

^c The instrument seismic category and QA level information provided in the UFSAR tables may have been upgraded to meet the Pressure Boundary Integrity (PBI) or other requirements. The instrument seismic category and QA level information is available in the Fermi 2 Central Component Database.

^d The instrument accuracy information provided in the UFSAR tables is a bounding value.

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TABLE 7.6-2 GENERAL INSTRUMENTATION INFORMATION

	<u>FPCC Conductivity</u>	<u>FPCC Pump Discharge Pressure</u>	<u>FPCC Pump Low Suction Pressure</u>	<u>FPCC Refueling Bellows Leakage Rate</u>
Design classes quality/seismic category ^c	III/NA ^a	III/NA	III/NA	III/NA
Power supply	120-V ac inst. bus	120-V ac local power bus	130-V dc @ SWGR	120-V ac local power bus
No. of channels	2	2	2	1
Alarm setpoint(s) ^b	1 per demineralizer	1 per pump 90 psig decreasing = low	1 per pump -10 ft H ₂ O	5 gpm
Control logic	NA	NA	1/1	NA
Instrument range	0 to 10 micromhos/cm	0 to 200 psig	0.908 - 34.05 ft H ₂ O _g	2 to 20 gpm
Instrument accuracy ^d		±4 psig	±3 ft H ₂ O _g	±1 gpm

^a NA = Not Applicable.

^b Nominal value or refer to technical specification for setpoint information (as applicable).

^c The instrument seismic category and QA level information provided in the UFSAR tables may have been upgraded to meet the Pressure Boundary Integrity (PBI) or other requirements. The instrument seismic category and QA level information is available in the Fermi 2 Central Component Database.

^d The instrument accuracy information provided in the UFSAR tables is a bounding value.

	<u>FPCC Gate Seal Level</u>	<u>FPCC Pool Water Level</u>	<u>FPCC Surge Tank Level</u>
Design classes quality/seismic category ^c	III/NA ^a	III/NA	III/NA
Power supply	120-V ac local power bus	120-V ac local power bus	120-V ac local power bus
No. of channels	1	1	3; High, Low and Low-Low level switches
Alarm setpoint(s) ^b	5 gpm	High + 3 in. Low, 4 in. (normal = 0 in.)	High = 250 ft ³ Low = 100 ft ³
Control logic	NA	NA	1/1
Instrument range	2 to 20 gpm	8 in. H ₂ O	NA
Instrument accuracy ^d	±1 gpm	±1/2 in. H ₂ O	±1/2 in. H ₂ O

^a NA = Not Applicable.

^b Nominal value or refer to technical specification for setpoint information (as applicable).

^c The instrument seismic category and QA level information provided in the UFSAR tables may have been upgraded to meet the Pressure Boundary Integrity (PBI) or other requirements. The instrument seismic category and QA level information is available in the Fermi 2 Central Component Database.

^d The instrument accuracy information provided in the UFSAR tables is a bounding value.

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TABLE 7.6-3 SPDS SAFETY PARAMETERS AND ASSOCIATED PRIMARY VARIABLES

Safety Parameter	Primary Variable
Core cooling	Reactor water level
Fuel integrity	Reactor coolant sample analysis, offgas pretreatment radiation
Reactivity	Startup range monitor log count rate
Reactor coolant system integrity	Reactor pressure, drywell pressure, drywell sump collection rate, RPV isolation, safety/relief valve position
Containment integrity	Containment pressure, containment isolation valve positions, containment oxygen concentration, suppression pool/wetwell/torus level, drywell temperature
Radioactivity effluent to environment	Radiation level at plant release points

TABLE 7.6-4 PARAMETERS ASSOCIATED WITH SPDS DISPLAYS

Reactor Water Level

- Wide Range Div I
- Wide Range Div II
- Narrow Range Div I
- Narrow Range Div II
- Fuel Zone Range Div I
- Fuel Zone Range Div II
- Shutdown Range

Reactor Pressure

- Wide Range Div I
- Wide Range Div II
- Dome Pressure Wide Range
- Dome Pressure Narrow Range

Neutron Monitoring

- | | |
|--------|-------|
| APRM 1 | SRM A |
| APRM 2 | SRM B |
| APRM 3 | SRM C |
| APRM 4 | SRM D |

Main Steam Line Radiation

- Containment High Range Rad Mon Div I
- Containment High Range Rad Mon Div II

Drywell Pressure

- Wide Range Div I
- Wide Range Div II
- Narrow Range Div I
- Narrow Range Div II

Primary Containment 02 Level Div I

Primary Containment 02 Level Div II

Primary Containment Water Level

(Elevation 545 feet to 650 feet)

Torus Water Level Wide Range (-156 in. to +44 in.)

Torus Water Level Narrow Range (-10 in. to +10 in.)

Channels B and D

Torus Pressure Div I

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TABLE 7.6-4 PARAMETERS ASSOCIATED WITH SPDS DISPLAYS

Suppression Pool Temperature
Drywell Temperature
Fuel Pool Div I Rad Mon A
Fuel Pool Div I Rad Mon C
SJAЕ Radiation Mon A
SJAЕ Radiation Mon B
Drywell Floor Drain Sump Level
Primary Containment Isolation Valves/Signal Status
Safety Relief Valve Status
SGTS Exhaust Fan Div I Status
SGTS Exhaust Fan Div II Status
Turbine Bldg Exhaust Fan Status
Radwaste Bldg Exhaust Fan Status
Reactor Bldg Exhaust Fan Status
Gaseous Effluent Radiation Monitors
 SGTS Div I Exhaust
 SGTS Div II Exhaust
 Reactor Bldg Exhaust
 Radwaste Bldg Exhaust
 Turbine Bldg Exhaust

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TABLE 7.6-5 SRM SYSTEM TRIPS

<u>Trip Function</u>	<u>Nominal^a Setpoint</u>	<u>Trip Action^b</u>
SRM Upscale (high)	10 ⁵ c/s	Rod block, amber light display, annunciator
SRM Instrument Inoperative	(c)	Rod block, amber light display, annunciator
Detector Retraction Permissive (SRM downscale)		Bypass detector full-in-limit switch when above present limit, annunciator, green light display, rod block when below preset limit with IBM range switches on first two ranges
SRM Period	50	Annunciator, amber light display
SRM Downscale	3c/s	White light display, annunciator, rod block
SRM Bypassed		White light display

^a Nominal setpoints are included for reference only. See Technical Specifications for actual operational values.

^b Also refer to Figure 7.6-17.

^c Operate-Calibrate Switch not in Operate, module interlocks open, detector-polarizing voltage below 300 V.

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TABLE 7.6-6 IRM SYSTEM TRIPS

<u>Trip Function</u>	<u>Nominal^a Setpoint</u>	<u>Trip Action^b</u>
IRM Upscale (High-High)	120/125 fs	Scram, annunciator, red light display
IRM Instrument Inoperative	(c)	Scram, annunciator, red light display
IRM Upscale (High)	108/125 fs	Rod block, annunciator, amber light display
IRM Downscale	5/125 fs	Rod block (exception on most sensitive scale), annunciator, white light display
IRM Bypassed	NA	White light display

^a Nominal setpoints are included for references. See Technical Specifications for actual operational values.

^b Also refer to Figure 7.6-17.

^c Operate-Calibrate Switch not in Operate, module interlocks open, detector-polarizing voltage below 80 V.

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TABLE 7.6-7 LPRM SYSTEM TRIPS

<u>Trip Function</u>	<u>Trip Range</u>	<u>Trip** Setpoint</u>	<u>Trip Action</u>
LPRM Downscale	0 percent to full scale	3 fs	APRM ODA* indication and annunciator
LPRM Upscale	0 percent to full scale	100 fs	APRM ODA* indication and annunciator
LPRM Bypass	Manual selection	NA	APRM ODA* indication and APRM averaging compensation

* Digital Operator Display Assembly

** Nominal Setpoints are included for reference only

TABLE 7.6-8 APRM FLOW FUNCTIONAL TRIPS

<u>Trip Function</u>	Nominal ^a <u>Setpoint</u>	<u>Trip Action</u>
Upscale	108 fs	Rod block, APRM ODA* indication and annunciator

* Digital Operator Display Assembly

^a Nominal values are included for reference only.

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TABLE 7.6-9 APRM SYSTEM TRIP

Trip Function	Trip Point Range	Nominal ^a Value	Action ^b
APRM Downscale	0 percent to full scale	5 percent of rated thermal power	Rod block, annunciator, APRM ODA*
APRM Upscale (Rod block) (two recirc loops)	Varied with flow, intercept and slope adjustable	(0.62W ^c +54.5 percent with a clamp of 108 percent) 12 percent of rated thermal power in startup mode	Rod block, annunciator, APRM ODA*
APRM Upscale (thermal) (two recirc loops)	Varied with flow, intercept and slope adjustable	(0.62W ^c +60.2 percent) with max of 113.5 percent of rated thermal power	Scram, annunciator, APRM ODA*
APRM Inoperative	OPER-INOP switch, module interlocks open, or self-test	Not in OPER mode or critical self-test fault	Scram, rod block, annunciator, APRM ODA*
APRM Bypass	Manual switch	--	White light
APRM Upscale (neutron)	0 percent to full scale	118 percent of rated thermal power 15 percent of rated thermal power in startup mode	Scram annunciator, APRM ODA*
OPRM Upscale Trip	Growth: 1.00-1.50 Amplitude: 1.05-1.50 Confirmation count: 2-25 (Trip) and 1.00-1.30 (Amplitude)	Growth: 1.30 Amplitude: 1.30 Confirmation Count: 14 (Trip) and 1.11 (Amplitude)	Scram, Annunciator, APRM ODA*
OPRM Upscale 1.05-1.50	Growth: 1.00-1.50 Amplitude: 1.20 Confirmation Count: 2-25 (Trip)	Growth: 1.20 APRM ODA* Confirmation Count 12 (Trip)	Annunciator, Amplitude:
OPRM Inoperative	0-44 (Min. OPRM Cells required)	<21 OPRM Cells operable	Annunciator, APRM ODA*
OPRM Enable	STP: 10-40% Flow: 50-100%	≥27.5% STP <60% drive flow	Annunciator, APRM ODA*

* Digital Operator Display Assembly

^a See Technical Specifications for actual operational values

^b Also see Figure 7.6-16.

^c W is recirculation loop flow.

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TABLE 7.6-10 RBM SYSTEM TRIPS

Trip Function	Nominal Setpoint ^(a)	Trip Action
RBM Downscale RBM ODA*	94%	Rod block, annunciator,
RBM Inoperative	(b)	Rod block, annunciator, RBM ODA*
RBM Upscale ^(c)	LTSP = 114.0; ITSP = 118.2; HTSP = 104.4	Rod block, annunciator, RBM ODA*
RBM Bypassed	Manual switch	RBM ODA*, White light Display

* Digital Operator Display Assembly

(a) Nominal setpoints for reference only. See Core Operating Limits Report (COLR) for actual operational values.

(b) OPER - INOP switch not in OPER, module interlocks open, too few inputs, failure to adjust gain or more than one rod selected.

(c) These setpoints are in percent of reference level (Refer to Figure 7.6-22(b) for additional information).

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TABLE 7.6-11 SPENT FUEL POOL LEAKAGE ALARMS IN CONTROL ROOM

Condition	Setpoint (nominal)
Fuel pool system temperature high	130 °F
Fuel pool water level low	4 in. below normal level
Fuel pool system trouble	Any alarm contacts from surge tank high or low, gate leakage, refueling bellows leakage, pump A or pump B discharge pressure low.
Fuel pool water level high	3 in. above normal level

TABLE 7.6-12 HAS BEEN INTENTIONALLY DELETED

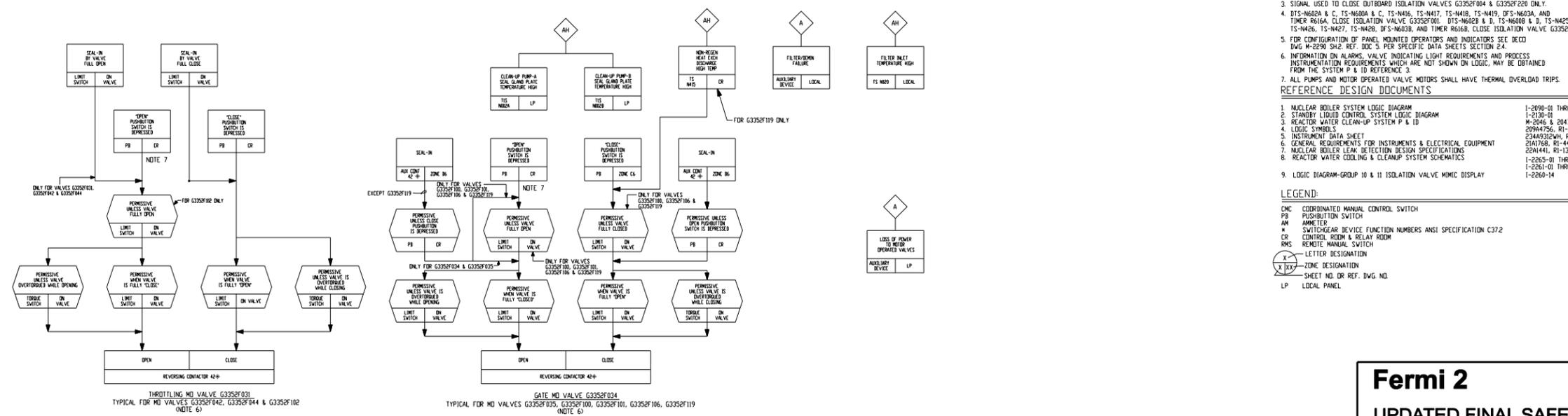
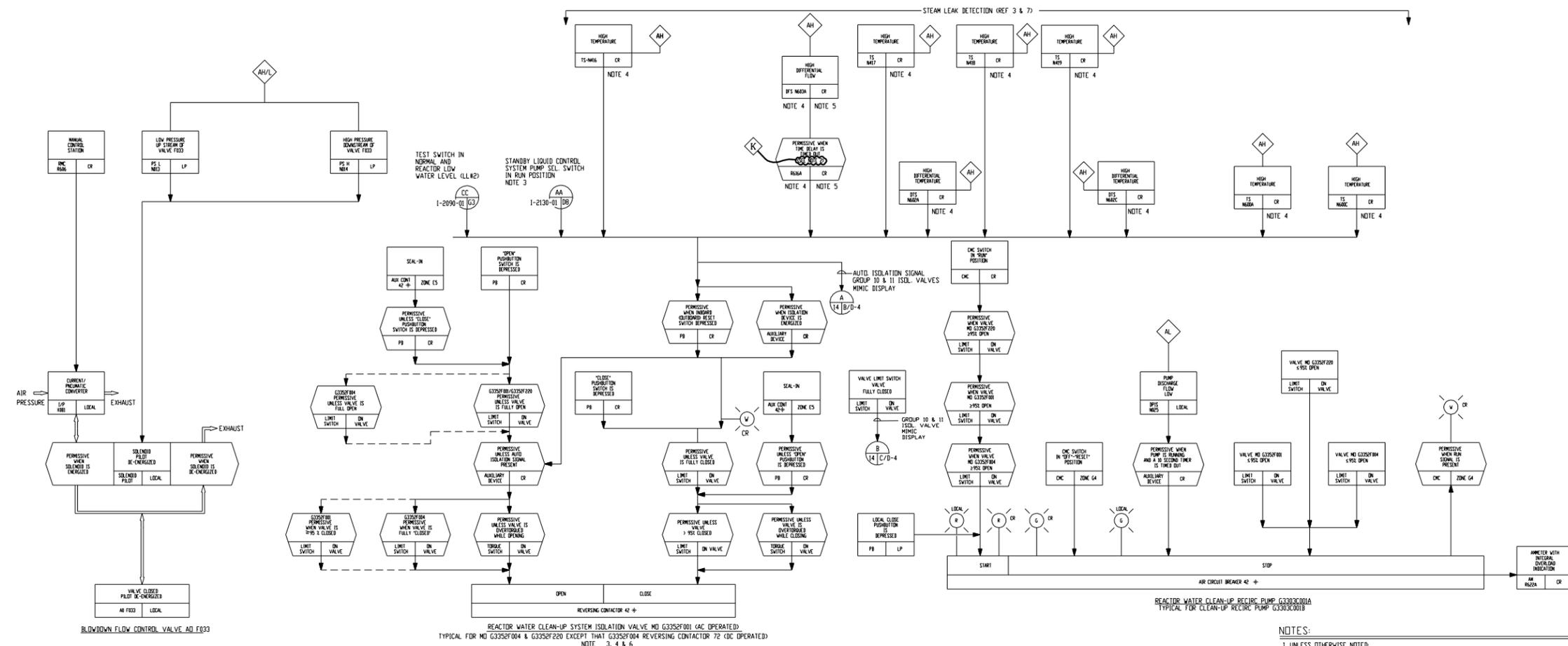
TABLE 7.6-13 HAS BEEN INTENTIONALLY DELETED

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TABLE 7.6-14 REACTOR WATER CLEANUP ANNUNCIATORS

<u>Function</u>	<u>Trip Point</u>
Reactor Water Cleanup Pump Low Flow	70 gpm
Reactor Water Cleanup Pump Seal Gland Plate Temperature High	250 °F
Reactor Water Filter-Demineralizer Inlet High Temperature	130 °F
Reactor Water Cleanup/Blowdown Line Pressure High	140 psi
Reactor Water Cleanup/Blowdown Line Pressure Low	5 psi
Reactor Water Cleanup Steam Leakage High Area Temperature	175 °F ^a
Reactor Water Cleanup Filter Demineralizer Trouble	Any alarm on load operated
Reactor Water Cleanup Valves Thermal Overload	Any valve over-load operated
Reactor Water Cleanup Differential High	55.1 gpm ^a
Reactor Water Conductivity High	Multiple setpoints

^a Nominal value - refer to Technical Specifications for setpoint information.

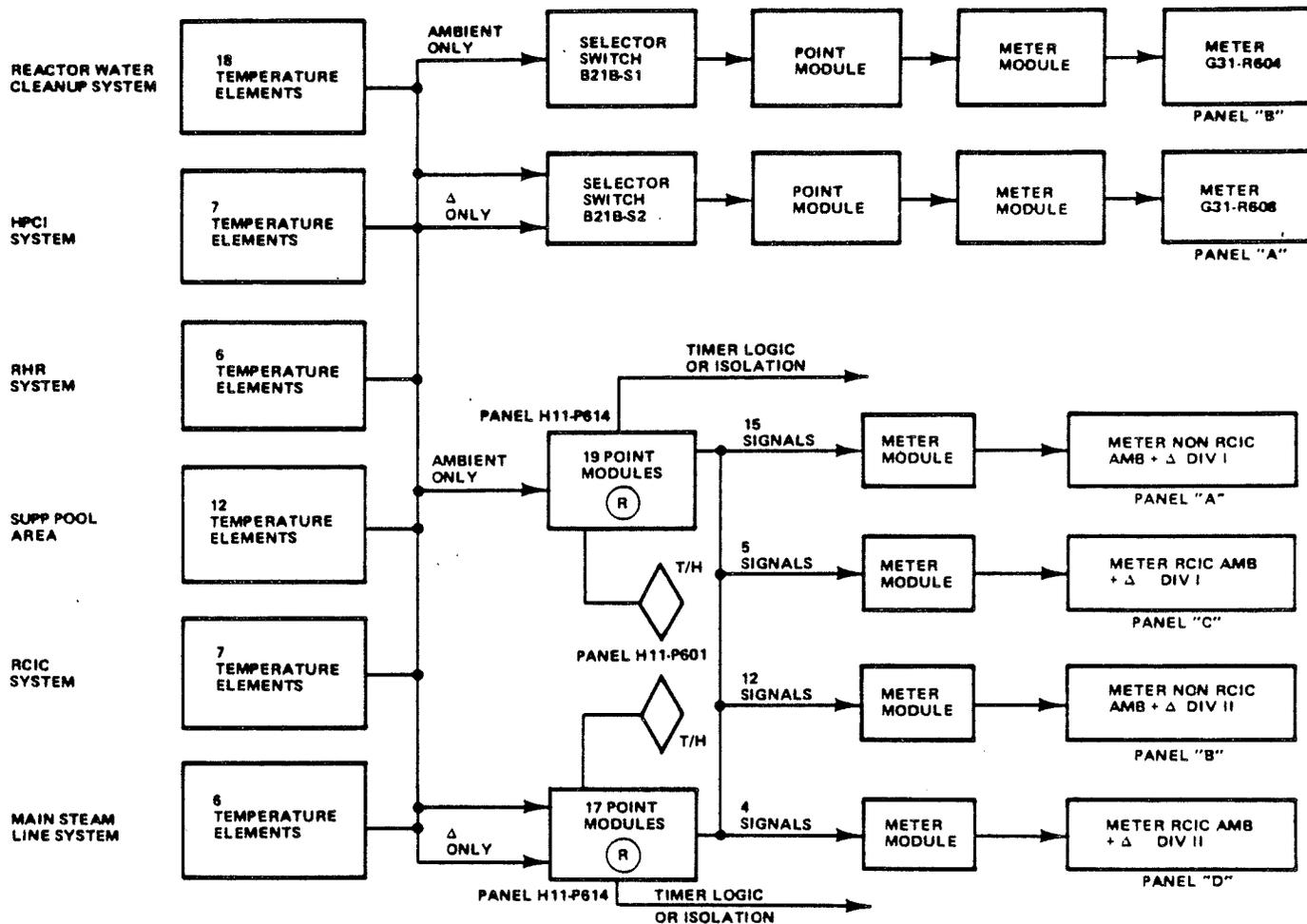


- NOTES:**
- UNLESS OTHERWISE NOTED: ALL INSTRUMENT PIS #s ARE PREFIXED G33. ALL VALVES & EQUIPMENT PIS #s ARE PREFIXED G3300.
 - FDR LOCATION, SETPOINTS AND IDENTIFICATION OF INSTRUMENTS SEE CENTRAL COMPONENT DATA BASE (CCED).
 - SIGNAL USED TO CLOSE OUTBOARD ISOLATION VALVES G3352F004 & G3352F220 ONLY.
 - DTS-M603A & C, TS-M426 & D, TS-M427, TS-M428, TS-M429, TS-M430, DFS-M603A, AND TIMER R616A, CLOSE ISOLATION VALVE G3352F001. DTS-M602B & D, TS-M603B & D, TS-M425, TS-M426, TS-M427, TS-M428, DFS-M603B, AND TIMER R616B, CLOSE ISOLATION VALVE G3352F004.
 - FDR CONFIGURATION OF PANEL MOUNTED OPERATORS AND INDICATORS SEE DECO DWG M-2250 SIA2, REF. DOC 3, PER SPECIFIC DATA SHEETS SECTION 24.
 - INFORMATION ON ALARMS, VALVE INDICATING LIGHT REQUIREMENTS AND PROCESS INSTRUMENTATION REQUIREMENTS WHICH ARE NOT SHOWN ON LOGIC, MAY BE OBTAINED FROM THE SYSTEM P & ID REFERENCE 3.
 - ALL PUMPS AND MOTOR OPERATED VALVE MOTORS SHALL HAVE THERMAL OVERLOAD TRIPS.
- REFERENCE DESIGN DOCUMENTS**
- | | |
|--|---|
| 1. NUCLEAR BOILER SYSTEM LOGIC DIAGRAM | I-2090-01 THRU -06 |
| 2. STANDBY LIQUID CONTROL SYSTEM LOGIC DIAGRAM | I-2130-01 |
| 3. REACTOR WATER CLEAN-UP SYSTEM P & ID | M-2046 & 2047 |
| 4. LOGIC SYMBOLS | 209A4756, R1-20 |
| 5. INSTRUMENT DATA SHEET | 2344932EWA, RA-4411 |
| 6. GENERAL REQUIREMENTS FOR INSTRUMENTS & ELECTRICAL EQUIPMENT | 22A1768, R1-446 |
| 7. NUCLEAR BOILER LEAK DETECTION DESIGN SPECIFICATIONS | 22A1441, R1-138 |
| 8. REACTOR WATER COOLING & CLEANUP SYSTEM SCHEMATICS | I-2265-01 THRU -06
I-2261-01 THRU -11
I-2260-14 |
- LEGEND:**
- CNC COORDINATED MANUAL CONTROL SWITCH
 - PB PUSHBUTTON SWITCH
 - AM AMMETER
 - M SWITCHGEAR DEVICE FUNCTION NUMBERS ANSI SPECIFICATION C37.2
 - CR CONTROL ROOM & RELAY ROOM
 - RMS REMOTE MANUAL SWITCH
 - X LETTER DESIGNATION
 - X [Z] ZONE DESIGNATION
 - LP LOCAL PANEL

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FIGURE 7.6-1

REACTOR WATER CLEANUP SYSTEM FCD

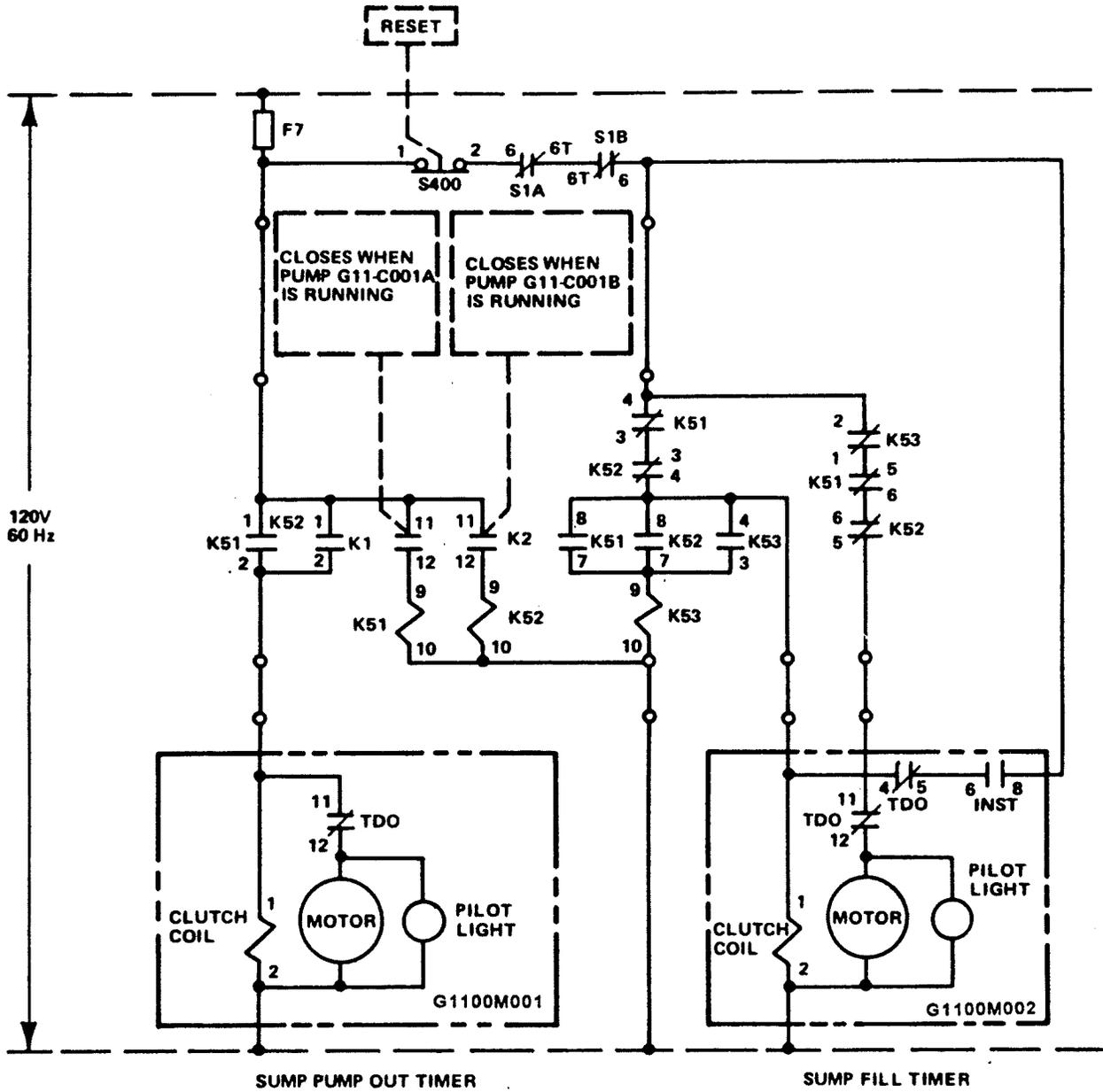


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FIGURE 7.6-2

AREA TEMPERATURE MONITORING SYSTEM
BLOCK DIAGRAM

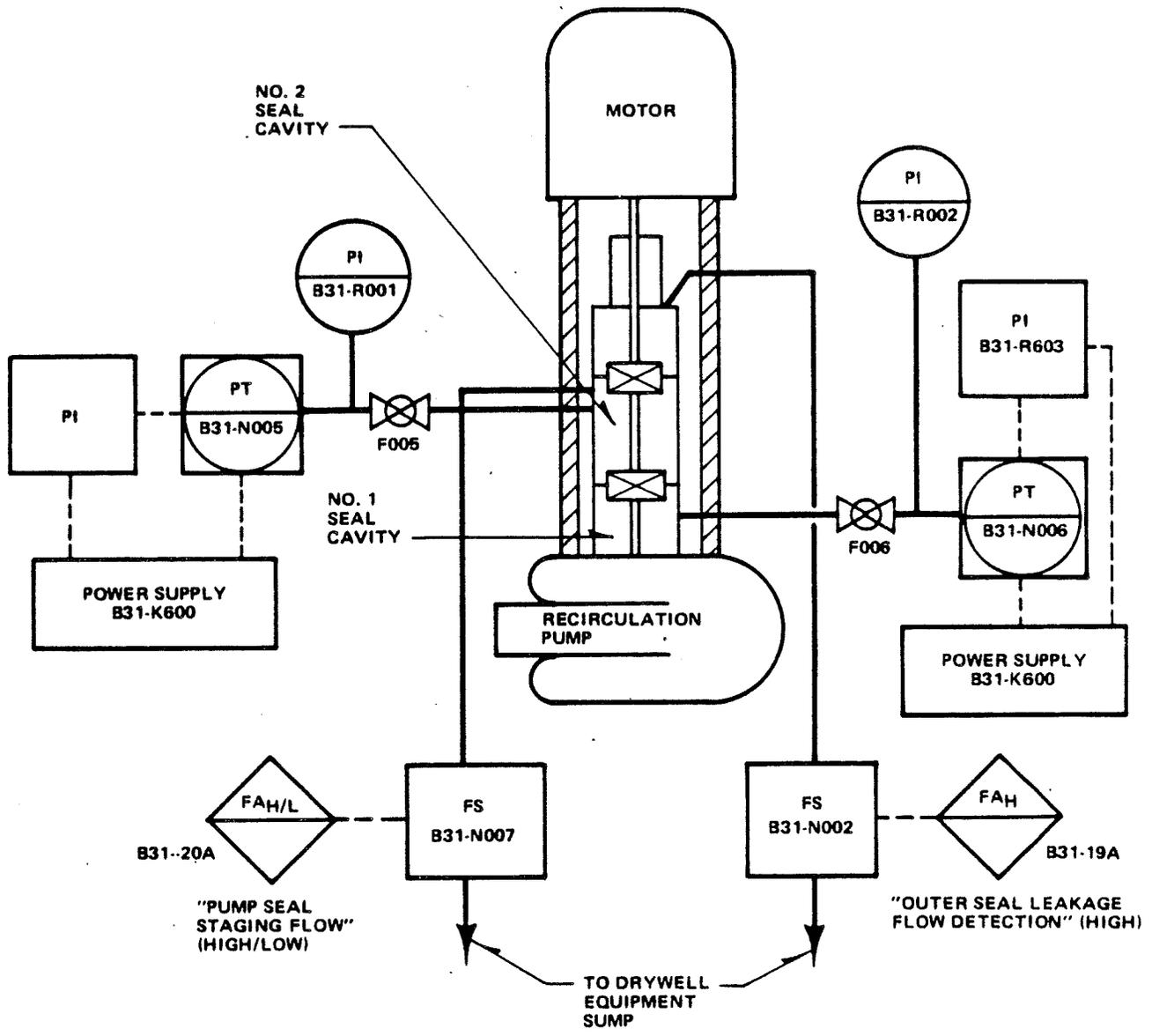


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FIGURE 7.6-4

EQUIPMENT AREA LEAK DETECTION SYSTEM
TYPICAL



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FIGURE 7.6-5
 RECIRCULATION PUMP LEAK DETECTION
 BLOCK DIAGRAM

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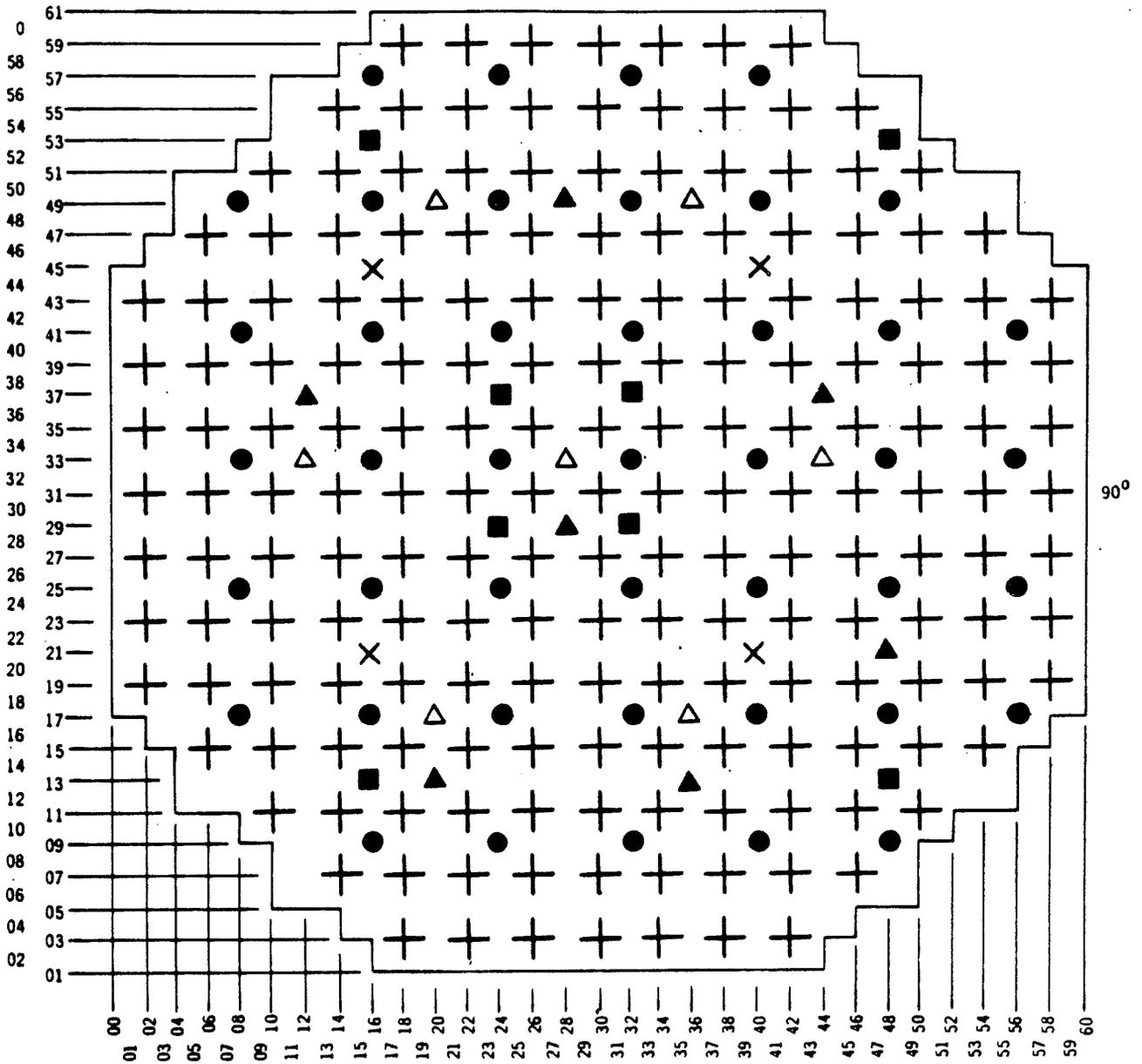
FIGURES 7.6-7, 7.6-8, and 7.6-9 HAVE BEEN DELETED

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FIGURE 7.6-10 HAS BEEN DELETED

TOP VIEW

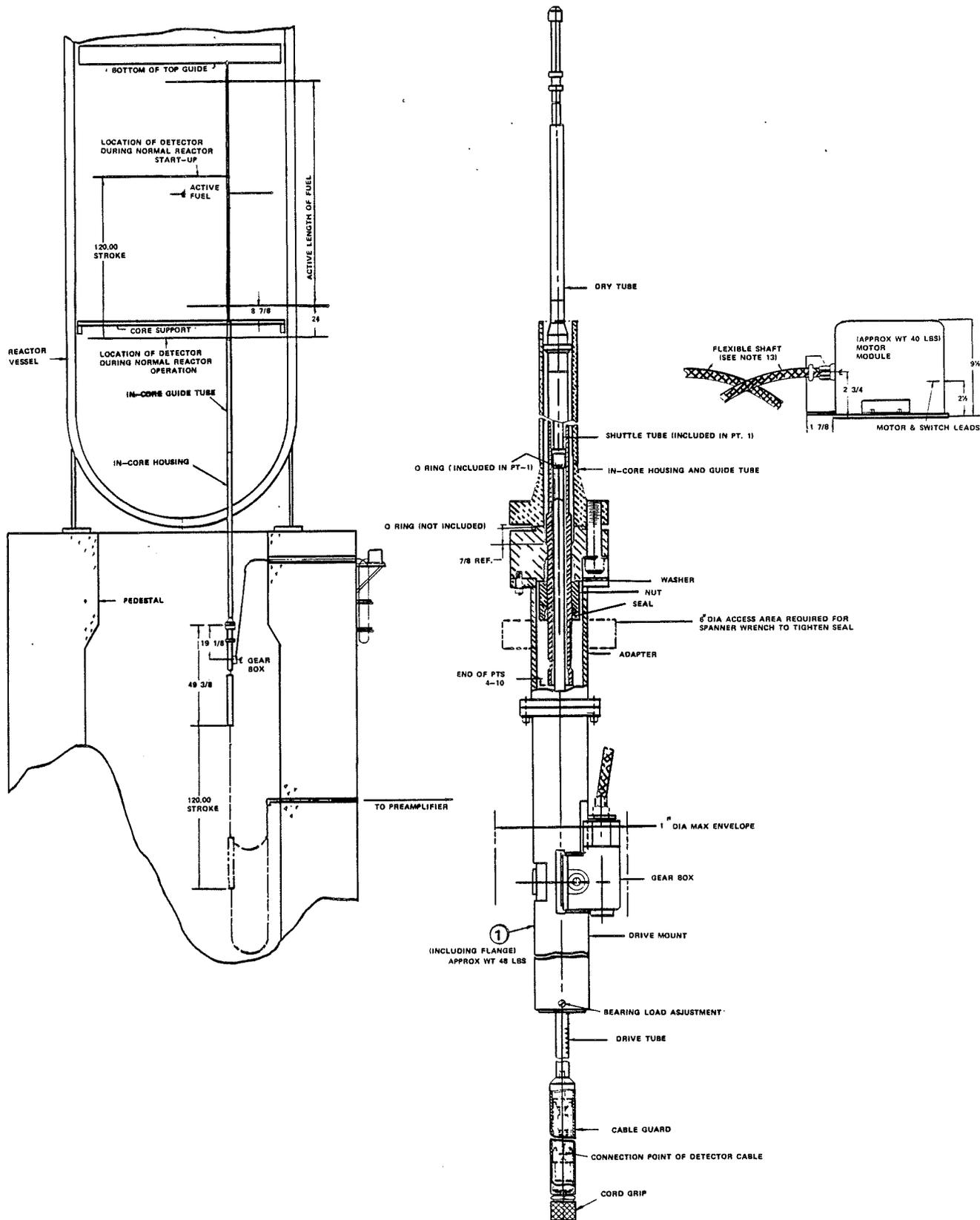
0°



- LOCAL POWER RANGE MONITORING SYSTEM (LPRM) 43
- × SOURCE RANGE MONITORING SYSTEM (SRM) 4
- INTERMEDIATE RANGE MONITORING SYSTEM (IRM) 8
- ▲ DESIGN NEUTRON SOURCE LOCATIONS 7
- △ SPARE NEUTRON SOURCE LOCATIONS 7

Note: All neutron sources removed from core during first refueling outage.

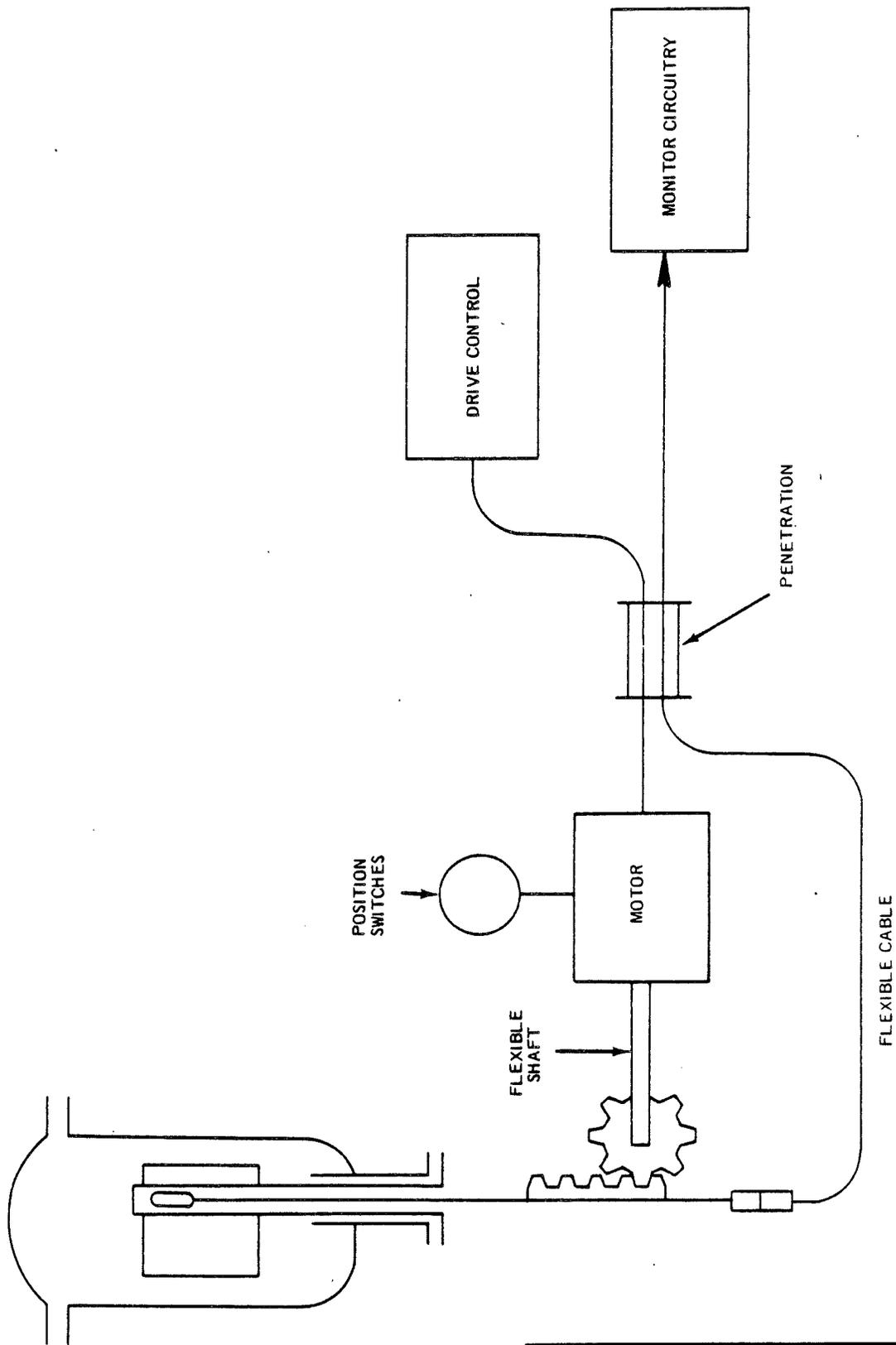
<p>Fermi 2 UPDATED FINAL SAFETY ANALYSIS REPORT</p>
<p>FIGURE 7.6-12 NEUTRON MONITORING SYSTEM CORE LOCATION</p>



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FIGURE 7.6-13

SOURCE/INTERMEDIATE RANGE MONITOR
 NEUTRON MONITORING UNIT

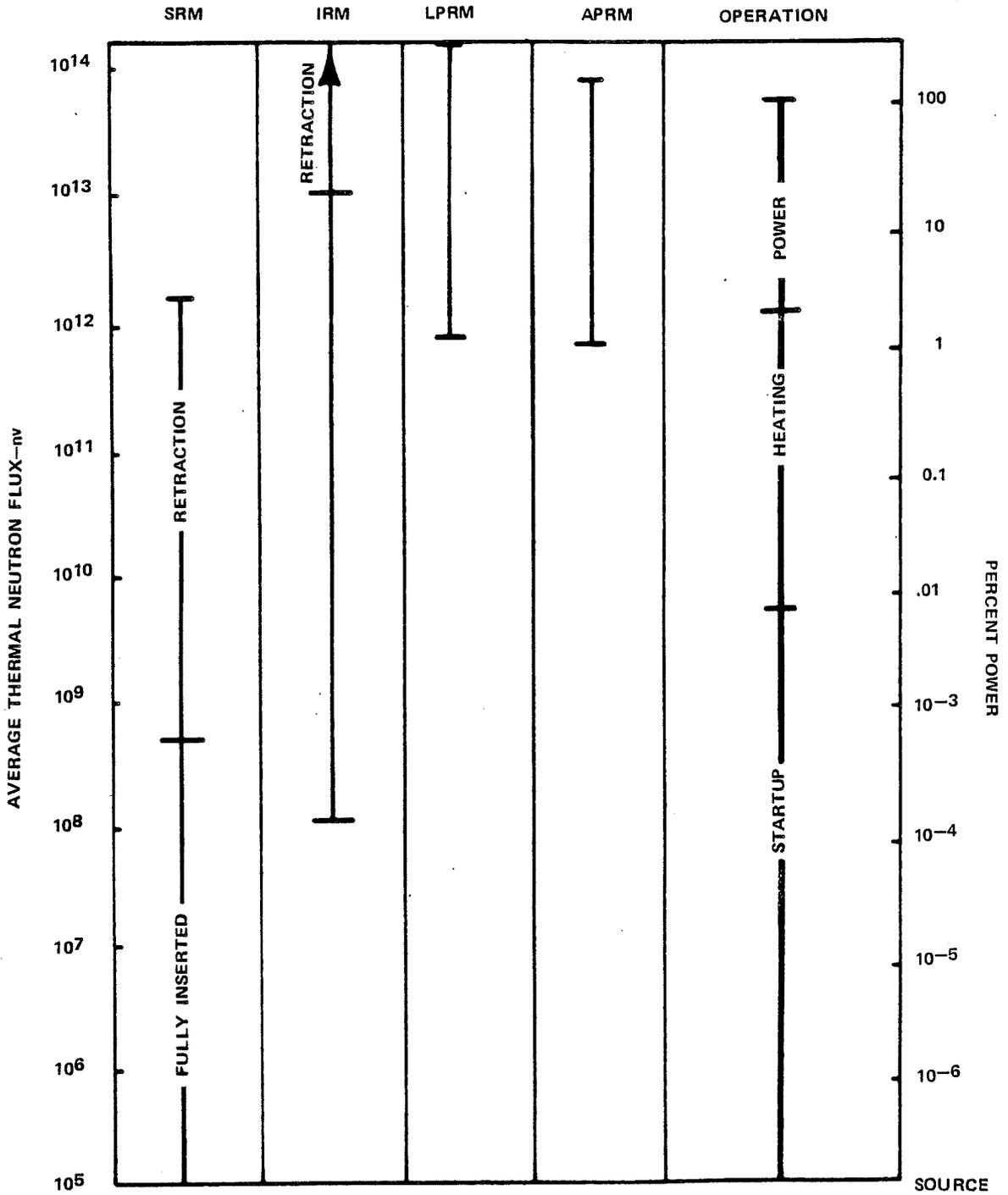


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FIGURE 7.6-14

DETECTOR DRIVE SYSTEM SCHEMATIC



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FIGURE 7.6-15

RANGES OF NEUTRON MONITORING SYSTEM

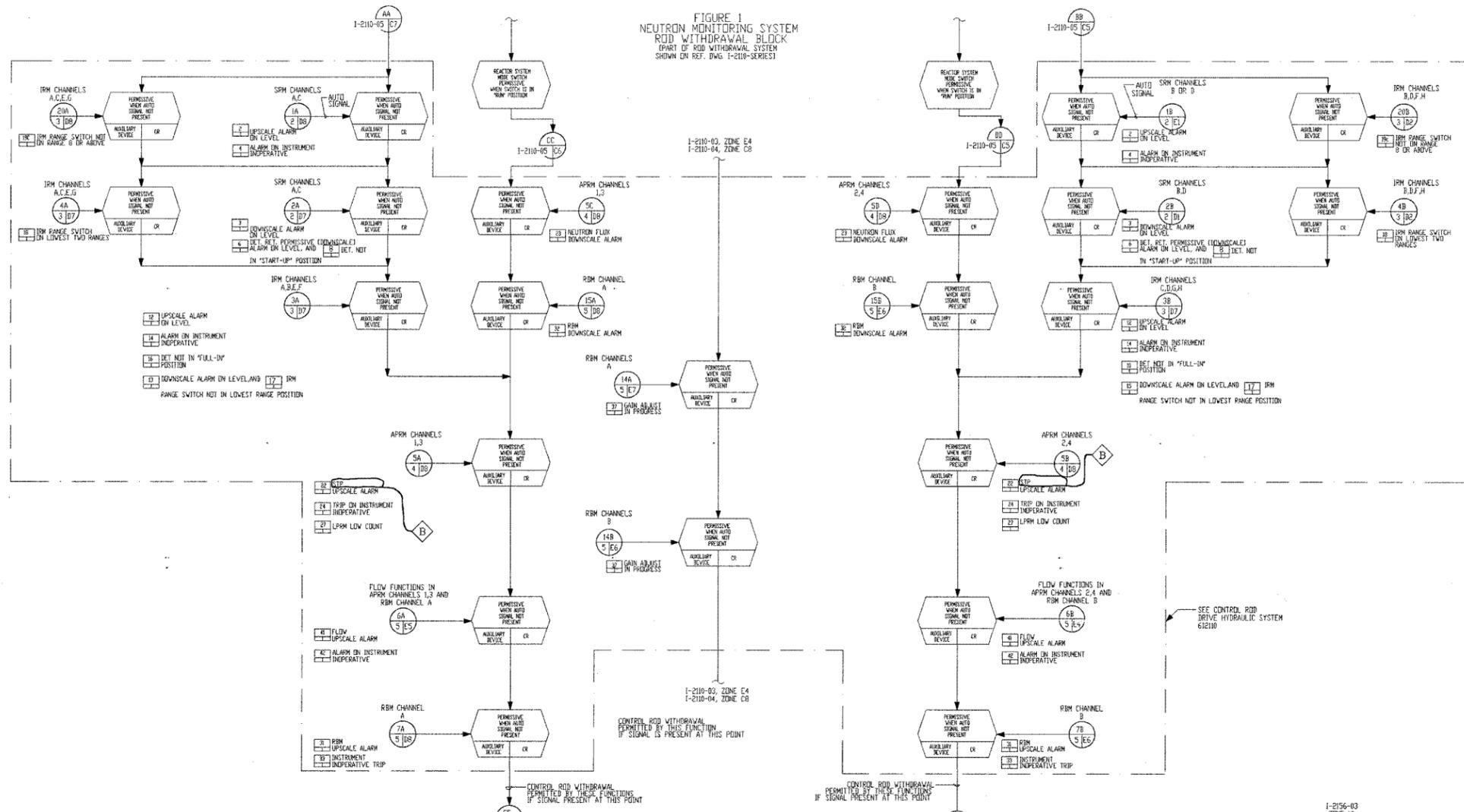


FIGURE 1
NEUTRON MONITORING SYSTEM
ROD WITHDRAWAL BLOCK
PART OF ROD WITHDRAWAL SYSTEM
SHOWN IN REF. DWG. I-2110-SR-02

- NOTES:
1. INPUTS TO COMPUTER ARE ISOLATED CLOSE TO ALARM CONTACTS.
 2. SEE I-2140-07.
 3. SEE I-2140-07.
 4. THE ENTIRE NEUTRON MONITORING SYSTEM IS A FULLY AUTOMATIC SYSTEM EXCEPT FOR MANUAL OPERATED SWITCHES.
 5. UNLESS OTHERWISE SHOWN ALL INSTRUMENT PIS NO.'S ARE PREFIXED C01. ALL VALVE AND EQUIPMENT NO.'S ARE PREFIXED C5100.
 6. CHANNELS A,C,E & G ARE FOR TRIP SYSTEM A. CHANNELS B,D,F & H ARE FOR TRIP SYSTEM B.
 7. FOR CONFIGURATION OF PANEL MOUNTED OPERATORS AND INDICATORS SEE DWG. I-2090-02, PUR. SPECIFICATION DATA SHEET SEC. 2.4.
- NOTES CONTINUED ON I-2140-07

- LEGEND:
- IRM - INTERMEDIATE RANGE MONITOR
 - RBM - RED BLOCK MONITOR
 - APRM - AVERAGE POWER RANGE MONITOR
 - SRM - SOURCE RANGE MONITOR
 - LRPM - LOCAL POWER RANGE MONITOR
 - APRM-ODA - APRM OPERATOR DISPLAY ASSEMBLY
 - RBM-ODA - RBM OPERATOR DISPLAY ASSEMBLY
 - OPRM - OSCILLATION POWER RANGE MONITOR
 - STP - SIMULATED THERMAL POWER

- REFERENCE DOCUMENTS:
1. NEUTRON MONITORING SYSTEM IED I-2146-01.02
 2. CONTROL ROD DRIVE HYD SYS LOGIC DIAGRAM I-2110-01
 3. NUCLEAR BOILER SYSTEM LOGIC DIAGRAM I-2090-01
 4. REACTOR PROTECTION SYSTEM IED I-2156-01.02.03.04
 5. PROCESS COMPUTER SYSTEM INPUT/OUTPUT REQUIREMENTS DESIGN SPEC. 2244758 (RI-3092)
 6. LOGIC SYMBOLS 20944756 (RI-20)
 7. NEUTRON MONITORING SYSTEM ARRANGEMENT 7302675 (RI-25)
 8. GENERAL REQUIREMENTS FOR INSTRUMENTATION AND ELECTRICAL EQUIPMENT 241158 (RI-446)

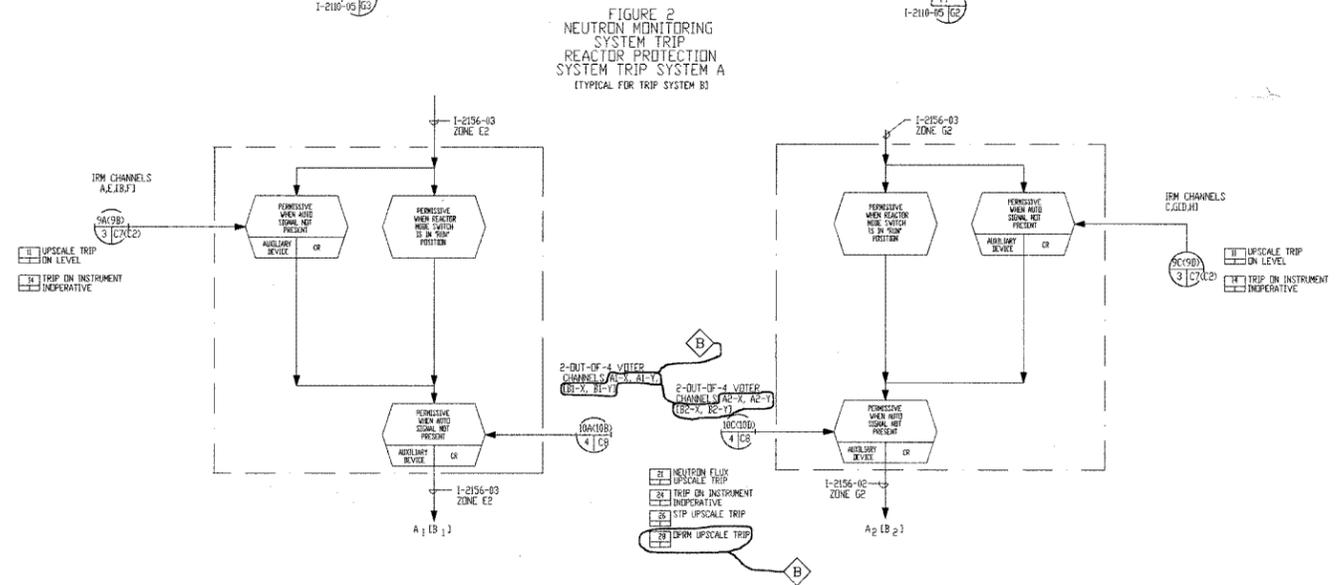


FIGURE 2
NEUTRON MONITORING SYSTEM TRIP
REACTOR PROTECTION SYSTEM TRIP SYSTEM A
(TYPICAL FOR TRIP SYSTEM B)

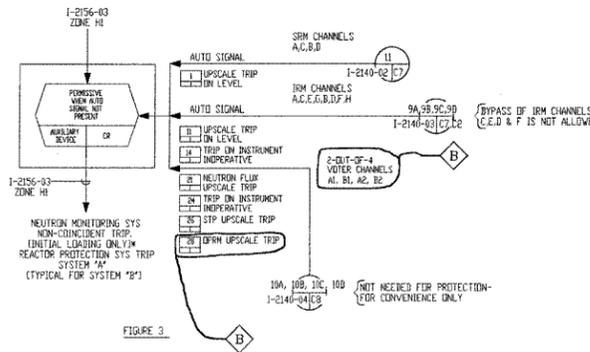
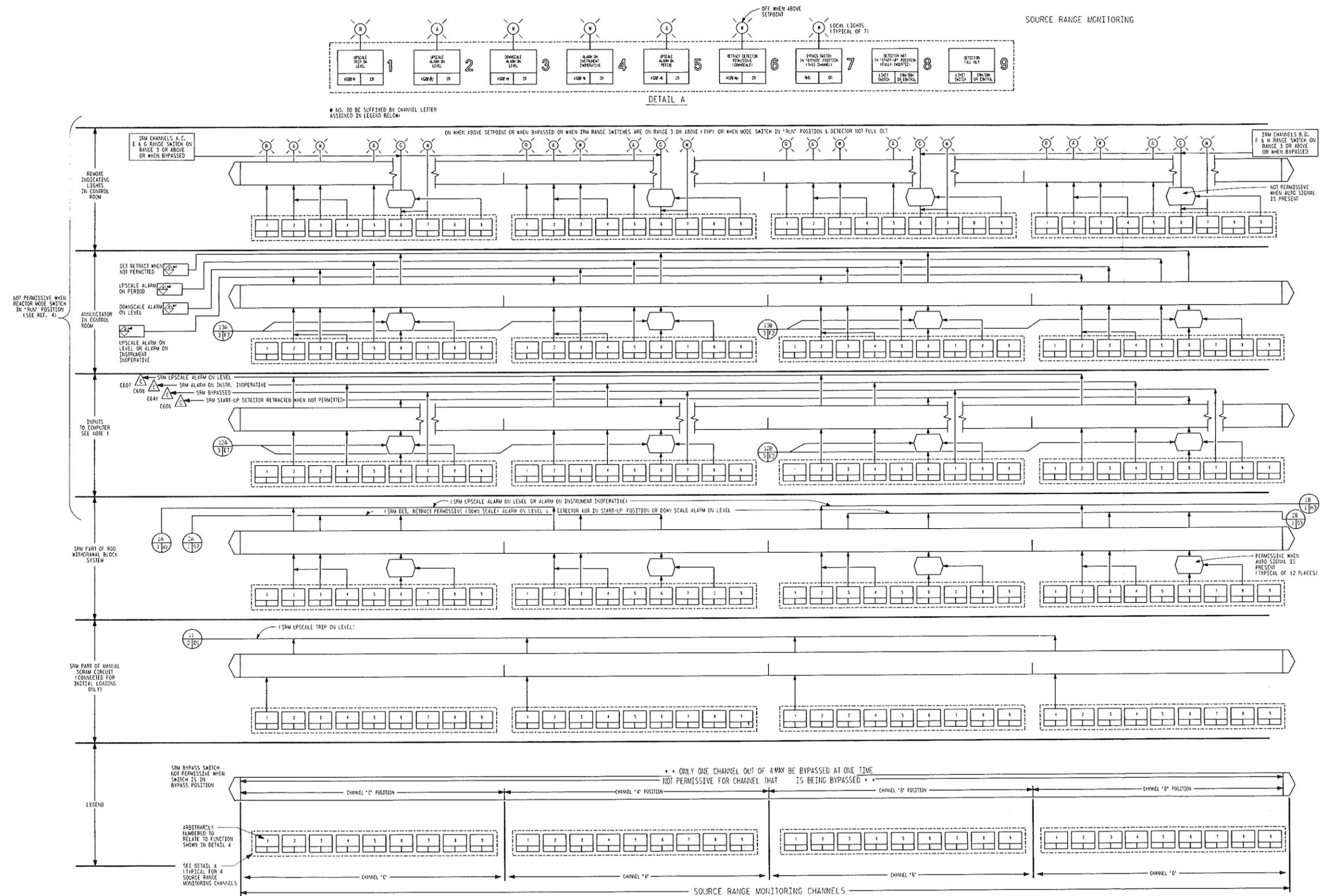


FIGURE 3

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FIGURE 7.6-16, SHEET 1
NEUTRON MONITORING SYSTEM FCD

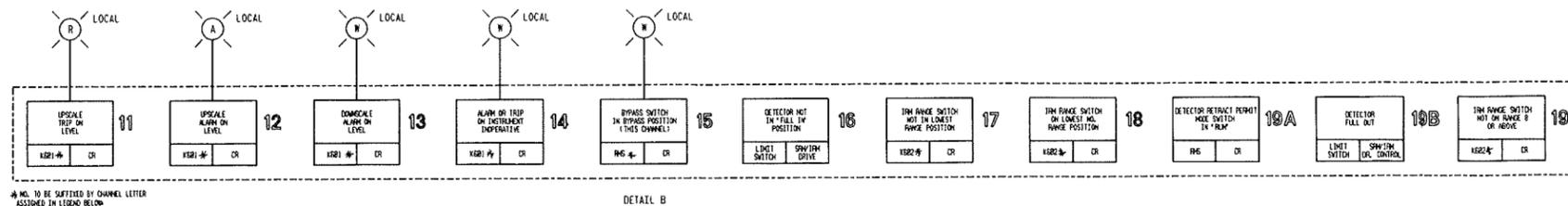


NOTE: SEE 61721-2140-1 (FIGURE 7.6-16, SHEET 1) FOR NOTES, REFERENCES AND LEGEND.

THIS FIGURE IS NOT UPDATED PERIODICALLY

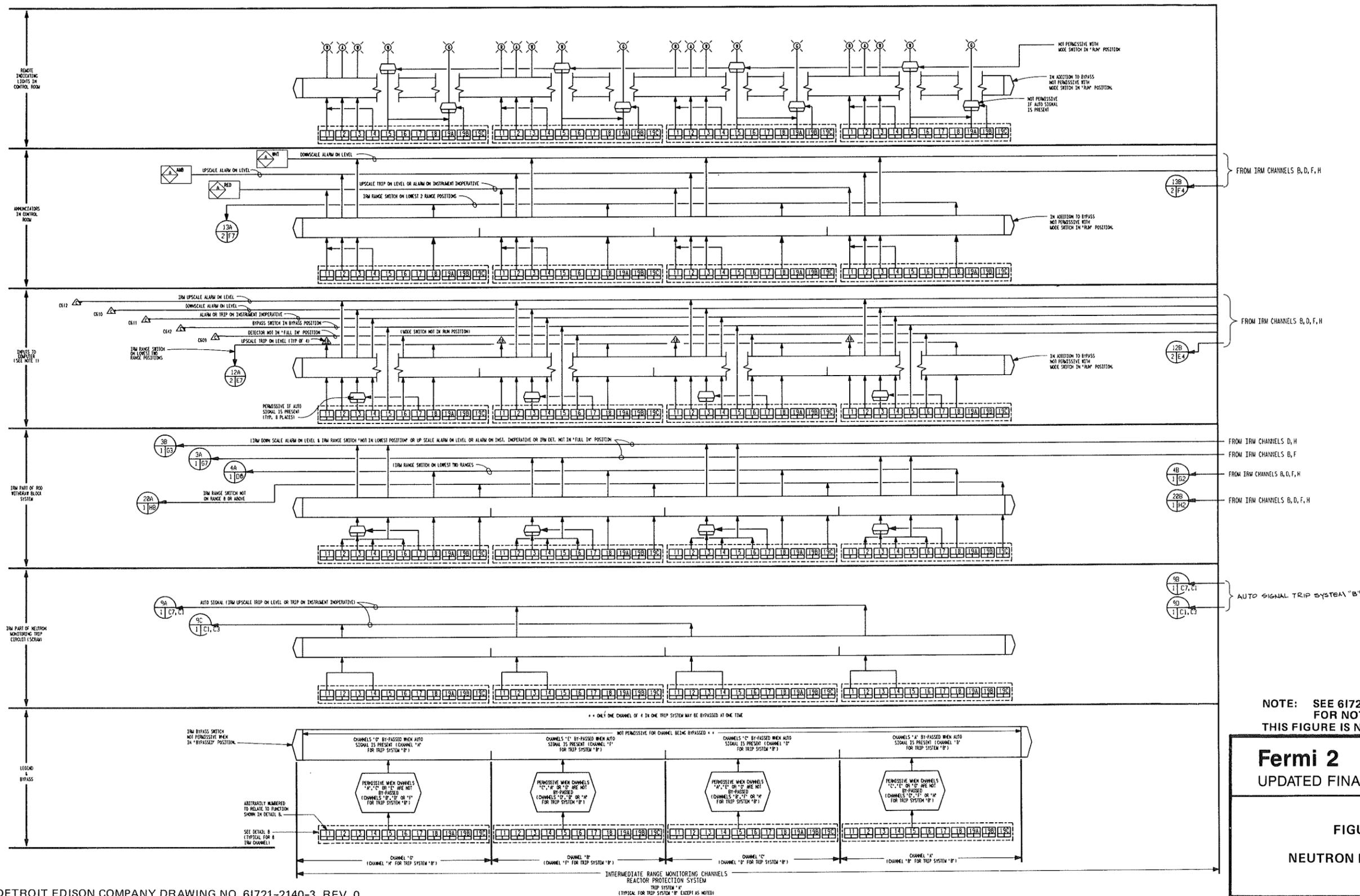
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FIGURE 7.6-16, SHEET 2
 NEUTRON MONITORING SYSTEM FCD



* NO. TO BE SUFFIXED BY CHANNEL LETTER ASSIGNED IN LEGEND BELOW

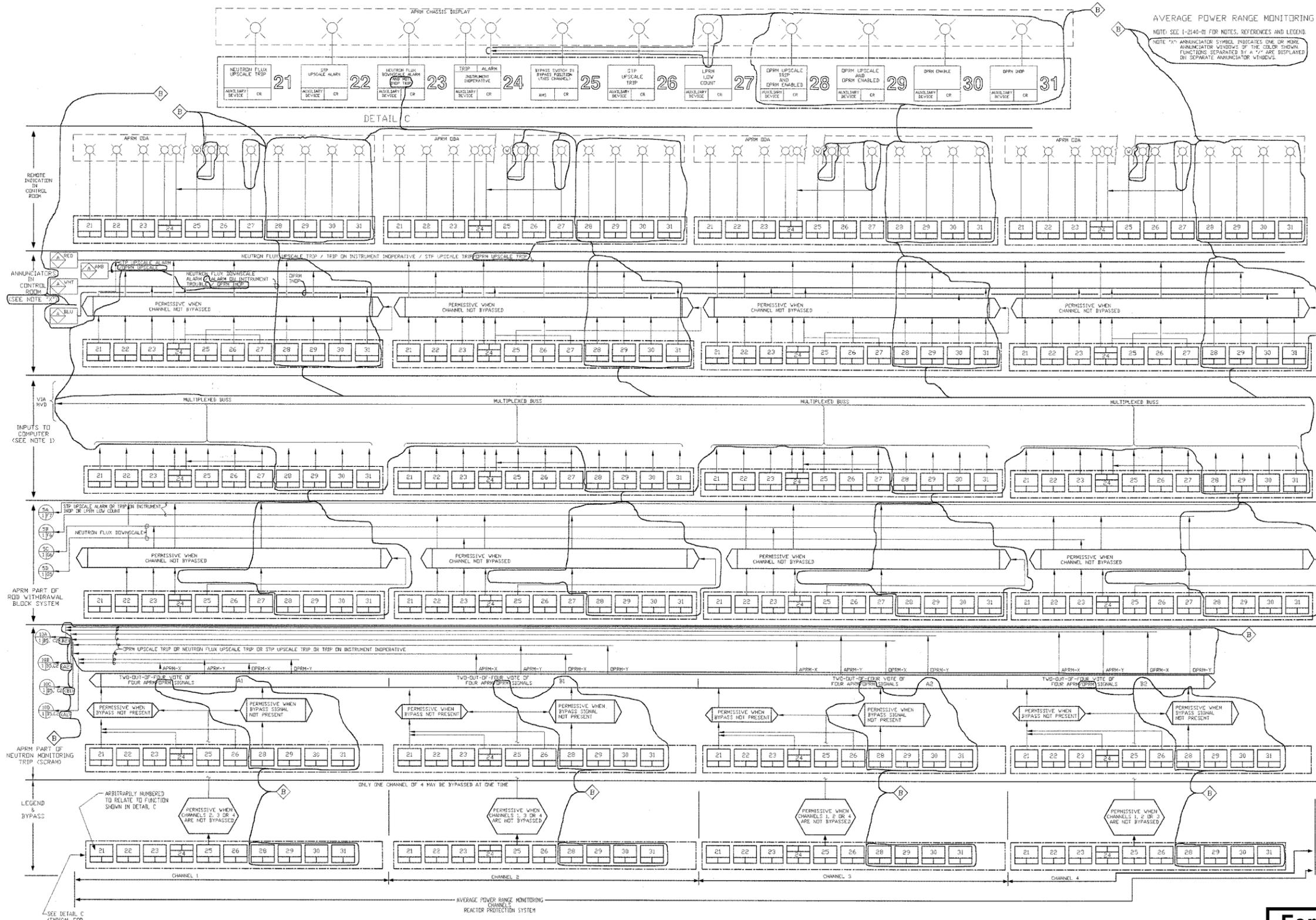
DETAIL B



NOTE: SEE 61721-2140-1 (FIGURE 7.6-16, SHEET 1) FOR NOTES, REFERENCES AND LEGEND. THIS FIGURE IS NOT UPDATED PERIODICALLY

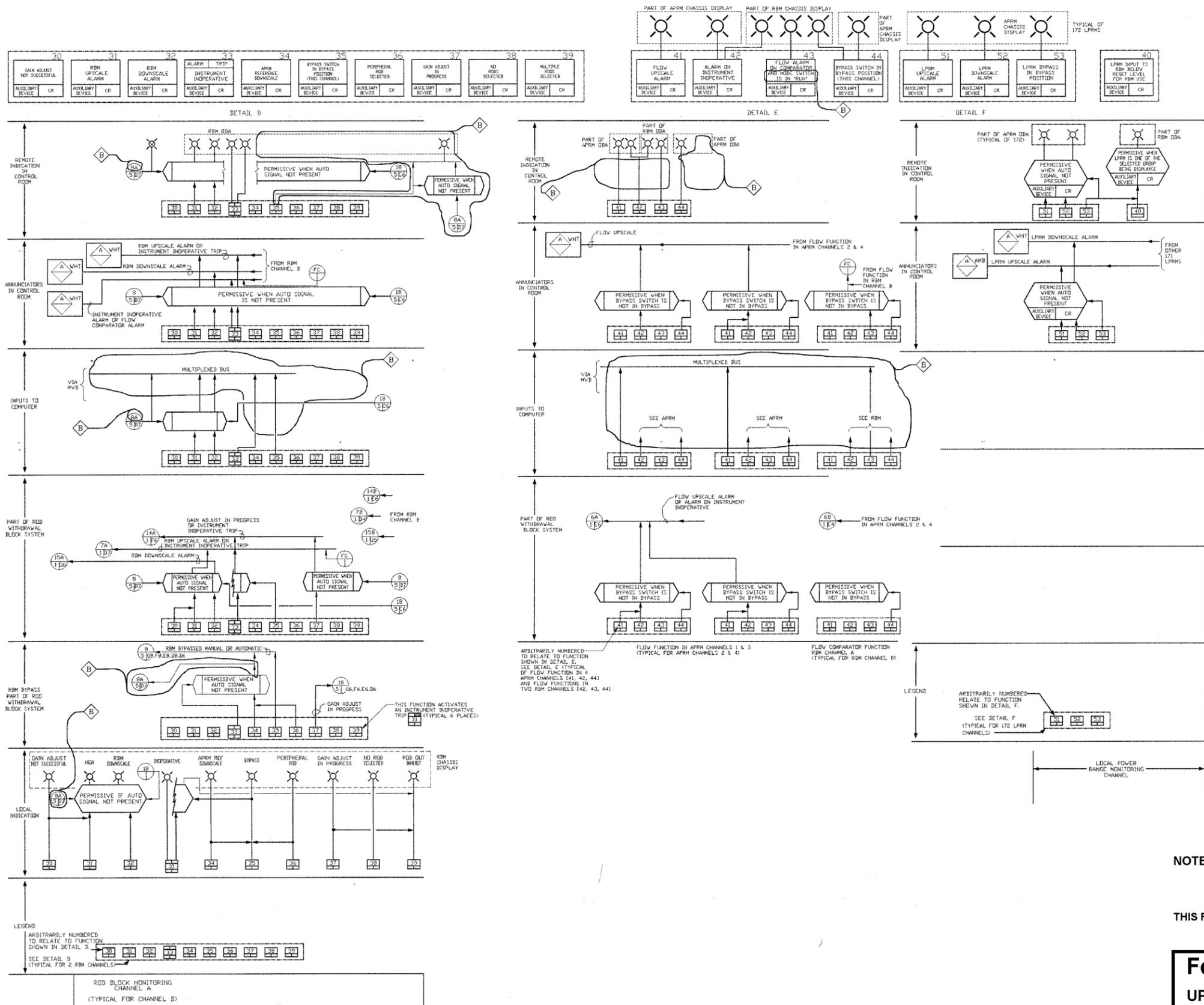
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FIGURE 7.6-16, SHEET 3
NEUTRON MONITORING SYSTEM FCD



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FIGURE 7.6-16, SHEET 4
NEUTRON MONITORING SYSTEM FCD



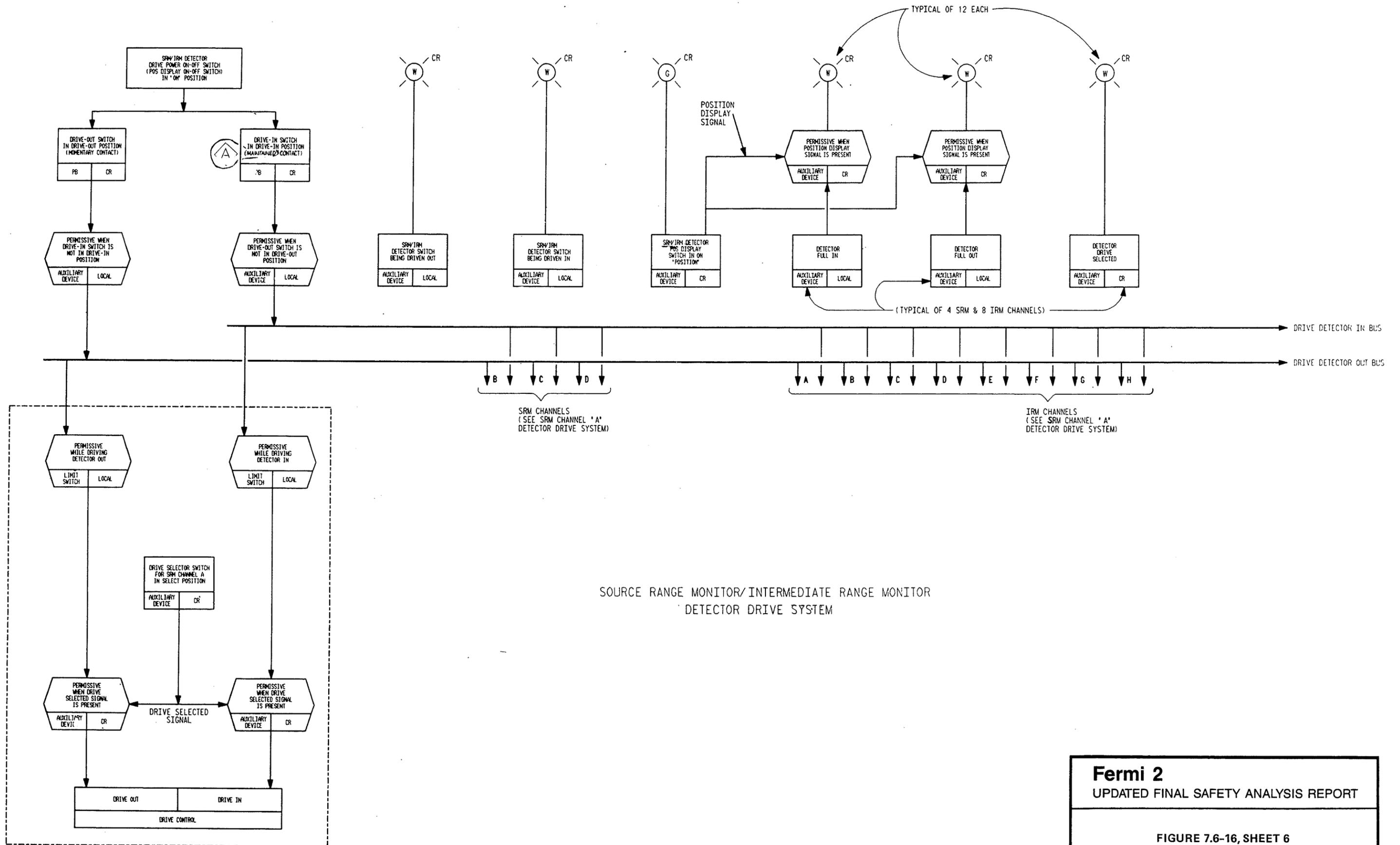
ROD BLOCK MONITORING
LOCAL POWER RANGE MONITORING
FLOW UNIT

NOTE: SEE 61721-2140-1 (FIGURE 7.6-16, SHEET 1)
FOR NOTES, REFERENCES AND LEGEND

THIS FIGURE IS NOT UPDATED PERIODICALLY

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FIGURE 7.6-16, SHEET 5
NEUTRON MONITORING SYSTEM FCD

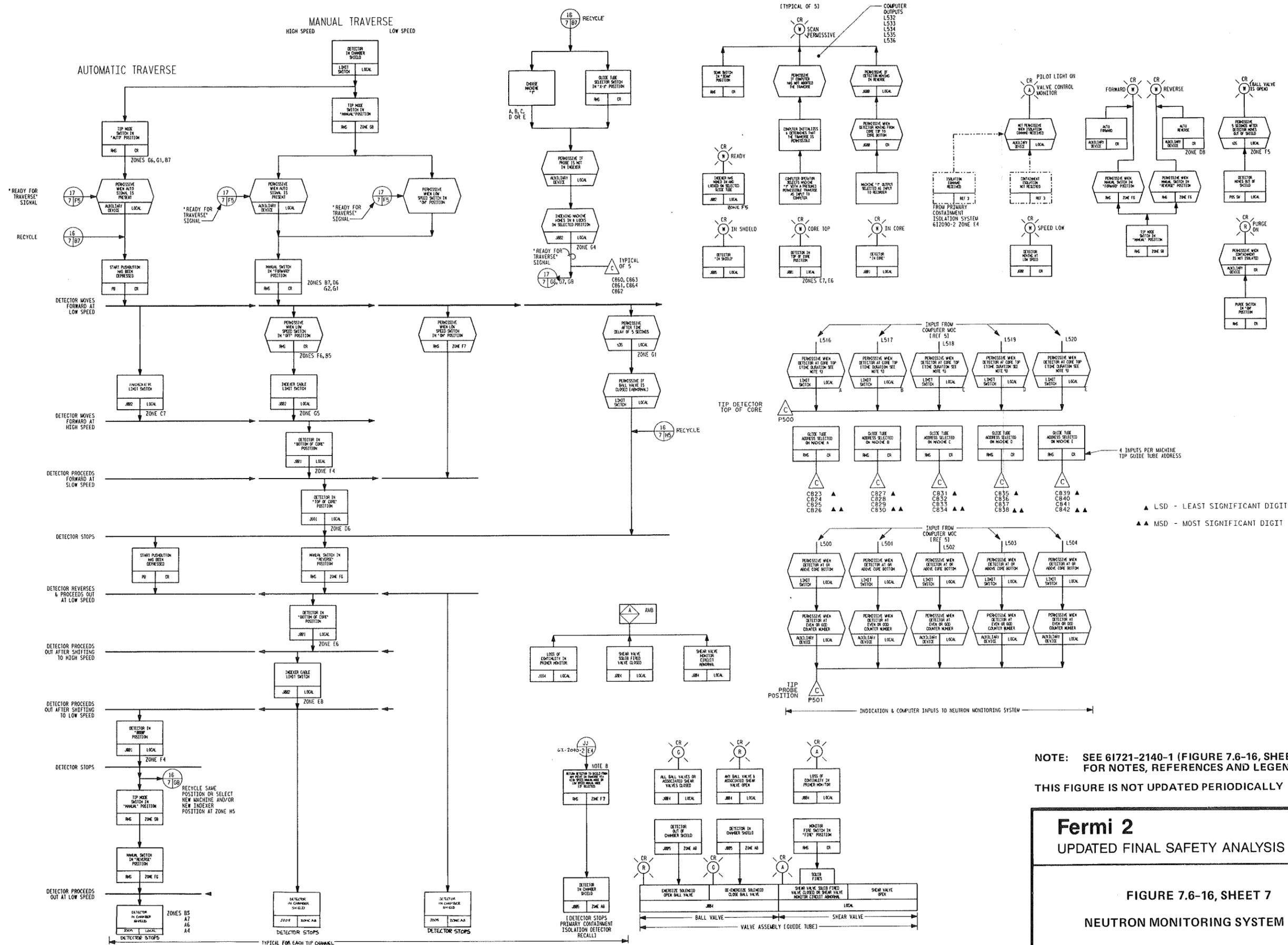


SRM CHANNEL "A"
TYPICAL DETECTOR
DRIVE SYSTEM

SOURCE RANGE MONITOR/INTERMEDIATE RANGE MONITOR
DETECTOR DRIVE SYSTEM

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FIGURE 7.6-16, SHEET 6
NEUTRON MONITORING SYSTEM FCD



NOTE: SEE 61721-2140-1 (FIGURE 7.6-16, SHEET 1) FOR NOTES, REFERENCES AND LEGEND.
THIS FIGURE IS NOT UPDATED PERIODICALLY

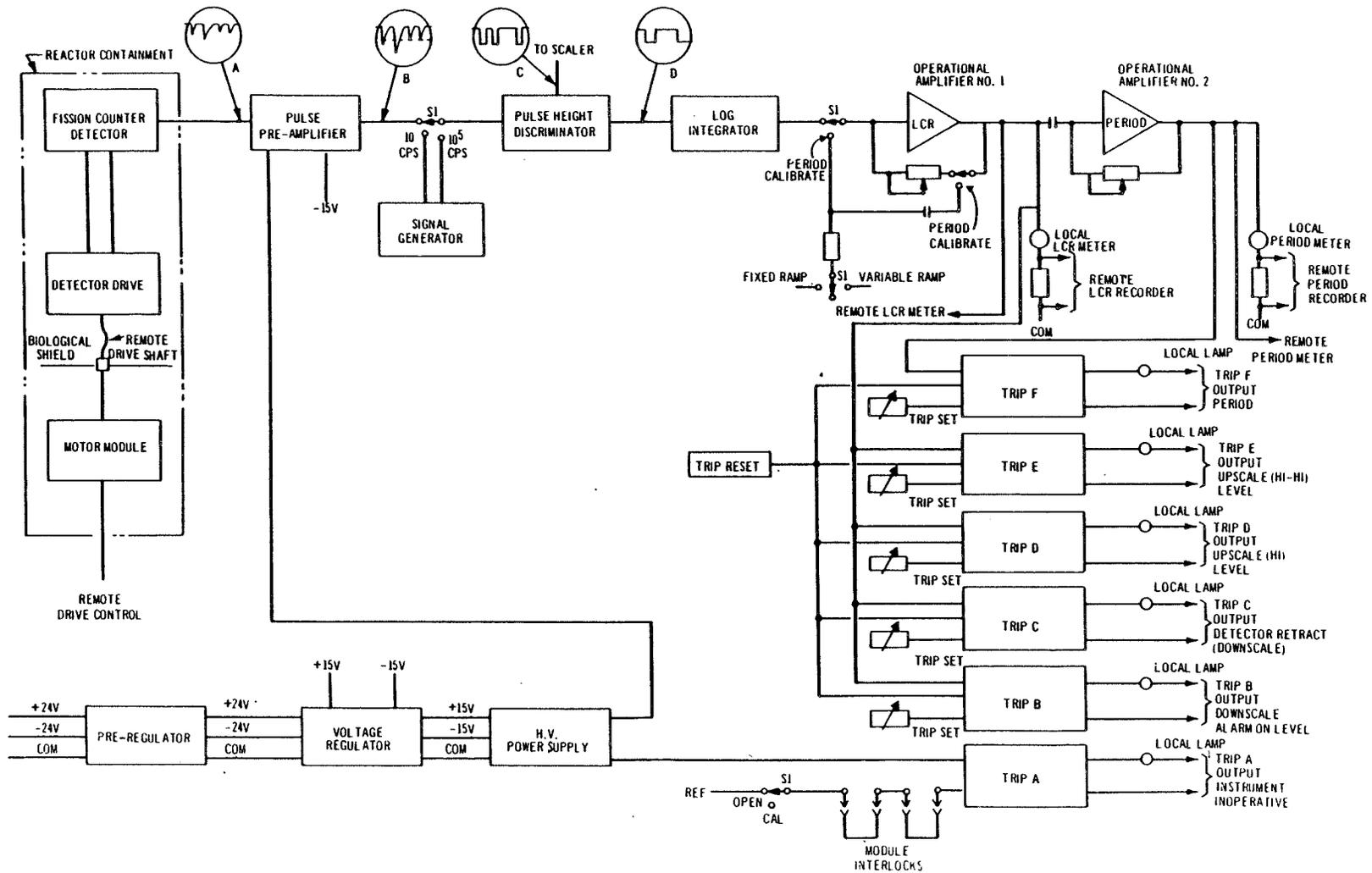
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FIGURE 7.6-16, SHEET 7
NEUTRON MONITORING SYSTEM FCD

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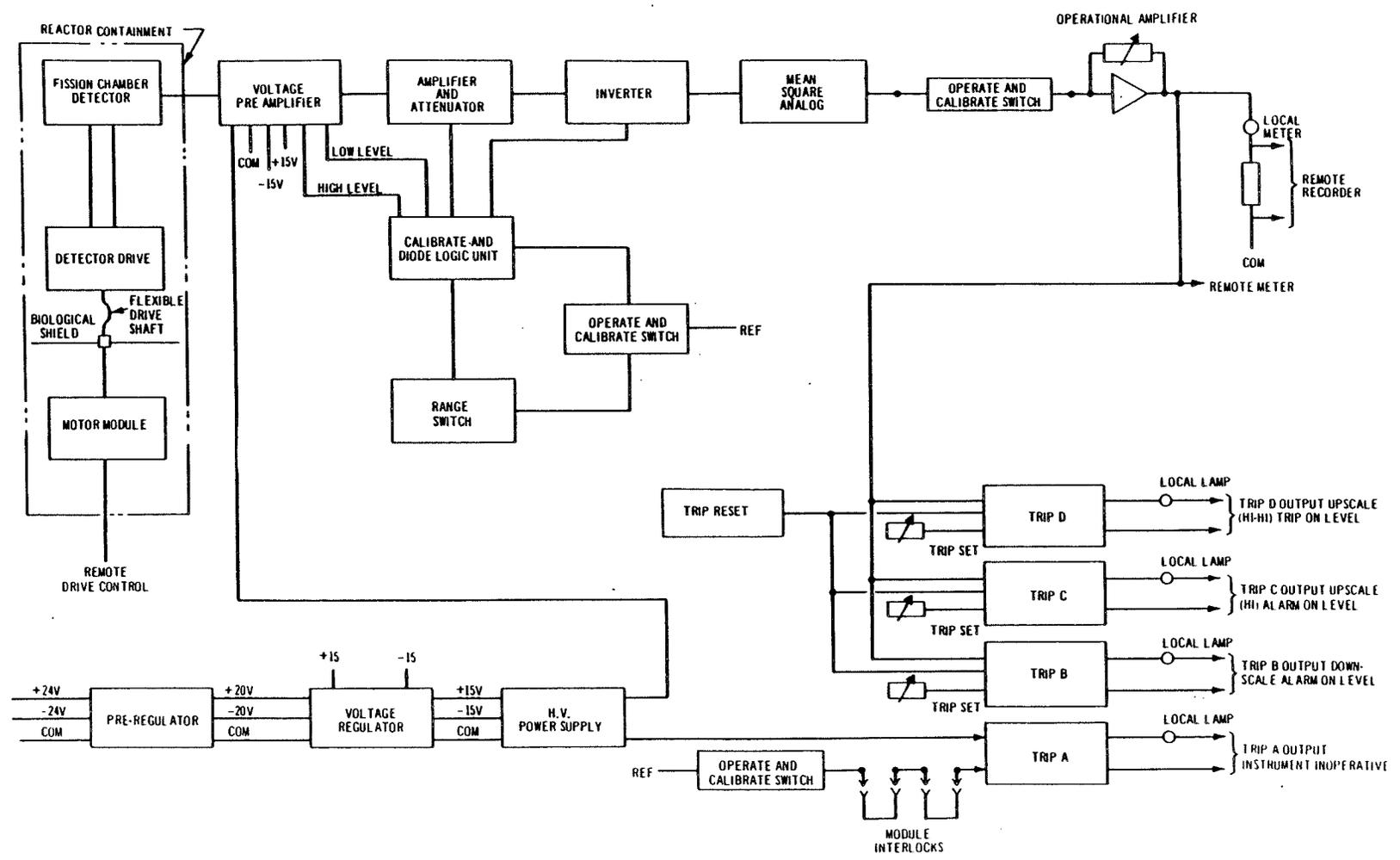
FIGURE 7.6-17

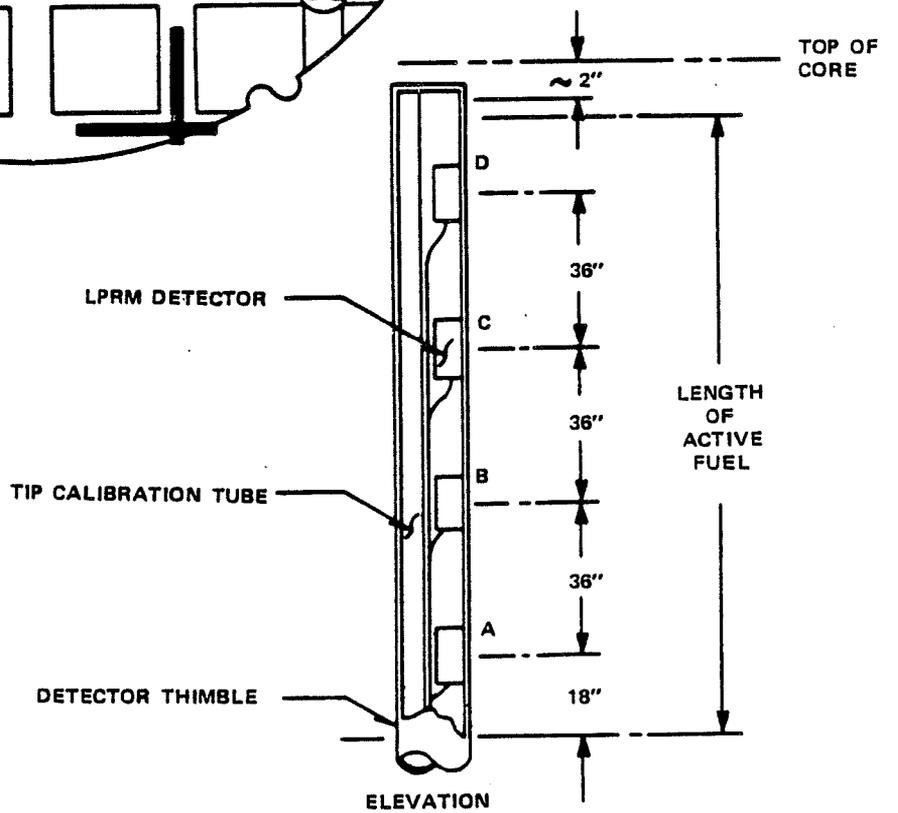
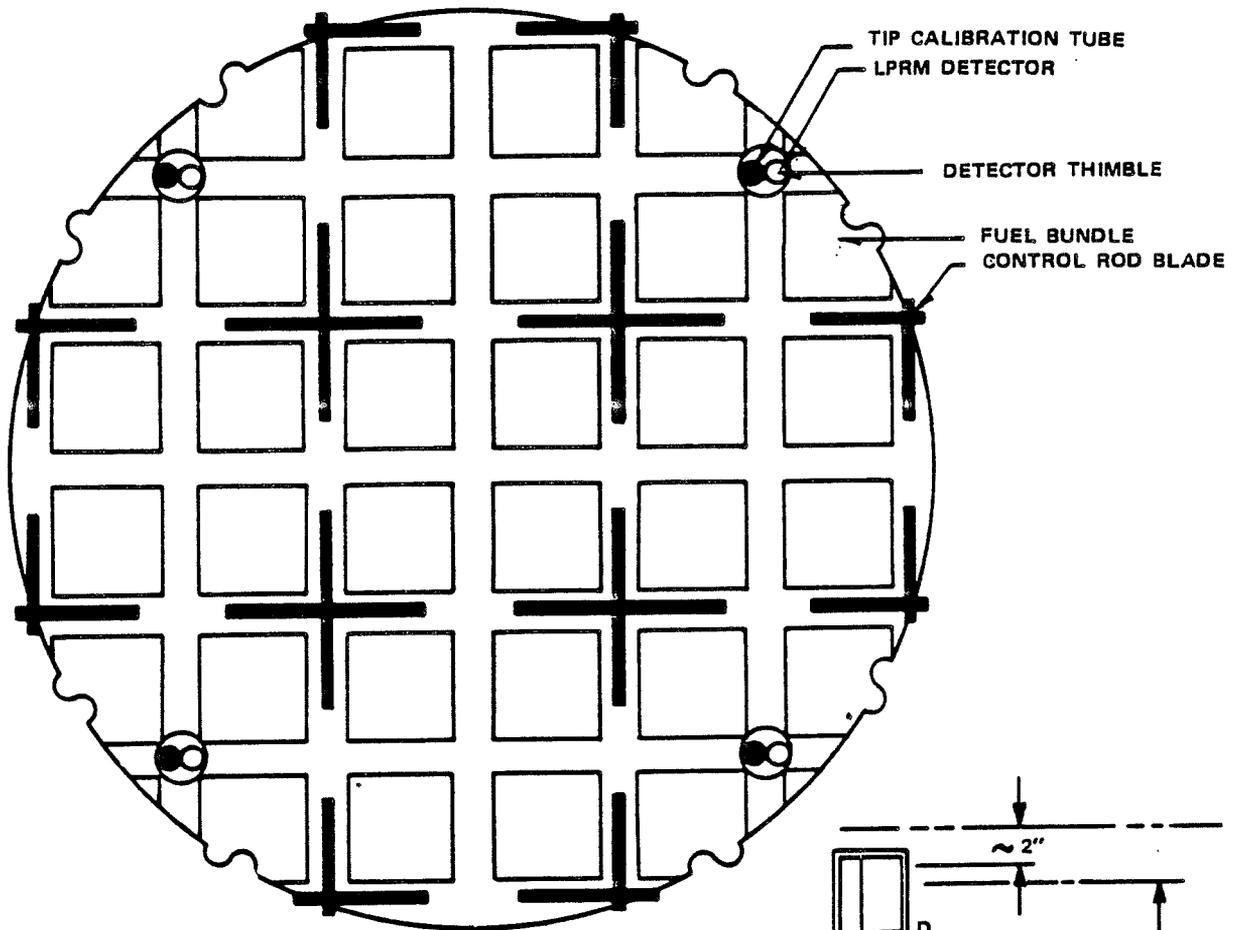
FUNCTIONAL BLOCK DIAGRAM OF SOURCE RANGE
 MONITOR CHANNEL



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 FUNCTIONAL BLOCK DIAGRAM OF INTERMEDIATE
 RANGE MONITOR CHANNEL

FIGURE 7.6-18



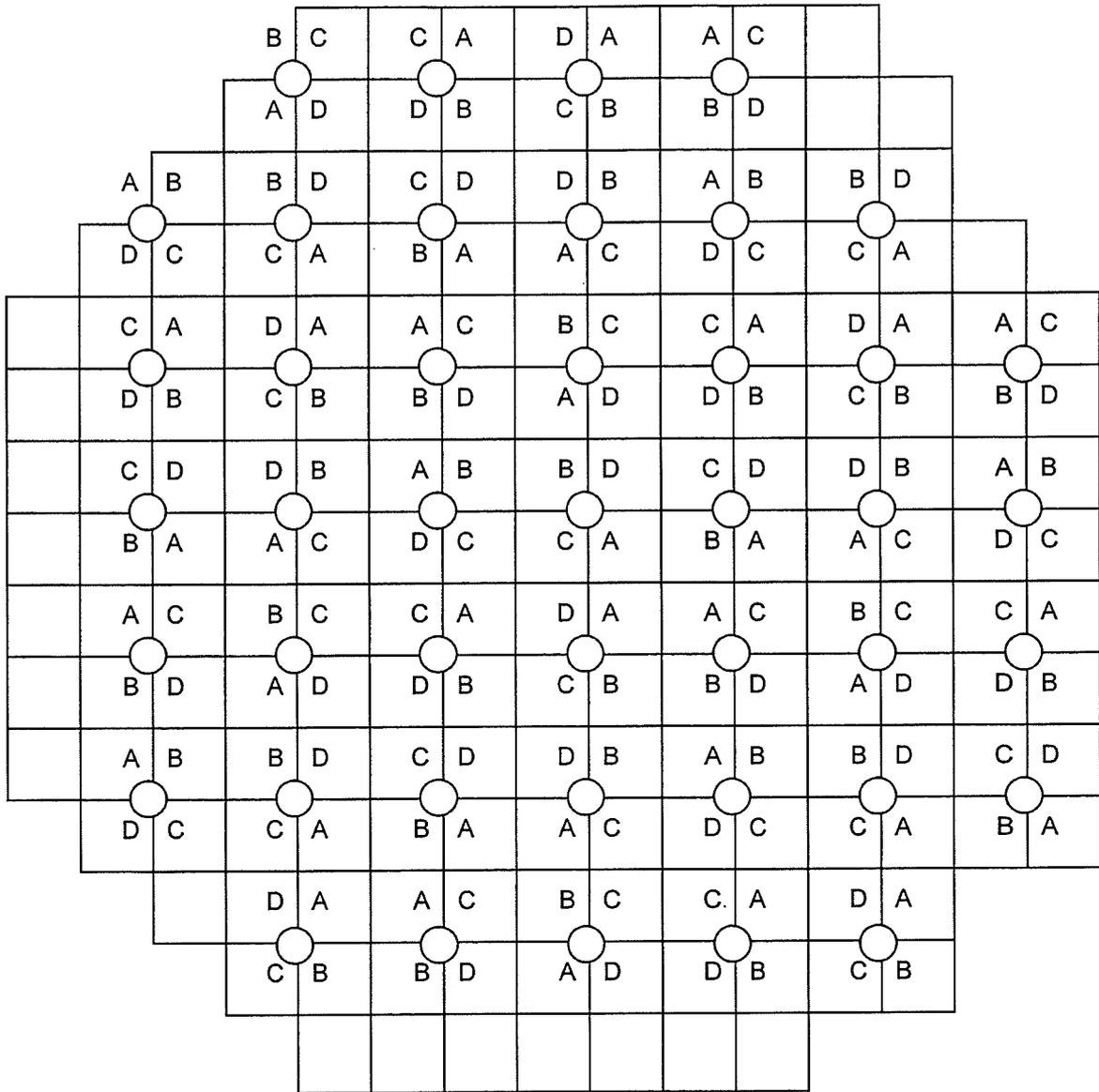


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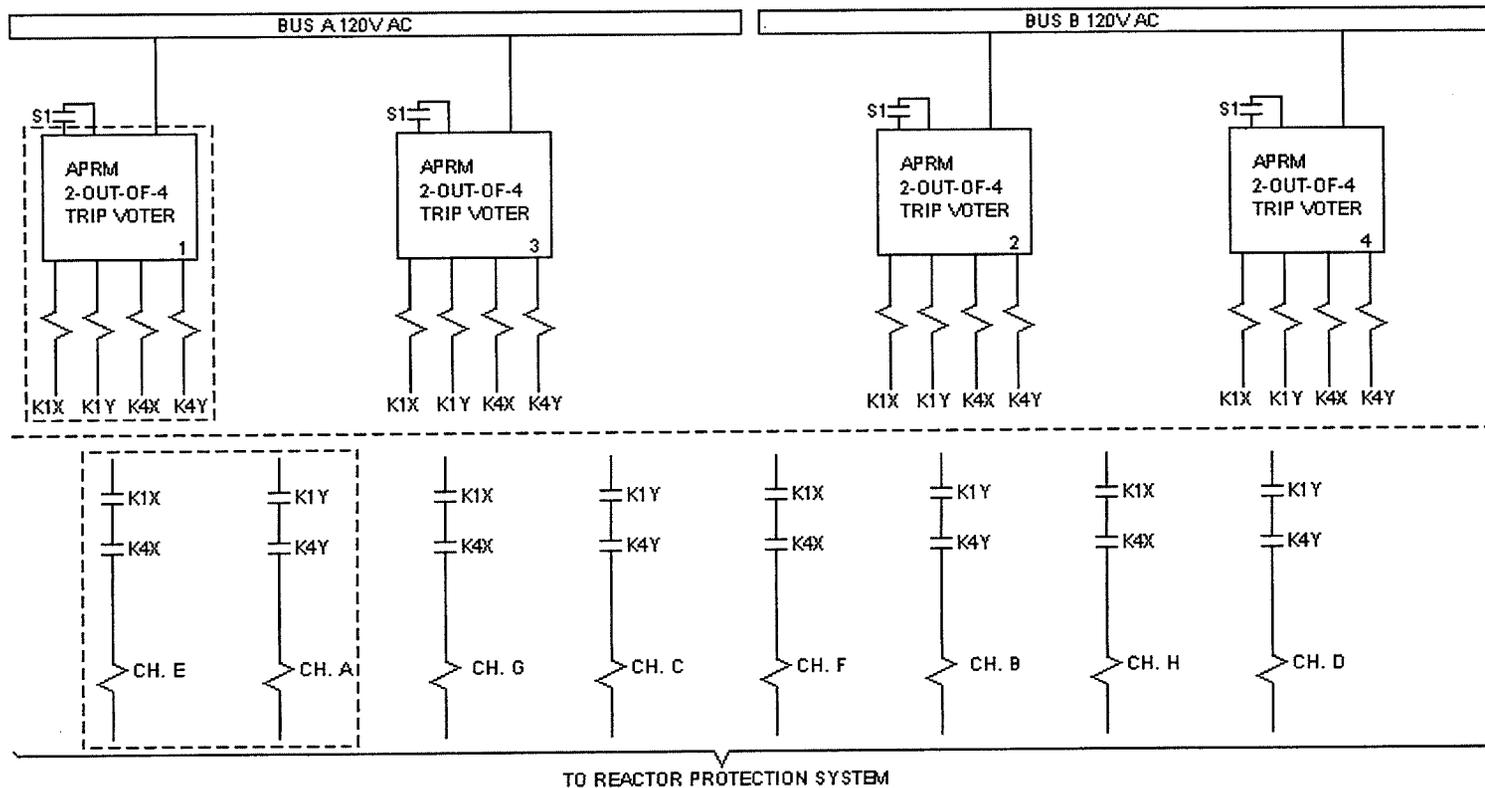
FIGURE 7.6-19

POWER RANGE MONITOR DETECTOR ASSEMBLY LOCATION



- CENTRAL CONTROL ROD
- UPPER LEFT LETTER IS LPRM CHAMBER ASSIGNED TO APRM 1
- UPPER RIGHT LETTER IS LPRM CHAMBER ASSIGNED TO APRM 2
- LOWER RIGHT LETTER IS LPRM CHAMBER ASSIGNED TO APRM 3
- LOWER LEFT LETTER IS LPRM CHAMBER ASSIGNED TO APRM 4

<p>Fermi 2</p> <p>UPDATED FINAL SAFETY ANALYSIS REPORT</p>
<p>FIGURE 7.6-20</p> <p>LOCAL POWER RANGE MONITOR TO AVERAGE POWER RANGE MONITOR ASSIGNMENT SCHEME</p>

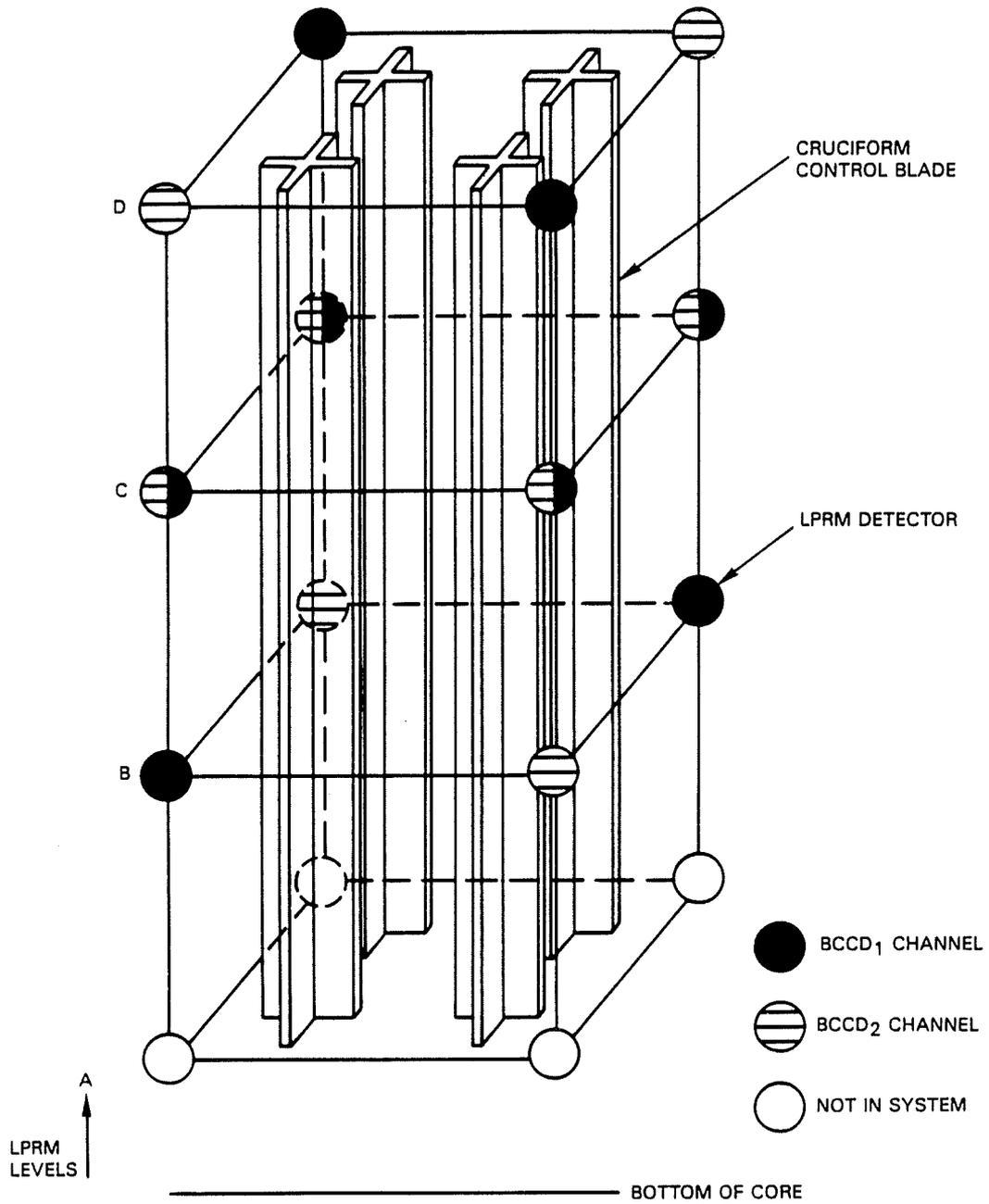


LEGEND:
 K1X & K1Y -- UPSCALE NEUTRON TRIP OR UPSCALE THERMAL TRIP OR INOP AND NOT BYPASSED
 K4X & K4Y -- OPRM TRIP AND NOT BYPASSED
 S1 -- REACTOR MODE SWITCH (CONTACT CLOSSES IN RUN MODE)

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FIGURE 7.6-21

TYPICAL AVERAGE POWER RANGE MONITOR
 CIRCUIT ARRANGEMENT FOR REACTOR
 PROTECTION SYSTEM INPUT

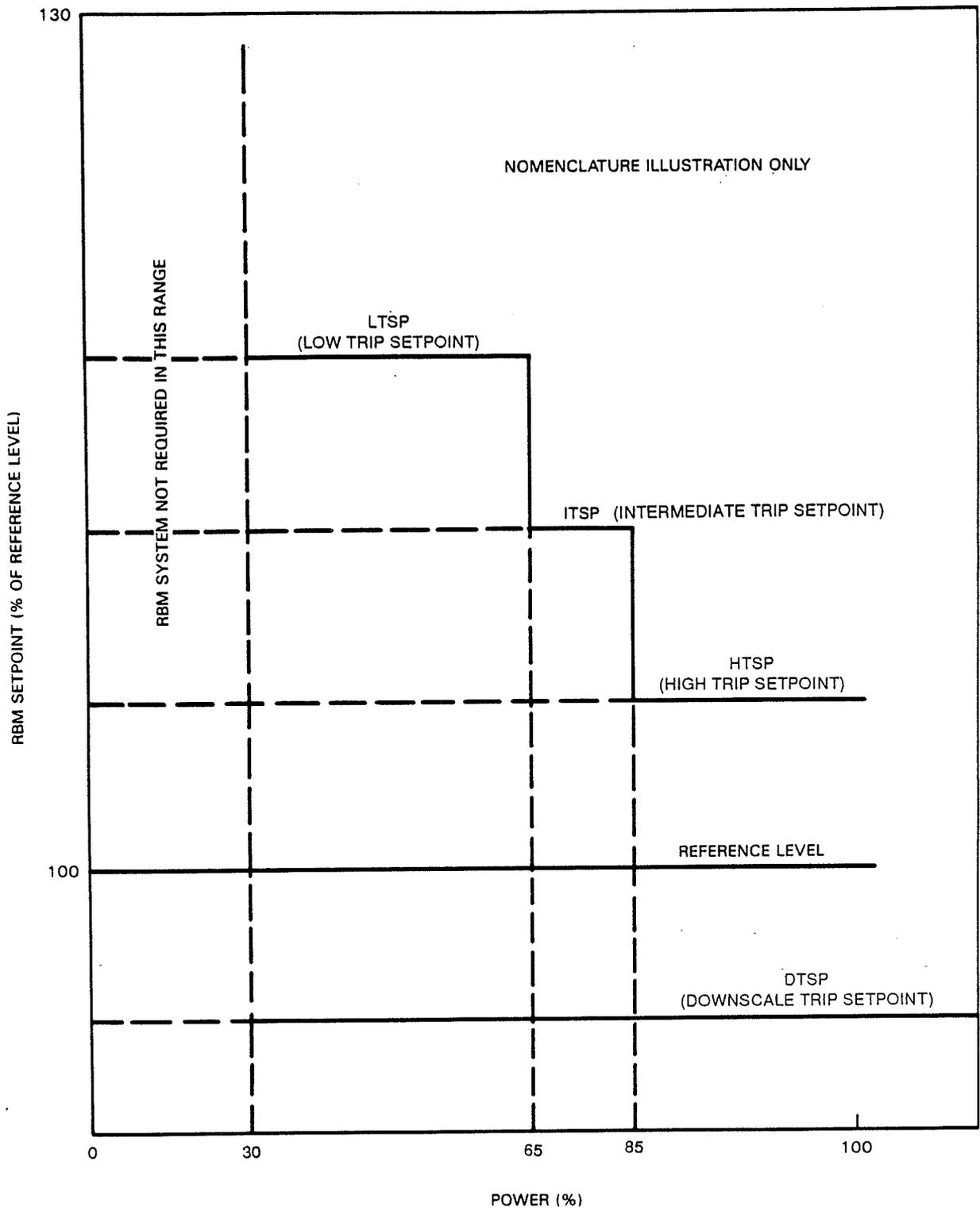


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FIGURE 7.6-22(a)

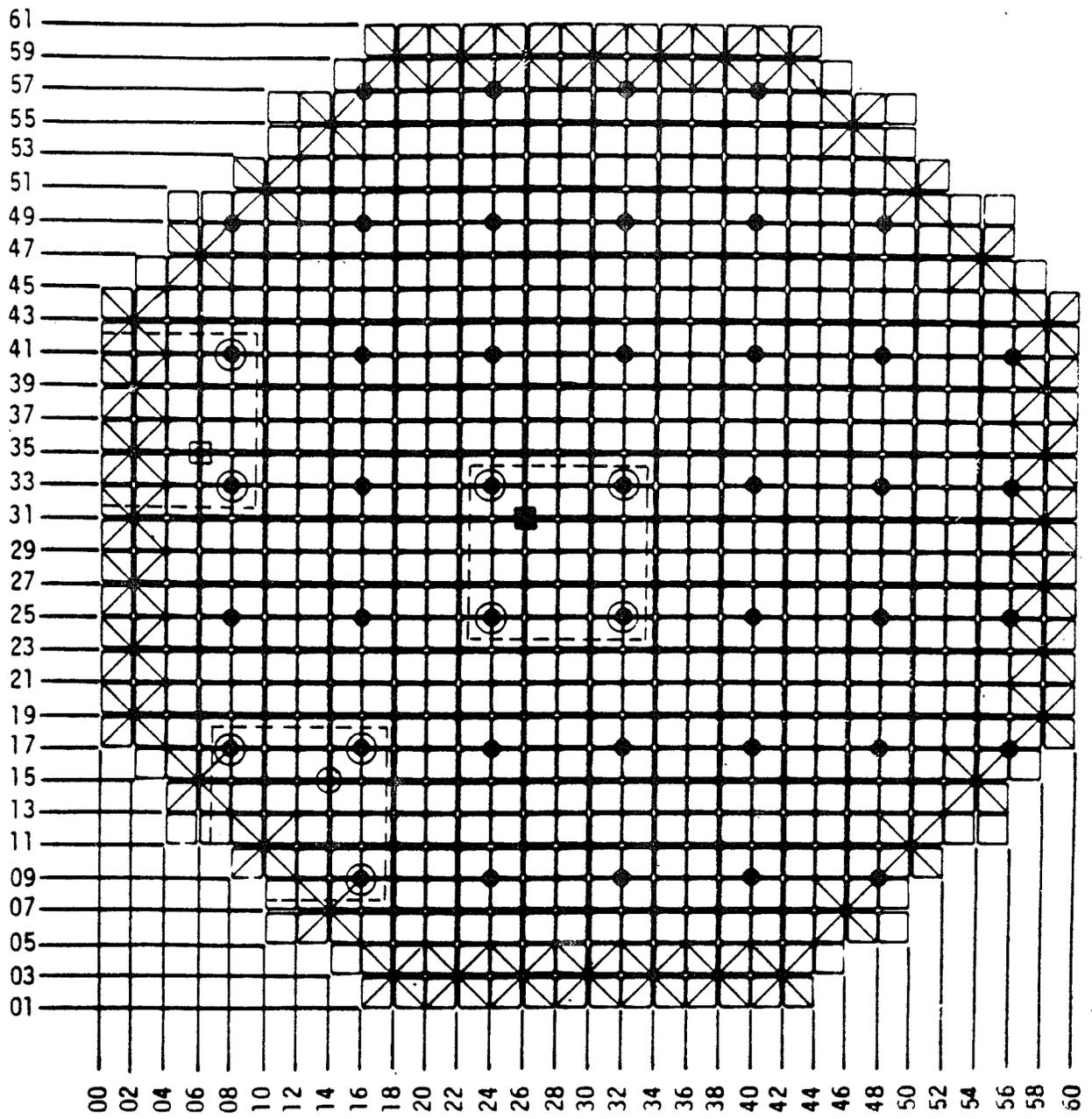
RBM BCCD₁/BCCD₂ LPRM ASSIGNMENT



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FIGURE 7.6-22(b)

TYPICAL POWER DEPENDENT
 RBM TRIP NOMENCLATURE



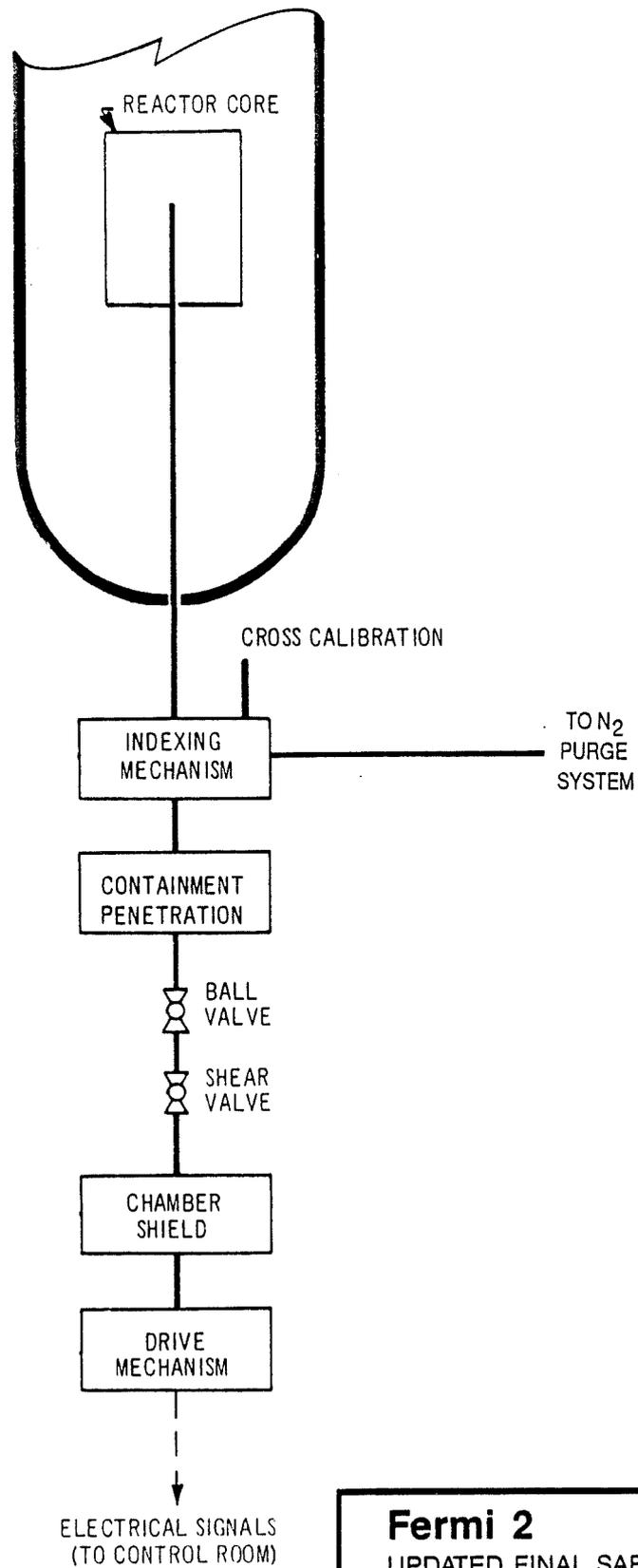
- LPRM ASSEMBLY ASSIGNED TO RBM
- ⊕ ROD SELECTION RESULTING IN TWO-ASSEMBLY ASSIGNMENT
- ⊗ ROD SELECTION RESULTING IN THREE-ASSEMBLY ASSIGNMENT
- ⊞ ROD SELECTION RESULTING IN FOUR-ASSEMBLY ASSIGNMENT
- * RBM AUTOMATICALLY BYPASSED (READING ZEROED)

NOTE: ASSIGNMENT IS AUTOMATICALLY INITIATED ON ROD SELECTION.

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FIGURE 7.6-22

ASSIGNMENT OF LOCAL POWER RANGE MONITOR
 ASSEMBLIES TO ROD BLOCK MONITORS

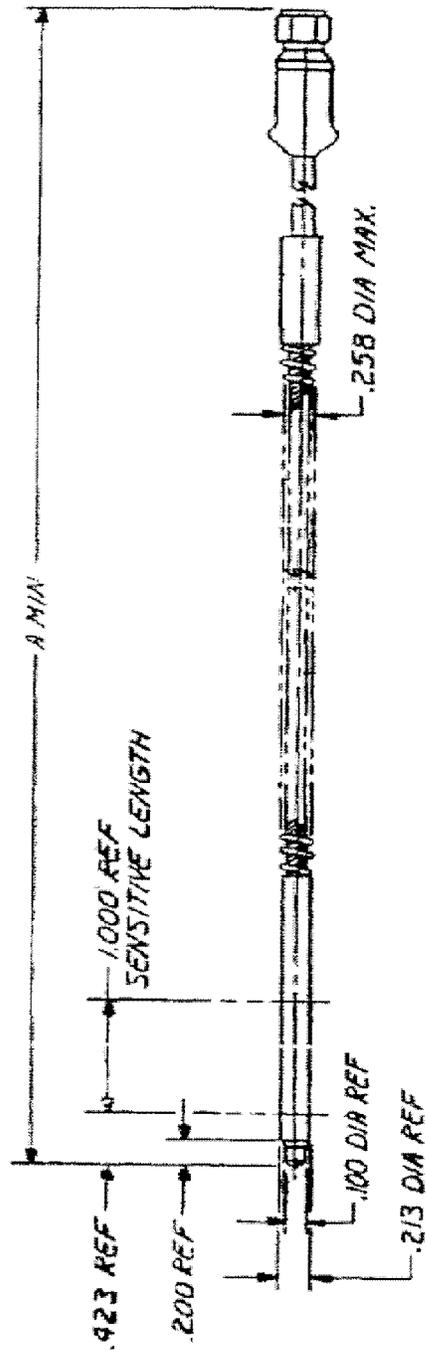


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FIGURE 7.6-23

TRAVERSING IN-CORE PROBE SUBSYSTEM
BLOCK DIAGRAM



PART	CAT. NO.	A - FEET
1	169CB9040001	130
2	169CB9040002	150
3	169CB9040003	170

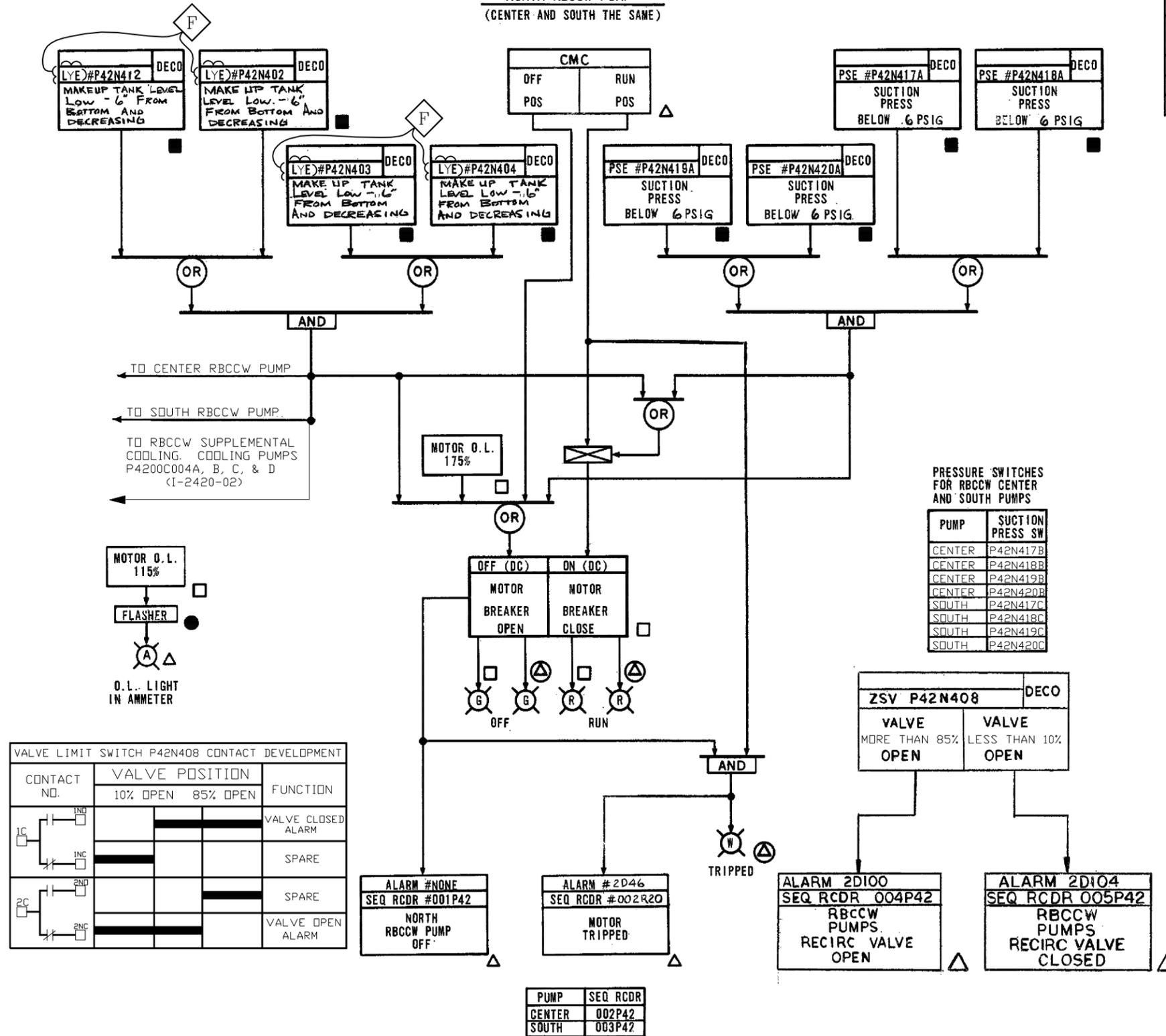
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FIGURE 7.6-24

TRAVERSING IN-CORE PROBE ASSEMBLY

NORTH RBCCW PUMP
(CENTER AND SOUTH THE SAME)



CONTACT NO.	VALVE POSITION		FUNCTION
	10% OPEN	85% OPEN	
1C	NO	NC	VALVE CLOSED ALARM
1C	NC	NO	SPARE
2C	NO	NC	SPARE
2C	NC	NO	VALVE OPEN ALARM

PUMP	SUCTION PRESS SW
CENTER	P42N417B
CENTER	P42N418B
CENTER	P42N419B
CENTER	P42N420B
SOUTH	P42N417C
SOUTH	P42N418C
SOUTH	P42N419C
SOUTH	P42N420C

PUMP	SEQ RCDR
CENTER	002P42
SOUTH	003P42

PUMP	MOTOR	FEEDER	PNL ITEM NO.	
	CODE NO	BUS NO.	POS	CMC SWITCH
NORTH	P4200C001	72C	4C	2B165
CENTER	P4200C002	72F	2D	2B166
SOUTH	P4200C003	72E	3B	2B167

LOCATION SYMBOLS

- △ IN CONTROL ROOM
- ⊖ IN CMC SWITCH
- IN RELAY ROOM
- AT CONTROLLED EQUIPMENT
- AT SWITCHGEAR

NOTE:

1. REFER TO DECO STANDARD DWG. 6M-2290, SHT. 2, MODE-A- FOR DESCRIPTION OF MOTOR CONTROL AND ASSOCIATED FUNCTIONS.

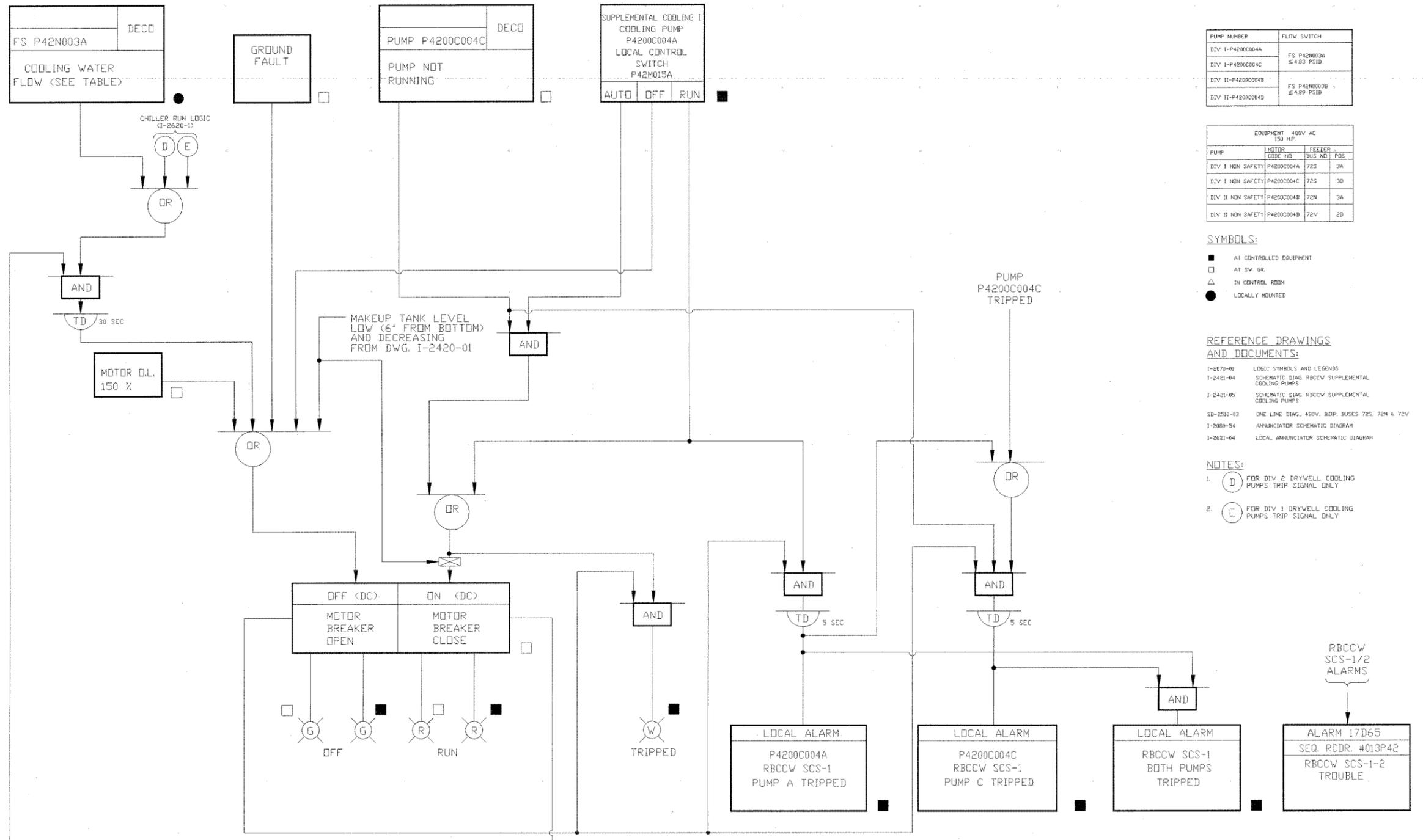
REFERENCE DWGS. & DOCUMENTS

- I-2070-01 LOGIC SYMBOLS & LEGENDS
- SD-2510-01 ONE LINE - 480VAC E.S.S. BUSS
- I-2421-01,02,03 SCHM. DIAG.-RBCCW PUMPS N, C, AND S
- M-2027 PIPING DIAG.-RBCCW & EECW SYSTEMS
- I-2005-01,02,03 ARRANGEMENT OF EQUIPMENT ON PANEL H11P602
- M-2001 LEGEND OF SYMBOLS & INSTR. IDENT. FOR PLANT SYSTEM DIAGRAMS
- I-2420-02, LOGIC DIAGRAM RBCCW SUPPLEMENTAL COOLING PUMPS

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FIGURE 7.6-25, SHEET 1

REACTOR BUILDING CLOSED COOLING WATER
 PUMP NORTH, CENTER, AND SOUTH LOGIC DIAGRAM



PUMP NUMBER	FLOW SWITCH
DIV I-P4200C004A	FS P42N003A ≤4.83 PSID
DIV I-P4200C004C	
DIV II-P4200C004B	FS P42N003B ≤4.89 PSID
DIV II-P4200C004D	

EQUIPMENT 480V AC 150 HP			
PUMP	MOTOR CODE NO	FEEDER BUS NO	FDS
DIV I NON SAFETY	P4200C004A	72S	3A
DIV I NON SAFETY	P4200C004C	72S	3D
DIV II NON SAFETY	P4200C004B	72N	3A
DIV II NON SAFETY	P4200C004D	72V	2D

SYMBOLS:

- AT CONTROLLED EQUIPMENT
- AT SW. GR.
- △ IN CONTRL. ROOM
- LOCALLY MOUNTED

REFERENCE DRAWINGS AND DOCUMENTS:

- I-2070-01 LOGIC SYMBOLS AND LEGENDS
- I-2421-04 SCHEMATIC DIAG. RBCCW SUPPLEMENTAL COOLING PUMPS
- I-2421-05 SCHEMATIC DIAG. RBCCW SUPPLEMENTAL COOLING PUMPS
- SD-2510-03 ONE LINE DIAG., 480V. BDP. BUSES 72S, 72N & 72V
- I-2080-54 ANNUNCIATOR SCHEMATIC DIAGRAM
- I-2621-04 LOCAL ANNUNCIATOR SCHEMATIC DIAGRAM

NOTES:

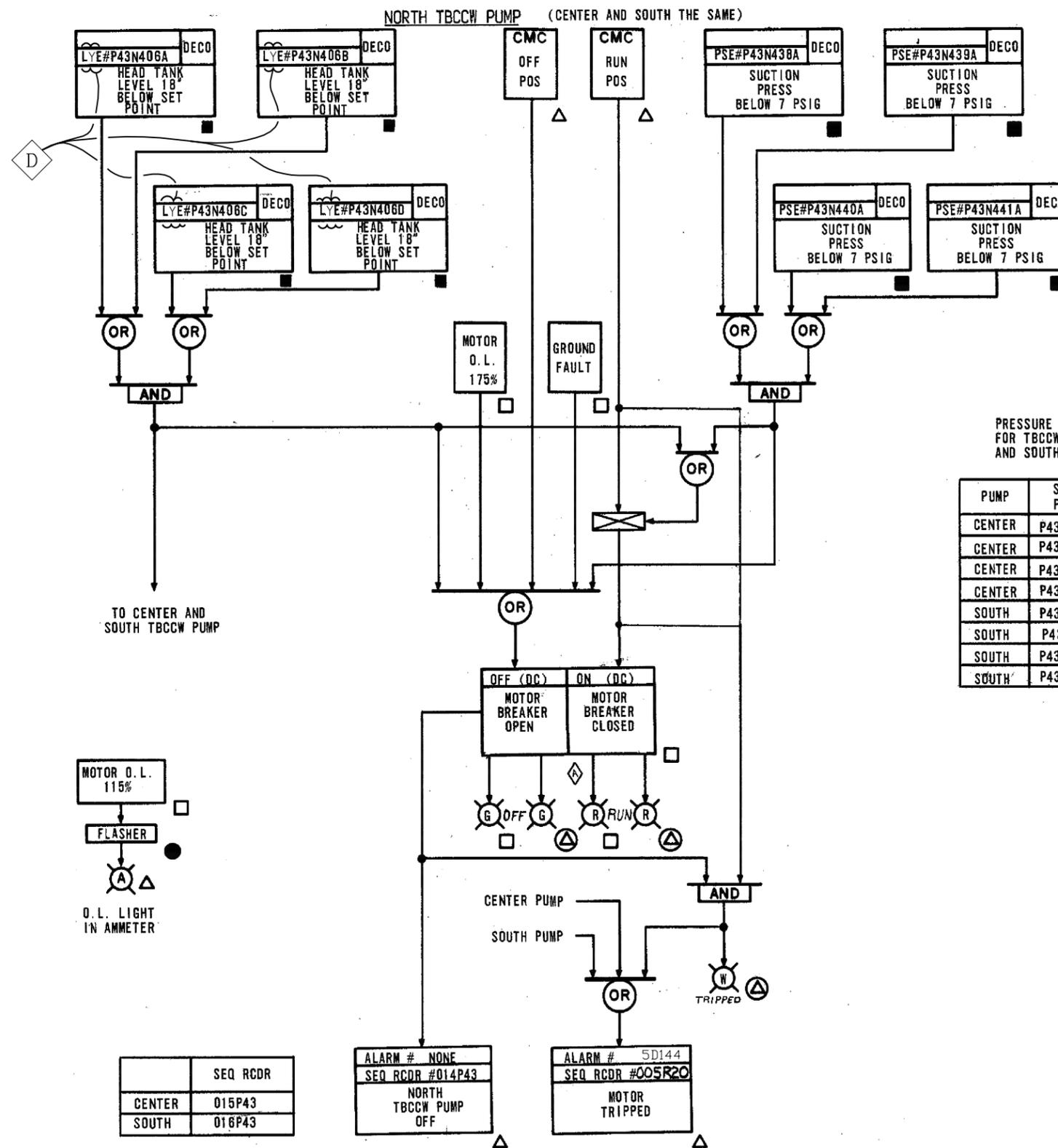
1. (D) FOR DIV 2 DRYWELL COOLING PUMPS TRIP SIGNAL ONLY
2. (E) FOR DIV 1 DRYWELL COOLING PUMPS TRIP SIGNAL ONLY

RBCCW SUPPLEMENTAL COOLING I COOLING PUMP P4200C004A

TYPICAL FOR PUMP P4200C004C AND DIV II PUMPS P4200C004B & C004D

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FIGURE 7.6-25, SHEET 2
RBCCW SUPPLEMENTAL COOLING PUMPS LOGIC DIAGRAM



ALARM NO.	SEQ RCDR NO.	ALARM DESCRIPTION	INIT DEVICE NO.	SETTING
5D1	012P43	TBCCW PUMPS RECIRC VALVE OPEN	P43N429	85% OPEN
5D2	013P43	TBCCW PUMPS RECIRC VALVE CLOSED	P43N429	10% OPEN

EQUIPMENT 480 VAC, 60 HP				
PUMP	MOTOR	FEEDER	PNL ITEM NO	
	CODE NO.	BUS NO.	POS	CMC SW.
NORTH	P4300C001	72M	2D	5854
CENTER	P4300C002	72N	3D	5856
SOUTH	P4300C003	72N	2D	5858

LOCATION SYMBOLS

- ▲ IN CONTROL ROOM
- ⊙ IN CMC SWITCH
- IN PUSHBUTTON
- IN RELAY ROOM
- AT CONTROLLED EQUIPMENT
- AT SWITCHGEAR

NOTE

1. REFER TO DECO STANDARD DWG 6M-2290 SHT 2, MODE "A" FOR DESCRIPTION OF MOTOR CONTROL AND ASSOCIATED FUNCTIONS.

PRESSURE SWITCHES FOR TBCCW CENTER AND SOUTH PUMPS

PUMP	SUCTION PRESS SW
CENTER	P43N438B
CENTER	P43N439B
CENTER	P43N440B
CENTER	P43N441B
SOUTH	P43N438C
SOUTH	P43N439C
SOUTH	P43N440C
SOUTH	P43N441C

REFERENCE DWGS & DOCUMENTS

- I-2070-01 LOGIC SYMBOLS & LEGENDS
- I-2431-01,02 SCHM DIAG TBCCW PUMPS N,C & S
- M-2008 TBCCW SYSTEM DIAGRAM
- I-2022-01,02,03 ARRANGEMENT OF EQUIPMENT ON PANEL H1P805
- M-2001 LEGEND OF SYMBOLS & INSTR IDENT FOR PLANT SYSTEM DIAGRAMS
- CECO CENTRAL COMPONENT DATABASE
- DECO DESIGN INSTR #22 REV A, TBCCW SYSTEM

	SEQ RCDR
CENTER	015P43
SOUTH	016P43

ALARM # NONE
SEQ RCDR #014P43
NORTH TBCCW PUMP OFF

ALARM # 5D144
SEQ RCDR #005R20
MOTOR TRIPPED

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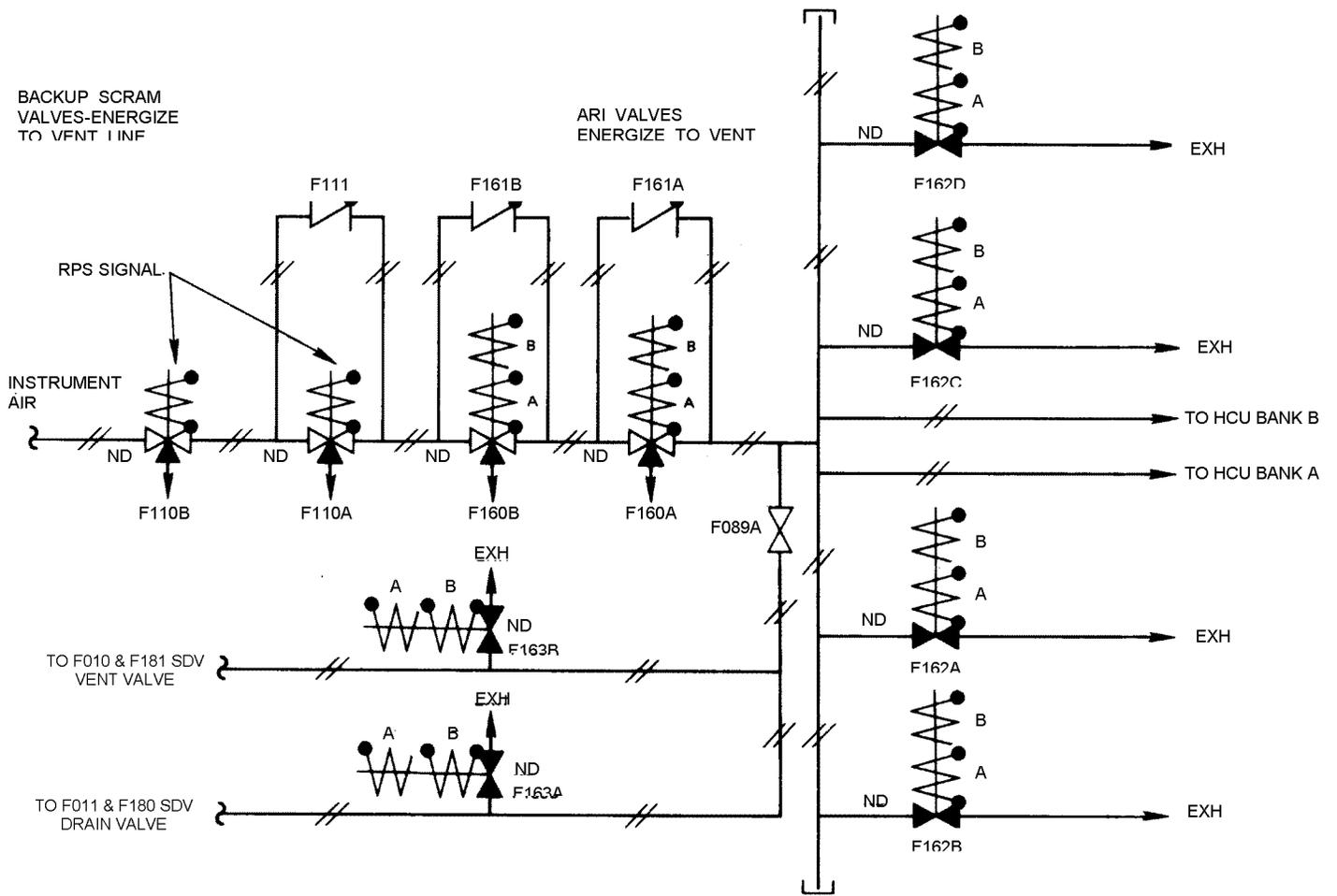
FIGURE 7.6-26

TURBINE BUILDING CLOSED COOLING WATER PUMP NORTH, CENTER, AND SOUTH LOGIC DIAGRAM

FIGURES 7.6-27 THROUGH 7.6-35
ARE INTENTIONALLY DELETED

BACKUP SCRAM
VALVES-ENERGIZE
TO VENT LINE

ARI VALVES
ENERGIZE TO VENT

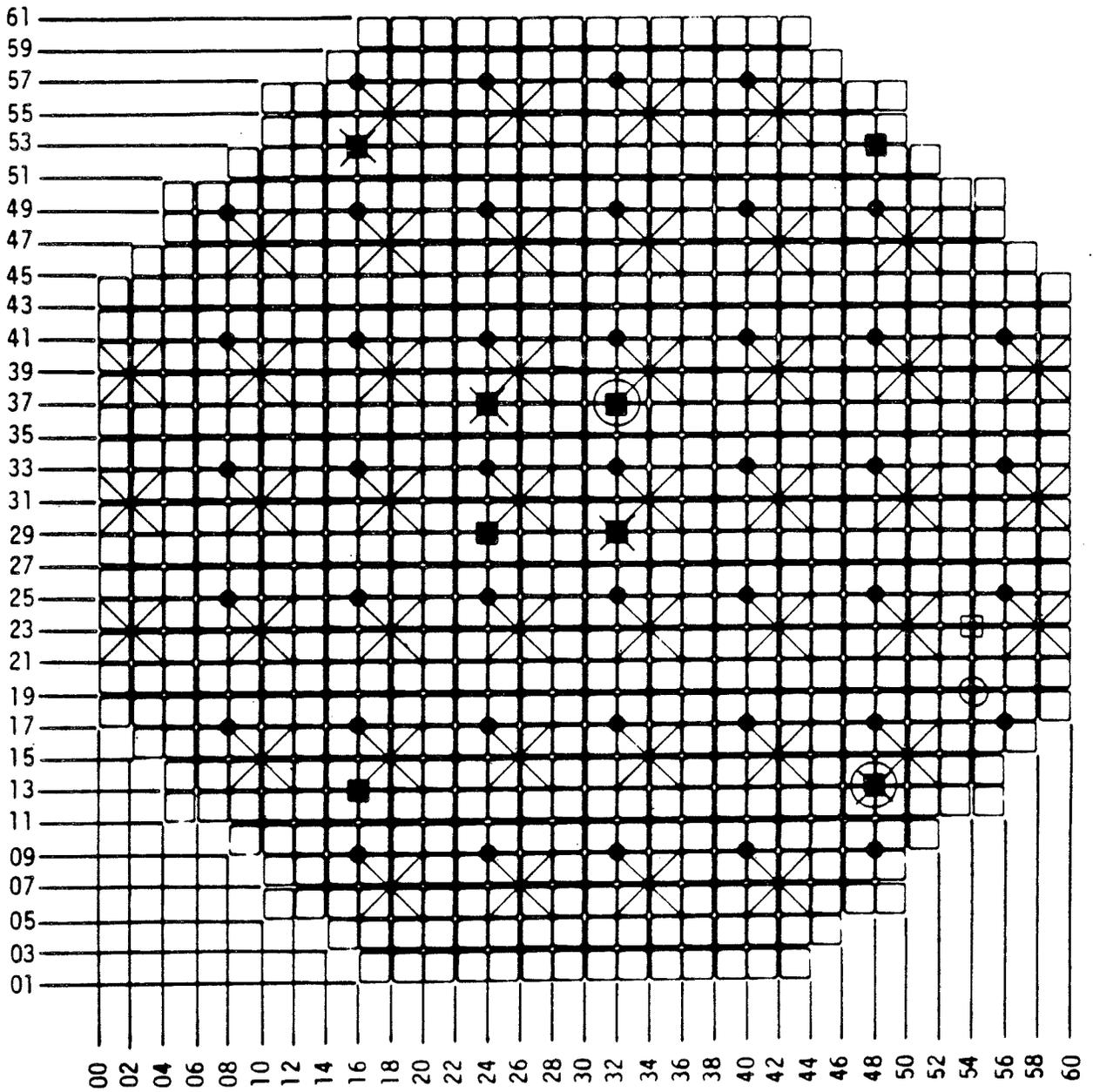


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FIGURE 7.6-36

ALTERNATE ROD INSERTION VALVE
ARRANGEMENT



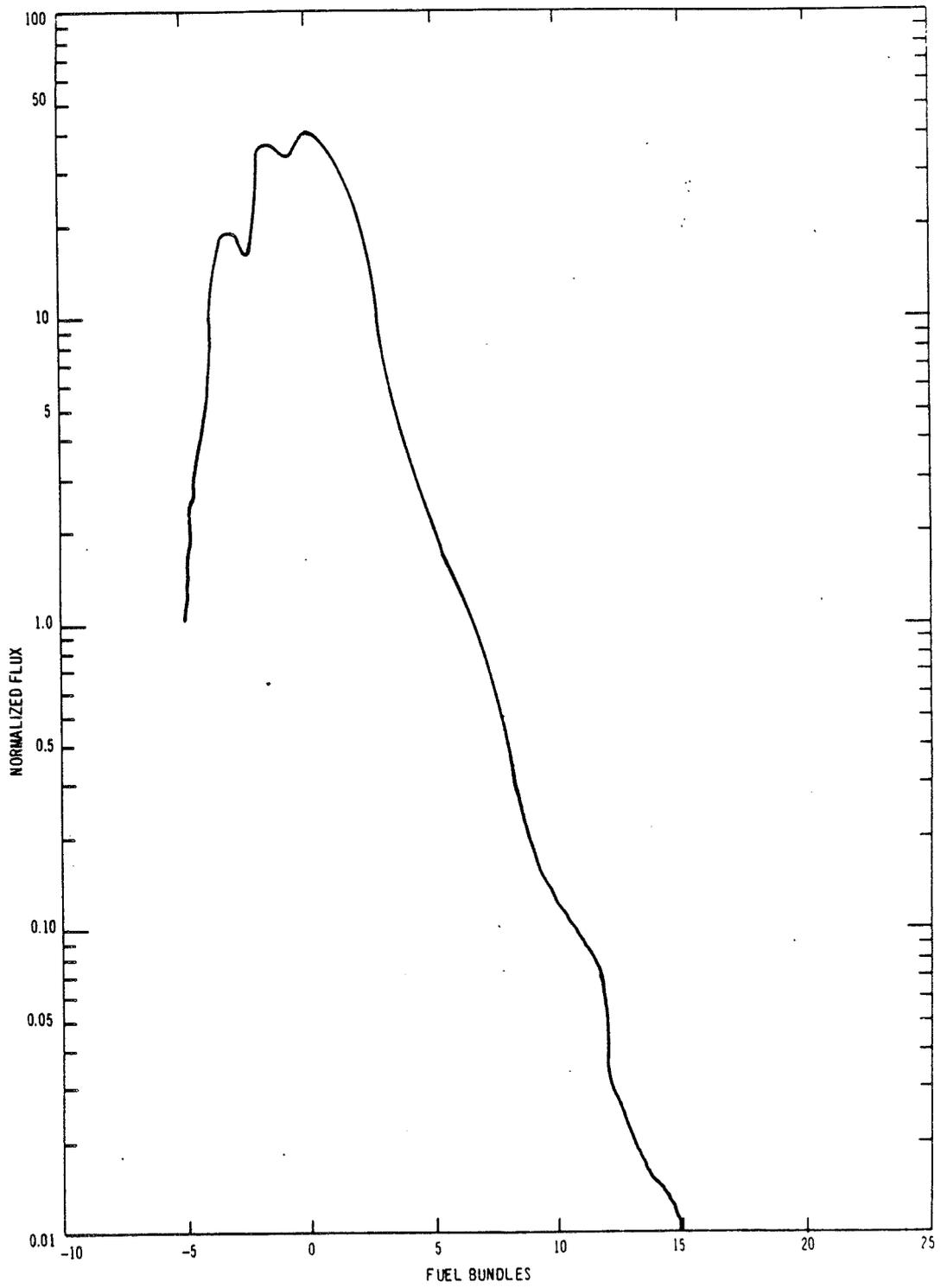
- | | | | |
|---|----------------------------|---|--|
|  | IRM DETECTOR TRIP SYSTEM A |  | NEXT CONTROL ROD WITHDRAWN IN SEQUENCE |
|  | IRM DETECTOR TRIP SYSTEM B |  | CONTROL ROD WITHDRAWN OUT OF SEQUENCE |
|  | CONTROL ROD WITHDRAWN |  | IRM DETECTOR BYPASSED |

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FIGURE 7.6-37

CONTROL ROD WITHDRAWAL ERROR



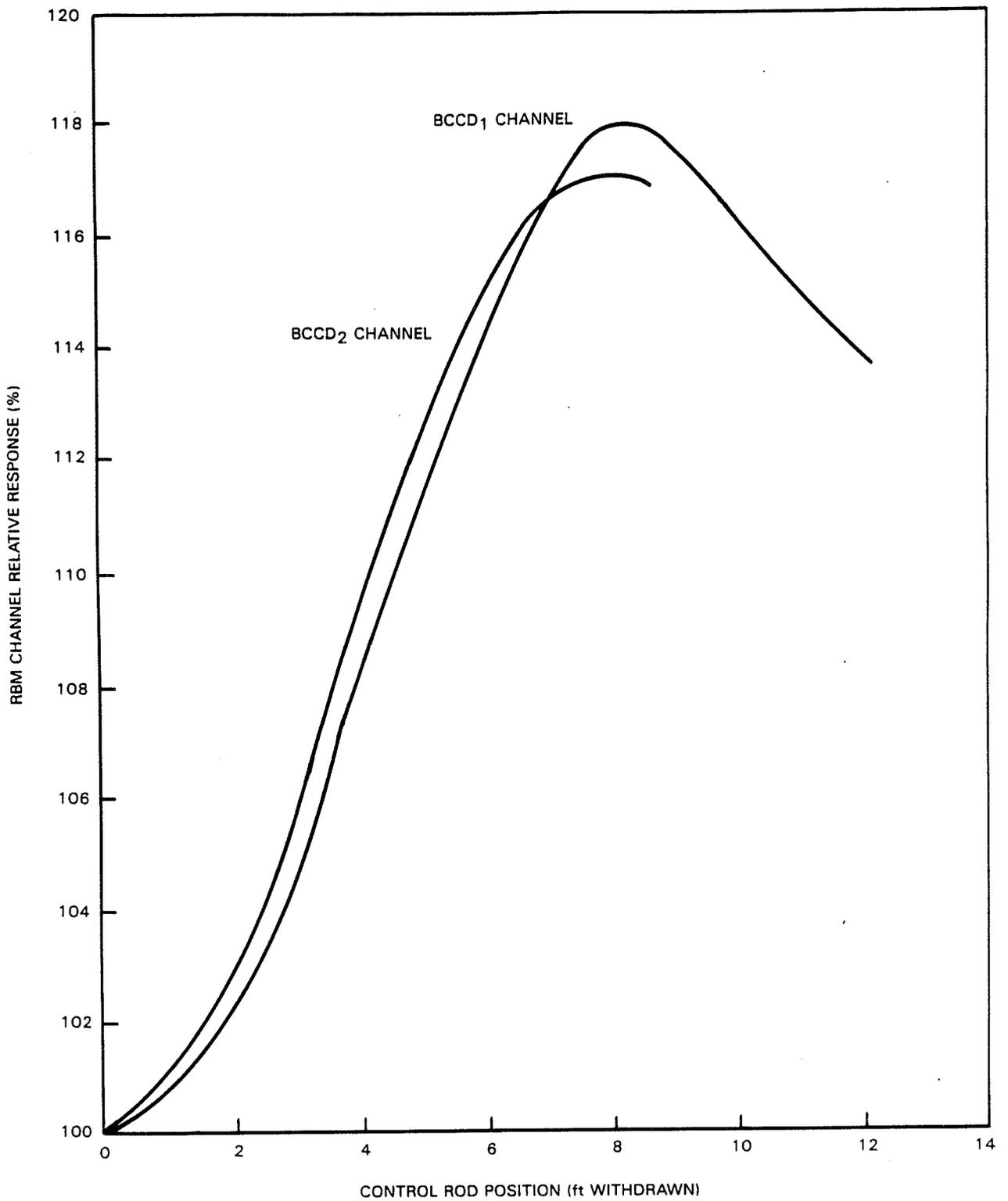
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FIGURE 7.6-38

NORMALIZED FLUX DISTRIBUTION FOR ROD
WITHDRAWAL ERROR

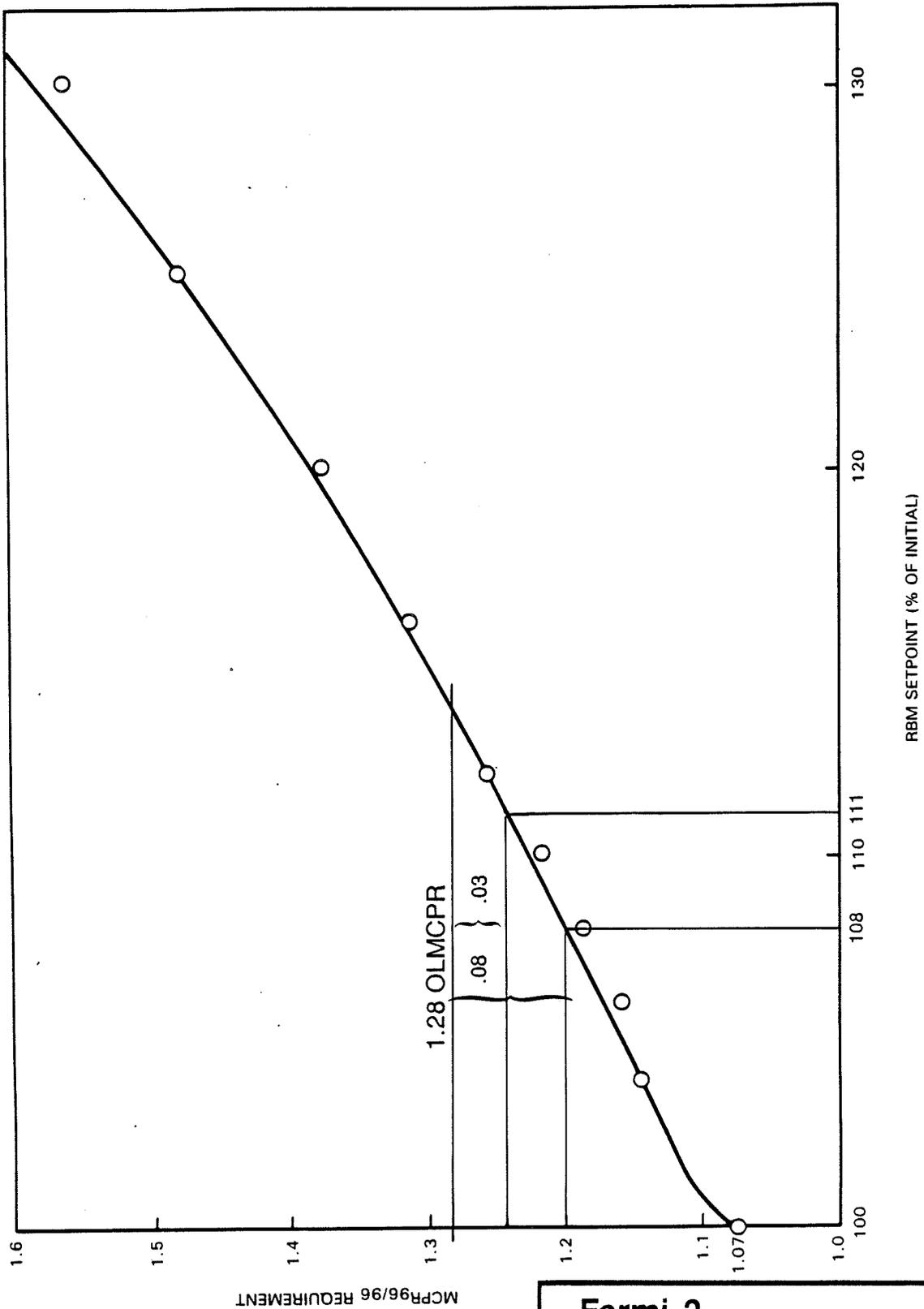
FIGURES 7.6-39 THROUGH 7.6-40
ARE INTENTIONALLY DELETED



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FIGURE 7.6-41

TYPICAL RBM CHANNEL RESPONSES,
 LPRM ASSIGNMENT, NO FAILED LPRMs



MCFR96/96 REQUIREMENT

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FIGURE 7.6-42

TYPICAL DESIGN BASIS RWE
MCFR REQUIREMENT
VERSUS RBM SETPOINT

7.7 CONTROL SYSTEMS NOT REQUIRED FOR SAFETY

7.7.1 Description

This subsection discusses control systems whose functions are not essential for the safety of the plant. These systems are the reactor manual control system (RMCS), recirculation flow control system (RFCS), feedwater control system, pressure regulator and turbine-generator controls, and the radwaste processing system controls.

7.7.1.1 Reactor Manual Control System Instrumentation and Control

7.7.1.1.1 Identification

The RMCS instrumentation and control consists of the electrical circuitry, switches, indicators, and alarm devices provided for operational manipulation of the control rods and the surveillance of associated equipment. This system includes the interlocks that inhibit rod movement (rod block) under certain conditions. The RMCS does not include any of the circuitry or devices used to automatically or manually scram the reactor; these devices are discussed in Sections 7.2 and 7.6. In addition, the mechanical devices of the control rod drive (CRD) and the CRD hydraulic system are not included in the RMCS. The latter mechanical components are described in Subsection 4.5.2.

7.7.1.1.1.1 Function

The objective of the RMCS is to provide the operator with the means for changing reactor power by manipulating the control rods.

7.7.1.1.1.2 Classification

This system is a power generation system, nonessential for safety, and is classified in Chapter 3.

7.7.1.1.2 Power Sources

7.7.1.1.2.1 Normal

The RMCS receives its power from the 120-V ac instrumentation and control power buses, either bus A or bus B. Each of these buses receives its normal power supply from the appropriate 480-V ac engineered safety feature (ESF) bus as described in Subsection 8.3.1. One subsystem, the control rod position indication system, is powered by the 120-V ac instrument bus, as described in Subsection 8.3.1.

7.7.1.1.2.2 Alternate

On loss of normal auxiliary power, the station diesel generator provides backup power to the 480-V ac ESF bus and the 120-V ac instrument bus.

7.7.1.1.3 Equipment Design

7.7.1.1.3.1 General

Figure 4.5-15 shows the layout of the CRD hydraulic system. Figure 7.7-1 shows the functional arrangement of devices for the control of components in the CRD hydraulic system. Although the figures also show the arrangement of scram devices, these devices are not part of the RMCS.

Control rods are moved by admitting water, under pressure, from a CRD water pump into the appropriate end of the CRD cylinder. The pressurized water forces the piston, which is attached by a connecting rod to a control rod, to move.

Three modes of control rod operation are used: insert, withdraw, and settle. Four solenoid-operated valves are associated with each control rod to accomplish the actions required for the operational modes. The valves control the path the CRD water takes to the cylinder. The RMCS controls the valves.

The settle mode of control rod operation is provided to decelerate the control rod at the end of either an insert cycle or a withdraw cycle. The settle action smooths out the control rod movement and prolongs the life of the CRD hydraulic system components. During the settle mode, the withdraw valve associated with the settle operation is opened or remains open while the other three solenoid-operated valves are closed. During an insert cycle, the settle action vents the pressure from the insert drive water supply line to the exhaust header, thus gradually reducing the differential pressure across the drive piston of the selected rod. During a withdraw cycle, the settle action holds open the discharge path for withdraw water while the withdraw drive water supply is shut off. This also allows for a gradual reduction in the differential pressure across the control drive piston. After the control rod has slowed down, the collet fingers engage the index tube and lock the rod in position.

The arrangement of control rod selection pushbuttons and circuitry permits the selection of only one control rod at a time for movement. A rod is selected for movement by depressing a button for the desired rod on the reactor control benchboard in the main control room (Figure 7.5-1).

The direction in which the selected rod moves is determined by the position of a switch, called the "rod movement" switch, which is also located on the reactor control benchboard. This switch has "rod-in" and "rod-out-notch" positions and returns by spring action to the "off" position. The rod selection circuitry is arranged so that a rod selection is sustained until the movement cycle of the selected rod has been completed. Reversion to the no-rod-selected condition is not possible, except for loss of control-circuit power until any moving rod has completed the movement cycle.

7.7.1.1.3.2 Insert Cycle

The following is a description of the detailed operation of the RMCS during an insert cycle. The response of a selected rod when the various commands are transmitted has been explained in Subsection 7.7.1.1.3.1. Figure 7.7-1 can be used to follow the sequence of an insert cycle.

A three-position rod movement switch is provided on the reactor control benchboard. The switch has a "rod-in" position, a "rod-out-notch" position, and an "off" position. The switch

returns by spring action to the "off" position. When a control rod is selected for movement, the operator places the rod movement switch in the "rod-in" position and then releases the switch. This action energizes the insert command for a limited time. Just before the insert command is removed, the settle command is automatically energized for a limited time. The insert command time setting and the rate of drive water flow provided by the CRD hydraulic system determine the distance traveled by a rod. The time setting results in a one-notch (6 in.) insertion of the selected rod for each momentary application of a "rod-in" signal from the rod movement switch. Continuous insertion of a selected control rod is possible by holding the rod movement switch in the "rod-in" position.

A second switch can be used to initiate insertion of a selected control rod. This switch is the "rod-out-notch-override" switch and is called the RONOR switch. The RONOR switch has three positions: "emergency-in," "notch override," and "off." The switch returns to the "off" position by spring action. By holding the RONOR switch in the "emergency-in" position, the logic maintains the insert command in a continuously energized state to cause continuous insertion of the selected control rod.

7.7.1.1.3.3 Withdraw Cycle

This subsection describes the detailed operation of the RMCS during a withdraw cycle. The response of a selected rod when the various commands are transmitted has been explained in Subsection 7.7.1.1.3.1. Figure 7.7-1 can be used to follow the sequence of a withdraw cycle.

When a control rod is selected for movement, the operator places the rod movement switch in the "rod-out-notch" position, which energizes the insert commands for a short time. Energizing the insert command at the beginning of the withdraw cycle is necessary to allow the collet fingers to disengage the index tube.

When the insert command is deenergized, the withdraw and settle commands are energized for a controlled period of time. The withdraw command is deenergized before the settle command; this tends to decelerate the selected rod. When the settle command is deenergized, the withdraw cycle is complete. This withdraw cycle is the same whether the rod movement switch is held continuously in the "rod-out-notch" position or is released. The timer that controls the withdraw cycle is set so that the rod travels one notch (6 in.) per cycle. Provisions are included to prevent further control rod motion in the event of timer failure.

A selected control rod can be continuously withdrawn if the rod movement switch is held in the "rod-out-notch" position at the same time that the RONOR switch is held in the "notch-override" position. When both switches are held in these positions, the withdraw and settle commands are continuously energized.

7.7.1.1.3.4 Control Rod Drive Hydraulic System Control

A motor-operated pressure control valve, two air-operated flow control valves, and four solenoid-operated stabilizer valves are included in the CRD hydraulic system to maintain smooth and regulated system operation. These devices are shown in Figure 4.5-15. The motor-operated pressure-control valve is positioned by manipulating a pushbutton in the main control room. The pushbuttons for this valve are located close to the pressure indicator that responds to the pressure changes caused by the movements of the valve. The air-operated flow control valves are automatically positioned in response to signals from an

upstream flow measuring device. The stabilizer valves are automatically controlled by the energization of the insert and withdraw commands. The control scheme is shown in Figure 7.7-1.

There are two drive-water pumps, one of which is a spare. They are controlled by switches in the main control room. Each pump automatically stops on indication of low suction pressure.

7.7.1.1.3.5 Rod Block Interlocks

General

Figure 7.7-1 shows the general functional arrangement of the rod block interlocks used in the RMCS.

To achieve an operationally desirable performance objective where most failures of individual components would be easily detected or would not disable the rod movement inhibiting functions, the rod block logic circuitry is arranged as two similar logic circuits. These circuits are energized when control rod movement is allowed. Each logic circuit receives input trip signals from a number of trip channels, and each logic circuit can provide a separate rod block signal to inhibit rod withdrawal.

The rod block circuitry is effective in preventing rod withdrawal, if required, during both normal (notch) withdrawal and continuous withdrawal. If a rod block signal is received during a rod withdrawal, the control rod is automatically stopped at the next notch position, even during a continuous rod withdrawal.

The components used to initiate rod blocks in combination with refueling operations provide rod block trip signals to these same rod block circuits. These refueling rod blocks are described in Subsection 7.6.1.1.

Rod Block Functions

The following discussion describes the various rod block functions and explains the intent of each function. The instruments used to sense the conditions for which a rod block is provided are discussed in Subsection 7.6.1.13. The rod block functions provided specifically for refueling situations are described in Subsection 7.6.1.1.

With the mode switch in the SHUTDOWN position, no control rod can be withdrawn. This enforces compliance with the intent of the shutdown mode.

The circuitry is arranged to initiate a rod block, regardless of the position of the mode switch, for the following conditions:

- a. Any average power range monitor (APRM) upscale rod block alarm. The purpose of this rod block function is to avoid conditions that would require reactor protection system (RPS) action if allowed to proceed. The APRM upscale rod block alarm setting is selected to initiate a rod block before the APRM high neutron flux scram setting is reached
- b. Any APRM inoperative alarm. This ensures that no control rod is withdrawn unless the average power range neutron monitoring channels are either in service or are correctly bypassed

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- c. Either rod block monitor (RBM) upscale alarm. This function is provided to stop the erroneous withdrawal of a control rod so that local fuel damage does not result. Although local fuel damage poses no significant threat in terms of radioactive material released from the nuclear system, the trip setting is selected so that no local fuel damage results from a single control rod withdrawal error during power range operation
- d. Either RBM inoperative alarm. This ensures that no control rod is withdrawn unless the RBM channels are in service or are correctly bypassed
- e. Any APRM indicating recirculation flow upscale. This ensures that no control rod is withdrawn unless the recirculation flow functions, which are necessary for the proper operation of APRM rod block function, are operable
- f. Deleted
- g. Scram discharge volume high water level. This ensures that no control rod is withdrawn unless enough capacity is available in the scram discharge volume to accommodate a scram. The setting is selected to initiate a rod block earlier than the scram that is initiated on scram discharge volume high water level
- h. Scram discharge volume high water level scram trip bypassed. This ensures that no control rod is withdrawn while the scram discharge volume high water level scram function is out of service
- i. The rod worth minimizer (RWM) can initiate a rod insert block and a rod withdrawal block. The RWM limits the worth of any control rod that could be dropped by regulating the withdrawal sequence. This system prevents the movement of an out-of-sequence rod in the 100 percent control rod density to the preset low power level, the RWM will allow only BPWS mode withdrawals or insertions. The rod block trip settings are based on the allowable control rod worth limits established for the design-basis control rod drop accident. Additional information on the RWM function is available in Subsection 7.6.1.20
- j. Rod position information system malfunction. This ensures that no control rod can be withdrawn unless the rod position information system is in service
- k. Rod movement timer malfunction during withdrawal. This ensures that continuous withdrawal of a control rod does not result from failure of the normal rod timer during the withdrawal portion of the timing sequence.

With the mode switch in the RUN position, any of the following conditions initiates a rod block:

- a. Any APRM downscale alarm. This ensures that no control rod will be withdrawn during power range operation unless the average power range neutron monitoring channels are operating correctly or are correctly bypassed. All unbypassed APRMs must be on scale during reactor operations in the RUN mode
- b. Either RBM downscale alarm. This ensures that no control rod is withdrawn during power range operation unless the RBM channels are operating correctly

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or are correctly bypassed. An RBM which reads downscale (downscale alarm) and not automatically bypassed by the APRM low power feature is considered to have failed. This results in the rod withdrawal permissive not being given unless this RBM is bypassed.

With the mode switch in the STARTUP or REFUEL position, any of the following conditions initiates a rod block:

- a. Any source range monitor (SRM) detector not fully inserted into the core when the SRM count level is below the retract permit level and any IRM range switch is on either of the two lowest ranges. This ensures that no control rod is withdrawn unless all SRM detectors are correctly inserted when they must be relied on to provide the operator with neutron flux level information
- b. Any SRM upscale level alarm. This ensures that no control rod is withdrawn unless the SRM detectors are correctly retracted during a reactor startup. The rod block setting is selected at the upper end of the range over which the SRM is designed to detect and measure neutron flux
- c. Any SRM downscale alarm. This ensures that no control rod is withdrawn unless the SRM count rate is above the minimum rate prescribed for low neutron flux level monitoring
- d. Any SRM inoperative alarm. This ensures that no control rod is withdrawn during low neutron flux level operations unless neutron monitoring capability is available in that all SRM channels are in service or are correctly bypassed
- e. Any intermediate range monitor (IRM) detector not fully inserted into the core. This ensures that no control rod is withdrawn during low neutron flux level operations unless proper neutron monitoring capability is available in that all IRM detectors are correctly located
- f. Any IRM upscale alarm. This ensures that no control rod is withdrawn unless the intermediate range neutron monitoring equipment is correctly upranged during a reactor startup. This rod block also provides a means to stop rod withdrawal in time to avoid conditions requiring RPS action (scram) in the event that a rod withdrawal error is made during low neutron flux level operations
- g. Any IRM downscale alarm except when range switch is on the lowest range. This ensures that no control rod is withdrawn during low neutron flux level operations unless the neutron flux is being correctly monitored. This rod block prevents the continuation of a reactor startup if the operator upranges the IRM too far for the existing flux level. Thus, the rod block ensures that the IRM is on scale if control rods are to be withdrawn
- h. Any IRM inoperative alarm. This ensures that no control rod is withdrawn during low neutron flux level operations unless neutron monitoring capability is available in that all IRM channels are in service or are correctly bypassed.

Rod Block Bypasses

To permit continued power operation during repair or calibration of equipment for selected functions that provide rod block interlocks, the limited number of manual bypasses that are permitted are:

- a. One SRM channel
- b. Two IRM channels (one on bus A and one on bus B)
- c. One APRM channel
- d. One RBM channel.

The permissible IRM bypasses are arranged in two groups, each having an equal number of channels. One manual bypass is allowed in each group. The groups are chosen so that adequate monitoring of the core is maintained when one channel is bypassed in each group. The arrangement allows the bypassing of one IRM in each rod block logic circuit.

These bypasses are effected by positioning switches in the main control room. A light in the main control room indicates the bypassed condition.

An automatic bypass of the SRM detector position rod block is effected as the neutron flux increases beyond a preset low level on the SRM instrumentation. The bypass allows the detectors to be partially or completely withdrawn as a reactor startup is continued.

An automatic bypass of the RBM rod block occurs when the power level is below a preselected level or when a peripheral control rod is selected. Either condition indicates that local fuel damage is not threatened and that RBM action is not required.

The RWM rod block function is automatically bypassed when reactor power increases above a preselected value in the power range.

Arrangement of Rod Block Trip Channels

Half of the total neutron monitoring equipment (SRM, IRM, APRM, RBM) provides input to one of the two rod block logic circuits and the other half provides input to the other logic circuit. Two of the flow functions from each of the two recirculation loops provides a rod block signal to one logic circuit and the other two flow functions for each recirculation loop provides an input to the other logic circuit. Scram discharge volume high water level signals are provided as inputs into both of the two rod block logic circuits. Both rod block logic circuits sense when the high water level scram trip for the scram discharge volume is bypassed. The rod withdrawal block from the RWM trip affects both rod block logic circuits. The rod insert block from the RWM function prevents energizing the insert bus for both notch insertion and continuous insertion.

The APRM rod block settings are varied as a function of recirculation flow. The RBM rod block settings are power dependent. Analyses show that the selected settings are sufficient to avoid both RPS action and local fuel damage as a result of a single control rod withdrawal error. Mechanical switches in the SRM and IRM detector drive systems provide the position signals used to indicate that a detector is not fully inserted. Additional detail on all the neutron monitoring system (NMS) trip channels is available in Subsection 7.6.1.13.

The rod block from scram discharge volume high water level uses one nonindicating float switch installed on the scram discharge volume. A second float switch provides a main control room annunciation of increasing level.

7.7.1.1.3.6 Inspection and Testing

The RMCS can be routinely checked for correct operation by manipulating control rods using the various methods of control. Detailed testing and calibration can be performed by using standard test and calibration procedures for the various components of the reactor manual control circuitry.

7.7.1.1.4 Environmental Considerations

The RMCS is not required for safety functions, nor is it required to operate after the design-basis accident (DBA). The RMCS is required to operate in the normal plant environments for power generation purposes only.

The hydraulic control units are located in the reactor building. The logic, control units, and instrumentation readouts are located in the main control room. The control rod position detectors are located beneath the reactor pressure vessel (RPV) in Zone 3 of the primary containment. The normal design environments encountered in these areas are listed in Table 3.11-5.

7.7.1.1.5 Operational Consideration

7.7.1.1.5.1 Normal

The RMCS is totally operable from the main control room. Manual operation of individual control rods is possible with a jog switch to effect control rod insertion, withdrawal, or settle. Rod position indicators, described in Subsection 7.7.1.1.5.2, provide the necessary information to ascertain the operating state and position of all control rods. Conditions that prohibit control rod insertion are alarmed by the rod block annunciator.

7.7.1.1.5.2 Operator Information

Instrumentation

Table 7.7-1 gives information on instruments for the RMCS. A large rod information display on the vertical portion of the reactor control benchboard is patterned after a top view of the reactor core, as shown in Figure 7.5-1. The display allows the operator to acquire information rapidly by scanning. Colored windows provide an overall indication of rod pattern and allow the operator to quickly identify an abnormal indication. The following information for each control rod is presented in the display:

- a. Rod fully inserted (green)
- b. Rod fully withdrawn (red)
- c. Rod identification (coordinate position, white)
- d. Accumulator trouble (amber)

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- e. Rod scram (blue)
- f. Rod drift (red).

Also available on digital operator display assemblies (ODAs), one for each two APRM channels, are local power range monitor (LPRM) readings as well as indications of LPRM low flux level and LPRM high flux level.

A separate, smaller display is located just below the large display on the vertical part of the benchboard. This display shows the positions of the control rod selected for movement and the positions of the other rods in the rod group. For display purposes, the control rods are considered in groups of four adjacent rods centered around a common core volume monitored by four LPRM strings. Rod groups at the periphery of the core may have less than four rods. The small rod display shows the positions, in digital form, of the rods in the group to which the selected rod belongs. A white light indicates which of the four rods is selected for movement. On either side of the four-rod position display are indicated (on RBM ODAs - one for each RBM) the readings of the 16 LPRM channels (four LPRM strings) surrounding the core volume common to the four rods of the group. The four-rod display allows the operator to easily focus his attention on the core volume of concern during rod movements. This arrangement eliminates the problems inherent in larger, full core displays where the operator must concentrate his attention on a small portion of a large display. The four-rod display also allows the operator to quickly investigate any volume of the core by simply selecting a control rod located in that volume.

The position signals of selected control rods, together with a rod identification signal, are provided as hardwired digital signals to the rod worth minimizer (RWM). These signals are then provided by the RWM as digital data to the Integrated Plant Computer System (IPCS) via the Multi-Vendor Data Acquisition System (MVD) component of the 3D-Monicores Computer System (3DM).

Control rod position information is obtained from reed switches in the CRD that open or close as a magnet attached to the rod drive piston passes during rod movement. Reed switches are provided at each 3-in. increment of piston travel. Because a notch is 6 in. long, indication is available for each half-notch of rod travel. The reed switches located at the half-notch positions for each rod are used to indicate rod drift. Both the rod selected for movement and the rods not selected for movement are monitored for drift. A drifting rod is indicated by an alarm and red light in the main control room.

The rod drift condition is also monitored by the IPCS. Reed switches are provided at locations that are beyond the limits of normal rod movement. If the rod drive piston moves to these overtravel positions, an alarm is sounded in the main control room. The overtravel alarm provides a means to verify that the drive-to-rod coupling is intact because the drive cannot be physically withdrawn to the overtravel position when the coupling is in its normal condition. Coupling integrity can be checked by attempting to withdraw the drive to the overtravel position.

The following main control room lights are provided to enable the operator to be aware of the conditions of the control rod drive hydraulic system and the control circuitry:

- a. Stabilizer valve selector switch position
- b. Insert command energized

- c. Withdraw command energized
- d. Settle command energized
- e. Withdrawal not permissive
- f. Notch override
- g. Pressure control valve position
- h. Flow control valve position
- i. Drive water pump low suction pressure (alarm only)
- j. Drive water filter high differential pressure (alarm only)
- k. High pressure of charging water to accumulator (alarm only)
- l. CRD temperature
- m. Scram discharge volume not drained (alarm only)
- n. Scram valve pilot air heater high/low pressure (alarm only).

7.7.1.2 Recirculating Flow Control System Instrumentation and Control

7.7.1.2.1 System Identification

7.7.1.2.1.1 Function

The objective of the RFCS is to control reactor power level, over a limited range, by controlling the flow rate of the reactor recirculating water (Figure 5.5-2). The control involves varying the speed of the recirculation pumps by varying the voltage and frequency of the ac supply to each pump motor. The ac supply is provided by a motor-generator set for each pump. Each motor-generator set consists of a squirrel-cage induction motor driving a variable-frequency generator through a variable-speed converter. The generator output is varied by varying the slip within the converter. Since flow rate is directly proportional to pump speed which is proportional to generator speed, generator speed is considered the controlled variable of the system. Manual input to the individual loop controllers is the reference input to the system.

The RFCS is also designed to limit the range and rate of change of pump speed, and to otherwise ensure proper operation and equipment protection.

7.7.1.2.1.2 Classification

This system is a power generation system, nonessential for safety, and is classified in Chapter 3.

7.7.1.2.2 Power Sources

The RFCS consists of Remote Distributed Control System – Reactor Recirculation (Remote DCS-RR) and Remote Input/Output – Reactor Recirculation (Remote I/O-RR). The RFCS has redundant power supplies and redundant processing units. Both the flow loops A and B

are controlled by Remote DCS-RR. The Remote DCS-RR is powered by 120 V ac instrument and control bus and 125 V dc power and Remote I/O-RR is powered by 120 V ac Bus A and Bus B for redundancy.

7.7.1.2.3 Equipment Design

7.7.1.2.3.1 General

Reactor recirculation flow is changed by adjusting the speed of the two reactor recirculating pumps by adjusting the frequency and voltage of the electrical power supplied to the recirculation pump motors.

Control of pump speed, and thus core flow, is such that at various control rod patterns, different power level changes can be accommodated. Refer to Section 4.4.3.5 for a discussion of BWR operation with recirculation flow control.

An increase in recirculation flow temporarily reduces the void content of the moderator by increasing the flow of coolant through the core. The additional neutron moderation increases the reactivity of the core, causing the reactor power level to increase. The increased steam generation rate increases the steam volume in the core with a consequent negative reactivity effect, and a new steady-state power level is established. When recirculation flow is reduced, the power level is reduced in the reverse manner.

Figure 7.7-2 is a simplified illustration of the RFCS. Figure 7.7-3 shows the system functional control diagram (FCD).

Each recirculation pump motor has its own motor-generator set for a power supply. A variable-speed converter is provided between the motor and generator of the motor-generator set. To change the speed of the reactor recirculation pump, the variable-speed converter varies the generator speed, which changes the frequency and magnitude of the voltage supplied to the pump motor so that the desired pump speed is attained. The RFCS uses a demand signal from the operator.

The RFCS is a digital microprocessor based distributed control system (DCS). The DCS features modular design consisting of input/output, redundant processor and communication. The DCS equipment is located in the relay room, control room and reactor building 4th floor.

The operator interface consists of manual/automatic (M/A) controllers, bar graph indicators, recorders, pushbuttons, control switches, indicating lights and video display. The controls permit the operator to operate in manual or automatic mode. An individual, independent M/A controller will provide speed control for each reactor recirculation pump. The manual mode of operation bypasses closed loop speed control. The M/A controller is placed in AUTO for control loop regulation by the feedback signal of generator speed, subject to the limiters. A manual runback indicating pushbutton is provided in the control room to manually runback the recirculation pumps. The M/A controller display includes speed setpoint, generator speed, and speed demand signal. A flat panel display with a touch screen is provided in the control room to access various system parameter data.

The system locks up the scoop tube on fault conditions. The system has the capability to monitor the initiation and clearance of the fault condition.

Each MG set has three magnetic pick-up speed sensors for redundancy, which provide speed feedback inputs into the Remote DCS-RR. The system transfers the M/A controller to MANUAL mode on either loss of two speed signals or communication failure with DCS processors.

A provision has been included in the Fermi 2 design to trip the recirculation pump motor-generator field breakers and drive motor breakers on receipt of ATWS initiation signals. The Fermi 2 RPT design employs two trip coils in each recirculation system motor generator set field breaker and drive motor breaker. This design provides redundant trips of both motor-generator sets following the transient and failure-to-scrum. To minimize the possibility of breakers being tripped inadvertently, the automatic trip signals are arranged in two-out-of-two logic.

The breaker automatic trip signal is a combined ARI/RPT logic. That is, a low reactor vessel water level (level 2) or high reactor vessel pressure signal will initiate the trip of both sets of field breakers. (Refer to Figure 7.7-3, sheet 4).

The RPT may be manually initiated by the same two pushbuttons in the control room (on a divisional basis) as ARI, the difference being that initiation of one division will trip both sets of breakers.

The RPT logic delays MG set field breaker trip on low reactor vessel water level for 9 seconds. This time delay was provided to account for the difference in the pump coastdown time if the field breaker is tripped rather than the motor-generator set drive motor, as was assumed in the LOCA analysis. The manual reset of the generator field breaker trip seal-in circuit does not have any time delay due to the rapid operation of the circuit breaker. The manual reset of the drive motor breakers will have a time delay because the reset logic is ARI logic.

7.7.1.2.3.2 Motor-Generator Sets

Each of the two motor-generator sets and its controls are identical; therefore, only one description is given of the motor-generator set. Figure 5.5-2 shows the general arrangement and rating of the motor-generator set. The motor-generator set can continuously supply power to the pump motor at any speed between approximately 19 percent and 96 percent of the drive motor speed. The motor-generator set is capable of starting the pump and accelerating it from standstill to the desired operating speed when the pump motor thrust bearing is fully loaded by reactor pressure acting on the pump shaft.

The main components of the motor-generator set are

- a. Drive motor - The drive motor is an ac induction motor that drives the input shaft of the variable speed converter
- b. Generator - The variable-frequency generator is driven by the output shaft of the variable-speed converter. During normal operation, the generator exciter is powered by the drive motor. The excitation of the generator is provided from an auxiliary source during pump startup
- c. Variable-speed converter and actuation device - The variable-speed converter transfers power from the drive motor to the generator. The variable-speed

converter actuator automatically adjusts the slip between the converter input shaft and output shaft as a function of the signal from the speed controller. If the speed controller signal is lost, the actuator causes the speed converter slip to remain "as is." Manual reset of the actuation device is required to return the speed converter to normal operation.

7.7.1.2.3.3 Speed Control Configuration

The speed control system (Figure 5.5-2) controls the variable speed converters of both motor-generator sets. The micro-processor-based scoop tube positioner directly interfaces with the speed control system. The motor-generator sets can be manually controlled individually. The control system configuration for each motor-generator set consists of a manual automatic transfer station, a speed control function, a signal failure alarm, a startup mode function and a speed limiter. Components in the new speed control system contain I/O modules feeding the redundant main processing units. The control system is comprised of an arrangement of discrete modules which run the main processing unit. The operator interface is manual/automatic setpoint stations.

Speed Control

There is one speed control for each motor-generator set. The speed control system transmits the signal that adjusts the motor-generator set variable-speed converter. The speed control for each motor-generator set compares the setpoint signal from the operator station to the feedback signal from triple redundant magnetic speed sensors for each motor-generator set. The control system adjusts its output to the speed converter so that the speed feedback signal is made to equal the setpoint signal. The speed controller setpoint signal is received during automatic operation and during motor-generator set manual operation or during pump startup from the startup signal generator.

System Trouble Alarm

There is one system trouble alarm for each motor-generator set. The system trouble alarm actuates an alarm in the main control room and acts to prevent any change of slip within the variable-speed converter.

Startup Mode

There are triple redundant magnetic speed sensors for each motor-generator set. The triple redundant speed sensors supply the setpoint signal to the speed control system. This function sets the motor-generator set variable speed converter for approximately 50 percent recirculation pump speed.

Speed Limiter

There are four speed limiter functions for each motor-generator set. Number 1 limiter is an adjustable high limit. The speed control setpoint signal is automatically limited by the Number 1 limiter if the recirculation pump main discharge valve is not fully open or if the feedwater flow is less than 20 percent of rated flow. Number 2 or 3 limiter acts on the position demand to the scoop tube positioner and is actuated when a feedwater pump is tripped and level is below the low alarm setpoint (Number 2 limiter) or when both heater drain pumps are not pumping forward to the suction of the feed pumps (Number 3 limiter).

A manual defeat of runback 2 and 3 logic is used during startups, shutdowns and single loop operation. Number 4 limiter limits speed controller setpoint signal of operating pump following a trip of a single recirculation pump. The limiters are enabled during manual operation of the operator station. A manual runback indicating pushbutton is provided in the control room to manually runback the recirculation pumps.

7.7.1.2.3.4 Recirculation Loop Starting Sequence

Each recirculation loop is independently put into operation by operating the controls of each recirculation loop as follows:

- a. Whenever the generator field breaker is open, the control system is automatically placed in startup mode. Startup mode bypasses the normal speed control circuits to position the variable-speed converter for startup. The minimum speed of the recirculation pumps is 20 percent as established by the mechanical stops. Startup operations of the plant are normally carried out with the recirculation pumps operating at approximately 30 percent speed. The power-versus-flow operating state for the reactor follows the 30 percent speed line for the normal control rod withdrawal sequence. (See Section 4.4.3.3.1 and Figure 4.4-3)
- b. The starting sequence is manually initiated by placing the drive motor control switch for one motor-generator set in the start position.
- c. Once the variable-speed converter has achieved its startup position, the following events occur:
 - 1. The auxiliary source of field excitation is engaged after a time delay
 - 2. The generator field breaker is closed after a time delay.
- d. When the generator field breaker is closed, the manual/automatic setpoint station is automatically transferred to give the desired initial generator speed (typically <30 percent of rated speed) after the startup sequence is complete.
- e. Deleted
- f. After recirculation pump start is sensed by a combination of field breaker position and generator output current, the generator is automatically transferred to self-excitation
- g. Recirculation flow is increased during startup by manually increasing recirculation pump speed
- h. Deleted

7.7.1.2.3.5 Inspection and Testing

The motor-generator set, and the speed control system are functioning during normal power operation. Any abnormal operation of these components can be detected during operation. The components that do not continually function during normal operation can be tested and inspected for calibration and operability during scheduled plant shutdowns. All the RFCS

components are tested and inspected according to the component manufacturer's recommendations. This can be done during scheduled shutdowns.

7.7.1.2.4 Environmental Considerations

The RFCS is not required for safety purposes, nor is it required to operate after the DBA. The system is required to operate in the normal plant environment for power generation purposes only. The following normal design environments are encountered by parts of the RFCS. The recirculation flow-control equipment in Zone 4 of the primary containment is the pump motor, which is subject to the environment specified in Table 3.11-5 under environmental conditions.

The control system hardware, operator controls, and instrumentation terminals are located in the main control room, relay room and reactor building (remote I/O) and are subject to the normal environments of these areas.

7.7.1.2.5 Operational Considerations

Indicators and alarms are provided to keep the operator informed of the status of the system so that he may quickly determine the location of malfunctioning equipment.

Temperature monitoring of equipment is recorded and alarmed if safe levels are exceeded. Indicators are provided to show pumping power requirements, motor-generator set speed, recirculation loop flow, valve positions, and analog control signal, all of which determine system status. Alarms are provided to alert the operator of malfunctioning control signals, excessive cooling water temperatures, inability to change pump speed, and the status of the motor-generator circulating lube-oil supply.

7.7.1.3 Feedwater Control System Instrumentation and Control

7.7.1.3.1 System Identification

7.7.1.3.1.1 Function

The feedwater control system automatically controls the flow of feedwater into the RPV so that the water in the vessel is maintained within predetermined levels during all modes of plant operation. The range of water level is based on the requirements of the steam separators, including limiting carryover and carryunder, which affects turbine performance and recirculation pump operation. The range of water level is also based on the need to prevent exposure of the reactor core. The feedwater control system uses water level, steam flow, and feedwater flow as a three-element control. Single-element control, based on water level only, is also available.

Normally, the signal from the feedwater flow is equal to the steam flow signal; thus, if a change in the steam flow occurs, the feedwater flow follows. The steam flow signal provides anticipation of the change in water level that would result from change in load. The level signal provides a correction for any mismatch between the steam and feedwater flow, which causes the level of the water in the RPV to rise or fall accordingly. Figure 7.7-4 shows the system IED.

7.7.1.3.1.2 Classification

This system is a power generation system, nonessential for safety, and is classified in Chapter 3.

7.7.1.3.2 Power Sources

The feedwater control system power is supplied by two (2) redundant uninterruptible power supplies. Interruptible instrument air and power is supplied to certain feedwater system (N21) control valves and operators.

7.7.1.3.3 Equipment Design

7.7.1.3.3.1 General

During normal plant operation, the feedwater control system automatically regulates feedwater flow into the RPV. The system is a distributed control system (DCS) using redundant processors and communication links. This system can be manually operated from the main control room.

The feedwater flow control instrumentation measures the water level in the RPV, the feedwater flow rate into the RPV, and the steam flow rate from the RPV. During automatic three-element operation, these measurements are used for controlling feedwater flow.

The optimum RPV water level is determined by the requirements of the steam separators. The separators limit water carryover in the steam going to the turbines and limit steam carryunder in water returning to the core. The water level in the RPV is maintained within ± 2 in. of the setpoint level. This control capability is achieved during plant load changes by balancing the mass flow rate of feedwater to the RPV with the steam flow from the RPV. The feedwater flow is regulated by adjusting the speed of the turbine-driven feedwater pumps to deliver the required flow to the RPV.

7.7.1.3.3.2 Reactor Pressure Vessel Water Level Measurement

Reactor pressure vessel water level is measured by two independent sensing systems. Two (2) redundant differential pressure transmitters in each system sense the difference between the pressure caused by a constant reference column of water and the pressure caused by the variable height of water in the RPV. A backfill system is installed on each reactor water level instrument reference leg in compliance with the requirements of USNRC Generic Letter 92-04 and Bulletin 93-03. The system provides a metered flow of water from the control rod drive system (CRD) to each leg to prevent the accumulation of the noncondensable gases in the reference legs and assure a high reliability of the water level indication. The backfill flow rate is low enough to not affect the performance of the instrumentation. The differential pressure transmitters are installed on lines that serve other systems (Subsection 7.6.1.2). The differential pressure signals are used for level indication and control. The amplifier transmits the level signal for indication and control. The RPV water level signals from each sensing system are indicated in the main control room. The level signal from either sensing system can be manually selected by the operator as the signal to be used for feedwater flow control.

The redundant level signals in the operator selected sensing system and both level signals from the nonselected sensing system are applied to a median signal selector. The median signal is compared against the operator selected level and will automatically assume the lead level control signal if the operator selected sensing system fails. The water level for control is continuously recorded in the main control room.

7.7.1.3.3.3 Steam Flow Measurement

Steam flow is sensed at each main steam line flow restrictor by differential pressure transmitter. These steam flow signals are indicated in the main control room. The signals are summed to produce a total steam flow signal for indication and feedwater flow control. The total steam flow signal is recorded in the main control room.

7.7.1.3.3.4 Feedwater Flow Measurement

Feedwater flow is sensed at a flow element in each feedwater line by differential pressure transmitters. Each feedwater signal is summed to provide a total mass flow signal for the feedwater control system. The total feedwater flow signal is also recorded and integrated in the main control room. In addition, feedwater flow is sensed by an ultrasonic flow meter in each feedwater line and processed in the associated central processing unit to provide mass flows to the Integrated Plant Computer System (IPCS) for the sole purpose of the IPCS heat balance calculation and is not used for any direct control function.

7.7.1.3.3.5 Feedwater Control Signal

The level control system produces the feedwater control signal through digital control logic, the master level controller and the reactor feed pump manual/automatic control stations. The signal can be controlled either manually or automatically.

The master level control and the reactor feed pump manual/automatic stations contain a setpoint meter, level indicator, and a manual output control indicator.

The master manual/automatic setpoint station contains a setpoint meter, level indicator and a manual output control with an indicator. Input to the control system is derived from either the single-element signal (level only) or the three-element signal. The three-element signal is the summation of steam flow, feedwater flow, and the selected reactor water level. Single-element or three-element level control is manually selected by the operator. When three-element level control is selected, automatic transfer to single-element level control will occur if one of the feedwater flow signals or two of the steam flow signals should fail. Manual level control is automatically initiated if the control system cannot provide automatic level control. During automatic operation of the feedwater control system, the level control system output is proportional to the level error in the system. During manual operation, output is set and indicated at the manual/automatic setpoint station.

The level demand signal from the master control is applied to the input of two manual/automatic stations that have capabilities to add or subtract a bias signal from the master level demand, when in automatic. The bias capabilities allow independent adjustment of the speed demand signals to the turbine-driven feedwater pumps during automatic

operation. During manual operation of the bias stations, the speed demand signal is manually adjusted by the operator.

Selection of automatic or manual control is made by the operator at a master manual/automatic setpoint station.

Normal Automatic Operation

The feedwater control system provides function block through the I/O modules to compute the three-element control signal to maintain RPV water level within a small margin of optimum water level during plant load changes. The total steam flow signal and the total feedwater flow signal are subtracted from each other to derive a flow error signal. When steam flow exceeds feedwater flow, the error signal is positive in polarity from its normal zero value. The flow error signal is multiplied by a gain factor referred to as mismatch gain. The mismatch gain determines how much level effect the flow error signal has when the error is 100 percent. The mismatch gain is used as a dynamic control system adjustment. The flow error signal is limited for +/-20 percent multiplied by the mismatch gain. The flow error signal is then subtracted from the selected level signal to provide the three-element control signal. When feedwater flow exceeds steam flow, the error signal is negative polarity. The three-element signal is modified further by a lead/lag function before being used for level control. The control system compares this signal against the level setpoint adjusted by the operator.

Following a reactor scram, the RPV water level controller setpoint is capable of being automatically lowered so that the Reactor Feed Pump Turbines do not overfill the reactor vessel. On receipt of a scram signal via a contact of an RPS auxiliary relay, the Post-Scram Reactor Water Level Setdown Logic lowers the level controller setpoint after the time delay. A momentary actuated switch in the control room allows the operator to reset the Post-Scram Reactor Water Level Setdown Logic after the scram signal is cleared.

Optional Automatic Operation

A single-element control signal (RPV water level) can be used to replace the above three-element signal. The operator manually transfers the level controller input to the "1 element control" signal. In the event of failure of the three-element signal, the control system will automatically transfer to single element. Reactor water level is then controlled in accordance with the controller setpoint.

Auxiliary Functions

The level control system also provides interlocks and control functions to other systems. When one of the reactor feed pumps is lost and coincident or subsequent low water level exists, reactor recirculation flow is reduced to within the power capabilities of the remaining reactor feed pumps. This reduction aids in avoiding a low-level scram by reducing the steaming rate.

Reactor recirculation flow is also reduced on sustained low feedwater flow to ensure that adequate net positive suction head (NPSH) is provided for the recirculation system.

Interlocks from steam flow and feedwater flow are used to initiate insertion of the RWM block. An alarm on low steam flow indicates that the RWM insertion interlock setpoint is being approached. Alarms from the control system are also provided for high and low water

level, reactor high pressure and failures. High reactor water level (L-8) from the nuclear boiler system trips the turbine driven feedwater pumps, see subsection 7.6.1.2.7.

7.7.1.3.3.6 Turbine-Driven Feedwater Pump Control

Feedwater is delivered to the RPV through turbine-driven feedwater pumps arranged in parallel. The turbines are driven by steam from the RPV. During normal operation, the feedwater control signal from the level controller is fed to the turbine control mechanisms. The turbine control mechanisms adjust the speed of their associated turbines so that feedwater flow is proportional to the feedwater control signal. Each turbine can be controlled by its manual/automatic transfer station. The master manual/automatic setpoint station and the manual/automatic station associated with each turbine speed controller are configured to have "bumpless transfer". The turbine-driven feedwater pump control has speed limiters to restrain maximum feedwater flow to 117 percent.

7.7.1.3.3.7 Inspection and Testing

All feedwater control system components can be tested and inspected according to the manufacturers' recommendations. This can be done prior to plant operation and during scheduled shutdowns. Reactor pressure vessel water level indications from the two water level sensing systems are compared during normal operation to detect instrument malfunctions. Steam mass flow rate and feedwater mass flow rate can be compared during constant load operation to detect inconsistencies in their signals.

7.7.1.3.4 Environmental Considerations

The feedwater control system is not required for safety purposes, nor is it required to operate after the DBA. This system is required to operate in the normal plant environment for power generation purposes only. The reactor feed pumps in the turbine building experience the normal design environments listed in Table 3.11-5.

7.7.1.3.5 Operational Considerations

7.7.1.3.5.1 Normal

All control stations are located in the main control room where, at the operator's discretion, the feedwater control system can be operated either manually or automatically. Manual control of the individual turbine-driven feedwater pumps is available to the operator in the main control room. Manual control of the individual turbine-driven feedwater pumps is used during control of the startup level control valve. The startup level control valve is used to supply feedwater during periods of low reactor pressure and/or flow demand. The startup control system will automatically hold reactor water level to an operator selected setpoint as a single-element unmodified control system. It can also be operated manually.

Subsequent to a scram, the feedwater flow demand is very low. To ensure adequate control at this low flow, the feedwater control system automatically diverts feedwater flow through the startup control valve when a scram occurs. A minimum flow recirculation line valve

automatically opens to maintain flow through each feedwater pump so that the pump is protected from overheating.

7.7.1.3.5.2 Operator Information

Indicators and alarms, provided to keep the operator informed of the status of the system, are discussed in Subsection 7.7.1.3.3.

7.7.1.4 Pressure Regulator and Turbine-Generator Instrumentation and Control

7.7.1.4.1 System Identification

7.7.1.4.1.1 Function

Power Generation - The pressure regulator system maintains constant main turbine inlet steam pressure.

7.7.1.4.1.2 Classification

The main turbine pressure regulator and bypass system is a conventional analog/hydraulic control system and is classified in Chapter 3.

7.7.1.4.2 Normal Power Sources

The main turbine pressure regulator control system is supplied by two independent 120-V ac instrument buses.

7.7.1.4.3 Equipment Design

7.7.1.4.3.1 System Description

Control and supervisory equipment for the turbine generator is conventional and arranged for remote operation from the main control room. Normally, the initial pressure regulator controls steam throttle valve position to maintain constant reactor pressure. The ability of the plant to follow system load demands is accomplished by adjusting reactor power level, either by changing flow in the reactor recirculation system (manually) or moving control rods (manually). However, the turbine speed governor, which is supplied by the turbine supplier, can override the initial pressure regulator. The steam valves close when an increase in system frequency or a loss of generator load causes the speed of the turbine to increase. In the event that the reactor is delivering more steam than the admission valves pass, the excess steam is automatically and directly bypassed to the main condenser by pressure-controlled bypass valves. Figure 7.7-5 is a simplified control diagram.

7.7.1.4.3.2 Steam Pressure Control

During normal plant operation, steam pressure is controlled by the turbine control valves. These control valves are positioned in response to either the pressure regulation signal or the turbine speed-load signal as selected by a "low value gate" circuit in the BOP turbine control

system. The change in steam production is sensed by the pressure regulator, which signals the turbine control valves to adjust position to accept the change in steam flow, thereby regulating steam pressure.

A main steam line resonance filter is included in the pressure regulator circuits to prevent cycling from false pressure signals. These false pressure signals could be caused by sonic resonances in the main steam lines.

7.7.1.4.3.3 Steam Bypass System

The steam bypass equipment is designed to control steam pressure when reactor steam generation exceeds turbine requirements such as during startup (speed raising and synchronizing), sudden load reduction, and cooldown. Capacity of the system is 23.5 percent of 105 percent of nuclear steam supply system (NSSS) rated steam flow, and sudden load reductions of up to 25 percent of rated power can be accommodated without reactor scram.

Normally, the bypass system valves are held closed while the pressure regulator controls the turbine control valves, directing all steam flow to the turbine. If the speed governor or the load limiter restricts steam flow to the turbine, the regulator controls system pressure by opening the bypass valves. If the capacity of the bypass valves is exceeded while the turbine cannot accept an increase in steam flow, the system pressure rise and RPS action causes shutdown of the reactor.

The bypass valves are the automatically operated, regulating type. They are proportionately controlled by the NSSS pressure regulator which compares the steam pressure signal with the turbine control valve signal to bypass excess steam to the main condenser.

Bypass valves and controls are designed so the valves close on loss of control system electric power or hydraulic pressure.

7.7.1.4.3.4 Turbine Speed/Load Control System

The turbine control system is discussed in Chapter 10.

7.7.1.4.3.5 Turbine Generator to Reactor Protection System Interface

The RPS initiates reactor scram when it is required by the particular monitored plant conditions (Section 7.2). Two such conditions are turbine stop valve closure and turbine control valve fast closure when reactor power is above 29.5 percent.

The turbine stop valve closure signal is generated before the turbine stop valves have closed more than 10 percent (opened less than 90 percent). This signal originates from position switches that sense stop-valve motion away from fully open. The switches are closed when the stop valves are fully open, and the switches open within 10 msec after the setpoint is reached. The switches are electrically isolated from each other and from other turbine plant equipment.

The control-valve-fast-closure signal is generated by the relay logic that initiates the fast control valve closure. Separate circuits are associated with each of the control valves. Relay contacts are closed whenever the control valves are not being closed in the fast mode, and these relay contacts open when the fast closure mode is initiated.

To avoid reactor scram due to turbine stop or control valve fast closure when power is below 29.5 percent of rated power, two independent sensing lines are provided from the turbine first-stage pressure transmitter/trip units, which supply power level logic contacts to the RPS. The pressure taps are located to provide a pressure signal proportional to turbine steam flow. The pressure taps are shared with other instrumentation sensors. All sensors have individual isolation or root valves.

7.7.1.4.3.6 Inspection and Testing

Testing controls for testing the turbine valve RPS interface signal switches are provided to:

- a. Actuate each stop valve individually to the 10 percent closed point with no interaction with other valves
- b. Actuate one stop valve to the ten percent closure point and simulate another stop valve at the 10 percent closure point in the following combinations: valves 1 and 2; valves 1 and 3; valves 2 and 4; valves 3 and 4
- c. Actuate one control valve at a time in the fast closure mode with no interaction with other valves.

7.7.1.4.4 Environmental Considerations

The pressure regulator and turbine-generator control system is not required for safety nor is it required to operate after the DBA. This system is required to operate in the normal plant environment for power generation purposes only.

Instrumentation and control on the turbines that experience the turbine building normal design environment is listed in Table 3.11-1. The logic, remote control units, and instrument terminals located in the main control room experience the environment listed in Table 3.11-1.

7.7.1.4.5 Operational Considerations

7.7.1.4.5.1 Normal

Two pressure control channels (A and B), operating redundantly, receive inputs from the pressure reference unit and from independent pressure transducers in the main steam lines upstream of the main steam stop valves. Main steam pressure is indicated on meters on the turbine control panel.

The pressure setpoints for the pressure reference circuit are produced by tandem potentiometers driven by a common motor. The motor is controlled by use of pushbuttons on the PRESSURE SETPOINT SELECTOR section of the main control panel. Desired setpoints for Channels A and B are indicated on meters on the main control panel. Pressure setpoint adjustment is limited to a maximum of 1 psi/sec by motor speed. In the event of failure of both regulators, alarm communication is provided in the main control room. Pushbutton operation is provided to remove the system from operation.

7.7.1.4.5.2 Operator Information

Nuclear Steam Supply System Control and Display

The NSSS pressure regulator has the following controls and information displayed in the main control room:

- a. Main steam pressure regulator setpoint A
- b. Main steam pressure regulator setpoint B
- c. Individual bypass valve position indicators
- d. Bypass valve test controls
- e. Pressure regulator selection control.

Balance of Plant Control and Display

A list of the conventional turbine-generator control and supervisory instrumentation provided for operational analysis and malfunction diagnosis is described in Section 10.2.

7.7.1.5 Gaseous Radwaste System Instrumentation and Control

7.7.1.5.1 System Identification

7.7.1.5.1.1 Function

The objective of the gaseous radwaste system is to process and control the release of gaseous radioactive wastes to the site environs so that the total radiation exposure to persons outside the controlled area is as low as practicable, and does not exceed applicable regulations.

7.7.1.5.1.2 Classification

This system is required for power generation only.

7.7.1.5.2 Power Sources

The 120-V ac instrument bus normally provides power for the gaseous radwaste system instrumentation.

7.7.1.5.3 Equipment Design

7.7.1.5.3.1 General

The radiation levels at the offgas delay pipe and at the discharge of the offgas system are continuously monitored by detectors described in Section 11.4. This system is also monitored by flow and temperature instrumentation and by a hydrogen analyzer to ensure correct operation and control and to ensure that hydrogen concentration is maintained below the flammable limit. Table 7.7-2 lists process instruments that cause alarms and whether or not they are indicated or recorded in the main control room.

7.7.1.5.3.2 Catalytic Recombiner Instrumentation

The catalytic recombinder vessel temperatures are monitored by thermocouples and are then recorded. High or low temperature is annunciated in the main control room. The standby recombinder is temperature controlled, maintained, monitored, and recorded. Any low temperature is annunciated in the main control room. Inlet process gas is monitored for pressure and temperature. If abnormal measurements are obtained, they are annunciated in the main control room.

7.7.1.5.3.3 Offgas Condenser Condensate Level Control

The offgas condenser condensate level is maintained at a given level within the condenser shell. A level control system is used to provide drainage of condensate from the condenser shell. High level is annunciated in the main control room.

7.7.1.5.3.4 Offgas System Flow Measurements

Offgas system flow measurements are made between the charcoal adsorbers and the absolute filter downstream of the ring water buffer tanks before discharge into the reactor building ventilation stack.

7.7.1.5.3.5 Hydrogen Analyzer Measurement System

One hydrogen analyzer is used to measure the hydrogen content of the offgas process stream in the delay pipe. The hydrogen concentration percentage output from the analyzer is indicated and recorded in the main control room along with alarm annunciation for high hydrogen concentration percentage in the offgas process stream.

The hydrogen analyzer system continuously withdraws a sample of the process offgas, analyzes the hydrogen content, and returns the sample gas to the delay pipe. A loss of ac power to the analyzer system stops the analyzer.

7.7.1.5.3.6 Charcoal Vessel and Vault Temperature and Flow Monitoring and Control

Each charcoal vessel is temperature monitored. High vessel temperature is alarmed and annunciated at 100°F in the main control room. The charcoal vessel vault is also temperature monitored and recorded in the main control room along with high temperature alarm and annunciation. Three refrigeration units maintain the vault at a nominal temperature of 70°F.

The charcoal vessel train is flow monitored at the outlet and is indicated and recorded in the main control room along with highflow alarm and annunciation.

7.7.1.5.3.7 Differential Pressure Measurements

Differential pressure measurements are made across the precoolers, the sandfilter, the chillers, the charcoal vessel train, and the absolute filters. High differential pressure is annunciated in the main control room.

7.7.1.5.4 Environmental Considerations

The offgas control system is not required for safety purposes, nor is it required to operate after the DBA. The offgas control systems are required to operate in the normal plant environment for power generation purposes only.

Radwaste instrumentation and controls located in the offgas equipment area are subject to the environment under design conditions listed in Table 3.11-5. The control circuitry, remote control units, and instrument terminals in the main control room experience the normal design environment also listed in Table 3.11-5.

7.7.1.5.5 Operational Considerations

7.7.1.5.5.1 General

No operator action is required on the equipment described unless an alarmed condition occurs. The offgas trip signal is taken from the relay room panel H11-P913. Employing contact-to-coil separation prevents HWC System operation from affecting the Offgas system. Operator indicators and alarms are described in Subsection 7.7.1.5.3.

7.7.1.5.5.2 Setpoints

Hydrogen Analyzer

A hydrogen level of ~1.5 percent alarms and annunciates in the main control room.

Flow

A high flow of approximately 70 scfm alarms and annunciates in the main control room.

7.7.1.6 Liquid Radwaste System Instrumentation and Control

7.7.1.6.1 System Identification

7.7.1.6.1.1 Function

The objective of the liquid radwaste system is to control the release of liquid radioactive waste material to the environs and to package these wastes in suitable containers for offsite shipment and burial.

7.7.1.6.1.2 Classification

Since this system is required for power generation only, it does not include any Quality Class 1 or Category I components with the exception of the drywell drain isolation valve controls. The closure of these valves is necessary for sealing the primary containment under postulated accident conditions. The initiating signal is from the containment and reactor isolation control system (Subsection 7.3.2.2).

7.7.1.6.2 Power Sources

The 120-V ac instrument power is used for the liquid radwaste system.

7.7.1.6.3 Equipment Design

7.7.1.6.3.1 General

The liquid radwaste system is designed to process liquid waste water to remove particulates, impurities, and other materials, and to return the processed water for plant usage. The resulting solid wastes are then packaged in suitable containers for offsite burial.

Only those portions of the liquid radwaste system related to safety are described herein.

7.7.1.6.3.2 Instrumentation and Control

The radiation levels of the waste materials packaged for burial are monitored by plant personnel and are not part of this control system. Wastewater is collected in various sumps throughout the plant and is pumped into the radwaste collection tanks where it is processed. Excess processed liquids that are discharged from the plant are radiation monitored, flow controlled, and recorded. The instrumentation and control system of the radwaste process is typical of a standard chemical and water treatment process. Tank levels are indicated and recorded in the radwaste control room and high tank levels are annunciated in the radwaste control room.

Radiation from the liquid releases is monitored and recorded with high and low/inoperative alarms in the radwaste control room and alarms only in the main control room.

7.7.1.6.3.3 Drywell Sumps Control

There are two sumps within the containment that collect waste water which is pumped out to the liquid radwaste system collector tanks. Each sump is equipped with two pumps that automatically start and stop on high and low sump levels, respectively. The pumps are alternately started on each high level signal. Each pump is equipped with a separate float switch in a separate float well and is electrically connected to provide level backup for the other pump if one float device should fail. A high-high level is provided by each float switch which will start both pumps and annunciate an alarm in the main control room. The liquid discharge lines to the radwaste collector tanks are provided with two isolation valves. When either isolation valve is closed, the sump pumps are interlocked to prevent their operation. The sumps are automatically isolated on high drywell pressure or low reactor water level (L3).

7.7.1.6.3.4 Reactor and Turbine Building Sumps

These sumps collect waste water from their respective areas and automatically pump out the sumps on level control. These are not safety systems. An alarm and annunciation in the radwaste control room will occur on a high-high sump level to allow the operator to take corrective action.

7.7.1.6.3.5 Tank Level and Process Control

All tanks containing waste liquids throughout the radwaste liquid processing system are provided with liquid level indicators or recorders and alarms, and annunciators in the radwaste control room for high liquid level to inform the operator that corrective action is to be taken. The process control is by an operator from the radwaste control room panel. The control system is designed for manual startup and automatic stop when a process is completed (i.e., tank liquid contents have been emptied to next process). Since this is a batch system, the operator has full control and responsibility for the system control process.

The Side Stream Liquid Radwaste Processing System (SSLRPS) operation is controlled from the local control panel in the Radwaste Building Basement. Tank liquid level indicators, recorders and alarms are provided in the local control panel. Radwaste Control Room is provided with a trouble indicating alarm, as a backup, to alert the Radwaste Control Room Operators when the system operation drifts from the normal range.

7.7.1.6.4 Environmental Considerations

The radwaste control systems are not required for safety purposes, nor are they required to operate after the DBA. The radwaste control systems are required to operate in the normal plant environment for power generation purposes only. This environment is listed in Table 3.11-1.

7.7.1.6.5 Operational Considerations

7.7.1.6.5.1 General

The operator is in full control of the process system batches. Indicators and recorders are provided for all liquid tanks to inform the operator of the status of the system. Alarms and annunciation are provided to inform the operator either that a tank must be emptied or processed, or that a particular piece of equipment has malfunctioned so that corrective action may be taken.

7.7.1.6.5.2 Setpoints

All tank levels are set to alarm and annunciate in a timely manner in order to avoid overflow. This allows sufficient time for the operator to take corrective action in the process control.

7.7.2 Analysis

7.7.2.1 General

This subsection demonstrates that the protection systems are capable of coping with all failure modes of the control system.

7.7.2.2 Reactor Manual Control System Instrumentation and Control

7.7.2.2.1 Conformance To General Functional Requirements

The circuitry used in the RMCS is completely independent of the circuitry controlling the scram valves. This separation of the scram and normal rod control functions prevents failures in the reactor manual control circuitry from affecting the scram circuitry. The scram circuitry is discussed in Section 7.2. Because each control rod is controlled as an individual unit, a failure that results in the energizing of any of the insert or withdraw solenoid valves can affect only one control rod. The effectiveness of a reactor scram is not impaired by the malfunctioning of any one control rod. Therefore, no single failure in the RMCS can result in the prevention of a reactor scram. Repair, adjustment, or maintenance of RMCS components does not affect the scram circuitry.

The RMCS is an operational system used for regulating power level and power distribution. This system is self-monitoring with the automatic rod blocks, operator annunciators, and operating status lights (such as the rod position indicators) as part of the system design. The rod blocks are an internal subsystem of this nonsafety system. As such they are designed to be single-failure-proof, but are not designed to stringent safety standards.

The RMCS receives rod block signals from the NMS to prevent improper rod motion that could result in reactor scram. Common LPRM, IRM, and SRM detectors are used, but the signal is physically and electrically isolated before use in the RMCS. This isolation is achieved through two separate relay trip units that prevent any feedback from the RMCS to the RPS. Subsections 7.6.1 and 7.6.2 describe this interface.

The performance of the RMCS is monitored by the RPS. If a variable, such as the neutron flux, which is controlled by the RMCS, exceeds specific limits, the RPS takes independent action to cause reactor shutdown.

It is thus seen that the RMCS is not required for safety nor for reactor shutdown, but only for changing plant power.

Accident analyses in Chapter 15 show that failures in the RMCS, such as continuous withdrawal of a control rod, do not result in any fuel damage. No fuel damage results from any single operator error or single equipment malfunction.

7.7.2.2.2 Conformance To Specific Regulatory Requirements

The RMCS meets the requirements of GDC 24 of 10 CFR 50, Appendix A.

No part of the RMCS is required for scram. The rod block functions provided by the NMS and the scram discharge volume high water level trip bypass signal interlocks are the only instances where the RMCS uses any instruments or devices used by the RPS. This includes relay contacts to the reactor mode switch and the scram discharge volume high level bypass switch. The rod block signals received from the NMS prevent improper rod motion before limits causing reactor scram are reached. Common LPRM, IRM, and SRM detectors are used, but physically and electrically separate trip signals are supplied to the RMCS and RPS systems. A description of this interface is contained in Subsections 7.6.1 and 7.6.2. The scram discharge volume high water level trip bypass signal interlocks with the RMCS to initiate a rod block. The interlock is performed using isolating relay contacts so that no failure in the control system can prevent a scram.

7.7.2.3 Recirculating Flow Control System Instrumentation and Control

7.7.2.3.1 Conformance To General Functional Requirements

The RFCS is designed so that coupling is maintained between a motor-generator set drive motor and its generator even if the ac power or a speed controller signal fails. This ensures that the drive motor inertia contributes to power supplied to the recirculation pump during the coastdown of the motor-generator set after loss of ac power, and also ensures that the generator continues to be driven if the speed controller signal is lost.

Transient analyses described in Chapter 15 show that no malfunction in the RFCS can cause a transient sufficient to either damage the fuel barrier or exceed the nuclear system pressure limits as required by the safety design basis.

7.7.2.3.2 Conformance To Specific Regulatory Requirements

Except for the recirculation pump trip function, there are no specific regulatory requirements for the RFCS. The RFCS is not a safety-related system and is not required for safe shutdown of the plant, nor is it required during or after accident conditions. The recirculation pump trip function meets the requirements of IEEE 323-1974 and IEEE 344-1975.

7.7.2.4 Feedwater Control System (Turbine-Driven Pumps) Instrumentation and Control

7.7.2.4.1 Conformance To General Functional Requirements

The feedwater is a power generation system for the purposes of maintaining proper RPV water level. Should the RPV water level rise too high, the feedwater pumps and plant main turbine would be tripped. This is an equipment protective action which would result in reactor shutdown by the RPS as outlined in Section 7.2. Lowering of the RPV water level would also result in action of the RPS to shut down the reactor. Further decrease would actuate the emergency core cooling system (ECCS). Loss of feedwater is analyzed in Chapter 15.

7.7.2.4.2 Conformance To Specific Regulatory Requirements

The feedwater control system is not a safety-related system and is not required for safe shutdown of the plant, nor is it required during or after accident conditions. The Feedwater Control System Contains QA Level 1 transmitters classified as NUREG-0588 Category 2B (mechanical) for pressure boundary integrity and Category 2C (electrical).

There is no interface with safety-related systems, with the exception of the Reactor Protection System which provides a Post Scram Signal to the Feedwater Control System.

7.7.2.5 Pressure Regulator and Turbine-Generator Instrumentation and Control

7.7.2.5.1 Conformance To General Functional Requirements

The pressure regulator and turbine-generator instrumentation and control is designed to maintain constant reactor pressure, to follow system load demand fluctuations, and to control turbine speed. Excessive reactor pressure swings caused by failure of this system would be dealt with by the RPS (Section 7.2) and/or the safety/relief valves.

7.7.2.5.2 Conformance To Specific Regulatory Requirements

The pressure regulator and turbine-generator instrumentation and control is neither safety-related nor required for the safe shutdown of the plant. It is also not required during or after accident conditions.

7.7.2.6 Gaseous Radwaste System Instrumentation and Control

7.7.2.6.1 Conformance To General Functional Requirements

The objectives of the gaseous radwaste system instrumentation and control are to indicate and alarm the level of radioactivity within offgas process lines, to provide a record of all radioactive plant site releases, and to initiate appropriate action that would prevent the release of radioactive materials to the environs that exceed the operational limits established in 10 CFR 20 and Regulatory Guide 1.21.

The flow recorder is provided to keep a record of all discharge volumes. The flow measurements and recording accuracies are within 5 percent of indication for the flows measured.

7.7.2.6.2 Conformance To Specific Regulatory Requirements

The gaseous radwaste system instrumentation and control is neither safety-related nor required for the safe shutdown of the plant. It is not required to operate after a DBA. The gaseous radwaste system instrumentation and control is required to operate in the normal plant environment for power generation purposes only.

7.7.2.7 Liquid Radwaste System Instrumentation and Control

7.7.2.7.1 Conformance To General Functional Requirements

The liquid radwaste effluent for discharge to the circulating water blowdown is flow controlled and monitored for activity level. The discharge flow shutoff valve is operated by a keylock switch that requires plant supervisory control of any releases. The flow is recorded in the radwaste control room.

The packaged wastes are stored in the plant in a storage area set aside for this purpose. The radioactivity and quantity is the responsibility of plant supervisory personnel. This complies with Regulatory Guide 1.21, Revision 0.

7.7.2.7.2 Conformance To Specific Regulatory Requirements

Section 11.2 discusses the conformance of the liquid radwaste system to specific regulatory requirements.

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TABLE 7.7-1 REACTOR MANUAL CONTROL SYSTEM INSTRUMENT SPECIFICATIONS

<u>Measured Variable</u>	<u>Instrument Type</u>	<u>Instrument Range</u>	<u>Accuracy^d</u>	<u>Trip Setting^a</u>
Drive water header pressure	Pressure indicator	0 to 2000 psig	±1/2 percent full scale	-
Drive water pump discharge pressure	Pressure indicator	0 to 2000 psig	±1/2 percent full scale	-
Drive water pump suction pressure	Pressure indicator	30 in. Hg to 60 psig	±1.5 percent full scale	-
Drive water filter differential pressure	Differential pressure switch (indicating)	0 to 75 psig	±1/2 percent full scale	25 psid, increasing
Cooling water header pressure	Pressure indicator	0 to 2000 psig	±1/2 percent full scale	-
Exhaust water header pressure	Pressure indicator	0 to 2000 psig	±1/2 percent full scale	-
Charging water accumulator header pressure	Pressure indicator	0 to 2000 psig	±1/2 percent full scale	-
Charging water header pressure	Pressure indicator	0 to 1800 psig	±1 percent full scale	-
Drive water pump suction Pressure	Pressure switch	30 in. Hg to 10 psig	±1 percent full scale	25 in. Hg, decreasing
Drive water system flow rate	Flow indicator	0 to 100 gpm	±1 percent full scale	-
Drive water header flow rate	Flow indicator	0 to 8 gpm	±2 percent full scale	-
Cooling water header flow rate	Flow indicator	0 to 80 gpm	±2 percent full scale	-
Stabilizing flow rate	Flow indicator	0 to 8 gpm	±0.5 percent full scale	-

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TABLE 7.7-1 REACTOR MANUAL CONTROL SYSTEM INSTRUMENT SPECIFICATIONS

<u>Measured Variable</u>	<u>Instrument Type</u>	<u>Instrument Range</u>	<u>Accuracy^d</u>	<u>Trip Setting^a</u>
Control rod drive Temperature	Temperature switch and monitor	0 to 500°F	±1 percent full scale	250°F
Control rod position (normal range)	Reed switches	Full in to full out every 3 in.	NA ^b	-
Control rod drive overtravel (withdraw direction)	Reed switches	NA	NA	2 in. beyond full out position
Insert bus time energized (for rod insertion)	Timer	-	-	2.8 sec
Insert bus time energized (for rod withdrawal)	Timer	-	-	0.62 sec
Withdraw bus time energized (for rod withdrawal)	Timer	-	-	1.5 sec
Settle bus time energized (for rod insertion)	Timer	-	-	4.4 sec
Settle bus time energized (for rod withdrawal)	Timer	-	-	5.8 sec
Rod block scram discharge volume high water level	Level switch		±3 in.	25 gal ^c
Rod block neutron monitoring system trip channels	Section 7.1.2.1.4, Neutron Monitoring System			
Rod block rod worth minimizer	Subsection 7.6.1.20, Rod Worth Minimizer System			
Rod block flow upscale	Section 7.1.2.1.4, Neutron Monitoring System			

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TABLE 7.7-1 REACTOR MANUAL CONTROL SYSTEM INSTRUMENT SPECIFICATIONS

<u>Measured Variable</u>	<u>Instrument Type</u>	<u>Instrument Range</u>	<u>Accuracy</u> ^d	<u>Trip Setting</u> ^a
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^a Nominal setting - see Technical Specifications for setpoint and allowable values.

^b NA = not applicable.

^c For 1/2 total instrument volume.

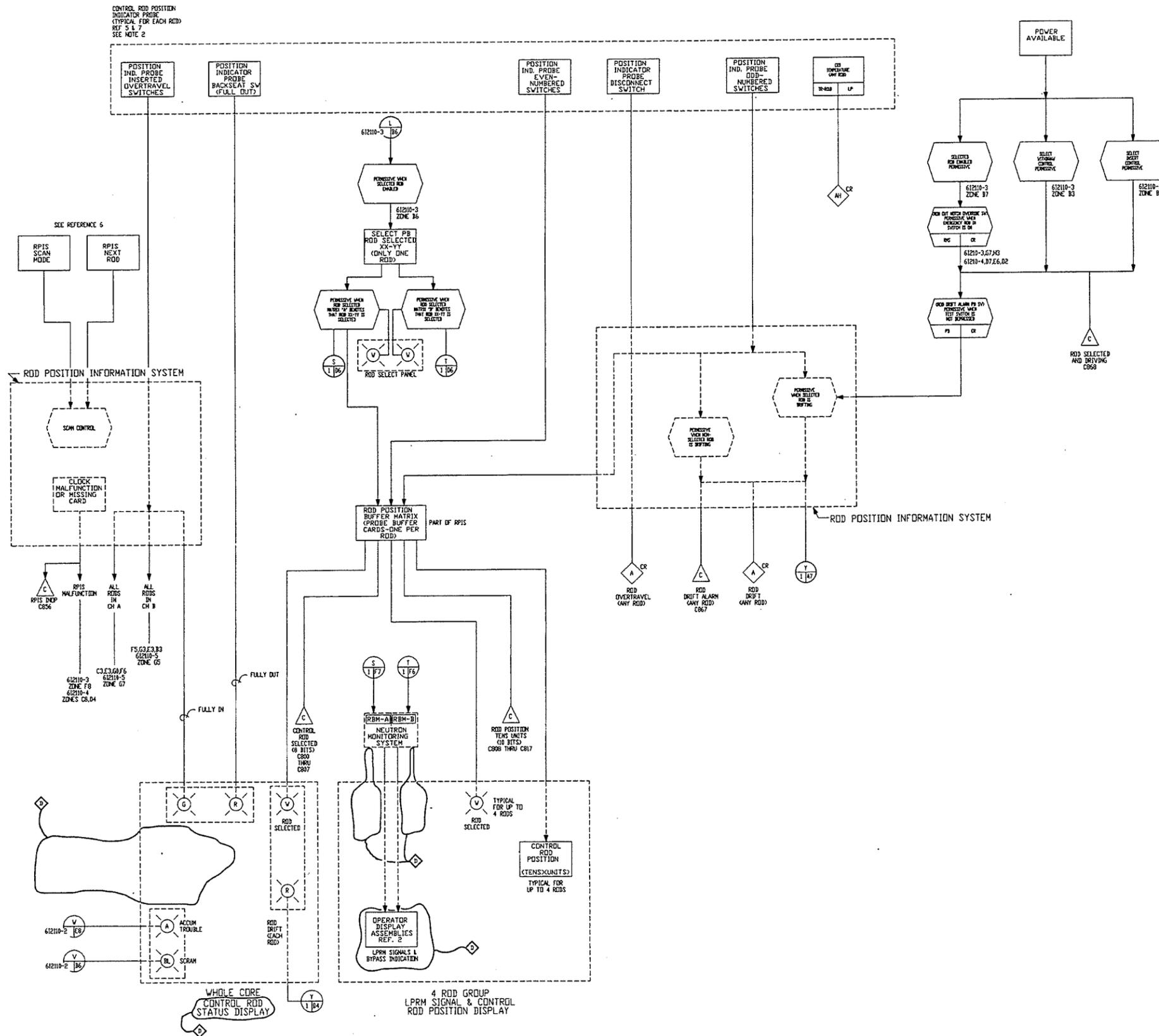
^d The instrument accuracy information provided in the UFSAR tables is a bounding value.

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TABLE 7.7-2 PROCESS INSTRUMENT ALARMS OFFGAS SYSTEM

<u>Parameter</u>	<u>Main Control Room</u>	
	<u>Indicated</u>	<u>Recorded</u>
Preheater discharge temperature – low		X
Recombiner catalyst temperature – high/low		X
Offgas condenser drain well level – high	X	
Offgas condenser gas discharge temperature – high		X
H ₂ analyzer – high		X
Precooler temperature – high		X
Chiller pressure ^a – high	X	
Charcoal bed temperature – high		X
Absolute filter pressure ^a – high	X	
Delay pipe pressure – high		X
Sandfilter pressure ^a – high	X	
Offgas system flow – high		X

^a Differential pressure.



- NOTES**
- THE ROD CIRCUIT SHALL BE DESIGNED TO INSURE THAT:
 - ONLY ONE ROD MAY BE SELECTED AT A TIME
 - WHEN MOVEMENT IS INITIATED, THE SELECTED ROD SHALL BE SEALED IN TO INHIBIT SELECTION OF ANOTHER ROD DURING THE MOVEMENT CYCLE (EXCEPT FOR LOSS OF CONTROL CIRCUIT POWER)
 - EACH CONTROL ROD, AS IT TRAVELS UP/INSERTED OR DOWN/WITHDRAWN PASSES A NUMBER OF SWITCHES. THE TOP TWO POSITION SWITCHES ARE CALLED "OVERTRAVEL" AND THE BOTTOM TWO SWITCHES ARE CALLED "WITHDRAW" (BACKSEAT & DISCONNECT). SWITCHES IN BETWEEN ARE DIVIDED INTO ODD (CRB7) AND EVEN (LATCH) POSITIONS. AS THE ROD TRAVELS OVER ANY SWITCH AN INDICATING SIGNAL IS ACTIVATED.
 - TOP CONFIGURATION OF PANEL MOUNTED OPERATORS AND INDICATORS SEE GH-2290-2, REFERENCE IS PURCHASE SPECIFICATION DATA SHEET SECTION 24.
 - SCRAM DISCHARGE VENT AND DRAIN VALVE CONTROL ROD INDICATORS SHALL INDICATE OPEN WHEN BOTH VALVES ARE OPEN AND CLOSED WHEN EITHER VALVE IS CLOSED.
 - ROD WORTH MONITORER FORWARD CONTROLLED LOGIC IS SPECIFIED IN THE RWM DESIGN SPECIFICATION (REF 14).

- REFERENCE DOCUMENTS**
- CONTROL ROD DRIVE HYD SYSTEM PLUD 642081, 645449
 - NEUTRON MONITORING SYSTEM IED 612146-1, 2
 - FEEDWATER CONTROL SYSTEM IED 612126-1
 - FEEDWATER CONTROL SYSTEM DESIGN SPECIFICATION 2243086, RI-884, RI-885
 - CONTROL ROD DRIVE HYDRAULIC SYSTEM DESIGN SPEC 2246249, RI-3784
 - PROCESS COMPUTER SYSTEM INPUT/OUTPUT REQUIREMENTS 2246958, RI-2892
 - POSITION INDICATOR PROBE CONNECTION DIAGRAM 10482506
 - LOGIC SYMBOLS 20944754; RI-20
 - NEUTRON MONITORING SYSTEM LOGIC DIAGRAM 612140
 - REACTOR PROTECTION SYSTEM IED 612156-1, 2, 3, 4
 - PIPING & INSTRUMENT SYMBOLS 1477567; RI-2507
 - GENERAL REQUIREMENTS FOR INSTR & ELECT EQUIP 2141768; RI-449
 - NUMAC-RWM OPERATION AND MAINTENANCE INSTRUCTIONS (GK 89430) VNU-731
 - NUMAC ROD WORTH MONITORER DESIGN SPECIFICATION 23451004; R4-758

- LEGEND**
- SWITCHGEAR DEVICE FUNCTION NUMBER USAS SPECIFICATION C372
 - RWM ROD WORTH MONITORER
 - LPRM LOCAL POWER RANGE MONITOR
 - RPIS ROD POSITION INFORMATION SYSTEM
 - LSP LOW POWER SET POINT
 - LPAP LDV POWER ALARM POINT
 - △ INPUT TO COMPUTER
 - NSC NEUTRON MONITORING SYSTEM
 - APRM AVERAGE POWER RANGE MONITOR
 - RBM ROD BLOCK MONITOR
 - IRM INTERMEDIATE RANGE MONITOR
 - CHC COORDINATED MANUAL CONTROL SWITCH
 - PB PUSHBUTTON SWITCH
 - AM AMPHETER
 - MATCH NUMBER DESIGNATION
 - ZONE DESIGNATION
 - SHEET NUMBER OR REFERENCE DRAWING
 - MATCH CIRCLE

SIGNAL	NOMINAL SPAN SETTING (SECONDS)	AVAILABLE RANGE (SECONDS)
TIME DELAY TO UNLATCH CONTROL	0.02 ±0.02	SEE GK-346528
UNLATCH CONTROL	0.65 ±0.15	SEE GK-346528
SWITCHING INTERVAL FROM UNLATCH CONTROL TO ROD WITHDRAW CONTROL	0 ±0.15	SEE GK-346528
ROD WITHDRAW CONTROL	1.50 ±0.20	SEE GK-346528
SWITCHING OVERLAP BETWEEN ROD WITHDRAW CONTROL & SETTLE CONTROL	0.07 ±0.01 -0.00	SEE GK-346528
SETTLE CONTROL	5.90 ±0.50	SEE GK-346528
TIME DELAY TO ROD INSERT CONTROL	0.02 ±0.02	SEE GK-346528
ROD INSERT CONTROL	2.90 ±0.20	SEE GK-346528
SWITCHING OVERLAP BETWEEN ROD INSERT CONTROL & SETTLE CONTROL	0.00	SEE GK-346528
SETTLE CONTROL	4.40 ±0.50	SEE GK-346528
CYCLE AUXILIARY	+8.0	SEE GK-346528
INSERT/WITHDRAW SELECT ENABLE	+8.5	SEE GK-346528

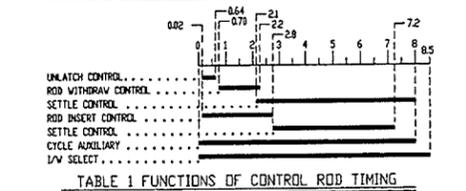
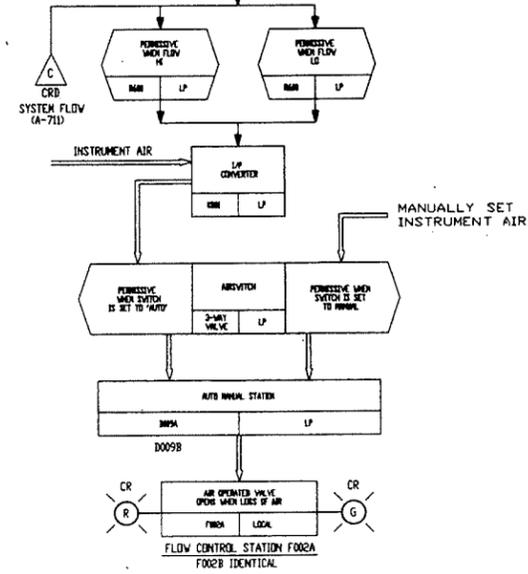
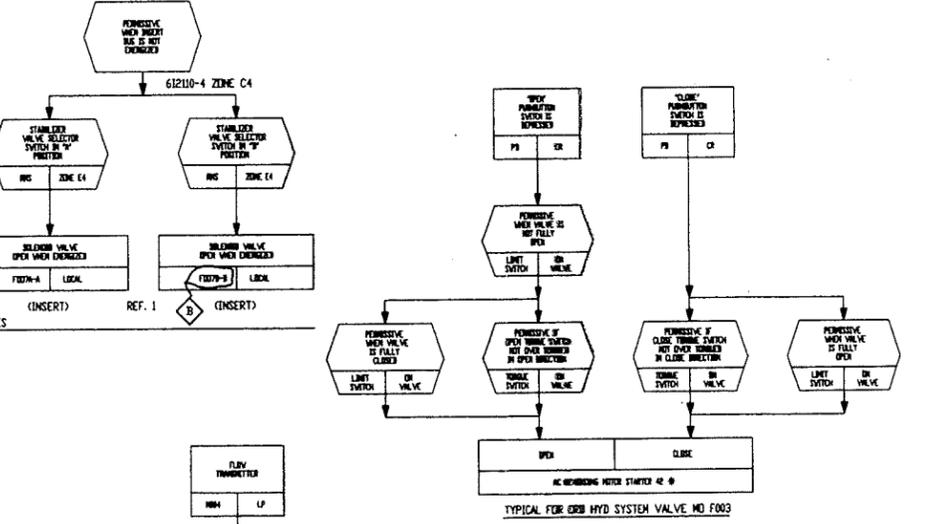
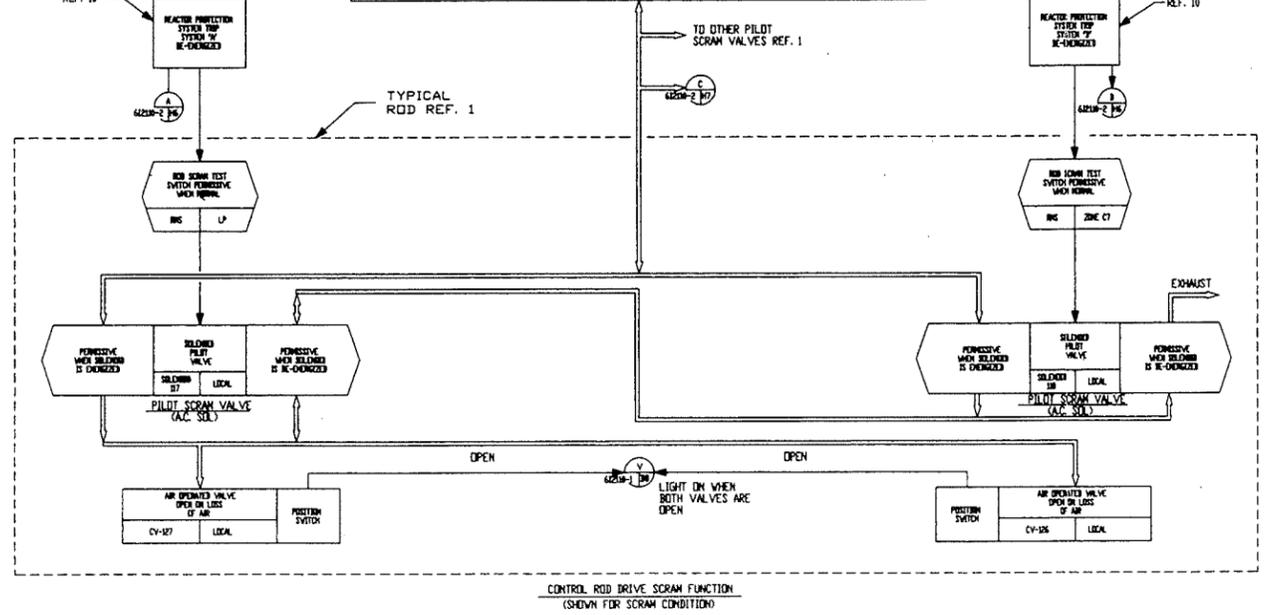
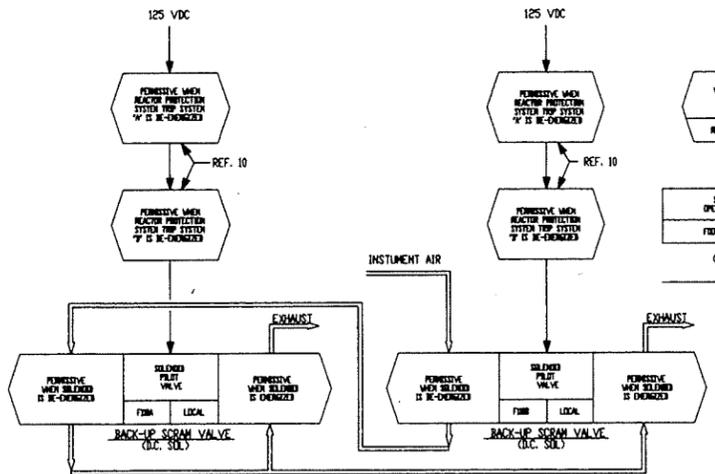
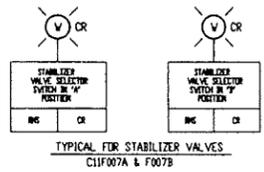
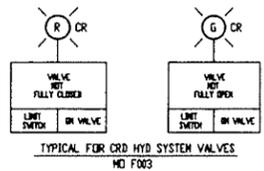
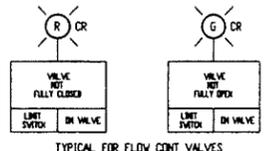
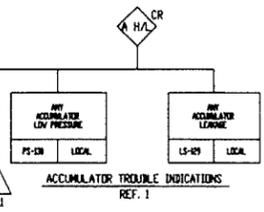
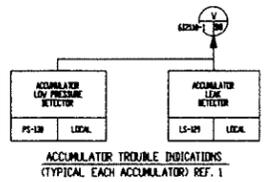
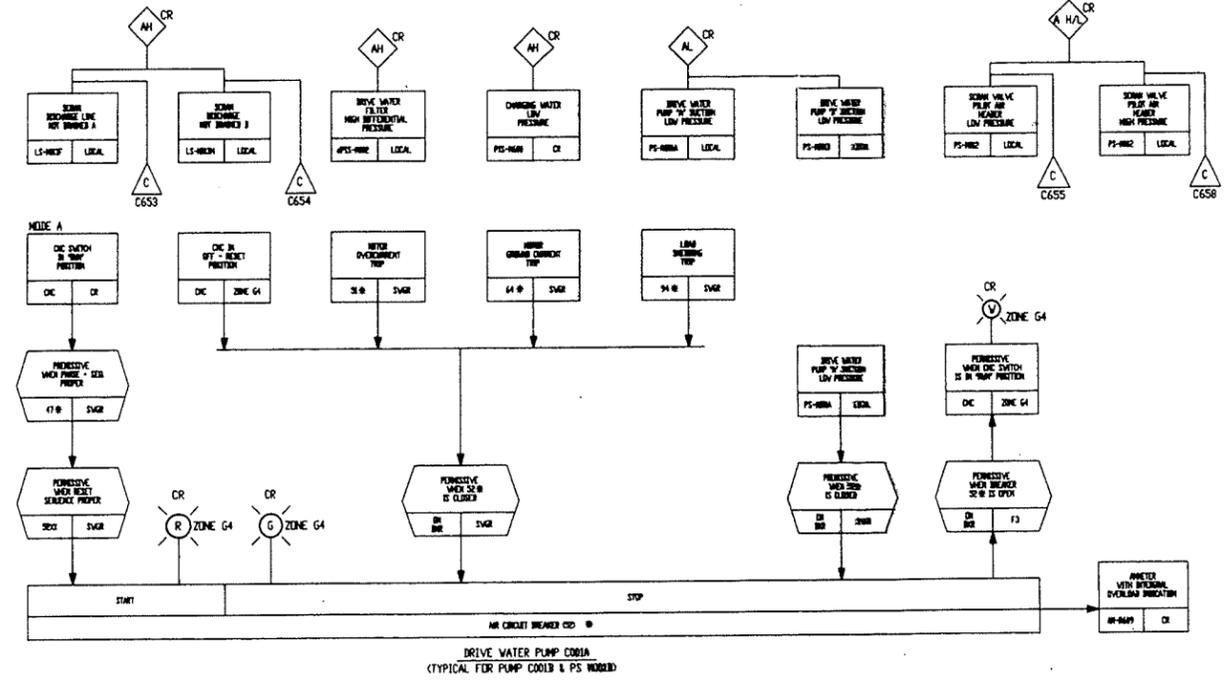
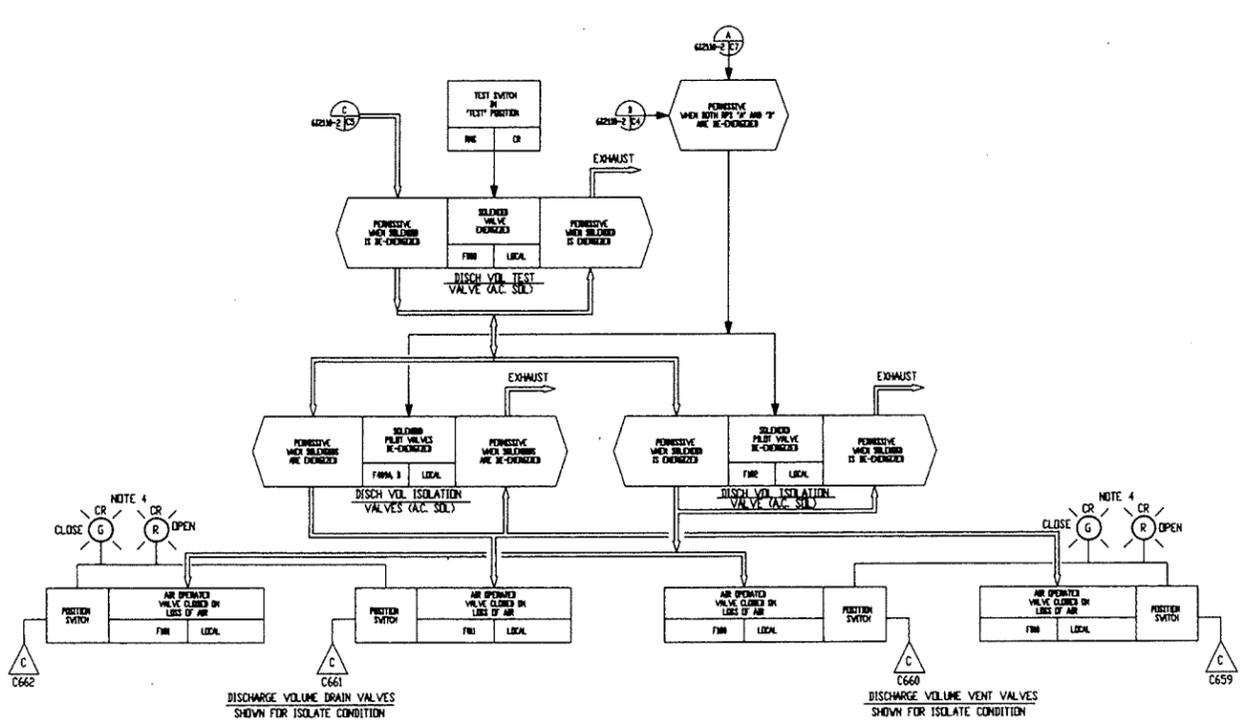


TABLE 1 FUNCTIONS OF CONTROL ROD TIMING

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FIGURE 7.7-1, SHEET 1
 CONTROL ROD DRIVE HYDRAULIC SYSTEM FCD



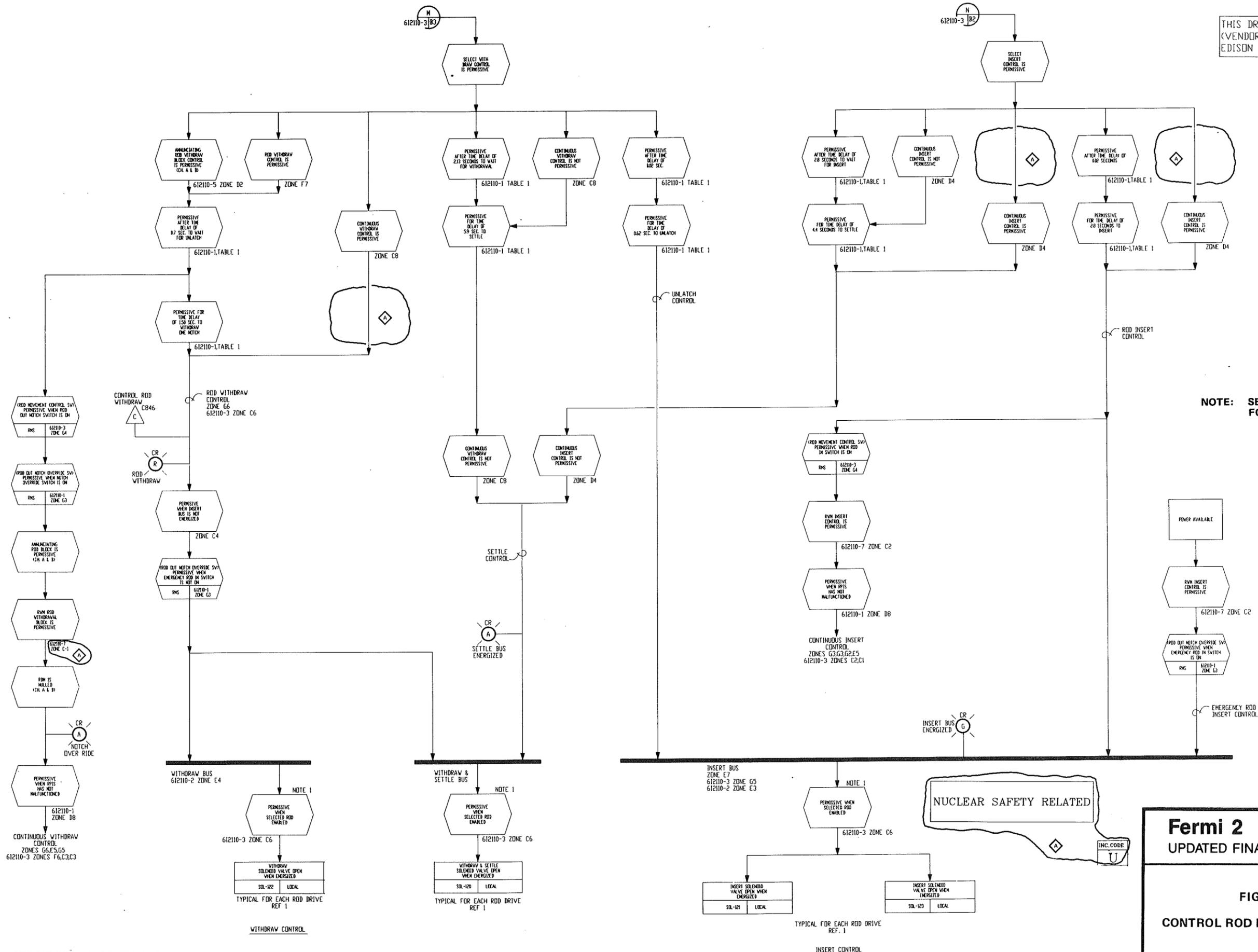
NOTE: SEE 61721-2110-1 (FIGURE 7.7-1, SHEET 1) FOR NOTES, REFERENCES AND LEGEND. THIS FIGURE IS NOT UPDATED PERIODICALLY

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FIGURE 7.7-1, SHEET 2

CONTROL ROD DRIVE HYDRAULIC SYSTEM FCD

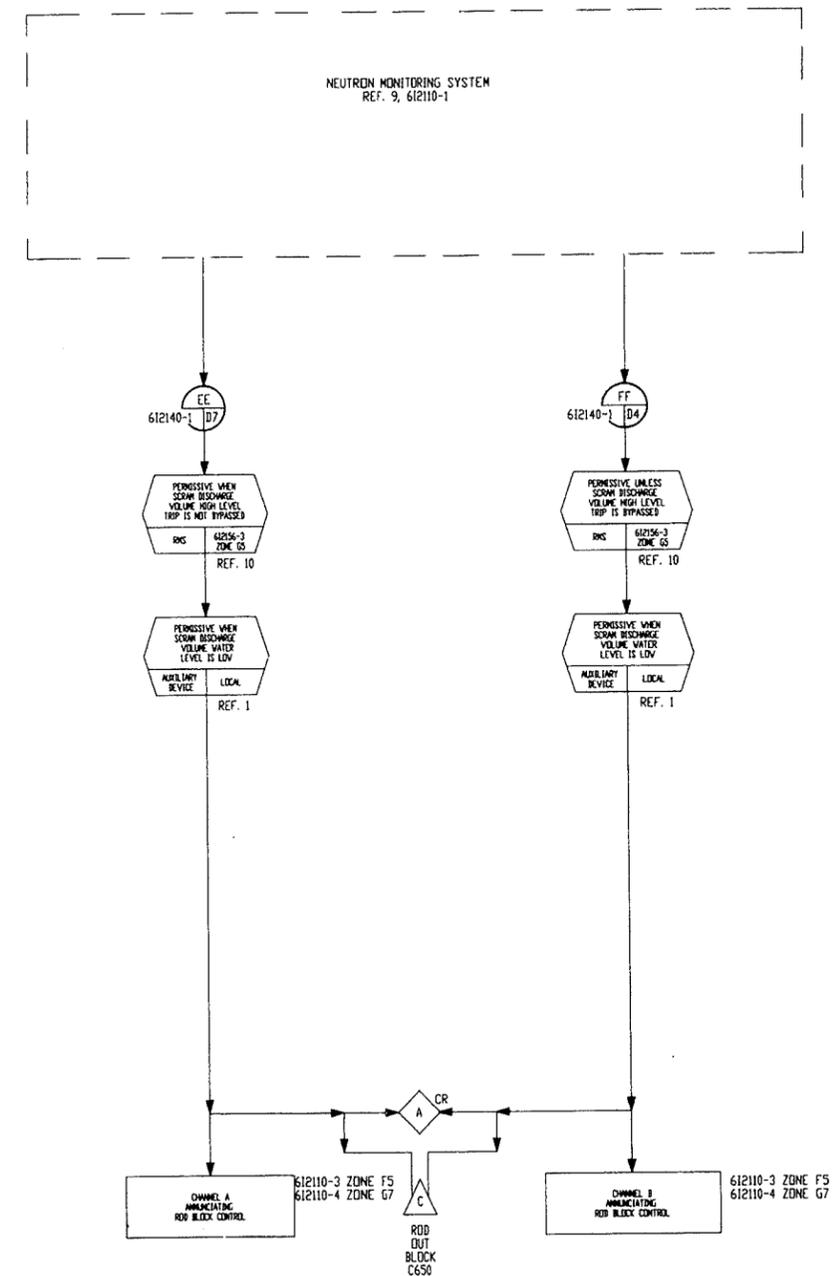
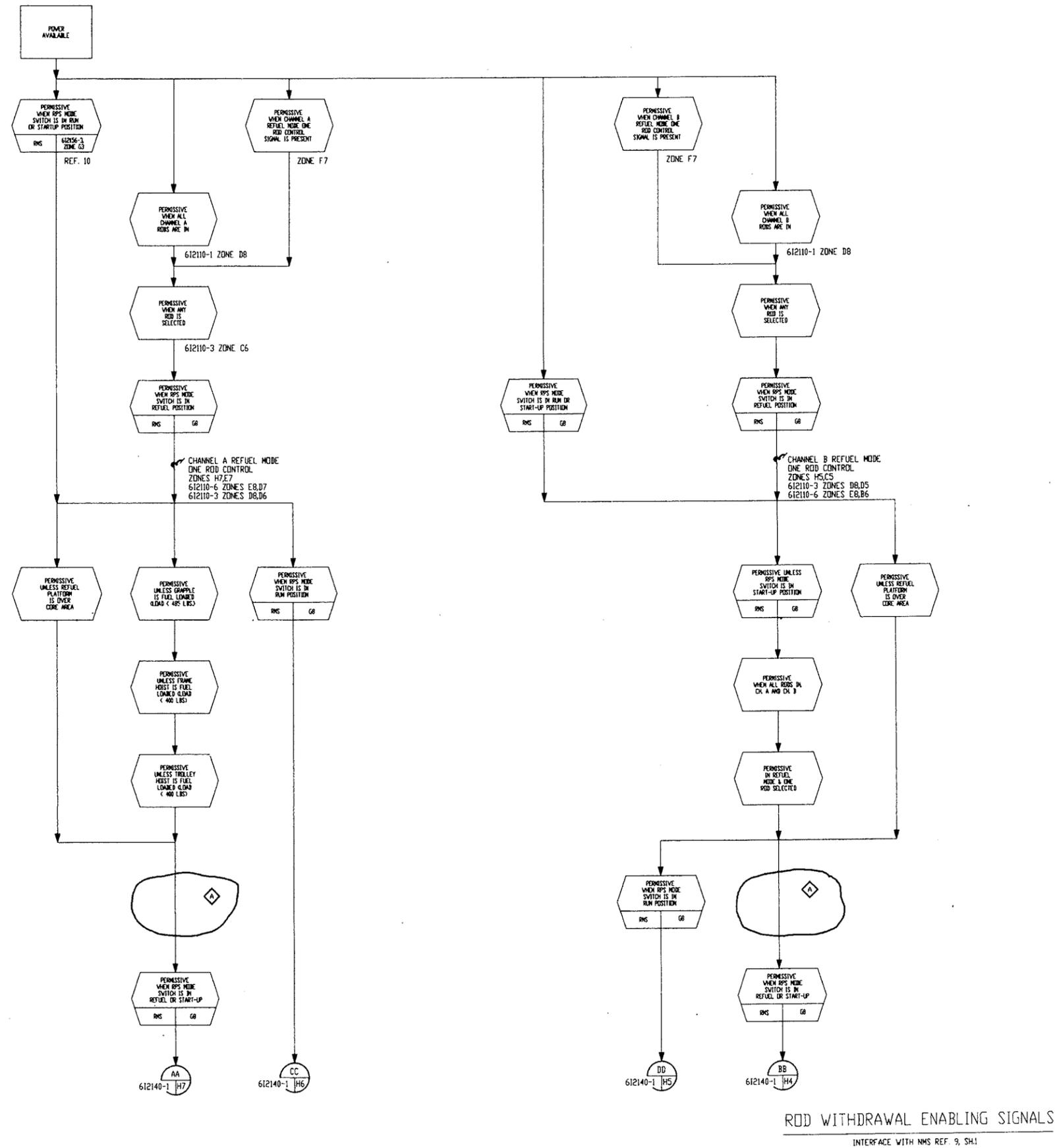
THIS DRAWING SUPERSEDES EDISON'S SITE MASTER
(VENDOR DRAWING 729E618AB SHT 4 REV 2) OF
EDISON FILE NO.R1-80



NOTE: SEE 61721-2110-1 (FIGURE 7.7-1, SHEET 1)
FOR NOTES, REFERENCES AND LEGEND.

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FIGURE 7.7-1, SHEET 4
CONTROL ROD DRIVE HYDRAULIC SYSTEM FCD

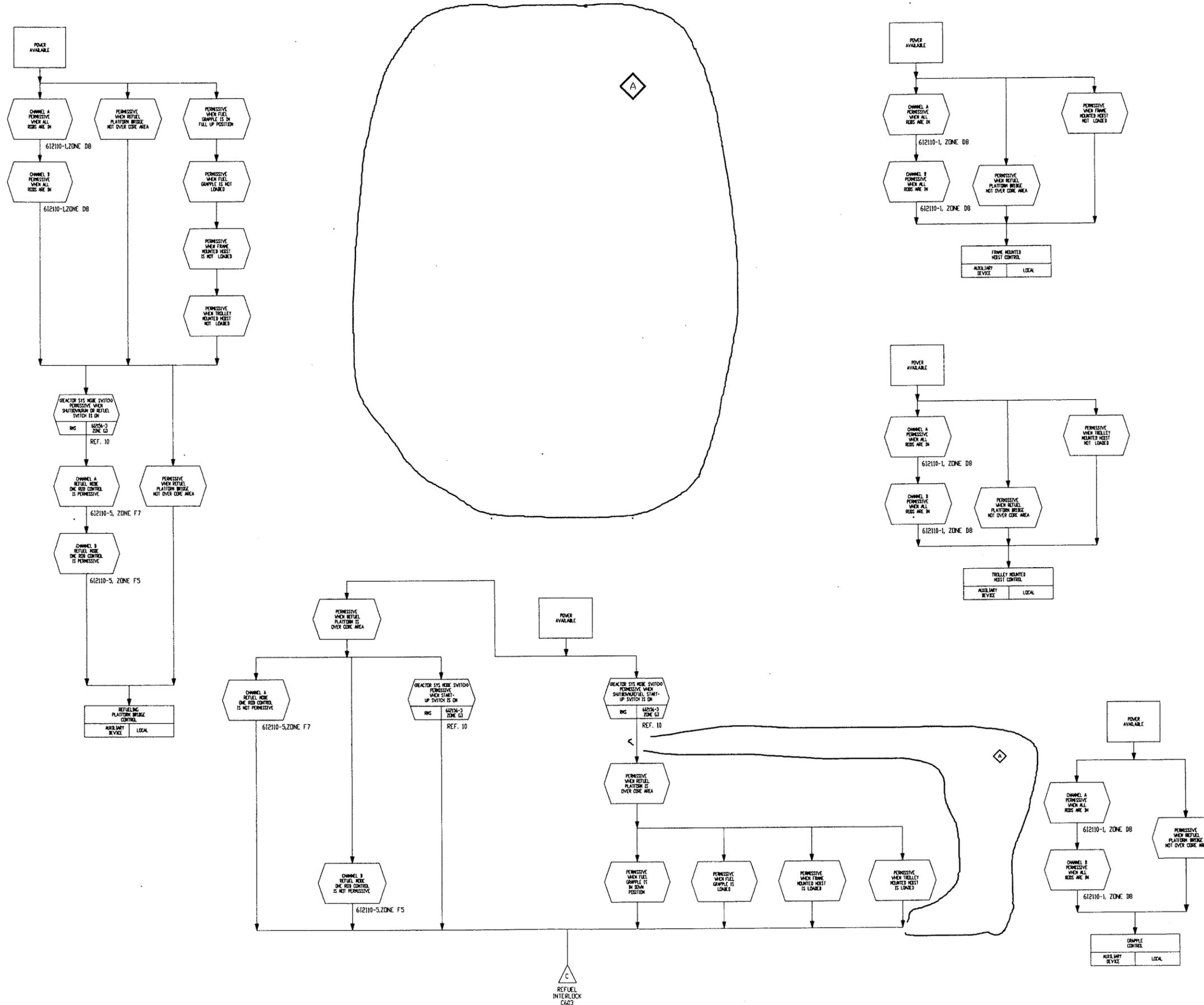


NOTE: SEE 61721-2110-1 (FIGURE 7.7-1, SHEET 1) FOR NOTES, REFERENCES AND LEGEND.

THIS FIGURE IS NOT UPDATED PERIODICALLY

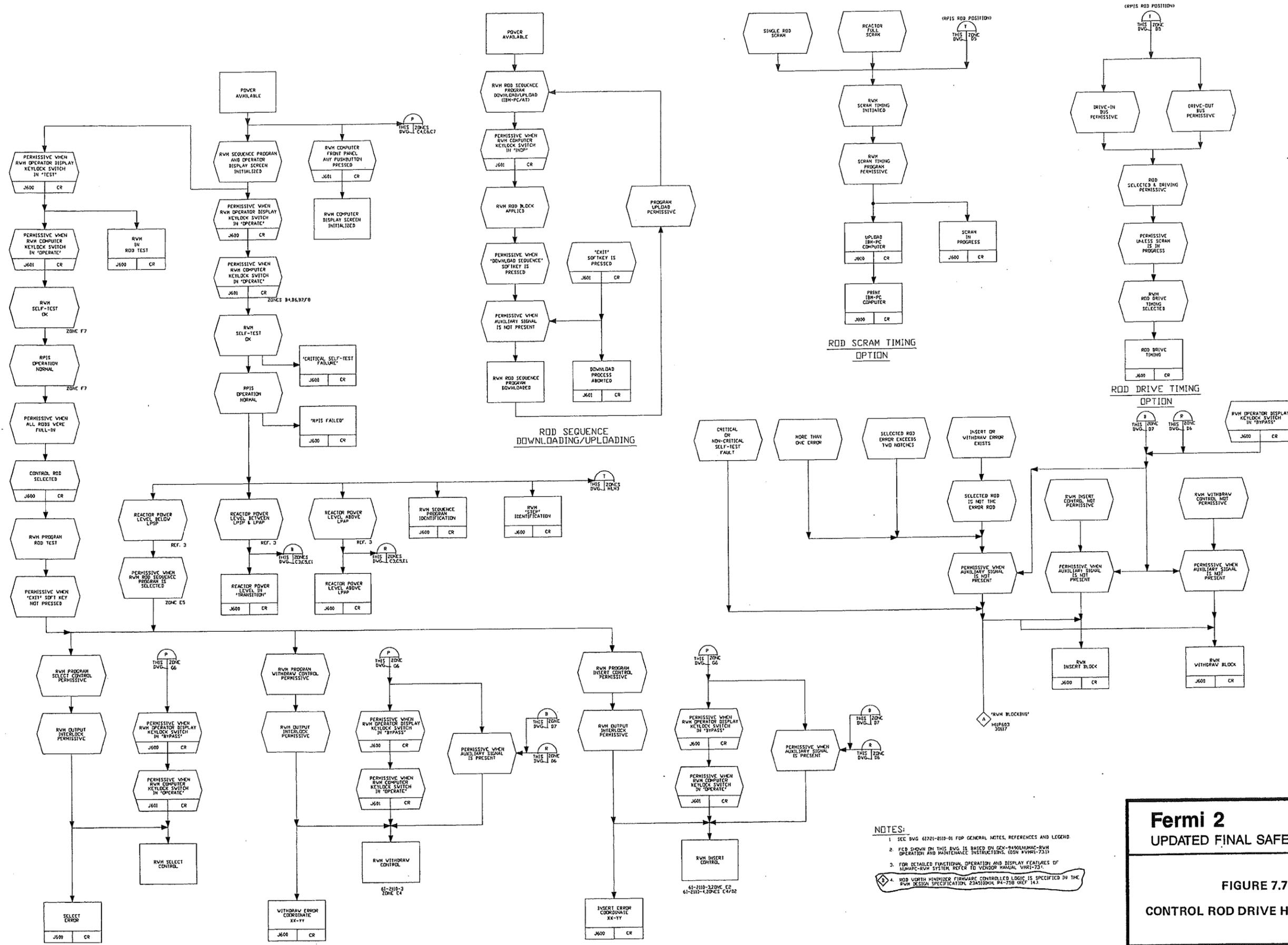
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FIGURE 7.7-1, SHEET 5
CONTROL ROD DRIVE HYDRAULIC SYSTEM FCD



NOTE: SEE 61721-2110-1 (FIGURE 7.7-1, SHEET 1) FOR NOTES, REFERENCES AND LEGEND.
THIS FIGURE IS NOT UPDATED PERIODICALLY

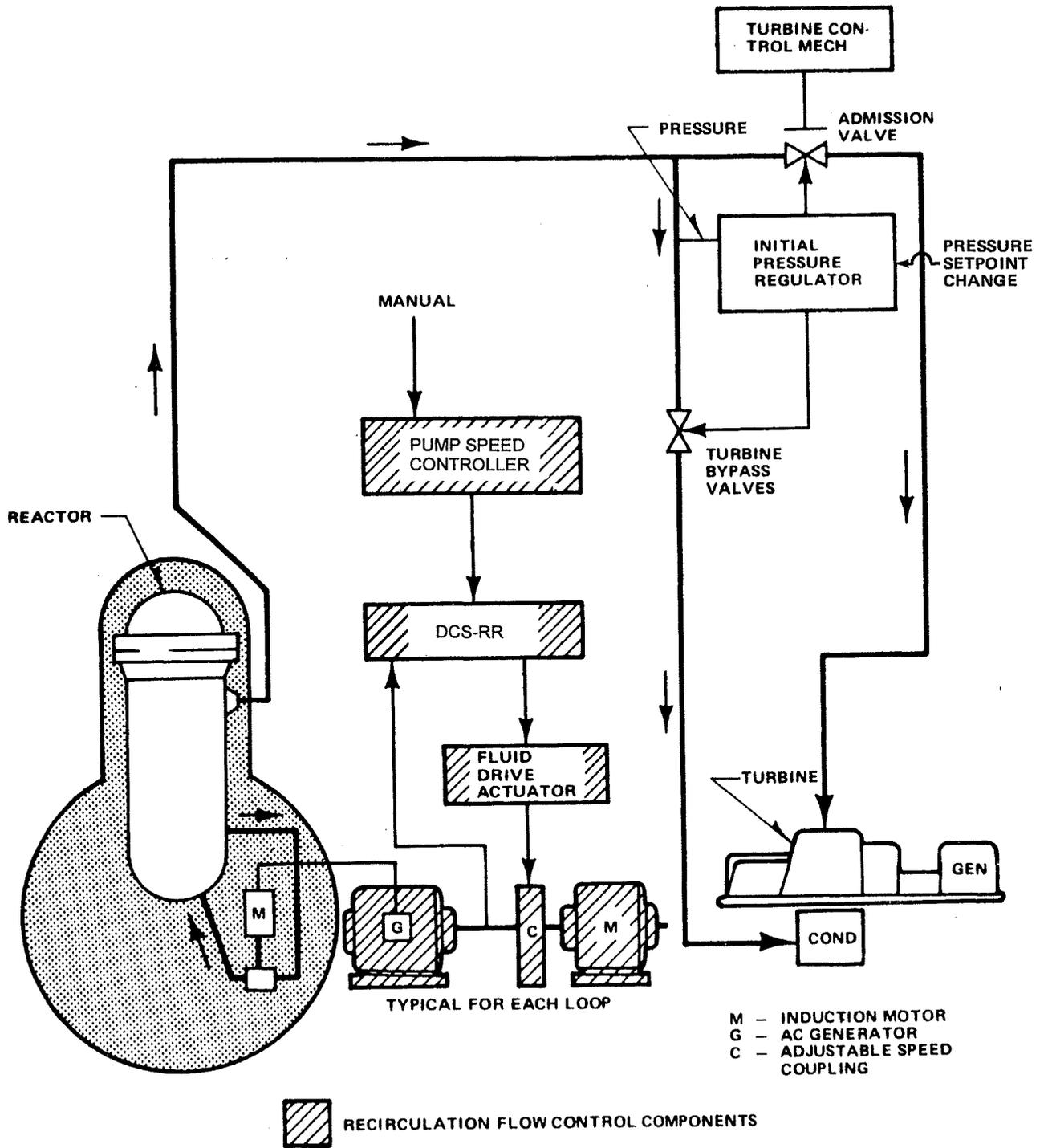
<p>Fermi 2 UPDATED FINAL SAFETY ANALYSIS REPORT</p>
<p>FIGURE 7.7-1, SHEET 6 CONTROL ROD DRIVE HYDRAULIC SYSTEM FCD</p>



- NOTES:**
1. SEE DWG 61721-2110-01 FOR GENERAL NOTES, REFERENCES AND LEGEND.
 2. FCD SHOWN ON THIS DWG IS BASED ON GEN-849043MAC-RVM OPERATION AND MAINTENANCE INSTRUCTIONS. (OSH 84901-731)
 3. FOR DETAILED FUNCTIONAL OPERATION AND DISPLAY FEATURES OF NONMPC-RVM SYSTEM REFER TO VENDOR MANUAL, VMSI-121.
 4. ROD WITH NUMBER FIRMWARE CONTROLLED LOGIC IS SPECIFIED IN THE RVM DESIGN SPECIFICATION, 23010004 PA-758 (REF 14).

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FIGURE 7.7-1, SHEET 7
CONTROL ROD DRIVE HYDRAULIC SYSTEM FCD

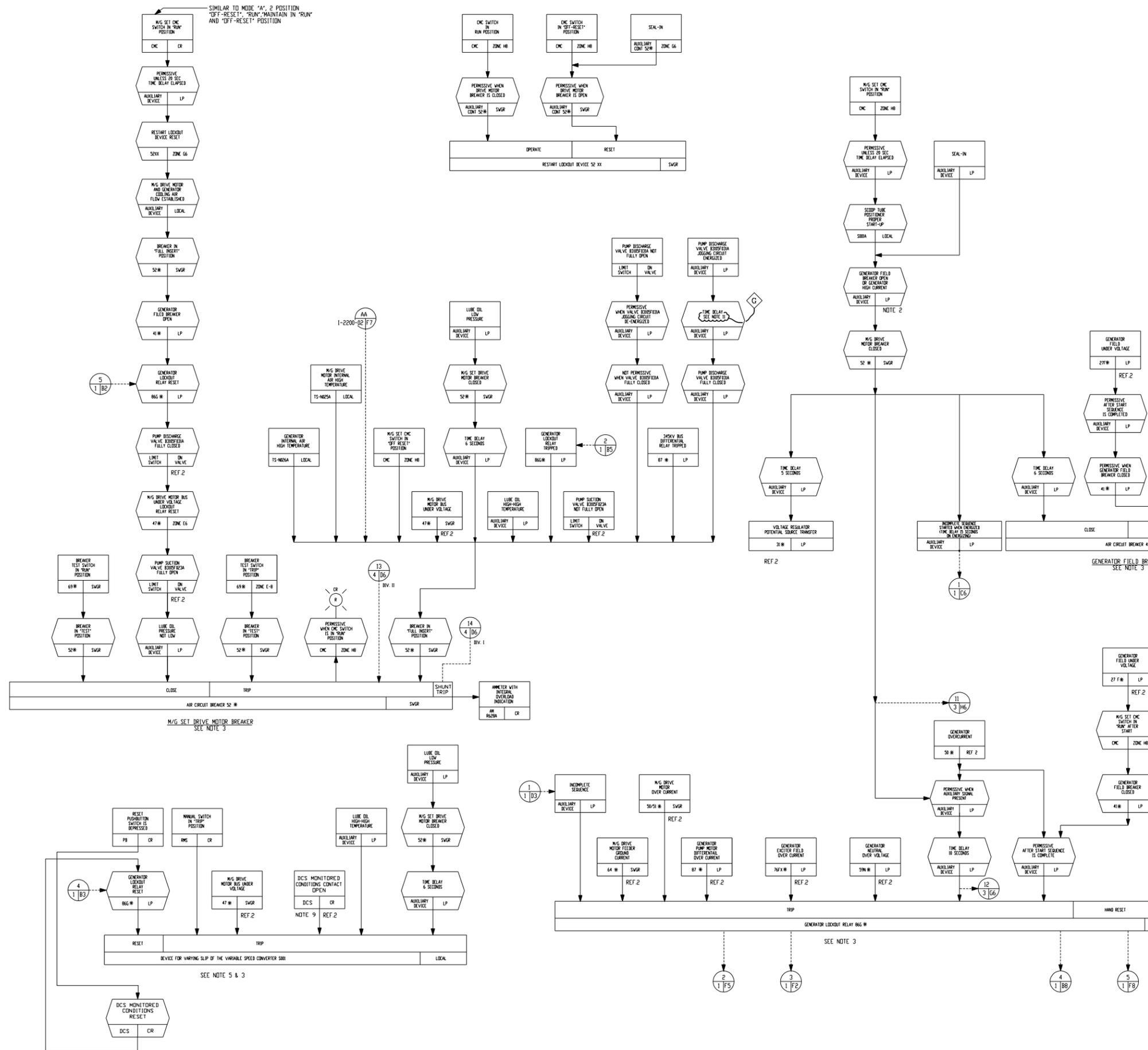


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FIGURE 7.7-2

RECIRCULATION FLOW CONTROL ILLUSTRATION



- NOTES:**
- AUXILIARY RELAYS, SWITCHES ETC. ARE NOT SHOWN EXCEPT WHERE NECESSARY TO CLARIFY THE FUNCTION.
 - THIS DEVICE SENSES WHEN THE PUMP HAS STARTED AND FUNCTIONS AFTER 2 SECOND TIME DELAY TO OPEN THE "SEAL-IN" TRANSFER VOLTAGE SUPPLY TO THE VOLTAGE REGULATOR AND INHIBIT THE INCOMPLETE SEQUENCE.
 - FUNCTION IS SHOWN FOR RECIRCULATION SYSTEM "A" (EXCEPT AS NOTED) AND IS TYPICAL FOR RECIRCULATION SYSTEM "B".
 - DELETED
 - THE TRIP SIGNALS SHALL CAUSE THE DEVICE TO MAINTAIN SLIP AT A VALUE EQUAL TO THE SLIP WHICH IS PRESENT AT THE TIME THE TRIP SIGNAL WAS RECEIVED. LOSS OF MEDIUM OF CONTROL (ELECTRICAL, HYDRAULIC, ETC.) SHALL ALSO CAUSE SLIP TO BE MAINTAINED AT THE SAME VALUE.
 - DELETED
 - FOR CONFIRMATION OF PANEL MOUNTED OPERATORS AND INDICATORS SEE DWG. 6M-2090, REF. 7, PURCHASE SPECIFICATION DATA SHEET SECT. 2.4
 - THIS DRAWING SUPERSEDES EDISON'S SITE MASTER (VENDOR DRAWING 72962626AB, SHT.1, REV.5) OF EDISON FILE NO. R1-200.
 - DISTRIBUTED CONTROL SYSTEM (DCS) PROVIDES OUTPUT TO LOCK SCOOP TUBE POSITIONERS FROM ABNORMAL CONDITIONS. REFER TO DWG. I-2106-02 FOR DETAILS.
 - UNLESS OTHERWISE SHOWN: ALL INSTRUMENT PIS NUMBERS ARE PREFIXED B31 ALL VALVE AND EQUIPMENT PIS NUMBERS ARE PREFIXED B3100
 - SEE CECD FOR SETPOINT INFORMATION

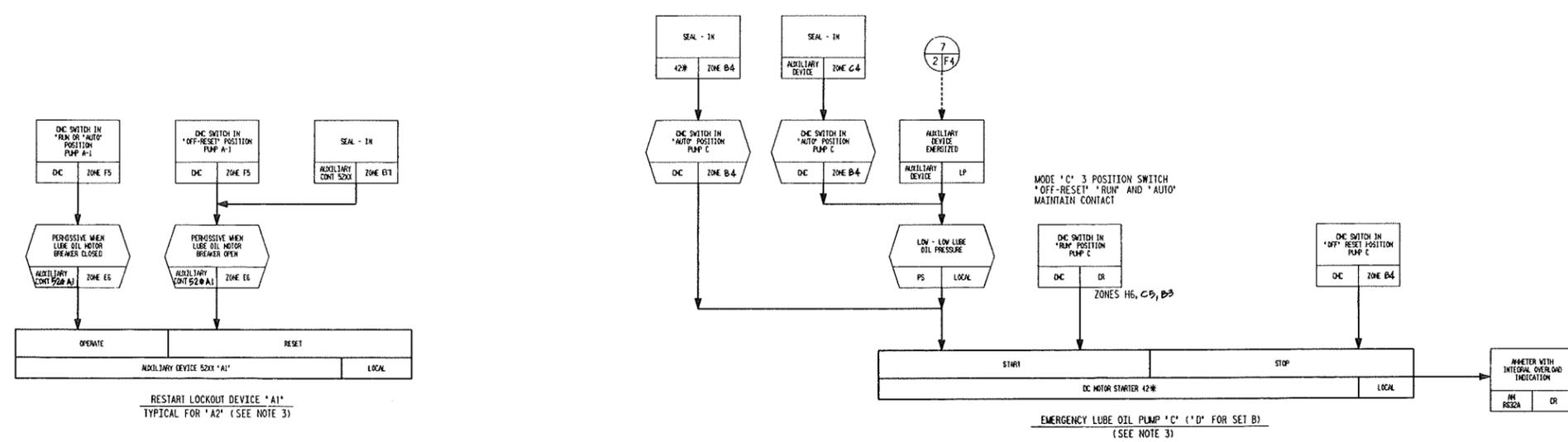
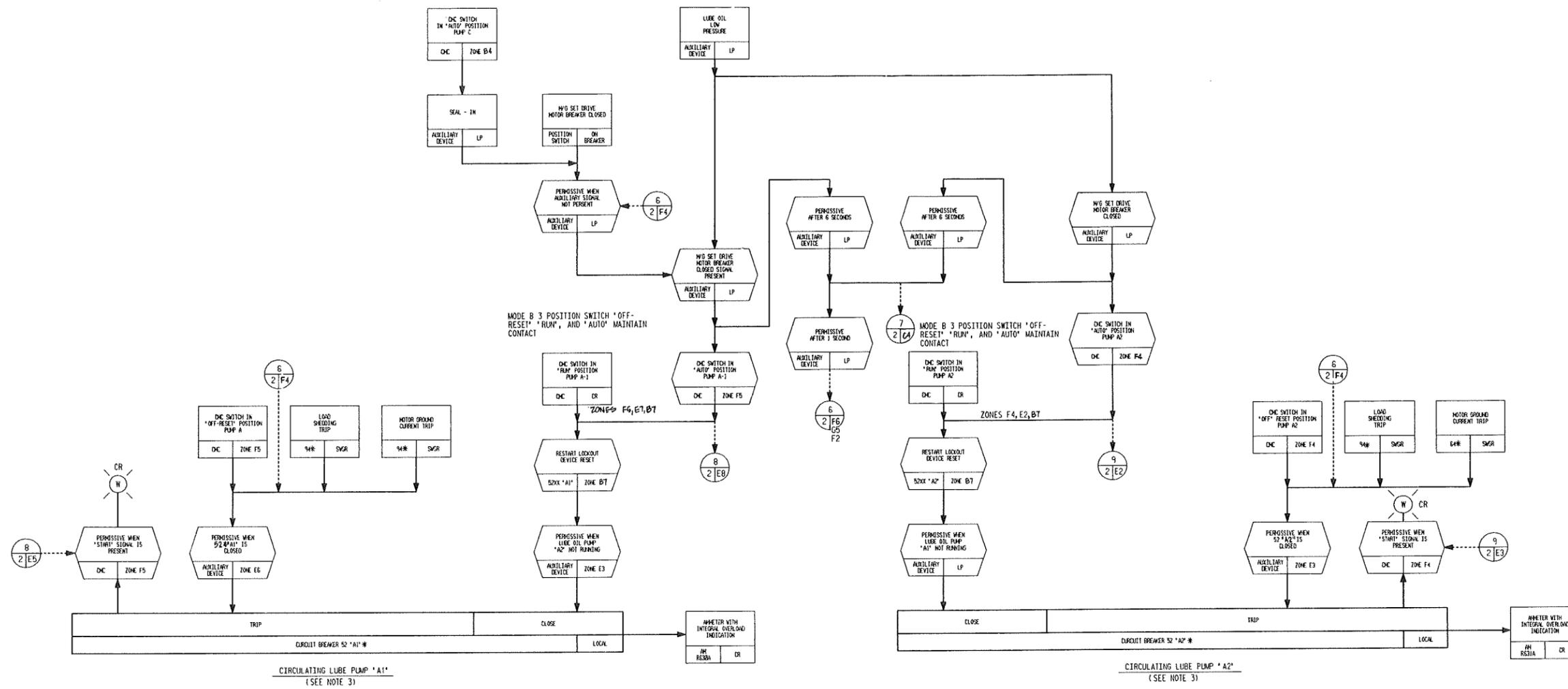
- REFERENCE DOCUMENTS**
- REACTOR RECIRCULATION SYSTEM DESIGN SPEC...2241458; R1-124
 - REACTOR RECIRCULATION SYSTEM P&ID ...M-2833, I-2106-01,02
 - NUCLEAR BOILER SYSTEM P&ID ...M-2089, M-2090, M-5538
 - RESIDUAL HEAT REMOVAL SYSTEM LOGIC DIAGRAM ...I-2200-01
 - LOGIC SYMBOLS ...209A4756; R1-20
 - REACTOR RECIRCULATION SYSTEM ELEMENTARY DIAGRAM...I-2105-01
 - GENERAL REQUIREMENTS FOR INSTR & ELEC EQUIPMENT...

- LEGEND:**
- AM = AMMETER
 - PB = PUSHBUTTON SWITCH
 - CMC = COORDINATED MANUAL CONTROL SWITCH
 - SWGR = M/G SET DRIVE MOTOR HIGH VOLTAGE SWITCH GEAR
 - * = SWITCHGEAR DEVICE FUNCTION NUMBER ANSI SPEC. C37.2
 - DCS = DISTRIBUTED CONTROL SYSTEM
- MATCH NUMBER DESIGNATION
 ZONE DESIGNATION
 SHEET NUMBER OR REFERENCE DRAWING
 MATCH CIRCLE
 LAST NO. USED 14

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FIGURE 7.7-3, SHEET 1

REACTOR RECIRCULATION SYSTEM FCD

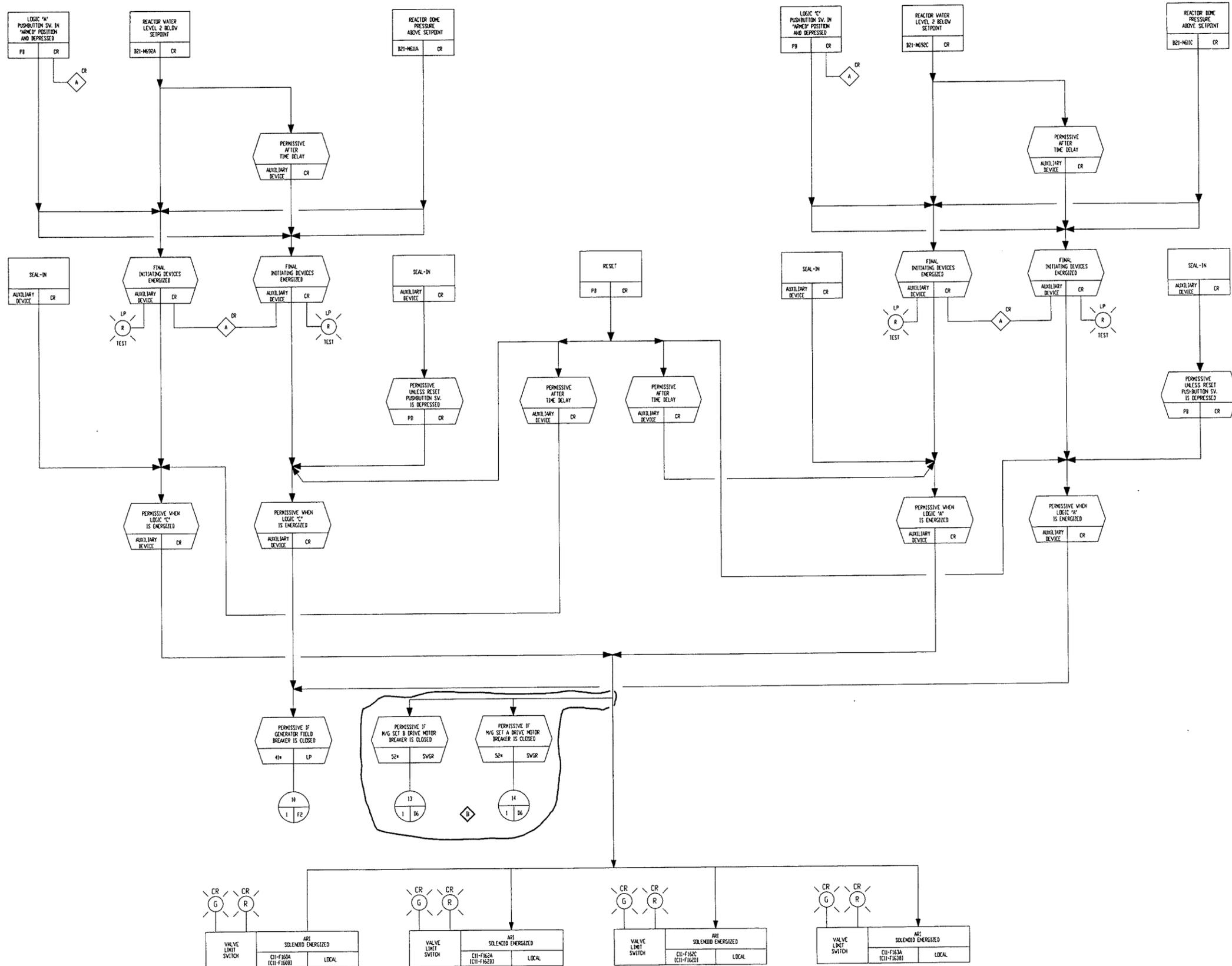


NOTE: SEE 61721-2100-1 (FIGURE 7.7-3, SHEET 1) FOR NOTES, REFERENCES AND LEGEND.

THIS FIGURE IS NOT UPDATED PERIODICALLY

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FIGURE 7.7-3, SHEET 2
REACTOR RECIRCULATION SYSTEM FCD

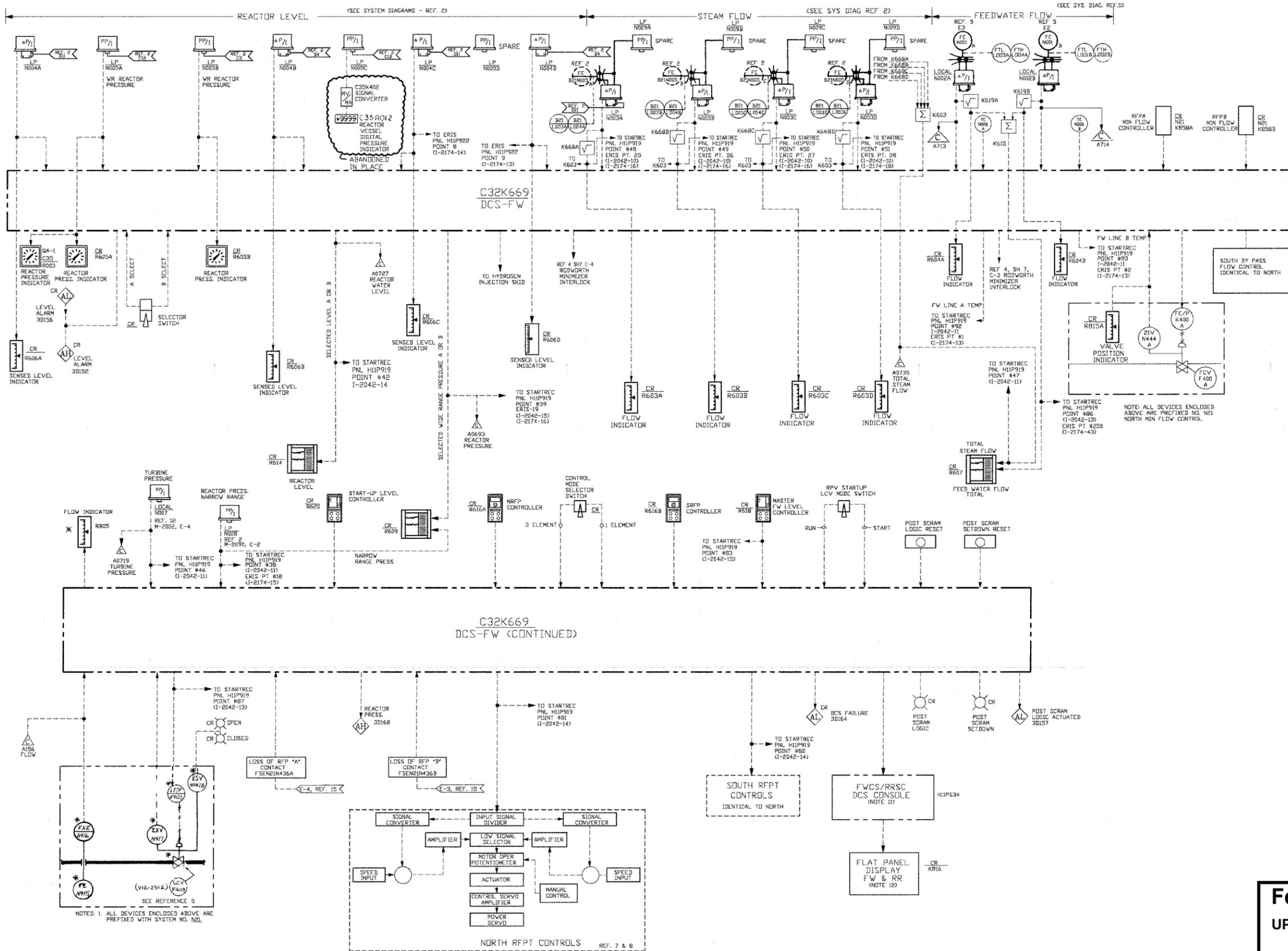


NOTES:
 1. FUNCTION IS SHOWN FOR ATWS (ONLY) DIVISION 1 LOGICS A & C1 AND IS TYPICAL FOR DIV1 LOGICS B & D1
 2. FOR OTHER NOTES, REFERENCES, AND LEGENDS, SEE 61721-2100-01

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FIGURE 7.7-3, SHEET 4
 REACTOR RECIRCULATION SYSTEM FCD

ATWS-ARI/RPT LOOP A
 (SEE NOTE 1)



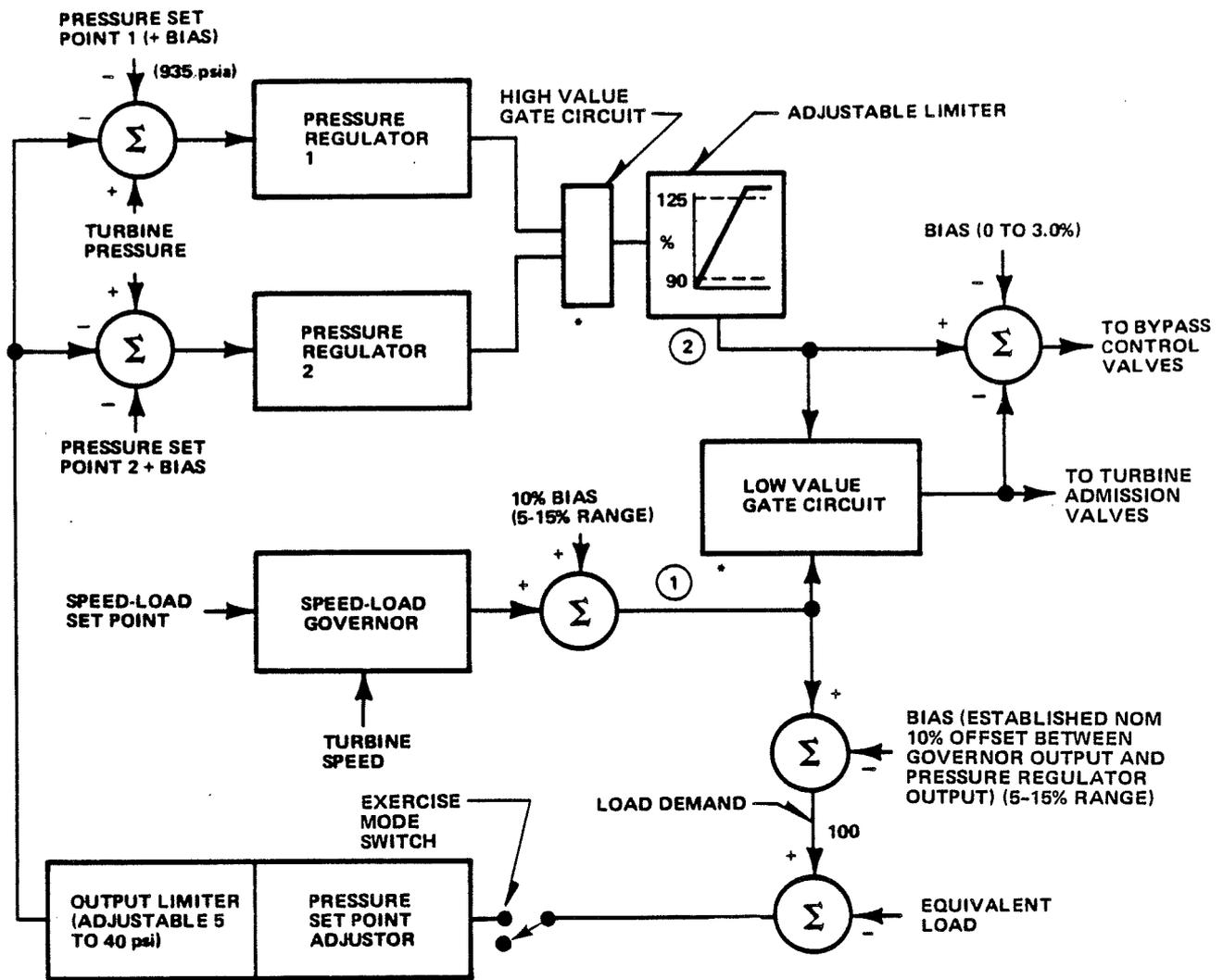
- NOTES:**
1. ALL EQUIPMENT AND INSTRUMENTS ARE PREFIXED BY SYSTEM NO. C32 UNLESS OTHERWISE SPECIFIED.
 2. FOR LOCATION & IDENTIFICATION OF INSTRUMENTS SEE CECO.
 3. FOR INSTRUMENT RANGES & SETPOINTS SEE CECO.
 4. FOR INSTRUMENT RANGES & SETPOINTS SEE CECO.
 5. THE POWER SOURCE FOR THE FEEDWATER INSTRUMENTATION AND CONTROL SYSTEM SHALL HAVE AT LEAST THE SAME DEGREE OF RELIABILITY AS THE POWER SOURCES FOR THE REACTOR/BOOSTER/CONDENSATE PUMPS.
 6. AE/CUSTOMER SHALL PROVIDE TWO NORMALLY OPEN CONTACTS PER TURBINE DRIVEN REACTOR FEED PUMP TO OPERATE WHEN TURBINE FEED PUMP UNIT HAS TRIPPED OR BEEN SHUT/DOWN. THE CONTACTS SHALL PREFERABLY BE DERIVED FROM A FLOW SWITCH MOUNTED ACROSS THE R.F.P. LOW FLOW RECIRCULATION FLOW ELEMENT PROVIDED BY THE AE/CUSTOMER OR R.F.P. DISCHARGE VALVES IF THESE AUTOMATICALLY RUN CLOSED ON R.F.P. TRIP G.E. (APED) REQUIRE THESE CONTACTS FOR INITIATION OF REACTOR RECIRCULATION PUMP RUN BACK IN EVENT OF A 1 OUT OF 2 R.F.P. TRIP AT HIGH LOADS.
 7. DELETED.
 8. DELETED.
 9. DELETED.
 10. PROCESS COMPUTER TO COMPUTE SQUARE ROOT AND PROVIDE DENSITY COMPENSATION.
 11. SEE DRAWING I-2105-21 FOR CONFIGURATION DETAILS.
 12. FLAT PANEL DISPLAY IS COMMON TO FWCS AND RRSC.

- REFERENCE DRAWINGS:**
1. FEEDWATER CONTROL SYSTEM DESIGN SPEC ----- GE 22A3086
 2. NUCLEAR BOILER SYSTEM P&ID ----- M-2089,2090
 3. REACTOR RECIRCULATION SYSTEM P&ID ----- M-2083, I-2106-01, 02
 4. CONTROL ROD DRIVE HYDRAULIC SYSTEM FCD ----- M-2081
 5. FEEDWATER SYSTEM P&ID ----- M-2023
 6. MAIN TURBINE GENERATOR TRIP LOGIC ELEM. ----- I-2339-2
 7. RFP TURBINE SPEED CONTROLLER MECH DIAG. ----- EDISON FILE S23-111
 8. RFP TURBINE SPEED CONTROLLER WIRING DIAG. ----- I-2313-03 & 04
 9. INSTRUMENT SYMBOLS ----- GE 921-D280 (R1-404)
 10. PIPING & INSTRUMENT SYMBOLS ----- EDISON FILE R1-25
 11. REACTOR FEED PUMP TURBINES TRIP LOGIC ELEM ----- I-2339-04
 12. MAIN AND HEAT STEAM SYS DIAG ----- M-2002
 13. REACTOR RECIRCULATION PUMP SPEED CONTROL DIAG ----- I-2106-02
 14. INDEX FOR STARTREC POINTS ----- I-2042-01 & 02
 15. DIAGRAM CONDENSATE SYSTEM - UNIT 2 ----- M-2004-1

- LEGEND**
- * - EQUIPMENT SUPPLIED BY OTHERS. (NOT GE)
 - △ - COMPUTER ANALOG INPUTS.
 - NPSH - NET POSITIVE SUCTION HEAD.
 - ◇ - START-UP CONTROL VALVE (THROTTLING TYPE VALVE).

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FIGURE 7.7-4
FEEDWATER CONTROL SYSTEM IED



*NOTE: HIGH AND LOW VALUE GATE TERMINOLOGY IS WITH REFERENCE TO THE MAGNITUDE OF THE INPUT SIGNAL TO THE GATE CIRCUIT UNDER NORMAL OPERATING CONDITIONS

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FIGURE 7.7-5
 SIMPLIFIED DIAGRAM OF TURBINE PRESSURE AND SPEED LOAD CONTROL REQUIREMENTS

7.8 EMERGENCY RESPONSE FACILITIES

7.8.1 Introduction

Edison has a technical support center (TSC), an operational support center (OSC), and an emergency operations facility (EOF) onsite; an alternative EOF is located offsite. The TSC is located in the office building annex. The OSC is located in the turbine building on the third floor close to the main control room. An alternative OSC is located in the machine shop. The EOF is located in the basement of the nuclear operations center approximately 6000 ft southwest of the reactor building outside the protected area. The alternative EOF is located at the Detroit Edison Western Wayne Center, 22 miles northwest of the Fermi 2 site. See Figure 7.8-1 for the location of the facilities within the owner controlled area.

7.8.2 Technical Support Center

7.8.2.1 General

The TSC has been established to provide the capability to display and transmit plant status information to individuals knowledgeable and responsible for engineering and management support of reactor operations in the event of an emergency condition. The TSC building is sited inside the protected area to lessen the time needed by personnel working in the plant to reach the building during an emergency condition. Other key factors considered in the selection of the TSC location included (1) the time needed by personnel working in the control room, in other plant areas, and at offsite locations to reach the TSC; (2) the availability of shielding to minimize exposure to direct radiation from the primary containment for personnel traveling to the TSC from the control room and other plant areas; and (3) the radiation protection (shielding) provided by existing plant structures to personnel arriving from offsite locations. The site chosen for the TSC also supports the efficient routine staffing of the building by plant operations and support groups who will provide added assurance that the systems and equipment necessary for TSC functioning will be maintained in a state of readiness.

The TSC is the emergency operations work area for designated Edison technical, engineering, and management personnel; other Edison personnel required to provide any needed technical support; and a small staff of NRC personnel. The TSC personnel will provide guidance and technical support for the Shift Manager in the control room. However, all control operations will be performed by licensed operators.

7.8.2.2 Design Basis

Information on plant status is provided to the TSC for use by technical and management personnel in support of command and control functions executed from the control room. The TSC does not affect the reliability or availability of the power plant and its safety systems.

The design bases for the TSC are as follows:

- a. Function. The onsite TSC will have the following functions:

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1. Provide plant management and technical support to plant operations personnel during emergency conditions
 2. Relieve the reactor operators of peripheral duties and communications not directly related to reactor system manipulations
 3. Prevent congestion in the control room
 4. Perform EOF functions until the EOF is activated.
- b. Activation time. The TSC is activated according to the RERP Plan and is made functional within prescribed times
 - c. Information availability. The TSC is equipped with Integrated Plant Computer System (IPCS) displays that provide information on plant status to support control room operations and emergency management. The IPCS also provides sufficient data for the assessment of offsite radiological and meteorological conditions. See Subsection 7.6.1.9.1.2.5.
 - d. Communications. The TSC is provided with the capability to communicate with all emergency facilities and locations to implement the Radiological Emergency Response Preparedness Plan
 - e. Habitability. The TSC is habitable during postulated radiological emergencies to the same degree as the main control room. Special shielding and heating, ventilation, and air conditioning (HVAC) systems are provided to minimize personal exposure and to ensure that NRC limits for whole-body exposure and airborne concentrations are satisfied
 - f. Size and layout. Adequate space is provided for proper functioning of the TSC emergency organization. Adequate space is also provided for equipment necessary for operation of the TSC. (See Figure 7.8-2)
 - g. Security. Normal plant security measures are maintained during the activation of the TSC
 - h. Access. The TSC is readily accessible to members of the TSC emergency organization arriving from both onsite and offsite locations. The exposure of personnel manning the TSC to potential direct radiation from the primary containment has been minimized by the selection of an appropriate TSC site and of appropriate access routes to the TSC
 - i. Fire protection. The TSC construction minimizes the use of combustible materials. Appropriate portable and permanent fire-extinguishing equipment is provided for the TSC and for the HVAC system. Fire-detection instrumentation is provided for automatic shutdown of the building's HVAC system
 - j. Record storage. Adequate space is provided within the TSC for permanent storage of records, diagrams, and design drawings that are considered necessary to support the functioning of the TSC during emergency conditions

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- k. Protection against natural phenomena. The TSC is sited and constructed to withstand the maximum postulated 100-year winds and 100-year floods.

7.8.2.3 Codes and Standards

The TSC building is designed and constructed according to the following codes and standards:

- a. ACI-318-77 - American Concrete Institute, Building Code Requirements for Reinforced Concrete
- b. AISC-1978 - American Institute of Steel Construction, Specification for the Design Fabrication and Erection of Structural Steel for Buildings
- c. ANSI A58.1-72 - American National Standards Institute, Building Code Requirements for Minimum Design Loads in Buildings and Other Structures, with
 1. Seismic Loadings conforming to Uniform Building Code (UBC) requirements for Zone 1
 2. Wind loads based on 100-year mean recurrence intervals for exposure type C.
- d. UBC-79 - Michigan Uniform Building Code, Seismic Zone 1
- e. ACI-531-79 - American Concrete Institute, Building Code Requirements for Concrete Masonry Structures.

The TSC is not classified as a nuclear safety-related facility. Its mechanical and electrical design bases are as follows:

- a. Mechanical systems: Quality Group D (includes mechanical system supports) design governed by the codes listed in Tables 7.8-1 and 7.8-2
- b. Electrical systems: Non-Class 1E.

7.8.2.4 Description

The TSC is located within the protected area of the Fermi 2 site, approximately 3-1/2 minutes walking time from the control room, as shown in Figure 7.8-3. It is located on the ground floor of the two-story office building annex, partially steel framed, with a 12-in.-thick reinforced-concrete ceiling slab on metal decking. The exterior walls, including labyrinths, are 12-in.-thick reinforced hollow-core concrete block filled with grout. The foundation incorporates spread footing under columns and strip footing under concrete block walls. There is a forced-air supply system, but no forced-air exhaust system; therefore, under normal operating conditions, the TSC will be under slight positive pressure with all entrance doors closed.

The site for the TSC was chosen to optimize the trade-off between travel time from the control room to the TSC and the radiation exposure of personnel enroute from onsite and offsite locations to the TSC. The Fermi 2 plant design was essentially completed at the time of TSC site selection, with the location of the primary containment and control room on the

northwest side of the plant, as shown in Figure 7.8-3. To satisfy the NRC guidelines for a 2-minute transit time from the control room to the TSC, it would have been necessary to locate the TSC to the north or west of the plant, which would increase the exposure of personnel traveling to the facility to radiation from the primary containment. Because the TSC could be activated at any time of day, the final site was selected so that members of the TSC emergency organization have access to the TSC from several locations, including the control room, offsite locations, and the plant supervisory offices located in the office building annex. The structures adjacent to the site offer the advantage of maximum shielding to the TSC and its access routes, thus providing acceptable, safe travel time from the control room, should this route have to be traveled.

The TSC, as shown in Figure 7.8-2, covers about 5000 ft². Approximately 2075 ft² of this area is devoted to occupancy by TSC personnel. The remaining space consists of rooms for records storage, toilets, HVAC equipment, telephone and communications equipment, and electrical equipment. Status boards and marking boards are conveniently located within the monitor room. Telephone jacks and electrical outlets are in the computer floor.

The NRC has defined four emergency action levels (see Subsection 7.8.2.13) to categorize the severity of various operational emergencies. Additional guidance is published in NUREG-0654, Appendix 1 (Reference 1). The TSC will be activated for events at or beyond the "alert" level. Upon activation, the TSC will be placed in operation after occupancy by a specified number and type of personnel (staffing of the TSC is described in the Fermi 2 Radiological Emergency Response Preparedness Plan) and after TSC communication, monitoring, and occupancy support systems have been energized.

7.8.2.5 Habitability

The TSC occupants are protected from radiological hazards, including exposure to direct radiation and airborne contaminants. Specific design features and administrative procedures ensure that the radiation dose received by TSC personnel does not exceed the limits and guidelines of General Design Criterion (GDC) 19 of 10 CFR 50, and NRC Standard Review Plan, Section 6.4 (Reference 2).

The contributions of several radiation sources are considered in calculating the dose equivalent to TSC personnel. Radiation exposure may derive from immersion in or inhalation of radio-activity in the TSC atmosphere as well as direct shine from sources outside the TSC shield envelope (e.g., reactor building, standby gas treatment system [SGTS] exhaust plume, TSC makeup filters). Shielding is used to reduce the dose equivalent from any single external source to a negligible level; that is, less than one-tenth of the allowable dose equivalent. The radiation shield design takes into account all shield penetrations, as well as potential radiation sources within the habitable area.

The HVAC system has been designed to facilitate the occupation of all necessary personnel for winter and summer environmental and radiological accident conditions. It is designed to maintain a habitable environment of the same quality as the control room, even though it is not rated safety related, seismic, or redundant. The HVAC system is capable of the following:

- a. Maintaining room temperature by removing all heat released by equipment, lights, occupants, and thermal transmission

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- b. Maintaining room temperature by replacing all heat loss due to thermal transmission, with no credit for lighting, occupants, or equipment
- c. Limiting the thyroid radioiodine dose received by personnel inside the TSC
- d. Introducing outside air into the TSC envelope will result in a slight positive pressure with respect to the outdoors.

Simplified process flow diagrams for the TSC HVAC system are provided as Figures 7.8-5, 7.8-6, and 7.8-7. The HVAC equipment consists of the following:

- a. Air-handling unit: a three-zone, multizone air-handling unit with an air delivery of 5500 cfm. The unit is equipped with a direct-expansion cooling coil with a cooling capacity of 15 tons. Pressure-equalizing baffles are provided in the hot deck, as each zone shall have its own electric heating coil
- b. Supply air filter: bag filters that are 80 to 85 percent efficient. The filter-element efficiency is based on the National Bureau of Standards' dust spot test
- c. Purge fan: a ceiling-mounted, vane-axial, 4900-cfm purge fan
- d. Steam generator: a self-contained, all-electric steam generator. The unit is capable of generating 20 lb/hr of steam for humidification
- e. Steam humidifiers: Each zone is provided with a steam humidifier. The steam for humidification is provided from the steam generator
- f. Air-cooled condensing unit: a floor-mounted, air-cooled condensing unit with a cooling capacity of 15 tons
- g. Electric heating coils: Duct-mounted heating coils for each zone are provided and are rated for their respective zones
- h. Toilet room exhaust fan: an air-line, duct-mounted exhaust fan with a 400-cfm capacity
- i. HVAC and electric equipment room air-handling unit: This unit has a capacity of 3600 cfm
- j. Duct-mounted electric heating coils and unit heaters are also provided for HVAC and electric equipment rooms.

The HVAC system for the TSC has an emergency makeup air system to filter a combination of outside makeup air and recirculation air for pressurization and to maintain the TSC dose within allowable limits. The emergency makeup air system consists of the following components:

- a. Prefilters capable of no less than 85 percent filtration efficiency based on the ASHRAE dust spot test
- b. A single-stage electric heating coil, capable of raising the air temperature and reducing the relative humidity of the airstream to 70 percent, or less, for the worst inlet condition
- c. High-efficiency particulate air (HEPA) filters capable of removing 99.97 percent of particulate matter 0.3 μm and larger in size based on a hot dioctyl

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phthalate (DOP) test. The HEPA filters are provided upstream and downstream of the charcoal adsorber. The HEPA filters need not be tested as specified in Regulatory Guide 1.52 and need not meet the quality assurance requirements of 10 CFR 50, Appendix B

- d. Two charcoal adsorbers (total charcoal thickness of 4 in.) that are capable of removing radioactive and nonradioactive forms of iodine are provided. The charcoal adsorbers are of the drawer type, filled with impregnated coconut shell where the depth of charcoal is 2 in. These adsorbers together have the iodine removal efficiency of not less than 99 percent. The charcoal adsorbers meet the requirements of Regulatory Guide 1.52, and of Table 7.8-3
- e. A belt-driven centrifugal fan located upstream of the filter unit is provided to maintain the filter unit at a positive pressure
- f. Each charcoal adsorber bank is provided with a two-stage continuous thermistor located across the discharge air path from each adsorber
- g. Each charcoal adsorber bank includes a fire protection system for extinguishing a charcoal fire
- h. The makeup air unit is provided with instrumentation as required in Table 4.2 of ANSI N509
- i. The makeup air unit is provided with drain connections for each compartment of the housing, which are piped to the side of the unit, valved, and drain to the sanitary sewer.

The filter train for the emergency makeup air system is designed to remove radioactive particulates and absorb radioactive iodine. Circulated air consists of a mixture of recirculated air and sufficient outside air to maintain the TSC at a positive pressure of $\geq 1/8$ -in. water gage. TSC doses will be maintained within allowable limits provided the introduction of outside air does not exceed 1000 cfm.

An area radiation monitor is provided to continuously measure and indicate the general area radiation levels in the TSC. Friskers are available at the entrance to the TSC to provide radiological access control of persons entering the TSC. The TSC has provisions for monitoring iodine by using specific cartridges that can detect iodine levels as low as 1×10^{-7} $\mu\text{Ci}/\text{cm}^3$. The TSC radiation monitoring equipment is calibrated according to Health Physics procedures and to manufacturers' recommendations.

Sufficient protective clothing is stored in the TSC for personnel. If additional clothing is required, it is available from various designated locations in the plant.

Stationery supplies and duplicating equipment are also available in the TSC.

7.8.2.6 Staffing

Staffing of the TSC is described in the Fermi 2 Radiological Emergency Response Preparedness Plan. The TSC emergency organization is activated for conditions involving an alert, site area emergency, and general emergency.

7.8.2.7 Communications

The communications system is discussed in detail in the Radiological Emergency Response Preparedness Plan.

The plant intercom (HiCom) system used for general plant operations also has extensions in the TSC and other site emergency facilities. There is a public address system within the TSC for general announcements to all TSC personnel.

The communications systems have been tested and their performance evaluated during practice drills and a full-scale exercise. The systems have been found satisfactory for implementing the emergency plan.

7.8.2.8 Instrumentation and Power Supplies

Electric power is furnished to the TSC via two independent non-Class 1E 480-V ac feeders derived from separate offsite sources, each sized to carry the entire TSC load. The feeders are connected to the TSC power distribution system through an automatic transfer switch. On complete loss of offsite power, combustion turbine generators (not associated with the emergency diesel generators) located at the site are capable of providing power. Since the TSC is powered from non-Class 1E sources, TSC loads or faults in the TSC power distribution system will not affect the plant's safety-related power distribution system.

The TSC power distribution system consists of a single motor control center that provides power to the HVAC equipment, lighting and instrumentation power supply transformers, and other TSC auxiliary loads, such as copying machines and microfilm viewers. A separate instrumentation power supply transformer protects the solid-state TSC IPCS data display equipment from power-line disturbances.

Lighting for the TSC consists of recessed fluorescent ceiling fixtures. Emergency battery-pack lighting is also furnished.

7.8.2.9 Information Systems

The TSC data display system, documentation, plant drawings, control room records, plant chemistry data, plant historical data, analytical data, verbal and recorded information provided by plant operations personnel, radiological assessment and other analyses available from offsite sources, and data provided by the radiological monitoring teams enable the TSC staff to determine the following:

- a. The plant status and dynamics before and during the accident
- b. The performance of accident mitigation functions
- c. The nature and trend of the accident
- d. The damage to the plant and equipment
- e. The status of the operation (including personnel activity in the plant)
- f. The amount of radioactive release to the environment
- g. The prevailing meteorological conditions

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- h. The radiation and radioactivity level of the environs
- i. The offsite dose assessment.

A computer-based data handling system, IPCS (see Subsection 7.6.1.9.1), is provided to supply emergency response information for display in the TSC. The IPCS is of high quality and reliability, and is non-Class 1E and nonseismic.

The IPCS displays emergency response information in the control room, TSC, and EOF. Recording, trending, and time-history plotting capabilities are provided within the system. In the event of a complete loss of offsite power, data will be retained by the IPCS during the outage for display once TSC power is restored.

Six workstations for data display are located in the TSC.

7.8.2.10 Records Storage

A file of copies of the following documents is maintained for use in the TSC:

- a. General arrangements
- b. Process and instrumentation diagrams
- c. Piping drawings
- d. Logic diagrams
- e. Electrical schematics
- f. Operating procedures
- g. Emergency procedures
- h. Technical Specifications
- i. Master instrument lists (retrieval by computer)
- j. Updated Final Safety Analysis Report
- k. Plant operating records
- l. Radiological Emergency Response Preparedness Plan and its implementing procedures
- m. Radiation exposure histories (retrieval by computer)
- n. Other documents sufficient to diagnose potential plant operating problems at the system level.

Other plant documents, or copies, are available for use by the TSC as needed during the course of an emergency. Such documents include those normally stored in other locations at the plant site, such as in the technical staff offices or in the records storage center next to the TSC. A conventional office copy machine and microfilm viewing and copying devices are located in the TSC for the purpose of copying documents likely to be used during an emergency.

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7.8.2.11 Fire Protection

The TSC is constructed using the following noncombustible materials:

- a. Exterior: reinforced concrete block walls, reinforced concrete roof, and hollow metal doors. Exposed structural steel has been fireproofed
- b. Interior: stud drywall-type partitions and suspended acoustical ceiling. All floor areas except the electrical equipment room, the HVAC room, and the monitor and rest rooms are carpeted. The monitor area has a computer floor with vinyl asbestos tile.

The following fire-protection equipment is provided:

- a. Portable carbon dioxide units
- b. A water deluge system for the charcoal filters in the emergency makeup filter unit
- c. Smoke detectors in the fresh air intake and in the return-air duct of the TSC HVAC system.

7.8.2.12 Evaluation

Edison has provided for a TSC separate from but near the control room. The location in the office building annex ensures operating personnel familiarity with the facility. The TSC has the capability to display and transmit plant status to individuals knowledgeable of and responsible for engineering and management support of plant operations in the event of an accident.

The overall data handling system is the IPCS. The emergency response capability of the IPCS includes measurements that permit assessment of reactivity control, reactor core cooling, reactor coolant system integrity, containment integrity, meteorology, and dose assessment. The IPCS incorporates features and recommendations from NUREG-0696 (Reference 3).

Upon activation, the TSC will provide the initial main communications link between the plant, the OSC, the NRC, and the offsite emergency response organizations until the EOF is available. The TSC will be habitable to the same degree as the control room during postulated accident conditions in accordance with the requirements of NUREG-0578 (Reference 4).

Records pertaining to as-built conditions and layout of structures, systems, and components are available to personnel in the TSC.

In summary, the TSC provides the integrated emergency response capability required by the NRC.

7.8.2.13 Emergency Action Levels

- a. Notification of unusual event: Events are in process or have occurred that indicate a potential degradation of the level of the safety of the plant

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- b. Alert: Events are in process or have occurred that involve an actual or potential substantial degradation of the level of the safety of the plant
- c. Site area emergency: Events are in process or have occurred that involve actual or likely major failures of plant functions needed for the protection of the public
- d. General emergency: Events are in process or have occurred that involve actual or imminent substantial core degradation or melting with imminent potential for loss of containment integrity.

7.8.3 Operational Support Center

The function of the OSC is to act as an onsite area, separate from the control room and the TSC, where licensee operations support personnel will assemble in an emergency. The OSC will:

- a. Provide a location where plant logistic support can be coordinated during an emergency
- b. Restrict control room access to those support personnel specifically requested by the Shift Manager.

The OSC is a designated area at the north end of the third floor (Elevation 643 ft) of the turbine building. The OSC is located on the same floor and in close proximity to the main control room. The area is served by normal building HVAC systems. Figures 1.2-13 and 1.2-14 show the location of the OSC relative to the main control room.

The OSC provides a designated assembly point for shift support personnel for assignment of duties to support emergency operations. Personnel such as instrument technicians and operators will be dispatched from this area. The OSC is activated for an alert, site area emergency, or general emergency condition. The emergency organization is described in the Radiological Emergency Response Preparedness Plan. An alternate OSC has been designated in the machine shop and is equipped as necessary with supplies and equipment to ensure continued support of the OSC emergency organization. The machine shop is located such that it would be highly improbable that both the primary and the alternate locations would not be habitable at the same time. The two OSC locations are served by separate HVAC systems.

The OSC and alternate OSC are equipped with communications systems and supplies, including protective clothing and equipment. Portable Health Physics equipment is provided to monitor radiological conditions in the OSC. Procedures have been established for control and for periodic inventories, recalibrations, and replenishments of perishable items. Communication to the OSC is via the emergency telephone communications network, radio communications using hand-held VHF transmitters, and the plant intercom system.

The OSC also has radiation monitoring capabilities, though its habitability requirements are not the same as those of the control room. The monitoring equipment consists of friskers, dose rate meters, and high range dosimeters. The OSC personnel also have available the use of self-contained breathing apparatus and partial sets of protective clothing. At the direction

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of the Emergency Director, the OSC supervisors shall relocate the OSC personnel to the alternate OSC and establish this as the staging area.

The OSC emergency organization, equipment, and communications systems have been evaluated during a full-scale exercise and have been found acceptable.

7.8.4 Emergency Operations Facility

7.8.4.1 General

The EOF functions as an operational support center with capabilities for the following:

- a. Management of overall licensee emergency response
- b. Coordination of radiological and environmental assessment
- c. Determination of recommended public protective actions
- d. Coordination of emergency response activities with federal, state, and local agencies

Facilities are provided in the EOF for the acquisition, display, and evaluation of radiological, meteorological, and plant system data pertinent to determining offsite protective measures. These facilities are used to evaluate the magnitude and effects of actual or potential radioactive releases from the plant and to determine offsite dose projections. The EOF is used to coordinate emergency response activities with those of local, state, and federal agencies, including the NRC. Edison personnel in the EOF will make protective action recommendations for the public to the state emergency response organization.

The EOF is located in the basement of the nuclear operations center (NOC), approximately 6000 ft southwest of the Fermi 2 reactor building, and has been designed for habitability in the event of a postulated accidental radioactive release from Fermi 2. Shielding and HVAC system design ensure that NRC regulations for personnel exposure are satisfied.

The EOF is activated for conditions involving an Alert or higher emergency classification. Emergency plan implementing procedures define the transition of responsibility from the control room to the TSC and the EOF until the latter facilities become functional. The NOC also provides space for managing recovery operations and media briefings.

An alternative EOF is located at the Western Wayne Center, 22 miles northwest of the Fermi 2 site. The facility has adequate communications equipment and sufficient space to accommodate the additional personnel required for continuity of dose projection and decisionmaking capability, including coordination of the offsite teams. Portable equipment is provided for the personnel to perform their assigned functions.

7.8.4.2 Description

The primary EOF is located in the NOC building. Besides housing the EOF, the NOC provides room for supporting personnel required for assistance to Fermi 2 operations. This includes licensing, data control, administrative support, and training personnel. Also contained in the NOC will be the recovery center, a media briefing area, and a food processing and service area. Over 60,000 ft² of space will be provided in the NOC for these

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support personnel. Detailed information about staffing and the emergency organization is contained in the Fermi 2 Radiological Emergency Response Preparedness Plan.

The EOF is located about 6000 ft southwest of the power plant, just west of Quarry Lakes, within the Edison-controlled property boundary (see Figure 7.8-1). The facility can be reached from two directions via roads under the control of Edison. Electrical power is available from either one of two major power substations. An emergency generator is also available to automatically restore power in the improbable event of the loss of both power supplies.

The NOC building has been designed for the following:

- a. Roof snow load: 40 lb/ft² minimum, plus provisions for drifted snow
- b. Floors: 150 lb/ft² for entire second floor and for first floor at EOF
- c. Stairs: 100 lb/ft²
- d. Wind load: conforms to ANSI A58.1-72, based on 100-year mean recurrence interval for exposure type C
- e. Seismic: conforms to UBC for Zone 1.

The construction is standard except for special concrete shield walls surrounding the EOF that provide a protection factor of approximately 20. The internal layout of the EOF is shown in Figure 7.8-10 and consists of space allocated for records, counting facilities, offices, NRC office space, communications equipment, and emergency power (batteries) for communications equipment.

Habitability of the EOF is provided by an HVAC system, which includes a HEPA filter.

Radiation detection alarms are set at approximately three times the background levels to provide an early visual and audible warning to the EOF occupants. Air sampling capability is also provided in the EOF, with the capability to detect iodine concentrations as low as $1 \times 10^{-7} \mu\text{Ci}/\text{cm}^3$.

The EOF also has available friskers, dose rate meters, dosimeters of legal record (DLRs), iodine air sampler/detectors, and dosimeters to monitor radiation levels. The quality and quantity of this instrumentation has been determined by surveys, data research, and professional experience. The radiation monitoring equipment used in the EOF is calibrated according to Health Physics procedures and to manufacturers' instructions.

The EOF backup laboratory is equipped with a high-resolution gamma spectroscopy system and other equipment required to perform chemistry and radiochemistry.

Emergency supplies and equipment are stored and maintained in the EOF in accordance with the implementing procedures for the Radiological Emergency Response Preparedness Plan. These are periodically inventoried and calibrated to ensure their availability if needed. These supplies include protective clothing, stationery supplies, and duplicating equipment.

The EOF communication network is described in detail in the Radiological Emergency Response Preparedness Plan.

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7.8.4.3 Information Systems

The EOF information systems are commensurate with the EOF functions of:

- a. Coordination of offsite response
- b. Coordination of radiological, meteorological, and environmental assessment
- c. Recommendations for protective actions.

The information required to perform the above functions includes:

- a. Assessment of plant status
- b. Radiation releases
- c. Meteorological data
- d. Atmospheric dispersion models
- e. Field monitoring for offsite radioactivity.

In addition to the information included in Subsections 7.8.2.9 and 7.8.2.10 for the TSC, the following are available for use in the EOF:

- a. Offsite population data
- b. Environmental radiological monitoring records
- c. State and local emergency response plans
- d. Evacuation plans.

7.8.4.4 Staffing

Staffing of the EOF is described in the Radiological Emergency Response Preparedness Plan.

7.8.4.5 Emergency Response Facilities Integration

During emergency conditions, it is essential that there be a continuous high level of interaction and communication among key personnel in the control room, emergency response facilities, and the NRC to ensure that all emergency actions are fully understood and coordinated.

The emergency response facilities are developed to function as an integrated system. Edison's emergency response facilities are designed to provide coordinated support to the control room during emergency operating conditions. These facilities are integrated into the Radiological Emergency Response Preparedness Plan to facilitate coordination with state and local emergency response facilities.

The system design of the emergency response facilities has the following functional criteria:

- a. The operation of any system or subsystem within the emergency response facilities does not degrade the performance or reliability of any reactor safety or control system or of any safety-related displays in the control room

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- b. The operation of any system or subsystem in the emergency response facilities does not degrade or interfere with the functional operation of other systems in those facilities
- c. The data acquisition hardware and software are protected against unauthorized manipulation or interference with input signals, data processing, data storage, and data output.

The emergency response function of the IPCS provides a fully integrated data processing system serving all emergency response facilities and systems.

The equipment to be used in the control room during an emergency is identified in Subsection 7.6.1.9. Subsection 7.6.1.9 also addresses the primary variables to be displayed.

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7.8 EMERGENCY RESPONSE FACILITIES

REFERENCES

1. U.S. Nuclear Regulatory Commission, Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants, NUREG-0654, Revision 1, November 1980.
2. U.S. Nuclear Regulatory Commission, Standard Review Plan, Section 6.4, "Habitability Systems," NUREG-75/087, Revision 1.
3. U.S. Nuclear Regulatory Commission, Functional Criteria for Emergency Response Facilities, for Interim Use and Comment, Draft NUREG-0696, July 1980.
4. U.S. Nuclear Regulatory Commission, TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations, NUREG-0578, July 1979; Revised August 1979 by NRC letter, enclosure 6.

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TABLE 7.8-1 GOVERNING CODES AND STANDARDS FOR HVAC SYSTEM COMPONENTS

Components	Codes and Standards
Fans	AMCA 210-74: Laboratory Methods of Testing Fans for Rating AMCA 211A-74: Certified Rating Program Air Performance AMCA 300-67: Test Code for Sound Rating
Motors	NEMA MG 1-74: Motors and Generators
Cooling coils	ARI 410-72: Standard for Forced Circulation Air-Cooling and Air-Heating Coils (nuclear safety related and nonnuclear safety related)
Isolation, modulation dampers, and damper operators	AMCA 500: Test Method for Louvers, Dampers and Shutters
Supply filter units	Applicable portions of ANSI N509-76: Nuclear Power Plant Air Cleaning Units and Components ASHRAE 52-68: Air Cleaning Devices Used in General Ventilation for Removing Particulate Matter, Method of Testing Regulatory Guide 1.52 (Revision 2, March 1978): Design Testing and Maintenance Criteria for Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Absorption Units of Light-Water-Cooled Nuclear Power Plants, including S&L Standard Position and excepting testing and quality assurance requirements.
Energy loads	ASHRAE Handbook and Product Directory, Fundamentals

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TABLE 7.8-2 GOVERNING CODES AND STANDARDS FOR MECHANICAL SYSTEM COMPONENTS

Components	Codes and Standards
Pressure vessel	ASME Boiler and Pressure Vessel (B&PV) Code, Section VIII, Division I
Heat exchangers	TEMA C and ASME B&PV Code, Section VIII, Division I
Piping	ANSI B31.1.0
Valves	ANSI B31.1.0
Pumps	Manufacturer's standards
Atmospheric storage tank	API-650, AWWA-D100, or ANSI B96.1
Storage tanks, 0 to 15 psig	API-620
Filter package	ANSI N509 and N510

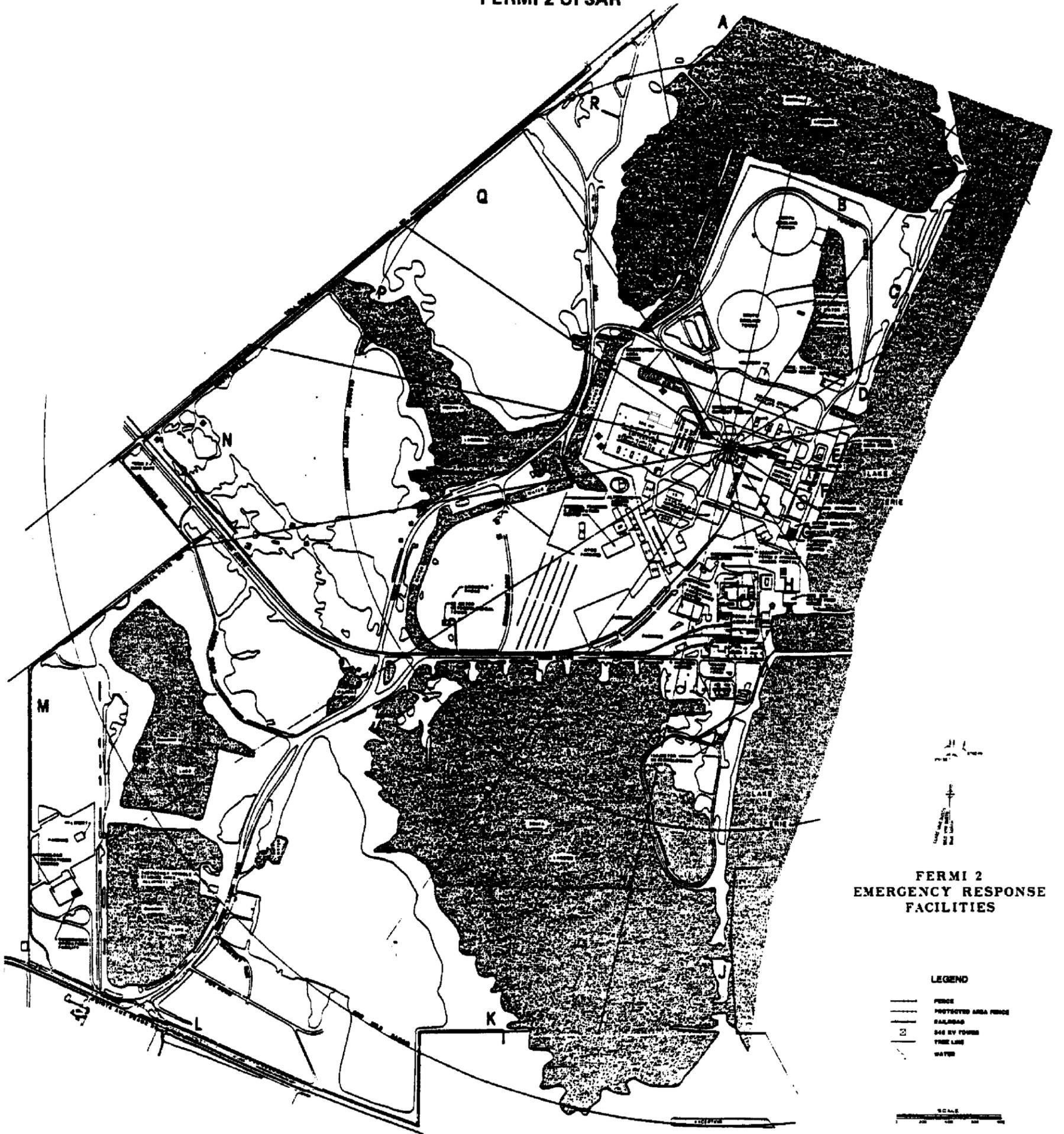
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TABLE 7.8-3 PERFORMANCE REQUIREMENT AND PHYSICAL PROPERTIES OF (UNUSED) ACTIVATED CARBON

Test	Method	Acceptance Value
<u>Performance Requirements</u>		
Molecular iodine, 30 °C, 95 percent RH ^a	ASTM D3803	0.1 percent penetration, maximum
Molecular iodine, 180 °C		99.5 percent retentivity, minimum
Methyl iodine, 30 °C, 95 percent RH		3 percent penetration, maximum
Methyl iodine, 80 °C, 95 percent RH ^a		1 percent penetration, maximum
<u>Physical Properties</u>		
Particle-size distribution	ASTM D2862	8 x 16 U.S. mesh Retained on No. 6 sieve: 0.1 percent maximum Retained on No. 8 sieve: 5.0 percent maximum Through No. 8, on No. 12 sieve: 60 percent maximum Through No. 12 on No. 16 sieve: 40 percent maximum Through No. 16 sieve: 5.0 percent maximum Through No. 18 sieve: 1.0 percent maximum
Ball pan hardness	ASTM D3802	92 minimum
C Cl ₄ activity (onbase)	ASTM D3467	60 minimum
Apparent density	ASTM D2854	0.38 g/cm ³ minimum
Ash content (onbase)	ASTM D2866	State value
Ignition temperature	ASTM D3466	330 °C minimum
Moisture content	ASTM D2867	State value
pH of water extract	Appendix D	State value

^a Tests shall be performed for qualification purposes only.

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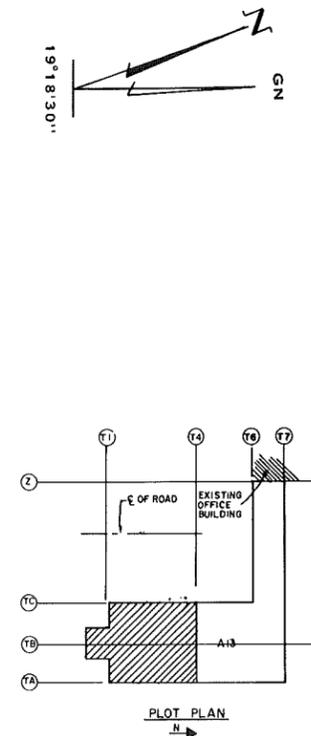
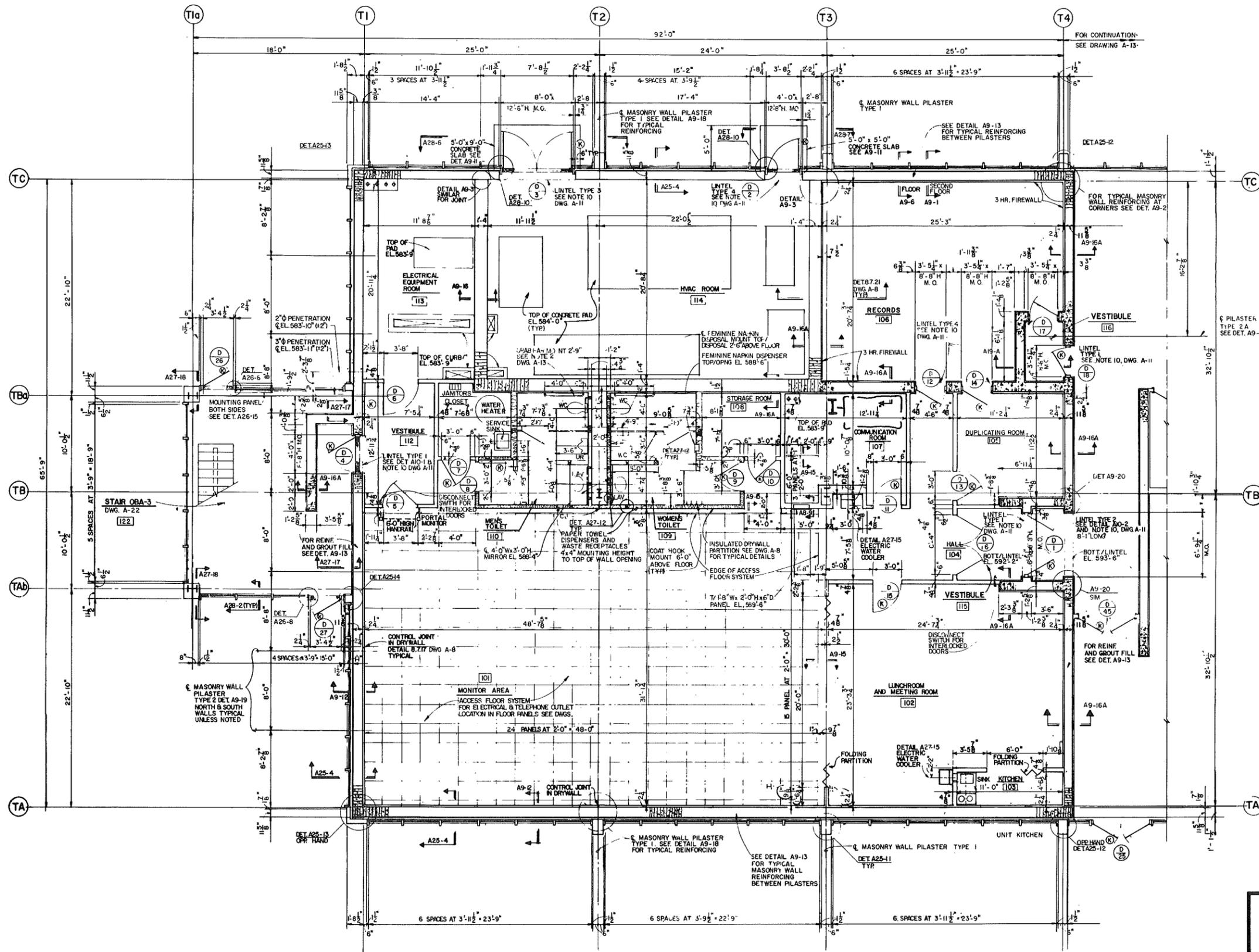


Fermi 2

UPDATED FINAL SAFETY ANALYSIS REPORT

FIGURE 7.8-1

EMERGENCY RESPONSE FACILITIES



NOTES	
1.	FOR ARCHITECTURAL GENERAL NOTES SEE DRAWING A-11
2.	FOR EQUIPMENT PANS SEE DRAWINGS S-1 AND S-3
3.	FOR ADDITIONAL JAMB LINING FOR OPENINGS IN ALL MASONRY WALLS SEE DET. A9-10
4.	FOR LINTEL "TYP" DETAILS SEE DWG A-10
5.	FOR PLUMBING FLOOR DRAIN SYSTEM SEE DWG A-29

REFERENCE DRAWINGS	
A-2	ELEVATIONS AND ROOM FINISH SCHEDULE.
A-5	SUSPENDED CEILING PLAN AND DETAILS.
A-6	DOOR AND HARDWARE SCHEDULE AND DETAILS.
A-8	DRYWALL AND LOUVER DETAILS.
A-9	SECTIONS AND DETAILS - SHEET 1.
A-10	SECTIONS AND DETAILS - SHEET 2.
S-1	FOUNDATION PLAN
S-3	CONCRETE SECTIONS AND DETAILS
E-1	CONCEALED CONDUITS & RACEWAYS
A-25	OFFICE R.D.S. ANNEK WALL SECTIONS & DETAILS
M-15	T.S.C. HVAC SYSTEM GRADE FLOOR PLAN
A-31	O&A B.T.S.C V&B DIAG EL.598'-6" & 613'-4"

GRADE FLOOR PLAN-EL.583'-6"
MONOLITHIC CONCRETE FLOOR

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FIGURE 7.8-2

TECHNICAL SUPPORT CENTER FLOOR PLAN

Security-Related Information - Withheld Under 10 CFR 2.390

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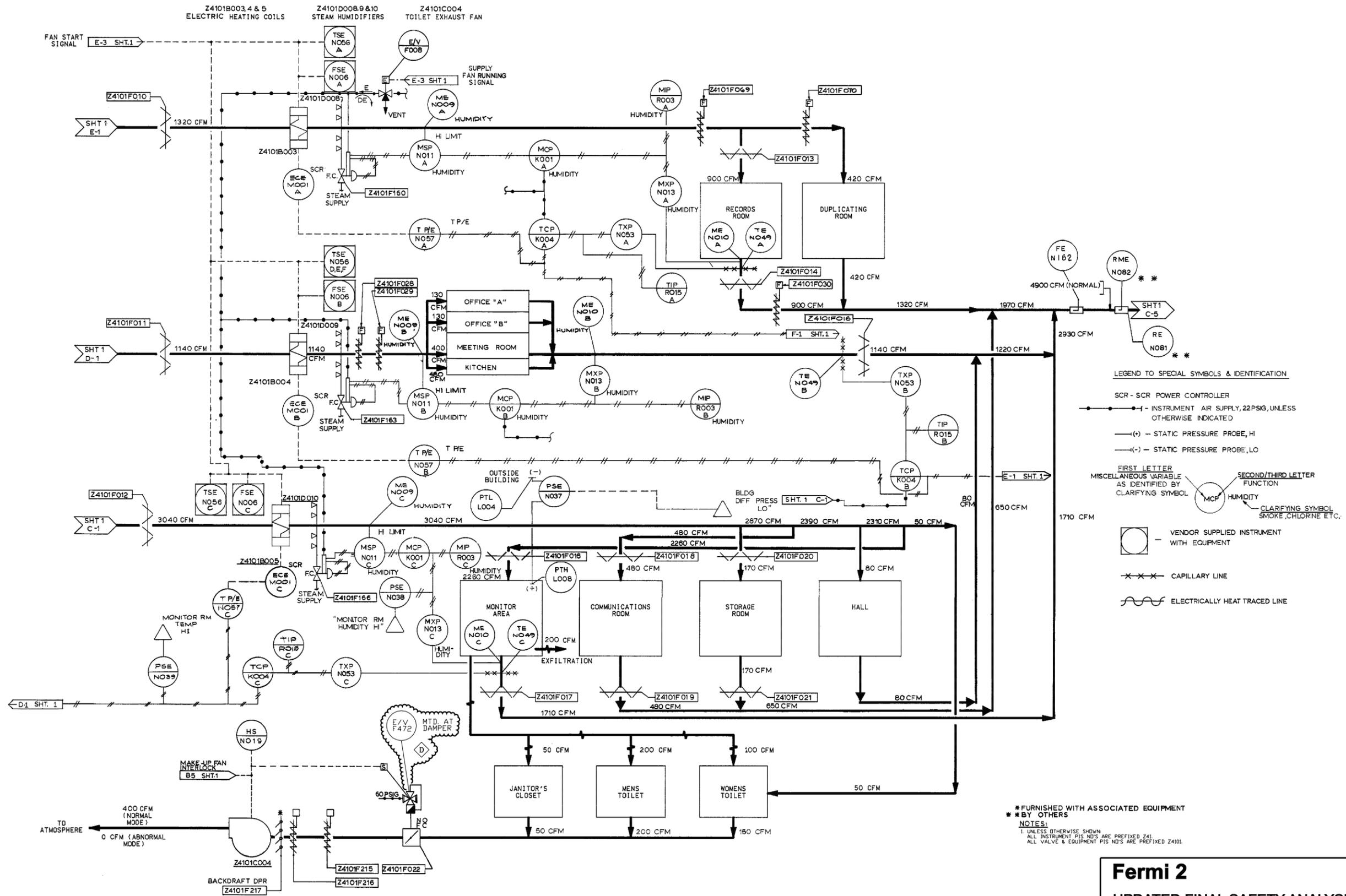
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FIGURE 7.8-3

PROPERTY PLAN – TECHNICAL SUPPORT CENTER

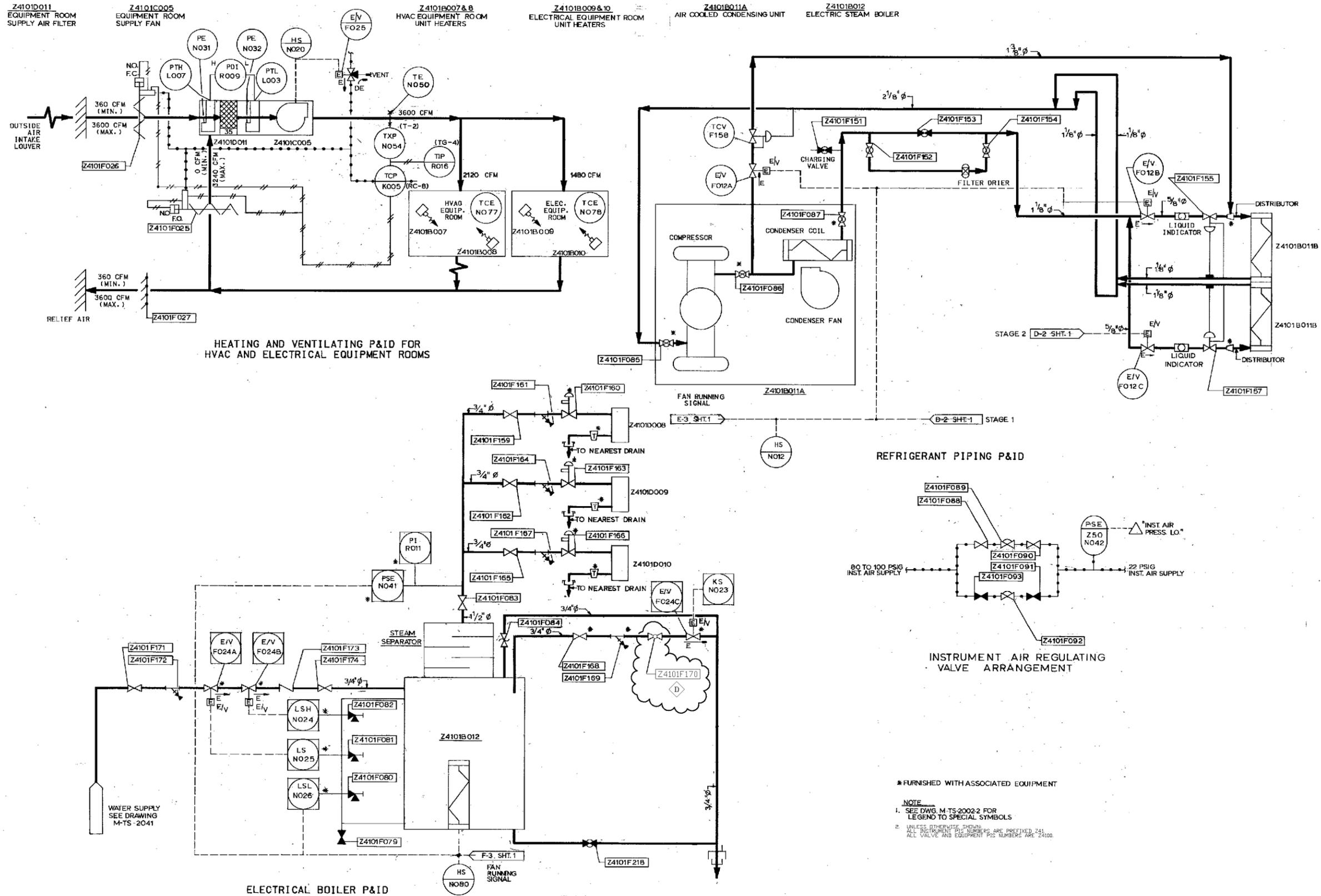
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FIGURE 7.8-4 HAS BEEN DELETED
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FIGURE 7.8-6
TECHNICAL SUPPORT CENTER HVAC SYSTEM
2VV02S



* FURNISHED WITH ASSOCIATED EQUIPMENT

NOTE
 1. SEE DWG. M-TS-2002-2 FOR LEGEND TO SPECIAL SYMBOLS
 2. UNLESS OTHERWISE SHOWN, ALL INSTRUMENT P&I NUMBERS ARE PREFIXED Z41. ALL VALVE AND EQUIPMENT P&I NUMBERS ARE Z4100.

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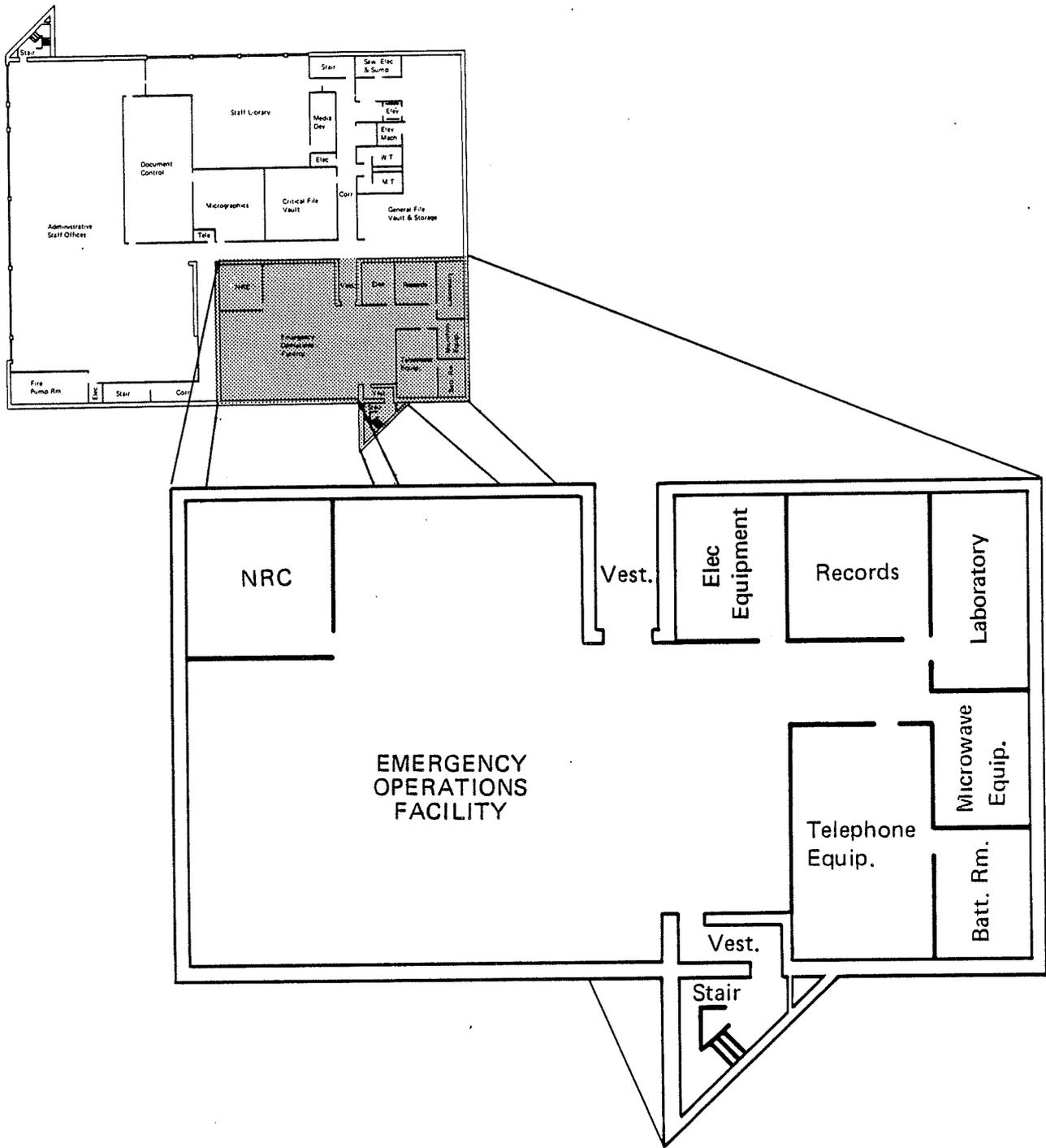
7.8-7
 TECHNICAL SUPPORT CENTER HVAC SYSTEM
 2VV03S AND 2VV04S

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FIGURE 7.8-10

FLOOR PLAN – EMERGENCY OPERATIONS FACILITY