Millstone Power Station Unit 2
Safety Analysis Report

Chapter 7

CHAPTER 7–INSTRUMENTATION AND CONTROL

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CHAPTER 7 - INSTRUMENTATION AND CONTROL

7.1 INTRODUCTION

The plant systems are instrumented to provide information on plant conditions at selected locations, to protect equipment and personnel from undesirable conditions and to control the plant during startup, operation, and shutdown. The principal control station for the plant is in the control room located in the reactor auxiliary building.

The plant is started up and shut down under remote manual control. Annunciators, indicators and recording devices will alert the operator and provide data on plant conditions.

Instrumentation and controls essential to plant safety are located in the control room. The instrumentation is arranged in groups on the control boards so that when corrective action is required, all pertinent indicators, recorders and controllers are within easy reach and view of the operator. The control board is a duplex benchboard. Visible and audible alarms located in the top section of the main control board annunciate and identify abnormal operating conditions. Telephone systems provide both in-plant and external communication. The control room and computer room are kept at a controlled temperature which is well within the design temperature requirements of the instruments.

To ensure reliability, components of established quality are selected and used in the instrumentation and control equipment. The protection instrumentation consists of four independent channels to permit system testing without reducing the degree of protection provided. Reliable sources of electrical power are provided to ensure safe and reliable plant operation (see Chapter 8).

The operation of the reactor within established limits is achieved by its inherent characteristics, instrumentation and control systems, by operational procedures and administrative controls. Departures from these limits are audibly and visibly annunciated in the control room. A reactor protective system is designed to initiate reactor trips to protect the core and the reactor coolant system pressure boundary.

The engineered safety features actuation system instrumentation provides the equipment necessary to initiate the required safety features functions. This system also monitors the power sources acting to assure the availability of emergency power for operation of at least the minimum engineered safety features (see Chapters 6 and 8). This system is provided with the necessary redundant circuitry and physical isolation so that a single failure within the system would not prevent the proper system action when required. This system is provided with test facilities and alarms to alert the operator when certain components trip or malfunction, or are inoperable. The controls are designed to automatically provide the sequence of operations required to initiate engineered safety features system operation with or without off-site power available.

Those instruments which have been classified Seismic Class I have been type tested or validated by computation to ensure operability.

The instrumentation, protection and control systems designed and built by C. E. for the Millstone Unit Number 2 are functionally identical to corresponding C. E. systems provided for Calvert Cliffs (NRC Docket Numbers 50-317 and 50-318 through Amendment 19) except for the modifications mentioned below.

A reactor coolant pump underspeed trip has been added (Section 7.2.3.3.1). This trip is not credited to provide protection during any accident analysis event.

Automatic axial flux tilt protection has been added through the addition of a Local Power Density Trip (Section 7.2.3.3.10) and modification of the Thermal Margin/Low Pressure Trip (Section 7.2.3.3.7).

The High Power Trip has been modified to use a variable setpoint (Section 7.2.3.3.2).

The trip logic matrix relay power supplies have been split so that each matrix power supply supplies only two relays, thus preventing a single power supply failure from causing a spurious reactor trip (Figure 7.2–2).

The range of the T_{cold} channel inputs to the RPS has been changed in support of the modification to the Thermal Margin/Low Pressure Trip (Section 7.2.3.3.7) from 515°F to 615°F to 465°F to 615°F. This was to assure the existence of a valid ΔT power signal during transients involving significant decreases in core inlet temperature.

The function of the Zero Power Mode Bypass (Section 7.2.3.3.12) has been expanded to include removing the ΔT power component of the power signal Q in order to allow for special physics startups when system temperatures are below the operation range. In addition, the Zero Power Mode deletes the RCP underspeed trip.

The Low Reactor Coolant Flow Trip (Section 7.2.3.3.3) acts on a ΔP signal.

The Power Trip Test Interlock (Section 7.2.4) has been expanded to provide protection against the Nuclear Instrumentation Test Select switch being left in an off-normal position after testing.

The Control Element Drive System (Section 7.4.2) has been modified to include a CEA Motion Inhibit.

The Reactor Regulating System (Section 7.4.1) has been modified to delete the automatic mode of CEA movement.

7.1.1 IDENTIFICATION OF SAFETY RELATED EQUIPMENT

7.1.1.1 Protective Systems

Protective Systems encompass electrical and mechanical devices and circuitry from sensors through actuation devices.

Reactor Protective System (Section 7.2)

Engineered Safety Features Actuation System (Section 7.3)

Auxiliary Feedwater Automatic Initiation System (Section 7.3)

Protective System Power Supply (Section 8)

7.1.1.2 Safe Shutdown Systems

The Reactor Protective System will safely bring the Reactor to a hot standby condition if any of the input parameters deviates from its preselected operating range or upon manual initiation by the operator. The equipment normally used to maintain the plant in hot shutdown is listed in Table 7.6-1. Additional equipment normally used to bring the plant to cold shutdown conditions is listed in Table 7.6-2.

7.1.1.3 Safety-Related Display Instrumentation

Reactor Protective System input parameter indication. (Section 7.2, 7.5)

Engineered Safety Features Actuation System input parameter indication. (Section 7.3, 7.5.1)

Engineered Safety Features monitoring instrumentation. (Section 6)

7.1.1.4 Other Safety-Related Systems

Shutdown Cooling Interlocks (Section 9.3)

Refueling Interlocks (Section 9.8)

Auxiliary Steam Detection and Isolation System (Sections 7.10 and 9.13)

7.1.1.5 Control Systems

The systems listed below are not required for safety. Protective system action ensures that design limits are not exceeded.

Reactor Coolant Pressure Regulating Systems (Section 7.4.3)

Pressurizer Level Regulating Systems (Section 7.4.4)

Reactor Regulating Systems (Section 7.4.1)

7.1.2 IDENTIFICATION OF SAFETY CRITERIA

The design bases, criteria, safety guides, information guides, standards, and other documents implemented in the design of systems listed in Section 7.1.1 are stated in the corresponding descriptive sections of the FSAR.

The following guides referenced in Section 7.0 of the "Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants." February 1972 were not available at the time of issue of the Millstone Unit Number 2 construction permit (December 11, 1970) and were therefore not included in the design or installation criteria used during construction. However, these guides are referenced in applicable description sections of the FSAR where the equipment provided meets the general guide requirements.

IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Systems," dated June 3, 1971.

IEEE Standard 308-1971, "Criteria for Class IE Electric Systems for Nuclear Power Generating Stations," dated September 16, 1971.

IEEE Standard 323-1971, "General Guide for Qualifying Class I Electric Equipment for Nuclear Power Generating Stations," dated April 1971.

IEEE Standard 336-1971, "Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment during the Construction of Nuclear Power Generating Stations," dated September 16, 1971.

IEEE 338-1971, "Periodic Testing of Nuclear Power Generating Station Protection Systems," dated September 16, 1971.

IEEE 344-1971, "Seismic Qualifications of Class I Electric Equipment for Nuclear Power Generating Stations," dated September 16, 1971.

Safety Guide 22, "Periodic Testing of Protection System Actuation Functions," dated February 17, 1972.

A description of the quality assurance procedures to be used during equipment fabrication, shipment, field storage, field installation, system and component checkout, and the records pertaining to each of these is contained in Sections 1.7, Appendix 1.B (located in the original FSAR dated August 1972), and 13.

It should be noted that the Local Power Density Trip is not a new trip but rather a modified version of the Axial Flux Offset trip described in the Calvert Cliffs FSAR. The name has been changed relative to the Calvert Cliffs FSAR in order to more accurately describe its function. The modifications made to the Thermal Margin, Axial Flux Offset and High Power trips described in the Calvert Cliffs FSAR can be categorized as follows:

1. Providing additional input to the trips, and

2. Providing additional processing equipment for the trips.

The Modifications were made to increase the operating flexibility over that provided by the equivalent trips described in the Calvert Cliffs FSAR.

Table 7.2.1.1–I shows the NSSS parameters that affect the acceptable fuel design limits mentioned above. Table 7.2.1.1–II shows what variables are monitored on both Calvert Cliffs and Millstone Unit 2, to determine the state of the NSSS parameters that affect the fuel design limits. It should be noted that the radial peaking factor is inferred from the core power measurement. This indirect measurement is accomplished through the Power Dependent CEA Insertion Limits (PDILS) given in the Technical Specifications. Also, the azimuthal tilt magnitude is not directly measured. The protective system set points are developed to: 1) take into account the worst core radial peaking factor that can result at any power level from CEAs full out up to CEAs inserted to their insertion limit, and 2) take into account an azimuthal tilt at its Technical Specification limit.

Table 7.2.1.1–III gives a comparison of the specific function of the subject trips, on Calvert Cliffs and Millstone Unit 2, with respect to the fuel design limits. The low flow trip is included in Table 7.2.1.1–III for completeness purposes only. Table 7.2.1.1–IV gives a comparison of the variables monitored, on Calvert Cliffs and Millstone Unit 2, for the trips listed in Table 7.2.1.1–III.

Thermal Margin Trip Modification:

The function of the Thermal Margin trip is the same on Millstone Unit 2 and Calvert Cliffs (see Table 7.2.1.1–III). The modification to the thermal margin trip consists of adding the axial flux offset as a measured input and the processing equipment needed to relate axial offset to thermal margin limits (see Table 7.2.1.1–IV). On Calvert Cliffs the combination of thermal margin and axial flux offset trip provides DNB and void fraction protection. This is accomplished by using the axial offset trip to limit the axial power distribution that can occur at any power level. This limiting distribution is then related to the thermal margin limit at that power level when generating the set points, since the thermal margin trip on Calvert Cliffs does not monitor the axial offset. The approach used on Calvert Cliffs does not allow credit to be taken for operation with axial offsets that are more favorable in terms of thermal margin than the offset that exists at the trip limit. Direct measurement of the axial offset in the thermal margin trip in Millstone Unit 2 allows this credit to be taken when it exists; thereby, 1) increasing margin to trip and improving operating flexibility, where 2) providing the same degree of protection as on Calvert Cliffs.

Axial Flux Offset Trip Modification (i.e., Local Power Density Trip):

The function of the axial flux offset trip on Calvert Cliffs is twofold:

1. assure that the fuel temperature limit (i.e, kw/ft limit) is not violated (See Table 7.2.1.1–III, and

2. limit the axial power distribution that can exist at any power level to that which was assumed in generating the thermal margin trip set points, as explained above.

The modifications made to the axial offset trip in Calvert Cliffs are:

1. a power dependent radial peaking penalty function has been included, and

2. the auctioneered higher of the ΔT and neutron flux power is used as the power input to generate the trip limits.

As shown in Table 7.2.1.1–I the radial peaking factor is one parameter that offsets the kw/ft of which the plant is operating. Table 7.2.1.1–II shows that this effect is inferred from the sensed power level. In Calvert Cliffs the axial offset trip setpoints are determined in a manner that is consistent with the worst core CEA insertion (radial peaking) that can occur at any power level. The trip functions to limit the axial peaking factor consistent with the assumed worst radial and measured power level such that the specified kw/ft limit is not exceeded. By generating trip setpoints in this manner, any change in CEA insertion limits during operation requires setpoints to be regenerated. The modification provided on Millstone Unit 2 is done to facilitate the regeneration of setpoints in the event that CEA insertion limits are modified.

In generating trip system setpoints, one must consider all measurement errors and uncertainties that can occur during anticipated operational occurrences. On Calvert Cliffs, the neutron flux power is used as the power input to the axial offset trip. One factor that is considered in generating trip setpoints is how much can this signal be distorted (i.e., uncertain) during transients that require this trip. Two major phenomena that result in neutron flux power measurement error are:

1. CEA shadowing: This effect results from the distortion of the radial power distribution. The out-of-core detectors “see” the fast neutron flux escaping from the peripheral fuel bundles. As CEAs are inserted or removed the power produced in the peripheral fuel bundles will vary relative to the core average power. The result is that the out-of-core detectors may indicate a power level different from the core average power.

2. Inlet Temperature Shadowing: This effect results from a change in density of the cold leg coolant which passes between the peripheral fuel bundles and the out-of-core detectors. The result is that as cold leg temperature decreases, from that value at which the out-of-core detectors had last been calibrated, the detectors will indicate a power level below the core average power. This is due to the fact that the incident neutron flux at the detectors will be decreased due to the increased coolant density.

The ΔT power, obtained from a calculation of hot and cold RTD measurements, is not influenced by these two phenomena. The use of the auctioneered higher of the ΔT power and neutron flux power provides a means of decreasing the uncertainty imposed on the core power measurement when generating the local power density trip setpoints.

The modifications made to the Calvert Cliffs axial flux offset trip on Millstone Unit 2 (local power density trip) provide 1) increased operating margin and increased system flexibility, while 2) still satisfying the same criteria (i.e., kw/ft protection) and providing the same degree of protection.

High Power Level Trip Modification:

The function of the high power trip on Calvert Cliffs is to provide 1) a maximum upper load limit on the NSSS, and 2) a reactor trip to assist the engineered safety system in the event of an ejected CEA accident. The function of the high power trip on Millstone Unit 2 is to provide:

1. an upper load limit on the NSS that is always a given percent of rated power above the steady-state operating power, and

2. assistance during an ejected CEA accident.

In the Calvert Cliffs design, the combination of the thermal margin and axial flux offset trips assure the integrity of the fuel design limits. If during a power excursion an axial flux offset or thermal margin trip does not occur, in which case neither would be required, a high power trip eventually results if the indicated power reaches 106.5% of rated.

As mentioned previously, the thermal margin and axial offset trip setpoints are generated by assuming that worst case radial peaks can occur at various power levels. This analysis must take into consideration the possibility of “carrying-up” high radial peaks to high power levels. This effect may occur when CEAs are in manual control and a power excursion ensues and must be accommodated up to the point of the high power trip at 106.5% or rated power. The high power trip then essentially provides a mechanism by which to limit the radial peaking that must be assumed in generating the thermal margin and axial offset trip setpoints. The modification to the high power trip on Millstone Unit 2 provides a means of limiting any power excursion to approximately 10% increase above the initial power level. The trip then provides a means of limiting the radial peaking that can be “carried-up” during a power excursion starting below 100% power. When the initial power level is 100% of rated, the 110% high power trip setpoint provides this assurance. Since the maximum power excursion is always limited by a set amount, credit can be taken for the reduced radial peak that may be “carried-up.” This credit is reflected in the generation of the thermal margin and local power density trip setpoints and results in increased operating margins. The modification of using the auctioneered high of the ΔT and neutron flux power as the trip variable for the high power trip is provided to increase system accuracy as explained previously.

The design criteria for the thermal margin, local power density and high power level trip are discussed in Section 7.2.1.2.

7.2 REACTOR PROTECTIVE SYSTEM

7.2.1 DESIGN BASIS

7.2.1.1 Functional Requirements

The reactor protective system (RPS) consists of the sensors, amplifiers, logic, and other equipment necessary to monitor selected nuclear steam supply system (NSSS) conditions and to effect reliable and rapid reactor shutdown if any one or a combination of conditions deviates from a preselected operating range. The system functions to protect the core and reactor coolant pressure boundary (RCPB). The Millstone Unit 2 Protective System is functionally identical to that provided for Calvert Cliffs (NRC Docket Numbers 50-317, 50-318 through Amendment 19), except for the following modifications:

A reactor coolant pump (RCP) underspeed trip has been added (Section 7.2.3.3.1). This trip is not credited to provide protection during any accident analysis event.

Automatic axial flux tilt protection has been added through the addition of a Local Power Density (LPD) Trip (Section 7.2.3.3.10) and modification of the Thermal Margin/Low Pressure (TM/LP) Trip (Section 7.2.3.3.7).

The High Power Trip has been modified to use a variable setpoint (Section 7.2.3.3.2).

The trip logic matrix relay power supplies have been split so that each matrix power supply supplies only two relays, thus preventing a single power supply failure from causing a spurious reactor trip (Figure 7.2–2).

The range of the T_{cold} channel inputs to the RPS has been changed in support of the modification to the TM/LP Trip (Section 7.2.3.3.7) from 515°F to 615°F to 465°F to 615°F. This was to assure the existence of a valid ΔT power signal during transients involving significant decreases in core inlet temperature.

The function of the Zero Power Mode Bypass (Section 7.2.3.3.12) has been expanded to include removing the ΔT power component of the power signal Q in order to allow for special physics startups when system temperatures are below the operating range.

The Low Reactor Coolant Flow Trip (Section 7.2.3.3.3 acts on a ΔP signal.

Signal Commons between Nuclear Instrumentation Channels have been isolated to prevent channel interaction. (Section 7.2.3.2.1)

The Power Trip Test Interlock (Section 7.2.4) has been expanded to provide protection against the Nuclear Instrumentation Test Select switch being left in an off-normal position after testing.

The function of the RPS on all Combustion Engineering (CE) supplied plants has always been threefold:

1. To prevent overpressurization of the primary coolant systems,

2. To assist the engineered safety system in mitigating the consequences of accidents (e.g., loss-of-coolant accident (LOCA)), and

3. To assure that acceptable fuel design limits are not exceeded during anticipated operational occurrences.

The RPS supplied with Millstone Unit 2 is, in this sense, functionally identical with that supplied on Calvert Cliffs.

The assurance that the protective system fulfills its threefold function is provided by choosing a complement of trips that monitor pertinent parameters that: 1) are related to specified limits, and 2) that are affected during accidents which may lead to a violation of these specified limits.

The particular trips under discussion (i.e., Thermal Margin, LPD and High Power) have as their prime function the assurance that acceptable fuel design limits are not exceeded during anticipated operational occurrences. Anticipated operational occurrences are defined in General Design Criteria (GDC), Appendix A of 10 CFR 50 as "...those conditions of normal operation which are expected to occur one or more times during the life of the nuclear power unit...".

Acceptable fuel design limits are described in Section 3.2.3.

7.2.1.2 Design Criteria

The RPS is designed to the following bases to assure adequate protection for the core:

a. Instrumentation conforms to the provisions of the Institute of Electrical and Electronic Engineers (IEEE), Criteria for Nuclear Power Plant Protection Systems (IEEE 279-1971).

b. No single component failure can prevent safety action.

c. Four independent measurement channels are provided for each parameter that can initiate safety action.

d. Channel independence is assured by separate connection of the sensors to the process systems and of the channels to vital instrument buses.

e. The four measurement channels provide trip signals to six independent logic matrices, arranged to effect a two-out-of-four coincidence logic having outputs to four independent trip paths.

f. A trip signal from any two-out-of-four protective channels on the same parameter causes a reactor trip.

g. When one of the four channels is taken out of service, the protective system logic may be changed to two-out-of-three coincidence for a reactor trip by bypassing the out of service channel.

h. The protective system AC power is supplied from four separate vital instrument buses.

i. Open circuiting, or loss of power supply for the channel logic, initiates an alarm and a channel trip.

j. The trip logic matrices assume the nonconducting state to provide a tripping function.

k. The RPS can be tested with the reactor in operation or shut down.

l. The manual trip system is independent of the automatic trip system.

m. Trip signals are preceded by pretrip alarms to alert the operator of undesirable operating conditions in cases where operator action can correct the abnormal condition and avoid a reactor trip.

n. The RPS components are independent of control systems.

o. All equipment, including panels, and cables associated with the RPS, are marked with colored markers or nameplates in order to facilitate identification.

p. Electrical circuit isolation is provided between the RPS, annunciators, and the plant computer.

q. There are no RPS instrumentation transmitters for which the trip set points are within 5 percent of the high or low end of the calibrated range, or within 5 percent of the overall instrument design range.

7.2.2 DISCUSSION

The RPS meets the general requirements of the applicable sections of the below listed guides although they were not available at this time the construction permit for Millstone Unit Number 2 was issued. (December 11, 1970).

IEEE Standard 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," dated June 3, 1971.

IEEE Standard 336-1971, "Installation, Inspection and Testing Requirements for Instrumentation and Electric Equipment during the construction of Nuclear Power Generating Stations," dated September 1971.

IEEE Standard 338-1971, "Trial Use Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems," dated September 16, 1971.

Regulatory Guide 1.22, "Periodic Testing of Protection System Actuation Functions," (Safety Guide 22) dated February 17, 1972.

Combustion Engineering Topical Report CENPD-11 "Reactor Protection System Diversity," W. C. Coppersmith, C. I. Kling, A. T. Shesler, and B. M. Tashjian CENPD, February 1971) demonstrates that functional diversity has been incorporated in the protective system design.

7.2.3 SYSTEM DESCRIPTION

7.2.3.1 General

As shown in Figures 7.2–1 and 7.2–2, the RPS consists of four trip paths operating through the coincidence logic matrices to maintain power to, or remove it from, the control element drive mechanisms (CEDM). Four independent measurement channels normally monitor each plant parameter which can initiate a reactor trip. Individual channel trips occur when the measurement reaches a preselected value. The channel trips are combined in six two-out-of-two logic matrices. Each two-out-of-two logic matrix provides trip signals to four one-out-of-six logic units, each of which causes a trip of the reactor trip switchgear in the AC supply to the CEDM power supplies. Each CEDM power supply source is separated into two branches.

Reactor trip is accomplished by deenergizing the CEDM coils allowing the shutdown and regulating CEAs to drop into the core by gravity. Reactor trip is initiated by the conditions described in Section 7.2.3.3; the reactor trip and pretrip alarm set points are listed in Table 7.2-1.

All protective systems that actuate reactor trip and engineered safety features (ESF) components conform to the regulation in effect at the time of procurement including relevant sections of the GDC, Appendix A to 10 CFR 50.

The cabinets of the RPS are appropriately tagged A, B, C, D to distinguish between channels. The RPS is distinguished from nonsafety related equipment by the use of colored nameplates. At termination points the incoming and outgoing cables of the RPS are appropriately tagged to identify the channel and to distinguish between channels.

Physical separation between channels is accomplished by feeding each of the four independent signal inputs into a separate cabinet. The four cabinets are separated from each other by fireproof barriers. Logic matrix and other interconnections between the four cabinets are made by running interconnecting wiring through rigid metallic conduits penetrating the barrier between each of the cabinets. All barrier penetrations are sealed with fireproof material.

The ability of the RPS to protect the core and the RCPB has been established by the analysis discussed in Chapter 14. The analyses show that the plant and fuel design limits are not exceeded during anticipated operational occurrences such as loss of load or inadvertent withdrawal of regulating control element assembly (CEA) groups. The provision of more restrictive set points to permit operation at less than full flow is described in Section 7.2.3.3.3 and where appropriate, such operation is considered in the analyses. No part of the RPS is dependent on plant instrument air systems.

Water is not used for cooling any part of the RPS.

7.2.3.2 System Components

7.2.3.2.1 Signal Generation

Four independent instrument channels are used to generate the signals necessary to initiate the automatic reactor trip action. The signal cable routing and readout drawer locations are separated and isolated to provide channel independence.

Isolation Assemblies provide independence and separation of signal common references through high impedance isolation of signals and reduction of common mode signals within FET repeating amplifiers. Isolation of signal common between channels prevents channel interaction. Isolation of signal common for signals originating outside the RPS cabinet prevents noise pick up.

7.2.3.2.1.1 Wide-Range Logarithmic Channels

The four wide range logarithmic channels each obtain signals from two high sensitivity fission chambers. The fission chambers are grouped axially and located on the reactor cavity wall around the reactor. The output from the fission chambers are conditioned and amplified in the cable vault amplifier assemblies and transmitted to the signal processing drawers in the control room. The signal processing drawers further process the detector signal into signals that represent the source range logarithm of count rate and the rate of change of count rate, and the wide range logarithm of reactor power and the rate of change of reactor power (see Figure 7.2–7).

7.2.3.2.1.2 Power-Range Safety Channels

The signals for each of the four power range safety channels are obtained from one of the four detector assemblies located on the reactor cavity wall around the reactor. Each assembly consists of two uncompensated ion chambers stacked vertically to monitor the full length of the core. The DC current signals from each set of ion chambers are fed separately and directly to the power range safety channel drawer assemblies located in the control room. The ion chambers cover the range from 0.1 percent to 200 percent power (see Figure 7.2–8).

7.2.3.2.1.3 Flow, Water Level, Pressure Temperature and Reactor Coolant Pump Speed

The flow, water level, pressure temperature and RCP speed signals are each generated by separate sets of transmitters. Flow is measured by monitoring the pressure difference between the hot leg piping and the steam generator outlet plenum. Steam generator water level and pressure are monitored in each steam generator. The reactor coolant system (RCS) pressure is measured in the pressurizer. Temperature measurements are taken from the reactor inlet and outlet piping in each loop.

Piping and connections for these transmitters are separated and isolated to provide independence. The output of each transmitter is an ungrounded current loop supplying signal receivers and bistable trip modules.

7.2.3.2.2 Logic

Refer to Figure 7.2–2 for the following discussion.

Each measurement channel which can initiate protective action operates a channel trip unit; each trip unit includes three sealed, electromagnetically actuated reed relays and associated contacts. Four trip units are provided for each trip condition, e.g., high pressurizer pressure.

The relays in each trip unit are numbered one, two, and three. The normally open contacts from the Number 1 relay group of Channel A are connected into a two-out-of-two logic matrix with Channel B relay contacts. (The normally open contacts are used for the logic ladders so that the relays are energized and the contacts closed under operating conditions).

The Number 2 and Number 3 relay contacts are similarly connected into two other two-out-of-two logic matrices with Channel C and Channel D relay contacts. With the number 2 and number 3 relay contacts of Channels, B, C, and D similarly arranged in BC, BD and CD combinations of two-out-of-two logic matrices, there is a total of six two-out-of-two logic matrices, forming a two-out- of-four coincidence logic with respect to the input channels.

At the output of each of the six trip logic matrices is a set of four sealed, electromagnetically actuated relays. These sets are designated the AB, AC, AD, BC, BD, and CD logic trips. The contacts from one relay of the logic trip set from each logic matrix output are placed in series with corresponding contacts from the remaining sets in each of the four trip paths. Each of these paths is the power supply line to one of the trip breaker control relays, K1 through K4, whose contacts provide actuation of under voltage and shunt trips on the trip circuit breakers, thus interrupting the AC power to the CEDMs. Deenergizing of any one trip breaker control relay interrupts (opens) one trip path and trips the two breakers controlled by that trip path. Deenergizing any set of four logic trip relays causes an interruption of all trip paths which results in a reactor trip. This logic is shown on the RPS Block Diagram, Figure 7.2–1.

The CEDMs are separated into two groups. The CEDM power supplies in each group are supplied in parallel, with three-phase AC power from the motor-generator sets. Two full-capacity motor-generator sets, each with a one-second ride through capability, are provided. The loss of either set does not cause a release of the CEAs. Each power supply source is separated into two branches. Each side of each branch line passes through two trip circuit breakers (each actuated by a separate trip path) in series so that, although both sides of the branch lines must be deenergized to release the CEAs, there are two separate means of interrupting each side of line. This arrangement provides means for the testing of the protective system.

If one of the trip units is to be removed for maintenance, the logic matrices may be changed from a two-out-of-four trip to a two-out-of-three trip by the operation of the logic bypass switch (shown on the output of the trip module, Figure 7.2–2). One key-operated switch is provided for each trip unit. Only one key is provided for the trips for any one variable to ensure that only one of the group of four units monitoring that variable could be bypassed at one time.

The operation of the key-operated switch to bypass the trip function of a single bistable trip unit is indicated by a light on the face of the bistable trip unit and an annunciator on the main control board. This meets the requirements of paragraph 4.13 in IEEE-279 in that it provides continuous indication of the bypass in the control room.

Where the trip is to be allowed only in selected power ranges, a neutron flux signal is utilized to inhibit the action of the trip units. A manually actuated inhibit action may, under administrative control, be applied to the low reactor coolant flow, thermal margin and low steam generator pressure trips for zero power testing. The inhibits on reactor coolant flow and thermal margin are automatically removed above a preset power. The inhibit on steam generator pressure is automatically removed above a preset pressure. An additional feature of this zero power inhibit is to remove the ΔT power component of the power signal Q which is described in Section 7.2.3.3.7. This prevents RCS temperature which would cause false trips on high power and LPD during low power testing. Protective system criteria are met by this use of neutron flux signals to provide multiple independent inhibit or reset signals.

7.2.3.3 System Operation

7.2.3.3.1 Reactor Coolant Pump Under-Speed

The trip is not credited in any accident analysis event. It is provided to improve response to a low-coolant-flow condition resulting from loss of supplied power to the RCPs.

RCP shaft speed is sensed by a magnetic sensor which transmits a signal to a frequency-voltage proportional transmitter. The voltage signal, which is proportional to pump shaft speed is then applied to a bistable trip unit. (See Figure 7.2–10). The trip is initiated by two-out-of-four coincidence logic from the four channels.

Each of the four RCPs is equipped with an independent speed sensing system, with each channel actuating independently.

The zero power mode bypass switch allows the trip to be bypassed below 10^{-4} percent power. The trip bypass is automatically removed prior to increasing reactor power to 10^{-4} percent power.

7.2.3.3.2 High Power Level

A reactor trip in power level Q (see Section 7.2.3.3.7) is provided to trip the reactor in the event of a reactivity excursion too rapid to result in a high pressure trip and to help prevent violation of the CEA position vs. power level assumed in the Thermal Margin and LPD trips. The high power trip setpoint can be set no more than a predetermined amount above the indicated plant power. Operator action is required to increase the setpoint as plant power is increased. The setpoint is automatically decreased as power decreases.

The setpoint and Q are compared in a bistable trip unit in each of the four safety channels. The high power trip is initiated by two-out-of-four coincidence logic from the four safety channels.

Figure 7.2–11 shows the operation of the system. If Q decreases, the setpoint Q_{TR} follows it, remaining above Q by a fixed, adjustable bias Q_b. If Q now increases, the setpoint remains at the minimum value of Q + Q_b last achieved, until reset by the operator.

The system must be capable of holding the setpoint Q_{TR} at the previous minimum of Q + Q_b indefinitely. This requirement precludes storing Q_{TR} by purely analog means. For this reason, the signal is stored as a digital word.

The reset circuit is designed to apply a momentary signal to the appropriate terminal of the digital storage device when a pushbutton is pressed. This causes Q_{TR} to achieve the current value of Q + Q_b. The reset circuit is buffered to permit locating the pushbutton outside the RPS.

The signal Q_{TR} is limited so that, regardless of the logic described above, it cannot go above or below limits set by potentiometers.

Other circuits generate a pretrip limit for the bistable trip unit, as well as a contact closure to alert the operator when power increases after reaching a minimum. The pretrip alarm provides audible and visual annunciation in addition to CEA withdrawal prohibit signals.

Q and Q_{TR} are processed and buffered for remote display on the main control board. Q is also taken to the Control Element Drive System (CEDS) and the Plant Computer for use in the power dependent insertion limit calculation.

7.2.3.3.3 Low Reactor Coolant Flow

This reactor trip is provided to protect the core against DNB in the event of a coolant flow decrease.

The flow measurement signals are provided by measuring the differential pressure across each of the two steam generators. Each steam generator differential pressure signal is proportional to the square of the steam generator mass flow rate. These signals are summed to provide a signal that is proportional to the square of the reactor vessel mass flow rate. The measured signal is compared to a pre-determined trip setpoint in the reactor protection system trip bistable. This configuration is shown in Figure 7.2–4, and is repeated in each of four redundant channels. A reactor trip is initiated when the measured value falls below the bistable trip setting in two-out-of-four coincident channels.

Pre-trip alarms are similarly provided to warn of decreased coolant flow conditions.

The Zero Power Mode Bypass switch allows this trip to be bypassed for subcritical testing of CEDMs. The trip bypass is automatically removed prior to increasing reactor power to 10^{-4} percent power.

Reactor coolant pump operating requirements and surveillance requirements are defined by Technical Specifications.

7.2.3.3.4 Low Steam Generator Water Level

An abnormally low steam generator water level indicates a loss of steam generator secondary water inventory. If not corrected, this would result in a loss of capability for removal of heat from the RCS.

The low steam generator water level reactor trip protects against the loss of feedwater flow accident (see Section 14.2.7) and assures that the design pressure of the RCS will not be exceeded. The trip set point specified in Table 7.2-1 assures that sufficient water inventory will be in the steam generator at the time of trip such that steam generator dryout does not occur before the auxiliary feedwater (AF) delivers sufficient flow to remove decay heat and recover steam generator water level.

A reactor trip signal is initiated by two-out-of-four logic from four independent channels. Each channel actuates on the lower of two signals from two downcomer level differential pressure transmitters, one on each steam generator. Audible and visual pretrip alarms are actuated to provide for annunciation of the approach to reactor trip conditions.

7.2.3.3.5 Low Steam Generator Pressure

An abnormally high steam flow from one of the steam generators (e.g., that which would occur as the result of a steam line break (SLB)) would be accompanied by a marked decrease in steam pressure. To protect against an excessive rate of heat extraction from the steam generators and subsequent cooldown of the reactor coolant following an SLB, a reactor trip is initiated by low steam generator pressure.

A reactor trip signal is initiated by two-out-of-four logic from four independent channels. Each channel actuates on the lower of two signals from two pressure transmitters: one on each steam generator. Audible and visual pretrip alarms are actuated to provide for annunciation of approach to reactor trip conditions.

Signals from these pressure transmitters initiate closure of the main steam isolation valves (MSIV) on a two-out-of-four coincidence of low pressure in either steam generator.

A bypass is provided for the low steam generator pressure trip to allow performance of zero power physics testing. Bypass is accomplished manually by means of a key-operated switch in each channel. The manual bypass is enabled only below a preset steam pressure and is automatically removed above this set point. Figure 7.2–5 is a functional diagram of this circuit.

The trip bypass is initiated manually by turning a switch to the BYPASS position. The bypass is removed, regardless of the manual switch position if the auctioneered higher of the steam generator pressures exceeds a predetermined set point. When the manual switch is in the "Off" position, there is no bypass of the low steam generator pressure trip function.

For setting and testing the bistable device by use of the trip tester, a TEST SELECT switch is provided to disconnect the signal not being tested. This can only be done if the manual bypass switch is in the OFF position.

The contact testing system consists of two pushbuttons, one for auto bypass removal test and one for manual bypass removal test. The purpose of these tests is to check the status of the bypass circuit contacts. These tests do not alter or change the contacts from either an open or closed position.

Pressing the AUTO TEST pushbutton completes a path through a contact to the light; thus the light being on indicates that bypass is allowed by the automatic removal circuit. Pressing the MAN TEST pushbutton similarly tests the manual contact. Pressing both pushbuttons energizes the light regardless of bypass status; this tests the light. The light is also on for both the manual and automatic contacts closed, i.e., when the trip bypass is in effect.

The reactor trip set point of 691 psia (Table 7.2-1) is sufficiently below the full load operating pressure so as not to interfere with normal operation, but still high enough to provide the required protection in the event of excessively high steam flow.

7.2.3.3.6 High Pressurizer Pressure

A reactor trip for high pressurizer pressure is provided to prevent excessive blowdown of the RCS by relief action through the pressurizer safety valves. A reactor trip is initiated by two-out-of- four coincidence logic from the four independent measuring channels if the pressurizer exceeds 2397 psia. This signal simultaneously opens the power-operated relief valves (PORV).

The trip signals are provided by four independent narrow range pressure transducers measuring the pressurizer pressure. Pretrip alarms are initiated if the pressurizer pressure exceeds 2350 psia as indication of the approach to reactor trip conditions.

7.2.3.3.7 Thermal Margin/Low-Pressure Trip

The TM/LP trip is provided for two purposes. The thermal margin portion of the trip, in conjunction with the low reactor coolant flow trip, is designed to prevent the reactor core safety limit on DNB from being violated during anticipated operational occurrences. The low pressurizer pressure portion of the trip functions to trip the reactor in the event of a LOCA.

A reactor trip is initiated whenever the RCS pressure signal drops below either 1865 psia or a computed value as described below, whichever is higher. The computed value is a function of the higher of ΔT power or neutron power, reactor inlet temperature, the number of RCPs operating and the axial offset. Consistent with the Technical Specifications, the minimum value of reactor coolant flow rate, the maximum azimuthal tilt, the maximum CEA deviation permitted for continuous operation are assumed in the generation of this trip function. In addition, CEA group sequencing in accordance with the Technical Specifications is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a High Power Level trip is assumed.

Figure 7.2–6 and 7.2–11 describe the operation of this trip system. The higher of the two inlet temperatures is used in the TM/LP calculation.

Figure 7.2–13 shows a block diagram of the thermal power calculation.

The calculation begins with the generation, by temperature transmitters, of currents representing the cold and hot leg temperatures in each loop. By forcing these currents through precision resistors and utilizing the resulting voltage drops, voltages representing cold leg temperatures (T_{c1} and T_{c2}) and hot leg temperature (T_h) are sent to the calculator. The latter signal is the average T_h for the two loops.

In the calculator, the higher cold leg temperature signal is selected and subtracted from the hot leg temperature signal to determine the temperature rise. The calculator generates terms proportional to the first and second powers of the temperature rise and to the product of temperature rise and cold leg temperature. These three terms represent thermal power for four pump operation and steady state conditions, accounting for coolant density, specific heat, and flow rate variations with temperature and power.

The sum of these terms represents the core power for four-pump operation under steady state or mild transient conditions.

The coefficient of the term proportional to the temperature rise ($K\sigma$) is set by the potentiometer labeled "ΔT Power Calibrate" on the Reactor Protective System Calibration and Indication Panel (RPSCIP) front panel. This factor is adjusted to make the thermal power calculation agree with the plant calorimetric calculation.

The thermal power (B) is subtracted from the nuclear power (ϕ), generated by the NI Channel, and the difference is displayed on a meter with a range of -10 percent to +10 percent of full power. The meter has adjustable upper and lower setpoints. The contacts energize a local light when the deviation goes outside the range defined by the setpoints.

To make the nuclear power signal agree with the thermal power and/or the plant calorimetric calculation, a potentiometer labeled "Nuclear Power Calibrate" is provided. This potentiometer adjusts the gain of the NI Channel from 0.8 to 1.33. An auctioneering circuit selects the higher of nuclear power or thermal power for use in the remainder of the system.

The signal Q, the maximum of nuclear or thermal power, is modified by a CEA position function. The resulting signal is then augmented by an axial factor which is generated in the LPD Trip section as shown on Figure 7.2–12 and described in Section 7.2.3.3.10. The resulting signal is called Q_{DNB}.

A pressure setpoint Pvar is calculated as a linear function Q_{DNB} and of the modified inlet temperature described above.

An auctioneering circuit selects the maximum of this calculated pressure setpoint and a constant pressure P_{min}, and sends the resulting signal to the trip unit as a downscale trip setpoint. Trip will occur if the primary pressure drops below the calculated setpoint or below 1865 psia, whichever is larger. A pretrip setpoint, 75 psi above the trip point, is also generated.

The trip signal is initiated by a two-out-of-four coincidence logic from four independent safety channels, and audible and visual pretrip alarms are actuated to provide for annunciation on approach to reactor trip conditions. The pretrip action also initiates a CEA withdrawal prohibit.

The zero power mode bypass switch allows this trip to be bypassed for low power testing. The trip bypass is automatically removed prior to increasing reactor power to 10^{-4} percent power.

The Thermal Margin trip setpoint is processed and buffered for remote display on the main control board in four dual indicators which compare the trip setpoint with indicated pressurizer pressure.

7.2.3.3.8 Loss of Turbine

The trip for loss of turbine is an equipment protective trip and is not required for reactor protection. (Refer to Chapter 14).

This trip is initiated above a preset power level, by actuation of 2 of 4 low hydraulic fluid pressure switches associated with the turbine-generator control systems. Its purpose is to help avoid the lifting of the steam generator safety valves during the system transient after a turbine trip, thus extending the service life of these valves. Since credit has not been taken for the equipment protective trips in the Safety Analysis of the plant, they do not fall within the scope of IEEE 279. In the case of the Millstone Unit Number 2 RPS, the design criteria listed in Section 7.2.1.2 (which includes IEEE 279) apply to all trip functions including the equipment protective trips.

7.2.3.3.9 High Containment Pressure

A trip is provided on high containment pressure in order to assure that the reactor is tripped concurrent with safety injection actuation.

Four pressure measurement channels provide analog signals to bistable trip units which are connected in a two-out-of-four coincidence logic to initiate the protective action if the containment pressure exceeds a preselected value.

7.2.3.3.10 High Local Power Density Trip

The minimum power level required to produce centerline melt in Zircaloy clad uranium fuel rods is defined as the Fuel Centerline Melt Linear Heat Rate (FCMLHR) limit and is expressed in KW/ ft. This FCMLHR is determined using the methodology of XN-NF-82-06(P)(A) Revision 1 and Supplements 2, 4 and 5 ("Qualification of Exxon Nuclear Fuel for Extended Burnup," Exxon Nuclear Company, October 1986.) The high LPD trip is provided to prevent the peak LPD in the fuel from exceeding the FCMLHR limit during anticipated operational occurrences thereby assuring that the melting point of the UO_2 fuel will not be reached.

A reactor trip is initiated whenever the axial offset exceeds either a high or low calculated setpoint as described below. The axial offset is calculated from upper and lower ex-core neutron detector channels. The calculated setpoints are generated as a function of the core power level with the CEA group position being inferred from the core power. The trip is automatically bypassed below 15 percent power.

Consistent with the Technical Specifications, the maximum azimuthal tilt and the maximum CEA deviation permitted for continuous operation are assumed in generation of the setpoints. In addition, CEA group sequencing in accordance with Technical Specification is assumed. Finally, the maximum insertion of CEA banks which can occur during any anticipated operational occurrence prior to a High Power Level Trip is assumed.

Figure 7.2–13 shows a block diagram of a typical channel. Circuits in the Power Range Safety Channel generate signals proportional to the sum of and the difference between the upper and lower detector outputs. An axial offset signal is formed as a linear function of the ratio of the difference to the sum and compared with upper and lower limits generated from a modified power signal described in Section 7.2.3.3.8. The offset signal is also used in the Thermal Margin Trip as previously described.

If the axial offset exceeds either calculated limit, a contact in the calculator opens and deenergizes the trip relays in an auxiliary trip unit. The pretrip relay is similarly released if a narrower envelope is exceeded.

7.2.3.3.11 Manual Trip

A manual reactor trip is provided to permit the operator to trip the reactor. Depressing two pushbutton switches on the control panel causes interruption of the AC power to the CEDM power supplies. The manual trip function is testable during reactor operation.

7.2.3.3.12 Bypass Operation

All trips are normally cleared before startup. (The loss-of-turbine trip is automatically bypassed below 15 percent power.) For some operations, it may be desirable to perform a reactor startup with some reactor parameters at values which would normally cause a trip. For these special operations, zero power mode bypass switches may be used to bypass the low flow, RCP underspeed, and the low TM/LP trip functions. Four bypass key switches are provided. Each bypass switch removes all three trip functions from one of the four protective system channels. These bypasses are automatically removed above 10^{-4} percent power. A manual bypass is provided to allow startup with a low steam generator pressure.

7.2.4 TESTING

IEEE 338-1971, "Trial Use Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems," September 1971, provides guidance for development of procedures, equipment and documentation of periodic testing. The bases for and the scope and means of testing are described in this section. Test intervals and their bases are included in the Technical Specifications. The organization for testing and for documentation is described in Chapter 13.

Since operation of the protective system will be infrequent, the system is periodically and routinely tested to verify its operability. A complete channel can be individually tested without initiating a reactor trip or violating the single failure criterion, and without inhibiting the operation of the RPS.

The RPS is capable of being checked from the trip unit input through the power supply circuit breakers of the CEDMs. The majority of the components in the protective system can be tested during reactor operation. The remainder of the components can be checked by comparison with similar channels or channels that involve related information. These components, which are not tested during reactor operation, will be tested during scheduled reactor shutdown to assure that they are capable of performing the necessary functions. Minimum frequencies for checks, calibration and testing of the RPS instrumentation are given in Section 4 of the Technical Specifications. Overlap in the checking and testing is provided to assure that the entire channel is functional. The use of individual trip and ground detection lights, in conjunction with those provided at the supply bus, assure that possible grounds or shorts to another source of voltage may be detected.

During reactor operation the measuring channels are checked by comparing the outputs of similar channels and cross-checking with related measurements. The trip units are tested by inserting a voltmeter in the circuit, noting the signal level, initiating a test input and noting signal level required to effect trip action. This provides the necessary overlap in the testing process and also enables the test to establish that the trip can be effected within the required tolerances. The test signal is provided by a test signal generator which is connected to the trip module at the signal input terminals. With the test signal generator connected, the desired signal is selected and then inserted into the trip unit by depressing the manual test switch. The test circuit permits various rates of change of signal input to be used. Trip action (opening) of each of the trip unit relays is indicated by individual lights on the front of the trip unit. The pretrip alarm action is indicated by a separate light.

The sets of logic trip relays at the output of each logic matrix are tested one at a time. The test circuits in the logic permit only one logic ladder to be opened and one set of relays to be held at a time; the application of hold power to one set denies the power source to the other sets. In testing a logic trip set (e.g., AB) a holding current is initiated in the test coils of the logic trip relays by turning the matrix relay trip test switch to "off" and depressing the matrix logic AB test pushbutton switch. Operation of the matrix trip test switch initiates a deenergizing current in the test coils of a parallel pair of trip unit relays. With the ladder logic relay contacts open, the logic trip relays may be deenergized one at a time (by rotating the matrix relay trip test switch) to open the associated trip breakers. Indicator lights on the trip status panel provide verification that coil operation and trip breaker actuation conditions have occurred.

The response time from an input signal to the protection system trip units through the opening of the trip circuit breakers is verified by measurement in accordance with the Technical Specifications and the Technical Requirements Manual.

Periodic testing can be carried out from the control room to ensure the continuity of the measurement loop. A supplementary signal is introduced into the measurement loop that is bypassed and the response to this signal is indicated on a meter in the protection system. This verifies the continuity of the loop and insures its operability.

The RPS is manufactured under strict engineering and quality control specifications. These specifications require that the equipment be inspected for workmanship, proper materials and channel separation as required by IEEE-279-1968. Furthermore, all intra- and inter-connection wiring is tested for continuity and an insulation test is performed between each conductor and chassis ground and between each individual pair of connectors. An operational test is performed on the system during which time input signals are simulated to ensure that the protective system is capable of producing the proper trip signals.

A Power Trip Test Interlock feature is provided to assure protection against the consequences of certain signal selection switches being left in off-normal positions at the completion of testing. The position of these switches, the Test Select switch in conjunction with the Test Enable switch in the Power Range Nuclear Instrumentation and the ΔT Power Calculator Test Switch in the RPSCIP, may be changed during testing without causing a trip by bypassing the High Power, TM/LP, and LPD Trips. If these switch positions should be changed during normal operation or if any of these channels that have been bypassed for testing should be returned to normal prior to returning the switches to normal, trips will occur on the affected channels.

The reactor trip circuit breakers are provided with the capability of disabling the shunt trip coils for time response testing of the undervoltage trip coils. This feature is provided by means of a handswitch located on the trip breaker switchgear. These handswitch contacts are shown in Figure 7.2–1 and 7.2–2. During this testing process, no more than one handswitch should be placed in the test position at the same time and annunciation is provided on C04 that warns the operator that the shunt trip coil is disabled.

7.2.5 SYSTEM EVALUATION

7.2.5.1 General

The RPS is designed to limit reactor power and coolant conditions to levels within the design capability of the reactor core. Instrument performance characteristics, response time, and accuracy are selected for compatibility with and adequacy for the particular function. Trip setpoints are established by analysis of the system parameters. Factors such as instrument inaccuracies, bistable trip times, CEA travel times, valve travel time, circuit breaker trip times, and pump starting times are considered in establishing the margin between the trip setpoints and the safety limits. The time response of the sensors and protective systems are evaluated for abnormal conditions.

Since all uncertainty factors are considered as cumulative for the derivation of these times, the actual response time may be more rapid. However, even at the maximum times, which are added to the CEA drop time, the system provides conservative protection.

The wiring in the protective system is grouped so that no single fault or failure, including either an open or shorted circuit, will negate protective system operation. Signal conductors are protected and routed independently.

Loss of or damage to any one path will not prevent the protective action. Sensors are piped so that blockage or failure of any one connection does not prevent protective system action. The process transducers located in the containment building are specified and rated for the intended service.

These components which must operate in the LOCA environment are rated for the LOCA temperature, pressure and humidity conditions. Results of type test are used to verify these ratings. In the control room the nuclear instrumentation and protective system trip paths are located in four compartments. Mechanical and thermal barriers between these compartments reduce the possibility of common event failure. Outputs from the components in this area to the control boards are buffered so that shorting, grounding, or the application of the highest available local voltage does not cause channel malfunction. Where signals originating in the RPS feed the computer, buffering is used to ensure circuit isolation; where the RPS is feeding annunciators, isolation is ensured through the use of relay contacts.

The protective system is designed such that the deenergized state initiates a channel trip. This feature ensures that if channel continuity is lost, that channel will fail in a safe condition. Module withdrawal is indicated by lights on the RPS panel and annunciators and alarms above the main console. The modules are not interlocked to prevent withdrawal but are set up such that withdrawal of one module causes a channel trip and withdrawal of a second module causes full trip. If a channel is in the bypassed condition, withdrawal of any other two modules of that parameter will cause a full trip since the system is in the two-out-of-three trip mode. Only one set of keys is available to the plant operator allowing only one of the four channels of any one parameter to be bypassed at any one time. Strict administrative control ensures that this requirement is not violated. Indication of test and bypass conditions or removal of any channel from service is given by lights on the protection system front panel and an audible/visible alarm annunciator. If a protection system channel is removed from service either by a failure in that channel or by deenergization for maintenance purposes, then that channel will go to the tripped condition which is indicated and alarmed. Bypasses are alarmed and indicated on the main control panel. Automatic removal of a bypass is also indicated by the main control panel annunciator.

The protective system is designed and arranged to be able to perform its function with a single failure of any component. Some of the faults and their effects are described below.

7.2.5.2 Analog Portion of the System

a. A loss of signal in a measurement channel initiates protective channel trip action for those parameters which normally trip on a decreasing input. These include Low RCP Speed, Low Reactor Coolant Flow, Low Steam Generator Water Level, Low Steam Generator Pressure, and Thermal Margin/Low Pressure pressure input. Parameters which trip on an increasing input, including high power level, high pressurizer pressure and high containment pressure, will not trip on a loss of input signal.

b. Shorting of the signal leads to each other has the same effect as a loss of signal. Shorting a lead to a voltage has no effect since the signal circuit is ungrounded. Periodic testing will determine if grounds or applications of potential to the signal circuit exist.

c. Single grounds of the signal circuit have no effect. Periodic checking of the system will assure that the circuit remains ungrounded.

d. Open circuit of the signal leads causes a channel trip signal.

7.2.5.3 Logic Portion of the Circuit

a. Inadvertent operation of the relay contacts in the matrices will be identified by the indicating lights.

b. Shorting of the pairs of contact in the matrices will prevent the trip relay sets from being released. Such shorts are detectable in the testing process by observing that the trip relay sets cannot be dropped out. Testing is accomplished by successive opening of the logic matrix contact pairs.

c. Shorting of the matrices to an external voltage has no effect since the matrix is ungrounded. The testing process will indicate accidental application of potential to the matrix. Equipment is provided to detect grounds on the matrices.

d. The logic matrices will each be supplied by two power sources. Loss of a single power source has no effect on operation. Loss of power to a logic matrix initiates a trip condition.

e. Failure of a logic trip relay set to actuate has no effect since there are six sets in series in the trip action and any one set initiating trip action will cause the action to be completed.

f. The failure of one trip breaker control relay in a trip breaker circuit has no effect since there are two trip breakers in series, either of which will provide the necessary action.

g. Single grounds in the trip breaker control relay circuits have no effect since the circuit is ungrounded. Ground detectors on each 125 VDC bus also indicate an accidental ground.

h. The AC circuit supplying power to the trip breaker control relay coil is fed from an isolation transformer. The circuit has a local ground detection system. Each of the four trip paths are fed from a separate 120 VAC vital instrument bus.

i. The CEDM power supply circuits operate ungrounded so that single grounds have no effect. A ground detection system is provided. The CEDMs are supplied in two groups by separate pairs of power supplies to further reduce the possibility of a CEA being improperly held. The CEDM load requirements are such that the application of any other local available voltage would not prevent CEA release.

7.2.6 SYSTEM RELIABILITY AND AVAILABILITY

7.2.6.1 Power Supply

The power for the protective system is supplied from four separate and independent vital 120 volt AC buses. Each vital bus is supplied from one of the two station battery systems through separate inverters (as shown in Figure 8.5–1). During normal operation, the battery chargers maintain a floating charge on each battery while at the same time, supply power to the vital inverters. Upon loss of auxiliary AC power, the batteries provide the power for inverter operation. In the event of loss of vital bus, the protective channel associated with the bus goes into a trip condition. Each vital bus also has automatic rapid transfer to a regulated instrument ac supply used as a backup.

The distribution circuits from the vital buses are provided with fuses and circuit breaker protection to assure that individual circuit faults are isolated close to the fault.

7.2.6.2 Environment Capability

RPS components were specified for environmental conditions existing in the area of the plant in which the components are installed.

Radiation design criteria for RPS components located within normally radioactive areas are specified at a gamma level of 1R/hour for 10 years, except for the main coolant resistance temperature detectors (RTDs), which are specified at 10R/hour for 40 years. Protective system equipment not located in normally radioactive areas or located in areas of very low activity has been specified accordingly. Periodic tests and calibration will assure detection of gradual equipment deterioration and will assure capability of the system to operate as required by the original design bases.

The design criteria for all electrical cable are that the cable shall not fail when subjected to any accident conditions after the long-term normal operating conditions. Cable details are specified in Chapter 8.

7.2.6.3 Seismic Capability

The specification for each RPS component or assembly includes the seismic requirements for that equipment.

These components are qualified by either of the two following methods.

In most cases, the supplier is required to qualify his equipment by calculation or testing, or a combination of both. This qualification is formally documented and submitted for approval prior to official acceptance of the equipment by Quality Control.

In other cases, tests or calculations are performed by independent consultants or laboratories who submit a formal report. Acceptance of the equipment from the supplier is contingent upon the proof of suitability as established by the results of those tests or calculations.

The choice of the analytical or experimental qualification procedure is determined by the size, shape and structural or functional complexity of the equipment in accordance with the criteria outlined in IEEE 344 “Guide for Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations.” Racks, panels or other supporting structures are generally qualified by analysis, while bistable trip units and other modules are generally qualified through testing. Tests and calculations are performed following the guidelines of IEEE 344, 1971.

Type testing is the preferred qualification method in accordance with IEEE-323, 1971, “General Guide for Qualifying Class I Electric Equipment for Nuclear Power Generating Station.” A report on the qualification of the RPS panels, racks, and equipment including the results of type tests, was submitted by CE in December of 1972 in the form of a topical report, CENPD-61, “Seismic Qualification of Category I Electric Equipment for Nuclear Steam Supply Systems.”

7.2.6.4 Physical Separation

The locations of the sensors and the points at which the sensing lines are connected to the process loop have been selected to provide physical separation of the channels, thereby precluding a situation in which a single event could remove or negate a protective function. Process transmitters located inside the containment and required for short term operation following a LOCA are rated for the intended service in the LOCA environment. The routing of cables from these transmitters is arranged so that the cables are separated from each other and from power cabling to minimize the likelihood of common event failures. This includes separation at the containment penetration areas. In the control room, the four nuclear instrumentation and protective system trip channels are located in individual compartments. Mechanical and thermal barriers between these compartments minimize the possibility of common event failure. Outputs from the components in this area to the control boards are buffered so that shorting, grounding, or the application of the highest available local voltages do not cause channel malfunction.

7.2.6.5 Bistable Trip Unit Drift

The bistable trip units have been specified and designed for a maximum drift of 32 mv.

TABLE 7.2-1 REACTOR TRIP AND PRETRIP SET POINTS

No.	Reactor Trip	Pretrip Alarm Set Point	Trip Set Point
1.	Reactor Coolant Pump Underspeed * (not credited)	N.A.	≥ 830 rpm
2.	High Power Level	2% below trip setpoint	≤ 10% above measured power Q
3.	Low Reactor Coolant Flow * 4-Pump Operation, %	N.A.	≥ 91.7
4.	Low Steam Generator Water Level, % (Auctioneered low of SG #1, SG #2)	54	≥ 48.5
5.	Low Steam Generator Pressure **, psia (Auctioneered low of SG #1, SG #2)	780	≥ 691
6.	High Pressurizer Pressure, psia	2350	≤ 2397
7.	Thermal Margin/Low-Pressure *	75 psia above trip set point	Variable trip set point with minimum of 1865 psia
8.	Loss of Turbine *** (Low Hydraulic Fluid Pressure) psig	N.A.	≥ 500
9.	High Containment Pressure, psig	N.A.	≤ 4.42
10.	Manual Trip (Push Buttons)	N.A.	N.A.
11.	Local Power Density	< FCMLHR	FCMLHR

* Manual inhibit permitted below 10^{-4} percent power: automatically removed prior to increasing reactor power to 10^{-4} percent power.

** Manual inhibit permitted below 800 psia: automatically removed above 800 psia.

*** Inhibited below 15% power.

TABLE 7.2.1.1–I PARAMETERS AFFECTING FUEL DESIGN LIMIT

I. DNBR and Void Fraction

1) Core Power
2) Core Inlet Temperature
3) Axial Power Distribution
4) Radial Peaking Factor
5) Primary System Pressure
6) Core Mass Flow Rate
7) Azimuthal Tilt Magnitude

II. Fuel Temperature (or Equivalent kw/ft)

1) Core Power
2) Axial Power Distribution
3) Radial Peaking Factor
4) Azimuthal Tilt Magnitude

TABLE 7.2.1.1–II PERTINENT NSSS PARAMETERS AND MONITORED VARIABLES

NSSS Parameter	Monitored Variable
Core Power	Neutron Flux Power/ΔT Power
Core Inlet Temperature	Cold Leg Temperature
Primary System Pressure	Pressurizer Pressure
Axial Power Distribution	Axial Flux Offset
Radial Peaking Factor	(Inferred from Neutron Flux Power/DT Power and Technical Specifications PDILS)
Core Mass Flow Rate	Steam Generator Differential Pressures
Azimuthal Tilt Magnitude	Value assumed consistent with Technical Specifications Limit

TABLE 7.2.1.1–III TRIP FUNCTIONS

Acceptable Fuel Design Limit	Calvert Cliffs	Millstone Unit 2
DNBR & Void Fraction	Thermal Margin Trip Axial Flux Offset Trip Low Flow Trip	Thermal Margin Trip High Power Trip Low Flow Trip
Kw/Ft (Fuel Temperature)	Axial Flux Offset Trip	Local Power Density Trip High Power Trip

TABLE 7.2.1.1–IV TRIP VARIABLE MONITORED

Trip	Calvert Cliffs	Millstone Unit 2
Thermal Margin	Pressurizer Pressure Neutron Flux Power/ΔT Power	Pressurizer Pressure Neutron Flux Power/ΔT Power
Core Inlet Temperature	Core Inlet Temperature	Core Inlet Temperature
Axial Flux Offset	Neutron Flux Power Axial Flux Offset	
Local Power Density		Neutron Flux Power/ΔT Power Axial Flux Offset
High Power	Neutron Flux Power	Neutron Flux Power/ΔT Power
Low Flow	Steam Generator Differential Pressure	Steam Generator Differential Pressure

FIGURE 7.2–1 REACTOR PROTECTIVE SYSTEM–BLOCK DIAGRAM

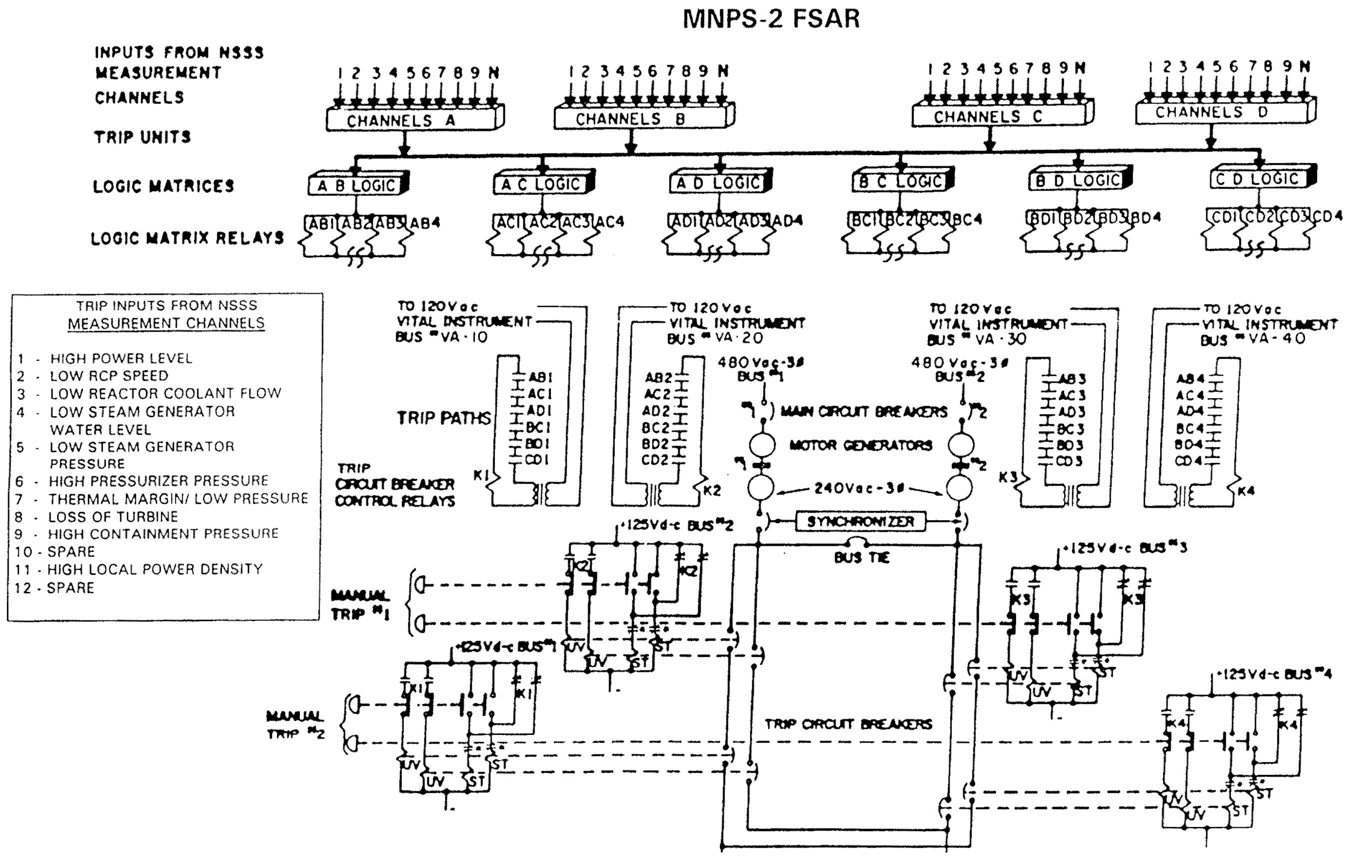


FIGURE 7.2–2 REACTOR PROTECTIVE SYSTEM FUNCTIONAL DIAGRAM

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

MNPS-2 FSAR

FIGURE 7.2–3 NUCLEAR INSTRUMENTATION SYSTEM FUNCTIONAL DIAGRAM

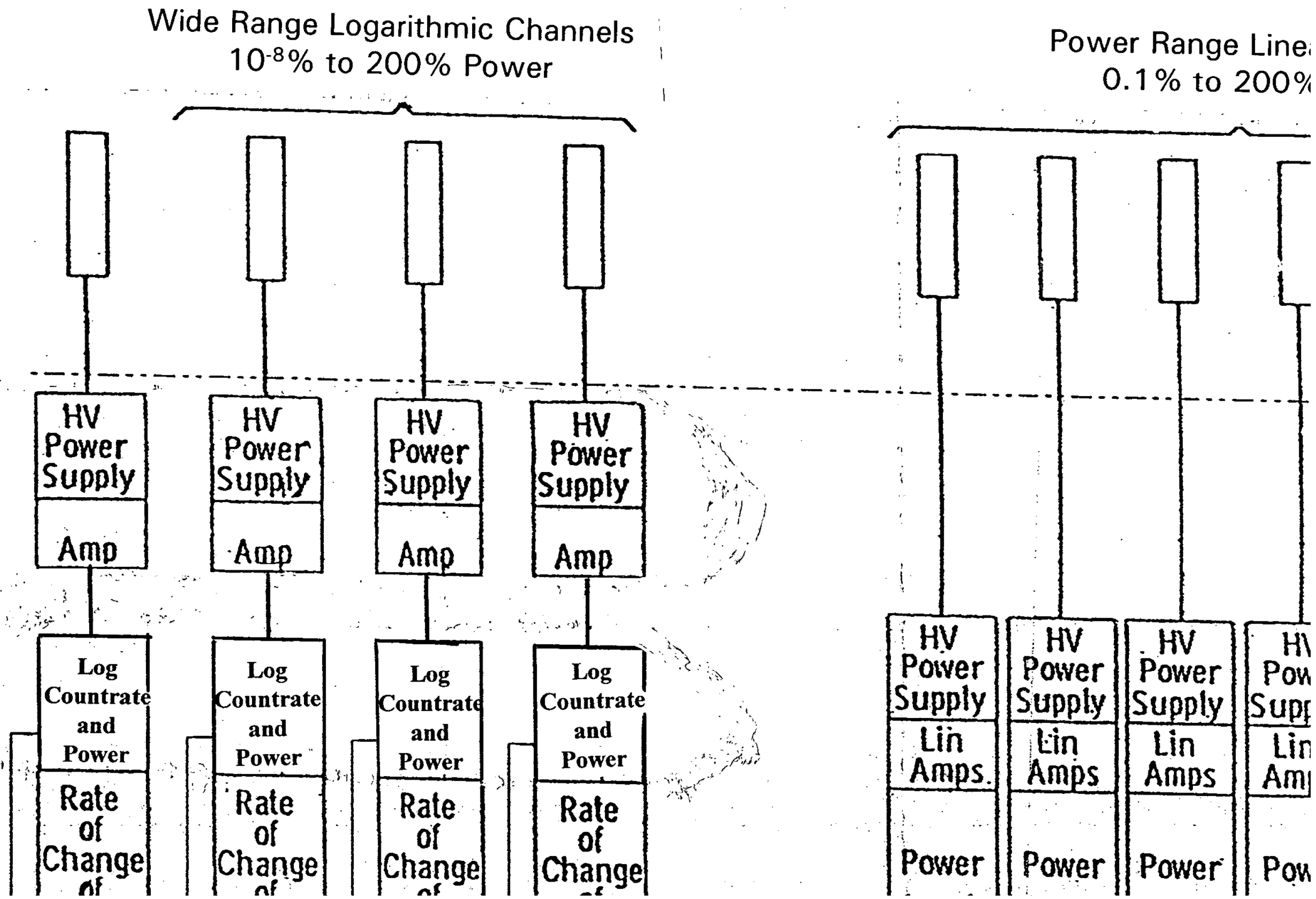


FIGURE 7.2–4 LOW FLOW PROTECTIVE SYSTEM FUNCTIONAL DIAGRAM

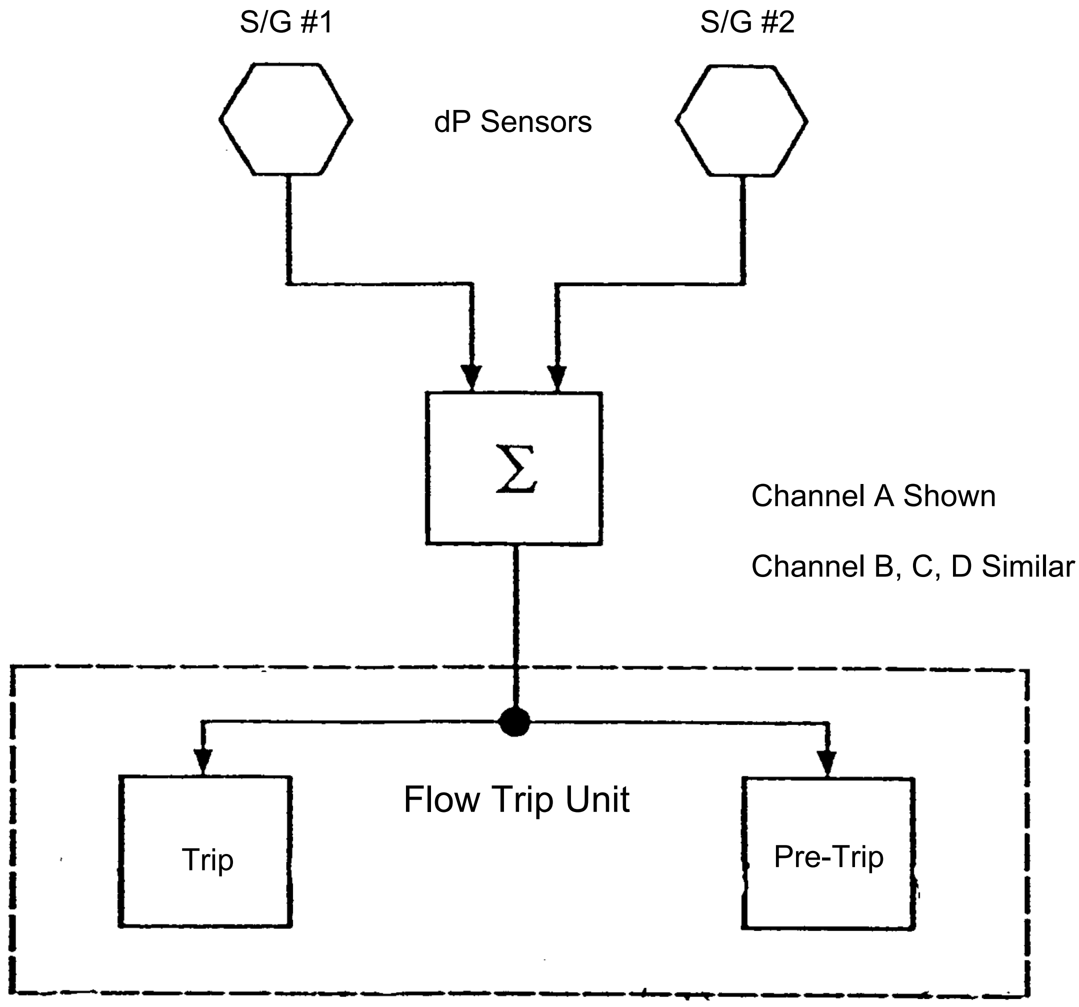


FIGURE 7.2–5 LOW STEAM GENERATOR PRESSURE REACTOR TRIP BYPASS FUNCTIONAL DIAGRAM

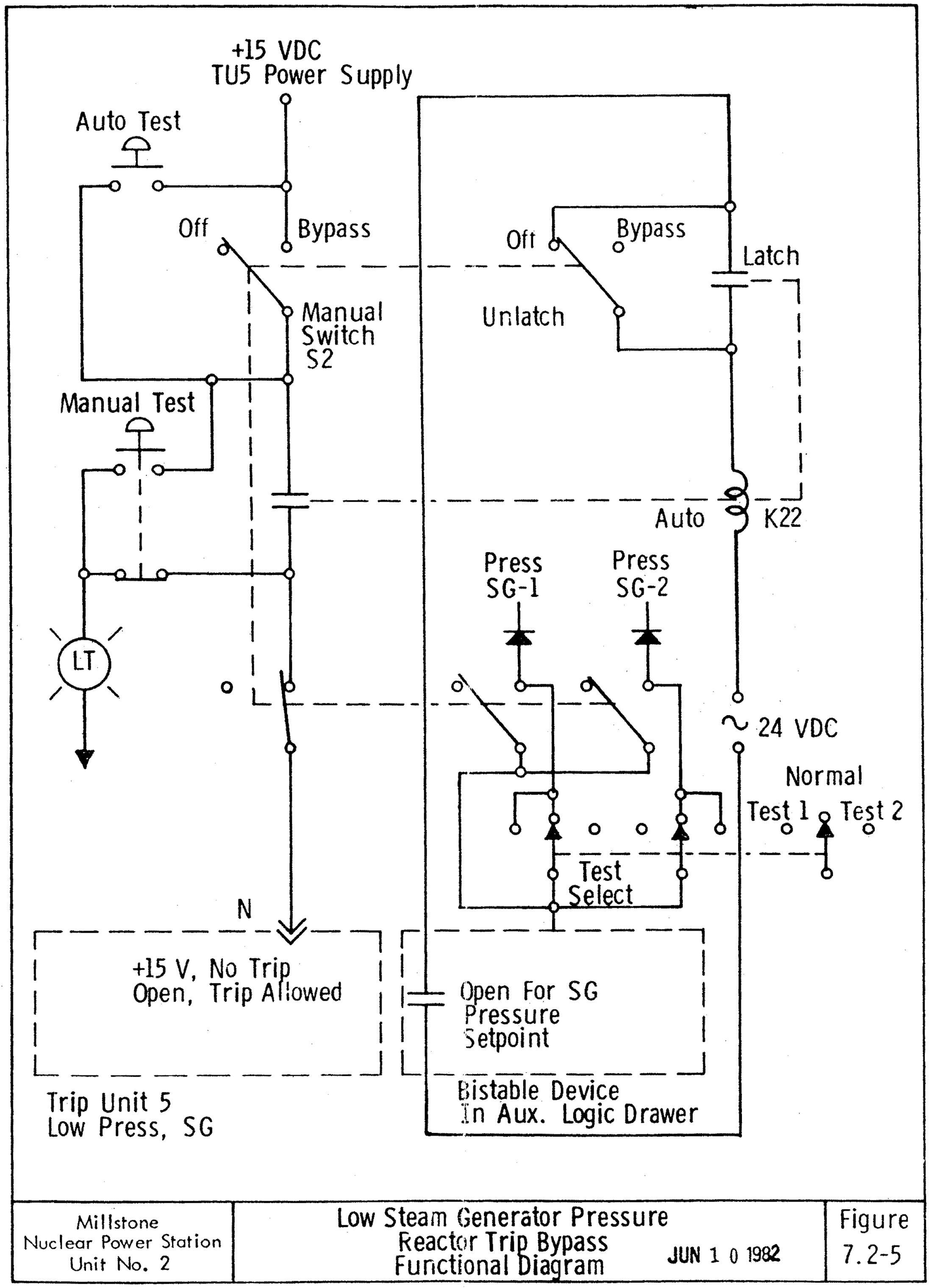
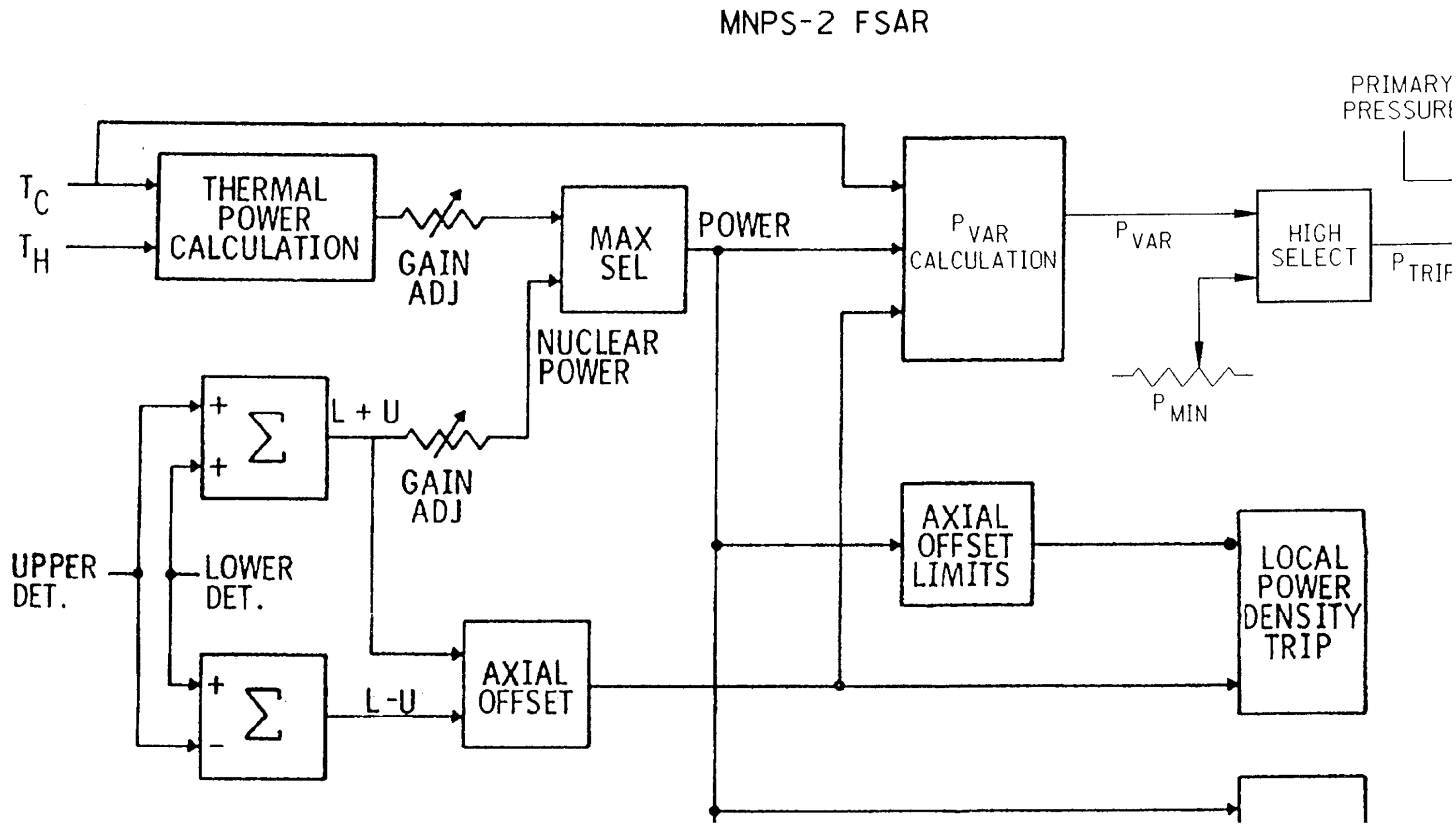
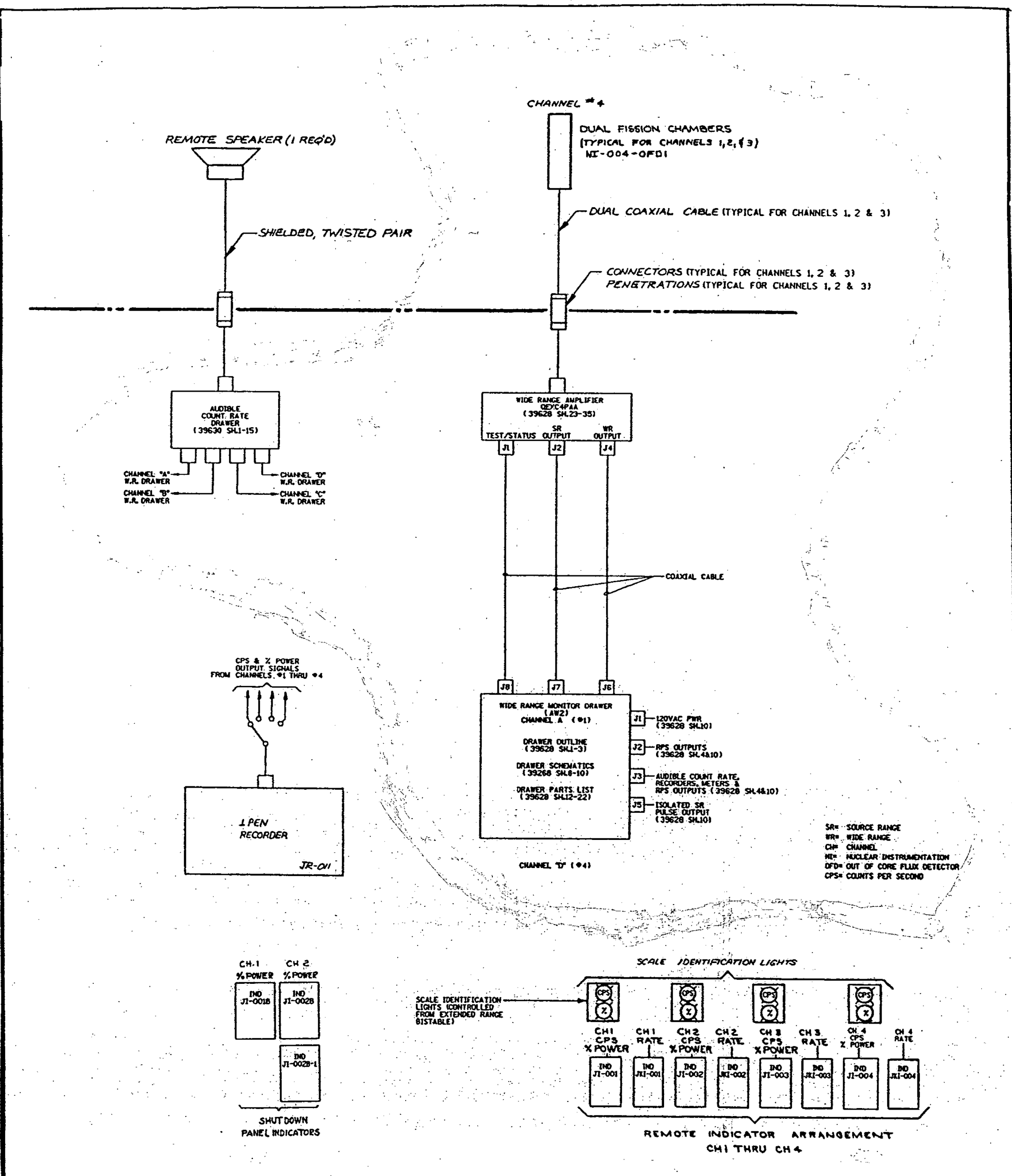


FIGURE 7.2–6 BLOCK DIAGRAM CORE PROTECTION TRIPS





Millstone Nuclear Power Station Unit No. 2	NEUTRON FLUX MONITORING SYSTEM LOGARITHMIC RANGE CHANNELS	Figure 7.2-7

October 2001 (00-50)

FIGURE 7.2–8 NEUTRON FLUX MONITORING SYSTEM POWER RANGE CHANNELS

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.2–9 NOT USED

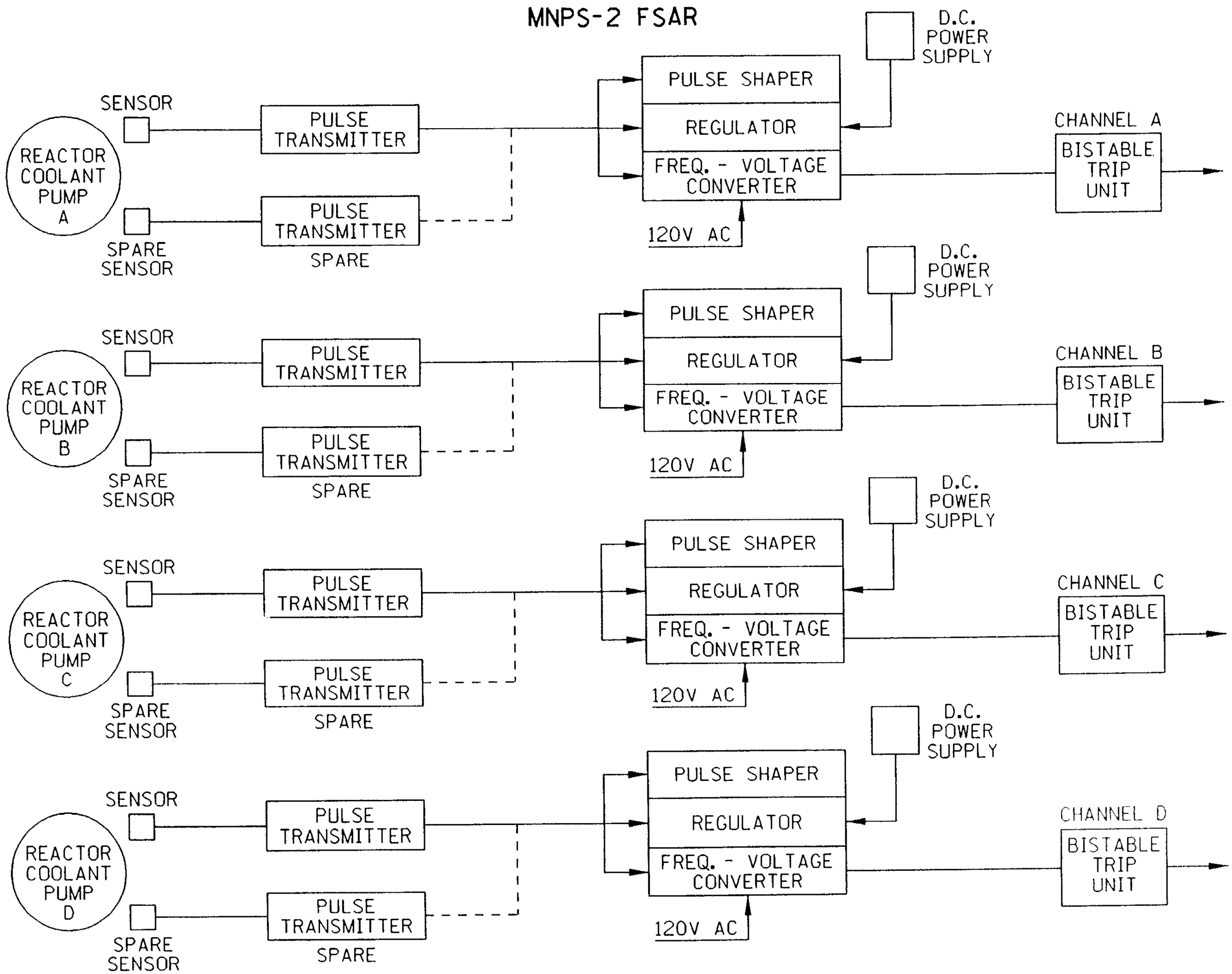


FIGURE 7.2-10
REACTOR COOLANT PUMP UNDERSPEED BLOCK DIAGRAM

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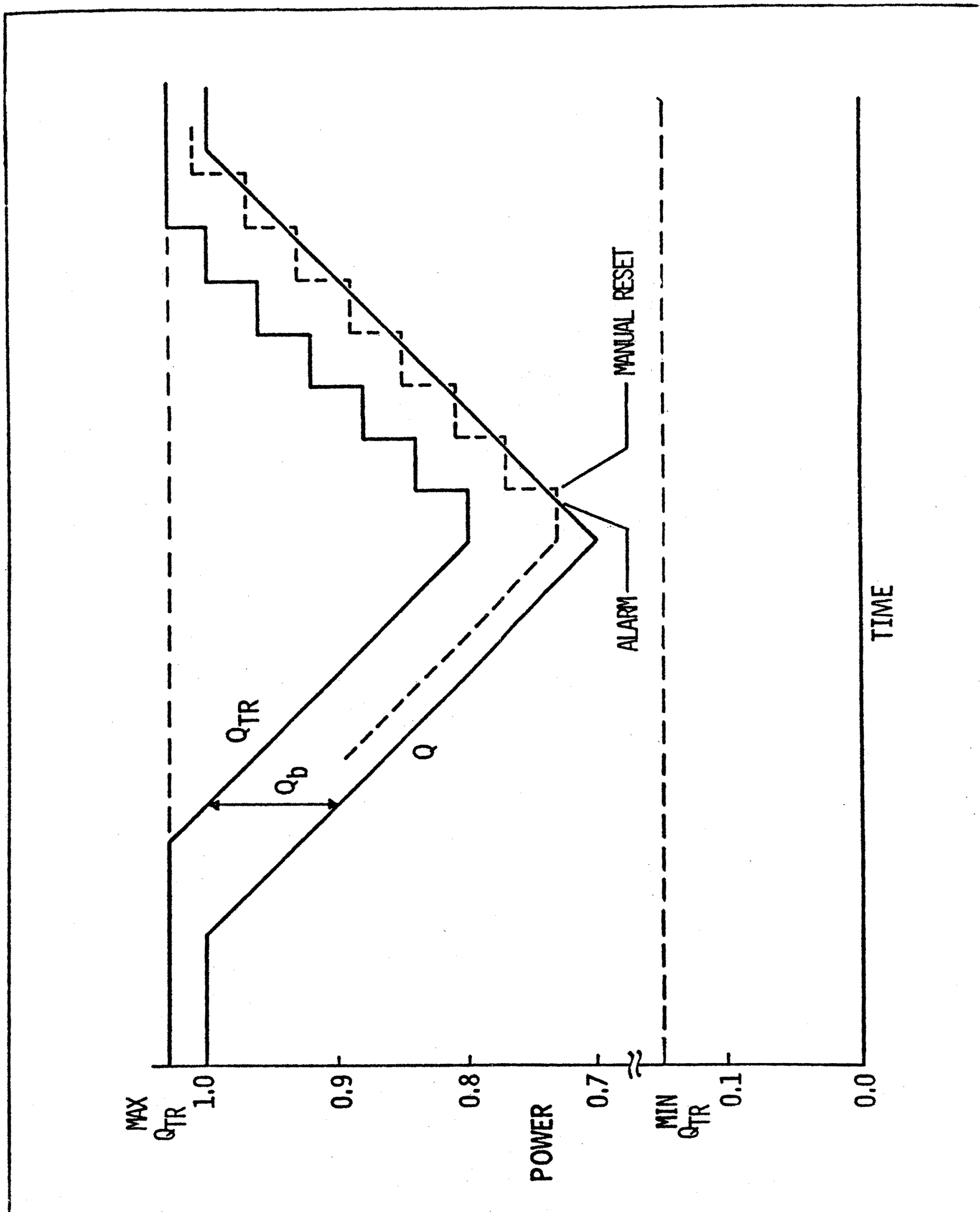
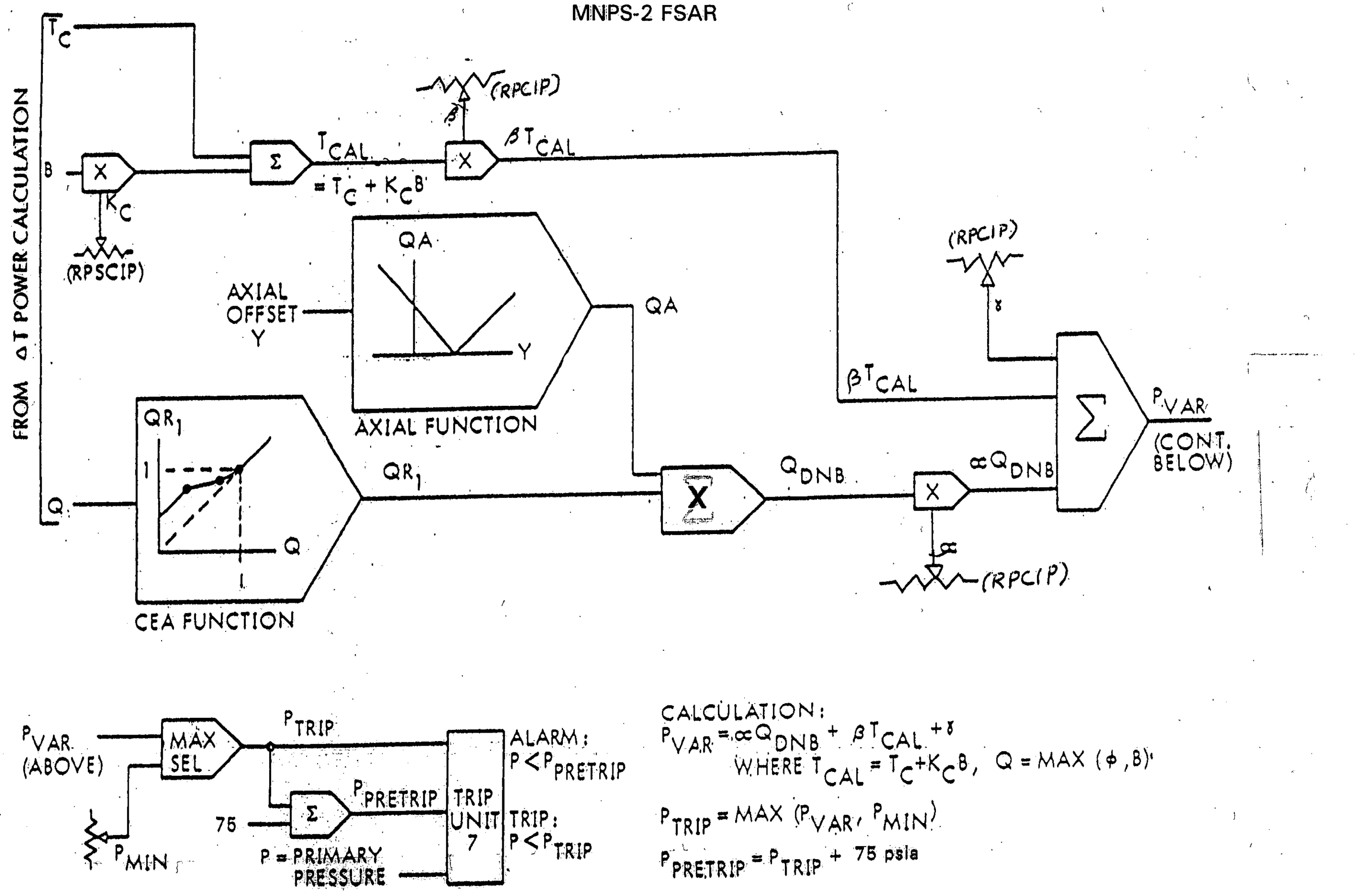


Millstone Nuclear Power Station Unit No. 2	Variable High Power Trip Operation	Figure 7.2-11
	JUN 1 0 1982	11-6



CALCULATION:

$P_{VAR} = \propto Q_{DNB} + \beta T_{CAL} + \gamma$

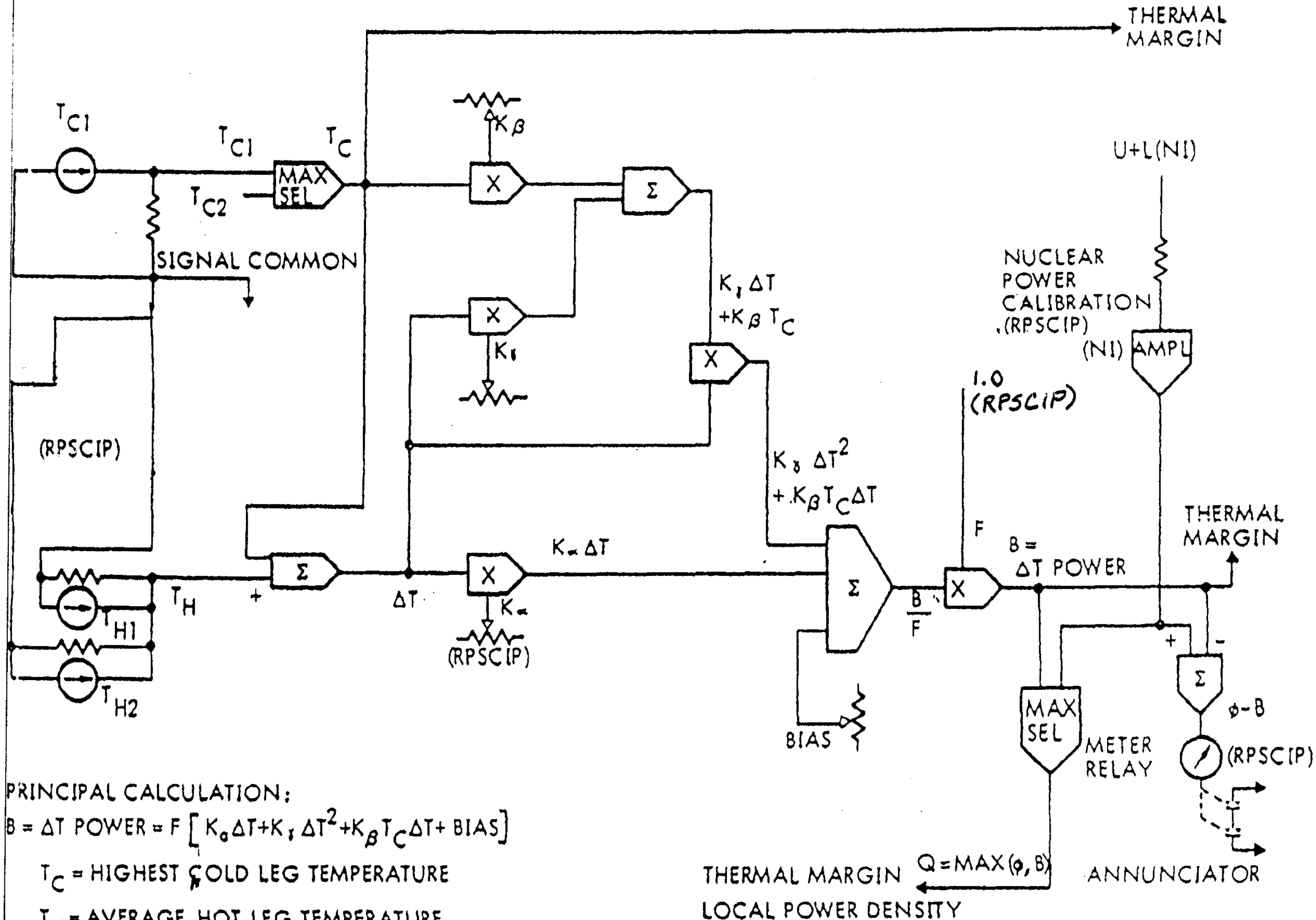
WHERE $T_{CAL} = T_C + K_C B$, $Q = MAX (\phi, B)$

$P_{TRIP} = MAX (P_{VAR}, P_{MIN})$

$P_{PRETRIP} = P_{TRIP} + 75$ psia

FIGURE 7.2-12

THERMAL MARGIN TRIP



PRINCIPAL CALCULATION:

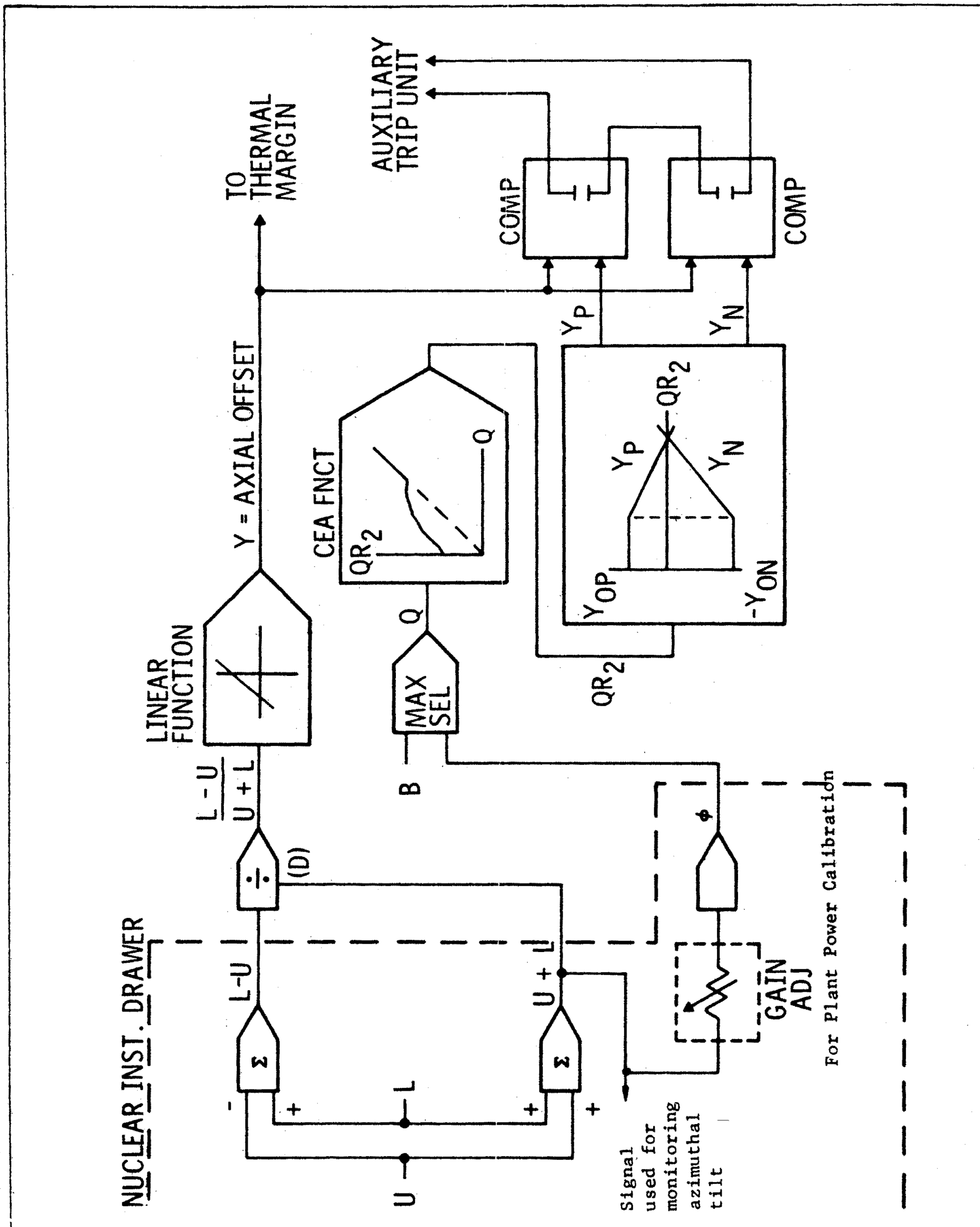
$B = \Delta T \text{ POWER} = F\left[K_\alpha \Delta T + K_\gamma \Delta T^2 + K_\beta T_C \Delta T + \text{BIAS}\right]$

T_C = HIGHEST COLD LEG TEMPERATURE

T_H = AVERAGE HOT LEG TEMPERATURE

$\Delta T = T_H - T_C$

FIGURE 7.2-13
ΔT POWER CALCULATION



Millstone
Nuclear Power Station
Unit No. 2

Local Power Density Trip

JUN 1 0 1982

Figure 7.2-14

11-4

7.3 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM

7.3.1 DESIGN BASES

The plant protection system consists of a sense and command design features involved in generating the signals used for reactor trip and engineered safety features actuations. The plant protection system consists of the Reactor Protection System (RPS) and the Engineered Safety Features Actuation System (ESFAS). The ESFAS is that portion of the plant protection system used to automatically and manually initiate the operation of the engineered safety features (ESF) systems and essential auxiliary supporting (EAS).

At Millstone Unit 2, the ESFAS is comprised of: (1) the Engineered Safeguards Actuation System (ESAS) and (2) the Auxiliary Feedwater Automatic Initiation System (AFAIS).

7.3.1.1 Functional Requirements

The Engineered Safety Features Actuation System (ESFAS) is designed to detect accident conditions and initiate the operation of systems and components important to safety. The ESFAS is designed for high functional reliability and in-service testability commensurate with the safety functions it performs.

7.3.1.2 Design Criteria

7.3.1.2.1 Single Failure, Redundancy and Independence

The ESAS and AFAIS are designed with sufficient redundancy and independence to assure that (1) no single failure results in loss of protective function and (2) removal from service of any component of a channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated.

The ESFAS is designed and constructed in accordance with Institute of Electrical and Electronics Engineers (IEEE) Standard 279, "Criteria for Nuclear Generating Station Protection Systems," 1971. Emphasis has been placed on the single-failure criteria, which requires that no single failure shall prevent the system from initiating the safety feature functions when true initiating conditions exist.

The system functions are implemented by means of redundant sensors, instrument loops, logic and actuation devices.

Independence is provided between redundant elements to preclude any interactions between channels during maintenance or in the event of channel malfunction.

Redundant elements are electrically isolated from each other such that events affecting one element are not reflected in any other redundant element.

The electrical isolation provided is capable of withstanding the application of the maximum credible voltage or current transient without degradation below an acceptable level.

Physical separation is in accordance with Section 8.7.

7.3.1.2.2 Modular Design

Interchangeability without preselection is provided for all corresponding Engineered Safeguards Actuation System (ESAS) modules or components with the exception of keys. All ESAS items removable from the equipment such as assemblies, subassemblies, electrical parts, modules and hardware are replaceable physically and electrically with corresponding items. The replacement of parts, when accomplished in a manner prescribed by the ESAS manufacturer, will not cause the equipment to depart from the original specified performance.

The Auxiliary Feedwater Automatic Initiation System (ASAIS) utilizes modular Foxboro equipment for the signal conditioning, bistable, and coincidence logic functions.

7.3.1.2.3 Module Withdrawal and Bypass

Withdrawal of any ESAS subunit or module in any of the sensor channels results in a trip of the affected channel and annunciation of the tripped channel. Means are provided by noninterchangeable key-operated switches to manually bypass any single analog channel bistable trip module of a four-channel parameter. Annunciation of ESAS channel bypass is provided.

The removal of an AFAIS module may not result in a trip as AFAIS is an energize to actuate design. Each AFAIS actuation channel contains provisions to bypass the sensor channel input to that actuation channel’s coincidence logic by means of a key-operated switch. Annunciation of channel bypass is provided.

With the exception of auxiliary feedwater automatic initiation control, loss of power to a sensor channel of the ESFAS will result in a trip of the sensor channel affected. A trip of the ESAS channel is annunciated. A trip of an AFAIS sensor channel is indicated by status lamps on C517 and C518.

The system is designed such that routine servicing and preventive maintenance can be performed without affecting operation or availability. Normal maintenance does not reduce performance below the minimum safety level.

7.3.1.2.4 Environment

The ESAS cabinets are installed in the control room which is normally air conditioned and suitable for computers or computer-grade equipment. However, the equipment installed in the ESAS cabinets is designed to function continuously at temperatures ranging from 40°F to 140°F and at a relative humidity of 95 percent. Additionally, the following control room equipment in the SPEC 200 cabinets is environmentally qualified for a temperature range of 40°F to 120°F and a relative humidity of 10 percent to 95 percent (with an 86°F maximum wet bulb): (a) plant process equipment that provides analog inputs to the ESAS and (b) plant process equipment that provides analog inputs and bistable and coincidence logic functions in the AFAIS.

7.3.1.2.5 Seismic Requirements

The ESFAS has been designed to function before, during and after a design basis earthquake (DBE). The ESFAS has been demonstrated to be seismically qualified by type tests and analysis. The original ESAS cabinet was evaluated by Consolidated Controls Corporation as documented in Engineering Reports No. 832 and NBR 863. The type tests and analysis on the remaining ESFAS equipment conform to the guidance of the IEEE standards governing seismic qualification described in Section 7.3.1.2.6 below.

7.3.1.2.6 Codes and Standards

The ESFAS and component parts conform to the requirements of the following IEEE standards and Nuclear Regulatory Commission (NRC) Regulatory Guide.

The auxiliary feedwater (AF) automatic initiation system and component parts and in containment mounted sensors conform to the 1974 edition of IEEE 323 and the 1975 edition of IEEE 344 and the standards listed below. The containment mounted sensors conform to the 1975 edition of IEEE 344 and the standards listed below.

a. IEEE 279	1971 Criteria for Protection Systems for Nuclear Power Generating Stations
b. IEEE 308	1970 IEEE Standard Criteria for Class IE Electrical Systems for Nuclear Power Generating Stations
c. IEEE 323	1971 General Guide for Qualifying Class I Electrical Equipment for Nuclear Power Generating Stations
d. IEEE 344	1971Seismic Qualification of Class I Electric Equipment for Nuclear Power Generating Stations
e. IEEE 336	1971 Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations
f. IEEE 338	1971 Periodic Testing of Nuclear Power Generating Station Protection Systems
g. Regulatory Guide 1.22	Periodic Testing of Protection System Actuation Functions
h. IE Bulletin No. 80-06	Engineered Safety Features Reset Controls

7.3.1.2.7 Testability

The ESAS system is designed to permit testing up to and including the actuation module during power operation.

The ESAS system has an automatic testing system. This system is described in Section 7.3.4.

7.3.1.2.8 Response Time

The overall response time of the ESAS system shall be measured from the sensor input to the trip device output terminals and will not exceed 500 milliseconds.

7.3.2 SYSTEM DESCRIPTION

7.3.2.1 General

The ESFAS detects accident conditions and initiates the safety features systems which are designed to localize, control, mitigate, and terminate such incidents. The ESAS was designed and constructed by Consolidated Controls Corp. in Bethel, Connecticut. The AFAIS equipment was supplied by the Foxboro Company. The engineered safety features actuation system is divided into four sensor channels (A, B, C, D), two actuation channels and actuation logic channels.

Each of the two ESAS actuation and logic channels includes an automatic load sequencer for sequentially loading the emergency diesel generators (DG) following a loss of normal power. A separate third channel is incorporated into the ESAS to actuate equipment that can be energized by either of the two electrical divisions.

All process variables are transmitted as analog signals. Loss of voltage on the 4.16 kV emergency bus is detected through the potential transformers by the ESFAS undervoltage modules.

Four essential or vital power sources are provided for the ESFAS. Two emergency DGs are provided to supply power to the actuated equipment of the protective systems in case of loss of offsite power.

If offsite power is available, the engineered safety features (ESF) equipment starts directly. If offsite power is not available, load shedding and sequencing are required for sequential loading of the DGs.

As a result of IE Bulletin Number 80-06, “Engineered Safety Feature Reset Controls,” the ESF Reset Controls have been modified so that ESF actuated equipment remains in its emergency, “actuated,” mode following reset of an ESFAS until deliberate operator action is taken.

7.3.2.2 Sensor Channels

The instrument channels monitor redundant and independent process variables and initiate a sensor channel trip when the variable or condition deviates beyond a set limit. Each of the actuation channels receives a signal from the following variables: (See Figures 7.3–1, 2, 7.3–3, 7.3–4.)

a. Pressurizer Pressure

Low pressurizer pressure during power operation is indicative of a loss-of-coolant accident (LOCA). It is measured with four redundant pressure transmitters. A pressure loss below 1714 psia on any two of four bistables in the ESF system will initiate a simultaneous safety injection actuation signal (SIAS), containment isolation activation signal (CIAS), and enclosure building filtration actuation signal (EBFAS). These signals will isolate all unnecessary lines at the containment penetration, initiate safety injection system (SIS) operation, and start the enclosure building filtration system (EBFS). The four pressure transmitters are also used for input signals to the RPS.

b. Containment Pressure

High containment pressure during power operation is indicative of a LOCA or main steam line break. It is measured with four pressure transmitters. An increase in containment pressure to 4.42 psig on any two of four bistables in the ESF system will initiate a simultaneous SIAS, CIAS, EBFAS, and MSIAS. Measurement of containment high pressure is a diverse means of sensing a loss of coolant condition. The transmitters are reverse acting type (increasing input gives a decreasing output signal) to permit fail safe operation.

With regard to power requirements, loss of instrument power results in a tripped bistable which would require only one more of the three remaining bistables to trip in order to get an actuation.

A failure mode analysis for loss of onsite instrument power is shown in Table 7.3.2.2-1.

A further increase in containment pressure to 9.48 psig will initiate a containment spray actuation signal (CSAS) which will start two containment spray pumps and open their respective discharge motor operated valves (MOV) to start spraying.

c. Containment Gaseous and Particulate Radiation

Two gaseous and two particulate monitors are used to detect the release of radioactive fission products to the containment atmosphere. If these monitors were to fail or were unavailable, grab samples are taken or portable continuous air monitoring equipment is used. The ESFAS logic will initiate containment purge isolation should any one of the four detectors exceed its set point. However, the fuel handling area radiation system automatic functions are not credited in the fuel handling accident analyses.

d. Steam Generator Pressure

Each steam generator pressure is sensed by four pressure transmitters. A drop in pressure to 572 psia on any two out of the four sensor channels on either steam generator will actuate a main steam isolation actuation signal (MSIAS) which automatically closes both MSIVs. The four pressure transmitters are also used for input signals to the RPS.

e. Fuel Handling Area Radiation

Fuel handling area high radiation is sensed by four redundant area radiation monitors located on walls adjacent to the spent fuel pool. Upon detection of high radiation due to a fuel handling accident from any two of the four monitors, an auxiliary exhaust actuation signal (AEAS) is generated which stops the spent fuel pool area outside air supply fan and diverts the exhaust to the EBFS.

f. Refueling Water Storage Tank Level

The safety injection pumps initially take suction from the refueling water storage tank (RWST). After the tank level has decreased to 46 inches as measured by two of four level sensing channels, a sump recirculation actuation signal (SRAS) transfers the safety injection pump suction to the containment sump for long-term recirculation.

g. Emergency Bus Undervoltage

The undervoltage protection provided for the emergency buses consists of two independent schemes, one for each 4,160 volt emergency bus, 24C(A3) and 24D(A4). Each scheme employs redundant design features and consists of two levels of protection. The Level 1 undervoltage protection (loss of voltage) is designed to detect a loss of voltage on the emergency buses. The Level 1 undervoltage logic isolates the emergency buss from all sources, initiates automatic loading shedding, and provides a start signal for the diesel generator associated with the emergency bus. The Level 2 undervoltage protection (degraded voltage) is designed to protect the safety-related equipment from operation under sustained low (degraded) voltage conditions. The Level 2 undervoltage logic provides a trip signal to the reserve station service transformer (RSST) supply breaker of the associated bus. Figure 7.3–1 illustrates the emergency bus undervoltage protection logic (see also FSAR Section 8.2).

Potential transformers on 4,160-volt emergency buses 24C(A3) and 24D(A4) provide voltage inputs to the ESAS undervoltage sensor logic. In the ESAS sensor cabinets, these analog inputs are compared to the trip setpoints for the Level 1 and Level 2 undervoltage protection. The Level 1 and Level 2 trip outputs associated with bus 24C(A3) are connected to the ESAS Facility 1 actuation logic cabinet.

The Level 1 and Level 2 trip outputs associated with bus 24D(A4) are connected to the ESAS Facility 2 actuation logic cabinet.

The trip setpoints and associated time delays for the Level 1 and Level 2 undervoltage protection are defined in the Technical Specifications and are as follows:

Level 1	2,912 volts with a time delay of 2 seconds
Level 2	3,700 volts with a time delay of 8 seconds

The time delay for the Level 1 undervoltage setpoint was chosen to:

1. allow sufficient time for protective relaying schemes to detect and clear system and station faults that cause bus voltages to drop below the Level 1 undervoltage setpoint and

2. ensure that the Level 1 undervoltage actuation functions occur in a time frame that is consistent with the MP2 safety analysis for design basis accidents.

The time delay for the Level 2 undervoltage setpoint was chosen to be:

1. sufficiently long as to prevent unnecessary trips of the circuit breaker connecting the emergency bus with the RSST source during voltage transient conditions (e.g., the starting of large motors) and

2. sufficiently short as to (a) prevent damage to safety-related AC loads and (b) ensure that the Level 1 undervoltage actuation functions occur in a time frame that is consistent with the MP2 safety analysis for design basis accidents.

h. Steam-Generator Level

Level in each of the two steam generators is sensed by four common and redundant differential pressure transmitters. (Each steam generator level transmitter also provides an isolated signal to the RPS for input to the low steam generator water level reactor trip as described in Section 7.2.) AF is initiated automatically on low steam generator level by a 2 out of 4 logic matrix from the four independent bistable channels. The level input signals from both steam generators are auctioneered at the channel level, so that each auto AF bistable channel receives the lower of the input signals from two level transmitters (one from each steam generator). If any two of the four (total), measurement channels detect a steam generator level decreasing below 26.8% (nominal), an automatic AF initiation signal is processed and the following occurs:

- each steam generator blowdown isolation valve receives a close signal
- a three (3) minute, twenty-five (25) second (nominal) time delay begins timing out
- and a control room alarm annunciates the automatic AF initiation signal.

Following the time delay, automatic AF start indication is provided on C05 and C21 and two auxililary relays per facility are energized. The auxiliary relays send a start signal to the respective facility motor driven AF pump and open signals to both AF regulating valves.

7.3.2.3 Actuation Channels

The two redundant and independent actuation channels monitor the sensor channel trips and by means of coincidence logic determine whether a protective action is required. The following actuation channels are initiated by the ESFASs: (See Figure 7.3–1, 7.3–3 and 7.3–4.)

a. SIAS under the condition: low-low pressurizer pressure 2-out-of-4 (or 2-out-of-3) or high containment pressure 2-out-of-4 (or 2-out-of-3) or manual SIAS (main control board pushbutton).

b. CSAS under the following conditions: SIAS (manual or automatic) and high-high containment pressure 2-out-of-4 (or 2-out-of-3) or manual CSAS (main control board pushbutton).

c. CIAS for automatic containment isolation valves under the conditions: low-low pressurizer pressure 2-out-of-4 (or 2-out-of-3) or high containment pressure 2-out-of-4 (or 2-out-of-3) or manual SIAS or CIAS (main control board pushbuttons).

d. EBFAS under the following conditions: low-low pressurizer pressure 2/4 (or 2-out-of-3) or high containment pressure 2-out-of-4 (or 2-out-of-3) or manual EBFAS or SIAS (main control board pushbuttons).

e. Containment purge valves close signal (CPVIS) under the following conditions: high containment radiation 1-out-of-4.

f. Containment hydrogen purge valves close signal under the following conditions: CIAS. In addition, two high range gamma-sensitive ion chambers are used to detect post-accident radiation levels in the containment. The ESAS CIAS logic in conjunction with the logic from the post-accident radiation monitors is combined together in an “OR” logic to initiate closure of the containment hydrogen purge valves. The containment hydrogen purge valve closure is generated either by one-out-of-two post-accident radiation monitors exceeding its setpoint, or by an ESAS CIAS signal.

g. MSIAS upon low steam generator pressure 2-out-of-4 (or 2-out-of-3) or high containment pressure 2-out-of-4 (or 2/3) or manual MSIAS (main control board push buttons).

h. SRAS for the following conditions: low-low RWST level 2-out-of-4 (or 2-out-of-3) or manual SRAS (main control board pushbutton).

i. AEAS for the following conditions: high radiation in the fuel handling area 2-out-of-4(or 2-out-of-3) or manual AEAS (main control board pushbutton or local pushbutton) in the absence of EBFAS.

j. Emergency bus load shed for the following conditions: Level 1 emergency bus undervoltage (loss of voltage), 2-out-of-4 (or 2-out-of-3) channels. FSAR Section 8.2 discusses the automatic load shedding of the emergency buses.

k. Emergency DG start for the following conditions: Level 1 emergency bus undervoltage (loss of voltage), 2-out-of-4 (or 2-out-of-3) channels or on a SIAS. FSAR Section 8.3 discusses the automatic start of the emergency DGs.

l. Trip of the RSST supply breaker to the associated 4160 Volt emergency bus for the following conditions: Level 2 emergency bus undervoltage (degraded voltage), 2-out-of-4 (or 2-out-of-3 channels).

m. Auxiliary feedwater automatic initiation system (AFAIS) signal under the following conditions: Steam Generator water level low, ≤ 26.8%, 2-out-of-4 (or 2-out-of-3).

7.3.2.4 Actuation Interface

For each of the groups in each ESAS actuation channel, separate interfacing logic and coincident logic are provided such that any isolated actuation of one group will not produce actuation of any of the others. The ESAS actuation interface modules ensure that the signals provided by the actuation logic are distributed to the safety equipment. The ESAS actuation modules provide signals to the final ESAS relays for the control of safety equipment.

The ESAS actuation signals produced in the logic modules are separated and test groupings have been selected to permit equipment actuation without upsetting normal plant operation.

The AFAIS actuation is arranged in two redundant actuation channels. The actuation outputs are arranged such that actuation of either of the two redundant actuation channels will result in actuation of both auxiliary feedwater valves. Each of the two redundant channels will actuate its associated motor-driven auxiliary feedwater pump.

7.3.2.5 Channel Trips

When an actuation channel trips, due to unit conditions or manual initiation, its logic seals in its tripped state until it is manually reset.

7.3.2.6 Manual Initiation

Each actuation channel (1 or 2) of each of the following actuation signals can be initiated by a pushbutton located on the main control board: SIAS, CSAS, CIAS, EBFAS, SRAS, MSIAS, and AEAS. Each pushbutton initiates all actuation groups of the associated signal within its respective actuation channel.

The manual actuation switches located on the main control board are designated “HS” on Figures 7.3–1 and 7.3–4.

Each train (A or B) of the auxiliary feedwater system (AFWS) may be manually initiated utilizing normal system manual controls.

7.3.2.7 Manual Reset

Each actuation channel (1 or 2) of each of the actuation signals described in Section 7.3.2.3 has an independent momentary contact pushbutton located on the ESAS cabinets for manual reset of the actuation signals. The reset switch has no effect as long as a trip condition input to the logic matrix exists. Each pushbutton resets all actuation groups of the associated signal within actuation channels 1 or 2.

Manual reset of the AF automatic initiation circuitry is accomplished by positioning the respective train control switch to the reset (momentary) position. The reset switch has no effect as long as a trip condition input to the logic matrix exists.

7.3.2.8 Automatic Sequencer

If offsite power is available, the equipment actuated by ESFAS signals is started as soon as the actuation signal is developed by the coincidence logic. When offsite power is not available, the equipment is started by a sequencer. Emergency DGs are provided for supplying power to ESFs in case of loss of offsite power. The load shedding and sequential actuation system initially blocks, and then unblocks in programmed steps, the actuation channels for the equipment requiring power to perform its intended function. The undervoltage system contains four redundant sensors. The trip outputs are delayed and, by means of two-out-of-four coincidence logic, load shed and diesel start are initiated.

The diesel load sequencers (1/channel) provide 10 independent (5/sequencer), time-separated loading signals as detailed in Figures 7.3–2A through 7.3–2D, and Tables 8.3-2 and 8.3-3.

The sequencer output signals provide the required time sequencing of the safety features and the loss-of-power shutdown equipment according to Figures 7.3–2A through 7.3–2D, and Tables 8.3-2 and 8.3-3.

Means for testing of the sequencers are provided.

Indicators located in the appropriate cabinet will show the state of each sequencer time step.

7.3.2.9 Annunciation

The ESFAS provides contact outputs to indicate status in the control room. The outputs are isolated so that faults occurring external to the actuation system cabinets will in no way degrade the protective function of the system.

The contacts provided are summarized, as follows:

a. Bistables

   A contact for each of the 52 ESAS sensor channel trip bistables, to indicate bistable trip, for use on individual windows of the main control board annunciation system.

   A contact for each of the four ESAS sensor channel parameters to indicate bistable bypass and conversion from 2-out-of-4 to 2-out-of-3 logic.

   A contact from each of the redundant AFAIS actuation logic trains to annunciate a channel in bypass and that an actuation train is converted from a 2-out-of-4 to a 2-out-of-3 logic.

   AFAIS bistable status is indicated by lamps on the status panel for each actuation channel on C517 and C518. Each panel has 4 status lamps, one for each of the four bistable inputs to the actuation channel. The panel for each actuation channel also indicates the "failed/not failed" status of the actuation channel microprocessor.

b. ESFAS Actuation

   A contact for each of the two channels for the following actuation signals to indicate an actuation trip. The actuation signals alarmed are SIAS, CIAS, CSAS, EBFAS, MSIAS, SRAS, AEAS and AFAIS. A contact for each of the actuation channels for the pressurizer low-low pressure, mainsteam isolation to and AFAIS to indicate manual blocking of the channel.

   A contact for each of the actuation channels for both the pressurizer low-low pressure and main steam isolation to indicate permission to block the actuation channel.

c. Sequencers

One contact per sequencer for indication of the automatic sequencing in progress.

One contact per sequencer for indication of sequencer malfunction.

7.3.2.10 Cabinets

a. The ESAS cabinets are located in the main control room behind the main control board (see Figure 7.6–1).

b. The AFAIS circuitry is located in cabinets C517, C518 and SPEC 200 Racks, RC-30A, B, C, D, A-1, B-1 in the main control room behind the main control board (see Figure 7.6–1).

c. The ESAS cabinets have hinged rear doors and open fronts with semiflush mounted devices.

d. Cabinets RC-30A, B, C, D, A-1, B-1 have hinged front and rear doors, removable side panels with nest-mounted modules. C-517 and 518 contain channel status lamps and key-operated channel bypass switches.

e. Any adjustment required for routine calibration is accessible from the cabinet front. Adjustments for routine calibration do not require removal of equipment from its housing. Adjustment controls have keyless locking mechanisms and/or they are recessed in the cabinet or applicable chassis.

f. Nameplates are used to identify all cabinets, channels, and assemblies. Individual channels are identified by color code throughout the system.

7.3.2.11 Analog Bistables

The ESAS analog comparator bistables are designed to meet the following specifications:

a. The trip-point adjustment of the analog channel bistables covers the input signal range from 0 through 100 percent. The adjustment is by means of a calibrated dial. The resolution of adjustment is ± 0.25 percent when suitable external measuring equipment is used in connection with setting the trip point.

b. The deadband or hysteresis of the bistable is internally adjustable.

c. The noise appearing in the system, such as power supply noise, switching transients, or noise included in wiring entering or leaving the cabinets, will not induce false outputs nor prevent true actuation.

d. The error of the bistable units will not be greater than 0.5 percent of span over a 30 day period for the combination of the effects of ambient temperature drift and supply voltage and frequency variations.

e. The bistables automatically reset when trip conditions no longer exist.

The AFAIS bistable is a Foxboro Spec 200 module with an adjustable setpoint.

7.3.2.12 Power Supply

The ESFAS and RC-30A, B, C, D, A-1 and B-1 cabinets are powered from four vital DC-AC inverters which are supplied from two vital 125 volt DC batteries. Each battery supplies two inverters. The four inverters supply power to the four sensor channels. Redundant power for the two actuation channels is supplied from the two redundant 125 volt DC batteries.

Each ESAS sensor cabinet is provided with a backup 120 volt vital AC source via isolation transformers internal to each sensor cabinet. Sensor Cabinet A is backed up by vital AC supplied from Sensor Cabinet D and vice versa. Sensor Cabinet B is backed up by vital AC supplied from Sensor Cabinet C and vice versa.

SPEC 200 Instrumentation Cabinets RC30A, B, C, and D, which provide most of the signal inputs to ESAS, are not provided with backup AC power. Loss of a vital DC bus with no credit of the non-vital Turbine battery results in loss of two inverters. The ESAS Sensor Cabinets would remain energized, and the associated SPEC 200 cabinets would de-energize. Loss of two inverters would result in all ESAS signals with the exception of LNP and SRAS.

A power supply interruption of up to 35 milliseconds or a voltage dip of 20 percent for up to 30 seconds will not produce actuation of a ESAS sensor channel bistable.

The availability of DC relay power for the ESAS is constantly monitored by the automatic test insertion (ATI) portion of the system. In the event of loss of DC power supply in either actuation channel, the ATI fault annunciation will sound. In addition, the power supply monitor lamp on the actuation cabinet front panel will extinguish.

Refer to Section 8.6 and Figure 8.5–1 for more detail on the vital instrument power supply system.

7.3.2.13 System Interface

The following redundant process variable signals are input to the ESFAS:

a. Pressurizer pressure

Range (psia)	1500-2500
Input (volts, DC)	1-5

b. Containment pressure

Range (psig)	60-0
Input (ma, DC)	4-20

c. Steam generator pressure

Range (psia)	0-1000
Input (volts, DC)	1-5

d. Fuel handling area radiation

Range	.1 mr/hr to 10^4 mr/hr logarithmic
Input (volts, DC)	5-1

e. Refueling water storage tank level

Range (%)	0-100 [1]
Input (ma, DC)	10-50

f. Emergency bus voltage

Range (vac)	0-120
Input (vac)	0-120

g. Steam generator water level

Calibrated range (inches of water column)	174.5-42.3
Input (ma, DC)	4-20

h. Containment radiation (Gaseous and particulate)

Range	10 cpm to 10^6 cpm
Input (volts, DC)	5-1

Isolated DC outputs from each sensor of items b and e above are provided for main control board indications.

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1. Note: Due to physical instrument tap locations, RWST level cannot be monitored below approximately 4%.

Pressure transmitters for items a, b and c above are shared for the RPS and the ESFAS.

The system has output relays that provide electrically isolated contacts for actuation of equipment as tabulated on Figures 7.3–2A through 7.3–2D.

7.3.3 SYSTEM OPERATION

7.3.3.1 Operational Bypasses

Means are provided for the operator to block actuation of the SIAS while the RCS is undergoing depressurization. Blocking is effected as follows:

a. Permitted only when the pressurizer pressure is less than 1850 psia on three out of four analog sensor channels.

b. Initiated independently for channels 1 and 2 by momentary contact closure switches on the main control panel.

c. Removed automatically when the pressurizer pressure exceeds the 1850 psia block permit pressure on two out of four analog sensor channels.

d. Block only the output of the 1714 psia pressurizer pressure trip 2/4 matrices.

e. Permission to block and any block initiations are annunciated on the main control board.

Means are provided for the operator to block actuation of the MSIAS during startup or shutdown. Blocking is effected as follows:

a. Permitted only when the steam generator pressure is below 700 psia on three out of four analog sensor channels.

b. Initiated independently for channels 1 and 2 by momentary contact closure switches on the main control panel.

c. Removed automatically when the steam generator pressure exceeds the 700 psia block permit pressure on two out of four of the analog sensor channels.

d. Block only the output of the 572 psia steam generator pressure trip 2/4 matrices. The High Containment Pressure MSIAS trip function is not blocked.

e. Permission to block and any block initiations are annunciated on the main control board.

Means are provided for the operator to block actuation of the AFAIS at any time. Blocking is effected as follows:

a. Initiated independently for Trains A and B by placing the respective control selector switch in the block position, on the main control board or hot shutdown panel.

b. Any block initiations are annunciated on the main control board.

7.3.3.2 Bistable Trip Bypass

By means of a key lock switch, any one of four redundant ESAS channel bistables can be bypassed such that trip signals in two of the remaining three channels are required to initiate an actuation channel output. The key is retained when the switch is in the bypassed position. Each group of four sensor channel bistables has a single noninterchangeable key with a registered combination such that only one sensor channel of a group is capable of being bypassed at any one time. An alarm is annunciated if any bypass is actuated.

For the AFAIS, the coincidence logic inputs associated with any one of the four redundant channel bistables can be bypassed by means of two key lock switches; one for Train A and one for Train B. In this condition, trip signals in two of the remaining three channels are required to initiate an AFAIS actuation. The keys are retained when the switches are in the bypassed position. The bypass keys are administratively controlled such that only one AFAIS trip channel can be bypassed in both trains of actuation logic at any one time. Train-specific alarms are annunciated if any bypass is actuated.

7.3.4 AVAILABILITY AND RELIABILITY

7.3.4.1 Special Features

Means for continuously monitoring and indicating the ESFAS status are provided by indicating lights on the front of the cabinets. All indicating lights have features for manually checking bulb function.

Means for monitoring the status of the AFAIS is provided by indicating lights on the main control board, RC-517, 518 and hot shutdown control panel and annunciators on the main control board.

7.3.4.1.1 Sensor Channel Surveillance

The functions indicated by a lighted bulb are as follows:

a. Bistable tripped

b. Power supply failure

For the AFAIS, the functions indicated by a main control board annunciator are as follows:

a. Sensor channel bypassed

7.3.4.1.2 Actuation Channel Surveillance

The ESAS functions indicated by a lighted bulb are as follows:

a. Tripped actuation subchannel

b. Sequential actuation system blocking

c. Sequencer tripped

For the AFAIS, the functions indicated by a lighted bulb are as follows:

a. Blocked actuation train

b. Actuation train - power available

c. Actuation train - initiated

d. SPEC 200 micro failure

The functions indicated by main control board annunciation are as follows:

a. Actuation train initiated

b. Actuation train blocked

c. Sensor channel bypassed

7.3.4.1.3 System Test Surveillance

In addition to the indicating lights, a matrix of indicating lights is furnished as part of the ESAS ATI to locate failures.

7.3.4.2 Tests and Inspection

7.3.4.2.1 Testing

ESAS and AFAIS are tested in accordance with plant procedures and Technical Specification requirements. These tests confirm the operability of the ESFAS, final actuated equipment, and all supporting subsystems and power supplies.

7.3.4.2.2 Testing Features

The ESFAS incorporates the following testing features:

a. Bistable trip test

A variable test input signal with an adjustable range consistent with the bistable setpoint range is applied at the analog channel inputs to verify bistable setpoint and operation. Means are provided for indication of proper return to normal operation following completion of test.

b. Actuation channel trip test

Each ESAS coincidence 2-out-of-4 matrix is provided with independent test switches. Operation of the test switches will cause an output of the associated coincidence matrix and trips the related actuation channel logic. The overlapping testing procedure and the arrangement of the matrix ensure that a protective action will occur if any combination of two sensor channels simultaneously trip.

The AFAIS actuation channel trip test is performed by producing an output from the logic matrix which trips the related actuation channel logic.

c. Automatic Test Feature

The ESAS is provided with an automatic test feature which tests all combinations of two of four bistable trip conditions for each parameter. The automatic test feature automatically indicates and identifies logic faults and verifies bistable availability approximately every twenty-seven seconds. Each bistable is tested to check that the bistable trip setpoint is functioning properly, and will process a trip signal. This is accomplished by inserting two test pulses one after the other above and below the trip setpoints. After each test pulse, the output of the bistable is compared to the input by the ATI. A fault is indicated by lamps on the ATI panel and an annunciation on the main control board panel CO1. The time duration of the test pulses are sufficiently short to prevent picking up the output actuation relay. The ATI monitor is automatically turned off during any undervoltage trip condition.

7.3.4.2.3 System Reliability

Components and modules used in the manufacture of the actuation system exhibit a quality consistent with the nuclear power plant 40 year design life objective and with minimum maintenance requirements and low failure rates. ESAS reliability (mean time between failure) has been specified as a minimum of 1 x 10^4 hours with manual testing on a 30 day schedule.

TABLE 7.3.2.2-1 ESAS FAILURE MODE ANALYSIS FOR CONTAINMENT PRESSURE CHANNELS

Postulated Failure	Failure Mode	Method of Detection	Effect on ESAS	Remarks
1. Loss of one containment pressure channel	Bistable goes to tripped state	Annunciation from tripped bistable	No detrimental effect on system	System logic becomes 1-out-of-3 mode.
2. Loss of one 125 volt DC battery with 120 volt regulated AC source operational	Two inverters' loads transfer to regulated AC source	Annunciation	Same as above	
3. Loss of one 125 volt DC. battery with 120 volt regulated AC source inoperative	Loss of power to two sensor channels and one actuation channel	Annunciation	With the exception of SRAs and LNP channels, all 2 out of 4 sensor channels are tripped. Final equipment on the other redundant channel is actuated to its safe state from remaining redundant power source.	Sensor Cabinets remain energized, but associated SPEC 200 Instrumentation Cabinets and radiation monitors de-energize.
4. Loss of one 125 volt DC battery with 120 volt regulated AC available coincident with a failure of a containment pressure channel	1 and 2 above	Separate annunciation of each condition.	1 above	1 above
5. Loss of one 125 volt DC battery with 120 volt regulated AC not available coincident with a failure of a containment pressure channel	1 and 3 above	Same as 4 above	3 above	3 above

MPS2 UFSAR

FIGURE 7.3–1 ENGINEERED SAFETY LOGIC

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.3–2A ENGINEERED SAFETY LOGIC ACTUATED EQUIPMENT TABULATION

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.3–2B ENGINEERED SAFETY LOGIC ACTUATED EQUIPMENT TABULATION

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.3–2C ENGINEERED SAFETY LOGIC ACTUATED EQUIPMENT TABULATION

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.3–2D ENGINEERED SAFETY LOGIC ACTUATED EQUIPMENT TABULATION

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.3–3 ENGINEERED SAFETY LOGIC SEQUENCER & CHANNEL 5 OUTPUT

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.3–4 ENGINEERED SAFETY LOGIC

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

7.4 REGULATING SYSTEMS

The Reactor Regulating System (RRS), Pressure Regulating System, Pressurizer Level System, and Control Element Drive System (CEDS) are functionally identical to those in the Calvert Cliffs Plants (AEC Docket Numbers 50-317 and 50-318 through Amendment 19) with the exception of the Control Element Assemblies (CEA) motion inhibit which has been added to the CEDS, described in Section 7.4.2 and the RRS which does not incorporate the automatic CEA functions.

7.4.1 REACTOR REGULATING SYSTEM

7.4.1.1 Design Bases

The RRS does not have any automatic CEA control functions. The RRS consists of subsystems which provide the reactor operator with indication of T_{avg}, T_{ref}, and T_{ref}-T_{avg} deviation alarm. The RRS also provides signal inputs to the pressurizer pressure level program and steam dump program.

7.4.1.2 System Description

The RRS is comprised of two functional subsystems. The first includes two independent neutron flux measurement channels and the second subsystem is comprised of various I/O modules connected to a distributed control system (DCS) forming a single fault-tolerant control and indication channel. Two circuits of a non-vital instrument bus provide redundant power to the I/O modules. The system consists of:

a. Steam Dump Program Function Generator;

b. Pressure Level Set Point Function Generator;

c. T_{ref} - Function Generator;

d. T_{avg} - T_{ref} Calculation;

The system includes the following inputs:

a. Loop 1 T_{hot}, Loop 2 T_{hot} signals;

b. Loop 1 T_{cold}, Loop 2 T_{cold} signals;

c. Two first-stage turbine pressure signals.

The system develops the following outputs:

a. T_{ref} and T_{avg} signals to recorders;

b. Deviation alarms for T_{avg} - T_{ref}.

c. Pressurizer level setpoint

d. Condenser steam dump valve control signal (modulating)

e. Quick Open signal to condenser and atmospheric steam dump valves

7.4.1.3 System Operation

A temperature program calculation establishes the desired reactor coolant average temperature reference (T_{ref}) based on a power reference signal from first-stage turbine pressure. T_{ref} varies linearly with power from a nominal temperature of 532°F at hot standby to an adjustable limit of 520°F to 580°F at 100 percent power. T_{AVG} varies linearly with power from a nominal temperature of 532°F at hot standby to a nominal value of 568.9°F at 100 percent power. Operators may select T_{AVG} to be calculated from the Loop 1 temperature inputs, the Loop 2 temperature inputs, or the Loop 1 & Loop 2 temperature inputs combined. A deviation alarm will be generated if the difference between T_{ref} and T_{avg} exceeds the engineering established limit.

A steam dump valve position demand signal is calculated as a function of T_{AVG}. In addition, a quick opening binary signal is calculated as a function of this valve position demand signal; both of these signals are transmitted to the steam dump valve control system upon the initiation of a turbine trip.

A pressurizer level setpoint program is calculated as a function of T_{AVG} and transmitted to the pressurizer level controllers, external to the RRS.

7.4.2 CONTROL ELEMENT DRIVE SYSTEM

7.4.2.1 Design Basis

The reactor is controlled by reactivity adjustments with CEAs and with boric acid dissolved in the reactor coolant. Rapid changes in reactivity are compensated for or initiated by CEA movement. Long-term variations in reactivity due to fuel burnup and fission product concentration changes are controlled by adjusting the boric acid concentration. Since this rate of addition produces slow changes in the reactor power level, operator action suffices to control the boron concentration change. The shutdown CEA group provides a hot shutdown margin of at least 1 percent reactivity, even if the most reactive CEA is stuck out of the core. Prohibits require that the shutdown CEA group is in the full withdrawn position before other CEAs can be withdrawn, thereby assuring a shutdown margin equal to or greater than the required minimum. The CEA motion inhibit and alarm is provided when further insertion of the regulating group of CEAs would reduce the amount of effective shutdown reactivity in the CEAs below specified limits:

CEA movement is effected by the Control Element Drive Mechanisms (CEDM) (see Chapter 3). The CEDS transmits manual signals from the CEDS control panel to the Coil Power Programmers (CPP), which develop the pulses for magnetic jack operation.

CEA withdrawal will be prevented when a high power pretrip alarm or a thermal margin pretrip alarm is present. All CEA motion is inhibited when CEA malposition limits are reached:

7.4.2.2 Design Criteria

a. Racks, panels and associated equipment meet the Electronic Industries Associated Standard RS-310. Standard Nuclear Instrument Modules are utilized as adopted by the United States Nuclear Regulatory Commission (TID-20893).

b. Each switch on the CEDS panel is individually mounted in its own plug-in receptacle to permit fast replacement. The logic in the CEDS logic cabinet is modularized into group and individual CEA modules. All group modules are identical and all individual CEA modules are identical.

7.4.2.3 System Description

A block diagram of the CEDS is shown in Figure 7.4–3.

The CEDS control panel is a selection panel. Three types of selections are made by this panel: control mode, CEA group, and the individual CEA within each group. All selections are made by pressing the appropriate pushbutton switch. Upon selection the switch will light and remain lit and closed until another selection within its scope is made. Eleven selections will always be made: one mode selection, one group selection, and nine individual CEA selections (one in each of the nine groups). Electrical interlocks are incorporated in each of these eleven scopes of selection. This permits only one selection to be made in each scope. A new selection within any scope automatically cancels the previous selection.

There are three different modes of control of CEAs: Manual Individual, Manual Group and Manual Sequential. Two of these modes, Manual Individual and Manual Group apply to both Shutdown and Regulating CEAs. Manual Sequential Mode applies only to the regulating CEAs.

The following limits are provided by the CEA supervisory function of the plant computer to prevent the reactor from reaching undesirable conditions:

a. Upper CEA limit;

b. Lower CEA limit;

c. Upper CEA group stop;

d. Lower CEA group stop.

The CEDS and its associated interfaces with the plant computer and CEA Position Display System (CEA PDS) contain design features that ensure the following actions:

a. Insertion of the regulating CEAs before the shutdown CEAs are inserted;

b. Simultaneous withdrawal of no more than two groups of CEAs;

c. Proper sequential withdrawal of CEAs.

No single equipment or operator failure will cause the CEDS to improperly carry out the above listed actions to the extent that safety limits as defined in the Technical Specifications are reached.

Three lines of defense are utilized to ensure that safety limits are not exceeded. First, the reactor is operated under strict administrative controls which dictate the proper CEA movement. Second, alarms are provided to warn the operator if CEA movement is improper. The third line of defense is the functions and design features described below.

The plant computer CEA supervisory function operates CEA permissive contacts that feed the CEDM logic system. This design feature determines which group or groups of CEAs will be moved in the manual sequential mode. The computer also generates alarms if its logic detects an out of sequence movement of CEAs, more than two-groups movement of CEAs, improper insertion of shutdown CEAs, withdrawal of regulating CEAs prior to withdrawal of shutdown CEAs, or deviation of individual CEAs from their group.

The setpoints of these alarms are chosen such that the operator is given sufficient time to take corrective action before a safety limit is reached without alarming for normally occurring conditions.

If an equipment failure or operator error should cause any of these alarm conditions to be reached, another alarm is also received from the Reed Switch Position Indication System described in Section 7.5.3.3. This second system, diverse in nature from the plant computer CEA supervisory function, provides the operator with continuous indication of the position of each individual CEA and provides alarms redundant to those supplied by the plant computer. In addition, this system provides actuation signals to the CEA motion inhibit circuitry which stops all CEA motion when an alarm set point is reached.

The actuator signals are computed from CEA position information from the reed switch position transducers for each CEA and a reactor power signal (for power dependent insertion limit only). The actuation signals are independent of any control system function that would result in CEA motion. The actuation signals are contact openings that fail open upon loss of power. The actuation signals are banded together and the composite signal is sent to the lift coil power switch for each CEA from contact multiplying relays. The actuation signal to the lift coil power switch will stop any power from being applied to the lift coil, so that regardless of CEA control system motion demand, CEA motion is inhibited.

Actuation of the CEA motion inhibit upon violation of power dependent insertion limits is disabled below 10^{-4} percent power. Power dependent insertion limits are not needed below this power level. The motion inhibit is automatically re-enabled as power increases past 10^{-4} percent.

Equipment protective prohibits are also provided to prohibit regulating group withdrawal and prevent the reactor from reaching undesirable conditions. These interlocks are summarized in Table 7.4-1.

CEA speed is a function of the coil power programmer cycle speed, the CEA control system group speed setting and the mechanical limitations of the magnetic jack mechanism.

The coil power programmer that sequences power signals to the magnetic jack coils of an individual CEA has an upper operating setting of forty inches per minute for regulating CEAs and twenty inches per minute for shutdown CEAs. This maximum speed would result only from a continuous demand for withdrawal from the CEA control system.

A continuous withdrawal signal from the CEA control system would result only during abnormal operating circumstances. During normal operating conditions a continuous demand for an increase in reactor power will result in a sequenced withdrawal of CEAs within a group. The average speed of CEAs within the group is determined by the speed setting of the group programmer in the CEA control system. The upper bound of the speed setting of the group programmer is forty inches per minute, but the normal operating speed setting of the group programmer is the same as that of the individual CEA coil power programmer. Maximum CEA speed is determined by the setting of the individual CEA CPP, and this speed cannot be increased by any setting of the group programmer.

The absolute maximum speed of withdrawal of any CEA is determined by the magnetic jack latching mechanisms and coil current decay time constants. With the present coil power sequencing cycle, the mechanical restraints of latch operating time and coil drop out time allow a maximum possible withdrawal speed of 45-50 inches per minute.

7.4.2.4 System Operation

The CEAs are divided into the following groups:

a. Shutdown: two groups;

b. Regulating: seven groups.

Each CEA remains stationary except when a raise or lower signal is present. In response to a signal, the regulating CEAs move at a speed of up to 40 inches per minute. The shutdown CEAs move at a fixed speed of 20 inches per minute.

The CEA position setpoints are shown on Figure 7.4–2.

The shutdown CEAs may be moved with either manual individual or manual group movement. A selector switch prevents withdrawal of more than one shutdown group at any time. The shutdown groups must be withdrawn above a lower limit before regulating group withdrawal is possible. A limit prevents group insertion of shutdown CEAs unless the regulating groups are fully inserted. These limits may be bypassed to allow the CEA groups to be withdrawn or inserted out-of-sequence during testing (e.g., rod drop time tests, rod worth measurements, etc.) or system maintenance using approved plant procedures.

Regulating CEA groups may be moved in manual group or manual sequential mode. Individual CEAs may be moved in the manual individual mode of control. Sequential group movement provides that when the moving group reaches a programmed low (high) position, the next group begins inserting (withdrawing); the initial group stops upon reaching its lower (upper) limit. This procedure, applied successively to all regulating groups, allows a smooth and continuous rate-of-change of reactivity.

Under sequential group control, when the regulating groups reach the “prepower dependent” insertion alarm point, this condition is annunciated. If sequential group insertion is continued, a “power dependent” alarm point limit is reached and a second alarm is initiated. These two programmed limits may be adjusted during the life of the plant and are provided to aid the operator in assuring adequate shutdown margin.

All CEAs are prevented from being withdrawn if either a high power or thermal margin/low-pressure (TM/LP) pretrip condition exists.

There is provision for manually bypassing the CEA motion inhibit circuitry at the CEDS operator’s console. The manual bypass requires a minimum of two operator actions to accomplish the bypass. The group(s) to be bypassed must be manually selected by the operator by depressing one pushbutton for each group. Actuation of any group bypass pushbutton results in an alarm at the Main Control Board alerting the operator of the bypass and in the bypass pushbutton illuminating red. All control pushbuttons on the console illuminate white with the exception of the “off” pushbutton that also inhibits all CEA motion and illuminates red. In addition to selecting the group to be bypassed, the operator must also depress the “CEA motion inhibit bypass” button to accomplish the bypass function. This pushbutton must be held depressed by the operator during the bypass operation, as release of this pushbutton will cause the bypass to be removed. Actuation of the “CEA motion inhibit bypass” pushbutton will also cause an alarm at the Main Control board, alerting the operator of the bypass, and in the pushbutton illuminating red.

7.4.2.5 Consequences of Single Failures

The Reactivity Control System contains two independent and diverse CEA position supervisory and control systems.

The pulse count system uses pulse counting techniques in the Plant Computer to determine and maintain a current record of the position of each CEA. These CEA positions are used to derive CEDS control inputs and CEA misalignment alarms. These alarms include regulating group sequencing and overlap violations, excessive CEA deviation within any group and excessive insertion of any regulating group below the prepower or power dependent limits. These alarms are outside the normal operation region of the CEDS and warn the operator that the CEAs are in an abnormal configuration while they are still well within fuel design limits. Each alarm is annunciated separately at the main control board.

The CEAPDS derives its CEA position information from a Reed Switch Position Transmitter (RSPT) which is mounted on each CEDM. The RSPT is a magnetically operated voltage divider which outputs an analog signal proportional to CEA position. This system provides a continuous display on the main control board of all CEA positions and monitors the positioning of CEAs to provide the same alarms as detailed in the description of the digital system. The CEAPDS only control function is to initiate a CEA motion inhibit if any one or combination of these alarm conditions are reached. As with the pulse count system, each alarm is annunciated separately at the main control board. The system is designed such that on a loss of electrical power each alarm is annunciated and CEA motion inhibit is initiated, (see Figure 7.4–9).

A single failure, for this response, has been defined as a component failure in either the active (energized) state or in the inactive (deenergized) state of the component, or a short circuit or open circuit in any signal line.

The Reactivity Control system is shown in block diagram form in Figure 7.4–8. The system has been separated into five functional blocks.

Functional block number 1 represents the integrated circuit logic elements and associated power supplies that determine motion demand for each of 61 CEDMs. The outputs of this block are raise and lower contact closure signals to the CPP and contact opening signals for the motion inhibit block circuit.

Single component failures within Figure 7.4–8 functional block number 1 may cause spurious raise or lower signals to the CPP timers. This spurious demand signal will cause CEA motion. This motion is monitored by the Control Element Assemblies Position Display System (CEAPDS) (functional block number 4). The CEAPDS monitors CEA deviations (highest to lowest CEA), out of sequence motion, violation of allowed CEA group overlap, power dependent insertion limits, shutdown group insertion prohibit and regulating group withdrawal permissives. When the CEAPDS senses motion that is in violation of the above conditions, the calculator will output a contact opening causing the motion inhibit relays to deenergize. When the motion inhibit relays deenergize they open a contact in the control circuit of the lift coil power switch of each CEDM, preventing the silicon controlled rectifiers from energizing. Deenergizing the lift coil power switch control prevents further motion, either insertion or withdrawal, regardless of demand from the CPP or CEDS logic. This inhibit does not affect the reactor trip capability of the CEAs as the reactor trip function will interrupt motive power input to all power switches, causing gravity insertion of all tripping CEDMs.

The setpoints of the CEAPDS are sufficiently conservative that the fuel design limits will not be reached for motion of any CEA or combinations of CEAs.

A single failure in functional block number 1 may also cause a spurious “off” signal to a lift coil power switch. This failure will cause the affected CEA to hold, regardless of motion demand. If motion is demanded of the group containing the affected CEA, the CEAPDS will sense the CEA deviation and cause all other CEA motion to be inhibited. The failure within functional block No. 1 cannot prevent the action of the inhibit circuitry since the circuitry of function block No. 1 is isolated from the inhibit circuitry by the use of relays that provide inherent coil to contact isolation, as shown on Figure 7.4–9.

A single failure within functional block number. 2 may cause a single CEA to raise or lower or drop. This failure will be sensed by the CEAPDS as a CEA deviation and the CEAPDS will cause all CEA motion to be inhibited as detailed above, before fuel design limits are reached.

A single failure within functional block number. 3 may cause a CEA to be held if it occurs within the lift coil power switch or a single failure may cause a single CEA to drop or hold if it occurs in one of the other coil power switches. These failures will be sensed by the CEAPDS as a CEA deviation and the CEAPDS will cause all CEA motion to be inhibited before fuel design limits are reached.

A single failure within functional block number 4 may cause a spurious motion inhibit signal which will block all CEA motion. This failure will not cause fuel design limits to be reached since all CEA motion is prevented.

A single failure within functional block number 4 may also prevent actuation of the motion inhibit signal. This failure, however, will not cause fuel design limits to be approached as it cannot cause CEA motion.

The integrity of the CEAPDS will be assured by periodic testing.

A single failure within function block number 5 may cause the motion of some or all CEAs to be inhibited spuriously. The CEAPDS will sense this failure as a CEA deviation and cause the remainder of the motion inhibit relays to deenergize, inhibiting the motion of all CEAs before fuel design limits are reached.

A single failure within functional block number 5 may also prevent the actuation of the motion inhibit signal for some or all CEAs. This failure will not cause fuel design limits to be approached as it does not affect CEA motion. The integrity of the motion inhibit relays is assured by periodic testing.

Since functional block number 5 is implemented with relay logic, there are no single failures which can both affect CEA motion and prevent the actuation of the motion inhibit signal.

As shown in detail above, no single failure in any of the five functional sections of the reactivity control system will cause fuel design limits to be reached.

7.4.3 REACTOR COOLANT PRESSURE REGULATING SYSTEM

7.4.3.1 Design Bases

The reactor coolant pressure regulating system is functionally identical to that in the Calvert Cliffs Plant (NRC Docket Numbers 50-317 and 50-318).

The reactor coolant pressure regulating system maintains system pressure within specified limits by the use of pressurizer heaters and spray valves. Pressurizer pressure and level sensors provide input to the system.

7.4.3.2 System Description

A high pressurizer pressure functions to open the pressurizer spray valves on a proportional basis, thereby reducing pressure. A low pressurizer pressure functions to energize heaters on a proportional or group basis to increase pressure. A high pressurizer water level energizes the backup heaters in anticipation of a low-pressure transient; a low pressurizer water level de-energizes all heaters, for heater protection.

Two channels of control are provided and the controlling channel is selected by a switch. Manual control of the heaters and spray may be selected at any time.

7.4.3.3 System Operation

Two pressure channels independent of those in the Reactor Protective System (RPS) provide suppressed range (1500 to 2500 psia) signals for control of the pressurizer heaters and spray valves. The output of either controller may be manually selected to perform the control function. During normal operation, a small group of heaters is proportionally controlled to maintain operating pressure. If the pressure falls below the proportional band all of the heaters are energized. Above the normal operating range the spray valves are proportionally opened to increase the spray flow rate as pressure rises. A small, continuous spray flow is maintained through the spray lines at all times to keep the lines warm to reduce thermal shock when the control valves open, and to aid in keeping the boric acid concentration in the coolant loops and pressurizer in equilibrium.

Outputs from the two pressure control channels are recorded in the control room and provide individual high and low alarms.

The control and alarm pressure setpoints are shown in Figure 7.4–4.

7.4.3.4 System Evaluation

Two individual channels are available for automatically regulating the pressurizer heaters and spray valves. Either channel may be used to control the pressure in the system, and the output from both channels is recorded in the control room. Individual high and low pressure alarms are provided.

7.4.4 PRESSURIZER LEVEL REGULATING SYSTEM

7.4.4.1 Design Bases

The pressurizer level regulating system is functionally identical to that in the Calvert Cliffs Plant (NRC Docket Numbers 50-317 and 50-318).

Pressurizer level is maintained by the action of the chemical and volume control system (see Section 9.2). The level set point is programmed as a function of coolant average temperature (T_{avg}). A low pressurizer level signal functions to reduce letdown flow proportionally and to start the available nonoperating charging pumps. A high level indication functions to increase letdown flow proportionally by opening the letdown control valves and stopping all but one charging pump. There are two independent automatic control channels with channel selection accomplished by means of a manual control switch. Automatic control is normally used during operation but manual control may be utilized at any time.

7.4.4.2 System Description

Two level channels provide pressurizer level signals for control of two specific functions:

a. A low level signal from either channel deenergizes all heaters;

b. A high level deviation signal from the controlling channel energizes the backup heaters and sets the proportional heater control to full power.

7.4.4.3 System Operation

The operating level in the pressurizer is programmed as a function of power to accommodate plant load changes and transients to minimize the changes in reactor coolant system (RCS) volume (see Figure 4.3–9).

The level programmer establishes a program level which is directly proportional to coolant average temperature, over the operating range of T_{avg}. The average temperature signal used by the level programmer is the signal used by the RRS.

The level controller compares the measured and programmed level signals and generates a proportional signal for regulating the letdown control valves. In addition, the level controller functions to start or stop additional charging pumps at low or high level set points. The outputs of either of two automatic control channels may be selected by the operator for level control in addition to manual control.

7.4.4.4 System Evaluation

Two level control systems are provided. The controllers are located in the control room. Both automatic and manual control of level is provided. Three charging pumps and two letdown control valves provide redundant means of increasing or decreasing reactor coolant inventory.

The variable pressurizer level control program maintains the proper coolant inventory by means of discharge or addition as required during plant load changes.

7.4.5 STEAM DUMP AND TURBINE BYPASS SYSTEMS

7.4.5.1 Design Bases

7.4.5.1.1 Functional Requirements

a. The steam dump to atmosphere systems (one each for the two steam generators) provide means for unit cooldown in the event that the condenser is not available.

b. The turbine bypass system provides a means for removing decay heat and pump heat at no load conditions and a means for unit cooldown.

c. The steam dump to condenser, steam dump to atmosphere, and bypass systems combined provide a means of dissipating excess Nuclear Steam Supply System (NSSS) stored energy and sensible heat following a simultaneous reactor and turbine trip from full load without lifting the secondary safety valves.

7.4.5.1.2 Design Criteria

The steam dump to atmosphere systems will have a minimum combined capacity of 15 percent of full load in order to meet the functional requirements of a.

The bypass system will have a minimum capacity of 5 percent of full load in order to meet the requirements of b.

The total steam dump to condensers, steam dump to atmosphere and turbine bypass system will have a minimum capacity of 55 percent of full load in order to meet the functional requirements of c.

The steam dump to atmosphere control valves can be controlled from the main control board or the hot shutdown panel external to the control room.

The steam dump to condenser control system, when in automatic operation, shall be limited in reactor coolant cooldown capability.

The steam dump to atmosphere system for one steam generator shall be redundant and independent of that for the other steam generator.

7.4.5.2 System Description

A block diagram of the steam dump to the condenser and turbine bypass system is shown on Figure 7.4–5.

The steam dump to condenser and turbine bypass valves can be controlled at their respective control room stations in either automatic or manual mode.

The steam dump to atmosphere control valves can be controlled from the main or auxiliary panels. Automatic and manual modes are available on the main control panel and manual mode is available on the hot shutdown control panel. The steam dump to atmosphere system for steam generator number 1 and steam generator number 2 reside partially on a distributed control system utilizing common fault tolerant control processors. Pressure sensor and control feedback input signals and control output signals are independent of each other.

The total steam dump and turbine bypass is sufficient to prevent the lifting of the secondary steam safety valves following a simultaneous reactor and turbine trip at full power. The capacity of each of the steam dump to condenser and turbine bypass valves is approximately 10 percent of full load for a total of approximately 40 percent of full load. The capacity of the steam dump to each atmosphere valves is 7.5 percent of full load, each, for a total of 15 percent of full load. The capacity of the steam dump to the atmosphere system is sufficient for plant cooldown after a full load trip even if for any reason the condenser is not available.

Excessive cooldown of the RCS by the steam dump to condenser valves, when in automatic operation, is prevented by a narrow range temperature signal which has a minimum output corresponding to 532°F. At this point, the steam dump to condenser flow demand will be zero while the bypass valve remains in operation to control header pressure. The turbine bypass system will tend to control the steam pressure to 900 psia during hot standby when the condenser is available.

7.4.5.3 System Operation

Steam is discharged from the main steam lines to the condenser by way of the dump to condenser and bypass valves in response to T_{avg} and secondary pressure signals. Inputs to the system are the T_{avg} turbine trip signal, main steam line pressure and condenser loss of vacuum.

Upon receipt of a turbine trip signal the steam dump to condenser controller generates a suppressed range signal proportional to the quantity T_{avg} -532°F. When T_{avg} is within a predetermined range, the signal is sent to the steam dump to condenser valves and the turbine bypass auctioneering unit via the steam dump to condenser vacuum permissive contacts. The turbine bypass pressure controller generates a suppressed range signal proportional to secondary pressure over the range of 800 to 1000 psia. The turbine bypass valve receives the higher of the steam dump to condenser controller or turbine bypass controller signals through an auctioneering unit. When T_{avg} exceeds the predetermined range, a quick opening signal from the RRS opens the steam dump to condenser valves and simultaneously opens the turbine bypass valve. In either case, loss of condenser vacuum will prevent opening of the turbine bypass or steam dump to condenser valves.

If it is necessary to boost the capacity of the bypass system near the end of cooldown, the dump to condenser valves may be operated on manual control. The operator may control plant cooldown by combined manipulation of the steam dump to condenser and turbine bypass controllers, as required.

The steam dump to atmosphere system is set to a preselected dump pressure below safety valve setting and above bypass system setting on the main control panel controller. The controller is equipped with proportional mode only to give rapid response without reset windup, while maintaining its capability to modulate smoothly. Standard switching procedures on the local panel transfer station allow the operator to control the steam dump to atmosphere manually with an adjacent pressure indicator for guidance. Either the automatic controller setpoint or the local station on manual control may be used for cooldown.

7.4.6 TURBINE GENERATOR CONTROL SYSTEM

7.4.6.1 Design Basis

7.4.6.1.1 Functional Requirements

The turbine generator is supplied with an electrohydraulic control (EHC) system. The EHC system provides speed and acceleration control for startup, load control, load limiting, emergency trip, manual trip and valve testing functions.

7.4.6.1.2 Design Criteria

The EHC system is designed to provide dependable and accurate startup, load control and emergency trip functions. Redundant electronic speed sensing for control and high pressure hydraulic fluid supply systems are incorporated to increase system reliability. A primary electronic overspeed protection system is provided with an independent emergency electrical overspeed protection system.

7.4.6.2 System Description

7.4.6.2.1 System

The General Electric Company EHC system consists of electrohydraulically operated turbine control and stop valves operated from an electronic control cabinet and an operator’s panel located on the main control board. The operator’s panel contains controls for startup, load control, operational testing of valves and overspeed trip system and manual trip. System alarms and indicators are provided for monitoring system operation.

The EHC system includes the following features:

a. Full arc admission

b. Manual set load limit

c. Initial pressure regulator

d. Chest warming control

e. Load set capability

The control and stop valve operators receive signals from the EHC control cabinet and use fire resistant hydraulic fluid for positioning.

A high-pressure fluid power unit supplies fluid under 1600 psi pressure for operation of servo valve power actuators. The fluid power unit consists of two redundant pumps, coolers and filters arranged in parallel.

Electric power for the electronic logic and trip circuits is provided by two redundant battery-backed uninterruptible power supplies. All power supplies are monitored to provide indication of out of tolerance voltage.

Three redundant primary speed signals are provided to permit speed and acceleration control using the median of the three speed signal values. An independent set of three redundant emergency speed signals are used for emergency overspeed protection. Failure of a single speed signal will result in a system alarm. The turbine will be automatically tripped should a loss of two primary speed signals or a loss of two emergency speed signals occur. Tripping is initiated at nominal 109 percent of rated speed. An emergency electrical overspeed trip circuit is provided to trip the turbine in the unlikely event of a failure of the primary overspeed trip system or during an overspeed trip test. The emergency electrical overspeed trip system is independent from the primary speed control and overspeed protection system and is set to trip at nominal 109.5 percent of rated speed.

The primary overspeed trip and the emergency overspeed trip systems constitute two separate and independent means of protecting the turbine against an overspeed.

7.4.6.3 System Operation

7.4.6.3.1 Startup

The startup sequence followed by the operator consists mainly of the following steps which are controlled from the turbine generator control panel, CO7:

a. High-pressure turbine chest warming

b. Increase turbine speed to 1800 rpm in discrete steps

c. Perform overspeed trip test

d. Synchronize generator to the line and close the generator breaker

e. Establish initial loading rate

f. Set load limit and initial pressure limit

g. Increase unit loading to final desired value

7.4.6.3.2 Normal Operation

Under normal operating conditions, with no variables out of limit, the EHC system automatically maintains the desired unit load set by the operator. Also see Section 10.2.3.2.

7.4.6.3.3 Abnormal Operation

The turbine is automatically tripped on the following signals:

a. Turbine overspeed (primary or emergency systems)

b. Low condenser vacuum

c. Excessive thrust bearing wear

d. Deleted by FSARCR 05-MP2-023

e. Loss of generator stator coolant

f. Low bearing oil pressure

g. Loss of two primary OR two emergency speed signals

h. Deleted by FSARCR 06-MP2-007

i. High steam generator level

j. Loss of primary and secondary control power

k. Low hydraulic fluid pressure

l. Low shaft pump discharge pressure

m. Main generator and transformer protection system trip

n. Reactor trip

o. Power to Load Unbalance

Four mechanically and electrically isolated pressure switches are furnished on the EHC hydraulic trip system to provide four redundant channels to the RPS. The four pressure switches are combined into 2 out of 4 logic in the RPS to trip the reactor on a turbine trip.

7.4.6.4 Availability and Reliability

The EHC system is designed with highly reliable components. Maximum use is made of solid state components in measuring and logic circuits. The EHC control cabinet is completely factory assembled, wired, tested and calibrated prior to shipment to the job site. Availability is increased through the use of redundant electrical and hydraulic power supplies.

7.4.7 FEEDWATER REGULATING SYSTEM AND FEEDWATER PUMP SPEED CONTROL

7.4.7.1 Design Bases

7.4.7.1.1 Functional Requirements

The feedwater regulating system (one for each steam generator) is designed to regulate the flow of feedwater to:

a. Maintain an inventory of water in the steam generator as required by the power production rate.

b. Assure a heat sink for the primary system.

The feedwater pump speed control is designed to maintain the speed of the feedwater pumps (2) as required by steam demand.

7.4.7.1.2 Design Criteria

Criteria for the design of the feedwater regulating system include the following:

a. Provide a means for both manual and automatic operating modes from the main control board.

b. Have proportional and reset control actions.

c. Provide automatic ramping to close the main feedwater regulating valves on turbine trip.

d. The control room portion of the system is to be factory wired and tested and shipped as a unit (one for each generator).

e. Provide recorder and computer display of steam generator level, steam flow and feedwater flow.

f. Provide a means for locking main the feedwater regulating valves in their last position on air and controller signal failure.

Criteria for the design of the feedwater pump speed control include the following:

a. Provide a means for both manual and automatic operating modes from the main control board on an individual pump basis.

b. Provide a redundant control architecture to minimize the effects of single point failures.

7.4.7.2 System Description

7.4.7.2.1 System

The feedwater regulating system inputs are steam generator level, steam flow, and feedwater flow. Each of these parameters is measured with differential pressure type instrumentation.

The Feedwater Control System for the Millstone Nuclear Generating Station - Unit 2 is based on utilizing SPEC 200 — a simplified package for electronic control. SPEC 200 provides a highly reliable complement of control room instrumentation.

In addition, the SPEC 200 control equipment is type-tested for qualification per IEEE Standards 323-1974 and IEEE 344-1975. These type-tests establish that the equipment can properly continue to perform its safety related functions before, during, and after specified design basis events. Such events include seismic disturbances that are considered by Foxboro to represent the most severe that would be anticipated at most nuclear power plant locations, as well as postulated changes to environmental conditions at the equipment site, both normal and abnormal.

The SPEC 200 Feedwater Control System has been configured to control the flow of feedwater into the steam generators, thus maintaining the water level within the desired range during all phases of plant operation. The control system includes both single and three element control to maintain steam generator level by positioning of the feedwater regulating valves.

The SPEC 200 analog instrumentation has been configured to meet the following functions:

- Single Element Control
- Three Element Control
- Main Feed Valve and Bypass Valve Control
- Process Measurement

A major consideration in meeting the above functions is to eliminate inadvertent plant trips due to the Feedwater Control System. Specifically, the effects due to transmitter failure, power supply disruption and operator error are taken into account and minimized.

All instrumentation is mounted in a total of two (2) SPEC 200 Racks, each rack will contain the same instrumentation with one (1) rack for each of the two Feedwater Control Elements.

All instrumentation is Class II (seismic only) as applicable to a Nuclear Power Plant.

SINGLE-ELEMENT LEVEL CONTROL

The single-element control consists of one control unit which receives the level signal and an external generated setpoint signal (from the “master controller”) and produces an output signal as a function of the difference between these signals. The rate and magnitude of the output signal change is determined by the control function. The single-element control also has an input from the RCS cold leg temperature to provide dynamic compensation for steam generator level changes at a low power level.

This control unit maintains steam generator level at a manually set value when steam flow is less than 15 percent.

The single-element control units’ output will track the output of the three element control whenever the system is on three element control.

This tracking feature is accomplished by forcing or switching both the high and low limits of the controller to be equal to the three element controller output.

THREE-ELEMENT FEEDWATER CONTROL

This is a three-element system in which the primary or level control unit functions to trim level and the secondary or feedwater control unit functions to maintain the balance between feedwater and steam flow.

Feedwater flow demand is set by steam flow with corrections added for changes in level. An increase in steam flow automatically increases the demand for feedwater flow. To compensate for changes in level due to shrink and swell, blow down or any other cause, the level control unit trims the feedwater setpoint signal in such a manner that the steam generator level is always returned to its setpoint.

The level control unit compares a measurement of level with its manually adjusted setpoint to develop an output signal. This signal is fed to one input of the computing unit where it is combined with 50 percent bias. When level is at the setpoint, output from the control unit will settle at about 50 percent to combine with the computing unit bias. Whenever the level is not at the control setpoint, the level control unit will alter the feedwater control unit setpoint to trim the flow of feedwater to the steam generator. Trimming action will continue until the level returns to the control setpoint.

The feedwater control unit compares a measurement of feedwater flow with the trimmed steam flow signal to develop an output signal to regulate the feedwater valve. Any change in steam or feedwater flow is immediately sensed as a discrepancy in the actual feedwater/steam flow relationship by the level control unit. This control unit, tuned to the response of the feedwater loop, will correct the feedwater flow to restore the feedwater/steam flow relationship.

The measurement input to the feedwater control unit is fed to the reset circuit of the level control unit to prevent reset windup of the primary control unit when the control station is in manual control, or when the level control unit cannot follow its setpoint.

Three element feedwater control is the most stable and dynamically responsive steam generator level control scheme that can be provided. This system maintains a mass balance between measured feedwater flow and steam generator level setpoint error compensated steam flow. The steam generator level setpoint error signal is added to the measured steam flow value in such a manner as to reduce steam generator level setpoint errors to a minimum amount. This system does not require an upset in steam generator level for the feedwater control to be modified. Note that whenever measured steam flow changes, there is an immediate change in feedwater flow, thereby inhibiting the steam generator level upset.

Note that measured steam flow is an input into the three element feedwater control system. Steam demands placed on the main feedwater system by steam generator blowdown, atmospheric dump valves (ADVs), and steam generator relief valves are not direct inputs into this mass balance equation. The slight steam generator level error caused by the continuous steam generator blowdown is compensated for by the steam generator level setpoint error signal. Large steam demands placed on the system by ADVs or relief valves opening cannot be adequately compensated for in this system. These conditions will result in significant steam generator level fluctuations and possible trips due to low steam generator water level. The main feedwater control system was designed to respond to routine plant transients and feedwater perturbations. This system was not designed to ameliorate accidents or respond to all off-normal events.

The transfer between three element and single element control will be accomplished automatically. The transfer from single-element to three-element occurs at 15 percent load, (900,000 lb/hr.) increasing. A 4 percent deadband is provided for stability which means the transfer back from three-element to single-element will occur at 11 percent load, (660,000 lb/hr.) decreasing. A tracking network is provided for bumpless/balanceless transfer between three-element and single-element control. The tracking is accomplished by switching the high and low limits to the output of the single-element controller.

As a security feature, the feedwater control system is provided with redundant differential pressure transmitters for feed and steam flows. One transmitter is arbitrarily designated as a “Main” and the other as a “ALT.” Failure of either redundant transmitter will be indicated on the main control board. Failure is a signal greater than 102 percent or less than minus 2 percent of full scale.

In order to minimize the impact on the feedwater control system of a transmitter failure, the average of the steam flow signal and the average of the feed flow signal will be calculated and used for control purposes. The operator has the ability to select either the “Main,” “Alt,” or “Both” transmitters on a main control board mounted selector switch. Selecting the “Both” position on the selector switch will average the signals.

As with steam and feedwater flow, the steam generator level measurement is redundant; a “Main” and “Alt.” When the selector switch is in the “Both” position the SPEC 200 control circuitry will automatically select the “low” steam generator level signal to provide the most conservative level measurement signal to the control system. A transmitter failure will be indicated on the main control board.

The main feedwater regulating valves are locked in their last position in the event of air failure to the valve positioner. The valves are also locked in the event of a low or no signal from the main control board controller to the valve positioner. These features are independent on the control systems for each of the steam generators.

A schematic and block diagram of the control system is shown in Figures 7.4–6 and 7.4–7.

The feedwater pump speed control system sets feedwater pump speed based on steam flow. The speed setpoint is high selected with a discharge pressure setpoint to ensure a minimum discharge pressure is maintained. The control system also features a high discharge pressure limiter.

7.4.7.2.2 Components

The feedwater regulating systems consists of differential pressure measuring devices for the steam generator level, steam flow and feedwater flow.

The main control room components, with the exception of display and switching devices, are contained in a factory wired, tested and packaged system. The system for each steam generator is independent from the system for the other generator.

The parameters of steam flow and feedwater flow are recorded on a dual pen recorder on the main control board. The steam generator level is recorded on a single pen recorder on the main control board.

A control station for each generator includes provision for manual-automatic bumpless transfer switching and display of the generator level signal and is mounted on the main control board. This controller includes the steam generator level setpoint mechanism and displays a continuous indication of the output signal to the feedwater control valve.

The feedwater pump speed control system consists of measuring devices for pump speed, suction flow, suction pressure, and discharge pressure. Each controller is mounted in a separate cabinet which also houses a two-line display panel. The control panel for the feedwater pump speed control system is mounted on the main control board. It allows for auto/manual transfer, speed adjustment, and indication of major pump parameters. A hydraulic actuator is used to manipulate the SGFPT steam admission valves.

7.4.7.3 System Operation

The feedwater regulating system for each of the steam generators has a level control station titled steam generator level controller. The operator has provisions for automatic or manual level control with appropriate bumpless transfer switching and display of the generator level signal and the controller output signal to the respective feedwater control valve.

Each of the parameters of steam generator level, steam flow and feedwater flow is recorded on the main control board for operator guidance.

The feedwater pump speed control system incorporates features which allow the operator to control the system in manual mode. The speed of the feedwater pump(s) is adjusted by speed lower and speed raise buttons to the main control board.

7.4.7.4 Availability and Reliability

In the feedwater regulating system all equipment is designed with highly reliable components. Maximum use of solid state components in the electronic instruments and, where possible piston-operated valves are employed to fail locked (as-is) on loss of air or power.

All instrumentation and controls, where practical, are installed outside of the containment structure and in locations accessible for inspection and maintenance.

As mentioned, the feedwater valves can be placed on manual control from the main panel. Feedwater valve bypass valve can also be actuated from the main panel.

The feedwater pump speed control system uses three isolated control modules, each with its own power supply, processor, and input/output cards. The software coordinates data exchange between modules to support double and triple redundant processing. This design of the feedwater pump speed control system is highly reliable as it provides a high level of fault tolerance and detection.

With the provision for manual operation from the main control board, it is possible to remove the majority of the electronic circuitry for repair or replacement. Operation on manual control is equivalent to having fixed speed turbines in this service.

7.4.8 REACTOR COOLANT SYSTEM LOW TEMPERATURE OVERPRESSURIZATION PROTECTION (LTOP) SYSTEM

7.4.8.1 Design Bases

The LTOP system consists of two redundant relief trains each with a power operated relief valve (PORV) and associated relief piping. This system is controlled through a series of pressure and temperature actuated devices and hand controlled switches in the main control board and it is powered by independent and redundant power supplies. The LTOP system when operated as described below, ensures that the limiting mass addition transients due to an inadvertent start of a high pressure safety injection (HPSI) and/or a charging pump or the limiting energy addition transient associated with the start of a reactor coolant pump (RCP) do not result in peak pressurizer pressures in excess of the 10 CFR 50, Appendix G beltline limits (normal operation) developed for up to 54 EFPY. The LTOP system is required to be operable and capable of mitigating RCS pressure transients whenever the RCS cold leg temperature is ≤275°F, which exceeds the recommended temperature provided by Branch Technical Position RSB 5-2, unless the RCS is properly vented with a vent size ≥ 2.2 in^2. Under normal conditions, the LTOP system relies on the two pressurizer PORVs with a setpoint of ≤ 400 psig.

7.4.8.2 System Description

The LTOP system utilizes a combination of automatic activation devices, manual handswitches, and control board alarms to alert the operators and to automatically open the power operated relief valves in the event of an overpressure transient. An alarm which is set sufficiently lower than the PORV setpoint will provide operator warning of an ongoing RCS pressure increase prior to PORV opening.

The evaluations performed to demonstrate that the flow relieving capability of the PORVs is adequate to mitigate the corresponding design bases transients include system corrections. These corrections include the valve overshoot resulting from the PORV stroke time and system process delay time, fluid conditions at the valve inlet, RCS hydraulic effects resulting from RCS flow and elevation differences in addition to the PORV loop instrument uncertainties. These corrections ensure that the 10 CFR 50, Appendix G limits are not exceeded during the following pressure transients:

a. Energy Addition - The transient is caused by the start of an RCP with a maximum secondary to primary side temperature differential of 50°F. Since the expansion of the primary coolant resulting from the heat addition varies with the RCS pressure and temperature conditions, a maximum RCS pressure of 340 psia and a maximum RCS temperature of 275°F were chosen to ensure that the entire LTOP temperature range is bounded by the analysis.

   The first RCP is not allowed to start unless a pressurizer bubble of ≥ 900 cubic feet is established to ensure that adequate RCS volume is available to accommodate any potential primary coolant expansion without resulting in water solid conditions. The 340 psia was chosen since this is the maximum expected RCS pressure when the Shutdown Cooling (SDC) System relief valves are aligned to the RCS. Since the calculated peak RCS pressure was less than the Appendix G allowable pressure, no credit was taken for the PORVs to mitigate the energy addition transient.

b. Mass Addition - This transient results from the inadvertent start of a HPSI and/or a charging pump during RCS water solid conditions. The maximum flow from the allowed pump combinations was assumed to occur concurrent with the loss of decay heat removal capabilities and the energizing of the pressurizer heaters. These conservative assumptions in combination with the pump restrictions discussed below, resulted in system peak pressures which are lower than the allowable 10 CFR 50 Appendix G limits:

   1. A maximum of one HPSI and two charging pumps may be capable of injecting into the RCS when the RCS cold leg temperature (T_c) is 190°F <

$T_c \leq 275°F$ or when T_c is $< 190°F$ when the RCS is vented through a 2.2 in^2 passive vent.

2. When the RCS is not vented with a vent size of $\geq 2.2\ in^2$, a maximum of one charging pump may be capable of injecting into the RCS when $T_c \leq 190°F$.

7.4.8.3 System Operation

During plant cooldown, the Appendix G pressure/temperature limit curves are used to decrease the RCS pressure and temperature down to 300°F and 400 psig. Prior to cooling the RCS below 275°F, normal operating procedures require the activation of the PORV "Low" setpoint at ≤400 psig by resetting the hand switches to the "Low" position. To assure that the PORVs are reset to the "Low" set point prior to cooling the RCS to <275°F, an alarm (reset to low) is activated when the RCS temperature approaches 280°F. Upon resetting a PORV handswitch to "low" (i.e., LTOP mode), the respective motor operated block valve (2-RC-403 or 2-RC-405) upstream of the PORV receives an open signal. This ensures that the PORVs have not been isolated and are capable of performing their function. While the PORV "Low" setpoint is at ≤400 psig, the overpressure transient alarm is activated when the RCS temperature is below a preselected value of ≥ 275°F and the RCS pressure exceeds a preselected value of ≤ 360 psig. The purpose of this overpressure transient alarm is to alert the operator that pressure is increasing, and action to control pressure should be taken to preclude PORV actuation.

During plant heatup, normal operating procedures require the RCS pressure be maintained below 400 psig until the RCS temperature is greater than 275°F to preclude coolant discharge through the PORVs. When the RCS temperature exceeds 275°F, normal operating procedures require the operator to reset the PORVs to the "High" setpoint relief of approximately 2385 psig. The low temperature transient alarm is also deenergized at this time. After the PORVs are reset to the "High" setpoint of approximately 2385 psig, normal plant heatup continues accordingly.

The start of an RCP significantly changes the flow characteristics in the RCS from one of stagnant or low flow conditions to that of forced/high flow conditions. When no RCPs are in operation and SDC is in service, the vessel inlet temperature is best represented by the temperature reading of the SDC return line water temperature since little mixing may occur between the hotter RCS coolant and the colder SDC coolant. However, once the first RCP is started, the vessel inlet temperature is best represented by the RCS cold leg temperature readings since the SDC flow mixes with the remainder of the reactor coolant.

The start of the first RCP may result in a step temperature increase which may be further amplified by the increase in RCS energy resulting from the added RCP heat input. To minimize the temperature differential between the SDC return coolant temperature and the RCS cold leg coolant temperature, it may be desirable to reduce the SDC heat removal rate which may result in a small RCS temperature increase. Since the accumulation of residual heat in the RCS may also result in increased RCS pressure, the energy addition evaluations assumed that no decay heat would be removed from the RCS for a period of up to 5 minutes following the start of an RCP.

This bounding assumption will allow the operators to reestablish the required SDC flow following the start of an RCP without exceeding the allowable pressure/temperature limits.

TABLE 7.4-1 PROHIBITS ON CEAS

Withdrawal Prohibit Condition

Pretrip Overpower

Thermal Margin / Low-Pressure Pretrip

Motion Inhibit condition

Group out of sequence movement

Individual CEA deviation

More than two Regulating group movement

Withdrawal of Regulating groups prior to complete withdrawal of all Shutdown groups

Insertion of Shutdown groups prior to insertion of all Regulating groups

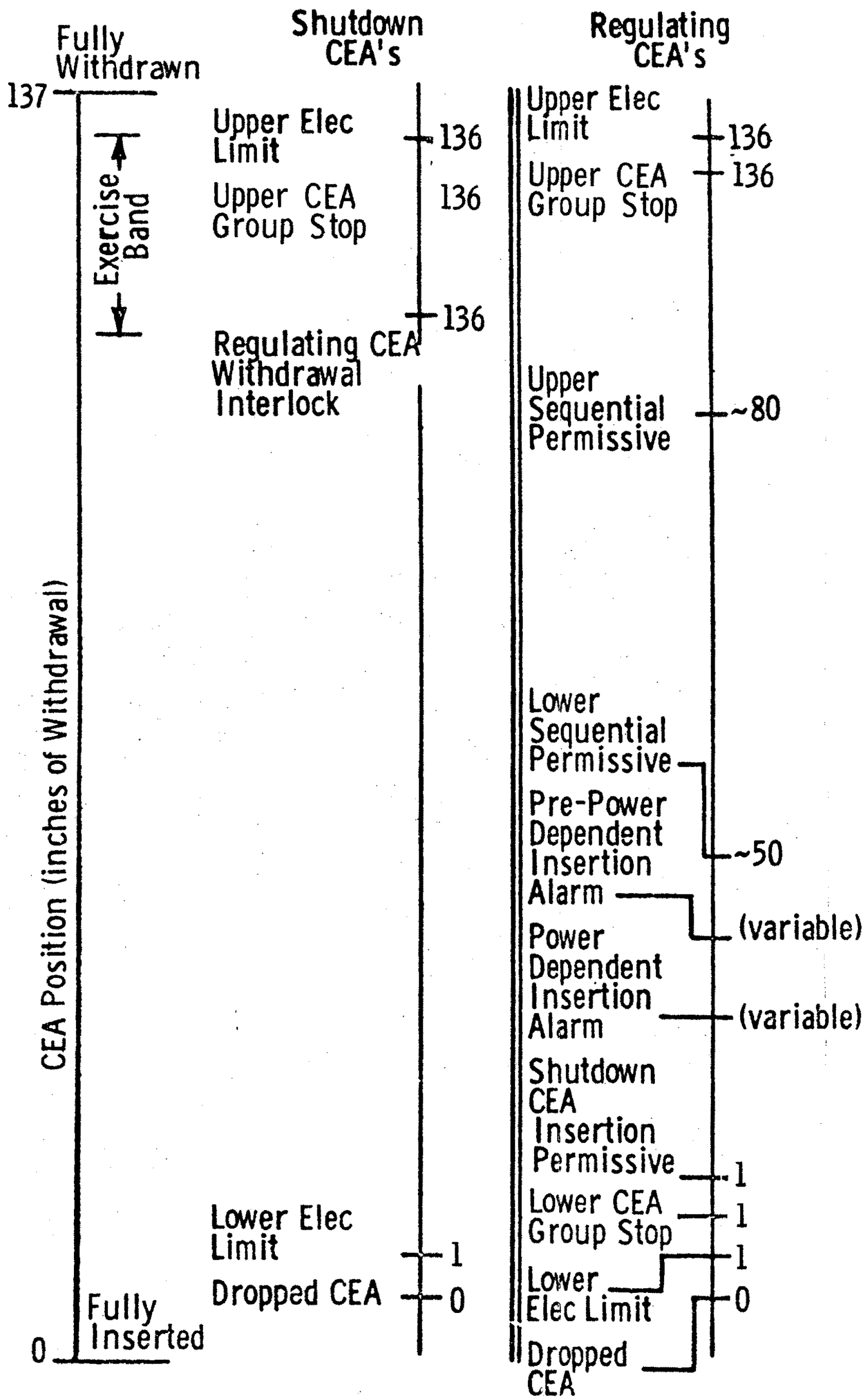
Input signal sources for the withdrawal prohibits:

Reactor Protective System: Pretrip Overpower Thermal Margin /Low-Pressure Pretrip

Input signal source for CEA motion inhibit:

Reed switch CEA Position Display System (CEAPDS)

<u>FIGURE 7.4–1 DELETED BY FSARCR 05-MP2-026</u>

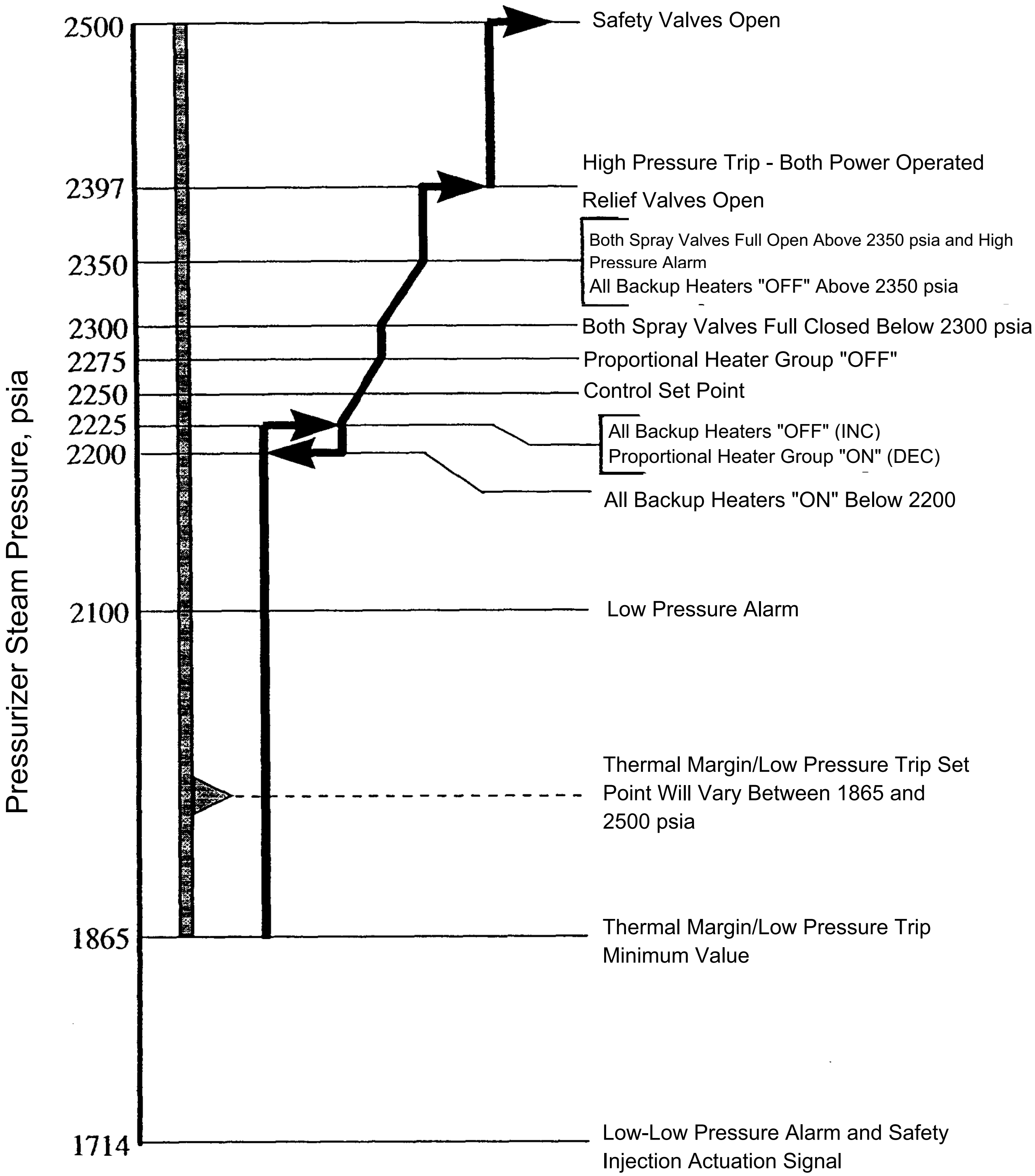


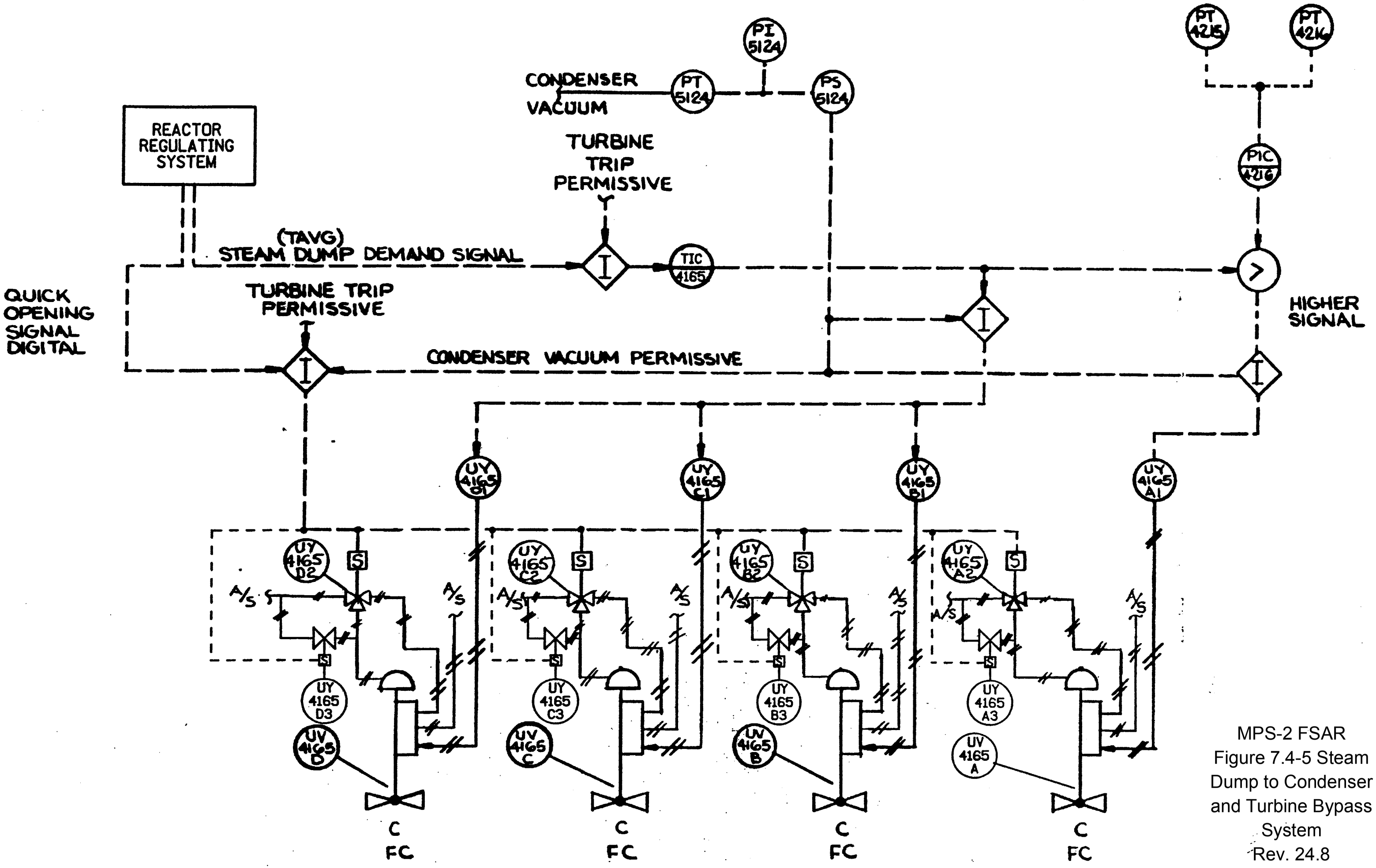
Millstone Nuclear Power Station Unit No. 2	CEA Position Setpoints JUN 1 0 1982	Figure 7.4-2

FIGURE 7.4–3 CEDS FUNCTIONAL BLOCK DIAGRAM

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.4-4 PRESSURE CONTROL PROGRAM





MPS-2 FSAR
Figure 7.4-5 Steam Dump to Condenser and Turbine Bypass System
Rev. 24.8

FIGURE 7.4–6 FEEDWATER CONTROL SYSTEM (SHEET 1)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.4–6 FEEDWATER CONTROL SYSTEM (SHEET 2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.4–6 FEEDWATER CONTROL SYSTEM (SHEET 3)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.4–6 FEEDWATER CONTROL SYSTEM (SHEET 4)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.4–6 FEEDWATER CONTROL SYSTEM (SHEET 5)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.4–6 FEEDWATER CONTROL SYSTEM (SHEET 6)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.4–6 FEEDWATER CONTROL SYSTEM (SHEET 7)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.4–6 FEEDWATER CONTROL SYSTEM (SHEET 8)

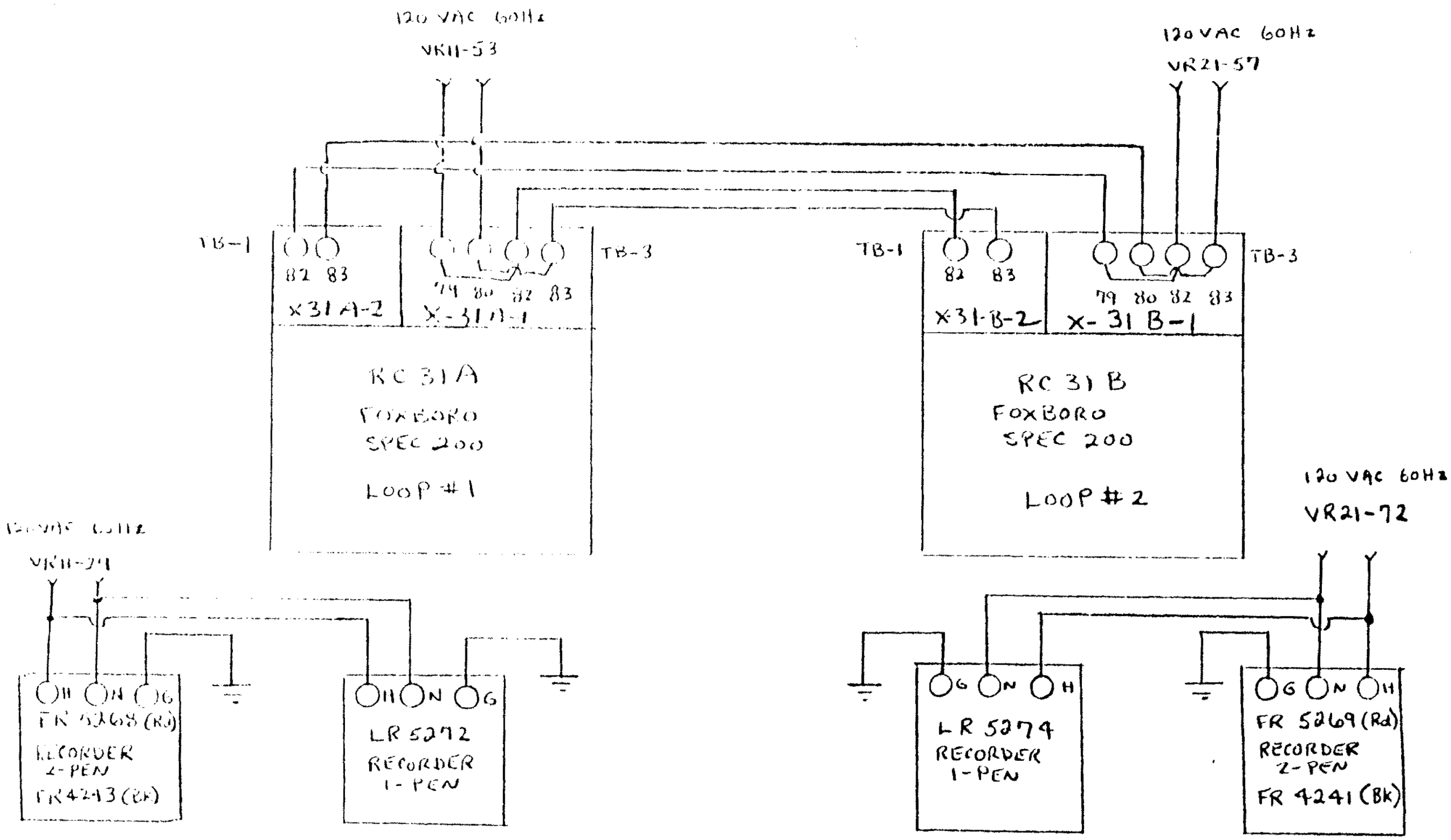
The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.4–6 FEEDWATER CONTROL SYSTEM (SHEET 9)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.4–6 FEEDWATER CONTROL SYSTEM (SHEET 10)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.



MP-2 FEEDWATER REGULATION SYSTEM: POWER DISTRIBUTION

FIGURE 7.4-7

JUN 2 9 1984

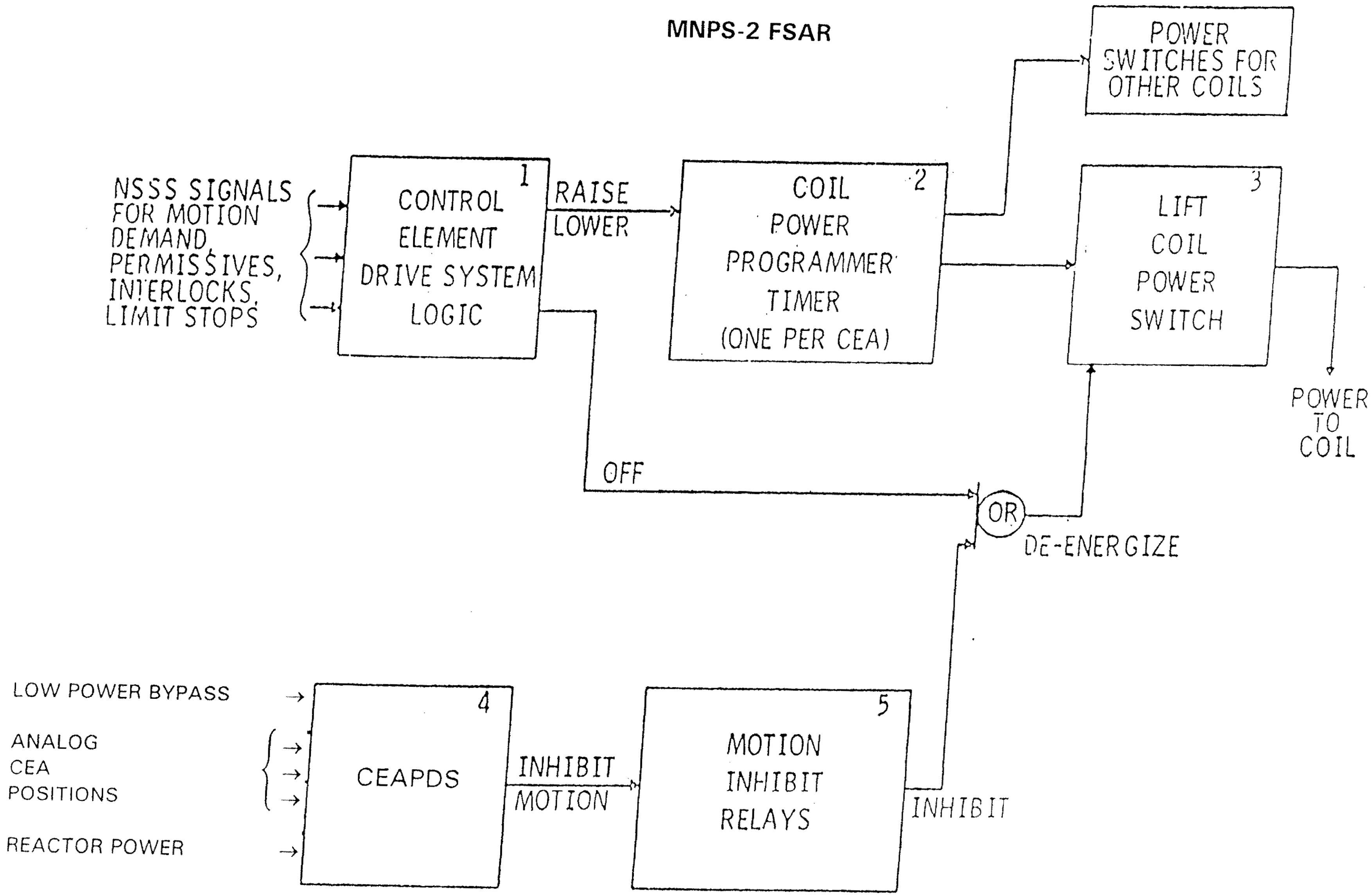


FIGURE 7.4-8
REACTIVITY CONTROL SYSTEM

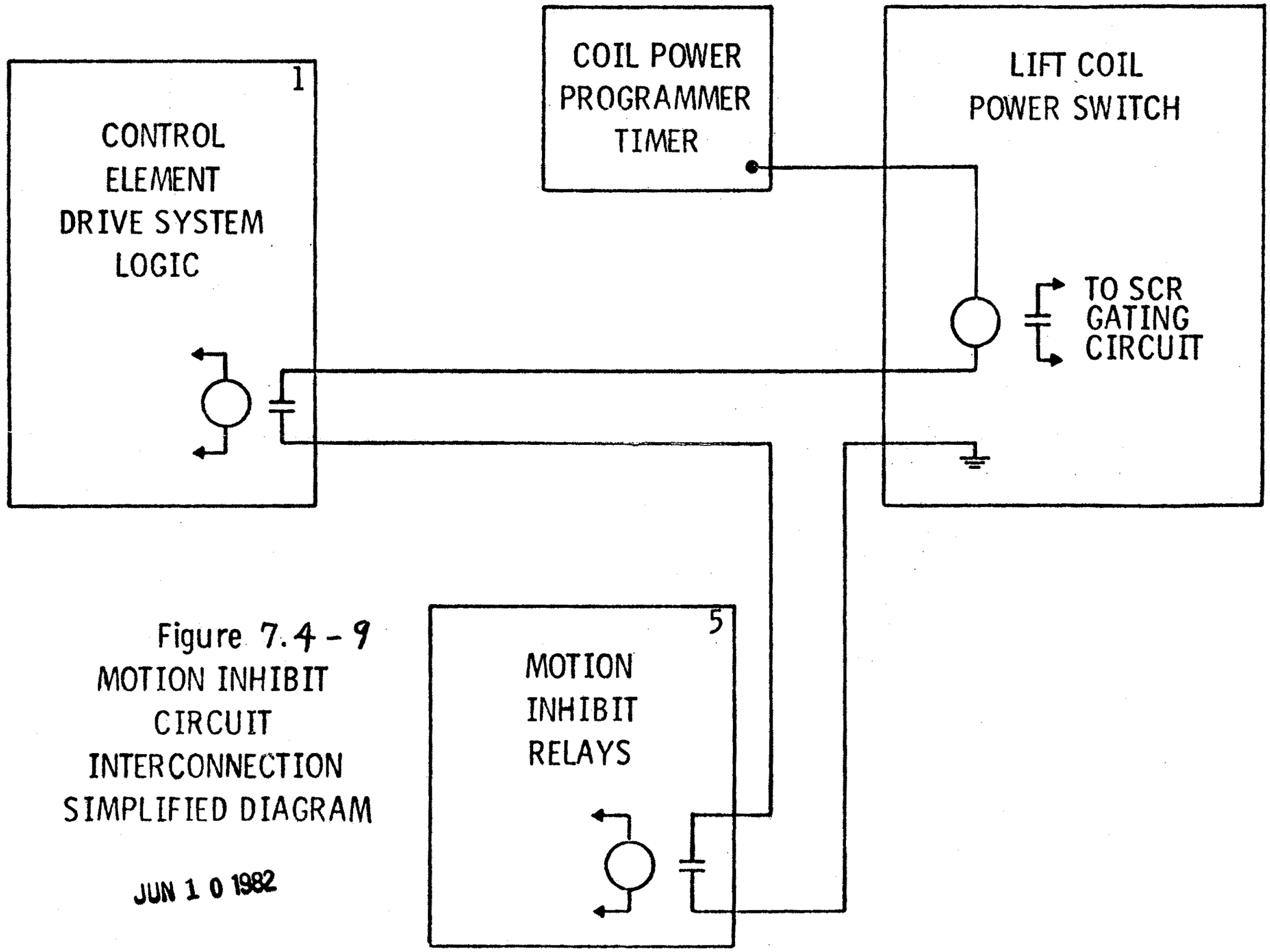


Figure 7.4 - 9
MOTION INHIBIT
CIRCUIT
INTERCONNECTION
SIMPLIFIED DIAGRAM

JUN 1 0 1982

7.5 INSTRUMENTATION SYSTEM

7.5.1 PROCESS INSTRUMENTATION

7.5.1.1 Design Bases

7.5.1.1.1 Functional Requirements

Non-nuclear process instrumentation devices are designed to perform one or more of the following functions:

Measurement

Display

Recording

Set Point Generation

Generation of Corrective Signal

Conditioning of Signal

These devices when connected together in a process instrument loop will monitor and control the process and alert the operator in the event the process variable exceeds beyond the allowable limits.

7.5.1.1.2 Design Criteria

The following criteria have been implemented in the design of the process instrumentation:

The process instrumentation shall measure and control temperature, pressure, flows, and level in all processes as required per 10 CFR 50, Appendix A, Criterion 13.

a. Alternate indicators and controls shall be located at other locations than the control room to allow reactor shutdown should the control room have to be evacuated. This provision is made in accordance with 10 CFR 50, Appendix A, Criterion 19.

b. Independent measurement channels shall be provided to monitor each process parameter required for the reactor protective system (RPS) and the engineered safety features actuation system (ESFAS) to meet the single-failure criterion per IEEE-279-1971.

7.5.1.2 System Description

7.5.1.2.1 System

Process instrumentation provided for reactor coolant temperature, pressure, level, and flow as well as the monitors for containment temperature, pressure, and radiation are described below. Major process instrumentation for the balance of the plant is tabulated in Table 7.5-1.

7.5.1.2.1.1 Reactor Coolant System

a. Temperature

The temperature measurements are made with precision resistance temperature detectors (RTD) which provide a signal to the remote temperature indicating control and safety devices. The following is a brief description of each of the temperature measurement channels.

Hot leg temperature: Each hot leg contains five temperature measurement channels. Four of these channels provide a hot leg temperature signal to the thermal margin/low-pressure (TM/LP) trip circuits. The other hot leg temperature measurement channel provides a signal to the loop T_{avg} computer in the reactor regulating system (RRS), to the Inadequate Core Cooling Monitoring System (ICCMS), and to a recorder. The five hot leg temperatures are indicated on the control panel.

Cold leg temperature: Each cold leg branch contains three temperature measurement channels. Two of the channels in each branch provide a cold leg temperature signal to the TM/LP trip circuits. These channels also provide cold leg temperature indications on the control panel. The third cold leg temperature measurement channel in two branches provides a signal to the loop T_{avg} computers and to the feedwater control system. This channel also provides a high alarm. The third channel in the other two branches provides an input to the ICCMS and Low Temperature Overpressurization Protection (LTOP) circuitry, and is recorded on the control panel.

Loop average temperature: The RRS receives a hot leg and cold leg temperature reading from both loops. The T_{avg} calculation receives hot and cold leg temperatures from Loop 1, Loop 2, or Loops 1 and 2 and provides average temperature outputs to a recorder. The temperature recorder is equipped with two pens. One pen records the average temperature and the other pen records the programmed reference temperature signal (T_{ref}), corresponding to turbine load (first-stage pressure).

b. Pressure

Pressure is measured by electromechanical pressure transmitters. The transmitter produces a DC current output that is proportional to the pressure sensed by the instrument. The DC current outputs are used to provide signals to the remote pressure-indicating, control and safety devices.

The following is a brief description of the pressure measurement channels:

Pressurizer pressure (protective action): Four pressurizer pressure transmitters provide independent, narrow range, pressure signals. These four independent pressure channels provide the signals for the RPS high pressure and TM/LP trips (see Section 7.2). The channels also provide the low-low pressure signal to initiate safety injection (see Section 7.3). All four pressure channels are indicated in the control room and RPS high pressure and TM/LP trip, associated RPS pretrip, and low-low trip alarms are annunciated. Figure 7.5–1 is a functional diagram of one of these channels.

Pressurizer pressure control (control action): Two pressure channels provide narrow-range signals for control of the pressurizer heaters and spray valves as described in Section 7.4. Outputs from the two pressure control channels are recorded in the control room and provide individual high and low alarms.

Pressurizer pressure (Low Range): Two pressure channels provide independent and redundant pressure signals to the shutdown cooling suction isolation valves and the power-operated relief valve (PORV). (See Sections 4.3.8.2.3 and 7.4.8.) These channels also provide an input to control room and hot shutdown panel indicators, computer and one signal to the subcooling margin monitor (see Section 9.3.4.1).

c. Pressurizer Level

Level is sensed by transmitters which measure the pressure difference between a reference column of water and the pressurizer water level. This pressure difference is converted to a DC current signal proportional to the level of water in the pressurizer. The DC current output of the level transmitters provides signals to the remote level indicating and control devices.

Two pressurizer level transmitters provide signals to the chemical and volume control charging and letdown system. In addition, signals are provided for pressurizer heater override control. These level transmitters are calibrated for steam and water densities existing at normal pressurizer operating conditions.

The selected pressurizer level control channel provides a signal for a level recorder in the control room. This recorder is a two pen recorder, with one pen recording actual level as sensed by the level control channel and the other pen recording the programmed level set point signal from the RRS. For additional details see Section 7.4.4.

d. Flow

An indication of reactor coolant flow is obtained by measuring the pressure drop between the hot leg piping and the outlet plenum of each steam generator. The pressure drop is sensed by differential pressure transmitters which convert the pressure difference to direct currents. The direct currents provide a signal to the remote flow indicating and safety devices.

Four independent differential pressure transmitters are provided in each reactor coolant loop to measure the pressure drop across the steam generators. The outputs of corresponding transmitters in each loop are summed by pairs to provide four independent signals representative of flow through the reactor core. These signals are indicated and supplied to the RPS for loss-of-flow determination. The differential pressure sensed by each transmitter is indicated in the control room. For additional details, see Section 7.2.

e. Subcooled Margin

See Section 7.5.4.4 for a description of the ICC System which provides subcooling information.

7.5.1.2.1.2 Containment

Containment temperature monitoring provisions include eight resistance temperature detectors in separate locations for measuring containment air temperatures. These RTDs have resistance-to-current transmitters which provide temperature signals to the plant computer for logging and for readout in the control room through an eight position selector switch. The RTD and cables are specified for continuous operation in the containment. The temperature transmitters are located outside the containment.

The capability of continuously monitoring pressure over the full range of postulated accidents is provided by four safeguards containment pressure transmitters which are located outside the containment. Dual redundant wide range containment pressure transmitters provide pressure monitoring beyond the range of the safeguards containment pressure transmitters. The wide range transmitters were procured and installed to the same criteria as the safeguards transmitters. These transmitters have sufficient range to measure between the normal operating conditions and the predicted peak pressure, due to either a LOCA or SLBA.

Dual-redundant containment level monitors are installed to measure water accumulation in the containment sump. These transmitters can monitor water depth up to seven feet above the sump (approximately $6x10^5$ gallons).

Containment radiation monitoring includes two gaseous and two particulate monitors of the types described in Section 7.5.6.3. The gaseous and particulate monitors are available except when the containment isolation actuation signal (CIAS) automatically closes valves in the sampling system.

This would preclude their use while the containment pressure is elevated during a LOCA or SLBA inside the containment.

Two containment high range area radiation monitors are located in the containment on the outside of the biological shield wall in the vicinity of the east and west electrical penetration areas. Each monitor has a range sufficiently wide to indicate activity levels following a serious accident. Each monitor is qualified to withstand the elevated pressures and temperatures and chemical spray associated with the LOCA or SLBA.

In addition, provisions for obtaining grab samples of containment atmosphere are included in the containment hydrogen monitoring sampling system.

7.5.1.2.2 Components

Standard commercially available instrumentation devices of high, proven quality have been used for all non-nuclear process loops. With the exception of a few multipoint strip chart recorders, all other display instruments are of the miniature type using the 10 to 50 ma, 4 to 20 ma, or 3 to 15 psig signals. All safety related instruments are designed to function in the most severe post-accident environmental condition to which they may be exposed.

7.5.1.3 System Operation

Each instrument loop is designed to display and/or control a certain process. Detailed description of major instrumentation systems are given in Section 7.5.1.2.1 and other appropriate sections where instrumentation is heavily involved in the processes.

In addition to the instrument channels listed in Table 7.5-1, the operator is provided with valve position lights, motor status lights, and alarms for the control of valves, circuit breakers and pumps required for the ESFASs. Additional ammeters are provided for motors of 250 horsepower and above.

7.5.1.4 Post-accident Monitoring

Instrumentation systems are provided for remote monitoring of system conditions during and following an accident to assure adequate public safety. While many parameters are available to the operator in the control room, those which would normally be used by the operator following an accident are those which: (1) aid in determining the nature of the accident, (2) can be used to follow the course of the accident and to aid in predicting its future course, (3) assure that the reactor trip systems (RTS) and engineered safety features (ESF) systems are functioning properly and that the plant is responding properly to these systems, (4) provide information for manual action by the operator.

Table 7.5-3 is a summary of the Regulatory Guide 1.97 (Rev. 2) information on file, that documents which instruments are credited with monitoring the associated variables. These indications of plant variables are required during accident situations to:

a. Provide information to permit the control room operator to take pre-planned action to accomplish safe plant shutdown.

b. Determine whether the RPS, ESF, and other systems important to safety are performing their intended function.

c. Provide information to the operator that will enable them to determine the potential for causing a gross breach of the barriers to radioactivity release, and to determine if a gross breach of a barrier has occurred.

By providing the instruments listed in this table, adequate instrumentation is provided for the operator to determine the nature of an accident, to follow the course of the accident, to determine what action is required to mitigate the consequences of an accident, and to determine the results produced by operator action.

7.5.2 NUCLEAR INSTRUMENTATION

7.5.2.1 Design Bases

The nuclear instrumentation monitors neutron flux over a range greater than ten decades with four independent and redundant channels.

Each of four wide range logarithmic channels provides: a) signals for power level and startup rate indication, b) a signal to the enabling circuitry of the RPS Zero Power Mode Bypass function, and c) a signal to the circuitry in the CEA Position Display System that blocks the CEA motion inhibit signals when reactor power is less than 10 E-4% power.

Four power range channels monitor neutron flux over the power range and provide four redundant, proportional signals to the RPS. These same four power range channels also are used to detect a dropped CEA by monitoring changes in reactor power.

Independent of the four wide range logarithmic channels and the four power range channels, the nuclear instrumentation system also includes two power-range channels that provide signals to the Reactivity Regulating System.

7.5.2.2 Design Criteria

The system is designed in accordance with the criteria of IEEE 279, 1971. In areas not covered or specifically identified by the criteria, the following criteria are used:

a. The nuclear instrumentation sensors are located so as to detect representative core flux conditions;

b. Four independent channels are used in each flux range;

c. The channel ranges overlap sufficiently to ensure that the flux is continually monitored from source range to 125 percent of rated power;

d. Power is supplied to the system from four separate AC buses. Loss of one bus deenergizes one power-range safety channel and one wide range logarithmic channel;

e. Loss of power to channel logic results in a channel trip;

f. All channel outputs are buffered so that accidental connection to 120 volts AC, or to channel supply voltage, or shorting individual outputs has no effect on any of the other outputs.

7.5.2.3 System Description

Ten channels of instrumentation are provided to monitor the neutron flux. The system consists of wide range logarithmic channels, power range safety and power range control channels. Each channel is complete with separate detectors, power supplies, amplifiers, and bistables to provide independent operation. The operating capability of the ten monitoring channels is greater than ten decades of neutron flux and is adequate to monitor the reactor power from shutdown through startup to greater than 125 percent of full power.

Four wide range logarithmic channels monitor the flux from source level to above full power. The flux signals, obtained from fission chambers, are conditioned and amplified in the cable vault amplifier assemblies and then transmitted to the signal processing drawers in the control room. The signal processing drawers further process the detector signal into signals that represent the source range logarithm of count rate and the rate of change of count rate, and the wide range logarithm of reactor power and the rate of change of reactor power. Audible count rate signals are available in the control room.

Four channels are designated as power range safety channels and provide signal outputs to the RPS. These channels operate from 0.1 to 200 percent of full power. Power level signals from these channels are supplied to the protective system. These four channels contain detectors composed of dual section ion chambers which monitor the full axial length of the reactor core at four circumferential positions equally spaced around the core. This arrangement enables detection of power tilts and imbalance.

Two separate power range control channels provide reactor power signals to the RRS cabinets. The channel output is a signal directly proportional to reactor power from 0.1 to 125 percent. The power signal is combined with the average coolant temperature, first stage turbine pressure, and pressurizer pressure signals as the control parameters to the RRS.

The gain of each channel is adjustable to provide a means for calibrating the output against a plant heat balance. Each control channel provides signals to remote indicators and the Anticipated Transients Without Scram (ATWS) Circuitry.

The comparator averager circuit is designed to provide the operator with an alarm upon failure of a linear power range safety channel or subchannel. In addition, this circuit can be used to alarm an off-normal azimuthal flux tilt condition. The comparator average is located in the rear of the RPS cabinet. It receives inputs from each of the four linear power range nuclear instrument channels and setpoints from the power ratio calculator. These setpoints are called High and High-High Deviation and are generated as a function of the average power sensed by all four power range safety channels. They represent a deviation band (absolute magnitude of deviation) that is allowed between the neutron flux measured by each channel and the average of all channels. The capability is provided for the operator to obtain two independently set alarms (High and High-High) representing different absolute magnitudes of deviation, in order to achieve a high degree of system flexibility to conform to changing operating conditions. The comparator averager measures the deviation between each channel and the average of the channels and then compares the measured deviations to the calculated allowable deviations represented by the setpoints. If the measured deviation exceeds the first setpoint, the High alarm is given. If the deviation continues to increase, the second setpoint may be reached, in which case the High-High alarm would be given. A High and High-High deviation alarm is provided for each channel.

Detector Cooling

Forced air cooling is provided for the eight neutron detectors and two spare detector wells which are located within the annular space between the reactor vessel and cavity wall. The remaining two detectors which are embedded in the cavity wall are not subjected to elevated temperatures and therefore do not require cooling.

Cooling air is provided by the containment air recirculation and cooling system (Section 6.5) to limit the maximum detector temperature below 200°F. Cooling air is supplied at 91°F to the base of the detector wells. The air is heated sensibly and discharged at 190°F through openings at the top of the detector wells.

If, however, a loss of air flow occurs, detector temperature would rise from 190°F to 260°F before attaining thermal equilibrium. The out of core detectors are designed to operate satisfactorily at a temperature of 300°F with no appreciable error. Therefore, even with loss of air flow, this value is not exceeded.

7.5.2.4 System Components

7.5.2.4.1 Wide-Range Logarithmic Channel Description

The wide range logarithmic channels are designed to provide to the operator the measure of the neutron flux level at the detector assembly and the measure of the rate of change of neutron flux level from source level (shutdown) to 200% of full power reactor operation.

The signal from the detector is composed of a series of charge pulses. The pulses result from alpha decay of the uranium coating in the detector, from gamma photon interaction with material in the electrodes of the detector, and from the fissioning of uranium atoms when a neutron is absorbed. The pulse signal from alpha decay and from gamma radiation is unwanted signal and can be eliminated by amplitude discrimination because the neutron pulse signal is much larger.

Fission chambers are used to detect the neutron flux because of their proven high reliability in a harsh environment and because of their ability to operate in a high gamma flux without damage and without loss of sensitivity.

The cable and interconnection assemblies between the detector and the amplifier are designed to provide protection against electromagnetic and electrostatic noise interference, and to provide immunity to the postulated environment in the containment during a design basis event (main steam line break or loss of coolant accident).

The amplifier assembly contains the signal conditioning circuitry, DC power supplies, and the high voltage detector excitation supply. The signal processor assembly provides the circuitry to further process the detector signal into signals which are a measure of the logarithm of countrate, the rate of change of countrate, the logarithm of reactor power and the rate of change of reactor power, and it provides outputs for each of these signals. Full scale and bias scaling adjustments to align the outputs to the full reactor power are all made in the signal processor.

The signal processor outputs isolated source range pulses to the audible countrate drawer. The audible countrate drawer provides an audible sound corresponding to the source range pulse outputs. A switch on the front panel allows you to select each of the four channels. Another switch allows you to divide the signal by 1, 10, 100, 1000, or 10000, in order to hear individual tones at any shutdown countrate. There is also a volume control on the front panel for the local speaker as well as an on/off switch and volume control for a remote speaker. The audible countrate drawer is considered non-class 1E, however, it has been qualified not to damage adjacent class 1E equipment during a seismic event.

The number of neutron pulses per unit time from the detector is proportional to the magnitude of the neutron flux at the detector. The magnitude of the neutron flux in the reactor core is proportional to the fission power being generated in the reactor. If the magnitude of the neutron flux at the detector is proportional to the magnitude of the neutron flux in the reactor core, then the pulse rate from the detector is proportional to reactor power.

The neutron flux monitor measures the number of pulses per unit time from the detector over the range from source level to the level where the error from countrate loss due to coincident pulses becomes unacceptable. From about two decades below the upper end of the countrate range to full reactor power, the neutron flux monitor measures the mean square value of the time variant signal from the detector. This mean square value is proportional to the average rate of neutron pulses and is not dependent on the pulses being individually identifiable. It provides good discrimination against alpha and gamma signal.

The derivative of the logarithm of reactor power provides a measurement that is proportional to the change in reactor power per unit time. The signals are displayed on the RATE meters in decades per minute (DPM).

Four bistables are used in each wide range logarithmic channel. One bistable initiates an alarm when either the amplifier or the wide range drawer low voltage power supply is degraded or lost, the magnitude of the high voltage supply in the amplifier is low, or the channel in test (CH IN TEST) switch is in the ON position.

The second bistable is used by the RPS to remove the zero power mode bypass above 10^{-4} percent power. The zero power manually actuated bypass allows CEA drop testing, or CEA withdrawal for other tests during shutdown. The trips bypassed are low flow, Reactor Coolant Pump Underspeed and TM/LP. These trips are automatically reset by the wide range logarithmic channels prior to increasing reactor power to 10^{-4} percent power.

The third bistable is used by the CEA Position Display System to block the CEA Motion Inhibit for Power Dependent Insertion Limit Violation when reactor power is less than 10^{-4}% power. A fourth bistable enables the extended range mode of operation of the wide range Logarithmic Channel at very low power levels.

7.5.2.4.2 Power-Range Safety Channel Description

The four power-range safety channels are capable of measuring flux linearly over the range of 0.1 percent to 200 percent of full power. The detector assembly consists of two uncompensated ion chambers for each channel. One detector extends axially along the lower half of the core while the other, which is located directly above it, monitors flux from the upper half of the core. The upper and lower sections have a total active length of approximately 12 feet. Integral shielded cable is used in the region of high neutron and gamma flux.

The DC current signals from both the upper and lower uncompensated ion chambers are fed directly to the rear of the linear power range monitors located in the RPS cabinets in the control room. The power range monitor houses the electronics that conditions the detector signals for front panel display, RPS inputs and bistable trips. Nuclear Power (upper + lower over 2) and subchannel deviation (upper-lower) signals are generated and output to the Reactor Protection System.

The Nuclear Power signal is also monitored by the rod drop circuit for a fast negative change in amplitude. Time delay and comparator circuits are used to generate a rod drop alarm whenever a negative change in the output signal level occurs in a corresponding period of time. This alarm condition is an indication that a rod had dropped from its proper position. Actuation of this alarm illuminates a LED on the front of the power range safety channel drawer, and actuates an alarm window on the main control board.

Each power range channel contains two bistables. One is used by the RPS to disable the Loss of Turbine and High Local Power Density trips below 15% power. The other bistable initiates an alarm when either the magnitude of the low voltage or high-voltage power supplies is degraded, or the TEST SELECT switch is not in the OFF position and the TEST ENABLE switch is in TEST ENABLE. The condition of each bistable is shown by a front panel LED.

7.5.2.4.3 Power-Range Control Channel Description

The two power range control channels are capable of measuring flux linearly over the range of 0.1 percent to 200 percent of full power. The detector assembly consists of two uncompensated ion chambers for each channel. One detector extends axially along the lower half of the core while the other, which is located directly above it, monitors flux from the upper half of the core. The upper and lower sections have a total active length of approximately 12 feet. Integral shielded cable is used in the region of high neutron and gamma flux. The DC current signals from both the upper and lower uncompensated ion chambers are fed directly to the rear of the linear power range control monitors located in the RRS cabinets in the control room. The power range control channels are connected with the RRS and provide outputs to remote indicators and the ATWS circuitry.

7.5.2.4.4 System Component Location

The nuclear instrumentation safety signal processing equipment is located in the RPS cabinet in the control room. Four cabinets designated as A, B, C and D each house one channel and one wide range logarithmic channel. Mechanical and thermal barriers between the cabinets reduce the possibility of common event failure. The detector cables are routed separately from each other. This includes separation at the containment penetration areas. The location of the neutron detectors is shown in Figure 7.5–2.

The nuclear instrumentation control signal processing equipment is located in the RRS cabinets.

7.5.3 CONTROL ELEMENT ASSEMBLIES POSITION INSTRUMENTATION

7.5.3.1 Design Bases

The principal purpose of the CEA position indication system is to provide the operator with reliable, comprehensible and timely information on CEA position.

7.5.3.2 Design Criteria

a. Position readouts of all CEAs may be obtained;

b. Continuous position readouts of any selected CEA in a group are available;

c. A means of alerting the operator to deviation of CEAs within a group is provided;

d. A permanent record may be made of position of any or all CEAs. The operator may obtain a record at any other desired time;

e. Separate “full-in” and “full-out” indication is provided for each CEA;

f. Redundant and diverse means of indicating CEA position is provided.

7.5.3.3 System Description

Three different display systems of CEA position are provided for the operator on the main control board. The pulse-counting CEA position indication system is one of the outputs of the CEA supervisory function of the plant computer. The reed switch CEA position display system provides continuous indication to the operator of the position of all CEAs in a bar chart form on a video display and numeric information of selected groups. The core mimic display CEA position indicating system provides CEA travel limit information to the operator.

7.5.3.4 System Components

7.5.3.4.1 Pulse-Counting Control Element Assemblies Position Indication System Description

The pulse-counting CEA position indication system infers the position of each CEA maintaining a record of the raise and lower control pulses sent to each magnetic jack mechanism. The plant computer counts these pulses to determine changes in CEA position. The bottom most reed switch is input to the process computer as indication of a fully inserted or dropped rod independent of pulse counts. The resulting inferred position of CEA is available for display on any plant process computer workstation. A printout is available, on operator demand, of the inferred position of all CEAs or of those CEAs within a given group. The plant computer also provides deviation information. If the deviation in position between the highest and the lowest CEA in any group exceeds a preset amount, the computer provides an alarm. The plant computer provides position information for CEA group position alarms.

7.5.3.4.2 Reed Switch Control Element Assemblies Position Display System Description

The reed switch CEA position display system (CEAPDS) utilizes a series of magnetically actuated reed switches, spaced at 1.5 inch intervals along the CEA housing and arranged with precision resistors in a voltage divider network, to provide voltage signals proportional to CEA position. The signals are displayed in bar chart form on a touchscreen monitor on the main control board. The touchscreen allows navigation to various display page information. The display and interlock functions are generated by the CEAPDS software. The distributed control system logic provides redundant CEA malposition indication and alarm functions which are used as input to the CEA motion inhibit circuitry. A CEA position backup readout display is available to provide position of all CEAs.

7.5.3.4.3 Core Mimic Control Element Assemblies Position Indication

A group of 61 light displays, arranged in a shape corresponding to the CEA distribution, is located on the main control board. Each display, which represents one control element drive mechanism (CEDM), contains four colored lights providing the information listed in Table 7.5-7.

7.5.4 IN-CORE INSTRUMENTATION

7.5.4.1 Design Bases

The primary function of the in-core instrumentation is to provide measured data which may be used in evaluating the gross core power distribution in the reactor core as an aid to reactor operations. This data may be used to evaluate thermal margins and to estimate local fuel burnup. No credit is taken for this system in the accident analysis of Chapter 14. The in-core detectors will be used to periodically calibrate the out-of-core detectors as defined in the Technical Specifications.

7.5.4.2 Design Criteria

a. Detector assemblies are installed in the reactor core at selected locations to measure core neutron flux and coolant temperature information during reactor operation in the power range;

b. Flux detectors of the self-powered type, with proven capabilities for in-core service, are used;

c. The information obtained from the detector assemblies may be used for fuel management purposes and to assess the core performance. It will not be used for automatic protective or control functions;

d. The output signal of the flux detectors will be adjusted for changes in sensitivity due to emitter material burnup and for undesirable background signals;

e. Each detector assembly is comprised of four local neutron flux detectors stacked vertically for axial monitoring, and one thermocouple at the assembly outlet;

f. Axial spacing of the detectors in each assembly and radial spacing of the assemblies permit an evaluation of the gross core power distribution through the use of In-Core Analysis Computer Program.

g. In accordance with the guidance of NUREG-0737 and Regulatory Guide 1.97 the Core Exit Thermocouples from each train are distributed such that all four quadrants of the reactor vessel can be monitored during and after a design basis accident following the loss of a single train.

7.5.4.3 In-Core Instrumentation System Description

The in-core instrumentation (ICI) system consists of 43 fixed in-core detector assemblies inserted into selected fuel assemblies. Each assembly contains four 40 cm long rhodium detectors, and one Cr-Al thermocouple. The detector assembly is illustrated in Figure 7.5–3. Outputs are fed to the plant computer in the control room for processing and logging.

Assemblies are inserted into the core through instrumentation nozzles in the top closure head of the reactor vessel. Each assembly is guided into position in the center of the fuel assembly via a fixed guide tube and instrument thimble assembly. A Grayloc seal forms a pressure boundary for each assembly at the instrument nozzle. The locations of the in-core detectors in the core are shown in Figure 7.5–4.

The neutron detectors produce a current proportional to neutron flux by a neutron-beta reaction in the detector wire. The emitter, which is the central conductor in the coaxial detector, is made of rhodium and has a high thermal neutron capture cross section. The useful life of the rhodium detectors is expected to be about three years at full power, after which the detector assemblies will be replaced by new units.

The data from the detectors are read out by the plant computer which scans all assemblies, processes and prints out the data periodically or on demand. The computer periodically computes integrated flux at each detector to update detector sensitivity factors to compensate for detector burnout.

7.5.4.4 Inadequate Core Cooling (ICC) System Description

The Inadequate Core Cooling Monitoring System (ICCMS) integrates the processing and display of:

1. Subcooled/Superheat

2. Core Exit Thermocouples (CETs)

   Note: The CETs are part of the ICIs.

3. Reactor Vessel Level Monitoring System (RVLMS) Heated Junction Thermocouple (HJTC) System for inventory tracking.

The information provided by this system allows the plant operators to monitor reactor coolant status during normal and abnormal plant conditions. The operator uses this information to take corrective action as needed and/or confirm that actions taken produce the desired result. Thus, the approach to, existence of, and recovery from ICC conditions can be monitored consistent with the provisions of NUREG-0737, Section II.F.2. The Millstone Unit Number 2 ICC system is designed as Category I (Class 1E) with redundant trains (train A and train B). Each train contains stand-alone processing electronics and displays, which monitor and alarm ICC as shown in Figure 7.5–7.

The ICCMS CET channel assignments are shown on Figure 7.5–4. The core exit thermocouples from each train are distributed such that all four quadrants of the reactor core can be monitored during and after an accident following loss of a single train. Data received from both channels of ICCMS are combined and displayed on the non-Class 1E plant process computer (PPC).

Subcooled/Superheat calculations use reactor coolant system (RCS) temperatures and pressures to calculate the degree of subcooling or superheat in the reactor coolant either in terms of temperature or pressure. The calculations are based upon the most conservative input temperature and pressures.

Note: Temperatures are input from the CETs, Unheated Thermocouples (UHTCs), and RCS hot and cold leg RTDs.

CETs, HJTCs, and UHTCs are provided with required reference junction temperature thermocouple compensation. All CETs, HJTCs, and UHTCs can be displayed on a touchscreen graphic local display unit and the PPC.

The RVLM module monitors coolant inventory in the region above the core. Redundant strings of HJTCs are arranged in the reactor vessel head area to provide indication of conditions at eight discrete levels. The system is a two-channel system each consisting of a string of eight sensors. The detector assembly is illustrated in Figure 7.5–6.

The primary means of displaying the ICC information is provided via the non-Class 1E plant process computer (PPC). The PPC receives ICC data transmitted via optical isolation provided by the ICCMS.

The ICCMS local display provides backup IE display of ICC information. Each ICCMS cabinet (train A and B), which is located adjacent to the Control Room, has a qualified class 1E touchscreen graphic display that provides the following ICC information.

1. Subcooled/superheat in °F (300°F subcooling to 45°F superheat).*

2. CETs (200°F to 2300°F).

3. Percentage level (0 to 100%) in the vessel above the core.

* Minimum displayed range

This ICC information is provided on the local ICCMS displays by means of display pages which are selected via touchscreen. The hierarchy of the display pages is shown in Figure 7.5–9.

Alarms are provided on main control boards from the ICC cabinets. There are four alarms: Saturation Trouble alarm, CET High alarm, Reactor Level Low alarm, and ICCMS Trouble alarm. Alarm status touchscreen buttons also are provided on all ICCMS display pages.

7.5.4.4.1 Inadequate Core Cooling System Software

The objective of the ICC monitor is to provide the operator with a simple indication of core cooling conditions. This objective is achieved by monitoring a number of reactor system parameters, performing certain calculations using these parameters in a digital computer, and providing the operator with the results of these calculations in real time. This section outlines the computer program that performs the required calculations.

Input Variables

The measured quantities used as input variables for the ICC monitor are RCS pressure, RCS hot and cold leg temperatures, CETs, HJTCs and UHTCs.

Calculations

A number of different calculations are performed using the input variables to generate output information for the operator's use. These calculations are characterized below:

1. All input values are converted to engineering units; °F for temperatures and psia for pressures. All engineering unit values are checked for sensor range limits and quality tags assigned before being used in calculations. CET, HJTC, and UHTC thermocouple processing, including reference junction temperature compensation, is also performed.

2. The HJTC Processing module calculates reactor vessel level based on eight incremental sensor pair locations, with each pair having heated and unheated thermocouples. A Reactor Level Low alarm will be generated if the calculated level exceeds the setpoint. HJTC heater control also is performed. Power to each of the eight heaters is individually controlled (on/off).

   The calculation includes provisions for the bypass of failed sensors and "substitution" of available, valid sensors for determining reactor level.

3. The CET processing module validates CET inputs used in the calculations and calculates the highest and second highest CET temperature (each quadrant and overall) from the validated set. This information is displayed on the applicable pages and is used in the Saturation Margin module. A CET High alarm is generated if the second highest (overall) CET temperature exceeds the setpoint.

4. The saturation margin module performs a number of calculations using all available, valid temperature and pressure inputs. The saturation temperature is calculated and then four temperature saturation margins are calculated. A saturation pressure margin is then calculated, and a Saturation Trouble alarm is generated if the saturation temperature margin exceeds the specified margin setpoint. (Adapted versions of the subroutines built by McClintock and Silverstri for the American Society of Mechanical Engineers (ASME) steam table are used for saturation calculations.)

5. Other calculations are performed which support various maintenance, diagnostic and test features of the ICCMS.

System Operation

The ICC monitor, once the initial startup is accomplished, is self sustaining. All program operations, from input through calculations to output, including periodic testing, are performed sequentially. Analog input points may be bypassed via touchscreen on the Signal and HJTC display pages, provided the Test Mode/Bypass keyswitch is enabled.

System Diagnostics

An online diagnostic function performs mathematical and Boolean logic checks and monitors the ICCMS application tasks for proper execution sequence. Resulting errors are logged, and a warm restart and ICCMS Trouble alarm will result if these checks fail.

Other on-line diagnostics include Input Processing range and quality checks, Plant Process computer and Remote Terminal data link health, cabinet temperature monitoring and Uniform Temperature Reference monitoring. Errors of this type are logged to the System Errors page, and an ICCMS Trouble alarm initiated. Cabinet temperature and reference temperature monitoring is performed as apart of the normal scan/alarm processing of the ICCMS.

System and Handler Software

C-based software performs the necessary ICC functions of data validation and conversion, selection of limiting pressures and temperatures, calculation of temperature and pressure margins, maintaining tabular and formatted data files, maintaining tabular and formatted system status files, and controlling communications.

System software is based on a real time, multitasking operating system. Applications programming uses the C language and utilities.

The applications program has two major divisions: one handling process calculations and the other handling input/output operations. Refer to Figure 7.5–8 for the following:

Data Acquisition

Data acquisition is performed by the Input Processing module. It provides the data validation process. The data is checked for bypassed, open, shorted, or out of range conditions. A different error code is stored in an array for each input depending upon the error. The data then is converted to engineering units according to standard temperature (or pressure) conversions.

Process Calculations

The process calculations include CET, HJTC, HJTC Heater Control, and Saturation Margin modules.

Communications

The Output Data Services module handles communication tasks with the PPC, the display units, and a maintenance diagnostic terminal connection.

Data is "broadcast" (one way) to the PPC via fiber optic modem approximately every 3 seconds. No "handshaking," either hardware or software, occurs. This eliminates the potential for unacceptable interaction between the 1E ICCMS and the non-Class 1E PPC. All acquired and calculated data are transmitted.

The ICCMS display communicates with the ICCMS processor via a serial connection. Communication is two way; i.e., both the ICCMS processor and display send and receive data interactively. The display and processor are both class 1E.

A maintenance diagnostic terminal serial connection also is provided. Specific file data are provided in response to a specific file request by a requesting terminal. The terminal is connected only during maintenance.

7.5.5 PLANT COMPUTER SYSTEM

7.5.5.1 Summary Description

The plant computer is an online digital computer designed to perform a variety of data acquisition and information processing output functions. The computer presents information to the plant operational and technical personnel on printers, main control board color monitors, man-machine interface workstations, and analog trend recorders.

h. In fulfilling the overall data acquisition requirements of the computer, selected process instrumentation inputs are used.

i. In addition to the output of operational and technical information, the computer operates digital output latched contacts, some of which are used for control and supervision of CEAs.

7.5.5.2 Functional Objectives

The objectives of the plant computer system are as follows:

a. To assist the plant operational and technical personnel in monitoring the performance and operation of the plant equipment;

b. To provide for the acquisition and logging of process data which are available during unusual plant conditions;

c. To monitor all CEA movements and to control CEA group sequencing in the automatic and manual sequential modes and display CEA position;

d. To assist plant personnel in the analysis of reactor core operation during both steady state and transient conditions;

e. To assist plant personnel in performing certain periodic surveillance tests;

f. To relieve the plant operational personnel of routine plant logging functions.

7.5.5.3 System Description

The major equipment associated with the plant computer system is located in the computer room. This equipment consists of:

a. Processors, in a redundant host configuration.

b. Disk drive cabinets (Storage Area Network (SAN)).

c. Cabinets for process input/output interface equipment, consisting of digital input, digital output, analog input and analog output cards.

d. Cabinets for peripheral and communication equipment.

e. Network hubs for connections between the processor, MMI workstations, printers and digital plant equipment.

Equipment is also available in the computer room for specific reactor engineering and plant performance applications. This equipment includes MMI workstations and printers. A firewall exists between the Millstone Unit Number 2 PPC network and Millstone Site Data Network (MSDN) which isolates traffic and provides access to PPC data on the MSDN.

Output information from the computer is displayed in the main control room for use by the plant operating personnel. The central section of the main control board is designed to provide the operator with overall surveillance of all computer initiated data displays. Two two-pen analog trend recorders and three color monitors are provided on the vertical panels of this section.

A minimum of four MMI workstations are located in the Main Control Room and serve as the man/machine interface for the plant operators. These workstations provide the plant operators with access to numerous computer system data display functions.

Hardcopy printout capability is provided in the Main Control Room, readily accessible to plant operators. Print output is provided to more than one printer such that the capability to produce important reports is not lost if a single printer fails. Most reports can also be directed to the requesting workstation for local file viewing in lieu of printed output. Simulated printer output is also available at control room workstations for alarm messages.

The power to operate the computer equipment is normally supplied by a 75 kVA three phase inverter. The inverter is normally supplied by a 480 volt station service AC power source through a Kirk-Key interlock scheme which allows for alignment to either emergency diesel generator. In the event of a loss of station service power, the inverter is automatically supplied by a battery. The battery, which has a 90 minute computer full load capacity, is separate from the station service batteries and is used only to supply the computer system loads. A second station service 480-120/208Y regulated supply provides for computer system loads during periods of inverter, battery charger or rectifier maintenance.

7.5.5.4 Functional Description

Functional objectives of the computer system are accomplished automatically, via computer programming and, on demand, via request through MMI workstations. Four basic types of functions are possible via the MMI workstations:

a. Printout request functions

b. Trend-display control functions

c. Data entry functions

d. Special operation functions

Printout requests enable the plant operator to demand a variety of printed logs such as a core map of CET temperatures, a post-trip review log, or a sequence of events log. Trend display control functions give the plant operator the ability to select and control the inputs to the analog trend recorders, and MMI workstations. Preset groupings of analog inputs or calculated values may be trended at interval multiples of one second using real-time data. Data entry functions give the operator the ability to modify stored data such as constants or assign temporary process alarm limit setpoints.

Computer functions which are performed automatically are as follows:

a. Analog and digital scanning

b. Sequence of events status checking and message logging

c. Alarming process out of limits and alarm message logging

d. Analog input averaging and engineering unit conversion

e. Selected nuclear and plant performance calculations

f. Video alarm message display

g. Video point display

h. CEA position pulse counting

i. Selected rod position digital display

j. Pre-post-trip review logging

k. Hourly plant data logging

l. Process variable pulse counting

m. CEA procedural limit alarming

n. CEA sequencing supervision

o. Hourly excore detector readout data logging

p. Daily excore detector readout data logging

q. Daily reactor coolant system (RCS) leakage data logging

r. Daily pump/fan motor run times

s. Updating incore correction factors

t. Incore detector reasonability checks

u. SPDS Screen (Section 7.5.5.5)

The following functions are automatic and are initiated by operator or reactor engineer MMI workstation requests.

a. Process value digital video display

b. Analog pen-recorder trending

c. Archive recording

d. Certain periodic surveillance test logging

The foreground/background capability of the redundant processor pair will permit the use of the idle computers for online modification and/or addition of programs. This will increase system availability by greatly reducing the need to take the computer out of service for program maintenance.

RCS mid-loop level hot leg number 1 (L-112) and hot leg number 2 (L-122) are displayed on the process computer.

7.5.5.5 Safety Parameter Display System

The Millstone 2 Safety Parameter Display System (SPDS) is a computer based system that serves as an aid to the operating crew during normal and emergency operating conditions. The SPDS design addresses the guidelines of Supplement 1 to NUREG-0737, “Requirements for Emergency Response Capability” and is based on the Emergency Operating Procedure (EOP) Safety Function Status Checks (SFSCs). It is not a Category 1 system and is not a replacement or substitute for safety instrumentation.

The primary purpose of the SPDS is to help the operating crew monitor the safety status of the plant. It does this by providing a concentrated and organized source of plant data that allows the users to compare existing plant conditions with the safety function limits listed in the EOPs to determine if the limits are violated. The safety functions monitored by the SFSCs include Reactivity Control, Vital Auxiliaries, Reactor Coolant System (RCS) Inventory Control, RCS Pressure Control, Core Heat Removal, RCS Heat Removal, Containment Isolation, and Containment Temperature and Pressure Control. The SPDS performs signal validation on redundant sensor data and displays quality tags for sensor data and calculated values.

The SPDS displays are divided into three-tier hierarchies that readily support the evaluation of the SFSCs. The three-tier hierarchies consist of a single Overview Display (top level), one or more displays for each safety function (mid-level), and one or more sensor data displays for each safety function display (data level). In addition to the hierarchical displays, displays of trends and graphs that support the evaluation of the safety status of the plant are accessible from menus and/or command buttons.

One dedicated display terminal in the control room is designated to continuously display the Overview display of critical plant variables. The SPDS Overview provides a continuous concise display of the critical plant parameters and the parameter values are color coded. It has EOP specific acceptance criteria on the parameters to alert the operators that a safety function may not be met. The parameter values change color and blink when they exceed the EOP acceptance criteria. Selection of specific EOP zzcan be achieved manually by authorized operators and setpoints for each safety function parameters will automatically change according to the selected EOP. Colored boxes at the bottom of the display will indicate EOP Critical Safety Function (CSF) status, (green - limits not exceeded, red - limits exceeded) for specific EOP selected (with the exception being EOP 2525). Upon a reactor trip, the Overview is automatically displayed with EOP 2525’s, Standard Post Trip Action, setpoints.

SPDS displays can be accessed through the site’s computer network at various locations. Personnel outside of the control room cannot influence the analysis performed by the SPDS, that is, they cannot enter manual inputs, change setpoints, or change the EOP/mode that has been selected from within the control room.

7.5.6 RADIOACTIVITY MONITORING

7.5.6.1 Area Radiation Monitoring System

7.5.6.1.1 Design Bases

7.5.6.1.1.1 Functional Requirements

The basic purpose of the area radiation monitoring equipment is to provide information for the protection of plant personnel from radiation.

7.5.6.1.1.2 Design Criteria

The area radiation monitors are sensitive to gamma radiation. Audible radiation alarm signals are actuated on the main control board and at the detector location whenever an abnormal increase in radiation levels occur. Exception: The Containment High Range Radiation Monitors, RM-8240 and 8241, do not have audible alarms at the detector locations. The alarm setpoint for each area radiation monitor is variable and is set at a level sufficiently above the normal ambient background radiation level in the respective area to avoid spurious alarming. Table 7.5-4 shows the range, for each area monitor.

As a minimum, the area radiation monitors have a dynamic range of at least one decade above the normal expected level.

7.5.6.1.2 System Description

7.5.6.1.2.1 Location

Area radiation monitors are located as follows:

- a. Containment personnel access hatch (elevation 38 feet, six inches)
- b. Containment refueling machine service platform (elevation 38 feet, six inches)
- c. Auxiliary building drumming and decontamination area (elevation 14 feet, six inches)
- d. Control room (elevation 38 feet, six inches)
- e. Auxiliary building - charging pump area (elevation 25 feet, six inches)
- f. Sampling room - auxiliary building (elevation (+) 14 feet, six inches)
- g. Radioactive waste processing area (elevation (-) 25 feet, six inches)
- h. Radioactive waste processing area (elevation (-) 45 feet, six inches) (2)

i. Spent fuel pool area (elevation 38 feet, six inches) (4)

j. Containment high range (elevation 14 feet, six inches) (2)

7.5.6.1.2.2 Components

Each area radiation monitoring system consists of a gamma sensitive detector located in the area to be monitored. Most also consist of a local indicator/alarm module, located in or adjacent to the area being monitored (except the containment high range) and a remote readout module in the control room.

7.5.6.1.3 System Operation

Area radiation monitors are calibrated on a periodic basis.

Calibration of the area radiation monitor is accomplished by placing the detector assembly in a reproducible, fixed geometry configuration and exposing the detector to a calibrated radioactive source.

Power for the area radiation monitoring system is provided by either the 120 volt regulated or the vital AC distribution system.

A "loss of power/channel failure" and/or high radiation is monitored for each area radiation monitoring channel, with common annunciation in the control room.

Channel performance and test is available to the operator. An electronic signal is used to verify the performance of the control readout instrumentation.

Alarm settings are normally based on desired level above background.

Two containment high-range area radiation monitors are located in the containment on the outside of the biological shield wall in the vicinity of the electrical penetration area, elevation 14 feet, six inches. Each monitor has a range sufficiently wide to indicate activity levels following a serious accident. Each monitor is qualified to withstand the elevated temperatures and pressures and chemical spray associated with the LOCA or SLBA. Both monitors provide continuous indication and both channels provide signals for recording. Local indication and alarm functions are not included for these monitors.

In the event of high containment radiation, the high range monitors will actuate the automatic closure of the containment hydrogen purge isolation valves.

Four area monitors in the Spent Fuel Pool area furnish a -5 to -1 volt output signal to the engineered safeguards system to initiate operation of the auxiliary exhaust actuation system (AEAS).

7.5.6.1.4 Availability and Reliability

Area radiation monitoring equipment is designed for continuous operation during the 40 year life of the plant. High quality, commercial grade components are used throughout the system. Servicing of the electrical system is simplified by the use of plug-in components.

7.5.6.2 Liquid Radiation Monitoring System

7.5.6.2.1 Design Bases

7.5.6.2.1.1 Functional Requirement

Liquid radiation monitors are used with alarms and indicators to give warning of abnormal radioactivity in process piping and to prevent releases in excess of allowable limits.

In conjunction with control circuit components, liquid radiation monitors initiate valve action to stop the release of liquid waste, upon detection of high radioactivity.

7.5.6.2.1.2 Design Criteria

Liquid radiation monitors are designed to detect radioactivity in process lines.

Liquid radiation monitors are also designed to detect radioactivity in liquids prior to their release to the environment. In conjunction with control circuit components, selected radiation monitors initiate closure of valves to prevent the release of radioactivity to uncontrolled areas in excess of allowable limits.

7.5.6.2.2 System Description

7.5.6.2.2.1 Systems

Liquid radiation monitor systems are provided for the following:

a. Steam generator blowdown sample

b. Reactor building closed cooling water (RBCCW)

c. Clean liquid waste discharge

d. Aerated liquid waste discharge

e. Condensate recovery tank

f. Deleted

g. Condensate polishing facility - waste neutralization sump discharge

7.5.6.2.2.2 Components

The components used for each liquid monitor are as follows:

1. A liquid sample chamber.

2. A gamma sensitive scintillation detector positioned on the outside surface of the sample chamber.

3. Lead shielding for the sample chamber and detector to reduce the effect of background sources of radioactivity.

4. Local indication with audio and visual alarm.

5. Control room indication and alarm for the monitors. (Control room alarm annunciator only for condensate polishing facility-waste neutralization sump discharge monitor.)

6. Pen recorders for continuous records.

7.5.6.2.2.2.1 Steam-Generator Blowdown Sample Monitor

The steam-generator blowdown sample monitor supplies high radiation/instrument failure signal to the control logic to initiate closure of: blowdown line isolation valves from steam generators 1 and 2, blowdown quench tank discharge valve, blowdown tank discharge valve, and steam generators 1 and 2 sample valves. It serves as the final effluent monitor for liquid blowdown releases. It may provide indication of gross primary to secondary leakage.

7.5.6.2.2.2.2 Reactor Building Closed Cooling Water Monitor

The RBCCW system continuously circulates water in a closed loop which could become radioactive in the event of inleakage from the components handling radioactive materials. The discharge from the RBCCW pump is continuously monitored for high radioactivity. High activity or instrument failure is alarmed.

7.5.6.2.2.2.3 Clean Liquid Waste Discharge Monitor

This monitor is provided to measure radioactivity in clean liquid waste being discharged to the circulating water. It is a final effluent monitor. High radioactivity, low sample flow, or instrument failure will initiate closure of valves in the discharge line and will actuate alarms.

7.5.6.2.2.2.4 Aerated Liquid Waste Discharge Monitor

This monitor measures the radioactivity in the aerated waste discharge line. It is a final effluent monitor. High radioactivity, low sample flow, or instrument failure will initiate closure of valves in the discharge line and actuate alarms.

7.5.6.2.2.2.5 Condensate Recovery Tank Monitor

This monitor looks for the presence of radioactivity in the auxiliary condensate system by monitoring the discharge from the condensate recovery tank. On a high radiation or instrument failure alarm it will automatically terminate flow from the condensate recovery tank.

7.5.6.2.2.2.6 Failed Fuel Monitor

Deleted

7.5.6.2.2.2.7 Condensate Polishing Facility – Waste Neutralization Sump Discharge

This monitor measures the radioactivity in the effluent from the Condensate Polishing Facility - Waste Neutralization Sump. It is a final effluent monitor. High or low flow is alarmed. It automatically isolates discharge upon a high radiation or instrument failure alarm.

7.5.6.2.3 System Operation

Liquid radiation monitors will be checked and calibrated on a periodic schedule using radioactive sources.

7.5.6.2.4 Availability and Reliability

Liquid radiation monitoring equipment is designed for operation over long periods with minimum of service. Spare sample chambers are available to replace contaminated chambers.

High-quality, commercial-grade components are used throughout the system. Servicing of the electrical system is simplified by the use of plug-in components.

7.5.6.3 Airborne and Steam Radioactivity Monitoring

7.5.6.3.1 Design Basis

7.5.6.3.1.1 Functional Requirements

An airborne (including steam) radioactivity monitoring system is provided to detect and measure the levels of airborne radioactivity at various locations, both within the plant and in plant effluents, to satisfy the requirements of 10 CFR, Part 20 including Sections 1301 and 1302 and Appendix B, Table 2, and Part 50, including the general design criteria (GDC) of Appendix A.

The system will indicate the plant areas in which increases in radioactivity have occurred so that the cause can be determined and corrected to ensure on site and off site safety during all plant operations.

Instrumentation is also included to provide a record of the level of radioactivity for the points monitored. (Except Control Room Inlet Monitors)

7.5.6.3.1.2 Design Criteria

The criteria which determines the type of airborne radiation monitoring equipment selected for each application is as follows:

1. The nature of the radioactive release.

2. The sensitivity and range requirements to monitor normal operating or potential accident radioactive levels.

3. The response requirements to alert and warn personnel of the radioactive hazard so that protective measures may be taken.

7.5.6.3.2 System Description

7.5.6.3.2.1 System

The location and types of radiation monitors provided are listed in Tables 7.5-6 and 7.5-6A and described below.

7.5.6.3.2.1.1 Unit 2 Stack Gaseous and Particulate Monitoring

A representative sample is continuously extracted from the Unit 2 stack by one of three isokinetic nozzles. Selection of sample nozzle is automatic, depending on the stack flow rate, as determined by the number of fans in operation.

The air extracted from the stack is directed through an off line particulate and gaseous monitoring systems with particulate and iodine grab samples for laboratory analysis.

A beta scintillation detector placed in the gaseous sample chamber detects and measures the radioactivity present in the sample volume passing through the chamber. A beta scintillation detector measures the radioactivity of the particulate filter.

A second monitoring assembly located at elevation 31 feet, six inches in the switchgear room in the turbine building is designed to measure high range post-accident gaseous releases. Additionally, this equipment will sample for particulates and iodine. A required sample flow is used to minimize particulate and iodine buildup. This flow is monitored in a gas sample chamber using two geiger-mueller detectors. Three separate particulate/iodine assemblies are monitored by a geiger-mueller detector for personnel protection and to ensure capability of laboratory counting. The cartridge assemblies sample in a sequential manner such that when one collects the maximum amount of radioactivity specified, the flow path is then switched to the second cartridge assembly. The cartridge assemblies are removed for laboratory analysis.

The high range effluent monitor has both local and remote (control room) indication and control. A recorder in the control room continuously records high range effluent radiation levels.

The high range effluent monitor is electrically interconnected to the normal range vent-stack monitor such that a high range release will put the normal range monitor into a purge mode for the duration of the high range release.

7.5.6.3.2.1.2 Containment Gaseous and Particulate Monitoring

Two redundant off line Seismic Class I particulate and gaseous monitoring and halogen sampling systems are used to monitor the containment atmosphere by extracting representative samples continuously from the containment atmosphere. Each redundant system consists of a particulate detector and a gaseous detector, which share some common components such as power supplies and microprocessors. The sample first passes through a fixed particulate filter, then passes through an iodine filter cartridge and then into a gas chamber before being exhausted back into the containment atmosphere. The iodine cartridge is removable for periodic laboratory analysis.

The redundant detection systems provide inputs to ESFAS from a total of four detectors, two particulate and two gaseous. High radiation and instrument failure alarm signals from any one of the four detectors will cause the ESFAS to initiate the containment purge isolation.

The containment isolation actuation signal (CIAS) closes the containment isolation valves for the sampling system, trips the sample fan and also closes local sample line isolation valves at each containment gaseous and particulate radiation monitor. The closed monitor sample line isolation valves prevent elevated pressure associated with occurrence of a LOCA or SLBA from affecting the containment gaseous and particulate radiation monitors once the containment isolation valves are reopened. Reopening the containment isolation valves under this condition will permit post-accident monitoring using the hydrogen analyzer and PASS which share the same sample lines with the containment gaseous and particulate radiation monitors.

Under post-accident conditions with the containment isolation valves reopened, the radiation monitor sample line piping system extending beyond the outermost containment isolation valve to and including the local monitor sample line isolation valves are considered part of an extended containment pressure boundary. The ANSI B31.1 (1967) piping system and associated valves which comprise this section of monitor sample line system are subject to testing and surveillance requirements typically applicable to ANSI B31.7 (1969) Class 2 piping systems and valve components. The local monitor sample line isolation valves do not provide a containment isolation function.

7.5.6.3.2.1.3 Radioactive Waste Ventilation Monitor

The radioactive waste airborne radioactivity monitoring system consists of four radiation monitors, each capable of continuously monitoring airborne radioactive particulates. One of the four also monitors for gaseous activity. In addition, each monitor contains a replaceable charcoal cartridge mounted in series and downstream of the particulate filter. The charcoal cartridge may be removed periodically for laboratory analysis of iodine activity.

The combined exhaust from the radioactive waste areas will be vented to the Unit Number 2 stack and will be monitored by the stack monitoring system (See Section 7.5.6.3.2.1.1).

7.5.6.3.2.1.4 Condenser Air Ejector Gaseous Monitor

The condenser air ejector system extracts the non-condensable gases from the condenser and exhausts to the Millstone stack. The presence of radioactivity in this line would indicate a primary-to-secondary tube leak in the steam generators. An adjacent-to-line gaseous monitoring system is provided to continuously monitor the air ejector vent line for the presence of radioactivity in the non-condensable gases. A high radiation or instrument failure alarm on this monitor will automatically isolate steam generator blowdown and effluents. This monitor may also provide indication of gross primary to secondary leakage and serves as a backup to the steam generator blowdown sample monitor.

7.5.6.3.2.1.5 Control Room Ventilation Gaseous Monitor

A fixed, GM, gaseous monitor is provided to continually monitor the control room atmosphere.

7.5.6.3.2.1.6 Filtered Waste Gas Monitor

When waste gases are to be released, the allowable radioactivity level of the release is established. An off line gaseous monitor is provided to monitor and record the radioactivity level of these releases. Should releases exceed the established level, the monitor will provide an alarm in the control room and initiate closure of valves (instrument failure alarm will also initiate valve closure) in the filtered waste gas discharge line.

7.5.6.3.2.1.7 Fuel Handling Exhaust Air Gaseous and Particulate Monitor

An off line airborne particulate and gaseous monitoring system is used to monitor the spent fuel pool ventilation exhaust air system.

The particulate and gaseous channels are monitored continuously. The monitor provides a charcoal sample for laboratory analysis of the iodine content.

7.5.6.3.2.1.8 Main Steam Line Monitors

Three monitors are provided to measure potential releases from the main steam lines following a steam generator tube rupture. These monitors each include an ion chamber mounted approximately twelve inches from the steam lines (one on each main steam line and one on an atmospheric dump line).

Remote equipment in the control room includes a six decade meter for each monitor, a multipen recorder, alarm lamps for each monitor at the control room auxiliary cabinet, and a common alarm window at the main control board, which indicates equipment failure and high radioactivity.

The two unshielded gamma scintillation detectors, one mounted on each main steam line near the containment penetrations, measure N-16 activity in the main steam lines. A common radiation monitor provides main control board annunciation for N-16 equipment failure. The common monitor also sends a signal to the plant process computer which calculates primary to secondary leak rate for each steam generator in gallons per day. The plant process computer records, displays, and provides main control board annunciation based on high leak rate. This monitoring system is Non-QA, and is designed to support Millstone's Primary to Secondary leak rate detection and monitoring program.

7.5.6.3.2.1.9 Control Room Inlet Monitors

Two redundant GM tubes are located on the inlet duct of the control room outside air supply. A high radiation or instrument failure alarm on either monitor will isolate normal control room ventilation and initiate use of the filtered recirculation system.

7.5.6.3.2.2 Components

When applicable, the various components used in the systems are as follows:

1. Isokinetic nozzles, designed in accordance with American National Standards Institute (ANSI) 13.1-1969, are used for extracting samples from process streams to assure representative sampling of particulates.

2. Sample chambers are fabricated of stainless steel to prevent corrosion and polished to reduce absorption (plate out) of radioactive materials.

3. The sample chambers and detection assemblies utilize lead shielding to prevent background radiation levels from affecting instrument sensitivity.

4. A radioactive check source, controlled from the readout module in the control room, provides a convenient operational check.

5. Readout instrumentation consists of log count ratemeters with wide range (five decade) capability. A failure alarm is provided to indicate loss of signal or power and an alarm is provided for high radiation level. On certain monitors an alert alarm is also provided.

6. Indication with visual and audible alarms are provided locally and on the radiation monitoring panel in the control room.

7. Multipoint recorders are used to record particulate and gaseous radiation levels continuously.

7.5.6.3.3 System Operation

Should the level of radioactivity exceed the alarm setpoint, an audible/visual alarm will be sounded at the local monitor, with the exception of the containment air radiation monitors which have no local alarm facility. In addition a visible and audible alarm will be annunciated on the main control board. Should a loss of signal or loss of power occur on any monitor, a visible and audible alarm will also be annunciated on the main control board.

The gaseous and particulate monitors will be calibrated by using calibrated sources placed in a fixed geometry.

Administrative procedures established for the operators will ensure that releases will be kept to as low as practicable as specified in 10 CFR Part 50.34a and that releases will be less than the limits established by 10 CFR, Part 20 Sections 1301 and 1302 and Appendix B, Table 2.

In the event an audible alarm is sounded by an airborne radiation monitor, the operator will take appropriate action in accordance with plant procedures.

7.5.6.3.4 Availability and Reliability

Gaseous and particulate monitoring equipment is designed for operation over long periods with a minimum of maintenance. The sampling pumps are designed to be cleaned by a simple flushing operation. Filter replacement is a simple operation and the stainless steel sample chamber can be easily decontaminated by air purging and/or chemical flushing.

High-quality commercial-grade components are used throughout the system. Servicing of the electrical system is simplified by use of plug-in components.

The radiation monitoring panel was seismically qualified as documented in the report, “Technical Data Report, TDR-4174, Seismic Testing of Millstone Point Unit 2 Radiation Monitoring Instrumentation” dated April 1, 1974.

7.5.7 LOOSE PARTS MONITORING

7.5.7.1 Design Bases

7.5.7.1.1 Functional Requirements

The Loose Part Monitor (LPM) is designed to provide an indication of free and loose parts within the reactor coolant system (RCS). The LPM monitors the outputs of eight accelerometers: four on the reactor vessel and two on each of the steam generators. Metal-to-metal impacting within the RCS causes high-frequency, exponentially-decaying vibrations on the RCS shell which are detected and recorded by the LPM. The LPM provides an alarm in the control room to alert the operator so that further analysis and possible action can be undertaken.

7.5.7.2 Design Criteria

The LPM provides the means for detecting and evaluating metallic loose parts through analysis of transient acoustic signals produced by loose part impacts.

The LPM monitors the outputs of eight accelerometers:

a. Two on the lower reactor vessel.

b. Two on the upper reactor head.

c. Two on the inlet side of the channel head (one above and one below the tubesheet) of steam generator 1.

d. Two on the inlet side of the channel head (one above and one below the tubesheet) of steam generator 2.

The detection and recording circuitry are designed to capture impact data automatically with a minimum of operator intervention required.

The LPM is adjusted to achieve the maximum sensitivity commensurate with an acceptably low level of false alarms.

7.5.7.3 System Description

The detectors are accelerometers selected for operation in environments of high temperature and high radiation. Two types of cables are used in containment. Near the accelerometers, in areas of high temperature, mineral insulated, stainless steel jacketed cables are used. Beyond this, and to the containment penetrations, double insulated, twisted, low noise cables are used. The containment penetrations are low noise, coaxial type. Junction boxes are provided both inside and outside containment for ease of calibration and testing.

Outside the containment, eight differential charge amplifiers convert the output of each accelerometer to a proportional voltage signal which is routed to the control room electronics.

The control room electronics consists of signal conditioners, waveform recorders, an IBM-AT compatible computer (in an industrial package), real time displays and indicators, and interface circuits. Three cathode-ray tubes, digital voltmeters and light emitting diodes are provided for monitoring the status of the incoming signals. A floppy disc is used for the storage of impact data. Audio amplifiers and a headset permit the operator to monitor the sounds from each of the eight detectors.

7.5.7.4 System Operation

During normal operation, the LPM is in the standby mode, ready to detect and capture impact data from a loose part.

When an incoming signal(s) exceeds the preset trigger level, the trigger circuit commands the waveform recorders to collect the impact data and commands the computer to transfer the recorded data to memory and to the on-board floppy disc. The waveform recorders maintain a 10 millisecond pre-trigger buffer and can capture pre-trigger, as well as post-trigger data. The alarm circuit notifies the operators that an event occurred, and the operators remove the floppy disc containing the captured data for further analysis and possible action.

7.5.7.5 Emergency Operation

In the event a loose part monitor alarm (annunciator) is sounded, the operator will check the LPM cabinet to verify the alarm. If the alarm can be attributed to valve testing, CEA movement or other known transient, the operator will reset the alarms and verify that they do not return immediately. The operator will then monitor all previously alarmed channels, using the headset, to verify that no unusual sounds are present. If the above conditions are met, the operator will:

1. Remove the LPM DATA AQ DISK from the "A" drive.

2. Insert a new (unrecorded) NU LPM DATA AQ DISK.

3. Arm the LPM for monitoring by momentarily toggling the POWER switch to either the RESET or OFF position.

4. Transmit the recorded disk to the engineering department for disposition.

All action will be considered complete after engineering department has been informed.

If the alarm(s) cannot be attributed to known plant transient conditions, the operator will retrieve the floppy disc containing the captured data and will insert a new floppy disc. The floppy disc will be identified and forwarded to engineering department for evaluation. The alarms will then be reset, if possible. If the alarms reset, with no further alarms or unusual audible sounds, the operator's action is complete. If the alarms do not reset or if there are unusual audible sounds present, the operator will contact plant management for resolution.

Based on an evaluation of the nature of the loose part data, engineering department will recommend the appropriate action to protect the nuclear steam supply system.

7.5.7.6 Availability and Reliability

The components are conservatively rated for long life in this application. Accelerometers and mineral insulated cables are designed for the high radiation and high temperature environment to which they are exposed. Two installed spare accelerometers and cabling are provided for the lower reactor vessel to reduce future radiation exposure and speed repair in the event of a detector failure.

All active components are located outside the containment for ease of accessibility during plant operation. Ready access to all electronic components, channel-to-channel parts interchangeability, and availability of a full complement of spare parts will result in high system availability during plant operation.

7.5.8 CHLORINE MONITORING SYSTEM (SYSTEM DELETED)

7.5.9 SECONDARY SIDE SAFETY RELIEF VALVE POSITION INDICATION

The Safety Relief Valve (SRV) position indication instrumentation complies with the requirements of Regulatory Guide 1.97, Rev. 2, D-18 variable.

The SRV instruments are designed to aid the operating crew in monitoring the status of the SRVs during normal and accident conditions.

The sensor/electronics which are located in a harsh environment comply with the Equipment Environment Qualification (EEQ) requirements of the Regulatory Guide 1.97 and are, therefore, treated as QA Category 1, Components.

The instrumentation provides signals to the Main Control Board annunciator and the plant computer to indicate when the SRVs are closed/not closed.

7.5.10 REACTOR COOLANT PUMP VIBRATION MONITORING SYSTEM

7.5.10.1 Design Basis

7.5.10.1.1 Functional Requirements

The reactor coolant pump (RCP) vibration monitoring system provides the capability to continuously monitor each RCP. The system has an alarm that is triggered if high vibration is detected. The system records vibration and shaft rotation data periodically and when an alarm is triggered.

7.5.10.2 Design Criteria

The RCP vibration monitoring system monitors the RCP pump, shaft and motor casing. Transducers are selected for vibration characteristics, ability to withstand the containment environment and maintainability.

System components are located outside of containment in normally accessible locations when required or feasible.

7.5.10.3 System Description

Motor vibration is monitored by two velomitor probes mounted near the upper motor bearing, 90° apart, that sense motor casing radial vibration.

Proximity probes monitor the RCP shaft. Two probes, mounted 90 degrees apart, sense shaft radial vibration. A third probe monitors shaft speed and provides a reference for vibration phase. The probes are mounted above the pump seal housing.

Cables connect the proximity probes to proximitors located in a junction box inside containment. Cables connect the velomitor probes to terminal strips in the junction box. Cables then carry the signals outside of containment to the RCP vibration monitoring system instrument rack, located in the west penetration room. Once in the instrument rack, the signals are digitized and transmitted to the RCP vibration monitoring system computer terminal. If high vibration is sensed, an alarm is sent to the main control board annunciator.

7.5.10.4 System Operation

During normal operation the instruments are continuously monitored and periodically recorded. When high vibration is detected, monitor alarm contacts actuate an alarm on the main board. “Acknowledge” and “Silence” options are available at the main board for the alarm. The alarm is cleared via the plant process computer.

7.5.10.5 Emergency Operation

An operating procedure provides instructions for operator response if an alarm is triggered.

7.5.10.6 Availability and Reliability

Components were selected and located for long term operability and maintainability. In addition, all active electronic components are readily accessible in the west penetration room. Spare parts are readily available for system components.

TABLE 7.5-1 MAJOR PROCESS INSTRUMENTATION

Reactor Coolant System (see Figure 4.1–1)							
Component			**Protective Function ***	**Control Function ***	**Indicated**	**Recorded**	**Alarmed**
Reactor coolant loops							
	Flow		X		X		L
	Temperature (hot leg)		X	X	X	X	H
	Temperature (cold leg)		X	X	X	X	H
RCS mid-loop level							L
Pressurizer							
	Pressure		X	X	X	X	H, L, LL
	Level			X	X	X	H, L, LL
	Temperature						
		Water			X		
		Surge Line			X		L
		Spray Line			X		L
		Relief / Safety			X		H
		Discharge			X		H
		Steam			X		
Quench tank							
	Pressure				X		H
	Level				X		H, L
	Temperature				X		H

* Protective sensors and channels are independent of control sensors and channels.

Engineered Safeguards and Containment (see Figure 6.1–1)				
Component		**Indicated**	**Recorded**	**Alarmed**
Low-pressure safety injection				
	Pressure	X		
	Flow	X		

Engineered Safeguards and Containment (see Figure 6.1–1)				
Component		Indicated	Recorded	Alarmed
	Temperature		X	
High-pressure safety injection				
	Pressure	X	H	
	Flow	X		
Safety injection tanks				
	Pressure	X	H,L	
	Level	X	H,L	
Containment spray header				
	Pressure			
	Flow	X		
	Temperature	X		

Component		Protective Function *	Control Function *	Indicated	Recorded	Alarmed
Refueling water storage tank						
	Level	X		X		H, L
	Temperature		X	X		H, L
Containment pressure		X		X		H, HH
Containment temperature				X		H **
Containment radiation		X		X	X	H

* Protective sensors and channels are independent of control sensors and channels.

** Plant computer alarm only.

Chemical and Volume Control System (see Figure 9.2–2)				
Components	Control Function	Indicated	Recorded	Alarmed
Letdown line Pressure	X	X		H, L

Chemical and Volume Control System (see Figure 9.2–2)				
Components	**Control Function**	**Indicated**	**Recorded**	**Alarmed**
Flow		X		H
Temperature	X	X		H
Volume control tank				
Pressure		X		H, L
Level	X	X		H, L, LL
Temperature		X		H
Charging line				
Pressure		X		
Flow		X		L
Temperature		X		
Boric acid batching tank				
Temperature	X	X		
Boric acid tanks				
Level		X		H, L, LL
Temperature	X	X		H, L
Boric acid charging lines				
Pressure		X		
Flow	X	X	X	H, L
Miscellaneous				
Filter/strainer D/P		X		
Ion exchanger D/P		X		

Sampling System (see Figure 9.6–1 and 9.6–2)	
Components	**Indicated**
Sampling lines	
Pressure	X
Flow	X
Temperature	X

Spent Fuel Pool Cooling and Purification System (see Figure 9.5–1)			
Component	**Indicated**	**Recorded**	**Alarmed**
Fuel pool			
Level	X [1]		H, L
Temperature		X	H
Heat exchanger flow path			
Pressure	X		
Temperature			H
Flow	X		L
Purification flow path			
Pressure	X		
Flow	X		
Filter D/P	X		H
Ion exchanger D/P	X		H
Ion exchanger strainer D/P	X		H

Notes: 1. Continuous Nonsafety augmented quality wide range level indication is provided remotely in the Cable Vault and East 480V Switchgear Rooms of the Auxiliary Building.

Shutdown Cooling System (see Figure 9.3–1)				
Components	**Control Function**	**Indicated**	**Recorded**	**Alarmed**
Shutdown heat exchangers				
Temperature (out)		X		
Discharge lines				
Pressure		X		
Flow	X	X		
Temperature		X	X	
Return lines				
Temperature			X	
Makeup water flow	X	X	X	H, L
Reactor coolant pump controlled bleedoff pressure	X			H, HH

Steam and Feedwater System (see Figure 10.3–1 and 10.4–2)					
Component	**Protective Function ***	**Control Function ***	**Indicated**	**Recorded**	**Alarmed**
Steam generator					
Level (N.R.)	X	X	X	X	H, L, HH
Feedwater inlet temperature			X		
Feedwater flow		X		X	
Steam flow		X		X	
Outlet Pressure	X	X	X	X	L
Level (W.R.)			X		
Condenser					
Hot well level		X	X	X	H, L
Absolute pressure					
Narrow range				X	H
Wide range		X	X		
Conductivity			X	X	H, HH

Steam and Feedwater System (see Figure 10.3–1 and 10.4–2)					
Component	**Protective Function ***	**Control Function ***	**Indicated**	**Recorded**	**Alarmed**
Turbine					
Throttle steam pressure			X	X	
First stage pressure		X	X		
Feedwater heaters level		X	X		H, L
Steam generator feedwater pumps					
Suction pressure			X		L
Discharge pressure		X	X		H
Speed		X			
Condensate pumps discharge pressure		X	X		
Condensate pump discharge total flow		X	X		
Auxiliary feedwater pumps					
Discharge pressure				X	
Feedwater regulator					
Differential pressure		X	X		
Auxiliary feedwater supply					
Flow			X		
Valve position			X		
Moisture separator drain tanks - Level		X			H, L
Reheater drain tank - Level		X		X	H, L
Condensate storage tank - Level			X		H, L, LL
Miscellaneous					
Main steam pressure		X	X	X	
Main steam conductivity			X	X	H
Auxiliary steam pressure			X		
Condensate makeup conductivity		X	X		

* Protective sensors and channels are independent of control sensors and channels.

Circulating Water System (see Figure 9.7–1)				
Components	**Control Function**	**Indicated**	**Recorded**	**Alarmed**
Condenser				
Inlet temperature		X		
Outlet temperature		X		
Service Water Pumps				
Discharge pressure	X	X		L

TABLE 7.5-2 OMITTED

TABLE 7.5-3 REGULATORY GUIDE 1.97 (REV. 2) ACCIDENT MONITORING INSTRUMENTATION

Regulatory Guide 1.97 VARIABLE	PARAMETER & INSTRUMENT (LOOP) ID	INSTRUMENT RANGE	COMMENTS
A-01	Pressurizer Pressure		
	LR: P-103, 103-1	LR: 0-1600 psia	
	HR: P-102A, B	HR: 1500-2500 psia	
A-02	Deleted		
A-03	RCS Hot Leg Temperature T-111X, T-121X	150-750°F	
A-04	Steam Generator Pressure P-1013A, B P-1023A, B	0-1000 psia	
A-05	Steam Generator Level		
	L-1113A, B L-1123A, B	0-100% (top of tube bundles to separators)	
	L-1114A, B L-1124A, B	0-460 inches (20 inches above tube sheet to top of moisture separators)	
A-06	Deleted		
A-07	Deleted		
	AE-8152, 8154		
A-08	Refueling Water Storage Tank Level L-3001, L-3002	4 to 100%	
A-09	RCS Cold Leg Temperature T-115 & T-125	0° - 750°F	

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TABLE 7.5-3 REGULATORY GUIDE 1.97 (REV. 2) ACCIDENT MONITORING INSTRUMENTATION

Regulatory Guide 1.97 VARIABLE	PARAMETER & INSTRUMENT (LOOP) ID	INSTRUMENT RANGE	COMMENTS
B-01	Neutron Flux		
	WR-LOG-A, D (DRWR)	10^{-8} to 200% FP	
	JI-001, JI-004 (MCB)	10^{-8} to 100% FP	
B-02	Control Rod Position	Full in or not full in	
B-03	RCS Soluble Boron Concentration	None	See Note 5
B-04	RCS Cold Leg Temperature T-115 & T-125	0°-750°F	
B-05	RCS Hot Leg Temperature T-111X & T-121X	150°-750°F	
B-06	RCS Cold Leg Temperature T-115 & T-125	0°-750°F	
B-07A	RCS Pressure		
	LR: P-103, 103-1	LR: 0-1600 psia	
	HR: P-102A, B	HR: 1500-2500 psia	
B-07B	Pressurizer Pressure (Wide Range) P-102B-1	0-3000 psig	
B-08	Core Exit Temperature 21 Sensors per Channel "A", 22 Sensors per Channel "B"	200-2300°F	
B-09	Coolant Level in Reactor HJTC-A, B	Top of core to top of vessel	
B-10	Degree of Subcooling ICCM Z1 & Z2	200°F Subcooling to 35°F Superheating	

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TABLE 7.5-3 REGULATORY GUIDE 1.97 (REV. 2) ACCIDENT MONITORING INSTRUMENTATION

Regulatory Guide 1.97 VARIABLE	PARAMETER & INSTRUMENT (LOOP) ID	INSTRUMENT RANGE	COMMENTS
B-11A	RCS Pressure		
	LR: P-103, 103-1	LR: 0-1600 psia	
	HR: P-102A, B	HR: 1500-2500 psia	
B-11B	Pressurizer Pressure (Wide Range) P-102B-1	0-3000 psig	
B-12A	Containment Sump Water Level (Wide Range) L-8242, 8243	0' to 7' 0 to 565,000 gallons	See Note 2
B-12B	Containment Sump Water Level (Narrow Range) L-9155 (L-9155A Backup)	0-100%	See Note 3
B-13	Containment Pressure P-8113, 8114	0-60 psig	
B-14	Containment Isolation Valve Position ZS-198, 505, 506, 516, 1060, 1062, 1064, 2525, 4246, 4248, 4250, 4251, 7311, 7312, 7690, 8121, 8122, 8124, 8150, 8151, 8377, 8378, 8379, 8380, 8656, 9015, 9016, 9125, 9126, 9150, 9151, 9230	Closed-Not closed	
B-15	Containment Pressure P-8238 & 8239	0-250 psia	
C-01	Core Exit Temperature 21 Sensors per Channel “A”, 22 Sensors per Channel “B”	200°-2300°F	
C-02	Radioactivity Concentration or Radiation Level in Circulating Primary Coolant	None	See Note 5
C-03	Analysis of Primary Coolant (Gamma Spectrum)	None	See Note 5

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TABLE 7.5-3 REGULATORY GUIDE 1.97 (REV. 2) ACCIDENT MONITORING INSTRUMENTATION

Regulatory Guide 1.97 VARIABLE	PARAMETER & INSTRUMENT (LOOP) ID	INSTRUMENT RANGE	COMMENTS
C-04A	RCS Pressure		
	Low range: P-103, 103-1	Low range: 0-1600 psia	
	High range: P-102A, B	High range: 1500-2500 psia	
C-04B	Pressurizer Pressure (Wide Range) P-102B-1	0-3000 psig	
C-05	Containment Pressure P-8238 & 8239	0-250 psia	
C-06A	Containment Sump Water Level (Wide Range) L-8242, 8243	0 feet to 7 feet, 0 to 565,000 gallons	See Note 2
C-06B	Containment Sump Water Level (Narrow Range) L-9155 (L-9155A Backup)	0-100%	See Note 3
C-07	Containment Area Radiation RM-8240 & 8241	10^0 to 10^8 R/hr	
C-08	Effluent Radioactivity-Noble Gas Effluent from Condenser Air Removal System Exhaust RM-5099, RR-9373	10^{-6} μCi/cc to 10^{-2} μCi/cc	
C-09A	RCS Pressure		
	Low range: P-103, 103-1	Low range: 0-1600 psia	
	High range: P-102A, B	High range: 1500-2500 psia	
C-09B	Pressurizer Pressure (Wide Range) P-102B-1	0-3000 psig	
C-10	Containment Hydrogen Concentration AE-8152, 8154	0-10%	
C-11	Containment Pressure P-8238, 8239	0-250 psia	

TABLE 7.5-3 REGULATORY GUIDE 1.97 (REV. 2) ACCIDENT MONITORING INSTRUMENTATION

Regulatory Guide 1.97 VARIABLE	PARAMETER & INSTRUMENT (LOOP) ID	INSTRUMENT RANGE	COMMENTS
C-12	Containment Effluent Radioactivity-Noble Gases from Identified Release Points-see Variable C-14	See Variable C-14	
C-13	Radiation Exposure Rate (inside buildings or areas, which are in direct contact with primary containment where penetrations and hatches are located) None	Deleted in Regulatory Guide 1.97, Rev. 3	See Note 6
C-14	Effluent Radioactivity-Noble Gases RM-8168 RM-8132B RM-8169	1 x 10^{-7} to 1 x 10^{5} μCi/cc	
D-01	RHR System Flow F-306	0-7000 gpm	
D-02	RHR Heat Exchanger Outlet T-303X, Y T-351Y	0-400°F	
D-03A	Accumulator Tank Level L-311, 321, 331, 341	0-100%	
D-03B	Accumulator Tank Pressure P-311, 321, 331, 341	0-250 psig	
D-04	Accumulator Isolation Valve Position Z-614, 624, 634, 644	Closed or open	
D-05	Boric Acid Charging Flow F-212	0-140 gpm	
D-06	Flow in HPSI System F-311, 321, 331, 341 (Backup variable: Pump Motor Current)	0-300 gpm	
D-07	Flow in LPSI System F-312, 322, 332 & 342 (Backup variable: Pump Motor Current)	0-2000 gpm	
D-08	Refueling Water Storage Tank Level L-3001, 3002	4-100%	
D-09	Reactor Coolant Pump Status P40A, B, C, D	0-600 amps	

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TABLE 7.5-3 REGULATORY GUIDE 1.97 (REV. 2) ACCIDENT MONITORING INSTRUMENTATION

Regulatory Guide 1.97 VARIABLE	PARAMETER & INSTRUMENT (LOOP) ID	INSTRUMENT RANGE	COMMENTS
D-10	Primary System Safety Relief Valve Positions Z-200, 201, 402, 404	Closed-not closed	
D-11	Pressurizer Level L-110X, Y	0-100%	
D-12	Pressurizer Heater Status (Proportional) L105 (AM-B0504), 106 (AM-B0609)	0-250 amps	
	Pressurizer Heater Status (Backup) L101, 102, 103, 104 (Lights)	On-off	
D-13	Quench Tank Level L-116	0-100%	
D-14	Quench Tank Temperature T-116	0-300°F	
D-15	Quench Tank Pressure P-116	0-100 psig	
D-16	Steam Generator Level		
	L-1113A, B	0-100% Top of tube bundles to separators	
	B L-1123A, B		
	L-1114A, B	0-460 inches (20 inches above tube sheet to separator)	
	B L-1124A, B		
D-17	Steam Generator Pressure P-4223 & 4224	0-1200 psia	
D-18	SRV Position FS-4225, 26, 27, 28, 29, 30, 31, 32, 33, 34, 35, 36, 37, 38, 39, 40	Open-closed	
D-19	Main Feedwater Flow F-5268A, B F-5269A, B	0-63 x 10^5 lbs/hr	

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TABLE 7.5-3 REGULATORY GUIDE 1.97 (REV. 2) ACCIDENT MONITORING INSTRUMENTATION

Regulatory Guide 1.97 VARIABLE	PARAMETER & INSTRUMENT (LOOP) ID	INSTRUMENT RANGE	COMMENTS
D-20	Auxiliary Feedwater Flow F-5277A, B F-5278A, B	0-600 gpm	
D-21	Condensate Storage Tank Level L-5282, LIS-5489, L-5280	0-100%	
D-22	Containment Spray Flow F-3023, 3024 (Backup variable: Pump Motor Current)	0-5000 gpm	
D-23	Heat Removal by Containment Fan Heat Removal System T-6082, 6086, 6090, 6093 (Backup variables: Containment Outlet Temp. & Flow) T-6031, 6032, 6033; F-6081, 6085, 6089, 6094)	0-200°F	
D-24	Containment Atmosphere Temperature T-8095, 8096, 8097, 8098, 8108, 8109 & 8110	0-350°F	
D-25	Containment Sump Water Temperature None		See Note 7
D-26	Make Up Flow-In (Charging) F-212	0-140 gpm	
D-27	Letdown Flow-Out F-202	0-140 gpm	
D-28	Volume Control Tank Level L-226	0-100%	
D-29	Component Cooling Water Temperature to ESF System T-6031, 6032, 6033	0-200°F	
D-30	Component Cooling Water Flow to ESF System F-6034, 6035	0-10,000 gpm	
D-31	High Level Radioactive Liquid Tank Level (PDT) L-9051	0-100%	
D-32	Radioactive Gas Holdup Tank Pressure P-9128	0-25 psig	

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TABLE 7.5-3 REGULATORY GUIDE 1.97 (REV. 2) ACCIDENT MONITORING INSTRUMENTATION

Regulatory Guide 1.97 VARIABLE	PARAMETER & INSTRUMENT (LOOP) ID	INSTRUMENT RANGE	COMMENTS
D-33	Emergency Ventilation Damper Position ZS-8000, 8001, 8002, 8003B, 8004, 8005, 8006, 8007, 8009, 8010, 8361, 9506, 9507, 9508	Open-closed	See Note 8
D-34	Status of Standby Power and Other Energy Sources Important to Safety (Hydraulic, Pneumatic) Various	Volts, Amps, Pressures	See Note 9
E-01	Containment Area Radiation (High Range) RM-8240, 8241	10^0 to 10^8 R/hr	
E-02	Radiation Exp. Rate Rm-7890, 7891, 7892, 7894, 7895, 7896, 7897, 7899, 8139, 8142, 8156, 8157	1 x 10^{-1} to 10^4 mR/hr	
E-03A	Common Vent-Noble Gas See Variable C-14	See Variable C-14	
E-03B	Plant Vent Flow		
	MP2: F-8412	10 to 10^5 cfm (MP2)	
	MP1: F-20-34	0 to 223,000 cfm (MP1)	
E-03C	Vent from Steam Generator or Steam Dump RM-4299A, B & C	10^{-1} μCi/cc to 10^3 μCi/cc	

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TABLE 7.5-3 REGULATORY GUIDE 1.97 (REV. 2) ACCIDENT MONITORING INSTRUMENTATION

Regulatory Guide 1.97 VARIABLE	PARAMETER & INSTRUMENT (LOOP) ID	INSTRUMENT RANGE	COMMENTS
E-04	(Particulates and Halogens) All Identified Plant Release Points	Sampler Particulate & Iodine filters are used for laboratory analysis	See Note 11
	RM-8132A/B	1 x 10^{-3} to 1 x 102 µCi/cc	
	RM-8168	1 x 10^{-7} to 1 x 105 µCi/cc	
	RM-8169		
E-05A	Radiation Exposure Meters (Continuous indication at fixed locations) None	None	See Note 6
E-05B	Airborne Radio-Halogens and Particulates None	None	See Note 12
E-05C	Plant and Environs Radiation None	None	See Note 12
E-05D	Plant and Environs Radioactivity None	None	See Note 15
E-06	Wind Direction	0-360°	See Note 13
	Speed	0-100 mph	
	Temperature	(-10)-(+18) °F	
	Various		

NOTES

NOTE (1) Deleted

NOTE (2) During normal operation, containment narrow range sump level is used to indicate level and initiate an alarm for the operator to manually start the sump pumps. Because the narrow range sump would be filled to capacity in the event of an accident, following an accident the containment wide range sump level would be used to monitor containment water level.

NOTE (3) Containment narrow range sump level, transmitter LT-9155A is an installed spare that is not connected to any indicator. It is a backup that can be easily terminated to replace the normal narrow range sump level transmitter LT-9155.

NOTE (4) Deleted.

NOTE (5) No existing instrument monitors this variable. Contingency plans to obtain and analyze samples of primary coolant are contained within Chemistry Department implementing procedures.

NOTE (6) Deleted on Regulatory Guide 1.97, Rev. 3.

Also, not cost-effective per NUREG/CR 2644.

NOTE (7) No existing instrument monitors this variable.

NOTE (8) The following is a list of dampers and limit switches for variable D-33:

2-HV-202, 2-HV-203A,2-HV-203B, 2-HV-206A, 2-HV-206B, 2-HV-207, 2-HV-208, 2-HV-210, 2-HV-211, 2-HV-212A, 2-HV-212B, ZS-8000, ZS-8001, ZS-8002, ZS-8003B, ZS-8004, ZS-8005, ZS-8006, ZS-8007, ZS-8009, ZS-8010, ZS-8361.

NOTE (9) The status is indicated by voltmeters, ammeters, watt meters, and status lights on the main control board. The status of the starting air for the diesels is alarmed in the control room. All sensors are located in a mild environment.

NOTE (10) Deleted.

NOTE (11) Particulate and iodine filters for these monitors are removed for laboratory analysis.

NOTE (12) Portable instruments are used to monitor this variable per Regulatory Guide1.97, Rev. 2.

NOTE (13) Actual wind direction indicates a 0°-540° range (1.5 revolutions) for computer averaging purposes.

NOTE (14) Deleted.

NOTE (15) Isotopic analysis via various on-site and off-site gamma (GeLi) spectrometers.

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TABLE 7.5-4 AREA RADIATION MONITORS

CHANNEL NO.	DESCRIPTION	RANGE mr/hr	DETECTION ASSEMBLY	A-AUDIBLE ALARM V-VISUAL ALARM METER INDICATION AND ALARM (A.V.) CONTROL ROOM	LOCAL	CONTROL LOGIC
RM-7890	Containment Personnel Access Match	1 x 10^{-1} to 1 x 10^{4}	Gamma-Scintillation	A, V	A, V	N/A
RM-7891	Containment Refueling Machine Service Platform	1 x 10^{-1} to 1 x 10^{4}	Gamma-Scintillation	A, V	A, V	N/A
RM-7892	Auxiliary Building Drumming and Decontamination Area	1 x 10^{-1} to 1 x 10^{4}	Gamma-Scintillation	A, V	A, V	N/A
RM-7894	Charging Pump Area	1 x 10^{-1} to 1 x 10^{4}	Gamma-Scintillation	A, V	A, V	N/A
RM-7895	Sampling Room	1 x 10^{-1} to 1 x 10^{4}	Gamma-Scintillation	A, V	A, V	N/A
RM-7896	Radioactive Waste Processing Area (Elevation (-)25 feet 6 inches)	1 x 10^{-1} to 1 x 10^{4}	Gamma-Scintillation	A, V	A, V	N/A

TABLE 7.5-4 AREA RADIATION MONITORS

CHANNEL NO.	DESCRIPTION	RANGE mr/hr	DETECTION ASSEMBLY	A-AUDIBLE ALARM V-VISUAL ALARM METER INDICATION AND ALARM (A.V.) CONTROL ROOM	LOCAL	CONTROL LOGIC
RM-7897	Radioactive Waste Processing Area (Elevation (-)45 feet 6 inches)	1×10^{-1} to 1×10^{4}	Gamma-Scintillation	A, V	A, V	N/A
RM-7899	Control Room	1×10^{-1} to 1×10^{4}	Gamma-Scintillation	A, V	A, V	N/A
RM-8139	Spent Fuel Pool Area	1×10^{-1} to 1×10^{4}	Gamma-Scintillation	A, V	A, V	2-out-of-4 Engineered Safety Features Actuation System
RM-8142	Spent Fuel Pool Area	1×10^{-1} to 1×10^{4}	Gamma-Scintillation	A, V	A, V	2-out-of-4 Engineered Safety Features Actuation System
RM-8156	Spent Fuel Pool Area	1×10^{-1} to 1×10^{4}	Gamma-Scintillation	A, V	A, V	2-out-of-4 Engineered Safety Features Actuation System
RM-8157	Spent Fuel Pool Area	1×10^{-1} to 1×10^{4}	Gamma-Scintillation	A, V	A, V	2-out-of-4 Engineered Safety Features Actuation System

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TABLE 7.5-4 AREA RADIATION MONITORS

CHANNEL NO.	DESCRIPTION	RANGE mr/hr	DETECTION ASSEMBLY	A-AUDIBLE ALARM V-VISUAL ALARM METER INDICATION AND ALARM (A.V.) CONTROL ROOM	LOCAL	CONTROL LOGIC
RM-9813	Drumming Area (Elevation (-)45 feet 6 inches)	10 to 10^6	Gamma-Scintillation	N/A	A, V	N/A
RM-8240	Containment High Range Radiation Monitor	10^3 to 10^{11}	Ionization Chamber	A, V	N/A	1-out-of-2 H2 Purge Valves Closure
RM-8241	Containment High Range Radiation Monitor	10^3 to 10^{11}	Ionization Chamber	A, V	N/A	1-out-of-2 H2 Purge Valves Closure

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TABLE 7.5-5 LIQUID PROCESS/EFFLUENT RADIATION MONITORS

CHANNEL NO.	DESCRIPTION *	DECADE RANGE CPM	AUTOMATIC ISOLATION FUNCTION
RM-4262 *	Steam Generator Blowdown Sample Monitor	10 to 10^6	Steam Generator Blowdown, Blowdown Tank Discharges & Blowdown Sample Valves
RM-6038	Reactor Building Closed Cooling Water Monitor	10 to 10^6	None
RM-9049	Clean Liquid Waste Monitor	10 to 10^6	Clean Liquid Waste Discharge
RM-9116	Aerated Liquid Waste Monitor	10 to 10^6	Aerated Liquid Waste Discharge
RM-9327	Condensate Recovery Tank Monitor	10 to 10^6	Condensate Recovery Tank Discharge
CND-RE245	Condensate Polishing Waste - Neutralizing Sump Monitor	10 to 10^6	Condensate Polishing Facility Waste - Neutralizing Sump Discharge

* All monitors include an off-line, gamma scintillation detector.

TABLE 7.5-6 AIRBORNE PROCESS/EFFLUENT RADIATION MONITORS

Channel Number	Description	Detector Type	Smapler Type	Decade Range
RM-8132A *	Unit 2 Stack Monitor Particulate	Beta Scintillation	Off-Line Part.	10 - 10^6 cpm
RM-8132B	Unit 2 Stack Monitor Gaseous	Beta Scintillation	Off-Line Gas	10 - 10^6 cpm
RM-8168	Unit 2 Stack Mid and High Range	GM Tubes	Off-Line Gas & Part.	10 x E -3 to 10 x E+5 µCi/cc
RM-8169	Unit 2 Millstone Stack Wide Range	GM Tubes	Off-Line Gas & Part.	10 x E -7 to 10 x E+5 µCi/cc
RM-8123A *	Containment Air Monitor - Particulate	Beta Scintillation	Off-Line Part.	10 - 10^6 cpm
RM-8123B	Containment Air Monitor - Gaseous	Beta Scintillation	Off-Line Gas	10 - 10^6 cpm
RM-8262A *	Containment Air Monitor - Particulate	Beta Scintillation	Off-Line Part.	10 - 10^6 cpm
RM-8262B	Containment Recirculating Air Monitor - Gaseous	Beta Scintillation	Off-Line Gas	10 - 10^6 cpm
RM-8434A *	Radwaste Vent Monitor - Particulate	Beta Scintillation	Off-Line Part.	10 - 10^6 cpm
RM-8434B	Radwaste Vent Monitor - Gaseous	GM Tube	Off-Line Gas	10 - 10^6 cpm
RM-5099	Condenser Air Ejector Discharge Monitor - Gaseous	Gamma Scintillation	Adjacent to Line Gas	10 - 10^7 cpm
RM-8011	Control Room Monitor - Gaseous	GM Tube	Off-Line Gas	10 - 10^6 cpm
RM-9799 A & B	Control Room Intake Duct	GM Tube	On-Line Gas	0.1 - 10^4 mr/hr
RM-8145A *	Fuel Handling Exhaust Air Monitor - Particulate	Beta Scintillation	Off-Line Part.	10 - 10^6 cpm

TABLE 7.5-6 AIRBORNE PROCESS/EFFLUENT RADIATION MONITORS

Channel Number	Description	Detector Type	Smapler Type	Decade Range
RM-8145B	Fuel Handling Exhaust Air Monitor - Gaseous	GM Tube	Off-Line Gas	10 - 10^6 cpm
RM-8997 *	Radwaste Vent Monitor - Particulate	Beta Scintillation	Off-Line Part.	10 - 10^6 cpm
RM-8998 *	Radwaste Vent Monitor - Particulate	Beta Scintillation	Off-Line Part.	10 - 10^6 cpm
RM 8999 *	Radwaste Vent Monitor - Particulate	Beta Scintillation	Off-Line Part.	10 - 10^6 cpm
RM-9095	Filtered Waste Gas to Millstone Stack Monitor - Gaseous	Beta Scintillation	Off-Line Gas	10 - 10^6 cpm

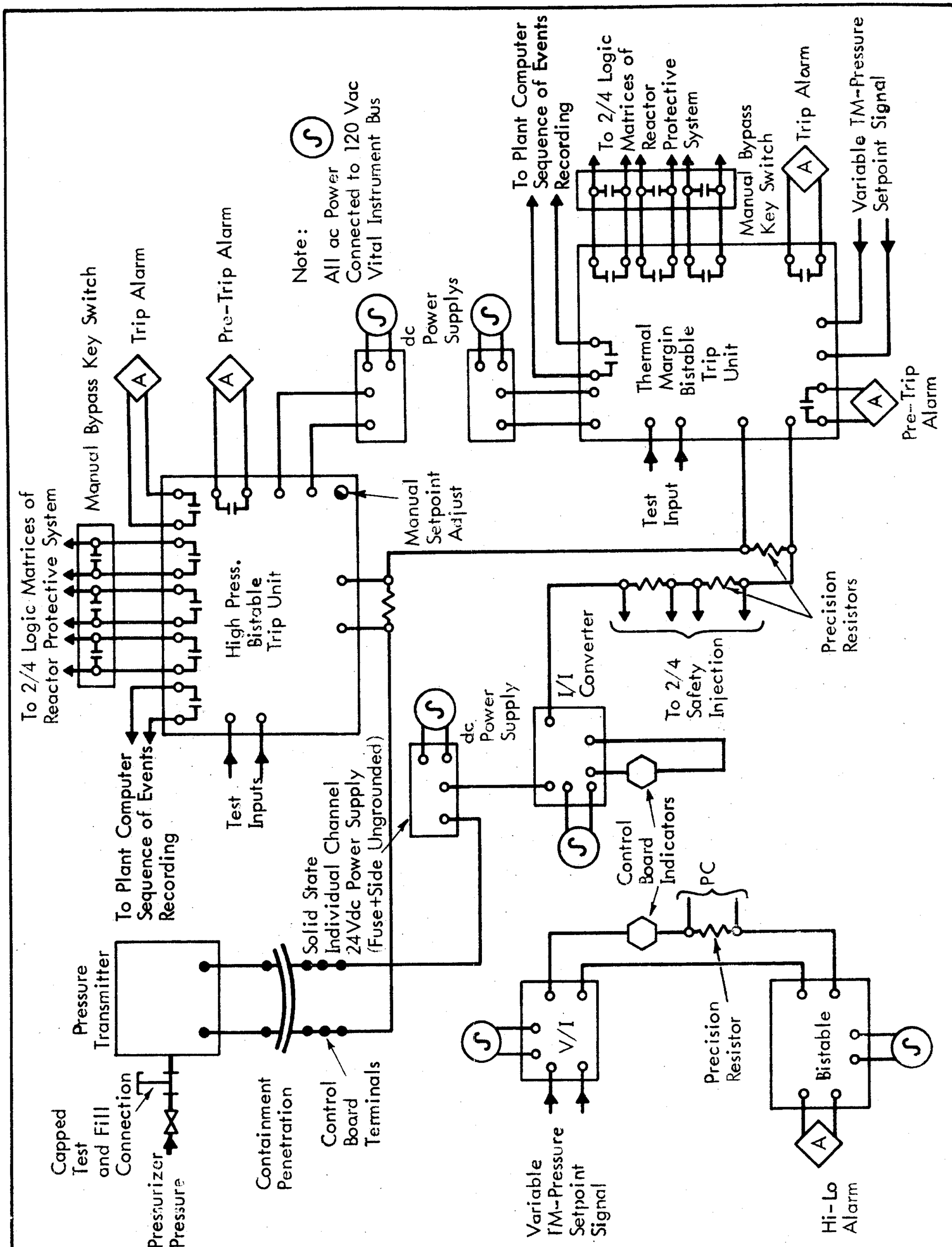
* Replaceable charcoal cartridge assembly is provided for laboratory analysis of iodine.

TABLE 7.5-6A PROCESS RADIATION MONITORS (STEAM)

Channel Number	Description	Detector Type	Sampler Type	Decade Range
RM4299A	Number 1 Steam Generator Main Steam Line Radiation Monitor	Ion Chamber	On Line Proximity	10^{-2} - 10^{4} R/HR
RM4299B	Number 1 Steam Generator Atmospheric Dump Steam Radiation Monitor	Ion Chamber	On Line Proximity	10^{-2} - 10^{4} R/HR
RM4299C	Number 2 Steam Generator Main Steam Line Radiation Monitor	Ion Chamber	On Line Proximity	10^{-2} - 10^{4} R/HR
RM4296A	Number 1 Steam Generator N-16 Monitor	Gamma Scintillator	On Line Proximity	10^{-7} - 10^{-1} μci/ml
RM4296B	Number 2 Steam Generator N-16 Monitor	Gamma Scintillator	On Line Proximity	10^{-7} - 10^{-1} μci/ml

TABLE 7.5-7 CEA POSITION LIGHT MATRIX

Light Color	Regulating CEAs	Shutdown CEAs
Red	Upper electrical limit	Upper electrical limit
Green	Lower electrical limit	Lower electrical limit
Amber	Dropped CEA	Dropped CEA
White	Between upper and lower limits	(Not Applicable)
Blue	(Not Applicable)	Exercise limit



Millstone Nuclear Power Station Unit No. 2	Pressurizer Pressure Measurement Channel Functional Diagram	Figure 7.5-1

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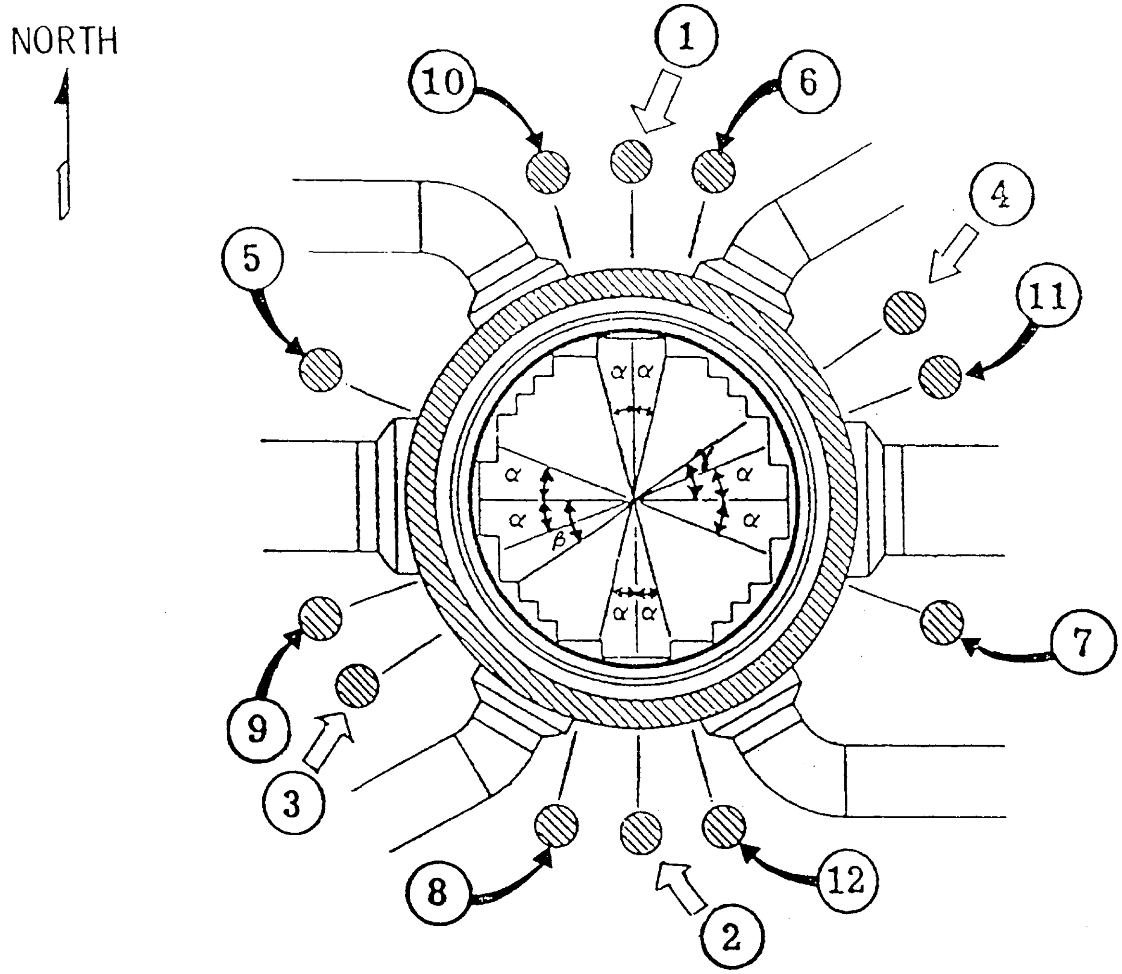
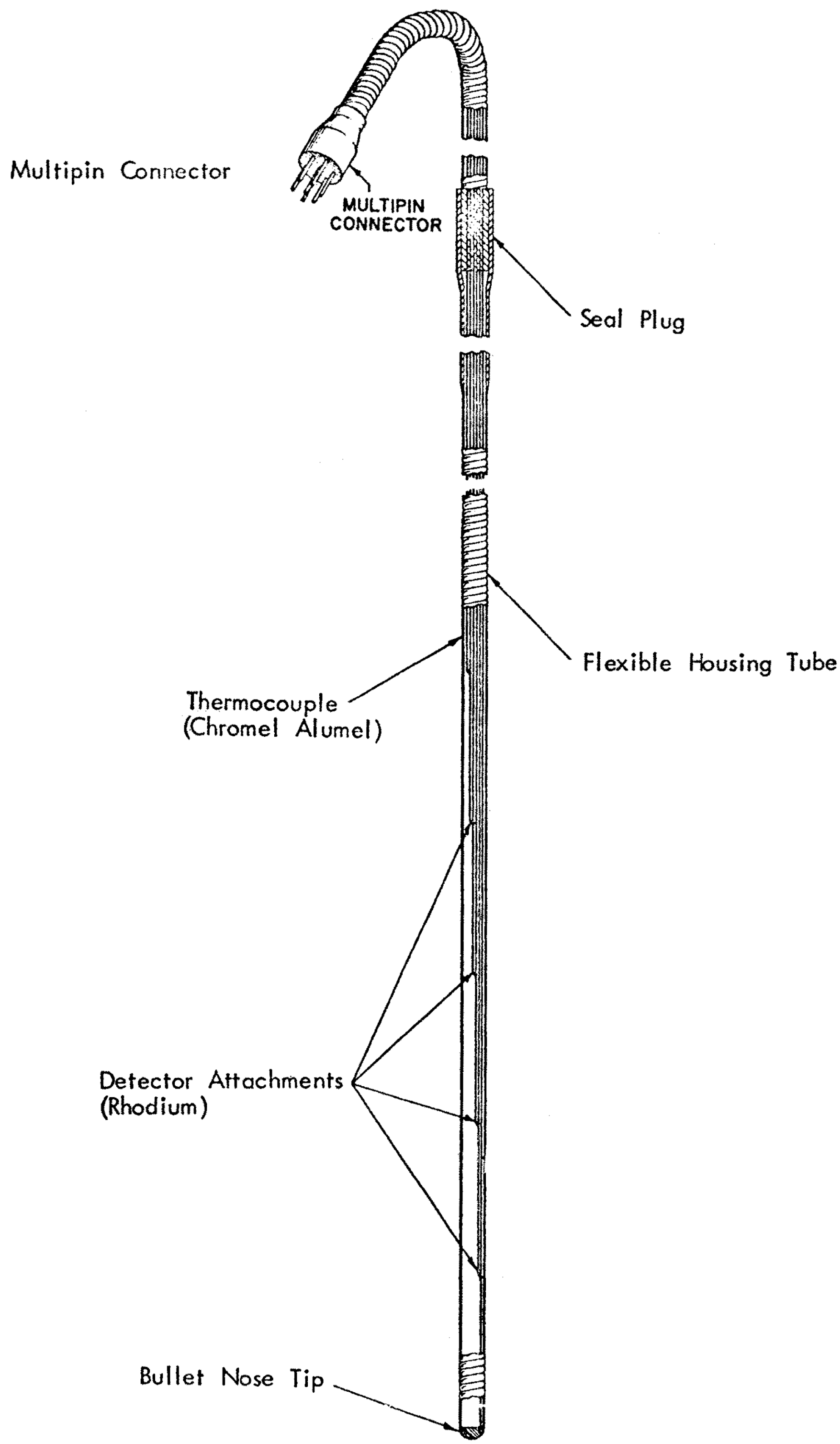


$\alpha = 17°$
$\beta = 37\frac{1}{2}°$
$\gamma = 25°$

WIDE RANGE LOGARITHMIC CHANNELS IDENTIFIED BY ⇨

CHANNELS	NUMBERS
WIDE RANGE LOGARITHMIC	1 - 4
CONTROL	10 & 12
SAFETY	5 - 8
SPARES	9 & 11

FIGURE 7.5-2
OUT-OF-CORE NUCLEAR DETECTOR LOCATIONS



Millstone Nuclear Power Station Unit No. 2	In-Core Nuclear Detector Assembly	Figure 7.5-3

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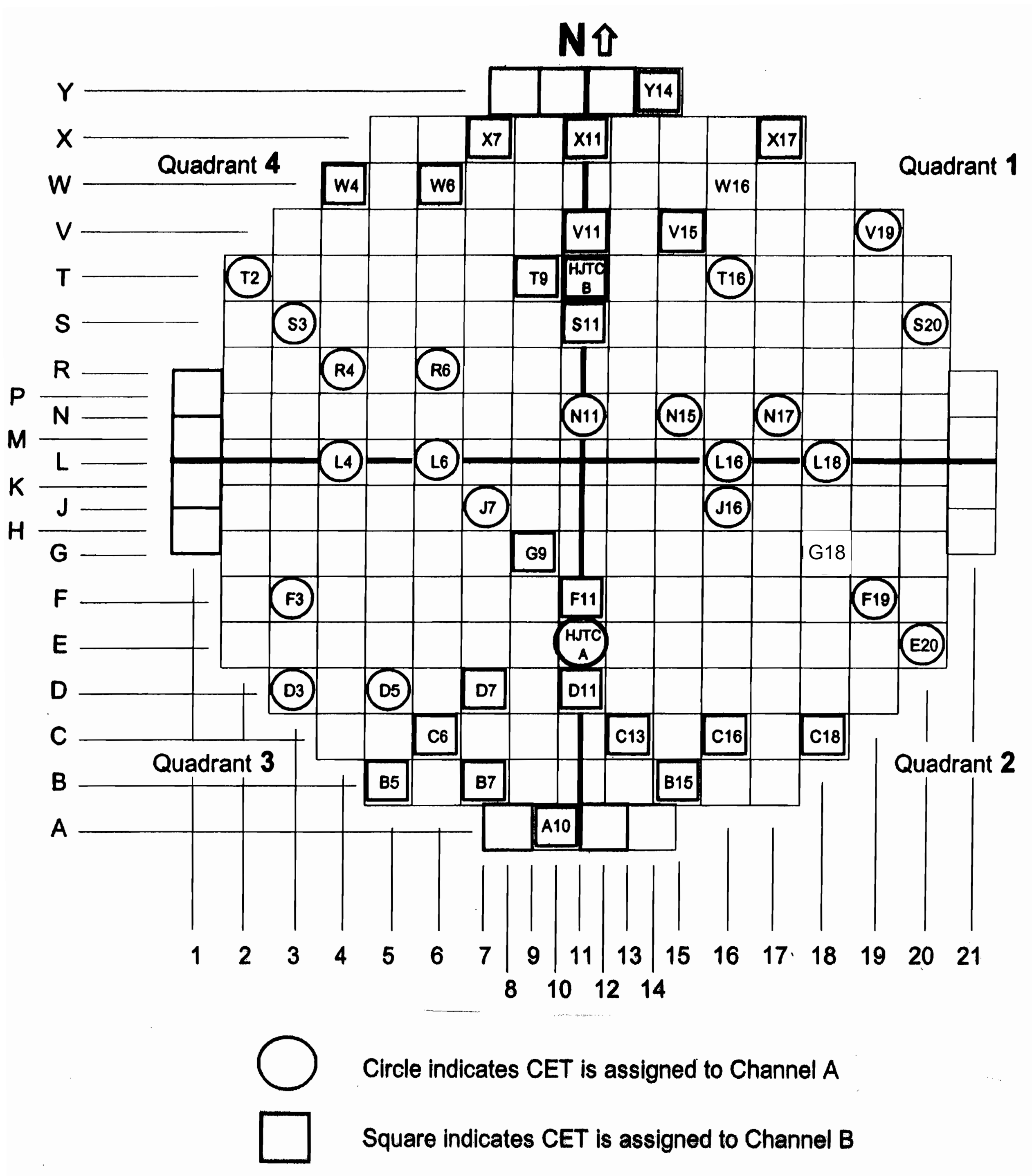
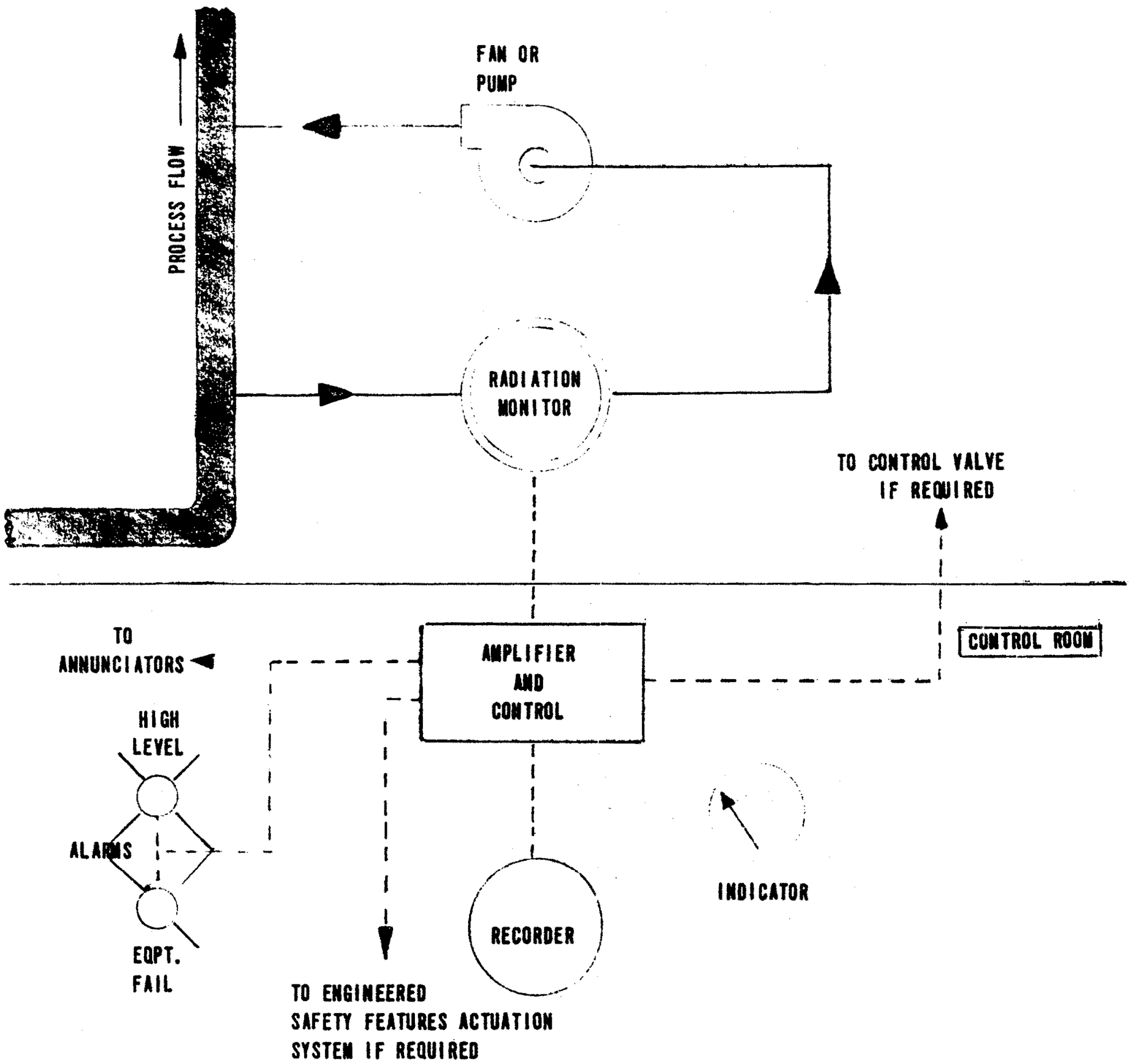
FIGURE 7.5-4 IN-CORE DETECTOR/CORE EXIT THERMOCOUPLE AND RVLM SENSOR LOCATIONS





MILLSTONE 2
PROCESS RADIATION MONITOR DIAGRAM (TYPICAL)

FIG. 7.5-5

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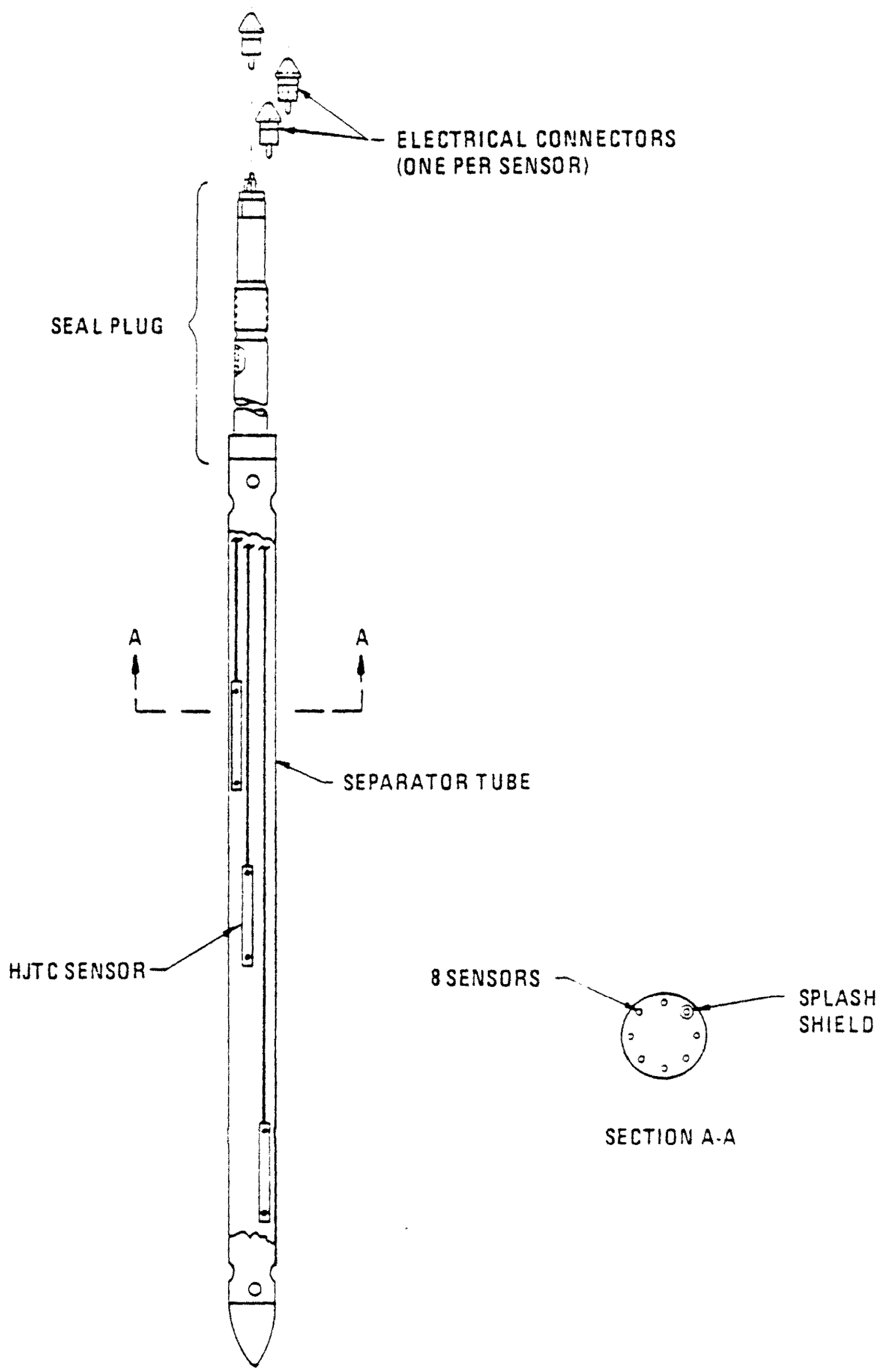

POWER SYSTEMS COMBUSTION ENGINEERING, INC.	HEATED JUNCTION THERMOCOUPLE PROBE ASSEMBLY	Figure 7.5-6

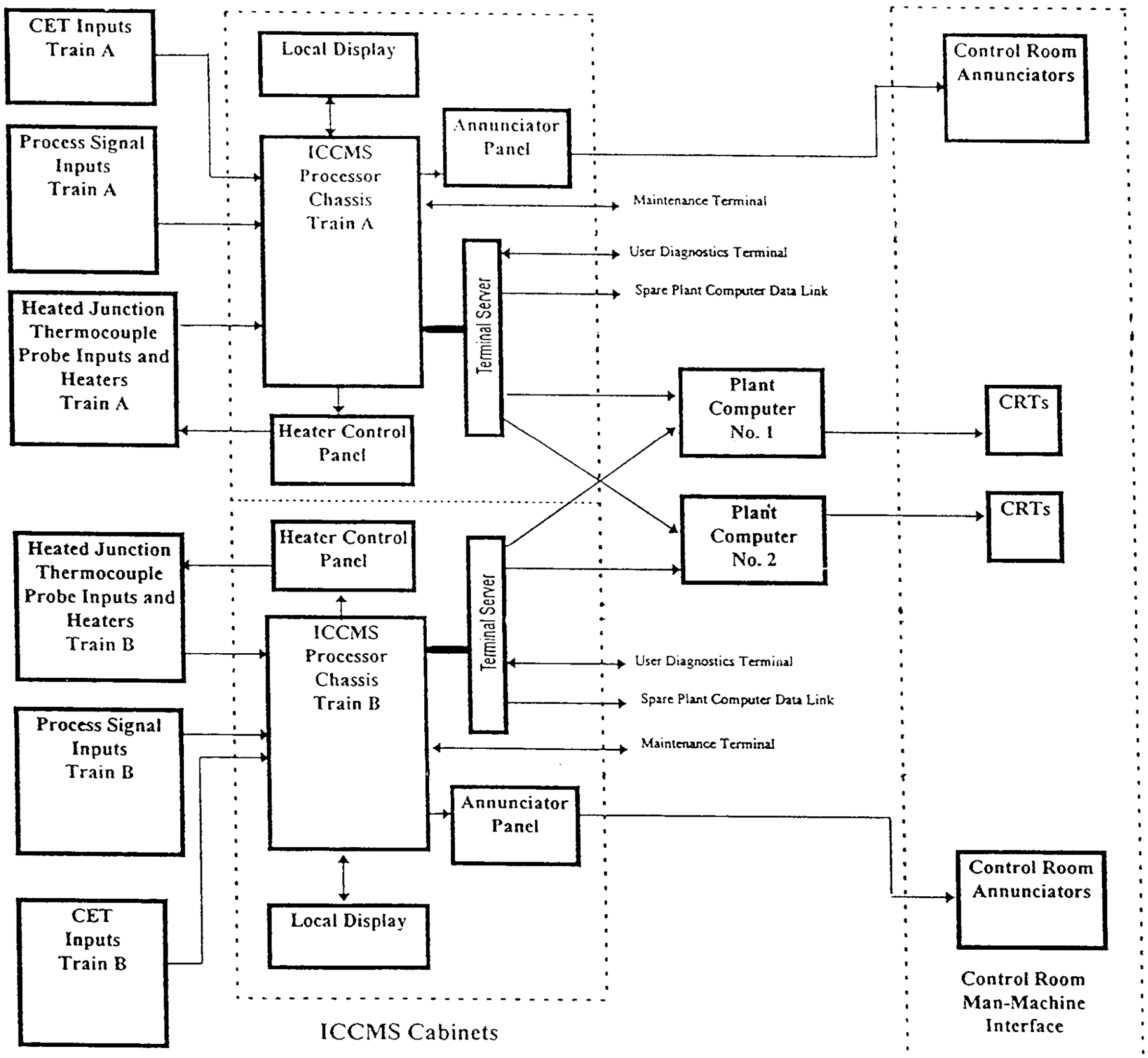

FIGURE 7.5-7
ICCMS FUNCTIONAL BLOCK DIAGRAM

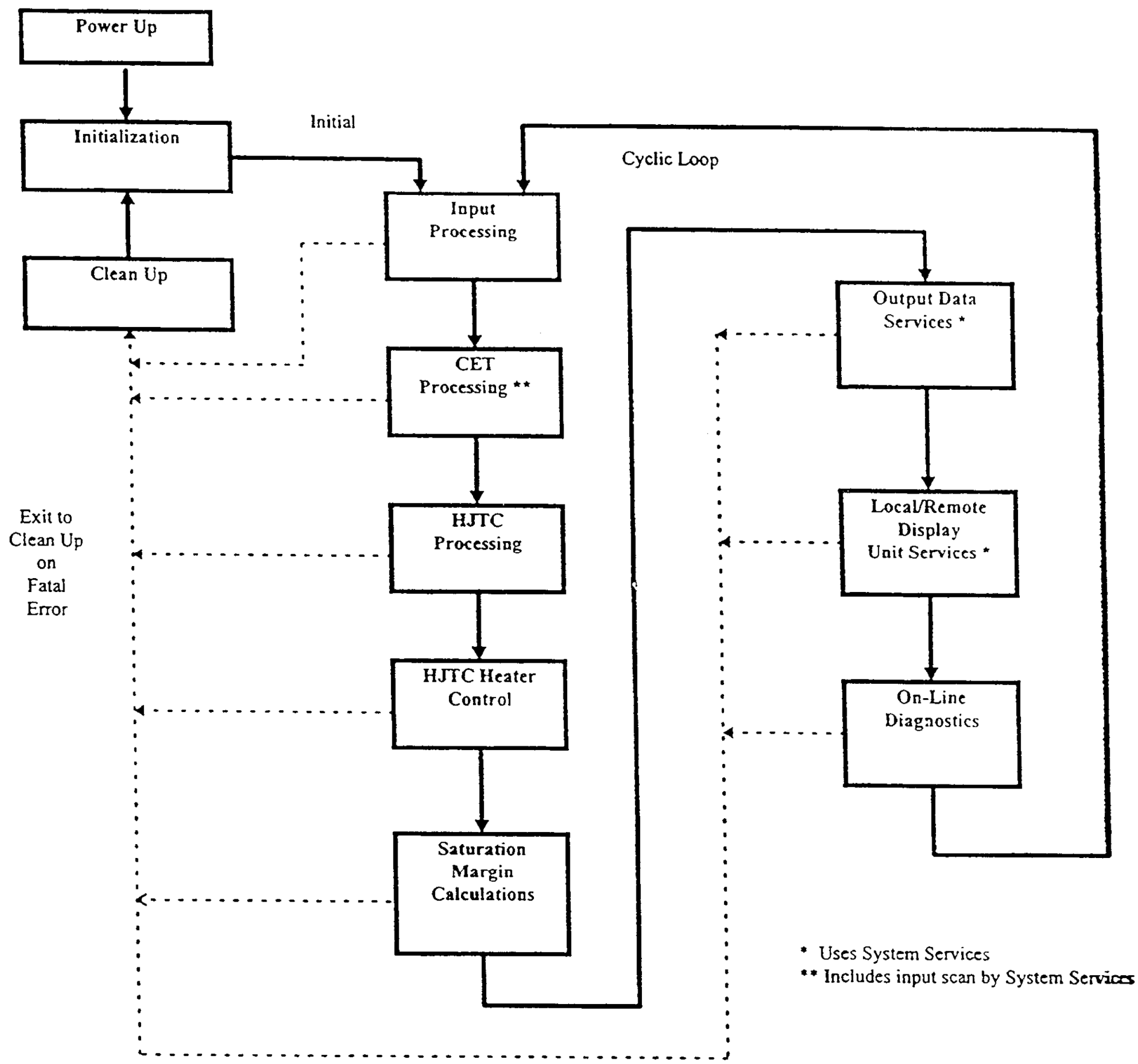

FIGURE 7.5-8
ICCMS DATA PROCESSING

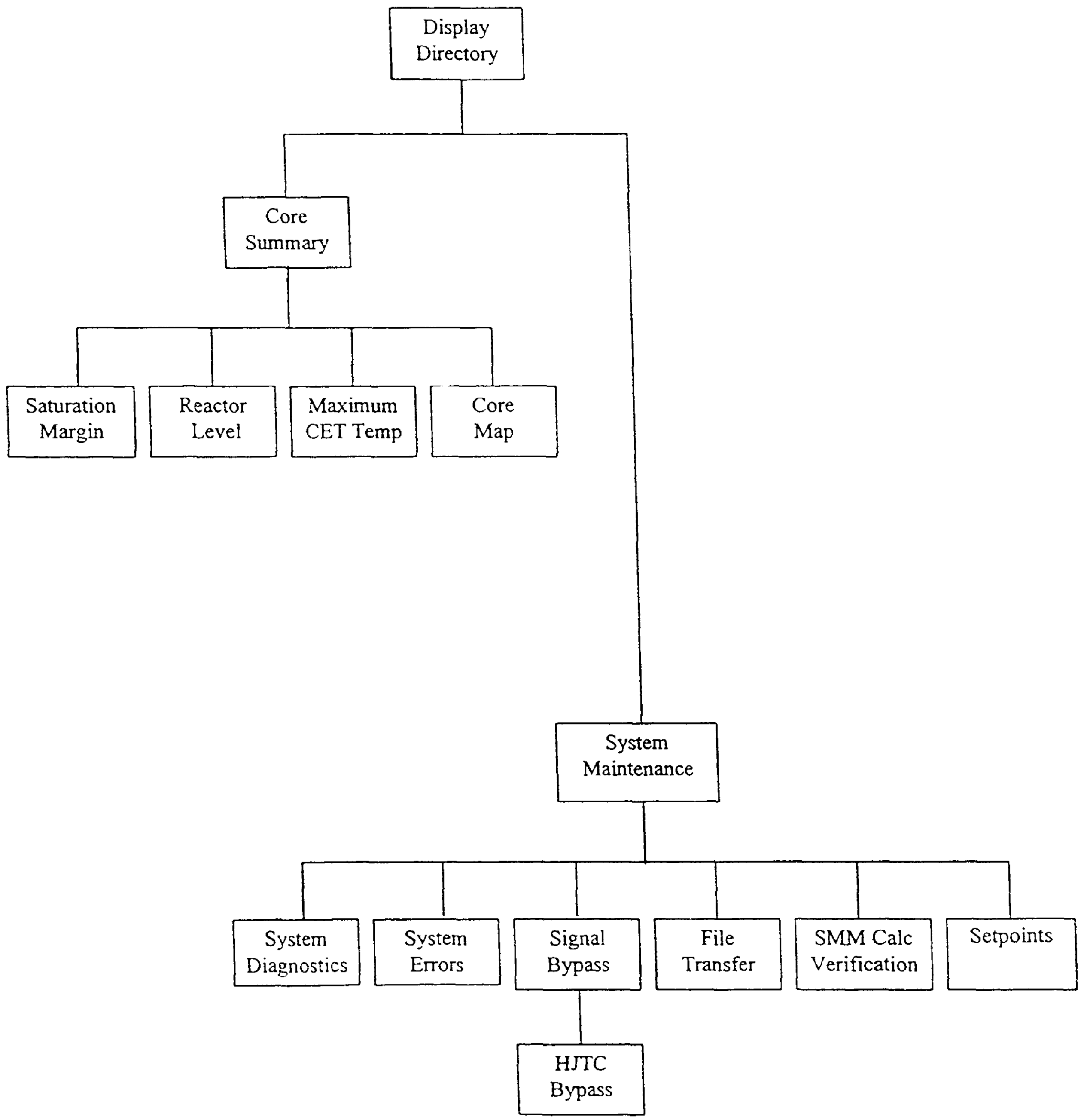

FIGURE 7.5-9
ICCMS DISPLAY HIERARCHY

7.6 OPERATING CONTROL STATIONS

7.6.1 CONTROL ROOM

7.6.1.1 Design Bases

7.6.1.1.1 Functional Requirements

The control room accommodates controls, alarms, indications, and instrumentation necessary to operate the nuclear power unit. This includes instrumentation for startup, normal operation, shutdown, and maintaining the plant in a safe condition under abnormal situations including loss-of-coolant accidents (LOCA).

7.6.1.1.2 Design Criteria

The following criteria have been implemented in the design of the control room:

a. The control room shall be equipped with adequate radiation protection to permit access and occupancy under accident conditions without personnel receiving radiation exposures in excess of 5 rem to the whole body, or its equivalent to any part of the body, for the duration of the accident as required by 10 CFR 50, Criterion 19.

b. Ventilation shall be provided to allow occupancy during and after a design basis accident (DBA).

c. Section 5.4.3 contains the structural design criteria for the auxiliary building in which the control room is located.

d. For Missile Protection, see Section 5.2.5.1.

7.6.1.2 Description

The Unit 2 control room is adjacent to the Unit 1 control room and is accessible from both the auxiliary and turbine buildings. It houses the enclosed walk-in duplex type main control boards, with integral consoles and miscellaneous instrument panels and racks, as well as its own air-conditioning and fire protection panels. All control boards which are safety related are designed to Seismic Class I requirements. In addition, the control room is equipped with separate, enclosed supervisor’s offices. Figure 7.6–1 shows the general arrangement of the control room.

The Unit 2 control room can be occupied under all credible accident conditions and is provided with redundant air conditioning systems, redundant filtration systems, an airborne radioactivity detector in the fresh air intake ductwork, and fresh air isolation dampers. A high radiation signal automatically switches the air conditioning system to the recirculation mode by closing the fresh air dampers, starting both filtration trains, and closing the exhaust dampers. The recirculation mode can also be manually actuated from the control room. Makeup outside air can be drawn through the high efficiency particulate air (HEPA) charcoal filter assembly at operator discretion if needed. An area radiation monitor is provided to indicate and alarm when a high radiation level occurs inside the control room.

The materials used in the construction of the control room will not support combustion. Electrical wiring is flame resistant. Portable CO_2 fire extinguishers are placed in readily accessible stations in the control room, and respiratory protective equipment is available to the operators at all times.

7.6.2 MAIN CONTROL BOARDS

7.6.2.1 Design Bases

7.6.2.1.1 Functional Requirements

The main control boards are designed for plant control during startup, normal operation, shutdown, and emergency operation.

7.6.2.1.2 Design Criteria

The following criteria are used in the design of the main control boards:

a. Protective systems are separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, independence requirements of the protection system.

b. Wherever redundant protective or safeguards channels of a system are provided, devices and related wiring including the incoming terminal blocks of one channel are isolated from other channels to ensure that failure of any one channel will not affect any of the remaining channels.

A Control Room Design Review (CRDR) was performed in response to Nuclear Regulation (NUREG) 0737, Supplement 1. The control panels were modified to conform to the NUREG 0700 design criteria. All devices are grouped according to their functions. Circuits are channelized and physically separated in accordance with facility codes assigned to each loop and device.

Each circuit and raceway within the board is given a unique identification and each wire is color coded for easy channel identification. For separation criteria details, see Section 8.7.3.1.

Control Room panel-mounted indicators and valve-position indicating lamps that are used to satisfy Regulatory Guide 1.97 (Rev. 2) requirements have been specifically encoded by marking the instrument labels to identify them as Regulatory Guide 1.97 indications. See Table 7.5-3 for a list of Regulatory Guide 1.97 (Rev. 2) instruments.

7.6.2.2 System Description

7.6.2.2.1 System

The main control boards are comprised of the following eight sections.

Boards Function	Designation
a. Engineered safeguards	CO 1
b. Chemical and volume control (CVCS)	CO 2
c. Reactor coolant system (RCS)	CO 3
d. Reactivity control	CO 4
e. Steam generator (SG) and feedwater control	CO 5
f. Plant auxiliaries	CO 6
g. Turbine-generator-exciter control	CO 7
h. Station service electrical	CO 8

The eight sections are arranged in an "L" shaped array from left to right in the listed order with the reactivity control boards section in the center corner (see Figure 7.6–1). The boards are enclosed walk-in duplex switchboard type with an integral bench-type control console in the front. All sections are built and analyzed to meet Seismic Class I specifications. (See Section 5.8.5 for analysis procedure.) The control board, including all mounted equipment, will remain structurally intact such that no equipment will become loose, separated, or dislocated when subjected to a design basis earthquake (DBE). Furthermore, those devices which are safety related will function during and after a DBE. Figures 7.6–2 through 7.6–8 show the control board arrangement.

Control and indicating instruments, switches and indicating lights are installed on the front face of the control boards and on console tops. Instrument power supplies, protection devices, isolation amplifiers, and miscellaneous blind instruments as well as some of the secondary instruments are mounted on both the rear face of the control boards, and the instrument racks located adjacent to the control panels. Annunciator panels are installed across the top of each of the vertical section of each board. (The annunciator is discussed in Section 7.7.)

7.6.2.2.2 Components

With the exception of a few multipoint recorders, most process instruments on the main control boards are miniature electronic type using the 10 to 50 ma and 4 to 20 ma signal levels.

All miniature recorders are mounted on racks which can be pulled out on chassis tracks for ease of maintenance.

The turbine electrohydraulic control (EHC) panel is located on the front face of CO 7 board section and is furnished as part of the General Electric (GE) turbine-generator package. Arrangement of controls and monitoring instrumentation are shown in Figure 7.6–7, Sheet 1.

Control element assembly (CEA) position indication and controls are located on the front face of CO 4 board section. A detailed description is given in Section 7.5.3.

In general, control switches for circuit breakers, pump motors, and fans are equipped with pistol-grip handles and rectangular escutcheon plates. Control switches that operate valves and dampers are typically equipped with knob operators and legend plates. Switches for voltmeters and ammeters typically have round knurled handles. Some control switches are key operated.

Switches that operate the drain valves for the main and auxiliary steam turbine systems are back-lighted push-button type. These switches are grouped together on their respective board section for ease of identification.

Indicating lights for valve position, motor operation, etc., are generally the filament type with covered lense. The colors of the lenses, as shown in the listing, indicate the status of equipment operation:

Red: Utilized to denote a component in its active state, (i.e., equipment running, valve open, breaker closed).

Green: Utilized to denote a component in its passive state, (i.e., equipment off, valve closed, breaker open).

Amber: Utilized to denote an off-normal condition, (i.e., relay trip or off-normal status of equipment or parameter).

White: Utilized to denote power available or automatic operational mode.

Blue: Utilized to denote permissive conditions for operation fulfilled.

Wire terminations, except coaxial, triaxial and certain other detector and plug-connected cables, are made with ring-tongue solderless compression-type connectors which are securely bolted to barrier-type terminal blocks.

Incoming and outgoing 10 to 50 ma and 4 to 20 ma current loop signals are terminated at Class 1E, Test/Disconnect Terminal Block Assemblies.

Each control board is equipped with two copper ground busses, one on the front and one on the rear side of the duplex control board extending the entire length with the steel structure and connected to the bus so as to effectively ground the entire structure. Circuits requiring grounding are separately and directly connected to the ground bus. Device cases not otherwise grounded are grounded through the device enclosures and the steel structure of the control board.

7.6.2.3 System Operation

The main control boards are designed to allow control of the plant during all modes of operation. This includes normal plant operation, startup and shutdown, as well as emergency operation.

7.6.2.4 Availability and Reliability

7.6.2.4.1 Special Features

For the convenience of operating personnel, all major processes are presented graphically. Instruments are either located on process mimic lines or are shown connected to certain processes by influence lines. SGs, turbines, tanks, and etc., are shown schematically in the process. Pumps and valves in most cases are represented by their respective control switches and indicating lights. All process lines and symbols are color coded.

For those pumps and valves associated with safety actuation signals that are provided with a manual override capability, override is possible only subsequent to the safety actuation signal. The switch must first be turned to the “safe” condition before it can override the safety command signal. This operating procedure is designed to prevent an operator from overriding a safety signal unintentionally. When a safety-actuated device is overridden, an indication is provided by the status panel. Override capability for the AFW System actuated components is as described in Sections 7.3.3.1 and 10.4.5.3.

Two independent sets of manual safety signal actuation switches and safety signal block switches are provided on the CO 1 engineered safeguard section. The switches are guarded so as to prevent inadvertent operation. Details of the manual actuation switch functions are given in Sections 7.3.2.3 and 7.3.2.6.

Two pairs of reactor trip pushbutton switches are provided on the CO 4 board section. Details are given in Section 7.2.3.3.12.

In addition to the position indicating lights or running lights for valves, pumps, fans and dampers, each safety-related equipment, which is automatically initiated to satisfy safety functions, is provided with a <u>white</u> and <u>blue</u> status light and a set of Engineering Safety Feature (ESF) annunciators. These windows will alarm during an ESF actuation if any safety-related component fails to relocate to its accident position.

An ESF Status Light Panel designated as COIX, is designed to provide continuous indicating of the status of ESF equipment under all normal plant operation. The position of safety-related control valve RB-402 is also continuously indicated on Panel C01X because it responds to the same CIAS closure signal as isolation valve CH-089. No other ESF position monitoring attribute is provided for valve RB-402. Also, the position of containment air radiation monitor sample line isolation valves AC-527, -528, -529 and -530 are continuously indicated on Panel C01X using a single blue and white status light for each pair of isolation valves, sample supply and return, serving each of the two redundant monitor safety trains.

The status panel is designed to meet the requirements of Institute of Electrical and Electronics Engineers (IEEE) Standard 279-1971 for removing ESF equipment from operation and indication of bypasses.

The ESF Status Panel is a free standing enclosed fabricated steel panel with rear access doors as shown in Figure 7.6–11. The status panel is located adjacent to the Engineered Safeguards Panel CO 1 (see Figure 7.6–1). The panel is divided into three separate sections with barrier plates to provide adequate separation between channels. Each piece of ESF equipment is provided with a white and a blue indicating light on the front of the panel.

a. Blue Status Light - The blue status light will light when its respective valve, pump or fan is in its "safe" position, i.e., the state of the equipment after actuation by an ESF signal. Examples would be closure of a containment isolation valve on a containment isolation actuation signal (CIAS) or starting a safety injection pump on a safety injection actuation signal (SIAS). The blue light will be continuously lit when the equipment is in its "safe" state during normal plant operation.

b. White Status Light - The white status light is located adjacent to the blue status light for each piece of ESF equipment. The white light is normally off. It is arranged to go on when power is not available to the actuating circuit due to a blown fuse, tripped or racked out circuit breaker or loss of power, or when the equipment is administratively bypassed for maintenance. In addition, "LOCK-OUT" position of key operated and safety-related switches are indicated by a white light on the status panel.

   The white lights are powered from redundant 125-volt DC vital battery circuits.

c. Status Panel Light Grouping - The white and blue status lights are grouped on the panel according to each ESF actuation signal such as CIAS, SIAS, enclosure building filtration actuation signal (EBFAS), etc. This method of display provides ease in operator verification after an ESF actuation signal that all equipment actuated by a particular ESF signal has gone to its safe state since all blue lights within that grouping should go on.

Provisions are made so that it is possible to conduct online testing of the main steam isolation trip valves to ensure that the valves are capable of performing their functions when a trip signal is initiated.

Computer inputs from instrument loops that are connected to the control boards are obtained through precision resistors connected in series in the loops. These resistors are located at the terminal blocks at each applicable board section.

7.6.2.4.2 Tests and Inspections

Continuity and dielectric tests will be conducted on each wire prior to plant operation. Additionally, each instrument channel will be given a functional test using simulated signals at the input terminals to prove the correct operation and proper polarity of all connected components in each channel.

7.6.3 RADIOACTIVE WASTE PROCESSING SYSTEM PANELS

7.6.3.1 Design Bases

7.6.3.1.1 Functional Requirements

The radioactive waste disposal system control panels are required to provide the controls, instrumentation, and alarms required to operate and monitor the waste process systems.

7.6.3.1.2 Design Criteria

The following criteria have been implemented in the design of the radioactive waste disposal system control panels.

a. Appropriate systems shall be provided for the radioactive waste systems and associated handling areas to detect conditions that may result in excessive radiation levels and to enable the operator to initiate appropriate control actions.

b. Means shall be provided for monitoring effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated incidents.

c. Same as Section 7.6.2.1.2.b.

7.6.3.2 System Description

7.6.3.2.1 System

Four radioactive waste processing control panels are provided, one for each type of radioactive waste.

Board Function	Designation
Clean liquid radioactive waste	C63
Gaseous radioactive waste	C61
Aerated liquid radioactive waste	C60
Condensate demineralizer waste	CDX

These control panels are located in the general vicinity of the respective equipment. The panels are free standing cubicles with instruments, switches and annunciators located on the front and full-height double doors in the rear. All devices are grouped according to the function and are channelized in accordance with facility codes assigned to them. Separation requirements are essentially identical to those stipulated in Section 8.7.3.1. Figures 7.6–9, 7.6–10 and 7.6–12 through 7.6-15 show the panel layout arrangements.

7.6.3.2.2 Components

The radioactive waste control panels have a mixture of pneumatic and electronic instruments, and some capillary type temperature indicating devices. With the exception of one large case recorder, all instruments are the miniature type.

The electronic instruments utilize 10 to 50 ma signals whereas the 3 to 15 psig signal is standard for the pneumatic instruments.

Each of the radioactive waste control panels is equipped with an annunciator system installed across the top of the panel. The annunciator is of solid-state design and is complete with logic modules, flasher, horn, power supply, and pushbutton switches. All alarms are annunciated at their local panel with a master alarm provided in the control room. Annunciator details are found in Section 8.7.3.1.b.

Control switches, indicating lights, terminals, terminal blocks and wiring are identical to those used for the main control boards. All instrument tubings are seamless, soft-annealed copper, one-quarter inch OD. All fittings are brass flareless compression types.

7.6.3.3 System Operation

The radioactive waste processing control panels are designed to control and monitor the disposal of radioactive wastes in a safe and efficient manner.

7.6.3.4 Availability and Reliability

7.6.3.4.1 Test and Inspection

Continuity and dielectric tests were conducted on all wires. Additionally, each instrument channel was given a functional test using simulated input signals. Pneumatic systems were also given a leakage test.

7.6.4 HOT SHUTDOWN PANEL

7.6.4.1 Design Bases

7.6.4.1.1 Functional Requirements

Numerous design features are provided to make control room inaccessibility a highly unlikely event. However, in the event the operator is forced to abandon the control room, the hot shutdown panel provides the instrumentation and controls (I&C) necessary to maintain the unit in the hot shutdown condition.

7.6.4.1.2 Design Criteria

The following criteria have been implemented in the design of the hot shutdown panel:

a. Equipment at appropriate locations outside the control room shall be provided with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls, to maintain the unit in a safe condition during hot shutdown as required by Design Criterion 19.

b. Circuits in the control room shall be maintained intact.

c. The reactor shall be tripped.

7.6.4.2 System Description

7.6.4.2.1 System

The Hot Shutdown Panel is located in the West 480 Volt Switchgear Room on the 36 feet 6 inches elevation. This panel, which is designated as C21, is normally not in use. All plant operation including emergency shutdown can be accomplished at the main control boards in the control room. However, in the event the operator is forced to abandon the control room and the reactor is tripped, it is possible for the operator to maintain the unit in the hot shutdown condition by controls and instrumentation provided on this panel. This panel is built and analyzed to meet seismic Class I specifications. (See Section 5.8.5 for analysis procedure.) The panel, including all mounted equipment, will remain structurally intact such that no equipment will become loose, separated, or dislocated when subjected to a DBE.

Major equipment normally required for hot shutdown are shown in Table 7.6-1.

The following indications and controls are provided on the hot shutdown panel:

<u>Indications</u>:

Pressurizer level

Pressurizer pressure

Steam generator level

Steam generator pressure

Steam generator auxiliary feedwater (AF)

Condensate storage tank (CST) level

Cold leg temperature

Wide range neutron flux

Letdown temperature

Letdown pressure

<u>Controls</u>:

Steam dump to atmosphere

Letdown flow

Pressurizer spray

Charging pump

Pressurizer backup heater

AF valve

AF pump

AF pump crossover valve

AF pump turbine speed

Main Steam to AF

Pump turbine stop valve

All controls and instrumentation are compatible with those provided on the main control board. The panel arrangement is shown on Figure 7.6–10.

Subsequent to a hot shutdown, it is possible to bring the unit to the cold shutdown condition safely (external to the control room) with the following additional provisions and procedures:

Boric acid gravity feed valves can be manually operated to effect boric acid flow to the charging pump suction.

Low-pressure safety injection (LPSI) pumps can be controlled by control switches provided on the associated 4,160 volt emergency switchgear cubicles.

Major equipment normally used for cold shutdown is shown in Table 7.6-2.

7.6.4.2.2 Components

Instruments on this panel are generally the miniature electronic type using the 10 to 50 ma signal. Control switches, indicating lights, terminals, terminal blocks, and wiring are similar to those used for the main control boards.

7.6.4.3 System Operation

The hot shutdown panel is provided for emergency operation only. In an event which forces evacuation of the main control room, the operators will be able to bring the plant safely to the hot shutdown condition by controls provided on this panel. It includes controls and instrumentation for the pressurizer heaters and sprays, charging pumps, and auxiliary feedwater system (AFWS).

Normally controlling instruments on this panel are set in the “By Pass” position, i.e., the main control board has direct control of the final elements. However, the system is connected in such a manner that it is possible to override the main board instruments and take control at this panel.

7.6.4.4 Availability and Reliability

7.6.4.4.1 Special Features

Since the hot shutdown panel is never used except in case of an emergency, full-height doors are provided to close off the panel front. Doors are normally closed and locked. An open door is alarmed in the control room.

To ensure maximum availability, two channels of controls and instrumentation are provided on this panel. One channel is capable of performing its function to maintain hot shutdown. Separation requirements are given in Section 8.7.3.1.

7.6.4.4.2 Tests and Inspections

To ensure the integrity and availability of the system in case of an emergency, the controls and instrumentation are inspected and functionally tested in accordance with the Technical Specification.

7.6.5 FIRE SHUTDOWN SYSTEM PANELS

7.6.5.1 Design Basis

7.6.5.1.1 Functional Requirements

The Fire Shutdown System is comprised of I&C panels which provide the means to achieve hot shutdown in the event of a fire in any single fire area. This capability is achieved through three distinct design control measures. The first control requires that the Fire Shutdown System panels are located in a different fire zone from the main control room. This will ensure that both control stations will not fail because of fire in the main control room. The second control method is to ensure that all I&Cs used for the Fire Shutdown System are electrically isolated from the I&Cs used in the main control board. This will ensure that I&Cs used at both locations will not be harmed by a control room fire. The third method is to control cable routing so that a fire in any plant fire zone cannot simultaneously cause a loss of vital indication and control at both the Fire Shutdown System panels and at the main control board.

7.6.5.1.2 Design Criteria

The Design Criteria for the Fire Shutdown System is contained in 10 CFR 50 Appendix R. A summary of the applicable criteria is given below:

- A single train of I&Cs must be available to achieve hot shutdown following a fire in a single fire zone.

- Equipment will be qualified to maintain qualification when interfacing to existing safety systems.

- The system must be electrically isolated to prevent electrical faults from affecting the equipment.

- The system shall accommodate postfire conditions where offsite power is not available for 72 hours.

- I&Cs shall functionally provide:

    - Reactivity control

    - Reactor cooling makeup

    - Decay Heat Removal

7.6.5.2 System Description

The Fire Shutdown System is divided into two distinct systems: the first system, referred to as the “Bottle-Up Panels”, provide the means to remove power from all valves which can cause a loss of primary system water inventory and secondary system steam; the second system, referred to as the “Fire Shutdown Panel”, provides the vital indication and control for critical shutdown systems.

7.6.5.2.1 Bottle-Up Panels (C70A, C70B)

The Bottle-Up Panels are located in the East 480 V Switchgear Room along the exit route from the main control room into the turbine hall. Bottle-Up Panel C70A is used to isolate Z1 control schemes and Bottle-Up Panel C70B is used to isolate Z2 control schemes.

Each Bottle-Up Panel contains five (5) isolation switches. In the “normal” position, the isolation switches connect field cables directly to the control room. This position permits control from the control room. In the “isolate” position, the isolation switches open all the field cable wires which removes all voltage sources from the valve schemes. This action forces the affected valves to their closed (failsafe) position.

The following isolation valves may be closed/isolated at the Bottle-Up Panels:

C70A (Z1 Schemes)	**C70B (Z2 Schemes)**
(2 HS's) Main Steam Isolation Valves (MSIV) (MS-64A & MS-64B)	MSIV'S (MS-64A & MS-64B) - (2 HS’s)

C70A (Z1 Schemes)	C70B (Z2 Schemes)
(1 HS) SG #1 Blowdown Isolation (MS-220A)	SG #2 Blowdown Isolation (MS-220B) -(1 HS)
(1 HS) Pressurizer Power Operated Relief Valve (PORV)* (RC-402)	Pressurizer PORV* (RC-404) - (1 HS)
(1 HS) SG #1 Atmospheric Dump Quick Open Permissive and Normal Control Operation Isolation	SG #2 Atmospheric Dump Quick Open Permissive Isolation - (1 HS)

7.6.5.2.2 Fire Shutdown Panels (C9, C10)

The Fire Shutdown Panels (C9, C10) are located in the turbine building at the fifty-four (54') foot level. Panel C9 contains instrumentation signal processing electronics, which condition field transducer signals, process the signals for meter display at Panel C10 and isolates and retransmits the signals for use in the control room. Panel C10 contains meters to display the C9 instrumentation signals and contains control switches and isolation switches. C10 switches both isolate circuits from the control room, and permit local manual control.

The Control Room Panels use facility 1 and Z1 I&C schemes to ensure that the plant can achieve hot shutdown for a fire at panels C9 and C10. For this scenario, hot shutdown can be achieved from the main control room by using the facility 1 and Z1 I&Cs.

Field I&C cables are routed directly from the instrumentation or control device to panels C9 and C10. Routing is controlled so that most of the facility 2 and Z2 cabling does not pass through the same fire zone with functionally redundant facility 1 and Z1 channels. Where the same fire zone must be shared, the Z2 cabling is wrapped in a three hour fire barrier.

At panel C10, the field signal is isolated by switches prior to cable routing for use in the main control room. These switches are two position maintained contact switches. In the “remote” position, the field cable wiring is directly connected to the control room cable which permits control scheme operation remotely from the control room or hot shutdown panel. In the “local” position, the control room connection is broken and the field cable wires are connected to a local control handswitch, a local power source and local fuses. This position permits manual control at panel C10 only.

Faults in the control scheme cable runs from panel C10 to the control room are removed by placing the scheme isolation switch into the “local” position. In the local position, local fuses and local power sources are used to assure power availability independent of any control room fusing or power sources.

The following indications and controls are provided at the Fire Shutdown Panels:

Indication:

Steam Generator number 2 Pressure

Steam Generator number 2 Level

Pressurizer Pressure

Pressurizer Level

Hot Leg Temp

Cold Leg Temp

AF Flow *

CST Level

Wide Range Nuclear Instrument *

Boric Acid Tank LVL T8A *

Boric Acid Tank LVL T8B *

Steam Generator number 2 Atmospheric Dump Pressure *

Controls:

Controller for Steam Generator number 2 Atmospheric Dump Valve (ADV) HIC 4224A *

Controller for AF Flow FW - 43B *

Charging Pump P18C Control

Charging Pump P18B Control

Charging Pump Header Isolation Valve CH - 429 *

Letdown Isolation Valve CH - 089

Charging Line Distribution Valve CH - 519 *

Auxiliary Spray Valve CH - 517

Terry Turbine Steam Supply Valve (SV - 4188)

Terry Turbine Speed Control (HS - 4192C)

* Indicator or control may not be available for an R-1 fire. Alternate methods of compliance are provided in the Millstone Unit 2, 10 CFR 50 Appendix R Compliance Report Components

I&C switches on panel C10 are generally the miniature electronic type. Indicating lights, terminals, terminal blocks, and wiring on panels C9, C10, C70A and C70B are similar to those used for the main control boards.

7.6.5.3 System Operation

The Fire Shutdown System Panels (C9, C10, C70A and C70B) can be utilized for any emergency event which requires control room evacuation. In an event which forces evacuation of the main control room, the operators will be able to bring the plant safely to the hot shutdown condition by controls provided on these panels. The fire shutdown system panels can be used for any emergency event, unlike the hot shutdown panel which would lose control features if fire damages circuits in the Control Room.

7.6.5.4 Availability and Reliability

Always available and highly reliable.

7.6.5.4.1 Special Features

Since the fire shutdown system panels are never used except in case of an emergency, full height doors are provided to close off the panel fronts. Doors are normally closed and locked. An open door is alarmed in the control room.

7.6.5.4.2 Tests and Inspections

To ensure the integrity and availability of the system in case of an emergency, the controls and instrumentation are inspected and functionally tested periodically.

7.6.6 MISCELLANEOUS LOCAL CONTROL PANELS

7.6.6.1 Design Bases

7.6.6.1.1 Functional Requirements

Local control panels for noncritical systems are located throughout the plant. Most of these local panels are part of packages furnished by mechanical and electrical manufacturers. Each panel contains the indications, controls, and alarms required for safe operation of the system.

TABLE 7.6-1 MAJOR EQUIPMENT NORMALLY USED FOR HOT SHUTDOWN

Pumps	
Auxiliary feedwater	10.4.5.4.4, 10.4.5.4.5, Table 10.4-1
Charging	9.2.2.2, Table 9.2-9
RBCCW	9.4.2.2, Table 9.4-1
Service water	9.7.2.2.2, Table 9.7-2
Reactor Coolant	4.3.3, Table 4.3-4
Valves	
Letdown	9.2.2.2, Table 9.2-4
Auxiliary feedwater regulating	10.4.5.3
Atmospheric dump	10.3.2.2, Table 10.3-1, Table 10.3-3
Pressurizer spray	4.3.7, Table 4.3-8
Storage Tanks	
Condensate storage	10.4.5.3
Heat Exchangers	
RBCCW	9.4.2.2, Table 9.4-1
Miscellaneous	
Pressurizer heaters	4.3.5
Containment cooling units	6.5
Diesel generator	8.3
Switchgear, 4160 volts	8.2
480 V emergency unit substation	8.4
480 V emergency motor control center	8.4
125 DC battery	8.5
125 VDC switchgear and distribution panels	8.5
AC/DC inverters and 120 VAC vital instrumentation buses	8.6
Reactor protection instrumentation	7.2

TABLE 7.6-2 MAJOR EQUIPMENT NORMALLY USED FOR COLD SHUTDOWN

Pumps	
Auxiliary feedwater	10.4.5.4.4, 10.4.5.4.5, Table 10.4-1
Charging	9.2.2.2, Table 9.2-9
RBCCW	9.4.2.2, Table 9.4-1
Service water	9.7.2.2.2, Table 9.7-2
Boric acid transfer	9.2.2.2, Table 9.2-11
Low pressure safety injection	6.3.2.2, Table 6.3-2
Reactor Coolant	4.3.3, Table 4.3-4
Valves	
Letdown	9.2.2.2, Table 9.2-4
Auxiliary feedwater regulating	10.4.5.3
Atmospheric dump	10.3.2.2, Table 10.3-1, Table 10.3-3
Pressurizer spray	4.3.7, Table 4.3-8
Storage Tanks	
Condensate storage	10.4.5.3
Boric acid storage	9.2.2.2, Table 9.2-10
Heat Exchangers	
RBCCW	9.4.2.2, Table 9.4-1
Shutdown	6.3.2.2, Table 6.3-3
Miscellaneous	
Pressurizer heaters	4.3.5
Containment cooling units	6.5
Diesel generator	8.3
Switchgear, 4160 volts	8.2
480 V emergency unit substation	8.4
480 V emergency motor control center	8.4

125 VDC battery	8.5
125 VDC switchgear and distribution panels	8.5
AC/DC inverters and 120 VAC vital instrumentation buses	8.6
Reactor protection instrumentation	7.2

Withheld under 10 CFR 2.390 (d)(1)

FIGURE 7.6–1 GENERAL ARRANGEMENT CONTROL ROOM PLAN ELEVATION 35 FEET 6 INCHES

[redacted]

FIGURE 7.6–2 C01, CRP, FRONT VIEW ARRANGEMENT SAFEGUARDS SECTION (SHEET 1)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–2 C01, CRP, FRONT VIEW ARRANGEMENT SAFEGUARDS SECTION (SHEET 2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–2 C01, CRP, FRONT VIEW ARRANGEMENT SAFEGUARDS SECTION (SHEET 3)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–2 C01, CRP, FRONT VIEW ARRANGEMENT SAFEGUARDS SECTION (SHEET 4)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–3 CRP FRONT VIEW ARRANGEMENTS (SHEET 1)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–3 CRP FRONT VIEW ARRANGEMENTS (SHEET 2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–3 CRP FRONT VIEW ARRANGEMENTS (SHEET 3)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–3 CRP FRONT VIEW ARRANGEMENTS (SHEET 4)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–4 C04 CRP, FRONT VIEW ARRANGEMENT REACTIVITY CONTROL SYSTEM (SHEET 1)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–4 C04 CRP, FRONT VIEW ARRANGEMENT REACTIVITY CONTROL SYSTEM (SHEET 2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–4 C04 CRP, FRONT VIEW ARRANGEMENT REACTIVITY CONTROL SYSTEM (SHEET 3)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–4 C04 CRP, FRONT VIEW ARRANGEMENT REACTIVITY CONTROL SYSTEM (SHEET 4)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–5 C05, CRP, FRONT VIEW ARRANGEMENT STEAM GENERATOR & FEEDWATER CONTROL (SHEETS 1)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–5 C05, CRP, FRONT VIEW ARRANGEMENT STEAM GENERATOR & FEEDWATER CONTROL (SHEET 2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–5 C05, CRP, FRONT VIEW ARRANGEMENT STEAM GENERATOR & FEEDWATER CONTROL (SHEET 3)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–5 C05, CRP, FRONT VIEW ARRANGEMENT STEAM GENERATOR & FEEDWATER CONTROL (SHEET 4)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–6 C06, CRP. FRONT VIEW ARRANGEMENT PLANT AUXILIARY (SHEET 1)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–6 C06, CRP. FRONT VIEW ARRANGEMENT PLANT AUXILIARY (SHEET 2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–7 C07, CONTROL ROOM PANEL, FRONT VIEW ARRANGEMENT TURBINE-GENERATOR (SHEET 1)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–7 C07, CONTROL ROOM PANEL, FRONT VIEW ARRANGEMENT TURBINE-GENERATOR (SHEET 2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–7 C07, CONTROL ROOM PANEL, FRONT VIEW ARRANGEMENT TURBINE-GENERATOR (SHEET 3)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–7 C07, CONTROL ROOM PANEL, FRONT VIEW ARRANGEMENT TURBINE-GENERATOR (SHEET 4)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–8 C08, CRP FRONT VIEW ARRANGEMENT STATION SERVICE ELECTRIC (SHEET 1)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–8 C08, CRP FRONT VIEW ARRANGEMENT STATION SERVICE ELECTRIC (SHEET 2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–8 C08, CRP FRONT VIEW ARRANGEMENT STATION SERVICE ELECTRIC (SHEET 3)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–8 C08, CRP FRONT VIEW ARRANGEMENT STATION SERVICE ELECTRIC (SHEET 4)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–9 LOCAL CONTROL PANELS (C60) & (C61)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–10 LOCAL CONTROL PANELS (C63) & (C21)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.6–11 ENGINEERED SAFETY EQUIPMENT STATUS PANEL LAYOUT (C01X)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

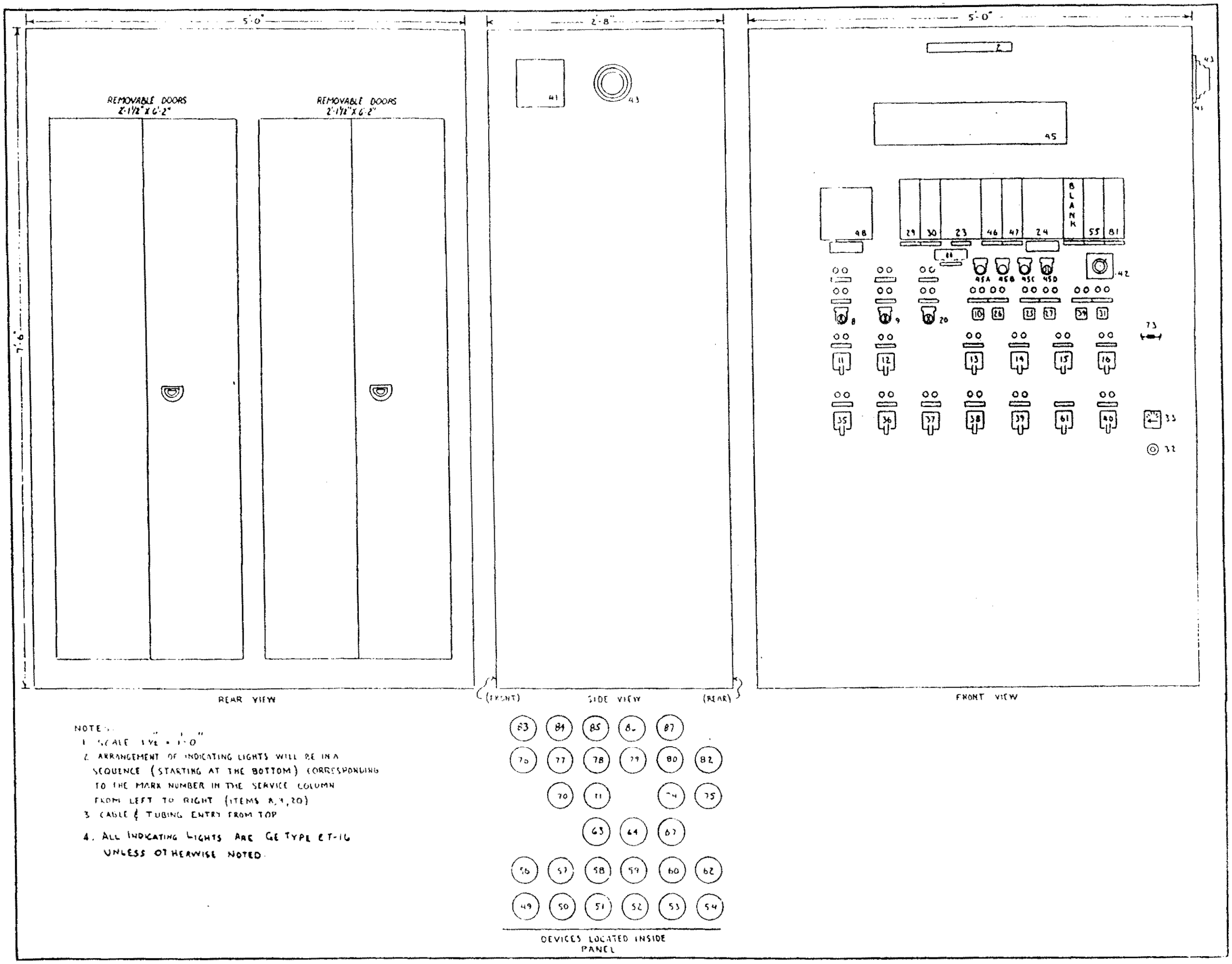

FIGURE 7.6-12
CONDENSATE DEMINERALIZER WASTE TREATING PANEL CDX (EXCERPT FROM DWG. 25213-30333 SH. 1)

ITEM NO.	MARK NO OR DEVICE NO	BY	DESCRIPTION	SERVICE	REMARKS
1	2CND-PNLCDX	S	STEEL PANEL 5'-0"W X 2'-8"D X 7'-6"H		
2		S	NAMEPLATE - CONDENSATE DEMINERALIZER WASTE TREATING PNL. 2CND-PNLCDX		
3					
3A					
3B					
3C					
3D					
4					
5					
6					
7					
8	43-2CNDN02	S	SELECTOR SWITCH ESK-3G DET AU9 GE-TYPE CR2940 W/2 SETS R&G LIGHTS	ACID NEUTRALIZATION FEED PUMP DISCHARGE VALVES 2CND-AOV242B,A	ESK-7JU
9	43-2CNDN03	S	SELECTOR SWITCH ESK-3G DET AU9 GE-TYPE CR2940 W/2 SETS R&G LIGHTS	CAUSTIC NEUTRALIZATION FEED PUMP DISCHARGE VALVES 2CND-AOV243B,A	ESK-7JV
10	PB-2CND-AOV244A	S	PUSHBUTTON ESK-3E, DET AG, CH-TYPE E30 W/ 1 SET R&G LIGHTS	WASTE NEUTRALIZING SUMP PUMP DISCHARGE VALVE AOV244A	ESK-7KT
11	1-2CND-P8	S	CONTROL SWITCH ESK-3J, DET BE, GE-TYPE SBI W/1 SET R&G LIGHTS	ACID NEUTRALIZATION FEED PUMP P8	ESK-6GQ
12	1-2CND-P10	S	CONTROL SWITCH ESK-3J, DET BE, GE-TYPE SBI W/1 SET R&G LIGHTS	CAUSTIC NEUTRALIZATION FEED PUMP P10	ESK-6GQ
13	1-2CND-P6A	S	CONTROL SWITCH ESK-3J, DET BE, GE-TYPE SBI W/1 SET R&G LIGHTS	WASTE NEUTRALIZATION SUMP PUMP P6A	ESK-6GN
14	1-2CND-P6B	S	CONTROL SWITCH ESK-3J, DET BE, GE-TYPE SBI W/1 SET R&G LIGHTS	WASTE NEUTRALIZATION SUMP PUMP P6B	ESK-6GN
15	1-2CND-P6C	S	CONTROL SWITCH ESK-3J, DET BE, GE-TYPE SBI W/1 SET R&G LIGHTS	WASTE NEUTRALIZATION SUMP PUMP P6C	ESK-6GP
16	1-2CND-P6D	S	CONTROL SWITCH ESK-3J, DET BE, GE-TYPE SBI W/1 SET R&G LIGHTS	WASTE NEUTRALIZATION PUMP PUMP P6D	ESK-6GP
17					
18					
19					
20	43-2CNDN01	S	SELECTOR SWITCH ESK-3G, DET AU9, GE-TYPE CR2940 W/2 SETS R&G LIGHTS	URC DISCHARGE VALVES TO WASTE NEUTRALIZATION PUMP 2CND-AOV241B,A	ESK-7JU
21					
22					UNUSED
23	2CND-FR246	S	ELECTRONIC 1 PEN FLOW RECORDER	WASTE NEUTRALIZATION SUMP PUMP DISCHARGE TO SEAL WELL	LOOP 2CND-246
24	2CND-AR297	S	ELECTRONIC 2 PEN PH RECORDER	WASTE NEUTRALIZATION SUMP PUMPS DISCHARGE	LOOP 2CND-297
25	PB-2CND-AOV245	S	PUSHBUTTON ESK-3E, DET AG, CH-TYPE E30 W/1 SET R&G LIGHTS	WASTE NEUTRALIZATION SUMP PP DISCHARGE VALVE TO SEAL WELL 2CND-AOV245	ESK-7JW
26	PB-2CND-AOV298A	S	PUSHBUTTON ESK-3E, DET AG, CH-TYPE E30 W/1 SET R&G LIGHTS	WASTE NEUTRALIZATION SUMP PP DISCHARGE RECIRC. VALVE 2CND-AOV298A	ESK-7JX
27	PB-2CND-AOV299	S	PUSHBUTTON ESK-3E, DET AG, CH-TYPE E30 W/1 SET R&G LIGHTS	WASTE NEUTRALIZATION SUMP PP TO CONDENSATE DEMINERALIZER LIQUID WASTE	ESK-7JW
28					
29	2CND-LI237A	S	ELECTRONIC LEVEL INDICATOR	WASTE NEUTRALIZATION SUMP (TK 10)	LOOP 2CND-237
30	2CND-LI237B	S	ELECTRONIC LEVEL INDICATOR	WASTE NEUTRALIZATION SUMP (TK 11)	LOOP 2CND-237
31	PB-2CND-AOV298B	S	PUSHBUTTON ESK-3E, DET AG, CH-TYPE E30 W/1 SET R&G LIGHTS	WASTE NEUTRALIZATION SUMP PP DISCHARGE RECIRC. VALVE 2CND-AOV298B	ESK-7JX
32		S	RECEPTACLE, CANNON TYPE XLR-4-13N	MAINTENANCE COMMUNICATION STATION	2570.000-267-008A
33		S	SELECTOR SWITCH, GAI-TRONICS SERIES 100, 4P, 6POS	MAINTENANCE COMMUNICATION STATION	2570.000-267-008A
34	PB-2CND-AOV244B	S	PUSHBUTTON ESK-3E, DET AG, CH-TYPE E30 W/1 SET R&G LIGHTS	WASTE NEUTRALIZATION SUMP PUMP DISCHARGE VALVE AOV244B	ESK-7KT
35	1-2CND-P11A	S	CONTROL SWITCH ESK-3J, DET BE, GE-TYPE SBI W/1 SET R&G LIGHTS	ACID FEED PUMP P11A	ESK-6HG
36	1-2CND-P11B	S	CONTROL SWITCH ESK-3J, DET BE, GE-TYPE SBI W/1 SET R&G LIGHTS	ACID FEED PUMP P11B	ESK-6HG
37	1-2CND-P12A	S	CONTROL SWITCH ESK-3J, DET BE, GE-TYPE SBI W/1 SET R&G LIGHTS	CAUSTIC FEED PUMP P12A	ESK-6HH
38	1-2CND-P12B	S	CONTROL SWITCH ESK-3J, DET BE, GE-TYPE SBI W/1 SET R&G LIGHTS	CAUSTIC FEED PUMP P12B	ESK-6HH

LEGEND

- △ CHANGE, ISSUE NUMBER INSIDE
- △ COMPLETE ADDITION
- △ COMPLETE DELETION
- S FURNISHED BY SELLER
- E FURNISHED BY ENGINEERS
- * EQUIPMENT PROVIDED BY INFILCO DEGREMONT INC.

NOTES:

1. ALL ELECTRONIC BISTABLES WILL HAVE A CONTACT OUTPUT ISOLATOR. AN ITE J10 RELAY WILL ALSO BE USED IN CASES WHERE THE ELECTRONIC BISTABLE IS USED FOR A CONTROL FUNCTION. REFER TO SKETCH 12552-I-1.

FIGURE 7.6-13

CONDENSATE DEMINERALIZER WASTE TREATING PANEL CDX (EXCERPT FROM DWG. 25213-30333 SH. 2)

ITEM NO.	MARK NO OR DEVICE NO	BY	DESCRIPTION	SERVICE	REMARKS
39	1-2CND-P7A	S	CONTROL SWITCH ESK-3J, DET 8E, GE-TYPE SBI W/1 SET R&G LIGHTS	WATER RECOVERY SUMP PUMP P7A	ESK-6HF
40	1-2CND-P7B	S	CONTROL SWITCH ESK-3J, DET 8E, GE-TYPE SBI W/1 SET R&G LIGHTS	WATER RECOVERY SUMP PUMP P7B	ESK-6HF
41		S	HORN, EDWARDS FLUSH-TYPE CAT. NO. 871-P1	ANNUNCIATOR	
42		S	TIMER, EAGLE MODEL 191-08-A6-00	40 MIN RANGE, STANDARD; PUSHBUTTON RESET TIMER	
43		S	BUZZER, EDWARDS ADAPTABUZZER CAT.NO. 340A-N5	TIMER	
45	2CND-ANN01	S	ANNUNCIATOR BETA SERIES 1211-A		2 HIGH X 8 WIDE
45A	PB	S	PUSHBUTTON; ESK-2K, DET BQ GE-TYPE CR2940	ANN "SILENCE"	
45B	PB	S	PUSHBUTTON; ESK-2K, DET BQ1 GE-TYPE CR2940	ANN "ACKNOWLEDGE"	
45C	PB	S	PUSHBUTTON; ESK-2K, DET BQ2 GE-TYPE CR2940	ANN "RESET"	
45D	1	S	CONTROL SWITCH; ESK-3P, DET CS1 GE-TYPE CR2940	ANN "LAMP TEST-GROUND TEST"	
46	2CND-AI297A	S	ELECTRONIC PH INDICATOR	WASTE NEUTRALIZATION SUMP (TK10) PUMP DISCHARGE	LOOP 2CND-297
47	2CND-AI297B	S	ELECTRONIC PH INDICATOR	WASTE NEUTRALIZATION SUMP (TK11) PUMP DISCHARGE	LOOP 2CND-297
48	2CND-LR237	N	ELECTRONIC 2 PEN RECORDER; L&N SPEEDOMAX H MARK III	WASTE NEUTRALIZATION AND WATER RECOVERY SUMPS	LOOP 2CND-237
49	2CND-LY237A	S	SIGNAL CONDITIONER I/E	2CND-LS237A1, A2 & 2CND-LT237A	LOOP 2CND-237
50	2CND-LY237B	S	SIGNAL CONDITIONER I/E	2CND-LS237B1, B2 & 2CND-LT237B	LOOP 2CND-237
51	2CND-LS237A1	S	ELECTRONIC BISTABLE	WASTE NEUTRALIZATION SUMP (TK10) LEVEL ALARM LO	LOOP 2CND-237
52	2CND-LS237A2	S	ELECTRONIC BISTABLE	WASTE NEUTRALIZATION SUMP (TK10) LEVEL ALARM HI	LOOP 2CND-237
53	2CND-LS237B1	S	ELECTRONIC BISTABLE	WASTE NEUTRALIZATION SUMP (TK11) LEVEL ALARM LO	LOOP 2CND-237
54	2CND-LS237B2	S	ELECTRONIC BISTABLE	WASTE NEUTRALIZATION SUMP (TK11) LEVEL ALARM HI	LOOP 2CND-237
55	2CND-LI300	S	ELECTRONIC LEVEL INDICATOR	WATER RECOVERY SUMP	LOOP 2CND-300
56	2CND-LY300	S	SIGNAL CONDITIONER I/E	2CND-LS300A, B, & 2CND-LI300	LOOP 2CND-300
57	2CND-LS300A	S	ELECTRONIC BISTABLE	WATER RECOVERY SUMP LEVEL ALARM HI	LOOP 2CND-300
58	2CND-LS300B	S	ELECTRONIC BISTABLE	WATER RECOVERY SUMP LEVEL ALARM LO	LOOP 2CND-300
59	2CND-AS297A	S	ELECTRONIC BISTABLE	WASTE NEUTRALIZATION SUMP (TK10) PH LEVEL HI ALARM	LOOP 2CND-297
60	2CND-AS297B	S	ELECTRONIC BISTABLE	WASTE NEUTRALIZATION SUMP (TK11) PH LEVEL HI ALARM	LOOP 2CND-297
61	43-2CND-P7A	S	SELECTOR SWITCH; ESK-3K DET 8H GE TYPE SBI	WATER RECOVERY SUMP PUMP P7A / P7B	ESK-6HF
62	74-2CNDN15	S	ALARM RELAY: ITE J13PA3012, 125VDC (3 NO. CONTACTS)	CONDENSATE DEMINERALIZER AUX CKT LOSS OF CONTROL POWER	ESK-7NY
63	86-AOV245		AUXILIARY RELAY; ITE J13P0612; 125VDC (6 NC. CONTACTS)	WASTE NEUTRALIZATION SUMP DISCHARGE VALVE	(23K-7JU) -32810 SH.30
64	86-AOV299		AUXILIARY RELAY; ITE J13P6012; 125VDC (6 NO CONTACTS)	WASTE NEUTRALIZATION SUMP TO LIQUID WASTE SYSTEM	(23K-7JW) -32810 SH.30
65					
66					
67	71X5-2CNDN15	S	AUXILIARY RELAY; ITE J13P3012, 125VDC (3 NO. CONTACTS)	CONDENSATE DEMINERALIZER AUX CKT	ESK-7NY
68					
69					
70	71X8-2CNDN15	S	AUXILIARY RELAY: ITE J13P3012, 125VDC (3 N.O. CONTACTS)	CONDENSATE DEMINERALIZER AUX CKT	ESK-7NY
71	71X9-2CNDN15	S	AUXILIARY RELAY; ITE J13P3012, 125VDC (3 N.O. CONTACTS)	CONDENSATE DEMINERALIZER AUX CKT	ESK-7NY
72					
73		S	HANDSET HANGUP HOOK GIA-TRONICS PART NO 5884	MAINTENANCE COMMUNICATION STATION	
74	2CND-FY246	S	SIGNAL CONDITIONER I/E	2CND-FR246	LOOP 2CND-246
75	2CND-LS318A	S	ELECTRONIC BISTABLE	CAUSTIC STORAGE TK LEVEL LOW-LOW INTLK P12A,B	LOOP 2CND-318
76	2CND-LS318B	S	ELECTRONIC BISTABLE	CAUSTIC STORAGE TK LEVEL ALARM LOW	LOOP 2CND-318
77	2CND-LS322A	S	ELECTRONIC BISTABLE	ACID STORAGE TK LEVEL LOW-LOW INTLK P11A,B	LOOP 2CND-322

FIGURE 7.6-14
CONDENSATE DEMINERALIZER WASTE TREATING PANEL CDX (EXCERPT FROM DWG. 25213-30333 SH. 3)

ITEM NO.	MARK. NO OR DEVICE NO	BY	DESCRIPTION	SERVICE	REMARKS
78	2CND-LS322B	S	ELECTRONIC BISTABLE	ACID STORAGE TK LEVEL ALARM LOW	LOOP 2CND-322
79	2CND-LY318	S	SIGNAL CONDITIONER I/E	2CND-LS318A,B	LOOP 2CND-318
80	2CND-LY322	S	SIGNAL CONDITIONER I/E	2CND-LS322A,B	LOOP 2CND-322
81	2CND-LI318A	E	ELECTRONIC 2 POINT LEVEL INDICATOR	ACID & CAUSTIC STORAGE TANKS	LOOP 2CND-318
82	74-2CND-PNLCDX	J	AUXILIARY RELAY; ITE J13PA3012, 125VDC (3 N.O. CONTACTS)	2CND-PNLCDX LOSS OF 125VDC TO PANEL ALARM	ESK-7NV
83	7IX10-2CNDN15	J	AUXILIARY RELAY; ITE J13PA3012 125VDC (3 N.O. CONTACTS)	CONDENSATE DEMINERALIZER AUXILIARY CIRCUIT	ESK-7NV
84	7IX11-2CNDN15	J	AUXILIARY RELAY; ITE J13PA3012 125VDC (3 N.O. CONTACTS)	CONDENSATE DEMINERALIZER AUXILIARY CIRCUIT	ESK-7NV
85	7IX12-2CNDN15	J	AUXILIARY RELAY; ITE J13PA3012 125VDC (3 N.O. CONTACTS)	CONDENSATE DEMINERALIZER AUXILIARY CIRCUIT	ESK-7NV
86	2CND-FY346A	J	SIGNAL CONDITIONER (SQ. ROOT INTEGRATOR)	WASTE NEUTRALIZING SUMP DISCHARGE TO SEAL WELL	LOOP 2CND-346
87	2CND-FY346B	J	SIGNAL CONDITIONER (PULSE CONVERTER)	WASTE NEUTRALIZING SUMP DISCHARGE TO SEAL WELL	LOOP 2CND-346
88	2CND-FQ346	J	FLOW TOTALIZER	WASTE NEUTRALIZING SUMP DISCHARGE TO SEAL WELL	LOOP 2CND-346

FIGURE 7.6-15

CONDENSATE DEMINERALIZER WASTE TREATING PANEL CDX (EXCERPT FROM DWG. 25213-30333 SH. 4)

7.7 CONTROL ROOM ANNUNCIATION

7.7.1 DESIGN BASES

7.7.1.1 Functional Requirements

The Main Control Board Annunciator provides the operator with visual and audible indications from external contacts if a limiting condition is being approached or abnormal conditions exist for any system so annunciated.

7.7.1.2 Design Criteria

The Main Control Board Annunciation System has been designed to meet the following criteria:

1. Respond to a permanent, fixed time momentary, or time dependent momentary alarm condition.

2. Via visual and audible devices (lights and electronic horns) indicate this response on the front panels of the Control Board.

3. By visual and audible devices indicate a return to normal condition for the point in alarm.

4. Provide means to silence the initial alarm horn and still maintain a bona fide alarm indication via lights.

5. Provide means of testing alarm lights without interfering with the normal status of the various systems in the plant.

6. Provide auxiliary contact outputs to the plant computer.

7. Provide means to silence audible alarms after a reactor trip via a Master Silence Switch.

7.7.2 SYSTEMS DESCRIPTION

7.7.2.1 System

The annunciator windows are installed in panels across the top front of each board section. The annunciators in each panel are grouped in accordance with the function of each board section, in general, i.e., alarms for Engineered Safeguards on CO1, Chemical & Volume Control Systems alarms on CO2, etc. Board sections CO1, CO4, CO5, and CO8 have individual electronic horns and alarm push-button controls and horn.

The annunciator logic is of solid-state design, and housed in free-standing cabinets located external to the Main Control Board. The system is powered by a primary AC to DC power supply fed from IAC VR11 and a backup AC to DC power supply fed from IAC VR21, which are both backed by separate UPS batteries capable of carrying their respective loads for a minimum of one hour. IAC VR11 and IAC VR21 will automatically swap to their respective diesel generator backed AC source in the event of a failure of the UPS. This ensures the annunciator system with a reliable power source. The voltage across the field contacts is 125 VDC except for parts of nine circuits which go through the RPS cabinets. Interposing relays were installed in those circuits to lower the voltage to 28 VDC in the RPS cabinets and to reduce potential common mode noise. The annunciator's DC voltage is isolated from the IAC power supplies through the two redundant AC to DC converters.

Ground detectors are provided to alarm whenever a ground exists on any annunciator contact input circuit or any annunciator power supply. Portions of nine circuits which go through the RPS cabinets are not monitored by ground detectors. These circuits are protected by fuses. The assembled annunciators are tested per NEMA Standard ICS-2.42.

The Master Silence Switch is located in the Control Room at the SCO's desk (C17C).

7.7.3 OPERATION

The annunciator is equipped with six sets of push buttons; their function and operation are as follows:

"Acknowledge" push button — Upon receipt of an alarm, i.e., field contact off norm, depressing the "acknowledge" push button on the corresponding control panel changes the flashing window to steady and silences the audible alarm.

"Silence" push button — Depressing any one of the "silence" push buttons silences all alarm horns should the operator choose to do this prior to acknowledging an alarm. "Silence" push buttons do not operate to silence the audible tone that sounds when the field contacts return to normal.

"Reset" push button — After an alarm has been acknowledged and the field contacts have returned to normal, depressing the "reset" push button on the corresponding control panel section causes the slowly flashing bright window to change to steady dim lighted and silences the audible tone.

"Test" push button — Depressing the "test" push button on the corresponding control panel section shall perform a functional test of all annunciator alarm points and the audible alarm on that panel.

The alarm sequence is as follows:

Condition	Visual	Audible
Field contact normal	Dim lighted	Off
Field contact off-normal	Fast flashing bright	On
Field contact off-normal Silence Button depressed	Fast flashing bright	Off
Field contact return to normal before acknowledge	Fast flashing bright	On
Alarm acknowledged	Steady bright if field contact off-normal	Off
Field contact returned to normal after acknowledge	Slow flashing bright	On *
Field contact returned to normal, alarm reset	Dim lighted	Off

* Different Tone

Master Audible Silence Switch will turn off all audible alarms in "silence" mode.

Major system alarms are described with systems in their appropriate section.

7.8 COMMUNICATION SYSTEMS

7.8.1 DESIGN BASES

7.8.1.1 Functional Requirements

Communication systems are provided to meet the requirements for operation and maintenance of this generating unit. Further provision is made for routine and emergency communications between the unit operator and outside locations such as the system operator and public authorities.

7.8.1.2 Design Criteria

Industry standards for communication are observed, and precautions are taken so no failure of these systems will compromise the proper functioning of any protective system. Redundancy and separation are provided between the dial telephone system, the public address system, and the "walkie-talkie" radios. The power supply for all communication systems is from a dependable AC or battery source.

7.8.2 SYSTEM DESCRIPTION

7.8.2.1 Systems

a. Public Switched Network (off-site Dial Telephone System)

b. Intraplant Private Branch Exchange (PBX) telephones

c. Microwave system

d. Multiplexing/SONET System

e. Radio facilities

f. Carrier current

g. Maintenance system

h. Fuel handling system

i. Public address system

j. Evacuation alarm

7.8.2.2 Components

a. <u>Public Switched Network</u>

The public switched network is operated by various telephone companies and connects various outside agencies.

b. Intraplant Private Branch Exchange (PBX) Telephones

The intraplant switching network or private branch (PBX) is a telephone system consisting of standard telephones, multiline telephones, pico cellular phones, and a digital PBX.

c. Microwave System

A microwave system provides all three generating units at the Millstone site with an extremely reliable telecommunications medium. The microwave system links the Millstone site to other utility companies throughout New England.

The microwave system uses low-power radio signals that operate in frequency bands established for industrial users by the Federal Communications Commission. These frequency allocations fall in the 2, 6, and 18 GHz industrial microwave frequency bands. Two types of microwave communications equipment are in use:

Analog Frequency Division Multiplex Equipment

The Analog Frequency Division Multiplex (FDM) microwave equipment uses frequency modulation techniques to place the information that is being sent on the microwave radio. The amount of information that may be placed on the microwave system is set by Federal Communications Commission Rules and Regulations (Part 94) to be equivalent to 480 voice telephone channels. A voice channel is interpreted as a balanced four-wire circuit (2 wires for send and 2 wires for receive) which passes audio signals in the voice frequency range (300 Hz to 3,400 Hz) and has output and input impedances of 600 ohms. Also included with each voice channel is another nonvoice circuit which is referred to as an out-of-band signaling channel. The purpose of this channel is to reproduce contact type signals such as a phone being dialed or a telephone handset being lifted from the phone hookswitch.

The type of telecommunications traffic that is placed on the microwave system is the same type that would normally be placed on a dedicated, 4 wire, data grade telephone circuit. This would include some of the following uses:

- Dial repeating tie trunks or “tie lines” that connect the telephone PBX at one location within the system to a similar PBX at another location.

- Automatic ring down circuits for use as “hot line” dedicated phones; where lifting a phone at one end will cause the phone on the other end of the circuit to ring.

- Radio control circuits that provide control of remotely located radio transmitters from key areas within the Millstone complex. This includes radio control circuits which provide one-way control as required by radio paging transmitters as well as control circuits that provide two-way control for standard mobile radio operation.

- Data circuits that connect one computer with another or allow data gathering equipment to communicate with a central "host" computer.

- Data circuits that carry analog data also can benefit from the greater reliability offered by the microwave system. This type of telecommunications traffic includes telemetering of important analog quantities and reporting alarms that are remote from the Millstone site.

- Data circuits, which are used for protective relaying signals, provide the electric generating and transmission system with protection from catastrophic failure.

Digital Time Division Multiplex Equipment

The Digital Time Division Multiplex (TDM) microwave equipment uses amplitude modulation techniques to place the information that is being sent out on the microwave radio. The digital microwave system provides all of the capabilities offered by the analog microwave system with the addition of high-speed data channels that are capable of transmitting and receiving data at a rate of up to 56,000 bits per second. This data rate is very valuable when large blocks of data have to be transferred from one computer to another. The digital microwave system also provides digital service over North American Standard Digital Services - first level (DS-1) at 1.544 megabits per second. In addition to the high speed data handling ability offered by the DS-1 signal path, voice traffic can also be encoded and placed on DS-1 circuits in groups of up to 24 voice channels.

d. The Multiplexing/SONET system connected in a diverse ring configuration (fiber optic cable) and multiplexing equipment supports the station.

e. Radio facilities consist of multiple separable systems available to the unit operator.

f. Plug-in headsets for carrier current voice transmission over each of the three interconnecting 345 kV lines.

g. Equipment for the maintenance system consists of directly connected amplified outlet jacks wired to cover each of several working areas, with five channel selector switches at each station. Portable instruments consist of headsets with boom mounted transmitter, each with a plug to match the outlet jacks.

h. The fuel handling system provides a noise-canceling, dynamic type headset with amplifier mounted in enclosures located throughout the fuel handling area inside and outside the containment structure. This system includes an instrument on the reactor polar crane.

i. The public address system, manufactured by Gai-Tronics, consists of permanently installed loudspeakers throughout the unit. Cone type speakers are used indoors, and weather proof re-entrant horn type speakers are used outdoors. Each loudspeaker has its own integral amplifier. An amplifier failure would affect only its associated loudspeaker, permitting normal use of all others.

j. The evacuation alarm consists of audio frequency oscillators that supply a distinctive tone signal to the public address loudspeakers located throughout the unit. An ambient noise level device assures that the output sound level is sufficiently high for each location.

7.8.3 SYSTEM OPERATION

7.8.3.1 Normal Operation

a.& b. The intraplant switching network is directly coupled to the public switched network. Additionally, there is a Federal Telecommunication System (FTS 2000) installed. This system is a federal dial telephone network which is independent of the plant private branch exchange (PBX). It is also independent of the public switched network. FTS 2000 telephones are installed as follows:

- Emergency Notification System (ENS) - Unit Control Room, Technical Support Center (TSC), and Emergency Operating Facility (EOF)

- Reactor Safety Counterpart Link (RSCL) - EOF and TSC

- Management Counterpart Line (MCL) - EOF and TSC

- Local Area Network (LAN) - EOF and TSC

- Protective Measures Counterpart Link (PMCL) - EOF and Health Physics Network (HPN) - EOF

This network allows telephone calls to be made to the NRC, both in Bethesda and Region One.

c. Microwave System

The microwave equipment at the Millstone site interfaces with the remainder of the NU microwave telecommunications system through an active microwave repeater site located in Haddam, Connecticut. The Millstone site and all other microwave sites within the system are designed to function in the most hostile of weather conditions. The microwave antenna and tower equipment is designed to remain fully operational with a 40 pound per square foot wind load and a 0.5 inch of radial ice. This is equivalent to loading that results from a sustained 100 mile per hour wind with all system component dimensions exaggerated by the 0.5 inch of ice at all points plus the additional weight generated by the formation of ice 0.5 inch thick. The survival rating for equipment is, in actuality, greater than the rate corresponding to conditions described above.

Additionally, all sites are fenced and equipment is operated from 24 or 48 V DC power which is provided by high quality lead-calcium batteries which are float charged by industrial grade AC powered battery charges. The batteries are sized to provide complete power requirements for the microwave equipment for a period of 12 hours. The batteries are backed up by an uninterruptible power supply at Millstone site and by propane fueled generators at remote microwave sites.

The microwave system provides the Millstone site with two additional telecommunications networks which are completely separate from the off-site telephone system. The use of these diverse systems to share the telecommunications requirements of the Millstone site results in enhanced telecommunications reliability because a failure of either system does not completely interrupt off site telecommunications traffic. The microwave system also allows Millstone to access a modern telephone PBX located approximately 50 miles from the site at the NU headquarters in Berlin, Connecticut. In an emergency situation, NU personnel would be able to displace less critical microwave channels with the additional traffic from the Millstone site.

d. Multiplexing/SONET Systems

Millstone is supported by a Synchronous Optical Network (SONET) connected in a diverse ring configuration (fiber optic cable) and by multiplexing equipment. This equipment is located in the CPF building and in Building 475. Telecommunication traffic placed on the SONET system is the same type that would be placed on the microwave system. The multiplexer has the capability to reroute the traffic assigned through the different media: the SONET terminal, the microwave system and a leased circuit.

The SONET terminal is powered by a 48 VDC source with battery backup. The batteries can provide backup power for a period of 35 hours.

e. Multiple Radio Systems

The multiple radio systems include the following communication systems:

- O&M/security/trunked radio system

- Connecticut Valley Electric Exchange (CONVEX) Command Control Network (CCN);

- Waterford Police System;

- Tri-town UHF radio system;

- State Police system;

- Very High Frequency (VHF) radio paging system.

A dedicated radio remote-control console is provided in the Millstone 2 control room for communications with all associated onsite as well as offsite radio facilities (as outlined above). Its power source is lighting panel LPCI, a highly reliable non-vital panel powered by the Emergency Diesel Generator backed Computer Power Inverter D50A. Normally, all radio systems, except the unit's O&M system, are quiet to the unit operator unless selected by the operator for monitoring or operation. Tone alert, except on the O&M system, is provided to enable remotely located radio dispatchers to contact the control room operator.

The radio console installed in the Millstone 2 control room consists of two individual bays secured together as a consolidated unit. The total length of the equipment is 46 inches with a height of 43.75 inches and an overall depth of 29.5 inches. The console is an equipment enclosure housing audio amplifiers (T/R modules), tone generators (encoders), tone decoders, and dual power supplies.

The console generates low level audio and DC voltages only, for the single purpose of controlling remotely located base station radios.

The radio control center equipment is mounted in a single width housing with a beveled front, projected writing surface and panel turret.

The power supplies and termination panels for the control consoles are located in the lower portion of the equipment housing. Provisions are included on the rear-hinged termination panel for securing cable entries. The two upper bays contain the heart of the console radio control system. These two bays can be considered as left center and right center as viewed from the operator side of the console. The left side of the console contains the controls for police, site security, tri-town, operations/maintenance radios, and master control module. The emergency alert paging system occupies the right half of the console. The radio control panel is mounted directly in front of the radio dispatcher’s position. This provides the interface functions required between the operator and the console (microphone, speakers, volume controls, push-to-talk switches).

The microphone is a moving coil, dynamic unidirectional, that is uniform with frequencies of 80 to 13,000 Hz. The microphone is adjustable vertically and horizontally to accommodate different operators and is internally rubber-vibration-isolated to avoid physical damage.

The console contains two power supplies - a low voltage supply and a high voltage supply, each with an input voltage of 120 VAC. The low-voltage supply provides +24 VDC and is capable of handling up to 24 radio channels. It includes a nominal +13.8 VDC ±10 percent regulator which, in conjunction with an overcurrent protection circuit, can provide a maximum continuous output current of 1 amp. The power supply has an output current capability of 8 amps. The high voltage supply provides +175 VDC for keying up to 24 DC controlled radios.

The console contains 15 audio amplifiers (T/R modules) with expansion capability of 15 future modules. One T/R module is used with each radio control channel. The module contains both logic circuits and receive/transmit audio circuits. The logic circuits include channel select, keying, busy, and priority functions. The receive audio circuits include speech processing (using an audio compressor circuit), muting, audio gating, and a voice enabled call indicator. The transmit audio circuits include a preamplifier, tone mixing amplifier, gating circuits, and a transmit audio line driver.

There are three tone generators (encoders) with external pushbutton operator controls located in the console. A touch-tone encoder allows standard touch code 2-frequency tone codes to be transmitted from the communications console. It can be used wherever a coded signal is required for selective calling or data transmissions. The encoder front panel includes a light emitting diode (LED) indicator which alerts the operator that the transmission of a code can proceed. A programmable timing circuit automatically resets the encoder and unkeys the transmitter if the tone sequence is not entered within a predetermined time. The encoder and transmitter automatically reset if the operator fails to complete a code entry. All codes generated by the encoder are compatible with standard touch-tone equipment. A two tone sequential tone generator allows encoding pocket pagers and fixed receive monitors. The operating controls and indicators are located on the front panel of the unit. The encoder has 16 pushbuttons, a four digit call code display, a call indicator, and a talk indicator. A code is manually punched into the encoder keyboard and the sequence is automatically sent whenever desired. The transmitter stays on the air for a predetermined amount of time after the code transmittal terminates in order for a voice message to be sent out to the desired pager or monitor.

The console contains eight touch-tone decoders that activate indicator lights and a sonalert audible device. This alerts the operator of an incoming call on a particular channel. The audio circuits of the console are muted until activated by the decoder, and turned on to normal volume when a proper code sequence is decoded. The audio circuit and sonalert have to be manually reset ensuring that the operator does not miss an incoming call.

It should be noted that the Millstone 2 console is operationally identical to console in Units 3. This provides an expanded backup system for the communications system on the Millstone site.

O&M/Security/Trunked Radio System

This is a five channel digital based trunking radio system. The trunking repeaters and central controllers are installed in the Telecommunications Radio Housing adjacent to the Millstone stack. The repeater stations are capable of 75 watts of RF power output with continuous duty operation. The primary power source is 120 VAC. The backup power is provided by a diesel generator. Panel type antennas are mounted on the stack. A backup system, designed to come on line in the event of a failure at the primary site, is installed in B475-3. The Control Room has access to a dedicated channel from the communications console. The base station is located in the CPF Building. The directional antennas are located on the roof of the CPF Building and the installation design is such to withstand windspeeds up to and including 100 mph.

System control is from the radio consoles in Units 2 and 3, the security central and secondary alarm stations, and the Emergency Operations Facility (EOF).

A one channel, digital based 900MHz radio system is provided to the Site Security in the event of a failure of the primary 800MHz radio system. The 900MHz repeater is located in the CPF Building. The repeater is capable of 100 watts of RF power output with continuous duty operation. The primary power source is 120 VAC and the backup power source is battery. The individual portable radio is also equipped with a small antenna which provides “portable-to-portable” feature between the radios. An omnidirectional antenna is mounted on the roof of the CPF Building and the installation design is such to withstand windspeeds up to and including 100 mph.

Command Control Network

The CONVEX CCN is a two way radio system using tone alert signaling to provide communications among the control room, the CONVEX load dispatcher and other key operating facilities.

This system is controlled by the radio console in Units 2 and 3. The transmitter/ receiver base station is installed in the Condensate Polishing Facility (CPF). It is installed in an impact resistant, 41 inch cabinet bonded to electrical ground. AC voltage is the primary power source. The base station is fully solid-state incorporating integrated circuitry, located on plug-in modules or independent printed circuit boards. Highly reliable reed switches are used for antenna switching. The base station produces 13.8 VDC to supply power and draws little current. Unheated, temperature compensated plug-in oscillator modules are used for frequency control. The unit contains a continuous duty transmitter that can operate indefinitely on full power. There are five front mounted metering receptacles for ease of maintenance troubleshooting. The station is remotely controlled by tone frequencies. The wire line controlling the station need not have DC continuity for operation.

The base station is connected to the antenna via a jacketed one-half inch diameter semirigid coaxial cable. The cable is installed in cable tray OTX 850N which is dedicated to communication cables only. The cable ultimately terminates at the antenna mount on the CPF penthouse. The coaxial cable has the outer copper jacket bonded to ground before entry into the building. The coaxial cable has an impedance of 50 ohms and offers a combination of remarkable flexibility, high strength, and superior electrical performance. It includes a copper clad aluminum center conductor, low loss cellular polyethylene foam dielectric, corrugated copper outer conductor, and a protective black polyethylene jacket. The antenna is rigidly mounted to a permanent bracket secured to the parapet of the CPF penthouse. It is a highly directional r-f radiating device with a power gain of 5 dB. The antenna is designed to withstand severe environmental conditions. Radiating elements are made of three-quarter inch diameter tubing and reinforced with seven-eighths inch diameter sockets at the mounting boom. It contains direct ground lightning protection and has a wind rating survival of 97 mph. The installed antenna weighs 37 pounds.

Waterford Police Radio

The Waterford Police Department two-way radio system provides communications between the Waterford Emergency Communications Dispatcher and the Control Room. The system is controlled by the radio console in Units 2 and 3. The base station is located in the CPF telecommunications room.

The antenna is installed on the CPF Building penthouse. It is provided with lightning protection and has a wind rating of 150 mph.

Tri-Town UHF Radio System

The Tri-Town UHF radio system is an administrative two-way radio system used by three towns in the Millstone area. Each of these towns has the ability to call the control room using tone alert signaling.

The system is controlled by the consoles in Units 2 and 3 and base/control station and repeater relay station. The base/control station is located in the CPF Building. It contains two transmit frequencies, the second frequency being “talk-around” in the event of a repeater relay station failure. The unit is installed in an impact-resistant cabinet bonded to the electrical ground. The primary power source is 120

VAC with dual 13.8 VDC battery backup. The station is fully solid-state. The station is connected to the antenna via a one-half inch jacketed semirigid coaxial cable. The cable is 20 feet in length and is securely clamped to the building bulkhead with stainless steel clamps. The cable consists of a copper clad center conductor surrounded by a low loss foam dielectric. A corrugated copper outer conductor encloses this and the entire cable is jacketed with black polyethylene. The antenna is rigidly mounted to the building exterior wall. The antenna is a heavy duty, lightweight, two-stack collinear array designed to provide 5 dB of gain, broad bandwidth, and minimum pattern distortion. A binary cable harness is used to ensure equal in-phase power distribution to all radiating elements. The wind survival of the antenna is 125 mph, and all elements are operated at DC ground to ensure immunity from lightning damage.

The repeater relay station is located in the Telecommunications Radio Housing adjacent to the Millstone stack. The repeater is fully solid-state and has r-f control capabilities to turn the unit on and off. The cabinet is bonded to electrical ground and its primary power source is 120 VAC backed up by the security diesel.

The station is connected to the antenna via a seven-eighths inch jacket semirigid coaxial cable. The cable length is 150 feet and is securely clamped to the stack with stainless steel clamps. The cable consists of a copper clad center conductor surrounded by a low loss foam dielectric. A corrugated copper outer conductor encloses this and the entire cable is jacketed with black polyethylene. The antenna is rigidly mounted to the stack exterior wall. The antenna is a heavy duty, lightweight two-stack collinear array designed to provide 9 dB of gain, broad bandwidth, and minimum pattern distortion. A binary cable harness is used to ensure equal in-phase power distribution to all radiating elements. The wind survival of the antenna is over 125 mph, and all elements are operated at DC ground to ensure immunity from lightning damage.

<u>State Police Radio System</u>

The State Police radio system uses two frequencies. One frequency is used for radio tests and short duration communications. The second frequency is used for communications over extended periods of time. Tone alert signaling is used to allow State Police calls to the control room.

The system is controlled by the consoles in Units 2 and 3. The base station is a desk top style and is located in the CPF Building 212 telecommunications room. The station fully utilizes the advantages of solid-state circuits; reliability, small size, ruggedness, and low maintenance requirements. Efficient heat radiators ensure safe operating temperatures for the transmitter power amplifier stages, and the power supply regulator transistors. The stations primary power source is 120 VAC, and it is protected from overcurrent conditions.

The base station is connected to the antenna via a jacketed one-half inch diameter semirigid coaxial cable. The cable is installed in cable tray OTX 850N which is dedicated to communication cables only. The cable ultimately terminates at the antenna mount on the CPF Building 212 penthouse. The coaxial cable has the outer copper jacket bonded to ground before entry into the building. The coaxial cable has an impedance of 50 ohms, and offers a combination of remarkable flexibility, high strength, and superior electrical performance. It includes a copper clad aluminum center conductor, low-loss cellular polyethylene foam dielectric, corrugated copper outer conductor, and protective black polyethylene jacket. The antenna is rigidly mounted to permanent brackets secured to the parapet of the CPF Building 212 penthouse. The antenna is a unity power gain omnidirectional antenna with a wind rating survival of 100 miles per hour. The antenna uses a shunt-fed coaxial design in a rugged two piece construction. The lower section is enclosed in a heavy-wall fiberglass tube, and the upper fiberglass whip fastens via a protected one-half inch by 20 thread connector. The antenna has direct ground lightning protection and requires no ground plane elements for proper radiation. The antenna weighs 10 pounds.

<u>VHF Radio Paging System/Emergency Notification and Response System</u>

The Emergency Notification and Response System (ENRS) is an automated radiopaging, telephone, audio message and fax delivery system designed to meet the needs of nuclear power facilities, including the requirements of notification in accordance with 10 CFR 50 Appendix E and NUREG-0654. The system initiates emergency notification through telephones, radio pagers and fax machines using application software which features call back and notification verification, and status display and reporting capability as described in the Millstone Nuclear Power Station Emergency Plan. The system activates the Northeast Utility Wide Area Paging System (NUWAPS) and provides radio pages to Site Emergency Response Organization (SERO) personnel, and informational alphanumeric pages and faxed event information to State and Local Agencies. The Incident Report Form (IRF) is faxed by computer to outside agencies. ENRS is comprised of redundant telephone servers and client terminals located in the control room and Emergency Operations Facility (EOF). Client terminals are connected to telephone servers through the local area network and individual modems.

f. The carrier current system provides direct communications with the substations at the termination of each of the outgoing transmission lines.

g. The 5-channel amplified system is used for maintenance purposes such as instrument calibration, equipment adjustment, and the like. The layout provides point-to-point service; as between the control room and a station within the unit, or between two stations within the unit. The instruments are not permanently installed, but are the portable type that can be plugged into jacks conveniently located throughout the unit. This jack system covers working areas, the main control board, and the operator’s desk.

h. The fuel handling system telephones are located along the fuel transfer canal, and provide ready communication among those engaged in loading or unloading fuel. By means of coupling units to the crane power supply, a carrier system ties in a telephone on the polar crane. All stations are “common talk,” and ringing facilities are not included.

i. The public address facilities consist of a voice-paging system that provides communication for the Unit 2 area. A switch is provided so that the unit operator can mute all outdoor speakers at night. A paging adapter is furnished by the telephone company so that designated PBX stations under b. above can dial into the paging system of Unit 1, Unit 2, Unit 3, or all.

j. In the unlikely event that all personnel must evacuate the area, switches in the control room energize containment, site and plant evacuation alarms. A distinctive tone generated by audio frequency oscillators is broadcast through the public address system loudspeakers. This signal takes precedence over all other use of the paging system.

7.8.4 AVAILABILITY AND RELIABILITY

7.8.4.1 Special Features

“Walkie-talkie” radios and pico cellular phones are available for communications between the reactor polar crane and the operating floor of the containment structure. They are also available for other intra-plant uses.

Administrative procedures prevent hand held UHF radios from affecting the solid state reactor protection and/or Engineered Safety Features (ESF) systems.

The cables in the communication systems are independent from those of other systems and are shielded or isolated from power cables and any other sources of line noise which could adversely affect the audibility of the systems. The communication systems use twisted, balanced audio pairs to further reduce the effects of longitudinally induced magnetic noise.

7.8.4.2 Design Evaluation

The failure of any system does not cause the malfunction of the other systems. To ensure high power supply reliability, nonvital systems (requiring power) receive power from the 120/208 V nonvital bus (Section 8.3.1), the Technical Support Center (TSC) electrical distribution system, or the normal DC power system (Section 8.3.2). The plant-switched network is provided with a backup power system that is equipped with a rectifier and backup battery. The microwave system is provided with a separate battery-rectifier power system. The normal and emergency power supply systems for the SNETCO message network are located at the telephone company operating facilities.

7.8.4.3 Testing and Inspection

The design of the communication systems permits routine testing and inspection without disrupting normal communication facilities. The evacuation alarm system will be tested periodically in accordance with normal station procedure.

7.9 ANTICIPATED TRANSIENTS WITHOUT SCRAM CIRCUITRY

7.9.1 DESIGN BASES

7.9.1.1 Functional Requirements

On July 26, 1984, the Code of Federal Regulations (CFR) was amended to include the “ATWS Rule” (Section 10 CFR 50.62, “Requirements for Reduction of Risk from Anticipated Transients Without SCRAM [ATWS] Events for Light-Water-Cooled Nuclear Power Plants”). An ATWS is an expected operational transient (such as loss of main feedwater, loss of condenser vacuum, or loss of offsite power), which is accompanied by a failure of the Reactor Trip System (RTS) to shut down the reactor. The ATWS Rule requires specific improvements in the design and operation of commercial nuclear power facilities to reduce the likelihood of failure to shut down the reactor following anticipated transients and to mitigate the consequences of an ATWS event.

The 10 CFR 50.62 requirements applicable to pressurized water reactors manufactured by Combustion Engineering (CE), are:

1. Each pressurized water reactor must have equipment from sensor output to final actuation device that is diverse from the RTS, which will automatically initiate the Auxiliary (or emergency) Feedwater System (AFWS) and initiate a turbine trip under conditions indicative of an ATWS. This equipment must be designed to perform its function in a reliable manner and be independent (from sensor output to the final actuation device) from the existing RTS.

2. Each pressurized water reactor manufactured by CE must have a diverse SCRAM system from the sensor output to interruption of power to the control rods. This SCRAM system must be designed to perform its function in a reliable manner and be independent from the existing RTS (from sensor output to interruption of power to the control rods).

7.9.2 DISCUSSION

An ATWS is an anticipated operational occurrence (e.g., loss of main feedwater, or turbine trip) which is accompanied by a failure of the Reactor Protection System (RPS) to shut down the reactor. The limiting ATWS events are typified by a rapid Reactor Coolant System (RCS) heatup and pressurization to above 3200 psia before moderator reactivity feedback substantially reduces reactor power.

Historically, CE has performed an analysis of selected transients which provide sufficient characterization of the CE Nuclear Steam Supply System (NSSS) design to ATWS events. For assessing the results of the ATWS analysis, CE used the following generalized criteria:

1. Radiological release within 10 CFR 100 Guidelines

2. RCS pressure less than emergency limits (3200 psia)

3. Reactor fuel - no melting

4. Fuel cladding - no degradation

5. Containment Pressure - within design pressure

Based on CE's analysis, the consequence of a failure to SCRAM would lead to an RCS overpressurization, a violation to Criterion 2. Those ATWS events which would cause RCS overpressurization are:

1. Zero Power Control Element Assembly (CEA) Withdrawal

2. Loss of load

3. Loss of Main Feedwater (complete or partial)

The Millstone Unit 2 Diverse SCRAM System (DSS) and ATWS Mitigating System Actuating Circuitry (AMSAC) fulfills the NRC requirement addressed in 10 CFR 50.62. The DSS is diverse and electrically independent from the RTS and provides a redundant path of reactor and turbine trip by a high-pressure setpoint. The AMSAC is modified from the existing Auxiliary Feedwater Actuation System (AFAS) to mitigate an ATWS event by redundant Auxiliary Feedwater (AF) initiation from DSS.

The DSS reduces the ATWS probability and AMSAC provides some limited mitigation. However, DSS/AMSAC is not expected to fully mitigate all ATWS events. Therefore, the ATWS system is not a direct response to all accidents analyzed in Chapter 14.

Reference:

1. Analysis of Anticipated Transients Without Reactor SCRAM in Combustion Engineering's NSSS's, May 1976, Combustion Engineering, CENPD-158, Revision 1.

7.9.3 DESIGN CRITERIA

7.9.3.1 General

The systems and equipment required by 10 CFR 50.62 do not have to meet all of the stringent requirements normally applied to safety related equipment. However, this equipment is part of the broader class of structures, systems, and components important to safety defined in the introduction of 10 CFR 50, Appendix A (General Design Criteria [GDC]). Although the ATWS mitigation system is not required to meet all of the stringent requirements normally applied to safety related equipment per 10 CFR 50.62, the DSS/AMSAC is designed to Quality Assurance Category 1 requirements in accordance with the Quality Assurance Program (QAP).

The ATWS system is designed to the following bases:

a. No single component failure can prevent the performance of a safety function.

b. Channel independence is assured by separate connection of the sensors to the process systems and of the channels to vital instrument buses.

c. The four measurement channels provide trip signals to six independent logic matrices, arranged to effect a two-out-of-four coincidence logic having outputs to two independent trip paths.

d. A trip signal from any two-out-of-four protective channels on the same parameter causes an ATWS trip.

e. When one of the four channels is taken out of service, the protective system logic may be changed to two-out-of-three coincidence for an ATWS trip by bypassing the removed channel.

f. Independence is provided between redundant elements to preclude any interactions between channels during maintenance or in the event of channel malfunction.

g. Redundant elements are electrically isolated from each other such that events affecting one element are not reflected in any other redundant element.

h. The tripping function is accomplished via an energize to trip logic path to preclude inadvertent plant trips due to component failure.

7.9.3.2 Electrical Independence

The ATWS rule requires that the DSS be electrically independent from the RPS. Both the RPS and DSS share four pressurizer pressure sensor channels. Each channel output is isolated between output to the RPS and output to the DSS. The isolation design is consistent with the present licensing basis for Millstone Unit 2. Subsequent DSS processing is totally electrically independent from the RPS.

7.9.3.3 Environmental Qualification

The DSS is designed to operate for Anticipated Operational Occurrence environment which means a normal containment environment and a mild control room environment. There is no requirement to qualify the DSS/AMSAC to Loss-of-Coolant Accident (LOCA) or High Energy Line Break (HELB) environment since these are not considered anticipated operational occurrences. The DSS electronics which share the same instrument rack and same power supply as isolation electronics for the RPS are qualified to IEEE-323-1983.

7.9.3.4 Seismic Qualification

The DSS need not work during or after a seismic event. However, all components that interface with Category 1E systems will be seismically restrained and electrically certified not to degrade the Category 1E equipment. The DSS electronics which share the same instrument rack and same power supply as isolation electronics for the RPS are qualified to IEEE-344-1975.

7.9.3.5 Annunciation and Display

The DSS status and bypass indications will be provided in the control room. Trip alarm and bypass will be annunciated on the main control board.

7.9.3.6 Testability

The DSS is designed as a four-channel system with individual bypass switch for each channel. The bypass switch will permit individual channel testing while the reactor is operating at power.

7.9.3.7 Diversity

The Millstone Unit 2 DSS design uses the existing RPS pressurizer pressure sensors to generate both the RPS and DSS actuation signals. Even though the DSS and RPS use common sensors, the DSS uses qualified, Category 1E electronics that have been analyzed to demonstrate isolation from the RPS. This minimizes the potential for adverse electrical interactions between the two systems.

Diversity of manufacturer exists for the DSS and RPS bistables, power supplies and matrix relays. The DSS and RPS initiation relays and final actuation devices are both manufactured by the same vendor; however, diversity of model/design principle exists.

The Millstone Unit 2, Diverse Turbine Trip (DTT) design is such that the DTT shares all circuit components with the DSS, up to, but not including, the final trip device. Thus, all of the information that is applicable to DSS components discussed previously are applicable to the DTT components, up to, but not including the final trip device. When the DSS causes a reactor trip, it also causes a turbine trip as the DSS interrupts power to the turbine trip undervoltage coils. The diverse SCRAM relays provide isolation between the Category 1E and Noncategory 1E portions of the circuits. Thus, with the implementation of the DSS, the existing turbine trip becomes a DTT due to the diversity between the DSS and the existing RPS.

The existing AFWS actuation circuitry, when installed at Millstone Unit 2, contained significant diversity from the RPS.

The AFWS design at Millstone Unit 2, was upgraded following the TMI-2 accident in accordance with the TMI Action Plan Items II.E.1.1, “Auxiliary Feedwater System Evaluation,” and II.E.1.2, “Auxiliary Feedwater System Automatic Initiation and Flow Indication,” of NUREG-0737, “Clarification of TMI Action Plant Requirements.” Diversity of manufacturer exists for the Diverse Auxiliary Feedwater (DAFW) bistables, matrix relays, initiation relays and power supplies. The RPS and DAFW initiation relays and final actuation devices are both manufactured by the same vendor; however, diversity of model/design principle exists.

7.9.4 SYSTEM DESCRIPTION

The Millstone Unit 2 scheme for mitigating ATWS events consists of a DSS, an AMSAC system, and a DTT system. The DSS provides a redundant path of reactor and turbine trip by a high pressurizer pressure setpoint. The AMSAC uses the existing AFAS to mitigate an ATWS event by redundant AF initiation from DSS.

The ATWS is designed to be highly reliable, resistant to inadvertent actuation, and easily maintained. Reliability is assured through the use of internal redundancy and industry proven system components. Inadvertent actuations are minimized through the use of internal redundancy, energize to trip design, and good human factors practices. The time delay on low steam generator level and the coincidence logic used also minimize inadvertent actuations. Figure 7.9–1 details the ATWS system.

7.9.4.1 Diverse Scram System

The DSS receives input from the existing RPS pressurizer pressure sensors. The pressurizer pressure signals are routed to signal processing instrumentation consisting of bistables and logic circuitry arranged in a two-out-of-four energize-to-actuate logic to trip the RPS Motor Generator (MG) set output contactors upon detection of conditions indicative of an ATWS event.

Visual and audible alarms provide the control room operator with indication of DSS initiation.

Provisions have been included to allow the bypass of any one sensor input thus converting the logic to two-out-of-three to allow for maintenance and testing of the sensors and associated signal processing electronics. All bypasses are annunciated on the Main Control Board.

7.9.4.2 Diverse Auxiliary Feedwater Actuating System

The existing AFAS has been modified to satisfy the ATWS requirement. The present AFAS is initiated upon low steam generator level (two-out-of-four logic) following a 3 minute, 25 second time delay. An ATWS event with reactor power greater than 20 percent will initiate automatic AF following a 10-second time delay. An ATWS event with reactor power less than 20 percent will initiate automatic AF following a 3 minute, 25 second time delay.

Visual and audible alarms provide the Control Room operator with indication of AFAS initiation.

7.9.4.3 Diverse Turbine Trip

The Millstone Unit 2 DTT design is such that the DTT shares all circuit components with the DSS, up to, but not including, the final trip device.

The existing turbine trip is actuated by the undervoltage relays on the Control Element Drive Mechanism (CEDM) power cables from the MG output. The DSS relays interrupt the MG output thus providing a DTT.

7.9.5 SYSTEM COMPONENTS

7.9.5.1 System Hardware and Interface

Figure 7.9–2 provides an overview of the DSS/AMSAC hardware and interconnections between components. Four pressurizer pressure channels each containing logic for high-pressure trip, provide the inputs to DSS matrix located in panel C100. Two Nuclear Instrument (NI) channels, which are indicative of reactor power level, interface with the AF initiation facilities Z1 and Z2 to provide a redundant AF initiation.

The DSS matrix combines four channel trip contacts in a two-out-of-four voting matrix. The Matrix is arranged so that two channels will issue a DSS signal. If a channel is removed out of service, the voting matrix automatically converts to a two-out-of-three voting configuration.

The output from the DSS matrix directly drives the DSS relays, 94A/DSS and 94B/DSS. Both relays provide for the redundant AF initiation and the tripping of the MG output contactors.

7.9.5.2 Pressurizer Pressure Channels

The Millstone Unit 2, design uses the existing RPS pressurizer pressure sensors (PT-102A-D) to generate both the RPS and DSS actuation signals. Even though the DSS and RPS use common sensors, the DSS uses qualified, Category 1E electronics that have been analyzed to demonstrate isolation from the RPS. This minimizes the potential for adverse electrical interactions between the two systems.

The High Alarm contact drives four relays on a SPEC 200 N-2A0-L2C-R relay isolator card which provides train isolation and interconnections for use in the DSS matrix. The High Alarm setpoint is 2400 psia.

7.9.5.3 Neutron Monitoring Channels

Two independent analog signals corresponding to 0 - 100 percent reactor power are routed from the Reactor Regulating System (RRS) Channel “X” and “Y” to qualified Category 1E electronics. A 20 percent reactor power alarm contact drives a relay on a SPEC 200 N-2A0-L2C-R relay isolator card which provides train isolation and interconnections to the AF initiation circuitry.

7.9.5.4 Diverse SCRAM Matrix

Figure 7.9–3 shows the scheme arrangement for the Diverse SCRAM (DS) Matrix and its logical interconnections.

There are six possible combinations to implement a two-out-of-four voting scheme. These six combinations are represented by the six vertical contact strings, each representing a voted combination. Each contact in a string is associated with a disable/enable contact from the channel keylock switch (HS-102).

The matrix is arranged to activate the DSS relays 94A/DSS and 94B/DSS only when power is available. A High Alarm on any two channels will close two contacts on a vertical string. If the HS-102 switch is closed for both contacts, then both relays will activate.

Both relays 94A/DSS and 94B/DSS are used to provide isolation between the DSS matrix, the AF initiation facilities and the MG contactors and to ensure a DS in the event of a single relay failure. Interposing (auxiliary) relays 94X/MG1 and 94X/MG2 provide the auxiliary contacts to operate the MG contactors and to provide additional isolation between the 480V MG control circuit and the DSS control circuit.

7.9.5.5 Auxiliary Feedwater Initiation

Two contacts from the DSS relay are used in the AF initiation scheme. The first contact parallels the steam generator low level - AF initiation relay contact and functions to start the 3 minute, 25 second timer to initiate AF. The second contact starts a 10 second timer which will be used to conditionally start AF initiation 10 seconds after receiving a DS signal with Reactor Power level ≥20 percent.

7.9.5.6 Motor Generator Contactors

The existing contactor schematic circuit is tripped by loss of motor power for longer than 2 seconds, high or low current through the exciter, synchronization errors or bus overvoltage.

Each MG contactor can be operated from either redundant DSS relay via auxiliary relay schemes. Auxiliary relays 94X/MG1 and 94X/MG2 are activated separately, by circuits containing parallel, normally open, contacts from the 94A/DSS and 94B/DSS relays. When activated by a DSS relay(s), both auxiliary relays provide signals to open each MG contactor via normally closed contacts connected in series within each contactor control circuit. For a postulated single failure in a DSS, auxiliary relay or associated contact, this control scheme ensures that the redundant DSS scheme will trip both contactors.

7.9.5.7 Power Supply

The DSS and AMSAC circuits use common AC power sources from the 120 VAC vital buses, VA10, VA20, VA30, and VA40, for all components from the sensors to the initiation relays.

The C100 matrix tripping relays (94A/DSS and 94B/DSS) are powered from 125 VDC power supplies (via VA10 and VA20 respectively). Each power supply is rated for 100 percent capacity and their outputs are diode auctioneered such that a single failure will not impact system operation.

7.9.6 SYSTEM OPERATION

The DSS/AMSAC provides a backup to the RTS for initiating a reactor and turbine trip and AF flow in the event of an anticipated transient.

An ATWS trip will occur when a high pressurizer pressure signal exceeds the setpoint of 2400 psia on two of the four inputs channels. Initiation of the ATWS will result in the following system response:

1. A reactor (and subsequent turbine) trip will be processed by opening of series DSS auxiliary relay contacts in the RPS MG Set outputs.

2. AF initiation will occur after a 10 second time delay if reactor power remains ≥20 percent, otherwise, AF initiation will occur following a 3 minute, 25 second time delay.

7.9.6.1 Bypasses

An individual bypass (keylock) switch is available for each of the four pressure channels. This bypass will allow on-line maintenance and testing capabilities of the DSS/AMSAC circuitry.

7.9.6.2 Annunciation and Display

Indication of individual pressure channel trip is available at the C100 status panel. An “Amber” lamp will be illuminated on the C100 status panel indicating a channel trip.

Annunciation alarms on main control board C04F are provided to alert the operator should an “ATWS” or “AAFWIS” (Automatic Auxiliary Feedwater Initiation System) initiation occur.

ATWS channel bypasses are annunciated on Main Control Board C04F.

7.9.6.3 Inadvertent Actuation

The DSS and AMSAC systems have been designed such that the frequency of inadvertent actuations is minimized. Reliability of this system is ensured through the use of redundancy, majority voting logic, bypasses and energize to trip circuitry.

The change slightly increases the likelihood of spurious reactor trip (i.e., a trip when no protection setpoints have been exceeded) because of the addition of electrical components which can interrupt power from the MG set to the CEDMs.

The inadvertent actuation of the DSS or malfunction of the MG output contactors would result in the interruption of power to the CEDM coils causing the rods to drop by gravity into the core. The undervoltage condition will also cause a turbine trip signal to be generated. Therefore, inadvertent DSS actuation would result in simultaneous turbine and reactor trip. An inadvertent DSS Actuation would resemble the interruption of power to or from the MG set as in the existing design. Therefore, a spurious actuation is neither a Design Basis Accident (DBA) nor a new accident.

The change could also result in automatic AFWS actuation, regardless of steam generator level (as in current design), if the DSS is inadvertently actuated and a high neutron flux signal (> 20 percent) is sustained for 10 seconds. This could increase the challenges to the Engineered Safeguards Features (ESF), and increase the potential for feedwater nozzle thermal shock, steam generator overfill, and so on. However, for this to be a concern, two low-probability events must occur. Inadvertent DSS actuation is of low probability. Given that DSS actuation has occurred, reactor SCRAM will occur and neutron flux will be below 20 percent within 10 seconds. Therefore, a second (low probability failure) must occur, that is, a high neutron flux signal because of miscalibration error or electrical component failure. Given that the plant was operating with neutron flux measurement calibrated in the first place, these failure modes are low probability. The conclusion is that inadvertent AFWS actuation, due to the change, is of insignificant probability compared to all other mechanisms for inadvertent actuation.

FIGURE 7.9–1 LOGIC DIAGRAM - ANTICIPATED TRANSIENT WITHOUT SCRAM

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.9–2 BLOCK DIAGRAM DIVERSE SCRAM SYSTEM (ATWS)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 7.9–3 SCHEMATIC DIAGRAM DIVERSE SCRAM SYSTEM ATWS

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

7.10 AUXILIARY STEAM LINE BREAK DETECTION/ISOLATION SYSTEM

7.10.1 DESIGN BASES

7.10.1.1 Functional Requirements

The auxiliary steam line break detection/isolation functions to isolate steam supply to specific safety related areas if steam leakage is detected due to an auxiliary steam line (or other) high energy line break.

7.10.2 DISCUSSION

Auxiliary Steam System

The auxiliary steam system is provided for building heating, freeze protection for outdoor water storage tanks, and radwaste process requirements. The system is nonnuclear safety (NNS). Steam is normally provided by the Unit 3 auxiliary steam system via a crosstie between the Units. Because the Unit 3 auxiliary steam system operates at 150 psig and the Unit 2 auxiliary steam system operates at 50 psig, a pressure reducing valve station, including isolation and relief valves are installed. In addition a condensate return line from the Unit 2 auxiliary feedwater surge tank is routed to the Unit 3 condensate system. Condensate is routed to the auxiliary boiler deaerator, when the Unit 3 auxiliary boilers are supplying auxiliary steam, or to the Unit 3 condensate surge tank when Unit 3 main steam is supplying auxiliary steam.

During Unit 2 plant shutdown, the Unit 3 auxiliary steam system provides steam to the Unit Number 2 auxiliary steam system. In the event both Units are shutdown the Unit 3 auxiliary steam boilers maintains the capability to provide house heating steam to both Units, as well as the steam needs for the site fire water storage tanks freeze protection. Temporary electric heating has been provided to the fire water storage tanks while auxiliary steam supply is unavailable.

Technical Specification Numbers 3/4.1.2 (Boration Systems - Modes 5 & 6) and 3/4.5.4 (Refueling Water Storage Tank (Modes 1, 2, 3 and 4) establish minimum temperature requirements for safety related tanks that are heated by the auxiliary steam system.

10 CFR 50, Appendix A, Design Criterion 4, requires structures, systems, and equipment important to safety to be designed for effects and be compatible with environmental conditions associated with effects of postulated pipe ruptures.

The auxiliary steam system is classified as a high energy fluid system. The auxiliary steam temperature detection system is installed to detect an auxiliary steam leak or line break in specific safety related areas shown on Figure 7.10–1. The detection system, if activated, will isolate the steam supply to these areas to ensure operability of safety related equipment needed to safely shut down the plant.

Pipe break analysis criteria and guidance are discussed in Section 6.1.4.

Guidance concerning equipment qualification related to high energy line break were provided by the NRC in November 1979 DOR Guidelines “Guidelines for Evaluating Environmental Qualification of Class 1E Electrical Equipment in Operating Reactors,” and NUREG-0588 “Interim Staff Position on Environmental Qualification of Safety Related Electrical Equipment.”

Regulations for environmental qualification of electrical equipment are specified in 10 CFR 50.49.

7.10.3 DESIGN CRITERIA

7.10.3.1 General

The following criteria have been used in the design of the auxiliary steam detection/isolation (ASDI) system:

a. The system shall have redundant, independent subsystems.

b. The system shall have suitable subsystem and component alignments to assure operation of the complete subsystem with its associated components.

c. Capabilities shall be provided to assure the system function with onsite power (assuming offsite power is not available) or with offsite electrical power.

d. A single failure in either subsystem shall not affect the functional capability of the other subsystem.

e. The system shall be designed to permit periodic inspection of important components, such as temperature detectors and automatic isolation valves to assure the integrity and capability of the system.

f. The ASDI system shall be designed to permit appropriate periodic pressure and functional testing to assure the operability and performance of the active components of the system, and the operability of the system as a whole. Under conditions as close to the design as practical, performance shall be demonstrated of the full operational sequence that brings the system into operation, including operation of applicable portions of the detection system.

g. The components of the detection system shall be designed to operate in the most severe environment to which it is exposed during an auxiliary steam line break.

7.10.3.2 Electrical Independence

The ASDI system is designed with two electrically independent channels in accordance with IEEE 279, Criteria for Protection Systems for Nuclear Power Generating Stations (1971).

7.10.3.3 Environmental Qualification

Temperature detectors associated with the ASDI system are qualified to withstand the predicted environment resulting from an auxiliary steam line break in the auxiliary and enclosure buildings. Detectors and associated components are qualified in accordance with IEEE 323, Qualifying Class 1E Equipment for Nuclear Power Generating Stations (1974). Isolation valves, associated controls, and other active components are designed to fail in the safe direction and are not required to be environmentally qualified.

7.10.3.4 Seismic Qualification

The ASDI system is seismically qualified to ensure detection of a steam line break in the required areas, transmission of the signal, and closure of isolation valves for the auxiliary steam supply to the Auxiliary Building. Components are seismically qualified to IEEE 344, Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Systems (1974).

7.10.3.5 Annunciation and Display

The ASDI system status and bypass indications are provided in the control room. Detection alarm and bypass are annunciated on the main control board.

7.10.3.6 Testability

The ASDI System is designed as a two channel system. Each channel is provided with a bypass switch to allow periodic test of each channel without causing isolation of the auxiliary steam supply to the Auxiliary Building.

7.10.4 SYSTEM DESCRIPTION

The ASDI System is shown schematically in Figure 7.10–1. The ASDI System consists of eight pairs of RTDs located in the areas of the auxiliary building as shown.

Temperature detectors (RTDs) are installed at locations easily accessible from floor level for maintenance and calibration. The temperature detector electronics are located in control cabinets C502 and C503 in the East 480 volt Loadcenter Room. Associated control relays and hand switches are located in panel C-80 in the main Control Room.

RTDs are resistance temperature detector type. The detection/isolation are set at a maximum threshold set point value of approximately 115°F to rapidly detect a potential steam line break or leak. The exact set point is determined based on maximum design temperature applicable to the respective area (see FSAR Section 9 for this basis) with a suitable margin included as needed to avoid spurious actuation and alarms. Actuation of any alarm bistable causes the respective steam isolation valve to close. Status indicating lights (red) are provided at C502/C503 panels for each temperature switch. A computer point and main control board alarm is provided for each Auxiliary Steam Detection Isolation System Actuation.

Each RTD of the pair in each environmental (EEQ) zone is independently powered and redundant to each other (single failure protected).

Each train of alarm switches has their control contacts wired in series. Reset of the system after an alarm switch actuates is accomplished via hand switch in the control room (panel C-80). The handswitch will reopen the auxiliary steam isolation valves after the affected RTD returns to normal ambient conditions. The RTDs are seismic and environmentally qualified.

Each RTD will actuate one of (2) automatic isolation valves to close. The position of the auxiliary steam isolation valves is indicated by red-green indicating lamps on the C80 panel in the control room. Maintenance/Bypass switch engagement is indicated by an amber lamp on C-80 panel and also annunciated on a common alarm window on panel CO6.

After an actuation of an isolation valve(s) and the alarm bistable resets, the annunciator will reset, but the auxiliary steam isolation valves will remain closed. The operator must take deliberate action by operating hand switch(es) on the C80 panel to re-open the closed valve(s).

The control scheme has a built-in maintenance/bypass keylock switch on the C80 panel which allows calibration of the bistables without closing the auxiliary steam isolation valves. This function is a “key locked” hand switch. The bypass condition is annunciated on panel CO6 when in the bypass position. An amber light on C80 panel indicates when the alarm bistable trip function has been bypassed. Procedural controls have been established and will be maintained to effect compensatory measures during periods when the trip function is bypassed.

The automatic isolation valves are located in the Turbine Building prior to the Auxiliary Steam penetration into the Auxiliary Building at the Turbine/Auxiliary building wall. The valves are independent and redundant to each other to meet single failure criteria. The valves are high performance (very low leakage), butterfly type with an offset disc and metallic seat.

The valves are air-operated, air to open, vent to close. They fail closed with spring assist on loss of instrument air or power. Each valve is provided with a high capacity, rapid discharge solenoid vent valve to ensure closure time of less than five seconds. The valve and air operator are seismically qualified. The solenoid vent valves and valve position limit switches are purchased EEQ <u>and</u> seismically qualified. However, these items do not have to be maintained in the MP2 EEQ master list since the isolation valves are not <u>required</u> to isolate if a steam line breaks in the Turbine Building. Each isolation valve has an air pressure regulator based on design pressure limitations of the air operator diaphragm. The regulator is located upstream of the solenoid vent valve and is nonsafety related.

The piping system is designed in accordance with ANSI B31.1 design criteria.

FIGURE 7.10–1 PIPING AND INSTRUMENTATION DIAGRAM AUXILIARY STEAM AND CONDENSATE

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

7.11 ENVIRONMENTAL QUALIFICATION OF ELECTRICAL EQUIPMENT IMPORTANT TO SAFETY

Since environmental conditions vary for different areas of the plant, there are several environmental zones. Safety related equipment and components are qualified to meet their performance requirements under normal, abnormal, and accident operating conditions based on the environmental zone in which the equipment is exposed.

The Millstone Electrical Equipment Qualification (EEQ) Program is a process that ensures the continued qualification of equipment which must function during and following the design conditions postulated for design basis accidents and the post-accident duration. This program has been developed to ensure that environmental qualification criteria are applied to electrical equipment important to safety in accordance with 10 CFR 50.49 and to document the process through which this qualification is demonstrated. It incorporates NRC Regulation, Regulatory Guides, and other positions and guidelines, as well as Institute of Electrical and Electronic Engineers (IEEE) Standards and sound engineering practices.

The Design Basis area of the EEQ program consists of the Environmental Qualification Master List (EQML) and the plant normal and accident environmental conditions. The EQML is the set of equipment required to be environmentally qualified. Environmental Parameters include temperature, pressure, relative humidity, chemicals, spray potential, submergence, accident duration, and gamma/beta radiation dose (where applicable). These parameters are given in terms of a time-based profile. Changes in equipment function or operating mode, substitution of new model numbers, and the addition or deletion of equipment in the plant, shall require a revision of the EEQ program documentation. Changes to the environmental parameters such as revisions to accident analyses, rerouting or modifying the high energy lines, or changes in the functional operation of the HVAC system may result in revisions to the EEQ program documentation.

Test Report Assessments (TRA), Equipment Qualification Records (EQR), and referenced design documents provide an auditable proof of the equipment qualification. The qualification process is based on the testing and/or analysis of same or similar equipment or material such that this equipment performance becomes the model or proof of how the installed equipment is anticipated to behave when exposed to design basis accident environmental conditions. Emulation of the tested equipment's internal, external, and maintained configuration is necessary to enable the test to represent the plant installed equipment. Products of the qualification verification include installation, maintenance, and procurement requirements that must be implemented to ensure that the installed equipment is the same as the equipment tested.