Consumers
Power
Company

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June 28, 1984

Harold Denton, Director
Office of Nuclear Reactor Regulation
US Nuclear Regulatory Commission
Washington, DC 20555

DOCKET 50-255 - LICENSE DPR-20 -
PALISADES PLANT - FINAL SAFETY ANALYSIS REPORT (FSAR) UPDATE, REV 0

Your October 29, 1982 letter issuing the Final Integrated Plant Safety Assessment Report (IPSAR), NUREG-0820, for Palisades Plant provided notification under the provision of 10 CFR 50.71(e)(3)(ii) that Consumers Power Company is required to file with the NRC, within twenty-four (24) months of our receipt of your letter, a complete Final Safety Analysis Report (FSAR) updated to within a maximum of six (6) months prior to the date of filing the revision.

Enclosed are one (1) original and twelve (12) copies of the Palisades Plant Final Safety Analysis Report (FSAR) Update, Revision 0. This FSAR Update is a complete revision of the 1980 FSAR as amended and revised. To the best of my knowledge, it accurately incorporates all changes made to the Palisades Plant or to the Palisades Plant procedures as described in the FSAR and all the other applicable information contained in licensing submittals made to the NRC through December 31, 1983.

The Palisades Site Emergency Plan, Security Plan, QA Program, Electrical Equipment Qualification Program and Fire Hazards Analysis are incorporated into the appropriate Chapters of the FSAR Update through reference. Other information too bulky or complex for direct incorporation is included as Appendices to the appropriate FSAR Chapters.

In preparing the FSAR Update, Consumers Power Company directed its consultant, to maintain in accordance with 10 CFR 50.71(e) the same level of detail as originally provided; to maintain essentially the same format as the original FSAR; and, to include additional information in the Chapters where available to conform as close as practicable to NRC Regulatory Guide 1.70 (Rev 3 11/78) - Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants.

Chapter 12 contains the information requested in NRC Generic Letter 84-14 regarding Replacement and Requalfication Training Program.

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Chapter 13, Initial Tests and Operation, is included in the FSAR Update for completeness only. It has not been updated but, rather, has been included in its entirety as originally submitted plus any amendments or revisions as clarified in the Questions and Responses (Item C, Question #2) referenced in the Statement of Considerations contained in the Final Rule (45FR30614).

Chapter 14 Update has not been included in this submittal as Consumers Power Company is still finalizing Chapter 14 and will submit it under separate cover at a later date but prior to the 24-month deadline established by 10 CFR 50.71(e)(3)(ii). Space has been reserved, however, in Volume 10 of the Update for Chapter 14.

We have revised the convention for designation of the FSAR "Sections" (1980 FSAR) to "Chapters."

Furthermore, because the Palisades Plant is in an extended refueling outage, the Update reflects the status of the Plant at its last day of normal operation with the inclusion of information on the following modifications scheduled to be completed by the end of the current refueling outage:

Auxiliary Building Addition
Service Water Pump Spray Deflectors
Electric System Expansion
Startup Power Load Shed
Emergency Radiation Assessment System (ERAS) -
 Containment Gamma and Hydrogen
Critical Functions Monitoring System
Auxiliary Feedwater System Upgrade
Heating, Ventilating and Air Condition Modification
Radwaste Solidifaction System

As a result of the extended refueling outage, several Facility Change Packages are still open and certain drawings are in various stages of revision. Update Figures 7-26, 7-62, 8-1 and 9-19 are currently in the revision process. The Update Figures represent the latest revision available from our Engineering Records Center. Revision of these figures will be submitted in Revision 1 of the Update which will be submitted by July 1, 1985.

Russell B DeWitt

Russell B DeWitt, Vice President
Nuclear Operations

CC Chief, Office of Nuclear Reactor Regulation
 Administrator, Region III, USNRC
 NRC Resident Inspector - Palisades

PALISADES PLANT
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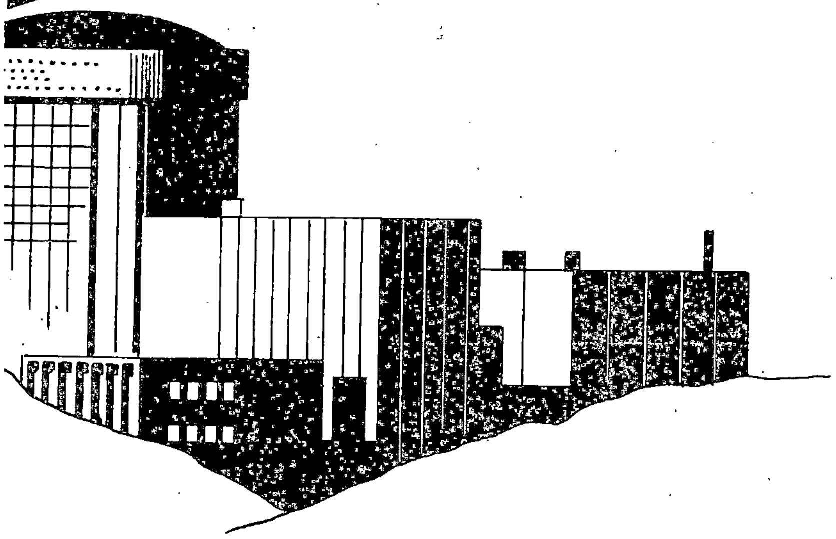

ISADES NUCLEAR PLANT

final safety analysis report

UPDATE

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CHAPTER 1

CHAPTER 1

INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

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CHAPTER 1

INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 INTRODUCTION

1.1.1 GENERAL

This update of the FSAR for the Palisades Plant is pursuant to the requirements of 10 CFR 50.71(e). It represents the current safety evaluation of the Palisades Plant by incorporating information relating to Plant modifications, revised analysis, additional studies and results of NRC special programs which became available subsequent to the release of the original FSAR on November 1, 1968.

Inasmuch as possible, the original (existing as of 1980) FSAR chapter format has been retained except for the inclusion of new material or rearrangement of original material for editorial purposes. All appendices and amendments of the 1980 FSAR have been incorporated into the Update text. The Security Plan, Emergency Plan and the QA Program have been omitted from the Update and should be consulted independently. Per the instructions of 10 CFR 50.71(e), the FSAR Update is self-standing, requiring that new information become part of the FSAR Update text rather than simply incorporating by reference only. New material not easily incorporable into the text may appear as new appendices. Most pages of the FSAR Update are new and cannot be directly compared to the 1980 FSAR.

1.1.2 LICENSING HISTORY

Consumers Power filed, on Docket 50-255, a Construction Permit and Operating License (CPOL) Application (which included the PSAR) to the AEC on June 2, 1966 for the Palisades Plant to be located near South Haven, Michigan. The application was for development of a 2,650 MWt (design core power) commercial nuclear-powered electrical generating facility to be operated at 2,200 MWt or an equivalent electrical output of 700 MWe. On March 14, 1967, the AEC issued Consumers Power a Class 104 Construction Permit CPPR-25, pursuant to Section 104(b) of the Atomic Energy Act, to construct a Combustion Engineering pressurized water reactor (PWR) with a full-power design rating of 2,650 MWt. Subsequent to the original CPOL/PSAR application, eight additional PSAR amendments were filed addressing NRC concerns and addition of the Technical Specifications. On November 1, 1968, Consumers Power filed with the AEC an Operating License (OL) Application (which included the FSAR) as Amendment 9 to the CPOL Docket 50-255, to operate the Palisades Plant at 2,200 MWt core power. Following submission of the initial FSAR as Amendment 9 to the CPOL, 23 subsequent amendments to the FSAR were submitted to the NRC. They were identified as Amendments 10 through 32 to the CPOL Docket 50-255, the most extensive of which was the Full-Term Operating License (FTOL) Application, Amendment 28. In addition, nine minor revisions were subsequently submitted to the NRC.

On March 24, 1971, the NRC issued Interim Provisional Operating License IDPR-20 to be effective for 1-1/2 years to operate Palisades up to 1 MWt. Subsequent amendments to IDPR-20 were issued on November 20, 1971 to operate up to 440 MWt (20% power); March 10, 1972 to operate up to 1,320 MWt (60% power); September 1, 1972 continued operation at 1,320 MWt (60% power); October 16, 1972 authorized operations for 2,200 MWt (100% power - limited to 60% power); and March 23, 1973 authorized operations for 2,200 MWt (100% power - limited to 85% power). That Operating License has since been amended numerous times to keep the Plant current with NRC standards and to reflect Plant modifications.

On January 22, 1974, Consumers Power requested conversion of the Provisional Operating License DPR-20 to a Full-Term Operating License to operate at 2,638 MWt (845 MWe gross) for a period of 40 years from the date of the issuance of the Construction Permit. As part of the FOL Application, which was submitted as Amendment 28 to Docket 50-255, a complete amendment to the FSAR was provided. This FSAR amendment included major revisions based upon:

1. Incorporation of information related to the once-through circulating water conversion to a closed-cycle system with mechanical draft cooling towers and water treatment chemistry changes

2. Increase in operating power to 2,638 MWt core power

3. Incorporation of information related to Radwaste System modifications implemented to obtain conformance to Appendix I of 10 CFR 50

NRC action on the request for an authorization to increase operating power and a Full-Term Operating License was delayed. The Provisional Operating License remained in effect indefinitely beyond its expiration date, however, under 10CFR2.109.

On August 12, 1977, Consumers Power requested that the Provisional Operating License limit of 2,200 MWt be increased to 2,530 MWt based upon reanalysis of safety evaluations and the improvements made with steam generator repairs. On November 1, 1977, the NRC granted Amendment 31 to DPR-20, authorizing operation of the facility at 2,530 MWt core power, which is the present Plant operating limit.

On February 21, 1991 the NRC issued the Full Term Operating License. This license was based on an Environmental Assessment dated October 22, 1990 and an SER issued as NUREG 1424 on November 21, 1990. The license expiration date is specified as midnight on March 14, 2007.

A chronological history of these license events is provided on Table 1-1.

1.2 GENERAL PLANT DESCRIPTION

1.2.1 PLANT SITE

The site for the Palisades Plant consists of approximately 432 acres on the eastern shore of Lake Michigan, in Covert Township, approximately four and one-half miles south of South Haven, Michigan. The area adjacent to the site is sparsely populated and is primarily farmland. The population along the lake increases during the summer months. See Subsection 2.1.2 for details on demography and Figure 2-2 for site layouts.

The exclusion area for Palisades is defined as the property boundary shown on Figure 2-2. The minimum exclusion distance for the site is approximately 2,300 feet (667 meters) and the nearest population center area of more than 24,000 residents is constituted by the cities of Benton Harbor and St Joseph which are approximately 16 miles south of the site.

1.2.2 PLANT ARRANGEMENT

Figure 1-1, Plant Site Plan and Plant Area Plan, displays the primary power block structures arrangement. The turbine building for the Palisades Plant is oriented parallel and adjacent to the shoreline of Lake Michigan, with the reactor containment building located on the east, or landward, side of the turbine building. The office and auxiliary facilities are situated east of the north end of the turbine building so that the entire complex is L-shaped. The reactor containment structure is located inside the corner of this "L." Equipment layouts are shown in Figures 1-2 through 1-25.

The containment building houses the NSSS, consisting of the reactor, steam generators, primary coolant pumps, pressurizer and some of the reactor auxiliaries which do not require access during power operation. The containment building is served by a circular bridge crane.

The turbine building houses the turbine generator, condenser, feedwater heaters, condensate and feed pumps, turbine auxiliaries and certain of the switchgear assemblies. The north end of the turbine building provides shop and warehouse space.

The auxiliary building and auxiliary building addition (radioactive waste building) houses the waste treatment facilities, engineered safeguards components, heating and ventilating system components, the emergency diesel generators, switchgear, laboratories, offices, laundry and the control room. The spent fuel pool and the new fuel storage facilities are located in a separate section of the auxiliary building (Chapter 9) which is under controlled ventilation whenever spent fuel is being moved or stored in that section. Fuel transfer to and from containment is through a fuel transfer tube.

The condensate and makeup demineralizer building (feedwater purity building) was constructed during the feedwater purity modification. It houses the raw water filtration system, the reverse osmosis pretreatment system, the makeup demineralizer system, various components of the condensate demineralizer system, regeneration chemicals handling system, feedwater purity service and instrument air, chemical storage and a boiler room. Because of continuing concern with resin leakage and sodium release, the condensate demineralizer system has been rendered inoperable and retired in place.

The intake structure houses the service water pump and the fire protection pumps. Prior to converting the Plant from once-through cooling to closed-cycle cooling, this building contained the circulating water pumps.

The cooling tower pump house contains two vertical pumps with sufficient head capacity to circulate the tube side condenser cooling water up to the cooling tower inlet near the tower top. The cooling tower basins are elevated some 20 feet above the Plant.

The circulating water cooling towers are cross-flow mechanical draft, located approximately 500 and 1,000 feet from the Plant. Each tower contains 18 cells and is designed for a 30°F range.

1.2.3 CONTAINMENT

The containment building uses a prestressed concrete design. The building is a vertical right cylindrical structure with a dome and a flat base. The building interior is lined with carbon steel plate for leak tightness. Inside the structure, the reactor and other NSSS components are shielded with concrete. An unlined steel ventilation stack is attached to the outside of the containment building and extends to an elevation equal to the top of the containment dome. Access to portions of the containment building during power operation is permissible.

The containment building, in conjunction with engineered safeguards, is designed to withstand the internal pressure and coincident temperature resulting from the energy released in the event of a DBA. The original structure design conditions are an internal pressure of 55 psig, a coincident temperature of 283°F and a leak rate of 0.1% per day by weight at design temperature and pressure. Actual containment conditions calculated to occur following accidents are discussed in Chapter 14.

The containment is equipped with two independent, full-capacity systems for cooling by air recirculation or building sprays after the postulated DBA. The recirculation system is designed to provide maximum containment atmosphere mixing. The cooling coils and fans are sized to provide adequate containment cooling following a DBA with three of the four units in service on emergency power. The building sprays supply borated water to cool and simultaneously remove some of the released fission products from the containment atmosphere. The spray system is sized to provide adequate cooling with two of the three containment spray pumps in service and the two shutdown heat exchangers in operation. Actual system capabilities and operating requirements for fans, coolers and sprays are discussed in Chapters 6 and 14.

The pumps initially take suction from the safety injection and refueling water storage tank. When this supply is depleted, the suction is transferred automatically to the containment sump. By the onset of this recirculation phase, trisodium phosphate is dissolved in the sump solution to neutralize the boric acid.

1.2.4 NUCLEAR STEAM SUPPLY SYSTEM (NSSS)

The NSSS consists of a pressurized water reactor with two closed loops. The principal components and supporting systems of the NSSS are the reactor vessel, internals, control rods, control rod drives, slightly enriched fuel, two "U" tube steam generators, four primary coolant pumps, primary system piping, pressurizer, quench tank, Chemical and Volume Control System, Safety Injection System, nuclear and process instrumentation, and the Reactor Protective System.

The NSSS uses chemical shim and control rods for reactivity control and supplies steam to a four-flow, tandem-compound, hydrogen-cooled turbine generator.

The NSSS is expected to have adequate margin to obtain an ultimate output of 2,650 MWt. The steam and power conversion equipment is designed for a maximum expected gross capability of 845 MWe. See Table 1-2 for equipment design. The Primary Coolant System operates at a nominal pressure of 2,060 psia. The primary coolant enters the upper section of the reactor vessel, flows downward between the reactor vessel shell and the core barrel, and passes through the flow skirt and into the lower plenum where the flow distribution is equalized. The coolant then flows upward through the core removing heat from the fuel rods, exits from the reactor vessel, and passes through the tube side of the two vertical "U" tube steam generators where heat is transferred to the secondary system. Two primary coolant pumps per steam generator return the primary coolant to the reactor vessel.

1. Reactor Vessel and Internals

The reactor vessel and its removable hemispherical closure head are fabricated from carbon steel and are lined with 308/309 stainless steel. In the areas of internal attachments, the interior is clad with Ni-Cr-Fe alloy. A fixed hemispherical head is attached to the lower end of the shell. The reactor vessel is supported on three pads welded to the underside of the coolant nozzles.

The reactor core is supported from the reactor vessel flange and is fueled with uranium in the form of slightly enriched UO_2 pellets. Zircaloy-4 tubing with a minimum wall thickness of 27.5 mils is used for the fuel cladding. The core contains 204 fuel bundles and 45 control rods.

A three-to-four batch, mixed central zone fuel management plan is employed and a further reduction in nuclear peaking is obtained by local enrichment zoning within the bundles. Boric acid dissolved in the coolant is used as the neutron absorber to provide long-term reactivity control. In order to reduce the boric acid concentration required at the beginning of the fuel cycle, and thus to make the moderator coefficient of reactivity more negative, mechanically fixed, burnable poison rods are utilized.

2. Steam Generators

The two steam generators are vertical shell and "U" tube units, each producing approximately 5.5×10^6 lb/h of steam at a normal operating pressure of 750 psia based on approximately 7910 active tubes in each steam generator.

The steam generated in the shell side of the steam generator flows upward through moisture separators which reduce its moisture content to less than 0.2%. All surfaces in contact with the primary coolant are either stainless steel or Inconel in order to maintain primary coolant purity.

3. Primary Coolant Pumps

The coolant in the primary loop is circulated by four primary coolant pumps of the single suction centrifugal type. The pump shafts are sealed by mechanical seals. The seal performance is monitored by pressure and temperature sensing devices in the seal water circulation system.

4. Primary System Piping

Each of the two loops which make up the Primary Coolant System consists of one 42-inch ID pipe and two 30-inch ID pipes. The larger pipe carries the water from the reactor to the steam generator. The flow from the steam generators is pumped to the reactor through the 30-inch ID pipes.

5. Pressure Control System

The pressure in the Primary Coolant System is controlled by regulating the temperature of the coolant in the pressurizer, where steam and water are held in thermal equilibrium. Steam is formed by the pressurizer heaters or condensed by the pressurizer spray to reduce pressure variations caused by expansion and contraction of the primary coolant due to primary system temperature changes.

Overpressure protection is provided by spring-loaded safety valves connected to the pressurizer. The discharge from the pressurizer safety valves is released under water in the pressurizer quench tank, where it is condensed and cooled. In the event that the discharged volume of steam exceeds the capacity of the quench tank, the tank relieves via a rupture disc to containment.

6. Reactor Control

The reactor is controlled by a combination of 45 control rods and dissolved boric acid in the primary coolant. Forty-one of the control rods are full length, and four partial-length rods are also provided. The part-length rods are maintained in the fully withdrawn position during reactor operation and do not insert following a reactor trip.

Boric acid addition or removal is used for reactivity changes associated with major changes in water temperature during start-up and shutdown, fuel burnup and xenon variations. Additions of boric acid also provide an increased shutdown margin during initial fuel loading, refuelings and approaches to cold shutdown condition. The boric acid solution is prepared and stored in two tanks and is maintained at a temperature sufficient to prevent precipitation. The tanks are connected to the charging pumps through automatic valving.

Control rod movement provides changes in reactivity required for power changes or for shutdown to a hot condition. The control rods are made of a silver-indium-cadmium alloy clad with stainless steel welded into a cruciform configuration. They are actuated by control rod drive mechanisms mounted on the head of the reactor vessel. The control rod drive mechanisms, which are rack-and-pinion units, are designed to permit rapid insertion of the control rods into the reactor core by gravity.

7. Chemical and Volume Control System

The purity of primary coolant is controlled by continuous purification of a portion, "letdown," of the total primary coolant volume. Coolant is removed from the primary system and is initially cooled in the regenerative heat exchanger. The coolant letdown is then reduced in pressure by orifices and letdown back pressure valves and again in temperature as it passes through the letdown heat exchanger. The letdown then flows through one of three demineralizers where corrosion and fission products are removed through a filter which traps particulate matter in the effluent from the demineralizer. It is then sprayed into the volume control tank.

The volume control system automatically controls the rate and amount of coolant returned to the Primary Coolant System to maintain the pressurizer level within a control band and thereby compensates for changes in volume due to primary coolant temperature changes. The volume control tank is sized to accommodate primary coolant inventory changes resulting from load changes from hot standby to full power. This mode of operation, using the volume control tank as a surge tank, decreases the quantity of liquid and gaseous waste which otherwise would be generated.

8. Chemical Treatment

Primary system makeup water is taken from the demineralized water storage system and from the concentrated boric acid tanks. The makeup water is pumped through the regenerative heat exchanger into the primary loop by the charging pumps.

Bleed from the primary system during a boron concentration reduction is routed to the radwaste liquid receiver tanks for processing through the Radwaste System before reuse in the Plant or disposal to the lake.

Chemical injection equipment is provided for the addition of corrosion control chemicals to the primary loop water. Hydrogen is added to primary coolant for oxygen scavenging through the volume control tank.

9. Nuclear Control and Instrumentation

a. Nuclear Plant Control

The reactor control system provides for start-up and shutdown of the reactor and for adjustment of the reactor power in response to turbine load demand. The NSSS is capable of following a ramp change from 15% to 100% power at a rate of 5% per minute and at greater rates over smaller load change increments up to a step change of 10%. This control is accomplished by manual rod motion. A temperature computing station calculates the reactor average temperature and a reference temperature value corresponding to turbine power. The reactor average coolant temperature and the reference temperature values are displayed to operators who manually adjust coolant temperature by moving control rods. Regulation of the primary temperature in accordance with this program maintains the secondary steam pressure and matches reactor power to load demand.

b. Reactor Neutron Monitoring

The nuclear instrumentation consists of out-of-core and incore flux monitoring chambers. Eight channels of out-of-core instrumentation monitor the neutron flux and six of the eight channels provide reactor protection signals during start-up and power operation. Two of the channels follow the neutron flux through the start-up range.

The incore monitors consist of rhodium neutron detectors and a thermocouple. This system provides information on neutron flux and temperatures in the core.

c. Reactor Protective System

The reactor parameters are maintained within acceptable limits by the inherent characteristics of the reactor, by the control rod system, by boron control and by the operating procedures. Departures from these limits are indicated audibly and visually in the control room. A Reactor Protective System initiates reactor shutdown if selected values of parameters are exceeded and an unsafe condition for Plant equipment or personnel could arise. The protective system is divided into four channels. Each channel receives trip signals from sensors when the relevant parameter values are exceeded and a two-of-four coincident logic system sends a "deenergize" signal to the control rod drive mechanism clutch power supplies.

The control rods are released and the reactor is shut down when the clutch power supplies are deenergized. Redundant sensors are provided for the reactor shutdown functions so that failure of any one sensor does not prevent a reactor trip.

d. Other Safety-Related Protection and Control Systems

While the Reactor Protective System protects the reactor core and the engineered safeguards controls protect against a Loss of Coolant Accident (LOCA), other safety-related Class 1E control and instrumentation systems are provided to allow a safe shutdown of the Plant, assure decay heat removal and protection of fluid systems boundaries. Such systems are reactor shutdown controls, primary coolant and other liquid boundaries overpressure protection, automatic auxiliary feedwater initiation and containment hydrogen control.

e. Process Instruments

Critical primary system parameters are monitored by redundant channels. Additional temperature, pressure, flow and liquid level monitoring is provided as required to keep the operating personnel informed of Plant conditions and to provide information from which Plant processes can be evaluated or regulated. The plant gaseous and liquid effluents are monitored for radioactivity. The levels are displayed and recorded and high values are annunciated.

Area monitoring stations are provided to monitor radioactivity at selected locations around the Plant. High-pressure or high-radiation conditions within the containment building initiate control action to isolate the containment.

10. Safety Injection System

Four safety injection tanks are provided, each connected to one of the four reactor inlet lines. Each tank has a volume of approximately 2,000 cubic feet containing approximately 1,000 cubic feet of borated water at a concentration of 1,720-2,500 ppm and pressurized by approximately 1,000 cubic feet of nitrogen at approximately 200 psia. In the event of a large Loss of Coolant Accident, the borated water is forced into the Primary Coolant System by the expansion of the nitrogen. The water in three tanks will adequately refill and reflood the entire core. In addition, borated water will be injected into the reactor vessel to cool the core via the same nozzles used by the SI tanks by two low-pressure and two high-pressure injection pumps taking suction from the 285,000-gallon safety injection and refueling water storage tank (SIRW). For maximum reliability, the designed capacity from the combined operation of one high-pressure and one low-pressure pump provides adequate injection flow for any Loss of Coolant Accident. Upon depletion of the storage tank supply, the high-pressure pump suction automatically transfers to the containment sump and the low-pressure pumps are shut down. One high-pressure pump has sufficient capacity to maintain the core water level at the start of recirculation. In the event of a DBA, at least one high-pressure and one low-pressure pump would receive power from the emergency power sources. Both high- and low-pressure injection pumps are located outside the containment building to permit access for periodic testing during normal operation. The pumps discharge into separate headers which lead to the containment. Test lines are provided to permit running the pumps for test purposes during Plant operation.

11. Shutdown Cooling System

The Shutdown Cooling System consists of a forced circulation heat removal loop which includes the low-pressure safety injection pumps and the shutdown heat exchangers. The system is designed to transfer heat from the Primary Coolant System to a closed loop cooling system during normal shutdown, refueling and maintenance operations.

Emergency shutdown cooling during a loss of normal and standby electrical power is accomplished by allowing natural circulation of the primary coolant to transfer heat from the core to the steam generators. The steam that is generated is released to the atmosphere as required. One of two auxiliary electric-driven feedwater pumps operating from either emergency diesel generator, or an auxiliary turbine-driven feedwater pump, supplies feedwater to the steam generators during this period. A 100,000-gallon supply of demineralized water available to these pumps is sufficient for eight hours of cooling adequate to maintain the core in a safe condition and reduce the primary coolant temperature and pressure to permit initiation of the Shutdown Cooling Systems. In addition, the Plant has the capacity for long-term cooling incorporating the ability to flush the reactor core and prevent post-LOCA boric acid precipitation.

12. Shielding

Shielding is provided so that radiation exposure of personnel will not exceed the recommended limits of 10 CFR, Part 20. The design of radiation shielding is dependent both on the extent of access required to a particular location and on the sources of radiation adjacent to that location.

The control room is shielded to permit continuous occupancy following any accidental release of radioactivity in the containment.

1.2.5 TURBINE GENERATOR

The turbine is an 1,800 r/min tandem-compound unit with external moisture separation and live steam reheating. The double-flow high-pressure element exhausts to two double-flow low-pressure elements through moisture separators and reheaters. The low-pressure elements discharge to the main condenser and the condensate is returned to the steam generators through six stages of feedwater heating. Steam is extracted for feedwater heating and for two auxiliary turbines which drive the two half-sized steam generator feed pumps.

The feedwater cycle is of the closed type with deaeration effected in the condenser. Feedwater heaters are arranged in two parallel trains, each with one high-pressure and five low-pressure heaters. Separate feedwater regulating valves control the flow to each of the two steam generators.

The 1,800 r/min, hydrogen inner-cooled generator is rated at 955,000 kVA at 75 psig hydrogen pressure, 0.85 power factor and 0.58 short circuit ratio. Field excitation is provided by a brushless exciter directly coupled to the generator shaft.

The turbine generator has a guaranteed capability of 811,776 kWe gross at 1.8 inches Hg absolute back pressure and 0.25% makeup with inlet steam conditions of 735 psia and 509°F. The maximum calculated capacity of the turbine generator is 841,817 kWe gross at 1.8 inches Hg absolute back pressure.

1.3 IDENTIFICATION OF CONTRACTORS

Consumers Power is the sole licensee for the continued operation of the Palisades Plant. As the owner and applicant, it bears the ultimate responsibility for the design, construction and safe operation of the Plant.

Consumers Power engaged Combustion Engineering, Inc (Combustion Engineering) to design and supply the nuclear fuel and the NSSS. The NSSS includes the primary system (eg, reactor vessel, steam generators, pressurizer, pumps), reactor auxiliary system components, nuclear and certain process instrumentation and the Reactor Protective System. Bechtel Corporation and its affiliate, Bechtel Company, were engaged to design and supply the balance of the Plant equipment, systems and structures. Bechtel Corporation performed the onsite construction of the original Plant, with technical advice and consultation provided by Combustion Engineering for installation of the NSSS. Subsequent to the initial Plant start-up and turnover to Consumers Power, several major modifications involving other contractors have been undertaken. Those contractors are identified in Section 1.5.

Under its contract with Consumers Power, Combustion Engineering furnished Bechtel with the design data for the NSSS. Bechtel and Consumers Power could request that Combustion Engineering make changes in the NSSS design, but Combustion Engineering did not need to accede to any such request if the proposed change, in Combustion Engineering's judgment, would be unsafe or technically unsound.

Because of the interdependence of the NSSS and certain balance-of-Plant equipment, systems and structures, Combustion Engineering furnished Bechtel with certain functional requirements for such balance-of-Plant items that affect the operability and maintainability of the NSSS or the nuclear safety of the Palisades Plant. As Bechtel's engineering work progressed, Combustion Engineering reviewed Bechtel drawings, specifications and data and Combustion Engineering was satisfied that Bechtel has understood and applied the functional requirements specified by Combustion Engineering and was satisfied that the balance-of-Plant items are compatible with the NSSS and with nuclear safety.

Palisades' original fuel vendor for cycle 1 was Combustion Engineering. Starting with cycle 2, Exxon Nuclear Corporation designed and manufactured all fuel for the reactor. Over the years, Exxon Nuclear has undergone the following company name changes: from Exxon Nuclear Corporation (ENC), to Advanced Nuclear Fuels (ANF) Corporation, to Siemens Nuclear Power (SNP), to the present name Siemens Power Corporation (SPC).

1.4 PRINCIPAL DESIGN CRITERIA

1.4.1 STATION DESIGN

Principal structures and equipment which are necessary either to prevent accidents or to mitigate their consequences were designed, fabricated and erected in accordance with applicable codes and to withstand the effects of the most severe earthquakes, flooding conditions, windstorms, ice conditions, temperature and other deleterious natural phenomena which could be expected at the site during the lifetime of this unit. Principal structures and equipment were sized for the maximum expected NSSS and turbine generator outputs.

All core physics and thermal hydraulics information contained in this report are based upon the reference core design of 2,650 MWt unless otherwise noted. The structures, systems and all postulated accidents are evaluated at either the 2,650 MWt design NSSS output or the licensed 2,530 MWt core output. Consult the specific FSAR chapters on systems or transient analyses for more detailed discussions. Section 5.1 details Palisades' conformance to all 64 General Design Criteria per 10 CFR 50, Appendix A. Section 5.2 specifies Design Codes, Structures/Systems/ Components Classification and establishes the basis for "CP Co Design Class" terminology.

1.4.2 REACTOR

1. The reactor is of the pressurized water type, designed to produce steam to drive a turbine generator. The reactor was initially operated at 2,200 thermal megawatts to produce steam at 770 psia and presently operates at 2,530 thermal megawatts, core power, producing steam at a nominal pressure of 770 psia.

2. The reactor is fueled with slightly enriched uranium dioxide contained in Zircaloy tubes.

3. The minimum departure from nucleate boiling ratio and maximum fuel center line temperature evaluated at the design overpower condition must be below values which could lead to fuel rod failures. The melting point of the UO_2 will not be reached during normal operation including expected transients.

4. Fuel rod clad thicknesses are designed to maintain cladding integrity throughout the anticipated fuel life. Fission gas release within the rods and other factors affecting design life must be considered for the maximum expected exposures.

5. The reactor and control system must be designed so that any xenon transients will be adequately damped.

6. The reactor must be designed to accommodate safely and without fuel damage tripping of the turbine generator, loss of power to the primary coolant pumps and station transients and maneuvers.

7. Power excursions which could result from any credible reactivity addition accident must not cause damage, either by motion or rupture, to the pressure vessel or impair operation of required safeguards.

8. Neutron absorption for reactivity control is provided by control rods and by dissolved boric acid in the coolant. The boron chemical shim system is completely independent of the control rod system.

9. For all operating conditions, the control rods are capable of providing an adequate shutdown margin at hot, zero power conditions following a trip, even with the most reactive rod stuck in the fully withdrawn position.

10. The boron chemical shim system is capable of adding boric acid to the primary coolant at a rate sufficient to maintain an adequate shutdown margin during primary system cooldown at the maximum design rate following a reactor trip.

11. The combined response of the fuel temperature coefficient, the moderator temperature coefficient, the moderator void coefficient and the moderator pressure coefficient to an increase in reactor thermal power is a decrease in reactivity. In addition, the reactor power transient remains bounded and damped in response to any expected changes in any operating variable.

12. The Primary Coolant Gas Vent System is designed to relieve steam or gas bubbles in the reactor vessel head and pressurizer areas of the Primary Coolant System. The system consists of a flow-limiting orifice on both the reactor vessel vent and pressurizer vent lines, solenoid valves, a pressure transmitter for pressure indication, and connecting piping. The primary vent path is directed into the open area of containment where adequate mixing with the containment atmosphere is assured.

Automatic and redundant reactor trips are provided to prevent anticipated plant transients from producing fuel or clad damage.

1.4.3 PRIMARY COOLANT AND AUXILIARY SYSTEMS

Heat removal systems are provided which can safely accommodate core heat output under all credible circumstances. Each of these heat removal systems has sufficient redundancy to provide reliable operation under all credible circumstances.

1.4.4 CONTAINMENT SYSTEM

The containment structure, including the associated access openings and penetrations, was designed to contain the pressures and temperatures resulting from a design basis accident (DBA) in which (a) the total energy contained in the Primary Coolant System water was assumed to be released into the containment through a double-ended break of one of the primary coolant pipes immediately adjacent to the reactor vessel outlet nozzle, (b) there was a simultaneous loss of external electric power, (c) heat was transferred from the reactor to containment by water supplied from the Safety Injection System but no credit was taken for cooling the fuel by water injected by the Safety Injection System, (d) the containment air recirculation and cooling system and the Containment Spray System function, and (e) the containment engineered safeguards do not operate until 30 seconds following the accident. For a discussion of the maximum hypothetical accident (MHA) refer to Section 14.22.

Means were provided for pressure and leak rate testing of the entire containment system including provisions for leak rate testing of individual piping and electrical penetrations that rely on gasketed seals, sealing compounds, or expansion bellows. Integrated leak rate testing is conducted according to Technical Specifications.

1.4.5 ENGINEERED SAFEGUARDS

Containment engineered safeguards systems with redundant features were incorporated in the Plant design which, in conjunction with the containment system and without relying upon the Emergency Core Cooling System, provide a high degree of assurance that the release of fission products to the environment following any credible Loss of Coolant Accident will not exceed the tolerances set forth in 10 CFR, Part 100.

An Emergency Core Cooling System was provided to prevent fuel and cladding damage that could interfere with adequate emergency core cooling and to limit the cladding water reaction to less than approximately 1% for all break sizes in the primary system piping up to the double-ended rupture of the largest primary coolant pipe, for any break location, and for the applicable break time. For discussion, refer to Chapter 6.

1.4.6 INSTRUMENTATION AND CONTROL

Interlocks and automatic protective systems were provided along with administrative controls to ensure safe operation of the Plant. A Reactor Protective System was provided which initiates reactor trip if the reactor approaches an unsafe condition.

Sufficient redundancy was installed to permit periodic testing of the Reactor Protective Systems and so that failure or removal from service of any one protective system component or portion of the system will not preclude reactor trip or other safety action when required.

1.4.7 ELECTRICAL SYSTEMS

Offsite and emergency sources of auxiliary electrical power were provided to assure safe and orderly shutdown of the Plant and the ability to maintain a safe shutdown condition under all credible circumstances. "Redundancy and Separation" criteria were incorporated into the associated cabling from these sources for "Safety-Related" systems/components.

1.4.8 RADIOACTIVE WASTES AND RADIATION PROTECTION

The radioactive waste treatment system was designed so that discharge of radioactivity to the environment is in accordance with the requirements of 10 CFR, Part 20, and Appendix I to 10 CFR 50.

The Plant was provided with a centralized control room having adequate shielding to permit occupancy during all credible accident situations. The radiation shielding in the Plant, in combination with Plant radiation control procedures, ensures that operating personnel do not receive radiation exposures in excess of the applicable limits of 10 CFR, Part 20, during normal operation and maintenance.

1.4.9 FUEL HANDLING AND STORAGE

Fuel handling and storage facilities were provided for the safe handling, storage and shipment of fuel and will preclude accidental criticality.

1.4.10 FIRE PROTECTION

A "Fire Protection Program" (FPP) consisting of Plant design considerations, fire detection and suppression equipment, and Plant procedures assures that the Plant can safely shut down after a major fire. The FPP complies with 10 CFR 50, Appendix R except where exemptions have been granted by the NRC.

1.4.11 CIRCULATING WATER SYSTEM

In order to minimize the environmental impact associated with "hot water" discharges, the circulating water provides condenser cooling water supplied from two mechanical draft evaporative cooling towers. Approximately 3% of the total flow is discharged as blowdown to be combined with a further temperature dilution flow before discharge to Lake Michigan. The discharges are within the Plant's NPDES Permit limitations.

1.4.12 SECURITY

Access and egress to all "protected" areas of the Plant are monitored/controlled through the utilization of card readers. Access to the Plant is controlled at the security entrance via metal detectors, guards and card readers. A physical security force is always present. Details of conformance are identified in the commission-approved physical security, safeguards contingency, and guard training and qualification plans.

1.4.13 EMERGENCY PLANNING

In the unlikely event of a Plant accident resulting in, or potentially capable of allowing, offsite releases of radioactivity in excess of federal regulations, a system of emergency warning sirens is in place. Established "emergency implementing procedures" in conjunction with the Plant's "Emergency Plans" have been developed to assure minimum risk to the general public in compliance with 10 CFR 50.54(q) and 10 CFR 50, Appendix E.

1.4.14 PLANT OPERATION

The plant's Facility Operating License requires operation to be in accordance with the Technical Specifications, which are contained in Appendix A to that license. Technical Specifications contain Safety Limits, Limiting Safety System Settings, Limiting Conditions for Operation, Design Features, and Administrative Controls, in accordance with the Code of Federal Regulations, Title 10, Part 50.36 (10CFR50). Operation of the Plant within Safety Limits and in accordance with the Limiting Safety System Settings and Limiting Conditions for Operation assures that plant operation will remain within the assumptions and initial conditions of the safety analyses. The Administrative Controls provide NRC requirements for plant staff Responsibilites, Organization, Qualifications, Procedures, Programs and Manuals, Reporting Requirements to the NRC, High Radiation Area Control, and Review and Audit functions.

1.4.15 STRUCTURES

Plant structures were designed in accordance with the design criteria identified in Chapter 5. Structures were identified as CP Co Design Class 1, 2 or 3 according to Section 5.2. Specific design criteria for containment is discussed in Section 5.8, and other CP Co Design Class 1 structures are discussed in Section 5.9.

1.5 MAJOR PLANT MODIFICATIONS (DESIGN/CONSTRUCTION)

Following initial completion of the Palisades Plant in 1971, several major facility modifications have been made to improve the safety and operability of the Plant. These modifications are briefly outlined below, with references to the appropriate FSAR Update section and identification of the designer/constructor.

CONDENSER RETUBING - BECHTEL/J A JONES

In 1974, due to condenser tube leakage problems, the Admirality tube section of the main condenser was replaced with 90-10 copper-nickel tubes.

CONDENSER REPLACEMENT - YUBA/TOWNSEND & BOTTUM

In 1990 the main condenser was replaced to eliminate copper-related corrosion concerns with the new steam generators. The replacement condenser is a Yuba design that contains Type 439 stainless steel tubing. See Subsection 10.2.3.1.

STEAM GENERATOR TUBE PLUGGING - CP CO/CP CO

Since 1973 a total of 2,044 tubes in Steam Generator A and 2,442 tubes in Steam Generator B have been plugged due to tube wall degradation resulting from the following: (1) secondary side standard phosphate water treatment, (2) intergrannular corrosion and (3) tube denting. An all-volatile secondary system water chemistry with boric acid has been implemented to reduce further tube degradation. See Subsections 4.3.4.1 and 4.3.4.2.

FEEDWATER PURITY BUILDING ADDITION - BECHTEL/J A JONES

A completely new secondary side feedwater (condensate) purity system was installed to provide full flow condensate demineralization system utilizing powdered ion exchange resins and on-line resin body feed capability. This new system is housed in the feedwater purity building addition. See Subsection 10.2.3.2. This system has since been deactivated.

COOLING TOWERS ADDITION - ECODYNE/ECODYNE

Initially, the Plant was designed for a once-through Circulating Water System for providing cooling water to the condenser. For environmental reasons, the system was converted in 1974 to a closed-cycle system using two mechanical draft cooling towers and blowdown dilution. A cooling tower pump house was constructed to enclose the cooling tower pumps. See Subsection 10.2.4.

RADWASTE SYSTEM MODIFICATIONS/AUXILIARY BUILDING ADDITION - BECHTEL/BECHTEL

During 1971-1973 the liquid waste management system was modified to reduce liquid discharges to "Near Zero" and meet the requirements of 10 CFR 50, Appendix I. The auxiliary building was expanded to enclose much of the new equipment required. The 1972-1973 service building addition, in conjunction with the aforementioned changes, was made to accommodate solid radwaste system changes designed by Protective Packaging Inc (PPI). This system was subsequently replaced by a new molten bitumen immobilizing system for waste concentrates. See Chapter 11 for details.

SPENT FUEL POOL STORAGE MODIFICATIONS - NUS/J A JONES 1977 - WESTINGHOUSE/WESTINGHOUSE 1987

In 1977, the spent fuel pool storage capacity was increased from a capacity of 272 assemblies to 798. In 1987, Amendment 105, dated July 24, 1987, authorized replacing existing racks with six high-density spent fuel racks that increased the storage capacity from 798 to 892 fuel assemblies. See Section 9.4 and Subsection 9.11.3.

HIGH-PRESSURE AIR ADDITION - BECHTEL/J A JONES

In 1977, the compressed air system was augmented by the addition of a high-pressure air system (325 psig) for supply to safety-related air-operated valves and components. See Section 9.5.

FIRE PROTECTION SYSTEM MODIFICATIONS - NUCLEAR SERVICES CORP/QUADREX/J A JONES

Following the 1975 Browns Ferry fire and subsequent NRC revised guidelines between 1976 and the present, Consumers Power has undertaken a series of studies and resultant Plant fire protection features modifications. This has included the addition of fire-fighting equipment, separation of cables, addition of fire stops, preparation of procedures, etc. See Section 9.6 and Chapters 7 and 8.

AUXILIARY FEEDWATER MODIFICATION - BECHTEL/BECHTEL

As a result of lessons learned at TMI, the Auxiliary Feedwater System has been upgraded to a safety-related system. See Sections 9.7 and 7.4.

METEOROLOGICAL PROGRAM IMPROVEMENTS - EG&G/CP CO

Several modifications were made to the onsite meteorological towers including the addition and relocation of new towers. A final meteorological program was attained in 1977 following a 1975 study by EG&G Environmental Consultants. See Subsection 2.5.2.3 for a description of the new tower and meteorological program.

AUXILIARY BUILDING TSC/EER/HVAC ADDITION - BECHTEL/BECHTEL

During 1983 an addition was added to the north side of the auxiliary building to house a technical support center (TSC), an electrical equipment room (EER) and a heating, ventilating and air conditioning (HVAC) area. The TSC was required to fulfill the guidelines of NUREG-0696, the HVAC area as a result of the control room habitability requirements of NUREG-0737, and the EER area as a result of loads placed on the electrical system by the addition of the TSC and HVAC areas. See Section 9.8 for discussion of the HVAC system, Chapter 8 for discussion of the electrical equipment and the Site Emergency Plan for the functional discussion of the TSC.

INTERIM OLD STEAM GENERATOR STORAGE FACILITY - BECHTEL/BECHTEL

In 1990, a reinforced concrete building was constructed for interim storage of two old steam generators. This facility is located in the controlled area of the site approximately 2,200 feet northeast of the containment building. The storage facility design provides sufficient radiation shielding such that the onsite and offsite dose rate will not exceed the limits defined in 10 CFR 20 and 40 CFR 190, respectively. The facility is designated as a secondary restricted area. The old steam generators will remain in this facility until an ultimate disposition method is selected.

INDEPENDENT SPENT FUEL STORAGE INSTALLATION (ISFSI) - PACIFIC SIERRA NUCLEAR

In 1993 Palisades constructed a suitable facility and began dry storage of spent nuclear fuel in casks under the General License provisions of 10CFR72. This facility is located north of the support building and is enclosed within the plant security fence. The ISFSI was planned to hold 25 Ventilated Storage Casks (VSCs) designed by Pacific Sierra Nuclear (later Sierra Nuclear) Corporation although other NRC-certified cask designs could also be utilized if the associated fuel handling equipment were procured.

1.6 INSERVICE INSPECTION

1.6.1 HISTORICAL BACKGROUND

The Palisades Nuclear Power Plant was built in the late 1960s and was placed in commercial service on December 31, 1971. During the first 40-month life of the Plant, in order to comply with Paragraphs 4.3 and 4.12 of the Technical Specifications (dated September 1, 1972) of the Provisional Operating License DPR-20 for the Palisades Nuclear Plant, which discusses ISI requirements of ASME Class 1 components and systems, the nondestructive examinations were performed to satisfy the requirements of the ASME Boiler and Pressure Vessel Code, Section XI, 1971 Edition, including the Winter 1972 Addenda (ASME B&PV Code, Section XI, 71W72a). In February 1976, the NRC amended Paragraph 55a (g) of 10 CFR 50 to require nuclear plants to upgrade their Technical Specifications in the areas of the ISI requirements and the functional testing of pumps and valves. By amending Paragraph 55a (g) and by invoking Regulatory Guide 1.26, the NRC required nuclear plants to upgrade their ISI program to include not only ASME Class 1 systems, but also ASME Class 2 and ASME Class 3 systems.

1.6.2 GENERAL

The Inservice Inspection Plan for the four 10-year inservice intervals was developed by Southwest Research Institute and Consumers Power Company, and reviewed and approved by Consumers Power Company for use at Consumers Power Company's Palisades Nuclear Power Plant. Subsequent updating to remain responsive to industry requirements is anticipated.

The start of the first 10-year interval coincides with the date of first commercial operation, December 31, 1971. The length of the first 3-1/3-year period was extended to October 30, 1976 by adding 18 months cumulative shutdown time between August 1973 and April 1975 in accordance with ASME B&PV Code, Section XI, IS-241, 71W72a. The second period ran to June 1, 1980 due to the 1979/1980 extended refueling outage. The third period extended to November 9, 1983 per ASME B&PV Code, Section XI, IWA-2400(c), 77S78a.

The second 10-year interval began November 10, 1983, and lasted until May, 1995.

The third 10-year interval began May 12, 1995, and is expected to end approximately May, 2005.

See Section 6.9 for details of the Inservice Inspection Program.

1.7 RESEARCH AND DEVELOPMENT REQUIREMENTS

The design of the Palisades Plant is based upon concepts which have been successfully incorporated in pressurized water reactor systems. However, certain Palisades specific Combustion Engineering development tests have been performed and are listed below. These tests were completed prior to initial Plant start-up.

1.7.1 FLOW MIXING AND FLOW DISTRIBUTION

Tests have been run to measure the flow mixing factor. Dye dispersion rate was measured in a series of full-scale and larger than full-scale mock-ups of various fuel bundle flow channel configurations. These tests were run in a cold-water test loop.

A larger than full-scale model of the Palisades bundle inlet region has been tested in a cold-water flow loop to determine the effect of minor flow maldistributions due to inlet structure nonuniformities, and to verify the effectiveness of certain steps taken to improve inlet flow.

1.7.2 CONTROL ROD TESTS

A series of tests were run to demonstrate the adequacy of the control rod and its guidance system. Cold-water flow testing of a slightly underscale model of four bundles and a cruciform control rod was performed. These tests were conducted with mechanical misalignments exceeding design values.

In addition to the cold-water tests, a test program has been performed using a full-scale model, including a prototype mechanism under reactor conditions of flow, temperature and pressure in the CE Utility Reactor Components Test Facility at Windsor. The purpose of this program was to assess the effects of mechanical misalignments, of thermal distortions which have been measured on model fuel bundles and on a prototype control rod, of cross flow and of upper limit conditions of axial flow and system pressure.

A series of mechanical tests have been run on structural components of the Ag-In-Cd control rod blade.

1.7.3 CONTROL ROD DRIVE MECHANISMS

An extensive development program has been completed on the control rod drive mechanisms. This program has included up to 130,000 feet of travel on various components and 350 full-height drops. The production mechanism design incorporates improvements derived from experience gained on this program.

1.7.4 FUEL BUNDLE DESIGN

Combustion Design

Cold-water flow-induced vibration tests on fuel rods and subassemblies and mechanically induced vibration tests in air on model bundles have been completed. An extensive hot flow test program, including the effects of forced cross flow, has been completed using essentially full-length, though not full cross-section, bundles. Total time at reactor conditions (or conditions believed to be more severe) exceeded 13,000 hours. These tests have been supplemented by a basic program on the mechanism of grid to fuel rod wear conducted in a static autoclave with mechanically induced relative motion.

These tests substantiate the adequacy of the fuel bundle design for its expected service.

In addition to this program, four full-scale model fuel bundles were tested by Combustion Engineering at reactor conditions (or more severe conditions) in the Utility Reactor Components Test Facility. Beginning with the second fuel cycle, Combustion Engineering fuel has not been used.

Current Fuel Design

Comparable developmental testings, as described previously, have also been performed by the current fuel vendor. Refer to Subsection 3.3.4.3 for details.

1.7.5 REACTOR VESSEL FLOW TESTS

A one-fifth scale model of the reactor vessel and its internals has been constructed and subjected to airflow testing at the Battelle Memorial Institute Laboratories at Columbus, Ohio. These tests have investigated flow distribution, pressure drop and the tracing of flow paths within the vessel for all four pumps running and various part-loop configurations.

1.8 SPECIAL MAJOR PROGRAMS

As a result of continued NRC concern with the "health and safety of the public" and its relationship to the safe operation of all nuclear facilities, the Palisades Plant and operations have been subjected to considerable NRC review.

The Inspection and Enforcement Branch of the NRC routinely provides IE Bulletins (for utilities to review and respond) regarding the identification of generic problems that could have a safety impact. In addition, the NRC on occasion establishes technical review programs as a result of legal mandates following court actions or NRC initiated programs resulting from unusual events in the industry.

Programs of special interest to Palisades resulting from these circumstances are discussed in the remainder of this section.

1.8.1 SYSTEMATIC EVALUATION PROGRAM

The SEP was initiated by the US Nuclear Regulatory Commission (NRC) to review the designs of older operating nuclear reactor plants; to confirm and document their safety. The review compared the as-built plant design with current review criteria in 137 different areas defined as "topics."

During the review, 47 of the topics were deleted from consideration by SEP, based on one of the three following reasons: (1) topic was part of the Unresolved Safety Issue Program (USI), (2) topic was part of Three Mile Island Action Plan Tasks, or (3) the topic was not applicable to the Plant. The remaining 90 topics were reviewed for Palisades and are listed in Table 1-3. Sixty-two of the 90 topics met current criteria or were acceptable on another defined basis. These topics are identified as Status Code S on Table 1-3.

1.8.1.1 Integrated Assessment (NUREG-0820)

Twenty-eight topics received further review and evaluation during the Integrated Assessment Program. A major part of the integrated assessment was the probabilistic risk assessment (PRA). PRA was used to determine which system failures would create an unacceptable risk because either a redundant system was not available or available systems were inadequate for the job required.

During the integrated assessment, several additional topics were found to be acceptable and required no further work. These items are identified as Status Code 4 in Table 1-3.

The remaining topics evaluated using PRA, were each found to require one or more of the following modifications:

1. Plant changes

2. Technical Specifications changes

3. Refined engineering analysis required

These items are identified as Status Code 1, 2 or 3 in Table 1-3. The disposition column of Table 1-3 shows where specific topics are addressed.

1.8.2 TMI ACTION ITEMS (NUREG-0737)

As a result of the incident at Three Mile Island Nuclear Power Plant, the NRC developed a list of requirements for other nuclear-powered generating stations. The list consists of 37 items, which are broken down into a total of 94 subitems as identified in NUREG-0737.

The list of items, compliance status and general description of how the item was to be resolved, are shown in Table 1-4. All of the 94 subitems have been closed out and are identified in the table as Status Code 1. The NRC SER for issuance of the Full Term Operating License, NUREG 1424 dated November 1990, confirms that the NRC views all TMI Action Items as being resolved for Palisades. Some requested changes to Technical Specifications may still be outstanding, however, awaiting NRC action.

1.8.3 PIPE SUPPORT BASEPLATE DESIGNS USING CONCRETE EXPANSION ANCHOR BOLTS (IE BULLETIN 79-02)

Nuclear Regulatory Commission IE Bulletin 79-02, addressed Seismic Category I pipe supports using concrete expansion anchor bolts (CEBs) for loadings obtained from analysis of Seismic Category I piping systems.

All baseplates or structural steel members using CEBs for large piping identified in the course of responding to IE 79-02 were evaluated. The evaluation was performed in accordance to the load combinations specified in Section 5.10. Acceptance criteria were as specified in the Bulletin for CEBs and Chapter 5 for baseplates or structural steel members. Those baseplates or structural steel members and CEBs which did not satisfy acceptance criteria were modified (or will be modified).

Approximately 3,000 accessible CEBs for large bore piping were inspected and load tested. More than 96% of this population satisfied the load testing.

Approximately 4% of the CEBs for large bore piping were inaccessible for full testing and inspection. These CEBs and their baseplates or structural steel members were evaluated. If these CEBs and baseplates or structural steel members did not satisfy acceptance criteria, they were either modified or the piping support system was revised to yield acceptable results.

Small bore piping supports were designed using a conservative chart method. A sample of CEBs used for support of small piping was inspected and tested. This sample consisted of more than 1,000 CEBs (more than 2/3 of the population). This testing and inspection program used in conjunction with the conservative chart method, yields an acceptable confidence level for small piping.

Thus, the inspection, testing and evaluation performed for baseplates or structural steel members and CEBs for Seismic Category I piping, satisfy the requirements of IE 79-02, and the modifications have been completed.

1.8.4 SEISMIC ANALYSIS FOR AS-BUILT SAFETY-RELATED PIPING SYSTEMS (IE BULLETIN 79-14)

The bulletin required an inspection of approximately 18,100 feet of large diameter safety-related piping, 1,550 pipe supports and piping components at the Palisades Plant. Small piping systems (2 inches or less in diameter) were also inspected, noted items evaluated and a sample of the small piping systems was evaluated.

The Palisades Plant systems were reviewed and it was determined that 23 systems had safety-related piping. Data on and sketches of the safety-related systems were completed, potential nonconformance items were listed and the as-built data were evaluated. Approximately 320 piping support changes have been completed: (1) 45 new supports were added, (2) 23 supports were removed, and (3) 252 supports were modified.

There were approximately 3,250 listed conditions that were either questionable or constituted a discrepancy. These items were evaluated and resolved. About 75% of the items related to lack of or nonconformance with existing drawings. The remaining 750 items related to hardware conditions, such as nuts/bolts loose or missing, spring cans bottomed out or without load and bent/broken or missing pipe support components.

Two Licensee Event Reports (LERs) were issued as a result of the program (LER 79-033 and LER 80-001). Corrective action has been completed on both items.

1.8.5 UNRESOLVED SAFETY ISSUES (NUREG-0410)

Of the unresolved safety issues (also called generic safety issues) identified by the NRC and discussed in NUREG-0410, -0510, -0649, and -0705, 19 were considered by the NRC to require investigation for their potential impact on the Palisades Nuclear Plant. The 19 unresolved safety issues considered are listed in Table 1-5 with a cross-reference to the appropriate FSAR chapter wherein they are discussed. All 19 of the unresolved safety issues have been assessed by Consumers Power Company and are considered to have no undue risk to the health and safety of the public while longer term generic review of these issues is being conducted.

USI A-46, Seismic Qualification of Equipment in Operating Plants, is currently being resolved by the Seismic Qualification Utility Group (SQUG). Consumers Power Company is a member of SQUG.

1.8.6 ENVIRONMENTAL QUALIFICATION OF "SAFETY-RELATED" ELECTRICAL EQUIPMENT (EEQ) (NUREG-0588) (USI A-24)

In order to assure the reliable functions of certain electrical equipment subjected to harsh environments following accident conditions, each licensee was requested to reevaluate all previously installed equipment. Pursuant to an NRC order issued August 29, 1980, Consumers Power Company engaged in the preparation of EEQ documentation. That information was submitted to the NRC on October 7, 1980 in a report entitled, "Environmental Qualification of Safety Related Electrical Equipment - Palisades Plant," September 1980. Revisions to the report were submitted to the NRC from October 29, 1980 to May 20, 1983. Industry resolution of this issue was embodied in new rule 10CFR50.49. For Palisades the NRC issued an SER on January 31, 1985 which contained the finding that the plant's environmental qualification program was in compliance with 10CFR50.49.

1.8.7 CONTROL ROOM HABITABILITY (NUREG-0696)

In 1983 Consumers Power Company completed several modifications to the control room HVAC system to satisfy the control room habitability requirements of NUREG-0737. These included extending the control room air intake from the then existing configuration, increasing the intake air duct to allow 100% makeup air, installing redundant charcoal filters, extending the control room habitability zone and replacing air intake and discharge dampers. The safety evaluation concluded that the systems will provide safe, habitable conditions within the control room under both normal and accident radiation and toxic gas conditions, including Loss of Coolant Accidents. See discussion in Section 6.10.

1.8.8 EFFECTS OF PIPE RUPTURE (SEP TOPICS III.5.A AND B)

In December 1972, the NRC initially raised the concern for the dynamic effects of pipe ruptures. In response, Consumers Power Company had an analysis done for postulated high-energy line breaks outside of containment. This report is entitled "Special Report No 6 - Analysis of Postulated High-Energy Line Breaks Outside of Containment" (SR-6), Revision 3.

The concerns of high-energy line breaks and moderate-energy line breaks were further reviewed for both situations inside and outside containment in SEP Topics III.5.A and III.5.B, respectively. The conclusions were that the criteria used to assess pipe breaks at the Palisades Nuclear Plant was in accordance with present-day criteria. For discussion, see Section 5.6.

1.8.9 STATION BLACKOUT (10 CFR 50.63) (USI A-44)

In 1988 the NRC issued 10 CFR 50.63 to define a loss of all onsite AC power sources (Station Blackout) as an event with which all plants must be able to cope. By letters dated April 17, 1989, December 11, 1989, March 27, 1990, and July 3, 1990, Consumers Power Company certified that the Palisades evaluation of the issue was completed in accordance with the specified guidance, NUMARC 87-00, and that Palisades possessed the required coping ability. In a letter dated May 20, 1991, the NRC issued its SER, which stated "...we find that the Palisades Plant conforms to the SBO rule, and the guidance of

Regulatory Guide (RG) 1.155, Nuclear Management and Resources Council (NUMARC) 87-00, and NUMARC 87-00 Supplemental Questions/Answers and Major Assumptions." NRC acceptance was contingent upon satisfactory resolution of several included recommendations. CPCo letter dated August 1, 1991 provided responses/commitments to resolve those recommendations. Final NRC closure was provided in an SER dated June 25, 1992.

TABLE 1-1
(Sheet 1 of 2)

CHRONOLOGICAL LICENSING EVENTS

Date	Docket 50-255 CPOL Application	FSAR	Operating License DPR-20	Other
June 2, 1966	CP Co Files CPOL Application	PSAR Included With Application		
March 14, 1967	AEC Issues CPPR-25			
November 1, 1968	CP Co Files Amendment 9 to CPOL Application	FSAR, Rev 0, Included	Initial Operating License Application for Operation at 2,200 MWt Core	
October 9, 1970				CP Co File Construction Stage Environmental Report
March 24, 1971			NRC Issues IDPR-20 (1 MW)	
August 18, 1971				CP Co Files ER Supplement (Special Report 4)
November 20, 1971			Amendment 1 - Authorization for 20% Power (440 MW)	
March 10, 1972			Amendment 2 - Authorization for 60% power (1,320 MW)	
June 1972				NRC Issues Final Environmental Statement
July 1972				CP Co Files Operating Stage ER
September 1, 1972	NRC Issues OL DPR-20		Amendment 3 - Authorized 60% Power	
October 16, 1972			Amendment 4 - Authorized 100% Power (Limited to 60% Power)	
March 23, 1973			Authorization for 100% Power (Limited to 85% Power)	
January 22, 1974	CP Co Files Amendment 28 to CPOL	Full Revision to FSAR (Denoted as Amendment 28)	CP Co Application for Full-Term Operating License at 2,638 MWt Core	
December 12, 1974				NPDES Permit Granted by Michigan DNR

TABLE 1-1
(Sheet 2 of 2)

CHRONOLOGICAL LICENSING EVENTS

Date	Docket 50-255 CPOL Application	FSAR	Operating License DPR-20	Other
August 2, 1976				Modified NPDES Permit Granted
August 12, 1977			CP Co Requests the OL Limit Be Increased to 2,530 MWt Core Power	
November 1, 1977			NRC Issues Amendment 31 to DPR-20 Granting Power Operation up to 2,530 MWt	
February 1978				NRC Initiated Systematic Evaluation Program (SEP)
October 1978				NRC Issues Final Addendum to the Final Environmental Statement
October 1982				SEP Integrated Plant Safety Assessment Final Report (NUREG 0820) issued
November, 1983				Supplement 1 to SEP Integrated Plant Safety Assessment Report issued.
June 28, 1984		Revision 0, Initial FSAR Update Pursuant to 10 CFR 50.71(e)		
October 22, 1990				NRC issues Environmental Assessment in support of POL to FTOL conversion
November 21, 1991				NRC issues SER (NUREG 1424) in support of POL to FTOL conversion
February 21, 1991			FTOL issued by NRC	

TABLE 1-2
(Sheet 1 of 12)

PALISADES ORIGINAL DESIGN CHARACTERISTICS (NOMINAL VALUES)*

Plant

Net Electrical Power Output @ 2,200 MWt	700 MWe
Net Electrical Power Output @ 2,530 MWt	740 MWe
Maximum Expected Gross Electrical Output @ 2,650 MWt	845 MWe

Nuclear Steam Supply System

Core Thermal Output	2,200 MWt Initial, 2,638 MWt Ultimate, 2,530 MWt Presently
Operating Pressure	2,100 psia Initial Rating, 2,250 psia for Ultimate Rating, 2060 present nominal rating
Design Pressure	2,500 psia
Primary Coolant Inlet Temperature	545°F Initial Rating, 537.3°F Presently
Primary Coolant Outlet Temperature	591°F Initial Rating, 582.7°F Presently
Pipe Size: Outlet (ID)	42"
(Wall Thickness)	4"
Inlet (ID)	30"
(Wall Thickness)	3"

*See detailed discussions in later chapters and Technical Specifications for actual required values for parameters and required design characteristics of Plant equipment.

Flow per Loop	62.5×10^6 lb/h original 72.3×10^6 lb/h present nominal
Number of Loops	2
Number of Pumps	4
Type	Vertical, Centrifugal, Mechanical Seals
Design Flow/Pump	83,000 gpm
Design Head	260'
Core	
Total Heat Output	2530 Mw_t
Heat Generated in Fuel	97.5%
Design Thermal Overpower	15%
DNB Ratio at Nominal Conditions	2.00
Minimum DNBR for Design Transients	
(XNB Correlation)	1.17
(ANFP Correlation)	1.154
(HTP Correlation)	1.141
Core Power Density	69.3 kW/Liter original 79.8 kW/Liter presently
Number of Fuel Bundles	204
Number of Fuel Rods/Bundle Initial Core Loading (A, B, C1 + Typical Reload Fuel)	212
Number of Fuel Rods/Bundle, Cycle 11	216 typical
Fuel Rod Pitch	0.550"
Fuel Clad Material	Zircaloy-4
Fuel Clad Thickness	0.0275" minimum

TABLE 1-2
(Sheet 3 of 12)

Number of Full-Length Control Rods	41
Number of Part-Length Control Rods	4
Control Rod Pitch	16.97"
Absorber Material	Silver-Indium-Cadmium
Control Rod Drive Type	Rack and Pinion
Equivalent Core Diameter 136.7"	
Total Uranium as UO_2	80 metric tons
Enrichment (Wt % U-235)	
Batch A	1.65
Batch Q (Average)	3.77
Reactor Vessel	
Inside Diameter	172"
Overall Length	40'-3/4"
Wall Thickness Without Clad	8-1/2"
Wall Material	SA-302
Cladding Thickness	3/16"
Cladding Material	SS-308/309
Design Temperature	650°F
Design Pressure	2,500 psia
NDT Temperature (Initial)	40°F
Total Weight	426 Tons

Steam Generators

Number of Units	2
Type	Vertical "U" Tube
Outside Diameter	20'-10"
Length	59'-2"
Number of Tubes	8,219 Initial Design
Tube OD	3/4"
Tube Material	Inconel
Shell Material	SA-302B and SA-516, Gr 70
Primary Side	
Tube Side Design Pressure	2,500 psia
Tube Side Design Temperature	650°F
Tube Side Operating Pressure	2,060 psia
Coolant Inlet Temperature (nominal)	582.7°F
Coolant Outlet Temperature (nominal)	537.3°F
Bottom Head Clad Material	SS-304
Secondary Side	
Shell Side Design Pressure	1,000 psia
Shell Side Design Temperature	550°F
Operating Pressure (Steam Generator Outlet at Plant Rating of 2,530 MWt Core)	760-770 psia presently
Operating Temperature	512°F
Quality	99.8%

TABLE 1-2
(Sheet 5 of 12)

Feedwater Inlet Temperature	435°F
Steam Flow/Steam Generator (10^6 lb/h)	5.485
Turbine Cycle	
Turbine Design	Tandem-Compound, 1 HP, 2 LP Turbines
Exhaust Pressure	1.8 inHg
Makeup	0.25%
Steam Atmospheric Dump (Rated Steam Flow)	35%
Steam Bypass to Condenser (Rated Steam Flow)	5%
Feedwater Heater Stages	6
Condensate Pumps - Number	2 - Half Capacity
Design Flow	9,250 gpm
Design Head	1,000'
Feedwater Pumps - Number	2 - Half Capacity
Design Flow	13,500 gpm
Design Head	1,920'
Condenser Circulating Pumps - Number	2 - Half Capacity
Design Flow	205,000 gpm/Pump
Design Head	90'
Generator	
Design Rating	955 MVA
Power Factor	0.85
Terminal Voltage	22,000

NSSS Auxiliary Systems

1. Chemical and Volume Control System

Normal Letdown Flow Rate	40 gpm
Maximum Letdown Flow Rate	133 gpm
Charging Pumps - Number	2 - Fixed Capacity 1 - Variable Capacity
Design Flow	40 gpm
Design Head	6,375'
Metering Pumps - Number	1
Design Flow Range	0 to 35 gpm
Maximum Outlet Pressure	120 psig
Regenerative Heat Exchanger - Number	1 - Full Capacity
Design Heat Transfer	6.6×10^6 Btu/h
Letdown Heat Exchanger - Number	1 - Full Capacity
Design Heat Transfer	19.1×10^6 Btu/h
Demineralizers - Number	2 - Purification 1 - Deborating
Nominal Rating	40 gpm
Maximum Flow	160 gpm
Resin Type	H-OH
Resin Volume	32 ft^3

TABLE 1-2
(Sheet 7 of 12)

Filter - Number	2
Type	Cartridge
Design Rating	120 gpm
Filter Size	2 Microns

2. Safety Injection System

Safety Injection Tanks - Number	4
Volume - Total	2,000 ft^3
Borated Water	1,000 ft^3
Nitrogen @ 200 psia	1,000 ft^3
Design Pressure	250 psia
Design Temperature	200°F
High-Pressure Pumps - Number	2 - Full Capacity
Rating, Each	300 gpm
Head	2,500'
Low-Pressure Pumps - Number	2 - Full Capacity
Rating, Each	3,000 gpm
Head	350'

3. Containment Air Cooling System

Air Coolers - Number	4 (3-safety related)
Rating, Each Safety Related Cooler (Btu/h) @ 283°F, 55 psig	87.2×10^6 (DBA)
Cooling Water Flow, Each	Set per FSAR Sect 9.1.2.3 (DBA)
Airflow	30,000 ft^3/min (DBA)

Rating (Safety Related Cooler)	1.98×10^6 Btu/h (Normal)*
(Non-safety Related Cooler)	1.38×10^6 Btu/h (Normal)**
Cooling Water Flow, Each	500 gpm (Normal)
Airflow	60,000 ft^3/min (Normal)
4. Containment Spray System	
Spray Pumps - Number	3 - Half Capacity
Rating, Each	1,340 gpm
Head	450'
Heat Exchangers (Shutdown Cooling Heat Exchangers)	2
Rating, Each (Btu/h)	83.5×10^6 @ 283°F
5. Hydrogen Recombiners	
Number of Recombiners	2 - Full Capacity
6. SIRW Tank	
Fluid Volume	285,000 Gallons
Boron Concentration	1720-2500 ppm
7. Shutdown Cooling System	
Auxiliary Feedwater Pumps - Number	3 - Full Capacity
Turbine Driven - Number	1
Rating	415 gpm
Head	2,730'

*Safety related cooler performance based on 115°F EAT & 81.5°F EWT

**Non-safety related cooler performance based on 104°F EAT & 75°F EWT

Motor Driven - Number	2
Approximate Rating (Pump P8A) (Pump P8C)	415 gpm 330 gpm
at approximate Head (Pump P8A) (Pump P8C)	2,730' 2,260'
Pumps - Use Low-Pressure Safety Injection Pumps	
Heat Exchangers - Number	2 - Half Capacity
Rating, Each	80.0×10^6 Btu/h
8. Component Cooling System	
Component Cooling Pumps - Number	3 - Half Capacity
Rating, Each	6,000 gpm
Head	164'
Heat Exchangers - Number	2
Rating, Each	50.5×10^6 Btu/h (Normal)
Rating, Each	85.0×10^6 Btu/h (Post-DBA)
9. Spent Fuel Cooling System	
Spent Fuel Pool Capacity	4 Cores
Volume	21,317 ft^3
Fuel Assemblies	892
Pumps - Number	2 - Half Capacity
Rating, Each	1,700 gpm
Head	64'

Heat Exchanger - Number	2
Rating	23×10^6 Btu/h
Filter - Number	1
Type	Cartridge
Rating	150 gpm
Size	25 Microns
Demineralizer - Number	1
Resin Type	H-OH
Bed Size	68 ft^3
Nominal Flow	150 gpm

10. Shield Cooling System

Pumps - Number	2 - Full Capacity
Rating, Each	180 gpm
Head	79'
Heat Exchanger - Number	1 - Full Capacity
Rating	2 x 10 Btu/h
Sets of Cooling Coils - Number	2 - Full Capacity

Conventional Plant Auxiliary Systems

1. Service Water System

Service Water Pumps - Number	3 - Half Capacity
Rating	8,000 gpm
Head	140'

2. Compressed Air System

Compressors - Number	3
Rating	200 scfm
Discharge Pressure	100 psig

3. High-Pressure Air Systems

Compressors - Number	3
Rating, Each	22.3 scfm
Discharge Pressure	325 psig

Containment

Type	Reinforced Concrete, Prestressed, Post-Tensioned
Diameter	116'-0" ID (Inside)
Height	190'-6" (Inside)
Liner - Material	A442 Plate
Thickness	1/4"
Design Pressure	55 psig
Design Temperature	283°F

TABLE 1-2
(Sheet 12 of 12)

Free Volume	1.64 x 10 ft^3
Leak Rate	0.2%/Day
Electrical Equipment	
Main Transformer - Rating	875 MVA
Voltage	345 kV
Diesel Generators - Number	2 - Full Capacity
Rating	3,000 kVA
Fuel Oil Capacity	1 Week Following a DBA
Station Battery - Number	2
Type	Lead Calcium
Number of Cells	59
Rating	125 V, 1,800 Ah
Chargers - Number	4
Inverters - Number	4
AC/DC Voltage	120/125
Rating	6 kVA

SYSTEMATIC EVALUATION PROGRAM (SEP)
AND INTEGRATED ASSESSMENT PROGRAM (IAP)
PALISADES PLANT - NUREG-0820

Topic	Title	Status(a)	IAP Disposition and Location Where Used in FSAR Update
II-1.A	Exclusion Area Authority and Control	4/N	(Evaluation acceptable.) Section 2.1.
II-1.B	Population Distribution	S	Section 2.1.2.
II-1.C	Potential Hazards or Changes in Potential Hazards Due to Transportation, Institutional, Industrial and Military Facilities	S	Sections 2.1.3 and 5.5.4.
II-2.A	Severe Weather Phenomena	S	Sections 8.7.1, 8.7.2 and 8.7.3.
II-2.C	Atmospheric Transport and Diffusion Characteristics for Accident Analysis	S	Section 2.5.
II-3.A	Hydrologic Description	S	Section 2.2.
II-3.B	Flooding Potential and Protection Requirements	3/C	(CP Co seiche evaluation established maximum flood level of 593.5 MSL accepted by staff October 7, 1982.) Sections 2.2.2 and 5.4.1.
II-3.B.1	Capability of Operating Plant To Cope With Design-Basis Flooding Conditions	3/C	(Same as II-3.B.) Section 5.4.1.
II-3.C	Safety-Related Water Supply (Ultimate Heat Sink (UHS))	3/C	(Same as II-3.B.) Sections 5.4.1 and 9.1.
II-4	Geology and Seismology	S	Sections 2.3 and 2.4.
II-4.A	Tectonic Province	S	Section 2.3.
II-4.B	Proximity of Capable Tectonic Structures in Plant Vicinity	S	Section 2.4.
II-4.C	Historical Seismicity Within 200 Miles of Plant	S	Section 2.4.
II-4.D	Stability of Slopes	S	Section 2.4.
II-4.F	Settlement of Foundations and Buried Equipment	S	Section 2.3.5.
III-1	Classification of Structures, Components and Systems (Seismic and Quality)	3/C	(Additional information supplied on radiography fracture toughness, valves, pumps and storage tanks accepted by staff June 27, 1983.) Sections 5.1.5 and 11.2.1; and Tables 5.2-1, 5.2-2 and 5.2-3.

Topic	Title	Status(a)	IAP Disposition and Location Where Used in FSAR Update
III-2	Wind and Tornado Loadings		
	- SIRW and Condensate Storage Tank	2/C	(New EOPs were issued in December 1986.) Sections 5.3.2 and 5.3.3.
	- Emergency Generators Supply and Exhaust Piping	4/N	(Loss of both diesel generators and offsite power could present safe shutdown problems, however, the probability of both generators being disabled, precludes any back-fitting requirements.) Sections 5.3.2 and 5.3.3.
	- Spent Fuel Pool Enclosure	4/N	(The ability of the Plant to shut down is not affected by the inability of the steel enclosure to withstand tornado loadings, precluding any backfit requirements.) Sections 5.3.2 and 5.3.3.
III-3.A	Effects of High Water Level on Structures	4/N	(The NRC concluded that the design approach used for Palisades CP Co Design Class 1 structures was adequate for resisting a design basis flood.) Sections 5.4.1 and 5.9.1.
III-3.C	Inservice Inspection of Water Control Structures		
	- Cooling Water System Structures Inspection	2/C	(Inspection began in 1982.)
	- Flood Protection Structures	4/N	(Inspection part of Preventative Maintenance Program.)
III-4.A	Tornado Missiles		
	- SIRW and Condensate Storage Tanks	2/C	(New EOPs were issued in December 1986.) Sections 5.3.3 and 5.5.1.
	- Emergency Diesel Supply and Exhaust Piping	4/N	(Same as III-2.) Sections 5.3.3 and 5.5.1.
	- Atmospheric Relief Stacks of Steam Relief Valves	4/N	(Same as III-2.) Sections 5.3.3 and 5.5.1.
	- Compressed Air System	4/N	(Same as III-2.) Sections 5.3.3, 5.5.1 and 9.5.2.1.
III-4.B	Turbine Missiles	S	Section 5.5.2.
III-4.C	Internally Generated Missiles	S	Section 5.5.3.
III-4.D	Site-Proximity Missiles (Including Aircraft)	S	Section 5.5.4.

TABLE 1-3
(Sheet 3 of 9)

Topic	Title	Status(a)	IAP Disposition and Location Where Used in FSAR Update
III-5.A	Effects of Pipe Break on Structures, Systems and Components Inside Containment	2/C	(No modifications necessary. Reference CP Co letter of 5/31/85.) Section 5.6.
	- Operability of Leakage Detection System	2/C	(Technical Specifications change submitted by CP Co letter of 5/23/85.)
III-5.B	Pipe Break Outside Containment	S	Section 5.6.1.
III-6	Seismic Design Considerations		
	- Small Piping	3/C	Supplemental SER issued July 12, 1991
	- Electrical Components	3/C	Supplemental SER issued July 12, 1991
III-7.A	Inservice Inspection, Including Prestressed Concrete Containments With Either Grouted or Ungrouted Tendons		
	- Tendon Force Acceptance Criteria	3/C	(Acceptance criteria developed for each tendon that varies with time. Reference CP Co 11/7/83 submittal.) Sections 5.8.4.2 and 5.8.2.
	- Inspect Tendon-End Anchorages	3/C	(CP Co submitted Technical Specifications changes that specify when tendon-end anchorages will be inspected during ILRT. Reference 5/4/87 and 9/16/87 CP Co submittals.) Sections 5.8.4.2 and 5.8.2.
III-7.B	Design Codes, Design Criteria, Load Combinations and Reactor Cavity Design Criteria	3/C	(Issue resolved. Reference CP Co letters of 6/28/84 and 5/3/85.) Sections 5.8.3, 5.9.1 and 5.10.2; and Table 5.2-2.
III-7.C	Delamination of Prestressed Concrete Containment Structures	3/C	(One-time delamination inspection of containment dome completed.) Section 5.8.8.
III-7.D	Containment Structural Integrity Tests	S	Section 5.8.8.4.
III-8.A	Loose-Parts Monitoring and Core Barrel Vibration Monitoring	3/C	(Program of "Reactor Internals Vibration Monitoring" instituted.) Program was discontinued in 1985 with NRC issuance of Technical Specifications Amendment 91.
III-8.C	Irradiation Damage, Use of Sensitized Stainless Steel and Fatigue Resistance	S	-
III-10.A	Thermal-Overload Protection for Motors of Motor-Operated Valves	S	Chapter 8.

TABLE 1-3
(Sheet 4 of 9)

Topic	Title	Status(a)	IAP Disposition and Location Where Used in FSAR Update
III-10.B	Pump Flywheel Integrity	S	Section 4.3.5.
IV-1.A	Operation With Less Than All Loops in Service	S	Section 7.2.3.3.
IV-2	Reactivity Control Systems Including Functional Design and Protection Against Single Failures	S	Section 7.4.1.2.
V-5	Reactor Coolant Pressure Boundary (RCPB) Leakage Detection	2/C	(Technical Specifications for operability of leakage detection systems completed December 15, 1983.) Section 4.7.
V-6	Reactor Vessel Integrity	S	Sections 4.3.3, 4.4.2 and 4.5.2.
V-7	Reactor Coolant Pump Overspeed	S	Section 4.3.5.
V-10.A	Residual Heat Removal System Heat Exchanger Tube Failures	S	Section 6.1.
V-10.B	Residual Heat Removal System Reliability		
	- Overpressurization Protection (OPS)	2/C	(Technical Specifications Amendment submitted on July 29, 1982 which would place OPS in service before shutdown cooling.) Section 7.4.2.
	- Safety Grade Systems for Safe Shutdown	2/C	(New EOPs were issued in December 1986.)
V-11.A	Requirements for Isolation of High- and Low-Pressure Systems	2/C	(Technical Specifications Amendment submitted on July 29, 1982 which requires verification of LPSI check valve closure before criticality after use of system.) Sections 6.1.3 and 7.4.2.
V-11.B	Residual Heat Removal System Interlock Requirements	S	Chapters 6 and 7.
VI-1	Organic Materials and Post-Accident Chemistry	S	-
VI-2.D	Mass and Energy Release for Postulated Pipe Break Inside	1/C	(See PRA study submitted by CP Co letter of 5/23/85; NRC review dated 2/28/86.)
VI-3	Containment Pressure and Heat Removal Capability	1/C	(Same as VI-2.D. NRC review dated 2/28/86.)

TABLE 1-3
(Sheet 5 of 9)

Topic	Title	Status(a)	IAP Disposition and Location Where Used in FSAR Update
VI-4	Containment Isolation System		
	- Manual Isolation Valve	1/C	(Change isolation valve on Penetration 44 to power operation scheduled for 1985 outage.)
	- Threaded Pipe Connection	1/C	(Modification of Penetration 19 threaded connection completed during 1983/84 outage.)
	- Containment Isolation Systems	1/C	(Modifications of Penetrations 13, 17, 17a, 21, 21a, 28, 29, 48 and 73 have been completed.)
VI-6	Containment Leak Testing	3/C	(Technical Specifications Amendment submitted March 3, 1982 for verification of airlock door seal integrity following opening. Approved by Amendment 126 dated June 1, 1989.)
VI-7.A.3	Emergency Core Cooling System Actuation System	S	Sections 7.3.2, 7.3.5 and 8.1.1; and Appendix 7A.
VI-7.B	Engineered Safety Feature Switchover From Injection to Recirculation Mode (Automatic Emergency Core Cooling System Realignment)	S	Section 6.1.2.2.
VI-7.C	Emergency Core Cooling System (ECCS) Single-Failure Criterion and Requirements for Locking Out Power to Valves, Including Independence of Interlocks on ECCS Valves	S	Section 7.3.
VI-7.C.1	Appendix K - Electrical Instrumentation and Control Re-reviews	S	Chapter 7, Sections 8.1.1 and 8.3.5.
VI-7.C.2	Failure Mode Analysis (Emergency Core Cooling System)	S	Section 7.2.3.10.
VI-7.D	Long-Term Cooling Passive Failures (eg, Flooding of Redundant Components)	S	Sections 5.4 and 9.1.3.
VI-7.F	Accumulator Isolation Valves Power and Control System Design	S	Chapters 7 and 8.
VI-10.A	Testing of Reactor Trip System and Engineered Safety Features, Including Response-Time Testing	3/C	(Requires response-time testing on a periodic basis.) Sections 7.3.2 and 7.3.5.
VII-1.A	Isolation of Reactor Protection System From Nonsafety Systems, Including Qualification of Isolation Devices	1/C	Section 7.2.9.
VII-1.B	Trip Uncertainty and Setpoint Analysis Review of Operating Data Base	S	-

TABLE 1-3
(Sheet 6 of 9)

Topic	Title	Status(a)	IAP Disposition and Location Where Used in FSAR Update
VII-2	Engineered Safety Features System Control Logic and Design	S	Chapter 6 and Section 7.3.1.
VII-3	Systems Required for Safe Shutdown	-	Sections 7.2.3, 7.2.7, 7.2.9, 7.3.1, 7.4.1, 7.4.2, 7.6.2, 8.1.1, 8.1.2, 8.1.3, 8.4.1 and 8.4.2; and Table 7-3.
	- Removal of Nonessential Loads as an Alternative to 10 CFR 50, Appendix A, GDC 17	2/C	(Procedure to remove nonessential loads from battery if the immediate sources of offsite power are available. Completed, see CP Co letter of 9/26/84.)
	- Component Cooling Water Surge Tank Level	1/C	(Installation of another level sensor to the component cooling water surge tank and control room indication completed.)
	- Boric Acid Heat Tracing	4/C	(The PRA study found that the boric acid system is a low contributor to risk and the impact of this issue on reactivity control is low, no backfit required.)
	- Pressure Sensor on CCW Pumps	4/C	(Because of redundancy in system failure indication, the PRA study concluded this issue has a small impact on availability, no backfit required.)
	- Adequate Seismic Category I Water Supply for AFW	2/C	(Same as III-2.)
VII-6	Frequency Decay	S	Section 14.7 and Chapter 8.
VIII-1.A	Potential Equipment Failures Associated With Degraded Grid Voltage	S	Sections 7.3.2.2 and 7.3.5.
VIII-2	Onsite Emergency Power Systems (Diesel Generator)	1/C	(Separate annunciator modification complete.) Section 8.4.
VIII-3.A	Station Battery Capacity Test Requirements	2/C	(CP Co proposed an amendment to the Technical Specifications to conduct battery test pursuant to NRC guidelines, see CP Co letter dated 5/13/85.) Section 8.4.2.
VIII-3.B	DC Power System Bus Voltage Monitoring and Annunciation	1/C	(Installation of (1) 125 V dc tie breaker open (both buses), (2) public address system inverter loss of voltage and (3) battery undervoltage (both batteries) is complete.) Sections 8.3.5 and 8.4.2.
VIII-4	Electrical Penetrations of Reactor Containment	4/C	(Evaluation acceptable based on similar SEP plants, modification unnecessary.) Section 8.5.1.
IX-1	Fuel Storage	S	Sections 9.4 and 9.11.

TABLE 1-3
(Sheet 7 of 9)

Topic	Title	Status(a)	IAP Disposition and Location Where Used in FSAR Update
IX-3	Station Service and Cooling Water Systems		
	- 2.7.1 Cooling of CCW Heat Exchanger	2/C	(Revision to EOP for FWS alignment acceptable and complete.)
	- 2.7.2 Flooding of Intake Safety Systems	2/C	Section 5.4.2.
IX-4	Boron Addition Systems (PWR)	S	Section 9.10.
IX-5	Ventilation Systems		
	- AFW Pump Room	2/C	(AFW pumps qualified for 160°F service.) Section 9.8.
	- Cable Spreading, Switchgear and Battery Rooms	2/C	(Ventilation fans for additional cooling to inverter cabinets, charger cabinets and auxiliary feedwater junction boxes completed during 1983/84 outage.) Sections 8.7.2, 8.7.3 and 9.8.
IX-6	Fire Protection	3/C	(11/7/83 NRC letter. FP will be reviewed generically outside the context of SEP.) Sections 7.4.1, 7.4.2, 7.6.2, 7.7.3, 7.7.4, 7.7.8, 8.3.5, 8.5.3, 8.7.2, 8.8; and Table 7-3; and Appendix 7B.
XIII-2	Safeguards/Industrial Security	S	-
XV-1	Decrease in Feedwater Temperature, Increase in Feedwater Flow, Increase in Steam Flow and Inadvertent Opening of a Steam Generator Relief or Safety Valve	S	Section 14.9.
XV-2	Spectrum of Steam System Piping Failures Inside and Outside Containment (PWR)	3/C	(See PRA study submitted by CP Co letter of 5/23/85, NRC review required. NRC review letter dated 2/28/86.) Section 14.14.
	- Failure of Main Feedwater Isolation	3/C	(No modifications necessary, pending NRC review. NRC review letter dated 2/28/86.)
XV-3	Loss of External Load, Turbine Trip, Loss of Condenser Vacuum, Closure of Main Steam Isolation Valve (BWR) and Steam Pressure Regulator Failure (Closed)	S	Section 14.12.
XV-4	Loss of Nonemergency AC Power to the Station Auxiliaries	S	Chapter 8.
XV-5	Loss of Normal Feedwater Flow	S	Section 14.13.

Topic	Title	Status(a)	IAP Disposition and Location Where Used in FSAR Update
XV-6	Feedwater System Pipe Breaks Inside and Outside Containment (PWR)	S	Section 5.6.
XV-7	Reactor Coolant Pump Rotor Seizure and Reactor Coolant Pump Shaft Break	S	Section 14.17.
XV-8	Control Rod Misoperation (System Malfunction or Operator Error)	S	Sections 14.2, 14.4, 14.6 and 14.16.
XV-9	Start-Up of an Inactive Loop or Recirculation Loop at an Incorrect Temperature, and Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate	S	Section 14.8.
XV-10	Chemical and Volume Control System Malfunction That Results in a Decrease in Boron Concentration in the Reactor Coolant (PWR)	S	Section 14.3.
XV-12	Spectrum of Rod Ejection Accidents (PWR)	2/C	(The staff found that radiological consequences of a rod ejection accident are acceptable and do not require any departure from existing analysis.)
XV-14	Inadvertent Operation of Emergency Core Cooling System and Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory	S	-
XV-15	Inadvertent Opening of a PWR Pressurizer Safety/Relief Valve or a BWR Safety/Relief Valve	S	-
XV-16	Radiological Consequences of Failure of Small Lines Carrying Primary Coolant Outside Containment	S	Sections 14.20 and 14.21.
XV-17	Radiological Consequences of Steam Generator Tube Failure (PWR)	S	Section 14.15.
XV-19	Loss-of-Coolant Accidents Resulting From Spectrum of Postulated Piping Breaks Within the Reactor Coolant Pressure Boundary	S	Section 14.7.
XV-20	Radiological Consequences of Fuel-Damaging Accidents (Inside and Outside Containment)	S	Sections 14.11 and 14.19.

TABLE 1-3
(Sheet 9 of 9)

Topic	Title	Status(a)	IAP Disposition and Location Where Used in FSAR Update
XVII	Operational Quality Assurance Program	S	-

(a)Staff Position Code

S - Reviewed and accepted during SEP
1 - Evaluated during integrated assessment, modification required
2 - Evaluated during integrated assessment, procedure development or Technical Specifications Change required
3 - Evaluated during integrated assessment, refined engineering analysis or continuation of ongoing analysis required
4 - Evaluated during integrated assessment; no further work required

Status Code

(N) - No action required
(C) - Action required complete and item resolved
(b) - Action required in progress

TABLE 1-4
(Sheet 1 of 8)

POST-TMI REQUIREMENTS FOR CONSUMERS POWER COMPANY'S
PALISADES PLANT - NUREG-0737

Item Number	Title	Description	Status(a)	Disposition
I.A.1.1	Shift Technical Advisor	1. On duty	1	Additional training produced.
		2. Technical Specifications	1	Additional training produced.
		3. Trained per LL Cat B	1	Additional training produced.
		4. Describe long-term program	1	Additional training produced.
I.A.1.2	Shift Supervisor Responsibilities	Delegate nonsafety duties	1	Defined Shift Supervisor's responsibilities.
I.A.1.3	Shift Manning	1. Limit overtime	1	Reviewed and revised procedures for shift manning.
		2. Min shift crew	1	To be addressed when final rule is issued.
I.A.2.1	Immediate Upgrading of RO and SRO Training and Qualifications	1. SRO experience	1	Training and qualification procedures at Palisades Plant have been modified to meet NRC criteria.
		2. SROs be ROs in one year	1	Training and qualification procedures at Palisades Plant have been modified to meet NRC criteria.
		3. Three-month training on shift	1	Training and qualification procedures at Palisades Plant have been modified to meet NRC criteria.
		4. Modify training	1	Training and qualification procedures at Palisades Plant have been modified to meet NRC criteria.
		5. Facility certification	1	Training and qualification procedures at Palisades Plant have been modified to meet NRC criteria.
I.A.2.3	Administration of Training Programs	Instructors complete SRO exam	1	Assure trainers are qualified at SRO qualifications level.

(a)Status Codes:

1 - Completed
2 - In Process

TABLE 1-4
(Sheet 2 of 8)

Item Number	Title	Description	Status(a)	Disposition
I.A.3.1	Revise Scope and Criteria for Licensing Exams	1. Increase scope	1	Include simulator examinations with training.
		2. Increase passing grade	1	Include simulator examinations with training.
		3. Simulator exams	1	Include simulator examinations with training.
I.C.1	Short-Term Accident and Procedures Review	1. SB LOCA	1	Complete per EOP upgrade.
		2. Inadequate Core Cooling a. Reanalyze and propose guidelines b. Revise procedures	1	Complete per EOP upgrade.
		3. Transients and accidents a. Reanalyze and propose guidelines b. Revise procedures	1	Complete.
I.C.2	Shift and Relief Turnover Procedures	Implement shift turnover checklist	1	Revised procedure for shift turnover.
I.C.3	Shift Supervisor Responsibility	Clearly define supervisor and operator responsibilities	1	Defined Shift Supervisor's responsibilities.
I.C.4	Control Room Access	Establish authority; limit access	1	Access controlled by Shift Supervisor or No 1 Control Operator.
I.C.5	Feedback of Operating Experience	Licensee to implement procedures	1	Assess operating experience and incorporate (if needed) into operating procedures, training and retraining programs.
I.C.6	Verify Correct Performance of Operating Activities	Revise performance procedures	1	Evaluate procedures and revise when appropriate.

(a)Status Codes:

1 - Completed
2 - In Process

TABLE 1-4
(Sheet 3 of 8)

Item Number	Title	Description	Status(a)	Disposition
I.D.1	Control Room Design Reviews	1. Assessment and schedule for correcting deficiencies	1	Complete - SER issued 9/14/89. See Note 1.
		2. Implementation of corrective actions	1	Continuing.
I.D.2	Plant Safety Parameter Display Console	1. Description	1	Complete - SER issued 6/29/87.
		2. Installed	1	Complete - SER issued 6/29/87. - CPCo Certification letter 7/10/89 - NRC Closure letter 4/19/90
		3. Fully implemented	1	Complete - SER issued 6/29/87. - CPCo Certification letter 7/10/89 - NRC Closure letter 4/19/90
II.B.1	Reactor Coolant System Vents	1. Design vents	1	Modification of gas venting system and revised procedures accordingly.
		2. Install vents (LL Cat B)	1	Modification of gas venting system and revised procedures accordingly.
		3. Procedures	1	Modification of gas venting system and revised procedures accordingly.
II.B.2	Plant Shielding	1. Review designs	1	Modification completed of facility to provide access to vital areas under accident conditions.
		2. Plant modifications (LL Cat B)	1	Modification completed of facility to provide access to vital areas under accident conditions.
		3. Equipment qualification	1	Modification completed of facility to provide access to vital areas under accident conditions.
II.B.3	Post-Accident Sampling	1. Interim system	1	Complete.
		2. Plant modifications (LL Cat B)	1	Complete.

(a)Status Codes:

1 - Completed
2 - In Process

TABLE 1-4
(Sheet 4 of 8)

Item Number	Title	Description	Status(a)	Disposition
II.B.4	Training for Mitigating Core Damage	1. Develop training program	1	Complete training program.
		2. Implement program a. Initial b. Complete	1	Complete training program.
II.D.1	Relief and Safety Valve Test Requirements	1. Submit program	1	Complete.
		2. RV and SV testing (LL Cat B) a. Complete testing b. Plant specific report	1	Complete.
		3. Block valve testing	1	Block valve requirement analysis complete, new block valves and PORVs installed during 1989 maintenance outage.
II.D.3	Valve Position Indication	1. Install direct indications of valve position	1	Complete.
		2. Technical Specifications	1	Modification of valve indicators and Technical Specifications changes as required (reference Amendment 67, 10/8/81).
II.E.1.1	Auxiliary Feedwater System Evaluation	1. Short term	1	Auxiliary Feedwater Systems modified as required.
		2. Long term	1	Auxiliary Feedwater Systems modified as required.
II.E.1.2	Auxiliary Feedwater System Initiation and Flow	1. Initiation a. Control grade b. Safety grade	1	Modification of instrumentation to level of safety grade complete.
		2. Flow indication a. Control grade b. LL A Technical Specifications c. Safety grade	1	Modification of instrumentation to level of safety grade complete.

(a)Status Codes:

1 - Completed
2 - In Process

TABLE 1-4
(Sheet 5 of 8)

Item Number	Title	Description	Status(a)	Disposition
II.E.3.1	Emergency Power for Pressurizer Heaters	1. Upgrade power supply	1	Modified system to provide emergency power.
		2. Technical Specifications	1	Modified system to provide emergency power.
II.E.4.1	Dedicated Hydrogen Penetrations	1. Design	1	Not required for Palisades Plant.
		2. Install	1	Not required for Palisades Plant.
II.E.4.2	Containment Isolation Dependability	1-4. Imp diverse isolation	1	Modified containment isolation circuitry.
		5. Containment pressure set point a. Specify pressure b. Modifications	1	Lower containment pressure set point to level compatible with normal operation.
		6. Containment purge valves	1	Consumers Power Company has committed to use valves only during outages.
		7. Radiation signal on purge valves	1	Isolate purge and vent valves on radiation signal.
		8. Technical Specifications	1	Isolate purge and vent valves on radiation signal.
II.F.1	Accident Monitoring	1. Noble gas monitor	1	Modifications made and Technical Specifications changes requested.
		2. Iodine/particulate sampling	1	Modifications made and Technical Specifications changes requested.
		3. Containment high-range monitor	1	Modifications made and Technical Specifications changes requested.
		4. Containment pressure	1	Modifications made and Technical Specifications changes requested.
		5. Containment water level	1	Modifications made and Technical Specifications changes requested.
		6. Containment hydrogen	1	Existing equipment meets the NRC's criteria.

(a)Status Codes:

1 - Completed
2 - In Process

Item Number	Title	Description	Status(a)	Disposition
II.F.2	Instrumentation for Detection of Inadequate Core Cooling	1. Subcool meter	1	Instrumentation modified during 1988 refueling outage.
		2. Technical Specifications (LL Cat A)	1	Issued as Amendment 129.
		3. Install level instruments (LL Cat B)	1	Instrumentation modified during 1988 refueling outage.
II.G.1	Power Supplies for Pressurizer Relief Valves, Block Valves and Level Indicators	1. Upgrade to emergency sources	1	No action required.
		2. Technical Specifications	1	No action required.
II.K.1	IE Bulletins	79-05	1	See Bulletin for specific disposition, no further action required.
		79-06	1	See Bulletin for specific disposition, no further action required.
		79-08	1	See Bulletin for specific disposition, no further action required.
II.K.2	Orders on B&W Plants	13. Thermal mechanical report	1	Analysis shows that vessel integrity will be maintained for such an event.
		17. Voiding in RCS	1	Existing equipment meets NRC requirements.
		19. Bench mark analysis of seq AFW flow	1	Not applicable to Palisades Plant.
II.K.3	Final Recommendations, B&O Task Force	1. Auto PORV isolation a. Design b. Test/install	1	Not applicable to Palisades. Palisades operates with the PORVs isolated.

(a)Status Codes:

1 - Completed
2 - In Process

TABLE 1-4
(Sheet 6 of 8)

Item Number	Title	Description	Status(a)	Disposition
II.F.2	Instrumentation for Detection of Inadequate Core Cooling	1. Subcool meter	1	Instrumentation to be modified. 1988 refueling outage.
		2. Technical Specifications (LL Cat A)	1	Instrumentation to be modified. 1988 refueling outage.
		3. Install level instruments (LL Cat B)	1	Instrumentation to be modified. 1988 refueling outage.
II.G.1	Power Supplies for Pressurizer Relief Valves, Block Valves and Level Indicators	1. Upgrade to emergency sources	1	No action required.
		2. Technical Specifications	1	No action required.
II.K.1	IE Bulletins	79-05	1	See Bulletin for specific disposition, no further action required.
		79-06	1	See Bulletin for specific disposition, no further action required.
		79-08	1	See Bulletin for specific disposition, no further action required.
II.K.2	Orders on B&W Plants	13. Thermal mechanical report	1	Analysis shows that vessel integrity will be maintained for such an event.
		17. Voiding in RCS	1	Existing equipment meets NRC requirements.
		19. Bench mark analysis of seq AFW flow	1	Not applicable to Palisades Plant.
II.K.3	Final Recommendations, B&O Task Force	1. Auto PORV isolation a. Design b. Test/install	1	Not applicable to Palisades. Palisades operates with the PORVs isolated.

(a)Status Codes:

1 - Completed
2 - In Process

Item Number	Title	Description	Status(a)	Disposition
		2. Report on PORV failures	1	Complete.
		3. Reporting SV and RV failures and challenges	1	RVs are isolated - are not challenged. SVs have never been challenged. Complete per CE Reports CEN-213 and -227.
		5. Auto trip of RCPs a. Propose modifications b. Modify	1	Complete Trip 2/leave 2 incorporated in EOPs.
		17. Emergency Core Cooling System outages	1	No action required.
		25. Power on pump seals a. Propose modifications b. Modifications	1	Item not applicable to Palisades Plant.
		30. SB LOCA methods a. Schedule outline b. Model c. New analyses	1	CE-CEFLASH 4AS acceptable.
		31. Compliance with CFR 50.46	1	Not required.
		44. Evaluate transient with single failure	1	Item not applicable to Palisades Plant.
III.A.1.1	Emergency Preparedness, Short Term	Short-term improvements	1	Emergency Plan will be changed on an as-needed basis.

(a)Status Codes:

1 - Completed
2 - In Process

TABLE 1-4
(Sheet 8 of 8)

Item Number	Title	Description	Status(a)	Disposition
III.A.1.2	Upgrade Emergency Support Facilities	1. Interim TSC, OCS and EOF	1	Emergency support facilities have been constructed/modified to meet NRC criteria.
		2. Design	1	Emergency support facilities have been constructed/modified to meet NRC criteria.
		3. Modifications	1	Emergency support facilities have been constructed/modified to meet NRC criteria.
III.A.2	Emergency Preparedness	1. Upgrade emergency plans to Appendix E, 10 CFR 50	1	Complete. Emergency Plan has been upgraded to meet Appendix E.
		2. Meteorological data	1	Complete. Emergency Plan has been upgraded to meet Appendix E.
III.D.1.1	Primary Coolant Outside Containment	1. Leak reduction	1	Leakage reduction program has been implemented.
		2. Technical Specifications	1	Leakage reduction program has been implemented.
III.D.3.3	In-Plant Radiation Monitoring	1. Provide means to determine presence of radioiodine	1	Existing Plant equipment meets NRC requirements.
		2. Modifications to accurately measure	1	Existing Plant equipment meets NRC requirements.
III.D.3.4	Control Room Habitability	1. Review	1	Control room HVAC was modified to satisfy NRC criteria.
		2. Modification	1	Control room HVAC was modified to satisfy NRC criteria.

NOTE 1: See Supplement 1 to NUREG-0737 for details.

(a)Status Codes:

1 - Completed
2 - In Process

TABLE 1-5

UNRESOLVED SAFETY ISSUES

Task No	Title	Reference FSAR Chapter(s)
A-1	Water Hammer	4 and 9
A-2	Asymmetric Blowdown Loads on the Primary Coolant System	6
A-4	Pressurized Water Reactor Steam Generator Tube Integrity	4
A-9	Anticipated Transients Without Scram (ATWS)	7
A-11	Reactor Vessel Materials Toughness	4
A-12	Fracture Toughness of Steam Generators and Primary Coolant Pump Supports	-(a)
A-17	Systems Interactions in Nuclear Power Plants	-(a)
A-24	Environmental Qualification of Safety-Related Electrical Equipment (EEQ)	8
A-26	Reactor Vessel Pressure Transient Protection	4
A-31	Residual Heat Removal Requirements	7 and 9
A-36	Control of Heavy Loads Near Spent Fuel	9
A-40	Seismic Design Criteria Short-Term Program	5
A-43	Containment Emergency Sump Reliability	6
A-44	Station Blackout	8
A-45	Shutdown Decay Heat Removal Requirements	6 and 9
A-46	Seismic Qualification of Equipment in Operating Plants	5
A-47	Safety Implications of Control System	8
A-48	Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment	6
A-49	Pressurized Thermal Shock	-(a)

(a)Not specifically addressed in FSAR.

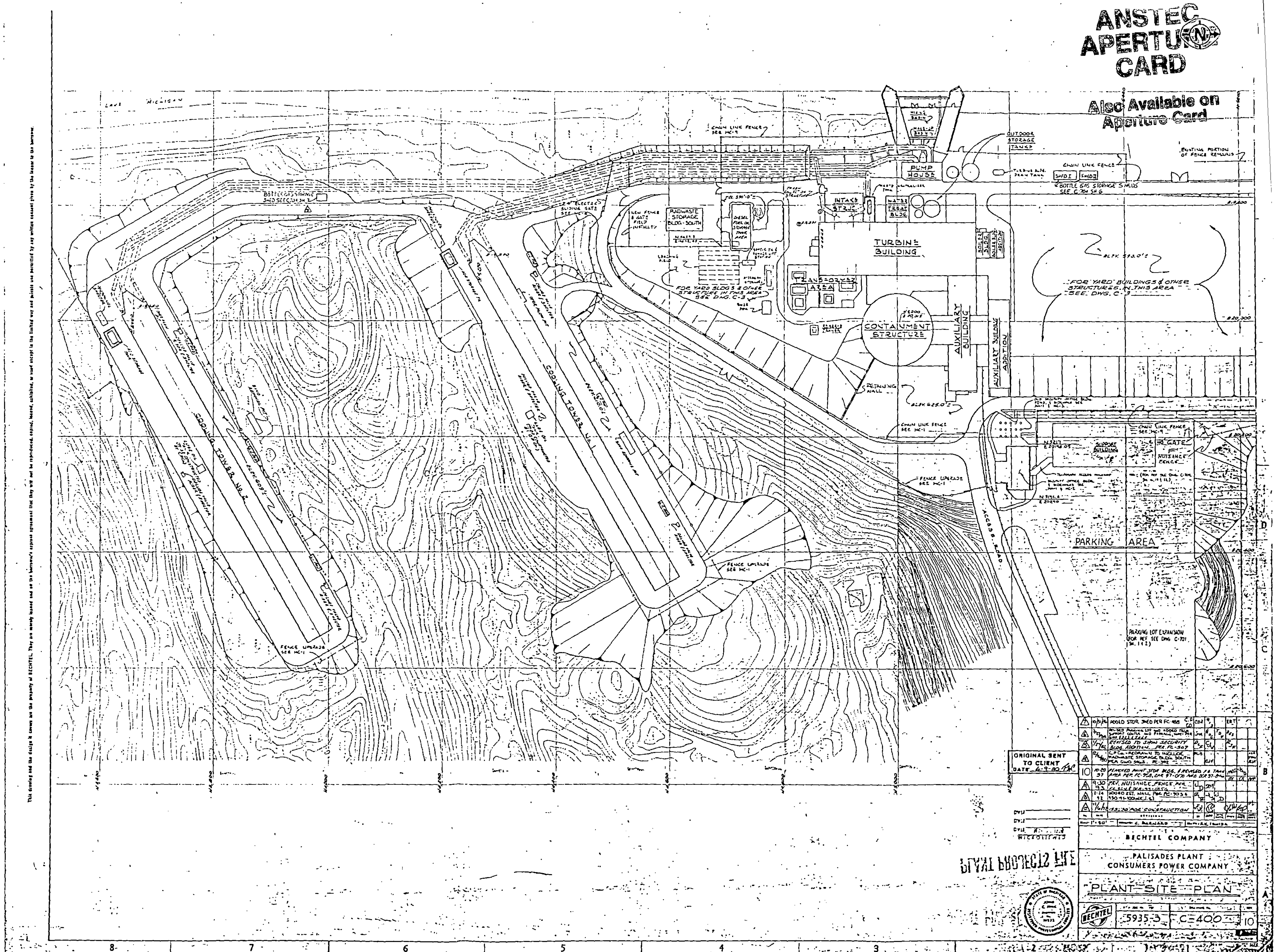

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FIG 01-01 SH 1
FSAR REV 20

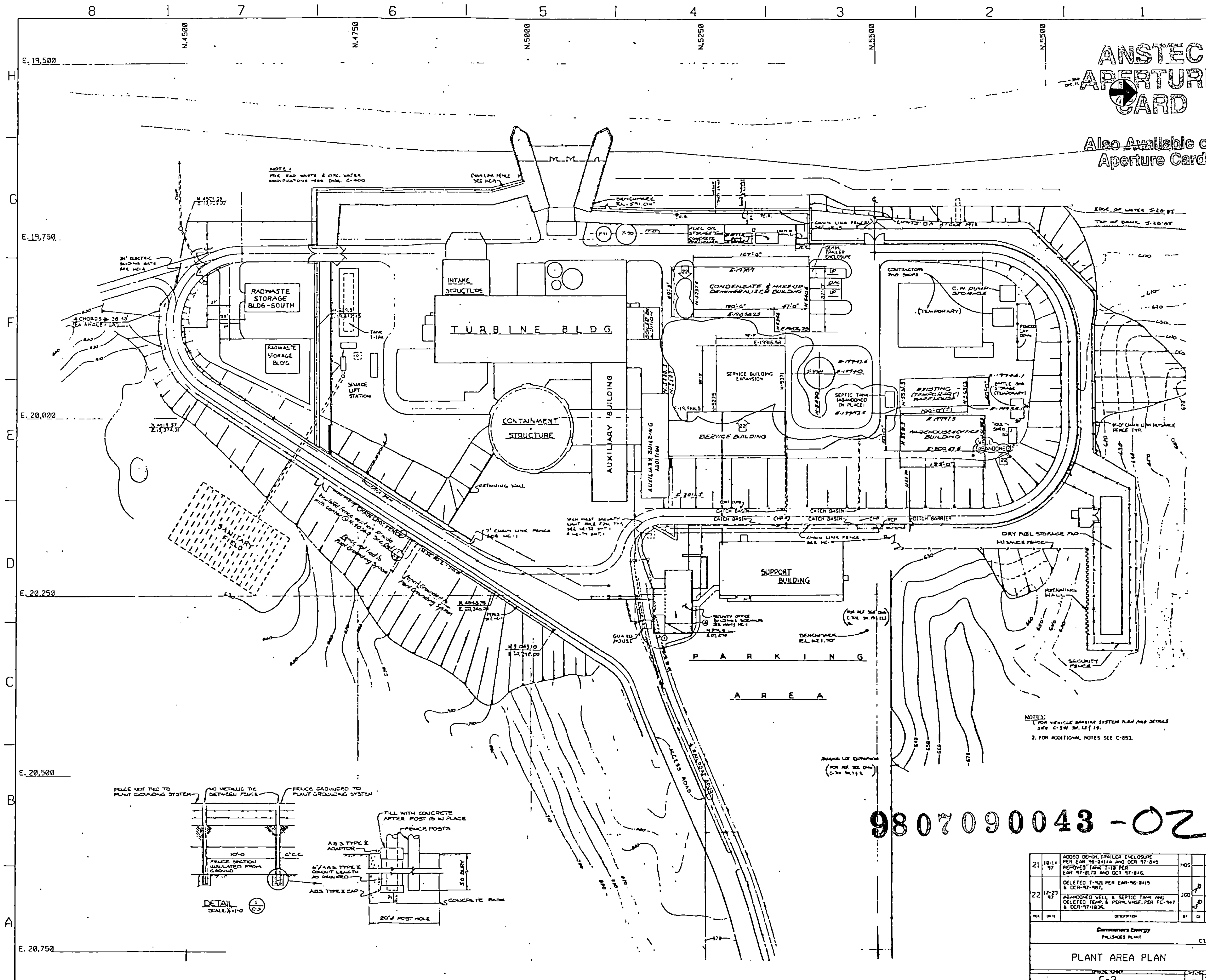
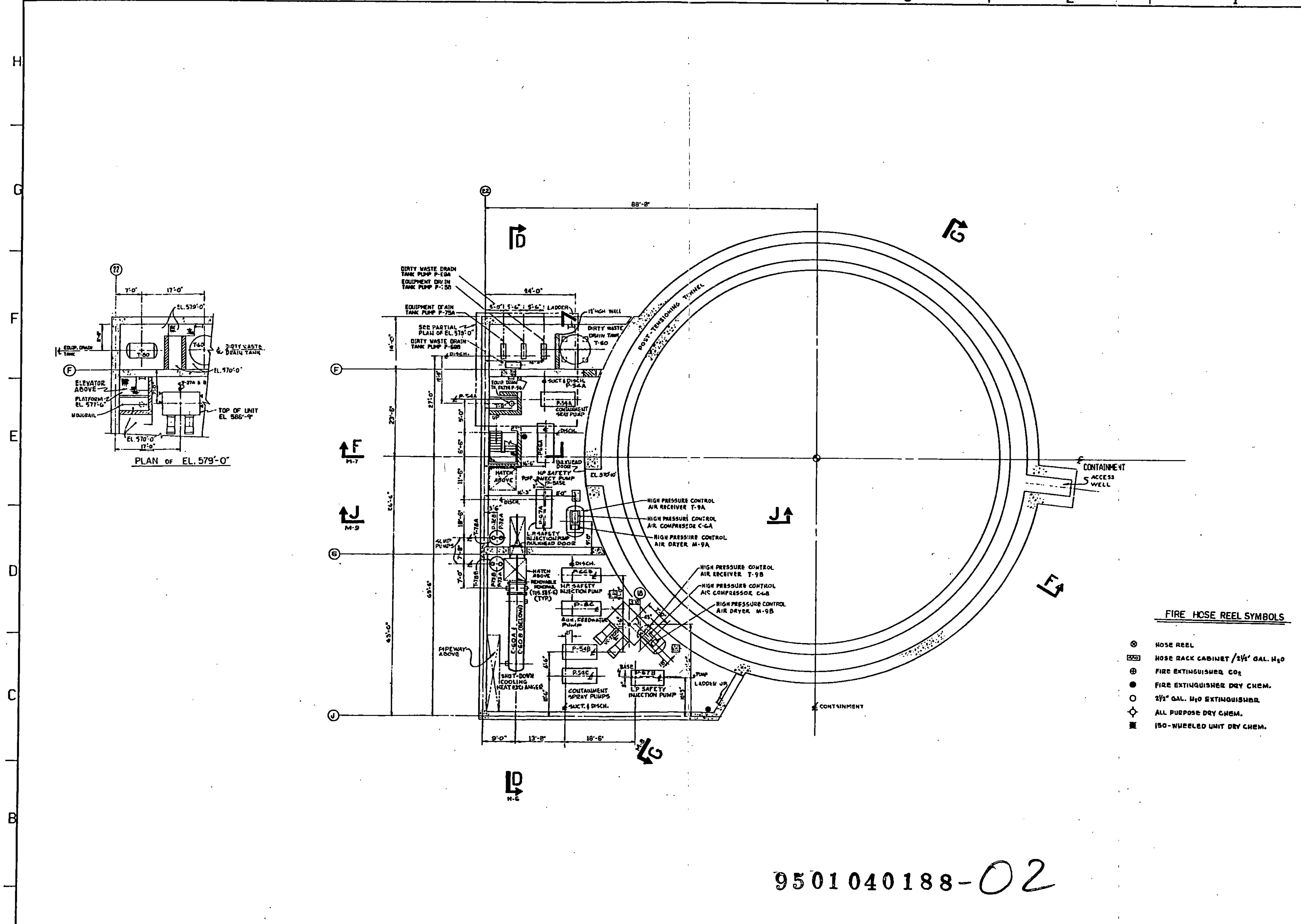


FIG 01-01 Sh 2 -
FSAR REV 20



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11	8-4-93	REDRAWN ON CAD PER DCR 93-1018	JFH	
REV.	DATE	DESCRIPTION	BY	APP

CONSUMERS POWER COMPANY
PALISADES NUCLEAR PLANT
COVERT, MICHIGAN

EQUIPMENT LOCATION
REACTOR BUILDING
PLAN OF EL. 570'-0"

M-1

Figure 1-2 FSAR Rev 17

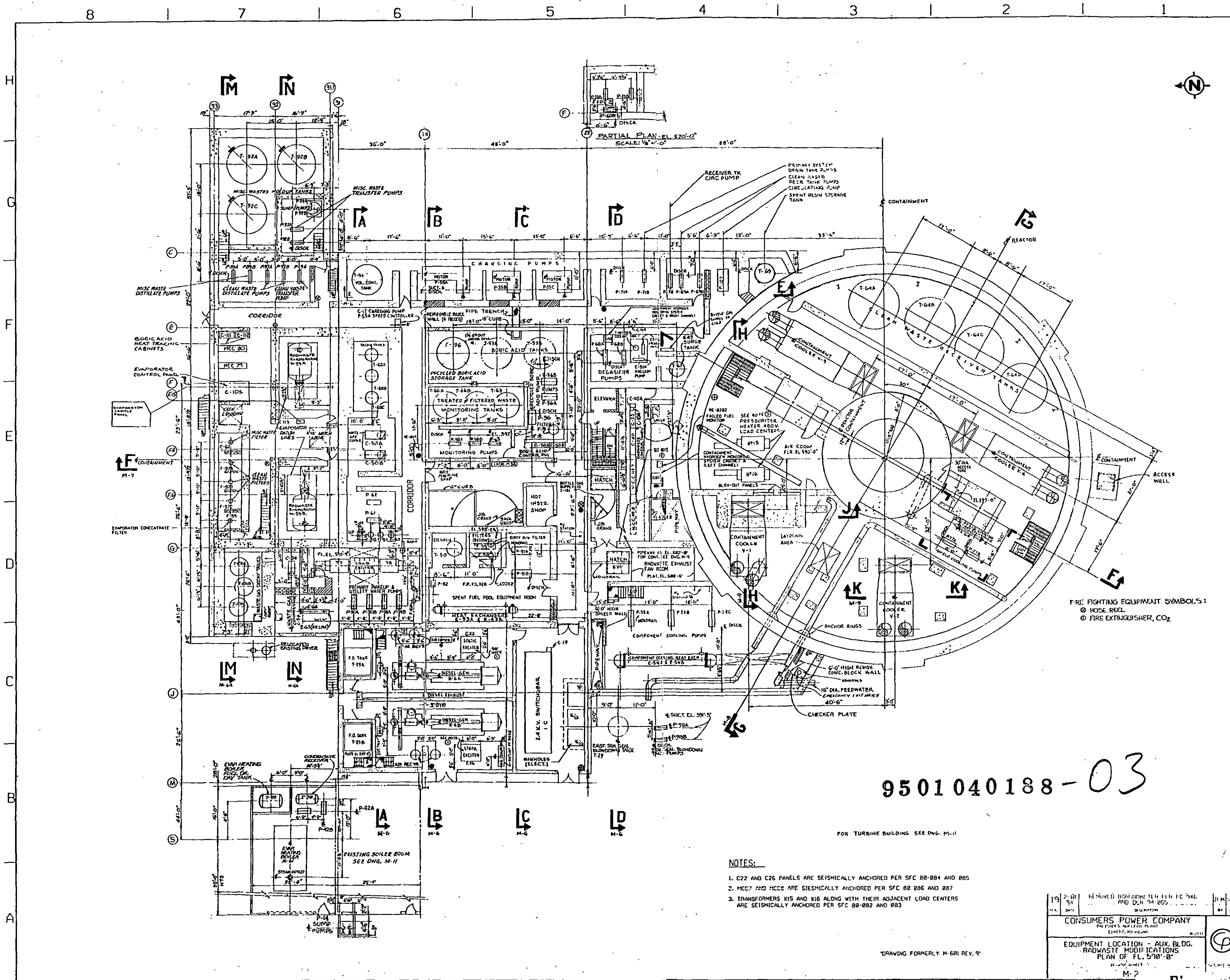

Figure 1-3 FSAR Rev 17

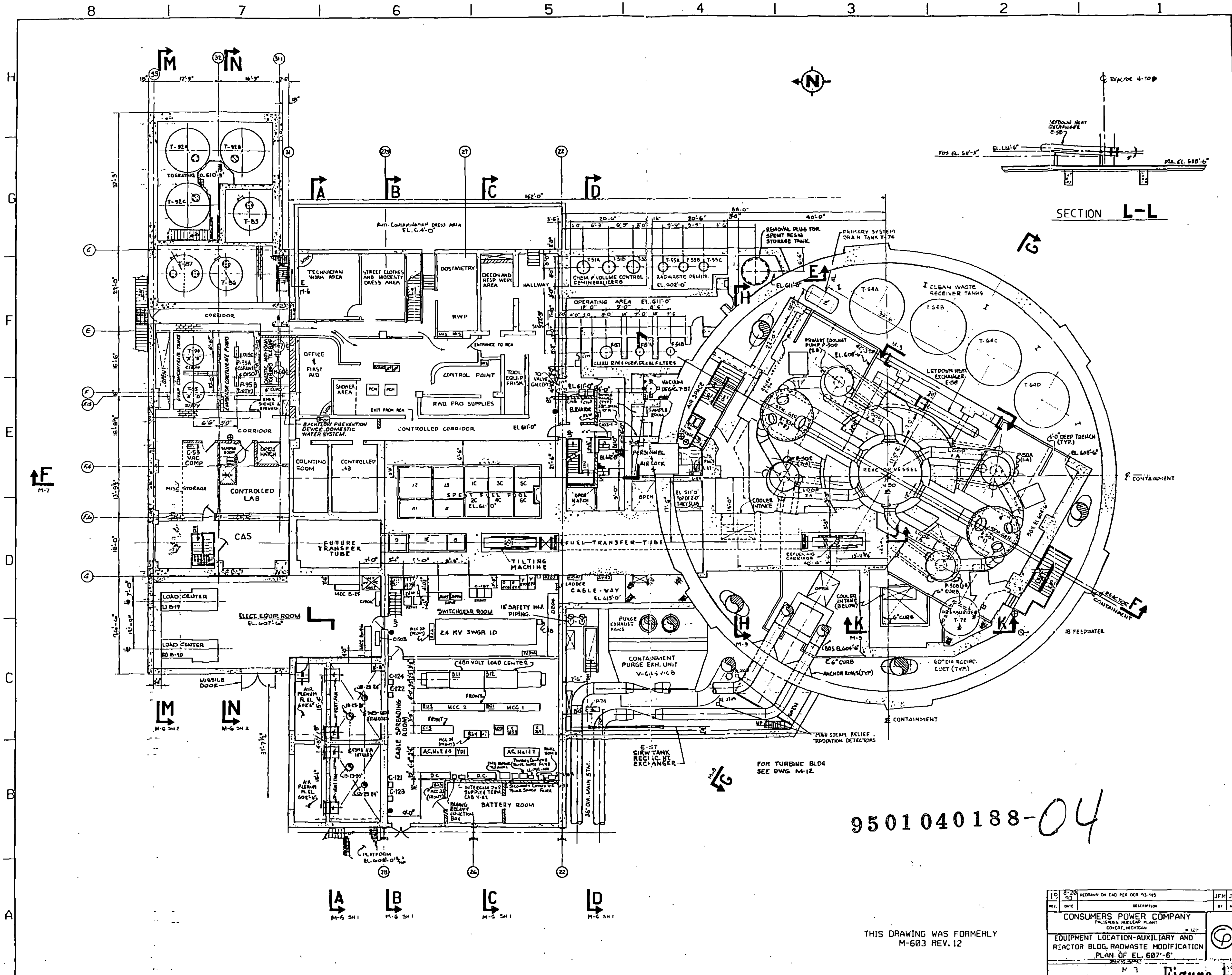

Figure 1-4 FSAR Rev 17

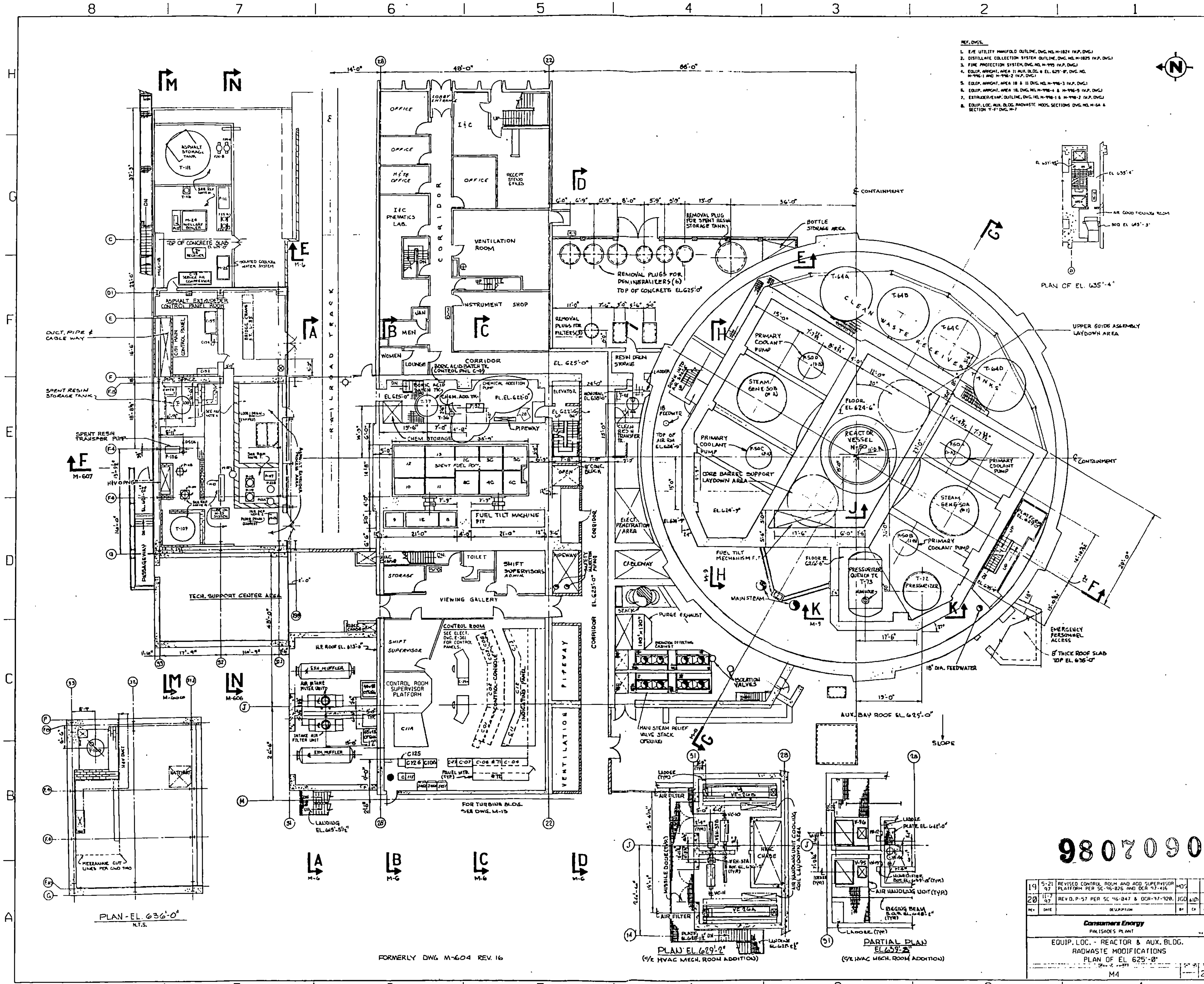

FIG 1-05

FSAR REV 20

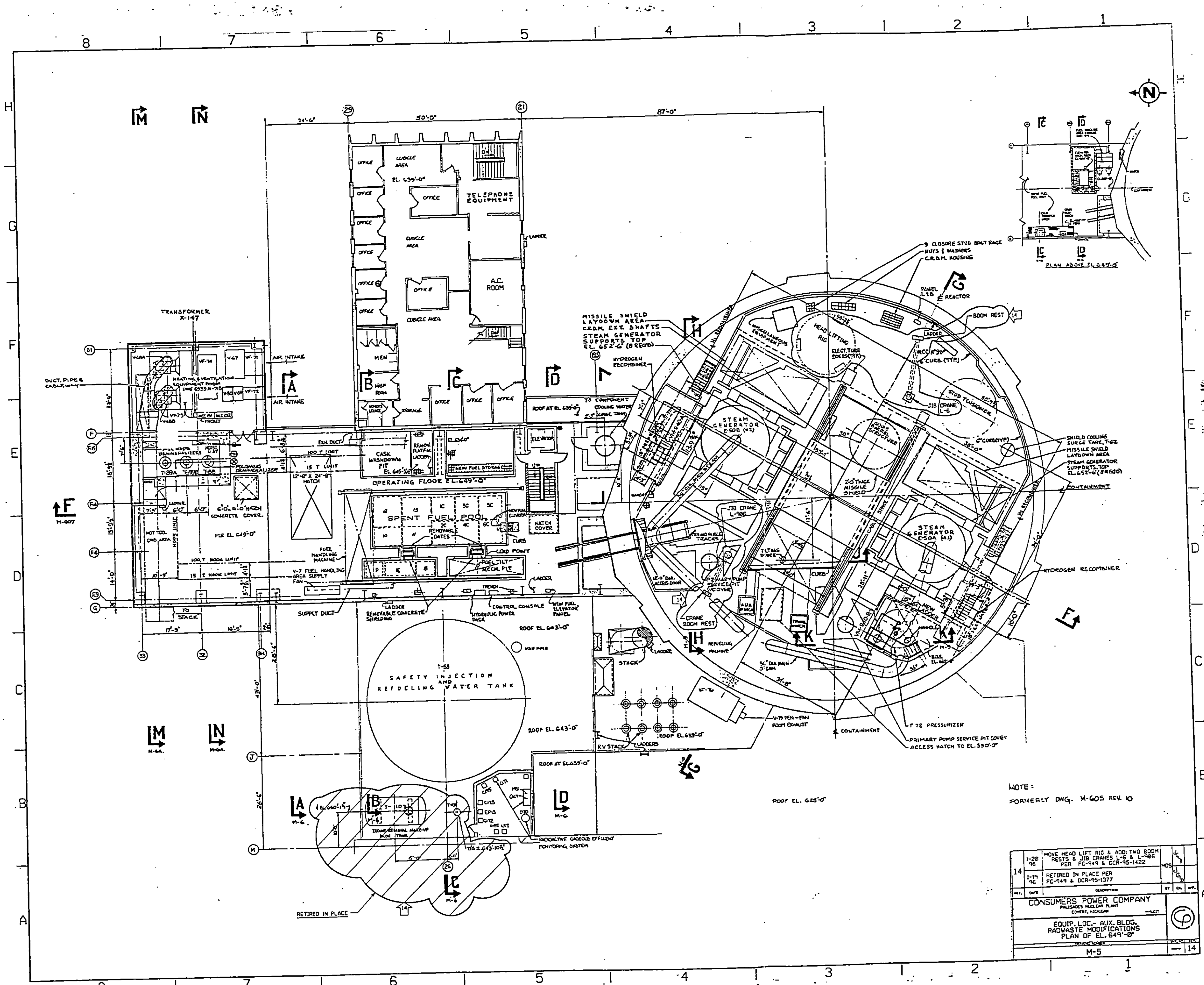

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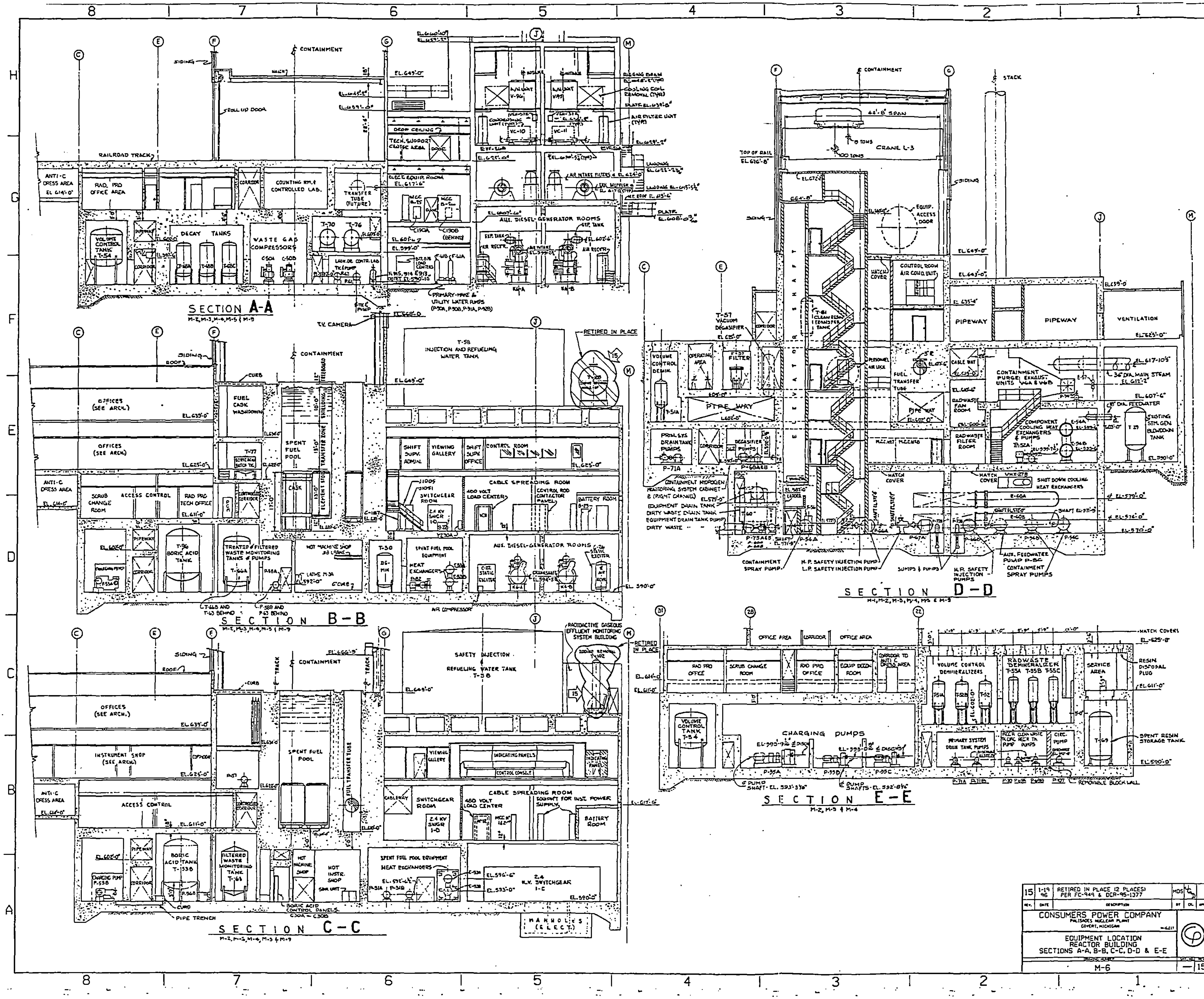
Figure 1-6
Rev 19



ANSTEC APERTURE CARD

Also Available on Aperture Card

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Figure 1-7
Rev 19

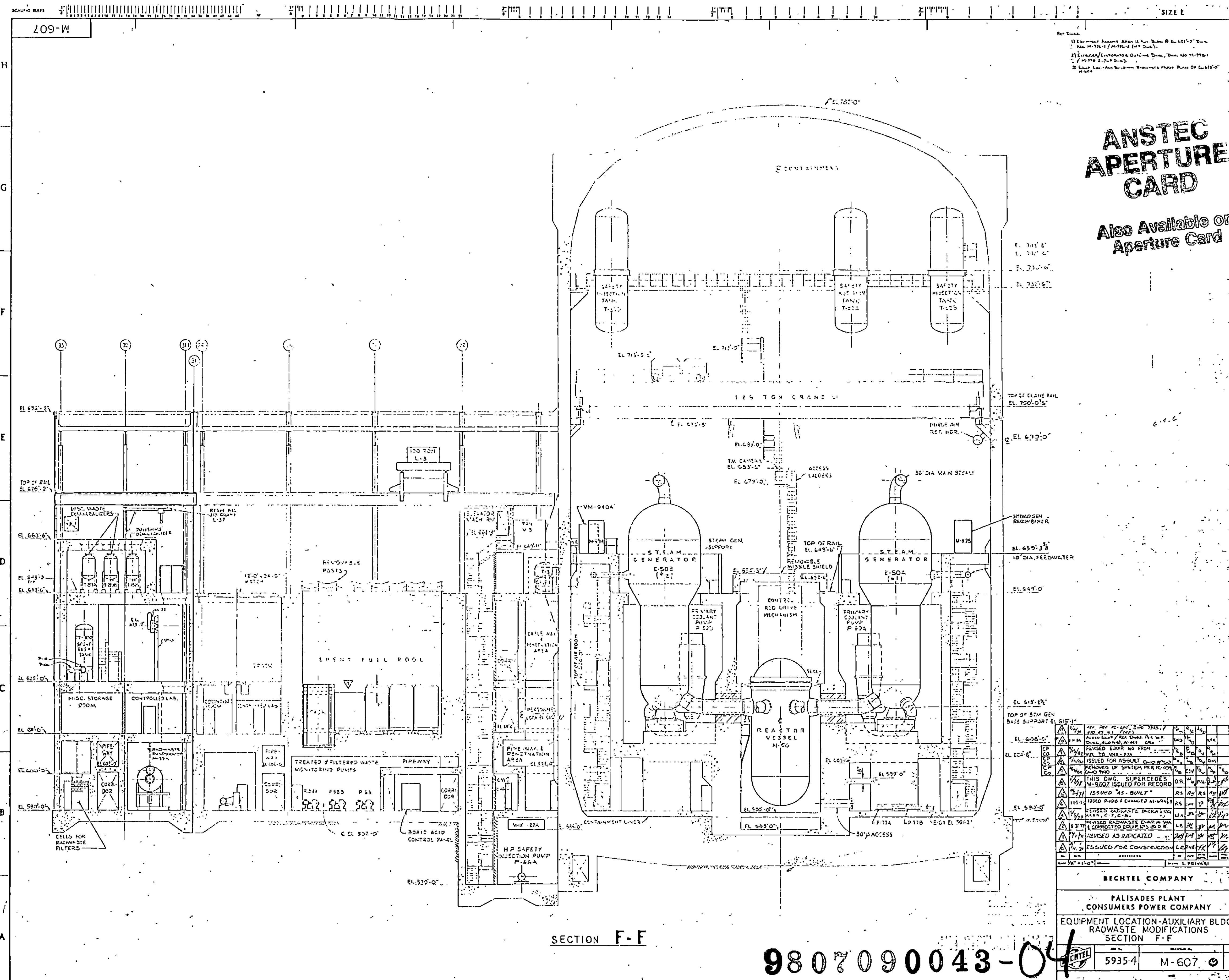

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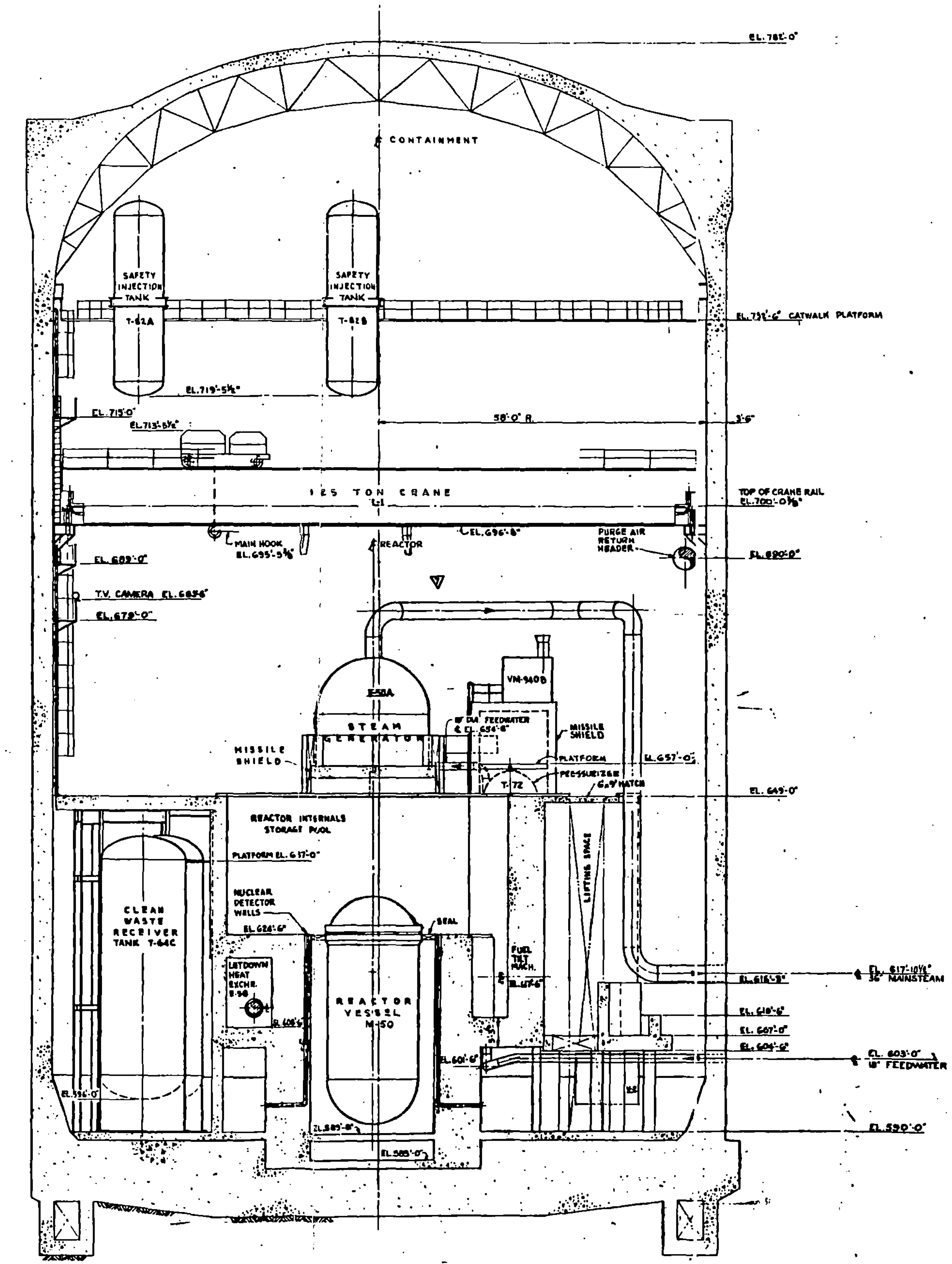
FIG 1-8

FSAR REV 20



SECTION G-G

M-1, M-2, M-3, M-4, M-5 & M-10

7	7/29/91	REV'D MAIN STM ELEV PER FC 911 & 950 91603 (PDB)			DMS	–	–	–
6	7/11/78	THIS DWG. SUPERCEDES M-9008 ISSUED FOR RECORD	OB		P.H.			
5		ISSUED "AS-BUILT"	RS		RS			
4	10/8/68	CHANGE PLATFORM ELEVATION TO 657'-0" (WAS 655'-0") DELETED LETDOWN HEAT EXCHR. E-58 ELEVATION.						
3	3/1/68	REVISED STM. GEN. No.						
2		ADDED LADDER & T.V. CAMERA; SHIELDING AT EL. 649'-0"	MG					
1	7/10/67	REVISED AS INDICATED		I.H.				
0	5/10/67	ISSUED FOR CONSTRUCTION	G.L.	I.H.				

3/8" = 1'-0" JMA

BECHTEL COMPANY

PALISADES PLANT
CONSUMERS POWER COMPANY

EQUIPMENT LOCATION - REACTOR BLDG
SECTION G-G

5935 | M-8 | 7

SI
APERTURE
CARD

Also Available On
Aperture Card

Figure No 1-9

FSAR Rev 13

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ANSTEC APERTURE CARD

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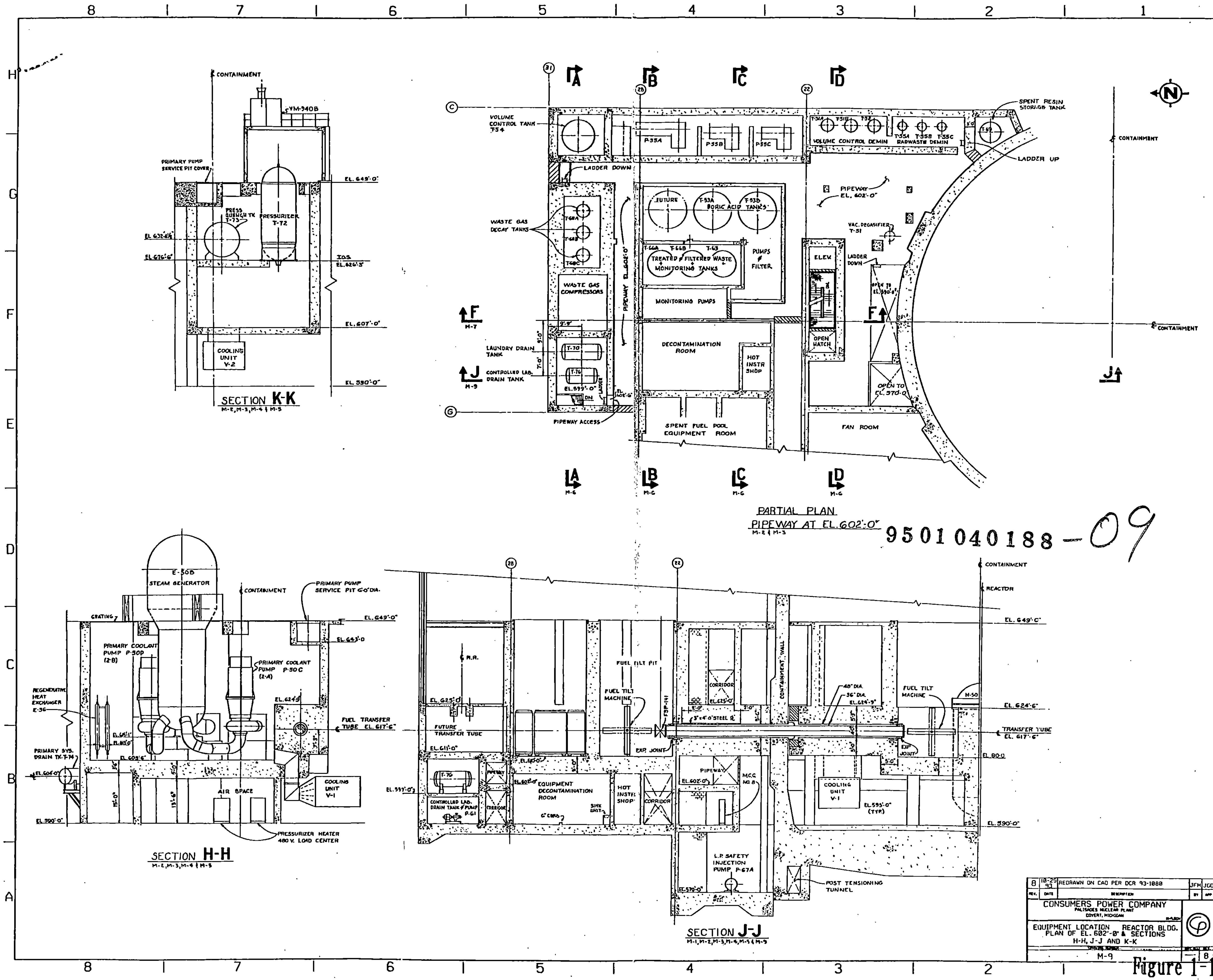

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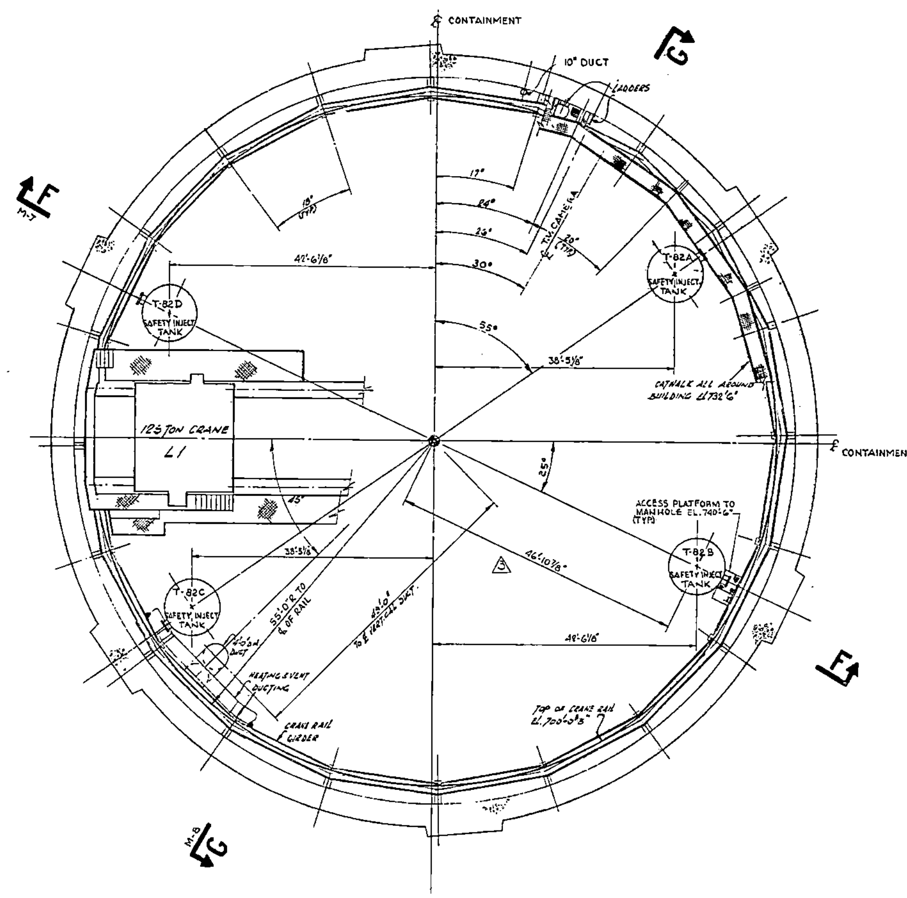
Figure 1-10 FSAR Rev 17



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CONSUMERS POWER COMPANY
PALISADES PLANT
FSAR UPDATE

EQUIPMENT LOCATION
REACTOR BUILDING
PLAN OF EL. 700'-0"

FIGURE NO 1-11

REVISION NO 0

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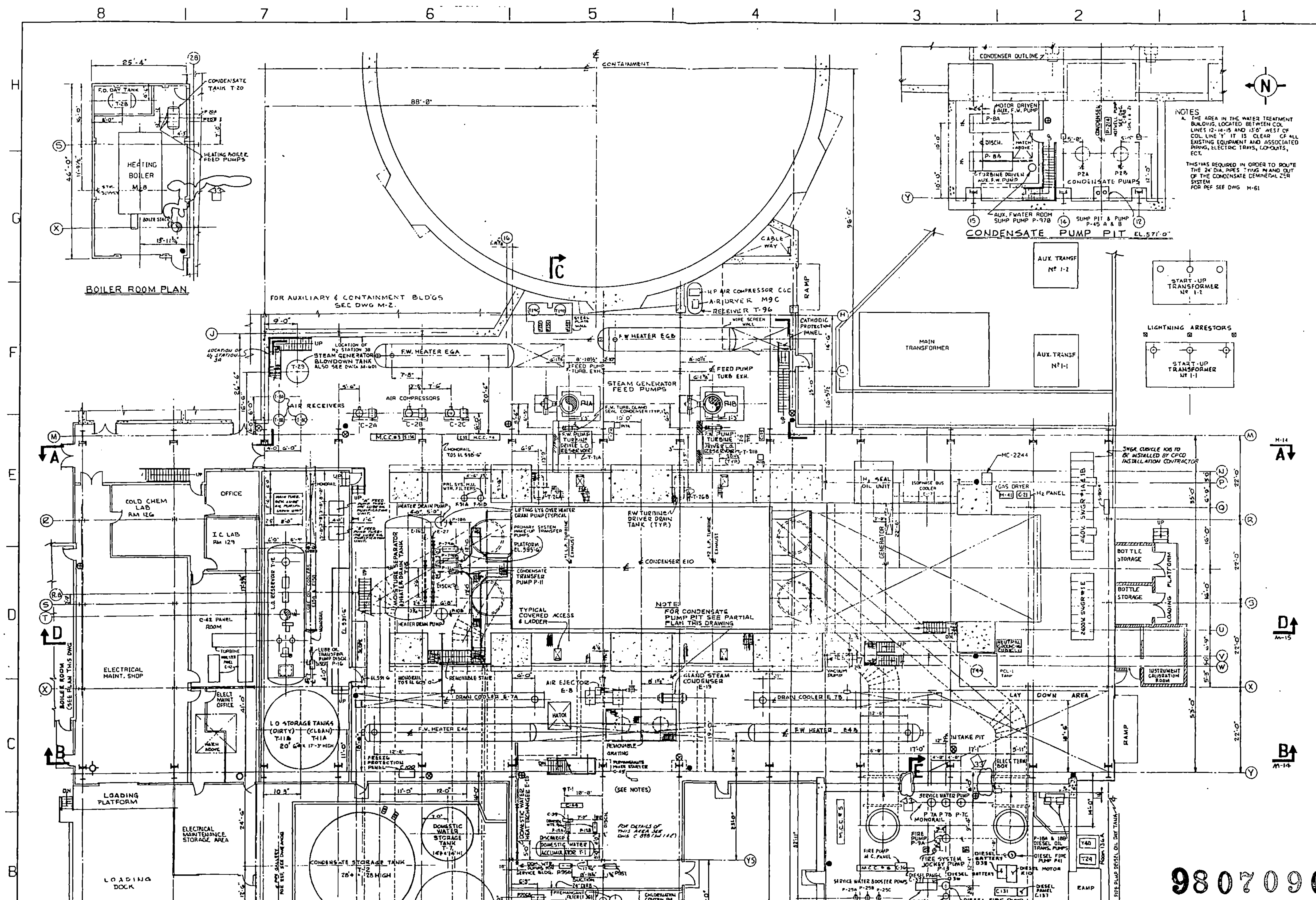

REV.	DATE	DESCRIPTION	BY	CH	APP
33	10-8 97	RELOCATED DOOR ABOVE SERVICE WATER PUMP @ (C-3) PER DCR 97-851.	MDS		
		REMOVED HOT WATER TANK & DIM'S. PER EAR-96-0832 AND DCR 97-793.			
32	8-22 97	ADDED STABILIZERS SEE M-11 SH. 3 PER SC 96-040 AND DCR 97-717.	MDS		
31	10-23 96	ADDED DIM'S. FOR CENTER LINES PER SC 94-053 AND DCP 96-276	MDS		
30	11-26 94	ADDED SUMP PUMP P-970 PER FES 94-170 & DCR 94-970	BMW		
		ADDED LOCATION OF MC-2244 FOR H2 GAS DRYER PER WO 24410423 & DCR 94-1030			

Consumers Energy
PALISADES PLANT

EQUIPMENT LOCATION
TURBINE BUILDING

M-11 | 1 | 33

9807090043-05

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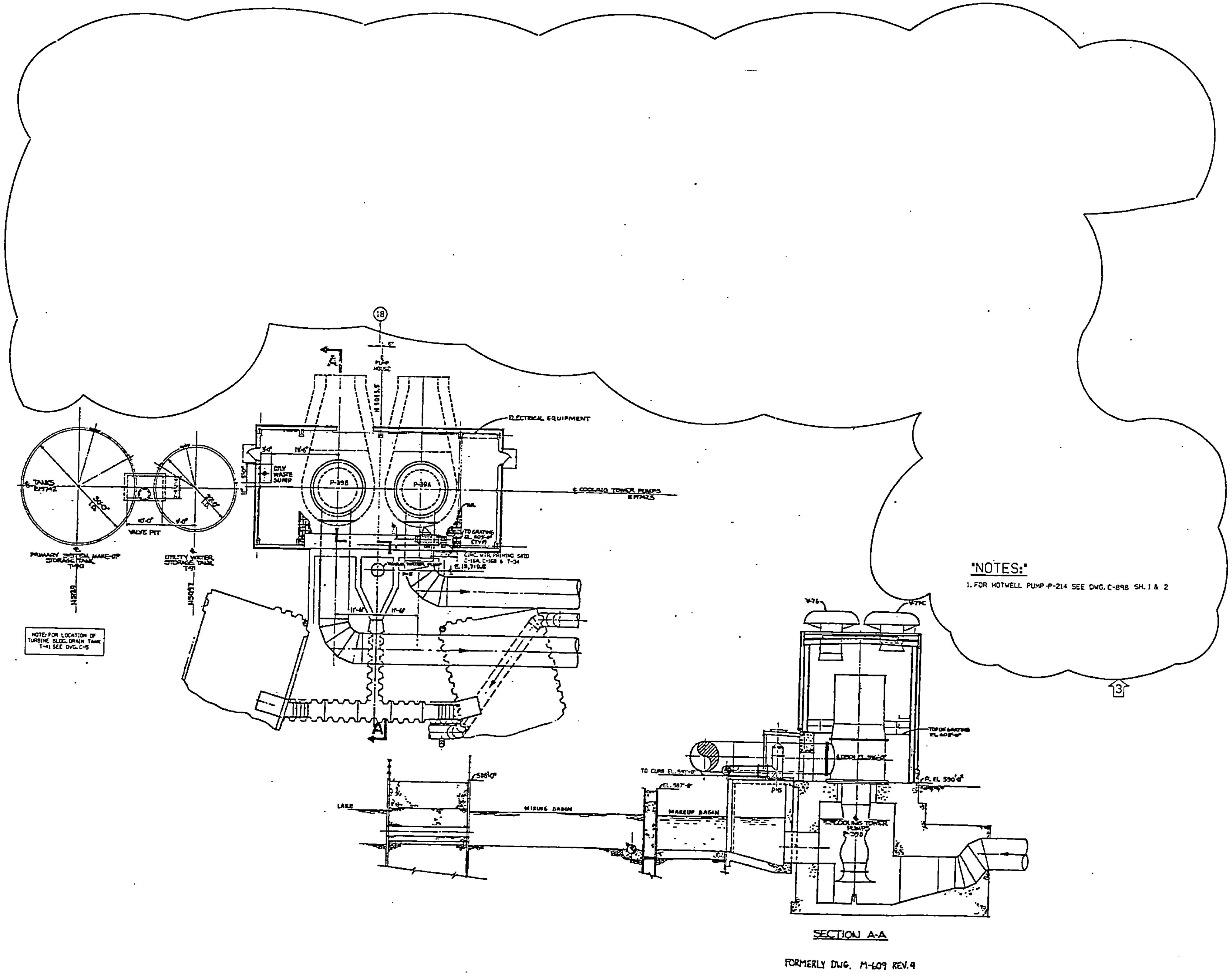
FIG 1-12 SHT 1
FSAR REV 20



3	10-22-96	REMOVED CONC. PUMP PIT & WATER TREAT. AREA PER SC-94-033 & SCR-96-775	JCO	MDS	
REV	DATE	DESCRIPTION	BY	CK	APP

CONSUMERS POWER COMPANY
PALISADES NUCLEAR PLANT
COVERT, MICHIGAN
M11-2.CIT

EQUIPMENT LOCATION-TURBINE BUILDING
PUMP STRUCTURE
PLAN OF EL. 590'-0"-SECTION

DRAWING NUMBER M11 | SHT. NO. 2 | REV 3

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Figure 1-12 Sh 2
Rev 19

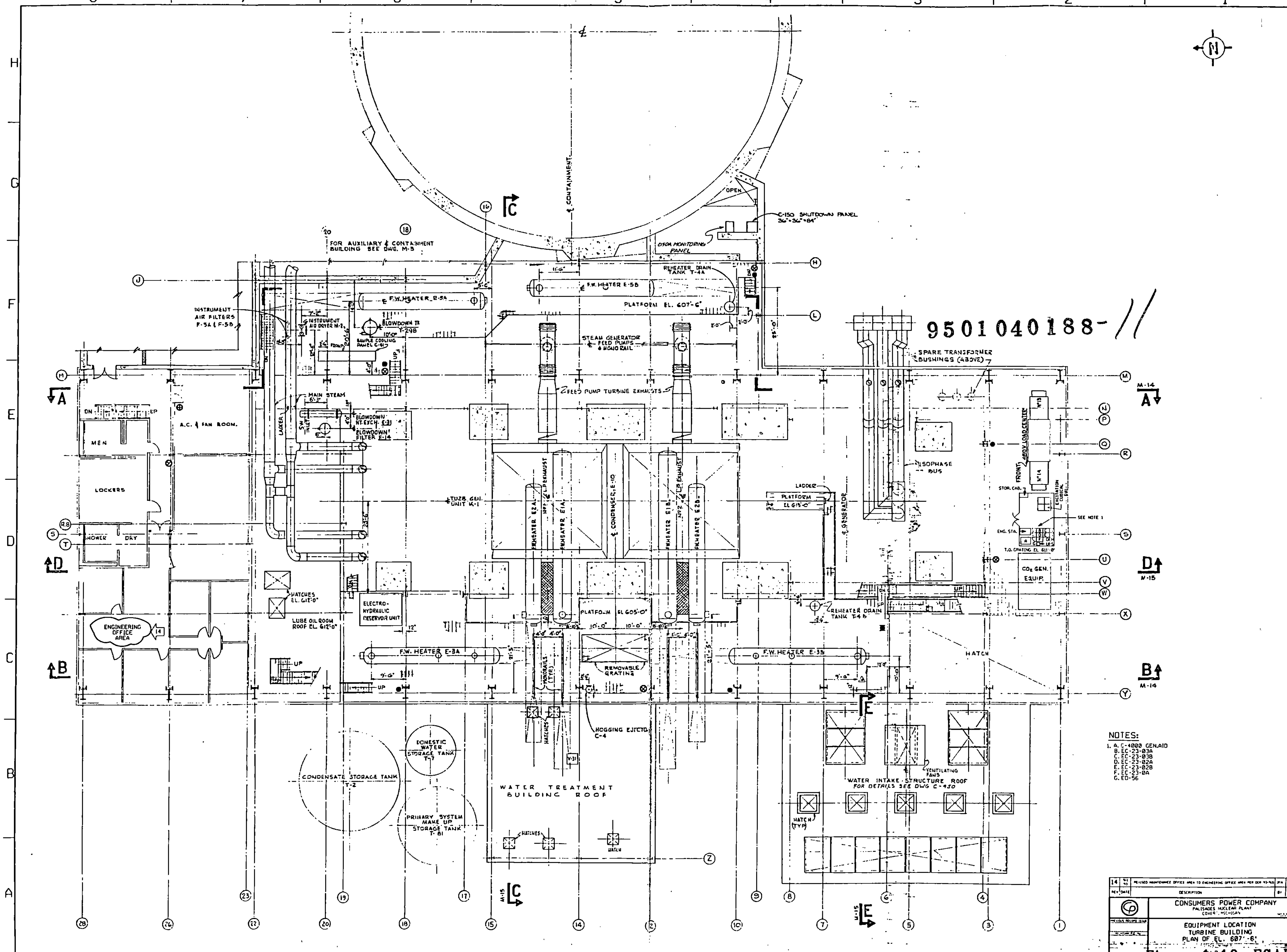

Figure 1-13 FSAR Rev 17

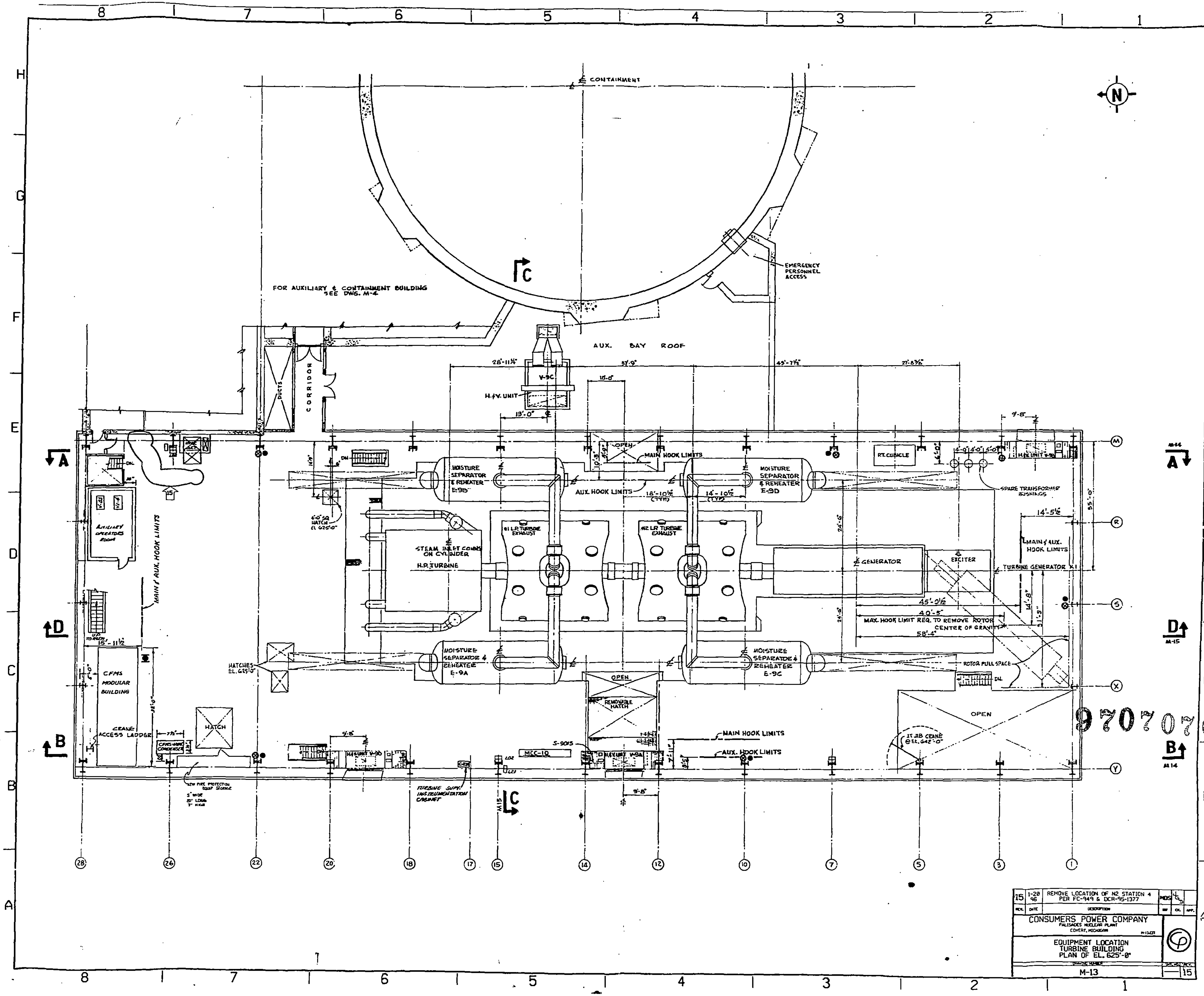

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Figure 1-14
Rev 19

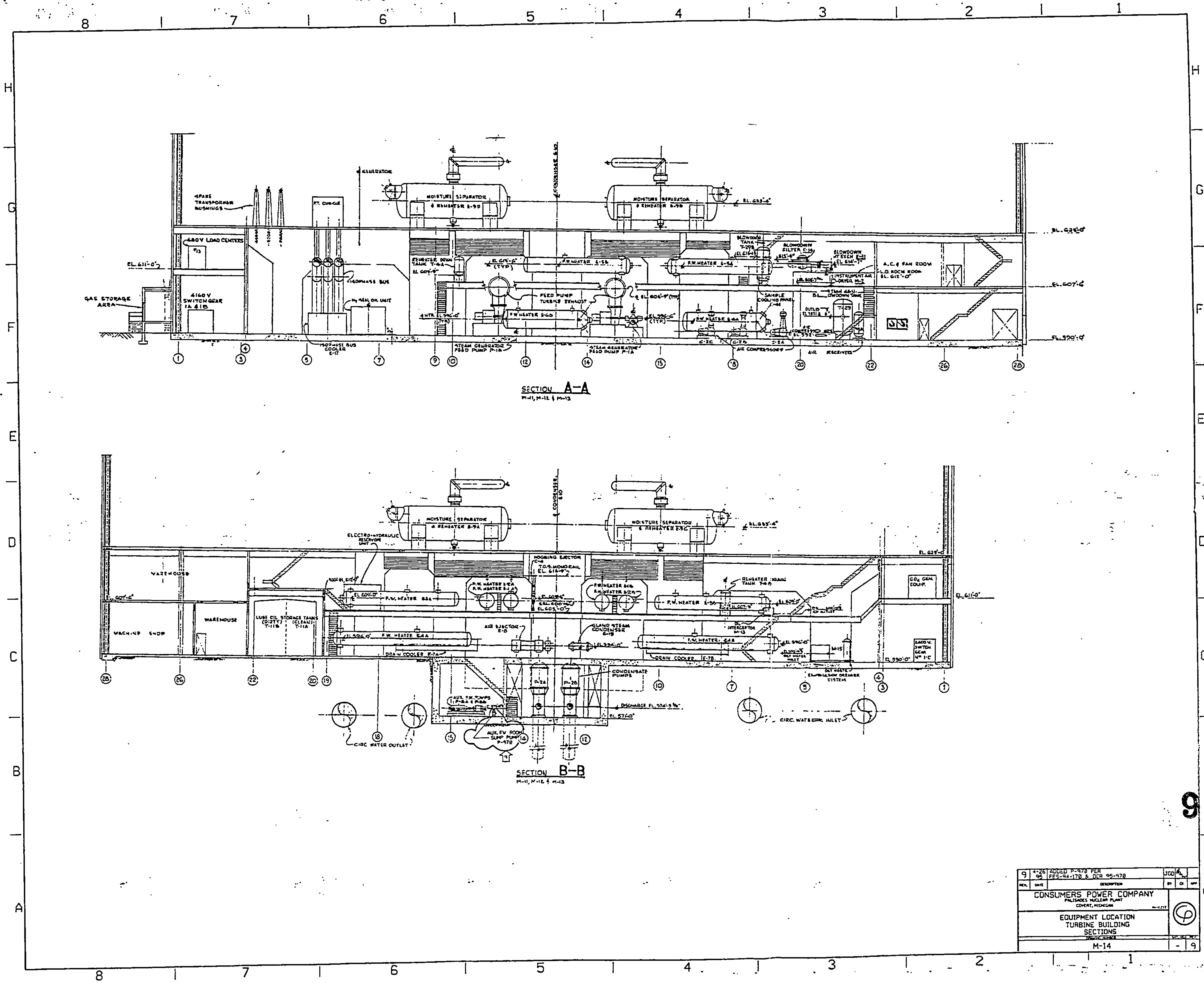

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Figure 1-15
Rev 19

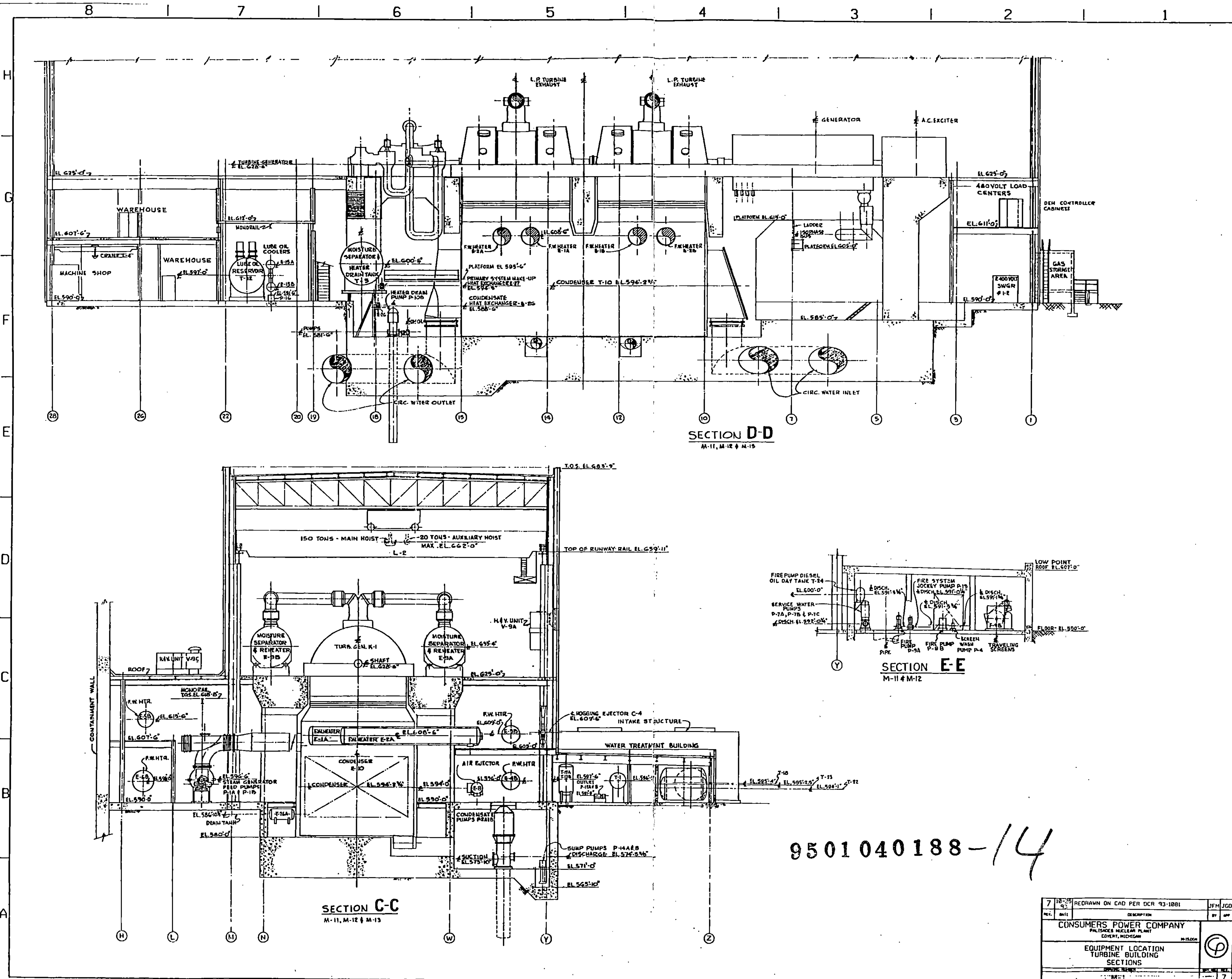

Figure 1-16 FSAR Rev 17

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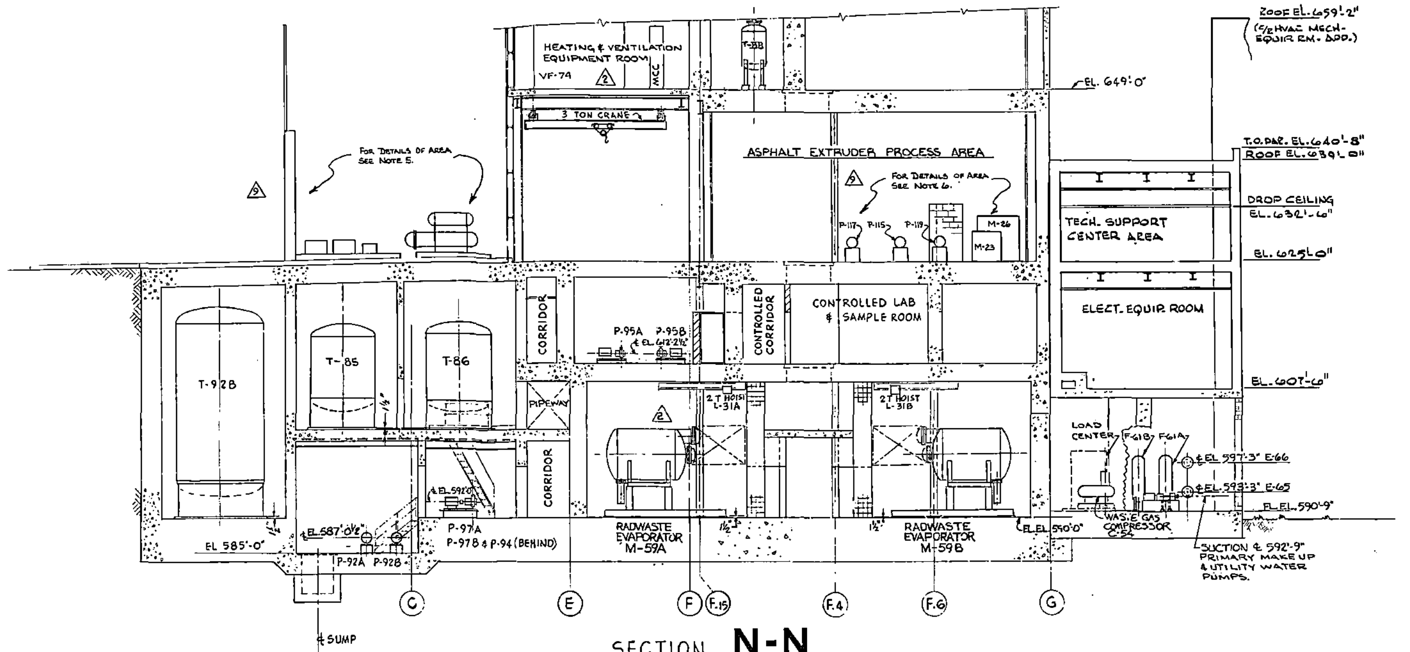
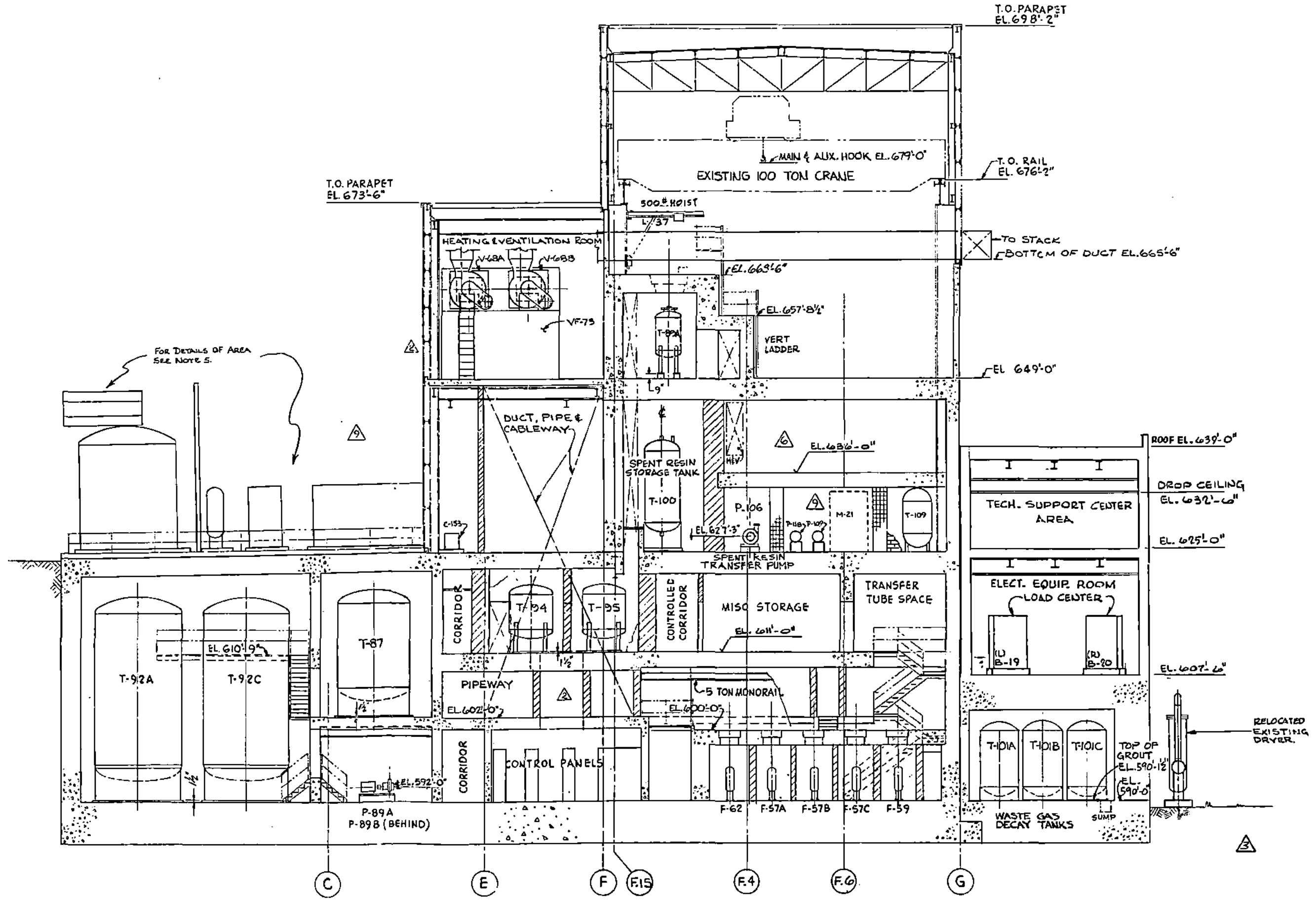
REF. DWGS.

1.) E/E UTILITY MANIFOLD OUTLINE DWG. NO. M-1024 (W.P. DWG.).
2.) DISTILLATE COLLECTION SYSTEM OUTLINE DWG. NO. M-1025 (W.P. DWG.).
3.) EQUIP. ARRANGEMENT AREA 11 AUX. BLDG. @ EL. 625'-0" DWG. NO. M-996-1 (W.P. DWG.).
4.) EQUIP. ARRGMT. AREA NO. 10 & 11 DWG. NO. M-996-3 (W.P. DWG.).
5.) EQUIP. ARRGMT. AREA NO. 10 DWG. M-996-4 & M-996-5 (W.P. DWG.)
6.) EXTRUDER/EVAPORATOR OUTLINE DWG., DWG. NO. M-998-1 & M-998-2. (W.P. DWG.).
7.) EQUIP. LOC. AUX. BLDG. RADWASTE MODS. PLAN OF EL. 625'-0", M-604



SECTION N-N
LOOKING SOUTH



SECTION M-M
LOOKING SOUTH

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CONSUMERS POWER COMPANY
PALISADES PLANT
FSAR UPDATE

EQUIPMENT LOCATION
AUXILIARY BUILDING
RADWASTE MODIFICATIONS
SECTIONS

FIGURE NO 1-22

DATE 10-31-86 REVISION NO 2

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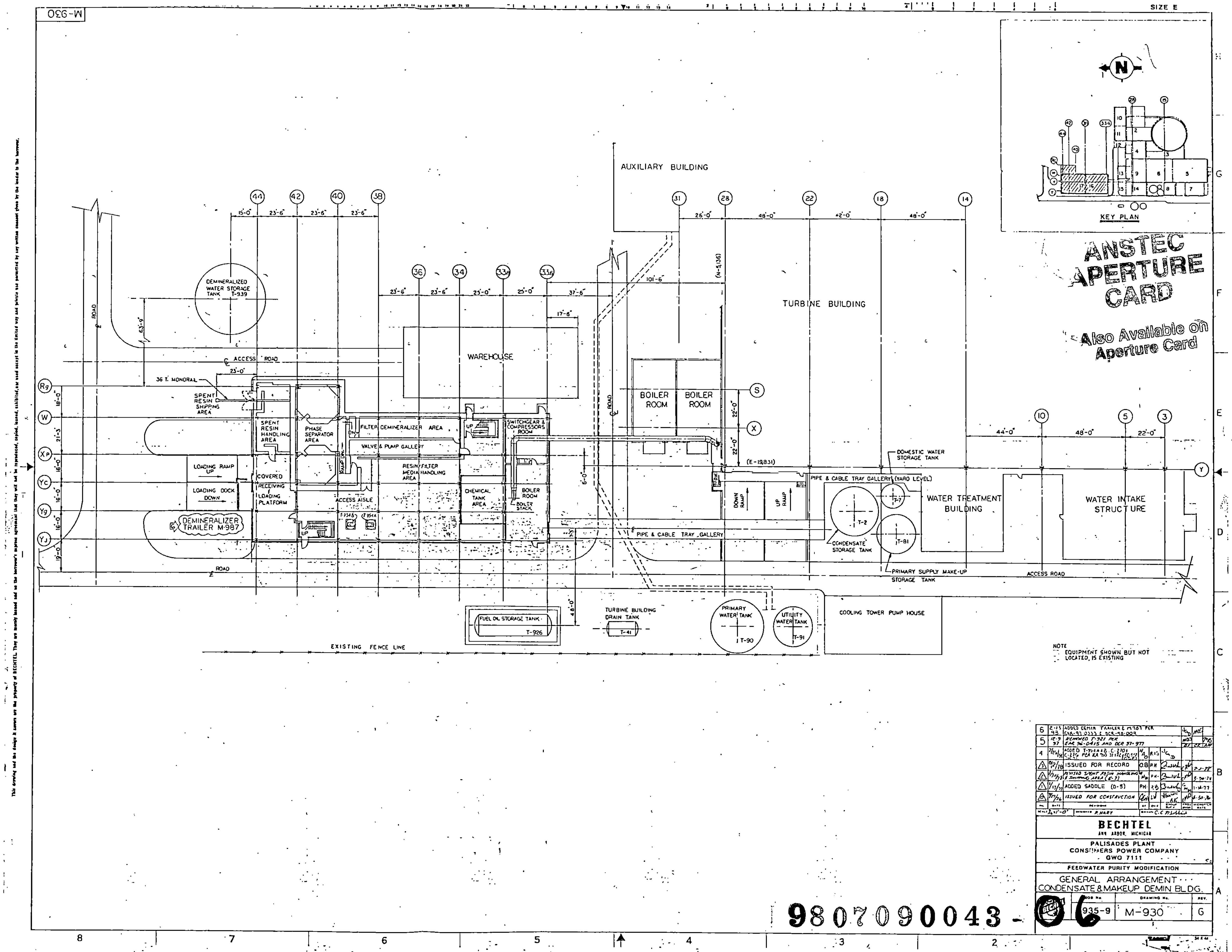
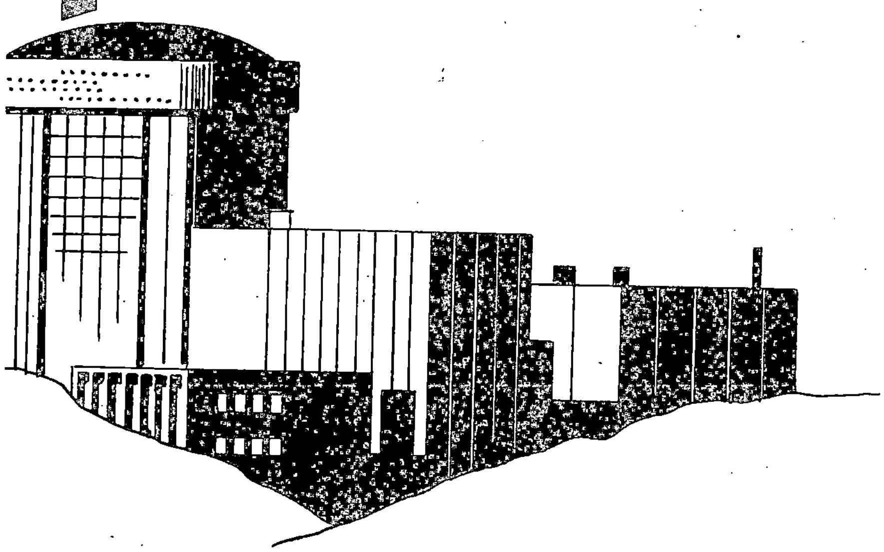


FIG 1-25
FSAR REV 20

CHAPTER 2



LISADES NUCLEAR PLANT

final safety analysis report

UPDATE

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5A.1 Special Report No 6 - Analysis of Postulated High-Energy Line Breaks Outside of Containment

VOLUME 4

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CHAPTER 7

INSTRUMENTATION AND CONTROLS

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 start-up, operation, and shutdown. The principal control station for the Plant is in the control room.

The Plant will be started up and shut down by manual control. In the power range, the operator will maintain manual control at all times. Annunciation, indication and recording will alert the operator and provide data on Plant conditions.

Major portions of the instrumentation and controls essential to Plant safety (Class 1E and safety related as discussed in Section 8.1) 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 of the operator. The control board consists of a control console and duplex vertical panels. Visible alarms with audible signals, located on the super-structure over the main control board, annunciate and identify abnormal operating conditions. A telephone system provides both intraplant and external communication. The control room is kept at a controlled temperature which is well within the design ambient temperature requirements of the instruments (Reference Section 6.10, Control Room Habitability).

All instrumentation and control equipment is made from highly reliable components. Environmental and seismic qualification of the instrumentation have been the subject of various reports (see Subsections 8.1.3, 8.1.4 and Class 1E classification in Section 5.2).

The protection instrumentation consists of independent multiple channels to ensure system flexibility while maintaining Plant safety at all times. A highly reliable source (Class 1E as discussed in Section 8.1) of electrical power is provided to ensure safe and reliable Plant operation.

The reactor operates within limits as a result of its inherent characteristics, the instrumentation and controls, the reactivity controls, and by operational procedures and administrative controls. Potential departures from these limits are audibly and visibly annunciated in the control room. A Reactor Protective System designed to protect the core initiates reactor trips.

This chapter describes safety-related as well as major nonsafety-related instrumentation and controls for systems described in other chapters. Special nonsafety-related instrumentation systems such as radiation monitoring, leak detection, fire detection and meteorological instrumentation are outlined in other chapters.

7.2 REACTOR PROTECTIVE SYSTEM

7.2.1 GENERAL

The Reactor Protective System (RPS) is a Class 1E system comprised of the sensor instrumentation, amplifiers, trip units, logic circuits, actuation circuits and other equipment as required to monitor selected nuclear steam supply system conditions, and is designed to reliably effect a rapid reactor shutdown (scram) if any one or combination of conditions deviates from a preselected operating range. The system functions to protect the reactor core.

The RPS controls are housed in four cabinets in the control room. The cabinets consist of the following components:

Four (4) Trip-Inhibit Switch Panel Assemblies

Four (4) Trip-Unit Assemblies (One for each channel A,B,C,D)

Twenty Eight (28) Bistable Trip Units (Seven in each channel)

Sixteen (16) Auxiliary Trip Units (Four in each channel)

Four (4) RPS Power Supply Assemblies

Four (4) Clutch Power Supply Assemblies

Four (4) RPS Bin Assemblies

Four (4) Trip Unit Interconnection Modules

One (1) Rod Drop Test Panel

Four (4) Auxiliary Logic Module Assemblies

Four (4) RPS Test Panel Modules

The following components are also part of the RPS cabinets and are part of the nuclear instrumentation described in Section 7.6:

Two (2) Nuclear Instrument Source/Wide Range Drawer Assemblies

Four (4) Power Range Safety Drawer Assemblies

One (1) Comparator Averager Assembly

Finally, a set of annunciators (non-class 1E) are also located on the above cabinets for operator convenience. An extension of the RPS is housed in an additional panel, also in the control room. This panel contains:

Four (4) Thermal Margin Monitors

Four (4) Reactor Power Calibration and Indication Assemblies
Another panel in the control room (C12) contains:

Four (4) pressure switch alarms (dual output each, one for PORV logic, one for ATWS logic)

7.2.2 DESIGN BASES

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

1. Instrumentation and controls for this Plant conform to the provisions of the General Design Criteria as indicated in Section 5.1 and to IEEE 279-1971.

2. No single component failure will prevent safety action.

3. Four independent measurement channels complete with sensors, sensor power supply units, amplifiers and trip units are provided for each safety parameter with the exception of loss of load and rate trips.

4. The channels are provided with a high degree of independence by separate connection of the sensors to the process systems and of the channels to preferred power supply buses. Separate raceways are used to ensure independence from cable faults.

5. The four normal measurement channels provide trip signals to four independent trip paths.

6. A trip signal from any two-out-of-four protective channels causes a reactor trip.

7. When one of the four channels is taken out of service for maintenance, the protective system logic can be changed to a two-out-of-three coincidence for a reactor trip by bypassing the removed channel. If the bypass is not effected, the out-of-service channel assumes a tripped condition, which results in a one-out-of-three channel logic.

8. The protective system ac power is supplied from four separate buses.

9. Open circuit, or loss of power supply of the channel logic, initiates an alarm and a channel trip.

10. All measurement channels and trip logic matrices assume the nonconducting state to provide a tripping function.

11. The RPS can be tested during reactor operation and when shut down.

12. The manual trip is totally independent of the automatic trip system.

13. Trip signals are preceded by alarms to alert the operator to undesirable operating conditions in cases where the operator could avert a reactor trip.

14. The RPS components are independent from the control channels.

15. Zero power mode bypass and low power nuclear channels bypass interlocks are provided with the same independence and single failures protection design as the trip circuits.

16. A diverse reactor trip to satisfy the 10CFR50.62 ATWS (Anticipated Transient Without SCRAM) rule was installed in 1990 (Reference 18). This trip uses a 2/4 pressurizer high pressure trip which is set higher than the normal pressurizer high trip. When one of the four signals of the ATWS trip is physically removed for maintenance, or any other reason, the protective system logic becomes 2/3.

    The new high pressurizer pressure trip modules are set up to trip at 2375+/-25 psi. This makes it secondary to the RPS trips which actuate at 2255+/-22 psi. Pressurizer safety relief valves are set at 2500, 2540, and 2580. The set points for the new modules were specifically chosen to avoid overlap with either RPS trips or safety relief valves.

    This modification was approved by the NRC by SER on December 5, 1989. It also covers diverse circuitry for both turbine trip and initiation of an auxiliary feedwater pump.

As shown in Figures 7-1 and 7-2, the RPS consists of four trip paths operating through coincidence logic to maintain power to, or remove it from, the control rod drive (CRD) clutches. Four independent measurement channels normally monitor each safety parameter. Individual channel trips occur when the measurement reaches a preselected value. A typical measurement channel functional diagram is shown in Figure 7-3. The channel trips are combined in multiple two-out-of-four logic. Each two-out-of-four logic system provides trip signals to one-out-of-six logic units, each of which causes a direct trip of the contactors in the ac supply to the CRD clutch power supplies. The common sides of the clutch power supplies are ungrounded.

Reactor trip is accomplished by de-energizing the magnetic clutch holding coils and releasing the full-length control rods to drop into the core. The four partial-length control rods are not equipped with magnetic clutches and are described in Chapter 3.

7.2.3 REACTOR PROTECTIVE SYSTEM ACTIONS

Rapid reactor shutdown is effected on the following conditions:

7.2.3.1 High Rate-of-Change of Power

A reactor trip for high rate-of-change of reactor power (neutron flux) is provided to protect the reactor against an uncontrolled control rod withdrawal while the core is critical but at very low power levels.

The rate-of-change of power is normally monitored at startup by two source range indications of the source/wide range nuclear instrumentation as discussed in Section 7.6.2.2.

A reactor trip is initiated if the rate-of-change of reactor power exceeds 2.6 decades per minute (dpm), over a range of about 10^{-4}% to 15% full power, by either of the two wide-range portions of the source/wide range nuclear instrumentation channels. The trip signal is automatically bypassed below 10^{-4}% and above 15% full power. Alarms for high rate-of-change of power are initiated at 1.5 dpm over the operating range of 10^{-4}% to 15% full power. In addition, manual rod withdrawal prohibits (between approximately 10^{-4}% and 15% full power from the two wide-range channels) prevent all regulating rods from being withdrawn, but do not prevent insertion.

Above 15% power the high rate-of-change of power trip unit is switched by the 15% bistable in the power range safety channels to perform Axial Shape Index (ASI) alarm functions. This function is derived in the thermal margin monitor using maximum selected power (Q_2) from either ΔT power (B) or calorimetric nuclear power (ϕ). As shown in Figure 7-7, the maximum power (Q_2) is used to determine a set point which is then compared with axial offset (Y) (see Subsection 7.2.3.5) to generate an alarm. The ASI alarm is applicable for positive as well as negative values and indicates that an axial offset administrative limit has been breached. The ASI alarm is used to monitor and protect the reactor from axial power distributions outside analyzed limits.

7.2.3.2 Variable High Power

A reactor trip using a variable high power (VHP) trip of the thermal margin monitors is provided to shut down the reactor when the indicated neutron power or thermal power (maximum selected) exceeds a predetermined value. This trip is available at full or part loop operation and is independent of power level or pump configuration. The high power trip signals are initiated by two-out-of-four coincidence logic from the four thermal margin monitors. During normal full power Plant operation with all coolant pumps operating, reactor trips are initiated when the reactor core power level exceeds a nominal value of 106.5% of indicated full power; this trip level represents a reactor power of no greater than 115% of full power when instrument and calorimetric errors are taken into account. Provisions are made in the thermal margin monitor to select VHP set points for three primary coolant pump operation.

The variable high power function is enabled at approximately 30% power with variable trip and pretrip set points set at 15% and 13.5% (Reference 21), respectively. Each increase of power level requires reset action of all four thermal margin monitor VHPT functions prior to the trip set point for continued escalation up to a maximum of 106.5% power. Upon power descension, the VHP trips and pretrips automatically decrease maintaining the predetermined set point margins. The pretrip alarms provide annunciation in addition to rod withdrawal prohibit signals. A block diagram of VHP is illustrated in Figure 7-7.

The neutron flux at the out-of-core detectors for a given reactor power level is affected by control rod position, radial core power distribution, and average coolant temperature. These effects are compensated for by performing periodic Plant heat balances and adjusting the calibration of the power range channels accordingly. Between secondary heat balances, an alarm has been added for operator convenience which indicates when power based on steam generator ΔT measurement is different than that based upon neutron monitoring by greater than a predetermined amount. The difference between the PCS hot and cold leg temperatures is compared to a power range safety channel output. This difference is displayed on a meter relay for each channel, and the difference alarmed when it exceeds a preset adjustable value. The alarm incorporates a time delay of up to eight seconds to dampen the effect of PCS flow oscillation on PCS elements. The meter relays and the associated calibration controls are located on the front of the reactor power calibration cabinet panel in the control room. The thermal power (B) is calculated in each of the four thermal margin monitors.

The thermal power signal (B) is then sent to the power comparator unit where it is subtracted from the nuclear power signal received from a power range safety channel. This difference may be either positive or negative and is sent to an indicating meter relay. The meter relay is equipped with adjustable high- and low-set points. When the absolute value of the deviation signal reaches the set point, an alarm is sounded on the Reactor Protective System alarm panel and a light is lit adjacent to the meter relay.

Periodic calibration of the system can be performed using a secondary Plant heat balance to determine actual reactor power. The nuclear power calibrate potentiometers and thermal margin monitor adjustable constants for ΔT power will be used to adjust these outputs to agree with the secondary Plant heat balance.

7.2.3.3 Low Primary Coolant Flow

A reactor trip is provided to protect the core from a power to flow mismatch. There are four primary coolant pumps, with flow in each measured by sensing differential pressure between the coolant pump suction line and the primary coolant input line to the associated steam generator. The flow measurement signals are provided by summing the output of the differential pressure transmitters to provide an indication of total coolant flow through the reactor. A reactor trip is initiated by two-out-of-four coincidence logic from either of the four independent measuring channels when the flow function falls below a preselected value.

Provisions are made in the Reactor Protective System to permit operation at reduced power if one pump is taken out of service. For this mode of operation, the low flow trip points are changed manually at the RPS bistable trip units to the allowable values for the selected pump condition, thus providing a positive means of assuring that the more restrictive settings are used.

Pretrip alarms provide indication of degrading coolant flow prior to the four pump trip set point, or three pump trip set point.

A key-operated bypass switch ("Zero Power Mode Bypass" switch, see Subsection 7.2.5.2) allows this trip to be bypassed for subcritical testing of control rod drive mechanisms. The trip bypass is automatically reset above 10^{-4}% full power.

7.2.3.4 High Pressurizer Pressure

A reactor trip for high pressurizer pressure is provided to prevent excessive blowdown of the Primary Coolant System by relief action through the safety valves.

The trip signals are provided by four narrow range independent pressure transducers measuring the pressurizer pressure. A typical channel diagram is shown on Figure 7-3.

A reactor trip is initiated by two-out-of-four coincidence logic from the four independent measuring channels if the pressurizer pressure exceeds a preset pressure (≤ 2,255 psia). A diverse reactor trip was added in 1990 to provide a two out of four coincidence pressurizer high pressure trip from four independent measuring channels if the pressurizer pressure exceeds a preset pressure (>2375 psia). See Figure 7-29A Sh. 1 and 2. Since no credit is taken for the relief capacity of the primary coolant power-operated relief valves in Chapter 14, the Plant operates with these valves isolated due to leakage problems.

Pretrip alarms are initiated if the pressurizer pressure exceeds a preset pressure (2,205 psia).

The Palisades Plant does not have a high pressurizer level trip since the requirement for this trip is based upon an assumption that a malfunction in the pressurizer level control system would cause the charging pumps to continue running and the letdown orifices to remain closed, which would result in filling the pressurizer solid with water. Immediately thereafter, a turbine generator load rejection would occur resulting in high pressurization of the Primary Coolant System (refer to the detailed analysis in Subsection 7.5.3.4).

The Plant design includes the use of two pressurizer level control channels. A separate level transducer is provided for each channel with a manual switching arrangement to select which channel supplies the control signals to the charging pumps and letdown orifices. The level alarm system is actuated from each level channel with no manual switching selection required.

With the pressurizer level at its full Plant load value, the level alarm is actuated at 15 inches above this point. Assuming that full charging system flow occurs and that there is no letdown flow, it would take approximately 35 minutes to completely fill the pressurizer. For the same conditions, except with the Plant at zero load, it would take 75 minutes.

The combination of redundant alarms, redundant controls and time required before operator action must be taken, is considered adequate protection for the primary system pressure boundary.

7.2.3.5 Thermal Margin/Low Pressure

A reactor trip is initiated by a continuously computed function of core power, reactor coolant maximum inlet temperature, core coolant system pressure and axial shape index to prevent reactor conditions from violating a minimum departure from nucleate boiling ratio (DNBR).

$$P_{Var} = \lambda\, Q_{DNB} + \beta\, T_{IN} + \gamma$$

Where: λ, β and γ are constants, T_{IN} is measured reactor inlet temperature, and

$$Q_{DNB} = QR_1 * QA$$

Where: $QA = F(Y)$
Y = Axial Offset
$QR_1 = F(Q_1)$
Q_1 = Max (ϕ or B)
B = ΔT Power
ϕ = Adjusted Nuclear Power

Figure 7-7 is a simplified functional diagram of this system. The signal representing core power (Q_1) is the auctioneered highest of the neutron flux power (ϕ) and the ΔT power (B) signal. This signal is used to calculate QR_1. The measured axial shape index signal (Y), which includes the adjustment for shape annealing and represents the peripheral axial shape index, is used to calculate QA. The QA, QR_1 and constant λ are multiplied together to generate a signal representing the first term in the P_{Var} equation.

The signal (T_{Cal}) representing reactor coolant inlet temperature (T_{IN}) is the highest measured cold leg temperature (T_C). The second term in the P_{Var} equation is then the product of constant β and T_{Cal}. The third term is constant γ. These three terms are summed to produce P_{Var}. This limit is then compared to a fixed low pressure trip limit (P_{Min}). The auctioneered highest of these signals (P_{Var} and P_{Min}) becomes the trip limit (P_{Trip}). P_{Trip} is compared to the measured reactor coolant pressure and a trip signal is generated in the RPS trip unit when P is less than or equal to P_{Trip}. A pretrip alarm is also generated when P is less than or equal to the pretrip setting $P_{Trip} + \Delta P$.

For three pump operation, power is limited to 34% of rated power for a maximum of 12 hours. During this mode of operation, the variable high power level trip in conjunction with the TM/LP P_{Min} limit and the secondary system safety valves (1,000 psia) assure that adequate DNB margin is maintained.

The logic shown in Figure 7-7 is for Channel A. There are three other functionally identical channels; B, C and D. The output of the TM/LP trip unit bistable is connected to the Reactor Protective System as shown on Figure 7-2.

A key-operated bypass switch ("Zero Power Mode Bypass" switch, see Subsection 7.2.5.2) allows this trip to be bypassed at low power level. The trip bypass is automatically reset above 10^{-4}% full power.

7.2.3.6 Loss of Load

A reactor trip will automatically be initiated after a turbine trip occurs. A turbine low auto stop oil condition occurs with all types of turbine trips. The reactor trip will be initiated when the turbine auto stop oil pressure decreases, causing the auto stop oil pressure switch contacts to close and energize two turbine trip auxiliary relays (see Figure 7-14, Sheet 2).

Each relay will provide a reactor trip signal to two of four protective system channels.

The loss of load reactor trip is an anticipatory trip which is not required to protect the reactor since the primary trip is high primary system pressure. As such, its measuring channels components are not required to be Class 1E and its circuits need not meet IEEE 279-1971. This trip is automatically bypassed when three of four power range safety channels indicate < 15% full power (see Subsection 7.2.5.1).

Isolation of Nonclass 1E turbine trip circuits from the Class 1E Reactor Protective System is provided by the turbine trip relays (see Subsection 7.2.9).

7.2.3.7 Low Steam Generator Water Level

Low steam generator downcomer water levels will cause a loss-of-heat-removal capability from the Primary Coolant System.

A reactor trip signal is initiated by two-out-of-four logic from four independent downcomer level differential pressure transmitters on each steam generator. The steam generator low-water level trip set point is at 25.71 feet above the tube sheet and the normal water level is 31.74 feet above the tube sheet (Reference 22). Thus, the reactor trip set point is 6 feet below the normal water level, which is at the center line of the feedwater ring. Pretrip alarms are actuated to provide for annunciation of approach to reactor trip conditions.

7.2.3.8 Low Steam Generator Pressure

A reactor trip on low steam generator secondary pressure is provided to protect against excessively high steam flow caused by a steam line break. The trip set point is ≥ 500 psia.

An abnormally high main steam flow from either steam generator will cause the secondary pressure to drop rapidly.

Four pressure transmitters on each steam generator actuate trip units which are connected in a two-out-of-four logic to initiate the reactor protective action if the steam generator pressure drops below a preselected value. Signals from two of the four indicating meter relays from either steam generator will close the main steam isolation valves on both steam generators. Pretrip alarms are also provided.

A key-operated bypass switch ("Zero Power Mode Bypass" switch, see Subsection 7.2.5.2) allows the reactor trip to be bypassed. This trip bypass is automatically reset above 10^{-4}% full power.

7.2.3.9 Containment High Pressure

A reactor trip is initiated on high containment pressure.

Four independent pressure switches actuate trip units which are connected in a two-out-of-four coincidence logic to initiate the reactor protective action when the containment pressure reaches 4 psig.

This reactor trip is in addition to the thermal margin/low-pressure trip to ensure that the reactor is tripped before the safety injection and containment spray are initiated (see Subsection 7.3.3.2).

Containment high-pressure circuitry related to the RPS does not involve any pretrip actions. A pretrip of 0.9 psig is initiated through containment pressure indications in the control room.

7.2.3.10 Manual Trip

A manual reactor trip is provided to permit the operators to trip the reactor (see Figure 7-1). Manual actuation of either of two independent reactor trip push-button switches in the main control room causes direct interruption of the ac power to the dc power supplies feeding the CRDM electromagnetic clutches.

One manual trip push button interrupts the control power to the holding coils of four contactors whose contacts break ac power to the clutch power supplies. The second push button interrupts power to the undervoltage coils of two circuit breakers which disconnect all ac power to the clutch power supplies. This system ensures diverse means of trip actuation.

7.2.4 SIGNAL GENERATION

Four instrument channels are used to generate the signals necessary to initiate the automatic reactor trip action except for the loss of load and high neutron flux rate-of-change trips where one and two measuring channels, respectively, are used. The signal cable routing and readout drawer locations in the RPS cabinets are separated and isolated to provide channel independence.

1. The high rate-of-change of neutron flux signals is generated by the two wide-range measuring channels (Figure 7-8) which monitor the flux from source level to 125% of full power. These channels receive signals from flux monitors in the biological shield around the reactor (refer to Section 7.6 for details).

2. The high neutron flux signals are supplied by four linear flux measuring channels (Figure 7-9) covering the flux range from 0% to 125% full power. These channels receive signals from ion chambers which monitor the full length of the core and are located in the biological shield around the reactor (refer to Section 7.6 for details).

3. The primary coolant pressure, flow, thermal margin, steam generator pressure and water level, and containment pressure trips are each actuated from signals generated by separate sets of sensors. The primary coolant pressure is measured in the pressurizer, flow is measured by monitoring the pressure difference across the steam generators, thermal measurements are taken from the reactor inlet and outlet piping in each loop and combined with primary coolant pressure, axial shape and power to determine thermal margin, steam generator water level and pressure are monitored in each steam generator, and containment pressure is obtained via appropriate pressure sensors (refer to Section 7.6 for details).

4. The electronic sensors (transmitters) are located in the air room inside containment with physical separation provided between each channel. The output of each transmitter is an ungrounded current loop (see typical loop, Figure 7-3) supplying signal receivers and trip units.

5. The trip units have three isolated outputs which feed the logic matrices. Additional outputs feed the pretrip alarm, the trip alarm, the plant data logger and the critical functions monitor (see Subsections 7.6.2.5 and 7.6.2.6). Provisions are also made for test inputs to the trip units for normal protective system testing and are described in Subsection 7.2.6. The protective system outputs referred to here are connected into logic matrices as shown in Figure 7-2.

7.2.5 LOGIC OPERATION

7.2.5.1 Trip Logic (See Figure 7-2)

The instrument channels which supply protective action, operate channel trip units in the corresponding channel cabinet of the Reactor Protective System; each unit includes three sealed, electromagnetically actuated reed relays and associated contacts. Four units are actuated for each trip condition, eg, high primary coolant pressure. The relays in each unit are numbered one, two and three. Each relay has a single-pole, double-throw (SPDT) contact. The normally open contacts of the No I relays in the Channels A and B trip units are connected into a two-out-of-two logic ladder matrix. (The normally open contacts are used for the logic ladders so that the relays are energized and the contacts closed under operating conditions.) The respective No 2 and No 3 relay contacts are similarly connected into separate logic ladder matrices. With the Channels C and D trip units arranged in a similar manner, there are a total of six independent matrices. These logic ladders are designated the AB, AC, AD, BC, BD and CD logic trips.

The output of each logic ladder is a logic trip set of four-sealed, electromagnetically actuated power reed relays. Each relay in these sets has an SPDT contact. The contacts from one relay of the set from each logic ladder 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 a power trip relay which interrupts the power to the CRDM clutches. De-energizing of any one power trip relay interrupts (opens) one trip path and effects a one half trip. De-energizing any set of logic trip relays (which results from trip action of any two channels through a ladder matrix) causes an interruption of all trip paths and a full trip.

The functions of Reactor Protective System Relays Kl, K2, K3 and K4 shown on Figure 7-1 are listed in Table 7-1.

7.2.5.2 Trip Bypass Logic

Four different types of trip bypasses (or inhibits), as discussed below, are used in the Reactor Protective System. An annunciator indicates the presence of any trip bypass.

Testing Bypass - The first of these bypasses ("Test Bypass") is used when it is desired to physically remove an individual trip unit from the system, or when calibration or servicing of a trip channel could cause an inadvertent trip. A key lock bypass switch is located above each trip unit. The key cannot be removed in the bypass position. Each trip bypass (high power, low pressure, etc) has a different lock cylinder combination; however, corresponding trips in each of the four protective channels have the same cylinder combination. Since only one key for each type trip is provided, an operator cannot bypass more than one channel at a time. Duplication of keys is prevented by using "ACE" type round key locks. Turning the key lock switch closes three separate contacts in parallel with the three normally open trip contacts of the trip unit. A light above the bypass switch indicates that the trip channel has been bypassed.

During the bypass condition, the system logic reverts from 2/4 channels required for trip to 2/3 channels required for trip. The Reactor Protective System continues to provide protective function in the event of a single failure in one of the unbypassed channels.

Nuclear Channel Bypass - The second type of trip bypass ("Nuclear Channel Bypass") is an automatic bypass. This is used in two cases where a bypass is required during start-up or shutdown. The high rate-of-change of power signal is bypassed when the reactor power is below 10^{-4}% of full power or above 15% of full power. The loss-of-load trip is bypassed during reactor start-up when power is below 15% of full power. Bypassing is automatically accomplished by contacts operated from the wide range portion of a nuclear instrumentation channel for the high rate-of-change of power bypass at 10^{-4}% full power and by contacts from the power range safety channel for the loss-of-load bypass at 15% full power. When power level is such that a bypass is required, contacts in parallel with the trip contact to the input of the high rate-of-change of power and the loss-of-load auxiliary trip units prevent reactor trip. The bypasses are arranged so that a given power range channel bypasses the corresponding rate-of-change power trip channel above 15% of full power, removes the bypass for loss of load above 15% of full power and enables the local power density alarm function of the rate-of-change auxiliary trip unit (K25 relays on Figure 7-10). A single wide range section of a nuclear instrumentation source/wide range channel feeds two rate-of-change of power level bypass circuits, each bypass circuit in turn bypasses a single rate-of-change of power trip channel (K26 relays on Figure 7-10). Removal of a power range safety channel or a wide range channel results in removal of the bypasses associated with that channel.

Zero Power Mode Bypass - The third type of trip bypass ("Zero Power Mode Bypass") is required when the Plant is shut down to permit maintenance or shutdown operation. This bypass is manually initiated when it is desired to raise the control rods for control rod drop testing or to assure a reserve shutdown margin during low power testing.

This bypass is effective for the following trips: low flow, low pressure SG No I and 2, and TM/low pressure.

This bypass is automatically removed by contacts operated from the wide-range logarithmic nuclear instrumentation channel when reactor power increases above 10^{-4}% full power; each wide-range logarithmic channel resets two RPS trip channels.

Four key lock switches, one located in each cabinet, control the initiation of the bypass. Each switch controls the initiation of the bypass for that cabinet. All four switches must be turned to make the bypass fully effective. This method of initiating the bypass was selected because four bypass switches, one in each protective channel cabinet of the Reactor Protective System, allow for complete channel separation as required by IEEE 279-1971.

Operation of a given bypass switch is shown in Figure 7-11. The bypass switch applies +15 V to the base of the transistor that drives the trip relays in the trip unit. The trip unit will remain in the untripped condition, regardless of the input signal level, as long as the +15 V is applied to the transistor base. Normally open contacts in series with the bypass switch allow auto bypass removal. Operation of the pretrip alarms is unaffected by the bypass switch.

ATWS System Bypass - The fourth type of trip bypass (ATWS system high pressurizer pressure trip) is necessary to permit maintenance on the ATWS trip system. This bypass is manually initiated from the control room via key switch HS-0102D on panel C12. The alarm for this bypass annunciates on the C06 panel.

7.2.5.3 CRDM Clutch Power Circuitry (See Figure 7-1)

The CRDM clutches are separated into two groups. The clutches in each group are supplied in parallel with low-voltage dc power by an ungrounded feed line. Two ac to dc converters supply each feed line so that one converter being cut off does not cause release of the clutches. The converters on each side are each supplied by a line from a preferred ac bus to assure a continued source of power. Each line passes through two interrupters (each actuated by a separate trip path) in series so that, although both ac lines must be de-energized to release the clutches, there are two separate means of interrupting each line. This arrangement provides means for the testing of the protective system including Relays Kl, K2, K3 and K4

7.2.6 TESTING

Provisions are made for monthly testing of the Reactor Protective System while the reactor is at operating power levels or when shut down. These tests cover the trip paths from sensor, to the output to the final CRDM clutches. The testing system is completely isolated from the protective system circuitry itself. Failure of any part of the testing system does not prevent proper operation of the trip circuitry. The system test does not prevent the protection function.

In response to Generic Letter 83-28 regarding the ATWS events at Salem Nuclear Power Plant, Safety Evaluation Reports (SERs) were written in response to inputs regarding testing of the RPS. The SERs concluded that the testing intervals and on-line functional test capabilities met NRC requirements for reliable and independent testing of trip contactors, see Reference 12. The stated basis for this conclusion was that Palisades' design meets Item 4.5.2 of Generic Letter 83-28.

Isolation of the testing circuitry is accomplished with an isolated test power supply and double coil relays. One coil is normally used in the protective system circuitry and the other coil is used in the testing system circuit. The double coil relays permit system testing without bypassing or inhibiting protective functions. Depending on the relay action required, the coil in the testing circuit, which is used to provide a magnetic flux in the relay core, aids or bucks the magnetic flux produced by the coil in the circuit of the protection system. This feature allows all trip test switches to be located in the circuitry of the test system, thus providing complete isolation of the two systems.

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, and initiating a test input which is also indicated on the voltmeter. 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 to the trip unit signal input circuit by an internal test signal generator which is incorporated in the test features of the RPS power supply. With the test signal generator functioning, a trip unit is selected for test and the test signal is increased or decreased (depending upon the type of trip unit) by depressing the manual test switch. The test circuit permits various rates-of-change of the signal tested by the coarse and fine adjustments of the test signal generator. 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 coincidence logic matrix are tested one at a time. The test circuits in the logic permit only one pair of coincidence matrix logic relays to be tripped while one set of matrix output relays can be held at the same time; the application of hold power to one set of matrix output relays denies the power source to the other sets. In testing a logic trip set, eg, 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 push-button switch. Operation of the matrix trip test switch de-energizes a parallel pair of trip unit relays. With the ladder-logic relay contacts open, the logic trip relays may be de-energized one at a time (by rotating the matrix-relay-trip test switch) to initiate a half-trip. Indicator lights on the trip unit relay coils and on the dc power supply ac feed lines provide verification that coil operation and half-trip conditions have occurred.

The capability to test the RPS Relays Kl through K4 associated with the RPS "trip/reset" function has been provided as described in Subsection 7.2.5.3. The nuclear channel bypass relays (K25 and K26), and their contacts described in Subsections 7.2.5.2, 7.6.2.2 and Figure 7-10, can be tested with the reactor at power. The K25 and K26 relay can be tested as part of the normal Reactor Protective System tests by varying the wide-range logarithmic channel output above and below 10^{-4}% full power and the power range safety channel output above and below 15% full power. In order to verify the loss-of-load trip function (see Subsection 7.2.3.6), a test switch is provided to energize the turbine trip relays while the unit is shut down.

7.2.7 EFFECTS OF FAILURES

The Reactor Protective System is designed and arranged to perform its function with single failures. Some of the faults and their effects are described below.

Analog portion:

1. A loss of signal in a channel initiates channel trip action for all trips except high rate-of-change of power, high-power level and high-pressurizer pressure.

Although high-pressurizer pressure will not trip, the loss of signal will cause a thermal margin/low-pressure trip.

2. Shorting of the signal leads to each other, has the same effect as a loss of signal. Shorting the leads to an ungrounded voltage source, has no effect since the signal circuit is ungrounded.

3. Single grounds on the signal circuit have no effect. Double grounds would tend to cause the channel to fail in the safe direction.

4. Open circuit of the signal leads has the same effect as a loss of signal.

5a. TMM VHPT function will cause a VHPT trip when power removed from TMM.

Logic portion:

5b. Inadvertent operation of the relay contacts in the matrices can be identified by the indicating lights.

6. Shorting of the pairs of contacts in the matrices prevents 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 matrix pairs.

7. 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.

8. The logic matrices will each be supplied by two power sources. Single loss of power source has no effect. Loss of power to a logic matrix initiates a trip condition.

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

10. The failure of one power trip relay in the power supply circuit has no effect since either of the two relays in series will provide the necessary action.

11. Single grounds in the power trip relay circuits have no effect since the circuit is ungrounded by a local isolating transformer. Local ground detectors also indicate should an accidental ground occur. Double grounds would cause the circuits to fail in the safe direction.

12. The ac circuit supplying power to the clutch power supplies is fed from an isolation transformer with a center tap grounded through a resistance to assure that single grounds will not prevent system action. The circuit has a local ground

detection system. The supply transformer center tap ground is used in a ground detection system to identify local ground and shorts around the power trip relay contacts when the relays are de-energized.

13. The dc clutch power supply circuits operate ungrounded so that single grounds have no effect. The clutches are supplied in two groups by separate pairs of power supplies to further reduce the possibility of clutches being improperly held. The clutch impedances and load requirements are such that the application of any other local available voltage will not prevent clutch release, eg, connection of the clutch supply circuit to the battery distribution circuit would cause the distribution circuit fuse to blow due to excessive current drain. Connection to a 115 volt ac circuit would have similar effects. Connection to available low-voltage dc, such as the nuclear instrumentation power supplies, would have no effect since these power supplies have insufficient capacity to supply the load.

<u>Diversity of Trip Functions</u>

Several Reactor Protective System functions back each other up for tripping the reactor. An example is high pressurizer pressure and thermal margin/ low-pressure trips.

Because the Technical Specifications permit operation with only two operable channels (if one of the inoperable channels is tripped), there are several scenarios in which the high-pressure trip could fail. The most simple of these is to fail the 125v DC bus which is assumed to be common to the two preferred AC buses that feed the remaining channels. Alternatively, assuming one of the four Reactor Protective System channels is bypassed but not tripped (operating in a two-out-of-three logic arrangement for reactor trip) and assuming the single failure of one 125v DC bus causes failure of two preferred AC buses that are needed to generate two high-pressure trip signals, the high pressurizer pressure reactor trip would not occur if required. However, the failure of two pressurizer pressure channels would also cause a thermal margin/low-pressure trip and thus protection of the reactor would still be maintained.

Other diversity features are described in Reference 1 and protect against common mode failures affecting similar components. In addition to the RPS diversities described in Reference 1 a diverse reactor trip based on pressurizer high pressure 2 out of 4 logic was added during the 1990 refueling outage as part of the ATWS rule (10CFR50.62).

7.2.8 POWER SOURCES

The power for the Reactor Protective System is supplied from four separate independent preferred ac buses. Each preferred bus is supplied from the battery system through an inverter to assure an uninterrupted, transient-free source of power (refer to Section 8.4).

Each preferred bus also has provision for connection to an instrument ac bus to permit servicing of the inverters.

The distribution circuits to the preferred buses are provided with circuit protective devices to assure that individual circuit faults are isolated.

For additional information on power sources for the Reactor Protective System, see Chapter 8.

7.2.9 PHYSICAL SEPARATION AND ELECTRICAL ISOLATION

7.2.9.1 Physical Separation

The sensors for the Reactor Protective System are located in the air room inside the containment building and physically separated to assure channel separation. The air room permits access to the sensors while the Plant is at power. The process transmitters located inside the containment, which are required for short-term operation following a design basis accident (DBA), are rated and have been tested under simulated DBA conditions (see Subsection 6.1.3.2.3) and have been tested for the safe shutdown earthquake (see Section 5.7). On June 21, 1991 the NRC granted exemption from requirements of Appendix R Section III.G.2.d which require 20 ft. separation for redundant instrumentation. This was in response to a CPCo analysis that the underlying purpose of Appendix R was met by considering the particular design and circumstances to the containment air room. The routing of cables from these transmitters is arranged so that the cables are separated from each other and from power cabling to reduce the danger of common event failures. This includes separation at the containment penetration areas. More detail on cable installation is discussed in Section 8.5. 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 reduce the possibility of common event failure and meet the intent of IEEE 384-1977 for interfaces with nonsafety-related systems as detailed in the following subsection.

7.2.9.2 Electrical Isolation

This subsection evaluates the adequacy of the Reactor Protective System electrical isolation from interfacing nonsafety-related systems. Such interfaces are intended to meet IEEE 384-1977 and 10 CFR 50 Appendix A, General Design Criterion 24, "Separation of Protection and Control Systems."

RPS interfaces are outlined in Figure 7-12. The plant data logger inputs are located in the data logger field remote station (FRS) in the cable spreading room (see Subsection 7.6.2.5). The critical functions monitor input cabinets and the Plant information processor (computer) are located in the main control room area (see Subsections 7.6.2.6 and 7.6.2.3, respectively). Other RPS outputs to nonsafety-related circuits (meters, recorders) go to control boards in the main control room (see

Subsection 7.6.2). Four basic types of isolation devices are used for isolation: optical isolators, thermistors/Zener diodes, operational amplifiers and resistors.

1. Analog RPS Outputs to Data Logger

Isolation for the RPS analog signals in the data logger is achieved by I k ohm thermistors except for neutron flux safety channels inputs which utilize operational amplifiers as isolation buffers. See Figure 7-12 for analog signals involved.

RPS analog signals connect to input conditioning cards in the FRS intracabinet assembly. The input conditioning cards include two 1 k ohm positive temperature coefficient thermistors and Zener diodes for isolation. The isolation method is considered adequate because:

a. There are four channels for each of the analog signals. If an isolation device fails and takes out one channel, the RPS will still operate properly on two-out-of-three logic.

b. All the analog signals interfacing with the data logger are on Channel A. Therefore, it is highly unlikely that the data logger could cause interchannel failures by tying two different channels together to completely eliminate the voting logic and thus inhibit a signal from reaching the RPS.

c. Each channel's power supply bus is alarmed to indicate if a problem exists on a particular channel so that it can be isolated and analyzed.

2. Digital RPS Outputs to Data Logger

Isolation for the RPS digital signals in the data logger is achieved by 36 k ohm input resistors and optical isolators. The RPS digital output signals to the data logger are by relay contact.

3. Analog RPS Outputs to Plant Information Processor

The neutron flux start-up rate of change of power is fed to the plant information processor (computer) via an operational amplifier as isolation buffer. The operational amplifier includes common mode rejection and fused output for protection against high voltage or excessive load. As such, it is considered adequate as an isolation device.

Analog outputs to the Plant information processor use 100 k ohm resistors as an isolation device. Engineering analyses, conducted in accordance with IEEE 384-1977, show that these resistors are adequate to protect the RPS from any short circuit, open circuit and application of the maximum voltage present on the nonsafety-related side of the resistors. As such, these resistors are qualified as isolation devices.

4. Analog RPS Outputs to Critical Function Monitor

The upper and lower power range neutron flux signals are fed to the Critical Function Monitor (CFM) via an operational amplifier as an isolation buffer. The operational amplifier is the same device as described in the above section. This signal is also isolated at the CFM input at the signal termination isolation cabinets which have qualified isolation devices per IEEE 384-1977.

5. Other RPS Outputs

Circuits other than those identified on Figure 7-12 (and other than the CFM) go to remote meters and auxiliary circuits from the power range safety channels and are isolated by operational amplifiers with 10 k ohm resistors at the inverting and noninverting inputs.

7.2.10 REACTOR TRIP AND PRETRIP SET POINTS

The Reactor Protective System trip set points are provided in the Technical Specifications; pretrip set points are determined by operating experience and are controlled administratively.

7.3 ENGINEERED SAFEGUARDS CONTROLS

7.3.1 INTRODUCTION

The engineered safeguards controls consist of equipment to monitor and select the available power sources and to initiate operation of certain load groups, and will initiate containment isolation when required. The system is designed on a two-independent-channel basis with each channel capable of initiating the safeguards equipment load groups to meet the minimum requirements to safely shut down the reactor and provide all functions necessary to operate the system associated with the Plant's capability to cope with abnormal events. The system is provided with the necessary redundant circuitry and physical isolation so that a single failure within the system will not prevent the proper system action when required in accordance with IEEE 279-1971 and 10 CFR 50, Appendix A, General Design Criteria 22 and 24. The control equipment is designed to withstand seismic loads described in Section 5.7 and is designed for Class 1E service. The system is provided with test facilities and alarms to alert the operator when certain components trip or malfunction or are not available or operable. The controls are interlocked to automatically provide the sequence of operations required to initiate engineered safeguards system operation with or without offsite power.

Certain critical parameters have four sensors utilizing a two-out-of-four logic to provide reliable operation with a minimum of spurious activations. Initiation level settings and their bases are provided in the Technical Specifications. The four sensors are physically isolated and operation of any two out of four will initiate the appropriate engineered safeguards action. This action is provided by combining the four sensors into relay matrices which provide dual-channel initiation signals. Actuation Channel A has all odd numbered relays. Channel B has all even numbered relays. Channel A receives its power from preferred ac Panel Y10; Channel B from Panel Y20 (see Section 8.3). Instruments Channel C has odd numbered devices and Instruments Channel D has even numbered devices. Channel C receives its power from preferred ac power Panel Y30; Channel D from Panel Y40. Physical separation between channels is maintained in the control panels to meet the intent of IEEE 384-1977 by locating devices in individual groups and providing barriers between groups. The cables for the two groups of channels (A/C and B/D) are run in separate raceways (see Section 8.5 for odd/even, left/right channels separation criteria).

Testing of major portions of the engineered safeguards control circuits is accomplished while the Plant is at power. More extensive circuit sequence and load testing may be done with the reactor shut down. The test circuits are designed to test the redundant circuits separately so that the correct operation of each circuit may be verified by either equipment operation or by sequence lights. The test circuit design is such that, should an accident occur while testing is in progress, the test will not interfere with initiation of the safeguards equipment required.

7.3.2 SAFETY INJECTION SYSTEM CONTROL CIRCUITS AND EQUIPMENT INITIATION

7.3.2.1 Design Basis

The control system is designed to automatically initiate the necessary engineered safeguards equipment upon a safety injection signal (SIS) with or without offsite power available. To assure reliability, the control system is designed on a two-channel concept with each channel initiating the operation of separate and redundant engineered safeguards load groups. The control system will function at all times and will operate in any mode of reactor operation.

7.3.2.2 Description and Operation

Description - The Safety Injection System logic is shown on Figures 7-13 and 8-17 and control schematics are shown on Figures 7-15 through 7-23.

Two independent and isolated circuits each initiate operation of redundant engineered safeguards equipment. These control circuits monitor whether offsite and/or emergency power is available and select load groups in accordance with the available power supply.

The safety injection signal is derived from pressurizer low-low pressure or containment high pressure. The pressurizer low-low pressure signal is derived from four pressure sensors installed on the pressurizer. Each sensor supplies a pressurizer pressure signal to a pressure indicator/alarm instrument (Figure 7-15). Each pressure instrument is connected to a latching-type auxiliary relay. The containment pressure signal is derived from four containment pressure sensors. Each containment pressure sensor is connected to a latching-type auxiliary relay. One pressure sensor and associated pressure instrument, as well as one containment pressure sensor, are supplied from each of the four preferred ac sources.

Either two out of four pressurizer low-low pressure or two out of four containment high-pressure signals initiate the SIS signal which, in turn, actuates two safety injection control circuits, each of which is supplied by a separate preferred ac source.

Within each control circuit, relays are provided to initiate redundant devices so that individual relay failure will not cause a complete circuit failure. Actuation of each safety injection control circuit can be performed manually via a safety injection initiate push button, one push button for each safety injection circuit. The SIS relay logic circuits control the loading sequence in duplicate control circuits. Failure of the control power on any one redundant circuit will be annunciated in the control room.

Containment spray activation requires the containment high-pressure signal to ensure the containment is sprayed only when needed (see Subsection 7.3.3).

If an SIS is accompanied by a loss of offsite power, the load sequencers will be initiated. There are two of these sequencers with each connected to a separate control circuit. The sequencers load the required equipment in sequence on the emergency generators so as to not exceed the emergency generators' capacity (see Section 8.4).

Operation - These circuits are safeguards circuits and operate only during shutdown or accident conditions. They have no function while the Plant is under normal operation. The shutdown sequence will vary depending on the presence or absence of an SIS signal and offsite power availability.

Shutdown Upon a Reactor Trip With Offsite Power Available - If no SIS condition exists at the time of the reactor trip, all auxiliary equipment will continue to operate from the offsite power source. Plant shutdown will be performed as necessary by the operator.

Shutdown Upon a Reactor Trip Without Offsite Power - Upon loss of offsite power during normal operation, each emergency generator will be started through its own separate control circuit. The emergency generator start is dependent upon undervoltage on the engineered safeguard buses (see Section 8.6). The bus loads will be shed by the pre-diesel load shedding relays. When the pre-diesel load shed relays have operated and the emergency generator voltage reaches a preset value, the buses will then be energized from the emergency generators. The sequencers will be energized to automatically start required normal shutdown equipment.

Safety Injection With Offsite Power Available - If offsite power is available at the time of initiation of the SIS, the SIS relays will initiate the simultaneous start of the engineered safeguards equipment.

Safety Injection Without Offsite Power - If offsite power fails, all loads will be shed at the time the diesel generators receive an automatic start signal. With load shedding completed, the diesel generator breakers will close automatically when generator voltage approaches a normal operating value. Closing of the breakers will reset the load shedding signals and start the sequencers. The sequencers will initiate operation of the engineered safeguards equipment required for design basis accident response.

Safety Injection System Block During Normal Shutdown - In order to avoid initiating operation of SIS equipment when the Primary Coolant System is depressurized, the SIS circuits must be blocked. Blocking is manual and is effective only when three of the four pressurizer pressure sensors are between the low-pressure and the low-low-pressure set points.

After the Primary Coolant System is placed back in service and the pressurizer pressure is restored to normal, the safety injection circuit block will be automatically reset when two or more of the four pressurizer pressure sensors detect normal operating pressure.

Testing - The availability of the control circuits may be tested at any time. Testing of these circuits will initiate the safeguards equipment unless their operation would adversely affect the normal Plant operation. A test sequence light indicator is provided to show that the initiating signal has energized the control circuit of the specific engineered safeguards equipment where either operation is not desirable during the test or equipment is already in operation. See Subsection 7.3.5 for further details on testing of the SIS.

7.3.2.3 Design Analysis

Reliability of the Safety Injection System control circuits is assured by redundant and diverse circuits, each initiating operation of redundant load groups. Diversity of initiation is provided by using two physically independent and diverse parameters (pressurizer and containment pressure) sensing an LOCA to activate SIS. Failure of control power to any one of the pressurizer low-pressure or containment high-pressure circuits will cause the circuit to fail in an SIS initiation signal mode. Failure of control power on any one redundant circuit will be annunciated.

7.3.3 CONTAINMENT HIGH PRESSURE AND HIGH RADIATION

7.3.3.1 Design Basis

The containment isolation control system is designed to isolate the containment upon occurrence of either containment high pressure or containment high radiation. The Containment Spray System is initiated upon containment high-pressure signal. The system is also designed to prevent inadvertent opening of the containment isolation valves.

The control system is designed on a two-channel concept with redundancy and physical separation. Each channel is capable of initiating containment isolation and operation of certain engineered safeguards.

7.3.3.2 Description and Operation

Description - The containment high-pressure and radiation control logic is shown on Figures 7-13, 7-24, 7-25 and the schematic on Figures 7-26 through 7-29.

One containment radiation monitor is located adjacent to each containment air cooler where radioiodines would condense along with water vapor in the event of relatively minor breaches of primary system integrity. Radiation monitor locations in the lower level of containment also allow for response from abnormal sump or 590-foot level water accumulations of radioactive coolant. Such conditions could be hypothesized for rupture of a letdown line after cooling by the letdown heat exchanger, leakage or overflow of the primary system drain tank, etc. In order to ensure ability to isolate containment on the premise that significant core damage might occur without loss of coolant, the location of two of the radiation monitors is also in the direct path of radiation emanating from abnormal concentrations of fission products passing through the letdown heat exchanger.

The sensors for containment pressure are located in the auxiliary building next to the containment building and in separate rooms to ensure channel separation.

Refer to Chapter 8, Subsection 8.1.3 for details of environment qualifications for the containment isolation instrumentation channels.

The controls consist of two independent and isolated groups of circuits. The four radiation sensors and four pressure sensors are each connected to an auxiliary relay. Four separate control circuits each consisting of one pressure and one radiation level sensor and their two auxiliary relays are connected to separate preferred ac buses. There are two separate initiation circuits which consist of two-out-of-four logic matrices and necessary auxiliary relays.

The containment isolation valves operate from the 125 volt dc source and are normally energized. It requires two high-radiation or two high-pressure signals to close the isolation valves. This prevents spurious signals from causing containment isolation.

Refueling accident high-radiation monitors are also provided in each of the two initiation circuits. These initiation circuits function on a one-out-of-two trip logic. Key switches are provided to lock in the monitors during refueling operations.

Operation - Coincident two-out-of-four high-radiation or two-out-of-four high-containment pressure signals from the auxiliary relays will trigger an alarm in the main control room, close all containment isolation valves not required for engineered safeguards except the component cooling line valves which are closed only by containment high pressure, and will isolate the control room ventilation system (see Section 9.8). High radiation detected by the refueling accident high-radiation monitors will also close all containment isolation valves not required for engineered safeguards when locked in by the respective refueling monitor key switches.

Coincident two-out-of-four high-radiation or two-out-of-four high-containment pressure signals from the auxiliary relays locks in the high-pressure and high-radiation circuits, respectively.

In order to deisolate the containment, the high-pressure and high-radiation circuits must be manually reset. At least three out of four pressure sensors must sense normal pressure, three out of four radiation sensors must sense normal radiation level or the refueling accident high-radiation monitors (when locked in) must sense normal radiation level before the operator can reset the pressure isolation circuits or the radiation isolation circuits. In accordance with NUREG-0578/0737, resetting the isolation circuits will not result in automatically opening the containment isolation valves; the operator must manually reopen each valve by placing each valve's hand switch from the "Open" position to the "Close" position and back to the "Open" position again. Resetting containment high-pressure will result in the component cooling line valves to reopen.

Containment high-pressure signal will initiate SIS, and start containment spray.

Containment high-pressure signal will also initiate a reactor trip with a two-out-of-four logic. This trip is in addition to the thermal margin/low-pressure trip (see Subsection 7.2.3.5) to ensure that the reactor is tripped before SIS and containment spray is initiated.

Containment high-pressure (CHP) signal will initiate closure of the main steam isolation valves to reduce the inventory blowdown from the intact steam generator in the case of a main steam line break, reducing the peak containment pressure and temperature as required in the accident analysis. CHP also closes the main and bypass feedwater regulating valves (Modification FC-906-1990). See Subsection 7.5.1.3.

Testing - The containment high-pressure detectors and auxiliary relays can be tested at power without actuating containment isolation by tripping one out of the four local pressure switches. Actuation of the auxiliary relay is annunciated in the control room. The detectors and auxiliary relays for containment high radiation are tested at power by using an internal test feature of the radiation monitor.

Testing the containment isolation circuits is done only during shutdown. RO-12, "Containment High Pressure (CHP) and Spray System Tests," functionally tests the CHP circuitry. Pressure is applied to two of the CHP pressure switches to simulate containment high pressure. All twelve 2/4 combinations (six left channel and six right channel) are tested separately. For the first test on each channel, proper operation is verified by verifying the following:

Containment Isolation Signal	-	Annunciator Lit
Containment Isolation	-	Valves Close
SIS Signal	-	Voltage Signal
Containment Spray System	-	Valves Open and Pumps Receive Start Signal
Control Room Ventilation	-	CR Air Filter Fans Start

The remaining five tests on each channel are performed in the same manner. However, only one component from each CHP relay is verified to operate properly. Actuation of containment isolation from the high-radiation channels is done by actuating the monitor by a radiation source thereby causing an actuation of the system. SIS is not actuated during this test. One of two redundant switches located in the control room, turned to test position, may be operated at a time to de-energize two of the four containment high-pressure channels which will cause containment isolation, initiate SIS, close feedwater regulating valves and start the containment spray pumps. The spray valves will not open in test position; if spray valves are not both closed fully, the spray pumps cannot be started in test position. The containment spray valves can be manually opened by means of their individual hand switches located in the control room. This second necessary manual operation to initiate the spray system will prevent inadvertent spraying inside containment with borated water.

Further information on testing of the above circuits is provided in Subsection 7.3.5.

7.3.3.3 Design Analysis

Reliability is assured by the redundancy and diversity in the initiating control circuits, in the design of the individual isolation valve control circuits and in the Containment Spray System and air cooler system control circuits.

The containment isolation signal receives diverse inputs from containment high-pressure and containment high-radiation signals. Diversity is ensured by the amount and location of containment radiation monitors ensuring effectiveness toward containment isolation in all postulated accident modes. This effectiveness has been verified by actual testing for the case of core damage assuming 1% failed fuel without LOCA (see Reference 2).

The containment isolation valves are closed following an isolation signal such that deliberate operator action is required to reopen the penetration (NUREG-0578, Paragraph 2.1.4).

The containment isolation signal is not received by essential systems. Essential systems are those critical to the immediate mitigation of any event that results in automatic containment isolation. Essential systems provide Primary Coolant System inventory and pressure control, reactivity control, core cooling, secondary heat sink, containment cooling (depressurization) and safe shutdown. Essential systems must be available in response to accident parameters without operator action and, therefore, should not be automatically isolated as part of the Containment Isolation System.

The essential systems are (Chapters 6, 9 and 10):

High-Pressure Safety Injection

Low-Pressure Safety Injection

Containment Spray (including recirculation)

Containment Critical Service Water

Charging

Auxiliary Feedwater

Main Steam

Note that main steam isolation will occur on containment high pressure or steam generator low pressure (see Subsection 7.2.3.8) based on containment pressurization considerations for a steam line break.

Nonessential systems receive the containment isolation signal, with two exceptions: instrument air and main feedwater. In both of these cases, isolation is affected in a way which is functionally equivalent to automatic containment isolation. Instrument air is at a higher pressure than containment pressure under LOCA conditions. Both instrument air and main feedwater have check valves prior to entering containment. Note that component cooling water provides cooling for primary coolant pump seals and isolates on containment high pressure only. Although not part of the isolation system, the feedwater regulating valves will also close on CHP due to a 1990 modification.

Failure in control source power to the pressure/radiation sensor relay circuit or to the redundant initiating circuit causes the circuit to fail in a mode to initiate isolation. Loss of control power will be annunciated in the control room, but isolation will not be effected unless a second failure occurs. Failure of source power to the control circuit of any isolation valve or failure of the pilot valve solenoid of an isolation valve will cause that isolation valve to close.

7.3.4 SAFETY INJECTION AND REFUELING WATER TANK LOW LEVEL

7.3.4.1 Design Basis

The SIRW tank low-level control system (recirculation actuation system (RAS)) is designed to transfer the suction of the safety injection and containment spray pumps to the containment sump when the SIRW tank is essentially empty, and to perform the functions required to recirculate and cool the water which has accumulated in the containment building sump, for post-accident cooling of the core. In the recirculation mode, it automatically provides component cooling water to the shell side of the shutdown cooling heat exchangers. The circuit is designed on a two-channel concept in accordance with IEEE 279-1971 with each channel initiating the operation of separate and redundant hydraulic loops.

7.3.4.2 Description and Operation

Description - The SIRW tank level control logic is shown on Figure 7-30 and the schematic is shown on Figures 7-26, 7-31 and 7-32.

The SIRW tank is provided with four level switches to detect low level with each connected to an auxiliary relay from separate preferred ac supplies.

A separate circuit is provided for control of the safety injection recirculation valves associated with one recirculation loop consistent with the two-channel concept. In addition, each circuit controls the operation of the component cooling water valves to each of the component cooling water heat exchangers and the service water valve for service water from each of the component cooling water heat exchangers.

Operation - The low-level control circuits have no normal or shutdown cooling operating function and will operate only after the SIRW tank has been essentially emptied. Coincident one-out-of-two (taken twice) low-level signals will automatically initiate the necessary valve operations to permit operation of the two recirculation loops and trip both low-pressure safety injection pumps to protect the pumps from low suction pressure. A manual bypass is provided so that the low-pressure safety injection pumps may be restarted if the operators deem this necessary for long-term core cooling.

A key switch is provided for each of the low-level control circuits. The switch contacts are in parallel with the SIRW tank low-level contacts, such that the minimum recirculation valves may be closed for chemical sampling without having low level in the SIRW tank. This operation is annunciated. The valves will not close if their control switch is in the "Open" position.

Testing - The RAS control circuit may be tested while the Plant is shut down. This test will initiate the operation of the valves and the trip signal of both LP safety injection pumps.

The test may be initiated by the test switches provided in the control room or can be initiated by actuating the level switches mounted at the SIRW tank. Operation of one of the two redundant three-position test switches on the control panel will de-energize two level switch auxiliary relay circuits and provide a one-out-of-two (taken twice) low-level signal which will initiate operation of the valves. This circuit will be maintained (sealed in) until the test switch is moved to the reset position. Releasing the test switch from reset will conclude the test and valve operators will return to the normal positions.

In addition, the valve operation may be manually tested individually with valve control switches.

Further information on SIRW tank low-level control circuits testing can be found in Subsection 7.3.5.

7.3.4.3 Design Analysis

Reliability of the low-level control (recirculation actuation system (RAS)) is assured by redundant control circuits, each controlling a redundant recirculation loop and cooling system valves. Each of the redundant control circuits is supplied from a separate preferred ac source. Failure of the power source in any one of the level switch circuits will cause the circuit to fail in a mode to initiate recirculation. Manual activation is also available from the control room using pump and valve control switches. Override of the RAS is annunciated.

7.3.5 ENGINEERED SAFEGUARDS TESTING

7.3.5.1 Design Bases

The engineered safeguards testing features, method and program meet 10 CFR 50, Appendix A, General Design Criterion 37; NRC Branch Technical Position ICSB 25; Regulatory Guide 1.22; and Standard Review Plan (NUREG-75/087).

Response time testing of engineered safeguards is consistent with the requirements of 10 CFR 50, Appendix A, General Design Criterion 21; Section 3.9 of IEEE 279-1971; and IEEE 338-1977.

The engineered safeguard functional testing program also demonstrates the full-functional operability and independence of the onsite power sources (see Section 8.4) and is performed during shutdown. This testing program simulates loss of offsite power in conjunction with a simulated safety injection actuation signal and simulates interruption and subsequent reconnection of onsite power sources. Steps are included to verify the proper operation of the load-shed system, load-shed bypass when the emergency diesel generators are supplying power to their respective buses and that there is no adverse interaction between the onsite and offsite power sources.

7.3.5.2 Testing Description

The details of the testing program, methods and acceptance criteria are provided in Appendix 7A.

7.4 OTHER SAFETY RELATED PROTECTION, CONTROL AND DISPLAY SYSTEMS

While the Reactor Protective System protects against reactor core damage and the engineered safeguards controls protect against a loss of coolant incident, other safety related (Class 1E service) control and instrumentation systems ensure a safe shutdown of the Plant, protection of primary coolant fluid boundaries, mitigation of anticipated events such as loss of feedwater and uncontrolled release of radioactive effluents. In addition, Plant parameters critical to safety are monitored with Class 1E instruments to ensure the operator can act in a timely fashion during abnormal conditions.

7.4.1 REACTOR SHUTDOWN CONTROLS

Refer to section 9.6.8.3 for information concerning reactor shutdown controls.

7.4.2 PRIMARY COOLANT BOUNDARIES PROTECTION

Leak detection from the Primary Coolant System is described in Chapter 4 and is considered as nonsafety related. The primary coolant safety valves are the protective devices with coolant at normal pressure. The following identifies and describes the safety related control and instrumentation provided to protect primary coolant fluid system boundaries during off-normal anticipated condition.

7.4.2.1 Primary Coolant Overpressure Protection System

1. Design Bases

   Without a low temperature overpressure protection system, pressure transients in the Primary Coolant System initiated while operating at low temperatures are not protected against and there are no pressure relief devices to prevent these transients from exceeding the Technical Specifications limits as required by 10 CFR 50, Appendix G pressure-temperature limits. The reactor has a pressure limit in excess of 2,500 psia above 430°F, but has a much lower limit at 200°F (see Technical Specifications limit in Section 3.1.2). The code safety valves with settings in the 2,500 psia range would not be able to relieve a pressure transient at low Primary Coolant System temperature without the limits of the Technical Specifications being violated.

   The Technical Specifications pressure limit drops off rapidly at lower temperatures because the reactor vessel material and welds have significantly less toughness at lower temperatures and are therefore more susceptible to flaw-induced failure. In addition, factors such as copper content in welds and neutron fluence levels affect the material toughness and contribute to the reduction in safety margin to vessel failure at low temperature conditions.

The Primary Coolant System Low temperature overpressurization subsystem (LTOP) has been designed to provide automatic pressure relief of the Primary Coolant System at temperatures lower than 435°F. This is not a set point, set points are defined in Figure 4-17.

In order to prevent overpressurization of the Shutdown Cooling System, the LTOP will be placed in service whenever this system is not isolated from the Primary Coolant System. This additional requirement precludes conditions that could lead to a loss of coolant incident outside containment as identified in Regulatory Guide 1.139.

It is also noted that, following a loss of offsite power, which is an anticipated operational occurrence (AOO), "feed and bleed" mode of Primary Coolant System operation may be used to remove decay heat from the reactor. In order to satisfy the basic requirements for safety equipment used in mitigation of AOOs, the OPS mechanical components, valves, etc, are CP Co Design Class 1.

2. <u>Design Description</u>

Two redundant and separate overpressurization protection channels are provided. Each channel consists of a CP Co Design Class 1 power-operated relief valve (PORV), a CP Co Design Class I PORV isolation (block) valve, valve actuators, position indications, a Class 1E primary coolant narrow range pressure sensor, and related instrumentation and controls. Pressure relief is accomplished by the automatic opening of the power-operated relief valves. Each PORV has sufficient capacity to protect the Primary Coolant System from overpressurization at lower temperatures. Limiting transients that each PORV is capable of handling are: (1) the start of an idle primary coolant pump when secondary water in the steam generator is up to 100°F hotter than the Primary Coolant System cold leg temperatures (refer to Technical Specifications Section 3.1.1.h); and (2) the start of an HPSI pump when the system is in the water solid condition.

Each pressure relief valve is blocked by an individual block valve. The PORVs are solenoid operated and designed to fail closed, while the block valves are motor operated (fail as is). Direct indication of PORV position is provided in the control room using acoustic sensors. The block valves, controlled by hand switches in the control room are also provided with a direct valve position indication. The PORVs block valves are closed in normal operation. The pressurizer safety valves are provided with similar acoustic sensors for direct valve position indication in the control room.

Power for the PORVs and their respective block valves is supplied by Class 1E MCC-1 for one channel and Class 1E MCC-2 for the other. All valves may be operated with either offsite or onsite power. The initiation channels are fed from redundant preferred ac power sources.

Three annunciators are provided for interface of the system with the operator. The first annunciator advises the operator of five conditions:

1. PCS temperature of 460°F with PORV block valves closed as a signal to arm the system

2. Loss of control power

3. Microprocessor failure

4. Inputs out of range

5. LTOP/SDC switch failure

The second signals an approaching high-pressure condition. The third annunciator advises the operator that PCS pressure has exceeded the trip setpoint as shown on Figure 4-17 and the PORVs should have opened.

A separate key-operated switch, concurrently with a switch opening the associated block valve, is provided for arming each channel. Indicator lights in each channel have been provided on the control room panel to inform the operator that (1) the block valves are open (white light), (2) the system is in the SDC mode (amber light), and (3) the overpressurization system has been activated (red light).

Testing - Refer to the Plant Technical Specifications.

3. Design Evaluation

The Primary Coolant System overpressure protection subsystem design meets the following requirements:

a. Operator Action - No credit is taken for operator action until 10 minutes after the operator is aware, through an activation alarm, that a pressure transient is in progress.

b. Single Failure Criterion - The primary coolant overpressure protection system is designed to protect the reactor vessel given a single failure in addition to the event that initiated the pressure transient. The power-operated relief valves and their associated block valves are powered from Class 1E 480 volt buses to allow emergency power for these valves.

c. Testability - The system is tested on a periodic basis per Technical Specifications.

d. Seismic and IEEE 279 Criteria - The electrical and instrumentation equipment meets IEEE 279-1971 criteria. The system is not vulnerable to a failure mode that would both initiate a pressure transient and disable the overpressure mitigating system. Such events as loss of instrument air and loss of offsite power have been considered.

e. Environment Qualification - The narrow range primary coolant pressure channels equipment is qualified to IEEE 323 including post-LOCA environment for equipment inside containment.

f. Operator Interfaces - The electrical instrumentation and control system provides a variety of alarms to alert the operator to (1) properly enable the low temperature overpressure protection system at the proper temperature during cooldown, (2) indicate if a pressure transient is occurring, and (3) indicate system problems due to loss of control power, microprocessor failure, inputs out of range and LTOP/SDC switch failure. .

g. Interlocks - Additionally, the electrical system provides positive assurance that the block valve upstream of each PORV is open when the system is enabled by wiring its position into the enable alarm. The enable alarm is not permitted to be activated until the temperature is less than 460°F.

7.4.2.2 <u>Other Primary Coolant Boundaries Protection</u>

1. <u>Primary Coolant vs LPSI System</u>

A high-low pressure interface, in which the low-pressure system must be isolated from the high-pressure system when the high-pressure system is at rated pressure, exists in the shutdown cooling line connecting the Primary Coolant System to the suction of the low-pressure safety injection pumps. Isolation is provided by two valves in series. Except for the areas of penetration into containment, the power and control circuits of both valves are routed through the same fire areas. To preclude the possibility of fire damage to the control circuitry of the shutdown cooling valves resulting in opening operations of both valves, an operating procedure is provided requiring the valves to be checked closed and the operating power disconnected from both before pressurizing the primary system greater than 260 psig.

2. Primary Coolant vs Gas Vent System

The primary coolant gas vent system allows the control room operator to remotely vent either the reactor vessel head space or the pressurizer vapor space (see supplemental details in Chapter 4). The vented vapor is released to the pressurizer quench tank for small quantities of vapor or to the containment atmosphere for large quantities. The vent isolation valves (remotely controlled) are series redundant with Class 1E controls and power including a separate key-operated control switch for each valve to ensure the primary coolant boundaries are maintained. The vent isolation valves are also parallel redundant to reduce the probability of a vent path failing to open since the valves are fail-close type in the event of loss of power. The valves and their controls are qualified to IEEE 332-1972 for inside containment devices, IEEE 344-1975 (seismic event) and IEEE 323-1974 (environmental and qualified life considerations) for all components. The isolation/opening controls meet IEEE 279-1971 including testability, using individual test and open/closed indication for each valve.

7.4.3 AUXILIARY FEEDWATER CONTROLS

The auxiliary feedwater (AFW) controls became Class 1E and automatic in response to NUREG-0578/0737. Alternate locations for controls were the result of the Systematic Evaluation Program of the Palisades Plant and compliance with Appendix R of 10 CFR 50.

7.4.3.1 Auxiliary Feedwater Initiation

1. Design Bases

Without an automatic initiation of the Auxiliary Feedwater System, the operator would have 17 minutes to restore flow to the steam generators upon loss of main feedwater to avoid opening of the primary coolant power-operated relief valves resulting in a small loss of primary coolant incident. Since the loss of main feedwater is considered as an anticipated operational occurrence, NUREG-0578 (Paragraph 2.1.7a) followed by NUREG-0737 (Section II.E.1.2) have required the installation of a Class 1E automatic initiation of the auxiliary feedwater pumps meeting IEEE 279-1971 criteria, assuring restoration of flow to the steam generators independent of operator action. Specific design criteria used in the design were to be as follows:

a. Delivery of AFW minimum required flow to the steam generators must occur within 200 seconds of sensor activation for AFW Pumps P-8A, P-8B or P-8C.

b. The AFW pumps must be able to be started manually with or without the auto-start signal present.

c. All circuits within the control system must meet IEEE 279-1971 including capability for on-line testing and a two-out-of-three (2/3) coincidence logic for low suction trip of the pumps.

d. The circuits and instruments utilized must be powered from Class 1E power sources that meet the power redundancy requirements.

e. 10 CFR 50, Appendix A, General Design Criteria 13, 20 and 34 are relevant to this system.

f. 10 CFR 50, Appendix R, and 10 CFR 50.48 are to be met.

Review of the above criteria centered on potential common mode failures associated with the location of the existing AFW pumps (as of 1981) in a single room. Design modifications were performed in 1983 to incorporate a third AFW pump from the original design of two (one electric and one turbine driven) in a location physically separate from the other two pumps.

During the 1990 refueling outage, to meet the criteria defined in 10CFR50.62, the ATWS rule, an auto start of the turbine driven auxiliary feedwater pump on loss of dc control power was added.

2. <u>Design Description</u>

The Auxiliary Feedwater System utilizes two motor-driven pumps and one turbine-driven pump to feed the steam generators (Chapter 10). One or more of these pumps are started manually when required. In addition, one of the motor-driven pumps is automatically started upon low steam generator level. The second motor-driven pump is started in the case of failure of the first pump. If these pumps fail to establish flow or are tripped for any reason, the turbine-driven pump is then automatically started. Refer to the logic symbols of Figures 7-33 through 7-35 for reading the figures corresponding to this subsection.

Motor-Driven AFW Pumps Controls - Loss of main feedwater to the steam generators is indicated and annunciated in the control room by one of the following:

a. Closing of both the high- and low-pressure trip valves on each of the main feedwater pump drive turbines

b. Low main feedwater flow to the main feedwater pumps

c. Low steam generator level (pretrip)

If no action by the operator is performed, low-level signals from two-out-of-four (2/4) steam generator level sensors on an OR logic between the two steam generators energize a timer relay if the motor-driven AFW pumps mode selector switches are in the "Auto" position (Figures 7-36 through 7-39). Upon completion of the timing cycle, the timer contacts actuate closing of the motor-driven AFW Pump A circuit breaker provided offsite or onsite standby (emergency generator) power is available. If offsite power is unavailable, the auto start is blocked until the normal shutdown or DBA sequencer allows loading the pump motor onto the emergency generator. The motor-driven AFW Pump C is called to start if Pump A circuit breaker trips or AFW flow does not materialize. The AFW low flow initiation logic is one-out-of-two (1/2) derived from redundant Class 1E voltage-type flow switches (Figures 7-43 and 7-44).

The 2/4 initiation logic is derived from current-type level switches installed in the redundant Class 1E Reactor Protective System steam generator level input channels (Section 7.2 and Figure 7-36). The mode selector switch position off the "Auto" position is indicated as an off-normal condition on the main control panel. The actuation timer is provided with a suitable time setting to block unwanted automatic starting due to normal transients in the steam generator level.

Manual trip of the pump can be accomplished by the circuit breaker control switch regardless of the presence of the automatic start signal. Automatic tripping will occur on pump low suction pressure via a two-out-of-three (2/3) logic from redundant Class 1E pressure sensors on the pump suction (Figures 7-41 and 7-42), or upon operation of the pump circuit breaker electrical protection (overcurrent/ground fault), or upon a load shed signal (Section 8.6). These trip functions are in force regardless of the presence of the automatic start signal. The 2/3 logic output will be sealed in, and the seal-in will be removed by manual closing of the pump circuit breaker.

Turbine-Driven AFW Pump Controls - A similar automatic start system is provided for this pump except that the timer for this system has a longer time delay setting to ensure the motor-driven pumps have had a chance to start and establish the auxiliary feedwater flow (Figures 7-36, 7-37 and 7-40).

An occurrence of low flow in either AFW line to the steam generators or motor-driven Pump C trip, together with a time delayed auto-start signal for the turbine-driven pump will open the turbine drive Steam Supply Valve B and start the pump. The steam supply valve control switch must be in the "Auto" position for the auto-start signal to be effective.

The AFW low flow initiation logic is one-out-of-two (1/2) derived from redundant Class 1E voltage-type flow switches (Figures 7-43 and 7-44). The Steam Inlet Valve B control switch position off the "Auto" position is indicated as an off-normal condition on the main control panel.

In the case of a successful start of one of the motor-driven pumps, the start of the turbine-driven pump is overridden by the reset of the AFW discharge line flow switches on a two-out-of-two (2/2) basis.

Manual starting of the turbine-driven pump can be accomplished at any time by its steam inlet valve control switch "Open" position. Manual trip can also be accomplished at any time by the same switch in the "Close" position. The control switch position in "Close" is indicated on the main control panel as an off-normal condition.

Automatic tripping of the pump will occur on pump low suction pressure with the same circuit as the motor-driven pumps via a two-out-of-three logic. This trip function is in force regardless of the presence of the automatic start signal. The trip function is not in force for operation of the valve from the C150 panel or upon loss of DC control power to panel D11 (ATWS). The 2/3 logic output will be sealed in, and the seal-in will be removed by manual closing of the pump turbine Steam Supply Valve B. Steam supply valve A is also tripped on pump low suction pressure. This is a manually opened valve only. The 2/3 logic output will also be sealed in and the seal-in will be removed by manually closing the pump turbine steam inlet valve A.

A diverse system has been added to automatically start the turbine driven pump on loss of dc control power to panel D11 (ATWS system, 10CFR50.62). This automatic start is annunciated on the control room panel C06 (ATWS TROUBLE/TRIP).

Operation - The AFW automatic initiation system is placed in operation when the Primary Coolant System is heated above 300°F. Operation of the system is normally from the control room; if the control room becomes unavailable, manual controls can be taken over from the Auxiliary Shutdown Control Panel C-150 (see Subsections 7.4.1 and 7.7.4). The AFW automatic initiation system status is annunciated on the main control board (Figures 7-52 and 7-53).

Testing - Both the motor-driven and the turbine-driven AFW pumps' automatic initiation circuits are individually tested from the main control room according to Technical Specifications requirements. After a start test switch (one for each pump) has been turned to the "Test" position, a white light indicates the test status and the test signal passes through the automatic initiation circuit, including the timers, and starts the applicable pump. The test signal is sent into the circuit after the steam generator low-level signals logic, and in the case of the turbine-driven pump, after the AFW low-flow signals.

The steam generator low-level signals logic (Figure 7-37) is provided with test push buttons for test of the coincidence logic and bistable trip modules.

The entire automatic initiation circuit is tested on line and the pumps themselves are tested on line by the test switches. Water is delivered to the steam generators during the test, thus checking the suction pressure and discharge flow switches operation. The steam generator level signals are checked as a part of the Reactor Protective System input channels test.

3. Design Evaluation

The automatic AFW initiation system meets all the criteria of NUREG-0578/0737 and IEEE 279-1971.

The automatic start signal originates from independent steam generators low-level switches on a 2/4 logic from either steam generator. These two logic signals are independent and diverse. All components of the initiating logics are Class 1E. Initiating signals and circuits are powered from Class 1E preferred ac or 125 volt dc sources. The motor-driven Pump A circuit and the turbine-driven pump are considered as a "left" channel while the motor-driven Pump C is considered a "right" channel. One low steam generator level logic circuit is powered from the "left" channel preferred ac power supplies while the other logic circuit is powered from the "right" channel. Loss of power to either logic circuit does not result in AFW pump start and leaves one circuit available for activation. Loss of one 125 volt dc power to a pump control circuit channel leaves one pump start available. Electric power for motor-driven Pump A is supplied by 2,400 volt Bus 1C while Pump C power is from 2,400 volt Bus 1D, respectively, left and right channels. The electrical equipment and instrument channels are then powered from emergency buses which meet the power redundancy requirements set forth in NRC BTP ASB 10-1.

The two-out-of-three low-suction pressure pump trip eliminates the possibility of an incorrect trip of both pumps due to failure of a single pressure switch. These trip logic components are themselves Class 1E and the three sensing channels meet IEEE 279-1971.

Automatic/manual initiation requirements of IEEE 279-1971 are met as described earlier including ability to start the pumps regardless of auto-start system status and testability.

The ATWS system for automatic start of the turbine driven pump is a non-1E system as defined by 10CFR50.62. However the equipment in the area of CV-0522B was purchased and installed as 1E equipment and it was mounted seismically.

The motor-driven AFW pumps are redundant to each other and to the turbine-driven pump to the extent that each pump is capable of supplying the auxiliary feedwater flow requirements to remove decay heat from the primary system and maintain the reactor in a safe condition.

The reliability of the system has been verified by a failure mode and effects analysis (see Reference 4). In addition, manual controls located outside the control room, in accordance with 10 CFR 50, Appendix R, and 10 CFR 50.48, have been provided.

In the case of offsite power loss as sensed by auxiliary contacts from the offsite power incoming circuit breakers (both open), the auto start of the motor-driven AFW pumps are included in the emergency generator's DBA and normal shutdown sequence. The turbine-driven pump will also be available in this case since all support equipment is powered directly or indirectly from the station batteries.

Instrumentation and control devices are qualified to IEEE 344-1975 and Regulatory Guide 1.100 seismic design requirements (deviations from the seismic design class are presented in Reference 7) and to IEEE 323-1974 and Regulatory Guide 1.89 environment and life qualifications requirements. Compliance with IEEE 279-1971 is summarized as follows:

a. No single component failure will prevent the auto-start signal from being initiated for at least one AFW pump.

b. Four independent steam generator level measurement channels complete with sensors, sensor power supply units, indicators and level switches are provided for each steam generator (diverse parameters).

c. The channels are provided with a high degree of independence by separate connections of the sensors to the process system and of the channels to preferred power supply buses. Separate raceways, conduits and junction boxes are used to ensure independence from cable faults.

d. The four normal (narrow-range) steam generator level measurement channels provide initiation signals to four independent initiation paths.

e. Elimination of spurious activation is provided by the 2/4 logic and the built-in time delays for actuation.

f. When one of the four channels is taken out of service for maintenance, the initiation logic reverts to either a 2/3 coincidence or a 1/3 coincidence.

g. The auto-start system instrument channels power is supplied from four separate buses.

h. Open circuitry or loss of power supply of one of the instrument channels initiates an alarm and a channel activation.

i. Loss of 125 volt dc power from one dc power source disables only one AFW pump auto start channel and this loss of power is alarmed.

7.4.3.2 Auxiliary Feedwater Flow Controls and Isolation

1. Design Bases

Reliable AFW flow and steam generator level instrumentation are necessary in order to adequately determine and control, from the control room or alternate shutdown stations, the performance of the Auxiliary Feedwater System since the operation of this system is considered as an anticipated operational occurrence by 10 CFR 50, Appendix A, GDC 13. NUREG-0578 and 0737 have identified the criteria to be met by this instrumentation and are outlined below.

In the event of a main steam line break inside containment, the AFW flow toward the affected steam generator must be terminated. This function must be performed using isolation valves in each steam generator's AFW supply line. The valves are isolated manually based on a low water level in one steam generator and excessive pressure differential between steam generators. The operator is instructed on manual steam generator isolation in the Emergency Operating Procedures.

Redundant Class 1E environmentally and seismically qualified, instrumentation channels' components have to be provided. The redundancy must meet IEEE 279-1971 criteria. The component qualification must meet IEEE 323-1974 and IEEE 344-1975 criteria. Power supplies must come from emergency sources.

2. <u>Design Description</u>

Four Class 1E AFW flow control channels are provided, fed from separate preferred ac sources, two for each steam generator (Figures 7-43 and 7-44). Two flow control channels relate to motor-driven AFW Pump A and turbine-driven Pump B, while the other channels relate to motor-driven Pump C. In each flow control channel, a flow indicating controller maintains constant flow rates to the applicable steam generator and provides flow indication in the main control room. The four channels are physically and electrically isolated including fire barriers according to IEEE 384-1977. The channels' components are qualified to IEEE 323-1974 and IEEE 344-1975.

The AFW flow control valves activated by these channels are CP Co Design Class 1. The flow control valves corresponding to operation with the turbine-driven AFW pump have a 8-hour motive nitrogen supply.

For FIC-0727 and FIC-0749, the flow controllers keep the control valves shut until one of the AFW pumps is started (Figures 7-46 and 7-47). This is accomplished using two flow set points on the controllers, one for shutdown (valve closed) and one for operation (valve opened for predetermined flow). Set point switching is provided by the motor-driven pumps' circuit breaker auxiliary contacts and the turbine-driven pump steam admission valve controls auxiliary contacts on an OR logic basis. This design allows timely and smooth opening of the AFW flow control valves without operator intervention.

For FIC-0736A and FIC-0737A, the flow controllers keep the control valves shut until an AFAS signal is received. This is accomplished by the program in the controller. The program looks for a pump start signal and an AFAS signal before automatically opening the valves. This design allows timely and smooth opening of the AFW flow control valves without operator intervention.

A separate Class 1E AFW flow indication channel for each AFW flow path (Figure 7-45) and a wide-range steam generator level indication channel for each steam generator are also provided allowing indication of flow independent from the control channel and monitoring of steam generator water level to cover all anticipated transients. These indication channels are qualified in the same way as the control channels and are also fed from preferred ac sources.

A "feed-only-good-generator" (FOGG) logic circuit (Figure 7-49) monitors the pressure differential between the steam generators using four independent and redundant Class 1E pressure sensors on each steam generator. These pressure sensors are also used by the Reactor Protective System and main steam isolation circuits (refer to Subsection 7.2.3.8). Concurrent excessive differential pressure between steam generators and low level in the depressurized steam generator initiates isolation of the depressurized steam generator by closing corresponding motor-operated isolation valves in the AFW supply lines (Figures 7-50 and 7-51). Two-out-of-four (2/4) differential pressure logic is used in coincidence with the output of the steam generator low-level logic described in Subsection 7.4.3.1. The isolation signal is generated through electronic bistable modules. Due to nuclear safety considerations, the automatic isolation feature of the FOGG system has been disabled and the operator is instructed by Plant Emergency Operating Procedures to isolate the affected steam generator using the flow control valves (see Section 9.7.2). The FOGG MOVs are de-energized and locked open.

The normally de-energized motor-operated automatic isolation valves are supplied from Class 1E 480 volt motor control centers. One isolation valve from each of the four discharge headers to the steam generators is supplied from the left channel of power and the other from the right channel to meet the single failure criterion.

<u>Operation</u> - Auxiliary feedwater flow indication, controls and isolation are normally from the main control room. In the event the control room must be evacuated, indication and controls can be taken over from either the Engineered Safeguards Auxiliary Panel C-33 or from the Auxiliary Hot Shutdown Control Panel C-150, depending on the nature of the emergency (see Subsections 7.4.1, 7.7.3 and 7.7.4). The controls at the alternate locations are manual. The locked open, de-energized motor-operated isolation valves are controlled locally at the valve by the operator.

If the pressure differential between the steam generators reaches the set point for actuation of FOGG circuit and the water level is low in one of the steam generators, the redundant FOGG signals close the motor-operated isolation valves to the depressurized steam generator. Due to nuclear safety considerations, the automatic isolation feature of the FOGG system has been disabled and the operator is instructed by Plant Emergency Operating Procedures to isolate the affected steam generator using the flow control valves (see Section 9.7.2). Isolation valve status is monitored in the control room. The

FOGG MOVs are de-energized and locked open, with control room position indication de-energized. The FOGG circuit status is annunciated on the main control board (Figures 7-52 and 7-53).

Testing - Testing of the flow control instrumentation is provided by actual system functional testing since the Auxiliary Feedwater System is used during normal Plant evolutions.

The FOGG logic circuit including bistable isolation modules is provided with test push buttons for test of the coincidence logic and isolation modules for on-line testing.

3. Design Evaluation

The performance of the AFW system can be assessed by the AFW flow indicators, two for each steam generator located in the control room and alternate stations outside the control room and a wide-range water level indicator for each steam generator. All components of the indication system are Class 1E, seismically and environmentally qualified, and as such exceed the requirements of NUREG-0578/0737.

The AFW flow control and isolation systems meet IEEE 279-1971 in the same fashion as the AFW initiation circuitry. Their adequacy is demonstrated by their components qualification as well as their redundancy features both with a dedicated channel activating a given AFW control or isolation valve and in location (control room versus alternate stations).

The automatic AFW flow control system limits the AFW flow to a predetermined amount. Testing has demonstrated that with flow well above the set point of the instruments, no water hammer will occur.

7.4.4 CONTAINMENT HYDROGEN CONTROLS

7.4.4.1 Design Basis

Hydrogen recombiners controls are Class 1E and meet IEEE 279-1971, 323-1974 and 344-1975. Both recombiners are designed to be manually turned on following an accident. Each recombiner has five banks of heaters powered from Class 1E Motor Control Center I or 2 and located inside the containment. Remote controls are located in the cable spreading room.

Hydrogen monitoring in containment is described in Section 9.9. This equipment has been qualified in accordance with Regulatory Guide 1.97, Category 1 requirements for post-accident monitoring service.

7.4.4.2 Description and Operation

Description - Containment hydrogen recombiners are described in Section 6.6.

Operation - The recombiners do not require any instrumentation inside the containment for proper operation after a Loss of Coolant Accident. This operation is assured by measuring the amount of electric power to the recombiner from the control panel. Calibration of the power consumption involves a correction factor for containment pressure and temperature to be used for operation after an accident. This correction factor is derived from manufacturer calibration curves. Presence of hydrogen above a given percentage as measured by the containment hydrogen monitor requires resetting the power output by a known value.

Testing - Testing of the recombiners controls is included in the Technical Specifications.

7.4.5 VENTILATION AND EFFLUENT RELEASES CONTROLS

Safety related automatic isolation controls of ventilation and effluent releases are summarized in this subsection. Details of the applicable control systems are given in Section 9.8 and Chapter 11.

7.4.5.1 Control Room

Control room ventilation is provided with an automatic isolation upon containment high-pressure or high-radiation signal (see Subsection 7.3.3) to protect the operating personnel from any radioactivity release during an accident. The control room ventilation controls are Class 1E and are fed from the preferred ac power sources. In the event of isolation, all the makeup air for the control room is drawn through a charcoal filter to provide equal to or greater than 0.125 inch of water positive pressure design in the air recycle to ensure no in-leakage of radioactivity.

7.4.5.2 Engineered Safeguards Pump Rooms

One radiation monitor is installed for each engineered safeguards pump room to provide a room isolation signal upon high radioactivity levels in the applicable room. The automatic isolation allows maintenance of acceptable dose levels at the site boundaries (Chapter 11). This system contributes to the leak detection capability described in Chapter 6.

7.4.5.3 Radwaste Area

One radiation monitor provides a ventilation shutdown signal for the area in the event of spillage. The area is maintained under negative pressure to prevent radioactive leakage out of the building with the supply fan and damper shut off while one of the exhaust fans is not shut off. Alarms in the control room warn the operator that one or more radwaste area ventilation fans have tripped off.

7.4.5.4 Fuel Handling Areas

The fuel handling areas are provided with radiation monitors to protect against radioactivity releases in the event of a fuel handling accident. Half of the ventilation is automatically shut off. The fuel handling building is then maintained under negative pressure to prevent leakage out of the building with a fan which is not shut off. Upon receiving an alarm from this monitor, the operator can manually trip the ventilation of the area.

7.4.5.5 Waste Gas Decay Tank

A high-radiation monitoring signal is provided to automatically close the waste gas decay tank discharge valve when the release rate exceeds the limits in the Offsite Dose Calculation Manual.

7.4.6 OTHER SAFETY RELATED DISPLAY SYSTEMS

Systems covered by this subsection are required to monitor Plant transients as identified in NUREG-0578/0737 and Regulatory Guide 1.97 and not discussed elsewhere. Appendix 7C provides a listing and an evaluation of all instrumentation contributing toward meeting Regulatory Guide 1.97.

7.4.6.1 Subcooled Margin Monitor

1. Design Basis

   Instrumentation detecting, displaying and annunciating physical parameters indicative of inadequate reactor core cooling is provided in the main control room as required by Regulatory Guide 1.97. Since this instrumentation involves the integrity of the core, it is classified as Class 1E, Seismic Category I and environmentally qualified. Through this system the operator is relieved from the time-consuming task of using steam tables along with primary coolant pressure and temperature observations during a transient when these parameters may be varying rapidly, thus eliminating the potential for corresponding operator errors.

2. Design Description

The subcooled margin monitor is made out of two redundant and separate channels designed in accordance with IEEE 279-1971. Each channel includes Class 1E primary coolant process signal inputs, an on-line microcomputer-based calculator, a display and an annunciator. Both display and annunciator are located on the main control board in the main control room.

The system provides a continuous indication of the °F of primary coolant margin from saturation conditions. One channel monitors Coolant Loop 1 and the other Coolant Loop 2. The two channels are powered from separate preferred ac sources for assured continuity of monitoring.

Each channel receives a pressure signal from a Primary Coolant System wide-range pressure transmitter and four wide-range temperature signals from given primary coolant hot and cold legs. The temperature ranges of the instruments are 50°F to 700°F and the pressure input is 0-3,000 psia. Temperature and pressure analog signals are converted to digital signals and entered to the microcomputer containing steam tables from which a saturation temperature and pressure are calculated. By comparing these values with the actual primary coolant values, a margin for saturation is calculated. Either the temperature or the pressure margin can be displayed.

IEEE 279-1971 separation requirements are met by the inclusion of isolation devices in accordance with IEEE 384-1977 between the temperature inputs and the applicable margin monitor channel circuit as well as interfaces with the primary coolant wide-range pressure channels.

Testing - The subcooled margin monitor has on-line testing features to allow periodic testing. Included are a number of tests which cover input, output and display functions. This testing is accomplished using only the display module on the control board and requires no external test equipment. One channel is tested at a time. Automatic return from test to normal status is provided if not returned by the operator. The system has also built-in self-diagnostic features which start flashing the display and sound an alarm to attract the operator's attention. The subcooled margin monitor testing requirements are included in the Technical Specifications.

3. Design Evaluation

The criteria of IEEE 279-1971 are met via separation, redundancy and testing features in the design. Interfaces, where inputs to the system mix more than one redundant channel, are provided with suitable isolators in accordance with IEEE

384-1977. On-line testing features ensure continuous availability with automatic removal of the test bypass. Finally, the system and its inputs have been qualified per IEEE 323-1974 and IEEE 344-1975.

7.4.6.2 Wide-Range Containment Pressure, Temperature and Water Level

1. Design Basis

Operator performance during transient and accident conditions must include recognition and response to containment atmosphere pressure conditions and potential inadequate core cooling. Highly reliable instruments covering wide-range pressure and water level are considered as essential aids to accident diagnosis and control by NUREG-0578/0737 and Regulatory Guide 1.97. Additional instrumentation for monitoring containment atmosphere temperature throughout the accident range is also required for proper diagnosis in conjunction with pressure and water level.

2. Design Description

Containment Pressure - Two redundant Class 1E continuous wide-range pressure transmitters fed from separate preferred ac power sources are installed and transmit signals to two recorders located in the control room with state-of-the-art accuracy and response time characteristics. The monitors have a range of 10 psia to 200 psia enveloping the range of -5 psig to at least three times the design pressure of the containment building (Chapter 5).

Containment Water Level - Two redundant Class 1E instrument channels (magnetic float devices) provide containment floor water level indication up to the 597-foot elevation. A diverse system provides two other redundant Class 1E instrument channels for containment sump water level indication and recording via level transmitters in the containment sump drain line to the dirty waste drain header. The combined range of these overlapping channels covers the maximum level theoretically obtainable during a Loss of Coolant Accident. Each of the measurement channels is fed from a separate preferred ac power source. These instruments have also state-of-the-art accuracy.

Containment Temperature - Four instruments are provided for monitoring the containment atmosphere temperature throughout the predicted accident range. Reactor cavity, steam generator space and containment dome temperature are indicated on the main control board up to 400°F. The system provides continuous display even though this is not a requirement. Sensors are resistance temperature detectors placed in appropriate locations in the containment. Indicators are mounted in a control panel containing Class 1E instruments within the main control room. Instrument power is provided by preferred ac Panel Y30 which receives its power from the Plant emergency power sources.

3. Design Evaluation

Both pressure and water level instrument channels meet IEEE 323-1974 and 344-1975. The process connections and the sensors, as well as all components of these channels, are Seismic Category I per Regulatory Guide 1.29 and meet Regulatory Guide 1.97 as required by NUREG-0578/0737.

7.4.6.3 Reactor Vessel Level Monitoring System

1. Design Basis

In accordance with the requirements of Generic Letter 82-28, a reactor vessel level monitoring system (RVLMS) is installed to provide indication for detection of inadequate core cooling (ICC) conditions. The system provides indication of level in the Primary Coolant System covering a range from near the top of the reactor vessel head to just above the reactor core. The RVLMS instrumentation system assists the operator in avoidance of ICC when voids in the Primary Coolant System and saturation conditions result from overcooling events, steam generator tube ruptures or small-break loss of coolant events.

2. Design Description

The RVLMS consists of two independent, physically separated, redundant and identical channels. Each channel includes one radcal level instrument (RLI) which extends from the incore instrument nozzle closure flange down through the guide tube attached to the control rod shroud, ending just above the fuel assemblies. The RLIs are inserted into existing incore instrument guide tubes. Each RLI terminates above the reactor vessel head in a multi-pin, 1E qualified electrical connector. Two 1E qualified in-containment cable assemblies, connected in series, are provided for each channel from the RLI connector to the containment penetration. Outside of containment, 1E qualified cable is provided for each channel to transmit signals and supply power between the containment penetration and the level and temperature recording indicator (LTRI) located in the main control room.

Each RVLMS channel is composed of a manometer tube and an RLI containing eight sensors. The manometer is created by cutting ports in the existing incore instrument guide tube and is divided into regions with hydraulic isolators which are part of the RLI. Two separate regions, the head region and the upper guide structure (UGS) region, are monitored by four sensors each. The head region extends from the reactor head instrument nozzle flange down approximately 5.5 feet to the first hydraulic isolator. The UGS region extends from the first hydraulic isolator to the second hydraulic isolator located approximately at the top of the fuel alignment plate. The porting patterns cut in the incore instrument guide tube were determined by tests performed to establish proper manometer performance.

The RLI is a stainless steel rod containing sensors which consist of differentially connected thermocouple pairs. One thermocouple junction is electrically heated and the other is not. In water the heat transfer from the junctions is high so that the differential temperature indication is low. In steam the heat transfer rate is reduced and the heated junction gets hotter, increasing the differential temperature indication. High differential temperature is the indication of water uncovery of the sensor.

Indication for the RVLMS is provided by two identical units (LTRIs), one for each channel, located in the control room. The level indication consists of two strings of vertical light-emitting diodes (LEDs). A green LED indicates covered and a red LED indicates uncovered for each sensor in the RLI. The sensors are numbered and their distance above the fuel is indicated. Level information is also displayed on adjacent strip chart recorders, one for the head region and one for the UGS region. The recorders display differential temperature information from the RLI sensors which can be used to diagnose potential degradation or failure of the sensors and aid in interpretation of level information. Two additional recorders in the same panel display the outputs of eight CETs (16 total in the two channels) to provide the required CET backup displays. The CET and RLI signals are isolated and transmitted to the Critical Function Monitoring System (CFMS) for the primary displays.

3. <u>Design Evaluation</u>

The RVLMS conforms to the design and qualification criteria for Category 2 instrumentation of Regulatory Guide 1.97. An evaluation of the RVLMS against Regulatory Guide 1.97 criteria is provided in Appendix 7C. RVLMS meets NUREG-0737 requirements for inadequate core cooling instrumentation as described in Reference 10. RVLMS instruments have had operability and testing requirements incorporated into Technical Specifications with Amendment 129 (see Reference 17). The NRC has accepted the RVLMS as Palisades' Inadequate Core Cooling Instrumentation (ICCI) in a letter dated January 12, 1987 (Reference 11). The stated basis for approval was that the RVLMS meets the requirements of Item II.F.2 of NUREG-0737.

7.5 NONSAFETY-RELATED REGULATING CONTROLS

7.5.1 DESIGN BASES

7.5.1.1 Reactor Regulating

Reactivity is controlled by a combination of chemical shim and CRDM motion. Variation of the chemical shim (boric acid) concentration provides long-term regulation and control. The Chemical and Volume Control System is used to increase or decrease boron concentration with concentration being measured by chemical analysis.

Control rod motion is used for short-term regulation. Sequential insertion or withdrawal of the control rods in the regulating groups is used for normal power regulation. During Plant start-up and shutdown and all cases where power is below 15%, manual control of the control rods is used.

Either of two independent channels may be selected to provide reactor average coolant temperature and a reference temperature value corresponding to turbine power. The reactor average coolant temperature and reference temperature values are displayed to operators who manually adjust primary coolant temperatures by moving the control rods.

The primary coolant average temperature is adjusted according to a preselected program. This program provides an average temperature which is linearly increasing with power.

Inputs to the temperature computing stations are primary coolant cold leg temperature, primary coolant hot leg temperature, and turbine first stage pressure. The temperature computing stations are two separate stations with each having separate inputs as listed above. Rod position instrumentation is covered later in this section and details of the CRDM are covered in Chapter 3.

Control rods are grouped into shutdown, regulating and part-length groups. The shutdown groups are the first to be withdrawn on start-up and they remain withdrawn throughout power operation to provide a definite shutdown margin at all times. The regulating groups are manually positioned. The part-length control rods are manually positioned individually or by group and cannot be tripped. Alarms exist in the control rod position instrumentation to annunciate deviation of control rods within any group except the part-length group. Any individual control rod may be positioned manually if required.

The part-length control rods are completely withdrawn from the core during power operation except for control rod exercises and physics tests. The insertion of part-length rods into the core, except for rod exercises or physics tests, is not permitted since it has been demonstrated on other CE plants that design power distribution envelopes can, under some circumstances, be violated by using part-length rods.

The regulating and shutdown control rods are inserted by gravity action (backed up by control rod rundown) on the receipt of a reactor trip signal.

Control rod withdrawal is prohibited in certain situations where reactor limits are being approached. These prohibitions are detailed in Subsection 7.5.2.1.

7.5.1.2 Primary Pressure Regulating

The primary pressure controls maintain primary system pressure within preset limits by the use of pressurizer heaters and spray valves.

Pressurizer pressure and level sensors provide inputs to the controls. When system pressure is low, heaters are energized on a proportional or group basis to raise pressure. Before the level is low enough to uncover the heaters, level sensors cause the power to be interrupted to the heaters. When system pressure is high, pressurizer spray valves are opened on a proportional basis to reduce pressure.

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

7.5.1.3 Feedwater Regulating

The feedwater controls maintain steam generator downcomer level within acceptable limits by positioning the feedwater regulating valves supplying each steam generator (see Subsection 10.2.3.3). The speed of the turbine-driven main feedwater pumps will also be controlled by the feedwater controls.

Automatic control of feedwater is provided when the plant is at power. Steam flow, feedwater flow and downcomer level are used in a three-element controller on each steam generator to maintain preset level during steady-state and transient operation above 25%. At low power levels (5% to 25%), feedwater flow is regulated automatically by a single-element controller acting on the feedwater regulating bypass valves according to steam generator downcomer level requirements.

Manual control of feedwater flow may be assumed by the operator at any time. In the event of a reactor or turbine trip and if the feedwater pump turbine drivers are in the automatic control mode, the speeds of the feed pumps are automatically ramped down to a lower value. Following or during the ramp-down, operators can manually control main feedwater flow or initiate auxiliary feedwater flow as necessary to restore and maintain desired steam generator levels.

In the event of low steam generator pressure < 500 psia or containment high pressure (CHP), the main feedwater regulating and regulating bypass valves are closed to prevent excessive flow into the steam generators. This ensures containment pressure is not exceeded during a main steam line break inside containment (refer to Subsections 7.2.3.8 and 7.3.3.3). The valve closing on CHP was added by FC-906 in 1990 when analysis disclosed that for a small steam line break, low steam generator pressure would not occur fast enough to prevent exceeding containment design pressure. Administrative control of the bypass of the steam generator pressure signal to close the main steam isolation valves, the main feedwater regulating and regulating bypass valves is facilitated by using push buttons on the panel to override the signal for manual take-over of the controls. In addition to the push buttons, manual take-over of the feedwater regulating bypass valves can be accomplished by using key-operated switches.

7.5.1.4 Pressurizer Level Regulating

Pressurizer level is maintained within an operating band by means of the Chemical and Volume Control System. Water is continuously drained and charged automatically to and from the volume control tank with control inputs from pressurizer level sensors. The level set point is programmed as a function of T_{avg} (average of hot and cold leg temperatures). There are two completely independent automatic control channels with channel selection by means of a manual control switch. Automatic control is normally used during operation but manual control may be utilized at any time.

This control system is one of the defenses against high pressurizer level or water solid conditions. The Reactor Protective System provides the other defenses through the high pressurizer pressure reactor trip protective function (refer to Subsection 7.2.3.4).

7.5.1.5 Steam Dump and Bypass

The steam dump system provides a means of dissipating excess NSSS stored energy and sensible heat following a turbine trip, without lifting the safety valves. See Subsection 10.2.1.3 for details. Steam is discharged from the main steam lines to the atmosphere via steam dump valves and to the condenser via a steam bypass valve. The steam dump and bypass valves are sized to prevent opening of the steam generator safety valves following a turbine trip at full load. The steam flow is regulated by the dump and bypass valves in response to T_{avg} and secondary pressure signals.

Inputs to the controls are T_{avg}, turbine trip signal and steam header pressure.

The steam dump and turbine bypass controls need to be operable to control steam generator pressure such that P-8C is able to provide sufficient flow to maintain heat removal capacity. Most of the documentation in the facility change package, FC-516, either showed P-8C had sufficient capacity at 1000 psia or showed reduced capacity and referred to a Combustion Engineering letter P-CE-5792 which allows a lower flow. P-CE-5792 bases the new flow rates on 100 degree F feedwater and 900 psia steam generator pressure maintained by the turbine bypass. Since the primary coolant system heat load in P-CE-5792 included decay heat and primary coolant pump heat, the analysis assumes offsite power is available.

It is an assumption of P-CE-5792 that the turbine bypass valve will be available. More levels of redundancy are added to the assumption by including the four atmospheric dump valves as alternates to the turbine bypass. The wording implies a reliance on non-Q equipment. An acceptable alternative to relying on non-Q systems is taking credit for operator actions within thirty minutes after the reactor trips. When offsite power is available, the plant operator can manually start another auxiliary feedwater pump to provide additional auxiliary feedwater flow or trip all four primary coolant pumps.

The controls for the turbine bypass and atmospheric dump valve controls were designed and installed to provide reliable service in normal plant ambient conditions. Under accident conditions (with the exception of a main steam line break) the solenoid valves, switches, transducers, etc. will be in essentially a normal environment. If a main steam line break occurs that can't be isolated (to restore normal conditions), the steam generator will blow down (and the dumps or bypass won't be needed).

7.5.1.6 Turbine Runback

The original design included a turbine runback upon detection of a dropped rod. Later analysis showed that at the beginning of reactor core cycle, turbine runback could have unacceptable effects on reactor performance. Thus, the turbine runback feature has been disabled.

Turbine runback upon detection of a dropped rod doesn't exist in the Digital Electrohydraulic (DEH) control system. However, there is an operator selectable runback for 5% or 50% which, when selected, will runback to 5% or 50% at the rate that the DEH system was originally running.

7.5.1.7 Turbine Generator Controls

The turbine generator controls are the means by which the turbine generator is made to meet the electrical load demand placed upon it. The turbine impulse chamber pressure, turbine speed, and electrical load are used as the control indices.

7.5.2 SYSTEM DESIGN

7.5.2.1 Primary Loop Temperature Instrumentation

A block diagram of the primary loop temperature instrumentation is shown in Figure 7-55. The instrumentation consists of:

1. Four high level isolated transmitters.
2. Two temperature computing stations.
3. Two primary coolant loop hot leg temperature alarm/indicators.
4. Two T_{AVE}-T_{REF} recorders.
5. One dual input primary loop differential temperature recorder.
6. One two position switch to select either of the two channels to provide the T_{AVE} signal to the steam dump controller and pressurizer level controllers.
7. One digital indicator to display T_{AVE} as selected by the switch.
8. Two voltage to current converters.

Inputs to the temperature computing stations are:

1. Loop 1 Temperature computing station.
    1. Primary coolant loop 1 cold leg temperature.
    2. Primary coolant loop 1 hot leg temperature.
    3. Turbine first stage pressure.
    4. Primary coolant loop 2 T_{AVE}.
2. Loop 2 Temperature computing station.
    1. Primary coolant loop 2 cold leg temperature.
    2. Primary coolant loop 2 hot leg temperature.
    3. Turbine first stage pressure.

All inputs are isolated from protective system instrumentation.

Outputs of the temperature computing stations are:

1. Loop 1 temperature computing station.
    1. Primary coolant loop 1 T_{AVE}.
    2. Primary coolant loop 1 differential temperature.
    3. T_{REF} (for primary coolant loops T_{AVE}).
    4. Deviation alarm; $(T_{AVE}-T_{REF1})(T_{AVE}-T_{REF2})$

2. Loop 2 temperature computing station.
    1. Primary coolant loop 2 T_{AVE}.
    2. Primary coolant loop 2 differential temperature.
    3. T_{REF} (for primary coolant loop T_{AVE}).
    4. Deviation alarm; $(T_{AVE} \pm T_{REF2})$

Two separate T_{AVE} channels are provided for pressurizer level control and steam dump control. Either channel may be selected. In each channel a temperature computing station establishes primary coolant temperature (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 550°F to 580°F at 100% power. T_{REF} is compared with the actual loop T_{AVE}, and provides a deviation alarm at a prescribed value.

Control rods are divided into the following groups:

1. Shutdown: Two groups

2. Regulating: Four groups

3. Part-Length: One group

The circuitry utilized for sequential control rod group movement during manual operation is indicated on the Rod Drive Control System Schematic Diagram, Figure 7-56.

Each control rod remains stationary except when a raise or lower signal is present. When such a signal is received, the control rod will move at a fixed speed of approximately 46 inches per minute.

The shutdown control rods may be moved in the manual control mode only with either individual control rod or individual group movement. A selector switch prevents withdrawal of more than one shutdown group at any time. The shutdown groups must be withdrawn above the lower limit of their exercise band before regulating group withdrawal is possible. The upper 10 inches of shutdown group control rod travel is designated as the exercise band and is provided so the shutdown control rods may be exercised while the reactor is at power. An interlock from the primary control rod position indication system synchro limit switches described in Subsection 7.6.2.3 prevents the shutdown groups from being driven down more than 10 inches unless the regulating control rods are fully inserted. Refer to Figure 7-57. This interlock is bypassed for a control rod rundown following a reactor trip. The reactor trip initiates rundown of the shutdown and regulating control rods through relays connected in parallel with the CRDM clutches (see Subsection 7.2.5.1).

Regulating control rods may be moved in manual control with sequential group movement. Automatic sequential control utilizes outputs from the primary control rod position indication system data processor described in Subsection 7.6.2.3. Individual and groups of control rods may be moved in manual control. Sequential group movement provides that when the moving group reaches a programmed low (high) position, the next group begins lowering (raising); 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.

When the regulating groups reach the "prepower dependent insertion limit" alarm point, this condition will be annunciated. If insertion is continued, a "power dependent insertion" alarm point is reached and a continuous alarm is initiated. This second, continuous alarm can only be silenced by removing the condition (withdrawing control rods or lowering power) or by the use of a key switch. These two programmed limits may be adjusted during the life of the Plant and are provided by the primary rod position and secondary rod position indication systems data processors to assure adequate shutdown margin according to Technical Specifications requirements (see Subsection 7.6.2.3).

All control rods will be prevented from being withdrawn if either a high power or high power rate-of-change pretrip condition exists.

Interlocks to prohibit regulating group withdrawal are provided to prevent the reactor from reaching undesirable conditions. These interlocks are summarized in Table 7-3.

The part-length control rods may be moved manually either individually or as a group. A selector switch prevents simultaneous manual movement of the part-length and any other control rods. The part-length control rods have upper and lower limits of travel.

The part-length control rods are completely withdrawn from the core during power operation except for control rod exercises and physics tests. The insertion of part-length rods into the core, except for rod exercises or physics tests, is not permitted since it has been demonstrated on other CE plants that design power distribution envelopes can, under some circumstances, be violated by using part-length rods.

7.5.2.2 Primary Pressure Regulating

Two independent pressure channels provide suppressed range (1,500 to 2,500 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 the backup heaters are on. The spray valves are modulated to maintain system pressure. A continuous flow is maintained through the spray lines at all times to keep the pipes warm and to assure that the pressurizer maintains the proper boron concentration.

The two pressure control channels are recorded in the control room and provide independent high and low alarms.

Two independent level channels provide pressurizer level signals for two specific functions:

1. A low-level signal from either channel will de-energize all heaters.

2. A high-level signal from either channel will energize the backup heaters.

Control and alarm pressure set points are shown on Figure 7-58.

7.5.2.3 Feedwater Regulating

The steam generators are operated in parallel on the feedwater and on the steam sides. Each generator has a three-element controller with inputs of feedwater flow, steam flow (corrected for pressure) and downcomer level (see Figure 7-58). Output of each controller when in automatic control is used to provide pneumatic signals to position the respective feedwater regulating control valve. The higher of the two signals provides a speed control signal to the main feedwater, turbine-driven pumps. This signal controls pump speed if the turbine speed control is in automatic. When Plant power is between 5% and 25%, feedwater is automatically controlled by a single-element controller monitoring steam generator downcomer level and positioning the feedwater regulating bypass valves. Three overrides are provided:

1. When contacts in the steam dump permissive switch are actuated on a main turbine trip, feedwater regulating control valves are maintained in the position which existed prior to the switch activation. The feedwater pumps, if in automatic speed control, ramp down to a fixed speed which corresponds to approximately 5% of full flow. Due to the slow rampdown, operators manually control main feedwater flow or initiate auxiliary feedwater flow as necessary to restore and maintain steam generator levels.

2. When an abnormally high-steam generator level is sensed by an independent downcomer level sensor, a signal is sent to close the associated feedwater regulating control valve and a control room alarm is annunciated.

3. During low-steam generator pressure < 500 psia, the main feedwater control valves and the bypass valves are closed automatically. The operator can manually take control of the bypass valves by isolating the low steam generator pressure signal using a key switch on the control panel.

4. During high containment pressure, the main feedwater regulating valves and their bypass valves are closed automatically (FC-906, 1990).

Manual control of the feedwater flow may be assumed at any time to circumvent malfunction of components within the system except for control of the feedwater regulating valve when the steam generator high level override, steam generator low pressure override, or Containment High Pressure override is present. Each of these override signals blocks the valve control signal between the level controller and the feedwater regulating valve, causing the valve to be driven closed. Otherwise, when in manual control, the operator in the control room can:

1. Position manually each feedwater regulating control valve
2. Control speed of the two main feedwater pumps
3. Open or close each feedwater stop valve
4. Position manually each feedwater bypass regulating valve

Operation of the Auxiliary Feedwater System is always available to supplement the above options (refer to Subsection 7.4.3).

7.5.2.4 Pressurizer Level Regulating

The operating level of the pressurizer is a programmed function of the primary coolant loop T_{AVE}. It is programmed to accommodate plant load changes and transients while keeping the primary coolant system volume changes to a minimum.

Programmed level is established by the pressurizer level controllers, which receive a T_{AVE} input from the primary loop temperature instrumentation. The pressurizer level controller establishes a set point directly proportional to T_{AVE}. Refer to Figure 4-10. The controllers also receive pressurizer level information from two level sensors. Controller output is sent to control charging pumps and letdown orifice valves. The outputs of either of two automatic control channels may be selected by the operator for level control in addition to manual control.

Controller action and program level are described in Section 9.10, Chemical and Volume Control System.

7.5.2.5 Steam Dump and Bypass

A block diagram of the steam dump and bypass controls is shown in Figure 7-60. The system consists of:

1. One steam dump controller

2. Two steam dump controllers (performs tracking in auto mode) at engineered safeguards panel.

3. Four atmospheric steam dump valves

4. One steam bypass pressure controller

5. One signal auctioneering unit

6. One turbine bypass valve

Inputs to the controls are:

1. T_{AVE}

2. Steam header pressure

3. Turbine trip (contacts in steam dump permissive switch)

4. Loss-of-condenser vacuum (contacts)

Outputs of the controls are:

1. Atmospheric steam dump valve position signal

2. Turbine bypass valve position signal

The steam dump controller generates a suppressed range signal proportional to (T_{avg} - 532°F); see Figure 7-60 for signal characteristics. Upon receipt of a turbine trip, this signal is supplied to open the atmospheric steam dump valves, and is an input to the turbine bypass auctioneering unit to simultaneously open the bypass valve. The position of the atmospheric steam dump and bypass valves is proportional to the signals supplied to them, thus providing a controlled relieving of excess pressure.

After a turbine trip, the steam dump controller's quick opening signal will cause quick opening for the dump and bypass valves.

The atmospheric steam dump valves will close proportionately as T_{avg} reduces and will close completely at 535°F. They will remain closed unless T_{avg} increases again to more than 540°F.

The turbine bypass valve receives the higher of the steam dump controller or steam bypass pressure controller signals through an auctioneering unit. This controller generates a suppressed range signal proportional to secondary pressure over the range 895 psia to 905 psia. Loss-of-condenser vacuum will prevent the opening of the bypass valve.

By manually changing the set point on the steam bypass pressure controller, the operator may control primary coolant temperature during Plant cooldown by use of the bypass valve. Once the turbine trips, the operator may manually control the atmospheric dump valves and still control the turbine bypass valve.

7.5.2.6 Turbine Generator Controls

The turbine generator controls (or electrohydraulic control system) controls steam flow to the turbine. The controls consist of the following five parts:

1. Operators Interface Panel
2. Digital Controller & Engineers Console
3. Steam valve servo actuator assemblies
4. High-pressure fluid supply
5. Emergency trip

The electronic controller performs basic digital computations on reference signals and turbine feed-back signals and generates an output to the actuators.

The operator's panel contains push buttons and switches which are used to change the reference input to the controller to vary the speed or load. Indicators provide continuous monitoring of steam admission valve position, load limit setting and control signal.

The servo valves position the governing valves by directing the flow of E-H oil to the actuators.

The high-pressure fluid controls operate the stop, governor, reheat stop and interceptor valves. The fluid is stored in a stainless steel reservoir and is furnished with a duplicate system of oil pumps, filters and heat exchangers. The fluid pressure is controlled by constant pressure, variable displacement pumps.

The emergency trip uses bearing lubricating oil as its control medium. The trip device is a diaphragm valve which is sensitive to the dump valve servo actuator position and releases the emergency trip oil to drain. This action is caused by the operation of trips located in the hydraulic mechanical system protective device unit: low-vacuum, low bearing oil pressure, thrust bearing failure, overspeed trip and loss of generator load, or manually with the overspeed trip lever. This action is also caused by operation of the solenoid trip which is actuated by the manual trip switch in the control room and by electrical system protective relays. This emergency trip is also activated by the ATWS system which was installed in 1990. When a pressurizer high pressure 2 of 4 logic condition is satisfied, the emergency trip solenoid valve 20/ET is activated.

A reactor trip results from a turbine generator trip only when the reactor is above 15% full power level. A turbine generator trip on closure of the main steam isolation valves (MSIVs) is provided to protect the MSIVs from experiencing the full differential pressure of a steam generator.

The turbine generator unit is controlled from the operator's interface panel. The panel shows which devices are controlling the turbine generator. The controller computes signals to position the governor valves. As the speed reference is changed during start-up, the speed transducer signal is compared to the reference speed setting. During load control, the controller also computes signals to position the governor valves. As the load reference is changed by the control operator, the controller compares the load reference to the feed back from the megawatt transducer, the first stage steam pressure transducer, and from valve curves contained within the computers memory. The difference or load error then sets the position of the governor valves servo actuators. The governor valve servo actuators change the steam flow to the turbine. The result is a change in turbine load which is detected by the transducers and is compared to the reference load setting.

The turbine generator controls are composed of the digital computer controller and the valve servo driver cards. The Valve Servo Driver accepts serial input from the local auto controller, plus LVDT inputs. It outputs to Moog Valve Coils and the Valve Position meter. Manual inputs and Manual functions, plus valve contingencies are also built-in capabilities. This system element was designed by hydraulic and electronic experts to yield maximum performance. It provides minimum tuning time to save valuable manpower time.

The reheat stop and interceptor valves can be tested while the turbine is loaded. A signal can be introduced to the Valve Servo Driver cards to cause the servo valves to exercise the stop and governor valves.

7.5.3 SYSTEM EVALUATION

7.5.3.1 Rod Drive Control System

Control rods are segregated into three groups: shutdown, regulating and part-length. The shutdown and part-length control rods remain raised throughout power operation to provide a shutdown margin at all times. When the reactor is tripped, all control rods except the part-length are inserted by gravity and backed up by control rod rundown.

Sequential insertion and withdrawal of regulating group control rods are employed for normal power regulation. The manual demand signal controls the application of power to the regulating group(s) selected for movement by the action of the group sequencing relays. The group sequencing relays are programmed by permissive actions initiated by the data processor, utilizing information from the primary rod position system.

The circuitry used for sequential control rod group movement is shown on the Rod Drive Control System Schematic Diagram, Figure 7-56. The insertion of the shutdown rods before the regulating rods is prevented by the contacts from the shutdown rod insertion permissive relays R/RS1 through R/RS4 or the exercise band limit switches and contacts from the shutdown group relay R/AD1. A single failure could cause the shutdown rods to be inserted beyond the exercise limit prior to the insertion of the regulating rods, only in conjunction with the operator selecting the shutdown group for insertion. Simultaneous insertion of the regulating and shutdown rods could occur only by energizing the rod run-down relay. Failure of a single CRDM clutch power supply "K" relay (Subsection 7.2.5.3) will not cause rod rundown. De-energization of either the shutdown (scram) or rod control bus will run down all control rods.

The simultaneous withdrawal of more than two groups of control rods could occur upon certain single failures in the control system. This could occur during the overlap period when two groups of rods are being withdrawn, so that the failure of the sequential permissive contact in the sequence relay circuit of a third group would permit three groups to be moving at once. Indication of the group(s) selected for rod motion and indication of the direction of rod motion for the group(s) selected is provided at the main control room console (refer to Figure 7-56). An out-of-sequence alarm initiated from the rod position is provided to alert the operator of an out-of-sequence condition.

Should more than two banks be withdrawn simultaneously, there are two aspects to consider: (1) reactivity addition rate and (2) effect on power distribution and, therefore, the DNB ratio. The reactivity addition rate will be increased but not exceed the maximum values shown in Figures 14.2-6 and 14.2-17. As shown in these figures, as reactivity addition rate is increased, the minimum allowable DNB ratio increases. An adverse effect must then come from a change in power shape relative to the shape applicable to the planned sequential withdrawal. Starting from full-power conditions, maximum bite, only two banks are inserted (Figure 3-3); and thus the three-group withdrawal incident cannot occur. Starting from lower power conditions, for example at 50% of full power, there are three groups inserted under maximum bite conditions; and at hot standby there are four groups inserted. Withdrawing various combinations of three banks during the overlap period, starting from these initial conditions, results in minimum DNB ratios greater than 1.3.

It is possible for a rod group to be withdrawn out of proper sequence if certain single failures occur in coincidence with specific operational situations. This could occur during the overlap period when two groups of rods are being withdrawn so that any failure, such as the de-energization of the group relay which stops the movement of the leading group, will interrupt the planned sequence. The out-of-sequence alarm will be actuated in such an event. Again, the nonsequential withdrawal is not a continuous withdrawal.

The major difference between a sequential and a nonsequential withdrawal is their respective power distribution during the withdrawal. Therefore, in order to examine the potential consequences of a nonsequential withdrawal, cases were studied for initial conditions of maximum bite (Figure 3-3) for full power, 50% of full power and hot standby. For these initial conditions, the most unfavorable of nonsequential withdrawals were studied. The maximum total peaking factors are equal to or less than those assumed for the accident (sequential withdrawal) analysis of Chapter 14, except for the hot standby condition. For hot standby, the maximum total peaking factor may be several percent greater than for the nonsequential case. In all cases, the axial power peak is lower in the core (and therefore more favorable from a DNB ratio standpoint) for the nonsequential cases relative to the associated sequential power distributions. For all cases analyzed, the DNB ratio is greater than 1.3.

7.5.3.2 Primary Pressure Regulating

Two independent 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. Independent high and low alarms are provided.

7.5.3.3 Feedwater Regulating

For power above 25% full power, conventional three-element, feedwater control is used with fail-as-is, feedwater control valves. Manual override of the automatic control is always available. Manual bypass valves and feedwater stop valves provide backup for feedwater valve failure. For power below 25% full power, and to facilitate start-ups, a single-element feedwater bypass valve controller is used. Manual override of this automatic control is also available.

Feedwater pump speed control is by automatic or manual means.

Feedwater flow from the condensate pumps will be shut off via closure of the feedwater regulating and bypass valves on low steam generator pressure (< 500 psia). 500 psia is above the maximum condensate pump delivery head such that the maximum of feedwater delivered to the steam generator will be no greater than that assumed in the safety analysis.

Accurate measurements of reactor power output use the feedwater flow instruments as a base for calorimetric calculations (see Subsection 7.2.3.2 for reactor power level measurement versus reactor trip function). These flow instruments' calibration is thus regulated by the Technical Specifications.

A long term operational affect on the feedwater flow instrumentation (by methods such as venturi fouling) causes them to indicate conservatively (indicated feedwater flow is greater than actual feedwater flow). In order to determine the actual feedwater flow, an ultrasonic flowmeter has been used. By comparing the feedwater flows from both the feedwater flow instrumentation and the ultrasonic flowmeter, a correction factor (the ratio of ultrasonic flowmeter flow to feedwater instrumentation flow) can be calculated. This correction factor can be multiplied by the feedwater flow instrumentation indicated flow and then inputted into the calorimetric calculation as the actual feedwater flow. This will result in the most accurate available measurement of reactor power.

7.5.3.4 Pressurizer Level Regulating

Two separate level control channels are provided with redundant level transmitters and controllers. Only one channel is used during operation. The controllers are located in the control room. Control can be accomplished by either automatic or manual operation. Three charging pumps and three letdown orifice valves provide redundant means of increasing or decreasing primary coolant water inventory. The variable pressurizer level control program maintains primary coolant discharge and addition required during Plant load changes.

The pressurizer level control system is sufficient to protect the Primary Coolant System fluid boundaries without a reactor trip on high pressurizer level to protect against a water solid condition. If a malfunction is suspected in the operating channel, operation can be switched to the other channel. If a failure of the controller output is postulated, a large "program minus actual level" difference will result. This will cause two orifice stop valves to close and will start all three charging pumps. When the pressurizer level increases 4.6% above the programmed level, two charging pumps will be secured and the two orifices' stop valves reopened. This failure will not result in a filled pressurizer.

If a failure in the level transmitter is postulated, a low-low level signal is initiated causing all orifice stop valves to close and all three charging pumps to operate. In order to fill the pressurizer, this condition would have to exist unchecked by operation action for a period of about 30 minutes (time required to fill the 700-ft^3 steam space in the pressurizer). During this period, an alarm would sound alerting the operator to the mismatch between charging and letdown flow. Also, a low-level alarm from the volume control tank would sound. (There are about 3,600 gallons stored in the volume control tank, and over 5,000 gallons are required to fill the 700-ft^3 steam space.) Continuous operation of the charging pumps is indicated by lights in the control room. The operator can switch to pressurizer level control Channel B (assuming Channel A is in service at the time) to determine if the reason for the extended charging operation is caused by a malfunction of the level transmitter. The operator can manually secure the charging pumps when the problem has been diagnosed, or allow the backup channel to assume control.

Assuming that no operator action was taken, and the pressurizer continued filling with water, and pressure continued to rise, the pressurizer pressure control system will maintain pressure at about 2,180 psia. At 1,700 seconds after the transient is initiated, the spray nozzle, which extends about 2 feet from the top of the pressurizer, becomes submerged by the rising water and is no longer effective (the effectiveness of the spray is reduced even before submergence of the spray nozzle, owing to the shape of the spray extending from the nozzle). At the level of nozzle submergence, a 40-ft^3 steam space remains at the top of the pressurizer. A simplifying and conservative assumption invoked for this analysis is that pressurizer pressure is maintained at 2,180 psia until it is completely filled, thereby neglecting the mitigating effect of the 40-ft^3 steam space in minimizing pressure during the incident.

After the pressurizer fills, it is assumed that all three charging pumps continue to deliver water into the Primary Coolant System at the maximum rate of 120 gpm and that all letdown orifices remain closed. The time required to increase system pressure from 2,180 psia to reactor trip pressure of 2,255 psia is approximately 2 minutes (owing to the compressibility of one-half million pounds of water at 578°F, more than 1,000 pounds of additional water is required to raise system pressure 220 psi).

Following the high pressurizer pressure reactor trip signal, the control rods are inserted 90% of travel within 2.7 seconds. The turbine admission valves are closed at 0.3 second, and all rods are fully inserted. Although the steam dump and bypass valves will open following turbine trip (and thereby reduce the pressure increase in the steam generator and subsequently reduce the increase in primary system temperature), credit is not taken for such action in this analysis. The maximum increase in the average temperature of the primary system following trip of the reactor and turbine is less than 0.5°F. This energy increase in the Primary Coolant System results in a Primary Coolant System pressure transient as shown on Figure 7-61. The maximum pressure increase is 26 psi and occurs approximately 4 seconds following trip. Since reactor outlet temperature is decreasing during the entire transient following trip, the primary system temperature increase is due entirely to the increase in steam generator pressure, causing an increase in primary coolant temperature exiting from the steam generators.

At four seconds following reactor trip, core heat flux is decreasing at a faster rate than primary coolant temperature exiting from the steam generators is increasing; and therefore, the Primary Coolant System pressure begins decreasing as shown in Figure 7-61.

If it is postulated that the pressurizer fills solid with water, owing to a malfunction in the pressurizer level control system concurrent with the assumption of no operator response to the various alarms and indications available, the maximum Primary Coolant System pressure during the transient is well below the hydrostatic test pressure of 3,125 psia. Because of the rapid response of the reactor protection system in causing a reactor trip at 2,255 psia, and because of the large heat sink supplied by the 241,000 pounds of liquid stored in the steam generators, the maximum pressure during the transient is 2,426 psia; and operation of the pressurizer safety valves is not required.

7.5.3.5 Steam Dump and Bypass

The steam dump valves can be operated from either the control room or from the engineered safeguards local panel. Automatic or manual control is provided at the control room station.

Inadvertent opening of the atmospheric dump valves is prevented by requiring that the turbine stop valves be closed before the dump valves can be opened. Excessive primary system cooldown by the dump valves when in automatic control is prevented by a narrow-range T_{avg} temperature signal which has a minimum output corresponding to 515°F.

Turbine bypass is available whether the turbine valves are open or closed and will limit the maximum steam pressure to 900 psia during hot standby.

7.5.3.6 Turbine Generator Controls

The electrohydraulic control used, is a conventional control system with many unit-years of operating experience. This type control has been refined and has proved to be very reliable and superior to earlier controls.

With the redundancy and safety features designed into the turbine control and protection system as described below, the probability of turbine overspeed occurring is very remote.

The steam required to produce turbine overspeed has to come from either the main steam system or flashing from feedwater heaters and moisture separators after a turbine trip. All feedwater heaters with sufficient energy to overspeed the turbine have extraction nonreturn valves and the moisture separator outlets have intercept valves to limit steam flow following a turbine trip. To have an uncontrolled source of steam from the main steam line, all of the following turbine control devices would have to fail:

1. Main governor and governing valves.

2. Overspeed protection controller. This is an acceleration response device which closes the turbine main governing valves and moisture seperator intercept valves.

3. Mechanical overspeed trip. This is a centrifugally actuated device which trips the turbine main stop, control, reheat stop and intercept valves (16 valves total).

The main governor and overspeed protection controller both control high-pressure fluid system which provides the motive force to operate the turbine steam valves. The high-pressure fluid system consists of duplicate oil pumps, filters and heat exchangers. The fluid reservoir is stainless steel to minimize the possibility of contamination.

The mechanical overspeed trip actuates the auto stop oil system which uses turbine oil as the control medium and is separate from the high-pressure fluid control system used for the main and auxiliary governing systems.

The turbine main stop and governing valves, and moisture separator intercept and reheat stop valves are all spring-loaded to fail closed.

The turbine overspeed event has been analyzed in Section 5.5.

7.6 NUCLEAR STEAM SUPPLY SYSTEM INSTRUMENTATION

This section primarily describes nonsafety-related instrumentation relevant to the systems discussed in Sections 7.2, 7.3 and 7.5. Safety-related instrumentation for these sections and Section 7.4 are mentioned only for clarity of text.

7.6.1 DESIGN BASES

7.6.1.1 Process Instrumentation

The nuclear steam supply system (NSSS) nonnuclear process instrumentation measures temperatures, pressures, flows and levels in the Primary Coolant System, secondary system, NSSS auxiliary systems and measures containment parameters such as pressure, sump level, hydrogen content, gamma radiation. Process variables required on a continuous basis for start-up, operation and shutdown of the Plant are indicated, recorded and controlled from the control room. Other instrumentation which is used less frequently or which requires a minimum of operator action is located near the equipment with remote alarms annunciated in the control room. Alternate indicators and controls are located at other locations than the control room to allow reactor shutdown and cooldown should the control room have to be evacuated as described in Section 7.4.

Four independent measurement channels are provided to monitor each process parameter required for the Reactor Protective System (refer to Section 7.2). Redundant channels are provided for engineered safeguards action to meet the single failure criterion (refer to Section 7.3). The four independent channels provide sufficient redundancy to ensure system action and to allow each channel to be tested during Plant operation. Class 1E instruments listed in Table 5.7-8 are designed to withstand seismic loads described in Section 5.7 and have been environmentally qualified as described in Chapter 8, Subsection 8.1.3.

Two independent channels are provided to monitor parameters required for critical control functions (refer to Section 7.5).

7.6.1.2 Nuclear Instrumentation

Eight channels of instrumentation are provided to monitor the neutron flux. The system consists of four power range safety channels, and two source range channels combined with two wide range channels. The source/wide range channels share high sensitivity fission chambers, enclosure, and power supply; while the power range channels are completely independent with each channel complete with separate detectors and power supplies. Each power range safety channel is also provided with a rod drop detection circuit and provides calibrated flux, upper and lower signals to its channelized thermal margin monitor and non-channelized critical function monitor. The operating range of the eight monitoring channels is greater than ten decades of neutron flux with channel overlap adequate to monitor the reactor power from shutdown through start-up to 200% of full power(10^{-8}% to 200% of full power). See Figures 7-8 and 7-9 for channel range and overlap.

The neutron flux detectors are located in instrument thimbles in the biological shield around the reactor vessel. Each start-up and wide-range detector is placed approximately 180° apart. The power range safety channel detectors are placed in thimbles approximately every 90° around the core.

7.6.1.3 Control Rod Position Instrumentation

The Palisades Plant Computer is a distributed computer system composed of a host computer and several nodes. Two of these nodes are the PIP node and the SPI node. The SPI system, composed of the SPI node plus the host computer, is redundant to the PIP node in the tasks of control rod measurement, control rod monitoring, and limits processing.

The PIP Node ("PIP") uses the output of a synchro to provide the rod position. Each of the 45 control rods has a synchro. Control rod position is visually displayed on a control room panel. This information is also passed to the host Computer and to the control room workstation. Position information is also used to initiate alarms under certain limiting conditions, to provide contact closures for control rod sequencing and control, and to monitor for excessive control rod position deviation between individual rods within a group. The PIP is capable of measuring and recording the time for a control rod to reach bottom after the control rod clutch is released during a control rod drop test.

The SPI system ("SPI") consists of the SPI node plus the host computer. The SPI node functions as an input module and all processing of information is done in the Host Computer. The SPI node gathers information on the control rod positions from the reed stack switches. Each of the 45 control rods has a reed switch stack. The host computer monitors the various limits associated with the control rods. These limits include the PDILs and rod sequencing. If a limit is exceeded, an alarm is annunciated on the control room workstation. It does this monitoring using the rod positions from the synchro transducers. If a synchro input card on the PIP were to fail, the host computer would use the rod position from the SPI input module in the monitoring of rod limits. The host computer also compares the position of the rod position from the synchro transducer and the rod position from the reed stack switches. If there is deviation in the positions greater than a preset limit, an alarm is annunciated on the control room workstation.

7.6.1.4 Incore Instrumentation

The primary function of the incore instrumentation is to provide measured data which may be used in evaluating the neutron flux distribution in the reactor core. This data may be used to evaluate thermal margins and to estimate local fuel burnup.

The bases for the design of the incore monitors are as follows:

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

2. Flux detectors of the self-powered type, with proven capabilities for incore service, are used.

3. The information obtained from the detector assemblies may be used for fuel management purposes and to assess the core performance. It is not used directly for automatic protective or control functions, however it is used for ex-core instrument calibration.

4. The output signal of the flux detectors will be calibrated or adjusted for changes in sensitivity due to emitter material burnup.

5. The detector assemblies are comprised of local neutron flux detectors (stacked vertically for axial monitoring) and a thermocouple.

Axial spacing of the detectors in each assembly and radial spacing of the assemblies permit representative neutron flux mapping of the core and monitoring of the fuel assembly coolant outlet temperatures.

The incore instrumentation is required to measure radial peaking factors for Technical Specifications limits monitoring. This assures that the assumptions used in the analysis for establishing DNB margin, linear heat rate and the TM/LP and high-power Reactor Protective System trip set points remain valid during operation.

The incore instrumentation must also provide a diverse monitoring of reactor core quadrant power tilt and linear heat rate, both parameters being monitored also by the excore nuclear instrumentation (Subsection 7.6.1.2). This diversity of monitoring assures that, in the event of an LOCA, the peak fuel cladding temperature will be acceptable and the minimum DNB will be maintained above acceptable levels (fuel damage will not exceed acceptable limits) during anticipated transients. Quadrant power tilt and linear heat rate are limited by Technical Specifications. Linear heat rate is monitored in the control room normally via the incore alarm system. When required, the quadrant power tilt is determined from calculations involving incore detector readings.

Sixteen of the incore detectors are provided with electrical connectors and cabling inside containment which has been environmentally qualified to the requirements of IEEE 323-1974. This provides assurance that the sixteen core exit thermocouples (4/core quadrant) will be available to provide indication of the approach to inadequate core cooling conditions following postulated accident conditions. Design of these core exit thermocouple instrument loops meets the intent of NUREG-0737 and Regulatory Guide 1.97.

7.6.1.5 Palisades Plant Computer (PPC)

This monitoring system is provided to display, print, and store plant process information. Functions provided include Sequence of Events (SOE) monitoring, Safety Parameter Display System (SPDS) and Emergency Response Data-link System (ERDS). It is part of and provides services to the PIP/SPI control rod monitoring system described elsewhere. It provides a link between the Incore neutron inputs and Incore analysis software.

Sequences of events for safety- and non-safety-related Plant parameters of the following systems are monitored, displayed, and recorded.

1. Reactor Protective System
2. Engineered Safeguards Controls
3. Reactor Shutdown Controls
4. Fluid Systems Protection
5. Regulating Controls
6. Primary Plant Process Instruments
7. Secondary Plant Process Instruments
8. Electrical Power Distribution

The PPC is a non-class 1E monitoring system.

The PPC includes and conforms to Critical Functions Monitoring System (CFMS) design. This design provides concise display of important parameters to control room operators. The PPC is designed to provide the same information to the Technical Support Center (TSC) and Emergency Operations Facility (EOF) to aid in emergency response management. The CFMS is a Safety Parameter Display System as described in Supplement 1 to NUREG-0737. In a letter dated April 19, 1990 the NRC found the Palisades' SPDS to be acceptable on the basis that it meets NUREG-0737 Supplement 1 requirements (References 15 and 16).

The PPC typically interfaces with Class 1E systems through electronic isolation devices, 100K ohm isolation resistors, relay contacts and the CFMS input termination/ multiplexer cabinets located in the control room. The CFMS control cabinets are designed to be seismically qualified to the criteria of IEEE 344-1975. The CFMS input cabinets also provide for separation and isolation of Class 1E and Nonclass 1E equipment in accordance with the requirements of IEEE 384-1977.

The SOE node, Cooling Tower Control System (CTCS) node, PIP node, SPI node and the D204 Battery backed power system include components located in the CP Co Design Class I portion of the auxiliary building and, as such, are required to be housed in cabinets qualified as Seismic Category I per Regulatory Guide 1.29 to prevent damage to other equipment through structural failure. This system has been classified as functional Nonclass 1E, as such interfaces of Class 1E components with the system must meet IEEE 384-1977 and be in accordance with 10 CFR 50, Appendix A, GDC24.

7.6.2 SYSTEM DESCRIPTION

7.6.2.1 Process Instrumentation

The following process instruments are associated with the reactor protective, reactor control or primary Plant controls. They are safety related or nonsafety related as indicated.

Temperature - Temperature measurements are made with precision resistance temperature detectors (RTDs) which provide a signal to the remote temperature indicating control and safety devices. Class 1E temperature channels in each primary reactor coolant leg are provided power from separate preferred ac buses.

The following is a brief description of each of the temperature measurement channels:

1. Hot Leg Temperatures - Class 1E

Each of the two Primary Coolant System hot legs contains four safety grade temperature measurement channels. Each of these channels provides a narrow-range (515°F-615°F) temperature signal to the thermal margin monitors which input to the reactor protection system. Two of these channels on each hot leg also provide wide-range (50°F-700°F) temperature signals to the subcooled margin monitors discussed in Subsection 7.4.6.1.

One of these channels from each loop provides narrow range input through an isolation device to the temperature computing station and an indicator as discussed in section 7.5.2.1. Both the narrow-range and wide-range signals are obtained from the same RTD through use of a dual-range RTD transmitter.

Indications for each of the narrow-range temperature channels are provided in the control room. Indication of one of the wide-range temperature channels on each hot leg is provided in the control room and at the auxiliary hot shutdown control panel (C-150). Two of the wide-range hot leg temperature channels are also available in the Critical Function Monitoring System (CFMS) computer.

2. Hot Leg Temperature - Nonclass

A hot leg control grade signal is obtained from a safety channel through an isolation device for each hot leg. These channels provide a narrow range signal (515°F - 615°F) to the temperature computing station. Indication of the control grade temperature measurements for each hot leg is provided in the control room. A high temperature alarm is provided by these channels to alert the operator to a high temperature condition.

3. Cold Leg Temperature - Class 1E

Each of the four Primary Coolant System cold legs contains two safety grade temperature measurement channels. Each of these channels provides a narrow range (515°F-615°F) temperature signal to the thermal margin monitors which input to the reactor protection system. A TC alarm is initiated by the thermal margin monitors if the maximum monitors TC (Class 1E) exceeds a TC max or TC min operator adjustable set point. One of these channels on each cold leg provides wide-range (50°F-700°F) temperature signals to the subcooled margin monitors discussed in Subsection 7.4.6.1. One of these channels from each loop provides narrow range input through an isolation device to the temperature computing station and an indicator as discussed in Section 7.5.2.1. One of these channels from each loop provides wide range input through an isolation device to LTOP. Both the narrow-range and wide-range signals are obtained from the same RTD through use of a dual-range RTD transmitter.

Indications for each of the narrow-range temperature channels are provided in the control room. Indication of one of the wide-range temperature channels from each Primary Coolant System loop is also provided in the control room and at the auxiliary hot shutdown control panel (C-150). One wide-range channel from each of the four cold legs is available in the CFMS computer.

4. Cold Leg Temperature - Nonclass

A cold leg control grade signal is obtained from a safety channel through and isolation device for each cold leg. These channels provide a narrow range signal (515°F - 615°F) to the temperature computing station.

5. Loop Average Temperature

Loop average temperature is computed through the computing station in each loop. The computing station receives inputs from the control channel hot and cold leg temperatures. It outputs to a control room recorder.

The temperature recorders are 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).

6. Loop Differential Temperature

The loop differential temperature (Nonclass 1E) is computed from the control channel hot leg and cold leg temperature detector signal. Each loop differential temperature is recorded in the control room.

Pressure - Pressure is measured by electronic 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 each of the pressure measurement channels:

1. Pressurizer Pressure (Protective Action) - Class 1E

   Four pressurizer pressure transmitters provide independent suppressed range pressure signals for initiation of Reactor Protective System trips on high-pressure and low thermal margin. In addition to the above trips, signals are provided for initiation of safety injection.

   These four independent pressure channels provide the signals for the Reactor Protective System high-pressure trip and the variable thermal margin/low-pressure trip. These channels also provide the low-low-pressure signal to the safety injection units. All four pressure channels are indicated in the control room and high, low, and low-low alarms are annunciated. Each channel is provided power from a separate preferred ac bus.

2. Pressurizer Pressure (Control and Indication) - Class 1E

   Redundant narrow-range pressure channels are provided for overpressure interlocks on the suction line valves for shutdown cooling. These interlocks provide additional assurance that the high-low pressure interface between the Primary Coolant System and the Shutdown Cooling System is not breeched when the Primary Coolant System is pressurized above the design pressure of the Shutdown Cooling System. These narrow-range pressure channels also initiate opening of the PORVs (less than 600 psia) when required for overpressure protection of the Primary Coolant System at low temperatures (see Subsection 7.4.2.1). Indication and recording of these narrow-range pressure channels is provided in the control room. Power to the pressure channels is provided from separate preferred ac buses.

   Redundant wide-range pressure channels initiate opening of the PORVs (greater than or equal to 600 psia) when required for overpressure protection of PCS at low temperatures and provide for indication of Primary Coolant System pressures as recommended by Regulatory Guide 1.97. Components of these channels located in a harsh environment are qualified to the requirements of IEEE 323-1974 to provide assurance that the indicating loops will continue to function during post-accident conditions. These pressure channels also provide input to the subcooled margin monitors described in Subsection 7.4.6.1.

3. Pressurizer Pressure (Control and Indication) Nonclass

Two independent pressure channels provide suppressed range signals for control of the pressurizer heaters and spray valves during normal operations. The output of either pressure control loop may be selected for primary pressure control by a selector switch located in the control room. These pressure channels are indicated and recorded in the control room and are powered from independent preferred ac buses.

Level - Level is sensed by level 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 outputs of the level transmitters provide signals to the remote level indicating, control and safety devices.

The following is a brief description of each of the level measurement channels:

1. Pressurizer Level

Two Nonclass 1E independent pressurizer level transmitters provide signals for use by 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 two pressurizer level control channels each provide a signal for level recorders in the control room. These recorders are two-pen recorders, with one pen recording actual level as sensed by the level control channel and the other pen recording the programmed level set point signal as calculated by the level controller.

One Class 1E pressurizer level transmitter provides a signal to a control room indicator. Indication is provided also on the Auxiliary Hot Shutdown Control Panel C-150. The level transmitter receives power from a preferred ac bus. The level transmitter is calibrated for cold conditions in the Primary Coolant System.

Pressurizer level is also measured by a second Class 1E level transmitter which indicates in the control room and on the Engineered Safeguards Auxiliary Panel C-33. This level transmitter is calibrated for cold conditions in the Primary Coolant System.

2. Steam Generator Level

Each steam generator has four Class 1E narrow-range level transmitters for Reactor Protective System channels and two Nonclass 1E transmitters for control function. Each protection channel is provided with physically separated sensing taps. Each channel has level indication in the control room. One of the four indications is also located on the Auxiliary Hot Shutdown Control Panel C-150. In addition, two Class 1E wide-range level channels per steam generator are available to ensure proper monitoring of steam generator level during operation with auxiliary feedwater (see Subsection 7.4.3.2). Indication from these last channels is located in the control room.

Flow - An indication of primary flow is obtained from measurements of pressure drops across each steam generator. These pressure drops are sensed by differential pressure transmitters which convert the pressure difference to dc currents. The dc currents provide a signal to the remote flow indicating and safety devices.

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

1. Primary Coolant Loop Flow Rates

Four independent Class 1E differential pressure transmitters are provided in each loop branch to measure the pressure drop across the steam generators. The outputs of one of these from each loop branch are summed to provide a signal of flow rate through the reactor core, which is indicated and supplied to the Reactor Protective System for loss-of-flow determination. The differential pressure sensed by each transmitter is indicated in the control room. The arrangement of the flow transmitters is shown on Figure 7-6.

7.6.2.2 Nuclear Instrumentation

Introduction - The Nuclear Instrumentation System consists of eight channels.

The combined source/wide range channels and power range safety channels are located in the Reactor Protective System cabinet in the control room. Four cabinets designated as A, B, C and D each house one channel of the protective system. Cabinets A and B each contain one power range channel. Cabinets C and D each contain one source/wide range channel and one power range safety channel. Mechanical and thermal barriers between the cabinets reduce the possibility of common event failure. The source and wide-range detector cables of a channel originate from the same preamp and are routed in the same cable tray. Each redundant source/wide range channel is separated and fed through different penetrations. The power range safety channel detector cables are routed separately from each other including penetration areas. The nuclear detector locations are shown in Figure 7-62.

The source range indications are derived from dual independent high sensitivity fission counters. The detector output signals from this dual fission chamber arrangement are amplified, discriminated and summed at a remotely mounted preamplifier. This conditioned signal is input to the source level source rate circuitry located in the control room. Audible count rate signals are available in the control room and in the containment building.

The wide range indications receive signals from one of the dual high sensitivity fission counters used also for source range detection. The detector signal is preamplified before input to the count rate and Campbell circuits in the source/wide range drawer.

Four channels are designated as power range safety channels and are connected to the Reactor Protective System. These channels operate from 0% through 125% of full power. Instantaneous nuclear power signals are input to the Thermal Margin Monitor (TMM) for use in variable high-power trip, thermal margin/low-pressure set point calculation and axial shape index alarm. Each of these four channels contains detectors consisting of dual-section ion chambers which monitor the axial length of the reactor core at four circumferential positions. They can detect axial flux imbalance conditions as calculated by the TMM axial shape index alarm. Comparison between the channels allows detection of radial flux imbalance. The gain of each power range channel is adjustable to provide a means for calibrating its output against a Plant heat balance.

The system is generally designed in accordance with the following criteria:

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

2. Multiple channels are used in all flux ranges.

3. The channel ranges overlap sufficiently to assure that the flux is continually monitored from source range to 200% of full power.

4. The power range safety channels are classified as 1E and are an integral part of the Reactor Protective System input channels.

5. Each of the power range safety channels is physically separated from the others. Left and right channels of each source/wide range channel are separated from each other.

6. Uninterrupted power is supplied to the system from four separate ac buses. Loss of a channel bus will disable one power range, and in the case of channels C and D, one source/wide range channel.

7. All channel outputs are buffered so that accidental connection to 120 volts ac, or to channel supply voltage, or shorting individual outputs does not have any effect on any of the other outputs.

Source Range Indication - The source range nuclear instrumentation portion uses a pulse signal from a pair of high sensitivity fission chambers. The use of two detector elements within the detector assembly permits high neutron visibility while operating in gamma fluxes up to 200 rem/h. System reliability is improved through the use of integral coaxial detector cables housed in a high-pressure moisture barrier. The output of each detector is amplified and summed in a remotely mounted preamplifier. The pulse signals are also discriminated against gamma pulses and again amplified to drive 300 feet of cable between the amplifier and signal processing drawer in the control room. Here, the pulse input is converted to a signal proportional to the logarithm of count rate. This signal drives a front panel meter, a remote recorder, and a remote meter (all 0.1 cps to 10^5 cps). An audio signal proportional to the count rate is connected to control room and containment loudspeaker. The channel also provides a shaped pulse output for attachment of a scaler.

The source range signal is differentiated to provide rate-of-change of power information (-1 to +7 decades/minute). This rate signal feeds a front panel meter, and a remote meter.

Internally generated pulse signals are available for testing count rate circuitry at predetermined cardinal points. A fixed ramp test signal is available for testing the rate-of-change circuitry.

No automatic protective function is assigned to the source range instrumentation portion.

Wide Range Indication - The wide range nuclear instrumentation portion uses Campbelling techniques and conventional pulse counting techniques to permit the single channel to monitor over 10 decades of flux from $10^{-8}\%$ full power to 200% full power. A single high-sensitivity fission chamber is used as the detecting element. Pulses from the detector pass to a remotely mounted amplifier where they are amplified for transmission to the signal processing drawers.

The amplifier drives approximately 300 feet of cable between the detector and the signal processing drawer in the control room. At the signal processing drawer, the pulse signal is simultaneously applied to two separate detection and amplification circuits. One circuit consists of a pulse counting circuit. The other circuit uses the ac component of the chamber signal rather than the dc component of the signal.

Using Campbell's Theorem, it can be shown that the output of square law detection of the ac portion of a random pulse signal is proportional to the pulse rate (see Reference 3). Because square law detection is used, the smaller gamma pulses produce a very small contribution to the overall signal. Within the mean square portion of the channel, the pulse signal is fed to a band-pass amplifier, a rectifier and filter and a dc log amplifier. The band-pass amplifier and rectifier provide effective square law detection. The output of the pulse counting type circuit is effective over the first five decades. The mean square circuit is effective over the remaining five decades.

By using the two techniques in one channel, a dc signal proportional to the logarithm of neutron flux over approximately ten decades is obtained. This signal drives a front panel meter (10^{-8}% full power to 200% full power), a remote meter, a remote recorder and trip units.

The log level signal is differentiated to provide rate-of-change of power information from -1 to +7 decades/minute. The rate signal feeds a front panel meter, a remote meter and trip units.

Detector high voltage is also monitored by a trip unit which initiates an alarm on decrease of detector voltage or channel trouble.

Channel test and calibration are accomplished by internally generated test signals. Pulse rates controlled by a crystal oscillator, check the count rate portion of the circuitry, and the mean square portion of the circuitry.

A ramp signal is available for check of the rate-of-change circuitry.

Each source/wide range channel contains eight trip units. Operation of the trip units is according to Table 7-4.

The contact output of each trip unit is fed to a single channel of the Reactor Protective System. Thus, with two source/wide range channel portions, a separate rate trip signal is fed to Channels A, B, C and D of the Reactor Protective System. The $< 10^{-4}$% of full power rate-of-change bypass is initiated by the wide-range channel level signal. The level signal is fed to two trip units set to trip above 10^{-4}%. Contacts from each trip unit open above 10^{-4}% to remove the rate trip bypass and enable ΔT power block in TMM via Relay K26 and to remove the zero power manually actuated bypass associated with a single channel (see Subsection 7.2.5.2).

The > 15% full power rate-of-change trip bypass and LPD alarm enable for a particular channel are initiated by a trip unit in the power range safety channel via Relay K25. Above 15% full power, the trip unit resets closing a contact of Relay K25 in parallel with the rate trip contact associated with that channel (A, B, C or D). This method of rate trip bypass permits maximum independence of rate trip channels.

The rate-of-change of power pretrip alarm utilizes a single trip unit (containing two sets of relay contacts) in each wide-range logarithmic channel. Each set of contacts feeds an auxiliary trip unit in one of the channels of the Reactor Protective System. The auxiliary trip unit in turn initiates the control rod withdrawal prohibit signal and pretrip alarm. The signal to the auxiliary trip unit is bypassed below 10^{-4}% and above 15% of full power to avoid spurious alarms and control rod withdrawal prohibits.

Power Range Safety Channels - The four power range channels measure flux linearly over the range of 0% to 125% 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 12 feet. The dc current signal from each of the ion chambers is fed directly to the control room drawer assembly without preamplification. Integral shielded cable is used within the region of high neutron and gamma flux.

The signal from each chamber (lower and upper detectors, Subchannels L and U, respectively) is fed to independent linear amplifiers (Figure 7-9). The output of each amplifier is indicated, compared and summed. The outputs of the L and U subchannels are sent to the thermal margin monitor for calculation of the ASI function. Internal to the drawers, the subchannel signals are summed. Signals are sent to the comparator averager, rod drop detection circuit, remote power level recorder/indicators, critical function monitor, and thermal margin monitor for VHPT, TM/LP and LPD function calculations.

The output from the comparator averager (grand average) is returned to each channel drawer and compared to each channel via two deviation comparators. Variable deviation set points are calculated from the grand average core power in the comparator averager using deviation set point potentiometers. The set point signals are entered in the deviation comparators for alarm setting at two levels. The two levels of deviation are alarmed at the channel drawer and also by remote alarms as percent average core power radial (quadrant) flux tilt, Level I, or Level 2, for operator action to ensure the Technical Specifications limits on radial peaking factors are observed.

The 0%-125% full-scale output of the power range safety channels is fed to the comparator averager which computes the grand average power level of all four channels, and to the trip unit which disables the logarithmic channel rate trip above 16.5% full power which also enables the ΔT power in the TMM. The summing circuit also has an X2 gain selector switch which disconnects the input of one ion chamber and doubles the gain for the other ion chamber to allow full-scale power indication should one ion chamber fail.

Channel calibration and test is accomplished by an internal current source which checks amplifier gain and linearity. A check of the level trip set point is provided by a current signal which is added to the normal detector output.

Each power range channel contains a single bistable trip unit set at 15% power. Operation of the trip units is according to Table 7-5.

Power Monitoring - In addition to panel meters for decades per minute (DPM) and percent power (logarithmic scale) from, respectively, the start-up and wide-range channels, wide-range linear percent power meters are provided fed from the wide-range channels. All the metering equipment is considered as nonsafety related because it is for operator control information only and is isolated from safety-related equipment in accordance with IEEE 384-1977.

7.6.2.3 Control Rod Position Instrumentation

PIP Node - The PIP node measures control rod positions by use of synchros. Outputs are provided for visual display on the main control board and for control rod control.

The major components are:

1. Forty-five control rod position synchros (one per control rod)

2. One node (PIP) containing a VAX computer and an input multiplexer

3. Seven visual displays with seven switches to select control rods within a group

The synchro for each control rod is geared to the control rod drive shaft below the control rod clutch. Full control rod travel corresponds to 264° of synchro rotation. Synchro output is transmitted to the PIP node which scans and converts synchro outputs into inches of control rod withdrawal. The resolution of this system is approximately $\pm$ 0.5 inch.

The PIP, located in the main control room area, performs the following functions:

1. Converts the signal from the synchros to control rod positions and checks these positions against limiting positions

2. Initiates alarms and interlocks under certain limiting control rod positions as detailed in Subsection 7.5.2.1 (control rods at upper and lower control rod stops, regulating control rods at prepower and power dependent insertion limits, 4 inch and 8 inch deviations within a group, and control rod groups out-of-sequence.)

3. Provides contact outputs under other control rod positions as detailed in Subsection 7.5.2.1 (these outputs are used as permissive conditions in the regulating control rod sequencing controls)

4. Provides visual displays of the control rod positions on the main control board

5. Calculates the control rod drop times

The operator normally has two means of displaying control rod positions from the PIP node:

1. Seven visual displays are mounted above the control rod drive controls on the main control console. There is one display for each control rod group; a selector switch at each display will allow position of any control rod in that group to be indicated.

2. The control room workstation. Selected screens on this workstation will display the synchro rod positions.

SPI System - The SPI system ("SPI") consists of the SPI node plus the Host Computer. The SPI node functions as an input multiplexer and all processing of information is done in the Host Computer. The SPI node measures control rod positions by use of control rod-actuated magnetic reed switches. The reed switch stack contains a number of series resistors to form a voltage divider network with reed switches connected at each junction. This stack is attached to the control rod extension housing. A magnet on top of the control rod extension will actuate the reed switches as the control rod moves. The output signal depends on the particular reed switch that is closed. The signal is directly proportional to control rod position. The resolution of the signal is ± 1.5 inches.

The outputs from all reed stacks are sent to the host computer. The host computer performs the following functions related to the control rods:

1. Initiates alarms under certain limiting control rod positions as detailed in Subsection 7.5.2.1 (control rods at upper and lower control rod stops, regulating control rods at prepower and power dependent insertion limits, 4 inch and 8 inch deviations within a group, and control rod groups out-of-sequence.)

The SPI system is completely independent of the PIP node as far as rod monitoring is concerned. If the PIP node were to fail, the SPI system would use the reed stacks in monitoring and processing of rod positions and limits.

Interlocks and Limit Signals - Limit switches independent of either the PIP node or SPI system are provided within the control rod drive mechanism. These switches, which are controlled by cams on the control rod synchro shaft, provide shutdown control rod insertion limit signals (interlock function discussed in Subsection 7.5.2.1) and control rod upper and lower electrical limit signals.

Additional Control Rod Position Indication - Located on a vertical panel immediately behind the main control console, is a group of 45 light displays arranged in a shape corresponding to the control rod distribution. Each display, which represents one control rod, contains four different colored lights. These lights give individual control rod information as indicated in
Table 7-6.

7.6.2.4 Incore Instrumentation

The incore instrumentation consists of a maximum of 43 fixed incore detector assemblies inserted into selected fuel assemblies. Only 43 incore locations out of 45 are available; two locations are reserved for use by the reactor vessel level monitoring system. Each incore detector assembly runs the length of the active core and contains a thermocouple and neutron detectors. Outputs are fed to the SPI node (see Subsection 7.6.2.3) in the control room.

The thermocouples are of Inconel sheathed, Chromel-Alumel construction and are located at the top end of each incore detector assembly so that the primary coolant outlet temperatures may be measured. The neutron detectors in the assemblies are short rhodium detectors equally spaced. These units with their cabling are contained inside a 0.311-inch nominal diameter stainless steel sheath. Sixteen of the detectors are provided with environmentally qualified electrical connectors and cabling inside containment to provide increased reliability of the thermocouple readout for monitoring the potential approach to inadequate core cooling conditions. The readout from these thermocouples goes through the Cutler-Hammer (CFMS) multiplexer and not the SPI node. Assemblies are inserted into the core through eight instrumentation ports in the reactor vessel head. Each assembly is guided into position in an empty fuel tube in the center of the fuel assembly via a fixed stainless steel guide tube. The seal plug forms a pressure boundary for each assembly at the reactor vessel head. 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 rhodium detectors are 40 cm long and are provided to measure flux at several axial locations in fuel assemblies. 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 information received by the SPI node is forwarded to the Host Computer. The host computer is where the processing of the incore information occurs. The host:

1. Corrects the raw incore values for detector burnup. (A detector background correction is made in the SPI node.)

2. Compares these corrected values to preset alarm limits. This comparison facilitates the monitoring of reactor core radial peaking factors, quadrant power tilt, and linear heat rate.

3. Initiates an alarm if the limits are exceeded.

Verification of incore channel readings and identification of inoperable detectors are done by correlation between readings and with computed power shapes using a computer program. The incore alarm system operability can be monitored through the SPI trouble alarm on the main control room panels and an alarm on the workstations.

Quadrant power tilt and linear heat rate can be determined from the excore nuclear instrumentation (Subsection 7.6.2.2), provided they are calibrated against the incore readings as required by the Technical Specifications. Quadrant power tilt calibrations of the excore readings are performed based on measured quadrant power tilt calculated using a computer program which determines tilts based on symmetric incore detectors and/or the integral power in each quadrant of the core (Reference 20). Linear heat rate calibration of the excore readings involves two intermediary parameters, axial offset and allowable power level, which can be determined by the incore readings. The Technical Specifications give limits on these parameters above a certain reactor power level to ensure that the core linear heat rate limits are maintained while using the excore instruments.

7.6.2.5 Palisades Plant Computer

System Layout - The plant computer consists of four intelligent input nodes, one direct connected multiplexor, multiple display workstations, printers and interconnecting hardware. The plant computer is a distributed system which communicates via Ethernet. There are separate Ethernet cabling systems for the Input nodes and for the Man Machine Interfaces.

The Man-Machine-Interfaces are Computer Workstations. At the very least, there are workstations in the Control Room, TSC, and EOF. The host computer in the CFMS trailer distributes all database and display information to the workstations. These workstations maintain a local copy of the database and displays in order to off-load the host. Page printers are located in the Control Room (CR), TSC, and EOF for prints of the workstation screens and reports from the host computer.

Four input nodes, PIP, SPI, SOE, and CTCS, are combinations of an input multiplexor and a computer. These nodes perform input processing including Analog to Digital Conversion, Sequence of Events time-stamping, and engineering units conversion. This processed data is assembled and passed to the Host computer. The host computer in turn performs alarm processing, event logging, historical recording and database distribution functions based on this data. Two nodes, the PIP and CTCS nodes, perform additional software tasks such that control rod monitoring and Cooling Tower Fans can be operated independent of host computer operability. The host computer runs several custom software modules such as CFMS processing, Incore monitoring, Rod monitoring, ERDS, Meteorological computer interface, and calculated point processing.

Identification of the PPC components and general location is shown in Figure 7-65. The host computer interfaces, for ERDS, Meteorological, EOF, and the backup alarm printer, are located in the CFMS trailer on the turbine deck. The communications hubs and the SOE node are located in the Cable spreading room below the control room. The control room has at least one permanently located workstation and several receptacles where portable workstation(s) can be connected. A page printer is located here. The PIP, SPI, and CTCS nodes are located in the control room also. The Cutler-Hammer input multiplexor is also located in the control room and communicates back to the CFMS trailer directly.

The power supply for the PPC host computer and SOE node includes a 125 volt dc subsystem (one battery, two chargers and one distribution panel) and a dc-to-ac conversion subsystem (two inverters, two static switches) with bypass transformers. Power is taken from the 480-volt MCCs 3 and 4. Only those components required to maintain minimal PPC functionality to the Control room, TSC, and EOF are powered from this system. Extra workstations and non-essential devices are powered from lighting panel power. The CTCS node is powered from the Instrument AC panel Y-01, while the PIP and SPI nodes are powered from the Preferred AC panels Y-20 and Y-40, respectively.

Interfaces - The Reactor Protective System is monitored by the SOE node. The interfaces are both analog and digital. Refer to Subsection 7.2.9.2 for details. Interfaces with the engineered safeguards controls and the Class 1E electrical distribution system are exclusively digital. They are provided via relay contact inputs from these controls, thus ensuring adequate electrical isolation as required by IEEE 384-1977 and 10 CFR 50, Appendix A, GDC24. Interfaces with the reactor shutdown control, and auxiliary feedwater controls are also exclusively digital via relay contacts. Interface with the fluid systems protection is via relay contact to the Data Logger for PRV-1043B and by direct connection from the valve indicating light to the Data Logger for PRV-1042B.

Interfaces with non-safety-related systems (regulating controls, primary and secondary plant process and Nonclass 1E electrical distribution) are both digital and analog. They do not require any special isolation means.

The PPC is comprised of reliable electronic gear fed from an uninterruptible type of power supply. Being a Nonclass 1E system, all safety systems interfaces have isolation means in accordance with IEEE 384-1977 and GDC24 either via relay coil-contact isolation or qualified electronic isolators. As described in Section 5.2, components located in the CP Co Design Class 1 portion of the auxiliary building (the PPC cabinets in the cable spreading room, and certain power supply subsystem components in switchgear room 1D), have been qualified as Seismic Category I (Section 5.7). The system battery enclosure in switchgear room 1D is equipped with a hydrogen evacuation system designed to provide a scavenging rate which precludes the formation of an explosive concentration.

The CFMS method or design was carried over from the stand alone CFMS replaced in 1995 into the User interface of the new PPC. The principal software function of the CFMS is to provide concise displays of Plant data, provide for trending of input data and to provide for historical data storage and retrieval. This information is available to system users at each of the various workstations. Access to the information is provided through workstations using keyboard, pointing device, and CRT. The CFMS software design provides a hierarchy of displays showing the status of the Plant's critical safety functions. The hierarchy starts with a top-level display showing individual boxes that give an indication of the status of each critical safety function. Lower-level displays give system overviews with current values of important process variables and more detailed mimic diagrams showing system line-up and indicating variables that are in alarm state by use of color of component symbols or variable values. Displays such as the Critical Function Matrix, event and alarm log, trends and others can be accessed with dedicated function keys on the keyboard. A small representation of the Critical Functions Matrix is visible from every display and indicate the overall status of each critical function.

The PPC provides historical storage and retrieval of Process data in order to assist plant personnel in process trending and post-trip or transient recreations. Historical data can viewed in the form of real-time trends, X-Y plots, and statistical reports. Historical data can be archived to disk or tape for later viewing. Sequence of events logs are also archived.

In addition, the PPC is data linked to the NRC's Emergency Response Data-link System (ERDS) via a NRC supplied modem and phone line. This data link is capable of sending a preselected group of PPC input variables to the NRC at a rate of once per 15 seconds.

Additional information on the PPC/CFMS is provided in References 8 and 9.

7.7 OPERATING CONTROL STATIONS

7.7.1 GENERAL LAYOUT

The operating control stations consist of the control room for centralized control during start-up, normal, shutdown and emergency operations; and auxiliary stations for emergency operation of the engineered safeguards and shutdown systems, normal operation of the radwaste system and normal operation of miscellaneous noncritical systems.

Radiation and shielding design of the spaces in which operating personnel occupancy is required, including adequate access to the vital areas for control of the Plant during and after an accident, has been provided. Refer to Chapter 11 for details of radiation zones.

An onsite technical support center (TSC) is located just outside the control room in an extension to the auxiliary building, provided with dedicated communications with the control room and other centers identified in the Palisades Emergency Plan, including a dedicated display terminal from the critical functions monitor (Subsection 7.6.2.6). The intent of this TSC is to meet NUREG-0696.

7.7.2 CONTROL ROOM

The control room is accessible from the auxiliary and turbine buildings and houses the control console, vertical duplex boards, cooling tower boards, and switchyard supervisory boards for operation and monitoring of all critical systems. The control console consists of three sections arranged as follows:

1. Center Section - Control devices, indicators and recorders, reactivity regulation, primary coolant components, shutdown cooling and Chemical and Volume Control System.

2. Left Section - Control devices, indicators and controllers for the Engineered Safeguards Systems, Service Water System, Component Cooling System and Containment Air Cooling Systems.

3. Right Section - Control devices, indicators, recorders and controllers for the main turbine and generator and feedwater control systems.

The vertical board is a totally enclosed walk-in panel. One part of the vertical board is arranged in three connected sections with equipment layout and physical separation similar to the control console. The equipment on the front of the vertical panel is located directly behind the console section having devices for that same system. Other control and monitoring equipment for other systems is located on the back of this board and on separate vertical duplex boards. The annunciators for the entire Plant are located across the top of the vertical panels providing visual and audible indication of off-normal conditions requiring operator action.

A program to review the human factors of the control room was initiated in late 1980 in response to NUREG-0660 and NUREG-0737, Supplement 1, and consistent with the guidance provided by NUREG-0700. The objective, as stated in NUREG-0660 was to improve the ability of control room operators to prevent accidents or to cope with them, should they occur, by improving the information provided to them.

Short-term improvements were completed during the 1983 refueling outage while long-term improvements were scheduled for subsequent outages. On September 14, 1989 the NRC issued a final SER on the program. Based on review of the CP Co Summary Report submitted on August 29, 1986 and onsite inspections, the SER concluded that the Palisades Plant meets all the requirements of Supplement 1 to NUREG-0737 for the Detailed Control Room Design Review.

Control console and panel sections listed in Table 5.7-8 and containing Class 1E devices are designed to withstand seismic loads described in Section 5.7. All other panels and equipment in the control room have been anchored to stay in place during a seismic event (see Section 5.10).

The control room is located in the CP Co Design Class 1 portion of the auxiliary building. Sufficient concrete shielding is provided to ensure safe occupancy of the control room during all normal and abnormal Plant conditions. The control room atmosphere is air-conditioned for personnel comfort and equipment cooling using redundant HVAC with Class 1E controls (see Section 9.8). The ventilation system is arranged for outside makeup air to be drawn through absolute and charcoal filters in the event of a DBA. The control room area is provided with an area radiation monitor. Temperature monitors are provided to warn high temperature condition in the vicinity of the Reactor Protective System and Engineered Safeguards System panels to alarm at 110°F since these systems are designed for 120°F ambient.

The present Thermal Margin Monitors (TMM) were originally qualified to 131°F. However, the location in the panel requires fan cooling. Analysis shows that (with forced-air cooling) 131°F is reached by the TMM when the control room ambient temperature is 106°F. Because the TMM portion of the RPS is no longer capable of operating at 120°F, an administrative limit of 90°F was imposed (Reference 13).

The only time that control room temperature is postulated to exceed 90°F would be during a station blackout when temperatures may approach 120°F (Reference 14). Under these conditions, the RPS would not be required to be operable. See Chapter 1 for additional information on station blackout.

All materials of construction used in the control room are noncombustible. Electrical wiring is flame-retardant as proven by applicable vertical flame tests.

Control Room Protection Against Fire

Cables related to safety-related control cabinets and consoles including all the systems required for Plant shutdown penetrate the floor directly into control panels and consoles.

The combustibles in the area consist mainly of electrical wiring insulation contained within the cabinets and a small amount of ordinary combustibles such as paper. An unmitigated fire in one of the control panels would probably be limited to one panel involving only one division of safe shutdown equipment due to the low combustible loading and the physical separation and barriers provided. In the event a larger fire should occur, capability to achieve a safe shutdown includes required instrumentation and centralized controls (Panel C-150) independent of the control room and not affected by a fire in the control room.

In addition, smoke detectors are installed in the walk-in cabinet enclosures and throughout the ceiling of the control room and adjacent offices. Smoke detectors are not installed in the main control consoles since they are designed with outlet ventilation openings at the top which will allow the control room operators to observe any smoke present from a fire.

Other fire protection features are described in Section 9.6.9 for this area.

The control room ventilation system can be controlled manually by the control room operator and will allow shutoff of the fans if smoke was observed entering the control room. Manual operation of the ventilation system for venting the control room is also available.

All cable penetrations into the control room have been sealed with flame-retardant material.

There are no concealed floor or ceiling spaces that contain cables in the control room (area over panels, only).

The control room is not used as a cable right of way.

In light of the above fire protection features, Consumers Power Company has received an exemption from the NRC for the requirement to have a fixed fire suppression system in the control room per 10 CFR 50, Appendix R, Section III G.3.

7.7.3 ENGINEERED SAFEGUARDS AUXILIARY PANEL (C-33)

The engineered safeguards auxiliary panel is located in the auxiliary building. This panel provides a second point of control, outside the control room, for reactor shutdown and contains devices to permit the following emergency operations in the event the control room must be evacuated:

1. Operation of auxiliary feedwater control valves.
2. Monitoring steam generator level.
3. Monitoring pressurizer pressure and level.
4. Operation of boric acid tank.
5. Operation of atmospheric steam dump valves.
6. Initiation of shutdown cooling by control of corresponding valves.
7. Control of component cooling water and service water.
8. Control of the motor-operated valve in the line from the SIRW tank to the suction of the charging pumps. This provides a large backup supply of shutdown concentrated borated water to the Primary Coolant System.

Engineered Safeguards Auxiliary Panel Area Protection Against Fire

The engineered safeguards auxiliary panel is not functional if all cable spreading room or control room equipment is inoperable due to fire. Panel C-150 is provided for such an emergency. The control function available from Panel C-33 is limited to that of positioning of valves. Starting and stopping of motors could be accomplished at the switchgear for each motor. The panel control devices are wired in parallel with the control room devices and thus the effects of control circuit damage would be common to either location. However, a fire damaging the circuitry of one channel would not prevent the ability to control equipment from the other channel from either control location.

Other fire protection features are described in Section 9.6.9 for this area.

7.7.4 AUXILIARY HOT SHUTDOWN CONTROL PANELS (C-150/C-150A)

In order to ensure use of sufficient components of the Auxiliary Feedwater System and sufficient process information to permit reactor hot shutdown control in the event a fire damages equipment and circuitry of the main feedwater system or the Auxiliary Feedwater System in the control room, cable spreading room, Engineered Safeguards Auxiliary Panel C-33 room, or the corridor between Switchgear Room 1-C and the charging pump rooms, Auxiliary Hot Shutdown Control Panels (C-150/C-150A) have been provided and located in the southwest electrical penetration room. These panels are comprised of two enclosures, the main enclosure C-150 and an auxiliary one called C-150A. The description below combines these two enclosures into one entity called "Panel C-150."

From this panel, control of the auxiliary feedwater valves is enabled by transfer (see Figure 7-54 for this transfer system) and control of auxiliary feedwater turbine steam supply Valve B. Indication of auxiliary feedwater flow to both steam generators, water level of both steam generators and pressurizer level are enabled by transfer. In addition, primary coolant pressure (pressurizer pressure) is displayed by a primary sensor dedicated to this use. Transfer of the above-mentioned systems is annunciated in the control room. See Subsection 7.4.1 for operation via Panel C-150.

Equipment controls that are required by the alternative dedicated method of achieving and maintaining hot shutdown (Subsection 7.4.1) are as follows:

1. Auxiliary feedwater valves
2. Auxiliary feedwater pump turbine-driven steam Valve B

Instrumentation systems displayed on the Auxiliary Hot Shutdown Control Panel are:

1. Source range flux monitor
2. Auxiliary feedwater flow
3. Pressurizer pressure
4. Pressurizer level
5. Steam generator level and pressure
6. Primary coolant temperatures (hot and cold legs)
7. Turbine-driven auxiliary feedwater pump low-suction pressure warning light
8. SIRW tank level

Equipment and equipment housings procured for and utilized to take the Plant to safe shutdown have been specified to be qualified in accordance with IEEE 344-1975 and IEEE 323-1974 when placed within or interfacing with safety systems.

Switches, which transfer control or instrument functions from the control room to the auxiliary shutdown control panel, alarm in the control room when the devices in the Auxiliary Hot Shutdown Control Panel are enabled.

The transfer switches of the Auxiliary Hot Shutdown Control Panel provide access to the Auxiliary Feedwater System for hot shutdown only. No other means of achieving hot shutdown exists if a fire damages the control room or the cable spreading room.

Wiring, including power sources for the control circuit and equipment operation for the alternate shutdown method, is independent of equipment wiring in the postulated fire areas.

Alternate shutdown power sources, including all breakers, have isolation devices on control circuits that are routed through the postulated fire areas, even if the breaker is to be operated manually.

Procedures are provided for taking the Plant to hot shutdown via the Auxiliary Hot Shutdown Control Panel in the event a fire prevents use of the control room.

Spare fuses are available for control circuits where fuses may be required in supplying power to control circuits used for the alternate shutdown method and may be blown by the effects of a cable spreading room fire. The spare fuses are located convenient to the existing fuses. The shutdown procedure informs the operator to check these fuses.

Testing - Periodic testing of the auxiliary shutdown system will be in accordance with Regulatory Guide 1.22 and IEEE 338.

7.7.5 RADWASTE SYSTEM LOCAL CONTROL PANEL

The Radwaste System control panels are located in the auxiliary building and are accessible through the Access Control. These panels contain all the control and monitoring devices to initiate, control and monitor the Radwaste System components.

7.7.6 MISCELLANEOUS LOCAL CONTROL STATIONS

Miscellaneous noncritical systems are controlled from local stations throughout the Plant.

Off-normal conditions in all systems are alarmed on the respective local panel and on the main control panel annunciators.

7.7.7 FEATURES WHICH ENHANCE SAFE OPERATION

Steel fire barriers are incorporated in the main control console, main control panel, and engineered safeguards auxiliary panel. The barriers are arranged to separate control circuits similar to the separation of equipment power supplies.

All panels and consoles are totally enclosed and constructed from steel plate welded to steel frames. Fire retardance and rigidity for mounted equipment and cabling is provided by this construction.

7.7.8 IN-PLANT COMMUNICATION SYSTEM

The in-plant communication system is comprised of five subsystems: a telephone system, a public address system, an intercommunications system, a sound-powered phone system and an onsite/offsite radio system.

Paging over a Plant-wide public address system is initiated by dialing a preselected number from any of the 400 telephone stations. Plant-wide paging can also be accomplished by a microphone in the control room. Paging from the control room station overrides paging from any other station.

An intercommunications system permits common talking between the control room, the fuel pool area and inside containment as an aid in the fuel handling operation. A closed circuit television system with a screen in the control room aids in monitoring fuel handling.

The sound-powered phone system is intended for equipment test and calibration which require vocal communication between the equipment and control room. It can be used also by personnel in the Technical Support Center during an emergency.

The onsite/offsite radio system includes a base station in the turbine building, a remote base station in the control room for onsite communication with portable transceivers. A repeater unit is located onsite for use by security personnel.

Communications between operators performing manual actions and those providing feedback are not a requirement but rather an enhancement to aid in bringing the plant to either hot or cold shutdown conditions.

7.7.9 OUT-OF-PLANT COMMUNICATION SYSTEM

In addition to the telephone system which provides normal external communications for the Plant, there is a power controller radio system.

Additional offsite communications are provided by the onsite/offsite radio system. This system has offsite communications with the Emergency Operations Center in South Haven and with a repeater in the site emergency vehicle.

Direct communications with the Michigan State Police in South Haven are by means of a base station located in the secondary security alarm station. A second remote station is located in the central security alarm station.

All emergency communications are addressed in the Emergency Plan.

7.8 QUALITY CONTROL

For a discussion of the Quality Assurance Program, see Chapter 15.

7.8.1 SPECIFICATIONS

The following tests and inspections were performed as a minimum by the Reactor Protective System supplier:

1. Surface examination
2. Isolation requirements
3. Point-to-point continuity test
4. Insulation test

Upon completion of the above production tests, the following operation tests were performed:

1. Bistable trip unit operating tests
2. Bistable trip unit environmental tests
3. Auxiliary trip unit operating tests
4. Auxiliary trip unit environmental tests
5. Reactor protection system operating tests

The following standards were incorporated in whole or in part in the specifications:

National Bureau of Standards TID-20298 - Standard Nuclear Instrument Modules

IPCEA S-61-402 (NEMA WC-5-1961)

NEMA IC1-2.42

EIA RS-310, Racks, Panels and Associated Equipment

7.8.2 SUPPLIER'S QUALITY CONTROL

The quality control procedures as required by the applicable engineering specification of all vendors to Combustion Engineering Nuclear Power Department (CENPD) were reviewed by the Combustion Engineering (CE) quality assurance organization. The functions and organization of CE quality assurance organization were as given in Chapter 15.

All major components of the Reactor Protective System fabricated by outside vendors were treated in the manner described in the above reference. As such, and as detailed in the above reference, the quality control procedures of the various RPS vendors mentioned above were the responsibility of the manager of quality control of the manufacturing services department.

7.8.3 REACTOR PROTECTIVE SYSTEM SHOP TEST

The complete RPS was tested at the vendor's plant prior to shipment to the jobsite. This test was witnessed and approved as satisfactory by members of the design group of the CENPD. The test was also witnessed by Consumers Power personnel.

The purpose of the test was as follows:

1. To assure that the RPS functions as designed throughout <u>all</u> modes of operation prior to shipment
2. To assure that the nuclear instrumentation system operates properly and is properly interfaced with the RPS

Summary of Tests:

1. Trip Logic Test
2. Test Capability Check
3. Trip Inhibit Test
4. Nuclear Instrumentation Operability Test
5. Nuclear Instrumentation Interface Test
6. Zero Power Mode Test
7. Low Flow Protection Test
8. Miscellaneous Features Test
9. Bus Failure Test
10. Ground Fault Detection Test
11. System Response Time Test
12. Power Demand and Design Data Test
13. Temperature Rise Test

The above tests were performed in accordance with test procedures approved by CE.

7.8.4 SHIPPING AND STORAGE

The cabinet assembly was skid mounted (suitable for sling hoisting) for shipping and was packaged such that it would withstand:

1. Normal handling and weather during shipping without suffering damage

2. Unforeseen prolonged storage conditions in nonair-conditioned warehouse

7.8.5 RELIABILITY

The CE reliability group performed an independent review and analysis of the Reactor Protective System (RPS) during the design phase of the program. A mathematical model of the system was generated and a reliability review followed using the model as a basis for numerical predictions. This consisted of identifying the major reliability components of the design, calculating the electrical operating stresses of each critical item at the environment specified, and calculating the reliability prediction of the bistable trip unit and RPS.

Failure mode effects and sensitivity analysis were performed on each major item to assess the design approach used by the designer and to reveal all areas of potential reliability jeopardy and safety hazards. The results of this analysis highlighted those areas earmarked as critical and effected closer examination of the design by the responsible design engineer into those areas.

Parts and stress application analysis was performed on the electronic components to ascertain proper selection of intended parts, to review their usage in the design, and to assure that each item is properly derated in its application for reliability purposes. A separate review of each applicable detailed drawing and specification for the trip unit BTU and RPS was conducted. Particular emphasis was placed on details for design interfaces and adequacy of controls to assure proper compliance to detail requirements.

7.8.6 RECORDS AND CERTIFICATION

The following records were kept:

1. Trip unit operating and environmental test results

2. Supplier's inspection and test reports

3. Records of all operational tests performed on the system

In addition, complete records were kept on the extensive testing performed on the prototype trip unit developed in the CE laboratories which is identical to the trip units used in the Reactor Protective System.

The following certifications of satisfactory fabrication and test performance were made:

1. Certification that the reactor protection system has been fabricated in accordance with the applicable engineering specifications and references

2. Certification of satisfactory completion of all tests and inspections

7.8.7 FIELD QUALITY CONTROL

Upon receipt of equipment in the field, inspection was made for conformance to specifications, codes and drawings. Checks were made for completeness of shop inspection and material certification if required by specification and/or damage in transit. A record of this inspection was made on a special form. If deficiencies were present, they were noted on the forms and sent to the Job Engineer for correction. Copies of the forms were filed with the Quality Assurance Engineer and the Project Engineer. If the item was not immediately set in its final position, a notation was made on the form regarding the storage of the item and the protection provided. Field storage was determined by standard construction practices supplemented by any special requirements specified by the manufacturer. Periodic site inspections were made of equipment to ensure that proper storage procedures were being maintained.

The Consumers Power construction personnel made an independent check of equipment as it arrived onsite and observed proper handling and storage until it was placed. After installation, continual checks were made to ensure that it was properly protected until ready for initial operation.

When the item was set or installed, inspection was employed to confirm that electrical and instrumentation items were properly set and connected. A record of this inspection was made on a special form. Any deficiencies were noted on the form for correction in a similar manner as when the equipment was received. A visual checkout of wiring was made by System Protection and Laboratory Services Department personnel to assure that it was wired in accordance with the connection diagrams.

During the construction phase of installing cable and wiring, routine checks were made by both Bechtel and Consumers Power to assure that proper cableway routing and separations were being maintained as required for the protective and engineered safeguards systems. This included verification of separate penetration areas for these systems which enter the containment building.

Prior to preoperational testing, function tests of all electrical controls and instrumentation were performed by System Protection and Laboratory Services Department personnel and the Plant Instrumentation Group. This functional testing verified that with the equipment energized, it functioned as it should with an overall checkout being made from sensor through control and output device. In addition, instruments were calibrated during the functional test including most sensors although in some cases, sensors were calibrated prior to installation and this was considered adequate. These calibrations and system tests were recorded on special forms.

During the preoperational testing program, the correct functioning and calibration of all control systems was given a final verification by performance of a preoperational test on all Plant systems. The test procedures for each of these systems defined all instrumentation and control functions in the respective systems and the observed response was noted on the test forms. Any deviations from design specifications were noted on the forms and these were corrected prior to considering the test complete.

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TABLE 7-1

REACTOR PROTECTIVE SYSTEM RELAYS

Contact	Function	Contact	Function
K1-1	Data Logger	K2-1	Data Logger
K1-2	Rod Rundown(a)	K2-2	Rod Rundown(a)
K1-3	Trip Reset(b)	K2-3	Trip Reset
K1-4	Trip Reset	K2-4	Trip Reset
K1-5	Diesel Generator Preload Shed (1-1)	K2-5	Diesel Generator Preload Shed (1-2)
K1-6	Trips Turbine	K2-6	Trips Turbine
K1-7	Trip Reset Lockout(c)	K2-7	Trip Annunciator
K1-8	CFMS	K2-8	CFMS
K3-1	Data Logger	K4-1	Data Logger
K3-2	Rod Rundown	K4-2	Rod Rundown
K3-3	Trip Reset	K4-3	Trip Reset
K3-4	Trip Reset	K4-4	Trip Reset
K3-5	Diesel Generator Preload Shed (1-1)	K4-5	Diesel Generator Preload Shed (1-2)
K3-6	Trips Turbine	K4-6	Trips Turbine
K3-7	Trip Reset Lockout	K4-7	(Spare)
K3-8	CFMS	K4-8	CFMS

(a) Rod Rundown - The control rods receive a "rods in" signal following a reactor trip which causes any rod with a "stuck" clutch to be driven to the bottom of the core.
(b) Trip Reset - The reactor trip reset push button must be depressed to permit reactor start-up following a trip.
(c) Trip Reset Lockout - The reactor trip cannot be reset within 30 seconds following a reactor trip. This function prevents the reactor trip being reset while the control rods are still descending following a reactor trip.

TABLE 7-2

SAFE SHUTDOWN INSTRUMENTS

Refer to Section 9.6.8 for information concerning post-fire safe shutdown instruments.

TABLE 7-3

REGULATING RODS WITHDRAWAL INTERLOCKS

Withdrawal Prohibit Conditions	Manual Individual Control Mode	Manual Sequential, or Group Control Mode
Pretrip Overpower	X	X
High Start-up Rate (Between 10^{-4}% and 15% Full Power)	X	X

TABLE 7-4

SOURCE/WIDE RANGE NUCLEAR INSTRUMENT CHANNEL TRIP UNIT ACTIONS

Trip Unit	Input Signal	Action	Approximate Set Point
1	Low High Voltage	Audible and Visible Alarm	410 vdc Decreasing HV
2	Log Power Level	Bypass Rate-of-Change of Power Trip	$< 10^{-4}$% Full Power
		Disable Zero Power Mode Bypass (Effective for One Protective Channel)	$> 10^{-4}$% Full Power Enable ΔT Power Block in Thermal Margin Monitor
3	Log Power Level	Bypass Rate-of-Change of Power Trip	10^{-4}% Full Power
		Disable Zero Power Mode Bypass (Effective for One Protective Channel)	$> 10^{-4}$% Full Power Enable ΔT Power Block in Thermal Margin Monitor
4	Log Power Level	Spare	-
5	Rate-of-Change of Power	Pretrip Signal and Rod Withdrawal Prohibit (Effective for Two Protective Channels)	1.5 Decades/Minute (Bypassed $< 10^{-4}$% and > 15%)
6	Rate-of-Change of Power	Trip Signal to Reactor Protective System (Effective for One Protective Channel)	2.6 Decades/Minute (Bypassed $< 10^{-4}$% and > 15%)
7	Rate-of-Change of Power	Trip Signal to Reactor Protective System (Effective for One Protective Channel)	2.6 Decades/Minute (Bypassed $< 10^{-4}$% and > 15%)
8	Rate-of-Change of Power	Spare	-

TABLE 7-5

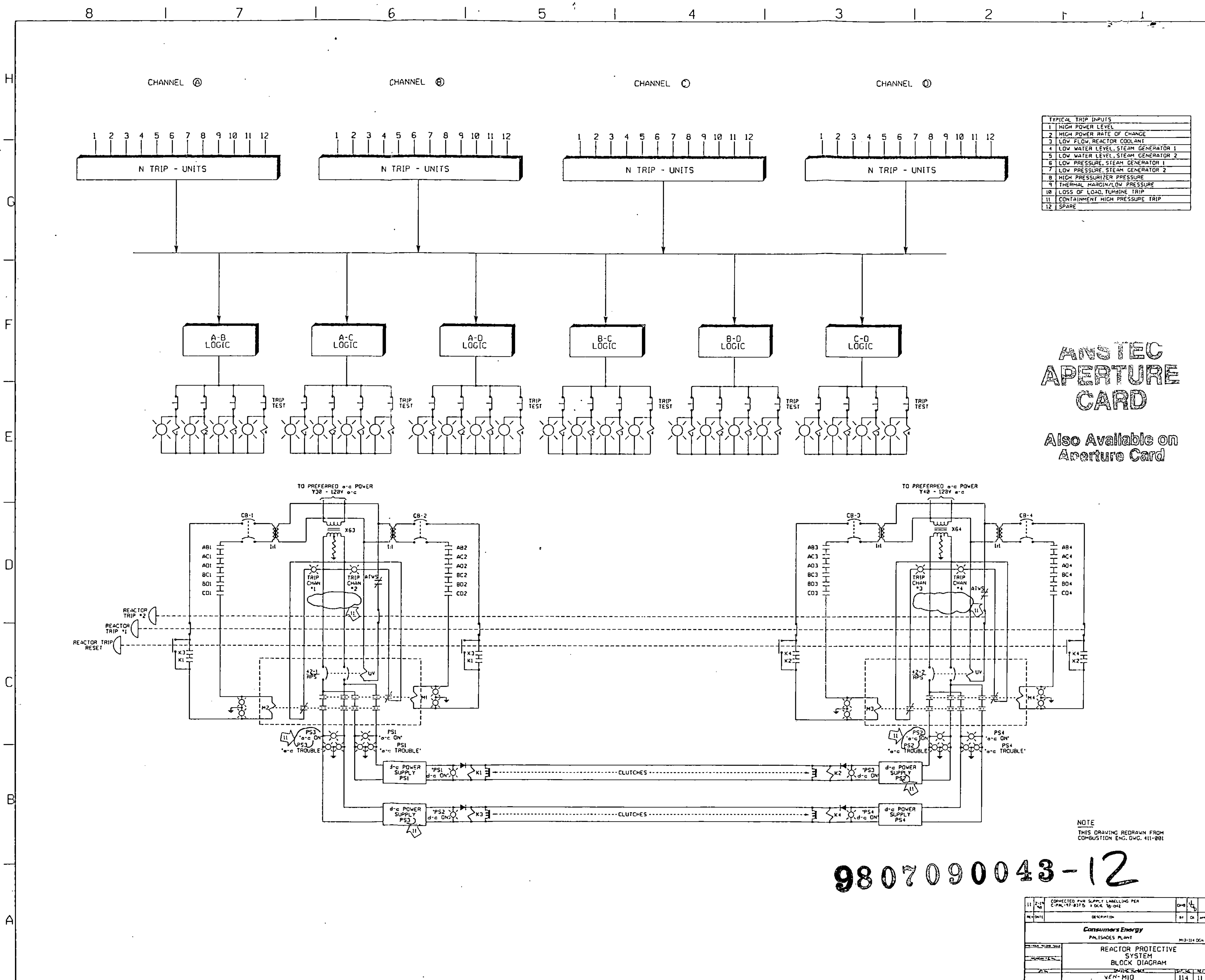
POWER-RANGE SAFETY CHANNEL TRIP UNIT ACTIONS

Trip Unit	Input Signal	Action	Approximate Set Point
1	Detector Voltage, Module Interlock, Operate Calibrate Switch	Audible and Visible Alarm	3.5% Below Normal Operating Voltage
2	Power Level	Spare	
3	Power Level	Spare	
4	Power Level	Spare	
5	Power Level	Spare	
6	Power Level	Spare	
7	Power Level	Spare	
8	Power Level	Rate Trip Inhibit to Logarithmic Channel	>15% Full Power
		Enable ASI Alarm	>15% Full Power
		Bypass Loss-of-Load Trip	<15% Full Power

TABLE 7-6

CONTROL ROD POSITION LIGHT MATRIX

	Control Rod Position		
Color	Shutdown Control Rods	Regulating Control Rods	Part-Length Control Rods
Green	Lower Electrical Limit	Lower Electrical Limit	Lower Electrical Limit
White	Between Shutdown Control Rod Insertion Limit and Lower Electrical Limit	Between Upper Control Rod Stop and Lower Electrical Limit	Moving
Amber	NA	Between Upper Control Rod Stop and Upper Electrical Limit	Between Upper Control Rod Stop and Upper Electrical Limit
Blue	Between Upper and Shutdown Control Rod Insertion Limit	NA	-
Red	At Upper Electrical Limit	At Upper Electrical Limit	At Upper Electrical Limit



REV	DATE	DESCRIPTION	BY	CK	APP
11	2-19-98	CORRECTED PWR SUPPLY LABELLING PER C-PAL-97-0378 & DCR 98-012	DHB		

Consumers Energy
PALISADES PLANT
M10-114 DGN

REACTOR PROTECTIVE SYSTEM BLOCK DIAGRAM

VEN-M10 114 11

FIG 7-1
FSAR REV 20

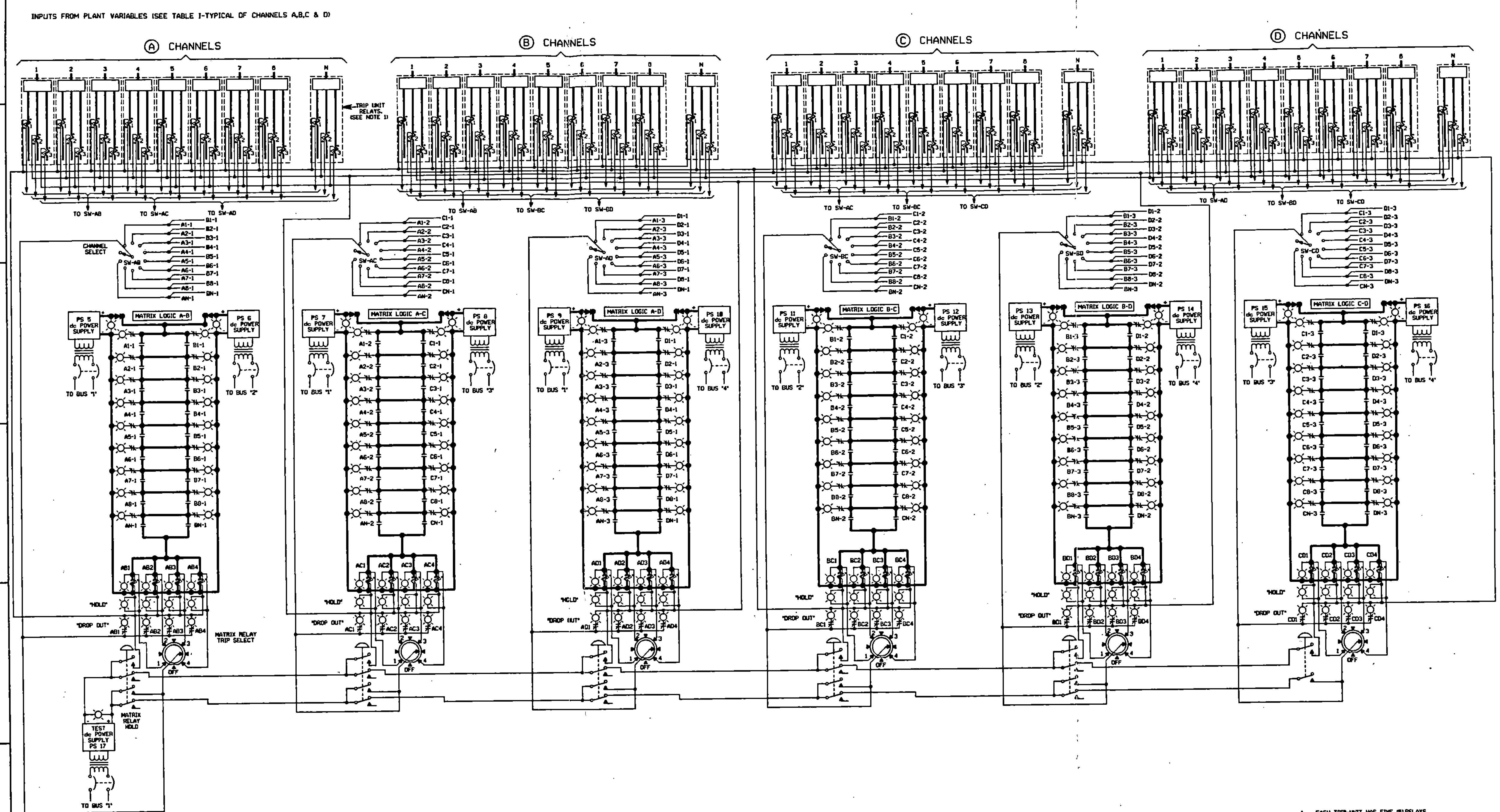


TABLE I	
TYPICAL TRIP INPUTS	
1	HIGH POWER LEVEL
2	HIGH POWER RATE OF CHANGE
3	LOW FLOW, REACTOR COOLANT
4	LOW WATER LEVEL, STEAM GENERATOR 1
5	LOW WATER LEVEL, STEAM GENERATOR 2
6	LOW PRESSURE, STEAM GENERATOR 1
7	LOW PRESSURE, STEAM GENERATOR 2
8	HIGH PRESSURIZER PRESSURE
9	THERMAL MARGIN/LOW PRESSURE
10	LOSS OF LOAD, TURBINE TRIP
11	CONTAINMENT HIGH PRESSURE TRIP
12	SPARE

TABLE 1	
MATRIX RELAY DESIGNATION	
DESIGNATION ON DWG. NO. 2966-E-2855 AND 2966-E-2859	DESIGNATION ON DWG. NO. 2966-E-2858
K1	AB1 THRU CD1
K2	AB2 THRU CD2
K3	AB3 THRU CD3
K4	AB4 THRU CD4

1. EACH TRIP-UNIT HAS FIVE (5) RELAYS, THREE (3) AS SHOWN FOR TRIPPING LOGIC MATRICES A-B THRU C-D, ONE (1) FOR REMOTE PRE-TRIP ALARM ANNUNCIATION, ONE (1) FOR REMOTE TRIP ALARM ANNUNCIATION AND FOR SEQUENTIAL EVENTS RECORDING.
2. ALL CONTACTS SHOWN FOR RELAYS IN DE-ENERGIZED CONDITION.

9005080040-08

SI APERTURE CARD

Also Available On Aperture Card

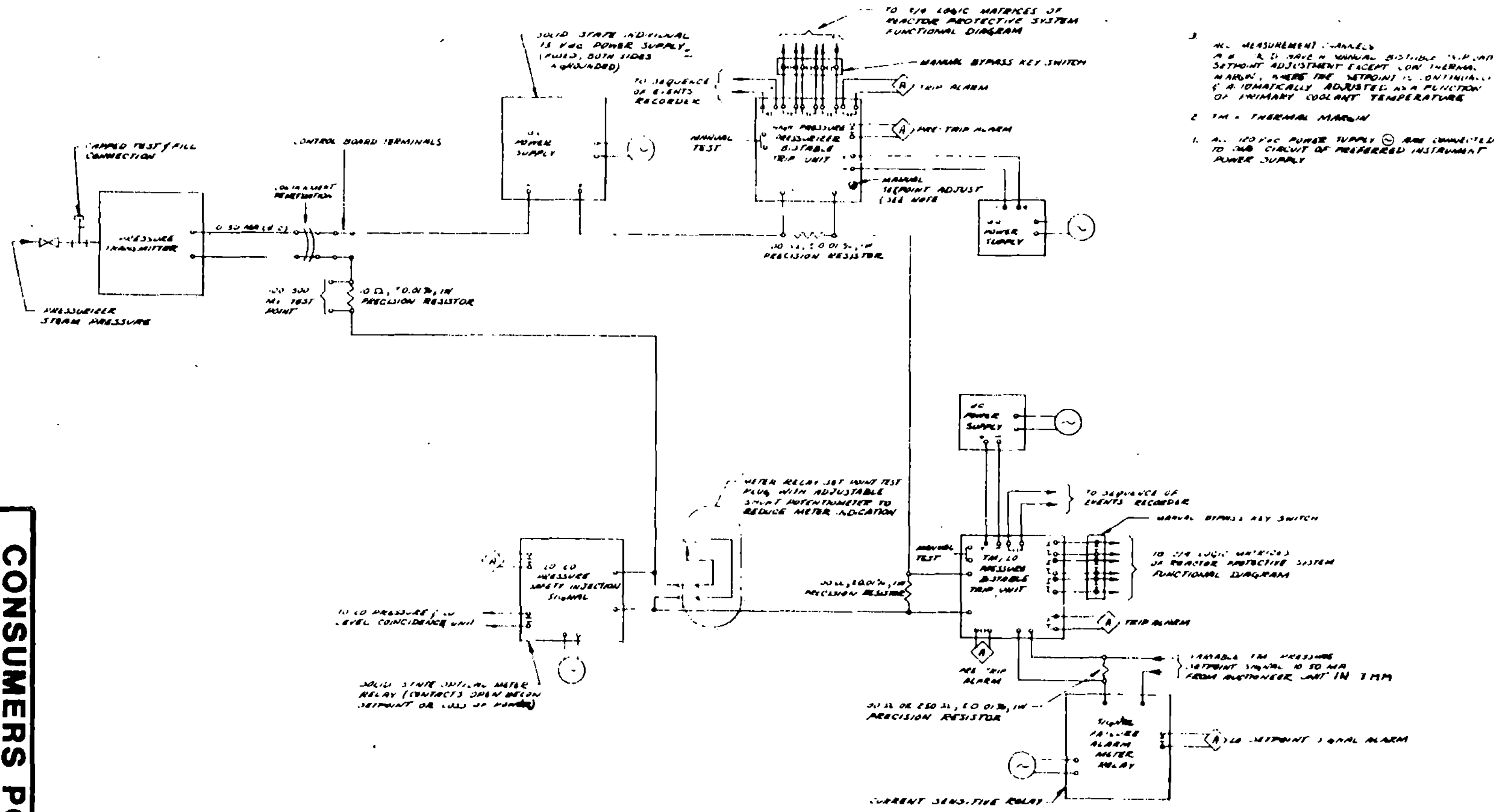
NO	DATE	REVISIONS	BY	CH'D	APP	ENG'R
F	6/10/84	RE-NUMBERED TO SHEET 113 TO PLACE DWG. IN PROPER SEQUENCE	JLS	JGD	TDV	KWB
E	6/27/83	REDRAWN FOR FSAR UPDATE	JLS	JGD	TDV	

PALISADES PLANT
CONSUMERS POWER COMPANY

REACTOR PROTECTIVE SYSTEM FUNCTIONAL DIAGRAM

JOB NO.	DRAWING NO.	REV.
0950	COMBUSTION ENGINEERING 2966-E-2850	F

950Y181•M1-0 SH.113

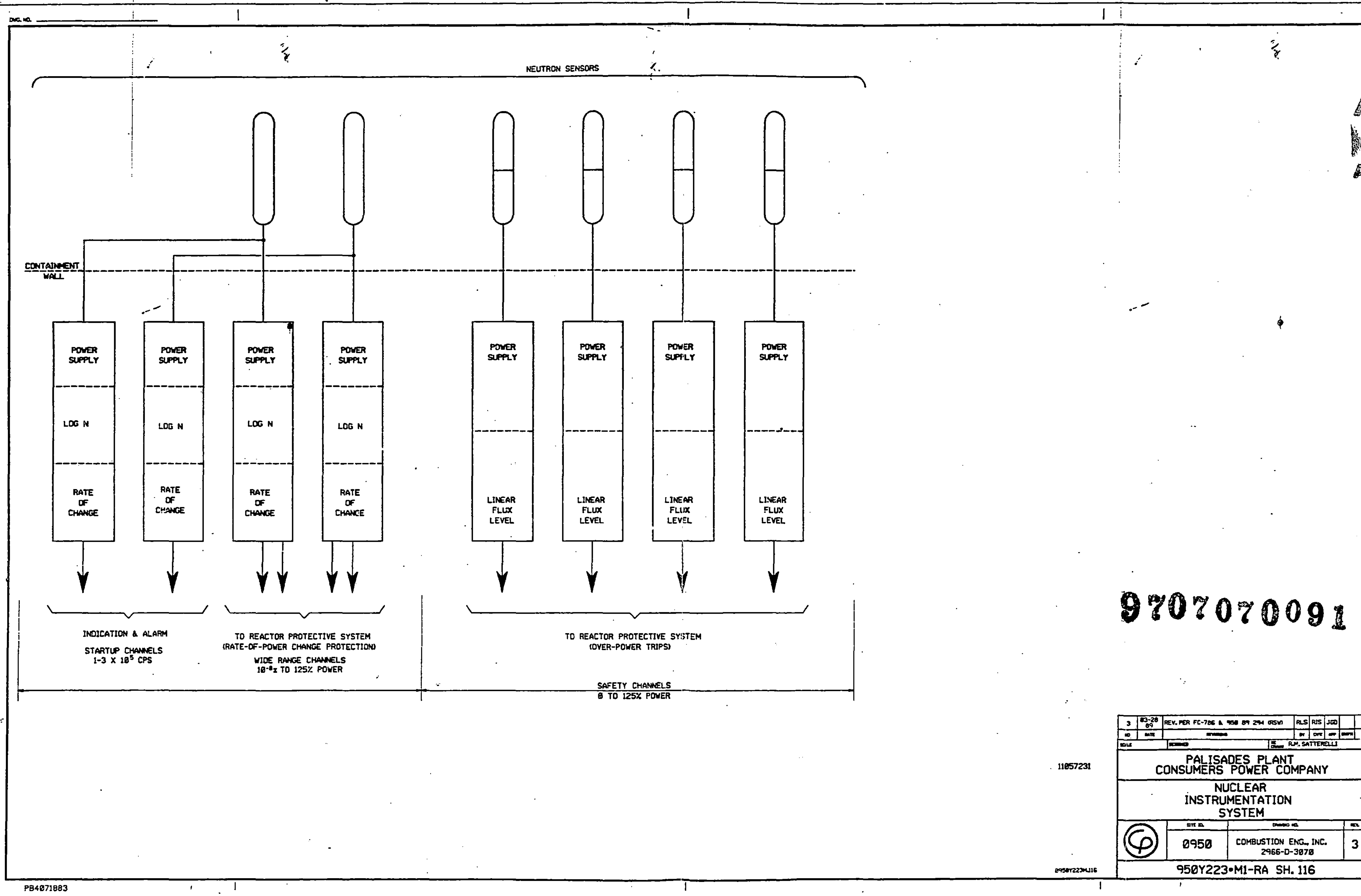


CONSUMERS POWER COMPANY
PALISADES PLANT
FSAR UPDATE

TYPICAL MEASUREMENT CHANNEL
FUNCTIONAL DIAGRAM

FIGURE NO 7-3

REVISION NO 8

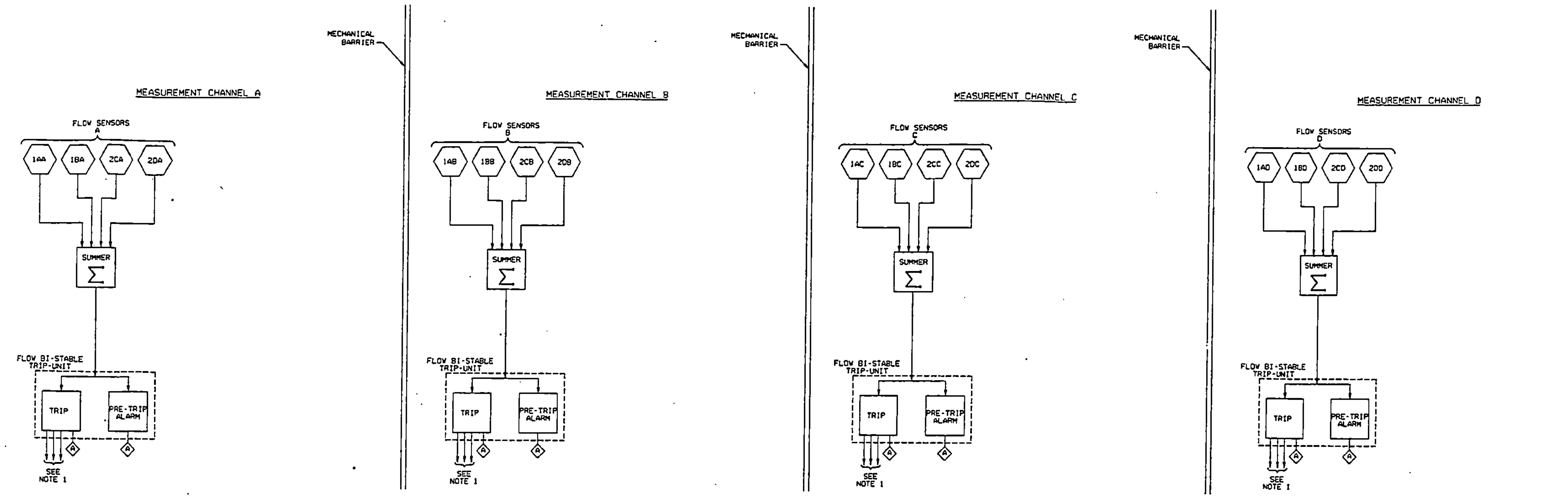


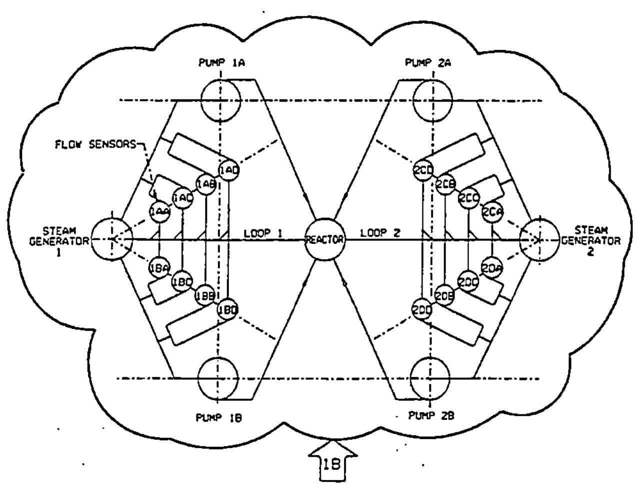
9707070091-02

Figure 7-4
Rev 19

FIGURE 7-5

(Deleted)





NOTES:

1. TO 2-OUT-OF-4 LOGIC MATRICES. REACTOR PRTECTIVE SYSTEM FUNCTIONAL DIAGRAM. (DWG 2966-E-2850)
2. FLOW TRIP SETPOINT SELECTOR (FTSS) SWITCH DELETED PER FC-888 (REF 92)

VERIFICATION STATUS: COMPLETE

REV.	DESCRIPTION				
01	INITIALLY RELEASED "AS BUILT" AND ADDED VERIFICATION STATUS: COMPLETE LABEL PER REV REQ. NO. 91-715	LANGLAIS 9-18-91			
1A	ISSUED NEW DWG PER FC-888 & DCR 92-605 (JBL)	DLT 11-10-92			
1B	REV'D. FLOW SENSORS & REMOVED "AS BUILT" STAMP PER C-PAL-95-1541 & DCR-96-846.	JKO 11-11-96	DHB		

UNLESS OTHERWISE SPECIFIED
DIMENSIONING & TOLERANCING PER ANSI Y14.5M-1982
DIMENSIONS APPLY AT 68°F(20°C)
ALL DIMENSIONS ARE IN INCHES
TOLERANCES:
ONE PLACE DECIMAL ±.1
TWO PLACE DECIMAL ±.02
THREE PLACE DECIMAL ±.005
BREAK CORNERS .015 APPROX. R OR CHAM - FILLETS .015 TO .030 R
MACHINED FINISH AA125 MICRO IN.
ANGLES: ±0° 30' CHAM: ±5°
DO NOT SCALE DRAWING

COMBUSTION ENGINEERING
NUCLEAR POWER SYSTEMS

CUSTOMER: CONSUMERS POWER PALISADES NUCLEAR PLANT

DRAWN BY: JOE LANGLAIS 9-18-91

TITLE: LOW FLOW PROTECTIVE SYSTEM FUNCTIONAL DIAGRAM

SUPERSEDES: 2966-E-2862 (M1-0 SH 1196)
DWG. NO. E-87789-489-012 01
JOB NO. 681771
SCALE: NONE

2 91-715 JOE LANGLAIS 10-7-91 INTERGRAPH 8778-489/E-012 VEN-M1-O SH. 4016 REV. 1B

ANSTEC APERTURE CARD

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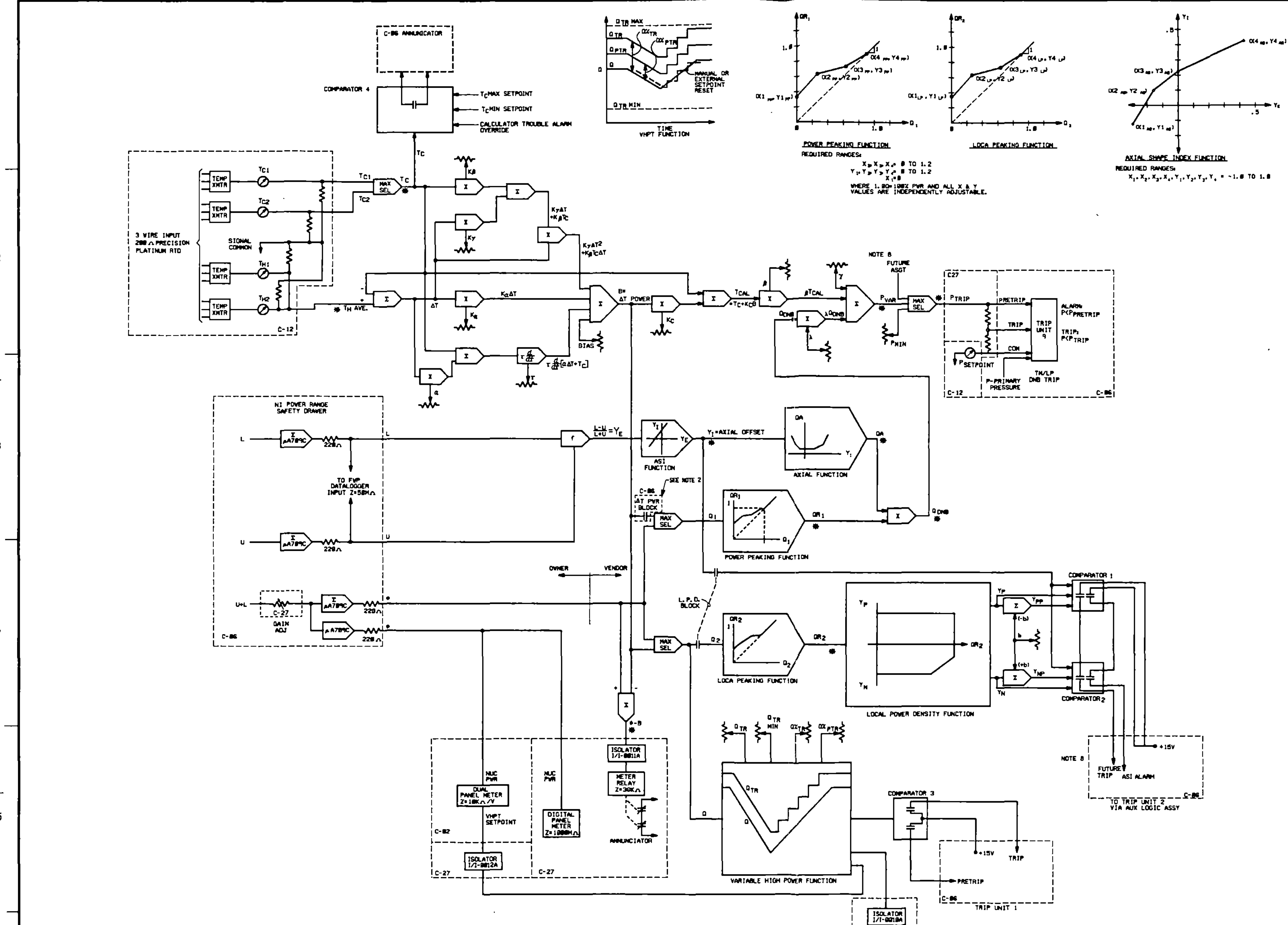
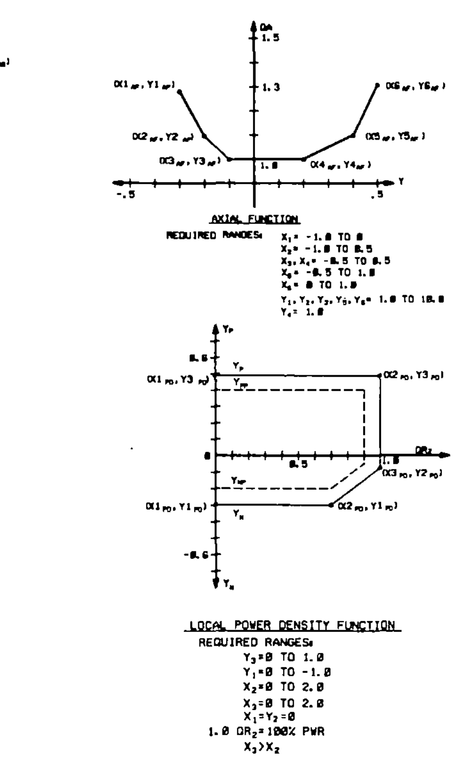
9707070091 -03

Figure 7-6
Rev 19





NOTICE:

THIS DRAWING IS CONTAINED IN SPEC J-54 WHICH ALSO FORMS A PART OF GAMM-METERICS, THERMAL MARGIN MONITOR MANUAL (NO.070) VOL IV, APPX A.
ALL UPDATES TO THIS DRAWING SHALL BE INSERTED IN THESE DOCUMENTS & COPIES FORWARDED TO GAMMA-METRIC FOR UPDATING THEIR MANUAL (REF: PO CP 11-2809-Q).

INTERNAL CALCULATIONS:

$B = \Delta T\ POWER = K\alpha\Delta T + K\gamma\Delta T^2 + K\beta T_C\Delta T + \tau\frac{d}{dt}(\alpha\Delta T + T_C) + BIAS$

T_C=HIGHEST COLD LEG TEMPERATURE
T_H=AVERAGE OR ACTIVE LOOP HOT LEG TEMPERATURE
$\Delta T = T_H - T_C$
$P_{VAR} = \lambda Q_{DNB} + \beta T_{CAL} + \gamma$
WHERE $T_{CAL} = T_C + K_C B$; $Q = MAX(\phi, B)$
$P_{TRIP} = MAX(P_{VAR}, P_{MIN})$

$Y_E = \frac{L-U}{L+U}$ **
** IF L+U < 0.50V (7% POWER) THEN $Y_E = 0$

NOTES:
1. CHANNEL A AS SHOWN
   CHANNELS B, C&D SIMILAR
2. ΔT PWR BLOCK IS CLOSED ≥ 10^{-4}% POWER. CONTACT INPUT TO TMM HAS BEEN PERMANENTLY SHORTED CLOSED.
3. L.P.D. BLOCK CLOSES AT 15% POWER ASCENDING AND OPENS AT 14.5% DESCENDING.
4. FOR DIGITAL TRIP UNITS +15V=NO TRIP/PRETRIP , 0V=TRIP/PRETRIP.
5. DIGITAL CONTACT RATING=0.5A @ 125VDC.
6. DOTTED LINES INDICATE EXISTING EQUIPMENT.
7. # DEPICTS HISTORICAL TRENDING DATA POINT.
8. FUTURE ASGT (ASYMETRICAL STEAM GENERATOR) AND FUTURE ASI TRIPS ARE FEATURES OF THE TMM WHICH ARE NOT USED.

ANALOG INPUTS/OUTPUTS

SIGNAL	DC RANGE	PARAMETER
U	0 +10 V	0 +125% PWR
L	0 +10 V	0 +125% PWR
•	0 +10 V	0 +125% PWR
P_{TRIP}	10-50 mA	1500-2500 PSIA
• -B	0 +10V	-10% 0 +10% PWR DEVIATION
TC1/TC2	1-5 V	515 - 615 °F
T_H AVE	1-5 V	515 - 615 °F
VHPT SETPT	0 +10 V	0 +125% PWR
P_{ASGT} (future)	0 +10 V	1500-2500 PSIA

DIGITAL OUTPUTS

	FUNCTION		
	PRETRIP	TRIP	ALARM
COMPARATOR 1		$Y_P \le Y_I$ *	$Y_{PP} \le Y_I$
COMPARATOR 2		$Y_N \ge Y_I$ *	$Y_{NP} \ge Y_I$
COMPARATOR 3	$Q > Q\%_{PTR}$	$Q > Q\%_{TR}$	
COMPARATOR 4			$T_C MAX > T_C$
			$T_C MIN < T_C$

* TRIP IS DISABLED

ADJUSTABLE CONSTANTS •

	CONSTANT	RANGE REQUIRED
ΔT POWER EQUATION	K_β	1×10^{-6} TO 5×10^{-3}
	K_γ	-1×10^{-5} TO 1×10^{-5}
	K_α	1×10^{-3} TO 3×10^{-3}
	α	0 TO 10
	τ	0 TO 1
	BIAS	-0.10 TO 0.10
TH/LP TRIP PRESSURE	K_C	0 TO 0.1
	λ	1000 TO 3000
	β	10 TO 30
	γ	-13000 TO -4000
	P_{MIN}	1500 TO 2000
ASI ALARM	h	0 TO 1.0
VHPT FUNCTION	Q_{TR} MAX	0-125% POWER
	Q_{TR} MIN	0-50% POWER
	$Q\%_{TR}$	0-20% ABOVE Q
	$Q\%_{PTR}$	0-20% ABOVE Q

• ADDITIONAL CONSTANTS ARE SPECIFIED ABOVE FOR EACH FUNCTION

9107300279-09

REV	DATE	DESCRIPTION	GWO / FC*	BY	CK	APP	PROJ ENGR
1	3-5 91	REV'D NOTE *2 PER GWO 8614 & FC-829 & 950 91 240 (EH)	6034 / 620	RLS	RIS	JGD	

PREVIOUS RECORD ISSUE

CONSUMERS POWER COMPANY
JACKSON, MICHIGAN

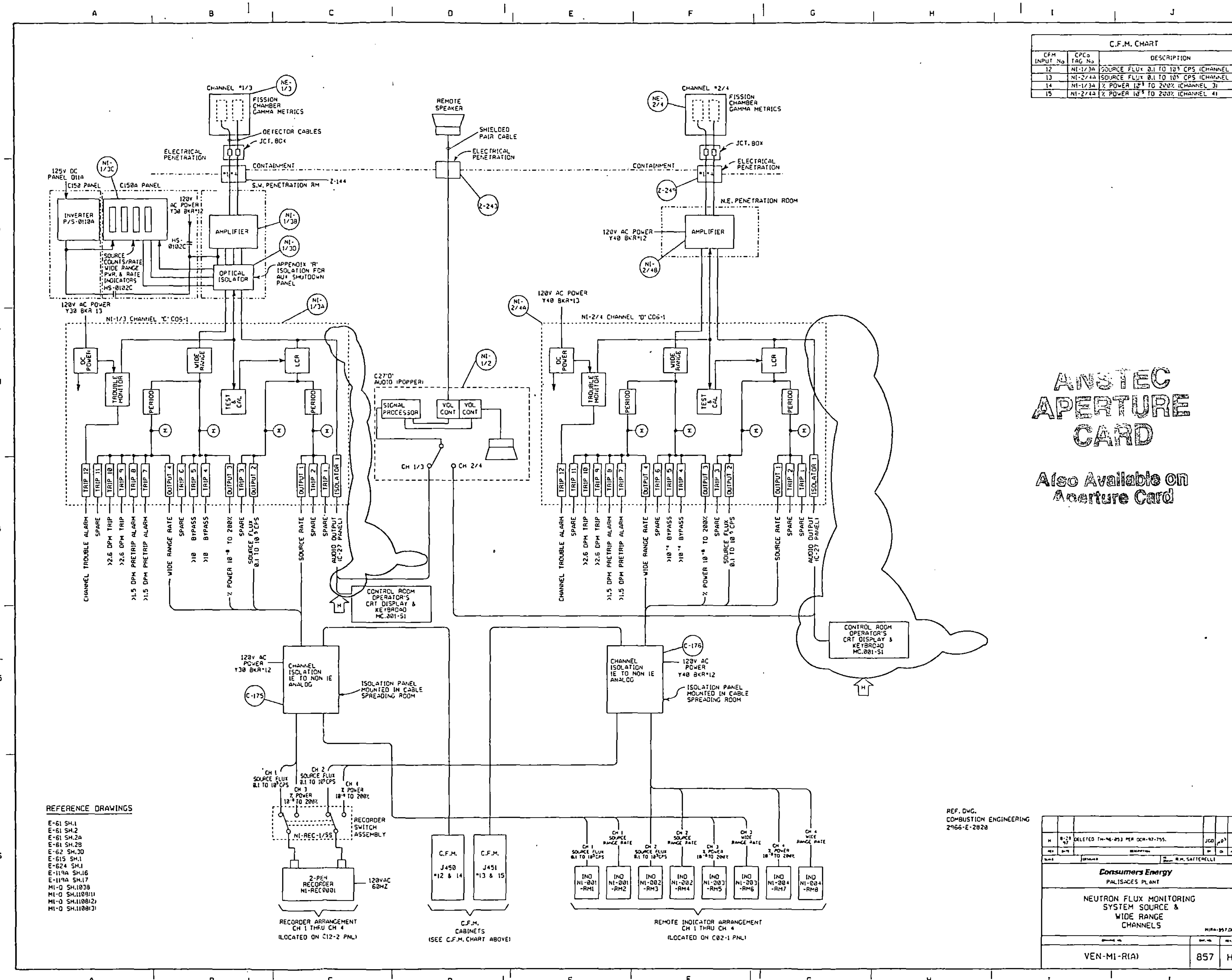
0950JL130.001

LOGIC FOR THERMAL MARGIN MONITOR

PALISADES PLANT

MICHIGAN P.E. NO.	UFI NO.	LOC	BASE	SHEET	REV
	SYS:02400	950	JL-130	1	1

Figure No. 7-7 FSAR Rev 12



ANSTEC APERTURE CARD

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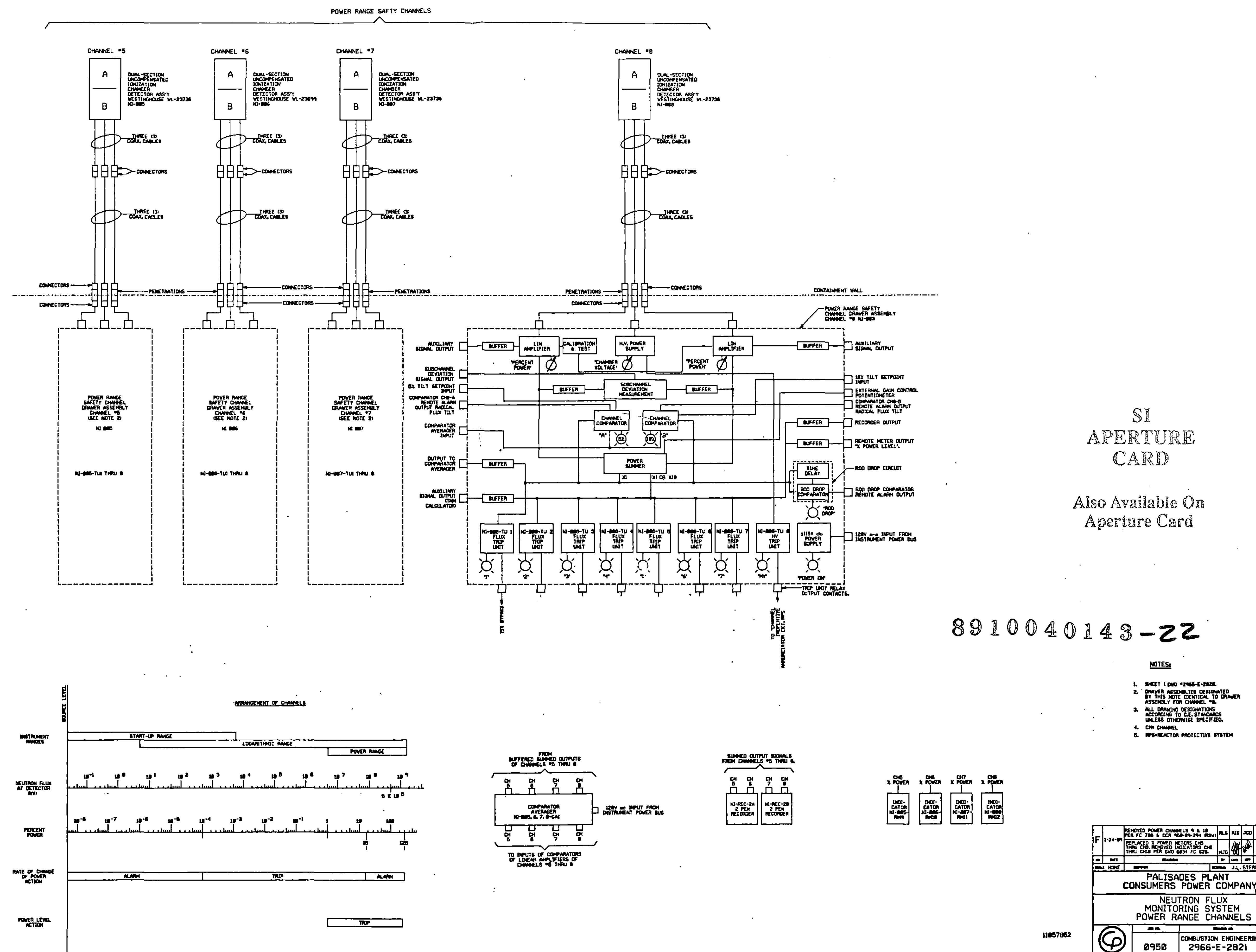
FIG 7-8
FSAR REV 20

9807090043 - 13



SI APERTURE CARD

Also Available On Aperture Card

8910040143-22

F	1-24-89	REMOVED POWER CHANNELS 9 & 10 PER FC 786 & DCR 950-89-294 (RSW) REPLACED % POWER METERS CH5 THRU CH8. REMOVED INDICATORS CH5 THRU CH08 PER GWO 6834 FC 628.	RLS MJG	RJS	JGD

SCALE NONE — J.L. STERRETT

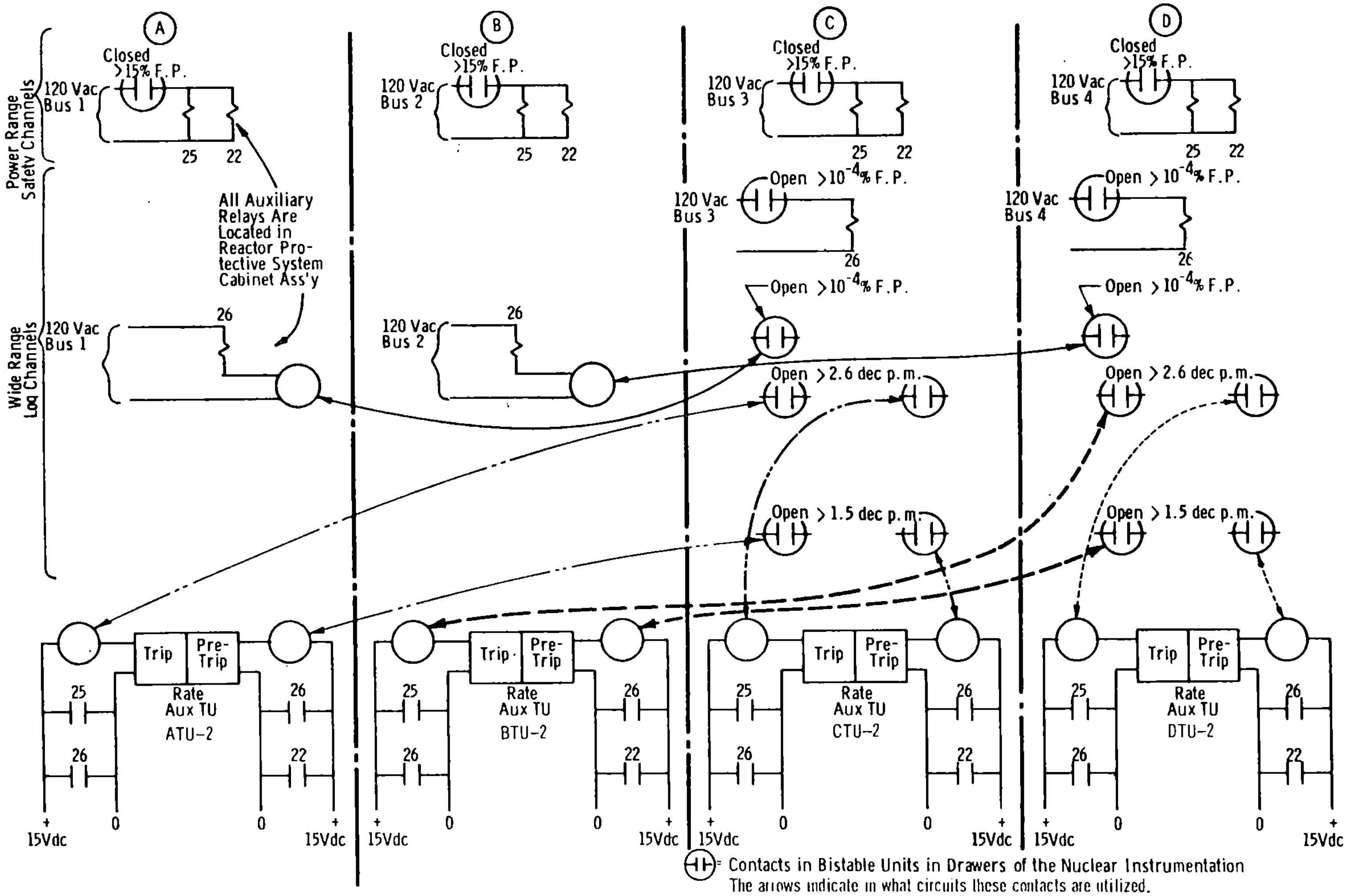
PALISADES PLANT
CONSUMERS POWER COMPANY

NEUTRON FLUX
MONITORING SYSTEM
POWER RANGE CHANNELS

0950 — COMBUSTION ENGINEERING 2966-E-2821 — F

950Y223•M1-RA SH.858

Figure No. 7-9 FSAR Rev. 9

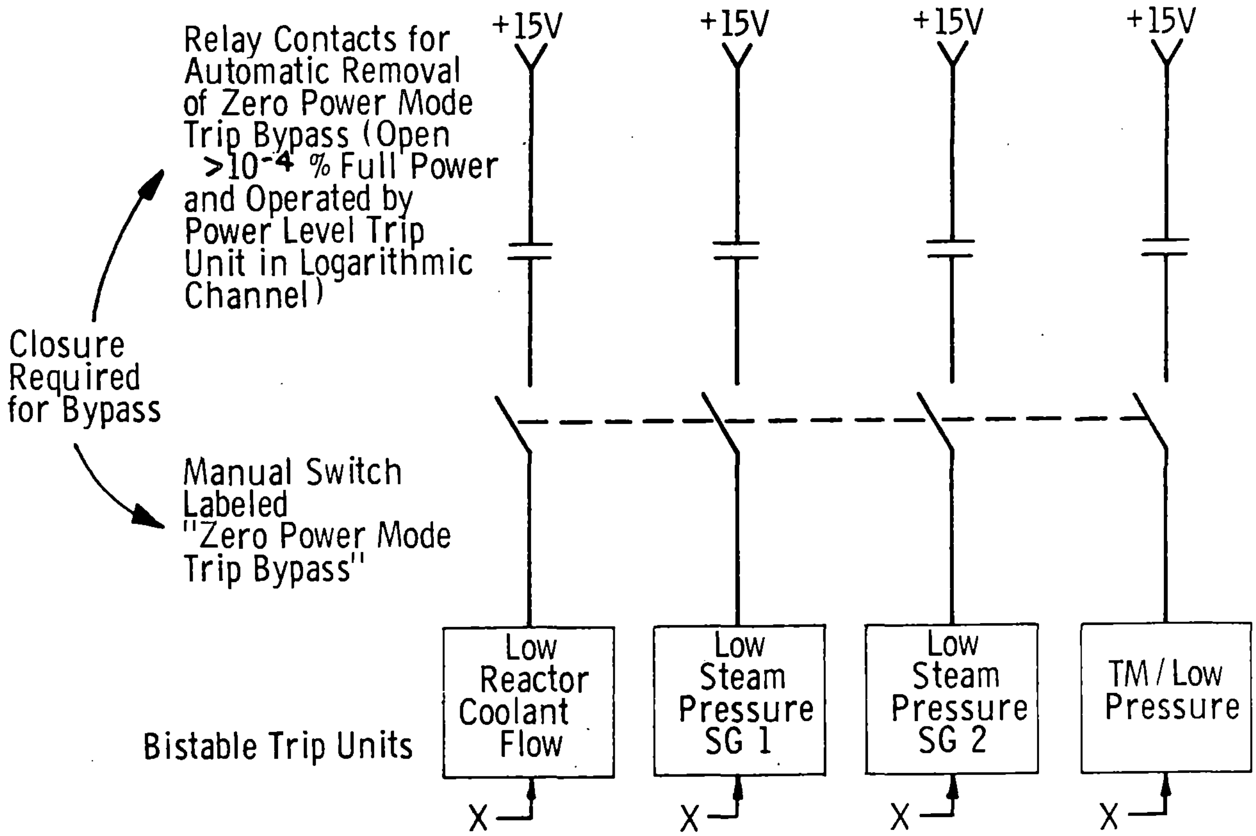


CONSUMERS POWER COMPANY
PALISADES PLANT
FSAR UPDATE

POWER RATE OF CHANGE TRIP AND PRE-TRIP INTERFACE WITH RPS

FIGURE NO 7-10

REVISION NO 0



X = Analog Input Signal

With +15V Applied to Bistable Trip Unit: No Trip Regardless To Level of Input Analog Signal

Without +15V Applied to Bistable Trip Unit: Trip According to Level of Input Analog Signal

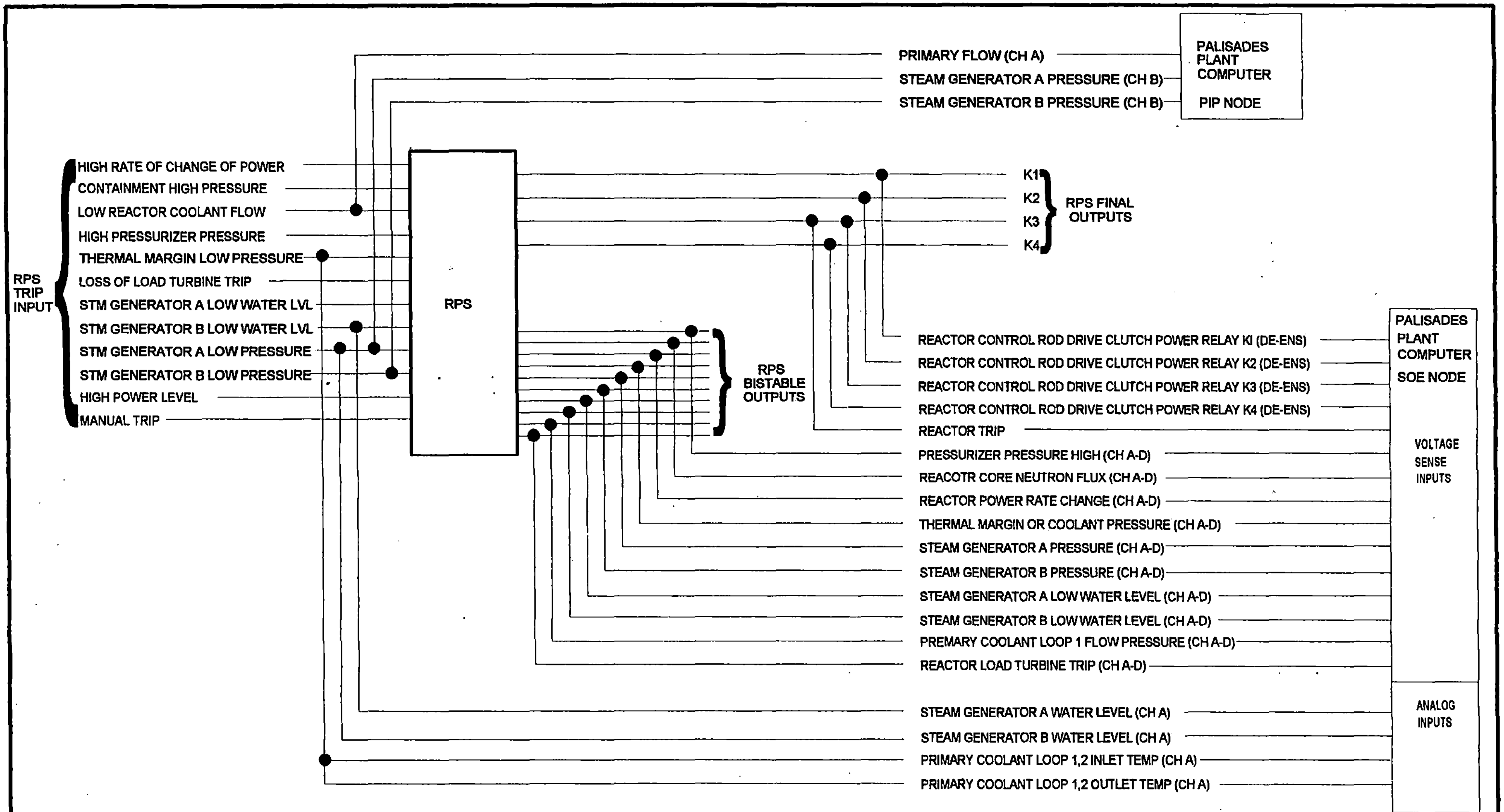
Same Arrangement for Other 3 Channels

CONSUMERS POWER COMPANY
PALISADES PLANT
FSAR UPDATE

ZERO POWER MODE BYPASS

FIGURE NO 7-11

REVISION NO 0

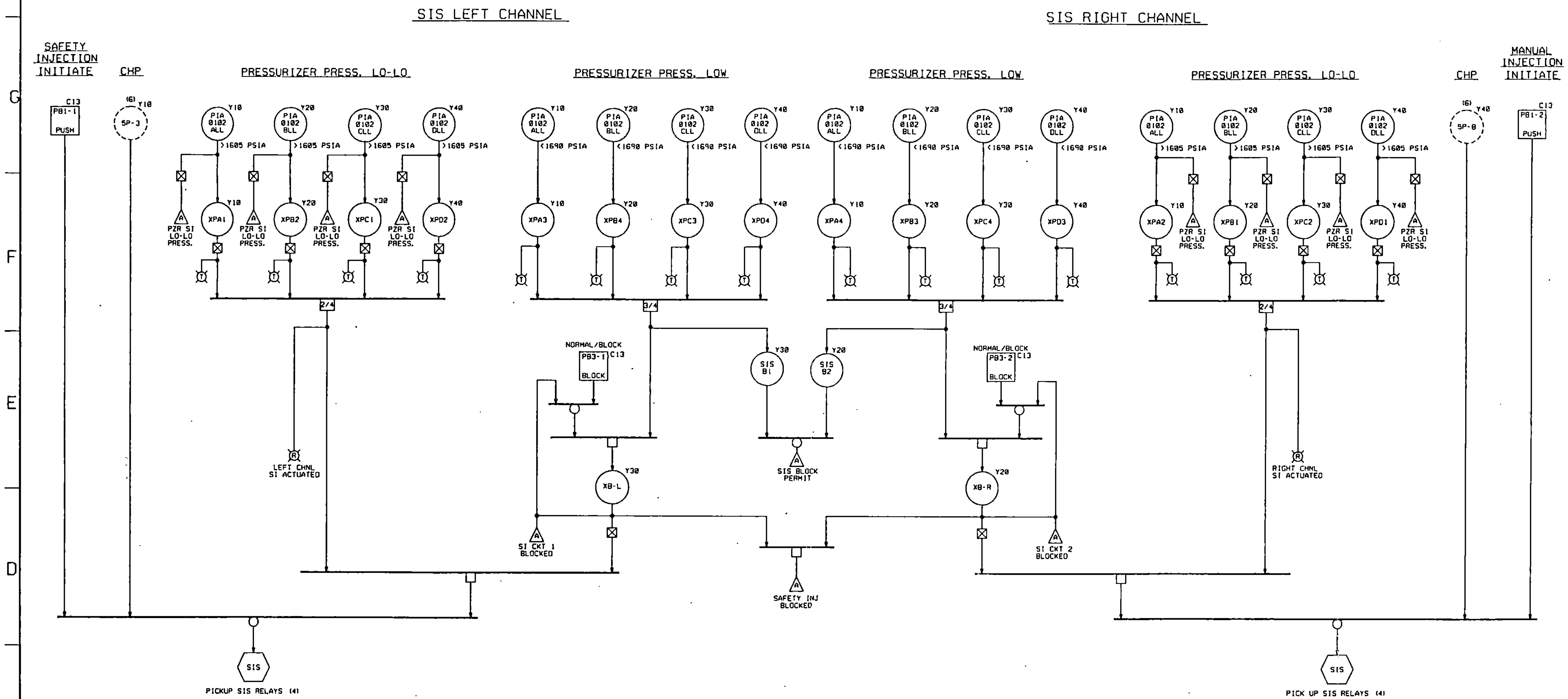


CONSUMERS POWER COMPANY
PALISADES PLANT
FSAR UPDATE

REACTOR PROTECTIVE SYSTEM INTERFACES

FIGURE NO 7-12

Rev 18



NOTES:

EACH PRESSURE-INDICATOR-ALARM (PIA) CHANNEL HAS THREE OUT PUTS:

1. CONTACTS 20-22 CLOSE ON DECREASING PRESSURE (1690 PSIA IS NOMINAL VALUE) ENERGIZING RELAYS XP 3 AND XP 4 IN THE SI BLOCK LOGIC.
2. CONTACTS 17-19 OPEN ON DECREASING PRESSURE (1605 PSIA IS NOMINAL VALUE) ACTUATING A PRESSURIZER SAFETY INJ. SIGNAL LO-LO PRESS' ALARM.
3. CONTACTS 14-16 OPEN ON DECREASING PRESSURE (1605 PSIA IS NOMINAL VALUE) DE-ENERGIZING RELAYS XP 1 AND XP 2 IN THE SI INITIATION LOGIC.

REF: E-206, E-209

TEST LIGHTS ARE INSIDE C12L

18

ANSTEC APERTURE CARD

Also Available on Aperture Card

9807090043 -14

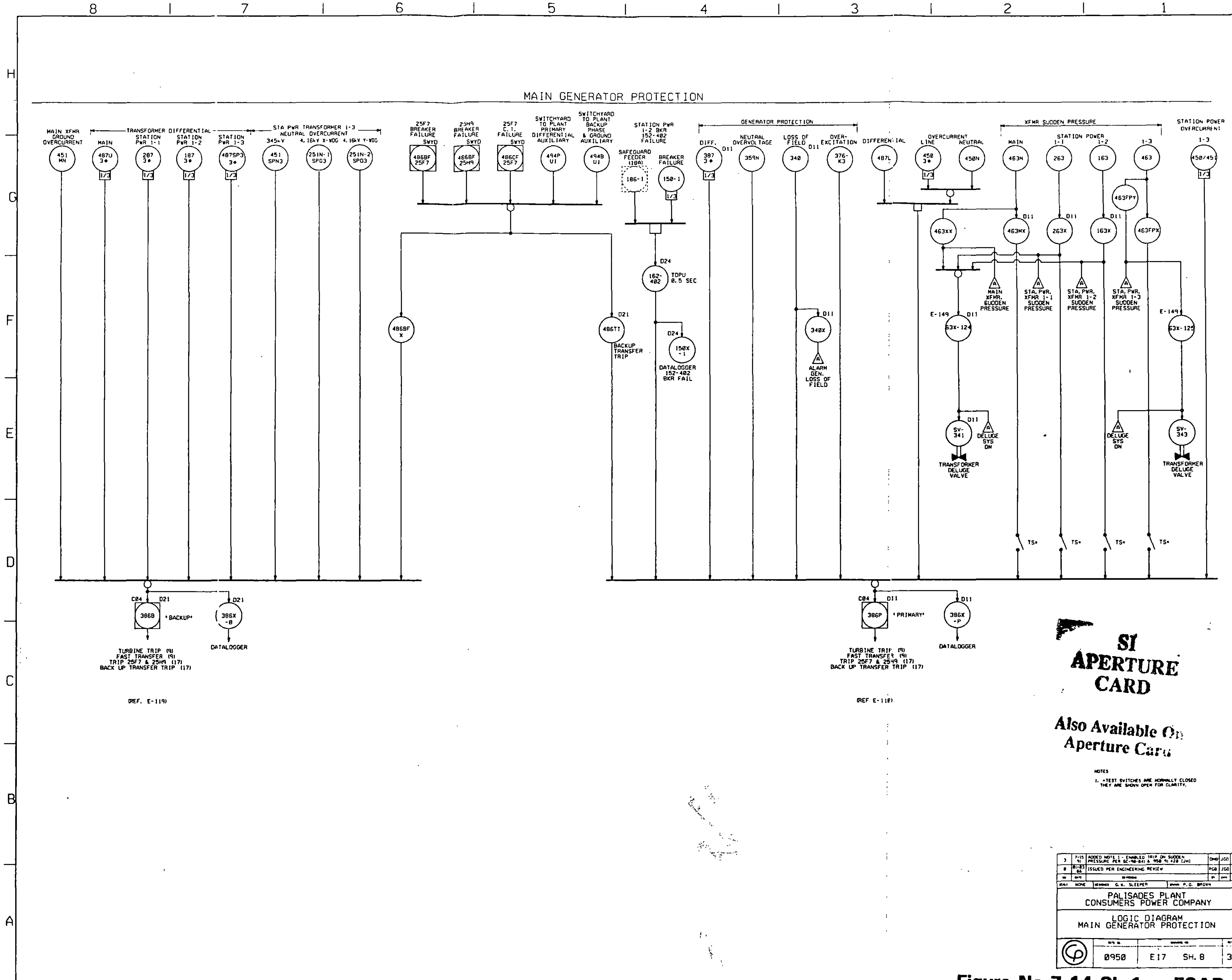
18	10-10 97	REV'D. NOTES PER DCR-97-850	JGD		
REV.	DATE	DESCRIPTION	BY	CK	APP
SCALE NONE	DESIGNED B.N. YOUNG		DRAWN P.G. BROWN		

PALISADES PLANT
CONSUMERS POWER COMPANY E17-3.DGN

LOGIC DIAGRAM
SAFETY INJECTION INITIATION

	0950	E-17 SH. 3	18

FIG 7-13
FSAR REV 20

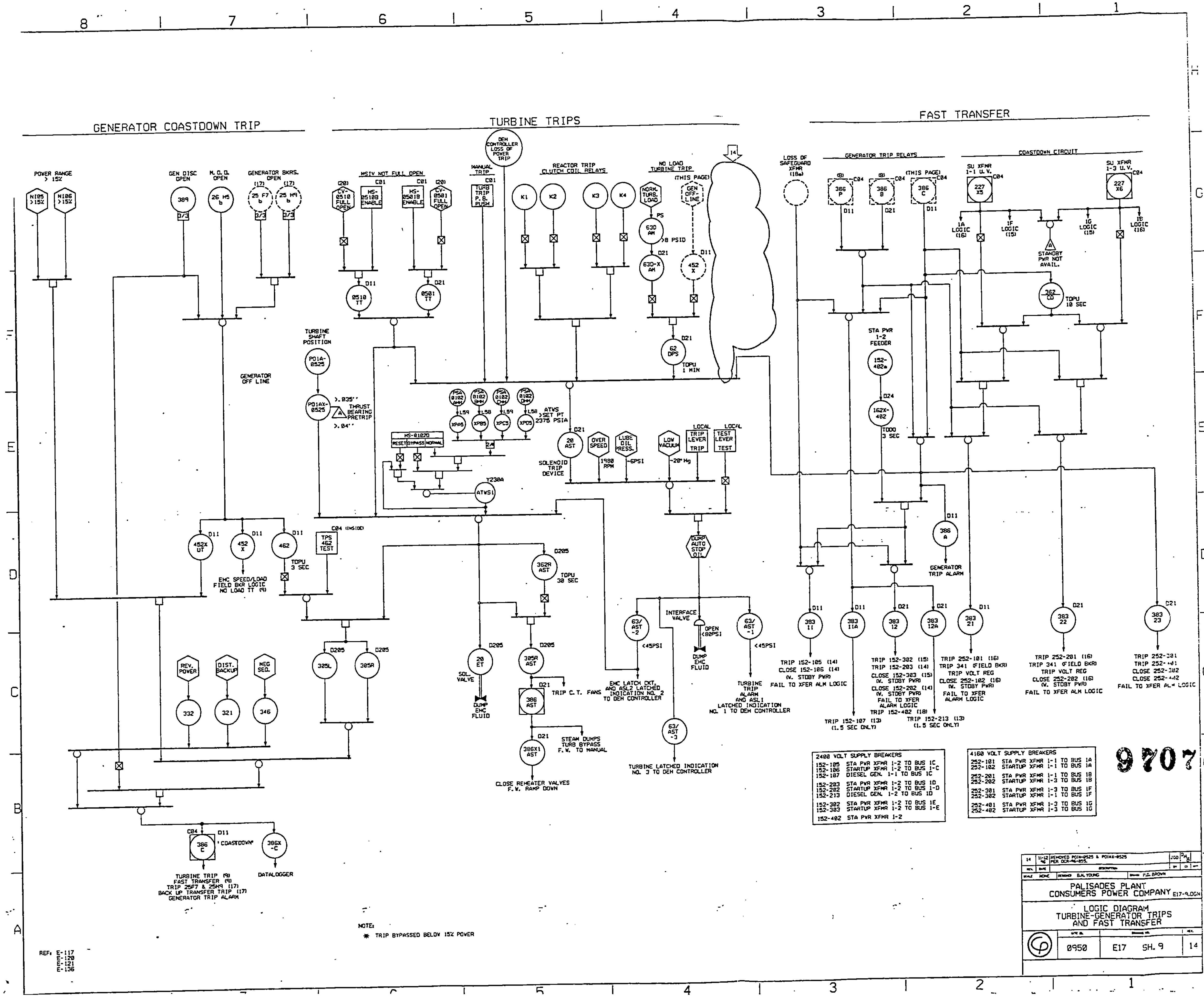


SI APERTURE CARD

Also Available On Aperture Card

Figure No 7-14 Sh 1 FSAR Rev 13

9203060145-12



2400 VOLT SUPPLY BREAKERS	
152-105	STA PWR XFMR 1-2 TO BUS 1C
152-106	STARTUP XFMR 1-2 TO BUS 1-C
152-107	DIESEL GEN. 1-1 TO BUS 1C
152-203	STA PWR XFMR 1-2 TO BUS 1D
152-202	STARTUP XFMR 1-2 TO BUS 1-D
152-213	DIESEL GEN. 1-2 TO BUS 1D
152-302	STA PWR XFMR 1-2 TO BUS 1E
152-303	STARTUP XFMR 1-2 TO BUS 1-E
152-402	STA PWR XFMR 1-2

4160 VOLT SUPPLY BREAKERS	
252-101	STA PWR XFMR 1-1 TO BUS 1A
252-102	STARTUP XFMR 1-1 TO BUS 1A
252-201	STA PWR XFMR 1-1 TO BUS 1B
252-202	STARTUP XFMR 1-3 TO BUS 1B
252-301	STA PWR XFMR 1-3 TO BUS 1F
252-302	STARTUP XFMR 1-1 TO BUS 1F
252-401	STA PWR XFMR 1-3 TO BUS 1G
252-402	STARTUP XFMR 1-3 TO BUS 1G

14 | 11-12-96 | REMOVED P01A-0525 & P01AX-0525 PER DCR-96-055. | JCO

SCALE NONE | B.N. YOUNG | P.G. BROWN

PALISADES PLANT
CONSUMERS POWER COMPANY E17-9.DGN

LOGIC DIAGRAM
TURBINE-GENERATOR TRIPS
AND FAST TRANSFER

0950 | E17 SH. 9 | 14

9707070091-05

Figure 7-14 Sh 2
Rev 19

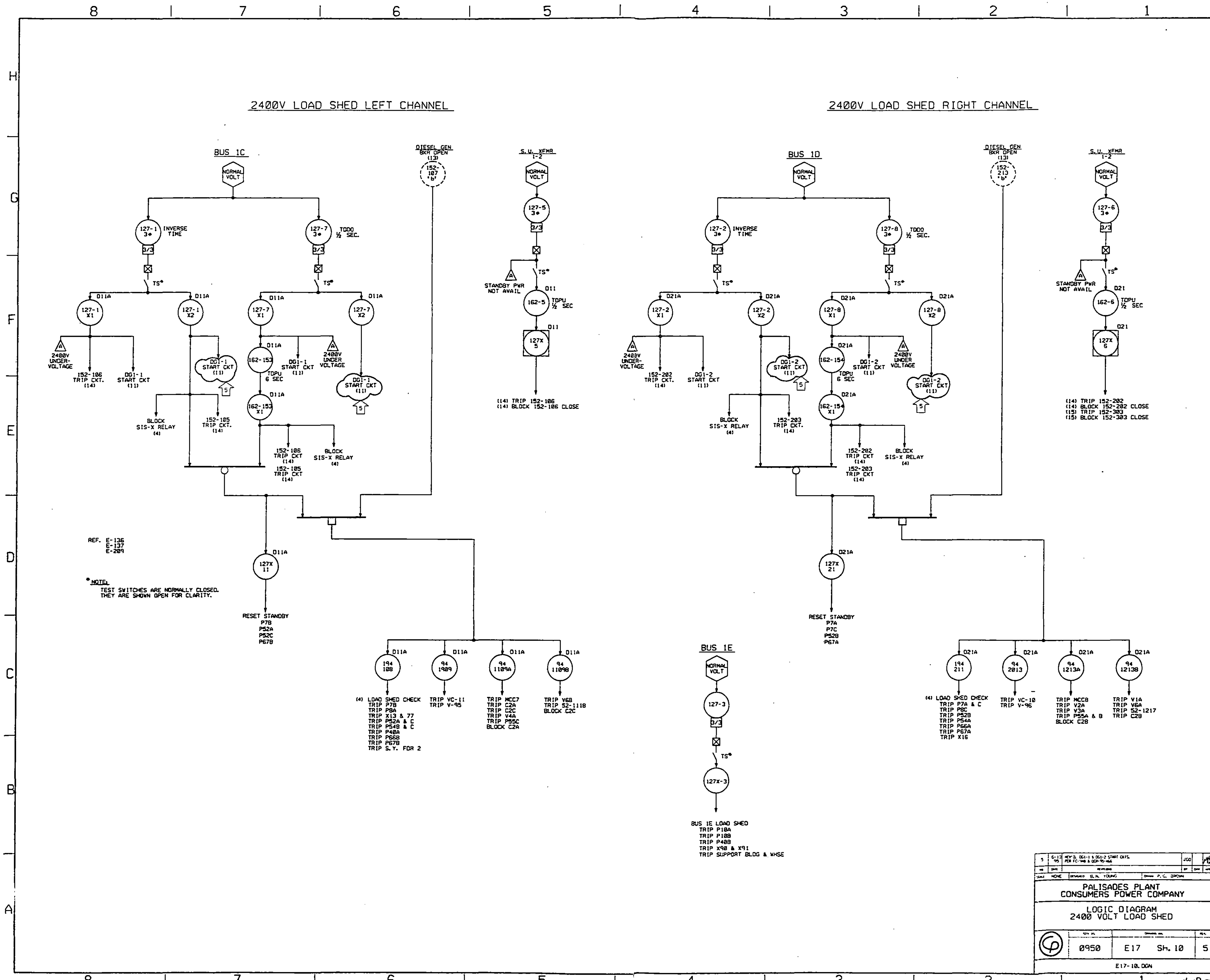
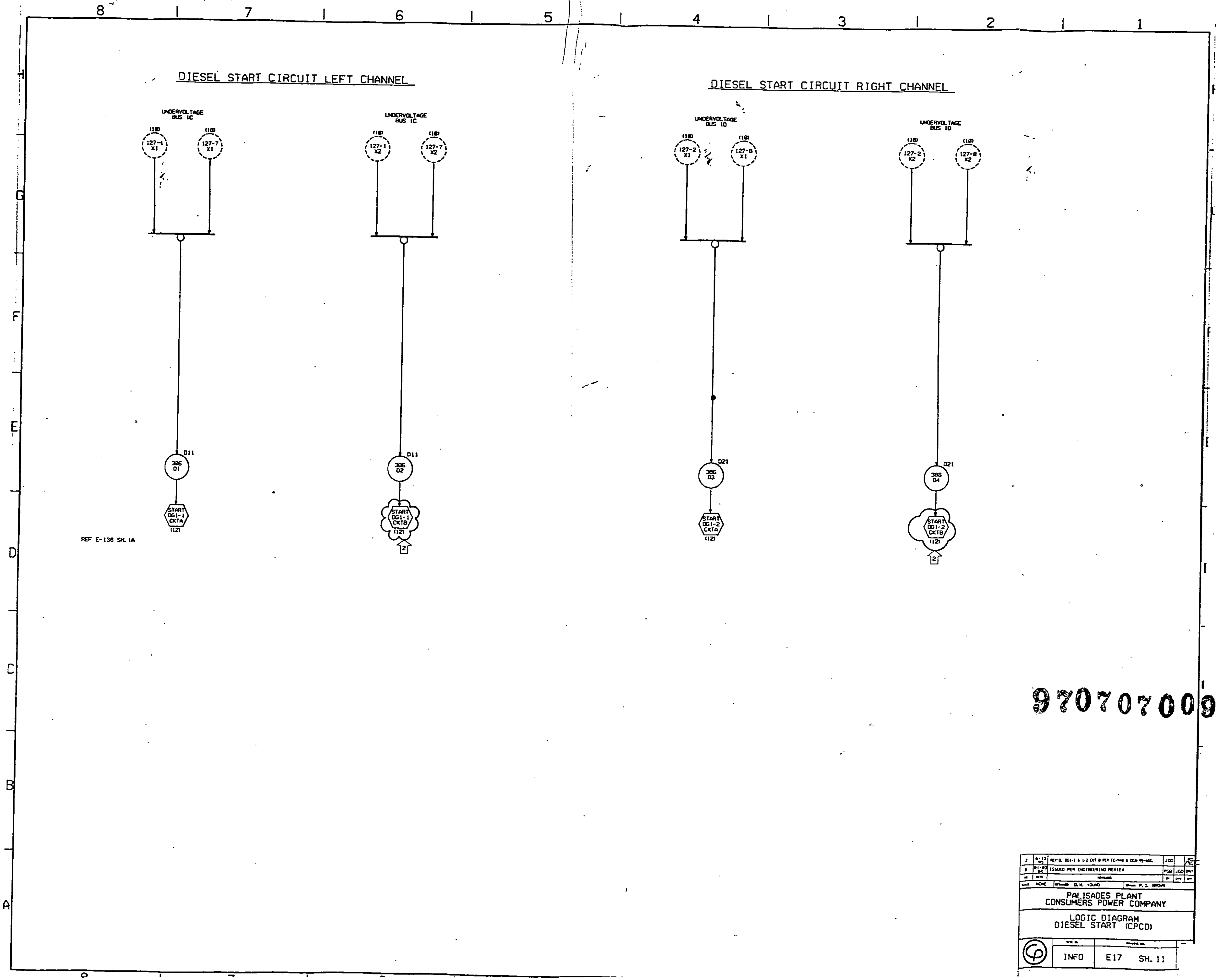
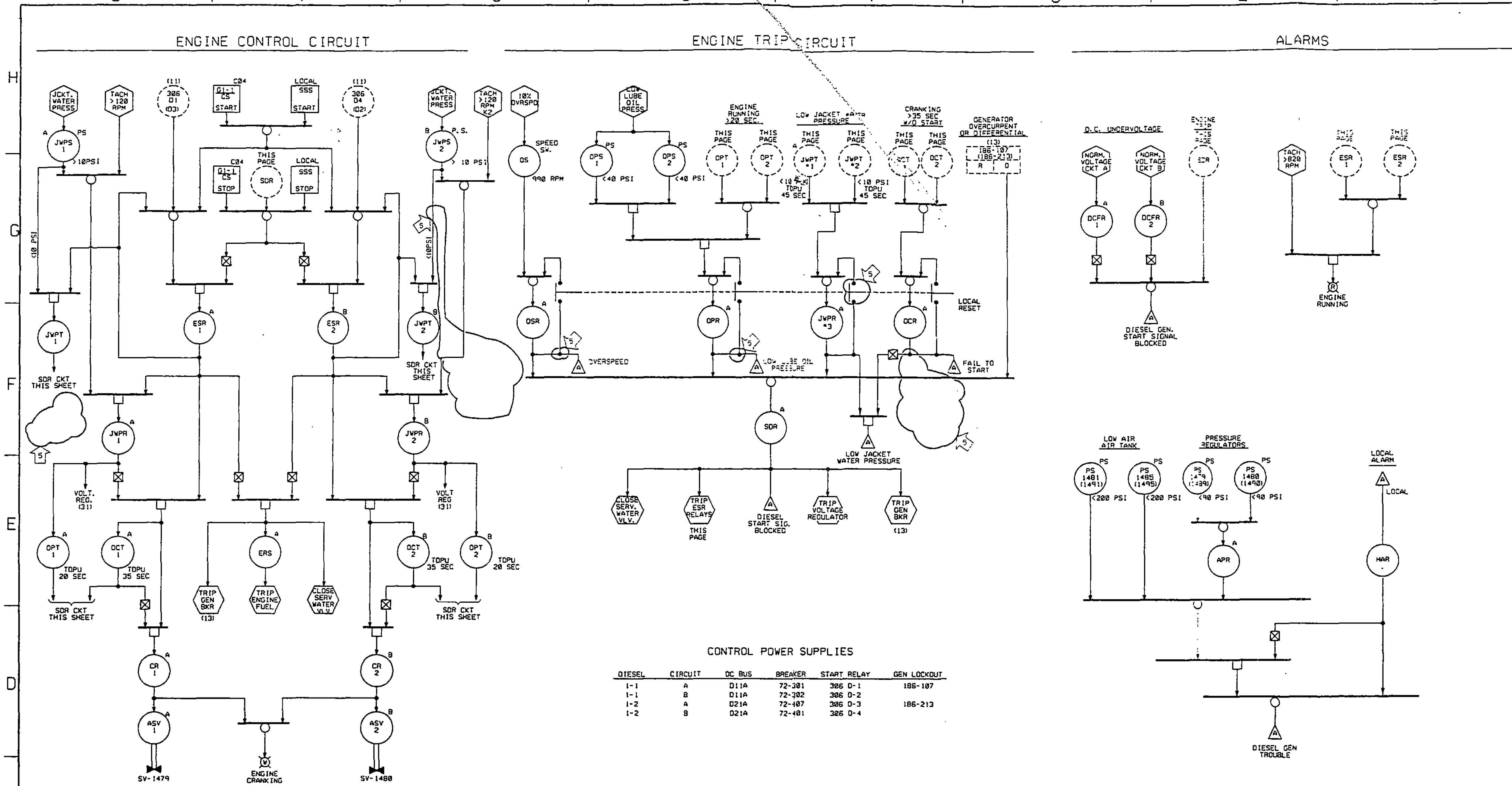


Figure 7-14 Sh 3 FSAR Rev 18



9707070091 -06

Figure 7-14 Sh 4
Rev 19



CONTROL POWER SUPPLIES

DIESEL	CIRCUIT	DC BUS	BREAKER	START RELAY	GEN LOCKOUT
1-1	A	D11A	72-301	306 D-1	186-107
1-1	B	D11A	72-302	306 D-2	
1-2	A	D21A	72-407	306 D-3	186-213
1-2	B	D21A	72-401	306 D-4	

ABBREVIATIONS

APR	AIR PRESSURE RELAY
ASV	AIR START VALVE
CR	CRANKING RELAY
DCFR	D.C. FAILURE RELAY
ERS	ENGINE RUN SOLENOID
ESR	ENGINE START RELAY
JWPR	JACKET WATER PRESSURE RELAY
JWPS	JACKET WATER PRESSURE SWITCH
OCR	OVER CRANK RELAY
OCT	OVER CRANK TIMER
OPR	OIL PRESSURE RELAY
OPS	OIL PRESSURE SWITCH
OPT	OIL PRESSURE TIMER
OS	OVERSPEED SWITCH
OSR	OVERSPEED RELAY
SDR	SHUT DOWN RELAY
HAR	HORN ALARM RELAY
JWPT	JACKET WATER PRESSURE TIMER

REV	DATE	DESCRIPTION
5	7-7-95	REVISED ENGINE CONTROL AND TRIP CIRCUITS PER FC 944 & DCR 95 503

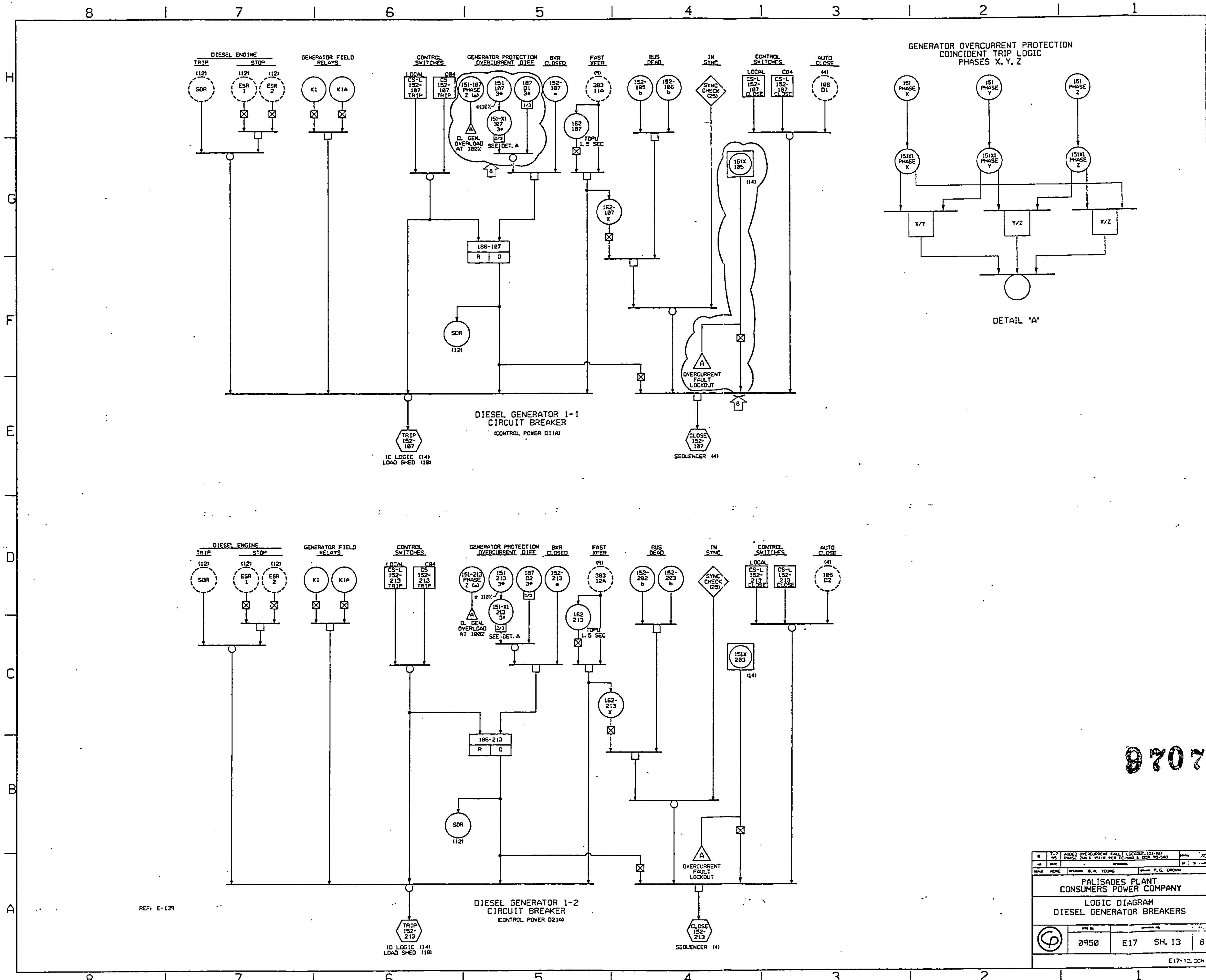
CONSUMERS POWER COMPANY

LOGIC DIAGRAM
DIESEL ENGINE CONTROL,
TRIPS & ALARMS

E17

Figure 7-14 Sh 5 FSAR Rev 18

9602290093-08



9707070091-07

Figure 7-14 Sh 6
Rev 19

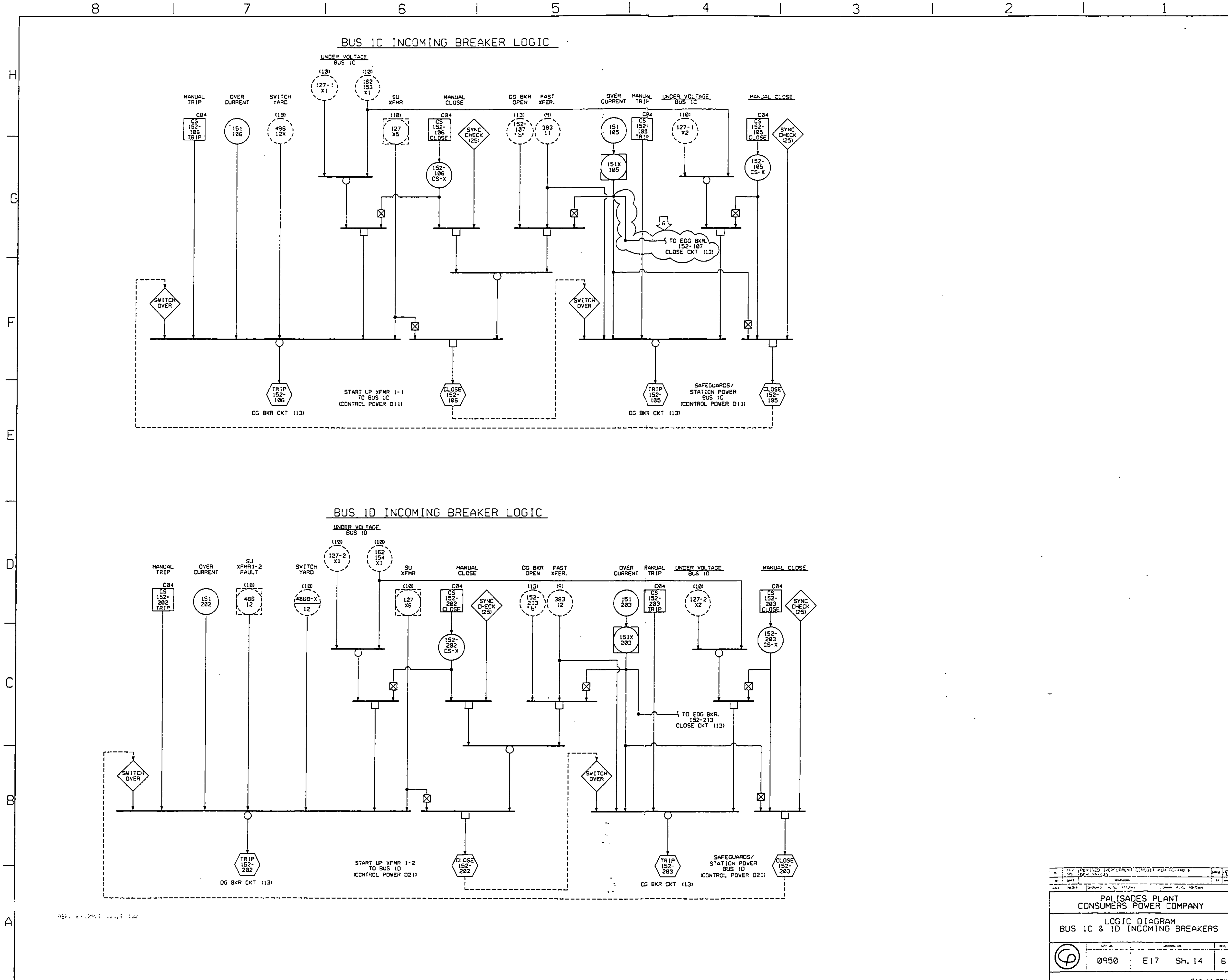


Figure 7-14 Sh 7 FSAR Rev 18

9602290093-09

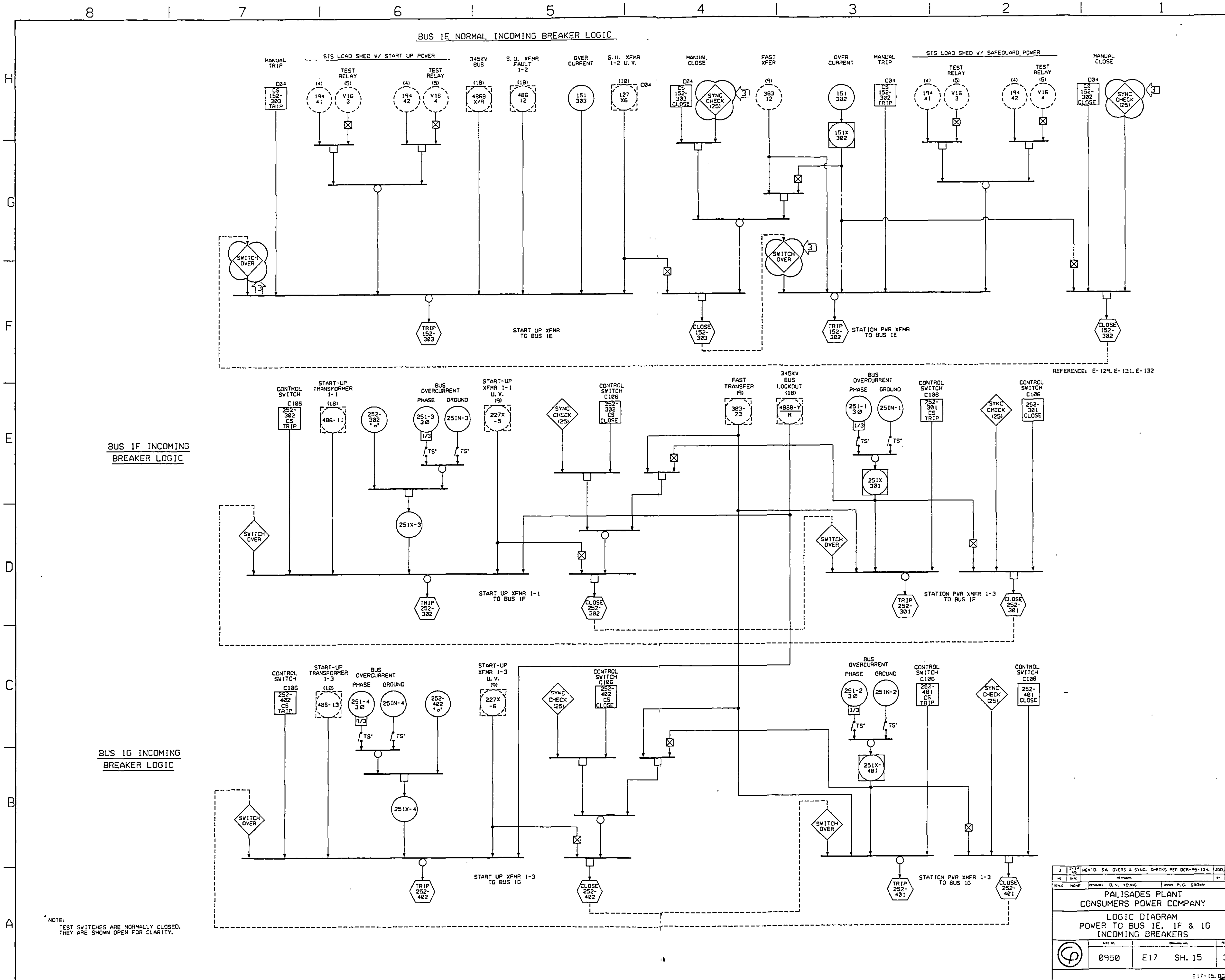
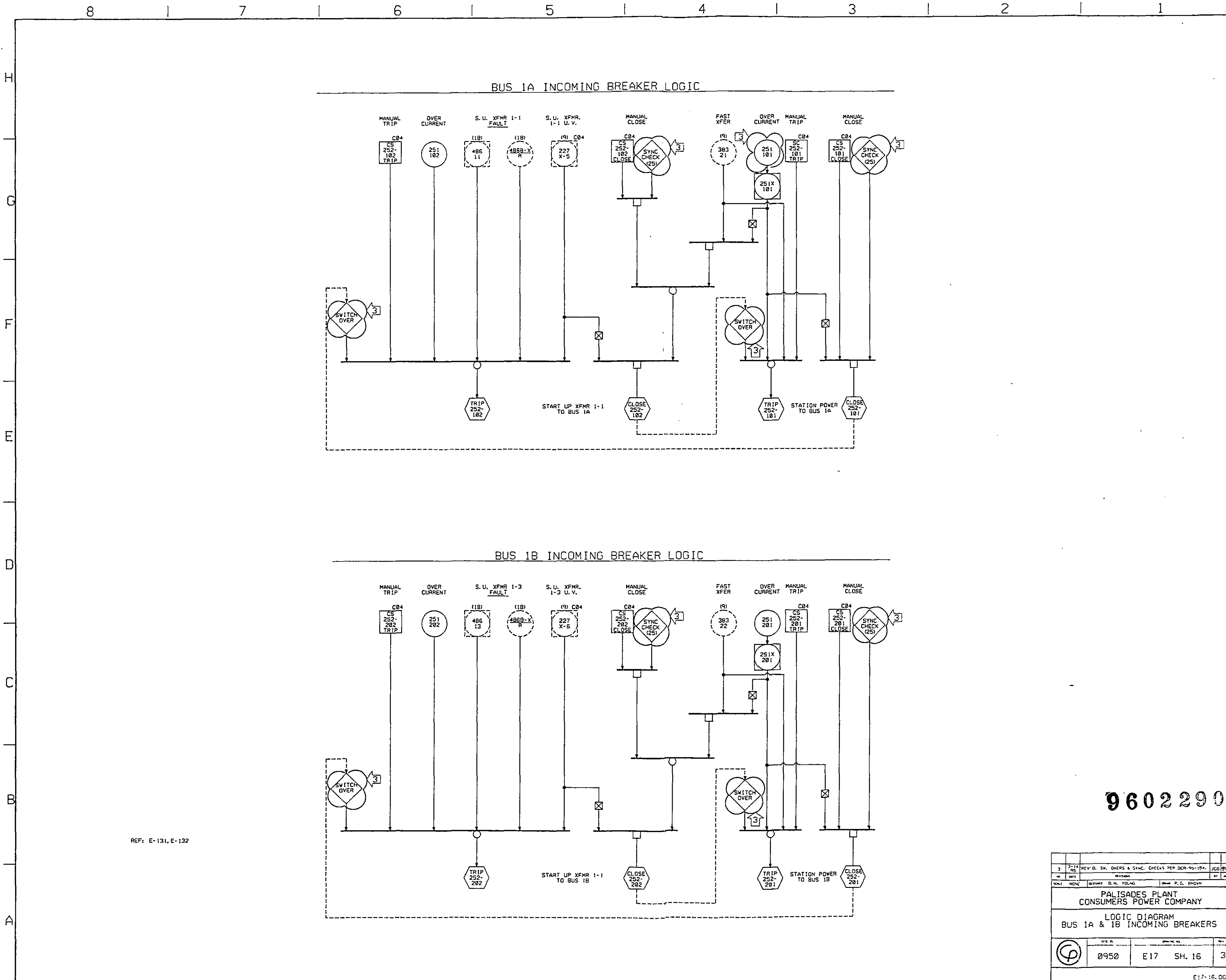


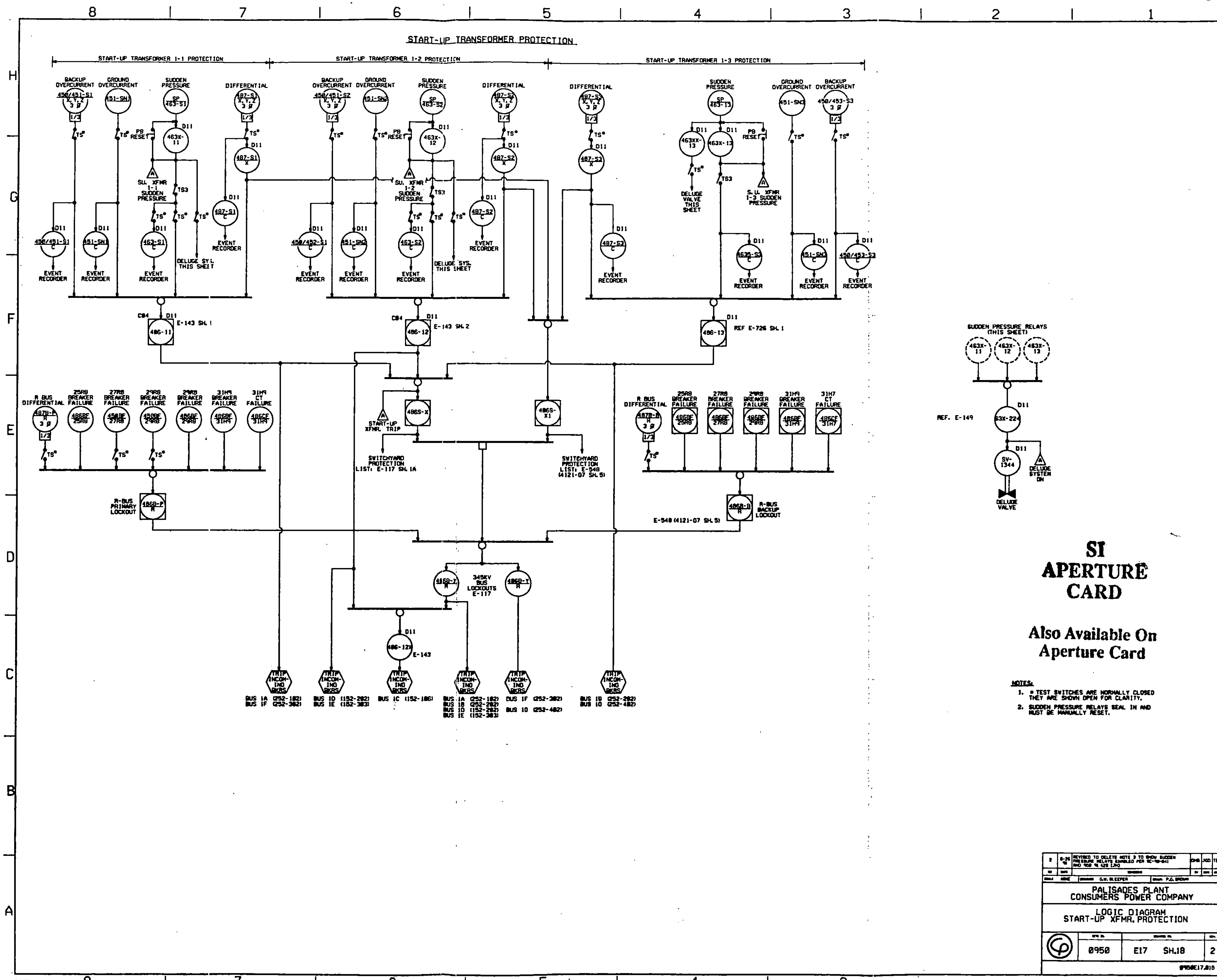
Figure 7-14 Sh 8 FSAR Rev 18

9602200093-10



9602290093

Figure 7-14 Sh 9 FSAR Rev 18



SI APERTURE CARD

Also Available On Aperture Card

Figure No 7-14 Sh 11 FSAR Rev 13

9203060145 -14

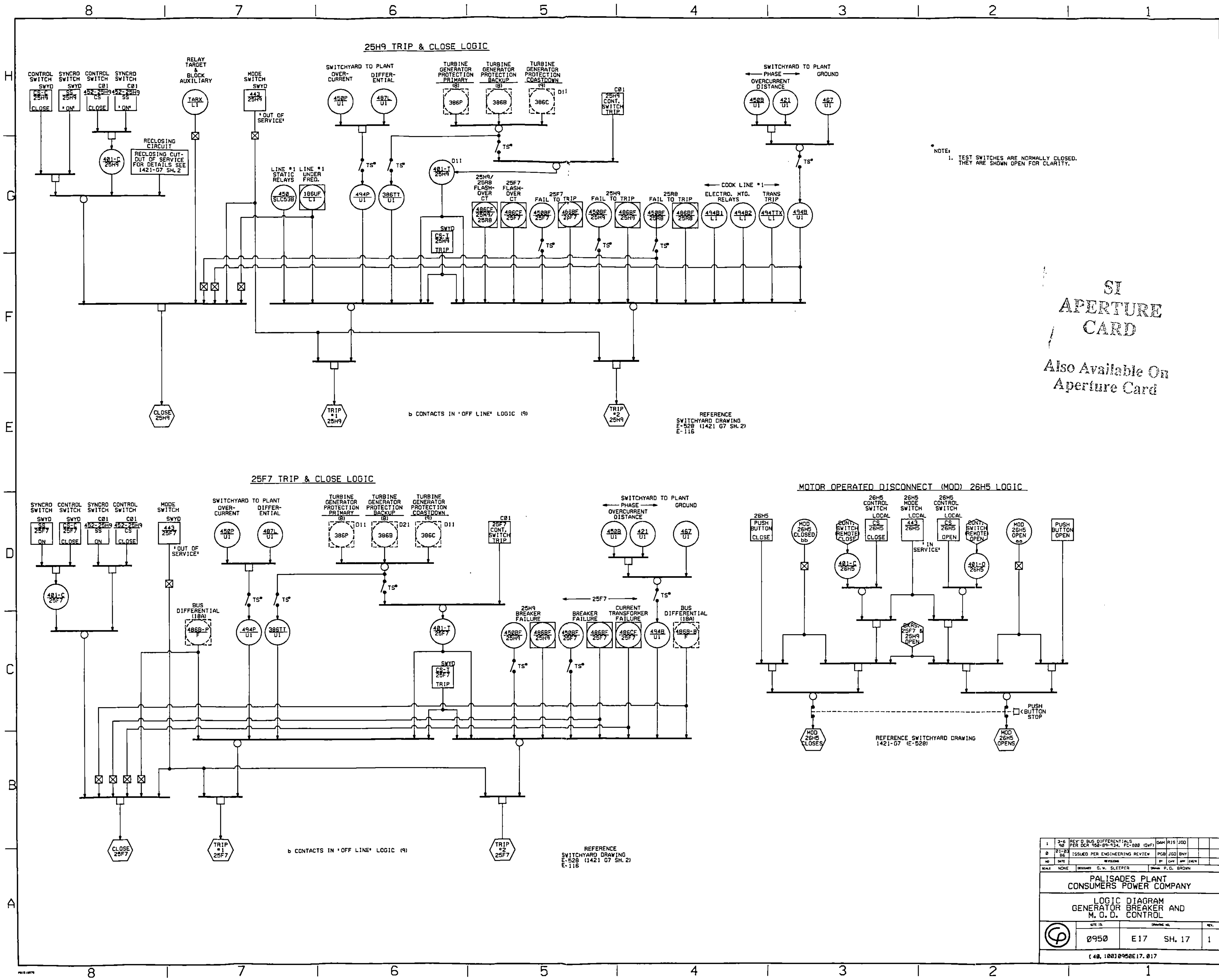
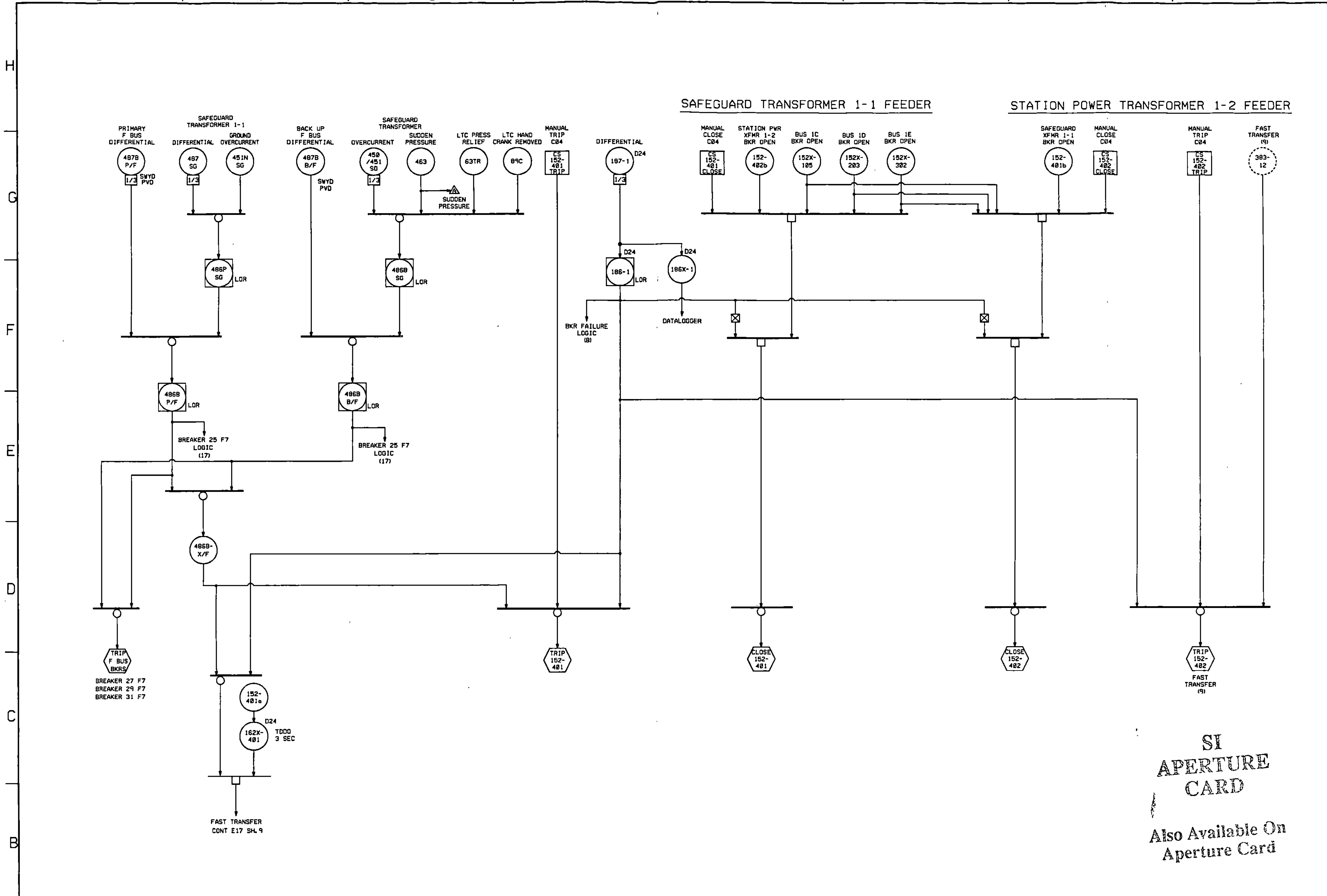


Figure No. 7-14 sh 10 FSAR Rev. 11

9012260101-22



SI APERTURE CARD

Also Available On Aperture Card

0	12-14 89	ISSUED FOR RECORD	8303 B00	MJG		
REV	DATE	DESCRIPTION	GWO FC*	BY	CK	PROJ ENGR

CONSUMERS POWER COMPANY
JACKSON, MICHIGAN
0950E17.18A

PREVIOUS RECORD ISSUE
MICHIGAN P.E. No.
UFI No.

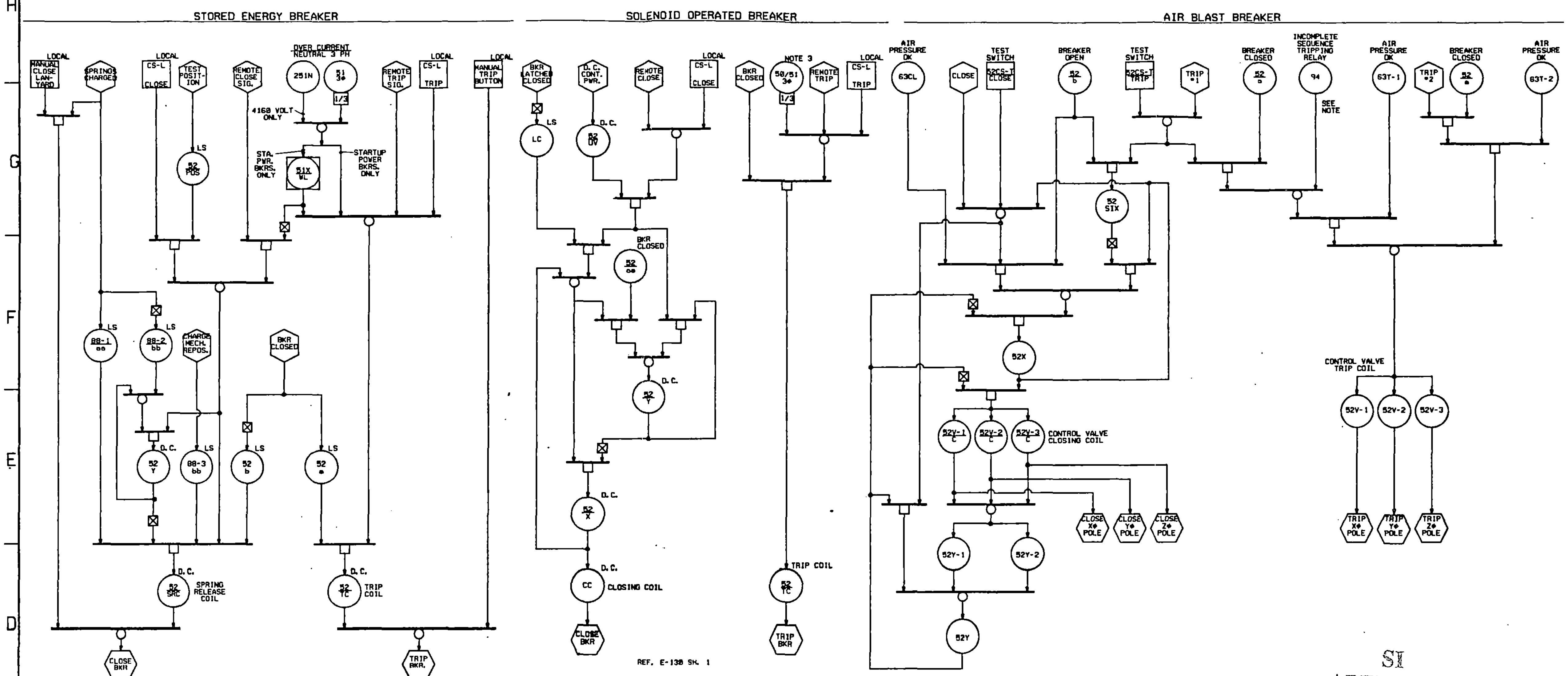
LOGIC DIAGRAM
SAFEGUARD TRANSFORMER 1-1
PALISADES PLANT

LOC	BASE	SHEET	REV
0950	E17	18A	0

L D

Figure No. 7-14 sh 12 FSAR Rev. 11

9012260101-24



NOTES:

1. 88-1 & 88-2- OPERATED BY A PIN ON THE CLOSING SPRING LINKAGE. WHEN THE SPRINGS ARE CHARGED aa CLOSES. ON SPRINGS DISCHARGE bb CLOSES TO START THE MOTOR AND CHARGE SPRINGS.

   88-3- OPERATED BY CAM FOLLOWERS. CONTINUES TO ENERGIZE MOTOR AT REDUCED SPEED AFTER SPRINGS ARE CHARGED. WHEN OPERATING CAM FOLLOWERS ARE IN PROPER POSITION aa OPENS DEENERGIZING MOTOR. THE bb CLOSES COMPLETING THE SRC CIRCUIT.

2. REF. E-129

3. PCP & D.G. BREAKERS ALSO HAVE 3 PHASE DIFFERENTIAL (87) RELAY PROTECTION.

NOTES:
"94" RELAY IS ACTUATED WHEN BREAKER DOES NOT FULLY CLOSE ON DEMAND

SI APERTURE CARD

Also Available On Aperture Card

ISSUED PER ENGINEERING REVIEW

SCALE NONE — B.N. YOUNG — P.D. BROWN

PALISADES PLANT
CONSUMERS POWER COMPANY

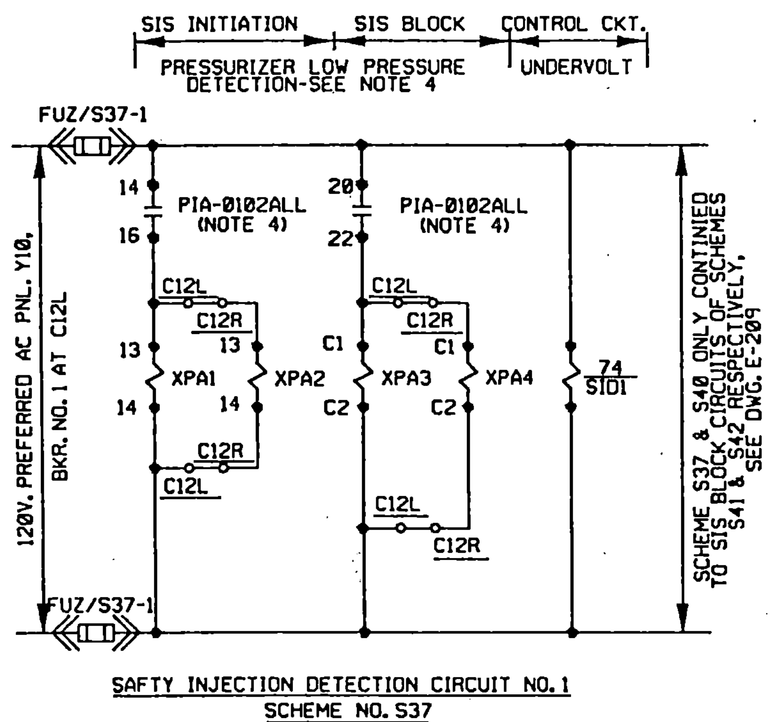
LOGIC DIAGRAM
CIRCUIT BREAKER OPERATION

0950	E17 SH. 19	0

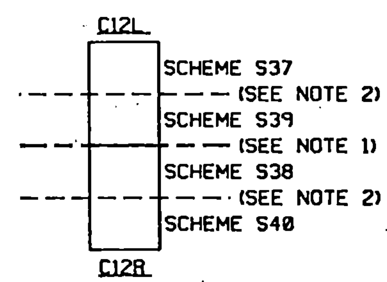
(40,100)0950E17.019

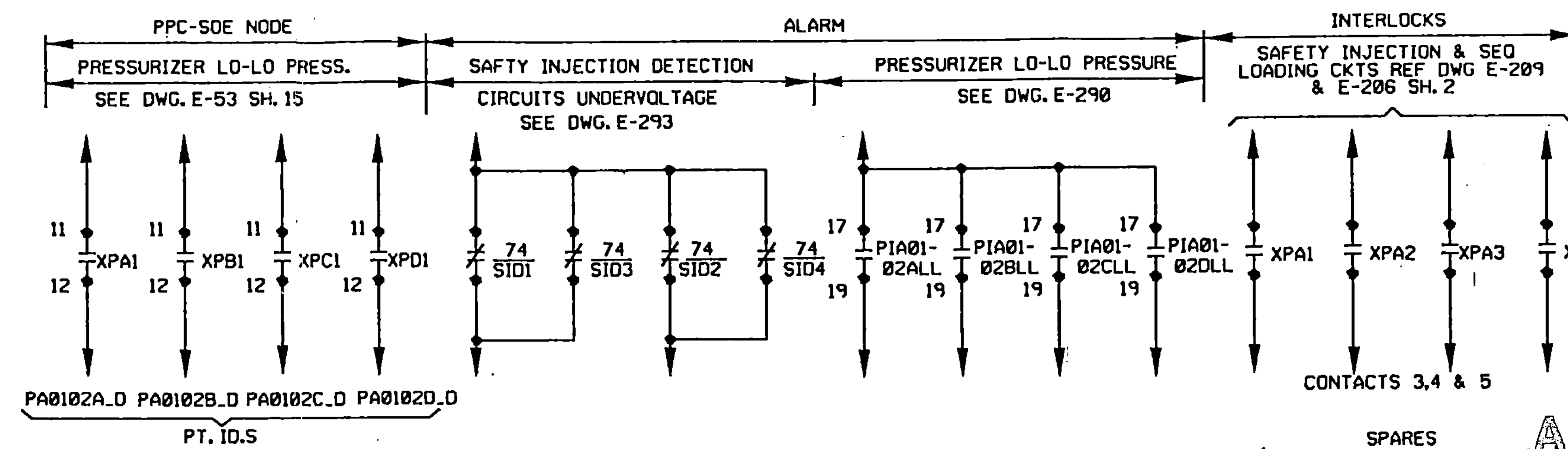
Figure No. 7-14 sh 13 FSAR Rev. 11

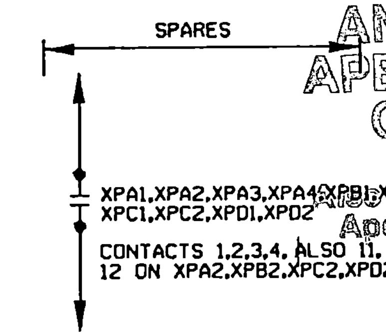
9012260101-25



SAFTY INJECTION DETECTION CIRCUIT NO. 1
SCHEME NO. S37







NOTES:
1. SEPARATE RACEWAY SYSTEM
2. BARRIER BETWEEN CHANNELS WITHIN RACEWAY.
3. REF. C E. DWG. 2966-D-3106 (FP5935NSSS-285) 2966-D-3117(FP-5935NSSS-287),HARLO DWG. E-1266 (FP M201-58)
4. FOR PIA INSTRUMENTATION WIRING, SEE DWG. E-84 SH. 2
CONTACT 14-16 OPENS ON DECREASING PRESSURE.
CONTACT 17-19 OPENS ON DECREASING PRESSURE.
CONTACT 20-22 OPENS ON INCREASING PRESSURE.
15
5. 4-4 LAMP LIGHT UNITS MOUNTED ON RELAY CABINETS. LENSE ENGRAVED WITH RELAY NO'S.
6. 4 LAMP MICRO-SWITCH LIGHT UNIT MTD. ON LEFT RELAY BOX.

ADAPTER TABLE

DESCRIPTION	SCHEME NO.	PRESSURIZER LEVEL & PRESSURE RELAYS		AUXILIARY RELAYS								SUPPLY PWR.		U.V. RELAY 74/	FUSE NO.
				LOW PRESSURE RELAYS				SIS BLOCK RELAYS							
		DEVICE	LOCATION	DEV.	LOC.	DEV.	LOC.	DEV.	LOC.	DEV.	LOC.	PNL.	BKR.		
S.I. DETECTION CKT. NO. 1	S37	PIA-0102ALL	C12L	XPA1	C12L	XPA2	C12R	XPA3	C12L	XPA4	C12R	Y10	1	SID1	FUZ/S37-1
S.I. DETECTION CKT. NO. 2	S38	PIA-0102BLL	C12R	XPB1	C12R	XPB2	C12L	XPB3	C12R	XPB4	C12L	Y20	1	SID2	FUZ/S38-1
S.I. DETECTION CKT. NO. 3	S39	PIA-0102CLL	C12L	XPC1	C12L	XPC2	C12R	XPC3	C12L	XPC4	C12R	Y30	1	SID3	FUZ/S39-1
S.I. DETECTION CKT. NO. 4	S40	PIA-0102DLL	C12R	XPD1	C12R	XPD2	C12L	XPD3	C12R	XPD4	C12L	Y40	1	SID4	FUZ/S40-1

REV	DATE	DESCRIPTION	BY	CK	APP
15	10-10-97	REV'D. NOTE 4 PER DCR-97-850.	JGO	MDS	

CONSUMERS POWER COMPANY
PALISADES NUCLEAR PLANT
COVERT, MICHIGAN
E206-1.DGN

PREVIOUS RECORD ISSUE
MICHIGAN P.E. No.
UFI No.

SCHEMATIC DIAGRAM
SAFETY INJECTION SIGNAL
AUXILIARY CIRCUITS

DRAWING NUMBER	SHT. NO.	REV
E-206	1	15

FIG 7-15
FSAR REV 20

9807090043 -15

APPENDIX 7A

ENGINEERED SAFEGUARDS TESTING

ENGINEERED SAFEGUARDS TESTING

7A.1 TESTING DESCRIPTION

7A.1.1 TESTING PROGRAM

7A.1.1.1 Overall Testing

Procedure QO-1, "Safety Injection" demonstrates the operability of the Safety Injection System (SIS) initiation circuitry by using the internal testing capability of the system. This procedure tests the performance of the SIS circuits during a simulated SI, both with and without offsite power available. This test is run quarterly with the Primary Coolant System pressure greater than 1,400 psia. Other monthly test procedures are also conducted at power:

MI-4, "Pressurizer Low Pressure SIS Initiation Functional Check"

MI-5, "Containment High-Pressure Test"

Specific procedures are provided for testing with the reactor shut down. These procedures are listed below:

QO-8B, "ESS Check Valve Operability Test"(Cold shutdown)

RT-8C+D, "Engineered Safeguards System"

RO-12, "CHP Spray System Tests"

RI-14, "SIRW Tank Level Switch Interlocks Tests"

Procedures RT-8C+D are conducted at every refueling to determine the operability of the Emergency Power System, engineered safeguards system and the manual safety injection feature of the engineered safeguards controls. This is determined by verifying correct sequencing and loading of safeguards equipment (including all ECCS pumps) when an SIS actuation is simulated (by manually initiating a simulated SIS) coincident with a simulated loss of offsite power. The manual SIS feature is verified by alarm indication after the SIS push button is depressed. This test duplicates, as close as practical, the integrated performance required from the engineered safety features actuation system (ESFAS), the engineered safety features (ESF) and their auxiliary support systems as defined in the design criteria of Subsection 7.3.5.1. SIS circuit design is such that each redundant circuit (or channel) is tested separately, so that the correct operation of each circuit can be identified.

Left channel procedure RT-8C and right channel procedure RT-8D are performed during refueling outages to demonstrate the operability of the normal shutdown sequence through testing the equipment starting circuits and contact closure times. Resetting SIS while bus 1C or 1D is being fed exclusively by the diesel generator, initiates the NSD Sequencer. Sequencer timing is verified. Loads may or may not be started. NSD and DBA Sequencer testing provides an adequate overlap to ensure proper equipment operation and meet Technical Specification testing requirements.

Procedure RI-14 requires that the RAS be tested by lifting the safety-related SIRW tank level switch probes above the water level to effect an auto RAS actuation. Upon actuation, the control room recirculation initiation alarm is verified. This alarm is wired to a contact on one of the SIRW tank low-level relays. Upon RAS initiation, these relays energize, and among other things, cause the alarm. This procedure tests all possible one-out-of-two (taken twice) level switch combinations.

Quarterly Procedure QO-2, "Recirculation Actuation System" (which, like RI-14, is conducted also during shutdown) also tests the RAS. In this test, however, a test switch is used to simulate the RAS initiation condition (low SIRW tank level). Upon placing this switch in the "test" position, the SIRW tank low-level relays are energized to effect RAS initiation. Actual component response verifications are made in this procedure. All required component response verifications are made except for the closure of the low-pressure safety-injection pump minimum recirculation valves, which is done at shutdown. The component response verifications include:

1. Low-pressure safety injection pumps trip

2. SIRW tank isolation valves close

3. Containment sump valves open

4. Component cooling water heat exchanger main service water valves open while the heat exchanger bypass service water valves close

5. Component cooling water heat exchanger cooling water valves open

Quarterly Procedure QO-38 provides verification for full stroke exercising of the containment sump check valves.

7A.1.1.2 Pump Testing

Pump head and flow, are tested for the ECCS pumps during the quarterly inservice inspection pump tests. During these tests, it is required that the pumps operate for at least 15 minutes. These tests are:

QO-16 for the containment spray pumps,
QO-19 for the high-pressure injection pumps, and
QO-20 for the low-pressure safety injection pumps.

The high-pressure safety-injection pumps, low-pressure safety-injection pumps (shutdown cooling pumps) and the containment spray pumps are tested quarterly. According to Test Procedures QO-16, QO-19 and QO-20, the method of starting each pump is alternated between the control room and the local breaker every quarter.

7A.1.1.3 Instrumentation Testing

In accordance with Procedures D/WO-1 "Operators Daily/Week/Biweekly Items" and SHO-1 "Operators' Shift Items," all of the safety-injection and containment spray initiation instrumentation that features control room indication (such as pressurizer-pressure channels and containment high-radiation channels), are checked either daily or each shift. The safety-related containment high-pressure circuits do not feature control room indication and are, therefore, tested monthly along with the other SIS and containment spray initiation circuitry. The procedures listed below are used to perform these tests.

MI-4, "Pressurizer Low Pressure SIS Initiation Functional Check"

MI-5, "Containment High-Pressure Initiation Test"

MI-6, "Area Monitors, Operational Check"

As previously mentioned, Procedure QO-1 (which is performed quarterly) tests the operation of the starting circuits and verifies that the active components (ie, pumps and valves) operate satisfactorily upon receipt of the SIS signal.

7A.1.1.4 Engineered Safeguards Passive Devices Testing

Procedure SHO-1 requires that the safety injection tank level and pressure be checked each shift during power operation.

Procedure QO-8B is used during cold shutdown to full flow test the LPSI check valves, to part stroke the HPSI/RHPSI check valves, and to stroke the hot leg injection check valve. The safety injection flow indication and safety injection tank leakoff control system are used to verify flow and monitor the test pressures between the valves. Primary system drain tank level instrumentation is used to verify flow through the hot leg injection valve.

7A.1.2 TEST METHODS

Since the engineered safeguards equipment being initiated varies according to whether power is available from the offsite source or the diesel generator, mode selector switches are provided so that either the normal SIS or the design basis accident (DBA) portions of the circuit can be tested separately. Individual momentary-type push buttons are provided to simulate the SIS in each of the redundant control circuits. The test is in progress only as long as the push button is depressed. Releasing this push button during a test will automatically reset the SIS or DBA sequence relays.

A momentary-type push button is provided to simulate the SIS in each of the redundant control channels. Procedure QO-1 calls for the use of this SIS test push button as a means of system initiation. QO-1 utilizes the left push button for left channel testing and the right push button for right channel testing. As described in QO-1, the test is terminated upon releasing the push button. The SIS relays or DBA sequencer will reset automatically.

Testing in the "without offsite power" mode does not initiate load shedding, since load shedding is purely a function of actual voltage on the emergency buses. Each component that features load shed input circuitry, utilizes a load shed "a" contact in its trip circuits. This "a" contact closes to provide component trip whenever the emergency bus de-energizes.

Procedure QO-1 simulates the SIS by requiring that the momentary test push buttons be depressed. Upon depressing the button, the test requires that the operation verifies proper load response. An alternate method of initiating the SIS is by tripping two-out-of-four pressurizer low-low pressure instruments in the SIS initiating circuit matrix. Procedures RT-8C + D actually call for this method of SIS initiation.

Procedure QO-1 simulates the loss of offsite power and sequences the loads. Procedures RT-8C + D verify bus shedding and actual sequence loading of components by causing an actual loss of power to each of the Class 1E buses.

7A.1.3 ACCEPTANCE CRITERIA

As previously described, the procedures used to test the Safety Injection System are QO-1 and RT-8C + D. The acceptance criteria for each of these are stated in the procedures.

Acceptance criteria for pump shutoff head at minimum recirculation flow and pump operability is included in the quarterly inservice inspection pump tests as given earlier.

7A.2 TESTING OVERLAP EVALUATION

7A.2.1 SAFETY INJECTION

The SIS is initiated by either pressurizer low-pressure or containment high-pressure conditions. During refueling, Procedure RI-3, "High Pressurizer Pressure Channels Calibration," requires that a test pressure be input to the safety-related pressurizer pressure sensors (transmitters). Upon reaching the proper set point by varying the pressure input, the appropriate pressurizer pressure indicator and alarm (PPIA) unit actuation is verified. Procedure RI-7, "Low-Pressure SIS Initiation Logic," overlaps with RI-3, in that RI-7 requires that a signal generator, connected into the various pressurizer pressure current loops (upstream of the PPIAs) be used to activate various combinations of the four PPIA units (one per channel) to produce the two-out-of-four pressurizer low-pressure trip inputs to the SIS. The SIS initiation logic is verified by the safety injection test indicating lamps, which are energized by safety injection auxiliary relays, and by verifying voltage which would energize the SIS output relays. Actual actuation of the SIS output relays takes place during the performance of RT-8C, "Engineered Safeguards System - Left Channel," and RT-8D, "Engineered Safeguards System - Right Channel." During the performance of RO-12, "Containment High Pressure (CHP) and Spray System Tests," only the SIS signal is verified. SIS is not initiated. RO-12 in conjunction with RT-8C and RT-8D provides proper testing overlap to ensure CHP will initiate safety injection. Monthly Procedure MI-4, "Pressurizer Low-Pressure SIS Initiation Functional Check," also tests these circuits in a method similar to Procedure RI-7. In MI-4, however, the 2/4 combinations are not verified. MI-4 simply verifies one channel at a time.

RO-12 requires that a test pressure be inserted into the safety-related CHP sensors (pressure switches). Upon reaching the proper set point, CHP is initiated. The CHP signal provides an initiation signal to the SIS circuitry. However, the containment high-pressure (CHP) input to the SIS is isolated during the performance of RO-12. Therefore, the initiation signal is verified by voltage readings. This minimizes the impact on plant equipment.

The SIS is tested quarterly per Section QO-1, "Safety Injection System," to demonstrate the operability of the SIS circuitry by using the internal testing capability of the system. This test overlaps Tests RI-7 and RO-12 in that, during the QO-1 test, the internal test circuits are used to simulate a safety-injection condition with and without offsite power available which energizes the SIS output relays. Upon initiation, all of the engineered safeguards loads are verified to respond appropriately.

During refueling, Procedures RT-8C + D, "Engineered Safeguards System," require that an actual loss of voltage occurs in the emergency buses concurrent with a trip of the PPIA units to effect a full design basis accident. Proper sequencing and timing of the sequenced loads are verified. Also verified, is the appropriate response of the other engineered safeguards loads. This test overlaps Tests RI-7, RO-12 and QO-1.

7A.2.2 CONTAINMENT ISOLATION

The containment isolation function is verified in one "system test." As previously mentioned, Procedure RO-12 requires that a test pressure be inserted into the CHP sensors (pressure switches). In addition to verifying SIS initiation, the procedure also requires that containment isolation (CI) actuation be verified. The verification is to be made by logging the response of at least one containment isolation valve for each of the CHP relays that energize upon receipt of the CHP condition as sensed by the pressure switches (sensors). The response is checked by valve position indication lights. Although overlap is not a problem with this test, the test does not verify the leak tightness of the containment isolation valves to the CHP condition. This is verified via the Containment Leak Rate testing program (Procedure RO-32).

In addition, the containment high-radiation input to containment isolation is verified. Procedures RI-86F, RO-11 and MR-6 are used. RI-86F, "Containment Isolation Monitor Calibration," utilizes a known external radiation source to verify the proper safety-related area monitor output. A current source is also used to simulate detector signal input and is adjusted until the high (trip) is reached. Verification of trip is through a RIAX relay contact initiating a control room alarm. During the conduct of RO-11, "Containment High Radiation Test," the radiation monitors are de-energized in each possible 2-out-of-4 combination. This causes the high (trip) circuit to de-energize in the radiation monitors to complete the logic actuation. The response of all the containment isolation valves is verified.

Valid verification of trip action with sufficient testing overlap is provided with the RI-86F and RO-11 procedure combination. The trip circuitry in the radiation monitors is fully functionally checked by increasing a simulated input to above the trip setpoint. The bistable comparison between the input and trip setpoint causes the output relay to de-energize. This in turn causes the RIAX relay in containment isolation 2/4 scheme to de-energize thus alarm. Removing power to the radiation monitor will also de-energize the output relay and cause RIAX actuation.

Procedure MI-6, "Area Monitor Functional Checks," requires verification of high alarm or trip setpoint on a monthly basis. Verification is by adjusting an internal current to the input of the meter amplifier to check the pre-trip and trip setpoints. An internal circuit prevents actual trip relay actuation in this mode. The setpoints are only adjusted if acceptance criteria is violated.

7A.2.3 RECIRCULATION ACTUATION

Proper overlap exists in the recirculation actuation testing program. Test RI-14 requires that the safety-related SIRW tank level switch probes be lifted which energize the SIRW tank low-level relay as procedurally verified by a control room alarm which is energized by these relays. Procedure QO-2 utilizes a test switch to energize the SIRW tank low-level relays, to actuate the required loads. Load response is then procedurally verified. These two tests overlap sufficiently to enable them to be combined to provide a valid system verification. Although the tests are performed only during shutdown, they duplicate (as closely as practical) the integrated performance of the RAS which is required in the event of an accident. This is consistent with the intent of 10 CFR 50, Appendix A, General Design Criterion 37; Regulatory Guide 1.22 and Standard Review Plan, Section 7.3, Appendix A.

The RAS is not designed to be tested while the reactor is at power. Shutting the SIRW tank outlet valves would result in a violation of the Technical Specifications, since shutting either of these valves would eliminate a source of water to more than one high-pressure or low-pressure safety-injection pump. It should be noted that upon RAS initiation, both of the SIRW tank outlet valves receive an auto closure signal and the containment sump isolation valves receive an auto open signal. Therefore, should an RAS initiation occur during normal conditions, the ECCS (or portions thereof) would be aligned to take suction from an empty sump for some period of time.

APPENDIX 7B

DELETED
(per 7-60-R13-286)

APPENDIX 7C
REGULATORY GUIDE 1.97 INSTRUMENTATION

The attached tables provide a comparison of the instrumentation provided at Palisades against the requirements of Regulatory Guide 1.97 Revision 3. These tables provide the following information for each parameter: Item, Tag Number, Variable Description, Type and Category, Existing and Required Instrument Ranges, QA Requirements, Environmental and Seismic Qualification, Redundance, Power Supply Display Location and a Comment section which provides a schedule for instrument loop upgrade or justification for acceptability of existing noncomplying instrumentation. The information provided in each column of the table is defined as follows:

ITEM: Consists of the RG 1.97 category followed by a sequential number. Item numbering is consistent with the RG 1.97 ordering of parameters.

TAG NO: This column lists component ID's of the sensors, indicators, power supplies, displays and recorders in each instrument loop selected to compare against the RG 1.97 requirements.

VARIABLE DESCRIPTION: Provides a description of the variable as taken from RG 1.97 Revision 3 Table 3.

VARIABLE TYPE: Lists the variable type as defined by RG 1.97.

VARIABLE CATEGORY: Lists the variable category as defined by RG 1.97

QA REQUIREMENT: Describes compliance with the RG 1.97 QA requirement for the specific variable category. Descriptions provided in this in this column are defined as follows:

- PRE-QA: Equipment was procured and installed prior to the establishment of a formal QA program. Equipment considered acceptable based on successful operating experience.
- COMPLY: Equipment was procured and installed under auspices of Palisades QA program.
- N/A: QA requirement not applicable to this category.

ENVIRONMENTAL QUALIFICATION: Describes compliance with the RG 1.97 EEQ requirements for the specific variable category. Descriptions provided in this column are defined as follows:

- COMPLY: Equipment located in a harsh environment are included in the Palisades Plant Equipment Qualification List and/or the equipment is located in a mild environment.
- N/A: EEQ requirements not applicable to this category.

SEISMIC QUALIFICATION: Describes compliance with the RG 1.97 Seismic requirements for the specific variable category. Descriptions provided in this column are defined as follows:

- COMPLY: Equipment is seismically qualified to criteria described in FSAR.
- N/A: Seismic qualification not applicable to this category.

REDUNDANCE: Indicates the number of instrumentation loops meeting the RG 1.97 requirements provided for this variable. Indicates N/A if redundance not applicable for this category.

POWER SUPPLY: Describes the type power provided to energize the instrument loop. Descriptions provided in this column are defined as follows:

PREFERRED 1E: Instrument loop powered by battery backed 1E ac power.

RELIABLE NON-1E: Instrument loop powered by Non-1E ac which is capable of being energized from the standby power sources and for which procedural guidance is provided for bus restoration.

1E BATTERY: Instrument loop is 1E dc powered.

Non-1E: Instrument loop ac power not backed up by standby power supply.

DISPLAY LOCATION: Indicates control room (CR) panel where variable is indicated. For the Technical Support Center (TSC) and Emergency Offsite Facility (EOF) indicates if the parameter is available via the Critical Function Monitor System (CFMS) computer.

COMMENT: Provides additional information where required. Also provides justification for acceptability of existing instrumentation not in compliance with RG 1.97 requirements.

ITEM	TAG NO	VARIABLE			INSTRUMENT RANGE		QA REQUIRE-MENT	ENVIRONMENTAL QUALIFICATION	SEISMIC QUALIFICATION	REDUNDANCE	POWER SUPPLY	DISPLAY LOCATION			COMMENTS
		DESCRIPTION	TYPE	CATA-GORY	EXISTING	REQUIRED						CR	TSC	EOF	
A01	Inputs Loop 1 TE0112CC TT0112CC I/I0112CC TE0112CD TT0112CD I/I0112CD TE0112HC TT0112HC I/I0112HC TE0112HD TT0112HD I/I0112HD PT0105A SMM0114 Inputs Loop 2 TE0122CC TT0122CC I/I0122CC TE0122CD TT0122CD I/I0122CD TE0122HC TT0122HC I/I0122HC TE0122HD TT0122HD I/I0122HD PT0105B SMM124	Degrees of Subcooling	A	1	200°F Sub-cooling to 35°F superheat	200°F Sub-cooling to 35°F superheat	Comply	Comply	Comply	2 Channels	Preferred 1E	C12	CFMS	CFMS	Indication used to initiate trip of primary coolant pumps. Also used to allow termination or throttling of SIS flows.

ITEM	TAG NO	VARIABLE			INSTRUMENT RANGE		QA REQUIRE-MENT	ENVIRONMENTAL QUALIFICATION	SEISMIC QUALIFICATION	REDUNDANCE	POWER SUPPLY	DISPLAY LOCATION			COMMENTS
		DESCRIPTION	TYPE	CATA-GORY	EXISTING	REQUIRED						CR	TSC	EOF	
A02	PT0105A P/S0751C PI0105C PTR0112 PT0105B P/S0751D PI0105A	Pressurizer Pressure	A	1	0-3000 PSIG	0-4000 PSIG	Comply	Comply	Comply	2 Channels	Preferred 1E	C12 C02	CFMS	CFMS	Indication used to initiate trip of primary coolant pump following small break LOCA. See Note 13.
A03	Stm Gen A LT0757A P/S0757A LI0757A LT0757B P/S0757B LI0757B STM GEN B LT0758A P/S0758A LI0758A LT0758B P/S0758B LI0758B	Steam Generator Level	A	1	100% to - 138% (Equivalent to tube sheet to steam separators)	Tube Sheet to Steam Separators	Comply	Comply	Comply	2 Channels /Steam Generator	Preferred 1E	C12	CFMS	CFMS	Indication used to determine steam generator with ruptured tube to be isolated. Indication also used to initiate once thru cooling on low/low level.

ITEM	TAG NO	VARIABLE			INSTRUMENT RANGE		QA REQUIRE-MENT	ENVIRONMENTAL QUALIFICATION	SEISMIC QUALIFICATION	REDUNDANCE	POWER SUPPLY	DISPLAY LOCATION			COMMENTS
		DESCRIPTION	TYPE	CATA-GORY	EXISTING	REQUIRED						CR	TSC	EOF	
A04	Stm Gen A PT0751C P/S0751C PIC0751C PT0751D P/S0751D PIC0751D Stm Gen B PTO752C P/S0751C PIC0752C PT0752D P/S0751D PIC0752D	Steam Generator Pressure	A	1	0-1200 PSIG	From Atmospheric to 20% above lowest safety valve setting	Comply	Comply	Comply	2 Channels /Steam Generator	Preferred 1E	C12	CFMS	CFMS	Indication used to determine ruptured steam generator to allow isolation following steam line break. Lowest relief valve setting 985 psig.
A05	LT0103 P/S0103 LI0103A LI0103B LT0102 P/S0751A LIA0102A LI0102B	Pressurizer Level	A	1	0-100% (Equivalent to top to bottom of vessel)	Top to Bottom of Vessel	Comply	Comply	Comply	2 Channels	Preferred 1E	C12 C02	CFMS	CFMS	Indication used to allow termination or throttling of SIS flows.
A06	AE2401L AIT2401L AI2401L AE2401R AIT2401R AI2401R	Containment Hydrogen Concentration	A	1	0-10 Vol% or 0-20 Vol% (switch selectable) from -2 to 60 PSIG	0-10 Vol% (capable of operating rom -2 to 60 PSIG)	Comply	Comply	Comply	2 Channels	Preferred 1E	C11A	CFMS	CFMS	Indication used to determine when to initiate hydrogen recombiners.

ITEM	TAG NO	VARIABLE			INSTRUMENT RANGE		QA REQUIREMENT	ENVIRONMENTAL QUALIFICATION	SEISMIC QUALIFICATION	REDUNDANCY	POWER SUPPLY	DISPLAY LOCATION			COMMENTS
		DESCRIPTION	TYPE	CATAGORY	EXISTING	REQUIRED						CR	TSC	EOF	
B01	NE-1/3 NI-1/3B NI-1/3D NI-1/3A EC-175 NE-2/4 NI-2/4B N-I2/4A EC-176	Neutron Flux	B	1	10^{-8} to 200% of full power	10^{-6} TO 100% of full power	Comply	Comply	Comply	2 Cannel	Preferred 1E	C06	CFMS	CFMS	
B02	CRD-1 Thru-45	Control Rod Position	B	3	Full in or not full in, analog position for any of 45 Control Rods	Full in or not full in	N/A	N/A	N/A	N/A	Preferred 1E	C12 C02	CFMS	CFMS	These indications are shown on the core map display and rod position display.
B03	AE0203 AR0203	RCS Soluble Boron Concentration	B	3	0-2050 ppm	0-6000 ppm	N/A	N/A	N/A	N/A	Reliable Non-1E	C02	CFMS	CFMS	Range is acceptable for normal operation. For post accident analyzer is isolated and boron analysis performed by grab sample.
B04	-	PCS Cold Leg Water Temperature	B	3		50°F to 400°F									Covered by Item B06.
B05	Loop 1 TE0112HC TT0112HC PTR0112 Loop 2 TE0122HD TE0122HD PTR0122	PCS Hot Leg Water Temperature	B	1	50°F to 700°F	50°F to 700°F	Comply	Comply	Comply	2 Channels	Preferred 1E	C12	CFMS	CFMS	

ITEM	TAG NO	VARIABLE			INSTRUMENT RANGE		QA REQUIRE-MENT	ENVIRONMENTAL QUALIFICATION	SEISMIC QUALIFICATION	REDUNDANCE	POWER SUPPLY	DISPLAY LOCATION			COMMENTS
		DESCRIPTION	TYPE	CATA-GORY	EXISTING	REQUIRED						CR	TSC	EOF	
B06	Loop 1 TE0112CC TT0122CC PTR0112 Loop 2 TE0122CD TT0122CD PTRO122	PCS Cold Leg Water Temperature	B	1	50°F to 700°F	50°F to 700°F	Comply	Comply	Comply	2 Channels	Preferred 1E	C12	CFMS	CFMS	
B07		PCS Pressure (Pressurizer Pressure)	B	1		0-400 PSIG									Covered by A02.
B08		Core Exit Temperature	B	3		0°F to 2300°F									Covered by Item C01.
B09	LE0101A LTRI0101A LE0101B LTRI0101B	Coolant Inventory	B	1	Top of Core to Top of Vessel	Bottom of Hot Let to Top of Vessel	Comply	Comply	Comply	2 Channels	Preferred 1E	C11A	CFMS	CFMS	
B10		Degrees of Subcooling	B	2		200°F subcooling 35°F superheat									Covered by A01.
B11		PCS Pressure (Pressurizer Pressure)	B	1		0-4000 psig									Covered by Item A02.
B12	LT0383 P/S1812A LPIR0383 LT0382 P/S1805A LPIR0382	Containment Sump Water Level (Narrow Range)	B	2	0-100% (Bottom to Top of Sump)	Narrow Range (Sump)	Comply	Comply	N/A	2 Channels	Preferred 1E	C13	CFMS	CFMS	

ITEM	TAG NO	VARIABLE			INSTRUMENT RANGE		QA REQUIRE-MENT	ENVIRONMENTAL QUALIFICATION	SEISMIC QUALIFICATION	REDUNDANCE	POWER SUPPLY	DISPLAY LOCATION			COMMENTS
		DESCRIPTION	TYPE	CATA-GORY	EXISTING	REQUIRED						CR	TSC	EOF	
B13	LE0446A LIT0446A LPIR0383 LE0446B LIT0446B LPIR0382	Containment Water Level (Wide Range)	B	1	0-100% (Exceeds Maximum Expected Water Level by 1.5 ft)	Wide Range (Plant Specific)	Comply	Comply	Comply	2 Channels	Preferred 1E	C13	CFMS	CFMS	
B14	-	Containment Pressure	B	1		0 - Design Pressure									Covered by C12.

ITEM	TAG NO	VARIABLE			INSTRUMENT RANGE		QA REQUIRE-MENT	ENVIRONMENTAL QUALIFICATION	SEISMIC QUALIFICATION	REDUNDANCE	POWER SUPPLY	DISPLAY LOCATION			COMMENTS
		DESCRIPTION	TYPE	CATA-GORY	EXISTING	REQUIRED						CR	TSC	EOF	
B15	POS0155 POS0738 POS0739 POS0767 POS0768 POS0770 POS0771 POS0910 POS0911 POS0939 POS0940 POS1001 POS1002 POS1004 POS1007 POS1036 POS1037 POS1038 POS1044 POS1045 POS1064 POS1065 POS1101 POS1102 POS1103 POS1104 POS1358 POS1501 POS1502 POS1503 POS1910 POS1911 POS2009 POS2083 POS2099 POS3001 POS3002 POS3029A POS3029B POS3030A POS3030B POS1805 POS1806 POS1807 POS1808 POS1813 POS1814	Containment Isolation Valve Position	B	1	Closed - Not Closed	Closed - Not Closed	Comply	Comply	Comply	Redundant Isolation Method for each flow path. Redundant Position Indication for each valve not provided.	1E Battery	Vario us	CFMS	CFMS	See Note 14.

ITEM	TAG NO	VARIABLE			INSTRUMENT RANGE		QA REQUIRE-MENT	ENVIRONMENTAL QUALIFICATION	SEISMIC QUALIFICATION	REDUNDANCE	POWER SUPPLY	DISPLAY LOCATION			COMMENTS
		DESCRIPTION	TYPE	CATA-GORY	EXISTING	REQUIRED						CR	TSC	EOF	
B16		Containment Pressure	B	1		-5 PSIG to Design Pressure									Covered by Item C12

ITEM	TAG NO	VARIABLE			INSTRUMENT RANGE		QA REQUIRE-MENT	ENVIRONMENTAL QUALIFICATION	SEISMIC QUALIFICATION	REDUNDANCE	POWER SUPPLY	DISPLAY LOCATION			COMMENTS
		DESCRIPTION	TYPE	CATA-GORY	EXISTING	REQUIRED						CR	TSC	EOF	
C01	INCE-14H02 INCE-16G07 INCE-21N11 INCE-25RO8 INCE-32Q04 INCE-42R17 INCE-43X14 INCE-52M13 INCE-66J16 INCE-71H13 INCE-73B13 INCE-76G17 INCE-86J10 INCE-81J07 INCE-36R07 INCE-41Q16 LTRI0101A LTRI0101B	Core Exit Temperature	C	1	0°F to 2300°F	200°F to 2300°F	Comply	Comply	Comply	2 Channels (8 Thermo-couples per Channel)	Preferred 1E	C11A	CFMS	CFMS	See Note 15.
C02	None	Radioactivity Concentration or Radiation Level in Circulating Primary Coolant	C	1	-	½ Tech Spec Limit 50 to 100 times Tech Spec Limit									Online Analysis capability isolated during accident. Grab sample to be used to evaluate variable.
C03	None	Analysis of Primary Coolant (Gamma Spectrum)	C	3	-	10μCi/ml to 10Ci/ml or TID14844 source term in coolant volume									No online system for analysis available. Grab sample to be used to evaluate variable.
C04		PCS Pressure (Pressurizer Pressure)	C	1	-	0-4000									Covered by Item A02.
C05		Containment Pressure	C	1	-	-5 PSIG to design pressure									Covered by Item C12.
C06		Containment Sump Water Level (Narrow Range)	C	2	-	Narrow Range (Sump)									Covered by Item B12.

ITEM	TAG NO	VARIABLE			INSTRUMENT RANGE		QA REQUIRE-MENT	ENVIRONMENTAL QUALIFICATION	SEISMIC QUALIFICATION	REDUNDANCY	POWER SUPPLY	DISPLAY LOCATION			COMMENTS
		DESCRIPTION	TYPE	CATA-GORY	EXISTING	REQUIRED						CR	TSC	EOF	
C07		Containment Water Level (Wide Range)	C	1	-	Wide Range (Plant Specific)									Covered by Item B13.
C08		Containment Area Radiation	C	3	-	1 R/HR to 10^4 R/HR									Covered by Item E01.
C09	RE0631 RIA0631	Effluent Radioactivity Noble Gas Effluent from Condenser air Removal System Exhaust	C	3	10^0 to 10^5 CPM (Equivalent to 1×10^{-5} to 2×10^{-2} μci/cc	10^{-6} μc/cc to 10^{-2} μc/cc	N/A	N/A	N/A	N/A	Preferred 1E	C13	CFMS	CFMS	
C10		PCS Pressure (Pressurizer Pressure)	C	1	-	0-4000 PSIG									Covered by Item A02.
C11		Containment Hydrogen Concentration	C	1		0-10 Vol-% (Capable of Operating rom -5 PSIG to maximum design pressure)									Covered by Item A06.
C12	PT1812A P/S1812A LPIR0383 PT1805A P/S1805A LPIR0382	Containment Pressure	C	1	-14.7 to 185.34 PSIG	-5 PSIG to 3 times Design Pressure for Concrete	Comply	Comply	Comply	2 Channels 1E	Preferred	C13	CFMS	CFMS	Design Pressure 55 PSIG
C13		Containment Effluent Radioactivity Noble Gases from Identified Release Points	C	2	-	10^{-6} μ Ci/cc to 10^{-2} μ Ci/cc									Covered by item E03

ITEM	TAG NO	VARIABLE			INSTRUMENT RANGE		QA REQUIRE-MENT	ENVIRONMENTAL QUALIFICATION	SEISMIC QUALIFICATION	REDUNDANCE	POWER SUPPLY	DISPLAY LOCATION			COMMENTS
		DESCRIPTION	TYPE	CATA-GORY	EXISTING	REQUIRED						CR	TSC	EOF	
C14		Effluent Radioactivity Noble Gases (From building or areas where penetrations and hatches are located)	C	2	-	10^{-6} μ Ci/cc to 10^{3} μ Ci/cc									Covered by Item E03

ITEM	TAG NO	VARIABLE			INSTRUMENT RANGE		QA REQUIRE-MENT	ENVIRONMENTAL QUALIFICATION	SEISMIC QUALIFICATION	REDUNDANCE	POWER SUPPLY	DISPLAY LOCATION			COMMENTS
		DESCRIPTION	TYPE	CATA-GORY	EXISTING	REQUIRED						CR	TSC	EOF	
D01	FT0306 FIC0306	RHR System Flow	D	2	O-8000 GPM GPM (6000 GPM design Flow)	0-110% Design Flow	Pre-QA	Comply	N/A	N/A	Reliable Non-1E	C02	CFMS	CFMS	All components located in mild environment
D02	TT0351B TR0351 TE0351B	RHR Heat Exchanger outlet Temperature	D	2	0°F to 400°F	40°F to 350°F	Pre-QA	Comply	N/A	N/A	Reliable Non-1E	C12	CFMS	CFMS	
D03	LT0365 P/S0365 LIA0365 LM0365 LT0368 P/S0368 LIA0368 LM0368 LT0372 P/S0372 LIA0372 LM0372 LT0374 P/S0374 LIA0374 LM0374	Accumulator Tank Level	D	3	0-100% (Equivalent to 5% to 95% Tank Volume)	10% to 90% Volume	N/A	N/A	N/A	N/A	Preferred 1E	C13	CFMS	CFMS	Note 1
D04	PT0363 P/S0363 PIA0363 PT0367 P/S0367 PIA0367 PT0371 P/S0371 PIA0371 PT0369 P/S0369 PIA0369	Accumulator Tank Pressure	D	3	0 to 300 PSIG	0 to 750 PSIG	N/A	N/A	N/A	N/A	Preferred 1E	C13	CFMS	CFMS	SI Tank Designed for 300 PSIG. Note 1

ITEM	TAG NO	VARIABLE			INSTRUMENT RANGE		QA REQUIRE-MENT	ENVIRONMENTAL QUALIFICATION	SEISMIC QUALIFICATION	REDUNDANCE	POWER SUPPLY	DISPLAY LOCATION			COMMENTS
		DESCRIPTION	TYPE	CATA-GORY	EXISTING	REQUIRED						CR	TSC	EOF	
D05	VOP3041 VOP3045 VOP3049 VOP3052	Accumulator Isolation Valve Position	D	2	Closed or Open	Closed or Open	Pre-QA	None	N/A	N/A	1E Battery	C03	CFMS	CFMS	Note 2
D06	FT0212 FY0212 FIA0212 P/S0212	Boric Acid Charging Flow	D	2	0-140 GPM	0-110% Design Flow	Pre-QA	Comply	N/A	N/A	Reliable Non-1E	C02	CFMS	CFMS	Design Flow is 132 GPM. All components located in mild environment.
D07	FT0308 P/S0751A FI-0308A FI-0308B FM0308 FT0310 P/S0751B FI0310A FI0310B FM0310 FT0312 P/S0751C FI0312A FI0312B FM0312 FT0313 P/S0751D FI0313A FI0313B FM0313	Flow in HPI System (Flow to Cold Legs)	D	2	0-250 GPM per injection line	0-110% Design Flow	Comply	Comply	N/A	N/A	Preferred 1E	C13 C33	CFMS	CFMS	Design flow is 225 GPM per injection line.
D07A	FT0316 P/S0316 FI0316A FI0316B FT0317 P/S0377 FI0317A FI0317B	Flow in HPI System (Flow to Hot Legs)	D	2	0-350 GPM per injection line	0-110% Design Flow	Comply	Comply	N/A	1. N/A	Preferred 1E	C13			Design flow is 300 GPM per injection line.

ITEM	TAG NO	VARIABLE			INSTRUMENT RANGE		QA REQUIRE-MENT	ENVIRONMENTAL QUALIFICATION	SEISMIC QUALIFICATION	REDUNDANCE	POWER SUPPLY	DISPLAY LOCATION			COMMENTS
		DESCRIPTION	TYPE	CATA-GORY	EXISTING	REQUIRED						CR	TSC	EOF	
D08	FT0307 P/S0751A FI0307A FI0307B FM0307 FT0309 P/S0751B FI0309A FI0309B FM0309 FT0311 P/S0751C FI0311A FI0311B FM0311 FT0314 P/S0751D FI0314A FI0314B FM0314	Flow in LPSI System	D	2	0-2000 GPM per injection line	o-110% Design Flow	Comply	Comply	N/A	N/A	Preferred 1E	C13	CFMS	CFMS	Design flow is 1500 GPM per injection line.
D09	LT0331 P/S0331 LIA0331 LT0332A P/S0332A LIA0332A	Refueling Water Storage Tank Level	D	2	0-100% Equivalent to Top to Bottom	Top to Bottom	Pre-QA	Comply	N/A	N/A	Preferred 1E	C13	CFMS	CFMS	All components located in mild environment
D10	EAI2103 EAI2104 EAI2203 EAI2204	Reactor Coolant Pump Status	D	3	0-800 Amps	Electric Current	N/A	N/A	N/A	N/A	Same Bus as Powers Pump	C12	CFMS	CFMS	

ITEM	TAG NO	VARIABLE			INSTRUMENT RANGE		QA REQUIRE-MENT	ENVIRONMENTAL QUALIFICATION	SEISMIC QUALIFICATION	REDUNDANCE	POWER SUPPLY	DISPLAY LOCATION			COMMENTS
		DESCRIPTION	TYPE	CATA-GORY	EXISTING	REQUIRED						CR	TSC	EOF	
D11	FE1039 FM1039 FI1039 FE1040 FM1040 FI1040 FE1041 FM1041 FI1041 FE1042B FM1042B FI1042B FE1043B FM1043B FI1043B	Primary System Safety Valve Positions (including PORV and Code Valves)	D	2	Closed - Not Closed	Closed - Not Closed	Comply	Comply	N/A	N/A	Preferred 1E	C11A	CFMS	CFMS	
D12		Pressurizer Level	D	1		Top to Bottom									Covered By Item A05
D13	EAI1211 EAI1305	Pressurizer Heater Status	D	2	0-200 Amps Current	Electric	Pre-QA	Comply	N/A	N/A	Same bus as heaters	C02	CFMS	CFMS	All components in mild environment
D14	LT0116 LIA0116	Quench Tank Level	D	3	0-100%	Top to Bottom	N/A	N/A	N/A	N/A	Reliable Non-1E	C02	CFMS	CFMS	
D15	TE0116 TIA0116	Quench Tank Temperature	D	3	0°F to 350°F	50°F to 750°F	N/A	N/A	N/A	N/A	Reliable Non-1E	C02	CFMS	CFMS	Note 3
D16	PT0116 PIA0116	Quench Tank Pressure	D	3	0 to 25/100 PSIG Dual Range	0 to Design Pressure	N/A	N/A	N/A	N/A	Reliable Non-1E	C02	CFMS	CFMS	Rupture Disk at 100 PSIG.
D17		Steam Generator Level	D	1		From tube sheet to separators									Covered by Item A03
D18		Steam Generator Pressure	D	2		From atmospheric to 20% above lowest safety valve setting									Covered by Item A04

ITEM	TAG NO	VARIABLE			INSTRUMENT RANGE		QA REQUIRE-MENT	ENVIRONMENTAL QUALIFICATION	SEISMIC QUALIFICATION	REDUNDANCE	POWER SUPPLY	DISPLAY LOCATION			COMMENTS
		DESCRIPTION	TYPE	CATA-GORY	EXISTING	REQUIRED						CR	TSC	EOF	
D19	Stm Gen A FT0702 PT0702 LIC0701 FR0702 Stm Gen B FT0704 PT0704 LIC0703 FR0704	Safety/Relief Valve Positions or Main Steam Flow	D	2	Main Steam Flow 0-6 x 10^6 lbs/hr per Steam Generator	Closed - Not Closed	Pre-QA	Comply Note 6	N/A	N/A	Reliable Non-1E	C11	CFMS	CFMS	Design Flow 5.6 x 10^6 lbs/hr per Steam Generator
D20	Stm Gen A FT0701 PT0701 FR0701 LIC0701 Stm Gen B FT0703 PT0703 FR0703 LIC0703	Main Feedwater Flow	D	3	0-6 x 10^6 lbs/hr per Steam Generator	0-110% Design Flow	N/A	N/A	N/A	N/A	Reliable Non-1E	C11	CFMS	CFMS	Design Flow 5.6 x 10^6 lbs/hr per Steam Generator
D21	Stm Gen A FT0737 FM0737 FI0737 P/S0737A FT0749A FM0749A FI0749A P/S0727 Stm Gen B FT0727A FM0727A FI0727A P/S0727 FT0736 FM0736 FI0736 P/S0737A	Auxiliary Feedwater Flow	D	2	0-650 GPM	0-110% Design Flow	Comply	Comply	N/A	2 Channels per Steam Generator	Preferred 1E	C11 C01	CFMS	CFMS	Design Flow 415 GPM

ITEM	TAG NO	VARIABLE			INSTRUMENT RANGE		QA REQUIRE-MENT	ENVIRONMENTAL QUALIFICATION	SEISMIC QUALIFICATION	REDUNDANCE	POWER SUPPLY	DISPLAY LOCATION			COMMENTS
		DESCRIPTION	TYPE	CATA-GORY	EXISTING	REQUIRED						CR	TSC	EOF	
D22	LT2021 P/S2021 LIA2021 LT2022 P/S2022 LIA2022	Condensate Storage Tank Level	D	1	0-100% (Essentially Top to Bottom of vessel)	Plant Specific	Comply	Comply	Note 17	2 Channels	Preferred 1E	C13	CFM S	CFMS	All components located in a mild environment
D23	FT0301 FI0301A FI0301B FI0302 FI0302A FI0302B	Containment Spray Flow	D	2	0-3000 GPM	0-110% Design Flow	Pre-QA	Comply	N/A	2 Channels	Preferred 1E	C13 C33	CFM S	CFMS	Design Flow 2700 GPM All components located in a mild environment
D24	Cooler Valve <u>Position</u> POS0824 POS0847 POS0861 POS0862 POS0864 POS0865 POS0873 POS0870 Cooler Fan <u>Status</u> 52-1208 52-1209 52-1210 Service Water Pump <u>Status</u> 152-103 152-204 152-205	Heat removal by Containment fan heat removal system	D	2	Indicating lights	Plant Specific	Comply	Comply	N/A	N/A	Reliable 1E	C08	CFM S	CFMS	Status determined by confirming cooler valves aligned, fans running and service water available. Only cooler inlet and outlet valves located in a harsh environment.

ITEM	TAG NO	VARIABLE			INSTRUMENT RANGE		QA REQUIRE-MENT	ENVIRONMENTAL QUALIFICATION	SEISMIC QUALIFICATION	REDUNDANCE	POWER SUPPLY	DISPLAY LOCATION			COMMENTS
		DESCRIPTION	TYPE	CATA-GORY	EXISTING	REQUIRED						CR	TSC	EOF	
D25	TE1812 TI1812 TE1813 TI1813 TE1814 TI1814 TE1815 TI1815 TT1815	Containment Atmospheric Temperature	D	2	0°F to 400°F	40°F to 400°F	Pre-QA	Note 7	N/A	4 Channels	Preferred 1E	C13	CFM S	CFMS	Maximum Containment Temperature 283°F.
D26	-	Containment Sump Water Temperature	D	2		50°F to 250°F									Not provided. Note 8
D27		Makeup Flow-in	D	2		0-110% Design Flow									Same as Charging Flow. See Item D06.
D28	FT0202 FIC0202	Letdown Flow-out	D	2	0 to 160 GPM	0 to 100% Design Flow	Pre-QA	Comply	N/A	N/A	Reliable Non-1E	C02	CFM S	CFMS	Design flow 132 GPM. All components in mild environment.
D29	LT0205 LA0205 LIC0205	Volume Control Tank Level	D	2	0 to 100% (Equivalent to top to bottom of tank)	Top to Bottom	Pre-QA	Comply	N/A	N/A	Reliable Non-1E	C02	CFM S	CFMS	All components in mild environment.
D30	TE0914 TIA0914 TE0916 TIA0916	Component Cooling Water Temperature to ESF System	D	2	0°F to 140°F	40°F to 200°F	Pre-QA	Comply	N/A	N/A	Preferred 1E	C08	-	-	Max temp 114°F.

ITEM	TAG NO	VARIABLE			INSTRUMENT RANGE		QA REQUIRE-MENT	ENVIRONMENTAL QUALIFICATION	SEISMIC QUALIFICATION	REDUNDANCE	POWER SUPPLY	DISPLAY LOCATION			COMMENTS
		DESCRIPTION	TYPE	CATA-GORY	EXISTING	REQUIRED						CR	TSC	EOF	
D31	CCW Pump Current EAI-116 EAI-1208 EAI-1109	Component Cooling Water Flow to ESF System	D	2	0-100 amp	0-110% Design Flow	Comply	Comply	N/A	N/A	Reliable 1E	C08	-	-	Note 9
	CCW Pump Pressure PT-0918 PIA-0918				0-150 psig										
	CCW Surge Tk Level LT-0920 LIA-0920				0-100%										
	SDC Hx Out Temp TE-0912 TI-0912 TE-0913 TI-0913				0-180°F										
D32	LT1012 LIA1012 LT1014 LIA1014 LT1016 LIA1016 LT1018 LIA1018	High Level Radioactive Liquid Tank Level	D	3	0-100% (Equivalent to Top to Bottom)	Top to Bottom	N/A	N/A	N/A	N/A	Reliable Non-1E	CFMS	CFMS	CFMS	

ITEM	TAG NO	VARIABLE			INSTRUMENT RANGE		QA REQUIRE-MENT	ENVIRONMENTAL QUALIFICATION	SEISMIC QUALIFICATION	REDUNDANCE	POWER SUPPLY	DISPLAY LOCATION			COMMENTS
		DESCRIPTION	TYPE	CATA-GORY	EXISTING	REQUIRED						CR	TSC	EOF	
D33	PT1160 PIA1160 PT1161 PIA1161 PT1162 PIA1162 PT1119 PIA1119 PT1120 PIA1120 PT1121 PIA1121	Radioactive Gas Holdup Tank Pressure	D	3	0 to 120 PSIG	0 to 150% Design Pressure	N/A	N/A	N/A	N/A	Reliable Non-1E	CFMS	CFMS	CFMS	Design Pressure 120 PSIG Note 10
D34	POS1657 POS1659 POS1663 POS1679 POS1713 POS1743 PO1745 POS1711 POS1658 POS1660 POS1664 POS1680 POS1714 POS1744 PO1746 POS1712 POS1757 POS1758 POS1768 POS1769	Emergency Ventilation Damper Position	D	2	Open - Closed Status	Open - Closed Status	Comply	Comply	N/A	N/A	1E Battery	C11A	-	-	Only Emergency dampers are in control room HVAC system All components in mild environment Note 18 Note 18

ITEM	TAG NO	VARIABLE			INSTRUMENT RANGE		QA REQUIRE-MENT	ENVIRONMENTAL QUALIFICATION	SEISMIC QUALIFICATION	REDUNDANCE	POWER SUPPLY	DISPLAY LOCATION			COMMENTS
		DESCRIPTION	TYPE	CATA-GORY	EXISTING	REQUIRED						CR	TSC	EOF	
D35A		Power supplies status of standby power	D	2		Plant Specific	Pre-QA	Comply	N/A	N/A	From Bus being monitored Power to alarm from 1E Battery bus	Various Location	-	-	All components located in mild environment
		Safeguards Xfmr Status													
	EVI-0015	Volts			0-3 KV										
	EAI-0015	Amps			0-3 KAMP										
		Startup Xfmr Status													
	EVI-0003	1-2 Volts			0-3 KV										
	EAI-0001X	1-2 Amps-X			0-3 KAMP										
	EAI-0001Y	1-2 Amps-Y			0-3 KAMP										
	EAI-0001Z	1-2 Amps-Z			0-3 KAMP										
		Safeguards Bus Status													
	EVI-0008	Volts			0-3 KV										
		2400 Volt 1E Bus Status													
	EVI-0001	1-C Volts			0-3 KV										
	EVI-0002	1-D Volts			0-3 KV										
	EAI-0003	1-C Amps			0-2 KAMP										
	EAI-0004	1-D Amps			0-2 KAMP										
	EWI-0007	1-C Power			0-5 MW										
	EWI-0009	1-D Power			0-5 MW										
		Diesel Gen Status													
	EVI-1107	1-1 Volt			0-3 KV										
	EVI-1107L	1-1 Volt			0-3 KV										
	EVI-1213	1-2 Volt			0-3 KV										
	EVI-1213L	1-2 Volt			0-3 KV										
	EWI-1107	1-1 Power			0-3 MW										
	EWI-1107L	1-1 Power			0-3 MW										
	EWI-1213	1-2 Power			0-3 MW										
	EWI-1213L	1-2 Power			0-3 MW										
	SPI-1107	1-1 Frequency			55-65 Hz										
	SPI-1107L	1-1 Frequency			55-65 Hz										
	SPI-1213	1-2 Frequency			55-65 Hz										
	SPI-1213L	1-2 Frequency			55-65 Hz										

ITEM	TAG NO	VARIABLE			INSTRUMENT RANGE		QA REQUIRE-MENT	ENVIRONMENTAL QUALIFICATION	SEISMIC QUALIFICATION	REDUNDANCY	POWER SUPPLY	DISPLAY LOCATION			COMMENTS
		DESCRIPTION	TYPE	CATA-GORY	EXISTING	REQUIRED						CR	TSC	EOF	
D35A Cont		480 Volt 1E Bus Status	D	2		Plant Specific	Pre-QA	Comply	N/A	N/A	From Bus being monitored Power to alarm from 1E battery bus	Various Locations	-	-	All components located in mild environment
	27-11 27-12 27-19 27-20	Bus 11 Bus 12 Bus 19 Bus 20			Undervoltage Alarm/ No alarm										
		480 Volt 1E MCC Status	D	2	Supply Bkr. Alarm/No Alarm										
	52-1906 52-2006 52-1112 52-1214 52-1901 52-2001	MCC 1 MCC 2 MCC 21+23 MCC 22+24 MCC 25 MCC 26													
		Preferred 1E 120 Volt Status			Undervoltage Alarm/No Alarm										
	27-1 27-2 27-3 27-4	Y 10 Y 20 Y 30 Y 40													
		1E DC Bus Status			Undervoltage Alarm/No Alarm										
	74-D1 74-D2	DC Bus 1 DC Bus 2													
D35B		Power Supplies Other Sources Important to Safety	D	2		Plant Specific	Pre-QA	None	N/A	N/A	1E Battery	Various	-	-	See Note 12
		High Pressure Air													
	PS0441 PS0439				Low Pressure Alarm/No Alarm										
		AFW Backup N2													
	PS2273 PS2274				Low Pressure Alarm/No Alarm										

ITEM	TAG NO	VARIABLE			INSTRUMENT RANGE		QA REQUIRE-MENT	ENVIRONMENTAL QUALIFICATION	SEISMIC QUALIFICATION	REDUNDANCY	POWER SUPPLY	DISPLAY LOCATION			COMMENTS
		DESCRIPTION	TYPE	CATA-GORY	EXISTING	REQUIRED						CR	TSC	EOF	
E01	RE2321 RIA2321 RE2322 RIA2322	Containment Area Radiation High Range	E	1	1 Rem/hr to 10^7 Rem/hr	1 Rem/hr to 10^7 Rem/hr	Comply	Comply	Comply	2 Channels	Preferred 1E	C11A	CFMS	CFMS	
E02	RE2300 thru RE2317 RIA2300 thru RIA2317	Radiation Exposure rate (inside buildings or area where access is required to service equipment important to safety).	E	3	10^{-4} Rem/hr to 10^4 Rem/hr	10^{-1} Rem/hr to 10^4 Rem/hr	N/A	N/A	N/A	N/A	N/A	C11	-	-	Radiation Monitors inside Auxiliary/Turbine buildings. Portable instrumentation will be used to determine radiation levels a long access routes and in areas where emergency maintenance is required.
E03A	RE2326 RIA2326 RE2327 RIA2327 FE1818 FT1818 FM1818 FR2318	Noble gases and vent flow rate (common plant vent)	E	2	$10^{-7}\mu c/cc$ to $10^5\mu c/cc$	$10^{-6}\mu c/cc$ to $10^4\mu c/cc$	Comply	Comply	N/A	N/A	Non-1E	C11A	CFMS	CFMS	All releases except steam gen relief valves or atmospheric dump valves through common stack.
E03B	RE2323 RIA2323 RE2324 RIA2324	Noble gases and vent flow rate (vent from steam gen relief valves or atmospheric dump valves)	E	2	$10^{-1}\mu c/cc$ to $10^3\mu c/cc$	$10^{-1}\mu c/cc$ to $10^3\mu c/cc$	Comply	See Note 16	N/A	N/A	Preferred	C11A	CFMS	CFMS	Flow rate not provided calculated in procedure based on primary temperatures
E04	-	Particulates and Halogens	E	3	$10^{-3}\mu Ci/cc$ to $10^2\mu Ci/cc$	$10^{-3}\mu c/cc$ to $10^2\mu c/cc$	N/A	N/A	N/A	N/A	N/A	-	-	-	Capability of sampling stack effluent provided. Onsite analysis capability provided.
E05	-	Airborne Radiohalogens and Particulates	E	3	$10^{-9}\mu Ci/cc$ to $10^{-3}\mu Ci/cc$	$10^{-9}\mu Ci/cc$ to $10^{-3}\mu Ci/cc$	N/A	N/A	N/A	N/A	N/A	-	-	-	Portable sampling and analysis equipment provided

ITEM	TAG NO	VARIABLE			INSTRUMENT RANGE		QA REQUIRE-MENT	ENVIRONMENTAL QUALIFICATION	SEISMIC QUALIFICATION	REDUNDANCE	POWER SUPPLY	DISPLAY LOCATION			COMMENTS
		DESCRIPTION	TYPE	CATA-GORY	EXISTING	REQUIRED						CR	TSC	EOF	
E06	-	Plant and Environs Radiation (Portable Instrumentation)	E	3	10-3 Rem./hr to $2x10^4$ Rem/hr Beta and Gamma	10^{-3} Rem/hr to 10^4 Rem/hr Photons 10^{-3} rad/hr to 10^4 rad/hr and low energy photons	N/A	N/A	N/A	N/A	N/A	-	-	-	Portable radiation monitors are provided
E07	-	Plant and Environs Radioactivity (Portable Instrumentation)	E	3	Isotopic Analysis	Isotopic Analysis	N/A	N/A	N/A	N/A	N/A	-	-	-	Samples collected remotely transported to multi channel analyzer
E08	-	Wind Direction	E	3	0-360° $\pm$ 5°	0-360° $\pm$ 5°	N/A	N/A	N/A	N/A	N/A	CFMS	CFMS	CFMS	Measured at 2 elevations 10 Meters and 60 Meters.
E09	-	Wind Speed	E	3	0-100 MPH $\pm$ .5 MPH	0-22 MPS (50 MPH) $\pm$ .2 MPS	N/A	N/A	N/A	N/A	N/A	CFMS	CFMS	CFMS	Measured at 2 elevations 10 Meters
E10	-	Estimation of Atmospheric Instability	E	3	-5°C to +10°C $\pm$ 0.15°C	Based on vertical temperature difference from primary met system (-5°C to 10°C $\pm$ 0.15°C)	N/A	N/A	N/A	N/A	N/A	CFMS	CFMS	CFMS	Based on temperatures at 2 elevations 10 Meters and 60 Meters
E11	-	Accident Sampling Capability (Analysis Capability on Site) Primary Coolant and Sump Containment Air	E	3	Grab Sample	Grab Sample	N/A	N/A	N/A	N/A	N/A	-	-	-	Note 11

1. Safety Injection Tank (SIT) level and pressure monitoring instrumentation changed to Category 3. Reference NRC Safety Evaluation. "Safety Injection Tank Pressure and Volume Instrumentation." dated Nov. 29, 1993.

2. Power to the SI tank isolation valves is locked out during normal operations to prevent inadvertent isolation of the SI tank from the primary coolant system (PCS). For postulated events, which result in rapid PCS depressurization to a point at which the SI tanks discharge their contents, there is no need to close the SI tanks' isolation valve. For events which result in a slowly decreasing primary system pressure, the containment environment would not be expected to be as severe as that resulting from a rapid PCS depressurization. For these less sever events, it is expected that both the SI tanks' isolation valves and vent valves' position indication would function properly to indicate isolation and/or venting of the tank. Therefore, based on the above, CPCo considers that qualification of the SI tank isolation valve position indication is not required.

3. Events which heat the quench tank result from the release of steam to the tank through underwater spargers. As the volume of water to be heated is large, the temperature of the tank will remain less than the saturation temperature corresponding to tank pressure. The maximum tank temperature is thus restricted to the saturation temperature corresponding to the tank rupture disk pressure setpoint of 100 psig which is 338°F.

4. Deleted.

5. Deleted.

6. The purpose of this indication is to monitor for potentially stuck open steam generator relief valves. Palisades has main steam flow instrument loops with flow transmitters located inside containment which can be utilized to monitor for excessive steam flows which would result from stuck open steam generator relief valves. The steam generator relief valves are located outside containment, and a Main Steam Line Break (MSLB) resulting from a stuck open relief valve would not result in a harsh environment inside of containment. Based on the above, it is concluded that the main steam flow instrument loops are located in a mild environment for events (MSLB outside containment) during which it would be required to function and, thus, are not subject to the provisions of 10CFR50.49.

7. Containment atmosphere temperature indicating loops are provided for routine surveillance of containment temperature during normal operations. The temperature elements associated with these loops are located in containment and have not been environmentally qualified. Qualified containment pressure indicating loop are, however, provided which can be utilized to assess containment temperature. During accident conditions, when energy is released to the containment the saturation temperature corresponding to the containment pressure provides a close approximation of the actual containment atmosphere temperature. As there are no post-accident operator actions required based on knowledge of containment temperature, CPCo considers that containment pressure is an acceptable alternative for monitoring post accident containment temperature.

8. The Palisades design does not contain provisions for monitoring sump water temperature. Monitoring containment pressure, however, allows an assessment of sump temperature as the sump temperature will be equal to or less (ie, slightly subcooled) than the saturation temperature corresponding to the containment pressure. As there are no operator actions required, based on knowledge of containment sump water temperature, CPCo considers that containment pressure is an acceptable alternative for monitoring sump water temperature.

9. CCW parameters monitored in the control room include: CCW pump current, CCW pump discharge pressure, CCW surge tank level and shutdown cooling heat exchanger inlet and outlet temperature (CCW side). These parameters are sufficient to allow determination of events such as flow blockage or pipe rupture. Flow blockage events, resulting in decreased system flow, would be indicated by increased CCW pump discharge pressure, decreased CCW pump current and increased differential temperatures across the shutdown cooling heat exchangers. Pipe rupture events would be indicated by decreasing levels in the CCW surge tanks. Providing these parameters in the control room precludes the need for CCW flow instrumentation.

10. The radioactive waste gas system consists of a waste gas surge tank, three air compressors and six waste gas decay tanks. The design pressure of the waste gas decay tanks is 120 psig and pressure-monitoring instrumentation is provided to indicate from 0 psig to design pressure (120 psig). This range of the pressure instrumentation is considered acceptable based on the following operating and design features which prevent the design pressure of the tank from being exceeded:

    a. During normal operation the tanks are isolated when reaching 95 psig (80% of design pressure).

    b. An alarm is provided to indicate when tank pressure exceeds 100 psig (83% design pressure).

    c. Two relief valves set at tank design pressure are available to relieve pressure when the tank is being filled. One relief is located on the discharge of the air compressors and the other on the tank. The total relief valve capacity exceed the design capacity of the pumping system. The relief valves relieve back to the waste gas surge tank.

    d. The tanks are located in an area of the plant where the environment would be unaffected during any design basis event. Thus, isolated tanks could not be overpressured by extreme temperature changes.

11. Consumers Power Company's submittal dated August 12, 1983 addresses the Palisades post-accident sampling system (PASS). A detailed comparison between Regulatory Guide 1.97, Revision 2 instrument range is provided on Pages 16-17. The accident sampling variables and ranges presented in Regulatory Guide 1.97, Revision 2 are identical with those presented in Revision 3 with the exception of the gross activity range. The Palisades post-accident sampling system gross activity range is identical to that required by Revision 2.

12. Other standby energy sources include emergency high pressure air systems utilized to operate selected air operated valves in the event that the normal instrument air system is disabled. Status of the various standby pressure sources is provided by local pressure indications and control room alarms actuated on low pressure. The accumulator tanks in the standby high pressure air systems are passive devices which function automatically to provide air to required valves through appropriate check valves in the event normal instrument air is lost. The status indications provided are used for monitoring during normal operations to assure that sufficient air is available in the accumulators to perform their function if required. Some of the components of the status indication system are located in harsh environments, however, as these indications are only required to determine system availability prior to use, CPCo considers that upgrading these components is not required.

13. The range of the installed PCS pressure instrument loops deviates from the 0-4000 psig range required by Regulatory Guide. The existing range of 0-3000 psig is considered to be adequate based on the improved readability of the smaller range and the modification installed to limit anticipated transient without scram (ATWS) events. (Reference: CPCo letter to NRC, "Response to NRC Interim Report, Conformance to Regulatory Guide 1.97," dated April 30, 1986).

14. Cable separation for the containment penetration isolation valve position indication listed below deviates from the guidance on redundancy provided in Regulatory Guide 1.97 and the Palisades design basis. A lack of position indication would not result in misleading the operator and by placing the valve handswitch in the closed position, the operator would assure that power is removed from the solenoid operator thus closing the valve. (Reference: supplemental Safety Evaluation, "Conformance to Regulatory Guide 1.97," dated January 11, 1994.)

CONTROL VALVE	PENETRATION NO	FUNCTION
CV1910, 1911	40	PCS SAMPLE
CV1064, 1065	25	CWRT VENT TO STACK
CV1002, 1007	47	PCS DRAIN TANK
CV1036, 1038	49	CWRT OUTLET
CV1044, 1046	69	CWRT PUMP SUCTION
CV1101, 1102	46	CONTAINMENT VENT HEADER TO WASTE GAS
CV1103, 1104	52	SUMP DRAIN TO DIRTY WASTE TANKS
CV1501, 1502	38	STEAM HEATING RETURN

PCS - Primary Coolant System
CWRT - Clean Waste Receiver Tank

15. Cable separation requirements for the core exit thermocouple cables between the reactor head and control room are not met. Justification for this deviation is provided in CPCo letter to the NRC, "Inadequate Core Cooling Instrumentation System," dated May 31, 1984.

16. The main steam line radiation monitors are located in an area defined as a harsh environment for a main steam line break outside of containment. The installed radiation monitors are not qualified to operate in this harsh environment. Justification for this deviation is based on the following:

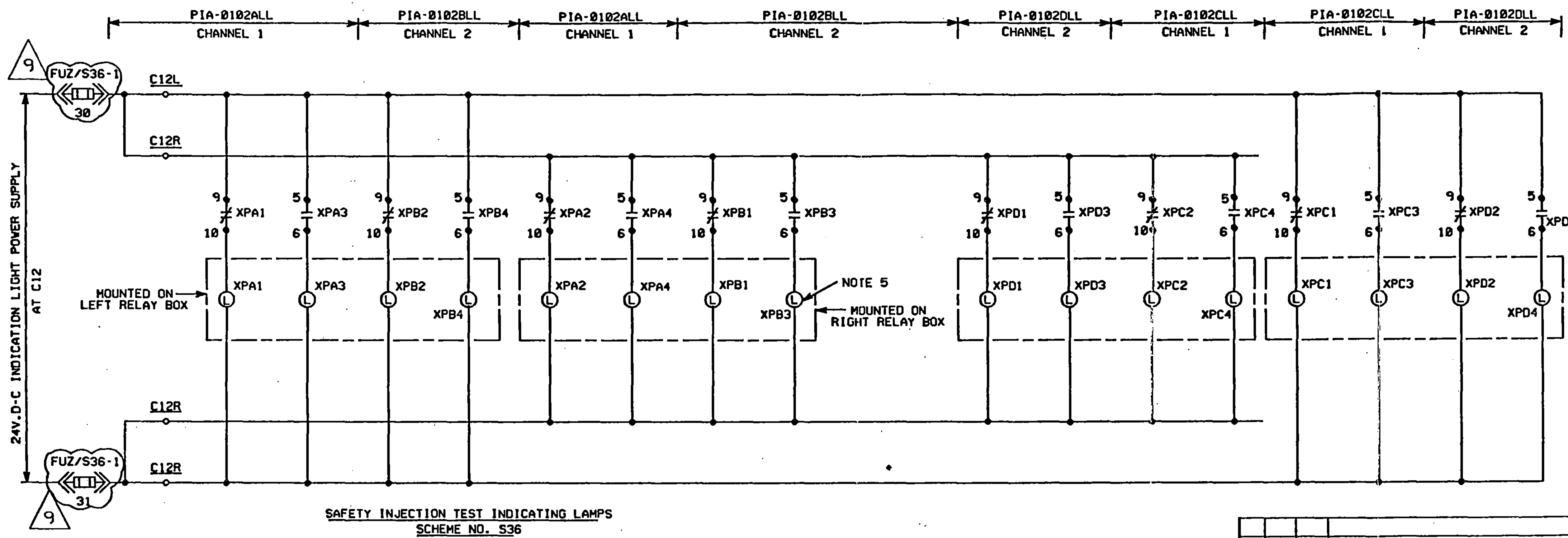
    a. The main steam radiation monitors are used to quantify radiation releases for a steam generator tube rupture. The steam generator tube rupture does not result in a harsh environment in the area containing the radiation monitors. Thus the main steam radiation monitors do not need to be environmentally qualified.

    b. The function of the main steam line radiation monitors is to calculate the potential offsite dose which could occur following a release through the steam generator safety relief or atmospheric dump valves. An alternate method of performing this calculation is provided by the backup High Range Effluent Monitors located on the auxiliary building roof. Use of this backup method is included in Plant Emergency Implementation Procedure.

    (Reference: Supplemental Safety Evaluation, "Conformance to Regulatory Guide 1.97," dates January 11, 1994)

17. Both channels of condensate storage level indication cabling are routed through the non-seismic turbine building. This condition has been reviewed and found acceptable as documented in Reference 24.

18. PO-1745 and PO-1746 provide position indication via limit switches in the motor operator. Power supply is from the motor operator's breaker cubicle via a step-down transformer.



SI APERTURE CARD

Also Available On Aperture Card

REV	DATE	DESCRIPTION	FC• SC•	BY	CK	APP
9	12-6 91	FUSE EDB NO. ADDED PER ENG. REVIEW DCR 950-91-883(MAL)		PRC	JGD	£DS

CONSUMERS POWER COMPANY
PALISADES NUCLEAR PLANT
COVERT, MICHIGAN

E206-2.009

PREVIOUS RECORD ISSU

MICHIGAN P.E. NO.

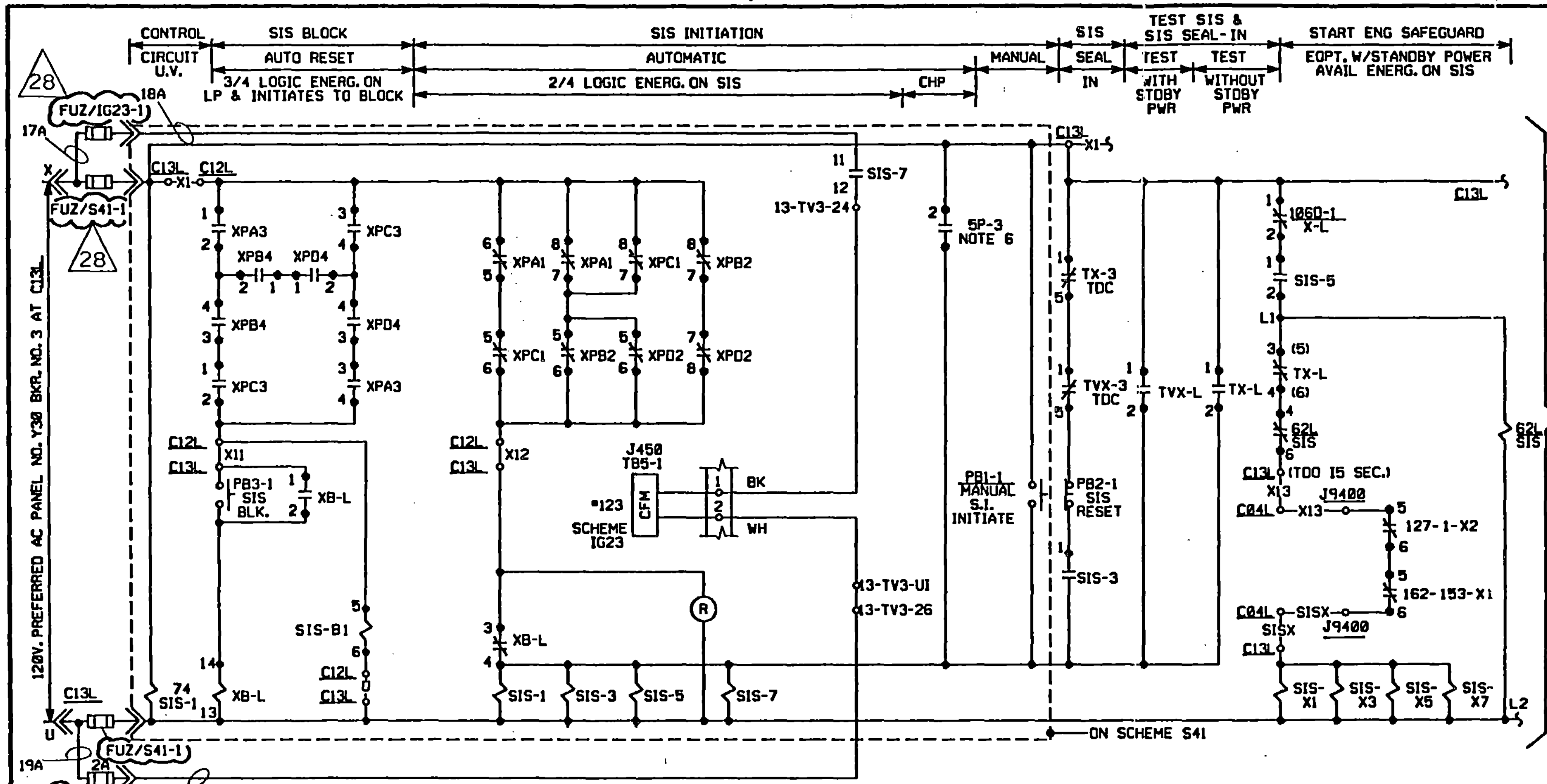
UFI NO.

SCHEMATIC DIAGRAM
SAFETY INJECTION SIGNAL
AUXILIARY CIRCUITS

DRAWING NUMBER	SHT. NO.	REV
E206	2	9

Figure No 7-16 FSAR Rev 15

9305130068 -27



SAFETY INJECTION & SEQUENCE LOADING CIRCUIT NO. 1
SCHEME NO. S41
ONE OF TWO IDENTICAL CIRCUITS

NOTES:

1. FOR SIS DETECTION AUX. RELAYS SEE DWG. E-206.
2. FOR SIS TEST SIGNAL RELAY CIRCUITS SEE DWG. E-210 SH. 1.
3. FOR WIRING BLOCK DIAGRAM SEE SH. 2 OF THIS DWG..
4. ALL WIRING #14 AWG..
5. FOR SIS BLOCK RELAYS REFER TO DWG. E-206.
6. FOR CHP RELAYS 5P-3 & 5P-8 SEE DWG. E-208.
7. FOR SCHEME S42 VARIATION SEE E-209 SH. 1A.
8. ( ) DENOTES RIGHT CHANNEL CONTACT NUMBERS.
9. SEQUENCER TEST CONNECTION WIRE IS #16 AWG.

SI APERTURE CARD

Also Available On Aperture Card

ADAPTER TABLE

DESCRIPTION	SCHEME NO.	PREF. AC. BKR.	BKR.	L.P.S.I.		AUXILIARY RELAYS SIS BLOCK	LOCATION	MAINTAINED SI INITIATION RELAYS	MOMENTARY SIS WITH STANDBY POWER RELAYS				SIS T/O RELAY	U.V. RELAY 74/	FUSE NO.		SIS BLK SWITCH
S.I. & SEQ. LOADING CKT. NO. 1	S41	Y30	3	XPA1, XPB2 XPC1, XPD2		XPA3, XPB4 XPC3, XPD4	C12L	SIS-1,3,5,7	SIS-X1	SIS-X3	SIS-X5	SIS-X7	62L SIS	SIS-1	FUZ/S41-1	FUZ/IG23-1	PB3-1
S.I. & SEQ. LOADING CKT. NO. 2	S42	Y20	3	XPA2, XPB1 XPC2, XPD1		XPA4, XPB3 XPC4, XPD3	C12R	SIS-2,4,6,8,10	SIS-X2	SIS-X4	SIS-X6	SIS-X8	62R SIS	SIS-2	FUZ/S42-1	FUZ/IG22-1	PB3-2

	REV	DATE	DESCRIPTION	FC* SC*	BY	CK	APP
	28	1-21 93	ADDED FUSE NO. S AND REV. ADAPTER TABLE TO SHOW FUSE NO. S PER DCR 92-1506 (JS)		JFM	JGD	DMB

CONSUMERS POWER COMPANY
PALISADES NUCLEAR PLANT
COVERT, MICHIGAN
E209-1.DGN

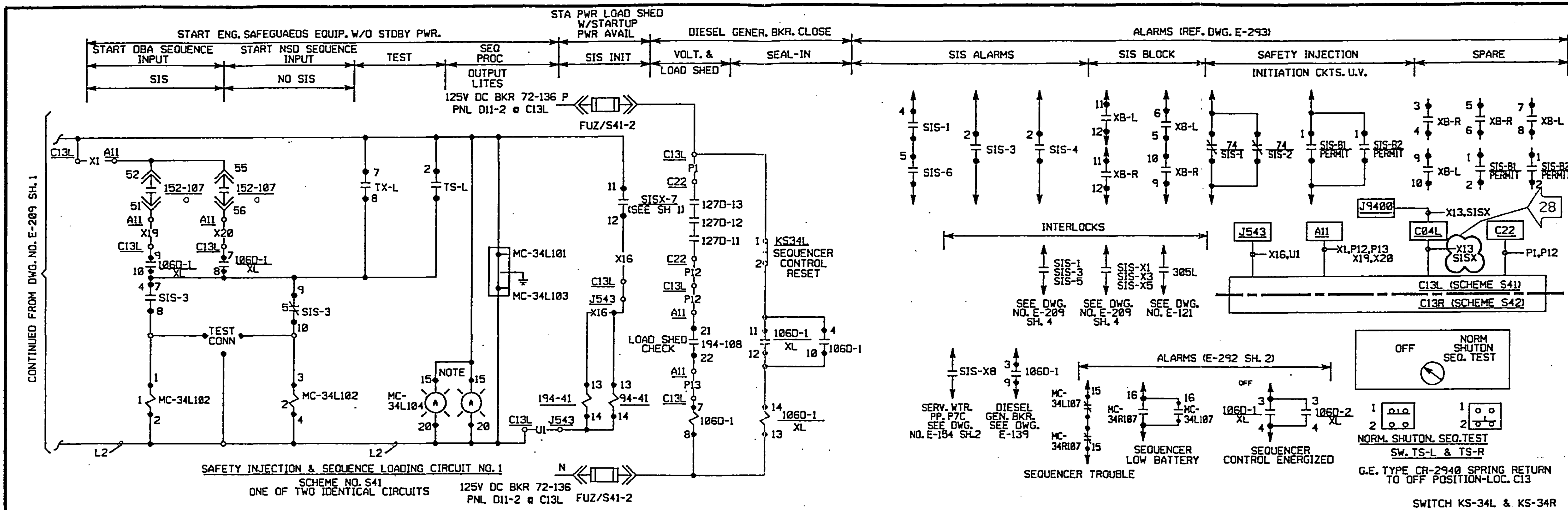
PREVIOUS RECORD ISSUE
MICHIGAN P.E. No.
UFI No.

SCHEMATIC DIAGRAM
SAFETY INJECTION & SEQUENCE
LOADING CIRCUITS

DRAWING NUMBER	SHT. NO.	REV
E209	1	28

Figure No 7-17 FSAR Rev 15

9305130068 -28



ADAPTER TABLE

DESCRIPTION	SCHEME NO.	LOCATION					PREF. A.C.	SEQ.	DIESEL ENGINE U.V.	DIESEL GEN.BKR. CLOSE RELAY	S.I. PB MAN. INIT.	S.I. PB RESET	S.I. PB BLOCK	BUS UNDERVOLTAGE RELAY	BUS UNDERVOLTAGE LOCATION	LOAD SHED RELAY	SEQUENCER CONTROL RELAY	SEQUENCER CONTROL RESET KEYSWITCH	NORM. SHUT-DN SEQ TEST PUSHBUTTON	CHP RELAY CONTACTS	DC PNL	DC BKR
S.I. & SEQ. LOADING CKT. NO. 1	S41	A11	C06L	C04L	C13L	C22	Y30	34-1,3,5	127D-1	106D-1	PB1-1	PB2-1	PB3-1	127-1-X2,162-153-X1	J9400	194-108	106D-1/XL	KS34L	TS-L	5P-3/2, 5P-0/1	D11-2	72-136
S.I. & SEQ. LOADING CKT. NO. 2	S42	A12	C06R	C04R	C13R	C26	Y20	34-2,4,6	127D-2	106D-2	PB1-2	PB2-2	PB3-2	127-2-X2,162-154-X1	J9401	194-211	106D-2/XL	KS34R	TS-R	5P-4/2, 5P-9/1	D21-2	72-236

* (DIESEL GEN.BKR. CLOSE RELAY) * (SEQUENCER CONTROL RELAY)

* SEE E-209, SH. 2A FOR RELAY TABLES.

SWITCH KS-34L & KS-34R

SEQUENCER CONTROL RESET
OFF RESET

1 2 1 2

MIRCO SW TYPE PTF
SPRING RETURN TO OFF POSITION
LOC. C13

ADAPTER TABLE

STATION POWER LOAD SHED W/STARTUP POWER AVAILABLE AND SIS INTIATION		D.G. LOADING SEQUENCER DBA/NSD PROCESSOR	D.G. LOADING SEQUENCER DBA INPUT	D.G. LOADING SEQUENCER NSD INPUT	FUSE NO.
194-41, 94-41, SISX-7	J543	MC-34L101	MC-34L102/1	MC-34L102/2	FUZ/S41-2
194-42, 94-42, SISX-8	J542	MC-34R101	MC-34R102/1	MC-34R102/2	FUZ/S42-2

NOTE: POWER PROVIDED TO DRIVE OUTPUT CARD INDICATING LITES. OUTPUTS WITH NO LOAD CONNECTED WILL HAVE LITE ON. OUTPUT WITH A LOAD CONNECTED WILL HAVE LITE ON WHEN OUTPUT IS CLOSED AND LITE OFF WHEN OUTPUT IS OPEN.

REV	DATE	DESCRIPTION	BY	CK	APP
28	4-13 95	REMOVED WIRES X3 & X19 (C04-C13) PER "APP. R" & DCR-95-234	JGD	MAB	—

SECTION HEAD APP. C.J. MCDONALD DATE 4/22/83	DR. J. L. STERRETT	CK. T.D. VOGT 4/22/83	
DIVISION HEAD APP. P. LOLICH 4/22/83	CONSUMERS POWER CO. PALISADES PLANT	SCHEMATIC DIAGRAM SAFETY INJECTION & SEQUENCE LOADING CIRCUITS	
DEPARTMENT HEAD APP. D.C. TARSI 4/22/83	E209-2.DGN	NO. E-209	SHEET 2

Figure 7-18
Rev 19

9707070091 -09

MC-34L102 (B817-116) SEQUENCER FUNCTIONS-LEFT CHANNEL

INPUT CONTACT NUMBER	INTER INPUT CONT NO	DESCRIPTION	DRAWING	TIME DBA	TIME NSD
1	10001	INPUT FOR DBA (10001)	E-209 SH.2	0(REF)	NA
2	10002	INPUT FOR NSD (10002)	E-209 SH.2	NA	0(REF)
3	10003	SPARE			
4	10004	SPARE			
5	10005	SPARE			
6	10006	SPARE			
7	10007	SPARE			
8	10008	SPARE			
9	10009	SPARE			
10	10010	SPARE			
11	10011	SPARE			
12	10012	SPARE			
13	10013	SPARE			
14	10014	SPARE			
15	10015	SPARE			
16	10016	SPARE			

MC-34R102 (B817-116) SEQUENCER FUNCTIONS-RIGHT CHANNEL

INPUT CONTACT NUMBER	INTER INPUT CONT NO	DESCRIPTION	DRAWING	TIME DBA	TIME NSD
1	10001	INPUT FOR DBA	E-209 SH.2	0(REF)	NA
2	10002	INPUT FOR NSD	E-209 SH.2	NA	0(REF)
3	10003	SPARE			
4	10004	SPARE			
5	10005	SPARE			
6	10006	SPARE			
7	10007	SPARE			
8	10008	SPARE			
9	10009	SPARE			
10	10010	SPARE			
11	10011	SPARE			
12	10012	SPARE			
13	10013	SPARE			
14	10014	SPARE			
15	10015	SPARE			
16	10016	SPARE			

MC-34L103 (B810-008) SEQUENCER FUNCTIONS-LEFT CHANNEL

OUTPUT CONTACT NUMBER	INTER OUTPUT CONT NO	DESCRIPTION	DRAWING	TIME DBA	TIME NSD
1	00001	OPEN HPSI VOP-3009	E-244 SH.1	0+0.3-0	NA
2	00002	OPEN HPSI VOP-3011	E-244 SH.4	0+0.3-0	NA
3	00003	OPEN LPSI VOP-3010	E-244 SH.1	0+0.3-0	NA
4	00004	CLOSE VOL CONT TANK VOP-2087	E-242 SH.1	0+0.3-0	NA
5	00005	START BORIC ACID PUMP P-56B	E-203	2±0.3	NA
6	00006	SPARE			
7	00007	SPARE			
8	00008	SPARE			

MC-34R103 (B810-008) SEQUENCER FUNCTIONS-RIGHT CHANNEL

OUTPUT CONTACT NUMBER	INTER OUTPUT CONT NO	DESCRIPTION	DRAWING	TIME DBA	TIME NSD
1	00001	OPEN HPSI VOP-3064	E-244 SH.1	0±0.3-0	NA
2	00002	OPEN HPSI VOP-3066	E-244 SH.4	0±0.3-0	NA
3	00003	OPEN LPSI VOP-3012	E-244 SH.1	0±0.3-0	NA
4	00004	OPEN BORIC ACID FEED PP VOP-2140	E-241	0±0.3-0	NA
5	00005	START BORIC ACID PUMP P-56A	E-203	2±0.3	NA
6	00006	SPARE			
7	00007	SPARE			
8	00008	SPARE			

MC-34L104 (B810-008) SEQUENCER FUNCTIONS-LEFT CHANNEL

OUTPUT CONTACT NUMBER	INTER OUTPUT CONT NO	DESCRIPTION	DRAWING	TIME DBA	TIME NSD
1	00009	OPEN HPSI VOP-3007	E-244 SH.1	0+0.3-0	NA
2	00010	OPEN HPSI VOP-3013	E-244 SH.4	0+0.3-0	NA
3	00011	OPEN LPSI VOP-3008	E-244 SH.1	0+0.3-0	NA
4	00012	OPEN BORIC ACID TANK VOP-2169	E-241	0+0.3-0	NA
5	00013	OPEN BORIC ACID TANK VOP-2170	E-241	0+0.3-0	NA
6	00014	SPARE			
7	00015	SPARE			
8	00016	SPARE			

MC-34R104 (B810-008) SEQUENCER FUNCTIONS-RIGHT CHANNEL

OUTPUT CONTACT NUMBER	INTER OUTPUT CONT NO	DESCRIPTION	DRAWING	TIME DBA	TIME NSD
1	00009	OPEN HPSI VOP-3062	E-244 SH.1	0±0.3-0	NA
2	00010	OPEN HPSI VOP-3068	E-244 SH.4	0±0.3-0	NA
3	00011	OPEN LPSI VOP-3014	E-244 SH.1	0+0.3-0	NA
4	00012	SPARE			
5	00013	SPARE			
6	00014	SPARE			
7	00015	SPARE			
8	00016	SPARE			

MC-34L105 (B836-016) SEQUENCER FUNCTIONS-LEFT CHANNEL

OUTPUT CONTACT NUMBER	INTER OUTPUT CONT NO	DESCRIPTION	DRAWING	TIME DBA	TIME NSD
1	00017	START SERVICE WATER PUMP P-7B	E-154 SH.1	10±0.3	10±0.3
2	00018	*START CCW PUMP P-52C	E-259	40±0.3	40±0.3
3	00019	START CHARGING PUMP P-55C	E-257 SH.2	2±0.3	2±0.3
4	00020	START LPSI PUMP P-67B	E-248	13±0.3	NA
5	00021	SPARE			
6	00022	SPARE			
7	00023	SPARE			
8	00024	SPARE			
9	00025	SPARE			
10	00026	SPARE			
11	00027	SPARE			
12	00028	SPARE			
13	00029	SPARE			
14	00030	SPARE			
15	00031	SPARE			
16	00032	SPARE			

MC-34R105 (B836-016) SEQUENCER FUNCTIONS-RIGHT CHANNEL

OUTPUT CONTACT NUMBER	INTER OUTPUT CONT NO	DESCRIPTION	DRAWING	TIME DBA	TIME NSD
1	00017	START SERVICE WATER PUMP P-7A	E-154 SH.1	10±0.3	10±0.3
2	00018	START CHARGING PUMP P-55B**	E-257 SH.1	19±0.3	19±0.3
3	00019	START CHARGING PUMP P-55A	E-257 SH.1	2±0.3	2±0.3
4	00020	LPSI PUMP P-67A	E-247	13±0.3	NA
5	00021	SPARE			
6	00022	SPARE			
7	00023	SPARE			
8	00024	SPARE			
9	00025	SPARE			
10	00026	SPARE			
11	00027	SPARE			
12	00028	SPARE			
13	00029	SPARE			
14	00030	SPARE			
15	00031	SPARE			
16	00032	SPARE			

MC-34L106 (B836-016) SEQUENCER FUNCTIONS-LEFT CHANNEL

OUTPUT CONTACT NUMBER	INTER OUTPUT CONT NO	DESCRIPTION	DRAWING	TIME DBA	TIME NSD
1	00033	START AUX FDWTR PUMP P-8A	E-196 SH.2	45±0.3	45±0.3
2	00034	START CCW PUMP P-52A	E-259	23+0.3	23+0.3
3	00035	START CONT SPRAY PUMP P-54C	E-251	19±0.3	NA
4	00036	START CONT SPRAY PUMP P-54B	E-251	2±0.3	NA
5	00037	SPARE			
6	00038	SPARE			
7	00039	SPARE			
8	00040	SPARE			
9	00041	SPARE			
10	00042	SPARE			
11	00043	SPARE			
12	00044	SPARE			
13	00045	SPARE			
14	00046	SPARE			
15	00047	SPARE			
16	00048	SPARE			

MC-34R106 (B836-016) SEQUENCER FUNCTIONS-RIGHT CHANNEL

OUTPUT CONTACT NUMBER	INTER OUTPUT CONT NO	DESCRIPTION	DRAWING	TIME DBA	TIME NSD
1	00033	START AUX FDWTR PUMP P-8C	E-196 SH.9	45±0.3	45±0.3
2	00034	START CC PUMP P-52B	E-259	23±0.3	23±0.3
3	00035	START SERVICE WTR PUMP P-7C	E-154 SH.2	26±0.3	26+0.3
4	00036	START CONT SPRAY PUMP P-54A	E-251	2±0.3	NA
5	00037	SPARE			
6	00038	SPARE			
7	00039	SPARE			
8	00040	SPARE			
9	00041	SPARE			
10	00042	SPARE			
11	00043	SPARE			
12	00044	SPARE			
13	00045	SPARE			
14	00046	SPARE			
15	00047	SPARE			
16	00048	SPARE			

NOTES

1. THE FOLLOWING CONVENTION WAS USED FOR EQUIPMENT NUMBERS.

```
        MC- 3 4 X X XX
CHANNEL         | |  |
(LEFT OR RIGHT)-' |  |
CHASSIS *         |  |
(ALWAYS 1)- - - --'  |
SLOT *               |
(01-07)- - - - - - --'
```

2. SEQUENCERS ARE GOULD MODEL 984-380 WITH 800 SERIES INPUT/OUTPUT MODULES.
3. FOR SEQUENCER INITIATION SCHEME SEE E-209 SH.2.

*IF PUMPS P-52A AND P-52B FAIL TO START.
** IF PUMPS P-55A AND P-55C FAIL TO START.

REV	DATE	DESCRIPTION	BY	CK	APP
25	11-1-96	REV MC-34L&R103, 104, 105 & 106 PER 950-96-810.	JGD	DMB	

APP. C.J. MCDONALD (SECTION HEAD) 4/18/83	DR. R.M. SATTERELLI	CK. T.D. VOGT 4/18/83
APP. P. LOLICH (DIVISION HEAD) 4/18/83	CONSUMERS POWER CO. PALISADES PLANT	SCHEMATIC DIAGRAM SAFETY INJECTION & SEQUENCER LOADING CIRCUITS
APP. D.C. TARSI (DEPARTMENT HEAD) 4/18/83	E209-3.DGN	NO. E-209 SHEET 3 REV. 25

Figure 7-19
Rev 19

9707070091-10

SIS (SAFETY INJECTION SIGNAL) RELAY TABLE

FUNCTIONS OF SIS RELAYS

CONTACTS		SIS RELAYS-LEFT (ODD NUMBERS)						SIS RELAYS-RIGHT (EVEN NUMBERS)							
		SIS-1		SIS-3		SIS-5		SIS-2	REF. DWG.	SIS-4	REF. DWG.	SIS-6	REF. DWG.	SIS-10 (SEE NOTE 4)	REF. DWG.
1	-| |-	CLOSE CONT'MT SUMP DRAIN ISOL. SV1103 **	E-235 SH.1	SEAL-IN CONTACT	E-209 SH.1	SIS WITH STANDBY PWR. AVAILABLE CKT. NO. 1	E-209 SH.1	CLOSE CONT'MT SUMP DRAIN ISOL. SV1104 **	E-235 SH.1	SEAL-IN CONTACT	E-209 SH.1	SIS WITH STANDBY PWR. AVAILABLE CKT. NO. 2	E-209 SH.1	OPEN SERV. WTR. DISCH. SV0861 **	E-216
2	-|/|-	CLOSE SERVICE WATER TO NON-CRITICAL ITEMS SV-1359	E-219 SH.3	S.I. INITIATION ALARM *	E-293 SH.4	CLOSE BORIC ACID MAKE-UP STOP VALVE SV2155	E-234 SH.1	SPARE		S.I. INITIATION ALARM *	E-293 SH.4	OPEN COMP. CLG. WTR. TO SEAL CLG. SUCTION VA. SV0913	E-239 SH.2	SPARE *	
3	-|/|-	TRIP BLOCK CHARGING PUMP P55C	E-257 SH.2	CLOSE BORIC ACID RECIRC. VALVE POC2130	E-90	CLOSE S.I. TANK RELIEF VALVE SV0347	E-245 SH.3	CLOSE SERVICE WTR. TO NON-CRITICAL ITEMS SV1359	E-219 SH.3	CLOSE BORIC ACID RECIRC. VALVE POC2136	E-90 SH.3	CLOSE COMP. CLG. WTR. TO FUEL POOL HEAT EXCH. SV0944A	E-239 SH.2	SPARE *	
4	-| |-	S.I. INITIATION ALARM	E-293 SH.4	START DBA SEQUENCER	E-209	CLOSE S.I. TANK RELIEF VALVE SV0338 **	E-245 SH.3	CLOSE S.I. TANK LEAKAGE VALVE SV3069 **	E-245 SH.3	START DBA SEQUENCER	E-209 SH.2	ENABLE OPENING COMP. CLG. WTR. TO FUEL POOL HEAT EXCH.	E-239 SH.2	CLOSE COMP. CLG. WTR. TO RADWASTE EVAPS SV0944 **	E-239
5	-|/|-	CLOSE BLOCK CHARGING PUMP P55C	E-257 SH.2	NO. SIS TO START NORMAL SHUT DN. SEQUENCER	E-209	OPEN COMPONENT CLG. WTR. TO SHUT DN. HEAT EXCH. SV0938	E-239 SH.1	OPEN COMPONENT CLG. WTR. DISCH. FROM SEAL CLG. SV0950	E-239 SH.2	NO. SIS TO START NORMAL SHUT DN. SEQ	E-209 SH.2	S.I. INITIATION ALARM *	E-293 SH.4	CLOSE COMP. CLG. WTR. TO RADWASTE EVAPS SV0977B	E-239
6	-|/|-	CLOSE S.I. TANK RELIEF VALVE SV0342	E-245 SH.3	CLOSE S.I. TANK RELIEF VALVE SV0346	E-245	CLOSE HPSI HOT-LEG PRESSURE LETDOWN VALVE CV3085	E-245 SH.4	CLOSE S.I. TANK RELIEF VALVE SV0347	E-245 SH.3	CLOSE S.I. TANK RELIEF VALVE SV0338	E-245 SH.3	OPEN SERV. WTR. DISCH. SV0873	E-216 SH.1	SPARE *	
7	-|/|-													CFM INPUT #124	E-209
8	-|/|-													CLOSE HPSI HOT-LEG PRESSURE LETDOWN VALVE CV3084	E-245
9	-|/|-													SPARE	
10	-|/|-													SPARE	
11	-|/|-													CLOSE S.I. TANK RELIEF VALVE SV0342	E-245 SH.3
12	-|/|-													CLOSE S.I. TANK RELIEF VALVE SV0346	E-245 SH.3

SIS-X (SAFETY INJECTION SIGNAL AUXILIARY) RELAY TABLE

FUNCTIONS WITH SIS & STANDBY POWER AVAILABLE

CONTACTS		SIS-X RELAYS-LEFT (ODD NUMBERS)						SIS-X RELAYS-RIGHT (EVEN NUMBERS)					
		SIS-X1	REF. DWG.	SIS-X3	REF. DWG.	SIS-X5	REF. DWG.	SIS-X2	REF. DWG.	SIS-X4	REF. DWG.	SIS-X6	REF. DWG.
1	-| |-	START L.P. INJ. PUMP P67B	E-248	OPEN H.P. INJ. LINE MOV 3011	E-244 SH.4	CLOSE VOLUME CONTROL TANK OUTLET MOV2087	E-242 SH.1	START L.P. INJ. PUMP P67A	E-247	OPEN H.P. INJ. LINE MOV 3062	E-244 SH.1	OPEN BORIC ACID PUMPED FEED MOV2140	E-241
2	-| |-	START L.P. INJ. PUMP P66B	E-249	OPEN L.P. INJ. LINE MOV 3008	E-244 SH.1	START BORIC ACID PUMP P56B	E-203	START H.P. INJ. PUMP P66A	E-249	OPEN H.P. INJ. LINE MOV 3066	E-244 SH.1	START BORIC ACID PUMP P56A	E-203
3	-| |-	CONT. SPRAY PUMP P54B ON STAND BY	E-251	OPEN H.P. INJ. LINE MOV 3010	E-244 SH.1	OPEN BORIC ACID GRAVITY FEED MOV2169	E-241	SPARE		OPEN H.P. INJ. LINE MOV 3068	E-244 SH.4	START CHG. PUMP P55A	E-257 SH.1
4	-| |-	OPEN H.P. INJ. LINE MOV 3007	E-244 SH.1	TRIP CONT. CLG. UNIT V4B **	E-218	OPEN BORIC ACID GRAVITY FEED MOV2170	E-241	CONT. SPRAY PUMP P54A ON STAND BY	E-251 SH.1	OPEN L.P. INJ. LINE MOV 3012	E-244 SH.1	START CHG. PUMP P55B	E-257 SH.1
5	-| |-	OPEN H.P. INJ. LINE MOV 3009	E-244 SH.1	SPARE		SPARE		TRIP CONT. CLG. UNIT V1B **	E-218	TRIP CONT. CLG. UNIT V2B **	E-218	START SERV. WTR PUMP P7A	E-154 SH.1
6	-| |-	OPEN H.P. INJ. LINE MOV 3013	E-244 SH.4	CONT. SPRAY PUMP P54C ON STAND BY	E-251	START CHG. PUMP P55C	E-257 SH.2	OPEN H.P. INJ. LINE MOV 3064	E-244 SH.1	OPEN H.P. INJ. LINE MOV 3014	E-244	TRIP CONT. CLG. UNIT V3B **	E-218

CONTACTS		SIS-X7	REF. DWG.	CONTACTS		SIS-8	REF. DWG.	CONTACTS		SIS-7	REF. DWG.	CONTACTS		SIS-X8	REF. DWG.
1	-| |-	START SERV. WTR. PUMP P7B	E-154 SH.1	1	-|/|-	OPEN SERV. WTR. DISCH. VALVE SV0867	E-217	1	-|/|-	OPEN COMP. CLG. WTR. SEAL CLG. SUCTION VALVES SV0913	E-239 SH.2	1	-| |-	START SERV. WTR. PUMP P7C	E-154 SH.2
2	-| |-	START COMP. CLG. PUMP P52A	E-259	2	-|/|-	CLOSE SERV. WTR. SUPPLY VALVE SV0869	E-217	2	-|/|-	OPEN COMP. CLG. WTR. DISCH. FROM SEAL CLG. SV0950	E-239 SH.2	2	-| |-	START COMP. CLG. PUMP P52B	E-259 SH.1
3	-| |-	START COMP. CLG. PUMP P52C	E-259	3	-|/|-	TRIP BLOCK CHARGING PUMP P55B	E-257 SH.1	3	-|/|-	OPEN SERV. WTR. DISCH. VALVE SV0867	E-217	3	-| |-	SPARE	
4	-| |-	SPARE		4	-|/|-	OPEN COMP. CLG. WTR. VA. TO SHUTDN. HEAT EXCH. SV0937 28	E-239 SH.1	4	-|/|-	CLOSE S.I. TANK LEAKAGE VALVE SV3069 28	E-245 SH.3	4	-| |-	SPARE	
5	-| |-	SPARE		5	-|/|-	OPEN SERV. WTR. DISCH. SV0864	E-216 SH.1	5	-|/|-	CLOSE SERV. WTR. SUPPLY VALVE SV0869	E-217	5	-| |-	SPARE	
6	-| |-	INITIATE STARTUP POWER LOAD SHED SYSTEM	E-209	6	-|/|-	CLOSE BLOCK CHARGING PUMP P55B	E-257 SH.1	6	-| |-	CFM INPUT #123	E-209	6	-| |-	INITIATE STARTUP POWER LOAD SHED SYSTEM	E-209 SH.2

GWO 8428
SUBMERGED ELECT. EQUIP. MOD.

NOTES
1. * N.O. CONTACT
2. ** N.C. CONTACT-OPENS ON SIS
3. FOR RELAY CIRCUITS, SEE E-209, SH. 1
4. SIS-10 RELAY HAS 12 CONTACTS

REV	DATE	DESCRIPTION	BY	CK	ENG	REV	DATE	DESCRIPTION	BY	CK	APP
						26	11-21 90	UPDATED PER SECOND GENERATION DWG. PER 950 90 770 (KHA)	RLS	RIS	JGD
28	12-17 96	REV'D. SIS-7 & 8 PER FC-969 & DCR-96-1004.	JGD	DHB		25	9-16 90	CHANGED E/P 2130 TO POC 2130 & E/P 2136 TO POC 2136 PER SC-87-001 & DCR 950-90-591	JSS	RIS	JGD

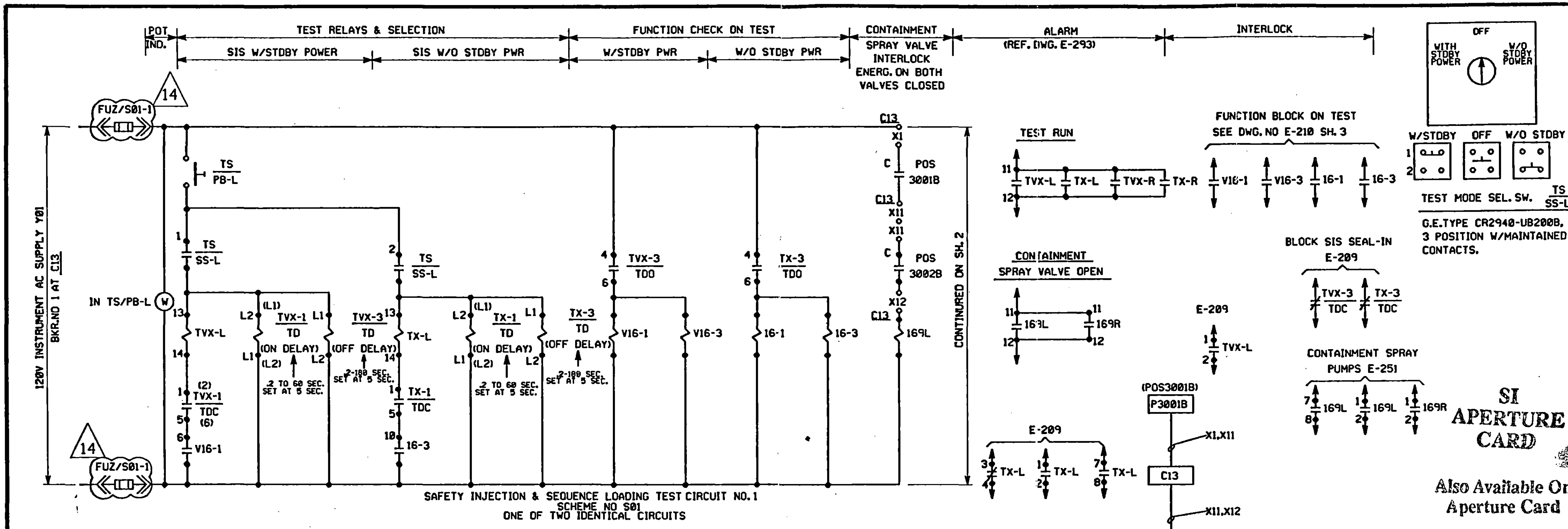
SECTION HEAD DATE
APP. C.J. MCDONALD 4/18/83
DIVISION HEAD
APP. P. LOLICH 4/18/83
DEPARTMENT HEAD
APP. D.C. TARSI 4/18/83

DR. J. L. STERRETT
CONSUMERS POWER CO.
PALISADES PLANT
E209-4.DGN

CK. T.D. VOGT 4/18/83
SCHEMATIC DIAGRAM
SAFETY INJECTION & SEQUENCE LOADING CIRCUITS
NO. E-209 SHEET 4

REV. 28
Figure 7-20
Rev 19

9707070091 -11



SAFETY INJECTION & SEQUENCE LOADING TEST CIRCUIT NO. 1
SCHEME NO S01
ONE OF TWO IDENTICAL CIRCUITS

ADAPTER TABLE

CIRCUIT NO.	SCHEME NO.	INST. A.C. PANEL	INST. A.C. BKR.	LOC.	SIS TEST RELAYS W/STDBY POWER			SIS TEST RELAYS W/O STDBY PWR			TEST&BLOCK RELAYS		SPRAY VALVE INTLK RELAYS	TEST MODE P.B.	TEST MODE S.S	FUSE NO.
1	S01	Y01	1	C13	TVX-L	TVX-1 TD	TVX-3 TD	TX-L	TX-1 TD	TX-3 TD	V16-1 V16-3	16-1 16-3	169-L	TS PB-L	TS SS-L	FUZ/S01-1
2	S02	Y01	2	C13	TVX-R	TVX-2 TD	TVX-4 TD	TX-R	TX-2 TD	TX-4 TD	V16-2 V16-4	16-2 16-4	169-R	TS PB-R	TS SS-R	FUZ/S02-1

REV	DATE	DESCRIPTION	FC* SC*	BY	CK	APP
14	12-5 91	FUSE EDB NO. ADDED PER ENG. REVIEW DCR 950-91-883(MAL)		PRC	JGD	CDS

CONSUMERS POWER COMPANY
PALISADES NUCLEAR PLANT
COVERT, MICHIGAN
E210-1.014

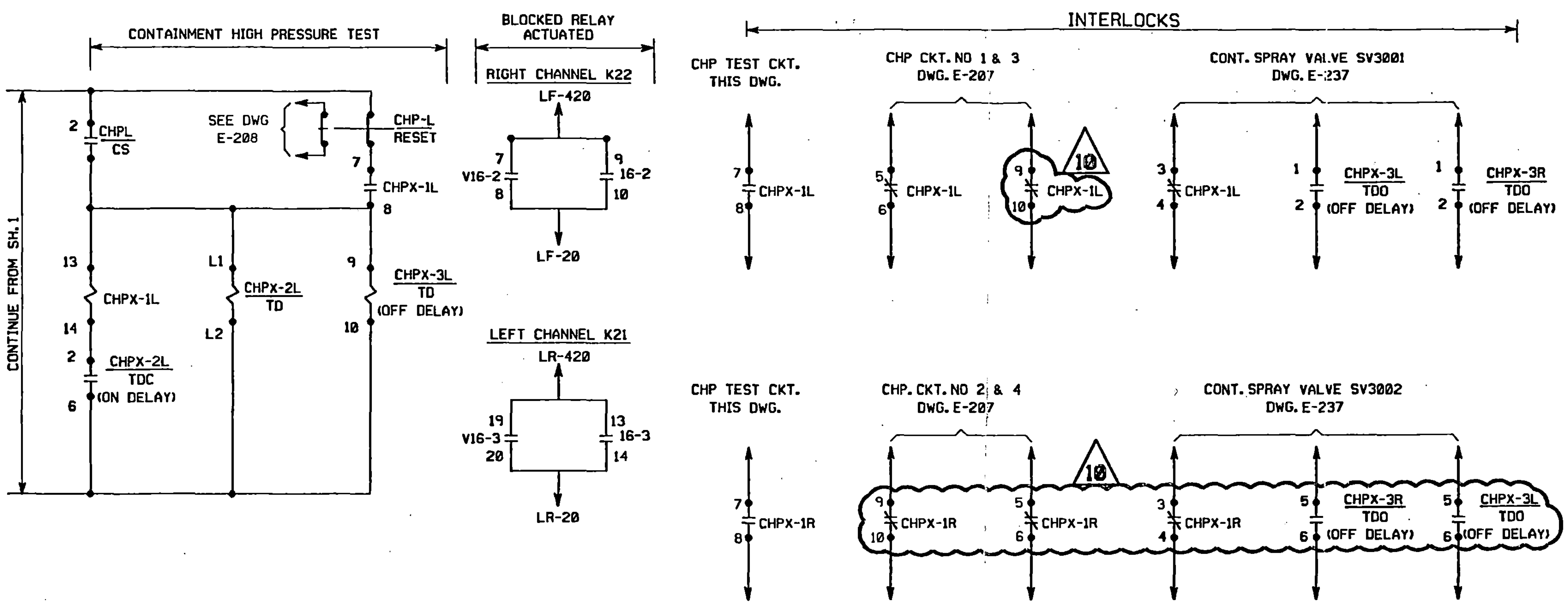
PREVIOUS RECORD ISSUE
MICHIGAN P.E. No.
UFI No.

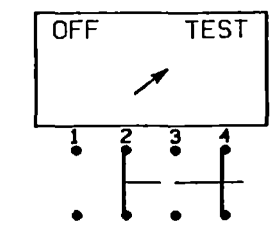
SCHEMATIC DIAGRAM
SAFETY INJECTION & SEQUENCE
LOADING TEST CIRCUIT

DRAWING NUMBER: E210 | SHT. NO. 1 | REV 14

Figure No 7-21 FSAR Rev 15

9305130068 - 31





CHPL/CS
GE. TYPE CR-2940
SPRING RETURN TO OFF POSITION

SI APERTURE CARD

Also Available On Aperture Card

NOTES

1. CHPX-2L / TD IS ON DELAY RELAY.
2. CHPX-3L / TD IS OFF DELAY RELAY.
3. ALL TIME DELAY RELAYS SET AT 5 SEC.

ADAPTER TABLE

CIRCUIT	SCHEME	INST. A.C. PANEL	INST. A.C. BKR	LOC	CHP TEST SW	CHP AUXILIARY TEST RELAYS
1	S01	Y01	1	C13	CHPL/CS	CHPX-1L,CHPX-2L,CHPX-3L
2	S02	Y01	2	C13	CHPR/CS	CHPX-1R,CHPX-2R,CHPX-3R

REV	DATE	DESCRIPTION	FC* SC*	BY	CK	APP
10	12-20-91	CHANGED INTERLOCK TERM. POINTS PER AS-BUILT GWO 7791 DCR 950-91-0090 CCP(GM)		PRC	JGD	DAB

CONSUMERS POWER COMPANY
PALISADES NUCLEAR PLANT
COVERT, MICHIGAN

E210-2.009

PREVIOUS RECORD ISSUE

MICHIGAN P.E. No.

UFI No.

SCHEMATIC DIAGRAM
SAFETY INJECTION & SEQUENCE
LOADING TEST CIRCUIT

DRAWING NUMBER	SHT. NO.	REV
E210	2	10

Figure No 7-22 FSAR Rev 15

9305130068 -32

DWG. NO.

TEST RELAY TABLES

FUNCTION ON TEST WITH STAND-BY POWER

CONTACT NO.	CONTACT	LEFT (ODD NUMBERS) RELAY V16-1	REF. DWG.	LEFT (ODD NUMBERS) RELAY V16-3	REF. DWG.	RIGHT (EVEN NUMBERS) RELAY V16-2	REF. DWG.	RIGHT (EVEN NUMBERS) RELAY V16-4	REF. DWG.
1	●┤├●	LOW PRESSURE SAFETY INJECTION PUMP P67B	E-248	●┤/├● BLKS BKR 52-7801 ⊛	E-135 SH.1	LOW PRESSURE SAFETY INJECTION PUMP P67A	E-247	CHARGING PUMP P55A	E-257 SH 1
2	●┤/├●			●┤/├● BLKS BKR 152-301 ⊛	E-132 SH.1			CHARGING PUMP P55B	E-257 SH 1
3	●┤/├●	BORIC ACID GRAVITY FEED VALVE ⊛ MOV 2169 18	E-241	●┤/├● BLKS BKR 152-102 ⊛	E-151	BORIC ACID PUMPED FEED VALVE MOV 2140 ⊛	E-241	BLOCK 30-1359	E-219 SH 3
4	●┤/├●	BORIC ACID GRAVITY FEED VALVE ⊛ MOV 2170 ⊛	E-241	●┤/├● BLKS BKR 152-302 ⊛	E-131 SH.1	BLOCKING RELAYS ACTUATED SEE NOTE1	E-210 SH. 2	MAINTAIN OPEN CV-1359 ⊛ SEE NOTE 2	E-219 SH 3
5	●┤/├●	MAINTAIN OPEN VOLUME CONTROL TANK MOV 2087 ⊛	E-242	●┤/├● BLKS BKR 52-7701 ⊛	E-135 SH.1			●┤/├● BLKS BKR 152-302 ⊛	E-131 SH.1
6	●┤├●	TEST START CIRCUIT NO 1	E-210	BORIC ACID PUMP P56B	E-203	TEST START CIRCUIT NO 2	E-210	●┤/├● BLKS BKR 152-303 ⊛	E-132 SH.1
7	●┤├●	SERVICE WATER PUMP P7B	E-154 SH 1	HIGH PRESSURE INJECTION PUMP P66B	E-249	SERVICE WATER PUMP P7A	E-154 SH 1	HIGH PRESSURE INJECTION PUMP P66A	E-249
8	●┤├●	CHARGING PUMP P55C	E-257 SH 2	MAINTAIN OPEN CV-1359	E-219 SH.3	SERVICE WATER PUMP P7C	E-154 SH 2	●┤/├● BLKS BKR 52-7701 ⊛	E-135 SH.1
9	●┤├●	COMPONENT COOLING PUMP P52A	E-259	BLOCK 30-1359	E-219 SH.3	BORIC ACID PUMP P56A	E-203	●┤/├● BLKS BKR 52-7804 ⊛	E-135 SH.1
10	●┤├●	COMPONENT COOLING PUMP P52C	E-259	BLOCKING RELAYS ACTUATED	E-210 SH. 2	COMPONENT COOLING PUMP P52B	E-259	●┤/├● BLKS BKR 152-102 ⊛	E-151

FUNCTION ON TEST WITHOUT STAND-BY POWER

CONTACT NO.	CONTACT	LEFT (ODD NUMBERS) RELAY 16-1	REF. DWG.	LEFT (ODD NUMBERS) RELAY 16-3	REF. DWG.	RIGHT (EVEN NUMBERS) RELAY 16-2	REF. DWG.	RIGHT (EVEN NUMBERS) RELAY 16-4	REF. DWG.
1	●┤├●	LOW PRESSURE SAFETY INJECTION PUMP P67B	E-248	BORIC ACID PUMP P56B	E-203	LOW PRESSURE SAFETY INJECTION PUMP P67A	E-247	CHARGING PUMP P55A	E-257 SH 1
2	●┤├●	CONTAINMENT COOLER RECIRC. FAN V4A	E-217	HIGH PRESSURE SAFETY INJECTION PUMP P66B	E-249	HIGH PRESSURE SAFETY INJECTION PUMP P66A	E-249	MAINTAIN OPEN CV-1359 ⊛	E-219 SH 3
3	●┤├●	SERVICE WATER PUMP P7B	E-154 SH 1			SERVICE WATER PUMP P7A	E-154 SH 1	SERVICE WATER PUMP P7C	E-154 SH 2
4	●┤/├●			BLOCK 30-1359	E-219 SH.3				
5	●┤├●	COMPONENT COOLING PUMP P52A	E-259	MAINTAIN OPEN CV-1359	E-219 SH.3	BLOCKING RELAYS ACTUATED	E-210 SH. 2	CONTAINMENT COOLER RECIRC. FAN V2A	E-216
6	●┤├●	COMPONENT COOLING PUMP P52C	E-259	CHARGING PUMP P55C	E-257 SH 2	CONTAINMENT COOLER RECIRC. FAN V1A	E-216	CONTAINMENT COOLER RECIRC. FAN V3A	E-216
7	●┤├●			BLOCKING RELAYS ACTUATED	E-210 SH. 2	BORIC ACID PUMP P56A	E-203	CHARGING PUMP P55B	E-257
8	●┤/├●	MAINTAIN OPEN VOLUME CONTROL TANK MOV 2087 ⊛	E-242	BORIC ACID GRAVITY FEED VALVE MOV 2169 ⊛	E-241	SPARE		BORIC ACID PUMPED FEED MOV 2140 ⊛	E-241
9	●┤/├●			BORIC ACID GRAVITY FEED VALVE MOV 2170 ⊛	E-241	SPARE		BLOCK 30-1359	E-219 SH 3
10	●┤├●			TEST START CIRCUIT NO. 1	E-210 SH. 1	COMPONENT COOLING PUMP P52B	E-259	TEST START CIRCUIT NO 2	E-210

⊛ BLOCKS OPERATION ON TEST

NOTES:

1. RELAY V-16-2 CONTACT GROUP #4 IS CHANGED TO NORMALLY OPEN.
2. CONTACTS CHANGED TO NORMALLY OPEN.

NO	DATE	REVISIONS	BY	APP
18	11-4-93	REVISED RELAY V16-1 CONTACT 3 TO MOV 2169 PER DCR 93-1297	JFM	

SCALE NONE DESIGNED DRAWN CHAVOUS

PALISADES PLANT
CONSUMERS POWER COMPANY

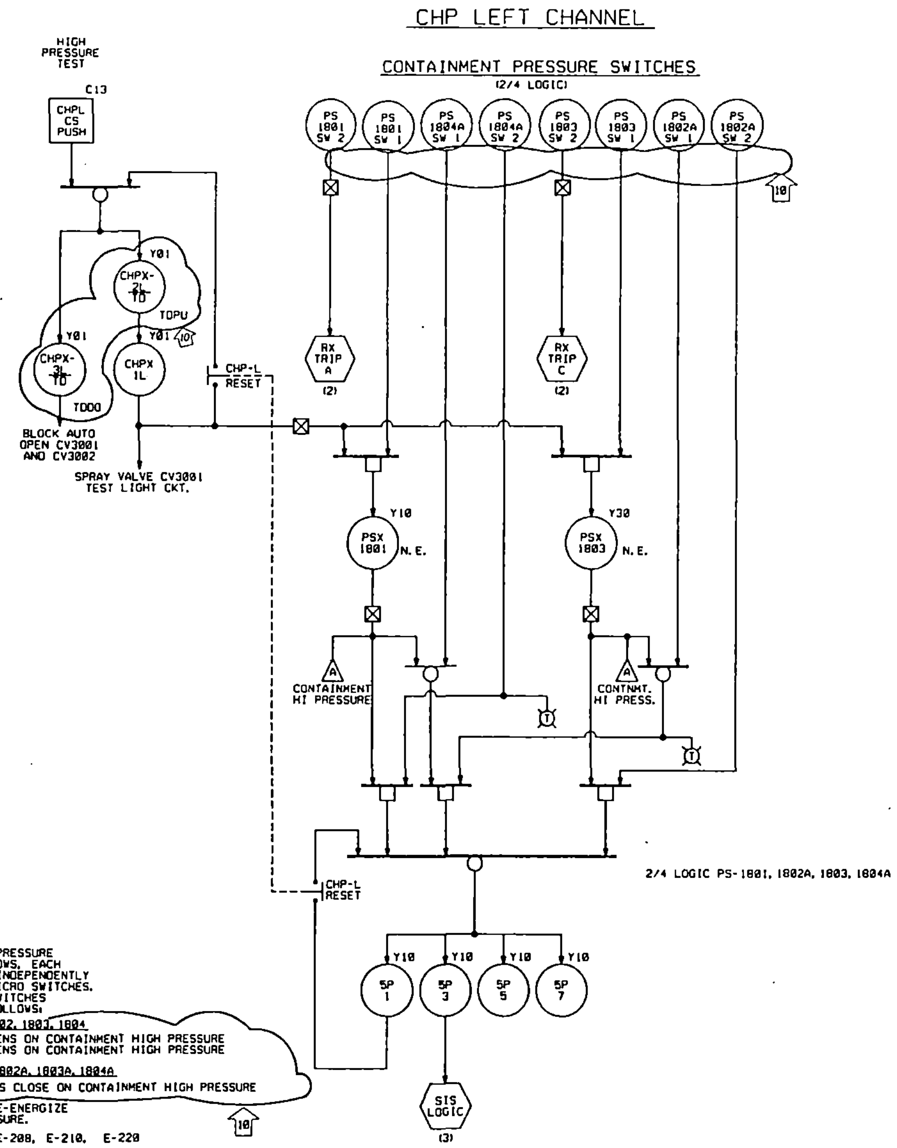
SCHEMATIC WIRING DIAGRAM
SAFETY INJECTION & SEQUENCE
LOADING TEST CIRCUIT

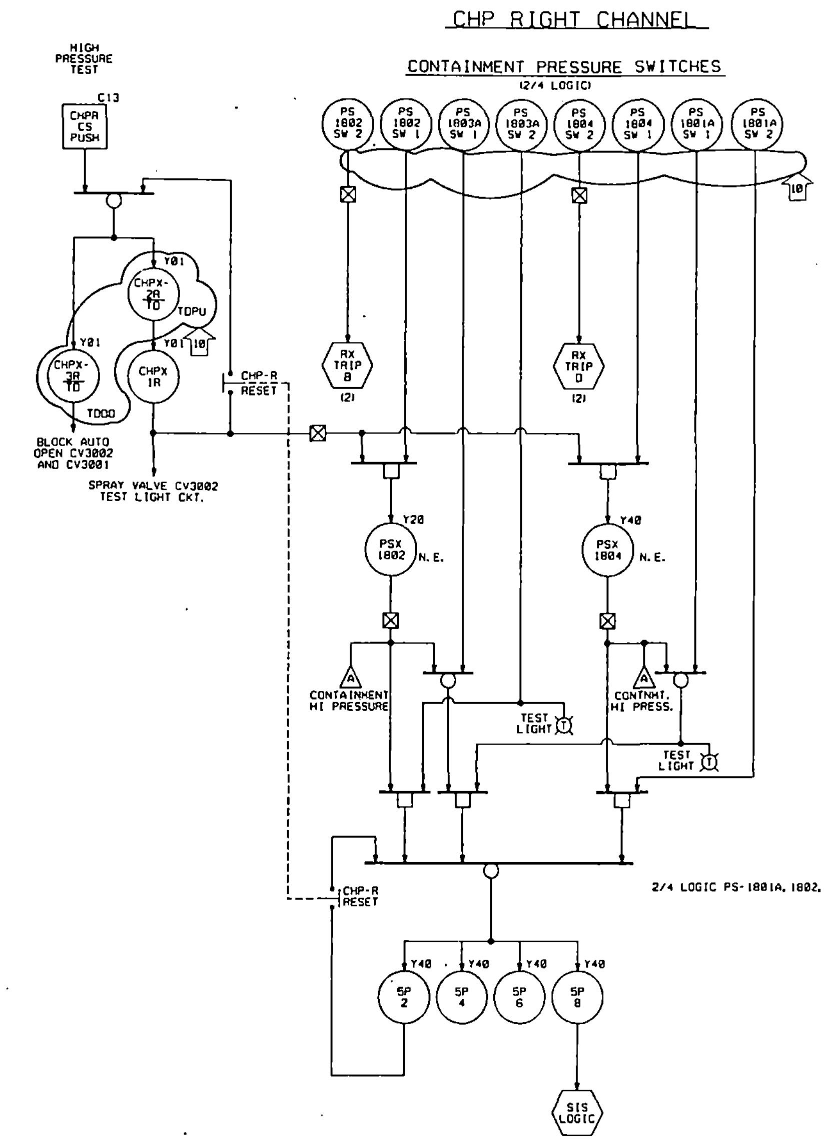
SITE ID.	DRAWING NO.	REV.
0950	E-210 SH.3	18

E210-3. DGN

Figure 7-23 FSAR Rev 17

9501040188-27





NOTE:

1. THERE ARE 8 PRESSURE SENSING BELLOWS, EACH OPERATE TWO INDEPENDENTLY ADJUSTABLE MICRO SWITCHES. THESE MICROSWITCHES ARE SET AS FOLLOWS:
   P.S. 1801, 1802, 1803, 1804
   1 CONTACT OPENS ON CONTAINMENT HIGH PRESSURE
   1 CONTACT OPENS ON CONTAINMENT HIGH PRESSURE
   P.S. 1801A, 1802A, 1803A, 1804A
   BOTH CONTACTS CLOSE ON CONTAINMENT HIGH PRESSURE
2. PSX RELAYS DE-ENERGIZE ON HIGH PRESSURE.
3. REF: E-207, E-208, E-210, E-220

5P-1	5P-3	5P-5	5P-7
CLOSE DEMIN WATER TO QUENCH TANK CV0155	CLOSE STM GEN E-50A SURFACE BLOWDN LINE CV0739	CLOSE NITROGEN ISOLATION VALVE CV1358	CLOSE PURGE EXHAUST FAN OUTLET CV1806
CLOSE PRI COOLANT PUMP CONTROLLED BLEED OFF CV2083	SIS INITIATION CKT 1 (3)	CLOSE CONTNMT BLDG HEAT STM DISCHARGE CV1501	CLOSE PURGE SUPPLY FAN VALVE CV1808
CLOSE PRI COOLANT LETDOWN ISOL VALVE CV2009	CLOSE CLEAN WASTE REC TANK ISOL VALVE CV1004	CLOSE CONTNMT BLDG HEAT STM DISCHARGE CV1503	CLOSE AIR SPACE PURGE FAN VALVE CV1814
CLOSE STM GEN E-50B SURFACE BLOWDN LINE CV0738	CLOSE CLEAN WASTE REC TANK ISOL VALVE CV1036	CLOSE CONTNMT HYD MON ISL VLV CV2413A & B. CV2415A & B	OPEN CONTAINMENT SPRAY VALVE CV3001
CLOSE STM GEN E-50A BOT BLOWDN LINE CV0771	CLOSE CLEAN WASTE REC TANK ISOL VALVE CV1037	CLOSE PRI SYS SAMPLING ISOLATION VALVE CV1910	CONTAINMENT ISOLATION ALARM
CLOSE STM GEN E-50B BOT BLOWDN LINE CV0770	CLOSE MSIV CV-0501 AND CV-0510 (20)	TRIP AIR ROOM PURGE FAN V-46	START CONTAINMENT SPRAY PUMP P-54B
CLOSE SHIELD COOL SURGE TANK FILL CV0939	CLOSE CLEAN WASTE REC TANK PUMP SUCTION CV1044		START CONTAINMENT SPRAY PUMP P-54C
CLOSE PRI SYS DRAIN TANK VALVE CV1001		CLOSE FW REG VALVE CV-0701	EVENTS RECORDER
CLOSE PRI SYS DRAIN TANK VALVE CV1002	CLOSE CONTAINMENT WASTE GAS VALVE CV1101	CLOSE FW REG BYPASS VALVE CV-0735	CLOSE COMPT CLG WATER DISCH FROM CONTMT SV0911
CIRCUIT SEAL-IN CONTACT	CONTROL ROOM HVAC EMERGENCY SYS ACTIVATION (TRAIN A)		CLOSE COMPT CLG WATER INLET TO CONTMT SV0910
CLOSE RECEIVER TANK VENT VALVE CV1064			CLOSE CONT. H2 MONITORING ISOL. VLVS. SV-2412 A & B AND SV-2414 A & B.
			CONTROL RM HVAC EMERGENCY SYSTEM ACTIVATION (TRAIN B)

5P-2	5P-4	5P-6	5P-8
CLOSE DEMIN WATER TO QUENCH TANK CV0155	CLOSE STM GEN E-50A SURFACE BLOWDN LINE CV0739	CLOSE NITROGEN ISOLATION VALVE CV1358	CLOSE PURGE EXHAUST FAN OUTLET CV1805
CLOSE PRI COOLANT PUMP CONTROLLED BLEED OFF CV2099	CLOSE MSIV CV-0501 (20) AND CV-0510	CLOSE CONTNMT HYD MON ISL VLV CV2412A & B. CV2414A & B	CLOSE PURGE SUPPLY FAN VALVE CV1807
CLOSE PRI COOLANT LETDOWN ISOL VALVE CV2009	CLOSE PRI SYS DRAIN TANK VALVE CV1007	CLOSE CONTNMT BLDG HEAT STM DISCHARGE CV1502	CLOSE AIR SPACE PURGE FAN VALVE CV1813
CLOSE STM GEN E-50B SURFACE BLOWDN LINE CV0738	CLOSE CLEAN WASTE REC TANK ISOL VALVE CV1004	CLOSE CONTNMT BLDG HEAT STM DISCHARGE CV1503	OPEN CONTAINMENT SPRAY VALVE CV3002
CONTAINMENT ISOLATION ALARM	CLOSE CLEAN WASTE REC TANK ISOL VALVE CV1037	CLOSE WASTE GAS SURGE TANK INLET CV1102	START CONTAINMENT SPRAY PUMP P-54A
CLOSE SHIELD COOL SURGE TANK INLET CV0939	CLOSE CLEAN WASTE REC TANK ISOL VALVE CV1038	TRIP AIR ROOM PURGE FAN V-46	SIS INITIATION CKT 2 (3)
CLOSE PRI SYS DRAIN TANK VALVE CV1001	CONTROL ROOM HVAC EMERGENCY SYS ACTIVATION (TRAIN B)		EVENTS RECORDER
CIRCUIT SEAL-IN CONTACT	CLOSE CLEAN WASTE REC TANK PUMP SUCTION CV1045	CLOSE FW REG VALVE CV-0703	CLOSE PRI SYS SAMPLING ISOLATION VALVE CV1911
CLOSE CONT. H2 MONITORING ISOLATION VALVES SV-2413 A & B AND SV-2415 A & B		CLOSE FW REG BYPASS VALVE CV-0734	CLOSE STM GEN E-508 BOT BLWDN LINE CV0768
	CLOSE RECEIVER TANK VENT VALVE CV1065	CONTROL RM HVAC EMERGENCY SYSTEM ACTIVATION (TRAIN A)	CLOSE STM GEN E-508 BOT BLWDN LINE CV0767
			CLOSE COMP CLG WATER FROM CONTMT SV0940
			CLOSE COMP CLG WATER INLET TO CONTMT SV0910

9807090043 -16

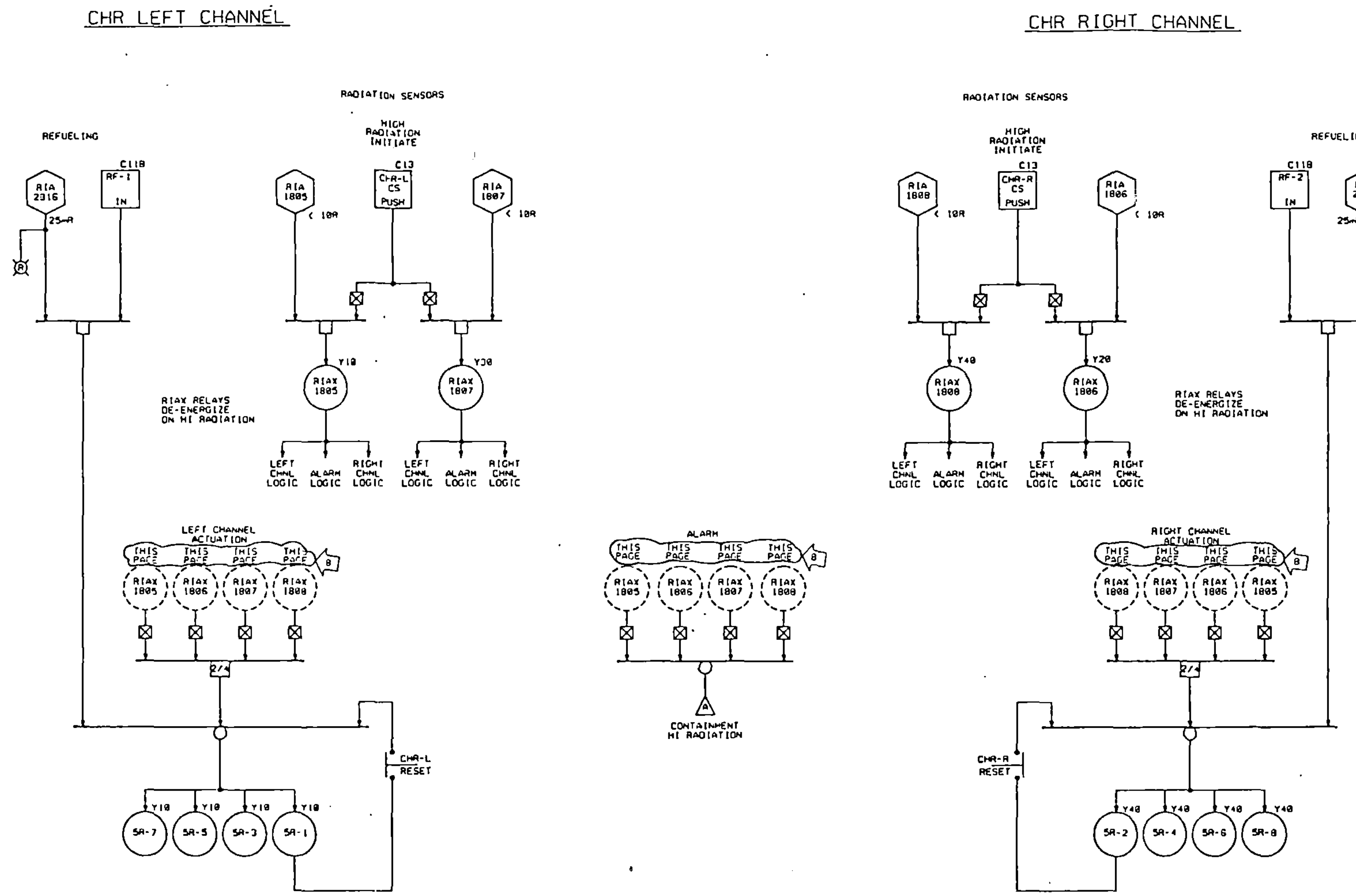
REV	DATE	DESCRIPTION	BY	CK	APP
10	10-10-97	REV'D. SETPOINTS & RELAY SETTINGS (SEE E-17 SH. 1) PER DCR 97 850.	JCD		

CONSUMERS POWER COMPANY
PALISADES NUCLEAR PLANT
COVERT, MICHIGAN

LOGIC DIAGRAM
CONTAINMENT HIGH PRESSURE

E-17 6 10

FIG 7-24
FSAR REV 20



5R-1	5R-3	5R-5	5R-7
CLOSE DEMIN WATER TO QUENCH TANK CV0155	CLOSE STM GEN E-50B SURFACE BLOWDN LINE CV0739	CLOSE NITROGEN ISOLATION VALVE CV1358	CLOSE PURGE EXHAUST FAN OUTLET CV1806
CLOSE PRI COOLANT PUMP CONTROLLED BLEED OF CV2083	CLOSE CLEAN WASTE REC TANK ISOL VALVE CV1004	CLOSE CONTAINMENT BLDG HEAT STM DISCHARGE CV1501	CLOSE PURGE SUPPLY FAN VALVE CV1808
CLOSE PRI COOLANT LETDOWN ISOL VALVE CV2009	CLOSE CLEAN WASTE REC TANK ISOL VALVE CV1036	CLOSE CONTAINMENT BLDG HEAT STM DISCHARGE CV1503	CLOSE AIR SPACE PURGE FAN VALVE CV1814
CLOSE STM GEN E-50A SURFACE BLOWDN LINE CV0738	CLOSE CLEAN WASTE REC TANK ISOL VALVE CV1037	CLOSE CONT HYD MON ISL VLV CV2413A & B, CV2415A & B	CONTAINMENT ISOLATION ALARM
CLOSE STM GEN E-50A BOT BLOWDN LINE CV0771	CLOSE CLEAN WASTE REC TANK PUMP SUCTION CV1044	CLOSE PRI SYS SAMPLING ISOLATION VALVE CV 1910	EVENTS RECORDER
CLOSE STM GEN E-50A BOT BLOWDN LINE CV0770	CLOSE CONTAINMENT WASTE GAS VALVE CV1101	TRIP AIR ROOM PURGE FAN V-46	CLOSE CONT. H2 MONITORING ISOLATION VALVES SV-2412A & SV-2412B SV-2414A & SV-2414B
CLOSE SHIELD COOL SURGE TANK FILL CV0939	CONTROL ROOM HVAC EMERGENCY SYS ACTIVATION (TRAIN A)	CLOSE CONTAINMENT SUMP DRAIN VALVE CV1103	
CLOSE PRI SYS DRAIN TANK VALVE CV1001		PREVENTS OPERATION OF ESS SUMP PUMP P-72B WHILE HS IN AUTO (E-285)	CONTROL ROOM HVAC EMERGENCY SYS ACTIVATION (TRAIN B)
CLOSE PRI SYS DRAIN TANK VLV CV1002		PREVENTS OPERATION OF ESS SUMP PUMP P-73B WHILE HS IN AUTO (E-285)	
CIRCUIT SEAL-IN CONTACT			
CLOSE RECEIVER TANK VENT VALVE CV1064			

5R-2	5R-4	5R-6	5R-8
CLOSE DEMIN WATER TO QUENCH TANK CV-0155	CLOSE STM GEN E-50B SURFACE BLOWDN LINE CV0739	CLOSE NITROGEN ISOLATION VALVE CV1358	CLOSE PURGE EXHAUST FAN OUTLET CV1805
CLOSE PRI COOLANT PUMP CONTROLLED BLEED OFF CV2099	CLOSE PRI SYS DRAIN TANK VALVE CV1007	CLOSE CONT HYD MON ISL VLV CV2412A & B CV2414A & B	CLOSE PURGE SUPPLY FAN VALVE CV1807
CLOSE PRI COOLANT LETDOWN ISOL VALVE CV2009	CLOSE CLEAN WASTE REC TANK ISOL VALVE CV1004	CLOSE CONT BLDG HEAT STM DISCHARGE CV1502	CLOSE AIR SPACE PURGE FAN VALVE CV1813
CLOSE STM GEN E-50A SURFACE BLOWDN LINE CV0738	CLOSE CLEAN WASTE REC TANK ISOL VALVE CV1037	CLOSE CONT BLDG HEAT STEAM INLET CV1503	CLOSE PRI SYS SAMPLING ISOLATION VALVE CV1911
CONTAINMENT ISOLATION ALARM	CLOSE CLEAN WASTE REC TANK ISOL VALVE CV1038	CLOSE WASTE GAS SURGE TANK INLET CV1102	CLOSE STM GEN E-50B BOT BLOWDN LINE CV0768
CLOSE SHIELD COOL SURGE TANK FILL CV0939	CONTROL ROOM HVAC EMERGENCY SYS ACTIVATION (TRAIN B)	TRIP AIR ROOM PURGE FAN V-46	CLOSE STM GEN E-50A BOT BLOWDN LINE CV0767
CLOSE PRI SYS DRAIN TANK VALVE CV1001	CLOSE CLEAN WASTE REC TANK PUMP SUCTION CV1045	CLOSE CONTAINMENT SUMP DRAIN VALVE CV1104	EVENTS RECORDER
CIRCUIT SEAL-IN CONTACT	CLOSE RECEIVER TANK VENT VALVE CV1065	PREVENTS OPERATION OF ESS SUMP PUMP P-73A WHILE HS IN AUTO (E-285)	
CLOSE CONT. H2 MONITORING ISOLATION SV'S SV-2413A & SV-2413B SV-2415A & SV-2415B		PREVENTS OPERATION OF ESS SUMP PUMP P-72A WHILE HS IN AUTO (E-285)	
		CONTROL ROOM HVAC EMERGENCY SYS ACTIVATION (TRAIN A)	

REF: E-207, E-208

ANSTEC APERTURE CARD

Also Available on Aperture Card

9807090043 -17

8	3-15-95	ADDED "THIS PAGE" TO ALARM, RIGHT & LEFT CHANNEL ACTUATION PER DCR 95-193			
SCALE NONE		B. N. YOUNG		P. D. BROWN	

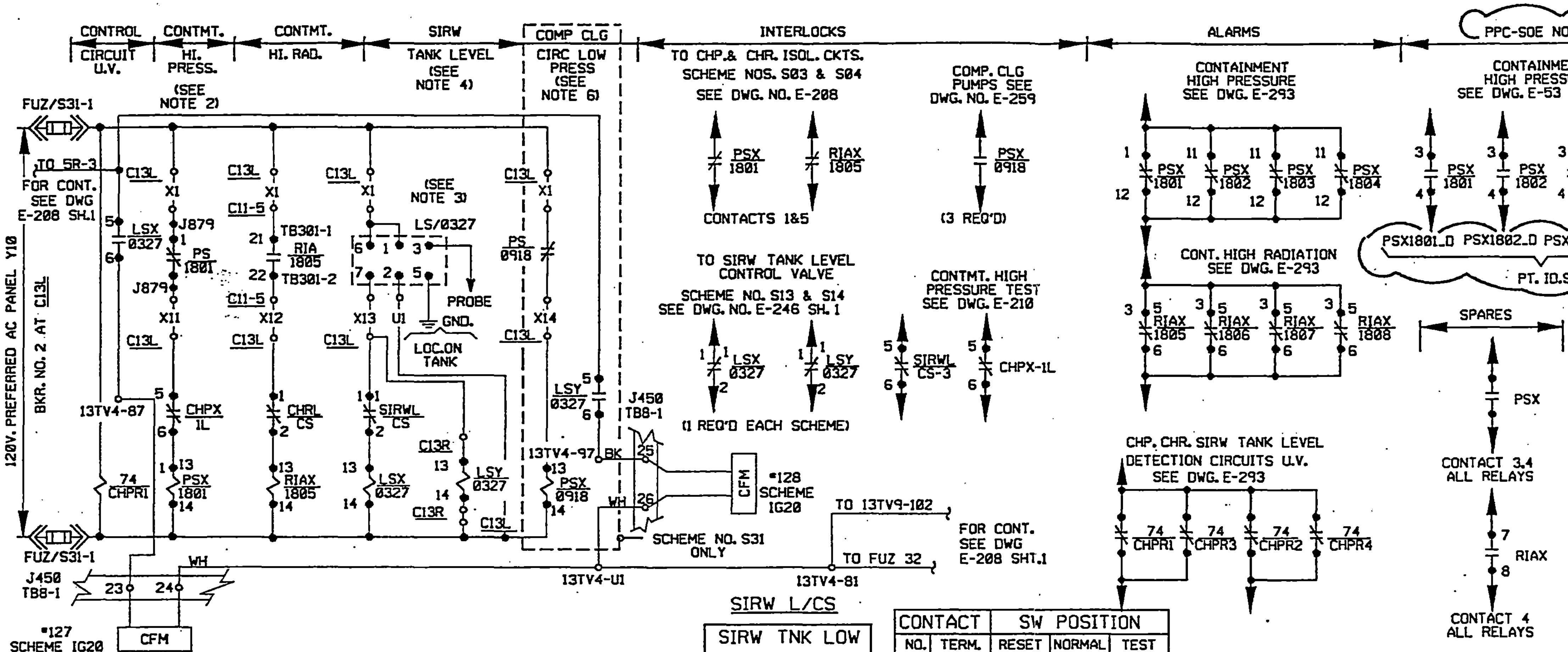
PALISADES PLANT
CONSUMERS POWER COMPANY E17-7.DGN

LOGIC DIAGRAM
CONTAINMENT HIGH RADIATION

E17 SH. 7 — 8

FIG 7-25
FSAR REV 20

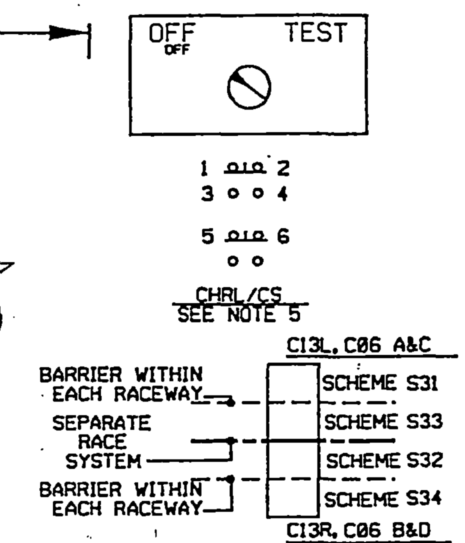
DWG. NO.



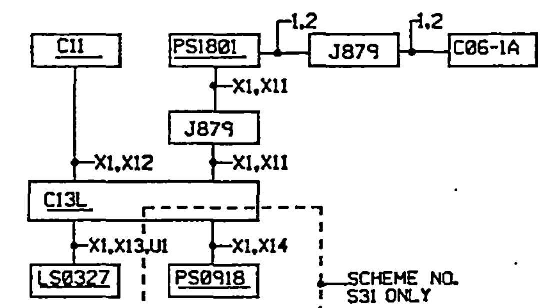
CONTAINMENT HIGH PRESSURE, HIGH RADIATION
SIRW TANK LOW LEVEL DETECTION-CHANNEL NO. 1 CIRCUIT
SCHEME NO. S31-(ONE OF FOUR IDENTICAL CIRCUITS)

1 2 NC 5 6
3 4 NO 7 8
9 10 NC 13 14
11 12 NO 15 16

CONTACT		SW POSITION		
NO.	TERM.	RESET	NORMAL	TEST
1	1 2	X	X	0
2	3 4	0	0	X
3	5 6	0	X	X
4	7 8	X	0	0
5	9 10	X	X	0
6	11 12	0	0	X
7	13 14	0	X	X
8	15 16	X	0	0



NOTES:
1. TEST ONLY WHEN PLANT IS SHUTDOWN.
2. PS/1801 OPENS ON CONTMT. HIGH PRESSURE.
3. RIA OPEN ON CONTMT. HI. RAD. (DWG. E-227).
4. LS OPENS ON SIRW TANK LOW LEVEL.
5. CHRL/CS-1S 2 POSITION SPRING RETURN TO OFF POSITION. GE TYPE CR2940.
6. PS/0918 CLOSES ON COMP. CLG. LOW PRESS.
7. SIRWL/CS IS 3 POSITION SPRING RETURN FROM LEFT & RIGHT TO NORMAL POSITION. GE TYPE CR104P, WITH BAR OPERATOR.



ADAPTER TABLE

CHANNEL NO.	SCHEME NO.	PREF. AC PNL. AND BKR.	PS & PSX	RIA & RIAX	RELAY	LS	LOCATIONS		CHP TEST RELAY CONTACTS	CONTROL SWITCHES & CONTACTS	LS AUX. RELAY NO.	LOC.	NO.	LOC.	UV RELAY 74/
1	S31	Y10-2	1801	1805	K1	0327	Y10	C13L	CHPX-1L/3	CHRL/CS-1 SIRWL/CS-1	LSX 0327	C13L	LSY 0327	C13R	CHPR1
2	S32	Y20-2	1802	1806	K2	0328	Y20	C13R	CHPX-1R/5	CHRR/CS-1 SIRWL/CS-5	LSX 0328	C13L	LSY 0328	C13R	CHPR2
3	S33	Y30-2	1803	1807	K3	0329	Y30	C13L	CHPX-1L/5	CHRL/CS-5 SIRWR/CS-1	LSY 0329	C13R	LSX 0329	C13L	CHPR3
4	S34	Y40-2	1804	1808	K4	0330	Y40	C13R	CHPX-1R/3	CHRR/CS-3 SIRWR/CS-5	LSY 0330	C13R	LSX 0330	C13L	CHPR4

CFM INPUT NO.	CPCO TAG NO.
127	LSX-0327
128	LSY-0327

SCHEME IG20 APPLIES TO CFM CIRCUIT ONLY

REV	DATE	DESCRIPTION	BY	CK	APP
25	7-31-95	REMOVED DATALOGGER, ADDED PPC-SOE NODE PER FC-933-06 AND DCR 95-520	DMB		fm

CONSUMERS POWER COMPANY
PALISADES NUCLEAR PLANT
COVERT, MICHIGAN

E207-1.DGN

PREVIOUS RECORD ISSUE

MICHIGAN P.E. No.

UFI No.

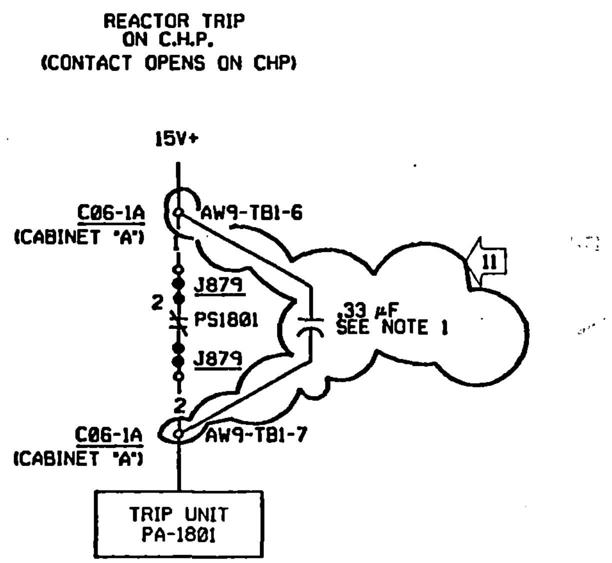
SCHEMATIC DIAGRAM
CONTAINMENT HIGH PRESSURE-
HIGH RADIATION & SIRW TANK LOW LEVEL

DRAWING NUMBER: E207 | SHT. NO.: 1 | REV: 25

Figure 7-26
Rev 19

9707070091-13

DWG. NO.



11

NOTES:

1. CAPACITOR TO FILTER NOISE PER FC-888

ADAPTER TABLE

REACTOR PROTECTIVE SYS. CABINET	JUNCTION BOX NO.	TRIP UNIT PA-	FUSE NO.
A (L)	J879	1801	FUZ/S31-1
B (R)	J868	1802	FUZ/S32-1
C (L)	J885	1803	FUZ/S33-1
D (R)	J870	1804	FUZ/S34-1

11

NO	DATE	REVISIONS	BY	CHK	ENGR
11	7-2-92	ADDED FUSE EDB NO. PER 950-91-883 (MAL) ADDED NOTE 1 PER 950-92-221 (ADS)	JJB	JGD	DMB
10	4-4-86	ADDED TRIP UNITS PER REVIEW	JLS	JGD	RSW

SCALE | DESIGNED | DRAWN

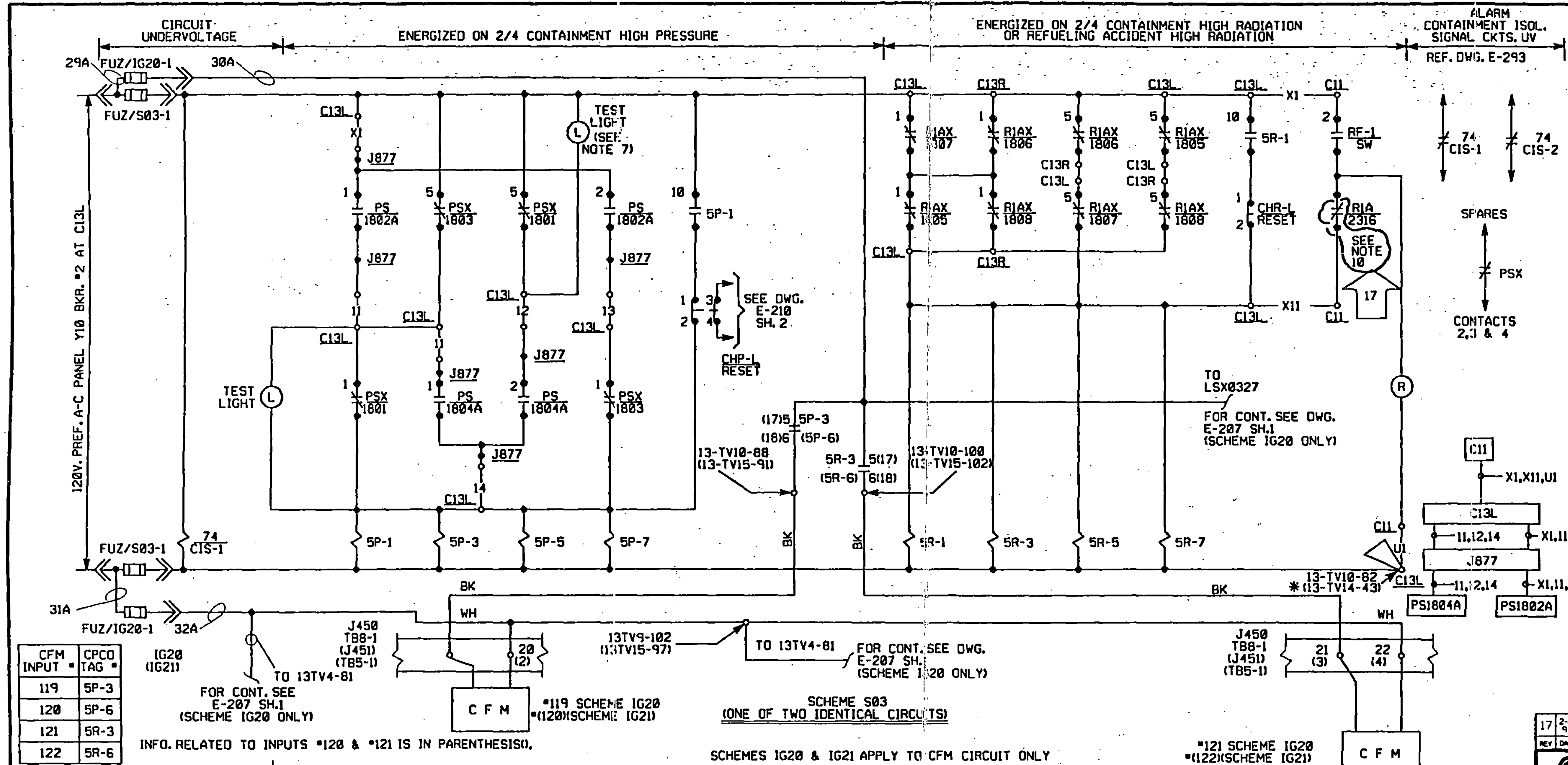
PALISADES PLANT
CONSUMERS POWER COMPANY

SCHEMATIC DIAGRAM E207-2.DGN
CONTAINMENT HIGH PRESSURE HIGH RADIATION & SIRW TANK LOW LEVEL

SITE ID.	DRAWING NO.	REV.
0950	E-207 SH. 2	11

Figure No 7-27 FSAR Rev 15

9305130068 -35



NOTES:

1. PS/1802A & PS/1804A ARE CONTAINMENT PRESSURE SWITCH CONTACTS CONNECTED DIRECTLY IN CIRCUIT #1 CONTACTS CLOSE ON HIGH PRESSURE.
2. PS/1801A & PS/1803A ARE CONTAINMENT HIGH PRESSURE SWITCH CONTACTS CONNECTED DIRECTLY IN CIRCUIT #2. CONTACTS CLOSE ON HIGH PRESSURE.
3. PS/1801, PS/1802, PS/1803 & PS/1804 ARE USED WITH AUX. RELAYS. THESE CONTACTS OPEN ON CONTAINMENT HIGH PRESSURE. (SEE DWG. E-207)
4. CONTAINMENT ISOLATION RELAYS ENERGIZED ON CONTAINMENT HIGH PRESSURE OR HIGH RADIATION.
5. • OPEN CONTACT WHEN RELAY IS DEENERGIZED. (REF. THIS DWG. SH. 2)
6. FOR 5P & 5R RELAY TABLE, SEE SH. 2
7. TEST LIGHT WILL BE ENERGIZED ON PRESSURE SWITCH TEST.

REFUEL RAD. MON. OUT — REFUEL RAD. MON. IN — OFF

RF-1 SW & RF-2 SW (KEY REMOVABLE IN "OUT" POSITION ONLY)

ADMINISTRATIVE CONTROL REFUELING ACCIDENT HIGH RADIATION SWITCH

G.E. TYPE CR2940 KEY OPERATED WITH MAINTAINED CONTACTS.

8. "R" LIGHT IS ON WHEN REFUELING HIGH RADIATION CIRCUIT IS IN SERVICE.
9. ✳ INDICATES JUMPERED BROKEN LINK
10. ALARM CONTACTS ARE FAIL-SAFE AND CHANGE STATE UPON RATEMETER POWER UP.

CFM INPUT *	CPCO TAG *
119	5P-3
120	5P-6
121	5R-3
122	5R-6

INFO. RELATED TO INPUTS *120 & *121 IS IN PARENTHESIS().

SCHEMES IG20 & IG21 APPLY TO CFM CIRCUIT ONLY

ADAPTER TABLE

CIRCUIT NO.	SCHEME NO.	PREF PANEL	A-C BKR.	AUX. RELAYS NO.S	AUX. RELAYS CONTACT	PRESSURE SWITCH NO.S	PRESSURE SWITCH CONTACT	ACTUATING RELAYS	RESET PUSHBUTTONS		UNDERVOLT RELAY	RIAX RELAYS	RIAX CONTACTS	REFUEL SWITCH	HI. RAD. CONT.	J-BOX	FUSE NO.
1	S03	Y10	2	PSX/1801 PSX/1803	1 & 5 1 & 5	PS/1802A PS/1804A	1 & 2 1 & 2	5P-1,5P-3,5P-5,5P-7 5R-1,5R-3,5R-5,5R-7	CHP-L RESET	CHR-L RESET	74/CIS-1	1805,1806, 1807 & 1808	1 & 5	RF-1 SW	RIA 2316	J877	FUZ/S03-1 FUZ/IG20-1
2	S04	Y40	2	PSX/1802 PSX/1804	1 & 5 1 & 5	PS/1801A PS/1803A	1 & 2 1 & 2	5P-2,5P-4,5P-6,5P-8 5R-2,5R-4,5R-6,5R-8	CHP-R RESET	CHR-R RESET	74/CIS-2	1805,1806, 1807 & 1808	2 & 6	RF-2 SW	RIA 2317	J853	FUZ/S04-1 FUZ/IG21-1

SEPARATE RACEWAY SYSTEM — SCHEME S03 / SCHEME S04

REV	DATE	DESCRIPTION	BY	APP
17	2-11-94	ADDED NOTE 10 PER SC-92-169 AND DCR 94-071	JFM	URP

CONSUMERS POWER COMPANY
PALISADES NUCLEAR PLANT
COVERT, MICHIGAN

SCHEMATIC DIAGRAM
CONTAINMENT ISOLATION ON HIGH PRESSURE OR HIGH RADIATION

DRAWING NUMBER E208 SHT. NO. 1 REV 17

9501040188-29

Figure 7-28 FSAR Rev 17

CONTAINMENT ISOLATION RELAY TABLE

CONTACTS	RELAYS 5P-1 & 5R-1	REF. DWG.	RELAYS 5P-3 & 5R-3	REF. DWG.	RELAYS 5P-5 & 5R-5	REF. DWG.	RELAYS 5P-7 & 5R-7	REF. DWG.
1	CLOSE DEMIN. WATER TO QUENCH TANK SV0155	E-235 SH. 2	CLOSE STM. GEN. E-50A TOP BLOWDN. LINE SV0739	E-235 SH. 3	DISABLE AUTO START OF ENG SFGRDS SUMP PUMP P-72B (5R-5 ONLY)	E-285	CLOSE PURGE EXHAUST FAN OUTLET SV1806	E-221
2	CLOSE PRI. COOLANT PUMP CONTROLLED BLEED OFF SV2083	E-235 SH. 2	•SIS INITIATION CKT. 1 (5P-3 ONLY)	E-209 SH. 1	CLOSE NITROGEN ISOLATION VALVE SV1358	E-235 SH. 3	CLOSE PURGE SUPPLY FAN VALVE SV1808	E-221
3	CLOSE PRI. COOLANT LETDOWN ISOL. VALVE SV2009	E-235 SH. 2	• CRITICAL FUNCTION MON. INPUTS: *119 (FOR 5P-3), *121 (FOR 5R-3)	E-208 SH. 1	CLOSE CONTAINMENT BLDG. HEAT STM. DISCHARGE SV1501	E-235 SH. 1	CLOSE AIR SPACE PURGE FAN VALVE SV1814	E-221
4	CLOSE STM. GEN. E-50B TOP BLOWDN. LINE SV0738	E-235 SH. 3	CLOSE CLEAN WASTE REC. TANK ISOL. VALVE SV1004	E-235 SH. 3	• CLOSE CV-0701 FW REG VLV (5P-5 ONLY)	E-69 SH. 1	OPEN CONTAINMENT SPRAY VALVE SV3001 (5P-7 ONLY)	E-237
5	CLOSE STM. GEN. E-50A BOT. BLOWDN. LINE SV0771	E-235 SH. 4	CLOSE CLEAN WASTE REC. TANK ISOL. VALVE SV1036	E-235 SH. 1	CLOSE CONTAINMENT BLDG. HEAT STM. INLET SV1503	E-235 SH. 3	• CONTAINMENT ISOLATION ALARM K11 *26	E-292 SH. 2
6	CLOSE STM. GEN. E-50B BOT. BLOWDN. LINE SV0770	E-235 SH. 4	CLOSE CLEAN WASTE REC. TANK ISOL. VALVE SV1037	E-235 SH. 3	CLOSE CONT. HYD. MON. ISL. VLV. SV-2413A & B, SV-2415A & B	E-916 SH. 1	• START CONTAINMENT SPRAY PUMP P-54B (5P-7 ONLY)	E-251
7	CLOSE SHIELD COOL SURGE TANK INLET SV0939	E-235 SH. 3	• CLOSE MSIV CV0501 & CV-0510 (5P-3 ONLY)	E-238 SH. 1	CLOSE CONTAINMENT SUMP DR. VALVE SV1103 (RELAYS 5R-5 ONLY)	E-235 SH. 1	• START CONTAINMENT SPRAY PUMP P-54C (5P-7 ONLY)	E-251
8	CLOSE PRI. SYS. DRAIN TANK VALVE SV1001	E-235 SH. 3	CLOSE CLEAN WASTE REC. TANK PUMP SUCTION SV1044	E-235 SH. 1	• CLOSE CV-735 FW REG BYPASS VLV (5P-5 ONLY)	E-69 SH. 1	CLOSE CONT. H_2 MONT. ISOL. VLV. SV-2412A & B & SV-2414A & B	E-916 SH. 2
9	CLOSE PRI. SYS. DRAIN TANK VALVE SV1002	E-235 SH. 1	• SPARE		CLOSE PRI. SYS. SAMPLING ISOLATION VALVE SV1910	E-235 SH. 1	CONTROL ROOM HVAC EMERGENCY SYSTEM ACTIVATION (TRAIN B)	E-271 SH. 8
10	• CIRCUIT SEAL-IN CONTACT	E-208	CLOSE CONTAINMENT WASTE GAS INLET VALVE SV1101	E-235 SH. 1	TRIP AIR ROOM PURGE FAN V-46	E-222	• CLOSE COMP CLG WTR DISCH FROM CONT SV0911(5P-7ONLY)	E-235 SH. 4
11	CLOSE RECEIVER TANK VENT VALVE SV1064	E-235 SH. 1	CONTROL ROOM HVAC EMERGENCY SYS. ACTIVATION (TRAIN A)	E-271 SH. 8	DISABLE AUTO START OF ENG SFGRDS SUMP PUMP P-73B (5R-5 ONLY)	E-285	• CLOSE COMP CLG WTR INLET TO CONT SV-0910(5P-7ONLY)	E-235 SH. 4
12	SPARE		• SPARE		• SPARE		PPC-SOE NODE PT. ID. KS5P_5R_D	E-53 SH. 8

CONTACTS	RELAYS 5P-2 & 5R-2	REF. DWG.	RELAYS 5P-4 & 5R-4	REF. DWG.	RELAYS 5P-6 & 5R-6	REF. DWG.	RELAYS 5P-8 & 5R-8	REF. DWG.	RELAY 5P-6 (CON'T)	REF. DWG.
1	CLOSE DEMIN. WATER TO QUENCH TANK SV0155	E-235 SH. 2	CLOSE STM. GEN. E-50A TOP BLOWDN. LINE SV0739	E-235 SH. 3	CLOSE CONTAINMENT SUMP DR. VALVE SV1104 (RELAYS 5R-6 ONLY)	E-235 SH. 1	CLOSE PURGE EXHAUST FAN OUTLET SV1805	E-221		
2	CLOSE PRI. COOLANT PUMP CONTROLLED BLEED OFF SV2099	E-235 SH. 2	• CLOSE MSIV CV-0501 & CV-0510 (5P-4 ONLY)	E-238 SH. 1	CLOSE NITROGEN ISOLATION VALVE SV1358	E-235 SH. 3	CLOSE PURGE SUPPLY FAN VALVE SV1807	E-221		
3	CLOSE PRI. COOLANT LETDOWN ISOL. VALVE SV2009	E-235 SH. 2	CLOSE PRI. SYS. DRAIN TANK VALVE SV1007	E-235 SH. 1	CLOSE CONT. HYD. MON. ISL. VLV. SV-2412A & B, SV-2414A & B	E-916 SH. 2	CLOSE AIR SPACE PURGE FAN VALVE SV1813	E-221		
4	CLOSE STM. GEN. E-50B TOP BLOWDN. LINE SV0738	E-235 SH. 3	CLOSE CLEAN WASTE REC. TANK ISOL. VALVE SV1004	E-235 SH. 3	CLOSE CONTAINMENT BLDG. HEAT STM. DISCHARGE SV1502	E-235 SH. 1	OPEN CONTAINMENT SPRAY VALVE SV3002 (5P-8 ONLY)	E-237		
5	• CONTAINMENT ISOLATION ALARM K11 *26	E-292 SH. 2	• SPARE		CLOSE CONTAINMENT BLDG. HEAT STM. INLET SV1503	E-235 SH. 3	• CLOSE COMP CLG WTR DISC CONMT SV-0940 (5P-8 ONLY)	E-235 SH. 4A		
6	CLOSE CONT. H_2 MONT. ISOL. VLV. SV-2413A & B & SV-2415A & B	E-916 SH. 1	CLOSE CLEAN WASTE REC. TANK ISOL. VALVE SV1037	E-235 SH. 3	CLOSE WASTE GAS SURGE TANK INLET SV1102	E-235 SH. 1	• START CONTAINMENT SPRAY PUMP P-54A (5P-8 ONLY)	E-251		
7	CLOSE SHIELD COOL SURGE TANK INLET SV0939	E-235 SH. 3	CLOSE CLEAN WASTE REC. TANK ISOL. VALVE SV1038	E-235 SH. 1	DISABLE AUTO START OF ENG SFGRDS SUMP PUMP P-72A (5R-6 ONLY)	E-285	• SIS INITIATION CKT. 2 (RELAYS 5P-8 ONLY)	E-209 SH. 1	• CLOSE FW REG VLVE CV-0703	E-69 SH.1
8	CLOSE PRI. SYS. DRAIN TANK VALVE SV1001	E-235 SH. 3	CONTROL ROOM HVAC EMERGENCY SYS. ACTIVATION (TRAIN B)	E-271 SH. 8	CONTROL ROOM HVAC EMERGENCY SYSTEM ACTIVATION (TRAIN A)	E-271 SH. 8	• CLOSE COMP CLG WTR INLET CONMT SV-0910 (5P-8 ONLY)	E-235 SH. 4		
9	• SPARE		CLOSE CLEAN WASTE REC. TANK PUMP SUCTION SV1045	E-235 SH. 1C	• CRITICAL FUNCTION MON. INPUTS: *120 (FOR 5P-6, *122 (FOR 5R-6)	E-208 SH. 1	CLOSE PRI. SYS. SAMPLING ISOLATION VALVE SV1911	E-235 SH. 1		
10	• CIRCUIT SEAL-IN CONTACT	E-208	• SPARE		TRIP AIR ROOM PURGE FAN V-46	E-222	CLOSE STM. GEN. E-50B BOT. BLOWDN. LINE SV0768	E-235 SH. 4		
11	• SPARE		CLOSE RECEIVER TANK VENT VALVE SV1065	E-235 SH. 1	SPARE		CLOSE STM. GEN. E-50A BOT. BLOWDN. LINE SV0767	E-235 SH. 4		
12	• SPARE		SPARE		DISABLE AUTO START OF ENG SFGRDS SUMP PUMP P-73A (5R-6 ONLY)	E-285	PPC-SOE NODE PT. ID. KS5P_5R_D	E-53 SH. 8	• CLOSE FW REG BYPASS VALVE CV-0734	E-69 SH.1

NOTES:

1. RELAYS WITH 8 OR LESS NC CONTACTS SHALL UTILIZE COIL TB113-3. RELAYS WITH 9 OR MORE NC CONTACTS SHALL UTILIZE COIL TB113-61.

• - NORMALLY OPEN CONTACT

REFERENCE DWGS:

LOGIC DIAGRAM	JLG-121(0) SH. 2
SCHEMATIC DIAGRAMS	E-271 SH. 8
	E-271 SH. 1
	E-223
	E-224
	E-916
	E-69 SH. 1

SPARE CONTACTS		
RELAY	NORMALLY OPEN	NORMALLY CLOSED
5R-1	—	• 12
5P-3	• 9, 12	—
5R-3	• 2, 7, 9, 12	—
5P-5	• 1., 7, 11, 12	—
5R-5	• 4, 8, 12	—
5R-7	• 4, 6, 7, 10, 11	—
5P-2	• 9, 11, 12	—
5R-2	• 9, 11, 12	—
5P-4	• 5, 10	• 12
5R-4	• 2, 5, 10	• 12
5P-6	• 11	• 1
5R-6	• 11	—
5R-8	• 4, 5, 6, 7, 8	—
5P-1	—	• 12

REV	DATE	DESCRIPTION	BY	CHK	APP
29	7-31 95	SPARED CONTACT NO. 12, RELAYS 5P-1, 5R-1, 5P-5, 5R-5, 5P-4 & 5R-4 AND CONTACT NO. 11 RELAYS 5P-6 & 5R-6 PER FC-949 & DCR 95-608 REMOVED EVENTS RECORDER, ADDED PPC-SOE NODE PER FC-933-06 AND DCR 95-520 (FCM)	DMB		ef lcm

CONSUMERS POWER COMPANY
PALISADES NUCLEAR PLANT
COVERT, MICHIGAN
E208-2.DGN

PREVIOUS RECORD ISSUE

MICHIGAN P.E. No.

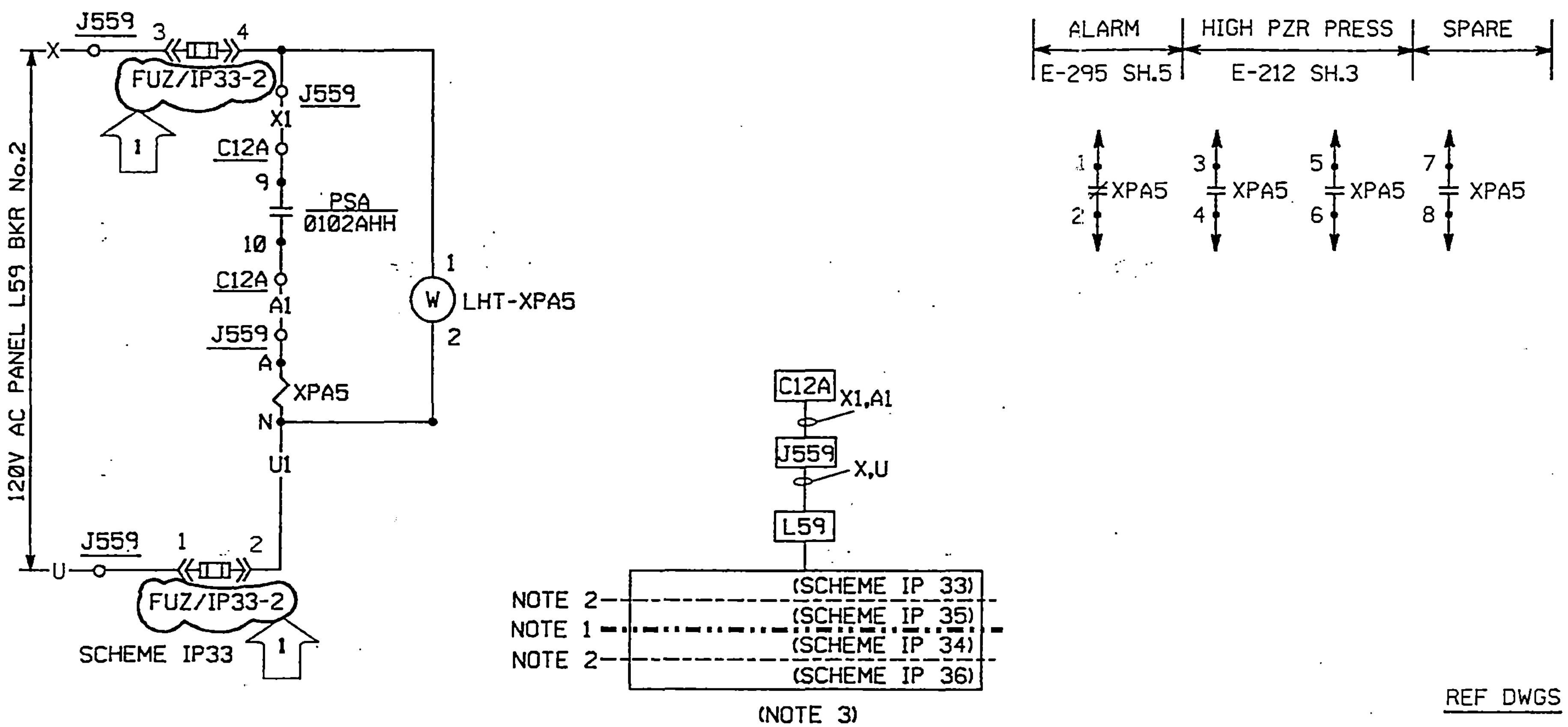
UFI No.

SCHEMATIC DIAGRAM
CONTAINMENT ISOLATION ON
HIGH PRESSURE OR
HIGH RADIATION

DRAWING NUMBER	SHT. NO.	REV
E208	2	29

Figure 7-29
Rev 19

9707070091 -14



NOTES

1. SEPARATE RACEWAY SYSTEM.
2. BARRIER BETWEEN CHANNELS WITHIN RACEWAY.
3. LEFT AND RIGHT CHANNEL SEPARATION IS PROVIDED FROM C12 TO J558 & C12 TO J559.
4. SEE SH.2B FOR CONTACT DEVELOPMENT.

REF DWGS

E84 SH 2 & 2B
E121 SH1A
E212 SH 3
E615 SH 1
E295

TRIP RELAY

ADAPTER TABLE

PRESSURIZER PRESSURE ATWS CIRCUITS	SCHEME	AC POWER SUPPLY		FUSE	ATWS MATRIX RELAY		VOLTAGE AVAIL. LIGHT		INDICATOR ALARM UNIT		FUSES
		PNL	BKR		ID	LOC	ID	LOC	ID	LOC	ID
CIRCUIT No. 1	IP 33	L59	2	FUA	XPA5	J559	LHT-XPA5	J559	PSA-0102AHH	C12-A	FUZ/IP33-2
CIRCUIT No. 2	IP 34	L58	8	FUB	XPB5	J558	LHT-XPB5	J558	PSA-0102BHH	C12-B	FUZ/IP34-2
CIRCUIT No. 3	IP 35	L59	3	FUC	XPC5	J559	LHT-XPC5	J559	PSA-0102CHH	C12-C	FUZ/IP35-2
CIRCUIT No. 4	IP 36	L58	9	FUD	XPD5	J558	LHT-XPD5	J558	PSA-0102DHH	C12-D	FUZ/IP36-2

REV	DATE	DESCRIPTION	BY	APP
1	11-4 93	ADDED FUSE ID #'S AND REVISED ADAPTER TABLE PER DCR 93-1299	JFM	JB

CONSUMERS POWER COMPANY
PALISADES PLANT
COVERT, MICHIGAN

E84-2A.DGN

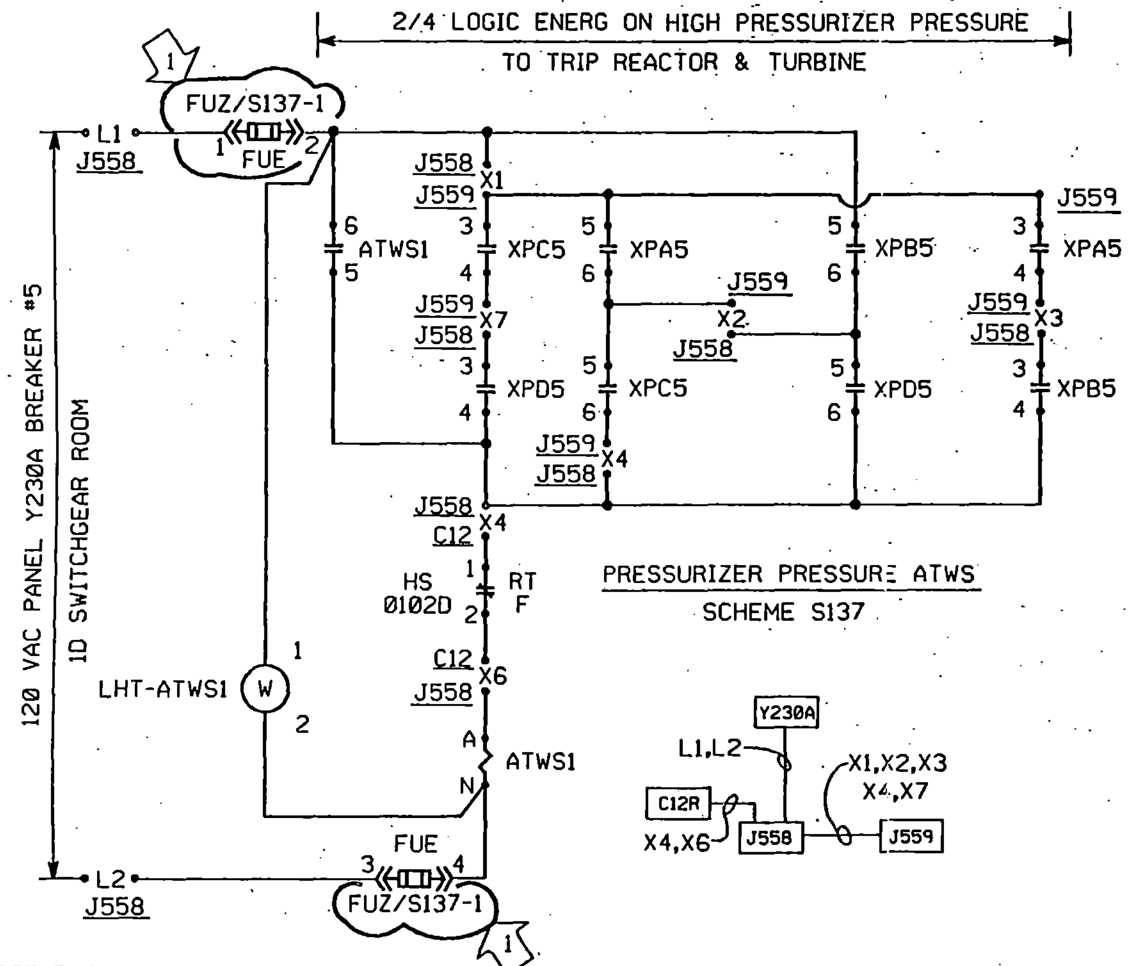
PREVIOUS REC ISSUE
MICHIGAN P.E. NO.
UFI NO.
SYSTEM: 02531
EQUIPMENT: ——

SCHEMATIC DIAGRAM
PRESSURIZER PRESSURE ATWS

DRAWING NO.	SHEET	REV
E-84	2A	1

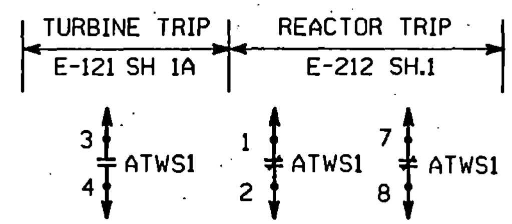
9707070091 -15

Figure 7-29A Sh 1
Rev 19



REF DWGS
E84 SH 2
E84 SH 2A
E121 SH 1A
E615 SH 1
E618 SH 558

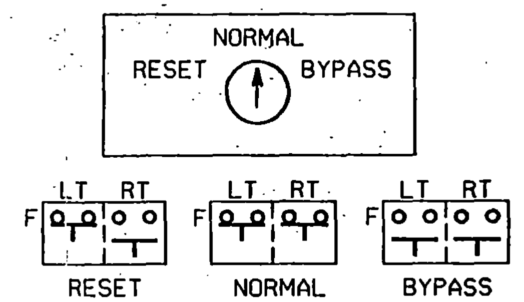
ATWS1 CONTACT	SCHEME	REF DWG	N.O.	N.C.	FUNCTION
1-2	S-65	E-212 SH 1		X	RX TRIP
3-4	T-28	E-121 SH 1A	X		TURB TRIP
5-6	S-137	E-212 SH 3	X		SEAL-IN
7-8	S-65	E-212 SH 1		X	RX TRIP



SWITCH DEVELOPMENT

CONTACTS		SWITCH POSITION RESET	NORM	BYPASS	FUNCTION (REF DWG)
LT F	1-2	X	X	0	BYPASS ALARM E295 SH.5
RT F	1-2	0	X	0	ATWS 1 RESET BYPASS THIS DWG

HS0102D



KEY OPERATED SELECTOR SWITCH HS 0102D
TYPE G.E. CR104PSK76A11C,
WITH MODIFIED CAM
3 POSITION SPRING RETURN
FROM LEFT. KEY REMOVABLE FROM CENTER
POSITION. VIEWED FROM FRONT OF PANEL

REV	DATE	DESCRIPTION	BY	APP
1	12-6 93	ADDED FUSE ID #'S PER DCR 93-1344	HMW	JOS

CONSUMERS POWER COMPANY
PALISADES PLANT
COVERT, MICHIGAN

E212-3.DGN

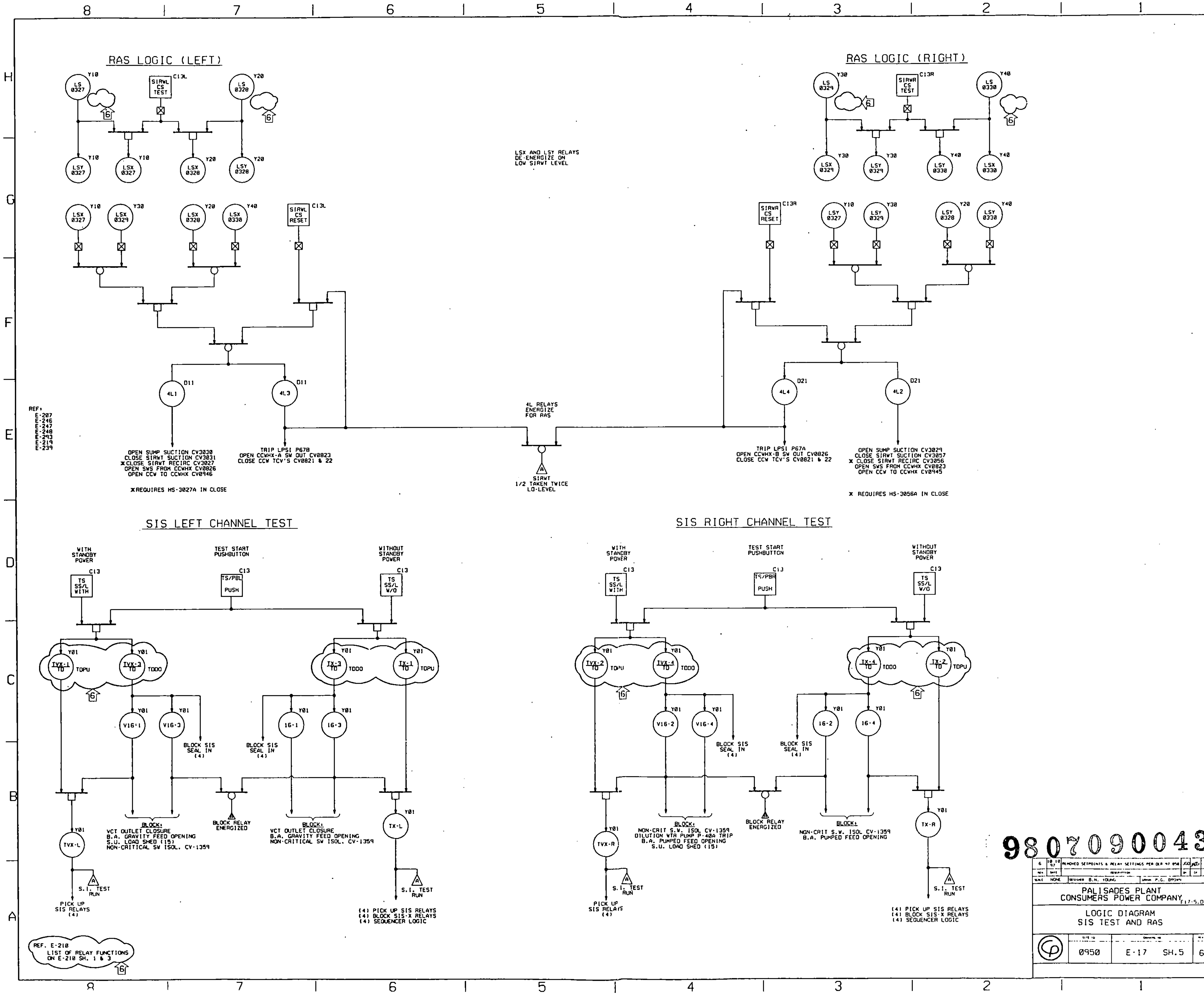
PREVIOUS REC ISSUE
MICHIGAN P.E. NO.

SCHEMATIC DIAGRAM
PRESSURIZER PRESSURE ATWS

UFI NO.	DRAWING NO.	SEQ	SHEET	REV
SYSTEM: 02531 EQUIPMENT: —	E-212	—	3	1

Figure 7-29 Sh 2 FSAR Rev 17

9501040188-31



ANSTEC APERTURE CARD

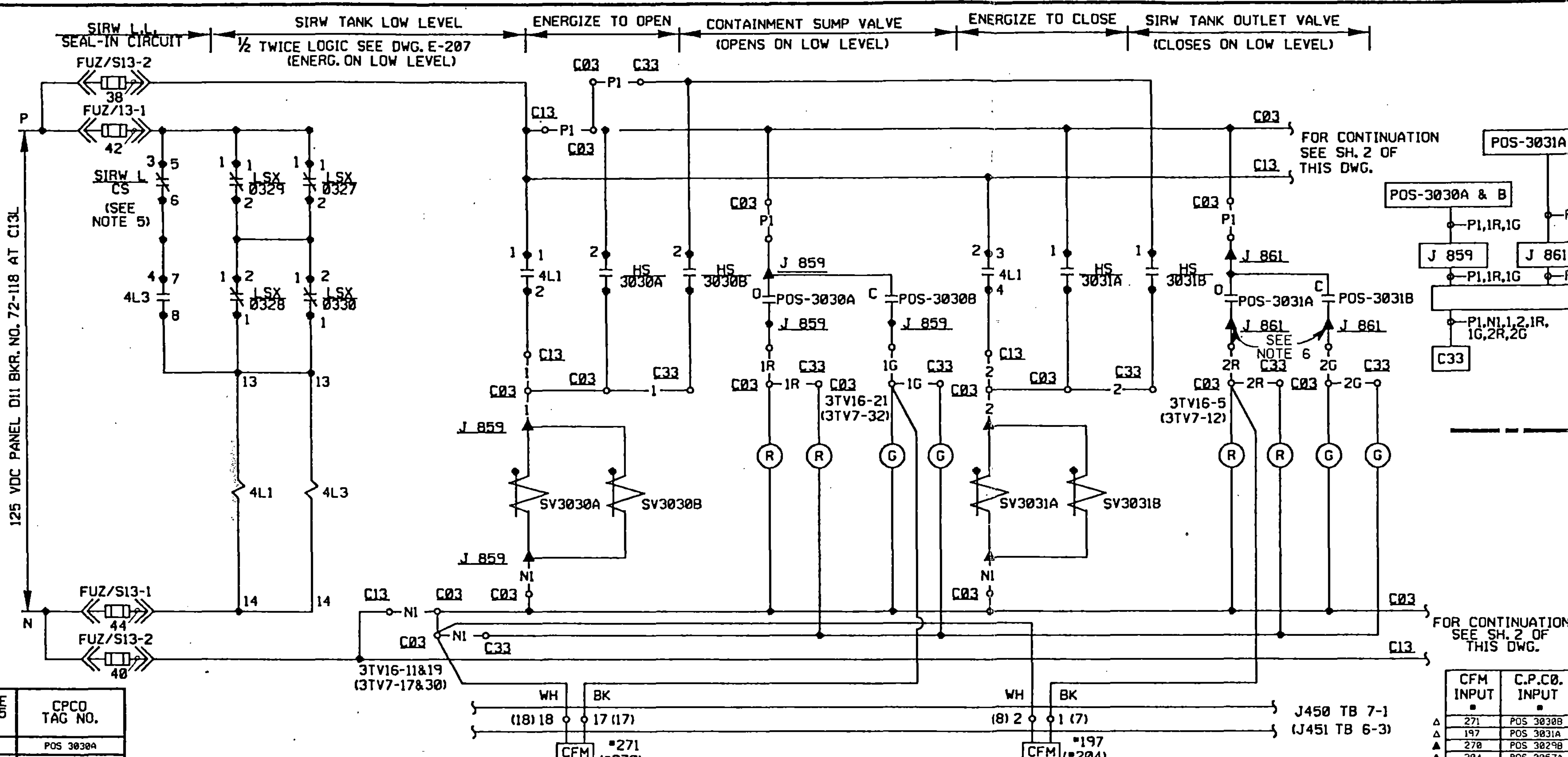
Also Available on Aperture Card

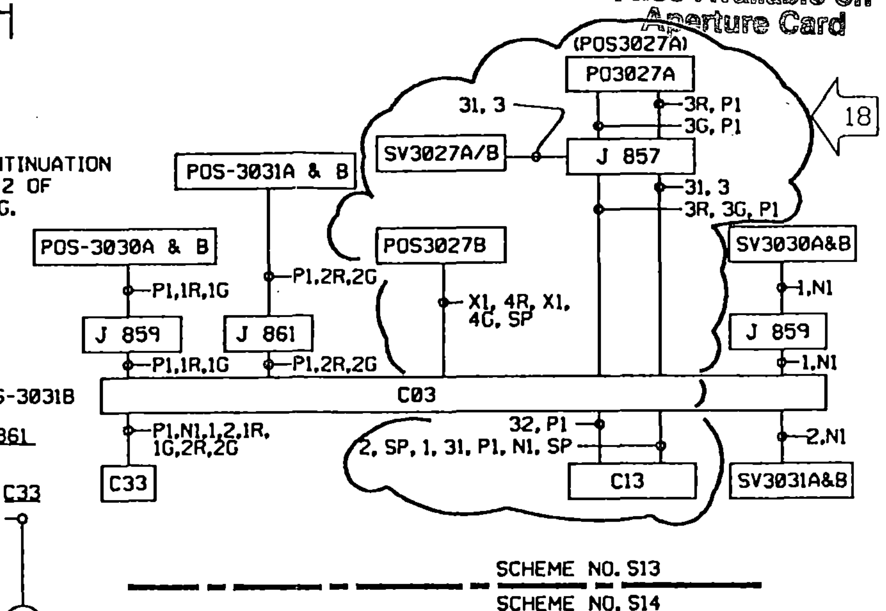
9807090043-18

FIG 7-30
FSAR REV 20

ANSTEC APERTURE CARD

Also Available on Aperture Card





SCHEME NO. S13

SCHEME NO. S14

NOTES:

1. SV3030B/3031A-ENERGIZE TO OPEN A/S VALVE
   SV3030A/3031B-ENERGIZE TO OPEN VENT VALVE.
2. FOR LSX RELAYS, SEE DWG. NO. E-207
3. INFO RELATED TO INPUT NO.'S *270 & *204 IS IN PARENTHESIS ( )
4. ▲ = RAYCHEM SPLICE.
5. SEAL-IN RESET CONTACT. SEE DWG E-207 SH.1 FOR SWITCH DEVELOPMENT.
6. RAYCHEM SPLICE FOR POS-3031A/B ONLY (SCHEME S13)

	CFM INPUT ■	C.P.CO. INPUT ■
△	271	POS 3030B
△	197	POS 3031A
▲	270	POS 3029B
▲	204	POS 3057A

△=J450

▲=J451

PPC-SOE NODE DWG. E53 SH.5 PT. ID.	CPCO TAG NO.
ZHC3030_D	POS 3030A
ZHC3029_D	POS 3029A
ZHC3031_D	POS 3031B
ZHC3057_D	POS 3057B

SIRW TANK LEVEL CONTROLLED VALVES-CIRCUIT NO.1

SCHEME NO. S13

ONE OF TWO IDENTICAL CIRCUITS

ADAPTER TABLE

CKT. NO.	SCHEME	LOC. NO.S			CONTAINMENT SUMP VALVE		SIRW TANK OUTLET VALVE		SIRW TANK MIN. RECIRC. VALVE		SIRW TANK L. LEVEL SEAL-IN		D.C. PNL. & LOCATION	D.C BREAKER	CONTROL POWER UNDERVOLT	RELAY CONTACTS								SIRW TANK L. LEVEL RELAYS	SIRWT LO-LEVEL BYPASS H.S.	FUSE NO.
					HS. SV. & POS	INTLK.	HS. SV. & POS	INTLK.	HS. SV. & POS	INTLK.						LSX 0327	LSY 0327	LSX 0328	LSY 0328	LSX 0329	LSY 0329	LSX 0330	LSY 0330			
1	S13	C03L	C13L	C33L	3030A & B	4L1 1	3031A & B	4L1 2	3027A & B	4L1 3	SIRW L CS-3	4L3 4	D11	72-118	74-S13	1		1		1		1		4L1 & 4L3	HS-3027B	FUZ/S13-1 FUZ/S13-2
2	S14	C03R	C13R	C33R	3029A & B	4L2 1	3057A & B	4L2 2	3056A & B	4L2 3	SIRW R CS-3	4L4 4	D21	72-218	74-S14		1		1		1		1	4L2 & 4L4	HS-3056B	FUZ/S14-1 FUZ/S14-2

REV.	DATE	DESCRIPTION	BY	CK	APP
18	6-13-97	REVISED SCHEME NO. S13 PER C-PAL-97-0417 PER DCR 97-0566	DMB		

SCALE | DESIGNED | REDRAWN J.L.STERRETT

Consumers Energy

PALISADES PLANT

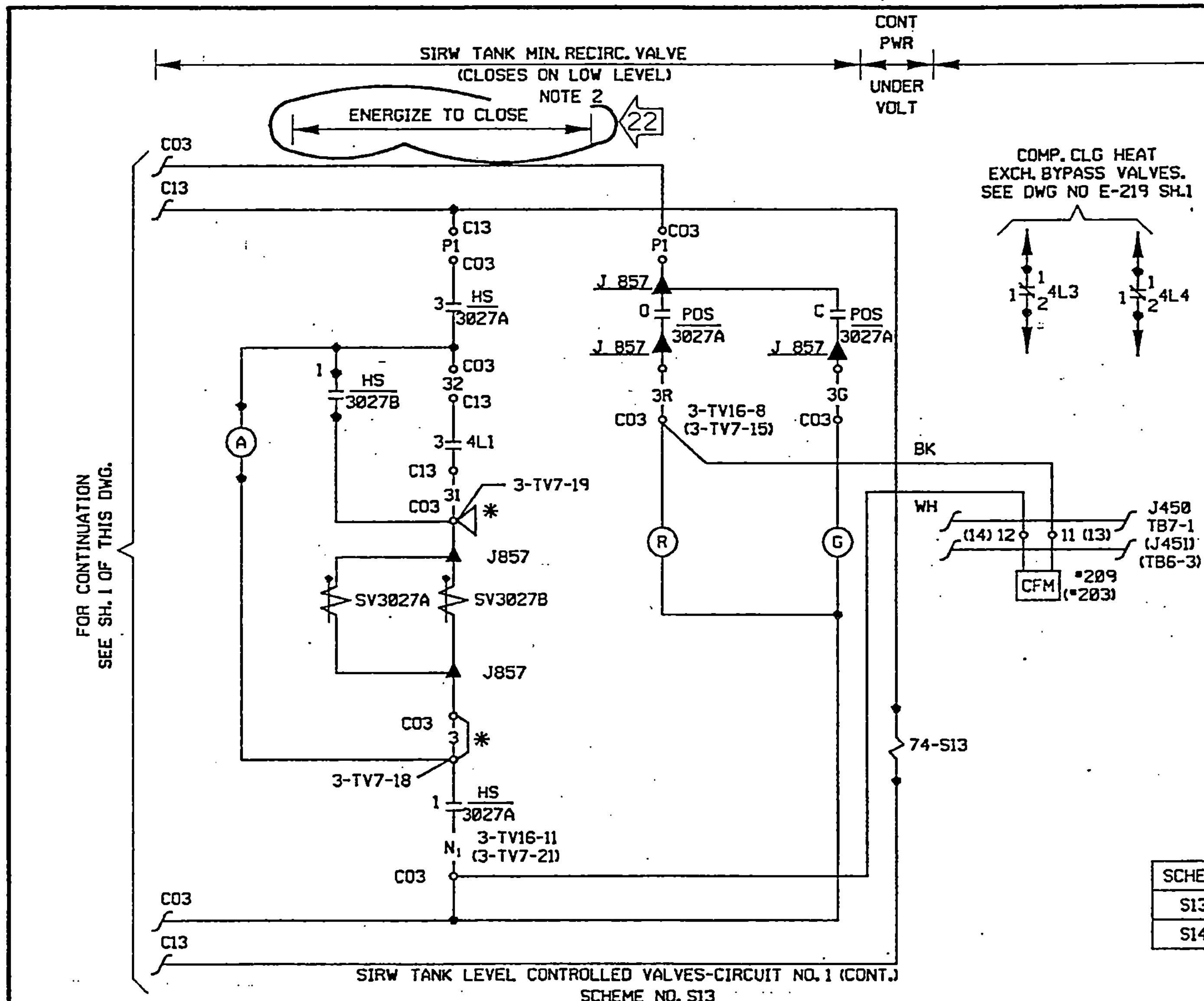
E246-1.DGN

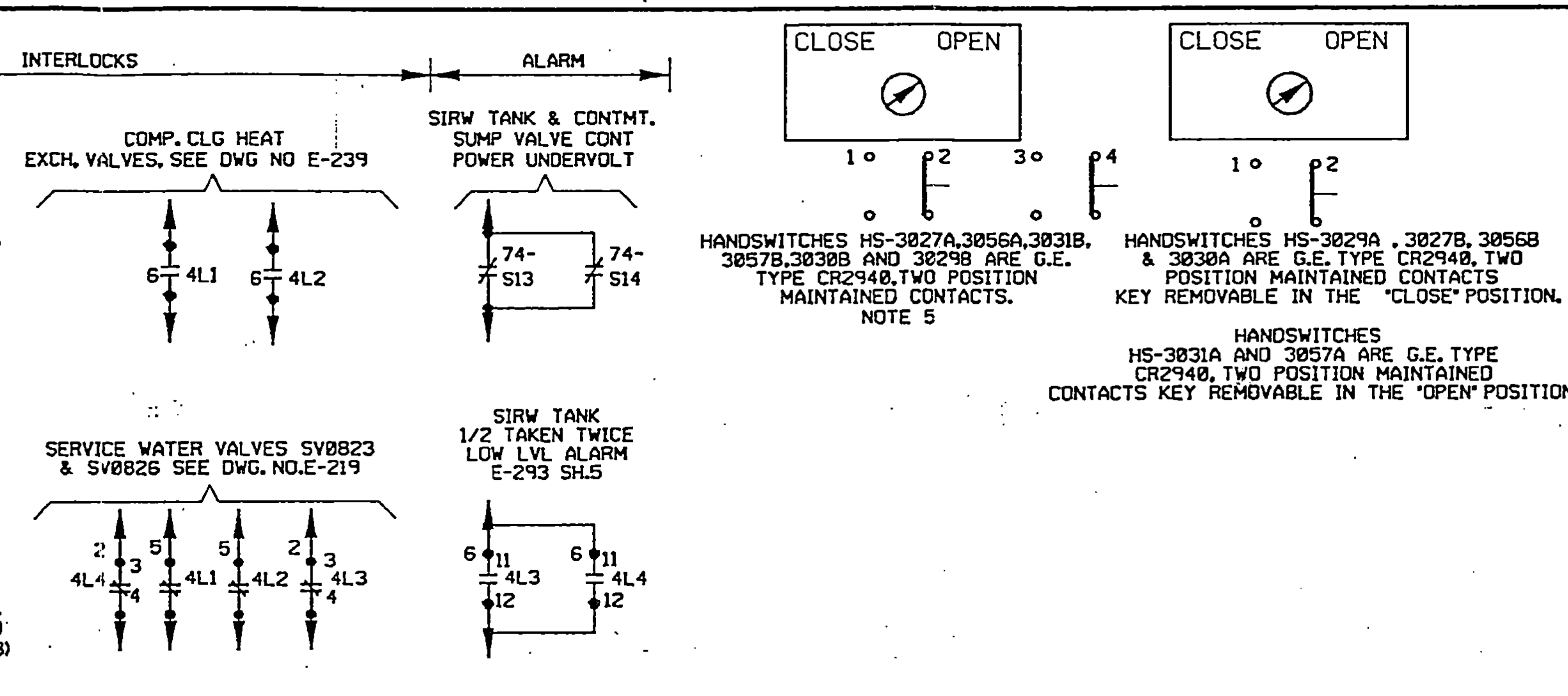
SCHEMATIC DIAGRAM
SIRW TANK &
CONTAINMENT SUMP VALVES

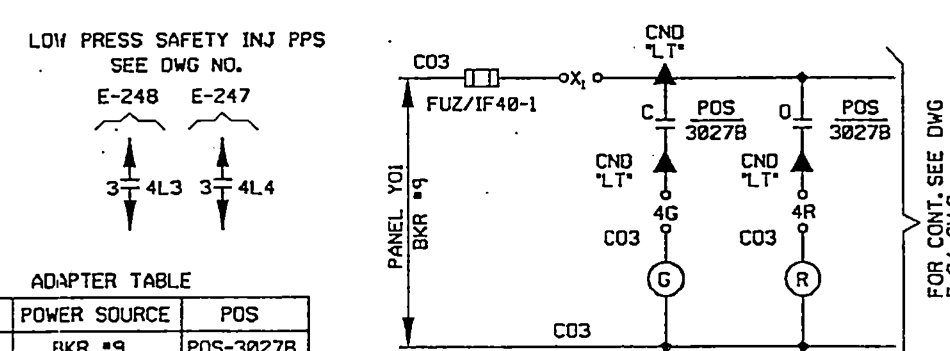
JOB NO.	DRAWING NO.	REV.
5935	E-246 SH.1	18

FIG 7-31
FSAR REV 20

9807090043 -19







ADAPTER TABLE

SCHEME	POWER SOURCE	POS
S13	BKR #9	POS-3027B
S14	BKR #10	POS-3056B

△=J450
▲=J451

	CFM INPUT NO.	C.P.CO. TAG NO.
△	209	POS-3027 A
▲	203	POS-3056 A

NOTE
1. SV-3027A- ENERGIZE TO OPEN A/S VALVE
SV-3027B-ENERGIZE TO OPEN VENT VALVE 22
2. REQUIRES OPERATOR ACTION ON H.S.
3. INFO. RELATED TO INPUT #203 ARE IN PARENTHESIS ( )
4. ▲ = RAYCHEM SPLICE.
5. HS-3027A AND 3056A HAVE A DOUBLE CONTACT BLOCK.
6. ✳ = INDICATES JUMPERED BROKEN LINK

ADAPTER TABLE

SCHEME	CONTAINMENT SUMP VALVE						SIRW. TANK OUTLET VALVE						SIRW. TANK MINI RECIRC. VALVE					
	SV.			POS.			SV			POS			SV.			POS.		
	SV	FIELD CONN	REF DWG	POS	FIELD CONN	REF DWG	SV	FIELD CONN	REF DWG	POS	FIELD CONN	REF DWG	SV	FIELD CONN	REF DWG	POS	FIELD CONN	REF DWG
S13	3030	J 859	E-618 SH.859	3030	J 859	E-618 SH.859	3031	CONDUIT "LT"	E-635 SH.5	3031	J 861	E-618 SH.861	3027	J857	E-635 SH.6	3027 A	J 857	E-618 SH.857
S14	3029	CONDUIT "LT"	E-635 SH.4	3029	J 804	E-618 SH.804	3057	J 852	E-618 SH.852	3057	J 852	E-618 SH.852	3056	J 850	E-618 SH.850	3056 A	J 850	E-618 SH.850

REV	DATE	DESCRIPTION	BY	CK	APP
22	1-29-97	REV'D. NOTE 1 PER DCR-97-068.	JGO	MDS	

SCALE | DESIGNED | REDRAWN R.M.SATTERELLI

PALISADES PLANT
CONSUMERS POWER COMPANY

SCHEMATIC DIAGRAM
SIRW TANK &
CONTAINMENT SUMP VALVES

E246-2.DGN

JOB NO.	DRAWING NO.	REV.
0950	E-246 SH.2	22

9707070091 -17

Figure 7-32
Rev 19

FUNCTION	SYMBOL INPUT OUTPUT	MEANING
AND		OUTPUT EXISTS WHEN ALL INPUTS EXIST.
OR		OUTPUT EXISTS WHEN ANY INPUT EXISTS.
COINCIDENCE MATRIX	A/B	OUTPUT EXISTS WHEN AT LEAST A OUT OF B INPUTS ARE PRESENT.
MEMORY	M R	OUTPUT EXISTS WHEN MEMORY INPUT IS APPLIED AND IS RETAINED UNTIL RESET INPUT IS APPLIED. M AND R DENOTE MEMORY AND RESET RESPECTIVELY.
NOT		OUTPUT EXISTS WHEN INPUT DOES NOT EXIST.
ON TIME DELAY		OUTPUT EXISTS FOLLOWING A TIME DELAY AFTER THE INPUT IS CONTINUOUSLY APPLIED. OUTPUT CEASES WHEN THE INPUT IS NOT PRESENT.
OFF TIME DELAY		OUTPUT EXISTS WHEN THE INPUT IS PRESENT AND CONTINUES TO EXIST FOR A TIME AFTER THE INPUT CEASES.

FUNCTION	SYMBOL INPUT OUTPUT	MEANING
CALIBRATION	CAL	CALIBRATING SET POINT INPUT TO PROVIDE A CALIBRATED SIGNAL.
LOW BISTABLE	L SP	DIGITAL OUTPUT EXISTS ONLY WHEN ANALOG INPUT IS LOWER THAN SET POINT.
HIGH BISTABLE	H SP	DIGITAL OUTPUT EXISTS ONLY WHEN ANALOG INPUT IS HIGHER THAN SET POINT.
TEST DEVICE	T TEST	TEST SIGNAL CAN BE INSERTED AUTOMATICALLY IN PLACE OF NORMAL SIGNAL.
ISOLATION	ISO	OUTPUT IS ELECTRICALLY ISOLATED FROM INPUT.
STATUS ARRAY INDICATING LIGHT	A	INDICATES EQUIPMENT STATUS

(REF JLG-102 SH.1 OF 3 REV.2)

CONSUMERS POWER COMPANY
PALISADES PLANT
FSAR UPDATE

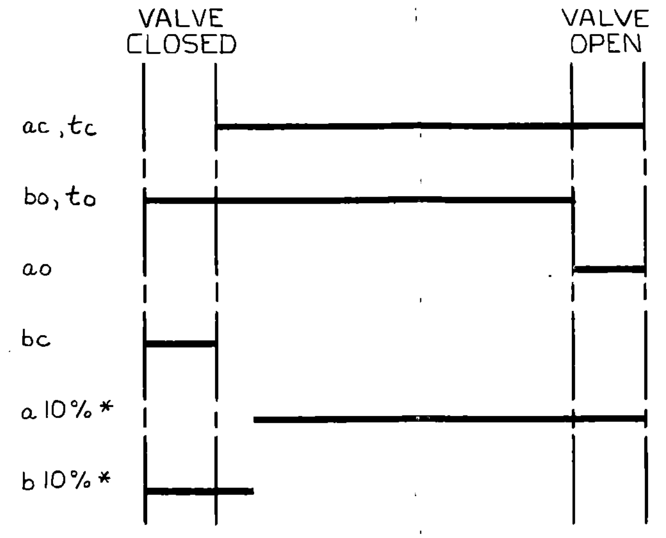
LOGIC DIAGRAM
LEGEND AND NOTES

FIGURE NO 7-33

REVISION NO 0

FUNCTION	SYMBOL (INPUT / OUTPUT)	MEANING
MANUAL INPUT		MOMENTARY HAND SWITCH INPUT TO LOGIC.
MANUAL INPUT		MAINTAINED HAND SWITCH INPUT TO LOGIC.
LOGIC CONTINUATION		DRAWING NUMBER CROSS REFERENCE, INDICATES SOURCE OF LOGIC INPUT OR DESTINATION OF LOGIC OUTPUT.
ALARM	(PANEL) (SCHEME/WINDOW)	ANNUNCIATOR WINDOW-WITH PANEL, SCHEME/WINDOW NUMBERS.
DATA LOGGER OR COMPUTER POINT	I-XXX	INPUT TO DATA LOGGER OR COMPUTER WITH IDENTIFYING NUMBER.
INDICATING LAMP	R	INDICATES EQUIPMENT OPERATION STATUS. R- RED: OPERATING, FLOWING OR INCREASING. G- GREEN: NOT OPERATING, NOT FLOWING OR DECREASING. W- WHITE: CONDITION NORMAL, PERMISSIVE, CONDITION REACHED OR CONTROL POWER AVAILABLE. A- AMBER: BREAKER TRIPPED, REMOTE OVERLOAD, ABNORMAL CONDITION OR STANDBY STATUS. B- BLUE: MOTOR THERMAL OVERLOAD
LOGIC OUTPUT ACTION	DEVICE CONTROLLED	INDICATES STATE OF CONTROLLED DEVICES GIVEN THE EXISTENCE OF THE LOGIC INPUT.

POSITION SWITCH DEVELOPMENT



INDICATES CLOSED CONTACT.

tc A NORMALLY CLOSED CONTACT THAT OPENS ON HIGH TORQUE IN THE CLOSING DIRECTION OF A MOTOR OPERATED VALVE.

to A NORMALLY CLOSED CONTACT THAT OPENS ON HIGH TORQUE IN THE OPENING DIRECTION OF A MOTOR OPERATED VALVE.

* % MAY BE ADJUSTED AS REQUIRED.

TI APERTURE CARD

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(REF JLG-102 SH.2 OF 3 REV.2)

CONSUMERS POWER COMPANY
PALISADES PLANT
FSAR UPDATE

LOGIC DIAGRAM
LEGEND AND NOTES

FIGURE NO 7-34

REVISION NO 0

ABBREVIATIONS:

- – CONTROL ROOM
- – LOCAL CONTROL PANEL
- – CONTROL SWITCH
- – HAND SWITCH
- – PUSH BUTTON
- CONTAINMENT HIGH PRESSURE

SR TO N – SPRING RETURN TO NORMAL
CHR CONTAINMENT HIGH RADIATION
AFAS – AUXILIARY FEEDWATER ACTUATION SYSTEM
FOGG A – FEED ONLY GOOD STEAM GENERATOR A ISOLATE STEAM GENERATOR B
FOGG B – FEED ONLY GOOD STEAM GENERATOR B ISOLATE STEAM GENERATOR A
SIS – SAFETY INJECTION SIGNAL
HVAC – HEATING, VENTILATION & AIR CONDITIONING
SS – SELECTOR SWITCH
POS – POSITION SWITCH
SV – SOLENOID VALVE
CV – CONTROL VALVE
MO – MOTOR OPERATED
PO – PNEUMATICALLY OPERATED
SP – SET POINT

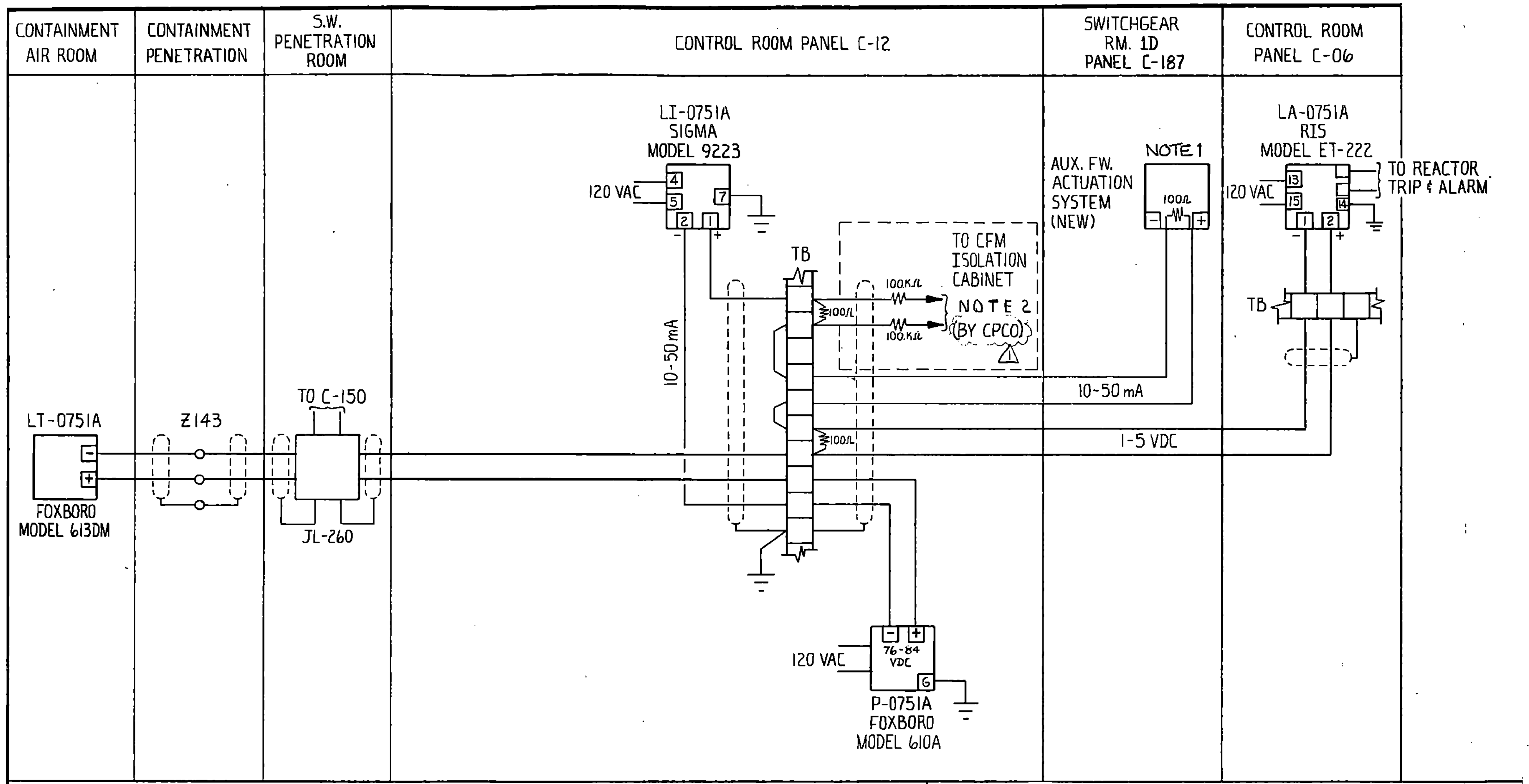
NOTES:

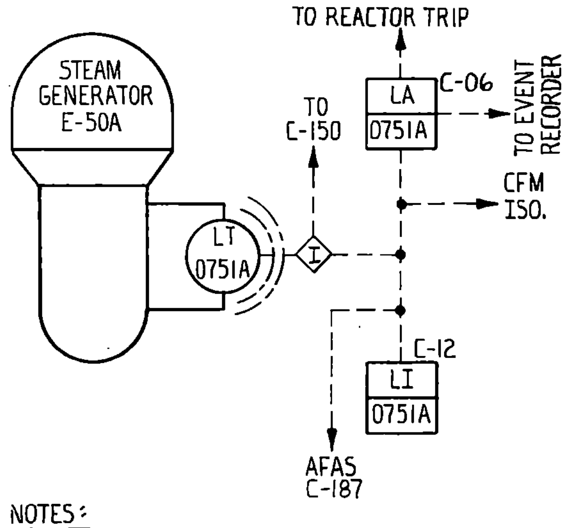
1. THESE LOGIC DIAGRAMS DESCRIBE EQUIPMENT/SYSTEM FUNCTIONS AND DO NOT NECESSARILY REFLECT THE ACTUAL HARDWARE IMPLEMENTATION.
2. THE STATE OF EQUIPMENT WILL NOT BE CHANGED BY A TRANSIENT OR LOSS OF POWER UNLESS OTHERWISE NOTED.
3. INHERENT EQUIPMENT INTERLOCKS, SUCH AS CIRCUIT BREAKER TRIP FREE AND REVERSING STARTER CROSS INTERLOCKS ARE NOT SHOWN UNLESS THE INTERLOCKS PERFORM A LOGICAL FUNCTION.
4. ANY SET POINTS SHOWN ON CONTROL LOGIC DIAGRAMS ARE APPROXIMATE. FOR EXACT VALUE REFER TO INSTRUMENT INDEX. 1
5. VALVE AND DAMPER RED AND GREEN LIGHTS ARE BOTH ON WHEN IN AN INTERMEDIATE POSITION.
6. REFER TO ELECTRICAL SCHEMATICS FOR DETAILS OF ELECTRICAL EQUIPMENT OVERCURRENT, SHORT CIRCUIT, AND DIFFERENTIAL PROTECTION.
7. EQUIPMENT WILL CHANGE STATE WHEN A CHANGE IS INITIATED AND WILL REMAIN IN THAT STATE UNTIL A CHANGE TO ANOTHER STATE IS INITIATED.

(REF JLG-102 SH.3 OF 3 REV.2)

CONSUMERS POWER COMPANY PALISADES PLANT FSAR UPDATE
LOGIC DIAGRAM LEGEND AND NOTES
FIGURE NO 7-35

REVISION NO 0





NOTES:

1. THE FOLLOWING ARE THE MODIFICATIONS TO THE EXISTING LOOPS:
   a. DISCONNECT EXISTING LS-0751A-D AND LS-0752A-D LOCATED IN JUNCTION BOXES J-578A-D.
   b. ADD 10-50mA INPUT TO THE AFAS CABINET C-187.
2. OUTPUT PROVIDED FOR LOOP LT-0751A AND LT-0752A ONLY.

REFERENCES:

1. P&ID: M- 207
2. ELECTRICAL SCHEMATIC: E- 78(Q)
3. VP: 12447/059-J447(Q)-31 THRU 34

ADAPTER TABLE

LT	LI	LA	POWER SUPPLY	PREFERRED AC	PENETRATION CANNISTER	STEAM GENERATOR	JUNCTION BOX	NOTES
LT-0751A	LI-0751A	LA-0751A	L-0751A	Y10	Z143	E-50A	JL-260	2
LT-0751B	LI-0751B	LA-0751B	L-0751B	Y20	Z243			
LT-0751C	LI-0751C	LA-0751C	L-0751C	Y30	Z146			
LT-0751D	LI-0751D	LA-0751D	L-0751D	Y40	Z246			
LT-0752A	LI-0752A	LA-0752A	L-0752A	Y10	Z143	E-50B	JL-261	2
LT-0752B	LI-0752B	LA-0752B	L-0752B	Y20	Z243			
LT-0752C	LI-0752C	LA-0752C	L-0752C	Y30	Z146			
LT-0752D	LI-0752D	LA-0752D	L-0752D	Y40	Z246			

THE INTENT OF THIS DRAWING IS TO DEPICT ONLY THE CHANGES REQUIRED FOR THIS MODIFICATION (SUBJOB 059). REFER TO SCHEMATIC DIAGRAM E-9078(Q). FOR COMPLETE SYSTEM CONFIGURATION.

TI APERTURE CARD

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(REF JLP-65 REV.1)

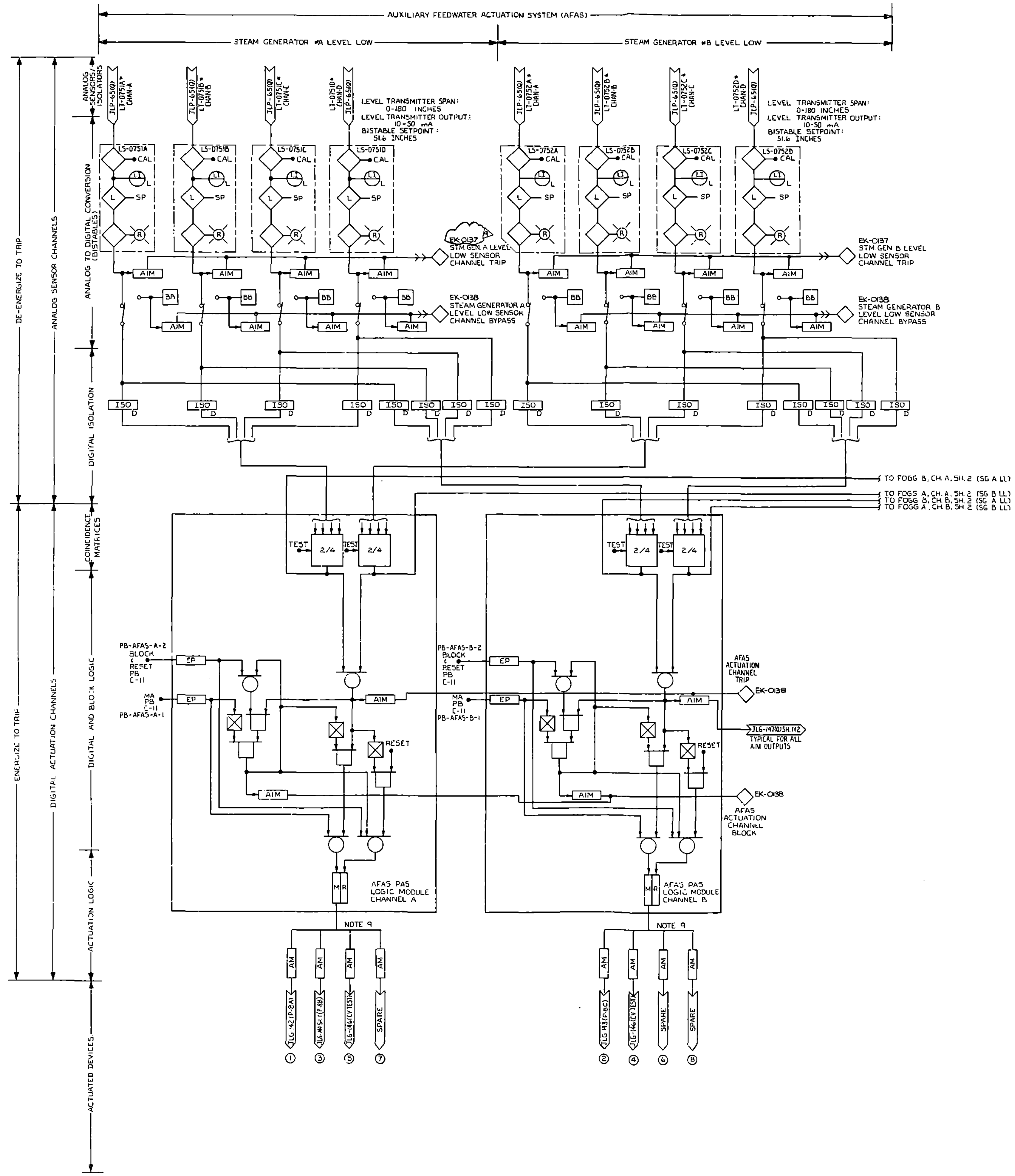
CONSUMERS POWER COMPANY
PALISADES PLANT
FSAR UPDATE

LOOP DIAGRAM
STEAM GENERATOR LEVEL CONTROL

FIGURE NO 7-36

REVISION NO 0

JLG-150 SH. 1



NOTES:

1. LEGEND (SEE LOGIC DIAGRAM JLG-102) FOR COMPLETE LEGEND OF LOGIC SYMBOLS.

FUNCTION	SYMBOL	MEANING
DIFFERENTIAL BISTABLE		DIGITAL OUTPUT EXISTS ONLY WHEN ANALOG INPUTS DIFFERENTIAL EXCEEDS SETPOINT.
CALIBRATION	CAL	CALIBRATION SET POINT INPUT TO PROVIDE A CALIBRATING SIGNAL
LOW BISTABLE	S.P.	DIGITAL TRIP OUTPUT EXISTS ONLY WHEN ANALOG INPUT IS LOWER THAN SET POINT
HIGH BISTABLE	S.P.	DIGITAL TRIP OUTPUT EXISTS ONLY WHEN ANALOG INPUT IS HIGHER THAN SET POINT
		TRIP MEMORY LIGHT
PROTECTION	EP	ELECTRICAL (FAULT) PROTECTION DEVICES
ISOLATION	ISO D	DIGITAL ISOLATION DEVICE
	ISO A	ANALOG ISOLATION DEVICE

ABBREVIATIONS:

- A – DRIVER AMPLIFIER
- AIM – ALARM INDICATING MODULE (OUTPUTS ARE NON CLASS IE)
- AM – ACTUATION MODULE
- BB – BYSTABLE BYPASS
- L – LOCAL
- LT – LEVEL TRANSMITTER
- M – MEMORY
- MA – MANUAL ACTUATION
- PAS – PROTECTIVE ACTUATION SUBSYSTEM
- PT – PRESSURE TRANSMITTER
- PI – PRESSURE INDICATOR
- R – RESET
- RLY – OUTPUT RELAY
- * – INDICATES EQUIPMENT SUPPLIED BY OTHER VENDORS.
- LI – LEVEL INDICATOR

2. AUTOMATIC TEST CIRCUITRY IS NOT SHOWN ON THIS LOGIC DIAGRAM. REFER TO VENDOR DRAWING 12447/059-J-447-43 FOR DETAILS OF AUTOMATIC TEST FUNCTION.
3. ALL SENSOR CHANNELS ARE DE-ENERGIZE-TO-TRIP.
4.
5. DIGITAL ACTUATION CHANNELS ALSO ARE PROVIDED WITH AN ELECTRONIC BLOCKING FUNCTION TO ALLOW TESTING OF COINCIDENCE LOGIC, AS SHOWN ON VENDOR DRAWING 12447/059-J-447-43
6. RELAY DRIVER OUTPUTS FROM DIGITAL ACTUATION CHANNELS MAY DRIVE MORE THAN 1 OUTPUT RELAY. TEST SWITCHES SHOWN ON ACTUATION MODULES (SHEET 2) MAY TEST MORE THAN ONE OUTPUT RELAY.
7. THIS DRAWING DESCRIBES THE FUNCTIONAL REQUIREMENTS OF THE AFAS AND IS NOT INTENDED TO NECESSARILY REFLECT THE ACTUAL HARDWARE DESIGN.
8. AFAS - INITIATES AUXILIARY FEEDWATER PUMPS IN THEIR RESPECTIVE SEQUENCE.
   FOGG A - ISOLATES STEAM GENERATOR B (ACTUATED BY SG A PRESS. > SG B PRESS. AND SG B LOW LEVEL)
   FOGG B - ISOLATES STEAM GENERATOR A (ACTUATED BY SG B PRESS. > SG A PRESS. AND SG A LOW LEVEL)
9. FINAL OUTPUT FROM PROTECTIVE SYSTEM ACTUATION MODULES IS NORMALLY "DE-ENERGIZED" AND CHANGE TO "ENERGIZED" FOR ACTUATION.
10. THE CONTENTS OF THIS LOGIC DIAGRAM EXCEPT AS NOTED BY AN ASTERISK (*) IS SUPPLIED BY:
    AUTOMATION INDUSTRIES, INC.
    VITRO LABORATORIES DIVISION
    MR 12447-059 J-447(Q)
    CABINET C-187
11. AFAS-FOGG ANNUNCIATOR WINDOW ENGRAVINGS ARE AS FOLLOWS:
    WINDOW No. 39 - AFAS-FOGG SUBSYSTEM TRIP
    WINDOW No. 40 - AFAS-FOGG SUBSYSTEM BLOCKED/BYPASSED
12. REFERENCE VENDOR DRAWINGS 12447/059-J-447(Q)-16
    12447/059-J-447(Q)-17
    12447/059-J-447(Q)-31
    12447/059-J-447(Q)-32
    12447/059-J-447(Q)-33
    12447/059-J-447(Q)-34
    FOR EXTERNAL CONNECTIONS INTERFACE WITH AFAS AND FOGG SYSTEMS.
13. AFAS-FOGG BISTABLE MODULES ARE IDENTIFIED AS LS-0751A AND DPS-0751A, ETC. FOR CALIBRATION PURPOSES ONLY AND ARE NOT INTENDED TO DEPICT DISCRETE INSTRUMENTATION OR DEVICES.

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APPENDIX N, SPECIFICATION 12447/059-J-447(Q)

No.	DATE	REVISIONS	BY	CHK	GROUP SUPV.	PROJ ENGR	MICROFILM DATE
8	9/12/90	REVISED PER DCR 950-90-0056 CCP (DEK)			JGD		
7	2/23/84	REVISED AS IND., ISSUED FOR RECORD					
6	11/10/83	REVISED AS INDICATED					
9	8/17/90	REV. EK-0139 TO EK-0137 PER 950-90-0056C CCP (DEK)			BAB		
1	6/25/82	ISSUED FOR USE					
0		ISSUED FOR FIDS					

SCALE NTS | DESIGNED D UGORCAK | DRAWN MCWOODS

BECHTEL
ANN ARBOR, MICHIGAN

PALISADES PLANT
CONSUMERS POWER COMPANY
GWO-8343

LOGIC DIAGRAM
AUXILIARY FEEDWATER ACTUATION SYSTEM

JOB No.	DRAWING No.	REV.
12447-059	JLG-150(Q) SH. 1	9

AUXILIARY FEEDWATER PHASE II SYS 03400 EQ 03430

SIZE E

Figure No. 7-37 FSAR Rev. 11

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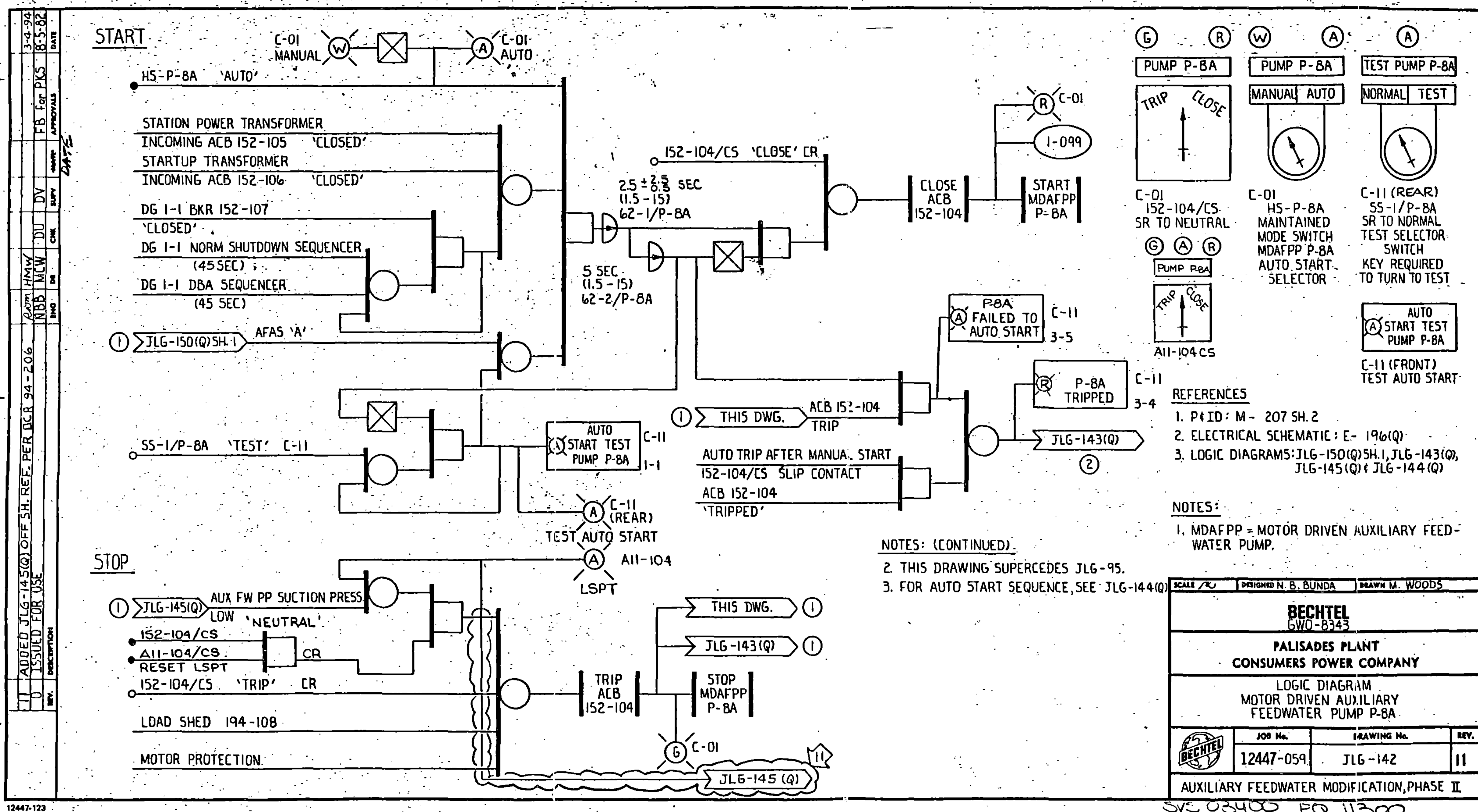

Figure 7-38 FSAR Rev 17

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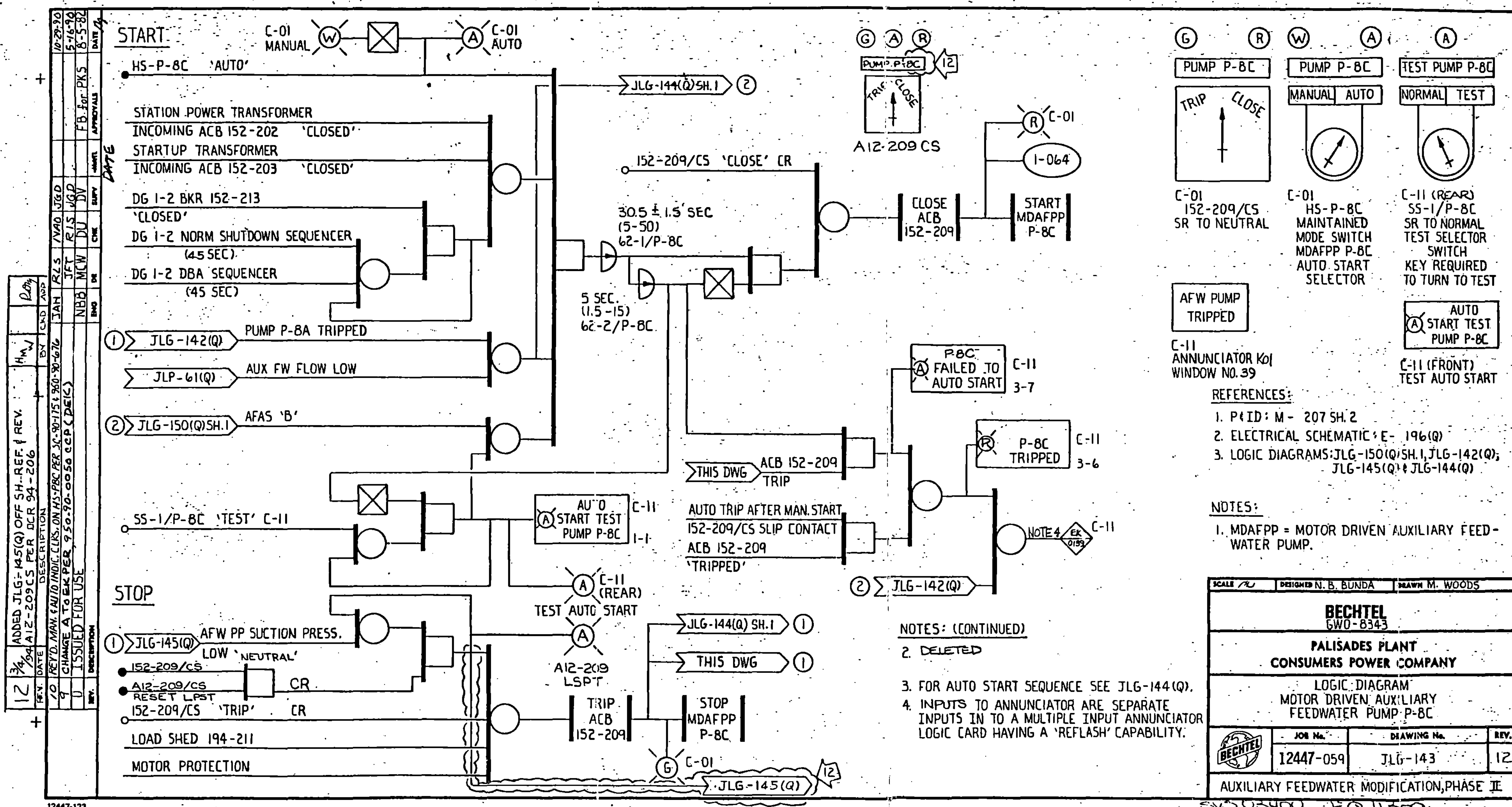

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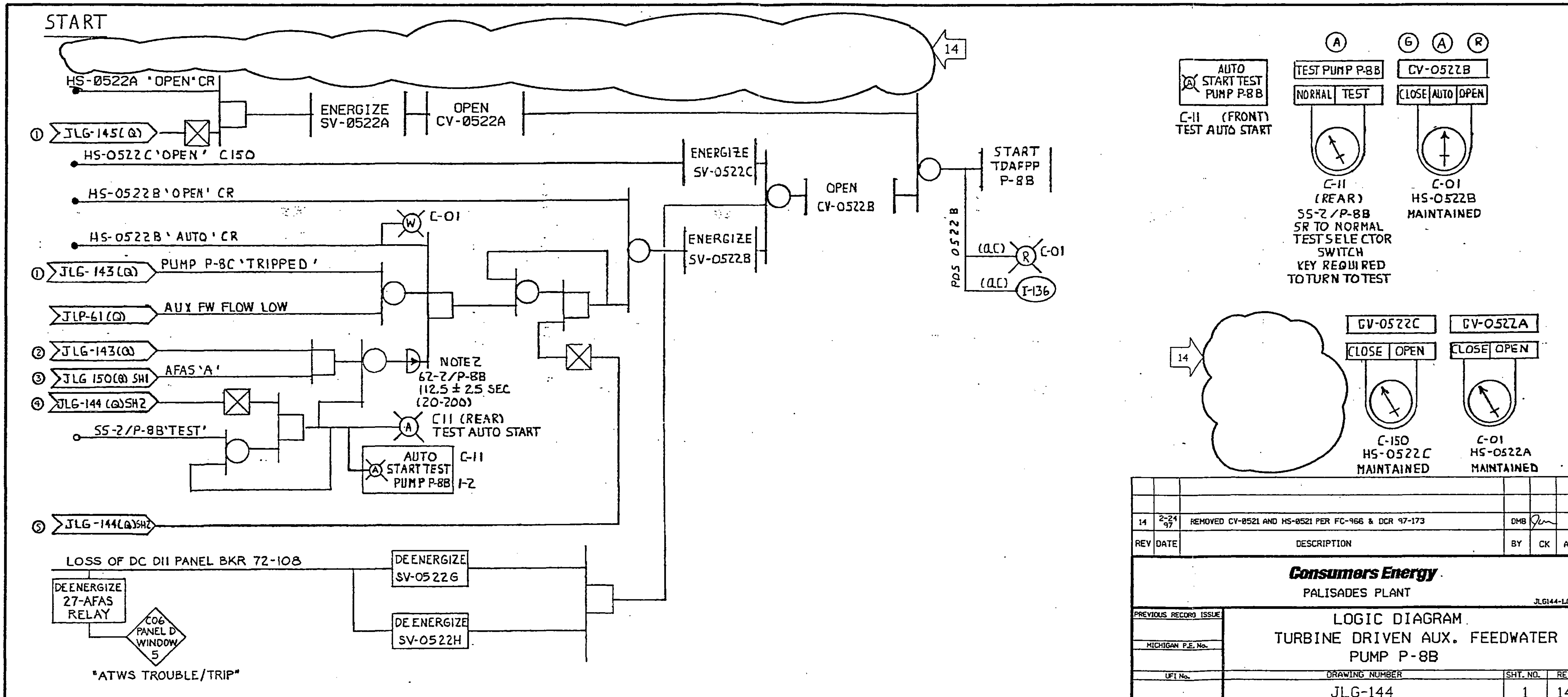
Figure 7-39 FSAR Rev 17



REV	DATE	DESCRIPTION	BY	CK	APP
14	2-24-97	REMOVED CV-0521 AND HS-0521 PER FC-966 & DCR 97-173	DMB		

Consumers Energy
PALISADES PLANT
JLG144-LCIT

LOGIC DIAGRAM
TURBINE DRIVEN AUX. FEEDWATER PUMP P-8B

PREVIOUS RECORD ISSUE	MICHIGAN P.E. No.	UFI No.	DRAWING NUMBER	SHT. NO.	REV
			JLG-144	1	14

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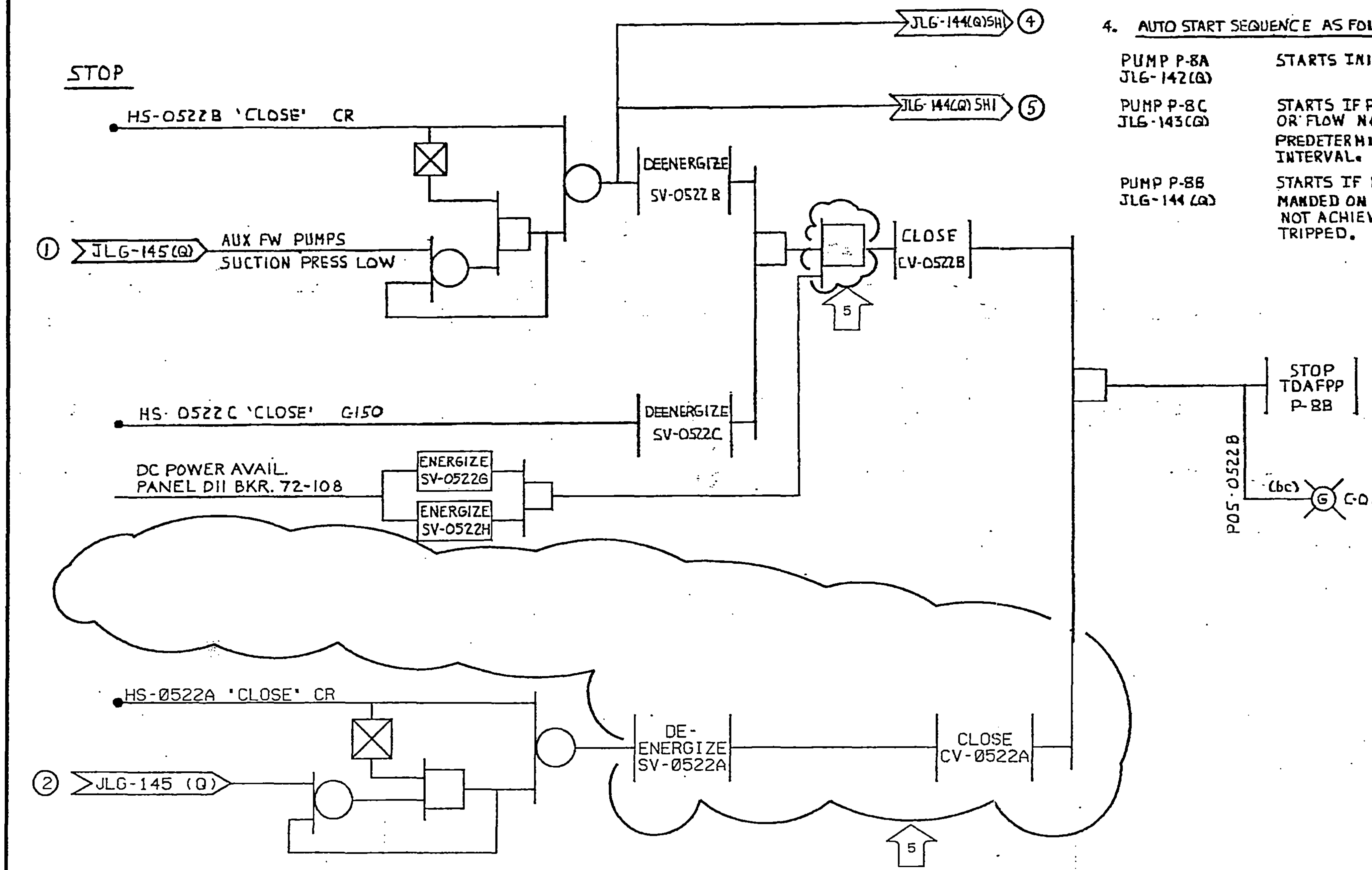
Figure 7-40 Sh 1
Rev 19



4. AUTO START SEQUENCE AS FOLLOWS:

PUMP P-8A JLG-142(Q)	STARTS INITIALLY
PUMP P-8C JLG-143(Q)	STARTS IF P-8A TRIPPED OR FLOW NOT ACHIEVED IN PREDETERMINED TIME INTERVAL.
PUMP P-8B JLG-144(Q)	STARTS IF P-8C WAS COMMANDED ON AND FLOW WAS NOT ACHIEVED OR P-8C TRIPPED.

REFERENCES:

1. P&ID: M-207 SH2 & M-205 SH2
2. ELECTRICAL SCHEMATIC: E-238(Q) SH.4
3. LOGIC DIAGRAMS: JLG-143(Q) & JLG-145(Q)

NOTES:

1. THIS DRAWING SUPERCEDES LOGIC DIAGRAM 12447-039-JLG-96
2. TDR DELAY TIME WILL BE GREATER THAN THE DELAY TIME OF OUTPUT RELAYS OF PUMP P-8C (JLG-143(Q))
3. TDAFPP = TURBINE DRIVEN AUXILIARY FEEDWATER PUMP

REV	DATE	DESCRIPTION	BY	CK	APP
5	2-24-97	REMOVED HS-0521 & ADDED CV-0522APER FC-966 AND DCR 97-173	DMB		

Consumers Energy
PALISADES PLANT

JLG144-2.CIT

PREVIOUS RECORD ISSUE
MICHIGAN P.E. No.
UFI No.

LOGIC DIAGRAM
TURBINE DRIVEN AUX FEEDWATER
PUMP P-8B

DRAWING NUMBER	SHT. NO.	REV
JLG-144	2	5

Figure 7-40 Sh 2
Rev 19

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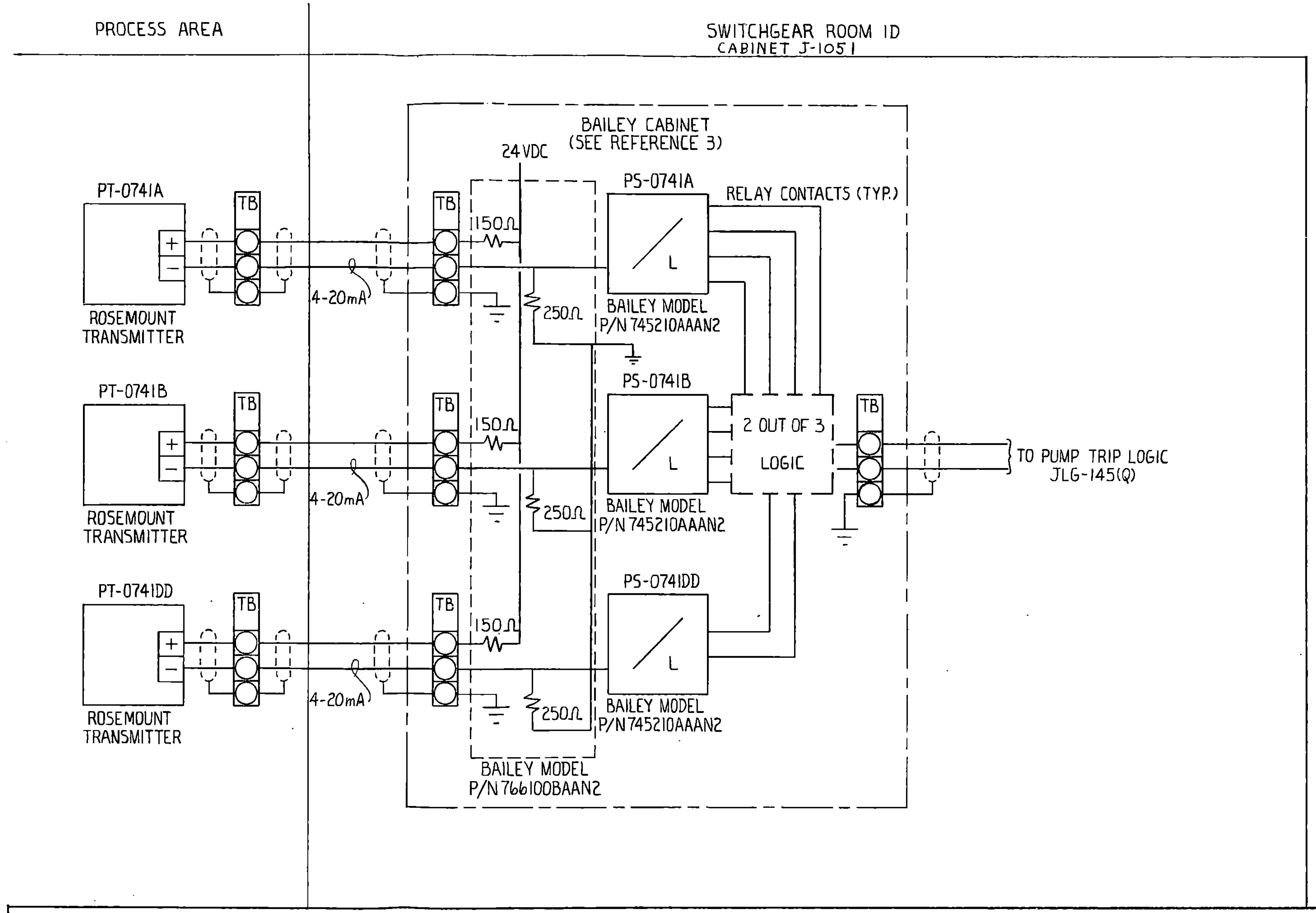

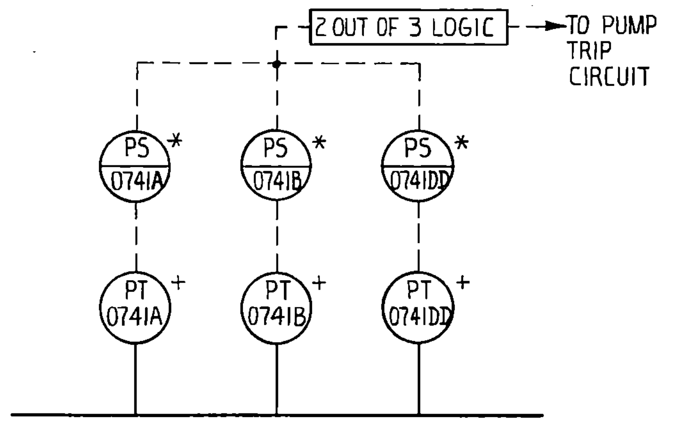

\+ SUPPLIED BY ROSEMOUNT M/R-12447/059-J-448(Q)

* SUPPLIED BY BAILEY M/R-12447/059-J-441A(Q)

REFERENCES:

1. P&ID: M- 207(Q) SH. 2
2. ELECTRICAL SCHEMATIC: E-81
3. FOR DETAILED CABINET WIRING INFORMATION SEE VENDOR PRINT WHICH IS IDENTIFIED IN THE ADAPTER TABLE BELOW.

ADAPTER TABLE

TRANSMITTER	PRESSURE SWITCH	VENDOR PRINT NO.	TRANSMITTER	PRESSURE SWITCH	VENDOR PRINT NO.	TRANSMITTER	PRESSURE SWITCH	VENDOR PRINT NO.	POWER CHANNEL	ASSOCIATED PUMP	CABINET
PT-0741A	PS-0741A	J-441A(Q)-6	PT-0741B	PS-0741B	J-441A(Q)-7	PT-0741DD	PS-0741DD	J-441A(Q)-8	LEFT	P-8A OR P-8B	J-1051
PT-0762A	PS-0762A	J-441A(Q)-15	PT-0762B	PS-0762B	J-441A(Q)-16	PT-0762C	PS-0762C	J-441A(Q)-17	RIGHT	P-8C	J-1052

TI APERTURE CARD

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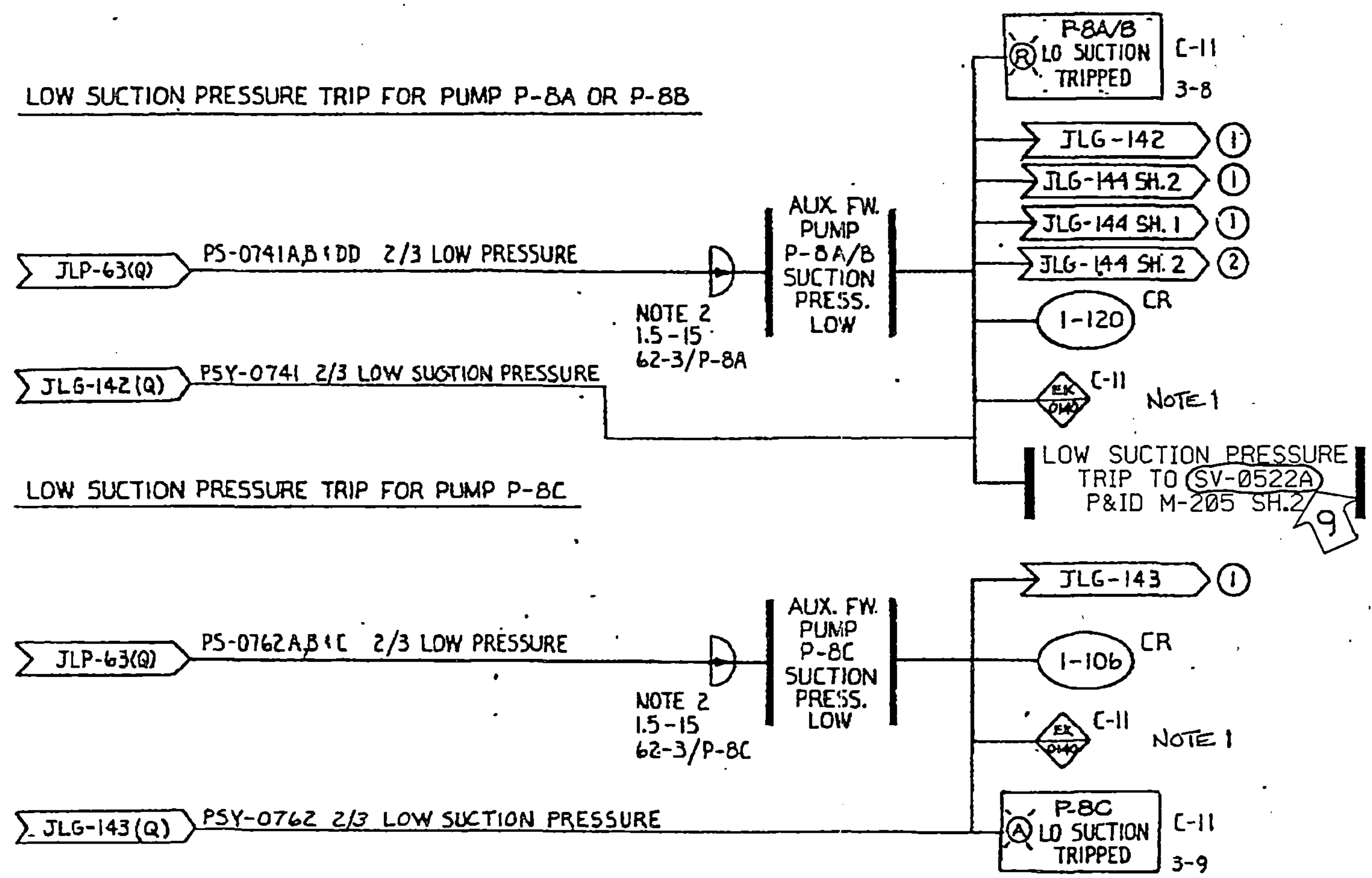
(REF JLP-63(Q) REV.1)

CONSUMERS POWER COMPANY
PALISADES PLANT
FSAR UPDATE

INTRFACE LOOP DIAGRAM
PUMP LOW SUCTION PRESSURE TRIP

FIGURE NO 7-41

REVISION NO 0



AUX FWP
LO SUCTION
PRESSURE

C 11 WINDOW 40
ANNUNCIATOR K01

REFERENCES:
1. P&ID: M- 207(Q) SH.2
2. ELECTRICAL SCHEMATIC: E- 81(Q)SH.3
E-238(Q) SH.4
3. LOGIC DIAGRAMS: JLG-142(Q), JLG-143(Q),
JLG-144(Q) & JLG-102

9

NOTES
1. INPUTS TO ANNUNCIATOR ARE SEPARATE INPUTS INTO A MULTIPLE INPUT ANNUNCIATOR LOGIC CARD HAVING A 'REFLASH' CAPABILITY.
2. TIME DELAY SETPOINT TO BE DETERMINED DURING START UP.

REV.	DATE	DESCRIPTION	BY	CK	APP
9	9-12 95	REV'D. REF'S. & SV0521 TO 0522A PER FC-954 & DCR-95-917.	JGD		DMK

CONSUMERS POWER COMPANY
PALISADES NUCLEAR PLANT
COVERT, MICHIGAN
JLG145.CIT

LOGIC DIAGRAM
AUXILIARY FEEDWATER PUMPS
LOW SUCTION PRESS TRIP

DRAWING NUMBER	SHT. NO.	REV.
JLG-145	---	9

Figure 7-42
Rev 19

9707070091 -20

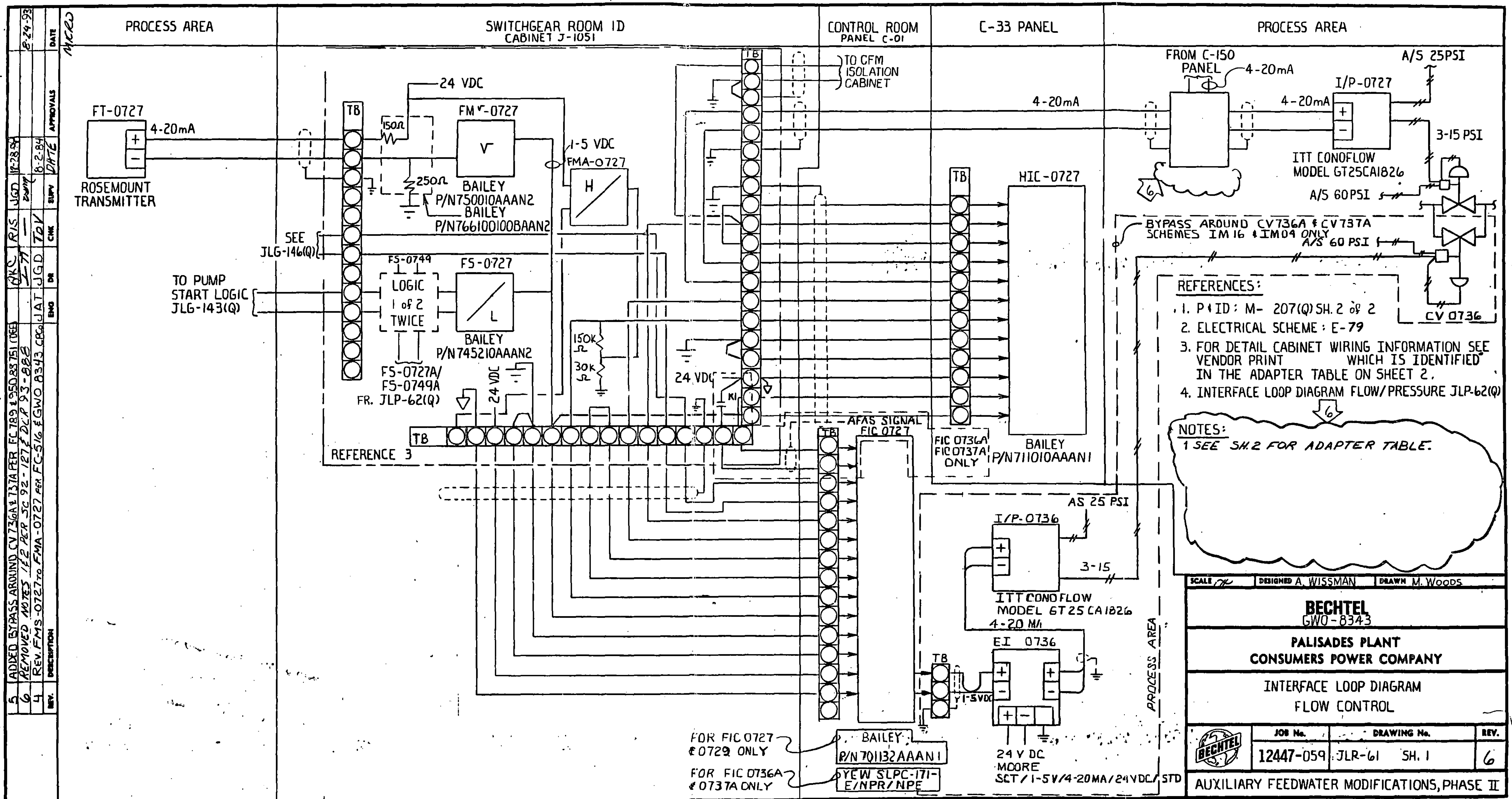


Figure No 7-43 FSAR Rev 16

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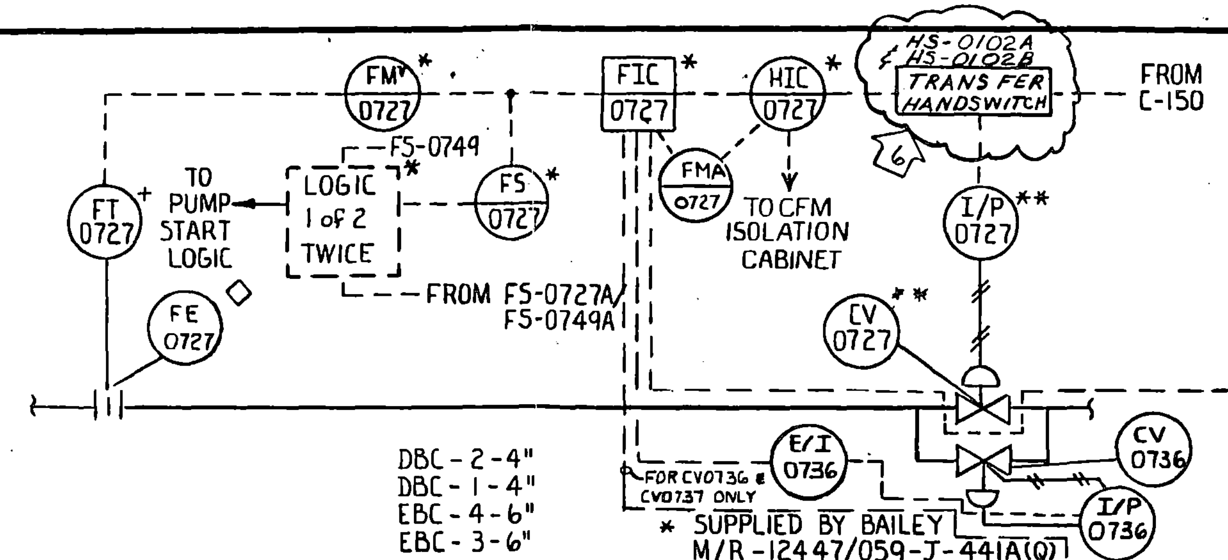

* SUPPLIED BY BAILEY M/R-12447/059-J-441A(Q)

** SUPPLIED BY COPES-VULCAN M/R-12447/059-J-436(Q)

\+ SUPPLIED BY ROSEMOUNT M/R-12447/059-J-448(Q)

◇ SUPPLIED BY FLOW TECH M/R-12447/059-J-451

ADAPTER TABLE

TRANSMITTER	SQUARE ROOT EXT.	FLOW SWITCH	TO PUMP LOGIC	FLOW INDICATING CONTROLLER	PANEL LOCATION	HAND INDICATING CONTROLLER	PANEL LOCATION	VENDOR PRINT	POWER CHANNEL	INSTR. PANEL	FIRE PROTECTION HIC	FIRE PROTECTION POWER CHANNEL	FIRE PROTECTION PANEL LOCATION	FIRE PROTECTION DRAWING REF.	I/P CONVERTER	CONTROL VALVE	PUMP LOGIC INTERFACE	NOTES	HIC MODE MONITOR SWITCH
FT-0727	FM^Y-0727	FS-0727	P-8C	FIC-0727	C-01	HIC-0727	C-33	J-441A(Q)-18&19	LEFT	J-1051	HIC-0727C	LEFT	C-150		I/P-0727	CV-0727	JLG-143(Q)	—	FMA-0727
FT-0749	FM^Y-0749	FS-0749	P-8C	FIC-0749	C-01	HIC-0749	C-33	J-441A(Q)-20&21	LEFT	J-1051	HIC-0749C	LEFT	C-150		I/P-0749	CV-0749			FMA-0749
FT-0736A	FM^Y-0736AA	FS-0736A	P-8B	FIC-0736A	C-01	HIC-0736A	C-33	J-441A(Q)-9&10	RIGHT	J-1052	—	—	—	—	I/P-0736A	CV-0736A	JLG-144(Q)		FMA-0736A
FT-0737A	FM^Y-0737AA	FS-0737A	P-8B	FIC-0737A	C-01	HIC-0737A	C-33	J-441A(Q)-11&12	RIGHT	J-1052	—	—	—	—	I/P-0737A	CV-0737A			FMA-0737A

6

CRITICAL FUNCTION MONITOR SYSTEM PANEL LOCATION	CONTROL VALVE (BYPASS)	I/P CONVERTER	SIGNAL CONVERTER
J450			
J450			
J451	CV0736	I/P0736	E/I 0736
J451	CV0737	I/P0737	E/I 0737

REV.	DESCRIPTION	ENG	DR	CHK	SUPV	APPROVALS	DATE
6	REV. FIRE PROT. NOTES & TRANSFER HANDSWITCH [illegible]						9-24-93
5	REV. FIC-0727 & ADAPTER TABLE PER FT 789 & 9508875I (DEF)		GKC	RJS			
D	ISSUED FOR USE	ATJ	MCW	EJV	JGD		12-28-88

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PALISADES PLANT
CONSUMERS POWER COMPANY

INTERFACE LOOP DIAGRAM
FLOW CONTROL

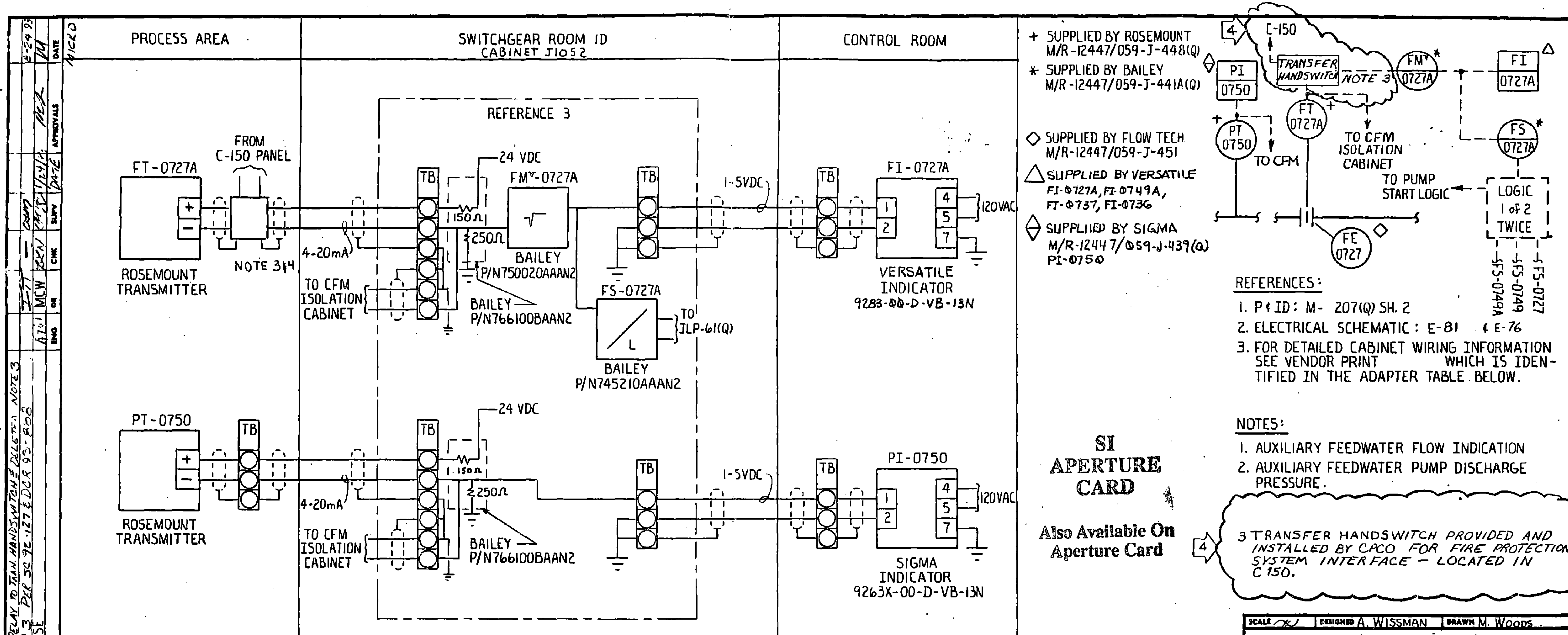
JOB No.	DRAWING No.	REV.
12447-059	JLP-61 SH. 2	6

AUXILIARY FEEDWATER MODIFICATIONS, PHASE II

12447-123

Figure No 7-44 FSAR Rev 16

9312220060 - 11



+ SUPPLIED BY ROSEMOUNT M/R-12447/059-J-448(Q)
✱ SUPPLIED BY BAILEY M/R-12447/059-J-441A(Q)
◇ SUPPLIED BY FLOW TECH M/R-12447/059-J-451
△ SUPPLIED BY VERSATILE FI-0727A, FI-0749A, FI-0737, FI-0736
◈ SUPPLIED BY SIGMA M/R-12447/059-J-439(Q) PI-0750

REFERENCES:
1. P&ID: M- 207(Q) SH. 2
2. ELECTRICAL SCHEMATIC: E-81 & E-76
3. FOR DETAILED CABINET WIRING INFORMATION SEE VENDOR PRINT WHICH IS IDENTIFIED IN THE ADAPTER TABLE BELOW.

NOTES:
1. AUXILIARY FEEDWATER FLOW INDICATION
2. AUXILIARY FEEDWATER PUMP DISCHARGE PRESSURE.
3 TRANSFER HANDSWITCH PROVIDED AND INSTALLED BY CPCO FOR FIRE PROTECTION SYSTEM INTERFACE - LOCATED IN C150.

SI APERTURE CARD

Also Available On Aperture Card

ADAPTER TABLE

TRANSMITTER	SQUARE ROOT EXTRACTOR	FLOW SWITCH	CABINET NUMBER	CABINET LOCATION	INDICATOR NUMBER	INDICATOR LOCATION	POWER CHANNEL	ASSOCIATED PUMP	VENDOR PRINT NUMBER	CRITICAL FUNCTION MONITOR LOCATION	NOTES
FLOW INDICATION											
FT-0727A	FMᵛ-0727A	FS-0727A	J-1051	SWGR RM 1D	FI-0727A	C-11	LEFT	P-8A OR P-8B	J-441A(Q)-22	J450	1
FT-0749A	FMᵛ-0749A	FS-0749A	J-1051	SWGR RM 1D	FI-0749A	C-11	LEFT	P-8A OR P-8B	J-441A(Q)-23	J450	1
FT-0737	FMᵛ-0737	FS-0737	J-1052	SWGR RM 1D	FI-0737	C-11	RIGHT	P-8C	J-441A(Q)-14	J451	1,3
FT-0736	FMᵛ-0736	FS-0736	J-1052	SWGR RM 1D	FI-0736	C-11	RIGHT	P-8C	J-441A(Q)-13	J451	1,3
PRESSURE INDICATION											
PT-0750	—	—	J-1052	SWGR RM 1D	PI-0750	C-01	RIGHT	P-8C	J-441A(Q)-9	J452	2

REV.	DESCRIPTION	DATE
4	REV TRANSFER RELAY TO TRAN. HANDSWITCH = DELETE NOTE 3, REV NOTE 4 TO 3 PER SC 92-127 & DCR 93-1006	2-24-95
U	ISSUED FOR USE	

SCALE NONE | DESIGNED A. WISSMAN | DRAWN M. WOODS

BECHTEL GWO-8343

PALISADES PLANT
CONSUMERS POWER COMPANY

INTERFACE LOOP DIAGRAM
FLOW & PRESSURE INDICATION

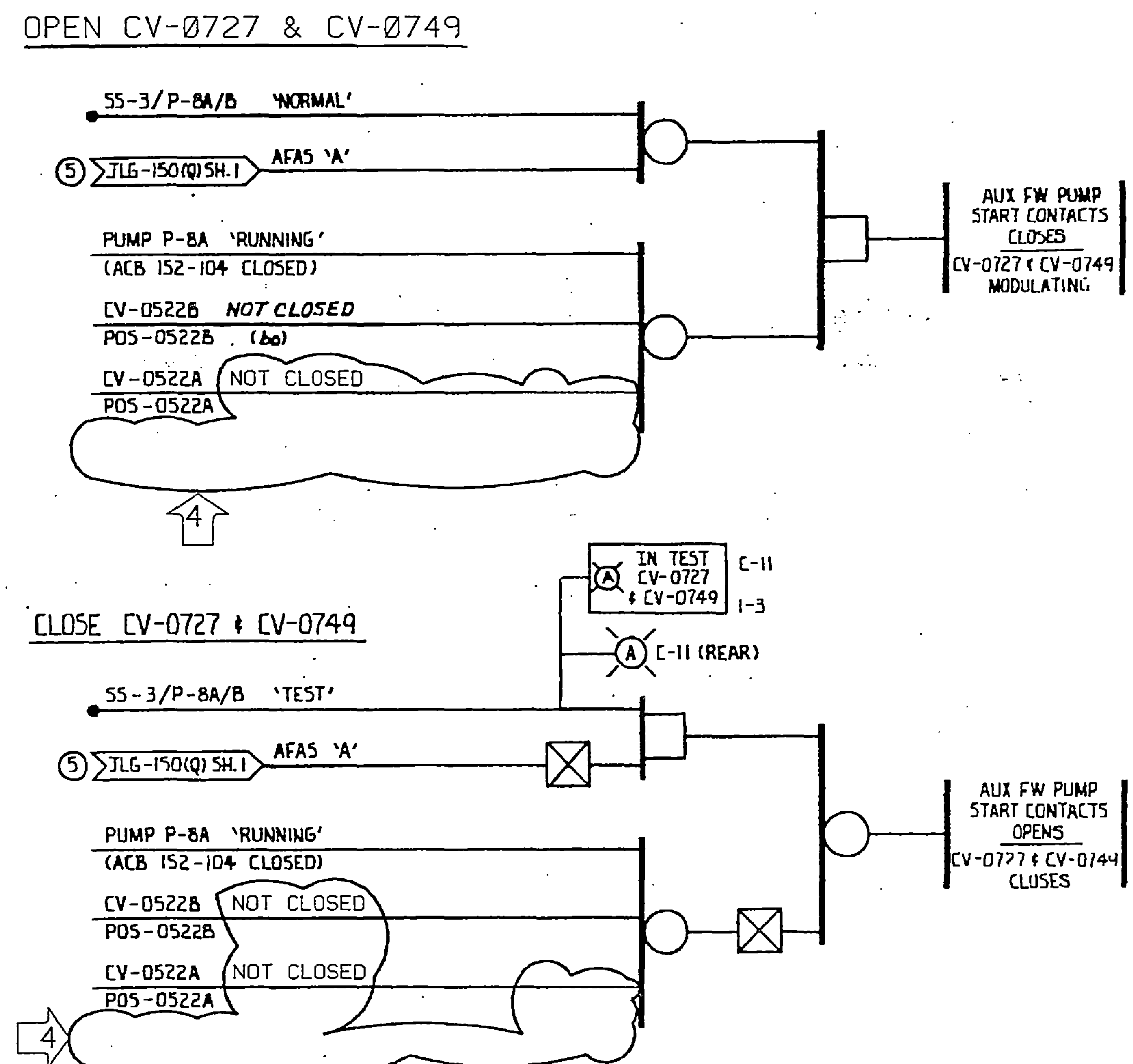
JOB No. 12447-059 | DRAWING No. JLP-62 | REV. 4

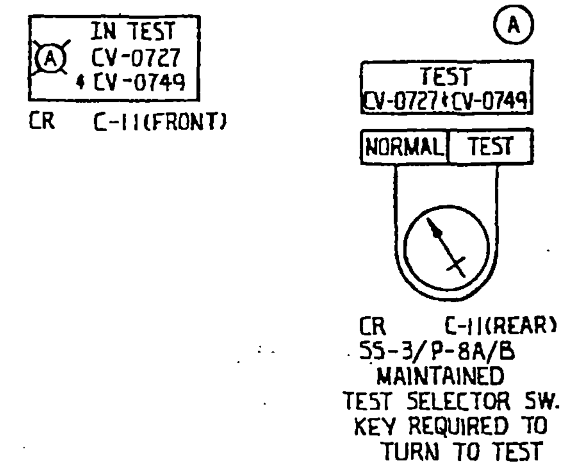
AUXILIARY FEEDWATER MODIFICATIONS PHASE II

12447-123

Figure No 7-45 FSAR Rev 16

9312220060 - 12





SCALE	DESIGNED D. UGORCAK	DRAWN M. WOODS

BECHTEL
GWO-8343

PALISADES PLANT
CONSUMERS POWER COMPANY

LOGIC DIAGRAM
AUXILIARY FEEDWATER FLOW CONTROL AND PUMP TEST LOGIC

JOB No.	DRAWING No.	REV.
12447-059	JLG-146(Q) SH 1 of 2	3

AUXILIARY FEEDWATER MODIFICATION, PHASE II

NOTES:

1. THIS DRAWING SUPERCEDES JLG-101(Q)
2. "PUMP RUNNING" CLOSES THE RELAY CONTACT SUPPLYING THE FLOW CONTROLLER WITH MANUAL SETPOINT, ENABLING CONTROL OF VALVE POSITIONS.

NOTES-CONTINUED

2. (CONT.)

"PUMP STOPPED" OPENS THIS CONTACT, BLOCKING THE MANUAL SETPOINT TO THE CONTROLLER, CAUSING THE VALVES TO BE DRIVEN CLOSED. THE TEST SWITCH BLOCKS THE "PUMP RUNNING" SIGNAL TO KEEP THE VALVES CLOSED DURING PERIODIC PUMP TESTING.

REV.	DATE	DESCRIPTION	BY	CK	APP
4	2-24-97	REVISED CV-0522A & B TO NOT CLOSED. DELETE CV-0521 PER FC-966 AND DCR 97-182 (REDRAWN ON CAD)	MDS	[illegible]	

REFERENCES:

1. P&ID: M-207(Q) SH 2
2. ELECTRICAL SCHEMATIC: E-196(Q), SH 7&8
3. LOGIC DIAGRAM: JLG-150(Q) SH. 1
4. LOOP DIAGRAM: JLP-61(Q)

Consumers Energy
PALISADES PLANT

JLG146-1.CIT

LOGIC DIAGRAM
AUXILIARY FEEDWATER FLOW CONTROL
AND PUMP TEST LOGIC

DRAWING NUMBER	SHT. NO.	REV.
JLG-146	1	4

Figure 7-46
Rev 19

9707070091 -21

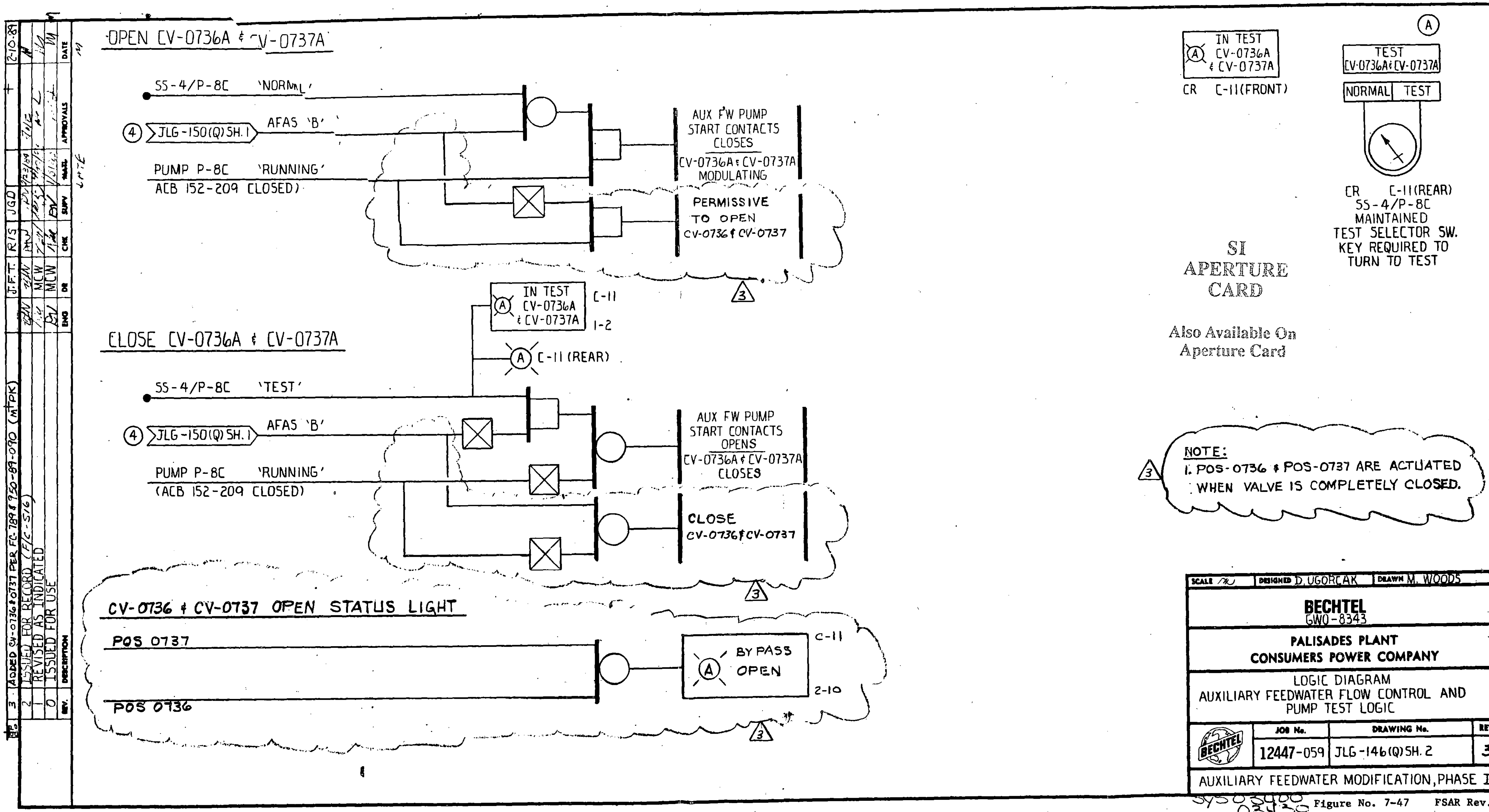
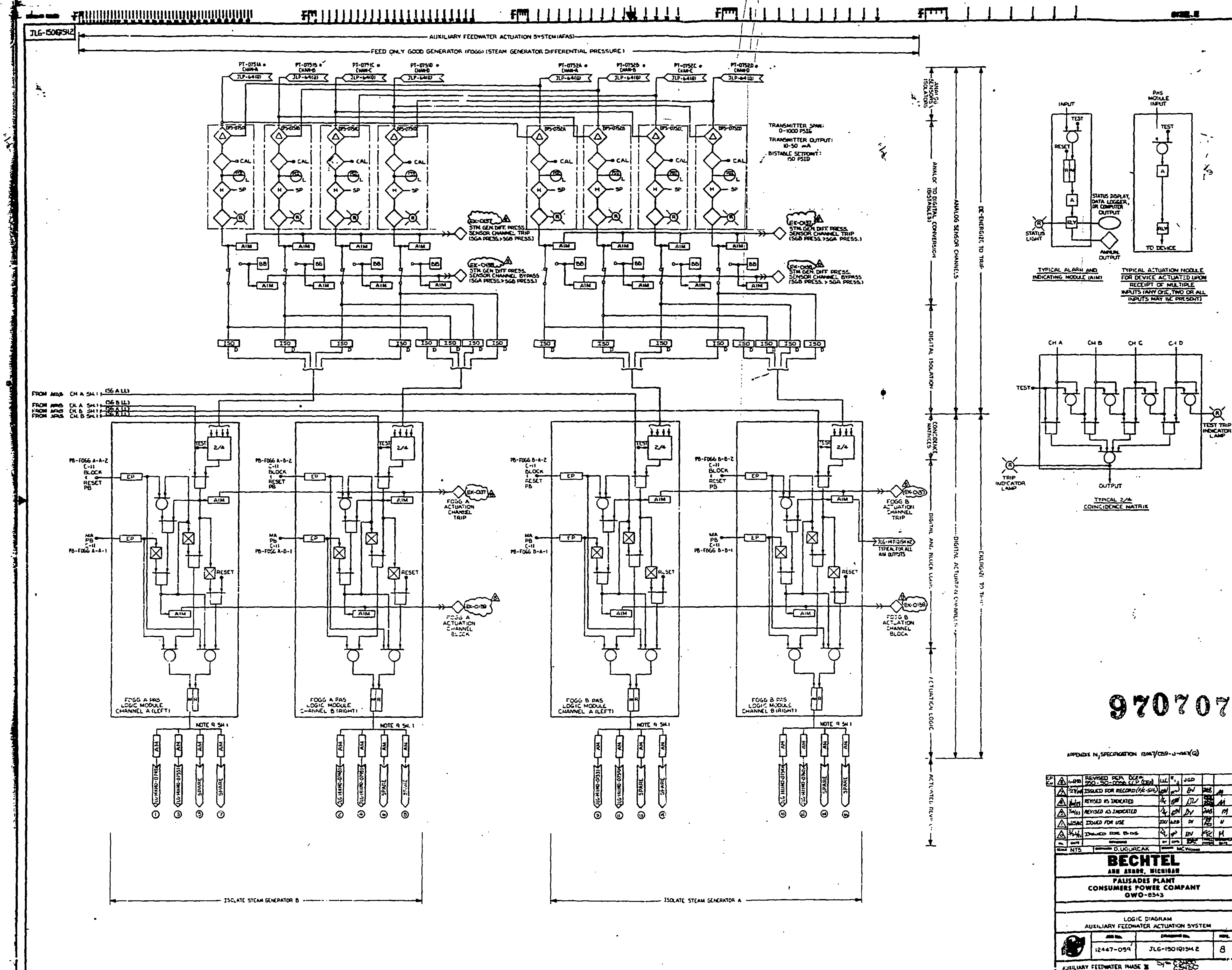


Figure No. 7-47 FSAR Rev. 9

8910040143-47

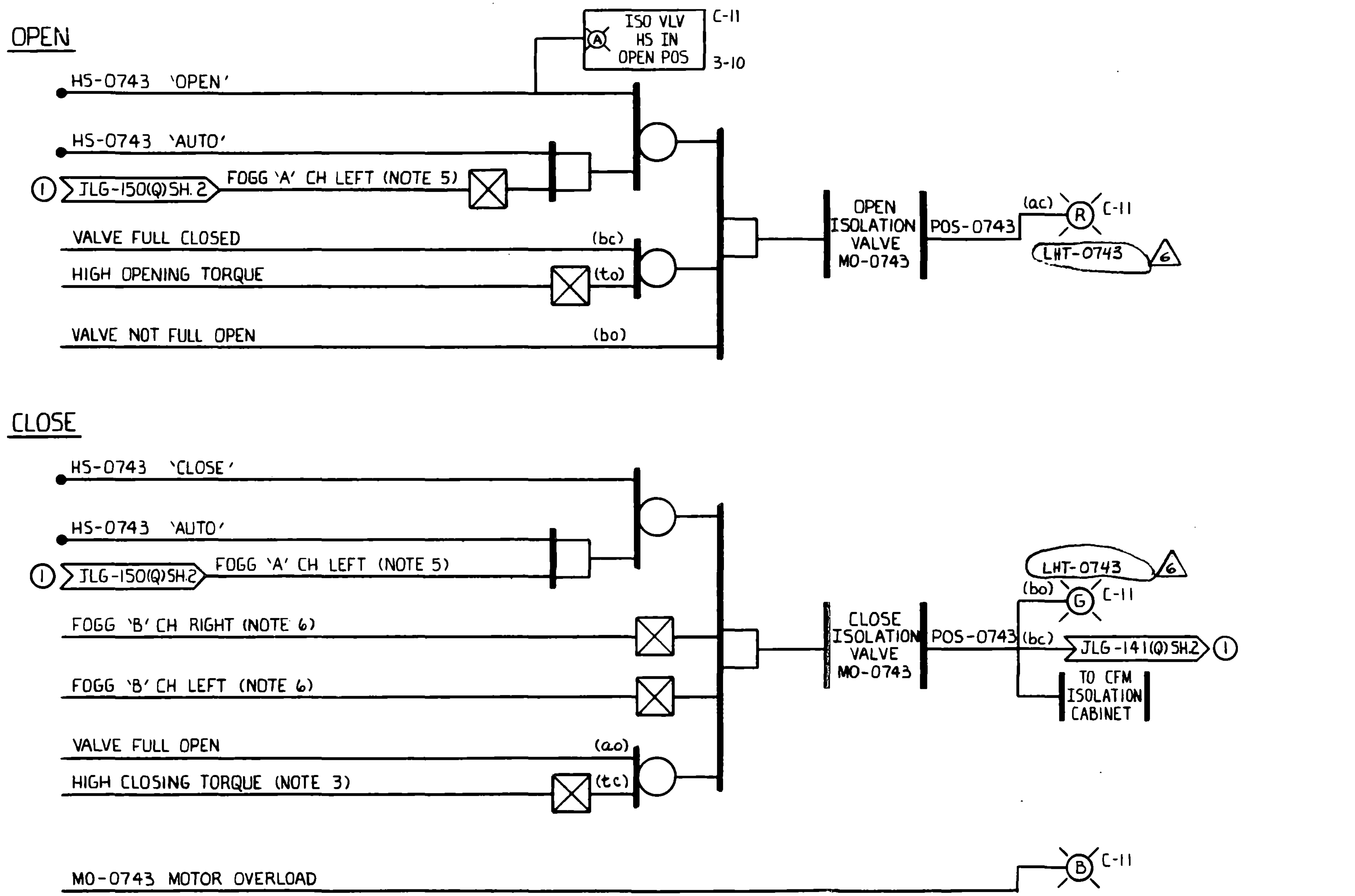
FIGURE NO. 7-48

(deleted)



9707070091-22

Figure 7-49
Rev 19



SI APERTURE CARD

Also Available On Aperture Card

REFERENCES:

1. P&ID: M- 207(Q) SH. 2
2. ELECTRICAL SCHEMATIC: E-1174(Q)
3. LOGIC DIAGRAM: JLG-150(Q) SH. 2

NOTES:

1. THE VALVE HAS AN IMMEDIATE REVERSAL FEATURE.
2. FOR ADAPTER TABLE SEE SH. 2
3. HIGH CLOSING TORQUE STOPS VALVE WITH NO POSITION SWITCH BACKUP.
4. INHERENT INTERLOCK: REVERSING STARTER CROSS INTERLOCK IS NOT SHOWN.
5. FOGG B = FEED SG B, ISOLATE SG A
   FOGG A = FEED SG A, ISOLATE SG B
6. ISOLATION OF SG 'B' NOT PERMITTED IF SG 'A' HAS BEEN ISOLATED.

REV.	DESCRIPTION	DATE
4	REVISED AS INDICATED	
5	ISSUED FOR RECORD (F/C-516)	5-16-90
6	ADDED LHT-0743 PER 950-90-0050 CCP (DEK)	

SCALE — DESIGNED N. B. BUNDA DRAWN M. WOODS

BECHTEL
GWO-8343

PALISADES PLANT
CONSUMERS POWER COMPANY

LOGIC DIAGRAM
AUXILIARY FEEDWATER-STEAM GENERATOR ISOLATION VALVES

JOB No.	DRAWING No.	REV.
12447-059	JLG-141(Q) SH. 1 of 2	6

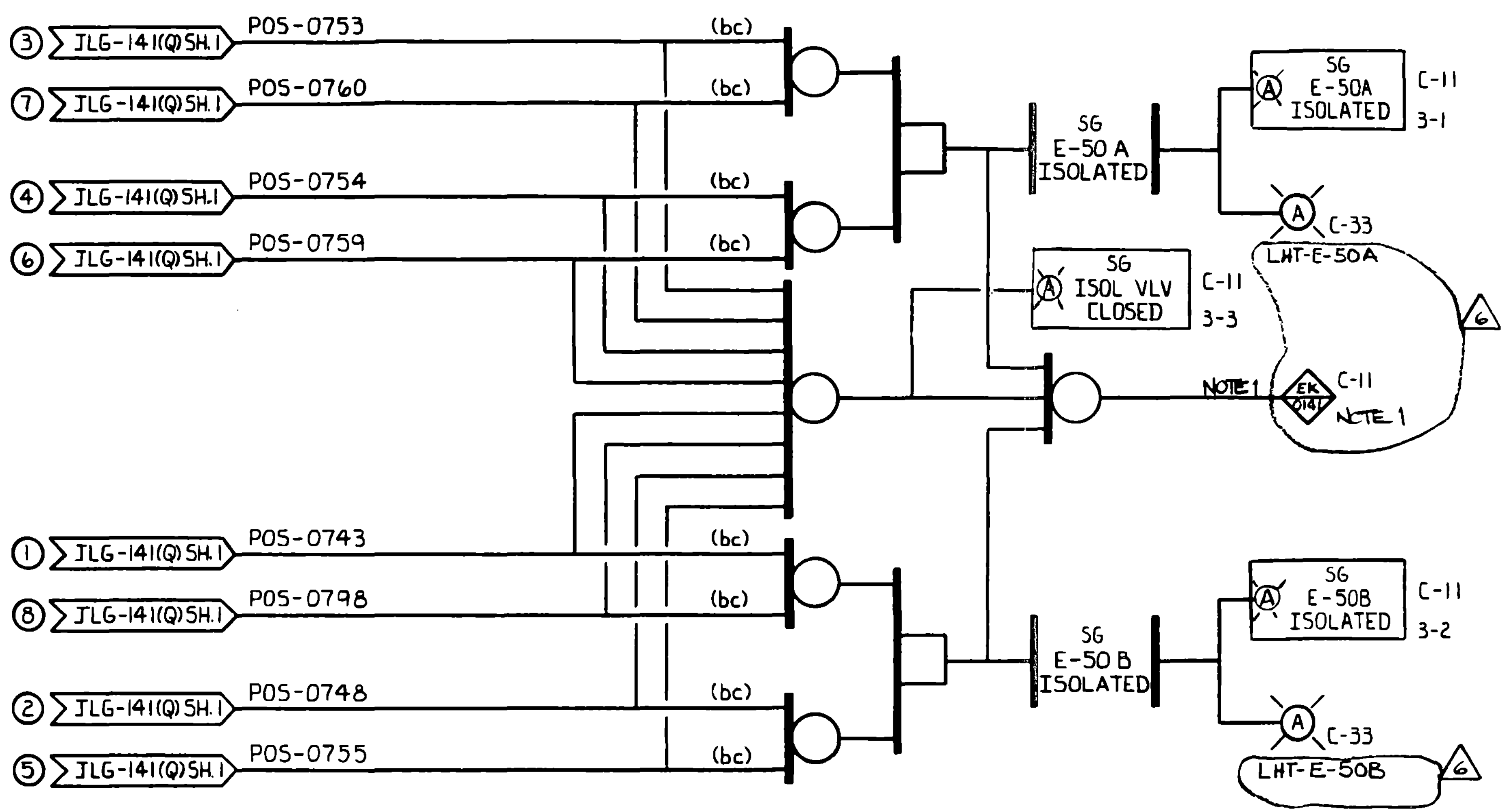
AUXILIARY FEEDWATER PHASE II

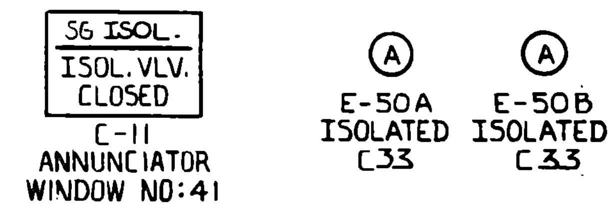
12447-123

SYS 03400 03430 EQ 13000 13100 13370

Figure No. 7-50 FSAR Rev. 11

9012260101-37





SI APERTURE CARD

Also Available On Aperture Card

NOTES:

1. INPUTS TO ANNUNCIATOR ARE SEPARATE INPUTS INTO A MULTIPLE INPUT ANNUNCIATOR LOGIC CARD HAVING A 'REFLASH' CAPABILITY.

ADAPTER TABLE

ISOLATION VALVE NUMBER	HAND SWITCH NUMBER	POSITION SWITCH NUMBER	ISOLATES STEAM GENERATOR	ISOLATION COMMAND	POWER SUPPLY CHANNEL	MATCH NUMBER JLG-150(Q) SH. 2	MATCH NUMBER JLG-141(Q) SH. 1	ISOLATION PERMISSIVE FROM	AUX. FW. PUMP NUMBER	LOCATION CS LIGHT	LOCATION LIGHT	LOCATION ALARM	LOCATION AFW MATRIX DISPLAY IND.	CFM INTERFACE CABINET
MO-0754	HS-0754	POS-0754	E-50 A	FOGG B	RIGHT	10	4	FOGG A	P-8C	C 11	C 33	C 11	C-11	J-451
MO-0759	HS-0759	POS-0759	E-50 A	FOGG B	LEFT	11	6	FOGG A	P-8C	C 11				J-450
MO-0760	HS-0760	POS-0760	E-50 A	FOGG B	RIGHT	12	7	FOGG A	P-8A/B	C 11				J-451
MO-0753	HS-0753	POS-0753	E-50 A	FOGG B	LEFT	9	3	FOGG A	P-8A/B	C 11				J-450
MO-0748	HS-0748	POS-0748	E-50 B	FOGG A	RIGHT	2	2	FOGG B	P-8C	C 11	C 33			J-451
MO-0755	HS-0755	POS-0755	E-50 B	FOGG A	LEFT	3	5	FOGG B	P-8C	C 11				J-450
MO-0798	HS-0798	POS-0798	E-50 B	FOGG A	RIGHT	4	8	FOGG B	P-8A/B	C 11				J-451
MO-0743	HS-0743	POS-0743	E-50 B	FOGG A	LEFT	1	1	FOGG B	P-8A/B	C 11				J-450

REV.	DESCRIPTION	ENG	DE	CHK	SUPV	MATE	APPROVALS	DATE
4	REVISED AS INDICATED							
5	ISSUED FOR RECORD (F/C-516)							
6	ADDED LHT-E-50A & B, REV'D A TO EK PER 950-90-0050CCP (DEK)		JFT	RIS	JGD			5-16-90

SCALE /NONE DESIGNED N. B. BUNDA DRAWN M. WOODS

BECHTEL
GWO-8343

PALISADES PLANT
CONSUMERS POWER COMPANY

LOGIC DIAGRAM
AUXILIARY FEEDWATER-STEAM GENERATOR ISOLATION VALVES

JOB No.	DRAWING No.	REV.
12447-059	JLG-141(Q) SH. 2 of 2	6

AUXILIARY FEEDWATER PHASE II

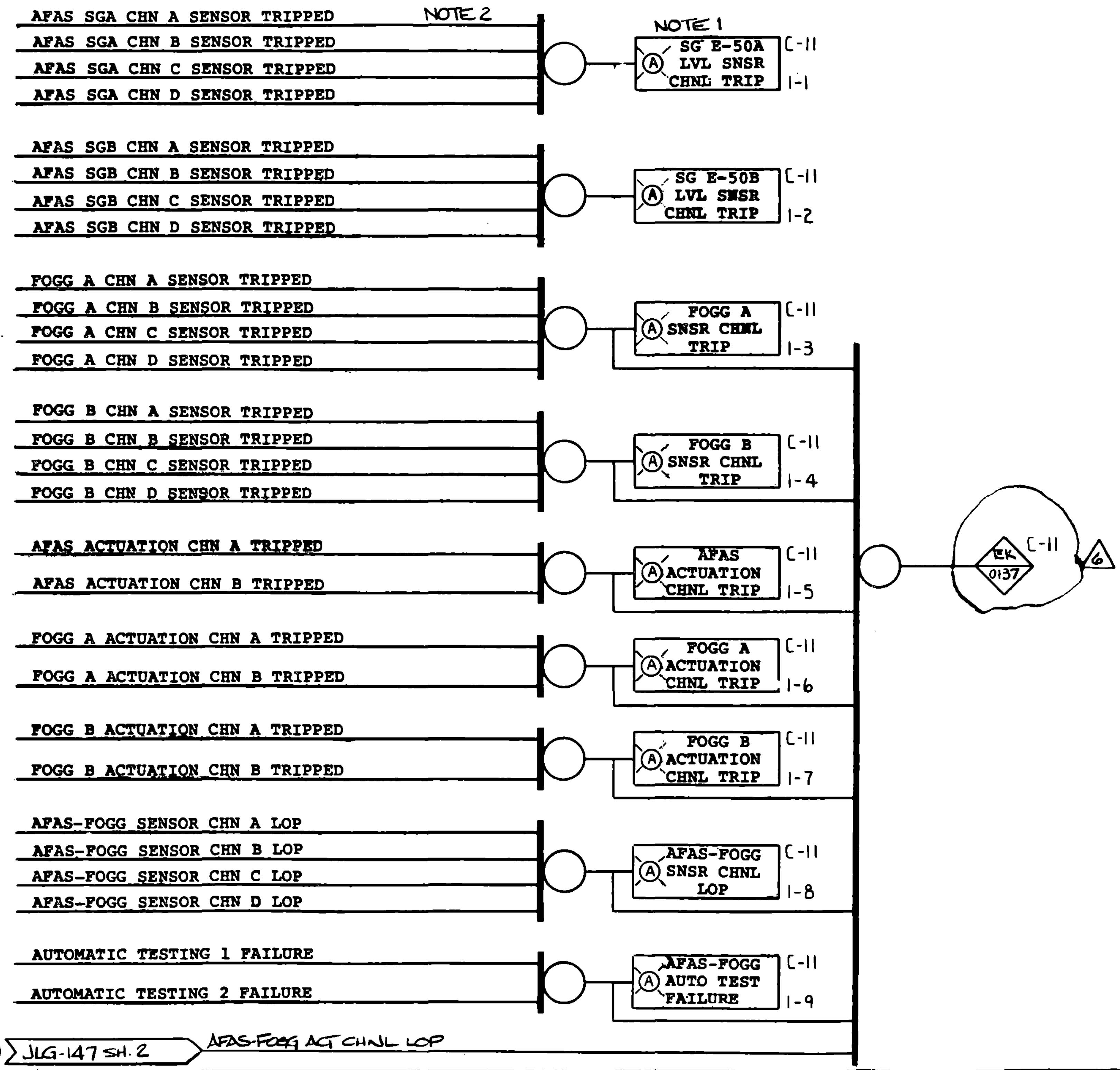
SYS 03400 03430 EQ 13000 13100 13310

12447-123

Figure No. 7-51 FSAR Rev. 11

9012260101-38

REV.	DESCRIPTION
4	REVISED AS INDICATED
5	ISSUED FOR RECORD (F/C-516)
6	CHANGED A TO EK PER 950-90-0050 ccP(DEK)



AFAS - FOGG SUBSYSTEM TRIP

C-11
ANNUNCIATOR K01
WINDOW NO. 37

SI APERTURE CARD

Also Available On Aperture Card

REFERENCES:

1. LOGIC DIAGRAMS: JLG-150(Q) SH. 1 & 2
2. VENDOR PRINTS: 12447-059-J-447(Q)-16, 17, 31, 32, 33 AND 34.
3. ELECTRICAL SCHEMATIC: E-1207 (Q)

NOTES:

1. INDICATOR WINDOWS ARE HOUSED IN A 3×10 ARRAY (PANEL C-11) FOR THE AFW SYSTEM.
2. ALL STATUS OUTPUTS FROM THE AFAS-FOGG ARE INDICATED ON PANEL C-187.
3. ANNUNCIATOR WINDOW IS TO HAVE REFLASH CAPABILITY.

SCALE NO	DESIGNED D. UGORCAK	DRAWN M. WOODS

BECHTEL
GWO-8343

PALISADES PLANT
CONSUMERS POWER COMPANY

LOGIC DIAGRAM
AFAS - FOGG
REMOTE DISPLAY & ANNUNCIATOR ASSIGNMENT

JOB No.	DRAWING No.	REV.
12447-059	JLG-147 SH. 1 of 2	6

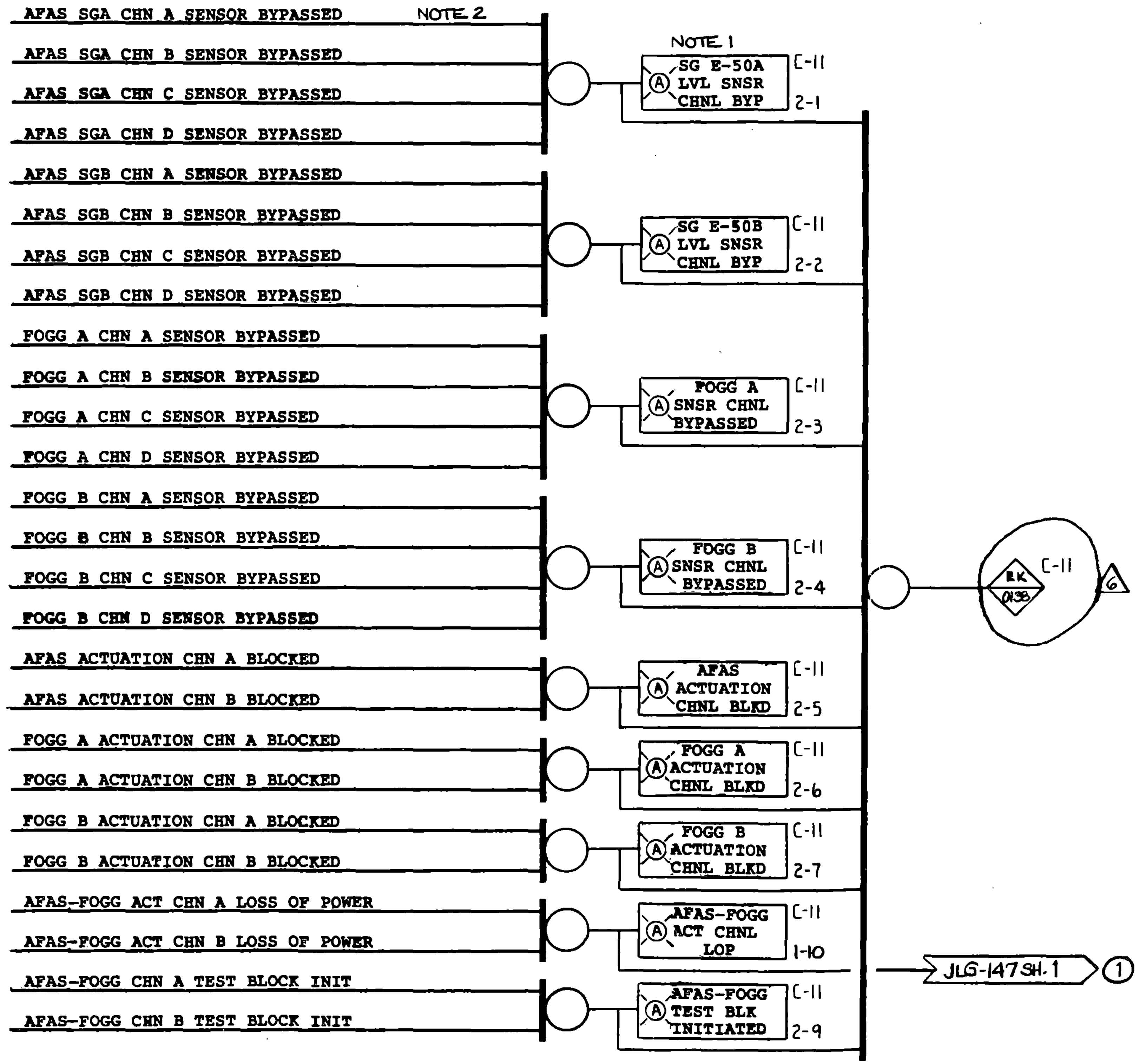
AUXILIARY FEEDWATER MODIFICATION PHASE II

12447-123

SYS 83400 83430 07071 EQ 500000

Figure No. 7-52 FSAR Rev. 11

9012260101-39



AFAS - FOGG SUBSYSTEM BYPASSED

C-11 ANNUNCIATOR K01 WINDOW NO. 38

SI APERTURE CARD

Also Available On Aperture Card

REFERENCES:

1. LOGIC DIAGRAMS: JLG-150(Q) SH. 1 & 2
2. VENDOR PRINTS: 12447-059-J-447(Q)-16, 17, 31, 32, 33 AND 34.
3. ELECTRICAL SCHEMATIC: E-1207(Q)

NOTES:

1. INDICATOR WINDOWS ARE HOUSED IN A 3×10 ARRAY (PANEL C-11) FOR THE AFW SYSTEM.
2. ALL STATUS OUTPUTS FROM THE AFAS-FOGG ARE INDICATED ON PANEL C-187.

REV.	DESCRIPTION
4	REVISED AS INDICATED
5	ISSUED FOR RECORD (F/C-516)
6	CHANGED A To EK PER 950-90-0050 & LP (DEK)

SCALE N/A | DESIGNED D. UGORCAK | DRAWN M. WOODS

BECHTEL GWO-8343

PALISADES PLANT
CONSUMERS POWER COMPANY

LOGIC DIAGRAM
AFAS-FOGG
REMOTE DISPLAY & ANNUNCIATOR ASSIGNMENT

JOB No.	DRAWING No.	REV.
12447-059	JLG-147 SH. 2 of 2	6

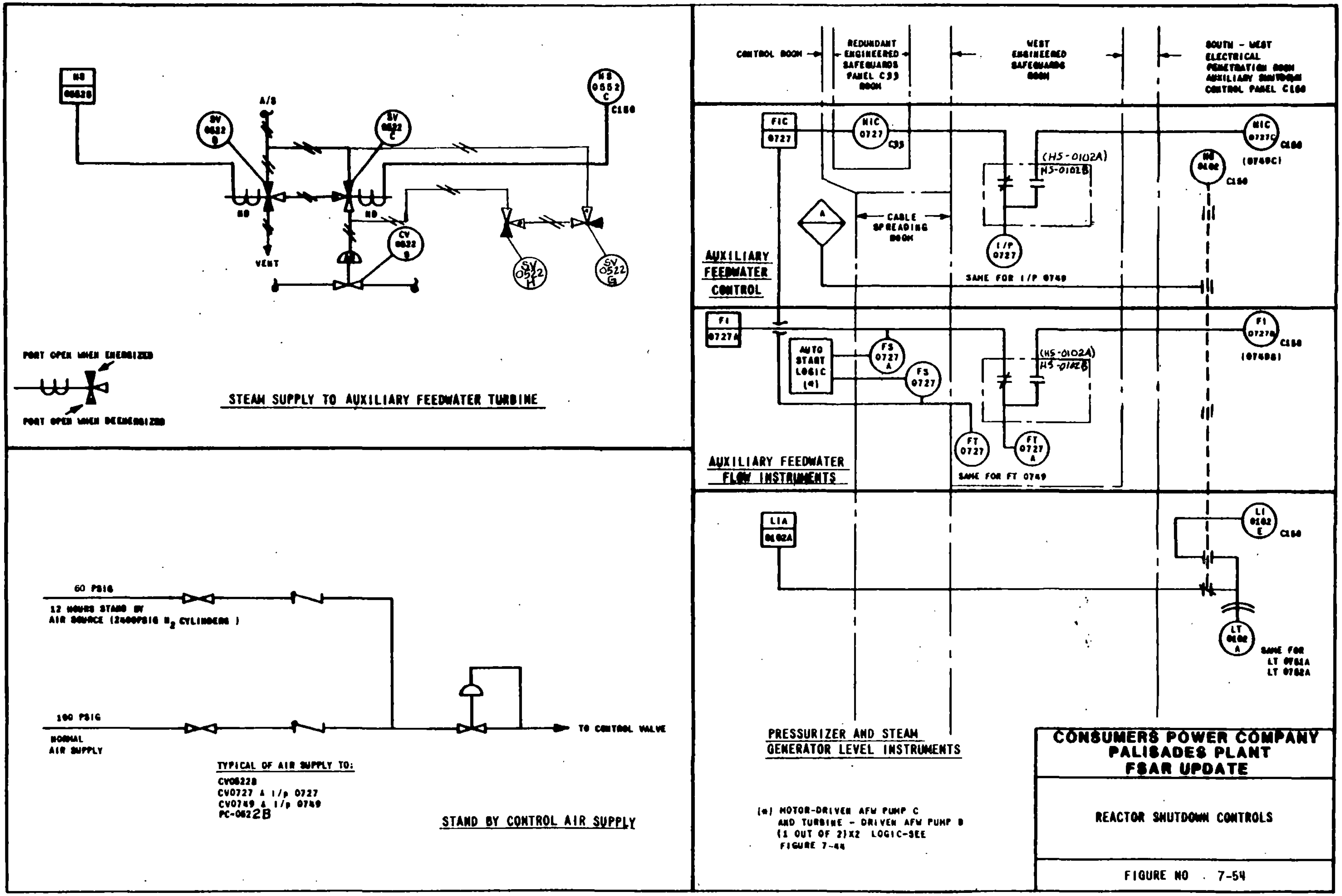
AUXILIARY FEEDWATER MODIFICATION PHASE II

12447-123

SYS 83400 07071 EQ 500060

Figure No. 7-53 FSAR Rev. 11

9012260101-40

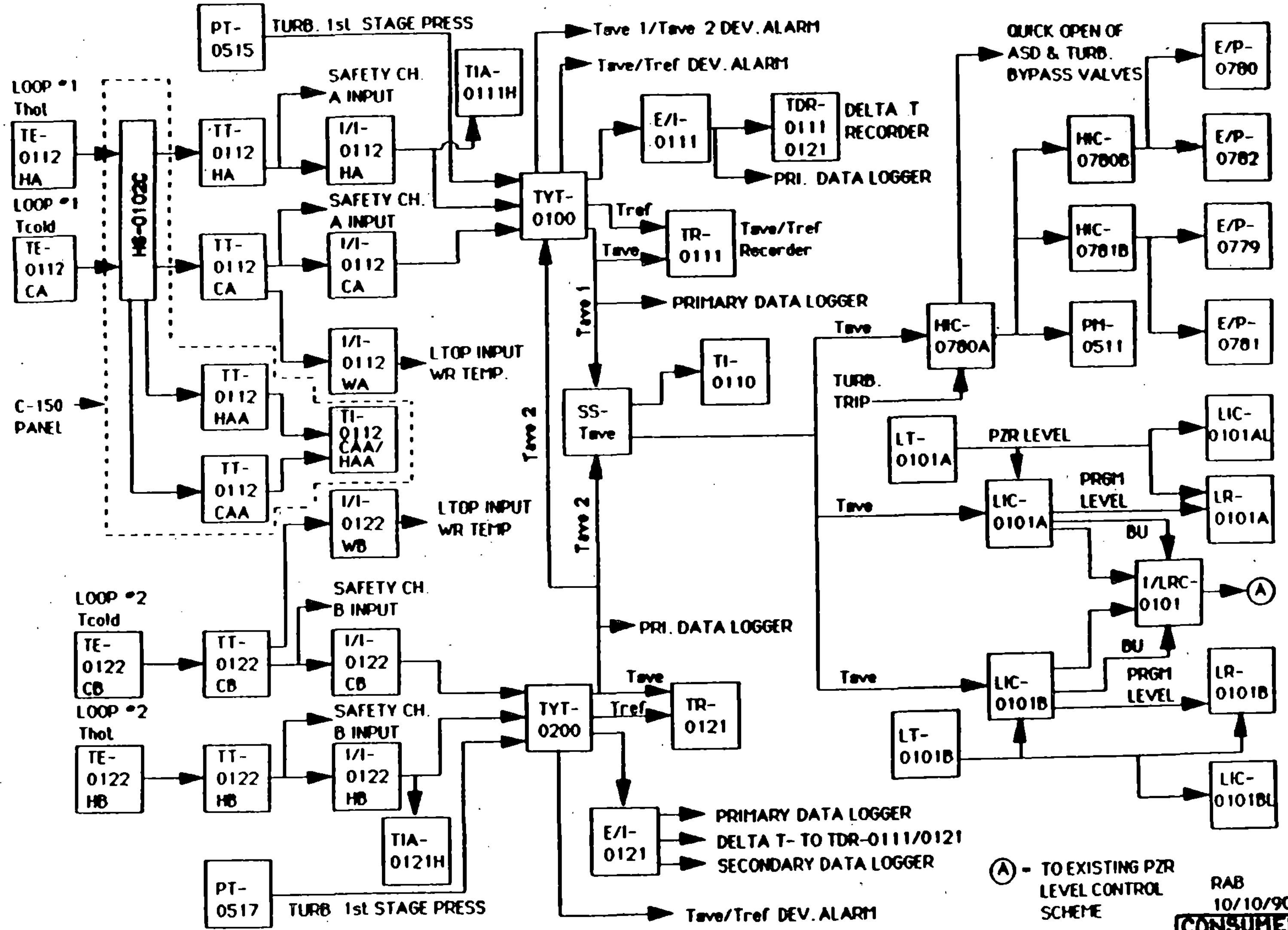


CONSUMERS POWER COMPANY
PALISADES PLANT
FSAR UPDATE

REACTOR SHUTDOWN CONTROLS

FIGURE NO. 7-54

REVISION NO 19

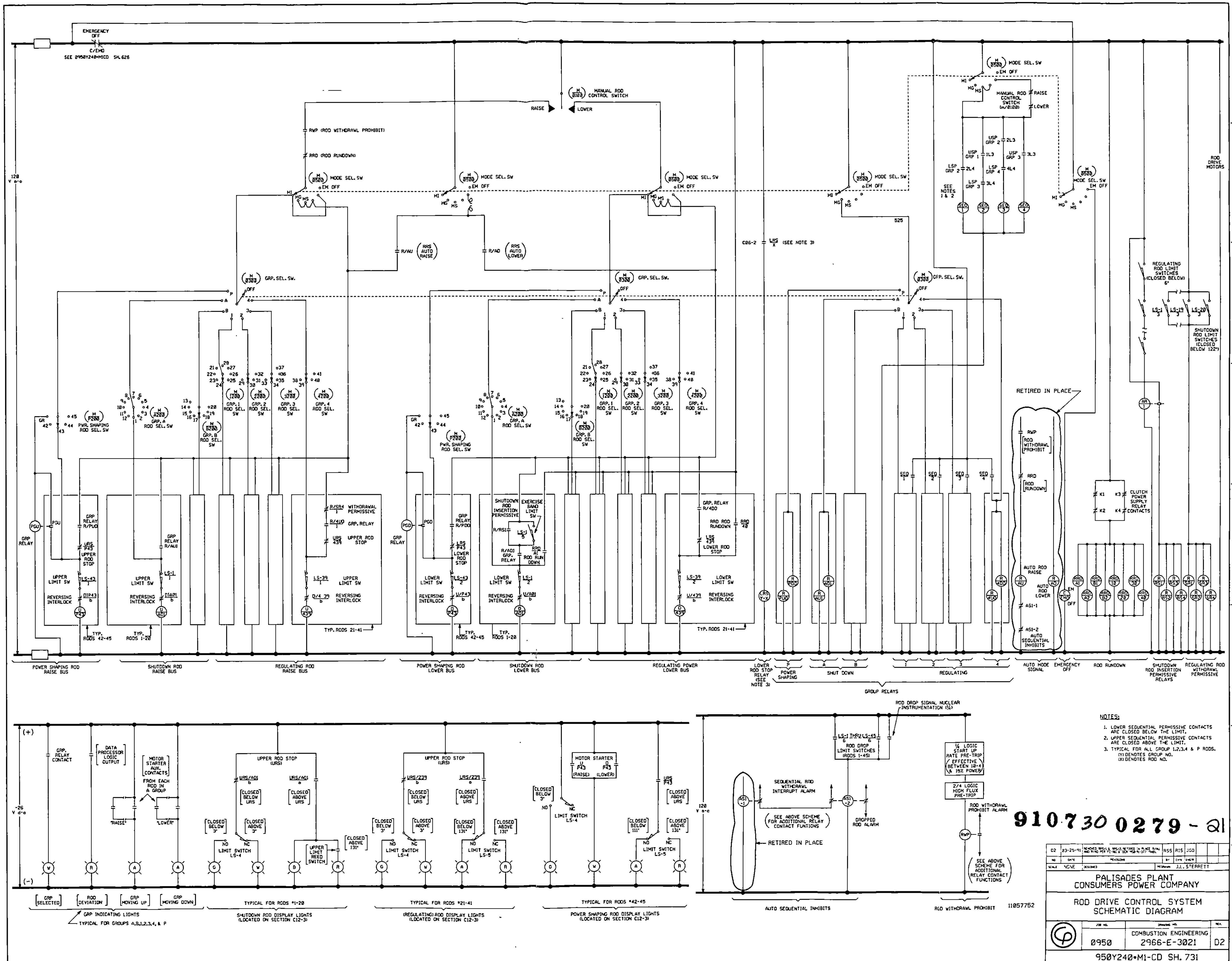


CONSUMERS POWER COMPANY
PALISADES PLANT
FSAR UPDATE

REACTOR REGULATING SYSTEM
BLOCK DIAGRAM

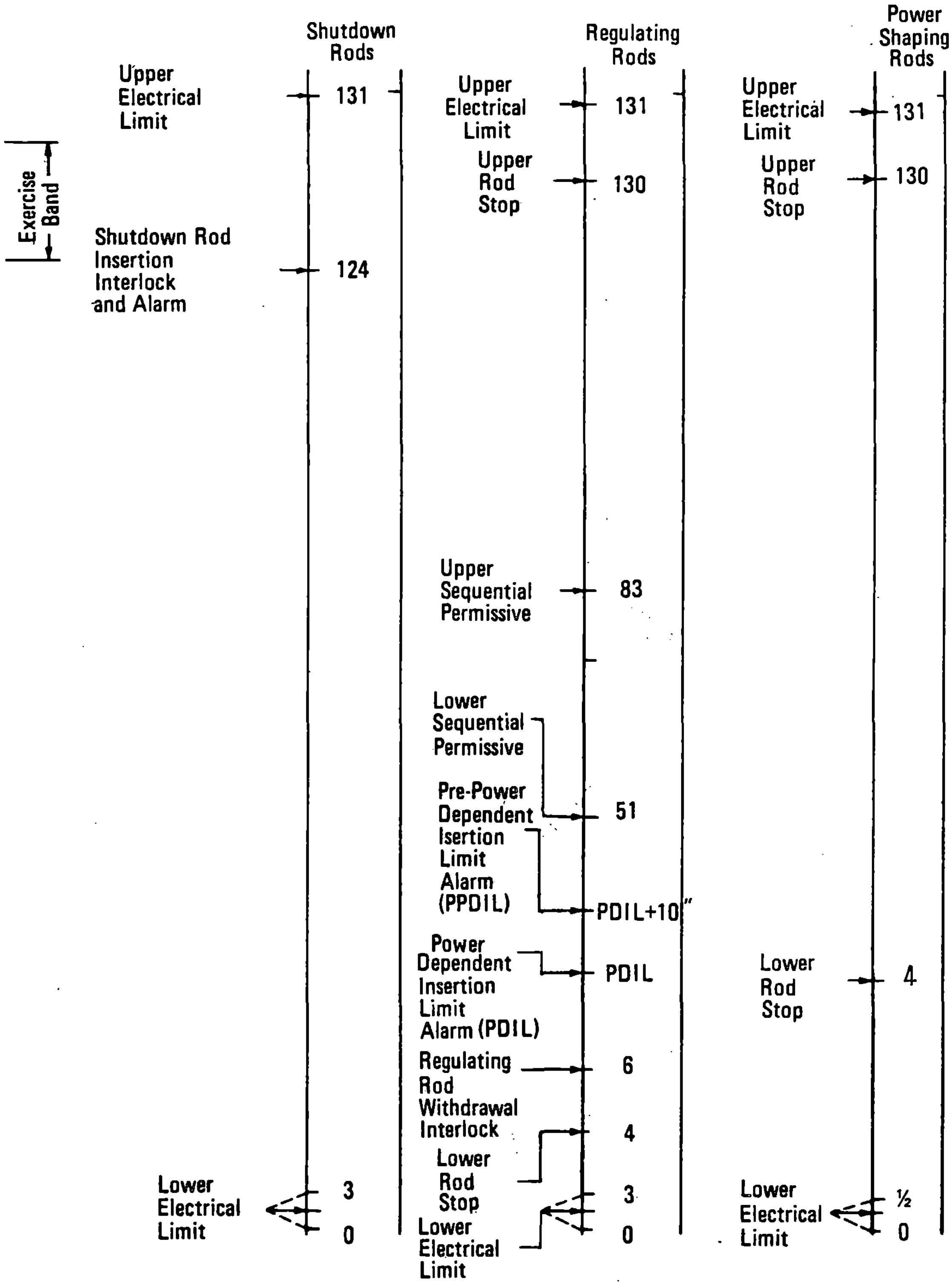
FIGURE NO. 7-55

REVISION NO. 12



9107300279-21

Figure No. 7-56 FSAR Rev 12



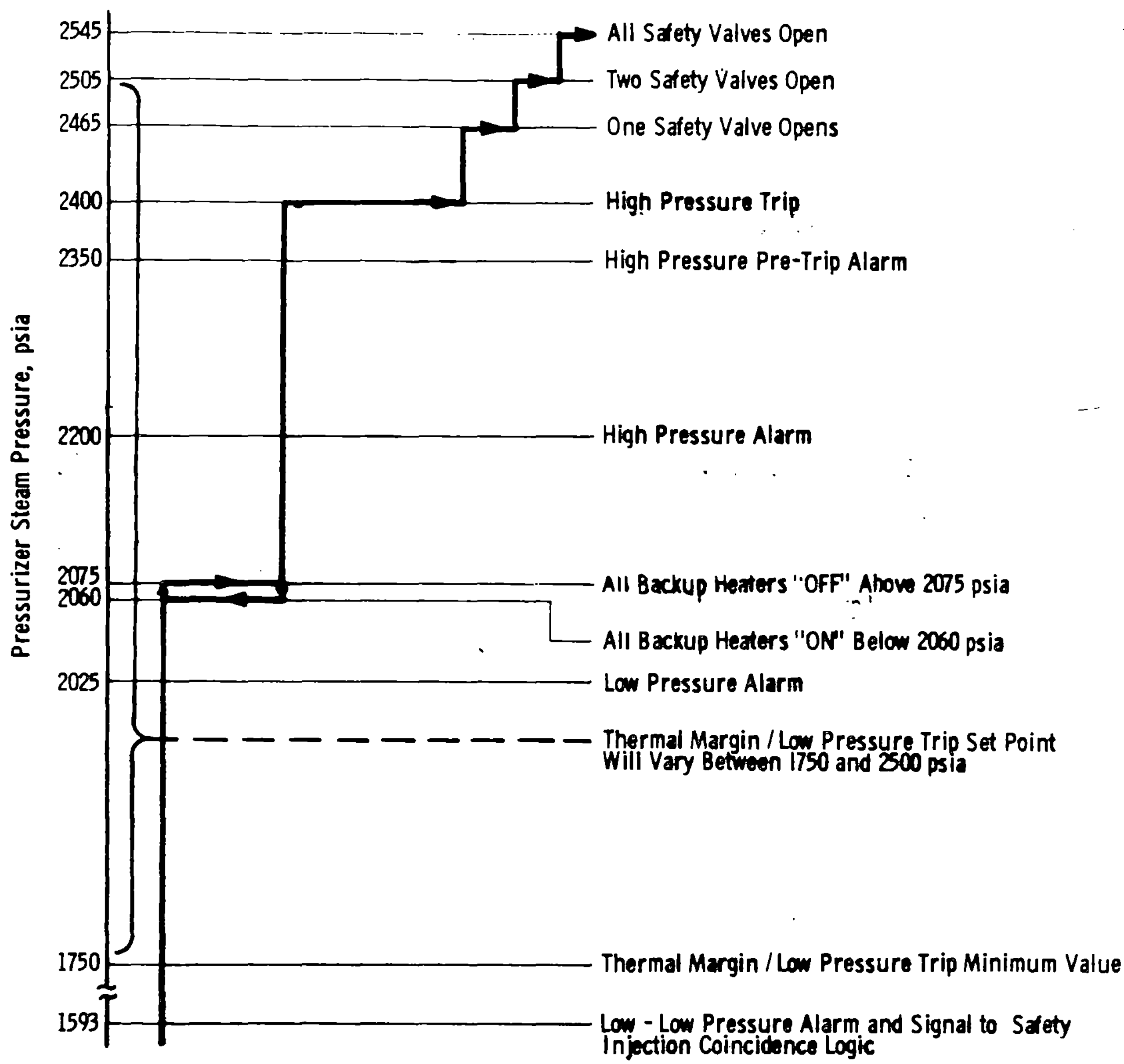
NOTE:
All Setpoint are in Inches From Bottom.

CONSUMERS POWER COMPANY
PALISADES PLANT
FSAR UPDATE

ROD POSITION SETPOINTS

FIGURE NO 7-57

REVISION NO 0



Pressurizer Normal Pressure Control

Controller Output	Action
100%	Both Spray Valves Open
67%	Both Spray Valves Closed
33%	Proportional Heater "OFF"
0%	Proportional Heater "On"

CONSUMERS POWER COMPANY
PALISADES PLANT
FSAR UPDATE

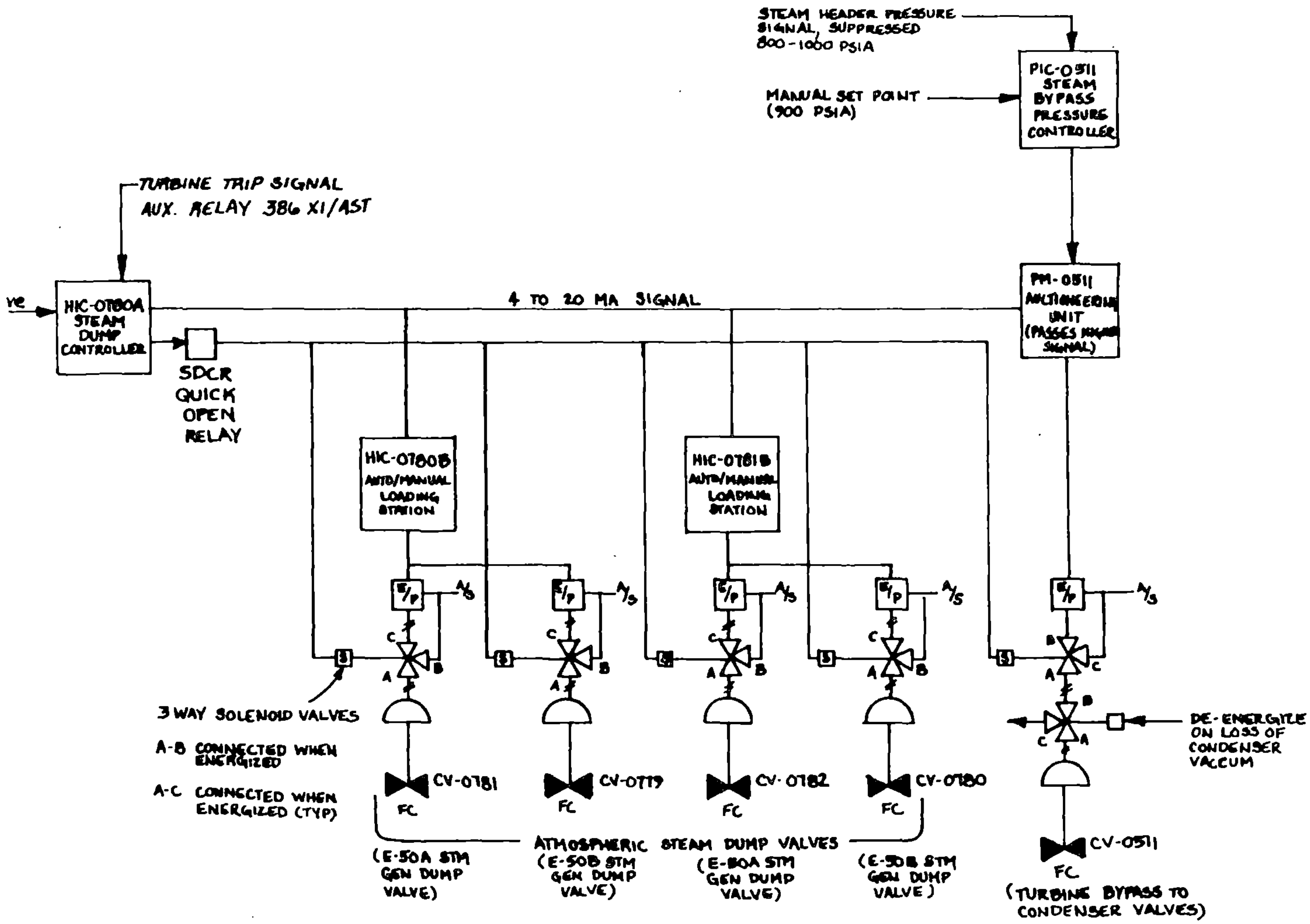
PRESSURE CONTROL PROGRAM

FIGURE NO 7-58

REVISION NO 12

FIGURE 7-59

(DELETED)

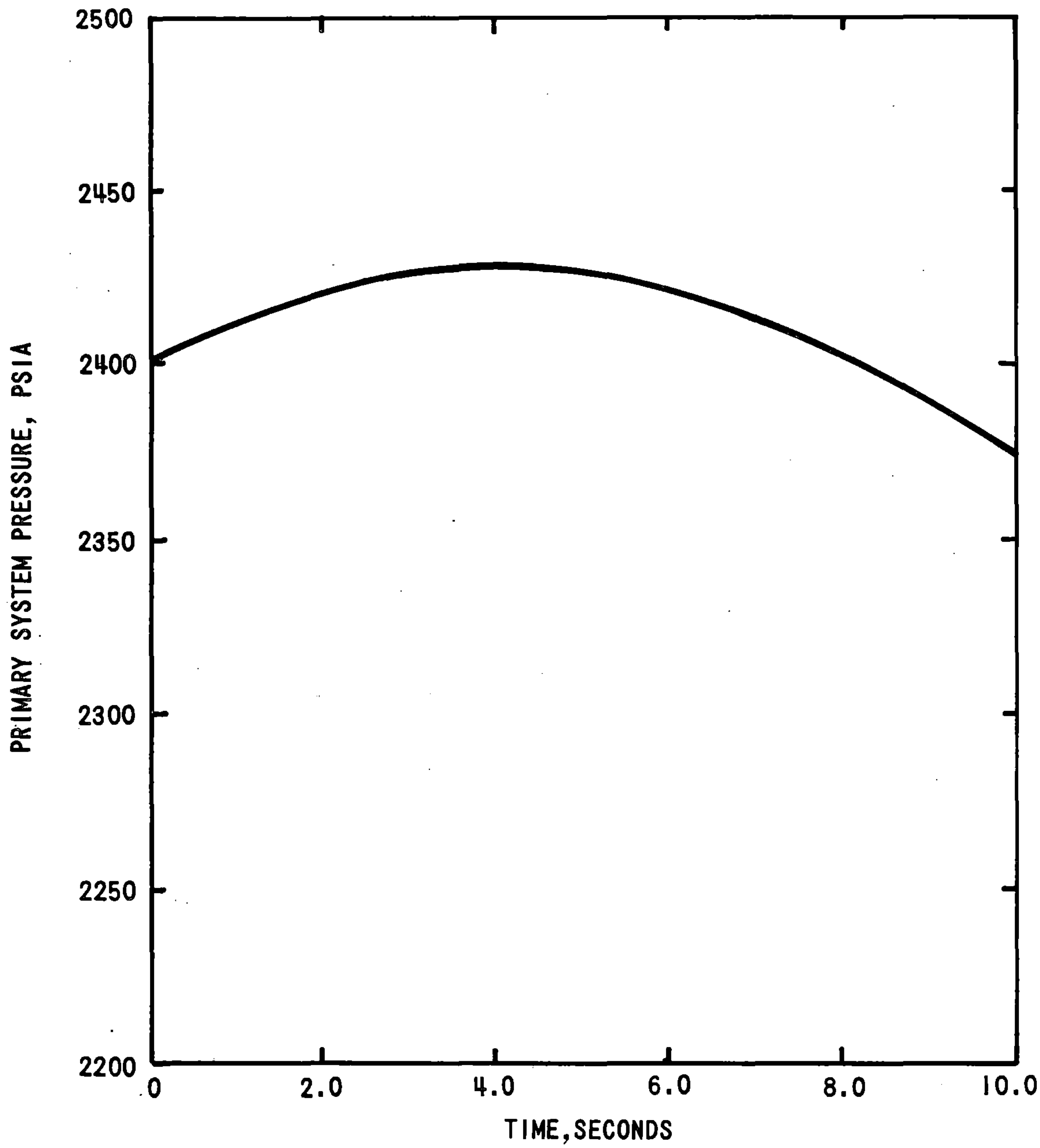


CONSUMERS POWER COMPANY
PALISADES PLANT
FSAR UPDATE

BLOCK DIAGRAM
STEAM DUMP AND BYPASS SYSTEM

FIGURE NO 7-60

REVISION NO 12



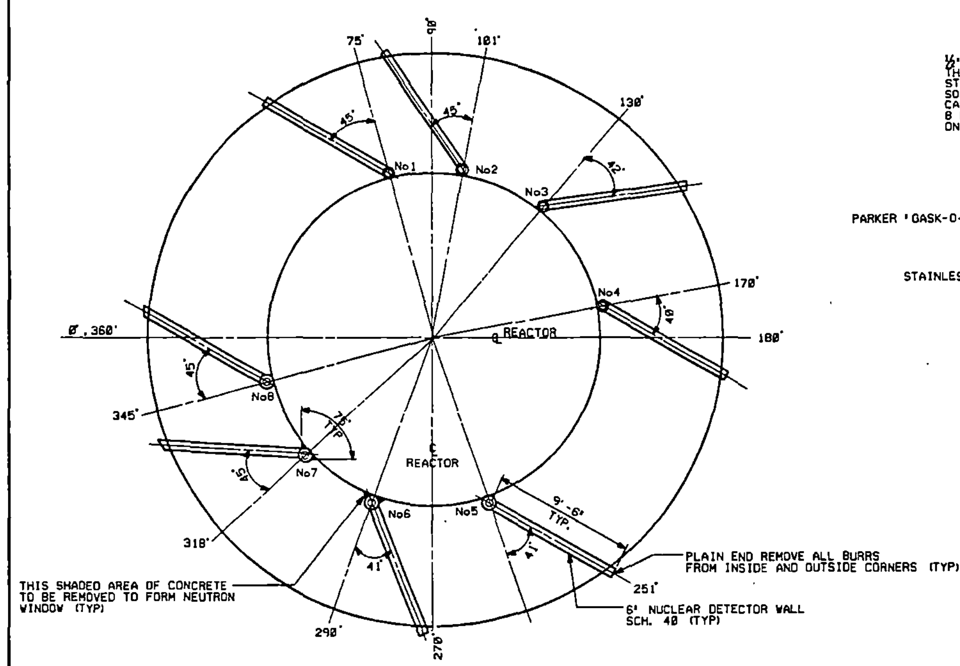
(REF P-ICE-900,10/9/69)

CONSUMERS POWER COMPANY
PALISADES PLANT
FSAR UPDATE

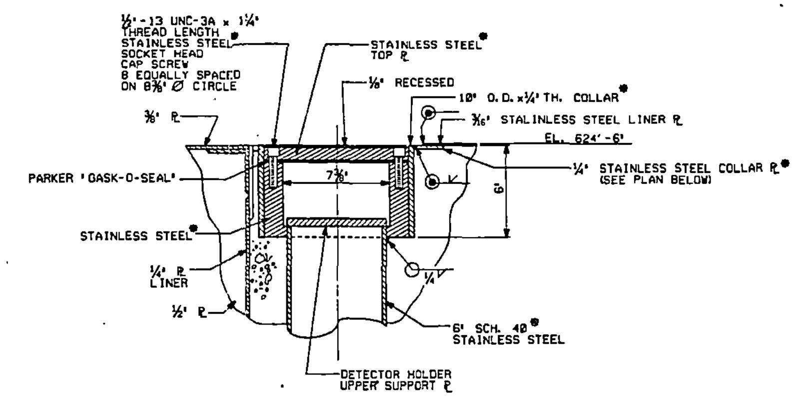
PRESSURIZER LEVEL CONTROL SYSTEM FAILURES STUDY MODE "A" FAILURE

FIGURE NO 7-61

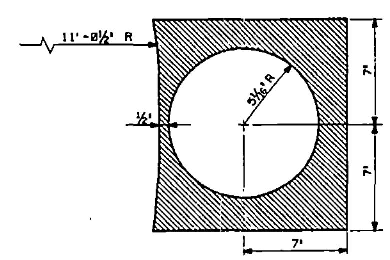
REVISION NO 0



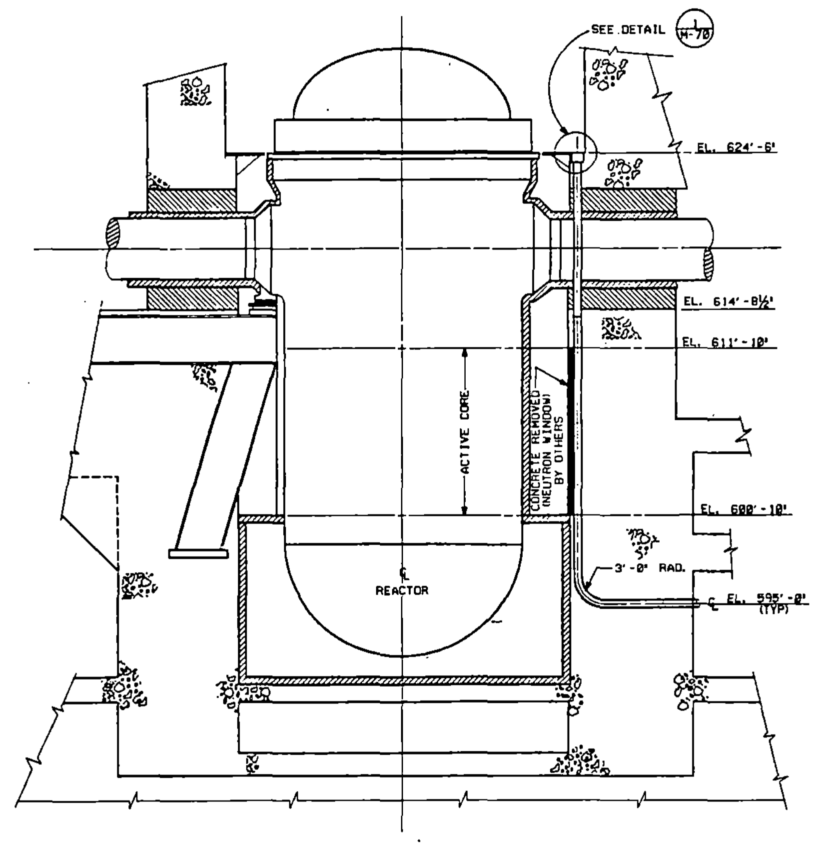
PLAN
SCALE: 1/4" = 1'-0"



DETAIL 1/M-70
SCALE: 3" = 1'-0"



PLAN *
(COLLAR PLATE)
SCALE: 3" = 1'0"



TYPICAL SECTION
SCALE: 1/4" = 1'-0"

NOTE:
* INDICATES MATERIAL SUPPLIED WITH DETECTOR TUBES

REFERENCE DRAWINGS:

1. FOR GENERAL ARRANGEMENT SEE DWG. SK-M-332
2. FOR PRIMARY SHIELD COOLING COILS SEE DWGS. M-71 AND M-72.
3. FOR LINER PLATE DETAILS SEE DWGS. C-92, C-153, C-154, AND C-155.

SI
APERTURE
CARD

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9005080040-18

NO	DATE	REVISIONS	BY	CHK	APP			
5	3-13-89	REVISED DETAIL 1 PER SC-87-064 & DCA 950-89-203 (TMP)	JSS	RIS	JGD			
4	5-31-85	REVISED PER FIELD COMMENTS FOR FSAR UPDATE	JLS	CWB	TDV	DLB		
3	11-18-83	REDRAWN PER FSAR UPDATE	PGB	JGD	TDV			

PALISADES PLANT
CONSUMERS POWER COMPANY

PIPING DRAWING
NUCLEAR DETECTOR WELLS

0950	M-70	5

0950M70.001

JS2053105

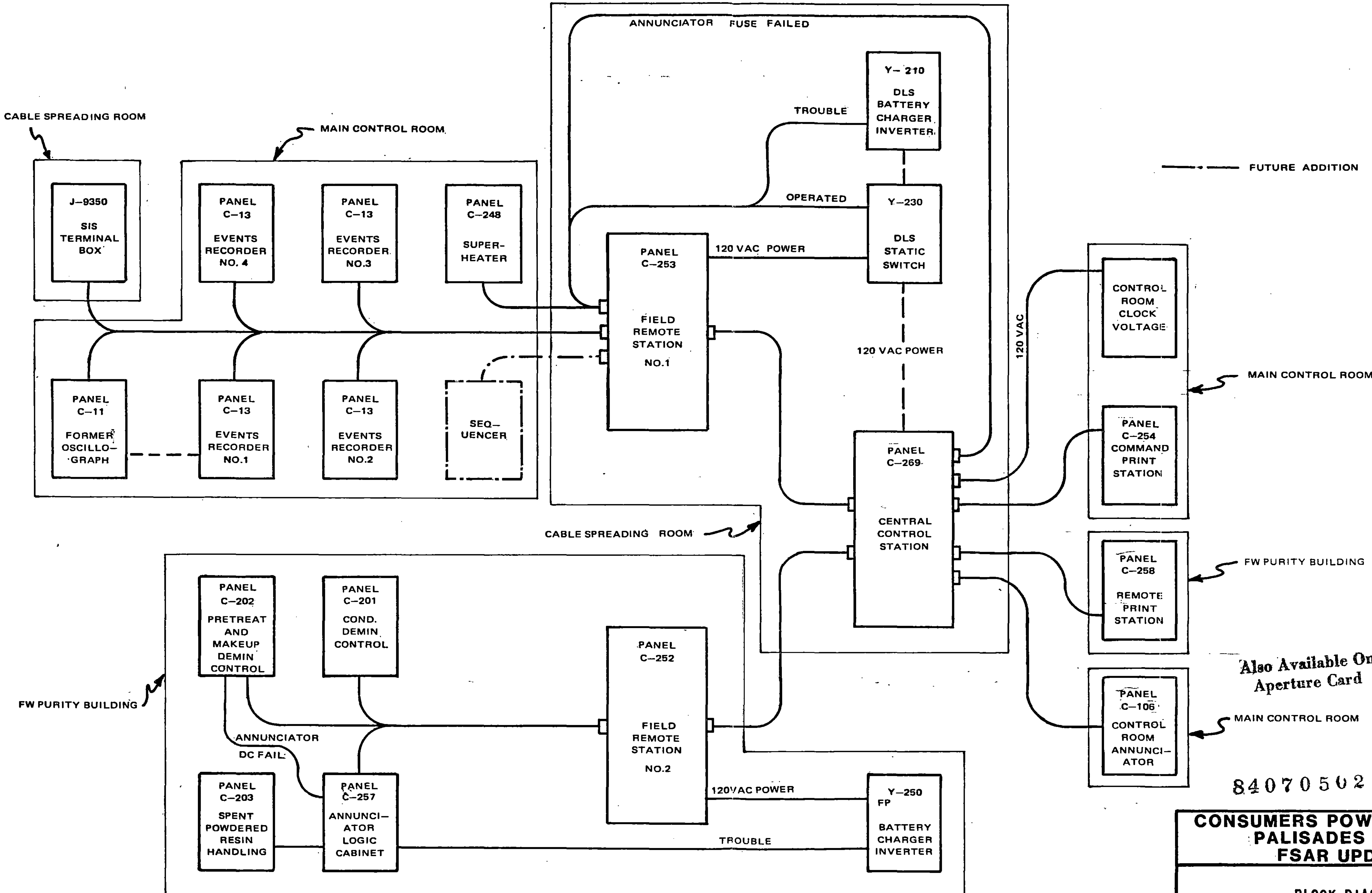
8 | 7 | 6 | 5 | 4 | 3

Figure No. 7-62

FSAR Rev. 10

FIGURE 7-63

(Deleted)



TI APERTURE CARD

Also Available On Aperture Card

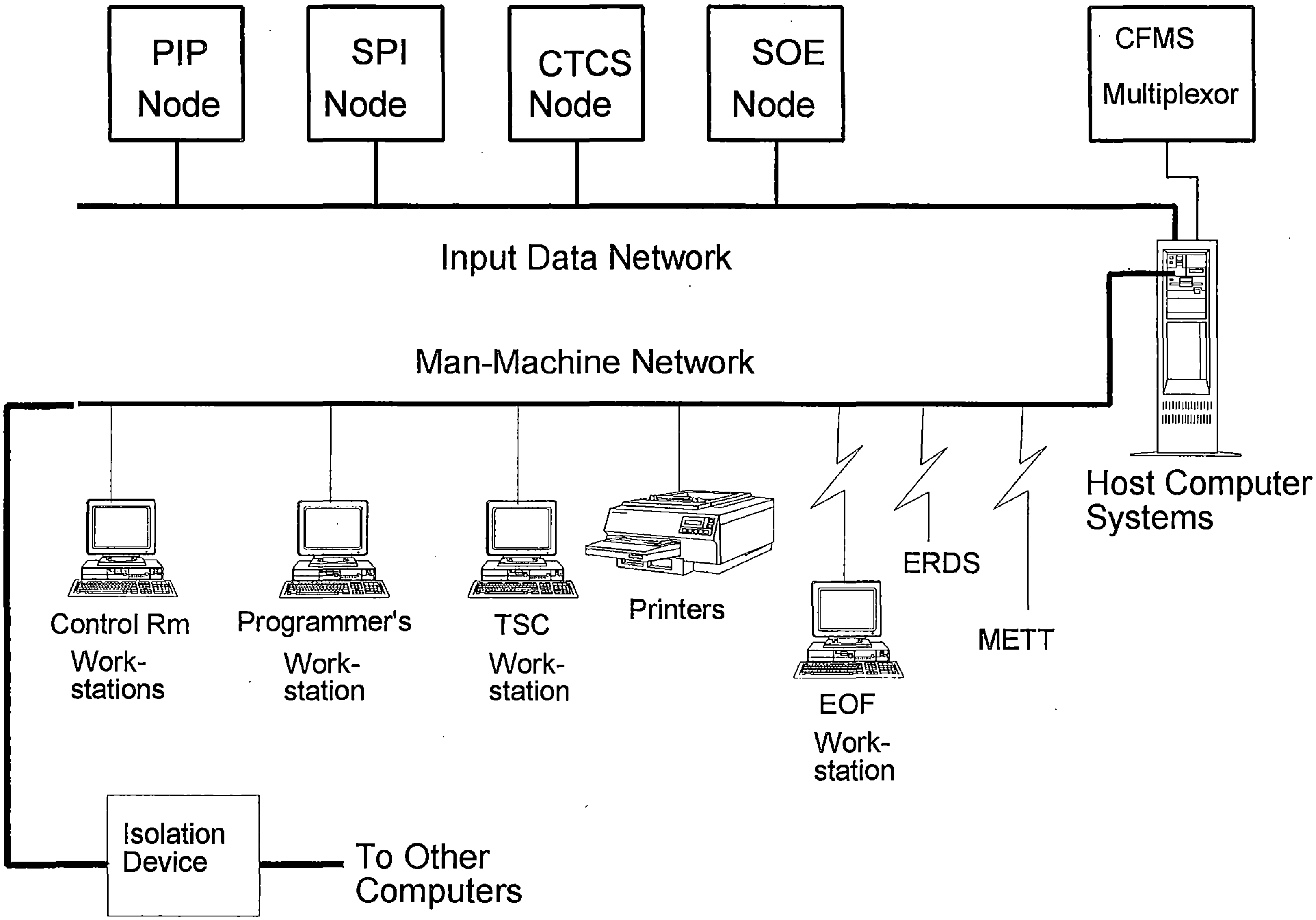
8407050272-55

CONSUMERS POWER COMPANY
PALISADES PLANT
FSAR UPDATE

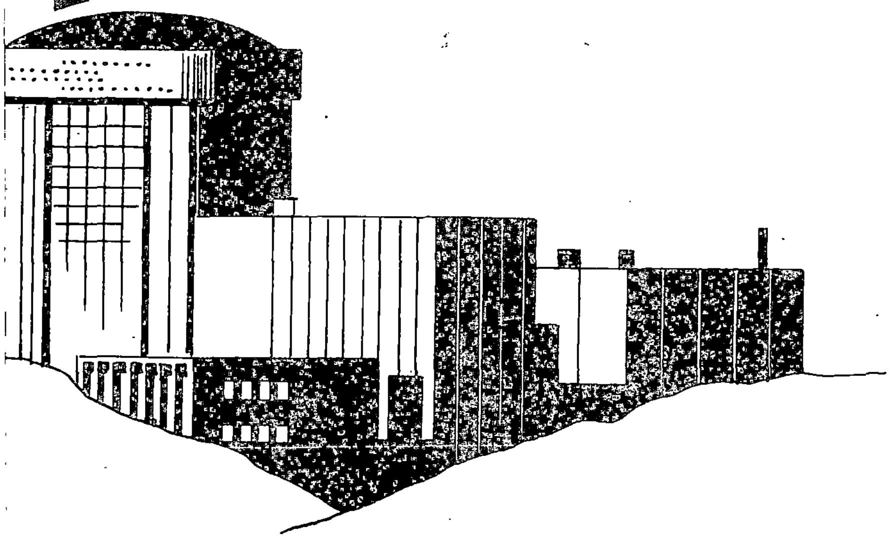
BLOCK DIAGRAM
DATA LOGGER SYSTEM

FIGURE NO 7-64

REVISION NO 0



Palisades Plant Computer Sys.
Figure 7-65 Rev. 20



·LISADES NUCLEAR PLANT

final safety analysis report

UPDATE

VOLUME 7

OTICE —

ARE OFFICIAL RECORDS OF THE
NT CONTROL. THEY HAVE BEEN
)R A LIMITED TIME PERIOD AND
) TO THE RECORDS FACILITY
SE DO NOT SEND DOCUMENTS
GH THE MAIL. REMOVAL OF ANY
/IENT FOR REPRODUCTION MUST
PERSONNEL. 50-255
6/28/84

\TE 8407060220

RECORDS FACILITY BRANCH

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VOLUME 2

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CHAPTER 4

PRIMARY COOLANT SYSTEM

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CHAPTER 8

ELECTRICAL SYSTEMS

8.1 INTRODUCTION

8.1.1 DESIGN BASIS

The Plant electrical system and the 345 kV switchyard are designed to reliably function and supply power during normal, abnormal and emergency conditions. This electrical power system is required to meet 10 CFR 50, Appendix A, General Design Criterion 17, for onsite and offsite power source requirements. The system will supply and distribute the electrical power necessary to operate the systems which preserve the Plant's three fission product barriers under all conditions of start-up, power generation and shutdown. The electrical system is divided into buses and subsystems to minimize the effects of any electrical fault and maximize the availability of onsite and offsite power sources.

The portion of the Plant electrical system which supplies the Plant engineered safeguards will be referred to as the engineered safeguards electrical system. The engineered safeguards electrical system is housed in CP Co Design Class 1 structures in accordance with Subsection 5.9.1.1 and provides a Class 1E service as defined in IEEE 308-1978.

The use of the term "Class 1E" in the FSAR, and elsewhere, means that the system or component meets the definition of 1E by providing a function listed in IEEE 308 as essential (under the "Class 1E" definition). Because the plant was designed and constructed prior to IEEE 308 requirements, some components/systems that are designated as Class 1E will not have all the necessary attributes to be certified as meeting Class 1E qualifications. Although original plant components/systems were designed and qualified (by analysis) to provide safety related functions, they may not meet all of the design criteria and testing requirements of IEEE 308 and other standards incorporated by reference therein.

The designation of these components/systems as 1E will assure that proper safety and quality reviews of function are performed prior to maintenance or replacement. When modifications are performed requiring equipment upgrades, change of function, etc, the new design is evaluated to determine whether upgrading to later standards including IEEE 308 would be practical and warranted. Systems and components originally designed and installed in conformance with 1E requirements will be maintained at least to those standards unless future changes in system/component function make classification as 1E no longer necessary.

The definition for safety related electrical equipment is the definition of equipment "important to safety" given in 10 CFR 50, Appendix A, when applied to the Palisades Plant. Class 1E equipment and systems are a subset of safety related equipment and systems.

Evaluation of safety-related electrical equipment (whether or not part of the Plant electrical system) for environmental and seismic service qualifications is discussed in Subsections 8.1.3 and 8.1.4.

Safety-related electrical equipment not part of the Plant electrical system is covered by Chapter 7.

The engineered safeguards electrical system consists of the necessary power sources, power distribution equipment, electrical and control subsystems, including cabling and raceway, to furnish power and controls to the required load groups upon a safety injection signal or containment isolation signal, to reliably shut down the reactor, and to remove reactor decay heat for long periods of time.

The system is designed to the two channel concept, defined as two independent electrical control and power systems supplying redundant engineered safeguards load groups. The engineered safeguards electrical system meets the single failure criteria, by which any single failure of a component within the system will not prevent the proper system action when required, and is intended to meet all other requirements identified in IEEE 279-1971 and IEEE 308-1978. Other standards such as IEEE 323-1974, IEEE 344-1975, IEEE 384-1977 and IEEE 383-1974 are intended to be met within the limits of practicability and consistent with original design features.

During the Systematic Evaluation Program, NRC and CPCo staff reviewed as built electrical design against licensing criteria current at the time. The purpose of the review was to determine if the designs in older plants provided a measure of safety comparable to that provided by design in newer plants; or as a minimum, were acceptable based on plant specific challenge and response capability. The adequacy of as built characteristics designed to promote safety related availability such as channel separation, isolation and independence was evaluated. In many cases, these reviews concluded that although the existing as built design did not feature the specific configuration required by current criteria, the design was considered acceptable. Summaries of these reviews are found in NUREG 0820, the NUREG 0820 Supplement, and individually docketed SEP submittals and related NRC Safety Evaluation Reports.

Separation, isolation and independence are discussed in Sections 7.4, 8.5, and 8.7. Physical separation specifically related to fire protection is described in Section 9.6.

Water spray fire protection is provided for raceways in areas described in Section 9.6.9.

8.1.2 DESCRIPTION AND OPERATION

The Plant electrical system is shown on Figure 8-1, Plant Single Line Diagrams. The 345 kV switchyard is shown on Figures 8-2 and 8-3, Substation Single Line Diagrams.

Six transmission circuits connect the site switchyard to the power system grid with two circuits on each of three sets of towers. The site switchyard is connected to the Plant electrical system using three circuits; one underground and two overhead. The Plant turbine generator is connected to the switchyard through Main Transformer 1 using one of the overhead circuits. The underground and other overhead circuits supply offsite power to the Plant electrical system.

The normal power source for the Plant 4,160-volt auxiliaries is the site turbine generator which supplies the reactor plant primary coolant pumps and condensate pumps via 21 - 4.16 kV Station Power Transformer 1-1. Also, the cooling towers and associated equipment loads are supplied via the 345 - 4.16 kV Station Power Transformer 1-3. When the turbine generator is out of service, offsite power is supplied to the primary coolant pumps and condensate pumps via 345 - 4.16 kV Start-Up Transformers 1-1 and 1-3. If the turbine generator is out of service for an extended period, the generator isophase bus motor-operated disconnect switch may be opened and Main Transformer 1 can be placed in service to supply 4,160-volt auxiliary power through the station power transformers.

2,400-volt reactor and turbine plant loads, including the engineered safeguards electric system, are normally supplied from either the offsite power source 345 - 2.4 kV Safeguard Transformer 1-1 or the offsite power source 345 - 2.4 kV Start-Up Transformer 1-2. Capability is also provided to power the 2,400-volt electrical system from the main turbine generator via 21 - 2.4 kV Station Power Transformer 1-2. If the turbine generator is out of service for an extended time, the generator isophase bus motor operated disconnect switch may be opened and Main Transformer 1 can be placed in service to supply 2,400 volt auxiliary power through Station Power Transformer 1-2.

The Plant auxiliary electrical system includes four 4,160 volt buses, four 2,400 volt buses, and several 480 volt load centers and 480 volt motor control centers. Certain 480 volt load center buses receive power from load center transformers energized from the 2,400 volt buses. Other 480 volt load center buses receive power from load center transformers energized from 4,160 volt Buses 1A, 1F and 1G. The 480 volt motor control centers are connected to the 480 volt load center buses.

Four preferred ac systems are energized from the station battery systems through inverters to power vital instrument and control loads. Two station batteries and four chargers supply the Plant 125 volt dc systems.

The nonvital instrumentation and controls are supplied from a 120 volt ac instrument bus. The instrument bus is normally supplied from one of two 480-120 volt transformers, each transformer being connected to a separate 480 volt motor control center. The transfer to the alternate source is automatic.

The engineered safeguards system includes two 2,400 volt buses (1C and 1D), four 480 volt load centers (11, 12, 19 and 20), eight motor control centers (1, 2, 21, 22, 23, 24, 25 and 26), two dc distribution centers, four battery chargers, two batteries, four preferred ac buses, four inverters and two diesel generators. The engineered safeguards electrical system is designed on a two-independent-channel basis. Each channel is capable of furnishing power to equipment load groups which meet the minimum requirements to safely shut down the reactor and is capable of providing sufficient electrical power for all functions necessary to operate the systems associated with the Plant's capability to cope with abnormal events. The system is provided with the necessary redundant circuitry and physical isolation so that a single failure within the system will not prevent the proper system action when required. The system is provided with test facilities and has alarms to alert the operator when certain components trip or malfunction or are not available or operable. Automatic controls and interlocks provide the proper sequence of operation of the engineered safeguards components with or without normal or offsite power.

Each channel of the engineered safeguards electrical system has access to the following power sources:

1. The offsite power source (Safeguard Transformer 1-1)

2. The offsite power source (Start-Up Transformer 1-2)

3. The onsite power source (Main Turbine Generator)

4. The emergency power source (one of two onsite emergency generators)

Each emergency generator may energize only its respective channel buses.

8.1.3 ENVIRONMENTAL QUALIFICATION OF ELECTRICAL EQUIPMENT

An order modifying the Palisades Plant Operating License, requiring submittal of environmental qualification information for safety-related electrical equipment, was issued by the NRC on August 29, 1980 (see Reference 1). The order was revised and reissued on September 19, 1980 (see Reference 2). In response to the Commission's orders, Consumers Power Company submitted a report entitled, "Environmental Qualification of Safety-Related Electrical Equipment - Palisades Plant, September 1980" (see Reference 3).

The report described the Electrical Equipment Qualification (EEQ) program and identified electrical equipment needed to mitigate the effects of an LOCA or main steam line break (MSLB) inside of containment or an MSLB outside of containment, and which are in areas that will experience the direct effects (high humidity, high temperature, high radiation, etc) of these accidents. The report lists the equipment and describes the expected post-accident conditions, describes the expected duration of the hostile environment, describes the expected effects on the affected equipment, and describes the method of qualification for each piece of equipment. Subsequent to the initial report, revisions to the report and additional information on the EEQ program were submitted to the NRC (see References 4, 5 and 6).

On April 25, 1983, the NRC issued its Safety Evaluation Report (SER) for the Palisades EEQ program (see Reference 7). The SER also contained a Technical Evaluation Report (TER) prepared by the Franklin Research Institute for the NRC. Consumers Power Company responded to the open items of the SER on May 20, 1983 (see Reference 8).

On February 14, 1984 (see Reference 9), Consumers Power Company submitted additional information in response to the Safety Evaluation Report (SER). The additional information responded to environmental qualification documentation deficiencies for safety-related equipment at Palisades noted in the TER and, in addition, the letter documented discussions held on January 10, 1984 with the NRC Staff regarding the methodology to be used by CP Co in complying with 10 CFR 50.49 which had become effective February 22, 1983.

On February 28, 1986, the NRC issued its Safety Evaluation Report for the Single Failure issue for the main steam isolation valves and feedwater isolation valves. The SER covers the concern that a main steam line break (MSLB) along with the single failure of one main steam isolation valve would allow both steam generators to blow down. In this review the NRC concluded that the Plant emergency operating procedures and training on the issue provide sufficient protection. They also concluded that the qualification of components inside containment for the temperature/pressure conditions resulting from a postulated double steam generator blowdown is not required. See Chapter 5 for additional information.

The present EEQ program involves the generation and maintenance of evidence that electrical equipment can perform its intended function throughout its installed life when exposed to harsh environmental conditions.

An EEQ Engineer ensures that new equipment is properly qualified, existing equipment qualification is properly maintained, and qualification files are accurate and auditable.

Vendor file E-48 contains a current list of EEQ equipment and environments. For each item a reference is provided to a vendor file containing the qualification report. EEQ file reports are developed by an EEQ Engineer using information provided by others. All changes to EEQ files are justified by engineering analyses and reviewed by an EEQ Engineer.

8.1.4 SEISMIC QUALIFICATION OF ELECTRICAL EQUIPMENT

The seismic design criteria for safety-related electrical equipment, instrumentation and raceways are provided in Section 5.7. Seismic Category I (Regulatory Guide 1.29) and Class 1E electrical equipment and raceways are listed in Table 5.2-4. Electrical equipment anchorage and raceway supports for the components listed in that table have been redesigned in the period 1979 to 1981 as Seismic Category I as defined in Regulatory Guide 1.29. Seismic adequacy of safety-related electrical components is determined by resolution of NRC Generic Letter 87-02 "Verification of Seismic Adequacy of Mechanical and Electrical Equipment in Operating Reactors, Unresolved Safety Issue A-46".

8.1.5 STATION BLACKOUT

In 10 CFR 50.63 the NRC defined the loss of all onsite and offsite ac power sources (station blackout) as an event with which all plants are required to cope. Such factors as redundancy of offsite power sources, severe weather potential of the site and diesel generator reliability were evaluated in accordance with NUMARC 87-00 guidance to define minimum battery capacity and instrumentation requirements. On May 20, 1991 the NRC found that the Palisades plant conformed with the SBO (Station Blackout) rule, Regulatory Guide 1.155, NUMARC 87-00, and NUMARC 87-00 Supplemental Questions/Answers and Major Assumptions.

Per NRC recommendations, load stripping of the station batteries is not initiated until 30 minutes after SBO. Actions necessary to isolate containment were identified, and an EDG reliability program (RG 1.155 Section 1.2) was developed. Evaluations and commitments to evaluate HVAC and heat tracing were approved by the NRC per the G.B.Slade, CPCo, letter of August 1, 1991. The addition of backup nitrogen to the atmospheric dump valves provides a minimum coping duration of four hours for SBO. These actions resolved recommendations made in the May 20, 1991 SER. This resolution was documented in a SER of June 25, 1992.

8.2 NETWORK INTERCONNECTION

8.2.1 DESIGN BASIS

The 345 kV switchyard is designed to be the interconnection point between the power plant and the power grid system which, in turn, interconnects the Michigan Power Pool with the American Electric Power Company's system grid. It is designed to function reliably under all conditions of power plant operation. It will furnish start-up power to the power plant, and reliably function and isolate trouble in the power system grid under power system normal and abnormal conditions.

High-speed clearing of faults and selective reclosing assure maximum availability of power and system grid stability. A joint Michigan Power Pool-American Electric Power transient stability study has been made on the Palisades Plant. The results of this study indicate that no system instability will result from the loss of the Palisades Plant. The sudden drop of a large load will not adversely affect the Palisades plant or the connected electrical system. Analyses demonstrate that the electric system can sustain contingencies equivalent to, or more severe than, an abrupt 1,000 MW loss of load without adverse effects to the system or connected units.

8.2.2 DESCRIPTION AND OPERATION

Description - Switchyard - The switchyard system is shown on Figures 8-2 and 8-4.

The switchyard operates at 345 kV and is arranged to give maximum availability of the power system grid. The equipment is selected to have the capability of isolating system and substation faults with a minimum effect on stability of the power system grid.

The switchyard is designed in a breaker-and-one-half arrangement with two main buses and connections for the generator main power transformer, the Plant safeguard transformer, the Plant start-up transformers, and six outgoing lines. Two of the outgoing lines are connected to the American Electric Power system and the remaining lines are connected to the Consumers Power Company system and the Michigan Power Pool. Each line has sufficient capacity to carry the entire output of the main turbine generator.

The switchyard 345 kV power circuit breakers, the circuit from the switchyard to the generator main power transformer, the circuit for the safeguard transformer, and the circuit from the switchyard to the start-up transformers are provided with disconnect switches to permit isolating any power circuit breaker or any circuit from the switchyard buses. See Table 8-1 for ratings and construction of the switchyard components.

High-speed relaying is provided for the circuit from the switchyard to the generator main power transformer and for the two switchyard main buses ("F" bus and "R" bus). The "R" bus relaying includes the circuit from the switchyard to the start-up transformers. The "F" bus relaying includes the circuit to the safeguard transformer. The six outgoing lines are each provided with high-speed relays. In addition, all 345 kV power circuit breakers are provided with relays to trip all adjacent breakers for a failed breaker condition.

Trip of both main 345 kV Plant output circuit breakers in the switchyard activates a main turbine generator trip which indirectly trips the reactor via the Reactor Protective System loss-of-load reactor trip channels when the Plant is above 15% power (Subsection 7.2.3.6). The turbine trip output also feeds transfer schemes for fast transfer of Plant distribution busses to standby power sources. This output is blocked for each standby power source if it is unavailable as sensed by transformer undervoltage (see Section 8.6).

<u>Description - Switchyard Control System</u> - The 240 volt and 120 volt, 60 hertz and 125 volt dc switchyard power supplies are shown on Substation Single Line Diagram, Figure 8-4. 2,400 volt Buses 1C and 1E supply the switchyard control power through two 2,400-240/120 volt, 60 hertz switchyard power transformers. Each of the transformers supplies half the 240/120 volt, 60 hertz power requirements for the switchyard; however, either transformer can be connected to carry the total load via a bus tie breaker. The ac load is divided among four power panels; the loss of one power panel will not affect operation of the other three and hence will not jeopardize the total 240/120 volt, 60 hertz auxiliary power in the switchyard. The 345 kV power circuit breakers have enough air stored in their high-pressure receivers to permit five breaker operations. A high-pressure hose may be used to cross-connect receivers upon loss of an air compressor and a spare compressor has been permanently installed in the switchyard as a backup supply via the high-pressure hose.

The 125 volt dc auxiliary power is supplied from a 60-cell battery which is located in the switchyard and can supply the switchyard dc power requirements for eight hours without recharging.

Two battery chargers are supplied to keep the battery fully charged and, under normal conditions, to supply the 125 volt dc power requirement. The battery circuit breaker can be used to isolate the battery from the dc power panels in event of a battery fault. The power panels can then be energized by either or both battery chargers. Each battery charger is fed from a separate 240 volt, 60 hertz power panel. The dc load is divided between two power panels; the loss of one power panel will not disrupt all the 125 volt dc auxiliary power in the switchyard.

Normal Operation - The switchyard normally operates energized with all breakers closed. Opening and closing of the line breakers can be accomplished locally in the switchyard relay house or remotely from the Plant main control room. The main generator breakers are normally controlled remotely from the Plant main control room; in an emergency, they can be controlled from the switchyard relay house. A supervisory panel in the Plant main control room monitors circuit breaker status.

Shutdown Operation - The switchyard furnishes power to the Plant whenever the turbine generator is not available.

Testing - The power circuit breakers may be removed from service and tested. Individual components and partial circuit tests may be carried out while the circuit breakers are carrying load.

The relays are supplied with test switches that will permit the removal of one relay or one set of relays from service for maintenance at any time. Because of the redundancy in the relay circuits, the power circuit will still be relay protected.

8.2.3 DESIGN ANALYSIS

The ratings and components of the 345 kV switchyard have been selected to ensure that conservative margins have been allowed between the maximum expected fault duty and the rating of the equipment.

Reliability is assured by the arrangement of the switchyard which utilizes the breaker-and-one-half scheme. With this scheme, any breaker may be removed from service without affecting the operation of the switchyard.

The switchyard is arranged so that parallel outgoing overhead line circuits are connected to different bays and staggered with respect to buses. If a bay or bus has to be removed from service, only a partial loss of capacity occurs.

The six transmission lines connecting the grid network to the switchyard are routed on three double-line tower poles. Each of the three double-line poles is routed on separate rights of way so that no single event such as pole falling or line breaking can simultaneously affect all lines in such a way that none of the lines can be returned to service in time to prevent fuel design limits or design conditions of the reactor coolant pressure boundary from being exceeded. See Figure 2-2 for layout of the transmission lines in the vicinity of the Plant.

All circuits or portions of the buses and overhead lines have primary and backup relaying. The outgoing lines have three sets of high-speed relays. The circuit breakers have dual trip coils on separate dc control circuits, and breaker failure relays to trip the adjacent breakers. The two redundant control circuits will operate even with one set of relays out of service.

At power levels below 15%, a loss of load without generator trip, can be tolerated without subsequent reactor trip. Bypass of the turbine generator trip signal below 15% power allows for initial synchronization to the transmission grid. Also, since station electrical loads are not being fed from the main generator, Plant auxiliaries are not affected if the generator is separated from the switchyard.

Reliability of the control circuit supply is assured by two full-size transformers, each capable of carrying the total load, and one capable of being fed from a diesel generator.

In the terminology of 10 CFR 50, Appendix A, GDC 17, the three circuits connecting the switchyard to the onsite Class 1E power system consists of two immediate and one delayed access circuit. One immediate access circuit consists of a 345 kV to 2,400 volt safeguard transformer, 2,400 volt underground cable and 2,400 volt bus, and cable. The other immediate access circuit consists of an overhead 345 kV transmission line, 345 kV to 2,400 volt start-up transformer and 2,400 volt bus and cable. The delayed access circuit consists of one overhead 345 kV transmission line, 345 kV to 23 kV main transformer, 22 kV isophase bus duct, 24 kV to 2,400 volt station power transformer and 2,400 volt bus, bus duct and cable. Each of the immediate access circuits is connected directly to one of the main switchyard buses whereas the delayed access circuit is capable of being powered from either bus or directly by one of the six incoming lines from the grid.

The 2,400 volt cables associated with each immediate access circuit are routed in physically separated locations to the Class 1E onsite power system. The physical separation between these two routes is such that no single event can simultaneously affect both circuits in such a way that neither can be returned to service in time to prevent fuel design limits or design conditions of the reactor coolant pressure boundary from being exceeded.

The delayed access circuit is established by opening the motor-operated disconnect switch in the isophase bus at the main generator (to establish the main transformer backfeed mode of operation). In accordance with 10 CFR 50, Appendix A, General Design Criterion 17, this delayed circuit must be designed to be available in sufficient time, following a loss of all onsite ac power supplies and the other offsite immediate access circuit, to assure that specified acceptable fuel design limits and design conditions of the primary coolant pressure boundary are not exceeded. Without onsite ac power to operate its motor, the time required to open the isophase bus disconnect switch with a hand crank is less than 30 minutes. The dc battery system is designed to supply the required shutdown loads, with total loss of ac power, for at least 4 hours (see Subsection 8.4.2.3).

8.3 STATION DISTRIBUTION

8.3.1 4,160 VOLT SYSTEM

8.3.1.1 Design Basis

The 4,160 volt system is designed to reliably function and supply power during normal, abnormal and accident conditions to the 4,160 volt station auxiliaries. The system will supply and distribute the 4,160 volt power for the primary coolant and the condensate pumps from either the station power or start-up transformers. Cooling tower 4,160 volt power is supplied normally from 1-3 station power transformer with backup power from the Start-Up Transformers 1-1 and 1-3.

8.3.1.2 Description and Operation

Description - The Plant's 4,160 volt system is shown on Figures 8-1 and 8-5.

The 4,160 volt system consists of Station Power Transformers 1-1 and 1-3, Start-Up (Standby) Transformers 1-1 and 1-3, four 4,160 volt Buses 1A, 1B, 1F, 1G and the incoming and motor feeder circuits.

The 4,160 volt system is divided into four sections. Buses 1A and 1B each supply two of the four primary coolant pumps and one of the two condensate pumps. Buses 1F and 1G each supply electrical load to one cooling tower.

The 4,160 volt buses consist of metal-clad switchgear with drawout air circuit breakers. All wiring in the switchgear and all system interconnecting cables pass the vertical flame resistance test in accordance with ASTM D470-59T. New installations will meet an equivalent flame test (eg, IEEE 383-1974, Section 2.5, ICEA-S-19-81, Section 6.19.6, etc). See Table 8-2 for ratings and construction of the 4,160 volt system components.

The 4,160 volt switchgear is provided with relay protection, grounding and the mechanical safeguards necessary to assure adequate personnel protection and to prevent or limit equipment damage during system fault conditions.

The 4,160 volt system has sufficient capacity to start and accelerate the largest motor when all other required motors are energized and carrying load.

Normal and Shutdown Operation - Power to Buses 1A, 1B, 1F and 1G during normal operation is furnished from the main generator via Station Power Transformers 1-1 and 1-3. During start-up or shutdown, power is furnished from the power system grid via the switchyard and Start-Up Transformers 1-1 and 1-3.

If the standby power source is not available, power may be furnished by the main transformer backfeeding the Plant auxiliary buses. This mode of operation is allowed during Plant shutdown.

After the main turbine generator is started, and when reactor power is approximately 20%, Buses 1A, 1B, 1F and 1G are manually transferred from the start-up transformers to the station power transformer. During normal operation, all 4,160 volt buses are energized.

Following a turbine trip, the 4,160 volt Buses 1A, 1B, 1F and 1G will automatically transfer to the standby source and all auxiliaries will continue to run. If the trip is accompanied by a failure in the standby source, the turbine generator will supply power to the primary coolant pumps for 10 seconds which is controlled by a fixed timer (362/CD). Refer to Section 8.6 for 4,160 volt bus automatic fast transfer features.

When the main turbine generator is to be shut down, 4,160 volt Buses 1A, 1B, 1F and 1G are manually transferred to the start-up source. When the buses are transferred, all loads continue to operate as required.

Operation of all 4,160 volt equipment is effected and monitored in the control room. Breaker status is indicated by red and green indicating lights. Electrical parameters, such as bus voltage, supply voltage, incoming amperes and motor amperes, are displayed in the control room. Important functions, such as incoming breaker trip, motor breaker trip, bus undervoltage, and failure of bus transfer, are annunciated in the control room.

Provisions are included for testing relays. A faulty relay may be bypassed to permit operation of equipment.

Testing - Components may be tested when the system is de-energized. Testing of parts of the system can be performed when the system is in operation and carrying load. When individual equipment is shut down, breakers may be put into the test position and the control circuit may be functionally tested and the breaker exercised. Protective relays may be removed and tested. Standby equipment may be placed into service and running equipment shut down and, in this manner, all components of the system may be tested.

8.3.1.3 Design Analysis

The ratings of the 4,160 volt system have been selected to ensure a conservative margin between the maximum expected fault duty and the rating of the switchgear.

The four Buses 1A, 1B, 1F, 1G load split permits part load operation on loss of one bus and also prevents a single fault from causing a loss of more than two primary coolant pumps. The 4,160 volt Buses 1A, 1B, 1F and 1G loads are normally supplied by the turbine generator but can be supplied by standby power. Additionally, Buses 1A and 1B can be supplied during initial turbine generator coastdown. During shutdown, after disconnecting the turbine generator, the Buses 1A, 1B, 1F and 1G loads can be supplied by the switchyard through the main transformer. The 4,160 volt loads are not required after a DBA.

8.3.2 2,400 VOLT SYSTEM

8.3.2.1 Design Basis

The 2,400 volt system is designed to reliably function and supply power during normal, abnormal and accident conditions. During start-up, normal operation and shutdown conditions, the system will supply power to the 2,400 volt auxiliary loads from either the offsite source through Safeguard Transformer 1-1, from offsite source through Start-Up Transformer 1-2, from the main generator through Station Power Transformer 1-2 or from the emergency diesel generators. Another source is available from the delayed access circuit through the main transformer and Station Power Transformer 1-2 when the unit is in a shutdown condition. Two of the four 2,400 volt buses are an integral part of the Plant engineered safeguards electrical system and are identified as Class 1E components. The engineered safeguard electrical system is designed to the two-channel concept wherein independent electrical controls and power systems supply redundant 2,400 volt engineered safeguard load groups. The 2,400 volt engineered safeguards electrical system meets the single failure criteria.

8.3.2.2 Description and Operation

Description - The Plant's 2,400 volt system is shown on Figures 8-1, 8-5 and 8-6.

The 2,400 volt system consists of Safeguard Transformer 1-1, Station Power Transformer 1-2, Start-Up Transformer 1-2, four ungrounded, delta connected 2,400 volt buses (1C, 1D, 1E and safeguard) and the feeder circuits to motors and 480 volt load centers.

2,400 volt Buses 1C, 1D and 1E have access to the following power sources:

1. Two immediate access circuits connected to independent offsite power sources.

2. One onsite power source (main generator).

3. One delayed access circuit.

The engineered safeguard electrical system also has access to the emergency diesel generator (see Section 8.4).

The first immediate access circuit consists of a 345 kV to 2,400 volt transformer (Safeguards Transformer 1-1) located in the switchyard. The high side of this transformer is connected to the "F" switchyard bus through a motor-operated disconnect switch. The low side of the transformer is connected to the Safeguard Bus located within a Non-Class 1E switchgear house located within the Plant protected area. Connections between the transformer and switchgear house are provided via direct buried cable along the route between the switchyard and protected area. This cable is buried beneath the towers carrying the other immediate and delayed access circuits to provide for physical separation. The Safeguard Bus is provided to allow for selection of this immediate access circuit or Station Power Transformer 1-2 as the source of power to the 2,400 volt buses.

The second immediate access circuit consists of a 345 kV transmission line between the switchyard and plant site and a 345 kV to 2,400 volt transformer (Start-Up Transformer 1-2) located within the Plant protected area. The high side of this transformer is connected to the "R" switchyard bus through a motor-operated disconnect switch. The low side of the transformer is connected directly to the 2,400 volt bus incoming breakers.

The onsite power source consists of the main turbine generator, the 22 kV isophase bus and the 21 kV to 2,400 volt Station Power Transformer 1-2. The low side of the 2,400 volt transformer is connected to the Safeguard Bus via an enclosed bus duct. The Safeguard Bus also provides for connections to the first immediate access circuit and allows for selection of the first immediate access circuit or the onsite power source as the source of power to the 2,400 volt buses.

The delayed access circuit consists of one 345 kV transmission line, the 345 kV to 23 kV Main Transformer, the 22 kV isophase bus and the 21 kV to 2,400 volt Station Power Transformer 1-2. The low side of the 2,400 volt transformer is connected to the Safeguard Bus via an enclosed bus.

The Safeguard Bus also provides for connection to the first immediate access circuit and allows for selection of the first immediate access circuit or the delayed access circuit as the source of power to the 2,400 volt buses. Switchyard power via the delayed access circuit can be provided from either of the two switchyard buses. Power can also be provided directly from the grid via Consumers Power Company's interconnection with the Indiana and Michigan Power Co (Cook Line 1). The delayed access circuit is established by opening the motor-operated disconnect switch in the isophase bus. This switch can be opened within 30 minutes. Opening the switch allows for backfeeding via the main and station power transformers following the operation of other switchyard and in-plant breakers.

The capabilities of the 2,400 volt buses are sufficient to permit Plant operation under reduced load with any 2,400 volt bus out of service. See Table 8-3 for ratings and construction of the 2,400 volt system components.

The 2,400 volt buses (1C, 1D and 1E) consist of metal-clad switchgear with drawout air circuit breakers. The 2,400 volt safeguard bus consists of metal-clad switchgear with drawout vacuum circuit breakers. All wiring in the switchgear and all system interconnecting cables pass the vertical flame resistance test in accordance with ASTM D470-59T. New installations will meet an equivalent flame test (eg, IEEE 383-1974, Section 2.5, ICEA S-19-81, Section 6.19.6, etc).

The 2,400 volt switchgear is provided with relay protection, grounding alarm and the mechanical safeguards necessary to assure adequate personnel protection and to prevent or limit equipment failure during system fault conditions. Single grounds will not impede operation of the system.

The 2,400 volt system has sufficient capacity to start and accelerate the largest motor when all other required motors are energized and carrying load.

Two of the 2,400 volt buses, 1C and 1D, supply power to engineered safeguards loads in addition to normal Plant loads and are part of the engineered safeguards electrical system. These engineered safeguards system 2,400 volt buses are designed to withstand Consumers Design Class 1 seismic acceleration forces per Section 5.7 without malfunction.

The engineered safeguards electrical system is divided into two channels so that multiple pieces of equipment with a common function are fed from opposite channels.

A separate emergency diesel generator supplies each of the emergency 2,400 volt buses (1C and 1D) and each bus supplies redundant equipment or loads consistent with the two-channel power concept.

The emergency buses are physically separated by being located in separate rooms within the Consumers Design Class 1 portions of the auxiliary building. Separation is maintained between the circuits of the two buses.

Electrical feeder cables from the emergency generators to the emergency buses and safeguards equipment motors are installed within the Consumers Design Class 1 portion of the auxiliary building or in underground ducts.

<u>Normal and Shutdown Operation</u> - Power to 2,400 volt Buses 1C, 1D and 1E during start-up, normal and shutdown operation is normally provided via the first immediate access circuit (Safeguard Transformer 1-1) powered from the switchyard "F" bus. Upon loss of this circuit, a fast transfer is provided to the second immediate access circuit (Start-Up Transformer 1-2) powered by the switchyard "R" bus. Refer to Section 8.6 for 2,400 volt bus automatic fast transfer features. Upon loss of both immediate access offsite power sources, the 2,400 volt Buses 1C and 1D are energized from the diesel generators. Bus load shedding, transfer to the diesel generator and energization of critical loads are performed automatically. Refer to Section 8.6 for 2,400 volt bus load shedding control.

With Safeguard Transformer 1-1 out of service, capability is provided to connect the 2,400 volt buses to the main generator via Station Power Transformer 1-2. When connected to Station Power Transformer 1-2, a reactor and/or generator trip will result in a fast transfer to the second immediate access circuit (Start-Up Transformer 1-2).

In the event that both immediate access circuits are out of service, power may be provided by the main transformer backfeeding the Plant 2,400 volt buses. This mode of operation is allowed during plant shutdown.

Operation of 2,400 volt equipment is normally effected and monitored in the control room. Breaker status is indicated by red and green indicating lights. All 2,400 volt breakers on Buses 1C and 1D are also capable of being controlled from the switchgear. Breaker 152-311, feeder to the Service Building Expansion, is not controlled or monitored in the Control Room.

Breakers 152-103, 107, 108 and 110 have special remote/local isolation switches to allow control in the event of fire in certain areas of the Plant. These switches are provided to ensure operability of safe shutdown equipment per 10 CFR 50.48 and 10 CFR 50, Appendix R. Post-fire shutdown capability is further enhanced by a remote/local switch provided for Breaker 152-106 which facilitates control of startup power (where available) to Bus 1C. Reference FSAR Section 9.6 for critical fire areas.

Important control circuits, such as bus transfer and load shedding, have white indicating lights to show circuit availability. Undervoltage relays initiate an alarm upon loss of potential. Electrical parameters, such as bus voltage, supply voltage, bus amperes and motor amperes, are displayed in the control room. Important functions, such as incoming breaker trip, motor breaker trip, bus undervoltage, bus ground, loss of incoming breaker 125 volt dc control voltage, and failure of bus transfer, are annunciated in the control room.

Provisions are included for testing relays. A faulty relay may be bypassed to permit operation of equipment.

Testing - Components may be tested when the system is de-energized. Testing of parts of the system can be performed when the system is in operation. When individual equipment is shut down, breakers may be put into the test position and the control circuit may be functionally tested and the breaker exercised. Protective relays may be removed and tested. Standby equipment may be placed in service after which the running equipment may be shut down and the breaker tested as above. In this manner, all components of the system may be tested.

8.3.2.3 Design Analysis

The ratings of the 2,400 volt system have been selected to ensure a conservative margin between the maximum expected fault duty and the fault rating of the switchgear.

Three buses (1C, 1D and 1E) permit part load operation on loss of one bus. The engineered safeguards loads can be supplied by either offsite power source, diesel power or from the switchyard through the main transformer after disconnecting the turbine generator.

To assure reliability, vital 2,400 volt Buses 1C and 1D obtain control power from separate dc sources. Incoming and feeder breakers on the 1C bus receive control power from Station Battery D01 via breakers in Panel D-11A. Station Battery DO2 via Panel D-21A supplies control power for incoming and feeder breakers on Bus 1D. These dc sources are physically isolated to preclude any event affecting the integrity of both sources. The breakers on the Safeguard Bus receive control power from dc Panel D-21. D-21 can be powered from Station Battery D02 and/or the associated battery chargers.

8.3.3 480 VOLT SYSTEM

8.3.3.1 Design Basis

The in-plant 480 volt system is designed to reliably function and supply power during normal, abnormal and accident conditions. Four load centers and eight motor control centers are an integral part of the Plant engineered safeguards electrical system and are identified as Class 1E components. The system is designed to the two-channel concept wherein independent electrical controls and power systems supply redundant 480 volt engineered safeguards load groups. The 480 volt engineered safeguards auxiliary power system meets the single failure criterion.

The cooling tower 480 volt system is designed such that all equipment for each tower is normally fed from separate 4,160 volt sources. Partial or complete failure of 480 volt supply to either cooling tower will not affect the other.

8.3.3.2 Description and Operation

Description - The Plant 480 volt system is shown on Figures 8-7 through 8-11, Single Line Meter and Relay Diagrams, 480 Volt Load Centers.

The in-plant 480 volt system is divided into load center buses and motor control centers. Power for each (except one) load center bus is supplied from a separate 2,400-480 volt station service transformer. The transformers fed from the 2,400 volt system are arranged so that each transformer of a double-ended load center unit is fed from a different 2,400 volt bus. One load center is fed from 4,160 volt Bus 1A through a 4,160-480 volt station service transformer.

The cooling tower 480 volt system shown on Figure 8-11 is divided into load center buses. Power for each load center bus is supplied from a separate 4,160-480 volt station service transformer fed from the 4,160 volt supply to the tower that each transformer services.

Power for the motor control centers is supplied from the load center buses.

Station Power Transformers 11, 12, 19 and 20, corresponding Load Centers and Motor Control Centers 1, 2, 21, 22, 23, 24, 25 and 26 are an integral part of the engineered safeguards electrical system. This equipment is arranged into two channels so that multiple pieces of equipment with a common function are fed from opposite channels. The capacities of the station power transformers and the 480 volt bus tie breakers for Load Centers 11 and 12 are sufficient to permit Plant operation with one transformer out of service for the length of time specified by Technical Specifications. During a bus tie between LC11 and LC12 the total combined load on LC11 and LC12 must be controlled such that it does not exceed the rating of the transformer remaining in service.

The 480 volt buses consist of metal-enclosed drip proof switchgear with drawout air circuit breakers. The motor control centers are NEMA 1 enclosures with drip hoods installed on the majority of the equipment. Drip hoods/shields were not installed on Motor Control Centers 1 and 2 because of space limitation. As an alternative, the metal seams at the top of these motor control centers were sealed with RTV sealant to make them waterproof from the fire sprinklers. All wiring in the switchgear and motor control centers and all system interconnecting cables pass the vertical flame resistance test in accordance with ASTM D470-59T or IPCEA S-61-402, Section 6.5. New installations will meet an equivalent flame test (eg, IEEE 383-1974, Section 2.5, ICEA S-19-81, Section 6.19.6, etc). See Table 8-4 for ratings and construction of the 480 volt system components.

The 480 volt switchgear and the motor control centers supplying engineered safeguards loads are designed to withstand Consumers Design Class 1 seismic acceleration forces per Section 5.7 without malfunction.

The 480 volt load centers and motor control centers are solidly grounded Y connected and are provided with the mechanical safeguards necessary to assure personnel protection and to prevent or limit equipment damage during system fault or overload conditions.

The 480 volt load center breakers are equipped with thermal magnetic or solid state trip devices. Motor control centers are equipped with thermal magnetic breakers for nonmotor loads, and magnetic breakers and starters with thermal protection for the motor circuits.

The 480 volt system has sufficient capacity to start and accelerate the largest motor when all other required motors on the system are energized and carrying load.

Starters in the 480 volt motor control centers assigned to engineered safeguards loads may be controlled from the control room, from local panels, or both. Status of these starters may be indicated by lights in the control room, at the local panels, or both. Other starters may be controlled at the motor control centers or at local panels. Motor overload condition for selected critical loads is annunciated in the control room.

The 480 volt engineered safeguards electrical system is installed in Consumers Design Class 1 portions of the auxiliary building.

Normal Operation - During normal operation, in-plant 480 volt power is supplied from the 2,400 and 4,160 volt systems. Following a turbine trip, all 480 volt loads will continue to run from the 2,400 volt system, and from the 4,160 volt system following transfer to the Start-Up Transformers.

Operation of some of the 480 volt equipment is monitored in the control room. Status of some of the breakers is indicated by lights in the control room.

During normal operation, all incoming bus breakers and motor control center feeder breakers are closed and all bus tie breakers are open. The status of these breakers will be changed only for emergency or maintenance.

Loss of Offsite Power - If the offsite sources fail, all cooling tower and nonessential loads will be shed. Load Centers 11, 12, 19 and 20 and Motor Control Centers 1, 2, 21, 22, 23, 24, 25 and 26, supplying engineered safeguards loads, are not shed and the essential loads will be supplied by the emergency generators through the 2,400 volt buses as described in Subsection 8.3.2.

After load shedding, the remaining in-plant load centers and motor control centers, which are not included in the engineered safeguards system, may be energized manually to serve loads which are not critical.

Shutdown Operation - No change in status of the incoming bus breakers and tie breakers is required during shutdown. Other breakers and the motor starters are operated manually or automatically as required by the shutdown sequence.

Operation After Loss of Coolant Accident - A Loss of Coolant Accident affects the 480 volt in-plant system only if accompanied by loss of offsite power in which case all loads except the engineered safeguards load centers and motor control centers are shed automatically. Upon return of power to 2,400 volt Buses 1C and 1D, additional 480 volt loads may be energized manually by the operator. Fuses are provided to ensure containment electrical penetrations overcurrent backup protection for circuit breakers in the MCC starters feeding submerged equipment not de-energized following a Loss of Coolant Accident (LOCA) and also to ensure that the operation of backup protection does not lead to interruption of the power supply to other safety-related equipment. Fuses are provided for electrical penetration backup protection in motor starters and 120 V ac control circuits to protect electrical penetrations against damage by overcurrent. Since MCC 9 is located within the containment, a backup circuit breaker has been provided in the feeder to this motor control center.

Testing - The individual components of the 480 volt system may be functionally tested at any time. Idle loads may be energized, then de-energized, to exercise starters and breakers and to functionally test the control circuits. Those loads which may not be stopped momentarily must be replaced by standby equipment during testing. The 480 volt load center circuit breakers may be placed in the test position if energization of equipment is not desired.

8.3.3.3 Design Analysis

The ratings of the 480 volt systems have been selected to ensure a conservative margin between the maximum expected fault duty and the fault rating of the equipment.

The 480 volt buses which are part of the engineered safeguards system are installed in Consumers Design Class 1 portions of the auxiliary building.

8.3.4 CONTROL ROD DRIVE POWER

8.3.4.1 Design Basis

The control rod drive power system is designed as a stable, reliable supply for the control rod drive motors.

8.3.4.2 Description and Operation

Two 480-208/120 volt, single phase, 60 hertz transformers furnish power for the Control Rod Drive System. Selection of the transformer is made through a transfer switch on the 120 volt side. The rod drive motors are supplied through individual contactors located in the contactor panel and are controlled from the control room.

Each transformer is supplied from a manually controlled circuit breaker in a different motor control center. The motor control centers are supplied with power from 480 volt load centers and arranged so that the power for each is fed from a different 2,400 volt bus.

Loss of control rod drive power does not affect the rod position. Loss of power is annunciated in the control room.

8.3.4.3 Design Analysis

Each transformer is conservatively rated. To assure reliability, the transformers are supplied from separate buses, each of which can be supplied by a separate emergency generator. Upon loss of one transformer, the other transformer may be manually placed in service to supply power to the rod drives. A loss of both transformers will not prevent a safe shutdown since the rods may be inserted by de-energizing the clutches which release the rods.

8.3.5 DC AND PREFERRED AC SYSTEMS

8.3.5.1 Design Basis

The dc and preferred ac systems are designed to furnish continuous power to the Plant instrumentation and control systems. The power supply is continuous even during disturbances in the auxiliary electrical system. The reliability of the system is assured by duplication of vital equipment and circuits; ie, dual circuits for identical purposes are supplied from alternate buses and either circuit can initiate the safety function required.

8.3.5.2 Description and Operation

Description - General - The 125 volt dc and 120 volt preferred ac systems are shown on Single Line Meter and Relay Diagram, Figure 8-12. Equipment is designed to withstand Consumers Design Class 1 seismic acceleration forces as described in Section 5.7 without malfunction. Equipment is provided with circuit breaker protection, grounding and the mechanical safeguards necessary to assure adequate personnel protection and to prevent or limit equipment failure during system fault conditions. See Table 8-5 for ratings and construction of the dc and preferred ac system components.

125 volt dc and 120 volt ac circuits going inside the containment to equipment susceptible to be submerged during a Loss of Coolant Accident have been provided with backup overcurrent protection, when required, to ensure the integrity of the containment electrical penetrations.

The systems are located in Consumers Design Class 1 portions of the auxiliary building.

DC System - The 125 volt dc system is divided into two independent and isolated systems.

Each system consists of a battery, switchgear, distribution panel, two chargers and instrumentation as shown on Figure 8-12. The switchgear bus is split into two sections, each of which can be fed by its own battery charger. Power to switchgear and the distribution panels is supplied by the station batteries and/or the battery chargers. Protection of the dc cabling in the case of a fire emergency is provided by separate dc distribution panels located in areas allowing emergency shutdown. The following design features assure the availability of 125 volt dc power for the operation of Diesel Generators 1-1 and 1-2, 2,400 volt Buses 1C and 1D, nonsafeguards Buses 13 and 14 and the Auxiliary Shutdown Control Panel C150 in the event a fire damages 125 volt dc distribution equipment in the cable spreading room (see Figure 8-13).

1. Fuses between each battery and its bus are located in their respective battery rooms.

2. In each battery room, a nonautomatic circuit breaker with a shunt trip is provided in the circuit between the battery fuse and its bus. The shunt trip device of these circuit breakers is a trip coil that is energized by battery voltage via the 125 volt dc distribution panel. The nonautomatic circuit breakers were specified for use in 125 volt dc systems and for a steady-state load of 400 amperes. They are qualified per IEEE 323-1974 and IEEE 344-1975. They do not contain fault detectors and are not intended to interrupt fault currents although they have that capability. They are manually operated open or close with the capability of being opened remotely via the shunt trip device.

   If the shunt trip push button is closed inadvertently, the battery will be separated from the principal 125 volt dc bus. An undervoltage relay has been installed on the battery and will detect the separation of the battery from its charging source. Operation of the relay is annunciated in the control room (see later "System Monitoring" description).

3. Distribution panels are provided in Switchgear Room 1-C and Diesel Generator 1-2 room connected to their respective batteries with a fuse located in the applicable battery room. Each distribution panel contains a push button for energizing the shunt trip of the above-mentioned circuit breaker.

4. From each of the distribution panels, circuits for operating and control power are provided for the corresponding diesel generator and 2,400 volt bus with routing avoiding the cable spreading room and the diesel generator and switchgear rooms of the other channel. In addition, the distribution panel serving 2,400 volt Switchgear Bus 1-C supplies the Auxiliary Hot Shutdown Control Panel C-150 and nonsafeguards 480 volt Buses 13 and 14.

The chargers are of the solid-state type. They have provision for two charge rates, one for floating and one for equalizing the battery. The chargers are provided with filters and surge protection to enable either charger to supply the dc loads including the operation of 2,400 volt circuit breakers with the battery disconnected. The two chargers on each 125 volt dc bus are fed from separate 480 volt motor control centers. The motor control centers are supplied with power from 480 volt load centers and arranged so that the power for each is fed from a different 2,400 volt emergency bus. Administrative controls limit the operation such that only one charger per battery is in service. This removes the possibility of a common mode failure affecting both emergency buses. The battery charger cabinets were specified to operate at a design ambient temperature of 104°F with only natural circulation cooling. However, to extend the life of temperature sensitive components, seismically mounted cooling fans powered from preferred AC power were installed.

Both dc systems are ungrounded and are equipped with ground detectors for continuous monitoring. Monitoring is also provided on other important system parameters, such as bus voltage and current. Abnormal conditions are annunciated in the control room.

Preferred AC System - The 120 volt preferred ac system has four separate buses to provide power for the four separate Reactor Protective System channels. Each bus is supplied by an inverter which is, in turn, supplied by a particular dc bus section. Each dc bus section ties together one battery charger and one inverter. Thus, each inverter with its associated charger can be separated, in emergency conditions, from the rest of the dc system for maximum reliability. Each inverter, one at a time, can be manually bypassed and its preferred ac bus supplied from the instrument ac panel via a bypass regulator. The preferred ac buses operate ungrounded and are equipped with ground detectors. Specific safety systems, such as the RPS neutron flux monitoring and clutch power supplies (Chapter 7), have separate floating grounds.

The preferred AC inverter cabinets were specified to operate at a design ambient temperature of 104°F with only natural circulation cooling. However, to extend the life of temperature sensitive components, seismically mounted cooling fans powered by the output of each preferred AC bus were installed.

In order to comply with the electrical isolation requirements of IEEE 384, Regulatory Guide 1.75 and Regulatory Guide 1.6, the bypass regulator output breakers are interlocked to preclude supplying more than one preferred ac bus at a time. The interlock scheme has a key lock on each output breaker. The four breaker locks utilize a single key which is maintained by the shift supervisor. The key must be inserted in the desired breaker locking device before that breaker can be closed, and with a breaker closed the key cannot be removed. Therefore, only one breaker can be closed at a time.

The bypass regulator is not fed from an uninterruptible supply. On a loss of offsite power, it will not power the connected AC bus. The AC bus will be dead for the ten seconds required for diesel generator starting (when the power is restored to the bypass regulator). This delay has been analyzed in the appropriate Chapter 14 events. What hasn't been analyzed is the additional delay that the sequencers take through an initialization routine. Because of this and other functions that are lost with preferred AC, any preferred AC bus that is on the bypass regulator is considered to be inoperable. A Technical Specification LCO applies when the Preferred AC bus is so powered.

Normal Operation - During normal operation, both sections of each dc bus are interconnected by a nonautomatic breaker. This ties together two inverters, one charger and one battery to the same bus, the second battery charger being left in standby. The battery on each bus is kept fully charged, floating at approximately 131 volts. The connected battery charger supplies the dc loads including the two inverters which, in turn, feed one preferred ac bus each.

Periodically, the charger voltage is raised to approximately 138 volts for battery equalizing.

Operation of circuit breakers in the dc and the preferred ac systems is manual with automatic trip for fault isolation. Tie breakers between the left and right sections of each dc switchgear bus and the battery isolation shunt trip breakers do not have an automatic trip for fault isolation.

Emergency Operation - On loss of normal and standby ac power, all dc loads will be supplied from the station battery. The normal dc loads will increase since there will be additional annunciators and additional loads, such as emergency lighting and oil lift pumps, required for primary coolant pump coastdown added to the battery load as a result of ac power failure. Aside from these steady loads, some intermittent dc power for breaker operation will be required. As soon as one of the diesel generators has started and is ready for loading, a battery charger will automatically resume operation and support the battery.

Testing - A test push button is provided at the dc control center to check the operation of the dc emergency lights. Testing of the batteries is described in Subsection 8.4.2.2.

System Monitoring - The dc and preferred ac power systems (ie, chargers, inverters, batteries and breakers) are controlled locally. In accordance with IEEE 308-1978, Paragraph 7.1.2, the operational status information is displayed locally (see Figure 8-12). The dc bus monitoring devices consist of:

1. Battery current (ammeter - charge/discharge); one per battery,

2. Battery charger output current (ammeter); one per charger,

3. Battery charger output voltage (voltmeter); one per charger,

4. DC bus voltage (voltmeter); one per bus,

5. DC bus ground current detector (milliammeter); one per bus, and

6. DC bus average ground current (recorder); one per bus.

All of the above indications are centrally located. Items 2 and 3 are located on the charger's front cabinet door. The remaining items are located on metering panels which are installed within approximately ten feet of the charger.

In accordance with IEEE 308-1978, Paragraph 7.1.3, IEEE 279-1971, Paragraph 4.13 and Part C of Regulatory Guide 1.47, administrative procedures and automatic control room display are also available. These procedures and displays provide the operator a complete and timely indication of system protective actions and system unavailability (such as deliberate actions to render inoperable a component of the dc power system).

Three control room annunciations are provided to alert the operator of dc power system unavailability. The first annunciation is for dc bus undervoltage or trouble. "125 Volt DC Bus Undervoltage/Trouble" alarm annunciates upon the following inputs:

1. 125 volt dc bus tie breaker open (either bus),

2. PA system inverter undervoltage (Bus D10 only),

3. Battery undervoltage (either battery), and

4. 125 volt dc bus undervoltage (either bus).

Alarm Input 1 results from a tie breaker position switch installed on both Bus D10 and D20 tie breakers. Should either (or both) tie breakers be opened, the position switch "a" contacts will energize an auxiliary relay to effect both a control room and a local annunciation.

Alarm Input 2 results from an undervoltage relay installed to monitor the input of the public address system inverter. Should this relay sense an undervoltage condition, both a control room and a local alarm will occur.

Alarm Input 3 results from an undervoltage relay installed between the battery and the battery's downstream fused disconnect. Two relays are installed, one for each battery (see Figure 8-12). The undervoltage relays inform the control room operator whenever the batteries are disconnected from the chargers (eg, the charger feeder breakers or the battery disconnects are open). The undervoltage alarm set point was chosen below the battery's normal "float" voltage on the charger and above the battery's voltage when disconnected from its charger. By providing a control room and local alarm at this set point, the operator is alerted to a system unavailability condition in a timely manner.

Alarm Input 4 is the bus undervoltage function indicating low voltage with regard to the dc loads.

The second annunciation is for battery chargers trouble. This control room alarm will occur whenever ac input or dc output power is lost on any one of the four chargers. Upon receiving the alarm, an operator can be dispatched to the battery chargers to determine which charger is affected utilizing the charger's voltmeter and ammeter indications, should it not already be apparent in the control room.

The third control room annunciation is for dc bus ground. Should the bus-to-ground current (milliamperes) on either bus exceed a predetermined set point, a control room alarm will be energized. At this point, an operator can be dispatched to the local metering panels to determine which bus is grounded by utilizing the local milliammeters. As previously mentioned, inputs to the control room alarms also provide a local alarm. Upon control room annunciation, an operator can be dispatched locally to determine which of the four inputs for dc bus trouble provided the alarm and on which dc system the problem has occurred; ie, Bus System D10 or Bus System D20. A local annunciation relay is provided for each system. Each relay indicates locally which alarm input caused the control room annunciation.

In addition to the above dc bus monitoring devices, there are alarms and indicators associated with the 120 volt ac preferred power panels. Since the preferred power panels take their normal power supply from the dc buses, the following alarms could be a further indication of dc bus trouble:

1. Preferred bus undervoltage - control room alarm (one per bus)

2. Ammeter (current input to inverter) - local indicator

3. Ammeter (current out of inverter) - local indicator

4. Frequency meter - local indicator

5. Ground detector - control room alarm (one per bus)

6. Voltage meter - local indicator

For Items 1 and 5 a common alarm (one per bus) indicating "preferred ac bus trouble" is utilized.

8.3.5.3 Design Analysis

Normal Operation - During normal operating conditions, the loading of both the dc and the preferred ac systems is much lower than the capacity of the system. Each of the two battery chargers provided on the dc bus is capable of supplying the normal dc loads on the bus and simultaneously recharging the battery in a reasonable time. A fully discharged battery can be recharged in less than nine hours by using two chargers.

Emergency Operation - Complete loss of all ac power analysis is given in Subsection 8.4.2.3.

In order to meet IEEE 308-1978, Paragraph 7.1.3 requirements, the control room features an assortment of dc power system alarms (ie, "Battery Chargers Trouble," "125 Volt DC Bus Ground," "125 Volt DC Bus Undervoltage/ Trouble" and "Preferred AC Bus Trouble"). The proper combination of these alarms will alert the operator of most conceivable malfunctions, misalignments or maladjustments which might occur to render any part of the system inoperable. Upon being alerted, the control room can dispatch an operator locally to specifically determine what the problem is and what system components may be affected.

8.3.6 INSTRUMENT AC SYSTEM

8.3.6.1 Design Basis

The 120 volt instrument ac system is designed to furnish reliable power to the Plant instruments other than those supplied from the dc and the preferred ac systems.

8.3.6.2 Description and Operation

Description - The instrument ac system is supplied by two three-phase transformers from Motor Control Centers 1 and 2 as shown on Figure 8-12. Both Motor Control Centers are automatically supplied by emergency generators upon loss of offsite power. An automatic transfer switch is provided to transfer supply to the instrument ac panel between the two power sources.

All panel breakers are equipped with thermal magnetic trip elements. The neutral of the instrument ac system is grounded. This system can only furnish power to one of the preferred ac buses at a time through a bypass regulator.

Operation - During normal operation, power is supplied to the instrument ac bus from Motor Control Center 1. Should the power fail from that source, the panel supply will automatically be transferred to Motor Control Center 2. When power to Motor Control Center 1 is restored, the panel supply will automatically transfer back. Transfer in either direction may be made manually.

8.3.6.3 Design Analysis

Each of the two instrument ac transformers is sized to supply the panel load and one preferred ac panel bus via the bypass regulator.

Testing - The operation of the transfer switch may be checked at any time without affecting Plant operation.

8.4 EMERGENCY POWER SOURCES

The emergency power sources are designed to furnish onsite power to reliably shut down the Plant and maintain it in a safe shutdown condition under all conditions, including DBA, upon loss of normal and standby power. The emergency power sources are part of the engineered safeguards electrical system and are identified as Class 1E systems. Reliability is assured by the two-channel concept wherein independent electrical controls and sources supply redundant ac and dc engineered safeguards loads.

8.4.1 EMERGENCY GENERATORS

8.4.1.1 Design Basis

The emergency generators are designed to provide a dependable onsite power source capable of starting and supplying the essential loads to safely shut down the Plant and maintain it in a safe shutdown condition under all conditions. The reliability of this onsite power is provided by its duplication wherein each emergency generator supplies redundant loads and each is capable of providing power to the minimum necessary safeguards.

8.4.1.2 Description and Operation

Description - There are two emergency diesel engine-driven generators of equal size. The generators have static-type excitation and are provided with field flashing for quick voltage buildup. Each generator is connected via a generator breaker to a separate 2,400 volt bus. The generator breaker control is shown on Schematic Diagram Figures 8-14 and 8-15. Synchronizing equipment is provided to permit connecting the generator to the 2,400 volt bus for parallel operation with the onsite or offsite power sources during testing of the emergency generators. The synchronizing equipment is automatically bypassed by breaker position interlocks to permit manual and automatic closing of the emergency generator breaker on a dead bus. The four 2,400 volt bus safeguard/station power and start-up transformer incoming breakers are interlocked to prevent automatic closing when the associated emergency generator breaker is closed. The incoming breakers can be closed manually only by using synchronizing equipment when the associated emergency generator breaker is closed.

Support systems associated with each diesel generator include a fuel oil system, air starting system, lube oil system, jacket water system, crankcase blower, two independent starting circuits and a load sequencer (see Figure 8-29). Supply of electric power for these systems is obtained from the generator they are supporting. Each system is located in a separate room from its redundant counterpart, except for the load sequencers which are located separate from one another in the main control room.

The diesel engines are designed for air start and a separate compressor and receiver are provided for each engine. There are two receivers and two air-start motors per engine. A separate fuel oil day tank is also provided for each engine. The diesel engines, fuel oil systems and air start systems are equipped with instrumentation to monitor all important parameters and annunciate abnormal conditions. Water and oil heaters are provided to maintain the engines in "start" readiness.

The emergency generators are equipped with the mechanical and electrical safeguards necessary to assure personnel protection and to prevent or limit equipment damage during operation or fault and overload conditions. The generators and their 2,400 volt breakers have overcurrent and differential protection. All wiring passes the vertical flame resistance test in accordance with IPCEA S-28-357, Paragraph 3.4. New installations will meet an equivalent flame test (e.g., IEEE 383-1974, Section 2.5, ICEA S-19-81, Section 6.19.6, etc).

The emergency diesel generators and their auxiliaries are designed to withstand CP Co Design Class 1 seismic acceleration forces per Section 5.7 without malfunction. The emergency diesel generators and their auxiliaries, except for the fuel oil transfer system, are installed in a CP Co Design Class 1 portion of the auxiliary building and the units are separated by a wall.

Each emergency generator supplies a separate 2,400 volt bus and a redundant group of engineered safeguards consistent with the two-channel power concept.

Diesel generator reliability is targeted at 0.95. This level is to be maintained by a Diesel Generator Reliability Program in response to commitments made by CP Co March 27, 1990 in regard to Station Blackout Rule 10 CFR 50.63 (Reference 10).

Diesel Generator Control Circuits - Physical separation and electrical isolation are maintained between the control circuits for the two diesel generators. The automatic start circuits are initiated by bus undervoltage. The control circuits, in addition to the "automatic" functions, are arranged for manual start-stop at the diesel and in the control room. The controls for the governor, voltage regulator, synchronizing and for the generator breaker are located in the control room.

Normal Operation - As shown on Figures 8-17, 8-19, 8-20 and 8-21, Diesel Generator 1-1 will start when an undervoltage is sensed on 2400 Volt Bus 1C and Diesel Generator 1-2 will start when an undervoltage is sensed on 2400 Volt Bus 1D. Section 8.6 provides additional details on undervoltage starting.

When the Plant is operating normally, the diesels may be started, synchronized with the 2,400 volt buses and loaded.

The status of the emergency generators is monitored in the control room. Important functions are annunciated in the control room.

The diesel generators may be started and shut down locally or from the control room.

Shutdown Operation - During shutdown operation, the emergency diesel generators will supply power only if the offsite power source fails. At this time, the automatic features will govern and normal shutdown sequencers will sequentially load the generators.

If the emergency generators fail to start, the Plant auxiliaries can be fed via the main transformer in a backfeed mode. Refer to Subsection 8.2.3 for discussion of backfeed.

Operation After Loss of Coolant Accident - The emergency generators are required to supply power only if the offsite power source fails. At this time, the automatic features will govern and DBA sequencers will sequentially load the diesels.

Operations During or After Fire - The 1-1 emergency diesel generator has three remote/local isolation switches (one for the output breaker and two for the diesel generator) to allow control in the event of fire in some plant fire areas . These switches are intended to ensure operability of safe shutdown equipment per 10 CFR 50.48 and 10 CFR 50, Appendix R. Operability of 1-2 emergency diesel generator after a fire could be restored by operation of slide links in control circuitry. Operation of these slide links is not required, however, by the Appendix R, Safe Shutdown Analysis. Operation of the switches or slide links is governed by Off Normal Procedures.

Testing - Automatic start and load sequencing of the emergency generators are tested as part of the safety injection testing. For details see Subsection 7.3.5. The emergency generators' start-up may be manually tested at any time to verify required voltage and frequency are obtained within acceptable limits and time. To verify load acceptance by the generator, the emergency generator breaker is closed manually and the engine loaded onto the 2,400 volt bus for parallel operation with the onsite or offsite power source. Refer to Technical Specifications for further details.

8.4.1.3 Design Analysis

The emergency generators have been selected to have sufficient capacity to supply the minimum necessary engineered safeguards loads with only one generator operating. In addition, each generator has enough reserve capacity to start and carry the largest single load that may be loaded on the bus by a control circuit malfunction.

The emergency generators are designed to reach required speed and voltage within 10 seconds after the receipt of a start signal.

To assure reliability, each emergency generator has two starting/control circuits, one for each of the two separate air starting motors. Although each starting circuit will initiate generator field flashing, the power for field flashing is provided from only the "B" circuit. Each engine's two starting/control circuits are powered from that engine's respective 125 VDC battery-backed channel, and are independent from the opposite diesel engine's two starting/control circuits. Each diesel engine's two starting/control circuits are actuated by redundant automatic start command signals. Physical separation is maintained between the two emergency generator units and their associated controls.

Each emergency generator and diesel engine is provided with several alarms, interlocks and trips. Each engine may be started and placed in service locally or from the control room. The generators may be synchronized from the control room so that they can be paralleled with the system for loading tests. Each diesel is located in a separate room as is shown on Figure 1-3. Each room has separate access doors.

Local alarms at each diesel are:

Prelube Oil Pump Failure
Low Lube Oil Pressure
High Lube Oil Temperature
Low Lube Oil Temperature
High Lube Oil Filter Differential Pressure
High Jacket Water Temperature
Low Jacket Water Level
Low Raw Water Pressure
Overspeed
Low Air Pressure
Overcrank
Low Lube Oil Level
Engine Trouble
High/Low Fuel Level
Low Jacket Water Pressure
Bus 1C (1D) Overcurrent Lockout

Any local alarm also results in annunciation of a Diesel Generator Trouble alarm in the control room.

Other diesel generator control room alarms include:

Diesel Generator Fail to Start
Diesel Generator Start Signal Blocked

The Diesel Generator Start Signal Blocked alarm is actuated by the following inputs:

Loss of DC control power
Overspeed trip
Low lube oil pressure trip
Low jacket water pressure
Overcurrent or differential current

In addition, a trip of the diesel generator breakers by anything other than a manual trip is annunciated separately in the control room as is low level in the main fuel oil storage tank, day tank hi-lo level, generator overload, and Bus 1C or 1D overcurrent lockout. The diesel generator breakers will be opened should there be an overload, or generator differential relay operation, or should the diesel shut down. Additionally, a short duration trip signal is provided to the diesel generator breakers whenever a signal is initiated to automatically fast transfer the normal source of power to the start-up transformer. This trip signal ensures that the diesel generator will not be placed in parallel with the offsite source while not in phase.

The diesel will be automatically tripped on generator differential or overcurrent relay action, engine overspeed, low lube oil pressure, or low jacket water pressure, and can be manually tripped at any time from the local station or from the control room.

There are no trips on either the generator or the engine which are bypassed while engineered safeguards systems are functioning. Since there are two emergency diesels provided, each with full capacity rating, the single failure criterion is met regardless of which diesel auxiliary component is assumed to fail. Even more pertinent is the fact that the trips are minimum in number but important in function for equipment protection. Emergency power availability is thereby enhanced by tripping the unit off for these faults which enables repair and return to service rather than burning out the generator or engine.

The diesel generator is designed to start and be ready for loading in ten seconds, and be capable of loading and carrying required safety-related loads within the times established for sequential loading. The worst case automatic loading sequence (with their associated nameplate HP) for each diesel is shown in Tables 8-6 and 8-7, with zero second being diesel start time.

The engines are rated at 3,500 brake horsepower (bhp), with a predicted overload capacity of 3,840 bhp for two hours.

The generator is rated at 2,500 kW at 0.8 power factor with a two-hour overload rating of 2,750 kW, a one-half hour overload rating of 3,125 kW and a one-minute overload rating of 3,750 kW. The recovery time for voltage to return to 90% of rated voltage after application of each load step is less than three seconds.

When any plant modification affects the diesel generator loading, the revised loading will be verified to not exceed either the engine or generator ratings.

Each diesel generator's fuel oil system is supplied fuel oil by a common transfer system. The transfer system consists of an underground fuel oil storage tank, a single supply line and two transfer pumps. The transfer system mechanical design is safety related; there is a single fuel oil supply line for both diesels, and the two transfer pumps are not physically separated. However, each diesel has a minimum day tank/belly tank capacity of 20 hours (at rated output power) and the day tanks can be refilled by tank truck.

IEEE 308-1978 requires sufficient fuel be onsite for the operation of one diesel for seven days assuming accident loads. Fuel oil storage tank T-10A has sufficient capacity to contain the required seven day supply of fuel oil for emergency diesel generator use. Additionally, plant operating procedures alert operations personnel to evaluate the fuel on hand, the probability of restoring offsite power, and the probability of getting additional fuel. Fuel conservation practices will be implemented if it is likely that seven days will elapse before offsite power is restored and additional fuel is received. The capability also exists to transfer fuel oil from another onsite storage tank, T-926, to storage tank T-10A.

When any plant modification affects the diesel generator loading, the fuel oil storage capacities will be verified to be adequate for the revised loading.

Either of two fuel oil transfer pumps are used for transferring fuel oil from the storage tank to the day tanks should additional fuel oil be required. In addition, a connection is available outside the diesel rooms to pump oil directly into the day tanks from an oil tanker truck.

Each emergency generator's fuel oil day tank has its own makeup control system which is redundant to and independent from that of the other. The emergency generator tank makeup control systems interface with the fuel oil transfer pump control systems. Each day tank makeup control system has its own separate control station and is powered from its respective channel. Each day tank makeup control system can independently demand makeup via either fuel oil transfer pump P-18A or P-18B.

The fuel oil transfer pump motors and their power and control circuits are non-Class 1E. These circuits are supplied from redundant channels but are not separate and independent. Neither transfer pump's control logic gives the emergency generator day tank makeup systems preference over or isolates those non-essential makeup systems which are also served. The arrangement of the fuel oil transfer pump power and control circuits requires operator action to establish fuel transfer via either pump during loss of off-site power conditions. This arrangement minimizes the potential for inadvertent fuel oil transfer to system ruptures caused by natural phenomena.

Transfer pump P-18A can be controlled either manually, or automatically via level controls. It is powered from non-Class 1E MCC-8, which is capable of being manually loaded onto EDG 1-2. Those portions of the P-18A control circuit which carry the demand signals from the two independent emergency generator day tank makeup control systems are not electrically isolated or separated from each other; nor are they isolated or separated from circuits for non-essential makeup systems. Transfer pump P-18B can only be controlled manually. It is powered from Class 1E MCC-1, which is automatically loaded onto EDG 1-1. Those portions of the P-18B control circuit which carry the demand signals from the two independent emergency generator day tank makeup control systems are not electrically isolated or separated from each other; nor are they isolated or separated from circuits for non-essential makeup systems.

Each diesel engine has its own self-contained jacket cooling and heating system. A jacket water pump is engine driven with a temperature controlled three-way valve which diverts part of the water through a jacket water heat exchanger which is cooled by the plant service water system. As is shown on Figure 9-1, each heat exchanger is fed from a separate critical service water header. The jacket water pump on each diesel is connected to a surge line running to a 30 gallon expansion and makeup tank located approximately 10 feet above the crankshaft centerline. Makeup water is from the primary system makeup water system. When the engines are not running, the jacket water is heated by two thermostatically controlled heating elements mounted in the engine jackets.

8.4.2 STATION BATTERIES

8.4.2.1 Design Basis

The batteries are designed to furnish continuous power to certain normal Plant control and instrumentation circuits, and to control and instrumentation circuits associated with the engineered safeguards systems. They are also used to supply emergency Plant lighting. Two identical batteries feeding separate dc control centers are provided to assure reliability.

8.4.2.2 Description and Operation

Description - The batteries are of the lead calcium type; the most reliable type presently known. Special reinforced seismically qualified battery racks with high impact cell spacers are provided to meet the seismic criteria of CP Co Design Class 1 and to prevent damage from shifting of the battery cells.

Each battery is housed in its own ventilated room in the CP Co Design Class 1 portion of the auxiliary building. A sail switch is mounted in the ventilation duct to warn the operator in the control room of a loss of battery room ventilation which could lead to accumulation of hydrogen.

Normal Operation - The batteries are kept fully charged at approximately 131 volts by the battery chargers. Periodically, the voltage is raised to approximately 138 volts for equalization of the charge on the individual battery cells. Since the batteries are normally connected to the dc switchgear, they will automatically absorb any sudden load changes that may occur on the system.

Emergency Operation - On loss of normal and standby ac power, the batteries will supply power to all preferred ac and dc loads, until one of the diesel generators has started and can supply power for the chargers.

System Monitoring - In order to ensure the availability of the batteries, several annunciations are provided in the control room to warn the operator of battery conditions (see Subsection 8.3.5.2). An operator is dispatched to the battery chargers and metering panels on a shiftly basis for a general inspection of the dc power system status. Proper trickle charging of the batteries by the chargers is monitored by the dc bus undervoltage alarm in the control room and by verification of circuit breakers position. In addition, the batteries are monitored on a monthly basis as required by Technical Specifications.

Testing - The batteries are tested in accordance with IEEE 450-1975, IEEE 308-1974, NRC BTP EICSB 6 and the "Standard Technical Specifications for Combustion Engineering Pressurized Water Reactors" (NUREG-0212). The tests are as follows:

1. At least once per 18 months, during shutdown, a battery service test is performed to verify that the battery capacity is adequate to supply and maintain in operable status all of the actual emergency loads for four hours.

2. At least once every 60 months, during shutdown, a battery performance test is performed to verify that the battery capacity is at least 80% of the manufacturer's rating.

Technical Specifications describe additional surveillance requirements for monthly and quarterly testing.

8.4.2.3 Design Analysis

The batteries have ample capacity to supply all dc loads and the preferred ac loads during a complete loss of ac power for at least four hours, assuming neither diesel emergency generator is available. The batteries are designed to furnish their maximum load down to an operating temperature of 70°F without dropping below 105 volts, and the equipment supplied by the batteries is capable of operating satisfactorily at this voltage rating. The sediment space in the individual battery cells is sized such that the battery cannot develop an internal short circuit during its normal life.

The worst battery loading case per EA-ELEC-LDTAB-009 (Reference 11) assumes that neither of the two battery chargers, which are available for each battery, is operating. This loading is based upon the required opening and closing sequences of the 4,160, 2,400 and 480 volt circuit breakers, and upon solenoid, inverter, emergency lighting, annunciator and dc motor operations.

The four-hour minimum used in the battery sizing design is conservative and allows ample time to place either of two chargers in service before adversely affecting the battery performance.

Battery calculations in accordance with the guidelines provided in NUMARC 87-00 verified that the Class 1E batteries have capacity to meet station blackout (SBO) loads for four hours. This assumes that loads not needed to cope with the station blackout are stripped. The loads and the stripping procedure are identified in Procedure EOP-3. EOP-3 requires monitoring and stripping loads prior to 30 minutes of SBO. The battery analysis shows manual load shedding at 30 minutes and plant procedures are consistent with this. NRC final approval was received in a SER (June 25, 1992).

In the event of a Loss of Coolant Accident and coincident loss of offsite power with emergency generators available, one charger for each battery will be energized automatically from an emergency generator to supply dc loads. Hence, the station battery will carry full load for approximately 10 seconds during a DBA and then will be supported by the battery charger.

8.4.3 TURBINE GENERATOR COASTDOWN

8.4.3.1 Design Basis

The coastdown circuits are designed to utilize the kinetic energy of the turbine generator to maintain primary coolant flow for approximately 10 seconds after a turbine generator trip when the trip occurs simultaneously with a power system grid failure.

8.4.3.2 Description and Operation

Description - The turbine generator voltage regulator temporarily maintains excitation during coastdown. A 362CD time delay relay that has been set at 10 ± 1 second is initiated by the 386C turbine coastdown relay. The coastdown control circuits, as shown on Figures 8-19, 8-21 and 8-28, consist of the necessary relays and components to maintain generator excitation and delay the tripping of Station Power Transformer 1-1 4,160 volt incoming breakers for the first 10 seconds of coastdown.

Operation - The coastdown circuits operate only when the turbine generator trips and there is no start-up transformer power. These circuit components act to delay tripping the 4,160 volt station power incoming breakers until after a 10 second time delay. The circuit components also act to remove the voltage regulator and exciter from service when the 4,160 volt station power breakers are tripped. Utilization of the turbine generator inertia is blocked whenever a fault occurs within the electrical system of the main generator.

8.4.4 EMERGENCY POWER SUPPLY FOR PRESSURIZER HEATERS

8.4.4.1 Design Basis

The pressurizer heaters' power supply is designed to supply one half of the heaters from an offsite power source and the other half from an emergency power source. The heaters connected to the offsite power source may be manually switched to an emergency power source to provide redundant emergency power to the heaters as required by NUREG-0737.

8.4.4.2 Description and Operation

Description - The pressurizer heater 2,400 volt power connections are shown on Figures 8-6 and 8-7. The pressurizer heater power supply is such that one half of the heater capacity (750 kW nominally) can be supplied from an offsite power source (via 2,400 volt Bus 1E); the other half of the pressurizer heaters can be supplied from the emergency power source (via 2,400 volt Bus 1D).

During 1980, modifications were made to the Pressurizer Heater Transformer 15 feeder breaker on 2,400 volt Bus 1E and the Dilution Water Pump A feeder breaker on 2,400 volt Bus 1C. This modification allows Pressurizer Heater Transformer 15 feeder to be manually switched from Bus 1E to Bus 1C, providing flexibility to operate half the heater banks from Bus 1C or 1D.

Operation - Switching the heaters from Bus 1E to Bus 1C requires that the Pressurizer Heaters Transformer breaker on Bus 1E be racked out, the Dilution Water Pump breaker on Bus 1C be opened, the dilution water pump leads be removed from the load side of the breaker, and the "jumper cable" be connected between Bus 1C and the load side of the Pressurizer Heaters Transformer breaker on Bus 1E. The control room then closes the Dilution Water Pump breaker on Bus 1C and the pressurizer heaters can be energized.

See Subsection 4.3.7 for additional operating details for the pressurizer heaters.

8.5 RACEWAY AND CABLING SYSTEM

8.5.1 DESIGN BASIS

8.5.1.1 Fire Protection Features

The design features of the raceway and cabling system for fire protection of safety-related cabling are described in this section. For additional fire protection-related features, refer to Section 9.6.

8.5.1.2 Electrical Penetrations of Reactor Containment

10 CFR 50, General Design Criterion 50, as implemented by Regulatory Guide 1.63 and IEEE Standard 317-1972, requires that electrical penetrations be designed so that the containment structure can accommodate, without exceeding the design leakage rate, the calculated pressure, temperature and other environmental conditions resulting from any Loss of Coolant Accident (LOCA).

8.5.2 DESIGN DESCRIPTION

As noted in Subsections 8.1.1 and 8.1.2, the engineered safeguards electrical power and control system buses are divided into two channels and the loads into two groups. Each channel consists of the following buses and power sources: one 2,400 volt bus, two 480 volt load centers, four 480 volt motor control centers, one dc distribution center, one battery, two battery chargers, two preferred ac buses, two inverters and one diesel generator. The power source for driven equipment and the control power for that system are supplied from the sources in one channel. Where redundant equipment is utilized in one load group, as in the case of the safety injection valves, the redundant equipment is supplied from the opposite channel.

The raceways and containment penetrations for these systems are also divided into two groups according to the separation criteria given in Subsection 8.5.3.2. Physical separation is maintained between the two raceway systems and between the two penetration areas. The interconnecting cables for any one channel are run in their respective raceway system.

The Reactor Protective System (Section 7.2) is divided into four channels supplied from the four preferred ac buses. The raceways for these systems are divided into two physically separated raceway systems and the two channels within one raceway system are further separated by a metal barrier within the raceway according to the separation criteria given in Subsection 8.5.3.2. Separate containment penetrations are used for like circuits.

The schematic diagrams provide all the information necessary for making circuit schedule and connection diagrams. The connection block diagram shown on each schematic diagram shows all the interconnecting wire, where separation is required between redundant circuits and how separation is to be effected. Scheme numbers and relay numbers are coded with odd numbers indicating Channel 1 and even numbers for Channel 2. The allocation for the power source is shown for each scheme.

The circuitry of functions which might be designated as nonsafety related are contained in the same safety-related cables serving the safety-related equipment. (Examples: remote indicating lights for valves, breakers, etc.) This circuitry has then been treated as "associated" circuitry within IEEE 384 definition and requirements.

The cable and wire connected to devices and instrumentation which are required to operate during a design basis accident (DBA) have been qualified to assure satisfactory operation through and following the accident. All essential cabling has been shown to have a 40 year qualified life. In-tray splices are not recommended but when required, use a qualified and tested design that is hermetically sealed to the expected environment (see subsection 8.1.3).

Cables installed in ventilated trays, conduit or underground ducts are thermally sized in accordance with NEC or IPCEA/ICEA ampacity values (depending on cable physical size) of concentric stranded insulated cable for the conductor operating temperature of the insulation. Insulation type may be of thermosetting, rubber or plastic. Ampacities are adjusted based on actual field conditions when possible. These adjustments may include, but not be limited to, conductor operating temperature, ambient temperature, cable overall diameter, tray depth of fill, conduit percent fill, and fire-stops. The basic methodology for applying ampacity adjustments is described in Reference 12. The methodology is further modified as described in Reference 13.

High-voltage cables are run in separate raceways from low-voltage power and control cables. Low-level signal cables are kept in separate raceways. Cables installed in stacked trays are arranged to have the highest voltage located in highest level tray and low-level signal cable in the lowest tray.

No documentation of the original design basis for the sizing of cables for short circuit has been found. Station power short-circuit levels and discussions with the original AE, however, indicate that the three-phase ac cable short-circuit protection design considered the fault current due to a fault at the load and used high speed fault clearing to prevent cable damage. This design sizes the protective device for a fault at the load to prevent the conductor temperature from exceeding 250°C for thermosetting insulated cable, 200°C for rubber insulated cable, and 150°C for plastic insulated cable. If a fault occurs on the cable, the entire cable upstream of the fault would be inspected and, depending on the fault location and resulting short-circuit current, the appropriate sections would be replaced. This is the present design criteria being used. The 125 volt dc protection design considers the fault current available at the source side of the feeder protective device.

8.5.3 DESIGN EVALUATION

The design of the Plant electrical penetrations is similar to those of other plants, the probability of electrical failure of the penetrations is low, and any resultant leakage path would be small. Analysis shows that electrical penetration failure contribution to containment failure by leakage is very small; i.e., the size of a potential leakage path is small and the probability of penetration electrical failure resulting in leakage is low.

The following subsections describe how the raceway and cabling system meets NRC BTP CMEB 9.5-1, Regulatory Guide 1.75 and Appendix R to 10 CFR 50.

8.5.3.1 Compliance With Regulatory Guide 1.75

The safety-related cabling system does not fully meet the requirements of Regulatory Guide 1.75 since the Plant was designed and constructed before the Guide was established. As a result, fixed automatic water fire suppression systems have been provided in areas of dense safety-related cables. Manual hose stations and portable extinguishers are provided as backup.

Although IEEE 383 was not in existence at the time the Palisades electrical cabling was purchased and installed, the cable was specified to meet the vertical flame tests in accordance with IPCEA S-19-81 and ASTM D470-59T. While such tests, as well as the IEEE 383 tests, provide a measure of comparability of fire retardance between various types of cables, they cannot be considered as indicative of their behavior when found in the configurations in the Plant. Cable insulation has thus been considered as combustible material for fire protection design considerations (subsections below, Section 8.7 and Section 9.6).

Starting in 1979, new cable installations utilize to the extent practical, cable construction that does not give off corrosive gases while burning.

8.5.3.2 Raceway and Cabling Separation Criteria

All circuits designated as belonging to a safety-related power distribution, Reactor Protective System, engineered safeguards or other safety-related system channel are run in separate raceway systems. The raceway system includes conduit, trays, wall and floor penetrations, containment penetration, panel wire troughs, etc. In general, the circuit isolation and separation requirements are met by the use of physically separate raceways which can afford fire and missile protection. The raceway systems are so arranged that a single failure cannot affect both channels of a power or control circuit. In designing the raceway system for "channeled" circuits and in the routing of these circuits, consideration has been given to the type of hazards that could be present in regard to potential fire, as well as size and type of missiles that may be generated by the equipment in the area. Physical separation (distance) has been considered as the most reliable method of providing the circuit separation and isolation. When raceways are run near one another, a fire barrier and/or a missile shield is provided between the raceway systems.

According to FSAR Amendment 15 of August 26, 1968, original Plant construction did not require rigorous separation of vital and nonvital cables. Amendment 15 states, "Nonvital cables, except for those required to operate during a DBA, are the same as the cables used in the engineered safeguards circuits. They are sized, rated, protected and, except for separation, use identical type raceways sharing, where convenient, the same trays as engineered safeguards cables." IEEE 384 would require that these "nonvital" cables be classified as "associated circuits" and designed and installed accordingly. Separation as required by IEEE 384 was not an original design requirement and the Plant is not in total conformance with this standard.

The following designations have been utilized for the Palisades Plant circuits:

Left	Right
Channel 1	Channel 2
Channel 3	Channel 4
System 1	System 2
System 3	System 4

Following is a description of the intended separation requirements by channel designation. A few circuits have been discovered that are not separated as described below. When deviations from separation requirements are identified they are evaluated for acceptability as-is or rerouted.

Left Circuits

Circuits designated as "Left" must be routed in trays and conduits separate from circuits designated as "Right," "Channel 2" or "Channel 4."

"Left" channel circuits may be run in Nonclass 1 building above ground if there exists a redundant "Right" circuit. Typical of a Nonclass 1 structure is the turbine building. The "Left" circuits may be routed with nonredundant circuits or circuits designated as a "System" circuit or with "Channel 1 or 3" circuits.

Right Circuits

Circuits designated as a "Right" must be routed in trays and conduits separate from circuits designated as "Left," "Channel 1" or "Channel 3." The circuits may not be routed in a Nonclass 1 structure. The circuits may be run with nonredundant circuits or with "Channel 2 or 4" circuits or with any "System" circuits.

Channel 1 Circuits

Circuits designated as "Channel 1" must be routed in raceways separate from "Channels 2 and 4" circuits and separate from "Right" circuits.

They can be routed in same raceways as "Channel 3" circuits but circuits must be separated by a barrier between them. "Channel 1" circuits may be run in Consumers Design Nonclass 1 structures if there are redundant Channels 2 and 4 circuits in a protected routing. A Channel 1 circuit may be run with nonredundant circuits, "Left" circuits or "System" circuits.

Channel 3 Circuits

Similar requirements as a "Channel 1" circuits.

Channel 2 Circuits

Circuits designated as "Channel 2" must be routed in raceways separate from "Channels 1 and 3" circuits and separate from "Left" circuits. They can be routed in the same tray, penetration, etc, as "Channel 4" if a barrier is provided between them. "Channel 2" circuits must not be run in Consumers Design Nonclass 1 structures. "Channel 2" may be routed with "Right" and nonredundant circuits or any "System" designated circuits.

Channel 4 Circuits

Similar requirements as "Channel 2" circuits.

System 1 Circuits

Circuits designated as "System 1" must be routed separately from "System 2" circuits.

A System 1 circuit may be run with any other like-type circuits; i.e., power, control or instrumentation circuits.

System 2 Circuits

Similar requirements as "System 1" circuits.

System 3 Circuits

Any circuits designated as "System 3" must be routed in separate raceways from "System 4" circuits.

A "System 3" circuit may be routed with any other like-type circuit.

System 4 Circuits

Similar requirements as "System 3" circuits.

Compliance With Appendix R to 10 CFR 50

In order to ensure operability of both Emergency Generators 1-1 and 1-2 in the event a fire damages control circuitry of the equipment in the control room or the cable spreading room, all support systems' power and control cabling for the generators, including the power source for the engine crankcase blowers are not routed through these areas. Furthermore, the terminations of control room and cable spreading room routed circuitry for generators and 2,400 volt Bus 1C and 1D switchgear positions are identified readily so that the sliding links of the terminal blocks can be opened to isolate the damaged circuitry. Isolating transfer switches are provided for breakers on Bus 1C which are required for hot shutdown operation. No slide link operations are credited for achieving or maintaining hot shutdown for 10 CFR 50 Appendix R.

In order to ensure operability of one emergency generator in the event of a fire in the emergency generator or switchgear room belonging to the other channel, the generator power cables, and control and instrument cables from the emergency generator are not routed in the opposite generator or switchgear room.

Other specific routing criteria have been established to ensure safe shutdown according to 10 CFR 50, Appendix R as described in Section 9.6.

8.5.3.3 Raceway and Cabling Fire Barriers

Fire barriers have been provided to separate the turbine building from the auxiliary building and to isolate one safe shutdown train from the associated redundant train or alternative safe shutdown equipment. Based on the type and quantity of combustibles present, the basic fire resistance of the barriers would prevent the spread of fire between fire areas.

Cable and cable tray penetration of fire barriers (vertical and horizontal) in safety-related areas of the Plant have been sealed to give protection at least equivalent to that of the fire barrier. The design of fire barriers for horizontal and vertical cable trays meets the requirements of ASTM E119, "Fire Test of Building Construction and Materials," including the hose stream test. Where fire barrier penetration seals require fire resistance characteristics, these seals in safety-related areas have been sealed to a rating equivalent to that required of the barrier. Piping penetrations of fire barriers are sealed in a similar fashion. The adequacy of seals has been demonstrated by testing [Factory Mutual Research Test Reports for Wall Penetrations (4/26/78) and Floor Penetrations (5/10/78)] and/or analysis.

The few cable tray sections in the containment building which communicate between the left and right cable tray systems are fitted with fire stops. These stops are intended to prevent a fire originating in one tray system from traveling along an interconnecting tray and affecting the other tray system.

8.5.3.4 Cable Spreading Room Protection Design

This area contains 480 volt dry-type transformers, 480 volt switchgear, cables for power, instrumentation and control for safety-related and nonsafety-related systems, and other equipment related to safety-related ac and dc power supplies. The significant combustible in this area consists of a large quantity of cable in open cable trays stacked three or four levels deep.

In order to meet the intent of Regulatory Guide 1.75 and NRC BTP CMEB 9.5-1, the following design features are provided for protection against a fire in the cable spreading room:

1. Fire detection provided by flow alarms in the sprinkler system.

2. Fire extinguishment provided by an automatic sprinkler system backed up by water hose stations and portable extinguishers.

3. Switchgear protected against flooding by mounting on curbs.

4. Physical separation and barriers used to separate redundant divisions of safe shutdown cables.

5. Capability independent of this area for achieving safe shutdown using centralized control and instrumentation (Panel C-150).

6. Smoke detectors to detect incipient fires.

7. Ladder to enhance manual fire-fighting capability.

8. Fire retardant coatings to close gaps in barriers.

9. Seal for cable penetrations on both sides.

10. Manual suppression capability to suppress fires in lower trays that may be shielded from sprinkler system water.

11. The wall to the turbine building is at least a 3-hour wall and to the adjacent switchgear room is at least a 2-hour wall. The floors and ceilings have a 3-hour fire resistance.

12. The ventilation ducts leading to the turbine building and into the battery rooms have fusible link fire dampers installed.

13. To ensure fire-fighting ability, there are two remote and separate entrances into the cable spreading room, one from the adjacent switchgear room; the cable trays are all installed above the floor-mounted switchgear cabinets starting about seven feet high and extending to the ceiling except for the vertical cable runs at the south side of the room; there are four-foot aisles between floor-mounted equipment for ladders or fire-fighting equipment.

8.5.3.5 Cable Penetration Rooms Protection Design

There are two cable penetration areas into containment totally separated from each other by distance and fire barriers, one area being at the north side and the other at the southwest side of the containment. When cables penetrate fire barriers, fire rated cable penetration seals are provided. Each area contains cables for safe shutdown equipment redundant to the other area.

The significant combustible in each of these areas is a moderate amount of electrical cable insulation stacked in open cable trays.

An unsuppressed fire in either of the penetration areas could cause damage to cables affecting one division of safe shutdown systems but would not cause loss of shutdown capability. Refer to Section 9.6.

Fire detection is provided by smoke detection and water flow alarms which are actuated by flow to the sprinkler system. Fire extinguishment is provided by an automatic sprinkler system in each area backed up by portable extinguishers and hose stations located in adjacent areas.

8.5.3.6 Raceway Runs Protection Design

Cable trays, raceways, conduit, trenches or culverts are used only for cables; no miscellaneous storage is present. No piping for flammable or combustible liquids or gases is installed in these areas.

The tunnel from the north penetration area to the Switchgear Room 1D and the cable spreading room can be vented with the HVAC system or manually vented with the door to the switchgear room from the outside. Also the tunnel can be vented with the door from the outside leading into the north penetration room.

All cables entering the control room pass through penetration seals and all terminate in the control room cabinets. There are no floor trenches or culverts in the control room.

8.5.3.7 Safety-Related Cabling Routing Via Nonsafety-Related Areas

Routing of electrical cables via nonsafety-related areas to the following safety-related equipment and areas: (1) auxiliary feedwater pumps and associated valves; (2) intake structure; and (3) electrical penetration area, does not allow a fire to cause loss of redundant equipment because of the following routing features:

1. The motor feeder for motor-driven Auxiliary Feedwater Pump A is in underground conduit from Bus 1C to the motor.

2. Steam Supply Valve A to the Auxiliary Feedwater Pump B turbine and associated circuitry is located in the reactor auxiliary building.

3. Steam Supply Valve B to the Auxiliary Feedwater Pump B turbine is located above the floor of the turbine room. The associated circuits route through cable trays and conduit in the area.

4. The Auxiliary Feedwater Pump B turbine driver speed control is mechanical and no electrical cable routing is involved.

5. Auxiliary feedwater flow control valves and circuits are located in the west safeguards room and the component cooling pump room.

6. Circuit routing for service water pumps is in conduit through the turbine room, or in underground duct.

7. Circuit routing to the southwest penetration area is via a combination of underground conduit and trays in the turbine building.

8. The turbine lube oil storage area is adequately separated in an interior structure inside the turbine building. The room has fire suppression and is adequately enclosed by fire walls. The room has a recessed floor to contain the single largest lube oil storage tank inventory without leakage.

The consequences of a fire in the turbine room are described in Section 9.6.

8.5.3.8 Containment Building Routing Protection

The containment building can be isolated to contain any fire source and the containment boundary will prevent any outside fire from entering the containment.

The lube oil system for each primary coolant pump has approximately 80 gallons of lube oil. The small amount of oil associated with each primary coolant pump does not justify a suppression system. It is not expected that a postulated fire at one pump would spread to another since an oil collection system is provided for the primary coolant pumps to collect any oil spill.

Cable trays, cables and penetration areas are separated into two divisions with fire stops in trays communicating between the two divisions (see Subsection 8.5.3.3).

An unmitigated fire in either cable penetration area could involve all the cables in one penetration area, but would not affect safe shutdown because only cables of one division are located in each penetration area.

As a result of this potential for fire, fire detection devices are provided in the reactor containment instrument room and cable penetration area; primary coolant pump bearing temperature and motor winding temperature readout are also available to give indication of a fire in the primary coolant pump area; portable carbon dioxide and water extinguishers are provided. Also, due to the difficulty of reaching some locations with a hand extinguisher, water hose stations are provided in containment with adequate hose to suppress fires in cables in cable trays.

8.5.3.9 Other Areas Routing Protection

For routing protection in switchgear rooms, emergency generator rooms and battery rooms, see Section 8.7. For routing protection in control station areas, see Sections 7.4 and 7.7.

All areas containing safe shutdown cabling have been designed to protect against a fire damaging both channels of safety cables and raceways, or an alternate method for the function is provided in a separate fire area.

8.6 AUTOMATIC TRANSFER, VOLTAGE PROTECTION AND LOAD SHEDDIN G CONTROLS

8.6.1 DESIGN BASIS

The automatic transfer control system is designed to monitor and select available offsite power sources and permit transfer of the 4,160 volt and 2,400 volt loads to the available offsite source upon loss of the normal power source. Redundant control circuits are provided for transfer of source power for the redundant 2,400 volt emergency buses (1C and 1D).

Voltage protection and load shedding features for safety-related buses at the 2,400 volt and lower voltage levels are designed in accordance with 10 CFR 50, Appendix A, General Design Criterion 17 and the following features:

1. Two levels of automatic voltage protection from loss or degradation of offsite power sources are provided. The first level provides normal loss of voltage protection. The second level of protection has voltage and time delay set points selected for automatic trip of the offsite sources to protect safety-related equipment from sustained degraded voltage conditions at all voltage levels in accordance with ANSI C8.4.1-1977, with coincidence logic to preclude spurious trips. Maximum time delays for second level trip do not exceed the maximum time delay assumed in Chapter 14 analysis for engineered safeguards actuation while allowing short duration bus voltage disturbances without trip and short enough to prevent damage or failure of safeguards systems and components. This second level of protection meets IEEE 279-1971, IEEE 308-1978, IEEE 501-1978 and its components are located in a controlled atmosphere not requiring environmental qualifications.

2. The voltage protection system automatically prevents load shedding of the safety-related buses when the emergency generators are supplying power to the safeguards loads. Automatic bypass and reinstatement is verified by periodic testing.

3. Control circuits for shedding of Class 1E and Nonclass 1E loads during a Loss of Coolant Accident themselves are Class 1E or are separated electrically from the Class 1E portions.

8.6.2 DESCRIPTION AND OPERATION

General Description - The automatic transfer, voltage protection and load shedding controls are shown on Figures 8-18 through 8-27. The controls for the safety-related 2,400 volt Buses 1C and 1D consist of redundant transfer, voltage protection and load shedding circuits connected to separate Plant batteries, circuit breaker controls for Bus 1C being fed from one battery and controls for Bus 1D being fed from the other battery. Separate voltage sensing units on each bus are utilized for each of the circuits.

During emergency conditions, a turbine or generator trip will trip circuit breakers for the nonvital 4,160 volt Station Power Transformers 1-1 and 1-3, initiating transfer to the 4,160 volt Start-Up Transformers 1-1 and 1-3. The transfer to the start-up source will be completed within ten cycles after initiation with a bus dead time of approximately one-and-one-half cycles. The fast transfer will permit all auxiliaries to continue to operate normally. During normal shutdown conditions, the 4,160 volt auxiliary power system is manually transferred to the start-up source.

The 2,400 volt system, which includes the emergency buses, is normally powered directly from offsite power via Safeguards Transformer 1-1. In this configuration, a turbine or generator trip will not result in a fast transfer of the 2,400 volt buses to the alternate source. Capability is provided in the design to allow powering the 2,400 volt buses from Station Power Transformer 1-2. When operating in this configuration, a turbine or generator trip will initiate a fast transfer to Start-Up Power Transformer 1-2. The transfer to the standby source will be completed within 10 cycles after initiation with a bus dead time of approximately one-and-one half cycles. The fast transfer will permit all auxiliaries to continue to operate normally.

The 2,400 volt auxiliary power system normally remains on the Safeguard Transformer source but can be manually transferred to the start-up source or, after transferring to the start-up source, can be manually transferred to the station power source.

In order to permit the main transformer backfeed mode of operation (Subsection 8.2.3), the fast transfer on turbine trip is blocked by opening the generator isophase bus disconnect switch. No automatic transfer is provided for a transfer from the start-up transformers to the station or safeguard transformers. This operation must be done manually.

4,160 Volt System - Automatic transfer of the 4,160 volt buses from the normal power source (Station Power Transformers 1-1 and 1-3) to the start-up power source (Start-Up Transformers 1-1 and 1-3) is initiated by turbine trip or generator trip.

Automatic transfer is blocked if the start-up transformer voltage is low. Low voltage is sensed by either 4,160 volt Undervoltage Relays 227-5 and 227-6, which actuate Lockout Relays 227-X5 and 227-X6. Blocking of transfer to each start-up transformer is independent, thereby permitting transfer of 4,160 volt buses if one transformer is inoperable.

Automatic transfer of a faulted bus will also be blocked if a safeguard/station power transformer incoming breaker is tripped on overcurrent. The lockout relay must be manually reset to close the faulted bus breaker after the overcurrent condition. When a turbine generator or reactor trip occurs with offsite power unavailable, turbine generator inertia is utilized to temporarily supply 4,160 volt Buses 1A and 1B. The generator excitation will temporarily be maintained. The generator excitation and primary coolant pumps will remain energized and primary coolant flow will be maintained for approximately 10 seconds. After the 10-second time delay, the generator excitation and the 4,160 volt station power bus incoming breakers will be automatically tripped (see Subsection 8.4.3 for details).

<u>2,400 Volt System</u> - The 2,400 volt system is normally powered directly from offsite power via Safeguards Transformer 1-1. Automatic transfer from this source to the standby source (Start-Up Transformer 1-2) is initiated by any fault which results in clearing of the Switchyard "F" bus, or by differential zone relaying which detects a fault between the Safeguards Bus and the 2,400 volt buses. If the 2,400 volt buses are connected to Station Power Transformer 1-2, an automatic transfer to the standby source (Start-Up Transformer 1-2) is initiated by a turbine trip or generator trip. The automatic transfer scheme in effect is determined by monitoring the status of the safeguard bus feeder breakers to indicate if Safeguards Transformer 1-1 or Station Power Transformer 1-2 is the source of power to the 2,400 volt buses.

Automatic transfer is blocked if the start-up transformer voltage is low. Low voltage is sensed by either 2,400 volt Relay 127-5 or 127-6, which actuate Lockout Relays 127-X5 and 127-X6. Blocking the transfer to the start-up transformer is independent for each 2,400 volt Bus 1C and 1D.

The automatic transfer will initiate a trip of the emergency generator breaker to prevent the generator from being paralleled automatically and the bus from being loaded onto the emergency generator if the transfer fails.

Automatic transfer of a faulted bus will also be blocked if a safeguard/ station power transformer incoming breaker is tripped on overcurrent. The lockout relay must be manually reset to close the faulted bus breaker after the over-current condition.

Each 2,400 volt Bus 1C and 1D is equipped with two levels of voltage protection relays, with their functional logic as shown in Figure 7-14, Sheet 3. The first level undervoltage Relays 127-1 and 127-2 are set at approximately 77% of rated voltage with an inverse time relay. These relays protect against a sudden loss of voltage as sensed on the corresponding bus using a three-out-of-three coincidence logic. The actuation of these relays will trip their respective incoming bus circuit breakers, start their respective emergency generators, initiate bus load shed, and activate annunciators in the control room. The emergency generator circuit breaker is closed automatically upon establishment of satisfactory voltage by the use of voltage protection Relays 127D-1 and/or 127D-2 (see Figures 7-14 Sheet 6 and 7-18).

The second level of voltage protection undervoltage Relays 127-7 and 127-8 are set at approximately 92% of rated voltage. These relays protect against sustained degraded voltage conditions on the corresponding bus using a three-out-of-three coincidence logic. These relays have a built-in 0.5 second time delay, after which their respective emergency generators will receive a start signal and activate annunciators in the control room. If a bus undervoltage exists after an additional six seconds, then the respective incoming bus circuit breaker will be tripped and a bus load shed will be initiated.

In addition to initiating the bus load shed feature, the undervoltage devices also initiate reset of the service water, component cooling water and low-pressure safety injection pump standby auto start feature. This feature, if enabled, would normally start the pump on detecting low system discharge pressure. Resetting this feature on bus undervoltage prevents closing the pump breaker onto a dead bus and assures that the breaker is ready to be closed by the appropriate sequencer after the diesel generator is connected to the bus.

2,400 volt Buses 1C and 1D load shedding is blocked via auxiliary contacts of the emergency generator circuit breaker when the generator is connected to the bus.

In order not to overload Safeguard Transformer 1-1 or Start-Up Transformer 1-2 during a Loss of Coolant Accident, 2,400 volt Bus 1E and selected Nonclass 1E 2,400 and 480 volt loads fed from Buses 1C and 1D are shed upon receipt of the safety injection signal. The start-up power load shed circuits are not considered Class 1E. Load shedding of selected non-Class 1E loads upon receipt of a safety injection signal provides reserve capacity and improves voltage transients on the running supply transformer. Safeguard Transformer 1-1 (10.5 MV A rated) has approximately 2.4 MVA steady state reserve capacity following load shed, SIS block loading and manual re-energization of certain non-Class 1E loads. Start-Up Transformer 1-2 (10.6 MVA rated) has approximately 2.9 MVA steady state reserve capacity following load shed, SIS block loading and manual re-energization of certain non-Class 1E loads. Adequate in-plant voltage (ie, above the second-level undervoltage relay settings) following SIS block is assured by administrative controls which limit Start-Up Transformer 1-2 load based on the offsite power supply voltage.

System Monitoring - Safeguards load centers have trip and undervoltage alarms in the control room as well as voltage monitors for supervision of system status and voltage adequacy.

Testing - Verification of the voltage protection system relays and load shedding overall operation is performed during Plant shutdown with the emergency generators testing in accordance with Technical Specifications requirements.

8.6.3 DESIGN ANALYSIS

8.6.3.1 Automatic Transfer System

The reliability of the automatic transfer control system is assured by two independent and separate circuits controlling their respective auxiliary system breakers. The circuit is designed so that a loss of control power will not cause a false transfer; loss of control power will be annunciated. This circuit is also designed to prevent both offsite power and emergency power from being paralleled automatically. Once "no offsite power" is detected, the offsite source is locked out and the emergency generator is used to energize the engineered safeguards. When offsite power returns, the start-up or safeguard/station power transformer incoming breakers may be closed manually through the synchronizing circuit.

8.6.3.2 Voltage Protection and Load Shedding Systems

The voltage protection and load shedding systems meet the criteria outlined in Subsection 8.6.1 as evaluated below:

The voltage trip set point has been set low enough such that spurious trips of the offsite source due to operation of the undervoltage relays are not expected for any combination of unit loads and normal grid voltages.

This set point at the 2,400 volt bus and reflected down to the 480 volt buses has been verified through an analysis to be greater than the minimum allowable motor voltage (90% of nominal voltage). Motors are the most limiting equipment in the system. MCC contactor pickup and drop-out voltage is also adequate at the set-point values. The analysis ensured that the distribution system is capable of starting and operating all safety-related equipment within the equipment voltage rating at the allowed source voltages. The power distribution system model used in the analysis has been verified by actual testing.

The time delays involved will not cause any thermal damage as the set points are within voltage ranges recommended by ANSI C8.4.1-1971 for sustained operation. They are long enough to preclude trip of the offsite source caused by the starting of large motors and yet do not exceed the time limits of safeguards actuation assumed in Chapter 14.

Once the emergency generator is connected to its bus, the load shed is blocked by interlocks and auxiliary relays and load shedding is reinstated upon a trip of the emergency generator.

Load shedding of 2,400 volt Bus 1E and other nonessential loads provide a more than adequate margin on Start-Up Transformer 1-2 and Safeguard Transformer 1-1 to ensure reliable power is available for all engineered safeguards loads.

Load shedding on offsite power trip and load sequencing once the diesel generator is supplying the safety buses are tested periodically. The load shed bypass circuit and a simulated loss of the diesel generator with subsequent load shedding are also tested. Calibration of the undervoltage relays verify that the time delay is sufficient to avoid spurious trips.

8.7 PHYSICAL SEPARATION, ELECTRICAL ISOLATION AND SUPPORT SYSTEMS

The physical and electrical separation of the two safety-related power distribution channels and their support systems is the subject of this section. Protection against fire is included also in this subsection. Protections against physical phenomena (flooding, tornado, missiles, etc) are described in Chapter 5.

8.7.1 ELECTRICAL ISOLATION (See Figure 8-1)

Electrical interconnections between safety-related Channels 1 and 2, between Channel 1 or 2 nonsafety buses and between Channel 1 or 2 and nonsafety loads are evaluated below according to Regulatory Guide 1.6 requirements. The objective of the evaluation is to identify single failures that could cause simultaneous failure of both channels.

The 2,400 volt Buses 1C and 1D (and, therefore, Diesel Generators 1-1 and 1-2) can be tied together only if either offsite Feeder Breakers 152-105 and 152-203 (normally closed) or Breakers 152-106 and 152-202 (normally open) are closed. On loss of the offsite source, these breakers each receive an automatic open signal from independent low voltage sensing devices located on Bus 1C for Breakers 152-105 and 152-106 and on Bus 1D for Breakers 152-202 and 152-203. Thus, there is no single failure that can cause the interconnection of Buses 1C and 1D, and the paralleling of the standby onsite power sources after loss of offsite power.

The 480 volt Buses 11 and 12 may be tied together by closing Breakers 52-1118 and 52-1217 (normally open). An interlock prevents the closure of these tie breakers if both breakers connecting the redundant sources to Buses 11 and 12 (Breakers 52-1102 and 52-1202, respectively) are closed; one of these two supply breakers must be open in order to close the tie breakers.

A tie breaker auto trip will occur whenever an undervoltage condition exists on 2,400 volt Bus 1C or 1D and the associated diesel generator output breaker is open. An undervoltage condition on Bus 1C opens Tie Breaker 52-1118 and an undervoltage condition on Bus 1D opens Tie Breaker 52-1217.

Load centers and motor control centers belonging to safety-related redundant channels are not intertied except as indicated above. Interties via nonsafety-related buses are described below:

Associated nonsafety-related 480 volt Bus 77 is interconnected with nonsafety-related Bus 78 by a single tie breaker. The supply breakers to Buses 77 and 78 are interlocked with the tie breaker such that one of the two supply breakers must be open to close the tie breaker. Thus, safety-related Bus 1C cannot be paralleled with nonsafety-related Bus 1E through Buses 77 and 78. Also, the load breaker on Bus 1C opens on loss of offsite power to prevent Diesel Generator 1-1 overload. Similar arrangements are provided for the interconnection between nonsafety-related 480 volt Bus 13 and 480 volt Bus 14 and for the interconnection between nonsafety-related 480 volt buses for switchyard loads.

Safety-related MCC 1 and MCC 2 both provide power to the instrument ac bus through an automatic transfer switch. Power to the instrument ac bus is normally supplied by MCC 1. With loss of MCC 1, the power source is automatically transferred from MCC 1 to MCC 2. The "break-before-make" design of the transfer switch prevents interconnection of the two motor control centers.

8.7.2 PHYSICAL SEPARATION

8.7.2.1 General

The physical separation of redundant equipment and cabling associated with the two safety-related power distribution channels meets 10 CFR 50, Appendix A, General Design Criterion 3. Review of electrical equipment fire-related design based upon the acceptance criteria of Appendix A to NRC BTP CMEB 9.5-1 has been performed and a summary included in this subsection. See Section 8.5 for generic raceway and cabling design, Sections 7.4 and 7.7 for control stations fire-related design, and Section 9.6 for fire protection equipment. These criteria were established to prevent a single fire in any area from disabling both redundant channels. In addition, 10 CFR 50.48, effective date February 17, 1981, required a reevaluation of all areas of the Plant to the separation criteria specified in Paragraph G of Section III of Appendix R to 10 CFR 50. These last criteria are met via alternate dedicated shutdown means (auxiliary shutdown systems) and associated circuits independent of cables, systems or components in the area, room or zone utilized by the normal shutdown. The descriptions of the auxiliary shutdown controls provided and associated circuits are included in Section 9.6.8 , together with their specific separation features.

The Palisades Plant has been designed with physical separation to prevent the spread of a fire in safety-related equipment areas such that the function of redundant engineered safeguards is not impaired. This separation is maintained by compartment isolation of Plant safety systems and by employing redundant equipment, controls and power supplies. In the event any single system is disabled, the redundant system and the protection afforded by compartment structural boundaries will ensure that the system safety function is not jeopardized.

An analysis of separation in the containment air room showed a nonconformance with the 20 ft. separation requirement of Appendix R Section III.G.2.d. On June 21, 1991, an exemption was granted by the NRC on the basis that the design of the room met the underlying purpose of the rule. A CPCo analysis using methodology approved by the

National Institute of Standards and Technology showed that existing separation was adequate for minimum safe shutdown instrumentation to survive the worst case fire.

8.7.2.2 Transformers

All high-voltage power transformers located within the Plant safety-related areas are dry type.

The main transformer, station power transformers and start-up transformers are located south of the containment vessel and east of the turbine building. While the transformers are within 50 feet of the containment building, there are no openings in the exterior walls of containment within 50 feet of the transformers and the fire resistance of these walls is in excess of 3 hours. All the transformers are provided with an automatic water deluge system.

8.7.2.3 Protection Against Water Damage

In order to protect electrical equipment against damage from fire suppression systems, floor drains for water removal are provided in sprinkler areas containing electrical equipment. Many areas have alternate drainage via equipment sumps, door accessways or stairwells-to-sump areas. Transformers and 480 volt vital load centers are mounted on pedestals. Switchgear are floor mounted and floor drains are provided since these areas have sprinklers. In addition, valves are available to isolate sections of the firewater piping inside buildings to preclude the buildup of water and thus prevent equipment from being incapacitated due to flooding or inadvertent operation of the fire suppression equipment.

All safety-related diesel day tanks have high curbs around the tanks or the tanks are enclosed in separate fuel oil rooms with door elevated well above floor level.

Cable trays are of the open-top ladder type made of galvanized steel with galvanized steel, painted channel or unistrut-type supports.

The safety-related cabinets housing electrical equipment in the vicinity of automatic sprinklers have been designed as drip proof, and all have had the top cable entry points in the cabinets sealed to prevent water ingress. The rooms in which this equipment is mounted are provided with floor drains to prevent flooding during sprinkler operation. Refer to Section 5.4 for a discussion on flooding.

The switchgear enclosures are louvered to limit water sprays from entering the enclosures.

8.7.2.4 Smoke Control

The power supply and controls for the ventilation systems used for smoke control are run outside the fire area served by the system with the exception of the cable spreading room and the two safety-related switchgear rooms. The normal exhaust fan for the cable spreading room and the two switchgear rooms is mounted in the cable spreading room. The emergency exhaust fan that serves the three above-mentioned rooms is mounted in Switchgear Room 1-D. All these rooms can also be readily vented with portable blowers.

All intakes are remote from locations where smoke could be exhausted.

The cable spreading room and the two switchgear rooms have both normal and emergency exhaust systems which exceed NFPA requirements. The emergency generator rooms have ample natural ventilation. Diesel day tank rooms are enclosed so that there would not be enough air to support combustion. All other locations have exhaust ventilation which exceeds the NFPA criteria.

8.7.2.5 Switchgear Rooms Protection

There are two redundant 2.4 kV switchgear rooms and an auxiliary electrical equipment room. Each switchgear room contains the switchgear to one of the redundant channels of safeguards equipment and the associated cables. A tunnel which contains cabling of one safety channel leads from Switchgear Room 1-D to the associated penetration area.

The combustibles in the switchgear rooms consist of a moderate amount of cable in open cable trays. Switchgear Room 1-C also contains piping of propane, hydrogen and acetylene. However, this piping only passes through Switchgear Room 1-C and does not service any equipment in this area. Additionally, damage to the piping is unlikely since the piping is routed near the ceiling and there is no rotating machinery in the area to present a missile hazard.

An unmitigated fire in one of the switchgear rooms could cause damage to and loss of equipment related to one channel of redundant systems but would not affect the redundant systems due to the barriers separating the rooms.

Both switchgear rooms are used as cable right of ways. Both rooms have closed head sprinkler systems to protect the cables.

The switchgear room for Bus 1C has a 3-hour fire barrier to the diesel generator room and to the turbine building.

The switchgear room for Bus 1D has a 2-hour barrier to the cable spreading room. Floors and ceilings have a 3-hour fire barrier.

Fire detection is provided by smoke detectors and flow alarms actuated by water flow in the sprinkler system. Fire extinguishment capability is provided by an automatic sprinkler system in the switchgear rooms and the cable tunnel, backed up by water hose stations and portable extinguishers.

Additional protection features are provided as follows:

- Smoke detectors for detection of incipient fires (both rooms);

- Cable penetration seals with flame retardant materials (both rooms);

- Redundant safeguards cabling is routed in each switchgear room according to the channel of power source. Local 125 volt dc distribution panels are provided such that the dc control power would not be affected in the case of switchgear room fire (see Subsection 8.3.5.2); and

- Dampers in ventilation duct penetrations of fire barriers.

8.7.2.6 <u>Emergency Generators Rooms Protection</u>

The redundant diesel generators are each housed in separate rooms, separated from each other by 3-hour fire-rated walls and doors, with a curb at each of the doors to prevent oil from seeping under the door.

The significant combustibles in each room are lube oil, diesel fuel and a small amount of electrical cable insulation. The day tank for each diesel is separated from the diesel generator room by a 3-hour rated fire barrier with diking provided to contain the complete inventory of the day tank.

An unmitigated fire in one of the rooms could cause loss of one diesel generator.

Fire detection is provided by flow alarms activated by water flow in the automatic sprinkler system. Fire extinguishment is provided by a fusible link-type automatic sprinkler system in each room backed up by water hose stations and portable extinguishers.

Floor drains are provided in both emergency generators rooms. Manual venting of smoke is provided by exhaust dampers mounted in the ceiling to the outside and by a door to the outside from the vestibule.

The day tanks are mounted in separate cubicles and are vented to the outside. The day tank cubicles are not provided with sprinkler systems. Portable fire extinguishers and hose stations serving the diesel generator rooms are available in case of a fire in the day tank cubicles. The day tank rooms do not have sufficient air to support combustion.

The transfer of diesel oil from the underground storage tank to the diesel generator day tanks can be stopped by manually tripping the power supply to the transfer pumps, isolating the supply either in the Diesel Generator 1-2 room or at the transfer pump discharge in the intake structure.

The diesel fuel oil storage tank is located outside and underground away from the auxiliary building and safety-related equipment.

8.7.2.7 Battery Rooms Protection

There are two separate battery rooms. They have a 2-hour fire barrier from the cable spreading room and from each other.

The significant combustibles in the battery room are the plastic battery cases and a small amount of electrical cable insulation. Hydrogen buildup from battery charging is precluded by a continuously operating ventilation system.

An unsuppressed fire in one of the battery rooms could cause the loss of one, but not both, of the batteries due to the low fire hazard and the fire barriers between the rooms. Hydrogen buildup on loss of ventilation flow is prevented by checking the batteries once a day and by maintaining the hydrogen concentration well below 2% volume through the ventilation exhaust system. A sail switch in the ventilation duct warns the control room of a loss of battery room ventilation.

Each room is maintained under negative air pressure with air intake from the cable spreading room through fusible link fire dampers. The battery rooms' ventilation exhausts to the outside.

Smoke detection is provided for these rooms. Fire extinguishment is provided by water hose stations located in adjacent areas and by portable extinguishers.

Considering the limited quantity of combustibles, manual fire protection is adequate to extinguish fires in these rooms.

8.7.3 SUPPORT SYSTEMS

8.7.3.1 <u>Ventilation</u>

Severe weather phenomena do not present a significant hazard to Plant electrical equipment (designed for 10° to 40°C) while the Plant is operating because simple air exchange will maintain adequate temperature control. It may be necessary to operate the diesel generators during the winter if the normal Plant heating systems fail.

Ventilation for each diesel generator room is supplied by two fans. The two fans are safety related and receive power from associated safety-related distribution systems. In addition to the normal safety related power source, vent fans V-24A and V-24B also may receive power from an alternate, non-safety related power source. The safety related power source for each fan is separated from the non-safety related power source by a manual transfer switch.

Ventilation for the remaining electrical distribution system rooms - the cable spreading room, the two 2,400 volt bus (switchgear) 1C and 1D rooms and the two battery rooms - is supplied from a single duct system. The duct system has one supply fan, one exhaust fan and one recirculation fan. The one recirculation fan is redundant to the supply and exhaust fans.

The cable spreading room can withstand a loss of ventilation for up to six hours before exceeding the upper design temperature limit. High temperature in the room is annunciated in the control room. One of the redundant fans can be connected to emergency power sources.

The 1C and 1D switchgear rooms are not affected by a loss of ventilation since no appreciable heat sources are contained in these rooms. The battery room redundant fans are powered from separate channels of safety-related sources and therefore are not vulnerable to a loss of offsite power.

Tests have demonstrated that the equipment serviced in these rooms would not be adversely affected by lack of ventilation during loss of offsite power and/or a safe shutdown earthquake as defined in Section 5.7.

8.8 MOTOR OPERATED VALVES

The Palisades Motor Operated Valve (MOV) Program satisfies the requirements of NRC Generic Letter 89-10 "Safety-Related Motor Operated Valve Testing and Surveillance" and its supplements. Generic Letter 89-10 was issued in June 1989 and superseded IEB 85-03 "Motor Operated Valve Common Mode Failures During Plant Transients Due to Improper Switch Settings." The purpose of Palisades MOV Program is to ensure safety-related MOVs are designed and maintained such that they will perform their design basis function for the life of the plant. Plant procedures identify those MOVs at Palisades subject to the requirements of GL 89-10 and describe the Palisades GL 89-10 Program as it applies to those valves.

Initial compliance to GL 89-10 consisted of a design basis review for each MOV in the Palisades GL 89-10 program to determine the worst case operating condition for each MOV. An evaluation was then performed to determine the ability to each motor operator to operate the valve under the worst case operating condition including operation at degraded voltage during worst case temperatures. Motor operator control switch settings were then calculated to ensure proper operation at the worst case operating condition.

MOV Diagnostic Testing under static system conditions was performed on each MOV to set control switches to the specified control switch setting. MOV diagnostic testing was also performed, where possible, under dynamic system conditions which duplicated, as much as possible, actual worst case operating conditions. Field data obtained from those MOVs tested under dynamic conditions were used to validate control switch settings in the initial design calculations. For MOVs where testing under dynamic system conditions was not possible, this validation was performed using other sources of test data (ie. best available industry test data, etc.).

The Palisades plan for long term compliance with GL 89-10 consists of periodic testing, both static and dynamic, of GL 89-10 MOVs to verify control switch settings, and monitoring MOV performance through use of a tracking and trending program.

8.9 LIGHTING SYSTEMS

In addition to the normal ac lighting, there are separate dc and ac emergency lighting systems provided in certain areas of the Plant.

Feeders to lighting panels from power source are carried in the cable tray system. Branch circuits from the lighting panels are carried in conduits. The emergency dc light circuits from the panel to the lights are in conduits dedicated to these circuits only. The lighting panels serving safety equipment areas are at various locations in the auxiliary and turbine buildings.

The feed to the emergency dc Panels D41 and D42 is from the D20 main dc distribution panel in the cable spreading area. The feeds to the emergency dc panels are in separate diverging trays, except at the location of emerging from the Source Panel D20 and are separated from the trays carrying the ac lighting panel feeders. The latter feeders are generally in trays separate from each other.

Fixed battery pack lights are provided for access/egress and at the location of all manual actions required to achieve and maintain hot shutdown per 10 CFR 50, Appendix R. This lighting is in addition to plant normal and emergency ac and dc lighting. The lighting is also available to support fire fighting activities when applicable.

In order to comply with 10 CFR 50, Appendix R, Section III.J, the emergency lighting has been designed to provide 8 hour operation in those areas deemed necessary for operation of safe shutdown equipment in accordance with the analysis of safe shutdown controls described in Section 9.6.8.

Emergency lighting inside containment is provided for personnel safety and to assist in safe handling of fuel during refueling outages. Their operability testing is covered by the Technical Specifications.

8.10 QUALITY CONTROL

For a discussion of the Quality Assurance Program, see Chapter 15. For field quality control, see Subsection 7.8.7.

REFERENCES

1. Crutchfield, Dennis M, Chief, Operating Reactors Branch 5, Division of Licensing, USNRC, to David P Hoffman, Nuclear Licensing Administrator, CP Co, "Environmental Qualification of Electrical Equipment," August 29, 1980

2. Darrell G Eisenhut, Director, Division of Licensing, USNRC, to David P Hoffman, Nuclear Licensing Administrator, CP Co, "Environmental Qualification of Electrical Equipment," September 19, 1980

3. "Environmental Qualification of Safety-Related Electrical Equipment - Palisades Plant - September 1980," October 7, 1980

4. David P Hoffman, Nuclear Licensing Administrator, CP Co, to Dennis M Crutchfield, Chief, Operating Reactors Branch 5, USNRC, "Revision 1 to October 7, 1980 Environmental Qualification of Electrical Equipment," October 30, 1980

5. David P Hoffman, Nuclear Licensing Administrator, CP Co, to Dennis M Crutchfield, Chief, Operating Reactors Branch 5, USNRC, "Revision 2 to October 7, 1980 Environmental Qualification of Electrical Equipment," June 11, 1981

6. David P Hoffman, Nuclear Licensing Administrator, CP Co, to Dennis M Crutchfield, Chief, Operating Reactors Branch 5, USNRC, "Environmental Qualification of Safety-Related Electrical Equipment - December 1981 (complete revision)," December 4, 1981

7. Dennis M Crutchfield, Chief, Operating Reactors Branch 5, USNRC, to David J VandeWalle, Nuclear Licensing Administrator, CP Co, "Safety Evaluation for Environmental Qualification of Safety-Related Electrical Equipment," April 25, 1983

8. Kerry A Toner, Senior Licensing Engineer, CP Co, to Dennis M Crutchfield, Chief, Operating Reactors Branch 5, USNRC, "Environment Qualification of Safety-Related Electrical Equipment - Response to SER Open Items and 10 CFR 50.49 Rulemaking," May 20, 1983

9. Brian D Johnson, Staff Licensing Engineer, CP Co, to Dennis M Crutchfield, Chief, Operating Reactors Branch 5, USNRC, "Environmental Qualification of Safety-Related Electrical Equipment Additional Information," February 14, 1984

10. Thomas C Bordine, Plant Licensing Administrator, CP Co to Nuclear Regulatory Commission, "Additional Information Related to Station Blackout Rule 10 CFR 50.63 (TAC No 68578)," March 27, 1990

11. EA-ELEC-LDTAB-009, "Battery Sizing for the Palisades Class 1E Station Batteries ED-01 & ED-02" (Internal NECO File # PAG-20-24-T)

12. "Ampacity of Cables in Single Open-Top Cable Trays", B L Harshe and W Z Black, IEEE Paper 94 WM 100-8 PWRD

13. Thomas C Bordine, Licensing Manager, Consumers Energy to Nuclear Regulatory Commission, "Response to NRC Request for Additional Information Regarding Cable Ampacity Adjustment Methodology," July 10, 1997

TABLE 8-1

SWITCHYARD SYSTEM
RATINGS AND CONSTRUCTION OF COMPONENTS

Breakers	- 345 kV Nominal
	- 2,000 A Continuous
	- 41,000 A Momentary
	- 25,000 MVA Interrupting
Insulators	- 1,050 kV BIL
Main Bus	- 3,000 A
Bay Bus	- 2,000 A
Disconnect Switches	- 345 kV Nominal
	- 2,000 A Continuous
	- 70,000 A Momentary

TABLE 8-2

4,160 VOLT SYSTEM
RATINGS AND CONSTRUCTION OF COMPONENTS

Bus	- 2,000 A Continuous Rating
Incoming Breakers	- 2,000 A, 350 MVA Interrupting
Feeder Breakers	- 1,200 A, 350 MVA Interrupting
Station Power Transformer 1-1 3-Winding Connected Delta-Wye-Wye	- 21-4.16-4.16 kV, 22.5/25.2 MVA, 55°/65°C
Start-Up Transformers 1-1, 1-3, 4-Winding Connected Wye-Wye-Wye With a Delta Tertiary	- 345-13.2-4.16-4.16 kV, 22.5/25.2 MVA, 55°/65°C
Station Power Transformer 1-3 4-Winding Connected Wye-Wye-Wye With a Delta Tertiary	- 345-2.4-4.16-4.16 kV, 22.5/25.2 MVA, 55°/65°C

TABLE 8-3

2,400 VOLT SYSTEM
RATINGS AND CONSTRUCTION OF COMPONENTS

Safeguard Transformer 1-1 2-Winding, Wye Grounded-Delta Automatic Load Tap Changer	- 354-2.52 kV, 3 Ph, 60 Hz, 10.5 OA/ 13.125 Future FA, 65°/65°C
Station Power Transformer 1-2 2-Winding, Delta-Delta	- 21-2.4 kV, 3 Ph, 60 Hz, 8/9 MVA OA, 55°/65°C
Start-Up Transformer 1-2 2-Winding, Wye Grounded- Delta	- 345-2.4 kV, 3 Ph, 60 Hz, 9.5/10.6 MVA OA, 55°/65°C
Bus	- 2,000 A Continuous Rating
Incoming Breakers	- 1,200 A Continuous, 150 MVA Interrupting
Feeder Breakers	- 1,200 A Continuous, 150 MVA Interrupting
Safeguard Bus	- 3,000 A Continuous Rating
Incoming Breakers	- 3,000 A Continuous, 350 MVA Interrupting

TABLE 8-4

480 VOLT SYSTEM
RATINGS AND CONSTRUCTION OF COMPONENTS
(Sheet 1 of 4)

480 V Load Centers 11, 12, 13, 14, 15, 16	
Transformers	- 750 kVA AA, 3 Ph, 60 Hz, 2,400-480 V
Bus	- 1,000 A Continuous, 50,000 A RMS Symmetrical (Load Centers 11, 12, 13,and 14) - 1,200 A Continuous, 25,000 A Asymmetrical (Load Centers 15 and 16)
Load Center Feeder Breakers - Interrupting Current	- 22,000 A RMS Symmetrical (Load Centers 11, 12, 13 and 14) - 18,000 A RMS Symmetrical (Load Centers 15 and 16)
480 V Motor Control Centers 1, 2, 3, 4, 5, 6, 7, 8, 9, 10	
Horizontal Bus	- 600 A Continuous, 25,000 A RMS Symmetrical
Vertical Bus	- 300 A Continuous, 25,000 A RMS Symmetrical
Molded Case Breakers - Interrupting Current	- 14,000 A RMS Symmetrical
480 V Load Center 17	
Transformer	- 2,000 kVA AA, 3 Ph, 60 Hz, 4,160-480/277 V
Bus	- 3,200 A Continuous, 65,000 A RMS Symmetrical
Load Center Feeder Breakers - Interrupting Current	- 50,000 A RMS Symmetrical

TABLE 8-4
(Sheet 2 of 4)

480 V Load Centers 19, 20	
Transformer	- 750/1,000 kVA AA/FA, 3 Ph, 60 Hz, 2,400-480 V
Bus	- Equal to Capacity of Largest Breaker
Load Center Feeder Breakers - Interrupting Current	- 14,000 A RMS Symmetrical (MCC 21 and 23) 22,000 A RMS Symmetrical (MCC 22 and 24)
480 V Motor Control Centers 21, 22, 23, 24	
Horizontal Bus	- 600 A Continuous, 42,000 A RMS Symmetrical
Vertical Bus	- 300 A Continuous, 42,000 A RMS Symmetrical
Molded Case Breakers - Interrupting Current	- 10,000 A RMS Symmetrical
480 V Motor Control Centers 25, 26	
Horizontal Bus	- 600 A Continuous, 42,000 A RMS Symmetrical
Vertical Bus	- 600 A Continuous, 42,000 A RMS Symmetrical
Molded Case Breakers - Interrupting Current	- 25,000 A RMS Symmetrical
480 V Load Centers 77, 78	
Transformers	- 500 kVA, 3 Ph, 60 Hz, 2,400-480 V
Bus	- Equal to Capacity of Largest Circuit Breaker
Load Center Feeder Breakers - Interrupting Current	- 22,000 A RMS Symmetrical

TABLE 8-4
(Sheet 3 of 4)

480 V Motor Control Centers 79, 80, 81, 82	
Horizontal Bus	- 600 A Continuous, 25,000 A RMS Symmetrical
Vertical Bus	- 300 A Continuous, 25,000 A RMS Symmetrical
Molded Case Breakers - Interrupting Current	- 14,000 A RMS Symmetrical
480 V Load Centers 71, 72, 73, 74, 75, 76 (Cooling Towers)	
Transformers	- 1,500 kVA, 3 Ph, 60 Hz, 4,160-480 V
Bus	- 2,000 A Continuous, 42,000 A RMS Symmetrical
Load Center Feeder Breakers - Interrupting Current	- 42,000 A RMS Symmetrical
480 V Load Centers 90, 91, 200	
Transformers	- 750 kVA, 3 Ph, 60 Hz, AA, 2,400-480 V
Bus	- 1,200 A Continuous, 50,000 A RMS Symmetrical (Load Center 200) - 1600 A Continuous (Load Centers 90 and 91)
Load Center Feeder Breakers - Interrupting Current	- 22,000 A RMS Symmetrical Momentary (Load Centers 90 and 91) - 30,00 A RMS Symmetrical Interrupting (Load Centers 90 and 91) - 50,000 A RMS Symmetrical (Load Center 200)

480 V Motor Control Centers 92, 93, 94, 95, 96, 97, 99	
Horizontal Bus	- 600 A Continuous, 42,000 A Symmetrical
Vertical Bus	- 300 A Continuous, 42,000 A Symmetrical
Molded Case Breakers - Interrupting Current	- 14,000 A RMS Symmetrical
	- 22,000 A RMS Symmetrical (Motor Control Center 99)

TABLE 8-5

DC AND PREFERRED AC SYSTEMS
RATINGS AND CONSTRUCTION OF COMPONENTS

DC Switchgear	
Main Bus	- 600 A Continuous, 10,000 A Momentary
Auxiliary Bus	- 300 A Continuous, 10,000 A Momentary
Breakers - Minimum Rating	- 10,000 A Interrupting at 120 V DC
Chargers	- Solid State, 200 A Continuous Output
DC Panel Breakers	- 10,000 A Interrupting at 125 V DC
Inverter	- 6 kVA Continuous Output
AC Panel Breakers	- 5,000 A Symmetrical at 120 V AC

TABLE 8-6

DIESEL NO 1-1 SEQUENCE START

Time(c) (Seconds)	Service	Nameplate hp
10	Miscellaneous 480 V Loads(a)	333
12	Boric Acid Pump P56B Charging Pump P55C Containment Cooler Recirc Fan V4A Containment Spray Pump P54B	30 75 75 250
16	HP Safety Injection Pump P66B	400
20	Service Water Pump P7B	350
23	LP Safety Injection Pump P67B	400
29(d)	Containment Spray Pump P54C	250
33	Component Cooling Pump P52A	300
50	Component Cooling Pump P52C (If Pumps P52A and P52B fail to start)	
55(b)	Auxiliary Feedwater Pump P8A	450
65	Control Room Air Handling Unit Fan V95	25

(a) Per EA-ELEC-LDTAB-005 Rev. 1, Table A.1 @ Time Zero.

(b) Sequencer set point, does not include five-second delay due to Auxiliary Feedwater Initiation Control Logic (Section 7.4).

(c) These times include 10 seconds for diesel start and generator breaker closing.

(d) Sequencer set point, does not include nominal fifteen-second delay of pump automatic start following receipt of Containment High Pressure Signal (Section 6.2.2.3).

TABLE 8-7

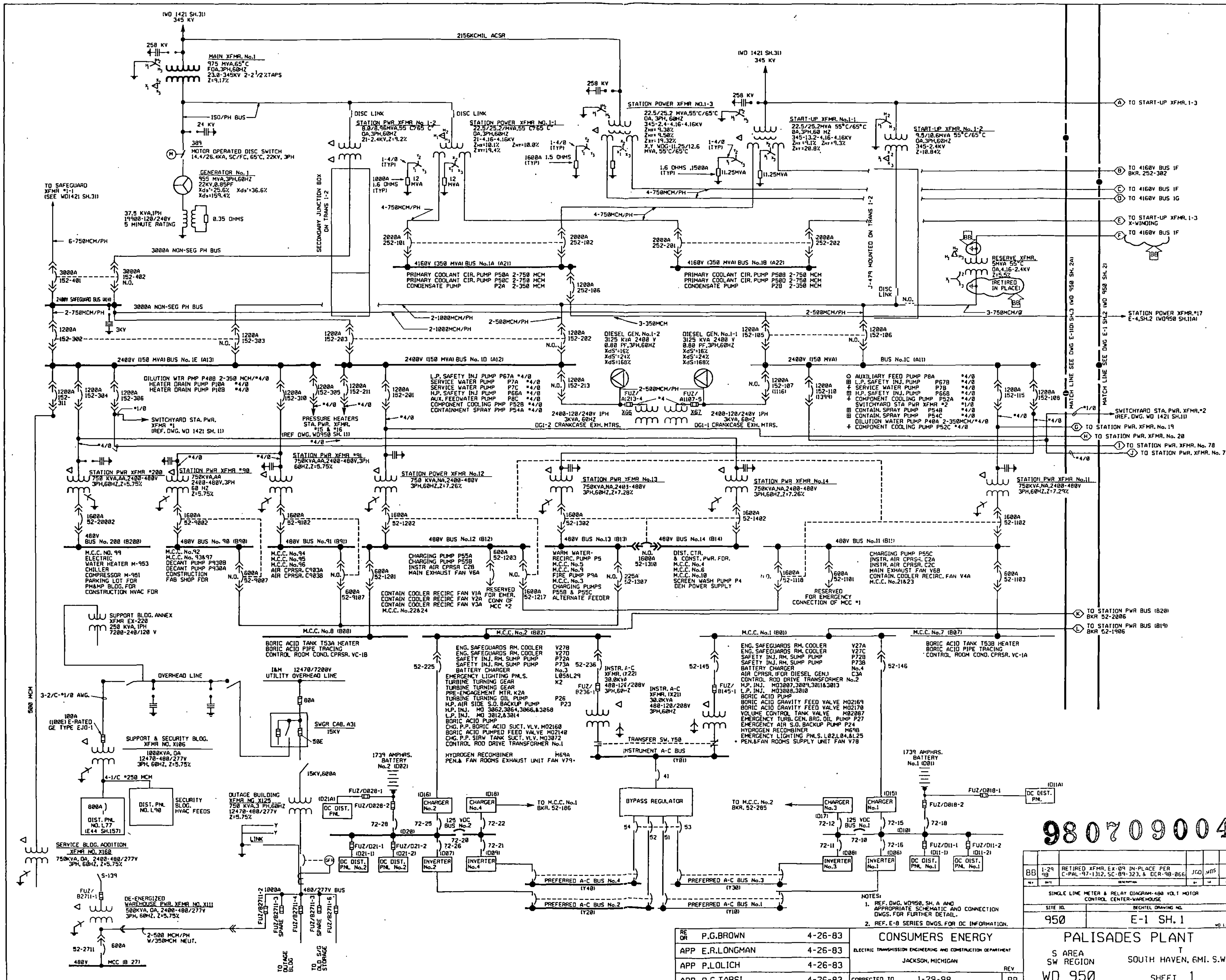
DIESEL NO 1-2 SEQUENCE START

Time(c) (Seconds)	Service	Nameplate hp
10	Miscellaneous 480 V Loads(a)	350
12	Boric Acid Pump P56A	30
	Charging Pump P55A	100
	Containment Cooler Recirc Fan V3A	75
	Containment Spray Pump P54A	250
16	HP Safety Injection Pump P66A	400
20	Service Water Pump P7A	350
23	LP Safety Injection Pump P67A	400
29	Containment Cooler Recirc Fans V1A and V2A	150
	Charging Pump P55B (If Pumps P55A and P55C fail to start)	75
33	Component Cooling Pump P52B	300
36	Service Water Pump P7C	350
55(b)	Auxiliary Feedwater Pump P8C (If Pump P8A fails to provide sufficient flow)	400
65	Control Room Air Handling Unit Fan V96	25

(a) Per EA-ELEC-LDTAB-005 Rev. 1, Table C.1 @ Time Zero.
(b) Sequencer set point, does not include delays due to Auxiliary Feedwater Initiation Control Logic (Section 7.4).
(c) These times include 10 seconds for diesel start and generator breaker closing.

Table 8-8

DELETED



ANSTEC APERTURE CARD

Also Available on Aperture Card

9807090043-20

FIG 8-1 SH 1
FSAR REV 20

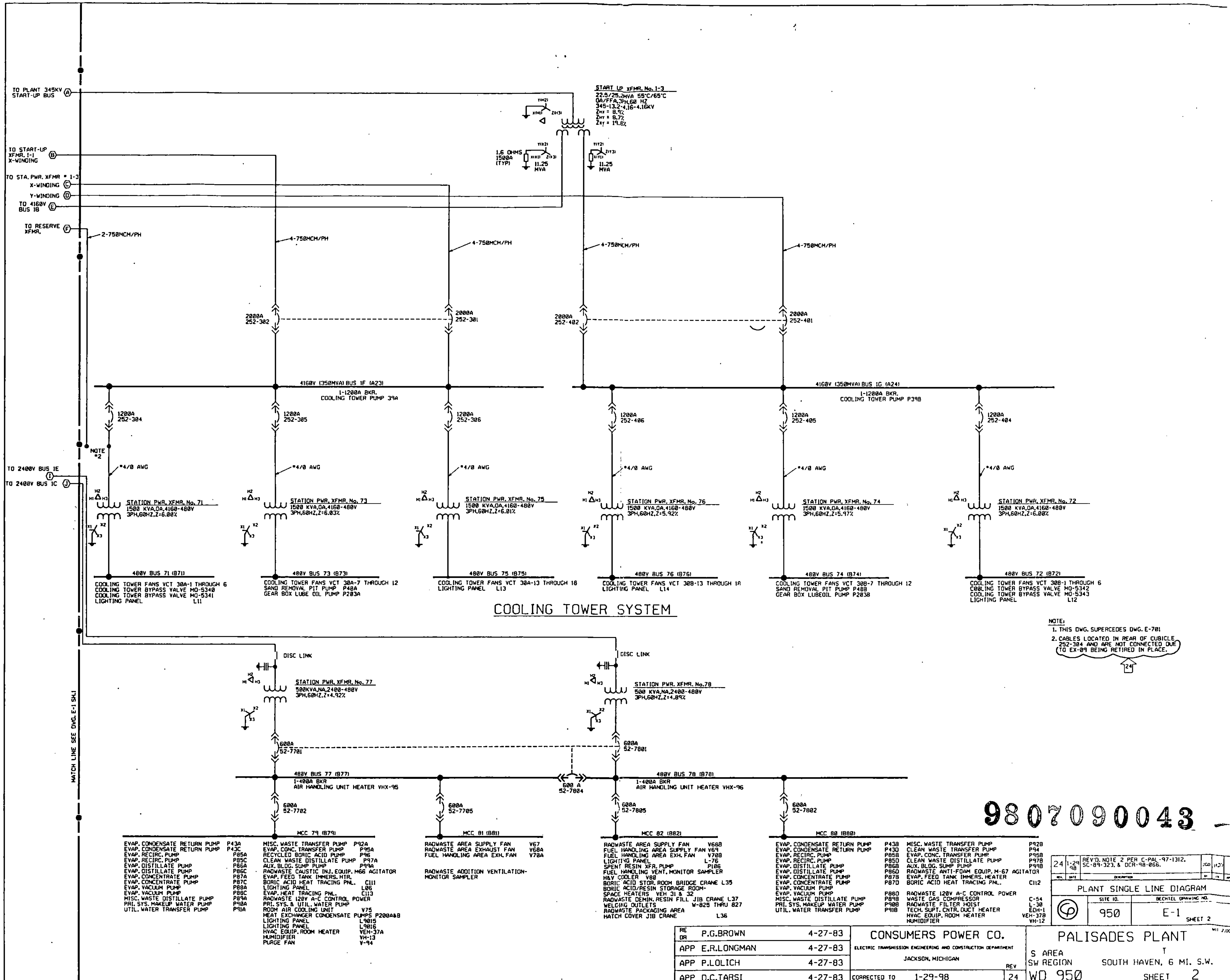
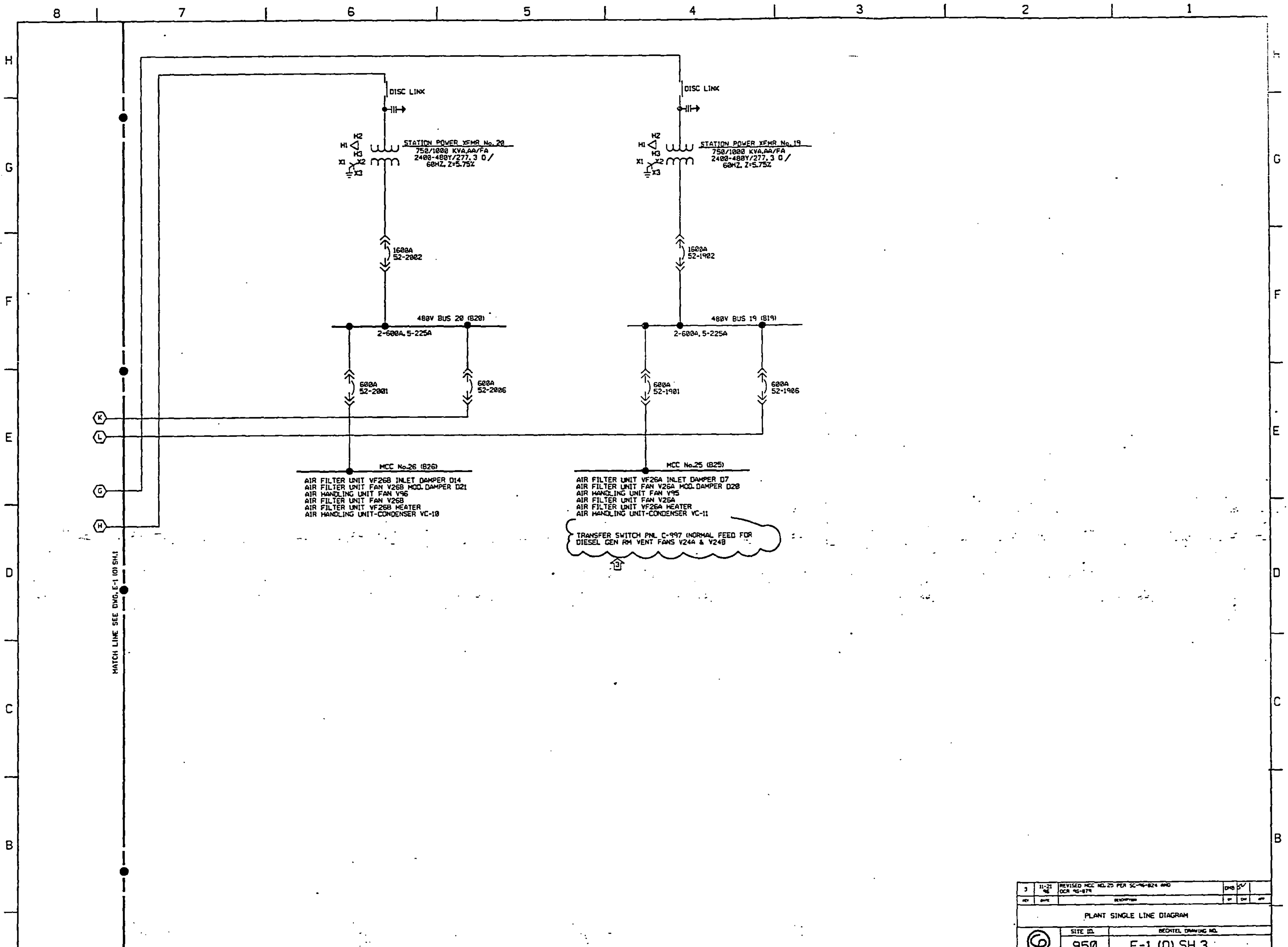


FIG 8-1 SH 2
FSAR REV 20



DR	P.G. BROWN	1-18-84	CONSUMERS POWER CO.
APP	E.R. LONGMAN	1-18-84	JACKSON, MICHIGAN
APP	P. LOLICH	1-18-84	
APP	D.C. TARSI	1-18-84	CORRECTED TO 11-21-96 REV 3

3 | 11-21-96 | REVISED MCC NO. 25 PER SC-96-024 AND DCR 96-079 | DMB

PLANT SINGLE LINE DIAGRAM

SITE ID. 950 — BECHTEL DRAWING NO. E-1 (Q) SH.3

PALISADES PLANT

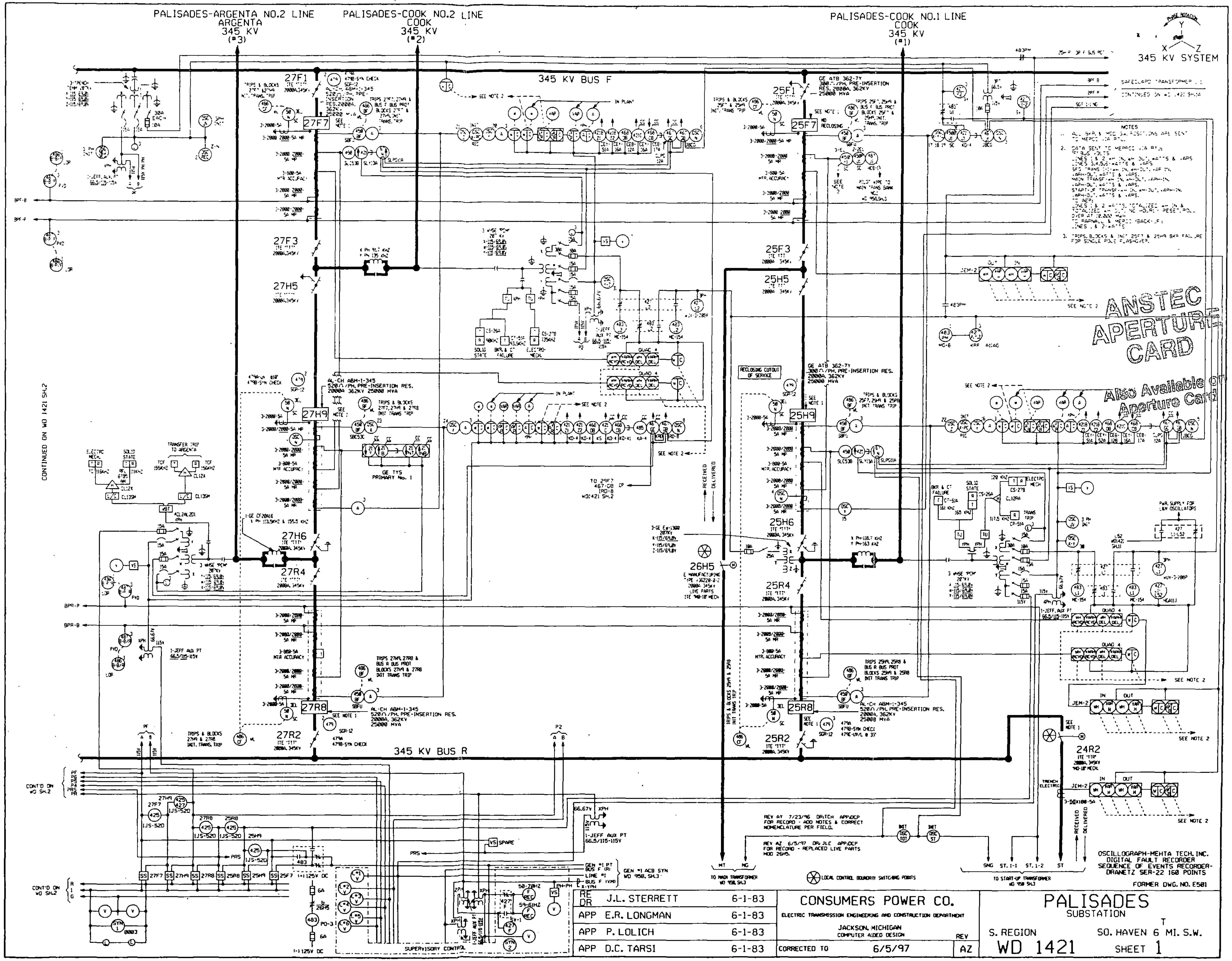
S. AREA
SW. REGION
SO. HAVEN, 6 MI. SW

WD 950 SHEET 2A

PB3012684

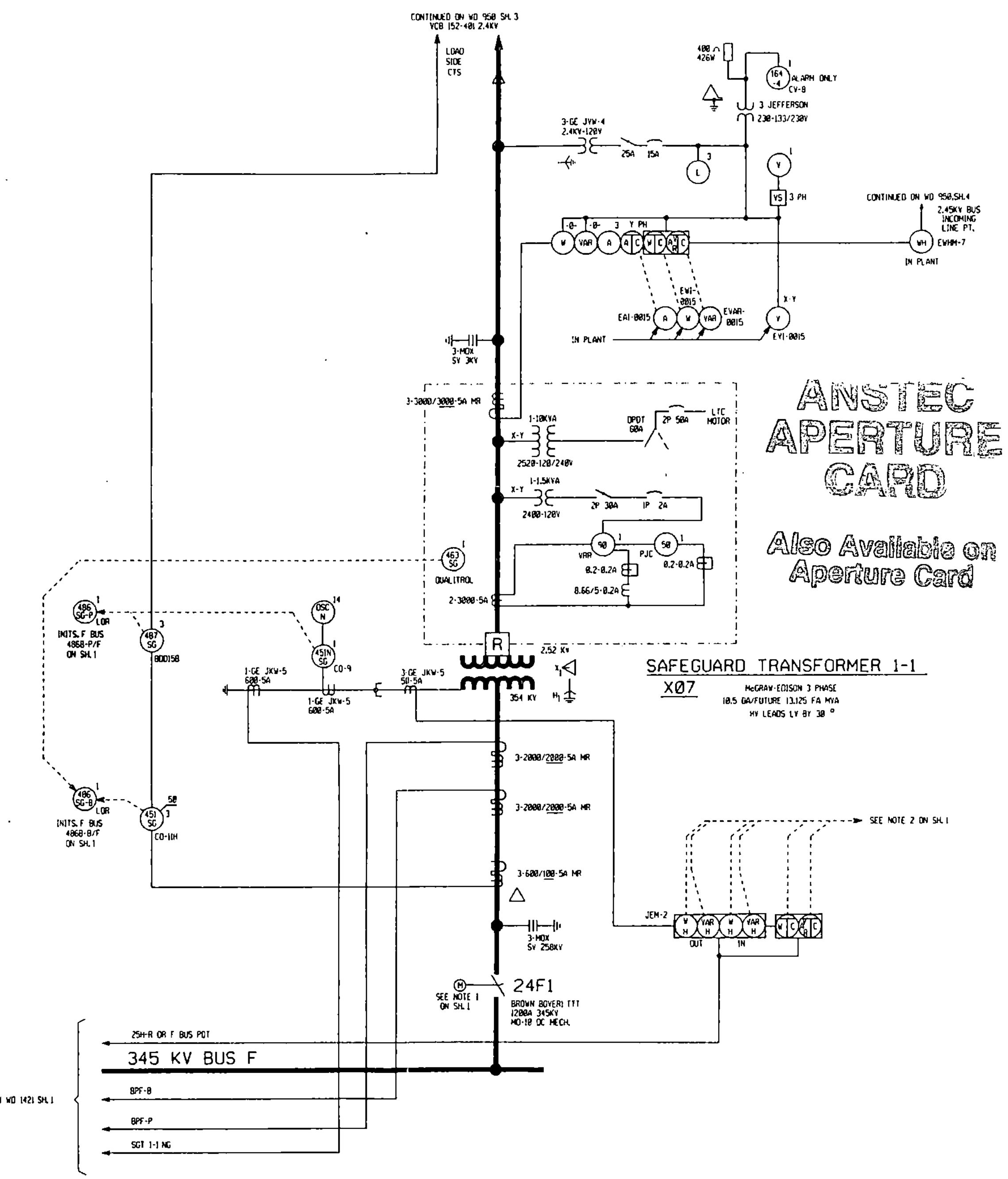
9707070093-02

Figure 8-1 Sh 3
Rev 19



9807090043 -22

FIG 8-2 SH 1
FSAR REV 20



REV.G 4-12-95 DR:NM APP:DCP FOR RECORD - REVISED PER FIELD AS BUILT.

REV.H 6-5-97 DR:JLC APP:DCP FOR RECORD - ADD ID NUMBERS TO TRANSF METERING.

9807090043 -23

DR	D.T. BINSACK/CDR	12-01-88	CONSUMERS POWER CO.	PALISADES SUBSTATION
APP	J.K. MASON	12-07-88	ELECTRIC TRANSMISSION ENGINEERING AND CONSTRUCTION DEPARTMENT	
APP	T.L. KRUMMREY	12-07-88	JACKSON, MICHIGAN (PC)WD1421.01A REV	S. REGION SO. HAVEN 6 MI. S.W.
APP	D.C. TARSI	12-07-88	CORRECTED TO 6-5-97 H	WD 1421 SHEET 1A

FIG 8-2 SH 1A
FSAR REV 20

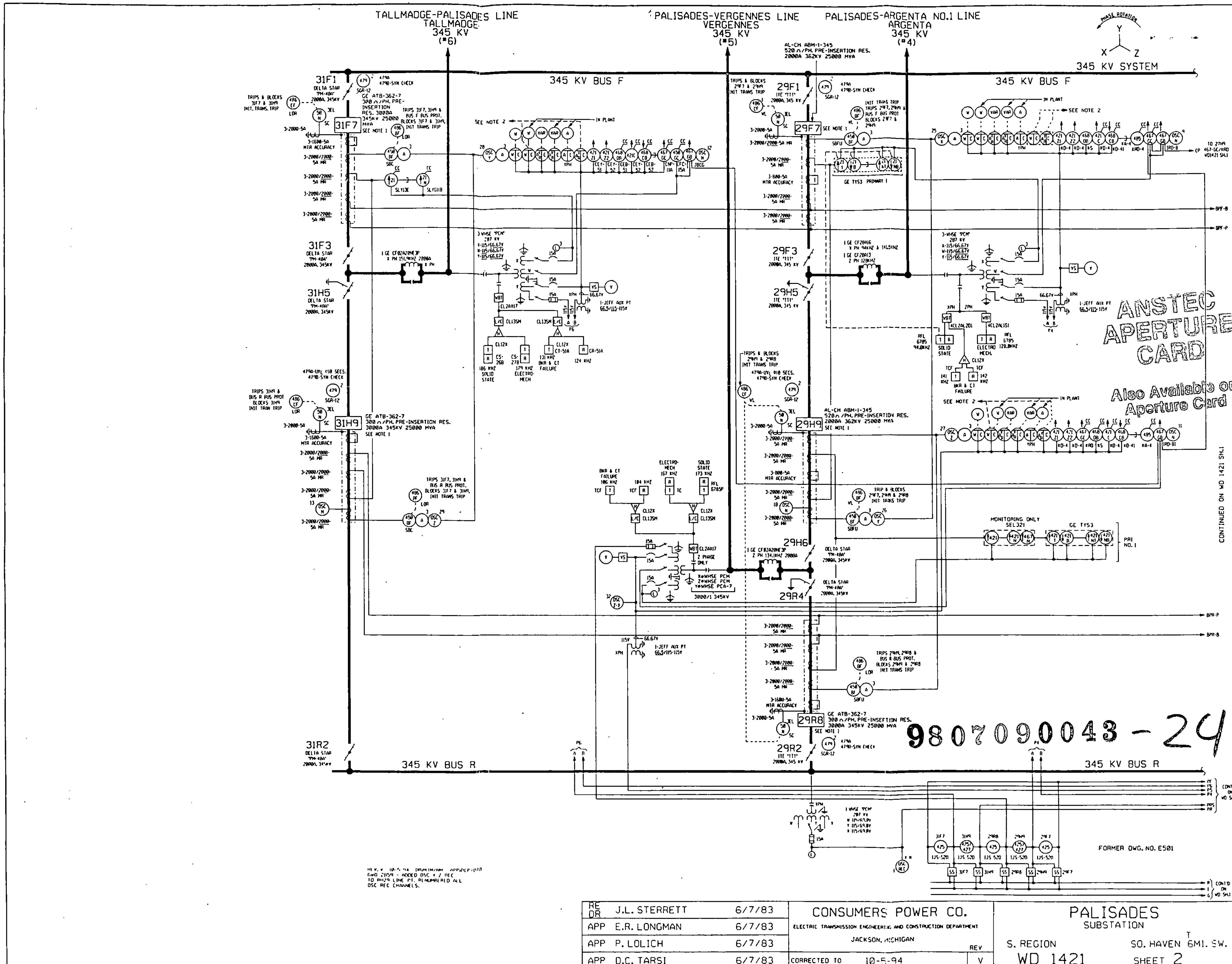


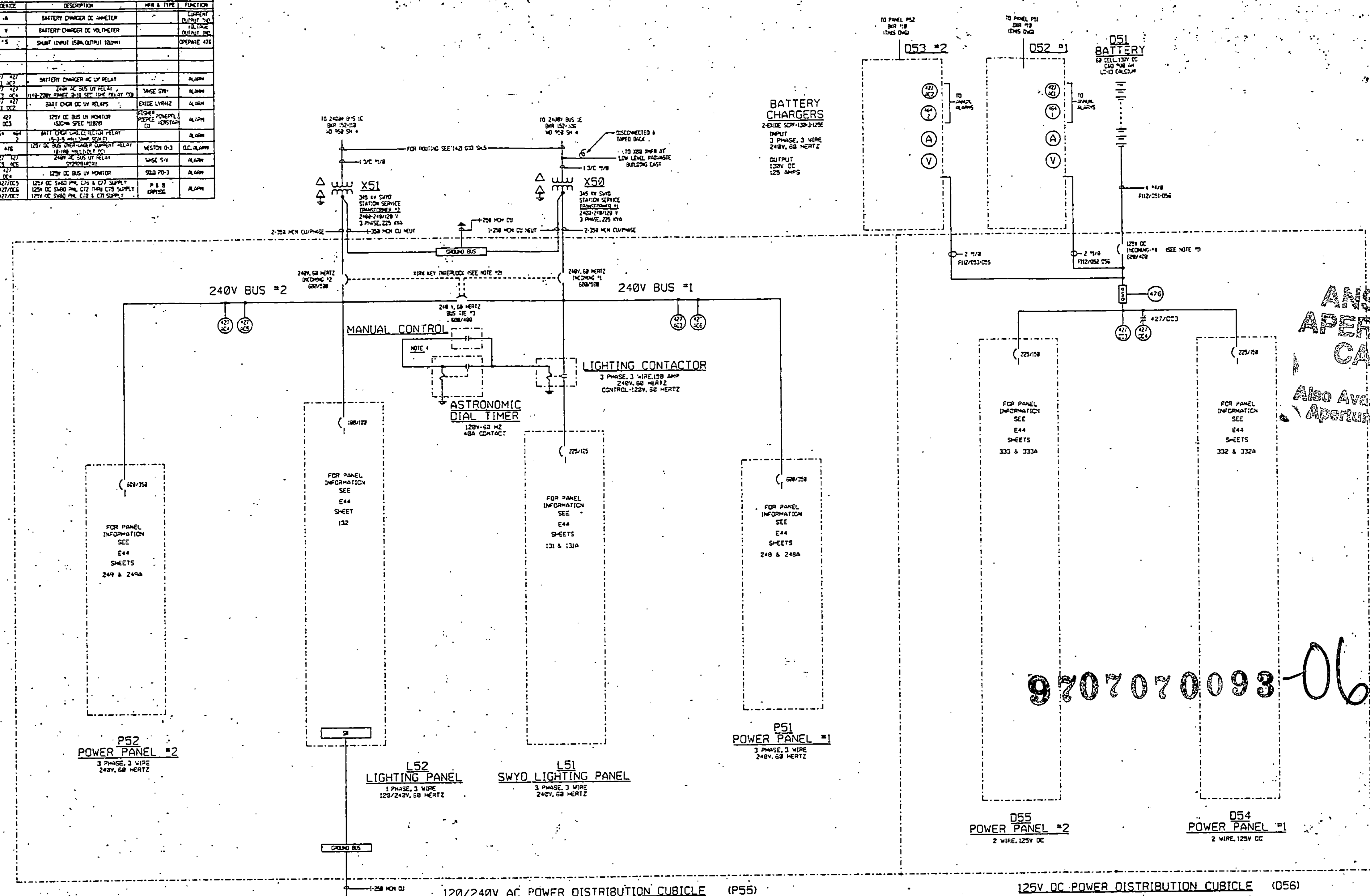
FIG 8-2 SH 2
FSAR REV 20

FIGURE 8-3

Deleted

(Has been renumbered as Figure 8-2 Sheet 2)

METER & RELAY TABLE			
DEVICE	DESCRIPTION	MFR & TYPE	FUNCTION
A	BATTERY CHARGER DC AMMETER		CURRENT OUTPUT IND.
V	BATTERY CHARGER DC VOLTMETER		VOLTAGE OUTPUT IND.
S	SHUNT INPUT 150A, OUTPUT 100MV		OPERATE 476
427 427 AC1 AC2	BATTERY CHARGER AC UV RELAY		ALARM
427 427 AC3 AC4	240V AC BUS UV RELAY (140-220V RANGE 2-10 SEC TIME DELAY TD)	WHSE SYV	ALARM
427 427 DC1 DC2	BATT CHGR DC UV RELAYS	EXIDE LVR412	ALARM
427 DC3	125V DC BUS UV MONITOR (SIGMA SPEC 1182B)	FISHER PIERCE CO / POWERPL. MONSTAR	ALARM
464 464 1 2	BATT CHGR GND. DETECTOR RELAY (5-0-5 MILLIAMP. SCALE)		ALARM
476	125V DC BUS OVER-UNDER CURRENT RELAY (0-100 MILLIVOLT DC)	WESTON 0-3	D.C. ALARM
427 427 AC5 AC6	240V AC BUS UV RELAY SV292944CA11	WHSE S-V	ALARM
427 DC4	125V DC BUS UV MONITOR	SOLO PO-3	ALARM
427/DC5 427/DC6 427/DC7	125V DC SWBD PNL C76 & C77 SUPPLY / 125V DC SWBD PNL C72 THRU C75 SUPPLY / 125V DC SWBD PNL C78 & C71 SUPPLY	P & B KRP100	ALARM



9707070093-06

FORMER DWG. NO. E-502

120/240V AC & 125V DC POWER DISTRIBUTION CENTER

NOTES

1. THE 125V DC SUPPLY CIRCUIT BREAKER HAS AN ADJUSTABLE MAGNETIC TRIP ONLY.
2. THE TWO 240V, 60 HZ SUPPLY AND ONE BUS TIE CIRCUIT BREAKERS ARE KIRK KEY INTERLOCKED SO THAT NO MORE THAN TWO OF THE BREAKERS CAN BE CLOSED AT ANY TIME.
3. REFER TO DWG. E-159, SH. 3 FOR SCHEMATIC DIAGRAM.
4. REFER TO DWG. E44 SHT. 132.

REVISED JUN - 3 1997 FOR RECORD

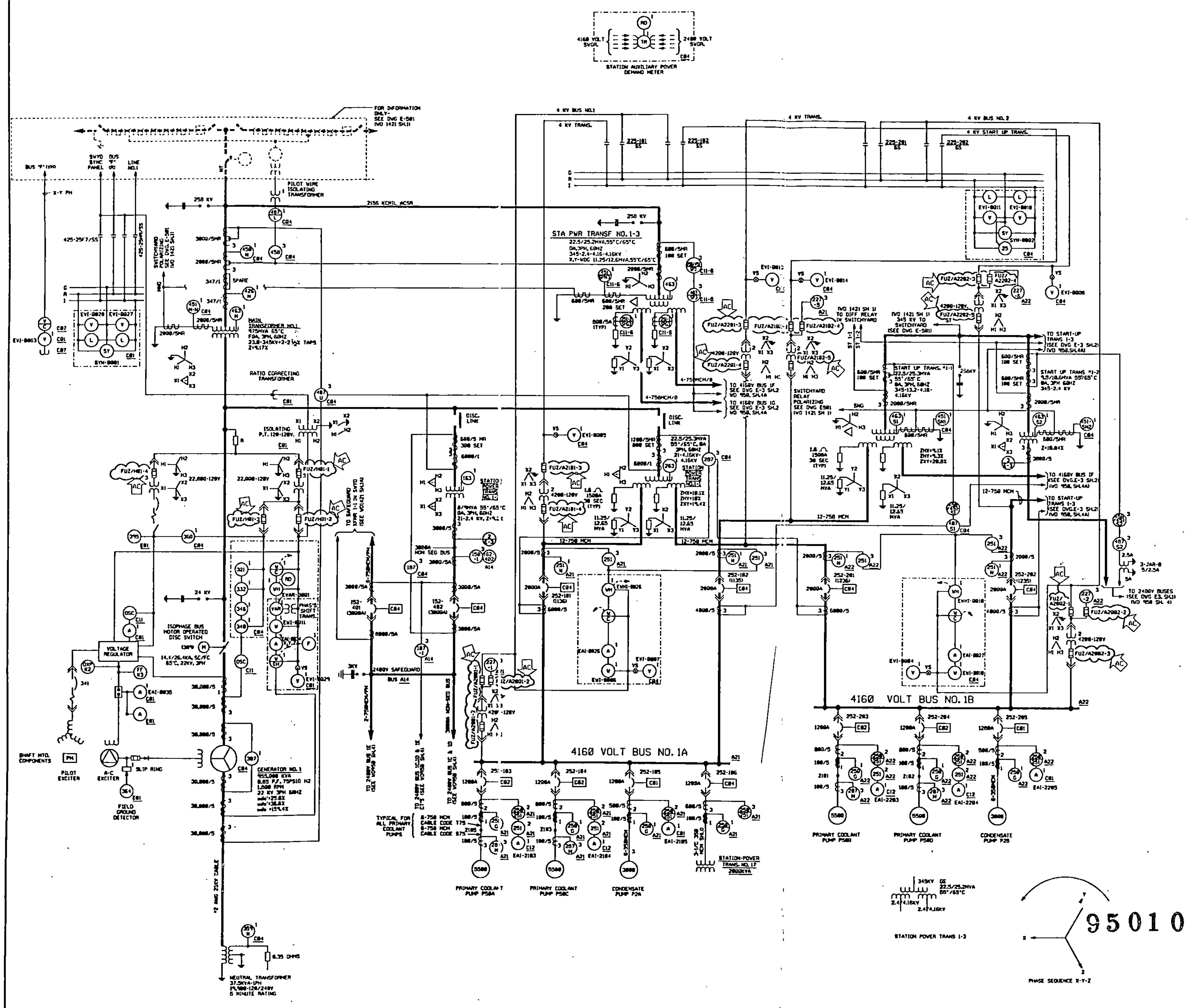
DR	J.L. STERRETT	6/16/83
APP	E.R. LONGMAN	6/16/83
APP	P. LOLICH	6/16/83
APP	D.C. TARSI	6/16/83

CONSUMERS POWER CO.
ELECTRIC TRANSMISSION ENGINEERING AND CONSTRUCTION DEPARTMENT
JACKSON, MICHIGAN
COMPUTER AIDED DESIGN

CORRECTED TO 06-03-97 REV W

PALISADES SUBSTATION
S. REGION
SO. HAVEN 6 MI. SW.
WD 1421 SHEET 11

Figure 8.4
Rev



METER AND RELAY TABLE

DEVICE	DESCRIPTION	FUNCTION
187	STA. POWER TRANS. *1-2 DIFFERENTIAL RELAY	TRIP GEN. LOCKOUT RELAY
163	STA. POWER TRANS. *1-2 SUDDEN PRESS RELAY	TRIP GEN. LOCKOUT RELAY
51 & 51N	STA PWR TRANSFORMER *17 PHASE & NEUTRAL OVERCURRENT RELAYS	TRIP ASSO. TRANS. CKT. BKR 252-106
227-5 & 6	4160V STANDBY SOURCE UNDERVOLTAGE RELAY	ALARM & BUS TRANSFER
227-1 & 2	4160V BUS UNDERVOLTAGE RELAY	LOAD SHEDDING
250/251	4160V MOTOR OVERCURRENT RELAY	TRIP ASSOCIATED MOTOR CKT. BKR.
251	4160V PHASE OVERCURRENT RELAY	TRIP ASSOCIATED FDR. CKT. BKR.
251N	4160V GROUND OVERCURRENT RELAY	TRIP ASSOCIATED FDR. CKT. BKR.
263	STA. POWER TRANS. *1-1 SUDDEN PRESS. RELAY	TRIP GEN. LOCKOUT RELAY
287	STA. POWER TRANS. *1-1 DIFFERENTIAL RELAY	TRIP GEN. LOCKOUT RELAY
287M	4160V MOTOR DIFF. RELAY	TRIP ASSOCIATED MOTOR CKT. BKR.
321	GENERATOR DISTANCE BACK-UP RELAY	TRIP GEN. INDIRECT TRIP LOCKOUT RELAY
332	GENERATOR ANTI-MOTORING RELAY	TRIP GEN. INDIRECT TRIP LOCKOUT RELAY
340	GENERATOR LOSS OF FIELD RELAY	TRIP GEN. LOCKOUT RELAY
346	GEN. NEGATIVE PHASE SEQUENCE RELAY	TRIP GEN. INDIRECT TRIP LOCKOUT RELAY
250G	4160V FDR. GROUND OVERCURRENT RELAY	TRIP ASSOCIATED MOTOR CKT. BKR.
395	GEN VOLTS/HERTZ RELAY	TRIP REGULATOR
359N	GEN. NEUTRAL OVERVOLTAGE RELAY	TRIP GEN. LOCKOUT RELAY
360	VOLTAGE BALANCE RELAY	TRIP VOLTAGE REGULATOR
359E	GEN. FIELD OVERVOLTAGE RELAY	ALARM
387	GENERATOR DIFFERENTIAL RELAY	TRIP GEN. LOCKOUT RELAY
376	GEN. FIELD OVERCURRENT RELAY	OPERATES BASE ADJUSTER
364	GEN. FIELD GROUND RELAY	ALARM
426M	MAIN TRANSFORMER TEMP. SW.	PUMP & FAN CONTORL
463 S1 & S2	START UP TRANS. SUDDEN PRESS. RELAY	TRIPS START UP TRANS. LOCKOUT RELAY
463M	MAIN TRANS. SUDDEN PRESS. RELAY	TRIP GEN. LOCKOUT RELAY
451HN	MAIN TRANS. GROUND O.C. RELAY	TRIP GEN. LOCKOUT RELAY
451 SN1 & SN2	STARTUP TRANS. GROUND O.C. RELAY	TRIPS START UP TRANS. LOCKOUT RELAY
487S1	STARTUP TRANS. *1-1 DIFF. RELAY	TRIPS START UP TRANS. LOCKOUT RELAY
487S2	STARTUP TRANS. *1-2 DIFF. RELAY	TRIPS START UP TRANS. LOCKOUT RELAY
487U	UNIT DIFFERENTIAL RELAY	TRIP GEN. LOCKOUT RELAY
487L	345KV LINE PILOT WIRE DIFF.	TRIP GEN. LOCKOUT RELAY
450	345KV LINE OVERCURRENT RELAY	INTERLOCKS 487L
450N	345KV LINE NEUTRAL O.C. RELAY	INTERLOCKS 487L
25	SYNCHRO-CHECK RELAY	
A	AMMETER	
F	FREQUENCY METER	
L	LAMP	
VS	VOLTMETER SWITCH	
SY	SYNCHROSCOPE	
V	VOLTMETER	
VAR	VARMETER	
SS	SYNCHRONIZING SWITCH	
W	WATTMETER	
W/C	WATT TRANSDUCER (GEN. NET WATTS)	
WH	WATTHOUR METER (WITH CONTACT FOR TOTALIZING)	
RD	PRINTING DEMAND METER	
TR	TOTALIZING RELAY	
OSC	OSCILLOGRAPH	
W/EH	WATT TRANSDUCER (EH SYSTEM)	
450/451 S1 & S2	START UP TRANSFORMER O.C. RELAY	TRIPS START UP TRANS LOCKOUT RELAY
V/C	VOLT TRANSDUCER	
450/451 SP3	STA. PWR. XFMR. *1-3 PHASE OVERCURRENT RELAYS	TRIP PRIMARY LOCKOUT (386P)
463	STA. PWR. XFMR. *1-3 SUDDEN PRESSURE RELAYS	TRIP PRIMARY LOCKOUT (386P)
487 SP3	STA. PWR. XFMR. *1-3 CURRENT DIFFERENTIAL RELAYS	TRIP BACK-UP LOCKOUT (386B)
451 SPN3	STA. PWR. XFMR. *1-3 HIGH-SIDE NEUT. GRD. RELAY	TRIP BACK-UP LOCKOUT (386B)
251-L-2 SPG3	STA. PWR. XFMR. *1-3 LOW-SIDE NEUT. GRD. RELAYS	TRIP BACK-UP LOCKOUT (386B)
187-1	SAFEGUARD FEEDER CURRENT DIFFERENTIAL RELAY	TRIP 401 & 402 BKR INIT 1C, 1D & 1E BUS LOAD TRANS TO ALL XFMR 1-2, INITIATES DATALOGGER
150-1	2400V SAFEGUARD BUS BKR 152-402 BREAKER FAILURE OVERCURRENT RELAY	WITH 187-1 OPERATION & BKR 152-402 OVERCURRENT INITIATES TIME DELAY RELAY 162-402
162-402	2400V SAFEGUARD BUS BKR 152-402 BREAKER FAILURE TIME DELAY RELAY	TRIPS PRIMARY LOCKOUT (386P) AFTER 0.5 SECONDS INITIATES DATALOGGER

NOTES

1/E-3 THE NUMBER ABOVE THE LINE IS THE MATCH CIRCLE NUMBER, AND THE NUMBER BELOW THE LINE INDICATES THE DRAWING ON WHICH THE CIRCUIT CONTINUES. NO NUMBER BELOW THE LINE INDICATES THAT THE CIRCUIT IS CONTINUED ON THE SAME DRAWING.

9501040188-40

NO	DATE	REVISION	BY	APP
AC	10/28/93	ADDED FUSE *'S PER DCR 93-1298	ITT	

SINGLE LINE METER & RELAY DIAGRAM
GENERATOR & 4160V SYSTEM

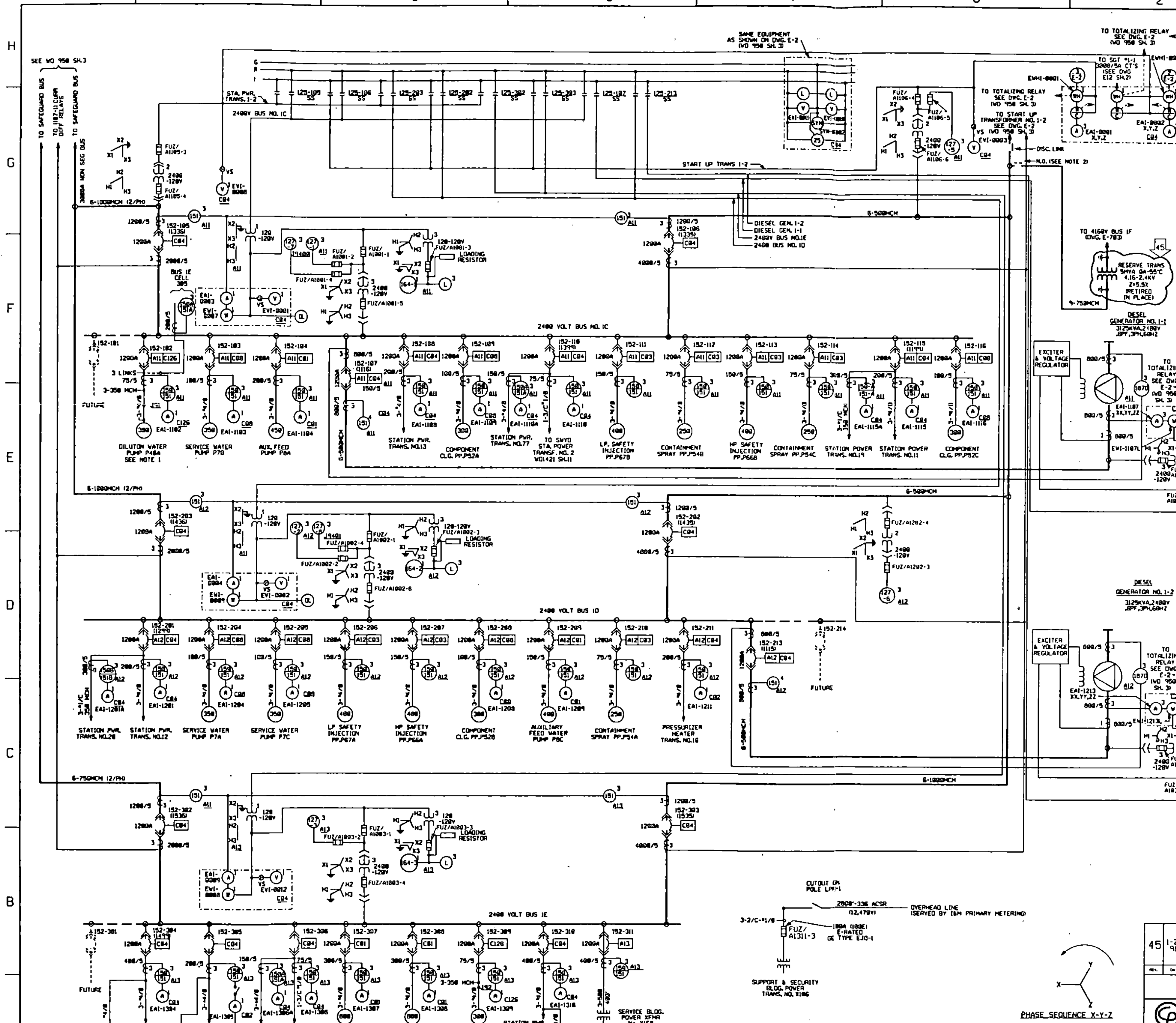
SITE ID. 0950 — BECHTEL DRAWING NO. E-2(

RE DR	E.E. MCCANN	8/30/83
APP	E.R. LONGMAN	8/30/83
APP	P. LOLICH	8/30/83
APP	D.C. TARSI	8/30/83

CONSUMERS POWER CO.
PLANT MODIFICATIONS AND MISCELLANEOUS PROJECTS DEPARTMENT
JACKSON, MICHIGAN
WO-3.DGN
CORRECTED TO 10-28-93 REV AC

PALISADES PLANT
S SW AREA REGION T SOUTH HAVEN, 6 MI. S.W.
WD 950

Figure 8-5 FSAR Rev 17



METER AND RELAY TABLE

DEVICE	DESCRIPTION	FUNCTION
127-5&6	2400V START UP SOURCE UNDERVOLTAGE RELAY	ALARM & BUS TRANSFER
127-1&2	2400V BUS U.V. RELAY (INVERSE TIME DELAY)	ALARM, LOADSHED, TRIP INC. BKRS & D.G. START
127D-(1,2,3) 127D-(21,22,23)	DIESEL GENERATOR UNDERVOLTAGE RELAY	GEN INCOMING BKRS AUTO-CLOSE
150/151	2400V MOTOR OVERCURRENT RELAY	TRIP ASSOCIATED FOR CIRCUIT BKR.
127-3	2400V BUS UNDERVOLTAGE RELAY	ALARM
151	2400V PHASE OVERCURRENT RELAY (TIME DELAY)	TRIP ASSOCIATED MOTOR CKT. BKR.
127-4	UNIT AUX. TRANS 1-2 U.V. RELAY	NOT USED
164-1,2&3	2400V BUS GROUND RELAY	ALARM
127-7&8	2400V BUS U.V. RELAY (DEFINITE TIME DELAY)	ALARM, LOADSHED, TRIP INC. BKRS & D.G. START
187D	DIESEL GENERATOR DIFFERENTIAL RELAY	TRIP DIESEL GEN.
DL	FWP DATALOGGER	BUS 1C & 1D VOLTS
150-A/151-A	2400 V MOTOR OVERCURRENT RELAY	TRIP ASSOC. FOR BKR LOAD CKT BUS OVER CUR.
150-B/151-B	2400 V MOTOR OVERCURRENT RELAY	TRIP ASSOC. FOR BKR LOAD CKT BUS OVER CUR.
25	SYNCHRO-CHECK RELAY	
A	AMMETER	
F	FREQUENCY METER	
L	LAMP	
VS	VOLTMETER SWITCH	
SYN	SYNCHROSCOPE	
V	VOLTMETER	
VAR	VARMETER	
SS	SYNCHRONIZING SWITCH	
W	WATTMETER	
W/C	WATT TRANSDUCER (GEN NET WATTS)	
WH	WATTHOUR METER (WITH CONTACT FOR TOTALIZING)	

NOTE

1. DILUTION WATER PUMPS P40A & P40B REPLACE CIRCULATING WATER PUMPS P3A & P3B RESPECTIVELY.
2. IN CASE RESERVE TRANSFORMER IS IN SERVICE, WITHDRAW DIFFERENTIAL RELAY 487S2.
3. THIS BREAKER MAY BEE USED FOR TEMPORARY POWER PER SPS-E-12. 45

QMO 7114 FEEDWATER PURITY MODIFICATIONS

QMO 8530 UNDERVOLTAGE MODIFICATION

PHASE SEQUENCE X-Y-Z

REV.	DATE	DESCRIPTION	BY	CK	APP
45	1-29-98	SPARED XFMR. EX-09 IN PLACE PER C-PAL-97-1312, SC-89-323, & DCR-98-066. ADDED NOTE 3 PER DCR-98-080.	JGD	MJT	

SINGLE LINE METER & RELAY DIAGRAM
2400 VOLT SYSTEM

WD-4.DGN

SITE ID. 950 — BECHTEL DRAWING NO. E-3 SH. 1

DR J. L. STERRETT	4 /27/83
APP E. R. LONGMAN	4 /27/83
APP P. LOLICH	4 /27/83
APP D. C. TARSI	4 /27/83

CONSUMERS POWER CO.
PLANT MODIFICATIONS AND MISCELLANEOUS PROJECTS DEPARTMENT
JACKSON, MICHIGAN

CORRECTED TO 1-2998 — REV 45

PALISADES PLANT
S SW AREA REGION T SOUTH HAVEN, 6 MI. S.W.
WD 950 SHEET 4

FIG 8-6
FSAR REV 20

9807090043-25

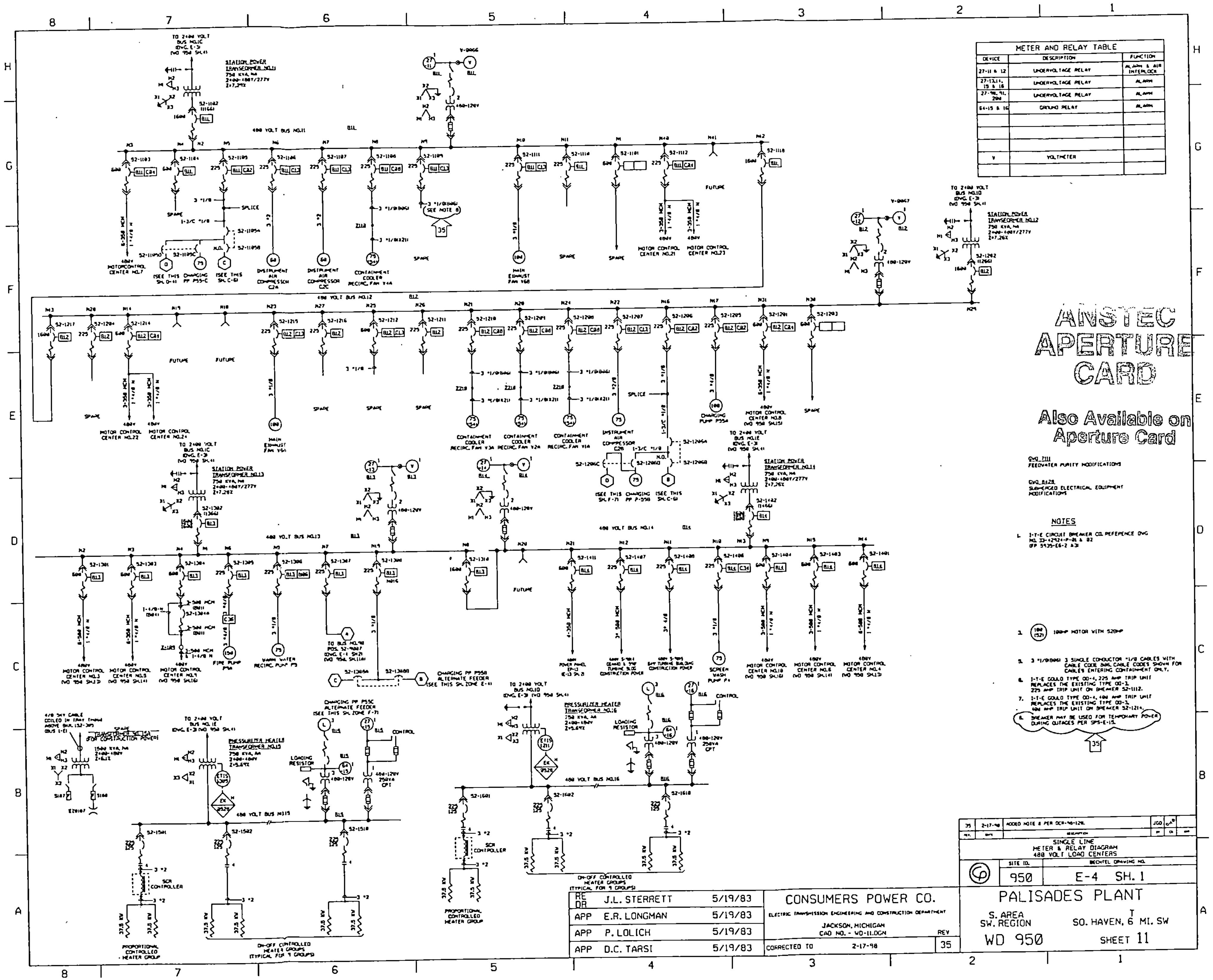


FIG 8-7
FSAR REV 20

9807090043 -26

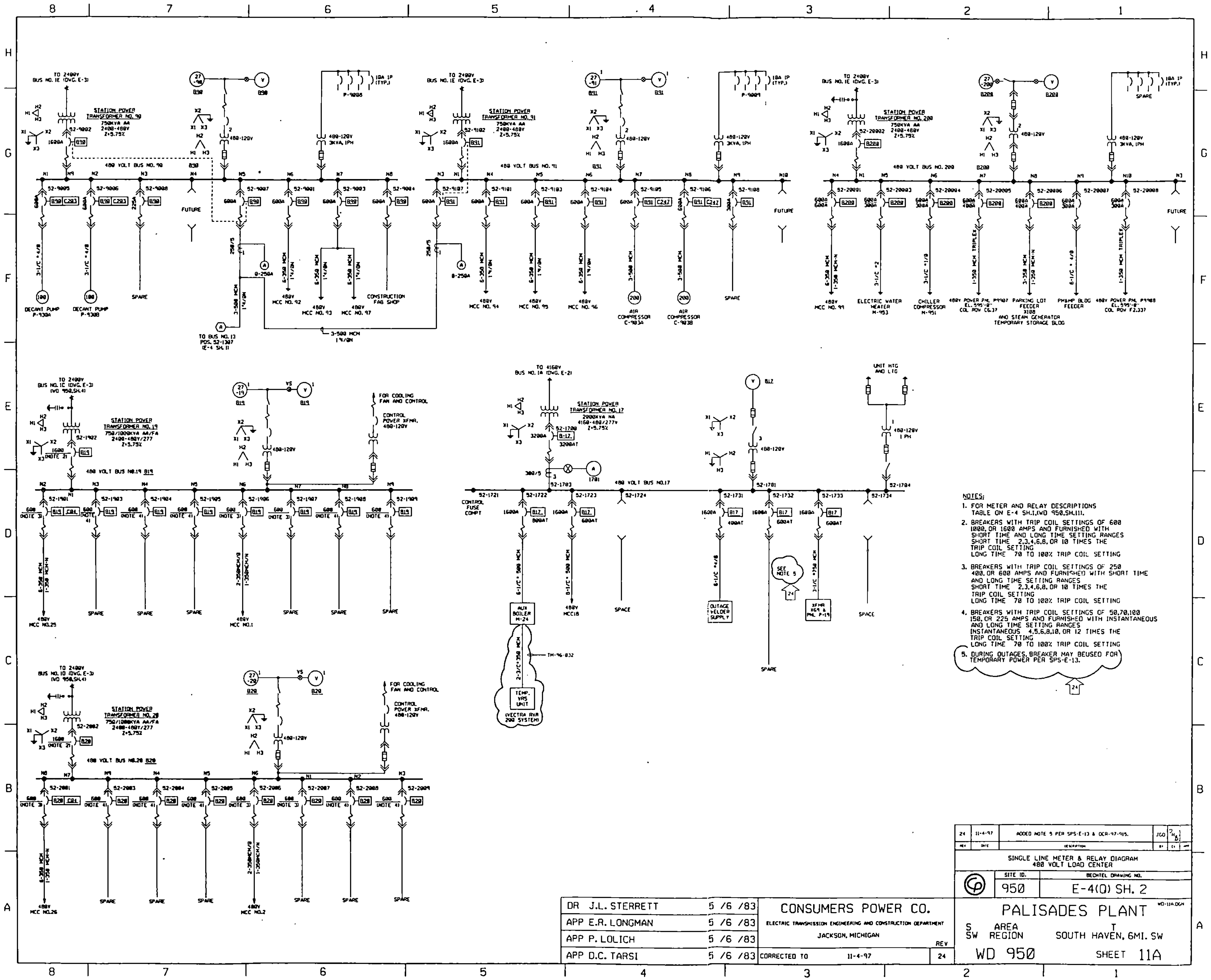


FIG 8-8
FSAR REV 20

9807090043 -27

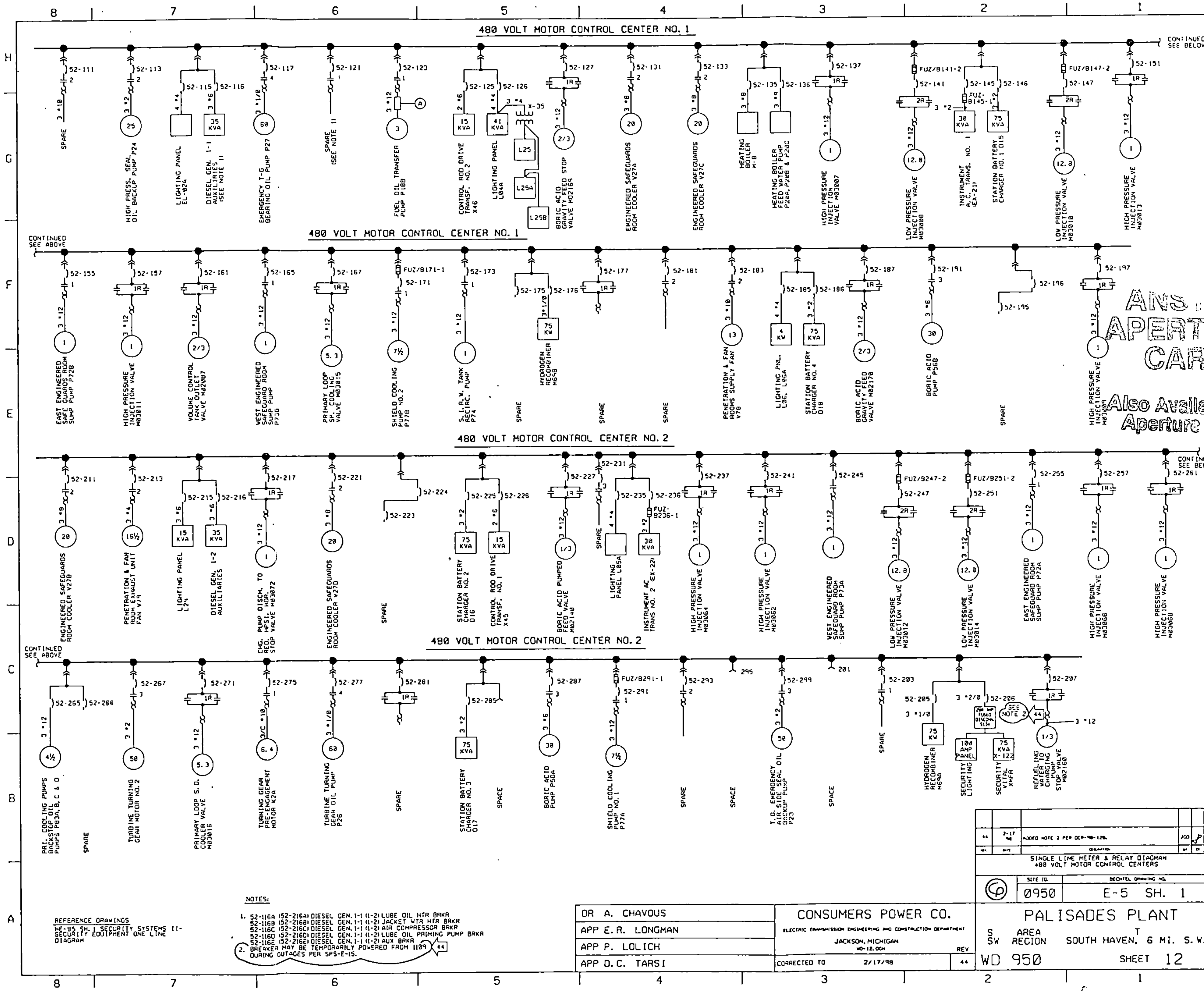
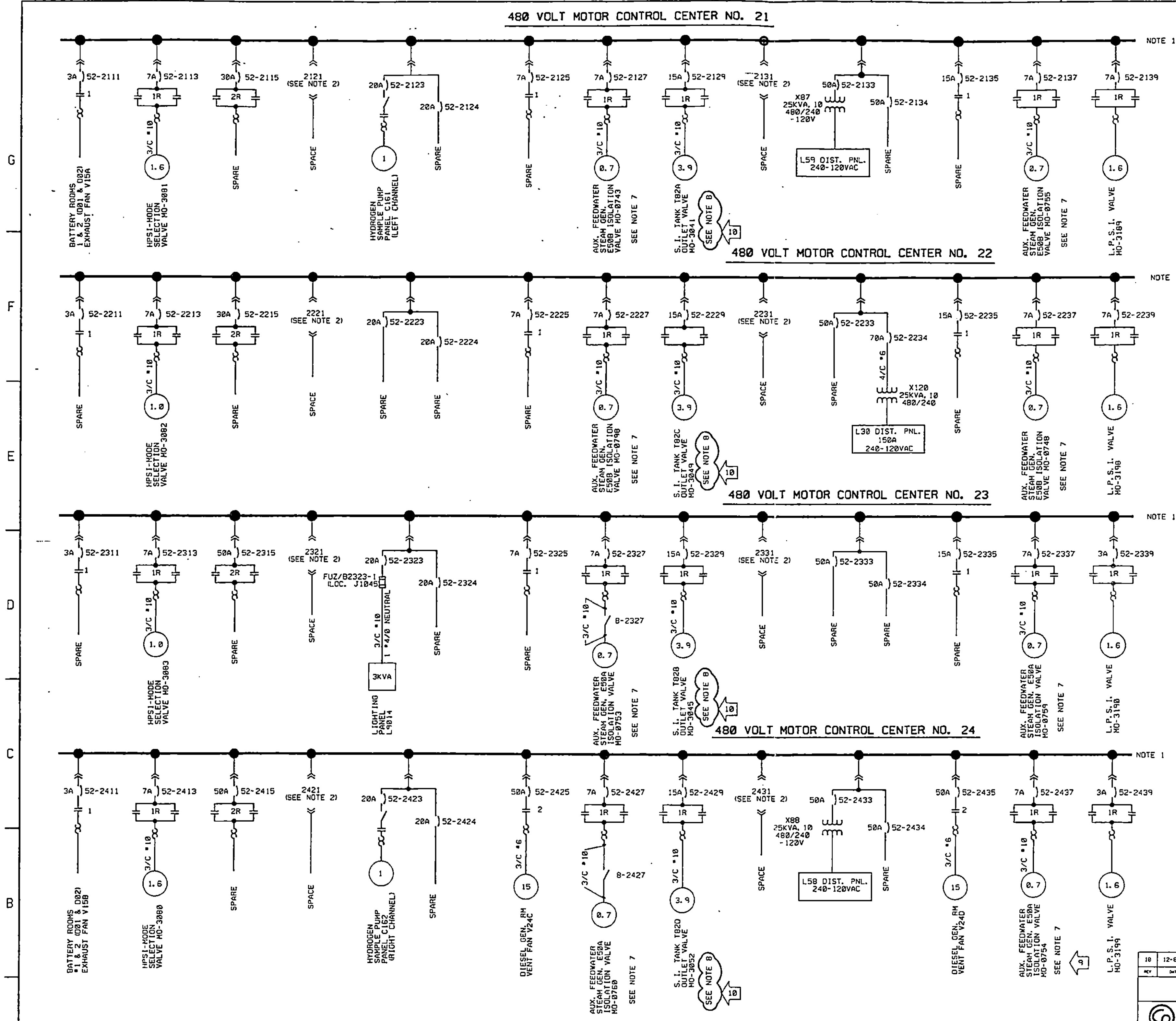


FIG 8-9 SH 1
FSAR REV 20

9807090043 - 28



NOTES

1. MCC 22 WILL BE LOCATED INSWGR. RM 1D AND MCC'S 21, 23 & 24 WILL BE LOCATED IN THE CABLE SPREADING ROOM. CHANNEL ASSIGNMENT WILL BE AS FOLLOWS: MCC'S 21 & 23-LEFT (RED)
MCC'S 22 & 24-RIGHT (GREEN)

2. THIS SPACE IS SUITABLE FOR MOUNTING ONE FEEDER UNIT WITH A MAX-BKR. RATING OF 100A (HORIZONTAL MOUNTING).

3. CPCO. CHANGE NOT PART OF GWO-7502, OR GWO-9062

4. FOR MCC'S 21, 22, 23, & 24 THE TORQUE VALUES ON BOLTED CONNECTIONS SHOULD BE AS FOLLOWS:

BOLT SIZE	RECOMMENDED TORQUE
7/16"	25-40 FT-Lb
1/2"	35-50 FT-Lb
9/16"	45-60 FT-Lb
3/8"	20-30 FT-Lb
1/4"	12-18 FT-Lb

5. REVERSING STARTERS IN MCC UNITS 2127, 2137, 2327, 2337, 2227, 2237, 2427 & 2437 PROCURED UNDER BECHTEL P.O. 12447/059-E-84 (Q)

6. FORMERLY E-5 SH. 1A

7. MOTOR OPERATOR DE-ENERGIZED AND LOCKED OPEN PER SC-95-041.

8. INDICATION & CONTROL ARE D.C. POWERED. 10

REV	DATE	DESCRIPTION	BY	CK	APP
10	12-8-95	ADDED NOTE #8 PER C-PAL-95-0922 & DCR-95-1328.	JGD		KWB

SINGLE LINE METER & RELAY DIAGRAM
480 VOLT MOTOR CONTROL CENTERS

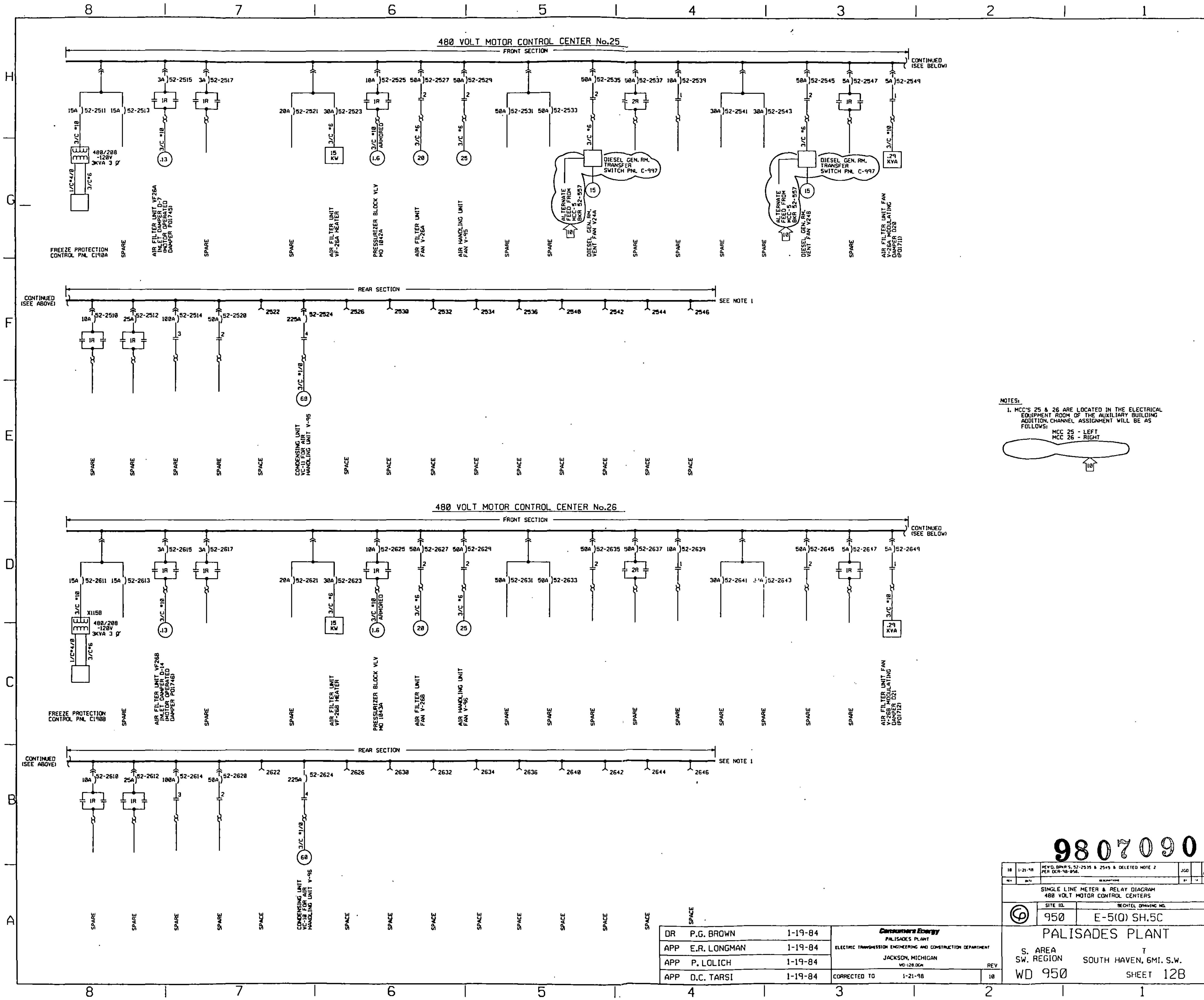
SITE ID. 0950 BECHTEL DRAWING NO. E-5 SH. 5B

RE DR	A. CHAVOUS
APP	E.R. LONGMAN
APP	P. LOLICH
APP	D.C. TARSI

CONSUMERS POWER CO.
ELECTRIC TRANSMISSION ENGINEERING AND CONSTRUCTION DEPARTMENT
JACKSON, MICHIGAN
WD-12A.DGN
CORRECTED TO 12-8-95 REV 10

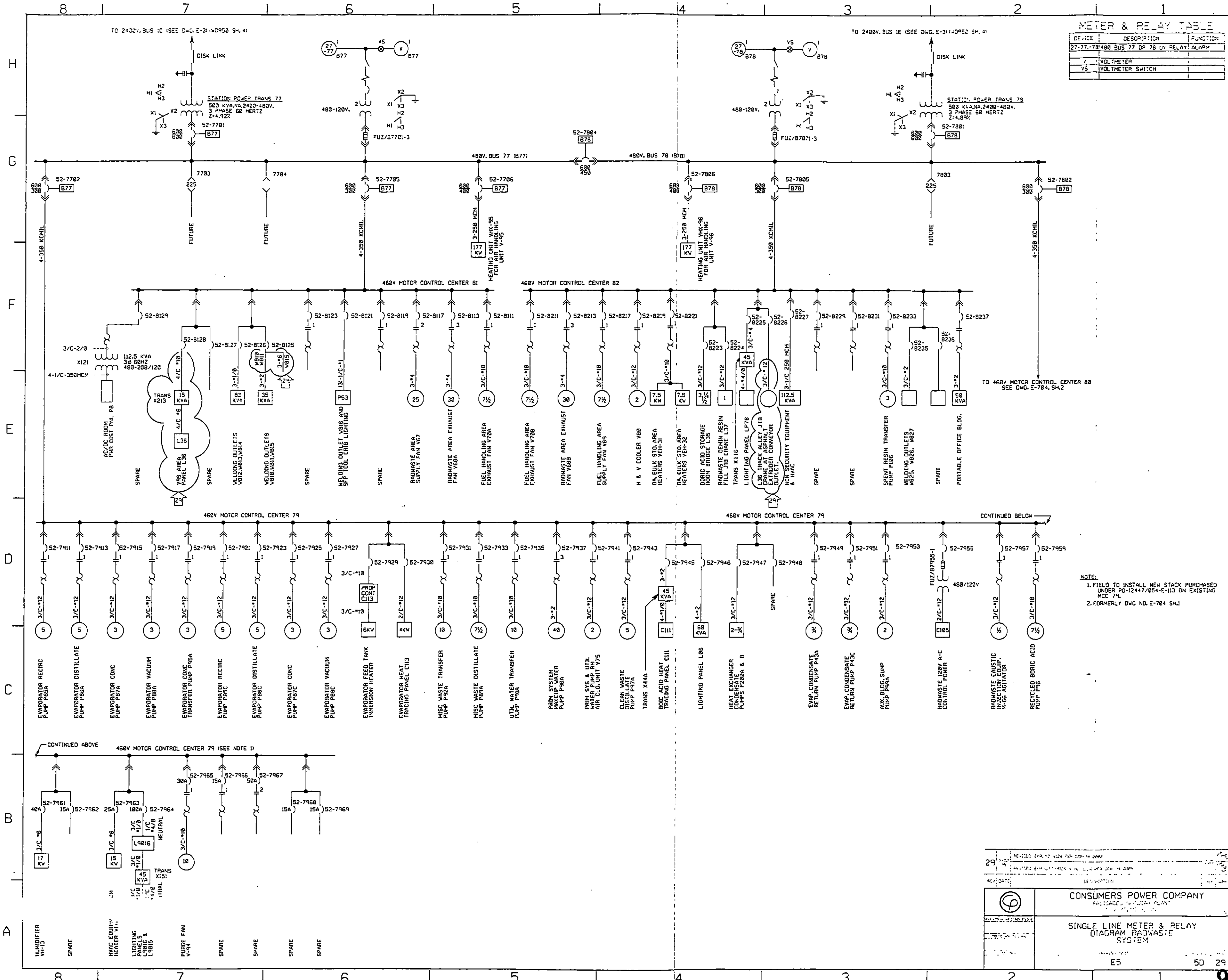
PALISADES PLANT
S SW AREA REGION SOUTH HAVEN, T 6 MI. SW
WD 950 SHEET 12A

Figure 8-9 Sh 2
Rev 19



9807090043 -29

FIG 8-9 SH 3
FSAR REV 20

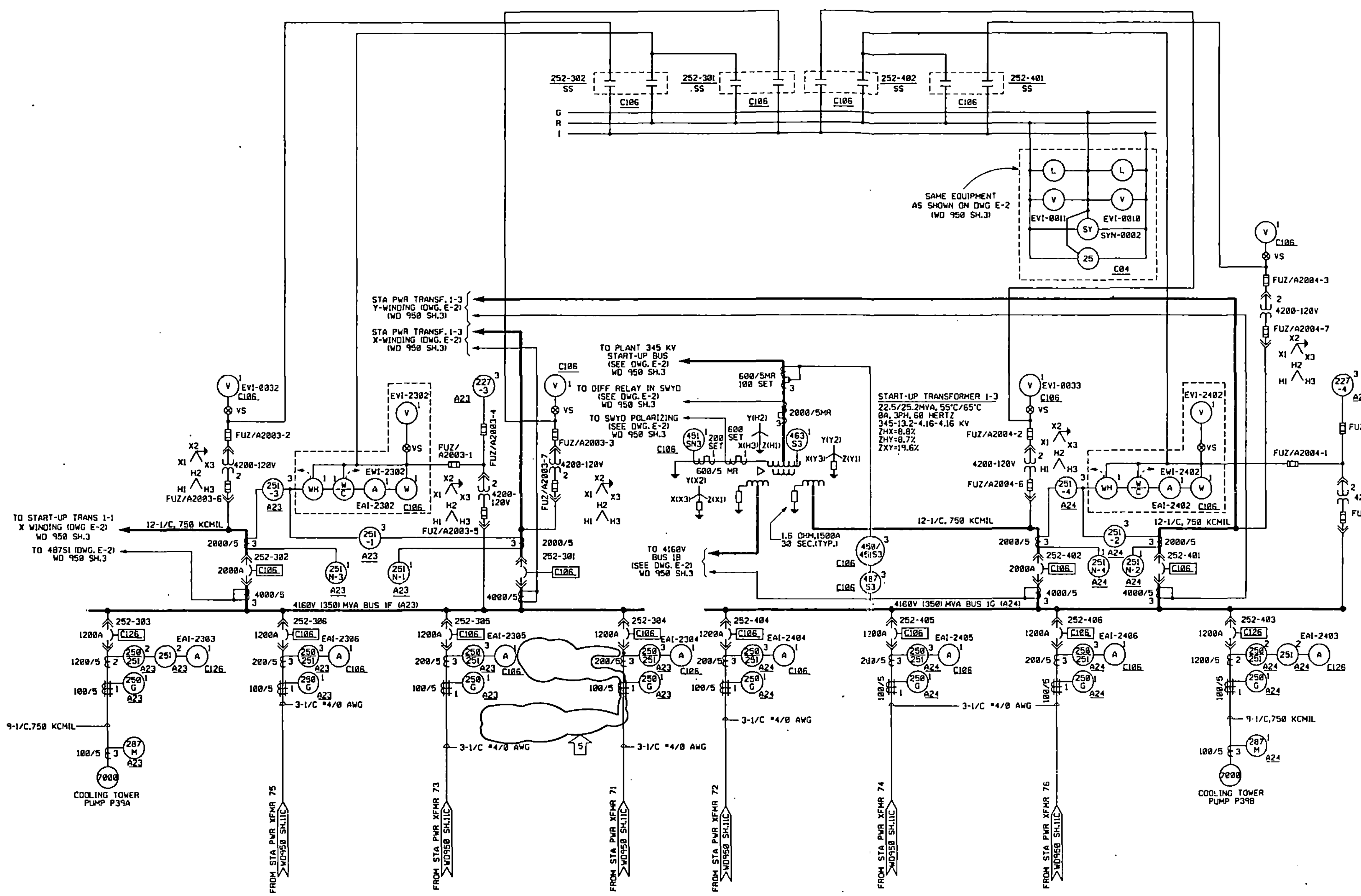


ANSTEC APERTURE CARD

Also Available on Aperture Card

Figure 8-10 FSAR Rev 18

9602290093-16



METER & RELAY TABLE

DEVICE	DESCRIPTION	FUNCTION
227-3,-4	4160V BUS UNDERVOLTAGE RELAY	SHED LOAD AND ALARM
250/251	4000V MOTOR OC RELAY	TRIP ASSOCIATED MOTOR CKT BKR
250/251	4160V TRANS FDR OC RELAY	TRIP ASSOCIATED CKT BKR
250G	4160V FDR GRD OC RELAY	TRIP ASSOCIATED CKT BKR
251	4000V MOTOR OC RELAY	TRIP ASSOCIATED MOTOR CKT BKR
251-1,-2,-3,-4	4160V BUS PHASE OC RELAY	TRIP ASSOCIATED FEEDER CKT BKR
251N-1,-2,-3,-4	4160V BUS NEUT RESIDUAL GRD OVERCURRENT RELAY	TRIP ASSOCIATED FEEDER CKT BKR
287M	4000V MOTOR DIFF RELAY	TRIP ASSOCIATED MOTOR CKT BKR
450/451S3	START-UP TRANS 1-3 DIFF BACK-UP OC RELAY	TRIP ASSOCIATED LOCKOUT RELAY
451SN3	START-UP TRANS 1-3 GRD OC RELAY	TRIP ASSOCIATED LOCKOUT RELAY
463S3	START-UP TRANS 1-3 SUDDEN PRESSURE RELAY	TRIP ASSOCIATED LOCKOUT RELAY
487S3	START-UP TRANS 1-3 DIFF RELAY	TRIP ASSOCIATED LOCKOUT RELAY
A	AMMETER	
V	VOLTMETER	
VS	VOLTMETER SWITCH	
W	WATTMETER	
WH	WATTHOUR METER (WITH CONTACT FOR TOTALIZING)	
W/C	WATT TRANSDUCER (GEN NET WATTS)	
SS	SYNCHRONIZING SWITCH	

NOTE: FORMERLY E-703 WD 950 SH.5

DWG. SPLIT TO BECOME E-3 SH.2 (WD 950 SH. 4A) AND E-4 SH.4 (WD 950 SH. 11C)

REV.	DATE	DESCRIPTION		
5	1-29-98	DELETED CABLE CONN. FROM EX-09 TO BRKR. 252-304 PER C-PAL-97-1312, SC 09-323, & DCR-98-066.	JCD	

SINGLE LINE METER & RELAY DIAGRAM
COOLING TOWER SYSTEMS

SITE ID. 950 | BECHTEL DRAWING NO. E-3 SH.2

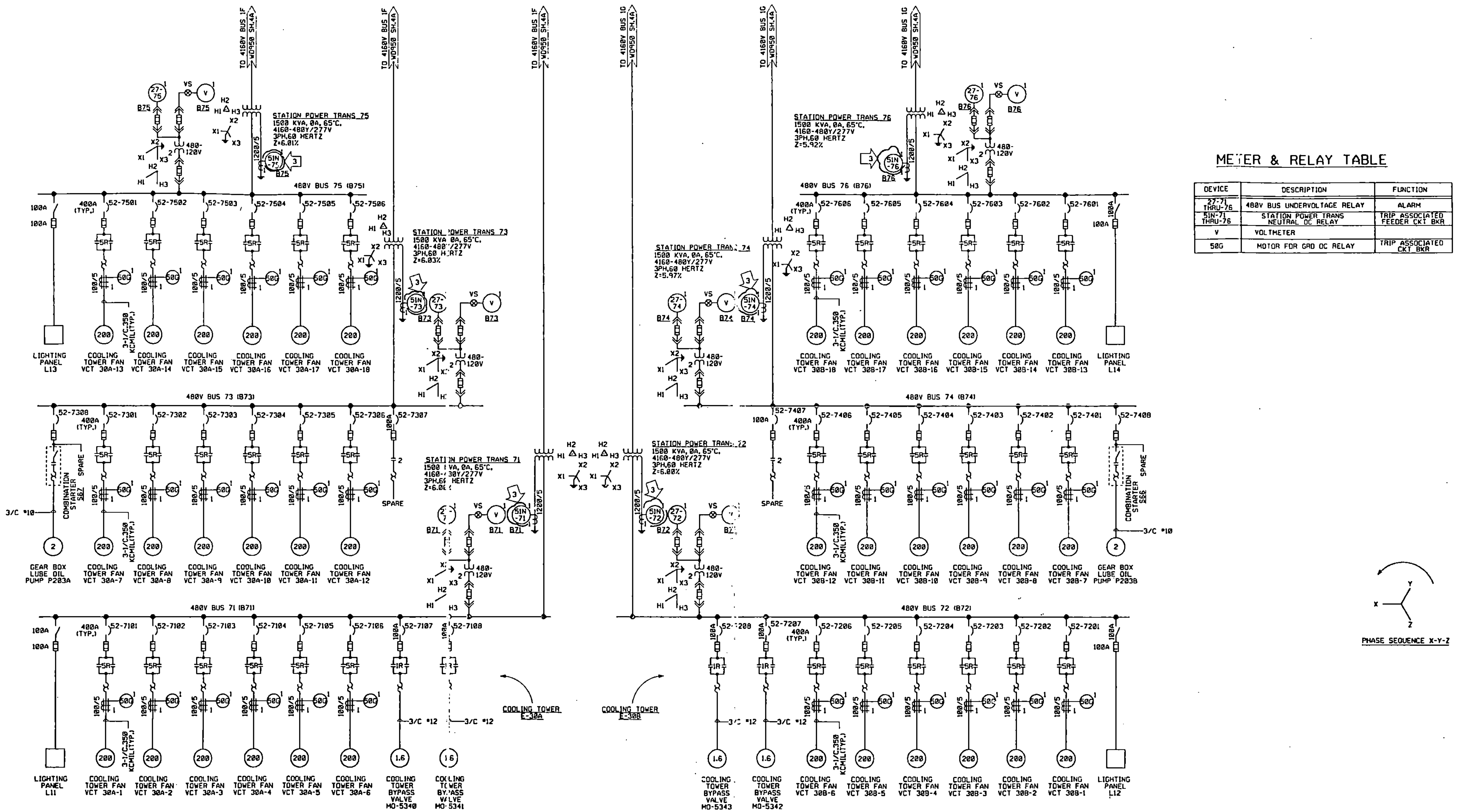
RE DR	R. S. STROUP	10/15/85
APP	E. R. LONGMAN	/ /
APP	P. LOLICH	/ /
APP	D. C. TARSI	/ /

CONSUMERS POWER CO.
JACKSON, MICHIGAN
WD-4A.DGN

CORRECTED TO 1-29-98 REV 5

PALISADES PLANT
S SW AREA REGION T SO. HAVEN. 6 MI. SW
WD 950 SHEET 4A

FIG 8-11 SH 1
FSAR REV 20



METER & RELAY TABLE

DEVICE	DESCRIPTION	FUNCTION
27-71 THRU-76	480V BUS UNDERVOLTAGE RELAY	ALARM
51N-71 THRU-76	STATION POWER TRANS NEUTRAL OC RELAY	TRIP ASSOCIATED FEEDER CKT BKR
V	VOLTMETER	
50G	MOTOR FOR GRD OC RELAY	TRIP ASSOCIATED CKT BKR

9501040188-48

NOTE: FORMERLY E-703 WD 950 SH.5
DWG. SPLIT TO BECOME E-3 SH.2 WD 950 SH. 4A AND E-4 SH.4 WD 950 SH.11C.

3	5-10-94	CORRECTED NEUTRAL OC RELAY NUMBERS ON LC'S 71-76 PER DCR 94-399	JFM	—	RKM

SINGLE LINE METER & RELAY DIAGRAM
COOLING TOWER SYSTEMS

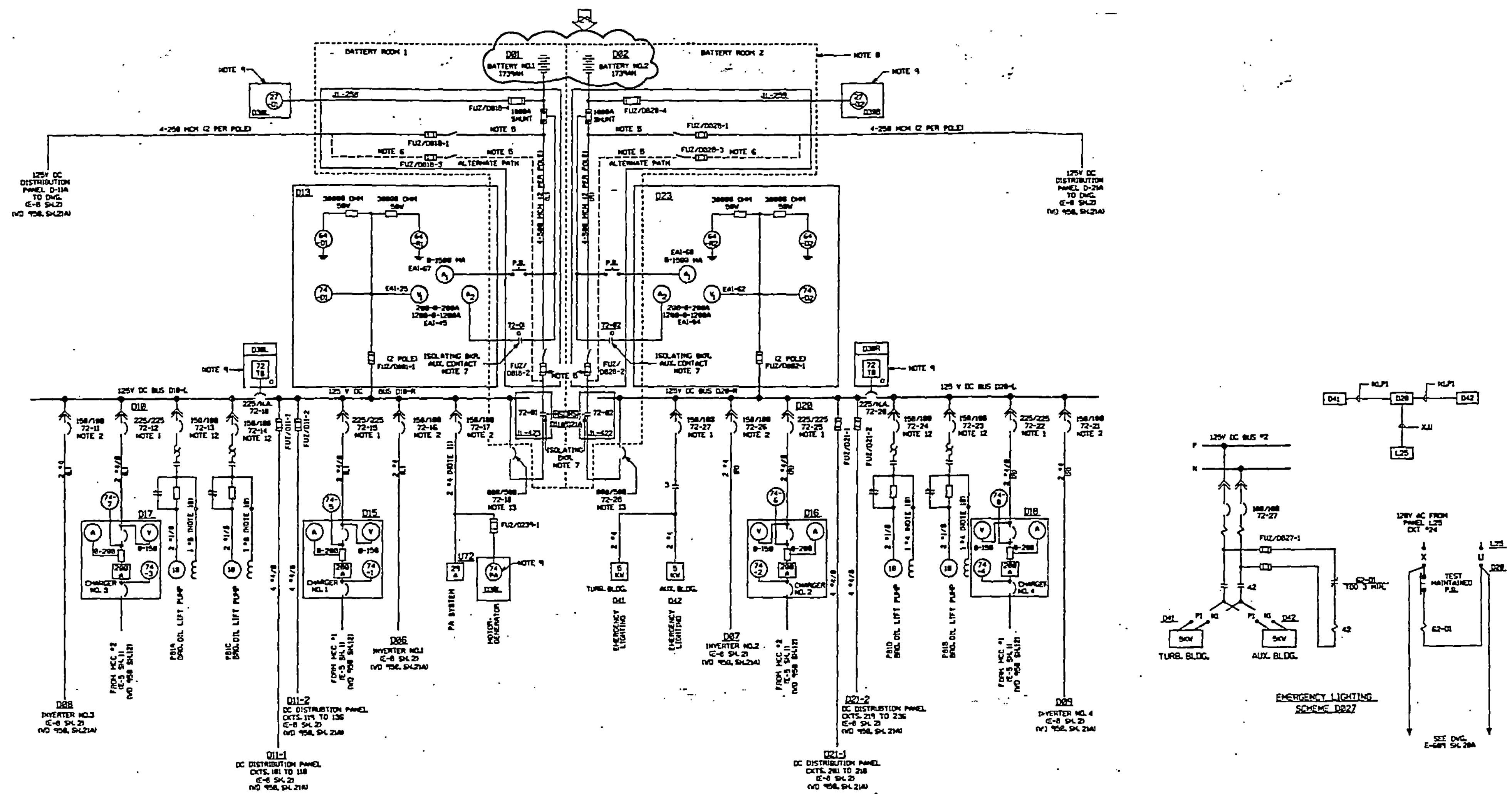
SITE ID.	BECHTEL DRAWING NO.
950	E-4 SH.4

RE DR	J.L. STERRETT	5-12-83
APP	E.R. LONGMAN	5-12-83
APP	P. LOLICH	5-12-83
APP	D.C. TARSI	5-12-83

CONSUMERS POWER CO.
ELECTRIC TRANSMISSION ENGINEERING AND CONSTRUCTION DEPARTMENT
JACKSON, MICHIGAN
WD-11C.DGN
CORRECTED TO 5-10-94 REV 3

PALISADES PLANT
S AREA SW REGION T SOUTH HAVEN,6 MI.SW
WD 950 SHEET 11C

Figure 8-11 Sh 2 FSAR Rev 17



NOTES:

1. BREAKERS WITH THERMAL MAGNETIC TRIP.
2. BREAKERS WITH ADJUSTABLE MAGNETIC TRIP ONLY.
3. BREAKERS WITH ADJUSTABLE MAGNETIC TRIP ONLY, STARTERS WITH THERMAL OVERLOAD ELEMENTS USED FOR ALARM ONLY, SET AT 115% MOTOR FULL LOAD CURRENT.
4. (L) INDICATES CIRCUITS ROUTED IN LEFT HAND CHANNEL & (R) INDICATES CIRCUITS ROUTED IN RIGHT HAND CHANNEL SEPARATED FROM EACH OTHER.
5. FUSES HOOK-STICK OPERATED.
6. ALTERNATE BY PASS POSITION FOR 300A FUSES.
7. NON-AUTOMATIC SHUNT TRIP BREAKER WITH AUXILIARY CONTACTS. BREAKER TRIPPED FROM D-11A (D-21A).
8. EQUIPMENT WITHIN DOTTED LINE IS LOCATED IN BATTERY ROOM. FOR MORE INFORMATION ON JL258 AND JL259, SEE DWG. E-625, SHT'S. 258 & 259.
9. RELAYS AND AUXILIARY SWITCHES USED TO MONITOR BREAKER LINE UPS. FOR MORE INFORMATION SEE FC-495 AND SCHEMES D239 AND D248 ON DWG. E-130 SH.2.
10. MIN. OF TWENTY-FIVE (25) FEET OF CONDUCTOR FROM STARTER IS 1/C #4 AWG. #4 AWG. SPLICED TO #8 AWG.
11. RECONDUCTORED WITH 2-1/C #4 AWG.
12. BREAKERS WITH THERMAL MAGNETICS TRIPS. STARTER WITH THERMAL OVERLOAD ELEMENTS USED FOR ALARM ONLY. SET @ 115% MOTOR FULL LOAD CURRENT.
13. BREAKER WITH THERMAL TRIP.

METER AND RELAY TABLE

DEVICE	DESCRIPTION	MFGR./TYPE	FUNCTION
62-01	TIMER FOR EMCY. LIGHTS	AGLAKE	CONTROL
64-01	DC BUS 1 GRND DETECTOR	WESTON #795.5-0-5MA	ALARM
64-02	DC BUS 2 GRND DETECTOR	WESTON #795.5-0-5MA	ALARM
64-R1	DC BUS 1 GRND RECORDER	ESTERLINE ANGUS MODEL A601C	GRD CURRENT MONITORING
64-R2	DC BUS 2 GRND RECORDER	ESTERLINE ANGUS MODEL A601C	GRD CURRENT MONITORING
64-1	PREF. AC BUS 1 GRND DET	ELECTRONIC DEVICE	ALARM
64-2	PREF. AC BUS 2 GRND DET	ELECTRONIC DEVICE	ALARM
64-3	PREF. AC BUS 3 GRND DET	ELECTRONIC DEVICE	ALARM
64-4	PREF. AC BUS 4 GRND DET	ELECTRONIC DEVICE	ALARM
74-01	DC BUS 1 LOSS OF V RELAY	POTTER & BRUMFIELD KRP11DG	ALARM
74-02	DC BUS 2 LOSS OF V RELAY	POTTER & BRUMFIELD KRP11DG	ALARM
A	AMMETER	WESTINGHOUSE 241	INDICATION
V	VOLTMETER	WESTINGHOUSE 241	INDICATION
F	FREQUENCY METER	J.B.T.	INDICATION
74-1	CHARGER 1 AC UNDER VOLTAGE RELAY	ELECTRONIC DEVICE	ALARM
74-2	CHARGER 2 AC UNDER VOLTAGE RELAY	ELECTRONIC DEVICE	ALARM
74-3	CHARGER 3 AC UNDER VOLTAGE RELAY	ELECTRONIC DEVICE	ALARM
74-4	CHARGER 4 AC UNDER VOLTAGE RELAY	ELECTRONIC DEVICE	ALARM
27-1	PREF. AC BUS 1 UNDER V RELAY	ELECTRONIC DEVICE	ALARM
27-2	PREF. AC BUS 2 UNDER V RELAY	ELECTRONIC DEVICE	ALARM
27-3	PREF. AC BUS 3 UNDER V RELAY	ELECTRONIC DEVICE	ALARM
27-4	PREF. AC BUS 4 UNDER V RELAY	ELECTRONIC DEVICE	ALARM
27-5	INSTR. AC BUS UNDER V RELAY		
A_1	MILLIAMMETER	SIGMA LUMIGRAPH 9278	INDICATION
A_2	AMMETER	WESTINGHOUSE KX 241	INDICATION
V_1	VOLTMETER	WESTINGHOUSE KX 241	INDICATION
74-5	CHARGER #1 DC UNDERVOLTAGE RELAY	POTTER & BRUMFIELD KRP11DG	ALARM
74-6	CHARGER #2 DC UNDERVOLTAGE RELAY	POTTER & BRUMFIELD KRP11DG	ALARM
74-7	CHARGER #3 DC UNDERVOLTAGE RELAY	POTTER & BRUMFIELD KRP11DG	ALARM
74-8	CHARGER #4 DC UNDERVOLTAGE RELAY	POTTER & BRUMFIELD KRP11DG	ALARM

9707070093 -12

REV.	DATE	DESCRIPTION	BY		
29	3-14-97	REV'D. 1800 TO 1739 AMPHRS PER FES-95-206 & DCR-97-294.	JGD	DMS	

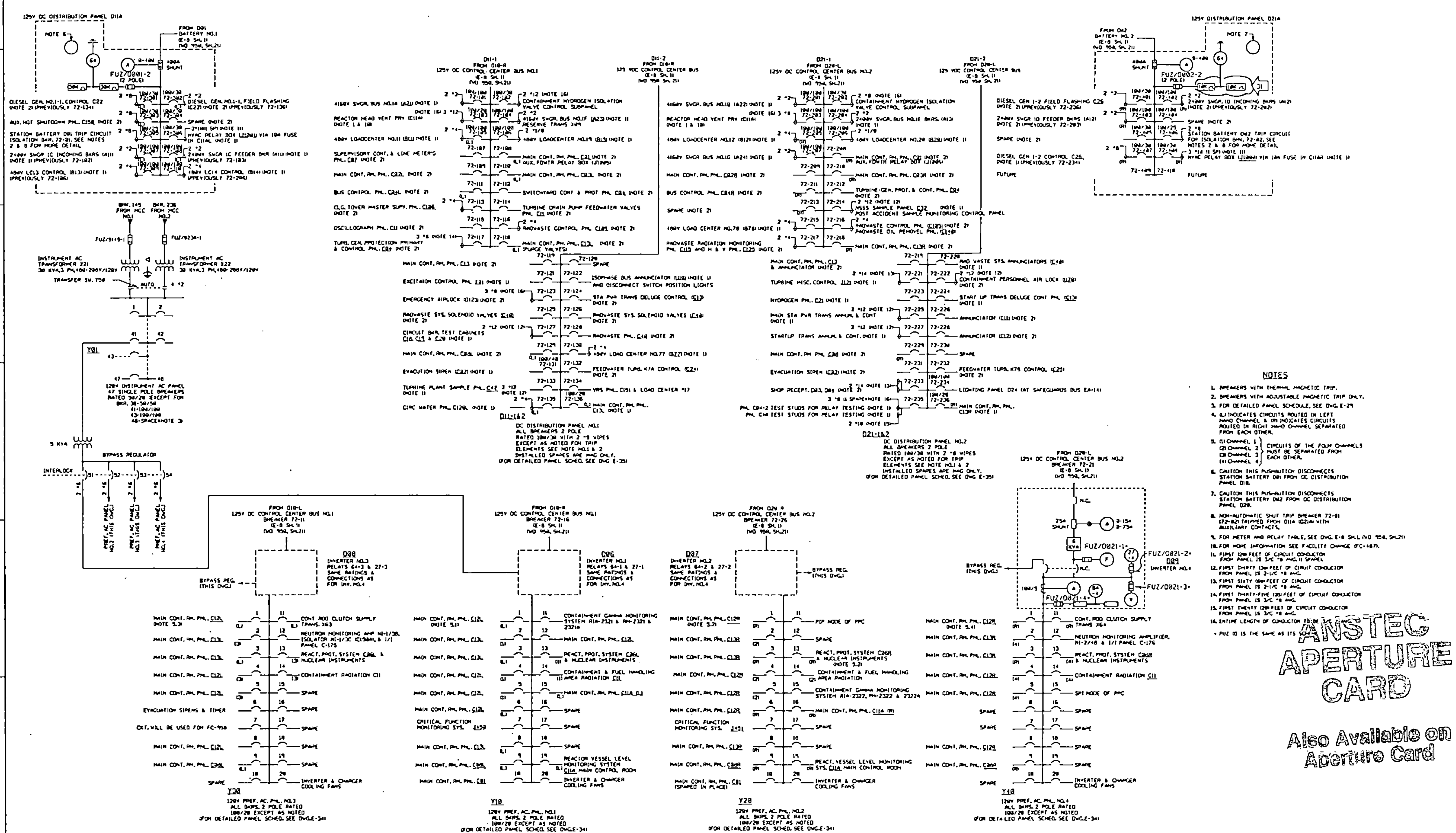
SINGLE LINE METER & RELAY DIAGRAM
125V DC
120V INSTRUMENT & PREFERRED AC SYSTEM

SITE ID: 950 — BECHTEL DRAWING NO.: E-8 SH. 1

CONSUMERS POWER CO.
PLANT MODIFICATIONS AND MISCELLANEOUS PROJECTS DEPARTMENT
JACKSON, MICHIGAN
WD-21.DGN

CORRECTED TO 3-14-97 — REV 29

PALISADES PLANT
S. AREA
SW. REGION
SO. HAVEN, 6 MI. SW
WD 950 — SHEET 21



9807090043 -31

REF. DRAWING
E-18 SH.1 & 2

REV	DATE	DESCRIPTION	BY	CK	APP
31	2-18-98	REVISED RESISTANCES IN D11A AND D21A PER SC-97-242 AND DCR 98-891	DMB		

SINGLE LINE METER & RELAY DIAGRAM
125V DC
120V INSTRUMENT & PREFERRED AC SYSTEM

SITE NO.	BECHTEL DRAWING NO.
950	E-8 SH. 2

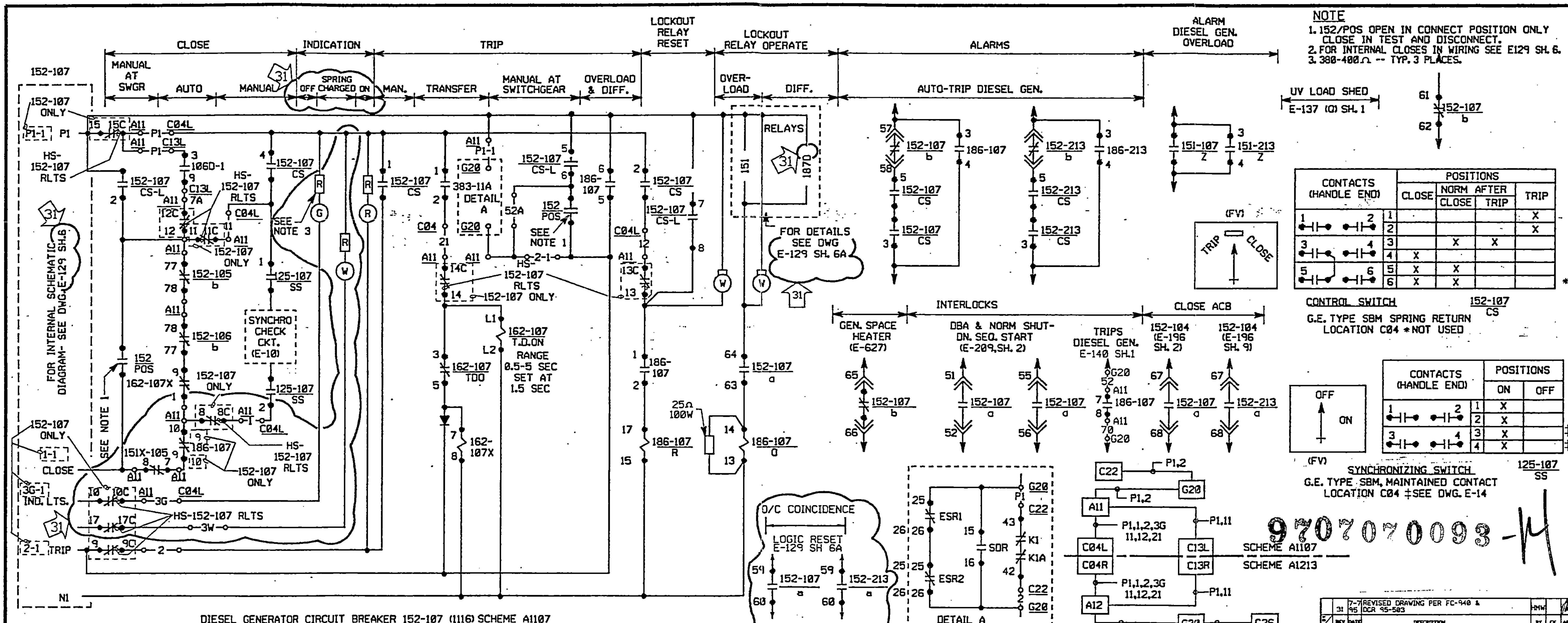
RE DR	J.L. STERRETT	5/16/83
APP	E.R. LONGMAN	5/16/83
APP	P. LOLICH	5/16/83
APP	D.C. TARSI	5/16/83

Consumers Energy
PALISADES PLANT
ELECTRIC TRANSMISSION ENGINEERING AND CONSTRUCTION DEPARTMENT
JACKSON, MICHIGAN
WD-21A.DCH

CORRECTED TO 2-18-98 REV 31

PALISADES PLANT
S. AREA
SW. REGION
SO. HAVEN, 6 MI. SW
WD 950
SHEET 21A

FIG 8-12 SH 2
FSAR REV 20



9707070093 -14

DIESEL GENERATOR CIRCUIT BREAKER 152-107 (1116) SCHEME A1107

ADAPTER TABLE

DESCRIPTION	A.C.B.		SCHEME	D. GEN. OVERLOAD & GEN. DIFF. PROT. L.O. REL.		BUS INCOMING A.C.B.'S			AUTO-CLOSE RELAYS		CONT. SW.	TRANSFER TRIP		TRIP BLOCK RELAY		LOCAL CONT. SW.		SYNCHRO. SWITCH		FUSE NO.	OVER CURRENT LOCKOUT	
	NO.	LOC		NO.	LOC	NO.	NO.	LOC	NO.	LOC	LOC.	383-	LOC	NO.	LOC	NO.	LOC.	NO.	LOC.		NO.	LOC.
DIESEL GEN. 1-1 CKT. BKR.	152-107(1116)	A11	A1107	186-107	A11	152-105	152-106	A11	106D-1	C13L	C04L	11A/1-2	C04L	162-107	A11	152-107 CS-L	A11	125-107/SS	C04	[illegible]	151X-105	A11
DIESEL GEN. 1-2 CKT. BKR.	152-213(1115)	A12	A1213	186-213	A12	152-203	152-202	A12	106D-2	C13R	C04R	12A/1-2	C04R	162-213	A12	152-213 CS-L	A12	125-213/SS	C04	[illegible]	151X-203	A12

	REV	DATE	DESCRIPTION	BY	CK	APP
	31	7-7-95	REVISED DRAWING PER FC-940 & DCR 95-503	HMW		

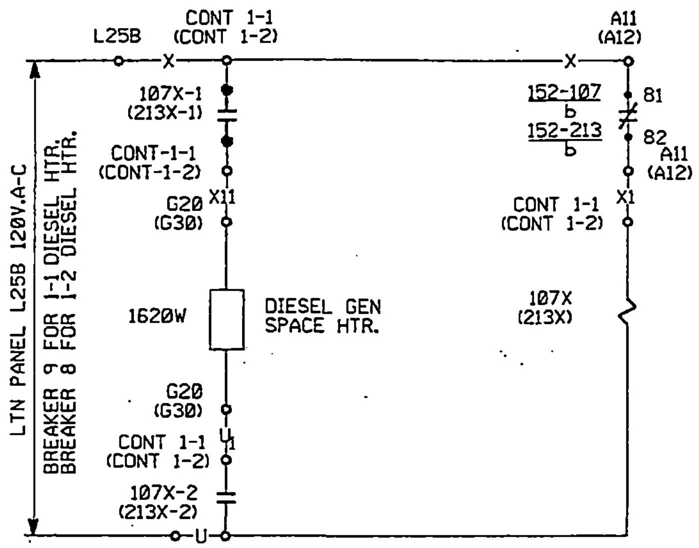
CONSUMERS POWER COMPANY
PALISADES NUCLEAR PLANT
COVERT, MICHIGAN

E139-1.DGN

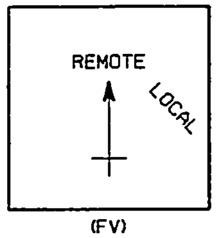
SCHEMATIC DIAGRAM
DIESEL GENERATOR BKR'S

DRAWING NUMBER	SHT. NO.	REV
E139	1	31

Figure 8-14
Rev 19



HS-152-107 RLTS



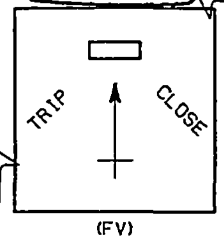
GE TYPE SBM
MAINTING CONTACT
LOCATION A11

CONTACTS (HANDLE END)		POSITIONS		LOCATION
		REMOTE	LOCAL	SCHEM. DWG.
1 1C 2C 2	1		X	E-129 SH 6
	2		X	E-129 SH 6
3 3C 4C 4	3		X	*
	4		X	*
5 5C 6C 6	5		X	*
	6	X		*
7 7C 8C 8	7	X		*
	8	X		E-139 SH 1
9 9C 10C 10	9	X		E-139 SH 1
	10	X		E-139 SH 1
11 11C 12C 12	11	X		E-139 SH 1
	12	X		E-139 SH 1
13 13C 14C 14	13	X		E-139 SH 1
	14	X		E-139 SH 1
15 15C 16C 16	15	X		E-139 SH 1
	16	X		*
17 17C 18C 18	17	X		E-139 SH 1
	18	X		*
19 19C 20C 20	19	X		*
	20	X		*

* NOT USED

16

152 - 107/CS-L 16



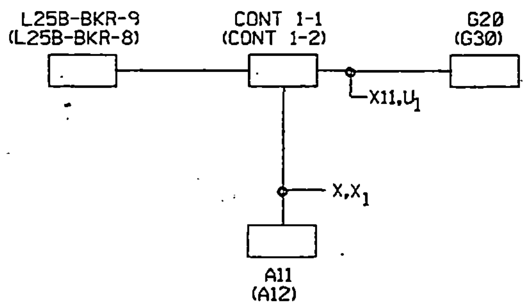
CONTACTS	POSITIONS			
	CLOSE	NORM AFTER		TRIP
		CLOSE	TRIP	
1-2	X			
3-4				*
5-6				X
7-8				X

16

CONTROL SWITCH

G. E. TYPE SBM, SPRING RETURN
LOCATION $\frac{152\text{-}107}{\text{CS-L}}$ & $\frac{152\text{-}213}{\text{CS-L}}$

* NOT USED



SCHEME A1107
(SCHEME A1213)

9707070093 -15

REV	DATE	DESCRIPTION	BY	CK	APP	REV	DATE	DESCRIPTION	BY	CK	APP
16	8-10 95	REVISED PER FC-940 & DCR-95 577 (RTD)	ALB		MO						

SECTION HEAD APP. C.J. MCDONALD 4/18/83
DIVISION HEAD APP. P. LOLICH 4/18/83
DEPARTMENT HEAD APP. D.C. TARSI 4/18/83

DR. J. L. STERRETT
CK. T.D. VOGT 4/18/83

CONSUMERS POWER CO.
PALISADES PLANT
E139-2.DGN

SCHEMATIC DIAGRAM
DIESEL GENERATOR BKR'S

NO. E-139 SHEET 2

REV. 16

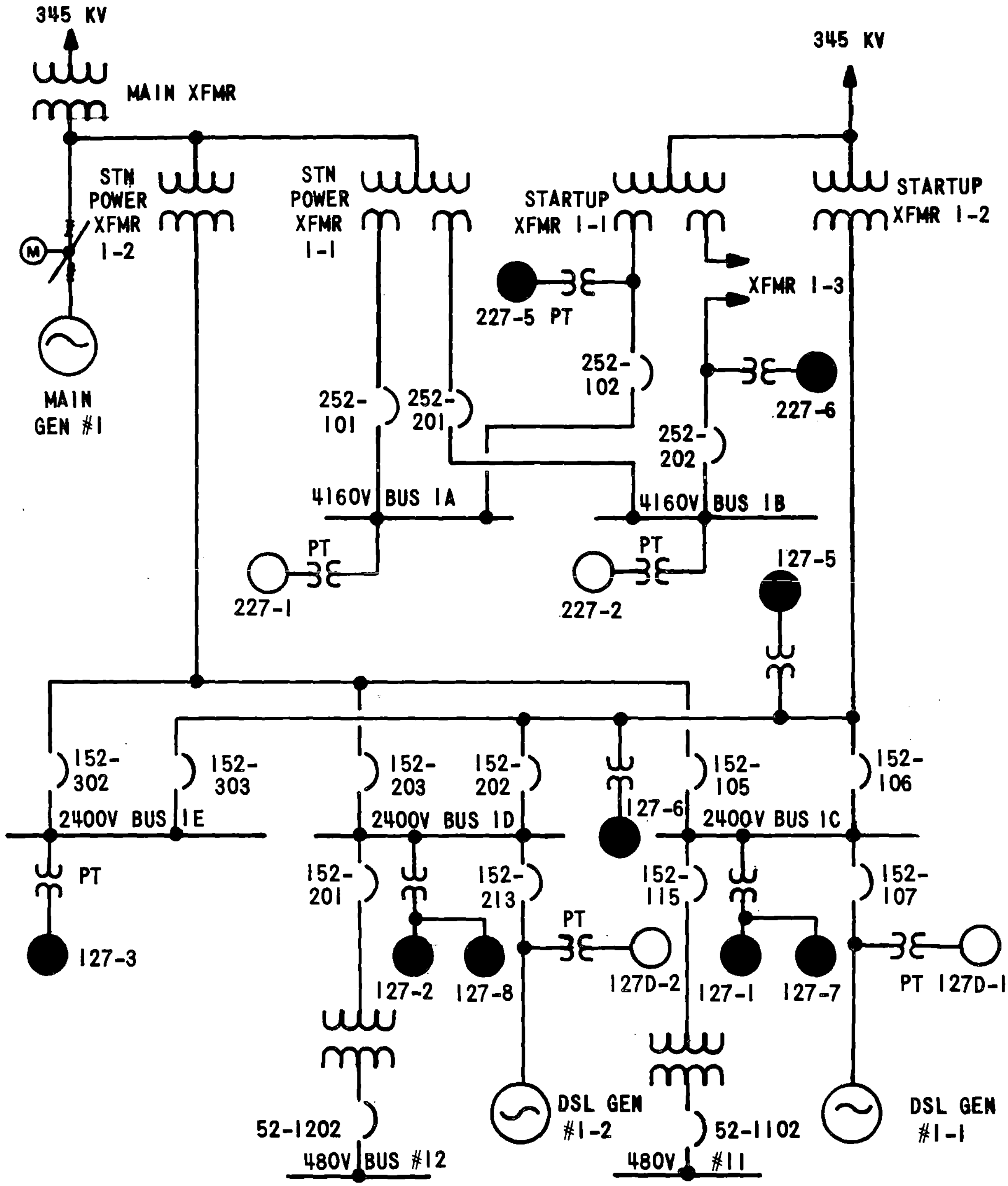
Figure 8-15
Rev 19

FIGURE 8-16

Deleted

FIGURE NO. 8-17

(Has been replaced by E17 Dwgs)
(Chapter 7 Fig. 7-14)



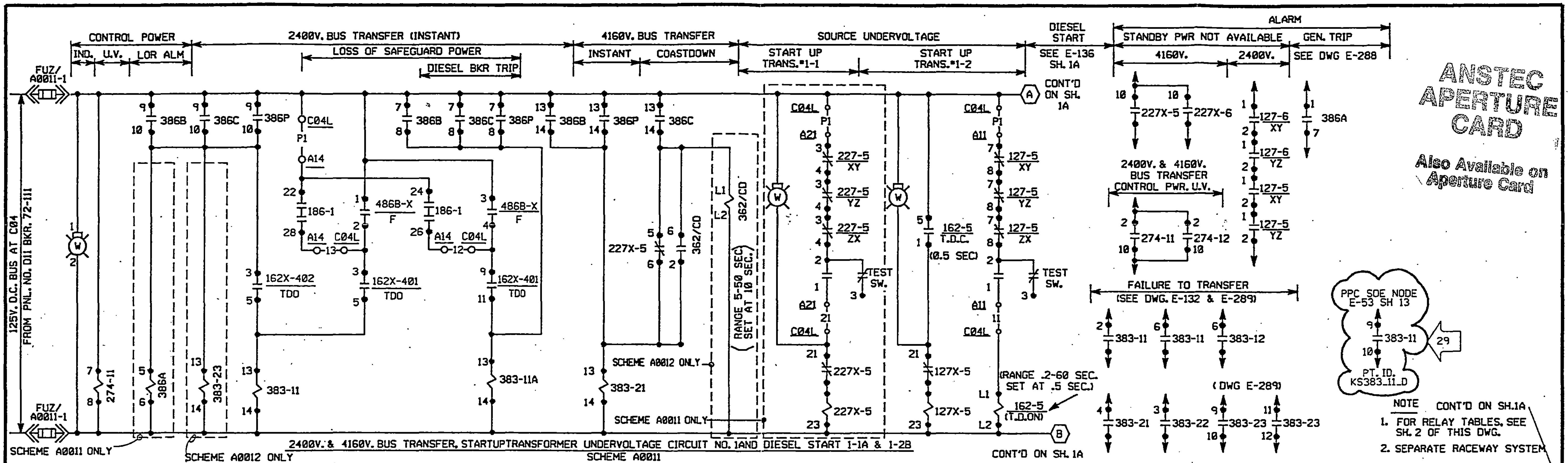
= VOLTAGE PROTECTION SENSOR

CONSUMERS POWER COMPANY
PALISADES PLANT
FSAR UPDATE

VOLTAGE PROTECTION SENSORS LOCATION

FIGURE NO 8-18

REVISION NO 11



NOTE CONT'D ON SH.1A

1. FOR RELAY TABLES, SEE SH. 2 OF THIS DWG.
2. SEPARATE RACEWAY SYSTEM

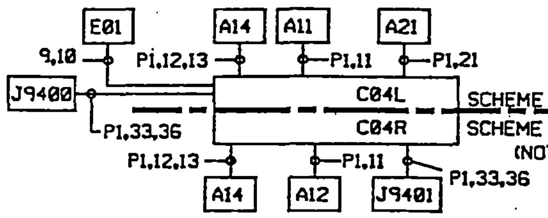
ANSTEC APERTURE CARD

Also Available on Aperture Card

ADAPTER TABLE

SCHEME	D-C SOURCE		START UP TRANS. #1-2 UNDERVOLTAGE RELAYS				S.U. TRANS. #1-2 AUX. U.V. RELAYS		START UP TRANS. #1-1 UNDERVOLTAGE RELAYS				S.U. TRANS. #1-1 AUX. U.V. RELAYS		2400V. BUS XFER RELAYS		4160V. BUS XFER RELAYS	
	BKR. NO.	LOC.	NO.	NO.	NO.	LOC.	NO.	LOC.	NO.	NO.	NO.	LOC.	NO.	LOC.	NO.	LOC.	NO.	LOC.
A0011	72-111	D11	127-5/XY	127-5/YZ	127-5/ZX	A11	127X-5	C04L	227-5/XY	227-5/YZ	227-5/ZX	A21	227X-5	C04L	383-11	C04L	——	——
A0012	72-211	D21	127-6/XY	127-6/YZ	127-6/ZX	A12	127X-6	C04R	——	——	——	—	——	——	383-12	C04R	383-23	C04R

SCHEME	4160V. BUS XFER RELAYS				CONTROL PWR. U.V.				INTERLOCKING RELAYS & CONTACT NUMBERS								START UP TRANS. #1-2 U.V. TIME DELAY REL.		DIESEL BKR TRIP	
	NO.	LOC.	NO.	LOC.	NO.	LOC.	NO.	LOC.	386B	386C	386P	362/CD	186-1	162X-401	162X-402	486B-X/F	NO.	LOC.	NO.	LOC.
A0011	——	——	383-21	C04L	274-11	C04L	386A	C04L	7 & 8 9 & 10 13 & 14	7 & 8 9 & 10 13 & 14	7 & 8 9 & 10 13 & 14	2 & 6	22-28 24-26	3-5 9-11	3-5	1-2 3-4	162-5	C04L	383-11A	C04L
A0012	383-23	C04R	383-22	C04R	274-12	C04R	——	——	19 & 20 11 & 12 15 & 16	19 & 20 11 & 12 15 & 16	19 & 20 11 & 12 15 & 16	1 & 5	44-46 52-58	4-6 10-12	4-6	5-6 7-8	162-6	C04R	383-12A	C04R



9707070093 -17

REV	DATE	DESCRIPTION	BY	CK'D	APP.
29	7-31-95	ADD: CONTACTS 9 & 10 & PPC SOE NODE PER FC-933-06 & DCR-95-518	MDS		JM

SECTION HEAD DATE
APP. C.J. MCDONALD 4/18/83
DIVISION HEAD
APP. P. LOLICH 4/18/83
DEPARTMENT HEAD
APP. D.C. TARSI 4/18/83

DR. J. L. STERRETT

CONSUMERS POWER CO.
PALISADES PLANT
E136-1.DGN

CK. T.D. VOGT 4/18/83

SCHEMATIC DIAGRAM
2400V. & 4160V. BUS TRANSFER

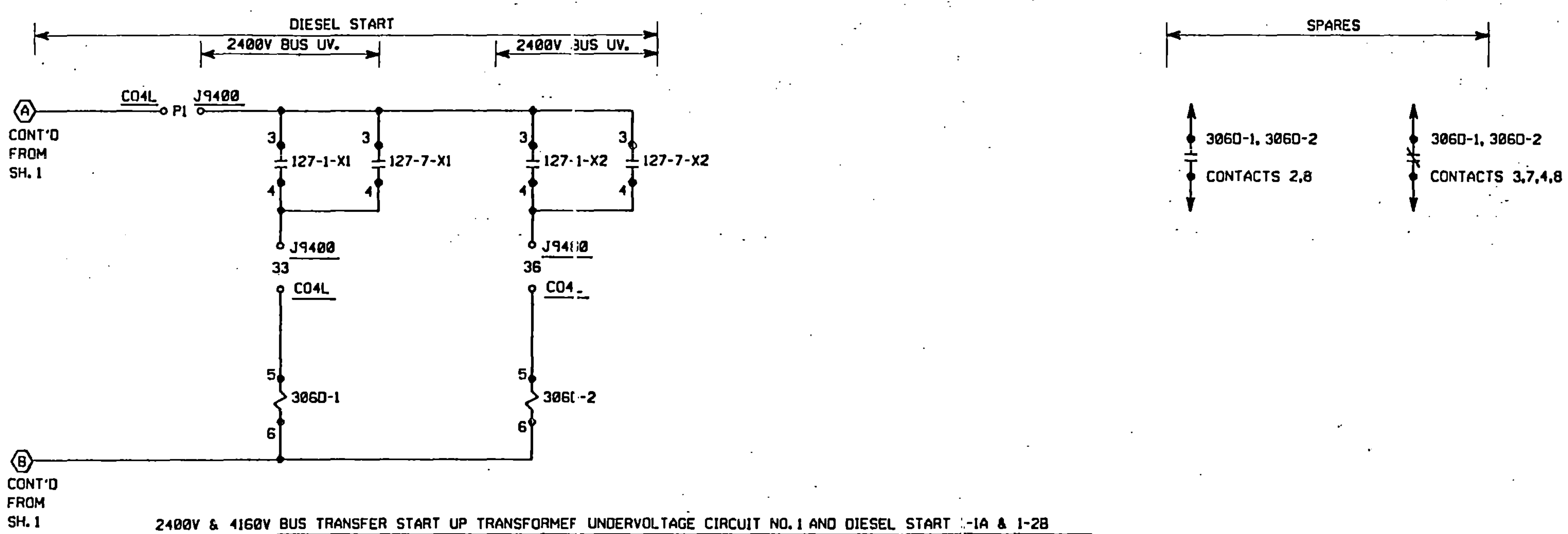
NO. E-136 SHEET 1

REV. 29

Figure 8-19
Rev 19

ANSTEC APERTURE CARD

Also Available on Aperture Card



2400V & 4160V BUS TRANSFER START UP TRANSFORMER UNDERVOLTAGE CIRCUIT NO. 1 AND DIESEL START 1-1A & 1-2B
SCHEME A0011

CONT'D FROM SH. 1

ADAPTER TABLE

DIESEL START	2400V BUS UV AUX. RELAYS & LOCATION		
3060-	AUX. RELAYS	LOCATION	FUSE ID
1 & 2	127-1-X1/2, 127-7-X1/2 & 127-1-X2/2, 127-7-X2/2	J9400	FUZ/A0011-1
3 & 4	127-2-X1/2, 127-8-X1/2 & 127-2-X2/2, 127-8-X2/2	J9401	FUZ/A0012-1

5

REV	DATE	DESCRIPTION	BY	APP
5	11-18-93	ADDED FUSE ID TO ADAPTER TABLE PER DCR 93-1302	ITT	

SECTION HEAD APP. C.J. MCDONALD 4/18/83	DR. R.M. SATTERELLI	CK. T.D. VOGT 4/18/83
DIVISION HEAD APP. P. LOLICH 4/18/83	CONSUMERS POWER CO. PALISADES PLANT	SCHEMATIC DIAGRAM 2400V & 4160V BUS TRANSFER
DEPARTMENT HEAD APP. D.C. TARSI 4/18/83	E136-1A.DGN	NO. E-136 SHEET 1A REV. 5

Figure 8-20 FSAR Rev 17

9501040188-51

2400V. & 4160V. BUS TRANSFER RELAY TABLE

CONTACTS		383-11 (HFA)	383-12 (HFA)	383-21 (HFA)	383-22 (HFA)	383-23 (HFA)	383-11A (HFA)	383-12A (HFA)
		FUNCTION	FUNCTION	FUNCTION	FUNCTION	FUNCTION	FUNCTION	FUNCTION
1 2	1	TRIPS BUS 1C INCOMING BKR. 152-105 FROM SAFEGUARD BUS (DWG. E-131)	TRIPS BUS 1E INCOMING BKR. 152-302 FROM SAFEGUARD BUS (DWG. E-131)	TRIPS BUS 1A INCOMING BKR. 252-101 FROM STA. PWR. TRANS.#1-1 (DWG. E-131)	TRIPS BUS 1B INCOMING BKR. 252-201 FROM STA. PWR. TRANS.#1-1 (DWG. E-131)	TRIPS BUS 1F INCOMING BKR. 252-301 FROM STA. PWR. TRANS. 1-3 (DWG. E-728 SH 2)	TRIPS DIESEL GENERATOR INCOMING BKR 152-107 (E-139 SH.1)	TRIPS DIESEL GENERATOR INCOMING BKR 152-213 (E-139 SH.1)
3 4	2	ALARM (DWG. E-289)	TRIPS BUS 1D INCOMING BKR. 152-203 FROM SAFEGUARD BUS (DWG. E-131)	SPARE	SPARE	TRIPS BUS 1G INCOMING BKR. 252-401 FROM STA. PWR. TRANS. 1-3 (DWG. E-728 SH 2)	SPARE	SPARE
29 5 6	3	CLOSES BUS 1C INCOMING BKR. 152-106 FROM STARTUP TRANS.#1-2 * (DWG. E-132)	CLOSES BUS 1D INCOMING BKR. 152-202 FROM STARTUP TRANS.#1-2 * (DWG. E-132)	CLOSES BUS 1A INCOMING BKR. 252-102 FROM STARTUP TRANS.#1-1 * (DWG. E-132)	CLOSES BUS 1B INCOMING BKR. 252-202 FROM STARTUP TRANS.#1-1 (DWG. E-132)	CLOSES BUS 1F INCOMING BKR. 252-302 FROM STARTUP TRANS. 1-1 (DWG. E-728 SH 1)	SPARE	SPARE
7 8	4	SPARE	TRIPS SAFEGUARD BUS INCOMING BKR 152-402 FROM SPT-#1-2 (DWG. E-131)	ALARM (DWG. E-289)	ALARM (DWG. E-289)	CLOSES BUS 1G INCOMING BKR. 252-402 FROM STARTUP TRANS. 1-3 (DWG. E-728 SH 1)	SPARE	SPARE
9 10	5	PPC SOE NODE PT. ID. KS383_11_D (DWG. E-53 SH. 13)	CLOSES BUS 1E INCOMING BKR. 152-303 FROM STARTUP TRANS.#1-2 * (DWG. E-132)	TRIPS MAIN GENERATOR EXCITATION BKR.#341 (DWG. E-125)	TRIPS MAIN GENERATOR EXCITATION BKR.#341 (DWG. E-125)	ALARM (DWG. E-289 SH 2)	SPARE	SPARE
11 12	6	ALARM (DWG. E-289)	ALARM (DWG. E-289)	TRIPS MAIN GENERATOR VOLTAGE REGULATOR (DWG. E-126)	TRIPS MAIN GENERATOR VOLTAGE REGULATOR (DWG. E-126)	ALARM (DWG. E-289 SH 2)	SPARE	SPARE

DATALOGGER CHANNEL 1-174

DIESEL START RELAY TABLE

CONTACTS		306D-1	306D-2	306D-3	306D-4	386A (HGA)
		FUNCTION	FUNCTION	FUNCTION	FUNCTION	FUNCTION
1 3 7	1	START DIESEL #1-1 CKT. A (DWG. M12-98)	START DIESEL #1-1 CKT. B (DWG. M12-98)	START DIESEL #1-2 CKT. A (DWG. M12-105)	START DIESEL #1-2 CKT. B (DWG. M12-105)	ANNUNCIATOR GENERATOR TRIP (E-288 SH.1)
2 4 8	2	SPARE	SPARE	SPARE	SPARE	SPARE

UNDERVOLTAGE AUXILIARY RELAY TABLE

CONTACTS		127X-5 (WL)	127X-6 (WL)	227X-5 (WL)
		FUNCTION	FUNCTION	FUNCTION
2 1	1	BLOCKS AUTO CLOSING OF BUS 1C INCOMING BKR. FROM START UP TRANS.#1-2 (DWG. E-132)	BLOCKS AUTO CLOSING OF BUS 1D INCOMING BKR. FROM START UP TRANS.#1-2 (DWG. E-132)	BLOCKS AUTO CLOSING OF BUS 1A INCOMING BKR. FROM START UP TRANS.#1-1 (DWG. E-132)
4 3	2	TRIPS BUS 1C INCOMING BKR. FROM START UP TRANS.#1-2 (DWG. E-132)	TRIPS BUS 1D INCOMING BKR. FROM START UP TRANS.#1-2 (DWG. E-132)	TRIPS BUS 1A INCOMING BKR. FROM START UP TRANS.#1-1 (DWG. E-132)
6 5	3	SPARE	BLOCKS AUTO CLOSING OF BUS 1E INCOMING BKR. FROM START UP TRANS.#1-2 (DWG. E-132)	BLOCKS TRIPPING OF TRANSFER RELAY 383-21 ** (DWG. E-136 SH. 1)
8 7	4	SPARE	TRIPS BUS 1E INCOMING BKR. FROM START UP TRANS.#1-2 (DWG. E-132)	TRIPS BUS 1F INCOMING BKR. (252-302) FROM START UP TRANS.#1-1 (DWG. E-728 SH. 1)
10 9	5	SPARE	SPARE	BLOCKS CLOSING OF BUS 1F INCOMING BKR. (252-302) FROM START UP TRANS.#1-1 (DWG. E-728 SH. 1)
12 11	6	SPARE	SPARE	
14 13	7	SPARE	SPARE	SPARE
16 15	8	SPARE	SPARE	SPARE
18 17	9	SPARE	SPARE	SPARE
20 19	10	SPARE	SPARE	ALARM (DWG. E-289)

NOTES

* IF STANDBY POWER AVAILABLE
** IN CASE OF COASTDOWN
*** N.C. CONTACT

9707070093 -18

REV	DATE	DESCRIPTION	BY	CK	APP
29	0-17 96	REV'D. 383-12 PER DCR-96-772.	JGD	DMB	

SECTION HEAD DATE
APP. C.J. MCDONALD 4/22/83
DIVISION HEAD
APP. P. LOLICH 4/22/83
DEPARTMENT HEAD
APP. D.C. TARSI 4/22/83

DR. J. L. STERRETT
CK. T.D. VOGT 4/22/83

CONSUMERS POWER CO.
PALISADES PLANT
E136-2.DGN

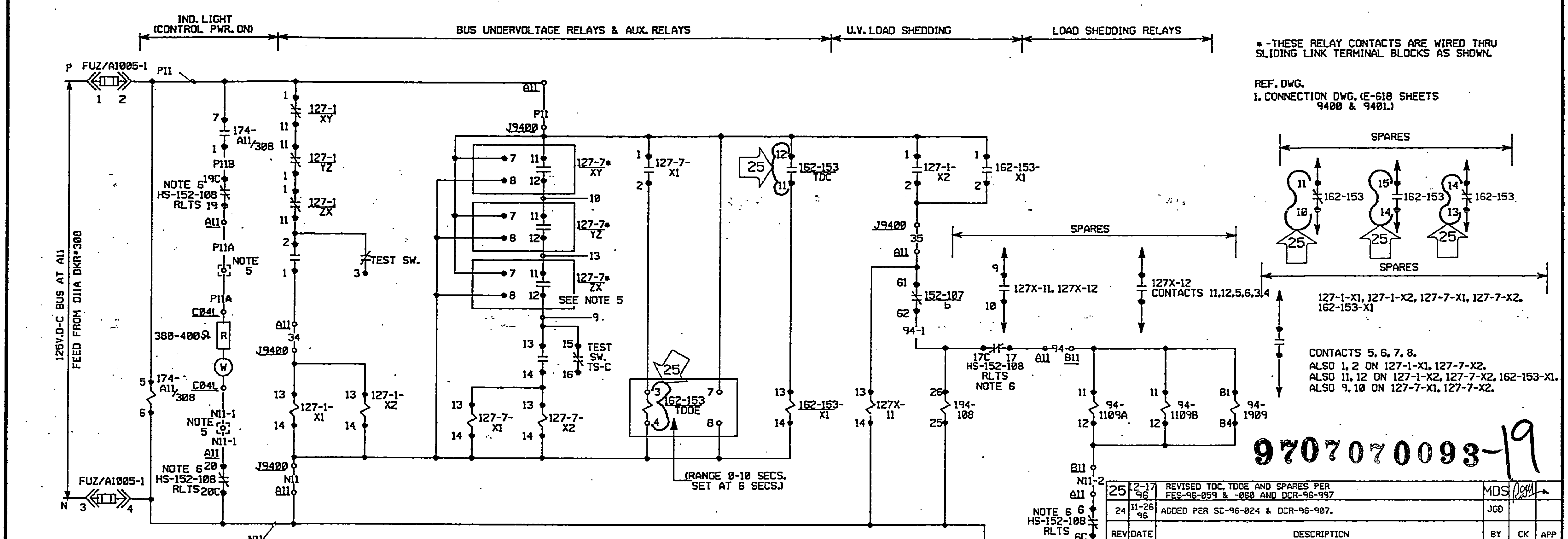
SCHEMATIC DIAGRAM
2400V. & 4160V. BUS
TRANSFER
NO. E-136 SHEET 2

REV. 29

Figure 8-21
Rev 19

ANSTEC APERTURE CARD

Also Available on Aperture Card



970707 0093-19

REV	DATE	DESCRIPTION	BY	CK	APP
25	12-17-96	REVISED TDC, TDOE AND SPARES PER FES-96-059 & -060 AND DCR-96-997	MDS		
24	11-26-96	ADDED PER SC-96-024 & DCR-96-907.	JGD		

NOTES:
1. FOR RELAY TABLES, SEE SHEETS 2, 2A, & 2B OF THIS DRAWING.
2. FOR ADAPTER TABLE, BLOCK DIAGRAM & ALARMS SEE SH. 1A OF THIS DRAWING.
3. FOR WIRING OF JUNCTION BOXES J9400 & J9401 SEE CONNECTION DWGS. E-618 SH. 9400 & 94-1.
4. RELAY CONTACTS CLOSE ON UNDERVOLTAGE CONDITION.
5. FOR SCHEME A1006 ONLY, TERM. BLOCK "E" IN 152-206.
6. CONTACTS SHOWN IN NORMALY CLOSED (REMOTE) CONTROL POSITION FOR SW. DEVELOPMENT SEE DWG. E-133 SH.1

CONSUMERS POWER COMPANY
PALISADES NUCLEAR PLANT
COVERT, MICHIGAN

E137-1.DGN

PREVIOUS RECORD ISSUE

MICHIGAN P.E. No.

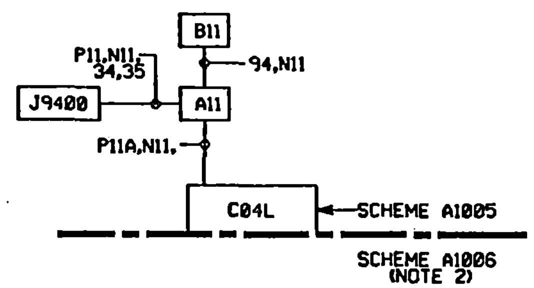
UFI No.

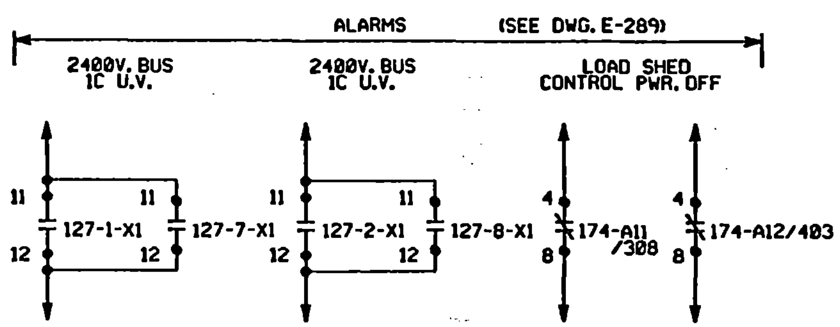
SCHEMATIC DIAGRAM
2400V. & 4160V BUS UNDER-VOLTAGE & LOAD SHEDDING

DRAWING NUMBER	SHT. NO.	REV
E-137	1	25

DESCRIPTION	SCHEME	BUS UV. RELAYS				BUS UV. AUX. RELAYS						LOAD SHEDDING RELAY				TIME DELAY RELAYS		CONTROL PWR. U.V.	IND LIGHT LOC	TEST SW. FOR BUS UV. RELAY		DIESEL GEN. BKR. "b" CONTACT		LOAD SHEDDING RELAY			D.C. SOURCE	FUSE NO.
		NO.	LOC.	NO.	LOC.	NO.	LOC.	NO.	LOC.	NO.	LOC.	NO.	LOC.	NO94-	LOC.	NO.	LOC.			NO.	LOC.	NO.	LOC.	BUS	NO.	LOC.		
2400V. BUS 1C 480V. BUS 11	A1005	127-1	A11	127-7	J9400	127X-11	A11	127-1-X1,X2	J9400	127-7-X1,X2 162-153X1	J9400	194-108	A11	1109A 1109B	B11	162-153	J9400	174-A11 /308	C04L	TS-C	J9400	152-107 b	A11	B-19	94-1909	B11	72-308	FUZ/A1005-1
2400V. BUS 1D 480V. BUS 12	A1006	127-2	A12	127-8	J9401	127X-21	A12	127-2-X1,X2	J9401	127-8-X1,X2 162-154X1	J9401	194-211	A12	1213A 1213B	B12	162-154	J9401	174-A12 /403	C04R	TS-D	J9401	152-213 b	A12	B-20	94-2013	B12	72-403	FUZ/A1006-1

8





SI
APERTURE
CARD

Also Available On
Aperture Card

NOTES:
1. FOR RELAY TABLES, SEE SHEETS 2,2A,2B OF THIS DWG.
2. SEPARATE RACEWAY SYSTEMS.

REV	DATE	DESCRIPTION	FC* SC*	BY	CK	APP
8	12-10 91	FUSE EDB NO. ADDED PER ENG. REVIEW DCR 950-91-898(MAL)		PRC	JGD	DMB

CONSUMERS POWER COMPANY
PALISADES NUCLEAR PLANT
COVERT, MICHIGAN

E137-1A.008

PREVIOUS RECORD ISSUE

MICHIGAN P.E. No.

UFI No.

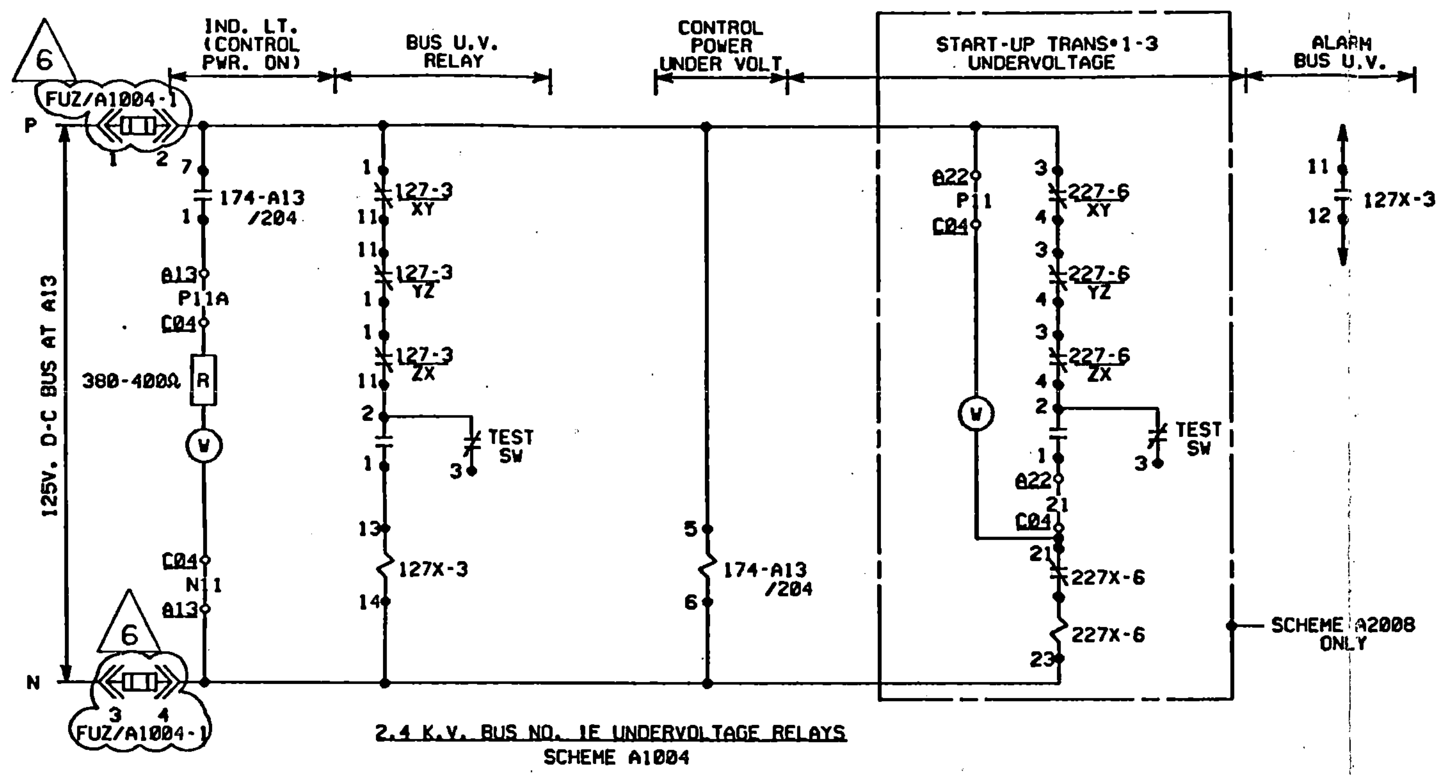
SCHEMATIC DIAGRAM
2400V & 4160V. BUS UNDER-
VOLTAGE & LOAD SHEDDING

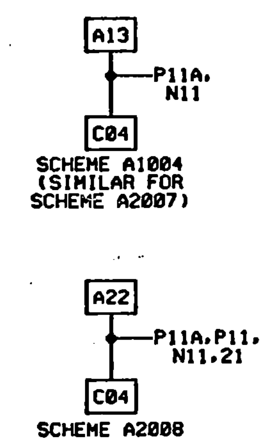
DRAWING NUMBER	SHT. NO.	REV
E137	1A	8

Figure No 8-23 FSAR Rev 15

9305130068 -55

NOTES:
1. FOR RELAY TABLES, SEE SHEET 2 OF THIS DWG.





SI APERTURE CARD

Also Available On Aperture Card

ADAPTER TABLE

DESCRIPTION	SCHEME	BUS U.V. RELAYS		INTLK. RELAY		CONTROL PWR. UV		START-UP TRANS.•1-1 UNDERVOLTAGE RELAYS		BUS U.V. AUX. RELAYS		FUSE NO.
		NO.	LOC.	NO.	LOC.	RELAY	IND. LOC	NO.	LOC	NO.	LOC.	
2400V. BUS 1E	A1004	127-3	A-13	——	——	174-A13 /204	C04	——	—	127X-3	A13	FUZ/A1004-1
4160V. BUS 1A	A2007	227-1	A-21	——	——	——	CD4L	——	—	227X-1	A21	FUZ/A2007-1
4160V. BUS 1B	A2008	227-2	A-22	227X-6	A22	274-A21	C04	227-6/XY;/YZ;/ZX	A22	227X-2	A22	FUZ/A2008-1

	REV	DATE	DESCRIPTION	FC• SC•	BY	CK	APP
	6	12-10 91	REMOVED FUSE SIZE PER ENG. REVIEW AND ADDED FDB NO. PER DCR 950-91-0981(MAL)		PRC	JOD	DMB

CONSUMERS POWER COMPANY
PALISADES NUCLEAR PLANT
COVERT, MICHIGAN
E137-1B.006

PREVIOUS RECORD ISSUE

MICHIGAN P.E. NO.

UFI NO.

SCHEMATIC DIAGRAM
2400V. & 4160V. BUS UNDER-VOLTAGE & LOAD SHEDDING

DRAWING NUMBER	SHT. NO.	REV
E137	1B	6

Figure No 8-24 FSAR Rev 15

9305130068 -56

ANSTEC
APERTURE
CARD

Also Available on
Aperture Card

UNDERVOLTAGE RELAY TABLE

CONTACTS	2400 VOLT. BUS					
	BUS #1C 127X-11(HFA)		BUS #1C 127X-12(HFA)		BUS #1D 127X-21(HFA)	
	FUNCTION	REF. DWG.	FUNCTION	REF. DWG.	FUNCTION	REF. DWG.
1-2	SPARE	—	SPARE	—	RESET SERVICE WATER PUMP P7A (152-204) (STANDBY)	E-154 SH.1
3-4	RESET SERVICE WATER PUMP P7B (152-103) (STANDBY)	E-154 SH.1	SPARE	—	RESET SERVICE WATER PUMP P7C (152-205) (STANDBY)	E-154 SH.2
5-6	SPARE	—	SPARE	—	RESET LOW PRESS SAFETY INJ PUMP P67A (152-206) (STANDBY)	E-247
7-8	RESET COMP COOLING PUMP P52A (152-109) (STANDBY)	E-259 SH.1	SPARE	—	SPARE	—
9-10	RESET COMP COOLING PUMP P52C (152-116) (STANDBY)	E-259 SH.1	SPARE	—	RESET COMP COOLING PUMP P52B (152-208) (STANDBY)	E-259 SH.1
11-12	RESET LOW PRESS SAFETY INJECTION PUMP P67B (152-111) (STANDBY)	E-248	SPARE	—	SPARE	—

CONTACTS	2400 VOLT. BUS				4160 VOLT. BUS			
	BUS #1D 127X-22(HFA)		BUS #1E 127X-3(HFA)		BUS #1A 227X-1(HFA)		BUS #1B 227X-2(HFA)	
	FUNCTION	REF. DWG.	FUNCTION	REF. DWG.	FUNCTION	REF. DWG.	FUNCTION	REF. DWG.
1-2	SPARE	—	TRIPS HTR. DRAIN PP. P10A 152-307	E-180				
3-4	SPARE	—	TRIPS HTR. DRAIN PP. P10B 152-308	E-180				
5-6	SPARE	—	TRIPS DILUTION WTR. PP. P40B 152-309	E-151	TRIPS PRI. COOLANT PP. P50A 252-103	E-183	TRIPS PRI. COOLANT PP. P50B 252-203	E-183
7-8	SPARE	—	TRIPS STATION PWR TRANS. #90 & 91 152-310	E-134	TRIPS PRI. COOLANT PP. P50C 252-104	E-183	TRIPS PRI. COOLANT PP. P50D 252-204	E-183
9-10	SPARE	—	TRIPS SUPPORT BLDG. & WHSE X106 152-311	E-134	TRIPS CONDENSATE PP. P2A 252-105	E-182	TRIPS CONDENSATE PP. P2B 252-205	E-182
11-12	SPARE	—	ALARM	E-289	ALARM	E-289	ALARM	E-289

NOTE

LOAD SHEDDING RELAY TABLE CONTINUED ON SHEET 2A.

LOAD SHEDDING RELAY TABLE

CONTACTS	480 VOLT. L.C. BUS			
	BUS #11 94-1109A(WEST)		BUS #12 94-1213A(WEST)	
	FUNCTION	REF. DWG.	FUNCTION	REF. DWG.
1-7	BLOCKS CLOSE OF INST. AIR COMP. C2A 52-1106	E-167	BLOCKS CLOSE OF INST. AIR COMP. C2B 52-1207	E-167
2-5	TRIPS 480V. MCC #7 BRR 52-1103	E-134	TRIPS CONTMT. COOLER RECIRC. FAN V2A 52-1209	E-216
3-6	TRIPS INST. AIR COMP C2A 52-1106	E-166	TRIPS CONTMT. COOLER RECIRC. FAN V3A 52-1210	E-216
4-8	TRIPS INST. AIR COMP. C2C 52-1107	E-166	TRIPS CHARG. PP. P55B 52-1206	E-257
9-13	TRIPS CONTMT. COOLER RECIRC. FAN V4A 52-1108	E-217	TRIPS CHARG. PP. P55A 52-1205	E-257
10-16	TRIPS CHARG. PP. P55C 52-1105	E-257	TRIPS 480V. MCC #8 BKR. 52-1201	E-134
CONTACTS	BUS #11 94-1109B(WEST)		BUS #12 94-1213B(WEST)	
	FUNCTION	REF. DWG.	FUNCTION	REF. DWG.
1-7	TRIPS TIE BKR 52-1118	E-135	TRIPS TIE BKR. 52-1217	E-135
2-5	BLOCK CLOSE OF INST. AIR COMP. C2C 52-1107	E-167	TRIPS CONTMT. COOLER RECIRC. FAN V1A 52-1208	E-216
3-6			TRIPS INST. AIR COMP. C2B 52-1207	E-166
4-8				
9-13	TRIPS MAIN EXH. FAN V6B 52-1111	E-225	TRIPS MAIN EXH. FAN V6A 52-1215	E-225
10-16			SPARE	

2400 VOLT BUS				CONTACTS
BUS #1C 194-108(CLARK)		BUS #1D 194-211(CLARK)		
FUNCTION	REF. DWG.	FUNCTION	REF. DWG.	
TRIPS DILUTION WTR. PP. P40A 152-102	E-151			1-2
TRIPS SERV. WTR. PP. P7B 152-103	E-154			3-4
TRIPS AUX. FEED PP. P8A 152-104	E-196 SH. 1	TRIPS SERV. WTR. PP. P7A 152-204	E-154	5-6
TRIPS COMP. COOLG. PP P52A 152-109	E-259	TRIPS SERV. WTR. PP. P7C 152-205	E-154	7-8
18 TRIPS SWYD. AUX. PWR. #2 152-118	E-133	TRIPS L.P. SAFTY INJ. PP. P67A 152-206	E-247	9-10
TRIPS L.P. SAFTY INJ. PP. P67B 152-111	E-248	TRIPS COMP. COOL'G. PP52B 152-208	E-259	11-12
TRIPS COMP. COOLG. PP. P52C 152-116	E-259	TRIPS PRESS. HTR. TRANS. #16 152-211	E-253	13-14
TRIPS H.P. SAFETY INJ. PP. P66B 152-113	E-249	TRIPS H.P. SAFETY INJ. PP. P66A 152-207	E-249	15-16
TRIPS CONTMT. SPRAY PP. P54B 152-112	E-251	TRIPS AUX. FEED PP. P8C 152-209	E-196 SH. 8	17-18
TRIPS CONTMT. SPRAY P54C 152-114	E-251	TRIPS CONTMT. SPRAY PP. P54A 152-210	E-251	19-20
CLOSES DIESEL GEN. 1-1 BKR. 152-107	E-209	CLOSES DIESEL GEN. 1-2 BKR. 152-213	E-209	21-22
18 TRIPS STATION PWR TRANS# 13 152-108	E-133			23-24

UNDERVOLTAGE AUX. RELAY TABLE

CONTACTS		227X-6(WL) FUNCTION	REF. DWG.
1	1-2	BLOCKS AUTO CLOSING OF BUS 1B INCOMING BKR. FROM START UP TRANS. #1-1	E-132
2	3-4	TRIPS BUS 1B INCOMING BKR. FROM START UP TRANS. #1-1	E-132
3	5-6	BLOCKS TRIPPING OF TRANSFER RELAY 383-22 **	E-136 (SH. 1)
4	7-8	TRIPS BUS 1G INCOMING BKR.(252-402)FROM START-UP TRANSF. #1-3	E-728 (SH.1)
5	9-10	BLOCKS CLOSING OF BUS 1G INCOMING BKR. (252-402) FROM START-UP TRANSF. #1-3	E-728 (SH.1)
6	11-12		
7	13-14	SPARE	
8	15-16	SPARE	
9	17-18	SPARE	
10	19-20	ALARM	E-289

** IN CASE OF COASTDOWN

9707070093-20

						SECTION HEAD APP. C.J. MCDONALD DATE 4/18/83	DR. J. L. STERRETT	CK. T.D. VOGT 4/18/83
18	11-27 96	REV'D. BUS #1C 194-108 PER SC-96-024 & DCR-96-907.	JGD	JC		DIVISION HEAD APP. P. LOLICH 4/18/83	CONSUMERS POWER CO. PALISADES PLANT E137-2.DGN	SCHEMATIC DIAGRAM 2400V. & 4160V. BUS UNDER-VOLTAGE LOAD SHEDDING
REV	DATE	DESCRIPTION	BY	CK	APP	DEPARTMENT HEAD APP. D.C. TARSI 4/18/83		NO. E-137 SHEET 2

REV. 18

Figure 8-25
Rev 19

ANSTEC APERTURE CARD

Also Available on Aperture Card

UNDERVOLTAGE AUX. RELAY TABLES

CONTACTS	2.4 K.V. BUS 1C									
	127-1-X1 (HFA) FUNCTION	REF. DWG.	127-1-X2 (HFA) FUNCTION	REF. DWG.	127-7-X1 (HFA) FUNCTION	REF. DWG.	127-7-X2 (HFA) FUNCTION	REF. DWG.	162-153X1 (HFA) FUNCTION	REF. DWG.
1 (1–2)	SPARE		U.V. LOADSHEDDING	E-137	BUS U.V. AUX. RELAY	E-137	SPARE		U.V. LOADSHEDDING	E-137
2 (3–4)	STARTS DIESEL GEN. 1-1 CKT. A	E-136	STARTS DIESEL GEN. 1-1 CKT. B 6	E-136	STARTS DIESEL GEN. 1-1 CKT. A	E-136	STARTS DIESEL GEN. 1-1 CKT. B	E-136	TRIPS BUS #1C ST-UP XFMR INCOMING BKR. 152-106	E-132
3 (5–6)	SPARE		BLOCK SIS-X INIT. (SEE NOTE 2)	E-209	SPARE		SPARE		BLOCK SIS-X INIT. (SEE NOTE 2)	E-209
4 (7–8)	SPARE		SPARE		SPARE		SPARE 6		SPARE	
5 (9–10)	TRIPS BUS #1C ST-UP XFMR INCOMING BKR. 152-106	E-132	TRIPS BUS #1C INCOMING BKR. 152-105	E-131	SPARE		SPARE		TRIPS BUS #1C INCOMING BKR. 152-105	E-131
6 (11–12)	ALARM	E-289	SPARE		ALARM	E-289	SPARE		SPARE	

CONTACTS	2.4 K.V. BUS 1D									
	127-2-X1 (HFA) FUNCTION	REF. DWG.	127-2-X2 (HFA) FUNCTION	REF. DWG.	127-8-X1 (HFA) FUNCTION	REF. DWG.	127-8-X2 (HFA) FUNCTION	REF. DWG.	162-154X1 (HFA) FUNCTION	REF. DWG.
1 (1–2)	SPARE		U.V. LOADSHEDDING	E-137	BUS U.V. AUX. RELAY	E-137	SPARE		U.V. LOADSHEDDING	E-137
2 (3–4)	STARTS DIESEL GEN. 1-2 CKT. A	E-132	STARTS DIESEL GEN. 1-2 CKT. B 6	E-136	STARTS DIESEL GEN. 1-2 CKT. A	E-136	STARTS DIESEL GEN. 1-2 CKT. B	E-136	TRIPS BUS #1D ST-UP XFMR INCOMING BKR. 152-202	E-132
3 (5–6)	SPARE		BLOCK SIS-X INIT. (SEE NOTE 2)	E-209	SPARE		SPARE		BLOCK SIS-X INIT. (SEE NOTE 2)	E-209
4 (7–8)	SPARE		SPARE		SPARE		SPARE 6		SPARE	
5 (9–10)	TRIPS BUS #1D ST-UP XFMR INCOMING BKR. 152-202	E-132	TRIPS BUS #1D INCOMING BKR. 152-203	E-131	SPARE		SPARE		TRIPS BUS #1D INCOMING BKR. 152-203	E-131
6 (11–12)	ALARM	E-289	SPARE		ALARM	E-289	SPARE		SPARE	

LOAD SHEDDING RELAY TABLE (CONT. FROM SH. 2)

CONTACTS	480 VOLT L.C. BUS			
	BUS 19 94-1909 FUNCTION	REF. DWG.	BUS 20 94-2013 (NOTE 1) FUNCTION	REF. DWG.
1 (T1–M1)	BLOCKS AIR HANDLING UNIT V-95	E-270 SH.13	BLOCKS AIR HANDLING UNIT V-96	E-270 SH.13
2 (T2–M2, R2)	SPARE		SPARE	
3 (T3–M3)	BLOCKS CONDENSING UNIT VC-11 START	E-270 SH.13	BLOCKS CONDENSING UNIT VC-10 START	E-270 SH.13
4 (T4–M4, R4)	SPARE		SPARE	

NOTES

1. LOAD SHEDDING RELAYS 94-1909 AND 94-2013 ARE AGASTAT SERIES GP POWER RELAYS.
2. FIELD CONVERT CONTACT (5-6) OF DEVICE 127-1-X2, 127-2-X2, 162-153-X1 AND 162-154-X1 FROM NORMALLY OPEN TO NORMALLY CLOSED AND READJUST HFA RELAY AS REQUIRED. (SEE DWG E-618, SH. 9400 & 9401)

9707070093

REV	DATE	DESCRIPTION	BY	CK	APP
5	1-17-90	ADDED CONN. TO CONTACT (5-6) OF 127X-1-X2, -2-X2, 162-153-X1 & -154-X1 RELAY CONTACT CHART PER GWO 8303 FC 888	MJG	GCG	RAL
6	6-13-95	REV'D. 127-1-X2, 127-2-X2, 127-7-X2 & 127-8-X2 PER FC-940 & DCR-95-466.	JGD		

SECTION HEAD APP. C.J. MCDONALD 4/18/83
DIVISION HEAD APP. P. LOLICH 4/18/83
DEPARTMENT HEAD APP. D.C. TARSI 4/18/83

DR. J. L. STERRETT | CK. T.D. VOGT 4/18/83

CONSUMERS POWER CO.
PALISADES PLANT
E137-2A.DGN

SCHEMATIC DIAGRAM 2400V. & 4160V. UNDER-VOLTAGE & LOAD SHEDDING

NO. E-137 SHEET 2A

Figure 8-26
Rev 19

2.4 K.V. BUS U.V. RELAY TABLE

CONTACTS	127-7 XY (ITE-27N) FUNCTION	REF. DWG.	127-7 YZ (ITE-27N) FUNCTION	REF. DWG.	127-7 ZX (ITE-27N) FUNCTION	REF. DWG.	127-8 XY (ITE-27N) FUNCTION	REF. DWG.	127-8 YZ (ITE-27N) FUNCTION	REF. DWG.	127-8 ZX (ITE-27N) FUNCTION	REF. DWG.
11 12	INDICATES U.V. ON BUS *1C PHASES X TO Y AFTER 1/2 SEC.	E-137	INDICATES U.V. ON BUS *1C PHASES Y TO Z AFTER 1/2 SEC.	E-137	INDICATES U.V. ON BUS *1C PHASES Z TO X AFTER 1/2 SEC.	E-137	INDICATES U.V. ON BUS *1D PHASES X TO Y AFTER 1/2 SEC.	E-137	INDICATES U.V. ON BUS *1D PHASES Y TO Z AFTER 1/2 SEC.	E-137	INDICATES U.V. ON BUS *1D PHASES Z TO X AFTER 1/2 SEC.	E-137
10 11	SPARE		SPARE		SPARE		SPARE		SPARE		SPARE	
13 14	SPARE		SPARE		SPARE		SPARE		SPARE		SPARE	
14 15	SPARE		SPARE		SPARE		SPARE		SPARE		SPARE	

2.4 K.V. BUS U.V. TIME DELAY RELAY TABLE

CONTACTS	162-153 (ABB-62T) 5 FUNCTION	REF. DWG.	162-154 (ABB-62T) 5 FUNCTION	REF. DWG.
10 11 12	TIME DELAY OF 6 SEC. TO AUX. RELAY 162-153X1	E-137	TIME DELAY OF 6 SEC. TO AUX. RELAY 162-154X1	E-137
13 14 15	SPARE		SPARE	

5

CONTACT DEVELOPMENT OF TEST SWITCHES FOR UNDERVOLTAGE RELAYS 127-7, 127-8

CONTACTS	TEST SW TS-CW FT-1) FUNCTION	REF. DWG.	TEST SW TS-DW FT-1) FUNCTION	REF. DWG.
A 1 2	ISOLATE UNDERVOLTAGE RELAY 127-7/XY	E-11	ISOLATE UNDERVOLTAGE RELAY 127-8/XY	E-11
B 3 4	ISOLATE UNDERVOLTAGE RELAY 127-7/YZ	E-11	ISOLATE UNDERVOLTAGE RELAY 127-8/YZ	E-11
C 5 6	ISOLATE UNDERVOLTAGE RELAY 127-7/ZX	E-11	ISOLATE UNDERVOLTAGE RELAY 127-8/ZX	E-11
D 7 8	SPARE		SPARE	
E NO SW.	—	—	—	—
F NO SW.	—	—	—	—
G 13 14	ISOLATE UNDERVOLTAGE AUX. RELAYS 127-7-X1,127-7-X2	E-137	ISOLATE UNDERVOLTAGE AUX. RELAYS 127-8-X1,127-8-X2	E-137
H 15 16	TEST & CALIBRATION OF U.V. RELAY 127-7	E-137	TEST & CALIBRATION OF U.V. RELAY 127-7	E-137
I 17 18	SPARE		SPARE	
J NO SW.	—	—	—	—

9707070093 -22

REV	DATE	DESCRIPTION	BY	CK	APP
5	12-17-96	REVISED SPARE CONTACTS FOR TIME DELAY PER FES-96-059 & -060 AND DCR-96-997	MDS		
4	8-2-86	CHANGED RELAY NO. TO REFLECT ACTUAL EQUIP. NO. 162-154.	DEB	JGD	JH

SECTION HEAD APP. C.J. MCDONALD DATE 4/18/83
DIVISION HEAD APP. P. LOLICH 4/18/83
DEPARTMENT HEAD APP. D.C. TARSI 4/18/83

DR. J. L. STERRETT
CK. T.D. VOGT 4/18/83

CONSUMERS POWER CO.
PALISADES PLANT
E137-2B.DGN

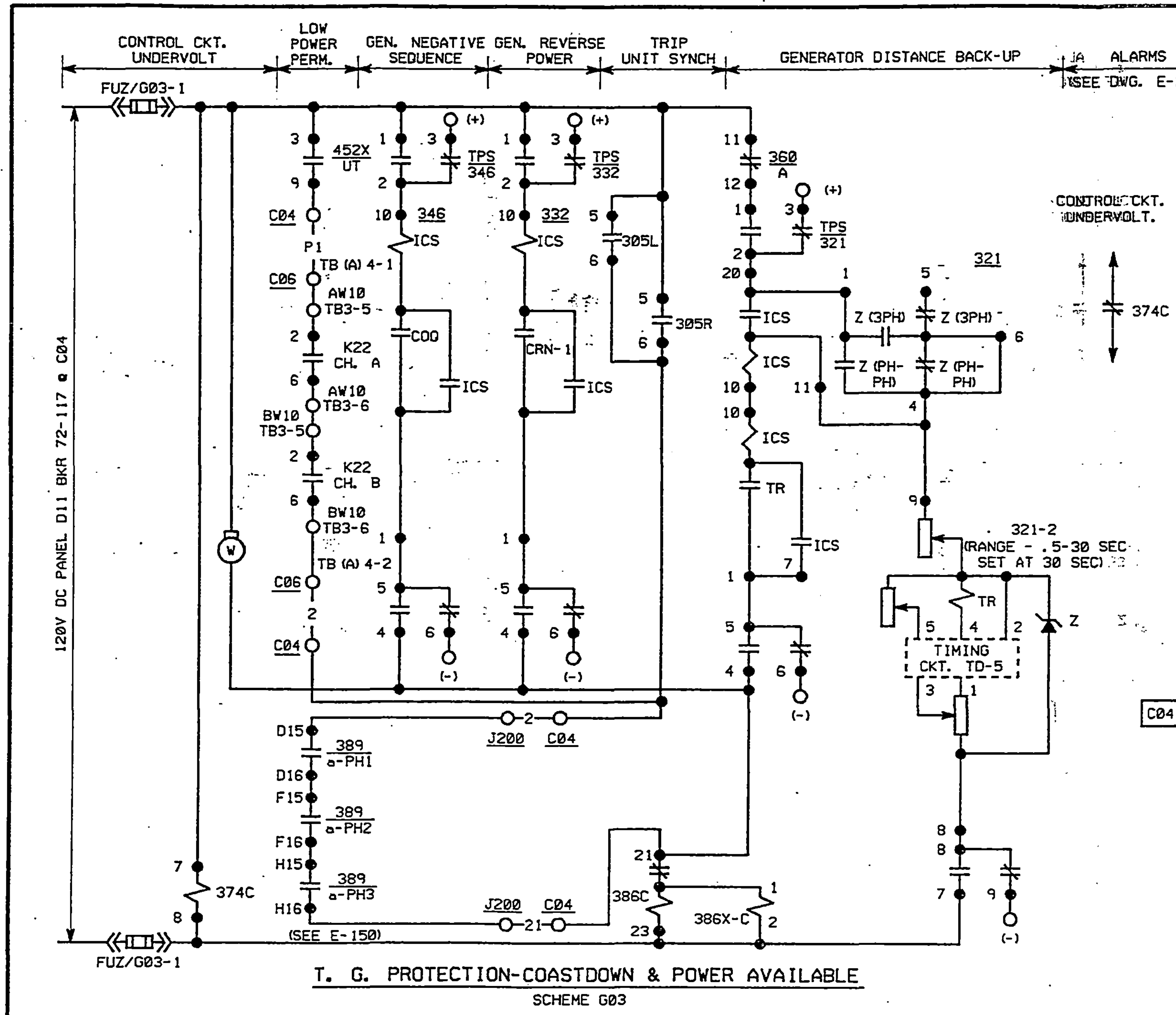
SCHEMATIC DIAGRAM
2400V. & 4160V. BUS UNDER-VOLTAGE & LOAD SHEDDING

NO. E-137 SHEET 2B

REV. 5

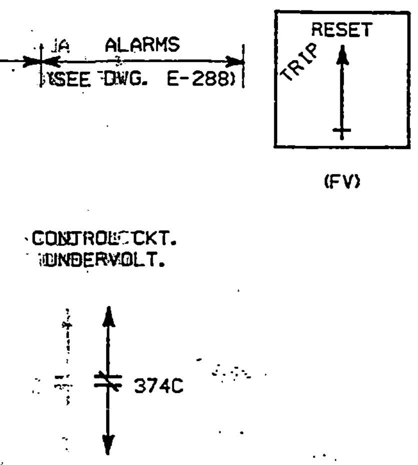
Figure 8-27
Rev 19

DWG. NO.



T. G. PROTECTION-COASTDOWN & POWER AVAILABLE

SCHEME G03

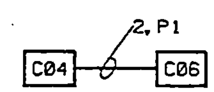


CONTACTS (HANDLE END)	POSITION RESET	POSITION TRIP	386P & 386B CONTACT FUNCTION	DWG. NO.	386C CONTACT FUNCTION	DWG. NO.
1 ●─┤├─● 2		X	TRIPS GEN. BKR. 25F7	E-116	TRIPS GEN. BKR. 25F7	E-116
3 ●─┤├─● 4		X	TRIPS GEN. BKR. 25H9	E-116	TRIPS GEN. BKR. 25H9	E-116
5 ●─┤├─● 6		X	BACK UP TRANSFER TRIP	E-116	BACK UP TRANSFER TRIP	E-116
7 ●─┤├─● 8		X	ENERGIZES 383-11A	E-136	ENERGIZES 383-11A	E-136
9 ●─┤├─● 10		X	ENERGIZES 383-11	E-136	ENERGIZES 383-11	E-136
11 ●─┤├─● 12		X	ENERGIZES 383-12, 383-23	E-136	ENERGIZES 383-12, 383-23	E-136
13 ●─┤├─● 14		X	ENERGIZES 383-21	E-136	ENERGIZES 383-21	E-136
15 ●─┤├─● 16		X	ENERGIZES 383-22	E-136	ENERGIZES 383-22	E-136
17 ●─┤├─● 18		X	TURBINE TRIP SOLENOID	E-121	TURBINE TRIP SOLENOID	E-121
19 ●─┤├─● 20		X	ENERGIZES 383-12A	E-136	ENERGIZES 383-12A	E-136

GENERATOR LOCKOUT RELAYS
TYPE 'WL' MANUALLY RESET

CONTACT	386X-P (AVR) FUNCTION	386X-P (AVR) REF	386X-B (AVR) FUNCTION	386X-B (AVR) REF	386X-C (AVR) FUNCTION	386X-C (AVR) REF
3 ●─┤├─● 4	SPARE	—	SPARE	—	SPARE	—
5 ●─┤├─● 6	PPC SOE NODE PT. ID. KS386X_P_D	E-53, SH 4	PPC SOE NODE PT. ID. KS386X_B_D	E-53, SH 7	PPC SOE NODE PT. ID. KS386X_C_D	E-53, SH 13

19 19 19



NOTES:

1 TEST SWITCH CONTACTS SHOWN IN TEST POSTION.

2 K22 CONTACTS CLOSE AT MORE THAN 15% POWER.

NO	DATE	REVISIONS	BY	CK'D	APP
19	7-28 95	REV. DATALOGGER TO PPC SOE NODE PER FC-933-05 & DCR-95-518	MDS		lm

SCALE NONE DESIGNED DRAWN J.L. STERRETT

PALISADES PLANT
CONSUMERS POWER COMPANY

SCHEMATIC DIAGRAM
TURBINE GENERATOR
PROTECTION-COAST DOWN

SITE ID.	DRAWING NO.	REV.
0950	E-120	19

E120.DGN

Figure 8-28
Rev 19

9707070093-23

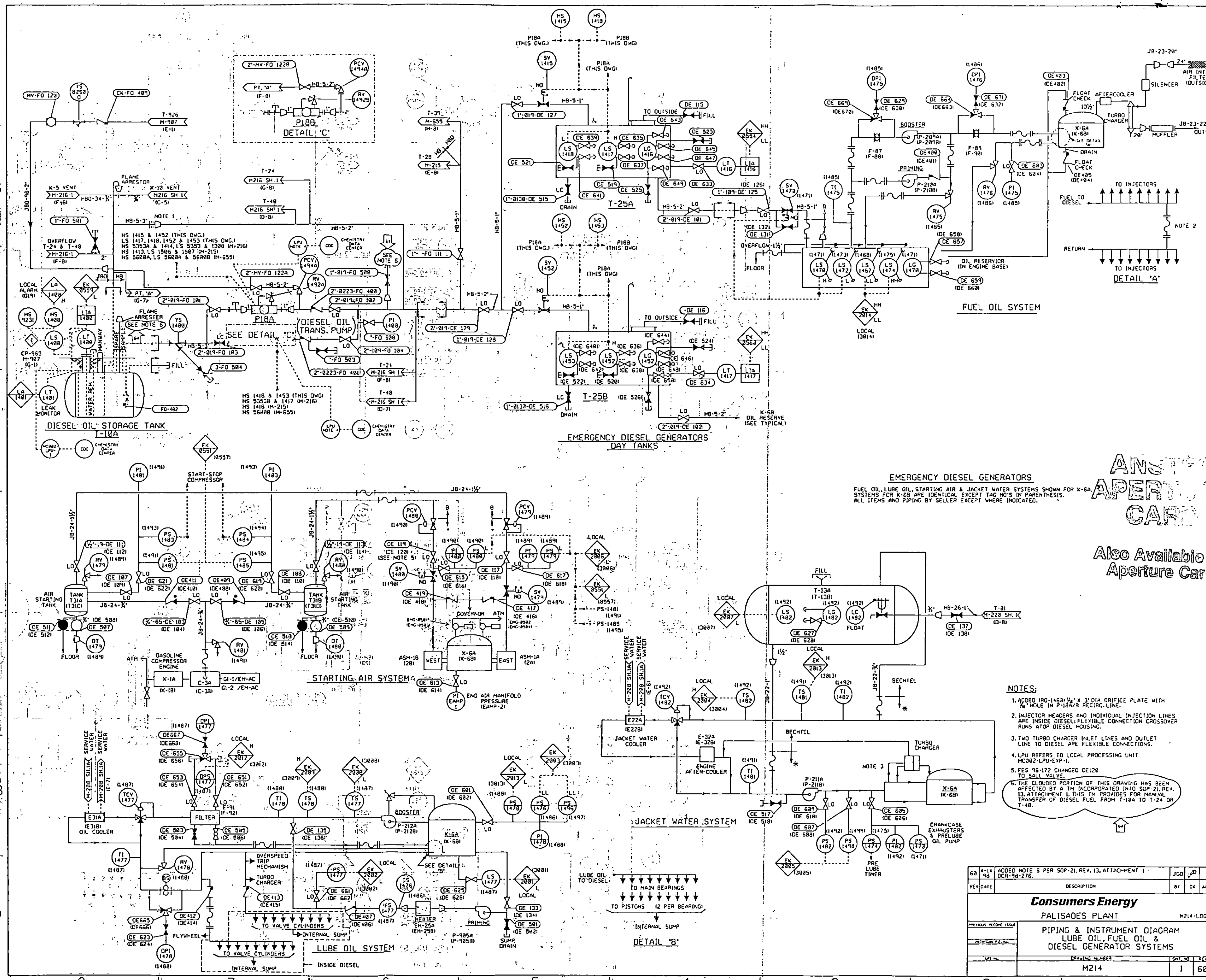
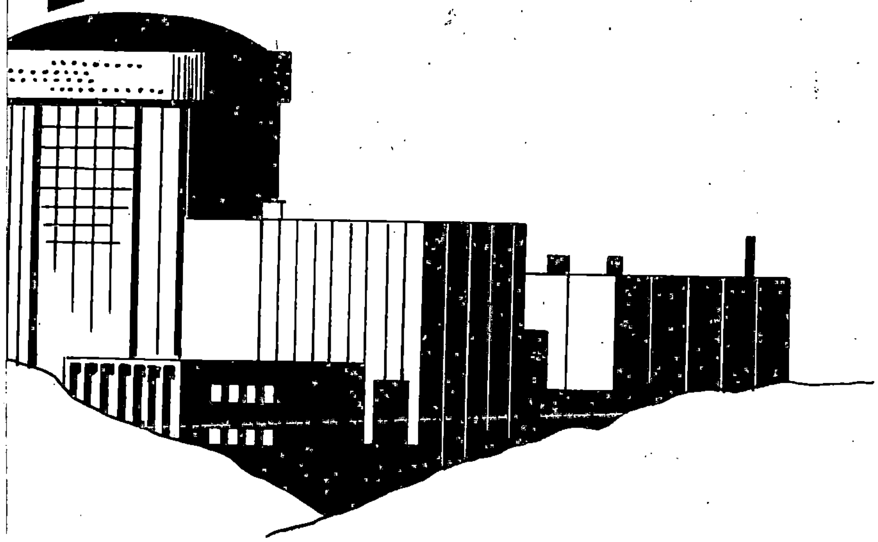


FIG 8-29
FSAR REV 20

9807090043 -32



LISADES NUCLEAR PLANT

final safety analysis report

UPDATE

VOLUME 8

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CHAPTER 9

AUXILIARY SYSTEMS

9.1 SERVICE WATER SYSTEM

9.1.1 DESIGN BASIS

The Service Water System is designed to supply lake water as the cooling medium (Ultimate Heat Sink) for removal of waste heat from the nuclear plant and steam plant auxiliary systems during normal, shutdown or emergency conditions. Separate service waterlines serve the Plant critical and noncritical systems. The critical system is divided into two independent trains, which are CP Co Design Class 1. The service water pumps are located in the CP Co Design Class 1 portion of the intake structure. The service water piping within the containment building is missile protected.

9.1.2 SYSTEM DESCRIPTION AND OPERATION

9.1.2.1 System Description

Three half-capacity electric motor-driven pumps draw screened and intermittently chlorinated Lake Michigan water from the intake structure (Figure 9-1). Two motors are connected to one 2.4 kV bus and the third motor is connected to a separate 2.4 kV bus. Each pump can be started or stopped remotely from the main control room or locally at the switchgear.

Each service water pump discharges through a simplex strainer into a common header. Each pump can be isolated from this header by a hand-operated valve in the pump discharge.

The common header of the service water pumps has a full-capacity takeoff at each end which supplies critical Plant systems. A third takeoff at one end of the common header supplies the noncritical auxiliary systems. The common header contains sectionalizing valves which can be closed from the main control room if isolation of a portion of the service water supply system is required.

The two critical service waterlines run underground by different paths from the intake structure to the auxiliary building. The two lines are joined in the auxiliary building by a double-valved crosstie. Each line has an isolation valve immediately upstream of the crosstie. These four valves permit the isolation of either critical line. Each valve is a fail open, diaphragm type and can be actuated remotely from the main control room or by a local handwheel.

Downstream of the crosstie, each line supplies cooling water to one set of the redundant components including an emergency diesel generator heat exchanger, a control room air-conditioning unit, an air compressor after-cooler and an engineered safeguards room cooler. In addition, Train A supplies cooling water to the component cooling water heat exchangers while Train B supplies cooling water to the containment air coolers.

The service water discharge from equipment carrying potentially contaminated fluid is continuously monitored for radioactivity, enabling radioactive leakage into the service water to be detected before service water is released to the lake.

Provisions are made to connect the fire system to the Service Water System as a partial backup.

In the event that water is not available to the intake structure due to a collapse of the intake crib or a similar type failure, approximately 17,000 gpm flow to the intake structure may be supplied from the mixing basin or makeup basin via the warm water recirculation pump P-5.

Systems supplied by the Service Water System and cooling water flow requirements for equipment supplied by the Service Water System are tabulated in Table 9-1.

In response to Generic Letter 89-13, a program was established to address the issue of biofouling of the service water system. Elements of this program include periodic inspections of the service water pump intake bay and service water system components, chlorination of service water, periodic flushing of infrequently used cooling loops and periodic verification of service water system flow rates and heat exchanger heat transfer capabilities (see Reference 6).

9.1.2.2 Component Description

Design ratings of components in this system are given in Table 9-2.

9.1.2.3 System Operation

1. Normal Operation

Two service water pumps are required to furnish the normal cooling water demand; the third pump will normally be on standby. Two pressure switches are provided in the discharge of each pump connecting to the starting circuits of the remaining two pumps. If the service water pressure falls below a preset value, one of the switches initiates automatic starting. The auto-start feature is automatically reset on bus undervoltage to prevent cycling the pump breaker onto a dead bus.

2. Shutdown Operation

Service water flow requirements during shutdown cooling will remain essentially the same as for normal operation. Both component cooling water heat exchangers are used to cool the primary coolant from 300°F to the refueling temperature. Service water flow is maintained to equipment on an as-needed basis (eg, VRS, FWP, containment air coolers.)

3. Post-DBA Operation

Either one or two service water pumps are required to provide cooling in the event of a DBA, depending on the accident events. If Plant offsite power sources are lost, all pump motors are automatically supplied with power from the emergency diesel generators with one pump on Diesel 1-1 and two pumps on Diesel 1-2. Cooling water demands can be met with one pump if only Diesel 1-1 is operating provided service water to containment is isolated, and with two pumps if only Diesel 1-2 is operating.

Service water through most noncritical systems is terminated by automatic closure of the noncritical header shutoff valve on a Safety Injection Signal (SIS), thus ensuring that all available service water is routed to the critical systems. The automatic shutoff valve can also be actuated remotely from the main control room or by a local handwheel. This does not isolate all non-critical loads; air compressors are not isolated because their availability for use is highly desirable, and their service water requirements are not significant (see Table 9-1).

On loss of instrument air, main valves to the CCW heat exchangers fail open, while the bypass valves failed closed to conserve service water. Hard stops are placed on these valves to prevent them from going full open and starving other critical services. Service water is continued to all critical systems' heat exchangers. Engineered safeguards pumps seal cooling is normally provided from the Component Cooling System; however, if that system is not operable, the Service Water System can be aligned for seal cooling.

The Service Water System is tested periodically to determine the flows to equipment on the critical headers. The system is aligned as it would be following a DBA coincident with a loss of offsite power, loss of a diesel generator, and a loss of instrument air.

The maximum allowed service water temperature was determined by analysis to be 81.5 °F. The service water temperature has exceeded 81.5°F on one occasion since 1982 for a very short duration (less than 4 hours). The NRC issued an SER in 1987 (Ref. 8) that recognized the possibility of elevated SW temperatures, but concluded that the likelihood for the incident of concern occurring was negligibly low. That incident being a DBA LOCA, loss of offsite power, loss of a diesel generator, plant at power, and Service Water inlet temperature above 80°F. Now, the maximum temperature is 81.5°F making it even less likely for the event to occur. The NRC also recognized that the time periods of elevated lake temperature are shorter than the time required to complete an action statement and therefore required no Technical Specification limit on ultimate heat sink temperature.

Even in the unlikely event that the incident of concern should occur the operators monitor SW system header pressure. If the header pressure reaches a predetermined low pressure corresponding to low flows to the various components the operators are procedurally instructed to align the Fire System to the Service Water System and start a fire pump. This action increases the flows to the critical components by approximately 10% as shown by past testing. This additional flow ensures that each component is capable of performing its design function for Service Water temperatures well in excess of any temperature ever measured.

9.1.3 DESIGN ANALYSIS

9.1.3.1 Margins of Safety

System reliability is achieved with the following features:

1. Each of the three service water pumps is capable of supplying 50% service water during normal, shutdown and post-DBA conditions.

2. Pump motor power is normally supplied from offsite sources with backup from the emergency diesel generators.

3. Redundant service waterlines supply critical systems. Loss of one header does not compromise Plant safety.

4. The fire pumps can be valved into the critical service water header thereby serving as a partial backup to the Service Water System.

9.1.3.2 Provisions for Testing and Inspection

Components of the Service Water System outside the containment building and aboveground are accessible for periodic inspection during Plant operation. Components inside the containment building are accessible only after Plant shutdown. This system is always in operation, but each service water pump can be periodically tested for auto-start by selection of one pump for standby service and tripping of one operating pump. After start-up, the piping is subject to the inservice inspection requirements of the ASME Boiler and Pressure Vessel Code, Section XI.

9.1.3.3 Discharge Line Rupture Analysis

The critical service water piping system consists of two main supply lines; one serving the component cooling heat exchangers, one serving the containment air coolers. One common discharge line returns the total service water to the cooling tower makeup basin. Each supply line also serves equipment comprising the minimum engineered safeguards for post-DBA cooling. The supply system is provided with valves to permit isolation of either of the main supply lines in the event of pipe failure.

The discharge piping system consists primarily of a 16-inch line from the containment air coolers and a 24-inch line from the component cooling heat exchangers. These two discharge lines are joined into a single 24-inch discharge line in the west engineered safeguards room. After the single 24-inch discharge line receives the service water from the engineered safeguards pumps seal coolers, the room air coolers and the control room air-conditioning units, the line is run underground to the discharge structure. The underground section receives service water from the air compressors, auxiliary building air-conditioning condensers, emergency generators and the noncritical equipment.

The containment air coolers 16-inch discharge line leaves the containment at elevation 611 feet 10 inches, turns down into the west engineered safeguards room through the component cooling equipment room floor at elevation 590 feet 0 inches and joins the 24-inch line from the component cooling heat exchangers. The single 24-inch discharge line is approximately 3 feet long in the west room. The single discharge line leaves the west room at elevation 579 feet 0 inches and is routed underground directly to the cooling tower pump suction/discharge basin at elevation 583 feet 0 inches.

Failure of the individual branch lines beyond their isolation valves and the single line have been analyzed to determine that failure in any part of the discharge piping will not result in significant flooding above the 590-foot floor. The amount of flooding has been based on the following:

1. Continuous spillage from a failure in a branch line will be backflow only. The supply can be valved off by closing the associated branch line valves.

2. Backflow through a ruptured branch line or spillage from the single line will be terminated at a water depth above the discharge elevation, of 583 feet, equal to the friction loss for the appropriate flow through the discharge line to discharge structure. The maximum friction loss and thus the maximum flood level occurs when the flow through the discharge line is the greatest.

With a service water flow of 16,386, a postulated slot rupture in the single 24-inch discharge line would result in a flood level elevation of 590 feet (Ref 12). All piping penetrations through the 590-foot floor are sealed and the maximum flooding depth will not result in any continuous leakage above the 590-foot floor.

All miscellaneous critical equipment connected to the common discharge line is provided with block valves in the individual discharges and can be isolated if required. The individual discharge lines are small and do not represent a potential large flooding source except for the emergency generators. These lines are each six inches, they do not penetrate the dividing wall, and the outlet block valves have been located outside of the generator rooms so that backflow through a rupture line associated with one generator will not enter the second generator room.

9.2 REACTOR PRIMARY SHIELD COOLING SYSTEM

9.2.1 DESIGN BASIS

The reactor Shield Cooling System is designed to remove heat from the biological shield surrounding the reactor vessel thereby limiting the thermal stresses in the structural concrete. It is not a safety-related system and was designed to CP Co Design Class 3 standards.

The system is designed to maintain concrete temperature below 165°F. The system is to assure that the concrete in the reactor cavity does not overheat and develop excessive thermal stress. One shield cooling pump and one set of cooling coils will be in operation whenever cooling is required to maintain the temperature of the concrete below approximately 165°F. The system is capable of removing 180,000 Btu/h. During normal Plant operation, the total heat load in the concrete is 120,000 Btu/h. This heat load consists of 85,000 Btu/h of convective and radiative heat losses from the reactor and 35,000 Btu/h of nuclear heat generated in the shield from the interaction of gamma rays and neutrons with the concrete.

9.2.2 SYSTEM DESCRIPTION AND OPERATION

9.2.2.1 System Description

The Shield Cooling System is a closed loop system consisting of two full-capacity sets of cooling coils, two full-capacity pumps, a heat exchanger, a surge tank, associated piping, valves, instrumentation and controls as shown on Figure 9-2.

All components of the system are located within the containment building.

Each set of shield cooling coils is composed of individual cooling coils embedded in the concrete shield. The distance between the inner surface of the concrete and center line of cooling coils is three inches.

The supply header to each set of cooling coils is provided with a diaphragm-operated, fail-open valve operated from the main control room. A check valve is provided in the discharge header for each set of coils to prevent flow from the coils in service into the coils which are out of service.

The closed loop system transfers heat to the Component Cooling Water System by means of the shield cooling heat exchanger. Demineralized water with a corrosion inhibitor is used in the shield cooling loop.

9.2.2.2 Component Description

Design ratings and design features of components of the system are given in Table 9-4.

9.2.2.3 System Operation

1. Normal Operation

During normal operation, one shield cooling pump and one set of cooling coils are in continuous service. The idle pump is in standby. The normal flow through the shield cooling coils is from 134 to 154 gpm. The shield cooling heat exchanger is in continuous service with the shield cooling water flowing through the tubes and component cooling water through the shell.

Both pumps can be started and stopped from the main control room. The standby pump starts automatically on low discharge header pressure.

The surge tank is installed at elevation 649 feet 0 inches in order to maintain an approximately constant suction head of 27 psig on the pumps. Makeup water to the tank is normally pumped from the primary system makeup storage tank through an on-off solenoid valve which is actuated by a level switch on the surge tank. The condensate storage tank can be used as an alternate makeup supply. High and low level in the tank is annunciated in the control room. The tank vents directly to the containment atmosphere and this protects the tank from overpressurization.

The temperature of the shield cooling water is regulated by manual adjustment of the component cooling water outlet header butterfly valve.

Temperature indication, high temperature (120°F) and low flow annunciation from the discharge of each set of coils are located in the control room. If the cooling coil set in operation becomes inopera-tive, the standby set is brought into operation by opening the inlet header control valve manually from the control room. Both pumps can supply cooling water to either set of coils.

The shield cooling system is used to maintain the concrete temperature below 165°F, thus preventing weakening of the structure through loss of moisture. The structure must remain intact during a DBA to preclude damage to the reactor building sump and the plugging of the suction lines to the engineered safeguards pumps. One pump and one set of cooling coils is more than adequate to remove the 120,000 Btu/hr heat load at Rated Power operation. The capacity of the reactor cavity concrete to sustain temperatures greater than 165°F is discussed in Reference 23.

The steady-state temperature profiles (reference 28) used in the design of the primary shield during normal operating condition and design basis conditions are shown in Figures 9-3 through 9-6.

2. Shutdown Operation

During hot reactor shutdown conditions, the operation of the system is the same as during normal operating conditions.

The temperature profiles in the primary shield are similar to those during normal operating conditions.

During cold shutdown of the reactor, one shield cooling pump will continue to operate during the initial hours. Subsequently, as the reactor temperature decreases to a point such that the resultant temperature to the shield concrete remains below approximately 165°F without cooling, the shield cooling pump can be stopped manually.

3. Post-DBA Operation

The cooling system is not required after a DBA.

9.2.3 DESIGN ANALYSIS

9.2.3.1 Margins of Safety

Each of the two sets of shield cooling coils is capable of removing 180,000 Btu/h. The heat exchanger is capable of removing 200,000 Btu/h of heat. However, the maximum heat load on the system is only 120,000 Btu/h. There-fore, there is an appreciable capacity margin in the Shield Cooling System.

Each pump is capable of handling 100% of the required cooling water flow for removing 180,000 Btu/h.

9.3 COMPONENT COOLING SYSTEM

9.3.1 DESIGN BASIS

The Component Cooling Water System, Figure 9-7, is designed to cool components carrying radioactive and potentially radioactive fluids. It provides a monitored intermediate barrier between these fluids and the Service Water System which transfers the heat to the lake. Thus, the probability of leakage of contaminated fluid into the lake is greatly reduced.

System components are rated for the maximum heat removal requirements that occur during normal, shutdown or accident operation as applicable. The parts of the system located inside containment are isolated in the event of a containment high-pressure signal (CHP). The component cooling water to the evaporators and spent fuel cooling system are isolated on SIAS. The system is designed to CP Co Design Class 1 requirements.

9.3.2 SYSTEM DESCRIPTION AND OPERATION

9.3.2.1 System Description

The system is a closed loop consisting of three motor-driven circulating pumps, two heat exchangers, a surge tank, associated valves, piping, instrumentation and controls. The Component Cooling System is shown on Figure 9-7. The system is continuously monitored by a process monitor which detects radioactivity which may have leaked into the system from the fluids being cooled.

The component cooling pump motors are connected to two separate 2,400 volt buses, with one pump on one bus and the remaining two on the other. The pumps can be started and stopped from the main control room and also locally at the switchgear.

System volume expansion and contraction due to start-up, shutdown and changes in load are accommodated by an elevated surge tank which also maintains a constant static head of approximately 28 psi on the pump suctions. The system can be vented to the auxiliary building through a diaphragm-operated three-way valve on the surge tank. The auxiliary building in turn vents to the outside atmosphere through the Plant ventilation exhaust stack. The other port on the three-way valve is connected to the gas collection header and is automatically transferred in the event the Component Cooling System contains radioactive gases due to leakage from radioactive systems being cooled. A relief valve discharging to the liquid radwaste system is provided on the surge tank to protect from overpressure.

The Component Cooling Water System uses demineralized water to which an inhibitor is added for corrosion control. Makeup to the system is automatically supplied from the primary system makeup storage tank.

Heat is transferred from the system to Plant service water by means of two component cooling heat exchangers. Cooling requirements are shown on Table 9-5. Service water from the critical service water-header is provided to the tube side of the heat exchangers and the rejected heat from the system is discharged by service water into the cooling tower makeup basin.

Four main supply lines are provided to the various areas of the Plant as follows:

1. To Shutdown Cooling Heat Exchangers

2. To Engineered Safeguards Pumps

3. To Spent Fuel Pool Heat Exchangers and Radwaste Equipment

4. To Services Inside the Containment

Supply valves in these lines are operable from the main control room and all, except the containment isolation valves and the fuel pool supply line valve, are operable from the Engineered Safeguards Auxiliary Panel.

Two full-capacity valves installed in parallel are provided in the line supplying component cooling water to the shutdown heat exchangers.

9.3.2.2 <u>Component Description</u>

Design ratings and construction features of system components are given in Table 9-6.

Material for components, connecting piping and valves in contact with component cooling water is carbon steel, cast iron or bronze.

9.3.2.3 <u>System Operation</u>

Flow requirements for various operational modes are shown in Table 9-7.

1. <u>Normal Operation</u>

   During normal operation, one or two of the three component cooling pumps and the two component cooling heat exchangers will be in service. The pump(s) runs continuously with the other pump(s) in "Standby." The number of pumps running is determined by the discharge header pressure. The auto-start feature is automatically reset on bus undervoltage to prevent cycling the pump breaker onto a dead bus.

Both component cooling heat exchangers are required to be in service during all plant operating modes above cold shutdown because both heat exchangers are required to remove accident heat loads and operation with only one heat exchanger could result in excessive flow rates (eg, if a safety injection signal is initiated) that damage the heat exchanger.

The temperature of the component cooling water at the heat exchanger discharge is controlled between 72 °F and 90°F by regulation of the service water flow. Gross adjustment required by seasonal temperature variations in the service water temperature is achieved by adjustment of hand indicating controllers (HICs), which position the heat exchanger service water outlet butterfly valves. Short-term fluctuations in CCW temperature are addressed by automatic temperature control of the globe valves that bypass the butterfly valves. High/low component cooling temperature is annunciated in the control room. The service water discharge temperature from each component cooling heat exchanger is indicated in the control room.

Makeup to the Component Cooling System is pumped to the surge tank from the primary system makeup storage tank through an automatic on-off diaphragm-operated valve which is actuated by a level switch on the surge tank. Tank low level is annunciated in the control room. Chemicals for corrosion control are added to the component cooling water chemical addition tank. By recirculation of cooling water through a recirculating header connecting the discharge header to the chemical addition tank, the chemical solution flows from the tank into the pump suctions where it mixes with the component cooling water and is gradually distributed throughout the Component Cooling System.

A radiation monitor in the pump discharge header detects radioactivity which may have leaked into the system from the equipment associated with the nuclear Plant. High activity is annunciated in the main control room. If the radioactivity level in the system reaches a preset level above normal background as detected by the radiation monitor, the three-way valve on the surge tank, if open to room atmosphere, is automatically closed to room atmosphere and opened to the vent gas collection header which is a portion of the gaseous radwaste treatment system.

2. Shutdown Operation

During shutdown cooling, two component cooling pumps and both component cooling heat exchangers cool the primary coolant from 300°F to cold shutdown. In this case, the remaining pump will act as a spare and is manually started from the control room.

Component cooling water is supplied to the shutdown cooling heat exchangers by remote-manual opening of the diaphragm-operated valves in the supply header from the main control room. At the start of the shutdown cooling operation, the component cooling water heat exchanger outlet temperature is at a maximum temperature of 90°F.

3. Emergency Operations

On initiation of the Safety Injection Signal (SIS) , the supply of component cooling water to the Spent Fuel Cooling System and to the radwaste evaporators will be cut off by automatic closure of supply and/or return line valves. This assures additional cooling capability by the system for the safety injection and containment spray water when it is recirculated through the shutdown heat exchangers, and for cooling the glands of the safety injection, charging and containment spray pumps. An SIS with right channel failure would not isolate CCW through the SFP and radwaste evaporators, which are non-essential loads. The resulting CCW flow still meets minimum flow requirements to essential loads or reduced flows have been shown to be acceptable (references 26 and 29). The valves in the gland cooling water supply and return headers for the ESS pumps do receive an SIS signal, however, they are normally in their SIS position. Therefore, no valve repositioning is required to provide CCW to the ESS pumps (reference 27).

If offsite power to the CCW pumps is available during an SIS, all three pumps will be started. If offsite power is not available during an SIS, the component cooling pumps are momentarily shed from the power supply buses. After the emergency diesel generators have energized the buses, two of the pumps are automatically started by the DBA sequencers. The third pump, though not required per accident analyses, is sequenced to standby and will only start on CCW system low pressure if neither of the other two pumps start. The system valve lineup for accident conditions is such that one pump is capable of providing required CCW flow to the two CCW heat exchangers at a higher pressure than the low pressure standby pump start setpoint.

If offsite power to the CCW pumps is lost and SIS is not present, then the pumps are shed from their respective bus. After the emergency diesel generators have energized the buses, two of the pumps are automatically started by the normal shutdown (NSD) sequencers. The third pump is sequenced later but only starts if a low pump discharge pressure is present, indicating that the other pumps have not started.

Upon receipt of an SIS , the valves in the component cooling water-supply lines to the shutdown cooling heat exchangers open. Upon receipt of a low-level signal from the SIRW tank, the valves in the component cooling heat exchanger main service water outlets and component cooling water inlets open to ensure maximum cooling water supply during the containment spray and safety injection recirculation mode; however, to conserve service water during RAS, the CCW heat exchanger outlet bypass valves close. Under this condition, neither the service water nor the component cooling water flows are modulated. The control valve in the supply line to the Spent Fuel Cooling System can be opened and closed remotely from the control room at the discretion of the operator in order to prevent overheating of the spent fuel pool.

The gland cooling water for the safety injection and containment spray pumps is backed up by service water from the critical Service Water System (see Subsection 9.1.2) in case of failure of the component cooling water supply. Low cooling water flow in the supply header to each engineered safeguards equipment room is annunciated in the control room. Changeover from one supply to the other is performed by manual operation of the Component Cooling Water Supply and Return Valves. The air supply for the Service Water Control Valves is then unisolated and one of the two Service Water Supply Valves along with the Return Valve are opened. The Service Water and Component Cooling Water Control Valves can be operated from the main Control Room or from the local Engineered Safeguards Auxiliary Panel. The air supply to the Service Water Control Valves is normally valved out of service to ensure a spurious operation of one of the Service Water Valves does not result in a loss of Component Cooling Water inventory.

The containment isolation valves in the Component Cooling System are designed to fail in the "Open" position in the event that air or control power is lost to the valve. This failure mode precludes the undesirable loss of component cooling water to the primary coolant pumps, control rod drive mechanisms, letdown heat exchanger or shield cooling coils during normal Plant operations. In the event of an accident which results in a containment high-pressure signal, the containment isolation valves in the Component Cooling Water System will close. If instrument air is also lost during this accident condition, the integrity of the containment will be maintained by the check valve in the supply header to containment and by the return header isolation valves which are provided with an air accumulator sized to place and maintain the valves in a "Closed" position.

A review was performed to assess the vulnerability of CCW piping within containment to failure caused by a high-energy line break (HELB). Failure of this piping, coupled with the CCW supply valve to containment (CV-0910) failing open due to a loss of air, could cause a complete loss of CCW inventory. The review concluded that this CCW piping within containment is not vulnerable to failure caused by a HELB (see Reference 7).

9.3.3 DESIGN ANALYSIS

9.3.3.1 Margins of Safety

1. Any one of three pumps is capable of supplying component cooling requirements during normal Plant operation. During shutdown, one pump can furnish at least 50% of the maximum shutdown cooling water requirements. For post-DBA operation, one pump can furnish 100% of the required capability for cooling the containment spray and safety injection recirculation water. For the Left Channel DBA condition, containment cooling is achieved using containment spray. For the right channel DBA, containment cooling is achieved using one spray pump and containment air coolers (VHX-1,2&3), which do not require CCW.

2. Pump motor power is normally supplied from offsite sources with backup supplied from the emergency diesel generators.

3. Two 50% capacity heat exchangers (based on maximum duty during shutdown cooling) are provided; each heat exchanger alone is capable of primary system cooldown at a reduced rate. Under post-DBA conditions, each heat exchanger is capable of handling at least 50% of the heat duty required, but neither is capable of handling 100% of the heat duty by itself without development of excessive flow conditions. Therefore, both heat exchangers are required to be operable at all times during normal operation.

4. Two full-capacity valves installed in parallel in the component cooling water-supply header to the shutdown heat exchangers ensure a reliable supply of cooling water for shutdown and for the post-DBA recirculation mode of operation.

9.3.3.2 Provisions for Testing and Inspection

Each pump was shop-tested in accordance with requirements of the Standards of Hydraulic Institute. The heat exchangers were each hydrostatically tested in accordance with the requirements of the ASME B&PV Code, Section III, Class C, 1965. Valve bodies were hydrotested in accordance with requirements of AWWA-C504.

Components of the Component Cooling System outside the containment are accessible for periodic inspection during Plant operation. Components inside the containment are normally accessible during Plant shutdown only.

Provisions are provided as necessary to facilitate ASME Section XI Code required tests as well as operability and performance tests.

9.4 SPENT FUEL POOL COOLING SYSTEM

9.4.1 DESIGN BASIS

The spent fuel pool cooling system removes decay heat from spent fuel stored in the spent fuel pool. The system was originally designed to remove the decay heat from one-third of the total core fuel elements.

The spent fuel pool cooling system is required to maintain the fuel pool water temperature less than 150°F with a minimum of one spent fuel pool cooling pump operating. The maximum spent fuel pool heat load resulting from off-loaded spent fuel shall be less than 28.64×10^6 Btu/hr regardless of whether the heat load is from a one-third core off-load or a full core off-load. A heat load less than 28.64×10^6 Btu/hr ensures that the spent fuel pool water temperature limit of 150°F is maintained. Heat is removed from the spent fuel pool by the spent fuel pool heat exchanger with component cooling water providing the cooling medium.

The fuel handling area, including the spent fuel pool, is a CP Co Design Class 1 structure and the entire spent fuel pool cooling system is a tornado protected, CP Co Design Class 1 system.

9.4.2 SYSTEM DESCRIPTION AND OPERATION

9.4.2.1 System Description

The fuel pool cooling system is a closed loop system consisting of two half-capacity pumps, a full-capacity heat exchange unit consisting of two heat exchangers in series, a bypass filter, a bypass demineralizer, a booster pump, piping, valves and instrumentation. The spent fuel pool cooling system is shown on Figure 9-8.

Materials used in the spent fuel pool cooling system are suitable for use with borated water (1% by wt boric acid). Fuel pool makeup water is supplied from the SIRW tank. A backup supply of water is available from the fire system. This would be utilized to replenish the pool water content in the event of considerable loss of pool water.

The clarity and purity of the water in the spent fuel pool are maintained by passing a portion of the flow through the bypass filter and/or demineralizer. Skimmers are provided in the spent fuel pool to remove accumulated dust from the pool.

Connections are provided for a temporary tie-in to the Shutdown Cooling System to provide a backup capability for the fuel pool heat exchangers.

The spent fuel pool cooling system is connected by valved piping to the reactor refueling cavity to cool the refueling cavity water during spent fuel transfer.

9.4.2.2 Component Description

Design ratings and construction of components are shown in Table 9-8.

9.4.2.3 System Operation

1. Normal Operation

   For normal refueling operations, one or both pumps may be operating depending on the heat load. If both pumps are initially operating, one of the two pumps may be stopped when the heat load has decreased to the point where it can be adequately dissipated by one pump. The pumps are started and stopped from the main control room or locally at the switchgear. A pressure switch on the discharge header annunciates low header pressure in the main control room. A manually controlled booster pump provides flow through the fuel pool filter and fuel pool demineralizer. The flow is regulated by manual adjustment of a butterfly valve in the main header in parallel with the filter and demineralizer.

2. Shutdown Operation

   During cold shutdown refueling condition, the reactor refueling cavity is filled with borated water from the SIRW tank. The reactor refueling cavity water can be cooled by the fuel pool heat exchanger and purified as needed by the filter and/or the demineralizer. The fuel pool cooling pumps empty the refueling cavity water into the SIRW tank after refueling is completed. The fuel tilting mechanism pits can be filled and emptied through the fuel pool cooling system piping interconnections and fuel pool pumps.

3. Post-Accident Operation

   In the event of a DBA, if auxiliary power is available, the pumps will continue to run if they were running before the accident. If no auxiliary power is available, the pumps are shed from the power supply buses. After the emergency diesel generators are running and the engineered safeguards equipment is in operation, the fuel pool cooling system can be operated intermittently from the main control room to remove decay heat generated by the spent fuel elements and thus prevent overheating of the spent fuel pool.

9.4.3 DESIGN ANALYSIS

9.4.3.1 Margins of Safety

Spent fuel decay heat values for the spent fuel pool cooling system heat load scenario below have been calculated using the methodology in NRC Branch Technical Position ASB 9-2, "Residual Decay Energy for Light-Water Reactors for Long Term Cooling." The analysis assumed that all available storage cells are filled (892 fuel assemblies - 11 x 11 Westinghouse rack in cask loading region), and that the full core off-load begins 216 hours (9 days) after reactor shutdown. The following chart shows the offload scenario (hypothetical) that bounds normal one-third core or full core offloads:

Parameter	Value
Heat Load, Btu/h	28.64×10^6
Maximum SFP Inlet Temperature, °F (2 SFP Pumps, 1840 gpm CCW Flow)	137.8
Maximum SFP Inlet Temperature, °F (1 SFP Pump, 1840 gpm CCW Flow)	145.3
Cooling Water Inlet Temperature, °F	90
Time To Reach 212°F (0 Pumps), h	3

NOTE: The analysis was performed for fuel stored in the main pool and the spare tilt pit. Therefore, the results given above apply to fuel stored in either location.

Failure of the outlet piping system would result in draining of the fuel pool to the outlet level which still maintains an adequate level of water in the pool for shielding and cooling requirements.

Failure of the inlet piping would result in no loss of water from the fuel pool as there is no downcomer by which a siphon could be started.

9.4.3.2 Provisions for Testing

When the cooling system is inoperative, starting and stopping of both pumps can be tested locally and from the main control room by manually operating the switches.

9.5 COMPRESSED AIR AND HIGH-PRESSURE AIR SYSTEM

9.5.1 DESIGN BASIS

The compressed air system is designed to provide a reliable supply of oil-free air for instruments and controls and for service air requirements. The design of the system is based on an estimated instrument air consumption of 80 scfm for the nuclear steam supply system and 115 scfm for the remainder of the Plant. Additional loads and system leakage make the total Plant requirements 250 scfm.

The high-pressure air system is sized to allow each valve operator, normally supplied by air from this system, to be stroked twice with the compressors inoperable and the initial pressure below the low-pressure alarm set point. This assures operability of those cylinder-operated safeguards valves necessary for accident conditions as long as the minimum pressure is maintained.

Compressed air needs for the condensate demineralizer building are supplied by two 100% capacity air compressors, which are independent of the Auxiliary Building Compressed Air System.

9.5.2 SYSTEM DESCRIPTION AND OPERATION

9.5.2.1 System Description

Two 200 scfm (C-2A and C-2C) and one 320 scfm (C-2B) nonlubricated compressors are provided, each with a separate receiver. C-2A and C-2C are cooled by critical service water, while C-2B is air cooled. C-2A and C-2C also have aftercoolers served with service water. C-2B has an air-cooled aftercooler. The air receivers are connected to the compressed air header. The receiver outlet air header branches into two separate air headers, one to the instrument air dryer and filter assembly, and one to the service air system. The system is shown on Figure 9-9.

Each air system header is divided into branch lines supplying the turbine building, the containment building, the intake structure and the auxiliary building.

A fail-open isolation valve is series with a check valve is located in the instrument air line outside the containment building. These valves are containment isolation valves. The isolation valve is designed to fail-open to provide a supply of instrument air to controls inside the containment.

Two banks of minimum 2000 psig nitrogen bottles provide limited backup of the compressed air system for operation of the Auxiliary Feedwater System valves. One bank of eight bottles is headered into the instrument air system and supplies 60 psig N_2 to the steam supply valve in the steam supply to the auxiliary feedwater pump turbine drive. The second bank, of four bottles, is piped to supply 60 psig N_2 to valves in the auxiliary feedwater supply piping to the steam generators.

The N_2 backup systems for the Auxiliary Feedwater System are required for the operability of auxiliary feedwater valves. Any failures of the AFW N_2 backup system will place the Plant in an LCO condition.

Four other nitrogen backup stations equipped with minimum 2,000 psig nitrogen bottles, located in the auxiliary and turbine buildings, provide backup of the compressed air system for operation of certain safety-related valves. Each station is sized to provide sufficient pressure to cycle each valve once and maintain the valve during a postulated accident in a desired position for a certain period of time or for a period of at least 5 days when a valve is required to hold position indefinitely. One station supplies 150 psig nitrogen to the 2 safety injection pump mini-flow stop valves. The remaining 3 stations supply 80 psi nitrogen to 2 containment spray isolation valves, 2 HPSI pump sub cooled suction valves, 2 service water containment isolation valves and 1 instrument air supply to containment valve. Nitrogen backup to all the valves except CV-0824 (service water containment isolation outlet) and CV-3070 / 3071 (HPSI pump cooling suction) are considered enhancements to normal air system and not necessary for the operability of that valve. However, the nitrogen backup is required to support post-fire safe shutdown (Appendix R).

The N_2 backup system is required to make CV-0824 and CV-3070 / 3071 operable. If CV-0824 becomes inoperable, then it places the Plant in an LCO per Technical Specification 3.4.4. CV-0824 is considered to be a valve directly associated with Service Water Pump P-7B. CV-3070 is required to have an operable N_2 backup system due to Appendix R fire concerns and is considered to be a valve directly associated with High Pressure Safety Injection Pump P-66B. Both CV-0824 and CV-3070 are supplied by the same N_2 backup station.

CV-3070 and CV-3071 are required to be operable during the recirculation mode to ensure the HPSI pump(s) have adequate available NPSH

One bank of air bottles provides a limited backup of the high pressure air system to CV-3018, P-66B HPSI Pump Discharge to Train 2. CV-3018 is required to have an operable backup air supply due to Appendix R fire concerns. The station is equipped with air bottles having a minimum pressure of 1800 psig and is located adjacent to CV-3018 in the West Safeguards Room. The backup air supply station is normally isolated and can be manually aligned to supply air at 150 psig. The station is sized to provide sufficient pressure to cycle CV-3018 once and maintain the valve in a desired position for 72 hours. It should be noted that the backup air supply is an enhancement and is not required for operability of CV-3018.

The nitrogen and air backup stations are shown on Figure 9-9, Sheet 2.

Nitrogen gas cylinders are also used to maintain an inert gas blanket on the internal connections of the containment electrical penetrations. See section 5.8.6.2.2 for a further description. The nitrogen bottle stations are shown on Figure 9-9, Sheet 3.

There is also a nitrogen backup system installed to provide backup of the instrument air system to the Main Steam System's Atmospheric Dump Valves (ADVs). The nitrogen backup system provides 90 psig nitrogen to the ADVs provided by the 230 psig bulk nitrogen system. The bulk nitrogen system was utilized as the source of the nitrogen rather than a bottle system due to the large number of bottles that would have been required to provide the four hour coping duration necessary to meet Station Blackout requirements. Use of the bulk nitrogen system allows the ADVs to exceed the Station Blackout coping duration requirements of 10 CFR 50.63 as recommended in Reg Guide 1.155. See Section 8.1.5 for an additional description of Palisades' response to Station Blackout requirements. The sizing of the pressure regulator provides sufficient nitrogen to fully open all four ADVs simultaneously. There is excess capacity for throttling the ADVs for the four hour coping duration. Nitrogen backup to the ADVs is not considered safety related per Reg Guide 1.155 and is not required for the operability of the ADVs. However, the nitrogen backup is required to support post-fire safe shutdown (Appendix R).

The high-pressure compressed air consists of three high-pressure, oil lubricated air compressors, each with its own refrigeration-type dryer and air receiver. The high-pressure air compressors provide high-pressure control air for cylinder-operated vital valves located in each of two engineered safeguards rooms and the turbine building. Though normally separated, these three compressors can, in an emergency, be crossconnected. The air receivers are sized to allow each valve operator, normally supplied by air, to be stroked twice with the compressor inoperable and the initial pressure (260 psi) far below the low-pressure alarm set point (300 psi). This assures operability of those cylinder-operated safeguards valves necessary for accident conditions, so long as the minimum pressure (260 psi) is maintained. Moisture is removed from the high-pressure air by refrigeration units in series with the compressors air-cooled aftercooler. Any remaining moisture is removed by periodic blowdown of the air receivers.

The Condensate Demineralizer Building compressed air needs are supplied by either of two full sized air compressors, each with an integral intercooler and separate aftercooler and receiver. Service air is piped directly from the receivers, while instrument air is routed from the receivers to a dryer and then piped to the instruments.

9.5.2.2 Component Description

Design ratings and construction of components are shown on Table 9-9.

9.5.2.3 System Operation

1. Compressed Air System

a. Normal Operation

A continuous supply of 80-100 psig instrument air is provided to hold power-operated valve actuators in the positions required for operating conditions and to provide air for modulating control valves. Two compressors (C-2A and C-2C) will operate at constant speed with interlocked controls such that both compressors will load and unload simultaneously in response to variations in the pressure in the common header. The third compressor (C-2B) will be on automatic standby.

Another operating mode is with C-2B operating and C-2A and C-2C in standby.

Each of the air compressors can be started, set up on standby and tripped by a separate control switch in the main control room.

The instrument air header downstream of the filters has a pressure switch which initiates the closing of a shutoff valve on the service air header in the event the instrument air pressure drops to 85 psig. In addition, low-pressure is alarmed in the control room.

b. Shutdown Operation

The system remains capable of supplying the necessary instrument air irrespective of whether offsite power is available or not. When offsite power is available, the system operation is similar to normal operation. In case of offsite power failure, the compressors can be restarted on emergency power from the diesels.

c. Post-DBA Operation

In the event of a DBA with loss of offsite power, the compressor motors are shed from the normal ac bus. Subsequently, the emergency diesel generators are started and the compressors can be manually started after all engineered safeguards equipment has started to provide the air supply. During the interim period, air stored in the three receivers is available to meet system requirements. If offsite power is available, the system operation is similar to normal operation.

2. High-Pressure Air System

The high-pressure air system is shown on Figure 9-10.

a. Normal Operation

Each high-pressure air compressor operates automatically as necessary to maintain a pressure of about 325 psig in its individual receiver tank.

b. Shutdown Operation

The system remains capable of supplying the necessary air irrespective of whether offsite power is available or not. When offsite power is available, the system operation is similar to normal operation. In the case of offsite power failure, the compressors can be restarted on emergency power from the diesel.

3. Condensate Demineralizer Building Compressed Air System

The condensate demineralizer building air compressors operate automatically as necessary to maintain a pressure of approximately 125 psi in the air receiver tanks.

4. Post-DBA Operation

In the event of a DBA with loss of offsite power, the compressor motors are shed from the normal ac bus. Assuming the air compressors are not manually restarted from the emergency diesel generators, enough air supply is available in the receiver tanks to provide system requirements. If offsite power is available or power from the emergency diesel generators is used, the system operation is similar to normal operation.

9.5.3 DESIGN ANALYSIS

9.5.3.1 Margins of Safety

Two of the three air compressors in the compressed air system are rated to deliver 200 scfm, while one supplies 320 scfm. The total requirement is 250 scfm. Therefore, during normal operating condition, approximately 200% margin of safety is incorporated into the system.

During the post-DBA condition, two compressors are in operation, which are adequate to supply all demands.

The high-pressure air system receiver tanks for the engineered safeguards rooms are sized such that after loss of an air compressor, all connected valves can be cycled in one direction with sufficient air capacity remaining to accomplish the valve operations necessary to shift the Safety Injection System to the recirculation mode of operation.

9.5.3.2 Provisions for Testing

Each compressor can be tested to ensure operability with manual "on-off" switches located in the main control room (one switch for each compressor).

9.5.3.3 Failure of instrument Air

Instrument air is primarily used for motive power for valve actuation and is not used in any reactor indication, control or protective circuit. The design of the system and redundancy of equipment and power supplies ensure that total loss of instrument air is highly improbable; however, attention has been given to the overall Plant design to ensure valve failures upon loss of air are consistent with the capability to maintain the Plant in a safe condition and to mitigate the consequences of any simultaneous incident or accident.

During normal Plant operation or Plant shutdown, only diaphragm air-operated valves are required to function or to be maintained in position. The diaphragm-type operator will function at pressures down to 30 psig. Assuming a stored air capacity of approximately 200 cubic feet is available (the volume of the three air receivers and air piping) and the system design usage rate of 195 scfm, 2.6 minutes is available from stoppage of all three air compressors until the diaphragm air-operated valves no longer function or assume their failed position.

When instrument air system pressure drops below 85 psig, a check valve in the nitrogen supply from the bulk nitrogen system opens and continues to allow the Atmospheric Dump Valves (ADVs) to be operated remotely. This system is intended to meet the four hour coping duration of Palisades' Station Blackout but in reality could provide operation of the ADVs for several days with normal initial bulk nitrogen tank level. The backup nitrogen pressure for the ADVs is higher than for the other diaphragm air-operated valves because the ADV operators begin to assume their loss-of-air position at 60 psig, which is higher than that for the other valves. There is also a solenoid operated valve for each ADV that is set to shut at 80 psig. This solenoid valve is set to ensure that upon a loss of instrument air pressure and backup nitrogen pressure, the ADV air accumulators do not depressurize to the point where a "rapid open" signal from the ADV controller would fail to get all four ADVs open in the required time. (Through other solenoid valves, the air accumulators only provide air to the ADV operators during a "rapid open" condition.) The backup nitrogen pressure must be high enough to prevent this solenoid valve from actuating, thus it is set at 85 psig.

When the instrument air supply drops to below 60 psig, a check valve in the nitrogen supply from the high-pressure bottles opens and continues to feed auxiliary feedwater valves. The bottles are designed to supply the valves for a minimum of 12 hours.

During a Design Basis Accident or post-DBA condition, operation of piston-type, air-operated valves may be desired. The piston air operator requires a minimum of 70 psig to function and, considering the same capacity and usage rate assumed above, will become inoperable or will assume its failed position in 1.4 minutes.

As discussed below, no failure of valves due to loss of instrument air precludes maintaining the Plant in a safe condition.

The positions of significant air-operated valves during normal reactor operation, reactor shutdown and loss of motive air are listed in Table 9-10.

Loss of instrument air during normal Plant operation may require manual trip of the main turbine due to the loss of control of various air operated valves. The turbine trip will induce the associated reactor trip. The Plant will then be maintained in a hot shutdown condition. Temperature of the primary system can be controlled through steam safety valve actuation while makeup water is added with the auxiliary feed pumps. In addition, the operation of the radioactive waste system is limited to accumulation of wastes in the drain tanks. Processing of waste or discharging of wastes is not possible upon loss of instrument air. Continued cooling of primary system components is possible in that the Component Cooling System and the Service Water System continue to function normally with the exception that the component cooling supply to the spent fuel pool cooling system is secured and maximum service water flow to the three safety-related containment air coolers is initiated (VHX-4 inlet valve fails closed). The spent fuel pool may be cooled, if required, through the temporary connection to the fire water system. No air-operated valve failure upon loss of instrument air precludes maintaining the Plant in a safe shutdown condition.

If the Plant is in the process of being cooled down using the Shutdown Cooling System or the primary system temperature is being maintained by the Shutdown Cooling System, loss of instrument air restricts continued cooling through failure of LPSI pump discharge crossover valve (CV-3055) and shutdown heat exchanger discharge valve (CV-3025). If there is significant decay heat, system temperature will increase until limited by the heat removal capability of the steam safety valves. In addition, the limitations discussed above under normal operation apply.

In the unlikely event of a Design Basis Accident occurring simultaneously with loss of instrument air, no valve failure will limit the ability of the engineered safeguards system to perform its function. Maximum cooling is initiated to the three safety-related containment air coolers upon loss of air (VHX-4 inlet valve fails closed) and the containment spray header isolation valves fail open. The HPSI pump subcooling valves are opened manually at the start of recirculation. No other air-operated valve operation is required of valves supplied from the compressed air system. Assuming the high-pressure air compressors were also lost, sufficient stored air capacity is available in the accumulators to open the containment sump suction valves after an SIRW low level is reached. Subcooling of the HPSI pump suction is required for the post-DBA condition.

9.6 FIRE PROTECTION

9.6.1 INTRODUCTION

Following a fire at the Browns Ferry Nuclear Station in March 1975, the Nuclear Regulatory Commission initiated an evaluation of the need for improving the fire protection programs at all licensed nuclear power plants. As part of this continuing evaluation, the NRC, in February 1976, published a report by a special review group entitled, "Recommendations Related to Browns Ferry Fire," NUREG-0050. This report recommended that improvements in the areas of fire prevention and fire control be made in most existing facilities and that consideration be given to design features that would increase the ability of nuclear facilities to withstand fires without the loss of important functions. To implement the report's recommendations, the NRC initiated a program for reevaluation of the fire protection programs at all licensed nuclear power stations and for a comprehensive review of all new license applications.

The NRC issued guidelines for fire protection programs which reflect the recommendations of NUREG-0050. The NRC issued 10 CFR 50, Appendix R guidelines in November 1980 for operating plants, reinforcing the requirements of 10 CFR 50, Appendix A, General Design Criterion 3.

As a result of these fire protection program guidelines, CP Co has:

1. Made major Plant modifications

2. Written new procedures and administrative controls

3. Changed operating practices

4. Amended the Technical Specifications via Amendments 37, 42, 60 and 64 to the Operating License DPR -20

5. In accordance with NRC recommendations in Generic Letter 86-10, removed the fire protection Limiting Conditions for Operation and the fire protection surveillance requirements from the Technical Specifications and included them in the FSAR (Section 9.6.7).

6. Added a condition to the Palisades Operating License which requires implementation and maintenance of the Fire Protection Program, as described in the FSAR.

Section 9.6 has been rewritten to be an overview of the Palisades Fire Protection Program.

Fire protection consists of fire suppression systems, equipment and procedures to provide protection to Plant equipment, structures and personnel from fire, explosions and the release of toxic vapors. Fire protection also provides a means of maintaining the integrity of safety-related systems. Fire suppression systems consist of automatic sprinkler systems, automatic deluge systems, standpipe and hose systems and fire extinguishers located around the Plant site. Fire protection procedures consist of implementing procedures established to inspect, test, train and operate the fire protection systems.

9.6.1.1 Other FSAR Sections Related to Fire Protection

The Palisades plant completed an Appendix R re-analysis in 1996. This re-analysis completely replaced the old Appendix R analysis. The plant systems, controls and instrumentation relied upon for shutdown following a fire are briefly summarized in Section 9.6.8. The fire area safe shutdown analysis, including details of components lost, and components relied upon for safe shutdown following a fire in any fire area of the plant, is presented in the Post-Fire Safe Shutdown Analysis (PSSA), which is included in the Fire Protection Program Report described below.

For a discussion on how the raceway and cabling system at Palisades meets NRC BTP CMEB 9.5-1, Regulatory Guide 1.75, and Appendix R to 10CFR50, refer to FSAR Section 8.5.3.

For a discussion of Fire Protection Training refer to FSAR Section 12.2.1.12.

9.6.1.2 Fire Protection Program

The Fire Protection Program encompasses several documents that are contained in the Fire Protection Program Report (FPPR). The FPPR contains or references the analyses and procedures which are the basis for the Fire Protection Program. The FPPR contents are summarized as follows:

1. Fire Protection Plan - This procedure was developed to satisfy Criterion 3 of Appendix A to 10CFR50 and utilizes some of the information provided in NRC guidance document "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls, and Quality Assurance," dated 8/19/77. The overall Fire Protection Plan identifies the various positions within the Palisades' organization that are responsible for the program, states the authorities that are delegated to each of these positions to implement those responsibilities, and summarizes the equipment utilized for fire protection, fire detection and suppression capability, and limitation of fire damage. The plan also describes specific features necessary to implement the program, such as administrative controls and personnel requirements of fire prevention and manual fire suppression systems, and the means to limit fire damage to structures, systems, or components important to safety so that the capability to safely shut down the plant is ensured.

2. Fire Hazards Analysis Report (FHAR) - The FHAR was performed by qualified fire protection and system engineers to:

   a. Provide an overview of the adequacy of fire protection features for each fire area,

   b. Identify the supporting analyses and exemptions for each fire area,

   c. Provide a general area description and describe the fire barriers defining each fire area, and

   d. Identify the potential for radiological releases from a fire in each fire area.

3. Post-Fire Safe Shutdown Analysis (PSSA) - The fire areas identified in the FHAR were evaluated to determine the effect of a single exposure fire on equipment and circuits utilized for safe shutdown of the plant and this is documented in the PSSA. The PSSA identifies necessary safe shutdown systems and components, safe shutdown circuits and their associated fault consequences, raceway locations, and determines the fire consequences and compliance strategies for each fire area. Safe shutdown performance goals were established to ensure the reactor will be safely shut down, cooled down, and maintained in a cold shutdown condition in the event of a fire. The specific performance goals that satisfy the criteria for safe shutdown in the event of a fire are:

   a. Reactor reactivity control,

   b. Reactor coolant inventory control,

   c. Reactor coolant pressure control,

   d. Decay heat removal,

   e. Process monitoring, and

   f. Supporting functions

   Safe shutdown logic diagrams, a safe shutdown equipment list, and other supporting calculations are identified in the PSSA evaluation. The PSSA compliance strategies for each fire area were utilized to develop plant operating procedures that ensure the ability to safely shutdown the plant in the event of a fire.

4. A listing and brief descriptions of NRC approved exemptions to the requirement of 10CFR50.48 and 10CFR50, Appendix R, is provided as a quick reference source.

5. A listing of the documents that comprise the Fire Protection Safety Evaluation Reports for Palisades is provided as a quick reference source.

6. Palisades summary responses to NRC guidance provide in Appendix A to Branch Technical Position APCSB 9.5-1 and Regulatory Guide 1.78 and 1.101.

9.6.1.3 <u>Changes to the Fire Protection Program</u>

1. Changes may be made to the Fire Protection Program without prior approval of the NRC only if the changes do not adversely affect the ability to achieve and maintain safe shutdown of the plant in the event of a fire.

2. Specific features of the approved Fire Protection Program may be altered subject to the following criteria.

   a. Such changes do not otherwise involve a change in a license condition or technical specification or result in an unreviewed safety question (see 10 CFR 50.59).

   b. Such changes do not result in failure to complete the Fire Protection Program as approved by the NRC. A current record of such changes shall be maintained in auditable form and shall be available to NRC inspectors upon request. This includes an analysis of the effects of the change on the Fire Protection Program. All changes to the approved Program shall be reported annually, along with the FSAR revision.

   c. Temporary changes to specific fire protection features which may be necessary to accomplish maintenance or modifications are acceptable provided interim compensatory measures are implemented.

   (Reference: Amendment 152 to the Technical Specifications)

9.6.2 DESIGN BASIS

Fire protection at the Palisades Plant uses a defense-in-depth concept of design, to provide a high degree of safety. The Plant is designed to prevent fires, detect and suppress quickly any fires that do occur, limit the damage and prevent safety-related functions and systems from being interrupted.

Fire protection is designed in accordance with the guidance of the National Fire Protection Association, the American Insurance Association, NEPIA (now American Nuclear Insurers) and the applicable codes and regulations of the State of Michigan.

The fire suppression water system may also provide a backup water supply to the following:

1. Auxiliary Feedwater Pumps Suction

2. Critical Service Water Supply

3. Spent Fuel Pool Fill

The diesel engine-driven fire pumps and the piping connecting the fire suppression water system to the Auxiliary Feedwater System are designed to CP Co Design Class 2 requirements (see Subsection 5.2.1.2). The remainder of the system is designed to CP Co Design Class 3 requirements. Appropriate valving is provided to separate the system, if required.

Administrative procedures (Fire Protection Implementing Procedure 7) have been developed and are used to monitor and control hazardous materials when required for use in safety-related areas and throughout the Plant. This ensures a minimum impact on Plant personnel and safety-related systems.

9.6.3 SYSTEM DESCRIPTION AND OPERATION

9.6.3.1 System Description

The building structure has been designed and arranged to prevent the spread of fire and to ensure integrity of redundant safe shutdown systems and areas. A complete description of fire areas, separations, and means of fire protection is detailed in the Fire Protection Program Report (FPPR).

The fire system is shown on Figures 9-12, 13 and 14. Fire suppression is provided by fixed water spray systems, such as sprinkler systems and deluge systems, fire hose reels and cabinets, portable fire extinguishers, fire barriers and fire detection systems. These fire suppression provisions are found throughout the Plant site.

Fire hoses from fire hydrants and a standpipe system will provide protection in accordance with the guidance of NFPA 14, 20 and 24.

The fire hydrant piping system is designed, installed and tested in accordance with the guidance of NFPA 24-1965, Outside Protection. The pumping supply system and fire pumps are designed and installed in accordance with NFPA 20-1959, Installation of Centrifugal Fire Pumps.

The standpipe system is designed, installed and tested as a Class II system in accordance with the guidance of NFPA 14-1963, Installation of Standpipe and Hose Systems.

Fixed water spray systems, such as wet pipe fusible link sprinkler systems, dry pipe fusible link sprinkler systems, and fixed fog deluge spray systems are designed, installed and tested in accordance with the guidance of NFPA 13-1968, Installation of Sprinkler Systems, and NFPA 15-1966, Water Spray Fixed Systems for Fire Protection. Indication of individual systems in various areas is indicated on an annunciator panel in the main control room.

Fixed fog deluge systems protect the main, start-up and station auxiliary transformers. Each of these deluge systems are automatically actuated and annunciated by a general alarm in the main control room. A manual operated fixed fog deluge system protects the charcoal filters used to maintain control room habitability.

The wet pipe and dry pipe fusible link sprinkler systems are provided in selected plant areas as identified in the FHAR. Actuation of any sprinkler system is annunciated by a general alarm in the main control room.

Readily accessible rack- or reel-mounted fire hose lines with fog-type nozzles are located throughout the Plant, so that all areas in the turbine and auxiliary buildings are within 75 feet (maximum) of a 35 psig (minimum) fog nozzle.

Fire detection is provided in the form of smoke and ultraviolet detectors. These detectors were located and installed in accordance with the guidance of NFPA 72E-1974. The fire detectors are located in selected plant areas as identified in the FHAR. Initiation of any of these detector zones alarms on the annunciator panel located in the main control room and in Switchgear Room 1D.

To comply with Appendix R, there is an ultraviolet fire detection system installed in the intake structure (Room 136). The fire detection control panel (C132) has battery backup and along with the detectors (11) gives alarm annunciation to the control room for a fire throughout the area. The ultraviolet detectors should eliminate nuisance alarms that could be generated by other types due to the heat from the exhaust system of the diesel driven fire pumps in the room.

Portable fire extinguishers are provided at convenient and accessible locations. The extinguishing media are pressurized water, CO_2 or dry chemical as appropriate for the service requirements of the area.

Water for the fire suppression system is supplied by one of three full-capacity fire pumps. Each fire pump is capable of providing water to the largest system demand plus fire hose streams in the area of demand. One fire pump is electrically driven; the other two are diesel engine-driven. Any fire pump will start automatically and can be manually started from the pump control panel . The diesel engine-driven fire pumps can also be manually started from the control room.

A jockey pump with local controls is provided to maintain the fire suppression system full and pressurized.

The fire pumps are housed in the Class 1 portion of the intake structure. The backup supply header to the auxiliary feedwater pumps is buried underground for protection against tornadoes. A cross-connection provided with two, series, hand-operated valves connects the fire pump discharge header to the suction header of the auxiliary feedwater pump. One cross-connection provided with a hand-operated valve connects the fire pump discharge header to each of the critical service waterlines. Both of the above cross-connections are protected from tornadoes. A header terminating in a blind flange is provided in the spent fuel pool heat exchanger room for emergency filling.

Electric cable fire protection is provided by approved fire barriers and fire stops, in addition to the above fire detection and automatic sprinkler systems where needed to augment separation requirements. See Section 8.5 for details of cable separation requirements and implementation.

9.6.3.2 Component Description

Design ratings and construction details of components in the systems are given in Table 9-12.

9.6.3.3 System Operation

The motor-driven fire pump starts automatically on a low fire system header pressure of 98 psig with the first diesel engine-driven fire pump being started at 83 psig. The second diesel-driven pump starts upon a pressure drop to 68 psig. The diesel-driven fire pumps are thus arranged to back up the motor driven fire pump in case the latter does not start.

The jockey pump operates continuously to keep the system pressurized at or above 100 psig. Excess jockey pump discharge flow is recirculated back to the intake crib. Should the jockey pump be removed from service for maintenance, the fire suppression water system header pressure will be maintained through operation of the motor-driven fire pump or through a temporary connection to a service water booster pump which takes suction from the Non-Critical Service Water Header. In the case of a failure of the jockey pump or the temporary connection, the fire suppression water system will be pressurized by automatic operation of the motor driven fire pump by tripping of one or all of the pump's header pressure switches. Operation of the motor-driven fire pump is annunciated in the control room to alert the operator of system usage.

The backup supply to the other systems is activated by locally starting a fire pump and hand opening the block valves.

9.6.4 TESTS AND INSPECTION

The fire suppression water system is provided with connections with test valves on the fire suppression header for periodic testing. All equipment is accessible for periodic inspection.

Procedures are in effect to maintain and test in accordance with the requirements of Section 9.6.7.2 and established standards.

9.6.5 SAFETY EVALUATION

9.6.5.1 Fire Protection Program Report (FPPR)

The Fire Protection Program Report includes a section titled, Fire Protection Safety "Evaluation Report," which contains the Safety Evaluation Reports (SERs) issued by the NRC and constitues NRC approval of Palisades' Fire Protection Program. These SERs are listed below.

1. "Fire Protection Safety Evaluation Report," dated September 1, 1978, evaluated Palisades' response to NRC guidelines issued pertaining to NUREG-0050 , "Recommendations Related to Brown's Ferry Fire."

2. "Supplement No. 1 to Fire Protection Safety Evaluation Report," dated March 19, 1980, declared acceptable two open items from the September 1, 1978 report concerning cable penetration firestop qualification and non-approved fire detectors.

3. "Supplement No. 2 to the September 1, 1978 Fire Protection Safety Evaluation Report," dated February 10, 1981, declared acceptable several open items from the earlier report concerning fire barriers, the reactor coolant pump oil collection system, and administrative controls for the fire protection program.

4. "Safety Evaluation by the Office of Nuclear Reactor Regulation Review in Accordance with Appendix R, Sections III.G and III.L," dated May 26, 1983, approved a plan for meeting the alternate shutdown requirements of III.G and III.L, approved Palisades' resolution of associated circuits issues, and noted Palisades' promise to prepare certain operating procedures and to verify the ability of the steam jet air ejector to achieve cold shutdown within 72 hours.

5. "Safety Evaluation by the Office of NRR - Installation of Switches for Appendix R," dated January 29, 1986, approved use of isolation switches in certain areas to protect equipment circuitry needed to safely shutdown in order to meet Section III.L.4.

6. "Safety Evaluation Regarding a Postulated Fire in the Charging Pump Room or Corridor," dated December 3, 1987, approved use of a containment spray pump in series with a high pressure safety injection pump to provide makeup to the primary coolant system in the event of a fire that disables the charging pumps.

7. "Safety Evaluation Related to Amendment No 122 to Provisional Operating License No DPR-20," dated May 19, 1989, approved a proposed Technical Specification amendment relating to operability and surveillance requirements for the alternate shutdown panel.

8. "Safety Evaluation Related to Amendment 152 to Facility Operating License No. DPR-20," dated August 21, 1992, approved transfer of requirements for fire protection systems from the Technical Specifications to the FSAR. Also approved was the addition of Fire Protection Program requirements to Technical Specification administrative control requirements.

In addition to the SERs listed above, the Fire Protection Program Report contains transmittals in which exemptions were granted by the NRC for certain 10 CFR 50.48 and 10 CFR 50 Appendix R requirements. Exemptions contained in the FPPR that do not appear in the SERs above are listed below.

1. Exemption from Section III.G.3 of Appendix R to 10 CFR 50 which required a fixed fire suppression system in the control room was approved February 8, 1983.

2. "Safety Evaluation by the Office of NRR Relative to Appendix R Exemptions Requested for the Palisades Plant," dated July 12, 1985, granted an exemption to Section III.G.3 which required fixed fire suppression systems in the Engineering Safeguards Panel Room and in the corridor between the Charging Pumps' Room and the 1C Switchgear Room.

3. Exemption from Section III.G.2 of Appendix R to 10 CFR 50 which required that redundant cables inside containment be separated by more than 20 feet with no intervening combustibles, or that fire detectors and a fixed fire suppression system be installed, or they be separated by a radiant heat shield. This exemption, dated July 23, 1985, approved the use of fire stops in lieu of the 20 foot separation with no intervening combustibles.

4. Exemption from Section III.G.2 of Appendix R to 10 CFR 50 as described above was granted June 21, 1991 in the containment air room on the basis of the configuration of the room and administrative controls that prevent local storage of combustibles.

9.6.6 PERSONNEL QUALIFICATIONS AND TRAINING

There is a five-man fire brigade onsite at all times. Procedures are in effect that provide fire brigade training and actions required. The Fire Protection Implementing Procedures cover the following topics:

Fire Protection Plan

Organization and Responsibility

Fire Emergency Responsibility and Response

Plant Fire Brigade

Fire Protection Systems and Equipment

Inspection and Testing of Fire Protection Systems and Equipment

Fire Suppression Training

Fire Prevention Activities

9.6.7 GENERIC LETTER 88-12

Generic Letter 88-12 provides guidance for the preparation of a license amendment request to implement Generic Letter 86-10. Such an amendment (1) institutes the standard license condition for a Fire Protection Program, (2) removes requirements for fire protection systems from Technical Specifications, (3) removes fire brigade staffing requirements from Technical Specifications, and (4) adds administrative controls to Technical Specifications that are consistent with those for other programs implemented by license condition.

This section of the FSAR contains the former Technical Specifications associated with fire detection systems, fire suppression systems, fire barriers, and the administrative controls that address fire brigade staffing.

The Palisades Operating License states that the plant shall implement and maintain in effect all provisions of the approved Fire Protection Program as described in the Final Safety Analysis Report (FSAR) for the facility and as approved in the NRC Safety Evaluation Report and the plant may make changes to the approved Fire Protection Program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

Thus, changes can be made to the limiting conditions for operation and surveillance requirements in FSAR Section 9.6.7 only if those changes do not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire, and if those changes do not involve an unreviewed safety question as defined in 10CFR50.59.

9.6.7.1 Requirements for Operation

9.6.7.1.1 Fire Detection Instrumentation

The fire detection instrumentation for each fire detection zone shown in Table 9-11 shall be OPERABLE.

APPLICABILITY: At all times when equipment in the fire detection zone is required to be OPERABLE.

ACTION: With the number of instruments OPERABLE less than that required by Table 9-11.

A. Except the detectors located inside containment, within one hour establish a fire tour to inspect the zone with the inoperable instrument(s) at least once per hour.

B. Restore the inoperable instrument(s) to an OPERABLE status within 14 days, or

C. Write a Condition Report to focus management attention on the need to restore the inoperable instuments(s) to OPERABLE status

D. When this Limiting Condition for Operation and/or associated action requirements cannot be satisfied, provisions relating to operating restrictions on the plant are not applicable.

E. For detectors located inside containment, restore the inoperable detector(s) to operable status within 24 hours, or within the next hour, and at least once per hour thereafter, view with the TV camera the zone containing the inoperable detector or view the zone located above the detector.

   If a fire in containment is confirmed, be in at least hot standby within the next 6 hours and in cold shutdown within the following 30 hours.

BASIS: OPERABILITY of the fire detection instrumentation ensures that adequate warning capability is available for the prompt detection of fires. This capability is required to detect and locate fires in their early stages. Prompt detection of fires will reduce the potential for damage to safety related equipment and is an integral element in the overall facility Fire Protection Program.

When a portion of the fire detection instrumentation is inoperable, frequent fire tours shall be established in the affected areas to provide detection capability until the inoperable instrumentation is returned to service.

(Formerly Technical Specification 3.22.1)

9.6.7.1.2 Fire Suppression Water System

The fire suppression water system required for fire sprinkler systems and fire hose stations, as defined in Sections 9.6.7.1.3 and 9.6.7.1.4 of the FSAR, shall be OPERABLE with:

A. Two fire pumps, one of which is the south diesel driven fire pump, each with a capacity of at least 1500 gpm with their discharge aligned to the fire suppression header.

B. An automatic initiation logic for each fire pump.

APPLICABILITY: At all times.

ACTION:

A. With the diesel driven fire pump located south of the fire protection assembly (radiant heat shield) in Room 136 or with two fire pumps inoperable, restore the inoperable equipment to operable status within 7 days, or, write a Condition Report to focus management attention on the need to restore redundancy in the system.

B. With the fire suppression water system otherwise inoperable:

1. implement the action statement in section 9.6.7.3 for inoperable fire sprinkler systems;

2. implement the action statements in section 9.6.7.1.4 for inoperable fire hose stations;

3. establish a backup fire suppression water system within 24 hours; and

4. Write a Condition Report to focus management attention on the need to restore the fire suppression water system to OPERABLE status, or

5. if 3 and 4.a above cannot be fulfilled, place the reactor in hot standby within the next 6 hours and in cold shutdown within the following 30 hours.

BASIS:

The operability of the fire suppression water system ensures that adequate fire suppression capability is available to confine and extinguish fires occurring in any portion of the facility where safety related equipment is located. The fire suppression water system consists of the water supply, sprinkler systems, and fire hose stations. The collective capability of the fire suppression system is adequate to minimize potential damage to safety related equipment and is a major element in the facility Fire Protection Program.

When portions of the fire suppression water system are inoperable, alternate backup fire fighting equipment is required in the affected areas until the inoperable equipment is restored to service.

When the fire suppression water system becomes inoperable, immediate corrective measures must be taken as this system provides the major fire suppression capability of the plant. The requirement for a Condition Report provides for prompt evaluation of the acceptability of the corrective measures to provide adequate fire suppression capability for the continued protection of the nuclear plant.

When a portion of the fire detection instrumentation is inoperable, the establishment of frequent fire tours in the affected areas is required to provide detection capability until the inoperable instrumentation is restored to OPERABILITY.

Those fire hose stations that are supplied by the service water system in containment will be used primarily during normal refueling operations.

(Formerly Technical Specification 3.22.2)

9.6.7.1.3 Fire Sprinkler System

The sprinkler systems located in the following areas shall be OPERABLE:

A. Cable Spreading Room 224

B. 1-C Switchgear Room 116A

C. 1-D Switchgear Room 223

D. 1-1 Diesel Generator Room

E. 1-2 Diesel Generator Room

F. Southwest Cable Penetrating Rooms 141 and 250

G. Cable Way Room 328

H. Intake Structure Room 136 and 136A

I. North Cable Penetration Room 332

J. Electrical Equipment Room 725

K. Charging Pump Rooms 104, 104A and 104B

APPLICABILITY:

Whenever equipment in the sprinkler protected area is required to be operable.

ACTION:

A. With one or more of the above required sprinkler systems inoperable, establish a continuous fire watch with backup fire suppression equipment in the unprotected area(s) within 1 hour. Restore the system(s) to operable status within 14 days, or write a Condition Report to focus management attention on the need to restore the system(s) to an OPERABLE status.

B. When the Limiting Condition for Operation and/or associated action requirements cannot be satisfied, any provision relating to the operating restrictions on the Plant are not applicable.

BASIS:

Refer to Basis Section 9.6.7.1.2

(Formerly Technical Specification 3.22.3)

9.6.7.1.4 Fire Hose Stations

The fire hose stations in the following locations shall be OPERABLE;

A. Corridor, Room 239

B. Viewing Gallery, Room 320

C. Corridor, Room 106

D. Corridor, Room 125

E. Fire Hose Hydrant Station #3

F. Turbine Building 590' Col Y-5

G. Turbine Building 590' Col Y-18

H. Spent Fuel Pool, Room 220

I. Turbine Building 609' Col H-9

J. North Stairway in containment 612' level

K. South Stairway in containment 612' level

L. Fire Hose Hydrant Station #5

APPLICABILITY:

Whenever equipment in the area protected by the fire hose station is required to be operable.

ACTION:

A. With the fire hose station inoperable, provide an additional fire hose for the unprotected area at an OPERABLE fire hose station within 1 hour, except J and K listed above.

B. With the fire hose station inside containment (J and K above) inoperable:

   1. when containment integrity is required, provide portable fire fighting equipment (eg, water fire extinguishers) at the entrance to containment within one hour.

   2. when containment integrity is not required, provide portable fire fighting equipment (eg, water fire extinguishers) at the fire hose station within one hour.

C. When this Limiting Condition for Operation and/or associated actions requirement cannot be satisfied, provisions relating to operating restrictions on the Plant are not applicable.

BASIS:

Refer to Basis Section 9.6.7.1.2.

(Formerly Technical Specification 3.22.4)

9.6.7.1.5 Fire Rated And Fire Protection Assemblies

All fire rated assemblies (walls, floors, ceilings, cable tray enclosures, cable wraps), fire protection assemblies (radiant heat shields) and fire barrier penetration sealing devices in fire rated assemblies (fire doors, fire dampers, cable, piping and ventilation duct penetration seals) which protect safety related fire areas or separate portions of redundant systems important to safe shutdown within a fire area, shall be OPERABLE.

APPLICABILITY: Whenever equipment on either side of the fire rated assembly is required to be operable

ACTION:

A. With one or more of the above required fire rated assemblies or fire barrier penetration sealing devices inoperable, a continuous fire watch shall be established on at least one side of the affected assembly within one hour, or verify the OPERABILITY of the minimum number of fire detectors or sprinkler flow switches listed in Table 9-11 on at least one side of the inoperable assembly and establish an hourly fire inspection or, install a portable detection system on at least one side of the inoperable assembly per plant procedures and establish an hourly fire inspection after installation.

B. With a fire protection assembly (radiant heat shields) located outside containment rendered inoperable, within one hour establish an hourly fire inspection in that area.

C. With a fire protection assembly located inside containment rendered inoperable, restore the assembly to operable status within 24 hours or be in at least hot standby within the next six (6) hours and in at least cold shutdown within the subsequent 30 hours.

BASIS:

The functional integrity of the penetration fire barriers ensures that fires will be confined or adequately retarded from spreading to adjacent portions of the facility. This design feature minimizes the possibility of a single fire rapidly involving several areas of the facility prior to detection and extinguishment. The penetration fire barriers are a passive element in the facility Fire Protection Program and are subject to periodic inspections.

(Formerly Technical Specification 3.22.5)

9.6.7.2 Testing Requirements

9.6.7.2.1 Fire Detection Instrumentation

Fire detection instruments located inside containment and associated alarms identified in Table 9-11 shall be demonstrated operable at least once each refueling outage.

Fire detection instruments located outside containment and associated alarms identified in Table 9-11 shall be demonstrated as operable at least once semiannually.

(Formerly Technical Specification 4.17.1)

9.6.7.2.2 Fire Suppression Water System

9.6.7.2.2.1 Fire Pump, Valve, and Hydrant Testing

The fire suppression water system shall be demonstrated OPERABLE:

A. At least monthly by starting each pump and operating it at least 15 minutes

B. At least monthly by verifying that each valve (manual, power operated or automatic) in the flow path, that is not locked, sealed or otherwise secured in position, is in its correct position

C. At least annually by performance of a system flush of the fire water hydrants

D. At least annually by cycling each testable valve in the flow path through at least one complete cycle of full travel

E. At least once each 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its operating sequence, and:

   1. verifying that each pump develops at least 1,500 gpm at 125 psig

   2. verifying that each pump starts (sequentially) to maintain the fire suppression water system pressure ≥ 90 psig

F. At least every 3 years by performing a flow test of the system in accordance with Chapter 5,Section 11, of the Fire Protection Handbook, 14th edition, published by the National Fire Protection Association.

9.6.7.2.2.2 Fire Pump Diesel Engine and Battery Testing

The (2) fire pump diesel engines and (2) starting 24 volt battery banks and charger shall be demonstrated operable:

A. At least every 7 days by verifying that:

1. the electrolyte level of each battery is above the plates, and

2. the overall battery voltage is ≥ 24 volts

B. At least every 3 months by verifying that:

1. a sample of diesel fuel oil from the main storage tank (T-10A) obtained in accordance with ASTM D4057-88, is within the acceptable limits with respect to kinematic viscosity, water content and sediment, as shown below:

Method	Test Method	Limit
kinematic viscosity	ASTM D445-88	2.0 - 4.3 cSt
water content and sediment	ASTM D2709-88	<0.05%

2. the specific gravity of the starting battery bank is appropriate for continued service of the battery.

C. At least every 18 months by verifying that:

1. the batteries, cell plates and battery racks show no visual indication of physical damage or abnormal deterioration, and

2. the battery-to-battery and terminal connections are clean, tight, free of corrosion and coated with anti-corrosive material.

NOTE: A diesel fire pump with an inoperable battery or battery bank can still be considered operable as long as the following are met:

1. One of the two battery banks on the diesel pump remains operable, and

2. The diesel fire pump is proven capable of staring and operating for a minimum time of not less than 15 minutes.

D. At least every 18 months, during shutdown, by:

1. subjecting the diesels to an inspection in accordance with procedures prepared in conjunction with the manufacturer's recommendations for the class of service.

(Formerly Technical Specification 4.17.2)

9.6.7.2.3 Fire Sprinkler Systems

The sprinkler systems defined in Section 9.6.7.1.3 shall be demonstrated OPERABLE to verify no blockage every 18 months by visual inspection of each accessible sprinkler head.

(Formerly Technical Specification 4.17.3)

9.6.7.2.4 Fire Hose Stations

9.6.7.2.4.1 Fire Hose Station Inspections During Plant Power Operations

Each fire hose station defined in Section 9.6.7.1.4 A through I and L shall be verified to be OPERABLE:

A. At least monthly by visual inspection of the station to assure all equipment is available;

B. At least every 18 months by removing the hose for inspection and reracking and replacing all gaskets in the couplings, as required;

C. At least every 3 years by:

1. partially opening each hose station valve to verify valve operability and no flow blockage, and

2. conducting a hose hydrostatic test at a pressure of 150 psig or at least 50 psig greater than the maximum fire main operating pressure, which ever is greater.

9.6.7.2.4.2 Fire Hose Station Inspections During Refueling Outages

Each fire hose station defined in Section 9.6.7.1.4, J and K shall be verified to be operable:

A. At least monthly during normal refueling outage by visual inspection of the station to assure all equipment is available:

B. At least each refueling outage by removing the hose for inspection and replacing all gaskets in the couplings, as required;

C. At least every 3 years by:

1. partially opening each hose station valve to verify valve operability and no flow blockage, and

2. Conducting a hose hydrostatic test at a pressure 150 psig or at least 50 psig greater than the maximum fire main operating pressure, which ever is greater.

(Formerly Technical Specification 4.17.4)

9.6.7.2.5 Fire Rated And Fire Protection Assemblies

9.6.7.2.5.1 Fire Rated and Fire Protection Assembly Inspections

Fire rated assemblies, fire protection assemblies and fire barrier penetration sealing devices (except fire doors), shall be verified OPERABLE at least once per 18 months by performing an inspection of:

A. The exposed surfaces of each fire rated assembly (visual).

B. The structural integrity of fire protection assemblies (visual).

C. All fire barrier penetration sealing devices (visual).

D. Fire dampers and associated hardware (functional where practicable).

9.6.7.2.5.2 Fire Door Inspections

At least once per six months, all fire doors shall be verified OPERABLE by visually inspecting the structural integrity, automatic hold-open, release, closing mechanism and latches and by verifying:

A. At least once per 31 days, the OPERABILITY of the fire door supervision system for each electrically supervised fire door.

B. At least once per seven days that each locked closed fire door is closed.

C. At least once per 24 hours, that doors with automatic hold-open and release mechanisms are free of obstructions.

D. At least once per 24 hours, that each unlocked fire door without electrical supervision is closed.

(Formerly Technical Specification 4.17.5.)

9.6.7.2.6 Emergency Lighting

Each DC light powered by station batteries outside of containment shall be verified OPERABLE at least once a year. These same type of lights inside of containment shall be verified OPERABLE prior to removal of the reactor head.

Each battery powered Emergency Lighting Unit (ELU) shall be verified functional at least once per year.

Technical Specification 4.7.3

Demonstration of OPERABILITY shall be performed every 24 months to prove minimum lighting duration of eight hours.

9.6.7.3 Fire Brigade

A Fire Brigade of at least 5 members shall be maintained on site at all times.* The Fire Brigade shall not include 3 members of the minimum shift crew necessary for safe shutdown or any personnel required for other essential functions during a fire emergency.

* The Fire Brigade composition may be less than the minimum requirements for a period of time not to exceed two hours in order to accommodate unexpected absence provided immediate action is taken to restore the minimum requirements.

(Formerly Technical Specification 6.2.2.e)

9.6.7.4 Training

The Palisades Fire Protection Staff is responsible for the development, revision, approval, and implementation of the fire brigade training program. The program shall, as practical, meet or exceed the requirements of NFPA 27-1975. Fire brigade drills shall be held at least quarterly.

(Formerly Technical Specification 6.4.2)

9.6.8 POST FIRE SAFE SHUTDOWN ANALYSIS

9.6.8.1 Introduction

This section summarizes the results presented in numerous plant design documents, including drawings, engineering analyses, and the Appendix R Program Manager database, which collectively comprise the Palisades plant post-fire safe shutdown analysis. This summary presents the highlights and significant conclusions of the safe shutdown analysis. It demonstrates that the Palisades plant can be safely shut down following a fire in any area of the plant.

9.6.8.2 Methodology

The general methodology for performing the post-fire safe shutdown analysis consisted of the following discrete steps: identify/verify fire area boundaries; Identify safe shutdown systems and components; identify safe shutdown circuits and their associated fault consequences; identify raceway locations; and determine fire consequences and compliance strategies. Each of these steps is discussed in more detail below.

9.6.8.2.1 Identify Fire Areas and Fire Area Boundaries

Existing plant fire area boundary drawings were reviewed to verify that the fire area boundaries were explicitly and completely shown. All safe shutdown components and cables were verified to have assigned room numbers and fire areas.

9.6.8.2.2 Identify Safe Shutdown Systems and Components

9.6.8.2.2.1 Performance Goals

The safe shutdown performance goals establish the overall functional criteria for achieving safe shutdown following a fire. Achievement of these performance goals ensures that the reactor will be safely shut down, cooled down, and maintained in a cold shutdown condition. The specific performance goals that satisfy the criteria for safe shutdown in the event of a fire are:

a. Reactor Reactivity Control
b. Reactor Coolant Inventory Control
c. Reactor Coolant Pressure Control
d. Decay Heat Removal
e. Process Monitoring
f. Support Functions

9.6.8.2.2.2 Methodology

Using the performance goals identified above, the systems or portions of systems that will be used to achieve these goals were identified. This was accomplished by a detailed review of various plant design documents, licensing basis documents, and operating and/or maintenance procedures.

Once the required systems were identified, the P&ID's were reviewed to identify specific components within those systems that are necessary to accomplish the safe shutdown functions. When the list of mechanical systems and components was completed, support systems and components, including power supplies, were identified and added to the list.

Finally, instrumentation required for process monitoring and support was identified and added to the list.

9.6.8.2.2.3 Safe Shutdown Logic Diagrams

The systems and components identified as required for safe shutdown are depicted on the safe shutdown logic diagrams. An overall systems level logic diagram depicts the systems which will be utilized to achieve the safe shutdown performance goals listed above. Component level logic diagrams depict the individual components within each system that will be utilized to achieve the safe shutdown performance goals.

9.6.8.2.2.4 Safe Shutdown Equipment List

A Palisades plant safe shutdown equipment list (SSEL) was developed. In addition to the Equipment ID number, component description, and system, the SSEL also lists the plant location (room number) of the component, whether it is part of a high-low pressure interface between the primary coolant system and another system, whether it is needed only for cold shutdown, the P&ID or single line diagram on which it is shown, and the safe shutdown logic diagram on which it is shown. The components' normal position, fail air position (if applicable), fail electric position (if applicable), the desired hot standby safe shutdown position, and the desired cold shutdown safe shutdown position are also identified in the SSEL.

9.6.8.2.3 Circuit Analysis and Raceway Location Identification

9.6.8.2.3.1 Circuit Analysis

The circuit analysis for safe shutdown components was performed using the methodology summarized as described below.

For each electrically powered or controlled component listed in the SSEL, the applicable electrical design drawings, such as schematic diagrams, single-line diagrams, control wiring diagrams, instrument loop diagrams, or vendor drawings, were reviewed and the cables which are associated with that component were identified. The basis for cable identification was the Palisades Plant Cable and Raceway Schedule (CRS).

Each cable was evaluated to determine the effect of each possible fire induced failure mode on the components' circuit, and the resulting fault consequences were identified. Fire induced cable failure modes considered include short circuits, hot shorts, open circuits, and shorts to ground, as appropriate. Fault consequences are dependent on the type of component, but in general include failures such as loss of power, loss of control, and spurious actuation/operation.

If a fire induced failure of the cable could place the component in a position other than its desired safe shutdown position, or could prevent the component from being placed in its desired safe shutdown position, then the cable was identified as an Appendix R cable.

9.6.8.2.3.2 Raceway Location Identification

The detailed plant raceway location and verification effort was documented and the process is summarized as described below.

Using the fire area/room boundary drawings, the plant electrical location drawings were marked to delineate the room boundaries. For each Appendix R cable identified by the circuit analysis described above, the raceways through which the cable is routed were identified. Next, the rooms through which the raceways containing Appendix R cables were routed were identified.

In order to obtain a higher level of confidence in the accuracy of the cable routing documentation, each cable's routing was verified on the plant electrical location drawings from its starting to its ending point. In the event of either ambiguity in the cable routing, or incomplete or contradictory plant documentation of the routing, the discrepancy was further evaluated and resolved.

9.6.8.2.4 Safe Shutdown Fire Area Analysis

The detailed methodology used to prepare the post fire safe shutdown assessment is summarized as described below.

For each fire area in the plant, a listing of all safe shutdown components located in the area and all Appendix R cables routed through the area was generated. Using this listing, a set of the safe shutdown logic diagrams was marked up to cross out affected components and annotated with a code which indicated the failure mechanism. Cascaded failures resulting from either power supply losses or air supply losses were then added to the listing manually. For example, if a power supply such as a motor control center were disabled as a result of fire damage to one of its feed cables routed through an area, all of the safe shutdown components powered from that MCC would have to be added to the listing and annotated with the code which indicates that the component was lost because its power supply was lost. Similarly, if an air compressor were lost, those air-operated valves which are connected to the air compressor discharge piping would be added to the listing and annotated with the code that indicates that the component was lost because its air supply was lost.

Once all affected components have been identified, listed, and marked off on the safe shutdown logic diagrams, fire consequences were assigned to each affected component. Fire consequences are standardized descriptions of the effects of fire on the affected component. More than one fire consequence may apply to a particular component.

The affected component listings and marked up logic diagrams were then reviewed to ensure that a success path is available for each safe shutdown function and Appendix R performance goal. Compliance strategies were assigned to each affected component in each fire area. Compliance strategies have been standardized. More than one compliance strategy may be assigned to affected components.

When the task was completed, the list of affected components with assigned fire consequences and compliance strategies were entered into the Appendix R Program Manager database. This database contains the results used to develop post-fire safe shutdown procedures and the basis for review of future plant modifications to ensure the ability of the plant to achieve and maintain cold shutdown in the event of a fire.

9.6.8.3 Reactor Shutdown Controls

This subsection provides a general description of the instrumentation and control features provided in the plant design which may be utilized to accomplish safe shutdown of the plant following a fire in any fire area. The section summarizes the plant specific safe shutdown analysis as provided in Section IV of the Fire Protection Program Report, Post-Fire Safe Shutdown Analysis (PSSA).

Certain fire areas of the plant are identified as alternate shutdown fire areas, and may require that shutdown be accomplished from outside the control room, due to postulated fire damage to instrumentation and controls located in the main control room. The following areas of the plant are currently identified as such: Control Room complex (Fire Area 1), Cable Spreading Room (Fire Area 2), 1-D Switchgear Room (Fire Area 3), Auxiliary Building (formerly referred to as the 590' corridor) (Fire Area 13), and the Electrical Equipment Room (Fire Area 21).

The Auxiliary Hot Shutdown Panels, C-150/C-150A, have been provided as a centralized location for controlling safe shutdown of the plant for the alternate shutdown fire areas described above. These panels are described in Subsection 7.7.4. In the following discussions, reference to "Panel C-150" indicates a combined reference to main panel C-150 and auxiliary panel C-150A.

The safe shutdown performance goals for achieving safe shutdown in the event of a fire are defined in the PSSA. The specific systems and components which will be utilized to accomplish those goals for specific fires are also described in the PSSA. The following sections provide general discussions of the instrumentation and controls which may be utilized to accomplish those functions. Refer to the PSSA for details.

9.6.8.3.1 Safe Shutdown Offsite and Onsite Power

Offsite and onsite power system features are discussed in Chapter 8.0.

In general, the PSSA assumes that offsite power will not be available following a fire in any fire area of the plant. Fires in many fire areas of the plant are postulated to cause a loss of offsite power as a result of fire damage to offsite power feeder cables or breaker control cables. Such areas include, but are not limited to the control room, the cable spreading room, and the turbine building. In some fire areas of the plant, where it is known that cables related to the supply or control of the offsite power supply are not located, the assumption of loss of offsite power is not made. This is in accordance with current regulatory practice.

1. One of the emergency diesel generators is demonstrated to be available to supply essential AC power for all fire areas where a loss of offsite power is assumed to occur.

2. Fires in several plant fire areas may damage circuitry for control of the operation of one or both of the emergency diesel generators, and/or the circuitry for the corresponding 2400 VAC switchgear breakers. Procedures exist for local manual control of the emergency diesel generators and the associated essential 2400 VAC breakers for critical safe shutdown components.

9.6.8.3.2 Reactivity Control and Maintenance of Primary Coolant Inventory

For hot standby operations, credit is taken for reactor trip from the control room for reactivity control at the time of fire initiation/detection.

The primary method of providing both reactivity control and PCS inventory control utilizes the charging system. This involves one of the three charging pumps drawing borated water from either a boric acid tank or from the SIRWT. An alternative method utilizes a HPSI pump in conjunction with a containment spray pump drawing borated water from the SIRWT. An engineering analysis has shown that the HPSI pump by itself can supply the required makeup without use of the containment spray pump. The specific instruments and components within these systems that are utilized for safe shutdown are addressed in the PSSA.

Charging pumps P-55A and P-55B are powered from Class 1E 480 VAC Bus 12, while charging pump P-55C is powered from Class 1E 480 VAC Bus 11. Pumps P-55B and P-55C can be powered from alternative power source 480 VAC Bus 13. This alternative source is credited for some fire areas.

Motor-operated valves within these systems can be locally manually operated using their handwheels in the event of control circuit damage. Essential air-operated valves within these systems can be placed in their desired positions by one of the following methods: 1) failing the electrical power source, causing the valve to assume its loss of power position, or 2) locally failing the air supply to the valve, causing the valve to assume its loss of air position, or 3) locally manipulating manual air supply valves to force the valve to its desired position (this capability has only been provided for specific AOV's, and is not generally available).

Existing procedures provide operator guidance for the maintenance of reactivity control and PCS inventory control using those systems and components credited in the Appendix R post-fire safe shutdown analysis.

9.6.8.3.3 Primary Coolant System Pressure Control

The plant will control pressure in the primary coolant system by means of the pressurizer heaters, if available. This is the preferred method of pressure control. A portion of the pressurizer heaters have a power source capable of being fed from the emergency generators (see Subsection 8.4.4). If the heaters are not available, the charging system can be used to maintain the PCS pressure within its desired operating range. The specific instruments and components within these systems that are utilized for safe shutdown are addressed in the PSSA.

Depressurization capability, if required, is provided by the pressurizer PORV's and block valves, or by the pressurizer aux spray. If the pressurizer heaters have been secured, then depressurization capability will generally not be required for PCS pressure control.

Existing procedures provide operator guidance for the maintenance of PCS pressure control using those systems and components credited in the Appendix R post-fire safe shutdown analysis.

9.6.8.3.4 Reactor Decay Heat Removal - Hot Standby/Hot Shutdown

The systems and equipment which may be utilized for decay heat removal while maintaining the plant at hot standby or hot shutdown include the auxiliary feedwater system, the main feedwater and condensate systems, the main steam system, the atmospheric steam dump valves (ASDV's), the hogging air ejector and the steam generator code safety valves. Instrumentation required includes the primary coolant hot and cold leg temperature, and steam generator level and pressure. The specific instruments and components within these systems that are utilized for safe shutdown are addressed in the PSSA. Details of the auxiliary feedwater controls and instrumentation are provided in Subsection 7.4.3.

The reactor possesses an alternate method of removing decay heat in the event of an extended loss of feedwater, besides the auxiliary feedwater system. The method is called "feed and bleed". It uses the HPSI system and/or the charging pumps to add coolant (feed) at high pressure to the primary coolant system. Decay heat increases the primary coolant pressure and energy is removed through the power operated relief valves and/or the PCS safety valves (bleed). This method is not credited for post-fire safe shutdown for any fire area in the current analysis.

Existing procedures provide operator guidance for decay heat removal using those systems and components credited in the Appendix R post-fire safe shutdown analysis.

9.6.8.3.5 Pressure Reduction and Cooldown

In the transition from hot shutdown to cold shutdown, decay heat removal is accomplished by feeding either or both steam generators via the auxiliary feedwater system with water from the condensate storage tank. The fire pumps and/or service water system provide an emergency backup source of water to the auxiliary feedwater pumps' suction in the event of depletion of condensate storage tank inventory. Steam release to the atmosphere is accomplished using the ASDV's or the hogging air ejector. This is the same method and systems/components used for hot standby/hot shutdown decay heat removal.

9.6.8.3.6 Reactor Decay Heat Removal - Cold Shutdown

When the primary coolant system pressure has been reduced to below 250 psig, then one or both of the low pressure safety injection (LPSI) pumps will be used to circulate primary coolant through the shutdown cooling (SDC) heat exchangers. Decay heat will be transferred to the component cooling system through the SDC heat exchangers, and subsequently to the service water system through the component cooling heat exchangers.

Motor-operated valves within these systems can be locally manually operated using their handwheels in the event of control circuit damage. Essential air-operated valves within these systems can be placed in their desired positions by one of the following methods: 1) failing the electrical power source, causing the valve to assume its loss of power position, or 2) locally failing the air supply to the valve, causing the valve to assume its loss of air position, or 3) locally operating the valve using its handwheel (this capability has only been provided for specific AOV's, and is not generally available), or 4) isolating the flowpath using a manual valve.

Credit is taken for making repairs to certain cold shutdown components for some fire areas, in accordance with the guidance provided for Appendix R. In all such cases, the required repair materials are maintained onsite, procedures exist to govern the repair, and adequate manpower can be provided within the required time frame to accomplish the repair and achieve cold shutdown within 72 hours. Refer to the PSSA for fire areas and components where repairs are credited.

Existing procedures provide operator guidance for decay heat removal using those systems and components credited in the Appendix R post-fire safe shutdown analysis.

9.6.8.3.7 Support Functions

Required support systems are dependent on the specific safe shutdown function being performed and on the specific plant system selected to perform the function. Plant systems required to support safe shutdown operations include the component cooling water system, the service water system, the instrument air system and the miscellaneous gas system (these provide the air supply for AOV's), the fuel oil system, the fire protection system (this provides an emergency backup source of water to the service water system, and an emergency backup source of feedwater to the auxiliary feedwater pump suction), and selected plant ventilation systems. Ventilation systems required for equipment operability and human habitability include the control room ventilation system, the switchgear and cable spreading room ventilation system, and the diesel generator room ventilation systems. Ventilation systems required solely for human habitability include the engineered safeguards room ventilation systems and the containment air coolers (CAC's). The specific instruments and components within these systems that are utilized for safe shutdown are addressed in the PSSA.

Existing procedures provide operator guidance for the operation of those systems and components credited in the Appendix R post-fire safe shutdown analysis.

9.6.8.3.8 Shutdown Process Monitoring

Process monitoring instrumentation credited for post-fire safe shutdown includes: pressurizer level and pressure, steam generator level and pressure, primary coolant system hot and cold leg temperature, source range neutron flux, charging line flow, shutdown cooling heat exchanger outlet temperature, LPSI pump discharge pressure, and SIRWT level. The specific instruments that are credited for safe shutdown are addressed in the PSSA.

9.6.9 FIRE AREA SUMMARY

Provided below is a summary of the fire protection features for each plant Fire Area. The key features are the rooms defining the area boundaries and fire protection features. In addition, the listing includes the applicability of 10CFR50, Appendix R exemptions and the location of post-fire safe shutdown equipment within the area.

9.6.9.1 Fire Area #1 Control Room

1. Fire Zones in Fire Area: None

2. Rooms in Fire Area: 320, 321, 322, 323, 324, and 325

3. Automatic Fire Suppression: No

4. Automatic Fire Detection: Yes. FSAR Section 9.6.7.1.1 is applicable

5. Fire Barriers: Adequate for the hazard in this area.

6. Exemption applicable to Fire Area: Yes

7. 86-10 Evaluation for Fire Area: Yes

8. Safe Shutdown Equipment in Fire Area: Yes

9.6.9.2 Fire Area #2 Cable Spreading Room

1. Fire Zones in Fire Area: None

2. Rooms in Fire Area: 224

3. Automatic Fire Suppression: Yes, FSAR Section 9.6.7.1.3 is applicable

4. Automatic Fire Detection: Yes. FSAR Section 9.6.7.1.1 is applicable

5. Fire Barriers: Adequate for the hazard in this area.

6. Exemption applicable to Fire Area: None

7. 86-10 Evaluation for Fire Area: Yes

8. Safe Shutdown Equipment in Fire Area: Yes

9.6.9.3 Fire Area 3 1-D Switchgear Room

1. Fire Zones in Fire Area: 3A and 3B

2. Rooms in Fire Area: Fire Zone 3A: 223
Fire Zone 3B: 328 and 332

3. Automatic Fire Suppression: Yes, FSAR Section 9.6.7.1.3 is applicable

4. Automatic Fire Detection: Yes. FSAR Section 9.6.7.1.1 is applicable

5. Fire Barriers: Adequate for the hazard in this area.

6. Exemption applicable to Fire Area: None

7. 86-10 Evaluation for Fire Area: Yes

8. Safe Shutdown Equipment in Fire Area: Yes

9.6.9.4 Fire Area #4 1-C Switchgear Room

1. Fire Zones in Fire Area: None

2. Rooms in Fire Area: 116A

3. Automatic Fire Suppression: Yes, FSAR Section 9.6.7.1.3 is applicable

4. Automatic Fire Detection: Yes. FSAR Section 9.6.7.1.1 is applicable

5. Fire Barriers: Adequate for the hazard in this area.

6. Exemption applicable to Fire Area: None

7. 86-10 Evaluation for Fire Area: Yes

8. Safe Shutdown Equipment in Fire Area: Yes

9.6.9.5 Fire Area #5 Diesel Room 1-1

1. Fire Zones in Fire Area: None

2. Rooms in Fire Area: 116 and 348

3. Automatic Fire Suppression: Yes, FSAR Section 9.6.7.1.3 is applicable

4. Automatic Fire Detection: No

5. Fire Barriers: Adequate for the hazard in this area.

6. Exemption applicable to Fire Area: None

7. 86-10 Evaluation for Fire Area: Yes

8. Safe Shutdown Equipment in Fire Area: Yes

9.6.9.6 Fire Area #6 Diesel Room 1-2

1. Fire Zones in Fire Area: None

2. Rooms in Fire Area: 116B, 148, 149, and 349

3. Automatic Fire Suppression: Yes, FSAR Section 9.6.7.1.3 is applicable

4. Automatic Fire Detection: No

5. Fire Barriers: Adequate for the hazard in this area.

6. Exemption applicable to Fire Area: None

7. 86-10 Evaluation for Fire Area: Yes

8. Safe Shutdown Equipment in Fire Area: Yes

9.6.9.7 Fire Area #7 Diesel Generator Day Tank 1-1

1. Fire Zones in Fire Area: None

2. Rooms in Fire Area: 146

3. Automatic Fire Suppression: No

4. Automatic Fire Detection: No

5. Fire Barriers: Adequate for the hazard in this area.

6. Exemption applicable to Fire Area: None

7. 86-10 Evaluation for Fire Area: None

8. Safe Shutdown Equipment in Fire Area: Yes

9.6.9.8 Fire Area #8 Diesel Generator Day Tank 1-2

1. Fire Zones in Fire Area: None

2. Rooms in Fire Area: 147

3. Automatic Fire Suppression: No

4. Automatic Fire Detection: No

5. Fire Barriers: Adequate for the hazard in this area.

6. Exemption applicable to Fire Area: None

7. 86-10 Evaluation for Fire Area: None

8. Safe Shutdown Equipment in Fire Area: Yes

9.6.9.9 Fire Area #9 Intake Structure

1. Fire Zones in Fire Area: None

2. Rooms in Fire Area: 136

3. Automatic Fire Suppression: Yes, FSAR Section 9.6.7.1.3 is applicable

4. Automatic Fire Detection: Yes. FSAR Section 9.6.7.1.1 is applicable

5. Fire Barriers: Adequate for the hazard in this area.

6. Exemption applicable to Fire Area: None

7. 86-10 Evaluation for Fire Area: Yes

8. Safe Shutdown Equipment in Fire Area: Yes

9.6.9.10 Fire Area #10 East Engineered Safeguards Area

1. Fire Zones in Fire Area: None

2. Rooms in Fire Area: 001, 001A, 001B, and 004

3. Automatic Fire Suppression: No

4. Automatic Fire Detection: Yes. FSAR Section 9.6.7.1.1 is applicable

5. Fire Barriers: Adequate for the hazard in this area.

6. Exemption applicable to Fire Area: None

7. 86-10 Evaluation for Fire Area: Yes

8. Safe Shutdown Equipment in Fire Area: Yes

9.6.9.11 Fire Area #11 Battery Room A

1. Fire Zones in Fire Area: None

2. Rooms in Fire Area: 225

3. Automatic Fire Suppression: No

4. Automatic Fire Detection: Yes. FSAR Section 9.6.7.1.1 is applicable

5. Fire Barriers: Adequate for the hazard in this area.

6. Exemption applicable to Fire Area: None

7. 86-10 Evaluation for Fire Area: Yes

8. Safe Shutdown Equipment in Fire Area: Yes

9.6.9.12 Fire Area #12 Battery Room B

1. Fire Zones in Fire Area: None

2. Rooms in Fire Area: 225A

3. Automatic Fire Suppression: No

4. Automatic Fire Detection: Yes. FSAR Section 9.6.7.1.1 is applicable

5. Fire Barriers: Adequate for the hazard in this area.

6. Exemption applicable to Fire Area: None

7. 86-10 Evaluation for Fire Area: None

8. Safe Shutdown Equipment in Fire Area: Yes

9.6.9.13 Fire Area #13 Aux. Bldg. El.590', Charging Pump Room, Waste Gas Room, Decontamination Room, Waste Gas Processing Room, Boric Acid Equip. Room, Spent Fuel Pool Equipment Room.

1. Fire Zones in Fire Area: 13A, 13B, 13C, 13D, 13E, 13F and 13G

2. Rooms in Fire Area:

   Fire Zone 13A: 106, 121A, 121B, 122, 150, 227, 228, 233, 234, 709, 741, and 802

   Fire Zone 13B: 100, 104, 104A, 104B, and 117

   Fire Zone 13C: 101, 102, 103, and 151

   Fire Zone 13D: 111 and 112

   Fire Zone 13E: 118, 119, 120 and 120A

   Fire Zone 13F: 107, 107A, 108, 109 and 110

   Fire Zone 13G: 113, 113A, 114 and 115

3. Automatic Fire Suppression: Yes, FSAR Section 9.6.7.1.3 is applicable

4. Automatic Fire Detection: Yes. FSAR Section 9.6.7.1.1 is applicable

5. Fire Barriers: Adequate for the hazard in this area.

6. Exemption applicable to Fire Area: Yes

7. 86-10 Evaluation for Fire Area: Yes

8. Safe Shutdown Equipment in Fire Area: Yes

9.6.9.14 Fire Area #14 Reactor Containment

1. Fire Zones in Fire Area: None

2. Rooms in Fire Area: 000, 003, 008, 142, 142A, 143, 143A, 144, 145, 236, 237, 334, 335, 336, 420, 423, 423A, 424, 800, 801, 810, 820, 821, 822, 823, 901, and 902

3. Automatic Fire Suppression: No

4. Automatic Fire Detection: Yes. FSAR Section 9.6.7.1.1 is applicable

5. Fire Barriers: Adequate for the hazard in this area.

6. Exemption applicable to Fire Area: Yes

7. 86-10 Evaluation for Fire Area: Yes

8. Safe Shutdown Equipment in Fire Area: Yes

9.6.9.15 Fire Area #15 Engineered Safeguards Panel Room

1. Fire Zones in Fire Area: None

2. Rooms in Fire Area: 002, 016, and 121

3. Automatic Fire Suppression: No

4. Automatic Fire Detection: Yes. FSAR Section 9.6.7.1.1 is applicable

5. Fire Barriers: Adequate for the hazard in this area.

6. Exemption applicable to Fire Area: Yes

7. 86-10 Evaluation for Fire Area: Yes

8. Safe Shutdown Equipment in Fire Area: Yes

9.6.9.16 Fire Area #16 Component Cooling Room

1. Fire Zones in Fire Area: None

2. Rooms in Fire Area: 123, 238, and 338

3. Automatic Fire Suppression: No

4. Automatic Fire Detection: Yes. FSAR Section 9.6.7.1.1 is applicable

5. Fire Barriers: Adequate for the hazard in this area.

6. Exemption applicable to Fire Area: None

7. 86-10 Evaluation for Fire Area: Yes

8. Safe Shutdown Equipment in Fire Area: Yes

9.6.9.17 Fire Area #17 Refueling and Spent Fuel Pool Area

1. Fire Zones in Fire Area: None

2. Rooms in Fire Area: 219, 220, 221, 222, 333, 418, 419, 421, 422, 501, 735, 736, 737, and 901

3. Automatic Fire Suppression: No

4. Automatic Fire Detection: Yes. FSAR Section 9.6.7.1.1 is applicable

5. Fire Barriers: Adequate for the hazard in this area.

6. Exemption applicable to Fire Area: None

7. 86-10 Evaluation for Fire Area: Yes

8. Safe Shutdown Equipment in Fire Area: Yes

9.6.9.18 Fire Area #18 Demineralizer Area

1. Fire Zones in Fire Area: None

2. Rooms in Fire Area: 153, 154, 229, 230, 231, 232, and 235

3. Automatic Fire Suppression: No

4. Automatic Fire Detection: No

5. Fire Barriers: Adequate for the hazard in this area.

6. Exemption applicable to Fire Area: None

7. 86-10 Evaluation for Fire Area: Yes

8. Safe Shutdown Equipment in Fire Area: Yes

9.6.9.19 Fire Area #19 Compactor-Track Alley

1. Fire Zones in Fire Area: None

2. Rooms in Fire Area: 329

3. Automatic Fire Suppression: Yes, FSAR Section 9.6.7.1.3 is not applicable

4. Automatic Fire Detection: No

5. Fire Barriers: Adequate for the hazard in this area.

6. Exemption applicable to Fire Area: None

7. 86-10 Evaluation for Fire Area: None

8. Safe Shutdown Equipment in Fire Area: None

9.6.9.20 (Fire Area #20 reserved for future use)

9.6.9.21 Fire Area #21 Electrical Equipment Room

1. Fire Zones in Fire Area: None

2. Rooms in Fire Area: 725

3. Automatic Fire Suppression: Yes, FSAR Section 9.6.7.1.3 is applicable

4. Automatic Fire Detection: Yes. FSAR Section 9.6.7.1.1 is applicable

5. Fire Barriers: Adequate for the hazard in this area.

6. Exemption applicable to Fire Area: None

7. 86-10 Evaluation for Fire Area: Yes

8. Safe Shutdown Equipment in Fire Area: Yes

9.6.9.22 Fire Area #22 Turbine Lube Oil Room

1. Fire Zones in Fire Area: None

2. Rooms in Fire Area: 132

3. Automatic Fire Suppression: Yes, FSAR Section 9.6.7.1.3 is not applicable

4. Automatic Fire Detection: No

5. Fire Barriers: Adequate for the hazard in this area.

6. Exemption applicable to Fire Area: None

7. 86-10 Evaluation for Fire Area: Yes

8. Safe Shutdown Equipment in Fire Area: None

9.6.9.23 Fire Area #23 Condensate Pump Room, Steam Generator Feed Pump Area, Main Generator Seal Oil System Area, Turbine Building General, and Gas Storage West Room

1. Fire Zones in Fire Area: 23A, 23B, 23C, 23D, and 23E

2. Rooms in Fire Area:

   Fire Zone 23A: 006

   Fire Zone 23B: 131

   Fire Zone 23C: 133 and 133A

   Fire Zone 23D: 125, 126, 127, 128, 129, 130, 134, 135, 137, 239, 240, 241, 242, 243, 244, 245, 246, 247, 248, 249, 251, and 337

   Fire Zone 23E: 140

3. Automatic Fire Suppression: Yes, FSAR Section 9.6.7.1.3 is not applicable

4. Automatic Fire Detection: No

5. Fire Barriers: Adequate for the hazard in this area.

6. Exemption applicable to Fire Area: None

7. 86-10 Evaluation for Fire Area: Yes

8. Safe Shutdown Equipment in Fire Area: Yes

9.6.9.24 Fire Area #24 Auxiliary Feedwater Pump Room

1. Fire Zones in Fire Area: None

2. Rooms in Fire Area: 007

3. Automatic Fire Suppression: No

4. Automatic Fire Detection: Yes. FSAR Section 9.6.7.1.1 is applicable

5. Fire Barriers: Adequate for the hazard in this area.

6. Exemption applicable to Fire Area: None

7. 86-10 Evaluation for Fire Area: Yes

8. Safe Shutdown Equipment in Fire Area: Yes

9.6.9.25 Fire Area #25 North and South Heating Boiler Rooms

1. Fire Zones in Fire Area: 25A and 25B

2. Rooms in Fire Area: Fire Zone 25A: 124

   Fire Zone 25B: 707

3. Automatic Fire Suppression: Yes, FSAR Section 9.6.7.1.3 is not applicable

4. Automatic Fire Detection: None

5. Fire Barriers: Adequate for the hazard in this area.

6. Exemption applicable to Fire Area: None

7. 86-10 Evaluation for Fire Area: None

8. Safe Shutdown Equipment in Fire Area: Yes

9.6.9.26 Fire Area #26 Southwest Cable Penetration Room

1. Fire Zones in Fire Area: None

2. Rooms in Fire Area: 141 and 250

3. Automatic Fire Suppression: Yes, FSAR Section 9.6.7.1.3 is applicable

4. Automatic Fire Detection: Yes, FSAR Section 9.6.7.1.1 is applicable

5. Fire Barriers: Adequate for the hazard in this area.

6. Exemption applicable to Fire Area: None

7. 86-10 Evaluation for Fire Area: Yes

8. Safe Shutdown Equipment in Fire Area: Yes

9.6.9.27 Fire Area #27 Radwaste Facilities Building VRS Rooms

1. Fire Zones in Fire Area: None

2. Rooms in Fire Area: 152, 700, 701, 702, 703, 704, 705, 706, 708, 710, 711, 712, 713, 714, 715, 716, 717, 718, 719, 720A, 720B, 721, 722, 723, 724, 726, 730, 730A, 730B, 730C, 730D, 730E, 731, 732, 733, 754, 755, 756, 757, 758, 759, 760, 761, and 762

3. Automatic Fire Suppression: Yes, FSAR Section 9.6.7.1.3 is not applicable

4. Automatic Fire Detection: Yes. FSAR Section 9.6.7.1.1 is not applicable

5. Fire Barriers: Adequate for the hazard in this area.

6. Exemption applicable to Fire Area: None

7. 86-10 Evaluation for Fire Area: Yes

8. Safe Shutdown Equipment in Fire Area: None

9.6.9.28 Fire Area #28 West Engineered Safeguards Room

1. Fire Zones in Fire Area: None

2. Rooms in Fire Area: 005

3. Automatic Fire Suppression: No

4. Automatic Fire Detection: Yes. FSAR Section 9.6.7.1.1 is applicable

5. Fire Barriers: Adequate for the hazard in this area.

6. Exemption applicable to Fire Area: None

7. 86-10 Evaluation for Fire Area: Yes

8. Safe Shutdown Equipment in Fire Area: Yes

9.6.9.29 Fire Area #29 Center Mechanical Equipment Room

1. Fire Zones in Fire Area: None

2. Rooms in Fire Area: 300B

3. Automatic Fire Suppression No:

4. Automatic Fire Detection: Yes. FSAR Section 9.6.7.1.1 is applicable

5. Fire Barriers: Adequate for the hazard in this area.

6. Exemption applicable to Fire Area: None

7. 86-10 Evaluation for Fire Area: Yes

8. Safe Shutdown Equipment in Fire Area: Yes

9.6.9.30 Fire Area #30 East Mechanical Equipment Room

1. Fire Zones in Fire Area: None

2. Rooms in Fire Area: 300A and 400A

3. Automatic Fire Suppression: No

4. Automatic Fire Detection: Yes. FSAR Section 9.6.7.1.1 is applicable

5. Fire Barriers: Adequate for the hazard in this area.

6. Exemption applicable to Fire Area: None

7. 86-10 Evaluation for Fire Area: Yes

8. Safe Shutdown Equipment in Fire Area: Yes

9.6.9.31 Fire Area #31 West Mechanical Equipment Room

1. Fire Zones in Fire Area: None

2. Rooms in Fire Area: 300 and 400

3. Automatic Fire Suppression: No

4. Automatic Fire Detection: Yes. FSAR Section 9.6.7.1.1 is applicable

5. Fire Barriers: Adequate for the hazard in this area.

6. Exemption applicable to Fire Area: No

7. 86-10 Evaluation for Fire Area: Yes

8. Safe Shutdown Equipment in Fire Area: Yes

9.6.9.32 Fire Area #32 SIRW Tank Roof Area

1. Fire Zones in Fire Area: None

2. Rooms in Fire Area: 808 and 809

3. Automatic Fire Suppression: No

4. Automatic Fire Detection: No

5. Fire Barriers: Adequate for the hazard in this area.

6. Exemption applicable to Fire Area: None

7. 86-10 Evaluation for Fire Area: Yes

8. Safe Shutdown Equipment in Fire Area: Yes

9.6.9.33 Fire Area #33 Technical Support Center

1. Fire Zones in Fire Area: None

2. Rooms in Fire Area: 320A

3. Automatic Fire Suppression: No

4. Automatic Fire Detection: Yes. FSAR Section 9.6.7.1.1 is applicable

5. Fire Barriers: Adequate for the hazard in this area.

6. Exemption applicable to Fire Area: None

7. 86-10 Evaluation for Fire Area: Yes

8. Safe Shutdown Equipment in Fire Area: Yes

9.6.9.34 Fire Area #34 Manhole #1

1. Fire Zones in Fire Area: None
2. Rooms in Fire Area: 190
3. Automatic Fire Suppression: No
4. Automatic Fire Detection: No
5. Fire Barriers: Adequate for the hazard in this area.
6. Exemption applicable to Fire Area: None
7. 86-10 Evaluation for Fire Area: Yes
8. Safe Shutdown Equipment in Fire Area: Yes

9.6.9.35 Fire Area #35 Manhole #2

1. Fire Zones in Fire Area: None
2. Rooms in Fire Area: 191
3. Automatic Fire Suppression: No
4. Automatic Fire Detection: No
5. Fire Barriers: Adequate for the hazard in this area.
6. Exemption applicable to Fire Area: None
7. 86-10 Evaluation for Fire Area: Yes
8. Safe Shutdown Equipment in Fire Area: Yes

9.6.9.36 Fire Area #36 Manhole #3

1. Fire Zones in Fire Area: None
2. Rooms in Fire Area: 192
3. Automatic Fire Suppression: No
4. Automatic Fire Detection: No
5. Fire Barriers: Adequate for the hazard in this area.
6. Exemption applicable to Fire Area: None

7. 86-10 Evaluation for Fire Area: Yes

8. Safe Shutdown Equipment in Fire Area: Yes

9.7 AUXILIARY FEEDWATER SYSTEM

9.7.1 DESIGN BASIS

The Auxiliary Feedwater System is designed to provide a supply of feedwater to the steam generators during start-up operations and to remove primary system sensible and decay heat during initial stages of shutdown operations. Equipment in the system is designed to CP Co Design Class 1 requirements (see Reference 3) with the exception of portions of steam supply piping to P-8B. As a result of lessons learned at TMI, the Auxiliary Feedwater System has been upgraded to a safety-related system.

9.7.2 SYSTEM DESCRIPTION AND OPERATION

9.7.2.1 System Description

The Auxiliary Feedwater System (AFW) supplies water to the secondary side of the steam generators for reactor decay heat removal when normal feedwater sources are unavailable. The system originally consisted of one electric motor-driven pump and one turbine-driven pump with piping, valves and associated instrumentation and controls. In 1983, a third high-pressure safety injection pump was converted to AFW service as the second electric motor-driven pump in the AFW system. Piping, valves and controls were added to provide redundancy of supply up to the containment penetrations where the redundant systems merge to form just two AFW lines - one to each steam generator (Figure 9-15). Each of the four lines in the redundant portion of the system, feeding the steam generators, contain one normally-closed, pneumatically-operated flow control valve and two locked open, de-energized motor-operated isolation valves; any one of the three pumps can feed one or both steam generators.

In 1988, flow control bypass valves were added around the flow control valves from P-8C. They were designed to allow continuous auxiliary feedwater at low flow rates during start-up and hot shutdown conditions. They are administratively controlled to operate only when the steam generator is cold or the level in the steam generator is 60% or greater to prevent the potential for water hammer.

All three AFW pumps normally take suction from the condensate storage tank. The minimum amount of water required in the condensate storage tank and primary coolant system makeup tanks combined (100,000 gallons) exceeds the amount needed for 8 hours of auxiliary feedwater pump operation for decay heat removal following a reactor trip. The condensate storage tank level is monitored in the control room. In addition, a low-level switch is provided to alarm at low water level which corresponds to a total tank inventory of 94,280 gallons. The primary system makeup tank provides an additional source of water to the AFW pump suction. A low-level switch is set to alarm at 65,580 gallons (total tank inventory) which assures that the required inventory of

100,000 gallons is available, even under gravity-feed conditions between the tanks (Reference 22). Control valves, CV-2008 and CV-2010 in the gravity feed line, can be manually operated to ensure adequated inventory is available at all times, including during loss of power and loss of instrument air. A crosstie from the fire system provides an additional backup water supply to the AFW pumps P-8A and P-8B. The third pump (Pump C) has additional backup supply water from the Service Water System.

Minimum flow recirculation is provided through breakdown orifices which are designed to pass minimum pump design flow at maximum pressure.

The two original pumps are located in a tornado-proof CP Co Design Class 1 portion of the turbine building. Pump C is located in west engineered safeguards room in the auxiliary building. The supply header from the condensate storage tank and the tank are not protected from tornadoes, but the backup supplies from the diesel engine-driven fire pump and Service Water System are located in a protected area. The discharge header from the auxiliary feedwater pumps in the turbine building to the auxiliary building is buried underground.

9.7.2.2 Component Description

Design ratings and construction of components are shown in Table 9-13.

9.7.2.3 System Operation

During the initial phase of primary system cooldown, the Auxiliary Feedwater System supplies water to the steam generators to remove reactor sensible and decay heat. Core decay heat is transferred from the reactor to the steam generators by natural or forced circulation of the primary coolant. Steam from the steam generators is discharged through the bypass valve to the condenser. The steam can be discharged to the atmosphere in the event that the main condenser is not operable.

Either motor-driven auxiliary feedwater pump can be operated to provide auxiliary feedwater to the steam generators during start-up. However, the flow control valves from P-8C have bypass valves which can control flow at lower flow rates. The level in the steam generators is maintained from the control room by remotely adjusting the auxiliary feedwater control valves in each respective steam generator auxiliary feed header.

The added motor-driven pump (Pump C) or the turbine-driven pump (Pump B) could supply auxiliary feedwater to the steam generators if Auxiliary Feedwater Pump A would fail. In the event of a loss of offsite power , the turbine-driven pump will start automatically or will be started from the control room and is used to supply feedwater to the steam generators. Upon a loss of DC power, the turbine driven pump will start automatically via the diverse start system added for ATWS during the 1990-91 refueling outage. The turbine-driven auxiliary feed pump and auxiliary feedwater control valves can also be operated locally. Driving steam for the turbine is supplied from the main steam header and the turbine exhaust steam is discharged to the atmosphere. The turbine can operate with steam generator pressures down to 38 psig and is protected by a 10% overspeed trip. Steam traps were added in 1989 to resolve concerns regarding the starting of the turbine without draining the steam supply piping and the turbine casing. In 1996, the steam supply piping was reconfigured. A moisture separator was added and the existing steam traps were modified to improve moisture removal.

Auxiliary feedwater flow to the steam generators will be automatically initiated on a low-steam generator water level. The normal valve positions on all valves of the suction side of the pumps, between the condensate storage tank and the pumps, are locked open and the steam admission valves to the turbine-driven pump are closed. The flow control valves fail open and the steam admission valves may be manually operated on the loss of air. The flow control bypass valves fail closed allowing control with the flow control valves. Safety grade flow rate indication for auxiliary feedwater flow to each steam generator is provided in the main control room. In the event of loss of offsite power, the motor-driven auxiliary feedwater pumps are enabled by the sequencers and started upon receipt of an AFAS signal. Power supplies for instrumentation and the motor-driven auxiliary feedwater pumps are discussed in Section 7.4 and Chapter 8, respectively.

In the event of loss or depletion of the water supply from the condensate storage tank, the backup water supplies from the fire system or Service Water System can be utilized by opening the hand valves in the crossties and, in the case of the fire systems, starting one of the fire pumps.

For any condition during which feedwater to the steam generators from the main feedwater pumps is interrupted and the reactor is tripped, sufficient feedwater flow is maintained by the motor-driven AFW pumps or the turbine-driven auxiliary feed pump to remove decay heat from the primary system and maintain the reactor in a safe condition.

In the event a steam line break occurs, the main feedwater pumps are inoperative. The turbine-driven auxiliary feed pump and the motor-driven auxiliary feed pumps are available to be used to maintain shutdown cooling flow to one steam generator. The Feed Only Good Generator (FOGG) actuation system monitors steam generator pressure. The steam line break will result in a lower pressure in the affected steam generator. The FOGG actuation system was designed to terminate AFW flow to that steam generator. However, due to nuclear safety considerations, the automatic isolation feature has been disabled (Reference 2) and the Emergency Operating Procedures direct the operators to isolate the affected steam generator using the flow control valves.

9.7.3 DESIGN ANALYSIS

A loss of feedwater event is the bounding condition for the Auxiliary Feedwater System. For P-8A and P-8B the required flow is 280 gpm (140 to each) at 985 psig to both steam generators or 280 gpm at 985 psig to one steam generator. For Pump P-8C, the required flow for the loss of feedwater event is 280 gpm (140 to each) at 885 psig to both steam generators or 280 gpm at 885 psig to one steam generator. Operation of the turbine bypass system or atmospheric dump valves is required to depressurize to 885 psig. The preceding flowrates will remove decay heat and pump heat from four operating primary coolant pumps.

If offsite power is not available, the primary coolant pumps will trip, reducing the primary system heat load. The above required flow rate maintains sufficient Steam Generator inventory to support decay heat removal with natural ciculation in the primary coolant system.

9.7.4 SYSTEM RELIABILITY

System reliability is achieved by the following features:

1. Two motor-driven and one turbine-driven pumps are provided, any of which satisfy the requirements of primary system cooldown.

2. Pump motor power is supplied from offsite sources with backup supplied from the emergency diesel generators (see Subsection 8.4.1).

3. Steam can be supplied to the turbine-driven pump from either steam generator.

4. The condensate storage tank capacity is 125,000 gallons and is monitored to maintain a minimum storage of 94,280 gallons. A backup supply from the primary system makeup tank, fire and makeup systems, and Service Water System is provided to the auxiliary feed pump suction.

5. The condensate pumps may be used to pump water through the normal feedwater train to the steam generators in the event of a failure of the auxiliary feedwater piping system. The steam generator pressure may be relieved by the steam dump system to accommodate this mode of operation.

A reliability and operability review has been conducted by Consumers Power Company, the NRC and their consultants. The findings demonstrate that the AFW meets the NRC's long-term safety requirements.

9.7.5 TESTS AND INSPECTION

1. The auxiliary feedwater pumps are tested periodically during Plant operation by starting each pump, establishing test reference conditions and monitoring pump performance.

2. Each nonautomatic valve in the flow path that is not locked, sealed or otherwise secured in position is periodically inspected to verify its correct position.

3. The diaphragm-operated flow control valves in the auxiliary feedwater pump discharge piping are exercised periodically during Plant operation to ensure proper functioning.

4. The automatic initiation function of the Auxiliary Feedwater System is periodically tested by simulating a low-steam generator level and observation of pump start. Operability of the diaphragm-operated 4-inch flow control valves is verified by initiating Auxiliary Feedwater flow and observing valve actuation to its correct position or by monitoring for proper Auxiliary Feedwater System flow. Operability of the 1½-inch bypass valves is verified by observing the valves closing when control is taken by the 4-inch flow control valves.

5. Pump and valve operability tests are conducted in accordance with Section XI of the ASME B&PV Code with applicable addenda as modified by relief requests.

6. All Auxiliary Feedwater System components outside containment are accessible for inspection during Plant operation.

7. A 48-hour endurance test has been performed on the auxiliary feedwater pumps. The results demonstrated that the pumps performed in an acceptable manner without exceeding design limits.

9.8 HEATING, VENTILATION AND AIR-CONDITIONING SYSTEM

9.8.1 DESIGN BASIS

1. The Heating, Ventilation and Air-Conditioning System is designed to provide a suitable environment for equipment and personnel. The path of air for ventilating systems in potentially radioactively contaminated areas runs from areas of low activity toward areas of progressively higher activity for ultimate discharge from the Plant via the ventilation stack. The condensate and makeup demineralizer building HVAC system is also designed so air flows from areas of low potential airborne radioactivity to areas of higher potential airborne radioactivity. High-efficiency air filters are provided for the exhaust.

2. The design is based upon the ambient conditions listed in Table 9-15.

3. The containment building, radwaste area and fuel handling area are designed for containment of radioactive particles. The exhaust air from these areas is ducted to high-efficiency filters to assure minimum activity levels for the stack discharge and to maintain containment of radioactive particles in those areas of possible contamination. The fuel handling area also has a charcoal filter in parallel with the high-efficiency filter which may be placed in operation during fuel handling operations or heavy load movements.

4. The control room Heating, Ventilation and Air-Conditioning (HVAC) System was modified in 1983 in response to NUREG-0737, Item III.D.3.4. The design bases for the system are as follows:

   a. During emergency mode operation, the control room HVAC system maintains a .125" of water-positive pressure in the control room, TSC and viewing gallery; it maintains a dry bulb temperature and relative humidity as indicated in Table 9-15. The mechanical equipment room is also maintained under positive pressure.

   b. The control room HVAC system is designed to permit periodic inspection, testing and maintenance of principal components with minimal interruption of normal control room operation.

   c. The control room HVAC system is designed to limit the radiation exposure of control room personnel during any of the postulated design basis accidents within the guidelines of 10 CFR 50, Appendix A, General Design Criterion 19.

   d. Throughout the duration of any design basis accidents, the control room HVAC system maintains control room environmental temperatures suitable for prolonged occupancy and continued operation of safety-related equipment.

e. The failure of an active component in the control room HVAC system, assuming a loss of offsite power, cannot impair the system's ability to meet the design bases discussed in Paragraphs c. and d. above.

f. The control room HVAC system is designed to remain functional during and after a safe shutdown earthquake.

g. The control room HVAC system is designed to permit rapid purging of the control room for smoke removal.

h. The control room HVAC system is designed to remain functional during and after a design basis tornado.

9.8.2 SYSTEM DESCRIPTION AND OPERATION

9.8.2.1 System Description

Plant equipment spaces are ventilated and cooled with ambient outside air, the outdoor design maximum being 95°F. The maximum space temperature varies from 75°F to 110°F except for the engineered safeguards room where the equipment is designed for 135°F and the Fan Room (Room 338) where maximum temperature reaches 120°F. A space temperature of 110°F is the design maximum for personnel occupancy. The ventilation systems are either induced draft using motor-driven roof exhausters or forced draft fan and duct distribution systems. Spot cooling of equipment is used where it is impractical to cool the entire space. Individual systems are as shown on Figures 9-17 through 9-19.

Airflow controllers are used to maintain negative differential pressures in equipment compartments and between controlled and noncontrolled spaces in the auxiliary building and auxiliary building addition. This negative pressure is used to induce infiltration into compartments thus producing a predictable direction of airflow toward areas of increasing radiation hazard. Final exhaust from these potentially contaminable compartments is discharged to atmosphere through the ventilation stack after filtering out radioactive particulate matter in a high-efficiency filter. The fuel handling area exhaust has a charcoal filter in parallel with the high-efficiency filter which is placed in operation during fuel handling operations.

The Plant normally uses extraction steam, from the low-pressure turbine, reduced to 15 psig for Plant heating. The two auxiliary heating boilers are available for use during Plant shutdown. Each boiler has the capacity to satisfy minimum shutdown heating requirements at the minimum outdoor design temperature of -10°F. A third boiler in the feedwater purity building can be tied into the system if required.

The total control room HVAC system, shown schematically in Figure 9-17, Sheets 6 and 7, consists of two trains of air handling, air filtering units and Continuous Air Monitor (CAM) units (Train A and Train B), the purge exhaust system, toilet exhaust system, and associated ductwork, dampers, instruments and controls.

To reduce noise levels, acoustical diffusers were installed at the HVAC duct outlet supplies to the office and Control Room Supervisor's areas. An acoustical silencer was installed in the main supply duct to the Control Room. Both modifications provided effective noise reduction in the main Control Room.

A CP Co Design Class 1 mechanical equipment room (MER) (located above the emergency diesels) is provided to house the HVAC equipment. The equipment room is divided into two compartments separated by a 3-hour fire barrier. All components of one train are located in one compartment to meet the redundancy criteria. The MER is part of the control room envelope.

The major components of Train A are an air handling unit (V-95), a condensing unit (VC-11), one Continuous Air Monitor unit (consists of sampling head RE-1818A, display/processing unit RIA-1818A and air pump P-968A), one charcoal filter unit (VF-26A) and associated fan (V-26A). Dampers associated with Train A are D-1, D-2, D-3, D-4, D-5, D-6, D-7, D-20 and Tornado Dampers TD-1 and TD-4.

The major components of Train B are an air handling unit (V-96), a condensing unit (VC-10), one Continuous Air Monitor unit (consists of sampling head RE-1818B, display/processing unit RIA-1818B and air pump P-968B), a charcoal filter unit (VF-26B) and an associated fan (V-26B). Dampers associated with Train B are D-8, D-9, D-10, D-11, D-12, D-13, D-14, D-21 and Tornado Dampers TD-2 and TD-5.

The purge exhaust system consists of Fan V-94, Isolation Dampers D-15 and D-16 and Tornado Damper TD-3. Toilet exhaust system Fan V-16 has Isolation Dampers D-17 and D-18 and Tornado Damper TD-6.

During normal mode operation, DPIC 1659 and 1660 control positive pressure in the control room envelope with respect to the relative pressure of the south hallway outside the control room viewing gallery.

Four humidity detectors are provided in the containment building to detect leakage from the Primary Coolant System and the main steam lines. The relative humidity measured by these detectors is indicated in the control room.

Cooling water for the containment air coolers, the engineered safeguards room coolers and the condensing units and water chiller is Plant service water.

The ventilation and air-conditioning control systems use pneumatic-type controllers with pneumatic-electric switching devices to interconnect the equipment and the controls. Instrument air for the controllers is taken from the Plant instrument air system.

9.8.2.2 Component Description

The design data for the major components in the control room HVAC system are listed in Table 9-16.

1. Air Handling Units V-95 and V-96

   Each air handling unit consists of a medium efficiency filter, an electric heating coil, a refrigerant cooling coil, a centrifugal fan and associated ductwork, instrumentation and controls. A steam injection grid is located in the supply ductwork after the airflow measuring unit for each air handling system.

2. Condensing Units VC-10 and VC-11

   The water-cooled condensing units supply refrigerant to the cooling coil of Air Handling Units V-95 and V-96, as required. Cooling water to the condensing units is supplied from the Plant service water.

3. Charcoal Filter Units VF-26A and VF-26B

   The 3,200 ft^3/min capacity charcoal pressurization/recirculation filter units are provided for emergency operation. Each filter train consists of medium efficiency prefilters, an electric heating coil, upstream high-efficiency particulate air (HEPA) filters, two banks of 2-inch carbon adsorber trays, downstream HEPA filters, a vaneaxial fan, an electric modulating damper and associated ductwork, instrumentation and controls.

4. Humidifiers VC-12 and VC-13

   A 50 lb/h capacity humidifier is provided for each air handling unit. The humidifier consists of a steam generator, with high water cutoff and steam dispersion tubes for installation in the ductwork.

5. Low Leak Isolation Dampers

Damper D-1 and D-2 are series dampers which provide isolation of the normal air intake on Train A. Dampers D-8 and D-9 are series dampers, of the same design, which provide isolation of the normal air intake on Train B. Dampers D-15 and D-16 are series smoke purge isolation dampers. All of these dampers are leakage class 1B ("Leaktight") and are designed for a maximum leakage of 2 scfm per sq ft. at design operating pressure of 4.0 inches of water. These dampers have air-operated actuators with fail-close features. Reference 24 calculates a conservative total unfiltered inleakage of 68.15 cfm, past the D-1, D-2 and D-15, D-16 dampers set for one E-HVAC train operation. For two-train E-HVAC operation a conservative total unfiltered inleakage of 111.65 cfm occurs past the D-1, D-2 and D-15, D-16 dampers sets.

6. Continuous Air Monitors

A Continuous Air Monitor unit in installed in each Control Room HVAC equipment room. Each CAM unit consists of sampling head RE-1818A (Train A), RE-1818B (Train B), display/processing unit RIA-1818A (Train A) RIA-1818B (Train B) and air pump P-968A (Train A) P-968B (Train B). Each CAM unit is installed with stainless steel tubing sample lines to monitor each outside air intake downstream of the outside air isolation damper and alarm on airborne radioactivity.

9.8.2.3 Codes

1. The work, equipment and materials for the original Plant HVAC system design conform to the requirements and recommendations of the following codes and standards as applicable:

   a. The work and materials conform to the American Society of Heating, Refrigeration and Air Conditioning Engineers Guide (ASHRAE).

   b. The fans conform to the Air Moving and Conditioning Association, Inc, standards, definitions, terms and test codes for centrifugal, axial and propeller fans.

   c. The work, equipment and materials conform to the National Fire Protection Association Pamphlet 90A, "Air Conditioning, Warm Air Heating, Air Cooling Ventilating System."

2. The work, equipment and materials for the control room HVAC modifications made in 1983 conform to the requirements and recommendations of the following additional guides, codes and standards, as applicable:

   a. Ventilation ductwork conforms to applicable sections of the Sheet Metal and Air Conditioning Contractors National Association (SMACNA) manual.

   b. Refrigerant cooling coils conform to the standards of the Air Conditioning and Refrigeration Institute (ARI) and to requirements for Seismic Category I equipment.

   c. Applicable components and controls conform to the requirements of Underwriters Laboratories (UL), the National Electric Manufacturers Association (NEMA) and the Institute of Electrical and Electronics Engineers (IEEE) Standards 323, 344 and 383.

   d. Charcoal filter units and the associated ductwork, dampers and controls conform to the applicable sections of American National Standard Institute (ANSI) Standard 509-1980 and Standard 510-1980.

9.8.2.4 Operation

1. The individual systems are as shown on Figures 9-17 through 9-19.

2. The operation of the air handling units for the turbine building is as follows:

   a. Each unit has one steam coil downstream of a mixing box. The mixing box dampers and steam coil are controlled to provide a 60°F supply air temperature.

   b. If the fan motor is shut off, the fresh air inlet dampers will close.

   c. During normal operation, air is supplied to the auxiliary feed pump room by one of the turbine building air handling units and air is exhausted back to the main turbine building space via an exhaust duct located in the ceiling of the auxiliary feed pump room.

3. Steam-operated unit heaters are provided to heat the turbine building, containment building and other areas as needed. When heat is required, a thermostat starts the fan automatically.

4. Roof exhausters are provided for the turbine building, feedwater area, intake structure and boiler room. Thermostats set at preset temperatures individually start the roof exhausters so that all roof exhaust fans will be operating at a maximum temperature of 104°F.

5. Wall supply fans in the feedwater area and in the vicinity of the condensate pumps are started at preset temperatures by thermostats mounted in the area.

6. The supply unit mounted on the intake structure will operate continuously supplying a mixture of outside and return air. Wall supply fans are started at preset temperatures by thermostats mounted in the area.

7. The redundant supply units for the diesel generator room supply fresh air as the cooling load requirements demand. These fans are started automatically in sequence by thermostats.

8. Operation of the air supply units for the fuel handling area and the radwaste area is as follows:

   a. Each air supply unit is equipped with steam coils, a preheat coil and a reheat coil, and an air filter.

   b. The preheat coil is energized by an outside air thermostat at temperatures lower than 40 °F.

   c. The reheat coil is controlled by a leaving air thermostat set at 60°F.

   d. A thermostat senses the preheat coil leaving air temperature and closes an alarm circuit on low temperature to signal faulty coil performance. The alarm is located in the control room HVAC panel.

   e. If the fan motor is shut off, the fresh air inlet dampers close.

9. The operation of the auxiliary building office air-conditioning unit is as follows:

   a. Air is recirculated and mixed with fresh air to provide a mixed air temperature of 60°F.

   b. The steam coil and chilled water cooling coil in the air-conditioning unit are controlled by a thermostat in the supply fan discharge flow path. .

c. The supply airflows remain nearly constant but the fresh airflow varies depending upon the setting of the occupancy selector switch and the mixed air thermostat.

d. If the fan motor is shut off, the fresh air inlet dampers close.

10. The duct heaters for both the auxiliary building systems and the turbine building offices are controlled by room thermostats to obtain the desired room temperatures.

11. The access control duct cooling coil is controlled by the same thermostat that controls the duct heater.

12. The control room HVAC system operates in three different modes to meet the design bases discussed in Subsection 9.8.1. The three modes of operation (normal, emergency and purge) are described in Paragraphs a. through c. below. Tornado protection is described in Paragraph d. below:

a. Normal Mode

During normal mode operation, either Train A or Train B operates to supply air to the control room, Technical Support Center (TSC) and viewing gallery, and maintains a positive pressure with respect to the surroundings.

When Train A is in operation, Air Handling Unit Fan V-95 supplies conditioned air to the control room, TSC and viewing gallery. The room differential pressure controller modulates Damper D-2 to bring in a sufficient amount of outside air to maintain a positive control room pressure. Control room temperature is maintained by two 2-stage thermostats located in the control room, which control Condensing Unit VC-11 by unloading cylinders. Dampers D-1 and D-2 are interlocked with the air handling unit fan to open when the fan is running and to close when the fan is stopped. A humidistat controls the humidity to 40% relative humidity (design basis is 50% relative humidity). Humidifiers are interlocked with a high limit humidistat and with the fan.

The Train A damper positions during normal mode operation are as follows:

(1) Dampers D-5, D-6 and D-7 close.

(2) Dampers D-1, D-3, D-4 and D-20 open.

(3) Damper D-2 modulates.

The Train B damper positions during normal mode operation are as follows:

(1) Dampers D-12, D-13 and D-14 close.

(2) Dampers D-8, D-10, D-11 and D-21 open.

(3) Damper D-9 modulates.

Dampers D-1, D-2 and D-4 isolate Air Handling Unit V-95 of Train A from the outside and Train B when Train A is not in operation.

Dampers D-8, D-9 and D-11 isolate Air Handling Unit V-96 of Train B from the outside and Train A when Train B is not in operation.

When Train B operates, Air Handling Unit V-96 supplies conditioned air to the control room, TSC and viewing gallery. The room differential pressure controller modulates Damper D-9 to bring in a sufficient amount of outside air to maintain positive pressure in the control room. Control room temperature is maintained by two 2-stage thermostats located in the control room, which control Condensing Unit VC-10 by unloading cylinders. Dampers D-8 and D-9 are interlocked with the air handling unit fan to open when the fan is running and to close when the fan is stopped. A humidistat controls the humidity to 40% relative humidity (design basis is 50% relative humidity). Humidifiers are interlocked with a high limit humidistat and with the fan. A smoke detector, downstream of outside air dampers in the outside air duct is provided to detect the smoke.

b. Emergency Mode

The emergency mode of operation is actuated either by a containment high-radiation or a containment high-pressure signal (Section 7.3), or manually from the control room. During emergency mode operation, the air handling units and the charcoal filter units of both Train A and Train B operate. The refrigerant Condensing Units VC-10 and VC-11 shut down and are manually started by the operator. The control room operator has the option to turn off either Train A or Train B. During an emergency signal, operation of Purge Fan V-94 and Isolation Dampers D-15 and D-16 is blocked. The toilet exhaust fan in the viewing gallery is shut off, and Fan Isolation Dampers D-17 and D-18 close. A manual switch to override each outside air duct damper (D-7 and D-14) is provided to isolate the control room from outside air and to allow 100% air recirculation.

The Train A damper positions during emergency mode operation are as follows:

(1) Dampers D-1, D-2, D-17 and D-18 close.

(2) Dampers D-3, D-4, D-5, D-6 and D-7 are fully open.

(3) Damper D-20 modulates.

Air Handling Unit V-95 takes return air from the control room, TSC and viewing gallery, conditions it, and returns it to maintain the desired temperature in the control room. Filter VF-26A (which has a capacity of 3,200 ft^3/min) admits 1,000 ft^3/min outside air through the emergency outside air intake duct and admits 2,200 ft^3/min from the control room return air duct. The 3,200 ft^3/min filtered air is then supplied to the control room via Air Handling Unit V-95. Positive pressure of .125" of water or greater is maintained in the control room by adding the 1,000 ft^3/min outside air to the system. A constant flow rate through the filter unit is maintained by modulating Damper D-20 to compensate for filter loading.

Train B emergency operation is similar to Train A. The Train B damper positions during emergency mode operation are as follows:

(1) Dampers D-8, D-9, D-17 and D-18 close.

(2) Dampers D-10, D-11, D-12, D-13 and D-14 are fully open.

(3) Damper D-21 modulates.

Air Handling Unit V-96 takes return air from the control room, TSC and viewing gallery, conditions it, and returns it to maintain the design temperature in the control room.

Filter VF-26B operation is similar to V-26A operation.
Filter VF-26B recirculates 3,200 ft^3/min and pressurizes the control room by bringing in 1,000 ft^3/min outside air.

c. Purge Mode

Smoke can be purged from the control room by Fan V-94. This fan is manually started by the operator, when required. When the purge fan is started (with V-95 running), Dampers D-15 and D-16 open; Return Damper D-3 closes; and Dampers D-1 and D-2 open fully to bring in 9,060 ft^3/min outside air and prevent recirculation. When the purge fan is started (with V-96 running), Dampers D-15 and D-16 open; Damper D-10 closes; and Dampers D-8 and D-9 open. Purge Fan V-94 exhausts 7,800 ft^3/min to the atmosphere, 160 ft^3/min is exhausted by the toilet exhaust fan and 1,100 ft^3/min exfiltrates.

When the purge fan runs in conjunction with Train A, the damper positions are as follows:

(1) Damper D-3 closes.

(2) Dampers D-1, D-2, D-15 and D-16 open.

When the purge fan runs in conjunction with Train B, the damper positions are as follows:

(1) Damper D-10 closes.

(2) Dampers D-8, D-9, D-15 and D-16 open.

d. Tornado Protection

Tornado dampers are provided in all the outside air intakes, the purge exhaust and the toilet exhaust ducts. During tornado depressurization, the tornado dampers close to isolate the HVAC system from the outside.

e. Control Room/TSC Envelope

Four vestibules are used to provide egress and ingress to the control room/TSC during post-accident operations. These vestibules are adjacent to Doors 108, 115, 175 and 52. Their function is to prevent air in-leakage.

f. Penetrations to the Control Room Envelope

Uncontrolled open penetrations to the control room/TSC envelope degrades the maintenance of positive air pressure. Therefore, administrative controls are used for maintenance activities requiring an open penetration. These controls assure prompt and secure closure of openings in the event of an emergency.

The Thermal Margin Monitor (TMM) was originally qualified to 131°F. However, the location of the TMM in the panel is such that cooling is required. Analysis shows that, with forced air cooling, 131°F is reached by the TMM when the control room ambient temperature is 106°F. Because the TMM portion of the RPS is no longer capable of operating at the nominal control room design temperature of 120°F, a control room administrative limit of 90°F was imposed.

Other portions of the Reactor Protective System located in the control room were designed to operate up to 135°F and 90% relative humidity. Individual components and modules of the RPS have been factory tested at design temperature and humidity conditions. With the exception of the TMM, the RPS cabinet (including all portions of the system located in the control room) has been tested for operation as a system at temperatures in excess of 135°F.

Other electronic equipment used in plant safety-related components can operate at 120°F continuously and at 140°F intermittently as proven by experience.

Cooling of safety-related equipment and controls located in rooms other than the main control room is maintained by systems designed with similar component redundancy as the control room air-conditioning system.

13. The engineered safeguards equipment room coolers are started by a signal from wall-mounted thermostats and provide cooling for the protection of the engineered safeguards equipment. Service water flow is continually maintained through the cooling coils by maintaining CV-0878 and CV-0825 electrically locked open. The valves are operated by cylinder lock hand switches preventing inadvertent closure.
Each room has one cooler with two fans, one powered from Class 1E MCC-1 and the other powered from Class 1E MCC-2. Emergency power for each cooler is supplied upon loss of offsite power from diverse sources. MCC-1 and MCC-2, which provide diverse power sources for the fans in each room, are automatically loaded on EDG 1-1 and EDG 1-2 respectively. Equipment located in these rooms are the HPSI pumps, containment spray pumps, LPSI pumps, shutdown cooling heat exchangers, high-pressure control air equipment, and related piping, valves and controls.

As a result of Consumers Power Company's evaluation of IE Bulletin 80-06, "Engineered Safety System (ESF) Reset Controls," circuitry modifications were made to ESF Room Cooler Valves SV-0825 and SV-0878 such that these valves do not close upon an ESF reset signal. In addition, to preclude inadvertent closure of the service water valves supplying cooling to the ESF room coolers, the hand switch controllers (HS-0825A and HS-0878A) for these valves were changed from hand switches without locks to hand switches with cylinder lock operators.

14. Two iodine removal filters were installed for preoutage containment atmosphere iodine removal. The units are freestanding and do not use inlet or outlet ductwork. The units are manually operated when required.

15. The containment purge and vent system is operated only during cold shutdown conditions. The operation of the system is as follows:

    a. One main exhaust fan (V-6A or V-6B) must be running.

    b. Purge isolation valves (CV-1805 and CV-1806, and/or CV-1807 and CV-1808) are opened.

    c. Air room supply isolation valves (CV-1813 and CV-1814) are opened and air room purge supply fan (V-46) is started. During cold weather heat is supplied by Heating Coil VHX-48.

    d. The purge system is stopped by reversing the above procedure.

    Occurrence of a containment high-pressure or high-radiation signal during this purge cycle automatically closes the isolation valves.

16. The radwaste area exhaust system operates as follows:

    a. Normally both fans, each rated at 50% of the normal flow, operate continuously. Dampers in the fan discharge modulate to maintain a uniform static pressure in the filter intake plenum.

    b. The filter intake pressure is the static pressure of a balanced airflow from all areas with access openings closed or in the normal condition. The ductwork is sized to permit airflows from the cells through access ports sufficient to permit entrainment velocities. Thus, if an access port or hatch cover is open, the air velocity through the opening is over 100 feet per minute and the fan discharge dampers will open to maintain the set static pressure at the filter intake plenum.

c. Hoods have high-efficiency particulate filters as an integral part of the hood, and booster fans are provided to offset the pressure drop through the filter.

d. In the event of an exhaust fan failure, the supply fan may be shut down and the negative pressure of the radwaste area will be maintained by the remaining exhaust fan.

e. In the event of failure of the radwaste area supply fan, one of the exhaust fans is automatically shut down but the pressure control apparatus will limit the amount of the negative pressure developed by the lack of supply air and prevent excessive pressure differentials.

f. In the event of a spillage of radioactive material in the radwaste area, the radiation monitor at the filter plenum senses the activity and stops the supply fan, closes the radwaste area supply Damper PO-1809, and stops the selected exhaust fan; however, a low flow alarm will override the high radiation signal and keep the standby exhaust fan running. The duct to access control remains open and is isolated from the radwaste area by Damper PO-1809.

g. In the event of significant airborne contamination in the engineered safeguards rooms, the supply and exhaust dampers of those rooms are closed on a signal from the individual non 1-E radiation monitor for each exhaust duct.

17. The fuel handling area exhaust system operates as follows:

a. During normal operation, one or both of the exhaust fans run, as required, and draw air through a prefilter and a high-efficiency filter.

b. During refueling operations, the exhaust air may be diverted through a prefilter, HEPA filter and a charcoal adsorber bed. This filter train is parallel to the normal high-efficiency filters and is isolated from it by the positioning of an inlet damper.

c. In the event of a fuel handling/cask (heavy load) drop accident in the spent fuel pool, the exhaust airflow is reduced to one-half by tripping the supply fan and closing the inlet damper and tripping one of the 50% capacity exhaust fans. The exhaust air flows through the high-efficiency particulate filter and a charcoal filter. See Sections 14.11 and 14.19 for specific assumptions of the Fuel Handling Building HVAC used in the safety analysis.

18. The operation of the auxiliary building addition fuel handling supply and radwaste supply is as follows:

   a. The supply unit for each area is equipped with a preheat coil, a reheat coil and an automatic filter.

   b. The preheat coil is controlled by a thermostat in the fresh air intake set at 35°F. The reheat coil is controlled by a leaving air thermostat to maintain a discharge temperature of 60°F.

   c. Another thermostat is provided in the leaving air stream which is set at 45°F and alarms in the control room when this temperature is reached to indicate faulty coil performance.

   d. If the fan motor is shut off, the fresh air inlet dampers close.

   e. The supply fans will trip on a high-radiation signal from radiation monitors located in the corresponding exhaust system ducts.

19. The operation of the auxiliary building addition fuel handling area exhaust and radwaste exhaust systems is as follows:

   a. The exhaust systems each consist of a filter package which contains a bank of roughing filters and a bank of HEPA filters. Air is drawn through the filter plenum by two exhaust fans.

   b. Normally both fans, each rated at 50% of the normal flow, operate continuously. Dampers in the fan discharges modulate to maintain a uniform static pressure in the filter intake plenum.

   c. In the event of an exhaust fan failure, the supply fan may be shut down and the negative pressure of the area served by the particular system will be maintained by the remaining exhaust fan.

   d. In the event of failure of a supply fan, one of the exhaust fans will shut down but the pressure controller in the filter intake plenum will limit the amount of negative pressure developed by the lack of supply air and prevent excessive pressure differentials.

   e. In the event of release of radioactive material in the area served by the system, the radiation monitor at the filter plenum senses the activity and trips the supply fan which in turn trips one of the exhaust fans. However, a low flow condition will override the high-radiation signal and keep the standby exhaust fan running.

20. The penetration and fan rooms' heating and ventilating system has been installed as part of the high-energy line work heating and ventilation. This system provides cooling air to the feedwater pipe penetration room and fan room. This system is not considered essential because the essential equipment located in this area is qualified to survive a main steam line break within this area. The system operates as follows:

a. The supply system consists of a supply fan, an air filter and an outside air damper.

b. The exhaust system consists of a prefilter, a high-efficiency filter and an exhaust fan.

c. The supply and exhaust systems run concurrently and are controlled by a thermostat located in the exhaust duct. The supply and exhaust fans are started when the exhaust air temperature is 90°F and stop when the exhaust air temperature is 70°F.

d. A differential pressure controller which measures differential pressure across the filters and filter inlet damper, modulates the filter inlet damper to maintain a preset negative pressure across the filters and dampers.

21. The electrical equipment, switchgear, cable spreading and battery rooms' HVAC system was modified in the 1983 outage to include the new electrical equipment room added as part of the control room modification work. This system formerly served the viewing gallery. This system operates as described below and serves the following areas:

Electrical Equipment Room
Cable Spreading Room
Bus 1D Switchgear Room
Bus 1C Switchgear Room
Battery Room

The cable spreading room, switchgear rooms and battery rooms are considered essential because they house the reactor protection and control system, the instrumentation for shutdown and cooldown, the emergency power (ac and dc) and control power for safe shutdown systems all of which are considered important to safety.

The ventilation system that services these areas is composed of V-33 and V-43 with supplemental ventilation supplied by V-47, none of which are safety grade. Supply Fan V-33 provides 18,500 scfm of air to the areas identified. Makeup air to V-33 is a blend of outside air and recirculated air from V-43. This blend is controlled by a mixed air temperature controller. When outside air temperature increases, the amount of recirculation is decreased, and the amount of makeup increases up to the full 18,500 scfm.

The duct branch that services the new electrical equipment room is equipped with a chilled water cooling coil to provide adequate cooling for the room. This cooling coil is controlled by a thermostat located in the electrical equipment room.

Separate from this two-fan ventilation system is a 30,000 scfm exhaust fan that takes suction on the cable spreading, Bus 1D switchgear and Bus 1C switchgear rooms only. When air temperature in the upper region of the rooms increases above 100°F, Temperature Switches 1824, 1825 and 1826 will initiate a control room annunciator none of which are safety grade. During normal plant conditions, the operator may manually start the supplemental Exhaust Fan V-47. Normally, the temperature will drop below the 100°F set point within 10 minutes and the operator will stop V-47. When the control room ventilation system is in its emergency mode, V-47 is not operated because its high suction capacity inhibits the ventilation system from maintaining its required positive pressure.

If the high-temperature alarm does not clear, other corrective measures available to the operator would be: check fan and damper operation, ensure heating steam controller and cooling controllers are functioning, ensure filter media is clear, block open doors, place fire protection smoke blowers in rooms as temporary air movers.

22. The containment building air coolers operate as follows under normal conditions:

The service water supply line for each safety-related cooler has an air-operated stop valve which is electrically locked open. The return line for each safety-related cooler has an 8" air-operated discharge valve which is usually (in cold weather) held closed and a 4" temperature control valve in a bypass line around the closed discharge valve. The non-safety related cooler (VHX-4) has air-operated valve in its service water supply and return lines that are normally open. The VHX-4 return line valve can be closed during cold weather to reduce the cooling occurring in containment. The service water supply and discharge valves for all the coolers go to their safety position upon loss of control power or instrument air. The 4" temperature control valves were modified by SC-93-054 to eliminate the automatic temperature control function. These valves may be manually operated locally using the provided air regulator but are typically kept full open (with the exception of VHX-4's TCV, whose air supply has been isolated per FC-713, failing it closed). Because the temperature control function was eliminated, the valve equipment ID's are now CV rather than TCV. The supply and 8" discharge valves may be manually operated from the main control room and the engineered safeguards local panel. The bypass valves can only be closed from the control room by isolating instrument air to containment.

Air is drawn through the containment air coolers by two matched vaneaxial fans with direct connected motors. One fan motor is rated for normal conditions and the second is rated for post-DBA conditions. During normal operation the total airflow through each cooler is 60,000 CFM. The fan motors, rated for the post-DBA condition are fed from the emergency power buses. All fans may be manually started or stopped from the main control room or at the individual breakers. For description of the operation under accident conditions, see Section 6.3.

23. The CRDM cooling system consists of two fans which draw in building ambient air and discharge it into a shroud around the CRDMs. This system was modified in 1977 to limit inflow of hot air into the service structure and, as a result, lower the temperature in the service structure.

24. The containment post-accident filter system is described in Section 6.5.

25. The main steam line and feedwater line containment penetration cooling system consists of two fans which draw air from the 590 ft elevation CCW room and discharges it through passages in the annulus between the pipe and concrete.

26. Operation of the condensate and makeup demineralizer building HVAC system is as follows:

    a. The heating and ventilation air handling units perform the ventilation and heating function for the condensate and makeup demineralizer building areas. The systems are designed to take outside air, mix it with return air as applicable, filter it at the air handling unit and distribute it to the building areas. Areas being served by the heating and ventilating air handling units are provided with thermostats for control of winter space temperatures. Exhaust airflows are from areas of low potential airborne radioactivity to areas of higher potential airborne radioactivity.

    b. The instrument room air-conditioning unit provides cooling and heating for the instrument room. The system is designed to take outside air, mix it with return air, filter it at the air-conditioning unit and deliver it to the instrument room. A space thermostat is provided for year-around temperature control. The air-conditioning unit and the return/exhaust fan are equipped for economizer control in the event ambient air conditions can satisfy interior air-conditioning heat loads.

    c. The boiler room supply fan and roof exhauster perform the ventilating function for the boiler room. The equipment supplies outside air and exhausts hotter room air. The fans are started and stopped by room thermostats or may be operated manually by a control switch.

d. The air compressor and switchgear room and the pipe gallery use wall louvers and roof ventilators for ventilation. The systems are started and stopped by room thermostats or may be operated manually with control switches.

e. Unit heaters are controlled individually by room thermostats.

27. Operation of the Volume Reduction and Solidification (VRS) system HVAC systems are as follows:

a. The VRS system control room air-conditioning unit provides cooling for the VRS system control room. The system is designed to take room return air, mix it with minimum outside air, filter and cool the air as necessary and deliver it to the VRS system control room. The air-conditioning unit is controlled from a room thermostat. Redundant condensers are provided for the air-conditioning unit for reliability.

b. The VRS system area supply air system consists of an air-conditioning unit with an air-cooled condenser. Outside air is drawn through the unit, filtered and cooled as necessary and delivered to the space. A space thermostat controls operation of the unit.

c. The VRS system area exhaust system consists of a duct system which is tied into the existing radwaste area exhaust system. The air supplied to VRS area is drawn through the space and exhausted to the outside through the radwaste area exhaust filter (VF- 73). Effluents from the drum filling station vent hood are exhausted to this system after passing through a prefilter , a HEPA filter, charcoal filter and another HEPA filter.

d. The VRS system area electrical equipment room uses roof ventilation for ventilation. The roof ventilators are started and stopped by a thermostat located in the space.

9.8.3 TESTS AND INSPECTIONS

Provisions for testing equipment performance are built into the critical apparatus such as exhaust systems, the engineered safeguards room coolers and the control room air-conditioning unit and refrigerant condensers. After the equipment is installed and operating, periodic tests may be performed to assure that filters and coils are not dirty or plugged and the unit is still performing as required.

The charcoal and high-efficiency filters for the control room and the fuel handling area are tested per the requirements of the Technical Specifications.

9.8.4 LOSS OF INSTRUMENT AIR TO VENTILATION DAMPERS

Table 9-17 lists the ventilation dampers, their function, and positions during operation of the Plant under normal, shutdown, abnormal conditions and loss of instrument air. Particular attention has been given to the failed position of dampers to ensure maximum safety of Plant personnel and minimum emission of possible contaminants to the environment.

The control room HVAC system damper positions for the various modes of operation are discussed in Subsection 9.8.2.4, Item 12.

The normal radwaste area and engineered safeguards rooms ventilation mode is with all dampers open, Supply Fan V-10 running, one or both exhaust fans (V-14A and/or V-14B) running, and the exhaust dampers (PO-1839 and PO-1840) controlled by filter intake pressure to maintain balanced airflow from all areas. A high-activity level at the filter intake plenum will actuate the radiation monitor (RE-1809) which will close the radwaste area supply damper (PO-1809), trip one exhaust fan (V-14A or B) if both are running, close the respective exhaust damper, and trip the supply fan (V-10) which will in turn close the supply damper. The remaining exhaust fan will maintain a slight negative pressure on the radwaste area to prevent leakage out of the building. The tripped exhaust fan will restart if 2.5 inches of water vacuum is not maintained in the exhaust plenum. A high-radiation level in either of the exhaust ducts from the safeguards rooms will automatically close the supply and exhaust dampers for the affected room. Continued cooling of air within the safeguards rooms is provided by the local cooling units.

The normal ventilation mode in the fuel handling area during reactor operation or reactor shutdown is Supply Damper PO-3007 open, Supply Fan V-7 operating, one or both exhaust fans (V-8A or V-8B) operating, and one or both gravity exhaust dampers open. No change in the normal ventilation mode occurs in the unlikely event of a DBA unless the DBA is accompanied by a loss of offsite power at which time the ventilation fans will be shed from their respective bus and the dampers will close. Upon a fuel building high-radiation area alarm, Fan V-7 is manually tripped which closes Damper PO-3007 and one exhaust fan is manually tripped closing its gravity damper. The remaining running fan continues to run maintaining a slight negative pressure on the fuel building to prevent leakage from the building. Upon loss of instrument air, Supply Damper PO-3007 will shut. The supply fan and one exhaust fan will be manually tripped to ensure no building leakage in the unlikely event a simultaneous release of activity occurs within the fuel building.

9.8.5 SAFETY EVALUATION

9.8.5.1 Introduction

The Heating, Ventilation and Air Conditioning Systems covered in this section are those which were evaluated by the NRC in SEP Topic IX-5 dated February 11, 1982.

In determining which systems to evaluate under this topic, the NRC used the definition of "systems important to safety" provided in Regulatory Guide 1.105. The definition states systems important to safety are those necessary to ensure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and maintain it in a safe condition, or (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the guidelines of 10 CFR Part 100, "Reactor Site Criteria." This definition was used to determine which systems or portions of systems were "essential." Systems or portions of systems which perform functions important to safety were considered to be essential.

9.8.5.2 Evaluation

1. Control Room Heating, Ventilation and Air Conditioning (CRHVAC) System

   The function of the CRHVAC system is to provide a controlled environment for the comfort and safety of control room personnel and to assure the operability of control room components during normal operating, anticipated operational transient and design basis accident conditions.

   This system was modified during the 1983 outage in response to NUREG-0737, Item III.D.3.4 concerns. The system is described in Subsection 9.8.2.

   A safety evaluation for this system is presented in Subsection 6.10.3, Control Room Habitability.

   Safety Analysis for dose consequences are presented in Chapter 14 for various accident conditions.

2. Spent Fuel Pool Area Ventilation System

   The function of the spent fuel pool area ventilation system is to maintain ventilation in the spent fuel pool equipment areas, to permit personnel access, and to control airborne radioactivity in the area during normal operation, anticipated operational transients and following postulated fuel handling accidents.

   Based on the fuel handling accident analysis in Section 14.19, it was determined that the system is nonessential.

3. Radwaste Area Ventilation System

The radwaste area ventilation system services most areas within the auxiliary building during normal operation, including the engineered safeguard equipment rooms (east and west), the charging pump room, the primary drain tank pump room and the boric acid control area. These areas house equipment (Emergency Safeguards systems and the Chemical and Volume Control System) which operates either post-accident or for the safe shutdown of the Plant. Therefore, the service conditions within these areas and the equipment that maintains those conditions are considered essential.

If loss of instrument air were to occur, the Supply Damper PO-3010 A, B, C and D, radwaste area Supply Damper PO-1809, the two Exhaust Dampers PO-1839 and PO-1840 and safeguard rooms supply and exhaust dampers would fail closed. This would result in a loss of ventilation for the Chemical and Volume Control System. In addition, if offsite power is lost, the radwaste area ventilation system would fail.

This event actually occurred during a Plant trip which occurred on September 24, 1977 when the switchyard "R" bus was automatically de-energized during an electrical storm. The CVCS was required to operate 4 hours and 34 minutes on diesel power with the radwaste area ventilation system de-energized. No temperature-related equipment failures occurred in the auxiliary and radwaste areas. This demonstrates that ample time exists for operator action. The probability of a sustained loss of instrument air is not credible since station demand is less than 250 ft^3/min and the Plant is equipped with two 200 ft^3/min compressors and one 320 ft^3/min compressor , two of which can be powered from Emergency Diesel 1-1, and the other from 1-2. In addition, the feedwater purity building houses two 876 ft^3/min compressors, which can be paralleled into the original Plant instrument air system by opening a single control room activated control valve. These two compressors do require offsite electrical power.

The loss of offsite power does have a higher probability of occurrence than loss of instrument air, but loss of offsite power also trips the reactor and only activates the engineered safeguard equipment. Thus, there is a minimum of operating equipment to add heat. A simplified analysis was performed on the charging pump room assuming the three pump motors provide the heat input. The only other heat input would be that radiated from the insulated process lines carrying 120°F water. Since the normal design ambient for the pump and motor is also 120°F, the terminal temperature from this source is equal to the design temperature. This demonstrates less than an 11°F rise in 6 hours and a temperature rise of only 0.4°F during the fifth hour. Thus, assuming an initial temperature of 80°F, it would take on the order of 83 hours to reach the design temperature of 120°F. This would allow ample time to restore offsite power or to install temporary air movers.

Although the Chemical and Volume Control System, a system required for Plant safe shutdown, relies on the radwaste area ventilation system for maintaining its operational service conditions, it was demonstrated that a short loss of the radwaste area ventilation system has no adverse effect on the Chemical and Volume Control System and that adequate time exists for corrective action. Although the present radwaste area ventilation system is susceptible to single mode failure, it was found that a safety grade ventilation system is not required for this area based on the long time available for operator corrective action.

4. Turbine Building Ventilation System

The only area in the turbine building which is considered to be essential to safety is the auxiliary feed pump room. During normal operation, air is supplied to the auxiliary feed pump room by one of the turbine building air handling units and air is exhausted back to the main turbine space via an exhaust duct located in the ceiling of the auxiliary feed pump room.

A single duct failure or loss of offsite power would result in the loss of ventilation of this room which could potentially cause the failure of both auxiliary feed pumps to perform when required.

However, tests were reported to the NRC in a November 1, 1982 letter which simulated loss of ventilation in the auxiliary feedwater pump room. These were conducted for both motor-driven auxiliary feedwater pump operation and steam-driven auxiliary feedwater pump operation. The tests verified that the room can safely withstand a loss of power to the ventilation units for a long period of time, at least 24 hours. In addition, a 1996 modification to the steam supply piping for P-8B provided a slight reduction in heat input to the room. Also, the simple measure of opening a door provides adequate ventilation for an indefinite period.

5. Engineered Safeguard Equipment Rooms (East and West)

The engineered safeguard equipment rooms are located in the auxiliary building. Equipment located in these rooms is:

HPSI Pumps

Containment Spray Pumps

LPSI Pumps

Shutdown Cooling Heat Exchangers

High-Pressure Control Air Equipment

Related Piping, Valves, Controls, etc

Each room has one cooler with two fans, powered from diverse sources.

The water source for the coolers is the Service Water System. Emergency power is supplied upon loss of offsite power. The two fans V-27C and V-27D, which service the west room, receive power from Class 1E MCC 1 (B01) and MCC 2 (B02), respectively. Fans V-27A and V-27B, which service the east room, are similarly powered from MCC-1 and MCC-2, respectively. Buses MCC-1 and MCC-2 are automatically loaded onto EDG 1-1 and EDG 1-2, respectively.

6. Electrical Equipment, Switchgear and Cable Spreading Rooms Ventilation System

The areas served and operation of this system are discussed in Subsection 9.8.2.

The cable spreading room, switchgear room and battery room are considered essential because they house the reactor protection and control system, the instrumentation for shutdown and cooldown, the emergency power (ac and dc) and control power for safe shutdown systems all of which are considered important to safety.

Ventilation tests were conducted in July and August of 1982 to investigate loss of offsite power to the ventilation fans. These were reported in a November 1, 1982 letter to the NRC.

The ventilation tests showed that certain equipment in the cable spreading room cannot withstand a loss of normal ventilation for an indefinite period of time. Upon loss of normal ventilation, the operator has sufficient time, however (up to six hours), to take action to ensure that the room's design temperature of 104°F is not exceeded. Such action consists of restarting normal cable spreading room ventilation fans V-33 and V-43, or restarting supplemental ventilation fan V-47 if the emergency mode of the control room ventilation system is not required. The other corrective measures described in Subsection 9.8.2 would also be utilized if necessary. Initiation of such action(s) would be the result of a control room annunciation when the cable spreading room temperature reaches 100°F. The cable spreading room fans V-33, V-43, and V-47 are all capable of being powered from an emergency diesel generator.

The 1C and 1D switchgear rooms are not affected by a loss of ventilation since no appreciable heat sources are contained in these rooms. The battery rooms, which are contained within the larger cable spreading room, were not tested since the battery room fans are powered from a safety-related source and, therefore, would not be vulnerable to a loss of offsite power.

During the conduct of the cable spreading room tests, it was observed that the inverter cabinets, the charger cabinets and auxiliary feedwater junction boxes (J-569 and J-570) require forced-air cooling. Therefore, V-928 was installed to cool the inverter and charger cabinets (auxiliary feedwater junction boxes no longer contain equipment requiring cooling)

7. Emergency Diesel Generator Room

There are two emergency diesel generator rooms. Ventilation to maintain suitable operating temperatures for the diesel and its associated electric control equipment within each room is provided by two separate ventilation systems and are supplied from safety-related power sources. The reliable operation of these ventilation systems is considered essential to Plant safety.

The major components of these systems are the two diesel generators (1-1 and 1-2), and two cooling fans for each generator room, V-24A, V-24B, V-24C and V-24D, respectively. An intake plenum is installed between the two fans in each room.

On the basis of extensive experience with this system involving normal periods of diesel operation for test purposes, this ventilation system has been demonstrated to be of adequate design.

8. Intake Structure Ventilation System

The intake structure ventilation system is addressed in this evaluation because it services the area where the three service water pumps are located. These pumps are considered important to safety. The system consists of seven supply fans, five wall-mounted units (V-21D-H) and two roof units (V-32A and B). These supply fans draw atmospheric air into the building. The air is then exhausted back outside through five roof-mounted exhaust fans (V-30A-E).

The intake structure ventilation system was originally sized to cool circulating water pumps in addition to the service water pumps. The existing ventilation system is now oversized for normal operation since the circulating water pumps have been removed.

In the event of a power failure, several mechanisms act to prevent any rapid heat buildup. All other heat loads within the structure are secured as the Plant is shut down, service water pipes containing cool lake water act as heat sinks, and the room is not airtight allowing some limited convective cooling to take place. If a system failure should occur, doors opening to the outside are available and should provide sufficient airflow even with multiple fan failures. Inspection of the intake structure at eight-hour intervals provides additional protection against excessive heat buildup.

9. <u>Penetration and Fan Room Ventilation System</u>

The penetration and fan room ventilation system provides cooling air to the feedwater pipe penetration room, the main steam pipe penetration room and fan room. Although this ventilation system services the main steam system and component cooling water system, both considered essential for a shutdown of the Plant, it is not considered essential as defined in Subsection 9.8.5.1, because the essential equipment located in this area is being qualified to survive a main steam line break within this area.

9.9 <u>SAMPLING SYSTEM</u>

9.9.1 DESIGN BASIS

The sampling systems are designed to permit liquid and gaseous sampling for analysis and chemistry control of the Plant primary and secondary fluids. Samples are used to determine if chemical and radiochemical concentrations are within the prescribed operating limits.

9.9.2 SYSTEM DESCRIPTION AND OPERATION

The sampling system is a collection of smaller subsystems which are designed to sample various Plant fluids. These subsystems are designated by the Plant systems or fluid sampled. Table 9-18 lists each subsystem.

The NSSS Sampling Station is located in the auxiliary building sample room.

High-temperature, high-pressure fluid samples taken from the Primary Coolant System are first passed through a delay coil to permit decay of short-lived radioactivity and then through a cooler, pressure reducing coil, flow controller and finally an analyzer or grab sample valve. All grab samples and bomb samples are taken to the chemistry lab for analysis.

Block and bleed valves, located on the reactor coolant and LPSI pump suction sample lines, provide the opportunity to backflush these lines through the sample coolers to reduce the dose rate and potential equipment contamination. The block valve is also controlled to shut on high temperature at the sample cooler outlet.

A parallel system, the post-accident sample & monitoring system (PASM) (Figure 9-21), is designed to sample primary coolant, low pressure injection pump discharge and containment atmosphere following an accident involving fuel damage. This system can continuously sample pH, conductivity and dissolved O_2 and can be monitored from outside the PASM room. The sample panel is faced with 7 inches of lead shot sandwiched between 1/2-inch steel plates to reduce the potentially high dose rates to tolerable levels. The PASM inline instrumentation is also used to monitor the PCS during routine plant operations.

In addition, liquid samples may be taken directly or diluted with demineralized water to reduce the activity levels. The undiluted sample may be injected into a shielded transportable sample flask for transport to the lab for analysis.

Dissolved gases are vacuum stripped from a pressurized primary coolant sample. A gas chromatograph can remotely sample for PCS dissolved H_2 . The gases can also be extracted via a syringe for transport to the lab and subsequent analysis. The waste gas decay tanks and volume control tank can also be sampled with the gas chromatograph.

The containment hydrogen monitoring system (Figure 9-22) consists of redundant, Class 1 monitors designed to continuously monitor the containment hydrogen concentration during post-accident conditions. Each monitor contains a sample pump, temperature, pressure and flow controllers, and a thermal conductivity cell. Piping from the containment to the H_2 analyzer panels are heat-traced and maintained at approximately 285°F to prevent condensation in the sample stream.

During normal Plant operation, the system is maintained at standby conditions permitting rapid start-up.

Following initial start-up and calibration, system operation may be initiated locally at the panel or remotely from the control room. Once initiated, operation is automatic.

The Turbine Plant Analyzer Station is located in the turbine building. This station contains sample pressure reducing and cooling equipment including valves, pressure regulators, pressure indicators, flow regulators, piping grab sample sinks and continuous analyzers for various parameters such as conductivity, dissolved oxygen, sodium, hydrazine, and pH. A data acquisition system, indicators and an annunciator, to alarm abnormal conditions, are located at the Turbine Plant Analyzer Station.

At the Turbine Plant Analyzer Station, sample streams are sent through continuous analyzers. These analyzers transmit their signals to indicators for continuous display on the local analyzer panel. A data acquisition system also receives the signals from the analyzers.

The Radwaste Sample Station (Figure 9-23), located in the auxiliary building sample room, provides sample streams for grab sampling or collection in sample bombs. The sample streams are radioactive or potentially radioactive fluids.

The Radwaste Addition Sample Station, located in the new radwaste building sample room, provides sample streams for grab sampling or collection in sample bombs. The sample streams are radioactive or potentially radioactive fluids.

Table 9-19 is a summary of sample points.

9.9.3 SYSTEM EVALUATION

The sampling system obtains a maximum of information from a number of separately located sample points and stations. All of the continuous sample analysis equipment is located near its sample conditioning equipment which permits rapid detection of deteriorating conditions of either the samples or the sampling equipment.

9.10 CHEMICAL AND VOLUME CONTROL SYSTEM

9.10.1 DESIGN BASIS

The Chemical and Volume Control System (CVC), a CP Co Design Class 1 system, is designed to:

1. Maintain the required volume of water in the Primary Coolant System over the range of full to zero reactor power without requiring makeup

2. Maintain the chemistry and purity of the primary coolant

3. Maintain the desired boric acid concentration in the Primary Coolant System

4. Pressure test the Primary Coolant System

The design parameters for the Chemical and Volume Control System and components are listed in Table 9-20.

9.10.2 SYSTEM DESCRIPTION AND OPERATION

9.10.2.1 General

The Chemical and Volume Control System is shown in Figure 9-24 and Figure 9-25. Primary coolant normally flows through the Chemical and Volume Control System as shown by the heavy lines in Figures 9-24 and 9-25. The letdown coolant from the cold leg of the Primary Coolant System passes through the tube side of the regenerative heat exchanger and is partially cooled. The cooled fluid is then partially depressurized as it passes through the letdown stop valves and orifices. The temperature and pressure of the letdown coolant are finally reduced to the operating requirements of the purification system by the letdown heat exchanger and back pressure valve, respectively. The coolant then passes through an ion exchanger and a filter and is sprayed into the volume control tank. The charging pumps remove the coolant from the volume control tank and return it to the Primary Coolant System by way of the shell side of the regenerative heat exchanger. The heat exchanger transfers heat from the letdown coolant to the charging coolant before the charging coolant is returned to the Primary Coolant System.

When the level in the volume control tank reaches the high level set point, the letdown flow is automatically diverted to the liquid radwaste system. When the level in the volume control tank reaches the low-level set point, makeup water, borated to the existing concentration of the Primary Coolant System, may be manually supplied to the suction of the charging pumps.

The volume control tank is designed and sized with a large enough capacity that with the level in the normal control band, the tank can accommodate a zero to full power increase or a full to zero power decrease.

The boric acid concentration and chemistry of the primary coolant are maintained by the Chemical and Volume Control System. Concentrated boric acid solution is prepared in a batching tank and is stored in two concentrated boric acid storage tanks. Two pumps are provided to transfer concentrated boric acid to a blender where the boric acid is mixed with primary makeup water in a predetermined ratio. The solution is introduced to the Primary Coolant System by the charging pumps.

Chemicals are introduced to the Primary Coolant System by means of a metering pump which pumps the chemical solution from a chemical addition tank and introduces it to the charging pump suction header.

The Primary Coolant System may be pressure tested for leaks by means of the variable speed charging pump. The system is also provided with connections for installing a hydrostatic test pump.

9.10.2.2 Volume Control

The CVC automatically adjusts the volume of water in the Primary Coolant System using a signal from the level instrumentation located on the pressurizer. The system reduces the amount of fluid that must be transferred between the Primary Coolant System and the CVC during power changes by employing a programmed pressurizer level set point which varies with reactor power level. The set point varies linearly with reactor power, defined for this purpose as the average primary coolant temperature measured across a steam generator. This linear relationship is shown in Figure 4-9. The control system compares the programmed level set point with the measured pressurizer water level. The resulting error signal is used to control the operation of the charging pumps and the letdown valves as described below. The pressurizer level control program is shown in Table 4-9.

The pressurizer level control program adjusts the charging rate of the variable capacity charging pump, normally in operation, to obtain a flow equal to the letdown flow through one letdown stop valve and orifice plus the total primary coolant pump seal bleedoff flow. If power changes or abnormal operations cause a large drop in the pressurizer level, one or both of the constant capacity charging pumps start to return the level to the normal control band. If conditions cause a large rise in the pressurizer level, additional letdown stop valves open to lower pressurizer level.

Since the normal letdown flow plus the primary coolant pump controlled bleedoff flow slightly exceeds the capacity of one constant capacity charging pump, one of two method of maintaining pressurizer level is used when the variable capacity charging pump is removed from service.

One method places one constant capacity charging pump in manual and allows the pressurizer level control program to cycle the second constant capacity charging pump on and off automatically to maintain level. One of the letdown orifice stop valves may be closed to reduce the cycling of the letdown orifice stop valves during this method. The second method places both constant capacity charging pumps in manual and allows the pressurizer level control program to maintain level by cycling the letdown stop valves.

The volume control tank level may be automatically controlled. When the level in the tank reaches a high-level set point, the letdown flow is automatically diverted to the liquid waste disposal system. When the level in the tank reaches the low-level set point, makeup water is manually or automatically supplied to the charging pumps. When the level in the tank reaches a low-low set point, the system automatically closes the outlet valve on the tank and switches the suction of the charging pumps to the safety injection and refueling water tank.

The volume control tank can store enough coolant below its normal operating level to compensate for a full to zero power decrease in the primary coolant volume without requiring makeup. The tank is supplied with hydrogen and nitrogen gas. Gases may be vented to the waste gas surge tank.

9.10.2.3 Chemical Control

The CVC purifies and conditions the primary coolant by means of ion exchangers, filters and chemical additives.

Filters located upstream of the purification demineralizers filter out large particles. The purification demineralizers contain a mixed bed resin which removes soluble nuclides by ion exchange and insoluble nuclides by impaction of the particles on the surface of the resin beads. A demineralizer post-filter is located downstream of the purification demineralizers to filter out resin material that may be carried over from the demineralizers. In addition, the filter may be operated as either a prefilter or a post-filter.

The primary coolant is chemically conditioned to the typical conditions shown in Table 4-16 by:

1. Hydrazine scavenging to remove oxygen during start-up

2. Maintaining excess hydrogen concentration to control oxygen concentration during operation

3. Chemical additives to control pH during operation

The chemical addition tank and metering pump are used to feed chemicals to the charging pumps which inject the additives into the Primary Coolant System. The concentration of hydrogen in the primary coolant is controlled by maintaining a hydrogen atmosphere in the volume control tank.

The chemical control system is designed to prevent the activity of the primary coolant from exceeding approximately 292 μCi/cc with failed fuel elements.

9.10.2.4 Reactivity Control

The boron concentration of the primary coolant is controlled by the CVC to:

1. Optimize the position of the control rods.

2. Compensate for reactivity changes in the temperature of the coolant, burnup of the core and variations in the concentration of xenon in the core (see Figure 9-27).

3. Provide a margin of shutdown for maintenance, refueling or emergencies.

The system includes a batching tank for preparing the boric acid solution, two tanks for storing the solution and two pumps for supplying boric acid solution to the makeup system.

Normally, the system adjusts the boron concentration of the primary coolant by "feed" and "bleed." To change concentration, the makeup (feed) system supplies either water or concentrated boric acid to the charging pumps, and the letdown (bleed) stream is diverted to the waste disposal system. Toward the end of a core cycle, the quantities of waste produced due to the "feed" and "bleed" operations become excessive. Then, the deborating demineralizer is used to reduce the boron concentration.

The system adds boron to the primary coolant and thereby decreases reactivity at a sufficient rate to override the maximum increase in reactivity due to cooldown and the decay of xenon in the reactor.

The control rods can decrease reactivity far more rapidly than the boron removal system can increase reactivity. The maximum equivalent reactivity insertion rate of the rods is 143 ppm/min; whereas the maximum boron reduction rate is only 3 ppm/min.

9.10.2.5 Pressure-Leakage Test System

The Primary Coolant System can be tested for leaks while the Plant is at power by monitoring pressurizer level and charging rate. The charging pumps may also be used to hydrostatically test the primary system at design pressure when the Plant is shut down.

9.10.2.6 Component Functional Description

The major components of the Chemical and Volume Control System perform the following functions:

1. Regenerative Heat Exchanger

   The regenerative heat exchanger transfers heat from the letdown stream to the charging stream. Materials of construction are primarily austenitic stainless steel.

2. Letdown Heat Exchanger

   The letdown heat exchanger cools the letdown stream from the tube side of the regenerative heat exchanger to a temperature suitable for entry into the purification demineralizer. Component Cooling System fluid is the cooling medium on the shell side of the letdown heat exchanger, with the letdown stream passing through the tube side. Materials of construction are primarily austenitic stainless steel.

3. Purification Demineralizers

   The two purification demineralizers provide a means of removing undesired ionic species such as activation/fission products and lithium from the primary coolant system. They are configured in one of two ways:

   1) One vessel is loaded with mixed bed resin in the borate/lithium form and the other vessel loaded with cation only resin in the hydrogen form. The borate/lithium demineralizer is used during normal operation to remove ionic specie without removing lithium. The cation demineralizer is placed in service periodically to remove the natural build in of PCS lithium. During PCS source term evolutions the cation only hydrogen form demineralizer is placed in service in series with the anion form deborating borate from demineralizer.

   2) One vessel is loaded with mixed bed resin in the borate/lithium form and the other vessel loaded with mixed bed resin in the borate/hydrogen form. In this configuration the borate/lithium form demineralizer is used during normal operation to remove ionic specie without removing lithium. The borate/hydrogen form demineralizer is placed in service periodically to remove the natural build in of PCS lithium. During PCS source term evolutions the borate/hydrogen form demineralizer is placed in service.

   Each unit is designed to handle maximum letdown flow of 120 gpm. The vessels and retention screens are constructed of austenitic stainless steel.

4. Deborating Demineralizers

The deborating demineralizer is used to remove boron from the primary coolant when this mode of operation is preferable to a feed and bleed operation. The anion resin is initially in the hydroxyl form and is converted to a borated form during boron removal. The unit is designed for the maximum letdown flow of 120 gpm, and the quantity of resin is sufficient to remove the equivalent of 50 ppm of boron from the entire Primary Coolant System. The vessel and retention screens are of austenitic stainless steel construction.

5. Purification Filters

The purification filters collect resin fines and insoluble particulates from the primary coolant. The filters will accommodate maximum letdown flow of 120 gpm. The filter housing is austenitic stainless steel.

6. Volume Control Tank

The volume control tank accumulates water from the Primary Coolant System. The tank has enough capacity to accommodate the variation in water inventory of the Primary Coolant System due to power level changes in excess of that accommodated by the pressurizer. The tank provides a gas space where hydrogen atmosphere is maintained to control the hydrogen concentration in the primary coolant. A vent to waste processing system permits removal of gaseous fission products released from solution in the volume control tank. The tank is of austenitic stainless steel construction and provided with overpressure protection. Level controls release coolant to the waste processing system on high level or operate borated water makeup valves on low level.

7. Charging Pumps

Three charging pumps supply makeup water to the Primary Coolant System. The pumps return coolant to the Primary Coolant System at a rate equal to the purification flow rate and the bleedoff rate. The charging pumps automatically start upon a safety injection signal and discharge concentrated boric acid into the Primary Coolant System. P-55B and P-55C automatically start upon low pressurizer level. The pumps are of the positive displacement type. All wetted parts, except seals, are of austenitic stainless steel. Two of the pumps are fixed capacity pumps while one (P-55A) is a variable capacity pump. Any two of the three pumps shall be capable of providing an output of 68 gpm, with a single pump providing a minimum of 33 gpm. The normal purification flow rate is 40 gpm. Accumulators are located on the suction and discharge of each pump to reduce pump induced vibrations.

8. Chemical Addition Tank

The chemical addition tank is used to prepare chemicals for primary coolant pH control, oxygen control, and source term reduction evolutions. These chemicals are added to the suction of the charging pumps with the metering pump. The tank is austenitic stainless steel.

9. Metering Pump

The metering pump is an air operated double diaphragm pump with wetted parts of austenitic stainless steel. The pump is used to inject a controlled amount of chemicals into the suction of the charging pumps.

10. Concentrated Boric Acid Storage Tanks

Each of the two concentrated boric acid tanks stores enough concentrated boric acid solution to bring the reactor to a cold shutdown condition at any time during the core lifetime. The combined capacity of the tanks will also be sufficient to bring the primary coolant to refueling concentration. The tanks are heated to maintain a temperature above the saturation temperature of the concentrated solution, and sampling connections are used to verify that proper concentration is maintained. The tanks are constructed of stainless steel.

11. Boric Acid Pumps

The two boric acid pumps supply boric acid solution at the desired concentration to the charging pumps through the blender. Upon a safety injection signal, these pumps line up with the charging pumps to permit direct introduction of concentrated boric acid into the Primary Coolant System. Each is capable of supplying boric acid at the maximum demand conditions. Each pump shall provide a minimum flow of 68 gpm. Wetted parts of the pumps are stainless steel.

12. Process Radiation Monitor

The process radiation monitor monitors the fluid from the primary coolant loop for high levels of activity which would provide an indication of failed fuel.

9.10.3 OPERATIONS

9.10.3.1 Start-Up

During start-up, the reactor is brought from cold shutdown to hot standby at normal operating pressure, zero power temperature, with the reactor critical at a low power level. While the primary coolant is being heated, and until the pressurizer steam bubble is established, the charging pumps in combination with the backpressure regulating valves in the CVCS system maintain pressure in the primary system. During the heatup and after the steam bubble is established, the operator adjusts the pressurizer water level manually, with the intermediate pressure letdown control valves, the letdown orifice bypass control valves and/or the letdown orifices. The level controls of the volume control tank automatically divert the letdown flow to the waste disposal system.

If the residual activity in the core is insufficient to reduce the oxygen in the primary coolant by recombining it with excess hydrogen prior to start-up, hydrazine is used to scavenge the oxygen at a primary coolant temperature between 130°F and 150°F. If required, chemicals are added to control the pH of the coolant.

The volume control tank is initially vented to the radioactive waste treatment system. After the tank is purged with nitrogen, a hydrogen atmosphere is established and the vent is secured.

Throughout start-up, one purification filter is in service to reduce the activity of wastes entering the radioactive waste treatment system. When the Primary Coolant System reaches hot standby temperature and pressure, one or both purification ion exchangers are put into service.

Depending on limitations placed on the shutdown margin, the boric acid concentration may be reduced during heatup. The shutdown group of control rods must be in the fully withdrawn position before the operator may start diluting the concentration of boric acid in the Primary Coolant System. The operator may inject a predetermined amount of primary makeup water by operating the system in the dilute mode. The concentration of boric acid in the primary coolant is measured by analyzing samples.

9.10.3.2 Normal Operations

Normal operation includes operating the reactor at hot standby and when it is generating power, with the Primary Coolant System at normal operating pressure and temperature.

During normal operation:

1. Level instrumentation on the pressurizer automatically controls the volume of water in the primary system by adjusting the charging rate of the variable capacity charging pump.

2. Instrumentation on the volume control tank automatically controls the level of water in the tank as described in Subsection 9.10.2.

3. The operator controls the hydrogen concentration and pH of the coolant as described in Subsection 9.10.2.3.

4. The operator may compensate for changes in the reactivity of the core by controlling the concentration of boric acid in the primary coolant. He may operate in three modes.

    a. In the dilute mode, the operator preselects a quantity of primary makeup water and introduces it into the charging pump suction at a preset rate. When the selected quantity of makeup water has been added, the flow is secured upon signal from the primary makeup water batch controller.

    b. In the borate mode, the operator preselects a quantity of concentrated boric acid and introduces it as a preset rate as described in a. above.

    c. In the blend mode, the operator presets the flow rates of the primary makeup water and concentrated boric acid for any blend between primary makeup water and concentrated boric acid. This mode is primarily used to supply makeup to the safety injection and refueling water tank.

9.10.3.3 Shutdown

Plant shutdown is a series of operations which bring the reactor plant from a hot standby condition at normal operating pressure and zero power temperature to a cold shutdown.

Before the plant is cooled down, the volume control tank is vented to the gaseous Radwaste System to reduce the activity and hydrogen concentration in the Primary Coolant System. The operator may also increase the letdown flow rate to accelerate degasification, ion exchange, and filtration of the primary coolant.

Before the plant is cooled down, the operator increases the concentration of boric acid in the primary coolant to the value required for cold shutdown. This is done to assure that the reactor has an adequate shutdown margin throughout its period of cooldown. However, the operator does not insert the shutdown group of control rods until he verifies the concentration of boric acid in the primary coolant by sample analysis.

During cooldown, the operator uses the charging pumps to adjust and maintain the level of water in the pressurizer. The operator can introduce a calculated combination of concentrated boric acid and primary makeup water through the blender into the charging pumps' suction. The flow ratio for each addition is manually selected and provided by the blender inlet valves and controllers. The operator may switch the suction of the charging pumps to the safety injection and refueling water tank, or a portion of the charging flow may be used as an auxiliary spray to cool the pressurizer, when the pressure of the primary system is below that required to operate the primary coolant pumps.

9.10.3.4 Emergency Operations

Under emergency conditions, the charging pumps are used to inject concentrated boric acid into the Primary Coolant System. Either the pressurizer level control or the safety injection signal will automatically start all charging pumps, with the exception that the pressurizer level control does not start P-55A, which is assumed to be operating during normal conditions. The safety injection signal will also cause the charging pump suction to be switched from the volume control tank to the discharge of the boric acid pump. If the boric acid supply from the boric acid pump is not available, boric acid from the concentrated boric acid tanks may be gravity fed into the charging line. If the charging line inside the reactor containment building is inoperative, the line may be isolated outside the reactor containment, and the Safety Injection System may be used to inject concentrated boric acid into the Primary Coolant System.

9.10.4 DESIGN ANALYSIS

1. System Reliability

The CVC is designed for reliability by the provision of redundant critical components to reduce dependence on any single component. Redundancy is provided as follows:

Component	Redundancy
Purification Demineralizer	Parallel Standby Unit
Purification Filters	Parallel Standby Unit
Charging Pump	Two Parallel Standby Units
Letdown Flow Control	Two Parallel Standby Orifices and Valves
Boric Acid Pump and Tank	Parallel Standby Unit

The charging and boric acid pumps are powered by the diesel generators under emergency conditions. One diesel generator supplies Charging Pumps A and B and Boric Acid Pump A. The other diesel generator supplies Charging Pump C and Boric Acid Pump B. Additionally, Charging Pumps B and C can be powered from an alternate power supply (480 volt, Bus 13). Charging Pump B can be powered from the Charging Pump C power supply due to a change made in October 1989 (refer to Section 7.4 for details). Physical separation and barriers are provided between the power and control circuits for the redundant pumps. Standby start features are provided so that at least one charging pump is running. If both diesels are available, both boric acid pumps will be running. The boric acid pumps and the charging pumps may be controlled locally at their switchgear. Separate power supplies for pump power and separate control circuits assure that this system satisfies the single failure criterion.

The boric acid solution is stored in heated tanks and piped in heat-traced lines to preclude precipitation of the boric acid. Two independent and redundant heating systems are provided for the boric acid tanks and lines. Low temperature alarms and automatic temperature controls are included in the heating systems. If the boric acid pumps are not available, boric acid from the concentrated boric acid tanks may be gravity fed into the charging line. If the charging line inside the reactor containment building is inoperative, the charging pump discharge may be routed via the Safety Injection System to inject concentrated boric acid into the Primary Coolant System.

9.10.5 TESTING AND INSPECTION

The operability of the system can be demonstrated by the periodic testing of active components and the cycling of all valves.

9.10.6 REGENERATIVE HEAT EXCHANGER

The Regenerative Heat Exchanger (RHX) is CP Co Design Class 1 and was designed according to the ASME Boiler and Pressure Vessel Code, Section III, Class C (ASME B&PV Code, Section III, Class C) vessel. There are two principal reasons for this:

1. A reliable charging path was the principal reason for originally considering Class A for this component. As the detailed design of the Palisades Plant evolved, it was found desirable to add a two-inch, high-pressure line from the charging pumps through one of the high-pressure safety injection headers and to the primary loop through the four safety injection headers. Thus, an alternate charging path was available. Also, it was felt desirable to have the ability to isolate the RHX by remote manual means. Therefore, isolation valves are located on the inlet and outlet lines of both the shell and tube sides of the RHX as shown on Figure 9-25. These valves can be operated from the control room.

2. The manufacturer of the Palisades RHX was unable to obtain approval from the ASME Code "N" stamp committee to produce ASME B&PV Code, Section III, Class A components. Combustion Engineering (CE) knew of no manufacturer of such heat exchangers who had met the requirements of the "N" stamp committee. CE and the vendor agreed to additional quality control inspections, to be provided by CE, as detailed in subsequent paragraphs.

Combustion Engineering assured that the following requirements were met, which were in addition to those required for a Class C vessel, and which would normally have been performed for a Class A vessel.

1. A fatigue analysis equivalent to the requirements of a Class A vessel was performed by the manufacturer or his consultant. This analysis was reviewed under the direction of a licensed professional engineer at CE to assure its accuracy.

2. The Quality Control requirements of ASME B&PV Code, Section III, Appendix IX, 1965, W67a were met except that shop inspection personnel, although experienced in inspection techniques, did not meet in all respects the qualifications of the applicable standards. Inspections were performed in accordance with written procedures which had been reviewed by CE Quality Assurance (QA) personnel. In addition, CE QA personnel witnessed certain predetermined inspections and also conducted random periodic surveillance inspections. Inspection records were kept at the manufacturer's office and also at Combustion Engineering. Certification of inspection compliance was transmitted to Consumers Power Company.

   In addition to the above, nondestructive testing was witnessed by CE QA personnel who were qualified to ASME B&PV Code, Section III, Appendix IX, 1965, W67a procedures. All nondestructive test procedures were reviewed by CE QA personnel and were deemed acceptable and in accordance with ASME B&PV Code, Section III, Appendix IX, 1965, W67a.

With the aforementioned changes in Plant design, additional analyses and quality control, we believe that Class C vessel classification of the regenerative heat exchanger was justified.

9.11 FUEL HANDLING AND STORAGE SYSTEMS

9.11.1 INTRODUCTION

The Fuel Handling System (Table 9-23) provides for the safe handling and storage of fuel under all foreseeable conditions, from receipt of unirradiated fuel at the Plant to shipment of irradiated fuel following radioactive decay. The design and construction of the system includes interlocks, travel and load limiting devices and other protective measures to minimize the possibility of mishandling or equipment malfunction that could cause damage to the fuel and potential fission product release. Power operation of the system components is supplemented by manual backup to ensure that the transfer of a fuel bundle can be completed in the event of a power failure. The fuel transfer and storage structures, the fuel handling equipment and the new fuel storage racks are CP Co Design Class 1. The high density spent fuel storage racks which replaced the existing racks and the frame supporting the fuel pool crane are Seismic Category I per USNRC Regulatory Guide 1.29.

9.11.2 NEW FUEL STORAGE

The new fuel bundles are stored in rigid racks in a dry pit next to the spent fuel cooling pool. The floor of the pit is open grating to avoid flooding and eliminate the possibility of criticality. The rack is a box-like structure consisting of 72 locations, 36 of which can hold new fuel. The other locations contain steel box beams and core plugs designed for neutron absorption in the event of a heavy mist over the pool such as that produced in fire fighting. The fuel racks can accommodate fuel assemblies enriched to 4.20 weight percent U-235 with 216 UO_2, Gd_2O_3-UO_2 fuel rods or metal rods (Reference 11).

The fuel racks are also used occasionally to store core plugs, poison clusters and spare control elements. During actual refueling, the new fuel bundles are often inspected and placed directly into the spent fuel pool, bypassing the new fuel storage racks.

9.11.3 SPENT FUEL STORAGE

9.11.3.1 Original Design

The spent fuel storage pool, located in the auxiliary building adjacent to the containment, is lined with stainless steel and has reinforced concrete walls and floor varying in thickness from 4-1/2 feet to 6 feet.

The original fuel racks were stainless steel with a center-to-center spacing of 11-1/4 inches. There were two 1/4-inch stainless steel plates between each pair of fuel assemblies. At design temperature, with no credit taken for soluble boron in the pool water, the maximum k_{eff} was less than 0.95. A recessed area was provided in the pool for a spent fuel shipping cask.

The spent fuel pool cooling system (see Section 9.4) is a closed loop system consisting of two half-capacity pumps, a full-capacity heat exchange unit consisting of two heat exchangers in series, a bypass filter, a bypass demineralizer, a booster pump, piping, valves and instrumentation.

The spent fuel pool cooling system is conservatively designed to maintain a pool average temperature at less than 150°F with 1/3 core of fully burned up fuel in the pool, 7 days after reactor shutdown. A single failure of the cooling system would increase the pool temperature by only 3°F. The water in the spent fuel pool is borated to ≥ 1,720 ppm. The entire spent fuel pool cooling system is tornado protected and is located in a CP Co Design Class 1 structure.

Fuel pool makeup water is supplied from the Safety Injection and Refueling Water (SIRW) tank. A secondary backup supply of water is available from the fire system. This would be utilized to replenish the fuel pool water inventory in the event of considerable loss of pool water.

Two fuel tilt pits are located in the fuel handling area adjacent to the spent fuel pool and connected to it by canals which are closed off by dam blocks. One tilt pit is used for normal fuel transfer activities. The second tilt pit originally was provided to accommodate an additional unit on the site.

9.11.3.2 Modified Spent Fuel Storage

In 1977, due to the lack of fuel reprocessing facilities, the spent fuel pool storage capacity was increased from a capacity of 272 assemblies to a capacity of 798 assemblies. This increase in capacity was achieved by removing the existing fuel and control rod racks and replacing them with new racks with smaller center-to-center spacing.

Each individual storage location consists of two concentric 1/8-inch austenitic Type 304 stainless steel square cans with the annular space occupied by B_4C neutron absorber plates to ensure subcriticality.

A rack assembly consists of a rectangular array of storage cans with a minimum 10-1/4 inches center-to-center spacing of the fuel assemblies. The array size of each rack was chosen to optimize the use of the pool space as shown in Figure 9-28. The racks are Seismic Category I per NRC Regulatory Guide 1.29 and are restrained to the pool wall at the top and bottom of each rack to prevent excessive movement of the racks under postulated seismic accelerations. Provisions are made in the design to accommodate thermal expansion.

The cask laydown area may contain one 11 x 11 rack which may be used to store fuel during full core off-loads. This rack may be removed to allow placement of the spent fuel shipping cask, to allow placement of equipment associated with dry fuel storage, or to allow the use of fuel inspection and repair equipment.

The second tilt pit is used for spent fuel and control rod storage and as an alternate cask laydown area. Control rods and dimensionally abnormal fuel assemblies may be stored in one rack with slightly larger cans than those used in the other racks. To minimize heat generation in the tilt pit, normally only fuel decayed for at least one year will be stored there. When fuel with a shorter decay time is stored in the tilt pit, thermal conditions are monitored to ensure that the design criteria is not exceeded.

The Nuclear Waste Policy Act of 1982 required owners of nuclear power plants to diligently pursue licensed alternatives to the use of federal storage capacity for the storage of the spent fuel expected to be generated by that plant before entering into a contract with the Federal Government to provide such storage.

A second modification to the spent fuel storage facility, in 1987, consisted of an increase of the spent fuel pool total storage capacity from 798 assemblies to 892 (see Reference 5). This increase in capacity was accomplished by removing 376 storage locations having a 10.25-inch center-to-center spacing, and replacing them with 470 storage locations having a 9.17-inch center-to-center spacing. The 9.17-inch center-to-center spacing has been accomplished by taking credit for burnup with poison. This "Credit for Burnup Racks With Poison" (CRBP) design will accommodate irradiated fuel that has sustained a predetermined design burnup (23,500 MWD MTU for 3.27 weight-percent initial enrichment). In addition, the new storage locations have more space in each location, and permit 2:1 consolidation if this method is chosen in the future to further expand storage capacity.

This second (1987) modification consisted of reracking approximately one-half of the spent fuel pool and North Tilt Pit dividing the spent fuel pool and North Tilt Pit into two regions, which are designated as Region 1 and Region 2 (see Figure 9-28). The reason only approximately one-half of the spent fuel pool was reracked was due to the need to maintain a portion of the storage capacity with larger center-to-center spacing to accommodate the storage of fuel with little or no burnup. The racks (Region 1), with 10.25 center-to-center spacing, have the required spacing to store such fuel.

Region 1 contains racks in the spent fuel pool having a 10.25-inch center-to-center spacing and a single rack in the North Tilt Pit having 11.25-inch x 10.69-inch center-to-center spacing. Region 2 contains racks in both the spent fuel pool and North Tilt Pit having 9.17-inch center-to-center spacing. Because of the larger center-to-center spacing, and the poison (B^{10}) concentration of Region 1 cells, Region I (NUS) spent fuel storage racks can accommodate fuel assemblies enriched to 4.40 weight percent U-235 provided that fuel assemblies having enrichment above 3.27 weight percent U-235 contain 216 UO_2, Gd_2O_3-UO_2, or solid metal rods (Reference 11). This assures the fuel enrichment limit assumed in the spent fuel analyses will not be exceeded.

The Region 2 racks contain a neutron absorbing material, boraflex, manufactured by the Brand Industrial Services, Inc, and fabricated to the Nuclear Criteria of 10 CFR 50, Appendix B. Boraflex is a silicone-based polymer containing fine particles of boron carbide in a homogeneous matrix. The boraflex used in the Region 2 racks contains a minimum B^{10} areal density of 0.006 gm/cm^2. Since the long-term stability of boraflex has not yet been resolved, the ability to maintain full core off-load capability of 204 vacant spent fuel storage cells, the ability to remove, replace or modify the largest Region 2 rack (which contains 121 storage cells) will be maintained. Thus the ability to replace all the Region 2 boraflex is maintained (see Reference 4).

Each Region 2 rack module is provided with adjustable leveling pads which are located at selected locations within the module. The pads are remotely adjustable from above. All support pads rest directly on the pool liner and/or adapter plates.

9.11.3.3 Structural Analysis

The spent fuel storage racks are designated Seismic Category I per NRC Regulatory Guide 1.29. Structural integrity of these racks was investigated using the loads and load combinations presented in Reference 1, which satisfy the requirements of NRC Standard Review Plan (SRP), Section 3.8.4. Stresses were computed at critical sections of the rack. A comparison of the computed versus allowable stresses indicates that the racks are structurally adequate. A discussion of this analysis, with emphasis on the seismic aspects, is found in Subsection 5.7.6.

The fuel pit and tilt pit floors and walls were analyzed to determine if they could support the fully loaded, high-density, spent fuel racks. A conservative analysis was performed for dead (fully loaded racks, hydrostatic), seismic (inertia of floors/walls, rack reactions, sloshing) and thermal loads. Forces and moments obtained at selected points were combined in accordance with the load combinations presented in Subsection 5.9.1.1.2. The maximum tensile/compressive stress was computed for each load combination and compared with the required yield strength of the structure ("Y"). For the seven walls and two floors analyzed, the minimum factor of safety was found to be 1.1 and the average factor of safety was found to be 4.5. Therefore, it was concluded that the fuel pit and tilt pit floors and walls have adequate strength to safely support the increased fuel storage.

The spent fuel pool structure was designed for ductile behavior (ie, with reinforcing steel stresses controlling the design). The acceptance criteria are stated in Chapter 5, Appendix A of the FSAR. These criteria apply in the structural reanalysis. Acceptance is based on maintaining structural integrity and ductile behavior of the pool structure.

The fuel racks (Region 2) were analyzed for normal and faulted load combinations in accordance with the NRC "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications." The results of this seismic and structural analysis show that the Region 2 racks meet all structural acceptance criteria adequately.

An analysis was performed to demonstrate that the Region 2 rack can withstand a maximum uplift load of 4,000 pounds. This load can be applied to a postulated stuck fuel assembly without violating the criticality acceptance criterion. Resulting stresses are within acceptable stress limits, and there are no changes in rack geometry of a magnitude which cause the criticality acceptance criterion to be violated.

In the unlikely event of dropping a fuel assembly, accidental deformation of the rack will not cause the criticality acceptance criterion to be violated. Criticality calculations show that k_{eff} less than 0.95 and the acceptance criterion is not violated.

Consistent with the criteria of the NRC "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," the racks were evaluated for overturning and sliding displacement due to earthquake conditions under the various conditions of full, partially filled, and empty fuel assembly loadings. The fuel rack nonlinear time history analysis shows that the fuel rack slides a minimal distance. This distance combined with the rack structural deflection and thermal growth is less than rack-to-rack or rack-to-rack clearances. Thus, impact between adjacent rack modules or between rack module and pool is prevented. The factor of safety against overturning is well within the values permitted by Section 3.8.5.II.5 of the Standard Review Plan.

9.11.3.4 <u>Prevention of Criticality During Transfer and Storage</u>

The Region 1 racks in the main pool are designed for a 10-1/4-inch center-to-center spacing with B_4C plates around each assembly, while the Region 2 racks are designed for 9.17-inch center-to-center spacing with boraflex sheets as a neutron-absorbing material. Borated water surrounds the spent fuel storage racks in the same concentration and to a level common to the refueling cavity and pool. The center-to-center distance of the storage racks in both the main pool and the tilt pit is such that a k_{eff} of less than 0.95 is maintained even in the event that unborated water was used to fill the storage areas.

The results of the criticality analysis are for the worst case situations, considering maximum variations in the position of fuel assemblies within the storage rack, neutron absorber positioning, variations in can dimensions, the most reactive temperature, calculational uncertainties and worst case accidents result in a k_{eff} less than 0.95 with a confidence level of 95%.

The Spent Fuel Pool storage racks and fuel elevator meet criticality requirements assuming pure fuel pool water and no other coincident occurrence such as a misplaced assembly (Reference 15). With 1720 ppm boron in the fuel pool, two coincident occurrences would be required (i.e., dilution and a misplaced assembly or a cask drop) in order to violate the criticality acceptance criteria.

In order to maintain a k-effective less than or equal to 0.95 for the tilt machine, 600 ppm boron is required. This is maintained by meeting normal refueling requirements outlined in Technical Specification 3.8 and associated procedures. Credit is taken for boron to maintain k-effective less than or equal to 0.95 with a margin of 1120 ppm being available due to the Technical Specification requirement of 1720 ppm.

9.11.3.5 Radiological Considerations

9.11.3.5.1 Radiation Shielding

Adequate shielding for radiation protection of refueling personnel is provided by the handling of irradiated fuel assemblies under 10 feet of water. An individual fuel rod may be handled under 8 feet of water and still maintain adequate shielding. Mechanical stops are provided on all handling equipment, which limit the height of withdrawal of the irradiated fuel assemblies, to maintain the low level of radiation required for unrestricted occupancy of the area by personnel. An annunciation of low water level is provided.

9.11.3.5.2 Pool Surface Dose

The additional spent fuel assemblies in the pool will result in an increase in dose rates in the spent fuel pool area due to a buildup of radionuclides in the pool water. To determine the amount of increase, a calculational model was devised that considered the presence of activated corrosion products, leakage of the isotopes from the fuel to the pool, the decontamination factor and flow rate of the pool purification system, the isotopic half-lives and the decay time of the fuel. Using this model, the pool's activity was predicted for the present pool capacity (272 assemblies) and for the increased capacity (798 original rerack assemblies). On the refueling platform, 5 feet above the center of the pool, the dose rate increased from 2.17 mrem/h for 272 assemblies to 3.24 mrem/h for 798 assemblies. (At poolside, 1 foot from the pool wall and 5 feet above the surface, the dose rate increased from 1.58 mrem/h to 2.34 mrem/h.) The increase in the pool capacity has a negligible effect on personnel exposure. Assuming an occupancy time of 504 man-hours per year at the refueling platform and 1,134 man-year poolside for refueling operations, and an additional 52 man-hours per year poolside for routine operations, the total incremental dose due to the expansion of pool capacity from 272 to 798 assemblies is 1.43 manrem per year.

To monitor dose rates in the spent fuel pool area, Thermoluminescent Dosimeters (TLD) have been mounted on a wall adjacent to the spent fuel pool since the beginning of Plant operations. The dose rate directly above the spent fuel pool has been measured during routine area surveys on the service platform. Survey sheets were examined for the periods of time between 1975 and 1983 during which the Plant was operating. Thirteen surveys were found with a record of the dose rates on the service platform directly above the spent fuel pool. These measurements ranged from 0.2 to 3.5 millirem per hour. The average dose rate was 1.5 millirem per hour. As with the TLD results, there is no correlation between the dose rate and the number of fuel bundles in the spent fuel pool.

9.11.3.5.3 Airborne Doses

The water evaporation rate, and hence tritium release to the environment around the spent fuel pool, is expected to change as a result of the following factors:

1. Lower calculated water temperatures for the updated FSAR in the spent fuel pool than those evaluated previously in the 1980 FSAR.

2. Higher water temperatures in the north tilt pit area relative to the main pool.

3. Increased water surface area due to utilization of the north tilt pit.

Calculations show that the overall evaporation rate will increase approximately 9%.

Airborne samples of the gross beta-gamma activities from the spent fuel pool were taken during normal operating periods from 1979 through 1983. As with other parameters examined, no correlation could be established between the gross airborne activities and the number of fuel bundles in the spent fuel pool.

9.11.3.5.4 General Area Doses

The adequacy of the spent fuel pool and tilt pit shielding was analyzed with the QAD and ANISN computer codes, to take into account storage of additional spent fuel.

Analyses have shown that the existing shielding is generally adequate to reduce effectively neutron and secondary gamma radiation in all expected areas of occupancy surrounding the pool. However, three areas in which fission product gamma dose rates have exceeded the FSAR radiation zoning criteria (Section 11.6) have been identified. These are (1) outside the north wall of the north tilt pit, (2) outside the north wall of the existing spent fuel pool, and (3) in the space directly below the spent fuel pool cask loading area.

When the north tilt pit is used to store fuel which has decayed for at least one year, it has been calculated that the expected gamma dose rate on the north wall of the tilt pit, which is 2 feet thick, will be approximately 14 rem/h. Studies show that approximately 7 inches of lead equivalent will be required in addition to the 2-foot-thick concrete wall to achieve dose rates consistent with the FSAR radiation zoning criteria. Assuming that the spent fuel pool will be used to store fuel which has decayed for at least 36 hours, it has been calculated that the expected gamma dose rate on the north wall of the pool will exceed 10 mrem/h. Assuming the cask loading area will be used to store fuel which has decayed for at least 36 hours, it has been calculated that the gamma dose rate under the pool floor adjacent to the cask loading area will exceed 200 mrem/h.

9.11.3.5.5 Protection Against Radioactivity Release

Protection against accidental radiation release from irradiated fuel is provided by the containment ventilation system and isolation capability, if required, of the spent fuel pit and auxiliary building ventilation system. Because of the submergence of a bundle in 10 feet of water and an individual rod in 8 feet of water, any released fission products will be diluted and partially retained by the pool water.

The ventilation air for both the containment and spent fuel pool atmospheres flows through absolute particulate filters before discharging to the Plant stack. The containment is normally isolated with purge air only when access to the air room is desired. In the event that the stack discharge should indicate a release in excess of the limits in the Offsite Dose Calculation Manual , an alarm is received in the control room and the ventilation flow path from containment is closed manually from the control room. The ventilation flow paths from the fuel handling area and radwaste area are also manually closed from the control room. In addition, the ventilation flow paths to and from containment are closed automatically upon containment high pressure or containment high radiation (Section 7.3).

During normal operation, the spent fuel pool area exhaust air is pulled through a prefilter and a high-efficiency filter with a particulate efficiency of 99.97% of 0.3 micron particles. The fuel building exhaust fans discharge to the main exhaust fan inlet plenum for ultimate discharge through the ventilation stack.

In the event of a fuel handling / cask (heavy load) drop accident in the spent fuel pool, the exhaust airflow is reduced to one-half by tripping the supply fan and closing the inlet damper and tripping one of the 50% capacity exhaust fans. The exhaust air flows through the high-efficiency particulate filter and a charcoal filter. See Sections 14.11 and 14.19 for specific assumptions of the Fuel Handling Building HVAC used in the safety analysis.

The radiological filter is in a bypass around the normal service filters and designed for a capacity of 10,000 ft^3/min and will retain 1,200 grams of methyl iodide. Its particulate efficiency is 99.97% for particles of 0.3 micron in size, and the filter medium has a test-proven efficiency for removal of radioactive iodine and iodine compounds as follows:

Radioactive Iodine I_2^{131}	99.5%
Radioactive Methyl Iodide $CH_3I_2^{131}$	95.0%

9.11.4 FUEL HANDLING SYSTEM

9.11.4.1 General

Refueling is accomplished by handling fuel bundles underwater at all times. The refueling cavity and spent fuel pool are filled with borated water to a common level during refueling. The use of borated water provides a transparent radiation shield, a cooling medium and a neutron absorber to prevent inadvertent criticality.

The Fuel Handling System transfers the fuel bundles between the refueling cavity and the fuel storage pool through a transfer tube. The refueling machine removes a spent fuel bundle from the core, transports it to the tilt machine and deposits it in the transfer carriage within the tilt machine. The carriage is then rotated from a vertical position to a horizontal position and moved through the transfer tube to the spent fuel storage area. The carriage is then rotated to a vertical position, the spent fuel removed and placed in a storage rack by the service platform. The service platform is also used to remove the fuel from the storage rack and deposit it in the shipping cask for shipment off the site or to deposit it in the Multi-Assembly Storage Basket (MSB) for storage at the Independent Spent Fuel Storage Installation. During all handling operations, a sufficient water shield is maintained over the top of the fuel bundle to restrict radiation exposure to operating personnel. The refueling water boron concentration is checked periodically to assure adequate shutdown margins. Water boron concentration is also checked prior to and during MSB fuel loading.

New fuel bundles are stored dry in the new fuel storage area. This area is provided with vertical racks to hold 36 replacement bundles. New fuel bundles are transported from the storage rack to the new fuel elevator by means of the fuel building overhead crane. The new fuel elevator receives the fuel bundle in its raised position and then travels to the bottom of the fuel pool. Then the fuel bundle will be picked up by the service platform for transportation to one of the designated storage spaces in the storage rack. During refueling the service platform transports the fuel bundle to the transfer carriage. A layout of the refueling system is shown in Figure 9-29.

The new fuel elevator contains an inspection position to allow examination of irradiated fuel. Fuel repairs can be conducted in the elevator. The elevator is also used to transfer neutron sources between fuel assemblies.

9.11.4.2 Fuel Handling Structures

The refueling cavity is a reinforced concrete structure lined with stainless steel that forms a pool above the reactor. During the refueling, the cavity is filled with borated water to a depth which limits the radiation at the surface of the water to 2.5 mrem/h.

To prevent leakage of refueling water from the cavity, the flange of the reactor vessel is temporarily sealed to the bottom of the refueling cavity. This seal is installed after reactor cooldown but prior to the removal of the reactor vessel head and the flooding of the refueling cavity.

The refueling cavity also provides storage space for the upper guide structure, irradiated incore instrumentation, miscellaneous refueling tools and the core support barrel when its removal is required. The reactor vessel head and missile shield are stored on the operating floor.

9.11.4.3 Major Fuel Handling Equipment

1. Reactor Vessel Head Lifting Device

   The head lifting device is composed of a removable spreader bar assembly and a three-part column assembly and the rigging necessary to lift and move the head to the storage area. The column assembly which remains attached to the head also provides a working platform for personnel during maintenance, and supports the three hoists which are provided for handling the hydraulic stud tensioners, the studs, washers and nuts.

2. Upper Guide Structure Lifting Device

   When installed, this device allows the main crane to lift the upper guide structure. Three bolts are threaded into the flange of the upper guide structure using a manually operated tool. Bushings on the lifting device engage the guide studs installed on the reactor vessel flange to provide guidance during removal and insertion of the guide structure. Work platforms are provided for operating personnel and brackets are attached to the lifting device for the storage of withdrawn incore instrumentation.

3. Refueling Machine

The refueling machine is a traveling bridge and trolley which spans the refueling cavity and moves on rails located on the working floor of the containment area. The bridge and trolley motions allow coordinate location of the fuel handling hoist and guide assembly over the fuel in the core. The hoist assembly contains a coupling device which when rotated by the actuator mechanism engages the fuel bundle or control rod to be removed. The hoist assembly is moved in a vertical direction by a cable that is attached to the top of the hoist assembly and runs over a sheave on the hoist cable support to the drum of the hoist winch. After the fuel bundle is raised into the hoist and the hoist into the refueling machine mast, the refueling machine transports the fuel bundle to its new location. The capability to perform In-Mast Sipping of fuel bundles was installed in the Refueling Machine for enhanced detection of fuel defects, in 1996. Horizontal seismic motion is restrained by the bridge and trolley flanged wheels. Vertical seismic upward motion is restrained by uplifters on both the bridge and trolley of both the containment and fuel pool cranes.

The controls for the refueling machine are mounted on a console which is located on the refueling machine trolley. Coordinate location of the bridge and trolley is indicated at the console by digital readout devices which are driven by encoders coupled to the guide rails through rack and pinion gears. A system of pointer and scales is provided as a backup for the remote positioning readout equipment, and manually operated handwheels are provided for bridge, trolley and winch motions in the event of a power loss.

During withdrawal or insertion of a fuel assembly, the load on the hoist cable is monitored at the control console to ensure that movement is not being restricted. Variations from normal loads in excess of hoist load setpoints will stop the motion of the hoist winch mechanism. A zoned mechanical interlock is provided which prevents opening of the fuel grapple and protects against inadvertent dropping of the fuel. A spreader device is provided which spreads adjacent fuel bundles to provide unrestricted removal and insertion. This spreader is part of the mast assembly and is piston-operated after grappling of the fuel bundle. Safety features of the refueling machine are as follows:

a. An anticollision device on the refueling machine mast which will stop bridge and trolley motion. This device consists of a hoop and limit switches to protect the mast from hitting vessel studs, guide structures or walls of refueling cavity.

b. Interlocks which restrict simultaneous operation of either the bridge and trolley or the hoist winch drive mechanism.

c. An interlock which prevents bridge and trolley motion with spreader device actuated.

d. An override switch which must be actuated after fuel hoist operation to allow bridge or trolley motion.

e. Overload and underload switches which stop fuel hoist motion.

f. Bridge and trolley speed restriction zones over the reactor core.

g. Fuel hoist speed restriction while fuel bundle is within the core.

h. An interlock which prevents positioning of refueling machine over the tilting machine unless the tilting machine is in the vertical position.

4. Tilting Machines

Two tilting machines are provided, one in the containment building and the other in the fuel building. The tilting machine installed in the containment building consists of a fabricated hollow rectangular structure, supported through a pivot to a triangular-shaped support base. This structure is closed at one end and open at the other, which allows the transfer carriage to move completely into the structure by riding on the rails attached to the inner sides. Hydraulic cylinders attached to both the box and the frame are provided to rotate the transfer carriage to a vertical position and then to a horizontal position, as required by the fuel bundle transfer procedure. Slots are cut in the top and bottom surfaces of the box to accommodate the transfer carriage drive cables during the tilting operation.

The tilting machine installed in the fuel storage area is essentially as described above except that the box structure is open at both ends to allow the insertion and transfer of the fuel assemblies. A track is, therefore, provided to mate with the end rollers of the transfer carriage to support the weight of the transfer carriage and the fuel assemblies during the tilting operations.

Interlocks are provided to ensure the safe operation of this equipment by (1) prohibiting the lowering of a fuel bundle unless the transfer carriage has been correctly positioned in the tilting machine, (2) preventing inadvertent rotation of the tilting mechanism while a fuel bundle is being lowered, and (3) deactivating the cable drive so that a premature attempt to move the transfer carriage through the refueling tube cannot be initiated.

5. <u>Transfer Carriage</u>

A transfer carriage is provided to transport the fuel bundles from the refueling cavity through the transfer tube to the spent fuel storage area. Two main structural members form the sides of the carrier from which are supported two fuel assembly cavities and the associated bracing. The carrier rolls on rails through the transfer tube. Stainless steel wire cables connect the carrier to a drive assembly which provides the motive force. The location of the cable connections is such that during the tilting operations, a minimum of cable slack will be encountered and this slack will be automatically taken up when the vertical or horizontal stop positions are reached. Rollers on one end transfer the load of the carrier and fuel assembly to the track of the tilting machine in the fuel storage area.

The carriage has been provided with two fuel bundle locations to minimize the time required for one complete fuel transfer cycle. After the transfer carriage containing a new fuel bundle is moved into the containment area and is tilted to the vertical position, the refueling machine can deposit a spent fuel bundle into one location and remove the new bundle from the other, thus allowing parallel operation of each piece of equipment. The fuel positions in the transfer carriage are located to allow the refueling machine to move from one position to the other by utilizing only bridge motion.

6. <u>Transfer Rails</u>

This is an assembly which contains the rails on which the transfer carriage rides when moving between the refueling cavity and fuel storage area. The rail supports seat on and are welded to the ID of the 36-inch diameter transfer tube and a groove is provided to mate with the key affixed to the supports which keep the rails aligned. The rail assemblies are fabricated to a length which will allow them to be lowered for installation in the transfer tube. A gap is left in the track at the 36-inch diameter valve on the fuel storage side of the transfer tube to allow closing of the valve.

7. <u>Communications</u>

Direct audible communication between the control room and the refueling machine operator is available whenever changes in core geometry are taking place.

This provision allows the control room operator to inform the refueling machine operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

8. Fuel Building Crane

The fuel building crane is a 100-ton indoor electric overhead traveling bridge, single trolley crane, with radio controlled operator unit. Table 9-24 describes specifics of the fuel building crane. The fuel building crane is used to handle the spent fuel cask. The spent fuel cask is described in Sections 9.11.5 and 14.11.

Codes and Standards

The crane was designed, constructed and erected in accordance with the requirements of:

a. Electric Overhead Crane Institute Specification 61 - Class A

b. American Welding Society Standard Specifications

c. National Electric Manufacturers Association

d. American Standards Association

e. National Electrical Code

f. National Fire Protection Association

Factors of Safety

The following minimum factors of safety, under static full rated load stresses and based on ultimate strength of material were provided:

Material	Factor of Safety
Cast Iron	12
Cast Steel	8
Structural Steel	5
Forged Steel	5
Cables	5
Weld	5 (Based on ultimate strength of metal in weld)
Stainless Steel	5

Explicitly, the factors of safety are:

a. Hooks, shear blocks, bridge and trolley drives, complete hoisting mechanism, trolley frames and structural steel parts, not including bridge girders - not less than a safety factor of 5

b. Bridge girders - not less than 5

c. Welds - not less than 5

d. Rope - not less than 5

Mechanical Stress Analysis

In addition to the usual design requirements given in the referenced codes, the equipment is designed to meet seismic requirements as stated below.

The stresses resulting from the following seismic loads combined with normal operating stresses in no case exceed the yield point of the component materials. The seismic load was calculated as 60% of the dead load applied in any horizontal direction and 15% of the dead load applied in either direction vertically. The criteria is only applied to the unloaded crane.

Positive means are provided to prevent the crane bridge, trolley or any other items normally held by gravity from becoming dislodged and falling on equipment or structures situated below the crane.

Brakes

The main hoist is equipped with two magnetic-operated two-shoe brakes. The two-shoe holding brakes are spring-set power-released magnetic brakes, used on the motor shaft and capable of exerting 380 ft-lb torque. The two-shoe brakes are capable of holding one-and-one-half times crane capacity when power to main hoist motor is off. The main hoist is controlled by a Flux Vector Drive system which controls the motor such that lowering and hoisting speeds can be maintained at low speeds for extended periods of time. The Flux Vector Drive is equipped with a dynamic brake which diverts excess current during lowering of the hoist with a load on the hook.

The auxiliary hoist is equipped with a magnetic-operated two-shoe brake identical to the main hoist holding brakes discussed above except that the brake is adjusted to exert 285 ft-lb torque. The brake is capable of holding one-and-one-half times full load motor torque when power to auxiliary hoist motor is off.

The bridge is equipped with a brake identical, except in size, to the magnetic-operated two-shoe main hoist holding brakes discussed above. The brake is capable of exerting 75 ft-lb torque. The brake is rated at 100% of full load motor torque and will automatically set when power is not available to the bridge motor.

The trolley is equipped with a brake identical, except in size, to the magnetic-operated two-shoe main hoist holding brakes discussed above. The brake is capable of exerting 50 ft-lb torque. The brake is rated at 100% of full load motor torque when power to the trolley drive is off.

All brakes are equally effective in both directions.

Two Blocking

Two blocking occurs when block and tackle meet.

Two blocking of main hoist could result, with full-rated load on hook, if both upper limit switches fail while hoist control is in the hoist position. Failure of both upper limit switches is not considered to be credible.

The main hoist magnetic-operated two-shoe brake operates on the shaft of the motor. A dc magnet on the brake overcomes spring pressure to release the brake when energized. The main hoist motor is rated at 50 hp, 1200 r/min, with 5-step control in either direction. The main hoist holding brake will not prevent two blocking of the hoist under rated load conditions.

Two blocking of auxiliary hoist could result with full-rated load on hook, if upper limit switch fails while hoist control is in the hoist position.

The auxiliary hoist motor is rated at 40 hp, 1,200 r/min, with stepless control in either direction. The rated full load torque for this motor in on the order of 180 ft-lb. The corresponding motor stall torque is 495 ft-lb, which exceeds the brake capacity of 285 ft-lb plus full load rating of 180 ft-lb. The auxiliary hoist holding brake will not prevent two blocking of the hoist under rated load conditions.

Hoist Drive System

For the 100-ton hoist, the hoist drive is driven by a 250:1 ratio gearbox.

Hoisting machinery consists of a continuous duty, phasor drive, Class H insulation Reuland motor which drives through necessary gear reductions to a winding drum. Gears in reduction units are mounted on short shafts and supported between bearings. The drum gear is pressed on and keyed to the hub of the winding drum. The hoist motor is flexibly coupled to the speed reducer.

The hoist drum is mounted on pedestal bearings supported on a trolley truck assembly.

The hoist drive motor and gearbox are attached to the trolley truck.

An essentially identical arrangement exists for the 15-ton auxiliary hoist drive system.

Limit Switches

The main hoist has control circuit screw-type upper and lower limit switches and a redundant block-operated, paddle-type upper limit switch. These limit switches serve to interrupt current to the motor when the hoist block reaches or exceeds a predetermined limit of travel, thus setting brakes. Limit switches are reset automatically by moving controller to opposite direction.

The auxiliary hoist has control circuit screw-type upper and lower limit switches capable of setting its brake.

Automatic reset-type limit switches of the forked lever type have been provided to limit travel of bridge on each end of the frame runway. The limit switch is reset by reversal of bridge direction of travel.

An equivalent arrangement to that discussed for the bridge has been provided to limit trolley travel at each end.

Finally, limit switches have been provided to prevent traversal of the fuel pool. Under fuel transfer cask handling operations, the limit switches may be bypassed by a key kept under strict administrative control to allow placing cask in loading area of pool.

Controls

Control of all crane functions is from a radio controlled station carried by the crane operator.

The radio controlled station weighs about 7 pounds and has a master key lock power (on-off) switch with additional key lock switches for fuel pool and cask laydown overrides.

The radio control station is housed in a NEMA I enclosure with four-dead man style, spring-return, detent rotary switches for speed control. The master main (on-off) switch is a heavy duty, toggle-type with a mechanical latch required for the on position.

The bridge and trolley drive controllers are three speed, full magnetic with protection, furnished with NEMA I steel enclosures, NEMA Class 162 unbreakable resistors, general duty master switches, and are mounted on the crane for ease of maintenance and convenience. They are standard GE Type IC 7427A reversing-plugging controllers. Movement of master switch to first point closes the correct directional contactor to place all starting resistance in the circuit. Accelerating points are controlled by automatic relays which cut out resistance until full speed is attained. Quick reversal of master switch results in immediate reversal of directional contactors but acceleration contactors are held open by plugging relay until motor has stopped and reversed. Some braking is accomplished by plugging motor; however, controlled stopping is accomplished through holding brakes.

The auxiliary hoist is provided with a GE IC 7415 silicon controlled rectifier (SCR) controller, providing stepless speed control in both hoist and lower directions. It is characterized by accuracy of speed control and smoothness of stopping. Reversing is accomplished by hoist and lower contactors in the primary circuit of the motor. Speed control is accomplished by varying the firing angle of the SCRs to result in sufficient ac voltage at the motor terminals to produce the required torque and speed. The required firing angle to do this is automatically controlled by employing a transistorized regulator. Use of the regulator makes it possible to control speed with essentially flat speed torque characteristics. The auxiliary hoist is also provided with a tachometer overspeed switch set at approximately 120% for overspeed protection.

The main hoist drive control is a Flux Vector Drive computer controller mounted in an existing NEMA 1 enclosure.

Electrical

The electrical systems furnished are 3 phase, 3 wire, 60 hertz, 460 volt, ac power. Power is provided through the main disconnect switch located on the crane to all motors, drives and controls. There is an additional fusible disconnect switch located on the spent fuel floor to control the power to the crane.

A main line contactor is provided and is operated by stop and reset buttons located conveniently for the operator. A control circuit transformer with fuses provides 110 volt control power to all control panels on the crane with the exception of the main hoist motor Flux Vector Drive system. Low voltage protection is included. Overload protection for the motors is included on the individual motor control panels at 125% overcurrent, with the exception of the main hoist motor which is controlled by the Flux Vector Drive computer controls.

The wire sizes are suitable for crane rated motors in accordance with the National Electrical Code. All insulation, conduit and fittings conform to the requirements of the National Electrical Code.

9. Spent Fuel Cask Lifting Device

When the shipment of spent fuel is feasible, a special spent fuel cask lifting device shall be used. This device shall conform to the standards of ANSI N14.6-1986 or 1978 and the recommendations of NUREG-0612.

9.11.4.4 System Evaluation

Underwater transfer of spent fuel provides ease and safety in handling operations. Water is an effective, economic and transparent radiation shield and a reliable cooling medium for removal of decay heat.

Basic provisions to ensure the safety of refueling operations are:

1. Gamma radiation levels in the containment and fuel storage areas are continuously monitored. These monitors provide an audible alarm at the initiating detector and in the control room indicating an unsafe condition. Continuous monitoring in the control room of reactor neutron flux provides immediate indication and alarm of an abnormal core flux level.

2. Violation of containment integrity is not permitted when the reactor vessel head is removed unless an adequate shutdown margin is maintained.

3. The required refueling boron concentration in the refueling cavity is sufficient to maintain the reactor subcritical by 5% $\Delta\rho$ with all control rods withdrawn. Administrative controls employed during the movement and placement of fuel within the refueling cavity ensure that the 5% $\Delta\rho$ subcriticality margin is maintained during Refueling Operations.

9.11.4.5 Test Program

In addition to the inspections and testing which were performed on individual components as they were fabricated, the major refueling items were shipped to Windsor, Connecticut, where they were assembled at a facility which allowed acceptance and performance testing of the equipment as a complete system. The testing facility simulated the refueling conditions of the Palisades site to allow assembly of the complete refueling configuration. The reactor tilting machine was positioned adjacent to a core and pressure vessel mock-up which was assembled in a pit. Rails were installed between the spent fuel pool tilting machine and the reactor tilting machine. With the refueling machine mounted on rails over the core mock-up, simulated refueling operations were performed as follows:

1. Indexing the refueling machine to the fuel assembly in the core

2. Engaging and lifting the fuel assembly into the fuel hoist

3. Indexing the refueling machine to the tilting machine and lowering the fuel assembly into the carriage

4. Operation of the transfer system to tilt the carriage to the horizontal, transfer it through the simulated refueling tube to the spent fuel pool tilting machine and to tilt the carriage back to the vertical

Heaters were installed in the bottom of the pit to simulate the turbulence caused by decay heat generation.

Subsequent to the completion of this test program at Windsor, the equipment was disassembled and shipped to the site. It was reassembled and sufficient tests performed to demonstrate that it met system requirements. This was part of the preoperational test program performed before fuel loading.

9.11.5 SPENT FUEL STORAGE AT THE INDEPENDENT SPENT FUEL STORAGE INSTALLATION

Approval to store spent fuel at the Palisades Independent Spent Fuel Storage Installation (ISFSI) is granted by the NRC via Subpart K of 10CFR72. Palisades chose the Pacific Sierra Nuclear VSC-24 system for the ISFSI. The VSC-24 system was determined by the NRC to meet the requirements of 10CFR72 by the NRC's issuance of the Certificate of Compliance on May 7, 1993. The license for each individual cask expires 20 years from the inservice date.

The VSC-24 system places a Multi-Assembly Sealed Basket (MSB) and a MSB Transfer Cask (MTC) in the cask loading area of the spent fuel pool where the MSB is loaded with spent fuel. Once loaded, the MSB is transported inside the MTC to the decontamination pit using the fuel building crane, where the MSB is seal welded, dried, backfilled with helium, and then structurally welded. Using the MTC and the crane, the MSB is then transported to track alley, where the MSB is lowered into a Ventilated Concrete Cask (VCC). A Load Distribution System (LDS) is installed in track alley to assure that the load is distributed to the walls supporting track alley. The loaded VCC, or Ventilated Storage Cask (VSC), is then transported along the LDS to a trailer which carries the VSC to the ISFSI where it is stored. The VSC-24 system is described in detail in the Pacific Sierra Nuclear Safety Analysis Report (Reference 21).

9.11.5.1 Description Of 10CFR72 License Items Which Interface With The 10CFR50 License

This section describes various ISFSI items licensed under 10CFR72 that interface with equipment licensed under 10CFR50. Items being evaluated under 10CFR72.48 that affect interfaces such as these require a 10CFR50.59 evaluation in addition to the 10CFR72.48 evaluation.

9.11.5.1.1 Multi-Assembly Sealed Basket And Transfer Cask

The MSB and MTC shell and internals are coated to prevent detrimental effects on fuel pool water chemistry during the time that the MSB and the MTC are in the pool. The MTC shell and internals and the outer shell of the MSB are coated with a paint that facilitates deconning the MTC and the MSB upon removal from the pool.

As preparation for fuel loading, an empty MSB is inserted into the MTC and filled with borated water. MTC-MSB gap shims are placed in the top of the MTC-MSB gap for shielding. A water supply hose is connected to the side of the MTC to allow clean borated water to be injected into the MTC-MSB gap during the entire time the MTC and MSB are submerged in the pool. This prevents the outside of the MSB from becoming contaminated due to contact with fuel pool water and loosened crud from the fuel assemblies.

The MTC and the MTC yoke are special lifting devices designed and fabricated to the requirements of NUREG 0612 and ANSI N14.6 per the Certificate of Compliance. The lifting of the MTC is performed under site specific heavy load requirements which conform to the recommended guidelines of NUREG 0612 and Generic Letter 85-11.

9.11.5.1.2 Impact Limiting Pads

Impact limiting pads (ILPs) are placed in the spent fuel pool and the decontamination pit when loading fuel. The required pressure ratings for the foam in each of the ILPs was determined by calculating the critical pressures on the slabs beneath the cask loading area and the cask washdown pit. These critical pressures were used to determine the minimum strength of the foam to be placed in the ILPs. Also, the bottom of the spent fuel pool ILP is designed with groove areas to prevent any load from bearing on the pool liner welds. Additional information on the impact limiting pads, including reference calculations, is found in section 14.11 "Postulated Cask Drop Accidents."

9.11.5.1.3 Security For The Independent Spent Fuel Storage Installation

The security system for the ISFSI was installed by Facility Change FC-925. The system was constucted and is maintained per the requirements of 10CFR50 and 10CFR72 Subpart H.

9.11.5.1.4 Lifting Equipment

The design and description of the rigging equipment used to handle the VSC-24 system components in the auxiliary building is described in Reference 18. Where applicable or required per the C of C, the SAR and the Technical Specifications, the design code requirements of NUREG-0612 and ANSI N14.6 were applied.

9.11.5.1.5 Spent Fuel Pool Boron And Temperature Limits

The C of C for the VSC-24 system requires that the spent fuel pool boron concentration be greater than or equal to 2850 ppm during cask loading and unloading activities (Reference 19). This ensures that a subcritical configuration is maintained in the case of accidental loading of the MSB with unirradiated fuel.

The reaction between carbo zinc primer and borated spent fuel pool water will generate a hydrogen gas and produce zinc borate precipitates which can result in boron depletion. Controls are provided during loading and unloading activities to prevent the hydrogen gas from reaching an explosive level and to assure that the boron concentration will not be reduced below the C of C limit of 2850 ppm.

An analysis was performed to determine the amount of time that exists until the MSB must be drained after removal of the loaded MSB from the fuel pool (Reference 20). The MSB must be drained before significant changes in moderator density occur due to heatup of fuel pool water in the MSB. The analysis, which uses a maximum fuel pool temperature of 100°F, calculated required MSB drain down times due to certain fuel heat loads. These drain down times are utilized as administrative limits in plant MSB loading procedures and are conservative with respect to the legal drain down time limits contained in the C of C.

The strength of the fuel pool ILP is based on a maximum pool water temperature of 100°F.

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Table 9-1

Service Water System Flow Requirements (GPM)[3]

Component Name	Normal Operation (Nominal)	Shutdown Operation (Nominal	DBA Operation SW = 85°F
Critical Service Water Header			
Containment Air Coolers	2000-7500	2000-7500	3935.5[7]
CCW Heat Exchangers[2]	<6000	<6000	4058.0[8] 4821.4[12]
Engineered Safeguards Room Coolers	400[4]	400[4]	70[9]
Emergency Diesel Generators	-	-	165[10]
Control Room Air Conditioning	12[5]	12[5]	39[11]
Instrument Air Compressors	10	10	-
Noncritical Service Water Header[6]			
Hydrogen Coolers	2610[1]	-	-
Exciter Air Coolers	370[1]	-	-
Turbine Lube Oil Coolers	2510[1]	-	-
Seal Oil Coolers	360	-	-
EHC Oil Coolers[5]	20	-	-
Isolated Phase Bus Cooler[5]	35	-	-
Main Feedwater Pump Lube Oil Cooler	78	-	-
Main Feedwater Pump Gland Cooler	40	-	-
Heater Drain Pump Cooling	50	-	-
VRS Isolated Cooling Water Cooler (E-68)	5	-	-
Blowdown Heat Exchanger (E-31)	60-420	-	-
Circ Water and Intake Basin Chlorinator	50	50	-
Hydrogen Dryer	2	-	-
Cooling tower Pump Seal and Bearing Water	94	-	-
Makeup Raw Water Supply	150	150	-
Condensor Vacuum Pump	-	15	-
Aux Building Addition Air Conditioning Unit	28.8	28.8	-
Ventilation Equipment Room Air Cooling Unit	28.8	28.8	-
Radwaste Area Compressor	10	10	-
Auxiliary Building Condensing Unit	130	130	-

Service Water System Flow Requirements (GPM)[3]

Component Name	Normal Operation (Nominal)	Shutdown Operation (Nominal	DBA Operation SW = 85°F
FWP Air Compressor	65.2	65.2	-
C-42 Panel and Sample Coolers	40	-	-
FWS Sample Cooler SCI-0710-C (to M-97(B))	40	-	-
Condensate Pumps	16	-	-
CD Bldg Boiler Sample (measured) Cooler	1.6	1.6	-
Radiation Monitors	10	-	-

NOTES:
(1) - Flow is temperature controlled.
(2) - DBA Requirement for Post-RAS mode only, flow is temperature controlled prior to RAS.
(3) - The flows listed for DBA operation are required flows at that SW temperature, the actual flow to each component is set periodically by Special Test T-216, which balances the system flows.
(4) - SW flows continuously to the ESGR Coolers. There is minimum heat load in the rooms during normal operation and slightly more heat load (SDC System) during shutdown operation. The 400 gpm is not a cooling requirement but is the approximate indicated flow.
(5) - Only one unit (or set of coolers) is operated at one time.
(6) - Flows listed here are from original Bechtel figures and have not been verified.
(7) - EA-GEJ-96-01 Rev 0. - Total of 3935.5 gpm to VHX-1,2,3 in left channel failure case.
(8) - EA-GEJ-96-01 Rev 0. - Total of 4058.0 gpm to both E-54A,B in right channel failure case.
(9) - EA-D-PAL-93-272F-01 Rev 0.
(10)- EA-D-PAL-93-272F-02 Rev 0.
(11)- EA-D-PAL-93-272E-02 Rev 0.
(12)- EA-GEJ-96-01 Rev 0. - Total of 4821.4 gpm to E-54A,B in left channel failure case.

SERVICE WATER SYSTEM
DESIGN RATINGS AND CONSTRUCTION OF COMPONENTS

1. Service Water Pumps

Type	Vertical Turbine With Water Lubrication
Number	3
Capacity (Each)	8,000 gpm*
Head	140 ft*
Pump Accelerating Time	4 Seconds @ 70% Voltage
Material	
Bowls	Cast Iron
Discharge Head	Carbon Steel
Bowl Shaft	416 SS
Line Shaft	1020 CS
Discharge Column	Carbon Steel
Impeller	Bronze
Motor	350 hp, 3 Ph, 60 Hz, 2,300 V
Codes	Standards of Hydraulic Institute, NEMA, ASA and ASTM

* The FSAR requirement of 8000 gpm and 140 ft of head is a design characteristic that was supplied to the vendor for individual pump performance.

2. Basket Strainers

Type	Simplex Multi-Basket
Number	3
Design Flow (Each)	8,000 gpm
Design Pressure	100 psig
Design Temperature	70°F
Screen Mesh	3/16 in Perforation
Material	
Body	Cast Steel
Baskets	304 SS

3. Piping, Fittings and Valves

Material	Carbon Steel or Bronze
Design Pressure	100 psig
Design Temperature	300°F
Piping and Fittings(a)	2-1/2 in and Larger - Butt- Welded Except at Flanged Equipment(a)
	2 in and Smaller - Socket Welded Except at Flanged Equipment
Valves(a)	2-1/2 in and Larger - Butt-Welded 150 lb(a)
	2 in and Smaller - Socket Welded 600 lb
Code	ASA B31.1-1955 ASA B16.5-1961

(a) These are considered to be classified as flanged equipment.

TABLE 9-3

Deleted

TABLE 9-4
(Sheet 1 of 4)

REACTOR PRIMARY SHIELD COOLING SYSTEM
DESIGN RATINGS AND CONSTRUCTION OF COMPONENTS

1. Shield Cooling Coils

Length (Each Coil)	Approx 24 ft
Spacing of Coils	9 in Center-to-Center
Number of Coil Sections in Each Set	3
Coil Diameter	3/4 in
Material	Seamless Carbon Steel
Design Pressure	75 psig
Design Temperature	220°F
Code	ASA B31.1

2. Shield Cooling Pumps

Type	Horizontal Centrifugal With Mechanical Seals
Number	2
Capacity (Each)	180 gpm
TDH	79 ft
Material	
Case	Cast Iron
Impeller	Bronze
Shaft	Carbon Steel

TABLE 9-4
(Sheet 2 of 4)

Motor	7.5 hp, 3 Ph, 60 Hz, 460 V, 3,485 r/min	
Codes	Standards of Hydraulic Institute, NEMA, ASA and ASTM	

3. Shield Cooling Heat Exchanger

Type	Horizontal Counterflow, Shell With Straight Tubes Rolled Into Fixed Tube Sheets	
Number	1	
Original Design Duty	200,000 Btu/h	
Original Heat Transfer Area	77 ft^2	
	Shell Side	Tube Side
Design Pressure	150 psig	125 psig
Design Temperature	200°F	200°F
Fluid	Component Cooling Water	Shield Cooling Water
Temperature In	90°F	100°F
Temperature Out	93.2°F	96.8°F
Material		
Shell	Carbon Steel	
Tubes	Admiralty	

TABLE 9-4
(Sheet 3 of 4)

Channels	Carbon Steel
Tube Sheets	Aluminum Bronze
Codes	ASME B&PV Code, Section III, Class C and ASME B&PV Code, Section VIII, Par UW-2 (a); TEMA Class C

4. Shield Cooling Surge Tank

Type	Vertical
Number	1
Design Pressure	50 psig
Design Temperature	200°F
Volume	1,700 Gallons (Based on Total Change in System Water Volume as a Result of Maximum Possible Change in Water Temperature From a Cold Start Condition at 60°F to 212°F)
Material	Carbon Steel
Code	ASME B&PV Code Section III, Class C and ASME B&PV Code, Section VIII, Par UW-2 (a)

TABLE 9-4
(Sheet 4 of 4)

5. Piping, Fittings and Valves

Material	Seamless Carbon Steel	
Design Pressure	125 psig	
Design Temperature	360°F	
	Not Embedded in Concrete	Embedded in Concrete
Construction		
Pipe, 2-1/2 in and Larger	Butt-Welded Except at Flanged Equipment	None
Pipe, 2 in and Smaller	Screwed Except at Flanged Equipment	Socket Welded
Valves, 2-1/2 in and Larger	Cast Iron, Flanged 125#	None
Valves, 2 in and Smaller	Bronze, Screwed 200# and Carbon Steel, flanged 150#	None
Code	ASA B31.1	

TABLE 9-5

COMPONENT COOLING SYSTEM HEAT LOADS

($x\ 10^6$ Btu/hr)

Component and (Number)	Normal	Shutdown Cooling		DBA	
		Initial	+30 h	SI	Post-RAS*
Shutdown Cooling HXs (2)	-	147.39 (max)	46.13	-	100.4 (max)
Primary and Auxiliary Systems Sample Cooling Coils (9)	0.3	0.15	Negligible	-	-
Letdown HX (1)	11.8	1.5	Negligible	11.8	-
CRDM Seal Coolers (45)	.07	.07	.07	.07	-
Charging Pumps (3)	0.20	0.12	0.12	0.20	0.20
Primary Coolant Pumps (4)	2.3	1.15	-	2.3	-
LPSI Pumps (2)	-	0.09	0.09	0.09	***
HPSI Pumps (2)	-	-	-	0.11	0.11
Containment Spray Pumps (3)	-	-	-		0.08
Spent Fuel Pool HX (1)	6.2	6.2	6.2**	-	-
Reactor Shield Cooling HX (1)	0.2	0.2	0.2	0.2	-
Waste Gas Compressors (3)	0.01	0.01	0.01	-	-
Vacuum Degasifier Pump Seal Water Cooler (1)	0.08	0.08	0.08	-	-
Radwaste Evaporators (2)	23.52	11.76	11.76	-	-
Total	44.77	168.72	64.66	14.77	100.79

* With containment high pressure

** Maximum heat load at 7 days after shutdown = 11.6 for 1/3 core off load

*** An additional Post-RAS heat load of up to $0.09X10^6$ Btu/hr could exist if LPSI pumps are used for post accident. However, LPSI pumps are not normally operating Post-RAS.

TABLE 9-6
(Sheet 1 of 4)

COMPONENT COOLING WATER SYSTEM
DESIGN RATINGS AND CONSTRUCTION OF COMPONENTS

1. Component Cooling Pumps

Type	Horizontal Centrifugal With Mechanical Seals
Number	3
Capacity (Each)	6,000 gpm (Based on Shutdown Cooling Requirements), Including Approximately 10% Wear Margin
Head	164 ft
Material	
Case	Carbon Steel
Impeller	Bronze
Shaft	Alloy Steel
Temperature Transient	Designed To Withstand Increase of 25°F in 1-1/2 Minutes. This May Occur When System Switches to Shutdown Cooling or Post-DBA Cooling From Normal Operation.
Motor	300 hp, 3 Ph, 60 Hz, 2,300 V
Time Required To Accelerate Pump to Full Speed at 70% Voltage	4 s
Codes	Standards of Hydraulic Institute, NEMA, ASA and ASTM

TABLE 9-6
(Sheet 2 of 4)

2. Component Cooling Heat Exchangers

Type	Horizontal, Counterflow, Shell Straight Tubes, Tubes Rolled Into Tube Sheets	
Number	2	
Original Design Duty (Each)	50.5×10^6 Btu/h (Normal) 94.8×10^6 Btu/h (At Start of Shutdown Cooling) 43.2×10^6 Btu/h (24 Hours After Shutdown Cooling) 85.0×10^6 Btu/h (Post-DBA)	
Original Heat Transfer Area (Each)	7,840 ft^2	
	Shell Side	Tube Side
Design Pressure	150 psig	125 psig
Design Temperature	200°F	200°F
Design Capacity (Each)	4700 gpm (Ref 13 and 14)	
Temperature Transient	Designed To Withstand Increase of 25°F in 1-1/2 Minutes. This May Occur When System Switches to Shutdown Cooling or Post-DBA Cooling From Normal Operation.	
Material		
Shell Side	Carbon Steel, Firebox Quality	
Tube Side	Admiralty	
Tube Sheet	Carbon Steel With Aluminum Bronze Overlay	
Codes	ASME B&PV Code, Section III, Class C, 1965 and ASME B&PV Code, Section VIII, Par UW-2 (a); TEMA Class C	

TABLE 9-6
(Sheet 3 of 4)

3. Surge Tank

Type	Vertical
Number	1
Design Pressure	25 psig
Design Temperature	140°F
Volume	1,230 Gallons (Based on Total Change in System Water Volume as Result of Maximum Possible Change in Water Temperature From Cold Start Condition at 60°F to 140°F)
Material	Carbon Steel
Code	ASME B&PV Code, Section III, Class C, 1965

4. Piping, Fittings and Valves

Piping Material	Carbon Steel, Seamless and Seam Welded (Seam Weld 100% Radiographed)
Design Pressure	150 psig
Design Temperature	140°F

TABLE 9-6
(Sheet 4 of 4)

	Outside Containment	Inside Containment
Construction		
Pipe, 2-1/2 in and Larger	Butt-Welded Except at Flanged Equipment, 10% of Circumferential Welds Examined by Radiography	Butt-Welded Except at Flanged Equipment
Pipe, 2 in and Smaller	Socket Welded Except Flanged Equipment	Screwed Except at Flanged Equipment
Valves (Except Butterfly), 2-1/2 in and Larger	Carbon Steel, Butt Weld Ends, 150#	Cast Iron, Flanged Ends, 125#
Butterfly Valves, 2-1/2 in and Larger	Carbon Steel, Flanged Ends, 150#	Carbon Steel, Flanged Ends, 150#
Valves, 2 in and Smaller	Carbon Steel, Socket Welded Ends, 600#	Bronze, Screwed Ends, 200#
Code	ASA B31.1-1955 ASA B16.5-1961	ASA B31.1-1955 ASA B16.5-1961

TABLE 9-7

COMPONENT COOLING SYSTEM
REQUIRED FLOW RATES (GPM)

Component and (Number)	Normal	Shutdown Cooling Initial	Shutdown Cooling +30 h	DBA SI	DBA Post-RAS*
Shutdown Cooling HXs (2)	-	5,000	5,000	-	4,973/5,493
Primary and Auxiliary System Sample Cooling Coils (9)	20	20	20	-	-
Letdown HX (1)**	1,000 (max)	70	-	1,000 (max)	-
CRDM Seal Coolers (45)	68	68	68	68	-
Charging Pumps (3)	32	22-32	22-32	32	32
Primary Coolant Pumps (4)	360	180	180	360	-
LPSI Pumps (2) #	8	8	8	8	8
HPSI Pumps (2) #	29	29	29	29	29
Containment Spray Pumps (3) #	24	24	24	24	24
Spent Fuel Pool HX (1)	500	500	500***	-	-
Shield Cooling HX (1)	126	126	126	126	-
Waste Gas Compressors (3)	6	6	6	-	-
Vacuum Degasifier Pump Seal Water Cooler (1)	8	8	8	-	-
Radwaste Evaporators (2)	2,136	1,068	1,068	-	-
Total	4,256	7,076	7,006	1,647	5,058/5,578

* With containment high pressure, loss of EDG 1-1/EDG 1-2.
** Flow set by temperature control
*** Increases to 935 gpm at 7 days after shutdown for 1/3 core off load.
\# The required flow rates are based on operation of the pumps under the worst case conditions. Actual flow requirements to prevent component degradation will vary upon operating conditions.

SPENT FUEL POOL COOLING SYSTEM
DESIGN RATINGS AND CONSTRUCTION OF COMPONENTS

1. Fuel Pool Cooling Pumps

Type	Horizontal Centrifugal With Mechanical Seals
Number	2
Capacity (Each)	1,700 gpm
TDH	64 ft
Temperature Transient from 60°F to 212°F in 5 Seconds	Designed To Withstand an Increase
Material	Stainless Steel
Motor	40 hp, 460 V, 60 Hz 3 Ph,
Code	Motor, NEMA; Pump, Standards of Hydraulic Institute

2. Spent Fuel Pool Cooling Heat Exchange Unit

Type	Horizontal Counterflow, With Straight Tubes Rolled Into Tube Sheets
Number	2 Shells in Series
Original Duty (Total)	23×10^6 Btu/h
Original Heat Transfer Area	4,080 ft^2
Component Cooling Water Temperature: In/Out	90/115°F
Spent Fuel Cooling Water Temperature: In/Out	125/110°F

Temperature Transient	Designed To Withstand an Increase From 60°F to 212°F in 5 Seconds
Material	
Shells	Carbon Steel
Tubes	Stainless Steel
Tube Sheets	Stainless Steel With SS 308L Weld Overlay
Codes	ASME B&PV Code, Section III, Class C and ASME B&PV Code, Section VIII, Par UW2 (a); TEMA Class C

3. Fuel Pool Recirculation Booster Pump

Type	Horizontal Centrifugal With Mechanical Seals
Number	1
Capacity	160 gpm
TDH	160 ft
Temperature Transient From	Designed To Withstand Increase 60°F to 212°F in 5 Seconds
Material	Type 316 Stainless Steel
Motor	15 hp, 3 Ph, 60 Hz 460 V
Code	Motor, NEMA; Pump, Standards of Hydraulic Institute

4. Fuel Pool Filter

Type	Cartridge With Replaceable Filter Element
Number	1
Design Flow	150 gpm
Design Pressure	200 psig
Design Temperature	220°F
Temperature Transient	Designed for Increase From 60°F to 212°F in 5 Seconds
Filter Rating	25 Microns Nominal
Material	Stainless Steel
Codes	ASME B&PV Code, Section III, Class C and ASME B&PV Code, Section VIII, Par UW2 (a)

5. Fuel Pool Demineralizer

Type	Mixed Bed
Number	1
Design Flow	150 gpm
Design Pressure	200 psig
Design Temperature	220°F
Temperature Transient	Design for an Increase From 60°F to 212°F in 5 Seconds
Resin	Equivalent Capacity Mixture of Nuclear Grade Cation and Anion

Material	Type 304 Stainless Steel for Vessel and Integral Parts
Codes	ASME B&PV Code, Section III, Class C and ASME B&PV Code, Section VIII, Par UW2 (a)

6. Piping, Fittings and Valves

Material	Stainless Steel
Design Pressure	125 psig
Design Temperature	150°F
Joints, 2 in and Larger	ButtWelded Except at Flanged Equipment
1-1/2 in and Smaller	Socket Welded Except at Flanged Equipment
Valves, 2 in and Larger	Stainless Steel, Butt Weld Ends, 150#
11/2 in and Smaller	Stainless Steel, Socket Weld Ends, 150#
Butterflies, All Sizes	Stainless Steel Flanged, 150#
Code	ASA B31.1
Welds	100% Radiographically Checked

TABLE 9-9
(Sheet 1 of 5)

INSTRUMENT AIR SYSTEM
DESIGN RATINGS AND CONSTRUCTION OF COMPONENTS

1. Compressed Air System

a. Air Compressors

1.	Type	Vertical, Nonlubricated, Reciprocated, Reciprocating, Double Acting, Water Cooled
	Number	2
	Design Capacity (Each)	200 scfm
	Design Pressure	100 psig
	Motor	60 hp, 3 Ph, 60 Hz, 460 V
	Code	Motor, NEMA
2.	Type	Rotary Tooth, Lubricated, Air Cooled
	Number	1
	Design Capacity	320 SCFM
	Design Pressure	100 psig
	Motor	75 hp, 3 Ph, 60 Hz, 3570 rpm
	Code	Motor, NEMA

b.	Aftercoolers	
1.	Type	Shell and Tube
	Number	2 (1 per Reciprocating Compressor)
2.	Type	Cooling Coil
	Number	1 (Rotary Tooth Compressor)
c.	Air Receivers	
	Type	Vertical
	Number	3
	Design Pressure	125 psig
	Capacity	57 ft^3
	Code	ASME B&PV Code, Section VIII
d.	Air Dryer	
	Type	Silica Gel Absorbent, Electric Heater Reactivated
	Number	1
	Capacity	205 scfm
	Outlet Moisture Content With Saturated Air Inlet	-40°F Dew Point at 100 psig
	Code	ASME B&PV Code, Section VIII

e. Piping and Valves	
Upstream of Dryer	Carbon Steel Piping and CI, or Bronze Valves
Downstream of Dryer	Copper Piping and Bronze or Stainless Steel Valves Except at Containment Penetration and at Isolation Valves (Carbon Steel)
Code	ASA B31.1
2. High-Pressure Air System	
a. Air Compressors	
Type	Single Acting, Air Cooled
Number	3
Design Capacity (Each)	22.3 scfm
Design Pressure	325 psig
Motor	10 hp, 440 V, 3 Phase
Code	Motor, NEMA
b. Air Dryer	
Type	Refrigeration
Number	3 (1 per Compressor)
Capacity (Each)	25.5 scfm
Moisture Content	-40 °F
Code	ASME B&PV Code, Section VIII

c.	Air Receivers	
	Type	Horizontal
	Number	3 (1 per Compressor)
	Design Pressure	350 psig
	Capacity	57.7 ft^3
	Code	ASME B&PV Code, Section VIII
d.	Aftercoolers	
	Number	3 (1 per Compressor)
	Type	Air Cooled
e.	Piping	
	Material	Carbon Steel
	Code	ASA B31.1 (Seismic Class I Supported From Receivers to Operators on Engineered Safe-guards Systems)

3. Condensate Demineralizer Building Air System

a.	Air Compressors	
	Type lubricated	Two stage reciprocating, oil
	Number	2
	Design Capacity (Each)	876 scfm
	Design Pressure	125 psi
	Motor	200 hp, 460 V, 3 Phase

TABLE 9-9
(Sheet 5 of 5)

b.	Air Dryer	
	Type	Pressure Swing
	Number	2
	Capacity (Each)	120 scfm
	Pressure	150 psig
	Dew Point	-40°F Dew Point at 125 psig
c.	Aftercooler	
	Number	2
	Type	Water Cooled
d.	Receiver	
	Type	Horizontal
	Number	2
	Design Pressure	150 psig
	Design Temp	650°F
	Capacity	50 ft^3
	Code	ASME B&PV Code, Section VIII

4. Nitrogen Backup Stations

a. Nitrogen Bottles

Pressure	2400 psig
Number	25

TABLE 9-10
(Sheet 1 of 5)

EFFECT OF LOSS OF AIR TO AIR-OPERATED VALVES

O Open
C Closed

Valve No	Valve Description	Normal Position	Shutdown Position	Position After Loss of Air
Primary Coolant System				
0148	Quench Tank Drain	C	C	C
0150	Quench Tank N_2 Supply	C	C	C
0152	Quench Tank Vent	C	C	C
0155	Quench Tank Spray	C	C	C
1057	Pressurizer Spray Valve	C	C	C
1059	Pressurizer Spray Valve	C	C	C
Chemical and Volume Control System				
2001	Letdown Stop	O	O	O
2002	Letdown Orifice Bypass Stop	C	C	C
2003	Letdown Orifice Stop	O	C	C
2004	Letdown Orifice Stop	C	C	C
2005	Letdown Orifice Stop	C	C	C
2009	Letdown Containment Isolation	O	C	C
2012	Letdown Pressure Control	O	O	C
2014	Radiation Monitor Stop	O	O	C
2023	Ion Exchanger Bypass	To Ion Ex	To Ion Ex	To Bypass
2056	Letdown to VCT or RWTS	To VCT	To VCT	To VCT
2080	VCT Vent	C	C	C
2083	PCP Bleedoff Cont Isolation	O	C	C
2111	Charging Line Stop	O	O	O
2113	Loop 1A Charging Line Stop	O	O	O
2115	Loop 2A Charging Line Stop	O	O	O
2117	Pressurizer Auxiliary Spray	C	C	C
2130	Boric Acid Recirc Control	O	C	C
2136	Boric Acid Recirc Control	O	C	C
2153	Boric Acid Makeup Stop	C	C	C
2155	Makeup Stop	C	C	C
2165	Primary Water Makeup Stop	C	C	C
2191	PCP Bleedoff Relief Stop	O	O	As Is (Accumulator)
2202	Letdown Orifice Bypass Control	C	C	C

TABLE 9-10
(Sheet 2 of 5)

Valve No	Valve Description	Normal Position	Shutdown Position	Position After Loss of Air
Safety Injection, Containment Spray and Shutdown Cooling				
3001(b)	Containment Spray Isolation	C	C	O
3002(b)	Containment Spray Isolation	C	C	O
3003	SI Tank Fill and Drain	C	C	C
3004	SI Tank Fill and Drain	C	C	C
3006	Shutdown HX Bypass	O	O	O
3018(a)(f)	HPSI Pump Dischg (Redundant)	C	C	C
3025	Shutdown HX Discharge	C	C	C
3027(a)(b)	Pump MiniFlow Stop	O	O	C
3029(c)	Containment Sump Suction	C	C	As Is
3030(c)	Containment Sump Suction	C	C	As Is
3031(c)	SIRW Tank Isolation	O	O	O
3036	HPSI Dischg (Redundant)	O	O	O
3037(a)	HPSI Dischg (Normal)	C	C	C
3038	SI Line Pressure Control	O	C	C
3039	SI Tank Fill and Drain	C	C	C
3040	SI Tank N_2 Supply	C	C	C
3042	SI Line Pressure Control	O	C	C
3043	SI Tank Fill and Drain	C	C	C
3044	SI Tank N_2 Supply	C	C	C
3046	SI Line Pressure Control	O	C	C
3047	SI Line Pressure Control	O	C	C
3048	SI Tank N_2 Supply	C	C	C
3050	SI Tank N_2 Supply	C	C	C
3051	SI Tank Purge	C	C	C
3055(a)	LPSI Pump Dischg Crossover	C	C	C
3056(a)(b)	Pump MiniFlow Stop	O	O	C
3057(a)	SIRW Tank Isolation	O	O	O
3059(a)	HPSI Dischg (Normal)	O	O	O
3063	SI Tank Purge	C	C	C
3065	SI Tank Purge	C	C	C
3067	SI Tank Purge	C	C	C
3069	Check Valve Leakage Drain	O	O	O
3070(b)	Cooled Suction HPSI Pump	C	C	C
3071(b)	Cooled Suction HPSI Pump	C	C	C
3073	SIRW Tank Temp Cont	O or C	O or C	C
3212(c)	Shutdown HX Isolation	O	O	O
3213(c)	Shutdown HX Isolation	O	O	O
3223(c)	Shutdown HX Isolation	O	O	O
3224(c)	Shutdown HX Isolation	O	O	O

TABLE 9-10
(Sheet 3 of 5)

Valve No	Valve Description	Normal Position	Shutdown Position	Position After Loss of Air
Feed and Condensate System				
0727(b)	Auxiliary Feed Control	C	O	O
0730A	Feedwater System	C		
0736	Auxiliary Feed Control Bypass	C	C	C
0736A	Auxiliary Feed Control	C	O	O
0737	Auxiliary Feed Control Bypass	C	C	C
0737A	Auxiliary Feed Control	C	O	O
0749(b)	Auxiliary Feed Control	C	O	O
0779(e)	Atmospheric Steam Dump	C	C	C
0780(e)	Atmospheric Steam Dump	C	C	C
0781(e)	Atmospheric Steam Dump	C	C	C
0782(e)	Atmospheric Steam Dump	C	C	C
Service Water System				
0823	Component Cool HX Dischg	O	O	O
0824(b)	Return From Containment Coolers	O	O	O
0825	Eng Safe Room Cooler Supply	C	C	O
0826	Component Cool HX Dischg	O	O	O
0835	Turbine LO Cooler Stop Bypass	C	C	O
0836	Turbine LO Cooler Stop Bypass	C	C	O
0838	Normal Cont Cooler Control	O	O	C
0839	Generator H_2 Cooler Stop Bypass	C	C	O
0843	Normal Cont Cooler Control	O	O	C
0844	Critical Service Wtr Header Iso	O	O	O
0845	Critical Service Wtr Header Iso	O	O	O
0846	Critical Service Water Header CrossConnect	O	O	O
0847(b)	Supply to Containment Coolers	O	O	O
0852	Generator Exciter Cooler Supply Bypass	C	C	O
0857	Critical Service Water Header CrossConnect	O	O	O
0861	8" Return From Cont Coolers	C	C	O
0862	Containment Cooler Supply	O	O	O
0863	Normal Cont Cooler Control	O	O	C
0864	8" Return From Cont Coolers	C	C	O
0865	Containment Cooler Supply	O	O	O
0867	8" Return From Cont Coolers	C	C	O
0869	Containment Cooler Supply	O	O	C
0870	Containment Cooler Supply	O	O	O
0872	Normal Cont Cooler Control	O	O	C
0873	8" Return From Cont Coolers	C	C	O
0876	Diesel Generator Cool Supply	O	O	O
0877	Diesel Generator Cool Supply	O	O	O

TABLE 9-10
(Sheet 4 of 5)

Valve No	Valve Description	Normal Position	Shutdown Position	Position After Loss of Air
0879	Backup Cool Safeguards Pumps	C	C	C
0880	Backup Cool Safeguards Pumps	C	C	C
0884	Diesel Generator Cool Supply	C	C	O
0885	Diesel Generator Cool Supply	C	C	O
1318	Service Water Pump Header Iso	O	O	O
1319	Service Water Pump Header Iso	O	O	O
1359	Noncritical Service Water Header Isolation	O	O	C
Component Cooling System				
0909	Letdown HX Return	O	O	O
0910	Component Cool to Cont Isolation	O	O	O
0911	Component Cool From Cont Isola tion	O	O	As Is(Ac cumulator)
0913	Supply Safeguards Pumps	O	O	O
0915	Comp Cool Surge Tank Vent	O	O	C
0918	Comp Cool Surge Tank Makeup	C	C	C
0937	Supply to Shutdown HX	C	O	O
0938	Supply to Shutdown HX	C	O	O
0940	Component Cool From Cont Isolation	O	O	As Is (Ac cumulator)
0944	Supply to Radwaste Evaporator	O	O	C
0944A	Supply to Spent Fuel HX	O	O	C
0945	Supply to Comp Cool HX	O	O	O
0946	Supply to Comp Cool HX	O	O	O
0947	Supply to Safeguards Pumps	O	O	O
0948	Supply to Safeguards Pumps	O	O	O
0949	Supply to Safeguards Pumps	O	O	O
0950	Return From Safeguards Pumps	O	O	O
0951	Return From Safeguards Pumps	C	C	C
0977B	Return From Radwaste Evaporator	O	O	C

TABLE 9-10
(Sheet 5 of 5)

Valve No	Valve Description	Normal Position	Shutdown Position	Position After Loss of Air
Main Steam, Main and Auxiliary Turbine Systems				
0501	Main Steam Isolation Valve	O	O	As Is (Accumulator)
0510	Main Steam Isolation Valve	O	O	As Is (Accumulator)
0511	Steam Bypass Valve	C	Open for Bleed	C
0522A	Steam to Aux Turbine Feed Pump	C	O	C
0522B(b)	Auxiliary Feed Steam	C	O	C

Service and Instrument Air Systems

Valve 1211(b), Instrument Air Supply to Containment, is open during reactor operation or reactor shutdown and fails open on loss of air.

Process Sampling System

Air-operated process sampling valves are normally closed unless sampling a specific point. All air-operated valves fail closed.

Radioactive Waste Treatment System

All air-operated valves in the radioactive waste treatment system, including liquid and gas discharge stop valves, fail closed upon loss of instrument air.

Heating, Ventilation and Air Conditioning

Reference Subsection 9.8.4.

Shield Cooling System

During normal reactor operation and reactor shutdown, one of two air-operated shield cooling supply valves is open. Upon loss of instrument air, both supply valves fail open.

(a)Air supplied by high-pressure air system.
(b)Nitrogen bottle backup.
(c)Air supplied from high-pressure air system with backup from instrument air.
(d)Air supplied from turbine room high-pressure air system.
(e)Bulk nitrogen backup.
(f)Manually operated air bottle backup

TABLE 9-11

FIRE DETECTION INSTRUMENTATION
(PAGE 1 OF 4)

	INTRUMENT LOCATION	DETECTORS	TYPE OF DETECTORS	MINIMUM INSTRUMENTS OPERABLE
1.	Cable Spreading Rm, Col M-28	1	Waterflow	1
2.	1-D Switchgear Rm , Col G-28; Col G-22; Col G- 22	3	Water Flow Sw	3
3.	1-1 Diesel Generator Rm , Col J-28	1	Water Flow Sw	1
4.	1-2 Diesel Generator Rm , Col M-28	1	Water Flow Sw	1
5.	Turbine Bldg 590" Col H-9	1	Water Flow Sw	1
6.	Control Room and Room 325	7	Smoke	5
7.	Control Room Adj Offices Rms 324 & 320	2	Smoke	1
8.	Cable Spreading, Room 224	8	Smoke	6
9.	Refueling & Spent Fuel Area, Rm 220	4	Smoke	2
10.	1-D Switchgear Rm , Rm 223	4	Smoke	3
11.	North Penetration, Rm 332	2	Smoke	1

TABLE 9-11

FIRE DETECTION INSTRUMENTATION

	iNTRUMENT LOCATION	DETECTORS	TYPE OF DETECTORS	MINIMUM INSTRUMENTS OPERABLE
12.	1-C Switchgear Rm , Rm 116A	2	Smoke	1
13.	Southwest Cable Penetration, Rm 250	2	Smoke	1
14.	Engineered Safeguards Panel Area, Rm 121	3	Smoke	2
15.	Stairwell Outside Engineered Safeguards Panel Area, Rm 016	1	Smoke	1
16.	Component Cooling Pump, Rm 123	2	Smoke	1
17.	Safeguard Area, Rm 4	3	Smoke	2
18.	Safeguard Area, Rm 5	2	Smoke	1
19.	Corridor 106 on 590' Elevation, Rm 106	6	Smoke	4
20.	Charging Pump, Rm 104	2	Smoke	1
21.	Containment, Interior North Penetration Area, Rm 332	3	Smoke	2

TABLE 9-11

FIRE DETECTION INSTRUMENTATION
(PAGE 3 OF 4)

	INTRUMENT LOCATION	DETECTORS	TYPE OF DETECTORS	MINIMUM INSTRUMENTS OPERABLE
22.	Containment, Interior SW Penetration Area, Rm 141, 250	3	Smoke	2
23.	Containment Instrument Air Room	3	Smoke	2
24.	Auxiliary Feed Pump Room 570' Level of Turbine Bldg, Rm 007	1	Smoke	1
25.	Battery Rm 225A	1	Smoke	1
26.	Battery Rm 225B	1	Smoke	1
27.	HVAC Equipment Rooms & Chase:		Water Flow Sw	1
	West Mechanical Equipment Room300	1	Smoke	1
	East Mechanical Equipment Room 300A	1	Smoke	1
	Duct Chase, Rm 300B	1	Smoke	1
28.	Air Handling Unit V-95 & V-96 Inlet Ducts, Rm 300, 300A	2	Smoke	2
29.	Electrical Equipment Room, Rm 725	4	Smoke	3
30.	Technical Support Center, Rm 320A	2	Smoke	1

TABLE 9-11

FIRE DETECTION INSTRUMENTATION
(PAGE 4 OF 4)

	iNTRUMENT LOCATION	DETECTORS	TYPE OF DETECTORS	MINIMUM INSTRUMENTS OPERABLE
31.	Intake Structure, Room 136	11	Ultraviolet	7
32.	North Cable Penetration Room	1	Water Flow Sw	1
33.	Electrical Equipment Room, Rm 725	1	Water Flow Sw	1
34.	Charging Pump Rooms 104, 104A, and 104B	1	Water Flow Sw	1

TABLE 9-12
(Sheet 1 of 3)

FIRE PROTECTION SYSTEM
DESIGN RATINGS AND CONSTRUCTION OF COMPONENTS

1. Fire Pump, Motor Driven

Type	Vertical Turbine
Number	1
Capacity	1,500 gpm
Discharge Pressure	125 psig
Material	
Discharge Head	Cast Iron
Impeller	Bronze
Motor	150 hp, 460 V, 3 Ph, 60 Hz
Codes	Underwriters Lab Label Motor, NEMA; Pump, Standards of Hydraulic Institute

2. Fire Pump, Diesel Engine Driven

Type	Vertical Turbine
Number	2
Capacity	1,500 gpm
Discharge Pressure	125 psig
Material	
Discharge Head	Cast Iron
Impeller	Bronze
Gear Drive	Reduction Ratio 1:1, 200 hp Rating

Diesel Engine	150 hp
Codes	Underwriters Lab Label Diesel Engine, NEMA; Pump, Standards of Hydraulic Institute

3. Fire System Jockey Pump

Type	Vertical Turbine
Number	1
Capacity	50 gpm
Discharge Pressure	110 psig
Material	
Discharge Head	Cast Iron
Impeller	Bronze
Motor	7-1/2 hp, 460 V, 3 Ph, 60 Hz
Codes	Motor, NEMA; Pump, Standards of Hydraulic Institute

4. Piping, Fittings and Valves

   a. To Auxiliary Feedwater Pump Suction Header and Critical Service Waterlines

Material	Seamless Carbon Steel
Design Pressure	125 psig
Design Temperature	100°F
Construction	Butt-Welded Except at Flanged Equipment
Valves	Carbon Steel, Butt-Weld Ends, 150#, or Cast Iron, Flanged End, 175#, Underwriters Lab Label

TABLE 9-12
(Sheet 3 of 3)

b. To Spent Fuel Pool Blind Flange and Normal Fire Protection Service

	Underground	Aboveground
Material	Cast Iron	Carbon Steel
Design Pressure	150 psig	125 psig
Design Temperature	100°F	100°F
Construction	Mechanical Joint	Butt-Welded Except at Flanged Equipment
Valves	Cast Iron, Mechanical Joint, 175#, Underwriters Lab Label	Cast Iron, Flanged End, 175#, Underwriters Lab Label

TABLE 9-13
(Sheet 1 of 3)

AUXILIARY FEEDWATER SYSTEM
DESIGN RATINGS AND CONSTRUCTION OF COMPONENTS

1. Motor-Driven Auxiliary Feedwater Pump (P-8A)

Type	Horizontal Centrifugal, With Packed Glands
Number	1
Capacity	415 gpm
Head	2,730 ft
Material	
Case	4.6% Chrome Alloy Steel
Impeller	Bronze
Shaft	11%-13% Chrome Alloy Steel
Motor	450 hp, 3 Ph, 60 Hz, 2,300 V
Codes	Motor, NEMA; Pump, Standards of Hydraulic Institute, 11th Edition, 1965

2. Turbine-Driven Auxiliary Feedwater Pump (P-8B)

Type	Horizontal Centrifugal, With Packed Glands
Number	1
Capacity	415 gpm
Head	2,730 ft
Material	
Case	4.6% Chrome Alloy Steel
Impeller	Bronze
Shaft	11%-13% Chrome Alloy Steel

TABLE 9-13
(Sheet 2 of 3)

Turbine	Single Stage, Axial Flow, Exhaust to Atmosphere, 450 hp
Codes	Turbine, NEMA; Pump, Standards of Hydraulic Institute, 11th Edition, 1965

3. Motor Driven Auxiliary Feedwater Pump (P-8C)

Type	Horizontal Centrifugal, With Mechanical Seals
Number	1
Capacity	330 gpm
Head	2260 ft
Material	
Case	18 Cr, 8 Ni Stainless Steel
Impeller	12 Cr Stainless Steel
Shaft	12 Cr, .6 Mo Stainless Steel
Motor	400 hp, 3 Ph, 60 Hz, 2,300 V
Codes	Motor, NEMA; Pump, Standards of Hydraulic Institute, 11th Edition, 1965

4. Piping and Valves

a. Pump Suction

	Underground	Aboveground
Material	304 Stainless Steel	Carbon Steel
Design Pressure (Minimum)	50 psig	50 psig
Design Temperature (Minimum)	100°F	100°F
Construction	Welded Except at Flanged Equipment Connections	
Valves 2-1/2 in and Larger	-	Carbon Steel, Butt-Welded, 150#

Valves 2-1/2 in and Smaller	-	Carbon Steel, Socket Welded, 600#	
Code	ASA B31.1-1955 ASA B16.5-1961	ASA B31.1-1955 ASA B16.5-1961	

b. Pump Discharge

	Underground	Aboveground	
Material	304 Stainless Steel	Carbon Steel	
		Upstream of FW Control Valve	Downstream of FW Control Valve
Design Pressure (Minimum)	1,440 psig	1,337 psig	1,100 psig
Design Temperature (Minimum)	100°F	100°F	100°F
Construction	Welded Except at Flanged Equipment Only		
Valves 2-1/2 in and Larger	-	Carbon Steel, Butt-Welded, 600#	
Valves 2-1/2 in and Smaller	-	Carbon Steel, Socket Welded, 600#	
Code	ASA B31.1-1955 ASA B16.5-1961	ASA B31.1-1955 ASA B16.5-1961	

c. Auxiliary Turbine Steam Supply

Design Pressure	1,000 psig
Design Temperature	550°F

Piping and Valves Same as for Aboveground Pump Discharge Piping

TABLE 9-14

Deleted

TABLE 9-15

DESIGN BASIS AMBIENT CONDITIONS
(PAGE 1 OF 2)

	Winter, °F		Summer, °F		Equipment Qualification (°F)	
	Outside	Inside	Outside	Inside	Normal	Shutdown
Turbine Building						
Oper Floor	-10	50	95	104	---	---
Turbine Building						
Piping Area	-10	50	95	110	---	---
Turbine Building						
Shops and Offices	-10	65	95	104	---	---
Containment Building (See Note 1)	-10	50	95	104	Note 2	80
Auxiliary Building						
Penetration and Fan Rooms	-10	50	95	104	---	---
R123 (590' Component Cooling)					100	100
R238 (607'-6" Containment Purge Exhaust)					110	110
R338 (625' Containment Purge Air Fan)					120	120
R001 (East Eng Safeguards East End)					80	80
R004 (Eng Safeguards)					72	90
R005 (Eng Safeguards)					72	90
R118 (Receiver Tank and Circ Pump)					80	80
R120 (Vac Degasifier)					80	80
R121A (590' Passageway)					90	104
R121B (590' Primary Coolant Sample Heat Exchanger)					80	80
R150 (602' Pipeway)					92	92

TABLE 9-15

DESIGN BASIS AMBIENT CONDITIONS
(PAGE 2 OF 2)

	Winter, °F		Summer, °F		Equipment Qualification (°F)	
	Outside	Inside	Outside	Inside	Normal	Shutdown
Auxiliary Building						
Radwaste Area and Radwaste Area Addition	-10	50	95	110	---	---
Fuel Handling Area and Fuel Handling Area Addition	-10	50	95	104	---	---
Office Area	-10	75	95	75	---	---
Control Room	-10	75, 50% RH	95	75, 50% RH	---	---
Condensate and Makeup Demineralizer Building						
Process and Equipment Area	-10	50	90	104	---	---
Covered Receiving and Loading Area	-10	50	90	104	---	---
Boiler Room	-10	50	90	104	---	---
Pipe Gallery	-10	50	90	104	---	---
Instrument Room	-10	75	90	75	---	---

Note 1: Original equipment design was based on these conditions. To allow for elevated service water temperatures a higher building design temperature was specified for the containment air coolers. Chapter 14 contains containment temperature assumptions for analyzed accident situations.

Note 2: Dependent on elevation and location (EA-BDM-88-02[1], EA-RGB-92-05[2], EA-RGB-94-02)
[1]EQ File E-48, Sheet 6EW
[2]EQ File E-48, Sheet 3M

Note 3: The Penetration and Fan Room (Rooms 123, 238, and 338) have a design temperature of 120°F per M-391, Specification for Installation of Ventilation Equipment and Ductwork Penetration and Fan Rooms. Actual maximum temperatures for these rooms have been measured for EEQ and are 100°F for room 123, 110°F for room 238 and 120°F for room 338.

TABLE 9-16
(Sheet 1 of 3)

CONTROL ROOM HVAC SYSTEM
MAJOR COMPONENT DESIGN DATA

Makeup/Recirculation Air Filter Units	
Quantity	Two - 100% Capacity Each
Capacity	3,200 ft^3/min
Filters (per Filtering Unit)	
Prefilter	
Quantity	3
Media	Glass Fiber or Knitted Pad
ΔP of wg (Clean)	0.25 in
HEPA Filter	
Quantity	6
Media	Glass Fiber
ΔP of wg (Clean)	1.25 in (Maximum)
Charcoal Filter Trays	
Quantity	18
Media	Activated Carbon, 4 in. Bed Depth (Two-2 inch deep Trays in Series)
ΔP of wg (Clean)	1.2 in (Maximum)
Fan Type	Vaneaxial
Fan Static, Pressure at Rating, in wg	10
Motor	20 hp, 460 V, 3 Ph
Filter Test Efficiency	
HEPA	99.97% of Particulate

NOTE: All electrical equipment is Class 1E unless otherwise noted.

TABLE 9-16
(Sheet 2 of 3)

Carbon Adsorber	99.9% of Elemental Iodine
Electric Heating Coil	
Type	Nickel/Chromium
Capacity	15 kW, 480 V, 3 Ph
Air Handling Unit V-95 or V-96	
Type	Package (Filter, Cooling Coil, Fan)
Capacity	12,500 ft^3/min
Cooling Coil	
Type	Direct Expansion Refrigerant
Capacity (Total)	603,500 Btu/h
Heating Coil (Nonclass 1E)	
Type	80% Nickel and 20% Chromium
Capacity	177 kW, 480 V, 3 Ph
Fan	
Type	Centrifugal
Total Pressure, wg	3.85 in
Motor	25 hp, 460 V, 3 Ph
Filter Type, Media	6 in Thick Moderate Efficiency Prefilter
ΔP of wg (Clean)	0.25 in
Refrigerant Condensing Unit	
Type	Water-Cooled Reciprocating
Refrigerant	R-22
Compressor	
Type	Reciprocating, 4 Cylinder
Motor	60 hp, 460 V, 3 Ph

NOTE: All electrical equipment is Class 1E unless otherwise noted.

Capacity	554,400 Btu/h @ 85 °F, 39 gpm Service Water
Condenser Water Flow	Set per T-216
Smoke Purge Exhaust Fan V-94 (Nonclass 1E)	
Type	Vaneaxial
Capacity	7,800 ft^3/min
Motor	7-1/2 hp, 460 V, 3 Ph
Exhaust Fan V-16 (Existing) (Nonclass 1E)	
Type	Centrifugal
Capacity	160 ft^3 /min
Motor	1/12 hp, 120 V, 1 Ph
Humidifiers VH-12 and VH-13 (Nonclass 1E)	
Type	Steam Generator
Capacity	50 lb/h (17 kW, 480 V, 3 Ph)

NOTE: All electrical equipment is Class 1E unless otherwise noted.

TABLE 9-17
(Sheet 1 of 6)

VENTILATION DAMPERS: FUNCTIONS AND POSITIONS
FOR VARIOUS MODES OF PLANT OPERATION

Damper	Description	Normal Position	Shutdown Position	Position After Auto Actuation	Position After Loss of Air
Control Room (see Figure 7-24)					
D-1	Normal Outside Air - Train A	Open - Train A Close - Train B	Open - Train A Close - Train B	Close on CHP or CHR	Close
D-2	Normal Outside Air - Train A	Modulate - Train A Close - Train B	Modulate - Train A Close - Train B	Close on CHP or CHR	Close
D-3	Normal Recirc Air - Train A	Open	Open	Open on CHP or CHR	Open
D-4	Supply Air Back Draft Dampers - Train A	Open - Train A Close - Train B	Open - Train A Close - Train B	Open - Train A Close - Train B on CHP or CHR	NA (Back Draft)
D-5	Charcoal Filter Unit Supply Air - Train A	Close	Close	Open - Train A Close - Train B on CHP or CHR	NA (Back Draft)
D-6	Charcoal Filter Unit Return Air - Train A	Close	Close	Open - Train A Close - Train B on CHP or CHR	Open
D-7	Charcoal Filter Unit Outside Air - Train A	Close	Close	Open - Train A, (Manual Close Avail) Close - Train B on CHP or CHR	NA (Elect Op FAI)

TABLE 9-17
(Sheet 2 of 6)

Damper	Description	Normal Position	Shutdown Position	Position After Auto Actuation	Position After Loss of Air
D-8	Normal Outside Air - Train B	Open - Train B Close - Train A	Open - Train B Close - Train A	Close on CHP or CHR	Close
D-9	Normal Outside Air - Train B	Modulate - Train B Close - Train A	Modulate - Train B Close - Train A	Close on CHP or CHR	Close
D-10	Normal Recirc Air - Train B	Open	Open	Open on CHP or CHR	Open
D-11	Supply Air Back Draft Damper - Train B	Open - Train B Close - Train A	Open - Train B Close - Train A	Open - Train B Close - Train A on CHP or CHR	NA (Back Draft)
D-12	Charcoal Filter Unit Supply Air - Train B	Close	Close	Open - Train B Close - Train A on CHP or CHR	NA (Back Draft)
D-13	Charcoal Filter Unit Return Air - Train B	Close	Close	Open - Train B Close - Train A on CHP or CHR	Close
D-14	Charcoal Filter Unit Outside Air - Train B	Close	Close	Open - Train B, (Manual Close Avail) Close - Train A on CHP or CHR	NA (Elect Op FAI)
D-15	Purge Fan Isolation	Close	Close	Close on CHP or CHR	Close

TABLE 9-17
(Sheet 3 of 6)

Damper	Description	Normal Position	Shutdown Position	Position After Auto Actuation	Position After Loss of Air
D-16	Purge Fan Isolation	Close	Close	Close on CHP or CHR	Close
D-17	Exhaust Fan V-16 Isolation	Open	Open	Close on CHP or CHR	Close
D-18	Exhaust Fan Isolation	Open	Open	Close on CHP or CHR	Close
D-19	Number Not Used	-	-	-	-
D-20	Charcoal Filter Flow Control - Train A	Open	Open	Modulate - Train A Open - Train B on CHP or CHR	Open
D-21	Charcoal Filter Flow Control - Train B	Open	Open	Modulate - Train B Open - Train A on CHP or CHR	Open
Radioactive Waste Area and Engineered Safeguards Rooms					
PO-3010	Fresh Air Supply	Open	Open	Close on Trip of Fan V-10	Close
PO-1809	Radwaste Area Supply	Open	Open	Close (RE-1809)	Close
PO-1839	Radwaste Area Exhaust	Open	Open	Close on Trip of Fan V-14A	Close
PO-1840	Radwaste Area Exhaust	Open	Open	Close on Trip of Fan V-14B	Close

TABLE 9-17
(Sheet 4 of 6)

Damper	Description	Normal Position	Shutdown Position	Position After Auto Actuation	Position After Loss of Air
PO-1817	East Safeguards Room Supply	Open	Open	Close (RE-1810)	Close
PO-1810	East Safeguards Room Exhaust	Open	Open	Close (RE-1810)	Close
PO-1812	West Safeguards Room Supply	Open	Open	Close (RE-1811)	Close
PO-1811	West Safeguards Room Exhaust	Open	Open	Close (RE-1811)	Close
Auxiliary Building Addition Radwaste Area Ventilation System					
PO-8006	Fresh Air Supply	Open	Open	Close on Trip of Fan V-67	Close
PO-8016A	Radwaste Add Exhaust	Open	Open	Close on Trip of Fan V-68A	Close
PO-8016B	Radwaste Add Exhaust	Open	Open	Close on Trip of Fan V-68B	Close

Fuel Handling Area

Damper PO-3007 is normally open during reactor operation or reactor shutdown and fails closed on loss of instrument air.

TABLE 9-17
(Sheet 5 of 6)

Damper	Description	Normal Position	Shutdown Position	Position After Auto Actuation	Position After Loss of Air
Auxiliary Building Addition Fuel Handling Area Ventilation System					
PO-8001	Fresh Air Supply	Open	Open	Close on Trip of Fan V-69	Close
PO-8013A	Fuel Handling Add Exhaust	Open	Open	Close on Trip of Fan V-70A	Close
PO-8013B	Fuel Handling Add Exhaust	Open	Open	Close on Trip of Fan V-70B	Close
Penetration and Fan Rooms Heating and Ventilation System					
PO-8035	Outside Air Supply	Open	Open	Close (RIA-5710)	Close
PO-8036	Exhaust	Open	Open	Close (RIA-5710)	Close
Containment					
CV-1813	Air Space Purge Supply	Close	Open	Close on CHP or CHR	Close
CV-1814	Air Space Purge Supply	Close	Open	Close on CHP or CHR	Close
CV-1806	Cont Purge Exhaust	Close	Open	Close on CHP or CHR	Close
CV-1805	Cont Purge Exhaust	Close	Open	Close on CHP or CHR	Close

TABLE 9-17
(Sheet 6 of 6)

Damper	Description	Normal Position	Shutdown Position	Position After Auto Actuation	Position After Loss of Air
CV-1808	Cont Purge Exhaust	Close	Open	Close on CHP or CHR	Close
CV-1807	Cont Purge Exhaust	Close	Open	Close on CHP or CHR	Close

TABLE 9-18

SAMPLING STATIONS

NSSS Sampling Station

Post-Accident Sampling System

Containment Hydrogen Monitoring System

Turbine Analyzer Panel

Radwaste Sampling Station

Waste Gas Sample Panel

Radwaste Addition Sampling System

TABLE 9-19
(Sheet 1 of 2)

SAMPLE POINT SUMMARY

1. NSSS Sample Station	
Pressurizer Vapor Phase	Grab Sample, Bomb
Pressurizer Liquid Phase	Grab Sample, Bomb
Primary Coolant Hot Leg	Grab Sample, Bomb
Quench Tank Liquid Phase	Grab Sample, Bomb
Quench Tank Vapor Phase	Grab Sample, Bomb
Purification Ion Exchange Inlet	Grab Sample
Purification Filters Outlet	Grab Sample
LPSI Pumps Discharge	Grab Sample
Purification Ion Exchange Outlet	Grab Sample
SI Drain Tank	Grab Sample
Containment Spray Pumps Discharge	Grab Sample
SIRW Tank Recirculation	Grab Sample
HPSI Pumps Discharge	Grab Sample
2. Radwaste Sampling Station	
Primary System Drain Tank Recirc	Grab Sample
Equipment Drain Tank Recirc	Grab Sample
Vacuum Degasifier Pump Discharge	Grab Sample
Receiver Tank Pumps Discharge	Grab Sample
Receiver Tank Circ Pumps Discharge	Grab Sample
Radwaste Demin Tanks Outlet (3)	Grab Sample
Treated Waste Mon Tanks Recirc (2)	Grab Sample
Controlled Chem Lab Drain Tank	Grab Sample
Filtered Waste Monitor Tank Recirc	Grab Sample
Dirty Waste Drain Tank Recirc	Grab Sample
Component Cooling Pumps Discharge	Grab Sample
3. Turbine Analyzer Panel	
Steam Generator Blowdown (2)	Grab Sample, Conductivity, pH, Sodium, Hydrazine
Feedwater Heater Train (2)	Grab Sample, Conductivity, pH, Oxygen, Sodium Hydrazine
Condensate Pumps Discharge (2)	Grab Sample, Conductivity, pH, Oxygen, Sodium, Hydrazine
Heater Drains Discharge (2)	Grab Sample, pH, Oxygen
Primary Storage Tank	Grab Sample, Conductivity, pH, Sodium
Condensate Pump P-11 Discharge	Grab Sample, Conductivity, ph, Sodium
Blowdown Demineralizer (3)	Grab Sample

TABLE 9-19
(Sheet 2 of 2)

4. Waste Gas Sample Panel	
Volume Control Tank	Bomb
Waste Gas Surge Tank	Bomb
Waste Gas Decay Tanks (6)	Bomb
Spurt Resin Storage Tank	Bomb
5. Radwaste Addition Sampling System	
Radwaste Polishing Demineralizer Discharge	Grab Sample, Conductivity
Clean Waste Transfer Pump Discharge	Grab Sample
Clean Waste Distillate Pump Discharge	Grab Sample
Misc Waste Distillate Pumps Discharge (2)	Grab Sample
Misc Waste Demineralizer Tank Discharge (2)	Grab Sample, Conductivity (1)
Misc Waste Transfer Pumps Discharge (2)	Grab Sample
Misc Waste Filter Inlet	Grab Sample
Misc Waste Filter Discharge (2)	Grab Sample
Primary System Makeup Water Pump Discharge	Grab Sample
Utility Water Transfer Pump Discharge	Grab Sample
Spurt Resin Storage Tank Gas	Bomb
Waste Gas Decay Tanks (3)	Bomb
Radwaste Evaporator Distillate (2)	Grab Sample
6. Post-Accident Sample & Monitoring System	
Containment Atmosphere	% H_2 and % O_2 via Remote Monitoring
Primary Coolant Hot Leg	pH, Conductivity and Dissolved O_2 via Continuous Remote Monitoring
LPSI Pump Discharge	pH, Conductivity and Dissolved O_2 via Continuous Remote Monitoring
Primary Coolant Off-Gas	% H_2 via Remote Monitoring
Waste Gas Decay Tanks	% H_2 via Remote Monitoring
Volume Control Tank	% H_2 via Remote Monitoring
7. Containment Hydrogen Monitor	
Containment Atmosphere (2)	% Hydrogen

TABLE 9-20
(Sheet 1 of 14)

CHEMICAL AND VOLUME CONTROL SYSTEM DESIGN PARAMETERS

1.1 General

Normal Letdown Flow	40 gpm
Normal Purification Flow Rate	40 gpm
Normal Charging Flow	44 gpm
Primary Coolant Pump Controlled Bleedoff (4 Pumps)	4 gpm
Normal Letdown Temperature at Loop	547.8°F
Normal Charging Temperature at Loop	425°F
Ion Exchanger Operating Temperature	120°F

1.2 Regenerative Heat Exchanger - E56

Quantity	1
Type	Shell and Tube, Vertical
Normal Heat Transfer	6.6 x 10^6 Btu/h
Code	ASME B&PV Code, Section III, Class C, 1965
Shell Side (Charging)	
Fluid	Primary Coolant, 6-1/4 Wt % Boric Acid, Maximum
Design Pressure	2,735 psig
Design Temperature	650°F
Material	Stainless Steel

Tube Side (Letdown)

Fluid	Primary Coolant, 1 Wt % Boric Acid, Maximum
Design Pressure	2,485 psig
Design Temperature	650°F
Material	Stainless Steel

Operating Parameters - Regenerative Heat Exchanger

	Normal	Maximum Unbalanced Charging With Heat Transfer	Maximum Purification	Maximum Unbalanced Letdown
Tube Side (Letdown)				
Flow - gpm	40	40	120	120
Inlet Temp - °F	547.8	547.8	547.8	547.8
Outlet Temp - °F	251	160	319	449
Shell Side (Charging)				
Flow - gpm	43	133	123	33
Inlet Temp - °F	120	120	120	120
Outlet Temp - °F	416	246	367	523
Heat Transfer - Btu/h	6.3×10^6	7.9×10^6	14.9×10^6	6.9×10^6

1.3 Letdown Orifice - RO2003, RO2004 and RO2005

Quantity	3
Capacity (Each)	40 gpm
Design Pressure	2,485 psia
Design Temperature	650°F
Normal Temperature of Fluid	250°F
Maximum Temperature of Fluid	450°F
Normal Downstream Pressure	470 psia
Normal Upstream Pressure	1,970 psia
Material	Stainless Steel
Fluid	Primary Coolant, 1 Wt % Boric Acid, Maximum

1.4 Letdown Heat Exchanger - E58

Quantity	1
Type	Shell and Tube, Horizontal
Design Heat Transfer	19.1×10^6 Btu/h
Code	ASME B&PV Code, Section III, Class C
Tube Side (Letdown)	
Fluid	Primary Coolant, 1 Wt % Boric Acid, Maximum
Design Pressure	650 psig
Design Temperature	550°F
Material	Stainless Steel

TABLE 9-20
(Sheet 4 of 14)

Shell Side (Cooling Water)

Fluid	Component Cooling Water
Design Pressure	150 psig
Design Temperature	250°F
Material	Carbon Steel

Operating Parameters - Letdown Heat Exchanger

	Normal	Maximum Unbalanced Charging With Letdown	Maximum Purification	Maximum Unbalanced Letdown
Tube Side (Letdown)				
Flow - gpm	40	40	120	120
Inlet Temp - °F	251	160	319	449
Outlet Temp - °F	120	120	120	139
Heat Transfer - Btu/h	2.6×10^6	1.1×10^6	11.9×10^6	19.1×10^6
Shell Sides (Cooling Water)				
Flow - gpm	66 - 111	23 - 40	500 - 1,000	591 - 960
Inlet Temp - °F	65 - 90	65 - 90	65 - 90	65 - 90
Outlet Temp - °F	144-137	133-130	113 - 114	130-130

1.5 Process Radiation Monitor - Element RE-0202

Quantity	1
Design Pressure	200 psig
Design Temperature	250°F
Normal Operating Pressure	20 psig
Normal Operating Temperature	120°F
Normal Flow Rate	0.5 gpm
Code	ASA B31.1

1.6 Ion Exchangers - T51A, T51B and T52

Quantity	3
Type	Flushable
Design Pressure	200 psig
Design Temperature	250°F
Normal Operating Pressure	20 psig
Normal Operating Temperature	120°F
Resin Volume	32 ft^3
Normal Flow Rate	40 gpm
Maximum Flow Rate	120 gpm
Decontamination Factor, Minimum	10
Retention Screen	80 US Mesh
Code for Vessel	ASME B&PV Code, Section III, Class C

Material	Stainless Steel
Fluid	1 Wt % Boric Acid, Maximum

1.7 Purification Filters - F54A and F54B

Quantity	2
Type of Elements	Synthetic Fiber
Retention	0.1 to 6.0 Micron Absolute; 1.0 Micron Nominal (or finer)
Design Pressure	200 psig
Design Temperature	250°F
Design Flow	120 gpm
Normal Flow	40 gpm
Maximum Flow	160 gpm
Code for Vessel	ASME B&PV Code, Section III, Class C, 1965
Material	Stainless Steel
Fluid	1 Wt % Boric Acid, Maximum

1.8 Volume Control Tank - T54

Quantity	1
Type	Vertical, Cylindrical
Design Pressure, Internal	75 psig
Design Pressure, External	15 psig
Design Temperature	250°F
Internal Volume, Minimum	4,170 gal

Operating Pressure Range	0 to 75 psig
Normal Operating Pressure	10 psig
Normal Operating Temperature	120°F
Normal Spray Flow	40 gpm
Blanket Gas	Hydrogen or Nitrogen
Code	ASME B&PV Code, Section III, Class C, 1965
Fluid	6-1/4 Wt % Boric Acid, Maximum
Material	Stainless Steel

1.9 Spray Nozzle (Volume Control Tank)

Quantity	1
Type	Medium Angle, Full Cone
Design Pressure	200 psig
Design Temperature	250°F
Normal Spray Flow	40 gpm
Maximum Spray Flow	120 gpm
Fluid	1 Wt % Boric Acid, Maximum
Material	Stainless Steel

1.10 Variable Speed Charging Pump - P55A

Quantity	1
Type	Positive Displacement
Design Pressure	2,735 psig
Design Temperature	250°F
Flow Rate Range	33 to 53 gpm
Normal Flow Rate	44 gpm
Normal Discharge Pressure	2,200 psig
Normal Temperature of Pumped Fluid	120°F
Maximum Discharge Pressure (Short Term)	2,900 psig
NPSH Required	7.65 ft (Ref. 25)
NPSH Available (Normal Suction From VCT)	30.39 ft (Ref. 25)
Maximum Pressure Pump Starts Against	2,485 psig
Driver Rating	100 hp
Type Variable Capacity Device	Fluid Drive
Fluid Drive and Pump Cooling Water Requirements	22 gpm, 15°F Rise
Materials in Contact With Pumped Fluid	Stainless Steel or Equivalent Corrosion Resistance
Fluid	6-1/4 Wt % Boric Acid, Maximum

1.11 Constant Speed Charging Pumps - P55B and P55C

Quantity	2
Type	Positive Displacement
Design Pressure	2,735 psig
Design Temperature	250°F
Flow Rate	40 gpm
Normal Discharge Pressure	2,200 psig
Normal Temperature of Pumped Fluid	120°F
Maximum Discharge Pressure (Short Term)	3,010 psig
NPSH Required	7.41 ft (Ref. 25)
NPSH Available P-55B/C (Normal Suction From VCT)	28.22/28.18 ft (Ref. 25)
Maximum Pressure Pump Starts Against	2,500psig
Driver Rating	75 hp
Pump Cooling Water Requirements	5 gpm, 15°F Rise
Materials in Contact With Pumped Fluid	Stainless Steel or Equivalent Corrosion Resistance
Fluid	6-1/4 Wt % Boric Acid, Maximum

TABLE 9-20
(Sheet 10 of 14)

1.12 Boric Acid Batching Tank - T77

Quantity	1
Internal Volume	580 gal
Useful Volume	457.4 gal
Design Pressure	Atmospheric
Design Temperature	200°F
Normal Operating Temperature	150°F
Type Heater	Electric Immersion
Heater Capacity	31.5 kw Minimum
Code	ASME B&PV Code, Section VIII
Fluid	6-1/4 Wt % Boric Acid
Material	Stainless Steel

1.13 Boric Acid Strainer - F10 (YS-0224)

Quantity	1
Type	Basket
Design Pressure	125 psig
Design Temperature	250°F
Screen Size	100 x 100 US Mesh
Design Flow	50 gpm
Material	Stainless Steel
Fluid	6-1/4 Wt % Boric Acid

1.14 Concentrated Boric Acid Storage Tanks - T53A and T53B

Quantity	2
Internal Volume	6,550 gal
Design Pressure	Atmospheric
Design Temperature	200°F
Normal Operating Temperature	140°F
Type Heater	Electrical, Dry Well Installation
Heater Capacity	Two Independent 4 kW Banks per Tank
Fluid	6-1/4 Wt % Boric Acid (HBO)
Normal Boron Concentration	10,936 ppm
Material	Stainless Steel
Code	ASME B&PV Code, Section III, Class C, 1965

1.15 Boric Acid Pumps - P56A and P56B

Quantity	2
Type	Centrifugal
Design Pressure	150 psig
Design Temperature	250°F
Design Head	225 ft
Design Flow	143 gpm
Minimum Flow	10 gpm

Normal Operating Temperature	160°F
NPSH Required	7.50 ft (Ref. 25)
NPSH Available P-56A/B	25.06/25.23 (Ref. 25)
Horsepower	30
Fluid	6-1/4 Wt % Boric Acid
Material in Contact With Liquid	Stainless Steel

1.16 Boric Acid Filter - F9

Quantity	1
Type Elements	Synthetic Fiber
Retention of 5 Micron Particles	98%
Design Pressure	150 psig
Design Temperature	250°F
Design Flow	140 gpm
Material	Stainless Steel
Liquid	6-1/4 Wt % Boric Acid
Code	ASME B&PV Code, Section III, Class C, 1965

1.17 Chemical Addition Tank - T56

Quantity	1
Capacity	10.5 gal
Design Pressure	Atmospheric
Design Temperature	200°F
Normal Operating Temperature	Ambient
Material	Stainless Steel
Fluid	Hydrazine (N_2H_4), LiOH, KOH, NH_4OH
Code	ASME B&PV Code, Section VIII

1.18 Chemical Addition Strainer - F58

Quantity	1
Type	Basket
Design Pressure	100 psig
Design Temperature	250°F
Screen Size	60 US Mesh
Design Flow	30 gph
Material	Stainless Steel
Fluid	Hydrazine (N_2H_4), LiOH, KOH, NH_4OH

1.19 Metering Pump - P57

Quantity	1
Type	Air Operated Double Diaphragm
Design Pressure	120 psig
Design Temperature	190°F
Design Flow Rate	0 to 35 gpm
Design Air Consumption	0 to 50 scfm
Normal Fluid Temperature	75°F
Material	Stainless Steel
Fluid	Hydrazine (N_2H_4), LiOH, KOH, NH_4OH

TABLE 9-21

(Deleted)

TABLE 9-22

DELETED

TABLE 9-23

FUEL HANDLING DATA

1. New Fuel Storage Rack	
Core Storage Capacity	1/6
Equivalent Fuel Assemblies	36
Center-to-Center Spacing of Assemblies	11 in
2. Spent Fuel Storage Pool	
Core Storage Capacity	4.3
Equivalent Fuel Assemblies	892
Number of Space Accommodations for Spent Fuel Shipping Casks	1
Center-to-Center Spacing of Assemblies	
Region 1	10-1/4 in
Region 2	9.17 in
Minimum k_{eff} With Unborated Water	0.95
3. Miscellaneous Details	
Wall Thickness for Spent Fuel Storage Pool	4 ft to 6-1/2 ft
Weight of Fuel Assembly	1,500 lb
Capacity of Refueling Water Storage Tank	285,000 gal
Quantity of Water Required for Refueling	250,000 gal

TABLE 9-24
(Sheet 1 of 2)

FUEL BUILDING CRANE

Main Hoist	5 ft/min at Full Load (5 Steps), 50 hp at 1200 r/min
Main Hoist Brake Capacity	380 ft-lb
Auxiliary Hoist	28 ft/min at Full Load (Stepless), 40 hp at 1,200 r/min
Trolley	75 ft/min at Full Load (3 Steps), 10 hp at 1,200 r/min
Bridge	100 ft/min at Full Load (3 Steps), 15 hp at 1,200 r/min
Service Class	Class A, Electric Overhead Crane Institute Specification 61
Lift Main Hoist	54 ft 0 in
Lift Auxiliary Hoist	108 ft 5 in
Span	44 ft 10 in Center-to-Center Rails
Bridge Travel	Approximately 100 ft
Lifting Tackle	Main Hoist - Rope 16 Parts 1-Inch SS, Drum 44-Inch Pitch Diameter, Sheaves 24-Inch Pitch Diameter
	Auxiliary Hoist - Rope 4 Parts 3/4-Inch SS, Drum 18-3/4-Inch Pitch Diameter, Sheaves 18-3/4-Inch Pitch Diameter
Girders	Welded Box Section
Runway Rail	100 lbs ASCE
Trolley Rail	175 lbs USS
Bridge Drive	Direct Drive Arrangement With Oiltight Center Gear Case
Trolley Drive	Direct Drive Arrangement With Oiltight Center Gear Case
Capacity in Net Tons	Bridge 100 Tons, Main Hoist 100 Tons, Auxiliary Hoist 15 Tons

TABLE 9-24
(Sheet 2 of 2)

Wheels	Bridge Has Eight 21-Inch Diameter Steel Hardened Treads Trolley Has Four 21-Inch Diameter Steel Hardened Treads
Bridge End Assembly	Rotating Axle
Bumpers	Rubber
Bearings	Antifriction Throughout
Gearing	Helical Gearing Heat-Treated Steel Throughout Except Trolley Traverse. All Gearing in Oiltight Casing

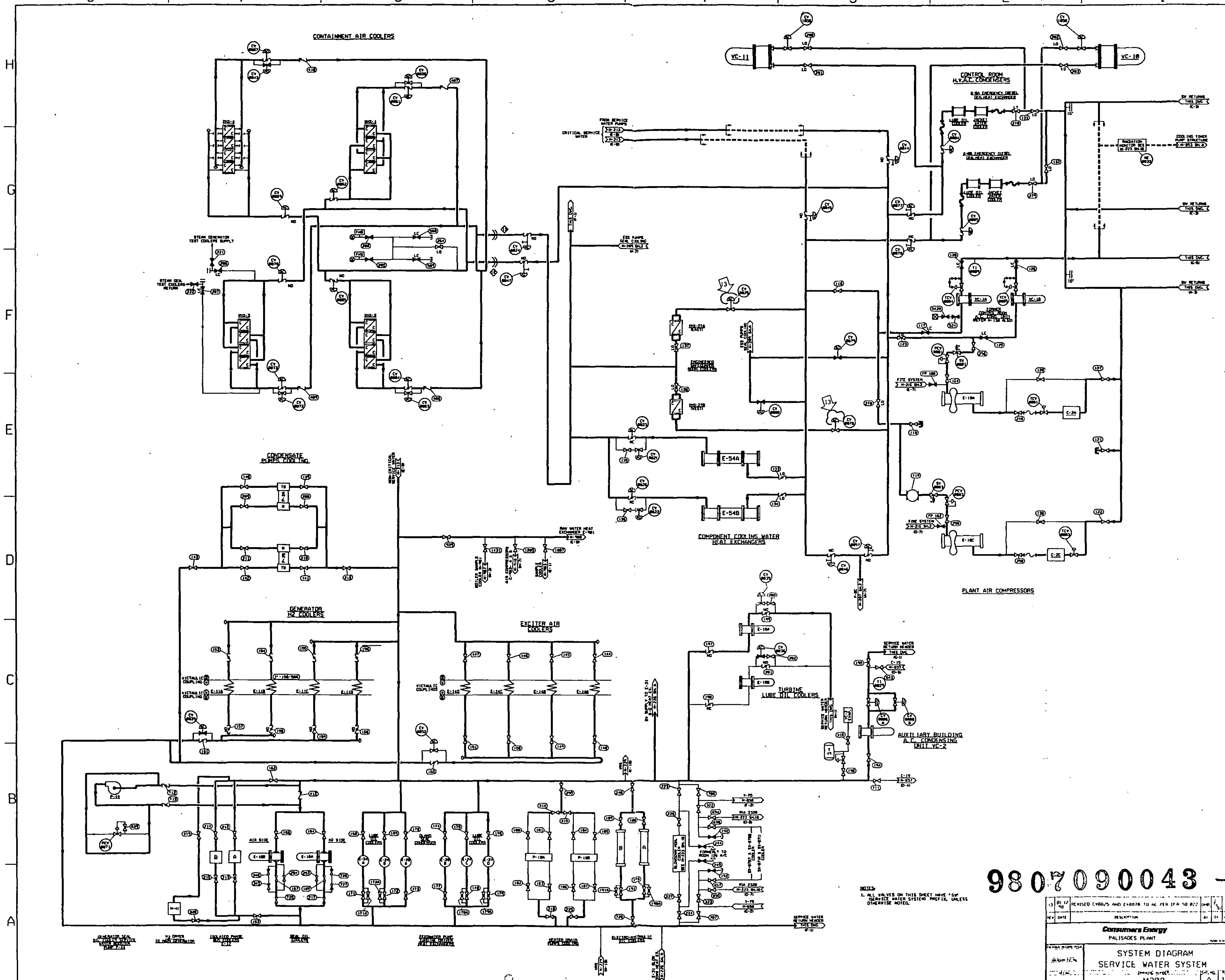

FIG 9-1 SH 1
FSAR REV 20

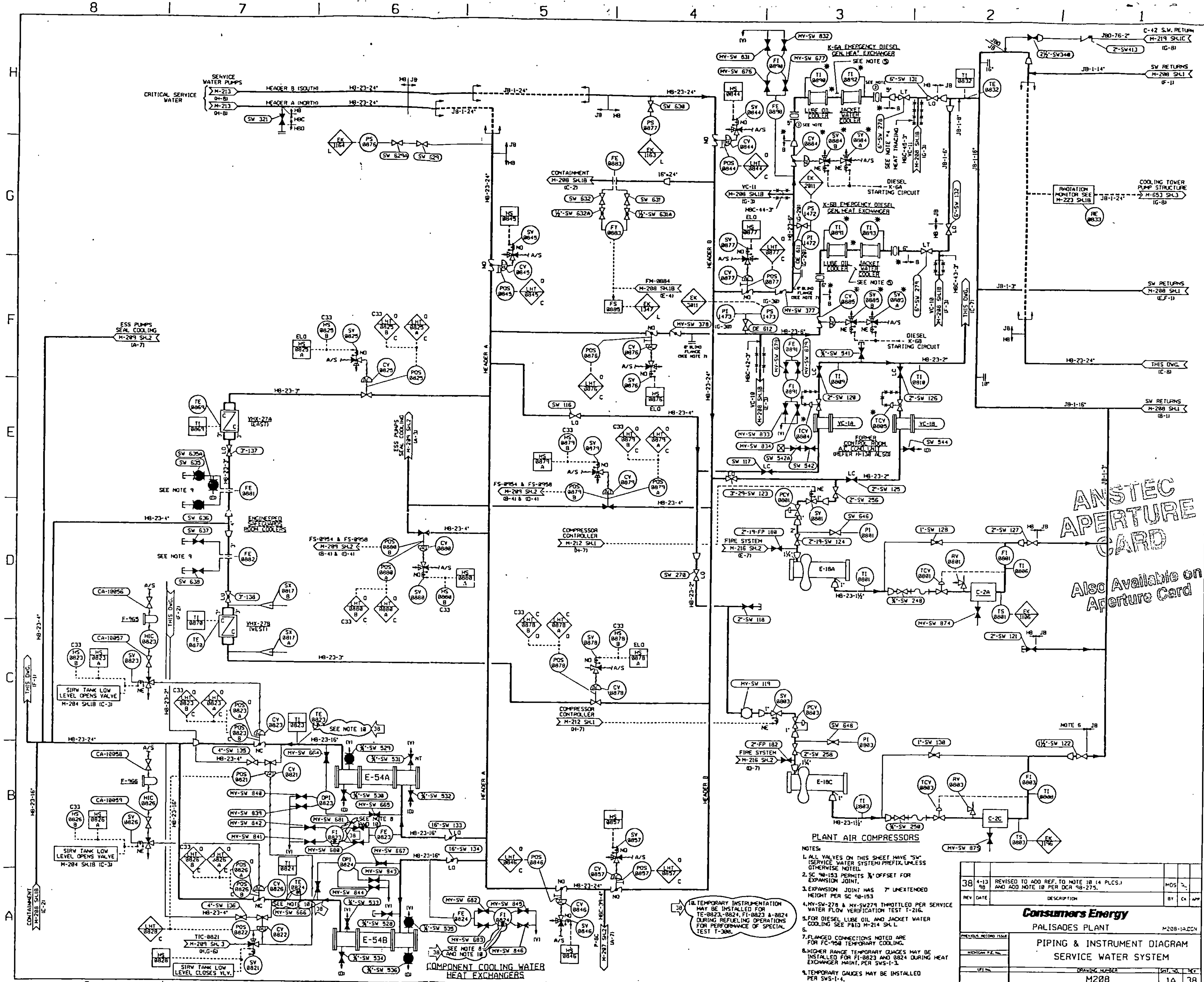

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FIG 9-1 SH 1A
FSAR REV 20

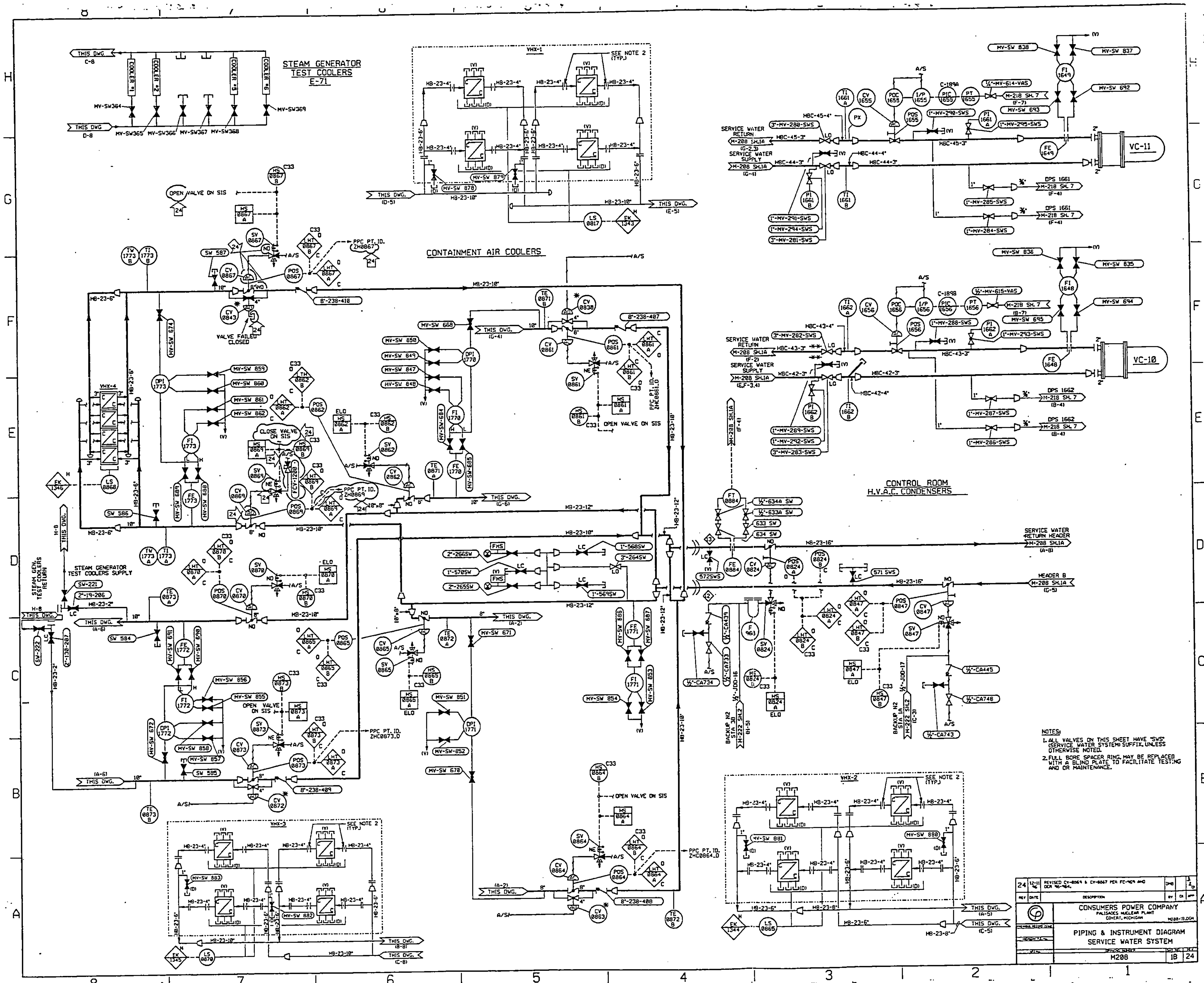

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Figure 9-1 Sh 1B
Rev 19

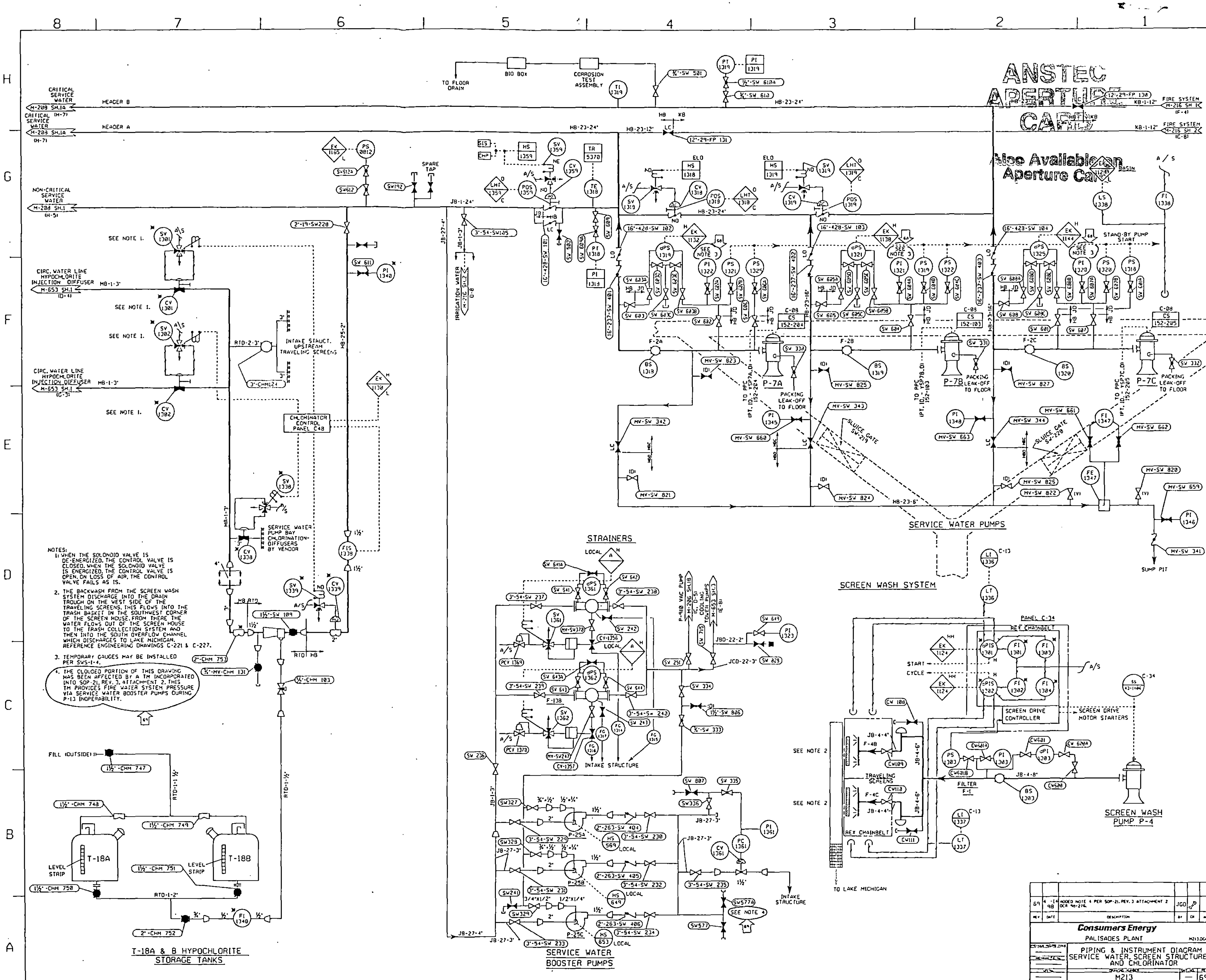

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FIG 9-1 SH 2
FSAR REV 20

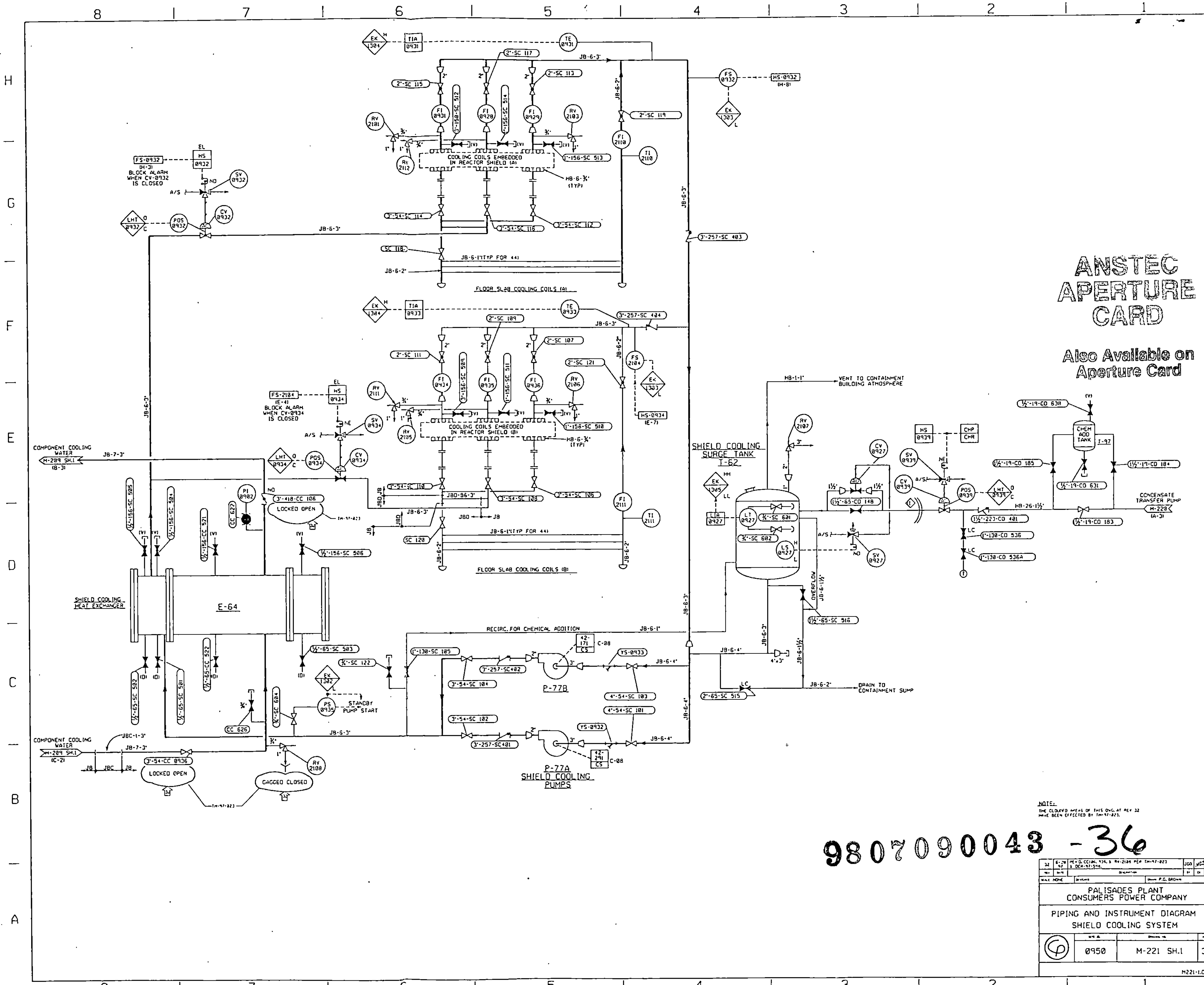
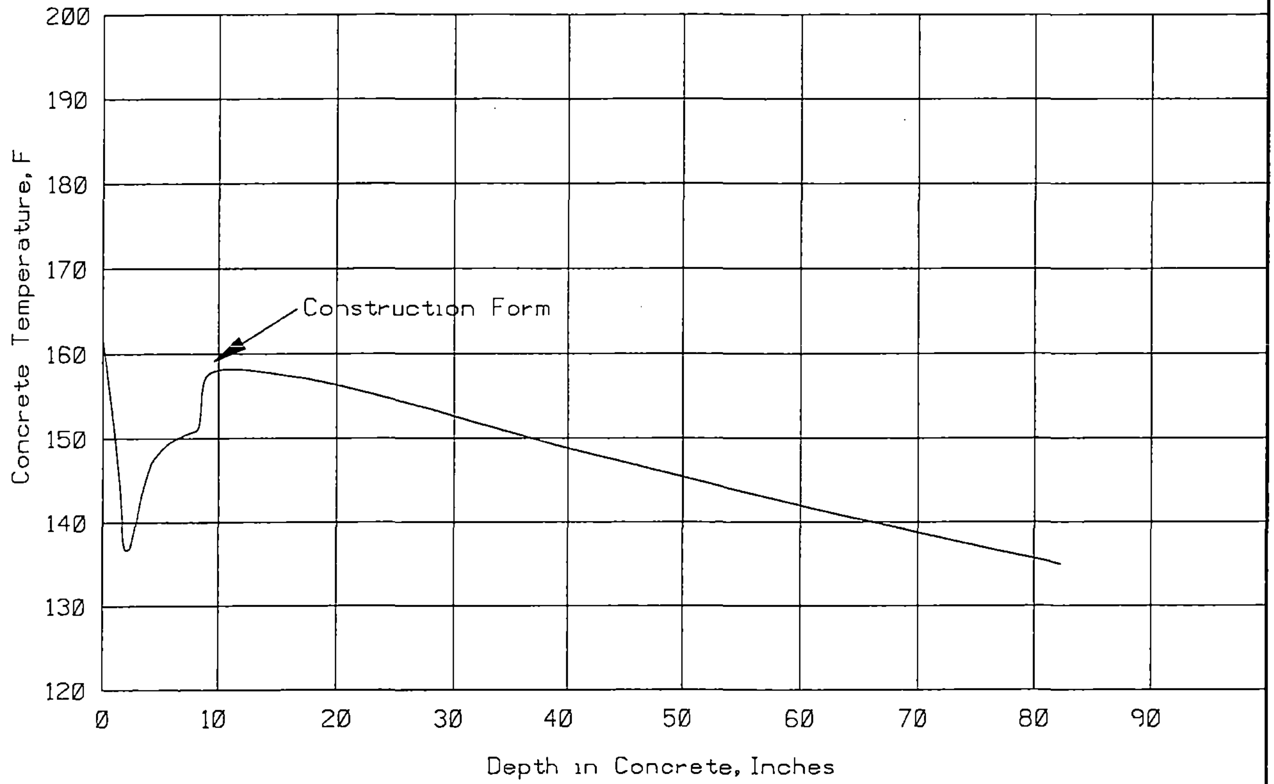


FIG 9-2
FSAR REV 2

FSAR Update

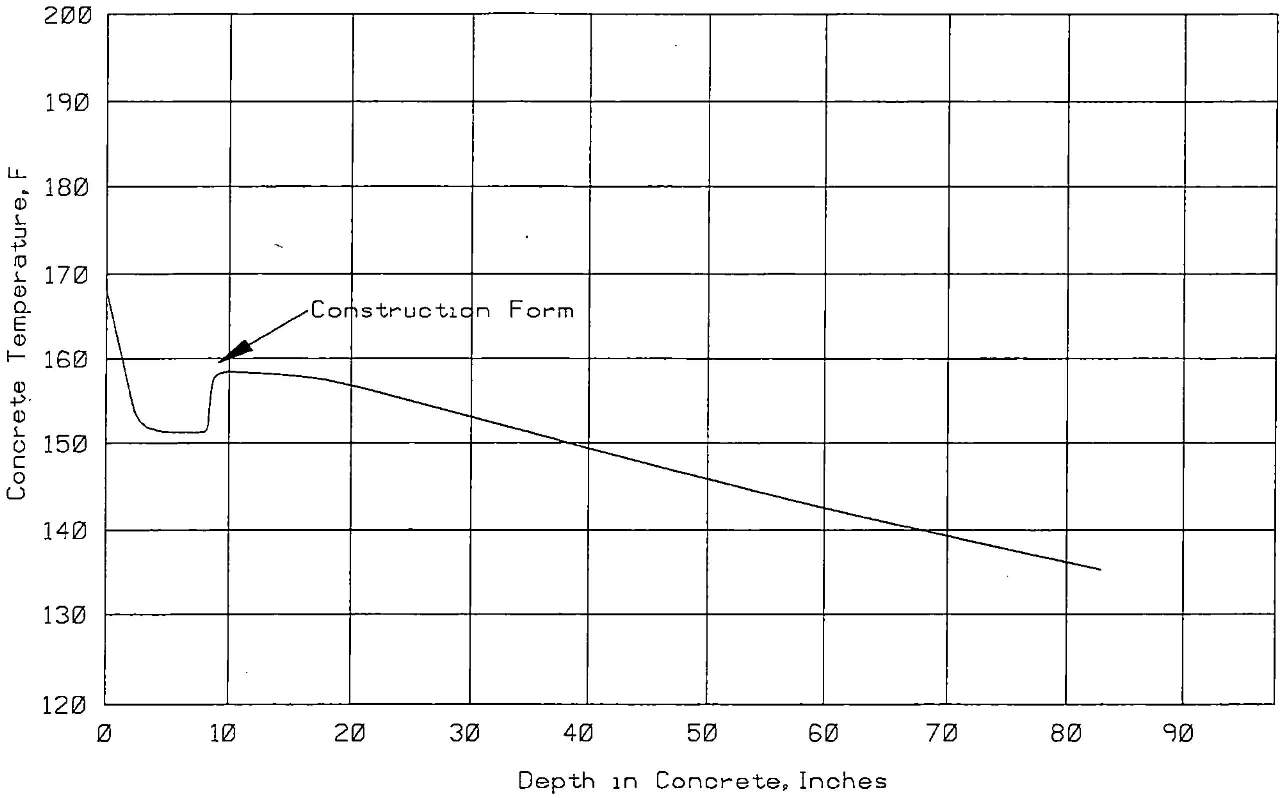


Consumers Energy
PALISADES PLANT
FSAR UPDATE

NOMINAL OPERATING CONDITIONS COOLING COIL CENTERLIN

Figure 9-3
Rev 19

FSAR Update

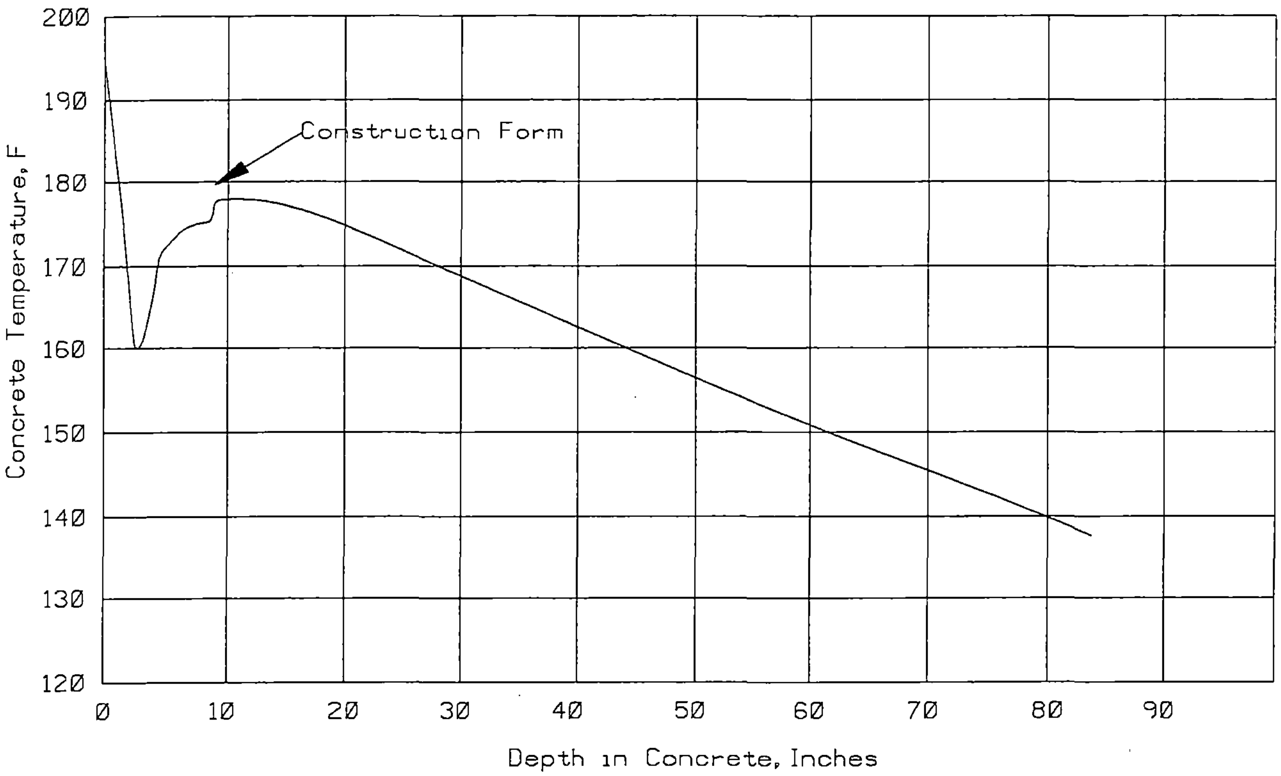


Consumers Energy
PALISADES PLANT
FSAR UPDATE

NOMINAL OPERATING CONDITIONS MID-SPAN OF TWO COOLING COILS

Figure 9-4
Rev 19

FSAR Update

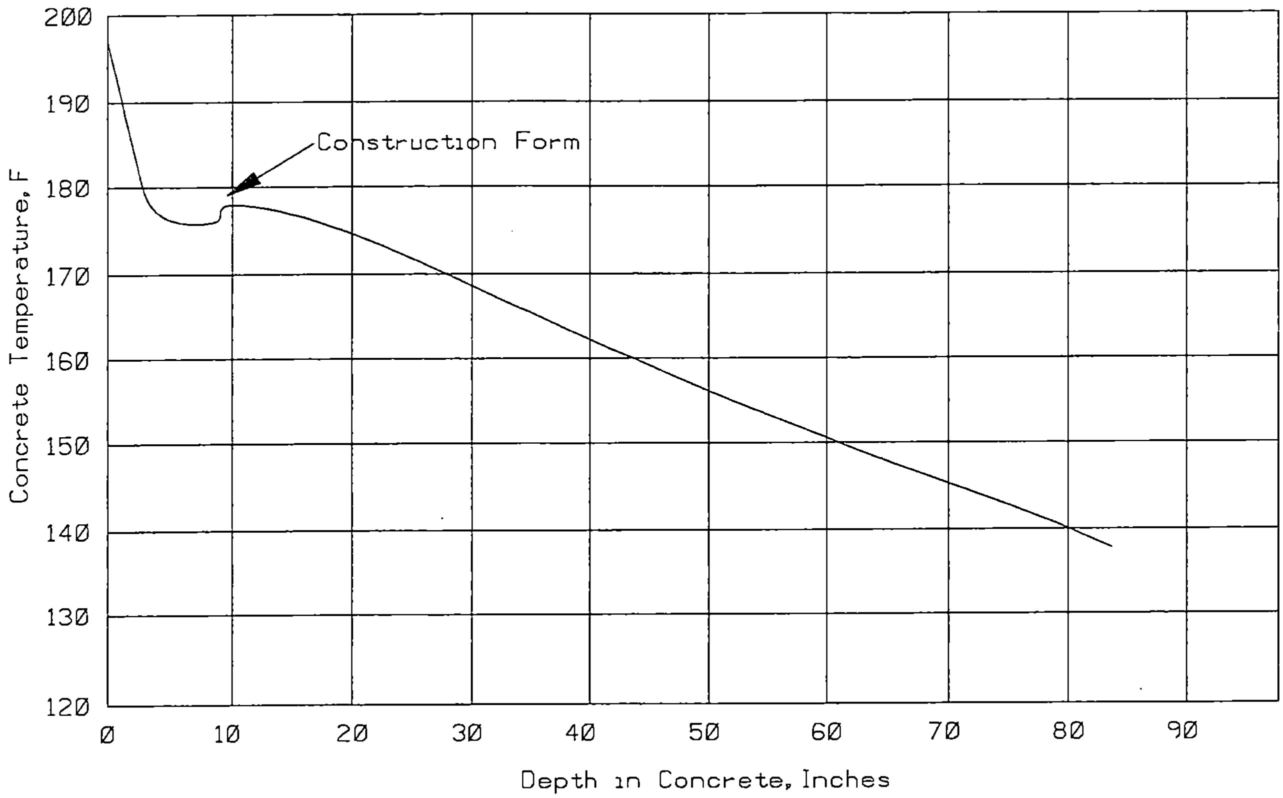


Consumers Energy
PALISADES PLANT
FSAR UPDATE

BOUNDING DESIGN
COOLING COIL
CENTERLINE

Figure 9-5
Rev 19

FSAR Update



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FSAR UPDATE

BOUNDING DESIGN CONDITIONS MID-SPAN OF TWO COOLING COILS

Figure 9-6
Rev 19

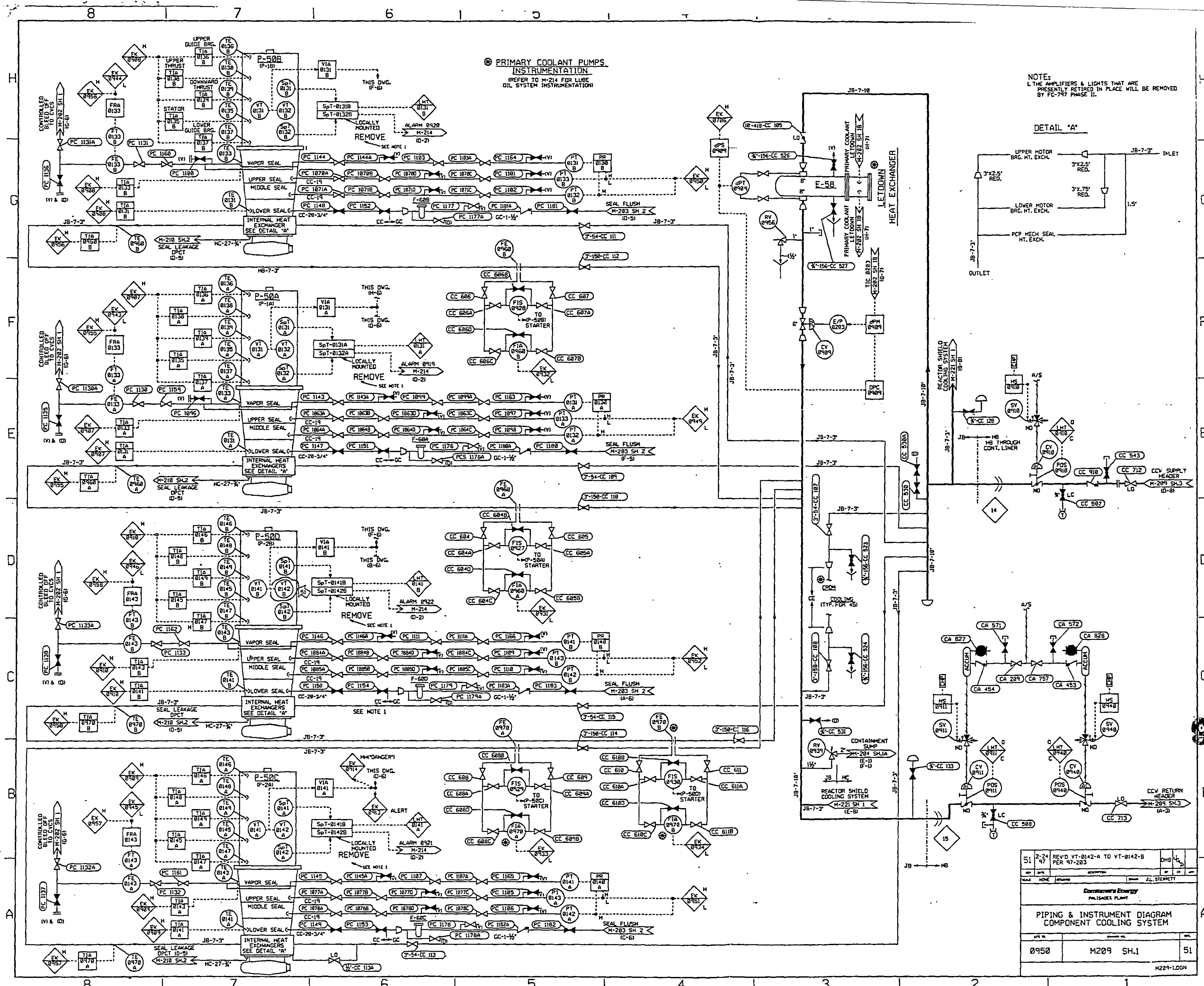

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Figure 9-7 Sh 1
Rev. 19

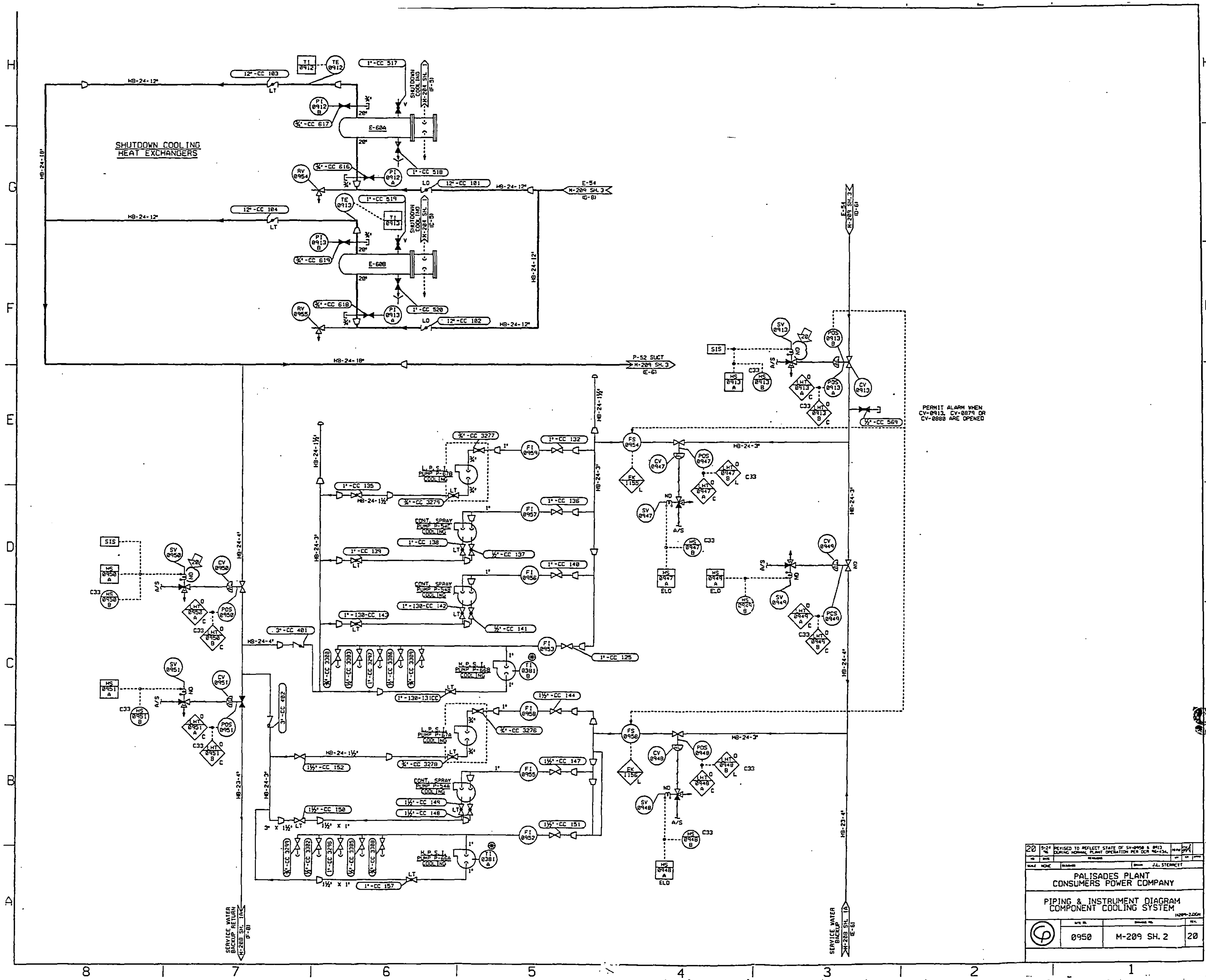

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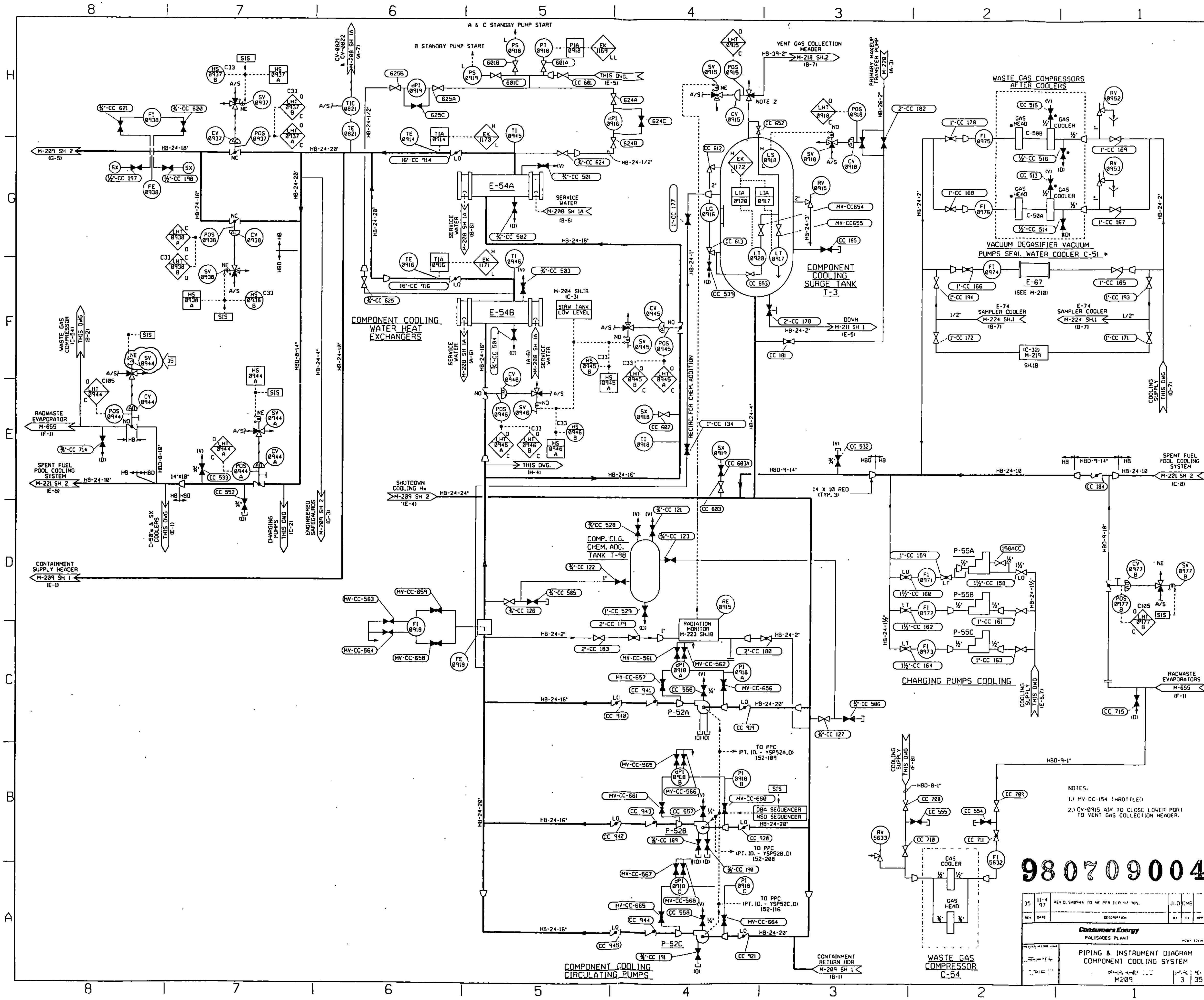
Figure 9-7 Sh 2

Rev 19



ANSTEC APERTURE CARD

Also Available on Aperture Card

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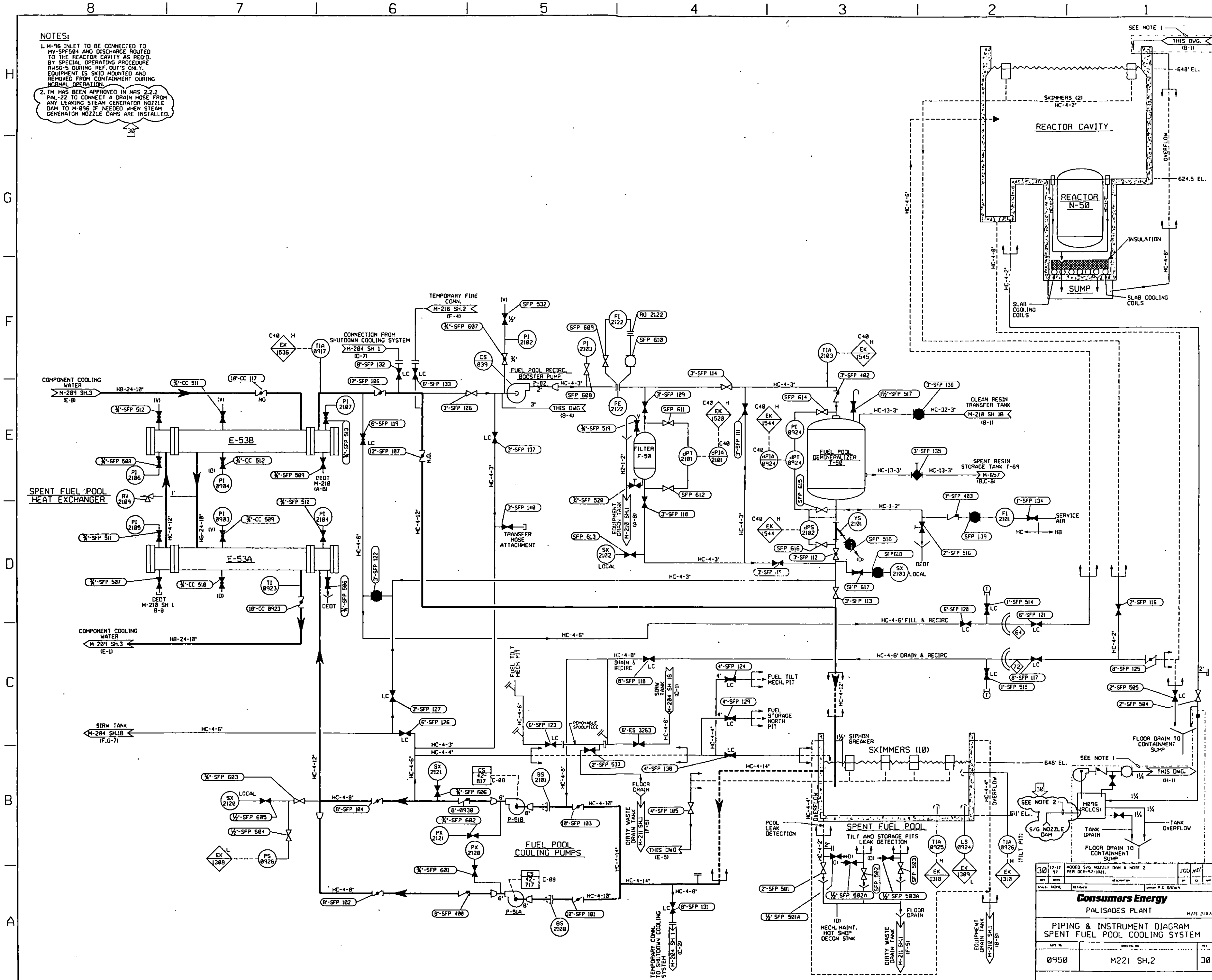
FIG 9-7 Sh 3
FSAR REV 2



ANSTEC APERTURE CARD

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FIG 9-8
FSAR REV 20

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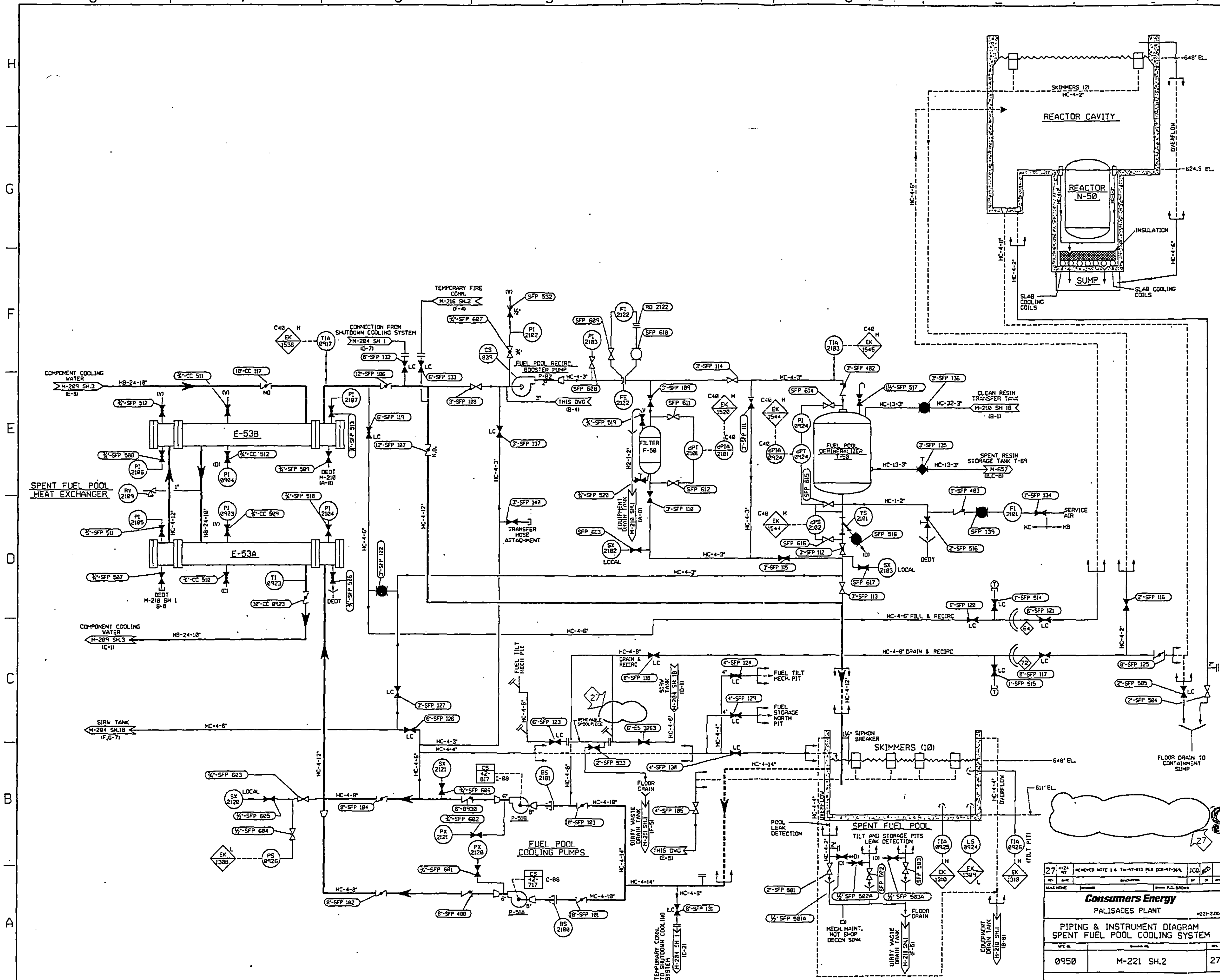
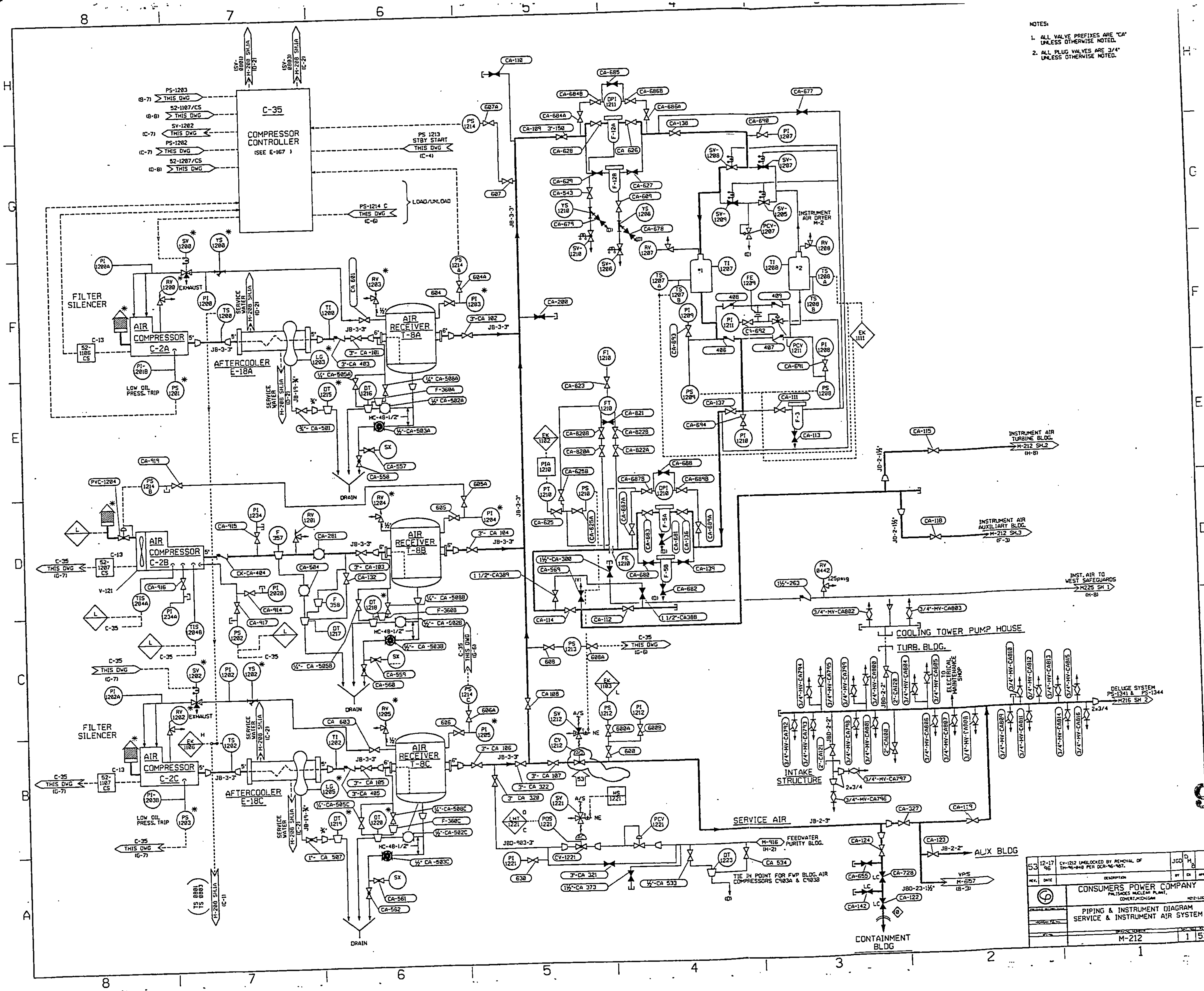


REACTOR CAVITY
SKIMMERS (2)
OVERFLOW
REACTOR N-50
INSULATION
SUMP
SLAB COOLING COILS
648' EL.
624.5 EL.
SPENT FUEL POOL HEAT EXCHANGER
E-53B
E-53A
COMPONENT COOLING WATER
FUEL POOL RECIRC. BOOSTER PUMP
FILTER F-50
FUEL POOL DEMINERALIZER T-58
CLEAN RESIN TRANSFER TANK
SPENT RESIN STORAGE TANK T-69
SERVICE AIR
TEMPORARY FIRE CONN.
CONNECTION FROM SHUTDOWN COOLING SYSTEM
TRANSFER HOSE ATTACHMENT
EQUIPMENT DRAIN TANK
SIRW TANK
FUEL TILT MECH PIT
FUEL STORAGE NORTH PIT
HC-4-6" FILL & RECIRC
HC-4-8" DRAIN & RECIRC
FLOOR DRAIN TO CONTAINMENT SUMP
SKIMMERS (10)
SIPHON BREAKER
SPENT FUEL POOL
TILT AND STORAGE PITS LEAK DETECTION
POOL LEAK DETECTION
FUEL POOL COOLING PUMPS
P-51B
P-51A
DIRTY WASTE DRAIN TANK
FLOOR DRAIN
MECH MAINT. HOT SHOP DECON SINK
TEMPORARY CONN. TO SHUTDOWN COOLING SYSTEM
611' EL.
Consumers Energy
PALISADES PLANT
PIPING & INSTRUMENT DIAGRAM
SPENT FUEL POOL COOLING SYSTEM
0950
M-221 SH.2
27

Fig
Re



ANSTEC APERTURE CARD

Also Available on Aperture Card

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Figure 9-9 Sh 1
Rev 19

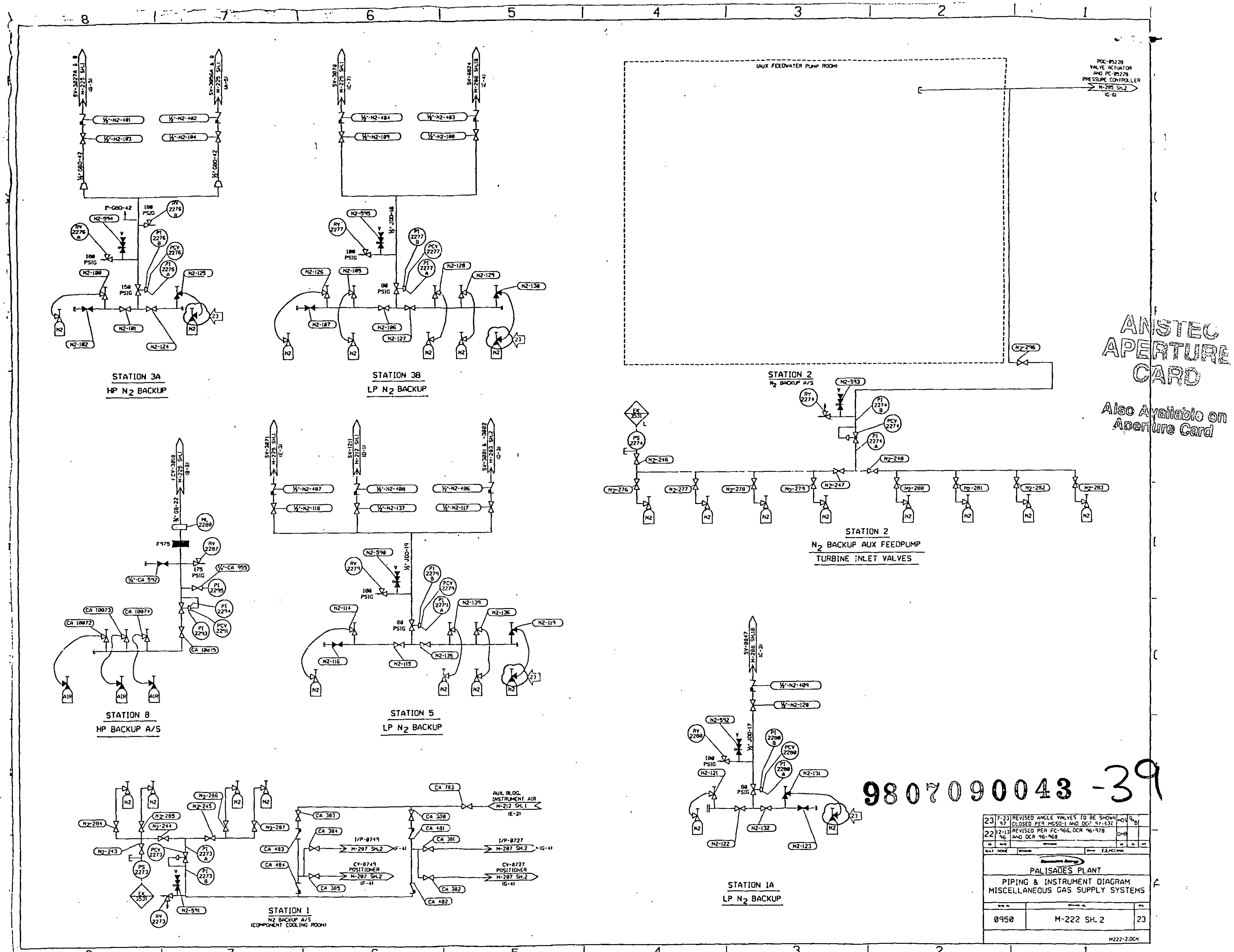
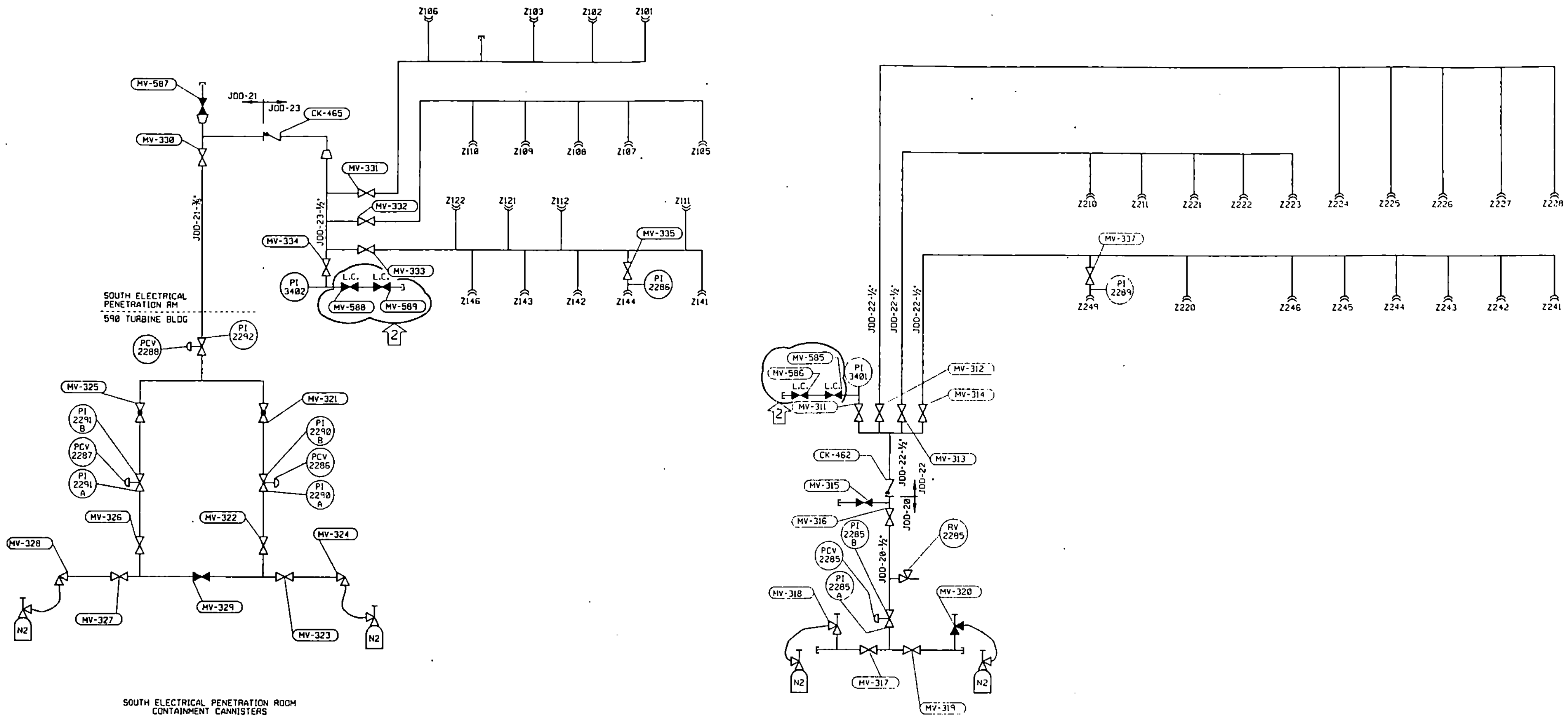


FIG 9-9 SH 2
FSAR REV 20



SOUTH ELECTRICAL PENETRATION ROOM CONTAINMENT CANNISTERS

LP NITROGEN STATION 7

STATION 6
LP NITROGEN
NORTH ELECTRICAL PENETRATION ROOM CONTAINMENT CANISTERS

NOTES
1. ALL VALVE SYSTEM DESIGNATIONS TO BE N_2 UNLESS NOTED OTHERWISE.

REFERENCE DRAWINGS:

VEN-M-110 SH. 363(1) NORTH PENETRATION ROOM N_2 TUBING TO PENETRATIONS ISOMETRIC

VEN-M-110 SH. 363(2) SOUTHWEST PENETRATION ROOM N_2 TUBING TO PENETRATIONS ISOMETRIC

9807090043 -40

2 REV'D. MV585, 586, 588, & 589 TO L.C. PER DCR-97-767.

PALISADES PLANT
CONSUMERS POWER COMPANY

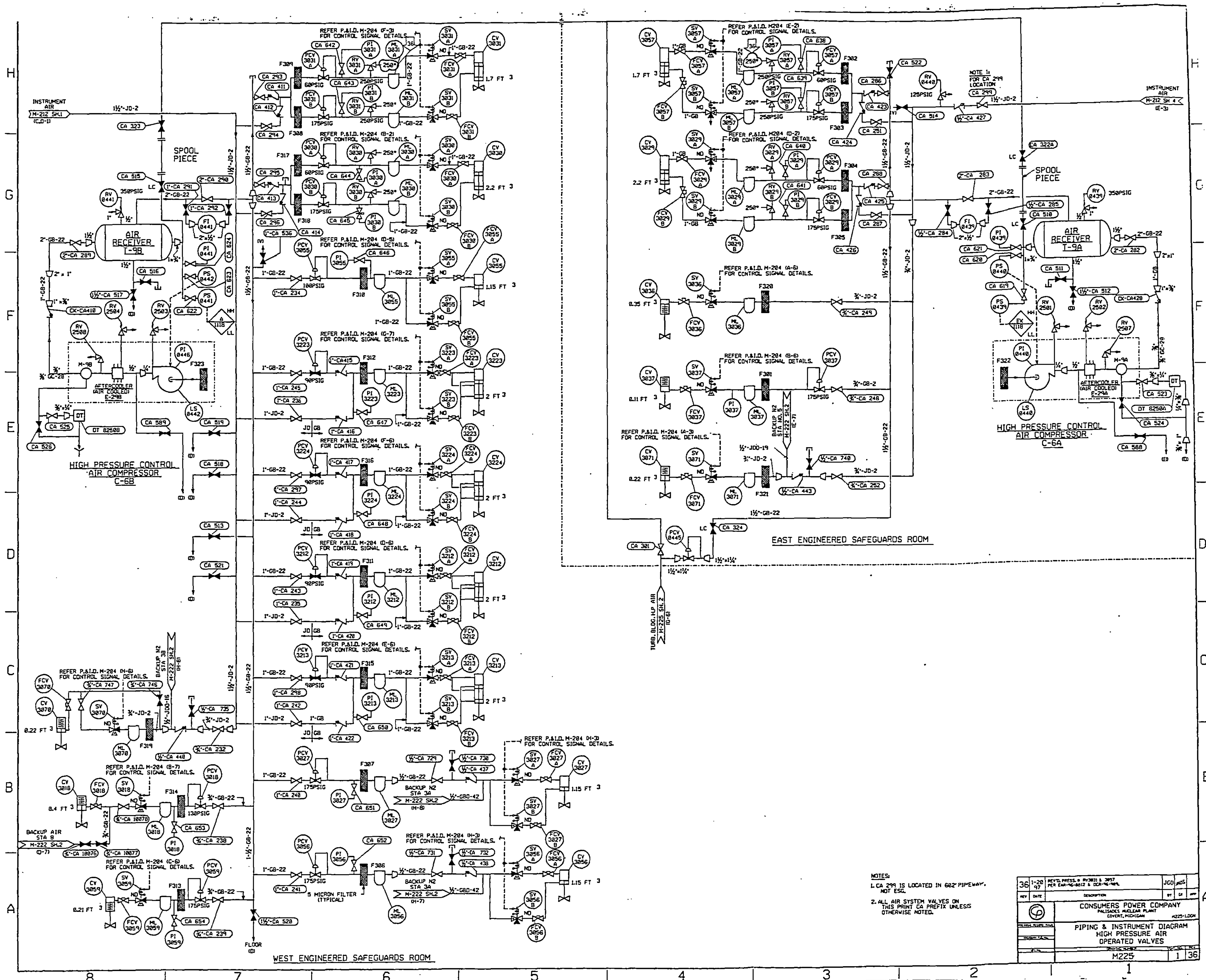
PIPING & INSTRUMENT DIAGRAM
MISCELLANEOUS GAS SUPPLY SYSTEMS

0950 M222 SH. 3 2

ANSTEC APERTURE CARD

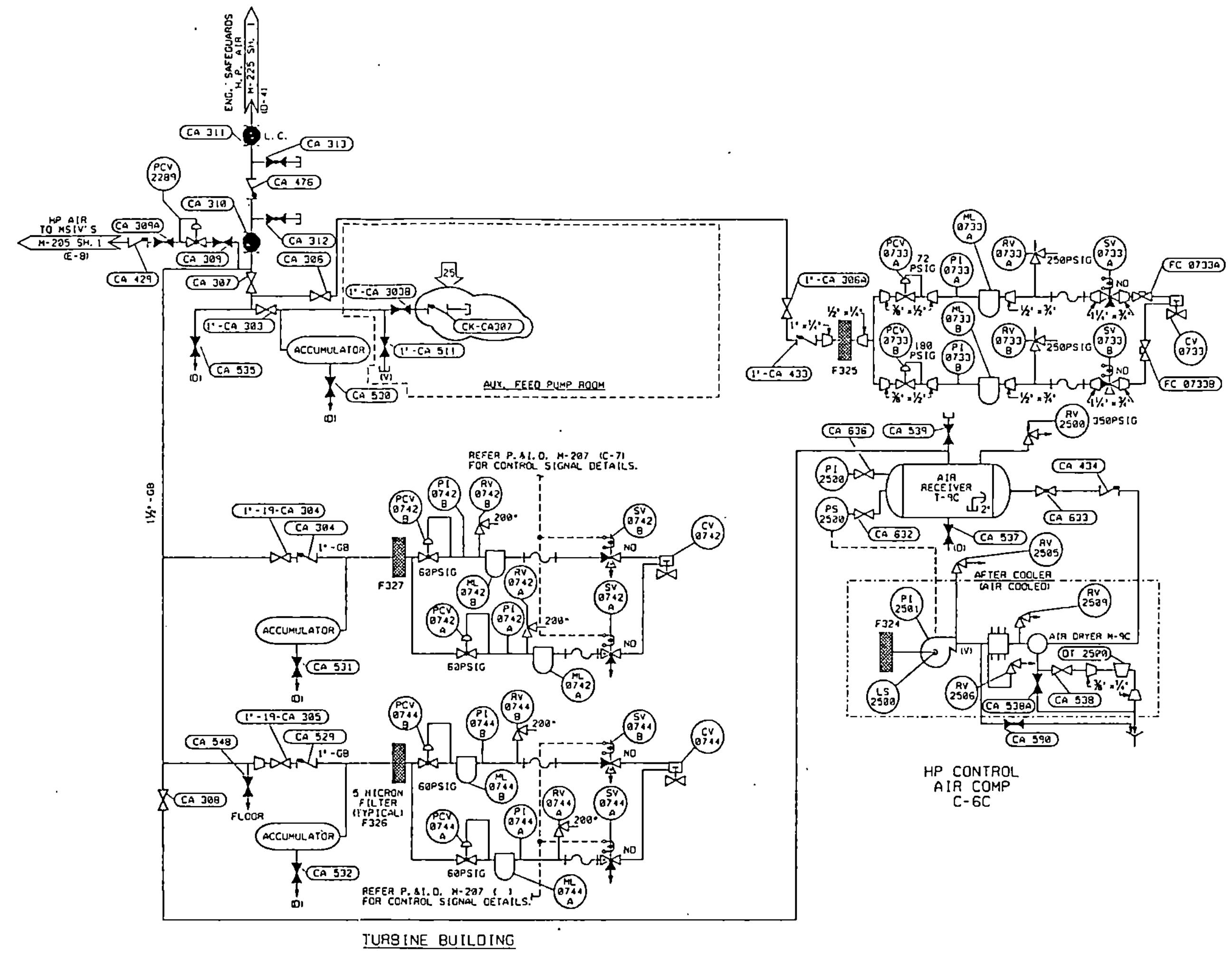
Also Available on Aperture Card

FIG 9-9 SH 3
FSAR REV 20



9707070096-12

Figure 9-10 Sh 1
Rev 19



9807090043 -41

NOTE
ALL AIR SYSTEM VALVES ON THIS PRINT CA PREFIX UNLESS OTHERWISE NOTED.

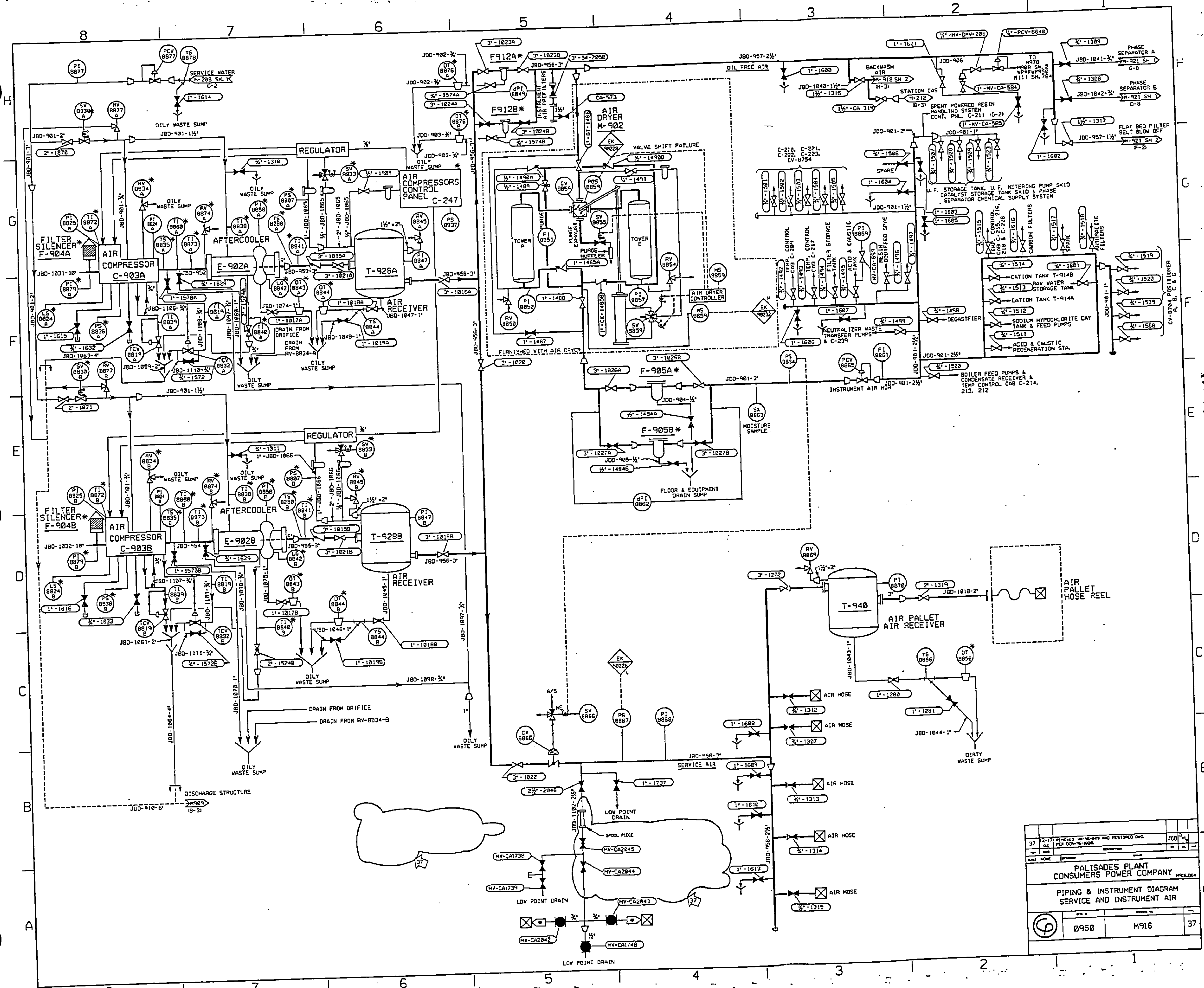
REV.	DATE	DESCRIPTION	BY	CHK	APP
25	7-2-97	ADDED CA307 PER C-PAP-97-0848, FC-966, & DCA-97-625.	JCO	MS	

Consumers Energy
PALISADES PLANT

PIPING & INSTRUMENT DIAGRAM
HIGH PRESSURE AIR
OPERATED VALVES

DRAWING NUMBER	SHT. NO.	REV.
M225	2	25

FIG 9-10 Sh 2
FSAR REV 20



ANSTEC APERTURE CARD

Also Available on Aperture Card

9707070096 -15

Figure
Rev

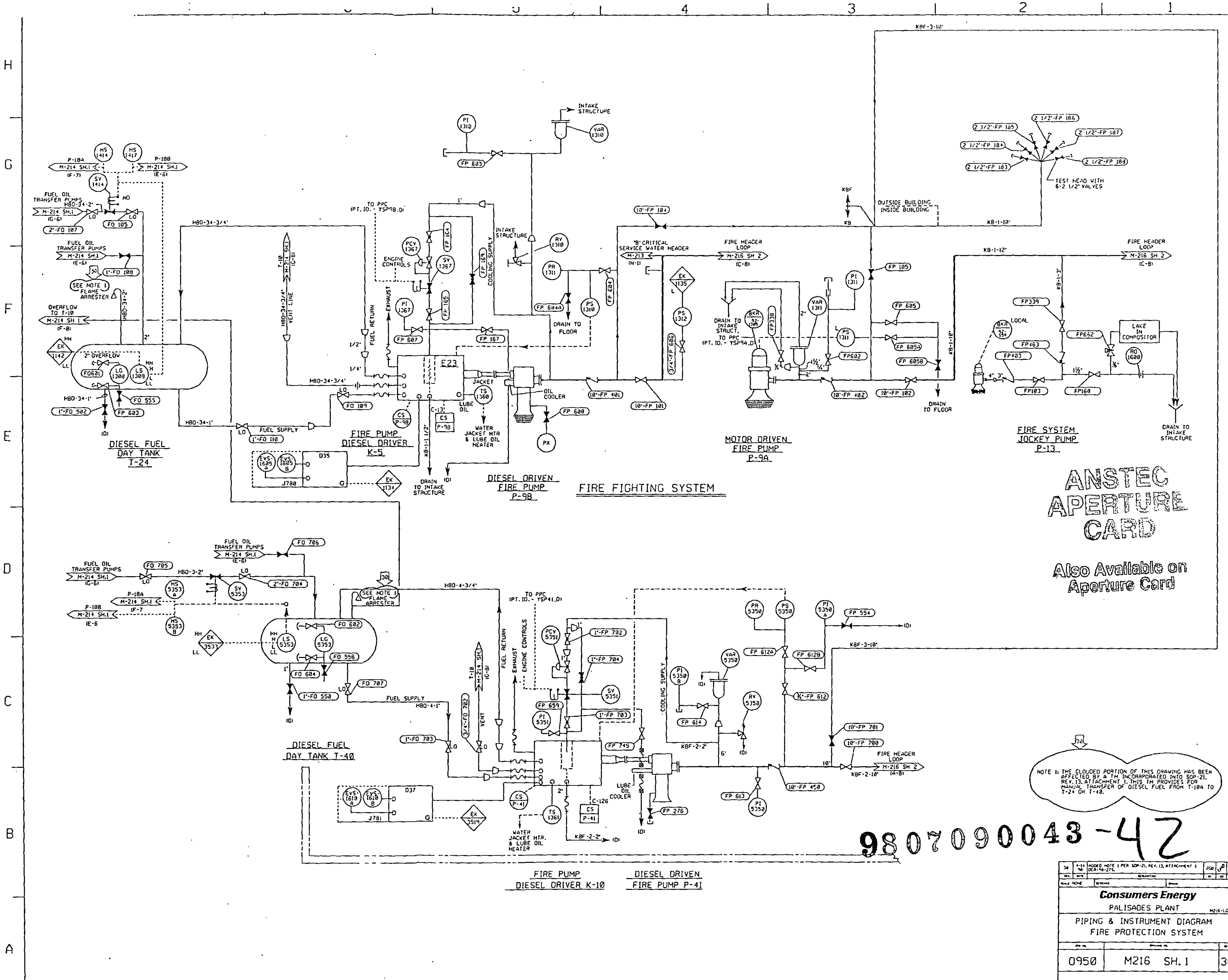


FIG 9-12 SH 1
FSAR REV 20

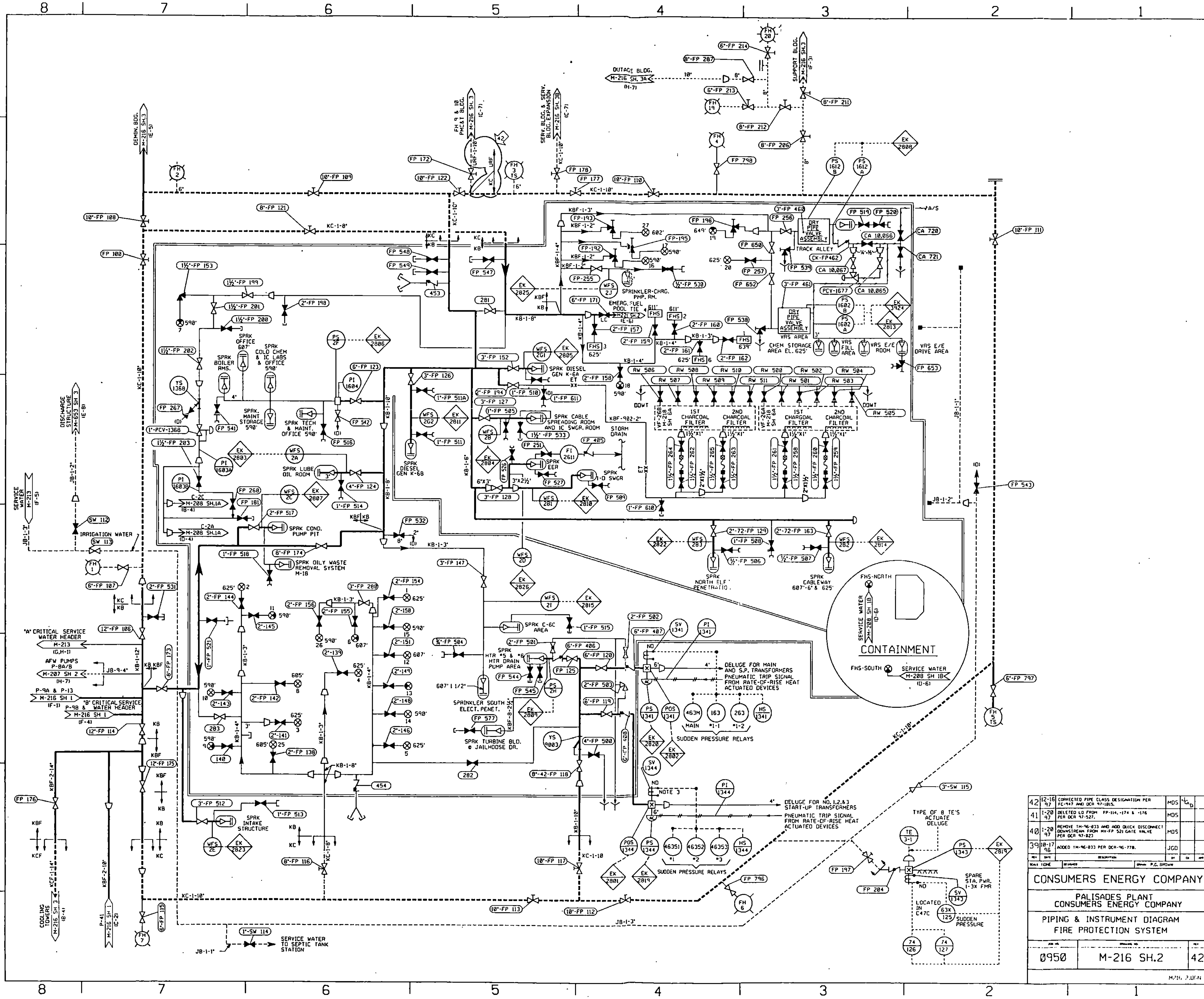
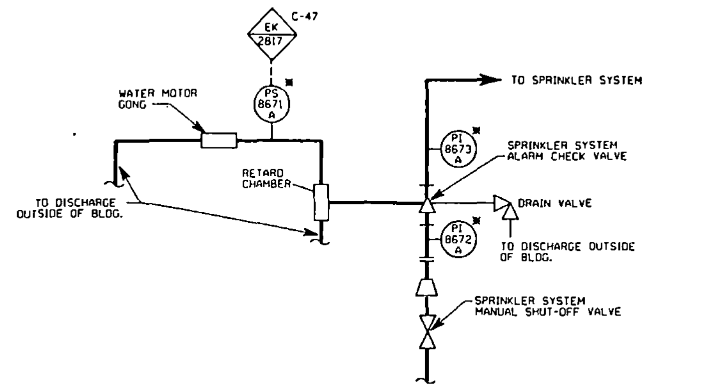


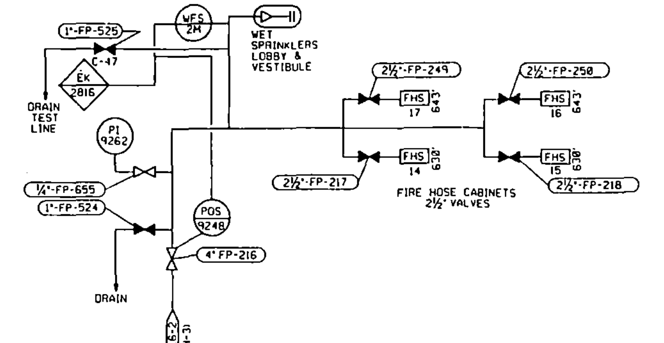
FIG 9-13
FSAR REV 20

9807090043 -43

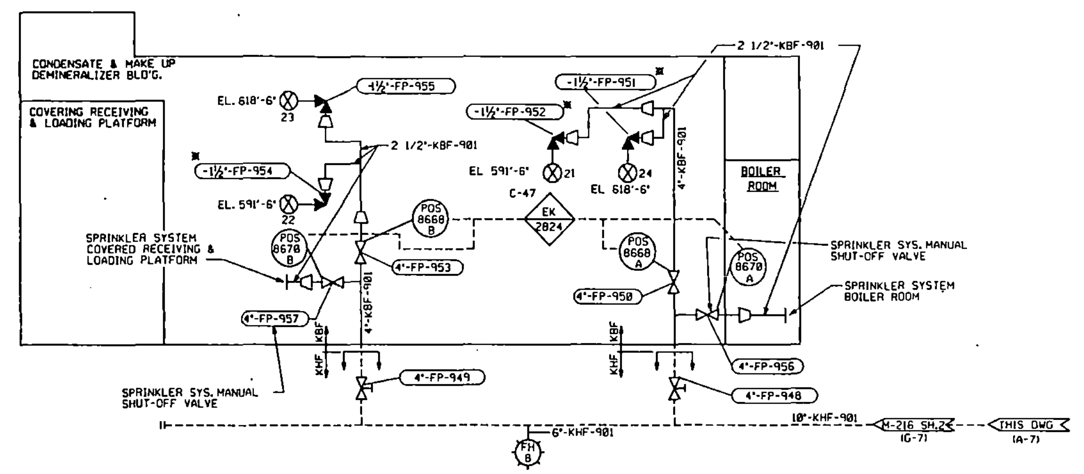


*TYPICAL SPRINKLER SYSTEM ALARM DETAILS (SEE TABLE)

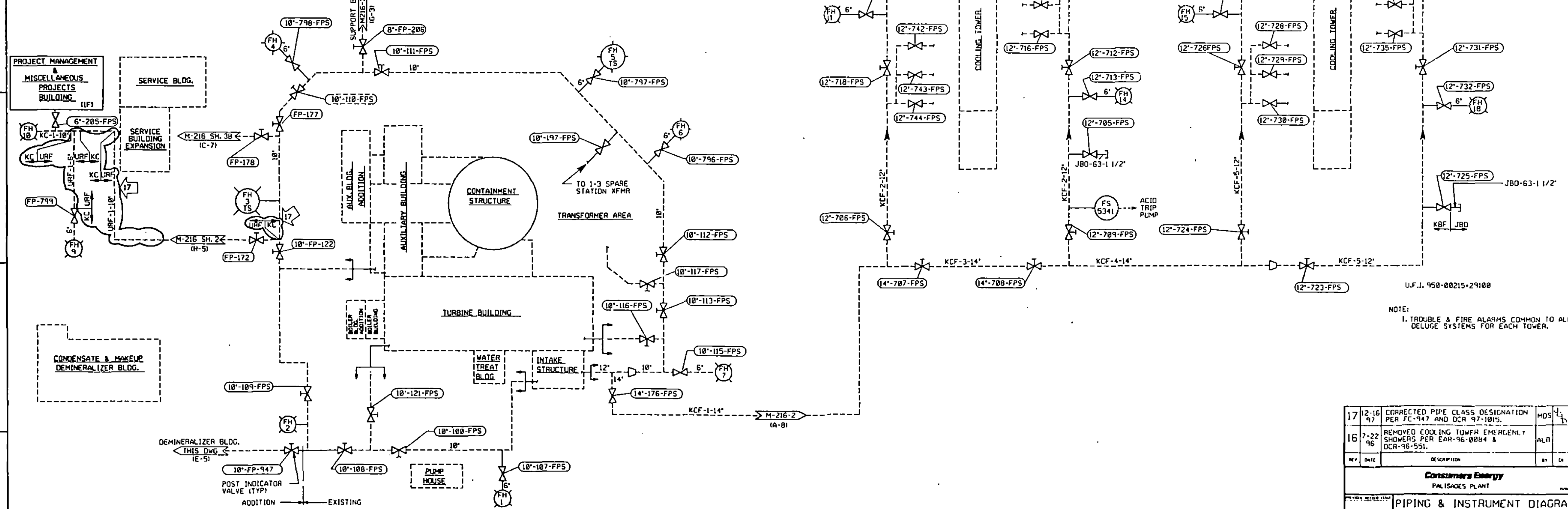
AREA / INSTR.	* FWP BOILER ROOM SYS.	* FWP COVERED REC & LOAD PLATFORM	*SERVICE BLDG-590 MECH.MAINT.	* SERVICE BLDG-607	* PM&MP BLDG-590 STRM.& 607 OFF. AREA	SUPPORT BLDG.	SERVICE BUILDING EXPANSION
PRESSURE SWITCH	PS-8671A	PS-8671B	PS-8675	PS-8674	N/A	N/A	N/A
PRESSURE INDICATOR SUPPLY	PI-8672A	PI-8672B	PI-8675	PI-8674	N/A	PI-9262	PI-1691
PRESSURE INDICATOR-SYSTEM USE	PI-8673A	PI-8673A	PI-8675A	PI-8674A	N/A	N/A	PI-1692
WATER FLOW SWITCH	N/A	N/A	N/A	N/A	WFS-2L	WFS-2M	WFS-4A
MAIN MANUAL SHUT-OFF VALVE	FP-956	FP-957	FP-279	FP-278	FP-205	FP-216	FP-311
INSP. TEST DRAIN VALVE	FP-534	FP-535	FP-536	FP-537	FP-522	FP-525	FP-565
C47 ALARM	EK-2817	EK-2818	EK-2812	EK-2812	N/A	EK-2816	EK-2812 EK-2821



P&ID DIAGRAM - SUPPORT BUILDING



P & I DIAGRAM CONDENSATE & MAKE-UP DEMINERALIZER BUILDING



U.F.I. 950-00215-29100

NOTE:
1. TROUBLE & FIRE ALARMS COMMON TO ALL DELUGE SYSTEMS FOR EACH TOWER.

REV	DATE	DESCRIPTION	BY	CK	APP
17	12-16-97	CORRECTED PIPE CLASS DESIGNATION PER FC-947 AND DCR 97-1015.	MDS		
16	7-22-96	REMOVED COOLING TOWER EMERGENCY SHOWERS PER EAR-96-0084 & DCR-96-551.	ALD		

Consumers Energy
PALISADES PLANT

PIPING & INSTRUMENT DIAGRAM
FIRE PROTECTION SYSTEM

M216 | 3 | 17

FIRE PROTECTION SYSTEM DIAGRAM

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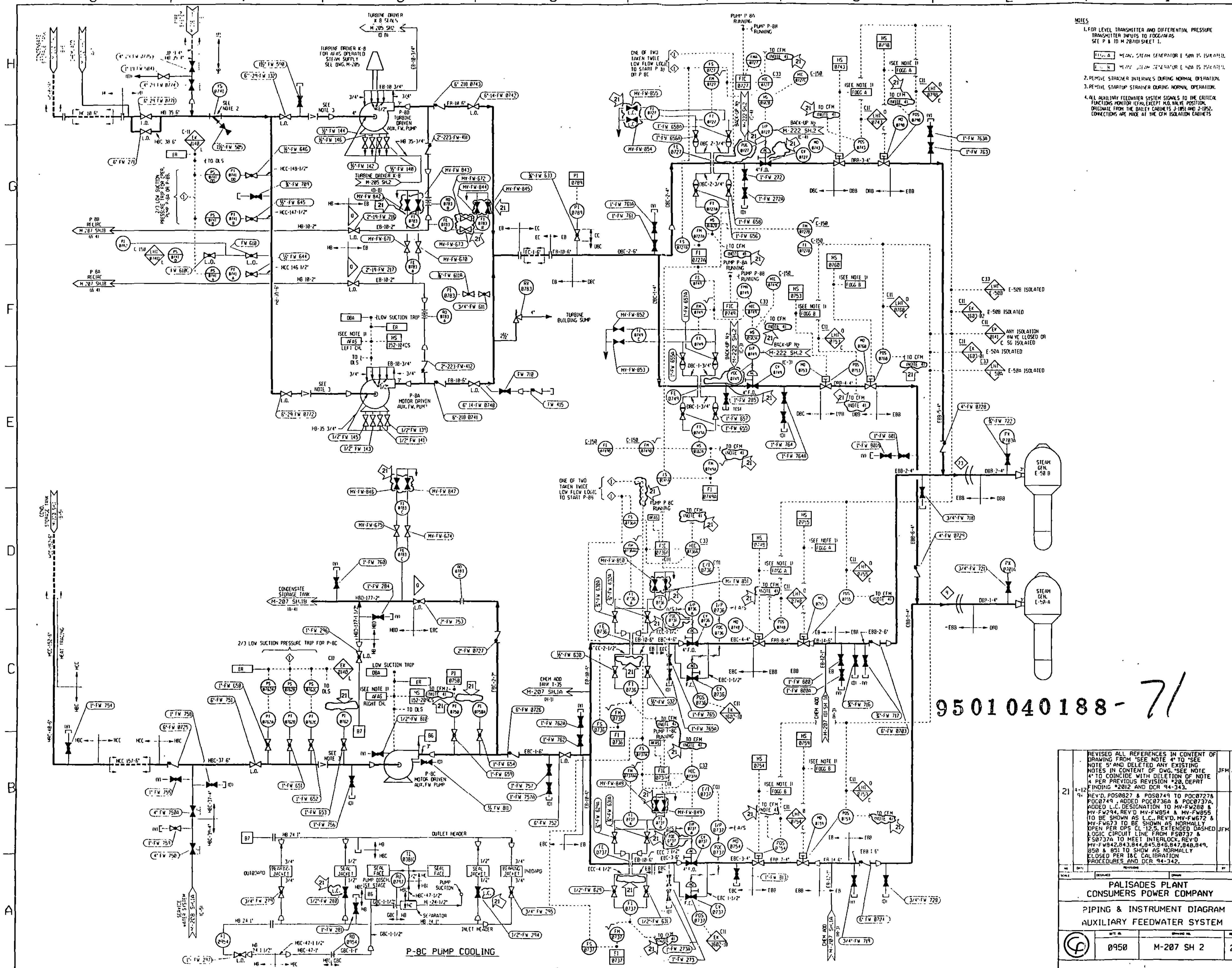
FIG 9-14
FSAR REV 20

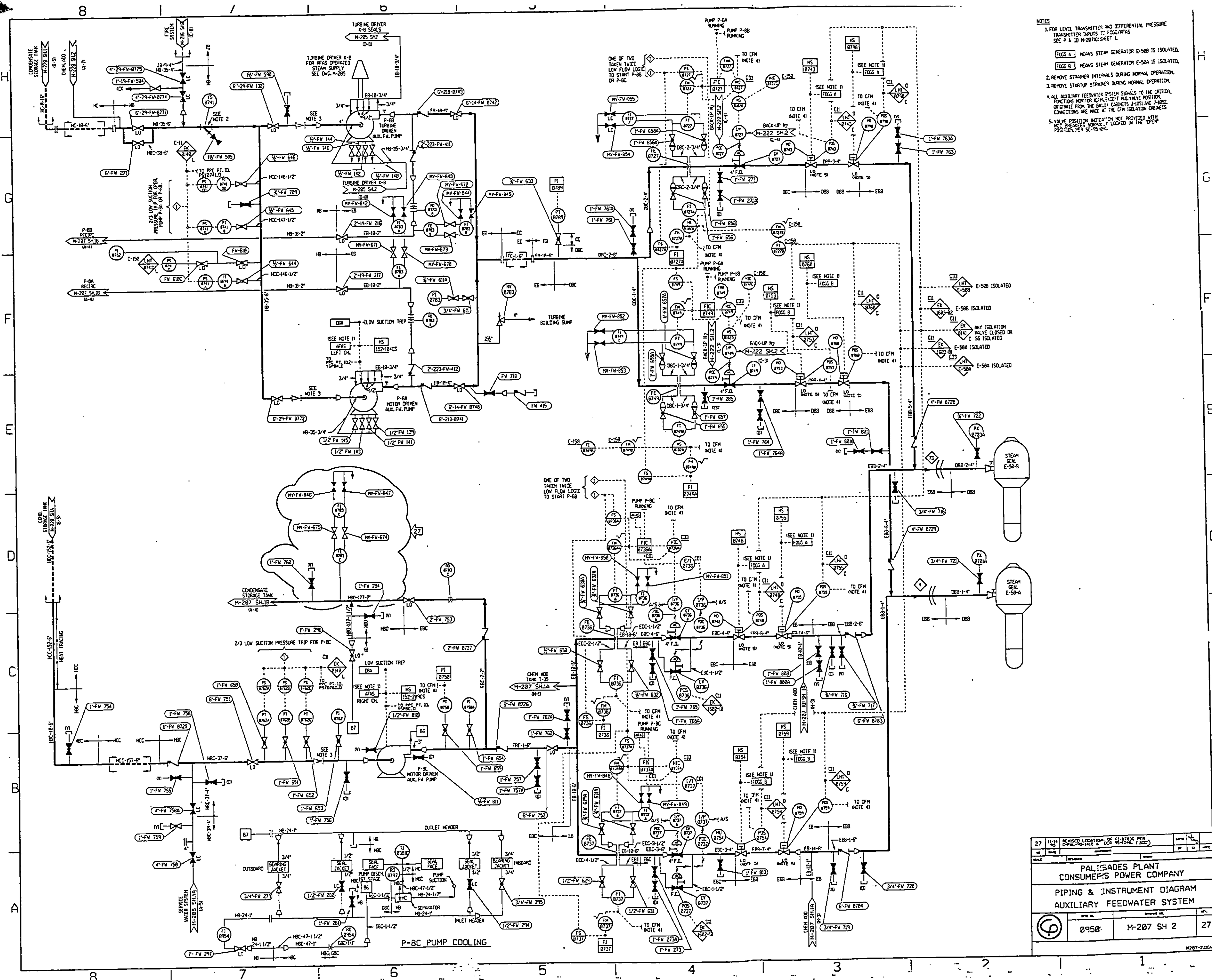
9807090043 - 44



ANSTEC APERTURE CARD

Also Available on Aperture Card

Figure 9-15 FSAR Rev 17



ANSTEC APERTURE CARD

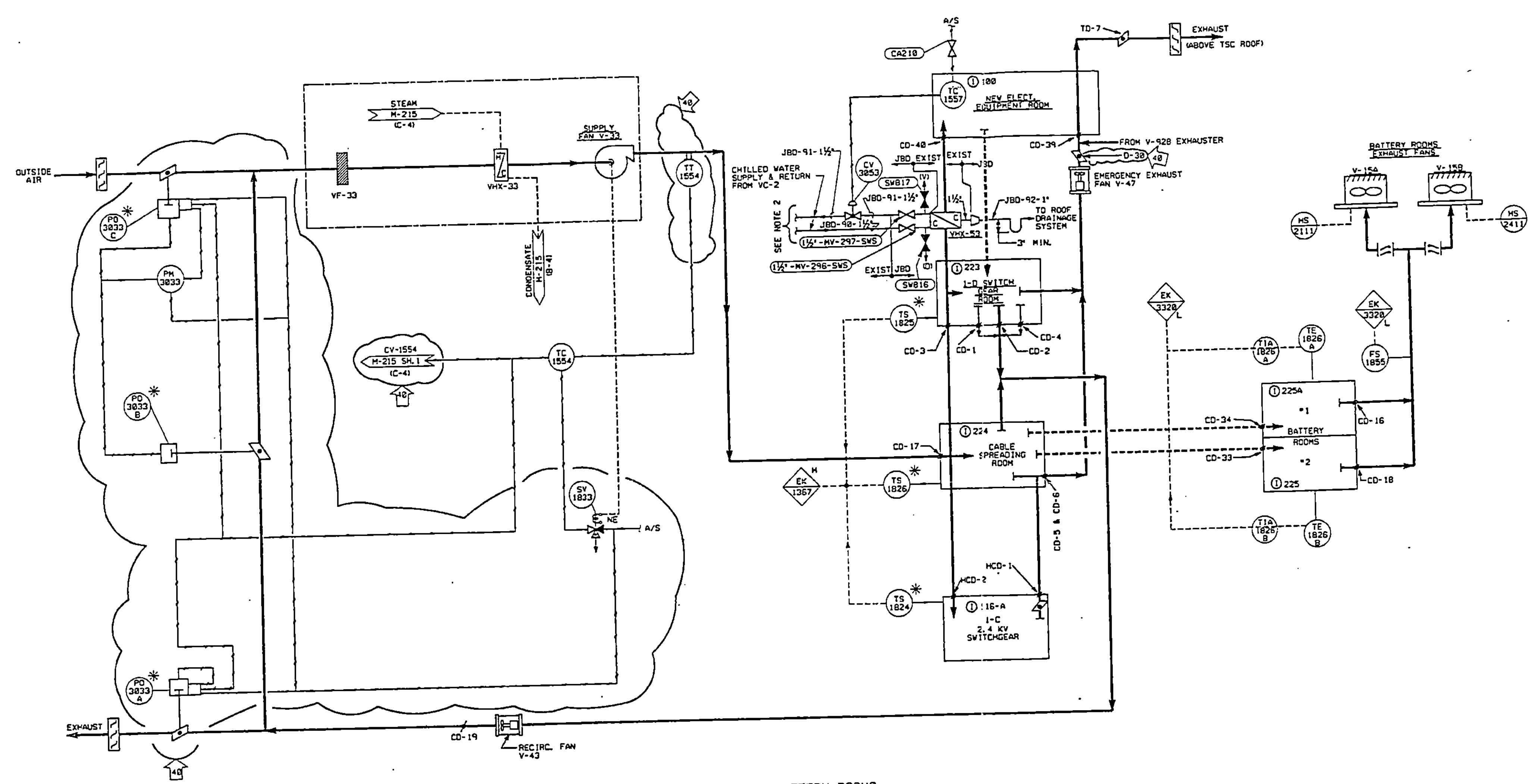
Also Available on Aperture Card

9707070096-19

Figure 9-15
Rev 1?

FIGURE 9-16

Deleted



ELECTRICAL EQUIPMENT, SWITCHGEAR, CABLE SPREADING & BATTERY ROOMS

NOTES:

1. DUCTWORK AND COOLING COIL FOR ELECTRICAL EQUIPMENT ROOM ARE NON-Q.
2. CHILLED WATER SUPPLY AND RETURN PIPING CONNECTION REFER TO M-110-2528.
3. THE FOLLOWING ITEMS PURCHASED UNDER 054 SUB JOB ARE "NON-Q". 1½"-MV-296-SWS AND 1½"-MV-297-SWS.

REV	DATE	DESCRIPTION
40	5-22-96	REV'D OFF PAGE REF. & ADDED PM-3033 PER DCR 96-415.

CONSUMERS POWER COMPANY
PALISADES NUCLEAR PLANT
COVERT, MICHIGAN

PIPING & INSTRUMENT DIAGRAM
HTG. VENT. & AIR CONDITIONING
SWITCHGEAR & CABLE SPREADING RMS.

M218 | 1 | 40

ANSTEC APERTURE CARD

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9707070096 -20

Figure 9-17 Sh 1
Rev 19

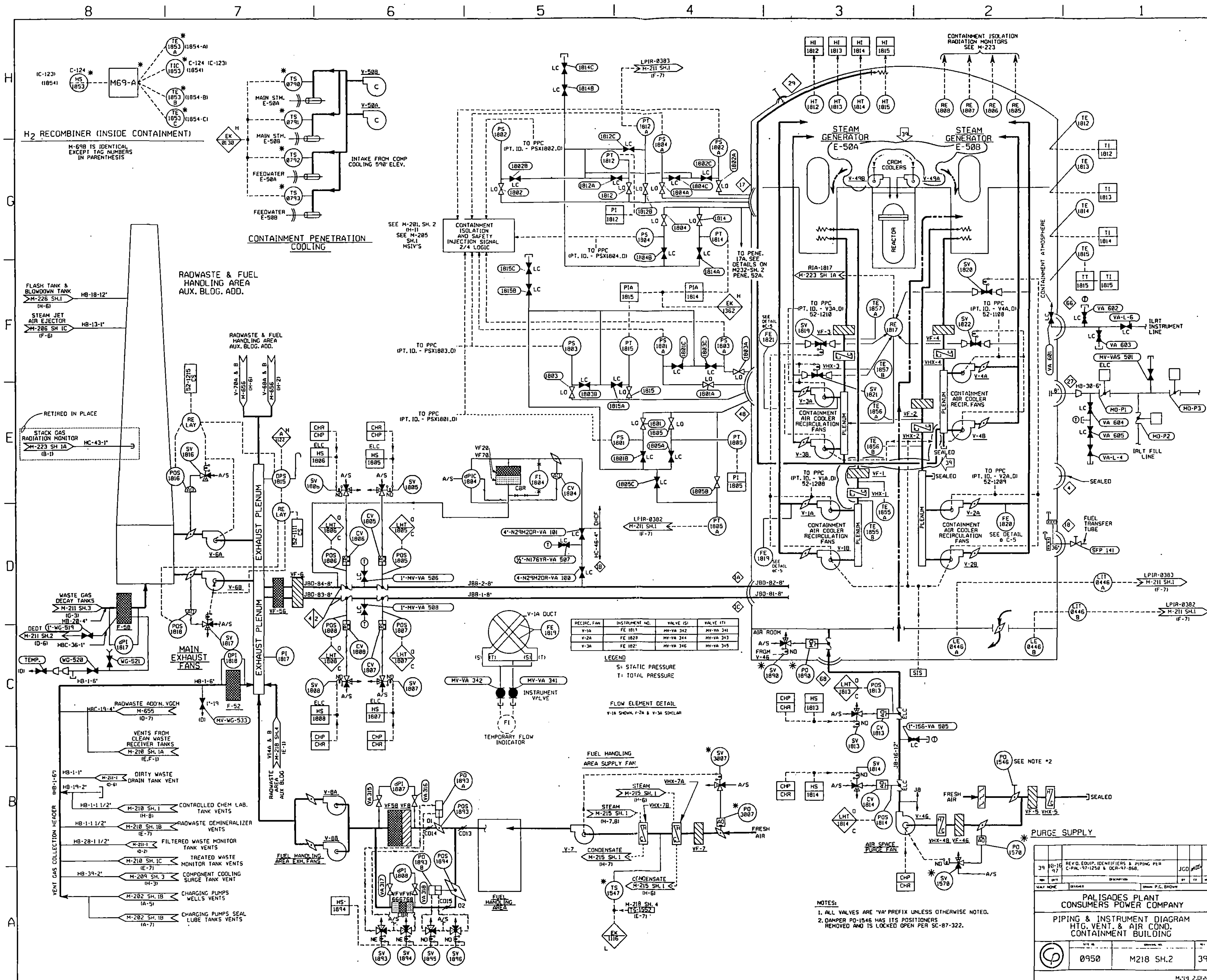

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FIG 9-17 SH 2
FSAR REV 20

9807090043 -45

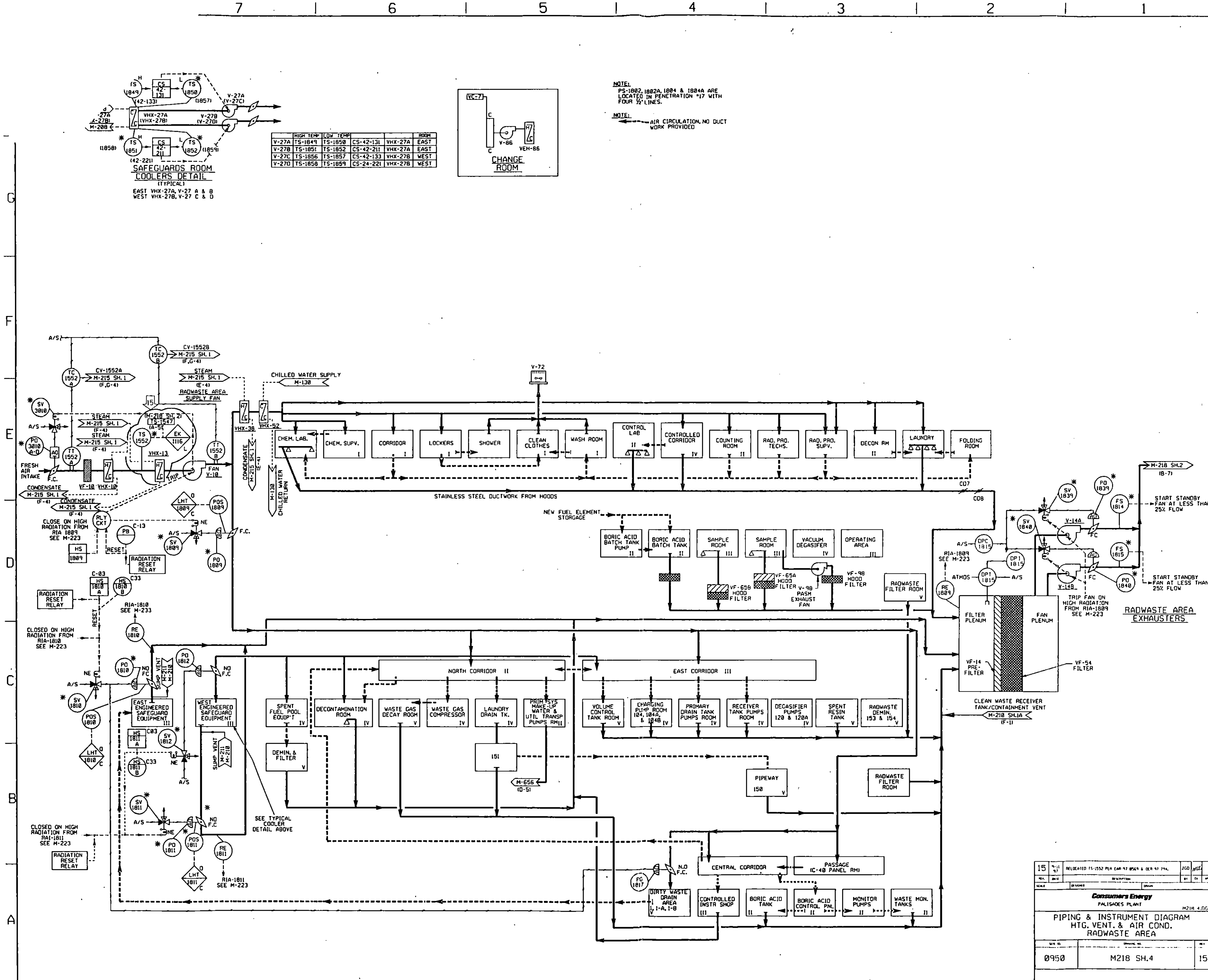


FIG 9-17 SH 4
FSAR REV 20

9807090043 -46

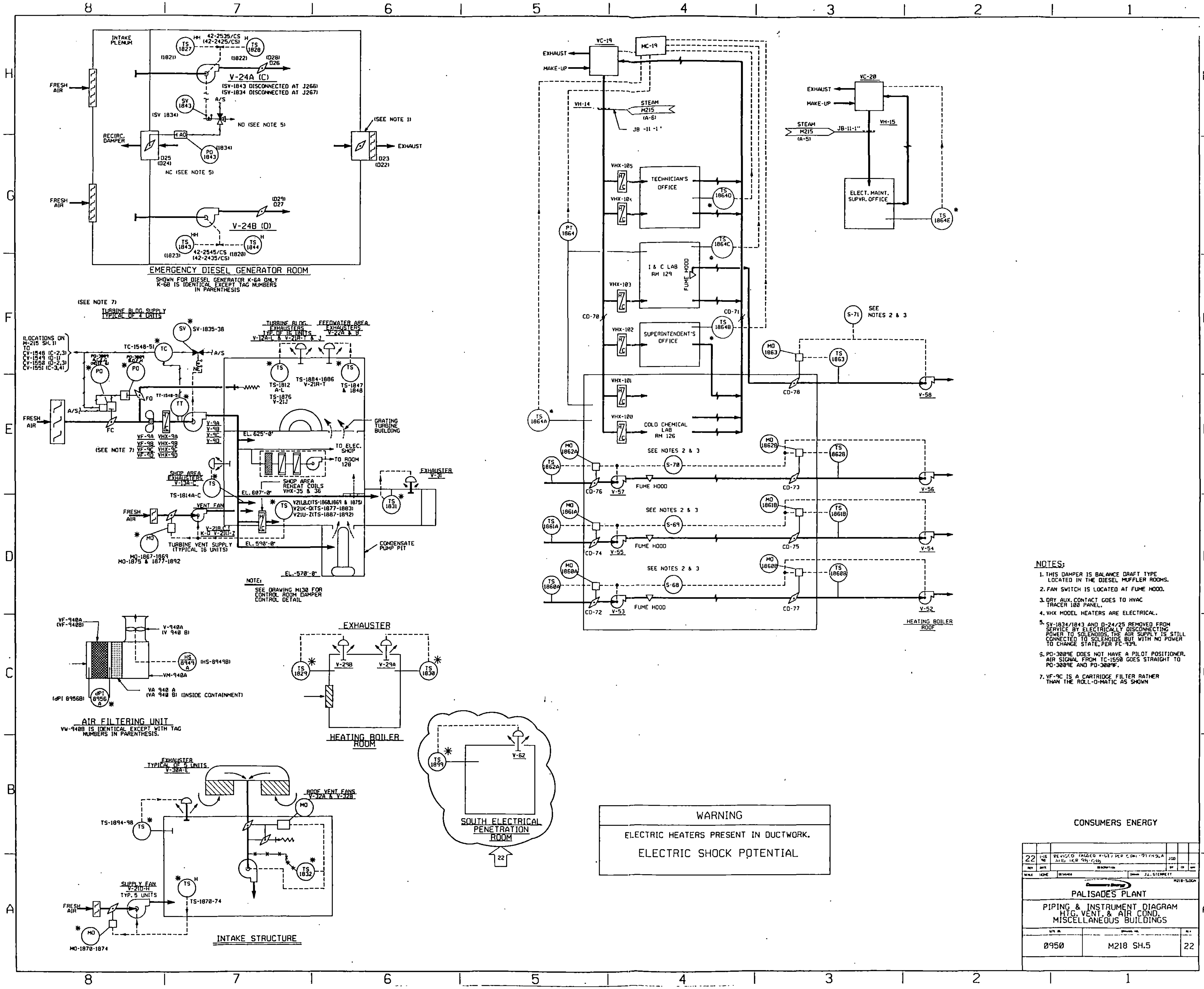


FIG 9-17 SH 5
FSAR REV 20

9807090043 -47

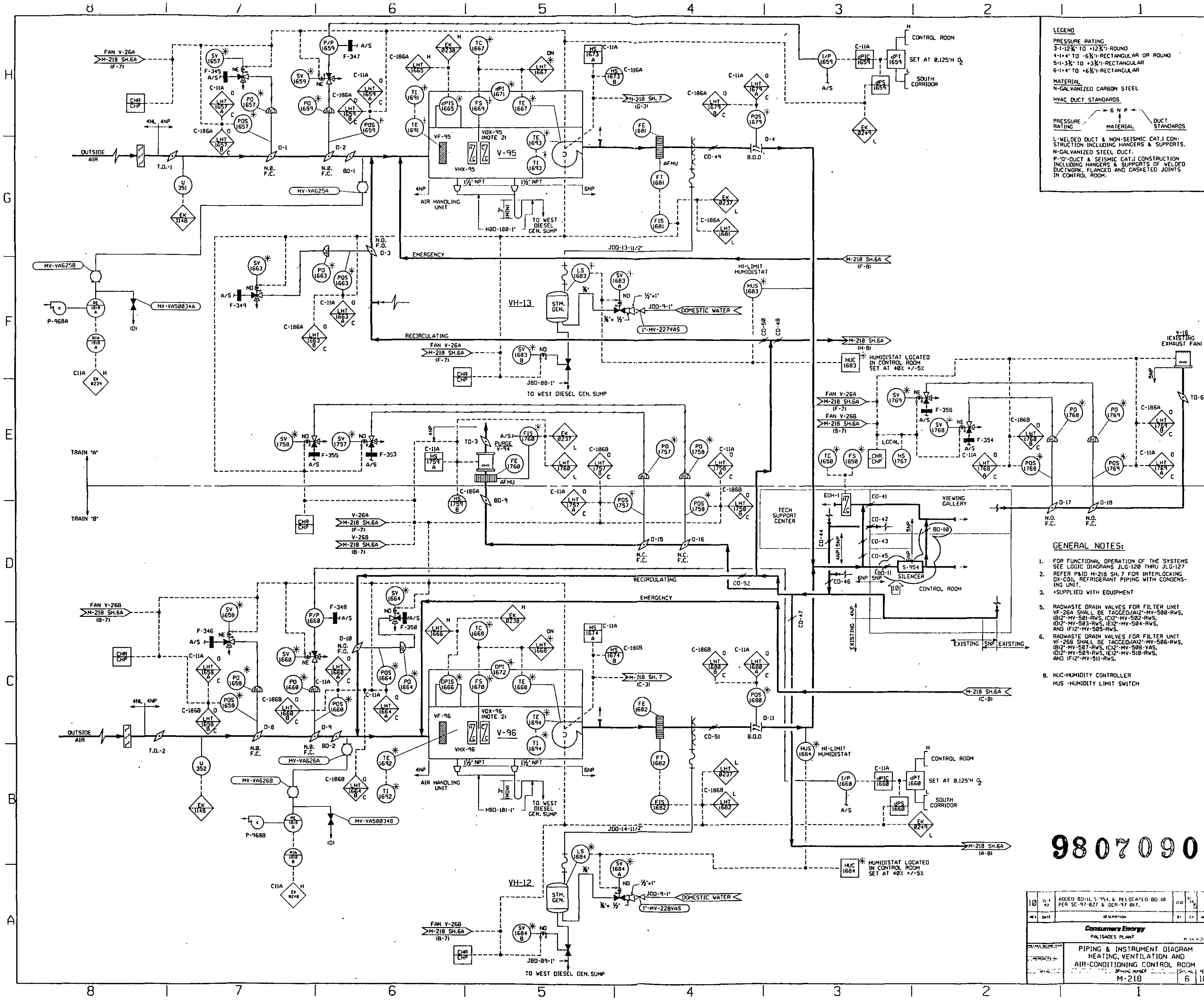
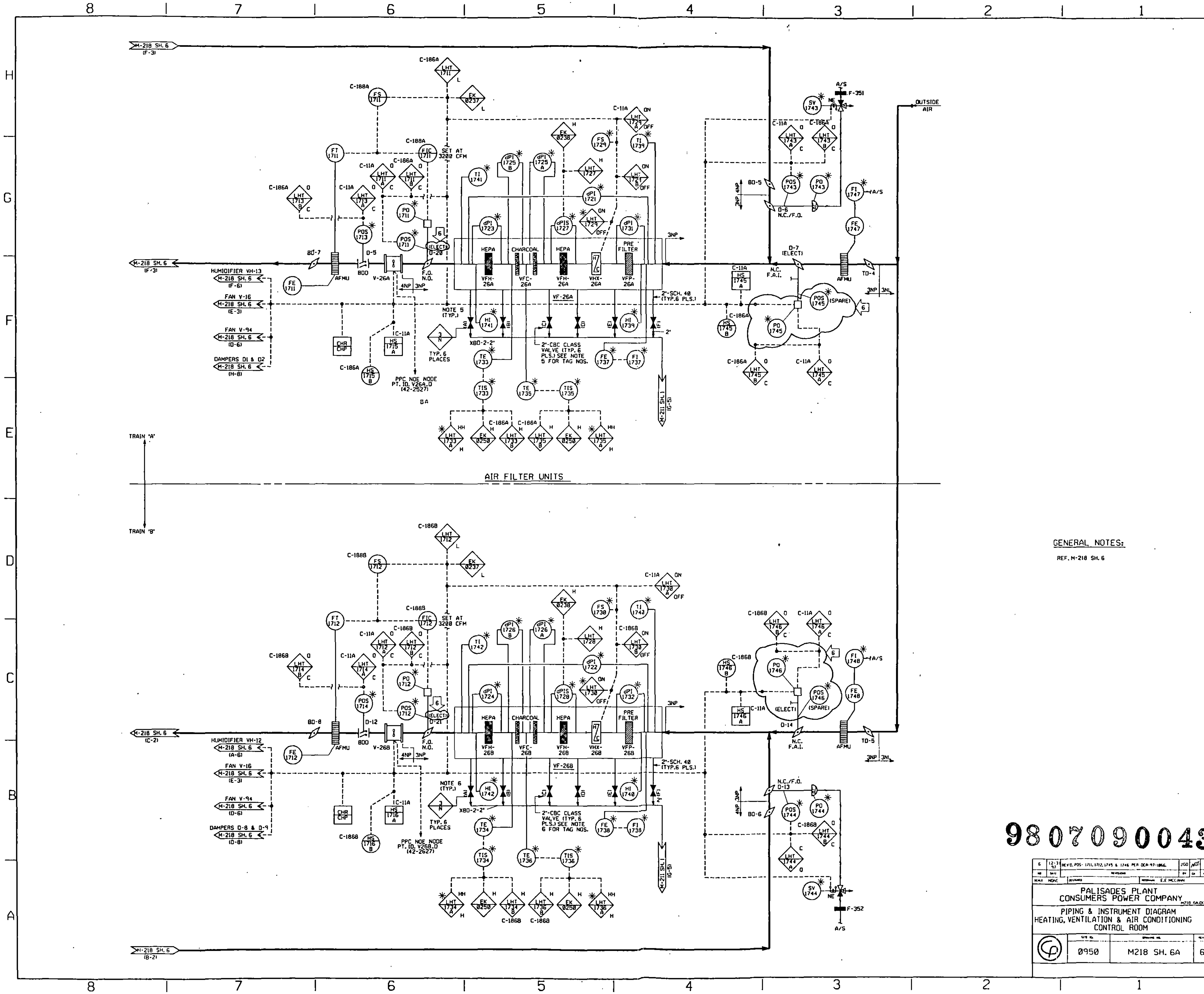


FIG 9-17 Sh 6
FSAR REV 20



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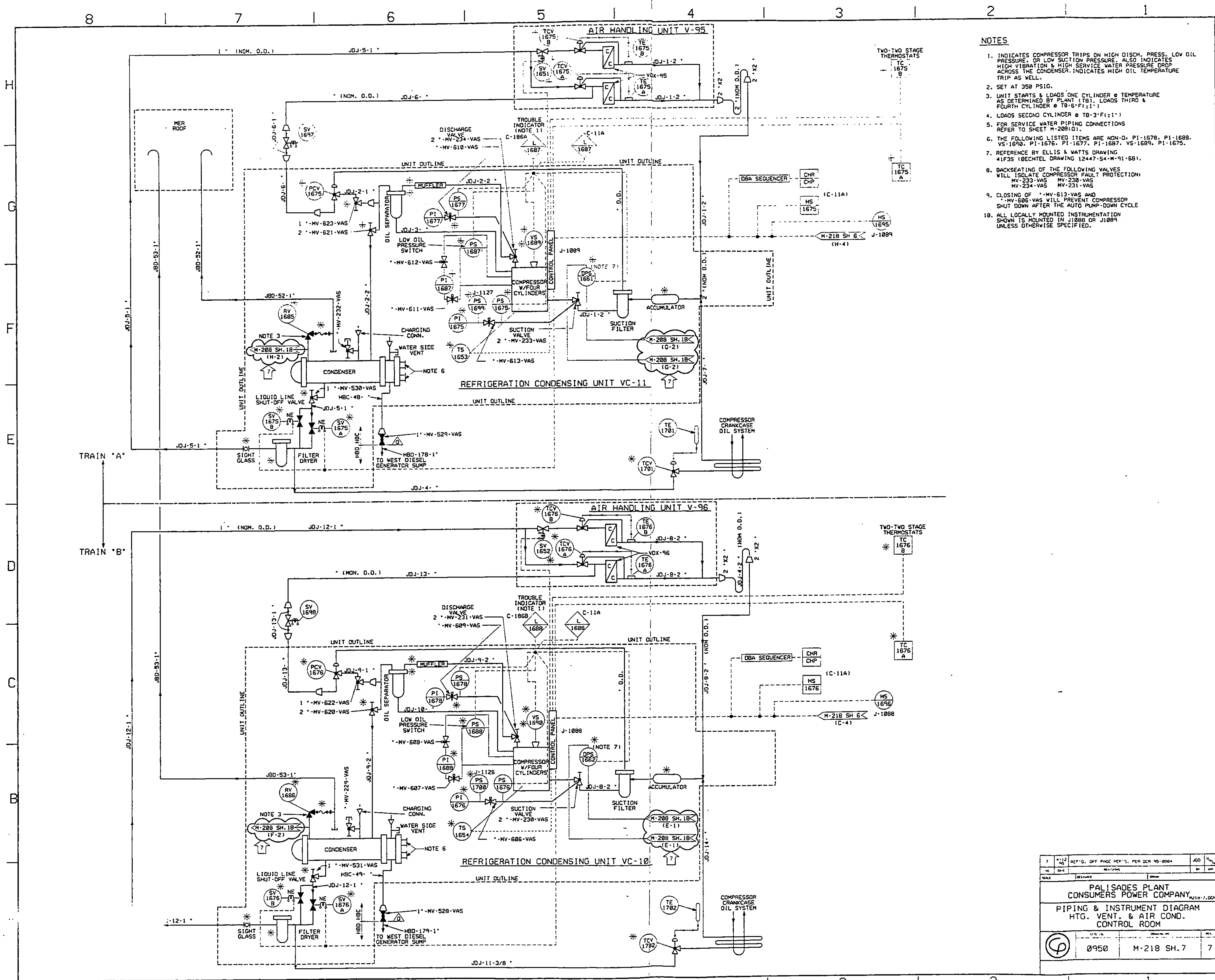
Also Available on Aperture Card

9807090043 -49

FIG 9-17 Sh 6A
FSAR REV 20

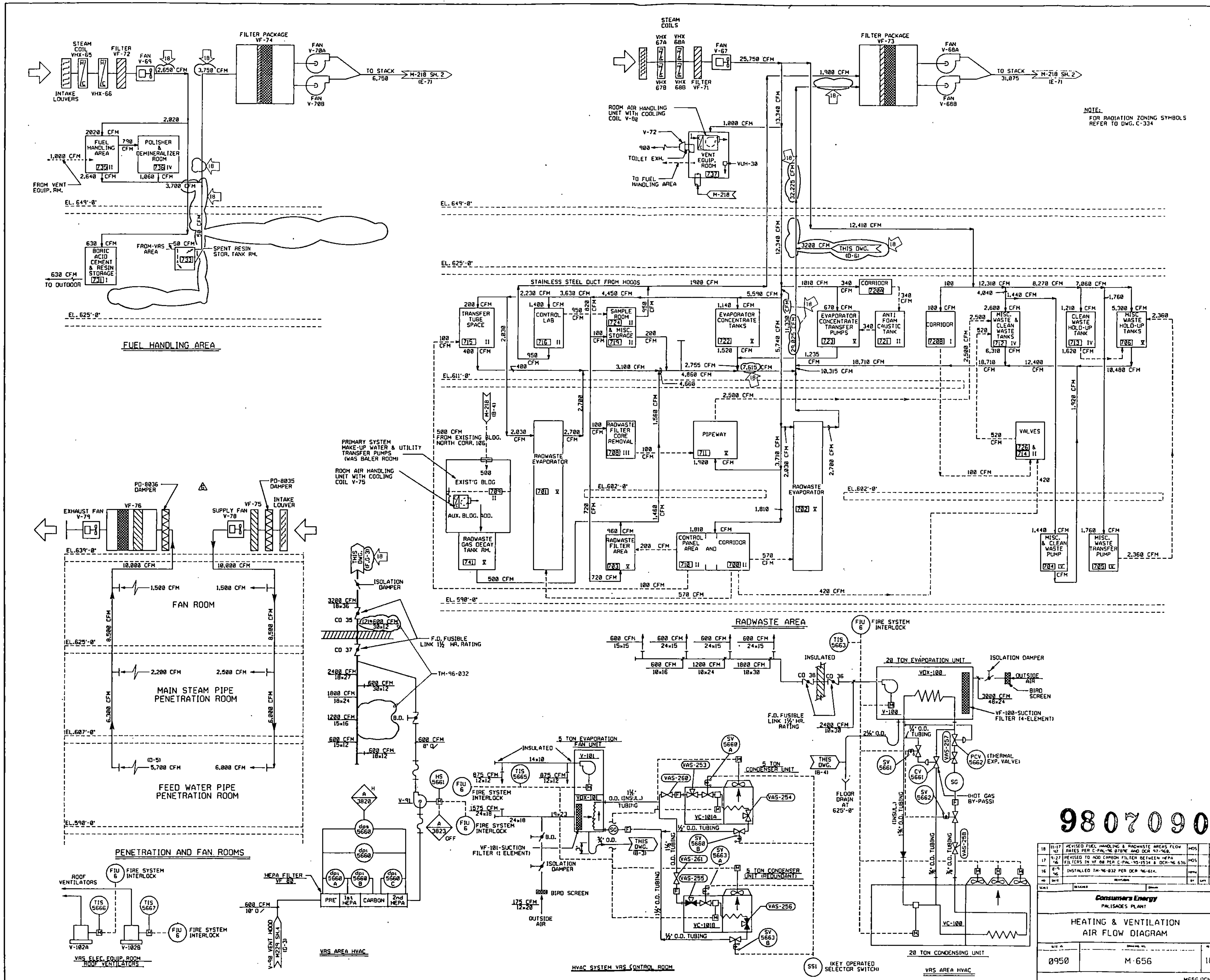


ANSTEC APERTURE CARD

Also Available on Aperture Card

Figure 9-17 Sh 7 FSAR Rev 18

9602290093-32

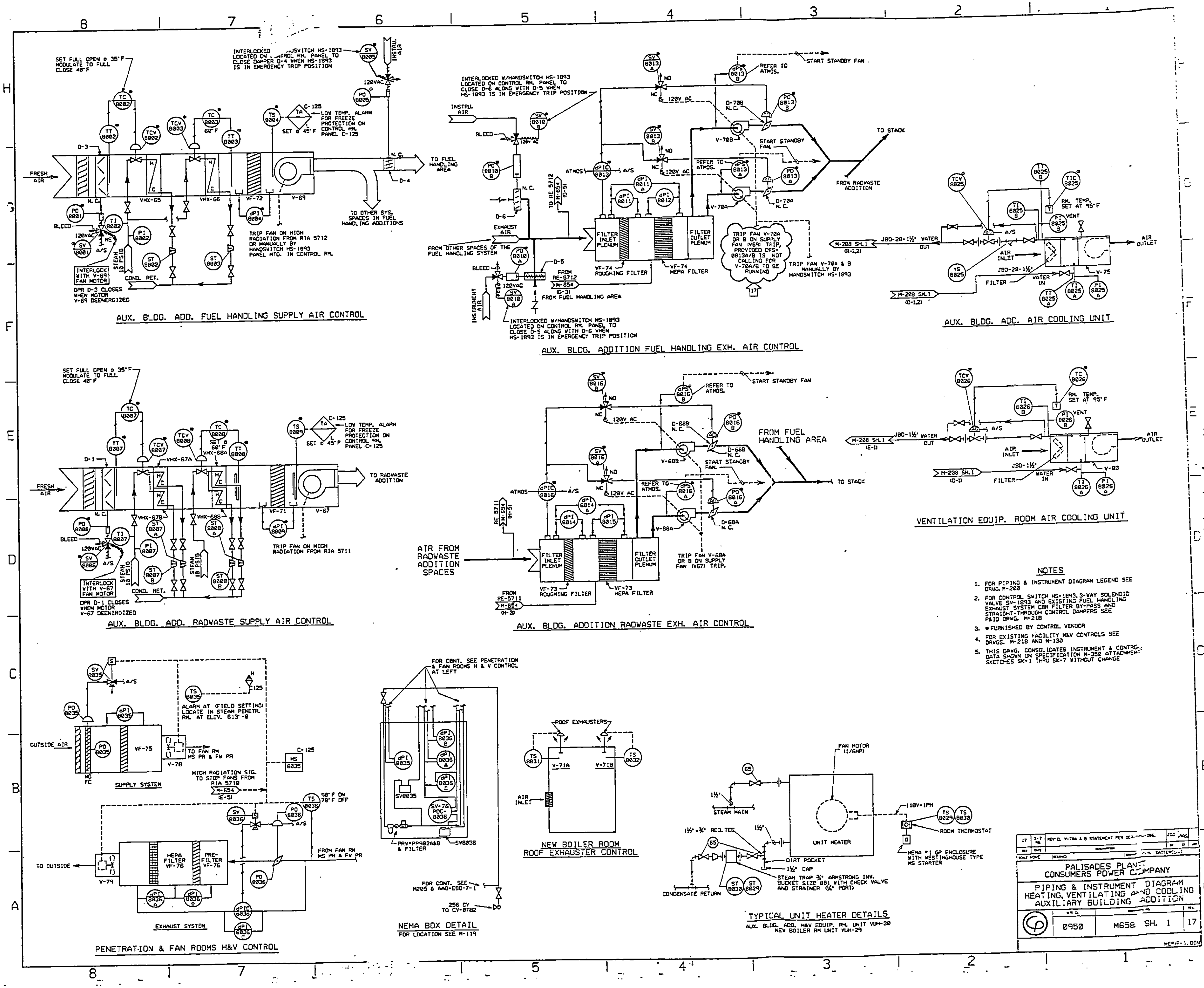


ANSTEC APERTURE CARD

Also Available on Aperture Card

9807090043 -50

FIG 9-18
FSAR REV 20

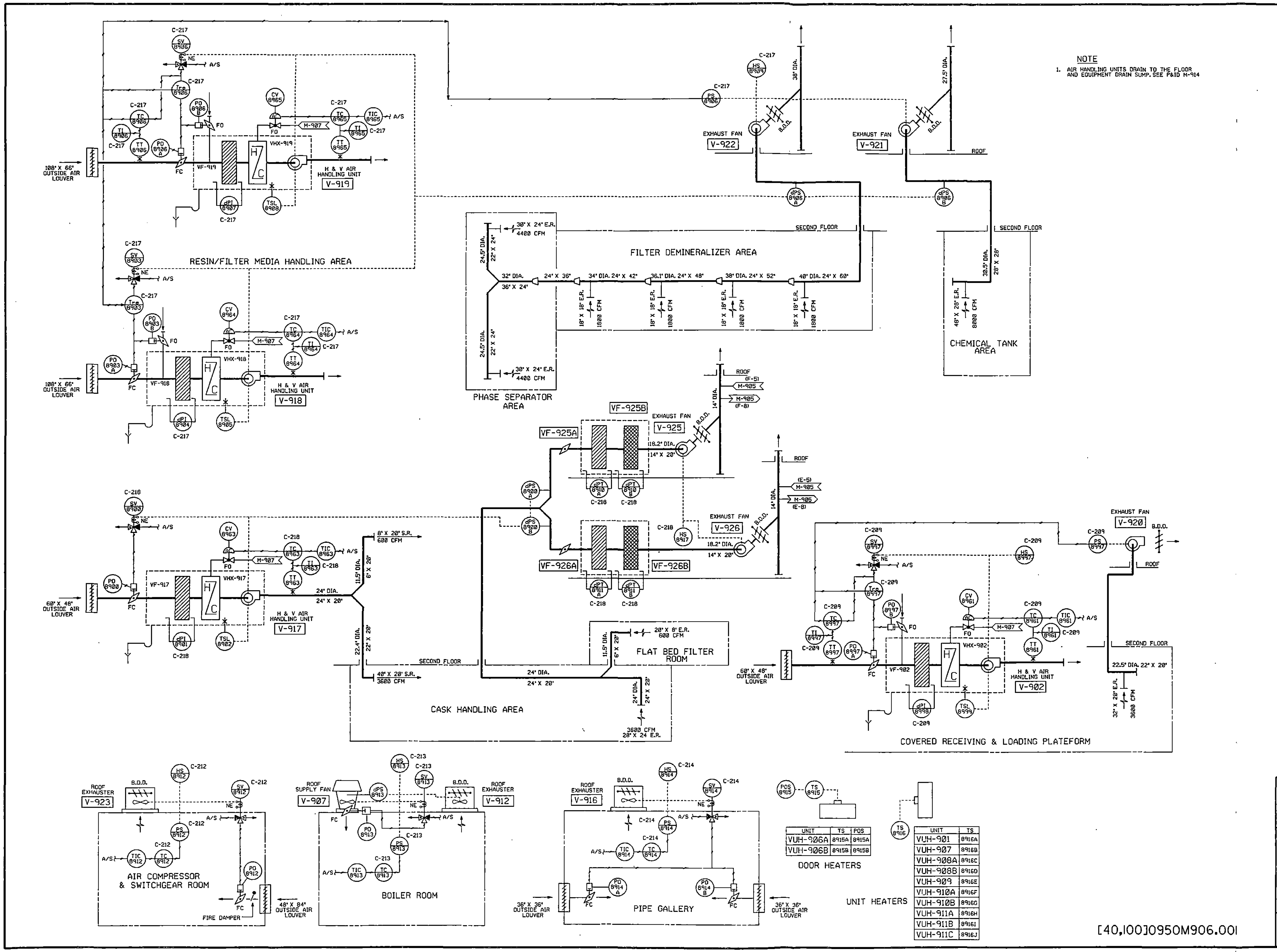


ANSTEC APERTURE CARD

Also Available on Aperture Card

9707070096-28

Figure 9-1
Rev 1



UNIT	TS	POS
VUH-906A	8915A	8915A
VUH-906B	8915B	8915B

DOOR HEATERS

UNIT HEATERS

UNIT	TS
VUH-901	8916A
VUH-907	8916B
VUH-908A	8916C
VUH-908B	8916D
VUH-909	8916E
VUH-910A	8916F
VUH-910B	8916G
VUH-911A	8916H
VUH-911B	8916I
VUH-911C	8916J

TI APERTURE CARD

Also Available On Aperture Card

(REF M-906 SH. 1 REV. 7)

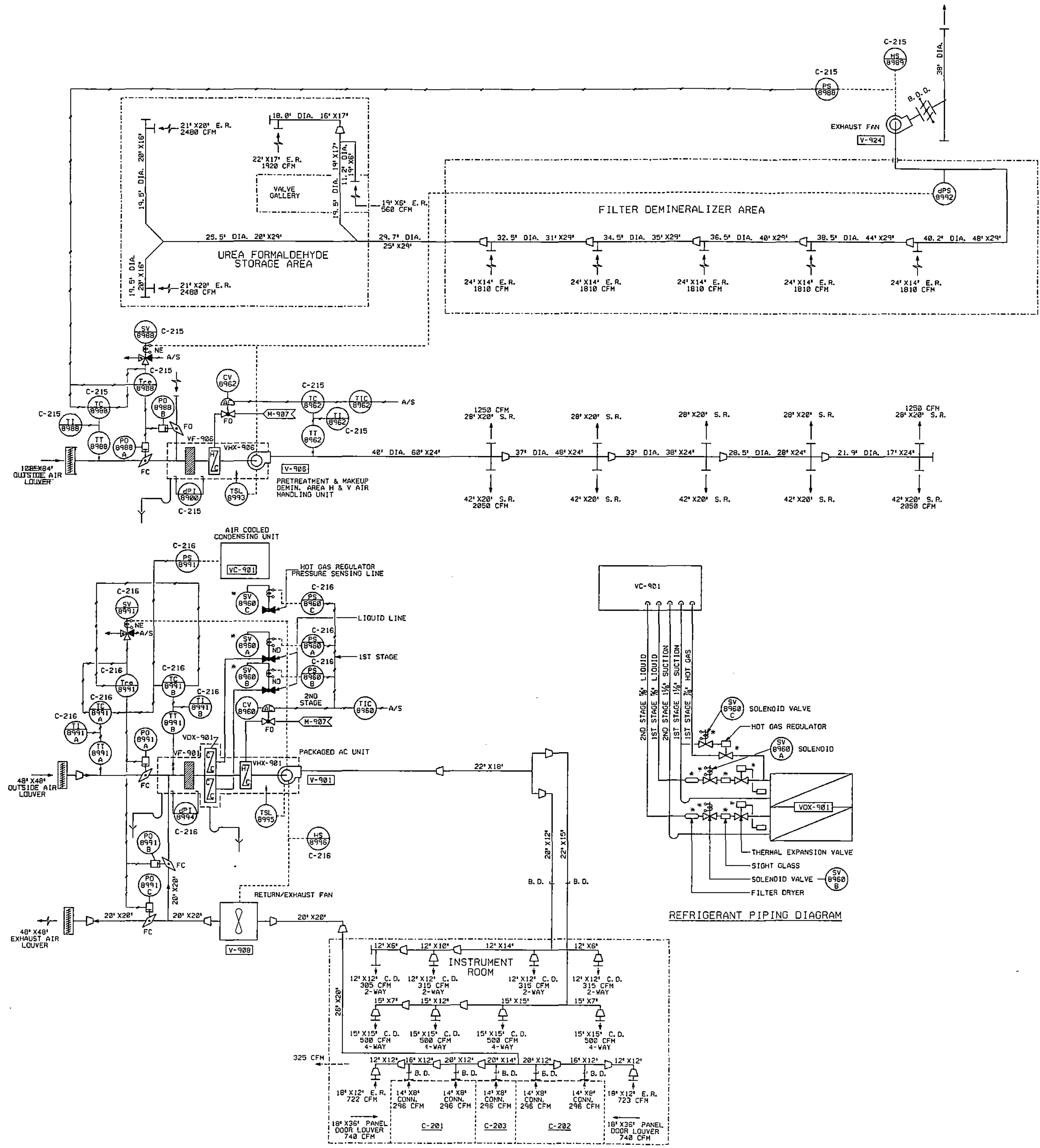
CONSUMERS POWER COMPANY
PALISADES PLANT
FSAR UPDATE

P & I DIAGRAM HVAC SYSTEM
CONDENSATE & MAKEUP DEMIN. BLDG.

FIGURE NO 9-20 SH. 1

REVISION NO 0

8407050287-29



NOTES

1. REVISIONS 0 THRU 6 WERE APPROVED BY SHEET 1 OF 2.

TI APERTURE CARD

Also Available On Aperture Card

(REF M-906 SH.2 REV. 7)

CONSUMERS POWER COMPANY
PALISADES PLANT
FSAR UPDATE

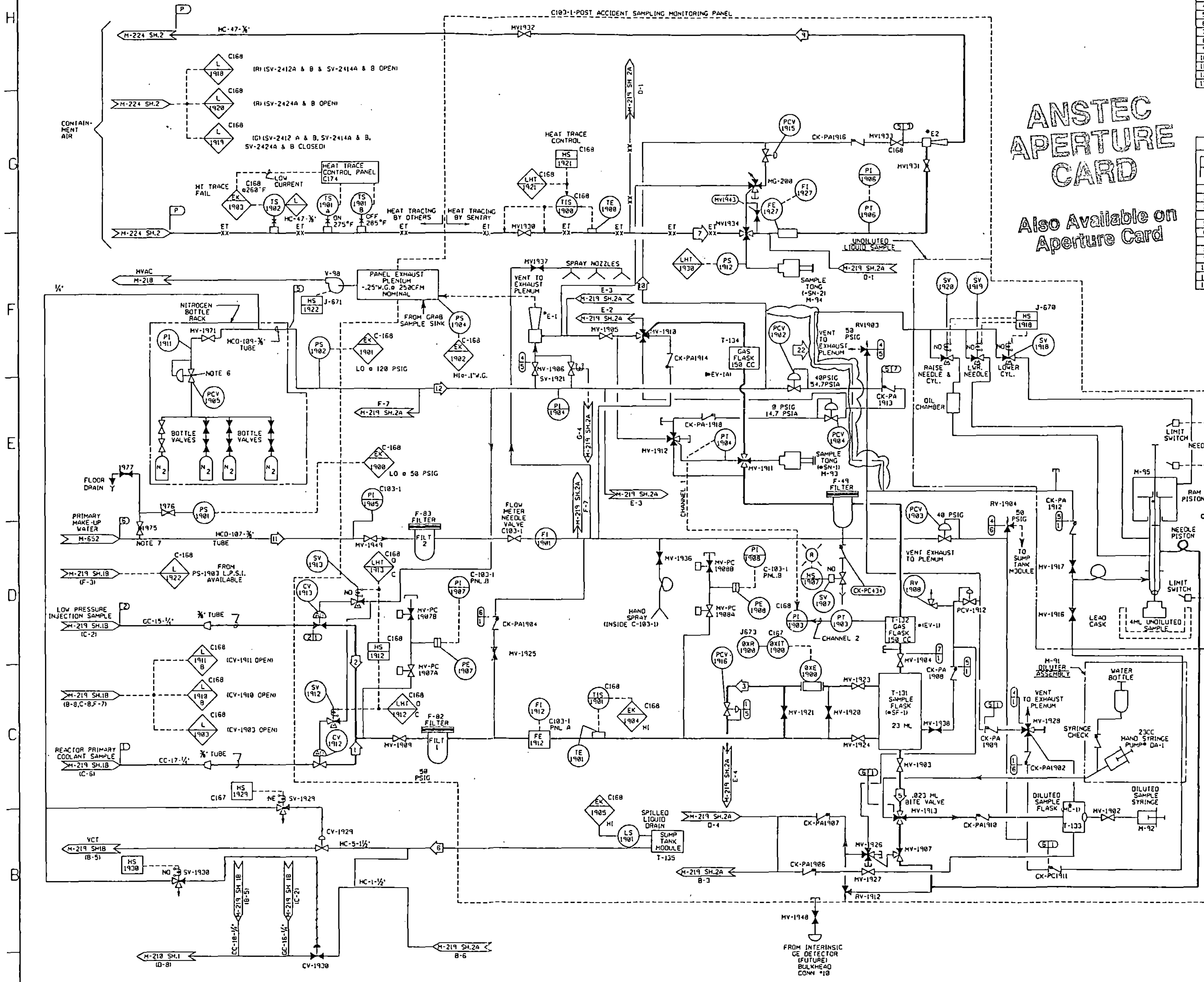
P & IDIAGRAM HVAC SYSTEM
CONDENSATE & MAKEUP DEMIN. BLDG.

[40,100]0950M906.002

FIGURE NO 9-20 SH.2

REVISION NO 0

8407050287-30



OPERATING DATA							
⇨	FLOW (CC/MIN)	PRESS (PSIG)	TEMP (F)	BY	CHKD	REMARKS	RE.
1	200	2085	105			PURGE FLOW 1400.	
2	200	160	105			PURGE FLOW 1400	
3	200	10	105			PURGE FLOW 1400	
4	40	10	105				
5	200	2085	105				
6	1400	0 TO 50	105				
7	500	-1	290			PURGE FLOW 5000CC/M	
8	25	-1	104				
9	28800	-1	104			PERGE FLOW 3300CC/M	
10	28300	100	80				
11	1400	60	150			PURGE FLOW 2500CC/M	
12	29000	120	80				
13	1400	0 TO 50	105				

DESIGN DATA								
#	NORMAL		TRANSIENT		BY	CHKD	REMARKS	REV
	PRESS PSIG	TEMP °F	PRESS PSIG	TEMP °F				
1	2485	650					SEE NOTE 7	
2	500	350					SEE NOTE 7	
3	55	300	240					
4	-.25	100						
5	150	100						
6	150	150						
7	150	300						
8	50	300						
9	0	100						
10								
11								
12								

ANSTEC APERTURE CARD

Also Available on Aperture Card

REFERENCE:

CONSUMERS POWER COMPANY
PALISADES PLANT

1. PIPING & INSTRUMENT DIAGRAM PROCESS SAMPLING SYSTEM M-219,SH.1
2. PIPING & INSTRUMENT DIAGRAM PRIMARY SYSTEM MAKE-UP WATER UTILITY WATER AND OILY WASTE SYSTEM M-652
3. PIPING & INSTRUMENT DIAGRAM GAS ANALYZING SYSTEM M-224,SH.1 & 2
4. PIPING & INSTRUMENT DIAGRAM HEATING, VENTILATION AND AIR CONDITIONING M-218

NOTES:

1. ALL SOLENOID VALVES ARE DESIGNED TO FAIL CLOSED
2. ADDITIONAL INFORMATION CONCERNING COMPONENTS IN THE SYSTEM CAN BE OBTAINED FROM:
   A. SYSTEM DESIGN DESCRIPTION 0330-1-775-51-01
   B. EQUIPMENT LIST, M-250
   C. INSTRUMENT LIST, M-311
   D. PIPING CLASS SUMMARY, M-259
3. DIAGRAM SYMBOLS AND INSTRUMENT LEGENDS ARE SHOWN ON PALISADES PLANT DWG M-200 1 OF 2
4. LIMITS OF NITROGEN SUPPLY PRESSURE ARE 110 AND 140 PSIG. LOW ALARM IS SET FOR 120 PSIG.
5. LIMITS OF DEMINERALIZED WATER SUPPLY PRESSURE ARE 50 AND 100 PSIG. TO CONTROL WASH DOWN SHOWER FLOW RATE, PRESSURE SHOULD BE SET FOR 60 PSIG.
6. INSTRUMENT TAGS MARKED WITH AN (*) ARE SENTRY TAG NUMBERS USED FOR COMPARISON WITH VENDOR DRAWINGS & PANEL GRAPHICS.
7. THE DESIGN TEMPERATURE RATING IS 105°F FROM THE OUTLET OF THE SAMPLE COOLERS PER FC-676.

22	8-19-97	REVISED ROUTING TO SV1920, -1919, -1918 PER FES 97-209 AND DCR 97-752.	MOS	
21	7-9-97	ADDED NO TO SV1907, 1912, 1918, 1919, 1920 & 1930 AND NC TO SV1929 PER DCR 97-636.	MOS	
20	01-12-93	REV. SYS. DESIGNATIONS ON CHECK VLV. PER DCR 93-013	ITT	

SCALE NONE — PG BROWN

M219-2.DGN

Consumers Energy
PALISADES PLANT

PIPING & INSTRUMENT DIAGRAM
FLUID SYSTEM DIAGRAM
POST ACCIDENT SAMPLE MONITOR SYS

0950 | M-219 SH.2 | 22

FIG 9-21
FSAR REV 20

9807090043 -51

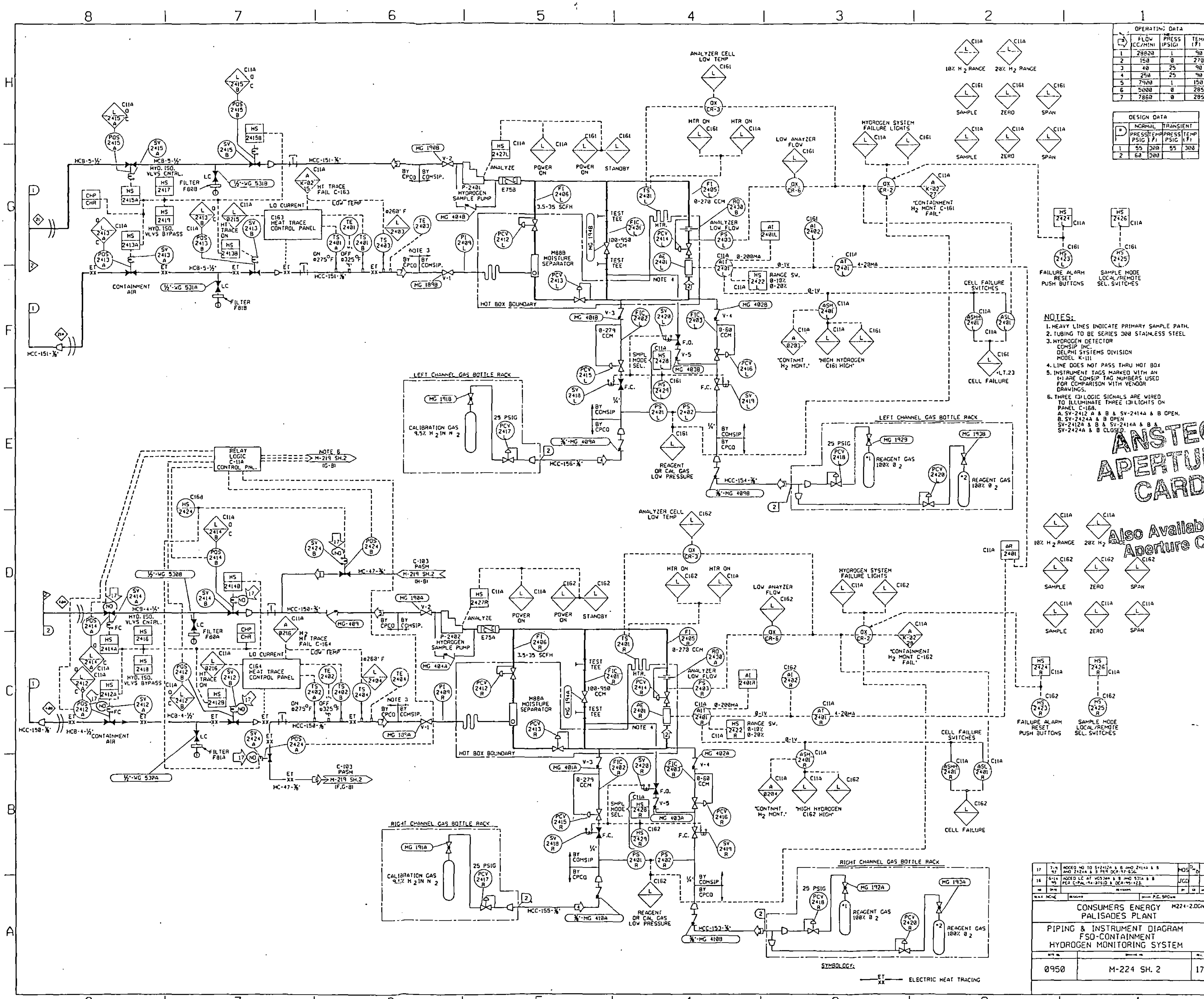
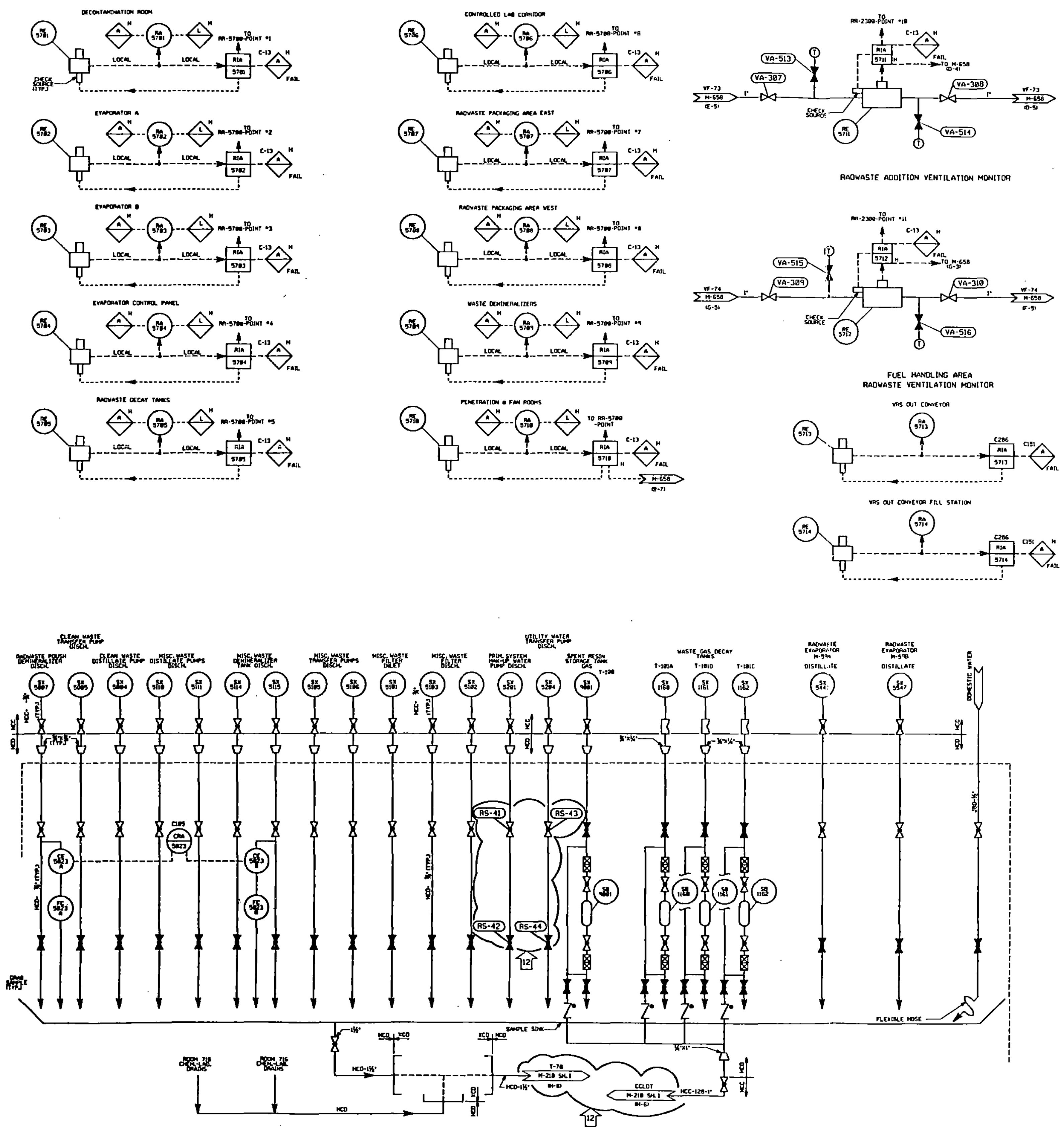


FIG 9-22
FSAR REV 20

9807090043 - 52



9312220060 -28

12	5-17-93	CORRECTED VLV. DESIGNATIONS RS-41-44 AND OFF PAGE REF PER DCR-93-384	RLA

PALISADES PLANT
CONSUMERS POWER COMPANY

RADWASTE ADDITION RADIATION
MONITORING & SAMPLING SYSTEMS

0950 | M-654 | 12

Figure No 9-23 FSAR Rev 1?

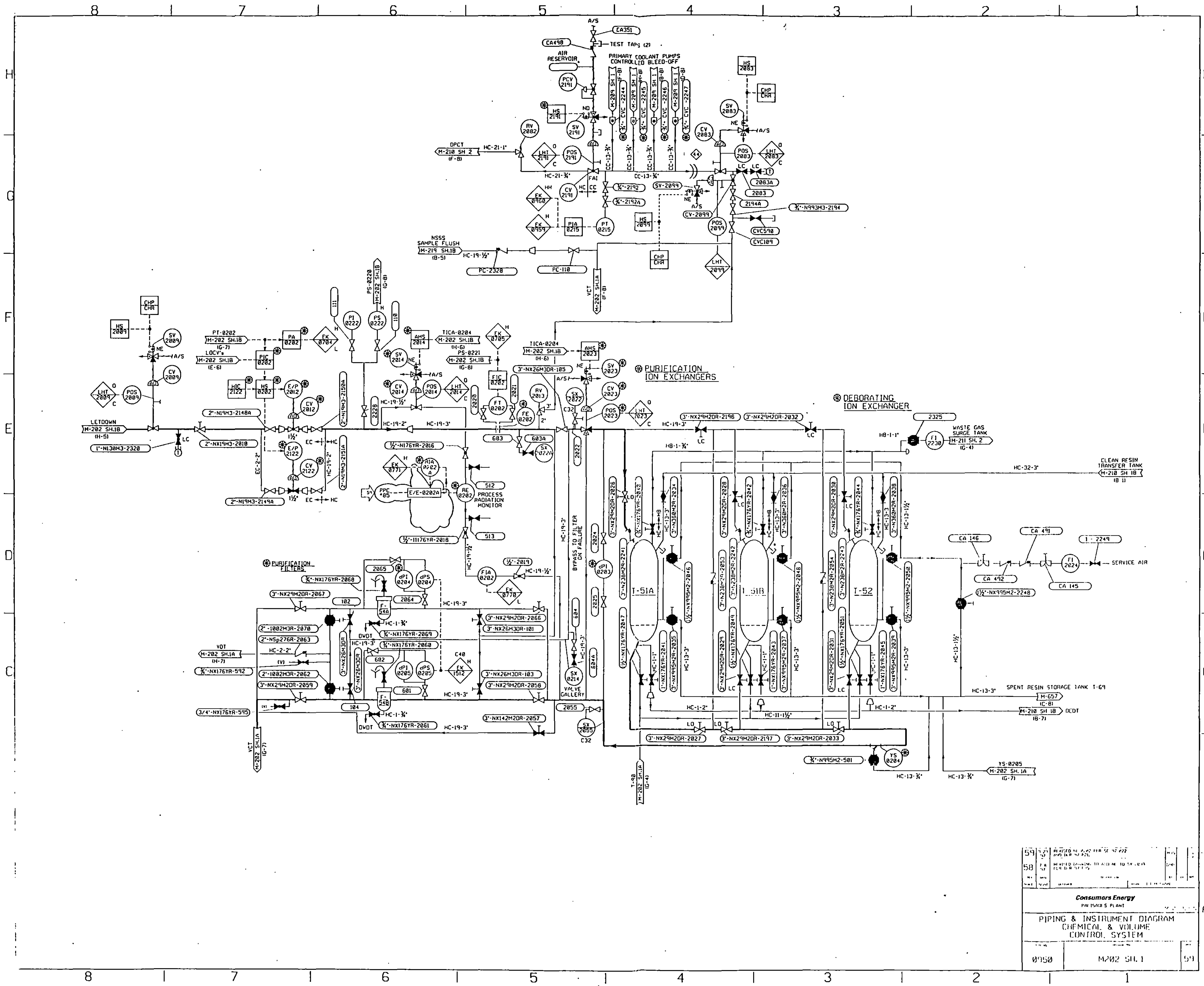
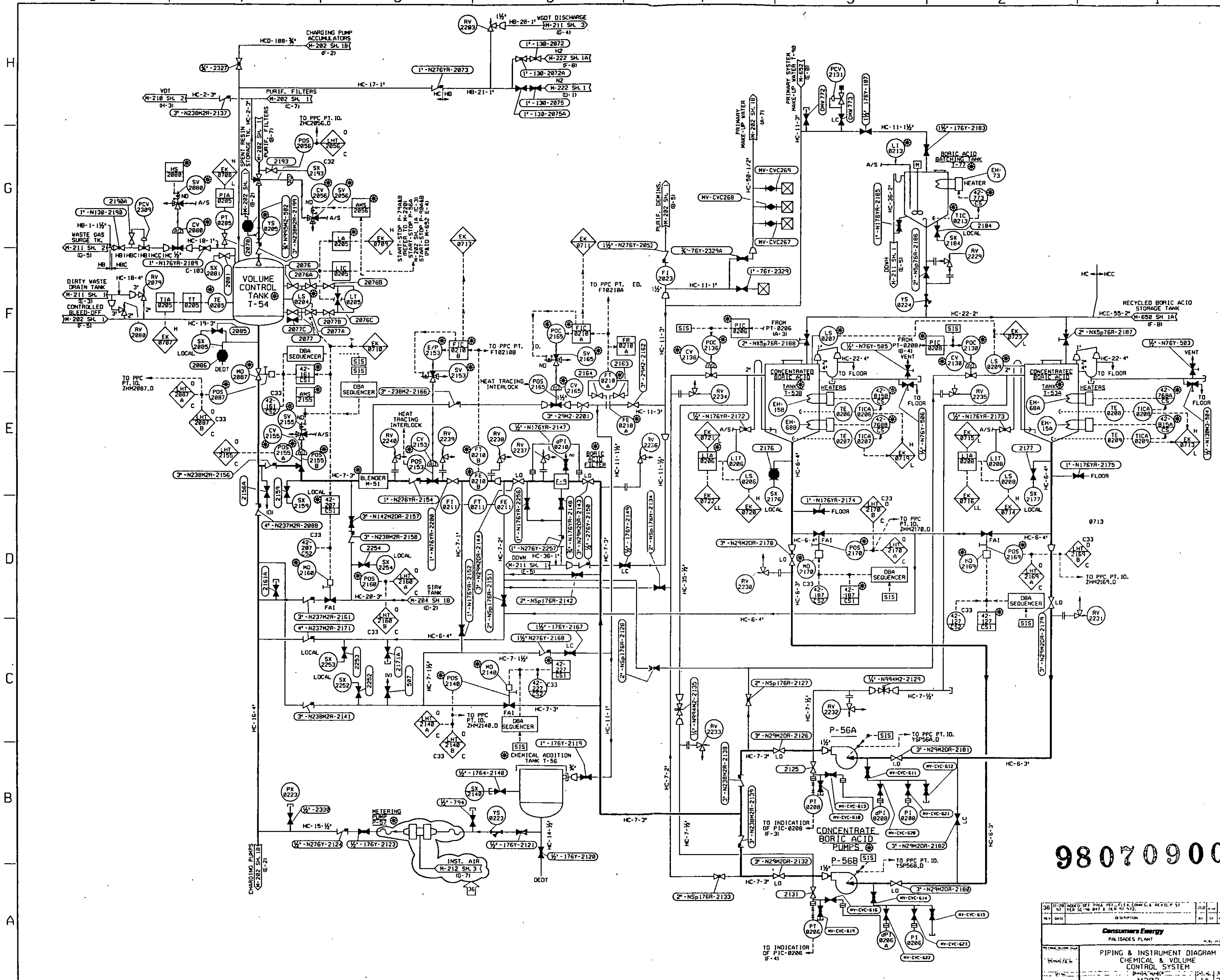


FIG 9-25 SH 1
FSAR REV 20

9807090043 -53

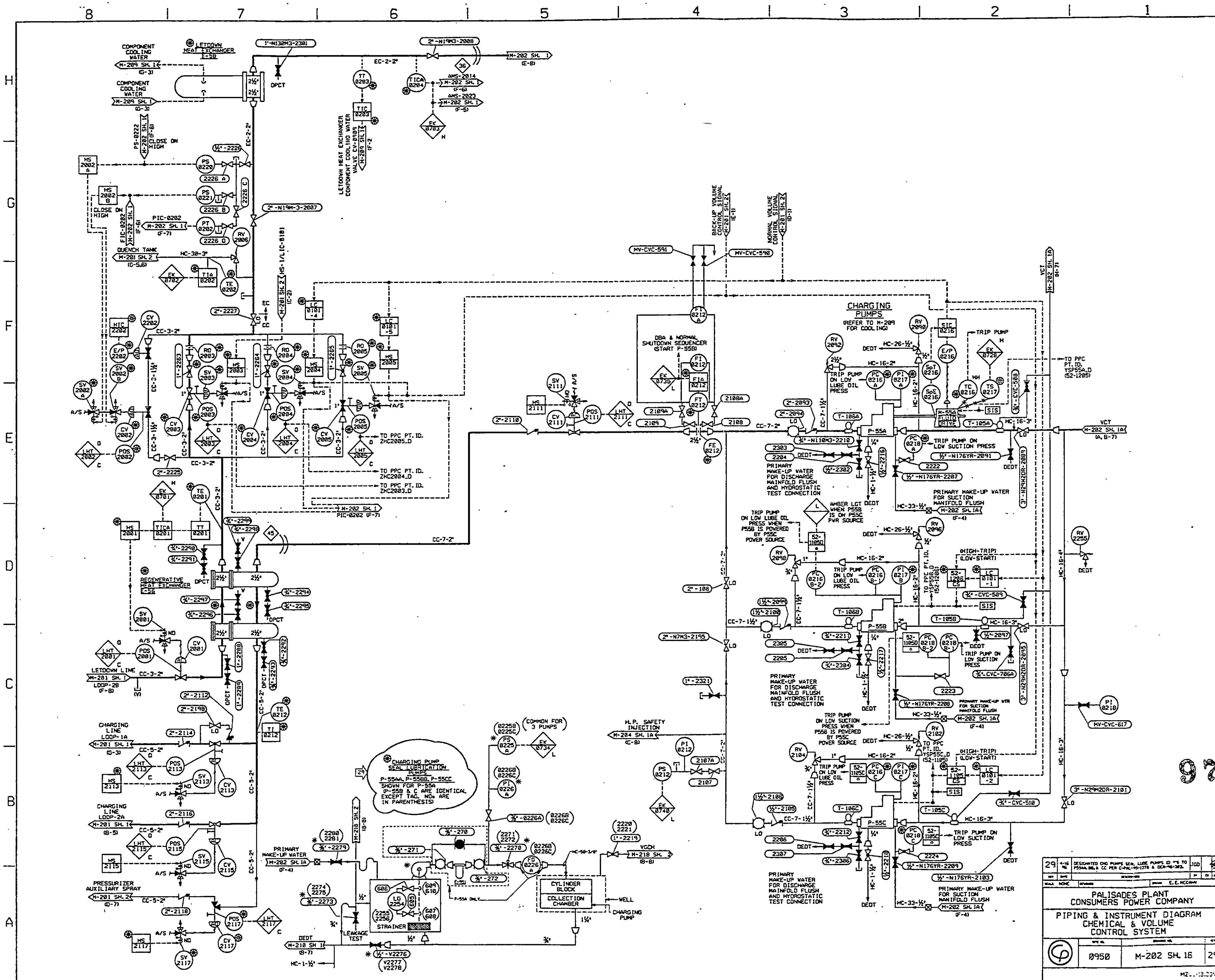


ANSTEC APERTURE CARD

Also Available on Aperture Card

9807090043 -54

FIG 9-25 SH 1A
FSAR REV 20

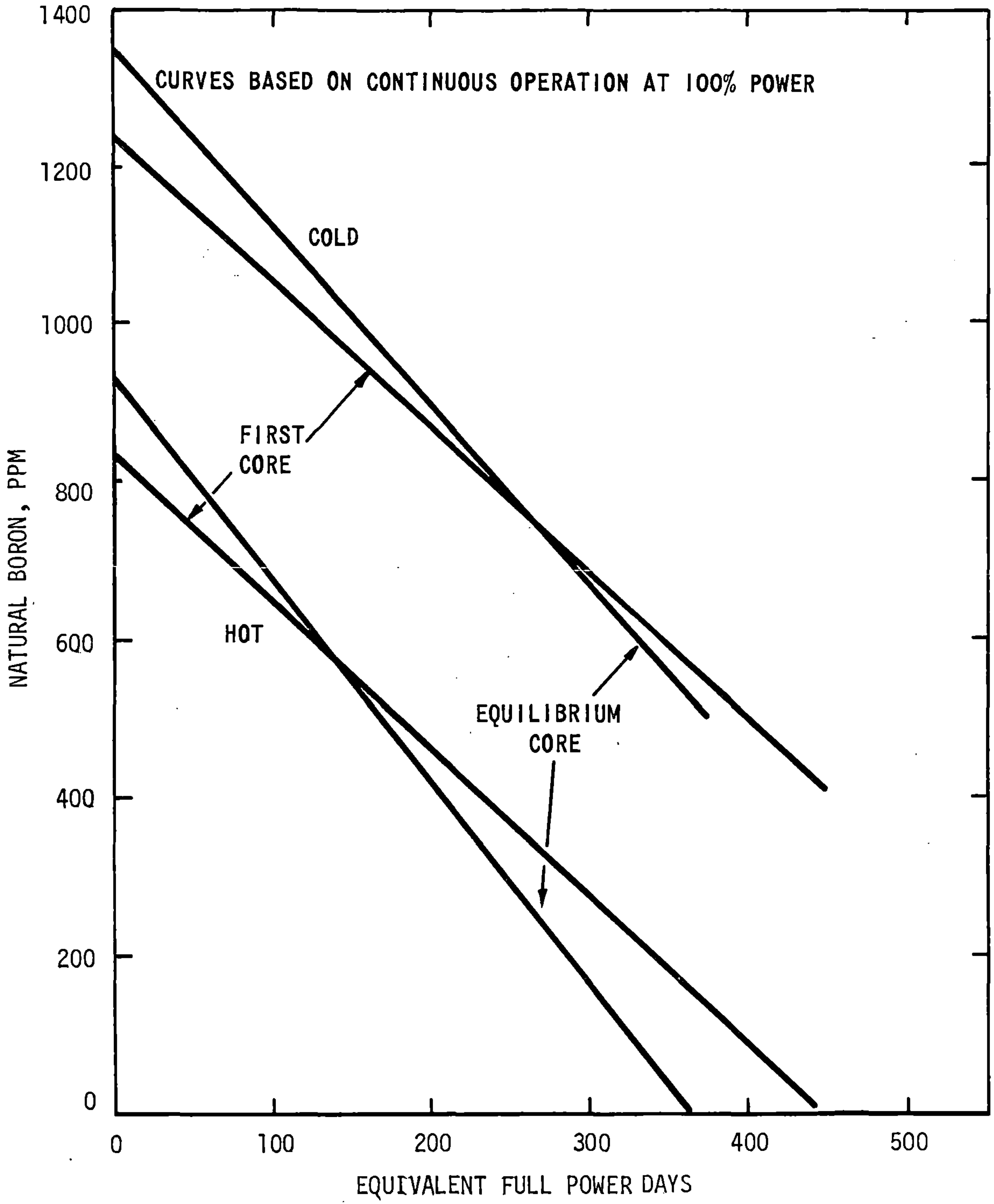


9707070096-38

Figure 9-25 Sh 1B
Rev 19

FIGURE 9-26

(DELETED)



CONSUMERS POWER COMPANY
PALISADES PLANT
FSAR UPDATE

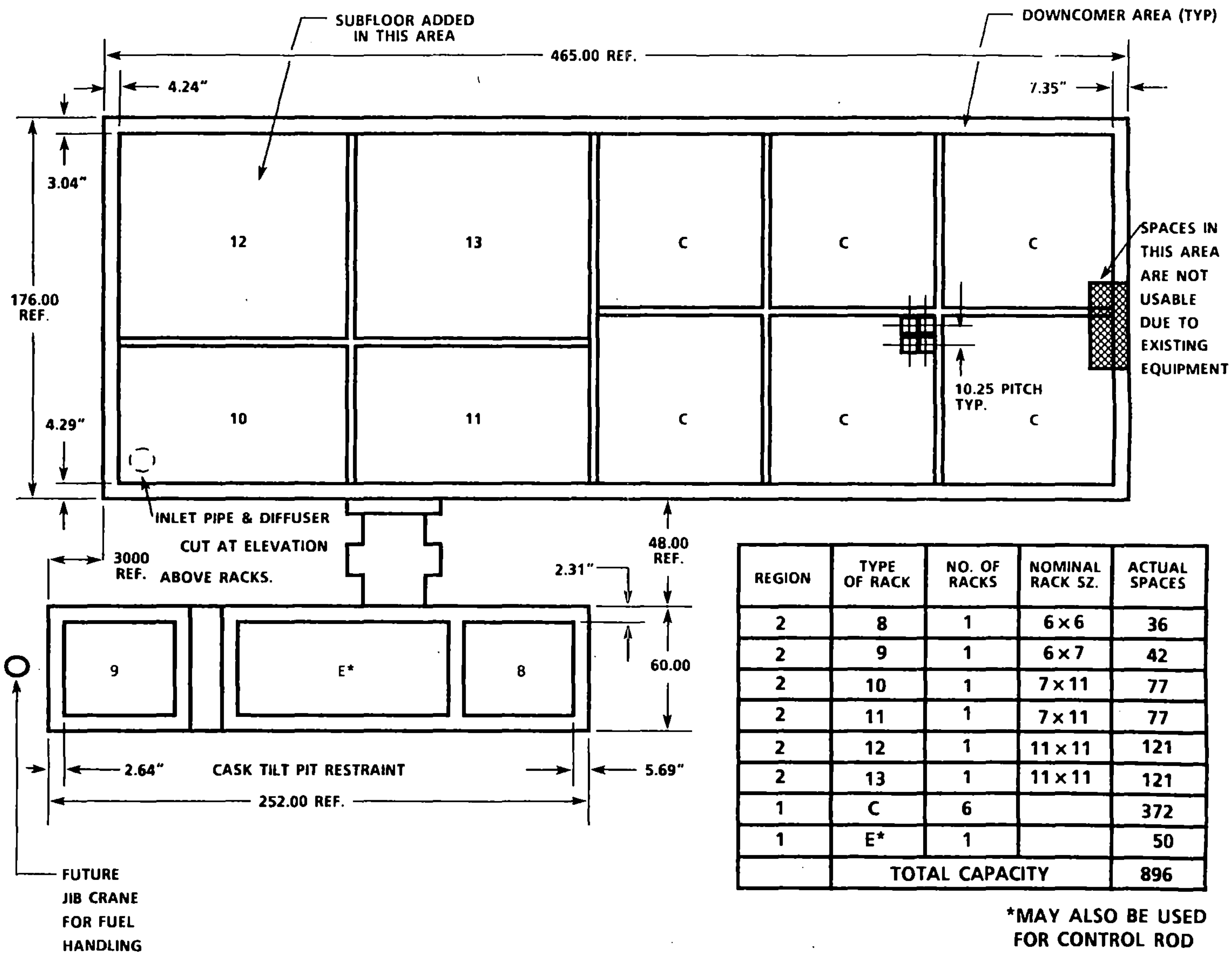
BORON CONCENTRATION

VS

CORE LIFETIME

FIGURE NO 9-27

REVISION NO 0



REGION	TYPE OF RACK	NO. OF RACKS	NOMINAL RACK SZ.	ACTUAL SPACES
2	8	1	6 × 6	36
2	9	1	6 × 7	42
2	10	1	7 × 11	77
2	11	1	7 × 11	77
2	12	1	11 × 11	121
2	13	1	11 × 11	121
1	C	6		372
1	E*	1		50
	TOTAL CAPACITY			896

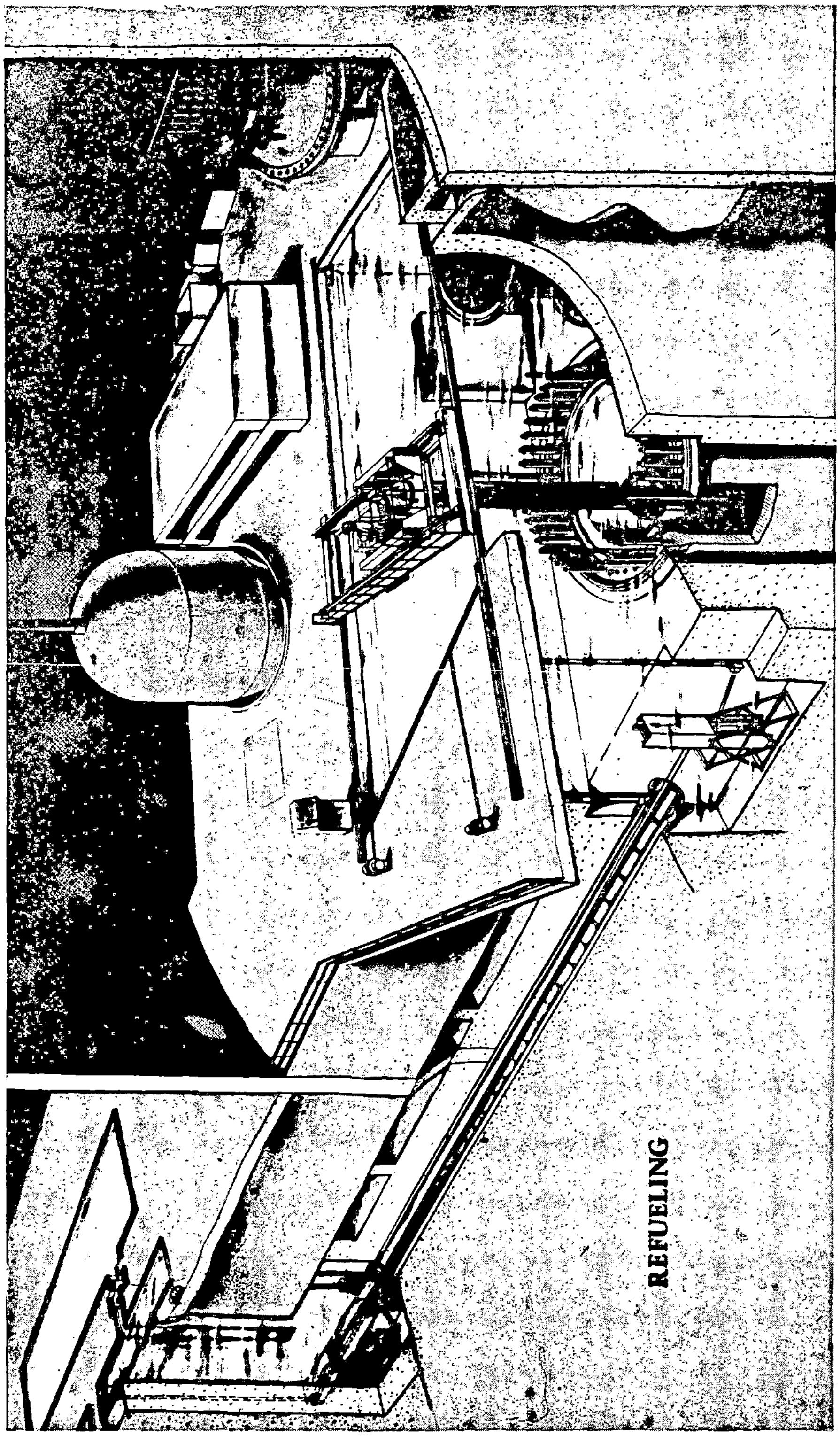
*MAY ALSO BE USED FOR CONTROL ROD STORAGE

CONSUMERS POWER COMPANY
PALISADES PLANT
FSAR UPDATE

PALISADES PLANT SPENT FUEL STORAGE RACK ARRANGEMENT

FIGURE NO. 9-28

REVISION NO. 6



CONSUMERS POWER COMPANY
PALISADES PLANT
FSAR UPDATE

FUEL HANDLING EQUIPMENT ARRANGEMENT

FIGURE NO 9-29

REVISION NO 0

FIGURE 9-30

(DELETED)

CHAPTER 10

STEAM AND POWER CONVERSION SYSTEM

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10.3.1	REACTOR AND/OR TURBINE TRIP	10.3-1
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10-3	Stud Mechanical Properties
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10-6	(deleted)
10-7	(deleted)
10-8	Cooling Tower Design Summary
10-9	Cooling Tower Pumps

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10-2	System Diagram, Extractions, Heater Vents and Drain Systems
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10-3 Sh 1A	Piping and Instrument Diagram Steam Generator Blowdown Modification
10-3 Sh 1B	Piping and Instrument Diagram Steam Generator Blowdown Modification
10-4	System Diagram, Feedwater and Condensate System
10-5	DELETED
10-6	System Diagram, Cooling Tower System
10-7	General Assembly, Electrical Generator
10-8	DELETED

CHAPTER 10

STEAM AND POWER CONVERSION SYSTEM

10.1 DESIGN BASIS

The steam and power conversion system is designed to receive steam from the NSSS and convert the steam thermal energy into electrical energy. A closed regenerative cycle condenses the steam from the main turbine and returns the condensate as heated feedwater to the steam generators.

The turbine generator has a maximum expected capacity of 845 MWe when operating with 6 stages of feedwater heating and the steam exhausting to a pressure of 1.8 inches Hg absolute.

Components from the steam generators up to and including the main steam isolation valves and main and auxiliary feedwater header isolation check valves were designed to CP Co Design Class 1 requirements. The main steam piping between the main steam isolation valve and various steam takeoff block valves is designed to CP Co Design Class 2 requirements. The remainder of the components and piping is designed to CP Co Design Class 3 requirements. See Section 5.2 for a discussion of classes.

10.2 SYSTEM DESCRIPTION AND OPERATION

10.2.1 SYSTEM GENERAL DESCRIPTION

The main steam, extraction steam, feedwater, condensate and steam generator blowdown systems are shown on Figures 10-1, 10-2, 10-3 and 10-4.

1. Main Steam System (Figures 10-1 and 10-2)

   Steam generated in the steam generators passes through two 36-inch headers and main steam isolation valves to the turbine stop valves. Each main steam header is provided with 12 spring-loaded safety valves and 2 atmospheric dump valves upstream of the main steam isolation valves (MSIVs). The safety valves discharge to the atmosphere and are in accordance with the requirements of the ASME B&PV Code, Section III. In addition, there is a steam bypass to condenser valve downstream of the MSIVs. The main steam line also supplies steam for the steam jet air ejectors, the heating steam for the reheaters, the secondary steam supply to the steam generator feed pump turbine drivers, and steam supply to the turbine-driven auxiliary feed pump turbine driver.

2. Moisture Separator-Reheaters

   Steam exhausted from the high-pressure turbines goes to the low-pressure turbines by way of four moisture separator-reheaters in which moisture in the wet exhaust steam is removed and drained to the moisture separator drain tank and the steam is reheated by main steam. The primary steam supply to the steam generator feed pump turbine drivers is extracted from the moisture separator-reheaters.

   Cycle steam flow through the moisture separator-reheater units is not controlled and is a function of load with all auxiliaries operating normally. Wet steam from the high-pressure turbine enters the reheater shell via the cycle steam inlet connection and flows through the shell. Moisture is removed from the wet steam when it flows upward through the demister mesh.

   The condensate removed from the wet cycle steam by the mesh is collected in the lower connection and drained to the moisture separator reheater drain tank.

   The dry cycle steam then passes over the finned tubes of the reheater bundle where it is superheated. This superheated steam continues upward and out the reheater through the cycle steam outlet connection.

Heating steam taken from upstream of the turbine throttle valve enters the upper portion of the tube bundle hemispherical channel head via the heating steam inlet connection. This steam circulates through the U-tubes, relinquishing its heat to the cycle steam flowing over the tubes before the majority of the heating steam exits the lower section of the channel head as liquid through the heating steam condensate outlet connection.

In 1983 a modification was made that installed a vent chamber over the bottom six rows of the tubes in the upper half of the tube sheet. This modification forces the heating steam entering the inlet chamber to flow through only the outer radius tubes. The condensate formed in these tubes drains from the lower section of the hemispherical head through the heating steam condensate outlet connection while the excess scavenging steam reverses direction and flows into the bottom leg of the tubes covered by the vent chamber. The condensate and remaining excess steam which exit from this third and fourth pass are collected by the vent chamber and exit the tube bundle hemispherical head through the scavenging steam vent condenser discharge connection. The lower chamber of the hemispherical head is not vented.

An elongated orifice device known as a "control section" was installed in the vent condenser discharge line to control the flow of condensate and excess steam to the feedwater heater/condenser. This control section is specially sized to pass the condensate accumulated in the third and fourth passes plus 2% of the heating steam.

3. Main Steam Dump and Bypass System (Figures 10-4 and 10-1, Respectively)

The main steam dump and bypass system consists of four automatically actuated atmospheric dump valves which exhaust to atmosphere and a turbine bypass valve which exhausts to the main condenser; the total capacities of the atmospheric steam dump and turbine bypass valves are 30% and 4.5%, respectively, of steam flow with reactor at full power. The capacity of the atmospheric steam dump valves is adequate to prevent lifting of the main steam safety valves following a turbine and reactor trip. The turbine bypass to the main condenser provides for removal of reactor decay heat following reactor shutdown. Although the steam dump system is arranged for automatic operation, the atmospheric dump valves may be manually controlled from either control room or engineered safeguards control panels.

The atmospheric steam dump valves have a back up nitrogen supply to allow steam generator pressure control during station blackout. This meets 10 CFR 50.63 requirements for coping without AC power by complying with Reg Guide 1.155, Station Blackout.

4. Main Steam Line Isolation

One main steam isolation valve is provided on each main steam header. The main steam isolation valves are closed on either a low steam generator pressure signal or a containment high-pressure signal. Closure of these valves will also result in a turbine-generator trip. Manual closure of one valve will cause automatic closure of the other valve. Each valve consists of a swing disc held open against flow by a pneumatic cylinder. The valves are provided to isolate the steam generators, in the unlikely event of a steam generator tube failure following a main steam line break accident, to prevent the uncontrolled release of radioactivity. Closure of these valves also prevents a rapid uncontrolled cooldown of the Primary Coolant System.

An auxiliary function of the main steam isolation valves is to prevent the release to the containment of the contents of the secondary sides of both steam generators in the event of the rupture of one main steamline inside containment. The valves are normally open, and close in five seconds upon receipt of a low steam generator pressure signal in a no-flow condition. When flow does exist, the valve will close in less than one second. An accumulator is provided to hold the valve open in case of a loss of air supply to the valve operator.

Four pressure transmitters on each steam generator actuate contacts in indicating meter relays which are connected in a two-out-of-four logic to close both main steam isolation valves. On low steam generator pressure only, automatic closing of the main steam isolation valves can be blocked by pushing both of two isolation block push buttons as the steam pressure is decreasing toward the isolation set point. The isolation block is automatically removed by a two-out-of-four logic when the steam generator pressure rises to 50 psi above the isolation set point pressure. Refer to Section 7.2 for further details on system controls.

5. Steam Generator Blowdown System (Figures 10-3 Shts 1-1B and 10-4)

The steam generator blowdown system is designed to process steam generator blowdown water. A minimum continuous blowdown of 5,000 lb/h per steam generator is required for effective steam generator chemistry control. During periods of severe condenser leakage, it is necessary to increase the blowdown rate considerably. Accordingly, the steam generator blowdown system is designed for continuous operation at up to 30,000 lb/h blowdown per steam generator. Other functions of the system include the capability to clean up the condenser hotwell prior to start up by recirculating the water through the blowdown demineralizers, and the capability to recirculate steam generator secondary side water, for treatment purposes, during cold shutdown conditions.

The steam generator blowdown system consists of flash tank, blowdown tank, two blowdown pumps, blowdown heat exchanger, blowdown filter, three blowdown demineralizers, piping, valves and instrumentation. The system is continuously monitored by a process monitor which detects radioactivity which may have leaked into the steam generator from the primary system.

During normal operation, the flash tank, blowdown tank, blowdown demineralizers, blowdown heat exchanger, blowdown filter, and one of the blowdown pumps will be in service. Under this condition, one pump is in "standby" and the other is in continuous service. The standby pump starts automatically on high blowdown tank level. The flash tank and the blowdown tank are vented to the plant heating/evaporator steam system. Blowdown water is pumped through the blowdown heat exchanger and filter (filter is optional) to the blowdown demineralizers and into the condenser. Alternate modes of operation include:

1. The ability to direct the blowdown through the blowdown demineralizers to the condensate storage tank;

2. The ability to direct the blowdown to the mixing basin while isolating the blowdown demineralizers;

3. The ability to direct the blowdown to the condenser while isolating the blowdown demineralizers;

4. And the ability to recycle the blowdown through the miscellaneous waste system and to the utility water storage tank.

A radiation monitor on the effluent of the blowdown tank detects radioactivity which may have leaked into the blowdown water through the steam generators. High activity is annunciated in the main control room. If the radioactivity level in the system reaches a preset level above normal background as detected by the radiation monitor, the recirculation and blowdown containment isolation valves and the mixing basin discharge valve all close and the blowdown vent monitor provides an alarm in the control room.

10.2.2 STEAM TURBINE

The turbine is an 1,800 r/min tandem compound, 3 cylinder, quadruple flow, indoor unit. Saturated steam is supplied to the turbine throttle from the steam generators through four stop valves and four governing control valves. The steam flows through a two-flow, high-pressure turbine and then through four combination moisture separator-reheaters in parallel, and then to two double-flow, low-pressure turbines that exhaust to the main condenser.

Turbine control is accomplished with a rapid response electrohydraulic control system (see Section 7.5). In the event of turbine trip initiated from a solenoid trip, overspeed, low bearing oil pressure, low condenser vacuum, thrust bearing failure or a manual trip, a signal is supplied from the turbine auto-stop oil system to the Reactor Protective System to trip the reactor.

The turbine lubricating oil system supplies oil for lubricating the bearings. A bypass stream of turbine lubricating oil flows continuously through coalescing filters to remove water and other impurities.

10.2.2.1 High-Pressure Turbine

The high-pressure turbine element is of a double-flow design; therefore, it is inherently thrust balanced. Steam from the four control valves enters at the center of the turbine element through four inlet pipes, two in the base and two in the cover. These pipes feed four double-flow nozzle chambers flexibly connected to the turbine casing. Each nozzle chamber is free to expand and contract relative to the adjacent chambers.

Steam leaving the nozzle chambers passes through the Rateau control stages and flows through the reaction blading. The reaction blading is mounted in blade rings which, in turn, are mounted in the turbine casing. The blade rings are center line supported to ensure center alignment while allowing for differential expansion between the blade ring and the casing. This design reduces casing thermal distortion and, thus, seal clearances are more readily maintained.

Steam exhausts from the high-pressure turbine base, through cross-under piping, to the four combined moisture separator steam reheater assemblies.

The high-pressure rotor is made of NiCrMoV alloy steel. The specified minimum mechanical properties are as shown in Table 10-1. The main body of the rotor weighs approximately 100,000 pounds. The approximate values of the transverse center line diameter, the maximum diameter and the main body length are 36 inches, 66 inches and 138 inches, respectively.

The blade rings and the casing cover and base are made of carbon steel castings. The specified minimum mechanical properties are shown in Table 10-2.

The bend test specimen shall be capable of being bent cold through an angle of 90° and around a pin 1 inch in diameter without cracking on the outside of the bent portion.

The approximate weights of the four blade rings, the casing cover and the casing base are 80,000 pounds, 115,000 pounds and 115,000 pounds, respectively.

The casing cover and base are tied together by means of more than 100 studs. The stud material is an alloy steel having the mechanical properties shown in Table 10-3.

The studs have lengths ranging from 17 inches to 66 inches. About 90% of them have diameters ranging between 2.5 inches and 4 inches. The total stud cross-sectional area is about 900 square inches and the total stud free-length volume is about 36,000 cubic inches.

10.2.2.2 Low-Pressure Turbine

The double-flow low-pressure turbine incorporates high-efficiency blading, diffuser-type exhaust and liberal exhaust hood design. The low-pressure turbine cylinders are fabricated from steel plate to provide uniform wall thickness, thus reducing thermal distortion to a minimum. The entire outer casing is subjected to low temperature exhaust steam.

The temperature drop of the steam from its inlet to the LP to its exhaust from the last rotating blades is taken across three walls: Inner Cylinder 1, a thermal shield and Inner Cylinder 2. This precludes a large temperature drop across any one wall except the thermal shield which is not a structural element, thereby virtually eliminating thermal distortion. The fabricated Inner Cylinder 2 is supported by the outer casing at the horizontal center line and is fixed transversely at the top and bottom and axially at the center line of the steam inlets, thus allowing freedom of expansion independent of the outer casing. Inner Cylinder 1 is, in turn, supported by Inner Cylinder 2, at the horizontal center line, and fixed transversely at the top and bottom and axially at the center line of the steam inlets, thus allowing freedom of expansion independent of Inner Cylinder 2. Inner Cylinder 1 is surrounded by the thermal shield.

The steam leaving the last row of blades flows into the diffuser where the velocity energy is converted to pressure energy, thus improving efficiency and reducing the rotational forces on the last rotating row of blades.

The low-pressure rotors are made of NiCrMoV alloy steel in accordance with ASTM A 470. The shrunk-on disc is also made of NiCrMoV alloy steel in accordance with ASTM A 471.

In the late 1970s, Westinghouse turbines developed a history of cracks forming in the disc bores and keyways of low-pressure turbines in nuclear plants. As a result of this generic problem, Westinghouse along with Westinghouse turbine owners and the NRC worked together to arrive at an acceptable solution. In order to minimize the possibility of disc rupture, it is necessary to periodically inspect the critical disc bore region. To arrive at a safe rational procedure for determining inspection intervals, a fracture mechanics approach for calculating the critical crack size is used. The crack's growth rate is predicted from disc yield strength and temperature using the regression equation derived from field data. Details for calculating the critical crack sizes, growth rates and criteria for inspection intervals are discussed in a proprietary Westinghouse Report MSTG-1-P, Criteria for Low Pressure Nuclear Turbine Disc Inspection, submitted to the NRC in June 1981. A summary of nonproprietary information related to "Turbine Missiles" is provided in Section 5.5.

10.2.2.3 Electrical Generator

The generator is made up of a housing, stator, rotor and shaft with sleeve bearings and ventilation blower, see Figure 10-7. The generator is a hydrogen inner cooled unit connected directly to the turbine and rated at 0.85 power factor. It is rated for 955 MVA and has the capability to accept the gross output of the turbine at rated steam conditions. Generator operation is supported by a hydrogen gas system, a seal oil system and a signal system.

1. Hydrogen Gas System - This system provides a safe means for transferring hydrogen to and from the generator, using carbon dioxide as a scavenging medium. In addition, this system maintains the desired gas pressure, cools the gas and dries the gas should moisture get into the machine from the seal oil system.

2. Seal Oil System - This system serves to lubricate the gland seals and to prevent hydrogen leakage from the generator, without introducing an excessive amount of air and moisture into the generator. The same oil is used in the seal oil system and the turbine bearing oil system. The turbine bearing oil system serves as a seal oil backup should the seal oil pump stop or if the seal oil pressure should drop below 8 psi above the generator gas pressure.

3. Signal System - This system provides the operator with signals on the operating conditions present in Table 10-4.

10.2.2.4 Exciter

The exciter is of the brushless type and is driven from the generator shaft. The exciter consists of a permanent magnet generator, an ac generator and a rectifier assembly mounted on a common shaft. The exciter is totally enclosed with suitable heat exchanger and means for circulating the air within the housing.

The high frequency power generated in the stator of the permanent magnet generator is used to supply power to the static Trinistat voltage regulator. The regulator output supplies excitation for the stator of the ac exciter. The rotor of the ac exciter is made with a multiphase winding and supplies high frequency power to the rectifier assembly. The dc output of the rectifier constitutes the main excitation power and is fed directly to the field winding of the generator by means of leads which pass through the coupling to the generator shaft.

10.2.3 CONDENSATE AND FEEDWATER

10.2.3.1 Condensate System

The condensate/feedwater cycle (Figure 10-4) is a closed system with deaeration accomplished in the main condenser. Steam is discharged from the low-pressure turbine and passes around the tube bank area (shell side) of the single pass main condenser to be condensed and deaerated. The main condenser originally contained 511,490 square feet of surface provided by 26,550, 1-inch, 70-foot-long Admiralty tubes, and by 1,426, 1-inch, 304 stainless steel tubes in the air cooler and impingement sections. In 1974, due to tube leakage problems, the entire Admiralty tube section was retubed with 90-10 copper-nickel tubes. In 1990, the main condenser, feedwater heaters E-5A/B & E-6A/B, and drain coolers E-7A/B were replaced to eliminate copper materials of construction in the secondary water/steam cycle. The new condenser contains 24,594 one inch, 70 ft long, 439 stainless steel tubes with an effective surface area of 449,282 square feet. The new heaters and drain coolers contain 304 stainless steel tubes. Other sources of steam and/or water which enter the condenser are:

1. High level dump from the feedwater heater tanks
2. High level dump from the reheater drain tanks
3. Drains from the air ejector
4. Drains from the gland seal condenser
5. Exhaust from the feed pump turbines
6. High level dump from the moisture separator reheater drain tanks
7. Bypass dump steam
8. Makeup from the condensate storage tank
9. Moisture separator-reheater scavenging steam vent chamber (on start-up only)
10. Gland seal steam spillover
11. Miscellaneous vents, steam traps and drains
12. Blowdown from the steam generators

Minimum condensate storage tank inventory is 94,280 gallons (See Section 9.7.2.1). This exceeds a requirement by the Station Blackout rule to have 57,100 gallons available to cope with loss of all AC power for 4 hours.

Noncondensible gases are removed from the main condenser during operation by the steam jet air ejectors, and during start-up by the condenser vacuum pump, hogging air ejector and the steam jet air ejectors. The condenser vacuum pump is used to establish a partial condenser vacuum during start-up and to allow testing of the main condenser for leakage while the Plant is shut down.

The deaerated condensate is transferred from the condenser hot well through a common header to the suction of two half-capacity, electric motor-driven condensate pumps. The two pumps discharge to a common line then diverge in two passes in parallel through the air ejector condenser and the gland seal condenser. The flow from each joins into a common line and is routed to the drain coolers and low-pressure feedwater heaters. The condensate flows from the low-pressure heaters to the suction of the two variable speed, turbine driven steam generator feed pumps. These pumps pump the feedwater through the high-pressure feedwater heaters to the steam generators.

During Plant start-up conditions, feedwater may be recirculated from the discharge of Heaters E-6A and B back to the condenser for purposes of recycling through the condensate demineralizers for system cleanup. This path is isolated during power operation.

10.2.3.2 Condensate Demineralizer System

In 1973, leaks developed in the steam generator tubes. Investigations showed that the problem was tube wastage caused by using phosphates for secondary water chemistry control. In 1974, it was decided to install a full flow condensate demineralizer system and institute a program of steam generator flushing to remove phosphates, continuous steam generator blowdown, start-up recirculation and volatile secondary water chemistry control. Consumers Power Company concurred with studies and tests performed during 1979 and 1980 that the Condensate Demineralizers would be of more detriment than help. Use of the Condensate Demineralizer System ceased in 1981. This system is currently considered as "retired in place."

The Condensate Demineralizer System was isolated from the Condensate System by replacing the inlet and outlet valves with blind flanges. The Condensate Demineralizer bypass valve was removed to avoid inadvertant closures.

10.2.3.3 Feedwater Regulating System

Two half-capacity feedwater pumps (Figure 10-4) are used to furnish the feedwater flow. Each turbine driver and pump must be started locally and brought up to speed before the driver can be controlled from the main control room. The suction and discharge pressures of the feedwater pumps are indicated and annunciated in the main control room. If the suction pressure falls (2 out of 3 logic) below a preset critical value, the pump will be automatically tripped. The turbine drivers will also be tripped from thrust bearing failure and overspeed. A manual trip is also available. Steam flow to each turbine driver is indicated and recorded in the main control room. The turbine speed is also indicated in the control room. Three modes of control are provided as follows:

1. Automatic Control in Conjunction With Feedwater Regulating Valves

   Each steam generator's three-element control unit will produce a demand signal for feedwater flow which is a function of the steam generator downcomer level error, trimmed by the difference between feedwater flow and compensated steam flow. The feedwater flow demand signal for each steam generator will be sent to the corresponding feedwater regulating valve controller and to the turbine driver speed control system. The feedwater regulating valve controller in combination with the turbine driver speed control system will function to control the level in each steam generator by modulating the feedwater flow. The regulating valve controller will automatically adjust the position of the regulating valves. The speed control system will select the signal from the steam generator requiring the higher feedwater flow. The selected feedwater demand which represents a speed demand for both turbine drivers is compared to a feedback signal from turbine speed in the speed controller of each driver. Any difference between the two signals will cause the speed controllers to produce a change in turbine speed in the appropriate direction. At low power levels (< 25% power) a single element control unit is used. See Section 7.5 for details.

   In 1980, additional automatic controls were placed on the feedwater regulating and bypass valves such that they would close on receipt of a low steam generator pressure (500 psia) signal. This modification was needed to prevent, in the event of a main steam line break, the possibility of a condensate pump supplying water to a depressurized steam generator causing overcooling of the Primary Coolant System. Overpressurization of containment was also a concern if the additional mass of feedwater were released as steam into containment through the broken steam line.

In 1990 additional automatic controls were placed on the main feedwater regulating valves and bypass valves such that they would also close on containment high pressure (CHP). This modification was needed when it was discovered that a small steam line break would result in high containment pressure, but not reduce steam generator pressure fast enough to close the valves in time to prevent exceeding containment design pressure.

2. Fixed Speed Control

The turbine driver speed control system can be divorced from the feedwater regulating system and operated automatically to maintain parallel operation of each turbine driver at a manually set speed. The feedwater regulating valve system will then function to control steam generator level by automatically throttling the discharge of the feed pumps as in 1. above.

3. Manual Control

The speed of each turbine driver may be manually adjusted from the control room.

The preceding operational modes consider simultaneous operation of both pumps. The system is designed to permit operation with one feed pump under all modes at reduced unit load.

If the turbine driver speed controllers are in Control Mode 1 or 2 at the time of a turbine trip, the turbine drivers will be automatically ramped down at a rate of 1.58% per second to a speed corresponding to 5% of full load feedwater flow. At the same time, the feedwater regulating valves will be locked in place at their respective existing positions.

With the loss of offsite power, the main feed pumps will be tripped from low suction header pressure which will result from loss of power of the condensate pumps from their supply buses. The auxiliary feedwater pumps will be available for service at the operator's discretion.

10.2.4 CIRCULATING WATER SYSTEM

Initially, the Plant was designed for a once-through condenser cooling Circulating Water System. The circulating water was taken from the lake through a submerged crib and a 3,300-foot-long pipe tunnel into the intake structure and pumped by two half-capacity motor-driven pumps through the condenser tubes to the discharge canal.

In 1974, the Circulating Water System was converted to a closed cycle system (Figure 10-6) using two mechanical draft cooling towers. The cooling tower system consists of two independent closed loops. Each loop supplies one-half of the main condenser with cooling water.

10.2.4.1 Cooling Towers

Cooling water is supplied to the main condenser by gravity flow from two 18-cell, induced draft cross-flow cooling towers. The cooling towers (Table 10-8) are designed for a 30°F range (inlet temperature minus outlet temperature). The cooling towers are erected to the south of the Plant (Figure 2-2) over concrete basins. The basin water level elevation is approximately 20 feet above the condenser inlet. Each tower basin supplies one-half of the condenser through a 90-inch pipe which connects to a 96-inch condenser inlet piping at the intake structure.

Two sets of fixed screens are provided at the outlet of each cooling tower to remove any debris which collects in the basin. Provisions for stop logs are also provided at the basin outlet.

The two cooling towers are located approximately 500 feet and 1,000 feet, respectively, from the Plant and 300 feet from the nearest transmission lines in order to minimize icing potential. They are spaced approximately 500 feet apart to prevent warm air recirculation between the towers.

Two half-capacity vertical wet pit cooling tower pumps (Table 10-9) are installed in the cooling tower pump building. The pumps receive heated circulating water from the condenser via the 96-inch condenser discharge piping and pump suction spillway provided to reduce velocity head. The cooling tower pumps return the circulating water to the cooling tower distribution headers through two 96-inch pipes. Motor-driven butterfly valves are provided in both the pump discharge and condenser inlet piping. The valves are provided for throttling during pump start-up and for maintenance isolation. The valves are interlocked with their corresponding pump motor breakers.

10.2.4.2 Makeup and Blowdown

The maximum expected blowdown rate is approximately 30,000 gpm, in addition to a full-load tower evaporation rate of 6,000 gpm each during the summer and 4,500 gpm each in the winter. In order to minimize the discharge temperature through the blowdown line, the blowdown is extracted from the circulating water piping just upstream of the condenser inlet and is discharged to the discharge mixing basin. The blowdown is controlled to keep the discharge temperature in compliance with the Plant's NPDES permit and to maintain 1-1/2 cycles of concentration when compared to the lake inlet concentrations. Organic antiscalant and sodium hypochlorite can be added to the Circulating Water System to prevent scaling and biological growth in the condenser tubes.

The NPDES permit limits the amount of total sodium hypochlorite that can be discharged to surface water systems. In order to meet the discharge limit and at the same time use sodium hypochlorite at an effective biocide concentration, sodium bisulfite is injected to the overflow weirs of the north and south make-up basins. Sodium Bisulfite is used as a de-chlorinator agent to reduce a total residual oxidant in the Lake out flow, to levels acceptable to the NPDES Permit.

Makeup water is provided by discharging the Plant service water effluent into the cooling tower makeup basin (discharge structure). The service water flow ranges from 14,200 gpm to 15,600 gpm, resulting in a surplus of makeup water which overflows the overflow weirs into the discharge mixing basin.

10.2.4.3 Dilution

Dilution water is normally added to the Circulating Water System on the inlet line downstream of the blowdown line. Two 30,000 gpm vertical dilution pumps provide this flow. In addition, dilution flow may be directed to the mixing basin on an as-needed basis.

In order to ensure proper mixing and surface discharge of the blowdown and excess service water returns, an extension from the existing discharge structure provides a mixing basin on the downstream side of the overflow weirs. Mixing is achieved by discharging the heated water sources plus the dilution flow to a controlled area at moderate velocity. The mixed discharge flows out of the mixing basin to the lake at a low velocity.

10.2.5 CODES AND STANDARDS

All components in the system are designed and fabricated in accordance with applicable codes; eg, the moisture separators-reheaters and the closed feedwater heaters are in accordance with the ASME B&PV Code, Section VIII, and the piping and valves are to ASA B31.1-1955, Code for Pressure Piping.

The components are similar to those which have experienced extensive service in operating power plants. Adequate protective devices and controls are provided to assure reliable and safe operation.

10.3 SYSTEM ANALYSIS

10.3.1 REACTOR AND/OR TURBINE TRIP

Following a reactor and/or turbine trip, the feedwater flow to the steam generator is ramped down to 5% of full flow in the first 60 seconds. Once the system transient has terminated, the operator, while monitoring the primary coolant temperature, can restore and maintain the steam generator level. The feedwater temperature will decrease to that of the stored condensate.

10.4 TESTS AND INSPECTIONS

Equipment, instruments and controls are regularly inspected in order to ensure proper functioning of systems.

The turbine governor and stop valves, reheat stop and intercept valves, bleeder trip valves and auxiliary feedwater pump may be tested while the turbine is in operation.

In addition, during the Plant shutdown period for refueling, equipment, instruments and controls can be checked and inspected.

10.4.1 PIPE WALL THINNING INSPECTION PROGRAM

In response to Generic Letter 89-08, the pipe wall thinning inspection program was initiated to meet or exceed the requirements of NUREG-1344, Appendix A.

A component susceptibility ranking has been broken down into 14 systems/ subsystems for ease of tracking. These systems are:

1. Main steam
2. Condensate
3. Feedwater
4. Steam generator blowdown
5. Heater drain pump discharge
6. Reheater drain tank
7. Numbers 5 and 6 heater drains
8. Moisture separator drain tanks
9. Numbers 1-4 heater drains
10. Heater vents
11. Extraction steam to Number 6 heater
12. Extraction steam to Number 5 heater
13. Extraction steam to Number 3 heater
14. Extraction steam to Numbers 1-2 heaters

Components in these systems are ranked according to projected wear rates obtained by modeling. Modeling factors for single-phase systems include piping material, fluid velocity, piping configuration, oxygen concentration, pH and temperature. Factors for two-phase systems include percent moisture, piping material, temperature, oxygen concentration, pH, piping configuration and fluid velocity.

Components were selected in a three-stage process. First, systems were selected based on material, velocity and temperature. Second, subsystem selection used temperature and velocity, water/steam quality and pH/chemistry. And last, component selection considered geometry, walkdowns and experience.

By letter dated April 19, 1980, the NRC accepted this program on the basis that it complied with NUREG-1344, Appendix A.

TABLE 10-1

HIGH-PRESSURE ROTOR PROPERTIES

Tensile Strength, psi, Min	100,000
Yield Strength, psi, Min (0.2% Offset)	80,000
Elongation in 2 Inches, %, Min	18
Reduction of Area, %, Min	45
Impact Strength, Charpy V-Notch, ft-lb (Min at Room Temperature)	60
50% Fracture Appearance Transition Temperature, °F, Max	50

TABLE 10-2

BLADE RINGS AND TURBINE CASING PROPERTIES

Tensile Strength, psi, Min	70,000
Yield Strength, psi, Min	36,000
Elongation in 2 Inches, %, Min	22
Reduction of Area, %, Min	35

TABLE 10-3

STUD MECHANICAL PROPERTIES

	Size, Inches		
	2-1/2 and Less	Over 2-1/2 to 4	Over 4 to 7
Tensile Strength, psi, Min	125,000	115,000	110,000
Yield Strength, psi, Min (0.2% Offset)	105,000	95,000	85,000
Elongation in 2 Inches, %, Min	16	16	16
Reduction of Area, %, Min	50	50	45

TABLE 10-4

SIGNAL SYSTEM OPERATING CONDITIONS

1. Hydrogen Purity - High or Low
2. Hydrogen Pressure - High or Low
3. Hydrogen Supply Pressure - Low
4. Water Detection - High
5. Hydrogen Temperature High / Generator Condition Monitor
6. Defoaming Tank Level - High
7. Air Side Seal Oil Pump - Off
8. Seal Oil Pressure - Low
9. Hydrogen Side Level - Low
10. Seal Oil Turbine Backup Pressure - Low
11. Hydrogen Side Seal Oil Pump - Off
12. Air Side Seal Oil Backup Pump Running

TABLE 10-5

(Deleted)

TABLE 10-6

(Deleted)

TABLE 10-7

(Deleted)

TABLE 10-8

COOLING TOWER DESIGN SUMMARY

Number of Towers	2
Number of Cells	18/Tower
Type	Induced/Cross Flow
Performance at Design Point	
Hot Water Flow, gpm	410,000
Approach Temperature, °F	107
Range, °F	30
Cold Water Temperature, °F	77
Fan Brake Horsepower, bhp	193
Pumping Head Above Sill, ft	90
Fans and Fan Motors	
Number of Fans	18
Number of Blades	8
Blade Diameter, ft	28
Blade Material	Fiberglass Reinforced Polyester
Tip Speed, ft/s	198
Fan Speed, r/min	135
Rated hp/Voltage	200/460V

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TABLE 10-9

COOLING TOWER PUMPS

Number of Pumps	2
Type	Vertical, Wet Pit
Design Flow, gpm	205,000
Design Head, ft	90
Design NPSH, ft	29
Pump Speed, r/min	253
Motor Type	Induction
Horsepower, hp	7,000
Voltage	4,160

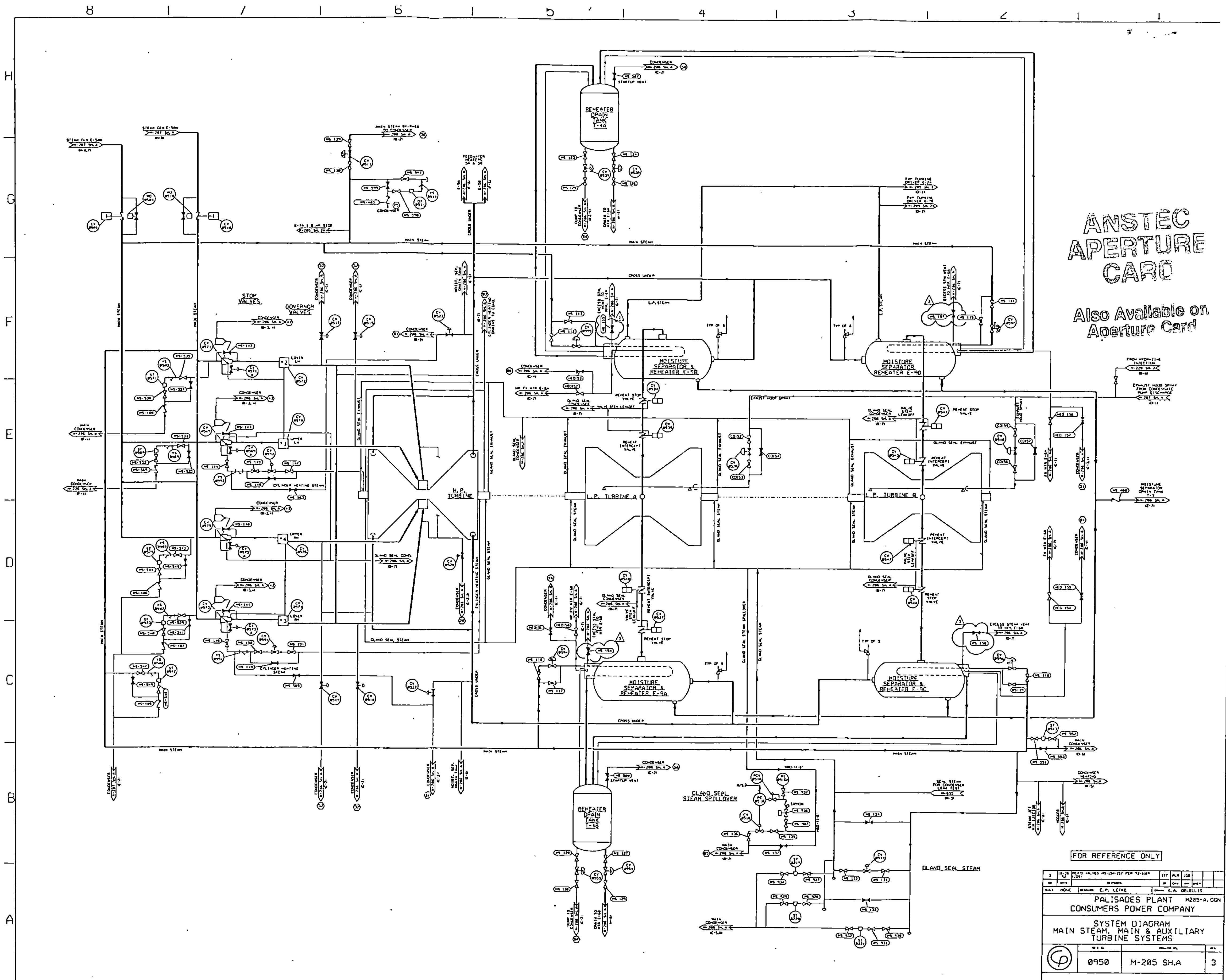

FIG 10-1
FSAR REV 20

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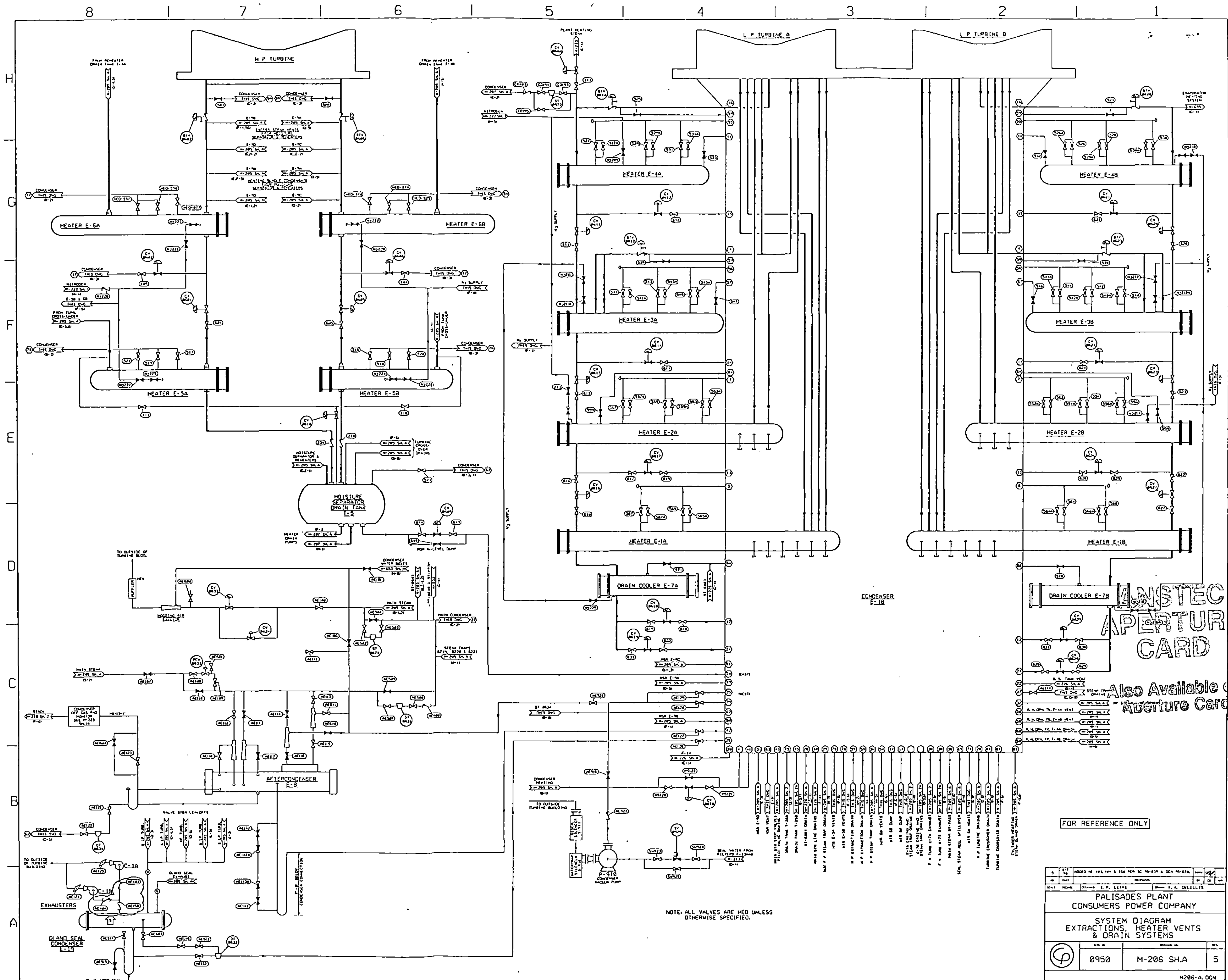

FIG 10-2
FSAR REV 20

APERTURE CARD
Also Available on Aperture Card

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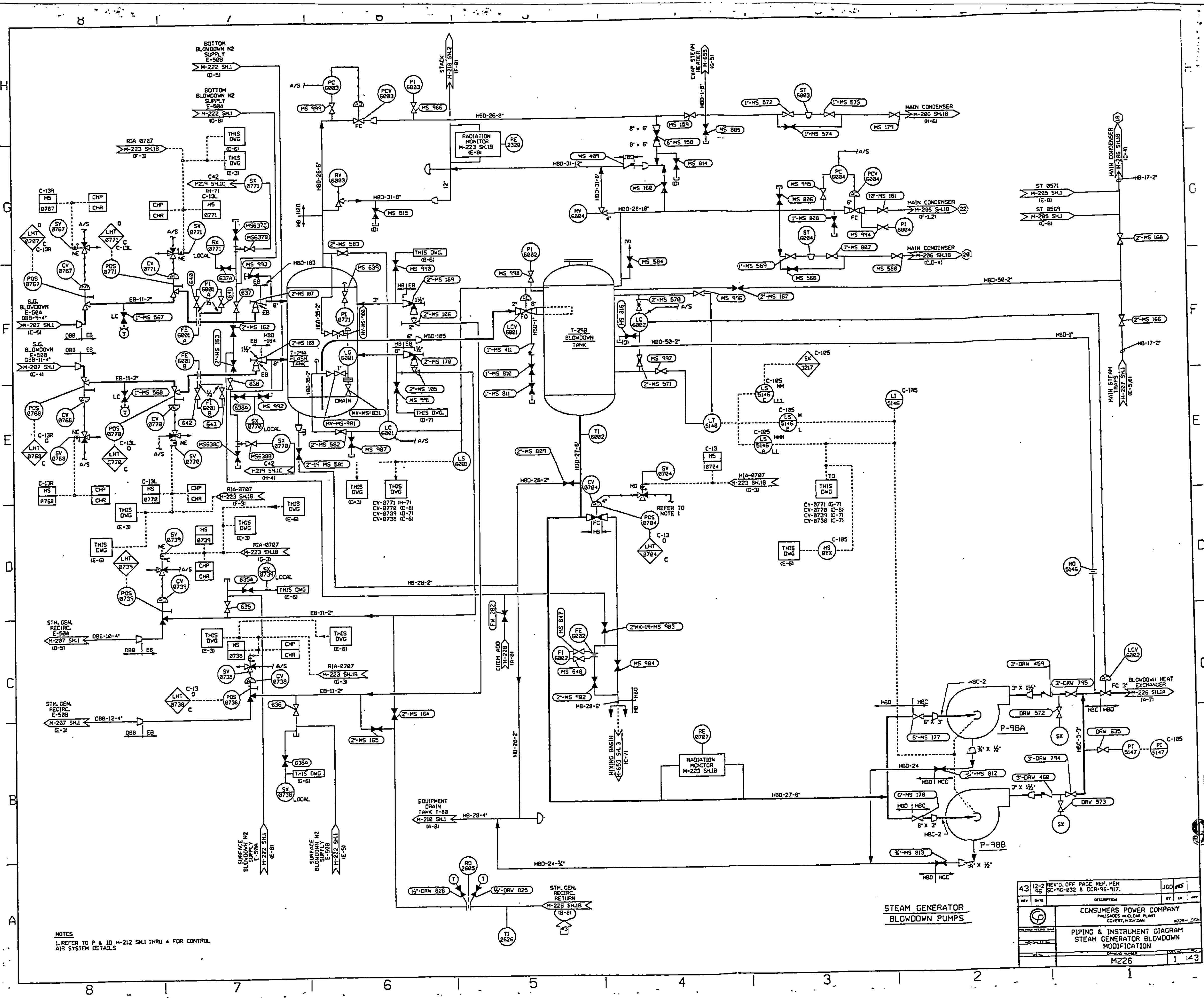
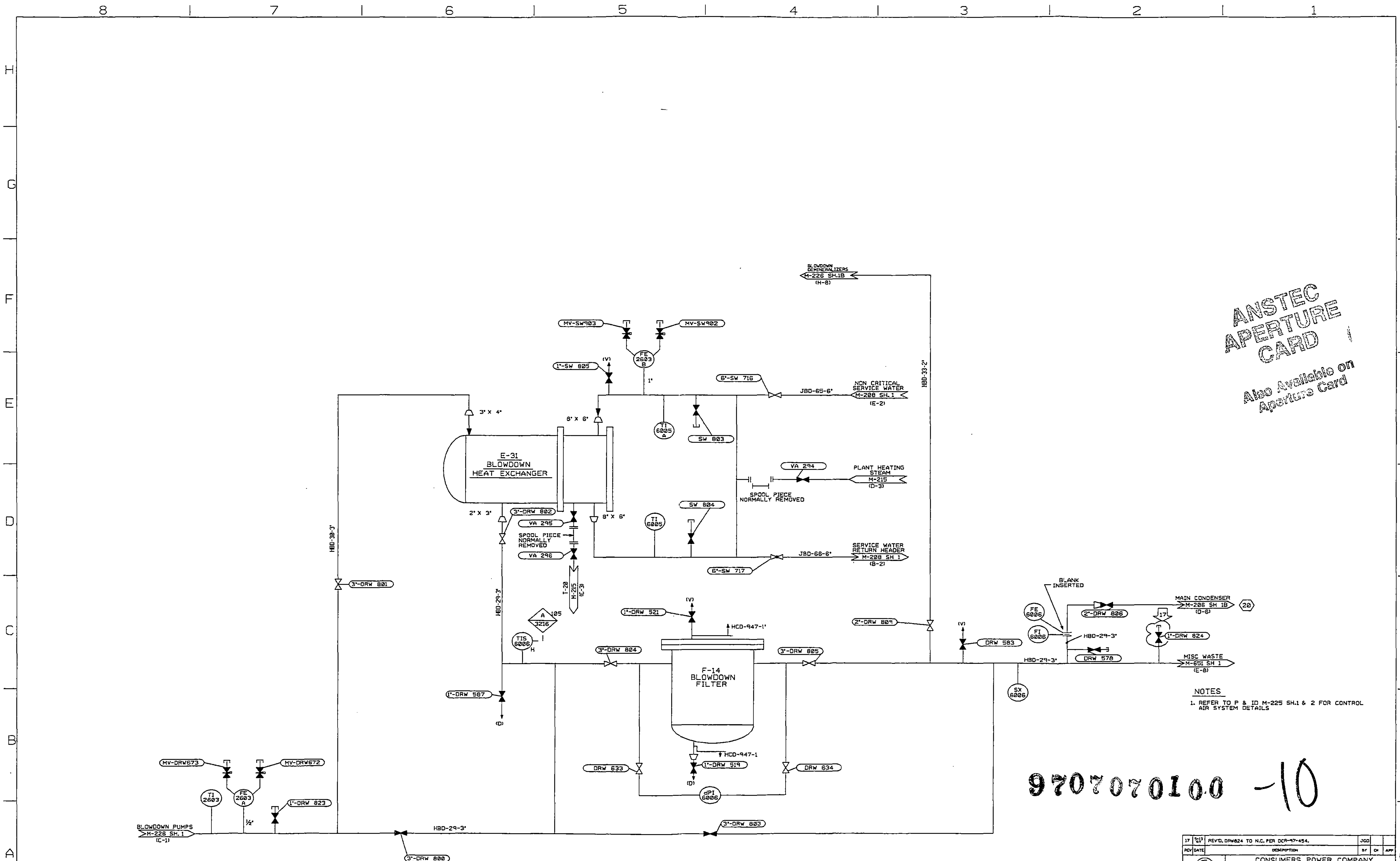


9707070100 -09

Figure 10-3 Sh 1
Rev 19



NOTES
1. REFER TO P & ID M-225 SH.1 & 2 FOR CONTROL AIR SYSTEM DETAILS

9707070100 -10

17		REV'D. DRW824 TO N.C. PER DCR-97-454.	JGD		
REV	DATE	DESCRIPTION	BY	CH	APP

CONSUMERS POWER COMPANY
PALISADES NUCLEAR PLANT
COVERT, MICHIGAN
M226-1A.DGN

PIPING & INSTRUMENT DIAGRAM
STEAM GENERATOR BLOWDOWN
MODIFICATION

M226 | 1A | 17

Figure 10-3 Sh 1A
Rev 19

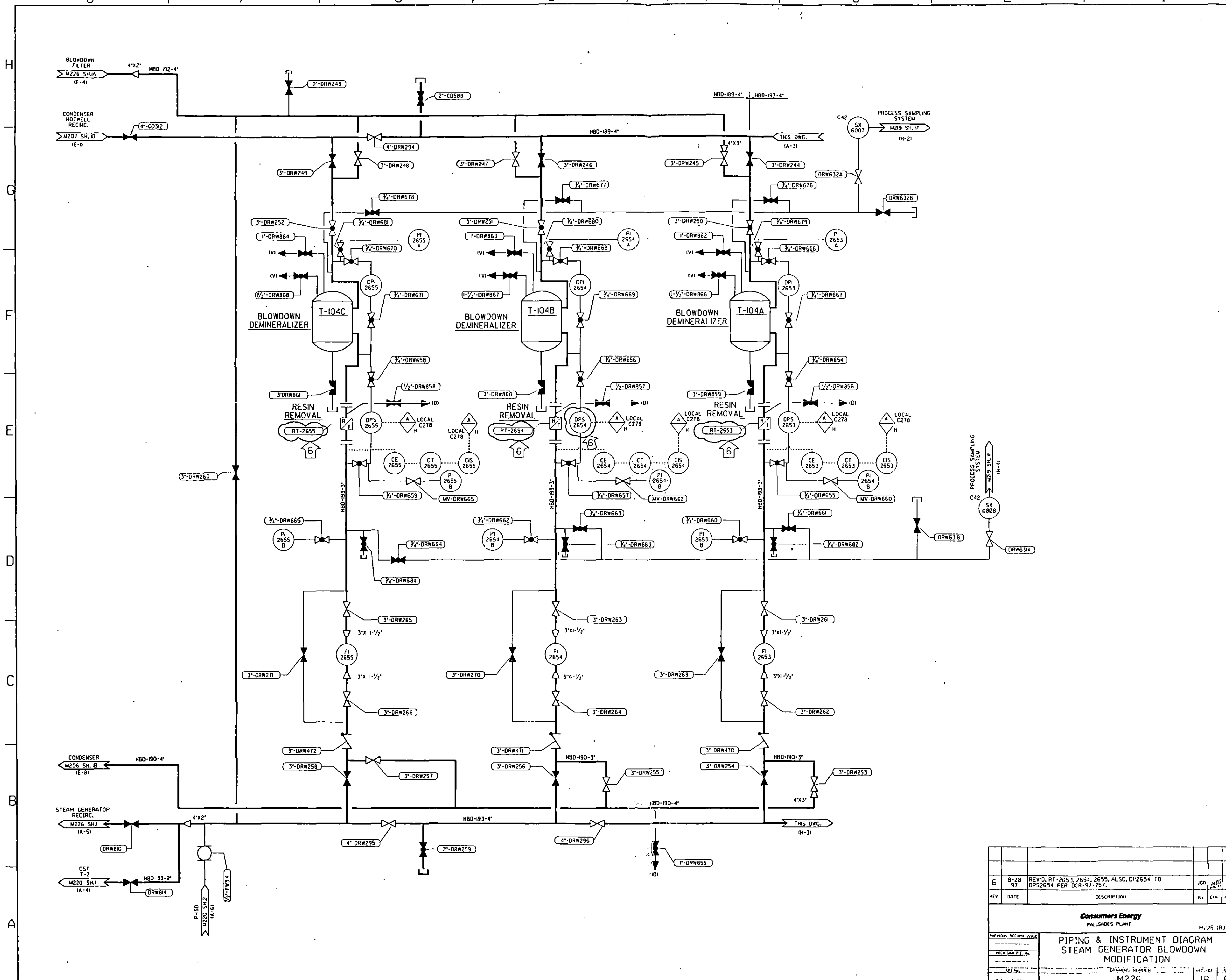


FIG 10-3 SH 1B
FSAR REV 20

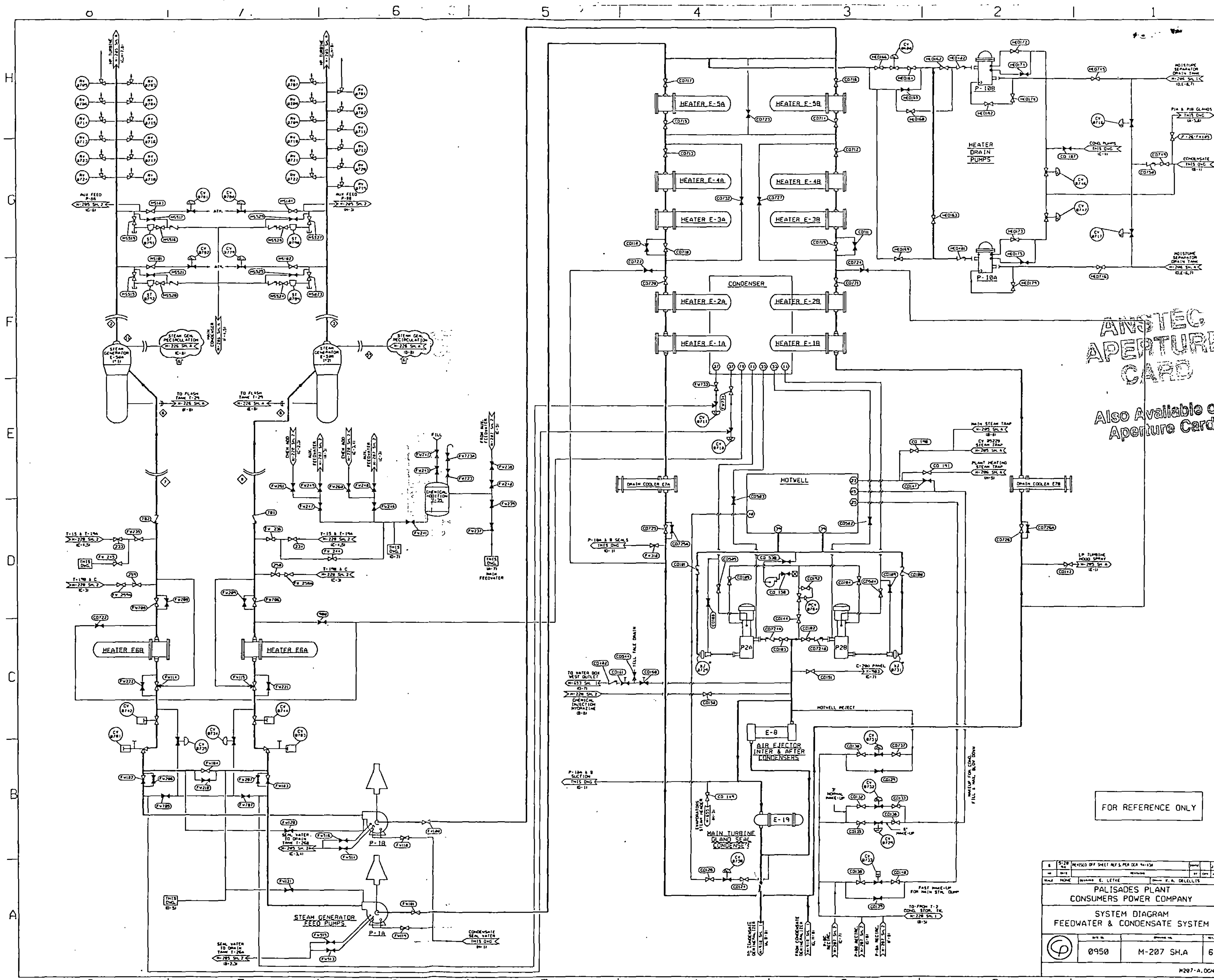


FIG 10-4
FSAR REV 20

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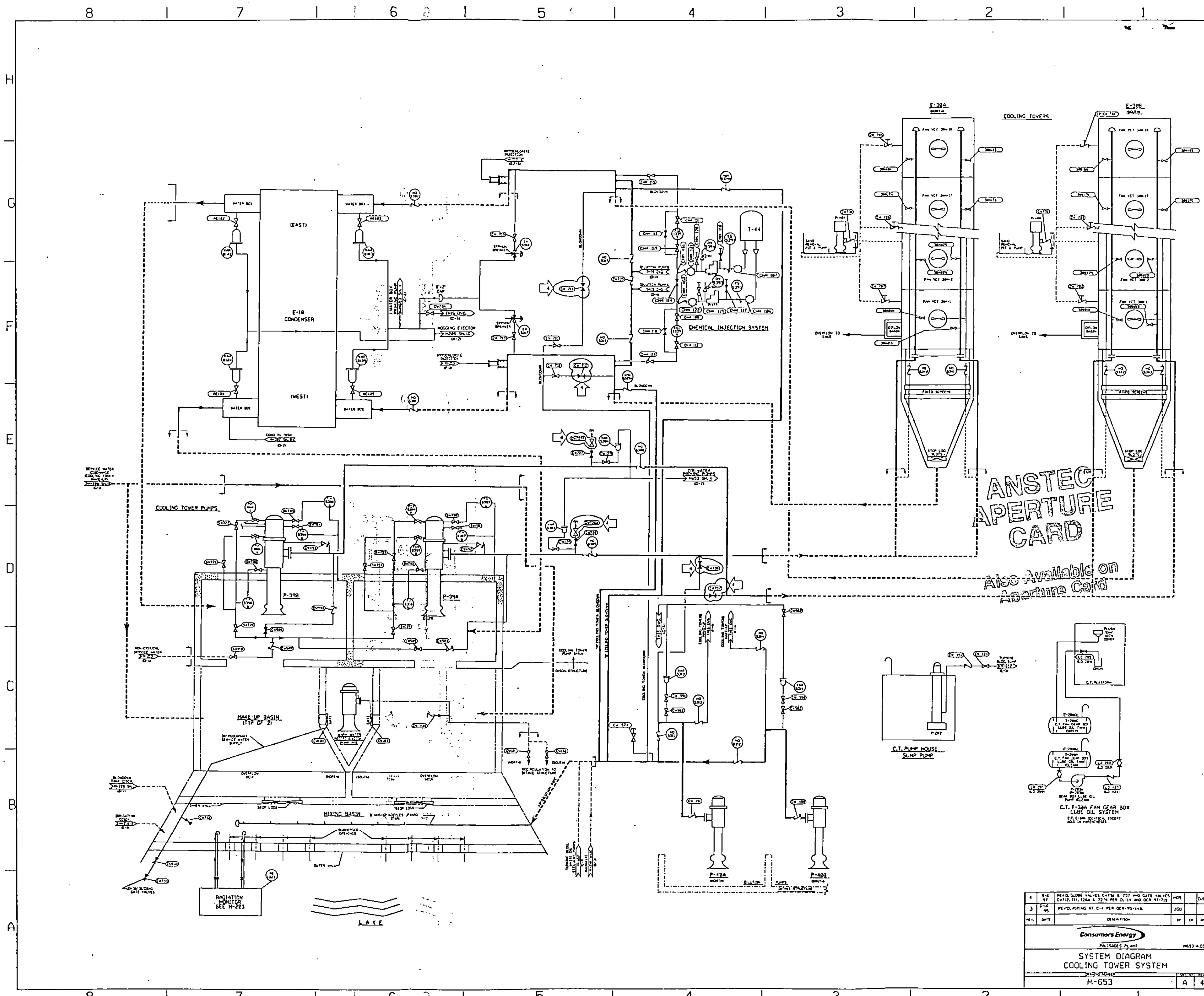
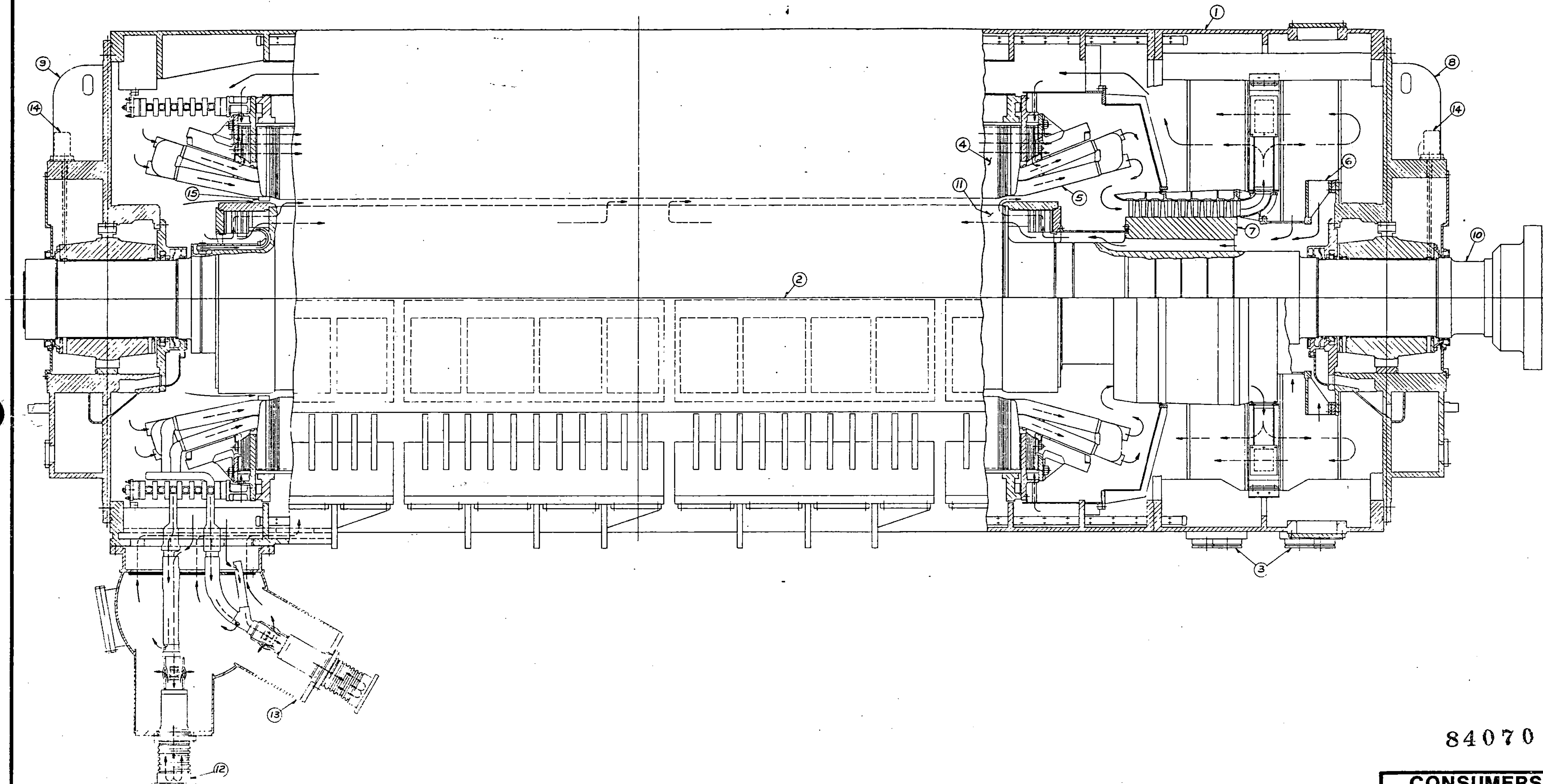


FIG 10-6
FSAR REV 20



IT	DESCRIPTION
1	FRAME
2	FRAME COVERS
3	HYDROGEN COOLERS
4	STATOR CORE
5	STATOR WINDING
6	BLOWER SHROUD SUPPORT ASSEMBLY
7	BLOWER
8	GLAND SEAL, BEARING & BRACKET T.E.
9	GLAND SEAL, BEARING & BRACKET E.E.
10	SHAFT
11	ROTOR WINDING
12	MAIN LEADS
13	MAIN LEAD BOX
14	VIBROMETER PICKUP (WHEN FURNISHED)
15	AIR GAP BAFFLE

TI
APERTURE
CARD

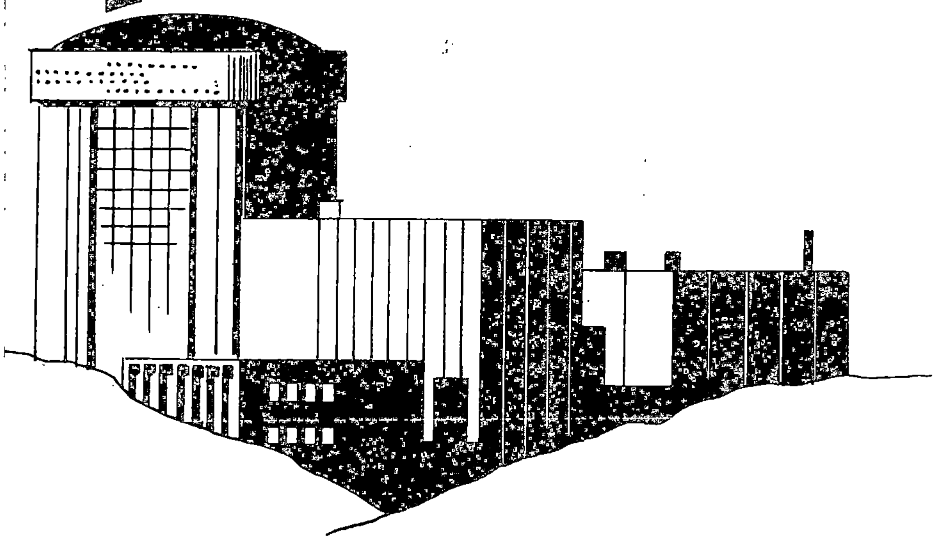
8407050291 -18

CONSUMERS POWER COMPANY
PALISADES PLANT
FSAR UPDATE

GENERAL ASSEMBLY ELECTRICAL GENERATOR

FIGURE NO 10-7

REVISION NO 0



ALISADES NUCLEAR PLANT

fina
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report

UPDATE

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5A.1 Special Report No 6 - Analysis of Postulated High-Energy Line Breaks Outside of Containment

VOLUME 4

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CHAPTER 11

RADIOACTIVE WASTE MANAGEMENT AND RADIATION PROTECTION

11.1 SOURCE TERMS

Liquid and Gaseous Waste Activity

The normal sources of radioactive wastes are fission and activation products generated within the Primary Coolant System during Plant operation. The radioactive waste systems are designed to safely process radioactive wastes from the Plant with a primary coolant activity based on 1% failed fuel rods and continuous purification during Plant operation. The calculated primary coolant activity by isotope for the 1% failed fuel rod condition is presented in Table 11-1. This table also presents activity in the primary coolant as observed during normal power operation and indicates that the total activity level normally remains at less than 1% of design.

Sources and Accumulation of Wastes

For the purpose of design evaluation, the activity of the sources to the clean waste system was assumed to be that of primary coolant for the 1% defective fuel rod condition as given in Table 11-1. Table 11-2 lists the maximum expected quantity of clean waste of significant activity from the major waste sources and the conservative assumptions used to derive the waste quantities. The dirty waste system was designed to handle radioactive wastes generated by a steam generator blowdown rate of 20 gpm concurrent with 1 gpm primary to secondary leakage. It was assumed that dirty waste has an activity equal to 1% of the primary coolant activity. (It should be noted that primary to secondary leak rate is currently limited to less than 0.3 gpm by the Technical Specifications.) Further, as shown by Table 11-1, primary coolant activity normally is less than 1/100 that of the design basis activity. These considerations combine to provide for an extremely conservative system design relative to normal operating conditions.

11.2 LIQUID RADIOACTIVE WASTE SYSTEM

11.2.1 DESIGN BASES

11.2.1.1 Design Objective

The Liquid Radioactive Waste System is designed to collect, store, process, monitor, and dispose of all liquid radioactive wastes from the Palisades Plant. The principal design criterion is to ensure that the general public is protected from exposure to radioactive waste products in accordance with Title 10, Code of Federal Regulations, Part 50 (10 CFR 50), Appendix I. The system was modified during 1971-1973 following voluntary agreement between Consumers Power Company and outside intervenors to reduce liquid discharges to Lake Michigan to essentially "Near-Zero" as achievable by use of modern processing equipment which permits recycling of most liquids. This modification consisted of the addition of clean waste evaporators and auxiliary equipment which were integrated into the Liquid Radioactive Waste System to provide clean waste recycle features reducing radwaste effluents.

11.2.1.2 Design Criteria

The Liquid Radioactive Waste System was originally designed and constructed as a CP Co Design Class 1 system and was located within the CP Co Design Class 1 auxiliary building, defined in Section 5.2. However, mechanical portions of the system not associated with "a loss of function which could cause an uncontrolled release of radioactivity" were designed as CP Co Design Class 2. Subsequent modifications to this system were undertaken during 1971-73 as described in Special Report 5 (see Reference 1) and Amendment 21 to Docket 50-255, dated February 26, 1971. Two buildings, the service building and the auxiliary building addition, were added during this time enclosing the new equipment. The auxiliary building addition is CP Co Design Class 1, the service building is CP Co Design Class 3, the liquid radwaste components are CP Co Design Class 2 and processing piping is CP Co Design Class 3, all per Section 5.2.

11.2.1.3 Codes

The original portions of the Liquid Radioactive Waste System were designed according to codes and criteria in effect at the time of initial design and construction of Palisades which was ASA B31.1-1961 for pressure piping and ASME, Section III, Class C, for vessels. These codes generally meet the suggested codes depicted for Quality Group D components and piping in NRC Regulatory Guide 1.26, Revision 1, issued September 1974, and Regulatory Guide 1.143, Revision 1, issued October 1979. A detailed listing of codes used on original components and piping is given in Table 11-3.

Liquid Radioactive Waste System modifications installed during the 1971-1973 service building addition were designed in accordance with the following applicable codes:

All nonatmospheric pressure bearing liquid radioactive waste process piping is designed to ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components, Nuclear Class 3, 1971 Edition, while all atmospheric radwaste drainage piping is designed to USAS B31.1.0-1967 Power Piping.

All tanks are designed to either ASME B&PV Code, Section III, Class 3, 1971, or API-620, 1970, or API-650, 1970, as follows:

ASME Section III: Pressure Vessels Over 15 Psig Design

API-620: 0 to 15 Psig Tanks (Tested at 5 Psig)

API-650: Atmospheric Tanks (Without Pressure Tests)

A more detailed listing of codes used for these components and piping will be found in Table 11-4.

In addition, it should be noted that the Systematic Evaluation Program (SEP) Topic III-1, "Quality Group Classification of Components and Systems (Palisades)," which was issued in December 1981 does not address liquid waste management system components and piping, all of which fall within Quality Group D, as defined in NRC Regulatory Guide 1.26. The Liquid Radioactive Waste System is now considered to be a Quality Group D system.

11.2.2 SYSTEM DESCRIPTION

The Liquid Radioactive Waste System is divided into three sections: (a) the clean waste section which processes high-activity, high-purity (low solids) liquid waste, (b) the dirty waste section which processes low-activity, low-purity (high solids) liquid waste and (c) the laundry waste section. The basic system is shown in Figures 11-1 and 11-2. Component ratings and descriptions are included in Tables 11-3 and 11-4.

11.2.2.1 Clean Waste Section

In the clean waste section of the Liquid Radioactive Waste System, wastes are collected, monitored, and processed by a combination of holdup (thereby permitting natural decay), filtration and ion exchange treatment (removal of insoluble particulates and soluble ions), evaporation (volume reduction), and under normal conditions, are stored for eventual recycle back to the Primary Coolant System. Quantities of clean waste are obtained from the Chemical and Volume Control System's bleed letdown (primary coolant - see Figure 9-25), and are generated due to Plant start-ups and shutdowns which require that primary coolant boron concentrations be varied to maintain the necessary shutdown margins.

Liquid waste from the Chemical and Volume Control System first passes through an ion exchanger and purification filter before entering the clean waste section. In the clean waste section, this stream passes through the vacuum degasifier and is then discharged into one of the clean waste receiver tanks. The vacuum degasifier removes hydrogen and fission product gases and discharges them to the waste gas surge tank located in the Gaseous Radioactive Waste System.

The primary system drain tank collects liquid waste from sources within the containment building as listed in Table 11-5. The primary loop drains provide the major source of liquid to the drain tank. Liquid waste from the primary system drain tank is also passed through the vacuum degasifier before entry into one of the clean waste receiver tanks.

The equipment drain tank collects liquid waste from outside the containment building as listed in Table 11-6. Liquid waste from the equipment drain tank is discharged to the clean waste receiver tanks through a filter.

The radiochemistry lab drain tank collects the liquid waste generated while sampling the primary coolant for chemical and radiochemical analysis. Liquid waste from the radiochemistry lab drain tank is discharged to the Solid Waste Management System for solidification or can be sent to the dirty radwaste evaporator for volume reduction.

Four 50,000-gallon clean waste receiver tanks located inside the containment building provide temporary storage for collected liquid waste to allow for natural decay and to permit sampling of liquid waste activity. Liquid waste from the receiver tanks is discharged through the clean waste filter to remove insoluble particulates and through the radwaste demineralizers to remove soluble ions.

Under normal conditions, the effluent from the clean waste demineralizers passes to the clean waste holdup tank or to a clean waste receiver tank, and then to the clean radwaste evaporator. The evaporator has the capability to process radioactive waste either continuously or in the batch mode. The clean radwaste evaporator concentrates boric acid and radioactive isotopes, except for gases and tritium. The evaporator concentrates are discharged to the clean waste concentrate tank, while its relatively pure distillate is discharged to the clean waste distillate tank.

The boric acid obtained from the clean radwaste evaporator is then pumped into the recycled boric acid storage tank where it is available to the Chemical and Volume Control System for reuse in the Primary Coolant System. However, if the boric acid cannot meet the quality requirements specified in the Plant's Technical Specifications, it is sent to the Solid Waste Management System where it is prepared for packaging and disposal.

The distillate from the radwaste evaporator is normally processed through a polishing demineralizer to further reduce boron and radioactivity concentrations, and is then sent to the primary system makeup storage tank where it is stored until used by the Chemical and Volume Control System in making up the primary coolant. If the distillate volumes become excessive under abnormal operating conditions or cannot meet the quality requirements specified in the Plant's Technical Specifications, provisions are available to send the distillate from the evaporator to the treated waste monitor tanks for holdup and sampling to determine the disposition of this liquid. Process routes available from these tanks include reprocessing through the clean radwaste evaporator, return to the clean waste receiver tanks for further decay and later processing, or to be routed to the circulating water discharge for dilution and release to the environment after reduction to levels that are as low as reasonably achievable.

11.2.2.2 Dirty Waste Section

In the dirty waste section of the Liquid Radioactive Waste System, wastes are collected, monitored and processed by a combination of filtration, evaporation and demineralization. The dirty waste drain tank collects liquid waste from the sources listed in Table 11-7.

Liquid from the dirty waste drain tank is processed through the miscellaneous waste filters while en route to the miscellaneous waste holdup tanks for temporary storage. Effluent from these tanks is fed to the dirty radwaste evaporator which is identical to that in the clean waste system. The distillate from the evaporator is sent to the miscellaneous waste distillate tank and then through either one or two demineralizers to the utility water storage tank for reuse in the Plant. If the distillate volumes become excessive under abnormal operating conditions or cannot meet the quality requirements specified in the Plant's Technical Specifications, provisions are available to reprocess evaporator distillate. Under extreme situations, discharge could be made to the discharge basin within limitations of the Offsite Dose Calculation Manual.

The boric acid concentrate obtained from the dirty radwaste evaporator is sent to the evaporator concentrate tank. The concentrates are sent to the Solid Waste Management System for solidification and disposal.

11.2.2.3 Laundry Waste Section

All liquid effluent obtained from the onsite laundering of anticontamination clothing and the water from the face-mask decontamination washing station is processed through a media filter and basket strainer, both connected in series, before being temporarily collected in the laundry drain tank. The tank's contents are then sampled to determine the disposition routing for this waste stream.

Normal system lineup permits transport of this waste stream from this storage tank through three series connected filters having successive ratings of 50, 25 and 5 microns before being deposited into filtered waste monitor tanks. The latter filters were installed by a field change in 1977. These filters minimize clogging or fouling problems in the dirty waste section.

Alternate system lineup is available to transport this waste stream directly to the Plant's discharge mixing basin following filtration by the three series connected filters, and monitoring to assure that the discharged effluent activity is below activity levels specified in Subsection 11.2.3.3. However, discharge to the mixing basin is usually made from the filtered waste monitor tanks rather than by the alternate direct-discharge lineup.

Additionally, the laundry waste section's volumes have been greatly reduced by installation of an independent dry cleaning machine in early 1980 which utilizes recyclable solvents for laundering rather than generating water-based soap and detergent wastes which formerly constituted most of this section's volume throughput.

Both dry cleaning and wet wash machines were removed from service in 1989. All laundry is now processed offsite by a vendor. The drain and filter units are still in place. Small amounts of detergent wastes are dumped into the drain system and are drained into the normal dirty waste system instead of discharging directly.

11.2.3 RADIOACTIVE RELEASES

The Liquid Radioactive Waste System was designed to reduce radioactive materials in liquid discharges from the Palisades Plant, except laundry waste discharges, to essentially zero (ie, actual lake background) under normal operating conditions, and to ensure that radioactive materials in liquid wastes discharged under normal conditions will be at a small fraction of the applicable limits set forth in Title 10 CFR, Part 50, Appendix I (10 CFR 50, Appendix I), as stipulated in Subsection 11.2.1 (see Appendix 11A).

A discussion on how the Liquid Radioactive Waste System's sections meet these requirements follows:

11.2.3.1 Clean Waste Section

The maximum anticipated annual quantity of clean liquid waste of significant activity to be processed through the liquid waste system before being recycled to the Plant is 724,300 gallons. As shown in Table 11-2, 586,600 gallons, or 81% of the total quantity of liquid waste, is obtained from the Chemical and Volume Control System and passes through one of three purification ion exchangers. The activity of this portion of the clean liquid waste is assumed to be reduced by a decontamination factor (DF) of 10 for each nuclide except noble gases and tritium for which a DF of 1 has been assumed.

Four 50,000-gallon clean waste receiver tanks provide temporary storage for liquid waste inside the containment building. It is expected that a 30-day hold period will normally be possible at any time during the fuel cycle. Sufficient capacity is always available for at least a 7-day hold period which allows for significant natural decay of radioactive isotopes.

The three clean waste demineralizers contain a mixed bed H-OH resin. Activity of liquid waste passing through each demineralizer is assumed to be reduced by a DF of 10 for each nuclide except noble gases and tritium. Fission product gases are assumed to be 100% removed by the vacuum degassing, by diffusion in the clean waste receiver tanks and by the evaporators. The filters upstream of the demineralizers are sized to substantially remove insoluble corrosion products. However, no credit in system evaluation has been assumed for use of these filters nor for the filtering effect of the radwaste demineralizers.

The decontamination factor of the evaporator has been conservatively assumed to be 10^3 for iodine and 10^4 for all other isotopes except tritium for which 1 has been assumed. Suppliers of radwaste evaporators have observed decontamination factors greater than 10^6 in testing and actual operating experience.

The distillate from the evaporator is expected to have a maximum concentration of 10^{-4} μCi/cm^3 (except for tritium) and 10 ppm of boron. This effluent can be processed through a polishing demineralizer to further reduce the boron or can be sent directly to the primary coolant storage tank prior to reuse in the Plant, or nonnormal discharge to the discharge mixing basin.

During normal operation, tritium builds up in the Primary Coolant System and the primary coolant makeup water storage tank due to recycling of primary coolant leakage. During refueling, a portion of the primary coolant is mixed with SIRW tank water and spent fuel pool water causing the tritium concentration to increase in these volumes also. Assuming a 40-year Plant lifetime and zero primary coolant leakage, the maximum tritium concentration was found to be 2.8 μCi/cm^3 in the primary coolant; 2.2 μCi/cm^3 in the refueling cavity water; 2.4 μCi/cm^3 in the SIRW tank; and 1.0 μCi/cm^3 in the spent fuel pool water.

Additionally, the concentration of tritium in the building air due to evaporation from the refueling cavity and spent fuel pool is monitored on a periodic basis. Chemical and Radiological personnel review the monitoring results and ensure the appropriate radiation safety controls are implemented in accordance with 10CFR20.

The amount of tritium released each year (over the Plant lifetime) due to evaporation in the containment building and spent fuel pool building is shown in Table 11-9.

Using the annual value of X/Q of 4.64×10^{-6} s/m^3, the maximum concentration of tritium at the site boundary from these sources is 3.3×10^{-11} μCi/cm^3. This is 0.033% of the 10 CFR 20 limit of 1×10^{-7} μCi/cm^3.

In order to estimate the maximum tritium dose to the general public, a hypothetical rupture of the SIRW tank was considered. The SIRW tank was chosen because its volume (33,420 ft^3) is larger than that of the primary system makeup tank or the spent fuel pool volume. The tritium activity in the SIRW tank was taken to be 2.8 $\mu Ci/cm^3$, the maximum predicted reactor coolant concentration over a 40-year operating lifetime. Thus, the total quantity of tritium that would be released to Lake Michigan in the event of a hypothetical rupture is 2.6×10^3 curies.

The average dose to an individual in nearby South Haven consuming this water was calculated to be 0.056 mrem. The conservative assumptions which were used in calculating this number are as follows:

Dilution by a factor of 1,000 in Lake Michigan (see Reference 2)

Surface plume of 10-foot depth

Daily intake for South Haven water supply - 3×10^6 gallons (see Reference 3)

Current past water supply intake = 0.21 mi/h (see Reference 2)

South Haven water supply service area - 6.15×10^3 people (see Reference 3)

Average daily water consumption - 2,200 cm^3 (see Reference 4)

Subsequent to the above analysis, a revised value of the atmospheric dispersion parameter, X/Q, has been calculated based on new meteorological data and utilizing the Computer Code "XOQDOQ." The updated X/Q of 3.0×10^{-6} s/m^3 is approximately 65% of the value used in the above analysis, further attesting to the conservative nature of the results.

The concentration of radioactive materials in the SIRW tank or any temporary outside tank is limited to 1,000 times allowable effluent concentration. This limit may be exceeded for operational flexibility if the conditions of the Offsite Dose Calculation Manual (ODCM), section III.k are met. This limit is based on the dilution factor of 1,000 (before reaching the South Haven city water intake) assumed for tank ruptures. The bases for the limit originate in NUREG-0472, Revision 2, Radiological Effluent Technical Specifications for PWRs, July 1979. The specification of the limit has been changed to a concentration rather than the curie limit in NUREG-0472.

The bases for Liquid Holdup Tanks in NUREG-0472 reads; "Restricting the quantity of radioactive material contained in the specified tanks provides assurance that in the event of an uncontrolled release of the tanks' contents, the resulting concentrations would be less that the limits of 10 CFR 20, Appendix B, Table 2, Column 2, at the nearest potable water supply and the nearest surface water supply in an unrestricted area."

11.2.3.2 Dirty Waste Section

As with the clean waste system, the miscellaneous waste demineralizers are assumed to provide a DF of 10 except for noble gases and tritium. Similarly, the decontamination factor of the dirty waste evaporator is assumed to be 10^3 for iodine and 10^4 for all other isotopes except tritium.

Table 11-8 summarizes the performance of the dirty waste system (by isotope) under design operating conditions of 1% defective fuel. No credit has been assumed for any decay in the miscellaneous waste holdup tanks. This table shows that the effluent from the demineralizers which will be recycled to the utility water storage tank is of negligible activity.

11.2.3.3 Laundry Waste Section

The radioactive constituents in the laundry waste effluent discharge which cannot be treated by the dirty waste section are released to Lake Michigan at levels not to exceed 2.5×10^{-8} μCi/cm^3 on an annual average basis. A dilution flow of up to 60,000 gpm is provided by dilution pumps and cooling tower blowdown streams. The anticipated annual quantity of laundry waste was 38,000 gallons prior to operation of the new dry cleaning machine in early 1980. Since that time, limited discharges from the laundry waste section have been made which are well below the specified discharge limits. A DF of five is assumed for the filters.

The laundry release limit of 2.5×10^{-8} uCi/cm^3 was removed from the Technical Specifications in Amendment 85, effective 1/1/85.

11.2.4 BALANCE OF PLANT (BOP) INTERFACE

The Liquid Radioactive Waste System interfaces with the remainder of the Plant as follows:

11.2.4.1 Clean Waste Section

The clean waste section obtains wastes as bleed from the primary loop letdown line, or inflow from the primary system drain tank, and equipment drain tank, which deposit into the clean waste receiver tanks.

Processed liquid effluent is recycled to the Plant's Chemical and Volume Control System from the primary system makeup storage tank. Recyclable boric acid is similarly returned to the Chemical and Volume Control System from the recycled boric acid storage tank.

11.2.4.2 Dirty Waste Section

The dirty waste section obtains wastes from open sumps and drains which empty into the dirty waste drain tank.

Processed liquid effluent is normally recycled to the utility water storage tank after passing through the miscellaneous waste demineralizer. However, provisions are available to cross feed the effluent to the clean waste section, and ultimately to the Plant via the Chemical and Volume Control System if strict primary water quality conditions can be met.

11.2.4.3 Laundry Waste Section

The laundry waste section obtains waste from the laundry drain tank. System is available but not used. There is no longer a limit for direct release of laundry to the environment (see 11.2.3.3).

11.2.5 SYSTEM EVALUATION

Under normal operating conditions, the expected release to the environment from the clean and dirty liquid waste sections is zero. All clean and dirty wastes are recycled or accumulated and prepared for shipment in accordance with applicable rules, regulations and orders of governmental authorities having jurisdiction and turned over to a carrier or carriers licensed by governmental authorities having jurisdiction for shipment to an authorized disposal area or areas.

As shown in Figure 10-6, 11-1, and 11-5, any discharged radwastes are released to the discharge mixing basin after proper monitoring. These wastes are mixed and diluted and then overflowed from the mixing basin into Lake Michigan.

During normal operation, the only radioactive liquid released will be processed miscellaneous waste. During abnormal operation, some processed clean or miscellaneous waste might also be released. As liquids are released from the Plant to Lake Michigan, they will be diluted in the discharge from the Circulating Water System.

Under normal operating conditions, such as (but not limited to) steam generator tube leakages, fire, or pipe breakage, clean and dirty wastes will be discharged to the environment after processing to reduce the discharged radioactivity to levels which are as low as practicable, and in any event, in accordance with the objectives of 10 CFR 50, Appendix I.

To assess the potential impact of these releases, the maximum individual doses from liquid effluents were calculated assuming 100% release of the liquid effluents from the clean and dirty waste systems.

Maximum individual doses from liquid effluents were calculated by the NRC LADTAP computer code, using models given in Regulatory Guide 1.109 (March 1976) (see Reference 5). Dose factors, bioaccumulation factors and the shore-width factor as given in Regulatory Guide 1.109 and in the LADTAP code were used, as were use factors for water and fish ingestion and for water-related activities.

Radioactive liquid wastes from Palisades are discharged to Lake Michigan after dilution with circulating water system discharge. This flow is via low velocity surface discharge at the shoreline.

Maximum individual doses were calculated for water and fish ingestion and for external exposure for shoreline use, swimming and boating. The water ingestion pathway was analyzed for an individual drinking water from the nearest municipal water intake, located at South Haven. This location is 5 miles from the Plant, where a dilution factor of 1,000 was assumed to apply (see Reference 2).

For the fish pathway, an effective dilution factor of 15 was used based on the following factors:

1. Sport fish likely to be taken in the area, rainbow trout, brown trout, lake trout and salmon, migrate on the order of 0.9 to 7 miles per day (see Reference 6). Thus, it is unlikely that a fish would be exposed to an undiluted concentration on an average basis.

2. According to Reference 6, it is unlikely that fish residing in a plume show increased concentrations as a result of their presence, since the majority of their uptake occurs through a food chain. Because of the migratory nature of these fish, the effective dilution appropriate for the food chain must be much greater than that existing at the Plant discharge.

For the purpose of this analysis, a factor of 15 is believed to be suitably conservative. Source terms and additional assumptions used in this analysis are presented in Appendix 11A.

The maximum calculated doses to individuals from liquid effluents are summarized in Table 11-10, which also presents the pertinent LADTAP input data used in this analysis.

11.3 GASEOUS RADIOACTIVE WASTE SYSTEM

11.3.1 DESIGN BASIS

The design basis for the Gaseous Radioactive Waste System is to efficiently store gaseous isotopes for a time period which will permit sufficient radioactive decay prior to their discharge to the environment within limitations of the Offsite Dose Calculation Manual.

Design and construction codes for components and piping, and applicable nondestructive testing requirements are listed in Tables 11-3 and 11-4. The design criteria for the Gaseous Radioactive Waste System was identical to the liquid portion (Subsection 11.2.1.2) with the exception that the gas decay tanks added during the 1971-73 modification were CP Co Design Class 1, per Section 5.2.

11.3.2 SYSTEM DESCRIPTION

The Gaseous Radioactive Waste System is divided into two sections: (a) the gas collection header which collects low-activity gases from liquids which have been previously degassed and/or vented in other waste handling steps, and (b) the gas processing section which collects gases from potentially high-activity sources. The Gaseous Radioactive Waste System is shown in Figure 11-3. Component ratings and descriptions are shown in Tables 11-3 and 11-4.

11.3.2.1 Gas Collection Header

Gases which may contain potentially radioactive gases are passed through the gas collection header where they are discharged through a high-efficiency filter to the suction side of the main vent exhaust fans. The gases are diluted by ventilation exhaust air and are discharged through the ventilation stack to the atmosphere. The primary sources of low activity gaseous waste include the atmospheric vents of the liquid radwaste drain, collection and monitoring tanks, and containment building via "D" clean waste receiver tank (RUD-1018 removed). This vent path from containment to the collection header is an alternate path for venting containment. The primary containment vent path is through the "D" clean waste receiver tank to the radwaste area exhaust fans (V-14A&B).

11.3.2.2 Waste Gas Processing System

The waste gas processing system collects all potentially high-activity gaseous waste. The gas surge tank collects and absorbs surges from the demineralizer vents, shield and cooling surge tank vent, quench tank vent, primary system drain tank vent, volume control tank vent, vacuum degasifier vent (which takes input from either of two degasifier pumps) and evaporator vents. The waste gas surge tank discharges to one of three compressors which compress the gas for storage and decay in one or more of six waste decay tanks. If activities greater than 1×10^{-5} $\mu Ci/cm^3$ (Xe-133) have not been detected, the waste gas surge tank can be discharged through a high-efficiency filter directly to the ventilation stack.

If the surge tank is discharging directly to the ventilation stack, a high-radiation condition (as identified by a continuously operating monitoring system taking samples from the discharge line) will automatically close the discharge valve which is upstream of the stack. On simultaneous occurrence of this high-radiation signal and high surge tank pressure, a waste gas compressor starts automatically and, taking suction from the surge tank, discharges to the decay tanks.

Three of the six waste gas decay tanks have a volume of 100 cubic feet each and are designed for 120 psig. The remaining three decay tanks (added during the 1971-1973 auxiliary building addition) have a volume of 225 cubic feet each and are also designed for 120 psig. The total Primary Coolant System gas inventory and the gas inventory of the volume control tank can be stored in two of the smaller tanks if required for a cold, degassed Plant shutdown.

11.3.3 RADIOACTIVE RELEASES

As described in Subsection 11.3.2.2, gaseous effluents entering the Plant's ventilation stack are radiologically monitored and flow controlled so that 10 CFR 20, Appendix B, Table 2 values used for the design basis are not exceeded. This discharge is then immediately diluted by mixing airflow from one of the two continuously operating ventilation fans which conservatively transport 75,000 ft^3/min of air up the stack.

The waste gases are controlled to limit dose rates at the site boundary to the limits imposed by 10 CFR 50, Appendix I. The requirements of Appendix I to 10 CFR 50, including limiting conditions of operation and process monitoring operability and surveillance requirements were implemented by the Radiological Effluent Technical Specifications, Amendment No 85, of the Plant Technical Specifications (see Reference 10). The effluent specifications were then relocated to the Offsite Dose Calculation Manual - Appendix A per NRC Generic Letter 89-09.

A listing of gaseous isotopic activity at the Plant boundary expressed as a fraction of 10 CFR 20 values used for the design basis for no-holdup and 60-day holdup periods is given in Table 11-11. Further analysis pertaining to calculated maximum offsite activity levels and doses will be found in Subsection 11.3.5.

11.3.4 BOP INTERFACE

As shown in Figure 11.3, the Gaseous Radioactive Waste System interfaces with nuclear systems as it processes waste gases which empty into the gas collection header and waste gas surge tank.

The gaseous wastes leave the Plant in diluted form from the Plant's ventilation stack. These waste gases are either sent directly to the stack without holdup or are temporarily stored in any one of six waste gas decay tanks under pressure depending on their radioactivity and isotopic content.

True BOP interface occurs with the Component Cooling Water System because this system provides the heat sink for the sensible heat formed in the waste gas compressor heads and their respective aftercoolers while compressing the waste gases.

11.3.5 SYSTEM EVALUATION

Maximum individual doses from gaseous effluents were calculated by the NRC GASPAR (see Reference 7) computer code, using models given in Regulatory Guide 1.109 (March 1976). The basic source term and meteorological data entering into the calculations are described in Appendix 11A.

Calculations of maximum individual doses from gaseous effluents have been made for the following exposure pathways:

1. External doses due to cloud immersion

2. External exposure to materials deposited on the ground

3. Internal exposure via food chain pathways, including vegetation, meat, cow milk and goat milk

4. Internal exposure via inhalation

All standard or default GASPAR parameter values were utilized, including dose conversion factors, food intake rates, stable element transfer coefficients and time delays.

Computed doses include the open site terrain correction factor for the appropriate distance, as given in Regulatory Guide 1.111 (March 1976). The occupancy and shielding factor of 0.7, as given in Regulatory Guide 1.109, was applied.

Maximum offsite air doses were determined, among overland locations, to be 0.95 mrem/yr for beta radiation and 0.36 mrem/yr gamma radiation at 0.48 mile in both the SSE and SSW directions.

Meteorological dispersion and deposition data were reviewed in conjunction with data pertaining to nearest residences, vegetable gardens, milk and meat animals within five miles to determined locations where specific exposure pathways would result in maximum doses. The GASPAR input data used to analyze doses at the locations so identified are presented in Table 11-12. Dose results for each location are presented in Table 11-13.

Including both the plume and ground contamination doses, the highest computed external dose rates are about 0.143 mrem/yr to the total body and 0.34 mrem/yr to the skin (both at 0.63 mile S). The highest computed dose due to nonnoble gas isotopes is 2.3 mrem/yr to the thyroid of a child (0.88 mile ENE at the nearest garden). Actual annual gaseous release data, analyzed by GASPAR using 5 year average, meteorological data along with yearly land use data, has demonstrated that gaseous effluents are a small fraction of 10 CFR 50, Appendix I limits.

11.4 SOLID WASTE MANAGEMENT SYSTEM

11.4.1 DESIGN BASIS

The design basis for the Solid Waste Management System incorporates the applicable regulatory requirements, including the following:

Title 10, Part 20 (10 CFR 20), "Standards for Protection Against Radiation"

Title 10, Part 50 (10 CFR 50), "Domestic Licensing of Production and Utilization Facilities"

Title 10, Part 50, Appendix I (10 CFR 50, Appendix I), "Numerical Guides for Design Objectives and Limiting Conditions for Operation Guides To Meet the Criterion as Low as is Reasonably Achievable for Radioactive Material in Light Water-Cooled Nuclear Power Reactor Effluents"

Title 10, Part 70 (10 CFR 70), "Domestic Licensing of Special Nuclear Material"

Title 10, Part 71 (10 CFR 71), "Packaging of Radioactive Material for Transport and Transportation of Radioactive Material Under Certain Conditions"

Title 49, Parts 170 and 171 (49 CFR 170 and 171), "Department of Transportation (DOT) Hazardous Materials Regulations"

NRC Information Notice No. 90-09, "Extended Interim Storage of Low Level Radioactive Waste by Fuel Cycle and Materials Licensees"(2/90).

NRC Generic Letter 81-38, "Storage of Low Level Radioactive Wastes at Power Reactor Sites"(11/81).

NRC Generic Letter 85-14, "Commercial Storage at Power Reactor Sites of Low Level Radioactive Waste Not Generated by the Utility"(8/85).

IE Circular No. 80-18, "10 CFR 50.59 Safety Evaluations for Changes to Radioactive Waste Treatment Systems"(8/80).

Design and construction codes used in this system are generally identical with those depicted in Subsection 11.2.1.3 for Quality Group D system components and piping. In addition, the asphalt volume reduction (VRS) system processing equipment is located in a CP Co Design Class 1 building. Onsite buildings, other than the Auxiliary and Service buildings used to process and store radioactive waste are engineered structures, but are not seismically qualified.

Spent nuclear fuel is not covered by this section. Discussion on this subject will be found in Section 9.11 (Fuel Handling and Storage).

11.4.2 SYSTEM DESCRIPTION

The Solid Waste Management System is designed to collect, process, package and store for future offsite disposal low-level liquid wastes (consisting of evaporator concentrates), spent ion-exchange resins and assorted solid wastes according to type of waste and levels of radiation activity present. The system is shown in Figure 11-4.

11.4.2.1 Original System

Until the auxiliary building addition was added to the auxiliary building in 1972-1973, the original system was designed to process and package only dewatered spent ion-exchange resins which were sluiced into the 400-cubic-foot spent resin storage tank. Liquid wastes were discharged to Lake Michigan via the Plant's discharge mixing basin, and miscellaneous solid wastes were shipped offsite in boxes or drums depending on whether or not the solid wastes could be compacted into 55-gallon drums. Dewatered spent resin and expended filter cartridges were shipped from the Plant site in metal drums until 1973.

11.4.2.2 1972-1973 Modification

The quantities of wastes processed by the system were increased when modifications to the liquid and solid waste systems occurred during the 1972-1973 service building addition. These changes were brought about by legal commitments to change the Plant to a "near-zero" release Plant. These were the addition of ion-exchange resins obtained from the radwaste polishing demineralizer and the liquid concentrates (or "Bottoms") obtained as a by-product from the clean and dirty radwaste evaporators. However, concentrates from the clean radwaste evaporator normally do not contribute to the waste stream, whereas concentrates from the dirty radwaste evaporator are always processed by the solid waste management system. Also, a second spent resin storage tank having a capacity of 200 cubic feet was installed to provide holdup time which would permit natural decay prior to the packaging operation.

After 1973 and until late 1980, processing and packaging of evaporator concentrates was accomplished by use of a solidification system designed by Protective Packaging, Inc (PPI). This system metered evaporator concentrates with a precise amount of liquid urea-formaldehyde polymer solution (such as Cyanaloc-62 (American Cyanamid) or equivalent) into disposable liners of approximately 50-cubic-foot capacity.

The PPI urea-formaldehyde-type solidification system was used extensively until it was dismantled in late 1980 to facilitate installation of another type of solidification system. The new solidification system utilizes molten asphalt as the immobilizing constituent to be mixed with the waste concentrates or spent ion-exchange resin. The new system was installed in the same packaging room formerly used, but extends eastward into a series of engineered rooms constructed above the former outdoor patio-roof area of the service building addition at elevation 625 feet 0 inch.

11.4.2.3 Interim Solid Waste System

In the interim period before start-up of the new solidification system, spent resins and filters were shipped offsite in dewatered state in approved high integrity containers manufactured from ultra-high molecular weight polyethylene. Liquid concentrates were physically transported in shielded 50-cubic-foot liners by vehicle to a temporary packaging area in the north end of the feedwater purity building. Processing and solidification of the concentrates were accomplished by commercial waste contractors utilizing a proprietary formula cement grout as the immobilizing constituent. Filter cartridges continued to be packaged damp dry as previously described. In 1983, the 55-gallon drum dry waste compactor was replaced with a box compactor designed to compress miscellaneous dry solids into 96-cubic-foot steel boxes.

11.4.2.4 Volume Reduction and Solidification System

The Werner & Pfleiderer volume reduction and solidification (VRS) system installed in 1983 utilizes one VRS-T120 extruder/evaporator to process the concentrates from the boric acid and liquid waste evaporators and the bead form ion-exchange resin. The system also has connections for processing powdered ion-exchange resin from the feedwater purity building if these materials become a source of radioactive waste. The system is shown in Figure 11-4 and the process equipment is described in Table 11-14.

The VRS system is a one-step process for reducing the volume of radwaste while incorporating the radwaste into a solidified asphalt matrix. Liquid radwastes or resin slurries and the asphalt binder are metered to a steam-heated twin-screw extruder/evaporator. As the radwaste moves through the extruder, the water is evaporated from the waste while the waste solids are simultaneously reduced to micron-sized particles and homogeneously dispersed in the asphalt. The asphalt and waste mixture is discharged from the extruder to standard waste shipping containers and the product solidifies upon cooling. The system does not require the addition of chemical agents to promote solidification and no free water is present in the final product.

Asphalt Storage and Supply System

The asphalt storage and supply system includes a 9,000-gallon bulk storage tank equipped with external steam heating panels. The temperature of the asphalt in the tank is maintained at approximately 300°F so that the asphalt in the tank is a pumpable liquid. The tank is insulated to reduce heat losses and a metal enclosure is provided around the tank for wind and snow protection. A three-hour fire wall is located between the asphalt tank and the auxiliary building addition, and a dike is provided around the tank to contain the asphalt in the event of a leak or overfilling of the tank.

A metering pump feeds the required amount of asphalt to the extruder/ evaporator based on the type of radwaste being processed and the solids content of the waste. Feed of asphalt to the VRS system is interlocked with operation of the extruder/evaporator. The entire asphalt transfer system is supplied with steam heat tracing to maintain proper temperatures. The steam is supplied by an electric boiler.

Waste Collection and Feed System

The system is provided with the capability of collecting, concentrating and feeding bead resin. An evaporator concentrates pH adjustment and feed system is also provided. Connections for processing powdered resins are included with the VRS system. A spent filter handling system is not provided, but encapsulation of the filters with asphalt can be performed once the filters are delivered to the VRS system and placed in an empty container.

A spent bead resin tank is provided for handling resin slurries. The tank has a useful capacity of 1,500 gallons. When used in conjunction with the existing resin storage tank, sufficient capacity exists for approximately on year's quantity of resin. The use of the two resin tanks also permits segregation of high specific activity resins from low specific activity materials and, therefore, more effective use of waste containers.

A positive displacement progressing cavity metering pump feeds controlled amounts of the resin slurry to the extruder/evaporator. The metering flow rate is established manually based on the composition of the waste stream. The asphalt flow rate is automatically adjusted to the proper ratio of asphalt to solids in the end product.

A positive displacement progressing cavity metering pump (P-119) feeds controlled amounts of evaporator concentrates from the recirculation line to the extruder/evaporator. All lines and components are electrically heat traced to maintain the fluid at 160°F to prevent crystallization of the waste salts in the system. Flushing connections are also provided to reduce radiation levels after a batch of concentrates has been processed. Concentrates feed is interlocked with operation of the extruder/evaporator, the asphalt feed system and the concentrates recirculation system.

Variations of pH will have little or no effect on the consistency or solidification of the final product. However, in order to minimize potential corrosion of all equipment, waste streams are adjusted to a pH greater than 7.0 prior to their introduction to the VRS system. A pH monitoring system is provided in the concentrates recirculation loop on the suction side of the concentrates metering pump. Caustic (25% solution) is added to a recirculation loop on the discharge side of the metering pump in controlled amounts. The caustic will then be mixed with the waste stream in the concentrates tank. The batch addition of caustic will be continued until the required pH is achieved in the recirculation line, after which metering of the concentrates to the extruder/evaporator is permissible.

Extruder/Evaporator

Waste processing takes place in the extruder/evaporator. The feed materials (asphalt and waste) enter the extruder in the first barrel section where they are immediately combined by the mixing action of the intermeshing screws. The corotating twin screws are driven by a variable-speed motor within the range of 30 to 300 r/min. The design of the screws, material selection and regulation of imposed torque have been developed considering processing efficiency and component wear.

Steam from an auxiliary boiler heats the extruder, thus heating the feed materials to a temperature sufficient to evaporate the free water in the waste feed. The steam input is controlled to maintain a conservatively safe temperature below the flammability point of asphalt. Several separate heat zones are provided along the length of the process section to provide for control of the evaporation process. As the water evaporates, the solids that remain in the extruder are kneaded to microscopic size particles which are individually coated and homogeneously dispersed in asphalt. The process section of the extruder/evaporator operates near ambient pressure. The asphalt/waste mix is continuously discharged into a container where solidification occurs as the mass cools. The free water evaporated from the wastes is condensed in the steam dome coolers which are integral parts of the extruder/evaporator. The condensed liquid is drained by gravity to the distillate collection system.

Container Filling System

The extruder/evaporator discharges the asphalt/radsalt mixture into standard 55-gallon carbon steel DOT drums. The empty drums are placed on a chain-driven roller conveyor system for filling. The conveyor is controlled to locate a drum directly beneath the extruder/evaporator discharge port. Use of a reversible conveyor permits more than one pass filling without operator handling of containers during fill operations.

The filling operation can be monitored by the operator who has visual access through shielded windows and television cameras. Level in the drum being filled is monitored by two independent level detectors. The first detector indicates when a preselected drum level is reached and the second has two additional higher set positions. In this manner, the operator can terminate the filling of the drum at a prescribed level or allow the filling to continue until the level control system terminates filling. In addition, a timer cycle will provide for a sequential shutdown of the system if the fill cycle exceeds a preselected time period.

Container Handling System

The empty 55-gallon containers are manually placed on an accumulation section of an indexing chain conveyor. The drums are then fed by a pop-up roller transfer to a chain-driven roller conveyor. The roller conveyor positions an empty drum under the extruder/evaporator outlet. When full, the drum is moved on the roller conveyor to the cap and swipe station. This station has a rising support which rotates and automatically turns the drum during the swipe operation. The drums are finally removed from the conveyor and loaded onto pallets.

Vapor Control

Vapors in the fill area are removed via the ventilation hood installed around the extruder/evaporator discharge port. Air flow between the drum lip and vent hood periphery is maintained at 100 ft/min by the blower in the exhaust system. The vapors are passed over a mesh separator, an HEPA filter and a charcoal filter prior to being discharged through the normal Plant vent.

11.4.2.5 Radioactive Waste Storage Facilities

Michigan was denied access to the three existing burial sites in November 1990. Michigan has also been expelled from the Midwest Compact. The Palisades Plant will need to temporarily store radwaste until Michigan builds a disposal facility. It is assumed Palisades will have to store radioactive waste onsite at least through 2007. It is proposed to use existing buildings at East and South locations as well as a new building at East location for the storage of radioactive waste. In mid 1995, the Barnwell, S. C. waste disposal site lifted the ban on Michigan radioactive waste and shipping activities resumed. The majority of the radioactive waste stored on site was disposed of at the waste facility. The south radwaste building was emptied of waste and released for other activities.

Estimated Volumes and Activities

Approximately 2600 Curies of activity, dispersed in 2500 cubic feet of solid wastes are generated from the plant site in a normal year (ie, a year in which there are no extended outages or major nuclear repair work being performed).

These wastes can be separated into the following categories:

	Ci/yr	Ft^3/yr
Expended Filter Cartridges	25	200
Dewatered Ion-exchange Resin-shielded	50	200
Solidified Evaporator Concentrates	10	300
Dry Active Waste	10	1000
Irradiated Hardware	2500	10
Dewatered Ion-exchange Resin-unshielded	0.1	800

Packaging

Solid wastes not being solidified by addition of immobilizing additives are processed as follows prior to offsite shipment:

1. Secondary side spent ion-exchange resin is packaged in liners, and dewatered to the turbine building sump or Auxiliary Building prior to shipment/storage.

2. Primary side spent ion-exchange resin is packaged in high integrity containers (HIC), dewatered in Auxiliary Building and transported/transferred in specially designed shipping casks.

3. Dry active wastes such as contaminated clothing, rags, paper, towels, gloves, shoe coverings, plastics, wood and metal could be compressed into 94 cubic foot metal boxes by a B-400 Supercompactor, but preferably are shipped in sea-land containers to a vendor for volume reduction (incineration/supercompaction).

4. Expended filter cartridges are drained then transported to the shielded storage area in the East Radwaste building in specially designed casks. Filters are then transferred to HICs located in shielded vaults.

5. Irradiated hardware is dewatered in Auxiliary Building and stored.

6. Small amounts of liquid mixed waste and contaminated oil are stored in overpacks containing approved burial ground absorbents.

Facility Description

1. The East Radwaste Facility consists of two adjacent buildings connected by a shared roll-up door. Radioactive waste (bags, filters, wood, metal, etc.) is transported to the East Facility to be processed. Dewatered resin and evaporator concentrates are packaged to meet criteria for dry waste prior to leaving the Auxiliary Building. The Dry Active Waste (DAW) is usually transported in a covered trash wagon used exclusively for this purpose.

The West building is currently used primarily for storage of cement rad vaults for storage of packaged resin and filter HICs. This building is also used for temporary storage of large contaminated or retired plant equipment awaiting processing or packaging.

The East building is primarily used for processing Dry Active Waste. The building is serviced by a 2000 cfm HEPA ventilation unit which is operated during processing operations. This building also has a separate (HEPA Ventilated) room for dismantling large contaminated components retired by the plant. This building is also equipped with a B-400 Supercompactor for compressing DAW. This compaction unit has a self-contained HEPA ventilation unit that provides local contamination control.

The East building also contains a built-in cement shielded vault system for storage of high level filters, resin and DAW. All items placed in the vaults for storage are packaged in High Integrity Containers to maintain a contamination free area. This vault system is sectioned into two areas, one with 24" of concrete shielding and the other with 36" of shielding which maintains area radiation levels to near background levels.

2. The South Radwaste building is a 40' x 80' engineered steel building. In January 1992 the main floor of the building was elevated 24 inches (18 inches compacted sand with 6 inches cement cap) to prevent water intrusion from flooding, cooling tower overflows or excessive rainfall.

   This building is being released for other activities, but will be again designated as a storage building if waste disposal privileges are again denied. The required container surveillance and radiation area monitoring to show compliance with Generic Letter 81-38 will be instituted if used as a storage building. If so, this building will be used only to store DAW in metal boxes and steel drums, incinerator ash in HICs and solidified or dewatered evaporator concentrate in steel drums or HICs, packaged in accordance with NRC, DOT and burial site requirements. These metal boxes are to be stored around the outer walls of the main floor. Every box is equipped with risers (feet) to allow containers to be raised off the floor to prevent inadvertent water accumulation to cause external corrosion and possible degradation of container integrity. A series of 5" cement DAW rad vaults will be placed in the center section of the building to store solidified or dewatered evaporator concentrate drums or HICs. The incinerator ash will also be stored in the center section of the building.

Radiological Consequences

1. Gaseous--Accident releases from the radioactive waste storage buildings are not considered credible because of material, packaging and steel and/or cement shielding. However, to show compliance with Generic Letter 81-38 criteria, three accident cases were run as well as direct dose calculations to the site boundary. The accident cases were a small fraction of 10CFR100 limits and direct dose to the site boundary was less than 1 mrem/yr as required by Generic Letter 81-38.

2. Liquid--There are no liquid effluent consequences in radwaste building because all waste in the building meets dry radioactive material status except for a small amount of liquid mixed waste and some contaminated oil. This waste will be over-packed with absorbent material and in no event could the criterion per IE Circular 80-18 of MPC at the nearest water supply be approached.

Containers

These containers are selected based on structural strength, the ability to maintain container integrity during processing, packaging, storage and transportation. They will also demonstrate minimal corrosion effects from exposure to internal environment over a long period of time. All containers awaiting shipping to be buried, are stored inside the building to protect against corrosion from external environment. These containers comply with the requirements of 10CFR71 and 49CFR as well as burial ground criteria to prevent the need for repackaging prior to shipment. HIC lids are equipped with passive vents to allow depressurization of hydrogen, but do not permit migration of radioactive material. Additional semi-portable cement rad vaults to shield filled HICs will be placed in East Radwaste Building on an as-needed basis.

Monitoring Equipment

1. Area Monitor--The east building will contain an area monitor calibrated to read out in mR/hr (equivalent to mrem/hr). This area monitor provides local alarm and will initiate a phone alarm to the control room when dose rates in the area reach or exceed alarm set points.

2. Air Monitoring--The storage area at the east radwaste building is equipped with a continuous air monitor. This monitor has an adjustable visual and audible alarm. The air monitor alarm will also initiate a phone alarm to the control room.

Effluent Monitoring

The processing/sorting area at the east radwaste facility is equipped with a portable 2000 cfm ventilation unit with HEPA filter. The ventilation exhaust is equipped with a sample collection system. This system consists of a flow meter, vacuum pump and particulate filter sampler.

ALARA

Purification filter (F-54), activated incores and other higher level material will be stored in 36 inch concrete vaults in east radwaste building. Higher level DAW, resin (not T-104) and filters will be stored in 18 inch thick semi-portable concrete vaults. The design resin liner would have a 12 inch reading of 20.1 mrem/hr in the 18 inch vault (Reference 1). One inch steel donut shield is available if concrete vault is not adequate. Concentrate asphalt drums or HICs will be stored in a five inch concrete, or equivalent, vault. DAW boxes will be stacked with lower reading boxes on the outside to minimize dose outside the radwaste areas. These shielding materials and methods are provided to address ALARA principles and maintain low radworker exposure.

Volume Reduction

Volume reduction techniques are being used on a full time basis. An asphalt extruder is used to volume reduce evaporator concentrate waste. Our 400,000 lb. ram force compactor is close to state-of-the-art. Wood planing and grit blasting are used to decontaminate wood and metal. Vendors process most trash and wood (incineration), some metal (smelting) and return unprocessable waste and residue to the site for storage. The waste returned will be packaged to meet NRC, DOT and burial site requirements prior to storage.

11.4.3 RADIOACTIVE RELEASES

All liquid discharged to the environment from the Liquid Radioactive Waste System and solid wastes leaving the Solid Waste Management System are accounted for by quantity, activity and isotope inventory and are reported to the NRC on an annual basis. Packaged and immobilized low-level wastes meeting all applicable state and federal regulations could be shipped to licensed shallow-land burial grounds for internment. Waste shipments are made primarily by licensed commercial motor vehicles. All such shipments are reported to the NRC on an annual basis.

11.4.4 BOP INTERFACE

The distillate from the extruder/evaporator is cooled and filtered to remove trace quantities of organics. The distillate is then piped to existing floor drains in the process area for collection in the Liquid Radioactive Waste System. Makeup for the distillate tank is obtained from the utility water storage tank.

Primary makeup water for the auxiliary boiler system and the closed loop cooling water system is obtained from the primary system makeup tank and the demineralized water storage tank.

Service water is supplied to remove heat from the closed loop cooling water system.

Building services required are HVAC, fire protection and electrical power.

11.4.5 SYSTEM EVALUATION

The portion of the Solid Waste Management System which processes liquid concentrates and spent ion-exchange resin is comprised of pressure retaining components to assure containment of these waste streams.

The NRC has reviewed the process control for the VRS system and has concluded that the system meets the existing requirements for safety, reliability and end product characteristics. The solidified product has been accepted for transportation and burial. Solid wastes and compacted wastes are packaged to standards which will permit similar shipment offsite, although these wastes are not immobilized by introduction of an outside constituent.

The Solid Waste Management System is capable of meeting all criteria listed in Subsection 11.4.1, "Design Basis."

11.4.6 REQUEST TO RETAIN SOIL IN ACCORDANCE WITH 10CFR 20.302 (10CFR20.302 was in affect when this commitment was made.)

Consumers Power Company correspondence dated November 12, 1987 and January 25, 1988 requested authorization to dispose of contaminated soil in place as specified by 10CFR 20.302. The area known as the South Radwaste Area has been contaminated by numerous cooling tower overflows and contamination was redistributed by heavy rain showers. Although the majority of the radioactive material has been packed for waste shipment, a large volume of very low activity radioactive material remains. This volume of material would be very expensive to ship as waste.

The specific area contaminated is noted as Area B on the survey grid map in Reference 11. The entire area is fenced and is about 12,000 sq ft of soil exposed with the remainder buildings and asphalt. The inhalation pathway is for breathing suspended soil from this area. The radworker could receive 8.03E-04 mrem/50year maximum organ (liver) dose and the infant could receive 3.16E-05 mrem/50year maximum organ (liver) dose, both of which are insignificant. Direct dose to a radworker is less than 2E-03 mrem/hr. Occupancy in this area should not average more than 2 hours/week or 100 hours/year, which would result in a dose of <1 mrem/year.

The radwaste activities which caused the contamination of the soil were completely relocated or have been replaced with other methodology. The South Building has been deconned and is being used for other activities. Some fixed contamination is present in floor cracks and vaults. This has been documented for plant decommissioning.

The waste in this building has been shipped for disposal and the building released for other activities.

11.5 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEM

The process radiation and effluent radiological monitoring and sampling system was designed to assure that ionizing radiation levels are indicated and alarmed so that action, either automatic or manual, can be taken to prevent radioactive release from exceeding the limits of 10 CFR 20.

Detection devices are located in the various process systems and at selected positions throughout the containment and auxiliary buildings to monitor radiation levels and annunciate any abnormally high radiation activity. Instrument ranges and sensitivities are chosen to enable monitoring within the requirements of 10 CFR 20.

11.5.1 DESIGN BASIS

Radiation monitoring devices using proven photomultiplier, scintillation and geiger-type ionization detection chambers have been utilized in the process and effluent radiological monitoring and sampling system.

Each process instrument has been selected according to the type of radiation to be measured, and sized to encompass the entire range of radiation activity which corresponds to the Plant's design power levels and failed fuel criteria. A listing of the process sample points, instrument sensitivity, and other information pertaining to these instruments is presented in Table 11-15 and shown in Figures 9-23, 11-5 and 11-6.

Additionally, all monitors in the stack-gas, containment air, off-gas, waste gas, engineered safeguards areas ventilating system discharge, radwaste ventilation and radwaste liquid discharge systems have been supplied with check sources. The check source is to simulate a radioactive sample and serve as a check for both the readout and detector.

11.5.2 SYSTEM DESCRIPTION

The process and effluent radiological monitoring and sampling system is a collection of radiological instrumentation and, therefore, cannot be described as a unique system by itself except for the stack monitoring subsystem (described in Subsection 11.5.3).

The process sampling systems and area radiation monitors, individual detectors, power supplies, and readout devices are designed to operate with 0% to at least 95% relative humidity. Those devices located outdoors will withstand temperatures from -10°F to 110°F and those indoors will withstand 50°F to 104°F. In addition, those components required to operate in a DBA environment are designed for 100% relative humidity at 283°F.

Provisions are included to permit periodic testing while detection equipment is operational.

The aforementioned detection devices display their information in radiation monitoring equipment panels located inside the main control room. The panels provide mounting for indicators, power supplies and alarms for each of these radiation monitoring systems. Two of the panels are located beside the area radiation monitoring panel. The process liquids radiation monitoring panel and the gas radiation monitoring panel are fed by the instrument ac bus which, in the event of a loss of power, is fed by the diesel generators. The circulating water discharge monitor sample pump is powered from a normal lighting panel, with a low flow alarm powered from the instrument ac bus.

The type of detectors used and the information displayed are listed in Table 11-15. The sensitivity and alarm conditions for each instrument are also listed.

11.5.3 EFFLUENT MONITORING AND SAMPLING

Liquid effluents which are discharged from the Plant are monitored by a process sampling detector located in the circulating water discharge structure. The process sample is obtained from a continuously flowing (freeze protected) sample loop which is part of the monitoring system.

In 1983 a main steam relief monitoring system was installed to monitor accident releases in the event the atmospheric dump or safety valves lift. Two monitors, one viewing each main steam line, continuously monitor and record the activity present in the secondary steam. In the event of a steam release, an acoustic switch, triggered by the high noise level, automatically switches the recorder to a higher speed for greater resolution.

Gaseous effluents leaving the Plant via the stack discharge system are described by the stack monitoring system (Subsections 11.5.3.1 and 11.5.3.2). Abnormal gaseous releases detected by any of the process or area radiation monitors within the radiation controlled areas of the containment and auxiliary buildings are processed by engineered ventilation systems which ultimately discharge to the Plant's stack.

Also in 1983 the radioactive gaseous effluent monitoring system (RGEMS) was added in parallel to the original system. The RGEMS extends the monitored range of the stack effluent and provides capability for rapid filter change out.

11.5.3.1 Original Stack Monitoring System

Prior to 1983 the stack monitoring system consisted of an isokinetic nozzle, dual particulate samplers, flow control valve, pump, gas monitoring channel and a flow indicator/transmitter. This equipment is retired in place as obsolete and is no longer used as backup to RGEMS. Grab sampling is now used as a backup.

11.5.3.2 Radioactive Gaseous Effluent Monitoring System (RGEMS)

The RGEMS, installed in 1983, consists of normal range particulate/radioiodine filters, NaI gamma detector, scintillation chamber beta detector, and an accident range filter and ion chamber (refer to Figure 11-6). Flow through the system is provided by two 100% capacity diaphragm vacuum pumps. The flow is controlled by automatic flow control valves to maintain a constant flow rate of 2 scfm through the system.

During normal operation, 2 scfm of the stack effluent is routed through a particulate/radioiodine filter then through the beta detector. The filter is continuously monitored by the NaI detector to detect any buildup on the filter. The filter is changed and counted on a regular basis by Plant personnel.

On indication of abnormal stack effluent activity (alert level), a 15-second grab sample is automatically trapped in a sample bottle and an annunciator in the control room indicated the off-normal condition.

Following a high level indication, the normal sample loop is bypassed and the sample flow is split with approximately 0.02 scfm directed through the high-range filter and the balance of the 2 scfm through the ion chamber. To avert a too rapid buildup of activity on the filter, the capability exists to interrupt the sample flow through the filter periodically for periods ranging from 6 seconds to 54 seconds every minute. The continuous monitoring capability of the high-range noble gas monitor is not affected during filter flow interruption. A "high radiation" annunciator in the control room alerts the Plant operators to the condition.

Systems may be controlled either locally or remotely from the control room. Dual microprocessor controllers provide system control through the normal, alert and high operating modes. Normally the controller located in the control room provides full system control. In the event of failure the local controller takes control of the system functions.

Refer to Table 11-15 for details of the monitors.

11.5.4 SYSTEM EVALUATION

All process systems which contribute to Plant discharges are monitored prior to entering the various discharge systems. Each discharge system is also monitored, providing redundancy of radiation detection for Plant effluents. The radwaste area, containment air, waste gas, engineered safeguards pump room, and the off-gas radiation monitoring systems are backed up by the stack-gas monitoring system. The service water, radwaste liquids discharge, component cooling and the system generator blowdown radiation monitoring systems are backed up by the circulating water discharge monitor.

Testing and maintenance for all systems, circuit testing of readout equipment and power supplies can be performed from the panels located in the control room. The circuit being tested or repaired is inoperative during that time and acts as if it were a tripped channel. The containment high-radiation monitors are continuously monitored while in service for loss of power, loss of detector high voltage and for loss of detector signal.

11.6 RADIATION PROTECTION

11.6.1 GENERAL

11.6.1.1 Radiation Exposure of Personnel

The basis for the shielding design for normal Plant operation is the Title 10, Code of Federal Regulations, Part 20 (10 CFR 20), entitled "Standards for Protection Against Radiation." The exposure of workers to licensed radioactive materials is limited to values in 10 CFR 20.

All areas of the Plant are subject to these regulations. The areas are zoned according to their expected occupancy by Plant personnel and their designed radiation exposure level under normal operating conditions.

Allowable design for all accessible areas of the Plant is not to exceed a total effective dose equivalent of 5 rems per calendar year. For all areas outside the Plant, the total effective dose equivalent should not exceed 0.1 rem per calendar year.

The control room shielding design provides adequate protection during a Maximum Hypothetical Accident (MHA) such that the accumulated dose will be within 10 CFR 50 Appendix A, GDC-19 limits as long as an individual remains in the control room. For exposures obtained during Plant ingress and egress as well as during excursions to selected Plant areas, limits of 10 CFR 100 will not be exceeded.

11.6.1.2 Radiation Exposure of Materials and Components

No regulations similar to those established for the protection of individuals existed at the time of FSAR preparation for materials and components. Materials were selected on the basis that radiation exposure would not cause significant changes in their physical properties which would adversely affect their operation during the design life of the Plant. Materials for equipment required to operate under accident conditions were selected on the basis of the additional exposure received through the time of their required operation in the event of an MHA.

In 1980, an analysis was performed to qualify safety-related electrical equipment for operation in a radiation environment following an accident. See Subsection 8.1.3.

11.6.2 RADIATION ZONING AND ACCESS CONTROL

The radiation zoning of the Plant areas established in the plant design is shown on Figures 11-7, 11-8 and 11-9.

The following list identifies the different zones used for the design of the Palisades Plant (the terms "controlled" and "uncontrolled" are used in a generic sense to describe access controls):

Zone Designation	Design Dose Rate (mrem/h on a 40 h/Week Basis)	Description
I	≤ 0.5	Uncontrolled, unlimited access
II	≤ 1.5	Controlled, unlimited access. 40 h/week
III	≤ 15.0	Controlled, limited access for routine tasks. 6 h/week
IV	≤ 100	Controlled, limited access for short periods. 1 h/week
V	> 100	Controlled occupancy for very short periods. Occupancy during emergencies. Normally inaccessible

UNCONTROLLED areas are those that can be occupied by Plant personnel or visitors on an unlimited time basis with a minimum probability of health hazard from radiation exposure.

CONTROLLED areas are those where higher radiation levels and/or radioactive contamination which have a greater probability of radiation health hazard to individuals can be expected. Only individuals directly involved in the operation of the Plant will, in general, be allowed to enter these areas.

ACCESSIBLE areas are those that will encounter radiation dose rates of less than 100 mrem/h and which can be entered either through open passages or unlocked doors. These areas can be entered by all individuals who have passed through the Plant access control station.

INACCESSIBLE areas are those where dose rates above 1,000 mrem/h can be expected. These areas are either blocked off completely or can be entered only through locked doors. Access is limited to an intermediate degree to areas where dose rates are between 100 mrem/h and 1,000 mrem/h. Access to areas greater than 1,000 mrem/hr is controlled by Radiation Safety personnel or, or in an emergency, by Operations supervision.

Access restrictions for controlled areas may be enforced by removable concrete shielding blocks, locked doors, chains, etc. Access is supervised commensurate with radiation health risk. Areas with dose rates in excess of 1000 mrem/hr may be barricaded, posted, and a flashing light used as a warning device consistent with Technical Specifications 6.7.3.

11.6.3 GENERAL DESIGN CONSIDERATIONS

The shielding design considers three conditions:

1. Normal full power operation. This also includes shielding requirements for certain off-normal conditions such as the release of fission products from leaking fuel elements.

2. Shutdown. This condition deals mainly with the radioactivity from the subcritical reactor core, with radiation from spent fuel bundles during onsite transfer, and with the residual activity in the reactor coolant and neutron-activated materials.

3. Maximum hypothetical accident. This includes design to reduce radiation streaming from the containment purge penetrations, personnel airlock, equipment lock and other accident sources. This permits access to areas required for emergency operations.

11.6.3.1 Specific Design Values

The material used for most of the Plant shielding is ordinary concrete with a bulk density of 144 lb/ft^3. Only in a very few instances has steel or water been utilized as primary shielding material.

11.6.3.2 Reactor Core Data

The reactor core power rating was assumed to be 2,650 MWt with 1% failed fuel and an average coolant velocity of 12.8 ft/s in the core. The core and fuel dimensions are discussed in Chapter 3.

11.6.4 SHIELDING DESIGN

11.6.4.1 Containment Building Shell

The containment shell serves two main shielding purposes:

1. During normal operation, it shields the surrounding Plant structures and yard areas from radiation originating at the reactor vessel and the primary loop components. Together with additional shielding inside the containment, the concrete shell will reduce radiation levels outside the shell to below 0.5 mrem/h in those areas which are occupied by personnel either on a permanent or routine basis.

2. In the event of a maximum hypothetical accident, the shell shielding will reduce Plant and offsite radiation intensities, emitted directly from released fission products, to acceptable emergency levels. The concrete roof of the containment will effectively reduce contributions due to sky-shine. The environmental consequences associated with an MHA are discussed in Section 14.22.

11.6.4.2 Containment Building Interior

During operation, most areas inside the containment will be inaccessible because of dose rates greater than 100 mrem/h. The containment air room is an exception. Personnel are able to enter this small portion of the containment which houses equipment and instruments that need inspection during operation. Shielding surrounding the air space (space adjacent to personnel hatch) keeps radiation levels down to less than 15 mrem/h.

Large sections over the steam generator chambers and the reactor transfer and storage pool are open and unshielded. These openings cause a high dose rate at the refueling floor. Neutrons streaming out of unfilled transfer and storage pool increase the containment internal dose rate. The missile shield over the reactor and the center part of the pool restricts these exposures in other areas. The reactor vessel which is the major radiation source is surrounded by a concrete shield.

Additional concrete shielding is provided around equipment that carries primary coolant water. The shielding is governed by strong N^{16} decay gamma radiation. Extra shielding is added for protection at the entrance to the containment air space (space adjacent to personnel hatch).

After shutdown, most of the containment becomes accessible but for limited lengths of time due to residual fission and activation product activities. At the refueling floor inside the containment, dose rates will range from 1.5 to 40 mrem/h, depending on the location at the refueling floor and the interval of time after shutdown. These dose rates are due to neutron-activated materials and fission and activation products inside the Primary Coolant System, and radioactivity absorbed on surfaces which had prior contact with primary coolant.

For the transport of spent fuel elements, concrete and gravel shielding provides protection for the areas close to the transfer route of the fuel. Shielding is provided around the reactor internals storage pool, the steam generator compartments and the clean waste receiver tanks. This shielding is designed for personnel protection during storage of activated reactor internals and for protection during the refueling operation.

11.6.4.3 Auxiliary Building (Including Radwaste Building Addition)

All radioactive areas can be reached through service corridors which are entered from the access control station. Radiation exposures are minimized during normal equipment operation by use of reach rods which penetrate through the shield walls or have other remote operators. Gauges and other instruments which need visual checking from time to time can be inspected from the corridor or on the local or main control boards. Plant operating personnel are thus able to perform duties necessary for normal operation of the Plant with a minimum of radiation exposure. The different systems are isolated from each other by individually shielded chambers. Systems and equipment can be isolated for maintenance or repair with no significant radiation interference from other systems or equipment.

During operation, the major radiation sources to be expected are the tanks, pumps and piping containing contaminated drainage. The concentrations of radioactive fission and activation products in these systems are expected to be generally of the low to medium type (10^{-3} μCi/cm^3 to 10 μCi/cm^3). Concrete shielding walls provide protection for the adjacent basement operation areas. Concrete shielding is provided around the waste gas decay tanks.

The spent resin holdup tank chamber is not accessible; however, a portion of the shielding constructed of solid concrete blocks is removable for access.

The dirty waste filter, spent fuel pool filter and demineralizer are shielded by concrete for protection of the adjacent uncontrolled areas, such as the diesel generator room. The use of dirty waste and fuel pool filters has been discontinued because of ALARA concerns.

Other equipment chambers contain volume control equipment, treated and filtered waste, laundry and radiation chemistry drain tanks and pumps, radwaste fan and filters, etc. A special area is designated and designed for the decontamination of equipment. This decontamination room can be sealed off by doors in order to prevent the spreading of contamination during the cleaning operations.

The area above the radwaste filters, demineralizers and spent resin tank, which is fenced off and controlled, serves for removal of the spent resin, filter and demineralizer components. Each filter and demineralizer can be reached by a removable concrete hatch. The adjacent service building may have to be shielded during removal of contaminated materials by construction of removable shielding shadow walls.

The counting room has been shielded in order to reduce background radiation. For the room containing the ventilation system, concrete is used to shield against radiation from the ventilation filters and also against radiation streaming out of the containment through the piping penetrations for the main steam lines.

After shutdown of the Plant, the shutdown heat exchangers will become significant radiation sources. Fission product activity and neutron-activated corrosion products present in the primary coolant are the radiation emitters. The heat exchangers are accessible with a 5 to 100 mrem/h dose rate.

The environment outside the containment will remain much the same during operation and shutdown of the Plant. The transfer of spent fuel is the primary source of radiation during periods of shutdown. The relative closeness of fully accessible and uncontrolled areas requires especially heavy shielding for the areas surrounding the spent fuel pool.

In the event of an MHA, the engineered safeguards pump rooms will become inaccessible, particularly when recirculating water from the containment sump. Shielding has been placed between these high sources and the Plant control room to prevent excessive direct radiation shine into the control room.

The areas close to the containment and close to the spray and injection systems at elevation 590 feet 0 inch will experience radiation levels as high as several tens of rem/hr and these areas will not be entered for some time after an MHA.

The control room has concrete shielding for those sides which are in direct line of sight with the containment building. Together with the containment shell, the integrated whole body shine dose inside the control room would be less than 2.5 rem over a period of 30 days following an MHA.

11.6.4.4 Turbine Building

The turbine building is fully accessible and uncontrolled with dose rates much less than 0.5 mrem/h during normal Plant operation as well as during shutdown.

In the event of an MHA, some portions of the turbine building will not be entered by personnel since dose rates from the reactor building initially will be of the order of several rem/h for the unprotected portions of the building close to the reactor building. Other areas which require personnel entry for emergency procedures have been shielded to allow safe access.

11.6.4.5 General Plant Yard Areas

The radiation shielding design of the containment and auxiliary buildings protects all Plant yard areas from excessive radiation exposure. All yard areas which are frequently occupied by Plant personnel during normal operation and shutdown receive a radiation field of less than 0.5 mrem/h.

11.6.4.6 Other Buildings

Other buildings such as the feedwater purity building and the east and south radwaste buildings are designed so that shielding is provided around potential radiation sources.

Equipment storage buildings outside the protected area, such as the south and east storage buildings are controlled less than 0.5 mrem/hr at the building exterior or fence to control access.

In addition, adequate radiation shielding is provided at the interim old steam generator storage facility located outside the protected area boundary in a Zone I access area. The dose rate at the facility's exterior surface is less than 0.5 mrem/h in accordance with the requirements of 10 CFR 20 and Zone I classification. The shielding design also meets the requirements of 40 CFR 190 for offsite dose.

11.6.5 AREA RADIATION MONITORING SYSTEMS

11.6.5.1 Design Basis

This system consists of monitors, instrumentation and alarms that serve to warn Plant personnel of increasing radiation levels in various Plant areas. Reliance is placed on the process radiation monitoring system for early warning of a Plant malfunction resulting in increasing radiation levels that might result in a health hazard.

All of the electronic circuitry except for the detectors (Geiger-Mueller tubes or ion chambers) is solid state. The circuits and their components are designed to operate with 0% to 95% relative humidity. Existing Model 845 equipment has operating temperature limitations of -40°F to 140°F. The operating temperature limitations for the detectors and readout modules are 32°F and 140°F. The new 900 series area monitors are designed to operate in a relative humidity of 0% to 95% noncondensing atmosphere, with operating temperature limits for the detector of 0°F to 122°F. The remote alarm/indicator temperature limits are -22°F to 140°F and 32°F to 122°F for the ratemeter.

Detector ranges and sensitivities are chosen to enable monitoring within the requirements of 10 CFR 20 and the access control zoning. The area radiation monitoring system instruments and detectors have been chosen for their proven reliability in other Plants, and spare items and portable units are provided to permit operation during prolonged maintenance.

The radwaste buildings are outside the plant protected area and are not connected to the area monitor system. Portable monitors with local readouts will be used with auto dialer phone alarm notification equipment because of remote locations. The monitors have an operating temperature range of 12°F to 122°F and the detector probe -40°F to 167°F.

11.6.5.2 System Description

Thirty four area monitors monitor locations within the auxiliary and containment buildings. They provide both indication and warning of radiation levels in both normally radioactive and non-radioactive areas. The 34 area radiation monitoring system detectors are wall mounted, coaxial ion chambers. The associated meters are designed with range and sensitivity suitable for their location. High radiation levels and individual circuit failures are alarmed both visually and audibly on the area radiation monitoring panel. Readouts and display equipment for the area radiation monitoring system are located on panels in the control room. The area radiation monitoring panels receive power from the preferred ac buses or the instrument bus.

Three additional monitors have been placed in the radwaste storage and processing buildings. These units are stand alone units with local readouts and alarms. If high radiation levels are reached in a storage building, the local monitor alarm will sound and also initiate an auto dialer phone notification to the plant main control room.

Pursuant to NUREG-0737, two high range gamma monitors have been installed in the containment building. The monitors are ion chambers with the readout range extended to 10^7 rads/hour. These monitors are designed to provide a continuous readout of containment radiation levels for all conditions ranging from normal operation to hypothetical accident conditions. Calibration is performed by electronic equipment due to extreme dose rate range of the instruments.

The monitors have been selected with ranges and sensitivities appropriate for their service areas. Alarms set points are adjustable to enable monitoring to within the requirements of 10CFR20 and the access control zoning. The location, range, and sensitivity of the monitors are listed in Table 11-16.

11.6.5.3 Testing and Maintenance

Circuit testing of readout equipment and the power source can be performed from the control room. The circuit being tested or repaired is inoperative during this time and acts as a tripped channel. Radiation sources are used to calibrate the detectors and circuits.

11.6.6 HEALTH PHYSICS

11.6.6.1 Facilities

Because the radiation protection and Plant water chemistry analyses are closely intermingled, laboratory complexes are used for most analytical work. This area, which is primarily located in the auxiliary building, is made up of the following:

1. A sample room is located adjacent to the containment building personnel air lock. Here, all samples of primary coolant are taken. They are then carried to the laboratory for radiochemical analyses.

2. A modified radwaste sampling area.

3. A controlled laboratory where highly radioactive samples are analyzed or diluted.

4. A controlled laboratory where intermediate-level radioactive samples are analyzed.

5. A counting room where station samples are counted for activity level.

6. A counting room for low-level environmental samples.

7. The Health Physics office.

8. The Radiation Safety Supervisor's office.

9. A change room where personnel change into scrubs if anticontamination clothing is required.

10. A personnel decontamination facility associated with the access control area where personnel can be monitored for contamination and appropriate measures taken.

11. A change room for donning anti-contamination clothing.

The chemistry and controlled laboratories are provided with constant air-flow fume hoods which vent to the Plant stack. The auxiliary building also has a portable instrument storage area on the 611' elevation to provide instrumentation before entering radiation controlled areas of the Plant. This permits convenience in returning instruments as personnel return through this elevation to access control. The personnel decontamination facility in access control contains a shower and instruments for monitoring of decontamination activities.

All laundry is processed offsite. Storage for laundered clothes is provided in the anti-contamination clothing change room.

11.6.6.2 Tool and Equipment Decontamination Facility

Tools and equipment are decontaminated in a specially designed facility in the auxiliary building. This room contains two sink units, a decontamination pan bordered by a six-inch concrete curb, two jib cranes for equipment handling and a hot instrument shop to provide an area for repair and maintenance of contaminated instrumentation.

A cask washdown pit for decontaminating large components and fuel transfer casks is located on the upper level of the spent fuel pool. This pit is also used for miscellaneous decontaminating of tools and equipment associated with fuel pool work.

The plant has purchased and installed a Kleiber & Schultz Inc. decontamination unit for removing radioactive contamination from tools and equipment. The unit uses either a grit blast or high pressure water to clean the item. Components as large as 4-feet x 4-feet x 8-feet can be cleaned in the unit and is currently located inside the restricted area and the radiologically controlled area of the plant adjacent to the fuel pool on the 649' elevation. By-product Material License No 21-08606-08 has been revised primarily to allow decontamination of Big Rock Point tools and equipment. The plant will not accept radioactive material from licensees other than Consumers Energy.

11.6.6.3 Calibration Facility

A calibration facility is provided in a separate room adjoining the turbine building at the 590-foot elevation. The room has a deep well for storing the larger sources (greater than 1 curie) which are primarily used for survey instrument calibration. Smaller sources are stored in a pipe well and in storage cells constructed of shielding blocks. Provisions are made for low-level instrument calibration, dosimetry spiking and TLD spiking.

The room housing the calibration facility is kept locked and entrance is allowed only with the approval of the Radiological Services Department.

The J.L. Shepherd Model 89 is an instrument used for calibrating portable radiation instruments and contains a 400 Ci Cs-137 and a 130 mCi Cs-137 source. The J.L. Shepherd is located in the instrument auxiliary building equipment cage. The cage is normally locked and the J.L. Shepherd is padlocked when not in use. Keys to the cage and the J.L. Shepherd are controlled by the Radiological Services Department.

11.6.6.4 Radiation Control

Personal radiation exposure is kept to a practical minimum by a combination of shielding and access control.

11.6.6.5 Shielding

As previously discussed, shielding is designed to keep exposures to personnel to a practical minimum. As Plant operations progress, if it is found that the shielding in given locations is insufficient, either more shielding will be added to reduce exposure to the design rate or access will be limited.

11.6.6.6 Access Control

Access to the restricted area (as defined in 10CFR20, Section 20.1003) is limited on the basis of radiation levels or the presence of radioactive materials and contamination. The area outside the restricted area, but within the site boundary is designated the owner controlled area. There are several secondary restricted areas outside the primary fenced area (e.g. East Radwaste, South Radioactive Material Buildings).

Any area in which radioactive material is stored, handled or processed and in which radiation levels are in excess of those defined as a "radiation area" in 10 CFR 20 has access controlled for purposes of radiation protection. In general, all areas accessed through Door 105A are designated as radiation controlled areas. All entrances to these areas will be controlled by the use of one or more signs with headings such as RADIATION CONTROLLED AREA - DOSIMETRY REQUIRED FOR ENTRY, CAUTION CONTAMINATION AREA, CAUTION OR DANGER AIRBORNE RADIOACTIVITY AREA. The words "caution" or "danger" shall be used for radiation areas and high radiation areas. The words "grave danger" shall be used for any very high radiation areas.

Within the restricted area, access to areas of higher radiation exposure rate are further controlled and defined as:

1. Radiation Area

   Any area, accessible to personnel, in which radiation levels could result in an individual receiving a dose equivalent in excess of 5 millirems in 1 hour at 30 centimeters (12 inches) from the radiation source, or from any surface that the radiation penetrates, is designated as a radiation area. Personal exposure in a radiation area are kept to a minimum by both administrative procedures based on accumulated dose records and by limiting the time spent in the area. Radiation areas may be isolated by a rope or chain, but they are all posted with signs, including the following words: CAUTION - RADIATION AREA

2. High-Radiation Area

Any area, accessible to personnel, in which radiation levels could result in an individual receiving a dose equivalent in excess of 100 millirems in 1 hour at 30 centimeters (12 inches) from the radiation source, or from any surface that the radiation penetrates, is designated as a high radiation area. Personnel entrances to high radiation areas are typically controlled by the use of locked doors when dose rates are above 1000 mrem/hr. Consistent with Technical Specification 6.7.3, these areas may be barricaded, posted, and a flashing red light used as a warning device. Access is only by permission of the Radiological Services Department. When dose rates are greater than 100 mrem/h but less than 1,000 mrem/h, a barricade, but not necessarily a locked door, is required. All high-radiation areas are equipped with a sign or signs bearing the following notation: DANGER - HIGH RADIATION AREA

Radiation protection personnel make routine surveys of all accessible areas of the Plant to ascertain the current status of the radiation levels in these areas. Status sheets are posted showing radiation levels and significant radiation sources in the area.

3. Very High Radiation Area

Any area, accessible to personnel, in which radiation levels could result in an individual receiving an absorbed dose in excess of 500 rads in one hour at 1 meter from the radiation source, or from any surface that the radiation penetrates, is designated as a very high radiation area. In addition to the above requirements for high radiation areas, the Radiological Services Department shall institute additional measures to ensure that an individual is not able to gain unauthorized or inadvertent access to a very high radiation area. Access is only by permission of the General Manager Plant Operations, with controls, in addition to those for high radiation areas, shall be specified on the Radiation Work Permit.

Any very high radiation areas identified at the plant shall be posted with signs, including the following words: GRAVE DANGER - VERY HIGH RADIATION AREA.

11.6.6.7 Facility Contamination Control

Contamination of general Plant areas is controlled by using the step-off pad technique. A double step-off pad may be employed for jobs involving high levels of contamination. Plastic and absorbent paper is used to carry contaminated tools and equipment and to facilitate rapid cleanup. Automated frisker stations are located at access control so that personnel can check themselves to assure that they are not contaminated, and thereby carry contamination to other parts of the plant. Where automated frisker stations are not installed, or out of service, monitoring with a G-M count rate meter and pancake probe or equivalent will be used. As a final check a walk-through-type portal monitor is normally used at the exit from the restricted area in the security building (except during Site Emergency Plan activations or drills involving accountability actions).

Radiation protection personnel make routine contamination surveys of all accessible areas of the Plant. Any area found contaminated to an undesirable level will be taped or roped off, signs posted and cleanup will commence as soon as practical. Status sheets are posted showing contamination levels and the appropriate protective clothing to be worn when entering the area.

11.6.6.8 Personnel Contamination Control

Contamination of personnel is controlled and prevented by the use of several types of protective clothing when personnel enter contaminated areas.

1. Lab Coats - These are worn by laboratory personnel when analyzing radioactive samples, and by others when performing light work activity in, slightly contaminated areas as allowed by radiation safety personnel.

2. Cloth Coveralls - These are worn in most instances when entering contaminated areas.

3. Paper Coveralls - These are disposable and are sometimes used as an outer protector to the cloth coveralls.

4. Shoe Covers - Cloth covers are worn in most contaminated areas where dry contamination is encountered. In the case of wet contamination, either plastic shoe covers or rubbers are worn over the cloth shoe covers.

5. Gloves - Cloth or rubber gloves are worn in most contaminated areas where dry contamination is encountered. Rubber gloves are worn over plastic gloves in the event wet contamination is a possibility.

6. Plastic Suits - These are worn in areas where the potential exists for liquid contamination of personnel and are generally worn over cloth coveralls.

7. Head Protection - Cloth caps can be worn in contamination areas during light work activity or with respirators, cloth hoods for high-level contamination and plastic hoods for wet contamination.

In cases where a double step-off pad is used, two sets of protective clothing are normally worn.

Normally, most of the Plant is accessible to personnel in street clothes. The bomb shelter change room just inside from access control is the main change area for the radiation controlled part of the station (areas beyond access control).

11.6.6.9 Airborne Contamination Control

Airborne contamination is minimized by keeping floor contamination low and by reducing leaks as much as possible. In the event of levels of airborne contamination approaching or exceeding levels as specified in 10 CFR 20, Appendix B, Table 1, Column 3, an evaluation is performed on the use of process or other engineering controls (e.g., close-capture and controlled-flow ventilation, containment, isolation, dilution).

In the event process or other engineering controls cannot be practically exercised, then other actions are taken to limit internal exposure, consistent with maintaining the total effective dose equivalent (TEDE) ALARA, such as limitation of working times, control of access, use of respiratory protection equipment or other controls. A protection factor is determined for each type of respiratory protection equipment in use. See Table 6-1 of Reference 8.

Continuous air monitors are used in areas of potential airborne radioactivity or air samples are taken with a portable sampler and analyzed to evaluate actual work conditions in such areas of potential airborne radioactivity. Continuous air monitors are set to alarm when the airborne radioactivity reaches the applicable derived air concentrations (DAC) established for Co-60 in 10 CFR, Part 20, Appendix B, Table 1, Column 3. In areas where continuous air monitors are not provided, samples taken with portable air samplers will provide the basis for determining concentrations to ensure appropriate control and tracking of internal exposures.

11.6.6.9.1 Respiratory Protection Program

A Respiratory Protection Program has been established by means of Plant procedures based on the requirements of 10 CFR 20, Subpart H and the Nuclear Operations Department Radiation Safety Plan (see Reference 9). This program provides guidelines for personnel training in the proper selection and use of respiratory equipment. This program also sets forth the requirements for respirator control, inspection and maintenance.

Training

All personnel whose duties may require the use of a respirator are trained in the Respiratory Protection Program. A thoroughly knowledgeable instructor trains the personnel in each of the following:

1. Recognition of the need for respiratory equipment, consistent with maintaining TEDE ALARA, including interpretation of airborne sampling results to identify airborne concentrations.

2. Restrictions on personnel entry into areas requiring respiratory protection and including frequency of entry and duration of stay for different levels of airborne concentrations.

3. Selection of the respiratory equipment which provides the most effective protection against the type and level of radioactive airborne contaminant that may be present. A determination of the respiratory protection factors for each device.

4. Selection of respiratory equipment most suitable for the required work.

5. Preparation of the selected respiratory equipment for use.

    a. Inspection for cleanliness, damage and contamination from previous use.

    b. Instruction in the proper fitting procedure.

    c. Testing for proper fit.

The medical examination shall be determined by a physician prior to initial fittings of respirators, and either every 12 months thereafter or periodically at a frequency determined by a physician, that the individual user is medically fit to use the respiratory protection equipment.

Personnel having received training are required to remain current regarding changes to the Respiratory Protection Program. Formal retraining and requalification is performed for each individual on an annual basis.

Control, Inspection and Maintenance

A member of the Plant radiation safety staff has designated responsibility for the Respiratory Protection Program and maintains control of respiratory equipment when not in use:

1. To ensure the proper cleaning, handling and storage,

2. To prevent the use of contaminated, defective or faulty devices,

3. To determine if personnel requesting the equipment have been properly trained and fitted,

4. For proper equipment utilized for task and

5. For maintaining breathing air quality for self-contained and air line respiratory devices.

All respirators are inspected before and after each use. Respiratory equipment subjected to extended storage is inspected periodically.

All repairs and the steps required to maintain the respiratory devices ready for use, such as, cleaning, decontamination and filter cartridge replacement, are performed under the direction of the supervisor assigned responsibility for the Respiratory Protection Program.

11.6.6.10 External Radiation Dose Determination

All regularly assigned employees, contractors, and visitors frequenting radiation controlled areas of the plant are assigned individual dosimeters. Permanent dose records are kept on the above employees as required by 10CFR20. Primary thermoluminescent dosimeters are sent to a dosimetry analytical group for processing and interpretation. Personnel who have assigned dosimeters also may be assigned additional Plant dosimeters (normally a direct reading dosimeter or electronic dosimeter). These dosimeters are used to keep a daily total estimate of an individual's dose. Their use as a permanent record of an individual's dose will be restricted to times when thermoluminescent dosimeters are lost or damaged and when a large discrepancy exists between direct reading dosimeter/electronic dosimeter accumulations and thermoluminescent dosimeter interpretations. Assignment of permanent dose by other than primary dosimeter will be documented with an explanation.

Special high-range direct reading dosimeters, finger rings, wrist badges, neutron dosimeters, or thermoluminescent dosimeters are issued by Radiation Safety on the basis of need.

11.6.6.11 Internal Radiation Dose Determination

Internal radiation dose from the inhalation of airborne radioactive material is determined through tracking of DAC-hours and/or conduct of bioassays (including passive monitoring, whole body counting, or analysis of excreta). The effectiveness of process/engineering controls and the Respiratory Protection Program is evaluated by bioassays, or by urine analysis if tritium burdens require evaluation. At least one of these methods of uptake evaluation (DAC-hr tracking or bioassay) will be performed not less than once per year for all radiation workers who have been exposed to significant levels of airborne radioactivity or work routinely in contaminated areas of the plant (a documented passive monitoring program satisfies this requirement). Dose records of Plant radiation workers are available to their supervisors so that work can be planned accordingly and the effectiveness of the Health Physics Program can be evaluated. Bioassay results are maintained as a portion of these records in accordance with 10 CFR 20.2106.

11.6.7 RADIATION PROTECTION INSTRUMENTATION

11.6.7.1 Counting Room Instrumentation

The counting room instrumentation includes:

1. A multichanneled analyzer using a germanium crystal in a lead cave.

2. An alpha detector and scaler.

3. A liquid scintillation tritium counter.

4. A beta detector and scaler.

5. A gross gamma scintillation detector and scaler.

11.6.7.2 Portable Radiation Detecting Instrumentation

The portable radiation detecting instrumentation normally stored in the portable instrument storage area in access control includes:

1. A fast and thermal neutron count instrument.

2. High- and low-range G-M survey instruments.

3. Low-range ionization chamber instruments.

4. High-range ionization chamber instruments.

5. Extended probe high-range G-M tube.

6. A condenser r-meter and a set of standardized ionization chamber thimbles are available.

11.6.7.3 <u>Air Sampling Instrumentation</u>

The portable and mobile air sampling instrumentation includes:

1. High-volume air samplers equipped with a particulate filter.

2. Low-volume air samplers that can be equipped with both particulate and halogen filters.

3. Lapel air samplers for breathing zone samples.

4. Continuous Air Monitors including alarm functions.

11.6.7.4 <u>Personal Monitoring Instrumentation</u>

Personal monitoring instrumentation may include one or all of the following:

1. Direct reading pocket ion chamber dosimeters with a range of 0-200mrem, 0-1.5rem, and 0-5rem.

2. Thermoluminescent dosimeters with a range of at least 10mrem to 1,000rem.

3. Electronic direct reading dosimeters with a range of at least 0-1000rem.

11.6.7.5 <u>Emergency Instrumentation</u>

Instruments are kept in special areas of the Plant which are accessible in the event of an emergency. These instruments are checked and calibrated at regular intervals to assure their proper functioning. These instruments include:

1. Radiation detection equipment capable of measuring field from 0.1 mrem/h-1,000 rem/h.

2. A portable G-M type instrument to be used as a check for low-level contamination.

3. A high-volume air sampler.

4. Direct reading dosimeters with a range of 0-50 rem.

11.6.8 TESTS AND INSPECTIONS

11.6.8.1 Shielding

Visual inspections of Plant shielding were made during the construction phase. Because of the shielding's massive structure, these inspections were limited to detecting major defects. With the reactor in operation, radiation surveys are made to assure that:

1. There are no defects or inadequacies in the shielding that might affect personal exposures during normal operation and maintenance of the Plant.

2. Areas of the Plant are correctly posted and barricaded as radiation and high-radiation areas.

These surveys consist both of gamma and neutron monitoring where appropriate. Continued routine radiation surveys of all areas of the Plant will assure integrity of the shielding.

A study was made of the adequacy of Plant shielding following a TMI-type accident. As a result of this study, shielding was added and emergency procedures were modified.

11.6.8.2 Area and Process Radiation Monitors

Each area and process monitor is periodically tested to determine that:

1. The calibration of the monitor ensures that control room readout instrumentation indicates true radiation levels.

2. The alarm scale trip points function properly and that the alarms function properly.

3. Equipment actions that occur upon a high radiation signal are verified.

11.6.8.3 Continuous Air Monitors

Each continuous air monitor is periodically tested to determine that:

1. The calibration of the monitor is correct and that readout in counts per minute can be converted to air contamination in $\mu Ci/cm^3$.

2. Airflow is constant.

3. Trip alarm points are set and function properly.

11.6.8.4 Radiation Protection Instrumentation

The following instrumentations are tested and calibrated periodically:

1. Counting room instrumentation.

2. Portable instrumentation.

3. Air samplers.

4. Personal monitoring instruments. Extra primary thermoluminescent dosimeters are spiked with measured radiation levels as control samples without the knowledge of the analytical group to test the accuracy of their measurements. Secondary electronic direct reading dosimeters are calibrated using standardized sources and known geometries. Plant thermoluminescent dosimeters have individual element correction factors calculated using calibrated sources and known geometries. Direct reading pocket ion chambers are certified using calibrated sources and known geometries.

5. Emergency instruments.

11.6.9 CONTROL OF BYPRODUCT, SOURCE OR SPECIAL NUCLEAR MATERIAL (SNM) SOURCES

The control of byproduct, source or special nuclear material sources exceeding 100 millicuries is by approved Radiological Services Department procedures which contain information described in Regulatory Guide 1.70.

The ability to handle sources has been demonstrated at Palisades since the Provisional Operating License was issued. Personnel qualifications, facilities, and equipment and procedures for handling have also been established. Surveillance leak testing to determine source leakage was incorporated into the Technical Specifications. Subsequently, the Technical Specifications were revised to relocate this leak testing requirement to the Offsite Dose Calculation Manual.

In the NRC safety evaluation for Amendment 98, the Staff noted they had reviewed the Palisades' personnel qualifications, facilities, equipment and procedures for handling byproduct, source and special nuclear material and found them consistent with Regulatory Guide 1.70.3 and meeting the requirements of 10 CFR Parts 30, 40 and 70. The Staff further found on the basis of the Palisades' radiation safety program, previous reviews, and information provided by NRC, Region III, that Palisades has an adequate Health Physics organization and radiation protection program, and that personnel are adequately trained to handle the sealed sources licensed for Palisades. The Staff concluded that incorporation of flexible, yet controlled licensed provisions for the receipt possession, and use of byproduct, source and special nuclear material into the Palisades Operating License is acceptable.

Some examples of Palisades' sources are:

Isotope	Quantity	Form	Use
PuBe	5 Ci	Sealed Source	Instrument Calibration
PuBe	1 Ci	Sealed Source	Instrument Calibration
Cs-137	10 Ci	Sealed Source	Instrument Calibration
Cs-137	400 Ci	Sealed Source	Instrument Calibration
Cs-137	120 mCi	Sealed Source	Instrument Calibration
Cs-137	250 mCi	Sealed Source	Instrument Calibration

The primary storage location for sources is the Calibration Facility but other controlled locations can be used as necessary for the operation of the facility (as described in Subsection 11.6.6.3).

11.6.10 RADIOACTIVE MATERIAL STORAGE FACILITIES

Storage for reusable radioactive materials, besides the limited space provided within the plant radiologically controlled area, is provided by buildings within the owner controlled area. These buildings are engineered structures. These buildings are assigned to work groups who are responsible for maintaining inventory, housekeeping and accessibility of work groups into the storage buildings. They include the north, south and east storage buildings. There is also radioactive material storage at the northeast end of the Feedwater Purity Building. This area is also used for receipt, monitoring and storage of returned clean anti-contamination clothing. The Radiological Services Department oversees the movement of radioactive material to and from these buildings along with performing periodic radiation survey requirements. These buildings are maintained locked and entrance is only allowed with the approval of the Radiological Services Department.

REFERENCES

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3. "Municipal Water Facilities" Public Health Service Publication No 775, Volume 5, 1964.

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5. LADTAP, NRC NUREG/CR-1276, May 1980. Users' Manual for LADTAP 11-A Computer Program (computer program for calculating radiation exposure to man from routine releases of nuclear reactor liquid effluent).

6. Spigarelli, S A, Cesium-137 Activities in Fish Residing in Thermal Discharges to Lake Michigan, Health Physics 30, 411-413 (1976).

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8. "Manual of Respiratory Protection Against Airborne Radioactive Materials," NUREG-0041, October 1976.

9. Nuclear Operations Department Radiation Safety Plan.

10. John A Zwolinski, Chief, Operating Reactors Branch 5, USNRC, to David J VandeWalle, Nuclear Licensing Administrator, CP Co, "Radiological Effluent Technical Specifications," November 9, 1984.

11. CPCo's letters, T.C. Bordine to NRC Document Control Desk, November 12, 1987 and January 25, 1988.

12. Memorandum from L.J. Cunningham, DREP to T.R. Quay, T.V. Vambach, "Request for Additional Information (RAI)", March 15, 1988, April 7, 1989, and January 12, 1990.

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15. CPCo's letter, T.P. Neal to B. Holian, October 23, 1990. (Typo of 10/13/90 in original reference).

16. NRC Letter, Brian Holian to G.B. Slade, CPCo, June 7, 1991, "Approval and Conditions to Retain Soil in Place".

17. NRC Letter, Brian Holian to G.B. Slade, CPCo, June 7, 1991, "Approval and Conditions to Retain Soil in Place".

TABLE 11-1
(Sheet 1 of 3)

PRIMARY COOLANT FISSION AND CORROSION PRODUCT ACTIVITIES

Isotope	Design Activity (1% Failed Fuel) (Microcuries/ Cubic Centimeter)	Normal Activity (Microcuries/Cubic Centimeter)(a)	Design Inventory (Curies)	Normal Inventory (Curies)
H-3	2.80	0.21	860	65
Br-84	4.66×10^{-2}	-	13.4	-
Kr-85(m)	1.50	0.030	432	8.6
Kr-85	2.34	3.52×10^{-4}	674	0.10
Kr-87	0.81	0.037	233	10.6
Kr-88	2.48	0.060	714	17.3
Rb-88	2.44	0.122	702	35.1
Rb-89	6.10×10^{-2}	3.05×10^{-4}	0.18	9.0×10^{-4}
Sr-89	5.33×10^{-3})	2.28×10^{-5} (b)	1.53)	7.0×10^{-3} (b)
Sr-90	1.50×10^{-4})		4.3×10^{-2})	
Y-90	5.77×10^{-4}	-	0.17	-
Sr-91	3.70×10^{-3}	8.01×10^{-5}	1.06	0.023
Y-91(m)	0.24	1.76×10^{-3}	68.4	0.50
Mo-99	2.26	3.70×10^{-5}	649	0.010
Te-129	2.72×10^{-2}	-	7.84	-
I-129	5.66×10^{-8}	-	1.63×10^{-5}	-
I-131	4.40	0.114	1,266	32.8
Te-132	0.36	3.22×10^{-5}	102	9.1×10^{-3}

(a)Based on typical data from January 14, 1981; primary coolant at 17% of Technical Specifications limit for dose equivalent I-131.
(b)Total value consisting of summation of Sr-89 and Sr-90.

TABLE 11-1
(Sheet 2 of 3)

Isotope	Design Activity (1% Failed Fuel) (Microcuries/ Cubic Centimeter)	Normal Activity (Microcuries/Cubic Centimeter)(a)	Design Inventory (Curies)	Normal Inventory (Curies)
I-132	1.13	0.095	324	27.2
I-133	6.38	0.141	1,830	40.4
Xe-133	225	0.472	6.47×10^{4}	135.7
Cs-134	1.42	1.35×10^{-5}	725	6.9×10^{-3}
Te-134	2.63×10^{-2}	-	7.55	-
I-134	0.64	0.134	184	38.5
I-135	2.78	0.142	801	40.9
Xe-135	6.75	0.090	1,943	25.9
Cs-136	0.15	1.90×10^{-5}	43.4	5.5×10^{-3}
Cs-137	10.6	2.32×10^{-5}	3,064	6.7×10^{-3}
Xe-138	0.36	(b)	103	(b)
Cs-138	0.69	0.059	198	16.9
Ba-140	6.78×10^{-3}	5.09×10^{-4}	1.95	0.15
La-140	6.52×10^{-3}	6.97×10^{-4}	1.88	0.20
Co-60	1.1×10^{-3}	2.78×10^{-5}	0.37	9.4×10^{-3}
Fe-59	7.7×10^{-6}	1.41×10^{-5}	2.6×10^{-3}	4.8×10^{-3}
Co-58	7.9×10^{-3}	4.28×10^{-4}	2.67	0.14
Mn-56	2.3×10^{-2}	1.55×10^{-4}	7.77	0.052

(a)Based on typical data from January 14, 1981; primary coolant at 17% of Technical Specifications limit for dose equivalent I-131.
(b)Data for nuclides with less than 15 minute half-lives is not documented.

TABLE 11-1
(Sheet 3 of 3)

Isotope	Design Activity (1% Failed Fuel) (Microcuries/ Cubic Centimeter)	Normal Activity (Microcuries/Cubic Centimeter)(a)	Design Inventory (Curies)	Normal Inventory (Curies)
Mn-54	2.0×10^{-5}	2.70×10^{-5}	6.8×10^{-3}	9.2×10^{-3}
Cr-51	2.4×10^{-3}	5.35×10^{-4}	0.81	0.18
Zr-95	1.9×10^{-5}	1.08×10^{-5}	6.4×10^{-3}	3.6×10^{-3}
Total	275.7	1.708	79,663	497

(a)Based on typical data from January 14, 1981; primary coolant at 17% of Technical Specifications limit for dose equivalent I-131.

TABLE 11-2
(Sheet 1 of 2)

RADIOACTIVE WASTE QUANTITIES OF SIGNIFICANT ACTIVITY

Liquid Waste Sources	Quantity/Year-Gallons	Assumptions
Chemical and Volume Control System	69,500	Start-up after refueling
	900	Cold shutdown Day 1
	37,200	Start-up from cold condition Day 1
	13,200	Hot shutdown(a) Day 50
	1,200	Start-up from hot condition Day 50
	17,400	Hot shutdown(a) Day 140
	1,700	Start-up from hot condition Day 140
	32,300	Hot shutdown(a) Day 230
	2,000	Start-up from hot condition Day 230
	2,200	Cold shutdown Day 240
	104,300	Start-up from cold condition Day 240
	68,900	Hot shutdown(a) Day 270
	2,300	Start-up from hot condition Day 270
	17,500	Cold shutdown to refuel
	216,000	Boron reduction for cycle shim bleed
Total	586,600	
Primary System Drain Tank	1,500	Primary coolant pump seal leakage at 1 gpd per pump for four pumps
	1,000	Safety injection check valve leakage
	64,900	Drain primary system for maintenance at Day 240
	13,800	Drain primary system for refueling
	21,300	Drain refueling shield
Total	102,500	

(a)Hot shutdown quantities are based on maintaining the shutdown margin during xenon buildup by boron dilution and on Plant start-up eight hours after shutdown during the xenon buildup peak.

TABLE 11-2
(Sheet 2 of 2)

Liquid Waste Sources	Quantity/Year-Gallons	Assumptions
Equipment Drain Tank	11,400	Coolant charging pump seal leakage at 31 gpd maximum leakage
	7,000	Spent fuel pool overflow
	3,900	Spent resin tank overflow from resin flushing - 260 ft^3/year of resin replacement at 2 ft^3/ft^3 resin
	1,900	Demineralizers drain from resin charging - 260 ft^3/year of resin replacement at 1 ft^3/ft^3 resin
Total	24,200	
Radiochemistry Lab Drain Tank	11,000	12 samples per day at 2.5 gallons per sample
Total Liquid Waste of Significant Activity	724,300	

Gaseous Waste Sources	Quantity/Year-ft^3	Assumptions
Degas of Primary System	2,418	Degas to 5 cm^3/kg prior to 3 cold shutdowns including gas in volume control tank
Off-Gas Liquid Waste From Shutdowns and Start-Ups	867	See start-up and shutdown schedule above
Off-Gas for Chemical Shim Reduction	1,011	Boron dilution generates 216,000 gallons liquid waste with 35 cm^3/kg gas content
Off-Gas Liquid Drained for Refueling Maintenance and Miscellaneous	243	
Total Gaseous Waste of Significant Activity	4,539	

TABLE 11-3
(Sheet 1 of 12)

EQUIPMENT RATINGS AND CONSTRUCTION CODES - ORIGINAL EQUIPMENT

1. CLEAN WASTE RECEIVER TANKS (T-64A, B, C, D)

Number	4
Material	Type 304 Stainless Steel
Capacity (Each)	50,000 gal (7,000 ft^3)
Design Pressure	5 psig (Vapor Space)
Design Temperature	103°F
Code	API 620

2. PRIMARY SYSTEM DRAIN TANK (T-74)

Number	1
Material	Type 304 Stainless Steel
Capacity	900 gal
Design Pressure	50 psig
Design Temperature	250°F
Code	ASME B&PV Code, Section III, Class C

3. EQUIPMENT DRAIN TANK (T-80)

Number	1
Material	Type 304 Stainless Steel
Capacity	550 gal
Design Pressure	50 psig
Design Temperature	150°F
Code	ASME B&PV Code, Section VIII

4. SPENT RESIN STORAGE TANK (T-69)

Number	1
Material	Type 304 Stainless Steel
Capacity	3,000 gal (400 ft^3)
Design Pressure	125 psig
Design Temperature	200°F
Code	ASME B&PV Code, Section III, Class C

TABLE 11-3
(Sheet 2 of 12)

5. CONTROLLED CHEM LAB DRAIN TANK (T-76)

Number	1 Divided Into 2 Compartments
Material	Type 304 Stainless Steel
Capacity	900 gal (Total)
Design Pressure	50 psig
Design Temperature	150°F
Code	ASME B&PV Code, Section VII

6. TREATED WASTE MONITOR TANKS (T-66A, B)

Number	2
Material	Carbon Steel
Internal Coating	Baked Phenolic, 6-8 Mils
Capacity	5,500 gal
Design Pressure	10 psig
Design Temperature	150°F
Code	ASME B&PV Code, Section VIII

7. DIRTY WASTE DRAIN TANK (T-60)

Number	1 Divided Into 2 Compartments
Material	Carbon Steel
Capacity	3,800 gal
Design Pressure	10 psig
Design Temperature	150°F
Code	ASME B&PV Code, Section VIII Requirements

8. FILTERED WATER MONITORING TANKS (T-63)

Number	1 With 2 Compartments
Material	Carbon Steel
Internal Coating	Baked Phenolic, 6-8 Mils
Capacity	5,500 gal
Design Pressure	10 psig
Design Temperature	150°F
Code	ASME B&PV Code, Section VIII

TABLE 11-3
(Sheet 3 of 12)

9. WASTE GAS SURGE TANK (T-67)

Number	1
Material	Carbon Steel
Capacity	550 gal (80 ft^3)
Design Pressure	20 psig
Design Temperature	150°F
Code	ASME B&PV Code, Section III, Class C

10. WASTE GAS DECAY TANKS (T-68A, B, C)

Number	3
Material	Carbon Steel
Capacity	800 gal (100 ft^3)
Design Pressure	120 psig
Design Temperature	550°F
Code	ASME B&PV Code, Section III, Class C

11. LAUNDRY DRAIN TANK (T-70)

Number	1 Divided Into 2 Compartments
Material	Carbon Steel
Capacity	1,000 gal (Total)
Design Pressure	15 psig
Design Temperature	150°F
Code	ASME P&PV Code, Section VIII

TABLE 11-3
(Sheet 4 of 12)

12. VACUUM DEGASIFIER

a. Tank (T-57)

Number	1
Performance	Handle Solution With 0-50 Std cm^3 H_2 (kg of Liquid Waste. Reduced H_2 Concentration to 1/40 of Influent Concentration)
Design Flow	0-160 gpm
Design Pressure	75 psig
Design Temperature	160°F
Material	Stainless steel
Code	ASME B&PV Code, Section III, Class C

b. Vacuum Pumps (C 51A, B)

Number	2
Capacity	2.3 scfm
Type	Rotary, Water-Sealed, Water-Lubricated with Closed Cooling Loop
Motor	3 hp

13. RADWASTE DEMINERALIZER

a. Demineralizer tanks (T-55A, B, C)

Number	3
Material of Tank	Stainless Steel
Flow Rate (Each)	48 gpm (Rated), 100 gpm (Max)
Unit Flow Rate (at Ratings)	5.0 gpm/ft^2
Resin Volume (Each)	32 ft^3
Resin Type	Equivalent Capacity Mixture of Nuclear Grade Cation and Anion
Design Pressure	125 psig
Design Temperature	160°F
Code	ASME B&PV Code, Section III, Class C and ASME B&PV Code, Section VIII, Para UW-2 (a)

TABLE 11-3
(Sheet 5 of 12)

b. Clean Resin Transfer Tank (T-61)

Number	1
Material	PVC Lined Carbon Steel
Capacity	32 ft^3
Design Pressure	125 psig
Design Temperature	125°F
Code	ASME B&PV Code, Section VIII

14. CLEAN WASTE FILTERS (F-57A, B, C)

Type	Cartridge Type With Replaceable Elements
Number	3
Material, Container	Stainless Steel
Material, Filter Media	Polypropolene
Filter Media Rating	3 Micron
Flow Rate	100 gpm
Design Pressure	125 psig
Design Temperature	160°F
Code	ASME B&PV Code, Section III, Class C and ASME B&PV Code, Section VIII, Para UW-2 (a)

15. EQUIPMENT DRAIN FILTER (F-56)

Number	Cartridge Type with Replaceable Elements
Material, Container	Stainless Steel
Material, Filter Media	Polypropolene
Filter Media Rating	25 Microns
Design Pressure	125 psig
Design Temperature	212°F
Code	ASME B&PV Code, Section III, Class C and ASME B&PV Code, Section VIII, Para Uw-2 (a)

TABLE 11-3
(Sheet 6 of 12)

16. DIRTY WASTE FILTER (F-53)

Type	Cartridge Type With Replaceable Elements
Number	1
Material, Container	Stainless Steel
Material, Filter Media	Polypropolene
Filter Media Rating	200 Microns
Flow Rate	75 gpm
Design Pressure	100 psig
Design Temperature	160°F
Code	ASME B&PV Code, Section III, Class C and ASME B&PV Code, Section VIII, Para UW-2 (a)

17. LAUNDRY DRAIN FILTER (F-55)

Type	Cartridge Type With Replaceable Elements
Number	1
Material, Container	Stainless Steel
Material, Filter Media	Polypropolene
Filter Media Rating	25 Microns
Flow Rate	20 gpm
Design Pressure	100 psig
Design Temperature	160°F
Code	ASME B&PV Code, Section III, Class C and ASME B&PV Code, Section VIII, Para UW-2 (a)

TABLE 11-3
(Sheet 7 of 12)

18. PRIMARY SYSTEM DRAIN TANKS PUMPS (T-71A, B)

Type	Horizontal Centrifugal With Mechanical Seals
Number	2
Capacity (Each)	75 gpm
Head	80 ft TDH
Material	Type 316 Stainless Steel
Motor	5 hp, 3 Phase, 60 Hertz, 460 Volt
Codes	Motor NEMA; Pump, Standards of Hydraulic Institute

19. DEGASIFIER PUMPS (P-68A, B)

Type	Horizontal Centrifugal With Mechanical Seals
Number	2
Capacity (Each)	160 gpm
Head	190 ft TDH
Material	Type 316 Stainless Steel
Motor	20 hp, 3 Phase, 60 Hertz, 460 Volt
Codes	Motor NEMA; Pump, Standards of Hydraulic Institute

20. EQUIPMENT DRAIN TANK PUMPS (P-75A, B)

Type	Horizontal Centrifugal With Mechanical Seals
Number	2
Capacity (Each)	100 gpm
Head	266 ft TDH
Material	Type 316 Stainless Steel
Motor	25 hp, 3 Phase, 60 Hertz, 460 Volt
Codes	Motor NEMA; Pump, Standards of Hydraulic Institute

TABLE 11-3
(Sheet 8 of 12)

21. CONTROLLED CHEM LAB DRAIN PUMPS (P-61A, B)

Type	Horizontal Centrifugal With Mechanical Seals
Number	2
Capacity (Each)	10 gpm
Head	158 ft TDH
Material	Type 316 Stainless Steel
Motor	7.5 hp, 3 Phase, 60 Hertz, 460 Volt
Codes	Motor NEMA; Pump, Standards of Hydraulic Institute

22. RECEIVER TANK (CWRT) PUMPS (P-69A, B)

Type	Horizontal Centrifugal With Mechanical Seals
Number	2
Capacity (Each)	100 gpm
Head	200 ft TDH
Material	Type 316 Stainless Steel
Motor	15 hp, 3 Phase, 60 Hertz, 460 Volt

23. RECEIVER TANK (CWRT) CIRCULATING PUMP (P-70)

Type	Horizontal Centrifugal With Mechanical Seals
Number	1
Capacity (Each)	250 gpm
Head	121 ft TDH
Material	Type 316 Stainless Steel
Motor	15 hp, 3 Phase, 60 Hertz, 460 Volt
Codes	Motor NEMA; Pump, Standards of Hydraulic Institute

TABLE 11-3
(Sheet 9 of 12)

24. TREATED WASTE MONITOR PUMPS (P-58A, B)

Type	Horizontal Centrifugal With Mechanical Seals
Number	2
Capacity (Each)	150 gpm
Head	140 ft TDH
Material	Type 316 Stainless Steel
Motor	15 hp, 3 Phase, 60 Hertz, 460 Volt
Codes	Motor NEMA; Pump, Standards of Hydraulic Institute

25. SAFETY INJECTION ROOM SUMP PUMPS (P-72A, B AND P-73A, B)

Type	Vertical Centrifugal
Number	4 (2 Sets With 2 Pumps per Set)
Capacity (Each)	25 gpm
Head	35 ft TDH
Material	Stainless Steel
Motor	1 hp, 3 Phase, 60 Hertz, 460 Volt
Codes	Motor NEMA,

26. DIRTY WASTE DRAIN TANK PUMPS (P-60A, B)

Type	Horizontal Centrifugal With Mechanical Seals
Number	2
Capacity (Each)	75 gpm
Head	160 ft TDH
Material	Type 316 Stainless Steel
Motor	10 hp, 3 Phase, 60 Hertz, 460 Volt
Codes	Motor NEMA; Pump, Standards of Hydraulic Institute

TABLE 11-3
(Sheet 10 of 12)

FILTERED WASTE MONITOR PUMP (P-63)

Type	Horizontal Centrifugal With Mechanical Seals
Number	1
Capacity (Each)	75 gpm
Head	150 ft TDH
Material	Type 316 Stainless Steel
Motor	10 hp, 3 Phase, 60 Hertz, 460 Volt
Codes	Motor NEMA; Pump, Standards of Hydraulic Institute

28. WASTE GAS COMPRESSORS (C-50A, B)

a. Compressor

Number	2
Type	Single Head, Single Stage Diaphragm Type
Capacity (Each)	2.35 scfm at 14.7 psia Suction, 0.44 scfm at 7.5 psia Suction
Discharge Pressure	100 psig at Maximum Delivery
Material	Stainless Steel
Motor	2 hp, 3 Phase, 60 Hertz, 460 Volt, TEFC

b. Aftercoolers

Number	2
Type	Shell and Tube With Moisture Separator and Drain Trap
Material	Tube Side (Gas) Stainless Steel, Shell Side (Water) Carbon Steel

29. WASTE GAS COMPRESSOR (C-54)

a. Compressor

Number 1	
Type	Diaphragm Type
Capacity (Each)	5 scfm Avg Between Suction Pressures of 7.5 and 15.0 psia
Discharge Pressure	100 psig
Material	
Motor	7.5

b. Aftercoolers

Number	1

30. PIPING, FITTING AND VALVES

a. Liquid Systems

Material	Stainless Steel
Design Pressure	Floor Drains - Atmosphere, Process - 50 psig
Joints, 2" and Larger	Butt Weld
1-1/2" and Smaller	2,000 lb SW
Valves, 2" and Larger	Stainless Steel, Butt Weld Ends, 150 lb
1-1/2" and Smaller	Stainless Steel, 2,000 lb SW Ends, 150 lb
Butterfly Valves	Stainless Steel, Flangeless, 150 lb
Plug Vales	Stainless steel, FF Flanged, 150 lb
Code	ASA B31.1, Code for Pressure Piping, Including Applicable Nuclear Code Cases, ASA B16.5
Radiography	All Butt Weld in Nuclear Service Systems Rated for Higher Than 50 psig and 212°F Are Radiographed 100. All Butt Welds in Nuclear Systems Rated for Less Than 50 psig and 212°F Are Radiographed 10%

TABLE 11-3
(Sheet 12 of 12)

b. Gaseous Systems

Material	Carbon Steel
Design Pressure	100 psig
Joints, 2" and Larger	Butt Weld
1-1/2" and Smaller	3,000 lb SW
Valves, 2" and Larger	Carbon Steel, 150 lb BW
1-1/2" and Smaller	Carbon Steel, 600 lb, 3,000 lb SW
Butterfly	Carbon Steel, 150 lb, FF Flanged
Plug	Ductile Iron, 150 lb, FF Flanged
Code	ASA B31.1, Code for Pressure Piping, Including Applicable Nuclear Code Cases, ASA B16.5
Radiography	All Welds Are 100% Radiographed

TABLE 11-4
(Sheet 1 of 7)

EQUIPMENT RATINGS AND CONSTRUCTION CODES -
ADDITIONAL EQUIPMENT INSTALLED 1971-1973

1. CLEAN WASTE HOLDUP TANK (T-85)

Number	1
Material of Tank	SA240 Type 304 SS
Capacity	5,000 gal
Design Pressure	Atm
Design Temperature	212°F
Code	API-620, 1970 Exam ASME B&PV Code, Section III, 1971

2. CLEAN WASTE TRANSFER PUMP (P-94)

Type	Gould 3196
Number	1
Capacity	80 gpm
Head	85 ft
Material	Stainless Steel
Motor	5 hp
Code	ASME B&PV Code, Section III, Class 3, 1971

3. CLEAN AND DIRTY WASTE EVAPORATORS (M-59A, B)

a. Evaporator Vessel

Number	2
Material	316 SS
Capacity	1,600 gal
Design Pressure	15 psig
Design Temperature	200°F
Code	ASME B&PV Code, Section III, Class 3, 1971

b. Recirculation Pump

Type	Gould 3196 - 3 x 4-13
Number	4
Capacity	200 gpm
Head	30 ft
Material	316 SS
Motor	5 hp, 3 Phase, 60 Hertz, 460 Volt
Code	ASME B&PV Code, Section III, Class 3, 1971

TABLE 11-4
(Sheet 2 of 7)

c. Distillate Pump

Type	Gould 3196 - 1 x 1-1/2-6
Number	4
Capacity	25 gpm
Head	130 ft
Material	316 SS
Motor	5 hp
Code	ASME B&PV Code, Section III, Class 3, 1971

d. Concentrate Pump

Type	Gould 3196 - 1 x 1-1/2-6
Number	4
Capacity	25 gpm
Head	100 ft
Material	316 SS
Motor	3 hp
Code	ASME B&PV Code, Section III, Class 3, 1971

e. Vacuum Pump

Type	Nash
Number	4
Capacity	48 cfm @ 20" vacuum
Head	2 psig
	Material316 SS
Motor	3 hp
Code	N/A

4. CLEAN WASTE DISTILLATE TANK (T-86)

Number	1
Material	SA240 Type 304 SS
Capacity	5,000 gal
Design Pressure	Atm
Design Temperature	212°F
Code	API-650, 1970 Exam ASME B&PV Code, Section III, 1971

TABLE 11-4
(Sheet 3 of 7)

5. CLEAN WASTE DISTILLATE PUMP (P-97A, B)

Type	Gould 3196
Number	2
Capacity	80 gpm
Head	85 ft
Material	316 SS
Motor	5 hp - Pacemaker
Code	ASME B&PV Code, Section III, Class 3, 1971

6. CLEAN (MISCELLANEOUS) WASTE CONCENTRATE TANKS (T-94, T-95)

Number	2
Material	SA240 Type 304 SS
Capacity	1,500 gal
Design Pressure	Atm
Design Temperature	212°F
Code	API-650, 1970 Exam ASME B&PV Code, Section III, 1971

7. MISCELLANEOUS WASTE FILTER (F-59)

Type Elements	Cartridge Type With Replaceable
Number	1
Material, Container	304 SS
Material, Filter Media	Epoxy Impregnated Cellulose
Filter Media Rating	150 Micron*
Flow Rate	30 gpm
Design Pressure	50 psig
Design Temperature	212°F
Code	ASME B&PV Code, Section III, Class 3, 1971

8. CLEAN (MISCELLANEOUS) WASTE CONCENTRATE PUMPS (P-95A, B)

Type	Gould 3196, 1 x 2-8
Number	2
Capacity	30 gpm
Head	75 ft
Material	316 SS
Motor	3 hp - Pacemaker
Code	ASME B&PV Code, Section III, Class 3, 1971

*May be in service with element removed.

TABLE 11-4
(Sheet 4 of 7)

9. POLISHING (MISCELLANEOUS WASTE) DEMINERALIZERS (T-88, T-89A, B)

Number	3
Material of Tank	304 SS
Flow Rate	50 gpm (Rated), 100 gpm (Max)
Unit Flow Rate (At Max)	10 gpm/ft^2
Resin Volume	32 ft^3
Resin Type	Equivalent Capacity Mixture of Nuclear Grade Cation and Anion

10. MISCELLANEOUS WASTE HOLDUP TANKS (T-92A, B, C)

Number	3
Material	SA240 Type 304 SS
Capacity	20,000 gal
Design Pressure	Atm
Design Temperature	212°F
Code	API-620, 1970 Exam ASME B&PV Code, Section III, 1971

11. MISCELLANEOUS WASTE TRANSFER PUMPS (P-92A, B)

Type	Gould 3196, 2 x 3-13
Number	2
Capacity	80 gpm
Head	120 ft
Material	316 SS
Motor	10 hp - Pacemaker
Code	ASME B&PV Code, Section III, Class 3, 1971

12. MISCELLANEOUS WASTE DISTILLATE TANK (T-87)

Number	1
Material	SA240 Type 304 SS
Capacity	8,000 gal
Design Pressure	Atm
Design Temperature	212°F
Code	API-650, 1970 Exam ASME B&PV Code, Section III, 1971

13. MISCELLANEOUS WASTE DISTILLATE PUMPS (P-89A, B)

Type	Gould 3196, 1-1/2 x 3-6
Number	2
Material	316 SS
Capacity	80 gpm
Head	110 ft
Motor	7-1/2 hp - Pacemaker
Code	ASME B&PV Code, Section III, Class 3, 1971

14. RADWASTE SPENT RESIN STORAGE TANK (T-100)

Number	1
Material	304 SS
Capacity	200 ft^3
Design Pressure	125 psig
Design Temperature	200°F
Code	ASME B&PV Code, Section III, Class 3, 1971

15. WASTE GAS DECAY TANKS (T-101A, B, C)

Number	3
Material	Carbon Steel
Capacity	225 ft^3
Design Pressure	120 psig
Design Temperature	550°F
Code	ASME B&PV Code, Section III, Class 3, 1971

TABLE 11-4
(Sheet 6 of 7)

16. PIPING, FITTINGS AND VALVES

a. Liquid Radwaste System (Process)

Material	Stainless Steel
Design Pressure	125 psig and 150 psig
Fittings	
2-1/2" and Larger	150 lb Thickness To Match Pipe Butt Weld Ends
2-1/2" and Smaller	3,000 lb Socket Weld
Valves	
2-1/2" and Larger	150 lb Butt Weld Ends
2" and Smaller	150 lb Socket Weld Ends
Joints	
2-1/2" and Larger	Butt Weld Ends
2" and Smaller	Socket Weld Ends
Code	ASME B&PV Code, Section III, Nuclear Power Plant Components, Nuclear Class 3, 1971, per AEC Safety Guide 26, March 27, 1972

b. Atmospheric Radwaste Drainage Piping

Material	Stainless Steel
Design Pressure	Atmospheric
Fittings	
8" and Smaller	Butt Weld Ends
Joints	
8" and Smaller	Butt Weld
Code	ANSI B31.1.0 - 1967 Power Piping

c. Gaseous Radwaste System (Process)

Material	Carbon Steel
Design Pressure	150 psig
Fittings	
2-1/2" and Larger	150 lb Thickness To Match Pipe Butt Weld Ends
2" and Smaller	3,000 lb Socket Weld Ends
Valves	
2-1/2" and Larger	150 lb Butt Weld Ends
2" and Smaller	600 lb Socket Weld Ends
Joints	
2-1/2" and Larger	Butt Weld Ends
2" and Smaller	Socket Weld
Code	ASME B&PV Code, Section III, Nuclear Power Plant Components, Nuclear Class 3, 1971, per AEC Safety Guide 26, March 23, 1972

17. PRIMARY SYSTEM MAKEUP STORAGE TANK (T-90)

Number	1
Material	Carbon Steel
Capacity	200,000 gal
Design Pressure	Atmospheric
Code	API-650, 1970

18. UTILITY WATER STORAGE TANK (T-91)

Number	1
Material	Carbon Steel
Capacity	75,000 gal
Design Pressure	Atmospheric
Code	API-650, 1970

TABLE 11-5

PRIMARY SYSTEM DRAIN TANK

Letdown and Regenerative HX Drain
Shutdown Cooling Header Relief
SI Tanks Leakage
PCS Loop Drains
Quench Tank Drains
Flange Leak Detector Drain
Controlled Bleed-Off Relief
PCS Pump Seal Leakage

TABLE 11-6

EQUIPMENT DRAIN TANK

Spent Resin Storage Tanks
Primary Coolant Sample Drain
Clean Resin Transfer Tank
Radwaste Demineralizer Drains
Fuel Pool Overflow
Fuel Pool HX Drains
Fuel Pool Demineralizer Drains
Fuel Pool Filter Drains
Blowdown Drain Tank
HPI Header Relief
SIRW Tank Drain
SIRW Tank HX Drain
SI Tank Sample Flush
Waste Gas Surge Tank Drain
Chemical Addition Tank Drain
Charging Pump Relief and Drains
VCT Drain
Purification Filters Drain

TABLE 11-7

DIRTY WASTE DRAIN TANK

Emergency Shower
Access Control Area Sink
Radwaste Addition Area Sample Sink Drains
Clean Resin Transfer Tank Drains
Pump Leak-Offs
Auxiliary Building Floor Drains
Decontamination Pit Drains
Cask Washdown Area Drains
Controlled Chemical Laboratory Drain Tank Drains
Component Cooling Surge Tank Drain
Boric Acid Batching Tank Drain
Boric Acid Filter Drains
Treated Waste Monitor Tanks
VCT Relief
Vacuum Degasifier Relief
Turbine Building Sump Pump Discharge
Containment Sump Drains
Shutdown Cooling HX Drains
Spent Resin Storage Tank (T-100)
Primary System Makeup and Utility Water Tanks Overflow
Laundry Drain Tank Drains
Auxiliary Building Sump Pumps

11-8

LIQUID RADWASTE

Isotope	Half-Life	Clean Liquid Waste Activities ($\mu Ci/cm^3$): As Received in Receiver Tanks	Clean: After 30 Days' Decay	Clean: After Processing Through Two Demineralizers	Clean: After Processing Through Evaporator	Clean: After Processing Through Polishing Demineralizer	Dirty Liquid Waste Activities ($\mu Ci/cm^3$): As Received in Drain Tank	Dirty: After Processing Through Evaporator	Dirty: After Processing Through Demineralizers
Fission Products									
Br-84	32m	1.25×10^{-2}	(a)	(a)	(a)	(a)	4.66×10^{-4}	4.66×10^{-8}	(a)
Rb-88	18m	6.61×10^{-1}	(a)	(a)	(a)	(a)	2.44×10^{-2}	2.44×10^{-6}	2.44×10^{-8}
Rb-89	15m	1.66×10^{-2}	(a)	(a)	(a)	(a)	6.10×10^{-4}	6.10×10^{-8}	(a)
Sr-89	54d	1.44×10^{-3}	9.80×10^{-4}	9.80×10^{-6}	(a)	(a)	5.33×10^{-5}	(a)	(a)
Sr-90	28y	4.00×10^{-5}	4.00×10^{-5}	4.00×10^{-7}	(a)	(a)	1.50×10^{-6}	(a)	(a)
Y-90	64h	1.55×10^{-4}	6.51×10^{-8}	(a)	(a)	(a)	5.77×10^{-6}	(a)	(a)
Sr-91	9.7h	1.00×10^{-3}	(a)	(a)	(a)	(a)	3.70×10^{-5}	(a)	(a)
Y-91	58d	6.50×10^{-2}	4.53×10^{-2}	4.53×10^{-4}	4.53×10^{-8}	(a)	2.40×10^{-3}	2.40×10^{-7}	(a)
Mo-99	67h	6.12×10^{-1}	3.67×10^{-4}	3.67×10^{-6}	(a)	(a)	2.26×10^{-2}	2.26×10^{-6}	2.26×10^{-8}
Te-129	33d	7.36×10^{-3}	3.93×10^{-3}	3.93×10^{-5}	(a)	(a)	2.72×10^{-4}	2.72×10^{-8}	(a)
I-129	1.7×10^{7}y	1.52×10^{-8}	1.52×10^{-8}	(a)	(a)	(a)	(a)	(a)	(a)
I-131	8.0d	1.18	8.87×10^{-2}	8.87×10^{-4}	8.87×10^{-7}	8.87×10^{-8}	4.40×10^{-2}	4.40×10^{-5}	4.40×10^{-7}
Te-132	77h	9.75×10^{-2}	1.46×10^{-4}	1.46×10^{-6}	(a)	(a)	3.60×10^{-3}	3.60×10^{-7}	(a)
I-132	2.3h	3.05×10^{-1}	(a)	(a)	(a)	(a)	1.13×10^{-2}	1.13×10^{-5}	1.13×10^{-7}
I-133	21h	1.73	(a)	(a)	(a)	(a)	6.38×10^{-2}	6.38×10^{-5}	6.38×10^{-7}
Cs-134	2.3y	3.84×10^{-1}	3.78×10^{-1}	3.78×10^{-3}	3.78×10^{-7}	3.78×10^{-8}	1.42×10^{-2}	1.42×10^{-6}	1.42×10^{-8}
Te-134	44m	7.12×10^{-3}	(a)	(a)	(a)	(a)	2.63×10^{-4}	2.63×10^{-8}	(a)
I-134	52m	1.73×10^{-1}	(a)	(a)	(a)	(a)	6.40×10^{-3}	6.40×10^{-5}	6.40×10^{-7}
I-135	6.7h	7.53×10^{-1}	(a)	(a)	(a)	(a)	2.78×10^{-2}	2.78×10^{-6}	2.78×10^{-8}
Cs-136	13d	4.06×10^{-2}	8.20×10^{-3}	8.20×10^{-5}	(a)	(a)	1.50×10^{-3}	1.50×10^{-7}	(a)
Cs-137	30y	2.87	2.87	2.87×10^{-2}	2.87×10^{-6}	2.87×10^{-7}	1.06×10^{-1}	1.06×10^{-5}	1.06×10^{-7}
Cs-138	32m	1.87×10^{-1}	(a)	(a)	(a)	(a)	6.90×10^{-3}	6.90×10^{-7}	(a)
Ba-140	12.8d	1.84×10^{-3}	3.65×10^{-4}	3.65×10^{-6}	(a)	(a)	6.78×10^{-5}	(a)	(a)
La-140	40.2h	1.27×10^{-3}	(a)	(a)	(a)	(a)	6.52×10^{-5}	(a)	(a)
Corrosion Products									
Cr-51	27d	6.50×10^{-4}	3.02×10^{-4}	3.02×10^{-6}	(a)	(a)	2.4×10^{-5}	(a)	(a)
Mn-54	300d	5.22×10^{-6}	4.87×10^{-6}	4.87×10^{-8}	(a)	(a)	2.0×10^{-7}	(a)	(a)
Mn-56	2.58h	6.00×10^{-3}	(a)	(a)	(a)	(a)	2.3×10^{-4}	2.3×10^{-8}	(a)
Co-58	71d	2.07×10^{-3}	1.55×10^{-3}	1.55×10^{-5}	(a)	(a)	7.9×10^{-5}	(a)	(a)
Fe-59	45d	2.02×10^{-6}	1.27×10^{-6}	1.27×10^{-8}	(a)	(a)	7.7×10^{-8}	(a)	(a)
Co-60	5.2y	2.98×10^{-4}	2.95×10^{-4}	2.95×10^{-6}	(a)	(a)	1.1×10^{-5}	(a)	(a)
Zr-95	65d	4.96×10^{-6}	3.61×10^{-6}	3.61×10^{-8}	(a)	(a)	1.9×10^{-7}	(a)	(a)

Assumptions: Factor of 10 reduction in each demineralizer for all isotopes except tritium.
10^3 reduction for iodine in evaporator.
10^4 reduction for all other isotopes except tritium.
19% of clean waste is direct from primary system.
81% of clean waste is from Chemical and Volume Control System and passes through one ion exchanger prior to entering clean waste receiver tanks.
No holdup period assumed in Miscellaneous Waste Holdup Tank.
1% defective fuel.
Dirty waste activity is 1% of primary coolant activity.

(a)Less Than 10^{-8}

TABLE 11-9

MAXIMUM CALCULATED TRITIUM RELEASE DUE TO EVAPORATION FROM REFUELING CAVITY AND SPENT FUEL POOL

Yr of Oper	Conc in Contain Bldg (Ci/cm^3)	Release From Contain Bldg (Ci/Yr)	Conc in Spent Fuel Bldg (Ci/cm^3)	Release From Spent Fuel Bldg (Ci/Yr)	Total Release (Ci/Yr)
5	1.1×10^{-13}	18	8.5×10^{-14}	13	31
10	2.1×10^{-13}	35	2.2×10^{-13}	33	68
15	3.0×10^{-13}	50	3.6×10^{-13}	53	103
20	3.8×10^{-13}	63	4.9×10^{-13}	72	135
25	4.5×10^{-13}	75	6.1×10^{-13}	90	165
30	5.1×10^{-13}	85	7.1×10^{-13}	105	190
35	5.6×10^{-13}	93	8.0×10^{-13}	118	211
40	6.0×10^{-13}	100	8.6×10^{-13}	127	227

TABLE 11-10

LADTAP INPUT DATA AND RESULTS
MAXIMUM INDIVIDUAL DOSE CALCULATIONS

Exposure Pathway	Dilution Factor	Transit Time (h)	Usage Rates (kg/yr or h/yr) Adult	Teen	Child	Infant
Fish Ingestion	15	12	21.0	16.0	6.9	0.0
Water Ingestion	1,000	24	730.0	510.0	510.0	510.0
Shoreline Use	1	0	12.0	67.0	14.0	0.0
Swimming	1	0	12.0	67.0	14.0	0.0
Boating	15	0	52.0	52.0	29.0	0.0

Dose Results (mrem/yr)(a)

Exposure Pathway	Adults T Body	Adults Liver	Adults Skin	Teenager T Body	Teenager Liver	Teenager Skin
Fish Ingestion	5.25(-1)(b)	6.96(-1)	-	2.97(-1)	6.91(-1)	-
Water Ingestion	5.66(-4)	6.13(-4)	-	3.10(-4)	4.06(-4)	-
Shoreline Use	7.96(-3)	7.96(-3)	9.3(-3)	4.45(-2)	4.45(-2)	5.19(-2)
Swimming	8.24(-5)	8.24(-5)	-	4.6 (-4)	4.6 (-4)	-
Boating	1.19(-5)	1.19(-5)	-	1.19(-5)	1.19(-5)	-
Totals	5.34(-1)	7.04(-1)	9.3(-3)	3.43(-1)	7.36(-1)	5.19(-2)

(a)Doses to other individuals and organs are smaller than those presented.
(b)5.25(-1) = 5.25×10^{-1}

TABLE 11-11

ACTIVITY IN COOLANT AND GASEOUS WASTE

Isotope	Half-Life	Activity $\mu Ci/cm^3$ Coolant	Activity in Surge Tank $\mu Ci/cm^3$ H_2 As Received	60-Day Hold	10 CFR 20 $\mu Ci/cm^3$	Fraction of 10 CFR 20 Limit at Boundary No Hold	60-Day Hold
Kr-85(m)	4.4h	1.50	42.9	(a)	1×10^{-7}	.0065	(a)
Kr-85	10.4y	2.34	66.9	65.6	3×10^{-7}	.0034	.0034
Kr-87	78m	0.81	23.1	(a)	2×10^{-8}	.017	(a)
Kr-88	2.8h	2.48	70.9	(a)	(b)	(b)	(b)
Xe-133	5.27d	225	6,430	2.48	3×10^{-7}	0.32	.00012
Xe-135	9.2h	6.75	193	(a)	1×10^{-7}	.029	(a)
Xe-138	17m	0.36	10.3	(a)	(b)	(b)	(b)

(a)Less than 10^{-8}
(b)Unlisted in 10 CFR 20

TABLE 11-12

SPECIAL LOCATION GASPAR INPUT DATA

Number	Distance (Miles)	Direction	Description	Normal $X/Q(s/m^3)$	Depleted $X/Q(s/m^3)$	Deposition $D/Q(1/m^2)$
1	0.63	S	Residence	5.56(-6)	4.66(-6)	3.08(-8)
2	0.88	ENE	Garden	2.06(-6)	1.67(-6)	2.31(-8)
3	1.00	ESE	Meat Animal	2.13(-6)	1.69(-6)	1.49(-8)
4	2.50	NE	Goat Milk	2.60(-7)	1.85(-7)	8.14(-10)
5	2.75	ENE	Milk Cow	1.81(-7)	1.26(-7)	1.59(-9)

Parameter	Value for Appropriate Locations
Fraction fresh leafy vegetables grown locally	1.0
Fraction of year cows, cattle, goats on pasture	0.5
Fraction of vegetable intake grown in garden	0.76
Fraction of feed from pasture while on pasture	1.0
Air water content, g/m^3	8.0

TABLE 11-13

DOSE RESULTS FOR SPECIAL LOCATIONS
MAXIMUM INDIVIDUAL DOSES BY AGE GROUP AND ORGAN, mrem/yr(a)

Location(a)	Description	Adults			Teenagers			Children			Infants		
		T Body	Skin	Thyroid	T Body	Skin	Thyroid	T Body	Skin	Thyroid	T Body	Skin	Thyroid
1	Plume	1.13(-1)	3.38(-1)	1.13(-1)	1.13(-1)	3.38(-1)	1.13(-1)	1.13(-1)	3.38(-1)	1.13(-1)	1.13(-1)	3.38(-1)	1.13(-1)
	Ground	1.43(-1)	1.68(-2)	1.43(-2)	1.43(-2)	1.68(-2)	1.43(-2)	1.43(-2)	1.60(-2)	1.43(-2)	1.43(-2)	1.68(-2)	1.43(-2)
	Inhalation	9.25(-2)	9.20(-2)	3.40(-1)	5.13(-2)	5.09(-2)	2.58(-1)	5.22(-2)	5.16(-2)	3.25(-1)	5.57(-2)	5.49(-2)	5.23(-1)
2	Garden	2.05(-1)	1.99(-1)	1.71(0)	2.43(-1)	2.37(-1)	1.43(0)	5.02(-1)	4.95(-1)	2.30(0)	-	-	-
3	Meat	5.09(-2)	5.06(-2)	1.10(-1)	3.52(-2)	3.50(-2)	7.62(-2)	6.16(-2)	6.14(-2)	1.24(-2)	-	-	-
4	Goat Milk	1.45(-2)	1.36(-2)	2.58(-1)	1.84(-2)	1.72(-2)	3.84(-1)	3.66(-2)	3.46(-2)	7.60(-1)	6.84(-2)	6.49(-2)	1.81(0)
5	Cow Milk	6.79(-3)	6.33(-3)	1.84(-1)	9.47(-3)	8.77(-3)	2.76(-1)	2.03(-2)	1.90(-2)	5.48(-1)	3.89(-2)	3.74(-2)	1.31(0)

(a)See Table 11-12 for location data

TABLE 11-14
(Sheet 1 of 5)

EQUIPMENT RATINGS AND CONSTRUCTION CODES
VOLUME REDUCTION AND SOLIDIFICATION SYSTEM

1. ASPHALT STORAGE TANK (T-111)

Number	1
Material	Carbon Steel
Capacity	9,000 gal
Design Pressure	15 psia
Design Temperature	370°F
Code	API-650

2. RESIN DECANT TANK (T-109)

Number	1
Material	Type 304L Stainless Steel
Capacity	1,500 gal
Design Pressure	15 psia
Design Temperature	100°F
Code	API-650

3. DISTILLATE TANK (T-114)

Number	1
Material	Type 304L Stainless Steel
Capacity	100 gal
Design Pressure	15 psia
Design Temperature	250°F

4. COOLING WATER SURGE TANK (T-112)

Number	1
Material	Carbon Steel
Capacity	100 gal
Design Pressure	15 psia
Design Temperature	120°F

5. LUBE OIL RESERVOIR (T-113)

Number	1
Material	Carbon Steel
Capacity	150 gal
Design Pressure	15 psia
Design Temperature	160°F

TABLE 11-14
(Sheet 2 of 5)

6. COOLING WATER EXCHANGER (E-68)

Number	1
Material	Type 304 Stainless Steel Tubes Carbon Steel Shell
Capacity	600,000 Btu/h
Design Pressure	150 psia
Design Temperature	120°F
Code	ASME VIII, 1977, Summer 1979

7. DISTILLATE COOLER (E-69)

Number	1
Material	Type 304 Stainless Steel Tubes Carbon Steel Shell
Capacity	60,000 Btu/h
Design Pressure	20 psia
Design Temperature	250°F
Code	ASME VIII, 1977, Summer 1979

8. LUBE OIL COOLER (E-70)

Number	1
Material	Stainless Steel Tubes Carbon Steel Shell
Capacity	45,000 Btu/h
Design Pressure	200 psia
Design Temperature	160°F
Code	ASME VIII

9. EXTRUDER DOME COOLER

Number	1
Capacity	200,000 Btu/h
Design Pressure	20 psia
Design Temperature	370°F

10. DISTILLATE ROUGHING FILTER (F-22A, B)

Number	2
Material	Type 304L Stainless Steel
Capacity	10 gpm
Design Pressure	20 psia
Design Temperature	250°F
Code	ANSI B31.1

TABLE 11-14
(Sheet 3 of 5)

11. DISTILLATE POLISHING FILTER (F-23A, B)

Number	2
Material	Type 304L Stainless Steel
Media	14 x 40 Mesh Activated Charcoal in 100 Mesh Basket
Capacity	10 gpm
Design Pressure	20 psia
Design Temperature	250°F
Code	ANSI B31.1

12. ASPHALT STRAINER (F-25A, B)

Number	2
Material	Carbon Steel
Capacity	50 gpm
Design Pressure	170 psia
Design Temperature	370°F
Code	ANSI B31.1

13. LUBE OIL FILTER (F-21)

Number	1
Material	Carbon Steel
Capacity	15 gpm
Design Pressure	200 psia
Design Temperature	160°F

14. RESIN SLURRY TRANSFER PUMP (P-109)

Type	Positive Displacement Progressing Cavity
Number	1
Capacity	80 gpm
Head	29 psi
Material	Type 316 Stainless Steel
Motor	5 hp, 3 Ph, 60 Hertz, 460 V

15. RESIN SLURRY METERING PUMP (P-115)

Type	Positive Displacement Progressing Cavity
Number	1
Capacity	0.1-1.0 gpm
Head	14 psi
Material	Type 316 Stainless Steel
Motor	1 hp, 1 Ph, 115 V

TABLE 11-14
(Sheet 4 of 5)

16. RESIN DECANT TANK PUMP (P-118)

Type	Vertical In-Line Centrifugal
Number	1
Capacity	0-60 gpm
Head	25 psi
Material	Type 316 Stainless Steel
Motor	3 hp, 3 Ph, 230 V

17. EVAPORATOR CONCENTRATES METERING PUMP (P-119)

Type	Positive Displacement Progressing Cavity
Number	1
Capacity	0.1-1.0 gpm
Head	17 psia
Material	Type 316 Stainless Steel
Motor	1 hp, 1 Ph, 115 V

18. ASPHALT RECIRCULATION PUMP (P-111)

Type	Rotary Gear
Number	1
Capacity	35 gpm
Material	Steel
Motor	30 hp, 3 Ph, 230/460 V

19. ASPHALT METERING PUMP (P-117)

Type	Rotary Gear
Number	1
Capacity	0.1-1.0 gpm
Head	17 psi
Material	Steel
Motor	1 hp, 1 Ph, 115 V

20. DISTILLATE TANK DISCHARGE PUMP (P-114A, B)

Type	Centrifugal
Number	2
Capacity	7 gpm
Head	37 psi
Material	Type 316 Stainless Steel
Motor	3 hp, 3 Ph, 230/460 V
Code	ANSI B31.1

TABLE 11-14
(Sheet 5 of 5)

21. LUBE OIL CENTRIFUGAL PUMP (P-113A, B)

Type	Horizontal, Centrifugal
Number	2
Capacity	13.25 gpm
Material	Steel
Motor	5 hp, 3 Ph, 230/460 V
Code	ANSI B31.1

22. COOLING WATER CIRCULATION PUMP (P-112)

Type	Horizontal, Centrifugal
Number	1
Capacity	72 gpm
Head	105 psig
Material	Steel
Motor	15 hp, 3 Ph, 230/460 V

23. AUXILIARY BOILER

Number	1
Material	Stainless Steel
Capacity	1,540 lb/h Sat Steam at 450°F and 250 psig
Service	495 kW, 508 Amp, 3 Ph, 480 V

24. PIPING, FITTINGS AND VALVES

Material	Carbon Steel Type 304L Stainless Steel Type 316 Stainless Steel
Codes	
Valves	ANSI B31.1 and B16.34
Piping	ANSI B31.1
Fittings	ANSI B16.5

TABLE 11-15
(Sheet 1 of 4)

PROCESS RADIATION SERVICE AND EQUIPMENT

Process Radiation Monitoring Systems	Detection Equipment/ Sampling Equipment	Readout Equipment	Sensitivity	Alarm and Control
Service Water, RE0833	Scintillation detector/detector well in service water line to discharge structure.	Digital indicator $10\text{-}10^7$ CPM.	5×10^{-6} μCi/cm^3 of Cs-137 equivalent.	Alarm on high radiation, circuit failure.
Steam Generator Blowdown, RE0707	Scintillation detector/external to blowdown tank, drain to P98A/B suction.	Digital rate meter $10\text{-}10^7$ CPM.	5×10^{-6} μCi/cm^3 of Cs-137 equivalent.	Alarm on high radiation signal; isolates blowdown tank.
Radwaste Liquid Discharge, RE1049	Scintillation detector on radwaste liquid line to discharge structure.	Digital rate meter $10\text{-}10^7$ CPM.	5×10^{-6} μCi/cm^3 of Cs-137 equivalent.	Alarm on high radiation, cirtuit failure. Isolates radwaste release.
Component Cooling Water, RE0915	Scintillation detector/piping, valves, and detector housing. CC pump dP for flow.	Digital indicator $10\text{-}10^7$ CPM.	5×10^{-5} μCi/cm^3 of Cs-137 equivalent.	Alarm on high radiation, circuit failure; isolates component cooling water surge tank vent.
Circ Water Discharge, RE1323	Scintillation detector/piping, valves, sample pump and detector housing; circulating from mixing basin prior to discharge.	Digital indicator $10\text{-}10^7$ CPM.	4×10^{-6} μCi/cm^3 of Cs-137 equivalent.	Alarm on high radiation, circuit failure.
Off-Gas Monitoring, RE0631	Scintillation detector/piping, valves and detector housing; main condenser steam jet air ejector noncondensibles to stack.	Digital rate meter analyzer $10\text{-}10^7$ CPM.	1×10^{-5} μCi/cm^3 of Xe-133 equivalent.	Alarm on high radiation and circuit failure.

TABLE 11-15
(Sheet 2 of 4)

PROCESS RADIATION SERVICE AND EQUIPMENT

Process Radiation Monitoring Systems	Detection Equipment/ Sampling Equipment	Readout Equipment	Sensitivity	Alarm and Control
Radwaste Area Ventilation, RE1809	Geiger-Mueller tube/piping, valves, sample pump and detector housing; air monitoring prior to discharge to stack.	Linear rate meter 0-10^6 CPM.	4 x 10^{-5} $\mu Ci/cm^3$ of Xe-133 equivalent.	Alarm on high radiation and circuit failure; isolates radwaste vent system.
Engineered Safeguards Pump Rooms Vent, RE1810, 1811	Geiger-Mueller tube; piping, valves, sample pump and detector housing; to stack, 2 systems, East and West rooms.	Linear rate meter 0-10^6 CPM.	1 x 10^{-4} $\mu Ci/cm^3$ of Xe-133 equivalent.	Alarm on high radiation and circuit failure; isolates pump room vent supply and exhausts.
Waste Gas Radiation, RE1113	Geiger-Mueller tube/piping, valves and detector housing; from the waste gas surge tank and waste gas decay tanks to stack.	Linear rate meter 0-10^6 CPM.	1 x 10^{-4} $\mu Ci/cm^3$ of Xe-133 equivalent.	Alarm on high radiation and circuit failure; isolation waste gas surge tank and decay tanks.
Containment Building Gas Monitoring System, RE1817	Geiger-Mueller tube/piping solenoid valves and detector housing; from 5 sample locations on (4) cooler fan discharges.	Linear rate meter 0-10^6 CPM.	1 x 10^{-4} $\mu Ci/cm^3$ of Xe-133 equivalent.	Alarm on high radiation and circuit failure.
Failed Fuel, RE0202	Scintillation detector/detector housing in letdown line.	Linear rate meter, 0-10^6 CPM, local.	N/A	Alarm on high radiation, circuit failure.

PROCESS RADIATION SERVICE AND EQUIPMENT

Process Radiation Monitoring Systems	Detection Equipment/ Sampling Equipment	Readout Equipment	Sensitivity	Alarm and Control
Steam Generator Blowdown Vent, RE2320	Scintillation detector/in well on blowdown vent line.	Log count rate meter, $10\text{-}10^6$ CPM.	2×10^{-5} $\mu Ci/cm^3$ of Cs-137 equivalent.	Alarm on high radiation, circuit failure.
Turbine Building Sump, RE5211	Scintillation detector/piping, valves, and detector housing, sump pump discharge to drain.	Log count rate meter, $10\text{-}10^6$ CPM.	5×10^{-6} $\mu Ci/cm^3$ of Cs-137 equivalent.	Alarm on high radiation, circuit failure.
Radwaste Addition Vent, RE5711	Beta scintillation/dP across roughing filter used for flow.	Digital rate meter, $10\text{-}10^7$ CPM.	5×10^{-6} $\mu Ci/cm^3$ of Xe-133 equivalent.	Alarm on high radiation circuit failure; high radiation isolates radwaste addition vent.
Fuel Building Addition Vent, RE5712	Beta scintillation/dP across roughing filter used for flow.	Digital rate meter, $10\text{-}10^7$ CPM.	5×10^{-6} $\mu Ci/cm^3$ of Xe-133 equivalent.	Alarm on high radiation circuit failure; high radiation isolates fuel building vent.

TABLE 11-15
(Sheet 4 of 4)

PROCESS RADIATION SERVICE AND EQUIPMENT

Process Radiation Monitoring Systems	Detection Equipment/ Sampling Equipment	Readout Equipment	Sensitivity	Alarm and Control
RGEMS, RE2325, 2326, 2327	Scintillation detectors for beta, gamma, ionization chamber/piping, valves, filters, sample collection bottle; discharge to stack inlet.	Log count rate meter, recorded; stack flow, recorded; sample flow recorded; 10-10^6 CPM beta (log), 10-10^7 CPM (log), gamma 1-10^7 mr/h (ion chamber) or 4 x 10^6 Ci/s digital.	1 x 10^{-6} μCi/cm^3 of Xe-133 equivalent.	Alarm, set recorder speed, isolate samples on alert level. Alarm transfer flow to upper range on high radiation.
Main Steam, RE2323, 2324	Geiger-Mueller tube/in lead collimator adjacent to main steam lines.	Log count rate meter 10^1 to 10^6 cpm or 2 x 10^4 ci/s.	5 x 10^{-2} μCi/cm^3 of dose equivalent I-131	Alarm on high radiation.

TABLE 11-16

AREA RADIATION DETECTORS

Instrument(a), (b)	Range
Containment 590' Elev NW Side (b)	10^{-2}-10^{4} Rem/hr
Containment 590' Elev SW Side (b)	10^{-2}-10^{4} Rem/hr
Containment 590' Elev NE Side (b)	10^{-2}-10^{4} Rem/hr
Containment 590' Elev SE Side (b)	10^{-2}-10^{4} Rem/hr
East Engineering Safeguards Room	0.1-10^{7} mrem/hr
Radwaste Service Corridor	0.1-10^{7} mRem/hr
Radwaste Control Area (C-40 Panel)	0.1-10^{7} mrem/hr
2.4 kV Switchgear Room (1C)	0.1-10^{7} mRem/hr
Control Lab Corridor	0.1-10^{7} mrem/hr
Access Control Station	0.1-10^{7} mrem/hr
Outside Cont Personnel Air Lock 607' Elev	0.1-10^{7} mrem/hr
Containment Purge Unit Room-North	0.1-10^{7} mrem/hr
Radwaste Demineralizer Room Roof	0.1-10^{7} mrem/hr
Control Room/Turbine Bldg Corridor	0.1-10^{7} mrem/hr
Control Room Entrance	0.1-10^{7} mrem/hr
Turbine Floor East Side	0.1-10^{7} mrem/hr
Health Physics Offices	0.1-10^{7} mrem/hr
(Former Lunchroom)	
Spent Fuel Pool-South	0.1-10^{7} mrem/hr
Air Room 590' Elev	0.1-10^{7} mrem/hr
Inside of Cont Personnel Air Lock	1.0-10^{7} mrem/hr
Containment 649' Elev Rx Cavity	1.0-10^{7} mRem/hr
Containment 649' Elev Rx Cavity	1.0-10^{7} mRem/hr
Containment Hi range-left channel	1-10^{7} Rem/hr
Containment Hi range-right channel	1-10^{7} Rem/hr
Decontamination Room	0.1-10^{7} mrem/hr
Evaporator "A"	0.1-10^{7} mrem/hr
Evaporator "B"	0.1-10^{7} mrem/hr
Evaporator Control Panel C-105	0.1-10^{7} mrem/hr
Waste Gas Decay Tank T-101, A, B & C	0.1-10^{7} mrem/hr
Environmental Lab Entrance	0.1-10^{7} mrem/hr
Radwaste Packaging Area-East	0.1-10^{7} mrem/hr
Radwaste Packaging Area-West	0.1-10^{7} mrem/hr
Radwaste Demineralizer 649' Elev	0.1-10^{7} mrem/hr
Steam Dumps Area	0.1-10^{7} mrem/hr
Radwaste Monitors (c) Local Readout	
East Processing Building	1.0-10^{5} mRem/hr
East Storage Building	1.0-10^{5} mRem/hr
South Storage Building	1.0-10^{5} mRem/hr

(a) Sensitivity for all instruments except containment detectors has energy dependence of ± 20% of the actual radiation intensity over a photon energy spectrum of 100 keV to 2.5 MeV

(b) Sensitivity for containment detector instruments has an energy dependence of ± 15% of actual radiation intensity over a photon energy spectrum of 100 keV to 2.5 MeV.

(c) Sensitivity for the radwaste monitor is ± 30% of the actual intensity over a photon energy spectrum from 80 keV to 1.33 MeV.

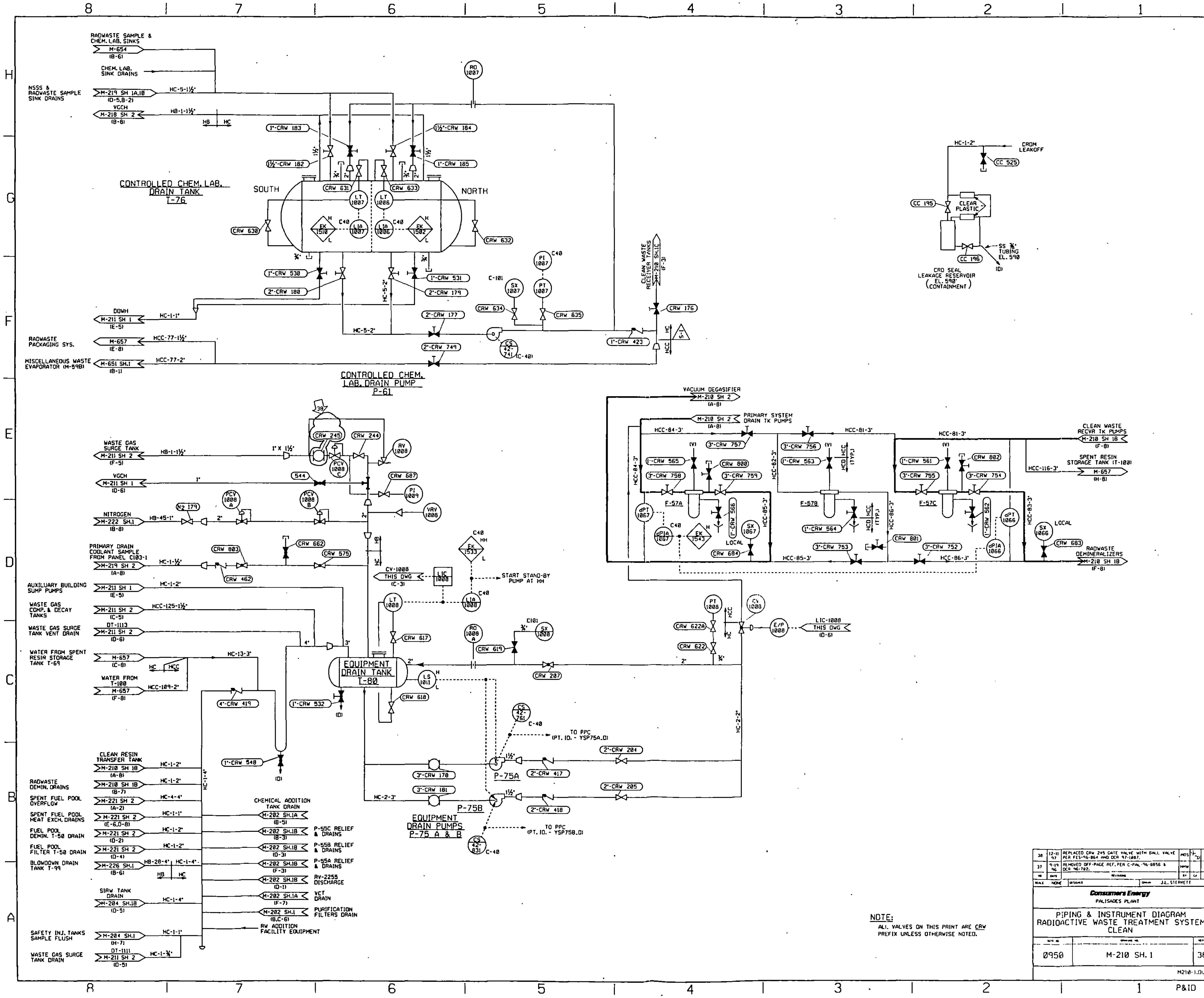

FIG 11-1 SH 1
FSAR REV 20

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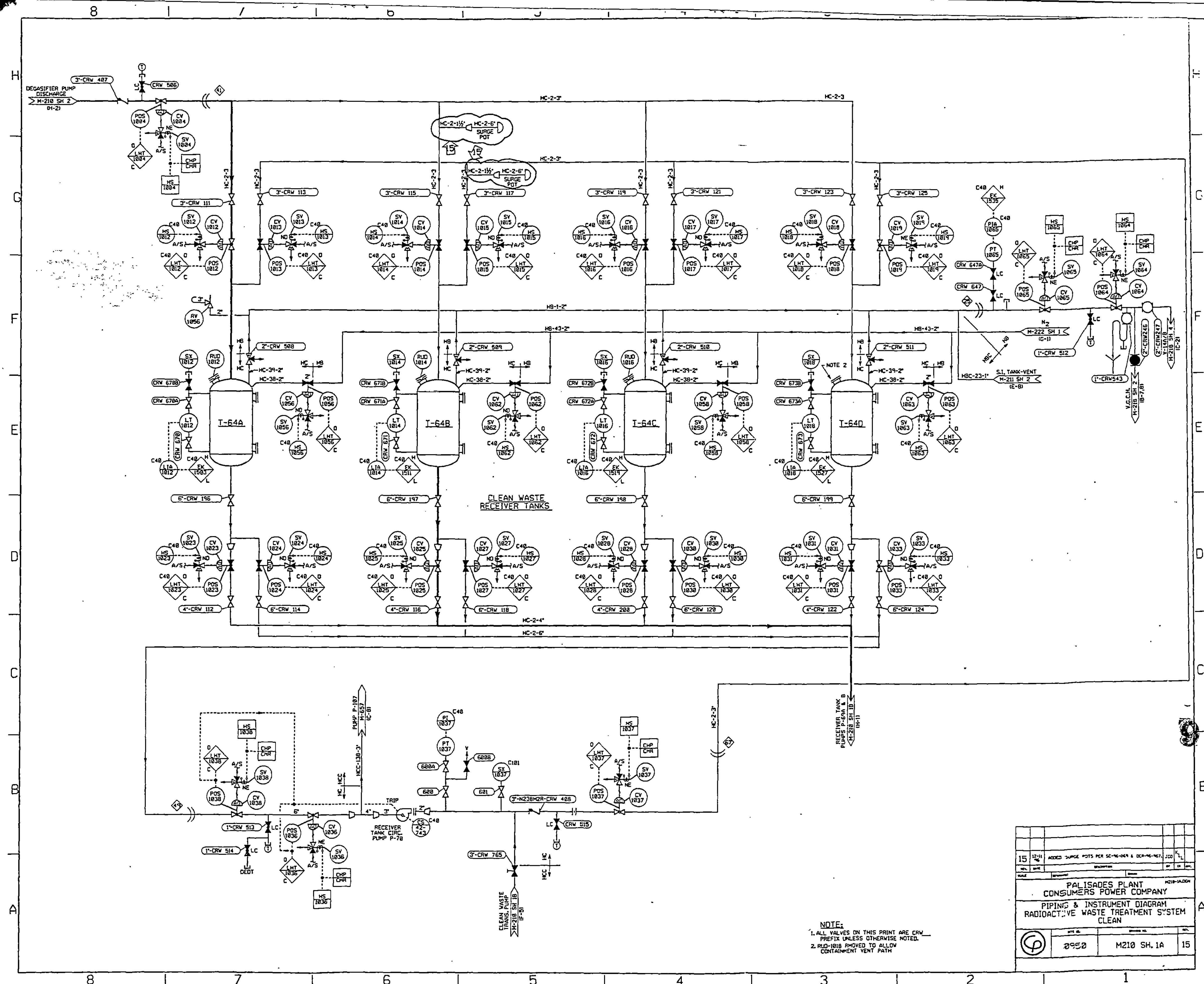

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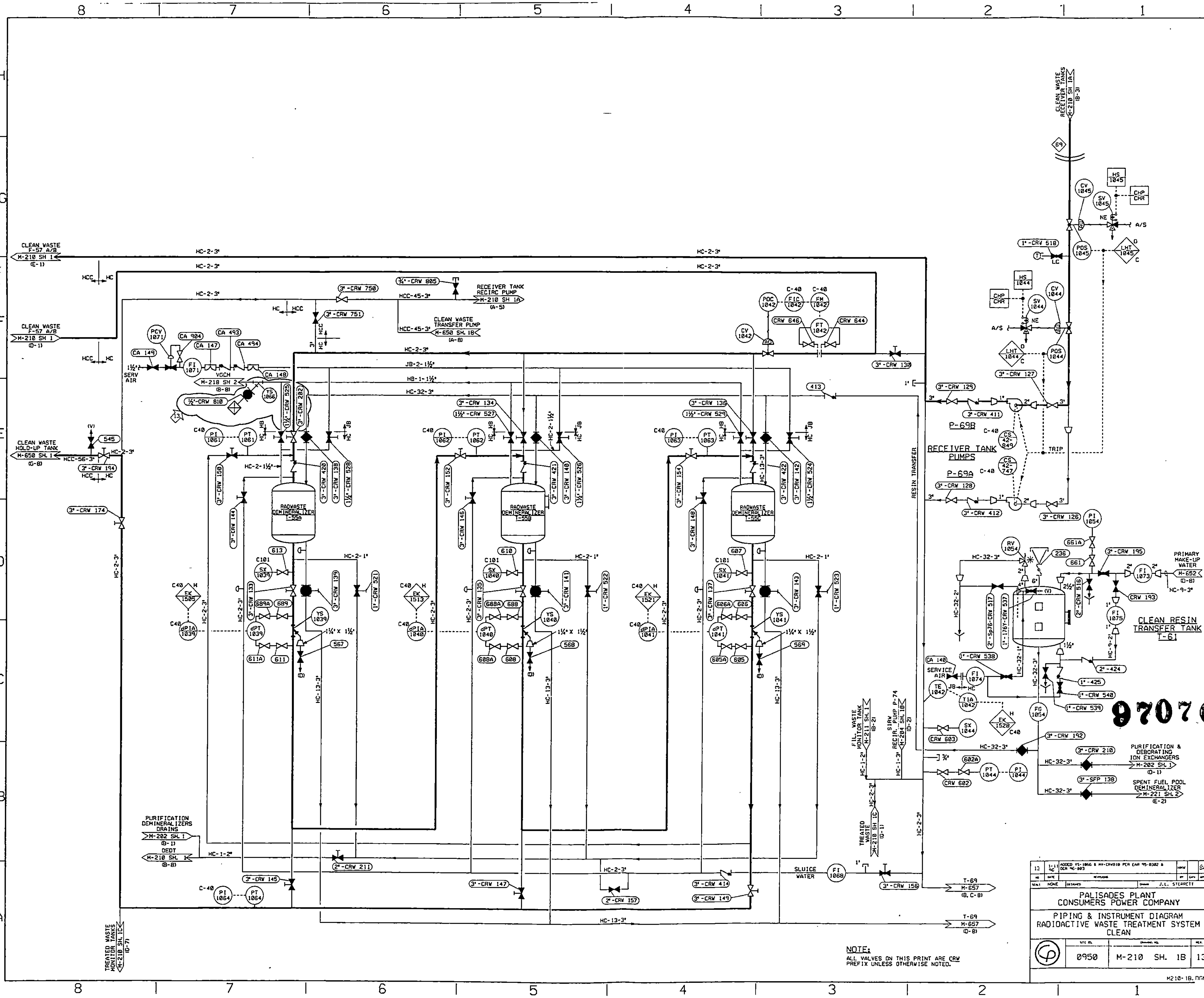
Figure 11-1 Sh 1A
Rev 19



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9707070102-03

Figure 11-1 Sh 1B
Rev 19

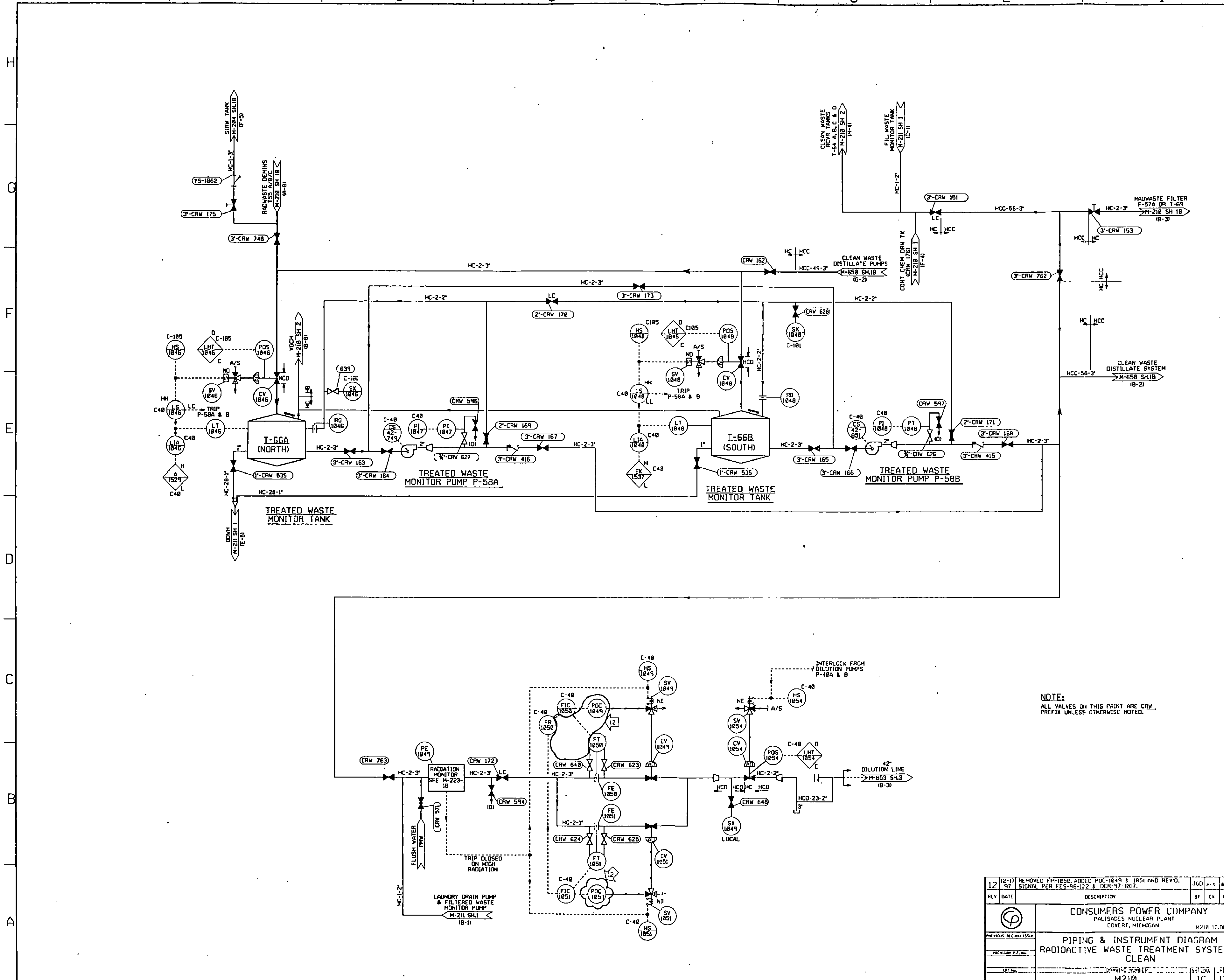

FIG 11-1 SH 1C
FSAR REV 20

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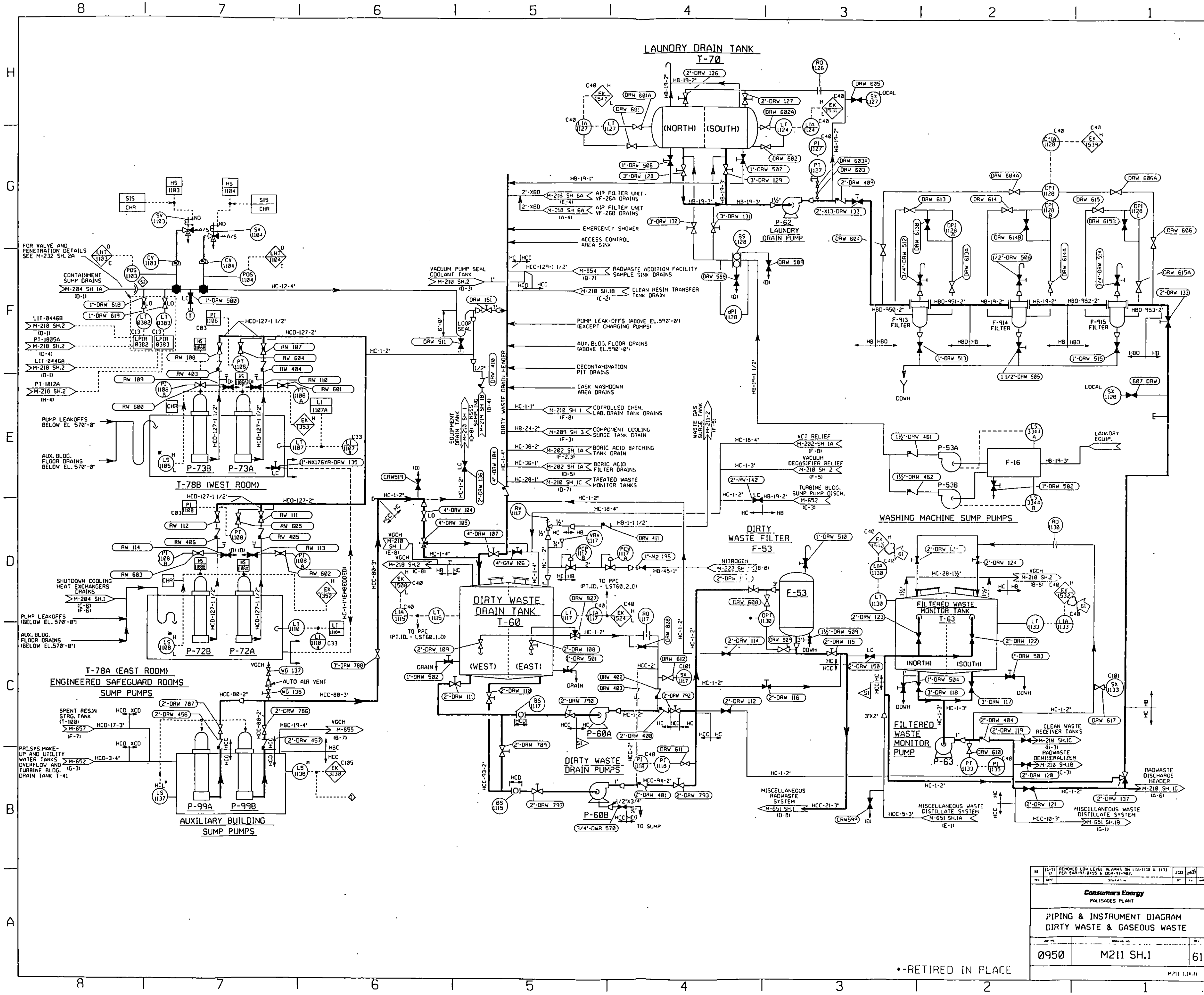

FIG 11-2 SH 1
FSAR REV 20

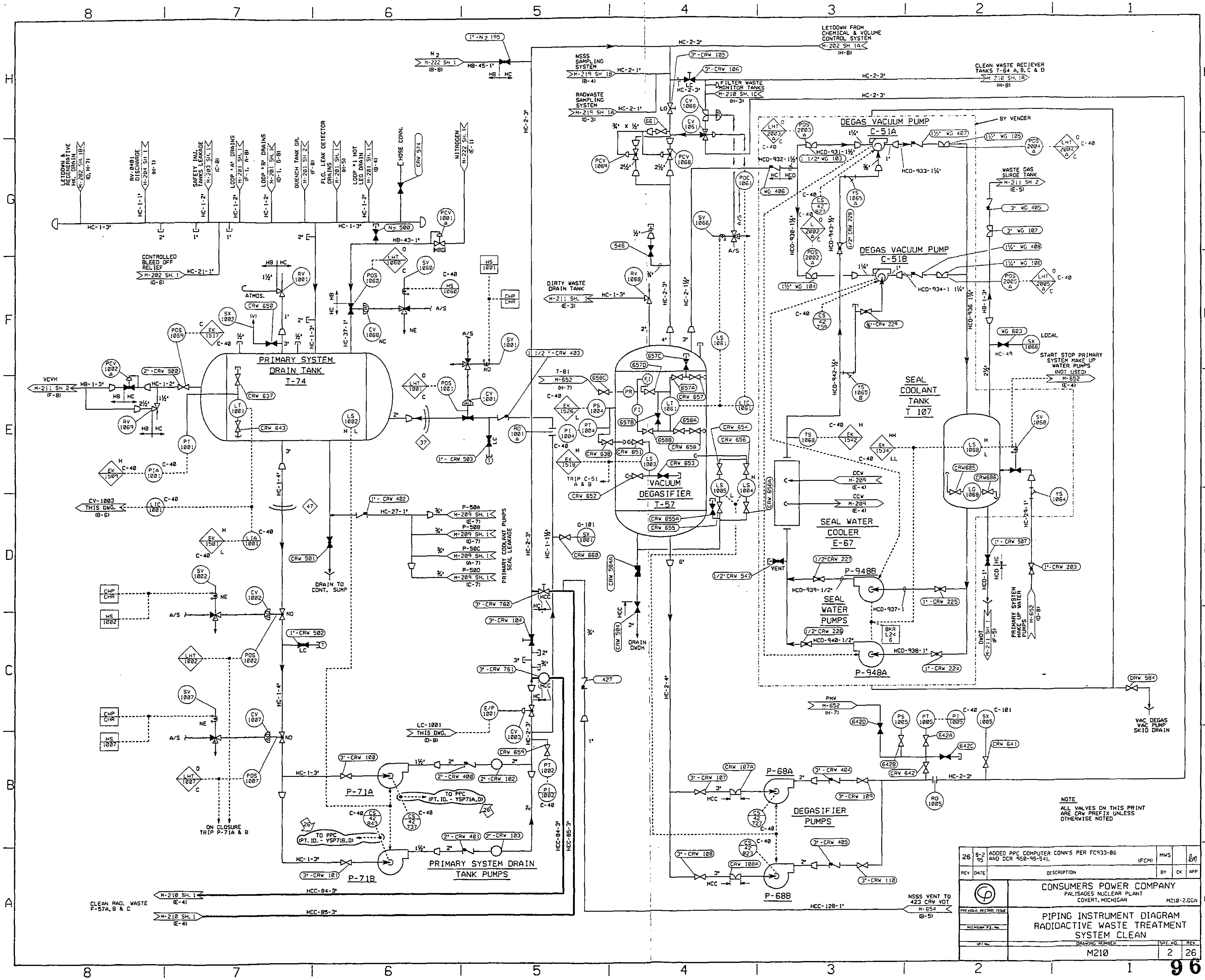

Figure 11-1 Sh 2 FSAR Rev 18

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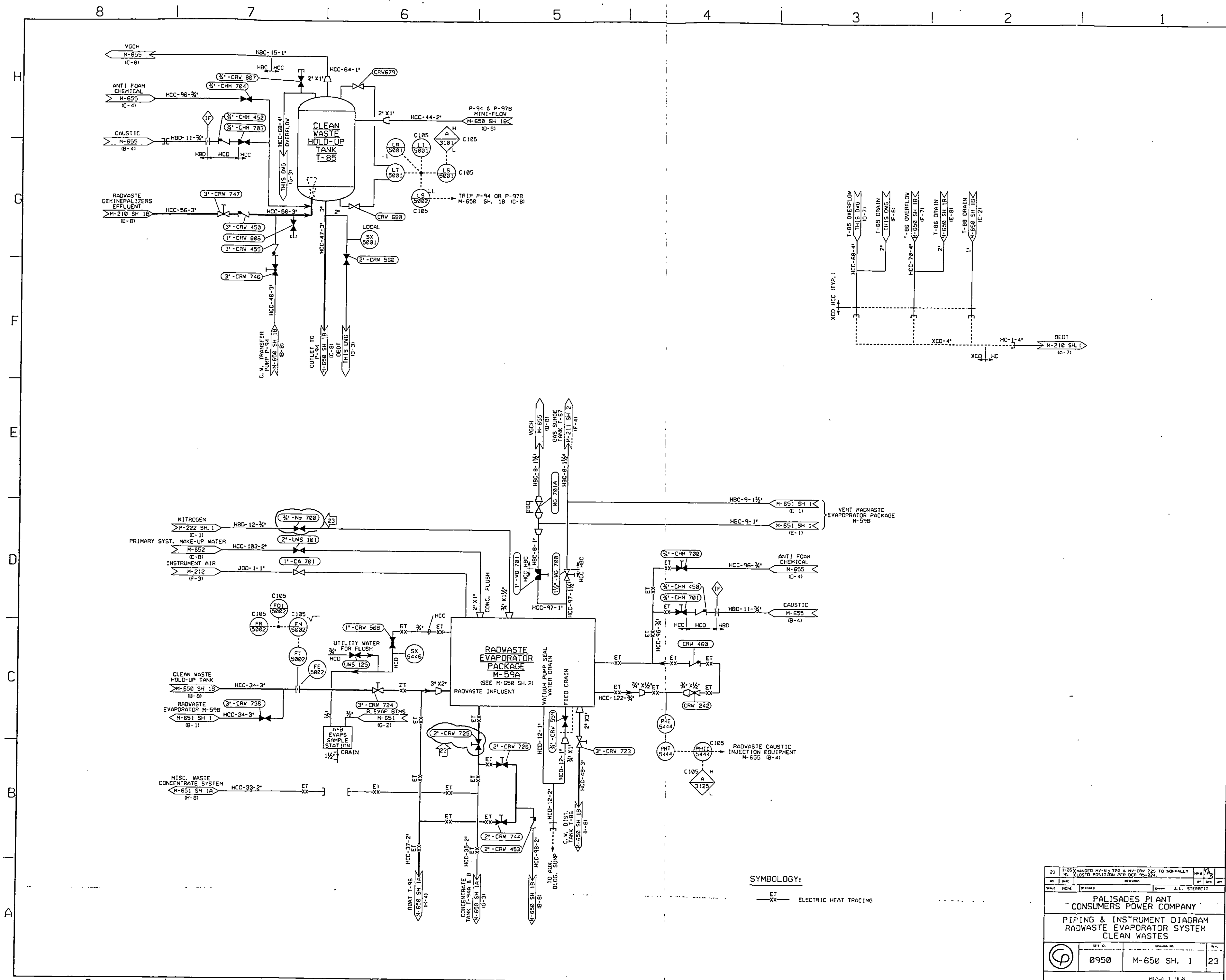

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Figure 11-1 Sh 3 FSAR Rev 18

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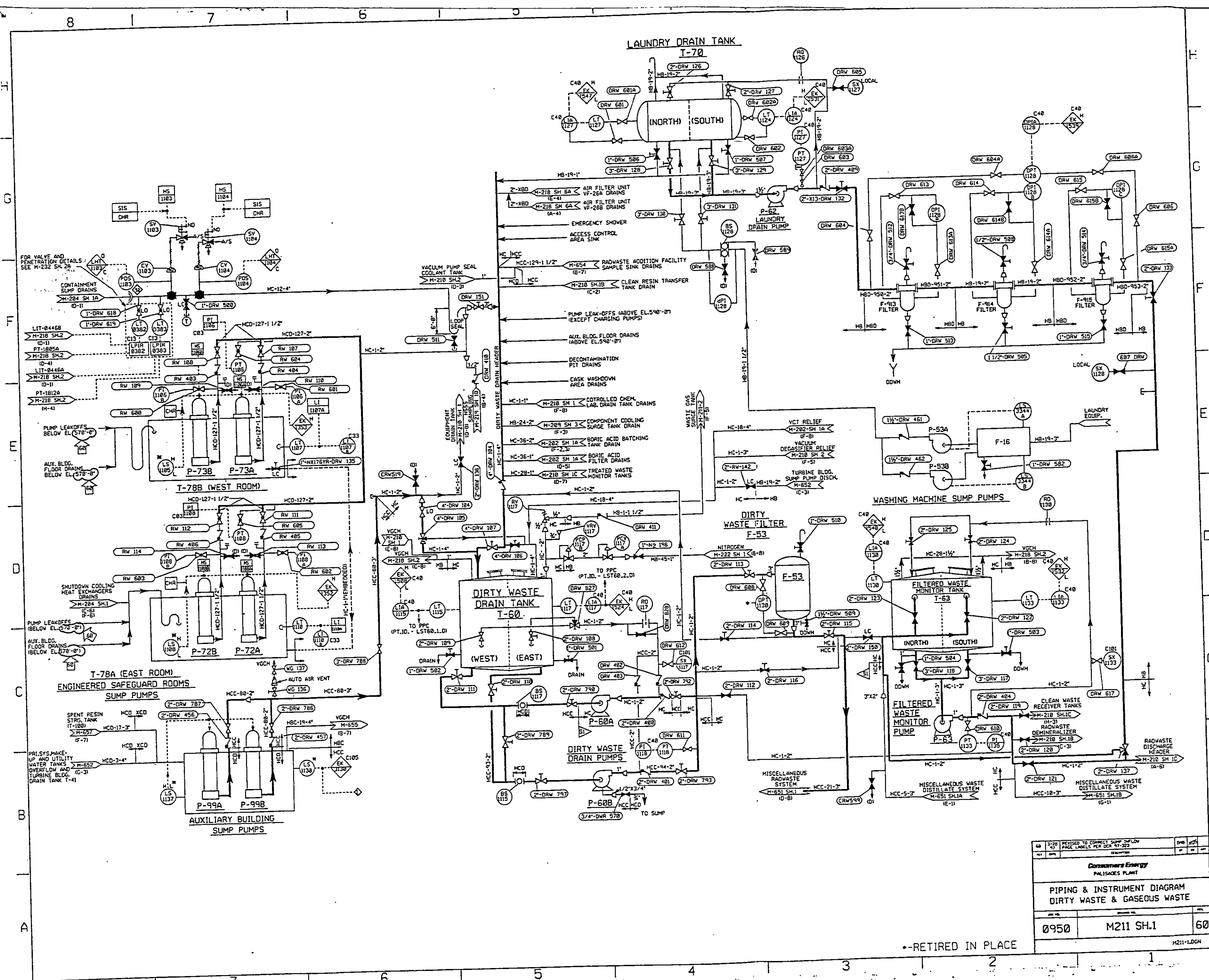

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Figure 11-2 Sh 1
Rev 19

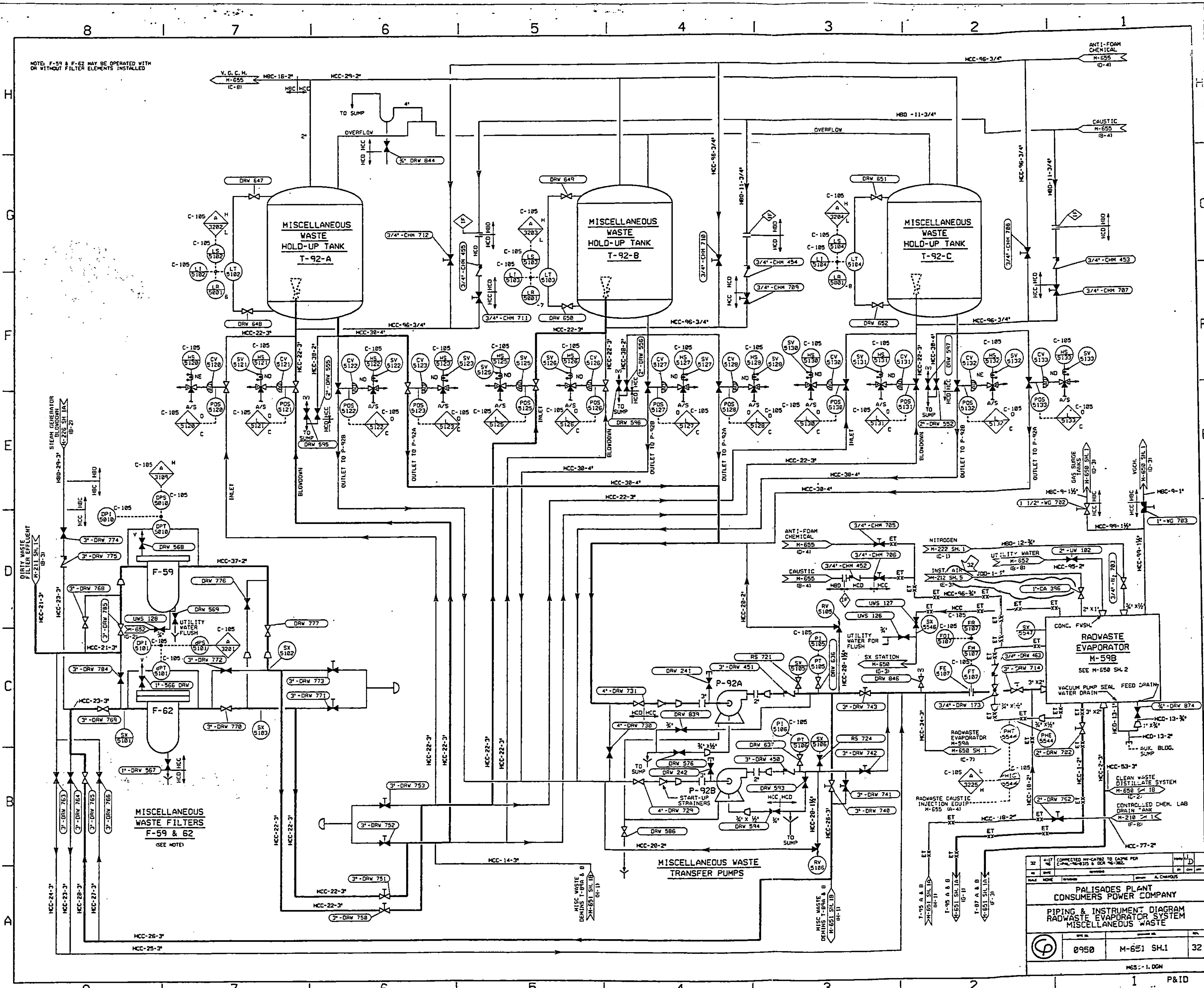

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Figure 11-2 Sh 2
Rev 19

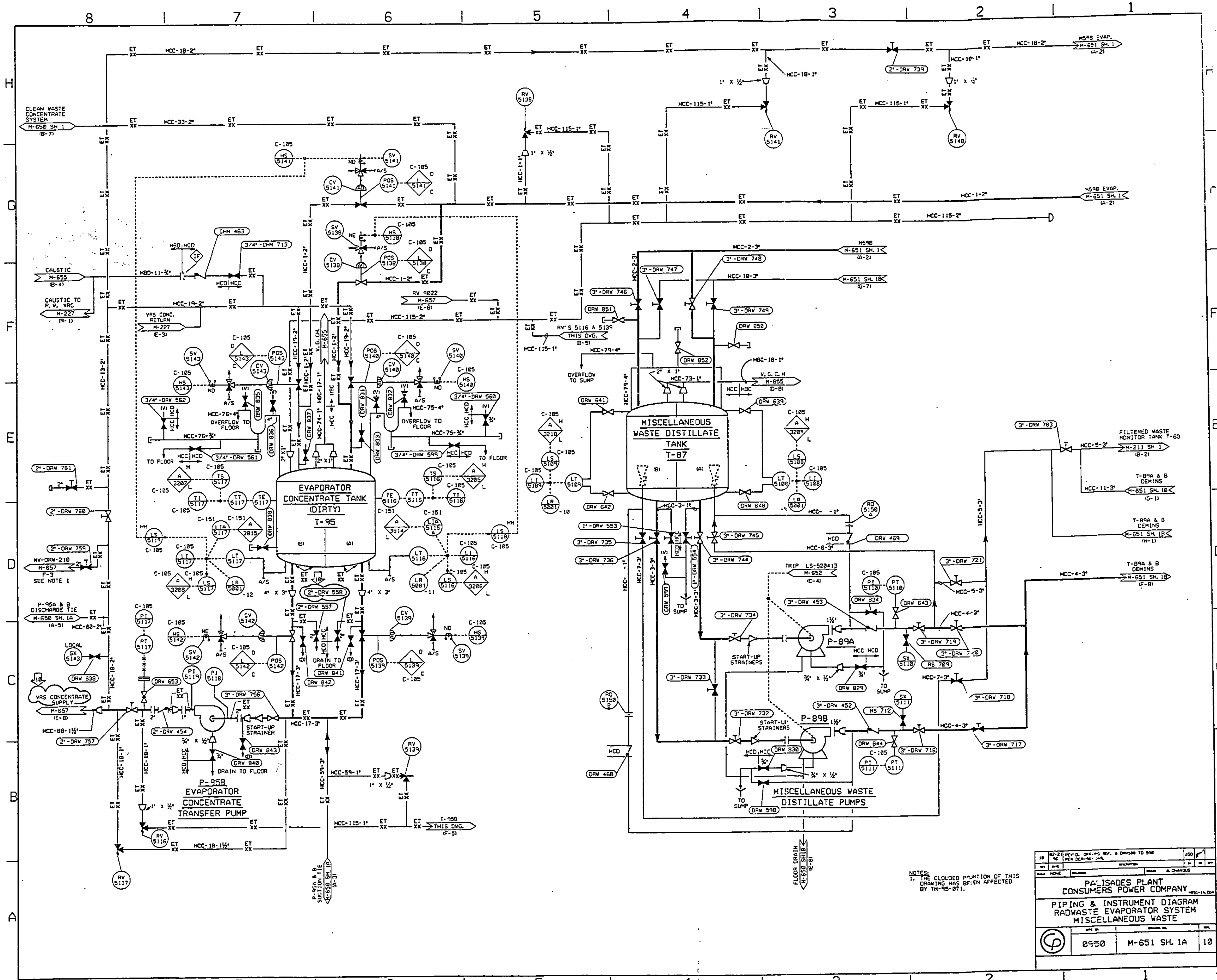

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Figure 11-2 Sh 2A
Rev 19

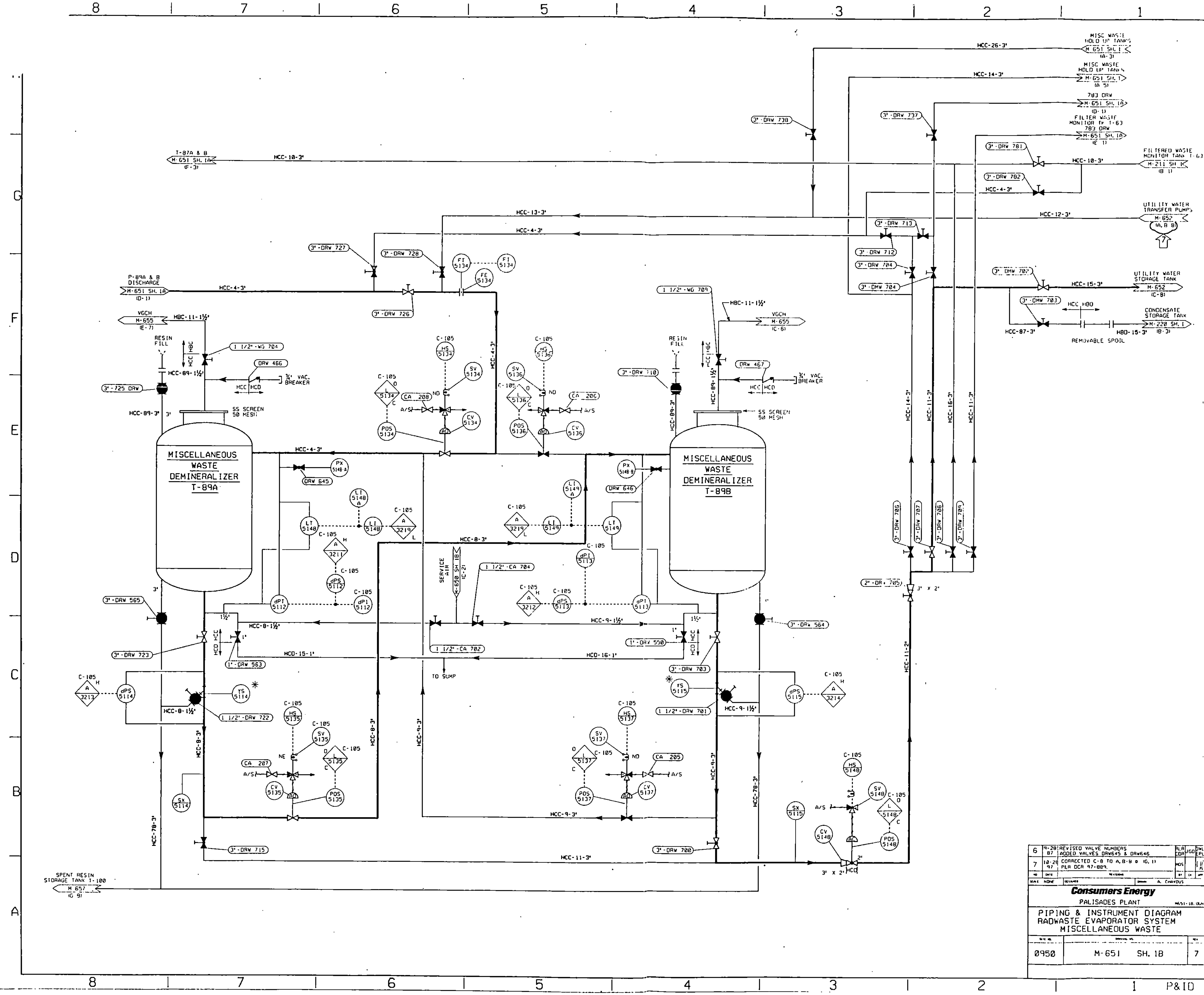

FIG 11-2 SH 2B
FSAR REV 20

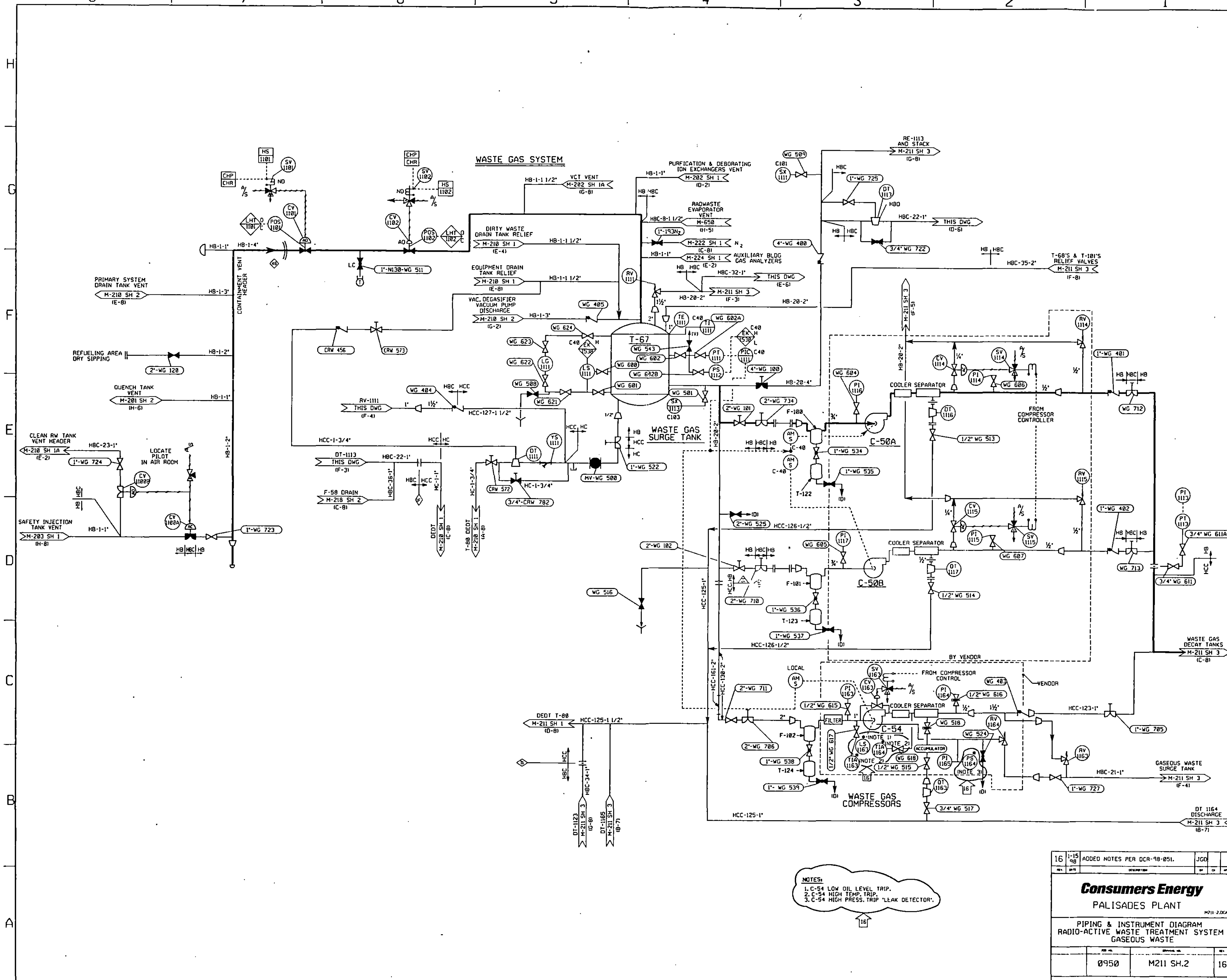

FIG 11-3 SH 1
FSAR REV 20

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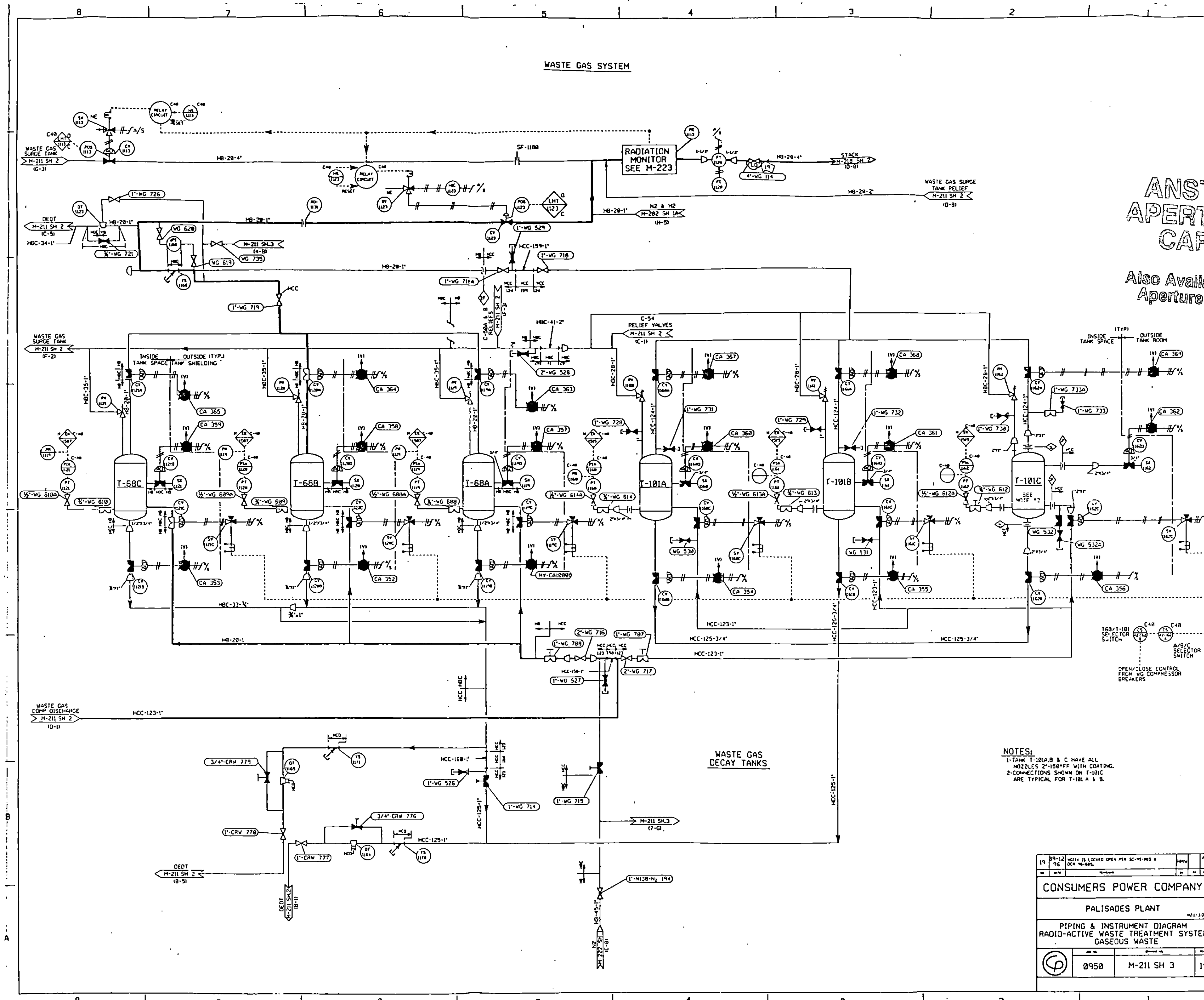

FIG 11-3 SH 2
FSAR REV 20

9807090043 - 65

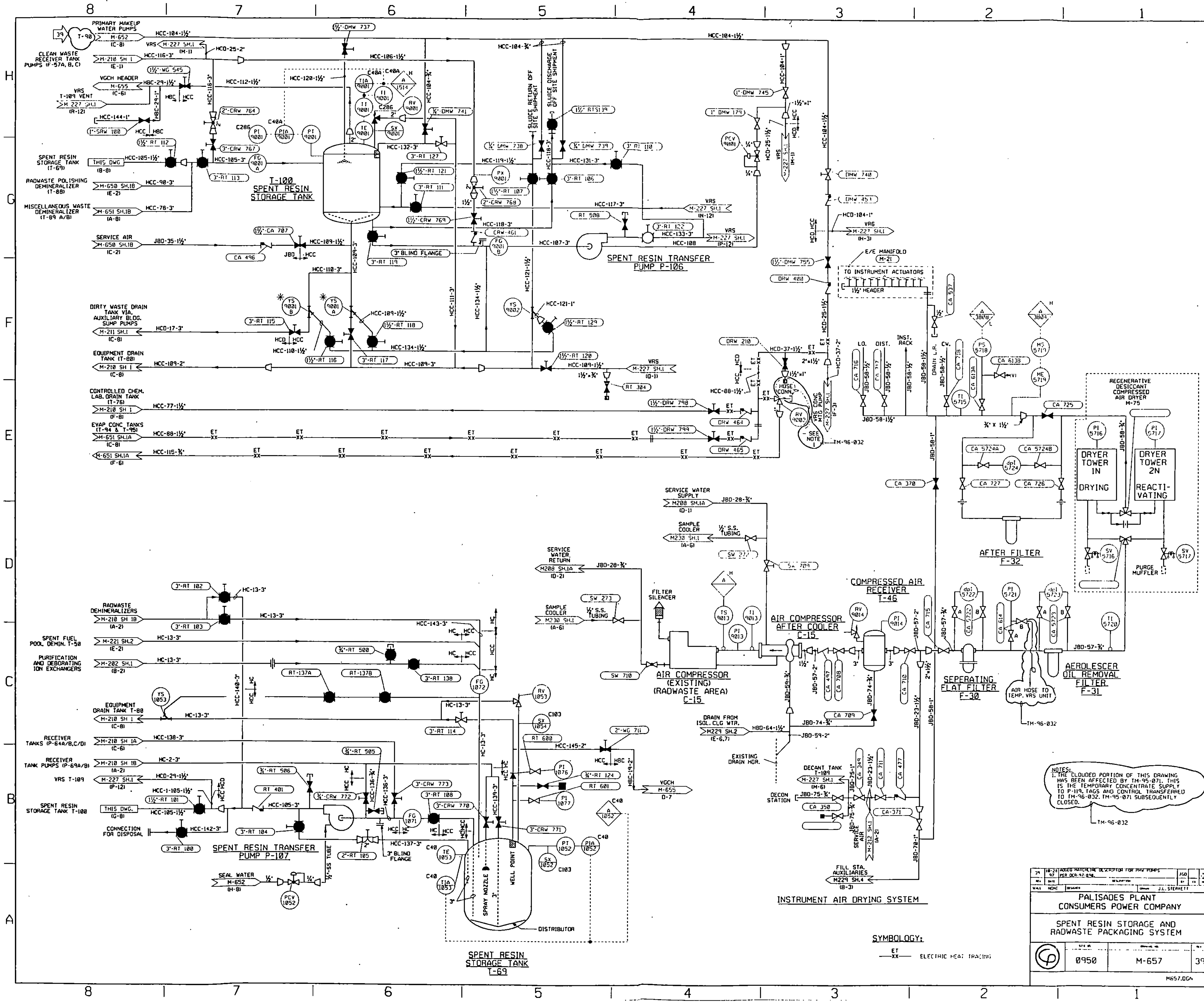

FIG 11-4 SH 1
FSAR REV 20

9807090043 -66

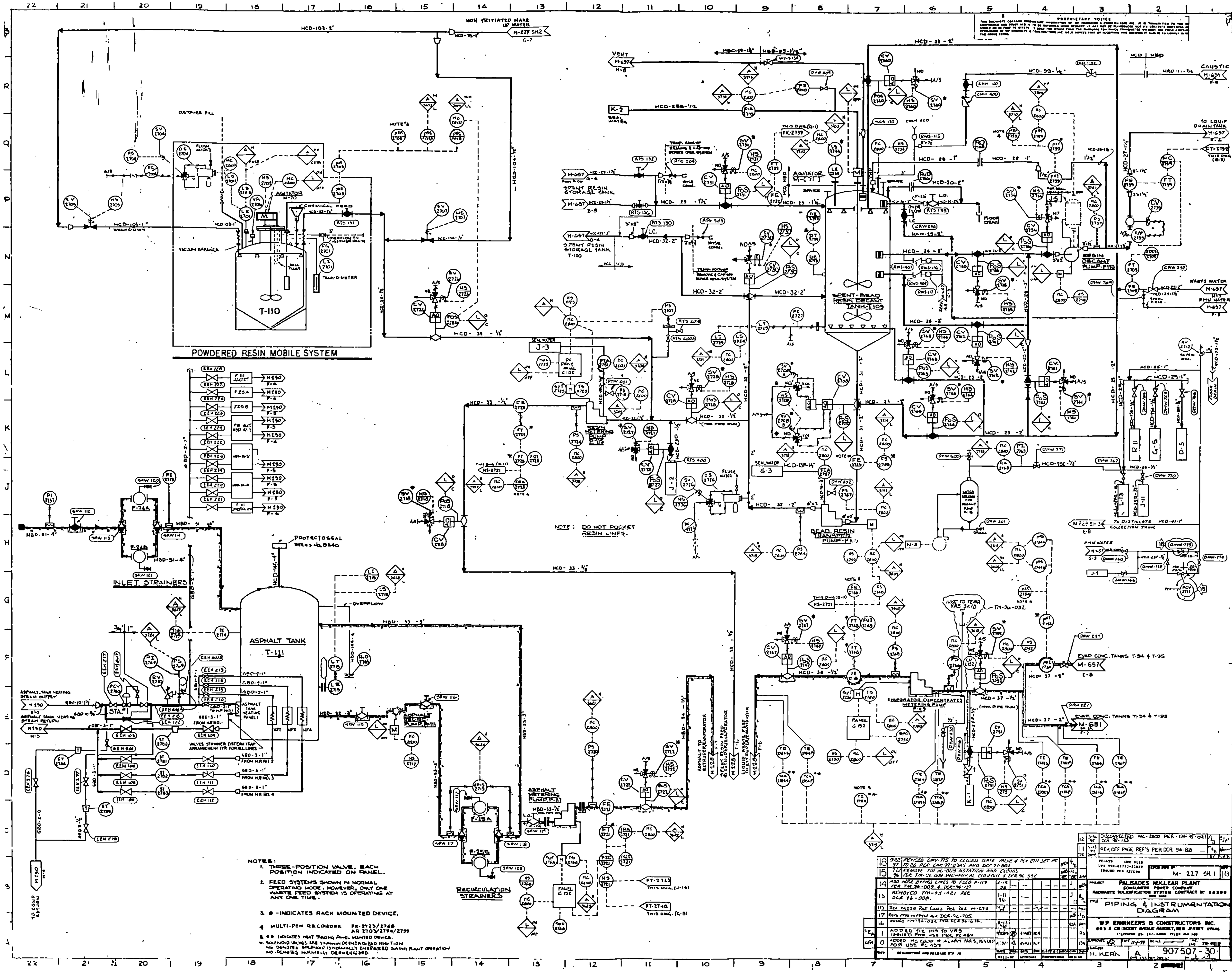

FIG 11-4 SH 2
FSAR REV 20

ANSTEC
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Also Available on
Aperture Card

9807090043 - 67

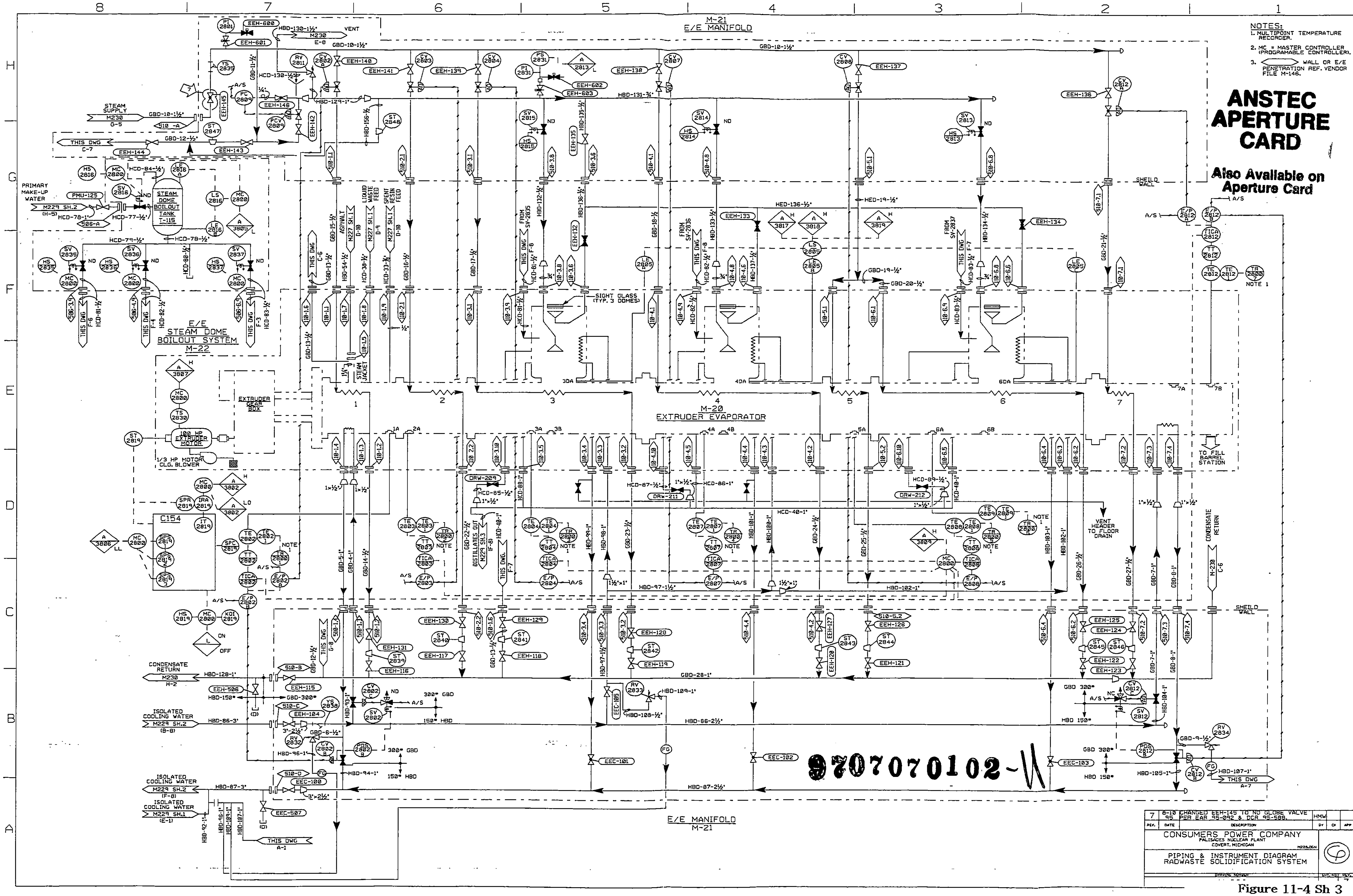
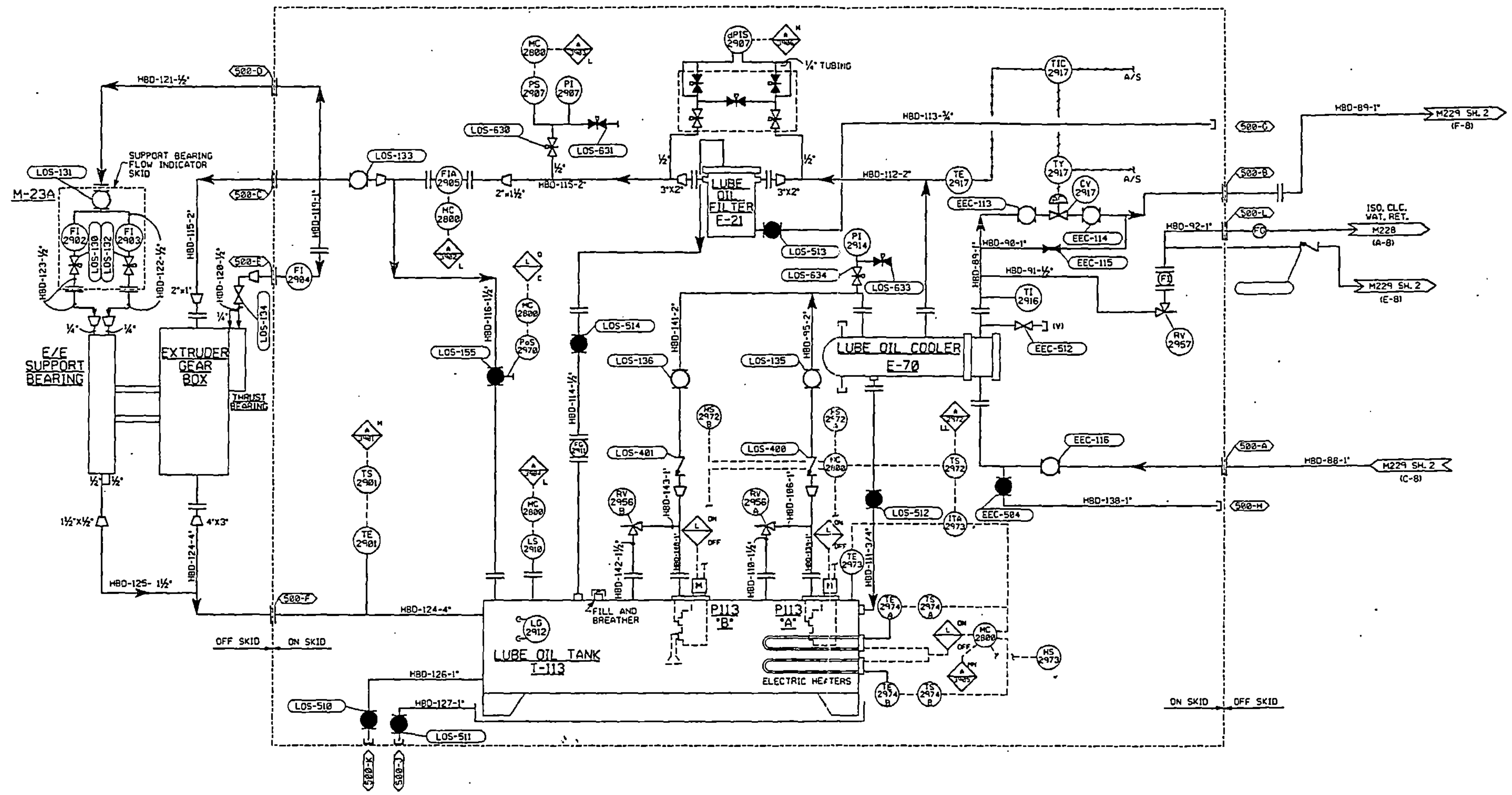


Figure 11-4 Sh 3
Rev 19



M-23
LUBE OIL SYSTEM

10	09-08-94	REDRAWN ON CAD PER DCR-94-821	JGD		
REV.	DATE	DESCRIPTION		CK	APP

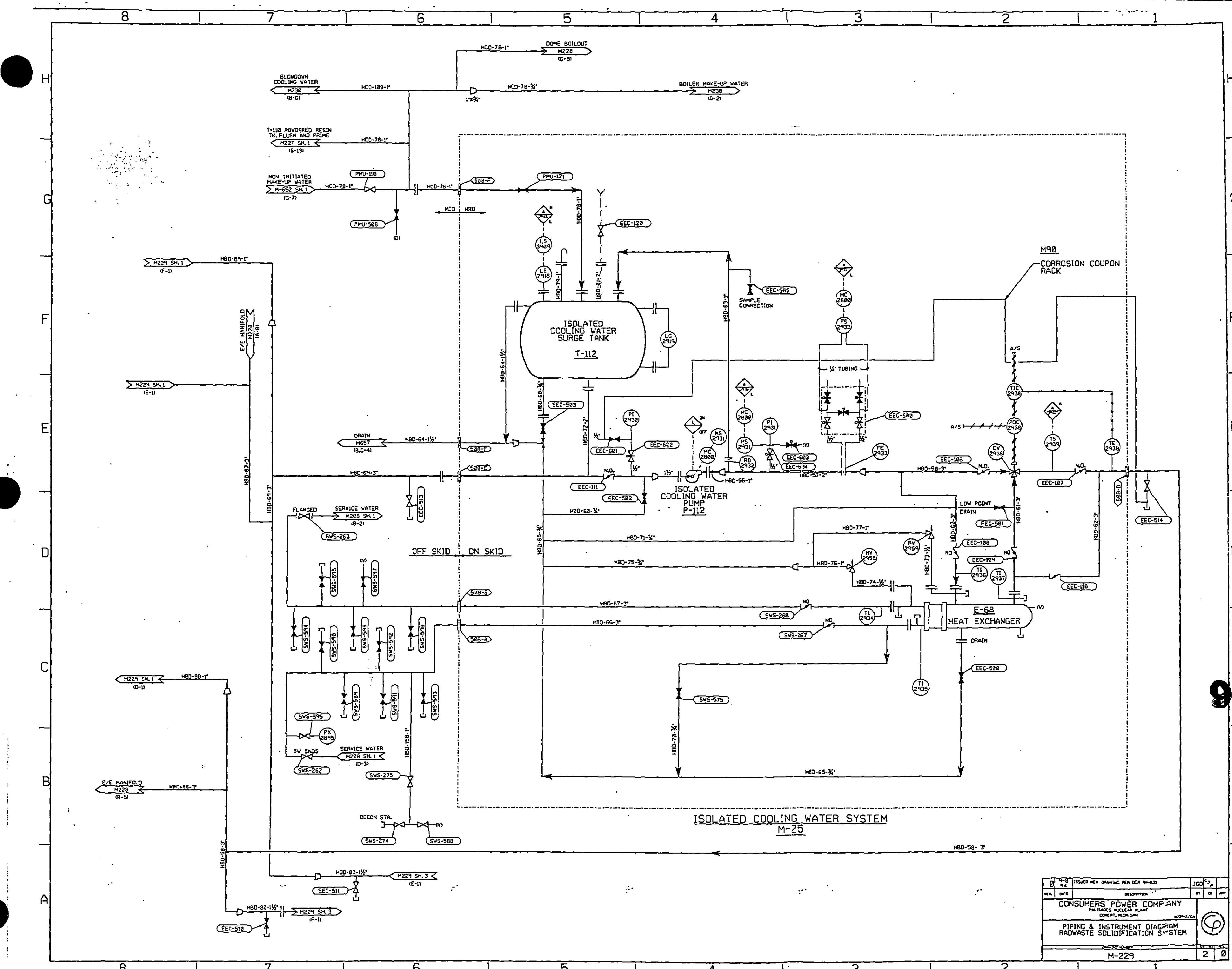
CONSUMERS POWER COMPANY
PALISADES NUCLEAR PLANT
COVERT, MICHIGAN

PIPING AND INSTRUMENT DIAGRAM
RADWASTE SOLIDIFICATION SYSTEM

M-229 | 1 | 10

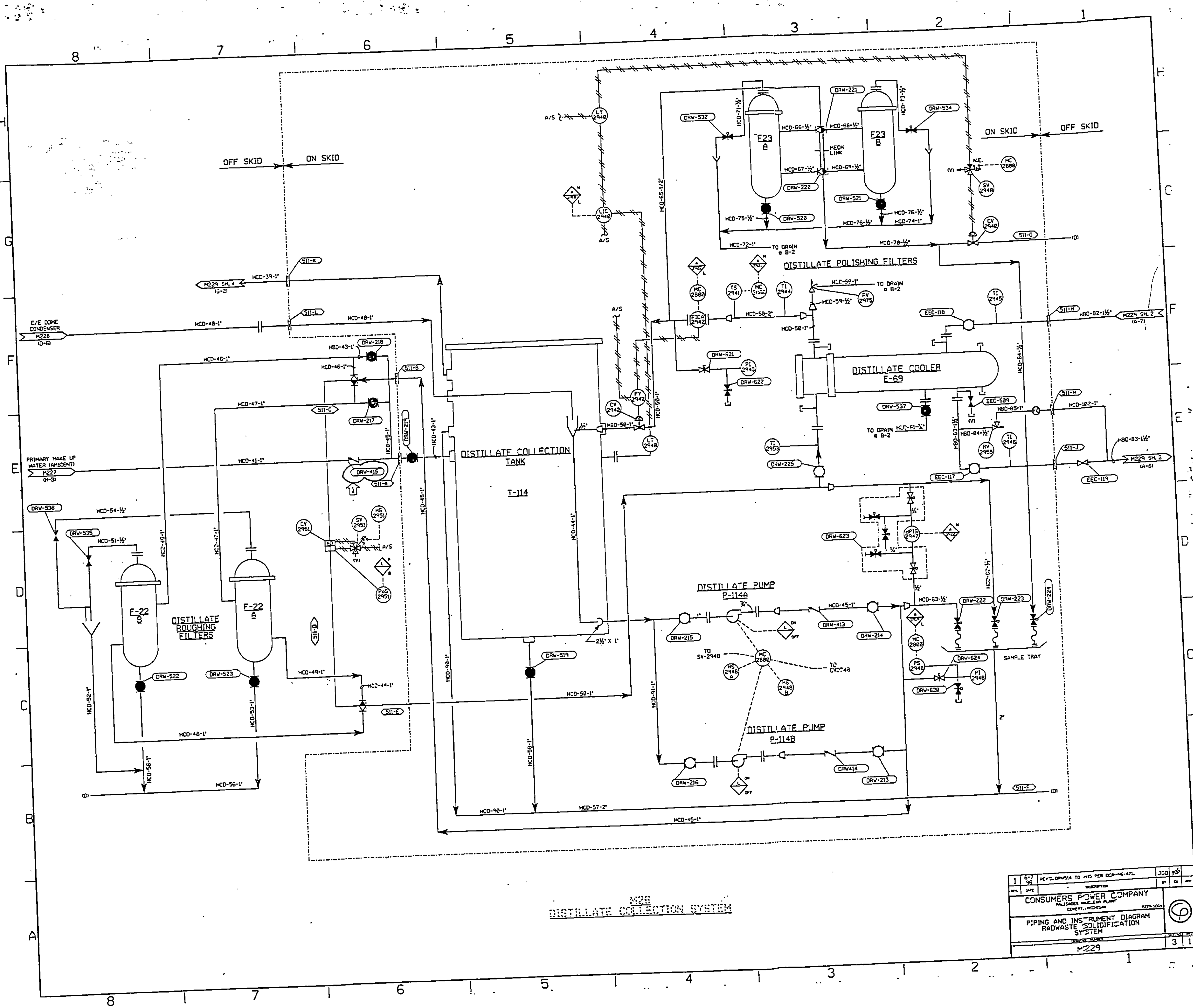
9707070102 -12

Figure 11-4 Sh 4A
Rev 19



9707070102 -13

Figure 11-4 Sh 4B
Rev 19

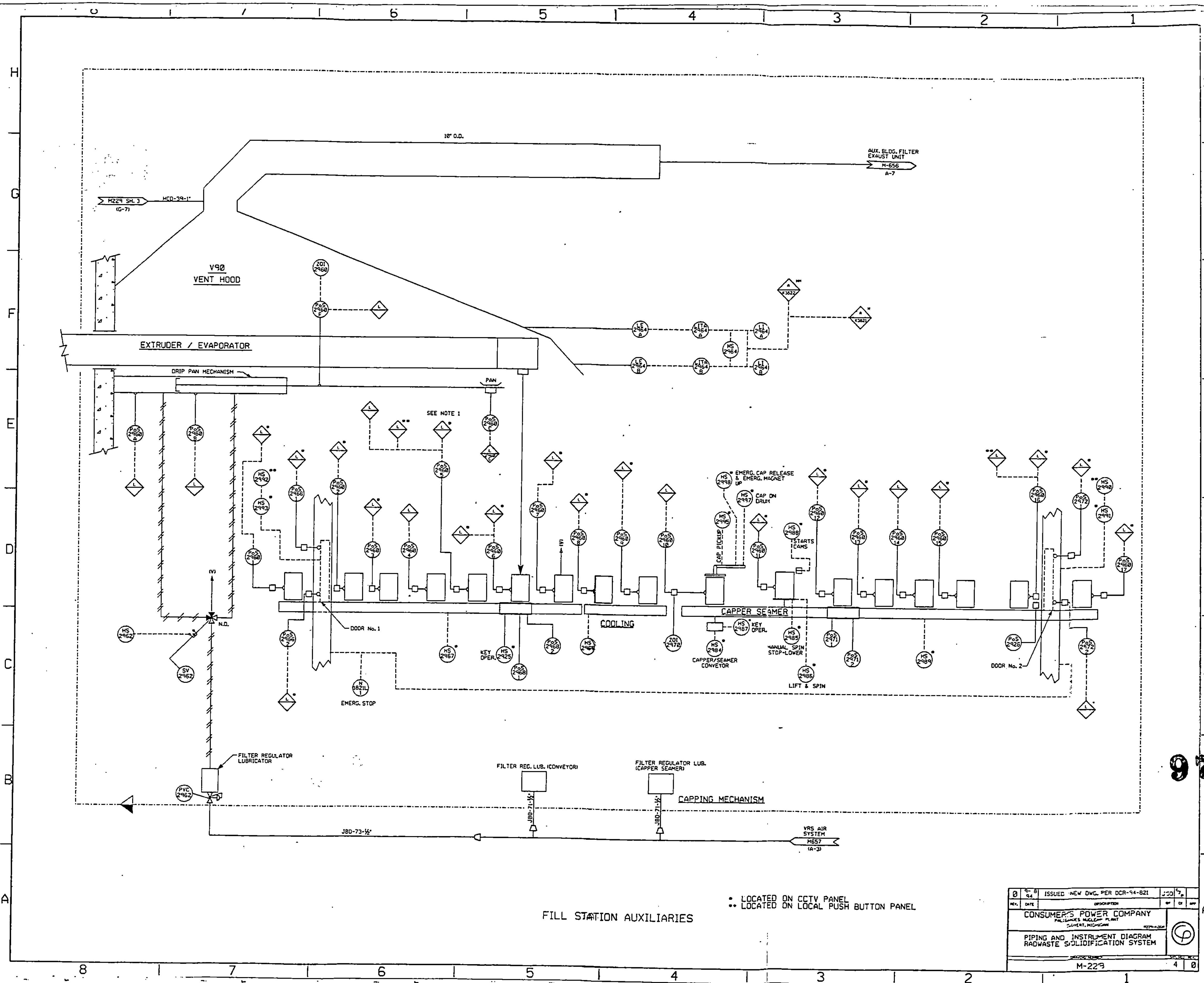


ANSTEC APERTURE CARD

Also Available on Aperture Card

9707070102 -14

Figure 11-4 Sh 4C
Rev 19

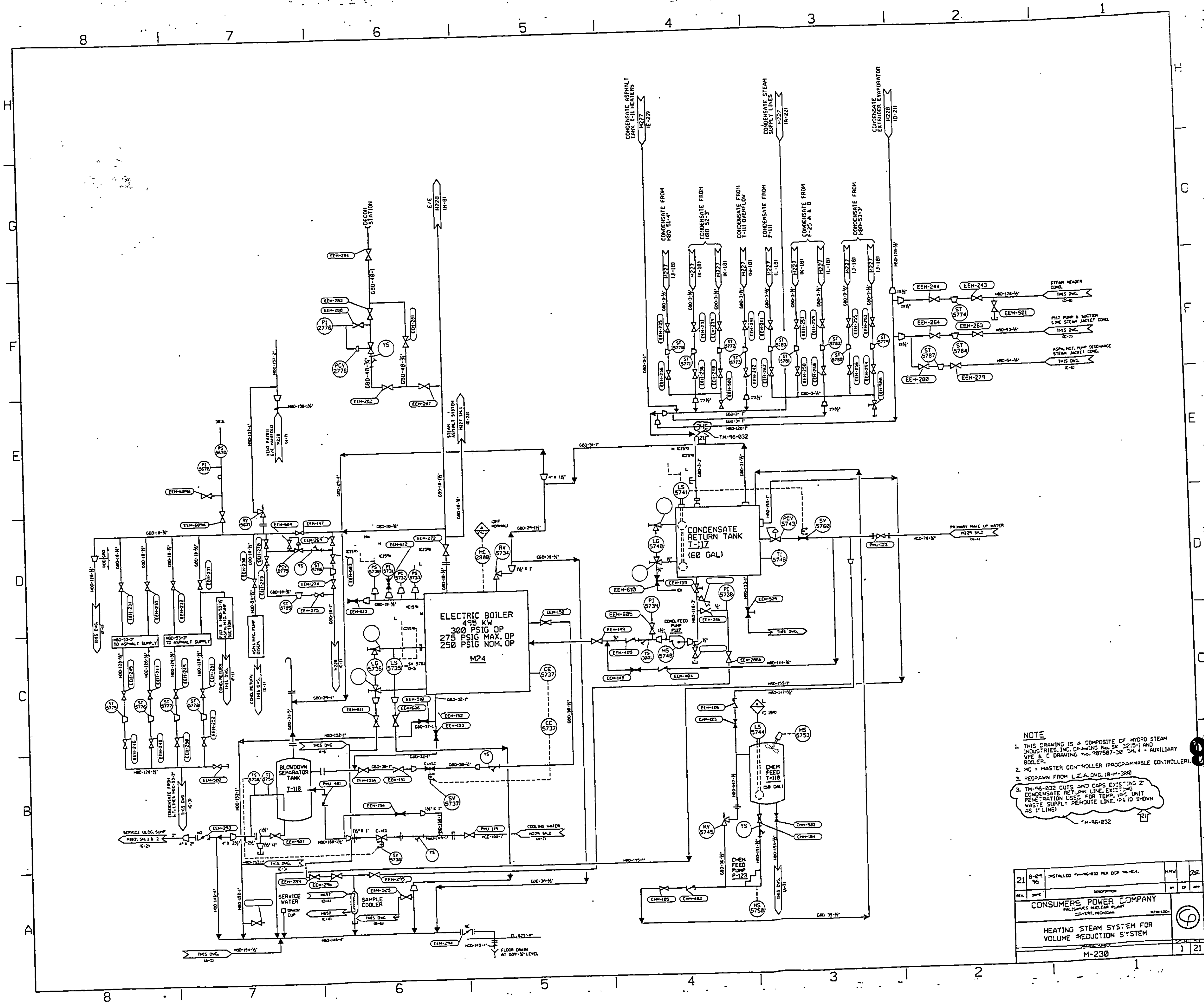


ANSTEC APERTURE CARD

Also Available on Aperture Card

9707070102 -15

Figure 11-4 Sh 4D
Rev 19



ANSTEC
APERTURE
CARD

Also Available on
Aperture Card

9707070102 -16

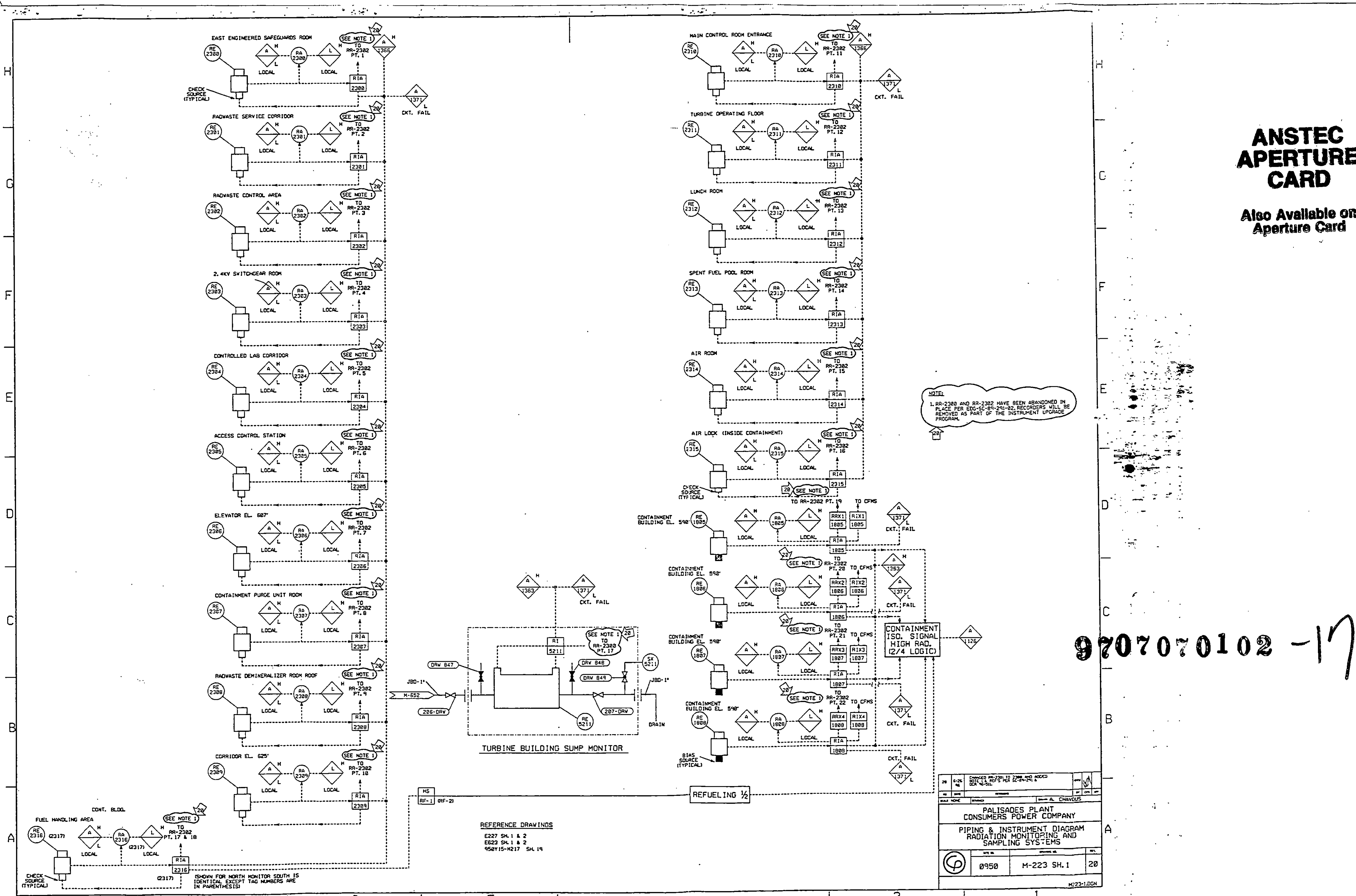
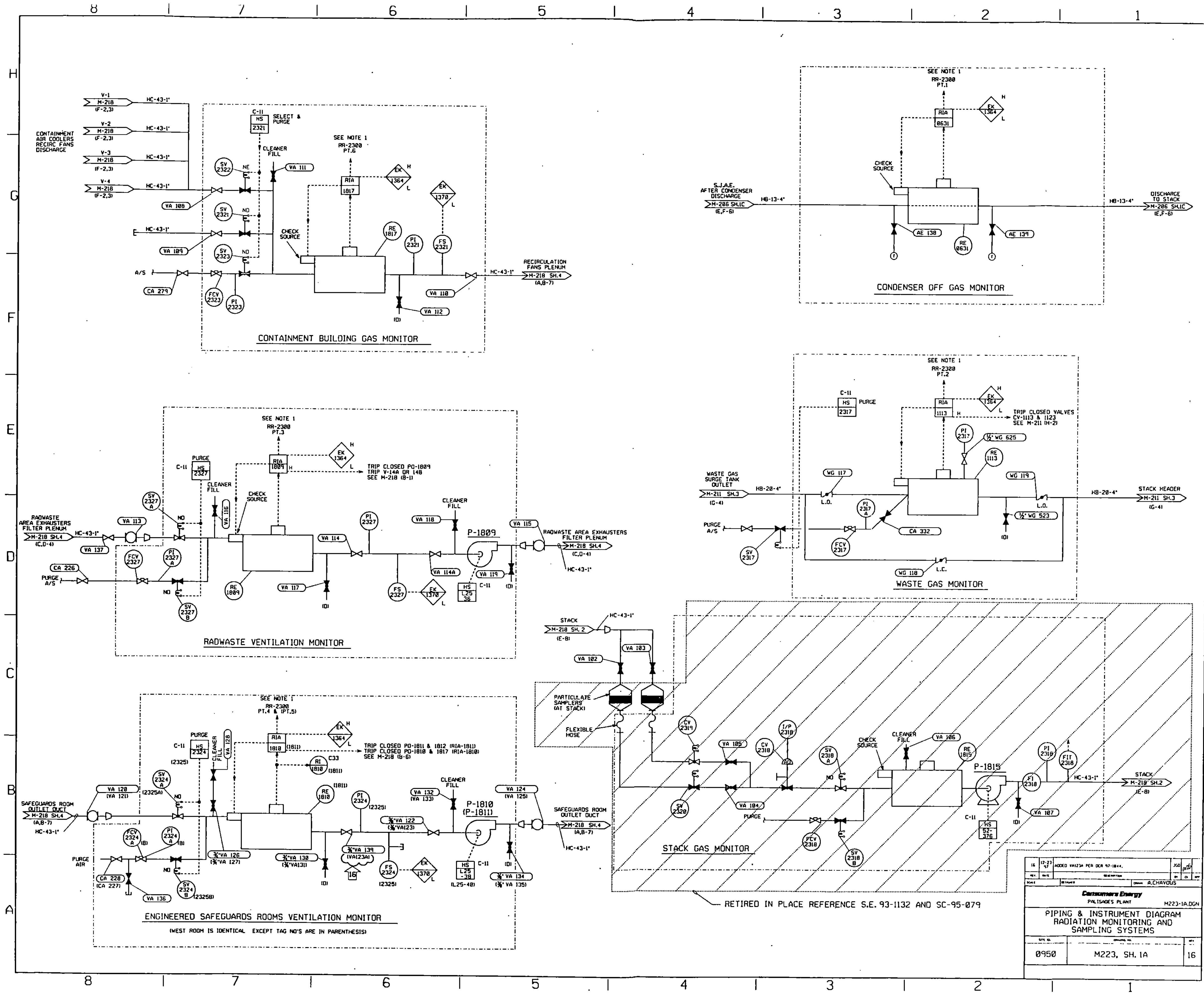


Figure 11-5 Sh 1
Rev 19



ANSTEC APERTURE CARD

Also Available on Aperture Card

FIG 11-5 SH 1A
FSAR REV 20

9807090043 - 68

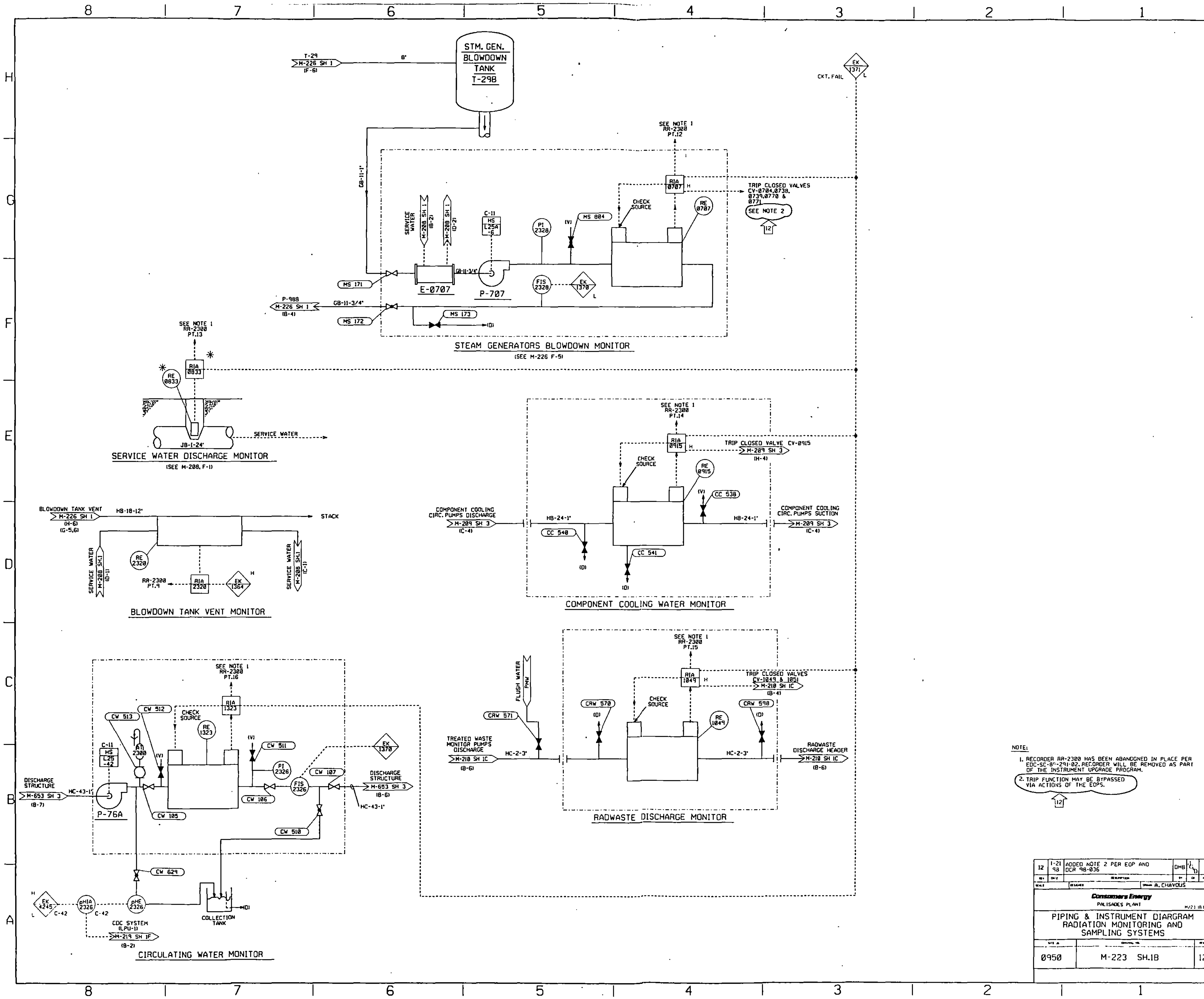
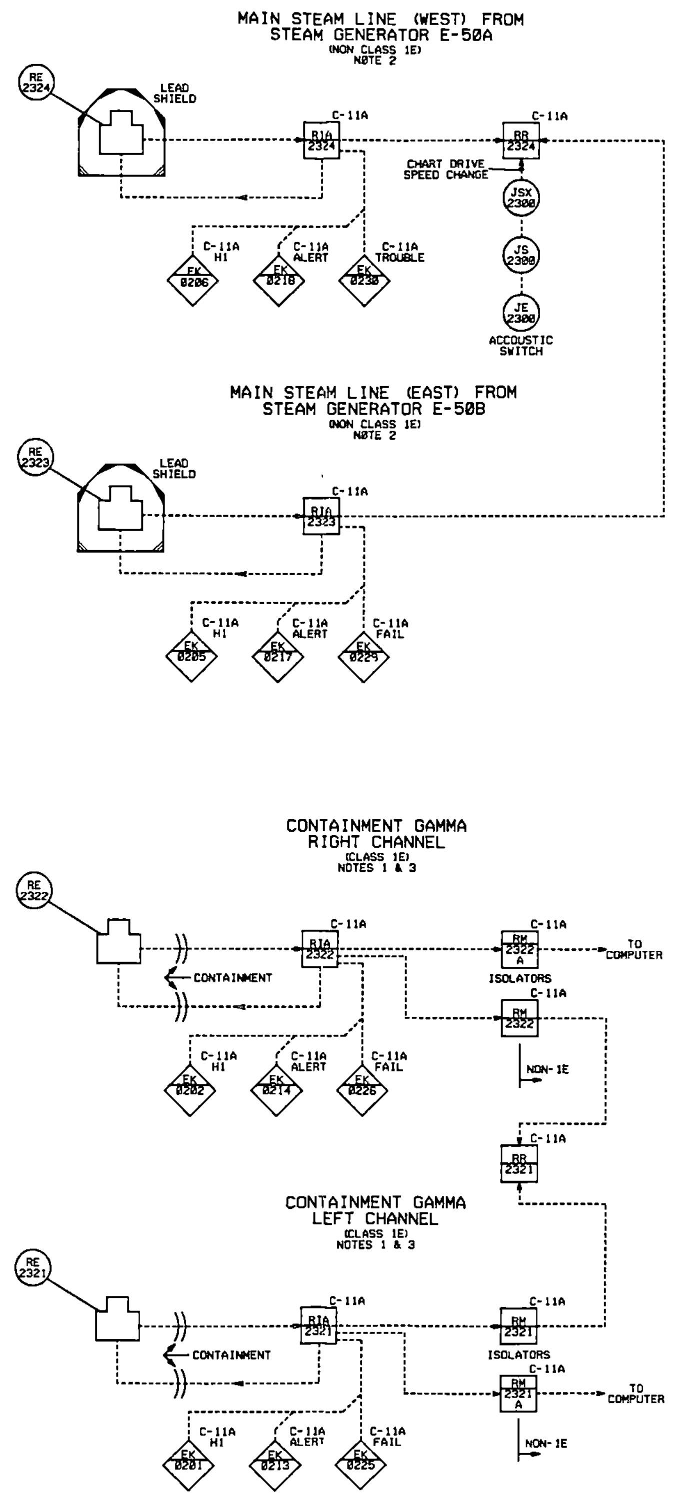


FIG 11-5 SH 1B
FSAR REV 20

9807090043 -69



SI
APERTURE
CARD

Also Available On
Aperture Card

NOTES:

1. REFER TO CAI DRAWING 0330-1-771-51-02-001, CPCO DWG. NO. M-165, SH. 0021 FOR CONTAINMENT GAMMA LOOP DIAGRAM.
2. REFER TO VICTOREEN DRAWINGS CPCO DRAWING NUMBERS M-169, SHTS. 0003 THRU 0007 AND SHTS. 0012 THRU 0014.
3. REFER TO VICTOREEN DRAWINGS CPCO DRAWING NUMBERS M-165, SHTS. 0003 THRU 0015.

FC-494-1
FC-494-5

NO	DATE	REVISIONS	BY	CHK	APP
4	7-13 90	REVISED ANNUNCIATOR NOMENCLATURE TO EK-0201, -0202, 0205, 0206, 0213, 0214, 0217, -0218, -0225, -0226, -0229, -0230, FROM A K-02-1, -2, -5, -6, -13, -14, -17, -18, -25 -26, -29, -30, PER DCR 950-90-0058 CCP (DEK)	LLC	RIS	JGD

SCALE NONE | DESIGNED | DRAWN J.L. STERRETT

PALISADES PLANT
CONSUMERS POWER COMPANY

PIPING & INSTRUMENT DIAGRAM
RADIATION MONITORING AND
SAMPLING SYSTEMS

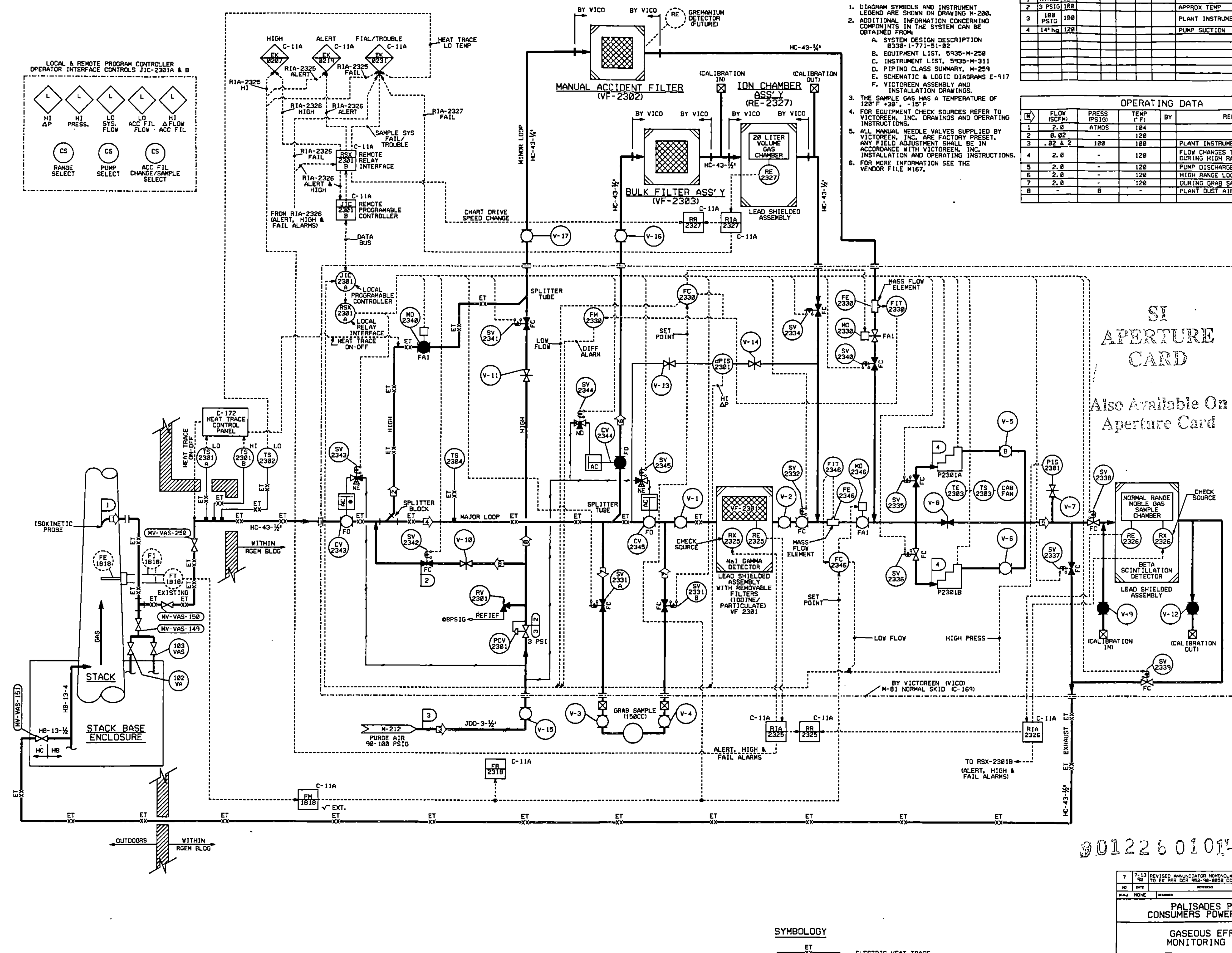
SITE ID.	DRAWING NO.	REV.
0950	M-223 SH. 3	4

0950M223.003

JS0032104

Figure No. 11-5 sh 2

9012260101-123



NOTES:

1. DIAGRAM SYMBOLS AND INSTRUMENT LEGEND ARE SHOWN ON DRAWING M-200.
2. ADDITIONAL INFORMATION CONCERNING COMPONINTS IN THE SYSTEM CAN BE OBTAINED FROM:
   A. SYSTEM DESIGN DESCRIPTION 0330-1-771-51-02
   B. EQUIPMENT LIST, 5935-M-250
   C. INSTRUMENT LIST, 5935-M-311
   D. PIPING CLASS SUMMARY, M-259
   E. SCHEMATIC & LOGIC DIAGRAMS E-917
   F. VICTOREEN ASSEMBLY AND INSTALLATION DRAWINGS.
3. THE SAMPLE GAS HAS A TEMPERATURE OF 120°F +30°, -15°F
4. FOR EQUIPMENT CHECK SOURCES REFER TO VICTOREEN, INC. DRAWINGS AND OPERATING INSTRUCTIONS.
5. ALL MANUAL NEEDLE VALVES SUPPLIED BY VICTOREEN, INC. ARE FACTORY PRESET. ANY FIELD ADJUSTMENT SHALL BE IN ACCORDANCE WITH VICTOREEN, INC. INSTALLATION AND OPERATING INSTRUCTIONS.
6. FOR MORE INFORMATION SEE THE VENDOR FILE M167.

DESIGN DATA

P	NORMAL		TRANSIENT			BY	CHKD	REMARKS	REV
	PRESS	°F	PSIG	°F	TIME				
1	ATMOS	104	-	-	-				
2	3 PSIG	100						APPROX TEMP	
3	100 PSIG	100						PLANT INSTRUMENT AIR	
4	14" hg	120						PUMP SUCTION	

OPERATING DATA

#	FLOW (SCFM)	PRESS (PSIG)	TEMP (°F)	BY	REMARKS	REV
1	2.0	ATMOS	104			
2	0.02	-	120			
3	.02 & 2	100	100		PLANT INSTRUMENT AIR	
4	2.0	-	120		FLOW CHANGES TO 1.98 SCFM DURING HIGH RANGE MODE.	
5	2.0	-	120		PUMP DISCHARGE	
6	2.0	-	120		HIGH RANGE LOOP	
7	2.0	-	120		DURING GRAB SAMPLE ONLY	
8	-	8	-		PLANT DUST AIR	

9012260101-124

NO	DATE	REVISIONS	BY	CHK	APP
7	7-13 90	REVISED ANNUNCIATOR NOMENCLATURE FROM A TO EK PER DCR 950-90-0058 CCP (DEK)	LLC	RIS	JOD

SCALE NONE | DESIGNED | DRAWN J.L. STERRETT

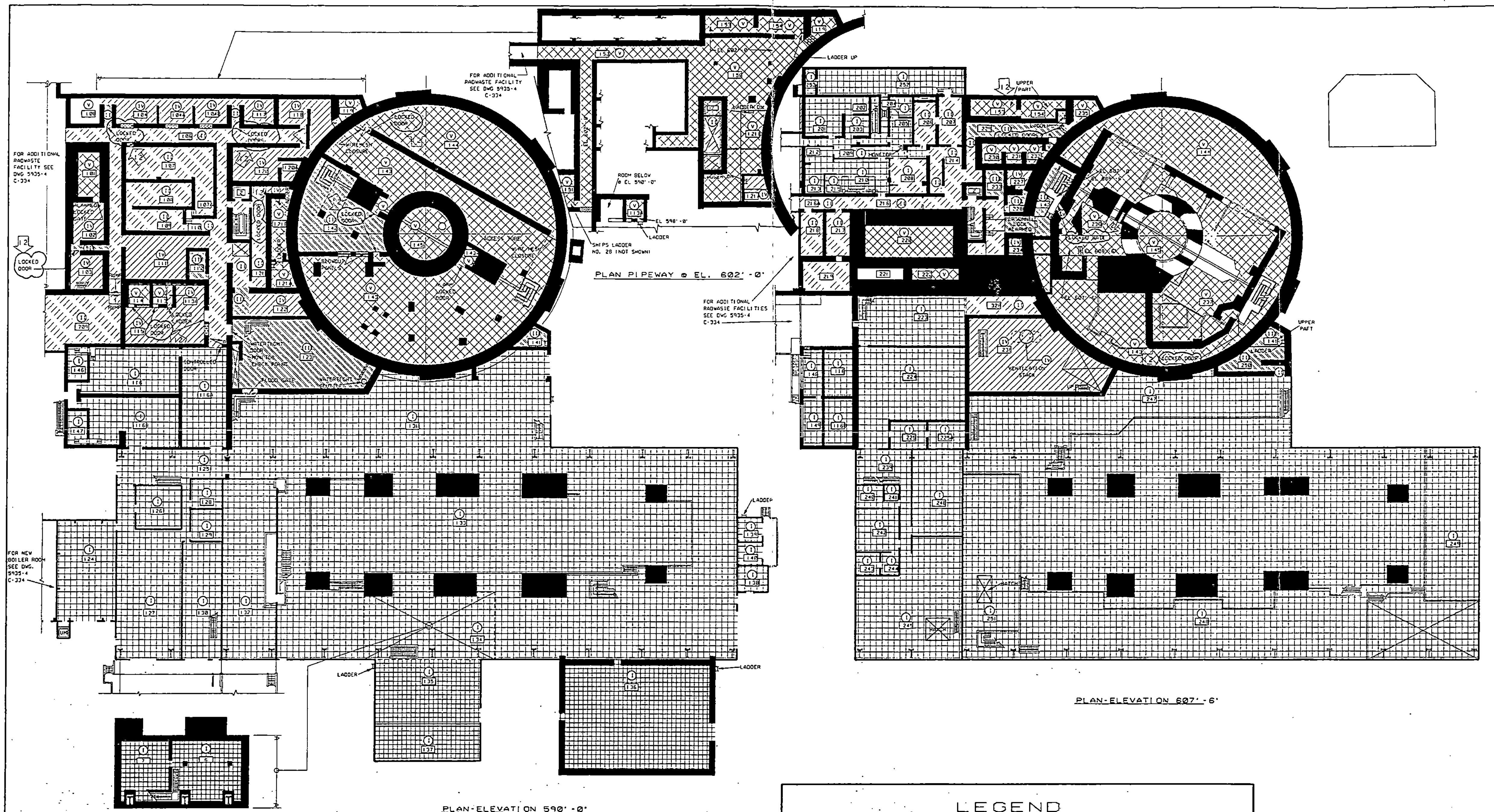
PALISADES PLANT
CONSUMERS POWER COMPANY

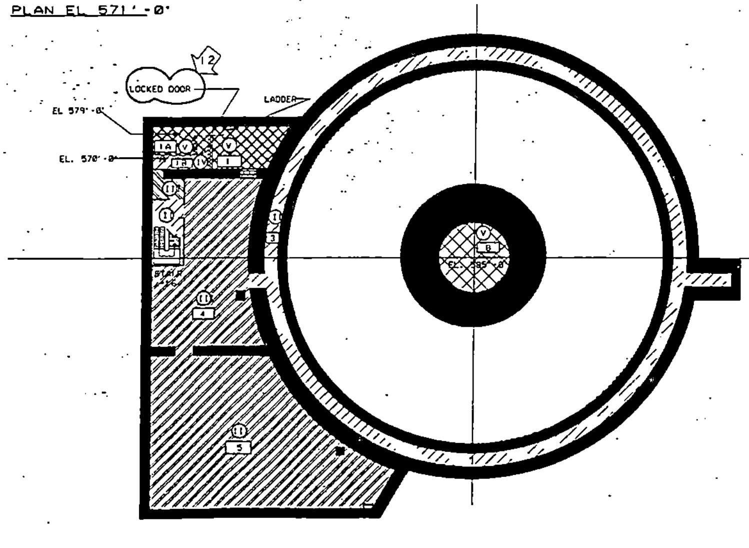
GASEOUS EFFLUENT
MONITORING SYSTEM

SITE NO.	DRAWING NO.	REV.
0950	M-223 SH. 2	7

0950M223.002

Figure No. 11-6 FSAR Rev. 11





LEGEND

SYMBOL	ZONE	DESIGN DOSE RATE MREM/HR.	DESCRIPTION
	I	≤ 0.5	UNCONTROLLED UNLIMITED ACCESS
	II	≤ 1.5	CONTROLLED UNLIMITED ACCESS 40 HRS/WEEK
	III	≤ 15	CONTROLLED LIMITED ACCESS FOR ROUTINE TASKS 6 HRS/WEEK
	IV	≤ 100	CONTROLLED LIMITED ACCESS FOR SHORT PERIODS 1 HR/WEEK
	V	> 100	CONTROLLED OCCUPANCY FOR VERY SHORT PERIODS OCCUPANCY DURING EMERGENCIES NORMALLY UNACCESSIBLE

RADIATION ZONES

NOTE:

1. DOORS SHOWN ARE THOSE THAT ARE REQUIRED BY ACCESS CONTROL ONLY.
2. ALL LOCKED DOORS OR LOCKED GATES FOR ZONE V AREAS SHALL BEAR THE RADIATION CAUTION SYMBOL AND THE WORDS. 'CAUTION HIGH RADIATION AREA'
3. ALL DOORS OR GATES FOR ZONE III AND ZONE IV AREAS SHALL BEAR THE RADIATION CAUTION SYMBOL AND THE WORDS. 'CAUTION RADIATION AREA'
4. L-DOOR - LOUVER DOOR
GATE - WIRE MESH GATE(CLOSING THE FULL OPENING)

GRAPHIC SCALE

12	9/13/94	CORRECTED REV 11 FROM MANUAL TO CAD PER DCR 94-393C	CJP	JD			
NO	DATE	REVISIONS	BY	CH'D	APP	ENG'R	

SCALE | DESIGNED | DRAWN R.M. SATTERELLI

PALISADES PLANT
CONSUMERS POWER COMPANY

ARCHITECTURAL
ACCESS CONTROL
AND RADIATION ZONING
(PLANT IN OPERATION)

JOB NO.	DRAWING NO	REV.
950	C-42	12

C42.DGN

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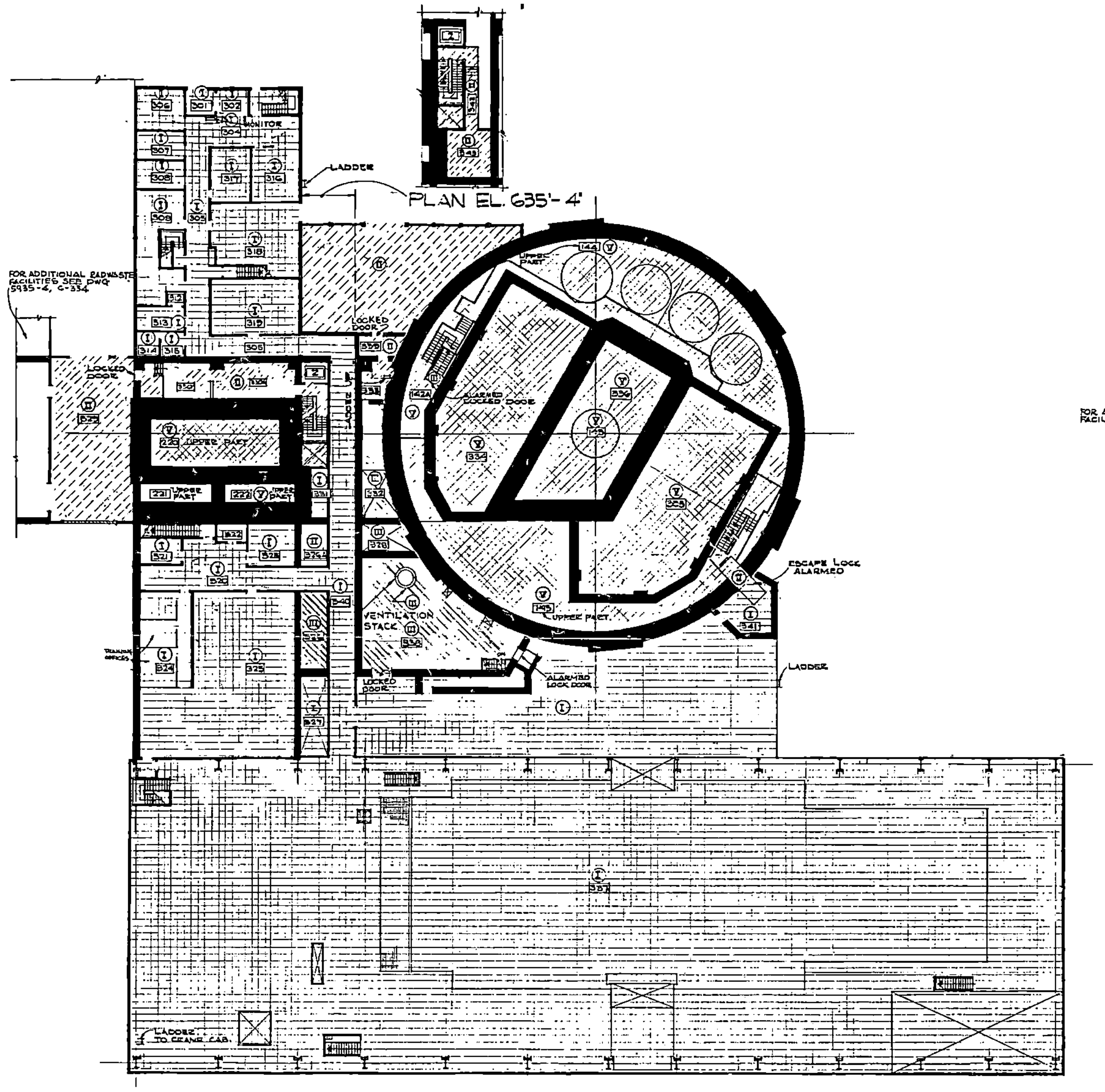
CARD

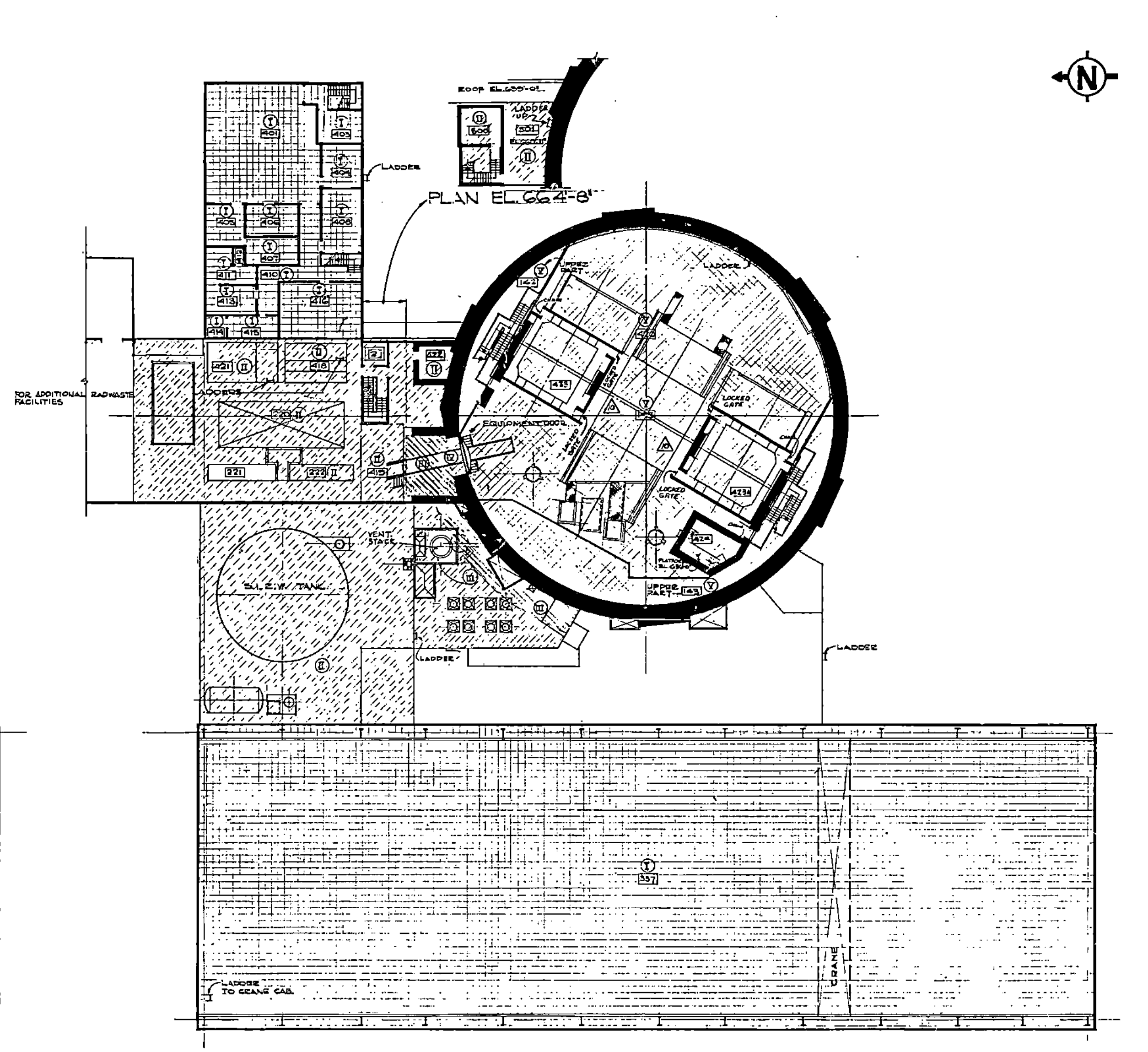
Also Available on
Aperture Card

Figure 11-7 FSAR Rev 18

9602290093-60



PLAN - ELEVATION 625'-0"



PLAN - ELEVATION 639'-0" & 649'-0"

NO.	ROOM OR AREA
1	EQUIPMENT DRAIN TANKS & PUMPS
1A	EQUIPMENT DRAIN TANK
1B	EQUIPMENT DRAIN PUMP
2	ELEVATOR SHAFT
3	POST-TENSIONING TUNNEL
4	INJECTION & SPRAY PUMPS
5	INJECTION & SPRAY PUMPS & SHUT-DOWN COOLING HEAT EXCHANGERS
6	CONDENSATE PUMPS
7	AUXILIARY F. W. PUMPS
8	SUMP
100	VOLUME CONTROL TANK
101	DECAY TANKS
102	WASTE GAS COMPRESSORS
103	LAUNDRY & RAD. CHEM. LAB. DRAIN PUMPS
104	CHARGING PUMP
104A	CHARGING PUMP
104B	CHARGING PUMP
105	
106	CORRIDOR
106A	CORRIDOR
107	BORIC ACID TANKS PUMPS & FILTER
107A	PUMPS & FILTER
108	TREATED & FILTERED WASTE MONITORING TANK
109	MONITORING PUMPS
110	BORIC ACID CONTROL PANEL
111	DECONTAMINATION ROOM
112	CONTROLLED INSTR. SHOP
113/113A	FILTER

NO.	ROOM OR AREA
114	DEMINERALIZER
115	SPENT FUEL POOL EQUIPMENT, PUMPS & HEAT EXCHANGER
116	DIESEL GENERATOR
116B	DIESEL GENERATOR
116A	SWITCHGEAR
117	PRIMARY SYSTEM DRAIN TANK PUMPS
118	RECEIVER TANK PUMPS & CIRC. PUMP
119	SPENT RESIN HOLD-UP TANK
120	DEGASIFIER, VACUUM PUMPS SURGE TANK
120A	SURGE TANK
121	PASSAGE
121A	PIPEWAY
121B	PRIMARY COOLING SAMPLE HEAT EXCHANGER
122	FAN ROOM
123	COMPONENT COOLING HEAT EXCHANGERS & PUMPS
124	BOILER ROOM
125	CORRIDOR
126	ELECTRIC SHOP
127	MACHINE SHOP
128	ELECTRICAL & MECHANICAL SUPERVISORS
129	TOOL CRIB
130	WAREHOUSE
131	FEEDWATER HEATERS & PUMPS
132	LUBE OIL RESERVOIR & STORAGE TANKS
133	CONDENSER
34	FEEDWATER HEATERS & PUMPS

NO.	ROOM OR AREA
135	DEMINERALIZERS & PUMPS
136	TRAVELING SCREENS & PUMPS
137	ACID, CAUSTIC & HYPO-CHLORITE TANKS
138	INSTRUMENT CALIBRATION STATION
139	GAS STORAGE
140	GAS STORAGE
141	CABLE-WAY
142	STAIRWAY AREA & PRESSURIZER HEATER
142A	PASSAGE
143	COOLING UNITS
143A	ACCESS TUBE ROOM
144	CLEAN WASTE RECEIVER TANKS
145	REACTOR CAVITY
146 & 147	TANK ROOM
148 & 149	AIR PLENUM
150	PIPEWAY AREA
151	LAUNDRY & RAD. CHEM. LAB. DRAIN TANKS
152	CORRIDOR
153	VOLUME CONTROL DEMINERALIZER
154	RADWASTE DEMINERALIZER
201	CHEMICAL LAB.
202	CHEMICAL STORAGE
203	CHEMICAL SUPV.
204	CORRIDOR
205	RADIATION PROTECTION SUPV.
206	DE CON ROOM
207	LAUNDRY

NO.	ROOM OR AREA
208	RAD PRO TECHS
209	CORRIDOR
210	LOCKERS
211	JANITOR
212	WASHROOM
213	CLEAN CLOTHES
214	CLEAN CLOTHES FOLDING
215	SHOWERS
216/216A	CONTROLLED CORRIDOR
217	CONTROLLED LAB.
218	COUNTING ROOM
219	FUTURE FUEL TRANSFER TUBE
220	SPENT FUEL POOL
221	FUTURE FUEL TRANSFER TUBE
222	FUEL TRANSFER TUBE
223	SWITCHGEAR
224	CABLE SPREADING
225/225A	BATTERY ROOM
227	VACUUM DEGASIFIER
228	SAMPLE ROOM
229	OPERATING AREA
230	RADWASTE & VOLUME CONTROL FILTERS
231	RADWASTE & VOLUME CONTROL FILTERS
232	RADWASTE & VOLUME CONTROL FILTERS
233	SAMPLE ROOM
234	PIPEWAY
235	SPENT RESIN REMOVABLE PLUG AREA

NO.	ROOM OR AREA
236	STEAM GENERATOR SUPPORT AREA
237	STEAM GENERATOR SUPPORT AREA
238	CONTAINMENT PURGE EXHAUST UNIT
239	CORRIDOR
240	MEN
241	WASHROOM
242	LOCKERS
243	SHOWER
244	DRYING ROOM
245	WAREHOUSE
246	AIR CONDITIONING & FAN ROOM
247	HEATERS
248	HEATERS
249	ELECTRICAL LOAD CENTER
250	ELECTRICAL PENETRATION ROOM
251	LUBE OIL ROOM ROOF
252	FALLOUT SHELTER
255	TOILET
301	ENTRY
302	VESTIBULE & MONITOR
304	LOBBY
305	CORRIDOR
306	PLANT SUPERINTENDENT
307	ASSIST. PLANT SUPERINTENDENT
308	TECHNICAL SUPERVISOR
309	I&C PNEUMATICS LAB
312	JANITOR
313	MEN

NO.	ROOM OR AREA
314	WOMEN
315	WOMEN LOUNGE
316	RECEPTIONIST, STENOGRAPHER & FILES
317	CONFERENCE ROOM
318	VENTILATION ROOM
319	INSTRUMENT SHOP
320	VIEWING GALLERY
321	SHIFT SUPERVISOR
322	TOILET
323	SHIFT SUPERVISORS
324	FUTURE CONTROL ROOM
325	CONTROL ROOM
326	PIPEWAY
326 A	PIPEWAY
327	VENT. AREA
328	CABLEWAY
329	RAILROAD CAR SHELTER
330	BORIC ACID TANK CHEMICAL MIXING TANKS
330A	BORIC ACID PUMPS
331	TILT MECHANISM ADJUST. AREA
332	ELECTRICAL PENETRATION AREA
333	CHEMICAL STORAGE AREA
334	STEAM GENERATOR
335	STEAM GENERATOR
336	CONTROL ROD DRIVE MECHANISM & REACTOR INTERNALS STORAGE POOL
337	TURBINE GENERATOR OPERATING FLOOR
338	CONTAINMENT PURGE AIR FAN ROOM
339	RESIN ADDITION TANK

NO.	ROOM OR AREA
340	CORRIDOR
341	ESCAPE LOCK ENTRY
342	PASSAGE
343	AIR CONDITIONING ROOM
401	CONFERENCE LUNCH ROOM
403	TELEPHONE EQUIPMENT
404	LIBRARY & FILES
405	KITCHEN
406	CHAIR STORAGE
407	STORAGE
408	AIR CONDITIONING ROOM
410	CORRIDOR
411	MEN
412	JANITOR
413	WASHROOM
414	WOMEN
415	WOMEN'S LOUNGE
416	ENGINEERING
418	NEW FUEL ELEMENT STORAGE
419	FUEL HANDLING AREA
420	FUEL HANDLING AREA
421	CASK WASHDOWN PIT
422	COMPONENT COOLING WATER SURGE TANK
423	STEAM GENERATOR
423A	STEAM GENERATOR
424	PRESSURIZER
500	MACHINE ROOM
501	FAN PLATFORM

TI
APERTURE
CARD

(REF C-43 REV.10)

CONSUMERS POWER COMPANY
PALISADES PLANT
FSAR UPDATE

ACCESS CONTROL & RADIATION ZONING
(PLANT IN OPERATION)

8711090308-112

FIGURE NO 11-8

DATE 10-30-87 REVISION NO 3

GENERAL NOTES

1. FOR ACCESS CONTROL AND RADIATION ZONING OF THE COMPLETE PLANT SEE DRAWINGS 5935-C-42 AND 5935-C-43

ROOM NAMES AND NUMBERS

ROOM NO.	FLOOR ELEVATION	ROOM OR AREA
700	590'-0"	CORRIDOR
701	590'-0"	RADWASTE EVAPORATOR
702	590'-0"	RADWASTE EVAPORATOR
703	590'-0"	RADWASTE FILTERS
704	590'-0"	MISC. & CLEAN WASTE PUMPS
705	585'-0"	MISC. WASTE TRANSFER PUMPS
706	590'-0"	MISC. WASTE HOLD-UP TANKS
707	590'-0"	BOILER ROOM
708	600'-0"	RADWASTE FILTER REMOVAL AREA
709	590'-0"	PRIMARY SYSTEM MAKE-UP WATER AND UTILITY WATER TRANSFER PUMPS
710	590'-0"	CONTROL PANELS AREA
711	602'-0"	PIPEWAY
712	602'-0"	MISC. WASTE & CLEAN WASTE TANK
713	602'-0"	CLEAN WASTE HOLD-UP TANK
714	603'-0"	VALVES
715	611'-0"	TRANSFER TUBE SPACE
716	611'-0"	CONTROLLED LAB NO. 2
717	611'-0"	PIPE WAY
718	611'-0"	PIPE WAY
719	611'-0"	MISCELLANEOUS STORAGE ROOM
720	611'-0"	CORRIDOR
721	611'-0"	STORAGE - ANTI-FOAM AND CAUSTIC TANKS
722	611'-0"	EVAP. CONCENTRATE TANKS
723	611'-0"	EVAP. CONCENTRATE TRANSFER PUMPS
724	611'-0"	SAMPLE ROOM
725	607'-3"	ELECTRICAL EQUIPMENT ROOM
726	611'-0"	VALVES
727		
728		
729		
730	625'-0"	
731	625'-0"	BORIC ACID, CEMENT AND RESIN STORAGE
732	625'-0"	ELECTRICAL AND DUCT SPACE
733	625'-0"	SPENT RESIN STORAGE TANK
734		
735	649'-0"	FUEL HANDLING AREA
736	649'-0"	DEMINERALIZERS
737	649'-0"	HEATING & VENTILATING EQUIPMENT
741	590'-0"	WASTE GAS DECAY TANK
730A	625'-0"	SPENT RESIN TRANSFER PUMP
730B	625'-0"	PROCESS MODULE
730C	625'-0"	OIL INTERCEPTER
730D	625'-0	RADWASTE FILTER (FUTURE)
730E	656'-0"	ELECTRICAL CONTROL PANEL AREA

NOTE:

1. DOORS SHOWN ARE THOSE THAT ARE REQUIRED BY ACCESS CONTROL ONLY.
2. ALL LOCKED DOORS OR LOCKED GATES FOR ZONE V AREAS SHALL BEAR THE RADIATION CAUTION SYMBOL AND THE WORDS "CAUTION HIGH RADIATION AREA".
3. ALL DOORS OR GATES FOR ZONE III AND ZONE IV AREAS SHALL BEAR THE RADIATION CAUTION SYMBOL AND THE WORDS "CAUTION RADIATION AREA".
4. L-DOOR = LOUVER DOOR
   GATE = WIRE MESH GATE (CLOSING THE FULL OPENING.

RADIATION ZONES

ZONE	DESIGN DOSE RATE mrem/hr.	DESCRIPTION
I	0.5	UNCONTROLLED UNLIMITED ACCESS
II	1.5	CONTROLLED UNLIMITED ACCESS 40 HRS/WEEK
III	15.0	CONTROLLED LIMITED ACCESS FOR ROUTINE TASKS, 6 HRS/WK
IV	100.0	CONTROLLED LIMITED ACCESS FOR SHORT PERIODS 1 HR/WK
V	100.0	CONTROLLED OCCUPANCY FOR VERY SHORT PERIODS. OCCUPANCY DURING EMERGENCIES; NORMALLY INACCESSIBLE

TI APERTURE CARD

Also Available On Aperture Card

(REF C-334 REV.6)

CONSUMERS POWER COMPANY
PALISADES PLANT
FSAR UPDATE

ACCESS CONTROL & RADIATION ZONING
(PLANT IN OPERATION)

FIGURE NO 11-9

REVISION NO 0

8407050294 -25