CLINCH RIVER

BREEDER REACTOR PROJECT

50-537

PRELIMINARY SAFETY ANALYSIS REPORT

VOLUME 6

PROJECT MANAGEMENT CORPORATION

6428

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CLINCH RIVER

BREEDER REACTOR PROJECT

PRELIMINARY SAFETY ANALYSIS REPORT

CHAPTER 6

ENGINEERED SAFETY FEATURES

PROJECT MANAGEMENT CORPORATION

CHAPTER 6.0 - ENGINEERED SAFETY FEATURES

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CHAPTER 6.0 ENGINEERED SAFETY FEATURES

6.1 GENERAL

The Engineered Safety Features are the provisions in the plant which are designed to prevent the occurrence or to mitigate the effects of serious accidents. For this plant, the Engineered Safety Features are listed in Table 6.1-1. Descriptions of the Containment Systems and the Habitability Systems are provided in Sections 6.2 and 6.3 of this chapter. Descriptions of the Cell Liner System and the Catch Pan/Fire Suppression Deck System are provided in Sections 6.4 and 6.5. Other Engineered Safety Features are described in their particular sections as identified in Table 6.1-1.

All Engineered Safety Features shall apply, as a minimum, all pertinent ASME Code requirements for materials properties. Any application of cold worked stainless steel shall require the material yield stress to be less than 90,000 psi. The fracture toughness of ferritic material shall meet code requirements.

For Engineered Safety Features requiring thermal insulation, the insulation requirements are discussed in the PSAR sections identified in Table 6.1-1. Controls placed on concentrations of leachable impurities in non-metallic thermal insulation used on components of Engineered Safety Features shall be in accordance with requirements of Regulatory Guide 1.36 "Non-metallic Thermal Insulation for Austenitic Stainless Steel".

Protective coatings used on Engineered Safety Features must be qualified for the particular application to assure that they do not degrade the operation of the ESF by delaminating, flaking or peeling. The requirements of Regulatory Guide 1.54 apply as noted in PSAR Section 1.1, Table 1.

Similarly, controls shall be placed on ESF component and system cleaning to assure protection against damage or deterioration by contamination. Regulatory Guide 1.37 shall apply as noted in PSAR Section 1.1, Table 1.

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TABLE 6.1-1

LIST OF ENGINEERED SAFETY FEATURES IN CRBRP

Engineered Safety Features	PSAR Section
Reactor Confinement/Containment	6.2
Reactor Containment Isolation System	6.2.4 & 7.3.1
RCB Annulus Filtration System	6.2.5 & 7.3.3
Reactor Service Building Filtration System	6.2.6 & 7.3.3
Steam Generator Building Aerosol Release Mitigation System	6.2.7, 7.3.4, 9.13.2, 9.6
Habitability Systems	6.3 & 7.3.3
Reactor Guard Vessel	5.2
Guard Vessels of PHTS Major Components	5.3
Residual Heat Removal System	5.6, 7.4.1 & 7.6.3
Cell Liner System	6.4, 3.8-B & 3A.8
Catch Pan & Fire Suppression Deck System	6.5, 3.8-C, 3A.9, 9.13.2
Ex-Vessel Storage Tank Guard Vessel	9.1.3
Ex-Vessel Storage Tank Anti-Siphon Devices	9.1.3

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6.2 CONTAINMENT SYSTEMS

6.2.1 CONFINEMENT/CONTAINMENT FUNCTIONAL DESIGN

6.2.1.1 Design Bases

The confinement/containment is designed to assure that an acceptable upper limit of leakage of radioactive material is not exceeded under design basis accident conditions. For purposes of integrity, the containment may be considered as the Containment Vessel and Containment Isolation System. Containment integrity is maintained by means of a steel containment shell and dome with a flat steel lined concrete bottom, a concrete confinement building; an air annulus between the vessel and the confinement and the foundation mat (which constitutes the bottom part of the containment).

The confinement/containment is specifically designed to meet the intent of 10CFR100 and the applicable CRBRP General Design Criteria listed in Section 3.8.2 and described in Section 3.1. This Subsection 6.2.1, Chapter 3 and other portions of Chapter 6 present information showing conformance of the design of the confinement/containment and related systems to these criteria.

The functional design and evaluation of the confinement/containment is based upon the design basis accident identified in Table 6.2-1 and the Site Suitability Source Term defined in Appendix 15A.

The design capability of the confinement/containment as described in the following section, 6.2.1.2, will provide a large margin of safety over the most severe accident identified in Table 6.2-1.

The functional design of the containment is based upon the accident input source term and assumptions as delineated in Table 6.2-1 with the additional assumption of a single failure during the accident in active Engineered Safety Features. In this case, the relevant active Engineered Safety Feature comprises only the Containment Isolation System. In the CRBRP, no design basis requirement for a post-accident containment heat removal system or cleanup system has been identified. However, a RCB cleanup system is provided for accidents beyond the design basis (see Reference 10b of Section 1.6).

6.2.1.2 System Design

The containment vessel is a free-standing, welded steel structure with a ellipsoidal spherical dome, a flat circular base and a vertical cylinder of which the lower portion is embedded in concrete up to the operating floor of the Containment Building.

The operating floor is at the 816-ft. elevation. The approximately 169 foot long, 186 ft. diameter cylindrical portion rises 83 feet above this level. The top of the ellipsoidal spherical dome is 158.3 feet above the operating floor. The configuration of the Containment Building and additional description are shown in the RCB General Arrangement's in Section 1.2 and Section 3.8.2.1.

The internal structure consists of concrete cells which contain the reactor vessel, Primary Heat Transport System (PHTS) and miscellaneous support systems. The containment inner cell structures are described in Section 3A.1 and are below the operating floor.

The free building volume above the operating floor is approximately 3,600,000 cu. ft. Free building volume below the operating floor, including cell voids, is approximately 1,400,000 cu. ft.

The design internal pressure for the containment is 10 psig, and the associated maximum allowable leakage rate is 0.1 volume-percent per day. Containment leak rate testing will be performed in accordance with 10CFR50, Appendix J. The design methods to assure integrity of the containment from the accident conditions are described in Section 3.8.2. A negative pressure is maintained in the confinement/containment annulus space and the confinement/containment penetrations are being designed to achieve a bypass leakage value of no more than 0.0006 volume-percent per day. Bypass leakage is defined as the leakage to the atmosphere from the reactor containment building (RCB) that is not subject to filtration and recirculation by either the Annulus Filtration System or the RSB HVAC system. Ninety-nine percent of the leakage from the RCB is filtered and recirculated by the Annulus Filtration System. Forty percent of the remaining one percent leakage that is not filtered and recirculated by the Annulus Filtration System, is directed to the Reactor Service Building where it is filtered by the RSB HVAC system. Thus, the bypass leakage is equivalent to 60% of the unfiltered leakage from the reactor containment building, or 0.006 percent of the maximum allowable RCB leakage rate of 0.1 volume-percent per day. Table 6.2-6 identifies each bypass leakage path and the total leakage for all paths in cubic feet per day.

The seal at each containment airlock consists of compression gaskets with a test connection provided to allow pressurizing of the air space between the seals for leak testing. The elastomer portion of each seal will provide a continuous barrier between the door and the door frame completely around the perimeter of the door. The testing system for the personnel and equipment airlock doors will detect failure or degradation of any seal by leakage from the pressurized air space between it and its companion seal.

The equipment/personnel airlock atmosphere is manually vented to the Annulus Filtration System, when the RSB railroad door is open. Administrative controls are provided to ensure that the airlock doors remain closed when the RSB railroad door is open. In addition, the inner and outer airlock doors are mechanically interlocked to prevent simultaneous opening of the inner and outer doors.

The confinement/containment annulus space above the operating floor is partitioned to provide a spiral flow path around the RCB, for a flowrate of approximately 400,000 SCFM of air. However, during normal operations, the total annulus air flow is less than 3,000 SCFM. This is the air flow which must be exhausted by the annulus pressure maintenance system to maintain a minimum of 1/4 in. W.G. negative pressure in the annulus with respect to the atmosphere. At this flow rate, the pressure variation in the annulus duct formed by the partitions is negligible, and therefore a uniform pressure is maintained throughout the annulus. The annulus pressure maintenance system will be tested prior to plant startup to verify that the above conditions are maintained.

Preliminary functional requirements for the RCB vacuum relief system have been developed and will ensure that the RCB External Design Pressure, 0.5 psig, will not be exceeded. The containment vessel shall be provided with two independent vacuum relief systems, each designed for the full relieving capacity. Each system shall include two series-connected self-actuated valves, one valve located on either side of the RCB shell. The valves shall be set to open on a negative differential pressure allowing the outside atmosphere to enter the containment vessel. The valves must close before the pressure in the vessel equalizes with the outside atmosphere in order to preclude the possibility of leakage out of containment.

The preliminary functional requirements for this system include:

1. The vacuum relief system is classified as Seismic Category I and meet the applicable requirements of Safety Class 2, as described in Section 3.2 of the CRBRP PSAR.

2. Installation of the vacuum relief system shall not result in any change to the RCB containment vessel design leak rate, i.e., 0.1% Vol Day @ 10 psig.

3. The design temperature of the vacuum relief system shall be 250°F.

4. The following flow/pressure conditions shall be met:

    a. Open Set Point Pressure - The open set point pressure for each valve is 3.5 inches water gauge negative differential pressure.

    b. Capacity - The minimum capacity of each valve set (two valves connected in series) is 200 SCFM at a maximum pressure differential across the valve set of 10 inches of water gauge differential pressure.

    c. Close Set Point Pressure - The valves close prior to reaching 1.75 inches of water gauge negative differential pressure.

5. The valves are designed to remain functional for integrated doses up to and including 1.1×10^7 rads gamma radiation. There are no accident transients in the CRBRP design basis that mandate a fast acting (full flow within seconds) vacuum relief system.

6.2.1.3 Design Evaluation

A spectrum of postulated in-containment sodium fires has been analyzed. Table 6.2-1 summarizes the results of the analysis for the most limiting fire investigated. This parametric analysis of postulated in-containment sodium fires has shown that the most limiting accident with respect to the containment building temperature and pressure retaining capability is the postulated failure of the primary sodium storage tank during maintenance, assuming the tank is full of sodium and the tank cell de-inerted and open to the upper containment volume.

The primary sodium in-containment storage tank is located below the containment operating floor (Cell 102A) in a normally inerted cell. The floor area beneath the tank is 850 ft^2. The cell walls are concrete, nominally 6 feet thick. The interior surfaces of the cell are protected with steel liners.

In the event that major maintenance requires draining of one of the primary loops, the tank will be used to store the sodium coolant. The maximum volume of sodium stored in the tank will be 35,000 gallons and the sodium temperature will be maintained at approximately 400°F. The cell atmosphere will remain inerted.

In order to identify a scenario that could present a challenge to containment integrity, a hypothetical set of conditions must be postulated to exist where the primary sodium storage tank contains its maximum volume of 35,000 gallons of sodium with the tank cell de-inerted and interfacing with the atmosphere of the RCB. The occurrence of this set of conditions is highly unlikely for the following reasons:

(1) Maintenance activities requiring draining of one or more PHTS loops will occur very few times over the life of the plant. Thus, Cell 102A will contain 35,000 gallons very few times over the life of the plant.

(2) If 35,000 gallons of sodium is present in Cell 102A, it will be at low temperature and low (atmospheric) pressure, and it will be pumped in and out of the cell at a relatively slow rate. These factors minimize the likelihood of any significant transient loads on the sodium containing boundary. The likelihood of any leak is minimized.

(3) De-inerting Cell 102A when it contains more than about 4000 gallons of sodium will be prohibited by administrative procedure.

(4) Opening any door or hatch in Cell 102A when it contains more than about 4000 gallons of sodium will be prohibited by administrative procedure. Violation of this administrative procedure is not likely because of the difficulty of opening the doors and hatches of the cell. In addition, there are only three scheduled entries into Cell 102A during the life of the plant. These entries are made to perform in-service inspection of the reactor overflow vessel and the primary sodium storage vessel. Thus, operators will not be accustomed to opening the cell; it is unlikely they would do so inadvertantly. The "door" to be used for planned entries is a massive concrete structure that must be moved with a fork-lift truck or equivalent. This door opens to Cell 105 which is below the operating floor. Other cell openings are closed even more permanently requiring extensive effort and use of heavy equipment (e.g., cranes) to open. Further, the number of doors and hatches will be minimized to those necessary for anticipated maintenance and repair activities. All of the above factors assure that it is highly unlikely that the Cell 102A would be in communication with the RCB at all. In the analysis, the door to be used for planned maintenance was assumed to be directly open to containment. Though this assumption was unrealistic, it provided an

upper bound on the amount of direct communication of Cell 102A with the upper containment and conservatively enveloped conditions that would occur if there was a large sodium inventory in Cell 102A. Any more direct communication between Cell 102A and containmnet is incredible.

(5) A non-mechanistic instantaneous failure of the primary sodium storage tank must be hypothesized whereby the total 35,000 gallons of sodium are spilled onto the floor of the tank cell with immediate commencement of sodium pool burning.

For purposes of evaluation, it is assumed that the hypothetical accident occurs near the end of plant life thereby maximizing the primary sodium coolant radiological activity. The radioisotope concentrations in the sodium coolant under these conditions are summarized in Table 15.6.1.4-4. The models and assumptions used in computing the coolant activity levels are discussed in Section 12.1.3. In addition the evaluation assumes that the tank cell environment interfaces directly with RCB environment via a hypothetical passageway equivalent in cross-sectional area to a tank cell door (21 $Ft.^2$).

The rate of sodium combustion with resultant temperature and pressure histories in the containment were computed using the GESOFIRE (Reference 1) computer code. The time behavior of the aerosol generated as a result of sodium combustion was computed with HAA-3 computer code (References 2 & 3). Descriptions of these codes are provided in Appendix A.

Sodium fire-burning rate with resulting containment pressures, temperatures and aerosol concentration are shown as a function of time in Figures 6.2-1 through 6.2-5. Table 6.2-1 summarizes the important design values of the containment and the significant results of the analyses. Table 6.2-3 provides an itemized listing of the radioactive constituents of the aerosol resulting from sodium burning.

Heat input to the containment results primarily from mixing of the hot tank cell atmosphere with upper containment atmosphere via the hypothesized 21-FT^2 passageway. The predominant mechanism for heat rejection from the containment atmosphere is natural convection heat transfer to the containment vessel steel shell. Heat then conducts through the steel shell and is transferred through the confinement annulus air space by convection and radiation. The mode of heat transfer from the outer surface of the concrete confinement to the atmosphere is by convection. The parameters pertinent to heat rejection from the containment atmosphere are itemized below:

Initial Containment Atmosphere Temp.	70°F
Initial Containment Steel Shell Temp.	70°F
Containment Steel Shell	
Thickness	1.4 inch (nominal)
Density	490 lb/ft^3
Specific Heat	0.12 BTU/lb-°F
Conductivity	25 BTU/hr-ft-°F
Containment/Confinement Annulus Air Space	
Thickness	5 Ft.
Density	0.071 lb/ft^3
Specific Heat	0.17 Btu/lb-°F
Effective Conductivity (Includes Convection)	1.1 Btu/hr-ft-°F
Concrete Confinement Building	
Thickness	4 Ft. cylinder; 3 Ft. dome
Density	147 lb/ft^3
Specific Heat	0.2 Btu/lb-°F
Conductivity	0.9 Btu/hr-ft-°F
Sink (Outside Atmosphere) Temp.	90°F

During heat transfer analysis, only the upper containment steel shell surface area is considered available for convective heat transfer. Heat transfer to the containment operating floor and structures within the upper containment is conservatively neglected. Heat transfer to the tank cell structures is calculated using GESOFIRE. The heat sinks available for heat rejection from the upper containment are Containment/Confinement Annulus Airspace, the Confinement Structure, and the outside atmosphere (assumed to be 90°F).

The containment gas pressure and temperature, the containment wall temperature, and the containment aerosol concentration, as a function of time, are based on the formation of 100% sodium-monoxide (Na_2O) during combustion, which results in a conservative estimate of heat generation from sodium burning.

Prior to containment isolation, containment is vented at its normal rate of 14000 cfm. The containment is isolated on activation signals from two sets of redundant radiation detectors. For the radiological analysis results reported in this section, the longer activation delay time of the two was used for additional conservatism. The Containment Isolation System is discussed in detail in Section 6.2.4.

Leakage from the containment after containment isolation is based on the relationship that leakage is proportional to $\sqrt{\Delta P}$ where ΔP is defined as the pressure differential across the containment vessel shell. Leakage, λ_ℓ, at a differential pressure of ΔP is computed from the following expression:

$$\lambda_\ell = (\sqrt{\Delta P}) \left(\sqrt{\frac{0.1\%\ \mathrm{VOL/DAY}}{10\ \mathrm{PSIG}}}\right)$$

This containment leakage is directed through the Containment/Confinement Annulus Filtration System except for the Confinement bypass leakage as described in Section 6.2.1.2.

A total of 3.4 kg of aerosol containing 2.5 Kg Na is released in the atmosphere over the 70 hours of containment overpressure. Release during specific time intervals is as follows:

Time (hr)	Mass Na Released (gm)
0-2	2485
2-8	2.5
8-24	7.9
24-70	15.1
>70	0

The potential off-site doses due to this release are summarized in Table 6.2-2.

Conservatively assuming no credit for the sodium fire protection system, sodium combustion is essentially complete approximately 550 hours after the postulated tank failure (See Figure 6.2-1). At this time, more than 99% of the total oxygen in the containment building available for combustion (62,400 lbs) has been reacted. The total sodium consumed during the 550 hours of burning is 179,700 lbs. Beyond 550 hours the rate of heat addition would be extremely slow and would not have a significant impact since the temperature, pressure, and aerosol concentration have already peaked and are decreasing at times greater than 100 hours.

The containment pressure history is shown in Figure 6.2-2. Three effects govern the shape of the curve. The first, which tends to increase pressure, is the increase in the containment gas temperature resulting from heat transfer from the burning sodium pool. The second, which tends to reduce pressure, is the decrease in the mass of the containment gas resulting from oxygen consumption. The third effect, the heat losses through the containment wall and cell structures also tends to reduce pressure. The peak containment pressure is 0.8 psig and occurs 20 hours after the postulated spill. A summary of the heat sinks for the analysis is given in Table 6.2-4.

The containment gas and wall temperatures, as a function of time, are shown in Figures 6.2-3 and 6.2-4, respectively. The predominant mechanism for heat transfer to the containment wall is convection with the containment gas atmosphere. As such, the behavior of these two temperatures with time is very similar as the figures indicate. The peak containment gas temperature, 138^{o}F, occurs 60 hours after the spill. The peak containment wall temperature, 122^{o}F, occurs 70 hours after the spill and is significantly below the design temperature of 250^{o}F, precluding the need for an emergency cooling system for the containment. Both containment gas and wall temperatures then decrease gradually to less than 100^{o}F, 470 hours after the postulated spill.

The mechanisms for the depletion of the suspended aerosol concentration with time include: 1) settling due to gravity, 2) wall plating, and 3) agglomeration due to Brownian Motion and/or agglomeration due to gravity.

The peak aerosol concentration, 11×10^{-6} gm/cc, occurs shortly after the start of the fire. The aerosol concentration decreases to less than 1×10^{-6} gm/cc 550 hours after the postulated spill. The peak radioactive content of the aerosol is 1.18×10^{-9} Ci/cc.

To decrease the temperature gradient between the steel containment vessel above elevation 816, which is exposed to the containment atmosphere, and that portion of the vessel protected by the concrete below that elevation, insulation is applied ot the inside of the containment vessel. The thickness of the insulation is 6 inches and it extends from elevation 816 to elevation 825. This insulation is required to maintain the allowable stresses in the steel containment vessel to acceptable levels for the RCB design basis accident.

The storage tank cell gas pressure and temperature and the cell wall and floor liner temperatures as functions of time following the event are shown in Figures 6.2-6 through 6.2-9. The storage tank cell pressure transient is essentially identical to that of the upper containment. The cell gas temperature peaks at about 815^{o}F, 30 hours after the start of the postulated fire. It then decreases gradually and is down to about 200^{o}F at 500 hours after the start of the fire.

The temperatures of the cell steel liner wall portion and the bottom portion (exposed to the sodium spill pool) are shown in Figures 6.2-8 and 6.2-9, respectively. The liner wall temperature peaks at 620^{o}F about 30 hours and then gradually decreases to about 200^{o}F at 470 hours following the start of the event. The temperature of the liner floor portion peaks at about 1000^{o}F at 30 hours and falls off to 220^{o}F at about 500 hours after the start of the fire.

From the pressure transients shown in Figures 6.2-2 and 6.2-6, it can be seen that the differential pressure between the cell and the upper containment is minimal. Any resultant pressure loading on the cell structure is therefore not expected to be of any significance. The temperature effects associated with the temperature transients as shown in Figures 6.2-8 through 6.2-9 will be the major component of accident loading during the postulated event. The cell concrete structure design requirement is such that its structural integrity will be maintained during and following the postulated accident.

The potential 2-hour, site boundary and 30-day low population zone (LPZ) whole body and organ doses resulting from this postulated accident are itemized in Table 6.2-2. A large margin (greater than a factor of 10^{2} exists between each of the potential doses and the applicable guidelines. The peak temperature and pressure resulting from this event is within the containment design temperature and pressure.

6.2.1.4 Testing and Inspection

Testing of the containment and appurtenances shall be conducted in accordance with the requirements of NE-6000 of ASME-III and Division 2 of ASME-III, as applicable, and of Appendix J to 10CFR50 with the qualifications stated in the following paragraph.

The word "pressure" should be omitted from all references to "reactor coolant pressure boundary." The 8 foot personnel and equipment airlock which will be opened frequently during normal operation will be tested automatically after each opening by utilization of an on line testing system. The on-line testing system meets the intent of the Type B tests specified in 10CFR50, Appendix J, Sections III.B.1.C and III.D.2.

The tests to be performed will be as follows:

1. <u>Construction Stage and Preoperational Tests</u>

   A. <u>Testing of Welds in Bottom Liner Plates</u>

   Prior to pouring of internal concrete, the channel-to-liner plates shall be examined using the liquid penetrant or magnetic particle examination method in accordance with the Code followed by 100 percent vacuum-box testing for leaktightness.

B. Leak Channel Test

Also prior to pouring of internal concrete, the channel-to-liner plate and related pressure barrier shall be tested for leaktightness by pressurizing the channels to containment design pressure (10 psig). If any indicated loss of channel test pressure occurs within 2 hours, as evidenced by a test gauge, the channel-to-liner welds will be soap bubble tested.

C. Preliminary Leak Test (PLT)

A PLT may be performed, prior to acceptance pneumatic tests as a means of locating major leaks. If used, a test pressure determined in accordance with NE-6112 of ASME-III shall be used.

All nondestructive examination of strength welds shall be completed prior to the PLT. The PLT shall be conducted after completion of construction of the vessel, including piping and electrical penetrations air locks, and equipment hatch.

D. Pneumatic Pressure (Strength) Test

A pneumatic pressure test at a test pressure of 11.5 psig shall be performed. This test shall be conducted in accordance with the requirements of NE-6300 of ASME-III.

E. Leakage Rate Tests

Subsequent to the pneumatic pressure test, leakage rate tests for measuring the containment overall integrated leakage rate shall be performed. These tests shall be conducted in accordance with the requirements of Appendix J to 10CFR50. The test pressure for the tests required in III.A.4(2) of Appendix J shall be 10 psig. The maximum allowable leakage rate for the containment is 0.1% (weight)* per 24 hours.

2. In-Service Surveillance

Periodic leakage rate tests may be performed at any time during the containment vessel 30 year design life. The periodic containment integrated leakage rate requirements are to be developed to meet the requirements of 10CFR50, Appendix J for Type A tests. Test requirements for penetrations using resilient seals or expansion bellows will meet the basic intent of 10CFR50, Appendix J, Type B. Reference section 16.4.4.

*It should be noted that a weight percent leak rate is the same as a volume percent leak rate.

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6.2.1.5 Instrumentation Requirements

Instrumentation required for containment consists of three general areas:

1. Containment Isolation

   Radiation monitors are used for automatic containment isolation. Detailed descriptions of instrumentation for this system are presented in Sections 7.3 and in 6.2.4.

2. Seismic Instrumentation

   A detailed description of the Seismic Instrumentation Program is provided in Section 3.7.4.

3. Containment Pressure, Temperature and Hydrogen

   Instrumentation channels for monitoring containment air pressure, temperature and hydrogen concentration will be provided for Control Room indication and operator observation. Monitors for containment vessel temperature will also be provided to the Control Room.

4. Airlock Door Seal Area Leak Rate Monitoring Instrumentation

   A desription of the Airlock Door Seal Area Leak Rate Monitoring Instrumentation is provided in Section 7.5.15.

6.2.1.6 Materials

Material selection shall be made in the design such that the radiolytic or pyrolytic decomposition products, if any, will not interfere with safe operation of the plant. All materials will meet the applicable requirements of the ASME Code, RDT Standards, ASTM Standards, ACI Code, and other nationally recognized materials codes and standards. Containment building materials are discussed in Section 3.8.

6.2.2 Containment Heat Removal

There is no requirement for a safeguard heat removal system for the containment following a containment design basis accident under post-accident conditions.

6.2.3 Containment Air Purification and Cleanup System

There is no design basis requirement for a safety related containment atmosphere cleanup system. In the event that a design basis accident causes a release of radioactivity to the containment atmosphere, the containment isolation will be effected and radioactive materials will be contained.

An air purification and cleanup system for mitigation of design basis post-accident radiological consequences is not required. The confinement/containment design leakage rate provides adequate safety margin for all the design basis accidents identified and described in Chapter 15.

Building cleanup procedures following a sodium fire accident will require personnel to be equipped with protective clothing, and auxiliary breathing air to avoid inhalation of particulates that may be dislodged from walls and equipment during the cleanup operation. An air purification and cleanup system is not required to enable safe entry into containment for post-accident cleanup.

6.2.4 Containment Isolation Systems

The containment isolation systems provide the means to close valves in lines penetrating containment to assure that the containment provides the required barrier to release of radioactive gas or particulate matter. Closure of valves may be manually or automatically initiated depending on the type of line penetrating containment. The design bases, key features of the design, and references to the appropriate chapter for complete description of the system and valving are included in this chapter. The automatic containment isolation is described in Section 7.3.

6.2.4.1 Design Bases

The provisions for containment isolation shall meet the requirements of the CRBRP Design Criteria, specifically criteria 45, 46, 47 and 48 (Refer to Chapter 3.). The penetrations requiring isolation are specified in Table 6.2-5 and Figure 6.2-10. Since different categories of lines penetrating containment are identified, a single design basis is not appropriate. Therefore, the bases for the categories will be specified.

For isolation of lines which are directly connected to the containment atmosphere, the containment isolation shall meet the requirements of CRBRP Design Criteria 45 and 47. Specifically, each line penetrating containment which is connected directly to the containment atmosphere shall include two automatic isolation valves, one inside and one outside the containment. The valves and associated piping shall be designed with the capability to periodically test the operability of the isolation valves and to determine if valve leakage is within acceptable limits. The valves and associated actuators shall close on loss of air or electrical power. The valves included in this category are delineated in Table 6.2-5 and identified in Table 6.2-5A. Response characteristics of these valves are dictated by: the primary sodium storage tank fire during maintenance; and the provision of the design for the Site Suitability Source Term. Since these are either extremely unlikely events or Site Suitability Source evaluations, the guidelines of 10CFR100 apply. Meeting these guidelines is a function of the air transport time, detection time, valve closure time, and the point at which the incident is sensed. For the present design concept with a 10 second air transport time, and 3 second detection time, closure of the inlet and exhaust valves in 4 seconds is necessary to meet Site Suitability Source Term. However, the critical feature is the interrelationship between the three variables (transport time, closure time, and sensor location). Therefore, the closure time is not an isolated design requirement but rather is a part of integrated system performance. The 10 second air transport time is a design requirement and represents the air transport time between the sensor and the in-board isolation valve of the H/V lines at normal operation containment pressure. The design of the activation sensors of the containment isolation is discussed in detail in Section 7.3.1.

The lines penetrating containment which are connected to the reactor coolant boundary, primary cover gas space, or inerted cell atmospheres are the argon supply lines and the nitrogen supply and exhaust to CAPS. The design includes two automatic valves for each penetration. Automatic action is specified if response is necessary within 10 minutes. Otherwise, remote manual initiation is provided. The provisions for containment isolation for these lines meet the requirements of CRBRP GDC 45 and 46 as detailed below.

The design may utilize several existing valves on branch lines within containment to provide the isolation function.

Argon and nitrogen supply lines shall have two automatically initiated containment isolation valves. One valve is located inside the containment, while the other is located outside of the containment as close as practical to the containment. The valves are back pressure regulated valves which close automatically if the supply side pressure drops below a preset limit. This assures that breaching of the gas system boundary outside of containment results in isolation of the line penetrating containment. Remote manual actuation is also provided. Closure of the valves for loss of supply side pressures assures that failures in the system outside of containment combined with postulated events within the containment cannot result in release of radioactive gases.

Nitrogen exhaust line to CAPS shall have two automatically initiated containment isolation valves. One valve shall be located inside while the other is outside of containment as close as practical to the containment. Remote manual actuation is also provided.

For closed systems penetrating containment, the requirements of GDC 45 and 48 shall be met. Specifically, each line penetrating containment which is part of a closed system other than the IHTS shall have at least one isolation valve (either automatic, locked closed, or capable of remote manual operation). The isolation valve shall be located outside of containment as close to the containment as practical. Provisions for testing the operation of the isolation valves and to determine that valve leakage is within acceptable limits shall be included. The valves shall close on loss of electrical signal or loss of air. The valves included in this category are delineated in Table 6.2-5. A summary of design criteria is presented in Table 6.2-5A.

The IHTS boundary is maintained intact as a containment boundary by providing in-depth protection for the piping and components. The seismic category 1 Steam Generator Building (SGB) provides a barrier capable of withstanding the extremes expected in the environmental conditions. It assures that tornado wind loadings and missiles will not result in damage to the IHTS. The SGB design also incorporates internal structures to provide missile protection for the IHTS (from internally generated missiles), separation between heat removal paths, piping restraints and impingement shields. These design features provide assurance that events initiated externally to the IHTS will not result in loss of the IHTS boundary integrity. The IHTS piping and components are also designed to maintain their integrity following the Safe Shutdown Earthquake (SSE).

The integrity of the IHTS boundary will be monitored through a combination of in-service inspections, sodium to air leak detection, and maintenance of a 10 psi (minimum) IHTS to PHTS Δp in the IHX and leak detection for IHX leaks (IHTS level probes). These mechanisms provide assurance that leakage between the primary and intermediate systems, as well as leakage from the IHTS to the air will be detected early and corrective action initiated before significant amounts of IHTS sodium are leaked.

The in-depth protection and integrity monitoring provisions assure that the containment function of the IHTS will be maintained through all expected environmental conditions and external accidents. However, even if a loss of integrity were to occur it will not result in unacceptable off-site radiological consequences. Section 15.6.1.5 presents the results of an evaluation of the potential radiological consequences of an undetected leak in the IHX. This evaluation shows that the site boundary doses are considerably less than the dose limits given in 10CFR100.

The chilled water lines penetrating containment shall have two remote manual isolation valves: one located inside and one located outside the containment as close as practical to the containment. Since this closed system is not connected to the containment atmosphere nor does it contain radioactive materials, failures in the system outside of containment cannot result in a radiological release.

The penetrations associated with the Primary Sodium Removal and Decontamination System (see Section 9.2) shall be isolated to meet the requirements of CRBRP General Design Criteria 45 and 47.

6.2.4.2 <u>System Design</u>

The design features of the containment isolation valving are summarized on Table 6.2-5. A summary of valving categories and applicable GDC is provided in Table 6.2-5A. For each line penetrating containment, Table 6.2-5 shows: the systems; the PSAR section in which the system is described in detail; the number of valves required; the type of valve required; the normal operational status; the primary means of actuation; the secondary means of actuation; and the required valve closure times. The piping and instrument diagrams are included in the PSAR chapter describing the system. Note that all valves are located as close to the containment as practical. The parameters sensed to initiate closure of the lines directly connected to the containment atmosphere are described in Section 7.3. The closure times for the system and the evaluation of adequacy are discussed in Section 7.3.

All lines penetrating containment up to and including the isolation valve outside of containment are located in areas which provide protection against missiles due to natural phenomena. In general, these lines and the associated isolation valves are not located in areas which contain equipment whose failure would result in the generation of missiles. By locating the equipment in areas which do not include equipment that can generate missiles, the valves and associated actuators are protected. Specific missile protection for containment isolation will be provided if necessary.

The lines penetrating containment do not contain high pressure fluid nor are they routed closely to lines which do contain high pressure fluid (detailed discussions are provided in Section 3.6 of this PSAR). These lines are adequately supported to provide the necessary performance in normal operation and to sustain seismic induced forces. The combination of adequate support and low pressure provides assurance that pipe whip due to postulated line failure will be of low magnitude and will not cause valve, actuator, or control damage.

The protection of valves, actuators, and controls against jet impingement is dependent on the lines in the immediate area and the fluids involved. The designs will provide protection if postulated line failures resulting in deleterious impingement of fluid is a credible event for a particular isolation valve.

All containment isolation valves will be designed to operate in the worst accident environment necessitating closure of the specific valve.

Because the systems operating pressures are low and the closure times required for the containment isolation valves are 4 seconds, the dynamic forces resulting from inadvertent closure under operating conditions will not challenge the integrity of the valves or connecting piping.

6.2.4.3 Design Evaluation

The containment isolation features of the design of lines penetrating containment provides the necessary assurance that the containment system will provide the barrier to release or spread of radioactive gas or particulate matter.

Automatic isolation of the lines connected to the containment atmosphere is provided by the design. The initiation features described in Section 7.3 and the isolation valves for the containment ventilation lines and air lines, respond to the assumed events to prevent releases in excess of the guidelines of 10CFR100 for extremely unlikely events such as sodium fires or assumed increases in pressure and activity levels.

Earlier in the design both temperature and pressure were parameters considered for initiation of the CIS; however, both were eliminated. Calculations show that substantial (for detection purposes) pressure and/or temperature rises are delayed many hours after the initiation of accidents, as shown in Figures 6.2-2 & 6.2-3. Further, the maximum pressure is in the range of 1 psig and the maximum temperature is 138°F. In addition the direct measurement of the parameter of concern, radioactivity, is the most desirable. It should be noted that containment isolation can be initiated by either a high radiation signal from the head access area radiation monitors, or a high radiation signal from the containment exhaust radiation monitors. Different (diverse) detector types are provided in these separate locations. Each set of detectors provides inputs to a separate CIS logic train. (See Section 7.3.1.).

Periodic on-line testing capabilities are included. The valves and associated actuators are located in areas which are protected from tornado generated missiles and which are designed to withstand the seismic forces.

For the lines connected to the reactor coolant boundary, the two valves, either manually or automatically actuated as appropriate, provide the necessary protection. The different classes of the lines are separately discussed below.

The argon and nitrogen supply line valves provide a double barrier which is automatically activated on loss of the ex-containment boundary. The valves and associated actuators are located in protected areas and are testable. Remote and local manual initiations are provided. The nitrogen exhaust line to CAPS has two automatically initiated valves. The valves provide two barriers following closure. The valves and associated actuators are located in protected areas and are testable.

For the remainder of the penetrations, two valves are provided as barriers to release. Manual initiation will be adequate to prevent releases exceeding the guideline values.

For lines of closed systems penetrating containment, one isolation valve located outside of containment as close as practical to containment is provided. A single valve meets Criteria 48 and provides the necessary capability to limit the release of activity. For lines which do not contain radioactive fluids, the closed system provides the first boundary while the isolation valve provides the second boundary to release of activity. Therefore, in all cases, there are two boundaries which effectively limit the release of activity from a postulated event. The valves and associated actuators are located in protected areas and are testable. Manual initiation of isolation is provided.

6.2.4.4 Tests and Inspections

The periodic test capability is described in Section 7.3.

6.2.5 Annulus Filtration System

6.2.5.1 Design Bases

The Annulus Filtration System is designed to ensure than an acceptable upper limit of leakage of radioactive material is not exceeded under the site suitability source term conditions.

The functional design and evaluation of the Annulus Filtration System is based upon the site suitability source term, as identified in Section 15.A. The design capability of the annulus filtration system as described in the following section will provide a large margin of safety over the containment design basis accident identified in Table 6.2-1.

6.2.5.2 System Design

The RCB annulus filter system design shall satisfy the following criteria:

(1) The containment/confinement annulus space shall be maintained under 1/4 inch W. G. negative pressure during normal plant operation and accident conditions.

(2) Capability shall be provided to filter the containment/confinement annulus exhaust during normal operation.

(3) Capability shall be provided to filter the RCB ventilation exhaust air through the annulus filter system during refueling operations, when the RCB/RSB refueling hatch is open.

(4) Capability shall be provided to filter and recirculate the annulus air during accident conditions. For every 1000 CFM filtered exhaust air (required for the maintenance of 1/4 in. W. G. negative pressure) not less than 3500 CFM air shall be recirculated through the filters.

(5) The recirculating duct system shall be designed to accomplish proper mixing in the annulus in accordance with USNRC Standard Review Plan Section 6.5.3.

(6) The Annulus Filtration System shall fully comply with USNRC Regulatory Guide 1.52.

(7) The filter system shall be designed to achieve a minimum of 99% particulate and 95% adsorbent efficiency.

Radiation monitoring equipment associated with the annulus filtration system is described in Section 12.2. By maintaining the annulus at a minimum of 1/4" water gauge negative pressure with respect to the outside atmosphere, the bypass leakage (that fraction of annulus radioactivity which leaks from the confinement building without being filtered) can be maintained at less than 1%.

6.2.5.3 Design Evaluation

The Annulus Filtration System features of the design provide the necessary assurance that the radioactivity released as a result of the site suitability source term will not exceed the guidelines of 10CFR100.

The annulus pressure maintenance fans have been sized at 3000 CFM, which has conservatively been determined to be greater than the total leakage into the annulus from all sources, including the dampers (vents and cap) provided at the top of the Confinement Building and the dampers provided at the 816' - 0" elevation for the Annulus Cooling System (leakage based on a negative 1/4" w.g. pressure).

Analysis will be conducted to substantiate that the annulus space will remain under a 1/4" W.G. negative pressure considering the effects of heat transfer, barometric pressure change, inleakage and wind loads. The results of this analysis will be provided in the FSAR.

Two 100% redundant filter-fan units consisting of a demister, heating coil, prefilter bank, HEPA filter bank, adsorber bank, pressure maintenance and exhaust fan, annulus recirculation fan, with associated ductwork and accessories, are provided for the annulus exhaust, recirculation and filtering. This insures that no single active failure will prevent 100% operation of the Annulus Filtration System. The Annulus Filtration System is described in Section 9.6.2.2.4.

6.2.5.4 Tests and Inspections

The Annulus Filtration System shall be tested per the requirements of Regulatory Guide 1.52. In addition, preoperational testing will be performed to establish baseline in-leakage ratio and to verify the ability of the system to maintain a uniform negative pressure in the annulus. Periodic surveillance will be performed to ensure that the performance of Annulus Filtration System remains satisfactory and that it retains its ability to maintain a uniform negative pressure.

The results of periodic containment penetration leak testing will be used to verify leakage assumptions used for radiological accident analysis.

6.2.6 Reactor Service Building (RSB) Filtration System

6.2.6.1 Design Basis

The RSB filtration system is designed as an Engineered Safety Feature (ESF) to filter the RSB exhaust air in order to mitigate the consequences of the Site Suitability Source Term (SSST) event.

The system is designed to function continuously.

6.2.6.2 System Design

The RSB is maintained at a minimum 1/4" negative water gauge pressure as described in Section 9.6.3.1.1.

The RSB Filtration System is used and designed to maintain the RSB at a minimum of 1/4" negative water gauge pressure and filter the RSB exhaust under all conditions except when the railroad door is open. A network of ducting is utilized in supplying and exhausting air to various floor elevations and/or cells in the RSB. This mode of operation exhausts 18,000 CFM of air through the missile protected exhaust on the Reactor Service Building (RSB).

During accident conditions the RSB Filtration System will automatically shift to an air recirculation mode of operation exhausting that amount of air (~1700 CFM) required to maintain a minimum of 1/4" negative water gauge pressure.

The filter system will be designed as a Safety Class 3 system and will meet the requirements of Regulatory Guide 1.52. The filter system will be designed to achieve a minimum of 99% particulate and 95% adsorbent efficiencies.

6.2.6.3 Design Evaluation

The RSB filter system is designed to filter 18,000 CFM of air of which 1700 CFM of air is exhausted while 16,300 CFM of air is recirculated during accident conditions. The exhausted air is designed to offset building in leakage air while maintaining 1/4" negative water gauge pressure.

Analysis will be conducted to substantiate that the RSB will remain under a 1/4" W.G. negative pressure considering the effects of heat transfer, barometric pressure change, inleakage and wind loads. The results of this analysis will be provided in the FSAR.

Two (2) 100% redundant filter fan units consisting of a demister, heating coil, pre-filter bank, adsorber bank, HEPA filter bank, cleanup filter fan, with associated ductwork and accessories, are provided for the RSB exhaust, recirculation, and filtering. This insures that no single active failure will prevent 100% operation of the RSB filtration system. The RSB filtration system is described in Section 9.6.3.1.1.

The system ducting is designed to exhaust air from all potentially radioactive areas. Capability exists to isolate the supply and exhaust air flow to the areas where an accident has occurred and to maintain these areas at a greater negative pressure than other areas.

This capability is designed to prevent the spread of airborne radioactivity from contaminated to clean areas within the building.

6.2.6.4 Test and Inspection

The RSB filtration system will be tested per the requirements of Regulatory Guide 1.52. Visual inspection will be conducted on installation.

6.2.7 STEAM GENERATOR BUILDING AEROSOL RELEASE MITIGATION SYSTEM FUNCTIONAL DESIGN

6.2.7.1 Design Bases

The Steam Generator Building Aerosol Release Mitigation System is designed to assure that release of a maximum of 630 lbs of sodium aerosols from the Steam Generator Building is not exceeded in the event of a design basis leak in one of the three loops in IHTS piping. This limit is obtained by releasing through a controlled vent area a maximum of 440 lbs of aerosols during the first five minutes of the accident. Between 5 minutes and 5000 seconds, 90 lbs of aerosols may be released through building cracks. Beyond 5000 seconds, an additional 100 lbs of aerosols could be released through building cracks.

A release of aerosols through the controlled vent area stack is required to maintain building overpressures below the 0.7 psig setpoint for opening of the large (360 Ft^2) steam vent louvers.

The functional design and evaluation of the SGB Aerosol Release Mitigation Features are based upon the design basis accident described in Section 15.6.1.5 of the PSAR.

6.2.7.2 System Design

Controlled release of aerosols from the Steam Generator Building (SGB) is accomplished by closure of SGB HVAC outlets and venting through a controlled area vent stack, both actions being initiated from either of a redundant set of safety-related aerosol smoke detectors located in the SGB HVAC exhaust stack. Aerosols are released from the controlled area vent stack for five minutes to assure until peak pressures in the SGB are within acceptable limits, at which time the vent path is closed to the external atmosphere.

The SGB Aerosol Release Mitigation Features consist of redundant sets of safety-related aerosol detectors (see Section 9.13.2) located in each SGB loop HVAC exhaust duct, redundant relief dampers to each loop controlled area vent stack, and redundant closure dampers in each controlled area vent stack.

Each aerosol detector set consists of three detectors provided power by three 1E uninterruptible power sources. These detectors trip when the sodium aerosol concentration in the SGB HVAC exhaust is 10^{-7} gm/cc. When two of the three detectors in either set sense an aerosol concentration of 10^{-7} gm/cc, a signal is provided to activate the I&C logic for the SGB aerosol release mitigation features. Within 10 seconds of receipt of an aerosol detection signal, the SGB building HVAC system will be closed to the outside atmosphere, the relief dampers to the controlled vent area will open, the controlled vent area closure devices will remain in their normally open position, and the remaining nuclear island building (RCB & RSB) HVAC systems will be closed to the outside atmosphere. The controlled vent area closure devices close five minutes after receipt of the trip signal from the aerosol detectors, with a total release to the outside atmosphere of no more than 440 lbs. of aerosol. Beyond five minutes aerosols will continue to be released to the outside atmosphere from the SGB via building penetration and equipment cracks (e.g. construction openings, pipe cable trays, dampers and louvers). To ensure that building cracks will plug with aerosols and limit further releases of aerosols, cracks are constrained to no more than 1/8 inches in width with a total summation of crack area for each IHTS loop not exceeding 10 ft^2. The total amount of aerosols released through building cracks is less than 200 lbs, with no more than 100 lbs. being released for times greater than 5000 seconds after detection of aerosols in the SGB.

6.2.7.3 Design Evaluation

The IHTS design basis sodium spill accident is described in detail in Section 15.6.1.5. Leakage from breaks in the IHTS sodium piping up to and including the Moderate Energy Fluid System (MEFS) break as defined in branch technical position MEB 3-1 were investigated with the maximum amount of aerosols being released to the outside atmosphere for the MEFS break (630 lbs.).

The release of 630 lbs. of aerosols from the SGB results in less than 60 lbs. of aerosols being ingested into the Air Blast Heat Exchanger (ABHXs) based on dispersion factors contained in the Murphy and Campe formula (Ref. 15.6.1.5-1). This amount of aerosols will not preclude the ABHXs from removing spent fuel decay heat with the Ex-Vessel Storage Tank containing its maximum load of spent fuel (1.8 Megawatts).

The Protected Air Cooled Condensers (PACCs) are prevented from starting in the presence of high sodium aerosol concentrations via detection signals from safety-related sodium aerosol detectors located in the air inlet to each PACC unit. In each PACC air inlet are located two sets of three safety-related aerosol detectors with each detector in a given set provided power by a separate 1E Uniterruptable Power Source. Detection of aerosol concentrations $\geq 10^{-7}$ gm/cc precludes a PACC unit from starting for a time period of 5000 seconds. During this time frame, decay heat would be removed via SGHARS without compromising the Protected Water Storage Tank 30 day supply requirement. Beyond 5000 seconds no more than 100 lbs. of sodium aerosols will be released from the SGB. This amount of aerosols ingested into a PACC unit will not preclude the unit from performing its decay heat removal function. At the end of the 5000 second shutdown period, an affected PACC unit will automatically restart.

The RSB and RCB HVAC systems will be isolated from the external atmosphere and placed in a recirculating cooling mode, within 10 seconds of generation of an aerosol trip signal from the aerosol detectors located in a SGB HVAC exhaust duct. During this 10 second time frame, aerosols released from the SGB are ingested into the RSB and RCB air filled cells resulting in a maximum suspended aerosol concentration of 15 milligrams/$meter^3$ (equivalent sodium monoxide) and 0.1 grams/$meter^2$ maximum deposition. Dispersion of aerosols from the SGB to the various RSB & RCB HVAC inlets was determined from the Murphy-Campe relationship (Reference 15.6.1.5-1). These aerosol concentrations are specified for qualification of 1E electrical equipment exposed to non-severe environments.

The diesel generator building is presently being relocated. Preliminary analyses indicate that the diesels can be located in the vicinity of the nuclear island complex such that minimum filtering of the diesel combustion intakes will exclude ingestion of sodium aerosols. Internal building 1E equipment will be protected from external atmosphere sodium aerosols via complete closed-loop internal recirculating cooling HVAC systems.

6.2.7.4 <u>Testing</u>

Detailed test requirements for the Steam Generator Building Aerosol Release Mitigation Features have not been developed at this time. Requirements will be incorporated into the PSAR as they are developed.

6.2.7.5 <u>Instrumentation Requirements</u>

Instrumentation required for SGB Aerosol Release Mitigation Features consist of the following:

1. <u>SGB Controlled Venting</u>

   Two sets of three safety related aerosol detectors provided with separate 1E Uninterruptable Power Sources are located in each SGB loop HVAC exhaust duct. (Total of eighteen (18) detectors).

   Controlled area vent closure devices are powered by 1E motors, each motor provided power by a separate 1E division of power.

2. <u>Isolation of HVAC Systems</u>

   Upon detection of aerosol by 2/3 of the aerosol detectors in either of the two sets of aerosol detectors located in the SGB exhaust ducts, isolation signals are provided to the nuclear island building HVAC systems whereby the HVAC systems are isolated from the external atmosphere and placed in a recirculating cooling mode.

3. <u>Protection of PACCs</u>

   Two sets of three safety-related aerosol detectors provided with separate 1E Uninterrupted Power sources are located in the air inlet of each PACC unit. (Total of eighteen (18) detectors).

References:

*(1) General Electric Specification #23A2843, "The GESOFIRE Code."

*(2) R.S. Hubrer et al. "HAA-3 User Report," AI-AEC-13088, Atomics International, March 30, 1973.

*(3) NUREG/CR-1724, "Proceedings Of The CSNI Specialists Meeting On Nuclear Aerosols In Reactor Safety," October, 1980.

*References annotated with an asterisk support conclusions in this section. Other references are provided as background.

TABLE 6.2-1

CRBRP CONTAINMENT

Design Basis Sodium Pool Fire Accident

I. General Information

1. Internal design pressure, 10 psig
2. External design pressure, 0.5 psig
3. Design temperature, 250^{o}F
4. Free volume, 3.6 x 10^{6} Ft^{3} (above operating floor)
5. Design leak rate of steel containment, 0.1% Vol/Day at 10 psig
6. Design ambient temperature, 70^{o}F

II. Sodium Pool Fire Accident Summary

1. Location of fire: Open, de-inerted Primary Sodium Storage Vessel Cell
2. Mass of Sodium Burned: 179,700 lbs
3. Duration of fire: 550 hours
4. Energy released into containment: 6.8 x 10^{8}BTU
5. Maximum density of aerosol: $11x10^{-6}$ gm/cc
6. Constituents of aerosol: See Table 6.2-4
7. Containment atmosphere maximum temperature: 138^{o}F
8. Containment wall maximum temperature: 122^{o}F
9. Containment maximum pressure: 0.8 psig
10. Time of containment maximum atmosphere temperature: 60 hours
11. Time of containment maximum wall temperature: 70 hours
12. Time of containment maximum pressure: 20 hours

TABLE 6.2-2

REACTOR CONTAINMENT DESIGN BASIS ACCIDENT

POTENTIAL OFF-SITE DOSES

		Dose (Rem)	
Organ	10CFR100	Site Boundary (0.42 mi) 2-Hour	Low Population Zone (2.5 mi) 30-Days
Whole Body**	25	2.14E-2*	3.43E-3
Thyroid	300	8.01E-2	1.28E-2
Bone	150+	2.88E-1	4.61E-2
Lung	75+	1.61E-2	2.57E-3

*2.14E-2 = 2.14×10^{-2}
**Includes both inhalation and external gamma exposure
+Not covered in 10CFR100; used as guideline values

TABLE 6.2-3

CONSTITUENTS OF CONTAINMENT AEROSOL
FOLLOWING FAILURE OF IN-CONTAINMENT
PRIMARY SODIUM STORAGE TANK DURING MAINTENANCE
(Values for Peak Aerosol Concentration)

Isotope	pCi/cc	gm/cc	Isotope	pCi/cc	gm/cc
Na 24	3.58E+0	4.12E-19*	Te 127	1.62E+0	6.18-19
Na 22	2.87E+1	4.59E-15	Te 127m	1.62E+0	1.72E-16
Rb86	1.14E+1	1.41E-16	La 140	3.15E-1	5.67E-19
Cs 137	6.98E+2	8.05E-12	Ce 141	5.19E-1	1.82E-17
Cs 136	8.70E+1	1.17E-15	Ce 144	3.71E-1	1.16E-16
Cs 134	8.78E+1	6.78E-14	Pr 144	3.71E-1	4.92E-21
Sb 125	3.98E+0	3.80E-15	Pr 143	2.73E-1	4.08E-18
I 131	1.74E+1	1.40E-15	Nc 147	1.14E-1	1.73E-18
Te 132	3.45E+0	1.13E-17	Pm 147	2.11E-1	2.28E-16
I 132	3.27E+1	3.15E-18	Pu 238	1.33E-1	7.77E-15
Te 129m	4.86E+0	1.61E-16	Pu 239	3.51E-2	5.72E-13
Te 129	4.86E+0	2.33E-19	Pu 240	4.59E-2	2.01E-13
Sr 89	7.96E-1	2.74E-17	Pu 241	3.90E+0	3.84E-14
Sr 90	5.64E-1	4.12E-15	Pu 242	9.78E-5	2.56E-14
Y 90	5.64E-1	1.04E-18	Np 238	1.49E-6	5.75E-24
Y 91	2.30E-1	9.39E-18	Np 239	6.82E-3	2.93E-20
Zr 95	4.34E-1	2.07E-17	Am 241	1.36E-2	3.98E-15
Nb 95	4.34E-1	1.11E-17	Am 242m	5.35E-4	5.50E-17
Ru 103	5.78E-1	1.80E-17	Am 243	2.19E-4	1.09E-15
Ru 106	4.67E-1	1.40E-16	Cm 242	9.53E-3	2.88E-18
Sb 127	4.85E+0	1.79E-17	Cm 243	1.32E-4	2.51E-18
Ba 140	3.15E-1	4.34E-18	Cm 244	2.75E-3	3.35E-17
H3	1.94E+1	2.01E-15	Na-Ox	-	1.12E-5

*4.12E-19 = 4.12×10^{-19} Total 1.18E+3 pCi/cc 1.12 E-5 gm/cc

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TABLE 6.2-4

SUMMARY DESCRIPTION OF HEAT SINKS USED

FOR CONTAINMENT PRESSURE/TEMPERATURE ANALYSIS

Heat Sink Description	Location	Material	Surface Area	Thickness
Containment Vessel Shell	Above Operating Floor (El. 816' to El. 974')	Steel	107,363ft^2	1.4 inch (nominal)
Sodium Storage Vessel Cell:	Below Operating Floor (El. 733' to El. 816')			
Wall Liner		Steel	9,920ft^2	3/8 inch
Floor Liner		Steel	850ft^2	3/8 inch
Wall		Concrete	9,920ft^2	6 feet
Floor		Concrete	850ft^2	6 feet
Containment/ Confinement Air Gap	Above Operating Floor (El. 816' to El. 984')	Air	--	5 feet
Confinement	Above Operating Floor (El. 816' to El. 984')	Concrete	114,830	4 feet cylinder; 3 feet dome

Table 6.2-5 Lines Penetrating Containment

Penetration	PSAR Section	Number of Valves Required	Type of Valve Required	Line Size	Actuation Signal	Loss of Actua-tion Power Position	Position Accident Conditions	Type of Valve Actuation Required	Valve Position Status at Normal Plant Operations	Primary Actuation	Secondary Actuation	Closure Time Seconds	Valve Configuration (See Fig. 6.2-10)
Decontamination Waste Water Return	9.2	2	Gate	3"	CIS	Closed	Closed	Auto-matic	Open	Auto-matic	Remote Manual	<4	B
IHTS Piping Loop No. 2 Inlet	5.4	0	N/A	24"	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
IHTS Piping Loop No. 2 Outlet	5.4	0	N/A	24"	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
IHTS Piping Loop No. 3 Inlet	5.4	0	N/A	24"	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
IHTS Piping L Loop No. 3 Outlet	5.4	0	N/A	24"	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
IHTS Piping Loop No. 1 Inlet	5.4	0	N/A	24"	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
IHTS Piping Loop No. 1 Outlet	5.4	0	N/A	24"	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A

Table 6.2-5 Lines Penetrating Containment (continued)

Penetration	PSAR Section	Number of Valves Required	Type of Valve Required	Line Size	Actuation Signal	Loss of Actuation Power Position	Position Accident Conditions	Type of Valve Actuation Required	Valve Position Status at Normal Plant Operations	Primary Actuation	Secondary Actuation	Closure Time Seconds	Valve Configuration (See Fig. 6.2-10)
Sodium Transfer Line (In-Cont. to Ex-Cont. Stor. Tank)	9.3	1	Globe	4"	N/A	N/A	Closed	Manual	Closed	Manual	N/A	<30	C
Sodium Transfer Line (EVS Fill & Drain)	9.3	1	Globe	3"	N/A	N/A	Closed	Manual	Closed	Manual	N/A	<30	C
NaK DHRS From Containment	9.3	1	Globe	6"	N/A	Fail In Place	Closed	Remote Manual	Closed	Remote Manual	Manual	<30	H
NaK DHRS To Containment	9.3	1	Globe	6"	N/A	Fail In Place	Closed	Remote Manual	Closed	Remote Manual	Manual	<30	H
RAPS to Cold Box	9.5	2	Globe	1-1/2"	CIS	Closed	Closed	Remote Manual	Open	Automatic	Remote Manual	<10	F
CAPS Inlet Header	9.5	2	Globe	3"	CIS	Closed	Closed	Remote Manual	Open	Automatic	Remote Manual	<10	F
RAPS to Recycle Argon Vessel	9.5	2	Globe	1-1/2"	CIS	Closed	Closed	Remote Manual	Open	Automatic	Remote Manual	<10	F

Table 6.2-5 Lines Penetrating Containment (continued)

Penetration	PSAR Section	Number of Valves Required	Type of Valve Required	Line Size	Actuation Signal	Loss of Actuation Power Position	Position Accident Conditions	Type of Valve Actuation Required	Valve Position Status at Normal Plant Operations	Primary Actuation	Secondary Actuation	Closure Time Seconds	Valve Configuration (See Fig. 6.2-10)
Post Accident Monitoring Sample Line	12.2	2	Gate	1"	N/A	Closed	Closed	Remote Manual	Closed	Remote Manual	Manual	<30	B
Post Accident Monitoring Sample Line	12.2	2	Gate	1"	N/A	Closed	Closed	Remote Manual	Closed	Remote Manual	Manual	<30	B
Post Accident Monitoring Sample Line	12.2	2	Gate	1"	N/A	Closed	Closed	Remote Manual	Closed	Remote Manual	Manual	<30	B
Post Accident Monitoring Sample Line	12.2	2	Gate	1"	N/A	Closed	Closed	Remote Manual	Closed	Remote Manual	Manual	<30	B
Post Accident Monitoring Sample Line	12.2	2	Gate	1"	N/A	Closed	Closed	Remote Manual	Closed	Remote Manual	Manual	<30	B
Post Accident Monitoring Sample Line	12.2	2	Gate	1"	N/A	Closed	Closed	Remote Manual	Closed	Remote Manual	Manual	<30	B
Instrument Air	9.10	2	Globe	2"	CIS	Closed	Closed	Automatic	Open	Automatic Back Pressure	Remote Manual	<4	B

Table 6.2-5
Lines Penetrating Containment (continued)

Penetration	PSAR Section	Number of Valves Required	Type of Valve Required	Line Size	Actuation Signal	Loss of Actuation Power Position	Position Accident Conditions	Type of Valve Actuation Required	Valve Position Status at Normal Plant Operations	Primary Actuation	Secondary Actuation	Closure Time Seconds	Valve Configuration (See Fig. 6.2-10)
Service Air	9.10	2	Globe	2"	CIS	Closed	Closed	Automatic	Open	Automatic Back Pressure	Remote Manual	<4	B
Emergency Chilled Water Supply	9.7	1 1	Ball Check	3" 3"	** N/A	FAI -	Open Open	Remote Manual -	Open -	Remote Manual -	Manual -	<10 -	I G
Emergency Chilled Water Return	9.7	2	Ball	3"	**	FAI	Open	Remote Manual	Open	Remote Manual	Manual	<10	I
Containment Vessel Equipment Hatch	6.2	N/A	N/A	44'6"	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Emergency Personnel Airlock	6.2	N/A	N/A	10'	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Equipment and Personnel Airlock	6.2	N/A	N/A	19'	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Decontamination Water Supply	9.2	2	Gate	3"	CIS	Closed	Closed	Automatic	Open	Automatic	Remote Manual	<4	B
Argon Supply Line	9.5	2	Globe	2"	Back Pressure	***	***	Automatic	Open	Automatic Back Pressure	Remote Manual	<10	D

Table 6.2-5
Lines Penetrating Containment (continued)

Penetration	PSAR Section	Number of Valves Required	Type of Valve Required	Line Size	Actuation Signal	Loss of Actuation Power Position	Position Accident Conditions	Type of Valve Actuation Required	Valve Position Status at Normal Plant Operations	Primary Actuation	Secondary Actuation	Closure Time Seconds	Valve Configuration (See Fig. 6.2-10)
Failed Fuel Monitoring Supply	9.5	2	Globe	2"	CIS	Closed	Closed	Automatic	Open	Automatic	Remote Manual	<10	F
Failed Fuel Monitoring Return	9.5	2	Globe	2"	CIS	Closed	Closed	Automatic	Open	Automatic	Remote Manual	<10	F
N_2 Supply Line	9.5	2	Globe	2"	Back Pressure	***	***	Automatic	Open	Automatic Back Pressure	Remote Manual	<10	D
Emergency Chilled Water Supply	9.7	1	Ball	3"	**	FAI	Open	Remote Manual	Open	Remote Manual	Manual	<10	I
		1	Check	3"	-	-	Open	-	Open	-	-	-	G
Emergency Chilled Water Return	9.7	2	Ball	3"	**	FAI	Open	Remote Manual	Open	Remote Manual	Manual	<10	I
Floor Drain Sump Discharge	9.15	2	Gate*	6"	CIS	Closed	Closed	Automatic	Open	Automatic	Remote Manual	<4	TBD

Table 6.2-5
Lines Penetrating Containment (continued)

Penetration	PSAR Section	Number of Valves Required	Type of Valve Required	Line Size	Actuation Signal	Loss of Actuation Power Position	Position Accident Conditions	Type of Valve Actuation Required	Valve Position Status at Normal Plant Operations	Primary Actuation	Secondary Actuation	Closure Time Seconds	Valve Configuration (See Fig. 6.2-10)
Normal Chilled Water Supply	9.7	1	Ball	8"	**	FAI	Open	Remote Manual	Open	Remote Manual	Manual	<10	I
		1	Check	8"	-	-	Open	-	Open	-	-	-	G
Normal Chilled Water Return	9.7	2	Ball	8"	**	FAI	Open	Remote Manual	Open	Remote Manual	Manual	<10	I
Normal Chilled Water Return	9.7	2	Ball	6"	**	FAI	Open	Remote Manual	Open	Remote Manual	Manual	<10	I
Normal Chilled Water Supply	9.7	1	Ball	6"	**	FAI	Open	Remote Manual	Open	Remote Manual	Manual	<10	I
		1	Check	6"	-	-	Open	-	Open	-	-	-	G
Normal Chilled Water Return	9.7	2	Ball	6"	**	FAI	Open	Remote Manual	Open	Remote Manual	Manual	<10	I
Normal Chilled Water Supply	9.7	1	Ball	6"	**	FAI	Open	Remote Manual	Open	Remote Manual	Manual	<10	I
		1	Check	6"		-	Open	-	Open	-	-	-	G
Normal Chilled Water Supply	9.7	1	Ball	6"	**	FAI	Open	Remote Manual	Open	Remote Manual	Manual	<10	I
		1	Check	6"		-	Open	-	Open	-	-	-	G
Normal Chilled Water Return	9.7	2	Ball	6"	**	FAI	Open	Remote Manual	Open	Remote Manual	Manual	<10	I
Containment Ventilation Air Exhaust	9.6	3	Butter-fly	24"	CIS	Closed	Closed	Auto-matic	Open	Auto-matic	Remote	<4	A

Table 6.2-5 Lines Penetrating Containment (continued)

Penetration	PSAR Section	Number of Valves Required	Type of Valve Required	Line Size	Actuation Signal	Loss of Actuation Power Position	Position Accident Conditions	Type of Valve Actuation Required	Valve Position Status at Normal Plant Operations	Primary Actuation	Secondary Actuation	Closure Time Seconds	Valve Configuration (See Fig. 6.2-10)
Containment Ventilation Air Supply	9.6	3	Butter-fly	24"	CIS	Closed	Closed	Auto-matic	Open	Auto-matic	Remote Manual	<4	A
Containment Purge Line (TMBDB)	9.6	2	Butter-fly	24"	****	Closed	Closed	Remote Manual	Closed	Remote Manual	Manual	TBD	TBD
Containment Purge Line (TMBDB)	9.6	2	Butter-fly	24"	****	Closed	Closed	Remote Manual	Closed	Remote Manual	Manual	TBD	TBD
Containment Vent Line (TMBDB)	9.6	2	Butter-fly	24"	****	Closed	Closed	Remote Manual	Closed	Remote Manual	Manual	TBD	TBD
Containment Vent Line (TMBDB)	9.6	2	Butter-fly	24"	****	Closed	Closed	Remote Manual	Closed	Remote Manual	Manual	TBD	TBD
Containment Vacuum Breaker		2	14"	N/A	Closed		Closed	Auto-matic	Closed	Auto-matic	Manual	<4	TBD
Containment Vacuum Breaker		2	14"	N/A	Closed		Closed	Auto-matic	Closed	Auto-matic	Manual	<4	TBD

Table 6.2-5
Lines Penetrating Containment (continued)

Penetration	PSAR Section	Number of Valves Required	Type of Valve Required	Line Size	Actuation Signal	Loss of Actuation Power Position	Position Accident Conditions	Type of Valve Actuation Required	Valve Position Status at Normal Plant Operations	Primary Actuation	Secondary Actuation	Closure Time Seconds	Valve Configuration (See Fig. 6.2-10)
Emergency Chilled Water Supply	9.7	1	Ball	2"	**	FAI	Open	Remote Manual	Open	Remote Manual	Manual	<10	I
		1	Check		-	-	Open	-	Open	-	-	-	G
Emergency Chilled Water Return	9.7	2	Ball	2"	**	FAI	Open	Remote Manual	Open	Remote Manual	Manual	<10	I
Emergency Chilled Water Return	9.7	2	Ball	2"	**	FAI	Open	Remote Manual	Open	Remote Manual	Manual	<10	I
Emergency Chilled Water Supply	9.7	1	Ball	2"	**	FAI	Open	Remote Manual	Open	Remote Manual	Manual	<10	I
		1	Check		-	-	Open	-	Open	-	-	-	G
Emergency Chilled Water Supply	9.7	1	Ball	2"	**	FAI	Open	Remote Manual	Open	Remote Manual	Manual	<10	I
		1	Check		-	-	Open	-	Open	-	-	-	G
Emergency Chilled Water Return	9.7	2	Ball	2"	**	FAI	Open	Remote Manual	Open	Remote Manual	Manual	<10	I
Fire Protection Standpipe		2		6"									
H_2 Sample Line Suction & Return	TBD	2 2	TBD	TBD 1"out	TBD	TBD	Open	TBD	Closed	Remote Manual	Manual	TBD	TBD

Table 6.2-5
Lines Penetrating Containment (continued)

Penetration	PSAR Section	Number of Valves Required	Type of Valve Required	Line Size	Actuation Signal	Loss of Actuation Power Position	Position Accident Conditions	Type of Valve Actuation Required	Valve Position Status at Normal Plant Operations	Primary Actuation	Secondary Actuation	Closure Time Seconds	Valve Configuration (See Fig. 6.2-10)
H_2 Sampl. Line Suct. & Ret.	TBD	2 2	TBD	TBD 1"out	TBD	TBD	Open	TBD	Closed	Remote Manual	Manual	TBD	TBD
Press. Instr. Line	TBD	N/A	N/A	1"	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A
Press. Instr. Line	TBD	N/A	N/A	1"	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A	N/A

*Preliminary designation of valve type.
**Upon confirmation of a chilled water leak inside containment and determination that the leak cannot be remotely isolated by valves inside containment, manual or remote manual closure of the containment isolation valves is accomplished.
***Open if gas pressure is present; closed on low gas pressure.
****These lines will be activated under the TMBDB scenario.
CIS-Containment Isolation Signal.
FAI-Fail as is.
Note-A summary of containment isolation valving categories and applicable GDC is provided in Table 6.2-5A.

TABLE 6.2-5A
SUMMARY OF CONTAINMENT ISOLATION
VALVING CATEGORIES AND APPLICABLE GDC

1. Lines connected directly to the containment atmosphere (Items 1-3) are provided with two isolation valves (one inside and one outside), which are automatically closed when necessary. This is in conformance with CRBRP GDC 47.

2. Lines connected to the reactor coolant boundary, primary cover gas spaces, or inerted cell atmospheres (Items 8-13) are provided with two isolation valves (one inside and one outside) which have automatic or remote manual closure activation. This is in conformance with CRBRP GDC 46.

3. Lines of closed systems which penetrate the containment and which are neither connected to the reactor coolant boundary nor to the containment atmosphere (Items 6, 7, 14, 15, 18, and 19) are provided with a single valve located outside of and close to the containment as practical. This is in conformance with CRBRP GDC 48.

   The Intermediate Heat Transport System lines are not provided with isolation valves. Justification of this position is included in Section 6.2.4.1.

4. Lines 4, 5, 16 and 17 are provided with two isolation valves, one outside and one inside.

5. The design of the Containment Isolation System will provide the capability of remote-manually closing all the containment isolation valves from the Control Room.

6. Indication of valve position status of all the containment isolation valves will be provided in the Control Room.

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TABLE 6.2-6
TABLE OF BYPASS LEAKAGE PATHS

Penetration Service	Location of Penetration[3]
Decontamination Waste Water Return	RSB
Caps Inlet Header	RSB
Post Accident Monitoring Sample Line	SGB
Post Accident Monitoring Sample Line	SGB
Na Trans. Line	SGB
Na Trans. Line	RSB
NaK DHRS (from)	RSB
NaK DHRS (to)	RSB
Post Accident Monitoring Sample Line	SGB
Post Accident Monitoring Sample Line	SGB
Post Accident Monitoring Sample Line	SGB
H_2 Sampling Line Suction & Return	SGB
H_2 Sampling Line Suction & Return	SGB
Pressure Instrument Line	SGB
Pressure Instrument Line	SGB
Containment Air Exhaust	RSB
Containment Air Supply	SGB
Containment Purge Line	RSB
Containment Purge Line	RSB
Containment Vent Line	SGB
Containment Vent Line	SGB
Instrument Air	SGB
Service Air	SGB

TABLE 6.2-6 (cont.d)
TABLE OF BYPASS LEAKAGE PATHS

Penetration Service	Location of Penetration[3]
Emergency Chilled Water (Supply)	SGB
Emergency Chilled Water (Return)	SGB
Decontamination Water Supply	RSB
Argon Supply Line (R82M0013)	RSB
Sample Gas Return (R82M0014)	RSB
Sample Gas Supply (R82M0015)	RSB
N_2 Supply Line (R82M0016)	RSB
Emergency Chilled Water (Supply) (R23M0024)	SGB
Emergency Chilled Water (Return) (R23M0025)	SGB
Floor Drain Sump Discharge (R27M0026)	RSB
Normal Chilled Water Supply (R23M0027)	SGB
Normal Chilled Water Return (R23M0028)	SGB
Normal Chilled Water Return (R23M0029)	SGB
Normal Chilled Water Supply (R23M0030)	SGB
Normal Chilled Water Return (R23M0031)	SGB
Normal Chilled Water Supply (R23M0032)	SGB

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TABLE 6.2-6 (cont'd)
TABLE OF BYPASS LEAKAGE PATHS

Penetration Service	Location of Penetration[3]
Normal Chilled Water Supply (R23M0033)	SGB
Normal Chilled Water Return (R23M0034)	SGB
Emergency Chilled Water Supply (R23M0035)	SGB
Emergency Chilled Water Return (R23M0036)	SGB
Emergency Chilled Water Return (R23M0037)	SGB
Emergency Chilled Water Supply (R23M0038)	SGB
Emergency Chilled Water Supply	SGB
Emergency Chilled Water Return	SGB
Fire Protection Standpipe	SGB
Liquid N_2	RSB
Emergency/Personnel Airlock	SGB

TOTAL MAXIMUM ALLOWABLE LEAKAGE[2]
(ACTUAL FT^3/DAY)

21.6

NOTES:

1) Total leakage is based on a free volume of 3.6×10^6 ft^3 (total volume above operating floor)

2) The maximum allowable leakage for any path, except for the emergency containment airlock, will not be greater than 20% of the total allowable leakage for all paths.

3) Legend for Building Designations:
RSB - Reactor Service Building
SGB - Steam Generator Building

Figure 6.2-1 Sodium Burning Rate-Primary Sodium In-Containment Storage Tank Failure During Maintenance

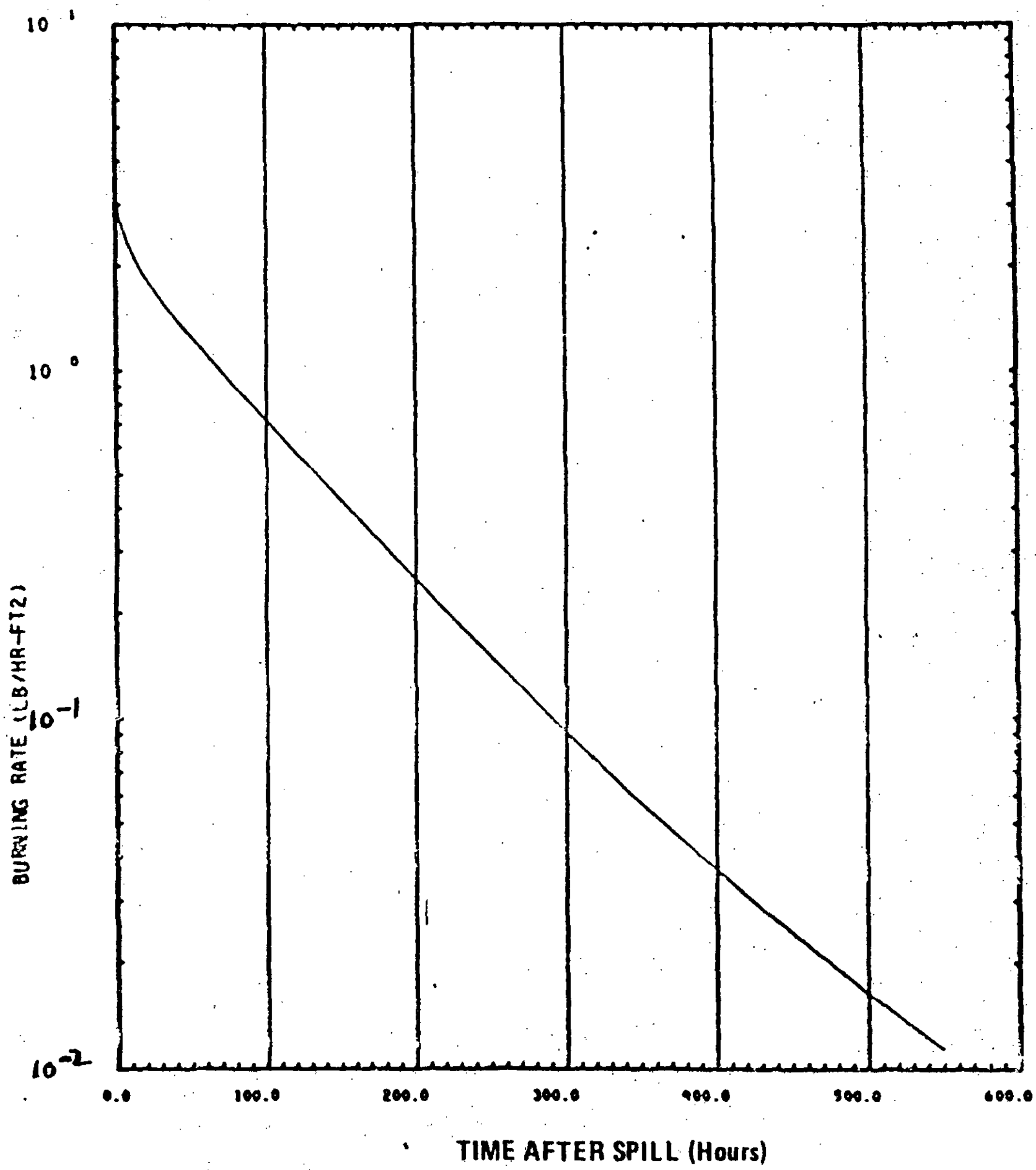

Figure 6.2-2 Containment Gas Pressure-Primary Sodium In-Containment Storage Tank Failure During Maintenance

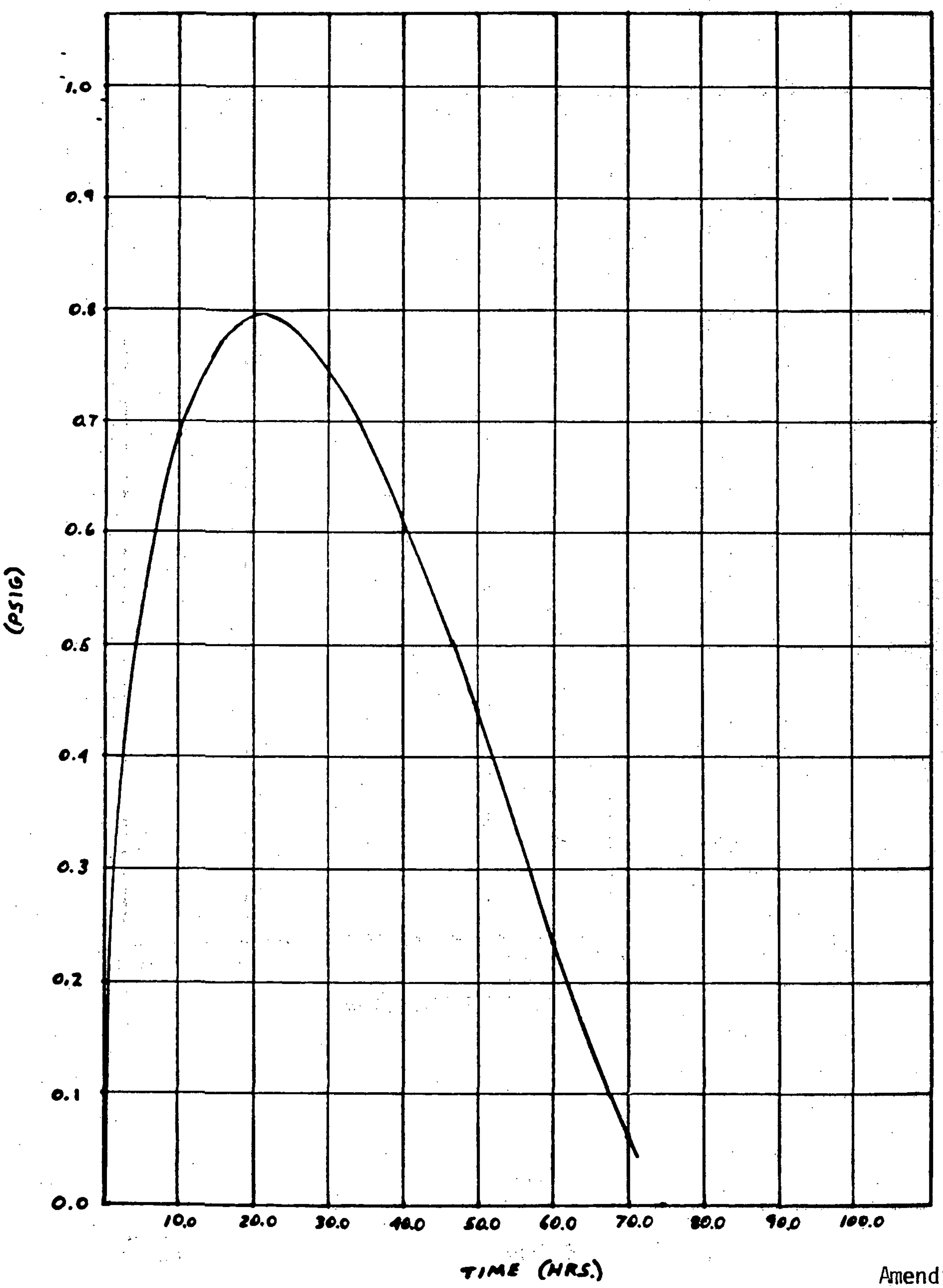

Figure 6.2-3 Containment Gas Temperature-Primary Sodium In-Containment Storage Tank Failure During Maintenance

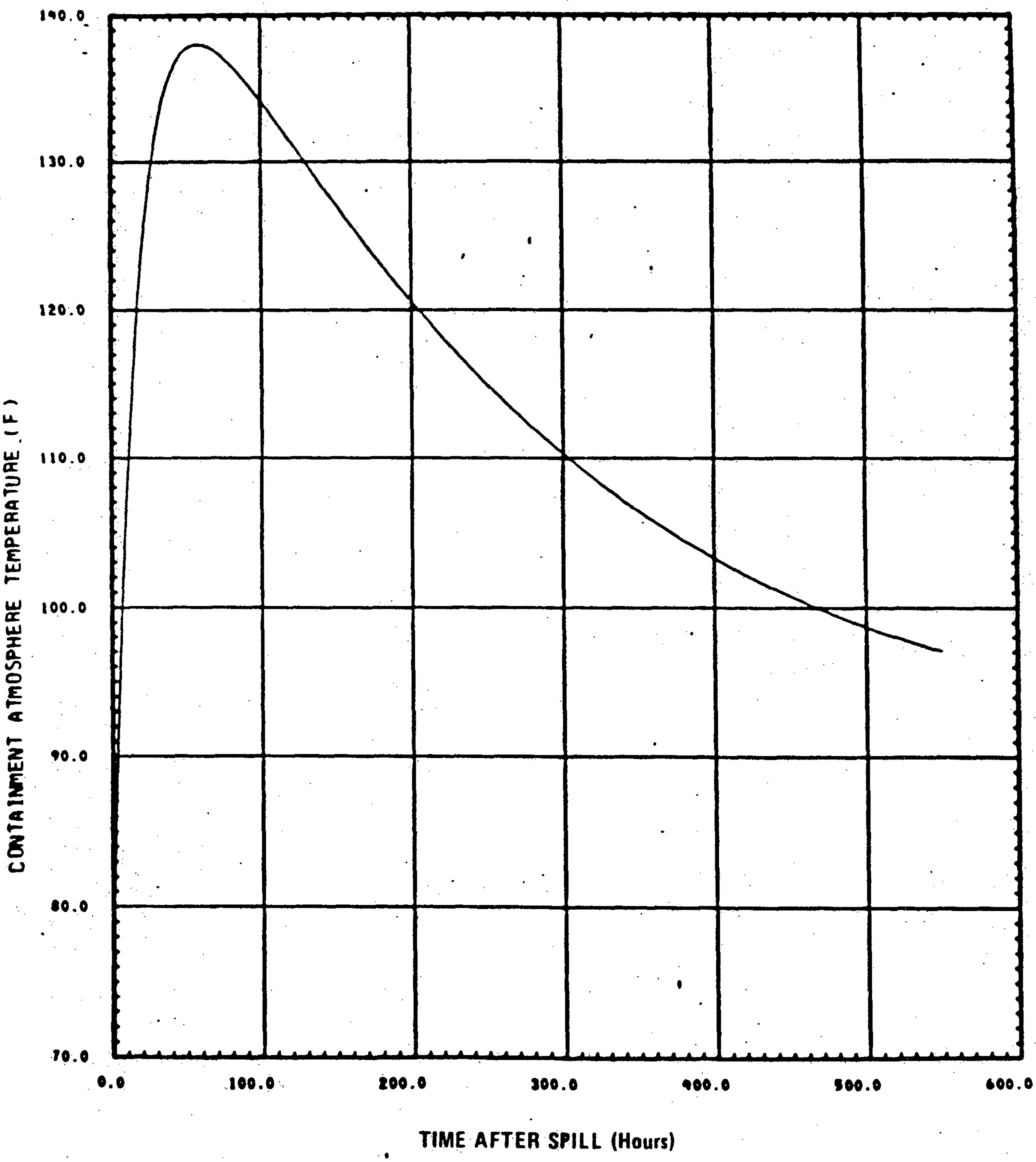

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Figure 6.2-4 Containment Wall Temperature-Primary Sodium In-Containment Storage Tank Failure During Maintenance

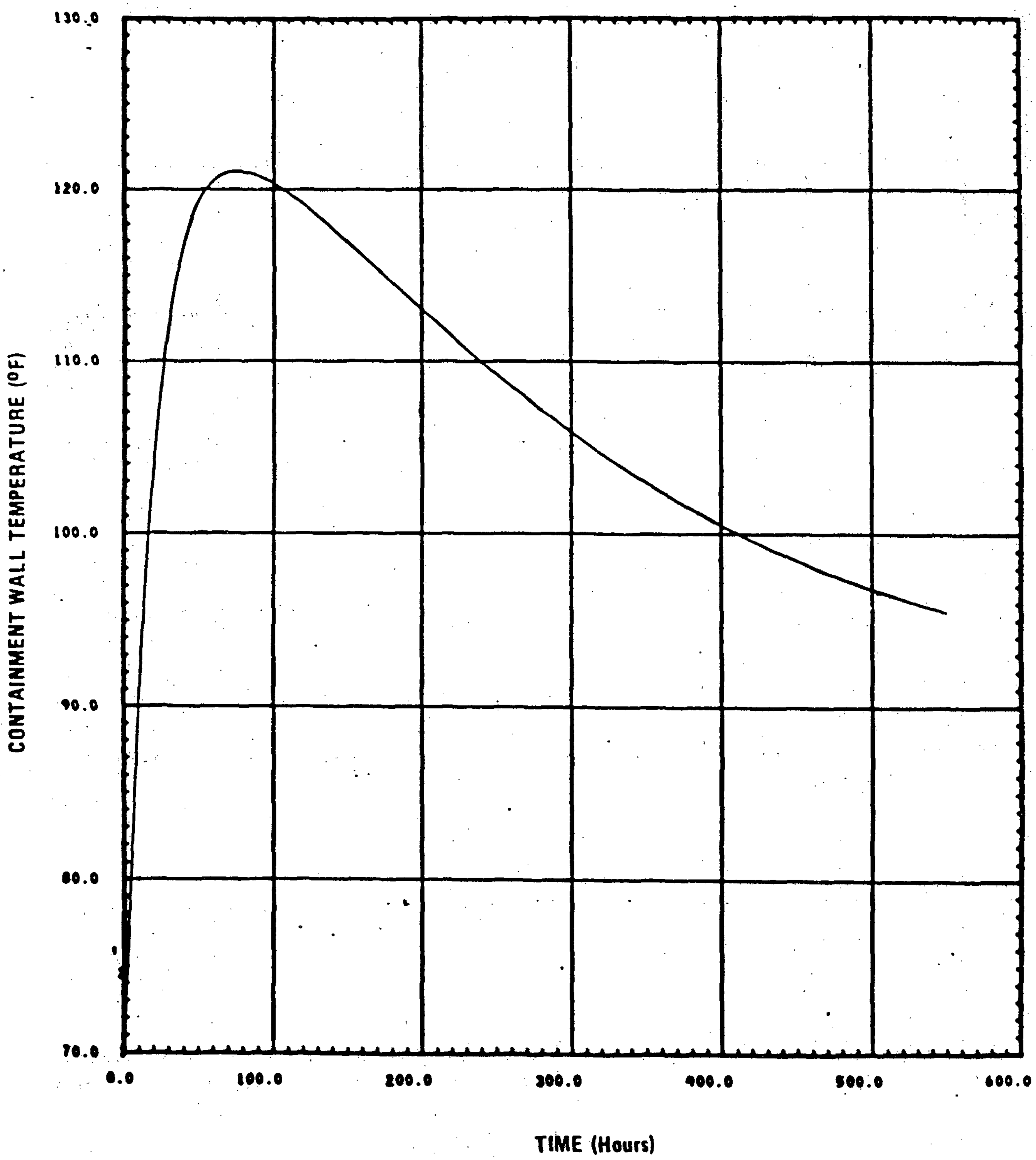

Figure 6.2-5 Containment Aerosol Concentration-Primary Sodium In-Containment Storage Tank Failure During Maintenance

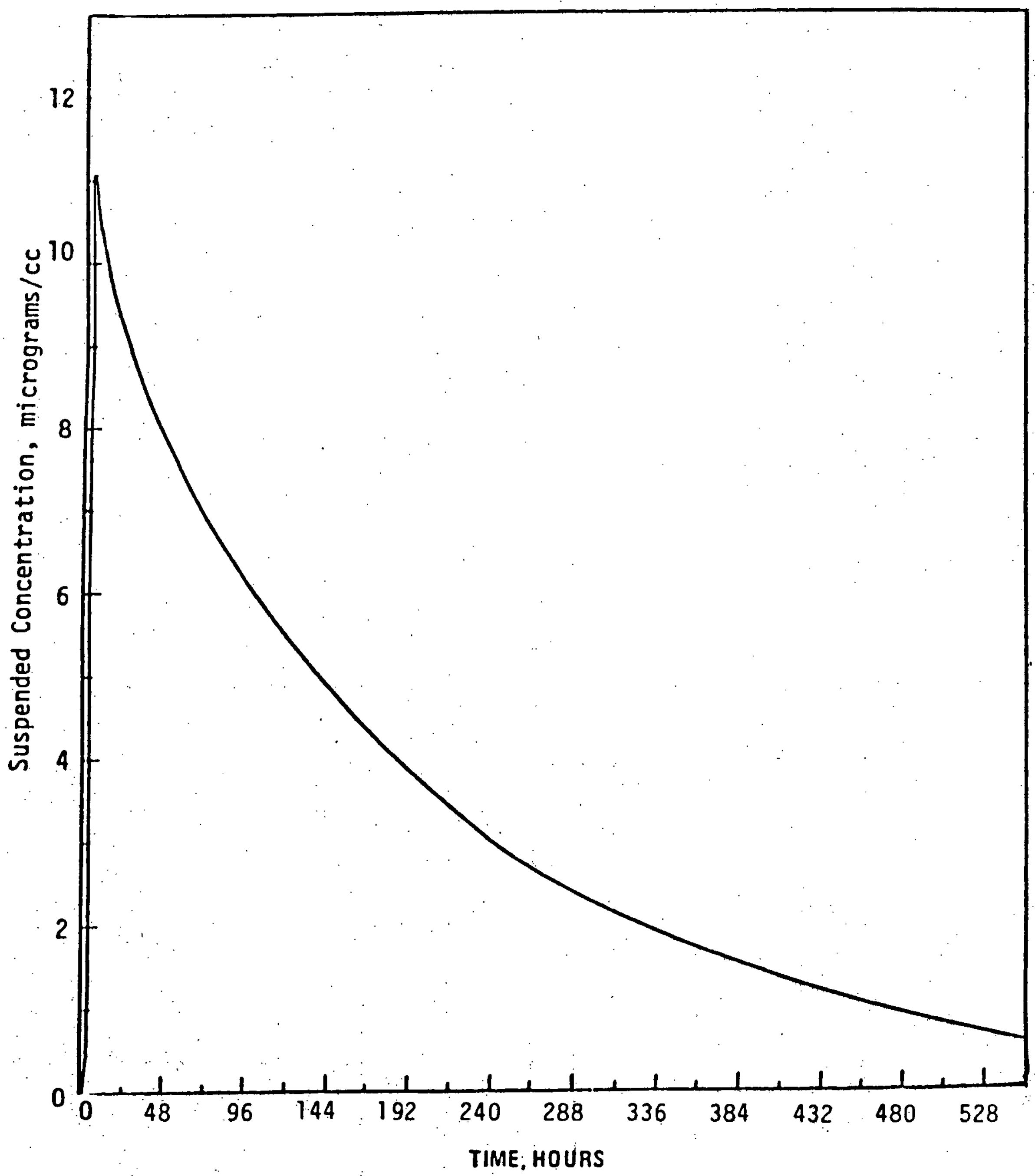

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Figure 6.2-6 Gas Pressure In Storage Tank Cell Following Primary Sodium Storage Tank Failure In Containment During Maintenance

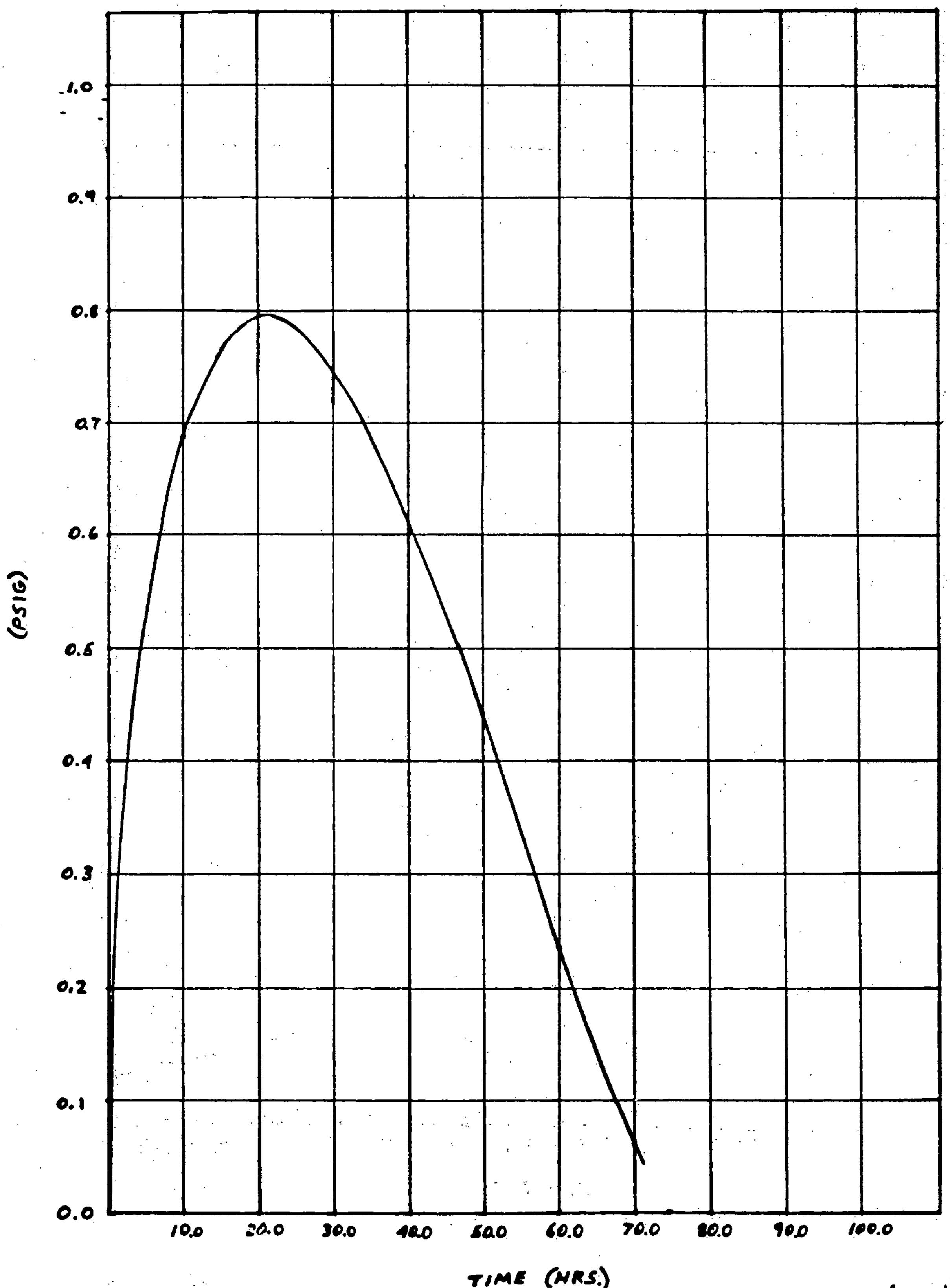

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Figure 6.2-7 Gas Temperature in Storage Tank Cell Following Primary Sodium Storage Tank Failure in Containment During Maintenance

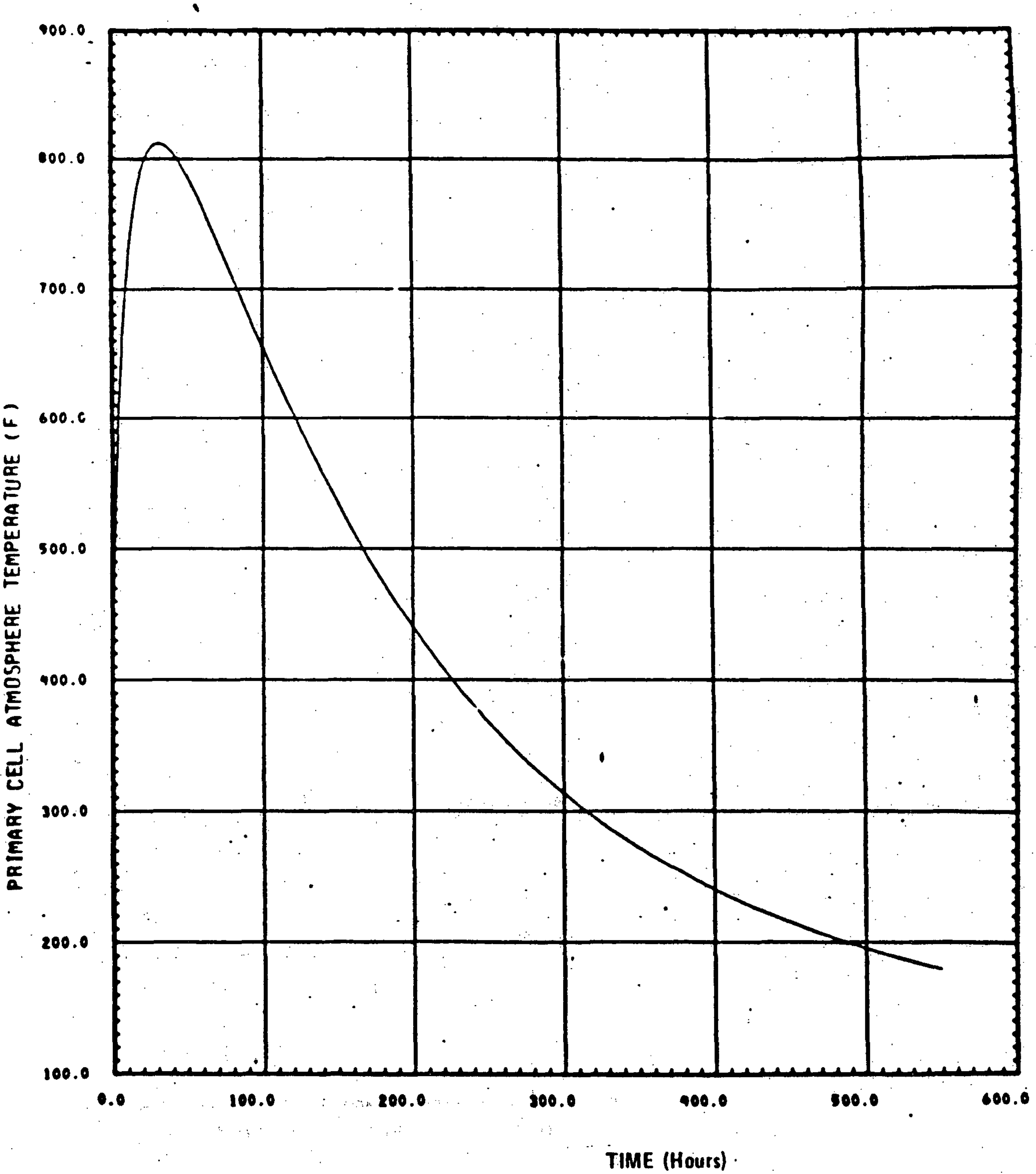

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Figure 6.2-8 Cell Liner (Wall) Temperature in Storage Tank Cell Following Primary Sodium Storage Tank Failure in Containment During Maintenance

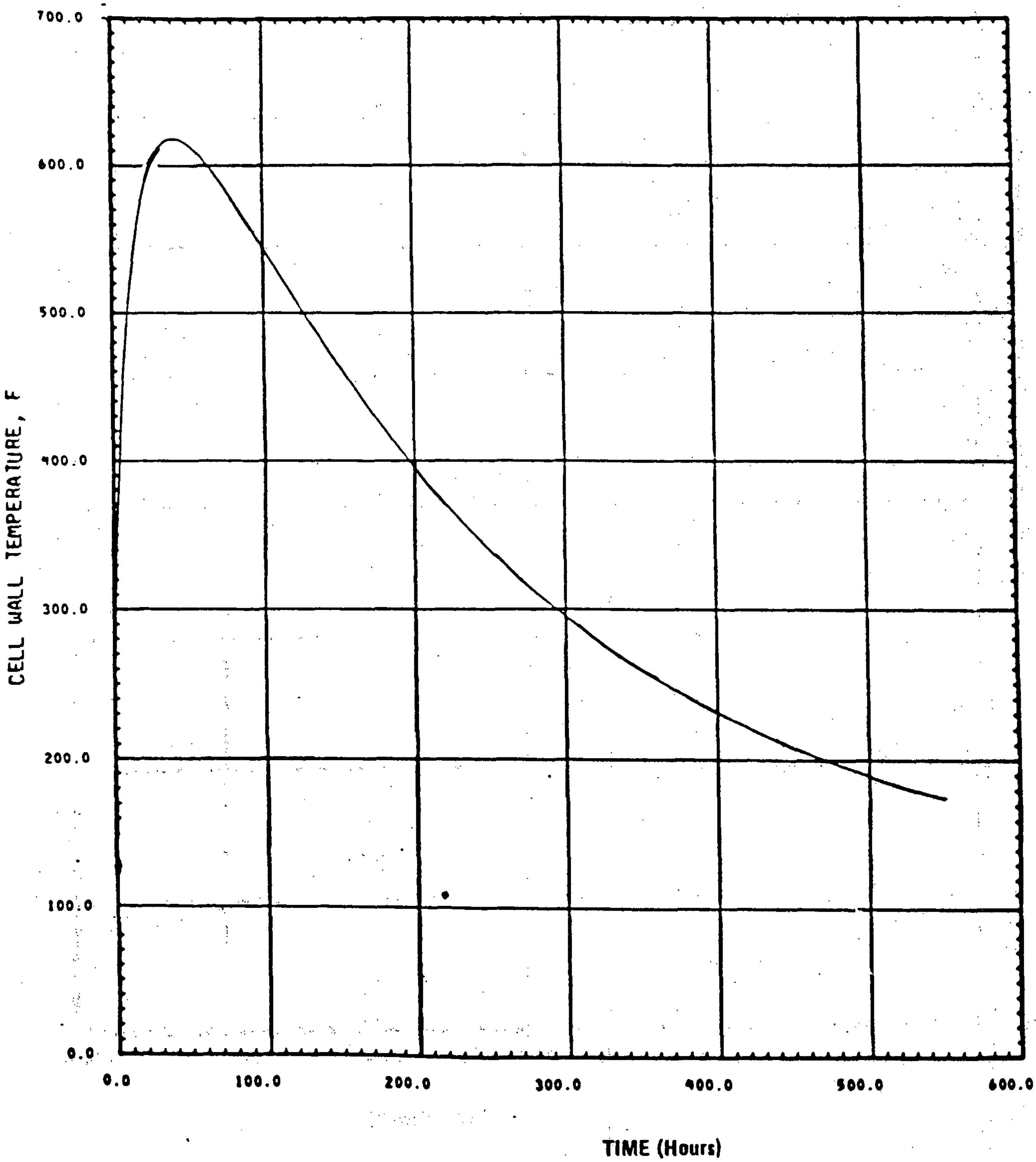

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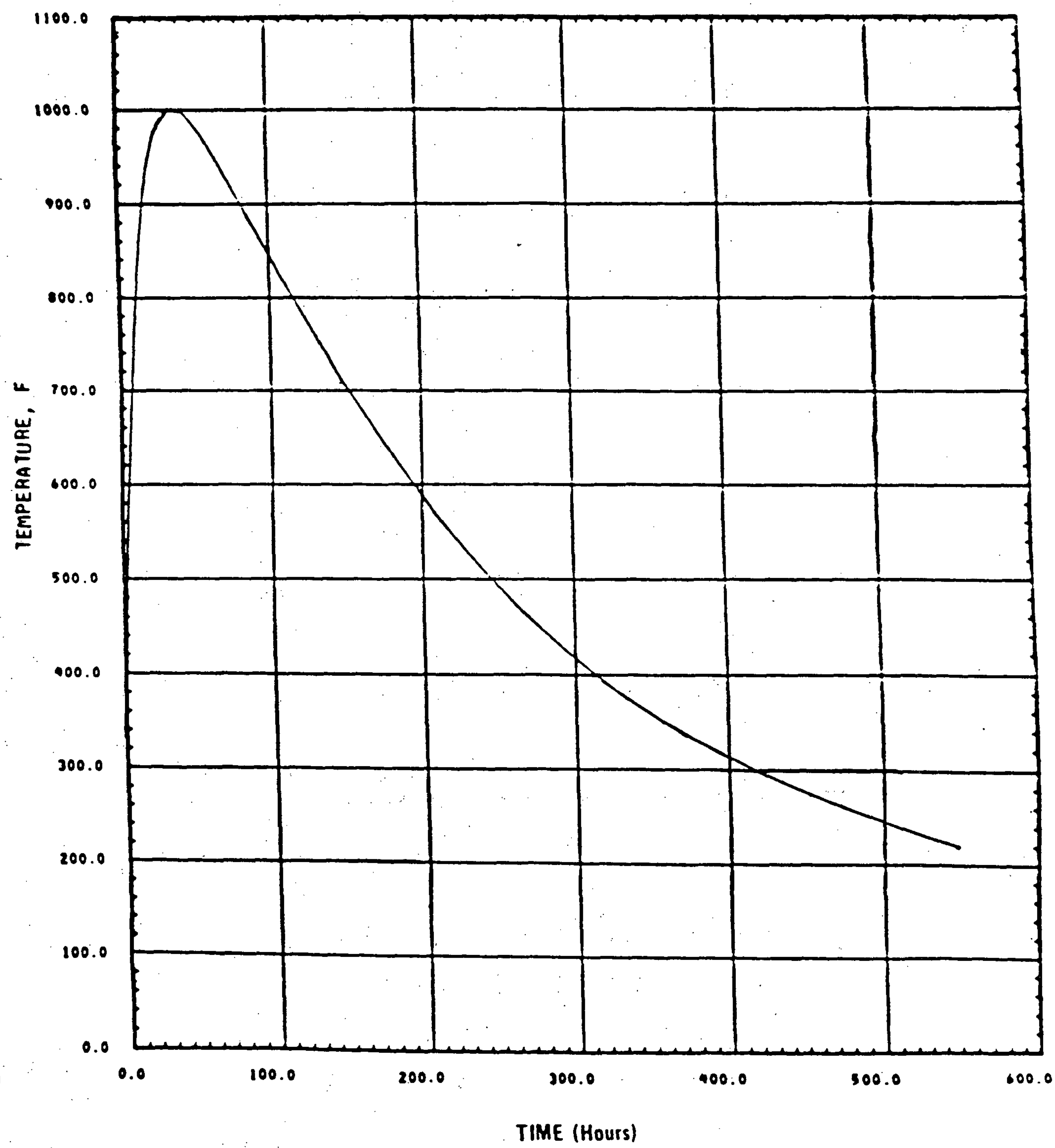
Figure 6.2-9 Cell Liner (Floor Surface) Temperature Following Primary Sodium Storage Tank Failure In Containment During Maintenance



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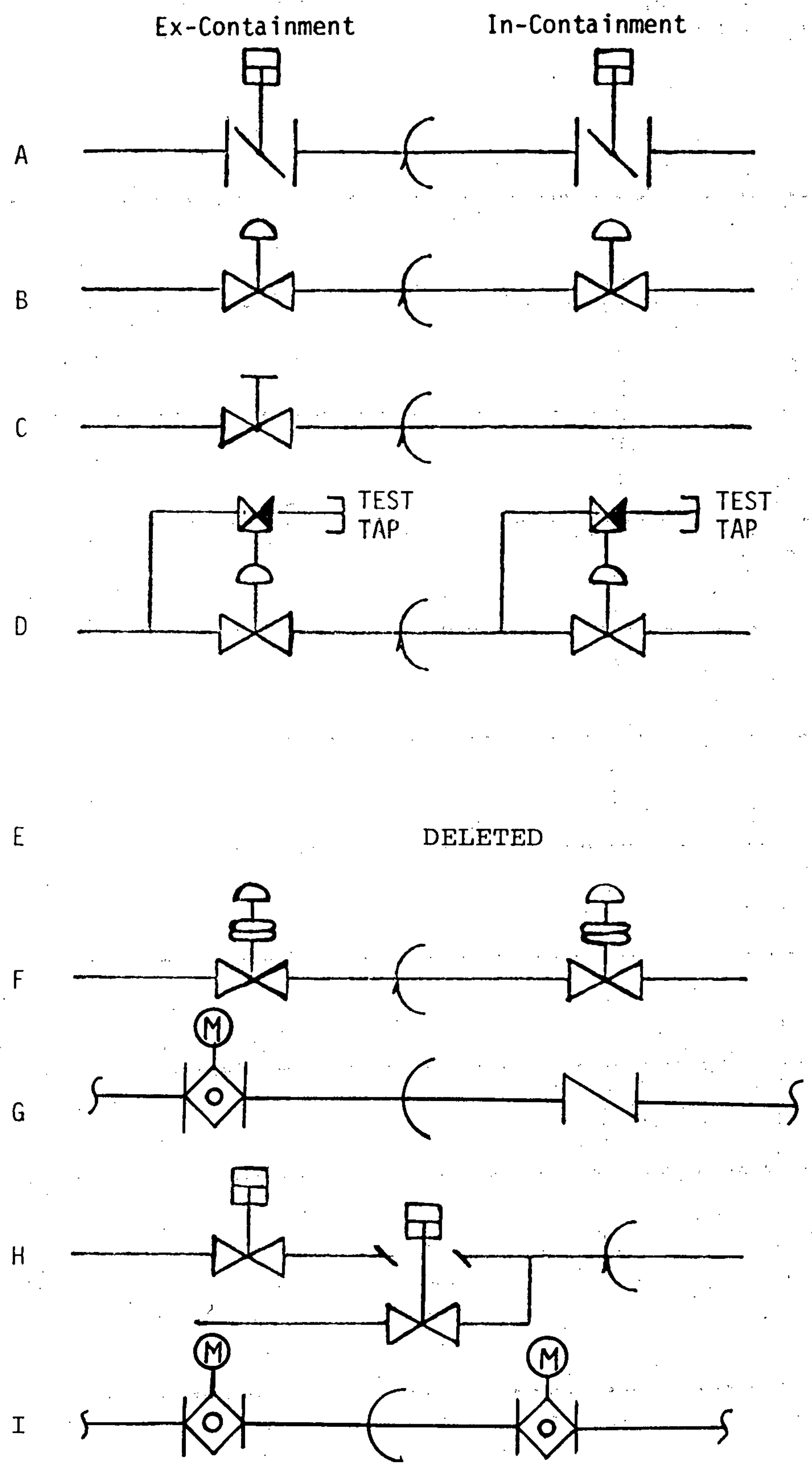

FIGURE 6.2-10 CONTAINMENT ISOLATION VALVE CONFIGURATIONS
(SEE TABLE 6.2-5)

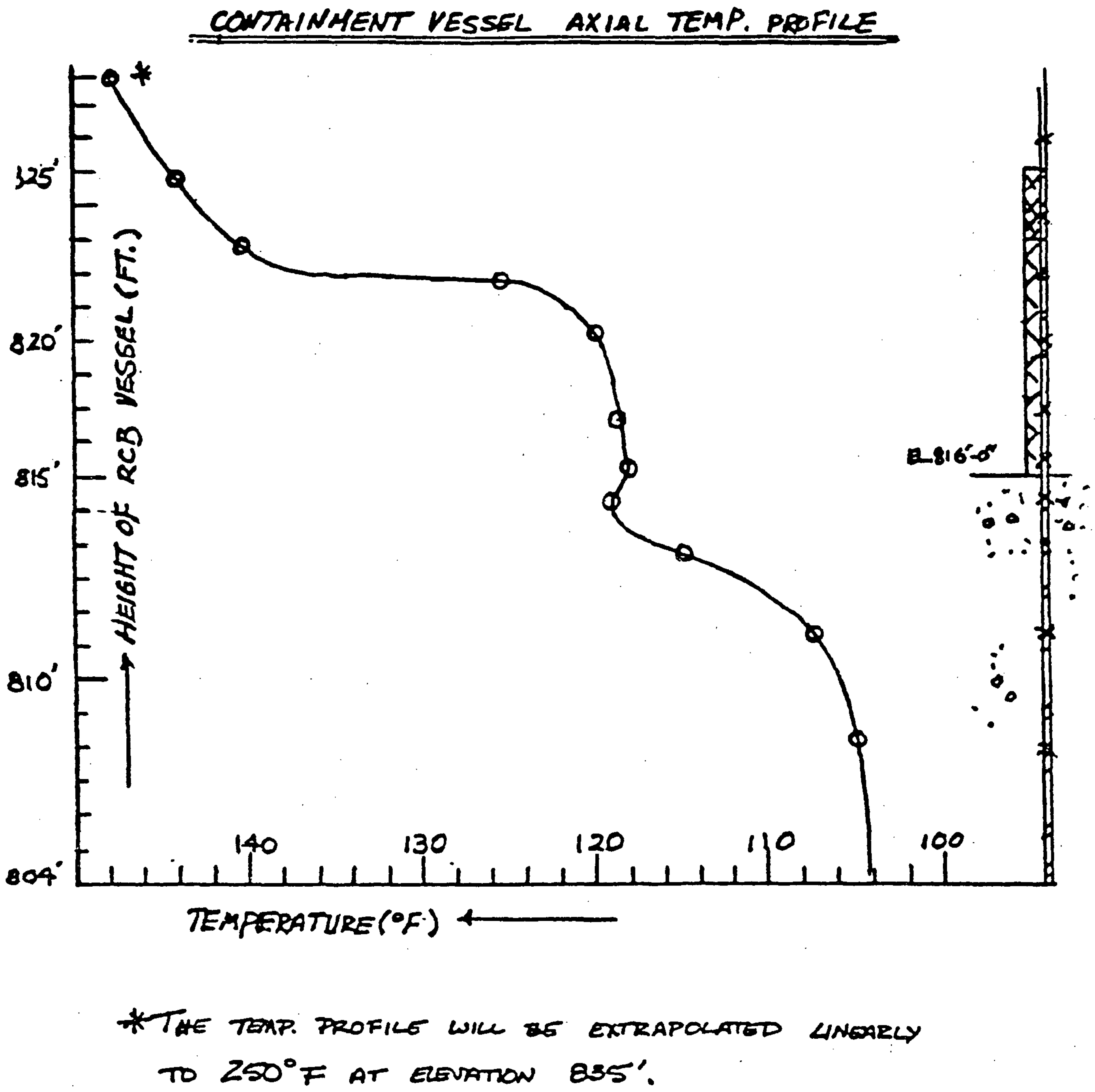

FIGURE 6.2-11

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APPENDIX 6.2A

OPEN CONTAINMENT CONCEPT

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APPENDIX TO PSAR SECTION 6.2
OPEN CONTAINMENT SYSTEM

Open Containment Concept

The CRBRP containment design as described in detail in Section 6.2 incorporates the features which support the "open containment" concept (i.e., operate with a continuous ventilation purge).

1.0 Slow Accident Development

Features incorporated in the design and evaluation which support these characteristics are consistent with current LMFBR technology. Unlike the LWR where the accident can be postulated which result in a very rapid increase in containment pressure and temperature, the postulated events for CRBRP as detailed in Section 6.2.1.3 and Chapter 15 are very slow developing events.

In the LMFBR there are no identified accidents within the design base capable of producing any rapid changes in containment temperature or pressure. For example, for the CRBRP design base accident, the containment atmosphere pressure changes by less than 1 psig over a period of almost 100 hours from accident initiation.

An additional contributor to the slow development of the accident is the design of the inner cell system. The areas within the CRBRP containment which contain radioactive material are heavily compartmented. The reactor coolant boundary is contained within separate, inerted concrete cells, with no interconnections with the containment atmosphere. Similarly, other radioactive items, such as cold traps, are within their own inerted cells. Thus, any release of radioactive material from the reactor coolant boundary and other radioactive components would be to a localized area only. While rapid changes may occur in the temperature, pressure, sodium aerosol concentration, etc., in these cells, such changes would be translated to the containment over a relatively long period of time, and would be substantially reduced by the relative volume of a cell as compared with containment volume.

The containment response (pressures, temperatures and aerosol concentrations) to the containment design basis accident is presented in PSAR Chapter 6.2, Figures 6.2-2 through 6.2-7. As indicated in these figures, the containment and cell pressures and temperatures increase very slowly. This is due to the slow development of the Na pool fire in the primary cell. These figures are developed from the event analysis described in Section 6.2.1.3 where the cell is deinerted and open to the containment atmosphere and the reactor is shut down.

However, under all plant operating conditions, the cells will be inerted and isolated from containment atmosphere. Under these conditions, a sodium spill and resulting fire will be contained in the cell and will not affect the balance of containment. In addition to the compartmented construction of the containment, the reactor coolant system boundary is at a very low pressure with substantial fractions maintained at atmospheric or even sub-atmospheric internal pressures.

These characteristics of LMFBR acidents are sufficiently different in nature from those of LWRs to justify consideration of a very different approach to the containment concept. Because of these characteristics, rapid isolation of containment following an accident, though provided in the design, is not essential from the viewpoint of offsite doses. To illustrate this point, Table 6.2A-1 shows the effect of a 1 hour and a 2 hour delay in containment isolation time for the containment design basis event. For these reasons, it is not considered necessary to restrict access to containment during normal operation, except from considerations of plant security and good management of the plant.

2.0 Containment Access Controls

Access to the containment will be controlled by the CRBRP Security Plan which is in compliance with current regulations pertaining to industrial security.

3.0 Human Factors and Human Error Considerations and ALARA

With the "open containment" concept and controlled access as stated in Section 2.0, the number of personnel in containment during periods of plant shutdown will be substantially reduced, since most of the system inspections and instrument calibrations will be conducted at power. In this way, the in-containment work load can be programmed on an orderly schedule such that only a few people would be inside containment at any given time.

These inspections and calibrations will be conducted by qualified personnel under close supervision using detailed procedures, thereby reducing the potential for error.

The "open containment" concept supports operations in the containment building by plant personnel without the encumbrance contributed by the use of a fresh air mask. Evaluations of the effect on the efficiency of personnel when using fresh air masks and the benefit derived in providing personal protection indicate that personnel efficiency will decrease in the range of 30% to 70% dependent upon the activity being performed, conditions under which the activity is being performed and individual respiratory rates, with an average decrease in efficiency of 50%. A decrease in personnel efficiency would result in an average increase in personnel exposure, which is contrary to the intent of the ALARA program. If a "closed containment" concept were used, the buildup of dose rates inside containment would be that from minor cover gas (argon) leaks, and bleed off from nitrogen operated components. Since this gaseous dose rate is a whole body dose rate, the use of respirators would not provide a decrease in exposure.

A tabulation of personnel activities required in containment is summarized in Table 6.2A-2.

Those activities performed in containment during operation are in the following general categories.

1. Surveillance activities
2. Inspections
3. Maintenance activities
4. Plant Alarm Responses

An estimate of the radiation exposure resulting from performing these activities in a closed containment during operation indicate a substantial increase in exposure as summarized in Table 6.2A-3, with zero days after isolation established as the base for "open containment". This gives a quantitative measure of the ALARA benefits of the concept selected.

4.0 NUREG-718, Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License

NUREG-718, Sections II.E.4.1 through II.E.4.4 are addressed in PSAR Appendix H.

TABLE 6.2A-1

SENSITIVITY OF OFF-SITE DOSES TO DELAY IN ISOLATING CONTAINMENT

Event	Dose Type	Dose Value (rem)		
		Nominal (PSAR)	1-hr delay	2-hr delay
Containment	2-hr Whole Body EB	0.02	0.3	1.1
DBA	Thyroid	0.08	1.1	4.1
	bone	0.3	4.1	15.0
	lung	0.02	0.2	0.8
	30-day Whole Body LPZ	3×10^{-3}	0.05	0.2
	Thyroid	0.01	0.2	0.7
	bone	0.05	0.7	2.4
	lung	3×10^{-3}	0.04	0.1

TABLE 6.2A-2

SUMMARY OF RCB ACTIVITIES NECESSARY DURING NORMAL PLANT OPERATION

Activity Classification	No. of Cells Entered	Time/Cell Est. M/H	Frequency of Activity
Periodic surveillance of electrical equipment located in RCB to ensure no abnormal conditions exist	15	0.75	1/shift
Operational checks on air handling units, unit coolers, fans, heating and cooling coils	6	0.5	1/shift
Surveillance of coolers, fans and panels	11	0.5	1/shift
Monitor Head Heating System control panel, record temperatures	1	0.2	1/shift
Monitor Plug Annulus Sealing System	1	0.2	1/shift
Operational check of PHTS I&C panel	1	0.25	1/shift
Cover gas sampling	3	0.20	1/day
Operation at plugging temperature indicator control panel	1	2.0	1/day
PA-IC System operability checks	all accessible cells	0.5	1/week
Inspection of Dry Standpipe System	14	0.1	1/week
Transfer operating HVAC equipment	6	0.12	1/week
Rotation of operating equipment for Control Rod System	1	.25	1/week
Detailed Inspection of Inert Gas Recirculation and Cooling System	11	.12	1/week
Monitor Head Heating System control panel and record temperatures	1	0.6	1/week

TABLE 6.2A-2

SUMMARY OF RCB ACTIVITIES NECESSARY DURING NORMAL PLANT OPERATION

Activity Classification	No. of Cells Entered	Time/Cell Est. M/H	Frequency of Activity
Monitor Plug Annulus Sealing System	1	0.3	1/week
Leak Detection System components inspection and operational checks	8	1.0	1/week
Sodium sampling	1	23.0	2/week
Perform visual inspection of containment isolation valves	various	.75 (all)	1/month
Scheduled maintenance on Chilled Water System components	1	1.0	1/month
Visual inspection of RCB sump pumps	1	0.5	1/month
Tests on secondary control rod drive mechanisms	1	0.5	1/month
Reactor Vessel Head Seal Service System operability check	1	2.0	1/month
Monitor Head Heating System and record temperatures	1	0.5	1/month
Auxiliary handling machine valve operator cycling	1	1.0	1/month
Leak Detection System component operational checks and maintenance	8	18	1/month
Equipment maintenance	7	1.0	1/month
Verify operability of cell atmosphere sampling and analysis	1	2.0	1/month
Instrument channel calibration and checks	3	18	1/month

There are many other activities required on a longer term basis which are not addressed

TABLE 6.2A-3

RADIATION EXPOSURE ESTIMATES DUE TO RCB
VENTILATION SYSTEM ISOLATION DURING PLANT OPERATION

Days After Isolating Ventilation	0	1	7	30	274
Best-Estimate RCB Dose Rate mrem/hr	0.002	0.013	0.038	0.12	1.01
Estimated Routine RCB Activities man hours/day*	30	30	30	30	30
Estimated Dose mrem/year	0.022	0.14	0.42	1.3	11

*Yearly average based upon routine plant operations excluding plant alarm resposes, surveillance activities not yet defined, and scheduled maintenance with a frequency greater than once every three years.

6.3 HABITABILITY SYSTEMS

A detailed description of the Control Room Habitability System is presented in this Section. The bases used in designing this system are defined in Section 6.3.1.1 and the system is described in Section 6.3.1.2. An evaluation of the provisions of the system is presented in Section 6.3.1.3. This is followed by a description of the testing and inspection to be performed on the habitability system and of the instrumentation to be employed and its functions. Conformance of the Control Room Filtration System to each position of USNRC Regulatory Guide 1.52 is presented in Table 6.3-1. The free air space volume serviced by the Control Room emergency ventilation system is presented in Table 6.3-2. 1

6.3.1 Habitability System Functional Design

6.3.1.1 Design Bases

The functional design of the habitability system of the Control Room is based on a postulated release as described in Section 12.1.2.4. Two widely separated Control Room air intakes were added to the design in order to mitigate the consequences of extremely low probability accidents beyond the design basis, such as hypothetical core disruptive accidents.

In addition to the direct dose, the inhalation dose is calculated due to the radioactivity which is released from the containment/ confinement. 22

The source term associated with the above is described in Section 12.1.3, "Control Room Design Source". The detailed information on energy release for fission products, i.e. noble gases, halogens, volatile solids and remaining fission products as well as the isotopes of plutonium, are presented in Tables 12.1-30 and 12.1-31 of Section 12.1.3. 22

The fission product inventories can be grouped into four (4) classes:

Noble Gases	(Xe and Kr)
Halogens	(Br and I)
Volatile Solids	(CS, Sb, Te, and Sr)
49 Remaining Fission Products	

To summarize, the Control Room shielding design source term consists of the following percentages of the end-of-cycle, equilibrium core fission product inventories: 25

100% Nobles Gases
25% Halogens
1% Volatile Solids
1% Remaining Fission products
1% Plutonium

The specific nuclide quantities that constitute the source term
can be determined from Table 12.1-32 which itemizes the isotopic curie
49 content.

The source volume is the RCB volume above the operating floor
and the source term is assumed to be homogeneously distributed in this
volume. The fission products release rate from the containment to the
containment annulus is 0.1% volume per day. The release from the
annulus is through annulus filters (99% efficient) and 1% through
49 bypass leakage.

Na-24 is not included in the Control Room shielding design
source term. The effect of excluding Na-24 is negligible. There are
approximately 13 curies of Na-24 per pound of sodium coolant at equilibrium conditions. There is no mechanism for discharge of sodium into
the upper containment. Nevertheless, even if 1000 pounds of Na (~13,000
curies Na-24) were to be arbitrarily included in the thermal margin beyond
49 design base source term, its contribution would be insignificant compared
to the more than 10^8 curies of noble gases already included.

49 The Control Room shielding design source term includes the
release of 1.0% of the plutonium fuel into the reactor containment
building. Although the plutonium isotopes do not effect the shielding
requirement for the control room, they are considered in establishing
control room ventilation requirements. The plutonium source terms are
provided in Section 12.1.3.

49 25

The habitability system is required to maintain an environment
within the Control Room at acceptable working conditions following the
postulated major release described above and to conservatively comply
49 with the intent of the CRBRP GDC 17 described in Section 3.1. The provisions of the system are described in Section 6.3.1.2, and the basic
requirements and considerations for the design are as follows:

1. Adequate shielding enclosing the Control Building to
limit the accumulated radiation dose to operating
49 personnel to 5 rem whole body or its equivalent following
the postulated releases.

   The shielding provided will also maintain radiation
   exposures as low as reasonably achievable, and within
   the criteria of 10CFR20 during normal plant operation. 22

2. A Heating, Ventilation and Air Conitioning (HVAC) System to keep the main control room slightly pressurized at all times and at temperatures, humidities, and air purity levels adequate for conducting safe, efficient plant control operations.

3. A low leakage enclosure for the main Control Room and its adjoining rooms to provide the capability for keeping a positive air pressure level within the enclosure.

4. Two alternate and widely separated air intakes and redundant filter units, to limit the amount of contamination entering the Control Room.

5. Airborne hazard monitors that detect unsafe concentrations of smoke, toxic chemicals and unsafe radiation levels, annunciate the presence of the hazard and automatically 22
49 transfer the HVAC system to its accident mode of operation.

6. Appropriate fire suppression equipment.

7. Office and living accommodations appropriate for long term occupancy.

8. An amply stocked inventory of emergency equipment and supplies.

9. Installation of two door vestibules to prevent unfiltered air entering the main Control Room.

49 The habitability system is also required to permit continuous control room occupancy under accident conditions. These provisions enable the operators to remain on duty without relief for as long as required. It is, therefore, not necessary to plan on routine shift changes by the control room personnel. However, if conditions necessitate a change in personnel, this can be accomplished without undue radiation exposure.

The calculated radiation exposure for either ingress or egress
49 at 24 hours after the accident is less than 1 mrem. This estimate is based on direct exposure from fission product gases, halogens, volatile solids, fission products, and activated sodium evenly distributed throughout the reactor containment building free volume. These sources consti-
49 tute direct shine dose only. 25

The ingress or egress exposure resulting from the radioactivity (annulus leakage previously stated) is approximately 3.50 mrem external exposure (whole body dose). The internal exposure to the lungs, thyroid,
49 and bone is less than 1.5 mrem, 9 mrem, and 28 mrem, respectively.

The operator ingress or egress is to be made by motor vehicle to a point adjacent to the control building portal. The car is assumed to travel at an average speed of 15 mph in both directions. The operator walks between the control building and the motor vehicle at an average rate of 4 mph, and personnel will be equipped with supplied air breathing apparel.

49

6.3.1.2 System Design

The Control Room Habitability System is designed to provide a safe, comfortable and appropriately equipped location for personnel controlling plant operations during normal operation and during accident
59 conditions. Features incorporated into this Habitability System to assure these aspects are described below.

A concrete enclosure and special sealed doors are important features in this habitability system. The details of this shielding and the shield wall thicknesses are described in Section 12.1.2.4. Bases used in the design and analyses are also presented in Section 12.1.2.4. Factors considered in these design analyses include thermal margin beyond design basis requirements and the associated activity releases
49 and gamma shine from the containment/confinement. In addition, other features, such as two widely separated air intakes are provided to mitigate the consequences of low probability accidents beyond the design basis.

Another important feature of the Control Room Habitability System is the HVAC system. The HVAC system for the Control Room contains
59 two 100% capacity air conditioning units and two 100% capacity exhaust fans, with one unit each normally operating and the other unit on standby; and two 100% capacity filter units. Complete details of the system are presented in Section 9.6.1. A P&ID of the system is also provided and shown in Figure 9.6-1. This HVAC system contains several aspects that are significant to the Control Room Habitability System. One of these aspects is that this system has full capacity redundancy to assure the capability for controlling the environment after any single component or subsystem failure. A second aspect is the capability to maintain the Control Room at a slightly greater pressure than any of its surroundings at all times. A third aspect of interest to the habitability system is the filter units that purify both make-up and partially recirculated air flows during postulated accidents. Each filter unit in the HVAC system contains bag filters,
59 two banks of HEPA filters and a bank of carbon adsorbers. The filter capability is discussed in Section 9.6. A fourth aspect is the capability for selecting emergency pressurizing air during accidents from widely separated intakes, one located at the SE Corner of the Control Building roof at approximately elevation 880' and the other at the NE corner of the
49 Steam Generator Building Auxiliary Bay at approximately elevation 858'. This along with instrumentation provided, allows selection of the cleaner air source during such periods.

A third important feature of the Main Control Room habitability System is its low leakage enclosure. This enclosure is formed by:

1. Monolithic reinforced concrete floor, walls and roof described in Paragraph 3.8.4.1.2.

2. Low leakage seals for all electrical lines penetrating the enclosure.

3. Low leakage double doors and door seals.

4. Low leakage ventilation system isolation valves. The isolation valves will be bubble tight type.

A fourth important feature of the Control Room Habitability System is the capability for operating in two different modes to assure the maintenance of a positive air pressure level in the control room at all times. The normal operating mode will be the mode utilized during all normal plant operations and during most minor accidents. The accident mode of operation will be initiated by the receipt of a containment isolation signal through the plant protection system or by the detection of smoke, toxic chemicals or high levels of radioactivity in the control room air intake ducts. These detectors and controls will be designed and installed in accordance with engineered safety feature standards. The fire detection instrumentation is discussed in Section 9.13. Detailed information on design bases and location of radiation monitors for the Control Room is presented in Section 12.1.4 and Table 12.1-48. Detailed information on airborne radioactivity monitors for the Control Room is presented in Section 12.2.4 and Table 12.2-3. Fixed-type airborne radioactivity monitors will be provided to continuously monitor the control room breathing air for airborne radioactivities including particulate, gaseous, and iodine activities.

Offices, living accommodations and emergency equipment and supplies are also important features in the Control Room Habitability System.

Fire protection for the main control room is provided by the use of non-combustible equipment in the room and by administrative control over the use of papers, manuals, and log sheets for day-to-day operations. Protection is also afforded by small and hand fire extinguishers for local fire protection.

Face masks and self contained breathing apparatus will be provided to permit emergency operation. Smoke detectors will be installed in the control room. Upon detection of smoke in the control room, an alarm will be sounded. The detectors do not affect operation of the ventilation system.

The operator is responsible for taking appropriate action to extinguish a fire. If he is unable to do so, he may transfer control at the backup control area. Safe shutdown can be achieved and maintained from the backup control area even with the Control Room completely evacuated.

All equipment in the Control Room is designed to operate normally in the temperature range 50°F to 104°F. The safety-related equipment will be designed to operate in the temperature range from 40°F to 120°F. The operator is responsible for judging whether or not fire equipment requires transfer of control to the backup control area.

6.3.1.3 Design Evaluation

The Control Room Habitability System has several features that collectively provide the capability needed to satisfy CRBRP General
49 Design Criterion 17. These features are:

1. Shielding enclosing the Control Room Analyses presented in Subsection 12.1.2 show that this shielding reduces the control room personnel doses from external sources postulated to be associated with thermal margin beyond design basis requirements to a small fraction of that permitted.

2. Low leakage enclosure for the main Control Room. Such an enclosure provides the capability for keeping a positive pressure within the main Control Room at all times.

3. Positive pressure level maintained in the main Control Room at all times. Such a capability assures that in-leakage of contaminated air is minimized.

49 4. Widely separated intakes, one located at the SW corner of the Control Building roof at approximately elevation 880' and the other at the NE corner of the Steam Generator
49 Building Auxiliary Bay at approximately elevation 858' are provided for pressurizing the Control Room during accident conditions. During such conditions, air will be supplied from the less contaminated intake. Radiation
49 monitors indicate the radiation levels in both pressurizing air streams, thus allowing the selection of the cleaner of the two air streams for continued operation. The selection of intakes will allow a cleaner air mass into the Control Room and will reduce contamination within the Control Room significantly. 22

5. Properly sized emergency pressuring air flow rate. The air intakes are designed to be sufficient for main Control Room pressurization during all operating modes.

6. Filters and adsorbers incorporated into the HVAC system.
59 The filter units will help provide the capability for reducing the radiation exposures below the limits of the CRBRP General Design Criterion 17. The 30 day accummulated doses to individual control room personnel for the whole
49 body and critical organs are: 22

Whole Body	7.7E-2 Rem
Beta Skin	4.8E-1 Rem
Thyroid	5.1E-1 Rem
Lung	8.0E-2 Rem
Bone	1.3E-0 Rem

7. Airborne hazards monitor and control subsystem. The automatic detection of smoke, toxic chemicals and radiation levels and the capability to annunciate and to switch the HVAC system to the accident mode of operation, when indicated, to help maintain an acceptable air purity level in the Control Room.

8. Accessibility of the HVAC system low leakage isolation valves. This feature provides easy access from the control room for repair and maintenance of isolation valves during normal and emergency conditions.

9. Full coverage goggles and protective clothing are provided in the Control Room emergency equipment stores.

10. Credit for occupancy is taken when determining the dose received by Control Room personnel. Guidelines for an individual occupancy are:

    100 percent occupancy during the first 24 hours,

    60 percent occupancy beginning with the second day and ending after the fourth day,

    40 percent occupancy beginning with the fifth day and ending after the thirtieth day.

11. Office and living accommodations provided adjacent to the Control Room. These facilities provide a capability for long-term occupancy by Control Room personnel needed for accident control operations.

6.3.1.4 <u>Testing and Inspection</u>

Tests and inspections conducted on the Control Room Habitability System will be mainly concerned with the HVAC System, the capability to keep a positive pressure within the Main Control Room, and the operation of the airborne hazards monitors. The scope includes pre-operational and periodic tests. The pre-operational test objectives will be to demonstrate that the HVAC System, the Control Room enclosure and the airborne hazard monitors are capable of detecting hazards and are capable of establishing and maintaining acceptable conditions for safe, long-term occupancy. In this testing, the capability for performing all necessary functions will be verified. The periodic tests will be scheduled to be performed during the plant lifetime in accordance with the Technical Specifications presented in Subsection 16.4.5.

Standard testing methods will be employed for these tests. System functional testing will be accomplished by simulating signals to specific HVAC controls and observing and/or measuring the response. Flow, thermal, and humidity tests will be accomplished with instrumentation accurate to at least ± 5 percent.

The acceptance standards set for this testing specify that the equipment must demonstrate a capability to perform within rated levels. Appropriate corrective measures needed to comply with the Technical Specification will be planned for all test failures and shortcomings.

6.3.1.5 Instrumentation Requirement

Several kinds of instrumentation are utilized in the Control
Room Habitability System. Redundant radiation monitors, toxic gas detectors,
59 and smoke detectors, are installed in the outside air intake ducts to
detect harmful concentrations of these airborne hazards. Thermostats
and humidistats are positioned in the main control room to control HVAC
system operations. Static pressure differential sensors are installed
59 in the filter units to measure the pressure change across each air filter
bank. Temperature sensors are utilized for duct heater element control
to keep the incoming air above specified limits. Flow sensors are installed
59 downstream from each control room air handling unit to sense air flows
and initiate startup of the standby redundant HVAC train. Details of
this instrumentation are provided in Sections 9.6, 9.13, 12.1.4, and
12.2.4.

6.3.1.6 Effects of Sodium Combustion Products or Other Toxic Gases on the Habitability System

The Control Room operators are protected from the effects of sodium combustion products or other toxic gases listed in Table 6.3-3, by the Control Room Habitability System.

6.3.1.6.1 Sodium Combustion Products

The release of sodium and subsequent sodium fires resulting in the release of Na_2O, Na_2O_2, NaOH, NaH and Na_2CO_3 to the atmosphere is described in Section 7.2.1.1 of the Environmental Report. The effect of the release of sodium combustion products to the atmosphere on the habitability of the Control Room will be analyzed in accordance with Regualtory Guide 1.78. The results of this analysis will be presented in the FSAR.

All stationary sodium and NaK piping, tanks and other components are located within Seismic Category I, tornado hardened buildings. Due to the above arrangement, sodium/NaK pipe or tank rupture caused by external missiles for these components (tornado or turbine) is extremely unlikely and not considered.

Any release of sodium or NaK combustion products to the atmosphere is preceded by a sodium/NaK leak and a subsequent fire. The sodium leak detection system, as described in Section 7.5.5 of the PSAR provides the first early warning to the Control Room of a probable sodium/NaK combustion product release to the atmosphere. If the sodium-NaK leak occurs in a cell containing an air atmosphere and the leak starts a fire, the sodium fire detection system will provide the second early warning to the Control Room of a probable sodium/NaK combustion product release to the atmosphere.

59

During the lifetime of the plant it is expected that in a limited number of occasions Na or NaK will be transported to the site for refill. Sodium is transported to the site in the solid state contained in tank cars. At the sodium unloading station the sodium is melted and pumped into storage tanks located in tornado hardened buildings. NaK is transported to the site in the liquid state, contained in drums and stored on the site in tornado hardened buildings. During transportation or sodium unloading a liquid metal Na or NaK) release can be postulated due to external missiles or tank or durm failures. The probability of this accident is very low, since liquid metal transportation to the site will take place only a few times during the life of the plant. The detectors located in the Control Room air intakes will be able to detect sodium or NaK combustion products if an accident occurs, resulting in a release of liquid metal combustion products to the atmosphere.

To prevent the accumulation of higher than permissible sodium/NaK combustion product concentrations in the Control Room, the HVAC System air intake will be isolated. The HVAC System will operate in a recirculating mode and the minimum outside air, required for the Control Room pressurization, will be taken through the Control Room Filter Unit. The isolation of the Control Room and the start of the HVAC system emergency operation will be automatically initiated by one of the above warning signals. The selection of the proper signal will be based on the Regulatory Guide 1.78 evaluation and will be presented in the FSAR.

6.3.1.6.2 Toxic Gases

The Control Room is classified as Type B construction in accordance with Regulatory Guide 1.78 and is constructed such that the leakage rate is about 0.06 air changes per hour. The Control Room is pressurized to maintain a minimum of 1/4 inch water positive pressure. The Control Room HVAC System air intakes will be provided with redundant, Seismic Category I detectors for those toxic materials which can enter the Control Room air intake in unacceptable concentrations. The Control Room HVAC System air intake and discharge ducts are provided with redundant quick acting low leakage isolation valves. The closing time of the isolation valves is less than 4 seconds. The closing of the isolation valves automatically sets the Control Room HVAC System into a recirculating mode. Two redundant Control Room filter units (as described in Section 9.6.1 of the PSAR) are provided for the maintenance of the Control Room positive pressure during isolation periods.

The toxic materials stored on the site or in onsite facilities are identified in Table 6.3-3. Storage vessels containing toxic materials are provided with instrumentation that will detect a leak, set off an alarm in the Control Room and automatically isolate the Control Room HVAC System air intake and discharge ducts, and set the Control Room HVAC System into a recirculation mode if the extent of the leak warrants it. The effects of accidents resulting in the release of toxic materials from the outside storage facilities on the Control Room habitability will be evaluated on the basis of the above described Control Room features, in accordance with Regulatory Guide 1.78, and will be presented in the FSAR.

Nearby facilities within a 10-mile radius of the plant are described in Section 2.2.1.1 of the PSAR. Early detection and telephone warning, as necessary, will be provided between these facilities and the CRBRP Control Room. Upon a significant toxic gas release and warning from the nearby facilities, the Control Room HVAC System intake and discharge ducts will be isolated, and the Control Room HVAC System will be set into a recirculation mode by the Control Room operators. A maximum postulated airborne release, involves anhydrosu hydrofluoric acid released from the ORGDP and is described in Section 2.2.2.

Transportation accidents involving barge traffic on the Clinch River are not postulated, since the frequency of traffic is extremely low and does not involve toxic materials. Transportation accidents involving railroad traffic are not postulated, since no major railroad encroaches the 5-mile radius of the site. Highway traffic accidents can involve the release of toxic materials, since this is the transportation route to the nearby ORNL and ORGDP facilities. The worst probable accident would involve a 15 ton tank car within a 1.5 mile distance from the Control Room, transporting Hydrofluoric acid. In the event of such an accident, the hydrofluoric detectors located in the Control Room HVAC System intakes, provide an alarm in the Control Room, isolating the Control Room HVAC system intake and discharge ducts and setting the HVAC system into a recirculating mode. The effect of such an accident on the Control Room habitability will be evaluated in accordance with Regulatory Guide 1.78 and will be presented in the FSAR.

TABLE 6.3-1

CONFORMANCE OF THE CONTROL ROOM FILTRATION SYSTEM WITH RESPECT TO EACH POSITION OF U. S. NRC REGULATORY GUIDE 1.52

Regulatory Position 1: Environmental Design Criteria	Control Room Filtration System
a. Each engineered safety feature (ESF) atmosphere cleanup system (ACS) is based on conditions resulting from DBA.	The control room filtration system is designed to limit doses resulting from the Control Room design source term described in Section 12.1.2.4.
b. System design based on 30-day integrated radiation dose.	Filters are designed for a 30-day integrated dose.
c. Adsorber design, based on iodine concentration.	The adsorber will be designed to provide at least 95% filtration efficiency for elemental and methyl iodine.
d. Compatibility of ACS with other ESF's.	Not applicable.
e. Components of systems designed for both the lowest and highest outdoor temperatures.	Not applicable. 1

Regulatory Position 2: System Design Criteria	Control Room Filtration System
a. Redundancy of ACS if designed for mitigation of accident doses.	Two 100% redundant filtration units are provided consisting of a prefilter bank, HEPA filter bank, charcoal filter bank, after HEPA filter bank, centrifugal fan and the required ductwork, dampers and instrumentation as shown in Figure 9.6-1. 22
b. Physical separation of redundant ACS's.	The redundant filter units are located in separate missile protected cells.
c. ACS designated Seismic Category I to prevent release of fission products.	The redundant filter units are designed to Seismic Category I.
d. ACS pressure surge protection.	Not applicable.
e. ACS construction materials effective performance if exposed to radiation.	All components of the Control Room filtration system are designed to withstand the radiation effects of the postulated DBA, for at least 30 days.
f. Maximum volumetric air flow rate per ACS train.	The redundant filter units are designed for a capacity of 8,500 CFM. 22
g. ACS instrumentation provided.	Instrumentation is provided for the redundant filter units to detect and alarm any malfunction.
h. Electrical distribution and power supply conforming to IEEE standards.	The power supply and electrical distribution system for the Control Room filter units is designed to all applicable IEEE standards listed in the reference table of Regulatory Guide 1.52.

TABLE 6.3-1 (cont'd)

Regulatory Position 2: (Continued) System Design Criteria	Control Room Filtration System
i. Avoid bypassing unfiltered air.	Upon Control Room isolation signal from the radiation, toxic chemical or smoke detectors, and air intake supply duct of the Control Room air conditioning units is isolated by quick acting, redundant automatic isolation valves, and all outside air, required for pressurization of the Control Room, is supplied from one of the outside air intakes directly through one of the redundant filter units. 22 1 22
j. Radiation protection for workers in order to perform maintenance.	An approved package unit will be provided, provisions will be made to prevent workers from radiation exposure during removal of activated components, i.e., bagging or casking.
k. Minimization of meteorological effects on outdoor air intakes.	The outdoor air intakes are provided with louvers to minimize the effects of high winds, rain, snow, ice, etc. 22
l. ACS housing and ductwork maximum total leak rate limitations.	Seismic Category I equipment and ductwork will be provided, filter housing and intake ductwork designed to less than 1% leakage.
m. Performance of ACS not classified as an ESF.	Not applicable

TABLE 6.3-1 (cont'd.)

Regulatory Position 3: Component Design Criteria and Qualification Testing	Control Room Filtration System
a. Demister performance and qualification requirements.	An approved filter train with qualified demister will be provided.
b. Effective operating conditions of adsorption units.	An approved filter train with Seismic Category I heater and controls will be provided.
c. Requirements of prefilter materials subjected to radiation.	Prefilters are designed to satisfy all applicable codes listed in the reference table of Regulatory Guide 1.52.
d. HEPA filter requirements and standards of performance.	HEPA filters are designed and tested to satisfy all applicable codes including military specifications MIL-F-51068C and MIL-F-50179A, UL Standard 586, and AACC CS-1.
e. Filter and adsorber mounting frames (design and construction).	HEPA and adsorber mounting frames are Seismic Category I structures, designed in accordance with the recommendations of Section 4.3 of ORNL-NSIC-65 and will satisfy all other applicable codes listed in the reference table of Regulatory Guide 1.52.
f. System filter housings - design and construction.	The Control Room filter unit housings are Seismic Category I structures, designed in accordance with the recommendations of Sections 4.5.5, 4.5.7, and 4.5.9 of ORNL-NSIC-65 and will satisfy all other applicable codes listed in the reference table of Regulatory Guide 1.52.
g. Water drain recommendation.	Water drains are provided for each component section of the filter housing. 1

Regulatory Position 3: (Continued) Component Design Criteria and Qualification Testing	Control Room Filtration System
h. Consideration of radiation effects on various materials.	All filter mounting frames will be designed and constructed in accordance with the recommendations of Section 4.3 of ORNL-NSIC-65 and will satisfy all other applicable codes listed in the reference table of Regulatory Guide 1.52.
i. Removal of gaseous iodine by adsorber (carbon) material.	The adsorber is designed to satisfy AACC CS-8T, RDT Standard M-16-1T and all other applicable codes listed in the reference table of Regulatory Guide 1.52.
j. Fire prevention in adsorber (from auto-ignition)	No water sprays are provided for charcoal filter bank, charcoal has an auto-ignition temperature of 644F with an air flow velocity of 40 fpm. A duct connecting the redundant filter trains before the supply fans is provided to permit a minimum amount of air to pass through a dirty train to remove some of the radioactive induced heat in the charcoal filters. In addition, the adsorber units are enclosed in separate fire rated enclosures and are provided with isolation dampers. This feature provides the capability to isolate the unit, confine the fire and extinguish the fire through oxygen depletion. 22
k. System fans provided with sufficient capacity and pressure.	The Control Room filter unit fans are designed with sufficient static pressure to operate at the maximum system pressure drop.
l. ACS fan or blower should operate under the environmental conditions postulated.	The Control Room filter unit supply fans are designed to withstand all the effects of a DBA.
m. Ductwork designed in accordance with ORNL* recommendations.	Ductwork is designed as Seismic Category I and will be in accordance with the recommendations of ORNL-NSIC-65, Section 2.8.

* ORNL-NSIC-65, Oak Ridge National Laboratory, January 1970. 1

Regulatory Position 3: (Continued) Component Design Criteria and Qualification Testing	Control Room Filtration System
n. System contains minimum of ledges, protrusions, crevices, etc., which impede personnel or create a hazard.	Approved ducts and filter housings will be provided. 1

Regulatory Position 4: Maintenance	Control Room Filtration System	
a. Personnel safety and ready removal of the elements.	A bagging or casking procedure will be employed.	
b. Easy access to components.	Approved standard design.	
c. Filter housings fitted with hinged personnel access doors.	Approved standard design.	22 1
d. Definite mounting frame separation distances.	A distance of five (5) linear feet is provided from mounting frame to mounting frame between components.	22
e. Provision for aligning and supporting filter elements.	Approved standard design.	
f. Adequate separation of filters in the same bank.	Approved standard design.	
g. Use of materials - handling facilities.	Dollies will be provided, the layout of the Control Room filter units and surrounding access areas is shown on Figures 1.2-23 and 1.2-44.	
h. Permanent test probes with external connections.	Approved standard design.	
i. Periodic operation of standby ACS.	Proper test procedures will be established.	
j. ACS components installed after active construction.	Standard procedure.	
k. Adequate vaportight lighting in the filter housing.	Approved standard design.	
l. Electrical water and compressed air services should be nearby but not inside filter housing.	Approved standard design.	
m. Ledges and sharp corners should be avoided.	Standard procedure.	1

TABLE 6.3-1 (cont'd.)

Regulatory Position 5: In-Place Testing Criteria	Control Room Filtration System
a. Testing the air flow distribution to the filters (HEPA and adsorbers).	Acceptance and periodic tests during plant operation will be performed.
b. The in-place testing of HEPA filters should conform to ANSI standards or be replaced.	Testing of HEPA filters will conform to ANSI N101.1 and all other applicable codes listed in the reference table of Regulatory Guide 1.52.
c. The adsorbers should be leak tested.	Testing of the activated carbon adsorber section will conform to NRC Report DP-1082 and all other applicable codes listed in the reference table of Regulatory Guide 1.52. 1

TABLE 6.3-1 (cont'd.)

<u>Regulatory Position 6: Laboratory Testing Criteria for Activated Carbon</u>	<u>Control Room Filtration System</u>
a. If the activated carbon adsorber meets the regulatory requirements, the adsorber section should be assigned the decontamination efficiencies. If it does not meet the regulatory requirements the carbon should not be used in ESF adsorbers.	Inplace testing of activated carbon will be provided.
b. The efficiency of the activated carbon adsorber section should be determined by laboratory testing of representative samples of the activated carbon exposed simultaneously to the same service conditions as the adsorber section.	Samples are periodically tested under the same service conditions as the postulated operating conditions for the ESF adsorbers.
c. The user should prepare detailed procedures for each required field and laboratory test suggested by the regulatory guide.	Test procedures will be provided in accordance with the recommendations of USNRC Regulatory Guide 1.52. 1

TABLE 6.3-2
FREE AIR SPACE VOLUME SERVICED BY THE
CONTROL ROOM EMERGENCY VENTILATION SYSTEM

Cell No.	Title	Elev.	Volume Cu. Ft.
431	Control Panel Area	816'-0"	140,700
432	Computer Room	816'-0"	8,640
433	Shift Suprvsr. & Engr. Rm.	816'-0"	2,520
434	Men's Room	816'-0"	720
435	Women's Room	816'-0"	720
436	Instrument Shop	816'-0"	3,420
437	Charts & Storage	816'-0"	1,296
440	Corridor	816'-0"	2,250
442	Corridor	816'-0"	864
421	Future Equipment	831'-0"	4,320
422	Observation Room	831'-0"	4,680
423	Future Equipment	831'-0"	4,500
429	Corridor	831'-0"	2,520
425	Men's Room	831'-0"	1,080
426	Women's Room	831'-0"	1,080
427	Lunch & Life Support Area	831'-0"	4,750
411A	Control Rm. HVAC Cell	846'-0"	26,950
			211,010

1

Amend. 1
July, 1975

TABLE 6.3-3
ON SITE TOXIC MATERIAL STORAGE

Chemical	Tanks	Capacity of Tanks Gallon	Location	Horizontal Distance To Control Room Intake	Vertical Distance To Control Room Intake	Toxicity Limit Mg/M3	Remarks
Liquid Argon (150 psig)	3	1500	East of SGB	300 ft.	60 ft.	Asphyxiant	
Liquid Argon (175 psig)	3	1500	East of RSB	250 ft.	60 ft.	Asphyxiant	
Liquid Nitrogen (230 psig)	4	3000	East of SGB	300 ft.	60 ft.	Asphyxiant	
Liquid Nitrogen (150 psig)	2	6000	East of RSB	250 ft.	60 ft.	Asphyxiant	
Sodium Hypochlorite (15% solution)	1	7000	At Cooling Towers	650 ft.	60 ft.	-	
Sodium Hypochlorite (15% solution)	1	2500	At River Water Pump House	2500 ft.	125 ft.	-	
Sulphuric Acid	1	6000	TGB, El. 816'	350 ft.	70 ft.	2	Inside Bldg.
Sodium Hydroxide	1	6000	TGB, El. 816'	350 ft.	70 ft.	-	Inside Bldg.
Hydrazine (30% solution)	1	300	TGB, El. 816'	300 ft.	70 ft.	1.3	Inside Bldg.
Ammonia	1	3000	TGB, El. 816'	350 ft.	70 ft.	35	Inside Bldg.
Sodium Hydroxide	2	20,000	RSB, El. 790'	350 ft.	85 ft.	-	Inside Bldg.
Dowtherm J	5	55	Warehouse	350 ft.	70 ft.	276	Storage for makeup —
Argon + 5% Hydrogen (100 psig)	12	100	RSB, El. 733'	200 ft.	140 ft.	Asphyxiant	Inside Bldg.
Liquid Nitrogen (150 psig)	2	1500	RSB, El. 733'	200 ft.	140 ft.	Asphyxiant	Inside Bldg.
Liquid Nitrogen (100 psig)	3	100	RSB, El. 765'	200 ft.	108 ft.	Asphyxiant	Inside Bldg.
Liquid Argon (150 psig)	1	1900	RSB, El. 790'	200 ft.	86 ft.	Asphyxiant	Inside Bldg.
Dowtherm A	1	600	RSB, El. 741'	300 ft.	135 ft.	47	Inside Bldg.

6.4 Cell Liner System

A detailed description of the cell liner system including design bases, system design, design evaluation, testing and inspection is contained in Sections 3.8-B and 3A.8.

6.4.1 Design Base

The cell liner design criteria and design bases are described in Section 3.8-B. The criterion associated with the cell liners as Engineered Safety Features are that the liners will prevent sodium or NaK concrete reactions under design basis accident coolant spill conditions (Section 15.6) by containing the liquid metal spill.

6.4.2 System Design

The design description of the cell liner system is included in Sections 3A.1 and 3A.8 of the PSAR. Section 3A.1 describes the cell liner functional design bases. Section 3A.8 presents a general Design Description and a discussion of Development Testing Programs in Support of the Cell Liner Design.

6.4.3 Design Evaluation

The evaluation of the functional design of the cell liner system is described in Section 3A.8.3. The referenced section provides the results of detailed computer analyses performed on the cell liner system to assure integrity under Design Basis Accident Conditions.

6.4.4 Tests and Inspections

The cell liner systems testing and inspection methods and requirements are described in Appendix 3.8-B. This appendix describes the construction inspection methods to be utilized and the inservice inspection and surveillance requirements used to assure continued integrity of the cell liner system.

The construction inspection methods identified in Appendix 3.8-B are consistent with the examination methods specified for containment liners in ASME B&PV Code Section III, Division 2. The inservice inspection and surveillance examination of the cell liners identified in Appendix 3.8-B is based on the draft ASME B&PV Code Section XI, Divison 3 "Rules for Inspection and Testing of Component of Liquid Metal Cooled Plants", and exceed those specified for Class 3 components. While the referenced codes are not directly applicable to the cell liner system, they have been utilized as guidance in establishing cell liner inspection and surveillance methods consistent with an Engineered Safety Feature (ESF).

6.4.5 Instrumentation Requirements

The cell liner system employed in the mitigation of sodium coolant spills in the inerted equipment cells is a passive structural system and does not require instrumentation.

6.5 Catch Pan

A detailed description of the catch pan system including design bases, system design, design evaluation, testing and inspection are contained in the PSAR and are referenced in this section.

6.5.1 Design Base

The catch pan system design criteria and design bases are described in PSAR Section 3.8-C.

6.5.2 System Design Description and Evaluation

The design description of the catch pan system describing the system design features is included in Section 9.13.2 of the PSAR. A detailed description and evaluation is given in Section 3A.9.

6.5.3 Tests and Inspections

The catch pan system testing and inspection methods and requirements are described in Appendix 3.8-C. This appendix describes the construction inspection methods to be utilized and the inservice inspection and surveillance requirements used to assure continued integrity of the catch pan system.

6.5.4 Instrumentation Requirements

The catch pan system employed in the mitigation of sodium coolant spills in the air filled cells is a passive strucutal system and does not require instrumentation.

CLINCH RIVER

BREEDER REACTOR PROJECT

PRELIMINARY SAFETY ANALYSIS REPORT

CHAPTER 7

INSTRUMENT & CONTROLS

PROJECT MANAGEMENT CORPORATION

CHAPTER 7.0 INSTRUMENTATION AND CONTROLS

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7.0 INSTRUMENTATION AND CONTROLS

7.1 INTRODUCTION

This chapter includes a description of the Instrumentation and Control Systems provided for the CRBRP. Particular emphasis is placed on the description of safety-related systems, which include the Plant Protection System and the safety-related display instrumentation required to maintain the plant in a safe shutdown condition. The Plant Protection System includes all equipment to initiate and carry to completion reactor heat transport and balance of plant shutdown, decay heat removal and containment isolation. Safety-related display instrumentation assures that the operator has sufficient information to perform required manual safety functions and monitor the safety status of the plant. Major control systems not required for safety are described and analysis is included to demonstrate that even gross failure of those systems does not prevent Plant Protection System action. Analysis is also included to demonstrate that the requirements of the NRC General Design Criteria, IEEE Standard 279-1971, applicable NRC Regulatory Guides and other appropriate criteria and standards are satisfied.

7.1.1 Identification of Safety-Related Instrumentation and Control Systems

Table 7.1-1 lists the Safety-Related Instrumentation and Control Systems and includes the definition of Safety-Related Equipment from Section 3.2.1. The entire Plant Protection System, including the Reactor Shutdown System, the Containment Isolation System and the Shutdown Heat Removal System is safety-related. The Reactor Shutdown System input variables are described in Section 7.2. The Containment Isolation Instrumentation and Control System is described in Section 7.4 and Section 7.6. The instrumentation which provides signal input to the Plant Protection System is also safety-related and is described in Section 7.5. Safety-Related Display Instrumentation, which assures that the operator has sufficient information to monitor the safety status of the plant and maintain it in a safe shutdown condition, is discussed in Sections 7.5 and 7.9. Other safety-related instrumentation and control systems including Emergency Chilled Water System, Emergency Plant Service Water System, and Fuel Handling and Storage Interlocks are described in Section 7.6.

7.1.2 Identification of Safety Criteria

In addition to meeting the requirements of the CRBRP General Design Criteria (refer to Section 3.1), the safety-related I&C systems will be designed to meet the applicable requirements of the Regulatory Guides and IEEE Standards listed in Tables 7.1-2 and 7.1-3. The means of compliance with the guides and standards applicable to all safety-related instrumentation and control equipment are described in paragraphs 7.1.2.2 through 7.1.2.11. Compliance with guides or standards applicable to specific I&C systems or equipment are described in the paragraphs related to those systems. The instrument error and other performance consideration are addressed in the description of individual subsystems.

7.1.2.1 Design Basis

The Plant Protection System (PPS) includes the Reactor Shutdown System (RSS), the Containment Isolation System and the Shutdown Heat Removal Systems. The Reactor Shutdown System consists of a Primary and a Secondary System either of which is designed to initiate and carry to completion trip of the control rods and sodium coolant pumps to prevent the results of postulated fault conditions from exceeding the allowable limits. Table 4.2-35 shows the basis for Primary and Secondary RSS performance for the defined fault categories. The performance limits for the fuel and cladding are identified in Section 4. The Reactor Shutdown Systems are described in Section 7.2.

Two diverse Reactor Shutdown Systems have been provided for CRBRP to ensure that the reactor is protected from the consequences of all anticipated and unlikely events even if one of the Reactor Shutdown Systems fails. The two Reactor Shutdown Systems have been made diverse in order to reduce the probability that a common mode failure will prevent a reactor shutdown from taking place. This diversity extends from the sensors used as input to the two systems, through the logic utilized, to the actuation devices required to trip the two different control rod designs.

Table 7.1-4 lists the principal diverse design features present in the two systems. These different design features are discussed in more detail in Section 7.2.1.1. When combined with the separation, qualification and other design requirements arising from the Regulatory Guides listed in Tables 7.1-2 and 7.1-3, these designs provide protection against degradation of performance arising from common mode initiators.

The Containment Isolation System (CIS) is designed to react automatically to prevent or limit the release of radioactive material to the outside environment. The system acts to isolate the interior of the containment by closing the containment isolation valves in the event that radioactive material is released within the containment. Radiation monitors within the containment boundary are used to activate the CIS. A description of this system is given in Section 7.3.

The Shutdown Heat Removal Instrumentation and Control System is designed to provide assurance against exceeding acceptable fuel and reactor coolant system damage limits following normal and emergency shutdowns. The description of this instrumentation and control is given in Section 7.4 for the removal through the auxiliary steam/water system (Steam Generator Auxiliary Heat Removal System (SGAHRS) and Outlet Steam Isolation System (OSIS) and Section 7.6 for removal through the NaK to air system (Direct Heat Removal System (DHRS).

Sufficient instrumentation and associated display equipment will be provided to permit effective determination of the status of the reactor at any time. Section 7.5 provides a description of the instrumentation provided. The above design bases have been applied to the PPS instrumentation listed in Table 7.5-1 and described in Section 7.5. In Section 7.9, a description of the control room, control room layout, operator-control panel interface, instrument and display groupings and habitability are given.

In the areas where the rupture of the steam or feedwater lines can occur, the field instrumentation and control shall be qualified to survive the resulting higher temperature and pressure transient.

The design of the PPS will include provisions for testing using a monitor. The PPS Monitor is an automatic test device which will test the PPS logic by monitoring the Primary PPS logic as test pulses are inserted into the Primary PPS. The Monitor will also trigger the initiation of the test pulses.

The design of the PPS Monitor will meet the following criteria:

1. The design of the PPS logic test system will provide two independent Monitors such that failures in one Monitor will not propagate to the other Monitor. The Monitors will be used to check the test results of each other.

2. Self-test features will be provided for each Monitor.

3. The automatic test of the PPS logic will be considered as a safety related function and will be performed by Class 1E devices. Thus, the PPS Monitors will be Class 1E.

4. Each Monitor will be independent from any other monitor. If isolation devices are required they will be considered IE and designed to the requirements of IEEE 279 and qualified in accordance with IEEE 323.

5. A QA program meeting the requirements of 10CFR50 Appendix B (see Section 17.1) will be applied to the PPS Monitor.

Failure modes and effects analyses will be performed on the PPS logic test system and in particular on the PPS Monitors. The design of the PPS logic test system will prevent any common mode failures identified in the FMEAs.

7.1.2.2 Independence of Redundant Safety-Related Systems

To assure that independence of redundant safety-related equipment is preserved, the following specific physical separation criteria are imposed for safety-related instrumentation.

- o All interrack PPS wiring shall be run in conduits (or equivalent) with wiring for redundant channels run in separate conduits. Only PPS wiring shall be included in these conduits. Primary RSS wiring shall not be run in the same conduit as secondary RSS wiring. Wiring for the CIS may be run in conduits containing either primary RSS wiring or conduits containing secondary shutdown system wiring, but never intermixed. Expanded criteria for physical separation of the CIS are given in Section 7.3.2.2.

- o Wiring for other safety-related systems may be run in conduits containing either primary RSS wiring or conduits containing secondary RSS wiring, but never intermixed, provided that no degradation of the separation between primary and secondary RSS results.

- Wiring for redundant channels shall be brought through separate containment penetrations with only PPS wiring brought through these penetrations. Primary RSS wiring shall not be brought through the same penetration as secondary RSS wiring. Wiring for the CIS and other safety-related systems will be brought through the same penetration as the RSS wiring with which it is routed.

- Instrumentation equipment associated with redundant channels shall be mounted in separate racks (or completely, metallically enclosed compartments). Only PPS channel instrumentation shall be mounted in these racks. Primary RSS equipment shall not be located in the same rack as secondary RSS equipment.

- The physical and electrical separation between DC and AC uninterruptible power supplies, conduits, equipment or racks of instrument channels of safety divisions 1, 2 and 3 shall meet the requirements of IEEE 384 and Reg. Guide 1.75. Redundant instrument channels in the primary RSS shall be physically separated from one another in accordance with the requirements of IEEE 384 and Reg. Guide 1.75.

  Physical separation of primary and secondary RSS of the same division (channel) shall meet the requirements of IEEE 384 and Reg. Guide 1.75 in non-hazard areas. In other areas of the plant primary and secondary RSS equipment and cables of the same division may be located in the same hazard zone.

  Functional capability is maintained in the event of single design basis events which might impact more than one sensor by alternate protective functions as indicated in Table 7.2-2.

- The wiring from a PPS buffered output which is used for a non-PPS purpose may be included in the same rack as PPS equipment. The PPS wiring shall be physically separated from the non-PPS wiring. The amount of separation shall meet the requirements of IEEE 384-1974.

- Electrical power for redundant PPS equipment shall be supplied from separate sources such that failure of a single power source does not cause failure of more than one redundant channel. The power sources and associated wiring shall be separated, as specified in Section 8.

The criteria for cable tray fill, cable derating, cable routing in congested or hostile areas, fire detection and protection in cable areas, and cable markings are defined in Section 8. Separation of redundant safety-related equipment within the control boards is described in Section 7.9.

7.1.2.3 Physical Identification of Safety-Related Equipment

The Plant Protection System equipment will be identified distinctively as being in the protection system. This identification will distinguish between redundant portions of the protection system such that qualified personnel can distinguish whether the equipment is safety-related and, if so, which channel. Color coding, cabinet and wire labeling and other techniques as appropriate will be used.

7.1.2.4 Conformance to Regulatory Guides 1.11 "Instrument Lines Penetrating Primary Reactor Containment" and 1.63 "Electric Penetration Assemblies in Containment Structures for Watercooled Nuclear Power Plants"

There are no instrument lines as defined in Regulatory Guide 1.11 which penetrate primary reactor containment. All electric penetration assemblies in the containment vessel will be designed, constructed and installed in accordance with Regulatory Guide 1.63 and IEEE Standard 317-1972.

7.1.2.5 Conformance to IEEE Standard 323-1974 "IEEE Standard for Qualifying Class IE Equipment for Nuclear Power Generating Stations"

All Class IE equipment will be qualified to confirm the adequacy of the equipment design under normal, abnormal, and postulated accident conditions for the performance Class IE functions. This will be accomplished through a disciplined program discussed in Reference 13 of PSAR Section 1.6, "CRBRP Requirements for Environmental Qualification of Class 1E Equipment." 61

7.1.2.6 Conformance to IEEE Standard 336-1971 "Installation, Inspection and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations"

The installation, inspection and testing of the instrumentation, electrical and electronic equipment during construction will conform to the requirements of IEEE Standard 336-1971. The quality assurance program for the safety-related instrumentation and control equipment will conform to the requirements of Regulatory Guide 1.30. Refer to Chapter 17 for a description of the quality assurance program.

7.1.2.7 Conformance to IEEE Standard 338-1971 "Periodic Testing of Nuclear Power Generating Station Protection System"

Capability for periodic testing is provided to ensure that the Plant Protection System and safety-related instrumentation and equipment meet the necessary performance objectives. This testing capability includes provisions for on-line-testing, testing during shutdown, and calibration as appropriate.

The basic objective of the testing is to assure that the Plant Protection System is performing within its specifications; or conversely, to detect functional failures of redundant components or degradation of important performance features. Further, neither the testing of the equipment nor the features incorporated to permit testing will compromise the independence of redundant components. Since both the Reactor Shutdown Systems uses 3 redundant channels, on-line testing is possible. Specific testing features are described with the associated hardware.

7.1.2.8 Conformance with Regulatory Guide 1.22, "Periodic Testing of Protection System Actuation Functions"

Plant Protection System actuation devices are periodically tested with the RSS during reactor operation. Since the Reactor Shutdown System scram circuit breakers and scram solenoid valves are arranged in 2 out of 3 coincidence logic, scram circuit breakers or solenoid valves may be individually tested during reactor operation. The HTS breakers are arranged to allow testing by using a test breaker to maintain power to the coolant pumps while testing the breaker. While the Reactor Shutdown Systems can be tested through the scram circuit breakers or solenoid valves, the primary and secondary rod release capability cannot be tested during reactor operation since dropping a single control rod will initiate a reactor scram. Scram actuator and control rod insertion times will be functionally tested every time the plant is shut down for refueling. The Containment Isolation System actuated equipment can be tested during reactor operation since containment isolation can take place without necessitating a reactor scram. Shutdown Heat Removal actuation system can be tested on line.

Specific details concerning test and calibration for each protection system channel will not be available until the FSAR. However, the capability for on line functional testing will be provided for all protection system channels. The design will satisfy the requirements of Section 4.10 of IEEE 279-1971 and the recommendations of Regulatory Guide 1.22. The functional performance of a protection system channel will, in general, be tested by inserting a signal in the instrument channel as near as is practical to the sensor. For all tests requiring disconnection of the sensor or modification of the analog input signal, the channel will be placed in a safe condition by tripping the channel comparator associated with the protection system channel under test.
Exception to this rule is made for the functional test of the PPS nuclear flux channels. Since addition of the nuclear flux test signal to the analog signal always drives the channel under test toward a safe (i.e., tripped) condition, the comparator output is not placed in trip before the functional test begins. All protection system instrument channels are functionally tested by varying the magnitude of the test signal thru the trip point to verify that the

comparator trips, then readjusting its magnitude to reset the comparator. After this functional test is completed, the test signal is removed from the instrument channel, and the instrument channel operation is restored. Calibration checks to assure that the protection system channel meets its performance requirements will be accomplished at periodic intervals during regularly scheduled shutdowns. Actuated equipment will, in general, be testable on line. In cases where this is not practical (for example, a control rod cannot be dropped during operation without scramming the reactor), the recommendations of EICSB 22 will be met.

7.1.2.9 Conformance to Regulatory Guide 1.53 "Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems"

Any single failure within the protection system will not prevent proper protection action at the system level when required. The Plant Protection System is periodically tested so that failures are detected. Test schemes will be designed to duplicate as closely as possible the operation being tested. Design precautions include two independent, redundant diverse shutdown systems, each capable of shutting down the reactor; physical independence between redundant channels; and physical barriers utilized for separation between redundant channels. The use of fire retardant materials in construction, fire retardant cable and wire insulation jackets, and physical separation between redundant circuits is relied upon to prevent or mitigate the consequences of a fire.

7.1.2.10 Conformance to Regulatory Guide 1.62 "Manual Initiation of Protective Functions"

The Plant Protection System will provide for manual actuation of each protective action at the system level. The manual initiation of a protective action will result in a Protection System response identical to automatic actuation of the same protective action. For example, manual trip buttons permit operator initiation of reactor scram and containment isolation. The amount of equipment common to both manual and automatic initiation is minimized. Manual initiation of protection actions is designed to go to completion once initiated. No single failure within automatic, manual, or common portions of protective subsystems will prevent initiation of a protective system action by manual or automatic means.

7.1.2.11 I&E Information Notice 79-22 "Qualification of Control Systems"

Safety system design features will be included to mitigate any malfunctions of non-safety grade control equipment which occur as a result of high energy line breaks, such that the effects of such malfunctions will not cause control system failures to complicate any event beyond the PSAR analysis.

TABLE 7.1-1

SAFETY-RELATED INSTRUMENTATION AND CONTROL SYSTEMS*

Reactor Shutdown Systems

Includes all RSS sensors, signal conditioning calculation units, comparators, buffers, 2/3 logic, scram actuators, scram breakers, control rods, back contacts on scram breakers, HTS shutdown logic, coolant pump breakers, and mechanical mounting hardware (equipment racks).

Containment Isolation System

Includes radiation monitoring sensors, signal conditioning, comparators, 2/3 logic, containment isolation valve actuators and valves.

Decay Heat Removal System Instrumentation and Control System

Includes initiating sensors, signal conditioning, calculation units, comparators, logic, auxiliary feedwater pump actuators and controls including feedwater turbine pump, PACC DHX actuators and controls, steam relief valve actuators and valves; sensors, signal conditioning, logic and actuators related to decay heat removal functions of DHRS including control of sodium and NaK pumps and air blast heat exchangers; and sensors, signal conditioning, logic and actuators related to removal of heat from the EVST.

Sodium-Water Reaction Pressure Relief System (SWRPRS)

The instrumentation, initiation and control logic which achieves adequate isolation and blow-down of the waterside of a superheater or evaporator in the event of a sodium/water reaction is Class 1E. The instrumentation used to initiate the isolation and blow-down valves are the rupture disc pressure detectors located downstream of the rupture discs. The other pressure and temperature instrumentation distributed throughout the sodium/water reaction pressure relief subsystem is used for status indication and is not Class 1E.

*The Clinch River Breeder Reactor Plant (CRBRP) safety-related structures, systems, and components are designed to remain functional in the event of a Safe Shutdown Earthquake (SSE). These include, but are not limited to, those structures, systems and components which are necessary:

a. To assure the integrity of the Reactor Coolant Boundary;
b. To shutdown the reactor and maintain it in a safe shutdown condition;
c. To prevent or mitigate the consequences of accidents which could result in potential off-site exposures comparable to the guideline exposures of 10CFR100.

A detailed listing of all Class 1E Instrumentation and Control equipment is provided in Reference 13 of PSAR Section 1.6.

NOTE: Class IE equipment loads are identified in Chapter 8.

Other Safety-Related Instrumentation and Control

Includes Instrumentation and Controls for portions of the following functions to assure the plant is maintained in a safe shutdown condition:

- o Emergency Chilled Water System
- o Emergency Plant Service Water System
- o Instrumentation necessary to assure plant is maintained in safe shutdown status (See Table 7.5-4)
- o Heating, Ventilating, and Air Conditioning System
- o Recirculating Gas Cooling System
- o Process and Effluent Radiation Monitors and Samplers (See Table 11.4-1).

Solid State Programmable Logic System

TABLE 7.1-2

LIST OF REGULATORY GUIDES APPLICABLE TO SAFETY RELATED INSTRUMENTATION AND CONTROL SYSTEMS

1.6	Independence Between Redundant Power Sources and their Distribution Systems (as discussed in Sections 8.3.1.2 and 8.3.2.2)
1.11	Instrument Lines Penetrating Primary Reactor Containment
1.12	Instrumentation for Earthquakes
1.22	Periodic Testing of Protection System Actuation Functions
1.28	Quality Assurance Program Requirements (Design and Construction)
1.29	Seismic Design Classification
1.30	Quality Assurance Program Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment
1.32	Criteria for Safety Related Electric Power Systems for Nuclear Power Plants
1.40	Qualification Tests of Continuous Duty Motors Installed Inside the Containment of Water Cooled Nuclear Power Plants
1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems (as discussed in Section 7.5.12)
1.53	Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems
1.62	Manual Initiation of Protective Actions
1.63	Electric Penetration Assemblies in Containment Structures for Water-Cooled Nuclear Power Plants
1.64	Quality Assurance Program Requirements for the Design of Nuclear Power Plants
1.73	Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants
1.75	Physical Independence of Electric System
1.78	Control Room Habitability During Chemical Release (as discussed in Section 6.3).
1.89	Qualification of Class IE Equipment for Nuclear Power Plants (as discussed in Section 3.11).

TABLE 7.1-3

LIST OF IEEE STANDARDS APPLICABLE TO
SAFETY-RELATED INSTRUMENTATION AND CONTROL SYSTEMS

IEEE-279-1971	IEEE Standard: Criteria for Protection Systems for Nuclear Power Generating Stations
IEEE-308-1974	Criteria for Class 1E Power Systems for Nuclear Power Generating Stations
IEEE-317-1976	Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations
IEEE-323-1974	Qualifying Class IE Electric Equipment for Nuclear Power Generating Stations
IEEE-323-A-1975	Supplement to the Foreword of IEEE 323-1974
IEEE-336-1971	IEEE Standard: Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During Construction of Nuclear Power Generating Stations
IEEE-338-1977	Criteria for the Periodic Testing of Nuclear Power Generating Station Safety Systems
IEEE-344-1975	IEEE Std. 344-1975, IEEE Recommended Practices for Seismic Qualification of Class 1 Equipment for Nuclear Power Generating Stations
IEEE-352-1975	General Principles for Reliability Analysis of Nuclear Power Generating Station Protection Systems
IEEE-379-1972*	IEEE Trial-Use Guide for the Application of the Single-Failure Criterion to Nuclear Power Generating Station Protection Systems
IEEE-383-1974	Standard for Type Test of Class 1E Electric Cables, Field Splices, and Connections for Nuclear Power Generating Station
IEEE-384-1974	IEEE Trial Use Standard Criteria for Separation of Class IE Equipment and Circuits
IEEE-420-1973	Trial-Use Guide for Class IE Control Switchboards for Nuclear Power Generating Stations
IEEE-494-1974	IEEE Standard Method for Identification of Documents Related to Class 1E Equipment and Systems for Nuclear Power Generating Station

* A systematic analysis to show the single failure criteria will be done on the RSS and ESF prior to the OL.

TABLE 7.1-4

RSS DIVERSITY

	Primary	Secondary
Logic:	Local Coincidence	General Coincidence
Sensors:	Inlet Plenum Pressure	Primary Loop Flow
	Primary Pump Speed	Primary Loop Flow
	Intermediate Pump Speed	Intermediate Loop Flow
	HTS Bus Frequency	HTS Bus Voltage
	Steam Flow	Steam Drum Level
	Feedwater Flow	Reaction Products Flow
	IHX Primary Outlet Temperature	Evaporator Outlet Sodium Temperature
Logic Isolation:	Photo Coupling	Direct Coupled
Equipment:		
o Circuitry	Integrated Circuits	Discrete Components
o Power Supplies	Separate vendors utilized	
o Potentimeters	Separate vendors utilized	
o Buffers	Light Coupling	Magnetic Coupling
o Control Rod Release	Circuit Breakers in 2/3 Logic Arrangement	Solenoid Operated Pneumatic Valve in a 2/3 Logic Arrangment

TABLE 7.1-4 (Continued)

44	C16-4	Plant Protection System Comparators, April 1972 - Amendment 1, June 1973
57	C17-5T	Metal-Sheathed, Mineral-Insulated Cable Bulk Material February 1973- Amendment 1, April 1974.
	E6-5T	Collapsible-Rotor, Roller Nut Control Rod Drive Mechanism for Sodium Service, March 1971, Amendment 1, December 1972- Amendment 2, August 1973- Amendment 3, September 1974.
	F2-2	Quality Assurance Program Requirements, August 1973, Amendment 1, December 1973 - Amendment 2, March 1974 - Amendment 3, July 1975.
41 44	F2-4T	Quality Verification Program Requirements, December 1974
	F3-2T	Calibration System Requirements, February 1973.
	F3-39T	Testing of High Temperature Cable for Nuclear Detectors, August 1971.
27	F7-2T	Preparations for Sealing, Packaging, Packing, and Marking of Components for Shipment and Storage, February 1969, Amendment 1, October 1971 - Amend 2, September 1972.

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7.2 REACTOR SHUTDOWN SYSTEM

7.2.1 Description

7.2.1.1 Reactor Shutdown System Description

The Reactor Shutdown System (RSS) consists of two independent and diverse systems, the Primary and Secondary Reactor Shutdown Systems, either of which is capable of Reactor and Heat Transport System Shutdown. All design basis events can be terminated without exceeding the specified limits by either system even if the most reactive control rod in the system cannot be inserted. To assure adequate independence of the shutdown systems, mechanical and electrical isolation of redundant components is provided. Functional or equipment diversity is included in the design of instrumentation and electronic equipment. The Primary RSS uses a local coincidence logic configuration while the Secondary RSS uses a general coincidence. Sufficient redundancy is included in each system to prevent single random failure degradation of either the Primary or Secondary RSS.

As shown in the block diagram of the Reactor Shutdown System, Figure 7.2-1, the Primary RSS is composed of 24 subsystems and the Secondary RSS is composed of 16 subsystems. Figure 7.2-2A is a typical Primary RSS instrument channel logic diagram. Each protective subsystem has 3 redundant sensors to monitor a physical parameter. The output signal from each sensor is amplified and converted for transmission to the trip comparator in the control room. Three physically separate redundant instrument channels are used. When necessary, calculational units derive additional variables from the sensed parameters, with the calculational units inserted in front of the comparators as needed. The comparator in each instrument channel determines if that instrument channel signal exceeds a specified limit and outputs 3 redundant signals corresponding to either the reset or trip state. The 3 outputs of each comparator are isolated and recombined with the isolated outputs of the redundant instrument channels as inputs to three redundant logic trains. The recombination of outputs is in a 2 out of 3 local coincidence logic arrangement.

Operating bypasses are necessary to allow RSS functions to be bypassed during main sodium coolant pump startup and ascent to power. Operating bypasses are accomplished in the instrument channels. For bypasses associated with normal three loop operation, the bypass cannot be instated unless certain permissive conditions exist which assure that adequate protection will be maintained while these protective functions are bypassed. Permissive comparators are used to determine when bypass conditions are satisfied. When permissive conditions are within the allowable range, the operator may manually instate the bypass.

If the permissive condition goes out of the allowable range, the protective function is automatically reinstated. The trip function will remain reinstated until the permissive conditions are again satisfied and the operator again manually initiates the bypass. Operator manual bypass control is not effective unless the bypass comparator indicates that permissive conditions are satisfied. A functional diagram of the Primary and Secondary bypass permissive logic is shown in Figure 7.2-2AA.

Two loop bypasses are established under administrative control by changing the hardware configuration within the locked comparator cabinets. These bypasses are also under permissive control such that the plant must be shutdown to establish two loop operation and if the shutdown loop if activated the bypass is automatically removed.

Bypass features included within the Primary and Secondary RSS hardware for two loop operation will be deactivated during all three loop operating modes so that the three loop operating configuration can not be affected by these bypass features either by operator action or by two loop hardware failure.

Bypass permissives are part of the Reactor Shutdown System (RSS), and are designed according to the RSS requirements detailed elsewhere in this section of the PSAR.

Continuous local and remote indication of bypassed instrument channels will be provided in conformance with Regulatory Guide 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems".

Figure 7.2-2B is a logic diagram of the Primary RSS logic trains. The outputs from the comparators and 2/3 functions are inputs to a 1 out of 24 general coincidence arrangement. The output of the 1/24 is an input to a 1 out of 2 with the manual trip function to actuate the scram breakers. The scram breakers are arranged in a 2 of 3. When 2 or more logic trains actuate the associated scram breakers, power to the control rods is open circuited and the control rods are released for insertion to shutdown position with spring assisted scram force. Open circuiting the control rod power initiates Heat Transport System shutdown.

In the Secondary RSS, the sensed variables are signal conditioned and compared to specified limits by equipment which is different from the Primary RSS equipment. The secondary logic is configured in general rather than local coincidence to provide additional protection against common mode failure. Each instrument channel comparator outputs its trip or reset signal to a 1 of 16 logic module. The 3 redundant secondary instrument channels from each subsystem feed 3 redundant logic trains, which are coupled to the secondary scram actuators. Figure 7.2-2D is a logic diagram for the Secondary RSS logic.

The Secondary RSS consists of 16 protective subsystems and monitors a set of parameters diverse from the Primary RSS as shown in Table 7.2-1. However, since a measure of nuclear flux is necessary in both the Primary and Secondary RSS, nuclear flux is sensed with compensated ionization chambers in the primary while fission chambers are used in the secondary. The Primary RSS monitors primary and intermediate pump speed while the Secondary RSS monitors primary and intermediate coolant flow. Similarly, the steam flow to feedwater flow ratio is used in the Primary RSS while the steam drum level is sensed for the Secondary RSS.

Figure 7.2-2C is a typical Secondary RSS instrument channel logic diagram. Each protective subsystem has 3 redundant sensors to monitor a physical parameter. The output signal from each sensor is conditioned for transmission to the trip comparator located in the control room. Redundant instrument channels are used. When necessary calculational units are placed in front of the comparators to derive additional variables. The output of the comparators are input to redundant logic trains in a general coincidence arrangement.

Bypass of secondary comparators is implemented in the same fashion as in the primary system except that different equipment is used to provide the permissive comparator function.

Figure 7.2-2D is a logic diagram of the Secondary RSS logic trains. The outputs from the instrument channels are input to a 1/16 general coincidence arrangement. The 1/16 output controls the solenoid power sources through isolated outputs. Isolated outputs are also provided to initiate a shutdown of the Heat Transport System and the primary control rod motor-generator sets. A trip latch-in function is provided to assure that once initiated, the scram will go to completion. The remaining redundant logic trains provide the other two signals for the 2/3 function.

Figure 7.2-2 shows the RSS interface with the Heat Transport System (HTS) pump breaker control. Two HTS pump breakers are connected in series for each HTS pump. Each HTS breakers receives input from the Primary RSS and Secondary RSS pump trip logic. Upon receipt of a reactor trip signal from either Primary or Secondary RSS, the HTS pump breakers open to remove power from the Primary and Intermediate pumps.

Provisions are made to allow testing of the HTS breaker actuation function during reactor operation. A test breaker is used to bypass the main HTS breaker during a test condition. Test signals are then inserted through the Primary or Secondary RSS pump trip logic to open the main HTS breaker. Mechanical interlocks are provided on the bypass breakers to prevent more than one main HTS breaker in any loop from being bypassed at a time. Control interlocks are provided which make the breaker test inputs ineffective unless the bypass breakers are properly installed. Main HTS breaker and test breaker position status is supplied as part of the RSS status display on the main control panel.

The Secondary RSS also initiates a shutdown of the primary control rod motor-generator sets. This feature assures that all drive power to the primary rod control equipment is interrupted upon Secondary RSS actuation. Provisions are made to allow testing of this feature during reactor operation.

The RSS subsystems do not directly require the reactor operator or control system to implement a protective action. However, manual control devices to manually initiate each protective function are included in the design of the Plant Protection System.

Where signals are extracted from the Reactor Shutdown System, buffers are provided. These buffers are designed to meet the requirements of IEEE-279-1971. The buffers prevent the effects of failures on the non-IE output side from affecting the performance of the RSS equipment. The buffers are considered part of the RSS and meet all RSS criteria.

<u>System Testability</u>

Both Reactor Shutdown Systems are designed to provide on-line testing capability. For the Primary RSS, overlapping testing is used. The sensors are checked by comparison with redundant sensor outputs and related measurements. Each instrument channel includes provisions for insertion of a signal on the sensor side of the signal conditioning electronics and test points to measure the performance at the comparator (or calculational unit) input. Where disconnection of the sensor is unavoidable for test purposes, the comparator is tripped when disconnected. The instrument channel electronics including trip comparators and bypass permissive comparators are tested for ability to change value to beyond the trip point and provide a trip input to the logic. The comparators and logic are tested by the PPS Monitor. A set of pulsed signals are inserted from the monitor into the comparators associated with one subsystem and the logic output is checked by the Monitor to assure that logic trip occurs for the correct combinations of comparator trips. The logic and scram breakers are tested by manually tripping one logic train and observing that the corresponding breakers trip. HTS breakers are tested by maintaining power to the pump through a bypass circuit breaker and manually inserting a test signal to the pump trip logic.

For the Secondary RSS, insertion of the test signal into a channel causes the entire train (comparator, logic, and scram solenoid valves) to trip. Testing of the pump breaker trip is identical to that for the Primary RSS.

System Instrumentation

The instrumentation used by the RSS to detect the occurrence of off-normal plant conditions includes:

- Neutron Flux

  The Primary RSS uses three compensated ionization chamber power range nuclear sensors evenly spaced around the reactor vessel. The Secondary RSS uses three fission chamber wide range nuclear sensors evenly spaced around the reactor vessel. See Section 7.5.2 for detector details.

- Reactor Inlet Plenum Pressure

  The Primary RSS uses six pressure detectors, two per HTS primary loop, located as close as practical to the reactor vessel inlet plenum in the elevated primary cold leg piping. Each set of two detectors comprises an instrument channel. The outputs of the two detectors in each loop are auctioneered. The resultant output signal is provided to the comparator. See Section 7.5.2 for detector details.

- Sodium Pump Speed

  The Primary RSS uses three redundant tachometers per primary and intermediate HTS pump to measure pump speed. See Section 7.5.2 for detector details.

- Sodium Flow

  The Secondary RSS uses six permanent magnet flowmeters to measure HTS sodium flows. One flowmeter is located in each of the primary and intermediate cold legs. Each flowmeter provides three redundant measurements of loop flow. See Section 7.5.2 for detector details.

- Reactor Vessel Sodium Level

  The Primary RSS uses four sodium level detectors evenly spaced within the reactor vessel. Three of these detectors provide redundant active signals to the RRS. The fourth detector is used as a spare. See Section 7.5.3 for detector details.

- Underfrequency Relay

  The Primary RSS uses three underfrequency relays, one per coolant loop pump bus. The underfrequency relays are located on the HTS pump buses.

o Steam Flow

The Primary RSS uses three redundant steam mass flow signals per loop. Each steam supply loop has one venturi flowmeter, three differential pressure sensors, three temperature sensors, and three pressure sensors located between the superheater exit and the main steam header. Three redundant steam mass flow signals are generated by pressure and temperature compensation of the venturi flowmeter analog signal. See Section 7.5.2 for detector details.

o Feedwater Flow

One venturi flowmeter, three differential pressure sensors, and three temperature sensors per steam generator loop supply the Primary RSS with three redundant temperature-compensated feedwater mass flow signals. See Section 7.5.2 for detector details.

o IHX Primary Outlet Sodium Temperature

The Primary RSS uses three redundant thermocouples, mounted in three thermowells, per loop to measure the sodium temperature in the primary cold leg. See Section 7.5.2 for detector details.

o Steam Drum Level

The Secondary RS uses three redundant reference column level sensors to determine the water level in each steam drum. The level sensor is density compensated. See Section 7.5.2 for detector details.

o Evaporator Outlet Sodium Temperature

The Secondary RSS uses three redundant thermocouples per loop, mounted in thermowells. These thermocouples are provided to monitor the sodium temperatures in each intermediate cold leg. See Section 7.5.2 for detector details.

o Undervoltage Relay

The Secondary RSS uses three undervoltage relays, one per coolant loop pump bus. The undervoltage relays are located on the HTS pump buses.

o Sodium-Water Reaction

The Secondary RSS uses three redundant pressure sensors located in the reaction products vent line immediately downstream from each rupture disk to detect if the rupture disks have blown. See Section 7.2.1 for details.

The configuration of the instrumentation in the protective subsystems is described in Section 7.2.1.2.

Primary Shutdown System Logic

The Primary RSS Logic is implemented using integrated circuits to minimize the scram delay time. Other advantages include minimizing power consumption and space required and maximizing testability, maintainability and reliability. The Primary RSS Logic is arranged as shown in Figure 7.2-3.

In each logic train, twenty-four 2/3 coincidence logic circuits feed a 1/24 module, whose output is coupled to the final actuation logic and rod actuators by a transistorized power amplifier. When only one comparator of any or all protective functions is tripped, the logic signal output remains positive (reset). When any two comparators of a protective function trip and provide a negative logic signal to the protective logic, the output of the corresponding 2/3 module also trips to a negative logic signal. This negative logic signal in turn trips the 1/24 logic module which outputs a negative logic signal to the final actuation logic and removes power from the scram breaker undervoltage coil.

Light emitting diodes and phototransistors are utilized to provide complete electrical isolation at strategic points through the Primary RSS logic. There is no electrical connection between the comparator output and protective logic input. Consequently, an internal electrical fault in a single instrument channel or comparator cannot propagate to the other channels, protective functions, or logic trains of the protective system. Each logic train is electrically isolated from the other so that protective action can be initiated regardless of any internal electrical fault in a single logic train.

The equipment needed to implement the 24 protective subsystems of the Primary RSS includes the sensors, signal transmitters and amplifiers or equivalent, calculational units, comparators, logic isolators, 2/3 logic modules, 1/24 logic modules, logic drivers, final scram actuation circuitry and breakers, buffers, permissives and bypasses. A three section equipment cabinet is used to house the equipment for each of the three instrument channels including the calculational units, comparators, power supplies and buffers. A two section equipment cabinet is used to house the equipment for each of the three logic trains and single equipment cabinets house signal conditioning equipment for each channel. This arrangement of equipment within cabinets provides the necessary mechanical separation of redundant equipment.

Secondary Reactor Shutdown System Logic

The Secondary RSS Logic consists of the 16 protective subsystems arranged in a general coincidence configuration, as shown in Figure 7.2-4. In this arrangement, the outputs of instrument channel A comparators are directly coupled to a 1/16 logic circuit module in logic train A, as are the outputs of instrument channel B with logic train B and the outputs of instrument channel C with logic train C.

When the sensed parameter in an instrument channel exceeds its setpoint (trips), the comparator outputs a zero (trip) signal to the 1/16 logic module, which in turn outputs a zero (trip) signal to the scram latch and scram solenoid valves (see Figure 7.2-2D). The 1/16 logic module output voltage changes to zero regardless of the output of the other comparators. The output of the 1/16 logic modules are combined in a 2/3 coincidence by the 3 solenoid valves located within the Secondary RSS rod. Electrical isolation of the logic output to the solenoid drivers and Heat Transport System shutdown logic (HTS pump breakers) is shown in Figure 7.2-2D. Redundant isolated outputs are provided from each Secondary RSS logic train to the Secondary RSS pump trip logic where they are combined in a 2/3 logic. Trip signals are provided to the HTS pump breakers and to the primary control rod motor-generator sets when 2/3 of the redundant Secondary RSS channels are in a tripped condition.

The equipment of the Secondary RSS includes sensors, signal conditioning equipment (transmitters), calculational units, comparators, 1/16 logic modules, solenoid drivers, secondary final actuation logic and actuators, buffers, permissives and bypasses. The equipment is designed using hardware which is diverse from that used in the Primary RSS. Since each instrument channel is uniquely associated with a logic train, a four section equipment cabinet houses each of the instrument channel comparators, logic trains and solenoid drivers. Single equipment cabinets are used to house signal conditioning equipment for each channel. This arrangement of equipment within separate, completely metallically enclosed cabinets provides the necessary mechanical separation between redundant equipment.

Channel Output Monitoring

Channel output monitoring is included to provide the operators with early indication of anomalous instrumentation performance. This equipment is not safety related. If the output of one channel differs from either of the redundant channels by more than a preset amount, the channel output monitoring circuitry alarms this condition.

7.2.1.2 Design Basis Information

The RSS initiates and carries to completion Reactor, Heat Transport and Balance of Plant Shutdown if any of the off-normal plant conditions listed in Table 7.2-2 occur. The table also shows the frequency classification of the postulated fault, and the first Primary and Secondary RSS subsystems which act to terminate the fault. As detailed in Chapter 15, the RSS design described below provides the performance necessary to appropriately limit the results of the postulated events. Table 7.2-1 shows the Primary and Secondary RSS subsystems which use the instrumentation described previously to determine the off-normal conditions and trip the plant.

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7.2.1.2.1 Primary Reactor Shutdown System Subsystems

High Flux

The High Flux Protective Subsystem (Figure 7.2-5) initiates trip for positive reactivity insertions at or near full power. This subsystem assures that sustained operation will not occur with the fuel near incipient centerline melting. As shown on the figure, the subsystem compares the compensated ion chamber output signal with a fixed setpoint and initiates trip when the signal exceeds the setpoint. Analysis of the performance of this subsystem is based on worst case time response of the instrument of 50 milliseconds and worst case trip point of 115% of full power. This subsystem is never bypassed.

Flux-Delayed Flux

The Flux-Delayed Flux Protective Subsystems (Figure 7.2-5) initiate a trip for rapid sustained reactivity disturbances, either positive or negative, which occur anywhere in the load range. Two subsystems are provided; one for positive flux rates and one for negative flux rates. These subsystems prevent undesired thermal transients caused by rapid changes in power with flow held constant. As shown on the figure, the flux signal is compared with a signal proportional to the nominal load level as measured by pump speed and with the output of a long lag circuit whose input is flux. To initiate trip for increasing reactivity disturbances, the flux signal is a negative input to the comparator. For decreasing reactivity disturbances, the flux signal is positive and the output of the lag circuit is negative. The operation of this subsystem is such that the trip point is dependent on the initial condition, rate of power change, and magnitude of power changes. For a given initial power, there is a threshold magnitude of power change to cause a trip. A step change of smaller magnitude than the threshold value will not cause a trip. Power changes greater than the threshold value will initiate a trip with a lower total power change than a slower ramp rate. The trip equation constants are adjusted to provide the necessary protection for the range of normal power conditions without significantly impairing the plant operations. Worst case values of the constants, instrument response times and repeatabilities are used in analyzing the performance of the subsystem. This subsystem is never bypassed. The negative flux rate subsystem must be bypassed for plant startup. Nuclear flux is used as a permissive signal. If nuclear flux is less than 20% of ufll power flux a bypass can be manually instated. The bypass is automatically removed when power is increased above the permissive level.

Flux- Pressure

The Flux-Pressure* Protective Subsystem (Figure 7.2-5) initiates trip for positive reactivity excursions or reductions in primary flow over the load range. Two pressure sensors are used for each redundant channel of the system. This arrangement assures appropriate redundancy while providing effective plant operational characteristics since pressure sensor replacement

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*Formerly referred to as Flux-$\sqrt{\text{Pressure}}$ subsystem.

cannot be carried out on-line. The use of the high auctioneer automatically accommodates failure of a sensor. All six pressure sensor outputs are compared in the channel output monitoring circuitry to provide early indication of anomalous performance to the operating personnel. The subsystem performance is a function of initial operating level and is analyzed using worst case values for the instrumentation and electronics including response time for the pressure instrumentation of 150 milliseconds. This subsystem must be bypassed for plant startup. Nuclear flux is used as a permissive signal. If nuclear flux is less than 10% of full power flux a bypass can be manually instated. The bypass is automatically removed when power is increased above the permissive level.

HTS Pump Frequency

The HTS Pump Frequency Subsystem (Figure 7.2-6) provides protection for loss of pumping power for two or three HTS loops. Three underfrequency relays, one on each bus, are used as redundant channels. If two of three relays are tripped, reactor trip ensues. A time delay is used to allow the plant to continue through momentary power outages.

Primary-Intermediate Pump Speed Mismatch

The Primary-Intermediate Pump Speed Mismatch Subsystems (Figure 7.2-6) initiate trip for imbalances in heat removal capability between the primary and intermediate circuits within a heat transport loop. Three subsystems are included, one for each HTS loop. As shown in the figure, the primary and intermediate speed signals are normalized and subtracted. The absolute value of this difference is compared with a fixed bias and a linear ratio of the primary speed to determine trip initiation. The actual trip point is dependent on initial conditions. Worst case values are used for analysis including a 20 millisecond tachometer time constant. These subsystems must be bypassed to start the plant. The permissive signal used is the nuclear flux. If the nuclear flux is less than 10% full power flux, the subsystem can be bypassed manually. For two loop operation, provisions are made to bypass the function associated with the shutdown loop. A manual bypass is instated under administrative control by changing the hardware configuation. Two loop bypasses are also under permissive control. Nuclear flux must be less than 10% of full power flux at the time of instating and the primary HTS pump speed in the shutdown loop must be less than 15% of the speed producing nominal full flow or the two loop bypass is automatically removed.

Reactor Vessel Level

The Reactor Vessel Level subystem (Figure 7.2-5) prevents reactor operation unless the sodium level in the reactor vessel is at least 6 inches above the suppressor plate. The output of the level sensor is compared with a fixed setpoint to determine the need for a reactor trip. Worst case values are used in the analysis of the performance of this subsystem including a sensor time constant of 0.5 second. This subsystem is never bypassed.

Steam-Feed Flow Mismatch

The Steam-Feed Flow Mismatch subsystem (Figure 7.2-7) initiate reactor trip to prevent continued operation with large imbalances between the steam and feedwater flow for each HTS loop. One of these subsystems is included in each HTS loop. These subsystems protect the steam generators and drums against unacceptable thermal transients. As shown in the figure, each subsystem compares the steam flow and feedwater flow, both of which are multiplied by appropriate constants, in two individual comparators. If the difference between the two values exceeds the setpoint in either of the comparators, a trip is initiated. Increasing steam flow and decreasing feedwater flow fault events are sensed by the first comparator. The second comparator senses decreasing steam flow and increasing feedwater flow fault events. Analysis of this function is based upon worst case parameter values. A permissive is included which allows manual bypass of this subsystem for nuclear power less than 10%.

IHX Primary Outlet Temperature

The IHX Primary Outlet Temperature Subsystems (Figure 7.2-7) compare the sodium temperature in the primary cold leg of each IHX to a fixed set point. A reactor trip is initiated if the sodium temperature exceeds this set point. These subsystems assure that temperature increases in an intermediate loop sodium resulting from steam side fault events or intermediate flow reductions do not increase the reactor coolant temperature. There is one IHX primary outlet temperature subsystem per HTS loop. These subsystems are never bypassed.

Primary Low Flux

The Primary Low Flux Subsystem (Figure 7.2-5) obtains a measurement of nuclear power from the compensated ion chambers and compares it to a fixed setpoint. If nuclear power is greater than the setpoint, a reactor trip is initiated. A permissive module allows manual bypass of this subsystem during the ascent to power operation. This subsystem protects against positive reactivity disturbances occurring during startup.

7.2.1.2.2 Secondary Reactor Shutdown System Subsystems

Modified Nuclear Rate

The Modified Nuclear Rate Subsystem (Figure 7.2-8) initiates trip for rapid sustained reactivity disturbances which occur in the load range. Two subsystems are provided. One for positive flux rates and one for negative flux rates. These subsystems prevent undesired thermal transients caused by rapid changes in power with flow held constant. The reactor trip is based on flux rate measurements from the fission counters. A permissive is included which allows manual bypass of the negative rate subsystem for nuclear power less than 10%. The positive rate subsystem is never bypassed.

Flux-Total Flow

The Flux-Total Flow Subsystem (Figure 7.2-8) provides protection against increasing and decreasing flow and power events over the 40 to 100% load range. The primary flows of the three HTS loops are summed and multiplied by an appropriate gain. A nuclear power signal obtained from the fission counters is subtracted in the comparator from the total flow value and this difference is compared to a fixed set point. If the difference exceeds the set point, then a reactor trip is initiated. Analysis of this subsystem is based on worst case parameter values, including a 500 msec. time delay for the flow detectors. This subsystem is never bypassed.

Startup Nuclear

The Startup Nuclear Subsystem (Figure 7.2-8) compares the output of wide range log mean square voltage measurements of nuclear power from the secondary fission chambers with a fixed setpoint. If nuclear power is greater than the setpoint, a reactor trip is initiated. This system provides protection against any positive reactivity disturbances, which may taken place during startup. This trip function is located in the Secondary Shutdown System, as withdrawal of the Secondary Control Rods is interlocked so as to take place prior to criticality. The Secondary System therefore provides the major protection against reactivity transients during startup. Availability of this trip function prior to startup is ensured by the following design and procedural features. First, the Startup Nuclear Subsystem utilizes a wide range fission chamber, which provides a measurable output signal when the reactor is shut down. Second, the wide range log counting electronics, the wide range mean square voltage electronics, and the wide range linear power electronics will be tested prior to startup to ensure calibrated and compatible signal outputs.

It should be noted that these features will also confirm the availability of other trip functions supplied from the secondary fission chambers; in particular, the modified positive flux rate subsystems and the flux to flow subsystems described previously.

A permissive module is provided, which allows manual bypass of the Startup Nuclear Subsystem. The permissive signal for this bypass is taken from the linear power electronics, also supplied from the secondary fission chambers.

The Startup Nuclear Subsystem also provides a means to confirm that the primary power range DC linear channels are functioning prior to reaching significant power levels. At approximately 5% reactor power, a check of the primary power range DC linear channels will be performed to ensure that they are on scale. Administrative procedures will prohibit bypassing of the Startup Nuclear Subsystem if the primary power range DC linear channels are not functioning and indicating values consistent with the wide range linear power channels.

Primary to Intermediate Flow Ratio

The Primary to Intermediate Flow Ratio Subsystems (Figure 7.2-8) protect against an imbalance in the heat removal capability of the primary and intermediate loops. The heat removal capability of a particular loop is determined by measurement of the sodium flow within the loop. The Secondary RSS includes two of these subsystems, Primary Flow High and Primary Flow Low. In the Primary Flow High Subsystem, the output of the high primary flow auctioneer is compared to the summation of the outputs from the low intermediate flow auctioneer and a signal proportional to the total primary flow. When the high primary flow auctioneer signal exceeds the low intermediate flow auctioneer signal by an amount proportional to the total primary flow, a reactor trip is initiated.

Similarly in the Primary Flow Low Subsystem, a comparison is made between low primary flow and high intermediate flow. When the high intermediate flow auctioneer signal exceeds the low primary flow auctioneer signal by an amount proportional to the total primary flow, a reactor trip is initiated. These subsystems are manually bypassed during plant startup. The permissive signal used is based on reactor power. If reactor power is less than 10%, the subsystems can be manually bypassed.

Steam Drum Level

The Steam Drum Level Subsystems (Figure 7.2-9) measure steam drum water level and compare it to two individual fixed set points. A reactor tripp is initiated whenever the drum water level decreases below this fixed setpoint. There are three of these subsystems, one per HTS loop. Analysis of these subsystems are based upon worst case parameter values. For two loop operation, a manual bypass is instated under administrative control by changing the hardware configuation. Two loop bypasses are also under permissive control. Nuclear flux must be less than 10% of full power flux at the time of instating and the primary flow in the shutdown loop must be less than 15% of full flow or the two loop bypass is automatically removed.

HTS Pump Voltage

The HTS Pump Voltage Subsystem (Figure 7.2-9) provides protection for loss of pumping power for two or three HTS loops. Three undervoltage relays, one on each HTS pump bus, are used as redundant channels. If two of the three redundant channels are tripped, reactor trip ensues. A time delay is included to allow the plant to continue operation through momentary power outages.

Evaporator Outlet Sodium Temperature

The Evaporator Outlet Sodium Temperature Subsystems (Figure 7.2-10) compare the sodium temperature at the outlet of the evaporator in each HTS loop to a fixed set point. If this temperature exceeds the set point, a reactor trip is initiated. There are three of these subsystems, one per loop. These subsystems detect a large class of events which impair the heat removal capability of the steam generators. These subsystems are never bypassed.

Sodium Water Reaction

The Sodium Water Reaction Subsystems (Figure 7.2-10) detect the occurrence of a sodium water reaction within a superheater or evaporator module. There are three of these subsystems, one per loop. Each subsystem receives nine signals from the sensors in the reaction products vent lines of a steam generator. These subsystems are never bypassed.

7.2.1.2.3 Essential Performance Requirements

In order to implement the required protective functions within the appropriate limits, RSS equipment must meet several essential performance requirements. These essential performance requirements and the RSS equipment to which they apply are summarized below.

The RSS instrumentation will meet the essential performance requirements of Table 7.2-3. This table defines the minimum accuracy and time constants which will result in acceptable performance of the RSS.

Analysis of worst case RSS functional performance is based on the values given in Table 7.2-3.

The maximum delay between the time a protective subsystem indicates the need for a trip and the time the rods are released is 0.200 second. This time includes the delays due to the calculational units, comparators, logic, scram breakers, and control rod release.

The maximum delay between the time a protective subsystem indicates the need for a trip and the time the HTS sodium pumps are tripped is 0.500 second. This time also includes the delays due to the logic and HTS scram breakers.

The RSS is designed to meet these essential performance requirements over a wide range of environmental conditions and credible single events to assure that environmental effects do not degrade the performance of the PPS. The environmental extremes are documented in Reference 13 of PSAR Section 1.6. Provisions are incorporated within the PPS which provide a defense against the following incidents:

- Environmental Changes

All electrical equipment is subject to performance degradation due to major changes in the operating environment. Where practical, PPS equipment is designed to minimize the effects of environmental changes; if not, the performance at the environmental extremes is used in the analysis.

Measures have been taken to assure that the RSS electronics are capable of performing according to their essential performance requirements under variations of temperature. The range of temperature environment specified for all the electronic equipment considered here is greater than is expected to occur during normal or abnormal conditions. Electronics do not fail catastrophically when these limits are exceeded even though this is the assumed failure mode. The detailed design of the circuit boards, board mounting and racks includes free ventilation to minimize hot spots. Ventilation is a result of natural convection air flow.

The RSS is designed to operate under or be protected from a wider range of relative humidity than that produced by normal or postulated accident conditions.

Vibration and shock are potential causes of failure in electronic components. Design measures, including the prudent location of equipment, minimize the vibration and shock experienced by RSS electronics. The equipment is qualified to shock and vibration specifications which exceed all normal and off-normal occurrences.

The RSS comparators and protective logic are designed to operate over a power source voltage range of 108 to 132 VAC and a power source frequency range of 57 to 63 HZ. The maximum variation of the source voltage is expected to be ±10%. More extreme variations in the power source may result in the affected channel comparator or logic train outputting a trip signal. In addition, testing and monitoring of RSS equipment is used, where appropriate, to warn of impending equipment degradation. Therefore, it is not expected that changes in the environment will cause total failure of an instrument channel or logic train, much less the simultaneous failure of all instrument channels or logic trains.

The majority of the RSS electronics is located in the control building, and is not subjected to a radioactive environment. Any PPS equipment located in the radioactive areas (such as the head access area) will be designed to withstand the level of activity to which it will be subjected, if its function is required.

o Tornado

The RSS is protected from the effects of the design basis tornado by locating the equipment within tornado hardened structures.

o Local Fires

All RSS equipment, including sensors, actuators, signal conditioning equipment, wiring, scram breakers, and cabinets housing this equipment is redundant and separated. These characteristics make any credible fire of no consequence to the safety of the plant. The separation of the redundant components increases the time required for fire to cause extensive damage and also allows time for the fire to be brought to the attention of the operator such that corrective action may be initiated. Fire protection systems are also provided as discussed in Section 9.13.

o Local Explosions and Missiles

All RSS equipment essential for reactor trip is redundant. Physical separation (distance or mechanical barriers) and electrical isolation exists between redundant components. This physical separation of redundant components minimized the possibility of a local explosion or missile damaging more than one redundant component. The remaining redundant components are still capable of performing the required protective functions.

o Earthquakes

All RSS equipment, including sensors, actuators, signal conditioning equipment, wiring, scram breakers and structures (e.g., cabinets) housing such equipment, is classed as Seismic Category I. As such, all RSS equipment is designed to remain functional under OBE and SSE conditions. The characteristics of the OBE and SSE used for the evaluation of the RSS are found in Section 3.7.

Information Read-Out

Indicators and alarms are provided as an operating aid and to keep the plant operator informed of the status of the RSS. Except for the IHX primary outlet temperature analog indicators which are part of the accident monitoring system, all indicators and alarms are not safety related. The following items are located on the Main Control Panel for operator information:

Analog Indication

A. Secondary Wide Range Log MSV Power Level
B. Secondary Wide Range Linear Power Level
C. Primary Power Range Power Level
D. Reactor Vessel Level
E. HTS Pump Speeds

F. HTS Loop Flows
G. Reactor Inlet Pressure
H. IHX Primary Outlet Temperature
I. Evaporator Outlet Temperature
J. Steam Flows
K. Feedwater Flows
L. Steam Drum Level

Indicating Lights

A. Instrument Channel Bypass Permissive Status
B. Instrument Channel Bypass Status
C. Logic Train Trip/Reset Status
D. HTS Loop Trip/Reset Status
E. HTS Loop Test Status

Annunciators

A. Instrument Channel Trip/Reset information is provided for each function listed in Table 7.2-1.
B. Logic Train Power Supply Failure

Information is also available to the operator via the Plant Data Handling and Display System.

7.2.1.2.4 Reactor Shutdown System Power Supplies

The Primary and Secondary Reactor Shutdown Systems are powered from the three Class 1E 120/208 volt Vital (Uninterruptable) AC Power Systems. Redundant channels within each shutdown system are powered from separate independent load groups with one channel/logic train from each system connected to the same vital AC power system. This commonality between one set of redundant channels/logic trains is not considered to impact their independence because of the following design features:

o The design of the Vital AC Power System assures independence between the three redundant power divisions such that failures within a power division load group will not propagate to a redundant load group.

o Loss of one vital power division will result in tripping one logic train in each reactor shutdown system.

o Provision of isolation devices in the individual power supplies within the two reactor shutdown systems will prevent any circuit failure in one redundant channel/logic train of one system from affecting the proper safety function of the other system.

o Features will be provided within the Primary and Secondary systems to accommodate electrical surges from the AC vital power source without loss of safety function in either system.

An analysis will be performed to identify any transients which could originate from within one of the reactor shutdown systems and be coupled through the AC vital distribution bus to the other shutdown system. If significant transients are identified by this analysis, then a test program will be defined to confirm the continued operation of the remaining reactor shutdown equipment connected to the same vital power source. A description of any such test program will be provided prior to the application for an Operating License.

7.2.2 Analysis

The Reactor Shutdown System meets the safety related channel performance and reliability requirements of the NRC General Design Criteria, IEEE Standard 279-1971, applicable NRC Regulatory Guides and other appropriate criteria and standards.

The RSS Logic is designed to conform to the IEEE Standards listed in Table 7.2-4.

General Functional Requirement

The Plant Protection System is designed to automatically initiate appropriate protective action to prevent unacceptable plant or component damage or the release or spread of radioactive materials.

Single Failure

No single failure within the Reactor Shutdown System nor removal from service of any component or channel will prevent protective action when required.

Two independent, diverse reactor shutdown systems are provided, either of which is capable of terminating all excursions without allowing plant parameters to exceed specified limits. Each system uses three redundant instrument channels and logic trains. The Primary RSS is configured using local coincidence logic while the Secondary RSS uses general coincidence logic. To provide further assurance against potential degradation of protection due to credible single events, functional and/or equipment diversity are included in the hardware design.

The DC and AC Uninterruptible Power Supplies (UPS) to the redundant instrument channels and logic trains are provided from three respective redundant power divisions. The three divisions are physically and electrically independent such that loss of any one division will not prevent the other divisions from performing their safety function. The design of power supply equipment (inverters and battery chargers), which use solid state components, is such that it precludes the possibility of a fault in one power division to have any adverse affect on similar power supply of the other two divisions.

The inverters will be tested to demonstrate that a transient on the inverter output will have no affect on the input power supplies. Testing will be performed in accordance with ISA and ANSI C37.90.

Bypasses

Bypasses for normal operation require manual instating. Bypasses will be automatically removed whenever the subsystem is needed to provide protection. The equipment used to provide this action is part of the RSS. Administrative procedures are used to assure correct use of bypasses for infrequent operations such as two loop operation. If the protective action of some part of the system has been bypassed or deliberately rendered inoperative, this fact will be continuously indicated in the control room.

Multiple Setpoints

Where it is necessary to change to a more restrictive setpoint to provide adequate protection for a particular normal mode of operation or set of operating conditions, the RSS design will provide automatic means of assuring that the more restrictive setpoint is used. Administrative procedures assure proper setpoints for infrequent operations.

For CRBRP, power operation on two-loops will be an infrequent occurrence, and will only be initiated from a shutdown condition. While the reactor is shutdown, the RSS equipment will be aligned for two-loop operation which will include set down of the appropriate trip points. Sufficient trip point set down is being designed into the RSS equipment to adequately cover the possible range (conceptually from 2% to 100%) of trip point adjustment required. In addition, administrative procedures (specifically the pre-critical checkoff) will be invoked during startup to ensure that the proper RSS trip points have been set.

The analysis of plant performance during two-loop operation has not been completed to date. Therefore, the exact trip point settings for two-loop operation cannot be specified at this time. However, the range of trip point settings indicated above is adequate to ensure that trip points appropriate for the anticipated lowest two-loop operating power can be achieved.

In summary, the design of the RSS equipment trip point adjustments and other features for two-loop operation coupled with the anticipated two-loop operating power level and administrative procedures assure full compliance with Branch Technical Position EICSB 12 and satisfy Section 4.15 of IEEE std 279-1971.

Completion of Protective Action

The Reactor Shutdown Systems are designed so that, once initiated, a protective action at the system level must go to completion. Return to normal operation requires manual reset by the operator because the Primary RSS scram breakers or Secondary scram latch circuitry must be manually closed following trip. Trip signals must be cleared prior to closure of scram breakers.

Manual Initiation

The Reactor Shutdown System includes means for manual initiation of each protective action at the system level with no single failure preventing initiation of the protective action. Manual initiation depends upon the operation of a minimum of equipment because the manual trip directly operates the scram breakers of the solenoid scram valve power supply.

Access

Administrative control of access to all setpoint adjustments, module calibration adjustments, test points and the means for establishing a bypass permissive condition is provided by locking cabinets and other access design features of the control room and the equipment racks.

Information Read-Out

Indicators and alarms are provided as an operating aid and to keep the plant operator informed of the status of the RSS. Except for the IHX primary outlet temperature analog indicators which are part of the accident monitoring system, all indicators and alarms are not safety-related. The following items are located on the Main Control Panel for operator information.

Analog Indication

A. Secondary Wide Range Log MSV Power Level
B. Secondary Wide Range Linear Power Level
C. Primary Power Range Power Level
D. Reactor Vessel Level
E. HTS Pump Speeds
F. HTS Loop Flows
G. Reactor Inlet Pressure
H. IHX Primary Outlet Temperature
I. Evaporator Outlet Temperature
J. Steam Flows
K. Feedwater Flows
L. Steam Drum Level

Indicating Lights

A. Instrument Channel Bypass Permissive Status
B. Instrument Channel Bypass Status
C. Logic Train Trip/Reset Status
D. HTS Loop Trip/Reset Status
E. HTS Loop Test Status

Annunciators

A. Instrument Channel Trip/Reset information is provided for each function listed in Table 7.2-1
B. Logic Train Power Supply Failure
C. Two Loop Bypasses Instated

Most information is also available to the operator via the Plant Data Handling and Display System.

Annunciator for RSS Channel Trips

A visual and audible indication of all channel trip conditions within the RSS will be provided in the control room. These alarm conditions include any tripped RSS comparators in the Primary RSS or Secondary RSS. The Plant Data

Handling and Display system alerts the operator to significant deviations between redundant RSS analog instrumentation used to monitor a reactor or plant parameter for the RSS.

Control and Protection System Interaction

The Reactor Shutdown System and the Plant Control System have been designed to assure stable reactor plant operation and to protect the reactor plant in the event of worst case postulated Plant Control System failures. The RSS is designed to protect the plant regardless of control system action or lack of action. Isolation devices will be used between protection and control functions. Where this is done, all equipment common to both the protection and control function is classified as part of the RSS. Equipment sharing between protection and control is minimized. Where practical, separate equipment (sensors, signal conditioning, cabling penetrations, raceways, cabinets, monitoring etc.) is provided. The sharing of components does not lead to a situation where a single event both initiates an incident through Plant Control System malfunctions and prevents reactor shutdown system action. Where such a malfunction in a shared component requires action by the shutdown systems this will take place with any resulting core transient no greater than that associated with the normal operation of the shutdown systems. Also, this performance will not be affected by the removal from service of a shutdown system channel for testing and the simultaneous presence of a single random failure in the shutdown system in addition to the malfunction which initiates the transient (both are assumed to occur in the particular shutdown system affected by the malfunction).

Median select circuits are used by the plant control systems for those instrumentation channels common to both protection and control systems. This allows continued operation of the control systems during testing of the reactor shutdown systems. These circuits will be seismically qualified for an operational basis earthquake.

Periodic Testing

The Reactor Shutdown System is designed to permit periodic testing of its functioning including actuation devices during reactor operation. In the Primary RSS, a single instrument channel is tested by inserting a test signal at the sensor transmitter and verifying it at the comparator output. A logic train is tested by inserting a very short test signal in 2 comparator inputs and verifying that the voltage on the scram breaker trip coils decrease. Because of the time response of the undervoltage relay coils of the scram breakers and very short duration of the test signal, the reactor does not trip. In the Secondary RSS, an instrument channel can be tested from sensor to scram actuator by inserting a single test signal because of the general coincidence configuration of the 3 redundant channels. The primary and secondary rod actuators cannot be tested during reactor operation since dropping a single control rod will initiate a reactor scram. Scram actuators and control rod drop will be tested and maintained when the plant is shutdown (See Section 7.1-2). Whenever the ability of a protective channel to respond to an accident signal is bypassed such as for testing or maintenance, the channel being tested is placed in the tripped state and its tripped condition is automatically indicated in the control room.

Failure Modes and Effects Analysis

A Failure Modes and Effects Analysis (FMEA) has been conducted to identify, analyze and document the possible failure modes within the Reactor Shutdown System and the effects of such failures on system performance (see Appendix C, Supplement 1). Components of the RSS analyzed are:

- Reactor Vessel Sodium Level Input
- RSS Sodium Flow Input
- Pump Electric Power Sensor
- Compensated Ion Chamber Nuclear Input
- Fission Chamber Nuclear Input
- Primary Loop Inlet Plenum Pressure Input
- Sodium Pump Speed (Primary and Intermediate)
- Steam Mass Flow Rate Input
- Feedwater Mass Flow Rate Input
- Steam Drum Level Input
- Primary Comparator
- Secondary Comparator
- Primary Logic Train
- Secondary Logic Train
- Primary Calculational Unit
- Secondary Calculational Unit

- Scram Actuator Logic
- Heat Transport System (HTS) Shutdown Logic
- Control Rod Drive Mechanism (CRDM) Power Train
- RSS Isolation Buffer

Figures 7.2-3 and 7.2-4 provide assistance in locating the above system level components within the overall RSS.

The probability of occurrence of each failure mode is listed in the tables of Appendix C, Supplement 1, in the Probability Column. The effects of each potential failure mode have also been categorized in the tables in the Criticality Column. Even though the failure of an individual element may result in the inability to initiate channel trip, the provision of redundant independent instrument channels and logic trains assures that single random failures cannot cause loss of either the Primary or Secondary RSS thereby meeting the design requirements of IEEE 279-1971. The high reliability of components, redundant configuration, provision for on-line monitoring and on-line periodic testing further assure that random failures will not accumulate to the point that trip initiation by either Primary or Secondary RSS is prevented. All failure effects are therefore categorized as not causing any degradation or failure of a system safety function. The majority of the identified failure modes can be eliminated from consideration based on their low probability of occurrence and the insignificance of their criticality. They are included in the FMEA, however, to document their consideration.

TABLE 7.2-1

REACTOR SHUTDOWN SYSTEM PROTECTIVE FUNCTIONS

Primary Reactor Shutdown System	Number of Inputs[1]
o Flux-Delayed Flux (Positive and Negative)	2
o Flux-Pressure	1
o High Flux	1
o Primary Low Flux	1
o Primary to Intermediate Speed Mismatch	3
o HTS Pump Frequency	1
o Pump Electrics	1
o Reactor Vessel Level	1
o Steam-Feedwater Flow Mismatch	3
o IHX Primary Outlet Temperature	3

Secondary Reactor Shutdown System	Number of Inputs
o Modified Nuclear Rate (Positive and Negative)	2
o Flux-Total Flow	1
o Startup Nuclear Flux	1
o Primary to Intermediate Flow Mismatch	2
o Steam Drum Level	3
o Evaporator Outlet Sodium Temperature	3
o HTS Pump Voltage	1
o Sodium Water Reaction	3

[1] The Primary RSS can accept a total of 24 inputs and the Secondary RSS can accept 16 inputs. There are 7 spare Primary inputs.

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TABLE 7.2-2

RSS DESIGN BASIS FAULT EVENTS

Fault Events	Primary Reactor Shutdown System	Secondary Reactor Shutdown System
I. Anticipated Faults		
A. Reactivity Disturbances(1)		
Positive Ramps $\leq 5¢$/sec and Steps ≤ 10		
Startup	Primary Low Flux	Startup Nuclear
25-40% Power	Flux-Delayed Flux or Flux- Pressure	Modified Nuclear Rate or Flux-Total Flow
40-100% Power	Flux- Pressure	Flux-Total Flow
Full Power	High Flux	Flux-Total Flow
Negative Ramps and Steps	Flux-Delayed Flux	Modified Nuclear Rate
B. Sodium Flow Disturbances		
Coastdown of a Single Primary or Intermediate Pump	Primary-Intermediate Speed Mismatch	Primary-Intermediate Flow Ratio
Loss of 1 HTS Loop	Flux-Pressure	Primary-Intermediate Flow Ratio
Loss of 3 HTS Loops	HTS Pump Frequency	Flux-Total Flow

TABLE 7.2-2 (Continued)

	Fault Events	Primary Reactor Shutdown System	Secondary Reactor Shutdown System
57			
	C. Steam Side Disturbances		
	Evaporator Module Isolation Valve Closure	IHX Primary Outlet Temperature	Evaporator Outlet Na Temperature
	Superheater Module Isolation Valve Closure	Steam-Feedwater Flow Mismatch	Evaporator Outlet Na Temperature
	Water Side Isolation and Dump of Single Evaporator	IHX Primary Outlet Temperature	Evaporator Outlet Na Temperature
	Water Side Isolation and Dump of Single Superheater	Steam-Feedwater Flow Mismatch	Evaporator Outlet Na Temperature
	Water Side Isolation and Dump of Both Evaporators and Superheater	Steam-Feedwater Flow Mismatch	Evaporator Outlet Na Temperature
	Loss of Normal Feedwater	Steam-Feedwater Flow Mismatch	Steam Drum Level
47	Turbine Trip with Reactor Trip (Loss of Main Condenser or Similar Problem)	Steam-Feedwater Flow Mismatch	Steam Drum Level
	Inadvertent Opening of Evaporator Outlet Safety Valve	Steam-Feedwater Flow Mismatch	Steam Drum Level
	Inadvertent Opening of Superheater Outlet Safety Valve	Steam-Feedwater Flow Mismatch	Steam Drum Level
	Inadvertent Opening of Evaporator Inlet Dump Valve	IHX Primary Outlet Temperature	Evaporator Outlet Na Temperature

TABLE 7.2-2 (Continued)

Fault Events	Primary Reactor Shutdown System	Secondary Reactor Shutdown System
II. Unlikely Faults		
A. Reactivity Disturbances[2]		
Positive Ramps ≤35¢/sec and Steps ≤60		
Startup	Primary Low Flux	Startup Nuclear
25-40% Power	Flux-Delayed Flux or Flux- Pressure	Modified Nuclear Rate or Flux-Total Flow
40-100% Power	Flux-Pressure	Flux-Total Flow
Full Power	High Flux	Flux-Total Flow
B. Sodium Flow Disturbances		
Primary Pump Seizure	Primary-Intermediate Speed Mismatch	Primary-Intermediate Flow Ratio
Intermediate Pump Seizure	Primary-Intermediate Speed Mismatch	Primary-Intermediate Flow Ratio
Loss of 2 HTS Loops	HTS Pump Frequency	Primary-Intermediate Flow Ratio
C. Steam Side Disturbances[3]		
Steam Line Break	Steam-Feedwater Flow Mismatch	Evaporator Outlet Na Temperature
Recirculation Line Break	Steam-Feedwater Flow Mismatch	Steam Drum Level
Feedwater Line Break[3]	Steam-Feedwater Flow Mismatch	Steam Drum Level

Fault Events	Primary Reactor Shutdown System	Secondary Reactor Shutdown System
Failure of Steam Dump System	Steam-Feedwater Flow Mismatch	Steam Drum Level
Sodium Water Reaction in Steam Generator(3)	Steam-Feedwater Flow Mismatch	Sodium-Water Reaction
III. Extremely Unlikely		
A. Reactivity Disturbances		
Positive Ramps ≤$2.0/sec		
Startup	Primary Low Flux	Startup Nuclear
25-40% Power	Flux-Delayed Flux or Flux- Pressure	Modified Nuclear Rate or Flux-Total Flow
40-100% Power	Flux- Pressure	Flux-Total Flow
Full Power	High Flux	Flux-Total Flow
SSE	HTS Pump Frequency	HTS Pump Voltage
B. Sodium Flow Disturbances		
PHTS Leak	Manual Trip(4)	Manual Trip
IHTS Leak	IHX Primary Outlet Temperature	Primary to Intermediate Flow Ratio

(1) The maximum anticipated reactivity fault results from a single failure of the control system with a maximum insertion rate of approximately 4.1 cents per second.

(2) The maximum unlikely reactivity faults result from multiple control system failures leading to withdrawal of six rods at normal speed or one rod at the maximum mechanical speed.

(3) The PPS is required to terminate the results of these extremely unlikely events within the umbrella transient specified as emergency for the design of the major components.

(4) No automatic PPS protection is required for the DBE PHTS leak of 8gpm since the required response time is significantly greater than 30 minutes and safety related information systems are provided to inform the Operator of the presence of a PHTS leak. For additional protection (margin), a reactor sodium level trip subsystem is included in the Primary RSS to provide protection for PHTS leaks beyond the design basis.

TABLE 7.2-3

ESSENTIAL PERFORMANCE REQUIREMENTS FOR RSS INSTRUMENTATION CHANNELS*

Plant Parameter	Accuracy (% of span)	Response Time (msec)
Neutron Flux		
Primary	±1.0	<10
Secondary - Wide range linear DC	±1.0	<10
- wide range logarithmic (MSV)	±1.0(1)	<50
Reactor Inlet Plenum Pressure	±2.0	<150
Sodium HTS Pump Speeds	±2.0	<20
Sodium HTS Flow	±5.0	<500
Reactor Vessel Sodium Level	±5.0	<500
Undervoltage Relay	±1.0	<230
Steam Flow	±2.0	<500
Feedwater Flow	±2.5	<500
Evaporator Outlet Sodium Temperature	±2.0	<5000
Steam Drum Level	±1.0	<1000
IHX Primary Outlet Temperature	±2.0	<5000
Underfrequency Relay	±2.0	<200

* Note that these accuracy and response times relate to the performance of the instrumentation channels from the sensors up to the signal conditioning output.

In addition, as noted in Section 7.2.1.2.3, the reactor shutdown system logic, actuators and rod unlatch features require a further response time delay of 200 msecs.

(1) Equivalent linear full scale output.

TABLE 7.2-4

LIST OF IEEE STANDARDS APPLICABLE TO
THE REACTOR SHUTDOWN SYSTEM LOGIC (1)

IEEE 279-1971	IEEE Standard: Criteria for Protection Systems for Nuclear Power Generating Stations
IEEE 308-1974	Criteria for Class 1E Power Systems for Nuclear Power Generating Stations
IEEE 317-1976	Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations
IEEE 323-1974	IEEE Trial-Use Standard: General Guide for Qualifying Class 1E Electric Equipment for Nuclear Power Generating Stations
IEEE 323-A-1975	Supplement to the Foreward of IEEE 323-1974
IEEE 336-1971	IEEE Standard: Installation, Inspection and Testing Requirements for Instrumentation and Electric Equipment During Construction of Nuclear Power Generating Stations
IEEE 338-1977	IEEE Trial Use Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems
IEEE-344-1975	IEEE Standard 344-1975, IEEE Recommended Practices for Seismic Qualification of Class 1 Equipment for Nuclear Power Generating Stations
IEEE 352-1975	IEEE Guide for General Principles for Reliability Analysis of Nuclear Power Generating Station Protection Systems
IEEE 379-1972	IEEE Trial-Use Guide for the Applicaton of the Single Failure Criterion to Nuclear Power Generating Station Protection Systems
IEEE 384-1974	IEEE Trial Use Standard Criteria for Separation of Class 1E Equipment and Circuits
IEEE 494-1974	IEEE Standard Method for Identification of Documents Related to Class 1E Equipment and Systems for Nuclear Power Generating Station

(1) IEEE Standards applicable to the instrumentation and monitoring systems are listed in Section 7.5.

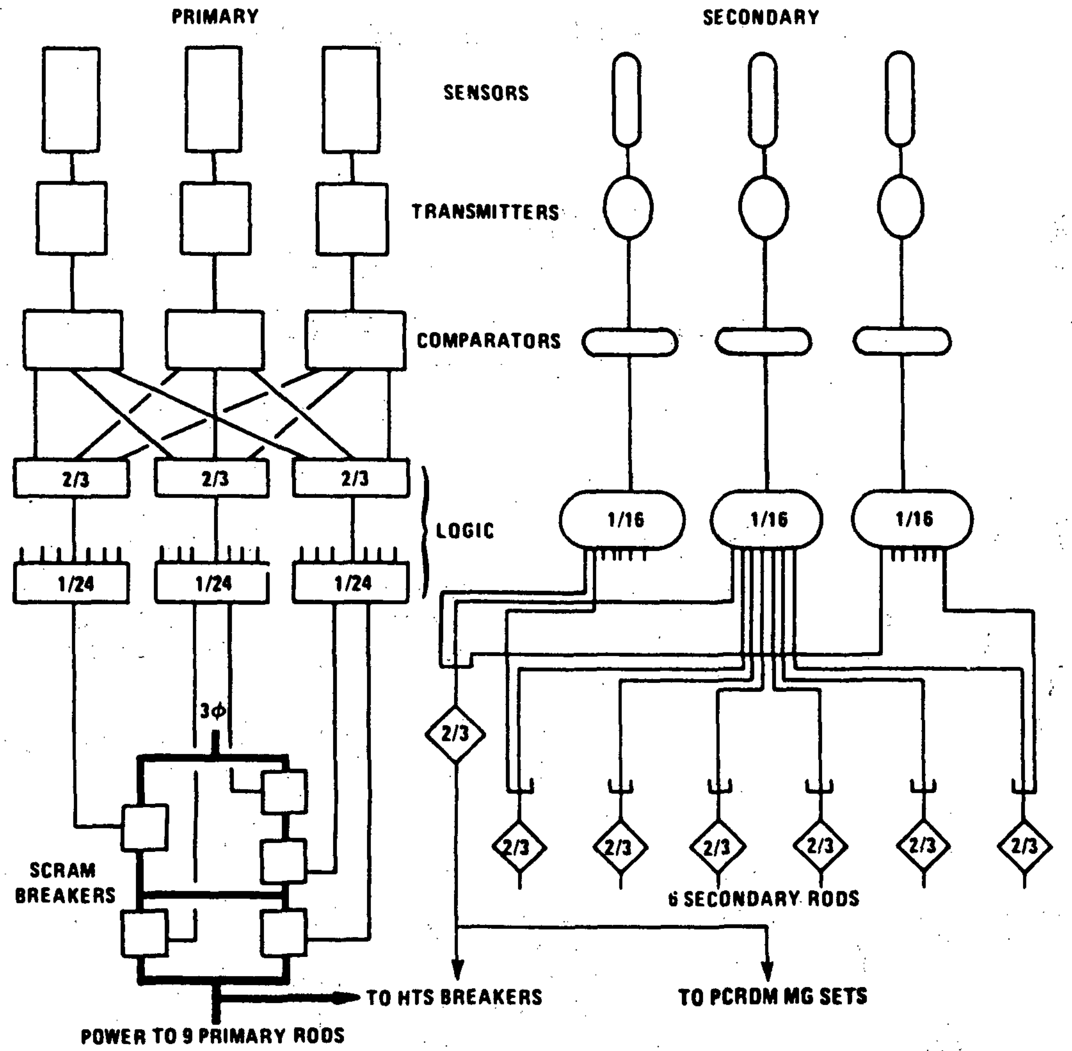

FIGURE 7.2-1. REACTOR SHUTDOWN SYSTEM

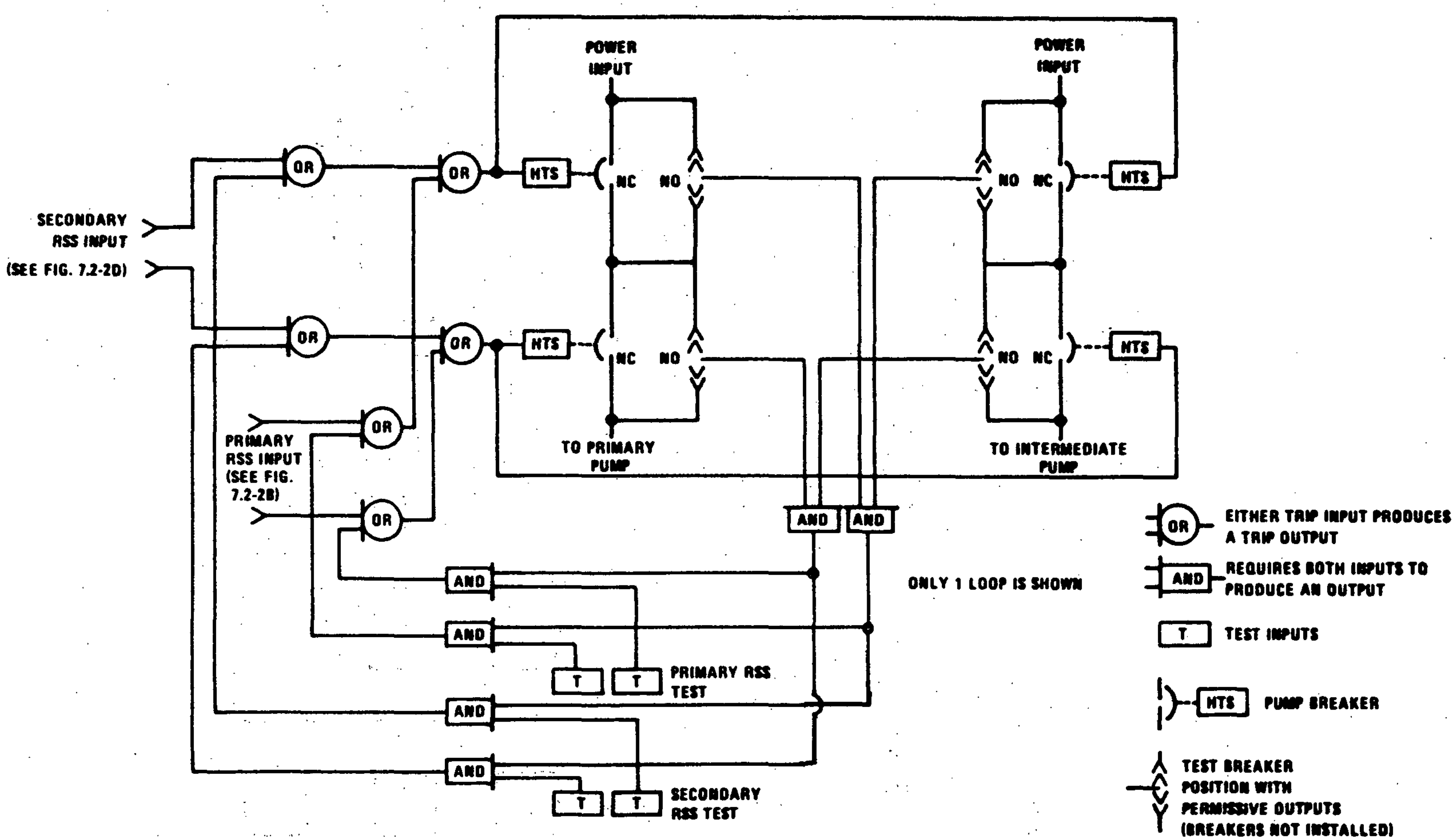

Figure 7.2-2. HTS Pump Breaker Logic Diagram

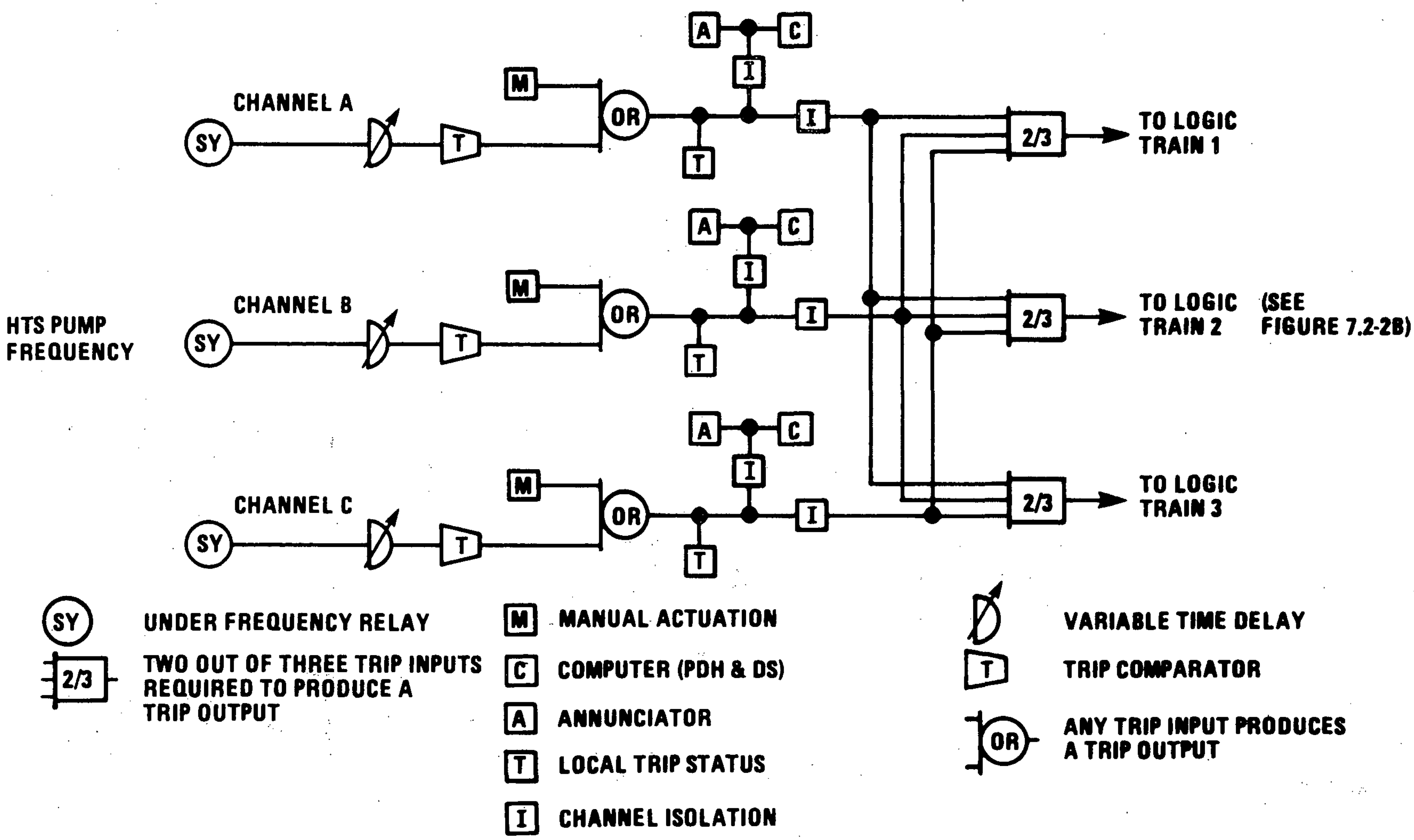

Figure 7.2-2A. Typical Primary RSS Instrument Channel Logic Diagram (HTS Pump Frequency Subsystem Shown)

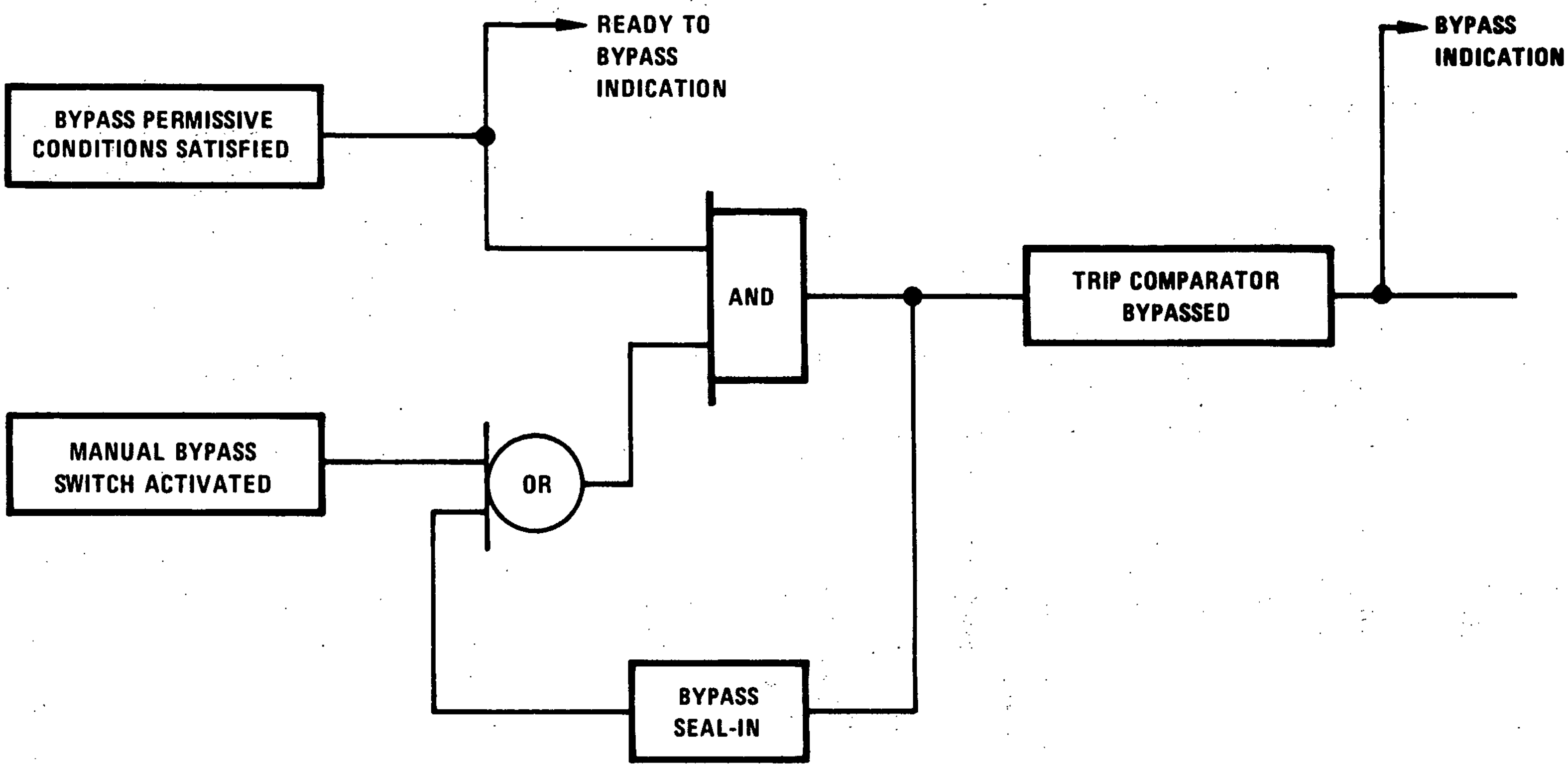

Figure 7.2-2AA. RSS Bypass Function Block Diagram

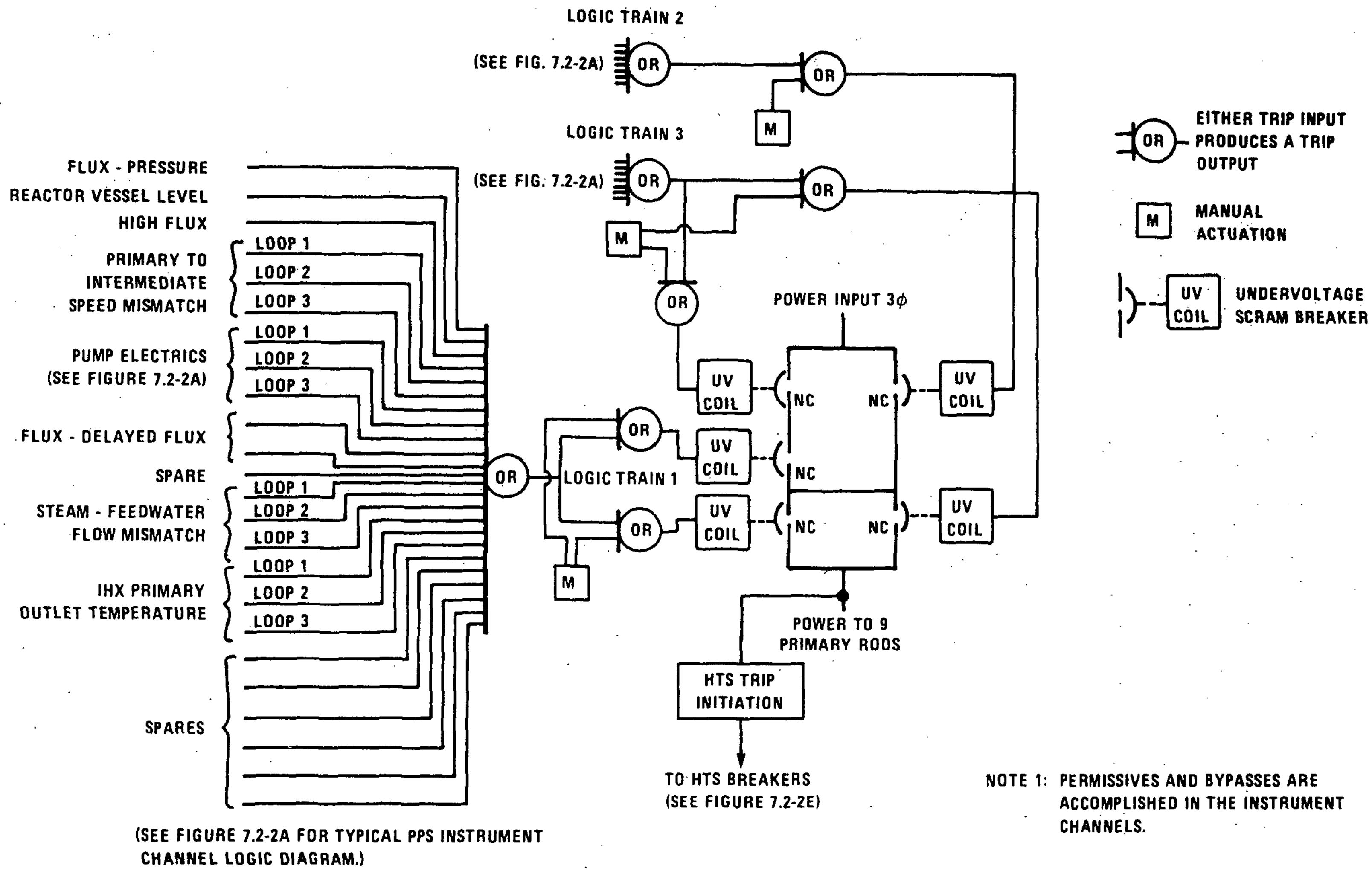

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Figure 7.2-2B. Primary RSS Final Actuation Logic Diagram

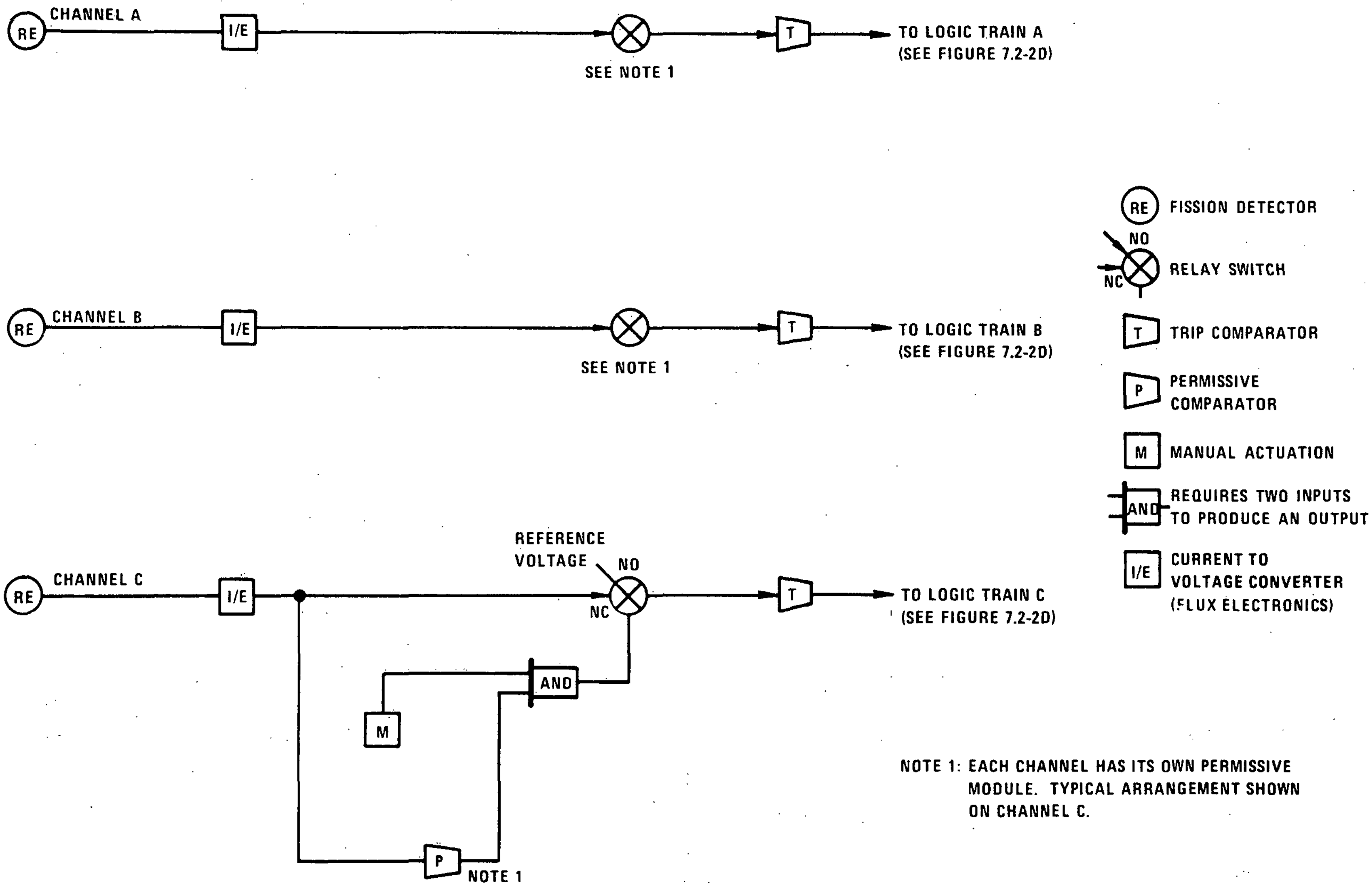


Figure 7.2-2C. Typical Secondary P.P.S. Instrument Channel Logic Diagram. (Startup Nuclear Shown).

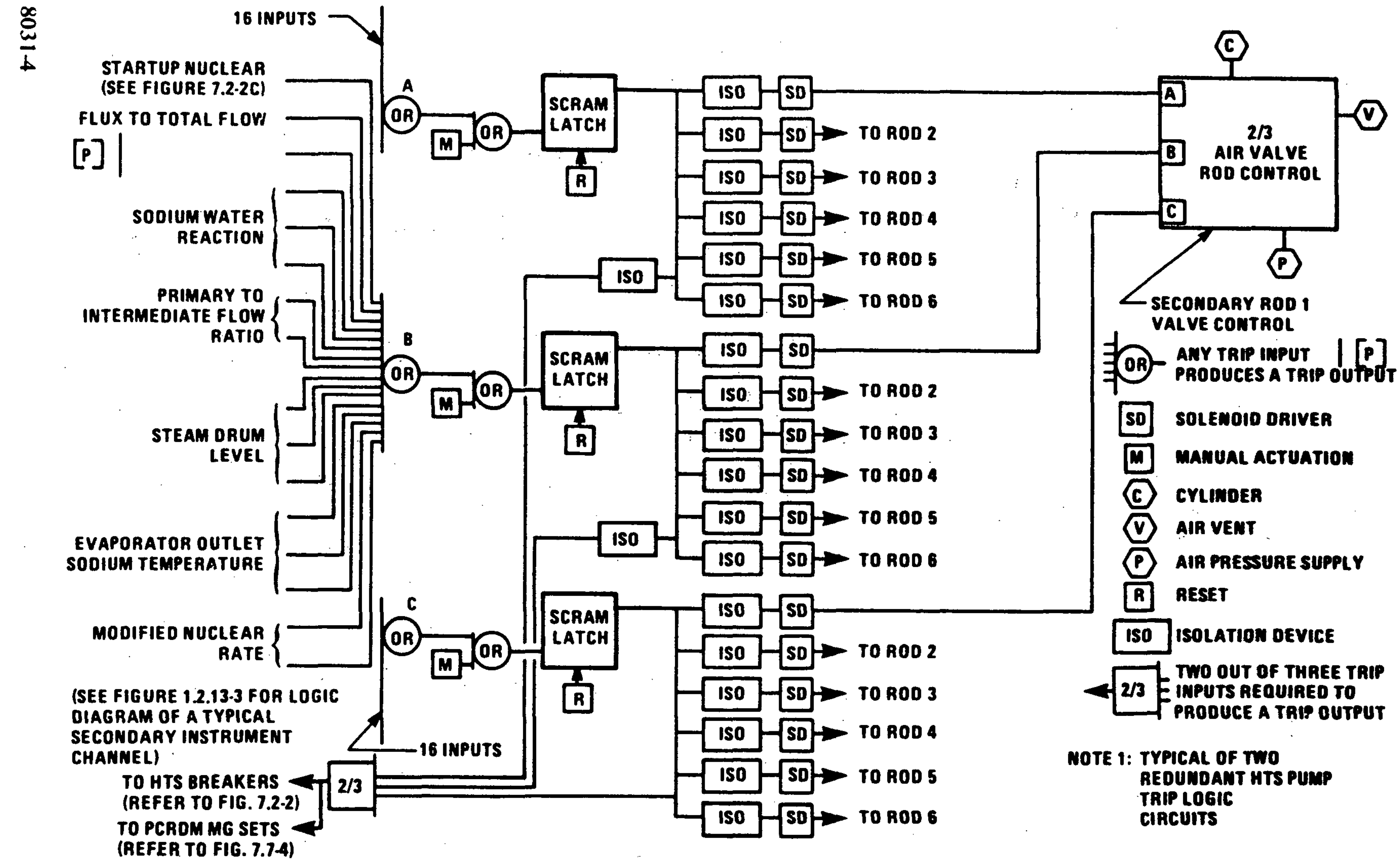


FIGURE 7.2-2D. SECONDARY RSS FINAL ACTUATION LOGIC DIAGRAM

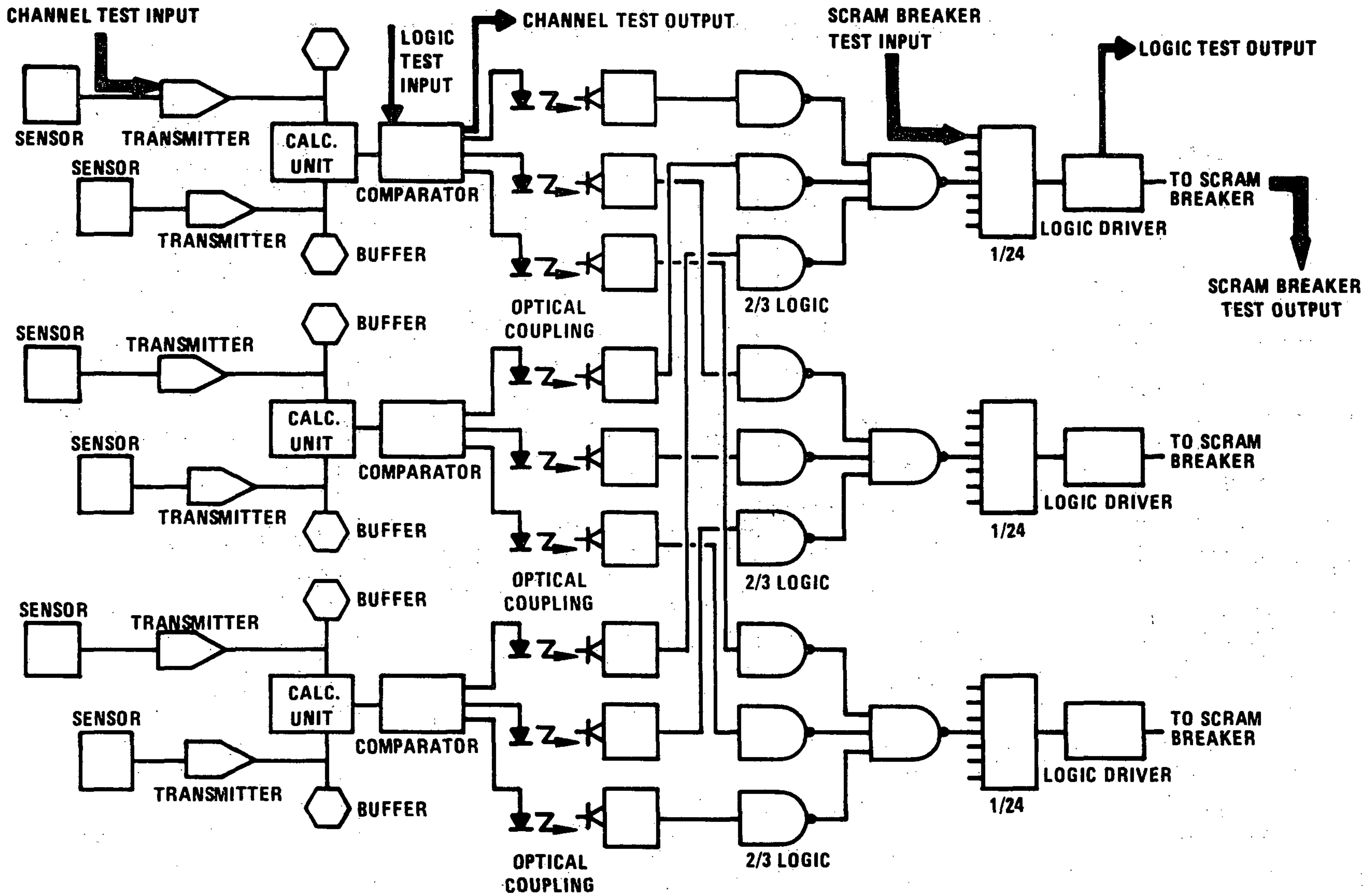


Figure 7.2-3. Typical Primary RSS Subsystem

4558-9

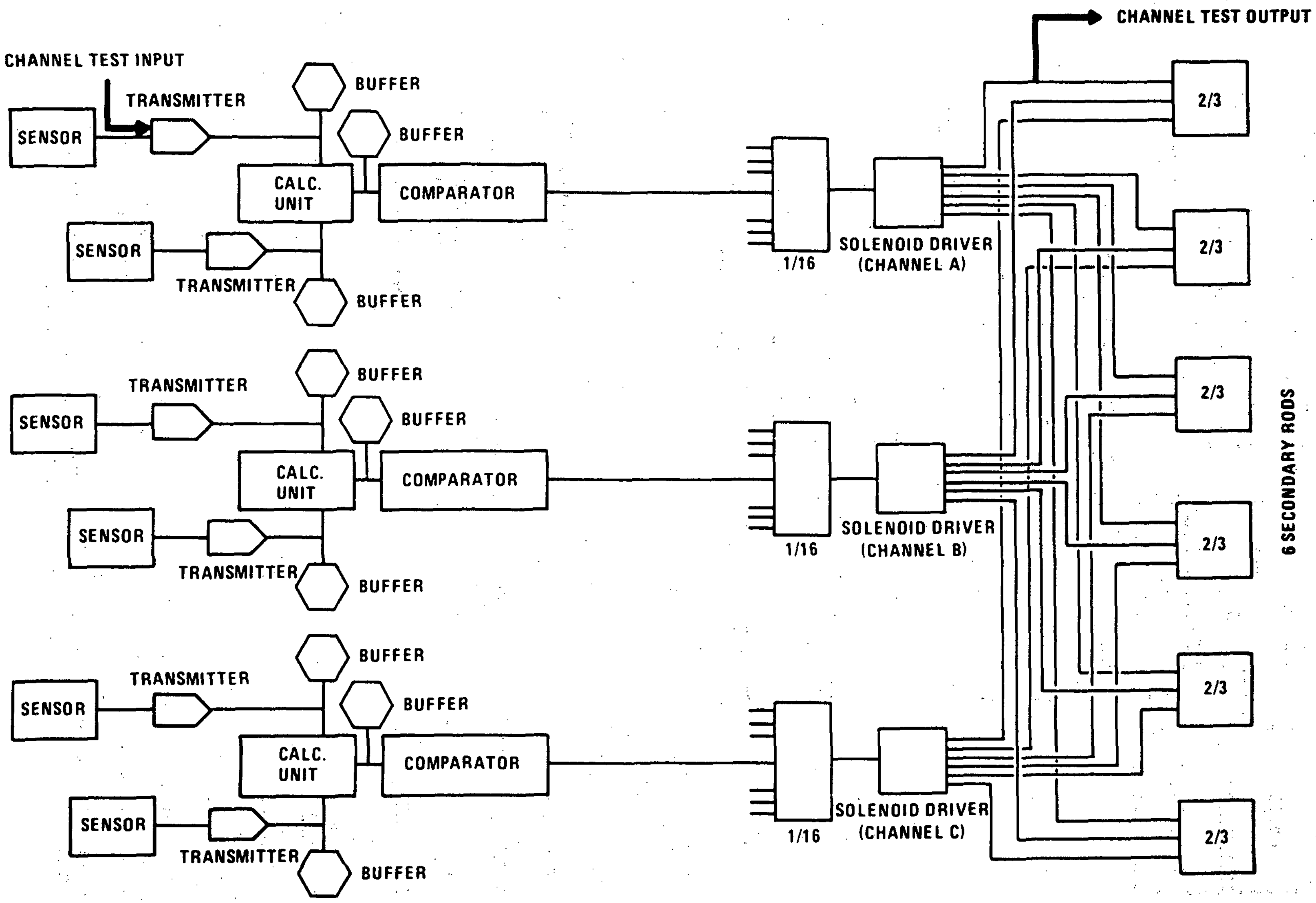


Figure 7.2-4. Typical Secondary RSS Subsystem

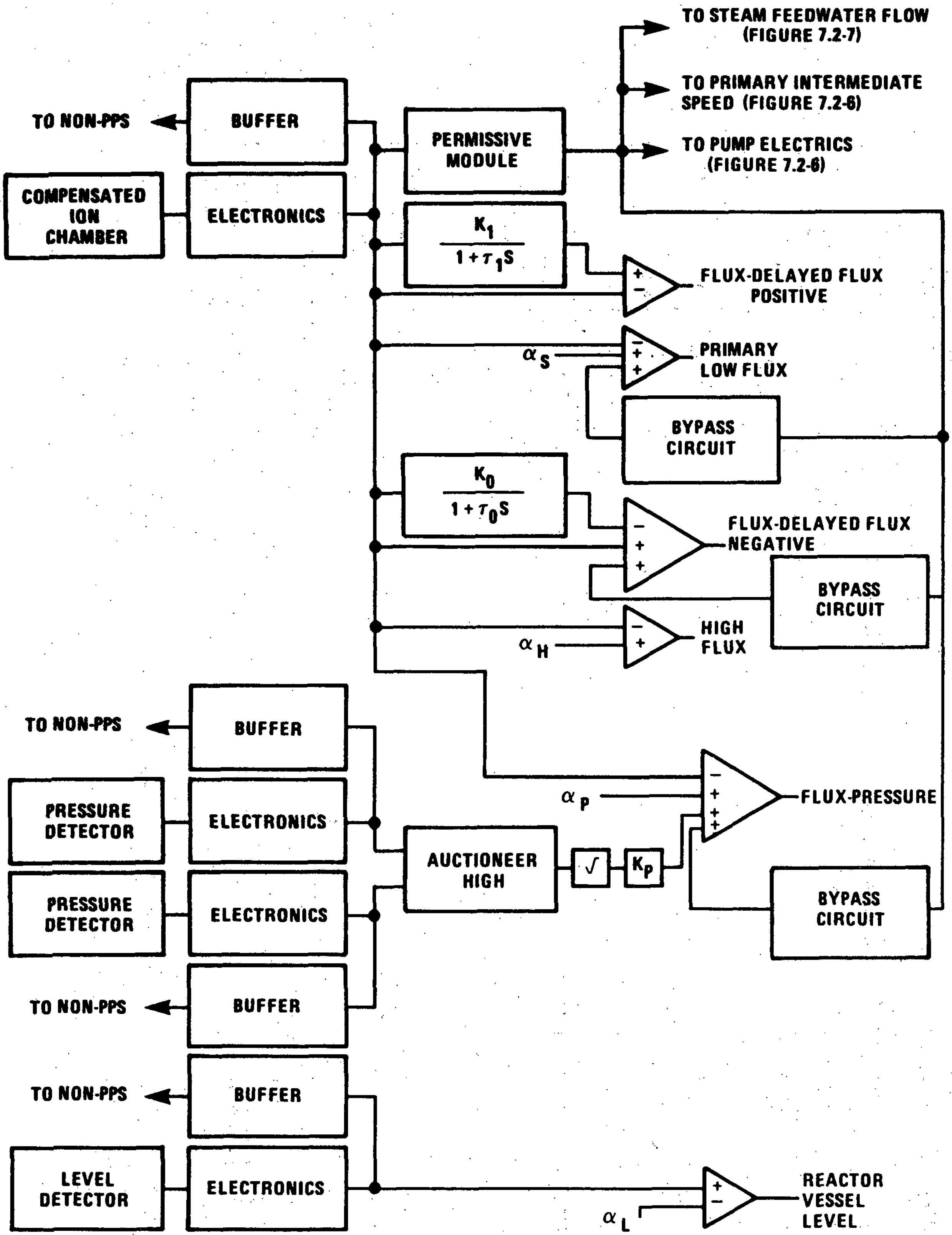


Figure 7.2-5. Functional Block Diagrams of the Flux-Delayed Flux, High Flux, Flux-Pressure, and Reactor Vessel Level Protective Subsystems, One Channel of Three Shown

7636-1

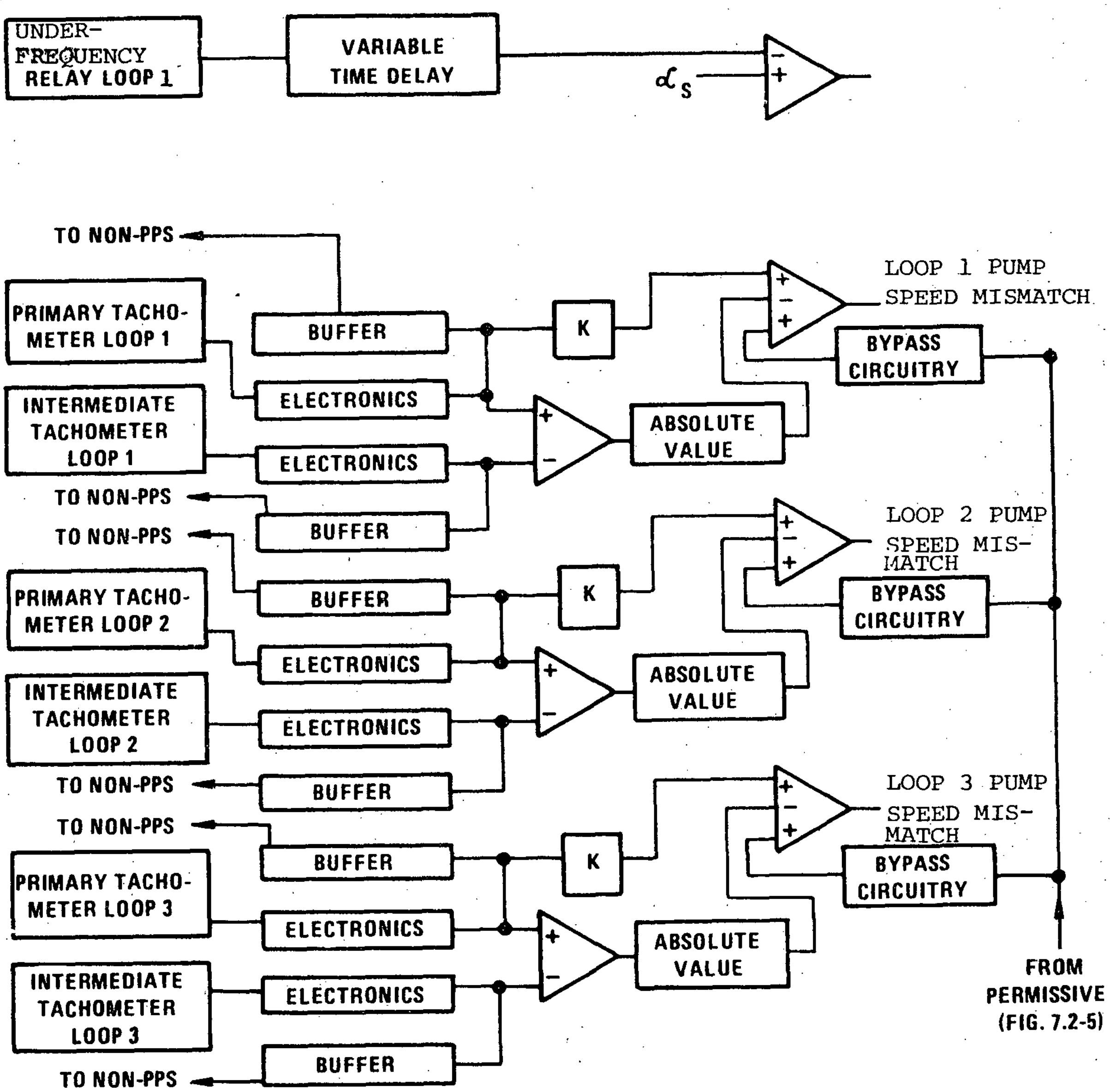


Figure 7.2-6. Functional Block Diagrams of the HTS Pump Frequency and Pump Speed Mismatch Protective Subsystems. One Channel of Three is Shown

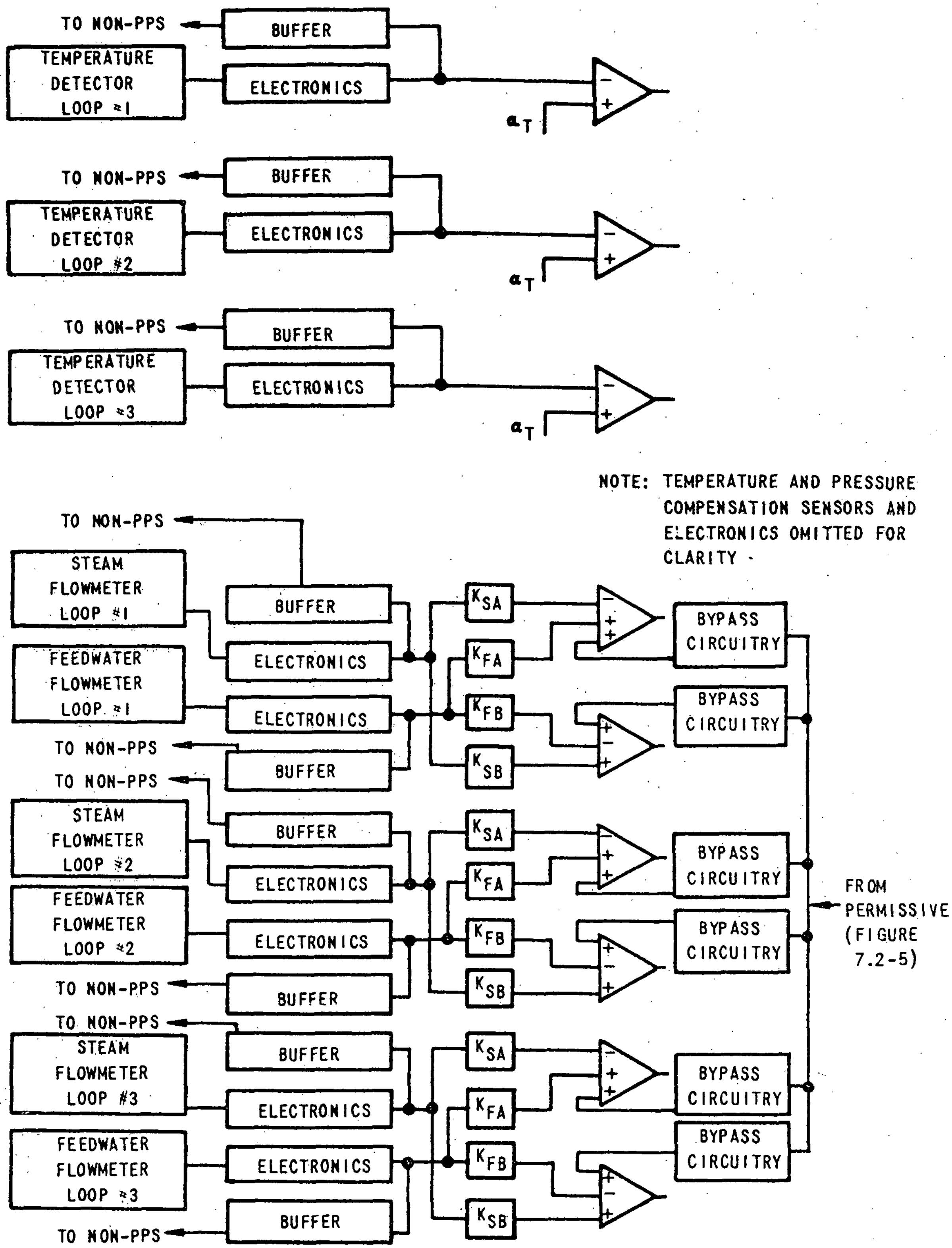


Figure 7.2-7. Functional Block Diagrams of the IHX Primary Outlet Temperature and Steam to Feedwater Flow Mismatch Protective Subsystem, One Channel of Three is Shown

6678-3

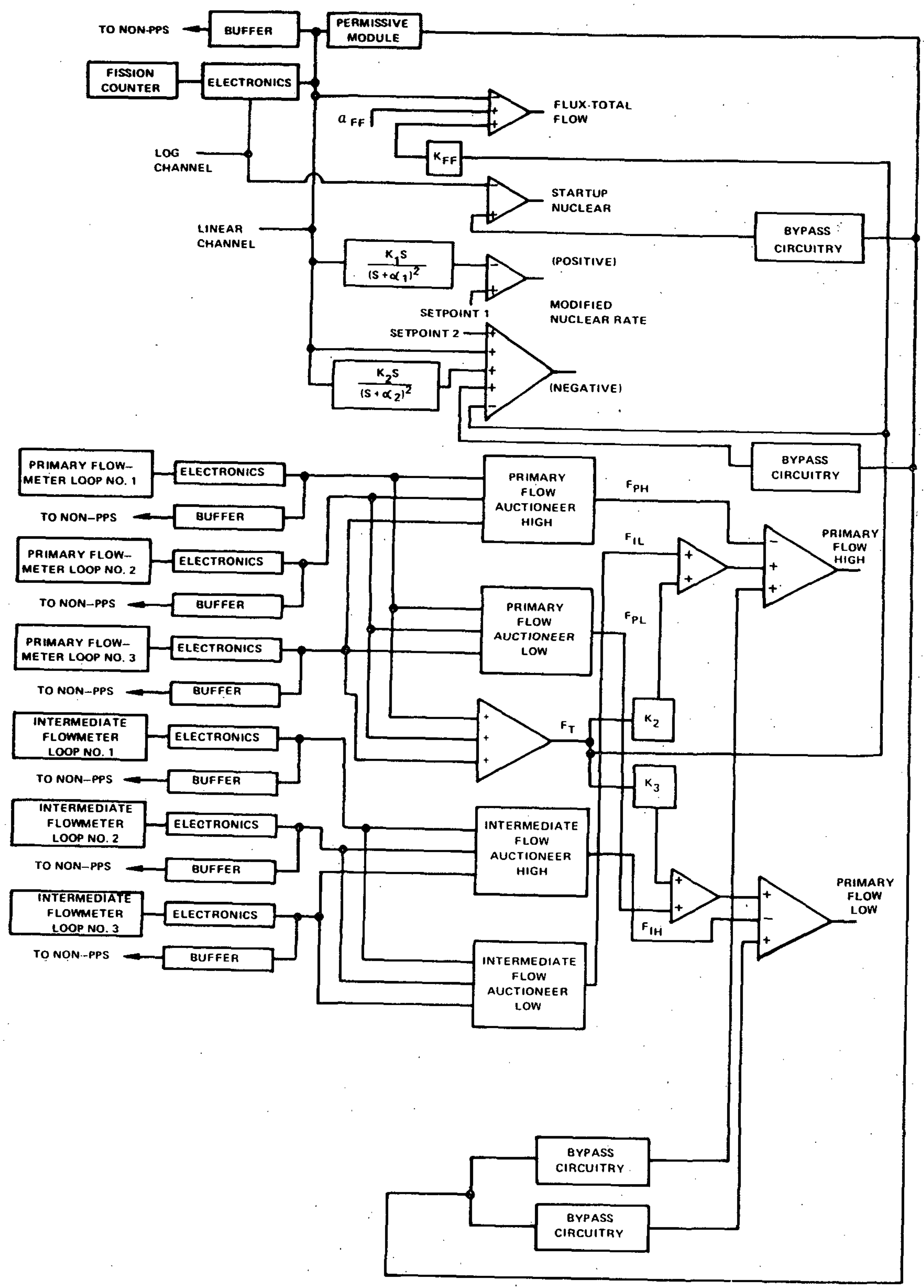


Figure 7.2-8. Functional Block Diagram of the Flux-Total Flow, Startup Nuclear, Modified Nuclear Rate, and Primary to Intermediate Flow Rate Protective Subsystems, One Channel of Three is Shown

4558-1

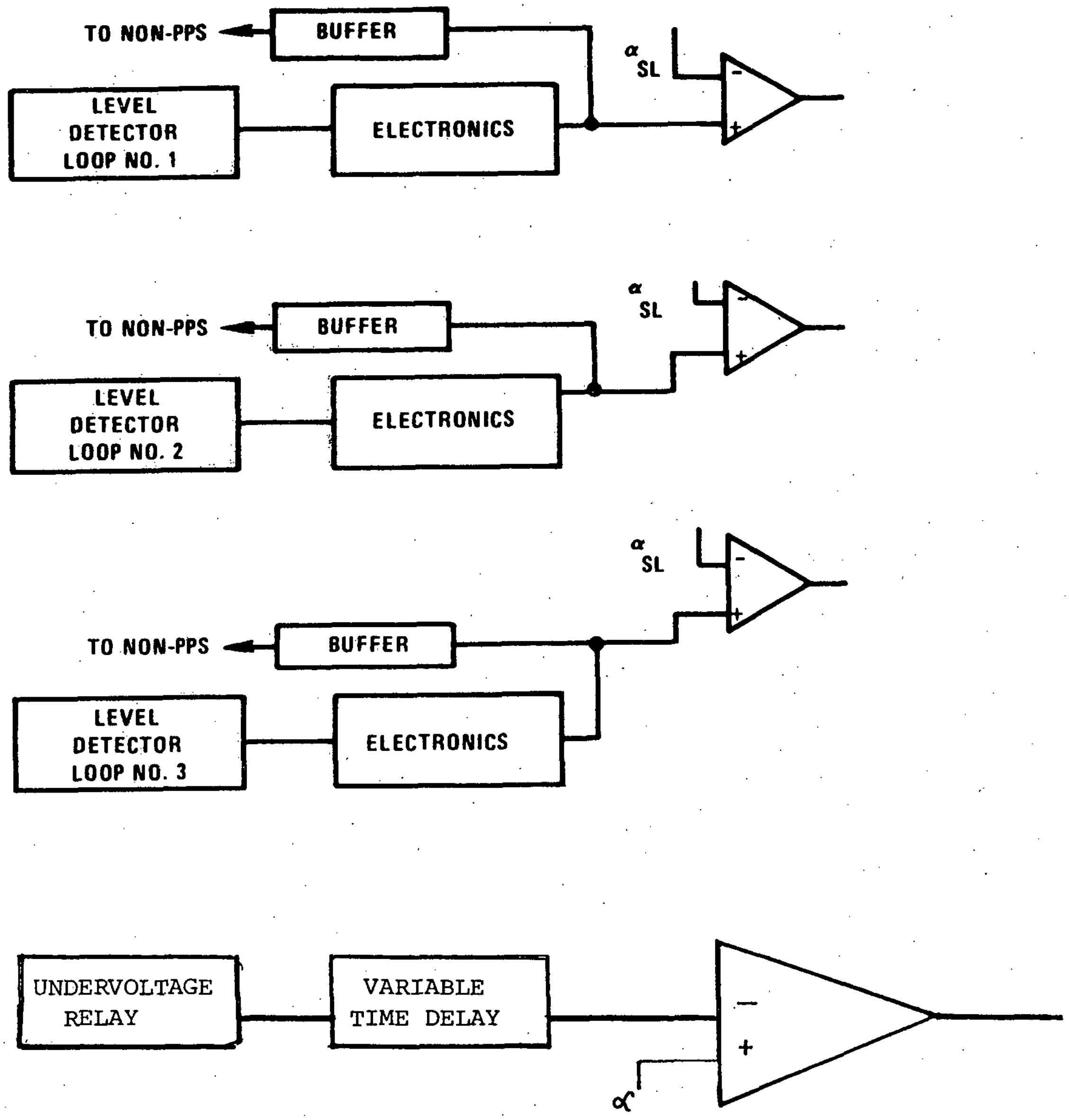
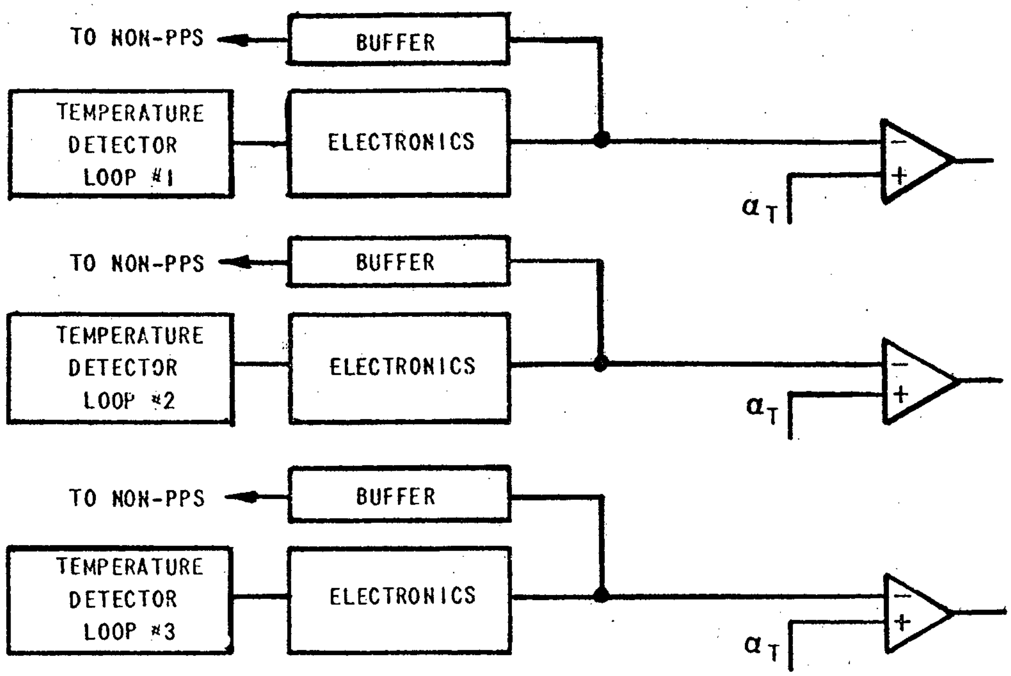
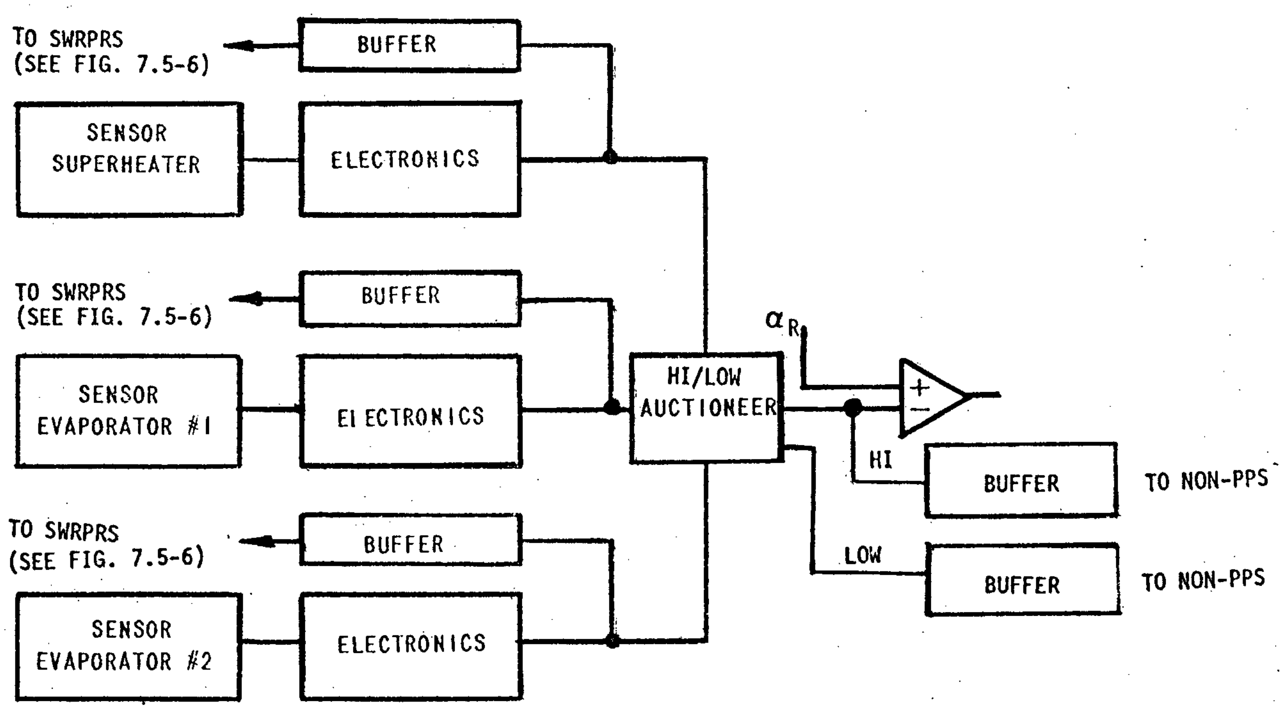


Figure 7.2-9. Functional Block Diagrams of the Steam Drum Level and HTS Pump Voltage Subsystems. One Channel of Three is Shown.



EVAPORATOR OUTLET TEMPERATURE



SODIUM WATER REACTION PROTECTIVE SUBSYSTEM

Figure 7.2-10 Functional Block Diagrams of the Evaporator Outlet Sodium Temperature and Sodium Water Reaction Protective Subsystems, One Channel of Three is Shown

6678-4

Table 7.2-5 PPS Sodium Flow Input - Failure Mode and Effects Analysis Criticality Analysis

PART ASSEMBLY OR PROCESS: PPS Sodium Flow Input				P/N: Plant Protection System (PPS)						
ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBABILITY	CRITICALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONSIBILITY DUE DATE
2	1	Permanent Magnet (1 per loop)	Creates magnetic field through which sodium flows	(1) Demagnetized	Physical shock, application of voltage or current	EU	I1	(1) Low signal in all 3 channels causes comparator trips and reactor scram.		
				(2) High field	Application of voltage or current	EU	I2	(2) Depending on magnitude, causes trip on high signal or causes loss of calibration in direction to retard trip on low flow.		
	2	Electrode (3 per loop)	Detects voltage induced by permanent magnet proportional to sodium flow	(1) Fails to low or zero output	Electrode connection to pipe loosens or is broken	U	I2	(1) Failure would cause incorrect (low) input to comparator. Failure could cause Flux-Flow or Primary-Intermediate Flow trip, or could retard channel comparator trip for channel specific events.		
				(2) Fails to high output	Application of external source with proper characteristics to flow detector.	EU	I2	(2) Failure would cause incorrect (high) input to comparator. Failure could cause Flux-Flow or Primary-Intermediate Flow trip or could retard channel comparator trip.		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

EXTREMELY UNLIKELY (EU) AN OFF-NORMAL CONDITION OF SUCH EXTREMELY LOW PROBABILITY THAT NO EVENT IN THIS CATEGORY IS EXPECTED TO OCCUR DURING THE PLANT LIFETIME BUT WHICH, NEVERTHELESS, REPRESENT EXTREME OR LIMITING CASES OF FAILURES WHICH ARE IDENTIFIED AS CONCEIVABLE.

CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

MINOR INCIDENT (MN) AN OCCURRENCE IN WHICH DAMAGE TO REPLACEABLE COMPONENTS IS NO GREATER THAN SOME SPECIFIED LOSS OF EFFECTIVE LIFETIME.

OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant(I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

6487-1

Table 7.2-5 PPS Sodium Flow Input - Failure Mode and Effects Analysis Criticality Analysis (continued)

PART ASSEMBLY OR PROCESS: PPS Sodium Flow Input (Continued)				P/N: Plant Protection System (PPS)						
ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBABILITY	CRITICALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONSIBILITY DUE DATE
2	3	Detector Wiring	Transmits detector signal to signal transmitter and control room.	(1) Fails to open circuit	Wire loose or severed	U	I2	(1) Same as 2.2-1		
				(2) Fails to short circuit	Insulation failure	U	I2	(2) Same as 2.2-1		
	4	Flow Transmitter	Amplifies and converts detector signal to standard PPS input signal.	(1) Fails to low or zero circuit	Internal failure (e.g., failed transister, capacitor)	U	I2	(1) Same as 2.2-1		
				(2) Fails to high current	Internal failure	EU	I2	(2) Same as 2.2-2		
	5	Power Supply	Supplies 25 V. DC power for instrumentation	(1) Fails to low or zero voltage	Internal failure (e.g., short circuit, open circuit, blown fuse)	A	I2	(1) Same as 2.2-1		
				(2) Fails to high voltage	Transformer primary winding short circuit	U	I2	(2) Same as 2.2-2		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

EXTREMELY UNLIKELY (EU) AN OFF-NORMAL CONDITION OF SUCH EXTREMELY LOW PROBABILITY THAT NO EVENT IN THIS CATEGORY IS EXPECTED TO OCCUR DURING THE PLANT LIFETIME BUT WHICH, NEVERTHELESS, REPRESENT EXTREME OR LIMITING CASES OF FAILURES WHICH ARE IDENTIFIED AS CONCEIVABLE.

CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

MINOR INCIDENT (MN) AN OCCURRENCE IN WHICH DAMAGE TO REPLACEABLE COMPONENTS IS NO GREATER THAN SOME SPECIFIED LOSS OF EFFECTIVE LIFETIME.

OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

6487-1

Table 7.2-6 Primary Pump Electric Power Sensor - Failure Mode and Effects Analysis Criticality Analysis

PART ASSEMBLY OR PROCESS: Primary Pump Electric Power Sensor				P/N: Plant Protection System (PPS)						
ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBA-BILITY	CRITI-CALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONS-IBILITY DUE DATE
3	1	Potential Transformer	Senses 13.8KV bus voltage	(1) Fails to low or zero voltage	Transformer open circuited.	U	I1	(1) Single channel comparator trips. Indicated by inter-channel comparison Safe Failure		
				(2) Fails to high voltage	Primary to secondary winding short circuit or short circuit across primary winding	EU	I2	(2) Failure prevents single channel comparator trip. Indicated by periodic testing. Bus voltage would normally go to 0 in loss of power event which would trip channel comparator even with short circuit applied.		
	2	Wiring from Potential Transformer to UV relay rack	Transmits sensed voltage signal to under-voltage relay.	(1) Fails to open circuit	Loose or broken wire	U	I1	(1) Same as 3.1-1 Safe failure		
				(2) Fails to short circuit	Insulation failure	U	I1	(2) Same as 3.1-1 Safe failure		
	3.	Undervoltage Relay (including auxiliary relays)	Determines whether bus voltage has dropped below a given set point. Auxiliary relays provide normally open contracts.	(1) Output contact permanently open	Failed UV relay	U	I1	(1) Same as 3.1-1 Safe failure		
				(2) Output contact permanently shorted	Output contacts welded	U	I2	(2) Failure prevents single channel comparator trip Indicated by periodic testing		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

EXTREMELY UNLIKELY (EU) AN OFF-NORMAL CONDITION OF SUCH EXTREMELY LOW PROBABILITY THAT NO EVENT IN THIS CATEGORY IS EXPECTED TO OCCUR DURING THE PLANT LIFETIME BUT WHICH, NEVERTHELESS, REPRESENT EXTREME OR LIMITING CASES OF FAILURES WHICH ARE IDENTIFIED AS CONCEIVABLE.

CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

MINOR INCIDENT (MN) AN OCCURRENCE IN WHICH DAMAGE TO REPLACEABLE COMPONENTS IS NO GREATER THAN SOME SPECIFIED LOSS OF EFFECTIVE LIFETIME.

OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant(I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

6487-1

Table 7.2-6 Primary Pump Electric Power Sensor - Failure Mode and Effects Analysis Criticality Analysis (continued)

PART ASSEMBLY OR PROCESS: Primary Pump Electric Power Sensor				P/N: Plant Protection System (PPS)						
ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBABILITY	CRITICALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONSIBILITY DUE DATE
3	4	Wiring from UV relay panel to control room.	Transmits open/closed contact signal to PPS	(1) Fails to open circuit.	Loose or broken wire	U	I1	(1) Same as 3.1-1 Safe Failure		
				(2) Fails to short circuit	Insulation failure	U	I2	(2) Same as 3.3-2		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

EXTREMELY UNLIKELY (EU) AN OFF-NORMAL CONDITION OF SUCH EXTREMELY LOW PROBABILITY THAT NO EVENT IN THIS CATEGORY IS EXPECTED TO OCCUR DURING THE PLANT LIFETIME BUT WHICH, NEVERTHELESS, REPRESENT EXTREME OR LIMITING CASES OF FAILURES WHICH ARE IDENTIFIED AS CONCEIVABLE.

CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

MINOR INCIDENT (MN) AN OCCURRENCE IN WHICH DAMAGE TO REPLACEABLE COMPONENTS IS NO GREATER THAN SOME SPECIFIED LOSS OF EFFECTIVE LIFETIME.

OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant(I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 Indicates a failure which tends to cause a component trip
I2 Indicates a failure which tends to prevent a component trip

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Table 7.2-7 Compensated Ion Chamber Nuclear Input - Failure Mode and Effects Analysis Criticality Analysis

PART ASSEMBLY OR PROCESS: Compensated Ion Chamber Nuclear Input				P/N: Plant Protection System (PPS)						
ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBABILITY	CRITICALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONSIBILITY DUE DATE
4	1	Compensated Ion Chamber	Converts neutron flux measurement to electrical signal	(1) Fails to low or zero output	Deteriorating detector Loose or broken wire Loss of thermalizing graphite block.	A	I2	(1) Failure prevents comparator trip. Indicated by Interchannel comparison.		
				(2) Fails to high output	Mechanical movement inside detector. Insulation deteriorates	U	I1	(2) Single channel comparator trips. Interchannel comparison and PPS status board. Safe failure.		
	2	Wiring	Transmits signal from sensor thru head compartment to transmitter	(1) Fails to open circuit	Loose or broken wire	U	I2	(1) Same as 4.1-1		
				(2) Fails to short circuit	Insulation failure	U	I2	(2) Same as 4.1-1		
	3	Compensated Ion Chamber Transmitter	Amplifies and converts detector signal to standard PPS signal	(1) Fails to low or zero voltage	Internal failure (e.g. transistor failure)	U	I2	(1) Same as 4.1-1		
				(2) Fails to high voltage	Internal failure (e.g. transistor failure)	EU	I2	(2) Same as 4.1-2 Safe failure.		
	4	Compensating Voltage Power Supply	Supplies compensating voltage to ion chamber	(1) Fails to low or zero voltage	Internal failure (e.g. transformer or rectifier failure)	U	I1	(1) Channel reads high. Same as 4.1-2 Safe failure.		
				(2) Fails to high voltage	Transformer winding short circuit	EU	I2	(2) Channel reads low Same as 4.1-1.		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

EXTREMELY UNLIKELY (EU) AN OFF-NORMAL CONDITION OF SUCH EXTREMELY LOW PROBABILITY THAT NO EVENT IN THIS CATEGORY IS EXPECTED TO OCCUR DURING THE PLANT LIFETIME BUT WHICH, NEVERTHELESS, REPRESENT EXTREME OR LIMITING CASES OF FAILURES WHICH ARE IDENTIFIED AS CONCEIVABLE.

CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

MINOR INCIDENT (MN) AN OCCURRENCE IN WHICH DAMAGE TO REPLACEABLE COMPONENTS IS NO GREATER THAN SOME SPECIFIED LOSS OF EFFECTIVE LIFETIME.

OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant(I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

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Table 7.2-7 Compensated Ion Chamber Nuclear Input - Failure Mode and Effects Analysis Criticality Analysis (continued)

PART ASSEMBLY OR PROCESS Compensated Ion Chamber Nuclear Input (Continued)				P/N Plant Protection System (PPS)						
ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBA-BILITY	CRITI-CALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONSIBILITY DUE DATE
4	5	Low Voltage Power Supply for Nuclear Electronics	Supplies ±15 V DC	(1) Fails to low or zero voltage	Internal failure (e.g. transformer, rectifier)	A	I2	(1) Channel reads low. Same as 4.1-1.		
				(2) Fails to high voltage	Transformer primary winding short circuit	U	I1	(2) Same as 4.1-2 Safe failure.		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

EXTREMELY UNLIKELY (EU) AN OFF-NORMAL CONDITION OF SUCH EXTREMELY LOW PROBABILITY THAT NO EVENT IN THIS CATEGORY IS EXPECTED TO OCCUR DURING THE PLANT LIFETIME BUT WHICH, NEVERTHELESS, REPRESENT EXTREME OR LIMITING CASES OF FAILURES WHICH ARE IDENTIFIED AS CONCEIVABLE.

CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

MINOR INCIDENT (MN) AN OCCURRENCE IN WHICH DAMAGE TO REPLACEABLE COMPONENTS IS NO GREATER THAN SOME SPECIFIED LOSS OF EFFECTIVE LIFETIME.

OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant(I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

6487-1

Table 7.2-8 Fission Chamber Nuclear Input - Failure Mode and Effects Analysis Criticality Analysis

PART ASSEMBLY OR PROCESS: Fission Chamber Nuclear Input

P/N: Plant Protection System (PPS)

ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBA-BILITY	CRITI-CALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONS-IBILITY DUE DATE
5	1.	Fission Chamber	Converts neutron flux measurement to electrical signal	(1) Fails to low or zero output	(1) Counter electrode or connecting wire breaks.	A	I2	(1) Failure prevents comparator trip. Indicated by Interchannel Comparison.		
				(2) Fails to high output	(2) Mechanical movement or insulation deterioration	U	I1	(2) Single Channel Comparator trips. Indicated by Interchannel Comparison and PPS status board. Safe failure		
	2.	High Voltage Power Supply for Fission Chamber	Provides high voltage necessary for neutron interaction in fission chamber	(1) Fails to low or zero voltage	(1) Internal failure (e.g. rectifier, fuse, or transformer failure)	A	I2	(1) Same as 5.1-1		
				(2) Fails to high voltage	(2) Transformer primary winding short circuit	U	I1	(2) Same as 5.1-2		
	3.	Wiring from Fission Chamber to Preamplifier	Transmits low level signal from detector to preamplifier	(1) Fails to open circuit	(1) Loose or broken wire	U	I2	(1) Same as 5.1-1		
				(2) Fails to short circuit	(2) Insulation failure	U	I1	(2) Same as 5.1-2		
				(3) Fails to ground	(3) Insulation failure	U	I2	(3) Same as 5.1-1		
	4.	Pre-amplifier	Amplifies low level signal to limit noise pick-up problems	(1) Fails to low or zero output	(1) Internal failure (e.g. resistor, transmitor)	U	I2	(1) Same as 5.1-1 except only counting and log MSV ranges are affected		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

EXTREMELY UNLIKELY (EU) AN OFF-NORMAL CONDITION OF SUCH EXTREMELY LOW PROBABILITY THAT NO EVENT IN THIS CATEGORY IS EXPECTED TO OCCUR DURING THE PLANT LIFETIME BUT WHICH, NEVERTHELESS, REPRESENT EXTREME OR LIMITING CASES OF FAILURES WHICH ARE IDENTIFIED AS CONCEIVABLE.

CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

MINOR INCIDENT (MN) AN OCCURRENCE IN WHICH DAMAGE TO REPLACEABLE COMPONENTS IS NO GREATER THAN SOME SPECIFIED LOSS OF EFFECTIVE LIFETIME.

OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant(I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

6487-1

Figure 7.2-8 Fission Chamber Nuclear Input - Failure Mode and Effects Analysis Criticality Analysis (continued)

PART ASSEMBLY OR PROCESS: Fission Chamber Nuclear Input (Cont.)

P/N: Plant Protection System (PPS)

ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBABILITY	CRITICALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONSIBILITY DUE DATE
5	4.			(2) Fails to high output	(2) Internal failure (e.g. transistor)	EU	I1	(2) Same as 5.1-2 except that trip occurs only in counting and log MSV ranges.		
				(3) D.C. output fails low or to zero	(3) Internal failure (e.g. resistor, capacitor)	U	I2	(3) Same as 5.1-1 except only power range channel is affected		
	5.	Wiring from Pre-amplifier to Nuclear Instrument-ation Cabinet	Transmits signal to main nuclear instrument-ation cabinet	(1) Pulse wiring fails to open circuit	(1) Loose or broken wire	U	I2	(1) Same as 5.4-1		
				(2) Pulse wiring fails to short circuit	(2) Insulation failure	U	I2	(2) Same as 5.4-1		
				(3) D.C. wiring fails to open circuit	(3) Loose or broken wire	U	I2	(3) Same as 5.1-1 except only power range is affected		
				(4) D.C. Wiring fails to short circuit	(4) Insulation failure	U	I2	(4) Same as 5.5-3		
	6.	Wide Range Counting Range Electronics	Amplifies and converts sensor signal to standard PPS input	(1) Fails to low or zero input	(1) Internal failure (e.g. transistor)	A	I2	(1) Same as 5.1-1 but only counting range is affected		
				(2) Fails to high output	(2) Internal failure (e.g. transistor)	U	I2	(2) Same as 5.1-2 but only counting range is affected		
	7.	Wide Range Log MSV Range Electronics	Amplifies and converts sensor signal to standard PPS input	(1) Fails to low or zero output	(1) Internal Failure (e.g. transistor)	A	I2	(1) Same as 5.1-1 but only log MSV range is affected		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

EXTREMELY UNLIKELY (EU) AN OFF-NORMAL CONDITION OF SUCH EXTREMELY LOW PROBABILITY THAT NO EVENT IN THIS CATEGORY IS EXPECTED TO OCCUR DURING THE PLANT LIFETIME BUT WHICH, NEVERTHELESS, REPRESENT EXTREME OR LIMITING CASES OF FAILURES WHICH ARE IDENTIFIED AS CONCEIVABLE.

CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

MINOR INCIDENT (MN) AN OCCURRENCE IN WHICH DAMAGE TO REPLACEABLE COMPONENTS IS NO GREATER THAN SOME SPECIFIED LOSS OF EFFECTIVE LIFETIME.

OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant(I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

6487-1

Table 7.2-8 Fission Chamber Nuclear Input - Failure Mode and Effects Analysis Criticality Analysis (continued)

PART ASSEMBLY OR PROCESS: Fission Chamber Nuclear Input (Cont.)				P/N: Plant Protection System (PPS)						
ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBA-BILITY	CRITI-CALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONS-IBILITY DUE DATE
5	7.			(2) Fails to high output	(2) Internal failure (e.g. transistor)	U	I2	(2) Same as 5.1-2 but only log MSV range is affected		
	8.	Power Supply	Provide ± 15 VDC for electronics in all 3 ranges	(1) Fails to low or zero voltage	(1) Internal failure (e.g. transformer, rectifier, fuse)	A	I2	(1) Same as 5.1-1		
				(2) Fails to high voltage	(2) Transformer primary winding short circuit	U	I1	(2) Same as 5.1-2 safe failure		
	9.	Wiring from Nuclear Instrument-ation Cabinet to Control Room	Transmit 1-5 VDC signal to PPS comparators	(1) Fails to open circuit	(1) Loose or broken wire	U	I1	(1) Same as 5.1-2 safe failure		
				(2) Fails to short circuit	(2) Insulation failure	U	I1	(2) Same as 5.1-2 safe failure		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

EXTREMELY UNLIKELY (EU) AN OFF-NORMAL CONDITION OF SUCH EXTREMELY LOW PROBABILITY THAT NO EVENT IN THIS CATEGORY IS EXPECTED TO OCCUR DURING THE PLANT LIFETIME BUT WHICH, NEVERTHELESS, REPRESENT EXTREME OR LIMITING CASES OF FAILURES WHICH ARE IDENTIFIED AS CONCEIVABLE.

CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

MINOR INCIDENT (MN) AN OCCURRENCE IN WHICH DAMAGE TO REPLACEABLE COMPONENTS IS NO GREATER THAN SOME SPECIFIED LOSS OF EFFECTIVE LIFETIME.

OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant (I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

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Table 7.2-9 Primary Loop Inlet Plenum Pressure Input-Failure Mode and Effects Analysis Criticality Analysis

PART ASSEMBLY OR PROCESS: Primary Loop Inlet Plenum Pressure Input				P/N: Plant Protection System (PPS)						
ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBABILITY	CRITICALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONSIBILITY DUE DATE
6	1.	Pressure Sensor (2 Per Loop)	Sensor pressure through bellows	(1) Rupture of bellows	(1) Defective or worn bellows	U	I2	(1) Sodium will plug in cooler capillary tube line. Sensor will not respond to changes in pressure. Auctioneer selects high signal from 2 inputs. Failure may prevent trip if line plugs within normal pressure range. Indicated by Interchannel Comparison.		
				(2) Bellows fail to move freely	(2) Work or temperature hardening of bellows	U	I2	(2) Slow or no response to changes in pressure. Failure may prevent trip. Indicated by Interchannel Comparison		
	2.	Capillary Tube	Transmits pressure from sensor to transducer	(1) Capillary tube ruptures	(1) Defective tube, wear, external force	EU	I1	(1) Loss of pressure driving signal. Auctioneer selects signal from redundant sensor. Safe failure.		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

EXTREMELY UNLIKELY (EU) AN OFF-NORMAL CONDITION OF SUCH EXTREMELY LOW PROBABILITY THAT NO EVENT IN THIS CATEGORY IS EXPECTED TO OCCUR DURING THE PLANT LIFETIME BUT, WHICH, NEVERTHELESS, REPRESENT EXTREME OR LIMITING CASES OF FAILURES WHICH ARE IDENTIFIED AS CONCEIVABLE.

CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

MINOR INCIDENT (MN) AN OCCURRENCE IN WHICH DAMAGE TO REPLACEABLE COMPONENTS IS NO GREATER THAN SOME SPECIFIED LOSS OF EFFECTIVE LIFETIME.

OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant (I): An occurrence which has no effect on safety or operating performance.
For redundant Components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

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Table 7.2-9 Primary Loop Inlet Plenum Pressure Input-Failure Mode and Effects Analysis Criticality Analysis (continued)

PART ASSEMBLY OR PROCESS (Cont.) Primary Loop Inlet Plenum Pressure Input				P/N Plant Protection System (PPS)						
ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBABILITY	CRITICALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONSIBILITY DUE DATE
6	2.			(2) Capillary tube static failure	(2) Pinched tube or frozen Nak	EU	I2	(2) Sensor will not respond to changes in pressure. Same results as 6.1-2.		
	3.	Strain Gauge Transducer	Converts pressure signal to low level electric signal (resistive measurement)	(1) Fails to low or zero output	(1) Internal failure	U	I1	(1) Auctioneer selects signal from redundant sensor. Safe failure.		
				(2) Fails to high output	(2) Crack in pressure plate	EU	I2	(2) Single comparator channel trip prevented. Failure indicated by Interchannel Comparison.		
	4.	Wiring from Transducer to Transmitter (4 leads)	Transmits signal from transducer to transmitter	(1) Fails to zero output	(1) Broken or loose wire. Insulation failure.	U	I1	(1) Same as 6.3-1 Safe failure.		
	5.	Statham Converter Transmitter	Converts resistive measurement to standard PPS input	(1) Fails to low or zero output	(1) Internal failure	U	I1	(1) Same as 6.3-1 Safe failure		
				(2) Fails to high output	(2) Internal failure	EU	I2	(2) Same as 6.3-2		
	6.	Wiring from Converter Transmitter to Control Room	Transmits signal to PPS equipment	(1) Fails to open circuit	(1) Loose or broken wire	U	I1	(1) Same as 6.3-1 Safe failure		
				2) Fails to short circuit	(2) Insulation failure	U	I1	(2) Same as 6.3-1 Safe failure		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

EXTREMELY UNLIKELY (EU) AN OFF-NORMAL CONDITION OF SUCH EXTREMELY LOW PROBABILITY THAT NO EVENT IN THIS CATEGORY IS EXPECTED TO OCCUR DURING THE PLANT LIFETIME BUT WHICH, NEVERTHELESS, REPRESENT EXTREME OR LIMITING CASES OF FAILURES WHICH ARE IDENTIFIED AS CONCEIVABLE.

CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

MINOR INCIDENT (MN) AN OCCURRENCE IN WHICH DAMAGE TO REPLACEABLE COMPONENTS IS NO GREATER THAN SOME SPECIFIED LOSS OF EFFECTIVE LIFETIME.

OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant (I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

6487-1

Table 7.2-9 Primary Loop Inlet Plenum Pressure Input - Failure Mode and Effects Analysis Criticality Analysis (continued)

PART ASSEMBLY OR PROCESS (Cont.) Primary Loop Inlet Plenum Pressure Input				P/N Plant Protection System (PPS)						
ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBABILITY	CRITICALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONSIBILITY DUE DATE
6	7.	Power Supply for Statham Converter Transmitter	Provides DC power for converter transmitter	(1) Fails to low or zero voltage	(1) Internal failure	A	I1	(1) Same as 6.3-1 Safe failure		
				(2) Fails to high voltage	(2) Transformer winding short circuit	U	I2	(2) Same as 6.3-2		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

EXTREMELY UNLIKELY (EU) AN OFF-NORMAL CONDITION OF SUCH EXTREMELY LOW PROBABILITY THAT NO EVENT IN THIS CATEGORY IS EXPECTED TO OCCUR DURING THE PLANT LIFETIME BUT WHICH, NEVERTHELESS, REPRESENT EXTREME OR LIMITING CASES OF FAILURES WHICH ARE IDENTIFIED AS CONCEIVABLE.

CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

MINOR INCIDENT (MN) AN OCCURRENCE IN WHICH DAMAGE TO REPLACEABLE COMPONENTS IS NO GREATER THAN SOME SPECIFIED LOSS OF EFFECTIVE LIFETIME.

OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant (I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

6487-1

Table 7.2-10 Sodium Pump Speed Input (Primary and Intermediate) - Failure Mode and Effects Analysis Criticality Analysis

PART ASSEMBLY OR PROCESS Sodium Pump Speed Input (Primary and Intermediate)				P/N Plant Protection System (PPS)						
ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBABILITY	CRITICALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONSIBILITY DUE DATE
7	1	Pulsing Digital Tachometer (including toothed gear on pump shaft and 3 EM proximity sensors per loop)	Changing reluctance of rotating gear teeth on pump shaft is measured by EM proximity sensor and converted to electrical signal	(1) Loss of one or more gear teeth	Mechanical failure or external force	EU	I2	(1) Affected pump speed sensor will indicate low. Failure may trip channel comparator or will be indicated by inter-channel comparison. Failure may prevent trip if coupled with change in corresponding loop speed (i.e., if primary speed indicates low and intermediate speed is low, comparator will not trip as it should).		
				(2) Foreign material in air gap between sensor and gear	Dirt, metal fillings buildup, short circuiting two gear teeth.	U	I2	(2) Affected pump speed may read high or low depending on permeability of material. Results same as 7.1-1.		
				(3) EM proximity sensor fails to low or zero output	Internal failure	U	I2	(3) Same as 7.1-1.		
				(4) EM proximity sensor fails to high output	Internal failure	EU	I2	(4) Affected pump speed sensor will indicate high. Same results as 7.1-1.		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

EXTREMELY UNLIKELY (EU) AN OFF-NORMAL CONDITION OF SUCH EXTREMELY LOW PROBABILITY THAT NO EVENT IN THIS CATEGORY IS EXPECTED TO OCCUR DURING THE PLANT LIFETIME BUT WHICH, NEVERTHELESS, REPRESENT EXTREME OR LIMITING CASES OF FAILURES WHICH ARE IDENTIFIED AS CONCEIVABLE.

CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

MINOR INCIDENT (MN) AN OCCURRENCE IN WHICH DAMAGE TO REPLACEABLE COMPONENTS IS NO GREATER THAN SOME SPECIFIED LOSS OF EFFECTIVE LIFETIME.

OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant(I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

6487-1

Table 7.2-10 Sodium Pump Speed Input (Primary and Intermediate) - Failure Mode and Effects Analysis Criticality Analysis (continued)

PART ASSEMBLY OR PROCESS Sodium Pump Speed Input (Primary and Intermediate)				P/N Plant Protection System (PPS)						
ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBABILITY	CRITICALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONSIBILITY DUE DATE
7	2	Wiring from sensor to transmitter	Transmit electrical signal from sensor to transmitter	(1) Fails to open circuit	Loose or broken wire	U	I1	(1) Affected pump speed sensor will indicate zero. Single comparator channel trip. Safe failure.		
				(2) Fails to short circuit	Insulation failure	U	I1	(2) Same as 7.2-1		
	3	Sodium Pump Speed Transmitter	Amplifies and converts digital speed signal to standard analog input for PPS	(1) Fails to low or zero output	Internal failure	U	I2	(1) Same as 7.1-1		
				(2) Fails to high output	Internal failure	EU	I2	(2) Same as 7.1-4		
	4	Wiring from transmitter to control room	Transmits analog signal from transmitter to control room	(1) Fails to open circuit	Loose or broken wire	U	I1	(1) Same as 7.2-1.		
				(2) Fails to short circuit	Insulation failure	U	I1	(2) Same as 7.2-1.		
	5	Power Supply for Transmitter	Provides power for transmitter electronics	(1) Fails to low or zero output	Internal failure	A	I2	(1) Same as 7.1-1		
				(2) Fails to high output	Internal failure	U	I2	(2) Same as 7.1-4		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

EXTREMELY UNLIKELY (EU) AN OFF-NORMAL CONDITION OF SUCH EXTREMELY LOW PROBABILITY THAT NO EVENT IN THIS CATEGORY IS EXPECTED TO OCCUR DURING THE PLANT LIFETIME BUT WHICH, NEVERTHELESS, REPRESENT EXTREME OR LIMITING CASES OF FAILURES WHICH ARE IDENTIFIED AS CONCEIVABLE.

CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

MINOR INCIDENT (MN) AN OCCURRENCE IN WHICH DAMAGE TO REPLACEABLE COMPONENTS IS NO GREATER THAN SOME SPECIFIED LOSS OF EFFECTIVE LIFETIME.

OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant (I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

6487-1

Table 7.2-11 Steam Mass Flow Rate Input - Failure Mode and Effects Analysis Criticality Analysis

PART, ASSEMBLY OR PROCESS: Steam Mass Flow Rate Input

P/N: Plant Protection System (PPS)

ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBABILITY	CRITICALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONSIBILITY DUE DATE
8	1.	Differential Pressure Transducer	Senses differential pressure across venturi flow tube and converts it to an electrical signal	(1) Fails to low or zero output	(1) Internal failure	U	I2	(1) Affected sensor will indicate low steam flow. Failure may trip channel comparator or will be indicated by Interchannel Comparison. Failure may prevent comparator trip if corresponding decrease in feedwater flow occurs.		
				(2) Fails to high output	(2) Internal failure	EU	I2	(2) Affected sensor will indicate high steam flow. Results are the same as 8.1-1 except that comparator trip may be prevented if a corresponding increase in feed-water flow occurs.		
	2.	Temperature Sensor	Senses steam temp. to compensate steam mass flow rate	(1) Fails to low or zero output	(1) Open circuit in temp. sensor	U	I2	(1) Same as 8.1-1.		
				(2) Fails to high output	(2) Application of hot short circuit	EU	I2	(2) Same as 8.1-2.		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

EXTREMELY UNLIKELY (EU) AN OFF-NORMAL CONDITION OF SUCH EXTREMELY LOW PROBABILITY THAT NO EVENT IN THIS CATEGORY IS EXPECTED TO OCCUR DURING THE PLANT LIFETIME BUT WHICH, NEVERTHELESS, REPRESENT EXTREME OR LIMITING CASES OF FAILURES WHICH ARE IDENTIFIED AS CONCEIVABLE.

CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

MINOR INCIDENT (MN) AN OCCURRENCE IN WHICH DAMAGE TO REPLACEABLE COMPONENTS IS NO GREATER THAN SOME SPECIFIED LOSS OF EFFECTIVE LIFETIME.

OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT

Insignificant(I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

6487-1

Table 7.2-11 Steam Mass Flow Rate Input - Failure Mode and Effects Analysis Criticality Analysis (continued)

PART ASSEMBLY OR PROCESS Steam Mass Flow Rate Input (Cont.)				P/N Plant Protection System (PPS)						
ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBABILITY	CRITICALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONSIBILITY DUE DATE
8	3.	Pressure Sensor	Senses steam pressure to compensate steam mass flow rate	(1) Fails to low or zero output	(1) Rupture in press. sensor	U	I2	(1) Same as 8.1-2.		
				(2) Fails to high output	(2) Special internal failure	EU	I2	(2) Same as 8.1-1.		
	4.	Wiring From Transducer to Transmitter	Transmit transducer signal to transmitter	(1) Fails to open circuit	(1) Loose or broken wire	U	I1	(1) Single comparator channel trip. Indicated by Interchannel Comparison or PPS Status board. Safe failure.		
				(2) Fails to short circuit	(2) Insulation failure	U	I1	(2) Same as 8.4-1 Safe Failure		
	5.	Wiring from Temp. Sensor to Transmitter	Transmits temp. signal to transmitter	(1) Fails to open circuit	(1) Loose or broken wire	U	I2	(1) Same as 8.1-1.		
				(2) Fails to short circuit	(2) Insulation failure	U	I2	(2) Same as 8.1-1.		
	6.	Wiring From Pressure Sensor to Transmitter	Transmits pressure signal to transmitter	(1) Fails to open circuit	(1) Loose or broken wire	U	I2	(1) Same as 8.1-2.		
				(2) Fails to short circuit	(2) Insulation failure	U	I2	(2) Same as 8.1-2.		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

EXTREMELY UNLIKELY (EU) AN OFF-NORMAL CONDITION OF SUCH EXTREMELY LOW PROBABILITY THAT NO EVENT IN THIS CATEGORY IS EXPECTED TO OCCUR DURING THE PLANT LIFETIME BUT WHICH, NEVERTHELESS, REPRESENT EXTREME OR LIMITING CASES OF FAILURES WHICH ARE IDENTIFIED AS CONCEIVABLE.

CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

MINOR INCIDENT (MN) AN OCCURRENCE IN WHICH DAMAGE TO REPLACEABLE COMPONENTS IS NO GREATER THAN SOME SPECIFIED LOSS OF EFFECTIVE LIFETIME.

OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant(I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

6487-1

Table 7.2-11 Steam Mass Flow Rate Input - Failure Mode and Effects Analysis Criticality Analysis (continued)

PART ASSEMBLY OR PROCESS Steam Mass Flow Rate Input (Cont.)				P/N Plant Protection System (PPS)						
ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBABILITY	CRITICALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONSIBILITY DUE DATE
8	7.	Steam Flow Transmitter	Compensates differential press. signal for temp. and pressure Amplifies and converts signal to standard PPS input.	(1) Fails to low or zero output (2) Fails to high output	(1) Internal failure (2) Internal failure	U EU	I2 I2	(1) Same as 8.1-1. (2) Same as 8.1-2.		
	8.	Wiring From Transmitter to Control Room	Transmits differential pressure signal from transmitter to control room	(1) Fails to open circuit (2) Fails to short circuit	(1) Loose or broken wire (2) Insulation failure	U U	I1 I1	(1) Same as 8.4-1 safe failure. (2) Same as 8.4-1 safe failure.		
	9.	Steam Flow Transmitter Power Supply	Provides power to differential pressure transmitter	(1) Fails to low or zero output (2) Fails to high output	(1) Internal failure (2) Transformer winding short circuit	A U	I2 I2	(1) Same as 8.1-1. (2) Same as 8.1-2.		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

EXTREMELY UNLIKELY (EU) AN OFF-NORMAL CONDITION OF SUCH EXTREMELY LOW PROBABILITY THAT NO EVENT IN THIS CATEGORY IS EXPECTED TO OCCUR DURING THE PLANT LIFETIME BUT WHICH, NEVERTHELESS, REPRESENT EXTREME OR LIMITING CASES OF FAILURES WHICH ARE IDENTIFIED AS CONCEIVABLE.

CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

MINOR INCIDENT (MN) AN OCCURRENCE IN WHICH DAMAGE TO REPLACEABLE COMPONENTS IS NO GREATER THAN SOME SPECIFIED LOSS OF EFFECTIVE LIFETIME.

OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant (I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

6487-1

Table 7.2-12 Feedwater Mass Flow Rate Input - Failure Mode and Effects Analysis Criticality Analysis

PART ASSEMBLY OR PROCESS: Feedwater Mass Flow Rate Input				P/N: Plant Protection System (PPS)						
ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBABILITY	CRITICALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONSIBILITY DUE DATE
9	1.	Differential Pressure Transducer	Senses differential pressure across venturi flow tube and converts it to an electrical signal	(1) Fails to low or zero output	(1) Internal failure	U	I2	(1) Affected sensor will indicate low feedwater flow. Failure may trip channel comparator or will be indicated by Interchannel Comparison. Failure may prevent comparator trip if corresponding decrease in steam flow occurs.		
				(2) Fails to high output	(2) Internal failure	EU	I2	(2) Affected sensor will indicate high feedwater flow. Results are same as 9.1-1 except that comparator trip may be prevented if a corresponding increase in steam flow occurs.		
	2.	Temperature Sensor	Senses steam temperature to compensate steam mass flow rate	(1) Fails to low or zero output	(1) Open circuit in temperature circuit	U	I2	(1) Same as 9.1-1.		
				(2) Fails to high output	(2) Application of hot short circuit	EU	I2	(2) Same as 9.1-1.		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

EXTREMELY UNLIKELY (EU) AN OFF-NORMAL CONDITION OF SUCH EXTREMELY LOW PROBABILITY THAT NO EVENT IN THIS CATEGORY IS EXPECTED TO OCCUR DURING THE PLANT LIFETIME BUT WHICH, NEVERTHELESS, REPRESENT EXTREME OR LIMITING CASES OF FAILURES WHICH ARE IDENTIFIED AS CONCEIVABLE.

CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

MINOR INCIDENT (MN) AN OCCURRENCE IN WHICH DAMAGE TO REPLACEABLE COMPONENTS IS NO GREATER THAN SOME SPECIFIED LOSS OF EFFECTIVE LIFETIME.

OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant (I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

6487-1

Table 7.2-12 Feedwater Mass Flow Rate Input - Failure Mode and Effects Analysis Criticality Analysis (continued)

PART ASSEMBLY OR PROCESS: Feedwater Mass Flow Rate Input (Cont.)

P/N: Plant Protection System (PPS)

ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBABILITY	CRITICALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONSIBILITY DUE DATE
9	3.	Wiring From Tranducer to Transmitter	Transmits transducer signal to transmitter	(1) Fails to open circuit	(1) Loose or broken wire	U	I1	(1) Single comparator channel trip. Indicated by Interchannel Comparison and PPS Status Board. Safe failure.		
				(2) Fails to short circuit	(2) Insulation failure	U	I1	(2) Same as 9.3-1 safe failure.		
	4.	Wiring From Temperature Sensor to Transmitter	Transmits temperature signal to transmitter	(1) Fails to open circuit	(1) Loose or broken wire	U	I2	(1) Same as 9.1-1		
				(2) Fails to short circuit	(2) Insulation failure	U	I2	(2) Same as 9.1-1		
	5.	Feedwater Flow Rate Transmitter	Compensates differential pressure signal for temperature. Amplifies and converts signal to standard PPS input	(1) Fails to low or zero output	(1) Internal failure	U	I2	(1) Same as 9.1-1		
				(2) Fails to high output	(2) Internal failure	EU	I2	(2) Same as 9.2-1		
	6.	Wiring from Transmitter to Control Room	Transmits differential pressure signal from transmitter to control room	(1) Fails to open circuit	(1) Loose or broken wire	U	I1	(1) Same as 9.3-1 safe failure		
				(2) Fails to short circuit	(2) Insulation failure	U	I1	(2) Same as 9.3-1 safe failure		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

EXTREMELY UNLIKELY (EU) AN OFF-NORMAL CONDITION OF SUCH EXTREMELY LOW PROBABILITY THAT NO EVENT IN THIS CATEGORY IS EXPECTED TO OCCUR DURING THE PLANT LIFETIME BUT WHICH, NEVERTHELESS, REPRESENT EXTREME OR LIMITING CASES OF FAILURES WHICH ARE IDENTIFIED AS CONCEIVABLE.

CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

MINOR INCIDENT (MN) AN OCCURRENCE IN WHICH DAMAGE TO REPLACEABLE COMPONENTS IS NO GREATER THAN SOME SPECIFIED LOSS OF EFFECTIVE LIFETIME.

OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant (I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

6487-1

Table 7.2-12 Feedwater Mass Flow Rate Input - Failure Mode and Effects Analysis Criticality Analysis (continued)

PART ASSEMBLY OR PROCESS Feedwater Mass Flow Rate Input (Cont.)				P/N Plant Protection System (PPS)						
ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBABILITY	CRITICALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONSIBILITY DUE DATE
9	7.	Transmitter Power Supply	Provides power to differential pressure transmitter	(1) Fails to low or zero output	(1) Internal failure	A	I2	(1) Same as 9.1-1		
				(2) Fails to high output	(2) Transformer winding short circuit	U	I2	(2) Same as 9.1-2		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

EXTREMELY UNLIKELY (EU) AN OFF-NORMAL CONDITION OF SUCH EXTREMELY LOW PROBABILITY THAT NO EVENT IN THIS CATEGORY IS EXPECTED TO OCCUR DURING THE PLANT LIFETIME BUT WHICH, NEVERTHELESS, REPRESENT EXTREME OR LIMITING CASES OF FAILURES WHICH ARE IDENTIFIED AS CONCEIVABLE.

CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

MINOR INCIDENT (MN) AN OCCURRENCE IN WHICH DAMAGE TO REPLACEABLE COMPONENTS IS NO GREATER THAN SOME SPECIFIED LOSS OF EFFECTIVE LIFETIME.

OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant(I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

6487-1

Table 7.2-13 Steam Drum Level Input - Failure Mode and Effects Analysis Criticality Analysis

PART ASSEMBLY OR PROCESS Steam Drum Level Input				P/N Plant Protection System (PPS)						
ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBA-BILITY	CRITI-CALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONS-IBILITY DUE DATE
10	1	Differential Pressure Column	Used as static water column to measure differential pressure	(1) Column fails to low water level	(1) Operator maloperation (inadvertent drainage of column). Column leak	U	I2	(1) Steam drum level appears high and prevent single comparator channel trip on low steam drum level.		
	2	Differential Pressure Transducer	Senses differential pressure between static water column and steam drum level and converts it to electrical indication of steam drum level	(1) Fails to low or zero output	(1) Strain gauge failure	U	I1	(1) Single comparator channel trip. Safe failure		
				(2) Fails to high output	(2) Cracked strain gauge plate	EU	I2	(2) Same as 10.1-1		
	3	Pressure Sensor	Senses steam drum pressure to compensate steam drum level	(1) Fails to low or zero output	(1) Rupture in pressure sensor	U	I1	(1) Steam drum level appears low. Single comparator channel trip. Safe failure.		
				(2) Fails to high output	(2) Special internal failure	EU	I2	(2) Same as 10.1-1		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

EXTREMELY UNLIKELY (EU) AN OFF-NORMAL CONDITION OF SUCH EXTREMELY LOW PROBABILITY THAT NO EVENT IN THIS CATEGORY IS EXPECTED TO OCCUR DURING THE PLANT LIFETIME BUT WHICH, NEVERTHELESS, REPRESENT EXTREME OR LIMITING CASES OF FAILURES WHICH ARE IDENTIFIED AS CONCEIVABLE.

CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

MINOR INCIDENT (MN) AN OCCURRENCE IN WHICH DAMAGE TO REPLACEABLE COMPONENTS IS NO GREATER THAN SOME SPECIFIED LOSS OF EFFECTIVE LIFETIME.

OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant(I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

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Table 7.2-13 Steam Drum Level Input - Failure Mode and Effects Analysis Criticality Analysis (continued)

PART ASSEMBLY OR PROCESS Steam Drum Level Input (cont)				P/N Plant Protection System (PPS)						
ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBA-BILITY	CRITI-CALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONS-IBILITY DUE DATE
10	4	Wiring from transducer to trans-mitter	Transmits transducer signal to transmitter	(1) Fails to open circuit	(1) Loose or broken wire	U	I1	(1) Same as 10.2-1 Safe failure		
				(2) Fails to short circuit	(2) Insulation fail-ure	U	I1	(2) Same as 10.2-1 Safe failure		
	5	Wiring from pressure sensor to transmitter	Transmits pressure signal to transmitter	(1) Fails to open circuit	(1) Loose or broken wire	U	I1	(1) Same as 10.3-1 Safe failure		
				(2) Fails to short circuit	(2) Insulation failure	U	I1	(2) Same as 10.3-1 Safe failure		
	6	Steam drum level trans mitter	Compensates steam drum level for steam drum pressure. Amplifies and con-verts sig-nal to standard PPS input	(1) Fails to low or zero output	(1) Internal failure	U	I1	(1) Same as 10.2-1 Safe failure		
				(2) Fails to high output	(2) Internal failure	EU	I2	(2) Same as 10.1-1		
	7	Wiring from transmitter to control room	Transmits steam drum level from transmitter to control room	(1) Fails to open circuit	(1) Loose or broken wire	U	I1	(1) Same as 10.3-1 Safe failure		
				(2) Fails to short circuit	(2) Insulation failure	U	I1	(2) Same as 10.3-1 Safe failure		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

EXTREMELY UNLIKELY (EU) AN OFF-NORMAL CONDITION OF SUCH EXTREMELY LOW PROBABILITY THAT NO EVENT IN THIS CATEGORY IS EXPECTED TO OCCUR DURING THE PLANT LIFETIME BUT WHICH, NEVERTHELESS, REPRESENT EXTREME OR LIMITING CASES OF FAILURES WHICH ARE IDENTIFIED AS CONCEIVABLE.

CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

MINOR INCIDENT (MN) AN OCCURRENCE IN WHICH DAMAGE TO REPLACEABLE COMPONENTS IS NO GREATER THAN SOME SPECIFIED LOSS OF EFFECTIVE LIFETIME.

OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant (I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

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Table 7.2-13 Steam Drum Level Input - Failure Mode and Effects Analysis Criticality Analysis (Continued)

PART ASSEMBLY OR PROCESS Steam Drum Level Input (cont)				P/N Plant Protection System (PPS)						
ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBA-BILITY	CRITI-CALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONS-IBILITY DUE DATE
10	8	Steam drum level power supply	Provides power to transmitter electronics	(1) Fails to low or zero output	(1) Internal failure	A	I1	(1) Same as 10.3-1 Safe failure		
				(2) Fails to high output	(2) Transformer winding short circuit	U	I2	(2) Same as 10.1-1		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

EXTREMELY UNLIKELY (EU) AN OFF-NORMAL CONDITION OF SUCH EXTREMELY LOW PROBABILITY THAT NO EVENT IN THIS CATEGORY IS EXPECTED TO OCCUR DURING THE PLANT LIFETIME BUT WHICH, NEVERTHELESS, REPRESENT EXTREME OR LIMITING CASES OF FAILURES WHICH ARE IDENTIFIED AS CONCEIVABLE.

CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

MINOR INCIDENT (MN) AN OCCURRENCE IN WHICH DAMAGE TO REPLACEABLE COMPONENTS IS NO GREATER THAN SOME SPECIFIED LOSS OF EFFECTIVE LIFETIME.

OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant(I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

6487-1

Table 7.2-14 Primary Comparator - Failure Mode and Effects Analysis Criticality Analysis

PART ASSEMBLY OR PROCESS Primary Comparator				P/N Plant Protection System (PPS)						
ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBABILITY	CRITICALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONSIBILITY DUE DATE
11	1.	Primary Comparator	Compares input signal with fixed or calculated set point. When comparison is unfavorable, comparator is tripped to zero output. When comparison is favorable, output is positive (reset).	(1) Fails to tripped output	(1) Open or short circuited out-put or internal failure	U	I1	(1) Comparator transmits trip signal to 1 channel of each of 3 logic trains. Failure indicated by PPS Status Board. Safe failure		
				(2) Fails to reset output	(2) Internal Comparator failure	EU	I2	(2) Disables 1 channel in each of 3 logic trains. Failure indicated by periodic testing.		
	2.	Power Supply	Supplies DC voltage to comparators	(1) Fails to low or zero voltage	(1) Internal Failure (e.g. fuse, rectifier)	A	I1	(1) Same as 11.1-1 except all 24 of A, B, or C comparators fail safe failure. Indicated by PPS Status Board.		
				(2) Fails to high voltage	(2) Transformer primary winding short circuit.	U	I1	(2) Overvoltage monitor trips power supply off, results as 11.1-1. Safe failure		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

EXTREMELY UNLIKELY (EU) AN OFF-NORMAL CONDITION OF SUCH EXTREMELY LOW PROBABILITY THAT NO EVENT IN THIS CATEGORY IS EXPECTED TO OCCUR DURING THE PLANT LIFETIME BUT WHICH, NEVERTHELESS, REPRESENT EXTREME OR LIMITING CASES OF FAILURES WHICH ARE IDENTIFIED AS CONCEIVABLE.

CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

MINOR INCIDENT (MN) AN OCCURRENCE IN WHICH DAMAGE TO REPLACEABLE COMPONENTS IS NO GREATER THAN SOME SPECIFIED LOSS OF EFFECTIVE LIFETIME.

OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant (I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

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Table 7.2-14 Primary Comparator - Failure Mode and Effects Analysis Criticality Analysis (continued)

PART ASSEMBLY OR PROCESS: Primary Comparator (Continued)

P/N: Plant Protection System (PPS)

ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBA-BILITY	CRITI-CALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONS-IBILITY DUE DATE
11	3.	Primary Comparator Bypass Circuit	Provides input to bypass comparator	(1) Fails to bypass (no voltage)	(1) Open bypass circuit	U	I1	(1) No bypass of comparator. A false scram may result should another failure occur.		
				(2) Fails to unbypass (bypassing voltage inadvertently applied to comparator)	(2) Total short circuit around bypassing switches and permissives	EU	I2	(2) Failure prevents comparator trip. Failure indicated by bypass light.		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

EXTREMELY UNLIKELY (EU) AN OFF-NORMAL CONDITION OF SUCH EXTREMELY LOW PROBABILITY THAT NO EVENT IN THIS CATEGORY IS EXPECTED TO OCCUR DURING THE PLANT LIFETIME BUT WHICH, NEVERTHELESS, REPRESENT EXTREME OR LIMITING CASES OF FAILURES WHICH ARE IDENTIFIED AS CONCEIVABLE.

CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

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OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant(I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

6487-1

Table 7.2-15 Secondary Comparator - Failure Mode and Effects Analysis Criticality Analysis

PART ASSEMBLY OR PROCESS Secondary Comparator				P/N Plant Protection System (PPS)						
ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBABILITY	CRITICALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONSIBILITY DUE DATE
12	1.	Secondary Comparator	Compares input signal with fixed or calculated set point. When comparison is unfavorable, comparator trips to zero output. When comparison is favorable, output is positive (reset).	(1) Fails to tripped output	(1) Open or short circuited output or internal failure	U	I1	(1) Comparator outputs trip signal to 1 of 3 general coincidence logic channels. Failure annunciated by PPS status board. Safe failure		
				(2) Fails to reset output	(2) Internal comparator failure	EU	I2	(2) 1 out of 3 general coincidence logic channels for subsystem cannot trip. Failure indicated by periodic testing.		
	2.	Power Supply	Supplies DC voltage to secondary comparators	(1) Fails to low or zero voltage	(1) Internal failure (e.g. fuse or rectifier	A	I1	(1) Same as 12.1-1 except that all comparators connected to power supply are affected. Safe failure		
				(2) Fails to high voltage	(2) Transformer primary winding short circuit	U	I1	(2) Overvoltage monitor trips power supply off. Results same as 12.2-1. Safe failure		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

EXTREMELY UNLIKELY (EU) AN OFF-NORMAL CONDITION OF SUCH EXTREMELY LOW PROBABILITY THAT NO EVENT IN THIS CATEGORY IS EXPECTED TO OCCUR DURING THE PLANT LIFETIME BUT WHICH, NEVERTHELESS, REPRESENT EXTREME OR LIMITING CASES OF FAILURES WHICH ARE IDENTIFIED AS CONCEIVABLE.

CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

MINOR INCIDENT (MN) AN OCCURRENCE IN WHICH DAMAGE TO REPLACEABLE COMPONENTS IS NO GREATER THAN SOME SPECIFIED LOSS OF EFFECTIVE LIFETIME.

OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant(I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

6487-1

Table 7.2-15 Secondary Comparator - Failure Modes and Effects Analysis Criticality Analysis (continued)

PART ASSEMBLY OR PROCESS: Secondary Comparato r (Continued)				P/N: Plant Protection Systems (PPS)						
ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBABILITY	CRITICALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONSIBILITY DUE DATE
12	3.	Logic Signal Generator	The negatively saturated transformer core is unsaturated by the positive (reset) output from the comparator. An AC reset logic signal is then transformer coupled to the logic train. When the comparator trips to zero output, the transformer core remains saturated, the AC reset signal is not coupled to the logic train and a logic train trip results	(1) Fails to low zero voltage	(1) Internal failure	U	I1	(1) Reset signal is not properly coupled. 1 logic train trips. Safe failure. Indicated by PPS Status Board.		
				(2) Fails to high voltage	(2) Internal failure	EU	I1	(2) Same as 12.3-1 Safe failure		
				(3) Fails to low or zero frequency	(3) Internal failure	U	I2	(3) Depending on magnitude of change same as 12.3-1 or 12.1-2.		
				(4) Fails to high frequency	(4) Internal failure	U	I2	(4) Same as 12.3-3.		
				(5) D.C. transformer bias fails to low or zero voltage	(5) Internal failure	A	I1	(5) Same as 12.3-3		
				(6) D.C. transformer bias fails to high voltage	(6) Special bias supply failure	U	I1	(6) Same as 12.3-3		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

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CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

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OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant(I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

6487-1

Table 7.2-15 Secondary Comparator - Failure Modes and Effects Analysis Criticality Analysis (continued)

PART ASSEMBLY OR PROCESS: Secondary Comparator (Continued)				P/N: Plant Protection System (PPS)						
ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBA-BILITY	CRITI-CALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONS-IBILITY DUE DATE
12	4.	Coupling Transformer	Couples comparator output to logic train	(1) Change in core permeability	(1) External physical shock, application of voltage or current	EU	I2	(1) Same as 12.3-1. or 12.1-2		
	5.	Secondary Comparator Bypass Circuit	Provides input to bypass comparator	(1) Fails to bypass (no voltage)	(1) Open bypass circuit	U	I1	(1) No bypass of comparator. A false scram may result should another failure occur.		
				(2) Fails to unbypass (bypassing voltage inadvertently applied to comparator).	(2) Total short circuit around bypassing switches and permissives	EU	I2	(2) Failure prevents comparator trip. Failure indicated by bypass light.		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

EXTREMELY UNLIKELY (EU) AN OFF-NORMAL CONDITION OF SUCH EXTREMELY LOW PROBABILITY THAT NO EVENT IN THIS CATEGORY IS EXPECTED TO OCCUR DURING THE PLANT LIFETIME BUT WHICH, NEVERTHELESS, REPRESENT EXTREME OR LIMITING CASES OF FAILURES WHICH ARE IDENTIFIED AS CONCEIVABLE.

CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

MINOR INCIDENT (MN) AN OCCURRENCE IN WHICH DAMAGE TO REPLACEABLE COMPONENTS IS NO GREATER THAN SOME SPECIFIED LOSS OF EFFECTIVE LIFETIME.

OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant (I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

6487-1

Table 7.2-16 Primary Logic Train - Failure Mode and Effects Analysis Criticality Analysis

PART ASSEMBLY OR PROCESS Primary Logic Train				P/N Plant Protection System (PPS)						
ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBABILITY	CRITICALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONSIBILITY DUE DATE
13	1	Interrack wiring from comparator to primary logic	Transmits comparator output to photo transistor input	(1) Fails to open circuit	(1) Loose or broken wire	U	I1	(1) Trip signal to 1 logic train. Indicated by PPS Status Board. Safe failure		
				(2) Fails to short circuit	(2) Insulation failure	U	I1	(2) Same as 13.1-1 Safe failure		
	2	Photo transistor	Isolates instrument channel signals before combining them in 2 of 3 logic	(1) Output fails to trip state	(1) Photo transistor failure	U	I1	(1) Same as 13.1-1 Safe failure		
				(2) Output fails to reset state	(2) Hot short	EU	I2	(2) Single logic train will not trip. Failure indicated by PPS monitor.		
	3	Wiring from phototransistor to logic trains	Transmits output of photo transistor to logic train.	(1) Fails to open circuit	(1) Loose or broken wire	U	I1	(1) Same as 13.1-1 Safe failure		
				(2) Fails to short circuit	(2) Insulation failure	U	I1	(2) Same as 13.1-1 Safe failure		
	4	Primary Logic Train (includes 2/3 and 1/24 modules)	Combines comparator outputs in 2/3 logic coincidence	(1) Output fails to reset state	(1) Internal Failure	EU	I2	(1) Same as 13.2-2		
				(2) Output fails to trip state	(2) Internal failure	U	I1	(2) Same as 13.1-1 Safe failure		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

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CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

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OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant(I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

6487-1

Table 7.2-16 Primary Logic Train - Failure Mode and Effects Analysis Criticality Analysis (continued)

PART ASSEMBLY OR PROCESS: Primary Logic Train				P/N: Plant Protection System (PPS)						
ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBABILITY	CRITICALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONSIBILITY DUE DATE
13	5	Primary Logic Drivers	Amplifies signal from logic train to scram breaker UV coil	(1) Fails to low or zero output	(1) Internal failure	U	I1	(1) One channel of scram breakers trip. Failure indicated by PPS Status Board. Safe failure		
				(2) Fails to high output	(2) Transformer winding short circuit	EU	I2	(2) Same as 13.2-2.		
	6	Primary Logic Power Supplies	Provides necessary dc voltages for logic modules	(1) Fails to low or zero voltage	(1) Internal failure transformer, rectifier or fuse)	A	I1	(1) Same as 13.1-1 Safe failure		
				(2) Fails to high voltage	(2) Transformer winding short circuit	U	I1	(2) Over voltage monitor on power supply trips voltage off. Same results as 13.1-1 Safe Failure.		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

EXTREMELY UNLIKELY (EU) AN OFF-NORMAL CONDITION OF SUCH EXTREMELY LOW PROBABILITY THAT NO EVENT IN THIS CATEGORY IS EXPECTED TO OCCUR DURING THE PLANT LIFETIME BUT WHICH, NEVERTHELESS, REPRESENT EXTREME OR LIMITING CASES OF FAILURES WHICH ARE IDENTIFIED AS CONCEIVABLE.

CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

MINOR INCIDENT (MN) AN OCCURRENCE IN WHICH DAMAGE TO REPLACEABLE COMPONENTS IS NO GREATER THAN SOME SPECIFIED LOSS OF EFFECTIVE LIFETIME.

OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant (I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

6487-1

Table 7.2-17 Secondary Logic Train - Failure Mode and Effects Analysis Criticality Analysis

PART ASSEMBLY OR PROCESS: Secondary Logic Train

P/N: Plant Protection System (PPS)

ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBABILITY	CRITICALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONSIBILITY DUE DATE
14	1	Interrack writing from comparator to logic	Transmits comparator output to logic train	(1) Fails to open circuit	(1) Loose or broken wire	U	I1	(1) Trip signal to 1 logic train. Failure indicated by PPS Status Board. Safe failure		
				(2) Fails to short circuit	(2) Insulation failure	U	I1	(2) Same as 14.1-1 Safe failure		
	2	Secondary Logic Train (1/16 modules)	Combines like out puts from all 16 comparators	(1) Output fails to trip state	(1) Internal failure	U	I1	(1) 1 logic train trips. Failure indicated by PPS Status Board. Safe Failure		
				(2) Output fails to reset state	(2) Internal failure	EU	I2	(2) 1/16 incapable of scram. Indicated by periodic testing.		
	3	Logic driver	Amplifies necessary signal for final activation logic	(1) Fails to low or zero output	(1) Internal failure	U	I1	(1) Same as 14.2-1 Safe failure		
				(2) Fails to high output	(2) Transformer winding short circuit	EU	I2	(2) Same as 14.2-2		
	4	Secondary Logic Power Supplies	Provides necessary voltage for logic train	(1) Fails to low or zero voltage	(1) Internal failure (eg transformer, rectifier or fuse failure)	A	I1	(1) Same as 14.2-1 Safe failure		
				(2) Fails to high voltage	(2) Transformer winding short circuit	U	I1	(2) Overvoltage monitor on power supply trips voltage off. Same results as 14.2-1. Safe failure.		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

EXTREMELY UNLIKELY (EU) AN OFF-NORMAL CONDITION OF SUCH EXTREMELY LOW PROBABILITY THAT NO EVENT IN THIS CATEGORY IS EXPECTED TO OCCUR DURING THE PLANT LIFETIME BUT WHICH, NEVERTHELESS, REPRESENT EXTREME OR LIMITING CASES OF FAILURES WHICH ARE IDENTIFIED AS CONCEIVABLE.

CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

MINOR INCIDENT (MN) AN OCCURRENCE IN WHICH DAMAGE TO REPLACEABLE COMPONENTS IS NO GREATER THAN SOME SPECIFIED LOSS OF EFFECTIVE LIFETIME.

OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant(I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends ~~to cause~~ a component trip
I2 indicates a failure which tends to prevent a component trip

6487-1

Table 7.2-18 Primary Calculational Unit - Failure Mode and Effects Analysis Criticality Analysis

PART ASSEMBLY OR PROCESS: Primary Calculational Unit				P/N: Plant Protection System (PPS)						
ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBABILITY	CRITICALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONSIBILITY DUE DATE
15	1.	Primary Calculational Unit	Derives secondary trip parameter for comparator (e.g., flux-delayed flux, flux √pressure, primary-intermediate speed, steam-feed-water flow)	(1) Fails to low or zero output	(1) Internal failure	A	I2	(1) Failure may be safe or unsafe depending on the particular protective function considered. 1 channel affected.		
				(2) Fails to high output	(2) Internal failure	A	I2	(2) Same as 15.1-1		
	2.	Primary Calculational Unit Power Supply	Supplies power to primary calculational units	(1) Fails to low or zero voltage	(1) Internal failure	A	I2	(1) Failure may be safe or unsafe depending on the particular protective function considered. Failure affects all calculational units of train A, B, or C.		
				(2) Fails to high output	(2) Internal failure	U	I2	(2) Same as 15.2-1		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

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CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

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OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant (I): An occurrence which has no affect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

6487-1

Table 7.2-19 Secondary Calculational Unit - Failure Mode and Effects Analysis Criticality Analysis

PART ASSEMBLY OR PROCESS Secondary Calculational Unit				P/N Plant Protection System (PPS)						
ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBA-BILITY	CRITI-CALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONS-IBILITY DUE DATE
16	1.	Secondary Calculational Unit	Derives secondary trip parameter for comparator (e.g. modified nuclear rate, flux-total flow, primary-intermediate flow)	(1) Fails to low or zero output	(1) Internal failure	A	I2	(1) Failure may be safe or unsafe depending on the particular protective function considered. 1 channel affected.		
				(2) Fails to high output	(2) Internal failure	A	I2	(2) Same as 16.1-1.		
	2.	Secondary Calculational Unit Power Supply	Supplies power to secondary calculational units	(1) Fails to low or zero voltage	(1) Internal failure	A	I2	(1) Failure may be safe or unsafe depending on the particular function considered. Failure affects all calculational units of train A, B, and C.		
				(2) Fails to high voltage	(2) Internal failure	U	I2	(2) Same as 16.2-1		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

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CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

MINOR INCIDENT (MN) AN OCCURRENCE IN WHICH DAMAGE TO REPLACEABLE COMPONENTS IS NO GREATER THAN SOME SPECIFIED LOSS OF EFFECTIVE LIFETIME.

OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant (I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

6487-1

Table 7.2-20 Scram Actuation Logic - Failure Mode and Effects Analysis Criticality Analysis

PART ASSEMBLY OR PROCESS: Scram Actuation Logic

P/N: Plant Protection System (PPS)

ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBABILITY	CRITICALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONSIBILITY DUE DATE
17	1.	Wiring from Logic Drivers to UV Relays	Transmit voltage from logic driver to UV relays	(1) Fails to open circuit	(1) Loose or broken wire	U	I1	(1) One breaker channel trips. Safe failure		
				(2) Fails to short circuit	(2) Insulation failure	U	I1	(2) Same as 17.1-1 Safe failure		
	2.	Manual Scram Relays	Manual interruption to voltage on UV relays	(1) Fails to deenergized position	(1) Loose or broken wire	U	I1	(1) Same as 17.1-1 Safe failure		
				(2) Fails in energized position	(2) Insulation failure or welding of relay contacts	U	I1	(2) Manual scram will go to completion through redundant relays. Automatic scram not affected.		
	3.	Manual Scram Switches	Permit manual scram by operator	(1) Fails in scram position	(1) Mechanical failure	U	I1	(1) Reactor scram Safe failure		
				(2) Fails in reset position	(2) Mechanical failure	U	I1	(2) Failed switch cannot initiate reactor scram. Redundant scram switch not affected.		
	4.	Scram Breaker Undervoltage Coil	Trip actuator for scram breaker	(1) Fails to open circuit	(1) Loose or broken wire	U	I1	(1) Same as 17.1-1 Safe failure		
				(2) Fails to short circuit	(2) Insulation failure	U	I1	(2) Same as 17.1-1 Safe failure		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

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CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

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OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant (I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

6487-1

Table 7.2-21 HTS Shutdown Logic - Failure Mode and Effects Analysis Criticality Analysis

PART ASSEMBLY OR PROCESS: Heat Transport (HTS) Shutdown Logic

P/N: Plant Protection System (PPS)

ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBABILITY	CRITICALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONSIBILITY DUE DATE
18	1.	Auxiliary Contacts on Scram Breakers	Initiate HTS Shutdown on reactor scram	(1) Contacts fail open	(1) Poor electrical Contact	U	I1	(1) 2/3 contact openings are required for HTS shutdown. Safe failure		
				(2) Contacts fail closed	(2) Insulation breakdown	U	I2	(2) 1 channel cannot open. HTS Shutdown from other channel contacts.		
	2.	Shutdown Relays	Coil indicates scram breaker position	(1) Fails to de-energized position	(1) Relay coil open circuited (loose or broken wire)	U	I1	(1) Same as 18.1-1 Safe failure		
				(2) Fails to energized position	(2) Relay blocked	U	I2	(2) Same as 18.1-2		
	3.	HTS Breaker Trip Coils	Energizing either primary or secondary trip coil trips HTS breaker	(1) Fails to open circuit	(1) Loose or broken wire	U	I2	(1) One coil incapable of breaker trip. Secondary coil or redundant breaker must provide trip.		
				(2) Fails to short circuit	(2) Insulation failure	U	I2	(2) Same as 18.3-1		
	4.	Wiring from Shutdown Relay Panel to HTS Breaker Trip Coils	Transmits trip voltage signal to HTS shutdown breaker	(1) Fails to open circuit	(1) Loose or broken wire	U	I2	(1) Same as 18.3-1		
				(2) Fails to short circuit	(2) Insulation failure	U	I2	(2) Same as 18.3-1		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

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CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

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Insignificant (I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

6487-1

Table 7.2-21 HTS Shutdown Logic - Failure Mode and Effects Analysis Criticality Analysis (continued)

PART ASSEMBLY OR PROCESS HTS Shutdown Logic (Continued)				P/N Plant Protection System (PPS)						
ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBABILITY	CRITICALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONSIBILITY DUE DATE
18	5.	Trip Coil Voltage Supply	Supplies tripping voltage to HTS breaker trip coils	(1) Fails to low or zero voltage	(1) Internal failure	A	I2	(1) Same as 18.3-1		
				(2) Fails to high voltage	(2) Internal failure	U	I2	(2) Breaker would function properly for reasonably high overvoltages. Extremely high voltages results similar to 18.3-1		
	6.	13.8 KV HTS Breakers	Provides HTS shutdown on per loop basis on trip of PPS scram breakers	(1) One breaker fails open	(1) Defective breaker or trip coil	U	I1	(1) HTS prime mover coasts down. Reactor scram follows. Safe failure		
				(2) One breaker fails closed	(2) Mechanical failure or blockage. Both trip coils fail open	U	I2	(2) HTS shutdown is dependent upon redundant breaker (series breaker)		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

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CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

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Insignificant(I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

6487-1

Table 7.2-22 Control Rod Drive Mechanism (CRDM) Power - Failure Mode and Effects Analysis Criticality Analysis

PART ASSEMBLY OR PROCESS: Control Rod Drive Mechanism (CRDM) Power Train

P/N: Plant Protection System (PPS)

ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBA-BILITY	CRITI-CALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONS-IBILITY DUE DATE
19	1	480 V 3ϕ Power Supply	Power Supply for CRDM M-G sets	(1) Fails to low or zero voltage	Fault in power distribution system	A	I1	(1) Primary control rods will fall. Safe failure.		
				(2) Fails to high voltage	Primary winding short circuit in stepdown transformer	U	I1	(2) Distribution system over-voltage relays would trip power supply. Same results as 19.1.1 Safe failure.		
				(3) Less than 1/2 second reduction or outage in voltage	Lightning strike or other fault which is cleared by breaker trip and reclose	A	I1	(3) M-G sets are designed to ride through outage with no effect. Longer outages result as in 19.1.1.		
	2	M-G Set Starter	Starts and disconnects M-G set	(1) Fails open	Failure of main electrical contacts	U	I1	(1) M-G set coasts down. Same results as 19.1.1. Safe failure.		
				(2) Fails closed	Mechanical blockage	U	I1	(2) No effect on plant safety, although M-G set may be damaged. Safe failure		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

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CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

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OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant (I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

6487-1

Table 7.2-22 Control Rod Drive Mechanism (CRDM) Power - Failure Mode and Effects Analysis Criticality Analysis (continued)

PART ASSEMBLY OR PROCESS: Control Rod Drive Mechanism (CRDM) Power Train (Continued)

P/N: Plant Protection System (PPS)

ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBA-BILITY	CRITI-CALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONS-IBILITY DUE DATE
19	3	M-G Set	Supplies voltage for CRDM operation	(1) Fails to low or zero voltage	Regulator failure and decreasing field current.	A	I1	(1) Same as 19.1.1 Safe failure.		
				(2) Fails to high voltage	Regulator failure and increasing field current.	U	I1	(2) Overvoltage monitor annunciates high voltage and high machine temp. CRDM operates properly until stator windings burn out and rods fall. Safe failure.		
				(3) Output frequency variation	Frequency variation to motor of M-G set	U	I1	(3) Motor frequency variation tied to TVA system. Underfrequency relays will trip power supply if variations become significant Safe failure.		
	4	M-G Set Output Breaker	Provides overcurrent protection and disconnects M-G set.	(1) Fails open	Failure in trip mechanism	U	I1	(1) Same as 19.1-1 Safe failure		
				(2) Fails closed	Failure in breaker mechanism	EU	I1	(2) Scram capability not affected. Equipment electrical protection and control performance affected. Safe failure.		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

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CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

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Insignificant(I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

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Table 7.2-22 Control Rod Drive Mechanism (CRDM) Power - Failure Mode and Effects Analysis Criticality Analysis (continued)

PART ASSEMBLY OR PROCESS: Control Rod Drive Mechanism (CRDM) Power Train (Continued)				P/N: Plant Protection System (PPS)						
ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBA-BILITY	CRITI-CALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONS-IBILITY DUE DATE
19	5	Scram Breakers	Drop control rods by open circuiting power to CRDMs.	(1) Fails open	Worn trip latch or undervoltage coil fails open circuited.	U	I1	(1) Same as 19.1.1 Safe failure		
				(2) Fails closed	Mechanical blockage	U	I2	(2) Single scram breaker failure requires that the other two breaker trains open properly for HTS shutdown.		
				(3) Fails with long time response on opening.	Spring failure or friction buildup.	U	I2	(3) Same as 19.5-2.		
	6	Wiring from M-G set to Control Cabinet	Transmits generated voltage to control cabinet	(1) Fails to open circuit	Loose or broken wire	U	I1	(1) Same as 19.1-1.		
				(2) Fails to short circuit	Insulation failure	U	I1	(2) MG output breaker trips. Same results as 19.1-1. Safe failure		
	7	3 to 6 phase transformer	Converts power for SCRS and CRDMs.	(1) Fails to low or zero voltage	Transformer failure (e.g., open windings)	U	I1	(1) Same as 19.1-1 Safe failure		
				(2) Fails to high voltage	Primary to secondary short circuit	U	I1	(2) Same as 19.3.2 except there is no overvoltage monitor. Failure indicated by high temp. alarms. Safe failure		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

EXTREMELY UNLIKELY (EU) AN OFF-NORMAL CONDITION OF SUCH EXTREMELY LOW PROBABILITY THAT NO EVENT IN THIS CATEGORY IS EXPECTED TO OCCUR DURING THE PLANT LIFETIME BUT WHICH, NEVERTHELESS, REPRESENT EXTREME OR LIMITING CASES OF FAILURES WHICH ARE IDENTIFIED AS CONCEIVABLE.

CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

MINOR INCIDENT (MN) AN OCCURRENCE IN WHICH DAMAGE TO REPLACEABLE COMPONENTS IS NO GREATER THAN SOME SPECIFIED LOSS OF EFFECTIVE LIFETIME.

OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant (I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

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Table 7.2-22 Control Rod Drive Mechanism (CRDM) Power - Failure Mode and Effects Analysis Criticality Analysis (continued)

PART ASSEMBLY OR PROCESS: Control Rod Drive Mechanism (CRDM) Power Train (Continued)

P/N: Plant Protection System (PPS)

ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBABILITY	CRITICALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONSIBILITY DUE DATE
19	8	SCR Bank	Supplies DC voltage to 6 stator windings of one CRDM	(1) SCR fails to zero output	SCR fails open circuit	U	I1	(1) Control Rod fed from affected SCR bank will drop. Scram follows. Safe failure		
				(2) SCR fails turned on	SCR or sequencer failure	U	I1	(2) No safety effect Scram breakers can still drop rods. CRDM controller will sense this condition and stop rod motion.		
	9	Sequencer	Controls firing of SCRs to control rod speed and direction	(1) Fails so all outputs go to zero voltage	Loss of DC control power	U	I1	(1) Same as 19.8-1 Safe failure		
				(2) Fails so that extra outputs are turned on	Misfiring output circuit	U	I1	(2) Same as 19.8-2 Safe failure		
				(3) Runs at fast speed.	Drive motor overspeed	U	I1	(3) If rod motion exceeds 9"/min by more than 10%, rod motion is held. Maximum mechanical speed of CRDM is 70"/min. Safe failure		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

EXTREMELY UNLIKELY (EU) AN OFF-NORMAL CONDITION OF SUCH EXTREMELY LOW PROBABILITY THAT NO EVENT IN THIS CATEGORY IS EXPECTED TO OCCUR DURING THE PLANT LIFETIME BUT WHICH, NEVERTHELESS, REPRESENT EXTREME OR LIMITING CASES OF FAILURES WHICH ARE IDENTIFIED AS CONCEIVABLE.

CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

MINOR INCIDENT (MN) AN OCCURRENCE IN WHICH DAMAGE TO REPLACEABLE COMPONENTS IS NO GREATER THAN SOME SPECIFIED LOSS OF EFFECTIVE LIFETIME.

OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant(I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

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Table 7.2-22 Control Rod Drive Mechanism (CRDM) Power - Failure Mode and Effects Analysis Criticality Analysis (continued)

PART ASSEMBLY OR PROCESS Control Rod Drive Mechanism (CRDM) Power Train (Continued)				P/N Plant Protection System (PPS)						
ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBA-BILITY	CRITI-CALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONS-IBILITY DUE DATE
19	10	Wiring from Control Cabinet to CRDM	Transmits voltage from sequencer to CRDM	(1) Fails to open circuit	Loose or broken wire	U	I1	(1) Same as 18.8-1. Safe failure as scram breakers can still drop rods.		
				(2) Fails to short circuit	Insulation failure	U	I1	(2) Same as 19.8-1. Safe failure as scram breakers can still drop rods.		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

EXTREMELY UNLIKELY (EU) AN OFF-NORMAL CONDITION OF SUCH EXTREMELY LOW PROBABILITY THAT NO EVENT IN THIS CATEGORY IS EXPECTED TO OCCUR DURING THE PLANT LIFETIME BUT WHICH, NEVERTHELESS, REPRESENT EXTREME OR LIMITING CASES OF FAILURES WHICH ARE IDENTIFIED AS CONCEIVABLE.

CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

MINOR INCIDENT (MN) AN OCCURRENCE IN WHICH DAMAGE TO REPLACEABLE COMPONENTS IS NO GREATER THAN SOME SPECIFIED LOSS OF EFFECTIVE LIFETIME.

OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant(I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

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Table 7.2-23 PPS Voltage Signal Buffer - Failure Mode and Effects Analysis Criticality Analysis

PART ASSEMBLY OR PROCESS: PPS Voltage Signal Buffer				P/N: Plant Protection System (PPS)						
ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBABILITY	CRITICALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONSIBILITY DUE DATE
20	1	PPS Voltage Signal Buffer	Provides non-PPS output for all PPS voltage signal inputs. Buffer isolates PPS from other systems.	(1) Open circuit to buffer input	Loose or broken wire	U	I2	(1) One PPS channel fails. Indicated by interchannel comparison.	-	
				(2) Short circuit to buffer input	Insulation failure	U	I2	(2) Same as 20.1-1		
				(3) Short circuit, open circuit, or voltage or current source applied to buffer output	Various failures	U	I1	(3) PPS is not affected. Safe failure.		
				(4) Buffer failure	Internal failure	U	I2	(4) 1 PPS channel not properly isolated from external systems.		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

EXTREMELY UNLIKELY (EU) AN OFF-NORMAL CONDITION OF SUCH EXTREMELY LOW PROBABILITY THAT NO EVENT IN THIS CATEGORY IS EXPECTED TO OCCUR DURING THE PLANT LIFETIME BUT WHICH, NEVERTHELESS, REPRESENT EXTREME OR LIMITING CASES OF FAILURES WHICH ARE IDENTIFIED AS CONCEIVABLE.

CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

MINOR INCIDENT (MN) AN OCCURRENCE IN WHICH DAMAGE TO REPLACEABLE COMPONENTS IS NO GREATER THAN SOME SPECIFIED LOSS OF EFFECTIVE LIFETIME.

OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant (I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 indicates a failure which tends to prevent a component trip

6487-1

Table 7.2-24 PPS Current Signal Buffer - Failure Mode and Effects Analysis Criticality Analysis

PART ASSEMBLY OR PROCESS: PPS Current Signal Buffer				P/N: Plant Protection System (PPS)						
ITEM NO.	PART, ASSEMBLY OR PROCESS NUMBER	PART, ASSEMBLY OR PROCESS NAME	PART, ASSEMBLY OR PROCESS FUNCTION	FAILURE MODE(S)	FAILURE CAUSE(S)	PROBA-BILITY	CRITI-CALITY	FAILURE EFFECT(S)	CORRECTIVE ACTION OR PREVENTATIVE ACTION	RESPONS-IBILITY DUE DATE
21	1	Buffer Input Resistor	Provides voltage input to PPS buffer	(1) Buffer resistor open circuited	Resistor failure or broken lead	U	I1	(1) PPS input signal goes to zero. Safe failure.		
				(2) Buffer resistor short circuited	Resistor or insulation failure	U	I1	(2) PPS signal not affected. Safe failure.		
	2	PPS Input Resistor	Provides voltage input to PPS	(1) PPS resistor open circuited	Resistor failure or broken lead	U	I1	(1) Same as 21.1-1. Safe failure.		
				(2) PPS resistor short circuited	Resistor or insulation failure	U	I1	(2) Same as 21.1-1. Safe failure.		
	3	PPS Voltage Signal Buffer	Provides non-PPS output which isolates PPS from other systems	(1) Open circuit to buffer input	Loose or broken wire	U	I1	(1) Same as 21.1-1. Safe failure.		
				(2) Short circuit to buffer input	Insulation failure	U	I1	(2) Same as 21.1-2. Safe failure.		
				(3) Short circuit, open, open circuit, or voltage or current source applied to buffer output	Various failures	U	I1	(3) Same as 21.1-2. Safe failure.		
				(4) Buffer failure	Internal failure	U	I2	(4) 1 PPS channel not properly isolated from external systems.		

PROBABILITIES: ANTICIPATED (A) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY MAY BE EXPECTED TO OCCUR ONCE OR MORE DURING THE PLANT LIFETIME.

UNLIKELY (U) AN OFF-NORMAL CONDITION WHICH INDIVIDUALLY IS NOT EXPECTED TO OCCUR DURING THE PLANT LIFETIME; HOWEVER, WHEN INTEGRATED OVER ALL PLANT COMPONENTS AND SYSTEMS, EVENT IN THIS CATEGORY MAY BE EXPECTED TO OCCUR A NUMBER OF TIMES.

EXTREMELY UNLIKELY (EU) AN OFF-NORMAL CONDITION OF SUCH EXTREMELY LOW PROBABILITY THAT NO EVENT IN THIS CATEGORY IS EXPECTED TO OCCUR DURING THE PLANT LIFETIME BUT WHICH, NEVERTHELESS, REPRESENT EXTREME OR LIMITING CASES OF FAILURES WHICH ARE IDENTIFIED AS CONCEIVABLE.

CRITICALITIES: CRITICAL INCIDENT (C) THERE IS A SAFETY HAZARD.

MAJOR INCIDENT (MA) AN OCCURRENCE IN WHICH DAMAGE NECESSITATES REPLACEMENT OR REPAIR OF REPLACEABLE COMPONENTS OR AT LEAST EXTENSIVE INSPECTION OF PERMANENT COMPONENTS BEFORE RESUMING OPERATION OF THE FACILITY.

MINOR INCIDENT (MN) AN OCCURRENCE IN WHICH DAMAGE TO REPLACEABLE COMPONENTS IS NO GREATER THAN SOME SPECIFIED LOSS OF EFFECTIVE LIFETIME.

OPERATIONAL INCIDENT (OP) AN OCCURRENCE IN WHICH NO SIGNIFICANT LOSS OF EFFECTIVE LIFETIME FOR ANY COMPONENT OCCURS UNLESS FAILURE OF THE COMPONENT CONSTITUTES THE INITIATING EVENT.

Insignificant (I): An occurrence which has no effect on safety or operating performance.
For redundant components
I1 indicates a failure which tends to cause a component trip
I2 [illegible]

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7.3 ENGINEERED SAFETY FEATURE INSTRUMENTATION AND CONTROL

Engineered Safety Features (ESF) are identified in Section 6.1. Table 6.1-1 lists these features and the sections in which they are discussed. Accident analyses are presented in Section 15 for the postulated events requiring ESF protection.

7.3.1 Containment Isolation System

7.3.1.1 System Description

The Containment Isolation System (CIS) is comprised of redundant instrumentation which senses the need for closure of certain valves in lines directly connected to the containment atmosphere, logic to initiate closure of the valves, manual initiation equipment, and the valves which are described in Section 6.2.4. Figure 7.3-1 shows a block diagram of the system. The Containment Isolation System is designed for automatic activation of these valves in lines directly connected to the containment atmosphere and valves which require closure in less than 10 minutes to remain within limits. Where closure time is not required in less than 10 minutes, manual actuation is provided. Radiation sensors are provided in two areas: the exhaust duct of the containment ventilation and the head access area. Three independent, redundant measurements are provided at each location. The sensor output is conditioned by the electronics and transmitted to the comparators where it is compared with a setpoint. If the signal is greater than the setpoint, a comparator trip is initiated. The logic for the automatic containment isolation is functionally identical to that used for the Primary Reactor Shutdown System. The comparator output is optically coupled to the logic. Within the logic, the two comparator outputs (one from the head access area, the other from the exhaust duct of the containment ventilation) are combined to feed 2/3 coincidence modules. Within the 2/3 modules, the three independent channels are combined. Any two in the tripped state results in closure of the isolation valves. One of the logic trains drives the in-containment automatic valves. The other logic train drives the ex-containment automatic valves. Figure 7.3-2 is a Logic Diagram of the system.

Each detector has a check source which is used for test. In addition to the comparators and electronics, buffers and power suppliers are provided for each channel. No permissives or bypasses are provided.

The provisions for on-line testing of electrical and mechanical equipment are included in the design. The test source is used to test the instrument channel through the comparator. Signals are inserted prior to the 2/3 module to test the remainder to the logic and the closure of the automatic valves. Since closure of the valves does not require reactor shutdown, this test can be performed during power operation.

Channel output meters are included to provide the operators with early indication of anomalous instrumentation performance. This equipment is not safety related.

There are no interlocks included in the design nor is there a necessity for sequencing the closure of the valves.

Manual initiation capability is provided locally at the CIS breaker cabinets and within the control room on the main control board to close all CIS valves. Capability to close the CIS valves is provided if loss of access to control room is assumed.

For all containment isolation valves, position indicating lights are provided in the control room.

7.3.1.2 Design Basis Information

The CIS initiates and completes closure of the identified isolation valves to prevent the results of the faults identified in Table 7.3-1 from exceeding the specified limits. Note that these limits apply to the CIS when the containment hatch is closed. Further, the design basis for the CIS is to provide appropriate design margin for postulated events and to assure that the radiological consequences of such events are within the guideline valves of 10CFR100.

7.3.1.2.1 Containment Isolation System Subsystems

Containment Exhaust Radiation

The Containment Exhaust Radiation Subsystem initiates automatic containment isolation valve closure if the radiation level in the containment exhaust exceeds preset limits. This subsystem assures that events releasing activity within the containment do not result in exceeding the limits (10CFR20 or 10CFR100 as appropriate) for exposures in unrestricted areas. As shown in Figure 7.3-1, the subsystem includes 3 radiation sensors located in the containment exhaust whose output is compared to a fixed setpoint. The subsystem is never bypassed. Worst case values for time response and repeatibilities will be used in the final analysis of the performance of this subsystem. These monitors have the following parameters:

Range 10^{-7} - 10^{-2} μCi/cc
Accuracy $\pm$20%

Head Access Area Radiation

The Head Access Area Radiation Subsystem initiates closure of the containment isolation valves in the event of large radiation releases in the head access area. Three radiation sensors are located in the head access area to provide early initiation and closure of the isolation valves to assure that releases from design basis events do not exceed the guideline values of 10CFR100.

These monitors have the following parameters:

Range - 0.1 - 10^4 mR/hr
Accuracy - ±40%

7.3.1.2.2 Essential Performance Requirements

To implement the required isolation function within the specified limits, the CIS must meet the functional requirements specified below:

The closure time requirement for the inlet and exhaust isolation valves is 4 seconds with a three second or less detection time in the heating and ventilating system. A 10 second transport time from sensing point to the valve exists (see Section 15.1.1). The 3 seconds includes sensor time response, comparator and logic time delays.

The CIS is designed to meet these requirements for the environmental conditions described in Section 7.2.1.

7.3.2 Analysis

The design of the CIS provides the necessary design features to meet the functional and performance requirements as described below. The CIS logic is designed to conform to the IEEE Standards listed in Table 7.3-2.

7.3.2.1 Functional Performance

The analyses in Sections 15.5 and 15.6 shows the results of the postulated fault conditions. These analyses assumed a closed containment where the events occurred with the containment hatch closed. For the limiting event, primary drain tank fire during maintenance, scoping analyses have been performed to determine the required closure time of the containment isolation valves. For the primary drain tank fire, closure within 20 minutes is adequate. Further, analyses to determine the required closure time under postulated accident conditions have been performed and are discussed in Section 15.1.1. These analyses are used to determine the available design margin. The results of this assumed condition do not exceed the guideline values of 10CFR100 if the main exhaust and inlet valves are closed within 4 seconds assuming the normal air transport time from the detector to the valve is 10 seconds or more, a 14,000 Cfm normal ventilation rate.

Since the automatic Containment Isolation System is designed to isolate within the above time response requirements, all of the design basis conditions are terminated within the necessary limits for the present design concept.

7.3.2.2 Design Features

The CIS instrumentation, controls and actuators are designed to meet the requirements of IEEE-279-1971. The analyses of compliance with these are summarized below.

Single Failure

No single failure within the CIS nor removal from service of any component or channel will prevent protective action when required. There are three independent instrument channels for each necessary measurement, two independent 2/3 logics, and two independent actuators provided (as shown in Figure 7.3-1).

Bypasses

No bypasses are provided.

Multiple Setpoints

Multiple setpoints are not required.

Completion of Protective Action

The automatic CIS is designed so that, once initiated, protective action at the system level must go to completion. Return to normal operation requires manual reset of the CIS breakers by the operator.

Manual Initiation

The CIS includes means for manual initiation of containment isolation at the system level. No single failure will prevent manual initiation of the containment isolation action.

Control and Protection Interaction

There are no shared components between the control system and the CIS.

The provisions for access, information read-out, annunciation of trips, and periodic testing are as specified for the Reactor Shutdown System in Section 7.2.2.

Physical Separation

The following criteria assure physical separation for the CIS.

There will be at least one containment penetration for each of the three Primary PPS instrument channel conduits and each of the three Secondary PPS instrument channel conduits which exit containment. All requirements for separation of PPS wiring through conduits will also apply to separation of PPS wiring through containment penetrations.

There are three categories of CIS cabling: cables between the radiation monitoring sensors and logic panels; cabling between the logic panels and the power breakers; and cabling from the breakers to the valve actuators.

Wiring for the three CIS instrument channels will be routed exclusively with the three Secondary PPS instrument channels.

CIS logic train actuation wiring will be routed through two separated and independent conduits. A conduit will contain only wiring from a single CIS logic train. No intermixing of CIS logic trains within a conduit will be permitted. CIS logic train 1 wiring will be routed from CIS logic panel 1 to CIS breaker 1. CIS logic train 2 wiring will be routed from CIS logic panel 2 to CIS breaker 2.

All of the inside containment isolation valve actuation wiring (both manual and automatic) will be routed through at least one separated and independent conduit from CIS breaker 1 through a separate and independent containment isolation valve actuation containment penetration. Inside containment isolation valve actuation wiring will be routed through separate and independent conduits from the inside of the containment isolation valve actuation containment penetration to the individual containment isolation valves. No other wiring will be routed through the conduit and containment penetration containing inside containment isolation valve actuation wiring.

All of the outside containment isolation valve actuation wiring (both manual and automatic) will be routed through at least one separated and independent conduit from CIS breaker 2 to the individual outside containment isolation valves. No other wiring will be routed through the conduit containing outside containment isolation valve actuation wiring.

Information Read-Out

Indicators and alarms are provided as an operating aid and to keep the plant operator informed of the CIS status. All indicators and alarms are not safety related. The following items are located on the Main Control Panel for operator information.

Analog Indication

A. Head Access Area Radioactivity
B. Containment Exhaust Radioactivity

Indicating Lights

A. CIS Breaker Trip/Reset Status
B. CIS Isolation Valve Position

Annunciators

A. Head Access Area High Radiation
B. Containment Exhaust High Radiation

7.3.3 ESF Atmosphere Clean-up Systems

The ESF Atmosphere Clean-up Systems are as follows:

a) RCB Annulus Filtration System

b) RSB Filtration System

c) Control Habitability System

These system are described in the following sections.

7.3.3.1 Design Criteria

The following design criteria are applicable to the ESF Atmosphere Clean-up Instrumentation and Control Systems:

A. Compliance with CRBRP General Design Criterion 11 as listed in Section 3.1.3.

B. Class 1E power supply, backed up by Diesel Generators, provide power to all components.

C. No single failure of an instrument, interconnecting cable or panel will prevent a key process variable from being controlled or monitored in redundant divisions.

D. Physical and electrical separation of redundant portions of the system is provided.

E. Manual initiation of each protective action is provided at the system level.

F. Instrumentation used in the control of ESF Atmosphere Clean-up Systems will function during and after an SSE.

G. Instrumentation used in the control of ESF Atmosphere Clean-up Systems will function during normal environmental conditions and during environmental conditions created by any design basis accident.

H. Capabilities for periodic testing and calibration of all instruments are provided.

I. Capabilities for remote operation of the RCB Annulus Filtration and RSB Filtration System are provided, should the Control Room become uninhabitable.

J. Capabilities are provided to monitor the bypass or inoperable status of components in accordance with NRC Regulatory Guide 1.47.

7.3.3.2 Monitoring Instrumentation

A. Typically, process variables are monitored and indication is provided in the Control Room and locally as follows (also refer to P&IDs in Section 9.6):

1. All valve and damper positions.

2. All fan operation status.

3. Filter unit humidity.

4. Filter unit discharge air flow (also recorded in the Control Room).

5. Filter unit adsorbent filter entering air and leaving air temperature (local only).

6. Filter unit differential pressure across each filter bank and across entire unit (local only).

7. Annulus to atmosphere differential pressure.

8. RSB confinement to atmosphere differential pressure.

9. Control Room HVAC unit supply air flow (also recorded in the Control Room).

10. Control Room HVAC unit supply air flow (also recorded in the Control Room).

11. Outside air inlet temperature.

12. Control Room main and remote air intake radioactivity level for radioactive gases, iodines and particulates.

13. Control Room differential pressure.

B. Typically, process variables are monitored and alarm annunciation is provided in the Control Room and locally as follows:

1. Motor thermal overload, high vibration or air flow low for each fan.

2. Differential pressure high across each filter bank and across entire filter unit.

3. Unit cooler or HVAC unit supply air temperature high or air temperature entering cooling coil low.

4. Smoke, ammonia, chlorine, hydrogen fluoride or radiation present in Control Room main or remote air intake.

5. Control switch in the local mode (Control Room only alarm only).

C. Typically, process variables are provided as inputs to the Plant Data Handling & Display System as follows:

1. Control Room and Computer Room humidity.

2. Annulus differential pressure.

3. RSB confinement differential pressure (four different cells).

4. Control Room differential pressure.

5. Air temperature entering and leaving each filter unit.

6. Air temperature entering and leaving each HVAC unit.

7. Inoperable or bypass status of components.

7.3.3.3 <u>Design Analysis</u>

The ESF Atmosphere Clean-up Systems are designed to meet the requirements of Section 7.1 and the IEEE Standards listed in Table 7.3-3 with the exception of alarm circuits and inputs to the Plant Data Handling and Display System (PDH&DS) which are non-Class 1E circuits.

7.3.3.4 <u>RCB Annulus Filtration System</u>

7.3.3.4.1 <u>Design Basis</u>

The RCB Annulus Filtration System is designed to ensure that an acceptable upper limit of release of radioactive material is not exceeded under the site suitability source term conditions.

7.3.3.4.2 <u>System Description</u>

Initiation and control of the RCB Annulus Filtration System is represented by Figures 7.3-3 through 7.3-9. The following initiation and control operations are provided:

1. Maintain the containment/confinement annulus space under 1/4 inch water gauge negative pressure with respect to outside atmosphere during normal plant operation and accident conditions.

2. Provide filtering of the annulus exhaust during normal operation.

3. Provide filtering of the RCB ventilation exhaust air through the annulus filter system during RCB/RSB open hatch refueling operations.

4. Initiate filtering and recirculating of the annulus air during accident conditions.

The RCB Annulus Filtration System is also described in Section 9.6.2.2.4.

7.3.3.5 Reactor Service Building (RSB) Filtration System

7.3.3.5.1 Design Basis

The RSB Filtration System is designed to filter the RSB exhaust air in order to mitigate the consequences of the RSB Radiation Release Term (SSST) event.

7.3.3.5.2 System Description

Initiation and control of the RSB Filtration System is represented by Figures 7.3-10 through 7.3-13. The following initiation and control operations shall be provided:

1. Filter RSB exhaust during all plant conditions.

2. Initiate RSB confinement, recirculation and filtering during accident conditions.

3. Maintain the RSB confinement at under 1/4 inch water gauge negative pressure with respect to outside atmosphere.

The RSB Filtration System is also decribed in Section 9.6.3.1.1.

7.3.3.6 Control Room Habitability System

7.3.3.6.1 Design Basis

The design basis for the Control Room Habitability System is detailed in Section 6.3.1.1.

7.3.3.6.2 System Description

A detailed description of the Control Room Habitability System is presented in Section 6.3.

Figures 7.3-14 through 7.3-24 represent initiation and control of the Control Room Habitability System. A summary of the functions performed by this system is as follows:

a) Maintain the Control Room at positive pressure to minimize the infiltration of radioactive or chemical contamination.

b) Initiate Control Room isolation when SGB ARMS signal is present, or when smoke or toxic gases are present at either the remote or main air intake, or radiation is above a fixed setpoint at the main air intake.

c) Initiate Control Room recirculation/filtration mode when containment isolation is initiated, when Control Room isolation is initiated or by manual initiation.

7.3.4 Steam Generator Building Aerosol Mitigation Release System (ARMS)

7.3.4.1 System Description

A description of the system and design basis is included in Section 6.2.7. Figures 7.3 (to be provided later) represent initiation and control of the SGB ARMS.

7.3.4.2 Design Criteria

The following design criteria are applicable to the SGB ARMS:

A. Compliance with CRBRP General Design Criterion 11 as listed in Section 3.1.3.

B. Class 1E power supply, backed up by Diesel Generators, provide power to all system components.

C. No single failure of an instrument, interconnecting cable or panel will prevent a key process variable from being controlled or monitored in redundant divisions.

D. Physical and electrical separation of redundant portions of the system is provided.

E. Manual initiation of each protective action is provided at the system level.

F. Instrumentation used in the control of ARMS system will function during and after an SSE.

G. Instrumentation used in the control of ARMS systems will function during normal environmental conditions and during environmental conditions created by any design basis accident.

H. Capabilities for periodic testing and calibration of all instruments are provided.

I. Capabilities are provided to monitor the bypass or inoperable status of components in accordance with NRC Regulatory Guide 1.47.

7.3.4.3 Design Analysis

The SGB ARMS is designed to meet the requirements of Section 7.1 and the IEEE Standards listed in Table 7.3-3.

TABLE 7.3-1

CONTAINMENT ISOLATION SYSTEM DESIGN BASIS

Event	Applicable Federal Regulation	Limit
Anticipated Fault	10CFR20 § 105	<2 millirem in any one hour
No examples of anticipated faults which lead to release of activity have been identified.		<100 millirem in any one week
Unlikely Fault	10CFR20 § 403b	<5 rem in any two hours
No examples are presently identified for the automatic containment isolation system design basis.		
Extremely Unlikely Faults & Design Margin*	10CFR100	<25 rem in any two hours
Examples include major sodium fires		<300 rem iodine doses in the thyroid in any two hours
		<75 rem to the lung
44		<150 rem to the bone

*The design basis for the CIS includes limiting the results of postulated accidents within the guideline values of 10CFR100. See Section 15.1.1.

TABLE 7.3-2

LIST OF IEEE STANDARDS APPLICABLE TO
THE CONTAINMENT ISOLATION SYSTEM LOGIC

IEEE-279-1971	IEEE Standard: Criteria for Protection Systems for Nuclear Power Generating Stations
IEEE-308-1974	Criteria for Class 1E Power Systems for Nuclear Power Generating Stations
IEEE-317-1976	Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations
IEEE-323-1974	IEEE Trial-Use Standard: General Guide for Qualifying Class 1E Electric Equipment for Nuclear Power Generating Stations
IEEE-323-A-1975	Supplement to the Forward of IEEE-323-1974
IEEE-336-1971	IEEE Standard: Installation, Inspection and Testing Requirements for Instrumentation and Electric Equipment During Construction of Nuclear Power Generating Stations
IEEE-338-1977	IEEE Trial-Use Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems
IEEE-344-1975	IEEE Standard 344-1975, IEEE Recommended Practices for Seismic Qualification of Class 1 Equipment for Nuclear Power Generating Stations
IEEE-352-1975	IEEE Guide for General Principles for Reliability Analysis of Nuclear Power Generating Station Protection Systems
IEEE-379-1972	IEEE Trial-Use Guide for the Application of the Single Failure Criterion to Nuclear Power Generating Station Protection Systems
IEEE-384-1974	IEEE Trial-Use Standard Criteria for Separation of Class 1E Equipment and Circuits
IEEE-494-1974	IEEE Standard Method for Identification of Documents Related to Class 1E Equipment and Systems for Nuclear Power Generating Station

TABLE 7.3-3

LIST OF IEEE STANDARDS APPLICABLE TO
ESF CLEAN-UP AND AEROSOL RELEASE MITIGATION SYSTEMS

IEEE-279-1971	IEEE Standard: Criteria for Protection Systems for Nuclear Power Generating Stations
IEEE-308-1974	Criteria for Class 1E Power Systems for Nuclear Power Generating Stations
IEEE-323-1974	IEEE Trial-Use Standard: General Guide for Qualifying Class 1E Electrical Equipment for Nuclear Power Generating Stations
IEEE-338-1977	Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems
IEEE-379-1972	IEEE Trial-Use Guide for the the Application of the Single Failure Criterion to Nuclear Power Generating Station Protection Systems
IEEE-383-1974	IEEE Standard for Type Test of Class 1E Electric Cables, Field Splices and Connections for Nuclear Power Generating Stations
IEEE-384-1974	IEEE Trial-Use Standard: Criteria for Separation of Class 1E Equipment and Circuits

TABLE 7.3-4

SYMBOLS

△A	ALARM
△IS	INOPERABLE STATUS MONITORING
Ⓡ	RED IND LITE
Ⓖ	GREEN IND LITE
Ⓦ	WHITE IND LITE
△C	COMPUTER INPUT

NOTES:

1) Control switches are spring return to auto from start with a maintained stop unless otherwise stated.

ABBREVIATIONS

SSPLS	- Solid State Programmable Logic System
CR	- Control Room (remote)
L	- Local (not Control Room)
T.D.	- Time delay
N.C.	- Normally closed
F.C.	- Fail closed
S.O.V.	- Solenoid operated valve
A.O.V.	- Air operated valve
MOD	- Motor operated damper
ZS	- Position switch
CIS	- Containment isolation signal
PPS	- Plant Protection System
E/H	- Electro-hydraulic

TABLE 7.3-4 Continued

TE	-	Temperature element
TT	-	Temperature transmitter
TIC	-	Temperature indicating controller
OAI	-	Outside air intake
TMD	-	Temperature modulated damper
RA	-	Return air
PDI	-	Pressure differential indicator
PDC	-	Pressure differential controller
PMD	-	Pressure modulated damper
PDISH	-	Pressure differential indicating switch high
FR	-	Flow recorder
FIC	-	Flow indicating controller
FSL	-	Flow switch low
FT	-	Flow transmitter
FMD	-	Flow modulated damper
FE	-	Flow element
M	-	Moisture
PB	-	Pushbutton
MUX	-	Multiplexing
AHU	-	Air handling unit

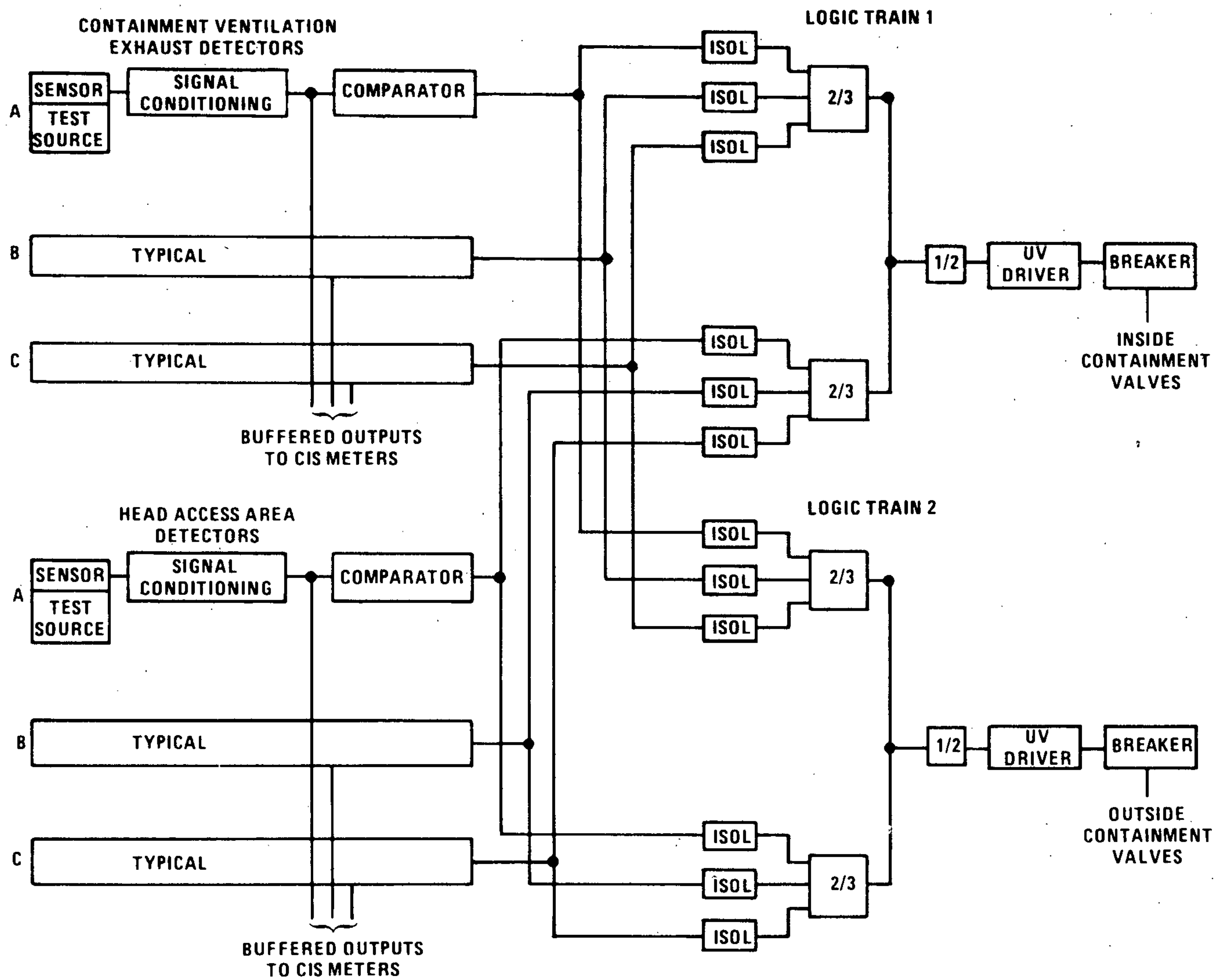

Figure 7.3-1. Containment Isolation System Block Diagram

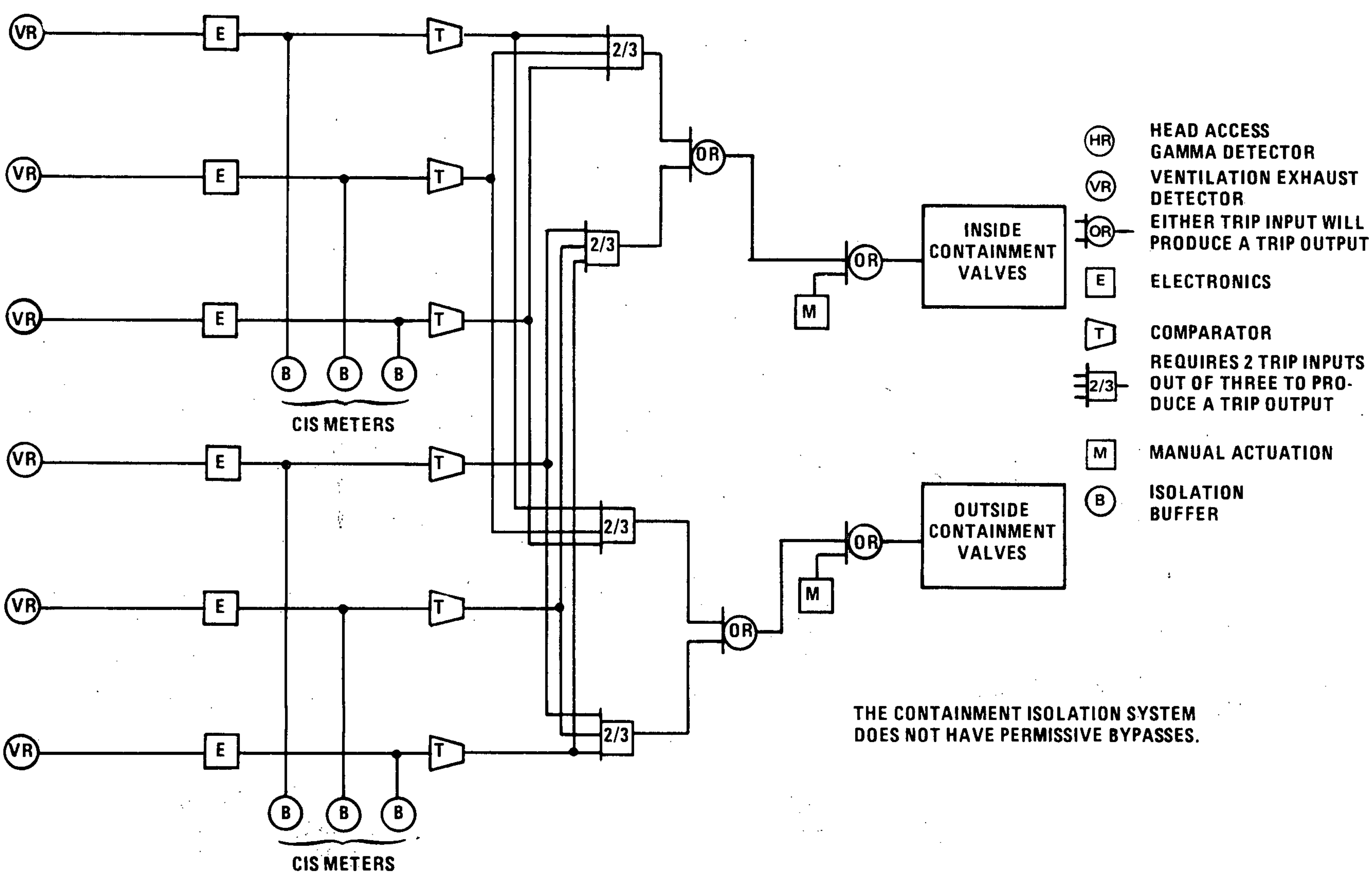

Figure 7.3-2. Containment Isolation System Logic Diagram

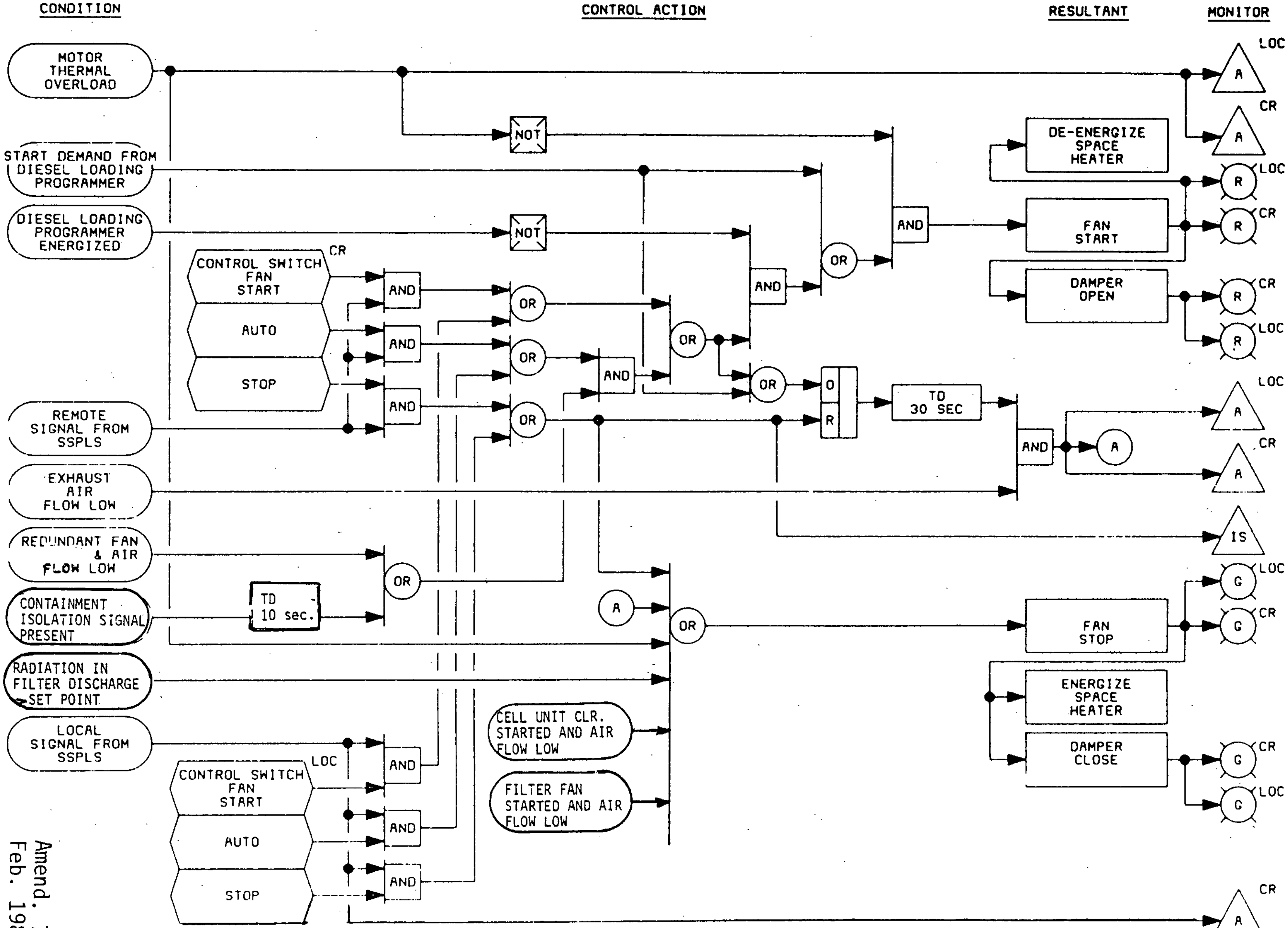

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FIGURE 7.3-3
FUNCTIONAL CONTROL DIAGRAM - ANNULUS PRESSURE MAINTENANCE FAN

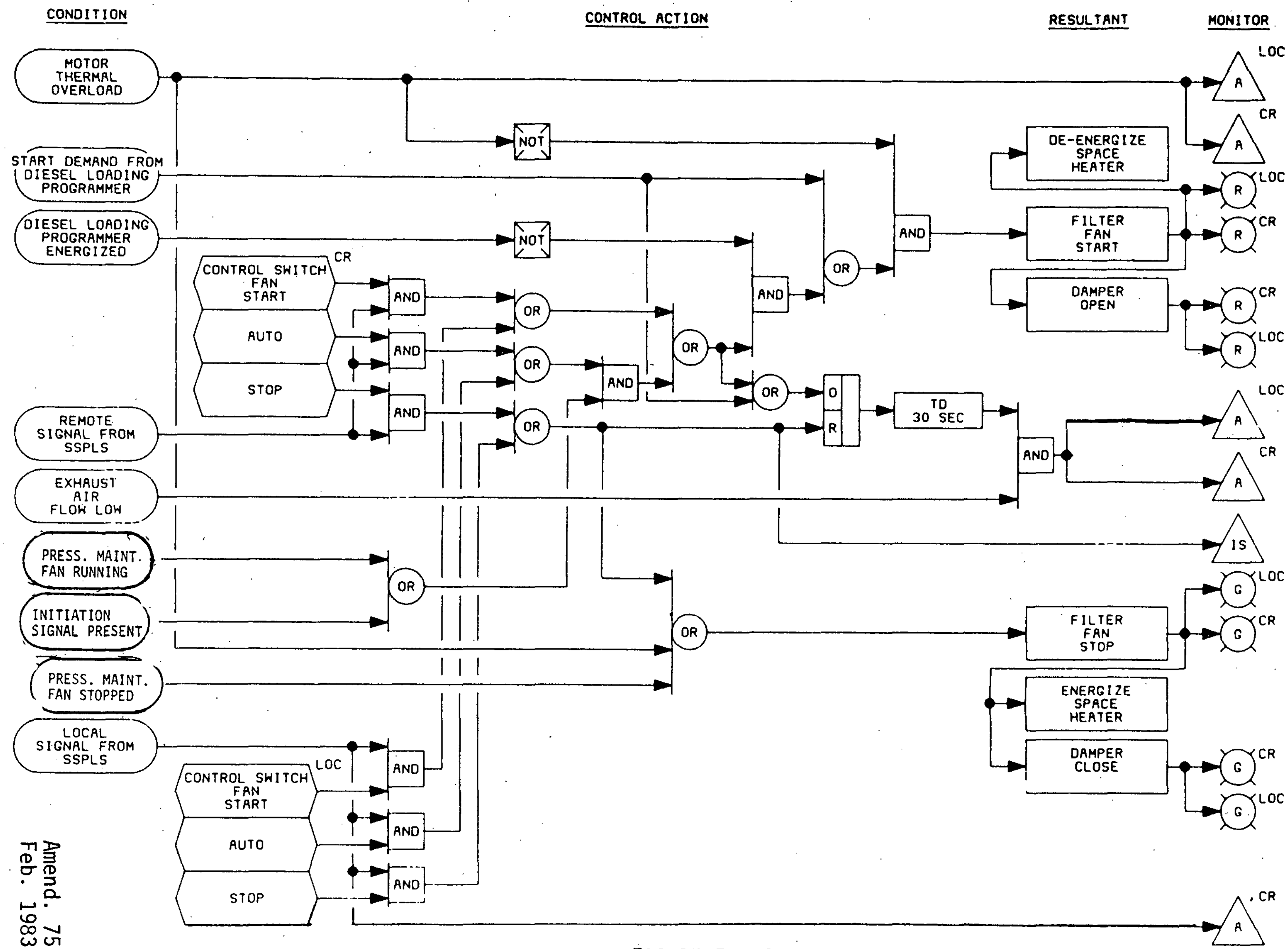

FIGURE 7.3-4
FUNCTIONAL CONTROL DIAGRAM ANNULUS FILTER FAN

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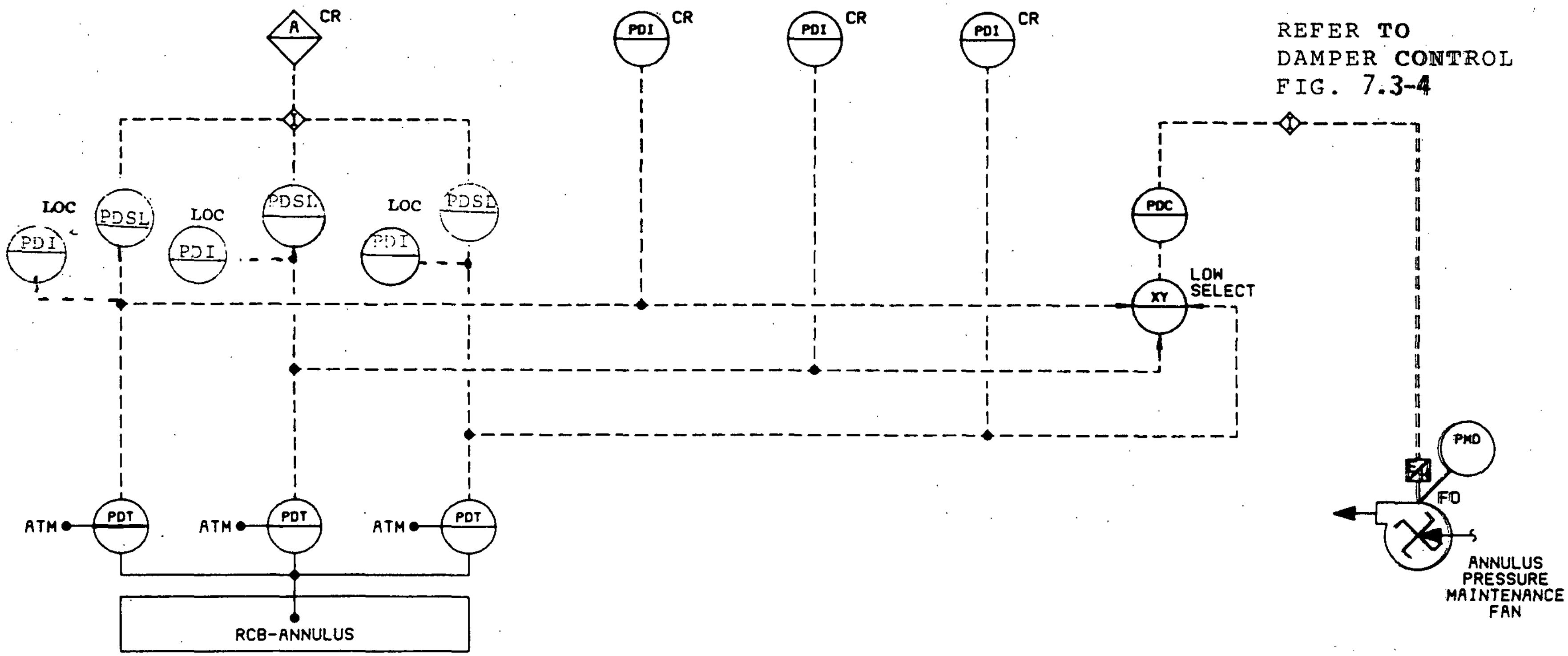

FIGURE 7.3-5
FUNCTIONAL CONTROL DIAGRAM ANNULUS
PRESSURE MAINTENANCE FAN INLET VANE MODULATION

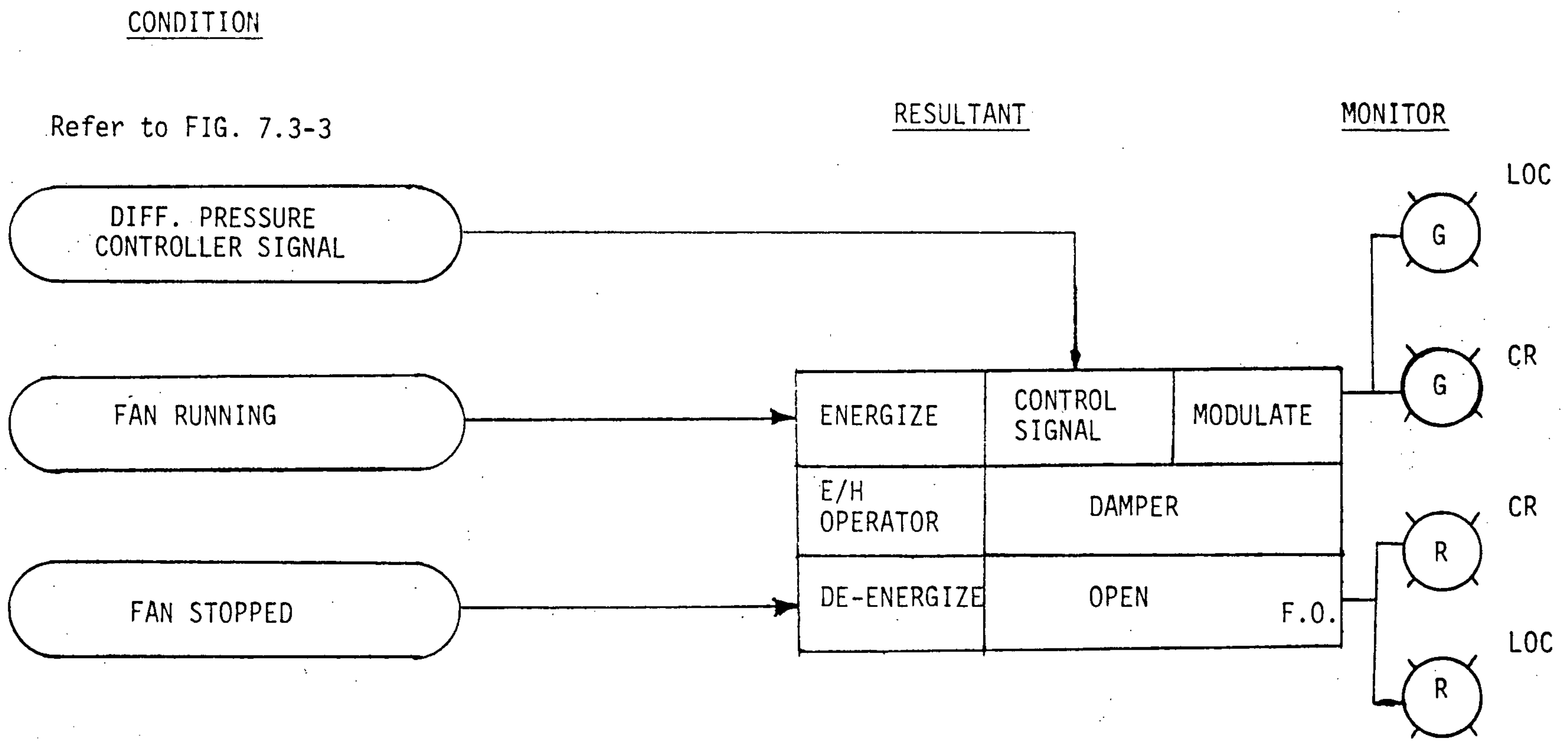

FIGURE 7.3-6
FUNCTIONAL CONTROL DIAGRAM ANNULUS PRESSURE MAINTENANCE
FAN INLET VANE MODULATION

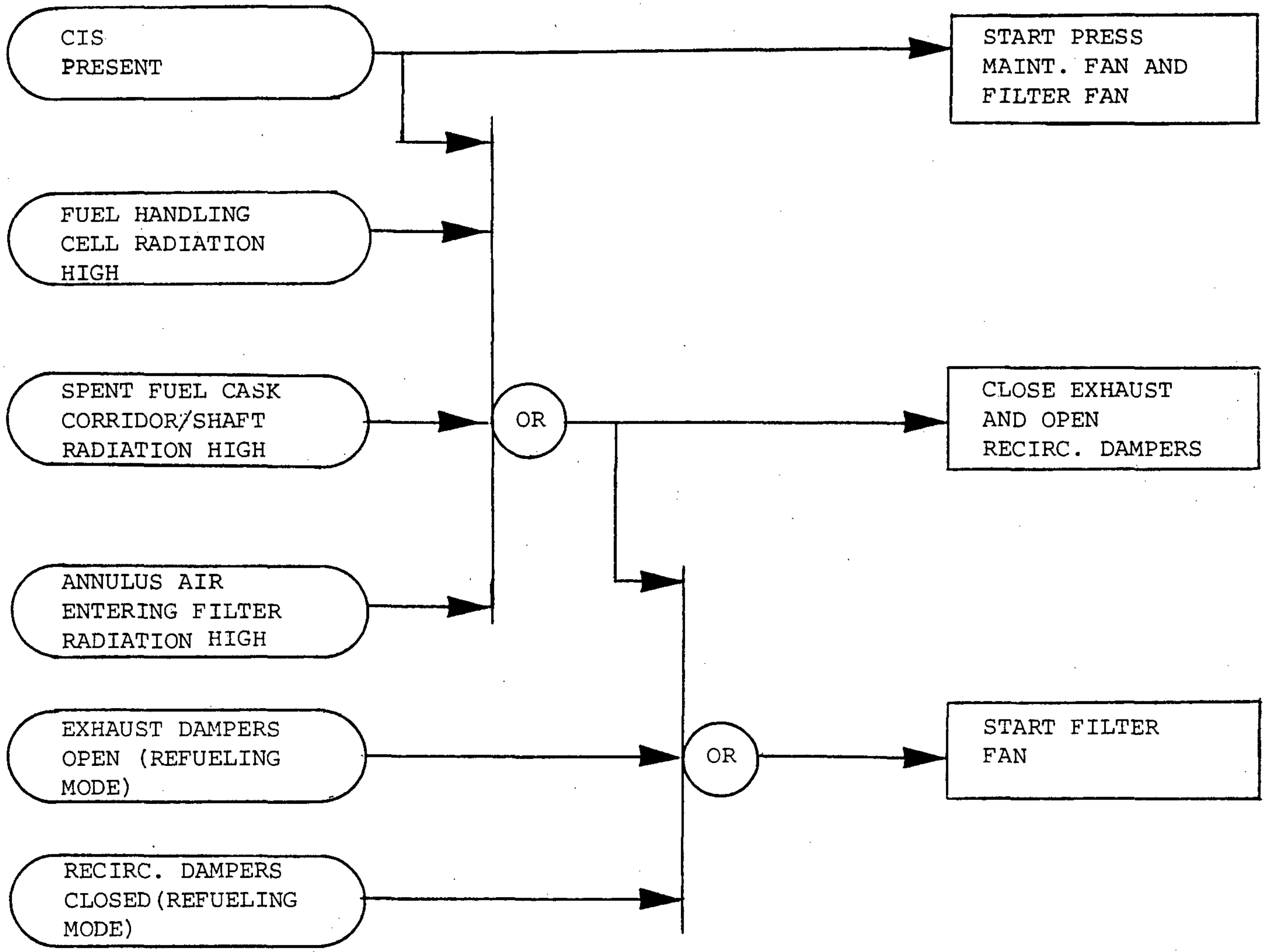

Figure 7.3-7

FUNCTIONAL CONTROL DIAGRAM-ANNULUS FILTRATION SYSTEM, FILTRATION AND/OR RECIRCULATION INITIATION

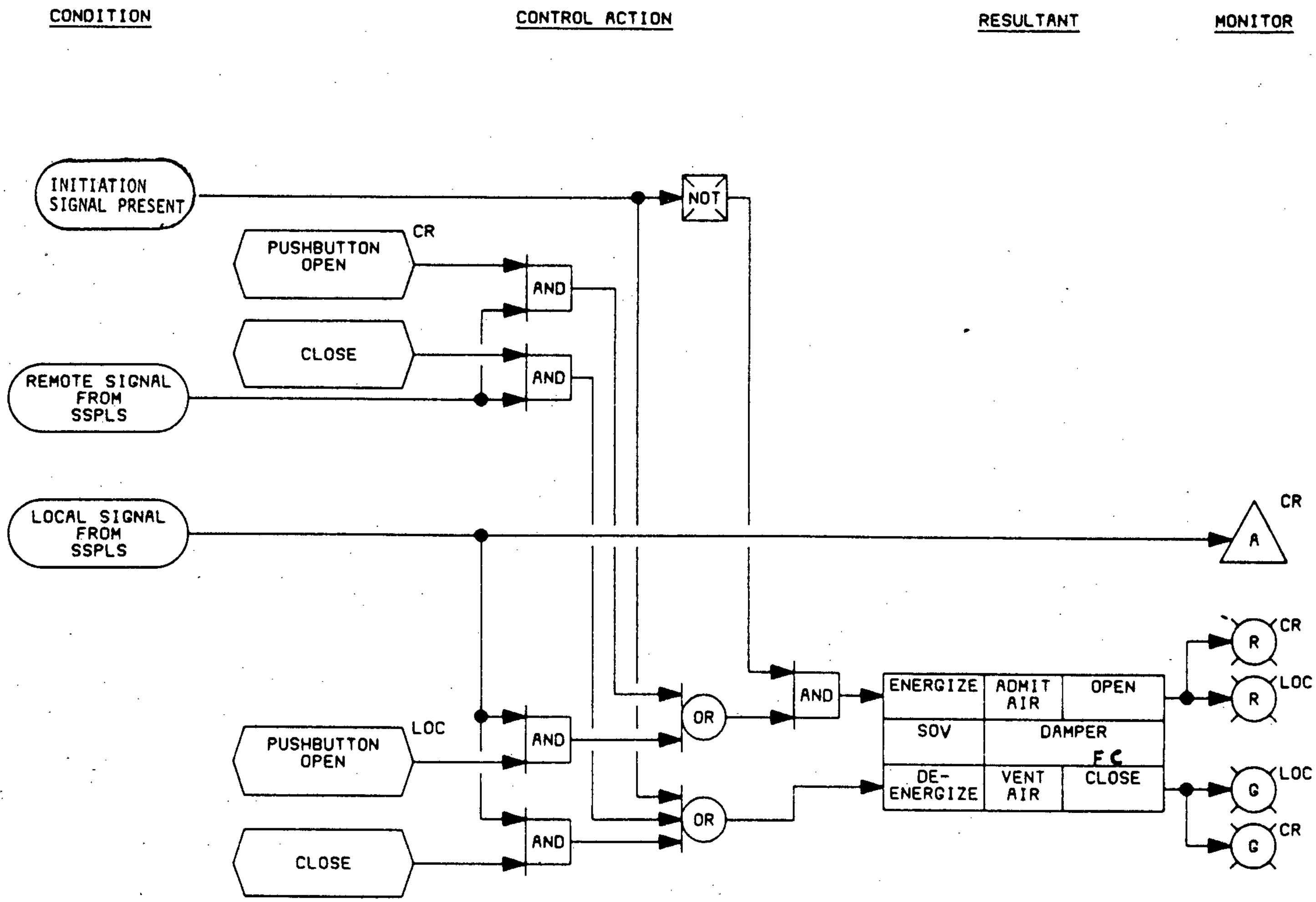

FIGURE 7.3-8
FUNCTIONAL CONTROL DIAGRAM ANNULUS
FILTRATION EXHAUST DAMPERS

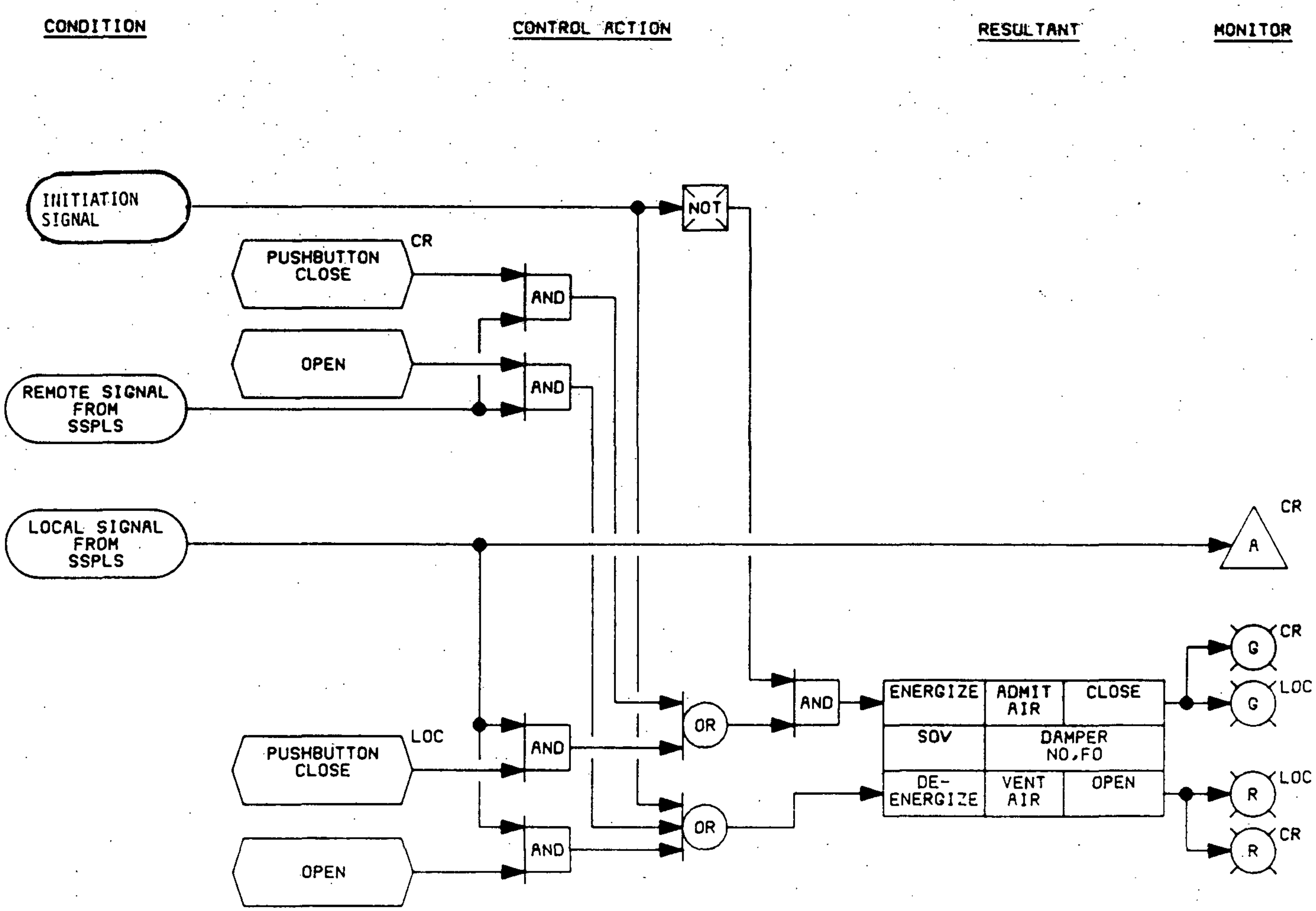

FIGURE 7.3-9
FUNCTIONAL CONTROL DIAGRAM ANNULUS
FILTRATION RECIRCULATION DAMPERS

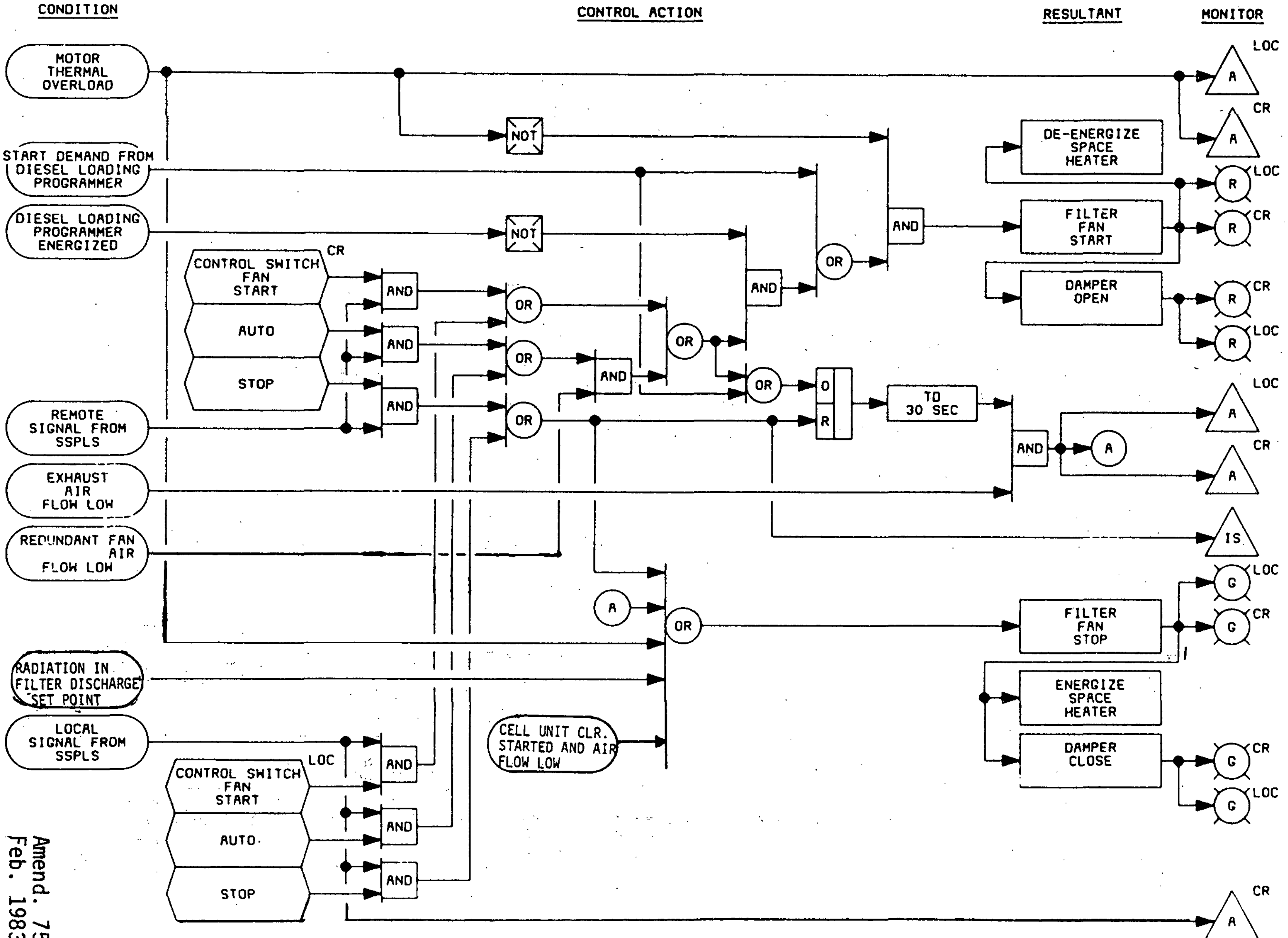

FIGURE 7.3-10
FUNCTIONAL CONTROL DIAGRAM RSB FILTRATION SYSTEM FILTER SUPPLY FAN

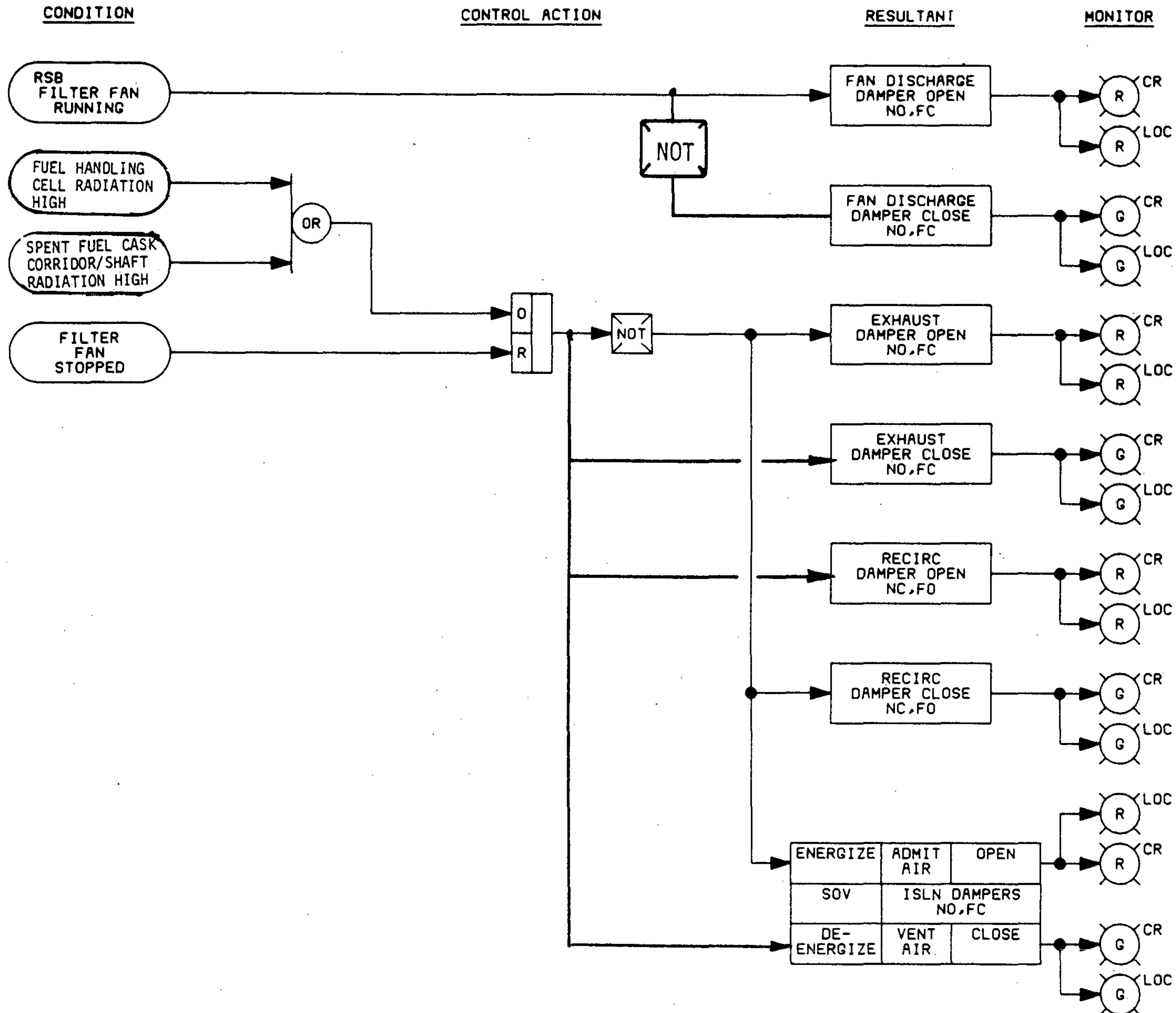

FIGURE 7.3-11
FUNCTIONAL CONTROL DIAGRAM RSB FILTRATION SYSTEM
DISCHARGE, EXHAUST, RECIRCULATION & CELL ISOLATION DAMPERS

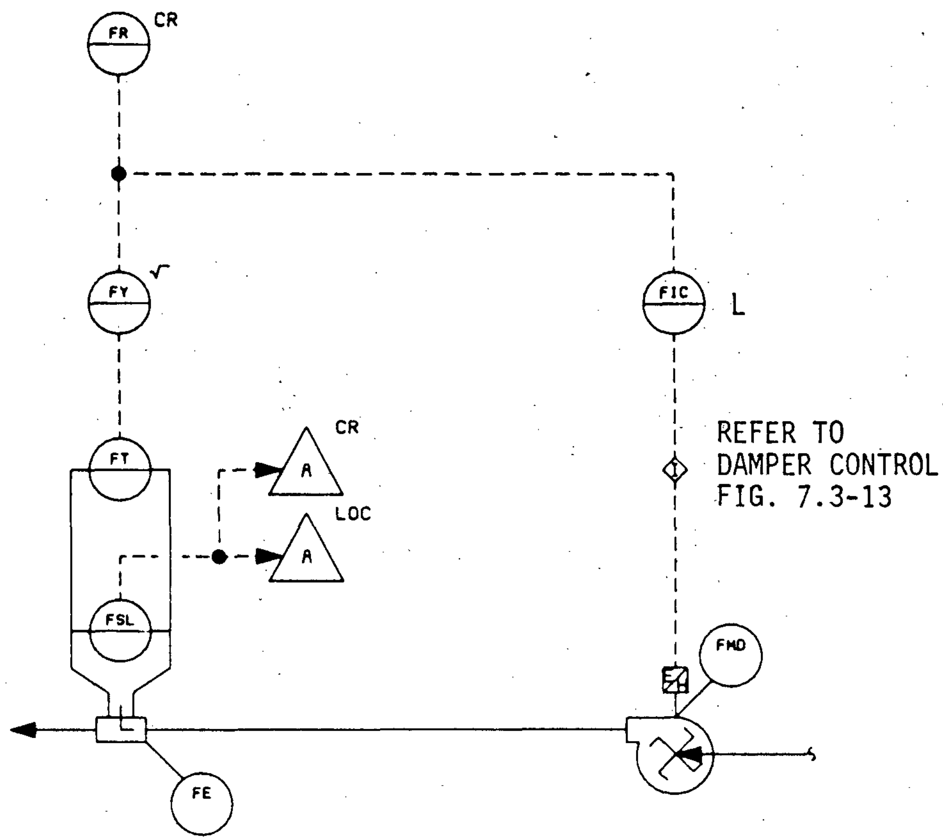

FIGURE 7.3-12
FUNCTIONAL CONTROL DIAGRAM RSB FILTER FAN INLET VANE MODULATION

CONDITION

Refer to FIG. 7.3-12

RESULTANT

MONITOR

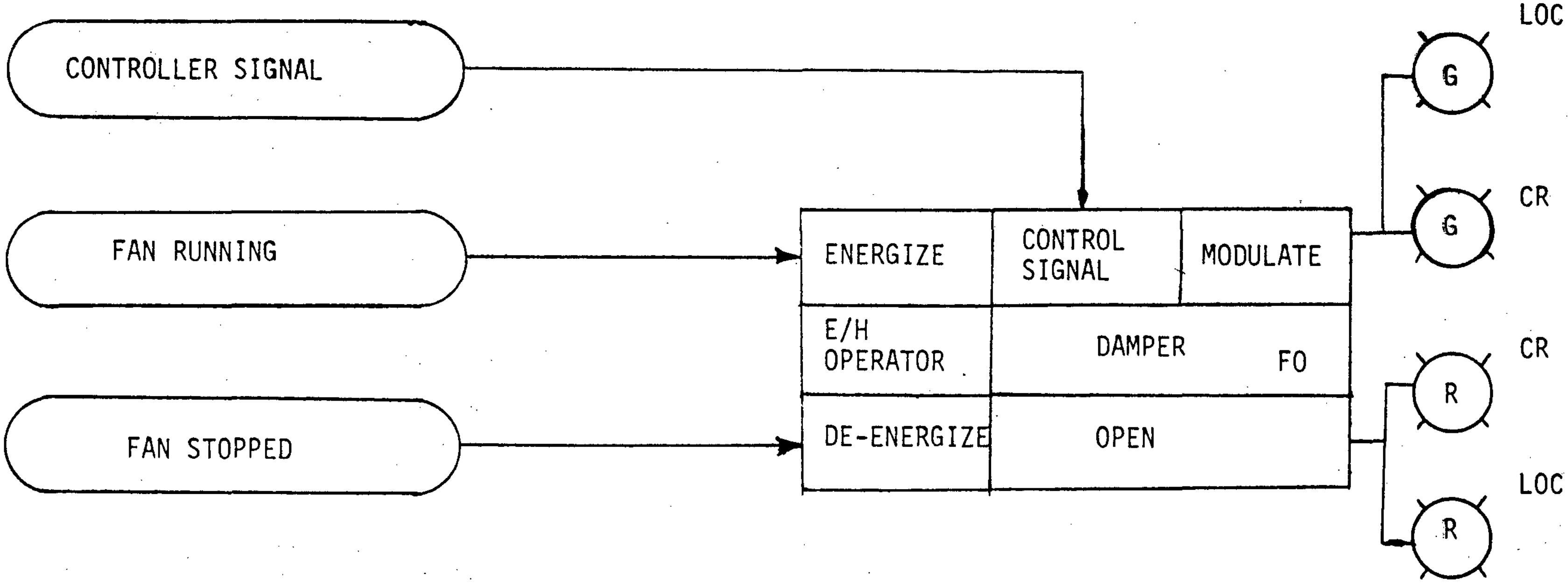

FIGURE 7.3-13
FUNCTIONAL CONTROL DIAGRAM RSB FILTER FAN INLET VANE MODULATION

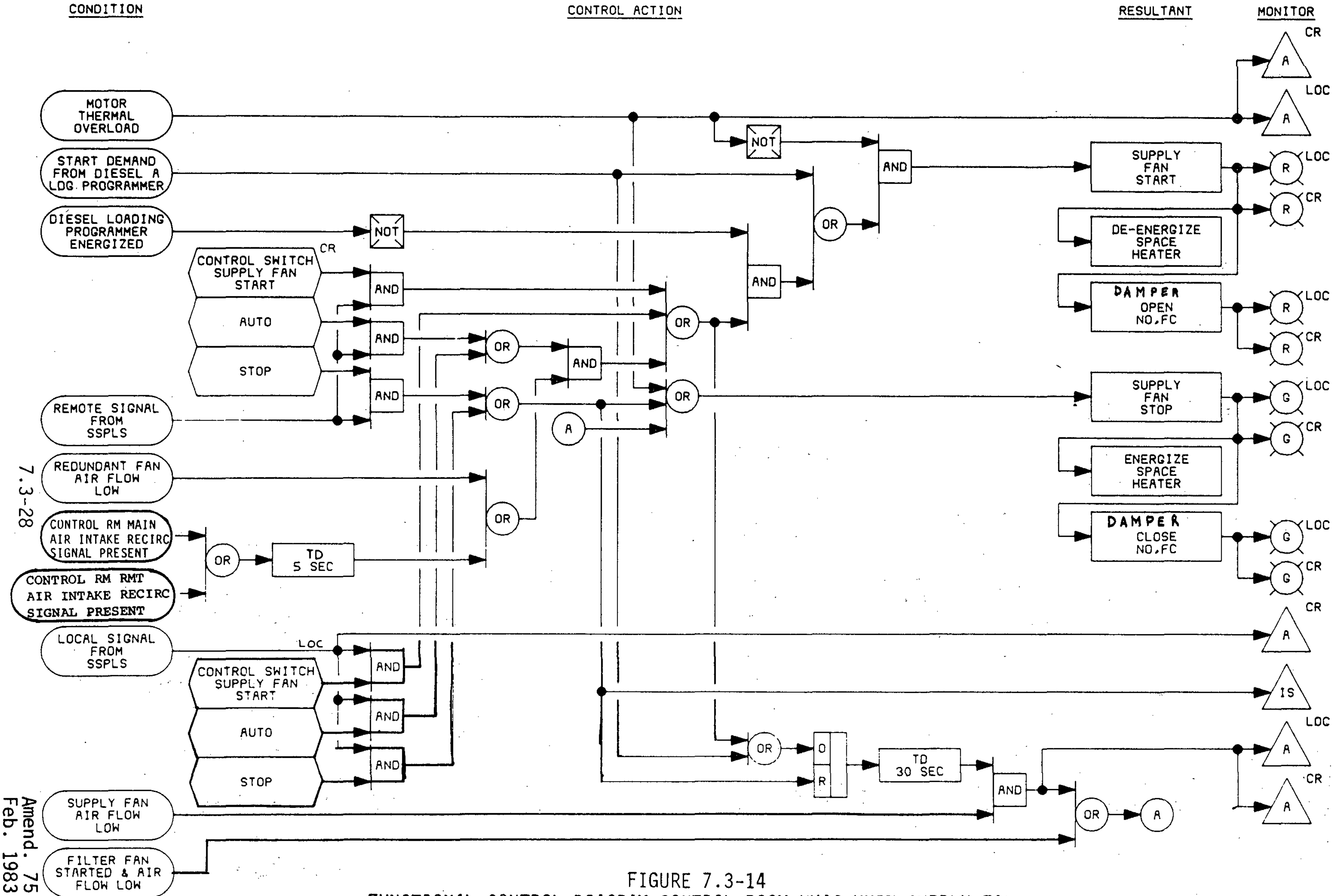

FIGURE 7.3-14
FUNCTIONAL CONTROL DIAGRAM CONTROL ROOM HVAC UNIT SUPPLY FAN

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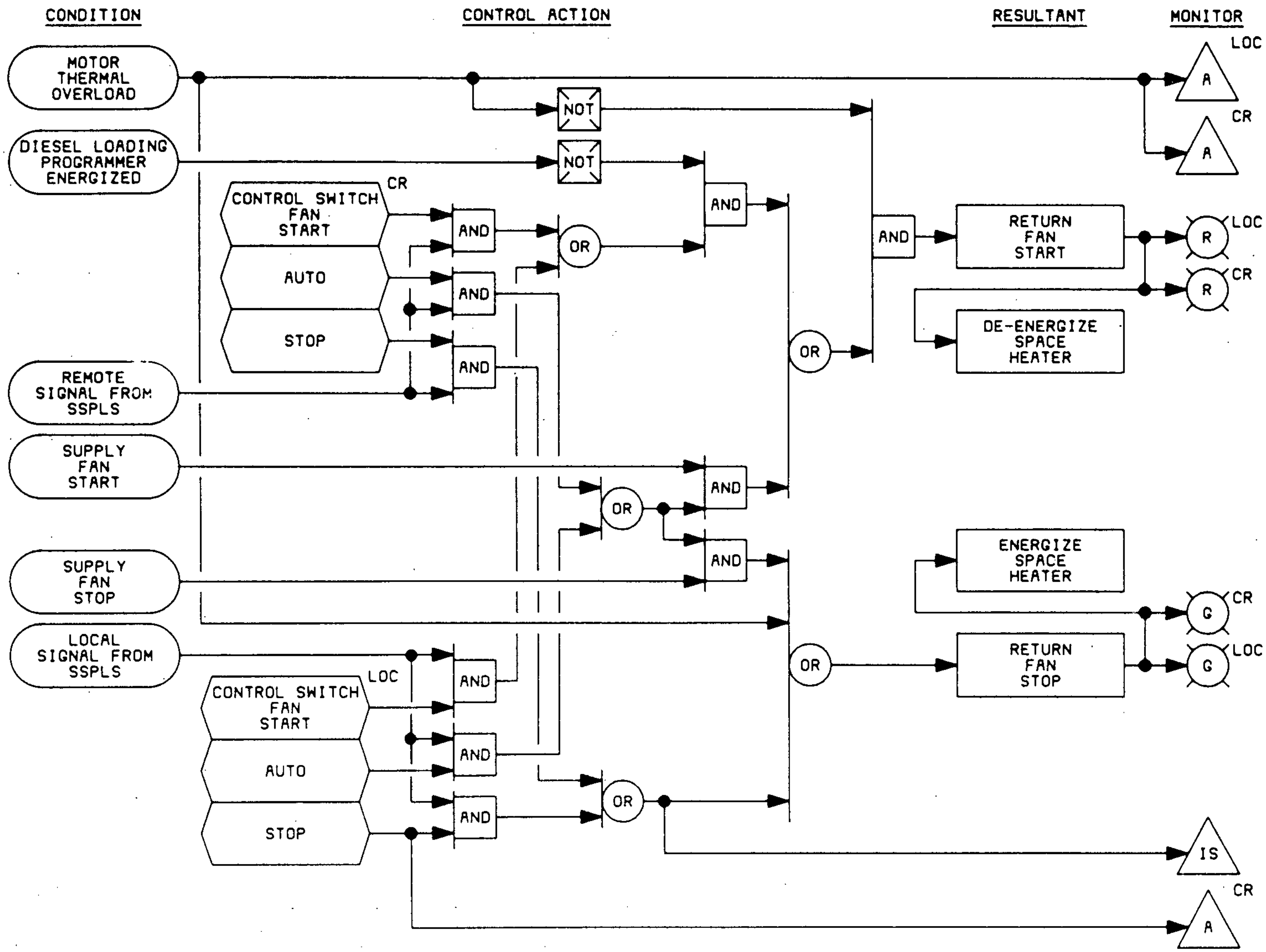

FIGURE 7.3-15
FUNCTIONAL CONTROL DIAGRAM CONTROL ROOM HVAC UNIT RETURN FAN

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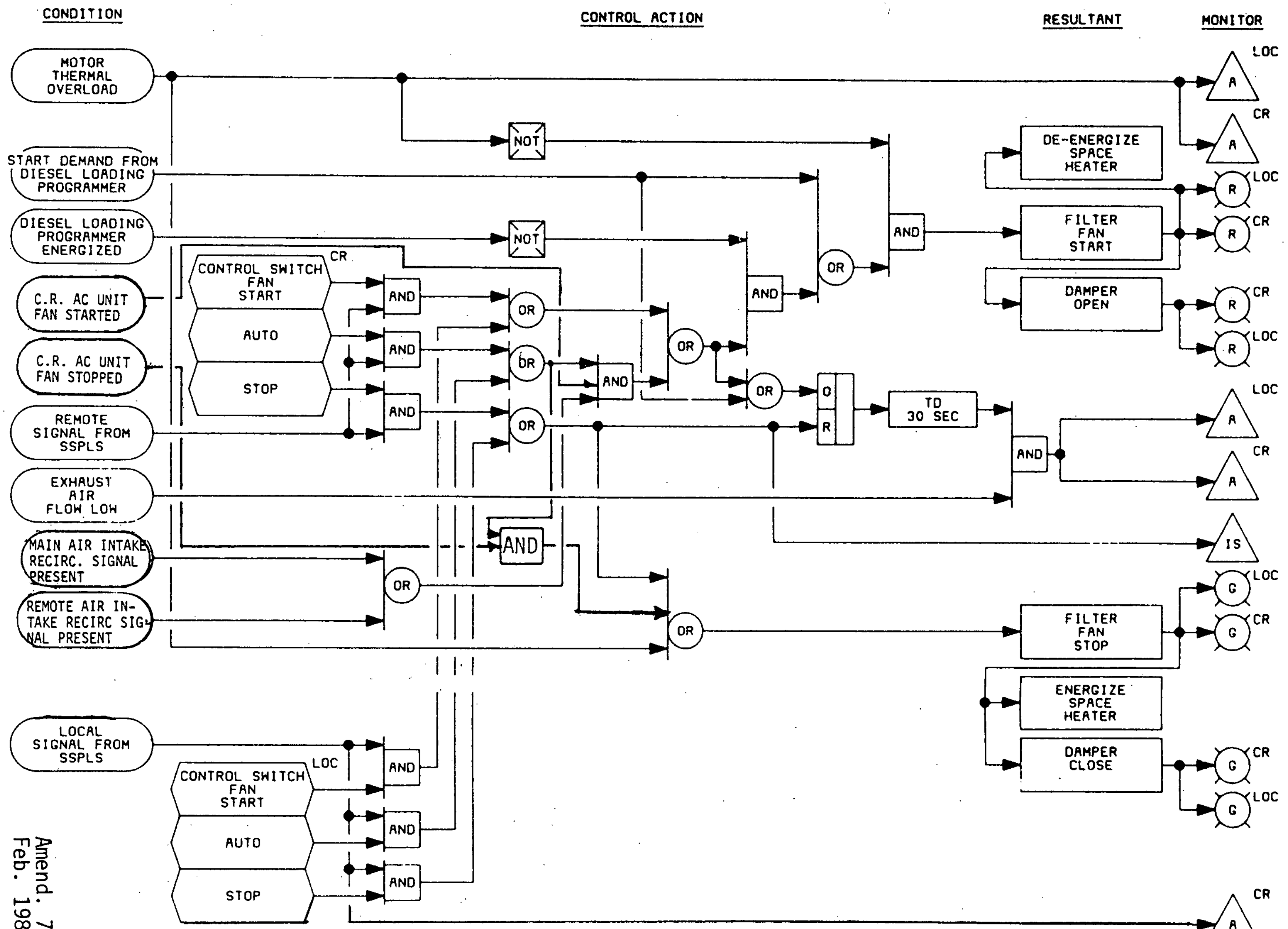

FIGURE 7.3-16
FUNCTIONAL CONTROL DIAGRAM CONTROL ROOM FILTER UNIT SUPPLY FAN

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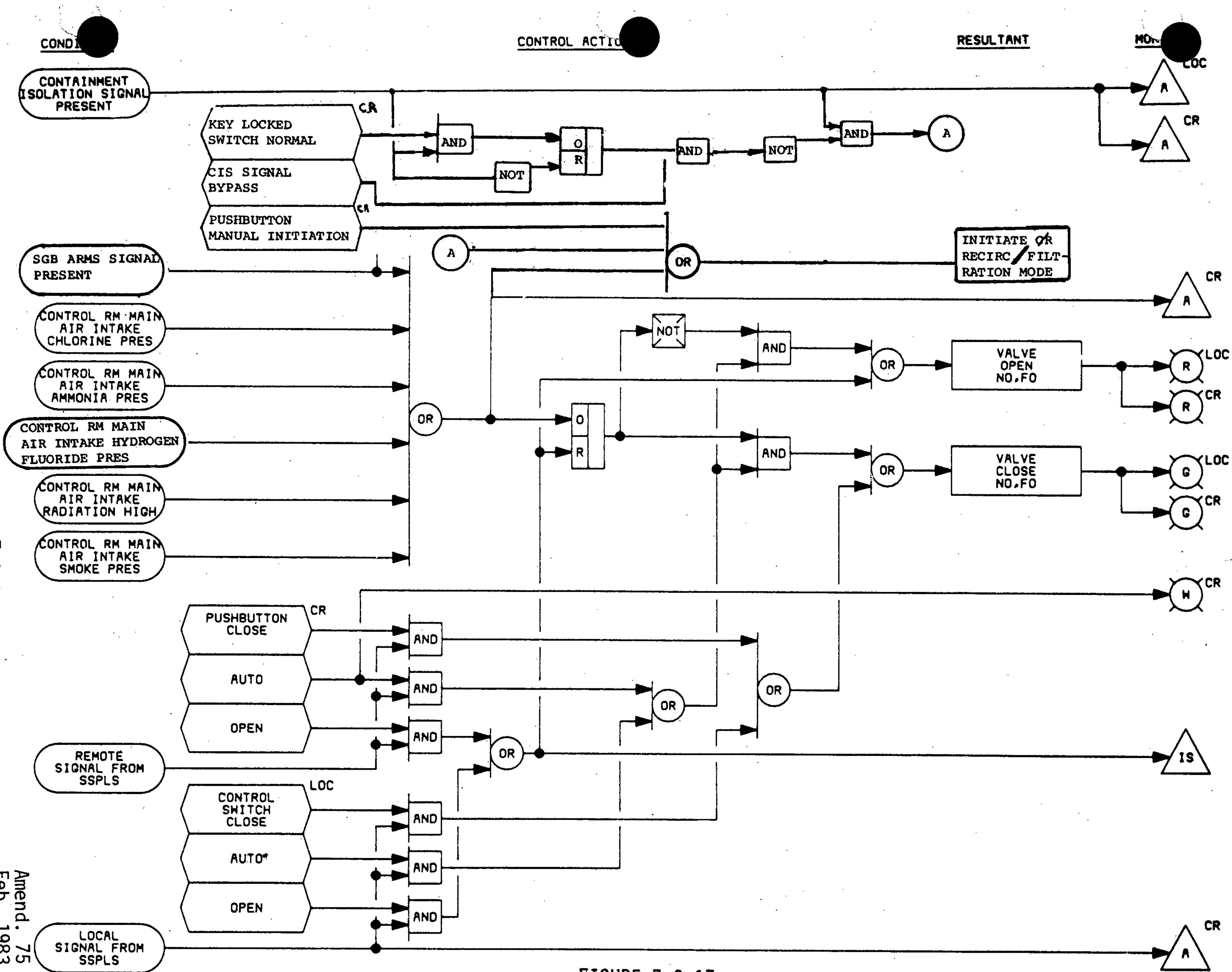

FIGURE 7.3-17
FUNCTIONAL CONTROL DIAGRAM CONTROL ROOM MAIN AIR INTAKE ISOLATION VALVES

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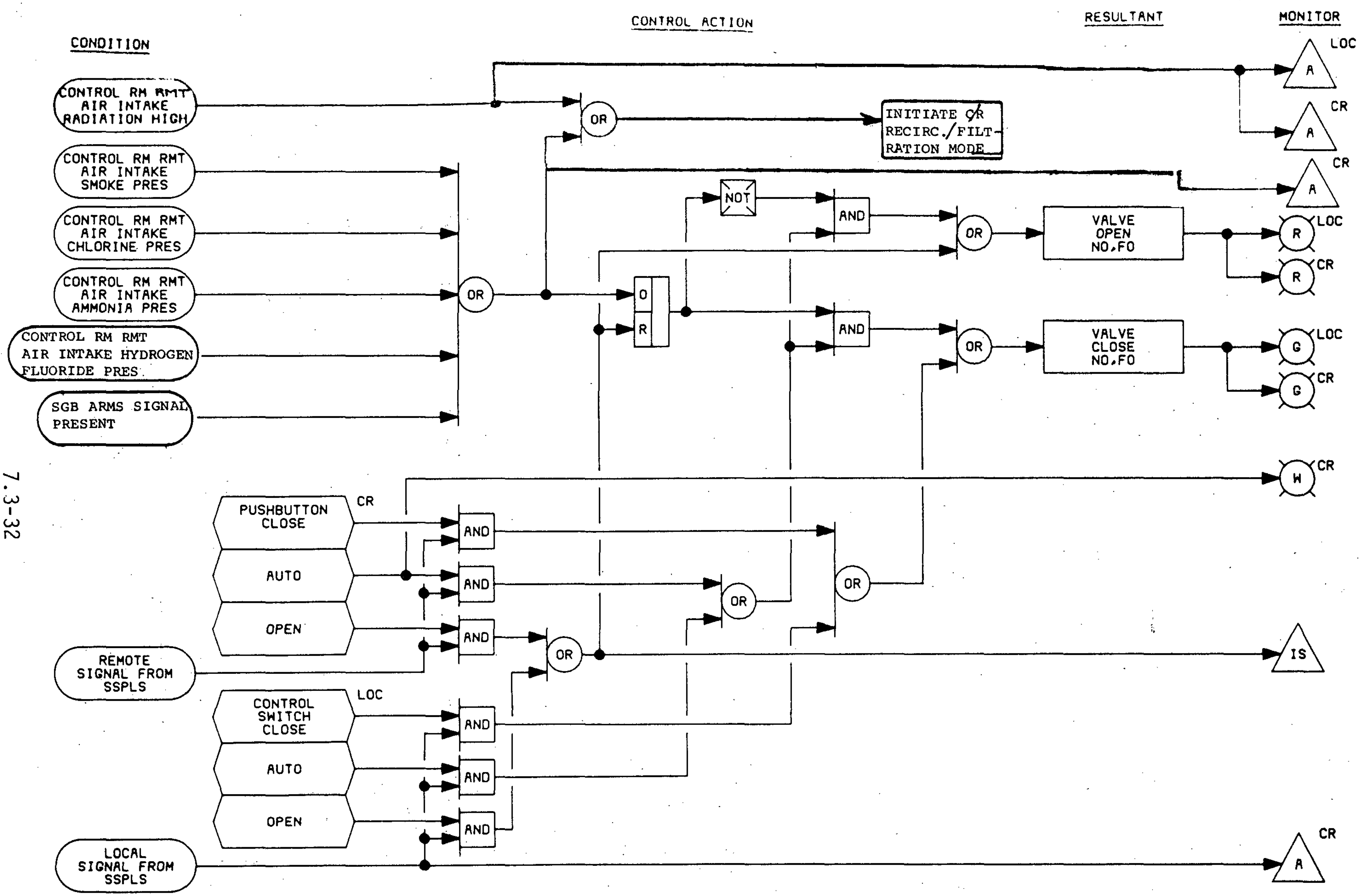

FIGURE 7.3-18
FUNCTIONAL CONTROL DIAGRAM CONTROL ROOM REMOTE AIR INTAKE ISOLATION VALVES

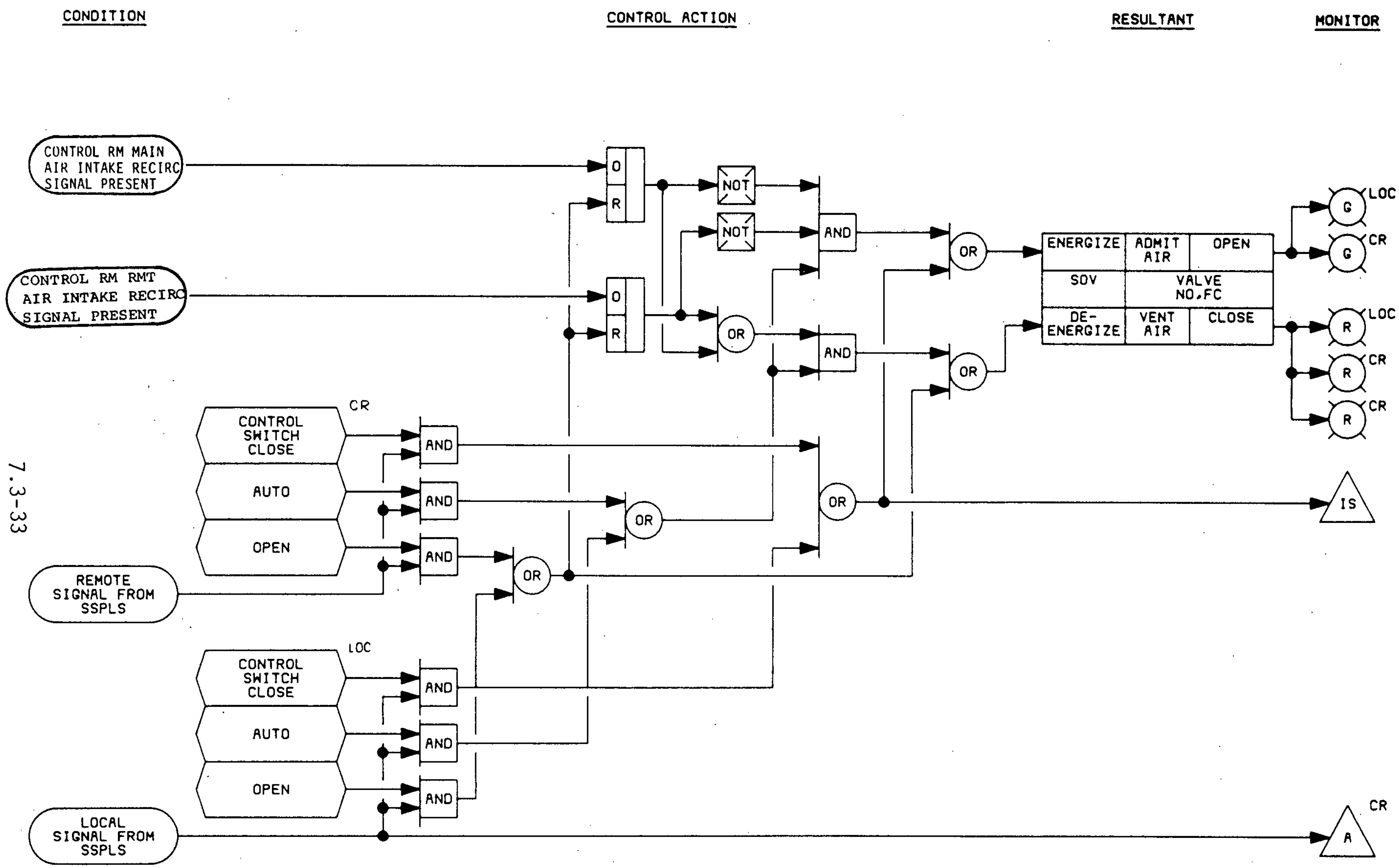

FIGURE 7.3-19
FUNCTIONAL CONTROL DIAGRAM CONTROL ROOM FILTER UNIT AIR INTAKE VALVES

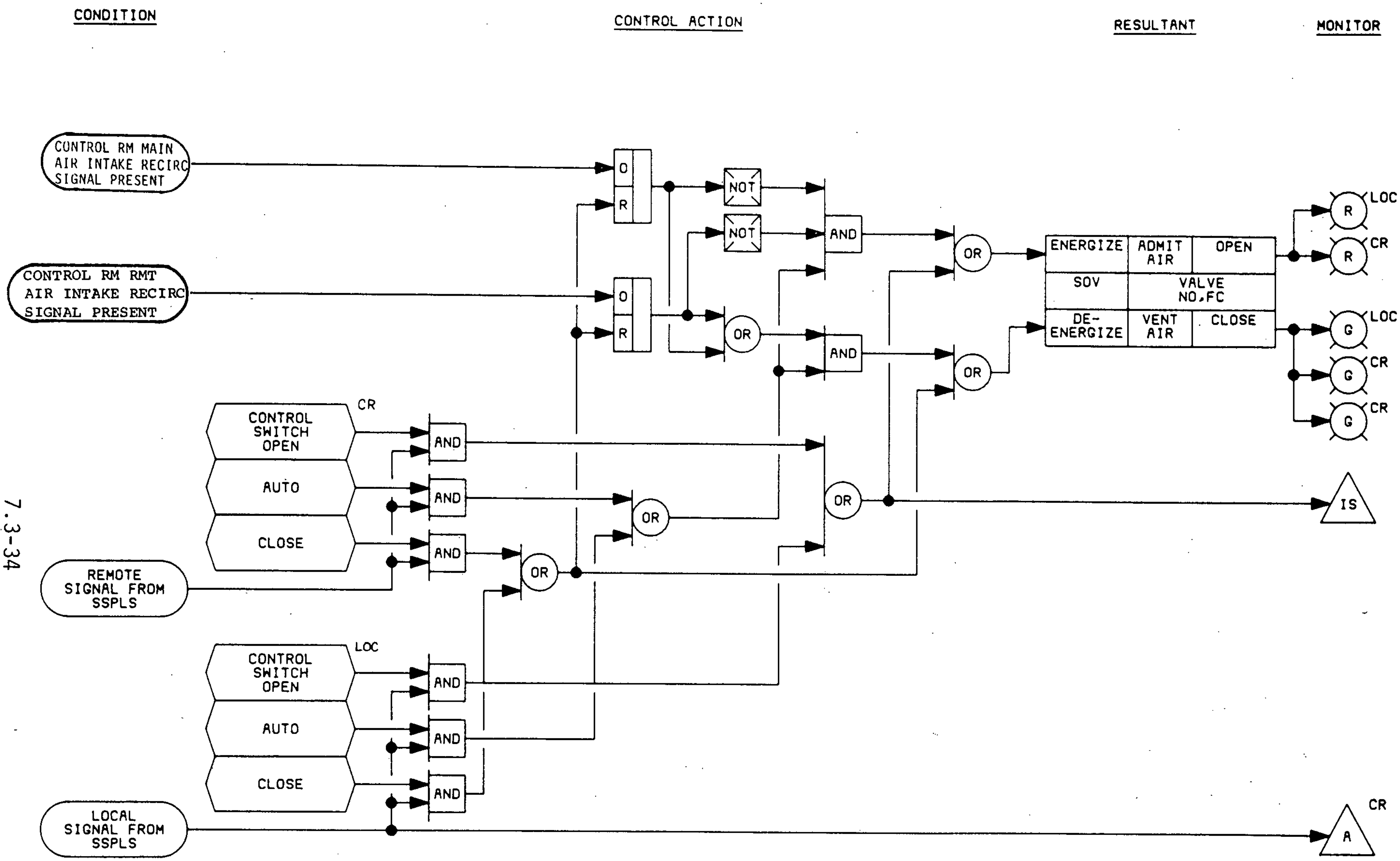

FIGURE 7.3-20
FUNCTIONAL CONTROL DIAGRAM CONTROL ROOM OUTSIDE AIR EXHAUST & HVAC UNIT OUTSIDE AIR INTAKE VALVES

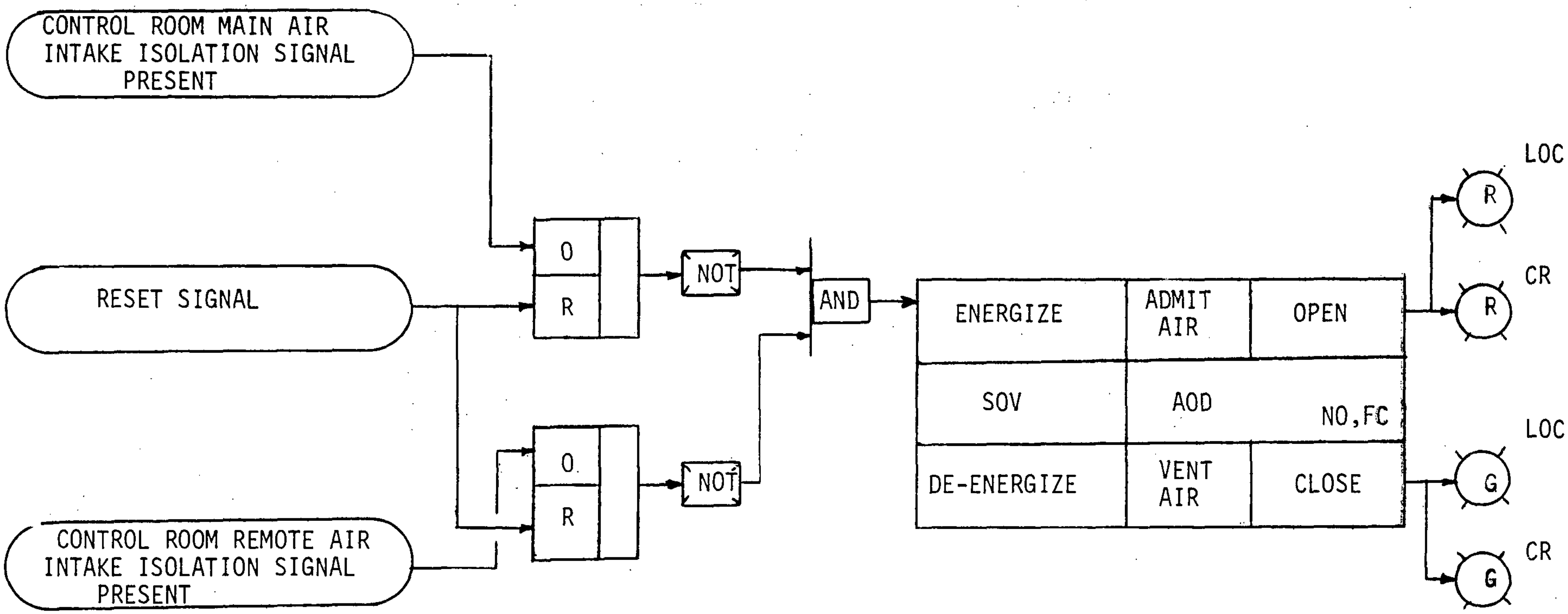

FIGURE 7.3-21

FUNCTIONAL CONTROL DIAGRAM CONTROL ROOM TOILET AND KITCHEN EXHAUST DAMPERS

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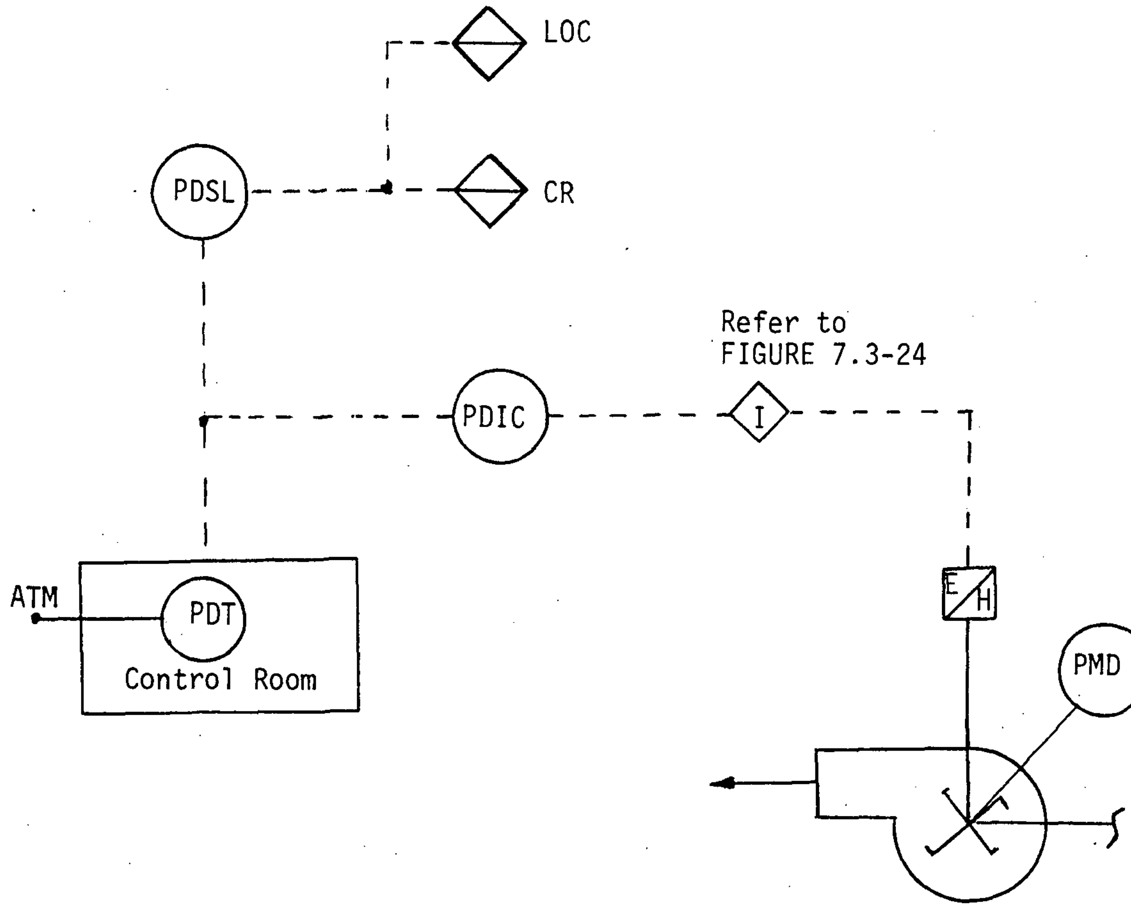

FIGURE 7.3-22 FUNCTIONAL CONTROL DIAGRAM
CONTROL ROOM HVAC UNIT FAN INLET VANE MODULATION

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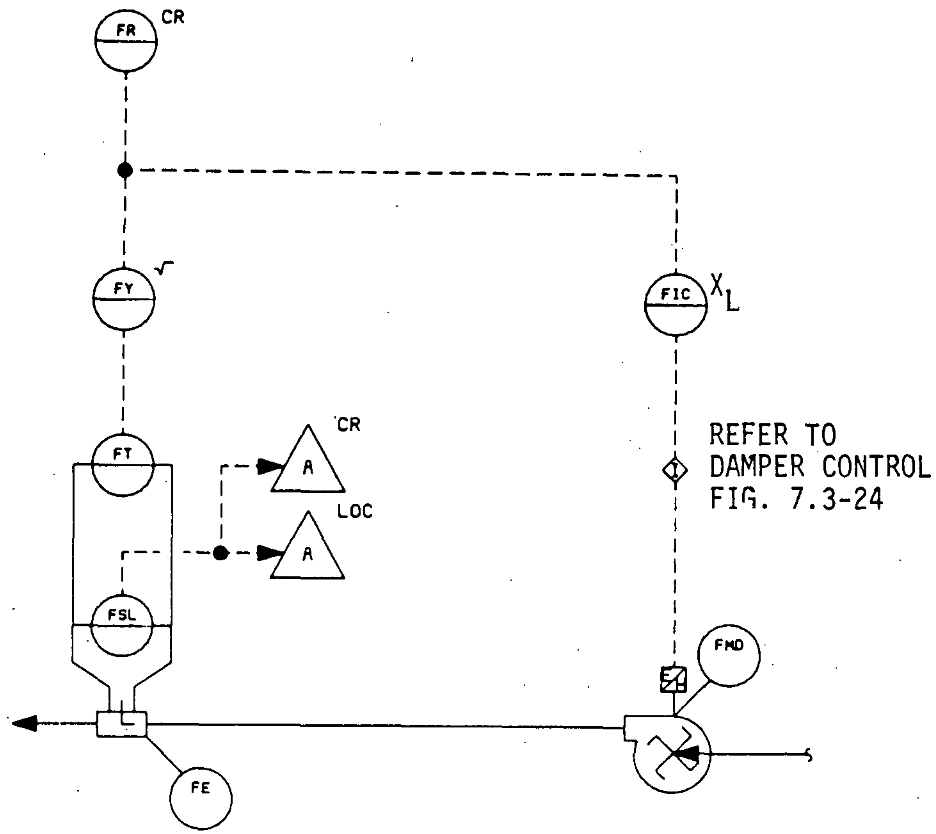

FIGURE 7.3-23
FUNCTIONAL CONTROL DIAGRAM CONTROL ROOM FILTER UNIT FAN
INLET VANE MODULATION

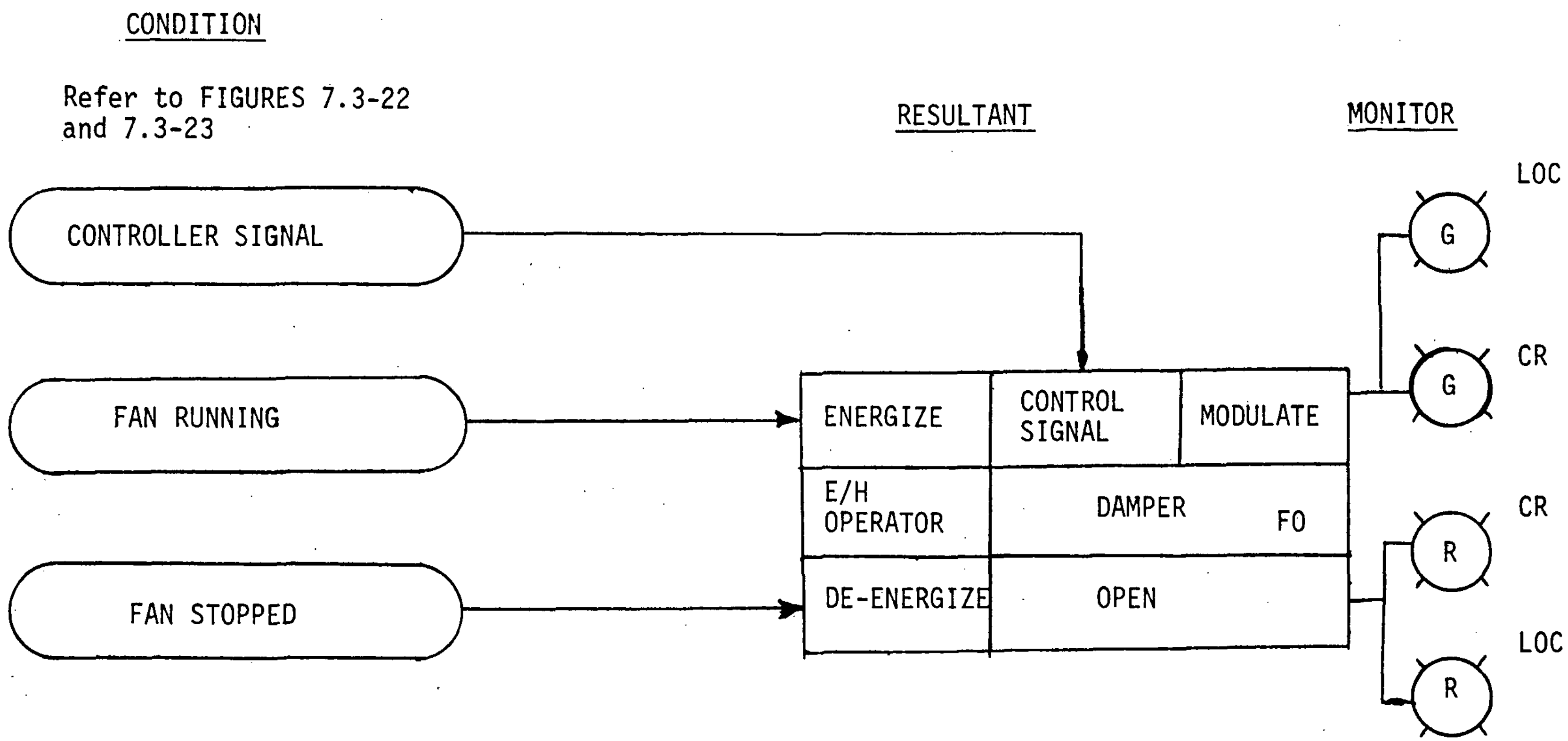

FIGURE 7.3-24 FUNCTIONAL CONTROL DIAGRAM
CONTROL ROOM HVAC UNIT AND FILTER FAN INLET VANE MODULATION

7.4 INSTRUMENTATION AND CONTROL SYSTEMS REQUIRED FOR SAFE SHUTDOWN

The Instrumentation and Control Systems necessary for safe shutdown are those associated with monitoring of core criticality, decay heat removal (SGAHRS portion), outlet steam isolation, and control room habitability.

Monitoring of core criticality is effected by the Flux Monitoring System (Section 7.5.1). The control room habitability is covered in Chapter 6. Thus, this section treats the control and instrumentation needs for decay heat removal by the Steam Generator Auxiliary Heat Removal System (SGAHRS) and outlet steam isolation by the Outlet Steam Isolation System (OSIS); control and instrumentation for Direct Heat Removal Service (DHRS) is discussed in Section 7.6.

7.4.1 Steam Generator Auxiliary Heat Removal Instrumentation and Control System

7.4.1.1 Design Description

7.4.1.1.1 Function

The SGAHRS (fluid system and mechanical components as described in Section 5.6.1, and electrical components as described below) provides the heat removal path and heat sink for the nuclear steam supply system following upset, emergency, or faulted events which render the normal heat sink unavailable.

The SGAHRS Instrumentation and Control System in conjunction with the PPS detects the need for, initiates, and controls the alternate heat removal path when the normal heat sink is unavailable. The SGAHRS nominal control setpoints shown in Table 7.4-2 are discussed in the following subsections.

The SGAHRS Instrumentation and Control System is designed to the IEEE Standards listed in Table 7.4-3.

7.4.1.1.2 Equipment Design

The mechanical system for which the SGAHRS I&C is provided is briefly described below.

When actuated, the SGAHRS draws water from a Protected Water Storage Tank and pumps it to each steam drum. Two supply lines are provided for each steam drum. One line is supplied by two half-sized, motor-driven feedwater pumps while the other is supplied by a full-sized, turbine-driven pump. Each supply line provides a flow control valve and an isolation valve at the inlet to each steam drum. The isolation valves are provided to isolate the auxiliary feedwater system from the steam generator system during power operation and to provide leak isolation during SGAHRS operation.

In addition, a Protected Air Cooled Condenser (PACC) supplied with each steam drum is placed into operation. This system rejects heat to the atmosphere via convection. Saturated steam is supplied to the condenser from the steam drum and saturated water is returned. This steam and water loop is driven by natural circulation. Each PACC unit consists of two tube bundles, two sets of louvers and two fans. Regulation of heat rejection is accomplished by controlling the air flow across the condensing tubes through adjustment of inlet louver and fan blade pitch positions. The air side flow is driven by either forced or natural convection.

The arrangement of SGAHRS equipment is shown in Figure 5.1-5 (SGAHRS P&ID). Instrumentation and controls are provided for the components described below:

- <u>Auxiliary Feedwater Pump Control</u> - Upon receipt of the SGAHRS initiation signal, (see Section 7.4.1.1.3), the two motor driven pumps are started, resulting in both pumps coming on line and operating at constant speed. In addition, the isolation valves in the steam supply lines from the steam drums to the turbine driven pump are opened. At the turbine inlet a pressure regulating valve reduces the steam supply pressure to the 1000 psig required by the turbine drive. The turbine drive mechanism is equipped with a governor to provide speed regulation. Each auxiliary feedwater pump can also be actuated manually at the operator's discretion.

  Each pump control includes a "Normal Long Term Cooldown (LTC)" mode selector. In "normal" mode, the pumps start on SGAHRS initiation. In the "LTC" mode, the operator may shutdown any or all AFW pumps provided the steam drum water level is above the trip point setting. When in the "LTC" mode, the pumps come on line automatically when the steam drum water level drops to a low level trip point.

- <u>Auxiliary Feedwater Flow Control</u> - The Auxiliary Feedwater Isolation Valves are opened upon receipt of the SGAHRS initiation signal. During SGAHRS operation, these valves close automatically upon indication of a sodium/water reaction, a high steam drum level, a steam drum pressure less than 200 psig, or AFW flow greater than 150% of full flow for 5 sec. This automatic closure occurs only in the affected loop. If the valves are closed by a high drum level signal they will reopen automatically when the drum level falls to the low drum level trip point. The flow to the steam drum is controlled with a control valve that is positioned by a single controller. Manual control of the Auxiliary Feedwater Flow Control valves is provided at the main control panel and at the local SGAHRS panel.

- Protected Air Cooled Condenser Control - The Protected Air Cooled Condenser louvers are opened and the fans started upon receipt of either a SCRAM or the SGAHRS initiation signal. The PACCs control the steam drum pressure to a variable setpoint with a nominal setting of 1400 psig by regulating heat rejection. Regulation of heat rejection is accomplished by controlling the air flow across the condensing tubes thru adjustment of inlet louver & fan blade pitch positions. During the airside forced convection mode of PACC operations, the air flow is varied by changing fan blade pitch with inlet louvers maintained at full open position. In the natural circulation mode of PACC operation, the air flow is varied by changing the position of the inlet louvers. The fan blades and inlet louvers are positioned by automatic controllers. Manual control of the inlet louver position and fan blade pitch is provided. Manual controls are also provided for the blower motors. The outlet louver is interlocked with the inlet louver. It opens automatically when the inlet louver actuator is energized. If a high concentration of sodium aerosol in each PACC cell is detected, redundant trip logic generates trip signals to
to shutdown the affected PACC system for approximately 1 1/2 hours.

- Pressure Controlled Bypass Valve - To prevent overheating of the Auxiliary Feedwater Pumps at reduced flow, each pump is provided with a bypass line from the discharge back to the Protected Water Storage Tank. The valve in the bypass line is normally open upon initiation during pump startup. After startup, the valve closes and then opens when pump discharge pressure rises to 1970 psig and closes when the pressure drops below 1820 psig.

- Auxiliary Feedwater Isolation Valves and Pump Inlet Isolation Valves - The isolation valves in each of the supply lines to the steam drums (AFW isolation valves) are provided to insure an uninterrupted supply of auxiliary feedwater to unaffected loops following failures in a loop which would otherwise limit the effectiveness of the auxiliary feedwater system. The isolation valves at the pump inlets are provided to prevent loss of water from the Protected Water Storage Tank (PWST) in the event of a failure between these valves and the AFW isolation valves and to allow switching suction from the PWST to the condensate storage tank.

- Superheater and Steam Drum Vent Control Valves - These valves are opened upon SGAHRS initiation and depressurize the steam drums to the valves respective setpoint levels. The superheater vent control valve setpoint is 1475 psig and the steam drum vent control valve setpoint is 1550 psig. The valves function to provide steam release during the venting period until the PACC units can remove the heat load in a closed loop manner.

o **PACC Noncondensible Vent Valve Control**

These valves are provided to vent noncondensibles from the PACC tube bundles during normal plant and PACC operation, and for PACC heatup following maintenance. Control of PACC noncondensible vent valves is by differential temperature control. The temperature differential between a concentration of non-condensible gases in a collection pipe and saturated steam temperature, measured in the PACC outlet heater, opens the vent valve. Venting is stopped when saturated steam enters the collection pipe and the temperature differential drops below the set point.

7.4.1.1.3 Initiating Circuits

The Reactor Shutdown System (see Section 7.2) provides redundant primary and secondary initiation signals to SGAHRS to sequentially start the three Auxiliary Feedwater Pumps and the three Protected Air Cooled Condensers when either a low steam drum level or high steam-to-feedwater flow ratio occurs in any one of the three Steam Generator System (SGS) loop subsystems. The PACCs are also initiated for all reactor scrams. However, initiation of the PACCs upon the occurrence of a reactor scram is not a safety-related function, but is provided in order to reduce the steam cooling of the superheater outlet tubesheet. In each subsystem, the three trip signals for low steam drum level and the three trip signals for high steam-to-feedwater flow ratio are each isolated and input to redundant two out of three logic networks. The outputs from the redundant logic networks are each isolated within the SGAHRS divisional control system and combined in a one-of-four logic to initiate SGAHRS. If two of three trip signals occur in any subsystem, the SGAHRS is initiated. The sequence of decay heat removal events is shown in Table 7.4-1. The scheme used for initiating the SGAHRS is shown in Figure 7.4-1.

Sensors are provided at the PACC airside inlet to detect sodium aerosols in the event of an IHTS loop sodium leak. When the aerosol concentration limit of 10^{-7} gm/cc is detected at any PACC inlet, the affected PACC will automatically shutdown for a 5000 sec. period and then will automatically restart. A manual override is provided for the operator to restart the PACC before the 5000 sec. period has ended if the operator has determined this can be done without adverse effect on PACC operation.

Since the automatic activation and control of auxiliary feedwater flow is necessary to assure decay heat removal, provisions are included in the design to assure that the automatic initiation takes precedence. A startup signal to the feedwater pumps overrides a manual control signal. Similarly, a signal to open the isolation valves overrides a manual closure signal.

7.4.1.1.4 Bypasses and Interlocks

Bypasses are required on the steam to feedwater flow mismatch and steam drum level subsystems to allow system reset and reactor startup without initiating SGAHRS. These bypasses will be implemented as described in the Reactor Shutdown System (Section 7.2).

The following are interlocks provided in the SGAHRS components control circuits:

(a) Each auxiliary feedwater pump (preferred) inlet valve may be closed only after the associated alternate inlet valve has been fully opened. The preferred inlet valve will open automatically anytime the alternate inlet valve start to close.

(b) A switch is provided on the back panel to permit the operator to bypass the sodium aerosol protection circuit of the PACC. The bypassed position is annunciated on the Main Control Room panel.

(c) The PACC outlet louver opens automatically whenever the inlet louver is not fully closed. When the outlet louver is fully open, the PACC blower may be started either automatically or manually.

7.4.1.1.5 Redundancy/Diversity

The SGAHRS (fluid system and mechanical components) is designed with suitable redundancy and diversity so that it can perform its safety functions following a single failure of an active component for anticipated, unlikely and extremely unlikely plant conditions. The design of SGAHRS relating to these objectives is discussed in Section 5.6.1.

Redundancy and diversity are also provided within the initiating circuitry of the SGAHRS control system. As shown in Figure 7.4-1, the system is actuated on two-out-of-three trip signals from either low steam drum level, or high steam-to-feedwater flow ratio.

The separation of SGAHRS equipment and cables from those of DHRS is described in Section 7.6.3.

7.4.1.1.6 Actuated Devices

All automatic valves and motors in the SGAHRS are provided with remote manual control capability, so that the entire system can be operated from the control room or the remote shutdown panels.

All isolation valves within the SGAHRS utilize an electrohydraulic actuator. All isolation valves are designed to fail to the position of greater safety upon loss of electrical power.

All required components of the SGAHRS instrumentation and control system operate on a vital electrical bus.

7.4.1.1.7 Testability

Instrumentation and controls for the SGAHRS are designed and arranged to allow for complete testability during reactor power operation. Bypassing of the actuated components (i.e., isolation valves and motors) is not required during testing as operation of these components during power operation poses no penalty on plant operation.

7.4.1.1.8 Separation

The SGAHRS instrumentation and control system, as part of the Decay Heat Removal System, is designed to maintain required isolation and separation between redundant channels (see Section 7.1.2.2).

7.4.1.1.9 Operator Information

Indicators and alarms are provided to keep the plant operator informed of the status of the SGAHRS. The following items are located on the Main Control Panel for operator information.

Analog Indication

- o Protected Water Storage Tank Level
- o Protected Water Storage Tank Temperature
- o Auxiliary Feedwater Flow (each loop)
- o Auxiliary Feedwater Pump Discharge Pressure
- o Drive Turbine Steam Inlet Pressure
- o Drive Turbine Speed
- o PACC Outlet Air Temperature
- o PACC Outlet Water Flow and Temperature
- o PACC Inlet Louver Position
- o PACC Fan Blade Pitch Position
- o Steam Drum Pressure and Water Level

Indicating Lights

- o PACC Outlet Louver Position
- o Position of all Isolation and Control Valves
- o Operating Status of all Motors
- o SGAHRS Initiation Logic Reset

Annunciators

- o Low Protected Water Storage Tank Level
- o Low Low Protected Water Storage Tank Level
- o High PWST Temperature
- o Simultaneous Opening or Closure of the AFW Pump Inlet Valve and the AFW Pump Alternate Inlet Valve
- o Flow Limiting of AFW
- o High AFW Supply Temperature
- o Acoustic Sensors Downstream of Superheater and Steam Drum Vent Control Valves
- o High/Low Drive Turbine Speed
- o High Drive Turbine Steam Inlet Pressure
- o Drive Turbine Group Alarm (Bearing and Lube Oil System)
- o AFW Pump Group Alarm (Bearing Temperature and Seal Cavity Pressure)
- o High Motor Bearing Temperatures
- o Transfer Switches on Local
- o SGAHRS Initiation Logic Trip
- o Na Aerosol Concentration High
- o Na Aerosol Control Bypassed
- o Acoustic Sensors Downstream of PACC Noncondensible Vent Valves
- o PACC Startup on Reactor Trip "on Test"
- o PACC Start-up Delay

Additional indicators and alarms are provided at the local instrumentation and control panels. Most information is also available to the operator via the Plant Data Handling and Display System (PDH&DS).

7.4.1.1.10 Instrumentation

Protected Water Storage Tank (PWST) Level

The PWST level is measured to monitor the water inventory available to be supplied to the steam drums in the event of loss of normal feedwater or the normal heat sink. The level is redundantly measured by two differential pressure sensors mounted across tap lines near the top and bottom of the tank. A PWST level measurement signal is provided to the Plant Control System (PSC), PDH&DS, Plant Annunciator System (PAS) and to a PAM recorder.

PWST Temperature

The PWST water temperature is measured to monitor the capacity of the water inventory to provide an efficient heat sink. The temperature is measured by a single chromel-alumel thermocouple. The temperature signal from the transmitter is provided to the PCS, PDH&DS and PAS.

Auxiliary Feedwater (AFW) Flow

The AFW flow is monitored to provide input to (a) restrict maximum flow through each control valve to less than 105 ± 5% rated AFW flow, and, (b) initiate automatic closure of AFW isolation valves in loops with AFW flow greater than 150% for 5 seconds. The flow in each of the AFW lines is redundantly measured by two differential pressure sensors across one venturi. This provides capacity for four flow measurements per loop. A flow measurement signal is provided to the PCS, PDH&DS, PAS and to a PAM recorder.

AFW Pump Discharge Pressure

The pressure of the water in the discharge line of the AFW pump is measured to provide the control of the valve in the recirculation line for AFW pump reduced flow operation. One pressure transmitter monitors the line pressure on the discharge side of each AFW pump. The pressure measurements are provided to the PCS and PDH&DS.

Drive Turbine Steam Inlet Pressure

The AFW Drive Tubine steam inlet pressure is measured to provide a control signal to modulate the pressure control valve. A single pressure transmitter is located between the turbine inlet and the control valve. The signal is provided to the PCS, PDH&DS and PAS.

Drive Turbine Speed

The AFW Drive Turbine speed is measured to provide a signal to the turbine speed governor and for initiating an overspeed trip. A single magnetic pickup provides signals to the PCS, PDH&DS and PAS.

Protected Air Cooled Condensor (PACC)

- o PACC Outlet Water Flow - sensed by one differential pressure sensor per loop across a venturi. Signals are provided to the PCS and the PDH&DS.
- o PACC Outlet Water Temperature - sensed by one chromel-alumel thermocouple per loop. Signals are provided to the PCS and the PDH&DS.
- o PACC Outlet Air Temperature - sensed by three chromel-alumel thermocouples per loop. Signals are provided to the PCS from only the "A" outlet.
- o PACC Inlet Louver Position - sensed by two louver position sensors per loop. Signals are provided to the PCS and the PDH&DS.
- o PACC Outlet Louver Position - sensed by two switches per louver. Signals are provided to the PCS and the PDH&DS.
- o PACC Fan Blade Pitch Position - sensed by one pitch position sensor per fan (i.e. - two per loop). Signals are provided to the PCS and the PDH&DS.

Isolation and Control Valve Positions

The position of each valve is sensed by two limit switches; one indicates the valve is open, one indicates the valve is closed. The "Open/Closed" position signal is provided to the PCS and the PDH&DS. The monitored valves are:

- PWST Fill Valve
- Alternate AFW Supply Valve
- AFW Pump Inlet Valve
- AFW Pump Alternate Inlet Valve
- AFW Pump Recirculation Valve
- AFW Control Valve
- AFW Isolation Valve
- AFW Pump Test Loop Isolation Valve
- Drive Turbine Steam Supply Isolation Valve
- Drive Turbine Pressure Control Valve
- Superheater Vent Control Valve
- Steam Drum Vent Control Valve
- Turbine Drive Governor Valve
- PACC Noncondensible Vent Valve

7.4.1.2 Design Analysis

To provide a high degree of assurance that the SGAHRS will operate when necessary, and in time to provide adequate decay heat removal, the power for the system is taken from energy sources of high reliability which are readily available. As a safety related system, the instrumentation and controls critical to SGAHRS operation are subject to the safety criteria identified in Section 7.1.2.

Redundant monitoring and control equipment will be provided to ensure that a single failure will not impair the capability of the SGAHRS Instrumentation and Control System to perform its intended safety function. The system will be designed for fail safe operation and control equipment where practical and will, in the event of a failure, assume a failed position consistent with its intended safety function.

Because there are three redundant decay heat removal loops, the instrumentation and controls associated with each individual loop (e.g., auxiliary feedwater flow and air cooled condenser control systems) do not independently meet single failure criteria. However, when taken collectively as a system, they provide the single failure capability required.

7.4.2 Outlet Steam Isolation Instrumentation and Control System

7.4.2.1 Design Description

7.4.2.1.1 Function

The Outlet Steam Isolation Subsystem (OSIS) provides isolation of steam system pipe breaks. Steam system isolation is a necessary function for safe shutdown in those pipe break conditions affecting the three steam supply systems and is provided if needed on a per loop basis. By definition, this zone of protection will include the high pressure steam supply system downstream from the individual loop check valves.

The OSIS Controls are designed to the IEEE Standards listed in Table 7.4-3.

7.4.2.1.2 Equipment Design

A high steam flow-to-feedwater flow ratio is indicative of a main steam supply leak downstream from the flow meter or insufficient feedwater flow. The superheater steam outlet valves and superheater bypass valves shall be closed with the appropriate signal supplied by the heat transport instrumentation system (Section 7.5). This action will assure the isolation of any steam system leak common to all three loops and also provide protection against a major steam condenser leak during a steam bypass heat removal operation.

7.4.2.1.3 Initiating Circuits

The OSIS is initiated by the SGAHRS initiation signal. The SGAHRS initiation signal is described in 7.4.1.1.3. This initiation signal closes the superheater outlet isolation valves in all 3 loops when a high steam-to-feedwater flow ratio or a low steam drum level occurs in any loop. In each Steam Generator System loop, the three trip signals for high steam-to-feedwater flow ratio and the low steam drum level are input to a two of three logic network. If two of three trip signals occur in any of the 3 loops, the OSIS is initiated, and all 3 loops are isolated from the main superheated steam system by closure of the superheater outlet isolation valves and superheater bypass valves.

7.4.2.1.4 Bypasses and Interlocks

Control interlocks and operator overrides associated with the operation of the superheater outlet isolation valves have not been completely defined.

Bypass of OSIS may be required to allow use of the main steam bypass and condenser for reactor heat removal. In case the OSIS is initiated by a leak in the feedwater supply system, the operator may decide to override the closure of certain superheater outlet isolation valves.

7.4.2.1.5 Redundancy and Diversity

Redundancy is provided within the initiating circuits of OSIS. The primary trip function takes place when a high steam-to-feedwater flow ratio is sensed by two of three redundant subsystems on any one SGS loop. The low steam drum level sensed by two of three redundant channels in any one loop provides a backup trip function. Additional redundance is provided by three independnt SGS steam supply loops serving one common turbine header. Any major break in the high pressure steam system external from the individual loop check valves will be sensed as a steam feedwater flow ratio trip signal in all three loops.

7.4.2.1.6 Actuated Device

The superheater outlet isolation and superheater bypass valves utilize a high reliability electro-hydraulic actuator. These valves are designed to fail closed upon loss of electrical supply to the control solenoid.

7.4.2.1.7 Separation

The OSIS Instrumentation and Control System, as part of the Decay Heat Removal System is designed to maintain required isolation and separation between redundant channels (see Section 7.1.2).

7.4.2.1.8 Operator Information

Indication of the superheater outlet isolation valve position is supplied to the control room. Indicator lamps are used for open-close position indication to the plant operator.

7.4.2.2 Design Analysis

To provide a high degree of assurance that the OSIS will operate when necessary, and in time to provide adequate isolation, the power for the system is taken from energy sources of high reliability which are readily available. As a safety related system, the instrumentation and controls critical to OSIS operation are subject to the safety criteria identified in Section 7.1.2.

Redundant monitoring and control equipment will be provided to ensure that a single failure will not impair the capability of the OSIS instrumentation and Control System to perform its intended safety function. The system will be designed for fail safe operation and control equipment, where practical, will assure a failed position consistent with its intended safety function.

7.4.3 Pony Motors and Controls

There are six pony motors, one in each primary and intermediate heat transport loop to provide sodium flow for decay heat removal. These motors through the use of a gear box are capable of providing five to ten percent sodium flow in five discrete steps by gear changes. Section 5.6 describes the interaction of the primary and intermediate heat transport loops with the SGAHRS to provide decay heat removal.

7.4.3.1 Design Description

The pony motors are 75 horsepower, 480 VAC, 3 phase, 60 Hz, totally enclosed fan cooled Class 1E motors. These motors are mounted on top of the sodium pump vertical drive motor. They are 1800 rpm motors which deliver power to the sodium pump via a reducing gear, an overrunning clutch, and the vertical motor shaft.

The overrunning clutch allows the pony motor to run continuously during all modes of plant operation and automatically drives the pump when the vertical motor speed decreases below the output speed of the reducing gear. Thus, after a reactor trip and pump (vertical drive motor) trip sodium flow does not decrease below pony motor flow.

During normal operation at pony motor speeds the external oil cooling system is in operation. However, the vertical dirve motor bearings are designed to start and operate continuously at pony motor speed without the external oil cooling system or high pressure lift pump.

The pony motor is controlled using both Non-class 1E and Class 1E circuit. The Non-class 1E circuit is isolated from the Class 1E circuit and is overridden by the Class 1E circuit.

Normal pony motor start is through a Non-1E permissive sequence circuit which first starts the vertical drive motor external circuit which first starts the vertical drive motor external lubricating oil cooling system and high pressure lube oil pump. When the oil system achieves flow and pressure the pony motor starts. Once started the Class 1E circuit takes over and the loss of the external lubricating oil system will not result in a pony motor trip. This method of starting is not classified as safety-related and is used for starting the pony motor during reactor shutdown periods after maintenance which requires the pony motor to be off.

The Class 1E controls start the pony motors without the use of the external lubricating oil cooling system or high pressure lube oil pump. This function is carried out by a start-stop switch on the main control panel in the control room. Once started by either the Class 1E or Non-class 1E control the pony motor will automatically restart following the loss of off-site power on the Class 1E diesels.

7.4.3.2 Initiating Circuits

The pony motor runs continuously during all modes of plant operations except during reactor shutdown for maintenance. During maintenance only one loop is permitted to be out of service. Therefore, there is no need for automatic or manual initiation circuits. However, the Class 1E start-stop switch is located on the main control panel.

7.4.3.3 Bypasses and Interlocks

There are no bypasses in the Class 1E control circuit.

The only condition which results in an interlock/automatic pony motor trip is a sodium-to-water leak in the steam generator modules. This results in an automatic trip of the affected intermediate heat transport loop pony motor only. The sodium-to-water leak trip is describe in 7.5.6.

7.4.3.4 Analysis

The pony motor and the Class 1E control circuit is designed to the IEEE Standards listed in Table 7.4-3 and is qualified in accordance with Section 1.6 Reference 9.

7.4.4 Remote Shutdown System

7.4.4.1 Design Description

7.4.4.1.1 Function

The Remote Shutdown System provides the means by which (1) safe shutdown conditions of the reactor plant can be established and maintained from locations outside of the Control Room in the event that the Control Room must be vacated; (2) hot shutdown conditions can be achieved and maintained; and, (3) if desired, the plant can be cooled to and maintained at the refueling temperature.

7.4.4.1.2 Design Basis

The Remote Shutdown system is designed to use equipment located outside of the Control Room to place the reactor and plant into a safe shutdown condition under the following conditions:

(a) The evacuation of the Control Room is not coincident with any other abnormal plant condition with the one exception that loss of offsite power may occur.

(b) No severe natural phenomena such as tornadoes, hurricanes, floods, tsunami and seiches (from 10CFR50, Appendix A, Criterion 2) with the exception of earthquake occur coincidently with the evacuation of the Control Room.

(c) The plant remains in an orderly shutdown status from the initiation of the evacuation of the Control Room to the time that command of the shutdown is re-established outside of the Control Room.

(d) The remote shutdown operations will be commanded from one location and will use plant systems operated in their local mode to effect the shutdown and decay heat removal.

(e) Plant instrumentation and control systems required for remote shutdown operations will have transfer switches located at the local panels to permit the plant operating personnel to select to operate from the local panels while isolating the remote controls or, conversely, to operate from the control room while isolating the local controls. The transfer of control of a plant system from the remote to the local mode is annunciated in the control room.

(f) Communications between the Remote Shutdown Monitoring Panel (RSMP), the command location for remote shutdown operations, and the SGAHRS panels and other local panels during remote shutdown operations will be by the Maintenance Communication Jacking (MCJ) system utilizing a sound-powered telephone. Plant telephone, paging and radio communication systems will also be available.

7.4.4.1.3 Remote Shutdown Operations

The RSMP will be located in Cell 272B of the 836'-0" level of the SGB. The RSMP will have indications (see Section 7.4.4.1.4) from which an operator can assess the progress of the shutdown, and it will be the location from which that operator will command the operation of the plant systems being operated in their local mode to effect shutdown.

The Division I, II and III SGAHRS (Section 7.4.1) local panels will be located in Cells 272A, B and C respectively, in close proximity to the RSMP, on the 836'-0" level of the SGB-1B. The SGAHRS, operated in its local mode, will be used to control the removal of heat from the reactor plant to achieve and stabilize the plant at the desired plant temperature (hot shutdown or refueling temperature). The local SGAHRS panels will have all of the controls and indications necessary to completely control the SGAHRS System. All signals from the Control Room to the SGAHRS panels are buffered to prevent faults occuring in the Control Room from propogating back to the SGAHRS panels. All SGAHRS component controls can be transferred to local at the local SGAHRS panels. Placing the transfer switches in "local" overrides all control functions in the Control Room.

The Division I, II and III OSIS local panels are located in SGB Cells 272A, B and C with the SGAHRS panels, and will be operated in the local mode when required to control heat removal from the plant in conjunction with the operation of SGAHRS. Isolation of OSIS panel controls from the Control Room is incorporated in the design. Steam drum drain and superheater outlet isolation valve controls can be transferred to local at the local OSIS panels.

Whenever any SGAHRS component control transfer switch is placed in the "local" position an alarm is initiated in the Control Room to alert the Control Room operator. The same statement is true for the steam drum drain controls and superheat outlet isolation valve controls on the OSIS panels.

If offsite power is lost coincident with having to achieve a safe shutdown condition in the reactor plant from outside of the Control Room, the diesel generators will automatically start and function in accordance with the design provided by the Building Electrical Power System (Section 8.3.1.1.1). Manual isolation by use of transfer switches will be done at the DG local panels. Manual control of the DG, if necessary, can also be accomplished at the DG local panels.

In the event that the Control Room must be vacated, reactor scram and SGAHRS operation will normally be initiated manually before Control Room evacuation. The operating personnel will move to the 836'-0" level of the SGB where the SGAHRS in the local mode will effect heat removal and stabilization of the plant temperatures. The plant shutdown will be directed by the operator at the RSMP who will also assign operating personnel not continuously occupied in operating SGAHRS to oversee or operate other systems as required. Primary sodium outlet temperature indication can be used for confirmation of reactor shutdown cooling.

Movement of personnel within the plant and access to building cells and local panels will be controlled by the facilities and procedures of the Industrial Security System.

7.4.4.1.4 Equipment Design

The RSMP is the only piece of equipment provided by the Remote Shutdown System. It will be a vertical sided, non-Class 1E cabinet assembly containing meters and a phone jack panel. The meters will receive buffered signals from the initiating systems and, thus, do not require transfer switches to isolate them from the Control Room. The phone jack panel will permit the operator at the RSMP to communicate with the five NSSS or Nuclear Island buildings by means of any of the three MCJ circuits provided in each of the buildings. In addition, communications among the buildings can be established through the phone jack panel on the RSMP.

The indications provided on the RSMP are as follows:

- For each primary heat transport system loop,

  1 - Pump outlet sodium temperature indicaton (3 total)
  1 - Reactor inlet sodium temperature indication (3 total)
  1 - Sodium pump shaft speed indication (3 total)

- For each intermediate heat transport system loop,

  1 - IHX outlet sodium temperature indication (3 total)
  1 - IHX inlet sodium temperature indication (3 total)
  1 - Sodium pump shaft speed indication (3 total)

- One reactor vessel sodium level meter (long probe)

- For each Diesel Generator (3 total)

  1 - Wattmeter
  1 - Frequency meter
  1 - Varmeter
  1 - Voltmeter with phase selector switch
  1 - Ammeter with phase selector switch

In addition to the foregoing indications, other indications used during remote shutdown operations that are not on the RSMP will be available as follows:

- <u>SGAHRS</u>

  Controls and indicators used for the operation of each SGAHRS division are located on the three seperate SGAHRS panels in cells 272A, B, and C. Each SGAHRS division is separate and redundant from the other divisions. See the response to Question CS421.04 for additional information about SGAHRS division assignments.

The following controls, indicators and alarms are on each SGAHRS panel.*

Controllers

Auxiliary Feedwater Flow
AFW Steam Turbine Steam Inlet Pressure
PACC Inlet Louver Position
PACC Fan Blade Position
Steam Drum Level
Steam Drum Vent
Superheater Vent

Analog Indicators

Protected Water Storage Tank Level
Protected Water Storage Tank Temperature
Auxiliary Feedwater Flow
Auxiliary Feedwater Pump Discharge Pressure
Steam Driven Turbine Steam Inlet Pressure
Steam Driven Tubrine Speed
PACC Outlet Air Temperature
PACC Outlet Water Flow and Temperature
PACC Inlet Louver Position
PACC Fan Blade Pitch Position
Steam Drum Pressure and Water Level

Annunciators

Protected Water Storage Tank Level
PWST Temperature
AFW Supply Temperature
Steam Driven Turbine Speed
Driven Turbine Steam Inlet Pressure
Steam Driven Turbine Bearing and Lube Oil Temperature
High Motor Bearing Temperatures
SGAHRS Initiation

o Diesel speed and fuel oil indications will be available at the diesel generator local control panels in the Diesel Generator Building.

*Each indicator, alarm and controller is repeated on each of the SGAHRS panels except for those associated with the AFW pumps. Panels A and B have the controls, alarms and indicators for motor driven AFW pumps A and B; Panel B has those associated with the steam driven AFW pump.

7.4.4.2 Design Analysis

The Remote Shutdown System provides the RSMP from which an operator can assess the progress of the plant shutdown and command the local operation of the plant systems (primarily SGAHRS) to effect the shutdown. It should be noted that the PACC subsystem of SGAHRS is automatically initiated by all reactor trips, and it remains in operation for the duration of the plant shutdown or as long as the reactor generates significant decay heat.

The Remote Shutdown System imposes no special requirements on the plant systems, but takes advantage of the following system design features:

- The ability to operate in both local and remote modes with isolation from and annunciation in the Control Room when operating in the local mode.

- The redundancy diversity, separation, isolation and reliability of the safety grade systems.

- The design and location of safety grade systems equipment that minimize the probability and effect of fires and explosions on the ability of the systems to perform their safety function.

- The redundant safety grade SGAHRS provides the capability to achieve and maintain hot shutdown and, if desired, to cool the plant to and maintain the plant at refueling conditions.

TABLE 7.4-1

SEQUENCE OF DECAY HEAT REMOVAL EVENTS FOR POSTULATED LOSS OF FEEDWATER

Event	Time From Previous Event (1)	Auto/Manual	Controlled Parameter	Controlling Signal
Scram Reactor & Trip Turbine. 59 Initiate Opening of AFW Isolation 58 Valves, SGAHRS Vent Control Valves, AFW Turbine Drive Steam Supply Isolation Valves, and Drive Turbine Pressure Control Valve. Start AFW Pump Motors. Initiate Start of Air Cooled Condensers.	0 Sec.	A	Initiate SGAHRS & OSIS	Plant Shut Down System High Steam to Feed Water Flow Ratio or Low Level
Steam Generator System Relief Valves Open.(2)	1 Sec.	A	SG System Pressure	High Steam System Pressure
AFW Drive Turbine Steam Supply Isolation Valve Full Open	2 Sec.	A	-------------------	----------------------
Electric Motor Driven AFW Pumps Attain Rated Head	7 Sec.	A	-------------------	----------------------
AFW Control Valves Regulate FW Flow	----	A	AFW Flow	Steam Drum Level
Vent Control Valves Regulate Steam Drum Pressure	----	A	Steam Venting	Steam Drum Pressure
Turbine-Driven AFW Pump Attains Rated Head	17 Sec.	A	Pump Speed	Pump Speed
Stop 2 of 3 AFW Pumps.(3)	Optional	M	-------------------	----------------------
Vent Control Valves Close and PACCs assume complete 54 heat load	~1 Hr.	A	Steam Venting	SG System Pressure

TABLE 7.4-1 (Continued)

Event	Time From Previous Event(1)	Auto/Manual	Controlled Parameter	Controlling Signal
PACC Fans and Louvers Stabilize PACC heat rejection rate	---------	A	Air Cooled Condenser Heat Rejection	Steam Drum Pressure
Stop AFW Pump	Optional	M	-------------------	-------------------
Decrease Steam Drum Pressure Setpoint for PACC	Optional	M	-------------------	-------------------
Plant Reaches Isothermal Conditions Corresponding to PACC Setpoint	--------	A	-------------------	-------------------

(1) For loss of off-site power event startup of AFW pumps initiated 20 sec. after loss of power.

(2) The SG System Relief Valves are open only a short time (<15 sec.) due to their high setpoint. As the system pressure falls, these valves will close first and the system pressure will be controlled by the SGAHRS Vent Control Valves only.

54 (3) Redundant pumps stopped when one motor-driven pump can assume the entire AFW flow requirement.

TABLE 7.4-2

SGAHRS NOMINAL SET POINTS

	Set Point
SGAHRS Primary Initiation Signal:	
High Main Steam to Main Feedwater Flow Ratio	1.3
SGAHRS Secondary Initiation Signal:	
Low Steam Drum Level (Inches from Normal Water Level)	-8
Turbine-Driven AFW Pump operating Speed (RPM)	4000
AFW Pump Recirculation Valve (52AFV-108) (Note 1)	
Pump Pressure - Open Valve (psig)	1970
Pump Pressure - Close Valve (psig)	1820
AFW Control Valve (52AFV-104 (Note 1):	
Steam Drum Level Control:	
Motor-Driven Pumps (In. from normal water level (note 6)	-4
Turbine-Driven Pumps (In. from normal water level (note 6)	-18
Flow Limiter (1lbm/hr)	264,500
Drive Turbine Pressure Control Valve (52AFV-121)(psig)(Note 1) Controls steam pressure to Drive Turbine at:	1000

TABLE 7.4-2
SGAHRS NOMINAL SET POINTS (cont'd)

	Set Point
AFW Isolation Valve (52AFV-103 (Note 1):	
Valve Closes on:	
High AFW Flow for 5 sec. (lbm/hr)	378,000
Low Steam Drum Pressure (psig)	200
High Drum Level (in. from normal water level (Note 2 and 6)	+8, +12
Sodium/Water Reaction	Indication
AFW Pump Test Loop Isolation Valves:	
Valves Close on the SGAHRS initiation signal	SGAHRS Initiation
Steam Drum Vent Valves (52AFV-117) (psig) (Note 1) control Steam Drum Pressure at:	1550
Superheater Vent Valves (52AFV-116) (psig) (Note 1) control Steam Drum Pressure at:	1475
Steam Drum Level at which AFW pumps automatically restart in long term cooldown mode (inches from normal water level):	
Motor-Driven Pump (52AFP002A)	-8
Motor-Driven Pump (52AFP002B) (Note 3 and 6)	-8
Turbine-Driven Pump (52AFP001) (Note 4 and 6)	-18
Protected Air Cooled Condenser (PACC) (psig) controls steam drum pressure at: During refueling and other long term cooldown operations: PACC nominal setpoint is 250 (psig)	1400
PACC Vent Valves (52ACV-129 (Note 1) Control by:	Note 5

TABLE 7.4-2
NOMINAL SET POINTS (Cont'd)

NOTES:

1. The capability for the operator to assume manual control of the indicated functions from either the control room or the local panel is provided.

2. Valves will reopen should steam drum level fall to the low level trip (-8 in. from normal water level). Valves in the motor-driven AFW pump loops close at +8 in. from normal water level while the valves in the turbine-driven AFW pump loops close at +12 in. from normal water level.

3. In the long term cooldown mode, the second motor driven pump automatically restarts after a 1-minute delay if steam drum level remains at -7 in. or lower.

4. Steam drum pressure must be above 1000 psig to initiate turbine operation.

5. PACC vent control valves are controlled by the temperature differential between the noncondensible gas collection pipe and the steam saturation temperature measured in the PACC outlet header.

6. Normal steam drum water level is 1 inch above drum centerline.

TABLE 7.4-3

LIST OF IEEE STANDARDS APPLICABLE TO
SGAHRS AND OSIS INSTRUMENTATION AND CONTROL SYSTEMS

IEEE-279-1971	IEEE Standard: Criteria for Protection Systems for Nuclear Power Generating Stations
IEEE-308-1974	Criteria for Class 1E Power Systems for Nuclear Power Generating Stations
IEEE-323-1974	IEEE Trial-Use Standard: General Guide for Qualifying Class 1E Electric Equipment for Nuclear Power Generating Stations
IEEE-323-A-1975	Supplement to the Foreword of IEEE 323-1974
IEEE-336-1971	IEEE Standard: Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During Construction of Nuclear Power Generating Stations
IEEE-338-1977	Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems
IEEE-344-1975	IEEE Standard 344-1975, IEEE Recommended Practices for Seismic Qualification of Class 1 Equipment for Nuclear Power Generating Stations
IEEE-352-1975	General Principles for Reliability Analysis of Nuclear Power Generating Station Protection Systems
IEEE-379-1972	IEEE Trial-Use Guide for the Application of the Single-Failure Criterion to Nuclear Power Generating Station Protection Systems
IEE-382-1980	IEEE Standard for Qualification of Safety-Related Valve Actuators
IEEE-384-1974	IEEE Trial Use Standard Criteria for Separation of Class 1E Equipment and Circuits
IEEE-494-1974	IEEE Standard Method for Identification of Documents Related to Class 1E Equipment and Systems for Nuclear Power Generating Station

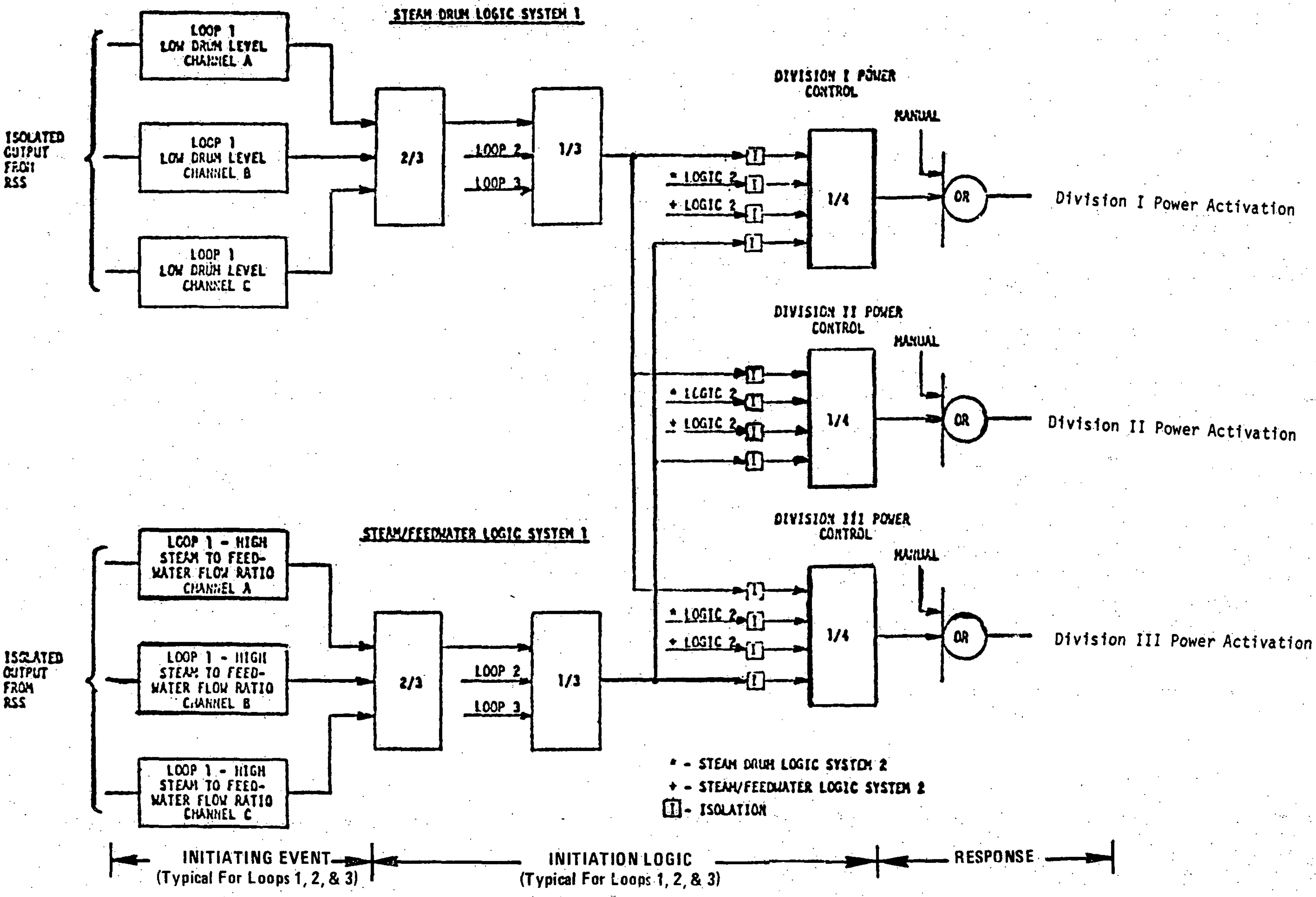

Figure 7.4-1 Steam Generator Auxiliary Heat Removal System Initiation Logic

7.5 INSTRUMENTATION AND MONITORING SYSTEM

The instrumentation and monitoring systems included in this Section are the Flux Monitoring System, the Heat Transport Instrumentation System, the Reactor and Vessel Instrumentation System, the Fuel Failure Monitoring System, the Leak Detection Instrumentation System, the Sodium-Water Reactor Pressure Relief System and the Loose Parts Monitoring System. Table 7.5-1 lists the measured parameters and instrumentation provided by these systems. The instrumentation which is safety related as defined in Section 3.2.1 is identified with an asterisk in column 2 of Table 7.5-1. Instrumentation and monitoring for TLTM parameters not included in the design basis are also discussed. These include containment hydrogen monitoring and containment vessel temperature and pressure monitoring.

7.5.1 Flux Monitoring System

The objective of the Flux Monitoring System (FMS) is to provide indications and electrical signals proportional to reactor power for reactor plant control and protection. The FMS meets its objective by means of neutron measuring instrumentation comprised of sensors and signal conditioning equipment which provide indications and signals for conditions of reactor shutdown, startup and full power operation.

Neutron sensors located around the periphery of the reactor guard vessel sense thermalized reactor leakage flux which is proportional to the reactor flux and thus to reactor power. Signals from the sensors are conditioned and then used to do the following:

- o Determine the flux status of the reactor from shutdown through startup and all power levels.
- o Provide signals to the Plant Protection System (PPS) to initiate reactor protective trips.
- o Provide signals to the Plant Control System (PCS) for reactor and plant control.
- o Provide neutron flux information for display, annunciation and recording.

A block diagram of the FM System is provided in Figure 7.5-1.

7.5.1.1 Design Description

The Neutron Flux Monitoring System provides three ranges of instrumentation: Source Range, Wide Range and Power Range. Each range of instrumentation is provided in three identical channels comprised of a detector, preamplifier (source and wide ranges), junction box (power range) and signal conditioning equipment. The Flux Monitoring System measures neutron flux proportional to reactor power over a span or more than ten decades from shutdown to above full power and provides indications and electrical outputs for plant protection, plant control, accident monitoring, data handling and display, recording, and annunciation.

The flux detectors will be within the reactor cavity and contained
in thimbles positioned approximately 120° apart on the periphery of the guard
vessel. Each detector will be separately installed in a dry thimble which
extends from the head compartment floor to below the reactor midplane to
7 position the detectors with the center of their active volumes
at or near the core midplane. The thimbles will be surrounded by neutron
thermalizing blocks and provided with gamma shielding at the detector
positions to provide suitable gamma and thermal neutron environments
for the detectors.

Preamplifiers and junction boxes for the system will be located in the head access area.

The signal processing equipment will be located in the control building and will be contained in equipment racks. Within these racks, there will be drawers containing the electronic signal conditioning equipment packaged such that each channel or range within a channel (in the case of the wide range channels) will occupy one drawer. The drawers will be mounted on slides and will be capable of being withdrawn for alignment and maintenance. The front of each drawer will have meters, switches and adjustments for alignment, tests and monitoring of the channel. The arrangement of drawers in the racks will satisfy the separation requirements for redundant PPS channels. (See Section 7.1.2 and 7.2.2)

Remote meters and switches will be provided to the reactor plant
operator to permit him to read the following parameters for each channel:
Source Range Channel logarithmic level and rate, logarithmic percent power
and rate for Wide Range counting and Mean Square Voltage (MSV) channels,
linear percent power for Wide Range and Power Range. Linear and Logarithmic
Source Range level will be provided on the IVTM control console. Audio Source
Range count rate will be provided in the control room and at the IVTM control
7 console. Logarithmic source range count rate will be provided in the Refueling
Communication Center.

The Flux Monitoring System detectors, cabling and signal conditioning equipment will be installed so as to preserve the separation requirements for redundant PPS channels (see Section 7.1.2 and 7.2.2).

Figure 7.5-2 presents the flux level coverage of the FMS instrumentation ranges based on the neutron flux at the detectors. The instrumentation ranges are shown to overlap so as to provide continuous indication from shutdown to more than full power.

7.5.1.1.1 Source Range

The design bases functions and operational requirements as stated in Section 4.3.2.1.5 are accomplished by incorporation of the design features described in the following paragraphs.

Each of the three Source Range channels will use a high sensitivity, dual
57 section BF_3 filled proportional counter assembly to sense thermal neutron flux
54 over the range of approximately 0.1 to 10^4 neutrons per cm^2 per second (nv).
This corresponds to a range from the fully shutdown fresh core conditions 1

to reactor low power (a few Kwt) operation. The sensitivity of the
54 BF_3 counters will be maintained at a minimum of 40 cps/thermal equivalent
57 nv at a gamma background count rate of approximately 1 cps. Shielding will be
provided to limit the gamma dose rate at the detectors due to prompt gammas
from the core and from decay of activation isotopes in the sodium coolant
and structural materials to less than 100R/hr. This will apply over the
operating range of the channel at all times when operation is required.
The BF_3 counters and associated cables will be designed to operate under
normal environmental conditions of 170°F maximum and atmospheric pressure
and under emergency conditions of 260°F maximum and 12 psig during con-
57 tainment testing. The design life goal for the counters is 3 full power
years based on a total fluence of 10^{18} nvt. In order to achieve this life-
time without retracting the counters, the operating voltage will be removed
and the anode of the counter shorted to ground through an appropriate
resistance when the flux level is above the operating range of the channel.

The output pulses from the detector assembly will be amplified by a dual
section pre-amplifier mounted in the head access area and routed to the signal con-
ditioning equipment in the control room racks. This equipment will consist
of dual amplifiers with pulse shaping networks to minimize gamma pulse pile-
up followed by discriminators to reject amplifier noise and gamma pulses.
57 The summed output of the discriminators will drive a count rate circuit followed
by a scaling amplifier which will produce an analog output signal pro-
portional to the logarithm of the input pulse rate. The analog signal
will be displayed on a log scale countrate level meter calibrated from
1 to 10^6 cps. The accuracy and repeatability of the channel will be $\pm$ 3% 17
and $\pm$ 1%, respectively, of linear equivalent full scale under the
worst case environmental conditions of temperature, pressure and input
power fluctuations to be encountered while channel operation is required.
The response time of the count rate circuits will vary with count rate,
being on the order of 30 seconds at the lowest expected average count
rate of approximately 4 cps which will occur during refueling at beginning of core
59 life conditions. The 1σ statistical counting error will be approximately $\pm 7\%$
54 at the 4 cps rate. The response time will decrease with increasing count
rate to approximately 0.1 seconds at a count rate of 3×10^5 cps produced
57 at a few kilowatts. The analog output signal of the scaling
amplifier also drives a differentiating circuit to produce a rate of
change of level signal which is displayed on a linear scale meter from
-1 to 0 to +3 decades per minute. Power supplies mounted in the equipment 17
drawers will provide detector excitation high voltage and low voltage
instrument power. Individual scaler timers will be provided in each
channel which will be driven from a buffered output of the discriminator summer
59 which precedes the countrate circuit. The scaler timers will be programmable
to count for a short preset time, stop, transfer the accumulated count to tempo-
rary storage for PDH&DS readout, immediately restart the count and repeat the
57 sequence to provide an accurate record of the count rate trace to implement the
inverse kinetics rod drop technique used to establish control rod worth. The
scaler-timers will also provide for counting of signal pulses for longer time
57 periods to accurately determine the system calibration constant which relates 17
subcritical reactivity to count rate.

A visual/audio linear count rate circuit will be provided for
operations at shutdown. This circuit will be provided with a switch 1

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57 through which it can be connected to the summed discriminator output of any of the three Source Range Channels, one at a time. The visual portion provides expanded scale indication of the flux level. The audio portion provides tone bursts whose rate changes (increase and decrease) with increasing and decreasing flux level.

Each channel will be provided with built-in counting level and rate of change calibration circuits for channel alignment and pre-operational testing to assure that the instrumentation circuitry is functioning properly.

The Source Range Channels are used to monitor the shutdown and startup flux only, no signals are provided to the PPS. Bistable comparators in each channel will activate individual annunciators in the control room to provide alarms if the specified minimum shutdown reactivity is exceeded during re-fueling or if detector excitation or instrument power is lost in any channel. To improve the operational effectiveness of the source range shutdown reactivity monitoring function, the reactivity alarm is inhibited during core assembly movement which could cause an erroneous alarm. The inhibit circuit is continuously monitored for proper operation and any malfunction of the inhibit circuit will activate a separate control room annunciator. An alarm will also be provided in the control room via the PDH&DS if any one channel deviates from the other two by a preset amount. The shutdown margin alarm bistables and the PDH&DS channel deviation alarm will operate off of the buffered outputs of the log count rate scaling amplifiers. The bistables which provide alarms upon loss of detector excitation voltage as loss of instrument power will operate off of H.V. sense signals developed by voltage divider networks corrected to the preamplifier H.V. outputs to the detector assemblies and off of instrument power supply output voltages, respectively.

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The operation of the source range channels will be under manual control of the reactor operators. These channels will be in operation continuously during reactor refueling and other shutdown conditions. During reactor startup, the source range channels will be used to monitor the core flux level until a predetermined overlap between the source range channels and wide range log count rate channels is obtained. The high voltage will then be removed from the source range detectors by the operator. He will do this by actuating a momentary contact pushbutton switch on the main control panel. When actuated, this switch will remotely interrupt the input power to the detector high voltage supplies, and ground the detector anodes through an appropriate resistor. The relay control circuit established by the momentary contact of this switch will also illuminate a green indicator light located adjacent to the switch. Upon reactor shutdown, the operator will interrupt the relay control circuit by actuating a second momentary contact pushbutton switch when the wide range log counting channels indications fall to a pre-determined level within the source range channels operating range. This action will remove the ground from the detector anodes, restore the detector excitation voltage, extinguish the green indicator light and illuminate a red indicator light. Inadvertent removal of, or failure to restore the operating voltage when needed will be prevented by procedural control, utilizing the separate on/off switches and with monitoring through the color of the illuminated indi-
57 cator light.

7.5.1.1.2 Wide Range

Each of the three channels of Wide Range Instrumentation will use a U^{235} fission chamber to sense neutron flux from low power to above full power by providing, within each channel, overlapping ranges of counting, mean square voltage (MSV) and direct current instrumentation. The counting and MSV instrumentation provides precent power indications on logarithmic scaled meters and rate of change of level on a linear scale from -1 to 0 to +3 decades per minute. These two overlapping ranges of instrumentation are designated as Category 1, Type B, Accident Monitoring equipment as defined in PSAR Section 7.5.11. The d-c instrumentation indicates percent power on a linear scale.

The signal conditioning equipment produces electrical signals which are indicative of the power levels and rate of change of power levels. These signals are used for plant protection, data handling and annunciation. The MSV and linear circuits outputs will be linear to at least 140 percent power and will have a significant positive response to as high a power level as required by the worst case power overshoot for which protection must be provided.

Built-in test circuits and controls will be provided to permit testing and aligning the equipment during plant operation and during plant shutdown.

7.5.1.1.3 Power Range

Each of the three channels of power Range Instrumentation will use B^{10} compensated ionization chambers to sense the neutron flux in a span of from less than one percent power to more than full power. The d-c current

Output of the detector will be processed in the signal conditioning equipment to provide linear indication of percent power and linear output signals for plant protection, plant control, data logging and annunciators. This instrumentation operates over the same flux span as the direct current circuitry of the wide range instrumentation to add redundancy and diversity to the wide range power measurements.

The output of this instrument will be linear to at least 140 percent power and will have no foldover to as high a power level as required by the worst case power overshoot for which protection must be provided.

Built-in test circuits and controls will be provided to permit testing and aligning the equipment during plant operation and plant shutdown.

7.5.1.2 Design Analysis

The Flux Monitoring System will be a functional subsystem of the Plant Protection System and will meet the safety related channel performance and reliability requirements of the CRBRP General Design Criteria, IEEE Standard 279-1971, applicable Regulatory Guides, criteria of Section 7.5.11 and other appropriate criteria and standards by complying with the applicable design requirements delineated in Section 7.1.2.

The FMS meets CRBRP General Design Criterion 21, which is applicable to instrumentation for normal and accident conditions, as follows:

- The shutdown flux level will be monitored at all times while fuel is in the core so as to provide safe operational control of the reactor during low power, normal shutdown, refueling and shutdown maintenance operations.

- The reactor flux will be continuously monitored during operation from shutdown to full power operation (i.e., overlap will exist between cascaded channels so that all power levels can be monitored without a gap in range).

- Reactor power operations will be continuously monitored with linear response to power up to at least 140% full power. Significant positive response will be provided to as high a power level as required by the worst case power overshoot for which protection must be provided. This positive response will be provided for as long as is required to seal in the scram trip.

- The FMS instrument response time delays will meet the response requirements of the Reactor Shutdown and Plant Control Systems.

- Indication of reactor power level and rate of change of power level will be provided to the operator. One set of meters and a selector switch will be provided for each range of instrumentation permitting the operator to select one channel at a time to be displayed on the related meters. Seven power level meters, five selector switches and three rate of change of power meters will be provided for the operator.

- o The source range level will be indicated in logarithmic counts per second and rate of change of level in decades per minute. Linear count rate will be provided at shutdown at the refueling console and at the FMS system panels in the control room. Audible count rate indication will be provided in the control room and in containment at the refueling console.

- o The wide ranges will be indicated as follows:

  Counting channels - Logarithmic percent power level and decades per minute rate of change.

  MSV channels - Logarithmic percent power level and decades per minute rate of change.

  DC channels - Linear percent power level.

- o The power range will be indicated in linear percent power level.

Preliminary Failure Mode and Effect Analysis results applicable to the FMS have been determined in an analysis of possible failure modes and their effects on the Reactor Shutdown System performance and are presented in Tables C.S. 1-4 and C.S. 1-5.

7.5.2 Heat Transport Instrumentation System

7.5.2.1 Description

The Heat Transport Instrumentation System provides sensors, associated signal conditioning equipment and controls other than Plant Control, for the Primary Heat Transport, the Intermediate Heat Transport, the Steam Generator, and the Steam Generator Auxiliary Heat Removal System (for a discussion of the SGAHRS safety-related display instrumentation see Section 7.4.1.1.10). The signals from the sensors are conditioned and then supplied to the Reactor Shutdown System logic, the Plant Control System, the Plant Data Handling and Display System, and the Plant Annunciator System as appropriate. The location of the Heat Transport Instrumentation is provided in Figures 5.1-2 and 5.1-4 (P&IDs).

7.5.2.1.1 Primary and Intermediate Sodium Loops

Reactor Inlet Pressure

The measurement is made by pressure elements installed in the cold leg of the primary loop piping just before it enters the reactor vessel. NaK filled capillaries from the pressure elements are connected to pressure transducers which develop electrical signals proportional to the pressure. These pressure transducers provide a secondary boundary if the bellows in the pressure elements should fail.

Each pressure transducer consists of a diaphragm which moves in response to pressure changes in the NaK filled capillary, a strain gage which converts diaphragm motion to resistance change, and a bridge and amplifier to convert strain gage resistance change to a standard signal.

Since pressure element replacement requires plant shutdown, two pressure elements per loop are provided.

The signals from the six, two per loop, pressure measurements are transmitted to the control room in three separate isolated PPS channels for use in the Reactor Shutdown System logic. The Reactor Shutdown System sup-
59 plies buffered signals to the PDH & DS.

Primary and Intermediate Loop Flow

The flow measurements are made by a permanent magnet flowmeter
located in the cold leg of each primary and intermediate loop (except
49 for intermediate loop 2, which has the PM flowmeter in the hot leg). The magnet assembly is in the shape of an inverted "U" which is suspended around the pipe. The magnet assembly is mounted rigidly to the building structure and is physically separated from the pipe.

Type K thermocouples are installed in the magnet structures to moni-
59 tor the magnet temperatures. The signals from these thermocouples are routed to a local panel. Provisions will be made to permit periodic monitoring of the magnetic flux of the flowmeters without disassembly or entrance into HTS cell. This is also accomplished at the local panel.

Four independent pairs of 3/8" (approx.) electrodes are attached to the pipe. The electrodes are of the same composition as the pipe so that thermal potentials are not developed. Three pairs of electrodes are connected to the conditioning equipment. The fourth pair is available for a portable measuring instrument or as an installed spare.

Flexible mica, polyimide and fiberglass insulated cables in separate conduits to meet PPS separation requirements are used to bring the four signals from each flowmeter assembly out of the Heat Transport System cell. The signals are then routed to signal conditioning equipment.

From the signal conditioning equipment, the signals are sent to the control room for the Reactor Shutdown System logic which in turn supplies
59 buffered signals to the PCS and the PDH & DS.

IHX Primary Outlet Temperature

The IHX primary outlet temperature measurement is made by three Chromel/Alumel thermocouples per loop installed in thermowells in the elevated section of the HTS cold leg piping nearest the IHX primary outlet. The thermocouples are 1/8" insulated junction swaged to 1/16" at the tip to provide the required time response. The thermowell is also swaged at the tip. The thermocouples are spring loaded against the bottom of the well. Although failures of the wells are not expected, as confirmed by tests and analysis, the head of the thermowell, including the cable penetration, is sealed to provide a secondary boundary for the sodium. Tests have shown that this system will provide a time response less than 5 seconds. Flexible mica, polyimide and fiberglass insulated thermocouple extension wires in conduit are used to bring the signals out of the Heat Transport System Cell. The signals are then routed to the containment mezzanine into reference junctions and signal conditioning equipment. The conditioned signals are transmitted to the control room for the Reactor Shutdown System logic. The Reactor Shutdown System provides buffered signals to the PCS and PDH & DS.

Primary and Intermediate Hot and Cold Leg Temperature

The primary and intermediate hot and cold leg temperatures are measured to determine and record operating conditions and to calorimetrically calibrate the permanent magnet flowmeters. The measurement is made by two duplex element resistance temperature detectors (RTDs) per loop, installed in thermowells. Although failures of the wells are not expected, as confirmed by tests and analysis, the head of the thermowell, including the cable penetration, is sealed to provide a secondary boundary for the sodium. The signals from the RTDs are routed to signal conditioning equipment which converts the resistance variation to a standard signal level for transmission to the PDH & DS.

Primary and Intermediate Pump Discharge Pressure

The primary and intermediate pump discharge pressure measurements monitor pump performance. In addition the primary pump outlet in conjunction with the intermediate IHX outlet pressure provide the primary loop/intermediate loop differential pressure. The measurements are made by pressure elements installed in the elevated section of the drain line from the discharge piping of the sodium pump. NaK filled capillaries from the pressure elements are connected to pressure transducers which develop electrical signals proportional to the pressure. These pressure transducers provide a secondary boundary if the bellows in the pressure elements should fail. The conditioned signal is supplied to the PDH & DS. Since this pressure element is located in an inerted cell and replacement would require entry into the cell and draining of the loop, two pressure elements per loop are provided.

Intermediate IHX Outlet Pressure

The intermediate IHX outlet pressure measurement is used to monitor the loop and IHX operational performance history. The measurements are made by pressure elements installed in the intermediate loop piping between the IHX and the superheater. NaK filled capillaries from the pressure elements are connected to pressure tranducers which develop electrical signals proportional to the pressure. The pressure transducers provide a secondary boundary if the bellows in the pressure elements should fail. The conditioned signal is supplied to the PDH and DS.

IHX Differential Pressure

The primary sodium pump discharge pressure and the IHX Intermediate Loop outlet pressure detectors are used to provide a differential measurement of the IHX Primary/Intermediate pressure difference, which is maintained above 10 psi during normal operating conditions. The differential pressure measurement is alarmed if the intermediate loop pressure drops to 10 psi above the primary loop pressure to alert the operator for corrective action to assure intermediate to primary differential pressure is maintained above the minimum required.

Intermediate Pump Inlet Pressure

The intermediate pump inlet pressure measurements provide a signal to monitor pump performance. Used with the pump outlet pressure, the differential pressure across the pump is obtained. In the primary loop, the reactor pressure is used for this surveillance. The measurements are made by pressure elements installed on the piping between the evaporators and the pump inlet. NaK filled capillaries from the pressure elements are connected to pressure transducers which develop electrical signals proportional to the pressure. The pressure transducers provide a secondary boundary if the bellows in the pressure elements should fail. The conditioned signal is supplied to the PDH & DS.

Intermediate Expansion Tank Level

Two separate level measurement channels are provided; both channels are used for indication in the control room and DH & DS and for alarm. Alarm channels provide a broad range measurement that covers possible high and low levels during plant operation as well as the IHTS fill level. The PDH & DS uses measurements for intermediate loop sodium inventory (see also Section 7.5.5). The level probes are designed to be replaceable.

Evaporator Sodium Outlet Temperature

Three thermocouple (as described above in the paragraph on IHX outlet temperature) channels are provided to measure the sodium temperature at the outlet of the evaporators in each loop. The thermocouples are placed just after the pipes from each evaporator join to form two single lines. These three signals are conditioned separately and provided to the Reactor Shutdown System logic. The Reactor Shutdown System in turn provides buffered signals to the PDH & DS.

7.5.2.1.2 Sodium Pumps

Sodium Level

Sodium level is measured in each pump tank. The signal provides indication and alarm. The alarm is used to notify the operator of abnormal operation and allow initiation of action to prevent pump damage. The signal is also provided to the PDH & DS where it can be used in calculation of sodium inventory.

Primary and Intermediate Pump Speed

Five primary and intermediate pump speed signals are provided. Three for the Reactor Shutdown System and two for the speed control system. These signals are obtained from the shaft of the vertical drive motor.

Three redundant signals are produced for each pump conditioned and transmitted to the control room as the three channels required for the Reactor Shutdown System. These signals are buffered and supplied to the PDH & DS.

Two redundant speed signals are provided to the pump speed control
59 equipment where it is used as a feedback signal.

Pony Motor

The pony motors are Class 1E motors and are supplied power from the Class 1E 480 VAC busses.

Non-class 1E signals are provided to the main control room to indicate the pony motors are running. These signals are pony motor speed indicators which are located in the main control panel and pony motor current which is available through the PDH & DS. Also start and stop lights are on the main control room panel which are from the pony motor starters.

During pony motor operation indication is available on the main control room panel from sodium flow.

7.5.2.1.3 Steam Generator

Sodium Flow

Venturi flowmeters are provided, one loop only, to accurately measure the sodium flow rate through each of the superheater outlet ports. The accurate flow data is used for determination of the performance characteristics typical of the superheaters and evaporators.

Sodium Temperature

The evaporator and superheater outlet temperature is monitored, on all three loops, by Resistance Temperature Detectors (RTD). The superheater inlet is monitored, on one loop only, also by an RTD for purposes of steam generator performance evaluation. These temperature sensors provide signals for the PDH & DS. The evaporator bulk outlet temperature is measured with three thermocouples and are part of the Reactor Shutdown System.

Sodium Pressure

For the purpose of steam generator performance evaluation, pressure is measured, in one loop only, at the superheater inlet, superheater outlet (both legs) and evaporator outlet. The type of pressure sensor used is the same as the one for intermediate pump inlet pressure. These pressure measurements provide pressure signals to the PDH & DS.

Steam and Water Flow

- Feedwater Mass Flow - sensed by three differential pressure elements across one venturi in the inlet line to each steam drum.

- The temperature corrected feedwater flow signals are supplied to the Reactor Shutdown System logic. The Reactor Shutdown
59 System provides buffered signals to PCS and PDH & DS.

- Steam Mass Flow - sensed by three differential pressure elements across one venturi in the outlet of each superheater. The temperature and pressure corrected mass signals are supplied to the Reactor Shutdown System logic. The Reactor Shutdown System pro-
59|29 vides buffered signals to PCS and PDH & DS.

- Steam Drum Blowdown Flow - sensed by flow orifice (differential
59 pressure) in the blowdown line for each steam drum. The signal is provided to the PDH & DS.

- Evaporator Inlet Flow - sensed by a differential pressure element across a venturi in the inlet line to one of the evaporators in one loop only. This is to aid in the performance evaluation of a typical evaporator module.

Steam and Water Temperature

- Feedwater Temperature - sensed by three resistance temperature detectors in the steam drum inlet line. The signal provides temperature compensation for the feedwater flow signal. Buffered
59 signals are supplied to the PDH & DS.

- Recirculating Water Temperature - sensed by a thermocouple detector in the recirculation pump discharge header. The signal is pro-
59 vided to the PDH & DS.

- Saturated Steam Temperature - sensed by a thermocouple detector in the outlet header from the steam drum. The signal is provided
59 to the PDH & DS.

- Superheat Steam Temperature - sensed by three resistance temperature detectors in the superheater outlet line. The signal provides temperature compensation for the steam flow. Buffered sig-
59 nals are supplied for PCS and PDH & DS.

- Evaporator and Superheater Inlet and Outlet Temperature - sensed by RTDs located at the inlet and outlet nozzles for one evaporator and superheater in one loop only. Used for performance evaluation for a typical generator module.

- Steam Drum Blowdown Temperature - sensed by a thermocouple located on the blowdown line. The signal provides temperature compensation for the steam drum blowdown flow and is also supplied
59 to the PDH & DS.

Steam and Water Pressure

- Feedwater Pressure - sensed by a pressure element located on the inlet line to each steam drum. The signal is used for PCS and is
59 supplied to the PDH & DS.

- Steam Drum Pressure - sensed by three pressure elements located on an appendage from the steam drum. The signal provides pressure compensation to steam drum level. Buffered signals are used for
59 PCS and PDH & DS.

- Recirculation Pump Discharge Pressure - sensed by a pressure element located on an appendage from the recirculation pump inlet header. This measurement is required for recirculation pump protection and
29 59 performance analysis by the PDH & DS.

- Superheat Steam Pressure - sensed by three pressure elements located on the loop output steam line. The signal is supplied to the Reactor Shutdown System logic. Buffered signals are supplied
59 to PCS and PDH & DS.

- Evaporator and Superheater Inlet Pressure - sensed by pressure elements located off the inlet nozzles for one evaporator and superheater in one loop only. The signal is supplied to the
59 PDH & DS for performance evaluation of a typical steam generator module.

Evaporator and Superheater Outlet Pressure

Sensed by pressure elements located off the outlet nozzle of each evaporator and superheater. The signal is used for SWRPRS Control and is
59 supplied to the PDH & DS for the performance evaluation of a typical steam generator module.

Steam Drum Level

Sensed by three differential pressure elements measuring the differential pressure between a reference column and the water head in the steam drum. This measurement is density compensated. The signal is supplied to the Reactor Shutdown System logic. Buffered signals are supplied to PCS and
59 PDH & DS.

Impurity Monitoring

Hydrogen and oxygen concentration monitors are utilized on the sodium side of the steam generators for detection of water-to-sodium leaks. The equipment is discussed in detail in Section 7.5.5.

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Indication and Alarm

Indication and alarms are provided to keep the operator informed of the status of system and equipment and to quickly determine location of malfunctioning equipment. All of the measurements discussed in this section are processed by the Plant Data Handling and Display System. In addition, the following parameters are continuously monitored in the main control room:

- Feedwater pressure, temperature, and flow.
- Steam drum pressure, temperature, and level.
- Superheat steam pressure, temperature, and flow.
- Turbine inlet pressure, themperature, and flow.
- 29 Recirculation Pump Discharge Pressure.

7.5.2.2 Analysis

Instruments part of the Reactor Shutdown System will comply with the PPS Design Requirements (See Section 7.1.2 and 7.2.1). The design analy-sis for the Reactor Shutdown System applies (Section 7.2.2), and a Failure Mode and Effects Analysis is performed as shown in Table C.S. 1-2 for PM flow meters; Table C.S. 1-6 for Sodium Pressure Sensors; Table C.S. 1-7 for Tachometers; Table C.S. 1-8 and C.S. 1-9 for the Pressure Sensors and Associated Temperature Sensors for Flow Measurements; and Table C.S. 1-10 for Steam Drum Level Instruments.

41 Non-PPS instruments will be duplicated or triplicated either to obtain more representative values of the parameter measured, or to permit continued parameter measurements upon failure of one instrument, when sensor replacement requires plant shutdown.

Built-in calibration means have been provided to ensure reliable and accurate monitoring throughout the plant life, e.g., the permanent magnet flowmeters can be calorimetrically recalibrated by the hot and cold leg temperature measurements after initial plant startup calibration.

Seismic considerations have been also included in the design of the Heat Transport Instrumentation, e.g., the clearance between the pipe and the magnet structure of the permanent magnet flowmeters is such that contact is prevented under all postulated conditions of vibrations, including a Safe Shutdown Earthquake (see Sections 3.7 and 7.2.1 for Seismic Design).

The design includes the requirements of sodium coolant contingencies by providing a double barrier against sodium spillage for the temperature and pressure sensors installed in the Primary and Intermediate Heat Transport coolant boundaries.

7.5.3 Reactor and Vessel Instrumentation

7.5.3.1 Description

The Reactor and Vessel Instrumentation System includes all in-vessel temperature, sodium level and vibration sensors for instrumenting the reactor parameters required for the Reactor Shutdown System, PCS, surveillance and design verification. It also includes signal conditioning equipment needed to make the sensor signal usable in the systems receiving the signal.

Table 7.5-2 shows the in-vessel instruments provided, their location, their quantity and purpose.

7.5.3.1.1 Sodium Level

A total of six sodium level sensors are provided. All of these sensors are mounted in wells to provide the physical barrier maintaining the integrity of the primary loop closed system. The sensors are induction type probes continuously sensitive over their entire length. Four of the units, located approximately equally spaced on the top of the reactor, are short with a sensing range of from 6 inches above the operating level to 24 inches below. Three of these provide the level signals to the three Reactor shutdown system logic and are thus isolated from each other and from non-PPS equipment. The fourth is an installed spare unit providing a means of maintaining the three operating channels without a shutdown in the event of failure of one of them. The remaining two level sensors are located close to one of the short units but provides a measuring range from 6 inches above the operating level to six inches below the top of the outlet nozzle. It has approximately sixteen feet of sensing length. The two signals are supplied to two indicators located on the main control panel and are monitored at all times, including refueling. These two wide range sodium level channels are Category 1, Accident Monitoring instruments.

7.5.3.1.2 Temperature

All in-vessel temperatures are sensed by 1/8 inch, chromel alumel, ungrounded, stainless steel sheathed thermocouples. Thirty wells are provided for thermocouples located in the sodium at the exit from the core. These thermocouples provide signals to the PCS and the PDH&DS. Additional wells are provided at the core exit (308), core periphery (2), and on parts of the upper internal structure (6) for thermocouples providing signals to the PDH&DS for surveillance and design verification. The 338 thermocouples provided at the core exit are Category 3, Accident Monitoring instruments.

7.5.3.1.3 Non-replaceable Instruments

Within the reactor vessel, four biaxial accelerometers are mounted on the upper internal structure so that they cannot be replaced. These sensors are not required to function beyond the first six months of operation although they are required to physically withstand the sodium environment for the life of the reactor. The signals provided by these sensors provide design verification information and the location of the sensors and their leads will not affect the safety of the plant.

7.5.3.2 Analysis

The in-vessel sensor mounting is designed to be operable during the combined stresses imposed by the reactor coolant velocity, vibration, pressure, environment, temperature, thermal shock, and radiation in order to provide operational lifetime that will not significantly affect the reactor availability. The sensors and their lead-out conductors are sufficiently rugged so that they will not be damaged during refueling and maintenance. Thermocouples used for measuring the sodium temperature are mounted to avoid close proximity to the structures so that the temperatures sensed will be that of the coolant and not be influenced by the structures.

The Sodium level instruments, which are part of the Reactor Shutdown System, will comply with the PPS Design Requirements (see Sections 7.1.2 and 7.2.1). The design analysis for the Reactor Shutdown System applies (Section 7.2.2) and a Failure Mode and Effects analysis is performed as shown in Table C.S.1-1.

7.5.4 Fuel Failure Monitoring System

The Fuel Failure Monitoring System (FFMS) is required to:

1. Detect and locate a fuel or blanket failure in the presence of up to four other existing failures. In addition, provide system design and accommodation capability to detect and locate a fuel or blanket failure in the presence of up to 59 previously failed pins.

2. Characterize the failed pins as to burnup and other information, to permit correlation with core and blanket history, thus enhancing location capabilities.

3. Detect less than 1.5 cm2 of fuel area exposed to sodium by providing a delayed neutron detector subsystem on each sodium loop of the Primary Heat Transport System.

The FFMS is comprised of several independently functioning subsystems, each providing information to the plant operations staff. This system does not provide reactor trip signals but does supply information for surveillance, display, and alarm purposes.

7.5.4.1 Design Description

The following subsystems make up the FFM system.

1. Cover Gas Monitoring

   This subsystem continuously samples the cover gas and determines, through gamma analysis:

   - the concentration of selected radioactive fission gases to inform the plant operations staff upon each instance of fuel or blanket pin cladding failure.

   - the concentration of radioactive fission gases to characterize the failed pins as to burnup and other information.

2. Reactor Delayed Neutron Monitoring

   This subsystem continuously monitors for the presence of fission products in the sodium coolant which decay with the emission of neutrons. A predetermined increase in the neutron signal from the Primary Heat Transport System sodium, above the normal background level, is taken as an indication of fuel contact with sodium.

   The Impurity Monitoring and Analysis System provides verification of fuel exposure to the sodium by removing sodium with a grab sampler and by subsequent laboratory analysis for fuel and fission product material.

3. Failed Fuel Location

   Stable (non-radioactive) xenon and krypton isotopes (that are not fission products) are placed in each fuel and blanket assembly pin. Each assembly has a unique ratio of isotopes which will be released to the cover gas upon failure of a pin in the assembly. Analysis of a processed sample of the cover gas, using a mass spectrometer, is used to identify the assembly containing the failed pin.

The FFM subsystems are described in greater detail in the following sections. A block diagram of the FFM system is provided in Figure 7.5-3.

7.5.4.1.1 Cover Gas Monitoring Subsystem

The Reactor Cover Gas Monitoring Subsystem (CGMS) detects a fuel or blanket failure in the presence of up to four existing failures. It continuously monitors the reactor cover gas for radioactive gases which are present as background (Ne-23 and Ar-41) and gaseous fission products which escape from failed fuel and radial blanket pins. The cover gas first passes through a sodium vapor trap, which is located in the Reactor Containment Building, then passes through a charcoal delay bed which is located in a shielded cell in the Reactor Service Building. The purpose of the delay bed is to concentrate the radioactive xenon and krypton fission gases relative to Ne-23 and Ar-41 in order to increase the signal to background ratio. Monitoring is done with a planar germanium (Ge) gamma detector which continuously monitors specific fission gas radioisotopes. An alarm in the main control room (plant annunciator) will be activated in the event of an abnormal activity level increase for any of these radioisotopes. A multichannel analyzer, including a minicomputer, analyzes the signal from the detector to display the entire gamma ray spectrum. The minicomputer with additional input of the reactor power provides basic characteristics of the failure, i.e., magnitude and burnup, which may be used to supplement the Failed Fuel Location Subsystem through correlation with core and blanket history.

Failure of the minicomputer does not affect the failure detection capability of the CGMS. In this case, the multichannel analyzer memory will still be functional; however, the characterization capability will be lost. During normal operation when there are no failures, the minicomputer is not used. However, if there is a fuel failure, which requires characterization, data from the multichannel analyzer can be recorded and analyzed manually. In addition, a gas grab sample could be obtained, and then analyzed in the Plant Service building Laboratory which has equipment equivalent to the CGMS multichannel analyzer and minicomputer. By using this equipment one can characterize the fuel failure and provide the desired information.

7.5.4.1.2 Reactor Delayed Neutron Monitoring Subsystem

The reactor delayed neutron monitoring subsystem consists of three monitoring channels, one channel for each of the three primary loops. Each channel contains signal conditioners, readout equipment, and an assembly containing multiple neutron detectors. Each detector assembly is mounted in a shield/moderator block adjacent to one of the three primary heat transport system hot leg pipes. The multiple detectors in each assembly provide sufficient countrate for detection as well as redundancy in case of a single detector failure. The delayed neutron monitoring subsystem is required to be in service at all times during plant operation. In certain situations, continued plant operation is permitted for short periods of time following the failure of a channel monitoring a single loop. In order to allow such operation, a number of prerequisite conditions must exist. These include requirements that the CGMS system is operational and shows no changes during the outage, and that the other two DND channels are also operational and provide steady output with no indication of change during the outage. The system design provides high availability since detector repair or replacement can be accomplished during operation and multiple detectors are in each source/moderator block. Also, the preamplifiers and signal conditioning equipment are accessible for maintenance, checkout, and repair or replacement at all times during reactor operation. This subsystem is seismically qualified for an operational basis earthquake.

Coolant sodium transported past the detector assembly, is continuously monitored for delayed neutrons emitted by decay of radioactive precursors in the sodium. The system sensitivity is dependent on the signal-to-background ratio of the system. Signal is defined as detected delayed neutrons produced by recoil of precursors from fuel exposed by cladding failure, or from fission of fuel washed out into the sodium through a failure. Background is defined as detected neutrons from known sources which are not initially related to failed fuel (fuel pin contamination, fissionable impurities in core structural materials, fissionable materials in the sodium, and neutrons from the reactor).

The shielding and moderator assembly provides 1) reduction of gamma interference from Na-24, 2) moderation of neutrons, 3) capability for remote insertion of a calibrated neutron source, 4) capability for insertion and removal of the detector assemblies from the reactor containment building operating floor without deinerting the PHTS cells.

Signals from the three BF_3 detectors are routed to individual inputs of a preamplifier. Each input of the preamplifier provides individual electronic discrimination against gamma-caused counts from each detector. After discrimination, the neutron-caused counts from the three BF_3 detectors, 1 per coolant loop, are combined in a summing circuit, and routed to a local control panel located in the Reactor Containment Building. The local control panel provides controls for remote adjustment of the preamplifier discriminators, a low voltage power supply, a detector-bias high-voltage supply, a comparator for low detector bias voltage alarm and a front panel function switch to select TEST, CALIBRATE or OPERATE modes. These signals are routed to a panel in the main control room displaying the output on three five decade logarithmic count rate meters and a three pen five decade logarithmic strip chart recorder.

Connections are provided to the Plant Data Handling and Display System for signal logging and high signal alarm, and for alarm on low detector bias voltage. A separate high signal alarm is provided to the Plant Annunciator.

7.5.4.1.3 Failed Fuel Location Subsystem

The Failed Fuel Location Subsystem (FFLS) locates a fuel or blanket failure in the presence of up to four existing failures. Stable xenon and krypton isotopes of differing ratios are placed within each fuel and radial blanket assembly pin in the total approximate amount of 2 standard cubic centimeters for the purpose of locating leakers. All individual pins of a given assembly contain the same unique mixture of tag gases. If a pin failure occurs, a fraction of the tag gases and fission gas in the pin escapes to the reactor cover gas plenum. The failure occurrence is detected by the Cover Gas Monitoring Subsystem (CGMS) in the Reactor Service Building which continuously measures the activity of the radionuclides in the cover gas. A failure alarm by the CGMS will initiate operation of the FFLS, which is also located in the Reactor Service Building. This operation involves the passing of approximately 5 standard cubic feet of cover gas through a charcoal trap which is cryogenically cooled with liquid nitrogen. This process preferentially traps the xenon and krypton gases. The trap is then heated and the concentrated gases are transported to the mass spectrometer which analyzes the tag gas isotope ratios. The failed assembly is identified by matching the results of the mass spectrometer analysis with previously determined analysis of all gas tags in the reactor, with suitable correction for burnout, production, and background.

A charcoal trap will obtain a sample of cover gas from the reactor plenum as soon as possible before the argon purge sweeps the tag gases out of the reactor plenum, and they are lost to the Radioactive Argon Processing System (RAPS) via the equalization line and overflow vessel (see Figure 9.5-2). The purging action by RAPS dilutes the released tag gas concentration in the reactor cover gas plenum by a factor of 1.7 per hour.

7.5.4.1.4 Test and Inspection

Prior to reactor startup, selected instruments are tested by inserting electrical test signals into the signal conditioner near the sensor. By observation of the indicators, recorders, or alarms, the operating personnel can determine whether the instrument is functioning properly. Each fission gas detector will be designed for the provision to be exposed to a calibrated, long-lived source, consisting of a radioactive isotope with a decay emission of different energy than those originating from fission gases, for testing the sensors. The fission gas detection systems will contain a provision for inserting a gas sample of known activity, for initial calibration of the detectors, and for correlating detector response to the calibrated test source with the gas sample activity. Each delayed neutron monitoring system will be designed for the provision to be exposed to a calibrated radioactive source for on-line calibration of the detector system.

The FFM detection systems will be in operation during reactor startup, normal operation, shutdown, and refueling, to provide information on fuel failures to reactor operation personnel. Normal background readings will be established for various operating modes (shutdown, startup, steady-state power), and this information, along with the use of calibrated sources, will be used to periodically calibrate and check out FFM systems and instruments.

7.5.4.2 Design Analysis

The reactor is designed to accommodate operation with failed fuel (Section 15.4 addresses operation with failed fuel). Thus the Fuel Failure Monitoring System is not required to prevent accidents associated with the release of radioactivity. The Radiation Monitoring System provides the means for detecting, measuring, and warning of excessive radiation levels within the plant, and of excessive radionuclide concentrations within and external to the plant. However, the FFM will be available during the various phases of reactor operation, including startup, shutdown and refueling, to provide information for fuel management to the plant operations staff. The ability to locate a failed assembly will permit the modification of the fuel management scheme to allow removal of failed assemblies during normal refueling cycles without severe downtime penalties. Detection and verification of fuel failure resulting in exposure of fuel to sodium will require the removal of exposed assemblies, either during planned shutdown or shutdown for that purpose, to minimize the maintenance problems resulting from excessive fuel contamination in the Primary Heat Transport System coolant and components. The Fuel Failure Location Subsystem will allow the leaking assemblies, detected by the Cover Gas Monitoring Subsystem, to operate as long as additional failures in the core do not interfere with the function of the tag gas location system.

7.5.5 Leak Detection Systems

7.5.5.1 Liquid Metal-to-Gas Leak Detection System

A Liquid Metal-to-Gas Leak Detection System is provided to detect and identify the location of leaks for the purpose of continuous surveillance of the liquid metal systems boundaries. Leak detection instrumentation is provided for the following systems: 28

a. Reactor Enclosure

b. Primary Heat Transport System

c. Intermediate Heat Transport and Steam Generator System

d. Impurity Monitoring and Analysis, Refueling and Auxiliary Liquid Metal System

7.5.5.1.1 Design Bases and Design Criteria For the Liquid Metal-To-Gas Leak Detection System

The design bases of the Liquid Metal-to-Gas Leak Detection System arises from the need to protect plant equipment, considerations of maintenance and plant availability, and the corrosion effects of sodium compounds on stainless steels at high temperatures.

Considering the significance of corrosion with respect to piping integrity, it is appropriate that the design criteria assure that the Liquid Metal-to-Gas Leak Detection System provide reliable detection for the Primary and Intermediate Heat Transport In-Containment Systems in a small fraction of the nominal time to penetrate the pipe by local corrosion. The effects of corrosion on the CRBRP PHTS piping have been thoroughly assessed in WARD-D-185 "Integrity of the Primary and Intermediate Heat Transport System Piping In-Containment," Reference 1.6 of the PSAR. In summary, leaks of 100 gm/hr may cause local corrosion in 3600 hrs and general corrosion in 18,000 hours at temperatures near 1000°F. At temperatures less than 700°F, the corrosion rate becomes extremely slow. The Leak Detection System will detect leaks of 100 gm/hr in pipes and components operating at temperatures greater than 700°F in less than 250 hrs.

Design Criteria have been established to guarantee reliable plant operation with pipe temperatures greater than 700°F. These include:

1. The PHTS and in-containment IHTS shall be monitored for leaks by diverse methods each capable of providing the required time response.

2. Capability shall be provided to procure a filter sample for laboratory analysis to provide a highly reliable confirmation method. Filter samples should be analyzed a minimum of once every 1000 hrs.

3. The Liquid Metal-to-Gas Leak Detection System must operate after an operating basis earthquake (OBE).

4. The leak detection system shall be equipped with provisions to readily permit testing for operability and calibration during plant operation.

5. A reliable self-monitoring provision shall be provided to detect component failure.

6. Upon loss of ability to fulfill the specified time response, the plant will be placed in a hot shutdown condition. Technical specifications will be imposed to assure IHTS leak detection capability during plant operation.

7. The system shall be qualified to operate in its environment.

8. The Liquid Metal-to-Gas Leak Detection System will detect a leak of 100 gm/hr from the IHTS in an air-filled all within 20 hours.

Additional Design Criteria of the Liquid Metal-To-Gas Leak Detection System required to protect plant and capital investment, limit maintenance and protect plant availability are outlined below:

1. The Liquid Metal-To-Gas Leak Detection System shall detect and locate liquid metal-to-gas leaks throughout the plant between the temperatures of 375 to 1000°F as required to fulfill continuous monitoring requirements of Appendix G, "CRBRP Plan For Inservice and Preservice Inspections."

2. The Leak Detection System shall be able to identify the general location of the leak.

This system is not needed for initiation of plant shutdown, for removal of decay heat or for reduction of off-site radiation exposure to acceptable levels; therefore, it is generally classified as a non-safety system. However, the leak detection capability in the PHTS, DHRS and EVST systems shall be Safety Related; i.e., Seismic Category I, 1E power, environmental qualification, and redundancy or diversity, but not both. (This requirement was established in December 1982 as a result of NRC's safety review. The remainder of this PSAR does not specifically reflect this requirement, but the leak detection technology and general systems arrangements identified in the PSAR reflect the technology and systems arrangements that will be used to satisfy this arrangement.)

The safety related instrumentation provided to accommodate liquid metal leaks is described in Section 7.5.3.1.1. The passive engineered safety features provided to mitigate the effects of liquid metal leaks are described in Section 3.8.

7.5.5.1.1.1 Design Description

General

Detection equipment is provided to monitor the primary and intermediate sodium coolant boundaries to identify comparatively small leaks when they occur.

The leak detection methods selected for the following installations are:

1. Particulate monitors (radiation detectors), Sodium Ionization Detectors (aerosol detectors), and chemical analysis for atmosphere monitoring in selected cells.

2. Plugging filter aerosol detectors (PFADs) for Main Heat Transfer System piping and guard vessels, major components, and for inerted cell atmosphere monitoring.

3. Contact detectors in the space between the bellows and the stem packing of the bellows sealed sodium valves.

4. Cable detectors in guard vessels and under major liquid metal components.

Of the types of leak detection devices that comprise the Leak Detection System, only sodium aerosol leak detection devices show a difference in their response when operated in an air atmosphere as opposed to an inert atmosphere. The time for a detector to respond to a leak in air is generally shorter than in an inert atmosphere. The electrical sensing types such as cable and contact detectors show no difference in response due to operating atmospheres. However, the potential for higher moisture content in air can result in greater inhibition to sodium flow when the leak is very small.

Considerations which materially affect detection times include: sodium leak rate, sodium temperature, and cell size. Test data (See Reference 5) confirm that sodium leaks of 100 gm per hour in an air or inert atmosphere can be detected by aerosol detection over the operating temperature ranges, within the detection time periods identified in Figure 5.1.1 of WARD-D-0185, "Integrity of Primary and Intermediate Heat Transport System Piping in Containment", (Reference 2, PSAR Section 1.6). Larger leaks (on the order of kg/min) will be readily detected by two or more systems in minutes.

The electrical sensing types of detectors (cables and contact) respond with an alarm when liquid metal causes an electrical short between the electrode and its protective sheath. The Sodium Ionization Detectors (SIDs) provide an alarm when the aerosol concentration reaches a level of about 10^{-11} gm/cc. The PFADs, which are integrating devices, respond with an alarm when the differential pressure across a filter has increased by 2 inches of water. The time for this response is related to aerosol concentration as shown on Figure 7.5-7. For example, at 1×10^{-11} gm/cc, the time response is approximately 250 hours. Both SIDs and PFADs have filters which are chemically examined for sodium on a monthly basis so that leaks which result in aerosol concentrations lower than 1×10^{-11} gm/cc will also be detected. A leak resulting in a concentration of approximately 2×10^{-13} gm/cc is detectable by chemical examination of the filter pads. The sodium aerosol concentration resulting from a 100-gm/h leak in inerted CRBRP cells ranging in volume from 15,000 to 115,000 ft^3 is shown on Figure 7.5-8. In the operating temperature range of 700-1000°F, the leak detection criteria are easily met with either SIDs or PFADs. Aerosol generation is greater in air-filled cells than in inerted cells. Therefore, sodium leaks can be detected faster. A 100 gm/hr leak in an IHTS air-filled cell can be detected within 20 hours compared to 250 hours from an equivalent inerted cell. In addition, during reactor operation, the radiation particulate monitoring system will detect leaks resulting in aerosol concentration of approximately 10^{-15} gm/cc in those cells containing primary sodium.

The aerosol detectors are connected to the PDH&DS so that the rate at which the signal is changing can be checked after a leak alarm is obtained. A rapid increase in PFAD differential pressure (less than 1 hour from normal reading to alarm) accompanied by leak alarms from other detectors in the same area would indicate a large leak (greater than 1 gpm). Conversely, a leak signal that took 10 to 100 hours or more to reach the alarm level would indicate a small (100-1000 gm/h) leak. The SIDs are calibrated so that aerosol concentration can be related to the signal level. Instruments are set to alarm at specific aerosol concentrations. The liquid metal-to-gas leak detection system is designed to function after an OBE. The radiation particulate monitoring system is designed to function after an SSE. All leak detection equipment will be tested periodically to demonstrate operability.

The increase in cell atmosphere temperature and pressure in the event leaks larger than 20 kg/min as detected by temperature and pressure sensors can provide an additional source of leak detection.

The ability to detect small leaks (100 gm/hr) by several methods in hours plus the ability to detect large leaks (>kg/min) in minutes will provide a highly reliable leak detection system that provides the operator information to enable shutdown to repair defects without extensive time for cleanup operations.

After a sodium or NaK leak has occurred, the Liquid Metal-to-Gas Leak Detection System equipment impacted by the leak will be either replaced or cleaned (pneumatic system rinsed with alcohol) to remove sodium leak residue products. The system will then be acceptance tested and calibrated in accordance with the preoperational test specification criteria utilized prior to inital plant startup.

Table 7.5-3 gives the primary and back-up methods of leak detection for the principal sodium systems and components in the plant. The methods shown in the table are related to the three sizes of leaks defined in Section 7.5.5.1.2. The principal methods of leak detection are described below.

Aerosol Monitoring

Aerosol monitoring will be performed by measuring the pressure drop across a membrane filter with a constant flow of gas sampled from the annular space between major piping and its insulation, from the space within guard vessels, and from cells containing liquid metal systems. Another cell aerosol monitoring method uses a sodium ionization detector. Liquid Metal aerosols or vapor are ionized by a hot filament and the ion current is measured. Increases in the ion current indicate a leak.

Based upon the experimental results, these methods provide for detection of leaks of 100 gm/hr and less, with a response time depending on temperature and the volume being monitored.

The major function of this instrumentation will be to provide indication of the presence of small leaks which do not present a significant contamination hazard, but which might result in undesirable long-term corrosion.

Contact Detectors (Spark-Plug)

Contact detectors consist of a stainless-steel-sheathed, mineral oxide-insulated, two-wire probe with the sensing end open and the wire ends exposed. Contact detectors are installed, for example, on bellows sealed valves with the sensing end between the bellows and the mechanical backup seal. A leak is detected by the reduction in circuit electrical resistance caused by sodium contacting the wire ends.

Cable Detectors

Cable detectors consist of stainless-steel-sheathed, mineral-oxide-insulated, cable with holes penetrating the sheath to permit leaked liquid metal to come in contact with the conductors. Cable detectors will be placed, for example, in the bottom of guard vessels and below large tanks.

Other Detection Methods

Pressure and temperature measurements available in the inerted cells (Section 9.5.1.5) will provide immediate indication of the presence of large leaks over the 20 kg/min size. In the case of systems containing radioactive sodium, the detection of airborne radioactivity arising from Na-24 or Na-22 in the aerosols will be performed by particulate radiation monitoring equipment (Section 11.4.2) which provides a sensitive detection method for aerosol concentrations as low as 10^{-15} gm/cc.

Chemical analysis provides positive detection capability for aerosol concentrations of approximately 10^{-13} gm/cc, depending on the leak integration period.

Other Backup Detection Method

Liquid Sodium Level Sensors in the reactor, the EVST, the IHTS expansion tank, and sodium storage tanks will provide indications of large leaks. Smoke detectors (Fire Protection System) will detect combustion products originating from sodium leaks in air (See Section 9.13.2).

Indication In Control Room

An audible group alarm is sounded in the control room upon indication of a leak or certain failures of contact, cable, or aerosol channels. The channel number producing the alarm and the location of the region covered by this channel are displayed on an annunciator on a local panel. This information will identify the leak as occurring in a specific major comonent or series of pipe sections, or specific bellow-sealed valve, or the cell containing the leaking system. The leak detection system uses the Plant Data Handling System for channel failure monitoring, data and trend logging; the sampling time interval will nominally be approximately 30 seconds.

No automatic isolation functions or reactor scram are initiated by the Liquid Metal-To-Gas Leak Detction System. Isolation or shutdown of a system showing a leak will be performed manually, following verification of the leak and review of the operating conditions.

7.5.5.1.2 Design Analysis

The Liquid Metal-to-Gas Leak Detection System will meet the appropriate requirements of CRBR Design Criterion 30, "Inspection and Surveillance of Reactor Coolant Boundary and Criterion 33, "Inspection and Surveillance of Reactor Coolant Boundary. Criterion 30 requires that means be provided for detecting and identifying the location of the source of reactor coolant leakage from the reactor coolant boundary to the extent necessary to assure that timely discovery and correction of leaks which could lead to accidents whose consequences could exceed the limits prescribed for protection of the health and safety of the public. Criterion 33 requires that means be provided for detecting intermediate coolant leakage from the intermediate coolant boundary. In order to demonstrate how the intent of the criteria will be satisfied, the instrumentation requirements met by this system for three different ranges of leaks are discussed. These ranges have been selected to analyze situations which cover the complete range of leak detection instruments. Section 15.6 discusses the consequences of leaks for the health and safety of the public.

Large Leaks

This category covers failures up to those resulting in a leak of 30 gpm or 100 kg/min. A significant physical characteristic of leaks of this size is that they would result in pressure and temperature changes in the primary cells if the leak occurs in PHTS pipe sections. This feature sets the lower boundary of the leak at about 20 kg/min; this being an estimte of the amount of sodium which would result in measurable changes in cell pressure and temperature. If the leak occurs in a guard vessel, continuity detectors will provide detection of these large leaks. Leaks of this magnitude would be detected in five minutes or less for the primary and intermediate heat transport system. The operator would then be able to initiate and complete plant shutdown within ten minutes after the start of the leak.

The pressure and temperature measurements available in the inerted cells will, in conjunction with the aerosol detectors, continuity detectors and radiation monitors, provide the response required for proper operator action in case of leaks of this magnitude.

Intermediate Leaks

Intermediate leaks were defined as those leaks which would not result in significant changes in cell pressures and temperatures but where the extent of the resulting contamination and plant maintenance makes plant shutdown desirable. The range of leak rates covered extends from the lower limit of the large leaks previously considered down to a leak of 100 gm/hr. The detection times for the wide range of leaks in this group would vary from a few minutes to several hours depending on the rate of leakage. Based upon experimental results, it is concluded that several systems would detect a leak of this magnitude in several hours at least and possibly in minutes.

Instrumentation capable of detecting leaks of this magnitude include radiation monitors, continuity detectors, and the different types of aerosol detectors.

Small Leaks

Small leaks at or below 100 gm/hr were defined as those events resulting in releases of sodium which do not pose a contamination or maintenance problem but might result in undesirable long-term corrosion (see Section 5.3.3). The methods for detecting leaks of this range are aerosol detectors and radiation monitors in the case of the primary system.

In the course of test programs, aerosol concentrations produced by leaks of down to 5 gm/hr were found to be within the detection capability of both a Sodium Ionization Detector and a Plugging Filter Aerosol Detector in test chambers. The test results show that leaks of this size can be detected in the range of one hour to 24 hours by annuli monitors depending upon the sodium temperature and gas environment. It is deduced from the test results that very small leak (<1 gm/hr) will be detected by annuli monitors in several days.

Tests during 1975 and 1976 showed that under environmental conditions typical of LMFBR operation, small leaks from typical piping configurations can be detected by both Sodium Ionization and Plugging Filter Aerosol Detectors. Continuity (cable or contact) detectors did not reliably detect small pipe leaks under these conditions. Testing in 1978 verified the performance of aerosol detectors using prototypic CRBRP cell atmosphere recirculation as well as pipe/insulation design.

It is deduced from the test results that the sodium vapor/aerosol systems will, in conjunction with existing radiation monitoring technology, provide adequate indication of the smallest sizes of leaks of interest.

Sodium Leaks into an Air Atmosphere

Test results indicate that the methods applicable to sodium leaks in inerted cells will also operate when applied in an air atmosphere. The additional use of smoke detectors and the accessibility of piping located in an air atmosphere to visual inspection assist in the selection of an effective sodium-to-air leak detection system.

7.5.5.2 Intermediate to Primary Heat Transport System Leak Detection

7.5.5.2.1 Design Description

The IHTS pressure is maintained at least 10 psi higher than the Primary Heat Transport System at the IHX to prevent radioactive primary sodium from entering the IHTS in the event of a tube leak. Maintaining a positive pressure differential across the IHX is a limiting condition for operation of the plant (Chapter 16 - Technical Specifications). This provides assurance that a zero or negative differential will not exist during any extended interval. A loss of this pressure or a reversal of it is not expected to occur except during accident conditions. Such an occurance would necessitate an orderly plant shutdown to correct the problem. Since a reverse differential cannot occur for a significant interval. the potential leakage of primary sodium into the intermediate system, through an IHX tube leak, is small.

Leakage of primary sodium into the IHTS, should it occur, will be detected by radiation monitors provided on the IHTS piping within the SGB. The radiation monitor system will provide an indication of the radiation level and will provide alarms for conditions of excessive radiation indicative of ingress of primary sodium. Since the only activity expected in the IHTS is a low level of tritium, the radiation monitors will be very sensitive to the presence of significant amounts of radioactive primary sodium in the intermediate system. For accidents which involve a loss of IHTS boundary integrity the radiological effects have been evaluated. The results of these evaluations are presented in Sections 15.3.2.3, 15.3.3.3 and 16.6.1.5.

Maintaining a positive pressure differential across the IHX assures that the leakage across the IHX tube barrier will result in an inflow of sodium into the primary system causing a loss of sodium inventory in the IHTS. The sodium inventory in the IHTS is monitored by tracking the sodium levels and correcting for loop temperature effects. Alarms are provided in the control room to alert the operator upon detection of a large loss of IHTS sodium inventory.

7.5.5.2.2 Design Analysis

Intermediate to Primary Heat Transport System leak detection is provided to comply with CRBRP General Design Criterion 36 "Inspection and Surveillance of Intermediate Coolant Boundary". In order to demonstrate now the intent of this criterion will be satisfied, an analysis of the minimum detectable leaks in the IHX is provided below.

The minimum detectable level change of sodium in the IHTS pump and expansion tank is approximately 3 inches which corresponds to about 150 gallons. In the event of a full-circumferential break of an IHX tube, the leak rate of intermediate sodium to the primary side of the IHX would be approximately 150 gpm. At this leak rate, the detection time would be about one minute assuming steady state temperature conditions.

Based upon a 3-inch level change, leakage of as low as 6.25 gph would fall within the detection threshold. Over long time periods, the sensitivity of the detection system will be reduced by an insignificant amount due to other potential leakages from the system. If leakage occurs due to piping or component leaks, the external leak detection system will detect the leakage. A second potential source is leakage through the four sets of dump valves which has a maximum expected rate of one to two gallons per day. Since this leakage rate is essentially two orders of magnitude smaller than the leakage threshold, it will not have a consequential effect on the detection sensitivity.

7.5.5.3 Steam Generator Leak Detection System

A Steam Generator Leak Detection System is provided to detect small (as low as 10^{-5} lb/sec) water-to-sodium and steam-to-sodium leaks in the steam generator modules, to identify the module in which the leak has occured, and to alert the control room operator enabling him to take manual corrective action to prevent the leak rate from increasing. Leak detection instrumentation is provided for:

1. Sodium exiting from the superheater and the evaporators.

2. Sodium filled vent lines from the evaporator vents and the superheater vent.

7.5.5.3.1 Design Description

General

Steam or water leakage into sodium increases the hydrogen and oxygen concentrate in the sodium stream. The leak detection is based on measurement of the hydrogen and oxygen concentration in the sodium.
47 Oxygen concentration measurements in the sodium are complementary to the hydrogen detectors, thus providing a diverse method to ensure early detection.

To provide a sensitive leak detection capability, the background concentrations of oxygen and hydrogen are maintained as low as practicable. The hydrogen in the sodium is removed by cold trap precipitation of sodium hydride. Oxygen in the sodium is removed in the same way by sodium oxide precipitation. The background concentration of hydrogen in the system is
33 31 13 50 normally maintained below 200 ppb through cold trap operation. Oxygen concentration is maintained at about 2 ppm.

Hydrogen Detectors

The measurement of hydrogen in the sodium is performed by allowing the hydrogen to diffuse through a thin walled nickel membrane, and detecting the hydrogen with an ion gauge and an ion pump. The monitor may operate in a static mode using the ion gauge to monitor steady state hydrogen concentration, since hydrogen content is directly related to the pressure measured in the chamber. The monitor may also operate in a dynamic mode, using the ion pump to constantly pump the chamber since then the hydrogen concentration is directly related to the ion-pump current.

47 34

Oxygen Detectors

Oxygen electro chemical cells are used to continuously monitor in-sodium concentration and consist of a reference oxygen electrode separated from the sodium by a solid electrolyte. The electrical potential drop between the reference electrode and the sodium measures the in-sodium oxygen concentration.

Detectors Location

The three steam generation loops utilize identical oxygen-hydrogen (O-H) detector modules at the following locations:

(a) One-leak detection module monitors evaporator outlets. This leak detection module under normal operation samples sodium that is a combination of sodium exiting evaporator A and evaporator B. However, by the use of valves in the sample lines, the leak detection module may be utilized to sample sodium exiting either evaporator A or evaporator B.

(b) Superheater vent line. This leak detection module under all conditions samples only sodium in the superheater vent line.

(c) One-leak detection module monitors evaporator vent lines. This leak detection module under normal operation samples sodium that is a combination of sodium in the evaporator A vent line and evaporator B vent line. However, by the use of valves in the sample lines, the leak detection module may be utilized to sample sodium in either evaporator A vent line or evaporator B vent line.

(d) One-leak detection module monitors superheater outlet pipes. This leak detection module samples sodium that is a combination of sodium exiting both the superheater outlets.

Figure 5.1-4 shows the instrumentation location.

Indication in Control Room

Measurements from the hydrogen and oxygen detectors are monitored by the Data Handling and Display System. Each channel is limit checked and its trend is limit checked. Low, intermediate and high alarms, and low, intermediate and high confirmed leak alarms, and channel failure alarms are also provided to the Plant Annunciator System.

System Operation

The leak detector detects two leak signal categories:

1. a strong signal in a single pass,

47 2. detection of a gradual concentration increase or decrease through several passes through the sodium.

Figure 7.5-4 illustrates typical first pass hydrogen concentration change as a function of water leak rate. As illustrated, a change in hydrogen concentration of a few ppb would be indicated at the detector for leak rates in the range of 10^{-4} lb/sec. Approximately one minute is required for the hydrogen to reach the detector and signal a leak. Detection capability can be extended to smaller leak sizes through the use of a rate of rise detection system. Several passes of sodium through the system would be required to allow the hydrogen concentration to build up. The sensitivity of this system will allow detection of leaks in the range of 10^{-5} lb/sec. Similarly, Figure 7.5-4A indicates the first pass oxygen
48 | 47 concentration as a function of water leak rate. Figure 7.5-5 illustrates the hydrogen concentration change with time for various sizes of leaks.

7.5.5.3.2 Design Analysis

A Steam Generator Leak Detection System is provided to comply with CRBRP General Design Criteria 4 which calls for provision of leak detection in the Steam Generators. In order to show how the criterion will be satisfied, a review of leak damage studies is presented with the
13 resulting instrumentation requirements.

Leak Damage Studies

Experimental studies have been conducted in the United States and in Europe over the past ten years which have given a broad base background to the understanding of the behavior of leakage damage effect. Most of the experimental data taken with 2-1/4 Cr- 1 Mo material have been obtained by injecting water or steam through hole type geometrics at selected target configurations (jet leaks). In general, results of these studies have indicated that both adjacent tube wastage and self wastage
45 13 are possible damage mechanisms, as described in Reference 1, 3 and 4.

Adjacent tube wastage will occur with the proper leak size and orientation. For very specific conditions and geometries, some experiments have been performed where adjacent tube wastage occurred very rapidly in a localized area. Relating these specific conditions to CRBRP, adjacent tube wastage could occur which would result in tube failure in a very short period of time, less than one minute. However, it is not likely that leaks would be optimized as to leak geometry, location and orientation, as those utilized for the experiments. In the event that this did occur, the steam generator rupture discs provide necessary protection.

13

The second class of damage is self-wastage around the leak site. Some experiments have noted that some very small leaks have experienced a sudden enlargement after a period of relatively steady operation as reported in Reference 1. The effect of this type of characteristic has
13 been studied by the GE/ANL Steam Generator Systems Development Program and
46 is reported in References 3 and 4.

Design Requirements

The design requirements for the Steam Generator Leak Detectors
13 have been selected as described below.

SGS Leak Detection Requirements

47

50		Hydrogen Detectors	Oxygen Detectors	
	Sensitivity	6 ppb	24 ppb	
	Range	0.04-2 ppm	0.1-10 ppm	23
	Response Time	$\leq$30 sec.	$\leq$30 sec.	

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Instrument Sensitivity

- The wastage rate studies for jet leaks show that leaks below 10^{-4} lb/sec persist without major damage for more than one loop transit time. The loop transit time can be calculated
47 from a 13.49 x 10^6 lbs/hr flow rate and 4 x 10^5 lbs sodium inventory in the IHTS loop; the hydrogen generated from the quantity of H_2O leaked in one transit time divided by the total sodium inventory yields an increase of 6.3 ppb in the
50 47 concentration of hydrogen, thus a 6 ppb sensitivity for the hydrogen detectors.

- A resolution of 3 ppb change in the hydrogen background con-
50 centration ranging from 60 - 200 ppb (i.e., a change of 3-4%) under steady-state SG operation is a design goal for the leak detector.

- The oxygen detector is as sensitive as the hydrogen detector. Taking into account an oxygen background concentration of
47 33 1ppm (with 2ppm maximum), the sensitivity is 24 ppb.

13

Instrument Range

- Detection capability of leaks up to 10^{-1} lb/sec.

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Instrument Availability

- Sodium loop leak detection capability provides continuous monitoring and indication of the impurity level whenever sodium and water/steam co-exist in the steam generator modules.

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Operation Requirements

- o In order to effect an orderly plant shutdown which minimize plant unavailability, the following operator actions are required.

Alarm	Leak Size (lb/sec)	Operator Action
Low	Less than 2×10^{-5}	1. Monitor Leak Data 2. Initiate Leak Location Procedure
Intermediate	2×10^{-5} to 6.5×10^{-3}	1. Monitor Leak Data 2. Initiate/Continue Leak Location Procedure
High	$>6.5 \times 10^{-3}$	1. Monitor Leak Data 2. Prepare to Initiate Shutdown or Scram 3. Initiate/Continue Leak Location Procedure
Confirmed Leak		
Low	Less than 2×10^{-5}	1. Monitor Leak Data 2. Reduce to 40% Power 3. Continue Leak Location Procedure
Intermediate	2×10^{-5} to 6.5×10^{-3}	1. Initiate Shutdown 2. Continue Leak Location Procedure 3. Blowdown Affected Module*
High	$>6.5 \times 10^{-3}$	1. Initiate Reactor Scram 2. Initiate Loop Blowdown of all 3 Modules

*If affected module cannot be identical, blowdown all 3 modules in the affected loop.

For leakages greater than about 0.1 lb/sec of water, the pressure buildup in the system will occur rapidly, causing the Sodium-Water Reaction Pressure Relief System to be activated (See Section 7.5.6).

7.5.6 Sodium-Water Reaction Pressure Relief System (SWRPRS) Instrumentation and Controls

7.5.6.1 Design Description

7.5.6.1.1 Function

The Sodium-Water reaction Pressure Relief System (SWRPRS) Instrumentation and Control System detects the inception of a large or intermediate water to sodium leak in any of the steam generator modules (see Section 5.5.2.6).

SWRPRS instrumentation and control which is used to initiate and control the SWRPRS function is 1E. Instrumentation which is used to monitor the status of SWRPRS serves no safety function and accordingly is non-1E.

For a large leak, three 1E pressure sensors (nine per loop) are provided immediately downstream from each pair of rupture disks in the superheater and evaporator's (two) reaction products vent line. The signals are transmitted to the PPS Secondary Shutdown System which initiates a reactor trip and PHTS and IHTS sodium pump trip. Buffered signals are transmitted to the SWRPRS trip logic which isolates the affected loop. A group alarm is transmitted to the Plant Annunciation System (PAS).

For intermediate leaks, three pressure sensors are provided in the IHTS sodium expansion tank equalization line to the sodium dump tank, downstream of the rupture disks. These signals are transmitted directly to the SWRPRS trip logic via a two-out-of-three coincidence logic which isolates the affected loop. Reactor trip and trip of the PHTS and IHTS sodium pumps is initiated via the PPS Primary Shutdown System as a result of a high steam-to-feedwater flow in the affected loop.

7.5.6.1.2 SWRPRS Trip Logic

There are three separate SWRPRS trip logics, one each loop. Thus, only the affected loop will be isolated leaving the other two loops for shutdown heat removal. The SWRPRS trip logic (Figure 7.5-6) and the remainder of this discussion addresses one loop only.

In parallel with sending signals to the PPS for reactor and sodium pump trip for large leaks, the SWRPRS instrumentation send buffered signals to the SWRPRS trip logic. The trip circuit develops a two-out-of-three coincidence logic from each steam generator module (one superheater and two evaporators). Each module is combined in a one-out-of-three coincidence logic which in turn is then combined in a one-out-of-two coincidence logic.

Upon receiving a signal from the large leak detection circuit, the intermediate leak detection circuit or a manual trip from the control room the following simultaneous actions occur in the faulted loop.

a. The IHTS sodium pump pony motor of the affected loop is tripped by deenergizing the contactor coil (large leak detection circuit only).

b. The SGS recirculating pump motor is tripped off the line by energizing the switchgear's tripping circuit.

c. The water/steam side of each evaporator and superheater is individually isolated by closure of their respective isolation valves. The main feedwater, auxiliary feedwater, and steam drum inlet and drain isolation valves are closed.

d. Water is removed from the evaporators by opening the valves between the evaporator inlet and the water dump tank. Power relief valves on the outlet line of each evaporator and the superheater are opened to provide a steam vent to the atmosphere.

e. Water dump and steam vent action is terminated by closure of all steam power relief and water dump valves when the units have been de-pressurized to 250 psig.

f. The water-steam side is then inerted by opening of the nitrogen purge valves which provide nitrogen to both units in the affected loop. A regulator on the nitrogen supply maintains the pressure at 200 psig. In the event of continued pressure buildup, the steam vent power relief valves will open at 300 psig and provide for another depressurization to 250 psig.

g. SWRPRS piping is purged by nitrogen following bursting of SWRPRS main rupture disks.

All isolation, dump, power relief, and purge valves are provided with controls and status indication in the Main Control Room to provide manual control at the plant operator's discretion. Alarms are provided in the PAS for the SGS isolation, dump and pressure relief valves to warm the operator of inadvertent
59 off-normal operation.

7.5.6.1.3 Bypasses and Interlocks

The control logic for the actuation of the Sodium-Water Reaction Pressure Relief System will be designed to insure reliability and freedom from spurious operation. A discussion of the bypasses and interlock functions and their safety classifications will be provided in the FSAR. Bypasses and interlock functions which are required for operation of the initiation and control logic will be Class 1E.

7.5.6.1.4 Sodium Dump

No automatic action is associated with the removal of sodium from the affected loop. However, sodium dump valves are provided for draining of sodium to the sodium dump tank, and can be initiated by operator action.

Drain valves are located in five piping runs between the IHTS sodium loop and the sodium dump tank. Each piping run contains a pair of drain valves arranged in series. Controls and indications for all these valves are located on the Main Control Board.

7.5.6.1.5 Monitoring Instrumentation

In addition to the instrumentation required for the initiating circuitry, the following parameters are measured to aid the plant operator in assessing the performance of the Sodium-Water Pressure Relief System:

- Pressure in the gas space between each pair of rupture disks is monitored to detect leakage, or failure of the sodium side rupture disk. Spark plug leak detectors are also provided in the gas space to detect rupture disk failure.

- Thermocouple elements are provided for monitoring surface temperatures of the reaction products separator tank, centrifugal separator, centrifugal separator drain tank, and the hydrogen igniter.

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57 • Pressure is monitored in the Reaction Products Separator Tank to alert the operator to off-normal conditions in the Reaction Products Vent System.

57 7.5.6.1.6 Sodium Dump Tank Instrumentation

A sodium dump tank is provided in each loop for drainage of sodium from the IHTS. Dump tank level is measured by inductive type level probes. Each tank is provided with two probes to meet the necessary range requirements. Each probe is connected to an excitation-conditioning module that provides local indication of sodium levels.

The conditioned signal from the wide-range probe is supplied to a trip unit, which provides a high-level alarm signal to the PAS. All level measurement channels provide inputs to the PDH&DS.

Surface thermocouples, which are part of the sodium dump tank trace heating system, are available for display as an indication of level and temperature. A pressure switch actuates an alarm and a rupture disc bursts to vent to atmosphere in case of excessive pressure buildup in the sodium dump tank.

57 7.5.6.1.7 Water Dump Tank Instrumentation

A water dump tank is provided in each loop to accept and store the water from the evaporators when rapid depressurization is required. Measured parameters for the water dump tank are level, pressure, and temperature.

Dump tank level is measured by a differential pressure transmitter that senses the difference in pressure caused by the variable height of water in the dump tank. The differential pressure signal is supplied to a local indicator, the PDH&DS, and a high-level alarm circuit that provides a signal to the PAS.

Pressure is measured on an appendage off the dump tank by a conventional pressure sensor-transmitter. The pressure signal is used for local indication, for the PDH&DS, and for high-pressure alarm to the PAS.

Dump tank temperature is monitored by chromel-alumel thermocouples attached to the surface of the tank. The signals are supplied to the multipoint temperature indicator.

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7.5.6.2 Design Analysis

Because of the large increase in pressure from the formation of reaction products during a large sodium-water reaction, rupture disc operation is necessary to prevent excessive pressure surges in the Intermediate Heat Transport System and possible primary boundary rupture at the Intermediate Heat Exchanger. Reaction products vent line sensors are part of the Reactor Shutdown System and as such meet the requirements of the Plant Protection System (see Section 7.1.2 and 7.2.2). The initiation of isolation and dump of the water side of the steam generators, trip of the recirculation pumps and inerting of the steam generators normally follows after rupture disc operation in a large sodium-water reaction and is desirable from the operational standpoint of minimizing the time to recover from the incident. However, the initiation of these actions after rupture disc operation is not necessary from a safety standpoint to assure protection of the core or the safety of the public.

All SWRPRS equipment associated with isolation or dump of evaporator or superheater modules is designed to assure that no credible single event can disable more than one of the three redundant decay heat removal paths. All electronic equipment is designed to withstand the Safe Shutdown Earthquake. The mechanical equipment associated with the 3 steam generators is physically separated in 3 different steam generators bays. Electrical and pneumatic supplies are arranged such that a single failure disables at most one decay heat removal path. Where practicable, the preferred failure position for equipment is in a direction to assure the safe operation of the SGAHRS.

SWRPRS equipment whose failure could cause loss of decay heat removal capability of the SGAHRS is safety related. This includes SWRPS initiation and control equipment which is used to initiate and blowdown the water side of the affected loop. Any credible single failure in the SWRPRS can lead to the failure of at most one of the three decay heat removal loops. Since the three decay heat removal loops are redundant and independent, the SGAHRS will meet the single failure criterion and the adequacy of the decay heat removal system following a credible single failure in the SWRPRS is assured.

7.5.7 Containment Hydrogen Monitoring

The objective of Containment Hydrogen Monitoring is to provide indication in the Control Room of the hydrogen concentration in the upper levels of containment.

25 7.5.7.1 Design Description

The hydrogen instrumentation consists of two fully redundant and independent analyzer channels. The containment atmosphere is sampled through an entry filter located near the top of the RCB. The air samples are pumped to the analyzer which is located in the SGB and operates on the principle of thermal conductivity. From there signals go to the Control Room where the hydrogen concentration readout is provided. This instrument is also required to perform functions for events which lie beyond the design basis for the plant. This instrument is further discussed in
57 this capacity in Sections 2.1 and 3.3 of Reference 10b of PSAR Section 1.6.

7.5.8 Containment Vessel Temperature Monitoring

The objective of Containment Vessel Temperature Monitoring is to provide indication in the Control Room of the containment vessel
25 temperature.

7.5.8.1 Design Description

The temperature instrumentation consists of two fully redundant and independent channels. Each channel consists of eight thermocouples mounted at various locations on the inside of the containment wall, with each thermocouple providing a signal to conditioning instrumentation in the SGB. The instrumentation sends a signal to the Control Room where individual readout is provided. This instrument is also required to perform functions for events which lie beyond the design basis for the Plant. This instrument is further discussed in this capacity in Sections 2.1 and 2.2 of
57 Reference 10b of PSAR Section 1.6.

7.5.9 Containment Pressure Monitoring

The objective of the Containment Pressure Monitoring System is to provide indication in the Control Room of the pressure inside the containment above the operating floor.

7.5.9.1 Design Description

The pressure instrumentation consists of a pressure detector inside the containment vessel. Signals will be provided to the display
25 and alarm panel in the Control Room so that continuous readout will be
provided to the plant operator. This instrument is also required to perform functions for events which lie beyond the design basis for the plant. This instrument is further discussed in this capacity in Sections
57 2.1 and 2.2 of Reference 10b of PSAR Section 1.6.

7.5.10 Containment Atmosphere Temperature

The objective of the Containment Atmosphere Temperature Monitoring System is to provide indication in the Control Room of the atmosphere temperature inside the containment building.

7.5.10.1 Design Description

The temperature instrumentation consists of two fully redundant and independent channels. Each channel consists of two thermocouples mounted on the RCB dome, with each thermocouple providing a signal to conditioning instrumentation in the SGB. The instrumentation sends a signal to the Control Room where individual readout is provided. This instrument is also required to perform functions for events which lie beyond the design basis for the plant. This instrument is further discussed in this capacity in Section 2.1 and 2.2 of Reference 10b of PSAR Section 1.6. Note: See Section 7.5.15.

7.5.11 Accident Monitoring Instrumentation

The Accident Monitoring Instrumentation is an integrated set of instruments made available to assess plant and environs conditions during and following accidents. A discussion of the functional requirements and general design requirements is provided below. Additional preliminary description of the application of Reg. Guide 1.97 is provided in Question Response CS760.06.

7.5.11.1 Description

Accident Monitoring parameters are monitored to perform the following functions:

- Provide primary information to permit manual actuation of safety systems. — Variable Type A

  Type A variables monitor the primary information required to permit the control room operator to take specific manually controlled actions for which no automatic control is provided and that are required for safety systems to accomplish their safety functions for Design Basis Accident events. Primary information is that which is essential for the direct accomplishment of the specified safety functions; it does not include those variables that are associated with contingency actions that may also be identified in written procedures.

- Indicate that safety functions are being accomplished (i.e., reactor shutdown, core cooling, containment integrity). — Variable Type B

  Type B variables provide information necessary to indicate whether plant safety functions are being accomplished.

- o Indicate potential for or breach of barriers to fission products release (i.e., fuel, cladding, primary boundary, containment). — Variable Type C

  Type C variables provide information to indicate the potential for, and/or the actual breach of barriers to fission product releases. The barriers are (1) fuel cladding, (2) primary coolant pressure boundary, and (3) containment.

- o Indicate operation of safety systems or other systems important to safety. — Variable Type D

  Type D variables provide information to indicate the operation of individual safety systems and other system important to safety. These variables are to help the operator make appropriate decisions in using the individual system important to safety in mitigating the consequences of an accident.

- o Indicate magnitude of release of radioactive materials and continuously access such release. — Variable Type E

  Type E variables provide information required for use in determining the magnitude of release of radioactive materials, and continually assessing such releases.

7.5.11.2 Instrumentation Design and Qualification

A graded approach to instrument requirements has been incorporated which emphasizes the importance to safety of a particular measured variable. Differenty categories for instrumentation have been identified as follows:

Category 1: Class 1E instrumentation which requires seismic and environmental qualification, single failure criteria, and Class 1E power source.

Category 2: Instrumentation which is environmentally qualified and powered from a reliable power source.

Category 3: Instrumentation of a high quality commercial grade.

7.5.11.2.1 Category 1

- o Each Category 1 parameter is monitored by at least two instruments. These instruments are referred to as:

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- o Principal Instruments, and
- o Redundant Backup Instruments

All Category 1 Principal Instruments are classified as 1E. Category 1 Redundant Backup Instruments are classified as 1E up to the isolation device from the sensor.

A third verification instrument is provided if a failure of the principal or redundant instrument will result in information ambiguity (that is the redundant displays disagree) that could lead operators to defeat or fail to accomplish a required safety function. This third instrument is called a verification instrument.

o The Principal Instrument indication will be located in the viewing area of the operator in the Control Room. The Redundant Backup Instrument indicator will be in the proximity of the Principal Instrument

Indicator to permit the operator to make comparisons. If a Verification Instrument is required, its indication will be accessible but not necessarily in the Control Room. (The plant computer may provide the Verification Instrument information).

o A minimum of one instrumentation channel for each Category 1 variable will be recorded unless it can be shown that recording that particular parameter will not provide benefit in analyzing the overall accident. The plant computer (non-1E) is the preferred method of recording. Special attention will be given to the logging frequency of each of these Category 1 parameters so that an adequate presentation of the parameter response during an event will be available.

A recorded pre-event history for these parameters is required for a minimum of one hour, and continuous recording of these instruments is required following an accident until such time as continuous recording of such information is no longer deemed necessary.

o The single failure criteria for Category 1 instruments is applied to the combination of the Principal Instrument and the Redundant Backup Instrument. The Verification Instrument is not taken into consideration when considering single failure criteria. No single failure within the Principal Instrument chain, and the Redundant Backup Instrument chain, their auxiliary supporting features, or their power sources, concurrent with the failures that are a condition of, or a result of a specific accident, will prevent the operator from being presented the required information.

- The Principal Instruments from sensor to indicator, and the Redundant Backup Instruments from sensor through the indication device will be qualified in accordance with PSAR Section 1.6 Reference 13, "Requirements for Environmental Qualification of Class 1E Equipment." They are qualified to provide the information needed by the operator to assess plant and environs conditions during and following design basis events.

- Instrumentation will continue to read within the required accuracy following, but not necessarily during, a Safe Shutdown Earthquake (SSE).

- The Principal Instrument (from sensor to indicator) and Redundant Backup Instrument (from sensor through the isolation devices) will be energized from Class 1E power and be supplied with battery backing where momentary interruption of the indication is not tolerable.

7.5.11.2.2 <u>Category 2</u>

- Each Category 2 instrument signal, will be, as a minimum, processed for display on demand.

- The Category 2 instrument indicators will be located to effectively support normal and emergency plant operations.

- The Category 2 instruments from sensor to indicator will as a minimum be qualified in accordance with Reference 13, PSAR Section 1.6, "Requirements for Environmental Qualification of Class 1E Equipment" except for seismic. They will be qualified to provide the information needed by the operator to assess plant and environs conditions during and following design basis events.

- The instrumentation will be energized from a highly reliable power source (not necessarily a Class 1E power supply). Where interruption of the power supply is acceptable station AC power may be used. Where momentary interruption is not tolerable, the non-1E UPS is used.

7.5.11.2.3 Category 3

- Each Category 3 instrument signal, will be, as a minimum, processed for display on demand.

- The location of the Category 3 instrument indication will be chosen to support normal and off-normal operations.

- The Category 3 instrumentation will be a high quality commercial grade.

7.5.11.2.4 General Requirements to Category 1, 2, and 3

- Servicing, testing, and calibration programs will be specified to maintain the capability of the monitoring instrumentation. For those instruments where the required interval between testing shall be less than the normal time interval between generating station shutdowns, a capability for testing during power operation shall be provided.

- Whenever means for removing channels from service are included in the plant design, the plant design will facilitate administrative control of the access to such removal means.

- The plant design will facilitate administrative control of the access to all setpoint adjustments, module calibration adjustments, and test points.

- The monitoring instrumentation design will minimize the development of conditions that would cause meters, annunciators, recorders, alarms, etc., to give anomalous indications potentially confusing to the operator.

- The instrumentation will be designed to facilitate the recognition, location, replacement, repair, or adjustment of malfunctioning components or modules.

- To the extent practicable, monitoring instrumentation inputs will be from sensors that directly measure the desired variables. An indirect measurement will be made only when it can be shown by analysis to provide unambiguous information.

7.5.11.3 Instrument Identification

All Category 2 instruments that are Types A, B, or C, and all Category 1 Principal and Redundant Backup instruments, will be specifically identified on their respective control panels so the operator can easily discern that they are for use under accident conditions. The above instruments will be yellow color coded (Federal Standard 595a, Chip Number 33793).

7.5.12 Inoperable Status Monitoring System

7.5.12.1 Design Description

The Inoperable Status Monitoring System (ISMS) provides for an automatic indication at the system level of the bypassed or deliberately induced inoperability of selected safety systems. The indication of inoperative and bypassed status will be provided in conformance with Regulatory Guide 1.47, "Bypassed and Inoperative Status Indication for Nuclear Power Plant Safety Systems". The Reactor Shutdown System (RSS) includes indication of bypassed status and is not part of ISMS. See Section 7.2 for the RSS details.

Specific capabilities of the ISMS include:

- o Monitoring of safety system variables and alerting the unit operator visually and audibly of a bypassed or deliberately induced inoperable safety system.

- o A manual capability to activate the safety system indicators.

No direct safety or protection function is performed by the ISMS. The list of status information and monitoring provided by ISMS is provided in Table 7.5-4. The data acquisition and processing components of the ISMS are located in the control room area behind the Main Control Panel (MCP). The unit operator interface controls and indicators are located so they can be seen from the MCP. The ISMS data acquisition system obtains the sensor status information processes the sensor data, and provides the status indication to the unit operator. The processed data is transmitted to the Plant Data Handling & Display System (PDH&DS) for presentation via cathode ray tube (CRT) displays in support of ISMS.

7.5.12.2 Design Analysis

The ISMS is completely designed to conform to the requirements of Regulatory Guide 1.47, "Bypassed and Indication for Nuclear Power Plant Safety Systems". The ISMS in combination with the bypass status indication portions of the Reactor Shutdown System provides the complete system level coverage of safety system indication and manual activation of indicators required by Regulatory Guide 1.47.

The safety systems provide the isolation devices for associated safety-related equipment so as to preclude any action of the ISMS from preventing the performance of a safety function.

7.5.13 LOOSE PARTS MONITORING SYSTEM

7.5.13.1 Design Description

The Loose Parts Monitoring System (LPMS) provides for the detection of loose parts occuring in components of the Reactor Enclosure, Primary heat Transfer, Intermediate Heat Transfer, and Steam Generator System heat transfer flow paths. No direct safety or protection function is performed by the LMPS. Implementation of a LPMS will conform to Regulatory Guide 1.133 except where differences between LWR and LMFBR technology require different methods. Specific design criteria of the LPMS will include:

- o Sensor locations will include as a minimum, Rv, PHTS Pump, IHX, IHTS Pump, SG Modules, Rv Head and/or Upper Plenum.

- o Sensors shall be located to detect loose parts of natural collection points for each of the above components.

- o Sensors shall be proven state of the art, consistent with LWR technology, modified as necessary for the CRBRP environment and shall be redundant.

- o Methods of mounting of sensors shall be either by direct mounting to components/piping or by attachment to suitable standoffs.

- o Sensitivity (threshold energy) shall be adequate to identify all loose parts that could potentially result in degradation of the above components by impacting.

- o A base line signature will be established for the LPMS and will be monitored either on a continuing or periodic basis.

- o Suitable audible indications/monitoring of the presence of loose parts shall be provided in the control room, and at other plant locations as appropriate.

7.5.14 Safety Parameter Display System

The primary function of the Safety parameter Display System (SPDS) will be to aid the operator in the rapid detection of abnormal operating conditions by providing a continuous indication of plant parameters or derived variables representative of the safety status of the plant. The system will have the capability of displaying the full range of important plant parameters and data trends on demand. The system will also indicate when plant parameters are approaching or exceeding process limits.

7.5.14.1 Description

The safety parameter displays will be provided by the state of the art Plant Computer described in Section 7.8. The backup to the SPDS is information provided by the Category I, accident monitoring instruments discussed in Section 7.5.11.

The safety parameter displays will be available in the control room as well as in the Technical Support Center (TSC). The displays will be readily accessible and visible to control room operators in the normal operating area but will not interfere with normal movement or with full visual access to other control room systems or displays. The safety parameter displays will be readable from the control room senior reactor operator's operating station.

The set of safety parameters to be displayed will be determined by ongoing design efforts. The displayed parameters will be selected consistent with Regulatory Guide 1.97, Rev. 2, as applied to CRBRP and discussed in Section 7.5.11, to enable the operator to determine the safety status of the plant process.

The display formats will be designed in accordance with human factors principles. Data trends will be available on demand, and the system will be capable of indicating when process limits are being approached or exceeded, for the selected set of parameters.

The safety parameter displays used in the control room will be designed to an operational unavailability goal of 0.1. The term unavailability is used to express a complete loss of the primary safety parameter displays.

7.5.14.2 Design Analysis

The SPDS will provide information to the Control Room and Technical Support Center consistent with the guidance provided in NUREG-0696.

7.5.15 Airlock Door Seal Area Leak Rate Monitoring

The objective of the Airlock Door Seal Area Leak Rate Monitoring System is to verify the seal integrity of the airlock doors and thus prevent containment leakage through the airlocks.

7.5.15.1 Design Description

The Airlock Door Seal Area Leak Rate Monitoring Instrumentation consists of a pneumatic unit to pressurize the airlock door seal area and a control unit that will sense the pressure and air flow into the seal area. The control unit then evaluates these signals and provides an alarm in the control room if the door seal area leak rate exceeds the allowable leak rate. One set of Leak Rate Monitoring Instrumentation will test both doors of an airlock.

References to Section 7.5

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3. Greene, D. A., J. A. Gudahl and J. C. Hunsicker, "Experimental Investigation of Steam Generator Materials by Sodium-Water Reactions, Volume 1, GEAP-14094, January 1976.

4. Gudahl, J. A. and P. M. Magee, "Microleak Wastage Test Results", GEFR-00352, March 1978.

5. Matlin, E., Witherspoon, J. E., Johnson, J. L., "Liquid Metal-to-Gas Leak Detection Instruments".

TABLE 7.5-1

INSTRUMENTATION SYSTEM FUNCTIONS AND SUMMARY

System	Measured Parameters	Instrument	Measurement Location	Purpose
Flux Monitoring	Source Range	BF_3	Thimbles on periphery of guard vessel	Determines or Provides: 1. Flux status at shutdown, startup and power levels 2. Signals to PPS logic (except source range) 3. Signals for reactor and plant control (D.C. linear power ranges) 4. Signals for display, accident monitoring, annunciation and recording
	Wide Range*	Fission Chambers	Thimbles on periphery of guard vessel	
	Power Range*	B-10, Compensated Ion Chamber	Thimbles on periphery of guard vessel	
Heat Transport Primary/ Intermediate Loops	Reactor Inlet Pressure*	Pressure Element	Cold leg primary loop	PPS and display PHTS performances
	Primary and Intermediate Flow*	PM Flowmeter	Cold leg of primary and intermediate loops (hot leg in intermediate loop 2)	PPS, Plant Control and Display, PHTS performance
	IHX Primary Outlet Temperature*	Thermocouple	Cold leg piping nearest to IHX primary outlet	Plant Control System (PCS), PPS, and Display
	Primary and Intermediate Hot and Cold Leg Temperature	Resistance Temperature (RTD)	Primary and intermediate hot and cold leg	Surveillance, display and use to calorimetrically calibrate PM flowmeters
	Primary and Intermediate Pump Discharge Pressure	Pressure Elements	Drainline from discharge piping of the loop's sodium pump	Surveillance, display and monitor differential pressure between primary & intermediate loops PHTS performance
	Intermediate IHX Outlet Pressure	Pressure Elements	Intermediate between IHX & Superheater	Surveillance, display & monitor differential pressure between intermediate loops

TABLE 7.5-1 (Continued)

System	Measured Parameters	Instrument	Measurement Location	Purpose
Heat Transport Primary/ Intermediate Loops (cont'd)	Intermediate Pump Inlet Pressure	Pressure Elements	Pipes between evaporator and pump inlet	Display pump performance
	Intermediate Expansion Tank Sodium Level	Level Probe	Intermediate expansion tank	Display-intermediate loop sodium inventory and alarm
	Evaporator Sodium Outlet Temperature*	Thermocouple	Downstream where the two evaporator outlets join into the header	PPS and display
Sodium Pumps	Sodium Level	Level Probe	Pump Tank	Display-used for sodium inventory and pump protection (alarm)
	Primary/Intermediate Pump Speed*	Tachometer	Main Shaft of each pump	PPS, display, pump speed control, performance
	Pony Motor Running	Speed Switch	Pumps	Display, performance
	Diagnostic Instrumentation	Various	Pumps	Display, pump performance
Steam Generator	Sodium Flow	Venturi	Superheater sodium outlet (1 loop)	Display & superheater & evaporator performance
	Sodium Temperature	Thermocouple	Superheater evaporator outlet (3 loops)	Display & steam generator performance evaluation
	Sodium Pressure	Pressure Element	1 loop-superheater inlet, outlet (both legs) and evaporator outlet (one leg)	Display & steam generator performance evaluation
	Feedwater Flow*	Venturi	Inlet line to steam drum (feedwater)	PPS, display & steam generator performance evaluation
	Superheat Steam Flow*	Venturi	Outlet of each superheater (steam)	PPS, display & steam generator performance evaluation
	Steam Drum Drain Flow	Orifice	Steam Drum Drain line for each steamdrum	Performance evaluation
	Evaporator Inlet Flow	Venturi	Inlet to one evaporator (1 loop)	Performance evaluation

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TABLE 7.5-1 (Continued)

System	Measured Parameters	Instrument	Measured Location	Purpose
Steam Generator (cont'd)	Feedwater Temp.*	RTD	Steam drum inlet (feedwater)	PPS, display & steam generator performance evaluation
	Recirc. Water Temp.	Thermocouple	Recirculation pump discharge header	Performance evaluation
	Steam Temp.	Thermocouple	Outlet header from steam drum (steam)	Performance evaluation
	Superheat Steam Temp.*	RTD	Superheater outlet line (steam)	PPS and Display
	Evap. & Superheater Inlet-Outlet Temp.	RTD	Inlet & outlet nozzles for 1 evaporator & superheater (1 loop)	Display & performance evaluation
	Blowdown Temp.	Thermocouple	Blowdown line for each steam drum	Performance evaluation
	Feedwater Pressure	Pressure element	Inlet line to each steam drum	Display and steam generator performance
	Steam Drum Pressure*	Pressure element	Appendage from steam drum	PPS and display
	Recirc. Pump Outlet & Inlet pressure.	Pressure element	Appendage from recirculation pump discharge and suction header.	Performance evaluation
	Superheat Steam* Pressure	Pressure element	Loop output steam line	PPS and display
	Evaporator and Superheater Inlet Pressure	Pressure element	Inlet nozzle for 1 evaporator and superheater (1 loop)	Performance evaluation
	Evaporator and Superheater Outlet Pressure	Pressure element	Outlet nozzle for evaporator and superheater in each loop	SWRPS and performance evaluation
	Steam Drum Level*	Differential Pressure Element	Differential pressure across steam drum	PPS and display

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TABLE 7.5-1 (Continued)

System	Measured Parameters	Instrument	Measured Location	Purpose
Reactor and Vessel Instru-mentation	Core Sodium Exit Temperature	Thermocouples	Selected fuel and blanket assemblies.	Display, control and accident monitoring - core outlet temperature
	Core Peripheral Temperature	Thermocouples	Core periphery - 2 locations	Display - Design verification
	Upper Internals Temperature	Thermocouples	Parts of upper internal structure 6 locations	Display - design verification, predict stress on various components
	Sodium Level above Core*	Level Probe	Reactor vessel plenum	PPS Display and Accident Monitoring
	Upper Internals Movement	Vibration Element	4 Biaxial on parts of appropriate structure	Display - measure vibrations induced by sodium flow
Fuel Failure Monitoring	Cover Gas Gamma Activity	Gamma Spectrometer	Sampling in RSB	Detect each instance of fuel clad failure and characterize failure
	Delayed Neutron Monitoring	BF_3 Counter	Shielded moderator assembly adjacent to each of the PHTS hot leg pipes	Detect fuel in PHTS
	Tag Gas Isotopic Composition	Mass Spectrometer	Gas tag sampling traps in RSB	Locate failed fuel
Leak Detection	Liquid Metal to Gas Leaks	Contact detectors cable detectors aerosol monitors	In various locations in sodium circuits	Identify location of liquid metal to gas leaks for continuous surveillance of liquid metal systems boundaries
Loose Parts Monitoring	Acoustic Noise	Accelerometers	Various locations in PHTS, IHTS, and Steam Generator System heat transfer flow paths	Detect loose part sizes that could degrade operability of components in heat transfer flow paths due to impacting.

TABLE 7.5-1 (Continued)

System	Measured Parameters	Instrument	Measurement Location	Purpose
Leak Detection (cont'd)	Intermediate to Primary Leak	Level probe	IHTS expansion tank	Detect leak in IHX
	Steam Generator Leaks	Hydrogen and Oxygen detectors	Sodium exiting either or both superheater outlets	
	Steam Generator Leaks	Hydrogen and Oxygen detectors	Sodium filled vent line from superheater	Detect small water-sodium and steam-sodium leak in steam generator, identify leaking module, provide signal for operator action
	Steam Generator Leaks	Hydrogen and Oxygen detectors	Sodium filled vent lines from either combined evaporator vents or evaporator A vent	
	Steam Generator Leaks	Hydrogen and Oxygen Detectors	Sodium exiting combined evaporator outlets or evaporator A outlet.	
Sodium-Water Reaction Pressure Relief	Rupture discs Operation*	Sensors	Downstream of rupture discs-before reaction products separation tank	PPS and SWRPS initiation
	Rupture discs Operation	Sensors	Downstream of rupture discs - before sodium dump tank	SWRPRS initiation
	Rupture discs integrity	Pressure Element	Gas space between rupture discs	Surveillance of discs
	SWRPS equipment Temperature	Thermocouples	Surface temperatures of reactor products separation tank, centrifugal separator, drain tank and hydrogen igniter	Surveillance
	Separation Tank Pressure	Pressure Elements	Reactor products separation tank	Surveillance
	Evaporator Water and sodium dump tanks level, pressure and temperature	Level, Pressure, Temperature Elements	Evaporator water and sodium dump tanks	Surveillance

*Safety Related.

TABLE 7.5-2

REACTOR AND VESSEL INSTRUMENTATION

Instrument	Measured Parameter	Location	Purpose
Thermocouple	Core Exit Sodium Temperature	One at each of 30 selected fuel and blanket assemblies	Control, surveillance and accident monitoring Core outlet temp.
		308 additional locations at selected fuel and blanket assemblies	Surveillance, Diagnostic and accident monitoring - Distribution of temperature across the core
Thermocouple	Core Peripheral Temperature	Two spaced locations on the core periphery	Design Verification - Distribution temp. around the core
	Upper Internals Temperature	Six on parts of the upper internal structure	Design Verification - Distribution of temp. to predict stress on various components
Sodium Level Detector	Sodium Level above the core	Four short units distributed equally around periphery	Protection and Control - Measures the operating level of the sodium in the reactor
		Two long units near one of the four short ones	Surveillance and accident monitoring - Measure the sodium level from operating level down to below the top of the outlet nozzle
Vibration Detector	Upper Internals Vibration	Four biaxial on parts of appropriate structure	Design Verification - Measure vibrations induced by sodium flow

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TABLE 7.5-3

SUMMARY OF SODIUM/GAS LEAK DETECTION METHODS

System	Equipment Monitored	Size of Leak (see Section 7.5.5.1.2)	Primary Method of Detection	Back-up Method of Detection
Reactor Enclosure	Reactor vessel and inlet, outlet, overflow nozzles	small	radiation monitoring aerosol monitoring	cable detector
		intermediate and large	aerosol monitoring radiation monitoring cable detector	sodium level in vessel cell temp. & pressure
PHTS	Major pipe sections	small	annuli aerosol monitoring, cell radiation monitoring, cell aerosol monitoring	sodium level in vessels
		intermediate and large	cell radiation monitoring, cell aerosol monitoring, annuli aerosol monitoring	cell temp. & pressure
	Pump housing, IHX shell, check valves	small	aerosol monitors and cell radiation monitoring	cable detectors
		intermediate and large	aerosol monitoring radiation monitoring cable detector	sodium level vessels cell temp & pressure

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TABLE 7.5-3 (Continued)

System	Equipment Monitored	Size of Leak (see Section 7.5.5.1.2)	Primary Method of Detection	Back-up Method of Detection
IHTS and Steam Gen.	Major pipe sections, pump housing, expansions tank, steam generators distribution lines, steam generators	small	aerosol monitoring	visual inspection
		intermediate and large	cable detector aerosol monitoring	vault smoke detector
Aux. Liq. Metal, Impurity Monitoring and Analysis, and Reactor Refueling	EVST, overflow and storage tanks	small	cell radiation monitoring (RCB only) & aerosol monitoring	sodium levels in vessels
		intermediate and large	cell radiation monitoring, aerosol monitoring, and cable detector	cell temp. & pressure
	Cold traps, heat exchangers, impurity monitoring, and other small equipment	all	cell radiation monitoring, aerosol monitoring, and cable detector	cell temp. & pressure (large leaks)

TABLE 7.5-4

POST ACCIDENT MONITORING

	Parameter	Sensor Location
	1) Reactor Sodium Level	RCB
	2) IHX Inlet Temperature	RCB
	3) IHX Outlet Temperature	RCB
	4) DHRS Cold Leg Temperature	RCB
	5) RCB Pressure	RCB
	6) RCB Temperature	RCB
	7) PWST Level	SGB
	8) Auxiliary Feedwater Flow	SGB
	9) Steam Drum Level	SGB
	10) Steam Drum Pressure	SGB
50	11) Deleted	
49	12) EVST Outlet Piping Temperature	RSB

Table 7.5-4

Safety Functions and Primary Systems Monitored by ISMS

SAFETY FUNCTION MONITORED	SYSTEMS/SUBSYSTEMS MONITORED	
Decay Heat Removal	SGAHRS	PACC
		Aux FW/Vent
		Steam Generator
		Heat Transport
	DHRS	PHTS Heat Transport
		Direct Heat Removal System
Fuel Storage Heat Removal	EVST Cooling	Forced Circulation
		Natural Circulation
Control Room Habitability	Control Room Filtration H&V	
Annulus Filtration	RCB/Annulus H&V	
RSB Filtration	RSB H&V	

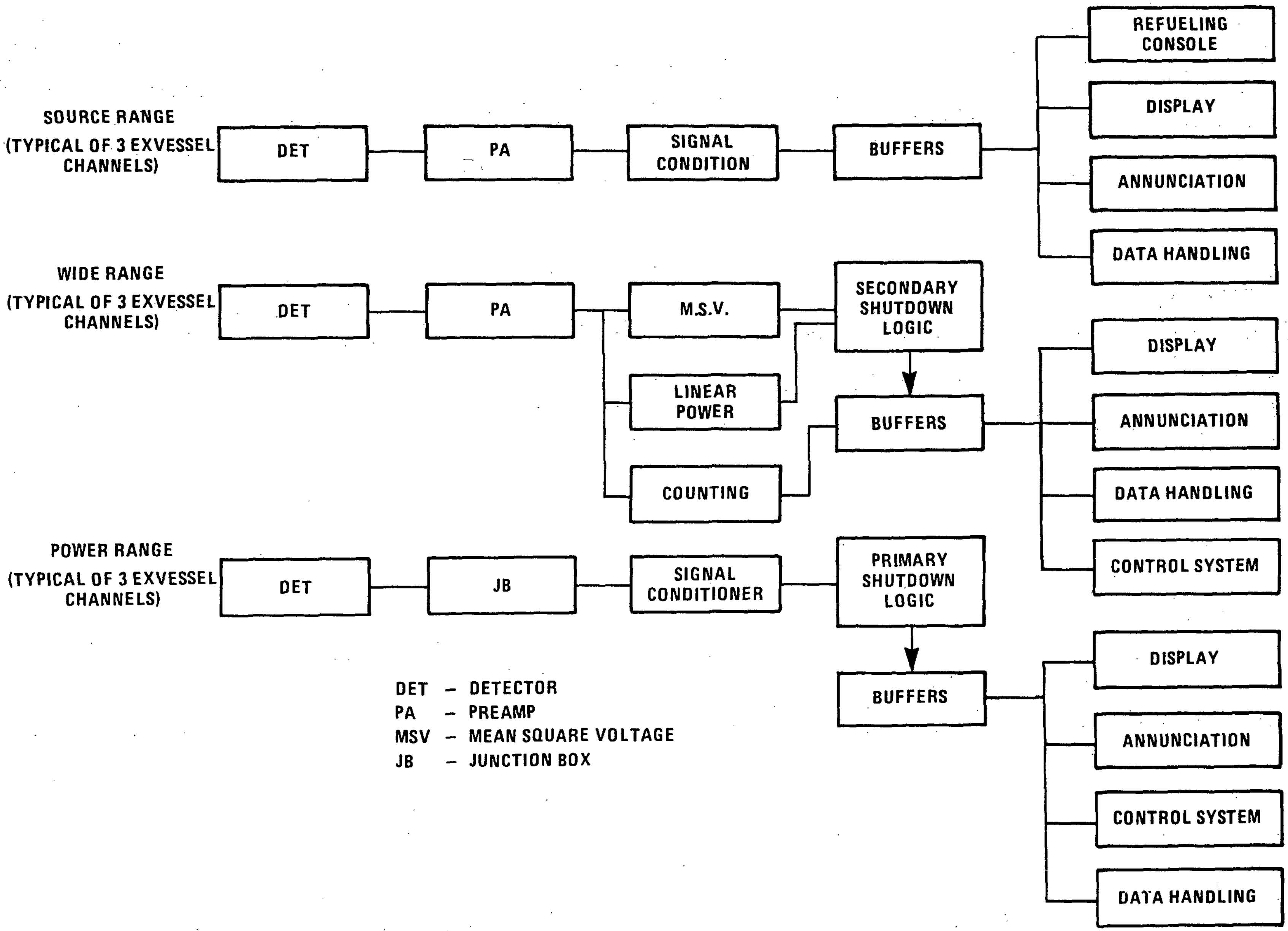

Figure 7.5-1. CRBRP Flux Monitoring System Block Diagram

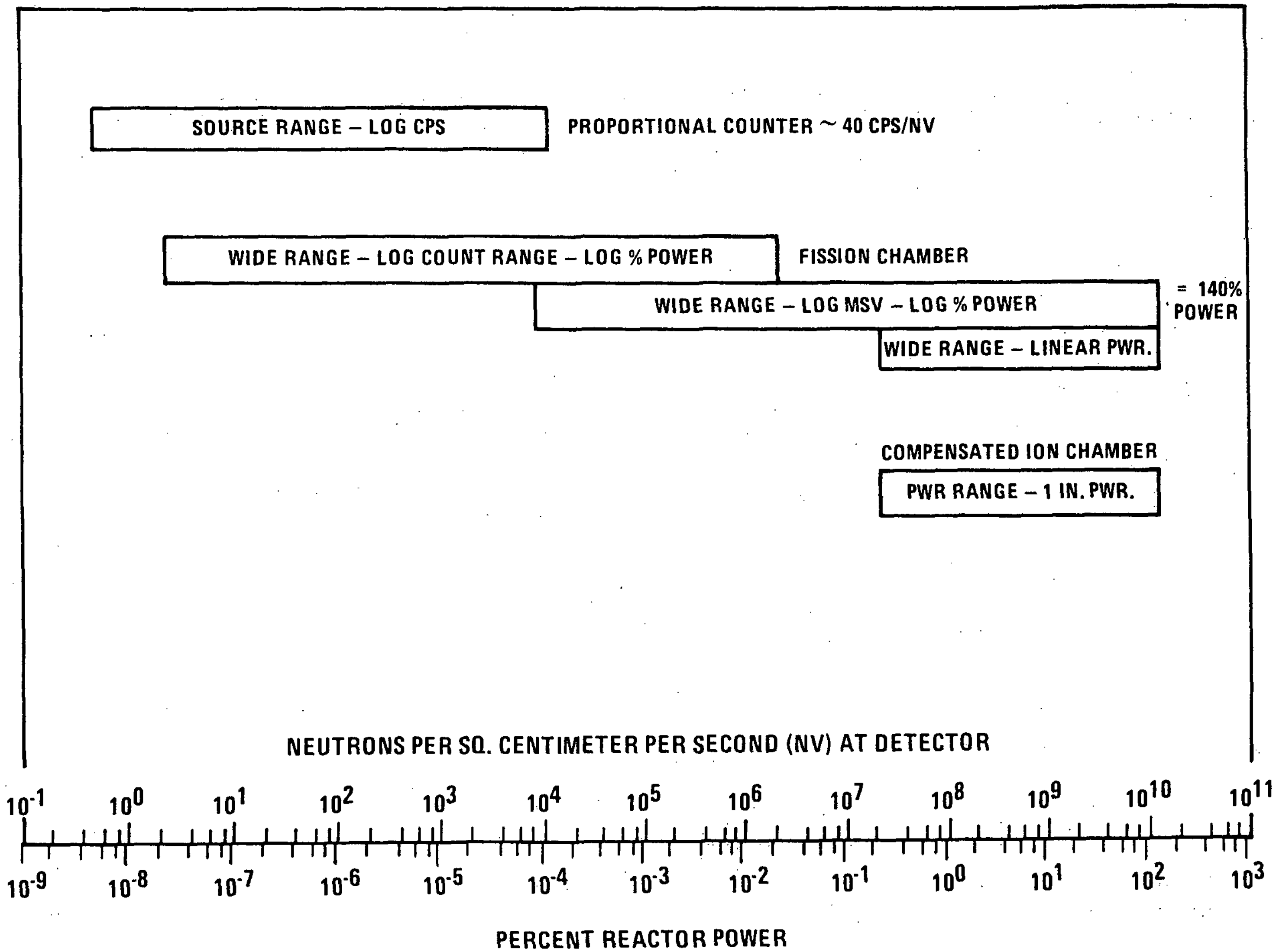

Figure 7.5-2. CRBRP Flux Monitoring System Instrument Range Coverage

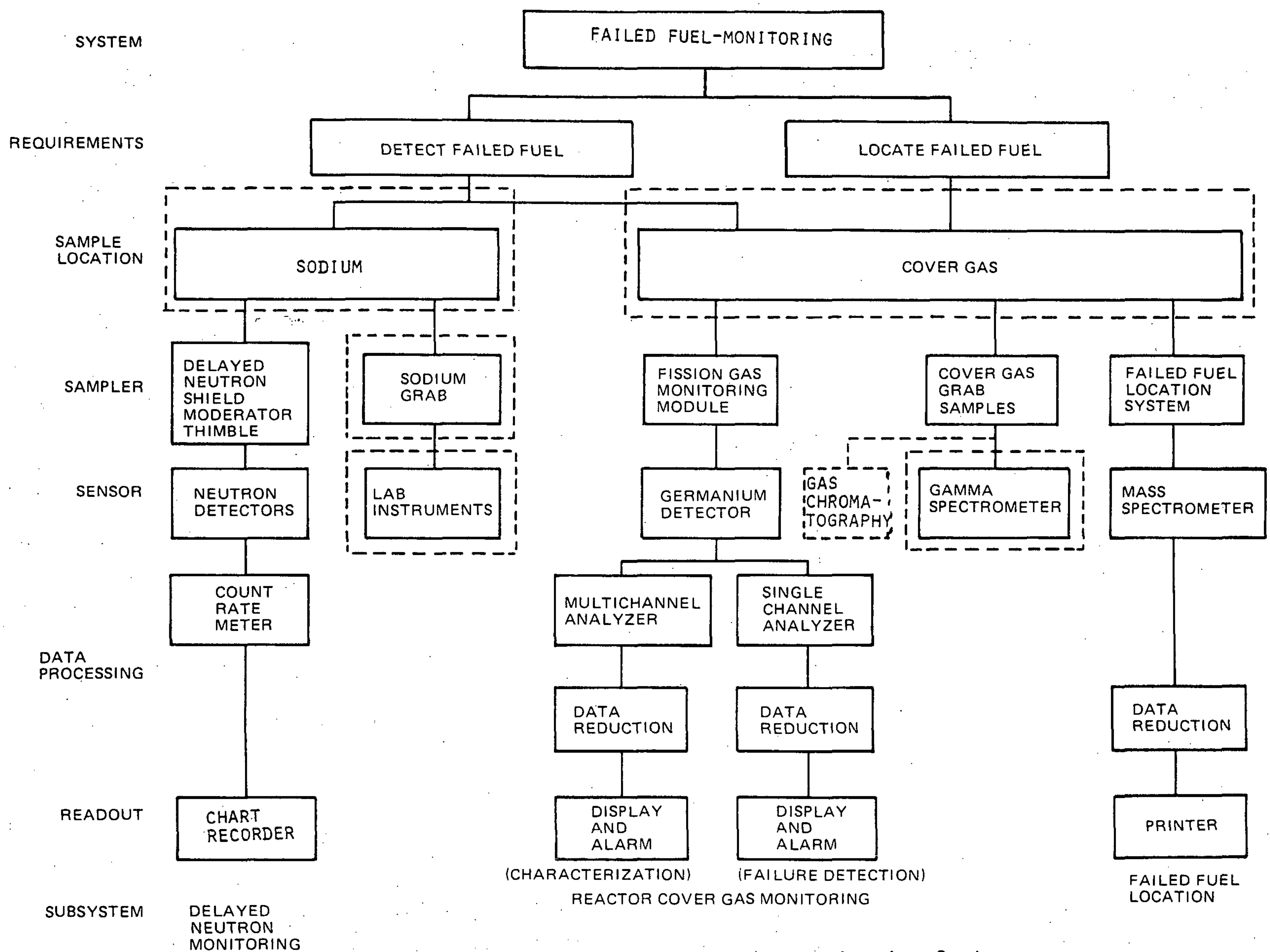

Figure 7.5-3. Fuel Failure Monitoring System

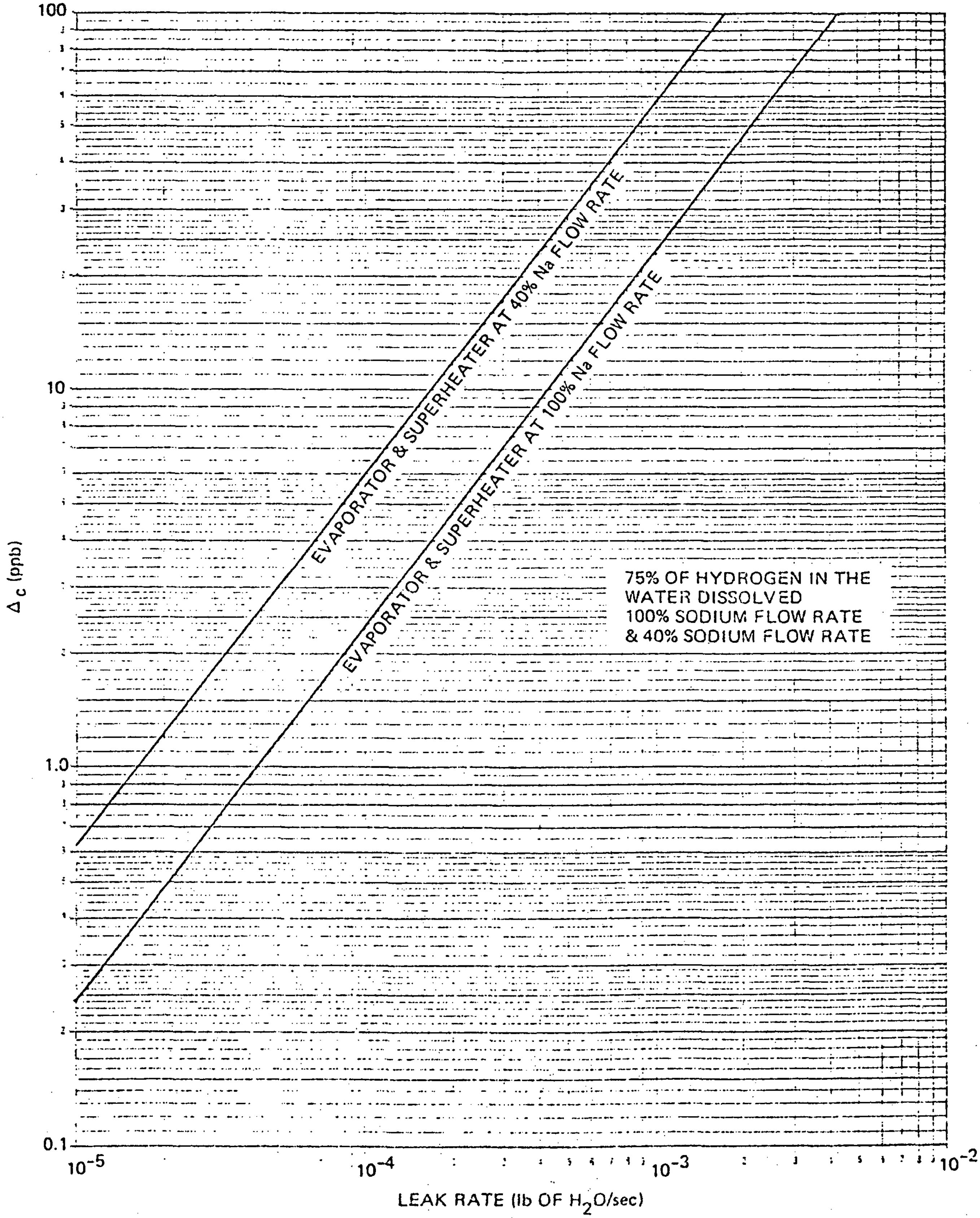

Figure 7.5-4 Main Sodium Stream First Pass Hydrogen Concentration Versus Leak Rate

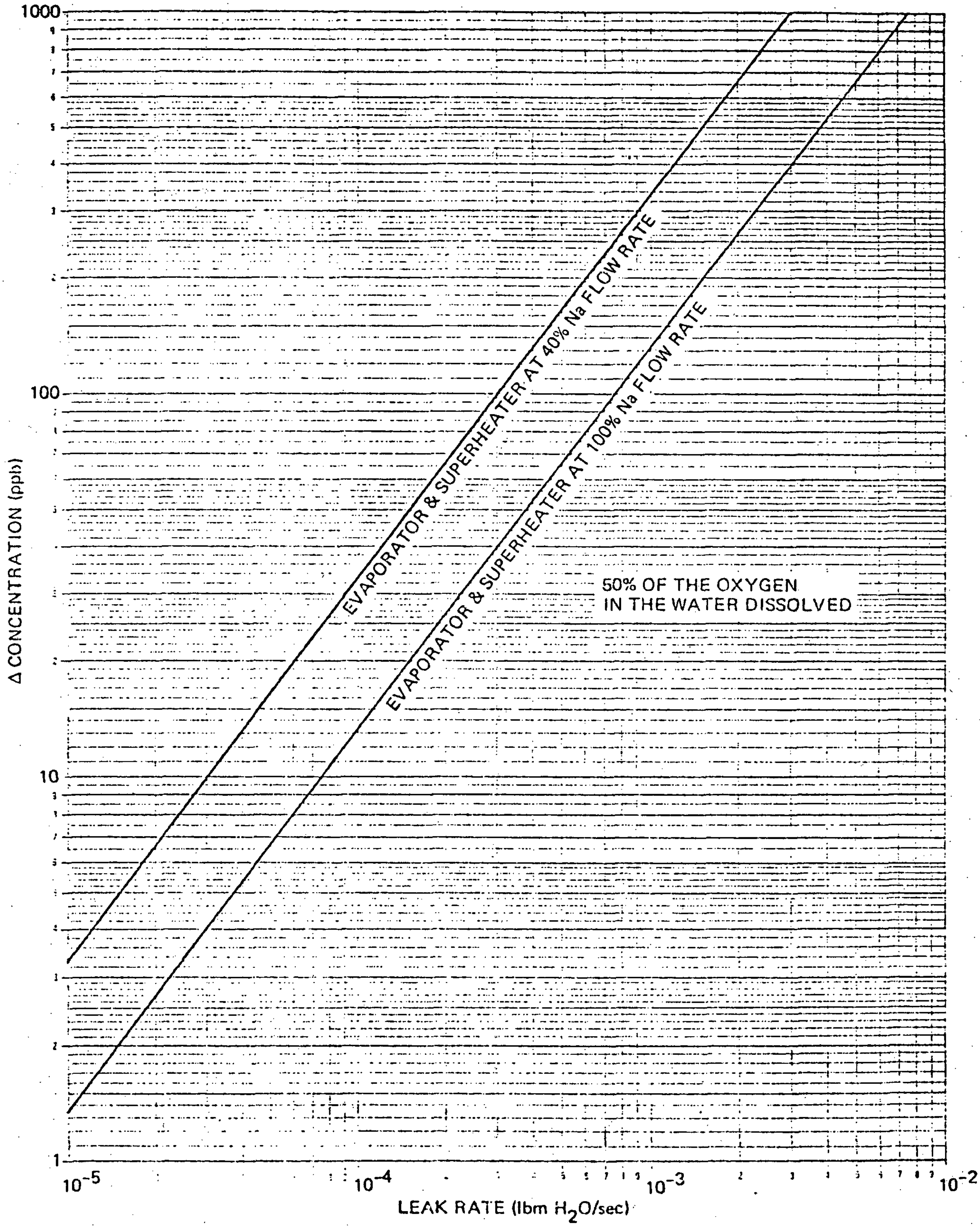

Figure 7.5-4a Main Sodium Stream First Pass Oxygen Concentration Versus Leak Rate

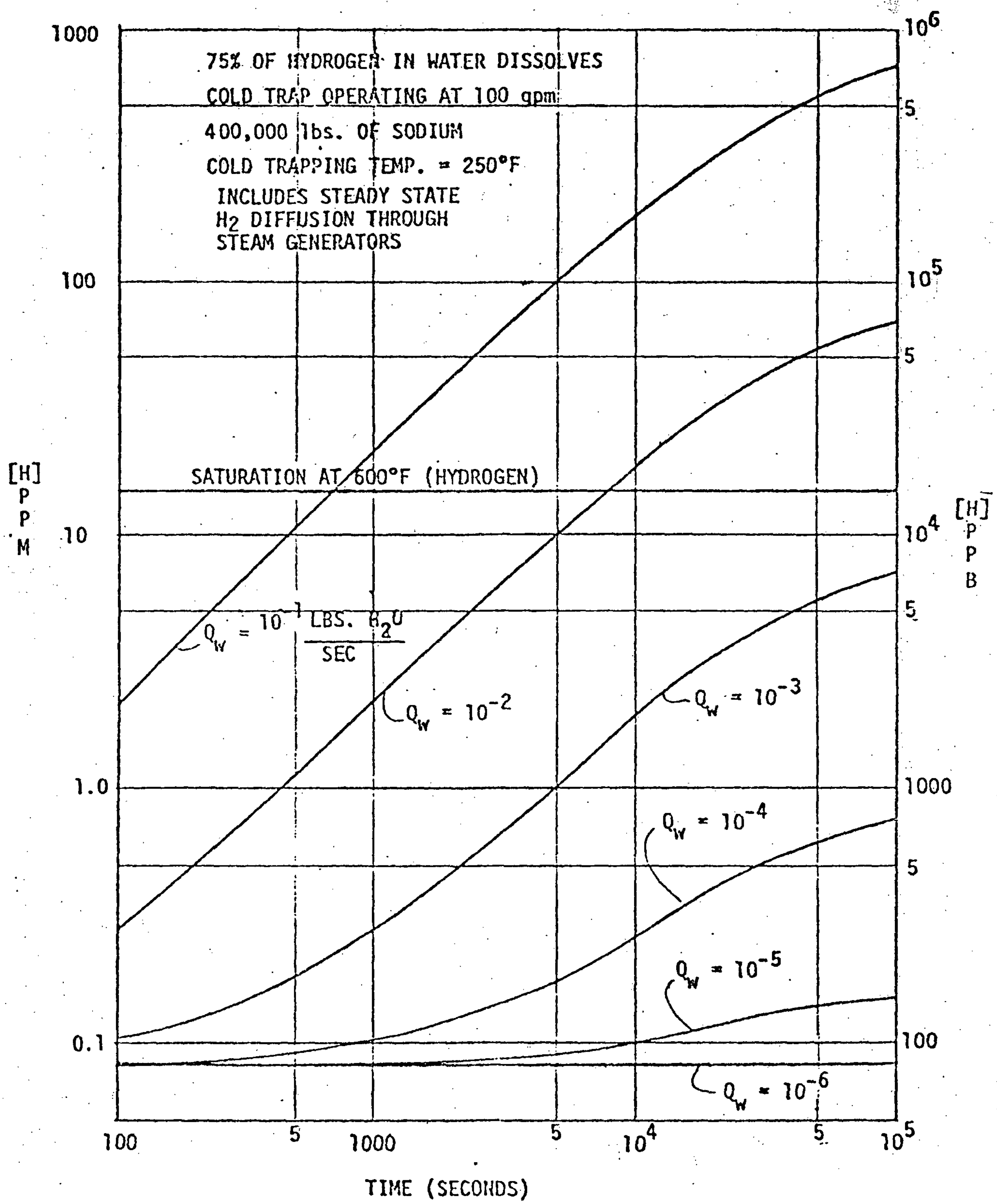

FIGURE 7.5-5

Hydrogen Concentration vs. Time For Various Water Leak Rates

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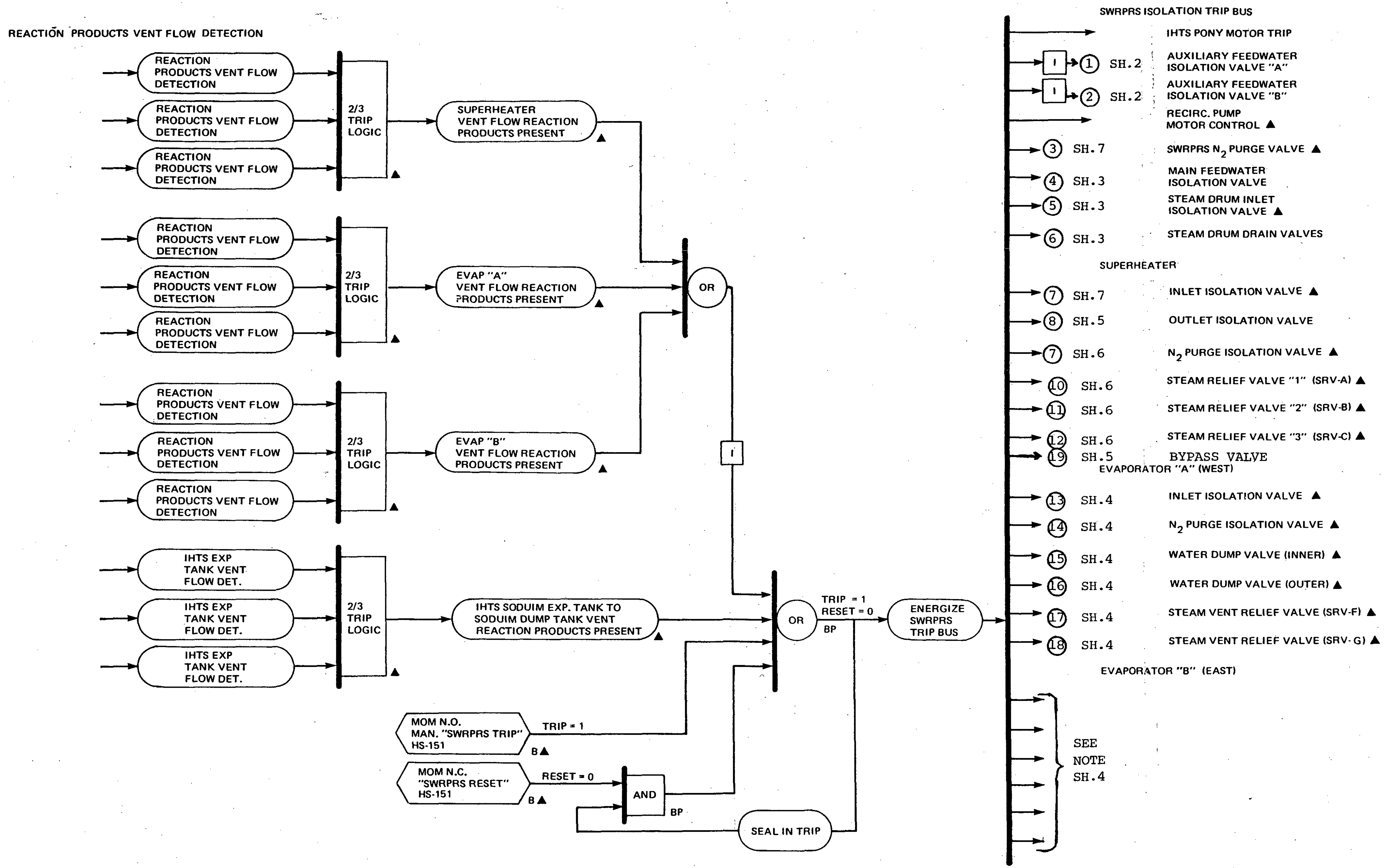

Figure 7.5-6 SWRPRS TRIP AND SWRPRS CONTROLLED ISOLATION VALVES CONTROL LOGIC DIAGRAM

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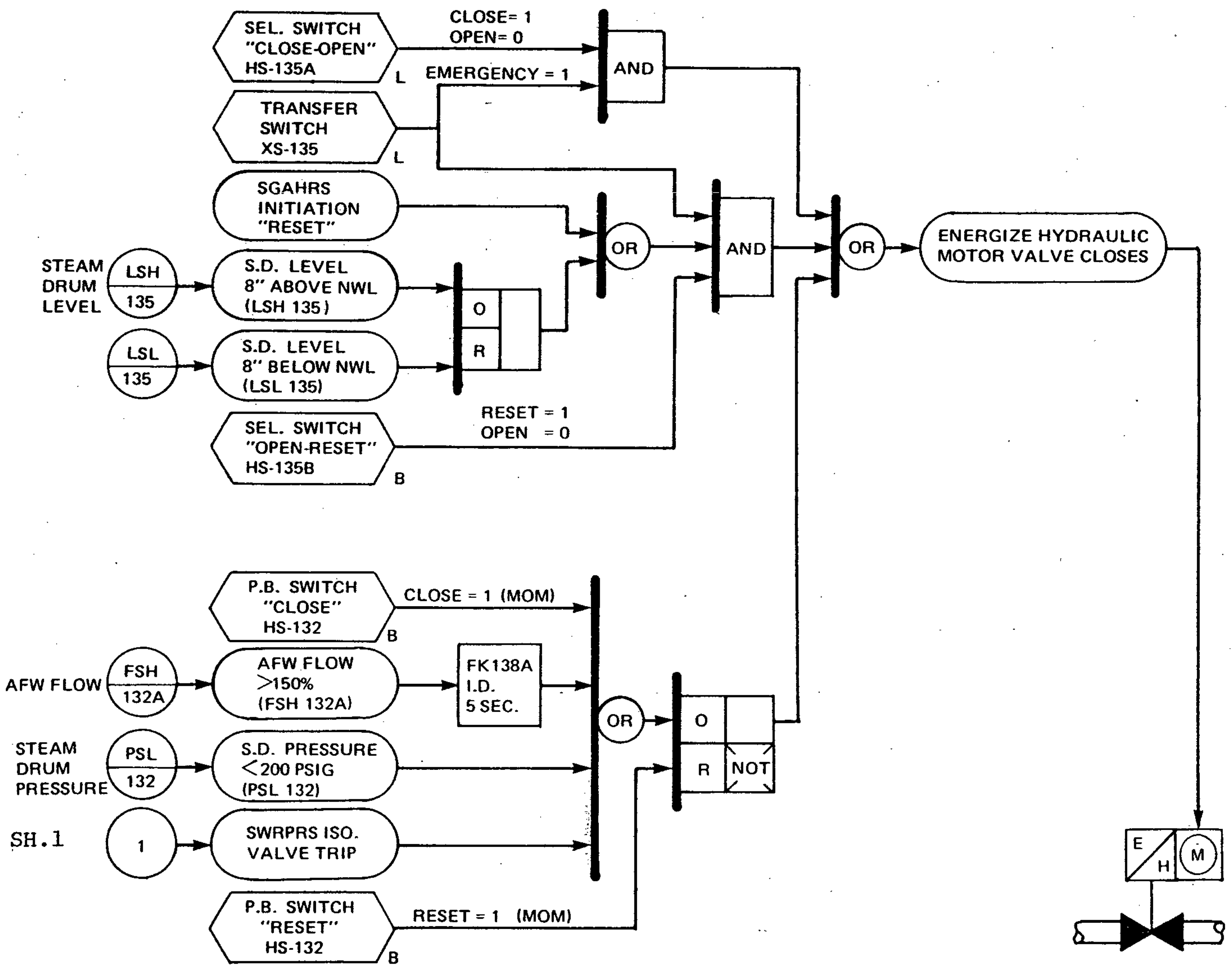
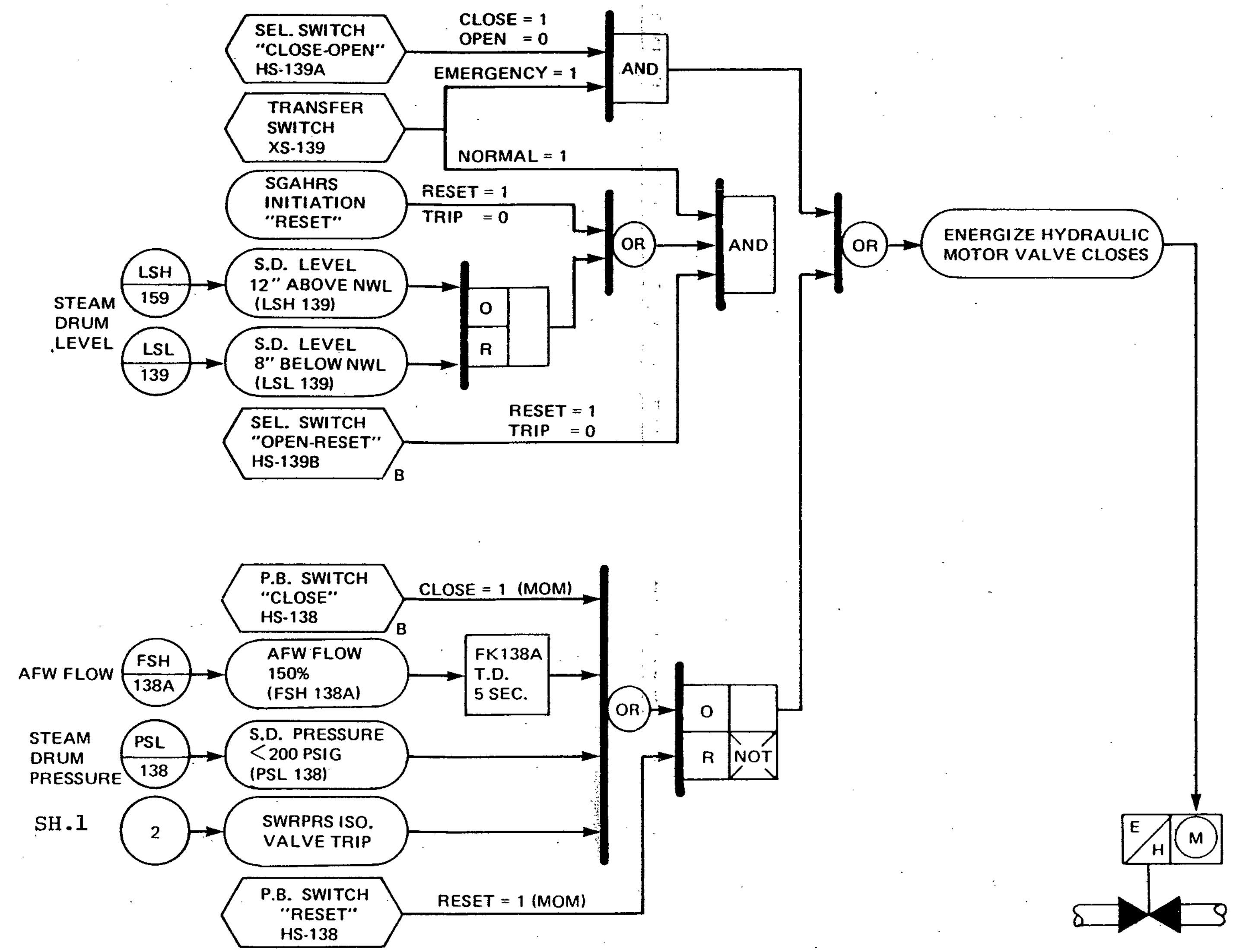
Mar. 1982



AUXILIARY FEEDWATER ISOLATION VALVE "A"



AUXILIARY FEEDWATER ISOLATION VALVE "B"

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80-433-07

Figure 7.5 - 6 SWRPRS TRIP AND SWRPRS CONTROLLED ISOLATION VALVES CONTROL LOGIC DIAGRAM

TABLE 1

LOOP NO.	1				2				3			
CELL NO.	241	221	224	207	242	222	225	208	243	223	226	209
INSTRUMENT NO.	151	152	153	154	251	252	253	254	351	352	353	354

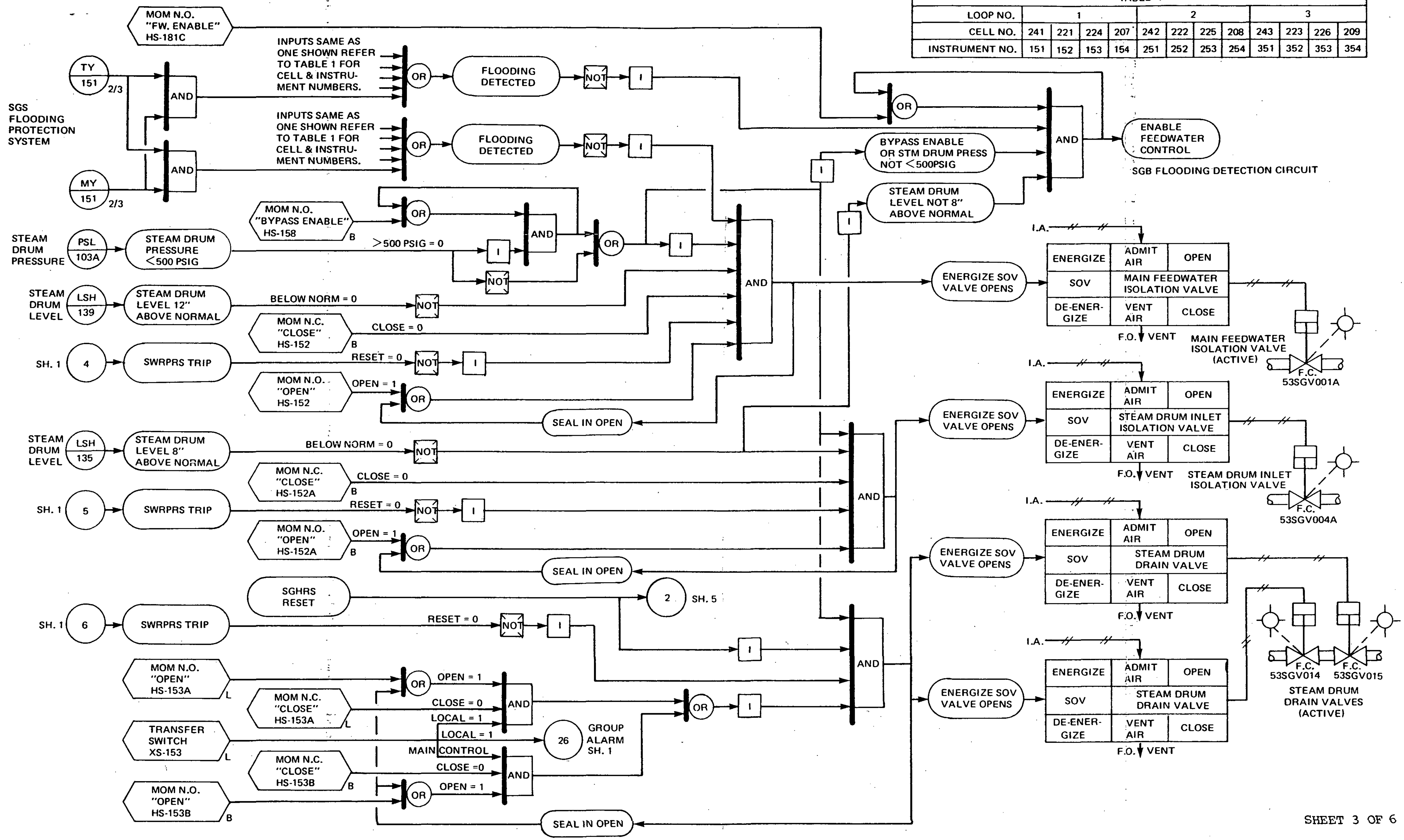

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Figure 7.5 - 6 SWRPRS TRIP AND SWRPRS CONTROLLED ISOLATION VALVES CONTROL LOGIC DIAGRAM

80-433-06

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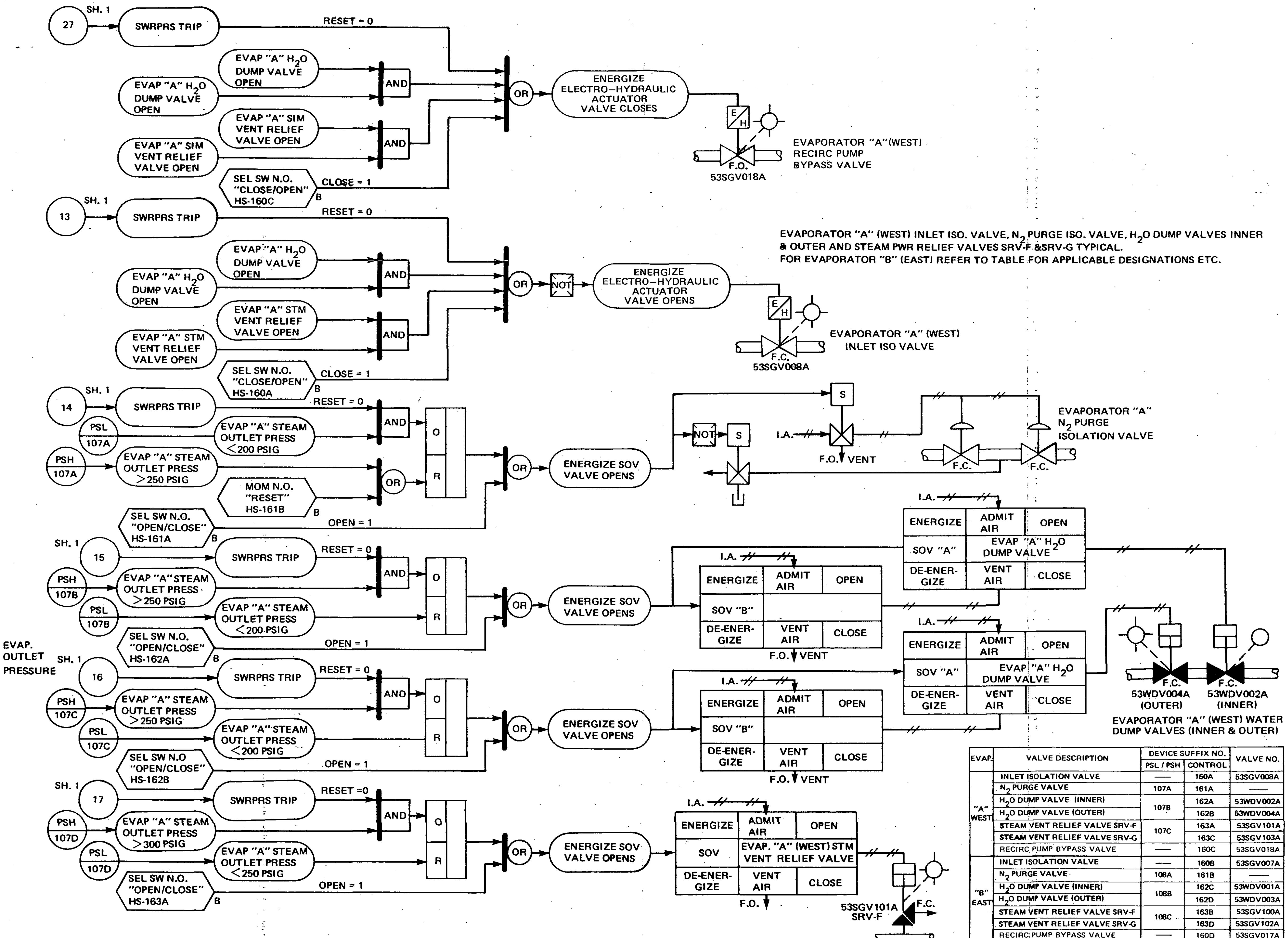
Mar. 1982



EVAP.	VALVE DESCRIPTION	DEVICE SUFFIX NO.		VALVE NO.
		PSL / PSH	CONTROL	
"A" WEST	INLET ISOLATION VALVE	—	160A	53SGV008A
	N_2 PURGE VALVE	107A	161A	—
	H_2O DUMP VALVE (INNER)	107B	162A	53WDV002A
	H_2O DUMP VALVE (OUTER)		162B	53WDV004A
	STEAM VENT RELIEF VALVE SRV-F	107C	163A	53SGV101A
	STEAM VENT RELIEF VALVE SRV-G		163C	53SGV103A
	RECIRC PUMP BYPASS VALVE	—	160C	53SGV018A
"B" EAST	INLET ISOLATION VALVE	—	160B	53SGV007A
	N_2 PURGE VALVE	108A	161B	—
	H_2O DUMP VALVE (INNER)	108B	162C	53WDV001A
	H_2O DUMP VALVE (OUTER)		162D	53WDV003A
	STEAM VENT RELIEF VALVE SRV-F	108C	163B	53SGV100A
	STEAM VENT RELIEF VALVE SRV-G		163D	53SGV102A
	RECIRC PUMP BYPASS VALVE	—	160D	53SGV017A

Figure 7.5 - 6 SWRPRS TRIP AND SWRPRS CONTROLLED ISOLATION VALVES CONTROL LOGIC DIAGRAM

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Mar. 1982

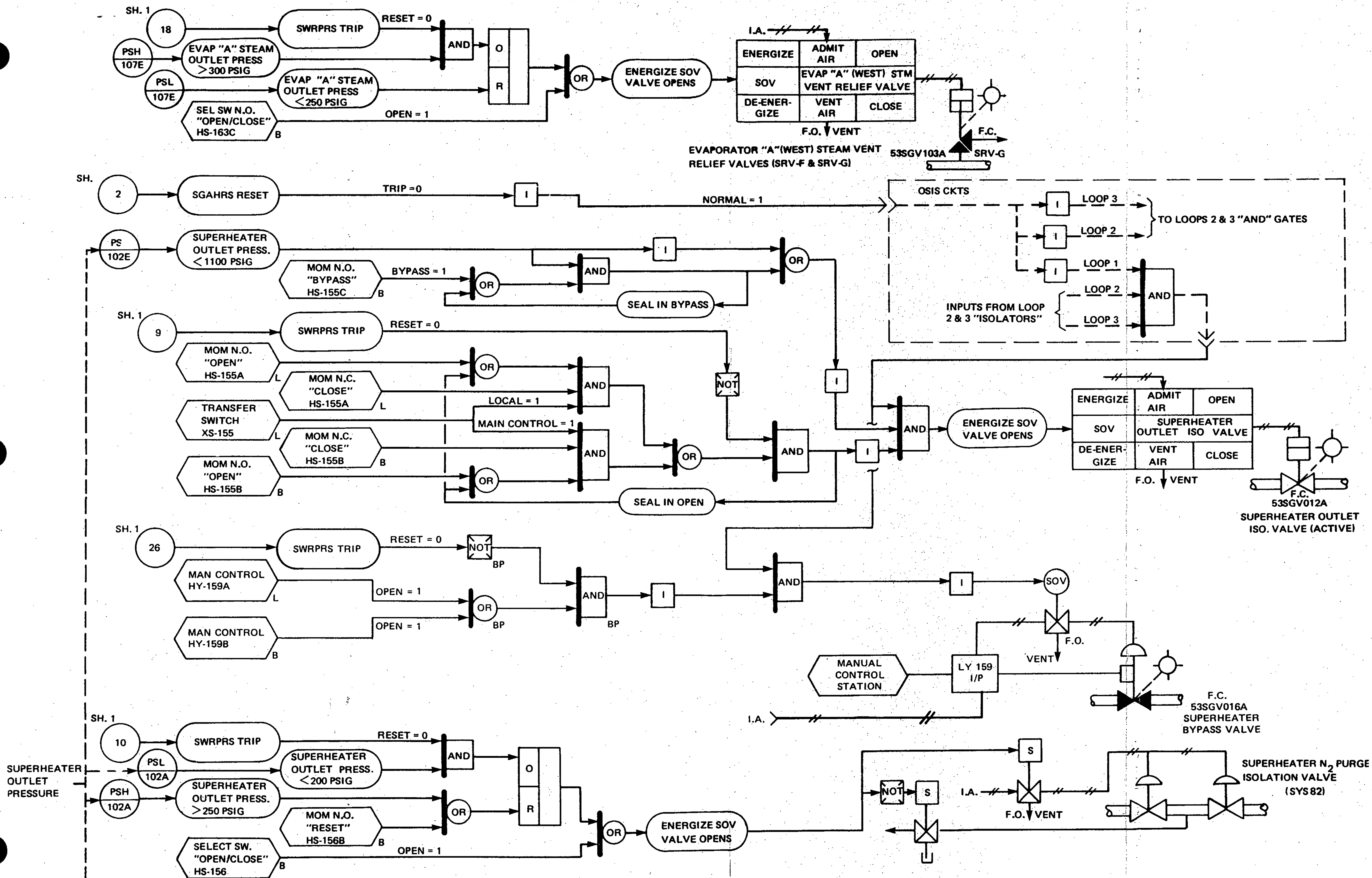

Figure 7.5 - 6 SWRPRS TRIP AND SWRPRS CONTROLLED ISOLATION VALVES CONTROL LOGIC DIAGRAM

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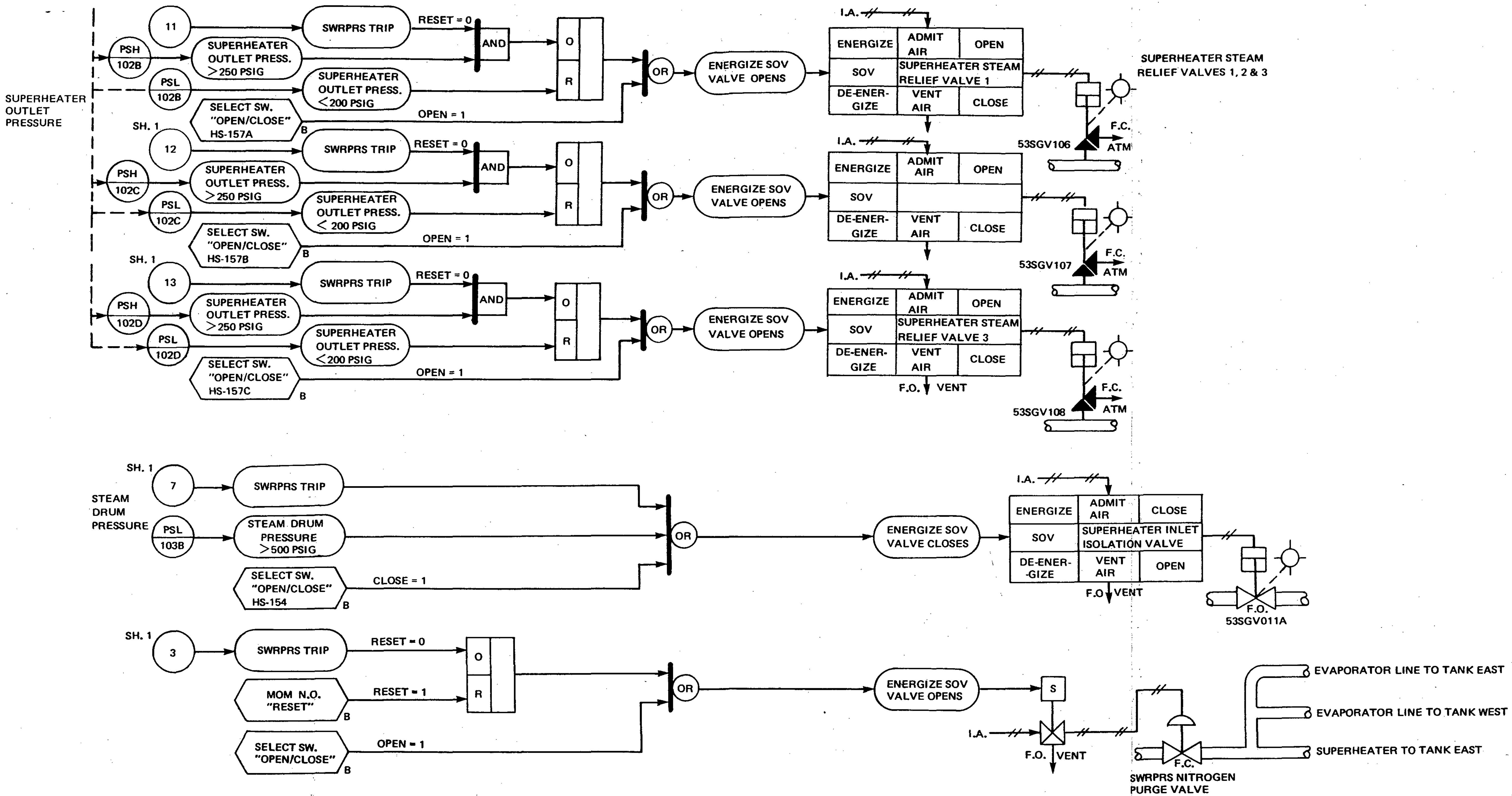
Nov. 1982



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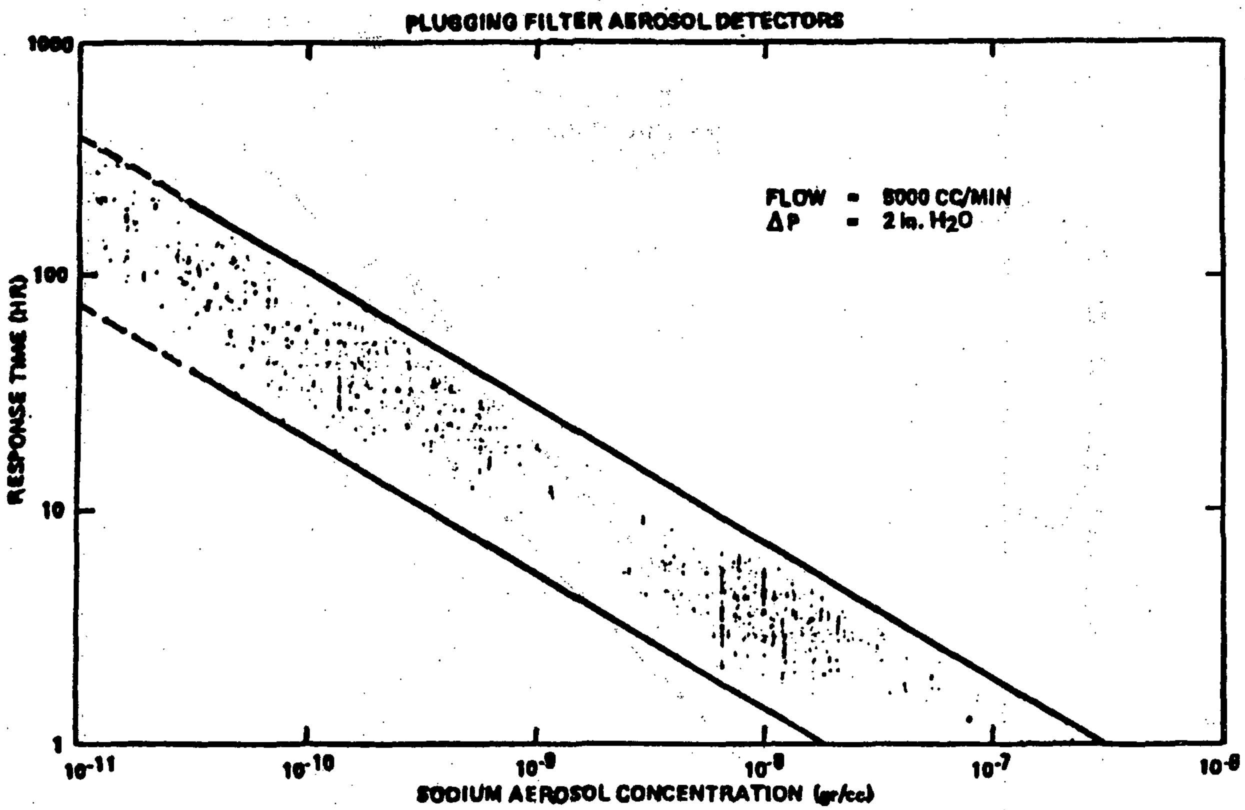
Amend. 67
March 1982

Figure 7.5 - 6 SWRPRS TRIP AND SWRPRS CONTROLLED ISOLATION VALVES CONTROL LOGIC DIAGRAM

Figure 7.5-7 Liquid Metal/Gas Leak Detection System
Response Time Vs. Sodium Aerosol Concentration (Inerted Cells)



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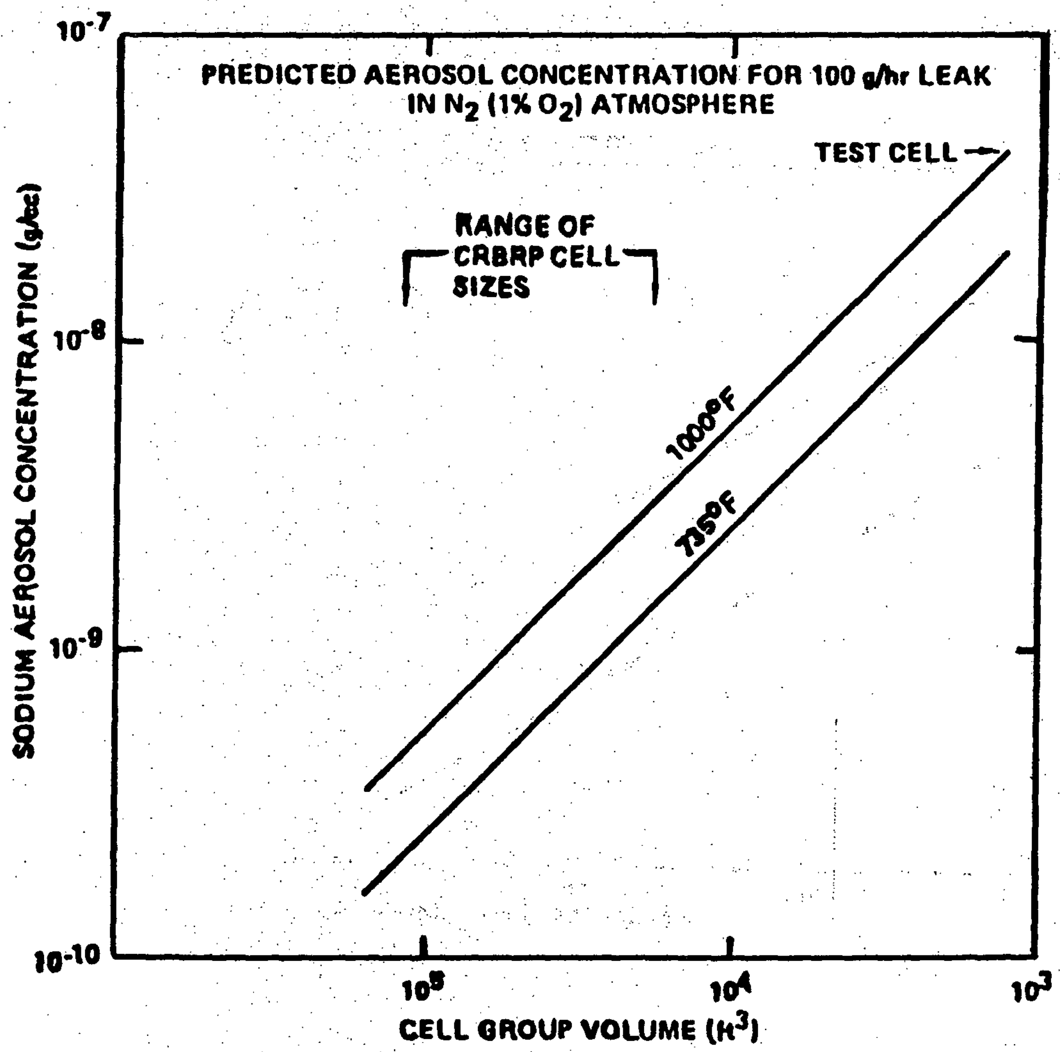
Oct. 1982

Figure 7.5-8 Liquid Metal/Gas Leak Detection System Predicted Aerosol Concentration for 100 g/HR Leak in N_2(1% O_2) Atmosphere



7.6 OTHER INSTRUMENTATION AND CONTROL SYSTEMS REQUIRED FOR SAFETY

The additional instrumentation and control systems required for safety which have not been discussed earlier in Chapter 7 are identified as the Emergency Plant Service Water, Emergency Chilled Water, Recirculating Gas Cooling, Heating, Ventilating, Air Conditioning, Instrumentation and Control Systems, and the Direct Heat Removal Service Instrumentation and Control. The Radiation Monitoring System also contains safety related components which are discussed in Chapter 11. The Emergency Plant Service Water, Emergency Chilled Water, Recirculating Gas Cooling, Heating Ventilating, Air Conditioning Systems, Fuel Handling, and DHRS Instrumentation and Control are discussed in this Section. Review of the functional control diagrams will require reference to the symbols, notes and abbreviations as shown in Table 7.6-1.

7.6.1 Emergency Plant Service Water and Emergency Chilled Water Instrumentation and Control System

7.6.1.1. Emergency Plant Service Water System (EPSW)

The EPSW System consists of two redundant divisions which supply cooling water to the diesel generators, the Emergency Chilled Water System and seismically qualified Non-Sodium Fire Protection System.

The Instrumentation and Control System is provided for automatic control of the Emergency Plant Service Water System, to monitor and indicate system process parameters during normal and off-normal conditions, and to provide signal inputs to Plant Data Handling and Display System.

Functional Control Diagrams for Emergency Plant Service Water System are identified in Figures 7.6-1, 7.6-2, 7.6-3 and 7.6-4.

7.6.1.2 Design Criteria

Design criteria that are applicable to Emergency Plant Service Water Instrumentation and Control System are as follows:

A. EPSW System is provided with Class 1E power supply, and is backed up by Diesel Generators to provide power during off-normal conditions.

B. No single failure of an instrument, interconnecting cable or panel will prevent a key process variable from being controlled or monitored in both redundant divisions.

C. Physical and electrical separation of redundant portions of Emergency Plant Service Water is provided.

D. System level manual initiation capabilities are provided in both divisions to perform all the actions performed by automatic initiation.

E. Instrumentation used in the control of Emergency Plant Service water will function during and after an SSE.

F. Instrumentation used in the control of Emergency Plant Service Water will function during normal environmental conditions and during environmental conditions created by any design basis accident.

G. Capabilities for periodic testing and calibration of all instruments are provided.

H. Capabilities are provided for remote shutdown, should the control room become uninhabitable.

I. Capabilities are provided to monitor the inoperable status of components in accordance with Reg. Guide 1.47.

J. Capabilities are provided to monitor the process variables to assess plant and environs conditions during and following an accident.

7.6.1.3 <u>Design</u>

Instrumentation and controls are provided for the following equipment in the EPSW System: EPSW Pumps; EPSW Makeup Water Pumps; Emergency Cooling Tower Fans and Temperature Control Valves. For a complete description of the EPSW System refer to Chapter 9.9.2.

7.6.1.3.1 <u>Control System</u>

A. Remote, auto and manual controls are provided in Control Room for EPSW Pumps, EPSW Makeup Water Pumps; Emergency Cooling Tower Fans.

B. Local, auto and manual controls are provided in Local Panels for EPSW Pumps; EPSW Makeup Water Pumps; Emergency Cooling Tower Fans; and Temperature Control Valves.

C. EPSW will start automatically under the following conditions:

i) On an Emergency Chilled Water System start demand;

ii) 20 seconds after the Diesel Generator Load Sequencer is actuated;

iii) when the system level manual control is initiated from Control Room.

7.6.1.3.2 <u>Monitoring Instrumentation</u>

The following process variables are monitored through indication and alarms:

A. EPSW Pump Discharge Temperature

B. EPSW Pump Discharge Pressure

C. EPSW Pump Pit Level

D. EPSW Makeup Pump Flow

E. Operating Basin Overflow

F. EPSW Makeup Pump Discharge Pressure

G. Emergency Cooling Tower Basin Level

H. EPSW Flow to Emergency Chillers

I. EPSW Temperature at the Discharge of Emergency Chillers

J. EPSW Flow from Diesel Generators Heat Exchangers

K. EPSW Temperature at the Discharge of Diesel Generators Heat Exchangers

L. Diff. Pressure Across Emergency Chillers

M. Transfer of Controlling Capabilities from Control Room to Local Panels

N. Pump and Fan Status

Process variables identified above with 'A' and 'H' are designated as accident monitoring variables to assess plant and environs conditions during and following an accident. Refer to Section 7.5.11 of PSAR for detailed requirements on Accident Monitoring.

7.6.1.3.3 Inputs to PDH&DS

The following process variables are provided as inputs to Plant Data Handling & Display System (Non-Safety System):

A. EPSW Discharge Temperature

B. Emergency Cooling Tower Basin Level

C. EPSW Temperature at the Discharge of Emergency Chiller

D. EPSW Temperature at the Discharge of Diesel Generator Heat Exchanger

E. EPSW Flow to Emergency Chiller

Inoperable status of EPSW Pumps; Makeup Pumps; and Cooling Tower Fans is also monitored through Inoperable Status Monitoring System.

7.6.1.1.3.4 Design Analysis

EPSW System is designed to operate automatically. The system is operated only during emergency conditions. EPSW System components are cascaded to operate in sequence. Starting of EPSW Pumps will operate EPSW Makeup Pumps and Cooling Tower Fans. System will not operate when the EPSW Pump pit level is low or when electrical fault exists.

The design of the EPSW System is in conformance with the following IEEE standards listed in Table 7.6-2.

7.6.2 Emergency Chilled Water (ECW) System

The ECW System consists of two redundant divisions which supply chilled water. Controls are provided for the following equipment in the ECW System: Emergency Chillers, Circulating Pumps, Expansion Tank Valve, Normal-to-Emergency Isolation Valves, Temperature Control Valves, and "Recirculating Gas Cooling System Heat Exchanger and Secondary Coolant Heat Exchanger" Leak Isolation Valves. Detailed description of these controls is given in the following paragraphs. For a complete description of the ECW System, refer to Chapter 9.7.2.

The ECW System cannot operate without support from the Plant Electrical Power System and the Emergency Plant Service Water (EPSW) System. The ECW System power supply is Class 1E and requires a diesel generator back-up. A detailed description of the diesel generators and the Plant Electrical Power System is given in Chapter 8. The EPSW System supplies service water to the ECW Chiller. A detailed description of the EPSW System is given in Chapter 9.9.2.

Functional Control Diagrams for Emergency Chilled Water System are identified in Figures 7.6-5, 7.6-6, 7.6-7, 7.6-8, 7.6-9 and 7.6-10.

7.6.2.1 Design Criteria

Design criteria that are applicable to Emergency Chilled Water Instrumentation and Control System are as follows:

A. ECW System is provided with Class 1E power supply, and is backed up by diesel generators to provide power during off-normal conditions.

B. No single failure of an instrument, interconnecting cable or panel will prevent a key process variable from being controlled or monitored in both redundant divisions.

C. Physical and electrical separation of redundant portions of Emergency Chilled Water is provided.

D. System level manual initiation capabilities are provided in both divisions to perform all the actions performed by automatic initiation.

E. Instrumentation used in the control of Emergency Chilled Water will function during and after an SSE.

F. Instrumentation used in the control of Emergency Chilled Water will function during normal environmental conditions and during environmental conditions created by any design basis accident.

G. Capabilities for periodic testing and calibration of all instruments are provided.

H. Capabilities are provided for remote shutdown, should the control room become uninhabitable.

I. Capabilities are provided to monitor the inoperable status of components in accordance with Reg. Guide 1.47.

J. Capabilities are provided to monitor process variables to assess plant and environs conditions during and following an accident. Refer to Section 7.5.11 of the PSAR for detailed requirements on Accident Monitoring.

7.6.2.2 Design

Instrumentation and controls are provided for the following equipment in the ECW System: ECW Pumps; ECW Chillers; ECW Isolation Valves.

7.6.2.2.1 Control System

A. Remote, auto and manual controls are provided in the control room for ECW Pumps, ECW Chillers and Valves.

B. Local, auto and manual controls are provided in local panels for ECW Pumps, ECW Chillers, and Valves.

C. ECW Pump will start automatically under the following conditions:

i) when NCW flow to ECW loop is low;

ii) 3 minutes after the diesel generator load sequencer is actuated;

iii) when the system level manual control is initiated from control room.

7.6.2.2.2 Monitoring Instrumentation

The following process variables are monitored through indication and/or alarms:

1. ECW Chiller Chilled Water Discharge Temperature
2. ECW Chiller Chilled Water Discharge Pressure
3. ECW Chiller Chilled Water Inlet Temperature
4. ECW Flow Status through Emergency Chiller
5. ECW Flow from Emergency Chiller
6. NCW Low Flow to ECW Loop
7. ECW Expansion Tank Low Level
8. ECW Expansion Tank High Level
9. ECW Expansion Tank Low Press

10. Transfer of Controlling Capabilities from Control Room to Local Panel Status

11. ECW Pump Operation Status

12. ECW Chiller Operation Status

13. ECW Isolation Valves Status

14. ECW Pump Thermal Overload Status

15. ECW Chiller Shaft Vibration

16. ECW Chiller Low Chill Water Temperature

17. ECW Chiller Condenser High Pressure

18. ECW Chiller Low Evap Pressure

19. ECW Chiller Bearing Oil High Temperature

20. ECW Chiller Motor High Temperature

21. ECW Chiller Compressor Discharge High Temperature

22. ECW Entering Secondary Coolant Heat Exchanger Temperature

23. ECW Leaving Secondary Coolant Heat Exchanger Temperature

24. DT-J Temperature Entering Secondary Coolant Heat Exchanger

25. DT-J Temperature Leaving Secondary Coolant Heat Exchanger

26. Secondary Coolant Expansion Tank High Pressure

27. Secondary Coolant Expansion Tank Low Pressure

28. Secondary Coolant Exp Tank Low Press Pump

29. Secondary Coolant Leakage Alarm

30. Secondary Coolant Pump Suction Pressure

31. DT-J Pump Discharge Press

32. ECW Expansion Tank Level.

Process variables identified above with 1, 2, 3, 4, 11, 12, 13 and 32 are designated as accident monitoring variables to assess plant and environs conditions during and following an accident. Refer to Section 7.5.11 of PSAR for detailed requirements on Post Accident Monitoring.

7.6.2.2.3 Inputs to PDH&DS

The following process variables are provided as inputs to Plant Data Handling & Display System (Non-Safety System):

A. ECW Temperature at the Inlet of Emergency Chiller

B. ECW Temperature at the Discharge of Emergency Chiller

C. ECW Flow from Emergency Chiller

D. ECW Chiller Trip Status

E. ECW Containment Isolation Valves Status

F. Secondary Coolant Expansion Tank DT-J Leakage

7.6.2.2.4 Design Analysis

ECW System is designed to operate automatically. The system is operated only during emergency condition. ECW System components are cascaded to operate in sequence. Low flow of NCW to ECW loop signal will align ECW isolation valves and operate ECW Pumps, Emergency Plant Service Water Loops, and ECW Chillers. System will not operate when the EPSW flow through chiller is not established or when electrical fault exists.

The design of the ECW System is in conformance with the IEEE Standards listed in Table 7.6-2.

7.6.3 Direct Heat Removal Service (DHRS) and Ex-Vessel Storage Tank (EVST) Cooling System Instrumentation and Control System

7.6.3.1 Design Description

7.6.3.1.1 Function

The DHRS (fluid system and mechanical components as described in Section 5.6, and electrical components as described below) provides a supplementary and separate means of removing long term decay heat from the reactor for the remote case in which none of the steam generator decay heat removal paths are available. DHRS is separate in function and equipment location from the Steam Generator Auxiliary Heat Removal System (SGAHRS), and there is no common sharing of instrumentation or controls between them.

The EVS Cooling System (described in Section 9.1.3) removes decay heat from fuel stored in the Ex-Vessell Storage Tank. The redundant liquid metal cooling circuits using forced convection heat rejection and one liquid metal cooling circuit using natural draft heat rejection provide this function.

The DHRS and EVST Cooling System Instrumentation and Controls are safety related and are provided to permit the monitoring of system conditions and to provide alarm indication of off-normal conditions. These are the same instrumentation and controls that are provided for EVST cooling (Section 9.1.3.1.5) and the reactor primary sodium overflow circuits (Section 9.3.2.5) with the addition of a few temperature monitoring instruments located on the NaK lines connecting the overflow heat exchanger with the EVST NaK cooling loops (see Figures 9.3-2 and 9.3-3).

7.6.3.1.2 Design Criteria

Design criteria that are applicable to DHRS and EVS Cooling System electrical equipment are as follows:

A. No single failure of an instrument, interconnecting cable or panel shall prevent a key process variable from being monitored.

B. DHRS valves shall be remotely operated and DHRS electrical equipment shall be controlled (see 5.6.2) from a panel in the Control Room to provide 1/2 hour start up capability.

C. Physical and electrical separation of redundant portions of DHRS (EVS cooling system, primary makeup pumps, instrumentation, and controls) shall be provided.

D. Electrical power supplied to electrical equipment shall be independent of off-site power.

E. Control instrumentation and electrical equipment shall function during and after an SSE.

F. Capability for periodic calibration and testing of electrical equipment shall be provided.

7.6.3.1.3 Equipment Design

As shown on Figure 5.1-7, the DHRS is part of the primary sodium processing, and the EVS Sodium Processing System. Description of the functioning of these systems for reactor decay heat removal is provided in Sections 9.1.3 and 9.3.2. The P&I diagrams are given in Figures 9.3-2 and 9.3-3. The EVST cooling system is described in Section 9.1.3 and the P&I diagram for the system is given in Figure 9.3-3.

DHRS and EVST cooling system electrical equipment meets the design criteria listed in Section 7.6.3.1.2 above in the following manner:

A. Control Systems

The following DHRS and EVST cooling system control functions are provided from separate, redundant control panels (local and main control room):

(1) Remote manual control of voltage to all NaK and sodium pumps.

(2) Remote manual control of ABHX dampers and fan speed.

(3) Remote manual override of pump and ABHX interlock circuits.

(4) Remote manual control of all valves required to provide DHRS and EVST cooling.

B. Monitoring Instrumentation

Some instrumentation required to monitor the functional performance of the decay heat removal process loops is redundant from the sensor out to and including the readout panel, so that a single failure of an instrument, interconnecting cable or panel does not prevent the process loop from being monitored. In those cases where a redundant sensor is not provided, separate indicators on separate panels are provided. Where redundant sensors are not provided, loss of the sensor does not prevent the acquisition of equivalent diagnostic information from other sensors on the process loop.

The following EVST cooling and DHRS process variables are monitored with completely redundant instrumentation (sensors, cabling, and panels):

*(1) EVST outlet sodium temperatures

* Required for post accident monitoring.

(2) Overflow vessel sodium level

*(3) Reactor overflow sodium inlet temperature

(4) EVST sodium pump control signal (each pump)

(5) EVST NaK pump control signal (each pump)

(6) Primary makeup pump control signal (each pump)

The following EVST cooling and DHRS process variables are monitored using a single sensor and redundant cabling and panels.

(1) EVST sodium flowrate (each loop)

(2) EVST NaK flowrate (each loop)

(3) Primary overflow makeup sodium flowrate (each pump outlet)

(4) EVST airblast heat exchanger fan speed setting (each loop)

(5) EVST NaK expansion tank level (each loop)

(6) EVST sodium inlet temperature (each loop)

(7) EVST sodium and NaK remotely operated valve position indicators (each loop)

(8) EVST airblast heat exchanger damper position indicator (each loop)

(9) DHRS heat exchanger bypass valve position indicator

(10) DHRS NaK expansion tank level

C. <u>Annunciation and Data Handling</u>

The following EVST cooling and DHRS process variables are connected to the plant annunciator system to alert the plant operator of off-normal conditions:

(1) Low sodium and NaK pump gas cooling flow rate (each pump)

(2) High EVST sodium, EVST NaK and primary makeup pump stator temperatures (each loop)

(3) Low EVST sodium, EVST NaK and primary makeup flowrate (each pump loop)

(4) High and low EVST sodium inlet temperature (each loop)

(5) High and low EVST NaK expansion tank level (each loop)

(6) High and low EVST sodium level

(7) High and low EVST sodium temperature

(8) Low sodium valve temperatures

(9) High and low DHRS expansion tank level

Key process variables that are connected to annunciators are also connected to the plant data handling and display system.

D. <u>Other Features</u>

Remotely operated valves in EVST cooling and DHRS circuits incorporate either "fail safe" or "fail in place" features and are provided with direct manual (reach rods on sodium valves) override capability in event of I&C or gas supply failures. DHRS valves are provided with accumulators to provide 1/2 hour startup capability for a period of 10 hours after the gas supply is lost.

Type 1E power is supplied to the equipment and instrumentation required to provide the safety-related functions of EVST cooling and DHRS as shown in Figure 7.6-11. This assures independence of off-site power.

Functional testing of all portion of DHRS that are not used during the course of normal operations will be tested on an annual basis during reactor refueling.

Equipment required to provide power to EVST and DHRS pumps, airblast heat exchangers and the monitoring instrumentation in the control panels shall be designed and tested to Seismic Category I requirements.

The DHRS control panel is configured to hi-light the DHRS coolant flow circuits with visual aids to assist the operator during DHRS initiation and operation.

E. <u>Separation of DHRS Equipment & Cables from SGAHRS</u>

The SGAHRS consists of three redundant systems, each system is powered from its respective Class 1E redundant power division 1, 2 or 3. The equipment and power supplies of the three divisions are physically and electrically independent such that the loss of any one division will not prevent the other divisions from performing their safety functions.

The DHRS equipment is powered from Class 1E, Divisions 1 and 2. The DHRS equipment of the two redundant divisions and their respective power supplies are physically and electrically independent.

In general, the cables, raceways and other equipment of the SGAHRS and DHRS are located in different areas of the plant, except in the Control Building and the Steam Generator Building where the cables of the same safety division of SGAHRS and DHRS share common raceways. However, the cables and raceways of redundant divisions are physically and electrically separated such that the loss of one safety division will not prevent the other divisions from performing their safety functions.

7.6.3.1.4 Initiating Circuits

Reactor decay heat removal through DHRS is initiated from the Control Room panel as described in Sections 9.1.3 and 9.3.2.

7.6.3.1.5 Bypass and Interlocks

When the DHRS is activated, automatic control of the EVST airblast heat exchangers is bypassed. All valves in this circuit are also operated on a direct or remote manual basis.

The flow in the primary sodium overflow makeup loop and EVST NaK loops, and the EVST airblast heat exchanger fan speed is set at maximum design rates.

The only interlocks remaining active in DHRS during this mode of operation are those associated with protection of the NaK and sodium pumps against high temperature in the pump stators. Manual override of this interlock can also be performed with the knowledge that pump damage and early failure could result.

7.6.3.2 Design Analysis

When DHRS is activated, all automatic controls are bypassed, the pumps and valves are remotely set to provide maximum flow through the DHRS loops, and the airblast heat exchangers are remotely set to provide maximum cooling capability. Control of the pumps and the airblast heat exchanger is provided from three separate locations: a field panel adjacent to or in a cell adjacent to the equipment, a local panel in same building as the equipment, and the control panel in the Main Control Room. The capability to provide power directly to the pumps, by bypassing all panel voltage and interlock control functions, is also provided so that no control function failure can keep DHRS electrical equipment from operating.

The EVST Cooling System is normally controlled from a local panel located in the Reactor Service Building. In the event of the loss of this local control, the EVST Cooling System equipment control is transferred to the Auxiliary Liquid Metal System panel located in the Main Control Room. All electrical equipment required for the functioning of the systems is classified as safety related and is qualified to IE requirements, and is provided with Class IE power supply, backed up by diesel generators to provide power during off-normal conditions.

7.6.4 Heating, Ventilating, and Air Conditioning (HVAC) Instrumentation and Control

The HVAC System provides heating and cooling to buildings, maintains pressures, temperatures, relative humidity and air purities within specified limits, and in conjunction with the Radiation Monitoring and Plant Protection Systems limits the release of airborne radioactive materials from the buildings.

The Instrumentation and Control System performs the following functions:

A. Provides control functions for the HVAC Systems in order to maintain the design operating and performance parameters during all system operating modes.

B. Provides monitoring of the HVAC System operating and performance parameters during all system operating modes.

C. Provides indication of failures in the HVAC System and/or deviations from the system design or performance parameters.

7.6.4.1 Design Criteria

The following design criteria are applicable to the safety-related HVAC Instrumentation and Control Systems:

A. Compliance with CRBRP General Design Criterion 11 as listed in Section 3.1.3.

B. Class 1E power supply, backed up by Diesel Generators, provide power to all safety-related HVAC System components.

C. No single failure of an instrument, interconnecting cable or panel will prevent a key process variable from being controlled or monitored in redundant divisions.

D. Physical and electrical separation of redundant portions of the HVAC System is provided.

E. Manual initiation of each protective action is provided at the system level.

F. Instrumentation used in the control of HVAC Systems will function during and after an SSE.

G. Instrumentation used in the control of HVAC Systems will function during normal environmental conditions and during environmental conditions created by any design basis accident.

H. Capabilities for periodic testing and calibration of all instruments are provided.

I. Capabilities for remote shutdown are provided, should the Control Room become uninhabitable.

J. Capabilities are provided to monitor the bypass or inoperable status of components in accordance with NRC Regulatory Guide 1.47.

K. Capabilities are provided to assess plant and environs conditions during and following an accident.

7.6.4.2 Design Description

Instrumentation and controls are provided for the safety-related HVAC Systems as described in this section. For a complete description, including process and instrumentation diagrams, refer to Section 9.6. Functional Control Diagrams for Heating, Ventilating and Air Conditioning Instrumentation and Control System are identified in Figures 7.6-12 through 7.6-22, and 7.3-3 through 7.3-24.

7.6.4.2.1 Control System

A. Fans

1. All fans in safety-related HVAC Systems are provided with remote manual controls in the Control Room.

2. All fans are provided with local manual controls on Local Panels.

3. Automatic control of fans is illustrated in the Functional Control Diagrams. Typically, fans will start automatically under any of the following conditions:

   i. Start demand from the Diesel Generator Load Sequencer will start fans;

   ii. Low air flow sensed in the discharge of a fan will start the redundant fan;

   iii. Starting of supply fans will start return fans;

   iv. Unit cooler fans will start when the safety equipment in the cell starts or when the temperature of the cell exceeds the setpoint;

   v. Diesel Generator emergency supply fans will start when the respective Diesel Generator starts.

B. Valves

o Containment supply and discharge valves will automatically close on a Containment Isolation Signal from the Plant Protection System.

C. Dampers

1. Dampers used to control a system parameter (i.e., temperature, pressure or flow) are energized by starting of the supply fan. Control of the damper is accomplished by measurement of the system parameter, comparison of the measurement to a required setpoint and transmission of a control signal by a controller, and modulation of the damper to maintain the required setpoint.

2. Dampers serving supply or exhaust fans are interlocked with the operation of the fans they serve and will open when fan starts and close when fan stops.

7.6.4.2.2 Monitoring Instrumentation

A. Typically, process variables are monitored and indication is provided in the Control Room and locally as follows:

1. All valve and damper positions.

2. All fan operation status.

3. Outside air temperature.

B. Typically, process variables are monitored and alarm annunciation is provided in the Control Room and locally as follows:

1. Motor thermal overload, high vibration or air flow low for each fan.

2. Unit cooler or HVAC unit supply air temperature high or air temperature entering cooling coil low.

3. Control switch in the local mode (Control Room only alarm only).

C. Typically, process variables are provided as inputs to the Plant Data Handling & Display System as follows:

1. Control Room and computer room humidity.

2. Containment differential pressure.

3. Annulus differential pressure.

4. RSB confinement differential pressure (four different cells).

5. Control Room differential pressure.

6. Air temperature entering and leaving each filter unit.

7. Air temperature entering and leaving each HVAC unit.

8. Cell temperature of each area being serviced by a unit cooler or HVAC unit.

9. Inoperable or bypass status of components.

D. The following process variables are classified as Accident Monitoring variables and are used to assess plant and environs conditions during and following an accident:

1. Annulus to atmosphere differential pressure.

2. RSB confinement to atmosphere differential pressure.

3. HVAC units discharge air temperature.

4. Filter units adsorbent filter leaving air temperature.

5. HVAC and filter units air flow low.

6. Damper and valve position indication.

7. Fan operation status indication.

7.6.4.3 Design Analysis

The HVAC Instrumentation and Control System is designed to perform the functions described in Section 7.6.4 while meeting the criteria listed in Section 7.6.4.1. All HVAC I&C circuits shall meet the requirements of Section 7.1 with the exception of alarm circuits and inputs to the PDH&DS which are Non-Class 1E circuits. The design of the HVAC Instrumentation and Control system is in conformance with the IEEE Standards listed in Table 7.6-2.

7.6.5 Steam Generator Building (SGB) Flooding Protection Subsystem

7.6.5.1 Design Basis

The SGB Flooding Protection Subsystem is provided to prevent flooding of SGAHRS equipment resulting from postulated SGS water/steam line ruptures, thereby assuring the availability of SGAHRS for reactor decay heat removal following water/steam line rupture events.

The SGB Flooding Protection Subsystem is designed to the IEEE Standards listed in Table 7.6-3.

7.6.5.2 Design Requirements

The SGB Flooding Protection Subsystem is designed to perform the following functions:

a) Detect the presence of large steam/water piping ruptures (see Section 15.3.3.1) by temperature and moisture sensors in each cell.

b) Detect water level flooding conditions in each cell by water level sensing elements.

c) Provide the signals to initiate the alarms and activate the equipment which provides the SGB flooding protection.

7.6.5.3 Design Description

7.6.5.3.1 Instrumentation

The SGB flooding protection instrumentation consist of temperature, moisture and water level instrument channels. The temperature and moisture instrument channels are Class 1E and the level channel is non-Class 1E.

For each cell in the SGB which contains steam and water piping three independent and redundant temperature and moisture instrumentation channels are provided. These signals are buffered and provided to two independent logic trains.

In addition, two water level instrumentation channels are provided in each cell.

7.6.5.3.2 Controls

The Flooding Protection Subsystem has a safety function and a non-safety function. The safety function is to detect a major pipe rupture and to isolate the feed water supply system and the affected loop. The non-safety function is to detect a small leak and annunciate in the main control room.

Upon detection of a major pipe rupture the startup and main feedwater control valves and the feedwater isolation valves are closed by two independent and separate class 1E logic trains. One logic train closes the startup and main feedwater control valves, the other the feedwater isolation valve.

Actuation of each logic train required concurrent two-out-of-three signals from both temperature and moisture from the same cell of any one of the four cells in each heat transport loop.

Small leaks are detected in each cell by measuring water level and by alarms on water level, temperature and moisture. Operator action is initiated upon annunciation in the main control room.

7.6.6 Recirculating Gas Cooling (RGC) Instrumentation and Control System

The RGC System provides heat removal capability and maintains the required cell temperature in the inerted cells where following components are located:

1. Sodium Makeup Pump and Vessel (Subsystem MA)

2. Sodium Makeup Pump and Pipeways (Subsystem MB)

3. Ex-Vessel Storage Loop 1 (Subsystem EA)

4. Ex-Vessel Storage Loop 2 (Subsystem EB)

As subsystems MA and MB cool two redundant sets of components, these subsystems are redundant trains for cooling sodium makeup systems, which supply forced circulation of gas through these systems. Similarly, for Ex-Vessel Storage Systems, subsystems EA and EB are redundant divisions.

The Instrumentation and Control System is provided for automatic control of the Recirculating Gas Cooling System, to monitor and indicate system process parameters during normal and off-normal conditions, and to provide signal inputs to Plant Data Handling and Display System.

7.6.6.1 Design Criteria

Design criteria that are applicable to safety-related subsystems MA, MB, EA and EB are as follows:

A. These subsystems are provided with Class 1E power supply and are backed up by Diesel Generators to provide power during loss of normal power.

B. No single failure of an instrument, interconnecting cable or panel will prevent a key process variable from being controlled or monitored in both redundant divisions.

C. Physical and electrical separation of redundant portions of these subsystems is provided.

D. System level manual initiation capabilities are provided in both the divisions.

E. Instrumentation used in control of these subsystems will function during and after an SSE.

F. Instrumentation used in the control of these subsystems will function during normal environmental conditions and during environmental conditions created by any design-basis accident.

G. Capabilities for periodic testing and calibration of all instruments are provided.

H. Capabilities are provided for remote shutdown, should the control room become uninhabitable.

I. Capabilities are provided to monitor the inoperable status of components in accordance with Reg. Guide 1.47.

J. Capabilities are provided to monitor process variables to assess plants and environs conditions during and following an accident.

7.6.6.2 Design

Instrumentation and controls are provided for the following equipment in RGC system: RGC fans, RGC subsystem isolation valves, subsystem cooler drain valves, and emergency chilled water isolation valves. For a complete description of the RGC system refer to Section 9.16.

7.6.6.2.1 Control System

7.6.6.2.1.1 Safety-Related Subsystem Operation

The subsystem components can be operated either from a back panel located in the control room or a local control panel. A transfer switch "local/remote" is located in the SSPLS (Solid State Programmable Logic System) cabinet for this purpose.

7.6.6.2.1.1.1 Fan Operation (Figure 7.6-41)

(1) The fan can be started either remotely from a pushbutton "start/stop" switch located on the back panel in the control room or locally from a pushbutton "start/stop" switch located on the local control panel provided the following conditions are satisfied:

(a) Absence of start inhibit from diesel generator load sequencer.

(b) Automatic isolation valves are open.

(2) The fan will start automatically when start deman signal is received from diesel generator sequencer and the Automatic Isolation valves are open.

(3) The fan can be stopped manually either from the local control panel or back panel depending on the Selector Switch position in the SSPLS cabinet.

(4) The fan will stop automatically under any of the following conditions:

(a) Either of the Automatic Isolation valves closes.

(b) High water vapor in the supply gas stream, provided no manual bypass for high water vapor is initiated.

(c) High water level in the cooler.

(d) Motor thermal overload.

The bypass identified in item (b) above is initiated by a Key Operated Selector Switch located on the Local Control Panel and administratively controlled.

(5) The fan start/stop indication is provided on the local control panel as well as on the back panel. "Fan stopped" is alarmed on the local control panel, and alarmed as "Fan trouble" in the computer. Bypass switch status is indicated on the local control panel. "Fan stopped" is alarmed as "Inoperable Status (IS)" in the computer located in the control room.

7.6.6.2.1.1.2 <u>Automatic Isolation Valve Operation</u> (Figures 7.6-39, 40 & 42)

The Automatic Isolation valves are fail-as-is valves.

(1) The Automatic Isolation valves can be operated from an "open-auto-close" switch (spring return to auto) located on the back panel and the local control panel.

(2) When the switch is in the "open" position, valves will open when all of the following conditions are satisfied:

(a) No high water level in the cooler.

(3) When the switch is in "auto" position, valves will open when fan start demand switch signal is received and all of the following conditions are satisfied:

(a) No high water level in the cooler.

(4) The valves can be closed manually by placing the switch in the close position.

(5) When the switch is in "auto" position, valves will automatically close under any of the following conditions:

(a) High water level in the cooler.

(b) The fan has stopped and there is no fan start demand.

(c) High temperature in return piping.

(6) The Automatic Isolation valve open/close indication is provided on the back panel and local control panel. Closure of the valve is alarmed as "Inoperable Status (IS)" in the computer located in the Control Room.

7.6.6.2.1.1.3 Drain Valve Operation (Figure 7.6-43)

The drain valves are fail open valves and locally operated.

(1) The drain valves can be opened/closed manually from a maintained pushbutton located on the local control panel.

(2) The drain valves will open automatically under any of the following:

(a) High water vapor in supply gas stream, provided no manual bypass for high water vapor is initiated.

(b) High water level in the cooler.

(3) The drain valves can be manually opened by pressing the "open" pushbutton.

(4) The drain valves can be manually closed by depressing the drain valve close pushbutton and when all of the following conditions are satisfied:

(a) No high water vapor in the supply gas stream.

(b) No high water level in the cooler.

(5) The drain valve open/close position is indicated on local control panel.

7.6.6.2.1.1.4 Chilled Water Valve Operation (Figure 7.6-44)

The chilled water isolation valves are fail open valves.

(1) The chilled water valves can be open/closed manually from momentary pushbuttons located on local control panel and back panel.

(2) The chilled water valves can be manually opened by depressing the chilled water valve open pushbutton and when all of the following conditions are satisfied:

(a) No high water vapor in the supply gas stream.

(b) No high water level in the cooler.

(3) The chilled water valves can be manually closed by pressing the "close" pushbutton.

(4) The chilled water valves will close automatically under any of the following conditions:

(a) High water vapor in the supply gas stream, provided no manual bypass for high water vapor is initiated.

(b) High water level in the cooler.

7.6.6.2.1.2 Safety-Related Subsystem EB

The operation of the Safety-Related Subsystem EB is the same as that of other safety-related subsystems (MA, MB, EA), however subsystem EB has a booster fan which is described below. The operation of the booster fan is not safety-related.

(1) The Booster Fan can be started manually from a "start/stop" pushbutton located on the local control panel.

(2) The Booster Fan will start automatically when main recirculating fan has started.

(3) The Booster Fan can be started manually by depressing the Booster Fan start pushbutton.

(4) The Booster Fan can be stopped manually be depressing the fan "stop" pushbutton.

(5) The Booster Fan will stop automatically under any of the following:

  (a) Fan of subsystem EB has stopped.

  (b) Booster Fan motor thermal overload.

(6) The Booster Fan start/stop status indication is provided on the local control panel. Booster Fan stopped is alarmed as "Booster Fan trouble" in the computer as well as group alarmed as "subsystem malfunction" on the back panel.

7.6.6.2.2 Monitoring Instrumentation

Indicators for water vapor in the gas and cooler level water are located in a local control panel (Figure 7.6-38). Indicating lights for the fans, the Automatic Isolation Valves, drain valves and emergency chilled water isolation valves are located both in a local control panel and the back panel (Figures 7.6-30 and 7.6-41), except the EB subsystem booster fan for which the indicating lights are on the local panel only.

Following process variables are monitored on local panels:

(a) High water vapor content.

(b) Cooler high water level (alarm only).

(c) Fan stopped (alarm only).

Safety-related subsystem malfunction is alarmed on the back panel when any of the following conditions is present (Figure 7.6-45):

(a) Cooler water level/water vapor content is high, provided no manual bypass for high water vapor is initiated.

(b) Cell environmental temperature is high.

(c) Cell exit gas temperature is high.

(d) Supply gas temperature is high.

(e) Gas pressure differential between supply and return is high.

(f) Fan vibration is high.

(g) Fan stopped.

(h) Fan motor thermal overload.

(i) Local control selected signal from SSPLS (Solid State Programmable Logic System).

Safety-related systems (MA, MB, EA, EB) malfunction alarms are grouped in two channels A and B and are provided on the main control panel (Figure 7.6-46).

Following process variables are monitored from control room:

(1) Water vapor or cooler water level (alarm only).

(2) Exit gas temperature (exit from cells) (alarm only).

(3) Primary heat transport system cells environment temperatures (indication only).

(4) Indicating lights for fans and automatic isolation valves.

(5) Supply and return gas temperatures (alarms only).

Process variables identified with items 3, 4 and 5 are also designated as accident monitoring variables to assess plant and environs conditions during and following an accident. Refer to Section 7.5.11 of PSAR for detailed requirements on post accident monitoring.

7.6.6.2.3 Inputs to PDH&DS

The following process variables are provided as inputs to Plant Data Handling and Display System (Non-Safety System):

A. High moisture in supply gas line.

B. Supply gas temperature.

C. Return gas temperature.

D. Gas differential pressure.

E. Fan stop status.

Inoperable status of subsystem fans (MA, MB, EA, EB) and isolation valves (two per subsystem) is also monitored through Inoperable Status Monitoring System.

7.6.6.2.4 <u>Design Analysis</u>

Refer to PSAR Section 7.1.2 for conformance to applicable IEEE Standards. RGC system is designed to operate automatically. The system and its safety-related subsystems are operated during normal as well as emergency conditions. The RGC system components are cascaded to operate in sequence. Starting a fan will open associated supply and return gas isolation valves. A subsystem will not operate when high water vapor or cooler high water level or electrical fault exists.

As discussed in Section 9.16 each subsystem of RGCs supplies cooling to redundant components, so no additional redundancy is provided in its components and instrumentation.

The systems are designed for fail safe operation and control equipment will assume a failed position consistent with its intended safety function.

The coolant supply to safety-related subsystems MA, MB, EA, EB is provided by Emergency Chilled Water System. The fan motors for these subsystems are provided with AC power from Class 1E power sources to continue operating during loss of offsite power, except for the booster fan of the subsystem EB which is not required to operate during loss of power condition. Subsystems MA and EA and the EM pumps cooled by these two subsystems are served by Class 1E power supply Division 1. Also, subsystems MA and EA are served by Emergency Chilled Water Loop "A". Subsystems MB and EB are served by Emergency Chilled Water Loop "B", and Class 1E power supply Division 2. The EM pumps cooled by subsystems MB and EB are also connected to Class 1E power supply Division 2. Automatic isolation valves are designed as fail open valves so as to be in their safety position upon loss of power.

Fan and Isolation Valve control switches are located in the local panels as well as in the back panels for subsystems MA, MB, EA and EB, except for booster fan. Thus, in case of control room evacuation the fans and valves can be controlled from outside the control rooms, using local panels.

The design of the Recirculating Gas System is in conformance with the IEEE Standards listed in Table 7.6-2.

TABLE 7.6-1

SYMBOLS

ALARM

INOPERABLE STATUS MONITORING

RED IND LITE

GREEN IND LITE

WHITE IND LITE

COMPUTER INPUT

NOTES:

1) Control switches are spring return to auto from start with a maintained stop unless otherwise stated.

ABBREVIATIONS

SSPLS	-	Solid State Programmable Logic System
CR	-	Control room (remote)
L	-	Local (not control room)
T.D.	-	Time delay
N.C.	-	Normally closed
F.C.	-	Fail closed
S.O.V.	-	Solenoid operated valve
A.O.V.	-	Air operated valve
MOD	-	Motor operated damper
ZS	-	Position switch
CIS	-	Containment isolation signal
PPS	-	Plant Protection System
E/H	-	Electro-hydraulic

TABLE 7.6-1 (Continued)

TE	-	Temp element
TT	-	Temp transmitter
TIC	-	Temp ind controller
OAI	-	Outside air intake
TMD	-	Temp modulated damper
RA	-	Return air
PDI	-	Pressure differential indicator
PDC	-	Pressure differential controller
PMD	-	Pressure modulated damper
PDISH	-	Pressure differential indicating switch high
FR	-	Flow recorder
FIC	-	Flow indicating controller
FSL	-	Flow switch low
FT	-	Flow transmitter
FMD	-	Flow modulated damper
FE	-	Flow element
M	-	Moisture
PB	-	Pushbutton
MUX	-	Multiplexing
AHU	-	Air handling unit

TABLE 7.6-2

LIST OF IEEE STANDARDS APPLICABLE TO EMERGENCY
PLANT SERVICE WATER, EMERGENCY CHILLED WATER, HVAC,
AND RECIRCULATING GAS INSTRUMENTATION AND CONTROL SYSTEM

a) IEEE Standard 279-1971

IEEE Standard: Criteria for Protection Systems for Nuclear Power Generating Stations

b) IEEE Standard 308-1974

Criteria for Class 1E Power Systems for Nuclear Power Generating Stations

c) IEEE Standard 323-1974

Qualifying Class 1E Electrical Equipment for Nuclear Power Generating Stations

d) IEEE Standard 338-1977

Criteria for Periodic Testing of Nuclear Power Generating Station Safety Systems

e) IEEE Standard 379-1972

IEEE Trial-Use Guide for the Applicability of the Single-Failure Criterion to Nuclear Power Generating Station Protection Systems

f) IEEE Standard 383-1974

Standard for Type Test of Class 1E Electric Cables, Field Splices and Connections for Nuclear Power Generating Stations

g) IEEE Standard 384-1974

IEEE Trial-Use Standard Criteria for Separation of Class 1E Equipment and Circuits.

TABLE 7.6-3

LIST OF IEEE STANDARDS APPLICABLE TO SGB FLOODING PROTECTION SUBSYSTEM

IEEE-279-1971	IEEE Standard: Criteria for Protection Systems for Nuclear Power Generating Stations
IEEE-323-1974	IEEE Trial-Use Standard: General Guide for Qualifying Class 1E Electric Equipment for Nuclear Power Generating Stations
IEEE-323-A-1975	Supplement to the Foreword of IEEE-323-1974
IEEE-336-1971	IEEE Standard: Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During Construction of Nuclear Power Generating Stations
IEEE-338-1971	IEEE Trial-Use Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems
IEEE-344-1975	IEEE Standard 344-1975, IEEE Recommended Practices for Seismic Qualification of Class 1 Equipment for Nuclear Power Generating Stations
IEEE-352-1972	IEEE Trial-Use Guide: General Principles for Reliability Analysis of Nuclear Power Generating Station Protection Systems
IEEE-379-1972	IEEE Trial-Use Guide for the Application of the Single-Failure Criterio to Nuclear Power Generating Station Protection Systems
IEEE-384-1974	IEEE Trial-Use Standard Criteria for Separation of Class 1E Equipment and Circuits
IEEE-494-1974	IEEE Standard Method for Identification of Documents Related to Class 1E Equipment and Systems for Nuclear Power Generating Station

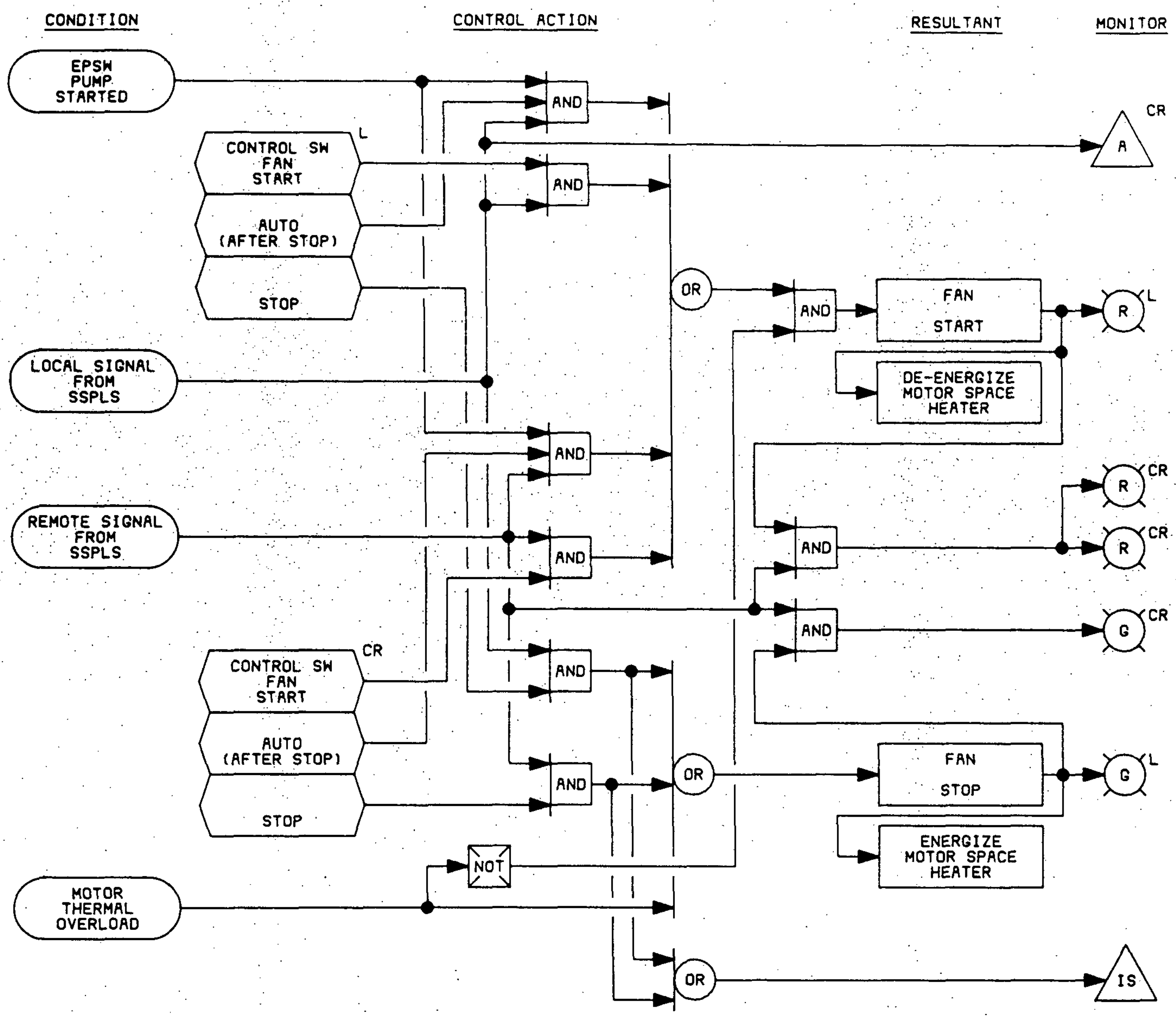


FIGURE 7.6-1
EMERGENCY COOLING TOWER FAN

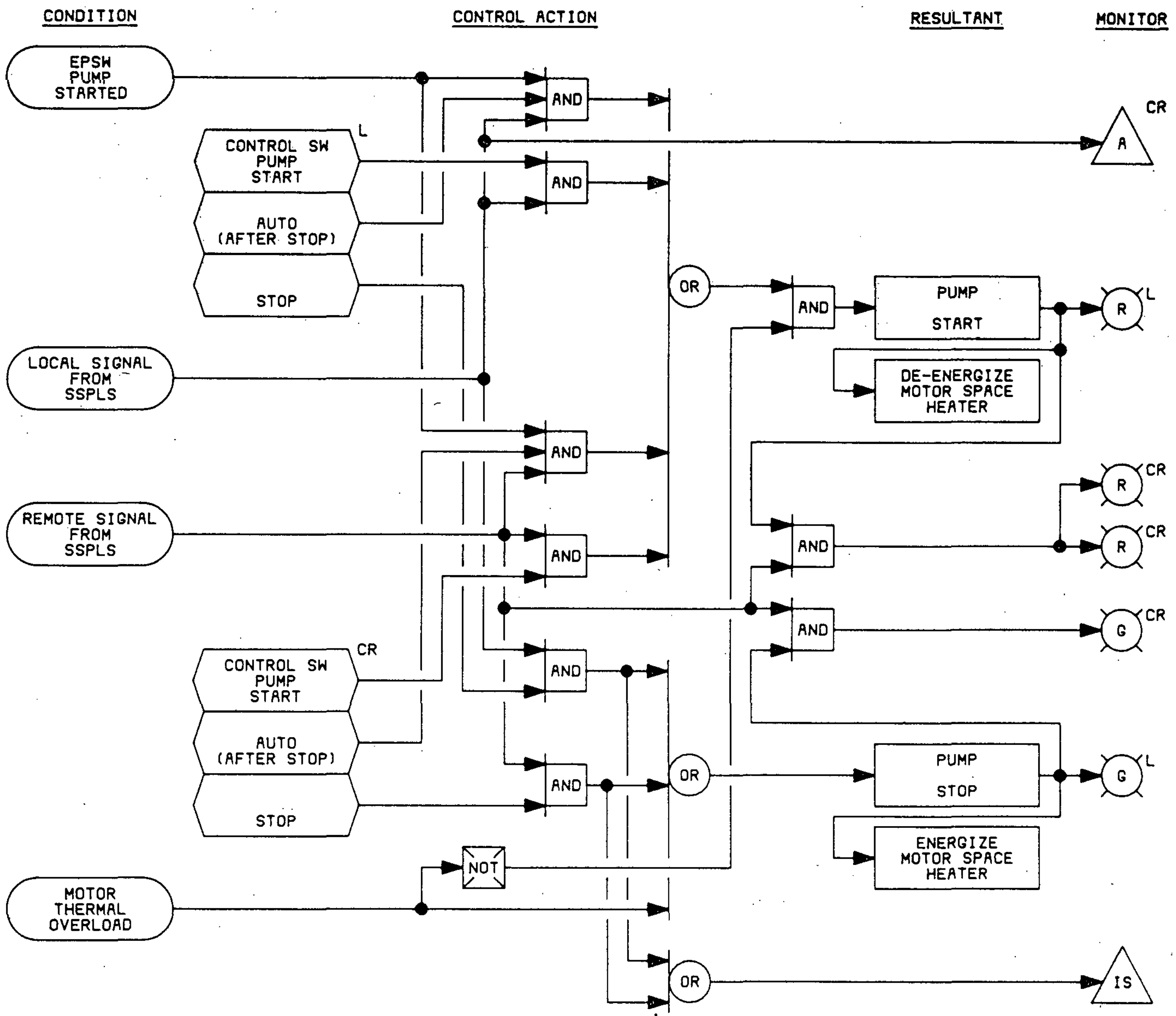


FIGURE 7.6-2
EMERGENCY PLANT SERVICE WATER MAKEUP PUMP

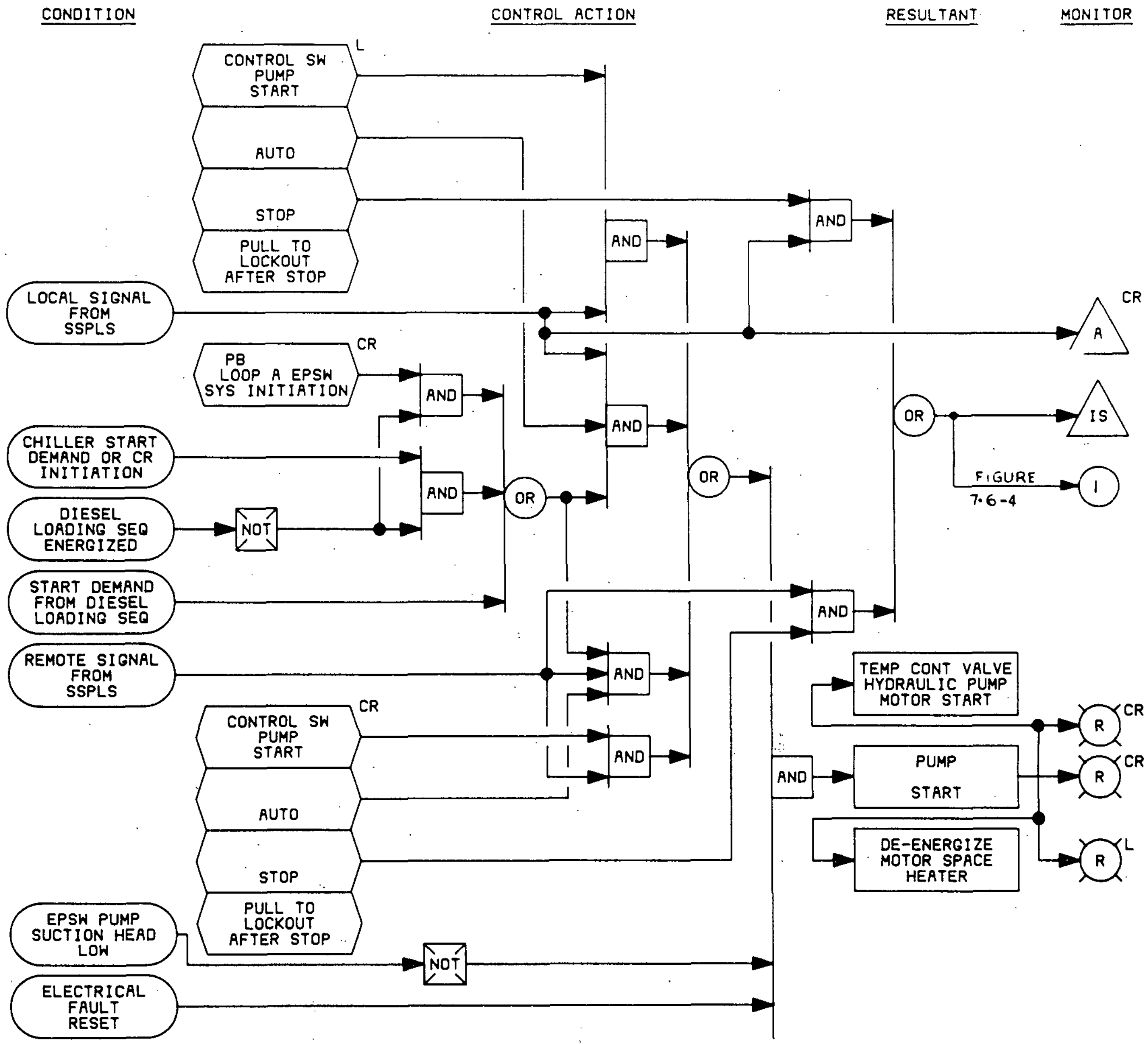


FIGURE 7.6-3
EMERGENCY PLANT SERVICE WATER PUMP START

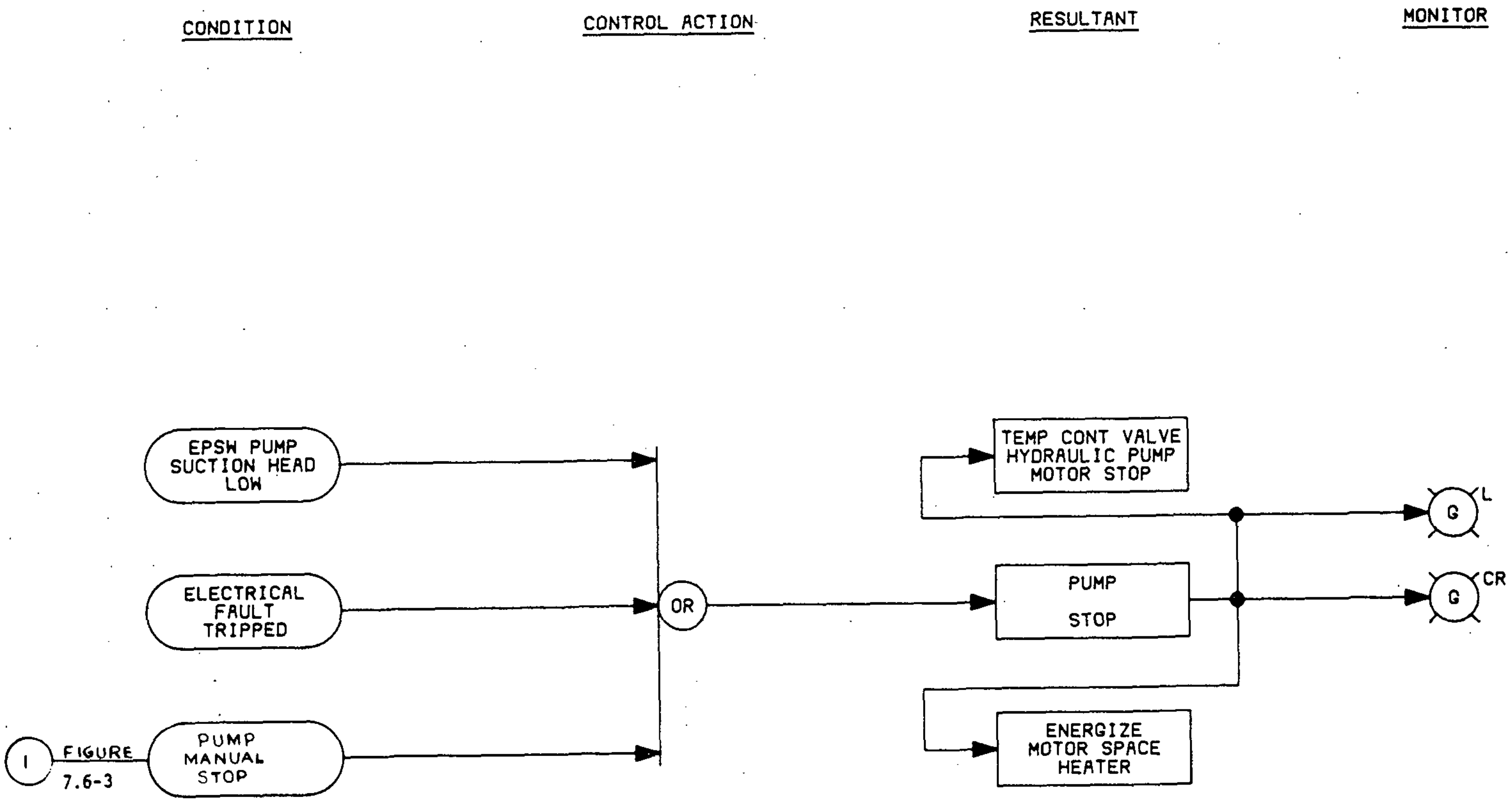


FIGURE 7.6-4
EMERGENCY PLANT SERVICE WATER PUMP STOP

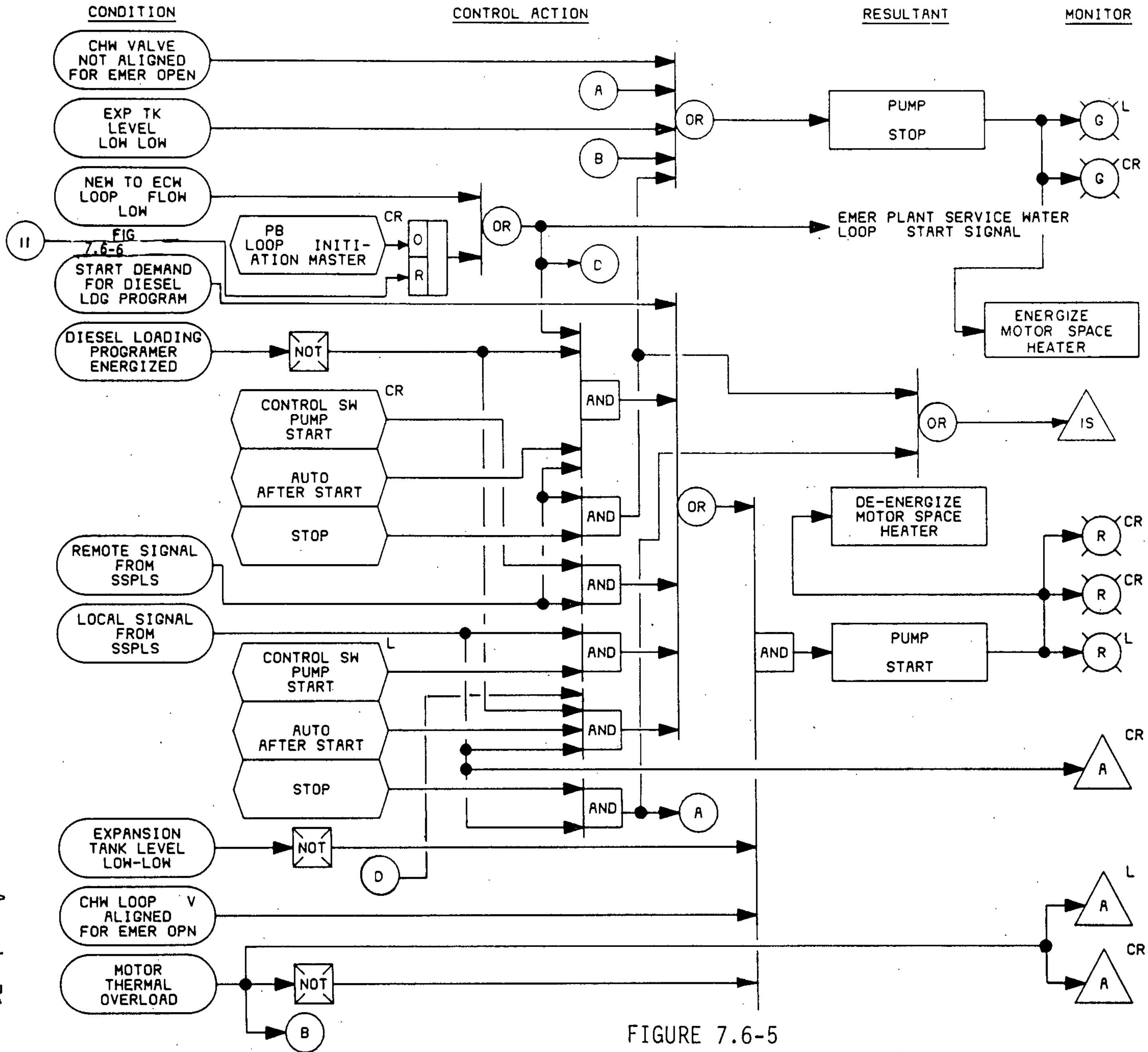


FIGURE 7.6-5
EMERGENCY CHILLED WATER PUMPS LOGIC

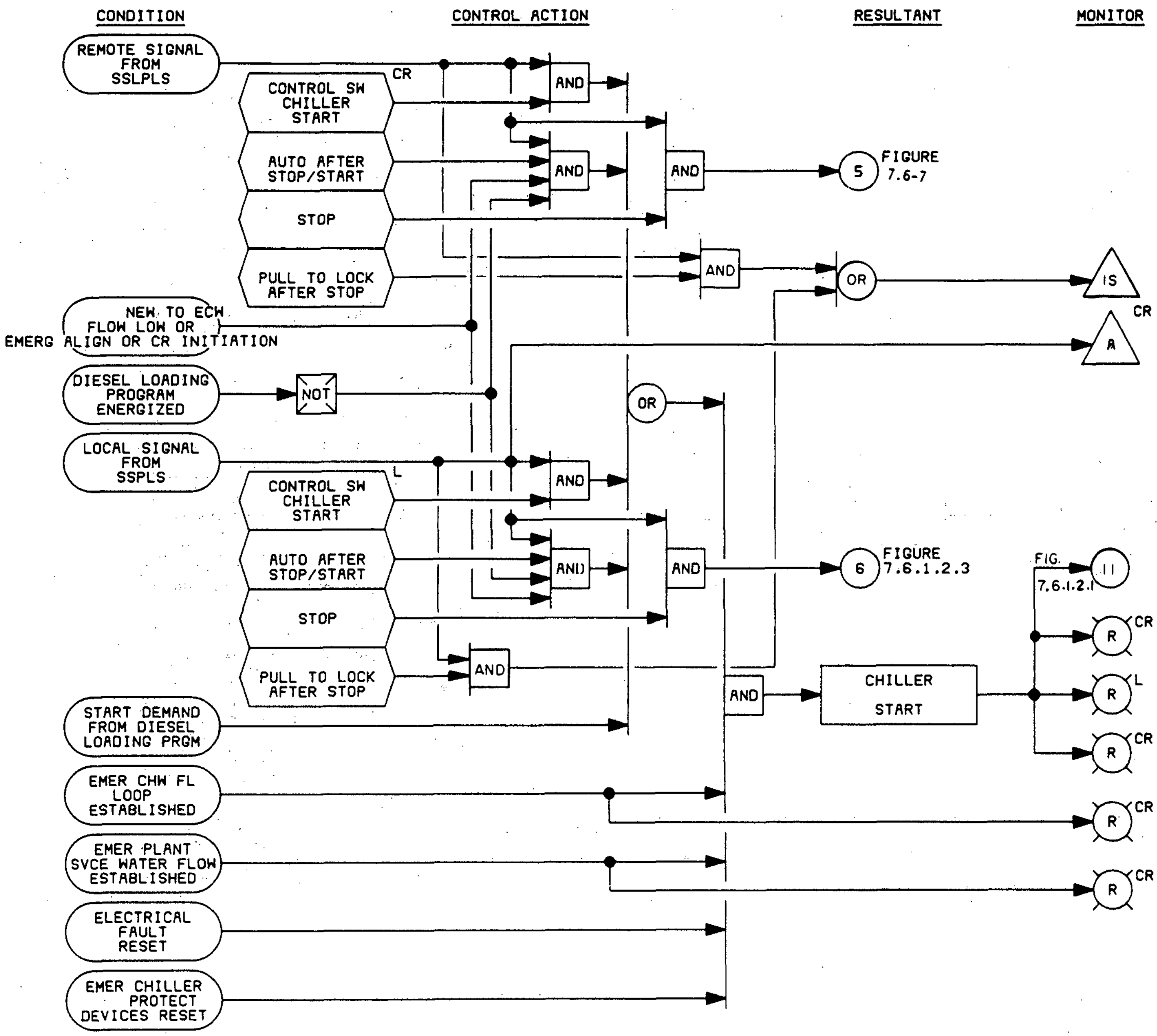


FIGURE 7.6-6
EMERGENCY CHILLED WATER CHILLER START

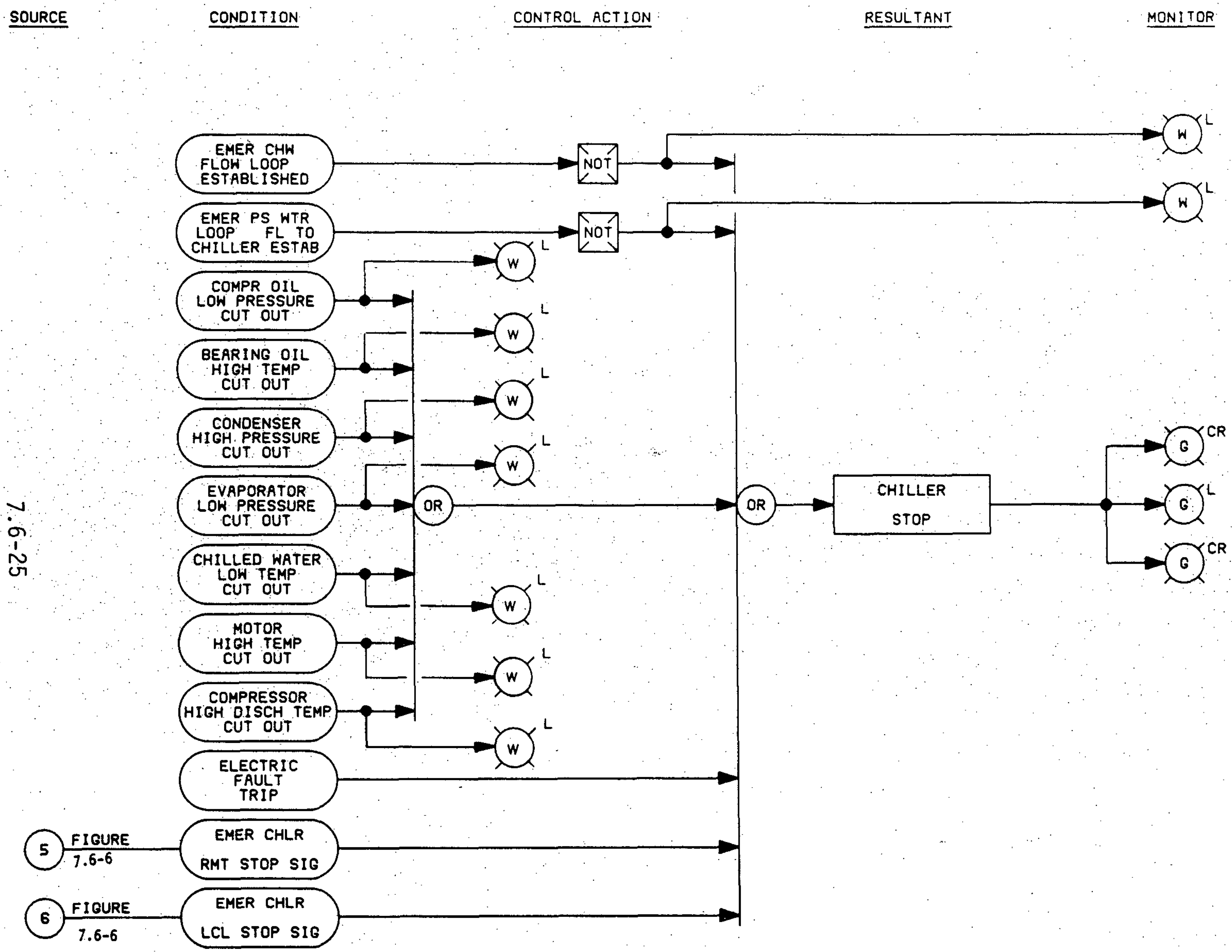


FIGURE 7.6-7
EMERGENCY CHILLED WATER CHILLERS STOP

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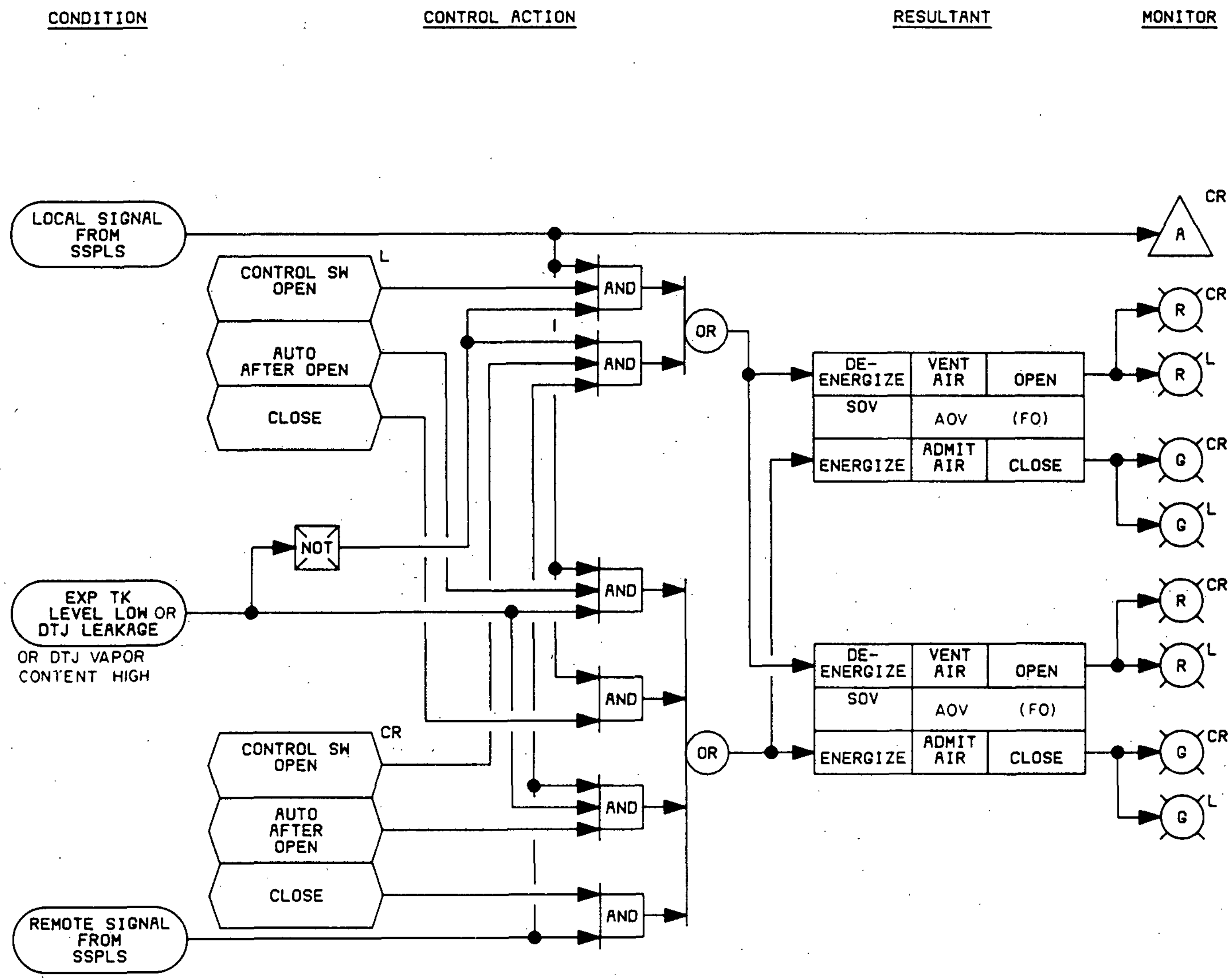


FIGURE 7.6-8
EMERGENCY CHILLED WATER ISOLATION VALVES
TO SECONDARY COOLANT LOOP

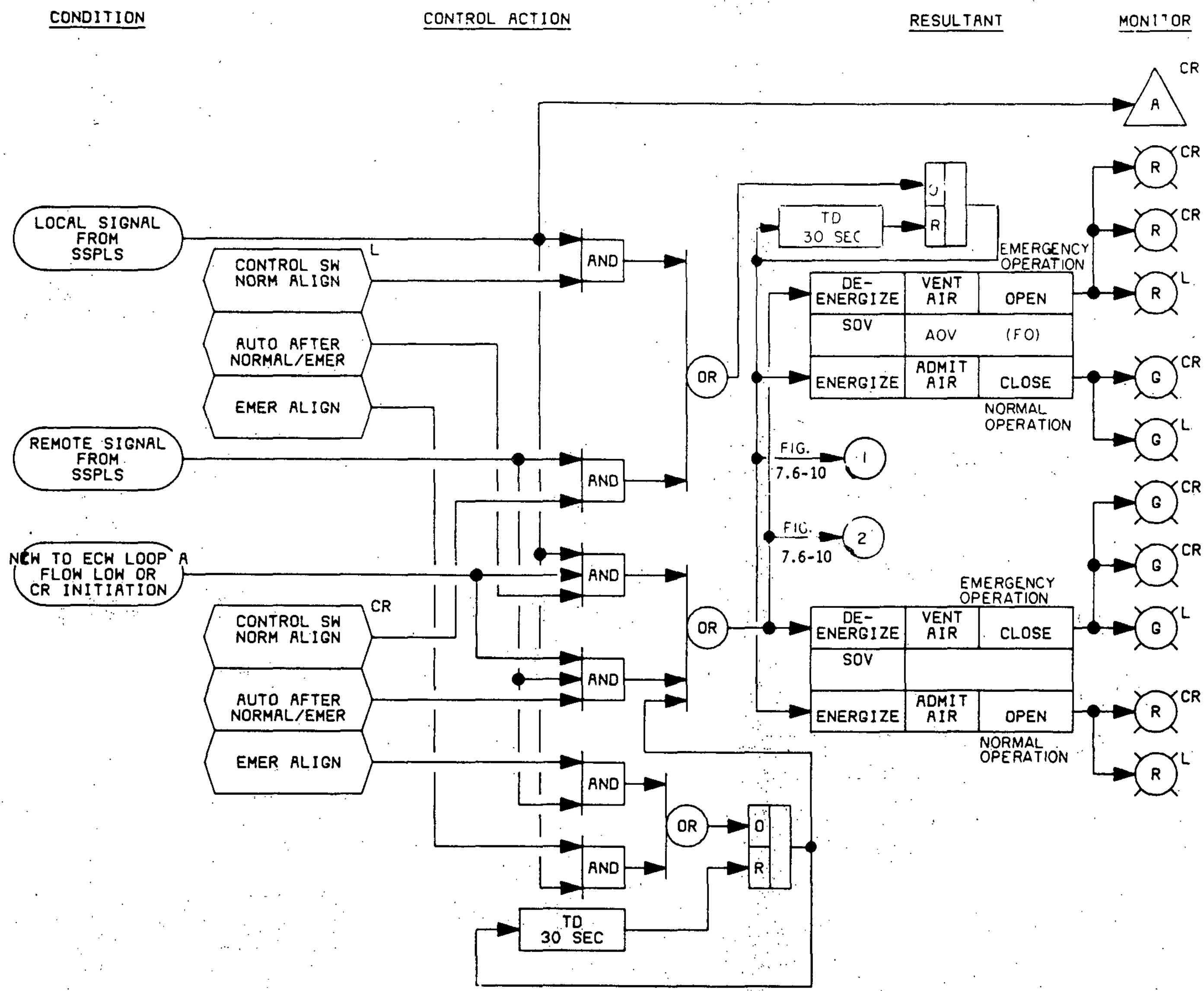


FIGURE 7.6-9
EMERGENCY CHILLED WATER SYSTEM NCW TO ECW ISOLATION VALVES

7.6-27

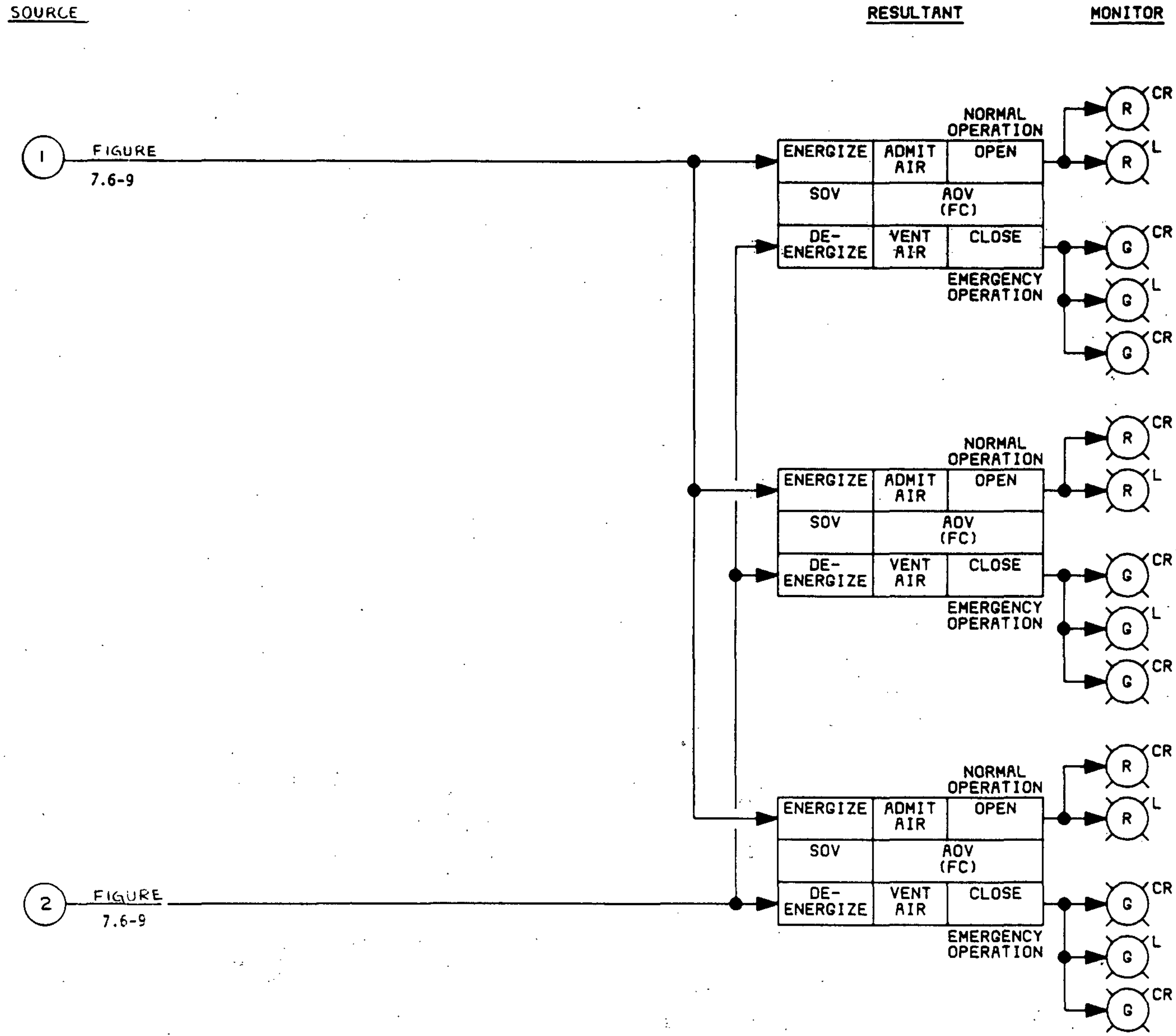
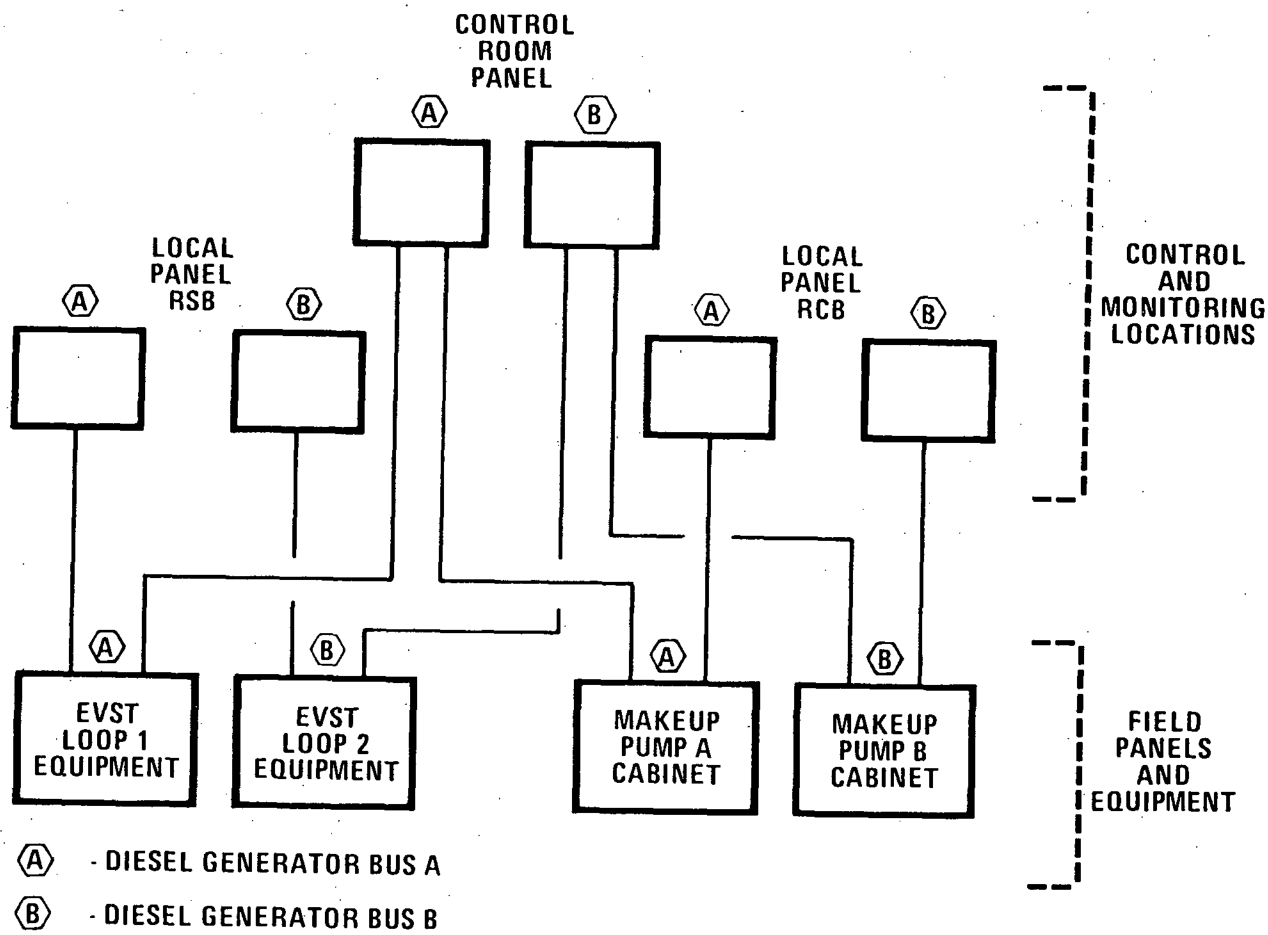


FIGURE 7.6-10
EMERGENCY CHILLED WATER SYSTEM LOOP A &
LOOP B AOV'S NORMAL & EMERGENCY OPERATION

FIGURE 7.6-11

SAFETY CLASS EQUIPMENT VITAL BUS HOOKUP



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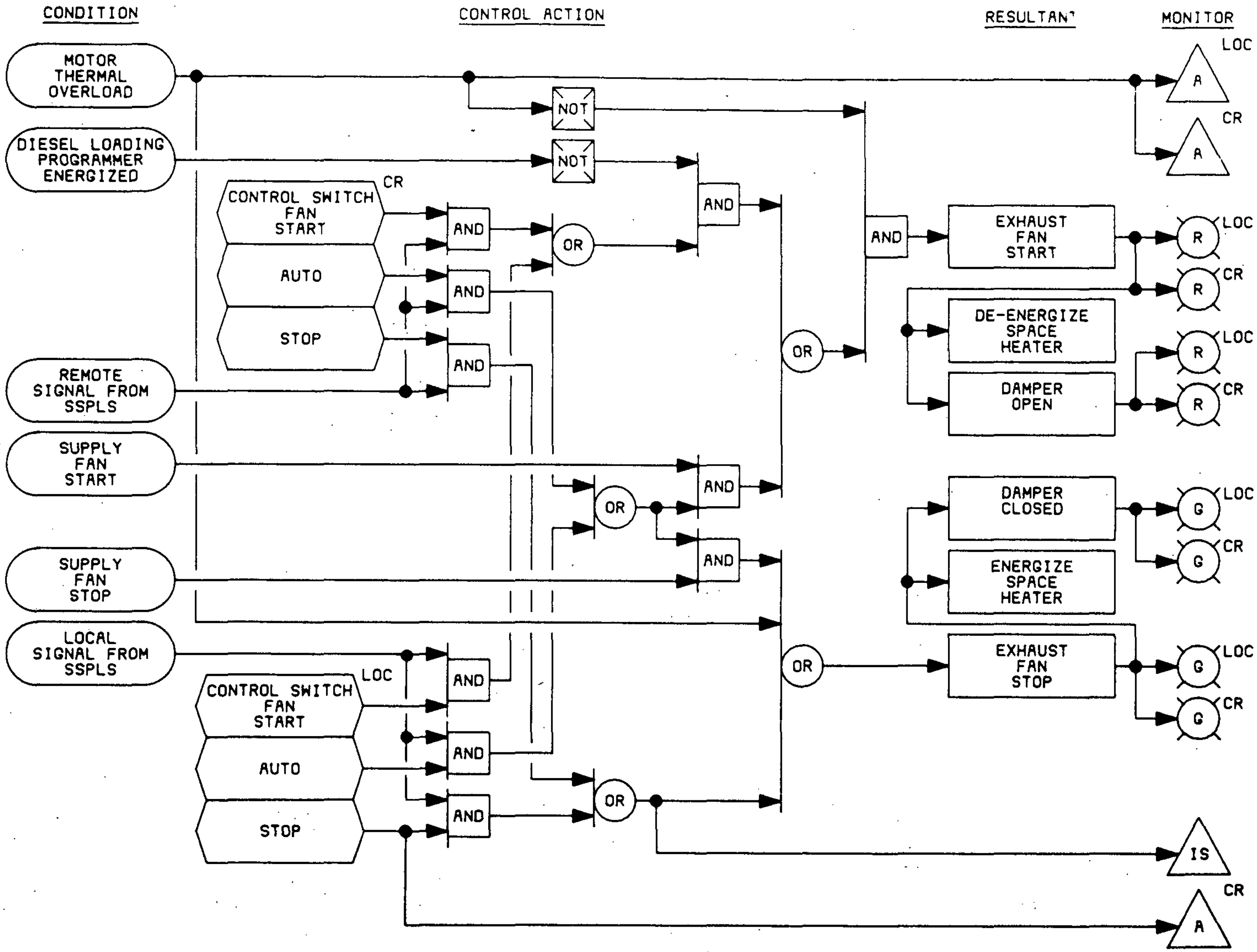

FIGURE 7.6-12
FUNCTIONAL CONTROL DIAGRAM TYPICAL HVAC
EXHAUST FAN

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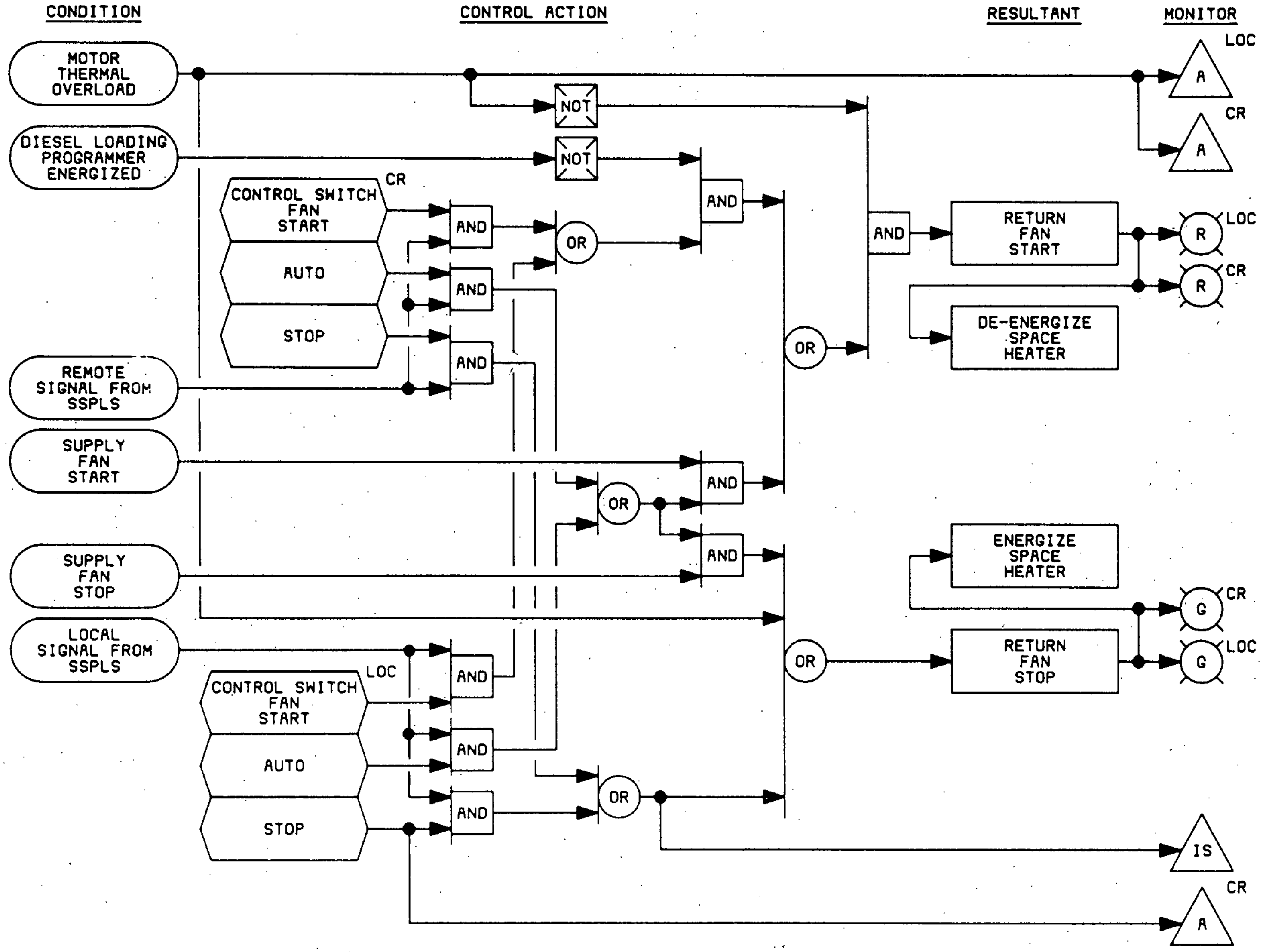

FIGURE 7.6-13
FUNCTIONAL CONTROL DIAGRAM TYPICAL HVAC
UNIT RETURN FAN

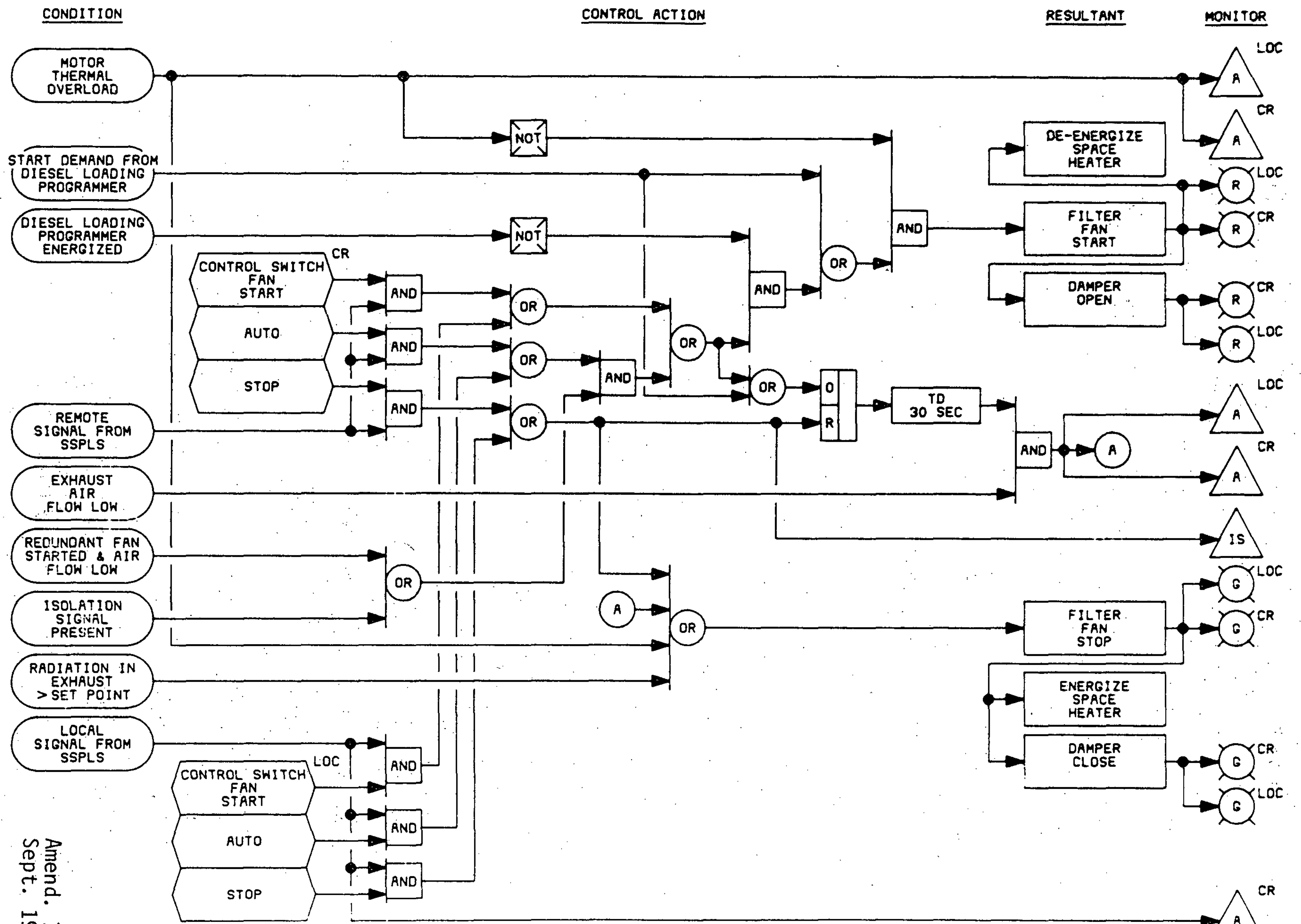

FIGURE 7.6-14
FUNCTIONAL CONTROL DIAGRAM TYPICAL FILTER
UNIT SUPPLY FAN

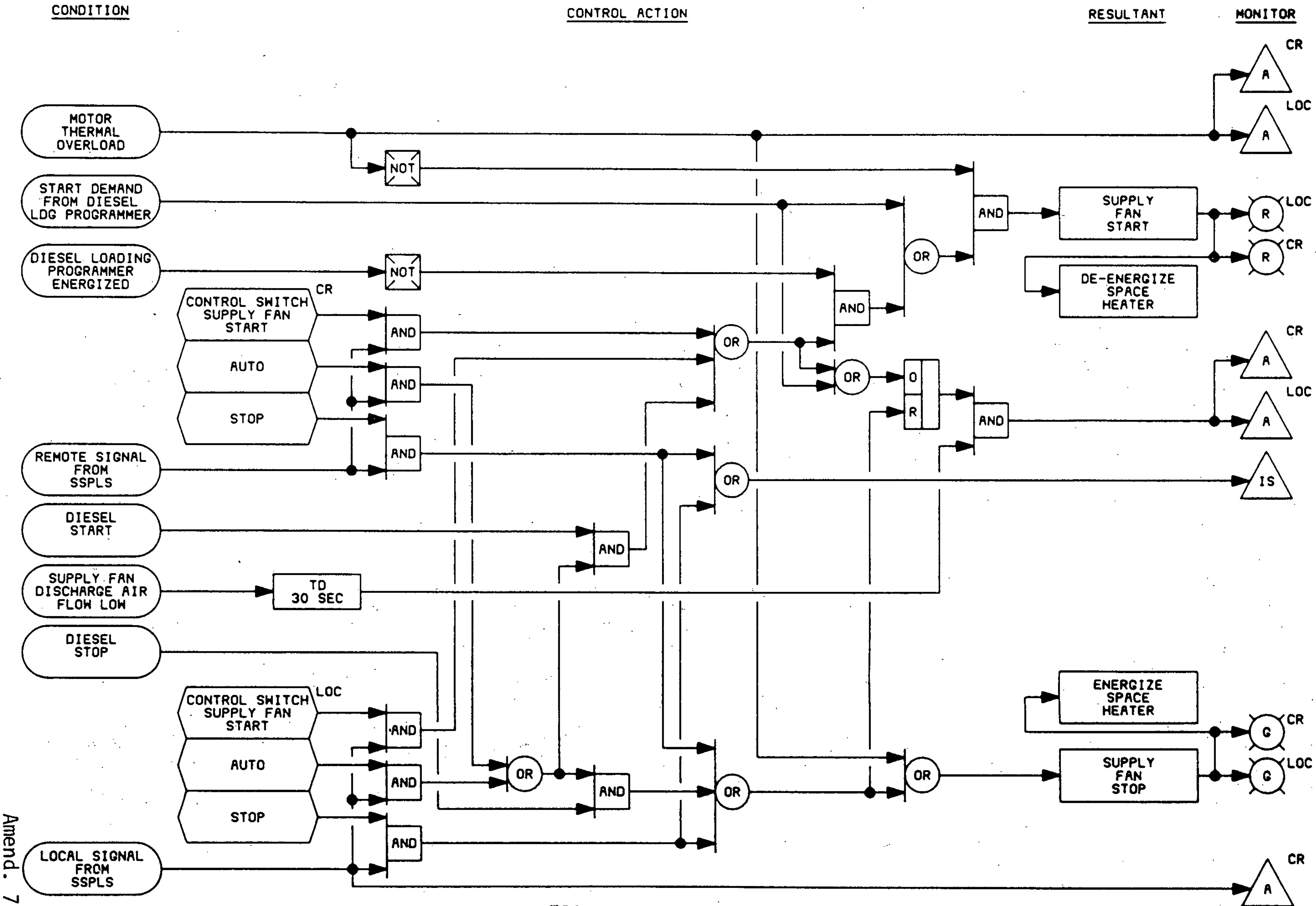

FIGURE 7.6-15
FUNCTIONAL CONTROL DIAGRAM DIESEL ROOM EMERGENCY SUPPLY FAN

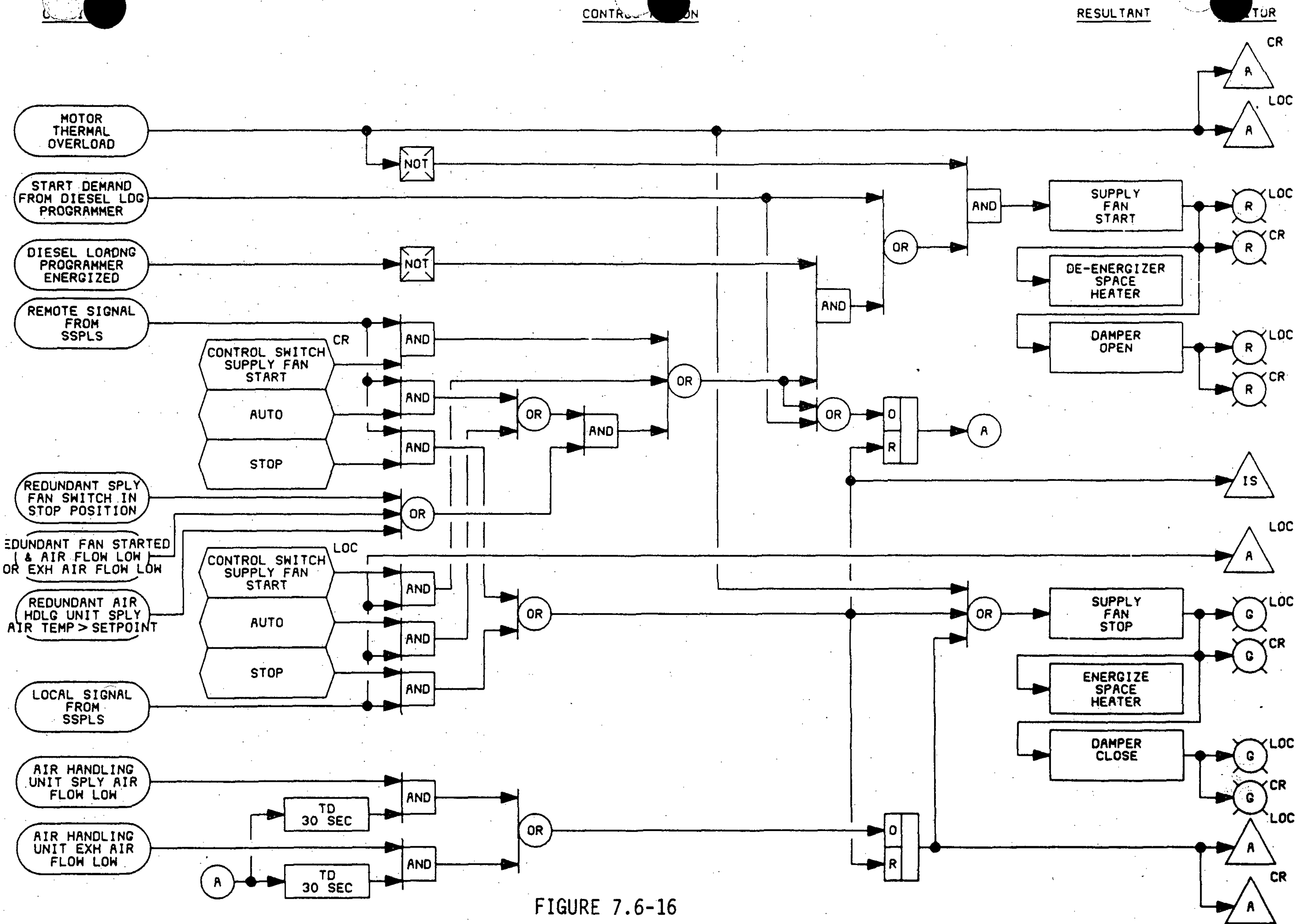

FIGURE 7.6-16

FUNCTIONAL CONTROL DIAGRAM 1 OF 2
REDUNDANT SUPPLY FANS FOR SGB-IB AIR
HANDLING UNIT

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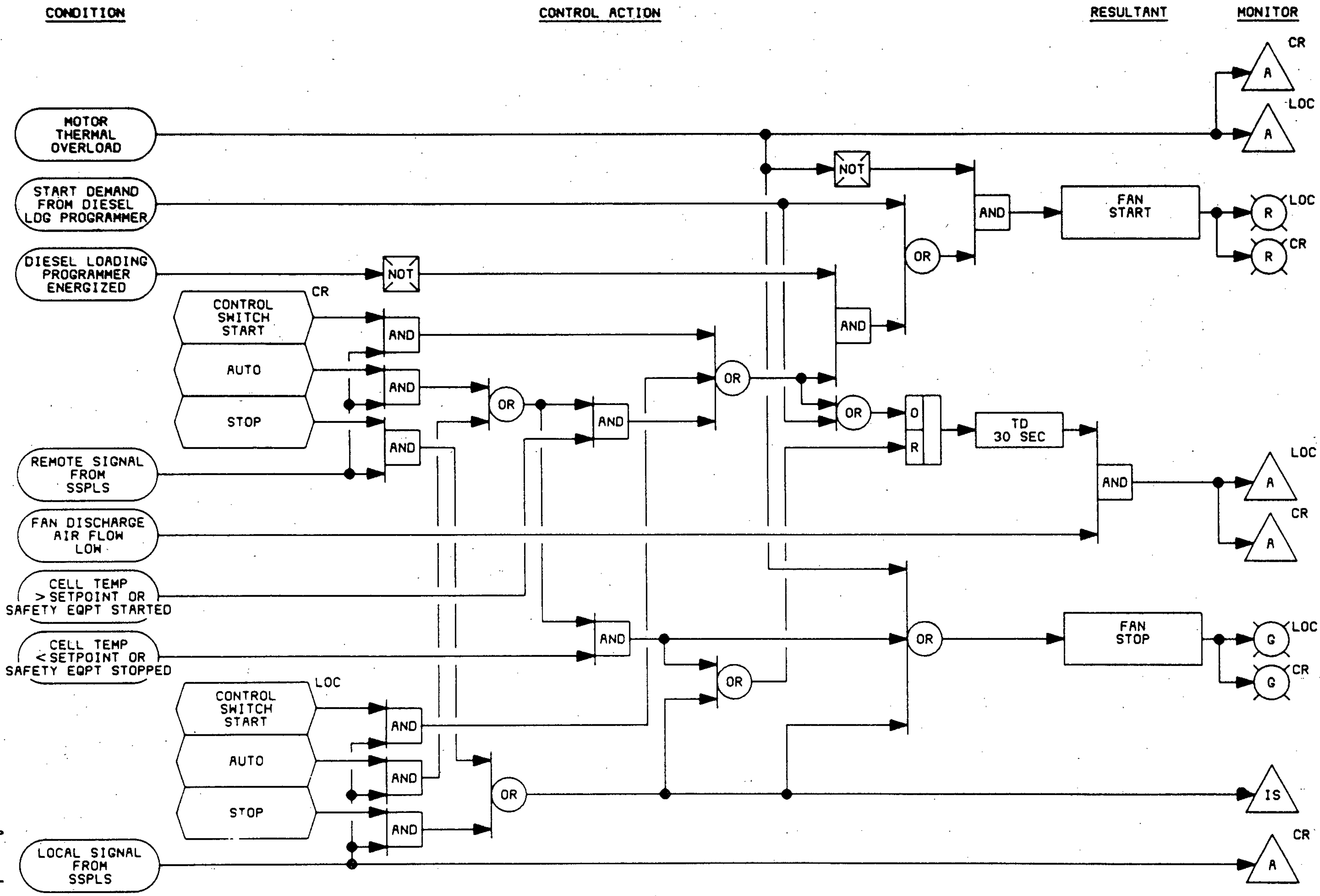


FIGURE 7.6-17

FUNCTIONAL CONTROL DIAGRAM TYPICAL UNIT COOLER SERVING CELL CONTAINING SAFETY RELATED EQUIPMENT

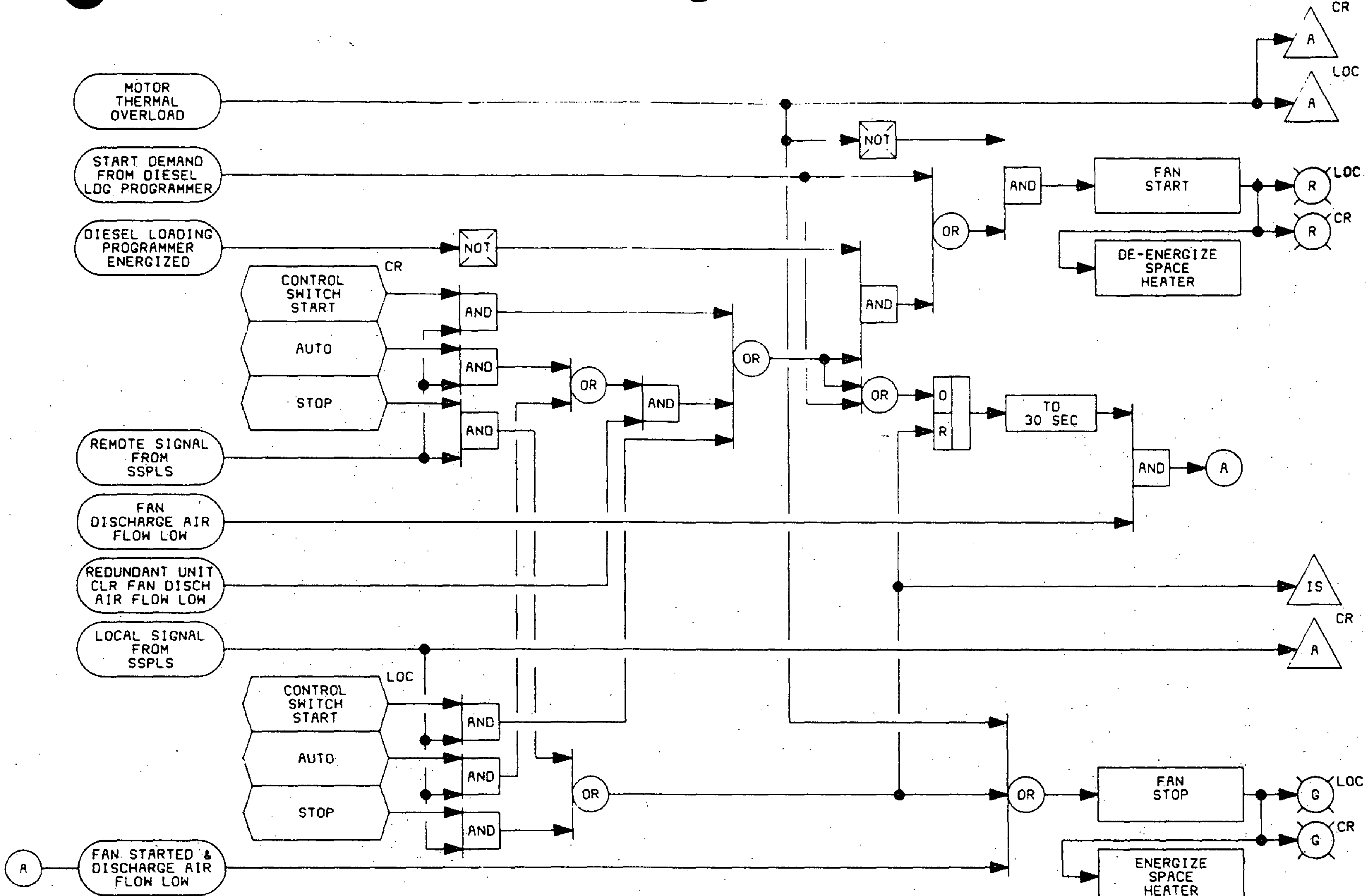


FIGURE 7.6-18

UNCTIONAL CONTROL DIAGRAM TYPICAL UNIT COOLER FAN SERVING CELL CONTAINING SAFETY RELATED EQUIPMENT WHERE REDUNDANT COOLERS ARE REQUIRED

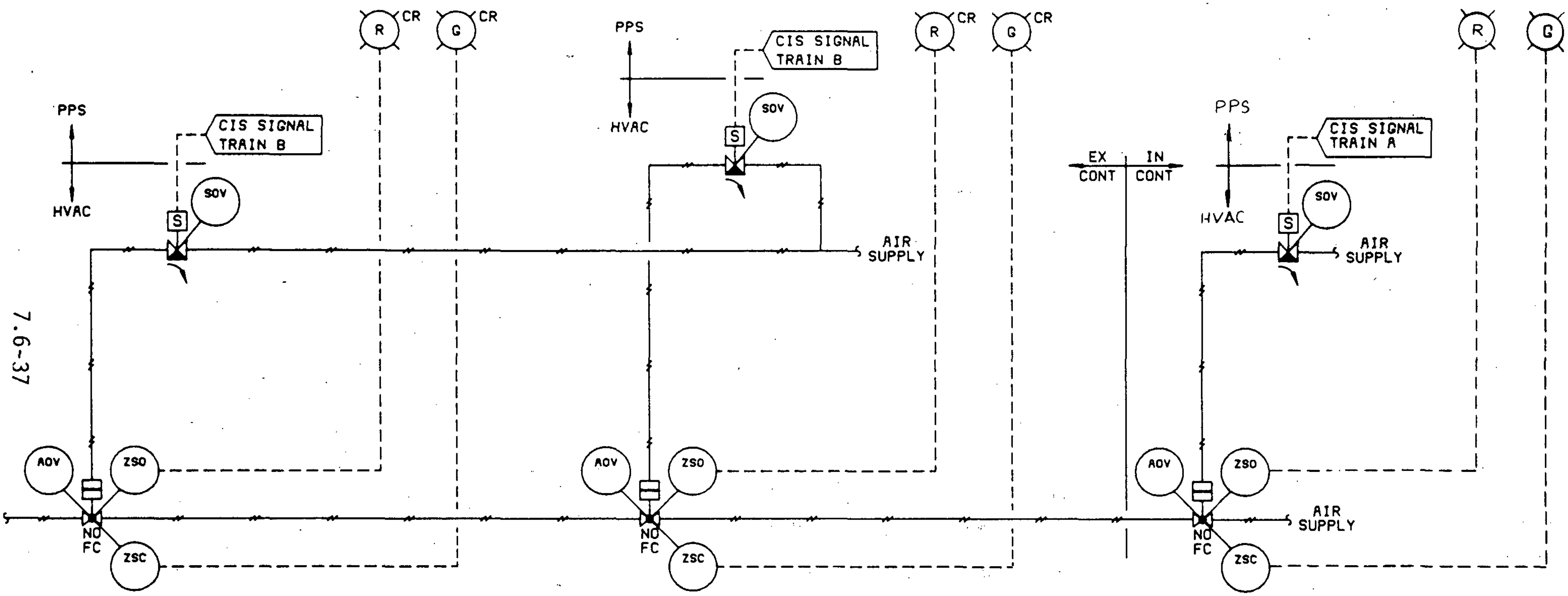
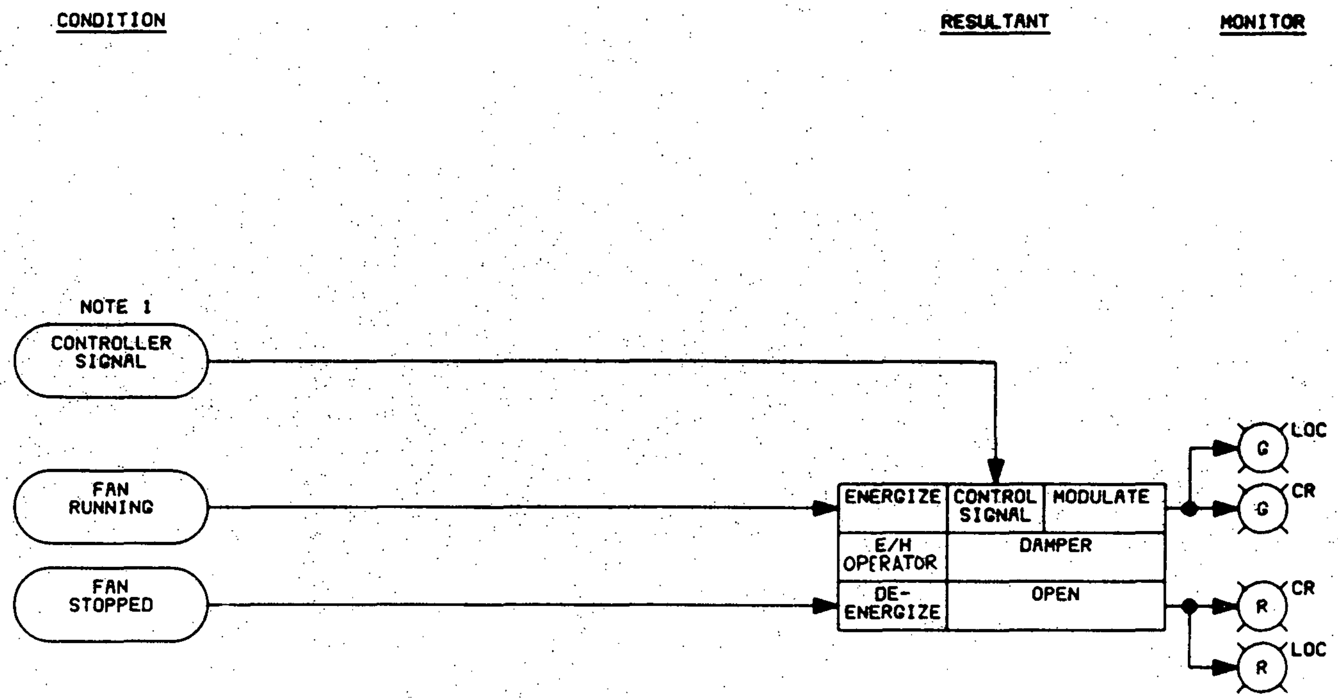


FIGURE 7.6-19

FUNCTIONAL CONTROL DIAGRAM RCB SUPPLY & EXHAUST CONTAINMENT ISOLATION VALVES



NOTES

1. DAMPERS MAYBE MODULATED FROM PRESSURE,FLOW OR TEMPERATURE SIGNAL. REFER TO FIGURES 7.6-35 THROUGH 7.6-37

FIGURE 7.6-20
FUNCTIONAL CONTROL DIAGRAM TYPICAL
PROCESS PARAMETER CONTROL OF DAMPER

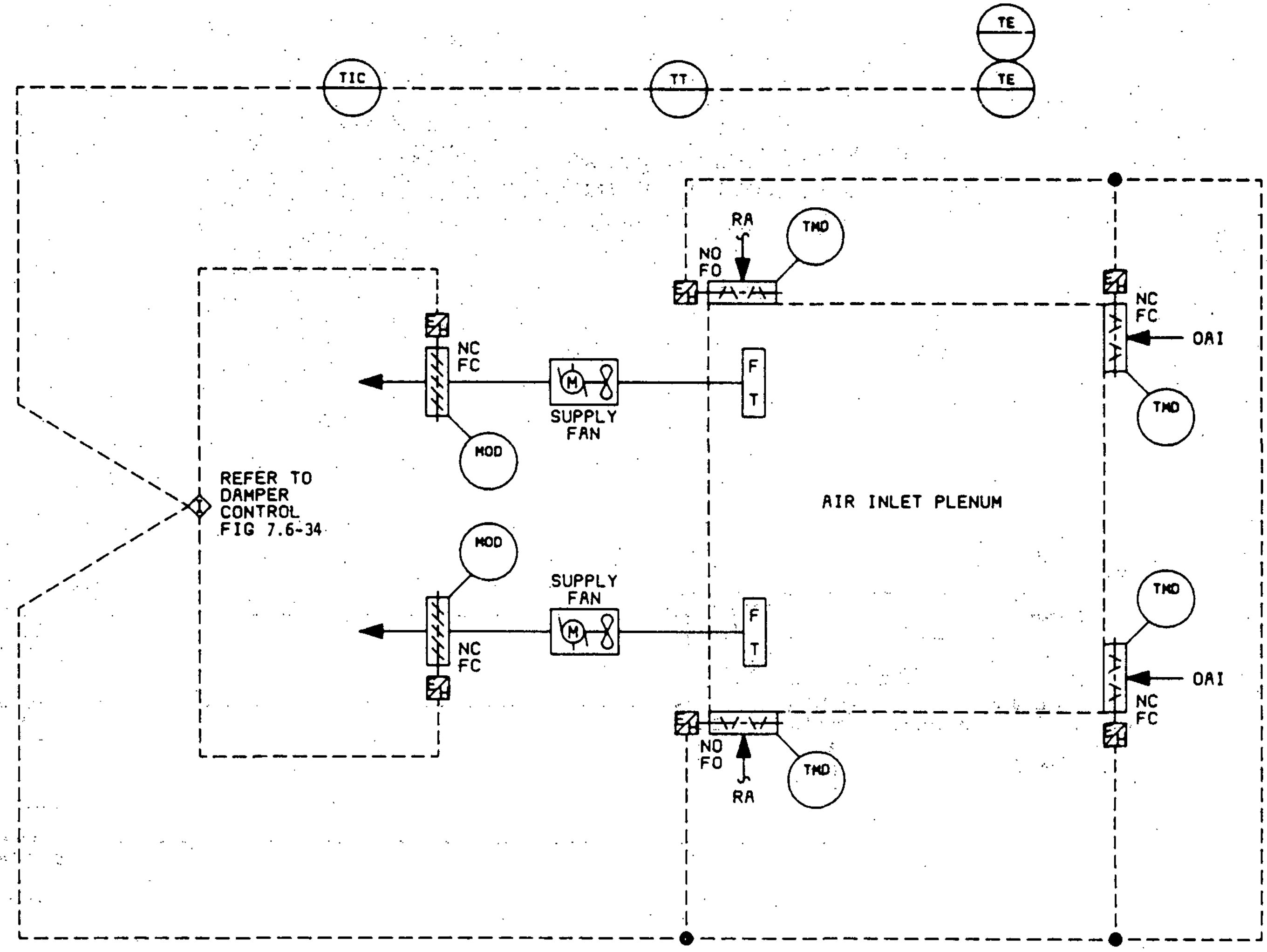


FIGURE 7.6-21
FUNCTIONAL CONTROL DIAGRAM DIESEL
GENERATOR EMERGENCY SUPPLY FAN
TEMPERATURE MODULATED DAMPERS

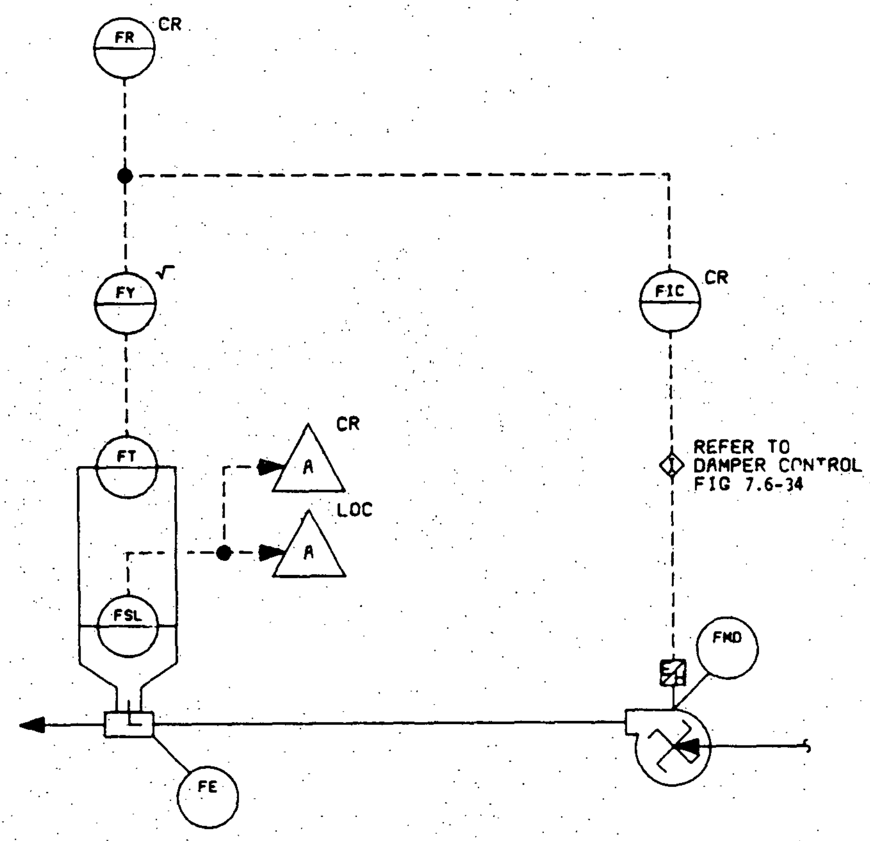


FIGURE 7.6-22
FUNCTIONAL CONTROL DIAGRAM TYPICAL FLOW
MODULATED VORTEX DAMPER

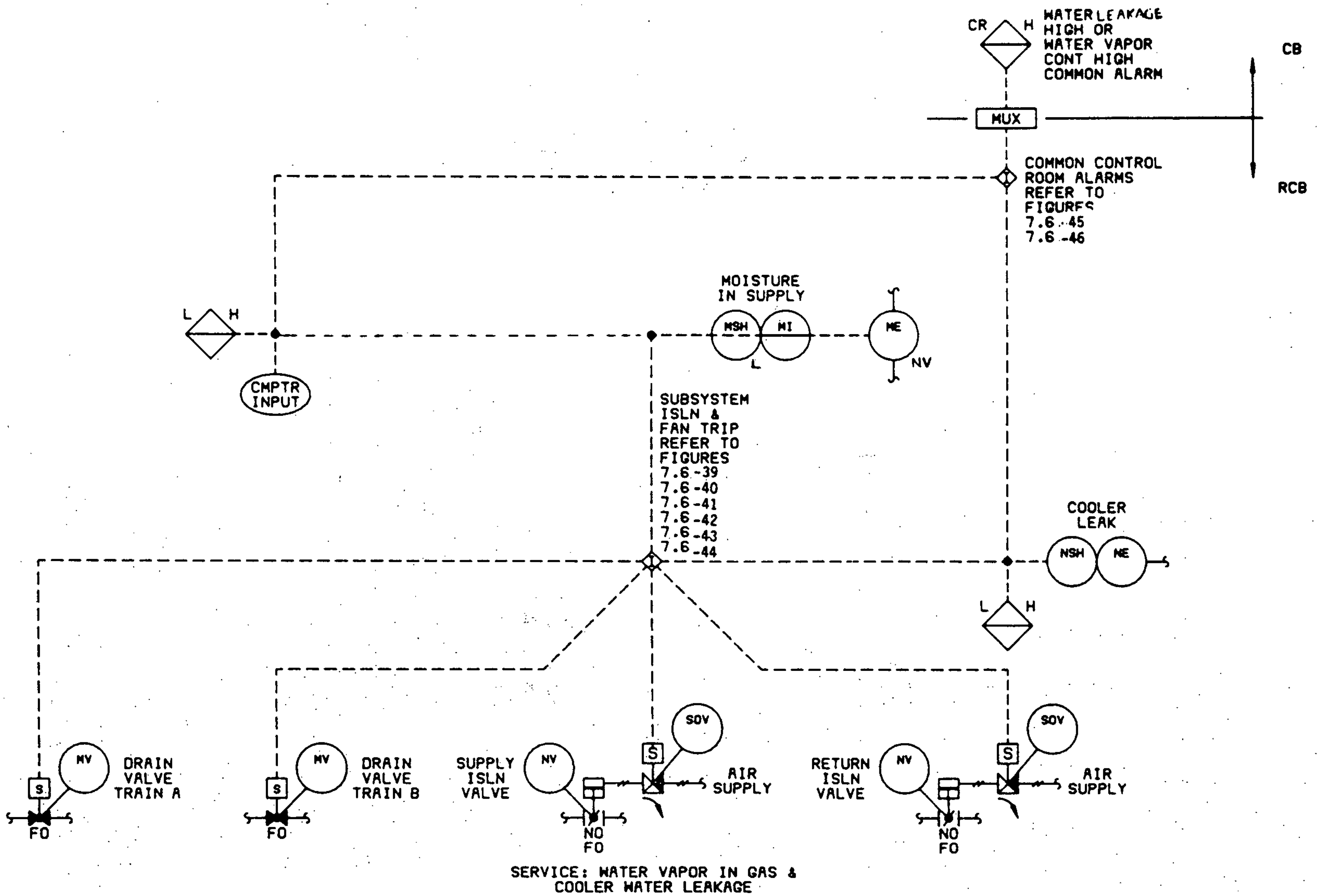


FIGURE 7.6-23
LOOP DIAGRAM
RECIRCULATING GAS COOLING SYSTEM
SUBSYSTEM MA (IDENTICAL FOR SUBSYSTEMS
MB, EA & EB)

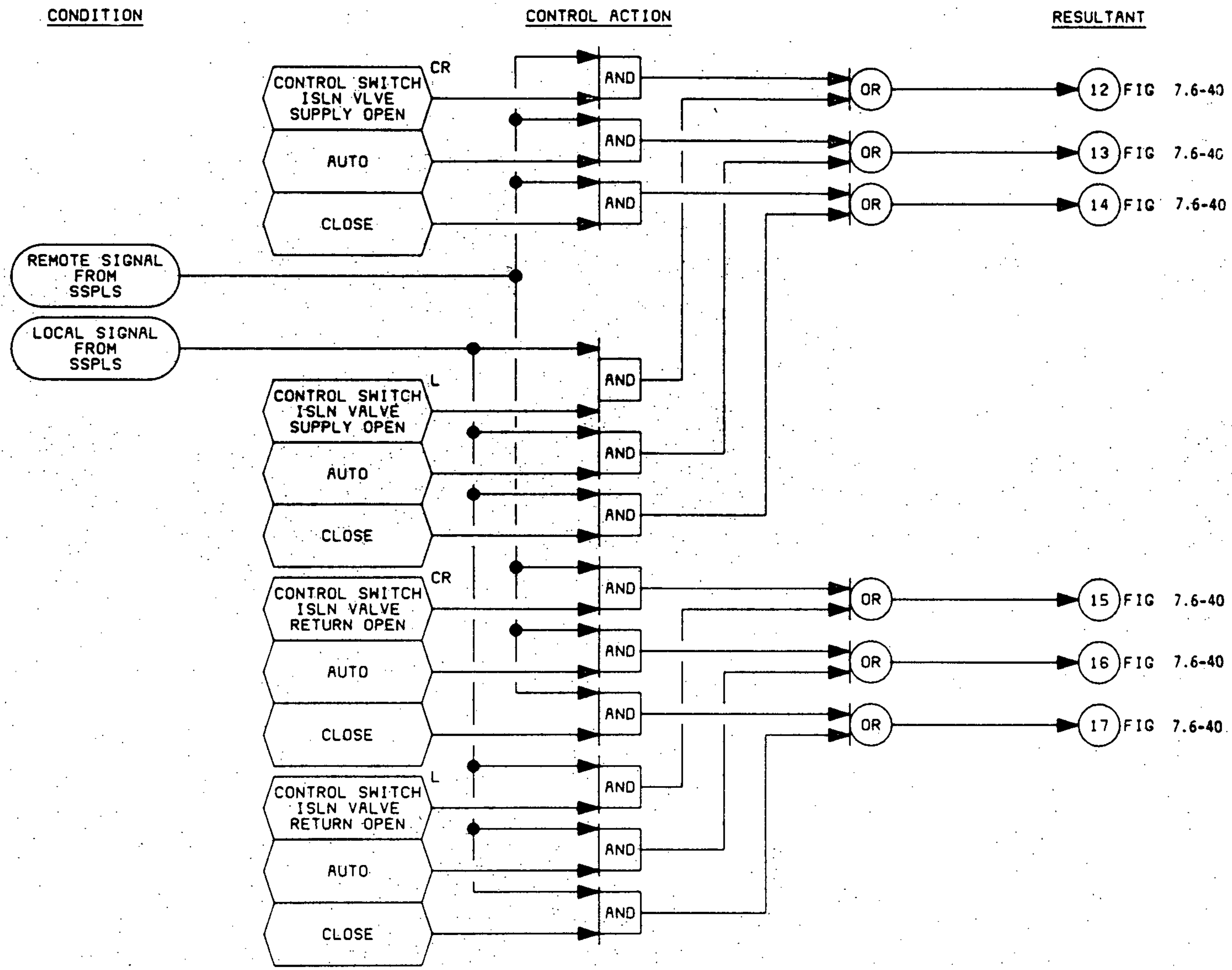


FIGURE 7.6-24
LOGIC DIAGRAM
SUBSYSTEM MA SUPPLY & RETURN ISOLATION VALVES

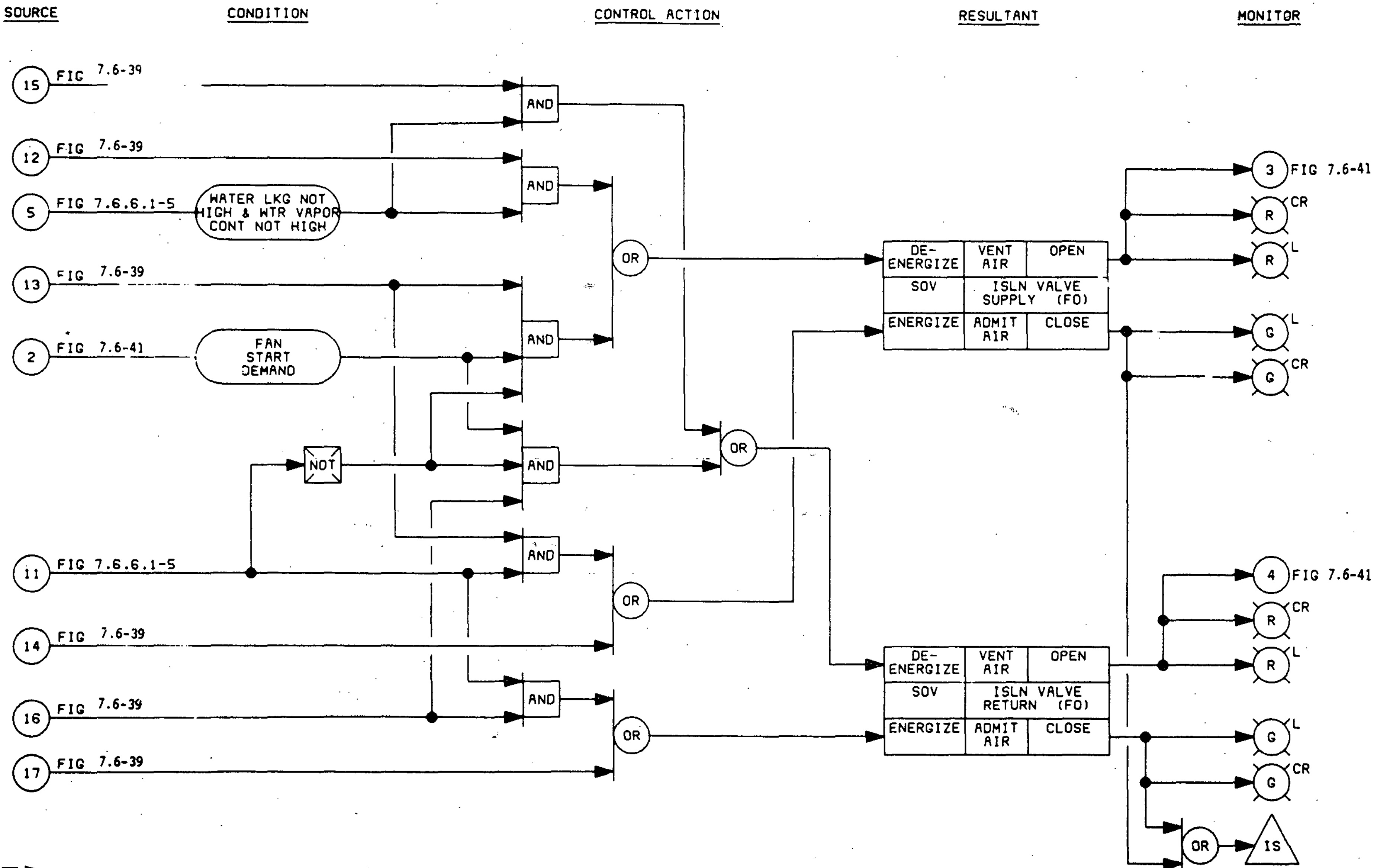


FIGURE 7.6-25
LOGIC DIAGRAM
SUBSYSTEM MA SUPPLY & RETURN ISOLATION
VALVES

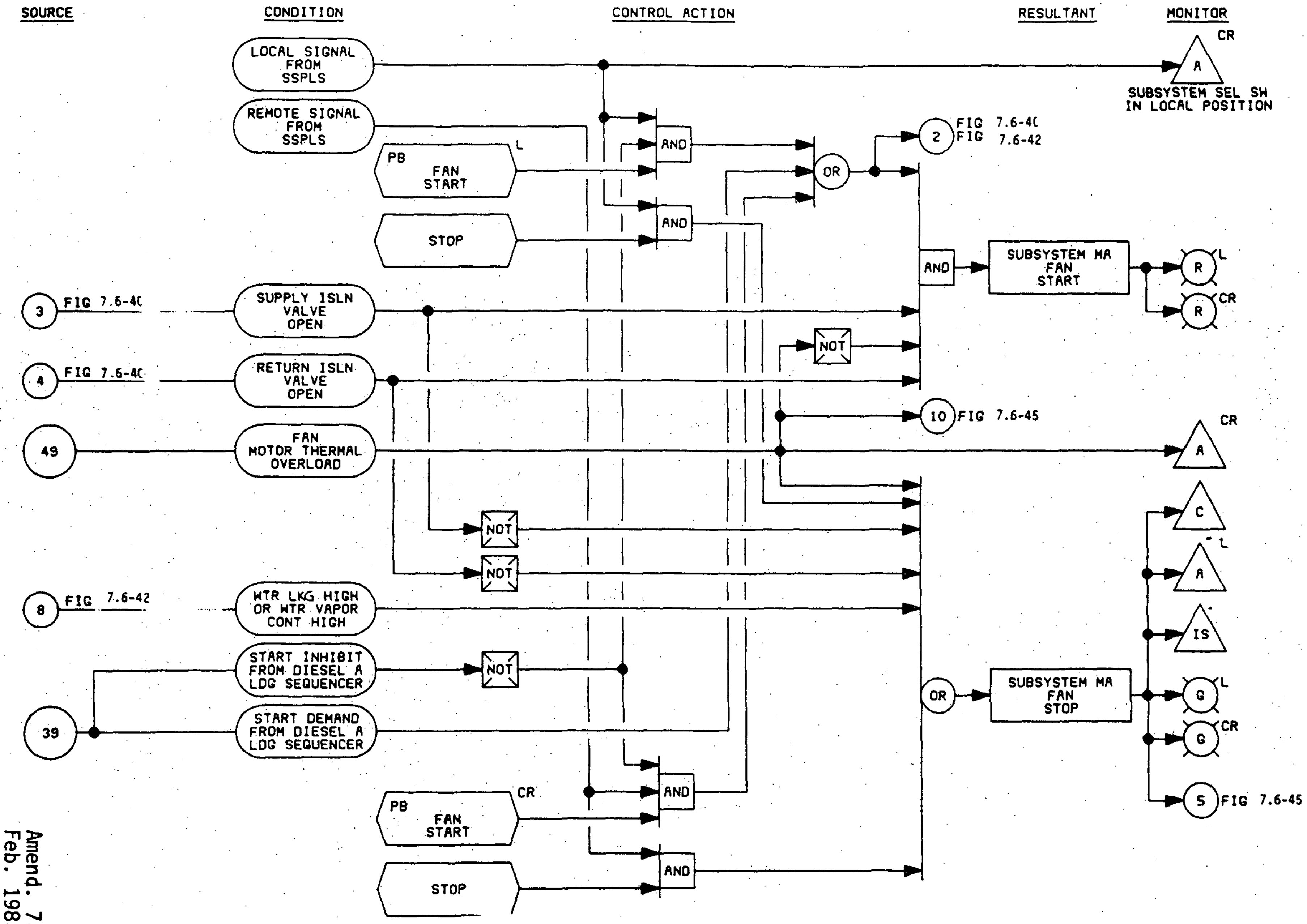


FIGURE 7.6-26
LOGIC DIAGRAM
SUBSYSTEM MA FAN

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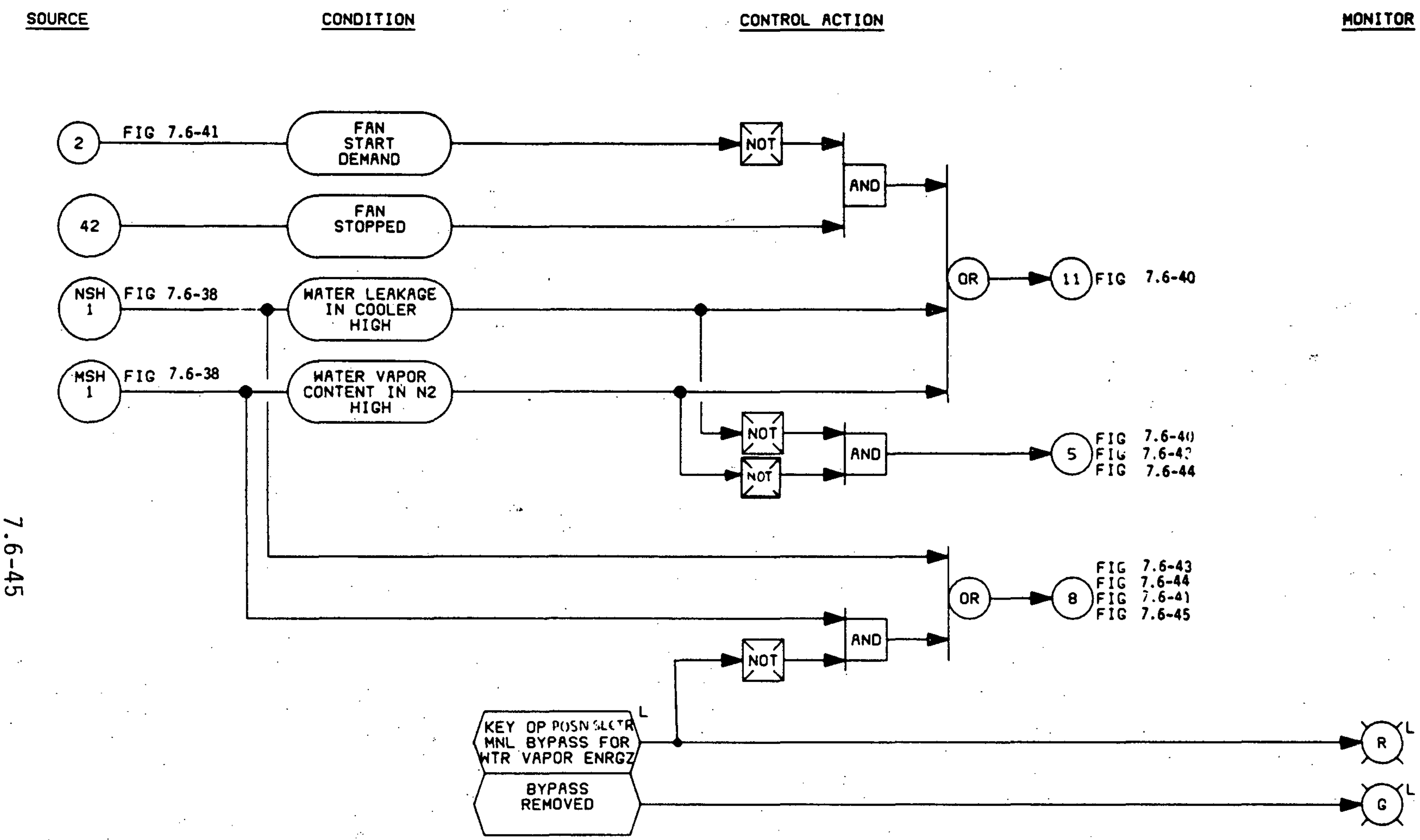

FIGURE 7.6-27
LOGIC DIAGRAM
SUBSYSTEM MA SUPPLY & RETURN
ISOLATION VALVES

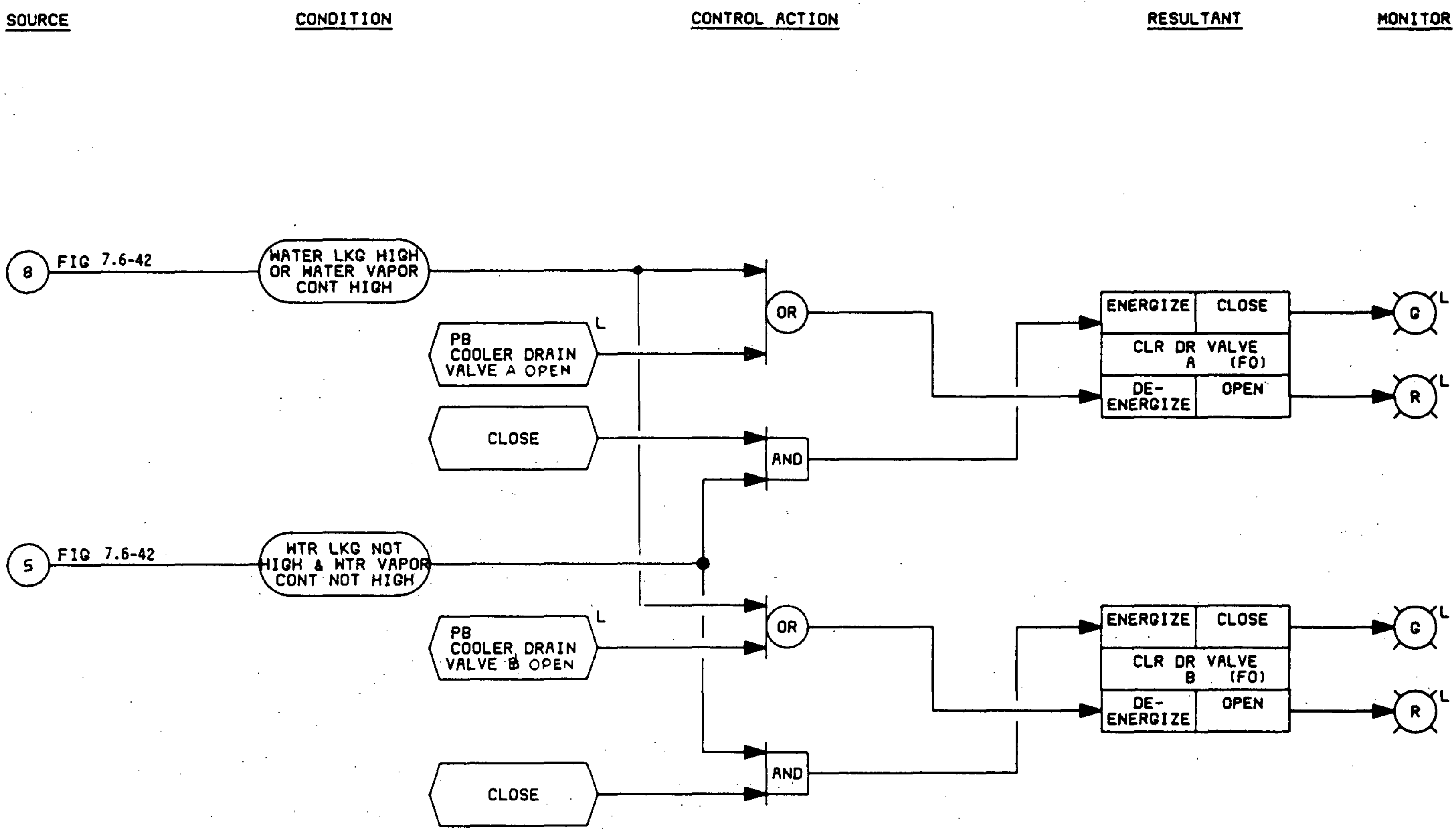

FIGURE 7.6-28
LOGIC DIAGRAM
SUBSYSTEM MA COOLER DRAIN VALVE

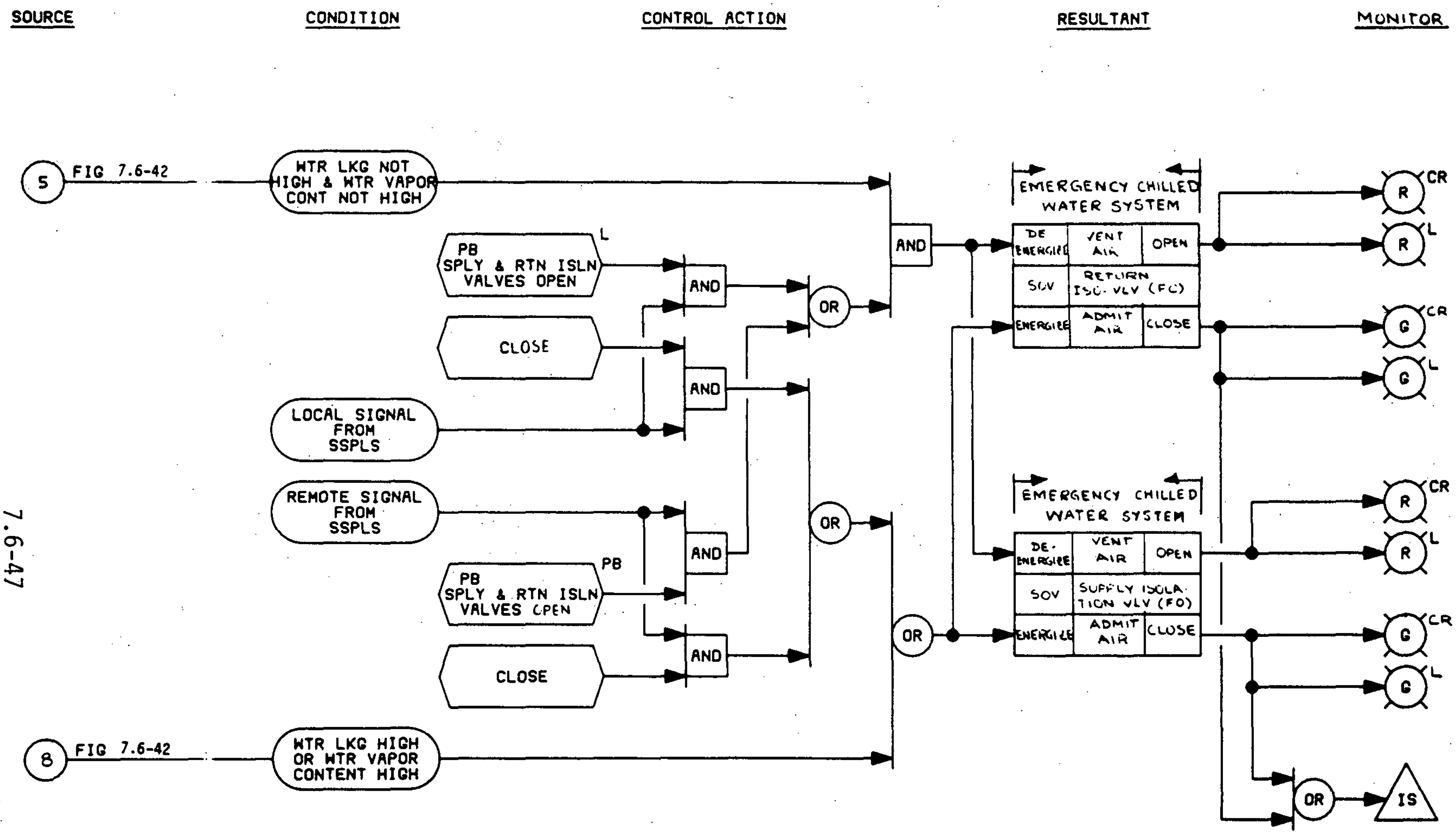

FIGURE 7.6-29
LOGIC DIAGRAM
SUBSYSTEM MA EMERGENCY CHILLED WATER
(TO COOLER) ISOLATION VALVE

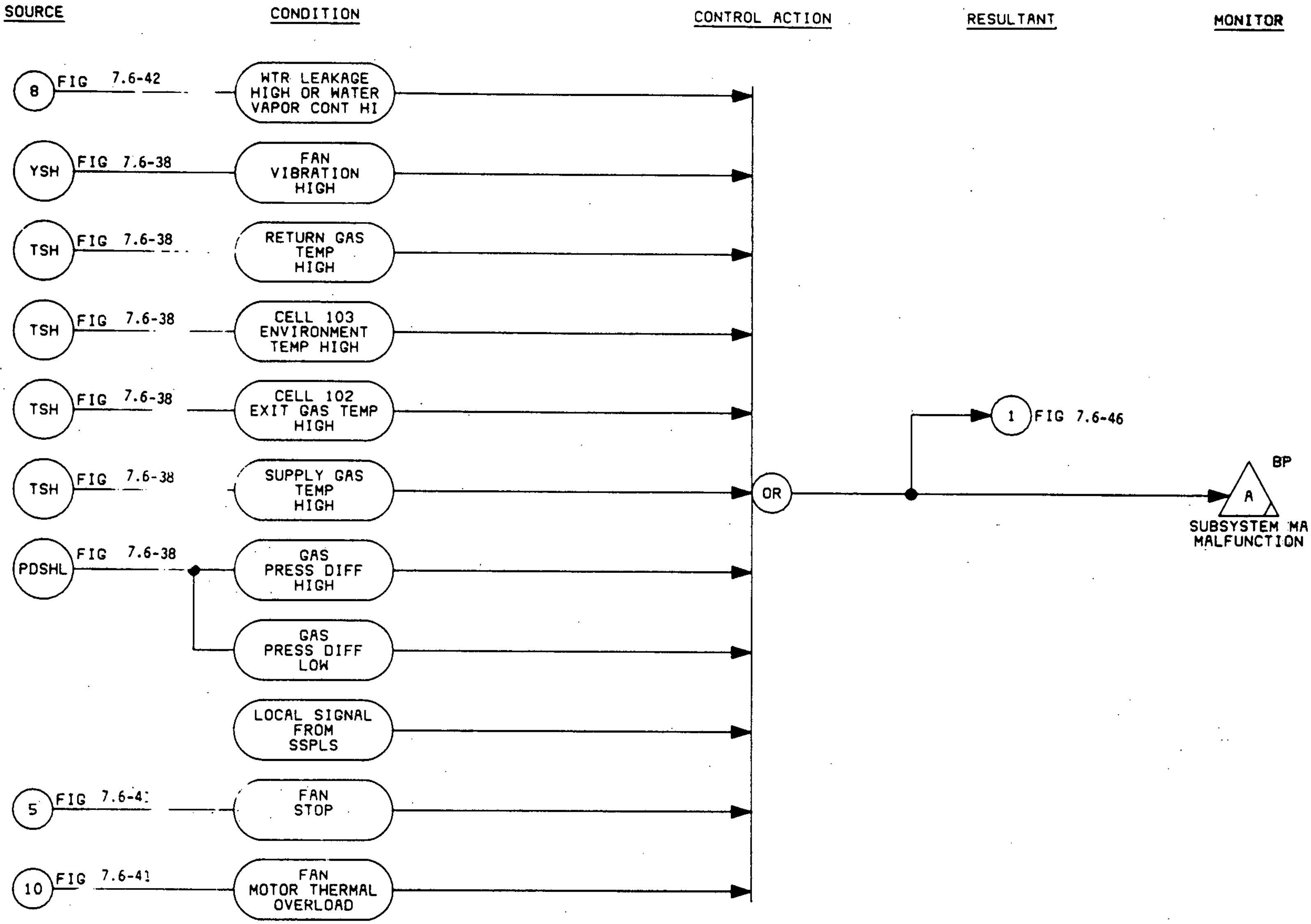

FIGURE 7.6-30
LOGIC DIAGRAM
SUBSYSTEM MA MALFUNCTION ALARM

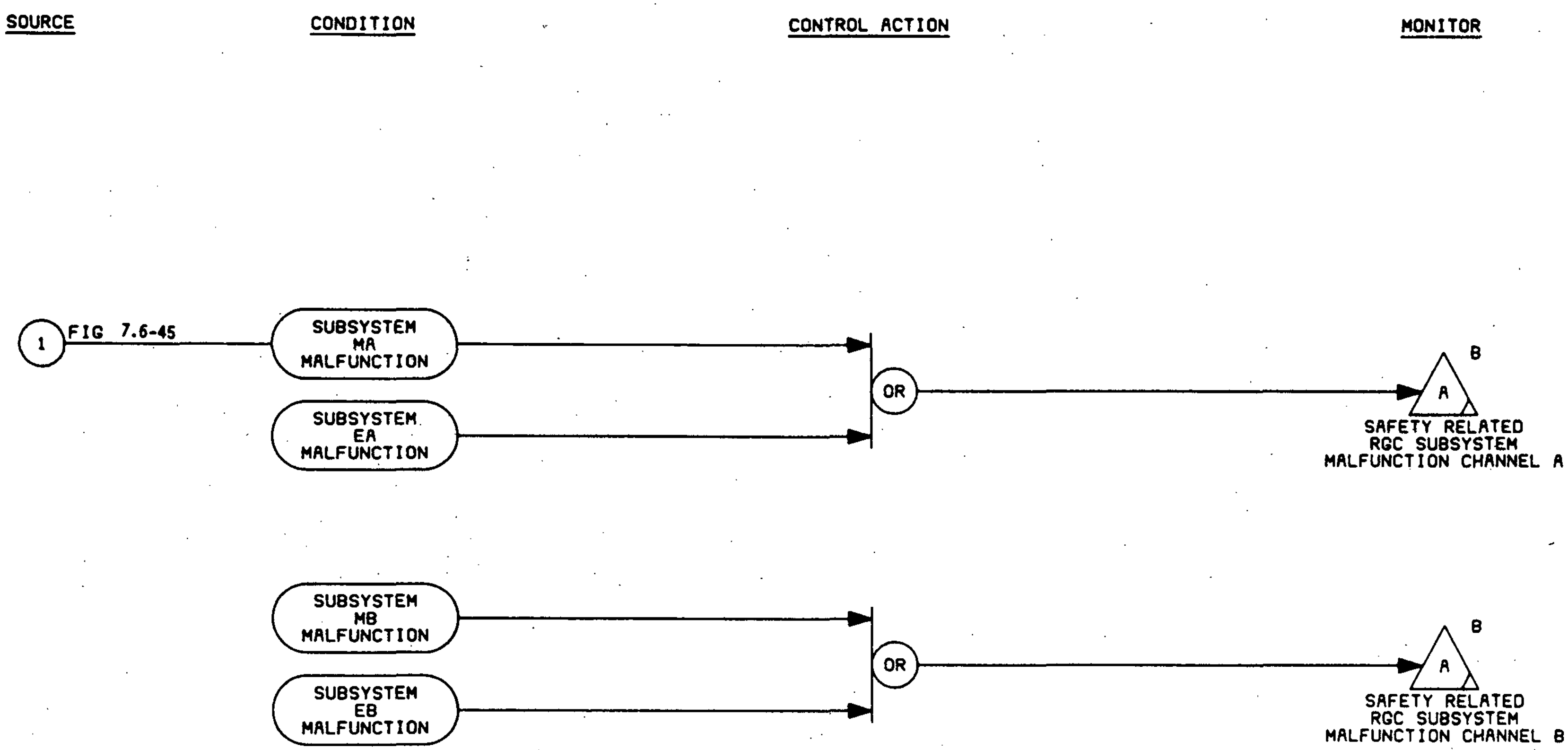

FIGURE 7.6-31
LOGIC DIAGRAM
RGCS SAFETY RELATED SUBSYSTEM MALFUNCTION
COMMON ALARM

7.7 INSTRUMENTATION AND CONTROL SYSTEMS NOT REQUIRED FOR SAFETY

The Plant Control System and other auxiliary instrumentation and control systems are not safety-related since these systems are not required to terminate or mitigate the consequences of postulated events to prevent unacceptable consequences nor will failure of these systems affect the operation or ability of the Plant Protection System to terminate the results of the PPS design basis events. The Plant Control System, which provides overall control and coordination of the Reactor Heat Transport, Nuclear Island Auxiliary and Balance of Plant Systems for all normal plant operating modes is described in Section 7.7 as are other auxiliary instrumentation and control systems. The Instrumentation and Control Systems not required for safety include:

- o Supervisory Control System
- o Reactor Control System
- o Primary & Secondary CRDM (Control Rod Drive Mechanism) Controller Systems
- o Rod Position Indication System
- o Rod Position Rod Block System
- o Sodium Flow Control System
- o Steam Generator Feedwater Flow Control System
- o Recirculation Flow Control System
- o Sodium Dump Tank Pressure Control System
- o Fuel Handling and Storage Control System
- o Nuclear Island Auxiliary Instrumentation and Control Systems
- o BOP Instrumentation and Control Systems

7.7.1 Plant Control System Description

The Plant Control System integrates the manual and automatic controls provided to maintain the plant at the desired power, temperature, pressure, and flow conditions for startup, load changing, rated power, standby, and shutdown conditions. The Plant Control System equipment includes: the control hardware for the Supervisory, Reactor, Sodium Flow, and Feedwater Control Loops; the Control Rod Drive Mechanism electrical and electronic processors; the Rod Position Indication and Display; and the inter-system interlocks and permissives. Automatic control of the power, sodium and steam temperatures, steam pressure, sodium flows, steam and water flows is provided for load changing and power operation above 40% thermal power. The loading/unloading is performed by changing the turbine load first with the nuclear steam supply following. The automatic control system maintains the temperatures, flows, and pressures according to a specified plant load profile shown in Figures 5.7-1 and 5.7-2. Below 40% thermal power the plant is controlled either manually or by flux control by the operator. Flux control refers to controller adjustment of control rod bank position activated by operator setting of the flux/power level.

The Plant Control System accomplishes the functions described above using a two level feedback control system. The Supervisory Control (top level) controls in accordance with the load demand from either the plant operator or the remote dispatcher. The core exit and steam temperature, power, flow, and steam pressure setpoints are established electronically according to the part load profile. These setpoints are the inputs for the Reactor and Sodium Flow Controllers (second level) which maneuver the control rods or sodium pump drives to attain the desired plant conditions. The Steam Generator Control System which includes the feedwater flow controls is also a second level controller but does not require an input from the Supervisory Control System. Figure 7.7-1 is a functional block diagram of this two level control system. Included on the diagram are the Supervisory, Reactor, Primary and Intermediate Flow, and Steam Generator Control which are described in detail below.

7.7.1.1 Supervisory Control System

A block diagram of the Supervisory Control System is shown in Figure 7.7-2. The Supervisory Controller uses steam temperature, steam pressure, steam flow, and load demand as input variables. The load programmer generates feed forward demand signals for the Reactor, Primary Flow and Intermediate Flow Controllers based on superheater exit steam flow. In addition, the Supervisory Control System contains trim controllers on steam temperature and steam pressure.

A feed forward Reactor Temperature Demand signal is produced by a temperature demand program function of plant load. This programmed temperature demand is then added to the controller output temperature trim demand at the input summer to the Reactor Temperature Controller.

A feed forward Intermediate Flow Demand signal is produced by a flow demand program function of plant load. This program flow demand is then added to the controller output flow trim demand to obtain the net intermediate flow demand for use by all three Intermediate Flow Controllers.

A primary flow demand signal is generated by a program function of plant load and supplied to each Primary Flow Controller.

The Flow Trim Controller provides the operator the capability from the main control panel of adjusting the HTS sodium flows. This controller adds any operator trim signals to the feedforward demands to obtain net demand signals for the primary and intermediate flow controllers.

The Steam Temperature Controller operates on the steam temperature error signal and provides a dynamic compensated temperature trim demand to the

Reactor Temperature Controller. The error signal is obtained by the algebraic sum of the temperature setpoint and the feedback measured steam temperature signal from the inlet to the Turbine Control Valve. The steam temperature setpoint at part load is the steam temperature profile shown in Figure 5.7-1.

The Steam Pressure Controller operates on the steam pressure error signal and provides a dynamic compensated flow trim demand to the Intermediate Loop Flow Controllers. The error signal is obtained by the algebraic sum of the steam pressure setpoint and the feedback measured steam pressure signal from the inlet to the Turbine Control Valve.

In addition, the supervisory control provides demand signals to the Turbine Control System.

7.7.1.2 <u>Reactor Control System</u>

A block diagram of the Reactor Control System is shown in Figure 7.7-3. The Reactor Controller uses a weighted core outlet temperature outer loop and an inner loop based on neutron flux to control the reactor outlet temperature by maneuvering the control rods. Multiple sensor inputs are used to assure control reliability. Redundant Rod Withdrawal Blocks based on high flux to flow deviation and high flux level are provided. The temperature setpoint is generated either manually or by the Supervisory Controller.

The Reactor Control System positions the control rods to reach the desired reactor thermal power and outlet temperature. The Inner Loop Flux Control, as part of the Reactor Control System, uses the Median Select Circuitry output as the feedback signal. The inputs to this circuit are three filtered and buffered power range flux signals. The feedback signal is then sent to a difference amplifier which, along with the power demand, generates the error signal for the Flux Control Compensation Network. The output of the compensation network is then sent to the Power Dead Zone and Saturation Circuit. This circuit sets up the region of "no control" and also the upper and lower limits of the control rod rates. Finally, the signal is divided into an analog magnitude signal and a digital direction signal for use as demands to the Control Rod Drive Mechanism Controller.

In the Outer Loop Temperature Control, thirty core exit thermocouples are used to generate the feedback signal (see Section 7.5.3). The thermocouple outputs are first individually filtered and then processed through an averaging and reject circuit. The feedback is compared with the temperature demand through a difference amplifier to generate the error signal. It is then sent to the Temperature Dead Zone Circuit followed by the Temperature Control Compensation Network. If the TEMP/FLUX Mode Switch is in the Temperature Mode, the signal becomes the power demand after going through the High Flux Limit Circuit. If the Mode Switch is in the Flux Mode, then the power demand is set by a manually adjusted potentiometer on the control panel. The High Flux Limit will keep the power demand below some preset limit. The AUTO/MAN Mode Switch is used to select the temperature demand from either the Supervisory Controller or a manually adjusted potentiometer on the control panel.

During plant startup, after criticality and hot standby conditions are reached, the operator may bring the reactor to 40% power by using either the Flux Control System or the Manual Rod Bank Control. Both Systems provide the capability to start the reactor up even in the presence of a limited positive bowing coefficient.

The Flux Control System is described above and shown in Figure 7.7-3; here, the operator can change power by adjusting the controller set-point. The Manual Rod Bank Control System interfaces with the Flux Control at the input to the magnitude and direction circuits shown in Figure 7.7-3. In this mode the operator will move the rods up or down in a bank just as the Flux Control System does but at a much smaller rate. The operator will use panel meter displays or
42 the data handling system as feedback.

7.7.1.3 Primary and Secondary CRDM (Control Rod Drive Mechanism) Controller and Rod Position Indication

The Primary Control Rod Drive Mechanism Control System transforms the bulk 3 phase power into the pulsed DC necessary to operate the Control Rod Drive Mechanism in response to input commands from the Reactor Control System. Interlocks and permissives are provided to prevent operating sequences of the control rods which would damage the equipment, and assure that the rods are maintained in the banked configuration required to maximize core performance. Rod Position Indication is provided redundantly for each rod to permit the operator to verify the reactivity status and operation of the control system. The Secondary CRDM Controller and Rod Position Indicators are described in Section 4.2.3.

7.7.1.3.1 Primary CRDM Control

The control rod drive mechanism is actuated by a 4 pole, 6 winding reluctance stepping motor. The mechanism lead screw has a thread pitch of 0.6 inch, and moves 0.025 inches for each pulse to the drive stator. A block diagram of the drive system is shown in Figure 7.7-4. Driving power is supplied from the site power system through redundant motor-generator sets, Reactor Shutdown System scram breakers, a 3 phase to 6 phase transformer, and banks of silicon controlled rectifiers (SCR's) in the individual controllers to the stator windings of the CRDM.

The primary rods are divided into 2 groups. One group of 3 startup
59 rods responds only to single rod manual control and during
reactor operation is normally fully withdrawn. A group of 6 control rods
respond to manual control or to an analog signal from the Reactor Control
57 System.

Rod speed demand limits are included in the reactor controller as well as rod speed limits in the individual Primary CRDM Controllers. Rod block interlocks are included in the "OUT" demand input as shown on Figure 7.7-5.

When the Reactor Shutdown System initiates a scram, the "Scram Breakers" open and interrupt the power to the Primary CRDM stator coils; the rotor collapses and disengages the rollers from the lead screw; and the CRDM drive train falls under the force of gravity and the scram assist spring to insert the control rods into the core. Failures within the sequence and controller units cannot prevent removal of the power required to hold the CRDM's in the withdrawn position. The components are described below. 19

Motor-Generator Set

Dual M-G sets provide the 3 phase power for CRDM operations.
57 When a latch signal is received at the voltage control, the output voltage
of a generator is increased. The M-G sets for the primary rods use
a 200 Hp motor and a 150 Kw generator.

Mechanism loads are shared by the two M-G sets; however, either M-G set has the capacity to power the entire load of the primary system. Controls are provided to synchronize the two M-G sets. The motor-generator sets are designed to provide sufficient inertia and voltage control to prevent rods dropping in the event of short power outages.

Each M-G set is shutdown when a scram signal is generated by the Secondary Reactor Shutdown System. This assures that the Reactor Control System can not continue to withdraw the primary control rods after the plant is shutdown by Secondary RSS action.

Generator output circuit breakers provide the necessary electrical protection for the generators and for system maintenance.

<u>Power Supplies and Transformers</u>

Since the CRDM controller use 6 phase AC power, one 3 phase to 6 phase transformer is provided for the primary rods. The transformer includes appropriate secondary side surge protection.

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Each CRDM controller requires control power to operate the interface
circuitry, programmer, gate drives, internal interlocks and display
equipment. As shown on Figure 7.7-4, redundant AC power sources
57 energize redundant DC logic power supplies whose outputs are auctioneered.
This design prevents failure of a power supply from causing a rod to
drop.

The power supplies are sized to provide sufficient capacity for all
of the CRDM controllers in the primary group. Transformer isolation,
including grounded Faraday shields, is used to prevent failures from
propagating into the controller electronics.

CRDM Motor Controller

The CRDM Motor requires DC energization of coils in the pro-
per sequence to develop the required setpoint motion. The sequence
of coil energization for rod motion is in a two coil-three coil
57 sequence. Thus a forward step is produced each time
a leading coil is energized and also when a trailing coil is
de-energized. To reverse the motion, the sequence is reversed.

The CRDM Controller uses six SCR's for each stator coil to half wave
rectify the 6 phase AC input power and supply DC output to a stator
coil. All six SCR's for a stator coil are turned on by one gate
drive unit. The Controller incorporates the logic necessary to
correctly sequence the gate drive units on and off, thereby
sequencing the coils in appropriate order. Separate controllers are
provided for each individual mechanism. Holds are provided when
input or output logic errors are detected.

57 In Single Rod Control Mode, the input circuitry to each controller accepts
on-off inputs for IN, OUT, and HOLD commands and provides the sequencer with
an IN pulse train, OUT pulse train, or HOLD DC output. The IN command steps a
single rod down in the core at a predetermined rate. The OUT command
steps a single rod up out of the core at a predetermined rate (not
necessarily the same as the IN rate) and the HOLD command maintains
the rod in its present position (no motion). The input circuitry
also incorporates adjustable speed settings for the IN, OUT, and
LATCH modes of CRDM operation and assures that an IN command takes
precedence over an OUT command. In addition to the adjustable speed
settings, the controller provides an independent speed limitation
which has a separate clock and power supply from that used by the
input circuitry. If the input circuitry called for a speed greater
57 than 10% above 9 inches per minute due to a postulated failure, the
speed limiter circuit will place the rod in the Hold Mode.

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In any automatic control mode, or in Group Manual mode, the mechanism controllers are operated in sequence one step at a time to keep the rod bank in required alignment. The sequence rate and direction are determined respectively by analog and digital signals from the reactor control system. If the selector sequence rate is higher than a predetermined trip point, an overspeed detector will alarm and place the controllers in HOLD. A functional block diagram of the control is shown in Figure 7.7-5.

Hold Bus

A Hold Bus Power Supply and transfer select circuitry are provided to allow any controller to be replaced without a plant shutdown. In the event of a controller failure, the mechanism controller in question can be switched out and transferred to a Hold Bus. Power to the Hold Bus Power Supply is provided downstream from the scram breakers. This ensures that if a scram is initiated, a rod on the Hold Bus will also scram.

7.7.1.3.2 Primary Rod Position Indication System

Two independent Rod Position Indicating Systems are provided for each primary control rod: An Absolute Position Indication System (ARPI) and a Relative Position Indication System (RRPI). These systems assure that the plant operators can continuously determine the position of the control rods.

The ARPI provides a direct measurement of rod position at any time and, unlike the RRPI, does not require re-zeroing after a scram or temporary loss of power. The system is solid state, utilizing ultrasonics and magnetics to provide a DC output indicative of rod position.

The sensor for this system consists of a tube extending down from the top of the motor tube and into the inside diameter of the PCRDM lead screw. A nickel-cadmium wire is stretched axially through the tube. As the lead screw translates, the flux from a torroidal magnet mounted on top of the lead screw intersects the wire at a point indicative of the rod position. Electrical pulses sent down the wire generate magnetic fields which, when they intersect the flux of the lead screw magnet, causes a torsional strain creating a sonic pulse which travels from the point of flux intersection upward. The sonic pulse is detected at the top of the wire, and the time of propogation is measured electronically. This propagation time is converted to a D.C. signal which is analagous to rod position.

This signal is read out on the main control panel by rod top and rod bottom indicator lights and a vertical bar graph indicator. It is also used to operate the rod out of alignment alarm, the rod position rod block system and rod control interlocks.

The Relative Rod Position Indication System provides a digital rod position indication on a CRT at the Main Control Board. Two pairs of magnetic coil pick-ups are mounted within each stator jacket above the stator and on opposite sides. A 6 pole magnetic section is attached to the mechanism rotor and rotates in the plane of the pick-up coils. Voltage pulses caused by the movement of the poles in the proximity of the pick-up coils are sent to a digital to analog converter. The D/A converter produces an analog signal which is a measure of rod position. This analog signal is sent to the PDH&DS and the rod position rod block system. The resolution of this signal is ±0.1 inch. Unlike the Absolute Position Indication System, this system must be reset after each scram and in the event of a power failure reset after power is restored. The pulses are also counted by an odometer type readout in the rod control equipment room.

7.7.1.3.3 Rod Position Rod Block System

The Rod Position Rod Block System provides protection against incorrect or invalid control rod positions when the plant is operating. As shown in Figure 7.7-6, rod position signals from the Relative Rod Position Indication (RRPI) and Absolute Rod Position Indication (ARPI) systems are used by two redundant trains of rod blocking logic. Each logic train outputs a rod block signal when (1) the position of one of the six row 7 control rods is more than a set distance above the average position of all the six row 7 rods comprising the operating bank or (2) the average position of the six row 7 rods exceeds a predetermined setpoint. A rod block signal from either of the two redundant logic trains results in all controllers for the six rods of the operating bank switching to the HOLD mode. Signals are also provided to the unit load controller of the supervisory control system to ensure that a plant loading or unloading operation is stopped upon the occurrence of a rod block. This prevents a reactor trip due to power/flow mismatches which may occur if sodium flow is allowed to change without a corresponding change in reactor power. In addition to the redundant logic trains, the rod block system includes:

1) Circuitry necessary to convert the pulses of the RRPI signal conditioners to an analog signal.

2) Deviation alarms which continually compare the RRPI signal and ARPI signal from each rod and from the rod position average circuit and provide a position fault alarm to the Plant Annunicator System when the two signals differ by a set amount.

3) A Low Power Bypass in each logic train which may be manually instated at low power to disable the rod block system. This bypass is provided to allow for control rod movement which is necessary to perform low power physics and startup testing. This bypass is automatically removed during the ascent to power.

4) A momentary manual override feature to allow the removal of the rod block so that the operating bank may be re-positioned if a rod block occurs. When the manual override feature is engaged, the operator may manually insert control rods to realign the operating bank. Withdrawal of control rods while the manual override feature is engaged is automatically prohibited.

5) Testing and bypass features to allow for the testing and maintenance of the RRPI, ARPI or one train of the rod block system during plant operation.

6) System alarm outputs which provide signals to the Plant Annunciator System when either train is bypassed or upon the occurrence of a rod block.

7.7.1.4 Sodium Flow Control System

The Sodium Flow Control System consists of six controllers used to drive the three primary and three intermediate sodium pumps. Each controller consists of a cascade system with an inner loop using speed as the feedback signal and an outer loop based on a flow feedback signal. The flow control range is 30 to 100% of rated flow. The flow setpoints are generated either manually or by the Supervisory Control.

Figure 7.7-7 is a block diagram of the flow/speed control loop which is typical of the six controllers in the system. The Speed Control System is an inner loop and used pump speed, which is sensed via a pump shaft mounted tachometer, as the feedback variable. The Speed Control System is limited internally by the torque limit circuit which sets both the accelerating and decelerating torque of the variable speed pump drive.

The demand to the Speed Controller is set by the FLOW/SPEED Mode Select Switch. In the Speed Mode, pump speed is set by a manually adjusted potentiometer; in the Flow Mode, pump speed is set by the Flow Controller. The Flow Controller uses the filtered, median select signal of three available redundant flow meter buffered PPS outputs as the feedback signal. This signal, along with the flow demand, is used to generate the error signal which is compensated through the Control Compensation Network and then limited by the High Speed Limit Circuit prior to being used as the speed demand signal. The demand to the Flow Controller is set by the MAN/AUTO Select Switch. In the automatic mode, the demand comes from the supervisory control, while in the Manual Mode, the demand comes from a manually adjusted potentiometer on the control panel.

7.7.1.5 Steam Generator, Steam Drum Level Control System

The steam drum level control system regulates the feedwater flow to the steam drum to maintain a constant water level in the steam drum during plant operation.

The control system consists of a three element (steam flow, feedwater flow and steam drum water level) controller and a median select module. Each of the input elements have three redundant measurement channels. The median select module selects the median signal of the three channels as the input to the controller.

Independent Class 1E high steam drum level trip logic trains are provided at 8 inches and 12 inches above steam drum normal water level. Each logic train also uses three redundant inputs and a median select module.

The steam drum level control signal, the 8 inch high level signal and the 12 inch high level signal, have separate buffered signals provided from the PPS instrument channels for isolation and independence.

The control logic is shown in Figure 7.7-1.

7.7.1.5.1 Feedwater Flow Control Valve Control

The startup feedwater control valve conntrols flow in the range of 0 to 15% of rated flow. The control loop for this valve is a single element controller, using drum water level to control valve position. The main feedwater control valve is closed during this operation. When the flow rate increases to approximately 15%, the control system will automatically open the main feedwater control valve and close the startup control valve. A deadband is provided for this switchboard point to prevent cycling from one valve to the other.

The control loop for the main valve is a three element controller, using drum normal water level, steam flow, and feedwater flow, to control the valve position. Drum drain flow rate, which remains essentially constant at all power levels, is a manual input to the controller. The controller compares steam flow to feedwater flow, and the resulting net flow error signal is combined with the drum water level error signal, to control the valve position. Drum water level is controlled within ± 2 inches of the normal water level. Three redundant buffered signals are provided from the PPS for steam flow, feedwater flow and steam drum level. The median signal of each element is provided to the steam drum level controller. Manual control of the startup and main feedwater control valves is provided in the control room. Instrumentation required by this control system is obtained as follows:

- Steam Drum Level - Water level is measured by a differential pressure transmitter which senses the difference between the pressure resulting from a constant reference column of water and the pressure resulting from the variable height of water in the steam drum. The measurement is density compensated.

- Steam Flow - Steam flow is sensed at a flow element in the outlet line from the superheater by a differential pressure transmitter. The differential pressure signal is compensated for temperature and pressure variations and linearized to provide a mass flow signal.

- Feedwater Flow - Feedwater flow is sensed at a flow element in the inlet line to the steam drum by a differential pressure transmitter. The differential pressure signal is corrected for temperature variations and linearized to provide a mass flow signal.

7.7.1.5.2 Main Feedwater Isolation

Isolation of the main feedwater supply is provided to mitigate the consequence of the loss of feedwater to a steam drum, a steam line break, or to prevent superheater flooding.

Isolation of the feedwater supply to the affected loop in the event of a steam generator system feedwater leak will ensure integrity of the feedwater supply to the two unaffected loops and mitigates the consequence of flooding damage to other equipment. This protection is provided by automatic closure of the steam drum isolation valve and both feedwater control valves upon sensing a low steam drum pressure (500 psig) signal and automatic closure of both feedwater control valves and feedwater valve isolation upon sensing a steam generator building flooding (temperature and humidity) signal.

In the event of a steam line break, steam drum dryout may occur and would result in damage to the steam generator loop upon re-introduction of feedwater. Protection against the re-introduction of feedwater is provided by the closure of the feedwater isolation, the steam drum isolation, and control valves on low steam drum pressure (500 psig) signal.

In the event of a failure in the drum water level control components, an overfilling condition might result in flooding of the steam drum and superheater. Protection against this is provided by three redundant water level sensors and by trip functions which close the feedwater valves at two steam drum levels. The first trip level, 8 inches above normal water level, closes the feedwater steam drum isolation valve, and the feedwater control valves. The second trip level, 12 inches above normal water level, closes the feedwater isolation valve.

Protection against flooding of the superheater during steam generator auziliary heat removal is discussed in Section 5.6.1.

7.7.1.5.3 Operational Considerations

Normal Operations

The steam drum level controller utilized for feedwater control valve operation is located in the control room back panels. The operator control station for the controller is located on the main control panel in the control room.

During normal operation the steam flow and feedwater flows mismatch is summed with the drum level signal and compared to level setpoint. The resultant signal is sent to the main feedwater control valve to provide a feedwater flow balanced to steam flow plus drain flow to maintain drum levels.

Operator Information

Indicators and alarms are provided to keep the operator informed of the status of the system under control. The following measurements are continuously displayed on the Main Control Board:

Superheat steam pressure, temperature and flow;

Steam drum pressure, temperature, and level;

Feedwater pressure, temperature and flow.

7.7.1.6 Recirculation Flow Control System

The reference design calls for constant speed recirculation pumps.

7.7.1.7 Sodium Dump Tank Pressure Control System

The Sodium Dump Tank Pressure Control System functions to maintain the argon cover gas pressure over the sodium dump tank within prescribed limits. Pressure control of the sodium dump tank is maintained by the supplying or venting of argon gas to or from the cover gas region via a two-inch line. Two pressure transmitters are located on this line, one for indication and off-normal alarm, and one for control of the argon supply and vent valves.

Control of the argon supply and vent values is accomplished by an "on-off" type pressure controller which cycles the supply and vent values to maintain the cover gas pressure between the lower and upper limits. Sufficient dead band is provided between lower and upper limit operation to prevent undue cycling of the supply and vent valves.

Operational Considerations

The pressure controllers for the sodium dump tanks are located at the local control panel in the Steam Generator Building. However, manual overrides for the supply and vent valves are provided in the main control room and may be utilized at the plant operator's discretion.

High and low pressure alarms alert the operator to off-normal conditions which may result from a malfunction of the pressure control system. Pressure data is provided to the Data Handling and Display System and is available for display upon call by the operator.

7.7.1.8. Steam Dump and Bypass Control System

The Steam Dump and Bypass Control System provides the necessary control and instrumentation hardware to operate the Turbine Bypass System as described in Section 10.4.4 and shown in Figure 10.3-1.

Redundant interlocks are provided to prevent bypass operation in the event the condenser is unable to accept steam flow (e.g., high condenser back pressure or high desuperheater outlet temperature).

At reactor power levels above 40% rated power, the Turbine Bypass Control System is operated in a load error mode where bypass valves are opened proportional to the difference between turbine demanded load and generated power. A valve position signal is provided to the Turbine Bypass Control System by the Supervisory Control System which makes the comparison. The circuitry includes a dead band with a 10% load setpoint.

A pressure control channel is provided for the regulation of main steam pressure following reactor trip, during decay heat removal operation and during turbine standby, loading and unloading operations. The pressure control mode is automatically selected for reactor power levels below 40%.

7.7.1.9 Fuel Handling and Storage Control System

The Fuel Handling and Storage (Reactor Refueling) Control System consists of electronic, electrical and mechanical hardware and software integrated into a coordinated system through a single common center utilized primarily for control of the refueling equipment used to load and unload core components and move them in and out of the plant as required. Emphasis is placed on optimizing control of the machines and facilities used repetitively during the annual refueling cycle [i.e., In-Vessel Transfer Machine (IVTM), Ex-Vessel Transfer Machine (EVTM) Reactor Rotating Plugs (RRP) and Ex-Vessel Storage Tank (EVST)]. See Section 9.1 for description of refueling equipment. However, control equipment for the other refueling equipment (i.e.,
20 Auxiliary Handling Machine (AHM) and Fuel Handling Cell (FHC) is also provided. Secondary functions of this system are: (1) data acquisition as required by the Plant Data Handling and Display System, (2) inventory of all core components in the plant and their locations, (3) providing system status indication and display to the refueling supervisor, (4) and monitoring of refueling equipment for out-of-limit conditions.

The Refueling Control System provides an efficient means of controlling the fuel handling machines used for reactor refueling. The control system design makes use of computer control for simple linear and rotary motions which are repeated many times during a refueling cycle. This automation reduces the possibility of operator error and provides for automatic inventory control.

The following definitions are important to an understanding of the Fuel Handling and Storage Control System design:

- EVENT - A change of state that occurs in essentially zero time, e.g., the closing of a limit switch.
- ACTION - A change of state that occurs between two events, e.g., the motion of a gate valve between the open and closed position.
- SEQUENCE - A series of actions which are always performed in the same order; for example, the various actions associated with pressurizing and depressurizing the seals associated with closing the EVTM closure valve constitute part of the closure valve sequence.

Three levels of control and automation are included in Fuel Handling and Storage Control System:

- Manual Systems (little or no automation) in which all actions are initiated individually by an operator. This is accomplished with pushbuttons, handcranks, or other appropriate means. Sufficient
59 information is displayed to the operator for verification that the preconditions for the next action have been satisfied.

- Semiautomatic systems in which each sequence is initiated manually by an operator, but actions within each sequence are initiated automatically by the control system. In each case, prior to initiating an action, the control system must have the capability for verifying that the previous action in the sequence has been successfully completed, and that any other preconditions for initiation of the next action have been satisifed. Information must be displayed to the operator for verification that the preconditions for sequence initiation have been satisfied.

- Moderately automated systems in which all actions within each sequence are initiated automatically, as in semiautomatic systems, and in addition, some sequences are initiated automatically. For automatic initiation of sequences, the control system must have the capability for determining which sequence should be initiated following completion of a previous sequence, as well as being capable of determining that all necessary conditions are met prior to sequence initiation.

59 Interlocks as described later in this section are incorporated to prevent the operator or automatic control equipment from initiating actions or sequences that result in equipment damage.

The following definitions are used to denote equipment location:

- CENTRAL - The control equipment is situated in a control room known as the Communications Center located at the far end of the gantry rails in the Reactor Service Building (RSB).

- LOCAL - The control equipment is situated in control consoles or racks located on or near the applicable refueling machine.

The degree of automation and centralization are interrelated. All manual systems are also local systems. Semiautomatic systems are local. Moderately automatic systems have central sequence and local action initiation. Whenever automatic control is used, a local manual backup is provided for use in case of failure in the automatic control equipment.

Figure 7.7-8 is a block diagram showing the interconnection of the control hardware and the man-machine interfaces. The EVTM, IVTM and EVST all have automatic local control of sequences with manual local backup control.

44 The Central Computer in the Communications Center handles the following items: Core component inventory control (both by core component location and by storage location contents), system level data acquisition from all

refueling machines, data storage and transmission to the Plant Data Handling and Display System, over-all system status information acquisition and display as required for Communications Center operating personnel, and monitoring of all interlock and annunciator status.

The EVTM computer functions as follows: receives and processes data from instrumentation and controls on the EVTM, and passes data to the Central Computer as required; provides action initiation for the EVTM; and displays data required for the EVTM operator.

The IVTM computer functions as follows: receives and processes data from instrumentation and controls on the IVTM, and passes data to the Central Computer as required; provides action initiation for the IVTM from sequence initiation commands given by the IVTM operator, processes data for core component identification done by the IVTM, and passes this data to the Central Computer.

Control consoles are located in the Communications Center and at or near all the machines. These consoles provide input for command and data output for the computers, display of required instrumentation and control data, manual backup to automatic control for the machines, annunciator displays, instrumentation signal conditioning and power distribution to the machines. Where required, key switches will be provided to override inter-
59 locks for maintenance, initial set-up, and calibration. In order to override these interlocks, it will be necessary for a keyswitch to be activated by the responsible operator; this action will also be annunciated in the Communications Center. All signals from control computers to control equipment can be manually initiated from the local console in the event of a computer failure.

Instrumentation and control cables are run from the machine locations to the local console and computer system and in some cases back to the Commu-
59 nications Center. Synchronous data links are run between the EVTM Computer and Central Computer, between the IVTM computer and Central Computer, and between the Central Computer and the Plant Data Handling and Display System. Cables are run through a flexible tray system along the gantry rails to the EVTM; and through flexible trays on the reactor head rotating plugs to the IVTM. Cables to the EVTM floor valves are run through the EVTM. All cables will be appropriately shielded and routed so as to minimize noise pickup from other cables or from other equipment (welding, etc.) which might be in operation during the refueling period.

The method of computer application (computer-generated setpoint control) used in the Fuel Handling and Storage Control System has several features which will tend to reduce operator errors and which facilitate the use
59 of an interlock system as described later in this section. Among these features is the computerized fuel inventory which enables cross-checking of every core

component movement prior to initiation. The type of core component is checked for compatibility with the intended destination. The destination for the core component is checked for occupancy and readiness to receive a particular core component. Core components can be identified by the IVTM to verify the type of core component prior to any movement into the reactor core or removal from the Reactor Vessel. The Central Computer monitors the operation of the other refueling machines and incorporates a software operational alarm system to add further depth to the design for operation without errors. The use of setpoint generation rather than direct digital control permits the IVTM and EVTM computer commands to be passed through a permissive hard-wired interlock system only if proper preconditions are met. In addition, the Central Computer monitors annunciator status and alarm failures. An alarm log can be displayed at all local computer CRT terminals.

Finally, a complete manual control capability is provided which also must work through the refueling interlock logic.

The analysis of the consequences of specific fuel handling events given in Section 15.5 has not identified a requirement for any specific safety interlocks.

Some interlocks are included in the design to preclude the possibility of major machine damage.

Typical interlocks are given below and in Table 7.7-1.

IVTM grapple/fuel element
EVTM grapple/fuel element
Rotating Plug drive system/IVTM grapple position
Rotating plug drive system/IVTM hold down sleeve
Rotating plug drive system/EVTM position
EVST drive motors/EVTM grapple position

Postulated Reactor Refueling System (RRS) accidents with potentially severe consequences were analyzed in detail to determine requirements for safety interlocks. The techniques employed included safety assurance diagrams, fault trees, mechanical and thermal analyses, and radiological release calculations. None of the analysis results showed off-site doses exceeding those presented in Section 15.5 or 15.7. The off-site doses in Section 15.5 and 15.7 resulting from postulated RRS accidents are all well below the 10 CFR 100 guideline exposures without taking credit for interlocks. It was there-
59 fore concluded that the RRS interlocks should not be designated as safety interlocks.

7.7.1.10 Nuclear Island Auxiliary Instrumentation and Control Systems

A number of Instrumentation and Control Systems, not discussed in Section 7.0, are provided in the plant to support various auxiliary systems. These systems do not perform a safety-related function, nor would their failure prevent the functioning of a safety-related system. These instrumentation systems, discussed in other sections of this report are:

System	Section
Recirculating Gas	9.16.5
Auxiliary Cooling Fluid	9.7.5
Inert Gas Receiving and Processing	9.5
Impurity Monitoring and Analysis	9.8.5
Auxiliary Liquid Metal	9.3

(This includes only those portions of the Auxiliary Liquid Metal system that are not associated with the Direct Heat Removal Service (DHRS) or the Spent Fuel Storage System (ex-vessel storage). The DHRS and the Spent Fuel Storage System are required for safety and their associated instrumentation and controls are discussed in Sections 7.6.3, 9.1.3 and 9.3.3).

7.7.1.11 Balance of Plant Instrumentation and Control Systems

A number of Instrumentation and Control Systems are provided to support various Balance of Plant Systems. These systems do not perform a safety-related function, nor would their failure prevent the functioning of safety-related systems.

7.7.1.11.1 Treated Water Instrumentation and Control System

The Treated Water System includes the Potable Water System, the Normal Plant Service Water System, the Secondary Service Closed Cooling Water System, the Emergency Plan Service Water System, the Normal and Emergency Plant Chilled Water Systems, and the Makeup Water Treatment System.

The Treated Water Instrumentation and Control System provides the instrumentation, control and indicating circuits and devices required for operation of the system. Section 9.9 covers this equipment.

Only the Emergency Plant Service and Chilled Water Systems are required for emergency safe shutdown after an accident or loss of AC power. Safety-related instrumentation for the Treated Water System is discussed in Section 7.6.1.

7.7.1.11.2 Waste Water Treatment Instrumentation and Control System

The Waste Water Treatment System provides for the treatment of non radioactive contaminated waste water, sanitary waste water and post-treatment facilities.

The Waste Water Treatment Instrumentation and Control System provides the instrumentation, control and indicating circuits and devices required for operation of the system.

7.7.1.11.3 Remaining Systems

Instrumentation and Control for the following systems are described elsewhere:

System	PSAR Section
Turbine Generator Control System	10.2.2
Feedwater and Condensate	10.4.7
Main Steam Supply	10.3
Circulating Water System	10.4.5
River Water Service System	9.9.7
Compressed Gas System	9.10
BOP Heating and Ventilating	9.6
BOP Fire Protection	9.13.1

7.7.2 Design Analysis

As described in Section 7.2.2, the Plant Control System is separate from the Plant Protection System; any failure of the Plant Control System will not affect the capability of the Plant Protection System but may result in a plant trip. Section 7.7.1 explains the interrelationship between the supervisory control and each of the plant sub-loop controls.

Where instrumentation signals are provided to the Plant Control System by the Plant Protection System (e.g., Nuclear Flux), multiple failures of the PPS sensors could cause the loss of instrumentation channels in both the Plant Protection and Control Systems. This could cause a Control System action that initiates a transient requiring Protection System action and could concurrently degrade the performance of one shutdown system. The consequences of this potential failure will be mitigated by diverse instrumentation in the second Reactor Shutdown System which, being independent, is unaffected by the sensor failures.

Postulated failures for the Plant Control System, their actuators, and sensors and the features included to mitigate results of these failures are described below.

7.7.2.1 Supervisory Control System

The function of the Supervisory Control System shown in Figure 7.7.2, is to relate the plant load demand to the second level (subloop) control system demands and to provide trim of the subloop controls to achieve the desired temperature or pressure operating conditions. Failures of this control system could result in either a combination of misdirected subloop control system demands, or a consistent, but erroneous, set of subloop control system demands.

The first case may be caused by a failure of at least one section (e.g., one or more programmers, one or more sensors, etc.) of the supervisory control. This will result in some of the subloop controllers being directed away from their desired profiles, while others would be controlling normally. An obvious result of this failure mode is a mismatch of some key plant variables. For instance, if the intermediate flow of a single HTS loop is at its desired flow and the primary flow is directed otherwise, the intermediate flow to primary flow ratio would be incorrect. The Plant Protection System would then trip the plant based on this erroneous ratio. In general, failures of this type, would result in activation of those Plant Protection subsystems which are based upon a ratio or mismatch of plant variables.

In the second type of failure mode, it is assumed that all plant variables are maneuvered in such a way that no mismatch occurs but that the general direction or rate of the control demands are wrong. This would result from a misinterpretation of the plant load demand or a gross failure of the entire Supervisory Control System. In general, those Plant Protection subsystems based on single variables (i.e., high flux, flux-delayed flux) would be activated under these conditions.

The Supervisory Control design uses multiple sensors and average/reject, auctioneer, or medial select circuitry to minimize the possibility that single sensor failure will result in inappropriate control system action. Failures in the electronic controllers can only affect the plant at the rate of change of the actuators (pump drives, control rod drive mechanisms, etc.). As shown in Chapter 15, the Plant Protection System acceptably terminates the results of all incidents involving incorrect actuator response. Therefore, the Supervisory Control System is inherently incapable of initiating a transient which is more severe than the PPS design basis.

7.7.2.2 Reactor Control System

The Reactor Control System shown in Figure 7.7-3 contains an Outer Core Exit Temperature Controller with an inner loop based on flux feedback. Failure in this system could result in erroneous movement of the control rods. This could result from failure in the sensors or feedback signal conditioning (i.e., flux or temperature), failures in the controller electronics, or a failure in the CRDM controller. The Reactor Control System has redundant sensors and average/reject or median select circuitry to prevent single sensor failures from initiating reactivity transients. Even though it is highly improbable, simultaneous multiple failures in PPS compensated ionization chamber instrumentation could cause the loss of flux instrumentation channels in both the Plant Protection and Control Systems, the consequences of this potential failure will be mitigated by diverse fission chamber instrumentation in the secondary reactor shutdown system. Rod withdrawal block circuitry and a rod position rod block system are provided which are independent of the normal control to prevent control electronic failures from causing reactor trip. Withdrawal blocks are initiated for both high power level and power-to-flow deviation. These withdrawal blocks operate directly on the Control Rod Drive Mechanism Controllers to stop outward rod motion of all primary rods. Withdrawal and insertion blocks are initiated by the rod position rod block system to prevent severe misalignment or incorrect positioning of the control rod bank. The Control Rod Drive Mechanism Controller and Rod Sequencer include overspeed detector and block circuitry to provide assured limitation of rod withdrawal speed even if reactor control failures and failures of the rod block or overspeed circuitry are postulated. The PPS acceptably terminates the results as shown in Chapter 15.

It is also considered that a failure of the Reactor Control System could result in improper banking of the control rods which is not severe enough to require action by the rod position rod block system. Under these conditions the reactor operator would have to readjust the out of bank rods manually. To aid the operation, the main control board is equipped with rod position indications for each rod and also an alarm if the rods deviate from the proper banking requirements.

7.7.2.3 Sodium Flow Control System

A block diagram of the Sodium Flow Control System is given in Figure 7.7-7 which is typical of the six HTS flow loops. The controller contains an outer flow loop with an inner loop based on pump speed. A failure of any of the six flow controllers would result in improper pump speed and, consequently, undesired sodium flow. Power to flow or primary to intermediate flow mismatch would occur resulting in a plant trip. Even though it is highly improbable, multiple failures in PPS flow instrumentation could cause the loss of flow instrumentation channels such that the secondary RRS fails to trip; the consequences of this potential failure to initiate control system action which requires Protection System action will be mitigated by the primary reactor shutdown system. Pump speed instrumentation; this is independent of the flow instrumentation and is, therefore, not affected by these failures.

7.7.2.4 Steam Generator Feedwater Flow Control System

The Steam Generator Feedwater Flow Control System contains a feedwater flow controller (typical of three). Failure of the controller would result in improper actuation of the Feedwater Control Valve and, consequently, an undesirable feedwater flow. This, in turn, would generate a change in drum level and a mismatch in steam flow to feedwater flow. The Plant Protection System has protective subsystems for each of these anomalies.

7.7.2.5 Balance of Plant Instrumentation and Control

The Balance of Plant Instrumentation and Control Systems described in Section 7.7.1.11 do not perform safety-related functions, nor would their failure prevent the functioning of a safety-related system.

The most severe transient introduced because of these controls is a turbine trip; the transient is described in Section 15-3.

Plant Fire Protection Instrumentation and Control System failure would not prevent the functioning of a safety-related system. Accident analysis for conventional and maximum fires is provided in Section 15.6 and 15-7.

59 TABLE 7.7-1

USE OF REFUELING INTERLOCKS

Event*	Cause	First Line of Defense	Second Line of Defense
Rotation of one or more rotating plugs with assembly partially inserted into core or storage location	Operator error when in manual control	Interlock prevents energizing rotating plug drive motors unless IVTM grapple and holdown sleeve are fully raised.	Redundant switch independently prevents energizing rotating plug drive motors unless IVTM grapple and holddown sleeve are fully raised.
Rotation of one or more rotating plugs while EVTM is over reactor	Operator error when in manual control	Interlock prevents energizing rotating plug drive motors if EVTM is within 14 ft of the reactor transfer port position, as indicated by rail switches.	Redundant interlock prevents energizing rotating plug drive motors if EVTM is within 14 ft. of the reactor transfer port position, as indicated by rail switches.

44 *Occurrence of Event could cause machine damage, but no unacceptable hazard to public health and safety.

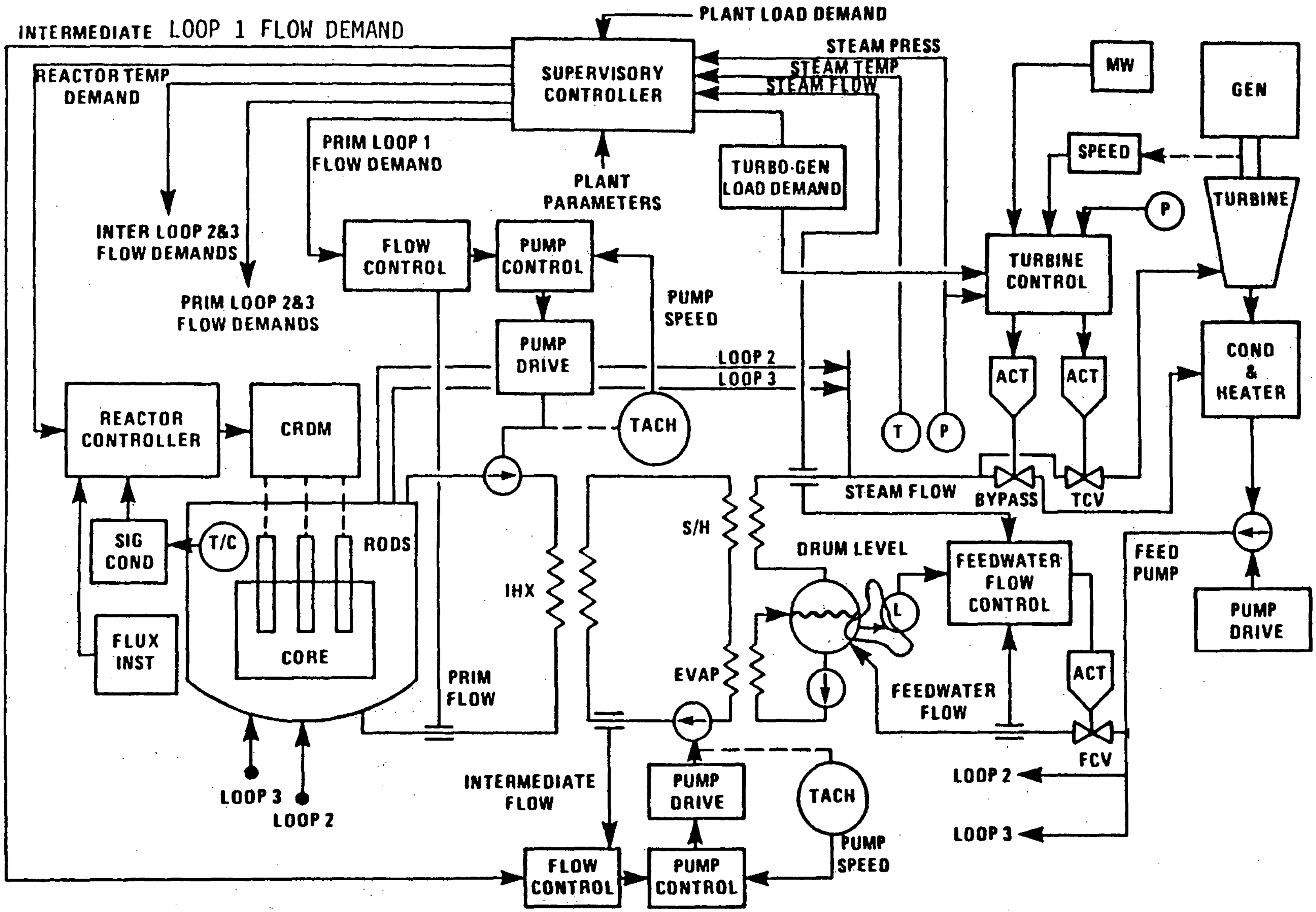

Figure 7.7-1. Plant Control System

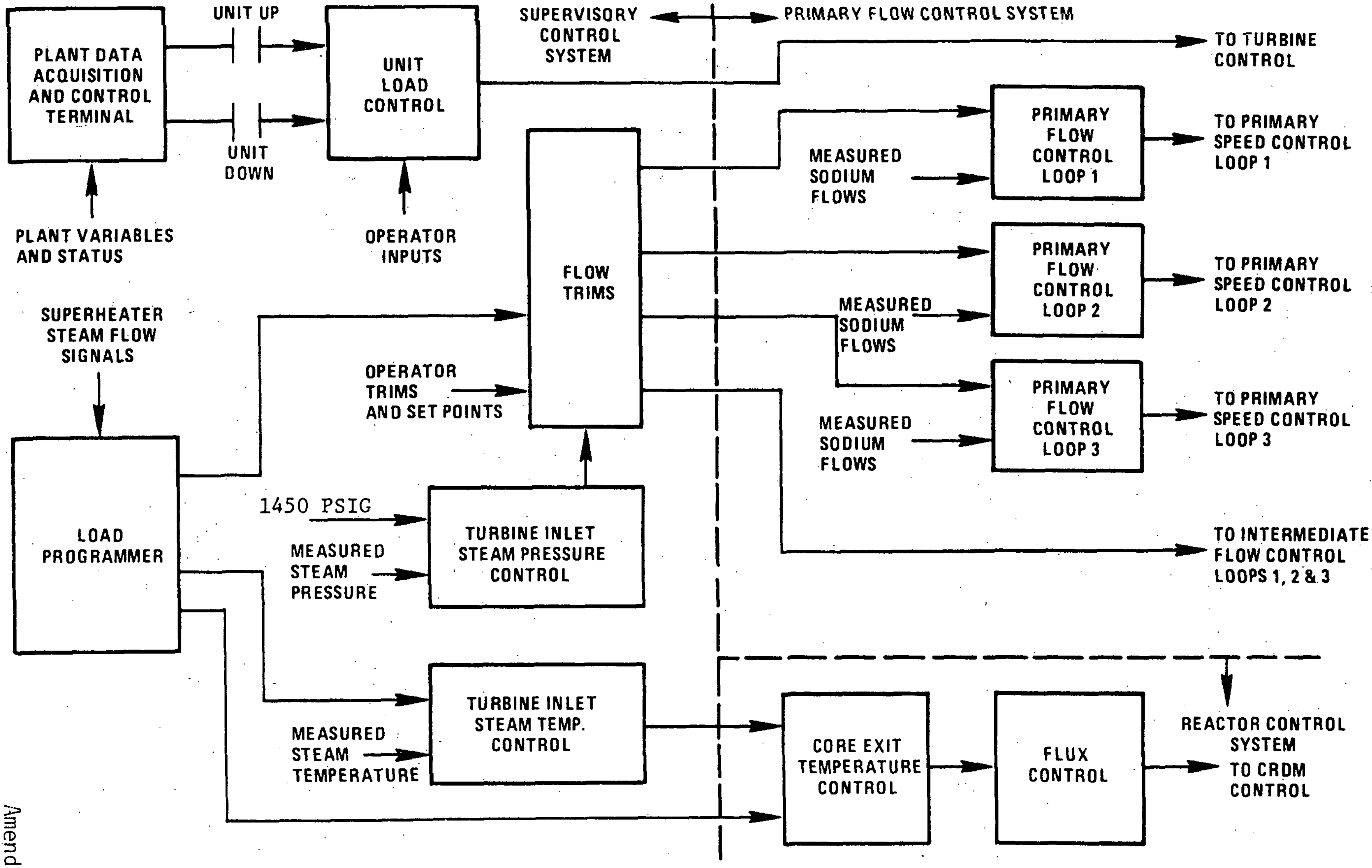

Figure 7.7-2 Supervisory Control System

8266-3

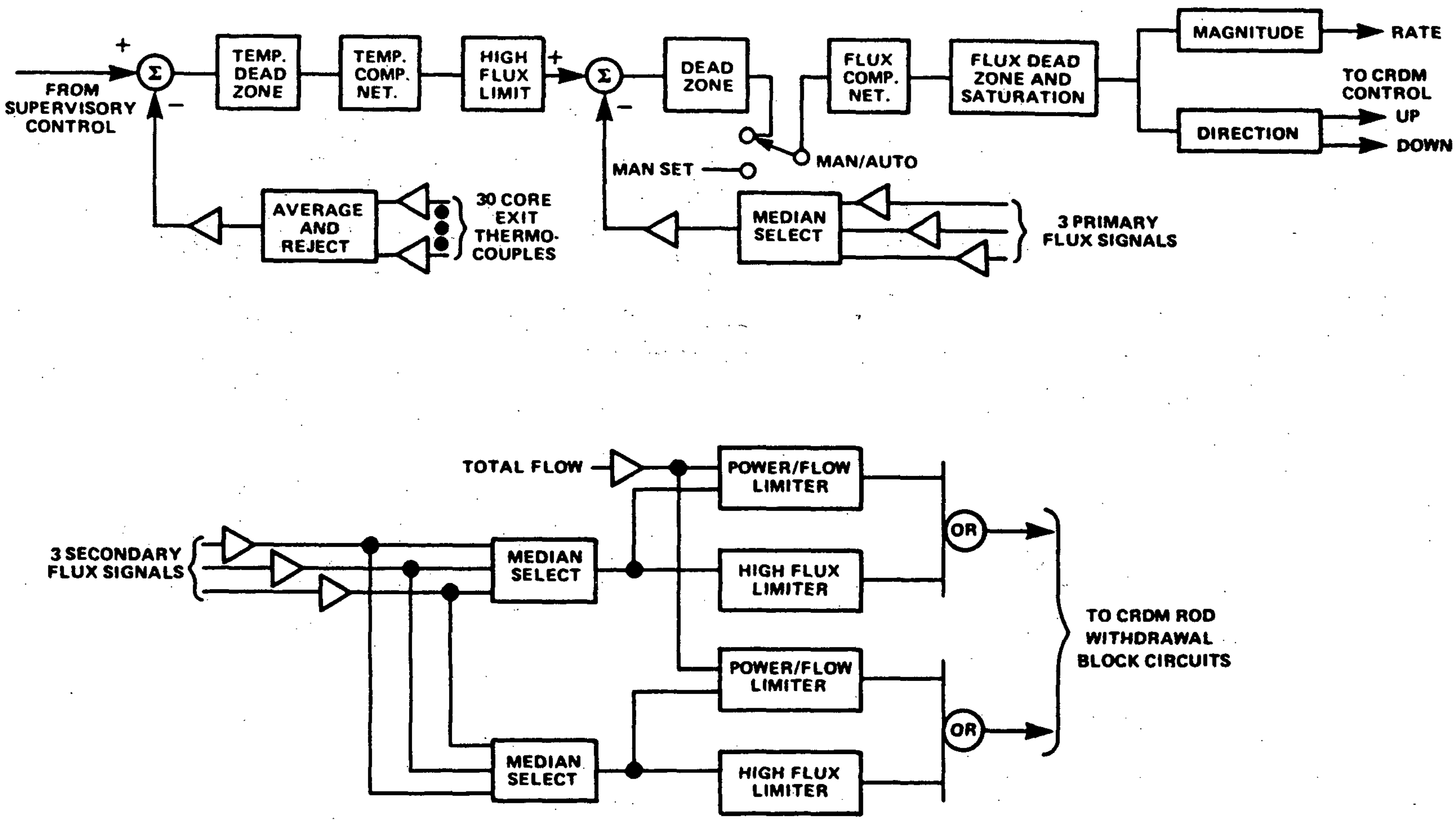

FIGURE 7.7-3. BLOCK DIAGRAM OF REACTOR CONTROL SYSTEM

8266-2

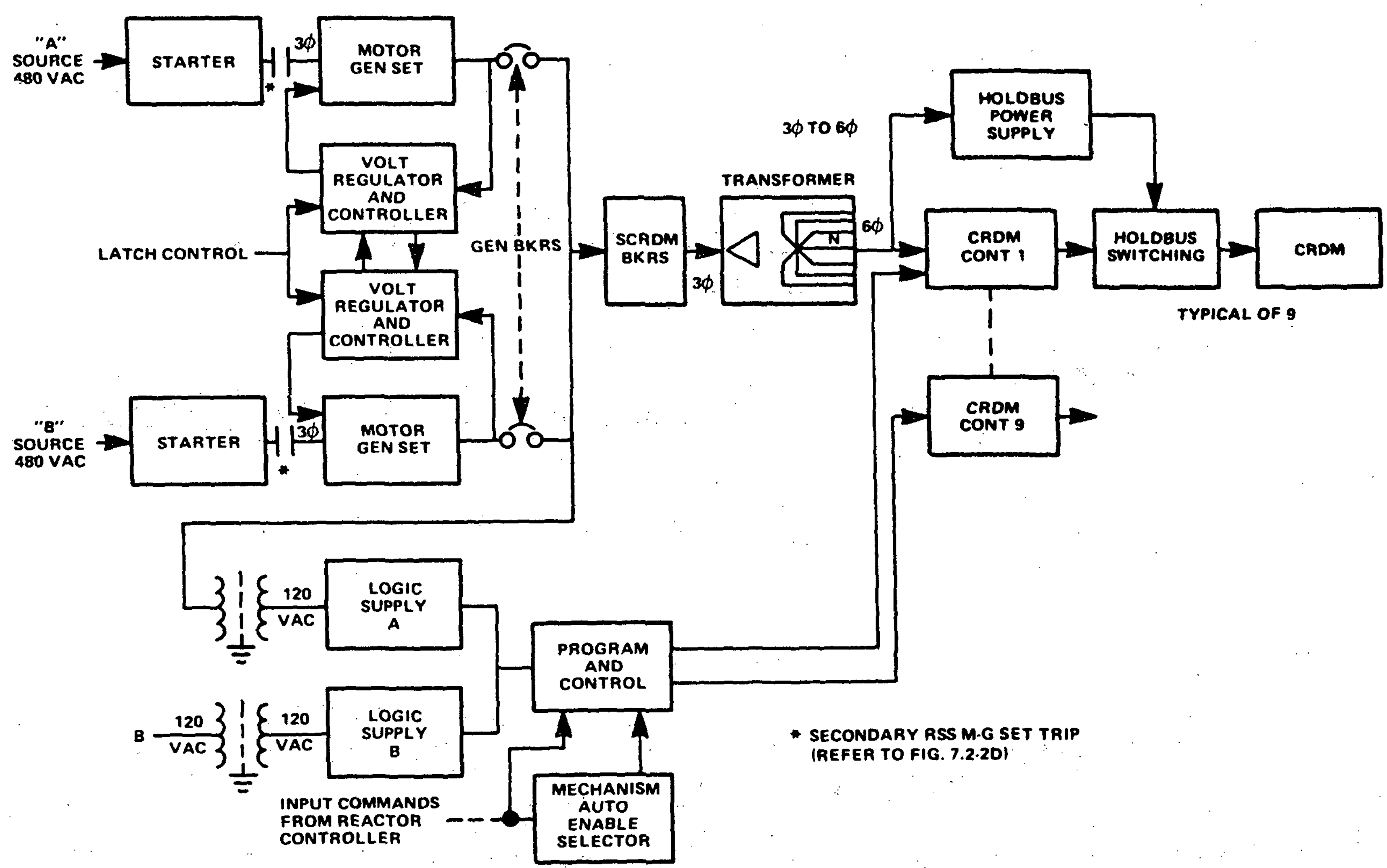

FIGURE 7.7-4. CRDM CONTROLLER AND POWER TRAIN FOR PRIMARY RODS

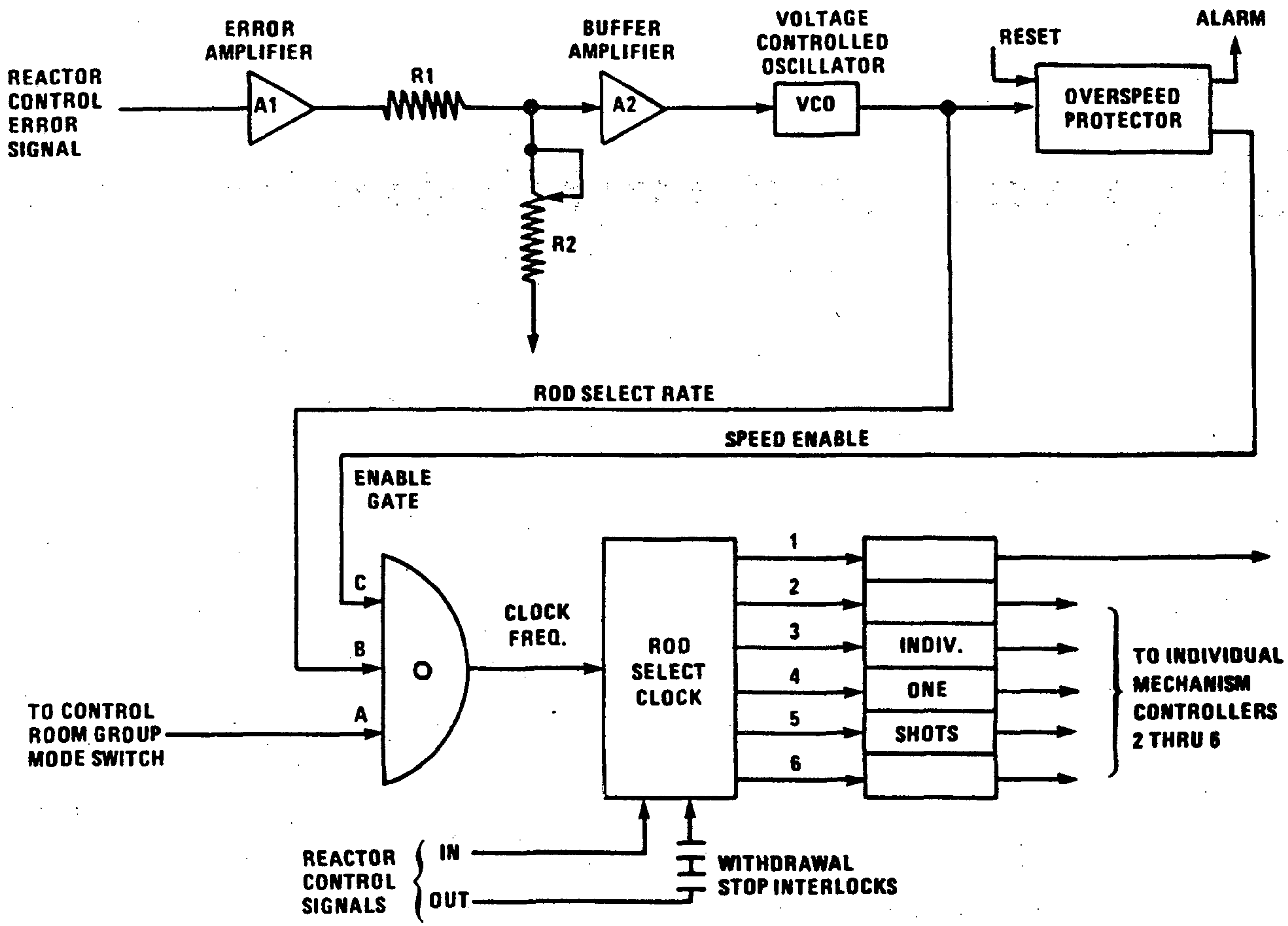

FIGURE 7.7-5. BLOCK DIAGRAM OF PRIMARY ROD GROUP CONTROL

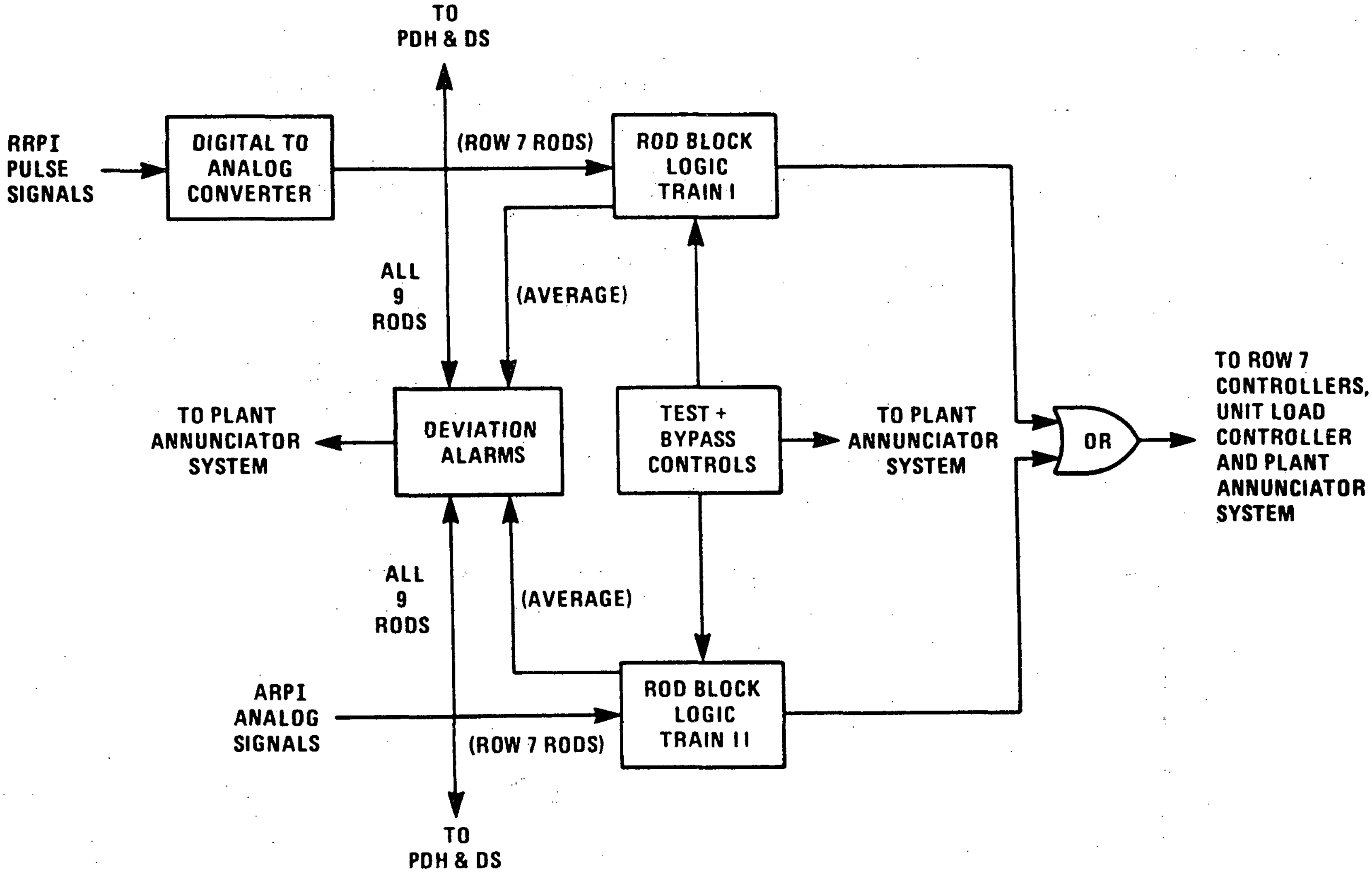

FIGURE 7.7-6 GENERAL BLOCK DIAGRAM FOR THE ROD POSITION ROD BLOCK SYSTEM

8266-1

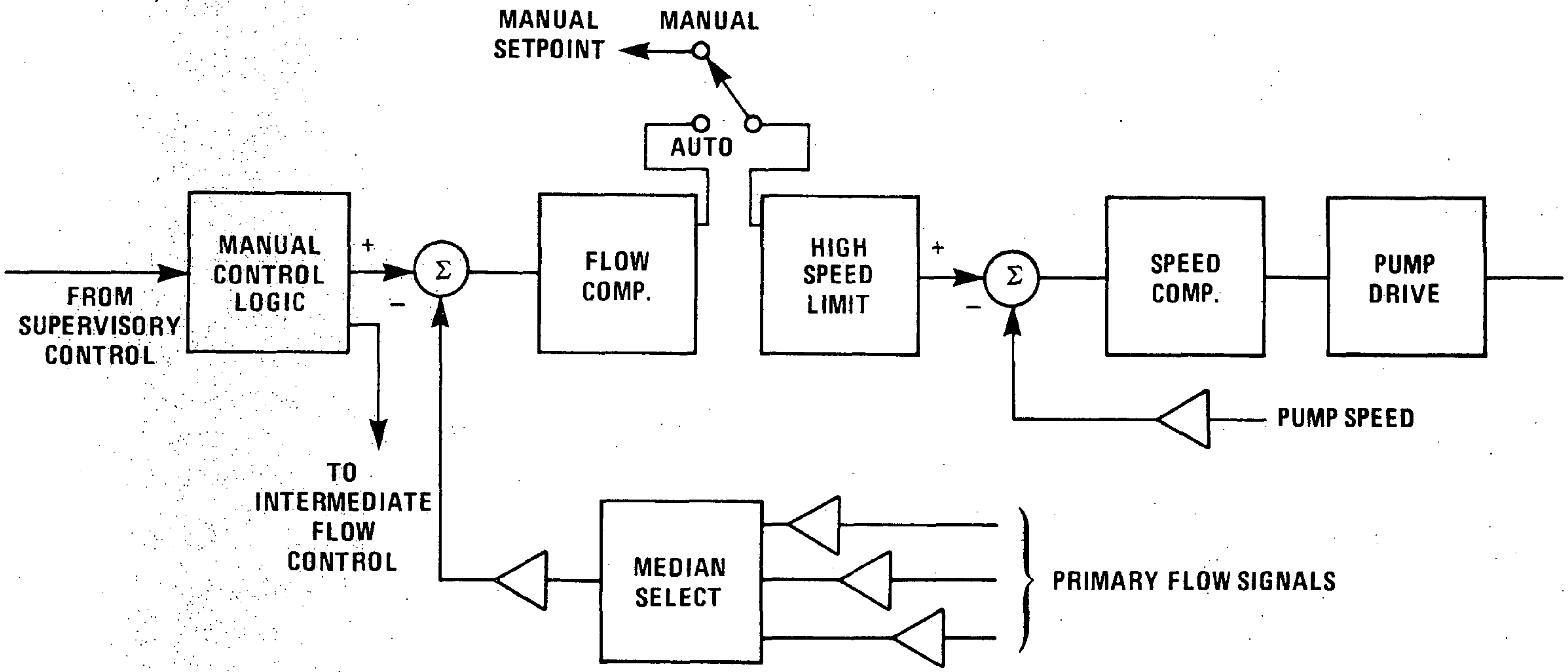

Figure 7.7-7. Sodium Flow Control System Flow/Speed Control

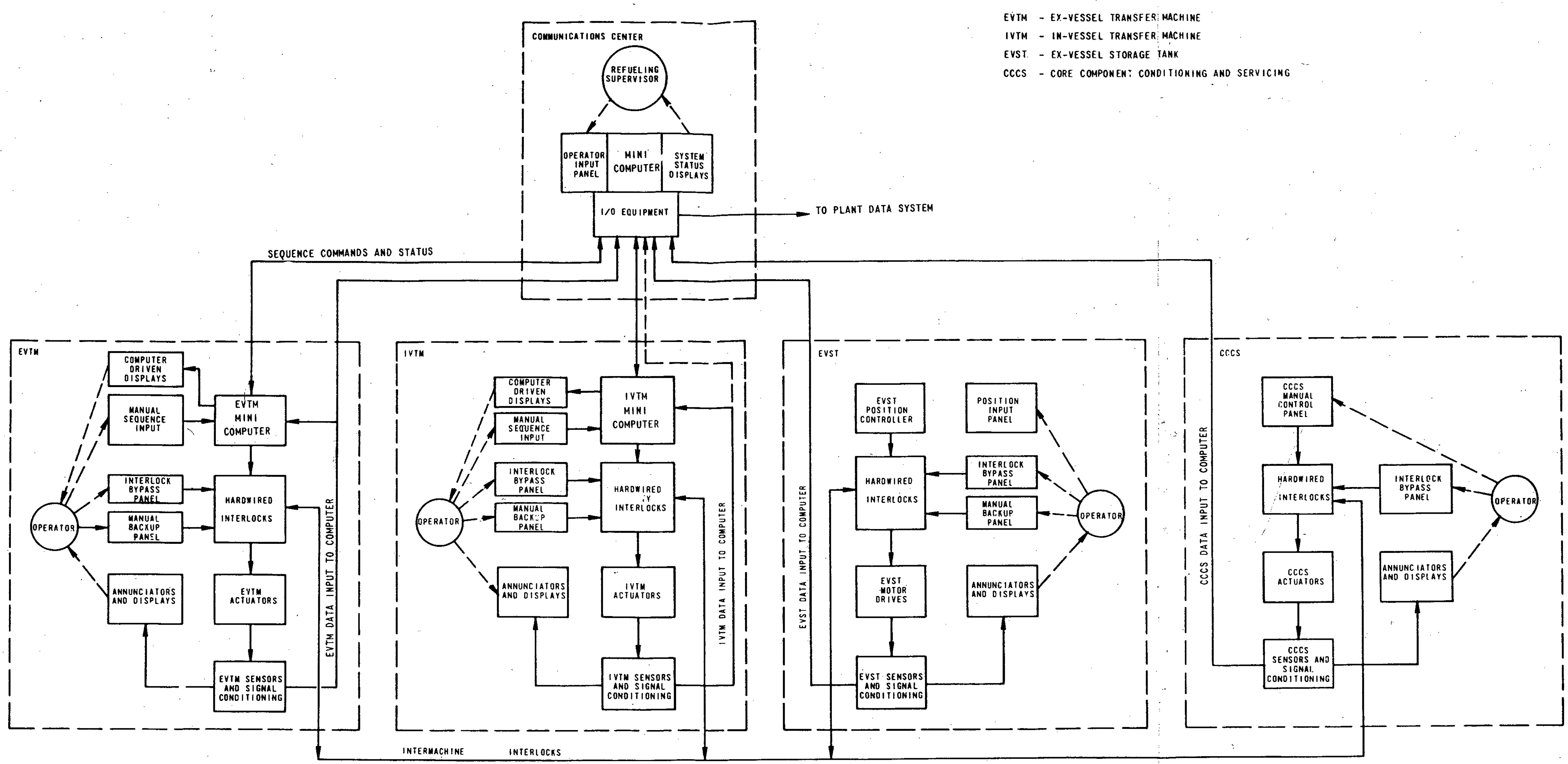

FIGURE 7.7-8. FUEL HANDLING AND STORAGE CONTROL SYSTEM

7.8 PLANT DATA HANDLING AND DISPLAY SYSTEM

7.8.1 Design Description

The Plant Data Handling and Display System PDH&DS is a distributive computer system which supports plant operations and performance by monitoring, limit checking, trending, and displaying plant information. It supplements other monitoring and displaying systems including Plant Annunciation and Plant Control. Other plant systems provide sufficient instrumentation and control to the operator such that the PDH&DS is not required for startup, operation, or shutdown of the plant. However, additional requirements may be placed on the plant operations should the PDH&DS be inoperative.

The PDH&DS uses redundant Central Processing Units (CPUs) to process plant information acquired from numerous Remote Data Acquisition Terminals (RDATs) located in various areas of the plant. Information is provided to the operator via Cathode Ray Tube (CRT) and hard copy units located in the control room. A block diagram of the PDH&DS is shown on Figure 7.8-1; Figure 7.8-2 depicts the system arrangement.

Specific functions of the PDH&DS include:

- Monitoring of most plant variables and alerting the operator when any selected variables exceed predetermined limits.
- Recording of the operating history of the plant.
- Performance parameter calculations including:

  overall plant heat balance

  reactor and equipment calorimetries

  plant protection system channel output monitoring

  shutdown margins

  control rod worth

  core assembly exit temperatures

  reactivity calculations

  sodium inventory calculations

  component efficiencies

- Providing CRT display units in the control room to present pertinent plant data for surveillance of plant protection and control systems, heat transport systems, SGAHRS, auxiliary systems and balance of plant.

- o Displaying measurements of variables not otherwise displayed in the control room to reduce the number of trips by plant personnel to read local indicators. This function provides a simple, efficient method for the operator to respond to group alarms.
- o Providing a mechanism to forewarn the operator of potential harmful conditions. Examples of these include high bearing temperature, detection of small sodium leaks and radiation levels. If the condition deteriorates further, the operator will be warned by the annunciator system.
- o Providing pre and post trip information for review.
- o Provides for acquisition of data for design verification of plant components.
- o Providing the above information to the operator via an integrated combination of state of the art human engineered color CRT displays (schematic diagrams, parameter lists, trend plots and other graphic representations as appropriate) and printed output.

7.8.2 Design Analysis

The PDH&DS is designed for high system availability by utilizing redundancy in processing and display.

The main processing part of this system is located adjacent to the control room in the computer room. Information generated by the DH&DS is presented in the control room. The data acquisition components of the system are located near sensor local panels. The data acquisition components multiplex sensor signals to reduce the number of control room panels and associated cabling.

Although the PDH&DS is designed for an extremely high availability, operating procedures of systems which normally use PDH&DS capabilities are written to allow operation of the plant with manual data recording and calculations should the PDH&DS not be available.

The operator interacts with the PDH&DS to obtain information, display and calculations via multiple CRTs and hard copy devices. CRTs are positioned on the operator's desk and at eye level on the Main Control Panel for convenient use by the operator. Human factors engineering is used in the displays (color, symbology, density, organization, format, etc.) and the displays are integrated with the Main Control Panel and operating procedures.

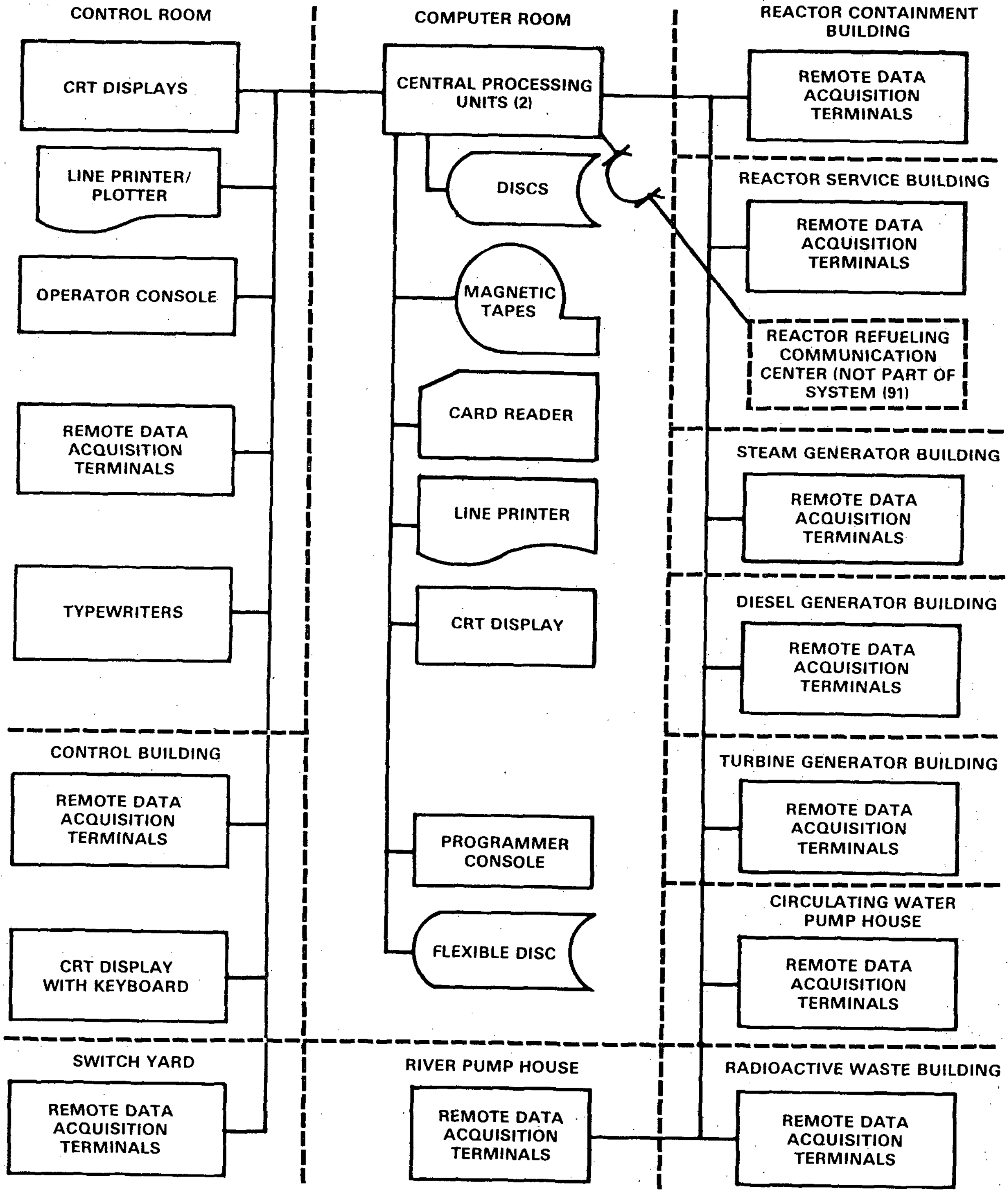

Figure 7.8-1 Plant Data Handling and Display System Schematic

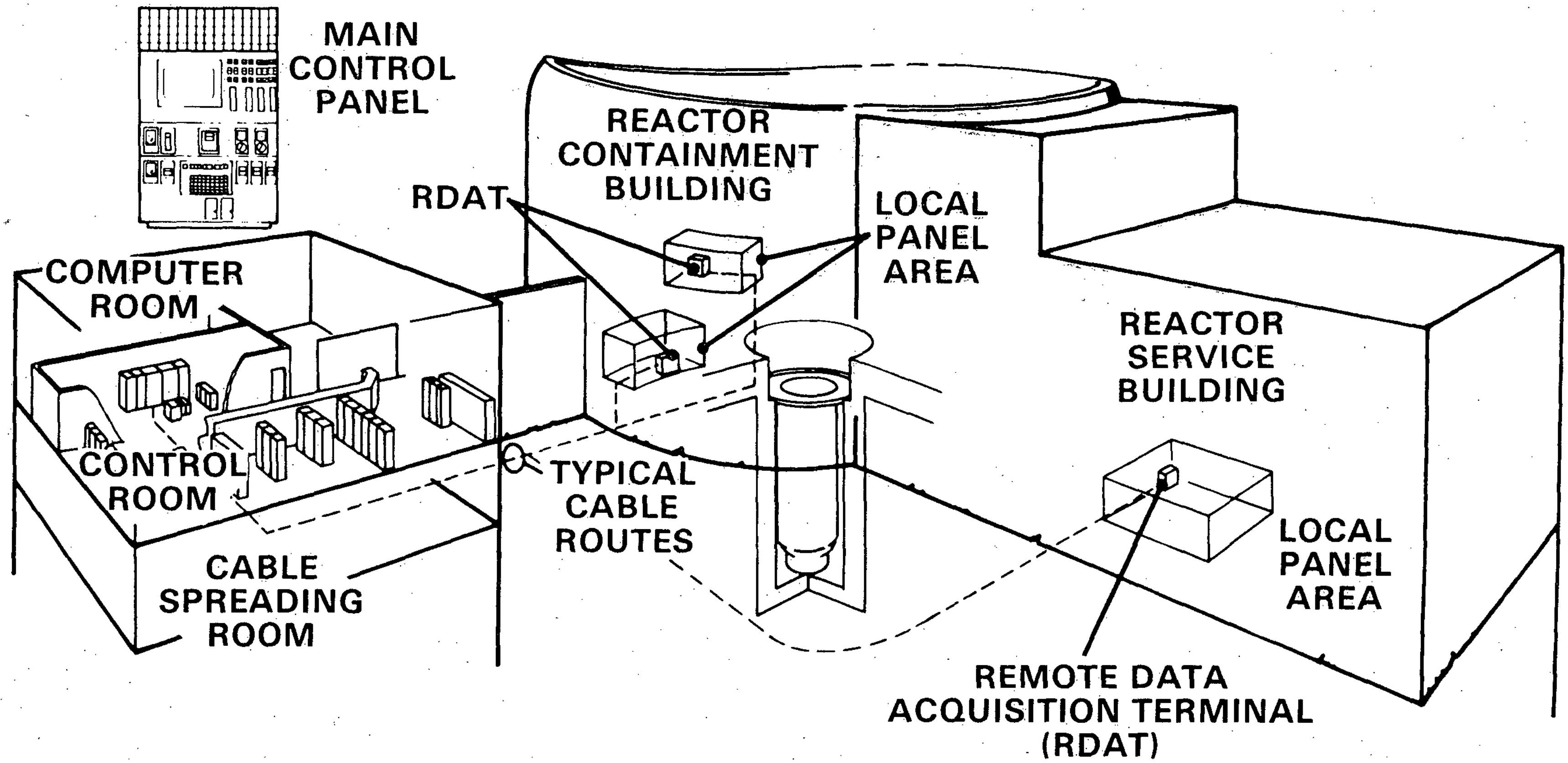

Figure 7.8-2 Plant Data Handling and Display System Arrangement.

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7.9 OPERATING CONTROL STATIONS

7.9.1 Design Basis

A control room is provided from which action can be taken to operate the nuclear power station safely under normal operation and off-normal conditions and to maintain it in a safe condition under postulated accident conditions. Adequate radiation protection is provided to permit access and occupancy under Extremely Unlikely Fault conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the postulated accident. The control room provides protection from substances such as sodium oxide which might be released to the local environment under Extremely Unlikely Fault conditions.

The basic criteria for inclusion of displays or controls in the control room shall be:

- The displays or controls necessary to support all normal plant operating conditions;
- The displays and controls necessary to respond to off-normal or casualty conditions which impact on power operations capability;
- The displays or controls necessary to prevent potential radiological hazards to offsite personnel;
- The displays necessary to the operator for detection of fire hazards; or
- The display and controls necessary to prevent potential damage to the plant.

The control boards are arranged functionally based on normal and off-normal operational considerations to minimize the number of operators required and to enhance the capability of the operational personnel to monitor and assure the safe status of the plant during all operations.

Remote control stations are provided outside of the control room to shut the plant down and maintain it in a safe condition assuming loss of control
57 room habitability. Access to the control room is controlled by card key to assure that only qualified personnel use the equipment provided to monitor and maneuver the reactor plant.

7.9.2 Control Room

7.9.2.1. General Description

The Control Room is located in the Control Building at approximately grade level. The Control Room proper occupies approximately a space of

57 70' x 75'. The remainder of the floor is devoted to auxiliary services for the control room operating personnel. The Control Building provides the necessary structural and atmospheric protection to allow continued habit-
57 ability that collectively satisfy CRBRP General Design Criterion 17. These features are summarized in Section 3.A.3.

The indicators, annunciators, and controls included in the Control Room provide the capability to operate the plant through all normal operational sequences and to respond to off-normal or emergency conditions without continuously manned remote stations.

7.9.2.2. Control Room Arrangement

The control room is arranged to provide an effective interface between the plant and the operating personnel (refer to Figure 7.9-1). Frequently used safety related instrumentation and controls are located on the
57 Main Control Panel. This equipment is grouped by operational category to assure that determination of plant condition and action to correct the condition are in close proximity. Less frequently used equipment and certain electronic equipment for which access control is desired are located in the rear panel area.

7.9.2.3 Main Control Board Arrangement

As shown in Figure 7.9-1, an open U-shaped main control panel is provided. The main control panel and the area it encloses form the central operating area for the plant. Equipment on the main control panel is arranged functionally and according to the power generation flow path. From left to right, the main control panel sections are as follows: emergency systems, plant protection system and engineered safety features, plant control and primary heat transport systems, intermediate heat transport and steam generator systems, steam generator auxiliary systems, turbine system and generator and switchyard. The equipment is arranged with annunciators at the top and display, controls, and switches in functional groups on the vertical and sloping bench sections. The size and arrangement of
57 equipment is based on the following guides:

- Displays, annunciation, switches, and control necessary to operate the plant without continuously manned remote stations are located on the main control panel or displayed through the Plant Digital Data Handling System and the cathode ray tube
57 (CRT) displays.

- Graphic or mimic displays are provided where warranted to enhance the operator/plant control interface and minimize the chances for inappropriate operator action.

- Physical separation of redundant safety related instrumentation equipment is incorporated.

- Physical, color, and geometric differentiation of displays and controls mounted on the board is provided to assure ease of recognition by operating personnel and minimize the chances for inappropriate actions.

- Arrangement and design of displays and controls is specified to provide arrays which permit determination of proper instrument comparison at a glance, where practical.

- Modular design of switches, controls, and indicators is used to permit ease of maintenance and minimum interference with operation. The equipment included on the main control board is summarized below (refer to Figure 7.9-1 and Table 7.9-1).

The arrangement of the instrumentation and control devices on the main control panel is as follows:

<u>Section A - Emergency, Plant Protection System, Engineered Safety Features and Plant Control Systems</u>

- Emergency Chilled Water
- Emergency Plant Service Water
- Reactor Shutdown
- Containment Isolation
- Sodium Water Reaction Pressure Relief Sub-System Status
- Sodium Dump
- Control Room Heating, Ventilating, and Air Conditioning
- Containment Instrumentation
- Flux Monitoring
- Primary and Secondary Manual Scram Switches
- Supervisory and Reactor Control
- Reactor Instrumentation
- Rod Control and Rod Position Indication

Section B - Primary and Intermediate and Steam Generator System Heat Transport Systems

- Primary Heat Transport
- Intermediate Heat Transport
- Steam Generator
- Feedwater and Auxiliary Flow

Section C - Feedwater, Conensate, Auxiliary Steam Generator, Turbine, Generator and Switchyard Systems

- Condensate
- Auxiliary Feedwater Pumps
- Protected Air Cooled Condenser
- Steam Generator Auxiliary Heat Removal Vent Controllers
- Turbine Control Panels
- Turbine Instrumentation
- Turbine Steam Bypass
- Circulating Water
- AC Bus Circuit Breaker Control
- Generator Syncroscope

The Following instrumentation and control panels, while not a part of the Main Control Panel, demand rapid operator response and have been arranged to permit operator scanning from the Main Control Panel:

- Failed Fuel Monitoring
- Sodium Leak Detection
- Sodium Fire Detection
- Non-Sodium Fire Detection

- Control Building Fire Detection
- Emergency Diesel Generators
- Switchyard and Station Electrical Distribution
- Direct Heat Removal Service

The layout of Section A of the main control panel is designed to minimize the time required for the operator to evaluate the system performance under accident conditions. Deviations from predetermined conditions are alarmed and/or indicated so that corrective action may be taken by the operator.

The control room also includes control and instrumentation equipment that is used infrequently or for which controlled access is desirable. Included in this control room back panel area are power distribution, chilled water, containment instrumentation, recirculating gas, heat transport, steam generator, heating ventilation and air conditioning, annunciator electronics, turbine, balance of plant, plant control, plant data handling and display system multiplexers, flux monitoring, radiation monitoring, reactor shutdown and containment isolation panels.

7.9.2.4 Main Control Board Design

The Main Control Panel is an open U-shaped, stand up vertical panel as shown in Figures 7.9-1 (plan view) and 7.9-2 (side view). There are 3 significant features of the control board mechanical design: seismic capability; separation of redundant safety related equipment and wiring; and modular construction of switch, indicator and control equipment.

Since the Main Control Panel includes safety related equipment, the sections including this equipment are designed to Seismic Category I and qualified in accordance with IEEE Std. 323 and IEEE Std. 344. Structures, wiring, wireways, and connectors are designed and installed to ensure that safety related equipment on the control panel remains operational during and after the SSE. The Main Control Panel is constructed of heavy gauge steel within appropriate supports to provide the requisite stiffness.

Within the boundaries of the Main Control Panel Sections, modules are arranged according to control functions. Fire retardant wire is used. Modular train wiring is formed into wire bundles and carried to metal wire ways (gutters). Gutters are run into metal vertical wireways (risers). The risers are the interface between external wire trays feeding the panel and Main Control Panel wiring. Risers are arranged to maintain the separated routing of the external wire trays. (See Figures 7.9-3 and 7.9-4).

Mutually redundant safety train wiring is routed so as to maintain separation in accordance with the criteria of IEEE Std. 384. A minimum of six inches air separation is maintained between wires associated with different trains. Where such air separation is not available, mechanical barriers are provided in lieu of air space.

The Main Control Panel protection system circuits are designed and selected to ensure that system performance requirements are met and channel integrity and independence are maintained as required by IEEE Std. 279. Power division separation and isolation are maintained in accordance with the requirements of IEEE Std. 308.

7.9.3 Local Control Stations

Local control panels are provided for systems and components which do not require full time operator attendance and are not used on a continuous basis. In these cases, however, appropriate alarms are activated in the Control Room to alert the operator of an equipment malfunction or approach to an off-normal condition.

7.9.4 Communications

Communications are provided between the Control Room and all operating or manned areas of the plant. In addition to public address and interplant communications and the private automatic exchange (used for in-plant and external communications) a sound powered maintenance communication jacking system is provided. Redundant and separate methods of communication between the control room and other TVA generating plants is also provided.

7.9.5 Design Evaluation

Following the Three Mile Island accident, a large task force was formed for the purpose of performing a thorough review of the CRBRP Control Room design. This overall review was divided into three parts; a planning phase, a review phase, and assessment and implementation phase. Following the task force effort, NUREG-0700 was issued. NUREG-0700 is similar in intent to the CRBRP Control Room design evaluation.

7.9.5.1 Planning Phase

In the planning phase the objectives and scope of the task force were identified, and criteria were established for personnel selection. A charter was developed which contained the scope and objectives, and personnel selection was accomplished.

The task force charter required a review of the Control Room design and the operating procedure outlines to ensure that the systems designs, the integration of the systems, and the man-machine interfaces properly supported safe operations of the plant during both normal and abnormal conditions. A task analysis was established for observing the operator conducting various duties. Specific items included in the review are:

1. Overall Control Room and individual panel designs and features, and their interface with the operator.

2. System and overall plant operating procedure outlines.

3. Administrative approaches for plant operations.

4. Recommendations from other Key System Review Task Forces.*

5. Recommendations made by NRC and other parties as a result of the Three Mile Island occurrence.

6. Computer utilization by the operators.

7. Operator training requirements.

8. Remote shutdown capabilities and safety system status indication in the Control Room.

Criteria were established for personnel selection of those to participate on the task force. Nuclear experience was considered necessary in the areas of design, analysis, operations, testing, maintenance, and training. Personnel whose background included sodium plants and light water plants were selected. Licensed and qualified operators were considered mandatory. Personnel with human factors education and experience both inside and outside the nuclear industry were included.

Human factors considerations were emphasized in the planning phase. Previous Control Room design efforts had attempted to optimize the man-machine interface. However, a major objective of the Control Room Task Force was to re-evaluate this interface. Prior to the evaluation effort a seminar was held, under the direction of three leading human factors personnel, to teach the Task Force disciplined methods for considering human factors. Based on this training and further assistance from human engineers, check lists were prepared to evaluate the man-machine interface.

*See Reference 7.9-1

7.9.5.2 Review Phase

In the review phase extensive analysis of plant events were conducted. Functional analyses were made of the operator in his response to automatic equipment actions, manual actions which had to be performed in the Control Room, and manual actions required by operators external to the Control Room. More than 200 walk-throughs of plant events were conducted.

The Control Room design and operating instructions were thoroughly reviewed in four areas:

1. Proper identification of systems to be operated from the Main Control Room.

2. Proper staffing of the Control Room.

3. Proper overall layout of the Control Room to enhance the man-machine interfaces and support the integrated operation of plant's systems.

4. Proper layout and design of individual Control Room panels, instruments, indicators, and controls to enhance the man-machine interface and support the integrated operations of the plant's systems.

A full scale mockup of the Control Room was used. The events chosen to be evaluated were carefully selected so they would umbrella all of the operations that are either expected to occur or might be postulated to occur over the life of CRBRP. The off-normal events include plant responses to single and multiple failures.

The methodology of performing this review consisted of using three groups of people; simulators, operators, and evaluators.

The Simulators analyzed the events which were to be evaluated prior to the walk-throughs and then, during the walk-through evaluations, simulated the control panel indicators. Some of these events had previously been analyzed via computer while other events required additional computer runs to enable mocking up the panel as it would appear to the operator. The control panels were mocked up by the Simulators to represent the changing plant conditions and the information flow into the Control Room during the event. This made the walk-through as realistic as possible.

The Operators played the part of the Control Room operators and carried out the steps of the procedure being evaluated. They touched each switch they were required to operate, and observed each indicator which was part of the particular event.

The Evaluators included a Human Factors Engineer and a Systems Engineer. Their function was to fill out the Operating Sequence Diagram and the evaluation sheets for each procedure and event reviewed.

As problems or concerns were encountered, recommendations were made. These were, in some cases, of a broad nature and reflected the need for reconsideration of decisions made in the four most important evaluation areas described above. Other problems and concerns related to specific details of the Control Room design or the procedure outlines.

7.9.5.3 Assessment and Implementation Phase

The evaluation and implementation of the recommendations started with a check of the consistency of all of the recommendations by the task force. Small models of the overall Control Room and Main Control Panel were made assuming all recommendations were incorporated into the design. The recommendations were modified based on the small model to provide a coordinated and consistent set of final recommendations. Senior Project Management reviewed the final set of recommendations and issued them to the Project line organization for assessment and implementation. The cognizant design engineers have two choices. They can either accept the recommendation if it is valid, and include it into the plant design via normal procedures, or reject the recommendation and provide adequate justification if the recommendation is invalid. Each assessment is reviewed and approved by senior project management.

7.9.5.4 Conclusions

The Control Room Task Force Design Review is documented in further detail in Reference 7.9-1. In September 1981, NUREG-0700 entitled "Guidelines for Control Room Design Review" was issued. A comparison between these two documents leads to the conclusion that although NUREG-0700 was issued after the Control Room Task Force Review, the intent of the NRC in promulgating NUREG-0700 is similar to the Project's intent in performing the Control Room Task Force Review, and the intent of NUREG-0700 is believed met by CRBRP.

7.9.5.5 Additional Committments Regarding Control Room Design

- A study of techniques to reduce alarm overload and assess their applicability to CRBRP alarm systems, will be submitted in conjunction with the FSAR.

- The main control panel layout, a completed human factors check list for the MCP (addressing applicable portions of Section 6 of NUREG-0700 as guidelines) and a summary report of the control room design process showing equivalence to NUREG-0700 Appendix B, will be submitted three months prior to initiation of main control room hardware fabrication.

- The functional basis of computer graphics and examples of typical top level graphics will be made available for audit prior to main control room panel fabrication.

Reference:

1. Summary Report on the Conduct of the Clinch River Breeder Reactor Plant (CRBRP) Key System Reviews, February 1982.

57 PAGES 7.9-7 and 7.9-7a

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Table 7.9-1

CONTROL ROOM ARRANGEMENT

Item	
1.	Main Control Panel Section A
2.	Main Control Panel Section B
3.	Main Control Panel Section C
4.	Not Used
5.	Not Used
6.	Not Used
7.	Class IE Division 2 Panel
8.	Class IE Division 3 Panel
9.	Class IE Division 1 Panel
10.	Switch Yard and Station Electrical Distribution
11.	Not Used
12.	Reactor Operator
13.	Sodium Leak Detection Panel
14.	Desk
15.	Chair
16.	Seismic Instrument Panel
17.	Flux Monitoring
18.	Radiation Monitoring Panel
19.	Radiation Monitoring Panel
20.	Radiation Monitoring Display Station
21.	PAIC Hand Set/Communication Jack and Switch
22.	HVAC System Recirculating Air Status Cabinet
23.	HVAC Control Rack
24.	Generator and Transformer Protection Panel

Table 7.9-1 (Continued)

<u>Item</u>

25. Turbine/Generator Supervisory Panel
26. Turbine (Electro Hydraulic Control) Equipment Bays
27. BOP Auxiliaries
28. Non-Sodium Fire Protection Rack
29. Sodium Fire Protection Zone Indicating Panel
30. Heat Removal and Conditioning Logic Rack
31. Steam Generator Logic Rack
32. Chilled Water Control Cabinet
33. Auxiliary Liquid Metal Control (Direct Heat Removal Service Panel)
34. Remote Annunciator Cabinets
35. Steam Plant Conditioning Rack
36. Steam Plant Logic Rack
37. Auxiliary Relay Panel
38. Termination Racks
39. PPS Containment Isolation Instrumentation Racks
40. Remote Multiplexing System (RMS) Central Control Unit
41. Primary PPS Buffers
42. Primary PPS Termination Cabinet
43. Primary PPS Comparator Panels
44. Primary PPS Logic Racks
45. Primary PPS Isolation Racks
46. Secondary PPS Buffers
47. Secondary PPS Termination Cabinets
48. Secondary PPS Comparator Panels

Amend. 69
July 1982

Table 7.9-1 (Continued)

Item	
49.	Secondary PPS Solenoid Drivers
50.	PPS Monitor Rack
51.	Not Used
52.	Not Used
53.	Desk Top Radio
54.	Reactor Control Panel
55.	Supervisory Control Panel
56.	Flow Control Panel
57.	Load Dispatch Panel
58.	Failed Fuel Readout Panel
59.	Computer Typewriters
60.	Computer Line Printer/Plotter
61.	Cathode Ray Tube Display & Keyboard
62.	Cathode Ray Tube Display & Keyboard
63.	Cathode Ray Tube Display & Keyboard
64.	Cathode Ray Tube Display & Keyboard
65.	Cathode Ray Tube Display & Keyboard
66.	PDH & DS Remote Data Acquisition Terminal
67.	Recirculation Gas Control Cabinet
68.	RMS Control Room Multiplexer
69.	RMS Control Room Multiplexer Termination Cab
70.	Not Used
71.	Containment Instrumentation

TABLE 7.9-1 (Continued)

Item	
72.	Plant Control System Switching Logic
73.	Auxiliary Equipment Isolation Logic
74.	Portable Radio Communications
75.	Not Used
76.	Not Used
77.	PDH&DS Remote Data Acq. Term
78.	Remote Annunciator Cabinets
79.	Manual Telephone Switchbard & PAC Heat Set
80.	CRT display and Keyboard
81.	Not Used
82.	Building Fire Protection Panel
83.	Not Used
84.	Patch Panel
85.	Containment Instrumentation Division II
86.	Inoperable Status Monitoring System
87.	Piping and Equipment Heating Control
88.	Piping and Equipment Heating Display
89.	Piping and Equipment Heating Typewriter
90.	TMBDB Panel
91.	Trace Heating Control & Instrumentation
92.	Remote Annunciator Cabinet
93.	Inert Gas System Control Panel
94.	Inoperable Status Monitoring System

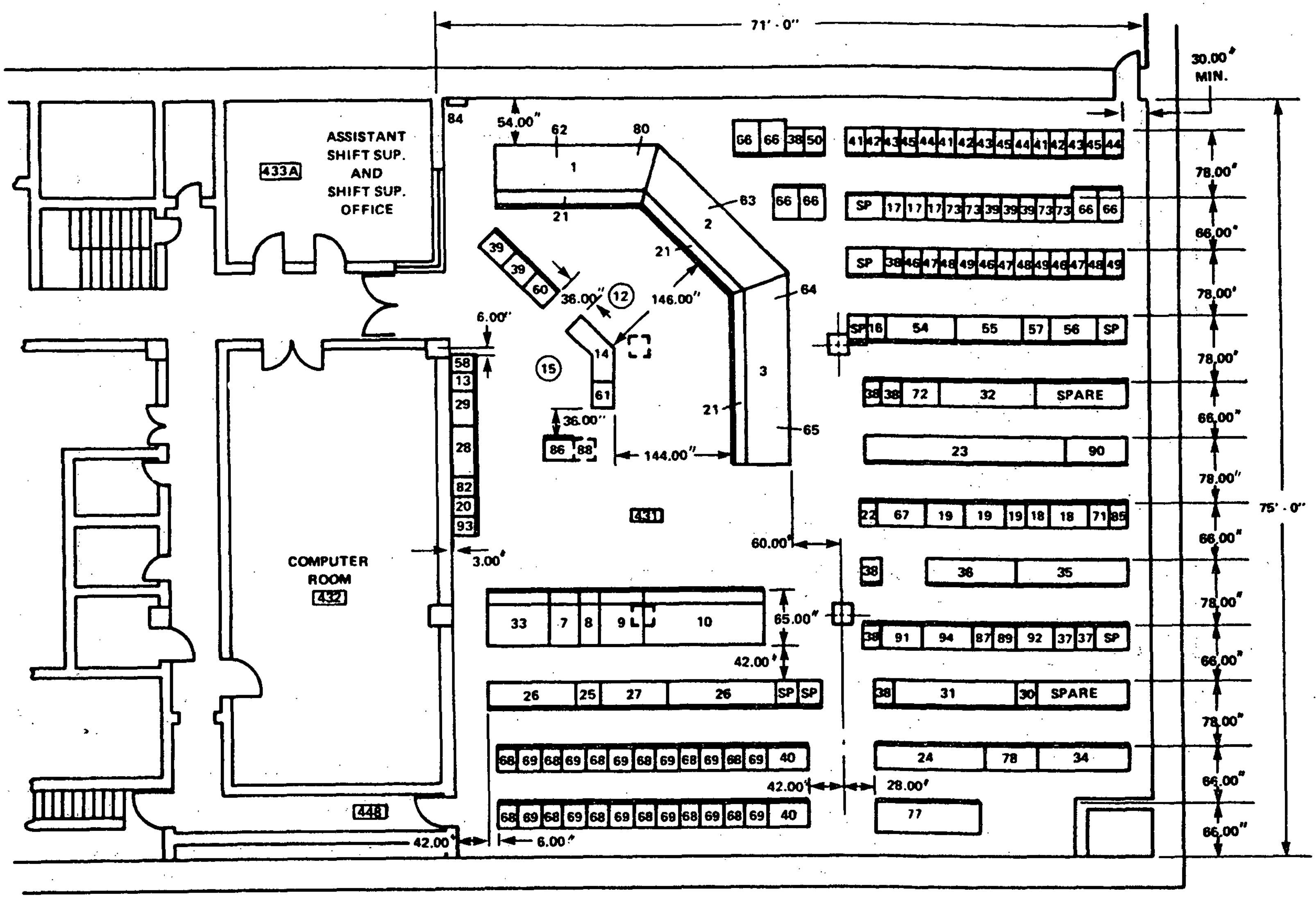

Figure 7.9-1. Control Room Layout

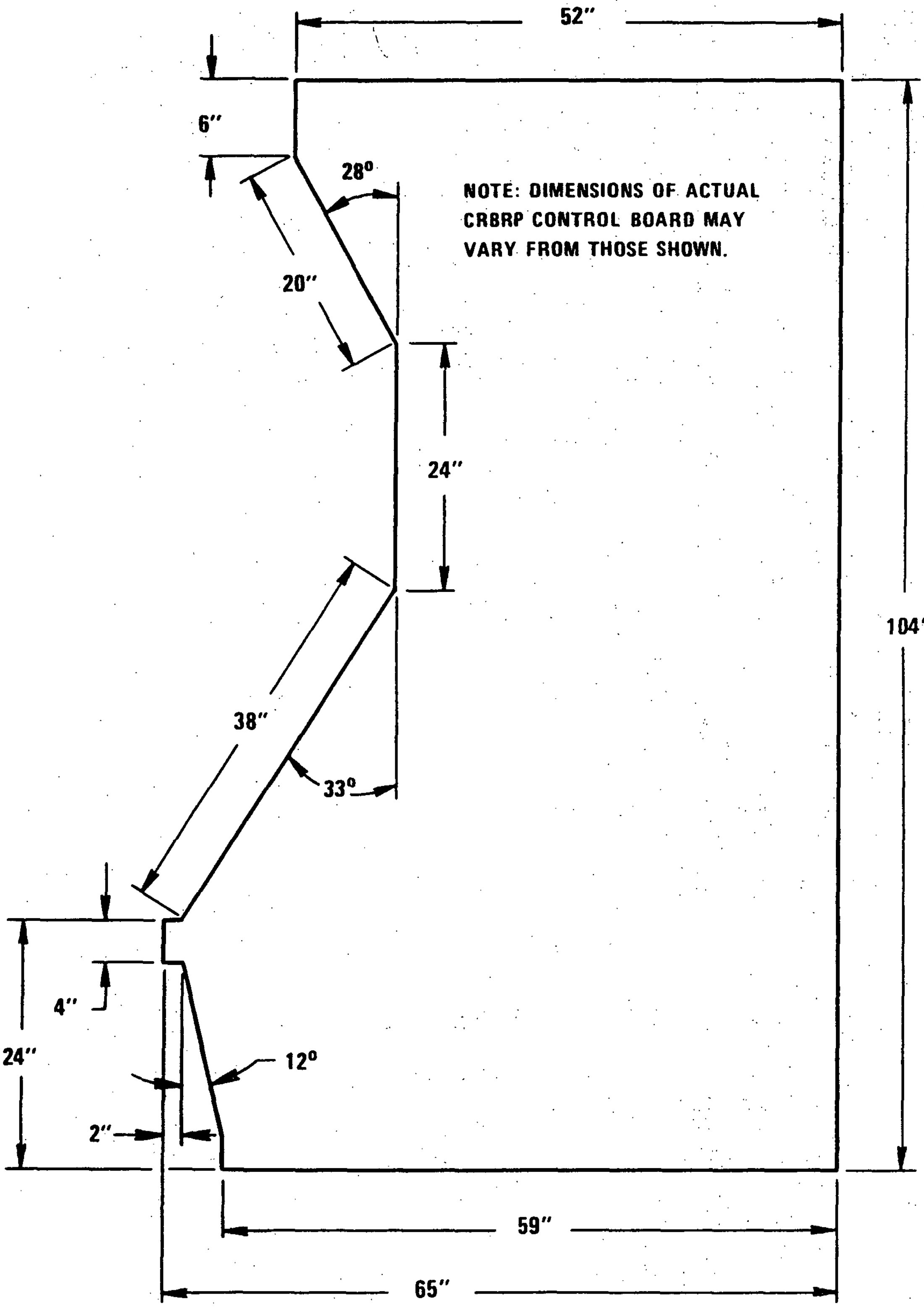

Figure 7.9-2. Typical Control Board (Side View)

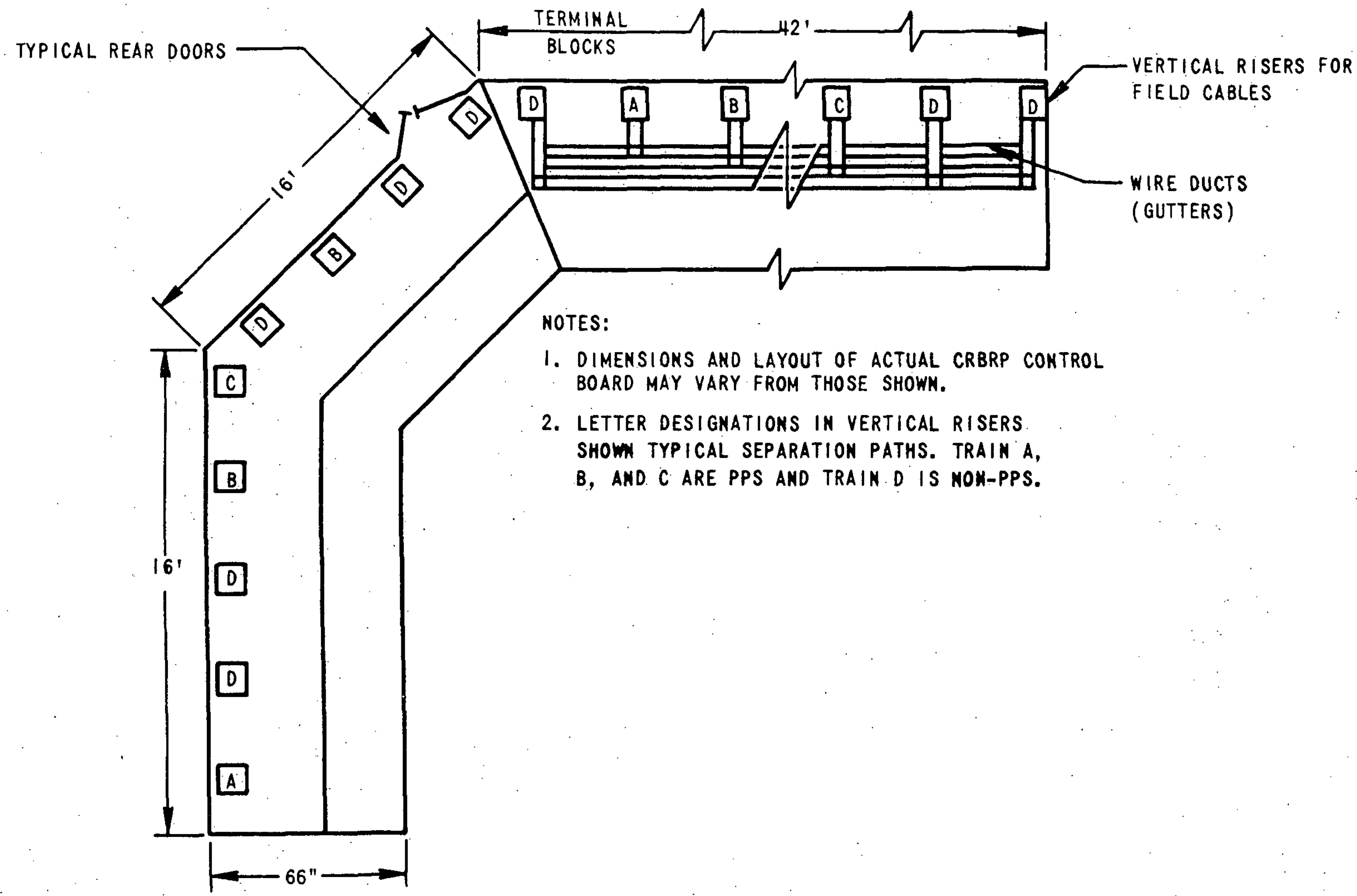

Figure 7.9-3. Main Control Baord Plan View

6685-3

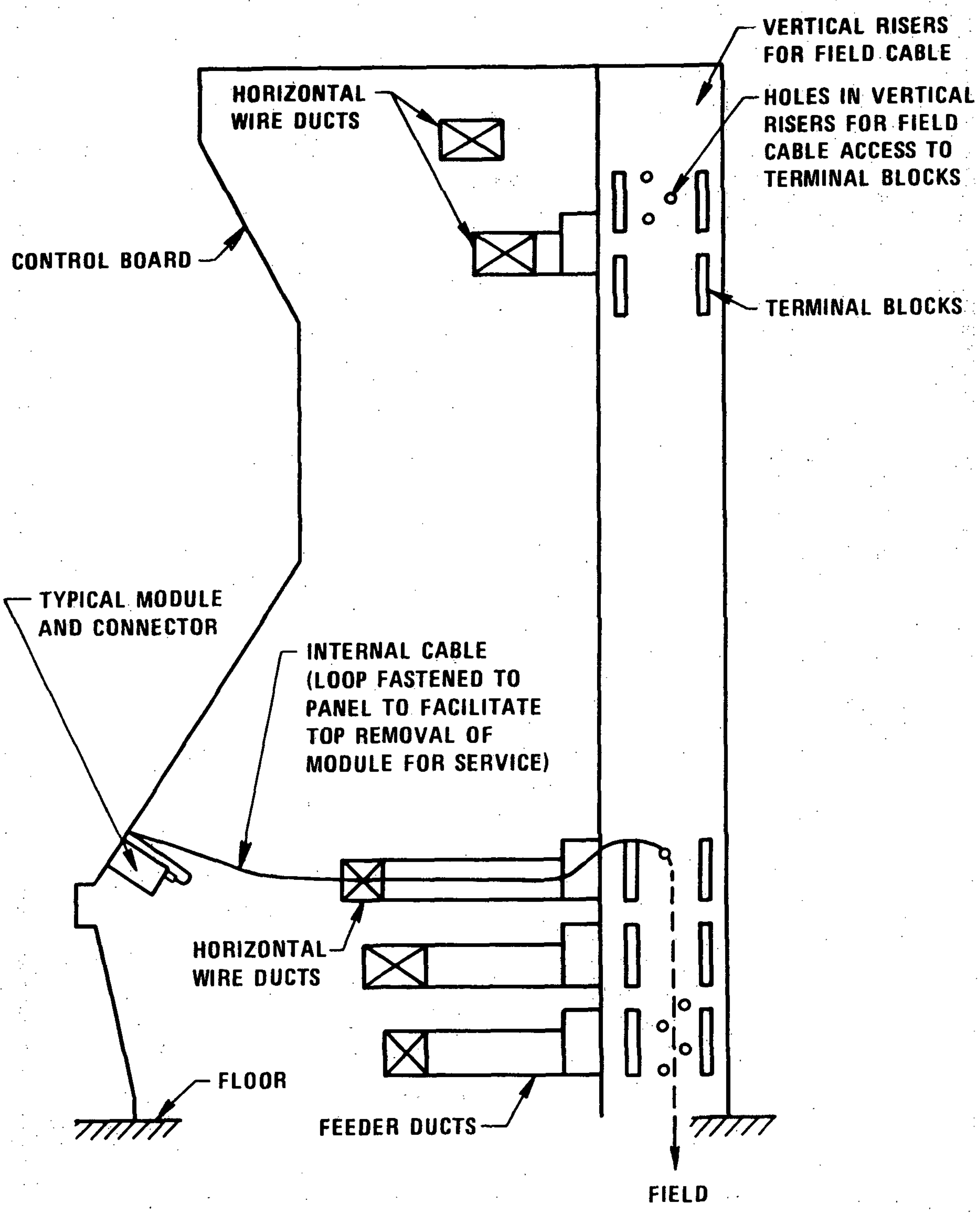

Figure 7.9-4. Typical Control Board Wiring Layout

CLINCH RIVER

BREEDER REACTOR PROJECT

PRELIMINARY SAFETY ANALYSIS REPORT

CHAPTER 8

ELECTRIC POWER

PROJECT MANAGEMENT CORPORATION

CHAPTER 8.0 ELECTRICAL POWER

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CHAPTER 8.0 ELECTRIC POWER

8.1 INTRODUCTION

8.1.1 Utility Grid and Interconnections

The State of Tennessee and parts of neighboring states are supplied with electric power by the Tennessee Valley Authority (TVA). TVA is a United States Government owned corporation which consists of interconnected hydro, fossil-fueled, combustion turbine, and nuclear power plants and transmission facilities. These plants comprise the TVA utility grid and are interconnected with other power companies in surrounding states (see Figure 8.1-1).

The CRBRP will be connected to the TVA 161KV grid using two separate and physically independent switchyards providing four separate connections to the TVA 161KV grid. The plant generating switchyard will be connected to the TVA 161KV power grid by two 161KV transmission lines, one connected to the 500-161KV Roane substation of TVA and designated as the Roane line, and the second to the Fort Loudoun Hydroelectric Plant and designated as the Fort Loudoun-1 line. The plant reserve switchyard will be connected to the TVA 161KV grid by two physically separate and electrically independent 161KV transmission lines, one connected to the Oak Ridge Gaseous Diffusion Plant (ORGDP) switchyard of DOE and designated as the K-31 line, and the other to the Fort Loudoun Hydroelectric Plant and designated as the Fort Loudoun-2 line.

8.1.2 Plant Electrical Power System (See Figures 8.1-2, 8.3-1, 8.3-2)

Power Supplies

The Plant Electrical Power System consists of power supplies and power distribution systems which provide flexibility in plant operation and availability under single contingency conditions. The power supplies are comprised of:

1) The Plant AC Power Supply which consists of the main generator supplying power to all plant loads through the generator circuit breaker and the unit station-service transformers (USSTs).

2) The CRBRP Preferred (off-site) AC Power Supply which consists of two 161KV transmission line connections from CRBRP generating switchyard to the TVA grid, each capable of supplying power to all plant loads through the main power transformer and the unit station service transformers.

3) The Reserve (off-site) AC Power Supply which consists of two physically independent 161KV transmission line connections from CRBRP reserve switchyard to the TVA grid. ("Reserve Power" as described here is "Preferred Power" according to IEEE Std. 308 terminology and 10CFR Part 50, Appendix A). Each line is capable of supplying power to all plant loads through the two reserve station service transformers (RSSTs).

4) The Standby (on-site) AC Power Supply* consists of three physically separate and electrically independent diesel generators. Two of these diesel generators which supply power to safety-related (Class 1E) Division 1 and 2 loads are redundant to each other. Either one of these three standby diesel generators can provide sufficient power to facilitate and maintain a safe plant shutdown. However, from the consideration of connected loads, Class 1E Divisions 1 and 2 provide power to redundant load groups and as such are referred to as redundant divisions. Class 1E Division 3 provides Class 1E power to Loop 3 of the Heat Transport System (HTS) and to certain plant Non-Class 1E loads. (The Non-Class 1E loads are connected through an isolation subsystem). Since not all the loads powered from Division 3 are identical or similar to those powered by Divisions 1 or 2, this division is not identified as redundant to Division 1 or 2. However, as far as the HTS is concerned, the Divisions 1, 2 and 3 power supplies are fully redundant serving the Loops 1, 2 and 3 Class 1E loads, respectively.

5) The DC Power Supply*, for Division 1, 2 or 3, consists of one independent 125 volt DC battery with its associated active and spare battery chargers and an inverter for 120/208 volt AC uninterruptible power supply (UPS). Each battery is capable of supplying power to DC loads and UPS loads of its associated safety division. Class 1E UPS is also referred to as vital AC power supply.

6) The 120/208 volt vital AC Power Supply*, for each Division 1, 2 or 3, consists of one independent inverter supplied by an independent DC system. Each inverter will supply power to vital AC loads of its associated safety Division. Division 1 and 2 vital AC loads are redundant to each other.

7) The Non-Class 1E DC Power Supply consists of two systems (Divisions A and B) each having one 125 volt DC battery dedicated for plant instrumentation and control. Two separate 125 volt DC batteries are dedicated for switchyard control and instrumentation and two 48 volt DC batteries are provided for the plant communication systems. Division A also has one 250 volt battery to provide power for DC motor loads. Each battery system is equipped with its own active and spare battery chargers, switchgear and distribution panels. 125 volt DC and 250 volt DC battery systems have inverters for 120/208 volt uninterruptible power supply (UPS). Non-Class 1E UPS is also referred to as Non-Class 1E essential power supply.

8) Two Non-Class 1E 125 volt DC Power Supplies (one for Division A and the other for Division B) will be provided complete with associated active and spare battery chargers for security systems, and the associated inverters for 480 volt AC UPS for security and lighting loads.

*This equipment is Class 1E as defined by IEEE Standard 308.

Distribution Systems

The Plant electrical power distribution system can be fed by the Plant, the CRBRP Preferred and the Reserve Power supplies and provides power to all Non-Class 1E and Class 1E loads. The Plant distribution system has been divided into two systems; the normal distribution (Non-Class 1E) system and the safety-related distribution (Class 1E) system. The safety-related distribution system can be fed by the Plant, the CRBRP Preferred, the Reserve and the Standby Power Supplies. The Standby Power Supplies are capable of providing the necessary power for all Class 1E loads.

Operation

When the main generator is operating, the safety-related and non-safety related distribution system receives power from the Plant Power Supply via the generator circuit breaker and unit station service transformers. In the event of a turbine trip, a reactor trip, a main generator fault, or a fault between the main generator and the generator circuit breaker, the generator circuit breaker will be tripped automatically. The unit station service transformers will remain connected to the CRBRP Preferred AC Power Supply (from 161KV generating switchyard) via the main power transformer and provide uninterrupted power through the Plant AC distribution system to the plant auxiliary loads. An electrical fault downstream of the generator circuit breaker will cause tripping of associated 161KV circuit breakers in the generating switchyard and circuit breakers of the 13.8KV and 4.16KV medium voltage (MV) switchgear. This will cause the loss of CRBRP Preferred AC Power Supply to the unit station service transformers.

Upon loss of power supply to the unit station service transformers, the MV switchgear of the Plant AC distribution system will be connected automatically to the Reserve AC Power Supply.

Upon loss of the CRBRP Preferred and the Reserve AC Power Supplies, the safety-related AC distribution system will receive power from the standby (on-site) diesel generators.

All Class 1E loads and their requirements are listed in Tables 8.3-1A, 8.3-1B, 8.3-1C, 8.3-2A, 8.3-2B and 8.3-2C.

8.1.3 Criteria and Standards

The following form the principal bases for the design of the electrical power system:

1) The entire Non-Class 1E electrical power system will be split into two load groups designated as Division A and Division B, each with its own power supply, buses, transformers and control power. Safety-related loads are divided into three load groups designated as Divisions 1, 2 and 3. 4.16KV buses of safety Divisions 1 and 2 will be fed from the same USSTs and RSSTs as Divisions A and B, respectively. Division 3 will be supplied from the USST or RSST of Division A.

1a) The Class 1E electrical system will comply with the single failure criterion.

2) Division 1 and Division 2 will consist of loads redundant to each other, and each will have one independent diesel generator connected to its bus as a standby source of power. A manual tie between these two diesel generator buses will be provided for extreme emergency conditions.

3) Safety Division 3 will be powered from a separate standby diesel generator and will be independent of safety Divisions 1 and 2. This Division will feed loop 3 decay heat removal system loads. It will have its own set of battery chargers, 125V storage battery and associated inverter. The inverter will provide 120/208V vital AC power for vital AC loads of Divison 3.

4) There will be no automatic paralleling of the diesel generators or automatic transfer of loads between the safety divisions.

5) Each safety division will have its own 125V DC control and instrumentation supply complete with battery chargers and storage batteries. Inverters will provide 120/208V vital AC power supply for each of these Divisions.

6) Each Division A or B of Non-Class 1E loads has its own independent 125V DC and 48V DC control and instrumentation supply complete with storage batteries and battery chargers. Division A will also have one 250V DC battery with its own battery chargers. 125V DC and 250V DC batteries will have inverters to supply 120/208V UPS for essential Non-Class 1E loads.

7) All storage batteries (except for security systems) will be sized to feed their connected loads for at least a two hour operation without recharging. Storage batteries for security systems are not described here.

8) Raceways will not be shared by Class 1E and Non-Class 1E cables. However, the Class 1E associated circuits will be treated as Class 1E circuits for cable routing purposes (in accordance with IEEE Std. 384 and Regulatory Guide 1.75), unless an acceptable isolation device is provided.

9) Special identification criteria will apply for Class 1E equipment, including cabling and raceways.

10) Separation criteria of IEEE Std. 384 and Regulatory Guide 1.75 will be used. These establish requirements for preserving the independence of redundant Class 1E electrical systems.

11) Class 1E equipment will be designed with the capability to be tested periodically.

12) Clinch River Breeder Reactor Plant (CRBRP) General Design Criteria 15 and 16 will be satisfied in the design of the Plant power distribution system (see Section 3.1).

13) The following NRC Regulatory Guides will be followed in the design of the Plant power distribution system, as discussed in the appropriate sections:

Regulatory Guide 1.6 (Safety Guide 6), Rev. 0, 3/71, "Independence Between Redundant Standby (On-site) Power Sources and Between Their Distribution Systems".

Regulatory Guide 1.9 (Safety Guide 9), Rev. 2, 12/79, "Selection of Diesel Generator Set Capacity for Standby Power Supplies".

Regulatory Guide 1.22 (Safety Guide 22), Rev. 0, 2/72, "Periodic Testing of Protection System Actuation Functions;"

Regulatory Guide 1.29, Rev. 3, 9/78, "Seismic Design Classification".

Regulatory Guide 1.30 (Safety Guide 30), Rev. 0, 8/72, "Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment".

Regulatory Guide 1.32, Rev. 2, 2/77, "Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants".

Regulatory Guide 1.40, Rev. 1, 10/79, "Qualification Tests of Continuous Duty Motors Installed Inside the Containment of Water Cooled Nuclear Power Plants".

Regulatory Guide 1.41, Rev. 0, 3/73, "Preoperational Testing of Redundant Onsite Electric Power Systems to Verify Proper Load Group Assignments".

Regulatory Guide 1.47, Rev. 0, 5/73, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems".

Regulatory Guide 1.53, Rev. 0, 6/73, "Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems".

Regulatory Guide 1.63, Rev. 2, 7/78, "Electric Penetration Assemblies in Containment Structures for Light-Water Cooled Nuclear Power Plants".

Regulatory Guide 1.68, Rev. 2, 8/78, "Initial Test Program for Water Cooled Nuclear Power Plants".

Regulatory Guide 1.73, Rev. 0, 1/74, "Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants".

Regulatory Guide 1.75, Rev. 2, 9/78, "Physical Independence of Electric Systems".

Regulatory Guide 1.89, Rev. 0, 11/74, "Qualification of Class 1E Equipment for Nuclear Power Plants".

Regulatory Guide 1.93, Rev. 0, 12/74, "Availability of Electric Power Sources."

Regulatory Guide 1.97, Rev. 2, 12/80, "Instrumentation for Light-Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident."

Regulatory Guide 1.100, Rev. 1, 8/77, "Seismic Qualification of Electric Equipment for Nuclear Power Plants."

Regulatory Guide 1.106, Rev. 1, 3/77, "Thermal Overload Protection for Electric Motors on Motor Operated Valves."

Regulatory Guide 1.108, Rev. 1, 8/77, "Periodic Testing of Diesel Generator Units Used as Onsite Electric Power Systems at Nuclear Power Plants."

Regulatory Guide 1.118, Rev. 2, 6/78, "Periodic Testing of Electric Power and Protection Systems."

Regulatory Guide 1.128, Rev. 1, 10/78, "Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants."

Regulatory Guide 1.129, Rev. 1, 2/78, "Maintenance, Testing and Replacement of Large Lead Storage Batteries for Nuclear Power Plants."

Regulatory Guide 1.131, Rev. 0, 8/77, "Qualification Tests of Electric Cables, Field Splices, and Connections for Light-Water Cooled Nuclear Power Plants."

Regulatory Guide 1.137, Rev. 1, 10/79, "Fuel Oil Systems for Standby Diesel Generators."

14) The following IEEE standards have been followed in the design, qualification and maintenance of the Plant power distribution system:

IEEE 308-1974, "Criteria for Class 1E Power Systems for Nuclear Power Generating Stations."

IEEE 317-1976, "Electric Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations."

IEEE 323-1974, "Qualifying Class 1E Equipment for Nuclear Power Generating Stations."

IEEE 344-1975, "Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations."

IEEE 384-1974, "Criteria for Independence of Class 1E Equipment and Circuits."

IEEE 387-1977, "Criteria for Diesel Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations".

IEEE 450-1980, "Recommended Practice for Maintenance, Testing and Replacement of Large Lead Storage Batteries for Generating Stations and Substations".

A discussion for each of the above Regulatory Guides, and IEEE standards is provided in Sections 8.2 and 8.3.

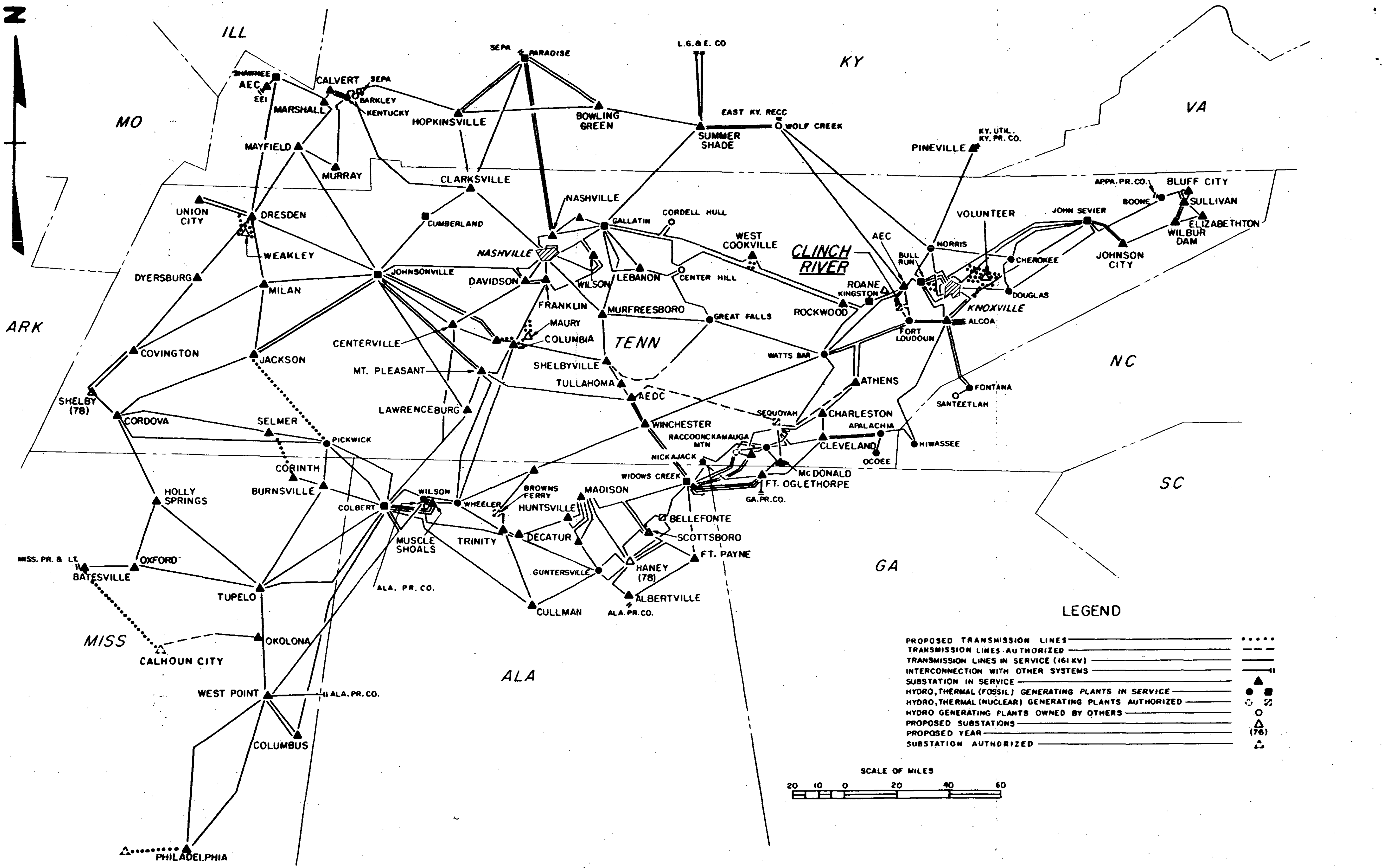

Figure 8.1-1. TVA 161 kV Transmission System

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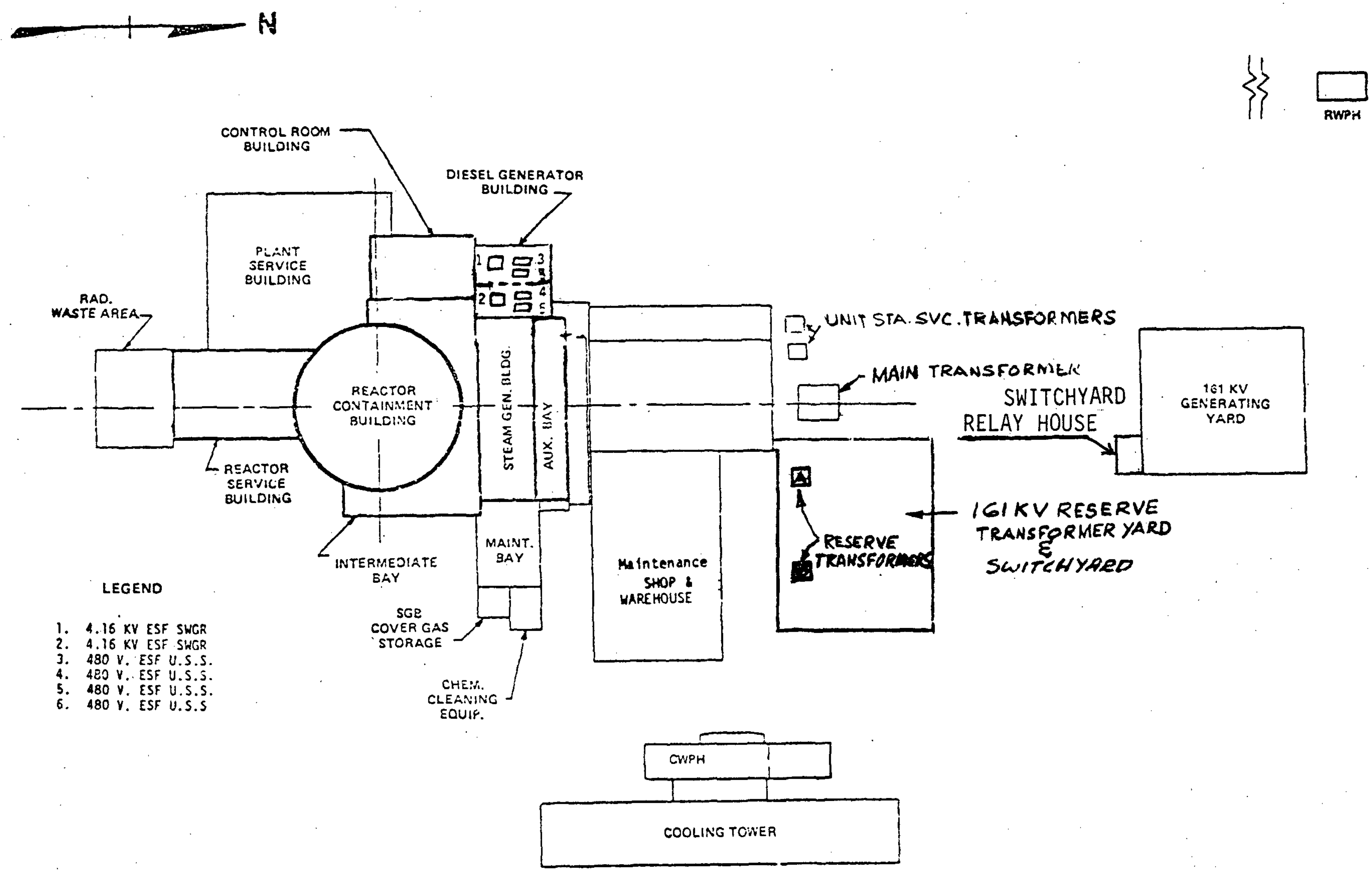

Figure 8.1-2. Location of ESF Switchgear and Unit Substations

8.2 OFF-SITE POWER SYSTEM

8.2.1 Description

The off-site power supply system provides a reliable source of AC power to the CRBRP. The system consists of the CRBRP Preferred AC Power Supply and the Reserve AC Power Supply, each having two independent 161KV connections to the TVA grid. All four of these 161KV grid connections are kept continously energized. Any one of these connections can supply the entire plant AC distribution system loads to facilitate and maintain a safe plant shutdown and startup.

8.2.1.1 Connection of the Switchyards to the Utility Grid

Off-site power will be supplied to the CRBRP by one of the four 161KV transmission lines (See Figure 8.2-1.). Two 161KV transmission lines to the Reserve Switchyard will be formed by opening and reterminating the DOE owned Fort Loudoun K-31 line. One of the lines formed by the retermination will be approximately four miles long, will connect the DOE K-31 substation bus in the reserve switchyard and will be designated as K-31 line. The K-31 substation is connected to the TVA grid by four other 161KV lines. Three of the lines connect to TVA's Kingston Steam Power Plant and the other to TVA's Bull Run Steam Power Plant. The second 161KV transmission line, formed by the retermination and designated as Fort Loudoun-2 line, will be approximately ten miles long and will connect to TVA's Fort Loudoun Hydroelectric Plant. The Fort Loudoun Hydroelectric Plant, with a generating capacity of 142,220KVA is connected to the grid by six 161KV lines. Two of the six lines connect to the CRBRP, and the other four lines connect to 161KV substations that are an integral part of the 161KV transmission network.

Two 161KV transmission lines to the generating switchyard will be formed by opening and extending the TVA owned Fort Loudoun K-33 161KV line. One of the these lines will be approximately eight miles long, and will connect CRBRP to the TVA Roane substation, and will be designated as the Roane line. The fourth line, designated as Fort Loudoun-1 line, will be thirteen miles long and will connect to TVA's Fort Loudoun Hydroelectric Plant described above.

8.2.1.2 CRBRP Preferred AC Power Supply (See Figures 8.2-1 & 8.3-1)

The CRBRP Preferred Power Supply consists of two 161KV transmission lines in the generating switchyard connected to the Plant distribution system through the main power transformer, and the unit station service transformers. Hereinafter this will be referred to as CRBRP Preferred AC Power Supply.

The generating switchyard consists of two main buses and three bays. The Roane line terminates in bay-1 and the Fort Loudoun-1 line terminates in bay-3. The main power transformer high voltage overhead line terminates in bay-2. The lines at bay-1 and bay-3 are connected to bus-1 through 161KV circuit breakers and to bus-2 through motor operated disconnect switches. The main power transformer high voltage overhead line terminates in bay-2 and is connected to bus-1 and bus-2 through 161KV circuit breakers.

The main generator is connected to the system through an isolated phase bus system from the main generator terminals through the generator circuit breaker to the low voltage terminals of the main power transformer and high voltage terminals of the unit station service transformers. During normal plant operation, power to the plant auxiliary loads will be provided from the main generator via the generator circuit breaker and the unit station service transformers, hereinafter called the Plant Power Supply.

In cases where the loss of the Plant Power Supply is caused by a turbine trip, reactor trip, a main generator fault, or a fault between the main generator and generator circuit breaker, the generator circuit breaker will trip. The unit station service transformers will remain energized and receive power from the CRBRP Preferred AC Power Supply via the main power transformer.

8.2.1.3 Reserve AC Power Supply (See Figures 8.2-1 & 8.3-1)

The CRBRP Off-site AC Power Supply in the reserve switchyard consists of two independent 161KV transmission lines (K-31 and Fort Loudoun-2 lines) connected to two independent reserve station service transformers. The K-31 line terminates in bay-6 and the Fort Loudoun-2 line in bay-8. Hereinafter this will be referred to as the Reserve AC Power Supply. The Reserve Power Supply, as described herein, is the preferred power supply in IEEE Std. 308 and 10CFR Part 50 terminology).

The reserve station service transformer yard consists of two reserve station service transformers, and it is physically combined with the reserve switchyard. The two reserve station service transformers are connected to reserve switchyard bus-2 and bus-1, respectively, by a rigid bus from transformer high voltage terminals.

In the event of the loss of the Plant Power Supply concurrent with the loss of the CRBRP Preferred AC Power Supply, the Plant distribution system will be transferred automatically to the Reserve Power Supply by a fast dead bus transfer scheme, as described in Section 8.3.1.1.

Offsite Power Supply Lines

The DOE owned K-31 Fort Loudoun 161KV transmission line, which will be opened and reterminated at the CRBRP consists of approximately ten miles of single circuit, self-supporting steel towers and approximately four miles of single circuit wood H-frame construction. Both sections of the line were designed to meet or exceed the National Electrical Safety Code heavy loading strength requirements. This ensures that the line has adequate structural integrity for wind and heavy icing conditions in excess of those that would be expected to occur in the area. The wire tensions for the conductors and shield wires were selected to ensure that vibration damage will not occur. Long experience with transmission lines in the Tennessee Valley area has verified that, where the tension is kept below 18 percent of the ultimate strength of the cable, vibration will not be a problem. Galloping of conductors has never been observed on lines in the eastern portion of the TVA system.

Both the steel tower and wood pole line sections have two overhead ground wires to provide lightning protection. The use of circuit breakers with high speed reclosing relays will clear the majority of the transmission line faults.

The TVA owned Fort Loudoun-Roane 161KV transmission line (presently operating as the Fort Loudoun K-33 line) will be looped into the CRBRP. This line is single circuit wood H-frame construction and was designed to conform with the heavy loading strength requirements of the National Electrical Safety Code. The design considerations incorporated into this existing line are identical to those outlined in the discussion of the DOE owned K-31 Fort Loudoun·161KV transmission line.

The new line construction, required to complete the CRBRP Roane and CRBRP Fort Loudoun-1 161KV lines will begin where the existing 161KV line crosses under a TVA 500KV line. Thus, the two lines will parallel the 500KV line along the side of Chestnut Ridge for approximately 2.7 miles before turning southwestward and paralleling the Fort Loudoun K-31 line into the CRBRP Switchyard. At the CRBRP, the K-31 (Reserve AC Power Supply) line will cross over the two power circuits in such a way that physical failure of either power circuit will not endanger the Reserve AC Power Supply. The loop line structures will be on 75 foot centers with 100 foot separation from the 500KV line. This will minimize the acreage for right of way and also reduce clearing requirements. The 500KV line will provide substantial shielding to the loop line from lightning.

The new construction which will be used for this loop connection, will provide adequate separation between the 161KV circuits to ensure that loss of one line will not jeopardize the integrity of the other circuit. The single circuit structures will be compact, self-supporting, narrow base towers suitable in strength. These narrow, silhouette structures will combine epoxy fiberglass crossarms on a new design stem of structural steel. The foundation will be composed of a precast concrete section made to fit an augered hole. This combination of steel and epoxy fiberglass will ensure minimum maintenance and structural reliabilty for weather element loading.

As previously stated, the northwestern end of the K-33 Fort Loudoun 161KV line will be terminated at TVA's Roane 500KV Substation.

There are no transmission lines that cross over the Fort Loudoun-1 CRBRP line and Fort Loudoun-2 CRBRP line that will connect to the generating switchyard and reserve switchyard, respectively. There are transmission lines which cross the remaining CRBRP lines; however, there is no single transmission line that crosses over more than one of the CRBRP transmission lines.

All transmission lines which cross the lines to the CRBRP, are designed to meet medium loading requirements of the National Electrical Safety Code (ANSI C2). In addition, design provides for wind loadings of approximately 85 mph winds on bare conductors and vertical loading strength based on approximately one (1) inch of radial ice. These loading conditions ensure adequate strength to provide reliability under the worst possible weather conditions encountered on TVA's transmission line system.

8.2.2 Analysis

8.2.2.1 General Analysis

The CRBRP Preferred AC Power Supply is designed to provide uninterrupted AC power to the plant distribution system. In case of a turbine trip or a reactor trip in the absence of an electrical fault, the generator circuit breaker permits automatic isolation of the main generator from the grid without disconnecting the CRBRP Preferred AC Power Supply. The 161KV rigid buses in the generating switchyard and the reserve switchyard provide reliability in the design. Since circuit breakers and disconnect switches to bus-1 are all normally closed, the CRBRP Preferred AC Power Supply is connected to the transmission system through two different paths. The circuit protective devices reduce the effect of system disturbances by promptly isolating the faulted section. This provides connection of the CRBRP Preferred AC Power Supply to at least one transmission line for most system disturbances. Each 161KV line is individually capable of supplying sufficient power for the startup as well as the safe shutdown of the plant and to maintain the shutdown conditions.

The Reserve AC Power Supply is designed to be available within a few cycles after the CRBRP Preferred AC Power Supply becomes unavailable. The 161KV disconnect switches and the circuit breakers are all normally closed so that each reserve station service transformer is connected to two 161KV transmission lines through two different paths. This arrangement provides connection of each reserve station service transformer to at least one transmission line for most system disturbances. Each 161KV line for the Reserve AC Power Supply is capable of supplying sufficient power to the plant distribution system for the startup as well as for the safe shutdown of the plant and to maintain the shutdown conditions. Each reserve station service transformer is capable of supplying sufficient power to the Plant distribution system for the safe shutdown of the plant and to maintain the shutdown conditions.

Compliance with CRBRP General Design Criteria 15 and 16 is discussed in Section 3.1. Compliance with Regulatory Guide 1.32, 1.93 and IEEE Std. 308-74 are discussed below.

Regulatory Guide 1.32, Rev. 2

The CRBRP Preferred and the Reserve AC Power Supplies are immediate access circuits.

The CRBRP Preferred AC Power Supply consists of two 161KV transmission lines in the generating switchyard connected to the main power transformer. In the event of a turbine trip when no electrical fault is present, the generator circuit breaker will open automatically and disconnect the Plant Power Supply. The Plant AC power distribution system will than be provided with power by the CRBRP Preferred AC Power Supply through the main power transformer without interruption.

In the event of non-availability of both the Plant and the CRBRP Preferred AC Power Supplies, the Plant AC distribution system will be transferred to the Reserve AC Power Supply. This transfer is performed within a period of 6 cycles by a fast dead bus transfer scheme as described in Section 8.3.1.1. Both reserve station service transformers are kept energized at all times during plant operation and are available to the Plant AC distribution system within a few cycles. This assures that the specified acceptable design limits are maintained.

<u>Regulatory Guide 1.93, Rev. 0 (12/74)</u>

The available off-site AC power sources consist of the CRBRP Preferred AC Power Supply and the Reserve AC Power Supply. Each of these two supplies provides two connections to the TVA 161KV grid. The two 161KV grid connections to the reserve station service transformers constitute the required independent off-site power sources. In addition, two 161KV grid connections to the generating switchyard provide an added reliability to off-site power, available through the main power and the unit station service transformers.

On-site AC power sources and on-site DC power sources comply with the requirements of CRBRP GDC15 for the availability of electric power sources.

Should an LCO condition be present on these power sources, the plant's continued operation will be restricted in accordance with the Regulatory Guide 1.93 recommendations.

<u>IEEE Standard 308-1974</u>

The Reserve AC Power Supply provides the two independent circuits of the IEEE Std. 308-1974 "preferred power supply." It connects the TVA 161KV grid to each of the two 4.16KV Class 1E switchgear buses through the reserve station service transformers. Hence, the safety-related AC distribution system has two physically separate and electrically independent sources available from the TVA grid.

The CRBRP Preferred and the Reserve AC Power Supplies, each has sufficient capacity to operate the loads applied during a design basis accident. Both the CRBRP Preferred and the Reserve AC Power Supplies are available during normal operation (see Section 16.3.9).

The availability of off-site power supplies to Class 1E buses is monitored on the Electrical Control Panel in the control room. In the event the incoming off-site power source has an undervoltage condition or any one of the protective relays is not reset, the condition will be alarmed on the Electrical Control Panel to alert the operator.

In addition, an amber light on the Electrical Control Panel in the control room will indicate that the off-site power supply line and its breaker are available for transfer of power from the other source if required.

8.2.2.2 Steady-State and Transient Analyses

Results of steady-state studies show that the 161KV off-site power sources of the Reserve AC Power Supply remain a reliable source to supply the Plant electric power distribution system for single contingency conditions including loss of the CRBRP generating unit, loss of one of the largest generating units on the system, loss of one offsite source, or the loss of a critical 500KV transmission facility. Steady-state studies also show that each 161KV line connected to CRBRP is capable of supplying the total Plant power distribution requirements for the startup as well as for the safe shutdown of the plant and to maintain the shutdown conditions.

Line flows and bus voltages can be obtained from the following steady state power flow diagrams.

Diagram	Conditions
Figure 8.2-2	Off-site Power Supply - System Normal
Figure 8.2-3	Off-site Power Supply - Loss of CRBRP Generation
Figure 8.2-4	Off-site Power Supply - Loss of One Cumberland Unit
Figure 8.2-5	Off-site Power Supply - Loss of K-31 Line Connection to the Reserve Switchyard
Figure 8.2-6	Off-site Power Supply - Loss of Fort Loudoun-2 Line Connection to the Reserve Switchyard

Results of transient stability studies show that the Reserve AC Power Supply remains a reliable source to supply the plant power distribution system for single contingency cases including loss of the CRBRP unit, loss of a critical 161KV transmission line, or loss of the largest generating unit on the TVA system.

Transient stability studies included single contingency conditions consisting of three-phase faults on transmission lines connecting the CRBRP unit into the 161KV transmission system and conditions of three phase faults on transmission lines connecting the Reserve AC Power Supply bus into the 161KV transmission system. These studies included unsuccessful reclosures in which lines were removed from service following disturbances on the system and stuck breaker conditions in which a complete bus section was disconnected automatically by backup breakers. These cases were considered to be the most severe conditions of postulated transmission disturbances. Results of these studies show that the transmission system remains stable with negligible disturbances to the 161KV off-site power supply.

The bus voltage and frequency at the respective 161KV off-site power supply bus are indicated on the following diagrams.

Diagram	Conditions
Figure 8.2-7	161KV Generating Switchyard Voltage Performance during Loss of CRBRP Unit - No Electrical Fault
Figure 8.2-8	161KV Generating Switchyard Frequency Performance during Loss of CRBRP Unit - No Electrial Fault
Figure 8.2-9	161KV Reserve Switchyard Bus Voltage Performance during Loss of Fort Loudoun-2 161KV Line Supplying Off-site Power under Most Severe Transmission Line Disturbance
Figure 8.2-10	161KV Reserve Switchyard Bus Frequency Performance during Loss of Fort Loudoun-2 161KV Line Supplying Off-site Power under Most Severe Transmission Line Disturbance

The most severe disturbance on the 161KV transmission system affecting the voltage and frequency at the Reserve AC Power Supply is indicated on Figures 8.2-9 and 8.2-10.

In none of the steady-state or transient stability cases were both 161KV connections to Reserve AC Power Supply incapacitated because of thermal overloads, voltage variation, or frequency deviations.

Table 8.2-1 provides a history of outages for the lines to be connected to the reserve switchyard. The present operating record of the Fort Loudoun-2 K-31 line, which is to be looped into the reserve switchyard to provide two independent off-site sources for the nuclear plant, is 4.61 outages per 100 year-miles. This compares favorably with 5.17 outages per 100 year-miles for TVA's 8,303 miles of 161KV transmission line in service during 1973. The operating record of each source line supplying the reserve switchyard is expected to be as good as or better than the average for the system total.

8.2.2.3 Effects of Electrical Faults

A fault between the main generator circuit breaker, the main power transformer and the unit station service transformers is sensed by one or more differential relays with overlapping zones. These relays trip the appropriate 161KV circuit breaker(s), the generator circuit breaker, the generator field circuit breaker and the 13.8KV and 4.16KV breakers connected to the Plant AC distribution system to isolate the fault. This causes loss of the Plant Power Supply and the CRBRP Preferred AC Power Supply (see Section 8.2.1.2 for a description of the CRBRP Preferred AC Power Supply).

The Reserve AC Power Supply is designed to be available within 6 cycles after loss of the CRBRP Preferred AC Power Supply (see Section 8.2.2.1 for a description of the Reserve AC Power Supply).

Each of the four 161KV transmission lines connected to the generating and reserve switchyards is protected by means of a directional comparison relay system and a back up relay system. The directional comparison relaying determines the fault location and trips the necessary 161KV breakers to isolate the fault. The back-up relaying will initiate a transfer-trip of the transmission line breaker at the distant end and the tripping of the local breakers necessary to isolate the line and the faulty breaker.

In the event of a turbine trip coupled with a fault in one of the two 161KV transmission lines in the generating switchyard, the faulted line is tripped, the generator circuit breaker is tripped and power is delivered to the Plant AC distribution system via the second 161KV transmission line. In the event of a turbine trip coupled with a fault in both 161KV transmission lines in the generating switchyard, the Plant Power Supply and the CRBRP Preferred AC Power Supply both are lost.

The generating switchyard is protected as follows: (See Figure 8.3-1 for identification of the 161KV circuit breakers and other switchyard equipment.)

a) The plant overall differential relay protects bay-2 bus and equipment and will trip circuit breakers 924 and 928 (if closed).

b) A fault in bay-1 or bay-3 will be cleared by transmission line relaying which will trip circuit breakers 914 and 934, respectively.

c) The bus-1 differential relay protects the equipment within the differential zone by tripping all the circuit breakers connected to bus-1.

d) A fault in bus-2 (if energized) will be cleared by the transmission line relays which trip circuit breaker 928.

The reserve switchyard is protected as follows:

a) A fault in bay-6 will be cleared by transmission line relays which will trip circuit breaker 978, associated medium voltage breakers, and transfer-trip the corresponding K-31 breaker. Similarly, a fault on the Fort Loudoun-2 line side of breaker 974, associated medium voltage breakers, and transfer-trip the corresponding Fort Loudoun-2 line breaker.

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b) A fault in between breakers 978 and 974 in bay-7 will be cleared by the differential relay which will trip circuit breakers 974 and 978.

Each of the two reserve station service transformers is provided with differential, temperature, sudden pressure and breaker failure timing relays. Upon sensing a fault, the relays trip the 974 and 978 switchyard circuit breakers, the corresponding 13.8KV and 4.16KV breakers, actuate the appropriate local fault initiating switch 96G1 or 98G1, and initiate a signal to trip and lock out the appropriate remote K-31 or Fort Loudoun-2 line breaker by a transfer-trip signal. In addition, backup ground fault protection is provided by a relay installed in the neutral circuit of each reserve station service transformer. Upon sensing a fault, these relays will trip circuit breaker 974 and 978 to effectively separate the two reserve station service transformers. After a time delay, the appropriate local fault initiating switch is actuated, and a signal is initiated to trip and lock out the K-31 or the Fort Loudoun-2 line remote breaker by a transfer-trip signal.

TABLE 8.2-1

FORT LOUDOUN K-31 OUTAGE HISTORY

NOTE: This transmission line, connecting with Fort Loudoun, was extended in December 1968 to establish the present Fort Loudoun K-31 connection.

Year	No. of Outages	Date of Outage	Cause of Outage	Type of Fault	Duration of Outage	Automatic Reclosing
1969	0					
1970	0					
1971	0					
1972	1	4/11/72	Lightning	AØ-CØ-Gnd.	Momentary	Successful
1973	1	7/10/73	Lightning	CØ-Gnd.	Momentary	Successful
1974	0					
1975	0					
1976	0					
1977	0					
1978	0					
1979	0					
1980	0					
1981	0					

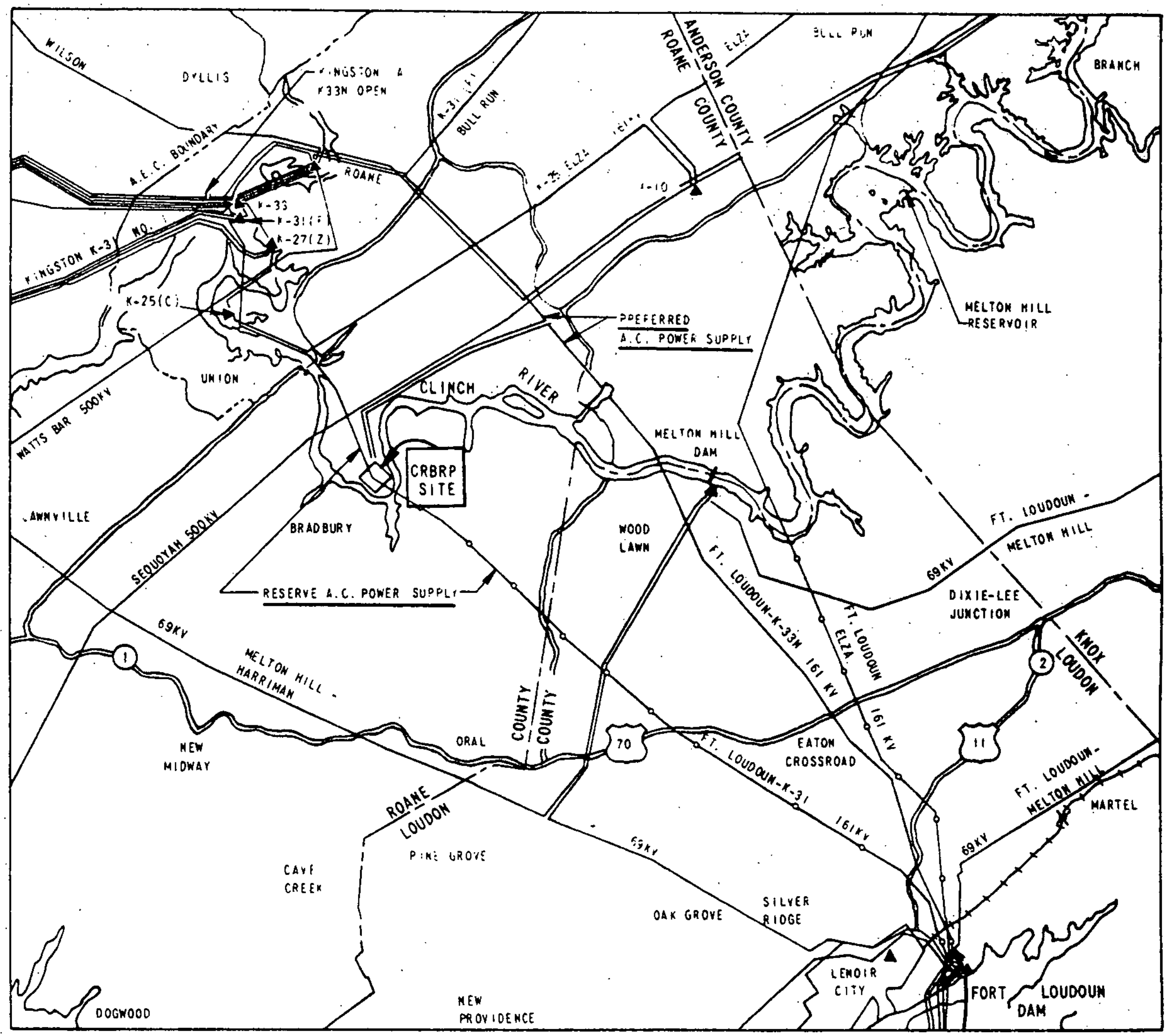
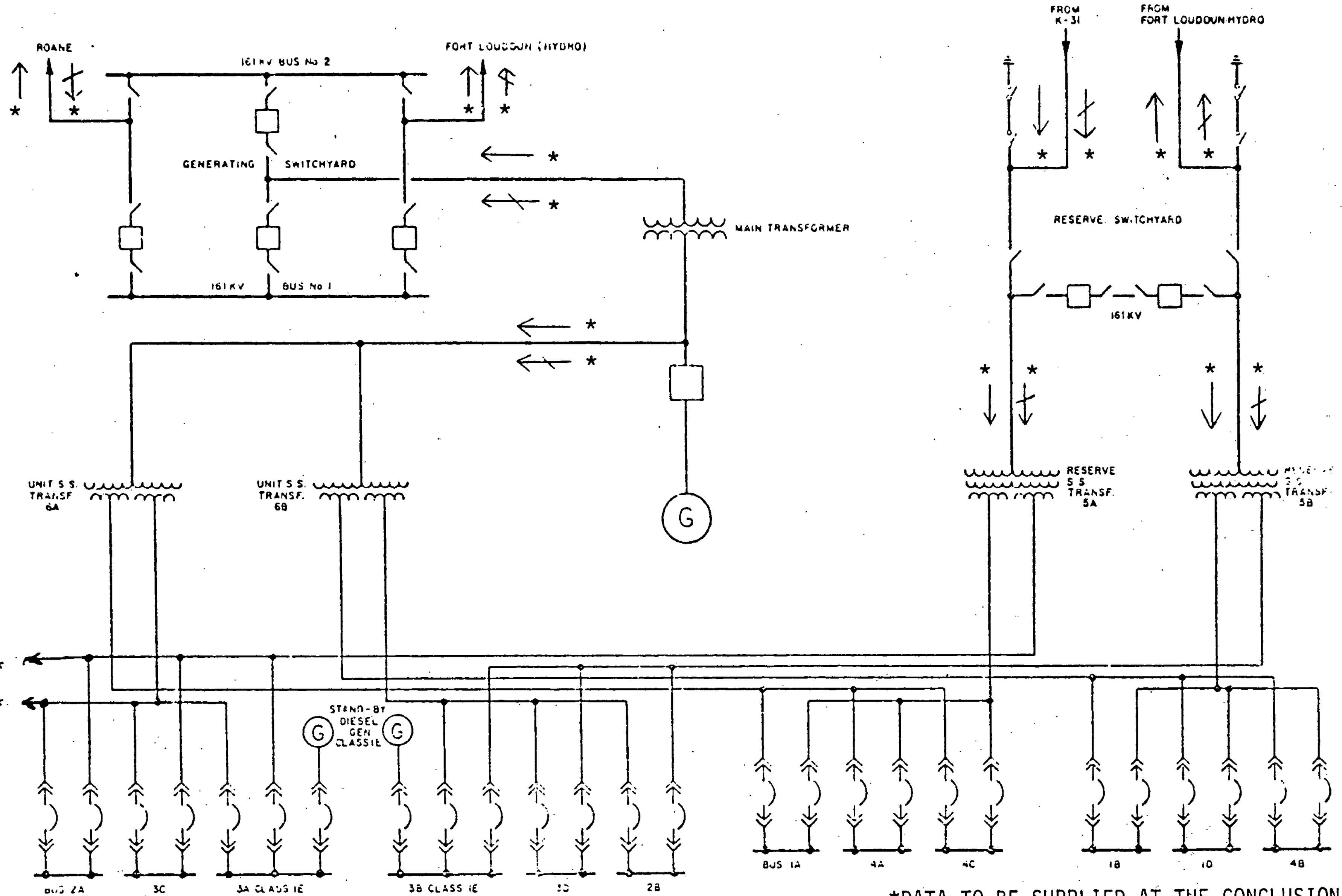


Figure 8.2-1. CRBRP Preferred and Reserve AC Power Supplies

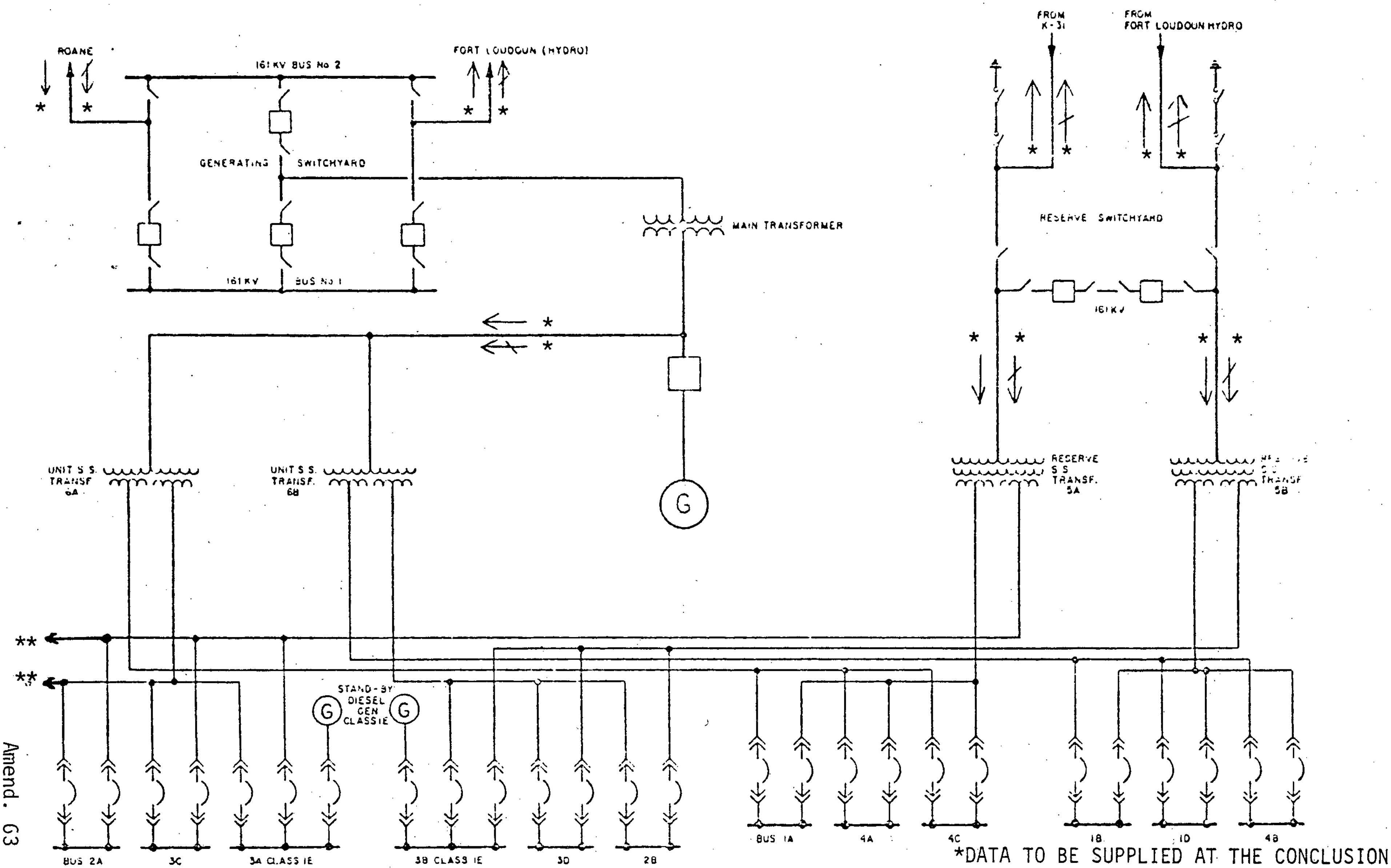


**To Division 1E Bus and DG.

*DATA TO BE SUPPLIED AT THE CONCLUSION OF LOAD STUDIES.

⟶ MW
⟶ MVAR

Figure 8.2-2 CRBRP Off-Site Power Supply - System Normal



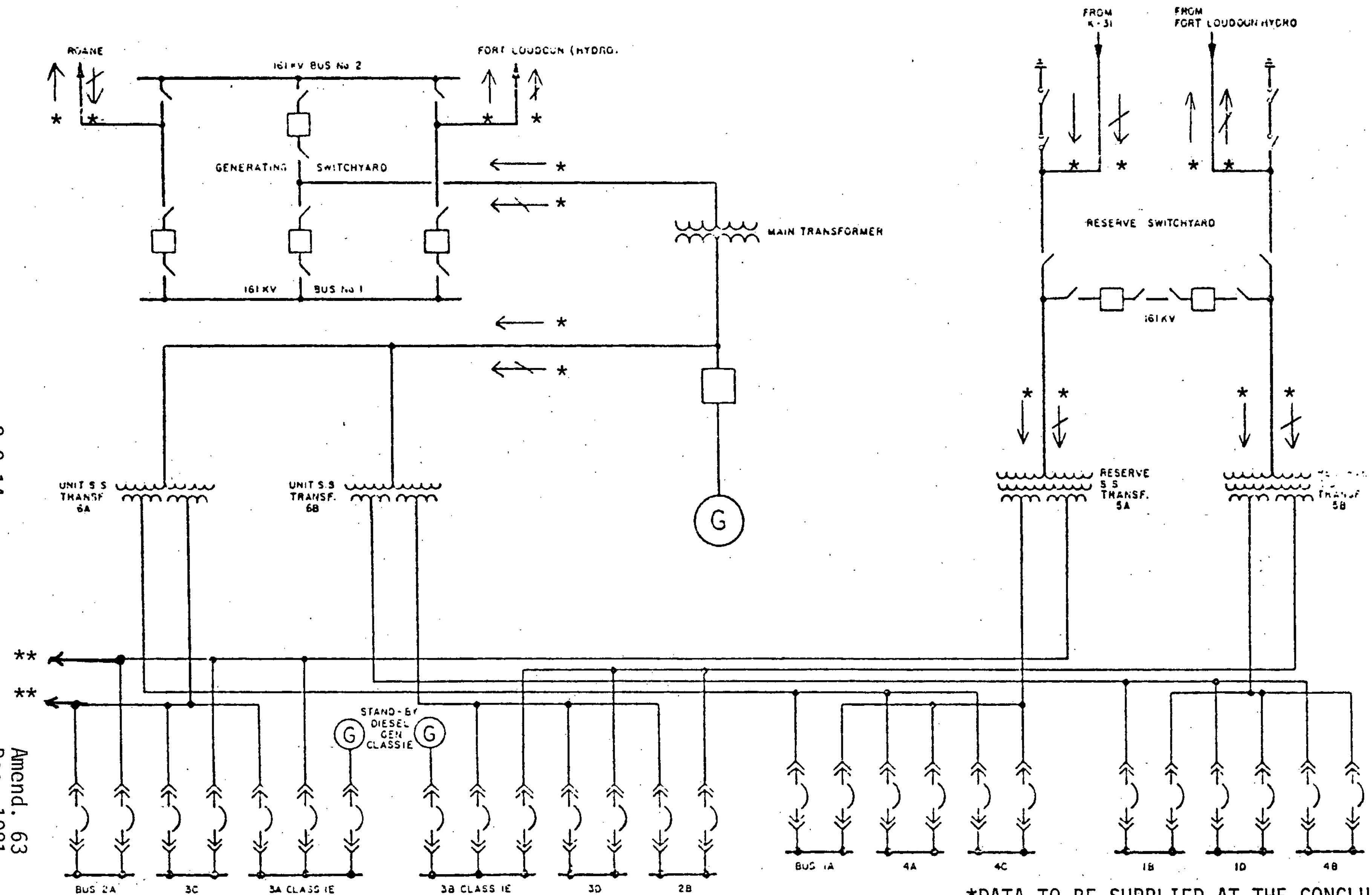
**To Division 1E Bus and DG.

*DATA TO BE SUPPLIED AT THE CONCLUSION OF LOAD STUDIES.

→ MW

↛ MVAR

Figure 8.2-3 CRBRP Off-Site Power Supply - Loss of CRBRP Generation



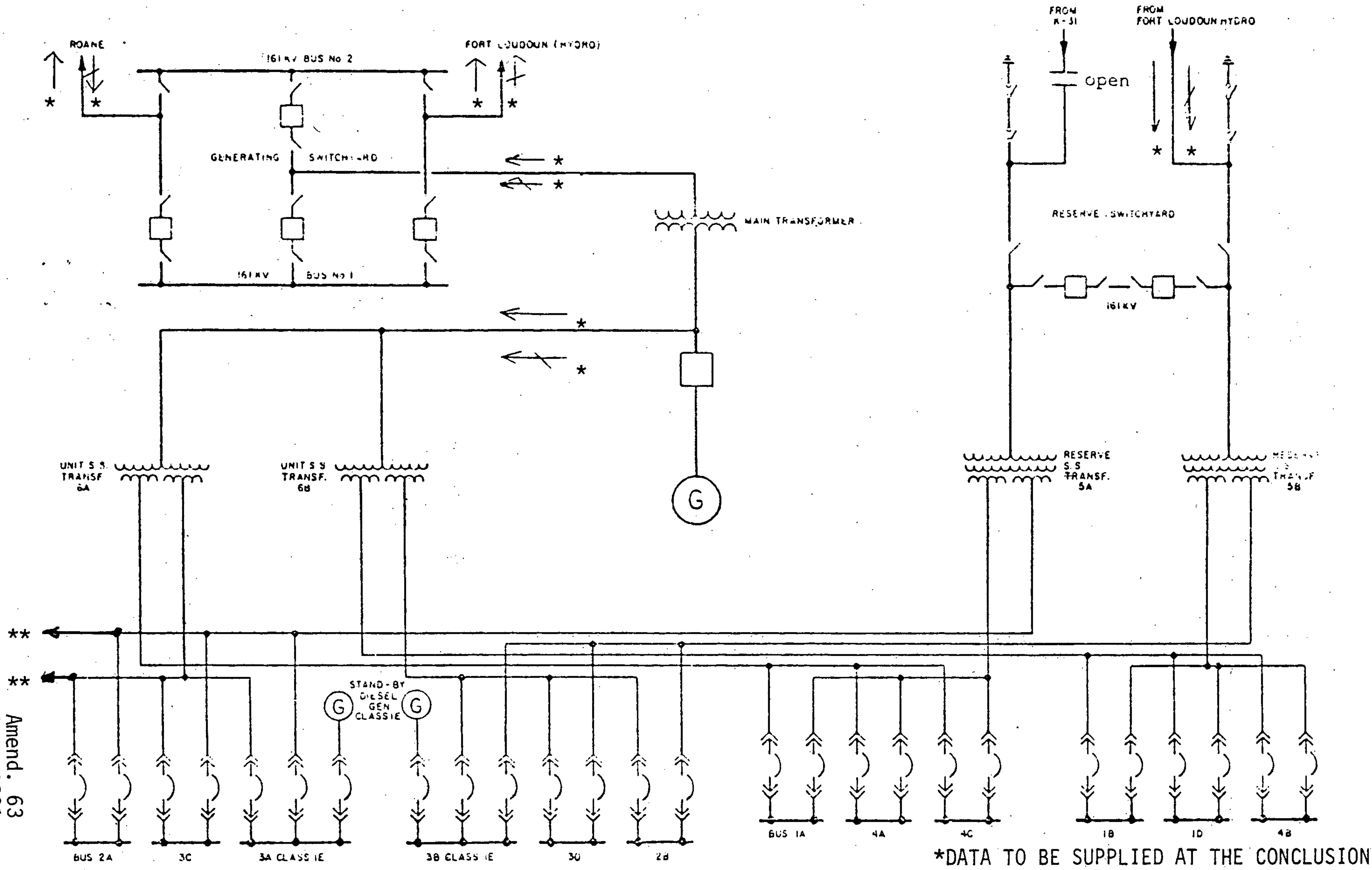
*DATA TO BE SUPPLIED AT THE CONCLUSION OF LOAD STUDIES.

**To Division 1E Bus and DG.

Figure 8.2-4 CRBRP Off-Site Power Supply - Loss of One Cumberland Unit

→ MW

⇸ MVAR

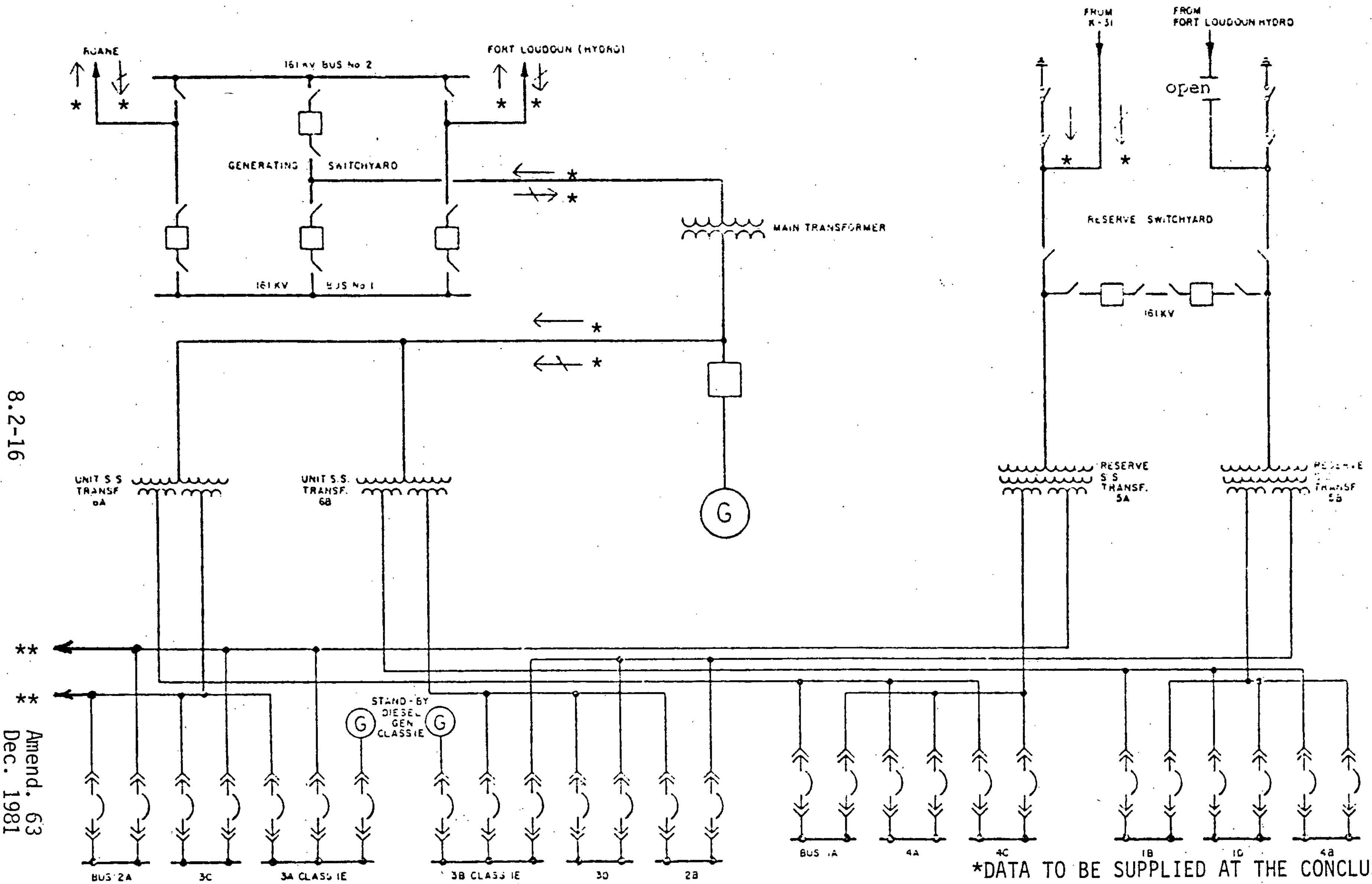


→ MW **To Division 1E Bus and DG.

MVAR

*DATA TO BE SUPPLIED AT THE CONCLUSION OF LOAD STUDIES.

Figure 8.2-5 CRBRP Off-Site Power Supply - Loss of K31 Connection to Reserve Switchyard



*DATA TO BE SUPPLIED AT THE CONCLUSION OF LOAD STUDIES.

**To Division 1E Bus and DG.

⟶ MW

⟶ MVAR

Figure 8.2-6 CRBRP Off-Site Power Supply - Loss of Fort Loudon Connection to the Reserve Switchyard

DATA TO BE SUPPLIED UPON COMPLETION OF LOAD STUDY

Figure 8.2-7. 161kV Generating Switchyard Voltage Performance During Loss of CRBR Unit - No Electrical Fault

DATA TO BE SUPPLIED UPON COMPLETION OF LOAD STUDY

Figure 8.2-8. 161kV Generating Switchyard Frequency Performance During Loss of CRBR Unit - No Electrical Fault

DATA TO BE SUPPLIED UPON COMPLETION OF LOAD STUDY

Figure 8.2-9. 161kV Reserve Switchyard Bus Voltage Performance During Loss of Fort Loudoun 161kV Line Supplying Offsite Power Under Most Severe Transmission Disturbance

DATA TO BE SUPPLIED UPON COMPLETION OF LOAD STUDY

Figure 8.2-10. 161kV Reserve Switchyard Bus Frequency Performance During Loss of Fort Loudoun 161kV Line Supplying Offsite Power Under Most Severe Transmission Line Disturbance

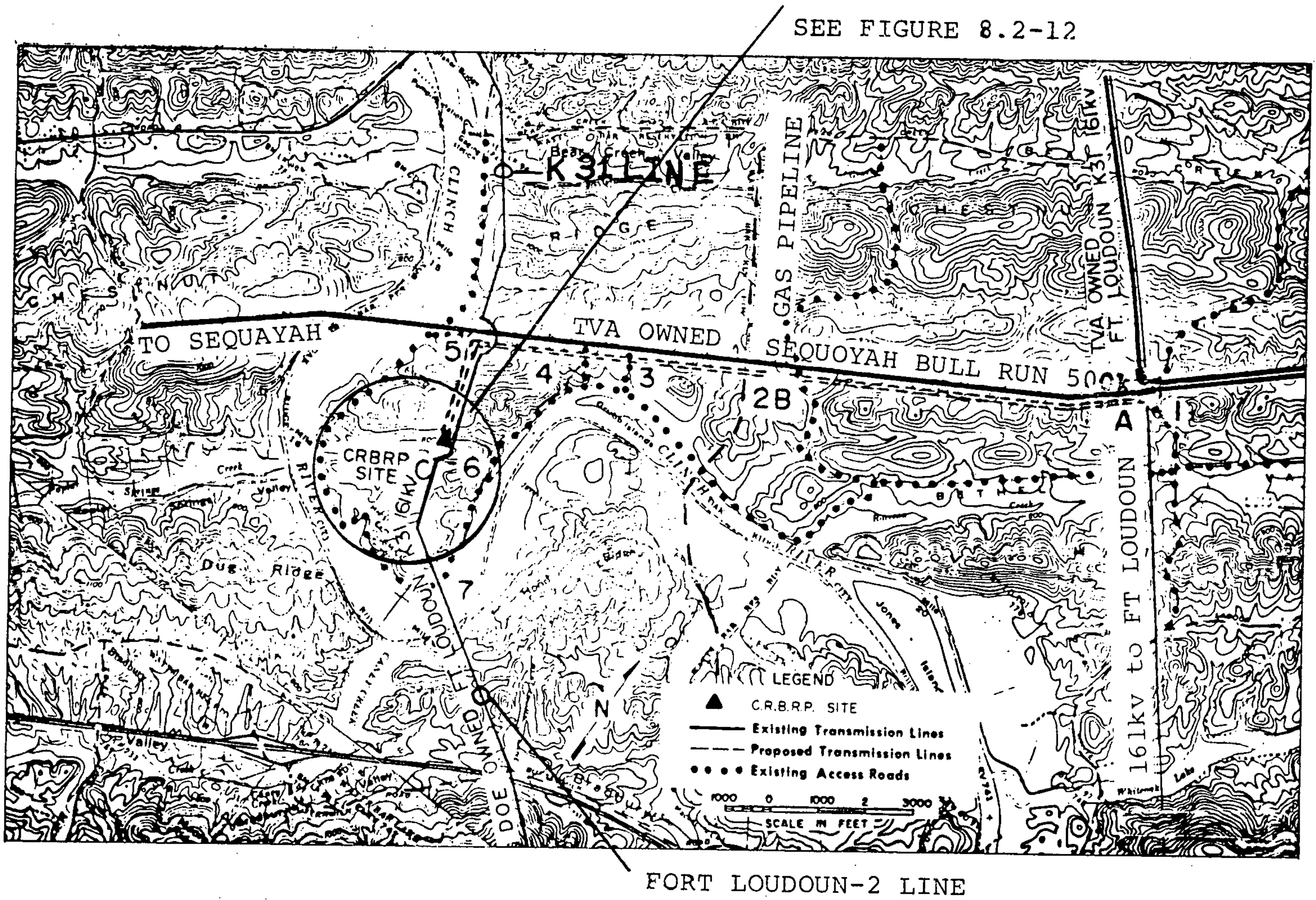
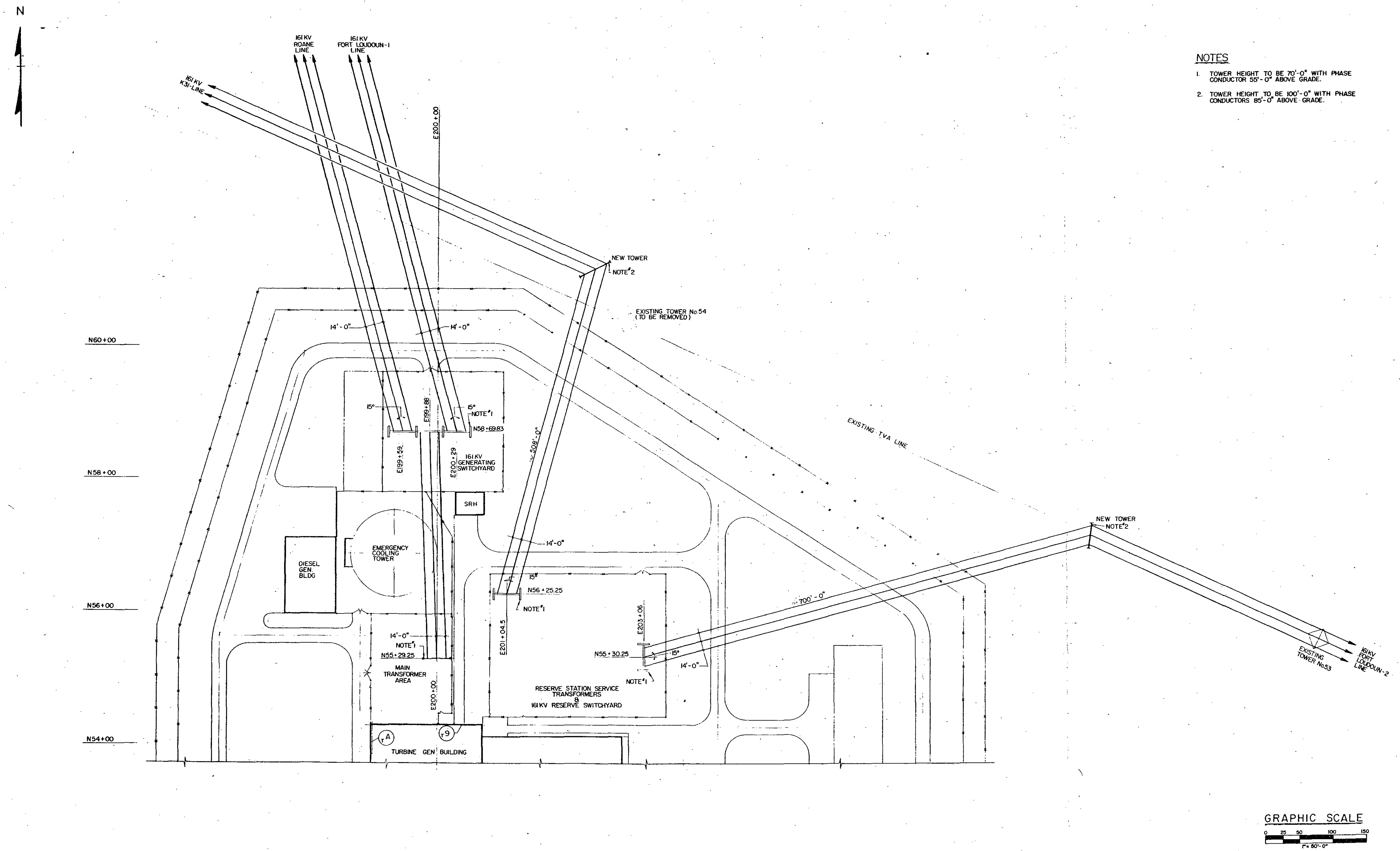
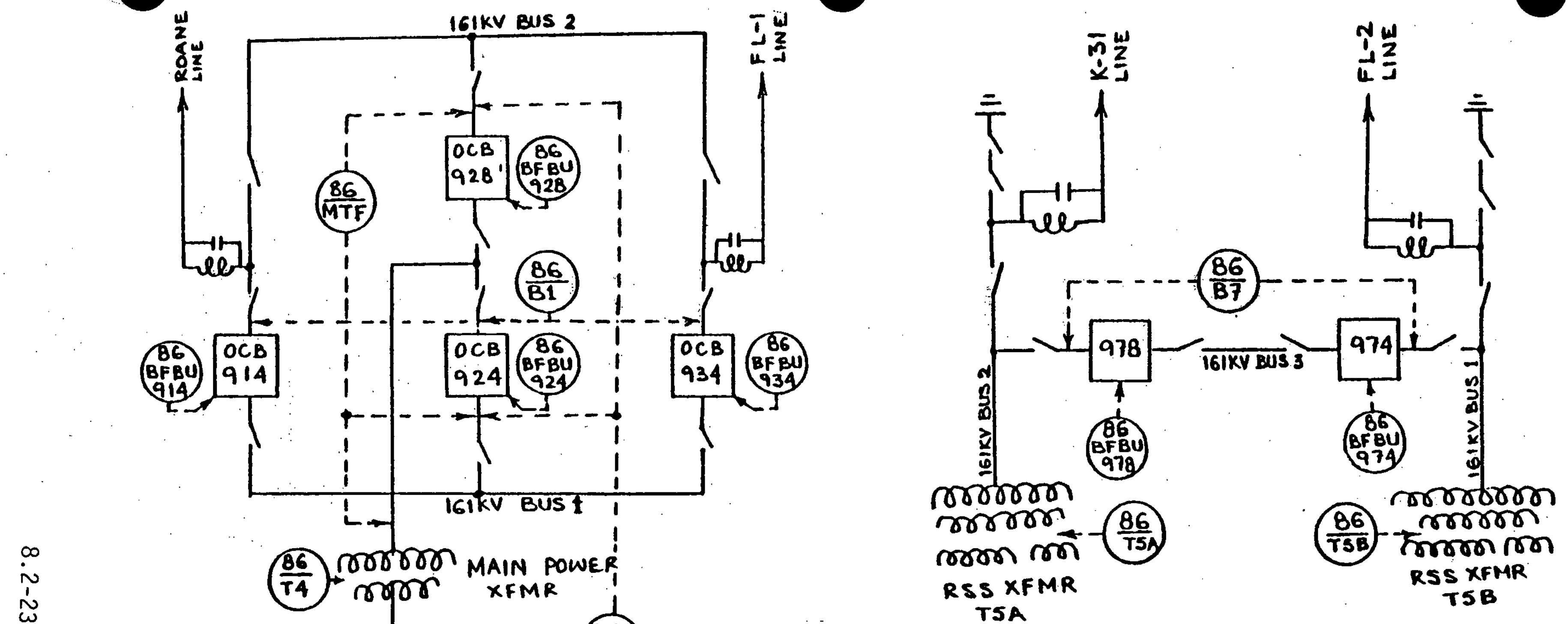
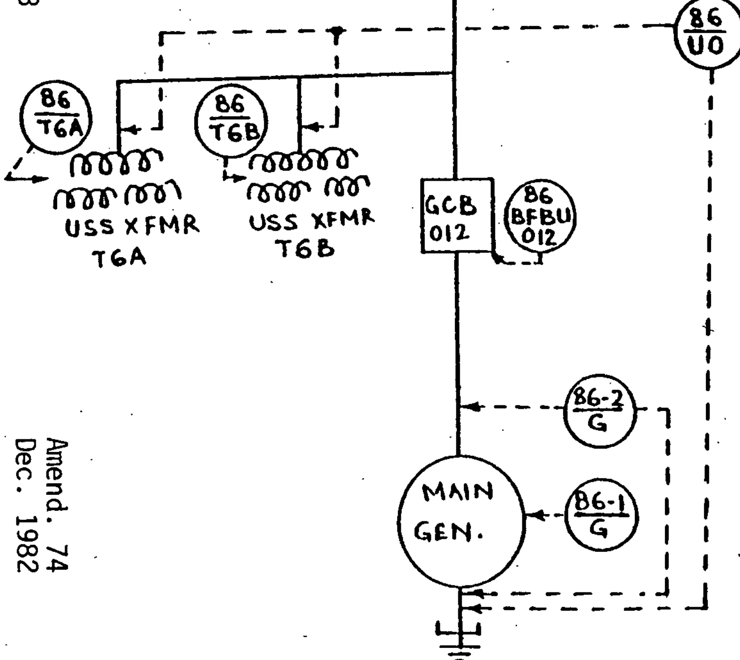


Figure 8.2-11 Proposed Transmission Line Route of the CRBRP Site Area



161 KV TRANSMISSION LINES FROM TVA GRID TO CRBRP
FIGURE 8.2 - 12

Division A 125V DC Distribution System Supplies Power to:

1. Carrier, transfer trip and primary line relays of GSY trans. line
2. GSY and main turbine generator lockout relays & GSY XFMR protection
3. GSY breakers and gen. circuit breaker's closing & primary trip coils
4. GSY spare breaker line relaying
5. Secondary line relaying of RSY trans. lines
6. RSY breakers' secondary trip coils and breaker failure of lockout relays

Division B 125V DC Distribution System Supplies Power to:

1. Carrier, transfer trip and primary line relays of RSY trans. lines
2. RSY XFMR protection and RSY lockout relays
3. RSY breakers closing and primary trip coils
4. Secondary line relaying for GSY trans. lines
5. GSY breakers secondary trip coils and breaker failure of lockout relays.

FIGURE 8.2-13

CRBRP POWER TRANSMISSION SYSTEM, PROTECTION & CONTROL

Note: Tripping of both GSY breakers 924 and 928 by any relay will transfer auxiliary loads to RSY power supply.

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8.3 ON-SITE POWER SYSTEMS

8.3.1 AC Power Systems

8.3.1.1 Description

The on-site power system consists of the following:

a) Non-Class 1E power distribution system which consists of two generally independent load groups (Division A and B). Each division is provided with its own:

- power supplies (13.8KV, 4.16KV, 480 volts, 277 volts, 208 volts and 120 volts AC)
- transformers
- cables and raceways
- 125 volts DC control and instrumentation power
- multiplexer system for control, alarm and indication
- 120/208 volts uninterruptible power supplies (UPS) for essential Non-Class 1E loads

b) Class 1E power distribution system which consists of three independent load groups (Division 1, 2 and 3). Class 1E Divisions 1 and 2 provide the two redundant safety related load groups. Each of the three load groups consists of its own:

- power supplies (4.16KV, 480 volts, 277 volts, 208 volts and 120 volts AC)
- standby (on-site) diesel generator
- transformers
- cables and raceways
- 125 volts DC control and instrumentation power
- 120/208 volts uninterruptible power supplies (UPS) for essential Class 1E loads
- solid state programmable logic system for control, diesel generator load sequencing, periodic testing, and alarm indications

Class 1E Division 3 provides power for Loop 3 decay heat removal system.

Each of these divisions is separated physically and electrically from the other two divisions as described in Section 8.3.1.4, and has the capability to shutdown the plant safely. However, from the consideration of connected loads, Class 1E Divisions 1 and 2 provide power to redundant load groups and as such are referred to as redundant divisions. Class 1E Division 3 provides Class 1E power to Loop 3 of the Heat Transport System (HTS) and to certain plant Non-Class 1E loads. (The Non-Class 1E loads are connected through an isolation subsystem). Since not all loads powered from Division 3 are identical or similar to those powered by Divisons 1 or 2, this division is not identified as redundant to Division 1 or 2. However, as far as the HTS is concerned, the Divisions 1, 2 and 3 power supplies are fully redundant serving the Loops 1, 2 and 3 Class 1E loads, respectively. Each of these three Divisions is capable of shutting down the plant safely.

13.8KV and 4.16KV Distribution System

During normal operation, plant auxiliary power is provided by two (2) 50 percent capacity (50 percent capacity of total plant electrical loads, but 100 percent of loads for one safety division) unit station service transformers (USSTs) fed from the main generator through the 22KV isolated phase bus and the generator circuit breaker.

NRC Regulatory Guidance

The following pages provide discussion of design features which address the guidance of NRC Regulatory Guides 1.6, 1.9, 1.22, 1.32, 1.40, 1.41, 1.47, 1.53, 1.63, 1.68, 1.75, 1.93, 1.100, 1.106, 1.118, 1.128, 1.129, 1.131 and 1.137. Regulatory Guides are further discussed in Appendix I of the PSAR.

Alternately, the USSTs can be powered from the generating switchyard through the main power transformer and the 22KV isolated phase bus. USSTs will also be used to provide the plant auxiliary power during plant startup. The main generator will be synchronized with the TVA grid at the generator circuit breaker. When the USSTs are not available, the plant auxiliary power will be provided by two (2) 50 percent capacity (50 percent of total plant electrical loads but 100 percent of loads for one safety division) reserve station service transformers (RSSTs) which are fed from the TVA grid via the Reserve Switchyard.

All four transformers (USSTs and RSSTs) will be kept continuously energized during normal plant operation.

Refer to Figure 8.3-1 for connections and ratings of the major equipment of the Plant AC distribution system.

Each medium voltage bus (13.8KV and 4.16KV) of the Non-Class 1E system has two transferrable incoming supplies. Each medium voltage bus (4.16KV) of the Class 1E system has three incoming supplies. One supply is from one of the two USSTs; the second supply is from one of the two RSSTs. The 13.8KV and 4.16KV switchgear buses distribute power to large and medium size motors and to 480 volt switchgear.

480 Volt Distribution System

Power for 480 volt auxiliaries is supplied from unit substation switchgear consisting of 13.8KV/480V or 4.16KV/480V transformers and associated low voltage switchgear. Each 480 volt switchgear is arranged as an independent radial system with each 480 volt bus fed by its own power transformer. No cross-ties are provided between 480 volt buses.

The 480 volt switchgear supplies 460 volt motor loads greater than 100hp and in general equal to or less than 250 hp, motor control centers, and power and control panels of the sodium heat tracing system.

The 480 volt motor control centers feed motor loads greater than 0.33 hp and equal to or less than 100 hp, distribution transformers, motor operated valves and other small loads.

277/480 Volt and 120/208 Volt Distribution System

Power to 277 and 120 volt loads is supplied from 277/480 or 120/208 volt distribution panels. These panels are fed from 480 volt motor control centers through 480-277/480 or 480-120/208 volt distribution transformers.

Vital (Uninterruptible) AC Power Supply System

Power for loads which must be continuously energized under all plant operating modes is supplied from 120/208 volt distribution panels which are fed from the Vital (uninterruptible) AC Power Supply Systems. Principal elements of Non-Class 1E and Class 1E Uninterruptible AC Power Supply sources are shown in Figure 8.3-2.

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8.3.1.1.1 Standby AC Power Supply

The Standby AC Power Supply is a Class 1E system which supplies AC power to the Class 1E and certain essential Non-Class 1E loads when the Plant AC Power Supply, CRBRP Preferred Power Supply, and Reserve AC Power Supply are not available.

The Standby AC Power Supply consists of three Class 1E diesel generators, each supplying power to its own safety group loads. Safety Division 1 and 2 are redundant to each other. The diesel generators are physically and electrically independent of each other. The Divisions 1 and 2 diesel generators supply power to redundant load groups. The diesel generators are sized in accordance with IEEE Standard 387-1977, supplemented by Regulatory Guide 1.9, Rev. 2. The total demand during an emergency condition when off-site AC power supplies are unavailable is within the continuous rating of each diesel generator as indicated in Tables 8.3-1A, 8.3-1B, and 8.3-1C.

Each diesel generator is installed in a separate and independent diesel generator room. These rooms are located in a Seismic Category 1 structure and are capable of withstanding missiles as described in Section 3.8.4.1.4. Auxiliary equipment, local control boards and excitation cubicles associated with each diesel generator are located in the same room with the diesel generator. Except for sensors and other equipment that must be directly mounted on the engine or associated piping, the controls and monitoring instrumentation for the diesel generator unit are installed on two (2) free standing floor mounted panels separate from diesel generator unit skids.

The Diesel Generator sets are installed on their own foundations which are isolated from the main building slab. The control panels are located on the floor area which is considered to be vibration free.

Cables for Standby AC Power Supplies will be installed in their own separate division of the Class 1E raceway system. Cables and raceways of the Standby AC Power Supply will be marked in a distinctive manner as described in Section 8.3.1.4.

The following support systems are those essential auxiliary systems or components required to start and operate the diesel generators.

a. The Safety-Related 125V DC Power System

Each diesel generator is furnished with an independent DC supply from the Safety-Related 125V DC Power System. (Section 8.3.2 describes the 125V Class 1E DC system).

b. The Diesel Generator Fuel Oil Storage and Transfer System

Fuel is provided for starting during initial operation using a shaft driven pump taking suction from a day tank. Fuel is provided for continuous operation using AC powered fuel transfer pumps taking suction from the underground storage tanks to replenish the day tank fuel supply. Each diesel generator is furnished with an independent fuel storage and transfer system. For details refer to Section 9.14.1.

c. Diesel Generator Cooling Water System

Each diesel generator is furnished with an independent cooling water support system. For details refer to Section 9.14.2.

d. The Diesel Generator Starting System

Each diesel generator is furnished with two independent and redundant air starting systems with individual starting air compressors, air tanks, and necessary valves and other accessories for cranking the engine. The two starting systems will be independent and arranged so that failure of one will not jeopardize proper operation of the other. For details refer to Section 9.14.3.

e. The Diesel Generator Lubrication System

Each diesel generator has a lubrication system integral with the diesel engine. For details refer to Section 9.14.4.

f. Diesel Generator Ventilation System

Room ventilation is not required for starting of the diesel generator. During continued operation under load and during testing, the ventilation system will provide adequate cooling air for proper operation of the local instrumentation, control and other equipment. For details refer to Section 9.6.5.

Operation

Each diesel generator will be automatically started upon sensing loss of voltage on the 4.16KV Class 1E bus by undervoltage relays or on an emergency signal as described in Section 8.3.1.1.2.

In order to ensure rapid start capability, the engine lube oil and jacket water systems for each engine will be maintained in a preheated (standby) condition at all times, and all essential elements of the system will be continuously maintained.

Each diesel generator unit operating under an emergency condition will only be tripped by sensing either of the following conditions (all other protection devices will be bypassed):

a. Engine overspeed.

b. Generator differential current.

All diesel generator unit protective devices will remain in force during unit testing.

Loading of each diesel generator is indicated in Tables 8.3-1A, 8.3-1B and 8.3-1C. The diesel generators will be designed to provide voltage and frequency within limits, as described in Section 8.3.1.2, which ensure proper performance of any of the loads during all possible steady-state or transient loadings.

Testing and Inspection

The equipment will be tested and inspected at the vendor's facility prior to delivery. The system will also be inspected during installation to confirm that design requirements have been met.

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Initial operational system tests will be performed with components installed and connected to demonstrate that the system operates within design limits and meets the performance specification, and to verify the independence between redundant AC power sources and load groups.

After being placed in service, the standby diesel generators and their respective associated supply systems will be inspected and tested periodically to detect any degradation of the system.

The preoperational tests and periodic testing after the diesel generator units are placed in service will be performed in accordance with Regulatory Guide 1.108 (Rev. 1, 8/77). Detailed step-by-step procedures will be provided for each test. The procedures will identify those special arrangements or changes in normal system configuration that must be made to put the diesel generator unit under test. Jumpers and other nonstandard configurations or arrangements will not be used subsequent to initial equipment startup testing. During periodic testing, the diesel generator units will be operated at a load in excess of a minimum of 25% of rated load.

The following tests will be performed as a minimum:

A. Testing of diesel generator units during the plant preoperational test program and at least once every 18 months (during refueling or prolonged plant shutdown) will be performed to:

   (1) Demonstrate proper startup operation by simulating loss of all AC voltage and demonstrate that the diesel generator unit can start automatically and attain the rated speed (450RPM) within 10 seconds. Verify that the generator voltage and frequency are at their rated values within 10 seconds after the start signal.

   (2) Demonstrate proper operation for design-accident-loading sequence to design-load requirements in Tables 8.3-1A, 8.3-1B, and 8.3-1C. Verify that at no time during the loading sequence will the frequency and voltage decrease to less than 95% of nominal and 75% of nominal, respectively. Verify that the frequency is restored to 98% of nominal and the voltage is restored to 90% of nominal within 4 seconds for each load sequence time interval.

   (3) Demonstrate full-load-carrying capability for an interval of not less than 24 hours, of which 22 hours will be at a load equivalent to the continuous rating of the diesel generator unit and 2 hours at a load equivalent to the 2 hour rating of the diesel generator unit. Verify that voltage and frequency are maintained at rated values. The test will also verify that the cooling system functions within design limits.

(4) Demonstrate proper operation during diesel generator unit load shedding, including a test of the loss of the largest single load (1050 hp) or of complete loss of load. Verify that, during recovery from transients resulting from the disconnection of the largest single load and of the complete loss of the continuous rated load, the speed of the diesel generator unit will not exceed 105% of nominal speed. (The overspeed trip set point will be at 110% of nominal speed.) Verify that the generator voltage will not exceed 110% of nominal voltage during and following the load rejection.

(5) Demonstrate functional capability at full-load temperature conditions by rerunning the test phase outlined in A.(1) and A.(2) above immediately following A.(3) above.

(6) Demonstrate the ability to (a) synchronize the diesel generator unit with off-site power while the unit is connected to the emergency load, (b) transfer this load to the offsite power, (c) isolate the diesel generator unit, and (d) restore it to standby status.

(7) Switching from one fuel oil supply system to another is not required and therefore not provided, because each diesel generator unit has its own independent fuel oil storage tank sized for a minimum of 7 days operation.

(8) Demonstrate that the capability of the diesel generator unit to supply emergency power within the required time is not impaired during periodic testing under "C" below.

(9) During the plant preoperational test program only, demonstrate the required reliability by means of any 69 consecutive valid tests (as defined in Regulatory Guide 1.108, Regulatory Position C.2.e) with no failures with a minimum of 35 tests per diesel generator unit.

B. Testing of redundant diesel generator units during normal plant operation will be performed independently (nonconcurrently) to minimize common failure modes resulting from undetected interdependences among diesel generator units. However, during reliability demonstration of diesel generator units during plant preoperational testing and testing subsequent to any plant modification where diesel generator unit interdependence may have been affected or every 10 years (during a plant shutdown), whichever is shorter, a test will be conducted in which redundant units are started simultaneously to help identify certain common failure modes undetected in single diesel generator unit tests.

C. Periodic testing of diesel generator units during normal plant operation will be performed to:

(1) Demonstrate proper startup and verify that the diesel generator unit can attain rated speed within 10 seconds. Verify that the generator voltage and frequency are at their rated values within 10 seconds after the start signal. This test will also verify that the components of the diesel generator unit required for automatic startup are operable.

(2) Demonstrate full-load-carrying capability (continuous rating) for an interval of not less than one hour. The test will also verify that the cooling system functions within design limits. This test will be accomplished by synchronizing the generator with the off-site power and assuming a load at the maximum practical rate.

D. After completion of the diesel generator unit reliability demonstration under A.(9) above, the interval for periodic testing under "C" above (on a per diesel generator unit basis) will be no more than 31 days and will depend on demonstrated performance. If more than one failure has occurred in the last 100 tests the test interval will be shortened in accordance with the schedule recommended by the Regulatory Guide 1.108, Regulatory Position C.2.d. The schedule will be included in Chapter 16, Technical Specifications.

E. Criteria for valid tests and failures for the tests under A.9 and "D" above will be based on the criteria recommended by the Regulatory Guide 1.108, Regulatory Position C.2.e.

8.3.1.1.2 Safety-Related AC Power Distribution System

The Safety-Related AC Power Distribution System is a Class 1E system which distributes power to Class 1E equipment that requires AC power. The Safety Related (Class 1E) Distribution System consists of three independent divisions (Divisions 1, 2 & 3).

The Divisions 1 and 2 distribute power to the two redundant Class 1E load groups. The Division 3 provides power to the Loop 3 decay heat removal system, the Division 3 PPS, control, instrumentation, alarms and indication. The power is distributed to the loads at the following voltage levels:

a. 4.16KV for large and medium sized motor-driven equipment.

b. 480 volts for motor-driven auxiliaries, heating and ventilation systems, and other miscellaneous loads.

c. 480/277 volts for lighting and other miscellaneous loads.

d. 120 volts for control, instrument power, lighting and other miscellaneous loads.

The Divisions 1, 2 and 3 are separated into three sets of cables, raceways, components and equipment, each set distributing power to its own load group. The three sets are physically and electrically independent of each other. Each division receives AC power from the Plant Power Supply, the CRBRP Preferred or Reserve AC Power Supplies or the Standby Onsite AC Power Supply.

Equipment ratings for the 4.16KV and 480 volt loads on the Safety-Related AC Power Distribution System are indicated in Figure 8.3-3. Power to 4.16KV Class 1E loads is distributed through Class 1E switchgear buses.

Power to 480 volt Class 1E loads is distributed from Class 1E 480 volt Unit Substation Switchgear which is supplied from unit substation transformers connected to the 4.16KV Class 1E switchgear buses. Each 480 volt Class 1E switchgear receives power from the 4.16KV switchgear bus of the same load group. The switchgear distributes power directly to Class 1E low voltage motors rated above 100 HP.

The 480 volt Class 1E switchgear also distributes power to Class 1E 480 volt motor control centers which contain motor starters or circuit breakers for loads such as small Class 1E motors, heating and ventilation system loads, feeder circuit breakers for vital instrument AC regulating transformers with a rating of 480-120/208V AC, and the 125 volt DC battery chargers. Certain Non-Class 1E loads are connected to the Division 3 Class 1E diesel generator system switchgear via a high impedance transformer isolation system as described in Section 8.3.1.2.14.

Motor control center loads such as motors are controlled by combination units which are installed in the motor control centers. These combination units consist of a starter or contactor for controlling the load and a circuit breaker for fault protection of the feeder cable and the load equipment. Motor loads are provided with overload protection. 120 volt AC control power for each motor control center starter or contactor is supplied from an individual control power transformer. These transformers receive power from the load side of the circuit breaker furnished in each combination starter or contactor.

Auxiliary devices required to operate equipment in one load group receive power from distribution panels in the same load group. This arrangement prevents the loss of electrical power in one load group causing loss of equipment in the other load group.

All components of the Safety-Related AC Power Distribution System are located in Seismic Category I structures. Each of the two redundant Class 1E 4.16KV switchgear buses is installed in a separate electrical equipment room. The electrical equipment rooms are located in a Seismic Category I structure. The switchgear for Division 3 loads is also located in an electrical area separate from Division 1 and 2 switchgear rooms.

Cables for the separate, redundant load groups (Division 1 and 2) of the Safety-Related AC Power Distribution System are installed in separate raceway systems. The independence criteria of the Class 1E raceway systems is described in Section 8.3.1.4. Cables and raceways of the Safety-Related AC Power Distribution System are marked in a distinctive manner as described in Section 8.3.1.5.

The following support systems are those essential components which are required for proper functioning of the Safety-Related AC Power Distribution System:

a. <u>The 125 Volt DC Power Supply</u>

DC power is required for control of the 4.16KV circuit breakers and 480 volt switchgear circuit breakers. Each group is furnished with an independent DC Supply from the DC Power Supply System as described in Section 8.3.2. DC power is not required to support the 480 volt motor control centers.

b. <u>Electrical Equipment Room HVAC Requirements</u>

Room ventilation is not required for initial operation of the safety-related electrical equipment. Adequate heating or cooling air is provided for proper functioning of safety-related local instrumentation, protective relays, control devices and other electrical equipment. Each HVAC unit is furnished with Class 1E power from the same division it serves. For details, refer to Section 9.6.

<u>Operation</u>

When the Main Generator is operating, the Safety-Related AC Power Distribution System receives power from the Plant Power Supply via the generator circuit breaker and unit station service transformers. In the event of a turbine trip, a reactor trip, a main generator fault, or fault in the isolated phase bus system between the main generator and generator circuit breaker, the generator circuit breaker is automatically tripped. The unit station service transformers will remain connected to the CRBRP Preferred AC Power Supply (161KV generating switchyard) via the main step-up transformer and provide uninterrupted power through the Plant AC Distribution System to the Safety-Related AC Power Distribution System. An electrical fault downstream of the generator circuit breaker will cause the tripping of the associated 161KV circuit breaker(s) located in the generating switchyard and subsequent loss of the power supply from the unit station service transformers.

Upon loss of the power supply from the unit station service transformers, the Plant AC Power Distribution System will be automatically connected to the Reserve AC Power Supply as described in Section 8.3.1.1.4.

Upon loss of the Reserve AC Power Supply, the undervoltage sensors in the 4.16KV Class 1E switchgear buses sense the loss of power and initiate the following:

1. Trip the circuit breakers connecting the Class 1E switchgear buses to the reserve station service transformers.

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2. Start the diesel generator and close the diesel generator circuit breaker, subject to the permissives in the Standby AC Power Supply.

The diesel generators will also start automatically on an emergency signal. However, if the CRBRP Preferred Power Supply is still available to the safety-related 4.16KV switchgear bus, each diesel generator breaker will remain open. The required safety-related loads will be sequenced on each Class 1E bus by a diesel generator load sequencer as shown on the diesel generator loading Tables 8.3-1A, 8.3-1B and 8.3-1C. The diesel generators will be shutdown manually by the plant operator in accordance with strict administrative controls.

Upon a successful start of the Standby AC Power Supply (diesel generators), voltage at the 4.16KV Class 1E switchgear bus will be restored. This permits automatic sequential loading on the Safety-Related AC Power Distribution System, as required to maintain safe shutdown of the plant. Each load which is required to start automatically will receive its own start signal from the diesel generator sequencer. The required loads will be connected, as shown in Tables 8.3-1A, 8.3-1B and 8.3-1C.

Upon loss of the power supply from the unit station service transformers or the reserve station service transformers, the Class 1E 480 volt switchgear will remain connected, unless intentionally tripped. Upon restoration of the voltage, the 480 volt Class 1E switchgear will be automatically re-energized.

On restoration of the CRBRP Preferred or the Reserve Power Supply, the 4.16KV Class 1E switchgear bus will be manually synchronized with one of these available power supplies and the incoming 4.16KV circuit breaker of the diesel generator will be opened and the diesel generator will be stopped by the operator under strict administrative controls.

Circuit Protection

Each supply circuit breaker of the Safety-Related AC Power Distribution System is protected for any of the following conditions:

a. Overcurrent

b. Ground fault

c. Transformer differential overcurrent

d. Input terminal undervoltage

The Class 1E diesel generators are each protected against the following:

1) Engine overspeed

2) Low engine oil pressure

3) Generator differential overcurrent

4) Generator overcurrent

5) Reverse power flow to generator

6) Generator loss of field

7) Generator Ground overvoltage

8) Generator field ground

Upon detection of one of the above conditions when the diesel generator is running under test with normal plant conditions, the lockout relay will operate, which will trip the diesel generator incoming bus circuit breaker. After automatic start under a plant emergency condition, all protective functions mentioned above will be bypassed except for the following:

1) Generator differential overcurrent

2) Engine overspeed

The diesel generator circuit breaker is included in the differential protective zone of the diesel generator. The diesel generator will be automatically tripped for a three phase fault or phase-to-phase fault on the generator or on the generator supply feeder. It will also be tripped automatically in case of a line-to-ground fault when the diesel generator is operating in test mode.

The design of the bypass circuitry will satisfy the requirements of IEEE Std. 279-1971 at the diesel generator system level and will include the capability for (1) testing the status and operability of the bypass circuits, (2) alarming in the Control Room abnormal values of the bypass parameters, and (3) manually resetting of the trip bypass function.

The following types of protection are provided for the Safety-Related AC Power Distribution System:

<u>4.16KV Switchgear</u>

1. Each 4.16KV Class 1E switchgear bus and the incoming lines are provided with undervoltage relays to detect an undervoltage condition. The relays initiate tripping of the circuit breakers connecting the Class 1E switchgear to the off-site power sources and automatic start of diesel generators.

2. In addition to the undervoltage relays provided to detect loss of off-site power to the safety buses, a second level of undervoltage protection with time delay is provided to protect the Class 1E equipment against sustained degraded voltage conditions of off-site power supplies. This second level of voltage protection will satisfy the following criteria:

a) The selection of undervoltage and time delay setpoints will be determined from an analysis of the voltage requirements of the safety related loads at all Standby On-site System distribution levels.

b) Two separate time delays will be selected for the second level of undervoltage protection based on the following conditions:

   1) The first time delay will be of a duration that establishes the existence of a sustained degraded voltage condition (i.e., something longer than a motor starting transient). Following this delay, an alarm in the Control Room will alert the operator to the degraded condition. The subsequent occurrence of an emergency signal will immediately separate the Class 1E distribution system from the off-site power system.

   2) The second time delay will be of a limited duration such that the permanently connected Class 1E loads will not be damaged. Following this delay, if the operator has failed to restore adequate voltages, the Class 1E distribution system will automatically be separated from the off-site power system.

c) The voltage sensors will be designed to satisfy the following applicable requirements derived from IEEE Std. 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations".

   1) Class 1E equipment will be utilized and will be physically located at and electrically connected to the Class 1E switchgear;

   2) An independent scheme will be provided for each division of the Class 1E power system;

   3) The undervoltage protection will include coincidence logic on a per bus basis to preclude spurious trips of the off-site power source;

   4) The voltage sensors will automatically initiate the disconnection of off-site power sources whenever the voltage set point and time delay limits have been exceeded;

   5) Capability for test and calibration during power operation will be provided.

   6) Annunciation will be provided in the Control Room for any bypasses incorporated in the design.

d) The Class 1E bus load shedding scheme will automatically prevent shedding during sequencing of the emergency loads to the bus. The load shedding feature will, however, be reinstated upon completion of the load sequencing action. The technical specifications will include a test requirement to demonstrate the operability of the automatic bypass and resinstatement features at least once per 18 months during shutdown.

e) The voltage levels at the safety-related buses will be optimized for the maximum and minimum load conditions that are expected throughout the anticipated range of voltage variations of the off-site power sources by appropriate adjustment of the voltage tap settings of the intervening transformers. The tap settings selected will be based on an analysis of the voltage at the terminals of the Class 1E loads. The analyses performed to determine the minimum operating voltages will typically consider maximum unit steady state and transient loads for events such as a unit trip, startup or shutdown; with the off-site power supply (grid) at minimum anticipated voltage and only the off-site source being considered available. Maximum voltages will be analyzed with the off-site power supply (grid) at maximum expected voltage concurrent with minimum unit loads (e.g. shutdown, refueling). A separate set of the above analyses will be performed for each available connection to the off-site power supply.

f) The analytical techniques and assumptions used in the voltage analyses cited above will be verified by actual measurement. The verification and test will be performed prior to inital full-power reactor operation on all sources of off-site power by:

   1) loading the station distribution buses, including all Class 1E buses down to the 120/208 volt level, to at least 30%;

   2) record the existing grid and Class 1E bus voltage and bus loading down to the 120/208 volt level at steady state conditions and during the starting of both a large Class 1E and Non-Class 1E motor (not concurrently);

   3) using the analytical techniques and assumptions of the previous voltage analyses cited above, and the measured existing grid voltage and bus loading conditions recorded during conduct of the test, a new set of voltages for all the Class 1E buses down to the 120/208 volt level will be established.

The voltage sensors will be designed to satisfy the applicable requirements of IEEE Std. 279-1971.

The Technical Specifications, Chapter 16, will include limiting condition for operation, surveillance requirements, trip set points with minimum and maximum limits, and allowable values for the second-level voltage protection sensors and associated time delay devices.

3. Each radial feeder into and out of the 4.16KV Class 1E switchgear buses is provided with phase and ground overcurrent protection. Motor protection is provided by relays that alarm in the Control Room on moderate overloads and trip on short circuit and locked rotor conditions.

4. When the diesel generator is supplying power to maintain safe shutdown of the plant, a line-to-ground fault on the 4.16KV Class 1E bus will be annunciated in the Control Room. No tripping is provided for this situation as the diesel generator grounding system is designed as a high resistance grounding system. The neutral of the generator is grounded through a single distribution transformer. A resistance is connected to the secondary winding of this transformer. The magnitude of the line-to-ground fault current will be limited to a very low value, which will not affect the safe operation of the diesel generator unit. Since possibility of damage to the equipment is minimized, continuity of service can be maintained.

5. Operation of a protective relay or the tripping of a 4.16KV circuit breaker will be annunciated in the Control Room.

480 Volt Switchgear

1. Each 480 volt switchgear bus is provided with an undervoltage relay to detect an undervoltage condition.

2. Each 480 volt switchgear circuit breaker is equipped with solid state trip devices for overcurrent and short circuit fault protection.

3. Operation of the undervoltage relay and tripping of the 480 volt circuit breakers are annunciated in the Control Room.

480 Volt Motor Control Centers

Molded case circuit breakers will provide time overcurrent and/or instantaneous short circuit protection for all connected loads. For motor circuits, the molded case circuit breakers will be equipped with an instantaneous trip only. Motor overload protection will be provided by heater-element trip units in the motor starter. Class 1E motors for the motor operated valves are also provided with thermal overload protection which will be bypassed under plant emergency conditions.

Testing and Inspection

The equipment will be inspected and tested at the vendor's facilities prior to delivery. The equipment will also be inspected during installation to confirm that the design requirements have been met. When installation is complete, pre-operational equipment tests and inspections will be performed.

Initial pre-operational tests will be performed with equipment and components installed and connected to demonstrate that the equipment is within the design limits and the system meets performance specifications. This test will also demonstrate that loss of the Plant Power Supply and Offsite (CRBRP Preferred and Reserve Power) AC power supplies can be detected.

Periodic equipment tests will be performed to detect any degradation of the system and to demonstrate the capability of equipment which is normally de-energized.

Periodic tests on the Class 1E 4.16KV switchgear and 480 volt switchgear circuit breakers will be performed by utilizing the following test methods:

a. The operability of circuit breakers carrying current under normal plant operation will be demonstrated by their performance in supplying power. In addition, the circuit breakers will be tested in "Test" position at regular intervals. During this test, the proper operation of the circuit breakers and the control circuits will be verified.

b. Testing of circuit breakers of the standby equipment will be performed by racking the circuit breakers in the "Test" position. In the "Test" position, the main contact of the circuit breaker are disconnected, but the auxiliary and the control circuits are maintained. This facilitates functional tests of the circuit breaker and its control circuit.

c. The operability of safety-related circuit breakers will be demonstrated by their performance in supplying power to safety-related loads during scheduled load performance tests. In addition, functional tests of the circuit breaker and its control circuit will be performed during plant refueling or prolonged plant shutdown.

Periodic tests of the transfer of power between the CRBRP Preferred Power Supply and Reserve AC Power Supplies will be performed during prolonged plant shutdown or during refueling to demonstrate that:

a. Sensors can properly detect loss of the CRBRP Preferred Power Supply and the Reserve AC Power Supplies.

b. Components required to accomplish the transfer from the CRBRP Preferred Power Supply to the Reserve AC Power Supply are operable.

c. Components required to accomplish the transfer from the Reserve AC Power Supply to the Standby On-Site AC Power Supply are operable.

d. Components required to accomplish the transfer from the Off-site Power Supply (simulating the unavailability of the off-site AC power supplies) to the On-Site Standby AC Power Supply are operable.

e. Instruments and protective relays are properly set and operating correctly.

Solid State Programmable Logic System (SSPLS)

- The SSPLS is a Class 1E System.

- The SSPLS controls and actuates safety-related, Class 1E equipment of the CRBRP. The types of equipment controlled are circuit breakers, reversing and non-reversing motor starters and solenoid valves.

- The SSPLS replaces conventional electromechanical relays with solid state electronic logic.

- The SSPLS consists of three Class 1E systems which are physically separated and electrically independent.

- SSPLS functions are performed in two types of cabinets; a logic cabinet (which contains the control logic, signal conditioning, auxiliary circuits and power supplies) and field termination cabinets (which contain field output relays and field wiring terminal blocks).

- The paired SSPLS cabinets are strategically located in different buildings of the plant with their associated equipment.

- Operator control functions interface with the SSPLS from Control Room panels or Local Stations.

- The diesel generator load sequencers are part of the SSPLS.

8.3.1.1.2.1 Division 3 Power Distribution System

The Division 3 Power Distribution System is a Class 1E system which distributes power to Loop 3 Primary Heat Transport System (PHTS) and Intermediate Heat Transport System (IHTS) Pony motors and Protected Air Cooled Condenser (PACC) blowers, and a Class 1E, Division 3, 120/208V Vital AC power distribution bus as indicated in Figure 8.3-2.

Cables of the Division 3 Power Supply are separated from cables of the remaining Class 1E Plant AC Power Supplies by routing them in conduits or cable trays in separate fire hazard areas.

All components of the Division 3 Power Distribution System are located in Seismic Category I structures.

Cables and raceways of the Division 3 Power Supply System are marked in a distinctive manner as described in Section 8.3.1.5.

Circuit Protection

Each supply circuit breaker of the Division 3 Power System is protected similarly to Divisions 1 and 2 system power distribution.

480 Volt Motor Control Centers

Molded case circuit breakers will provide time overcurrent and/or instantaneous short circuit protection for all connected loads. For motor circuits, the molded case circuit breakers will be equipped with instantaneous trip only. Motor overload protection will be provided by heater-element trip units in the motor starter.

Testing and Inspection

The equipment will be inspected and tested at the vendor's facilities. The equipment will also be inspected during installation to confirm that the design requirements have been met. When installation is complete, preoperational equipment tests and inspections will be performed.

Initial pre-operational tests will be performed with equipment and components installed and connected to demonstrate that the equipment is within design limits and the system meets performance specifications.

Periodic equipment tests will be performed to detect any degradation of the system and to demonstrate the capability of equipment which is normally de-energized.

8.3.1.1.3 Plant Power Supply

The Plant Power Supply generates and delivers AC power to the TVA 161KV grid. In addition, the Plant Power Supply is the normal source of AC power for the Class 1E and Non-Class 1E auxiliary loads when the plant is operating.

The Plant Power Supply includes the main generator, the generator circuit breaker, the interconnecting isolated phase bus, the two unit stations service transformers and associated power distribution system equipment. The main generator is connected to the main power transformer and unit station service transformers through the generator circuit breaker. Voltage from the main generator is stepped up by the main power transformer. Power is supplied to the 161KV transmission system as described in Section 8.2.

When the plant is operating, power will be supplied from the main generator to the Plant AC Distribution System through the generator circuit breaker and the unit station service transformers. Connections and ratings of major equipment for the Plant Power Supply are shown in Figure 8.3-1.

8.3.1.1.4 Plant AC Distribution System

The Plant AC Distribution System distributes power from the Plant Power Supply or the off-site AC power supplies to all facility auxiliaries that require AC power.

Power is distributed to auxiliary loads at four voltage levels:

A. 13.8KV and 4.16KV for large and medium size motors for motor driven auxiliaries.

B. 480 volt for motor driven auxiliaries, lighting, ventilation systems, and other miscellaneous loads.

C. 480/277 volt for lighting and sodium heat tracing loads.

D. 120/208 volt for control, alarm, instrument power, lighting, and other miscellaneous loads.

The Non-Class 1E Plant AC Distribution System is a two division distribution system, each division has its own equipment, components, raceway and cables, each distributing power to its own load group. Connections and ratings of major equipment of Plant AC Distribution System are shown in Figure 8.3-1.

Power to the 13.8KV loads is distributed from the 13.8KV switchgear buses which are supplied from the 13.8KV windings of the two unit station service transformers. The 13.8KV switchgear buses can also receive power from the 13.8KV windings of the two reserve station service transformers.

Power to the 4.16KV loads is distributed from the 4.16KV switchgear buses which are supplied from the 4.16KV windings of the two unit station service transformers. The 4.16KV buses can also receive power from the 4.16KV windings of the two reserve station service transformers.

The 13.8KV and 4.16KV switchgear buses distribute power to large and medium size motors and to 480 volt switchgear through transformers. Power to the 480 volt loads is distributed from 480 volt switchgear. The Class 1E switchgear are supplied through the Safety-Related AC Distribution System.

The 480V switchgear distribute power to larger 480 volt motors and to motor control centers.

Operation

When the unit is operating, the Plant AC Distribution System receives power from the main generator through the generator circuit breaker and the two 50% capacity (100% capacity for one safety-related Division's loads) unit station service transformers. In the event of a turbine or reactor trip, main generator fault or a fault in the isolated phase bus system between the main generator and generator circuit breaker, the generator circuit breaker will automatically open. The CRBRP Preferred

AC Power Supply will remain connected and provide uninterrupted power to the Plant AC Distribution System through the main power transformer and the unit station service transformers. An electrical fault downstream of the generator circuit breaker will cause tripping of the 161KV circuit breakers in the generating switchyard. This will result in the loss of the power supply from the unit station service transformers. Similarly, an event which trips the turbine or reactor concurrent with the loss of CRBRP Preferred Offsite Power from the generating switchyard will also result in the loss of the power supply from the unit station service transformers.

Fault sensing relays provided in the normal power supply and undervoltage sensors at each 13.8KV and 4.16KV switchgear bus will initiate the following upon detecting a fault or loss of bus voltage:

A. Trip the supply circuit breakers from the unit station service transformers.

B. Close the Reserve AC Power Supply circuit breakers from the two 50 percent capacity reserve station service transformers by means of a fast dead bus transfer scheme.

Primary and backup fault sensing relays have been provided in the normal power supply (Generator, generating switchyard, main step-up transformer and the Unit Station Service Transformers) and the reserve switchyard to perform the required protection of the electrical distribution system.

Each fault sensing relay provided in the normal power supply will actuate its respective lockout relay on sensing a fault in the normal power supply. The lockout relay will trip the normal generating switchyard (or CRBRP generator) power supply incoming circuit breakers on the medium voltage switchgear, including the 4.16KV Class 1E switchgear. The tripping of these circuit breakers will automatically initiate closing of the reserve offsite power supply (preferred power) incoming circuit breakers on the medium voltage switchgear using the early "b" contact of the normal power supply incoming circuit breakers.

The medium voltage switchgear busses are also provided with undervoltage sensors, which will also initiate tripping of the normal power supply circuit breaker and close the reserve power supply circuit breakers on sensing an undervoltage condition on the bus. In the case of Class 1E, 4.16KV medium voltage switchgear busses, the detection of an undervoltage condition will also result in an automatic start signal to the emergency diesel generator. However, if the automatic bus transfer to the reserve power supply restores the voltage to the medium voltage Class 1E switchgear busses, the circuit breaker connecting the diesel generator to the MV switchgear bus will remain open and the safety related loads will be powered from the reserve power supply.

A back-up breaker failure protection scheme is also provided in the event of the failure of the above protection scheme. On failure of the fault sensing relay(s), the fault sensing relay in the generating switchyard relaying scheme will actuate another lockout relay which will trip the 161KV circuit breakers in the generating switchyard, thereby isolating the medium voltage switchgear from the normal power supply and will initiate the closing of reserve offsite power supply incoming circuit breakers as described above. Additionally, in the event that a fast bus transfer is unsuccessful, a time delayed automatic bus transfer will be accomplished.

If this automatic closure of the reserve offsite power supply incoming circuit breaker(s) is not accomplished, the operator can manually close the reserve offsite power supply incoming circuit breaker(s).

The CRBRP design includes capability to test the transfer of power supplies among the plant power supply, the normal AC supply through the generating switchyard, the reserve AC supply through the reserve switchyard and the onsite standby diesel generator power supplies.

The sensors that detect the loss of power will be tested during plant operation or plant shutdown.

8.3.1.1.5 <u>120/208 Volt Vital (Uninterruptible) AC Power System</u>

The 120/208 volt Vital (Uninterruptible) AC Power system is a Class 1E system which is required to supply AC power to the Plant Protection System (PPS) controls, alarm and indication and other Class 1E loads for safe shutdown of the plant. The Plant Protection System (PPS), described in Chapter 7., generates signals to actuate reactor trip, and performs other supporting functions in the event of an emergency condition.

The system is divided into three separate and independent load groups (Divisions 1, 2 and 3), each receiving AC power from a separate inverter through a static transfer switch. Connections for the 120/208 volt Vital AC Power System are shown in Figure 8.3-2.

The normal source of power for the Vital AC Power Distribution buses are the inverters which are supplied from their associated division DC power supplies described in Section 8.3.2.

Each 120/208 volt Vital AC Power System Distribution bus can also recieve power from a Class 1E 480 volt motor control center which serves as a backup power source. Each of the distribution buses is connected to this motor control center through a static transfer switch and 480-120/208V AC regulating transformer. Failure of an inverter or its DC power source is sensed and the associated distribution bus is transferred automatically by the static transfer switch to the backup transformer supplied by the Class 1E 480 volt motor control center. The transfer is accomplished at high speed and does not degrade the performance of control and instrumentation loads.

Power for the vital AC loads is distributed from the three vital AC distribution buses through distribution panels.

Equipment in each of the three separate load groups of the 120/208 volt Vital AC Power System (inverters, static transfer switches, distribution buses, distribution panels and backup transformers) are installed in Seismic Category I structures. Cables for the PPS are installed in Class 1E raceways in accordance with PPS separation requirements described in Section 8.3.1.4. Cables and raceways of the Class 1E system are marked in a distinctive manner as described in 8.3.1.5.

The following types of protective devices are used:

A. <u>120/208 Volt Vital AC Inverter Supply</u>

1. The DC supply feeder to each inverter is protected by a power circuit breaker which operates without external control. The circuit breaker is equipped with a thermal magnetic trip device which provides fault protection for the circuit.

2. Each inverter is provided with an incoming circuit breaker equipped with a magnetic trip device to provide protection against internal fault. In addition, a current limiter is provided in the inverter output circuit to protect the inverter against overloads and to limit fault current contribution to external faults.

B. <u>480-120/208 Volt AC Supply Feeders</u>

Each backup 480-120/208 Volt AC supply feeder to the 120/208 volt Vital AC Power System is protected by a molded case circuit breaker which operates without external control. The circuit breaker is equipped with a thermal and magnetic trip device which provides short circuit and overload protection for the circuit.

C. <u>Vital AC Distribution Bus</u>

Each feeder from a Vital AC distribution bus to a distribution panel is protected by a fuse. In addition, each inverter is equipped with an undervoltage relay to annunciate a low voltage condition on the Vital AC bus.

D. <u>Feeders from Distribution Panels</u>

The outgoing feeders from each distribution panel are protected by fuses.

The protection system of the 120/208 volt Vital AC Power System is analyzed to ensure that the various adjustable devices are selective in their operation.

Testing and Inspection

The equipment is inspected and tested at the vendor's facilities. The system is also inspected during installation. When the installation is complete pre-operational equipment tests and inspections are performed to demonstrate that:

A. Components are correct and properly mounted.

B. Connections are correct and the circuits are continuous.

C. Components are operational.

D. Instruments and protective devices are properly calibrated and adjusted.

The initial system tests will also demonstrate that while supplied by the DC Power System or the backup 480 volt motor control center, the 120/208 volt Vital AC Power System can supply power to the design load as required.

Periodic tests will be performed to detect any deterioration of the equipment and to demonstrate the capability of equipment which is normally energized.

A manual bypass switch is provided to facilitate maintenance and testing of each static transfer switch and the inverter on a periodic basis. By using the bypass switch, the static transfer switch, and the inverter is isolated and taken out of service without power interruption.

Only one inverter and static transfer switch will be tested at a time to ensure that any postulated failure of the components during test will result in the loss of only one vital bus. In the unlikely event that an accident signal is generated during the test and if vital AC power from the bus under test is lost to the PPS, vital AC power from either of the remaining two buses enables the PPS to perform its function. Loss of two vital buses would cause the PPS to trip the Plant.

8.3.1.1.6 Remote Multiplexing System (RMS)

- The RMS is a Non-Class 1E system.

- The RMS is used as a wire-replacer. Signals enter the RMS, are converted to a digital data stream and replicated (under control of a Central Control Unit) at a predetermined multiplexer (field or Control Room) for presentation to logic panels for control purposes or at the main control panel (or local panel) for display or feedback information.

- Field Multiplexers (FMs) are located throughout the plant for interfacing field signals.

- Control Room Multiplexers (CRMs) are located in the Main Control Room for interfacing with Control Room signals.

- RMS functions are performed in two types of cabinets; an electronics cabinet (which contains signal conditions, modems, common logic and power supplies), and termination cabinets (which contain the field wiring terminal blocks).

- Two separate and independent RMS divisions are provided, (Non-Class 1E Division A and Non-Class 1E Division B). Each division uses redundant data transmission cables carrying time multiplexed signals between plant instrumentation alarm and control signals.

8.3.1.1.7 Motors

The following criteria will be used in sizing and selection of motors for safety-related loads:

a) Motor Size

- Motor size will be greater than the maximum horsepower required by the driven load under all operating conditions.

b) Starting Voltage

- Motors will be designed to start and accelerate their loads with 70 percent terminal voltage. This starting voltage will be consistent with the diesel generator limits.

c) Insulation

- Proper winding insulation will be selected for the specific environmental condition where the motor will be used.

d) Temperature Monitoring

- Large motors (250hp or larger) will be provided with thermocouples at each bearing. Motors 1000hp or larger will be provided with RTD's in the stator winding. All motors (irrespective of size) driving ASME code pumps will have bearing temperature determination.

8.3.1.2 Analysis

The plant power systems are in compliance with CRBRP General Design Criteria 15 and 16 as described in Section 3.1. On-site AC Power Systems are discussed below in regard to conformance with NRC Regulatory Guides 1.6, 1.9, 1.22, 1.29, 1.30, 1.32, 1.40, 1.41, 1.47, 1.53, 1.63, 1.68, 1.73, 1.75, 1.89, 1.93, 1.100, 1.106, 1.108, 1.118, 1.131 and IEEE Standards 308-1974, 317-1976, 323-1974, 334-1971, 383-1974, 384-1974 and 387-1977.

8.3.1.2.1 NRC Regulatory Guide 1.6, Rev. 0 (3/71)

The Class 1E safety-related loads are physically and electrically separated into three functionally redundant shutdown load groups (Divisions 1,2 and 3) such that loss of any two groups, including a single failure condition, will not prevent safe shutdown of the plant. Only one load group is required to shut down the plant safely.

Each AC load group will have connections to the CRBRP Preferred Power Supply, Reserve Power Supply and a Standby On-site AC Power Source. The Standby On-site AC Power source will have no automatic connection to any other redundant load group.

When operating from the Standby On-site sources, redundant load groups and the redundant Standby On-site sources will be independent of each other as follows:

a. The Standby On-site source of one Class 1E load group will not be automatically paralled with the Standby On-site source of another Class 1E load group under normal or emergency conditions.

b. No provisions exist for automatically connecting one Class 1E load group to another Class 1E load group.

c. No provisions exist for automatically transferring loads between redundant Class 1E power sources.

d. Manually connecting redundant load groups together will require at least one interlock to prevent an operator error that would parallel such Standby On-site power sources.

Each Diesel Generator unit consists of one diesel engine, one generator and required accessories.

The Standby On-Site Power Supply network has a provision to manually cross-connect the 4.16KV buses of the Division 1 and 2 power supplies in case of an extreme emergency. This connection will be put into service through strict administrative controls and must satisfy the following prerequisites:

a) There shall be a total loss of off-site power.

b) One of the two redundant diesel generators failed to start and it is determined to be inoperable.

c) Critical safety-related loads associated with the operative diesel generator have failed and become unavailable.

If the above prerequisites are met, loads of either redundant Division 1 or 2 can be connected to the diesel generator of the other division for safe shutdown of the plant and to maintain the plant in a safe shutdown condition.

Key and electrical interlocks and administrative controls will be provided to ensure:

- The two standby power supplies will never be paralled.

- The breakers for the cross connection are physically not in place and are located in locked compartments separate from the operative switchgear compartments.

- There will be an alarm in the Control Room when either or both tie breakers are in the operating switchgear compartment(s).

- The connection will be disabled if any one of the off-site power supplies or the disabled diesel generator becomes available.

See Figure 8.3-3 for details.

8.3.1.2.2 NRC Regulatory Guide 1.9, Rev. 2 (12/79)

Diesel Generator units are designed, qualified and tested in accordance with IEEE Std. 387-1977 supplemented by Regulatory Guide 1.9 recommendations as discussed below:

Each Diesel Generator unit has a continuous load rating (as defined in Section 3.7.1 of IEEE Std. 387) greater than the sum of the conservatively estimated loads needed to be powered by that unit at any one time. For mechanical equipment such as pumps, the electric motor drive ratings are calculated using conservative estimates of performance characteristics, e.g., pump runout conditions and motor efficiencies of 90 percent or less and power factors of 85 percent or less.

The predicted loads will not exceed the short-time rating (as defined in Section 3.7.2 of IEEE Std. 387) of the Diesel Generator unit.

In conjunction with Section 5.1.2 of IEEE Std. 387, each Diesel Generator unit is capable of starting and accelerating to rated speed, in the required sequence, all the needed Class 1E loads as indicated in Tables 8.3-1A, 8.3-1B, and 8.3-1C. The Diesel Generator unit design is such that at no time during the loading sequence will the frequency and voltage decrease to less than 95 percent of nominal and 75 percent of nominal, respectively. Voltage will be restored to within 10 percent of nominal and frequency will be restored to within 2 percent of nominal in less than 4 seconds for each load sequence time interval. During recovery from transients caused by step load increases or resulting from the disconnection of the largest single load (1050 HP) or of the complete loss of the continuous rated load, the speed of the Diesel Generator unit will not exceed the nominal speed plus 75 percent of the difference between nominal speed and the overspeed trip set point or 115 percent of nominal, whichever is lower.

In Section 5.4 of IEEE Std. 387, the qualification testing requirements of IEEE Std. 323 are used subject to the regulatory position of Regulatory Guide 1.89. (See Section 3.10 for seismic design criteria and 3.11 for environmental qualification criteria).

In conjunction with Section 5.5 of IEEE Std. 387, each Diesel Generator unit is designed to be testable during operation of the plant as well as while the plant is shut down. The design includes provisions so that the testing of the units will simulate the parameters of operation, outlined in Regulatory Guide 1.108, that would be expected if actual demand were to be placed on the system.

Testability is considered in the selection and location of instrumentation sensors and critical components (e.g., governor, starting system components) and the overall design includes status indication and alarm features. Instrumentation sensors are readily accessible and, where practicable, designed so that their inspection and calibration can be verified in place.

In conjunction with Section 5.6.2.2 of IEEE Std. 387, each Diesel Generator unit is protected against abnormal conditions as described in Section 8.3.1.1.2. After automatic start under a plant emergency condition, all protection circuits will be bypassed except the generator differential overcurrent and engine overspeed protection circuits.

In conjunction with Section 5.6.3.1 of IEEE Std. 387, in order to facilitate trouble diagnosis, each Diesel Generator unit surveillance system indicates which of the Diesel Generator protective trips is activated first.

In Section 6.3 of IEEE Std. 387, the requirements of IEEE 344-1975 for seismic analysis or seismic testing by equipment manufacturers is used subject to the regulatory position of Regulatory Guide 1.100.

The option indicated by "may" in Section 6.3.2(5)(c) of IEEE Std. 387 is treated as a requirement.

Section 6.5 and Section 6.6 of IEEE Std. 387 are supplemented by Regulatory Guide 1.108 as described in Section 8.3.1.1.1.

In Section 6.3.1 of IEEE Std. 387, the order of sequence of load tests described in parts (1) and (2) is as follows: Load equal to the continuous rating will be applied for the time required to reach engine temperature equilibrium, at which time, the rated short-time load is applied for a period of 2 hours. Immediately following the 2-hour short-time load test, load equal to the continuous rating will be applied for 22 hours.

8.3.1.2.3 NRC Regulatory Guide 1.22, Rev. 0 (2/72)

Design of the plant electrical power system will permit Class 1E components, which are part of Plant Protection Systems, actuation devices, auxiliary supporting systems, or actuated equipment, to be periodically tested to verify satisfactory protective device/system actuation. Components will be designed to permit testing of the actuation devices both during reactor operation and routinely, when the reactor is shutdown. Each of the Regulatory Guide recommendations for testing, testing methods, system response to bona-fide accident signal(s), and reliability of equipment not tested during reactor operation will be implemented.

8.3.1.2.4 NRC Regulatory Guide 1.29, Rev. 3 (9/78)

The Class 1E Electric Systems, including the auxiliary systems for the Onsite Electric Power Supplies, that provide the Class 1E electric power needed for functioning of nuclear safety related equipment are designated as Seismic Category I.

All electric devices and circuitry involved in generating signals that initiate protective action are designed as Class 1E.

All Class 1E equipment including the diesel generators, 4.16KV Switchgear, Unit Substations, Motor Control Centers, Control Room Panels, etc. are located inside Seismic Category I buildings and are designed as Seismic Category I.

All non-safety-related equipment located in Seismic Category I buildings are designed to maintain structural integrity under a Seismic event and will not become missiles.

Those portions of structures, systems or components whose continued function is not required but whose failure could reduce the functioning of any nuclear safety related equipment to an unacceptable safety level will be designed and constructed so that the SSE would not cause such a failure.

Seismic Category I design requirements will extend to the first seismic restraint beyond the defined boundaries. Those portions of structures, systems, or components which form interfaces between Seismic Category I and non-Seismic Category I features will be designed to Seismic Category I requirements.

For seismic design classifications, refer to Section 3.2.1.

8.3.1.2.5 NRC Regulatory Guide 1.30, Rev. 0 (8/72)

The Quality Assurance requirements for the installation, inspection and testing of instrumentation and electrical equipment during the plant construction, are those included in ANSI N45.2.4 supplemented by Regulatory Guide 1.30 as follows:

ANSI N45.2.4 will be used in conjunction with ANSI N45.2-1977.

ANSI N45.2.4 requirements will be considered applicable for the installation, inspection and testing of instrumentation and electric equipment during the plant operation.

8.3.1.2.6 NRC Regulatory Guide 1.32, Rev. 2 (2/77)

The electrical separation and independence of redundant (Divisions 1 and 2) and Division 3 Standby AC Power Supplies conform to IEEE Standard 308-1974 supplemented by Regulatory Guide 1.32 as follows:

Electrical independence between redundant Standby AC Power Supplies will be in accordance with Regulatory Guide 1.6 as described in Section 8.3.1.2.1.

Physical independence between redundant Standby AC Power Supplies will be in accordance with IEEE Standard 384-1974 supplemented by Regulatory Guide 1.75 as described in Section 8.3.1.2.14.

The selection of the diesel generator capacities will be made in accordance with Regulatory Guide 1.9 as described in Section 8.3.1.2.2.

8.3.1.2.7 NRC Regulatory Guide 1.40, Rev. 0 (3/73)

Class 1E motors in Heat Transport System (HTS) are the only Class 1E motors which are located inside the containment. Where test programs are required in the Liquid Metal Fast Breeder Reactor (LMFBR) design to verify the adequacy of specific design features in lieu of other verifying processes, these test programs will include qualification tests on prototype continuous-duty Class 1E motors installed inside the containment.

The procedure for conducting these qualification tests are those specified by IEEE Std. 334-1974.

When these qualification tests are used, they will be in agreement with Regulatory Guide 1.40, as follows:

1. A Liquid Metal Fast Breeder Reactor (LMFBR) auxiliary equipment that will be part of the installed motor assembly will be qualified in accordance with IEEE Std. 334-1974 to the extent applicable.

2. The qualification tests will simulate as closely as practicable those design basis events which:

   a. Require the motor to either drive equipment which mitigates the consequences of the event or provide auxiliary support to such equipment, and;

   b. Affect operation of the motor's auxiliary equipment.

8.3.1.2.8 NRC Regulatory Guide 1.41, Rev. 0 (3/73)

Preoperational tests will be performed to verify the independence of the redundant Standby AC Power Sources and of the Division 3 Standby AC Power Sources and between the redundant load groups described in Section 8.3.1.1.

The tests will be performed as follows:

a. The power sources to the Class 1E 4.16KV AC, and 125 Volt DC power distribution buses of the redundant load group not under tests are disconnected to verify that the operation of equipment under test is not affected.

b. The load group of the Safety-Related AC Power Distribution System under test is isolated from the Plant Power Supply and the Offsite AC Power Supplies, simulating an emergency actuation signal which starts the diesel generator under test.

c. The emergency actuation signal causes sequential starting of Class 1E loads as described in Section 8.3.1.1.

The tests will be of sufficient duration to attain steady-state operation of the Standby AC Power System as well as steady-state operation of the loads under test.

During the test, the Class 1E AC and DC power distribution buses not under test will be continuously monitored to verify absence of voltage at these buses.

The tests will be repeated for the redundant load group and source.

Functional performance of loads will be verified as follows:

During preoperational testing, functional performance of the auxiliaries will be verified by tests.

8.3.1.2.9 NRC Regulatory Guide 1.47, Rev. 0 (5/73)

Automatic indication of inoperability or bypassed conditions for standby AC power supplies, Class 1E DC power supplies and safety related components and systems will be provided in the Control Room.

8.3.1.2.10 Regulatory Guide 1.53, Rev. 0 (6/73)

The Class 1E electrical power systems will be designed to meet single failure criteria. The design will be in accordance with guidance provided in IEEE Standard 379-1972 as supplemented by Regulatory Guide 1.53.

The following describes compliance with Regulatory Guide 1.53 requirements:

a. Section 5.2 of IEEE Std. 379-1972 is adopted as follows:

   The detectability of a single failure is predicated on the assumption that the test results, in the presence of a failure are different from the results that would be obtained if no failure were present. Thus, inconclusive testing procedure such as "continuity checks" of relay circuit coils in lieu of relay operation is not considered as an adequate basis to classify as detectable all potential failures which could negate the functional capability of the tested device.

b. Section 6.2 of IEEE Std. 379-1972 is adopted as follows:

   Where a single mode switch applies signals to redundant channels, it is considered that the single-failure criteria is not satisfied if either (a) individual switch sections supply signals to redundant channels, or (b) redundant circuits controlled by the switch are separated by less than six inches without suitable barriers.

c. Sections 6.3 and 6.4 of IEEE Std. 379-1972 are interpreted as not permitting separate failure mode analyses for the protection system logic and the actuator system. The collective protection system logic-actuator system as applicable for the Class 1E electrical power systems is analyzed for single-failure modes which, though not negating the functional capability of either portion, act to disable the complete protective function.

8.3.1.2.11 NRC Regulatory Guide 1.63, Rev. 2 (7/78)

The electrical penetration assemblies in the containment vessel will be designed, constructed, qualified, installed and tested in accordance with IEEE Std. 317-1976, supplemented by Regulatory guide 1.63 positions as discussed herein.

The conductors and the electrical penetration assembly will be designed to withstand the maximum short-circuit currents versus time conditions that could occur given single random failures of circuit overload protective devices. The duration of rated short circuit current is based on the operating time of the secondary (backup) protective device or apparatus. The electrical penetration assemblies will be designed to maintain their mechanical and electrical integrity in accordance with IEEE Std. 317-1976, IEEE Std. 279-1971 and Regulatory Guide 1.63.

The dielectric-strength test qualification for medium voltage power conductors is in accordance with IEEE Std. 317-1976 supplemented by the impulse voltage test as described in Regulatory Guide 1.63.

Regulatory positions, C1, C2, C3, and C4 place additional restrictions on maximum short-circuit current, x/r ratios, maximum short-circuit current duration and impulse voltage qualification testing on the electrical penetration assemblies in addition to the requirements of IEEE Std. 317-1976. The project will comply fully with the requirements as set forth in IEEE Std. 317-1976 and as modified by Regulatory Guide 1.63.

8.3.1.2.12 NRC Regulatory guide 1.68, Rev. 2 (8/78)

Written procedures for preoperational and startup testing for the Plant AC Power Distribution System, Class 1E AC Power Distribution System, Standby AC Power Supplies and DC System will be developed. Format and content of these procedures will conform to the guidance given in Regulatory Guide 1.68. For test program description, see Chapter 14.

8.3.1.2.13 NRC Regulatory Guide 1.73, Rev. 0 (1/74)

All Class 1E electric valve operator assemblies, for installation inside the containment vessel, will be designed, constructed, qualified, installed and tested in accordance with IEEE Std. 382-1972 supplemented by Regulatory guide 1.73 requirements.

Each electric valve operator assembly will be designed and constructed to withstand the worst local environmental requirements (during normal or accident conditions) such as temperature, humidity, radiation, and sodium aerosol condition.

8.3.1.2.14 NRC Regulatory Guide 1.75, Rev. 2 (9/78)

The electrical equipment and circuits comprising or associated with the Class 1E power system, Class 1E protection systems and Class 1E equipment will be designed, qualified and tested in accordance with IEEE Std. 384-1975, supplemented by Regulatory Guide 1.75, "Physical Independence of Electric Systems," positions as discussed herein.

The system will be designed so that the redundant equipment and circuits are separated in accordance with the criteria set forth in paragraph 8.3.1.4.

The AC loads which are not Class 1E but are required for plant availability will not be connected to the redundant Class 1E, Divisions 1 and 2 4.16KV buses, but will be connected to Division 3 switchgear through an isolation device, which is designed as follows:

a. The isolation system will consist of a 4.16KV cirult breaker, a 4.16KV/480V high impedance transformer and a 480V circuit breaker as shown in Figure 8.3-3. The isolation system will be qualified as Class 1E up to the load terminals of 480V circuit breaker.

b. The impedance of the isolation system will be high enough so that for the worst possible fault (three phase bolted fault) on the Non-Class 1E 480V bus, the following conditions will be met:

   (1) The pick-up value of the overcurrent relays protecting the Class 1E 4.16KV main supply circuit breaker will exceed the maximum current (combined maximum load current and maximum fault current contribution) flowing through the supply circuit breaker by a 2:1 margin.

   (2) When the 4.16 KV Class 1E bus is being supplied from offsite power, the voltage at the bus will not drop below 80% of nominal. When the Class 1E bus is being supplied from on-site (standby) power supply, the voltage at the bus will not drop below 75% of nominal. The voltage levels of 80 and 75 percent of nominal are chosen to be the same as the allowable minimum voltage levels during the sequential loading of the 4.16kV Class 1E bus or during starting of the largest motor after the bus has been fully loaded.

c. The isolation system 480 volt and 4.16KV circuit breakers will perform redundant isolation functions. They will be stored energy devices and will be physically separated.

d. Diverse means (electro-mechanical and solid state) will be used for fault sensing and tripping of the isolation system.

e. The isolation system will be able to accept any single component failure concurrent with the worst fault on the Non-Class 1E 480V bus without unacceptable consequences. (This does not include short circuits on the 4.16KV portion of the isolation system since this is considered an extension of the Class 1E bus).

f. Protective devices have been provided in the design to clear any fault on the Non-Class 1E system such as phase to ground, phase to phase and three phase faults within a reasonable time such that there is no degradation to the Class 1E system.

   1) A phase to ground fault (which is the most likely mode of failure) on a Non-Class 1E circuit will have no effect on the Class 1E system since the isolation system includes a 4.16kV/480V delta-wye connected transformer with the high resistance grounded neutral. The neutral is grounded through a 55.4 ohm resistor which will limit the line to ground current to approximately 5 amperes. The Class 1E 480 volt and 4.16kV circuit breakers will be tripped to clear a ground fault in the case that the affected Non-Class 1E feeder circuit breaker fails to trip.

   2) Any phase to phase or three phase fault on the Non-Class 1E circuits will be isolated by instantaneous operation of the affected branch feeder circuit breaker. Back-up protection is provided by fast operation of the Class 1E 480 Volt circuit breaker (0.2-0.3 sec clearing time) or by the 4.16kV circuit breaker (0.6-0.7 sec clearing time). In addition undervoltage sensors are provided at the input terminals of the Class 1E 480 Volt circuit breaker. These undervoltage sensors will initiate tripping of the Class 1E 480 Volt and 4.16kV circuit breakers within five (5) seconds upon sensing the undervoltage caused by loss of power or failure of the circuit breakers to clear a fault.

   3) After the fault has been cleared the voltage at the 4.16kV bus will be restored to a minimum of 90 percent of nominal within (2) seconds, which will allow all connected loads to operate continuously.

g. The high impedance transformer used as an isolation device will be subjected to a short-circuit withstand test as part of the shop testing program at the manufacturer's facility. After the transformer has been energized a three phase fault will be applied at the secondary windings for the maximum duration of the fault. The purpose of this test is to demonstrate the mechanical and thermal capability of the transformer to withstand short-circuit stresses which the transformer could experience and to verify the transformer current limiting capability.

The system is designed to keep the number of associated circuits to a bare minimum. Associated circuits will comply with one of the following paragraphs of IEEE Std. 384-1974: 4.5(1), 4.5(2) or 4.5(3), and positions C.4 and C.6 of Regulatory Guides 1.75-1975.

The cable installation design prohibits the use of cable splicing inside the cable tray or conduit raceway system.

The physical identification of Class 1E equipment, cables and raceway systems are described in Section 8.3.1.5.

The design provides two separate cable spreading rooms, one above the Control Room and one below it. The design does not permit location of any high energy equipment in the cable spreading rooms as required by IEEE Std. 384-1974. The criteria for routing of circuits in the cable spreading rooms is given in Section 8.3.1.4 to assure physical separation.

The Divisions 1, 2 and 3 Class 1E Standby Diesel Generator units are described in Section 8.3.1.1.1. The diesel generator units and associated auxiliaries and control equipment are located in separate Seismic Category I structures having independent ventilating systems. The circuits related to redundant Standby Diesel Generators are routed in accordance with the criteria specified in Section 8.3.1.4, to assure physical separation.

The Non-Class 1E and Class 1E DC batteries and related uninterruptible power supply (UPS) equipment are described in Section 8.3.2. DC battery and associated UPS equipment of each safety division is separated from equipment of the other safety division by reinforced concrete walls. The Class 1E batteries and UPS equipment are located in Seismic Category I structures. The physical separation of circuits related to each separate division of batteries and UPS system is in accordance with the criteria described in Section 8.3.1.4.

Regulatory Positions C.2, C.3, C.4, C.7, C.8 and C.12 place additional restrictions on adequacy of separation of redundant circuits in addition to Section 4 and Section 5 of IEEE Std. 384-1974. The project will comply fully with the requirements as set forth in IEEE Std. 384-1974, as modified by Regulatory Guide 1.75.

8.3.1.2.15 NRC Regulatory Guide 1.89, Rev. 0 (7/74)

For a description of environmental qualification of Class 1E equipment, see Section 3.11.

8.3.1.2.16 NRC Regulatory Guide 1.93, Rev. 0 (12/74)

CRBRP design consists of two off-site power sources for the CRBRP Preferred Power Supply and two physically independent power sources for the Reserve Power Supply. Each one of these power sources is continuously available for the plant distribution system for safe shutdown. Further, On-site AC Power Supplies are available continuously from three physically separate and electrically independent diesel generators. Three physically separate and electrically independent On-site DC Systems are available to feed Class 1E Divisions 1, 2 and 3 loads, respectively. Divisions 1 and 2 AC loads are redundant to each other. This design satisfies the CRBRP General Design Criterion 15 (GDC 17).

The limiting conditions of operations (LCO) are met by the CRBRP design in accordance with Regulatory Guide 1.93. In the event of degradation of the electric power sources below the LCO conditions, restricted plant operations will be followed in accordance with the recommendations of Regulatory Guide 1.93.

8.3.1.2.17 NRC Regulatory Guide 1.100, Rev. 1 (8/77)

Seismic qualification of Class 1E equipment will be in accordance with IEEE Std. 344-1975 supplemented by Regulatory Guide 1.100, as described in Section 3.10.1.

8.3.1.2.18 NRC Regulatory Guide 1.106, Rev. 1 (3/77)

Motor thermal overload protection devices for safety related motor operated valves, which are in force during testing and normal operation, will be bypassed under plant emergency conditions. The bypass circuitry will conform to IEEE Std. 279-1971.

8.3.1.2.19 NRC Regulatory Guide 1.108, Rev. 1 (8/77)

The preoperational tests and periodic testing, after the diesel generator units are placed in service, will be performed in accordance with Regulatory Guide 1.108, as described in Section 8.3.1.1.1.

8.3.1.2.20 NRC Regulatory Guide 1.118, Rev. 1 (6/78)

CRBRP Design Criterion 16 (GDC 18) has been established to satisfy the requirements of IEEE Std. 279-1971 and 338-1977 and Regulatory Guide 1.118. This requires the design to provide for appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections and switchboards, to assess the continuity of the systems and the condition of their components to check the operability and functional performance of the system components such as on-site power sources, relays, switches, buses and the system as a whole under conditions as close to design as practical.

CRBRP design will comply with the above criteria and as a result, it will comply with IEEE Std. 279-1971, 338-1977 and Regulatory Guide 1.118. For details of compliance refer to Sections 8.3.1.1, 8.3.1.2, 8.3.1.2.1 and 8.3.1.3.

8.3.1.2.21 NRC Regulatory Guide 1.131, Rev. 0 (8/77)

The electric cables, field splices and connections will be designed, qualified and tested in accordance with IEEE Std. 383-1974, supplemented by Regulatory Guide 1.131, positions as discussed herein.

The medium voltage cables, low voltage power and control cables, and instrumentation cables are specified to be type tested and qualified to the requirements of IEEE Std. 383-1974 and IEEE Std. 323-1974 supplemented by Regulatory Guide 1.131 and to the design basis events described in Table 8.3-3. The field splicing of cables inside the cable tray or conduit raceway system is prohibited in accordance with the requirements of IEEE Std. 384-1974 supplemented by Regulatory Guide 1.75. The environmental conditions for all cables include the maximum sodium aerosol concentration, along with the values of pressure, temperature, radiation, chemical concentrations, humidity and time, and are specified as applicable to the design of the power plant.

8.3.1.2.22 IEEE Standard 308 - 1974

Class 1E AC and DC Power Supplies and distribution systems will be designed to conform to the requirements of Class 1E electrical systems as discussed below.

All Class 1E electrical equipment will be specified and qualified for the environmental conditions such that no design basis event will cause loss of electric power to any loads related to safety, surveillance or protection, thereby maintaining the safety of the plant at all times.

Loss of electric power to any single Class 1E equipment or to any Class 1E division will not cause damage to the fuel or to the reactor coolant system.

The Class 1E system is capable of performing its function when subjected to the effects of a design basis event at its location. (See Table 8.3-3) No significant radiation hazard to Class 1E loads has been identified for either normal or emergency conditions.

Class 1E loads are designed to perform their intended functions adequately for the variation of voltage or frequency in the Class 1E electric system.

Circuit breaker controls and indicating lights are provided both in the Control Room and locally for the incoming line and the diesel generator supply circuit breakers of the Class 1E 4.16 KV switchgear buses. Also, controls to start and operate the diesel generators are furnished locally and in the Control Room.

Each type of equipment used in the Standby AC Power Supply is qualified to perform its design function under normal and emergency environmental conditions as described in Section 3.11.

Supplementary design criteria of IEEE Std. 308 are addressed in the applicable section describing specific Class 1E equipment.

The tests and inspections performed for the equipment are described in Section 8.3.1.1.

8.3.1.2.23 IEEE Standard 317 - 1976

The electrical penetration assemblies which pass cables through the containment vessel are designed and tested in conformance to IEEE Std. 317 - 1976.

8.3.1.2.24 IEEE Standard 323 - 1974

The Class 1E AC and DC Power Supply and distribution systems will conform to IEEE Std. 323 - 1974 which includes requirements for an equipment qualification method and documentation. The qualification method establishes that each type of equipment is suitable for its application. The documentation will include the application requirements, the equipment specifications, and the data from the qualification methods used.

8.3.1.2.25 IEEE Standard 334 - 1971

Where qualification tests are required for prototype continuous-duty Class 1E Motors installed inside the containment, the procedures specified in IEEE Std. 334 - 1971 and NRC Regulatory Guide 1.40 will be followed.

8.3.1.2.26 IEEE Standard 383 - 1974

Electrical Cables, field splices, and connections will be in conformance with IEEE Std. 383 - 1974 supplemented by Regulatory Guide 1.131, Rev. 0. Type test will demonstrate satisfactory performance of the cable (cable assembly) by electrical and physical measurements appropriate to the type of cable during and after CRBRP's DBE. Environmental parameters specified in IEEE Std. 323 do not represent acceptable limits for CRBRP. The actual environmental parameters specific to CRBRP will be utilized in cable specifications.

8.3.1.2.27 IEEE Standard 384-1974

The separation criteria of IEEE Std. 384 - 1974 will be applied to insure that the physical independence of redundant Class 1E electrical systems is maintained. (See Section 8.3.1.4.)

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8.3.1.2.28 IEEE Standard 387 - 1977

The Standby AC Power Supply conforms to IEEE Standard 387-1977 which includes requirements for capability rating, independence, redundancy, testing, analyses, quality assurance, and identification.

8.3.1.3 Conformance with Appropriate Quality Assurance Standards

Assurance that equipment and workmanship quality is maintained throughout the construction process is provided by conformance to IEEE Standard 336 - 1971, "Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment during the Construction of Nuclear Power Generating Stations." The methods used to accomplish conformance are described by construction procedures and instructions and in Chapter 17.0 of this PSAR.

8.3.1.4 Independence of Class 1E Systems

The following criteria is used to preserve the independence of Class 1E systems. CRBRP separation criteria includes requirements for separating each of Safety-Related Divisions 1, 2 and 3 from each of the remaining Safety-Related Divisions in accordance with IEEE Standard 384-1974 and Regulatory Guide 1.75. Items being separated on these bases include Class 1E and associated cabling, circuits, equipment, raceways, and systems.

A. General Separation of Cables and Circuits by Voltage Class

A raceway contains cables of only one class. Classes are based on the nominal utilization voltage of the cable and/or vulnerability to spurious signals.

Voltage Classes are:

15KV Class - 13.8KV AC nominal power
5KV Class - 4.16KV AC nominal power
600V Class - 480-277 volt AC and 250 volt DC nominal power
Control - 120V/208V AC, 125V DC, 120V AC nominal power and control
Low level instrumentation including digital and analog signals

When cable trays are arranged in a vertical stack, the preferable arrangement is in order of voltage class, with the highest voltage at the top.

B. Cable Derating

Ampacity rating and group derating factors of cables are in accordance with the Insulated Power Cable Engineers Association Publications IPCEA-P54-440 and IPCEA-P46-426. Cables are selected to minimize deterioration due to temperature, humidity, and radiation during the design life of the plant. Environmental type tests will be performed during the design life of the plant. Environmental type tests will be performed on cables and terminations that are required to function in a hostile environment. The tests will include radiation exposure, heat aging, and electrical measurements to assure that the cable will function in the design environment for the required time. Cable derating as a result of fire stops/seals are included in the design.

C. Raceway Fill

Cable tray fill will be limited such that the summation of the cross-sectional areas of cables in a tray section will in general be not more than 40% of the usable cross-sectional srea of that tray section.

Conduits will be sized for a maximum percent fill of the inside area of the conduit in accordance with NFPA 70 "National Electrical Code" Art. 346.

D. Sealing Raceway Blockouts and Wall and Floor Penetrations

Fire stops will be installed for cable trays wherever the cables pass through fire walls and floors other than the Reactor Containment vessel. Cable and cable tray penetrations of fire baarriers are sealed to provide protection at least equivalent to that required of the fire barrier. Penetrations are qualified to meet the requirements of ASTM E-119, and IEEE Std. 634-1978. The actual fire ratings of stops and penetrations are determined by fire hazards analysis.

Fire stops, fire barriers, and air seals will be constructed of mastic type materials or elastomer modular construction materials qualified in accordance with IEEE Std. 623 and ASTM E-119. Fire stop/seal material will be compatible with insulation and conductor materials and will be shock, vibration, seismic, and radiation resistant in accordance with the area(s) penetrated.

E. Physical Separation of Class 1E Cables and Circuits

The separation design description for raceways, Class 1E circuitry and associated cabling and circuits given below incorporates the requirements of IEEE Std. 384-1974, Regulatory Guide 1.6 and NRC Regulatory Guide 1.75.

Load groups, cables and raceways of a safety-related system will be separated from load groups, cables, or raceways of other safety-related groups in accordance with the separation criteria described herein. This separation criteria will preclude a single failure within the safety-related system from preventing proper protective action at the system level when required. Raceways and cables will be classified by separation groups, namely Class 1E Division 1, Class 1E Division 2, Class 1E Division 3, and Plant Protection System. For the purpose of physical separation criteria Class 1E Division 1, 2 and 3 are treated as redundant divisions.

Cables designated in each division will be run in raceways separated from cables designated in other divisions and from Non-Class 1E cables. Associated cables and circuits will be separated as if they were Class 1E pursuant to the Class 1E division associated with these cables and circuits.

Each division of Class 1E equipment of Divisions 1, 2 and 3 are located in separate rooms which are separated by a minimum of 3 hours rated fire barriers.

F. Separation Criteria between Class 1E and Non-Class 1E and Associated Circuits

1. Separation of Cables and Circuits Within Safety-Related Panels

Within safety-related control boards and panels the separation between wiring of redundant divisions or of non-Class 1E wiring from Class 1E and associated Class 1E wiring will comply with at least one of the following:

i) A minimum separation distance of 6 inches vertical and horizontal will be maintained where the control board or panel materials are flame retardant.

ii) An analysis will be performed to determine the minimum separation distance. The analysis will be based on tests performed to determine the flame retardant characteristics of the wiring, wiring materials, equipment and other material internal to the control board or panel.

iii) Barriers will be installed in the event the above separation distances are not maintained.

Within safety-related control boards and panels, non-Class 1E wiring is not harnessed with Class 1E or associated Class 1E wiring. Associated Class 1E wiring is harnessed with Class 1E wiring of the same division.

2. Separation of Class 1E and Non-Class 1E Cables and Circuits

All Class 1E and non-Class 1E cables will be routed in raceways consisting of cable trays and conduits. Each raceway will contain cable(s) of one Class 1E safety division or a non-class 1E system only. For the purpose of cables, circuits, and raceways, the plant areas have been divided into six (6) separation zones as described in Section 8.3.1.4.

3. Separation Between Cable Trays and Conduits of Another Division

A Class 1E conduit will contain circuits of only one load division. In non-hazard zones exposed Class 1E conduits are separated from trays of another division as described in Section 8.3.1.4E. In all other separation zones the Class 1E conduits are not routed with trays of another division.

4. Criteria for the Separation of Third Division

The Class 1E electrical distribution system consists of three Class 1E divisions (Division 1, 2 and 3). Each of these divisions is designed to have physical and electrical independence from the other two divisions as described elsewhere in this section. Each of these divisions is provided with an on-site (standby) diesel generator and has the capability to shutdown the plant safely. However, from the consideration of connected loads, Class 1E Divisions 1 and 2 provide power to redundant load groups and as such are described as redundant divisions in Sections 8.1.2 and 8.3.1.1. Class 1E Division 3 provides power to heat removal system of Loop 3 and to important non-Class 1E loads through an isolation subsystem. Class 1E Division 3 as stated above has the capability to shutdown the plant safely; however, since all the loads powered from this division are not similar or identical to those powered by Division 1 or 2, this division has not been identified as redundant to Division 1 or 2.

5. Criteria for Separation Between Associated Cables and Circuits and Non-Class 1E Cables and Circuits

   The associated circuits as defined in paragraph 4.5 of IEEE Standard 384-1974 will be considered as Class 1E cables and circuits for the purpose of their routing and installation. The separation criteria between associated cables and non-Class 1E cables is the same as described in item 2 above for the separation between Class 1E and non-Class 1E cables. These cables, once identified as associated with a safety division, will be routed and installed in a raceway of that division. Each associated cable will be uniquely identified as described in Section 8.3.1.5.

6. Separation Criteria Before and After an Isolation Device

   The cables and circuits before an isolation device are Class 1E circuits and are routed in Class 1E raceway system in accordance with criteria described in item 2 above for physical separation of Class 1E cables. The cables and circuits after the isolation device are considered non-Class 1E cables and circuits and are routed in non-Class 1E raceway system.

The minimum separation maintained between cables and circuits of each division varies according to cable location with respect to potential hazards. The design intent is to provide separation greater than the minimum listed where consistent with a practical plant layout. Six general classifications of hazard zones or areas are defined for electrical separation consideration:

I. Non-Hazard Zones

   Areas in which the only potential hazard is a fire of an electrical nature.

II. Fire Hazard Zones

   Areas in which a potential fire hazard could exist as a consequence of the credible accumulation of a significant quantity of fixed or transient combustible materials as defined in PSAR Section 9.13.1.

III. Equipment Hazard Zone (Pipe Break Hazard Zone)

   Areas in which a potential hazard could exist as a consequence of postulated pipe break events in high energy lines.

IV. Cable Spreading Rooms

Areas just above and below the main control room where control and instrumentation cables converge prior to entering the control room.

V. Containment Electrical Penetration Areas

The areas and assemblies that allow cable passage through the Containment Building pressure boundary.

VI. Control Room

Continuously manned utilized by plant operators to monitor and control the plant.

Non-Hazard Zones

Redundant cables entering panels, cabinets or other equipment enter through separate openings.

In Non-Hazard Zones, no minimum vertical or horizontal physical separation is provided between conduits of the same division beyond that required for construction, installation or access clearances between conduits and/or metal enclosed ducts.

In Non-Hazard Zones, exposed conduits of different Class 1E divisions are routed as far apart as possible, preferably on opposite sides of the walls. Parallel routing of conduits of different divisions is avoided. If the design makes it unavoidable a minimum of one (1) inch spatial separation is provided between conduits of different Class 1E divisions as shown in Figure 8.3-6. When the safety related conduit cross or run parallel to another safety related tray the minimum horizontal and vertical clearance is the same as provided for cable trays of different Class 1E divisions. If this clearance is unobtainable, a barrier is provided between the safety-related conduits and cable trays of other Class 1E divisions as shown in Figure 8.3-6.

In Non-Hazard Zones, a minimum horizontal clear space of three feet is maintained between cable trays of different divisions as shown in Figure 8.3-6. If a horizontal clearance of less than three feet is unavoidable, a fire barrier is provided between the divisions as shown in Figure 8.3-6.

Vertical stacking of cable trays of different divisions is avoided wherever possible. Where cable trays of different divisions are stacked vertically, a minimum clear space of five feet is provided between the divisions as shown in Figure 8.3-6. If a vertical clearance of less than five feet is unavoidable, a fire barrier is provided between the divisions as shown in Figure 8.3-6.

Fire Hazard Zones

In fire hazard zones, Class 1E conduits, trays, wireways or raceways of only one safety division are routed. This division is suitably protected by fire barriers and fire protections systems to mitigate the effects of fire in this zone on the safety function of the other safety groups.

Equipment Hazard Zone (Pipe Break Hazard Zone)

To the extent practical, Class 1E cables are routed in areas remote from high energy piping or areas of potential sodium fires; if unavoidable, the following precautions are taken:

a) CRBRP has three (3) Class 1E Divisions with complete physical separation between divisions. Any damage to cable trays caused by pipe whip missles, jet impingement, or environmental effect will be limited to the same safety division to which the pipe belongs, and the two other divisions capable of safety shutting down the plant will remain unaffected.

   Additional protection will be provided against any single Class 1E Division cable tray damage due to high energy pipe whip missiles by restraint of high energy pipe lines in the vicinity of Class 1E raceways. The design of restraints and/or barriers will be determined by analysis to meet BTP APCSB 3-1.

b) Redundant Class 1E circuits are routed or protected such that a postulated event in one system and division cannot preclude the operation of the other redundant system or division.

c) In all areas of the plant, the separation between redundant Class 1E cable raceways takes into consideration the presence of rotating equipment, monorails, equipment removal paths and dropped equipment such that failure of rotating equipment will not cause failure of more than one safety division and any dropped equipment will not cause failure of any safety-related raceways.

The likelihood of rotating equipment missile damage to Class 1E equipment* is minimized or eliminated by one or a combination of the following:

i) Qualification of Non-Class 1E and Class 1E rotating equipment to prevent missiles during the worst case seismic event (which envelopes normal operating conditions) for that rotating equipment.

ii) Segregate rotating equipment from Class 1E equipment.*

iii) Provide missile protection by walls or barriers for the Class 1E equipment.*

iv) Provide redundant equipment* necessary to meet the single failure criterion.

*Equipment is construed here to include equipment, circuits, cabling, raceways, systems, etc.

See PSAR Section 3.5 for additional information on missile protection.

d) In general, Class 1E electrical distribution equipment (e.g., switchgear, motor control centers, etc.) is not located in areas where high energy piping or other similar hazards are located.

e) In general, no Class 1E raceways are installed in equipment hazard zones or potential missile areas. Where this is not practical, only one Class 1E division is installed in the area.

f) In all areas, Class 1E raceways of adequate construction are installed so as to minimize or eliminate the possibility of damage due to potential missiles or pipe whip. The damage potential of the missile or pipe whip is evaluated, and the physical separation between different safety division raceways is specified accordingly or the raceways are located. When physical separation is not practical, appropriately designed barriers are installed between redundant raceways. The separation of redundant Class 1E division circuits and equipment make effective use of inherent plant design features such as using different rooms or opposite side of rooms or areas.

Cable Spreading Rooms

a) Two Cable Spreading Rooms are provided, one above the Control Room for Division 1, and Non-Class 1E Division A and one below the Control Room for Division 2, and Non-Class 1E Division B. Division 3 cables within the Control Room and upper Cable Spreading Room areas are routed in raceways embedded in concrete floors and walls up to the point of entry to the Division 3 panels. There are no Division 3 cables or raceways in either the upper Cable Spreading Room or the lower cable Spreading Room.

b) Circuits routed in cable trays in the cable spreading rooms are limited to control and instrument functions. No power cables are routed through the Cable Spreading Rooms or the Control Room.

c) The Cable Spreading Rooms are protected against external missiles (there are no internal sources of missiles) such as high-pressure piping or rotating heavy machinery. The only potential source of damage to redundant division cables would be from fire generated by the cabling itself. A fire detection and suppression system is installed ensuring that potential for fire damage to cables is minimized in the Cable Spreading Room.

d) There is no cable tray or conduit connection between the upper and lower Cable Spreading Rooms.

Containment Penetration Areas

Three separate penetration areas are provided for all cables that must pass through the containment wall. Where possible, redundant Class 1E cables utilize electrical penetrations spaced horizontally rather than vertically. Cables through penetration of the primary containment are grouped such that failure of all cables in a single penetration will not prevent a protective action. Separation of Class 1E circuits is maintained through penetrations. No Class 1E cables share penetrations with Non-Class 1E systems, other than associated Class 1E cable systems.

Control Room

Cables in the Control Room are kept to the minimum necessary for operation of the Control Room. All cables entering the Control Room terminate there. Cables are not installed in culverts or floor trenches.

Cables are not routed in a concealed ceiling or under floor spaces unless installed in a solid enclosed steel raceway.

Separation of Non-Class 1E Cables (Non-Safety-Related)

The separation design description for Non-Class 1E circuitry from Class 1E and associated circuits given below incorporates the requirements of IEEE Standard 384-1974 and NRC Regulatory Guide 1.75.

The trays carrying Non-Class 1E circuits are separated form the trays carrying Class 1E and associated circuits by a minimum horizontal clear space of three (3) feet and a minimum vertical clear space of five (5) feet between the trays. If a horizontal clearance of less than three (3) feet is unavoidable, a fire barrier is provided between the safety and non-safety-related trays.

Vertical stacking of safety and non-safety-related trays is avoided wherever possible. Where safety and non-safety-related cable trays are stacked vertically and a vertical clearance of less than five (5) feet is unavoidable, a fire barrier is provided between non-safety and safety-related trays.

In Cable Spreading Rooms a minimum clear separation on one (1) foot horizontal and three (3) feet vertical is maintained between trays carrying Non-Class 1E cables and trays carrying Class 1E and associated cables. If the minimum horizontal or vertical separation does not exist, a fire barrier is provided.

Non-Class 1E and Class 1E exposed conduits are routed as far as possible, preferably on opposite sides of the walls. Parallel routing of safety-related and non-safety-related conduits is avoided. If the design makes it unavoidable a minimum of one (1) inch spatial separation is provided between non-safety-related and safety-related conduits.

Plant Protection System (PPS) Separation

The PPS will meet the separation requirements of IEEE Std. 384-1974 and Regulatory Guide 1.75 and the following:

a) All PPS wiring external to control panels is run in conduit, with wiring for redundant channels run in separate conduits. Only PPS wiring is included in these conduits. Primary Shutdown System wiring is not run in the same conduit as the Secondary Shutdown System wiring.

b) Wiring for each Primary PPS instrument channel (R1A, R1B, R1C) is routed in separate conduits.

c) Wiring for each Secondary PPS instrument channel (S2A, S2B, S2C) is routed in separate conduits.

d) There are dedicated containment penetrations for each of the three Primary PPS instrument channels and each of the three Secondary PPS instrument channels which pass through containment. All requirements for separation of PPS wiring in raceways are utilized for separation of PPS wiring through containment penetrations.

e) All wiring for the three Containment Isolation System instrument channels is routed exclusively with the three Primary PPS instrument channels, or exclusively with the Secondary PPS instrument channels or through three independent conduits.

f) The Primary PPS Logic Train Actuation wiring is routed through at least three separate conduits from three separate Primary PPS Logic Train Panels to the Primary PPS Scram Breakers. One conduit contains wiring from only one Primary PPS logic train.

g) The Secondary PPS Logic Train Actuation wiring is routed through at least three separate conduits from the Secondary PPS Logic Panels to the Secondary Control Rod Solenoid Valve Actuation wiring in the Head Access Area.

h) Containment Isolation System (CIS Logic Train Actuation wiring is routed through two independent conduits. One conduit contains wiring from only one CIS Logic Train. No intermixing of CIS Logic trains within a conduit is permitted. CIS Logic Train 1 wiring is routed from CIS Logic Panel 1 to CIS Breaker 1 in the Intermediate Building. CIS Logic Trian 2 is routed from CIS Logic Panel 2 to CIS Breaker 2 in the Intermediate Building.

i) The wiring from a PPS buffered output which is used for a non-PPS purpose may be included in a PPS rack. The PPS wiring is separated from the non-PPS wiring. The amount of separation is defined on an individual case basis; however, it is designed to meet the requirements of IEEE Std. 384-1974 and Regulatory Guide 1.75.

j) Containment isolation valve actuation wiring (for either manually or automatically initiated actuation) to the inside containment and the outside containment isolation valves are separated as Division 1 and Division 2 cabling, respectively.

k) Rigid, metallic, completely enclosed and unvented raceways are considered acceptable for any of the above applications as they are equivalent to rigid metal conduit, as defined in IEEE Std. 100 and NFPA 70.

l) The physical separation between PPS conduits, containment penetrations, or panels is in accordance with IEEE Std. 384-1974 and Regulatory Guide 1.75 to provide assurance that a credible single event cannot simultaneously degrade redundant protection channels or shutdown systems.

m) The Primary Steam Generator Auxiliary Heat Removal System (SGAHRS) channels and logic outputs are treated and separated as Primary PPS signals. The primary SGAHRS logic output is kept separated from the Secondary SGAHRS logic output channels. The Secondary SGAHRS channels and logic outputs are treated and separated as Secondary PPS signals. The Secondary SGAHRS logic output is kept separated from Primary PPS, CIS and non-PPS outputs. Redundant SGAHRS logic train outputs are separated form each other. The manual trip and reset inputs to each SGAHRS divisional latch logic are routed and separated as redundant PPS signals separated form the automatic SGAHRS logic outputs and all other PPS and non-PPS channels.

Sharing of Cables Trays and Routing of Non-Class 1E Cables

There are two classes of medium voltage cable trays, 13.8 KV and 4.16 KV, these cable trays carry only 15KV and 5 KV rated cables, respectively. 480/277 volt power cables, 120/208 volt AC and 125 volt DC cables carrying more than 20 amperes, and 250 volt power cables are run in common low voltage power (LV) trays. Instrumentation trays carry the following cables: input and output cables for the computer, shielded annunciator cables used with solid-state equipment and signal cables for thermocouples, strain gauges, vibration and radiation detectors, thermal converters and RTD's. All other cables are run in control trays. Within a load group division, the minimum spacing between trays stacked vertically is 9 inches, tray bottom to tray bottom. The minimum spacing between trays installed side by side, within a load group division shall be 6 inches. The trays will be constructed of steel. All cable tray systems located in Seismic Category I structures will have supports designed to meet Seismic Category I requirements.

All PPS cables are run in conduit or enclosed raceways. PPS analog circuits may be routed together in the same raceways, provided the circuits have the same characteristics, such as power supply, shutdown system (primary or secondary), and channel identity (A, B or C).

Vital instrument cables for the PPS may be routed together in the same raceways, provided the circuits have the same characteristics, such as power supply, shutdown system (primary or secondary), and channel-identity (A, B or C)

Automatic actuation and control power circuits for the PPS may be routed in the same raceways, provided the circuits have the same characteristics, such as power supply, shutdown system (primary or secondary), and train identity (PPS logic train I, II or III).

The circuits associated with the Standby AC Power System (Class IE electric systems) are separated into three independent divisions.

Administrative Responsibilities and Control for Assuring Separation Criteria

The raceway channel identification facilitates and ensures the maintenance of separation in the routing of cables and the connection of control boards, panels and other equipment. Class 1E cables will be inspected upon installation by construction personnnel for consistency with the design documents. Color identification of equipment and cabling will be used in this effort.

8.3.1.5 Physical Identification of Class IE and Non-Class 1E Equipment

Each Class 1E and Non-Class 1E cable and raceway is given a unique identification. This identification provides a means for distinguishing a circuit or raceway associated with a particular voltage or class as well as with a particular channel or division. The channel or division is provided by a coded digit of identification and is assigned on the basis of the following criteria:

Division 1

A Class 1E instrumentation, control, power cable or raceway associated with Load Group of Division 1

Division 2

A Class 1E instrumentation, control, power cable or raceway associated wth Load Group of Division 2

Division 3

A Class 1E instrumentation, control, power cable or raceway associated with Load Group of Division 3

There will be redundant divison color identification provided for all Class 1E cables. Color-coded markers will be provided at both ends of each Class 1E cable. Cables in Class 1E raceways will be color coded at a maximum of 5 foot intervals.

Within control panels where more than one division is present, wiring will be identified by that division designation; or, if enclosed by conduit, the conduit will be identified by division designation.

Within a cabinet or panel which is associated and identified with a single division, the internal wiring will be exclusively associated with the same division, and therfore, requires no further division identification.

Conduits, trays, and electrical equipment, of Class 1E and Non-Class 1E systems, will be identified by self-sticking markers with the color associated with its division.

The raceway system will be marked prior to installation of cables in a distinct permanent manner at intervals not to exceed 15 ft and at points of entry to and existing from enclosed areas.

Class 1E electric equipment labels will be color coded with the same division color coding used for Class 1E cables and raceways. Color coding will be permanent and prominently displayed.

Design drawings, operating documents and maintenance documents will provide distinct identification of Class 1E equipment. The applicable channel or division designation will also be identified. The identification scheme for Non-Class 1E cables and raceways will, in general, be similar to that used for the Class 1E circuits and raceway except that different color-coded markers will be used.

8.3.1.6 Grounding Requirements

The following criteria will be used in the design of electrical system grounding:

a) Recommendations of IEEE Standards 80 and 142 will be followed in the design of the grounding system.

b) Generator step-up transformer will be grounded using a grounding reactor.

c) 4.16KV and 13.8KV windings of the Unit Station and Reserve Station Service Transformers, will be low resistance grounded.

d) 480V Non-Class 1E system will be solidly grounded.

e) Class 1E system will be high resistance grounded.

f) DC system will remain ungrounded.

g) Ground grid design will meet safe touch and step potential requirements.

h) The plant grounding system will provide grounding for all electrical equipment and metallic objects subjected to accidental electrical connection.

8.3.2 DC Power System

8.3.2.1 Description

The CRBRP DC Power System as shown in Figure 8.3-2 is comprised of the following separate and independent power supplies:

- o Three Class 1E 125 volt DC systems, (Divisions 1, 2 and 3)
- o Two Non-Class 1E 125 volt DC systems, dedicated for plant control, alarm, indication and instrumentation (Divisions A and B)
- o One Non-Class 1E 250 volt DC system, Division A
- o Two Non-Class 1E 125 volt DC systems, dedicated for switchyard control, alarm, indication and instrumentation. (Divisions A and B)
- o Two Non-Class 1E 48 volt DC systems, dedicated for the Plant Communication System (Divisions A and B)
- o Two Non-Class 1E DC systems dedicated for the Plant Physical Security System. (Not shown in Fig. 8.3-2)

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8.3.2.1.1 Class 1E DC Power System

The Class 1E DC Power System as shown in Figure 8.3-2 is comprised of three independent DC Systems, rated at 125 volt DC. Each DC System consists of a battery, battery chargers, DC switchgear and distribution panels. All DC Systems are ungrounded and are provided with ground detectors to indicate if there is a ground positive or negative on the battery. A ground on one side of the battery will not cause equipment to malfunction, and will be annunciated in the control room. Each DC System is provided with undervoltage detection to annunciate in the control room for low voltage condition or power interruption at the buses. In addition, each battery charger is provided with an alarm relay which annunciates in the control room failure of AC supply to the charger.

Battery chargers, distribution buses and distribution panels are furnished with individual metal enclosures. Physical separation, and where required, fire barriers are provided between components and cabling of the redundant divisions so that a fire could not cause simultaneous interruption of power to the load groups. Ventilation for DC equipment is described in Section 9.6.1.

The Class 1E DC System provides a reliable source of continuous power for control, instrumentation, vital bus inverters and other plant DC loads. This system is comprised of three independent redundant Division 1, 2 and 3 125 volt DC Systems. Each Division supplies power for control of a redundant load group. See Tables 8.3-2A, 8.3-2B and 8.3-2C for listing of loads and figure 8.3-2 for system configuration.

Each 125 volt DC system consists of one battery, two solid state battery chargers, one DC switchgear and necessary instrumentation for system monitoring. The system is arranged so that the battery or any one battery charger can independently supply the system bus load. Each battery is provided with a 100 percent capacity spare battery charger. Power for the 125 volt DC system is normally supplied through the battery charger from a 480V AC motor control center (MCC). This MCC will automatically be supplied from the standby diesel generator on loss of Plant and Offsite AC Power Sources. The battery chargers, for each divison, are powered from the associated diesel generator of that safety division.

Since the Division 1, 2 and 3, 125 volt DC Power Systems are totally independent and serve redundant Class 1E loads, the complete loss of a division will not affect the ability of the plant to achieve safe shutdown and/or mitigate the consequences of an emergency condition.

The battery chargers for each division are 480V AC to 125V DC solid state chargers. Each Division has one active and one spare battery charger. Each battery charger is capable of carrying the normal steady state DC load for its division while simultaneously maintaining its battery in a fully charged "float" condition. The chargers are capable of supplying the

125 volt DC power requirements for all modes of plant operation while restoring the batteries from a discharge condition of 1.75 volts per cell to full charge within 12 hours. The charger size and instrumentation conform to the requirements of IEEE Std. 308.

Each battery charger is provided with an integrally mounted, incoming AC circuit breaker, and an output DC breaker with magnetic trip to isolate the charger from the DC bus in case of an internal fault in the charger. The output current of each battery charger will be monitored in the Control Room. A battery charger failure will be annunciated in the Control Room.

In the event of a loss or interruption of battery charger 125 volt DC output, the station batteries will maintain power to their respective DC systems. The ampere-hour capacity and duration of the batteries is in accordance with IEEE Std. 308. Batteries are capable of supplying for at least 2 hours, all DC power required to safely shutdown the plant and/or to limit the consequences of a design basis accident without recharging.

Batteries are connected to the DC bus through a circuit breaker. The current output of each battery will be monitored in the Control Room, and a high battery discharge will be annunciated in the Control Room.

Cables for the three redundant load groups of the Class 1E DC power systems will be installed in separate divisions of the Class 1E raceway system. The Class 1E cable and raceway independence criteria are described in Section 8.3.1.4. Cables and raceways of the Class 1E DC power system will be marked in a distinctive manner as described in Section 8.3.1.5.

8.3.2.1.2 Non-Class 1E DC Power System

The Non-Class 1E DC system is comprised of nine separate DC systems, six systems are rated at 125 volt DC, two systems are 48 volt DC and one system at 250 volt DC. See Figure 8.3-2 for system configurations.

Two of the 125 volt DC systems are dedicated for plant control, alarm and instrumentation. Each of these systems are comprised of one battery, three solid state battery chargers, one DC switchgear, distribution panels and the necessary instrumentation for system monitoring. The system is arranged so that the battery or any two battery chargers can independently supply the system bus load. Each battery is provided with one spare battery charger. Power for the 125V DC Systems is normally supplied through the battery chargers from the 480V AC motor control centers. The battery chargers are capable of supplying the 125V DC power requirements for all modes of plant operation while restoring the batteries from a discharge condition of 1.75 volts per cell to full charge within 12 hours. Each battery except those for the Plant Security System is sized to power required loads for a period of 2 hours without recharging in the event of loss of all AC power. The DC batteries for the Plant Security System are not described herein.

Two 125 volt DC systems are dedicated for switchyard power and control. Each of these systems are comprised of one battery, one solid state battery charger, one DC power panel and the necessary instrumentation for system monitoring. The system is arranged so that the battery or battery charger can independently supply the system bus load. Power for the 125V DC systems is normally supplied through the battery charger from a 480V motor control center. The battery charger is capable of supplying the 125V DC power requirements for all modes of plant operation while restoring the batteries from a discharge condition of 1.75 volts per cell to full charge within 12 hours. Each battery is sized to power the required loads for a period of 2 hours without recharging in the event of loss of all AC power.

One 250 volt DC system is used to provide power for DC motors of the back-up oil bearing pumps associated with the main turbine generator, the motor generator sets of the sodium pump drive system, and an AC inverter for loads of the fire protection and other systems. This system is comprised of one battery, three solid state battery chargers, one DC switchgear and the necessary instrumentation for system monitoring. The system is arranged so that the battery or any two battery chargers can independently supply the system bus load. One spare battery charger is provided for this battery. Power for the 250V DC systems is normally supplied through the battery chargers from 480V AC motor control centers. The battery chargers are capable of supplying the 250V DC power requirements for all modes of plant operation while restoring the batteries from a discharge condition of 1.75 volts per cell to full charge within 12 hours. The battery is sized to power required loads for a period of 2 hours without recharging in the event of loss of all AC power.

Two 48 volt DC systems are dedicated for the Plant Communication System. Each of these systems are comprised of one battery, one solid state battery charger, one DC power panel and the necessary instrumentation for system monitoring. The system is arranged so that the battery or battery charger can independently supply the system bus load. Power for the 48V DC system is normally supplied through the charger from 480V AC motor control center. Each battery charger is capable of supplying the 48V DC power requirements for all modes of plant operation while restoring the batteries from a discharge condition of 1.75 volts per cell to full charge within 12 hours. The battery is sized to accommodate all connected loads for a period of 2 hours in the event of loss of all AC power.

8.3.2.1.3 System Operations, Testing and Inspection

System Operations

The system steady-state voltage during normal operation will be determined by the output voltage of the battery charger. The output voltage level will be manually set to a predetermined value for float-charging the battery. When restoring charge to a discharged battery, the output voltage level of the battery charger can be manually increased, to the maximum permissible system voltage, for battery cell voltage equalization.

When the battery charger(s) is (are) unavailable, the battery supplies power to the system loads and the system voltage decays as the battery approaches its design minimum charge. The system voltage can vary from the low level associated with the design minimum charge of the battery, to a maximum level established by the battery charger.

All loads are designed to operate satisfactorily over the full operating range of the system voltage described above.

A voltmeter and ammeter will be mounted on each battery charger to monitor output voltage and current. The battery voltage and current will be monitored by indicating instruments located at the associated Class 1E DC distribution bus.

A ground detector will be provided for each channel of the Class 1E DC power system to annunciate, in the Control Room, the existence of a circuit ground in the battery supply, the output circuit of the battery charger, or the associated distribution system. An undervoltage relay will be connected to each Class 1E DC distribution bus to annunciate, in the Control Room, a low voltage condition or power interruption at the DC buses.

In addition, each battery charger will be provided with an alarm relay which provides annunciation in the Control Room for failure of the AC supply to the charger.

Each Class 1E battery and certain Non-Class 1E batteries are provided with spare battery chargers. The spare battery charger will be put in service in the event the active battery charger is under maintenance, testing or has failed.

Testing and Inspection

Periodic maintenance tests will be performed in accordance with IEEE std. 308-1974, 450-1975 and Regulatory Guides 1.32, on all DC system components to determine the condition of each individual subsystem. Batteries and battery chargers will be tested and inspected as follows:

A. The specific gravity, voltage, and temperature of the pilot cell and the overall battery voltage of each battery supply are measured and logged weekly during normal plant operation.

B. Cells are visually checked weekly for possible cracks, electrolyte leaks, or corrosion of terminals.

C. Voltage, specific gravity, and the liquid level of each cell and electrolyte temperature of every fifth cell are measured and logged monthly.

D. Each Class 1E battery will be subjected to a performance discharge test at three year intervals, during a refueling period, to ascertain that it is capable of delivering its rated capacity. The specific gravity and voltage of each cell will be measured and logged after the discharge.

Battery chargers in continuous service will be tested to demonstrate their operability and performance by supplying power to the loads. The standby battery charger will be tested during normal plant operation by putting the unit in service to demonstrate that it is operable and performs properly. Each battery charger will also be subjected to a weekly visual inspection and a performance test during each refueling period to ascertain that it is capable of operating within design limits.

Circuit breakers in continuous service will be tested to demonstrate their operating capability and performance by carrying electrical load. Circuit breakers of standby equipment will be tested for their operability by putting the standby equipment in service when plant operating conditions permit. All circuit breakers will be subjected to inspection, operational tests, and maintenance (if required) during prolonged plant shutdown periods.

For Class 1E DC System components, the manufacturer will perform operational tests or calculate data which will substantiate the capability of the equipment to carry their DC loads. Preoperational testing will be performed to verify that all components, automatic and manual controls and functions of the DC systems perform as required.

8.3.2.2 Analysis

Compliance with CRBRP General Design Criteria 15 and 16 is discussed in Section 3.1. The Class 1E DC Power Systems are discussed below to indicate the degree of conformance with: NRC Regulatory Guides 1.6, 1.32, 1.41, 1.47, 1.68, 1.75, 1.128, 1.129 and IEEE Standards 308, 317, 323, 384, 450, 484 and 485.

8.3.2.2.1 NRC Regulatory Guide 1.6, Rev. 0 (3/71)

Safety-Related DC Distribution System provides separation of the DC powered Class 1E loads into two redundant load groups so that loss of one group will not prevent the safe shutdown of the plant.

Each DC redundant load group is powered by a battery and a battery charger supply. The combined battery and battery charger supplies assigned to one redundant load group have no automatic or manual connection to the other redundant load group.

8.3.2.2.2 NRC Regulatory Guide 1.32, Rev. 2 (2/77)

The battery charger supply will have the capacity to furnish electric energy for the largest combined demands of the various steady-state loads, and the charging capacity to restore the battery from the design minimum charge state to the fully charged state, irrespective of the status of the plant during which these demands occur.

8.3.2.2.3 NRC Regulatory Guide 1.41, Rev. 0 (3/73)

Preoperational tests will be performed to verify the independence of the redundant DC power sources and between the redundant load groups described in Section 8.3.2.1. During these tests the power sources to the

Class 1E DC power distribution bus of the redundant load group not under test will be disconnected and the bus voltage continuously monitored to verify the absence of voltage at the bus.

The tests will be repeated for the redundant load group and sources.

The tests will be of sufficient duration to attain steady-state operation of the DC power sources as well as steady-state operation of the loads under test.

8.3.2.2.4 NRC Regulatory Guide 1.47, Rev. 0 (5/73)

Class 1E systems will be designed to meet the guidelines of this Regulatory Guide as described in Section 8.3.1.2.

8.3.2.2.5 NRC Regulatory Guide 1.68, Rev. 2 (8/78)

Written procedures for preoperational and startup testing for electrical distribution system will be developed to meet the guidelines of this Regulatory Guide as described in Section 8.3.1.2.

8.3.2.2.6 NRC Regulatory Gide 1.75, Rev. 2 (9/78)

Class 1E systems will be designed to meet the guidelines of this Regulatory Guide as described in Section 8.3.1.2.

8.3.2.2.7 NRC Regulatory Guide 1.128, Rev. 1 (10/78)

The Class 1E DC power system will be designed to conform to the requirement of IEEE Std. 344-1975 and 484-1975 supplemented by Regulatory Guide 1.128 per the following:

a) The location of the Class 1E DC system meets the design criteria of IEEE Std. 344 and supplemental Regulatory Guide 1.100.

b) Battery area ventilation system will limit hydrogen concentration to less than two percent by volume.

c) Fire protection in battery areas is in accordance with Regulatory Guide 1.120.

d) Any cell which, when unpacked, exhibits an electrolyte level 1/2 inch or more below the top plates will be replaced.

e) Class 1E batteries will be stored such that cells will not be exposed to low ambient temperatures or localized heat sources.

8.3.2.2.8 NRC Regulatory Guide 1.129, Rev. 1 (2/78)

Class 1E systems meet the guidelines of this Regulatory Guide as described in Section 8.3.2.1.1.

8.3.2.2.9 IEEE Standard 308 - 1974

The Class 1E DC power systems will be designed to conform to the requirements of Class 1E electric systems as indicated in Section 8.3.1.2.

8.3.2.2.10 IEEE Standard 317 - 1976

The electrical penetration assemblies which pass cables through the containment exterior wall will be designed and tested in conformance with IEEE Standard 317 - 1976.

8.3.2.2.11 IEEE Standard 323 - 1974

The Class 1E DC Power Systems conform to this standard as described in Section 8.3.1.2.

8.3.2.2.12 IEEE Standard 384 - 1974

The separation criteria of IEEE 384 - 1974 will be applied to insure the physical independence of redundant Class 1E electrical systems. (See Section 8.3.1.4)

8.3.2.2.13 IEEE Standard 450 - 1975

Periodic tests on the Class 1E 125 DC batteries will be performed in conformance with this standard. The periodic tests and test intervals are described in Section 8.3.2.1.1.

8.3.2.2.14 IEEE Standard 485 - 1978

Batteries are sized in accordance with IEEE Std. 485 with battery capacity temperature correction and aging factors. Each battery is sized for its maximum expected load duty cycle including a margin for load growth.

TABLE 8.3-1A

CLASS 1E DIVISION 1 DIESEL GENERATOR LOAD LIST

THE MACHINE SHALL ON RECEIPT OF START SIGNAL, (TIME 0), START AND ATTAIN RATED SPEED AND VOLTAGE WITHIN 10 SECONDS, AND ACCEPT LOADS IN ANY OF THE SEQUENCES STATED BELOW.

					APPLICABLE CONDITIONS OF OPERATION			
LOAD DESCRIPTION	NAME PLATE HP	KW (IN)	SEQUENCE OF OPERATION	DURATION OF OPERATION	UV (KW)	SGAHRS (KW)	DHRS (KW)	TMBDB (KW)
STANDBY LIGHTING SYSTEM PANELS		50	10sec	Continuous	50	50	50	50
BATTERY CHARGER (130VDC)		115	10sec	Continuous	115	115	115	-
BATTERY CHARGER (130VDC)		115	10sec	Back Up	-	-	-	-
FUEL OIL TRANSFER PUMP	5	4.2	10sec	Inter.	4.2	4.2	4.2	4.2
FUEL OIL TRANSFER PUMP	5	4.2	10sec	Back up	-	-	-	-
CONTROL ROOM UNIT SUPPLY FAN	75	52.0	10sec	Continuous	52.0	52.0	52.0	52.0
CONTROL ROOM FILTER UNIT FAN	20	11.6	10sec	Continuous	11.6	11.6	11.6	11.6
CONTROL ROOM RETURN FAN	30	17.6	10sec	Continuous	17.6	17.6	17.6	17.6
ANNULUS FILTER FANS	30	17.1	10sec	Continuous	17.1	17.1	17.1	17.1
ANNULUS PRESS. MAINT. FAN	7.5	4.9	10sec	Continuous	4.9	4.9	4.9	4.9

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TABLE 8.3-1A

CLASS 1E DIVISION 1 DIESEL GENERATOR LOAD LIST (Cont'd)

LOAD DESCRIPTION	NAME PLATE HP	KW (IN)	SEQUENCE OF OPERATION	DURATION OF OPERATION	APPLICABLE CONDITIONS OF OPERATION UV (KW)	SGAHRS (KW)	DHRS (KW)	TMBDB (KW)
SYST. 56 PANEL UNIT COOLER	1.5	0.8	10sec	Continuous	0.8	0.8	0.8	-
SYST. 56 PANEL UNIT COOLER	1.5	0.8	10sec	Continuous	0.8	0.8	0.8	-
REACTOR CONTAINMENT BLDG. INST. DIVISION 1		0.4	10sec	Continuous	0.4	0.4	0.4	0.4
REACTOR CONTAINMENT TMBDB INST. DIVISION 1		1.5	10sec	Continuous	1.5	1.5	1.5	1.5
REACTOR CONTAINMENT TMBDB INST. DIVISION 1		1.5	10sec	Continuous	1.5	1.5	1.5	1.5
SUMP DISCHARGE CV ISOLATION VALVE (INNER)	1.5	1.4	10sec	Continuous	1.4	1.4	1.4	1.4
SGAHRS AUX. FEED WATER PUMP DR. MTR.	1050	833	10sec	Continuous	833	833	-	-
SYSTEM 81 I&C PANEL A		0.6	10sec	Continuous	-	0.6	0.6	-
SYSTEM 81 I&C PANEL A		1.3	10sec	Continuous	-	1.3	1.3	-
SYSTEM 81 I&C PANEL A		1.6	10sec	Continuous	-	1.6	1.6	-
TOTAL LOAD FOR 10 SEC BLOCK				Continuous	1112	1115	282	161

TABLE 8.3-1A

CLASS 1E DIVISION 1 DIESEL GENERATOR LOAD LIST (Cont'd)

LOAD DESCRIPTION	NAME PLATE HP	KW (IN)	SEQUENCE OF OPERATION	DURATION OF OPERATION	APPLICABLE CONDITIONS OF OPERATION UV (KW)	SGAHRS (KW)	DHRS (KW)	TMBDB (KW)
BATTERY ROOM EXH. FAN	0.33	0.3	20sec	Continuous	0.3	0.3	0.3	0.3
BATTERY ROOM EXH. FAN	0.33	0.2	20sec	Continuous	0.2	0.2	0.2	0.2
DIESEL RM "A" EMERG. SUPPLY FAN	20	10.6	20sec	Continuous	10.6	10.6	10.6	10.6
DIESEL RM "A" EMERG. SUPPLY FAN	20	10.6	20sec	Continuous	10.6	10.6	10.6	10.6
AUX. FEED PUMP UNIT COOLER	7.5	5.2	20sec	Continuous	5.2	5.2	5.2	-
AUX. FEED PUMP UNIT COOLER	7.5	4.0	20sec	Continuous	4.0	4.0	4.0	-
EI&C CUBICLE UNIT COOLER	2	1.1	20sec	Continuous	1.1	1.1	1.1	-
EI&C CUBICLE UNIT COOLER	2	1.1	20sec	Continuous	1.1	1.1	1.1	-
ANNULUS FILTER HEATERS		133	20sec	Continuous	133	133	133	133
DAY TANK CELL EXH. FAN	0.125	0.18	20sec	Continuous	0.18	0.18	0.18	0.18
AIR COOLED CONDENSER BLOWER MOTOR	60	45	20sec	Continuous	45	45	-	-
AIR COOLED CONDENSER BLOWER MOTOR	60	45	20sec	Continuous	45	45	-	-

TABLE 8.3-1A

CLASS 1E DIVISION 1 DIESEL GENERATOR LOAD LIST (Cont'd)

LOAD DESCRIPTION	NAME PLATE HP	KW (IN)	SEQUENCE OF OPERATION	DURATION OF OPERATION	APPLICABLE CONDITIONS OF OPERATION UV (KW)	SGAHRS (KW)	DHRS (KW)	TMBDB (KW)
PHTS Na. PUMP PONY MOTOR	75	18.6	20sec	Continuous	18.6	18.6	18.6	-
IHTS Na. PUMP PONY MOTOR	75	18.6	20sec	Continuous	18.6	18.6	18.6	-
ARD BEARING FAN MTR. PRIM	2	2	20sec	Continuous	2	2	2	-
ARD BEARING FAN MTR. INTERM	2	2	20sec	Continuous	2	2	2	-
EMERGENCY PLANT SERVICE WATER PUMP	150	108	20sec	Continuous	108	108	108	108
COOLING TOWER FAN	25	15	20sec	Continuous	15	15	15	15
COOLING TOWER FAN	25	15	20sec	Continuous	15	15	15	15
COOLING TOWER FAN	25	15	20sec	Continuous	15	15	15	15
EMERGENCY PLANT SERVICE WATER MAKEUP PUMP	7.5	5.7	20sec	Continuous	5.7	5.7	5.7	5.7
TOTAL LOAD FOR 10 SEC AND 20 SEC BLOCKS				Continuous	1568	1571	648	475

TABLE 8.3-1A

CLASS 1E DIVISION 1 DIESEL GENERATOR LOAD LIST (Cont'd)

LOAD DESCRIPTION	NAME PLATE HP	KW (IN)	SEQUENCE OF OPERATION	DURATION OF OPERATION	APPLICABLE CONDITIONS OF OPERATION UV (KW)	SGAHRS (KW)	DHRS (KW)	TMBDB (KW)
SWGR A/C UNIT "A" (SUPPLY FAN)	25	15.2	1 min	Continuous	15.2	15.2	15.2	15.2
SWGR RETURN FAN "A"	10	6.1	1 min	Continuous	6.1	6.1	6.1	6.1
SG LOOP 1 SUPPLY FAN	200	137	1 min	Continuous	137	137	137	-
SG LOOP 1 SUPPLY FAN	200	137	1 min	Back up	-	-	-	-
SG LOOP 3 SUPPLY FAN	200	149	1 min	Continuous	149	149	149	-
SG LOOP 3 SUPPLY FAN	200	149	1 min	Back up	-	-	-	-
SGB-IB SUPPLY FAN	40	22	1 min	Continuous	22	22	22	22
SGB-IB SUPPLY FAN	40	22	1 min	Back up	-	-	-	-
SG LOOP 1 EXH. FAN	75	44	1 min	Continuous	44	44	44	-
SG LOOP 1 EXH. FAN	75	44	1 min	Back up	-	-	-	-

TABLE 8.3-1A

CLASS 1E DIVISION 1 DIESEL GENERATOR LOAD LIST (Cont'd)

LOAD DESCRIPTION	NAME PLATE HP	KW (IN)	SEQUENCE OF OPERATION	DURATION OF OPERATION	APPLICABLE CONDITIONS OF OPERATION UV (KW)	SGAHRS (KW)	DHRS (KW)	TMBDB (KW)
SG LOOP 3 EXH. FAN	75	51	1 min	Continuous	51	51	51	-
SG LOOP 3 EXH. FAN	75	51	1 min	Back up	-	-	-	-
SGB-IB EXH. FAN	15	10	1 min	Continuous	10	10	10	10
SGB-IB EXH. FAN	15	10	1 min	Back up	-	-	-	-
ABHX CELL UNIT COOLER	3	2.1	1 min	Continuous	2.1	2.1	2.1	2.1
ELECT. EQUIPMENT UNIT COOLER	2	1.2	1 min	Continuous	1.2	1.2	1.2	1.2
PRIMARY SODIUM MAKEUP PUMP "A"	75	63	1 min	Continuous	-	18	63	-
EVST NaK PUMP "A"	150	117	1 min	Continuous	-	77	117	-
EVST NaK PUMP "A" BLOWER	15	12.5	1 min	Continuous	-	12.5	12.5	-
EVST AIR BLAST HEAT EXCHANGER "A"	125	112.5	1 min	Continuous	-	63	112.5	-
TOTAL LOAD FOR 10 SEC THROUGH 1 MIN BLOCKS				Continuous	2006	2179	1391	531

TABLE 8.3-1A

CLASS 1E DIVISION 1 DIESEL GENERATOR LOAD LIST (Cont'd)

LOAD DESCRIPTION	NAME PLATE HP	KW (IN)	SEQUENCE OF OPERATION	DURATION OF OPERATION	APPLICABLE CONDITIONS OF OPERATION UV (KW)	SGAHRS (KW)	DHRS (KW)	TMBDB (KW)
EMERGENCY CHILLED WATER PUMP	75	54	3 min	Continuous	54	54	54	54
TOTAL LOAD FOR 10 SEC THRU 3 MIN BLOCKS				Continuous	2060	2233	1445	585
EMERGENCY CHILLER	667 KW	469	4 min	Continuous	469	469	469	469
EMERGENCY CHILLER CONTROL CENTER PANEL		5	4 min	Continuous	5	5	5	5
TOTAL LOAD FOR 10 SEC THROUGH 4 MIN BLOCKS				Continuous	2534	2707	1919	1059
EMERG. CHILLER ROOM UNIT COOLER	0.75	.53	5 min	Continuous	.53	.53	.53	.53
FAN MOTOR	100	55	5 min	Continuous	55	55	55	55

TABLE 8.3-1A

CLASS 1E DIVISION 1 DIESEL GENERATOR LOAD LIST (Cont'd)

LOAD DESCRIPTION	NAME PLATE HP	KW (IN)	SEQUENCE OF OPERATION	DURATION OF OPERATION	APPLICABLE CONDITIONS OF OPERATION UV (KW)	SGAHRS (KW)	DHRS (KW)	TMBDB (KW)
FAN MOTOR	60	35	5 min	Continuous	35	35	35	35
EVST SODIUM PUMP "A"	75	63	5 min	Continuous	-	63	63	-
TOTAL LOAD FOR 10 SEC THROUGH 5 MIN BLOCKS				Continuous	2624	2861	2072	1150
ANNULUS COOLING FAN	300	215.4	manual	Continuous	-	-	-	215.4
ANNULUS COOLING FAN	300	215.4	manual	Continuous	-	-	-	215.4
ANNULUS COOLING FAN	300	215.4	manual	Continuous	-	-	-	215.4
CONT. CLEAN-UP SCRB EXH. FAN	250	154	manual	continuous	-	-	-	154
CLEAN-UP FILTER HEATER		60	manual	Continuous	60	60	60	60
CLEAN-UP FILTER FAN	40	22.6	manual	Continuous	22.6	22.6	22.6	22.6
ANNULUS FILTER UNIT COOLER	1	0.8	manual	Continuous	0.8	0.8	0.8	0.8
CLEAN-UP FILTER UNIT COOLER	7.5	5.3	manual	Continuous	5.3	5.3	5.3	5.3

TABLE 8.3-1A

CLASS 1E DIVISION 1 DIESEL GENERATOR LOAD LIST (Cont'd)

LOAD DESCRIPTION	NAME PLATE HP	KW (IN)	SEQUENCE OF OPERATION	DURATION OF OPERATION	APPLICABLE CONDITIONS OF OPERATION UV (KW)	SGAHRS (KW)	DHRS (KW)	TMBDB (KW)
CLEAN-UP PUMP UNIT COOLER	3	1.9	manual	Continuous	-	-	-	1.9
CLEAN-UP SCRUBBER CELL UNIT COOLER	3	1.6	manual	Continuous	-	-	-	1.6
CLEAN-UP CHASE UNIT COOLER	0.5	.4	manual	Continuous	-	-	-	.4
SEISMICALLY QUAL. FIRE PUMP CONTROLLER	25	13.1	manual	Continuous	13.1	13.1	13.1	13.1
PANEL, LEAK MONITOR	1.14KW	0.76	manual	Continuous	0.76	0.76	0.76	0.76
PANEL, LEAK MONITOR	2.77KW	1.9	manual	Continuous	1.9	1.9	1.9	1.9
PANEL, LEAK MONITOR	2.78KW	1.9	manual	Continuous	1.9	1.9	1.9	1.9
TMBDB CIRCULATING WATER PUMP (HOLD 75001590)	300	220	manual	Continuous	-	-	-	220
TOTAL LOAD				Continuous	2731	2967	2179	2277
FUTURE LOAD (15% GROWTH)				Continuous	410	445	327	341
TOTAL LOAD INCLUDING FUTURE LOAD				Continuous	3140	3412	2506	2618

TABLE 8.3-1A

CLASS 1E DIVISION 1   DIESEL GENERATOR LOAD SUMMARY

LOAD SUMMARY

LOAD BLOCK	TIME ON	BUS UNDERVOLTAGE		SGAHRS		DHRS		TMBDB	
		LOAD KW	CUMUL. KW	LOAD KW	CUMUL. KW	LOAD KW	CUMUL. KW	LOAD KW	CUMUL. KW
1	10 sec	1112	1112	1115	1115	282	282	161	161
2	20 sec	456	1568	456	1571	366	648	314	475
3	1 min	438	2006	608	2179	743	1391	56	531
4	3 min	54	2060	54	2233	54	1445	54	585
5	4 min	474	2534	474	2707	474	1919	474	1059
6	5 min	90	2624	154	2861	153	2072	91	1150
	Manual	107	2731	106	2967	107	2179	1127	2277

TABLE 8.3-1B

CLASS 1E DIVISION 2 DIESEL GENERATOR LOAD LIST

THE MACHINE SHALL ON RECEIPT OF START SIGNAL, (TIME 0), START AND ATTAIN RATED SPEED AND VOLTAGE WITHIN 10 SECONDS, AND ACCEPT LOADS IN ANY OF THE SEQUENCES STATED BELOW.

LOAD DESCRIPTION	NAME PLATE HP	KW (IN)	SEQUENCE OF OPERATION	DURATION OF OPERATION	APPLICABLE CONDITIONS OF OPERATION UV (KW)	SGAHRS (KW)	DHRS (KW)	TMBDB (KW)
STANDBY LIGHTING SYSTEM PANELS		50	10sec	Continuous	50	50	50	50
BATTERY CHARGER (130VDC)		115	10sec	Continuous	115	115	115	-
BATTERY CHARGER (130VDC)		115	10sec	Back up	-	-	-	-
FUEL OIL TRANSFER PUMP	5	4.2	10sec	Inter.	4.2	4.2	4.2	4.2
FUEL OIL TRANSFER PUMP	5	4.2	10sec	Back up	-	-	-	-
CONTROL ROOM UNIT SUPPLY FAN	75	52	10sec	Continuous	52	52	52	52
CONTROL ROOM FILTER UNIT FAN	20	11.6	10sec	Continuous	11.6	11.6	11.6	11.6
CONTROL ROOM RETURN FAN	30	17.6	10sec	Continuous	17.6	17.6	17.6	17.6
ANNULUS FILTER FANS	30	17	10sec	Continuous	17	17	17	17
ANNULUS PRESS. MAINT. FAN	7.5	4.9	10sec	Continuous	4.9	4.9	4.9	4.9
SYST. 56 PANEL UNIT COOLER	1.5	0.8	10sec	Continuous	0.8	0.8	0.8	-

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TABLE 8.3-1B

CLASS 1E DIVISION 2 DIESEL GENERATOR LOAD LIST (Cont'd)

LOAD DESCRIPTION	NAME PLATE HP	KW (IN)	SEQUENCE OF OPERATION	DURATION OF OPERATION	APPLICABLE CONDITIONS OF OPERATION UV (KW)	SGAHRS (KW)	DHRS (KW)	TMBDB (KW)
REACTOR CONTAINMENT BUILD. INST. DIVISION II		0.5	10sec	Continuous	0.5	0.5	0.5	0.5
REACTOR CONTAINMENT TMBDB INST. DIVISION II		1.5	10sec	Continuous	1.5	1.5	1.5	1.5
REACTOR CONTAINMENT TMBDB INST. DIVISION II		1.5	10sec	Continuous	1.5	1.5	1.5	1.5
SUMP DISCHARGE CV ISOLATION VALVE (OUTER)	1.5	1.4	10sec	Continuous	1.4	1.4	1.4	1.4
SGAHRS AUX. FEED WATER PUMP DR. MTR.	1050	833	10sec	Continuous	833	833	-	-
SYSTEM 81 I&C PANEL 'B'		.25	10sec	Continuous	-	.25	.25	-
SYSTEM 81 I&C PANEL 'B'		.75	10sec	Continuous	-	.75	.75	-
SYSTEM 81 I&C PANEL 'B'		1.25	10sec	Continuous	-	1.25	1.25	-
LOAD FOR 10 SEC BLOCK				Continuous	1111	1114	281	161

TABLE 8.3-1B

CLASS 1E DIVISION 2 DIESEL GENERATOR LOAD LIST (Cont'd)

LOAD DESCRIPTION	NAME PLATE HP	KW (IN)	SEQUENCE OF OPERATION	DURATION OF OPERATION	APPLICABLE CONDITIONS OF OPERATION UV (KW)	SGAHRS (KW)	DHRS (KW)	TMBDB (KW)
BATTERY ROOM EXH. FAN	0.5	0.21	20sec	Continuous	0.21	0.21	0.21	0.21
BATTERY ROOM EXH. FAN	0.5	0.3	20sec	Continuous	0.3	0.3	0.3	0.3
DIESEL RM "B" EMERG. SUPPLY FAN	20	10.6	20sec	Continuous	10.6	10.6	10.6	10.6
DIESEL RM "B" EMERG. SUPLY FAN	20	10.6	20sec	Continuous	10.6	10.6	10.6	10.6
AUX. FEED PUMP UNIT COOLER	7.5	5.2	20sec	Continuous	5.2	5.2	5.2	-
AUX. FEED PUMP UNIT COOLER	7.5	4.0	20sec	Continuous	4.0	4.0	4.0	-
EI&C CUBICLE UNIT COOLER	2	1.0	20sec	Continuous	1.0	1.0	1.0	-
ANNULUS FILTER HEATERS		133	20sec	Continuous	133	133	133	133
DAY TANK CELL EXH. FAN	0.125	0.18	20sec	Continuous	0.18	0.18	0.18	0.18
AIR COOLED CONDENSER BLOWER MOTOR	60	45	20sec	Continuous	45	45	-	-
AIR COOLED CONDENSER BLOWER MOTOR	60	45	20sec	Continuous	45	45	-	-

TABLE 8.3-1B

CLASS 1E DIVISION 2 DIESEL GENERATOR LOAD LIST (Cont'd)

LOAD DESCRIPTION	NAME PLATE HP	KW (IN)	SEQUENCE OF OPERATION	DURATION OF OPERATION	APPLICABLE CONDITIONS OF OPERATION UV (KW)	SGAHRS (KW)	DHRS (KW)	TMBDB (KW)
PHTS Na. PUMP PONY MOTOR	75	18.6	20sec	Continuous	18.6	18.6	18.6	-
IHTS Na. PUMP PONY MOTOR	75	18.6	20sec	Continuous	18.6	18.6	18.6	-
ARD BEARING FAN MTR. PRIM	2	2	20sec	Continuous	2	2	2	-
ARD BEARING FAN MTR. INTERM	2	2	20sec	Continuous	2	2	2	-
EMERGENCY PLANT SERVICE WATER PUMP	150	108	20sec	Continuous	108	108	108	108
COOLING TOWER FAN	25	15	20sec	Continuous	15	15	15	15
COOLING TOWER FAN	25	15	20sec	Continuous	15	15	15	15
COOLING TOWER FAN	25	15	20sec	Continuous	15	15	15	15
EMERGENCY PLANT SERVICE WATER MAKEUP PUMP	7.5	5.7	20sec	Continuous	5.7	5.7	5.7	5.7
TOTAL LOAD FOR 10 SEC THROUGH 20 SEC BLOCKS				Continuous	1567	1570	647	475

TABLE 8.3-1B

CLASS 1E DIVISION 2 DIESEL GENERATOR LOAD LIST (Cont'd)

LOAD DESCRIPTION	NAME PLATE HP	KW (IN)	SEQUENCE OF OPERATION	DURATION OF OPERATION	APPLICABLE CONDITIONS OF OPERATION			
					UV (KW)	SGAHRS (KW)	DHRS (KW)	TMBDB (KW)
SWGR A/C Unit "B" (SUPPLY FAN)	25	15.2	1 min	Continuous	15.2	15.2	15.2	15.2
SWGR RETURN FAN "B"	10	6.1	1 min	Continuous	6.1	6.1	6.1	6.1
SG LOOP 2 SUPPLY FAN	200	141	1 min	Continuous	141	141	141	-
SG LOOP 2 SUPPLY FAN	200	141	1 min	Back up	-	-	-	-
SGB-IB SUPPLY FAN	60	37	1 min	Continuous	37	37	37	37
SGB-IB SUPPLY FAN	60	37	1 min	Back up	-	-	-	-
SG LOOP 2 EXH. FAN	60	37	1 min	Continuous	37	37	37	-
SG LOOP 2 EXH. FAN	60	37	1 min	Back Up	-	-	-	-
SGB-IB EXH. FAN	30	20	1 min	Continuous	20	20	20	20
SGB-IB EXH. FAN	30	20	1 min	Back up	-	-	-	-

TABLE 8.3-1B

CLASS 1E DIVISION 2 DIESEL GENERATOR LOAD LIST (Cont'd)

LOAD DESCRIPTION	NAME PLATE HP	KW (IN)	SEQUENCE OF OPERATION	DURATION OF OPERATION	APPLICABLE CONDITIONS OF OPERATION UV (KW)	SGAHRS (KW)	DHRS (KW)	TMBDB (KW)
ABHX CELL UNIT COOLER	3	2.1	1 min	Continuous	2.1	2.1	2.1	2.1
ELEC EQUIPMENT UNIT COOLER	2	1.2	1 min	Continuous	1.2	1.2	1.2	1.2
PRIMARY SODIUM MAKEUP PUMP 'B'	75	63	1 min	Continuous	-	18	63	-
EVST NaK PUMP 'B'	150	117	1 min	Continuous	-	77	117	-
EVST NaK PUMP 'B' BLOWER	15	12.5	1 min	Continuous	-	12.5	12.5	-
EVST AIR BLAST HEAT EXCHANGER 'B'	125	112.5	1 min	Continuous	-	63	112.5	-
TOTAL LOAD FOR 10 SEC THROUGH 1 MIN BLOCKS				Continuous	1827	2000	1211	581

TABLE 8.3-1B

CLASS 1E DIVISION 2 DIESEL GENERATOR LOAD LIST (Cont'd)

LOAD DESCRIPTION	NAME PLATE HP	KW (IN)	SEQUENCE OF OPERATION	DURATION OF OPERATION	APPLICABLE CONDITIONS OF OPERATION UV (KW)	SGAHRS (KW)	DHRS (KW)	TMBDB (KW)
EMERGENCY CHILLED WATER PUMP	75	54	3 min	Continuous	54	54	54	54
TOTAL LOAD FOR 10 SEC THROUGH 3 MIN BLOCKS				Continuous	1881	2054	1266	610
EMERGENCY CHILLER	667KW	469	4 min	Continuous	469	469	469	469
EMERGENCY CHILLER CONTROL CENTER PANEL		5	4 min	Continuous	5	5	5	5
TOTAL LOAD FOR 10 SEC THROUGH 4 MIN BLOCKS				Continuous	2355	2528	1740	1084

TABLE 8.3-1B

CLASS 1E DIVISION 2 DIESEL GENERATOR LOAD LIST (Cont'd)

LOAD DESCRIPTION	NAME PLATE HP	KW (IN)	SEQUENCE OF OPERATION	DURATION OF OPERATION	APPLICABLE CONDITIONS OF OPERATION UV (KW)	SGAHRS (KW)	DHRS (KW)	TMBDB (KW)
EMERG. CHILLER ROOM UNIT COOLER	0.75	0.53	5 min	Continuous	0.53	0.53	0.53	0.53
FAN MOTOR	60	36	5 min	Continuous	36	36	36	36
FAN MOTOR	60	38	5 min	Continuous	38	38	38	38
EVST SODIUM PUMP 'B'	75	63	5 min	Continuous	-	63	63	-
TOTAL LOAD FOR 10 SEC THROUGH 5 MIN BLOCKS				Continuous	2430	2666	1878	1159
REFUEL COMM CHILLER A/C UNIT	3	2.3	manual	Continuous	2.3	2.3	2.3	2.3

TABLE 8.3-1B

CLASS 1E DIVISION 2 DIESEL GENERATOR LOAD LIST (Cont'd)

LOAD DESCRIPTION	NAME PLATE HP	KW (IN)	SEQUENCE OF OPERATION	DURATION OF OPERATION	APPLICABLE CONDITIONS OF OPERATION UV (KW)	SGAHRS (KW)	DHRS (KW)	TMBDB (KW)
ANNULUS COOLING FAN	300	215.4	manual	Continuous	-	-	-	215.4
ANNULUS COOLING FAN	300	215.4	manual	Continuous	-	-	-	215.4
ANNULUS COOLING FAN	300	215.4	manual	Continuous	-	-	-	215.4
CONT. CLEAN-UP SCRB EXH. FAN	250	154	manual	Continuous	-	-	-	154
CLEAN-UP FILTER HEATER		171	manual	Continuous	171	171	171	171
CLEAN-UP FILTER FAN	40	22.6	manual	Continuous	22.6	22.6	22.6	22.6
ANNULUS FILTER UNIT COOLER	1	0.8	manual	Continuous	0.8	0.8	0.8	0.8
CLEAN-UP FILTER UNIT COOLER	7.5	5.3	manual	Continuous	5.3	5.3	5.3	5.3
CLEAN-UP PUMP CELL UNIT COOLER	3	1.9	manual	Continuous	-	-	-	1.9
CLEAN-UP SCRUBBER CELL UNIT COOLER	2	1.6	manual	Continuous	-	-	-	1.6
CLEAN-UP CHASE UNIT COOLER	0.5	0.4	manual	Continuous	-	-	-	0.4

TABLE 8.3-1B

CLASS 1E DIVISION 2 DIESEL GENERATOR LOAD LIST (Cont'd)

LOAD DESCRIPTION	NAME PLATE HP	KW (IN)	SEQUENCE OF OPERATION	DURATION OF OPERATION	APPLICABLE CONDITIONS OF OPERATION UV (KW)	SGAHRS (KW)	DHRS (KW)	TMBDB (KW)
SEISMICALLY QUAL. FIRE PUMP CONTROLLER	25	13.1	manual	Continuous	13.1	13.1	13.1	13.1
EARTHQUAKE MONITORING INST.		.5	manual	Continuous	.5	.5	.5	-0
PANEL LEAK MONITOR	3.86KW	2.57	manual	Continuous	2.57	2.57	2.57	2.57
PANEL LEAK MONITOR	2.72KW	1.84	manual	Continuous	1.84	1.84	1.84	1.84
PANEL LEAK MONITOR	2.61KW	1.73	manual	Continuous	1.73	1.73	1.73	1.73
CONTROL PANEL		0.15	manual	Continuous	.15	.15	.15	.15
TMBDB CIRCULATING WATER PUMP (HOLD 75001590)	300	220	manual	Continuous	-	-	-	220
TOTAL LOAD				Continuous	2649	2885	2097	2402
FUTURE LOAD (15% GROWTH)				Continuous	397	433	315	360
TOTAL LOAD INCLUDING FUTURE LOAD				Continuous	3046	3318	2412	2762

TABLE 8.3-1B

CLASS 1E DIVISION 2 DIESEL GENERATOR LOAD SUMMARY

LOAD SUMMARY

LOAD BLOCK	TIME ON	BUS UNDERVOLTAGE		SGAHRS		DHRS		TMBDB	
		LOAD KW	CUMUL. KW	LOAD KW	CUMUL. KW	LOAD KW	CUMUL. KW	LOAD KW	CUMUL. KW
1	10 sec	1111	1111	1114	1114	281	281	161	161
2	20 sec	456	1567	456	1570	366	647	314	475
3	1 min	260	1827	430	2000	564	1211	106	581
4	3 min	54	1881	54	2054	55	1266	29	610
5	4 min	474	2355	474	2528	474	1740	474	1084
6	5 min	75	2430	138	2666	138	1878	75	1159
	Manual	219	2649	219	2885	219	2097	1243	2402

TABLE 8.3-1C

CLASS 1E DIVISION 3 DIESEL GENERATOR LOAD LIST

THE MACHINE SHALL ON RECEIPT OF START SIGNAL, (TIME 0), START AND ATTAIN RATED SPEED AND VOLTAGE WITHIN 10 SECONDS, AND ACCEPT LOADS IN ANY OF THE SEQUENCES STATED BELOW.

LOAD DESCRIPTION	NAME PLATE HP	KW (IN)	SEQUENCE OF OPERATION	DURATION OF OPERATION	APPLICABLE CONDITIONS OF OPERATION UV (KW)	SGAHRS (KW)	DHRS (KW)	TMBDB (KW)
BATTERY CHARGER (130VDC)		115	10 sec	Continuous	115	115	115	_
BATTERY CHARGER (130VDC)		115	10 sec	Back up	_	_	_	_
LOADS REQUIRED TO SUPPORT OPERATION OF DIVISION 3 DIESEL GENERATION UNIT		Later	10 sec	Continuous				
AIR COOLED CONDENSER BLOWER MOTOR	60	45	10 sec	Continuous	45	45	-	-
AIR COOLED CONDENSER BLOWER MOTOR	60	45	10 sec	Continuous	45	45	-	-
PHTS Na. PUMP PONY MOTOR	75	18.6	10 sec	Continuous	18.6	18.6	18.6	-
IHTS Na. PUMP PONY MOTOR	75	18.6	10 sec	Continuous	18.6	18.6	18.6	-
ARD BEARING FAN MTR. PRIM	2	2	10 sec	Continuous	2	2	2	-
ARD BEARING FAN MTR. INTERM	2	2	10 sec	Continuous	2	2	2	-

TABLE 8.3-1C

CLASS 1E DIVISION 3 DIESEL GENERATOR LOAD LIST (Cont'd)

LOAD DESCRIPTION	NAME PLATE HP	KW (IN)	SEQUENCE OF OPERATION	DURATION OF OPERATION	APPLICABLE CONDITIONS OF OPERATION UV (KW)	SGAHRS (KW)	DHRS (KW)	TMBDB (KW)
ACS ISOLATION VALVES (3 Valves Each 1/2 hp)	1.5	1.2	10 sec	Intermittent	1.2	1.2	1.2	1.2
ACS ISOLATION VALVES (3 Valves Each 1/2 hp)	1.5	1.2	10 sec	Backup	-	-	-	-
ACS CONTROL POWER		.3	10 sec	Intermittent	.3	.3	.3	.3
ACS CONTROL POWER		.3	10 sec	Back Up	-	-	-	-
FHC SECONDARY COOLANT PUMP A	15	10	10 sec	Continuous	10	10	10	10
FHC SECONDARY COOLANT PUMP B	15	10	10 sec	Continuous	10	10	10	10
FHC ARGON COOLING BLOWER	50	41	10 sec	Continuous	41	41	41	41
FHC ARGON COOLING BLOWER	50	41	10 sec	Back up	-	-	-	-
STARTING AIR COMPRESSOR	30	25	Later	Inter.	25	25	25	-
STARTING AIR COMPRESSOR	30	25	Later	Backup	-	-	-	-

TABLE 8.3-1C

CLASS 1E DIVISION 3 DIESEL GENERATOR LOAD LIST (Cont'd)

LOAD DESCRIPTION	NAME PLATE HP	KW (IN)	SEQUENCE OF OPERATION	DURATION OF OPERATION	APPLICABLE CONDITIONS OF OPERATION UV (KW)	SGAHRS (KW)	DHRS (KW)	TMBDB (KW)
STARTING AIR COMPRESSOR	30	25	Later	Inter	25	25	25	-
STARTING AIR COMPRESSOR	30	25	Later	Back up	-	-	-	-
STARTING AIR COMPRESSOR	30	25	Later	Inter	25	25	25	-
STARTING AIR COMPRESSOR	30	25	Later	Back up	-	-	-	
VACUUM PUMP	3	3.0	Later	Continuous	3.0	3.0	3.0	3.0
VACUUM PUMP	3	3.0	Later	Back up	-	-	-	-
VACUUM PUMP	3	3.0	Later	Continuous	3.0	3.0	3.0	3.0
VACUUM PUMP	3	3.0	Later	Back up	-	-	-	-
VACUUM PUMP	3	3.0	Later	Continuous	3.0	3.0	3.0	3.0
VACUUM PUMP	3	3.0	Later	Back up	-	-	-	-
VACCUM PUMP	3	3.0	Later	Continuous	3.0	3.0	3.0	3.0
VACUUM PUMP	3	3.0	Later	Back up	-	-	-	-

TABLE 8.3-1C

CLASS 1E DIVISION 3 DIESEL GENERATOR LOAD LIST (Cont'd)

LOAD DESCRIPTION	NAME PLATE HP	KW (IN)	SEQUENCE OF OPERATION	DURATION OF OPERATION	APPLICABLE CONDITIONS OF OPERATION UV (KW)	SGAHRS (KW)	DHRS (KW)	TMBDB (KW)
VACUUM PUMP	3	3.0	Later	Continuous	3.0	3.0	3.0	3.0
VACUUM PUMP	3	3.0	Later	Back up	-	-	-	-
VACCUM PUMP	3	3.0	Later	Continuous	3.0	3.0	3.0	3.0
VACUUM PUMP	3	3.0	Later	Back up	-	-	-	-
VACUUM PUMP	3	3.0	Later	Continuous	3.0	3.0	3.0	3.0
VACUUM PUMP	3	3.0	Later	Back up	-	-	-	-
TURNING GEAR MOTOR	20	17.2	Later	Continuous	17.2	17.2	17.2	17.2
TURNING GEAR OIL PUMP	50	42.53	Later	Continuous	42.53	42.53	42.53	42.53
PIGGY BACK MOTOR	7.5	7.0	Later	Continuous	7.0	7.0	7.0	7.0

TABLE 8.3-1C

CLASS 1E DIVISION 3 DIESEL GENERATOR LOAD LIST (Cont'd)

LOAD DESCRIPTION	NAME PLATE HP	KW (IN)	SEQUENCE OF OPERATION	DURATION OF OPERATION	APPLICABLE CONDITIONS OF OPERATION UV (KW)	SGAHRS (KW)	DHRS (KW)	TMBDB (KW)
MISCELLANEOUS UPS SYSTEM		82	Later	Continuous	82	82	82	82
MISCELLANEOUS UPS SYSTEM		82	Later	Continuous	82	82	82	82
BATTERY CHARGERS (48VDC) (2@ 5.5 KW)		11	Later	Continuous	11	11	11	-
BATTERY CHARGERS (130VDC) (2@ 4.4 KW)		8.8	Later	Continuous	8.8	8.8	8.8	-
BATTERY CHARGERS (130VDC) (4@ 150 KW)		600	Later	Continuous	600	600	600	-
BATTERY CHARGERS (130VDC) (2@ 150 KW)		300	Later	Back up	-	-	-	-
BATTERY CHARGERS (260VDC)		112	Later	Continuous	112	112	112	-
BATTERY CHARGERS (260VDC)		112	Later	Back up	-	-	-	-
TOTAL LOAD				Continuous	1367	1367	1277	314
FUTURE LOAD (15% GROWTH)				Continuous	205	205	192	47
TOTAL LOAD INCLUDING FUTURE LOAD				Continuous	1572	1572	1469	361

TABLE 8.3-1D

DIESEL GENERATOR LOAD LIST NOTES

LOAD DESCRIPTION	NAME PLATE HP	KW (IN)	SEQUENCE OF OPERATION	DURATION OF OPERATION	APPLICABLE CONDITIONS OF OPERATION UV (KW)	SGAHRS (KW)	DHRS (KW)	TMBDB (KW)

NOTES: 1. Future Load Growth of 15% is proportionally distributed to each load block.

2. For all motors assume full voltage starting current to be 650% of motor's full load current.

3. Motor starting power factor and running power factor will be assumed as list below:

HORSEPOWER	STARTING POWER FACTOR	RUNNING POWER FACTOR
5 HP and Below	.62	.79
7.5 HP - 15 HP	.54	.79
20 HP - 30 HP	.45	.79
40 HP - 60 HP	.37	.81
75 HP - 125 HP	.32	.85
150 HP and 200 HP	.26	.85
250 HP - 350 HP	.22	.85
400 HP - 500 HP	.19	.85
600 HP and Above	.17	.85
900 HP and Above	.16	.85

NOTES: 4. The 1050 HP SGAHRS auxillary feed water pump drive motor will be started automatically at 10 sec. However, while the Diesel Generator is fully loaded this pump could be tripped and restarted.

5. At 9 seconds the magnetizing inrush current associated with two 1500 KVA (4160/480V, DELTA-WYE, Z=5.75%) and one 1000 KVA (4160/480V, DELTA-WYE, Z=5.75%) substation transformers will be imposed on each Division 1 and 2 Diesel Generator unit.

TABLE 8.3-2A

CLASS 1E DIVISION 1 125V DC LOAD LIST

LOAD DESCRIPTION	NORMAL (1) MAX. CONT. LOAD-AMPS.	EMERGENCY (2) AMPS.	EMERGENCY (2) DURATION	REMARKS
4.16 KV SWGR. AND 480 V USS BREAKER LOAD	22	158.8 22	FIRST 1 MIN. NEXT 119 MIN.	
D.G CONTROL PANELS LOCAL AND MCR (0.7 KW)	11.2	11.2	0-120 MIN.	
D.G FIELD FLASHING (7.5 KW)	-	60	FIRST 1 MIN.	
D.G. GOV. OIL BOOSTER PUMP, 1 HP	-	40 10	FIRST 1 MIN. NEXT 119 MIN.	
PHTS 1 DR PPS BREAKERS	–	12 68	FIRST 1 MIN. LAST 1 MIN.	
PHTS 2 DR PPS BREAKERS	–	12	FIRST 1 MIN.	
PHTS 3 DR PPS BREAKERS	–	12	FIRST 1 MIN.	
PHTS 1 DR PPS TEST BREAKERS	–	-	-	
PHTS 2 DR PPS TEST BREAKERS	–	-	-	
PHTS 3 DR PPS TEST BREAKERS	–	-	-	
IHTS 1 DR PPS BREAKERS	–	12	FIRST 1 MIN	
IHTS 2 DR PPS BREAKERS	–	12	FIRST 1 MIN	
IHTS 3 DR PPS BREAKERS	–	12	FIRST 1 MIN.	

TABLE 8.3-2A

CLASS 1E DIVISION 1 125V DC LOAD LIST (Cont'd)

LOAD DESCRIPTION	NORMAL (1) MAX. CONT. LOAD-AMPS.	EMERGENCY (2) AMPS.	DURATION	REMARKS
IHTS 1 DR PPS TEST BREAKERS	-	-	-	
IHTS 2 DR PPS TEST BREAKERS	-	-	-	
IHTS 3 DR PPS TEST BREAKERS	-	-	-	
CIS BREAKER PANEL 1 (0.15 KW)	1.2	1.2	0-120 MIN.	
120 VAC BUS 12N1E008A AS INVERTER LOAD	391	429 393 359	FIRST 1 MIN. NEXT 14 MIN. NEXT 105 MIN.	NOTE (3)
TOTAL IN AMPS	425	772 437 403 471	FIRST 1 MIN. NEXT 14 MIN. NEXT 104 MIN. LAST 1 MIN.	

TABLE 8.3-2B

CLASS 1E DIVISION 2 125V DC LOAD LIST

LOAD DESCRIPTION	NORMAL (1) MAX. CONT. LOAD-AMPS.	EMERGENCY (2) AMPS.	DURATION	REMARKS
4.16 KV SWGR. AND 480 V USS BREAKER LOAD	20	136.8 20	FIRST 1 MIN. NEXT 119 MIN.	
D.G CONTROL PANELS LOCAL AND MCR (0.7 KW)	11.2	11.2	0-120 MIN.	
D.G FIELD FLASHING (7.5 KW)	-	60	FIRST 1 MIN.	
D.G. GOV. OIL BOOSTER PUMP, 1 HP	-	40 10	FIRST 1 MIN. NEXT 119 MIN.	
PHTS 1 DR PPS BREAKERS	–	12 68	FIRST 1 MIN. LAST 1 MIN.	
PHTS 2 DR PPS BREAKERS	–	12	FIRST 1 MIN.	
PHTS 3 DR PPS BREAKERS	–	12	FIRST 1 MIN.	
PHTS 1 DR PPS TEST BREAKERS	–	-	-	
PHTS 2 DR PPS TEST BREAKERS	–	-	-	
PHTS 3 DR PPS TEST BREAKERS	–	-	-	
IHTS 1 DR PPS BREAKERS	–	12	FIRST 1 MIN	
IHTS 2 DR PPS BREAKERS	–	12	FIRST 1 MIN	
IHTS 3 DR PPS BREAKERS	–	12	FIRST 1 MIN.	

TABLE 8.3-2B

CLASS 1E DIVISION 2 125V DC LOAD LIST (Cont'd)

LOAD DESCRIPTION	NORMAL (1) MAX. CONT. LOAD-AMPS.	EMERGENCY (2) AMPS.	DURATION	REMARKS
IHTS 1 DR PPS TEST BREAKERS	-	-	-	
IHTS 2 DR PPS TEST BREAKERS	-	-	-	
IHTS 3 DR PPS TEST BREAKERS	-	-	-	
CIS BREAKER PANEL 1 (0.15 KW)	1.2	1.2	0-120 MIN.	
120 VAC BUS 12N1E008A AS INVERTER LOAD	382	420 385 354	FIRST 1 MIN. NEXT 14 MIN. NEXT 105 MIN.	NOTE (3)
TOTAL IN AMPS	414	741 427 396 464	FIRST 1 MIN. NEXT 14 MIN. NEXT 104 MIN. LAST 1 MIN.	

TABLE 8.3-2C
CLASS 1E DIVISION 3 125V DC LOAD LIST

LOAD DESCRIPTION	NORMAL (1) MAX. CONT. LOAD-AMPS	EMERGENCY (2) AMPS	DURATION	REMARKS
SGAHRS - STEAM TURBINE GOVERNOR CONTROL (1 KW)	4	4	0-120 MIN.	
120 VAC BUS 12NIE008C AS INVERTER LOAD	208	227 182 165	FIRST 1 MIN. NEXT 14 MIN. NEXT 105 MIN.	(NOTE 3)
D.G. CONTROL PANELS LOCAL AND MCR (0.7KW)	11.2	11.2	0-120 MIN.	
D.G. FIELD FLASHING (7.5KW)	-	60	FIRST 1 MIN.	
4.16kV SWITCHGEAR AND 480V USS BREAKER LOAD	4	38 4	FIRST 1 MIN. NEXT 119 MIN.	
TOTAL IN AMPS	227	340 204 184	FIRST 1 MIN. NEXT 14 MIN. NEXT 105 MIN.	

TABLE 8.3-2D

DC LOAD LIST NOTES

(1) Normal - (Maximum continuous load) plant normal operating load with power supply either from the battery or through the battery charger.

(2) Emergency - DC bus load during the emergency operation for specific time duration following a complete loss of AC system and with the battery supplying power to the load.

(3) The inverter connected to DC bus as load is considered the actual KW load and is converted to inverter input D.C. current as per "Application of Exide Static Uninterruptible AC Power Systems" Section 56.00a, Exide Bulletin 212, April 1971.

$$\text{DC Input Current} = \frac{1000\ (\text{Load KVA})\ (\text{PF})}{(\text{Inv. Efficiency})\ (1.75\ \text{V/C})\ (\text{No. of Cells})}$$

$$= \frac{1000\ (\text{Load KW})}{(0.8)\ (1.75\ \text{V/C})\ (\text{No. of Cells})}$$

TABLE 8.3-3

COMPONENT ARRANGEMENT CRITERIA

Equipment	Design Basis Event Considered	Applicable Criteria for Arrangement or Design
Diesel generator units	Tornado-wind, missiles and earthquake	The diesel generator units will be located in the diesel generator building which is Seismic Category I. The structure is designed to protect the diesel generator units from any credible tornado wind, missile and earthquake damage. The criteria applied to Category I structures associated with the design tornado are given in Chapter 3.
	Flood (natural)	The diesel generator units will be located above design flood level given in Section 3.4.
	Temperature and humidity	Proper ambient temperature or humidity is maintained in the Diesel Generator rooms as described in Section 9.6.5.
	Fires	The diesel generator units will be located in individual rooms, within the diesel generator building, separated by concrete wall barriers. The rooms will be individually protected by fire extinguishing systems as described in Section 9.13.1

TABLE 8.3-3 (Continued)

Equipment	Design Basis Event Considered	Applicable Criteria for Arrrangement or Design
Diesel generator units	Accident-generated missiles	The concrete wall barriers between the diesel generators will be designed to withstand any credible missile generated by a diesel generator casualty. The location and design of the diesel generator building will protect against any credible accident generated external missile.
	Fire protection system operation	The fire protection piping located in the individual diesel generator rooms is Seismic Category I. Each diesel is individually protected so that a fire in one diesel generator room will not actuate the components of the fire protection system in any other space.
	Accident-generated floods, sprays, or jets	The Diesel Generator Building is located away from any credible piping, and will be designed to protect the diesels from the worst credible accident-generated flood.

TABLE 8.3-3 (Continued)

Equipment	Design Basis Event Considered	Applicable Criteria for Arrrangement or Design
<u>4.16 KV AC Power Distribution System</u>		
a) 4.16 KV Class 1E Switchgear b) 4.16 KV/480 V Class 1E Unit Substations (including 480V Switchgear)	Tornado wind missiles, earthquake	The 4.16 KV Class 1E system is located in the Diesel Generating Building, Reactor Service Building and the Steam Generator Building. These buildings are Seismic Category I structures, designed to protect against any credible wind-tornado damage and earthquake.
c) Cable	Floods (natural)	The above buildings will be protected against design flood level.
	Temperature and humidity	Proper ambient temperature or humidity is maintained in the Class 1E switchgear room as described in Section 9.6.5.
	Fire	A description of the non-sodium fire and sodium fire protection and detection system for Category I structures is given in Sections 9.13.1 and 9.13.2.

TABLE 8.3-3 (Continued)

Equipment	Design Basis Event Considered	Applicable Criteria for Arrrangement or Design
		Category I structures are designed to withstand externally generated missiles. Physical separation of components in different power divisions will be such that no credible internal accident-generated missile will disable more than one safety division.
	Accident-generated floods, sprays, or jets	The arrangement of Class 1E systems will be such that no credible accident-generated spray or jet will disable more than one power division. The arrangement will also provide protection against the maximum possible accident-generated flood.
480 V Class 1E AC Power Distribution System a) 480 V Motor Control Center b) Containment Electrical Penetrations c) Cables	Tornado wind, earthquake	The 480 V Class 1E MCCs are located in Diesel Generator Building, Reactor Service Building, Steam Generator Building, Control Building and Emergency Cooling Towers. All of these buildings are Seismic Category I structures, designed to protect against any credible wind-tornado damage and earthquake.

TABLE 8.3-3 (Continued)

Equipment	Design Basis Event Considered	Applicable Criteria for Arrrangement or Design
	Floods (natural), temperature, humidity, fire and Accident-generated missiles, floods, sprays, or jets.	Identical with the 4.16 KV System.
Class 1E DC & Vital Plant Power Distribution System		
a) 125V DC Battery Chargers	Tornado wind, earthquake	Identical with the 480 V system.
b) 125V DC Batteries	Floods	Identical with the 4.16 KV system.
c) 125V DC Switchgear	Temperature-humidity	Identical with the 4.16 KV system.
d) 480-120/208V AC Transformers		
e) 120/208 AC Inverters		
f) 120/208 Vital AC Distribution Panels		
g) Cables	Fire	Identical with the 4.16 KV system.
	Accident-generated missiles	Each DC battery is separated from all others by a concrete wall compartment. The compartment will be ventilated to dissipate hydrogen given off by the battery in order to preclude a fire or explosion. The battery will also be separated from its associated DC and UPS equipment.
	Accident-generated floods, sprays, or jets	Identical with the 4.16KV system.

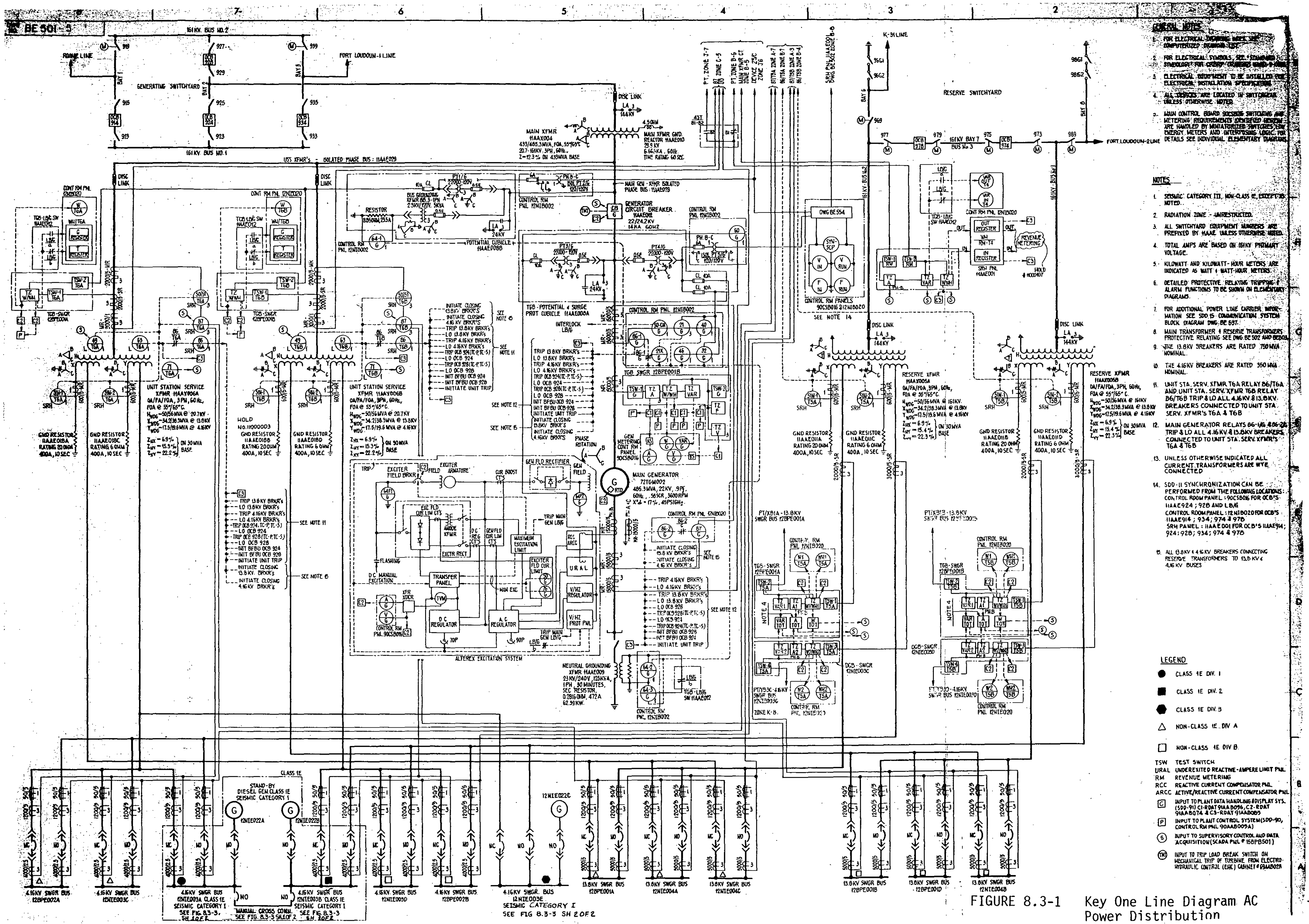

FIGURE 8.3-1 Key One Line Diagram AC Power Distribution

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(BE501-5)

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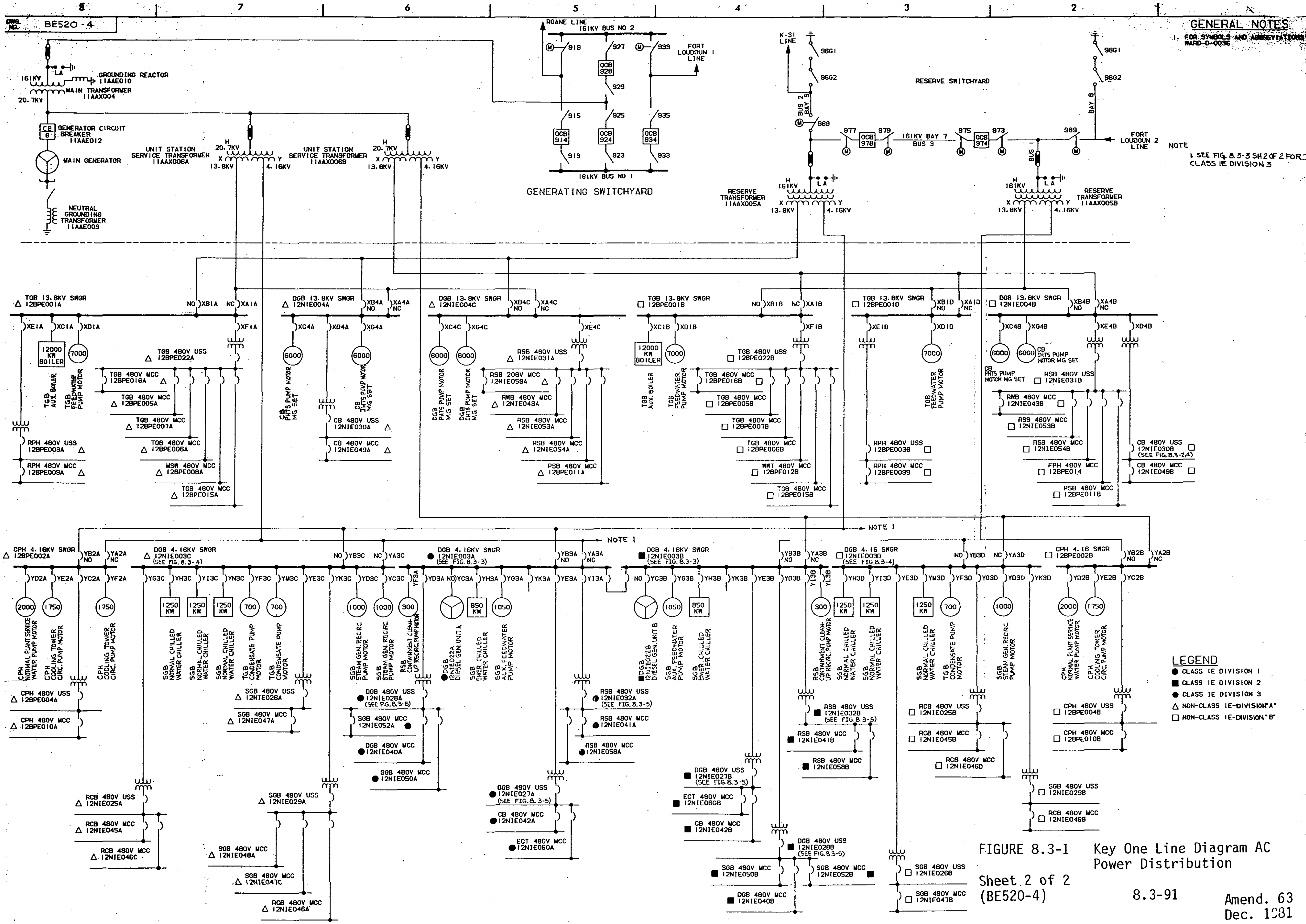

FIGURE 8.3-1 Key One Line Diagram AC Power Distribution

Sheet 2 of 2 (BE520-4)

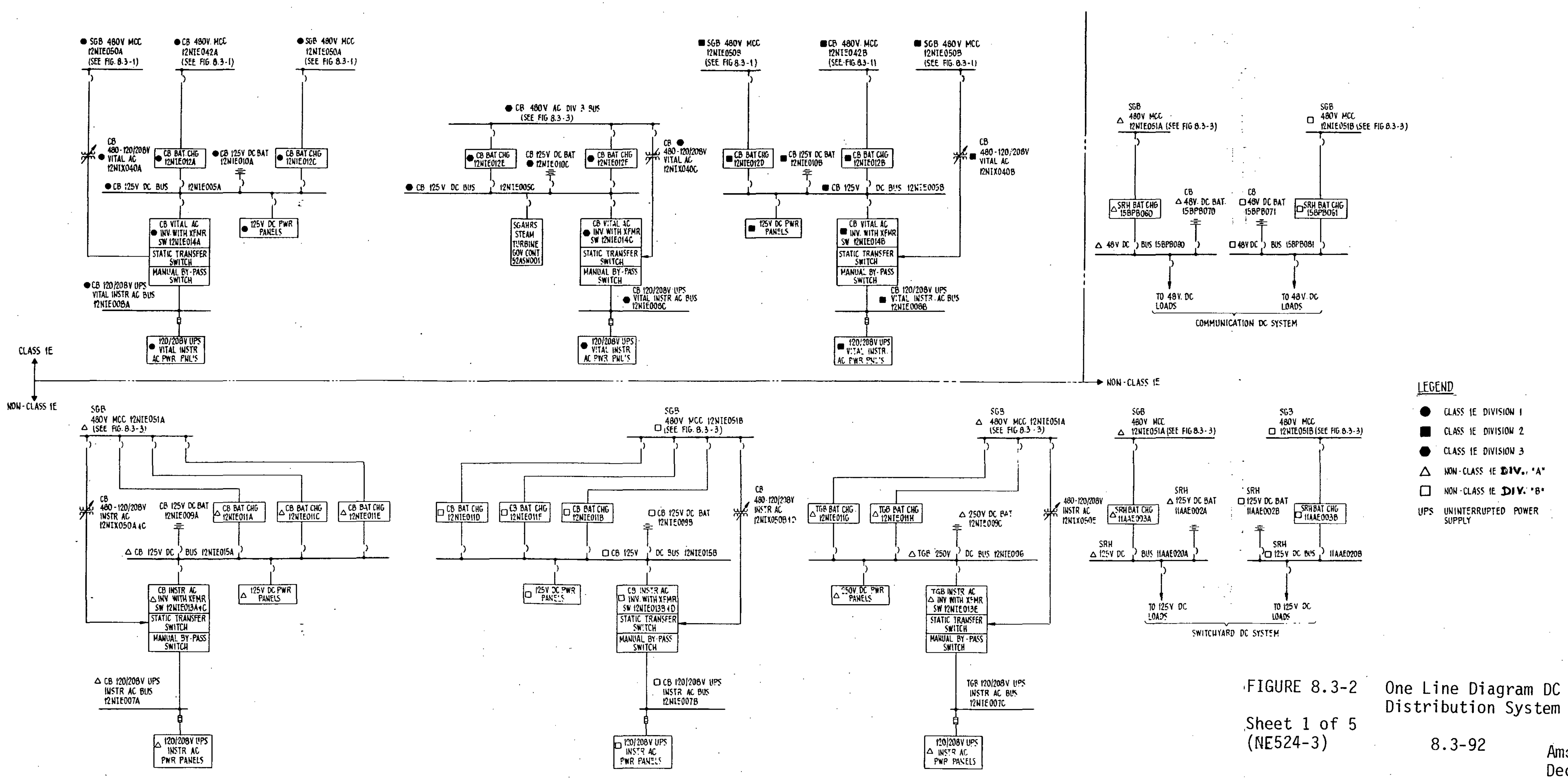

FIGURE 8.3-2 One Line Diagram DC Distribution System

Sheet 1 of 5
(NE524-3)

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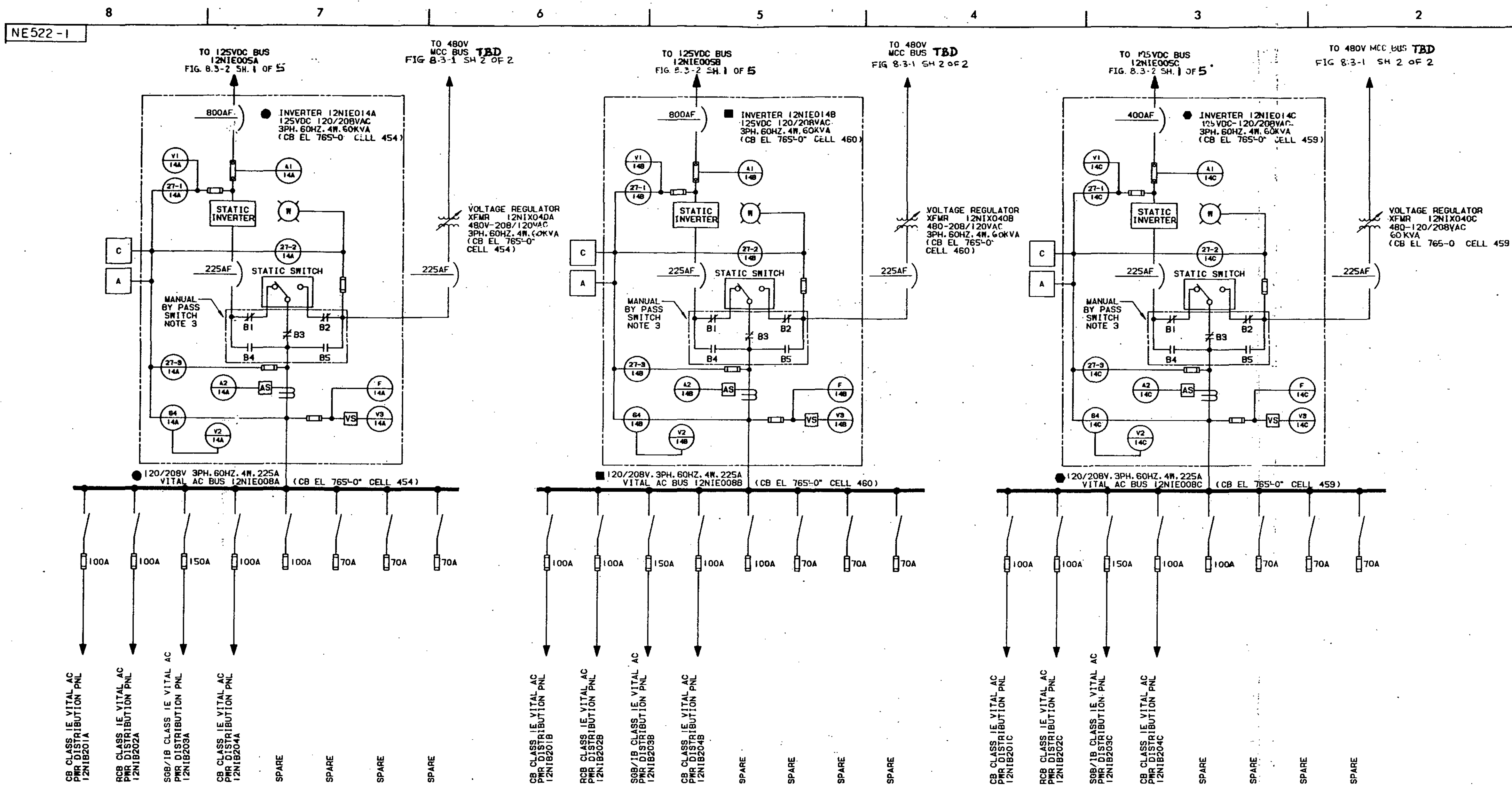

GENERAL NOTES

1. SYMBOLS AND ABBREVIATIONS WARD DOC D-0036
2. ALL DC BREAKERS ARE 2 POLE DEVICES AND AC BREAKERS ARE 3 POLE DEVICES UOS
3. MANUAL BY PASS SWITCH SHALL HAVE MAKE BEFORE BREAK CONTACTS
4. FUSE PANEL BOARDS HAVE FUSES SUITABLE FOR INTERRUPTING 10.000A SYMMETRICAL
5. UNINTERRUPTIBLE POWER SUPPLY IS UNGROUNDED
6. FUSES TO BE PROPERLY CO-ORDINATED WITH FUSES PROVIDED WITH STATIC TRANSFER SWITCH AND AC BREAKERS

NOTES

1. RADIATION ZONE: UNRESTRICTED UOS
2. CLASS IE. SEISMIC CATEGORY: I UOS

LEGEND

● - CLASS IE DIVISION 1
■ - CLASS IE DIVISION 2
● - CLASS IE DIVISION 3

MANUAL BY-PASS SWITCH CONTACT DEVELOPMENT

POSITION	CONTACTS				
	B1	B2	B3	B4	B5
REGULATED LINE					X
*	X	X			X
TEST	X	X			X
*	X	X	X		X
NORMAL	X	X	X		
*	X	X	X	X	
TEST	X	X		X	
*	X	X		X	
INVERTER				X	

* - INTERMEDIATE POSITION

FIGURE 8.3-2 One Line Diagram DC Distribution System

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(NE522-1)

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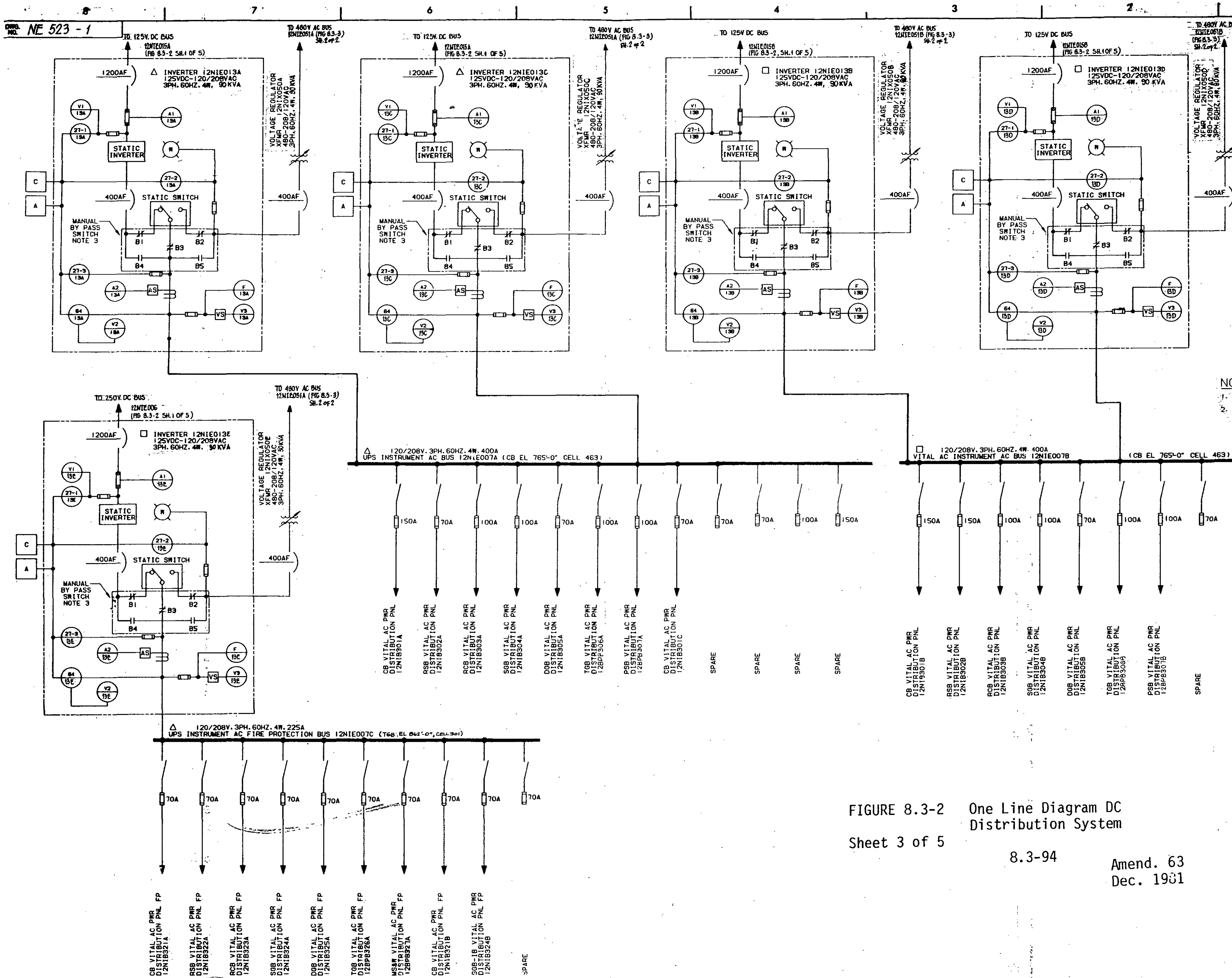
DWG. NO. NE 523 - 1



GENERAL NOTES

1. SYMBOLS AND ABBREVIATIONS HARD DOC D-0036

NOTES

1. RADIATION ZONE: UNRESTRICTED UOS
2. NON-CLASS IE SEISMIC CATEGORY UOS

LEGEND

△ - NON CLASS IE DIVISION A
□ - NON CLASS IE DIVISION B

MANUAL BY-PASS SWITCH CONTACT DEVELOPMENT

POSITION	CONTACTS B1	B2	B3	B4	B5
REGULATED LINE					X
*	X	X			X
TEST	X	X			X
*	X	X	X		X
NORMAL	X	X	X		
*	X	X	X	X	
TEST	X	X		X	
*	X	X		X	
INVERTER				X	

* - INTERMEDIATE POSITION

FIGURE 8.3-2 One Line Diagram DC Distribution System

Sheet 3 of 5

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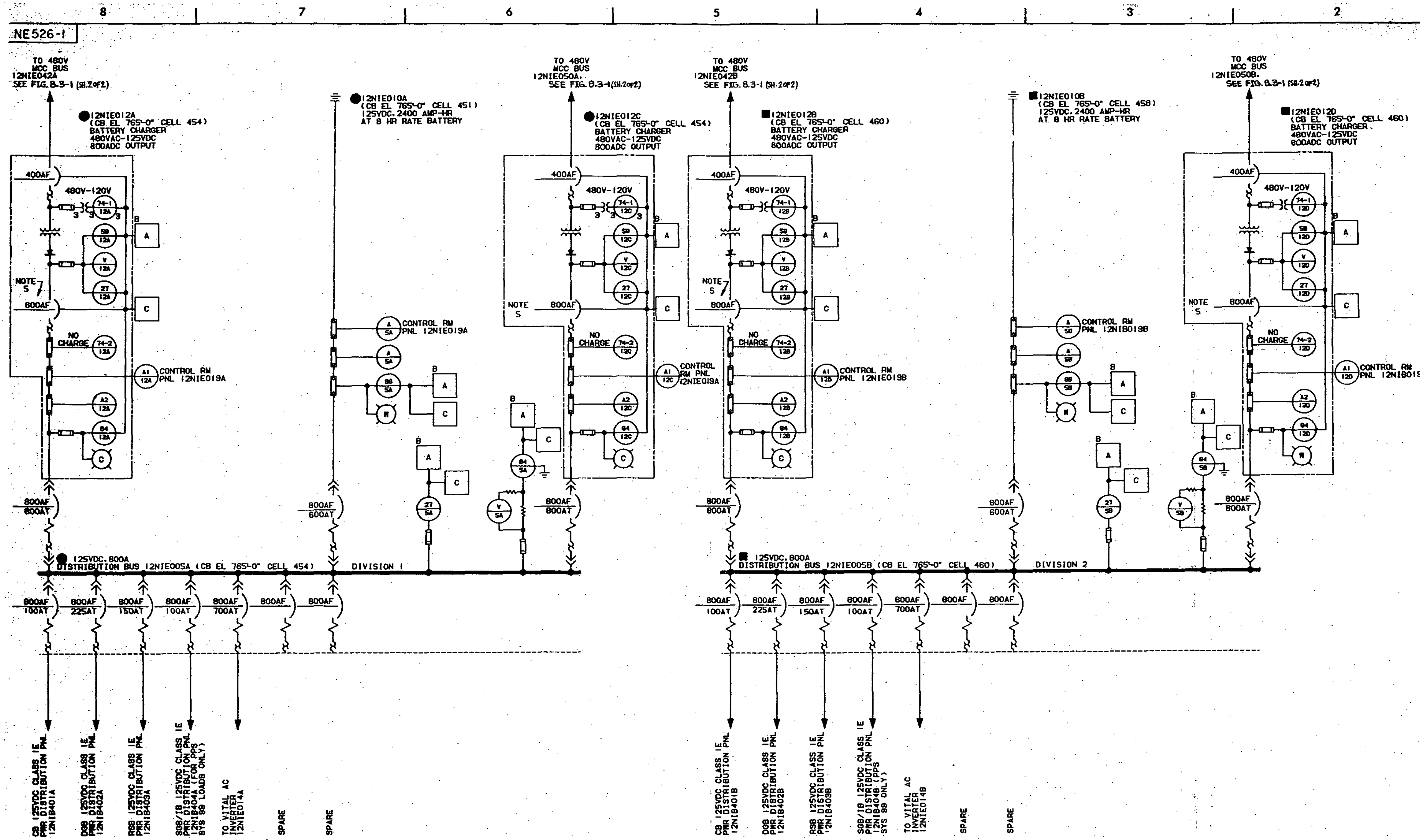

GENERAL NOTES

1. SYMBOLS AND ABBREVIATIONS WARD DOC D-0036
2. ALL DC BREAKERS ARE 2 POLE DEVICES AND AC BREAKERS ARE 3 POLE DEVICES UOS
3. FRAME SIZE AND TRIP RATING IN AMPERES ARE INDICATED ADJACENT TO SYMBOLS FOR AIR CURRENT BREAKERS
4. ACTUAL BATTERY AND BATTERY CHARGER RATINGS TO BE DETERMINED BY MANUFACTURER. INDICATED RATINGS ARE PURCHASERS CALCULATED VALUES
5. BATTERY CHARGER DC OUTPUT BREAKERS SHALL BE FURNISHED WITH REVERSE CURRENT PROTECTION

NOTES

1. RADIATION ZONE: UNRESTRICTED UOS
2. CLASS IE. SEISMIC CATEGORY: I UOS

LEGEND

● - CLASS IE DIVISION 1
■ - CLASS IE DIVISION 2

96 - HIGH DISCHARGE RELAY
PPS - PLANT PROTECTION SYSTEM

FIGURE 8.3-2 One Line Diagram DC Distribution System

Sheet 4 of 5
(NE526-1)

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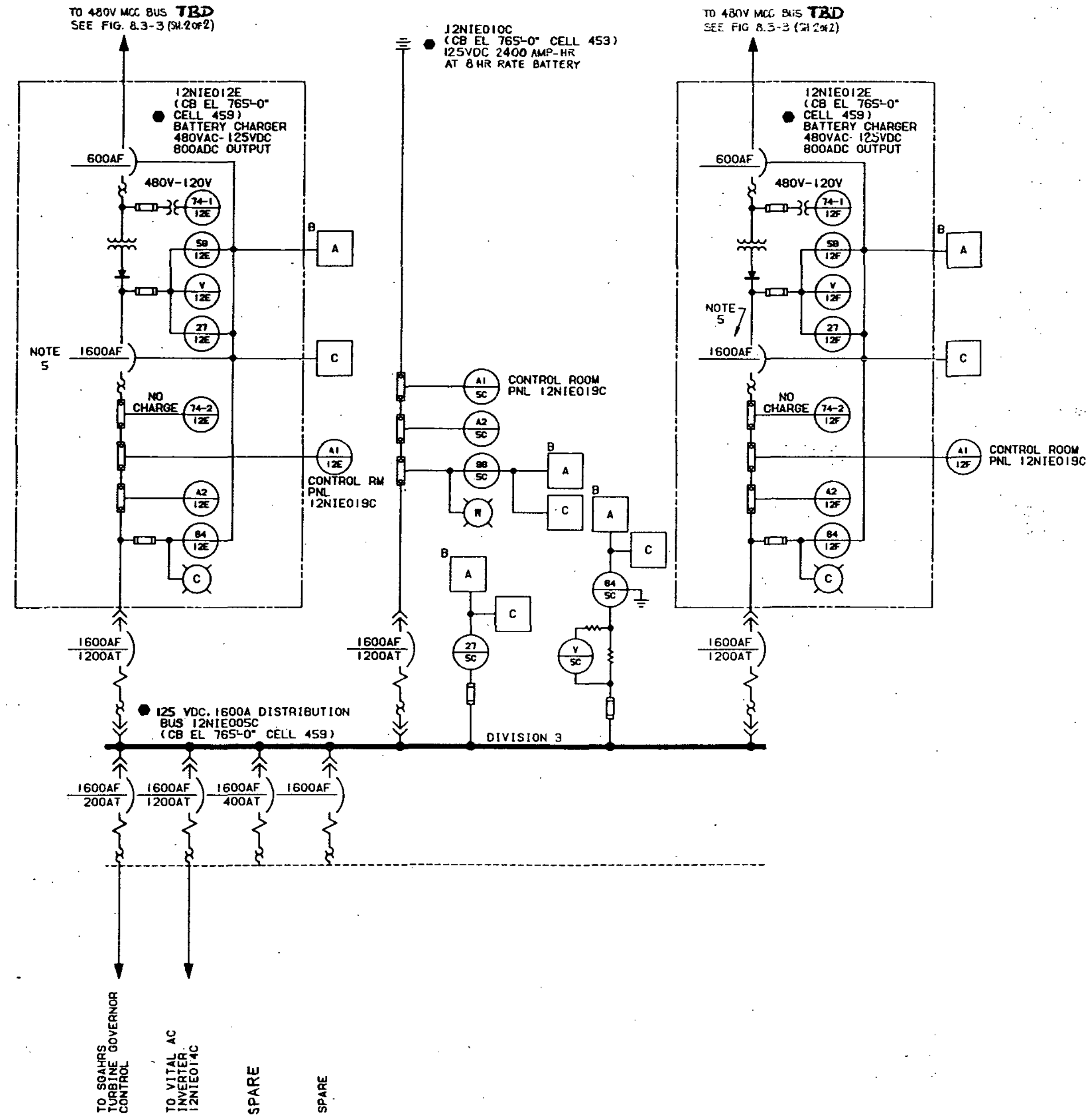
NE527-O



GENERAL NOTES

1. SYMBOLS AND ABBREVIATIONS WARD DOC D-003S 1
2. ALL DC BREAKERS ARE 2 POLE DEVICES AND AC BREAKERS ARE 3 POLE DEVICES UOS
3. FRAME SIZE AND TRIP RATING IN AMPERES ARE INDICATED ADJACENT TO SYMBOLS FOR AIR CURRENT BREAKERS
4. ACTUAL BATTERY AND BATTERY CHARGER RATINGS TO BE DETERMINED BY MANUFACTURER. INDICATED RATINGS ARE PURCHASERS CALCULATED VALUES
5. BATTERY CHARGER DC OUTPUT BREAKERS SHALL BE FURNISHED WITH REVERSE CURRENT PROTECTION

NOTES

1. RADIATION ZONE: UNRESTRICTED UOS
2. CLASS IE. SEISMIC CATEGORY: I UOS

LEGEND

● - CLASS IE DIVISION 3
96 - HIGH DISCHARGE RELAY

FIGURE 8.3-2 One Line Diagram DC Distribution System

Sheet 5 of 5
(NE527-0)

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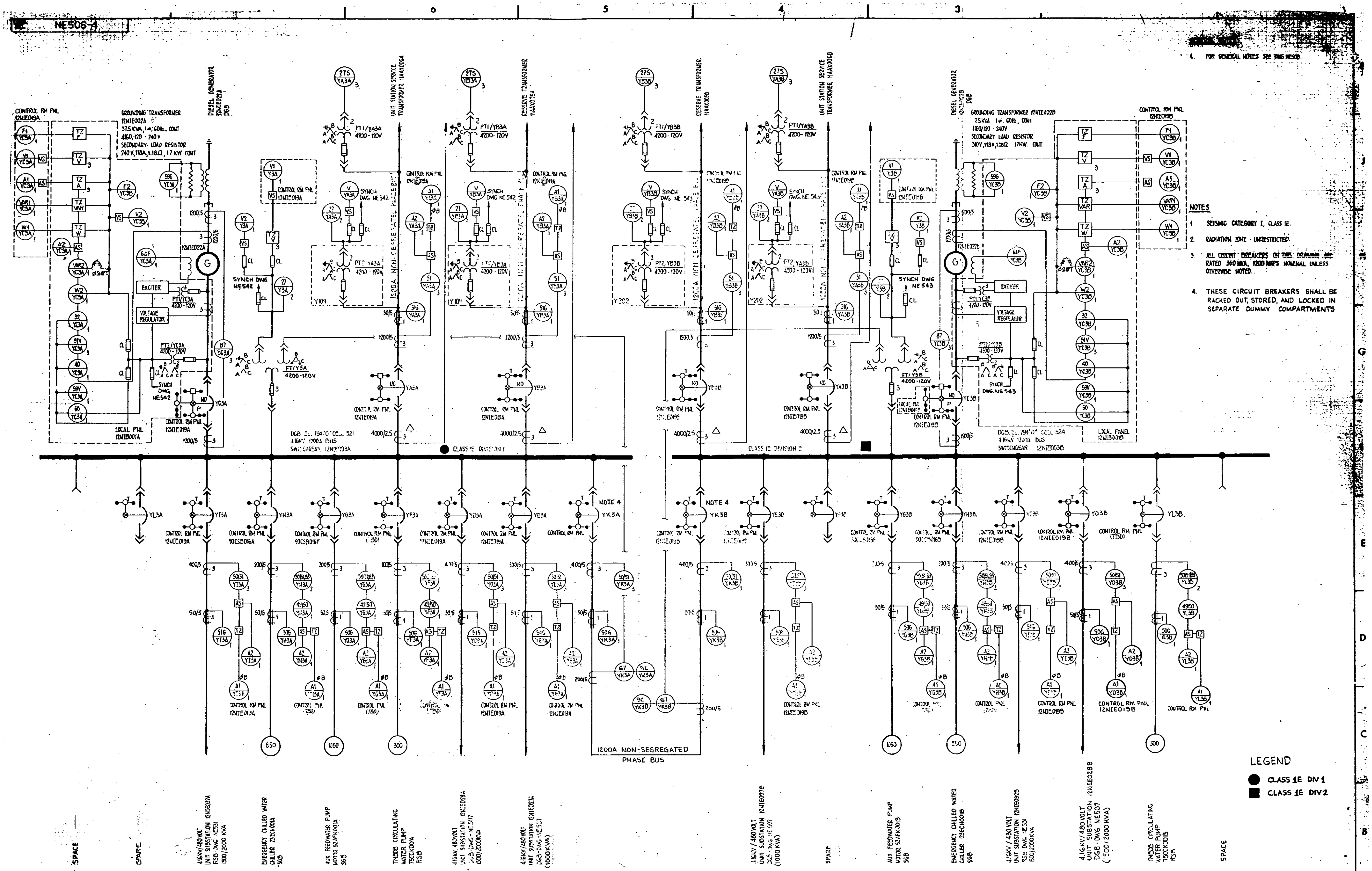


FIGURE 8.3-3 4.16KV Class 1E Switchgear One Line Diagram

Sheet 1 of 2 (NE506-4)

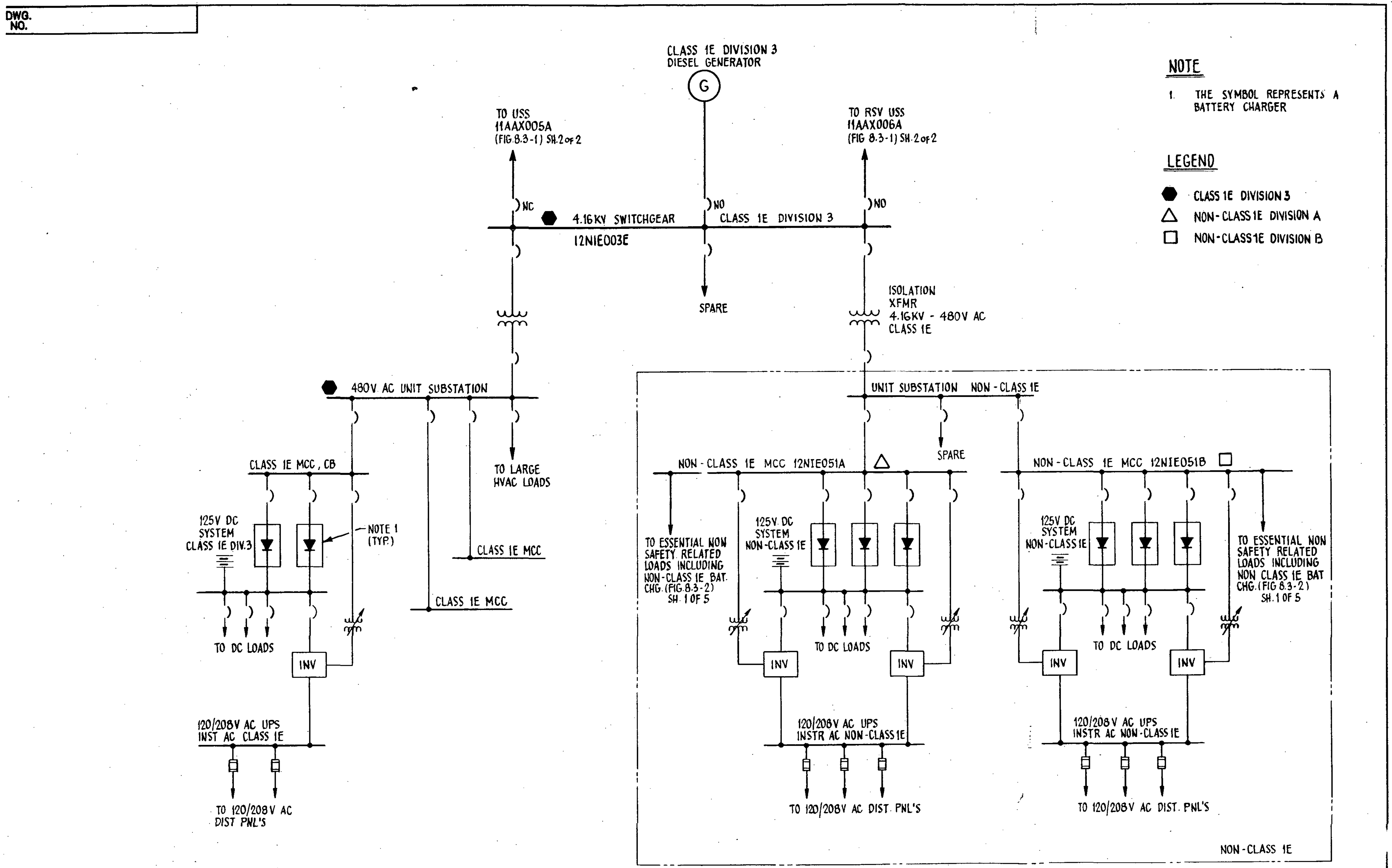
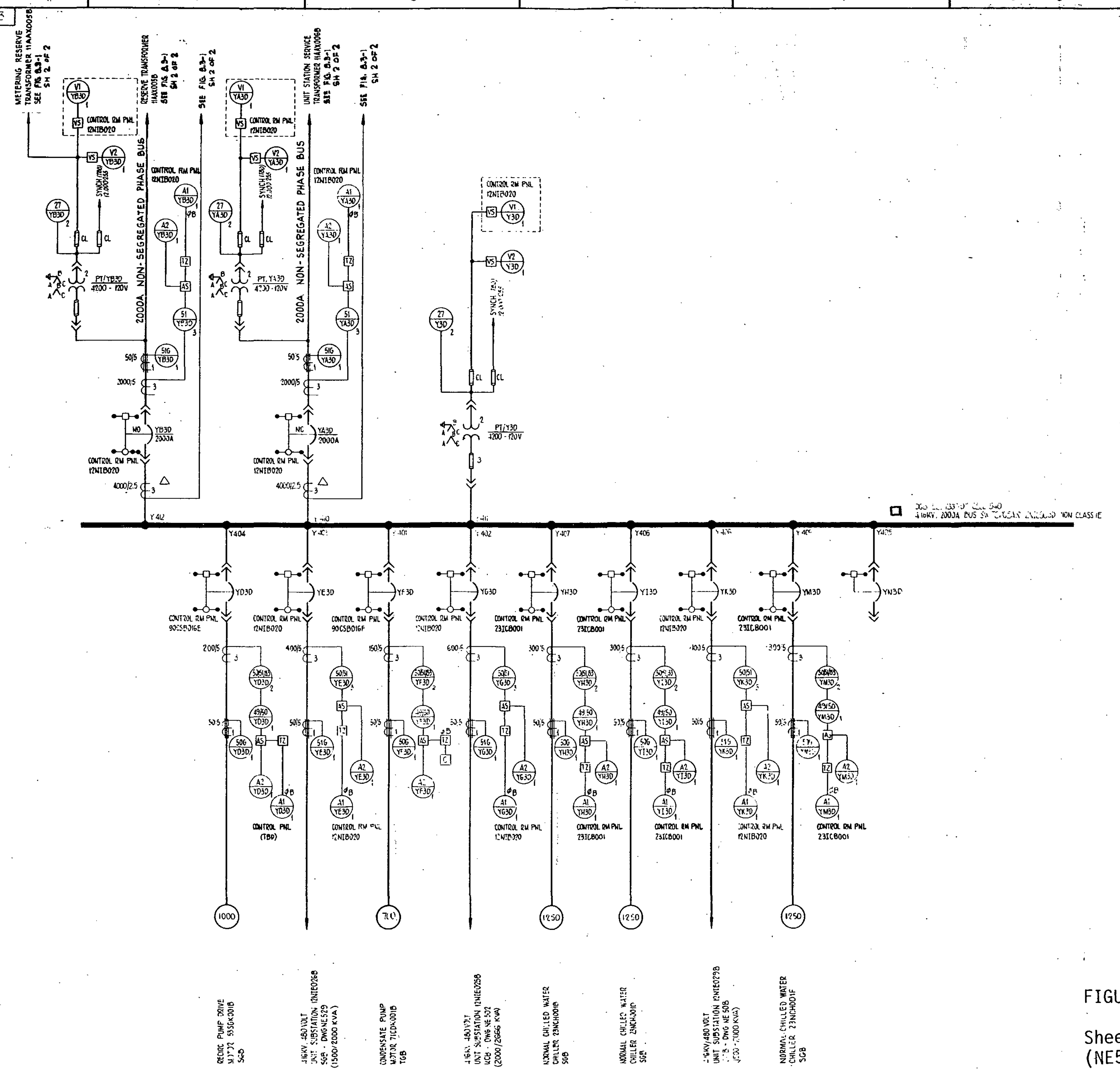


FIGURE 8.3-3 4.16KV Class 1E Switchgear One Line Diagram

Sheet 2 of 2



GENERAL NOTES

1. FOR ELECTRICAL DRAWING INDEX, SEE COMPUTERIZED DRAWING LIST.
2. FOR ELECTRICAL SYMBOLS, SEE "STANDARD SYMBOLOGY" FOR CRBRP DRAWINGS, WARD-D-0036.
3. ALL DEVICES ARE LOCATED IN SWITCHGEAR UNLESS OTHERWISE NOTED.
4. ELECTRICAL EQUIPMENT TO BE INSTALLED PER ELECTRICAL INSTALLATION SPECIFICATION.
5. INDICATED HP OF MOTORS ARE NAME PLATE RATINGS.
6. MAIN CONTROL BOARD 90CSB016 SWITCHING AND METERING REQUIREMENTS IDENTIFIED HEREIN ARE HANDLED BY MINIATURIZED SWITCHES. LOW ENERGY METERS, AND INTERPOSINGLOGIC. FOR DETAILS SEE INDIVIDUAL ELEMENTARY DIAGRAMS.

NOTES

1. SEISMIC CATEGORY III NON-CLASS IE.
2. RADIATION ZONE - UNRESTRICTED.
3. ALL CIRCUIT BREAKERS ON THIS DRAWING ARE RATED 350MVA 1200AMPS NOMINAL UNLESS OTHERWISE NOTED

LEGEND

□ Non Class 1E System B

FIGURE 8.3-4 4.16KV Switchgear One Line Diagram

Sheet 1 of 2
(NE503-3)

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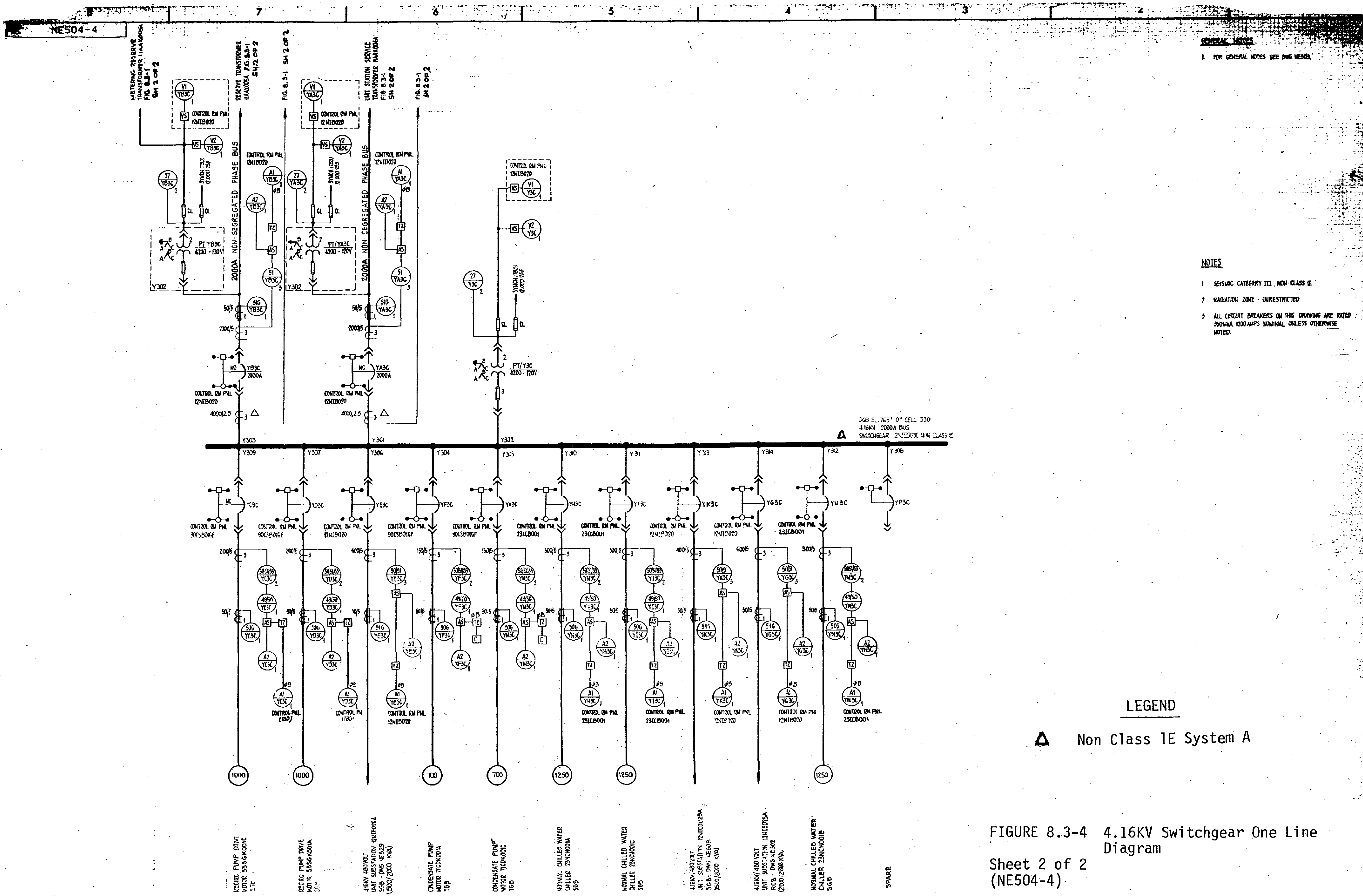

LEGEND

Δ Non Class 1E System A

FIGURE 8.3-4 4.16KV Switchgear One Line Diagram

Sheet 2 of 2
(NE504-4)

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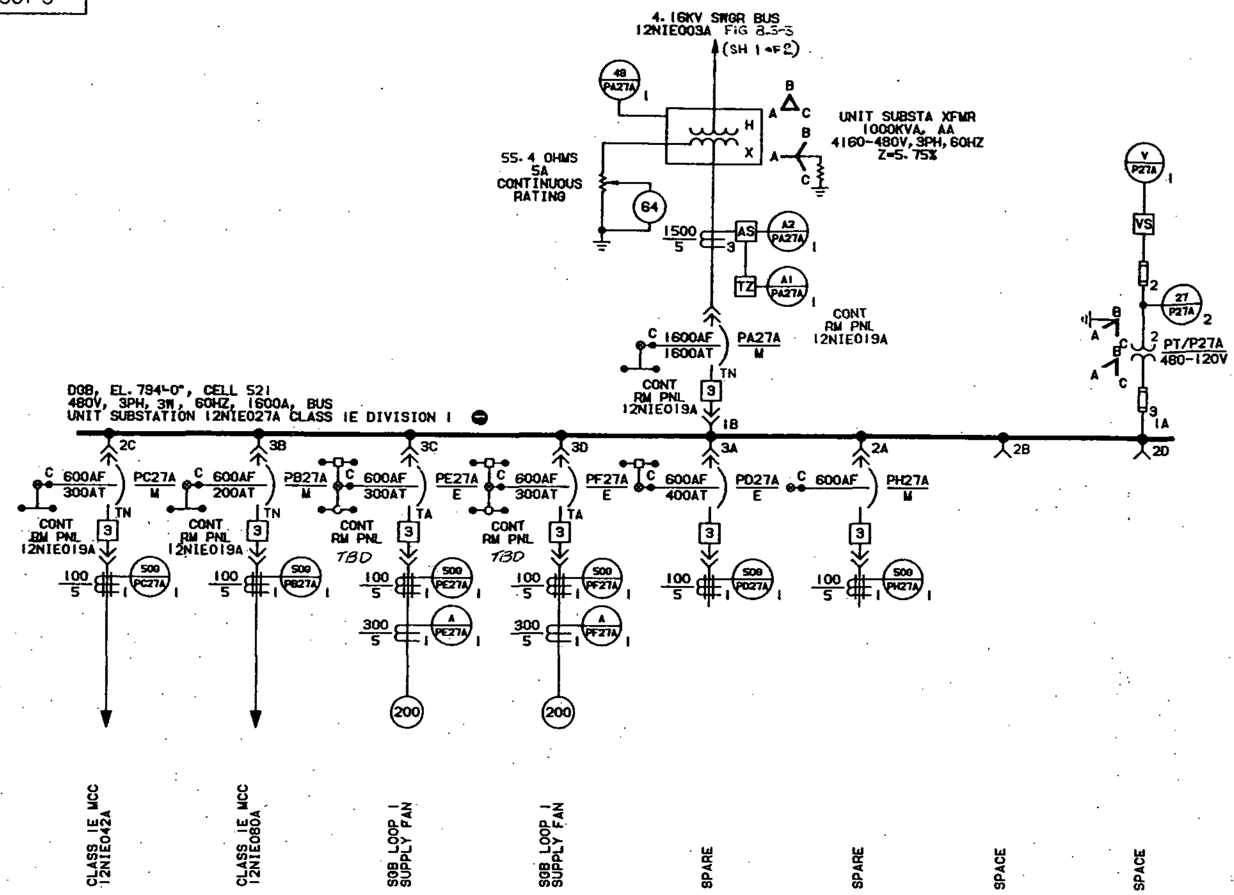
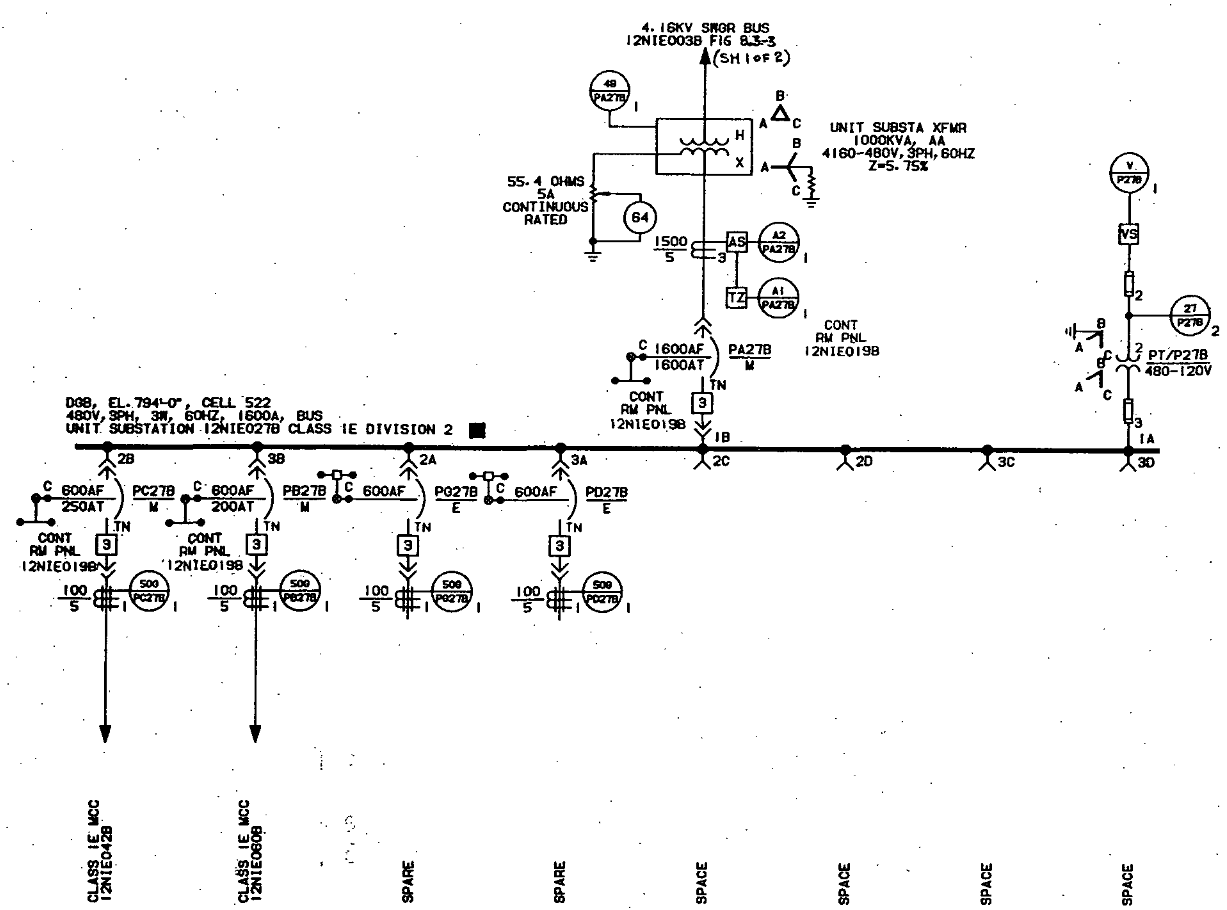
NE507-5





GENERAL NOTES

1. SYMBOLS & ABBREVIATIONS WARD DOC D-0036
2. FOR ELECTRICAL DRAWING INDEX. SEE COMPUTERIZED DRAWING LIST
3. ALL DEVICES ARE LOCATED IN UNIT SUBSTATIONS UOS
4. ELECTRICAL EQUIPMENT TO BE INSTALLED PER ELECTRICAL INSTALLMENT SPECIFICATIONS
5. ALL ELECTRICALLY OPERATED BREAKERS SHALL BE EQUIPPED WITH STANDARD PUSHBUTTONS ON THE BREAKER FRONT FOR TESTING PURPOSES
6. INDICATED HP OF MOTORS ARE NAME PLATE RATINGS
7. EACH CIRCUIT BREAKER SHALL BE EQUIPPED WITH AN ADJUSTABLE SOLID STATE TYPE SERIES OVER CURRENT TRIPPING DEVICE PROVIDING LONG TIME/SHORT TIME OVERCURRENT AND SHORT CIRCUIT PROTECTION (FOR MAIN AND MOTOR CONTROL CENTER FEEDER BREAKERS), AND LONG TIME/INSTANTANEOUS OVERCURRENT AND SHORT CIRCUIT PROTECTION (FOR INDIVIDUAL MOTOR FEEDER BREAKERS).
   ALL MANUALLY OPERATED BREAKERS. EXCEPT INCOMING CIRCUIT BREAKERS. SHALL BE EQUIPPED WITH SHUNT TRIP COIL
8. MAIN CONTROL BOARD SWITCHING UTILIZES MINIATURIZED CONTROL SWITCHES WITH INTERPOSING LOGIC. FOR DETAILS, SEE INIVIDUAL ELEMENTARY DIAGRAMS
9. DETAILED PROTECTIVE RELAYING TRIPPING AND ALARM FUNCTIONS ARE SHOWN ON ELEMENTARY DRAWINGS
10. THESE BREAKERS ARE PROVIDED WITH KIRK KEY INTERLOCKS SUCH THAT ONLY ONE OF THE TWO BREAKERS CAN BE IN OPERATING POSITION AT ANY TIME. THESE BREAKERS ARE PHYSICALLY LOCATED AT TWO DIFFERENT USS(12N1E028A & 12N1E028B)
11. THE TWO SETS OF BREAKERS PROVIDED WITH KIRK KEY INTERLOCKS ARE PROVIDED WITH TWO UNIQUE SETS OF KEYS.

NOTES

1. CLASS 1E. SEISMIC CATEGORY: I
2. RADIATION ZONE: UNRESTRICTED
3. COMPARTMENT NUMBER SHALL BE PREFIXED AS GIVEN BELOW:

USS	COMPT NO PREFIX
12N1E027A	PS11
12N1E027B	PS12

4. BREAKER SYMMETRICAL INTERRUPTING RATING CORRESPONDS TO THE FOLLOWING BREAKER FRAME SIZES

BREAKER FRAME SIZE	SYMMETRICAL INTERRUPTING RATING
600 AMPS	30.000 AMPS
1600 AMPS	50.000 AMPS

5. CLASS 1E FEEDER BREAKERS FOR CLASS 1E MCC SHALL NOT BE EQUIPPED WITH SHUNT TRIP COIL

LEGEND

CLASS 1E SYSTEM LOAD SHEDDING SYMBOLS:

TA-DEVISE TRIPS IN LOSS OF VOLTAGE. RECLOSES AUTOMATICALLY

TM-DEVICE TRIPS ON LOSS OF VOLTAGE. CLOSES MANUALLY

TN-DEVICE TRIP DOES NOT OCCUR ON LOSS OF VOLTAGE

● CLASS 1E DIV.1

■ CLASS 1E DIV. 2

FIGURE 8.3-5 480V Unit Substation One Line Diagram

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(NE507-5)

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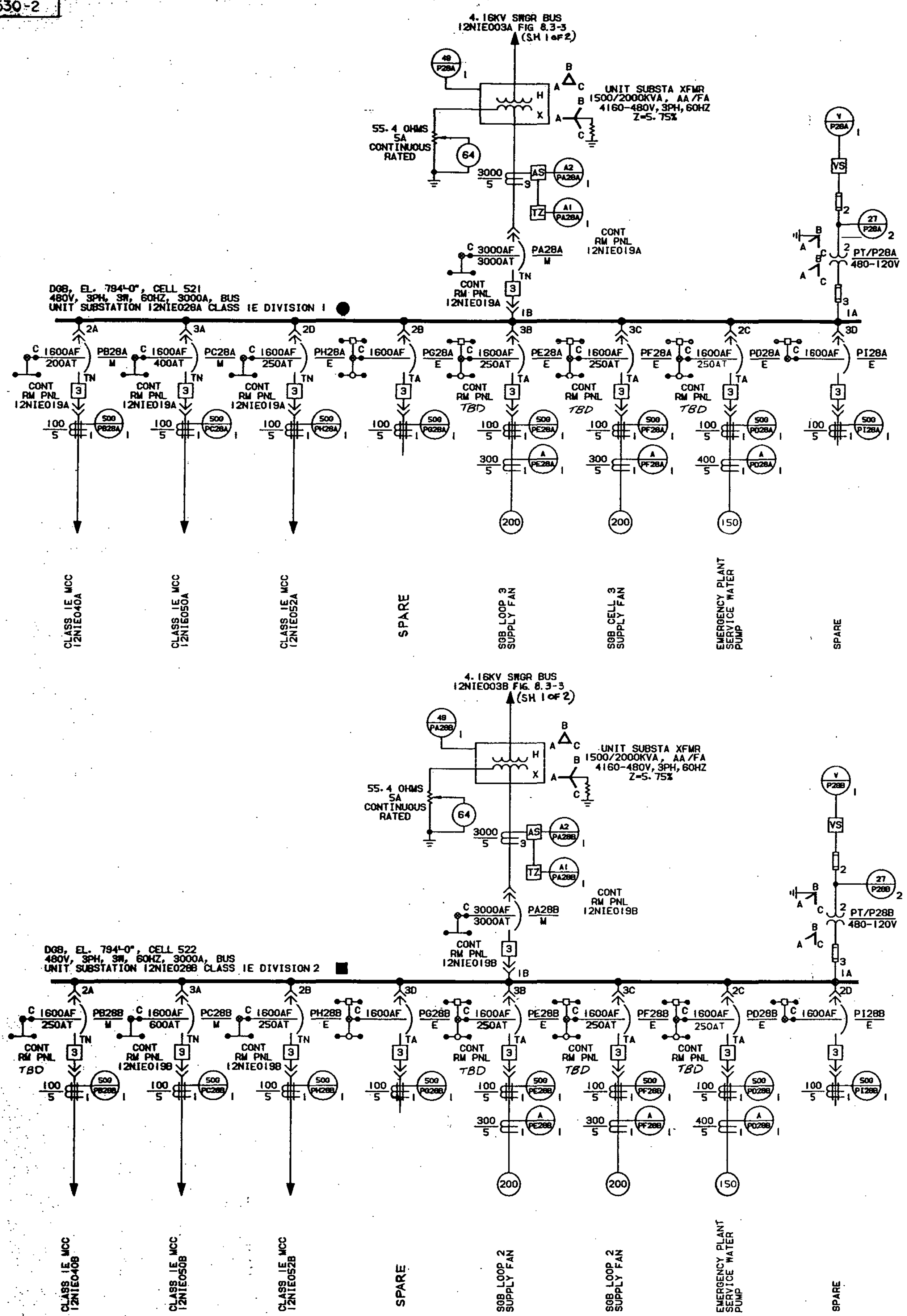
NE530-2



GENERAL NOTES

1. SYMBOLS & ABBREVIATIONS WARD-D-0036
2. FOR ELECTRICAL DRAWING INDEX, SEE COMPUTERIZED DRAWING LIST
3. ALL DEVICES ARE LOCATED IN UNIT SUBSTATIONS UOS
4. ELECTRICAL EQUIPMENT TO BE INSTALLED PER ELECTRICAL INSTALLMENT SPECIFICATIONS
5. ALL ELECTRICALLY OPERATED BREAKERS SHALL BE EQUIPPED WITH STANDARD PUSHBUTTONS ON THE BREAKER FRONT FOR TESTING PURPOSES
6. INDICATED HP OF MOTORS ARE NAME PLATE RATINGS
7. EACH CIRCUIT BREAKER SHALL BE EQUIPPED WITH AN ADJUSTABLE SOLID STATE TYPE SERIES OVER CURRENT TRIPPING DEVICE PROVIDING LONG TIME/SHORT TIME OVERCURRENT AND SHORT CIRCUIT PROTECTION (FOR MAIN AND MOTOR CONTROL CENTER FEEDER BREAKERS), AND LONG TIME/INSTANTANEOUS OVERCURRENT AND SHORT CIRCUIT PROTECTION (FOR INDIVIDUAL MOTOR FEEDER BREAKERS).
   ALL MANUALLY OPERATED BREAKERS, EXCEPT INCOMING CIRCUIT BREAKERS, SHALL BE EQUIPPED WITH SHUNT TRIP COIL
8. MAIN CONTROL BOARD SWITCHING UTILIZES MINIATURIZED CONTROL SWITCHES WITH INTERPOSING LOGIC. FOR DETAILS, SEE INIVIDUAL ELEMENTARY DIAGRAMS
9. DETAILED PROTECTIVE RELAYING TRIPPING AND ALARM FUNCTIONS ARE SHOWN ON ELEMENTARY DRAWINGS

NOTES

1. CLASS 1E, SEISMIC CATEGORY: I
2. RADIATION ZONE: UNRESTRICTED
3. COMPARTMENT NUMBER SHALL BE PREFIXED AS GIVEN BELOW:

USS	COMPT NO PREFIX
12N1E028A	PS13
12N1E028B	PS14

4. BREAKER SYMMETRICAL INTERRUPTING RATING CORRESPONDS TO THE FOLLOWING BREAKER FRAME SIZES:

BREAKER FRAME SIZE	SYMMETRICAL INTERRUPTING RATING
1600 AMPS	50,000 AMPS
3000 AMPS	65,000 AMPS

5. CLASS 1E FEEDER BREAKERS FOR CLASS 1E MCC SHALL NOT BE EQUIPPED WITH SHUNT TRIP COIL

LEGEND

CLASS 1E SYSTEM LOAD SHEDDING SYMBOLS:

TA-DEVICE TRIPS ON LOSS OF VOLTAGE, RECLOSES AUTOMATICALLY

TM-DEVICE TRIPS ON LOSS OF VOLTAGE, CLOSES MANUALLY

TN-DEVICE TRIP DOES NOT OCCUR ON LOSS OF VOLTAGE

● CLASS 1E DIV 1

■ CLASS 1E DIV 2

FIGURE 8.3-5 480V Unit Substation One Line Diagram

Sheet 2 of 4
(NE530-2)

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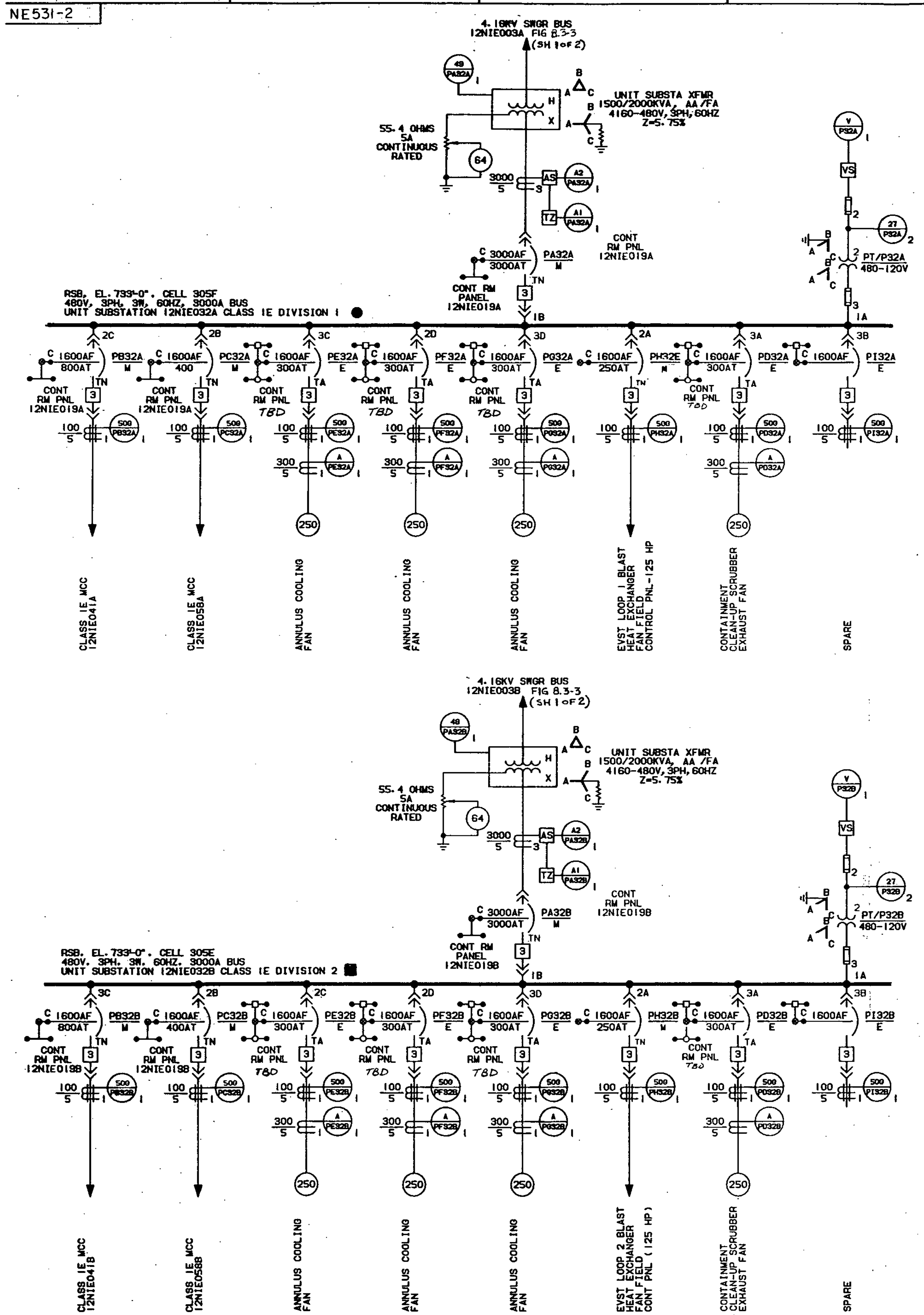
NE531-2



GENERAL NOTES

1. SYMBOLS & ABBREVIATIONS WARD-D-0036
2. FOR ELECTRICAL DRAWING INDEX. SEE COMPUTERIZED DRAWING LIST
3. ALL DEVICES ARE LOCATED IN UNIT SUBSTATIONS UOS
4. ELECTRICAL EQUIPMENT TO BE INSTALLED PER ELECTRICAL INSTALLATION SPECIFICATION
5. ALL ELECTRICALLY OPERATED BREAKERS SHALL BE EQUIPPED WITH STANDARD PUSHBUTTONS ON THE BREAKER FRONT FOR TESTING PURPOSES
6. INDICATED HP OF MOTORS ARE NAME PLATE RATINGS
7. EACH CIRCUIT BREAKER SHALL BE EQUIPPED WITH AN ADJUSTABLE SOLID STATE TYPE SERIES OVERCURRENT TRIPPING DEVICE PROVIDING LONG TIME/SHORT TIME OVERCURRENT AND SHORT CIRCUIT PROTECTION (FOR MAIN AND MOTOR CONTROL CENTER FEEDER BREAKERS). AND LONG TIME/ INSTANTANEOUS OVERCURRENT AND SHORT CIRCUIT PROTECTION (FOR INDIVIDUAL MOTOR FEEDER BREAKERS) ALL MANUALLY OPERATED BREAKERS. EXCEPT INCOMING CIRCUIT BREAKERS. SHALL BE EQUIPPED WITH SHUNT TRIP COIL
8. MAIN CONTROL BOARD SWITCHING UTILIZES MINIATURIZED CONTROL SWITCHES WITH INTERPOSING LOGIC. FOR DETAILS. SEE INDIVIDUAL ELEMENTARY DIAGRAMS
9. DETAILED PROTECTIVE RELAYING. TRIPPING AND ALARM FUNCTIONS ARE SHOWN ON ELEMENTARY DRAWINGS

NOTES

1. CLASS 1E. SEISMIC CATEGORY: I UOS
2. RADIATION ZONE: I UOS
3. COMPARTMENT NUMBER SHALL BE PREFIXED AS GIVEN BELOW:

USS	COMPT NO PREFIX
12NIE032A	PS21
12NIE032B	PS22

4. BREAKER SYMMETRICAL INTERRUPTING RATING CORRESPONDS TO THE FOLLOWING BREAKER FRAME SIZES:

BREAKER FRAME SIZE	SYMMETRICAL INTERRUPTING RATING
1600 AMPS	50.000 AMPS
3000 AMPS	65.000 AMPS

5. CLASS 1E FEEDER BREAKER FOR CLASS 1E MCC SHALL BE EQUIPPED WITH SHUNT TRIP COIL

LEGEND

CLASS 1E SYSTEM LOAD SHEDDING SYMBOLS:

TA-DEVICE TRIPS ON LOSS OF VOLTAGE. RECLOSES AUTOMATICALLY

TM-DEVICE TRIPS ON LOSS OF VOLTAGE. CLOSES MANUALLY

TN-DEVICE TRIP DOES NOT OCCUR ON LOSS OF VOLTAGE

● CLASS 1E DIV 1

■ CLASS 1E DIV 2

FIGURE 8.3-5 480V Unit Substation One Line Diagram

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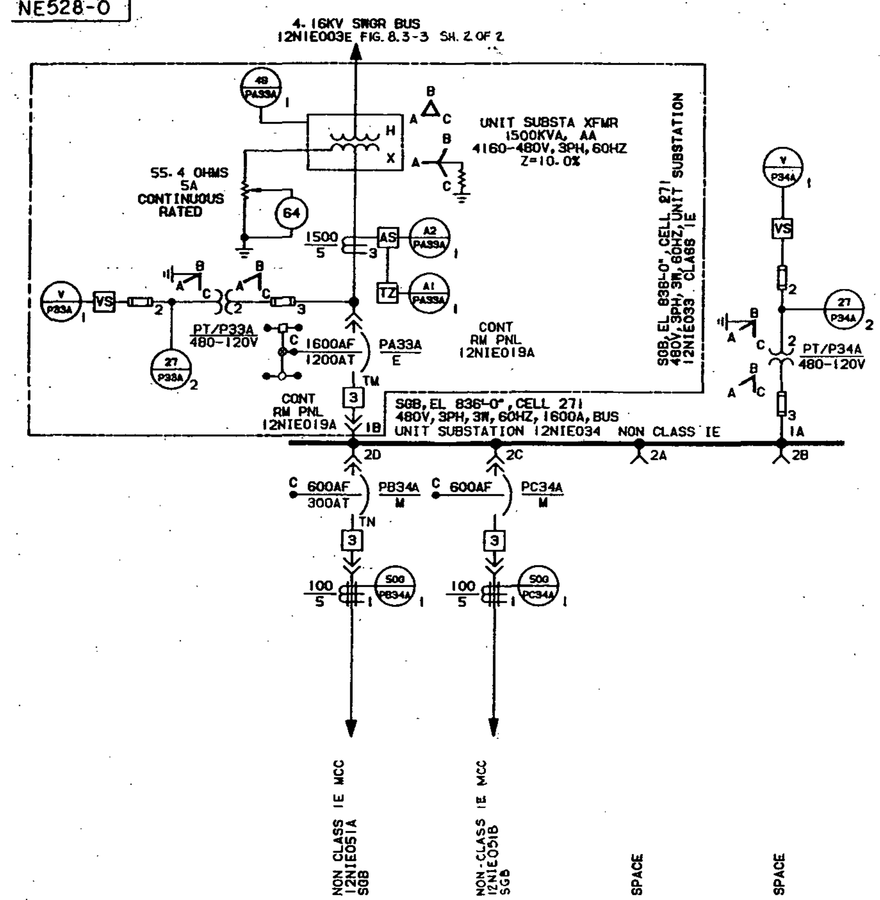

GENERAL NOTES

1. SYMBOLS & ABBREVIATIONS WARD-D-0036
2. FOR ELECTRICAL DRAWING INDEX, SEE COMPUTERIZED DRAWING LIST
3. ALL DEVICES ARE LOCATED IN UNIT SUBSTATIONS UOS
4. ELECTRICAL EQUIPMENT TO BE INSTALLED PER ELECTRICAL INSTALLATION SPECIFICATION
5. ALL ELECTRICALLY OPERATED BREAKERS SHALL BE EQUIPPED WITH STANDARD PUSHBUTTONS ON THE BREAKER FRONT FOR TESTING PURPOSES
6. INDICATED HP OF MOTORS ARE NAME PLATE RATINGS
7. EACH CIRCUIT BREAKER SHALL BE EQUIPPED WITH AN ADJUSTABLE SOLID STATE TYPE SERIES OVERCURRENT TRIPPING DEVICE PROVIDING LONG TIME/SHORT TIME OVERCURRENT AND SHORT CIRCUIT PROTECTION (FOR MAIN AND MOTOR CONTROL CENTER FEEDER BREAKERS). AND LONG TIME/ INSTANTANEOUS OVERCURRENT AND SHORT CIRCUIT PROTECTION (FOR INDIVIDUAL MOTOR FEEDER BREAKERS) ALL MANUALLY OPERATED BREAKERS, SHALL BE EQUIPPED WITH SHUNT TRIP COIL
8. MAIN CONTROL BOARD SWITCHING UTILIZES MINIATURIZED CONTROL SWITCHES WITH INTERPOSING LOGIC. FOR DETAILS, SEE INDIVIDUAL ELEMENTARY DIAGRAMS
9. DETAILED PROTECTIVE RELAYING, TRIPPING AND ALARM FUNCTIONS ARE SHOWN ON ELEMENTARY DRAWINGS

NOTES

1. CLASS IE, SEISMIC CATEGORY: I USS-12NIE033A & 12NIE033B NON-CLASS IE, SEISMIC CATEGORY: III 12NIE034A & 12NIE034B
2. RADIATION ZONE: UNRESTRICTED
3. COMPARTMENT NUMBER SHALL BE PREFIXED AS GIVEN BELOW:

USS	COMPT NO PREFIX
12NIE033A	PS23
12NIE033B	PS24
12NIE034A	PS25
12NIE034B	PS26

4. BREAKER SYMMETRICAL INTERRUPTING RATING CORRESPONDS TO THE FOLLOWING BREAKER FRAME SIZES:

BREAKER FRAME SIZE	SYMMETRICAL INTERRUPTING RATING
600 AMPS	30,000 AMPS
1600 AMPS	50,000 AMPS

LEGEND

CLASS IE SYSTEM LOAD SHEDDING SYMBOLS:

TA-DEVICE TRIPS ON LOSS OF VOLTAGE. RECLOSES AUTOMATICALLY

TM-DEVICE TRIPS ON LOSS OF VOLTAGE. CLOSES MANUALLY

TN-DEVICE TRIP DOES NOT OCCUR ON LOSS OF VOLTAGE

FIGURE 8.3-5 480V Unit Substation One Line Diagram

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(NE528-0)

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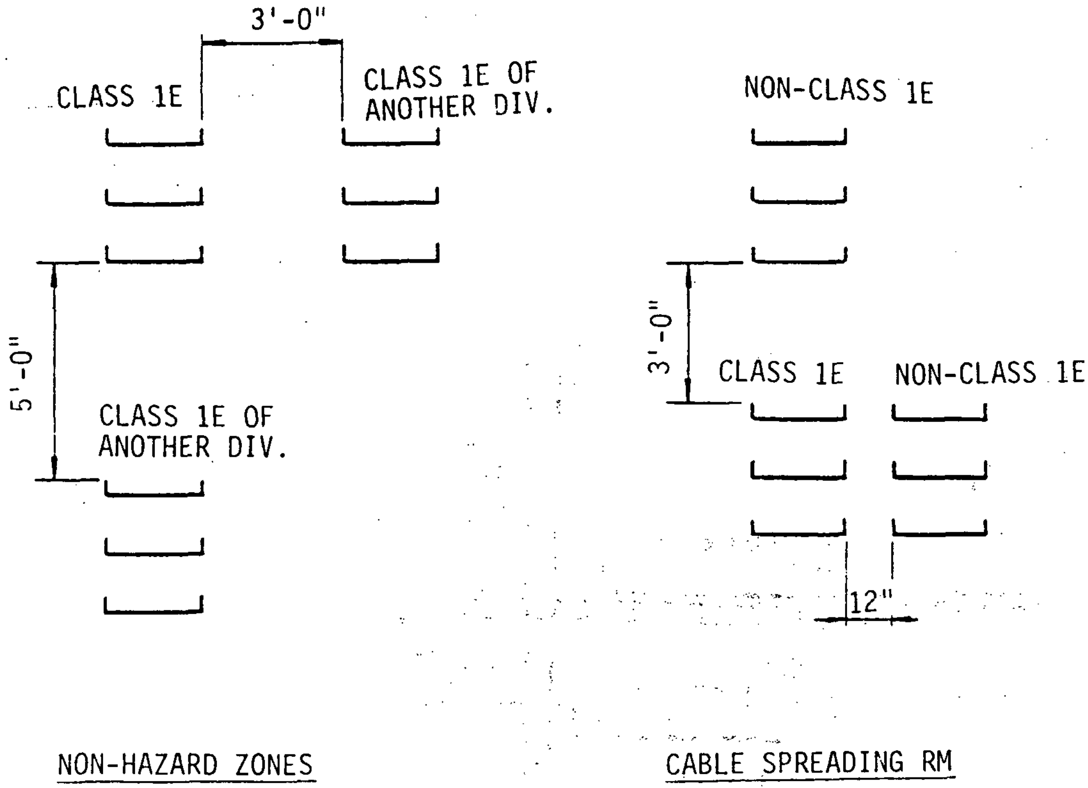


NOTE
THIS SEPARATION APPLIES FOR ALL DIVISIONS CLASS 1E CABLE TRAYS FROM EACH OTHER AND FROM NON-CLASS 1E CABLE TRAYS.

FIGURE 8.3-6 CABLE TRAY SEPARATION

SHEET 1 of 7

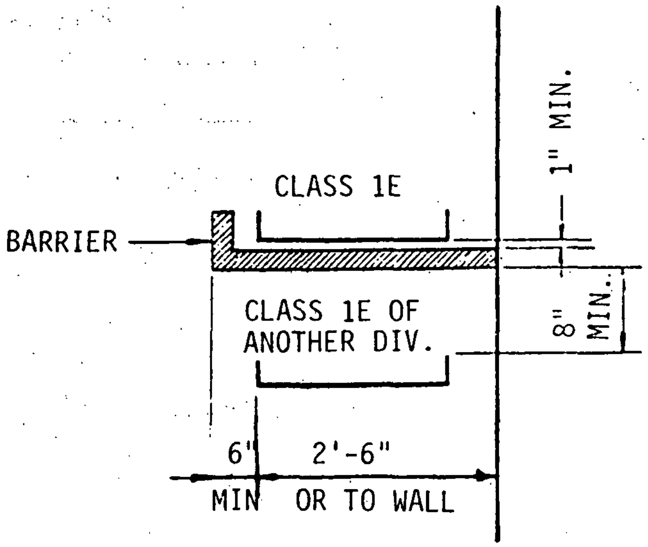


FIGURE 8.3-6 ARRANGEMENT WHEN VERTICAL SPATIAL SEPARATION IS NOT MAINTAINED NON-HAZARD ZONES

SHEET 2 of 7

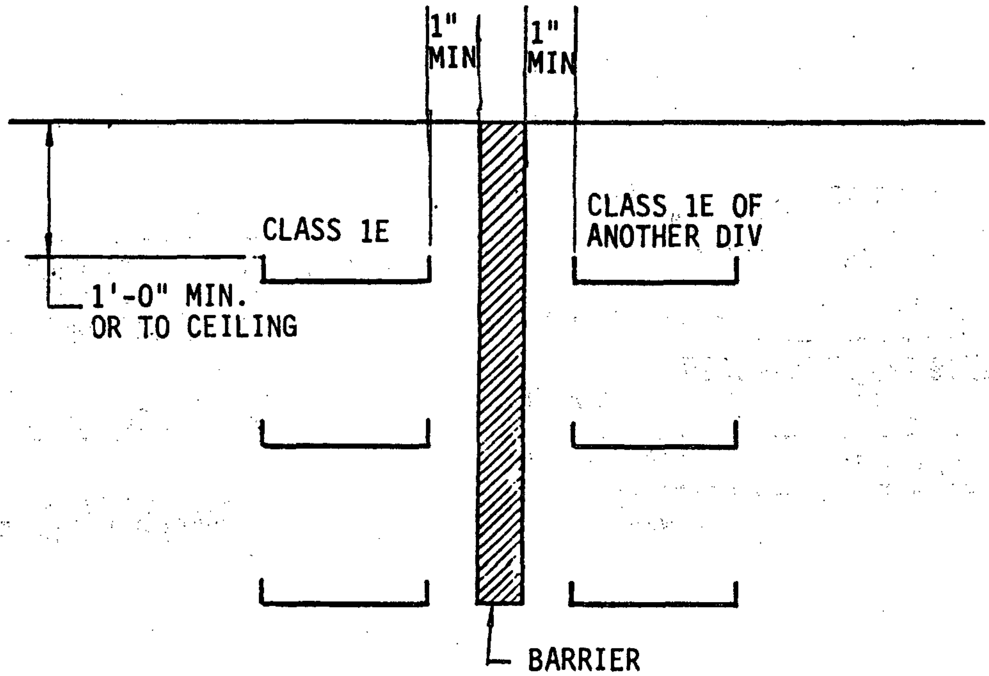


FIGURE 8.3-6 ARRANGEMENT WHEN HORIZONTAL SPATIAL SEPARATION IS NOT MAINTAINED NON-HAZARD ZONES

SHEET 3 of 7

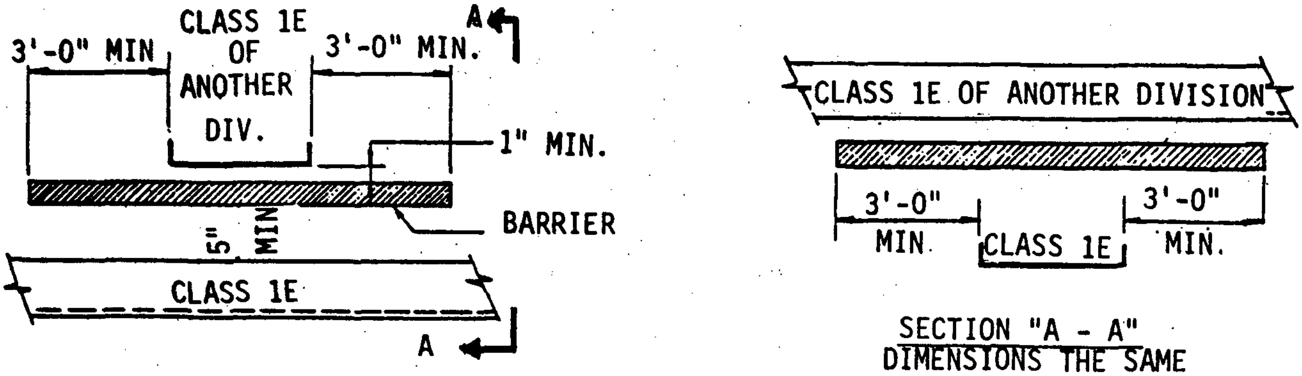


FIGURE 8.3-6 ARRANGEMENT FOR CLASS 1E CABLE TRAY CROSSINGS WHERE VERTICAL SEPARATION IS NOT MAINTAINED
NON-HAZARD ZONES

SHEET 4 of 7

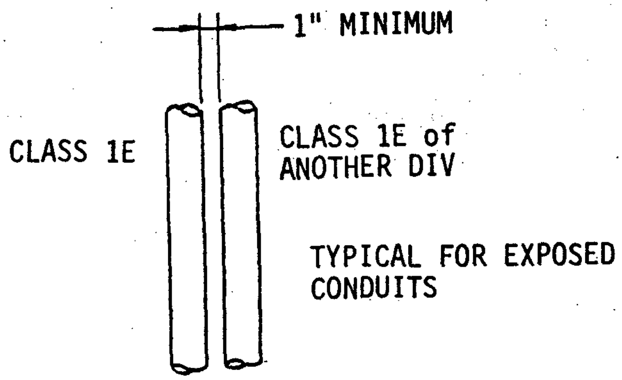
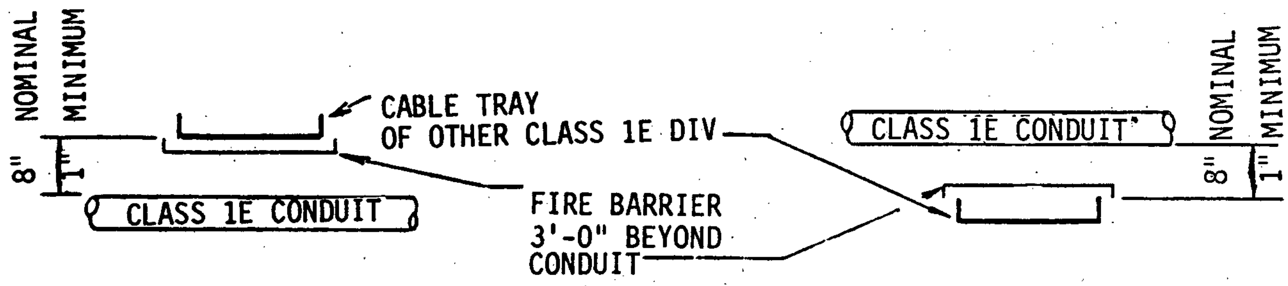


FIGURE 8.3-6 CONDUIT SPATIAL SEPARATION NON-HAZARD ZONES

SHEET 5 of 7



SAFETY CONDUITS CROSSING
OTHER CLASS 1E DIVISION
CABLE TRAYS FROM BELOW
NON-HAZARD ZONES

SAFETY CONDUITS CROSSING
OTHER CLASS 1E DIVISION
CABLE TRAYS FROM ABOVE
NON-HAZARD ZONES

FIGURE 8.3-6

SHEET 6 of 7

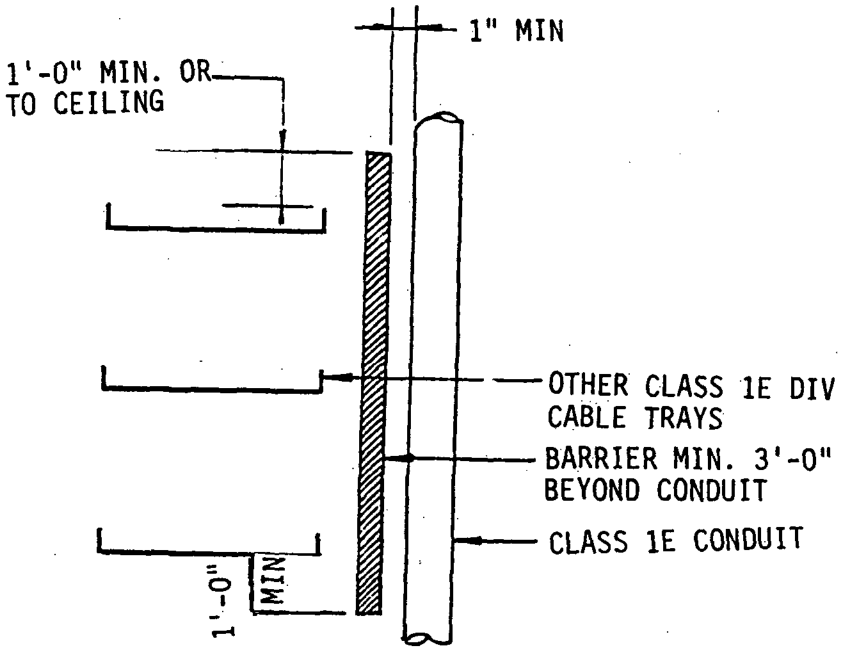


FIGURE 8.3-6 ARRANGEMENT FOR VERTICAL SAFETY CONDUITS CROSSING OTHER CLASS 1E DIVISION CABLE TRAYS NON-HAZARD ZONES

SHEET 7 of 7

CLINCH RIVER

BREEDER REACTOR PROJECT

PRELIMINARY SAFETY ANALYSIS REPORT

CHAPTER 9

AUXILIARY SYSTEMS

PROJECT MANAGEMENT CORPORATION

CHAPTER 9.0 AUXILIARY SYSTEMS

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9.1 FUEL STORAGE AND HANDLING

The Reactor Refueling System provides the means of storing, trans-
20 porting, and handling core assemblies and core special assemblies within the
44 CRBRP.* The following are defined as core assemblies:

1) Fuel assemblies
2) Inner blanket assemblies
3) Radial blanket assemblies
4) Control assemblies
5) Removable Radial shield assemblies

44 59 Of the new core assemblies arriving at the CRBRP, only new
fuel assemblies require shielding and criticality control, and present any
potential radiation safety hazard. However, after irradiation in the reactor
core, all core assemblies require shielding and removal of decay heat in
44 various degrees. Irradiated core assemblies containing fuel also require
criticality control and containment of fission gas in the event of leaks in
fuel rods.

The Reactor Refueling System consists of the facilities and equip-
ment needed to accomplish the normal scheduled refueling operations, and all
44 other functions incident to handling of core assemblies.

Its primary functions are as follows:

44 1) Receive, inspect, store, and prepare new core assemblies for insertion in reactor

2) Transfer core assemblies between facilities (i.e., ex-vessel
44 storage tank, reactor, fuel handling cell and new fuel unloading stations)

44 3) Transfer core assemblies between the core and in-vessel storage or transfer positions, or between core positions

44 4) Provide storage for irradiated core assemblies

20 *The purpose of core special assemblies is to facilitate core loading and
unloading. Due to the tight tolerances of the discriminator post with
20 respect to the socket, the vertical position of a core special assembly has
a limited deviation from the vertical center line when it is "standing
44 freely." This makes insertion or withdrawal of neighboring core assemblies
20 easier during core loading or unloading, respectively. Core special assem-
blies do not contain any fuel and are removed prior to reactor startup.
They have the same handling socket, except for identification number, as the
44 core assembly they replace.

5) Examine and prepare irradiated core assemblies for shipment.

44 6) Provide inventory control of all core assemblies.

The major equipment and facilities needed to perform these functions are as follows:

44 Function 1) New fuel shipping containers, new fuel unloading stations, and ex-vessel storage tank (EVST)

Function 2) Ex-vessel transfer machine (EVTM)

Function 3) In-vessel transfer machine (IVTM)

Function 4) EVST

44 Function 5) Fuel handling cell (FHC)

Function 6) Refueling instrumentation and control system, FHC and IVTM.

Major equipment and facilities needed to perform functions incident to the
44 handling of core assemblies and their functions are as follows:

1) Auxiliary handling machine (AHM)-handle equipment needed in fuel handling and serve as a maintenance cask for some components

2) RCB miscellaneous storage facilities and RSB plug storage facilities ·
44 store port plugs when removed from the reactor, EVST, or FHC

3) Floor valves - seal port openings in the reactor, EVST, and FHC when port plugs are removed

4) Fuel transfer port cooling inserts - provide cooling for removal of decay heat if an irradiated fuel assembly becomes immobilized in a port during transfer.

Figures 9.1-1 and 9.1-2 show the arrangement of major fuel handling and storage equipment in the Reactor Containment Building (RCB) and Reactor Service Building (RSB). Figure 9.1-1 also shows the flow path of new and spent core assemblies. New core assemblies enter the RSB by truck, are unloaded from their shipping containers, inspected and transferred by the EVTM to the sodium filled ex-vessel storage tank for preheating and storage. After reactor shutdown for refueling, the ex-vessel transfer machine transfers new core assemblies from the EVST to the reactor, and spent core assemblies from the reactor to the EVST, on a one-for-one basis. The in-vessel transfer machine, in conjunction with the reactor rotating plugs, transfers spent core
44 assemblies from the core to the in-vessel transfer positions, and new core

assemblies from the in-vessel transfer positions to the core, on a one-for-one basis. After completion of refueling and suitable period for decay, the spent core assemblies are removed from the EVST to the fuel handling cell, examined if desired, and loaded into the spent fuel shipping cask for shipment to a fuel reprocessor.

The design bases, description and safety evaluation of fuel handling and storage equipment and facilities are given in 9.1.1 (new fuel storage), 9.1.2 (spent fuel storage), 9.1.3 (spent fuel cooling) and 9.1.4 (fuel handling equipment). Section 9.1.4 also describes the movement of fuel assemblies through the plant in more detail.

All design bases were developed to conform with the CRBRP Design Criteria 53, 54, and 55 described in Section 3.1, and with the intent of Regulatory Guide 1.13. The Reactor Refueling System is designed to reduce the probability of operator mishandling or of maloperations that could cause fuel damage and potential fission product release, while limiting the in-plant buildup of airborne radioactivity during normal plant operations, so that the exposure to plant operators is minimized. In addition, specific attention was given in the selection of equipment design bases to ensure that no single active component failure can result in a loss-of-safety function. Additional margin is provided for protecting the public during refueling and fuel handling operations by the low leakage design of the RSB, and maintaining the RSB and RCB when the refueling hatch is open at a minimum of 1/4" W.G. negative pressure with respect to the outside atmosphere, with the exhaust discharged through a high efficiency filter train capable of efficiencies as high as 99% adsorbent efficiency and a 99.9% combined HEPA filter efficiency. For design basis events, the efficiencies of Section 6.2.6.2 are applied.

The fission product and transuranium elements inventories are developed from the nuclear design data and reference equilibrium core and blanket management schemes. The fission product inventory and its corresponding energy spectrum were derived using the fission yields contained in the ENDS/B IV and RIBD II code library and consider the cyclic operation and core/blanket management of the equilibrium cycle.

9.1.1 New Fuel Storage

New fuel is stored within the Reactor Service Building (RSB) in the ex-vessel storage tank (EVST), containing sodium. This storage facility contains both new and spent fuel. In addition to the EVST, new fuel assemblies are also temporarily retained in shipping containers on the RSB operating floor and below the operating floor in two new fuel unloading stations. Each of these two new fuel unloading stations can temporarily contain one shipping container with a new fuel assembly until unloaded. A conceptual drawing is provided in Figure 9.1-3.

This section will cover only new fuel storage in the unloading stations and in the shipping containers. Storage of new fuel in the EVST will be covered in Section 9.1.2 under spent fuel storage.

The new fuel shipping containers (NFSC) will be licensed separately; therefore, the following paragraphs are intended to assure that operations performed within the RSB do not exceed those for which the container is designed.

59

9.1.1.1 Design Basis

The conditions to which the new fuel shipping containers are exposed within the RSB will not exceed those of the applicable regulations to which containers must be designed.

9.1.1.2 Design Description

Each container can only hold 1 core assembly. The container measures about 20 in. in diameter; a conceptual design drawing is shown in
59 Figure 9.1-5. The containers are removed from the truck one at a time by
the RSB crane and lowered into a new fuel unloading station. The shipping cover is not removed from the container until it is installed in the unloading station.

Two new fuel unloading stations are located between the EVTM gantry-trolley rails in the RSB. Each station supports and positions, in a vertical orientation, a new fuel shipping container to permit loading or unloading of a new core assembly by the EVTM. The unloading station unloads the fuel assembly from the end in a vertical orientation.

59

Each station consists of a pit in the operating floor with a structural steel support for the shipping container. An adaptor and valve assembly is mounted on the top of the unloading station to provide an interface between the unloading station and the EVTM. The adaptor mates with and seals to the EVTM closure valve. Argon gas to inert and purge the shipping containers is provided from a nearby floor service station.

9.1.1.3 Safety Evaluation

44 The outer diameter of the shipping container limits the
center-to-center separation, and keeps the NFSC array subcritical, even if a large number of containers were as close together as physically possible and submerged in water. Each fuel assembly is removed from the shipping container by the EVTM only after the container is installed in the unloading
44 20 ing station.

9.1.2 Spent Fuel Storage

Spent fuel is stored within the Reactor Service Building (RSB) in
59 44 two areas; the ex-vessel storage tank (EVST) and the fuel handling cell (FHC).

The ex-vessel storage tank (EVST) provides safe and controlled storage under sodium for both new and spent fuel. New fuel is stored in the EVST under sodium, after an initial preheating in an argon gas environment in one of the EVST's 24 preheat positions. Temporary storage of spent fuel occurs
44 in a small transfer station in the fuel handling cell (FHC). The FHC is a sub-floor hot cell in which spent fuel assemblies are removed from Core Component Pots (CCP), inspected and measured, if desired, and transferred into the Spent Fuel Shipping Cask (SFSC).

This section covers the safety aspects of new and spent fuel stored in the EVST and spent fuel stored in the FHC. The SFSC itself will be
59 44 licensed separately. (Reference 2)

The design bases, description, and safety evaluation of the spent fuel storage will cover the following safety items: prevention of criticality, provision for adequate shielding protection against radioactivity release, prevention of mechanical damage, and fuel storage monitoring. Spent fuel decay heat removal is discussed in Section 9.1.3.

9.1.2.1 Safety Aspects of New and Spent Fuel Storage in the EVST

The EVST performs the functions of preparing new core assemblies for insertion into the reactor, and providing safe, controlled storage for irradiated and new core assemblies.

The EVST capabilities required to implement its functions are as follows:

1) Preheating of new fuel assemblies

2) Radiation shielding

3) Storing new and spent fuel assemblies under sodium

20 4) Cooling spent fuel assemblies (See Section 9.1.3)

5) Containing cover gas

6) Providing structural support and physical separation of fuel assemblies to maintain their subcriticality

26

The EVST is a sodium-filled storage facility with a two-tier
rotating turntable and a fixed head with fuel transfer ports (see
44 Figure 9.1-6). One EVST is provided, and is located in the RSB between
the EVTM gantry rails in a nitrogen-filled concrete vault. The major
EVST components are: (1) storage vessel, (2) guard tank, (3) closure
head assembly (4) rotating turntable, (5) support structure, and
(6) drive and controls.

The design of the EVST is similar to the FFTF intermediate
59 decay storage facility (IDS). The storage vessel is supported at its
upper flange, suspended into the surrounding guard tank. The guard tank is
bottom supported. The turntable is supported by a bearing and seal configuration
above the storage vessel flange. It holds storage tubes which provide approx-
imately 650 storage positions in two tiers. The entire turntable (except for
20 the approximately upper 11 ft. of flanged support) with the storage tubes is
59 under sodium.

9.1.2.1.1. <u>Design Bases</u>

59 Criticality of new and spent fuel assemblies stored in the
EVST is prevented by physical separation. The center-to-center spacing
between fuel assemblies in storage positions in combination with several
51 permanently installed neutron absorber assemblies is sufficient to
maintain the array, when fully loaded with new fuel of the highest
44 anticipated enrichment and immersed under sodium, in a subcritical con-
dition with the K_{eff} less than 0.95. The EVST design considers all
normal loadings in combination with the loads from a safe shutdown
earthquake (SSE) in maintaining the necessary physical separation.
The EVST head is designed to absorb the load of the heaviest piece of
equipment handled by the RSB bridge crane over the EVST: (a) for the
main hook, lowered at the maximum crane speed (5 fpm), and (b) for the
auxiliary hook, accidentally dropping from the maximum handling height
to which it is raised, onto the center of the striker plate without
affecting the integrity of the fuel separation lattice. The EVST is
44 located such that heavy equipment not belonging to the fuel handling
20 and storage system is not carried over it.

Shielding is provided in the EVST and in its containment vault for radiation protection to meet the radiation protection requirements specified in 10CFR20, to ensure that the integrated dose is below 125 mrem/quarter, and to meet the radiation zone criteria of Section 12.1.

The EVST is designed for containment of radioactive fission
gases. Radiation doses due to leakage and diffusion through seals and
penetration are limited to well below those of the guidelines of 10 CFR
100 when the entire fission gas inventory of one fuel assembly is
44 released into the EVST cover gas.

The EVST is so designed that movement of the rotating turntable will not occur while a core component pot is being withdrawn or inserted. This design condition prevents mechanical damage to the CCP or its contents.

Amend. 59
Dec. 1980

Monitoring instrumentation will be provided for the EVST and its associated areas for conditions that might result in a loss of the capability to remove decay heat and to detect excessive radiation levels.

9.1.2.1.2 <u>Design Description</u>

The EVST is shown in Figure 9.1-6. The fuel assemblies are separated from
each other by the storage positions which are part of the turntable. The
storage positions within the EVST are cylindrical tubes restrained and
59 supported by a steel gridwork. Each storage tube is held in place such that
all fuel assemblies have a center-to-center distance of 9 in. or more. Each
storage tube holds two core component pots containing either new or spent fuel
assemblies in a vertical array, i.e., one above the other in any order. A
maximum of 650 new and spent fuel assemblies can be stored in the EVST in nine
circular rows, although under typical refueling conditions, there will be 162
fuel assemblies and fewer than 150 radial blanket assemblies, all stored in
the upper tier. Nine storage positions in various rows of the upper tier, and
the nine corresponding positions in the lower tier contain B_4C-filled neutron
59 absorber assemblies and are inaccessible for fuel storage. The purpose of the
absorber assemblies is to limit the value of K_{eff} for the EVST to <0.95.

The closure head assembly consists of a 6.5 inch-thick steel striker plate, a 12-inch thick steel closure head, and a multi-sheet steel reflective thermal insulation. The closure head seals the upper end of the storage vessel and provides containment for the EVST cover gas. Striker plate and closure head are designed to support normal structural loads as well as the accidental impact loads given in the design bases.

The EVST is contained in a 26 ft square by 57 ft deep nitrogen-filled concrete vault located in the Reactor Service Building (RSB), as shown in Figure 9.1-1. The EVST vessel is filled with sodium, topped by argon cover gas. A guard tank, which precludes the loss of sodium, surrounds the storage vessel. The design and construction of the EVST is in accordance with applicable codes, standards, and specifications listed in Section 3.8 for Seismic Category I structures.

The top of the EVST is located at the operating floor level of the RSB, as shown in Figure 9.1-2. Sufficient axial shielding must therefore be provided so that the radiation level at the top of the EVST does not exceed 0.2 mrem/hr, (see plant radiation zone criteria presented in Section 12.1). This shielding is provided by 22 in. of steel in the closure head assembly and 87 in. of sodium above the fuel. Nine fuel transfer ports penetrate the head, as shown in Figure 9.1-6. Each fuel transfer port is provided with a shielded cooling sleeve (see Section 9.1.4.7) which extends from the closure head to a point between the thermal insulation and the sodium level. Lead shielded collars around the ports are located in the space between the head and striker plate. A floor valve adaptor is inserted into the fuel transfer port before a floor valve is mated to the EVST, and extends from the striker plate top to the cooling insert. The cooling sleeve, shield collar, and floor valve adaptor reduce the transient dose rate from a spent fuel assembly being transferred into the EVST from the EVTM to less than 200 mrem/hr at the surface. The port penetrations through the closure head are stepped to limit radiation streaming through the gaps. In order to allow sufficient time for inspection and maintenance of the main bearings and seals, shielding is provided to attenuate the direct and scattered radiation levels to less than 125 mrem/qtr.

The EVST internals, storage vessel, and guard tank thicknesses are based on structural considerations, but also attenuate radiation in the radial and downward directions. However, the bulk of shielding to reduce radiation levels in adjacent vaults is provided by the concrete vault walls which are discussed in Sections 3.8 and 12.1.

The fuel transfer port plugs in the EVST head have double, static elastomer seals. Large diameter metallic seals are between the storage vessel and the closure head. The operating floor striker plate has a seal at its mating surface with the side wall vault lining. The turntable driveshafts have double, dynamic elastomer seals. All seals in the EVST are double with capability for convenient leak testing by pressurizing the buffer space between seals in a pair. The effectiveness of the seals does not depend on the presence of a buffer gas, although it would mitigate an inner seal leak.

The EVST is designed with sensors and interlocks to prevent any unscheduled movement of the turntable while the EVTM is mounted on the EVST. The interlock allows the turntable to rotate only when the EVTM grapple is in the full up position. The EVST is designed to prevent excessive relative motion between the head and turntable during an SSE.

Temperature instrumentation and sodium level sensing probes will monitor cooling capability. High EVST sodium outlet temperature, and high or low sodium levels will sound an alarm. Other monitors will be provided in the EVST cooling system (see Section 9.1.3). Sodium leak detectors will monitor the space between the storage vessel and the guard tank. An argon cover gas activity monitor will be provided. An area monitor of the gamma scintillation type will measure the gamma radiation on the RSB operating floor above the EVST. The EVST design also includes the capability to add a temporary neutron detection system for confirmatory monitoring during EVST loading.

9.1.2.1.3 <u>Safety Evaluation</u>

The minimum center-to-center separation distance between storage tubes and the 9 storage positions permanently filled with B_4C will keep the storage array subcritical even if the EVST were completely loaded with new fuel assemblies of the highest reactivity. The B_4C neutron absorbers are designed such that they cannot be removed inadvertently, i.e. cannot be removed with the normal refueling equipment. Based on the calculations reported below the K_{eff} of this array, either with sodium or void of sodium, will be less than 0.95, as required.

Criticality calculations using a three-dimensional multigroup Monte
54 Carlo method (KENO-IV Code, see Appendix A) yielded a K_{eff} = 0.922±
0.025 for the upper limit EVST loading of 650 fresh core fuel assemblies
with the highest enrichment. Previously performed K_{eff} calculations
with fuel of lower fissile plutonium enrichment in an EVST lattice with-
out absorbers yielded close agreement between the two-dimensional,
multigroup diffusion method (DOT Code, see Appendix A) and the
54 51 44 KENO-IV Code. The uncertainties represent 95% confidence limits.

27

Calculations of the criticality of the EVST void of sodium were made using
a two-dimensional, multigroup, diffusion method (DOT codes, see Appendix A).
A k_{eff} value of 0.69 $\pm$ 0.04 was obtained. Virtually the same k_{eff} would be
obtained if argon, nitrogen, or air were to replace the sodium. The RSB
design and elevation of the operating floor prevent the entry of flood
or rainwater into the building. No significant amounts of fluids having
greater moderation than sodium are used in either the EVST or its sodium
44 cooling system.

The storage tubes are held in place at the upper end by a top grid
plate which precludes insertion of any other new fuel assembly between storage
tubes. Seismic forces will not change the separation distance between fuel
44 assemblies due to the restraints between storage tubes. The EVST closure head
assembly has sufficient strength to absorb the heaviest loads carried above
20 it without changing the separation lattice of the storage tubes.

Adequate shielding for radiation protection on the operating floor
is provided by the EVST design. Overall dose rates on the operating floor
are less that 0.2 mrem/hr which permits routinely occupied area access
49 (see Section 12.1).

Transient dose rates due to high-powered spent fuel assemblies are
less than 200 mrem/hr at the surface of the striker plate port. The inte-
49 grated doses are below the limits for unrestricted access (see also sub-
section 9.1.4).

Radiation shields near the EVST bearings and seals limit the dose to
125 mrem/quarter integrated over the time required to perform bearing or
seal maintenance. Additional safety factors are provided by the fact that
maintenance is not expected to be required more than once in three years.
The streaming dose rates from gaps around fuel transfer port plugs are less
than 2 mrem/hr, in accordance with the shielding criteria of 12.1.2.1. The
44 vault walls and floor provide radial and bottom shielding, as described in
49 20 Section 12.1.

The EVST has adequate seals to prevent excessive radioactive emis-
sions into other areas of the RSB. Section 15.5.2.4 analyzes a limiting
case and shows it to be acceptable. Radioactivity released from the EVST
does not exceed the limits specified in Sections 12.1.1 and 12.1.2. The
49 RSB has radioactivity monitors above the EVST to monitor radioactivity levels to
59 detect accidental release and to sound alarms.

The design prevents movement of the turntable sufficient to cause
20 failure of the CCP or to damage a new or spent fuel assembly due to either a
seismic event or inadvertent rotation.

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Dec. 1980

59 PAGES 9.1-10 through 9.1-19

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9.1.3 Spent Fuel Cooling and Cleanup System

There are two locations in the plant where spent fuel is stored. They are the Ex-Vessel Storage Tank (EVST) and Fuel Handling Cell (FHC), both of which are located in the Reactor Service Building.

The description of the spent fuel cooling and cleanup systems associated with these spent fuel storage systems is presented in this section.

9.1.3.1 Ex-Vessel Storage Tank Cooling and Cleanup System

Cooling and purification of the sodium within the Ex-Vessel Storage Tank is accomplished by the EVS Processing System, shown in Figure 9.3-3. This system is a subsystem within the Auxiliary Liquid Metal System, the balance of which is described in Section 9.3. The design bases, design description, safety evaluation and inspection and test requirements discussed in this section are applicable to the sodium cooling aspects of the EVST and to the EVS Processing System.

9.1.3.1.1 Design Bases

Sufficient cooling capacity is provided in the EVST to handle the unlikely event of a complete core unloading occurring with two annual loads of spent fuel assemblies already in the EVST.

The EVST is designed to maintain the sodium coolant level at a height which permits continued cooling of the spent fuel assemblies under normal conditions and in the extremely unlikely event of a primary vessel or cooling system leak or rupture. The system provides three independent means of heat removal, each of which can provide the required cooling. The system is designed such that no single failure, or operator error, can result in loss of two heat removal
59 paths.

Either of the EVS Sodium Processing System's two normal forced cooling circuits provide the capability to remove 1800 kW of heat while maintaining an EVST exit sodium temperature of approximately $510^{\circ}F$. The third (backup) cooling circuit provides the capability to remove 1800 kW while maintaining an exit temperature less than $775^{\circ}F$.

In the extremely unlikely event that both normal EVST heat removal circuits are unavailable due to a combination of an initiating event (active or passive failure) followed by an active failure, heat will be removed by a third (backup) natural convection heat removal circuit.

The system provides the capability to maintain the oxygen content of the sodium in the EVST at, or below, 5 ppm. The cold trap used for this service is separate from those used for reactor and primary loop sodium purification. Their removal and storage procedures would be the same (see Section 9.3.2.2.1); however, the EVS cold trap is expected to last for the life of the plant and would not be removed unless there was a serious malfunction.

Liquid metal NaK is used as the secondary coolant in each of the three EVS cooling loops. It has been used extensively in other plants (SIR, EBR-I, EBR-II, SRE, Fermi, Hallam, and FFTF) and there are no reports of any inherent problems with it. Oxygen in the EVS NaK cooling loops initially or entering the system later is removed by diffusion type cold traps (one on each loop). These are simple appendages on the NaK piping that are maintained at a lower temperature by forced air circulation. The temperature gradient and precipitation of impurity (primarily sodium oxide) establishes a concentration gradient which is the driving force for the transport of the impurity into the cold trap. Each of the diffusion cold traps is sized to store more than four times the maximum volume or anticipated impurities to be collected.

The system, working in conjunction with the Primary Sodium Storage and Processing System described in Section 9.3-2, provides a means of removing reactor decay heat in the event of loss of normal heat removal paths. These two systems, operating together, provide the Direct Heat Removal Service (DHRS). The DHRS is sized to limit the average bulk primary sodium temperature to approximately 1140oF when the DHRS is initiated one-half hour after reactor shutdown. Under this condition, all primary pump pony motors are assumed operational. When the DHRS is initiated twenty-four hours after shutdown, the average bulk primary sodium temperature is maintained below 900oF, assuming operation of a single primary pump pony motor. Total heat rejection capability of the EVS Sodium Processing System is based on removal of the required reactor decay heat in addition to the heat generated by spent fuel within the EVST. The maximum simultaneous EVST and reactor decay heat load is approximately 11-1/2 MW, with DHRS initiated one-half hour after reactor shutdown.

9.1.3.1.2 Design Description

The EVST design and operating decay heat loads and sodium coolant outlet temperatures are given in Table 9.1-1.

The major assemblies of the EVST important to decay heat removal, other than the cooling system itself, are the storage vessel, the guard tank and the internals. The internals, specifically the turntable, separate and support the spent fuel assemblies (contained in sodium-filled CCPs) permitting them to be satisfactorily cooled. The structural design of the turntable has already been discussed in Section 9.1.2.1.

The storage vessel has been classified as Safety Class 2 and is to be designed, fabricated and inspected in conformance with the appropriate codes and standards (see Section 3.2) to provide a leak-proof containment for the sodium coolant. The sodium level is maintained at a high enough elevation so that normal fluctuations due to changes in temperature or number of stored components do not uncover the top of the CCPs in which the spent fuel is stored. This level also prevents the loss of any of the EVST cooling systems should a leak occur from the vessel which floods the EVST guard tank. Should a leak occur in any of the EVST cooling systems, the resulting loss of sodium and EVST level drop will not result in disabling the other functional cooling circuits. During such leak events, the resulting sodium level outside the CCP's is in excess of that required to remove heat from the stored fuel assembly, at its highest possible location within the storage vessel, and to keep the fuel cladding temperature below the limits specified in Table 9.1-2. The EVST sodium inlet lines contain antisyphon devices which prevent a cooling system leak from lowering the vessel sodium below the minimum safe level. The EVST guard tank is sized to contain sodium leaked by the storage vessel and maintain the sodium level above the minimum safe level.

The EVS Processing System includes two independent forced convection cooling circuits, designated circuit Nos. 1 and 2, each of which can remove the required EVST heat loads.

During normal operation, one forced convection circuit is used for EVST cooling and the other is on standby. Each of the circuits is composed of two loops, one a sodium loop and the other a NaK loop. The sodium loop circulates sodium from the EVST through a sodium-to-NaK heat exchanger and back to the EVST. The NaK loop circulates NaK through the exchanger where it picks up EVST heat, to a forced-draft airblast heat exchanger, for dissipation of heat to the atmosphere, and back to the sodium-to-NaK heat exchanger. The system also includes a cold trap to provide purification of the EVST sodium.

The two forced convection cooling circuits are supplied with Class 1E electrical power. Standby electrical power is provided for both circuits in the event of loss of normal power (see Section 8.3.1.1.1). Standby power is supplied to the two circuits by different diesel generators.

In addition, the EVS Processing System consists of a third independent natural convection backup cooling circuit designated No. 3 which can also remove the required EVST heat loads. In the extremely unlikely event of loss of both normal cooling circuits, the backup natural convection cooling circuit is used to remove the required EVST heat loads. Sodium circulates from the EVST through a backup sodium-to-NaK heat exchanger and back to the EVST. The NaK loop circulates NaK through the exchanger where it picks up EVST heat, to a natural-draft heat exchanger, for dissipation of heat to the atmosphere, and back to the sodium-to-NaK heat exchanger.

The sodium nozzles in the vessel are all located in the upper elevation of the EVST vessel wall (see Figure 9.1-6). The cooling circuit inlet lines contain anti-siphon devices which would prevent a cooling system leak from lowering the EVST sodium level and disabling the other functional cooling circuits. The anti-siphon device is cleaned semi-annually, and it will be periodically functionally tested.

The EVST sodium outlet downcomers within the EVST terminate at different elevations above the stored fuel. Loop #2 (forced circulation) has two outlets; the highest outlet used for normal operation, and a second outlet at a lower elevation such that any sodium leakage from Loop #1 (forced circulation) will not uncover the Loop #2 outlet. Loop #1 has one outlet nozzle located at an elevation between the Loop #2 nozzles. The lower Loop #2 nozzle would be used only in off-normal conditions when both Loop #1 and the higher Loop #2 flow paths will not function. The third (backup) cooling circuit (Loop #3) has one outlet located below all Loop #1 and Loop #2 nozzles such that the Loop #3 outlet will not be uncovered by a leak in either Loop #1 or Loop #2. A leak in the Loop #3 piping will not uncover any of the other loop outlets because it is entirely elevated above its EVST nozzles.

The entire EVS processing system includes the following components:

EVST Sodium Pumps (2)

EVST Sodium Coolers (2)

EVST Backup Sodium Cooler (1)

EVST NaK Pumps (2)

EVST NaK Airblast Heat Exchangers (2)

EVST NaK Natural Draft Heat Exchanger (1)

EVST NaK Expansion Tanks (3)

44 EVST NaK Diffusion Cold Traps (3)

26 DHRS NaK Expansion Tank

EVST NaK Storage Vessel

EVS Sodium Cold Trap

44 Interconnecting Piping and Valves

All pumps, both for sodium and NaK service, are electromagnetic pumps. Heat exchangers are all-welded units. All pressurized fluid containment boundary components are of 300 series stainless steel. The normal cooling circuits plus the sodium loop of the backup cooling circuit are designed and fabricated to the requirements of Section III of the ASME Boiler and Pressure Vessel Code for Class 2 components. The backup cooling circuit EVST NaK expansion tank, the natural draft heat exchanger, and the NaK piping are designed and fabricated to the requirements of Section III of the ASME Boiler and Pressure Vessel Code for Class 3 components. All components are classified Seismic Category I and are housed in hardened
44 structures.

Each of the cooling circuits (the sodium and its associated NaK loop) are physically separated to preclude common failure due to potential accidents such as liquid metal leakage or fire. In addition, the sodium loop within each circuit is enclosed in an inerted cell to avoid possibility of a significant radioactive sodium fire. Shielding is provided to permit access for inspection or maintenance of any loop while the other loops remain operational. The cold trap is shielded separately from all three loops (see Section 12.1).

a) <u>Normal System Operation for Cooling of the EVST</u> – The
44 principal function of the EVS Processing System is to provide cooling for the EVST. Since its operation to provide reactor core heat removal is expected to occur, if at all, only once during plant life, EVST cooling is the normal mode of system operation. During normal EVST cooling operation
44 sodium is circulated at 400 gpm through one of the two normal sodium loops from the EVST to the EVST sodium pump, through the NaK-cooled EVST sodium cooler, and back to the EVST
44 (see Figure 9.1-10). A bypass flow of 60 gpm is circulated through the EVS sodium cold trap for purification, and 3 gpm through the plugging temperature indicator (provided by the Impurity Monitoring and Analysis System, described in Section 9.8).

At the design flow rate of 400 gpm, the normal loop is
designed to remove 1800 kW with EVST sodium inlet and
46 20 outlet temperature of 400 and $\sim$510^{o}F, respectively. The
EVS sodium cold trap can be valved into either of the normal
loops to provide essentially continuous EVST sodium purification.
On the basis of anticipated in-leakage, the cold trap
will maintain the oxygen content below 5 ppm in the EVST.

44

Normally, heat from EVST sodium is transferred in the EVST sodium
cooler to the associated NaK loop. NaK is pumped through the shell-
side of the sodium cooler and then to the airblast heat exchanger and
46 59 20 returned to the sodium cooler. With an EVST heat load of 1800
kW, NaK is circulated at a flow rate of approximately 400 gpm.
As the EVST heat load decreases with time, the ABHX flow rate
is decreased; air flow is controlled to maintain the EVST
sodium inlet temperature at 400^{o}F. The NaK cold leg temper-
ature, exiting from the airblast, is maintained below 400^{o}F by
controlling either the airblast fan speed or the setting of
air inlet dampers. NaK volumetric changes from temperature
44 variation are accommodated in the NaK expansion tank connected
to the high point of each loop. Oxide level in each NaK loop is
59 minimized by the diffusion cold trap.

On a periodic basis, EVST cooling is manually transferred
from normal Circuit No. 1 to Circuit No. 2 in order to equalize
operating time. The circuit not in use (both sodium and NaK
loops) is maintained full and at operating temperature, $\sim$400^{o}F
44 to be ready for immediate use, and is called the standby circuit.

During normal EVST cooling operation, the "crossover piping"
shown in Figure 9.1-10 is isolated by closed valves at each
of the EVST loops.

59

b) <u>Off-Normal System Operation for Cooling of the EVST</u> - In the
extremely unlikely event of loss of both normal cooling cir-
cuits, the dampers on the natural-draft heat exchanger will
47 be manually opened to initiate natural draft air flow. This
will induce NaK flow through the tubes of the natural-draft
heat exchanger and the shell of the backup sodium cooler.
This in turn induces sodium flow from the EVST through the
tubes of the backup sodium cooler and back to the EVST.

44

The third (backup) cooling circuit is designed to remove 1800 KW of heat while maintaining sodium temperatures within the EVST below 775°F. The damper position on the natural-draft heat exchanger is adjusted to control EVST sodium temperature under various heat loads. NaK volumetric changes from temperature variations are accommodated in the NaK expansion tank connected to the high point in the NaK loop. Oxide level in the NaK loop is minimized by the diffusion cold trap.

The backup circuit is maintained in a preheated condition and is ready for immediate operation. A small sodium and NaK flow is induced by slightly opening the dampers of the natural-draft heat exchanger and removing a minimum amount of heat during normal EVST cooling by circuits 1 or 2. Trace heaters are also provided for the sodium
59 piping and components.

c) <u>Operation During Reactor Decay Heat Removal</u> - Reactor decay heat removal is accomplished by using the combined heat removal capability of both of the normal NaK loops in the EVS Processing System. In conjunction with the primary sodium overflow heat exchanger, each of the NaK loops circulate approximately 400 gpm through each airblast heat exchanger. The circuits are interconnected during this operating mode to provide a total NaK flow of 800 gpm which is routed through the shell-side of the overflow heat exchanger in the Primary Sodium Storage and Processing System (described in Section 9.3). When the DHRS is initiated one-half hour after reactor shutdown, the NaK exits from the overflow heat exchanger at a maximum temperature of approximately 1000°F. At this temperature, the NaK airblast heat exchangers have a combined heat removal capacity of approximately 11-1/2 MW, which is sufficient to remove decay heat from both the reactor and the EVST.

During the DHRS mode of operation, one of the normal Na/NaK loops remains in use for EVST cooling. The NaK in this loop is circulated (400 gpm) from the airblast heat exchanger through the EVST sodium cooler prior to its flow to the overflow heat exchanger. The EVST, in effect, is cooled in series with the overflow heat exchanger. In this flow pattern, the EVST is located in the cold leg of the loop in order to minimize temperature rise of the EVST sodium. The sodium and NaK flow path in this mode of operation is shown schematically in Figure 9.1-11. Switchover from normal cooling (EVST) only to reactor decay heat removal (DHRS) is done remotely from the control room. Switchover is accomplished by opening the isolation valves at the connections to each of the normal EVST cooling loops. The DHRS NaK expansion tank is isolated and the EVST NaK pump is increased to 400 gpm each. The cover gas space in the two EVST NaK expansion tanks is cross-connected to equalize tank NaK levels.

9.1.3.1.3 Safety Evaluation

The EVST cooling capability can be provided by either of two identical, forced convection cooling circuits, each of which can remove 1800 kW while maintaining a maximum EVST sodium outlet temperature of $\sim$510°F.

In the extremely unlikely event that the normal circuits are unavailable, heat will be removed through a third independent (backup) natural convection cooling circuit. At 1800 kW this backup cooling circuit will maintain sodium temperatures within the EVST below 775°F.

The critical temperature in a fuel assembly, from the standpoint of safety, is the peak fuel cladding temperature. The normal and emergency limits are given in Table 9.1-2.

The peak fuel cladding temperatures shown in Table 9.1-2a, are within the limits. Hence, no damage to the stored fuel assemblies will occur.

The codes and standards to which the EVST vessel and the surrounding guard tank are designed and fabricated assure that leakage of sodium will be a very low probability event. At the minimum level, adequate cooling is maintained with no temperature increases from those shown in Table 9.1-1.

The EVS Processing System components are designed to accepted industrial and nuclear standards to insure structural integrity and operational reliability. The components, applicable design code and class, plus their seismic adequacy are listed in Table 9.3-1. Design temperatures and pressures are given in Table 9.3-7.

Each of the three sodium cooling loops is designed against the possibility of common-mode failure. Two pump suction lines are provided within the EVST for normal sodium circuit No. 2. The open end elevation of each is different, one high, one low. Each of the two lines is separately valved externally to the EVST. After the initial fill of the loop, the isolation valve in the low suction line is locked closed and remains closed (except for periodic testing) throughout the plant life. This low suction line is used only in the event of a major loop or vessel rupture. One pump suction line is provided within the EVST for normal cooling circuit No. 1. The open end elevation of this line is between those for circuit No. 2. This line is valved externally to the EVST, and is called a "high" pump suction line. During normal system operation, one of the normal cooling loops is operated using the "high" pump suction line. The suction line(s) in the standby normal loops are closed. In the event of a major failure (rupture) of the operating normal sodium cooling loop, the isolation valve in the pump suction line is closed by operator action from the control room, signalled by concurrent alarms, indicating low level in the EVST and a sodium leak within the cooling loop cell. If the isolation valve should not be closed the EVST sodium level could only be siphoned to the (high) pump suction outlet within the tank. Siphoning from the return line is prevented by an antisiphon vent in this line within the EVST. Each normal cooling loop return line antisiphen vent is located just below the normal EVST sodium level. In the event of a piping leak a spill drop in EVST sodium level will expose this vent which will ingest EVST argon cover gas and break the siphen. This will prevent uncovering an EVST suction line and permit EVST cooling to continue. If a failure of normal cooling loop occurs, as described previously, the standby normal cooling circuit can be immediately activated by valving in its lower pump suction and increasing pump flow to the design rate of 400 gpm. A leak anywhere in the EVS cooling system or the EVST cannot disable more than one of three cooling loops. See Figure 9.1-12A.

In the extremely unlikely event that the second normal loop cannot be activated after the first loop has experienced a failure, the third (backup) circuit will be brought into operation. One suction line is provided within the EVST for the backup cooling circuit. The open end elevation of this suction line is below the lower suction line of normal cooling circuit No. 2. Flow back to the EVST is through the fill/drain line. Siphoning from this return line is prevented because the entire backup loop is elevated above the sodium level in the EVST. The drain line for the Na heat exchanger in the circuit has a removable pipe spool at an elevation above the EVST sodium level to prevent siphoning through this circuit.

Failure of any component, in any of the sodium or NaK loops, can cause loss of only the circuit in which it is located. The normal standby or backup cooling circuit can then be put into operation within minutes to provide essentially continuous cooling of the EVST sodium. The potential radiological consequences of an extremely unlikely release of EVST sodium to an inerted cell is described in Section 15.

All components of the normal sodium and NaK loops which require electrical power are on the Class IE power system, to ensure continuous EVST cooling and reactor decay heat removal. In the event of complete loss of external power to the plant, power to both of the normal cooling circuits is provided by the plant diesels. Immediate activation of the diesel-powered supply is not necessary for the EVST sodium pumps since the sodium volume within the EVST provides a heat sink to minimize sodium temperature rise during loss of circulation. Sodium circulation can be lost for approximately 2 hours before the maximum sodium temperature in the upper portion of the EVST reaches 600^{o}F. Activation of the emergency power supply to the NaK pumps and airblast fans is required within 1/2 hour, however, to ensure the availability of DHRS for reactor decay heat removal.

The only "active" component in the backup loop is the damper on the natural draft heat exchanger. It is operated manually and, therefore, does not require connection to the emergency power system.

Isolation of all of the cooling circuits (sodium plus the associated NaK loop) in separately shielded, inerted cells precludes both radioactive sodium fire and the possibility of any failure in one loop imparing the operability of the other.

Isolation valves are provided in the suction and return lines to
44 each of the normal sodium loops, to permit loop isolation for inspection or maintenance. Prior to personnel access to a cell, the isolation valves will be closed and, if necessary (depending on Na activity level), the loop may
26 be drained. The loop isolation valves and loop high-point vents are located higher than the sodium level in the EVST. Thus, once the loop is vented
44 and drained, siphoning of the EVST cannot occur even if the isolation valves are accidentally opened during a maintenance operation.

The high-point vent in the sodium loop of the backup cooling circuit allows sodium to drain back to the EVST for inspection and maintenance. Since the entire backup loop is located higher than the sodium in the EVST,
44 siphoning cannot occur.

Instrumentation is provided to monitor and alarm off-normal con-
44 ditions in the sodium and NaK systems. The off-normal conditions include high temperature, low flow, and external leak detection. The operating pressure of the NaK system is maintained higher than that of the sodium system. Leakage of NaK-to-sodium is monitored and alarmed by abnormal level indication in the NaK system expansion tank, in conjunction with the level in the EVST.

9.1.3.1.4 Inspection and Test Requirements

Leak checks will be made on all of the systems prior to filling
59 with Na or NaK. Prior to spent fuel loadings, the system will be operationally tested to determine that the system will perform within design limits.

The equipment containing Na will be placed in inert atmosphere
59 cells that will be accessible for inspection. The three independent cooling loop components are separated by shield walls so that inspection and maintenance can be performed with the other loops remaining operational.
44

An in-service inspection device will be used to periodically check the structural integrity of the EVST vessel. Space for such a device is provided by allowing sufficient clearance between the storage vessel and guard vessel.
44 59

The two NaK airblast heat exchangers and NaK Natural Draft Heat
44 Exchanger will be located in air atmosphere cells and will be available for periodic visual inspection.

9.1.3.1.5 Instrumentation Requirements

Instrumentation and controls (I&C) are provided for operation, performance evaluation, and diagnosis of the EVS Sodium Processing System. These functions are required for off-normal, as well as for the full range 1 operation. Details of the I&C for the subsystem are shown in the piping and instrumentation diagram, Figure 9.3-3. DHRS instrumentation is discussed in Section 5.6.2.1.6. The following I&C is required to ensure safe operation of, and to prevent damage to, the EVS Sodium Processing System.

Temperatures at the inlet and outlet of all heat source and sink components, in conjunction with loop flow measurements, are provided for all systems to monitor their status. Critical temperatures and flows are alarmed to alert the operator to off-normal operations. All EM pumps are provided with winding temperature measurements and winding coolant low flow indication. These measurements are alarmed for off-normal conditions, and interlocked to automatically shut down the pump to prevent damaging it.

The EVST and the NaK expansion and storage tanks are provided with level measurements, which are alarmed for abnormal low and/or high level. This information, in conjunction with leak detection data, is utilized to diagnose external liquid metals leaks. The operator is alerted to NaK to sodium leakage by NaK expansion tank high-low level alarms. A differential pressure sensor and flow meter are provided to alert the operator to possible plugging of the cold traps or insufficient cold trap flow. All the bellows seal valves are provided with leak detectors (Section 7.5.5.1). All valves are provided with position indicators. Accessories (solenoids, pressure gauges, I/P converters, etc.) for remotely operated valves installed in inerted cells are installed off of the valves in adjacent accessible air cells. The stem portion of the sodium valve is monitored and alarmed for low temperature, to ensure free operation and protect the valve sodium seal from damage. To provide for continued operation and prevent possible system damage resulting from control system failures, hand controllers are provided for all controllers. The hand controller allows the operator to manually operate the system while the defect is repaired.

9.1.3.2 Fuel Handling Cell Spent Fuel Storage Cooling System

Decay heat generated by spent fuel assemblies stored in the spent fuel transfer station within the FHC is removed by natural convection heat transfer to the FHC argon atmosphere.

When fuel is transferred using the crane, a gas cooling grapple is used to remove decay heat from the assembly. The design basis, description, evaluation, and inspection and test requirements for these systems are discussed in the following sections.

9.1.3.2.1 Design Bases

Sufficient cooling capacity will be provided in the spent fuel transfer station to remove a maximum heat load of 18 kW from a total of three fuel assemblies in sodium-filled core component pots. During normal operation, a maximum of two fuel assemblies will be present in the FHC; the third position is used for transfer and will normally be empty or contain an empty CCP. Normally, the decay power of fuel assemblies handled in FHC will be limited to ≤6 kW each. However, under unusual conditions, it may be desired to examine a short-cooled assembly, i.e., with a decay power greater than 6 kW, but less than 15 kW. Under this condition, no more than a single fuel assembly shall be permitted in the FHC. The transfer station will be designed to cool a single fuel assembly of up to 15 kW decay heat without exceeding the normal cladding temperature limit.

The gas cooling grapple will have sufficient cooling capacity to maintain the cladding temperature of a fuel assembly below the normal cladding temperature limit with a decay heat load of 15 kW.

9.1.3.2.2 Design Description

The spent fuel transfer station shown in Figure 9.1-8 consists of a lazy susan assembly containing three transfer locations for core component pots (CCPs), a bearing and drive system for the lazy susan, a structural support frame and bracketry, heaters, insulation, and seismic restraints for the lazy susan. The transfer station is designed to ASME III/Sub NF3 and Seismic Category 1 requirements.

The spacing between the storage locations is determined such that adequate natural convection cooling is provided.

The lower portion of each storage location is a tapered cylindrical socket to support the CCP while providing a catch basin for sodium drippage. The cylindrical socket houses heaters on its outside which prevent sodium freezing when storing core assemblies with little or no decay heat.

The decay heat will be removed by natural convection to the FHC argon atmosphere, which in turn is cooled by the redundant argon circulation system. Under the worst case conditions the cladding temperature will not exceed 1100°F.

Cooling of the FHC argon atmosphere is provided by the Argon Circulation System, which has two loops, each consisting of a fan, gas heat exchanger and a piped distribution system. The heat exchanger removes heat from the argon gas and rejects it to the recirculating Dowtherm J System which rejects it to the Chilled Water System (Normal or Emergency as applicable) which in turn rejects the heat to the ambient air through the Emergency Cooling Tower in the emergency mode and through the Normal Cooling Tower in the Normal mode. The argon circulation system and supporting heat removal systems operate during normal plant operation, accident conditions, and periods of normal electrical power failure. The Argon Circulation System and Recirculating Dowtherm J System are Non-Class 1E systems supplied with standby electrical power by the same diesel generator (see Section 8.3.1.1.1). The chilled water system loops are Class 1E systems supplied with standby electrical power by diesel generators. One generator serves the argon circulation system loops, also (see Section 8.3.1.1.1). The low-pressure argon circulation system, including the shell of the cooler, is designed to ANSI B31.1, and Section VIII of the

ASME Boiler and Pressure Vessel Code, Seismic Category I, and is located within a hardened structure.

The crane handled gas cooling grapple, shown schematically in Figure 9.1-9, is mainly used to transfer bare fuel assemblies from the spent fuel transfer station to the spent fuel shipping cask. Design of the grapple finger actuation mechanism prevents actuation of the fingers to release a core assembly while the fingers are supporting the weight of the assembly. The crane hook includes a latch to prevent inadvertent disengagement of a cooling grapple from the hook. In the event of a loss of electric power, the crane will stop at its position at the time of the power failure. Design of the crane includes the capability for manual operation. Access to the crane for manual operation is through ports in the wall and roof closure.

Two redundant argon gas-cooling blowers are mounted on the upper end of the gas-cooling grapple. These blowers draw argon gas from the surrounding cell environment and blow it through the grapple and fuel assembly, discharging it back into the cell through the nozzles at the bottom of the fuel assembly. The argon gas flow rate will be large enough to maintain the cladding temperature of a fuel assembly below the normal cladding temperature limit for decay heat loads up to 15 kW. The blowers are non-Class 1E systems supplied with standby electrical power by the same diesel generator (see Section 8.3.1.1.1).

9.1.3.2.3 Safety Evaluation

A CCP containing a fuel assembly is cooled sufficiently by natural convection of the adjacent FHC atmosphere to maintain the peak fuel cladding temperature below the limits given in Table 9.1-2. The peak temperatures, given in Table 9.1-2A for normal operations in which the FHC atmosphere temperature is maintained by the argon circulation system and for the unlikely event of loss of cooling of the FHC atmosphere, are within the limits.

The argon cooling gas flow rate through the spent fuel assemblies while being handled by the gas cooling grapple is sufficient to maintain the maximum steady-state cladding temperature of a 15 kW fuel assembly below 600^{o}F. In the event of loss of argon cooling gas, sufficient time exists for the assembly to be transferred back to a Na-filled CCP in the spent fuel transfer station within the FHC before the fuel cladding reaches 1500^{o}F.

Adequate cooling of a spent fuel assembly suspended from the cooling grapple is maintained by the following means:

1) The grapple blowers are redundant to protect against loss of cooling capability by failure of one blower.

2) Each blower will be tested before beginning FHC spent fuel shipping operations to ensure its operability.

Evaluation of the loss of power for cooling systems for fuel assemblies in the FHC shows that the consequences are acceptable. In the event of loss of normal offsite power, operation of the cooling blowers would stop and the temperature of a suspended fuel assembly would rise. The loss of power would also prevent movement of the FHC in-cell crane to return the assembly to a sodium-filled CCP. The extent of the temperature rise would depend on the decay power of the assembly and the duration of the power loss. During normal operations with the maximum powered 6-kWt fuel assembly, there would be about 33 minutes before the peak cladding temperature would reach 1500°F. If normal power were not restored before the temperature limit was reached, it is assumed that fission products would be released into the FHC. They would normally be retained within the FHC, however, because the argon circulation system and supporting heat removal systems (supplied with electrical power from an onsite diesel generator), would continue operating and maintain the FHC atmosphere at a negative pressure relative to the surrounding cells. In the unlikely event of failure of the diesel generator supplying the argon circulation system, the FHC pressure would become positive relative to surrounding areas and fission products would leak to the building. No credit is taken in the accident analysis for the seals of the FHC. This event is enveloped by the event discussed in PSAR Section 15.5.2.3.

As an additional safeguard, high-powered assemblies (i.e., >6 kW) will not normally be handled in the FHC (this will be an anticipated event). In addition, special procedures will be followed whenever a fuel assembly with a decay heat between 6 and 15 kW is handled in the FHC. One of these procedures will ensure that no other spent fuel assemblies are being moved within the FHC whenever a fuel assembly with decay heat greater than 6 kW is being transferred or examined in the FHC.

These design features and procedures will ensure continued cooling of spent fuel assemblies during transfer, examination, and temporary storage in the FHC.

Cladding failure, resulting in the release of fission gas might occur; but as shown in Section 15.5.2.3 such a release is well within acceptable limits.

9.1.3.2.4 Inspection and Test Requirements

Prior to spent fuel loading, the FHC Argon Circulation System will be operationally tested to determine that all argon cooling and purification components perform within limits.

9.1.4 Fuel Handling System

The following subsections discuss the refueling procedure and the safety aspects of fuel handling equipment not discussed in the preceding sections of 9.1. The two major machines discussed are the ex-vessel transfer machine (EVTM), and the in-vessel transfer machine (IVTM). The design bases of fuel handling equipment cover the following specific safety aspects: prevention of criticality, sufficient cooling, protection against radioactivity release, prevention of mechanical damage, and provisions for adequate shielding.

9.1.4.1 Refueling Procedure

The sequence for handling fuel begins with receipt of new fuel in the hardened portion of the RSB by truck. The truck to be used for shipping new fuel assemblies will be a Safe Secure Trailer. The fuel assemblies are transported in shielded New Fuel Shipping Containers (NFSC), with six containers per shipment. Each NFSC contains one new fuel assembly (see Figure 9.1-5) and is designed to withstand the hypothetical accident of 10CFR71, characterized by a 30-ft drop onto a flat, unyielding surface. (When handled by the RSB bridge crane, a drop of 30 ft. is not possible since the maximum crane hook height above the RSB operating floor is 41 ft. and an NFSC is hooked on the end and is about 17 ft. long.) Shielding in the container is provided for the NFSC to accommodate the future use of LWR recycled fuel plutonium, which will be radioactive. The RSB crane is used to lift one NFSC at a time from the trailer and place it in a laydown area.

While at the laydown area, the NFSC are checked for damage. They are then transferred individually using the RSB crane to one of the two new fuel unloading stations (Figure 9.1-3). The stations are located between the EVST and FHC and are within the coverage of the EVTM. Each station can contain only one container. After an NFSC is placed in the new fuel unloading station, the NFSC cover is removed to provide access to the fuel assembly. A visual check is made to verify that the correct core assembly is contained in the NFSC.

Each new fuel assembly is inspected before transfer to the EVST for storage pending use. The inspection is performed using the shielded New Core Assembly Inspection Equipment (NCATE) (Figure 9.1-21), which is installed over the unloading station. The fuel assembly is lifted from the container using the davit crane hoist on the EVTM trolley (Figure 9.1-13), inspected, and set down again in the NFSC. The inspection consists of visual and mechanical identification, checking for shipping damage, and other visual and dimensional checks. While these operations are taking place, a second shipping container may be placed in the second unloading station to expedite the inspection operations. The inspection equipment could then be moved to the second station to complete inspection of the second assembly. After both assemblies are inspected, the inspection equipment is moved out of the way. Adequate shielding is provided on the NCATE to accomodate the future use of LWR recycle fuel plutonium (which is radioactive) and meet the requirements and radiation criteria of section 12-1.

The unloading station valve is closed and the station atmosphere changed from air to argon to be compatible with the EVTM and EVST. The EVTM is mated with the shipping container through an adaptor assembly. The EVTM removes the new fuel assembly and transfers it to the EVST. The fuel assembly is lowered into one of the argon filled preheat stations in the EVST. In the preheat station the new fuel assembly is slowly heated by the hot argon to approximately the same temperature as the sodium in the EVST storage vessel. After preheating, the assembly is transferred to a sodium-filled core component pot in one of the storage positions in the EVST. This transfer is accomplished by means of the EVTM. All new core assemblies are thus transferred to the EVST prior to reactor shutdown for refueling. All operations up to this point are performed while the reactor is operating and the equipment hatch between the RCB and RSB is closed. A minimum 1/4" W.G. negative pressure is maintained in the normal atmospheric areas of the RSB during the fuel handling operation. During RSB open hatch refueling operations, the outside air dampers for the RSB HVAC is manually closed and reopened when the 1/4" WG negative pressure in the RSB is achieved.

Refueling, operations of the reactor begin with reactor shutdown and cooldown to refueling temperatures. The control rod drive lines (CRDL) are disconnected from the control assemblies which are fully inserted in the core. The drive lines and the upper internals are then raised to clear the parting plane to allow the reactor head plugs to rotate. Concurrently, the reactor cover gas is purged and purified to reduce radioactivity levels in the gas to a very low level. The equipment hatch between the RCB and RSB is opened while these operations are in progress. In order to provide a minimum 1/4" W.G. negative pressure in the RCB and RSB after the containment vessel refueling hatch is opened, the RSB HVAC system fresh air inlet damper is closed and the RCB supply and exhaust fans are stopped. Once the required negative pressure for the RSB and RCB with open hatch is achieved, the RSB fresh air intake supply will be opened and the RCB supply fan is started to supply the required ventilation rate.

Two adapters and floor valves are installed on the reactor head. Then the fuel transfer and in-vessel transfer machine (IVTM) port plugs are removed using the EVTM and the auxiliary handling machine (AHM) respectively.

The AHM then installs the in-vessel section of the IVTM into the IVTM port in the small rotating plug (SRP) of the reactor head. Following this, the IVTM adapter and floor valve are removed and the drive section of the IVTM is installed with the RCB polar crane and connected to the in-vessel section. The reactor is now ready for refueling.

The EVTM and the IVTM work in conjunction during refueling to exchange spent core assemblies in the reactor with new core assemblies stored in the EVTS. Two operations occur simultaneously - the EVTM removes a new

core assembly in a sodium filled core component pot (CCP) from the ex-vessel
44 storage tank, while the IVTM removes a spent core assembly from the core and
deposits it in a CCP located in a transfer position outside the core, but
still within the reactor vessel. The EVTM then places the CCP containing the
44 new core assembly in the second transfer position inside the vessel, the
rotating guide tube rotates to the first transfer position, and the EVTM
44 removes the CCP containing the spent core assembly from the first transfer
position. During this time, while the EVTM is at the reactor, the IVTM and
rotating plugs must remain stationary. The EVTM then moves away from the
44 reactor to deposit the CCP with the spent core assembly in the EVST, while
the IVTM, operating in conjunction with the rotating plugs, installs the new
44 core assembly in the open core lattice position. These operations are
repeated until the refueling is completed.

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Spent blanket and control assemblies are exchanged first to increase
44 core negative reactivity. Then spent fuel assemblies are exchanged in inverse
59 51 order of decay power. As a consequence of this refueling sequence, time is
permitted for decay, thus reducing the decay heat which must be removed by the
EVTM during fuel handling. This sequence reduces the maximum spent fuel
assembly decay power handled by the fuel handling equipment to ~15 kw compared
to the 20 kw capability.

59 51 44

After refueling is completed, the IVTM is removed from the reactor and stored in the RCB. The fuel transfer and IVTM port plugs are reinstalled in the reactor head and both floor valves and adapters removed to their respective storage areas in the RCB. The reactor upper internals are lowered, and CRDL's are reconnected to the control assemblies. Finally, the hatch between the RCB and RSB is closed and leak checked, the RCB and RSB HVAC systems are returned to the normal operating mode, and the reactor is ready for startup.

After spent fuel has decayed for ~100 days in the EVST, it may be loaded into the spent fuel shipping cask. Control, radial shield, and some low-power blanket assemblies can be shipped offsite before the 100-day

cooling period, but fuel and high-power blanket assemblies are held until
59 20 they decay to less than 6 kw.

44 Spent assemblies to be loaded into the spent fuel shipping cask are transferred in sodium-filled CCP's from the EVST to the fuel handling cell by the ex-vessel transfer machine. In the FHC, they are stored temporarily in a three-position spent fuel transfer station. The assem-
44 blies are removed, one at a time, from the transfer station by a gas cooling grapple, the exterior is dimensionally and visually examined if desired, and residual sodium is drained prior to loading.

The spent fuel shipping cask (SFSC) is brought onsite by a special railroad car. The cask is removed from the railroad car, lowered down a shaft onto a transport dolly by the Reactor Service Building crane, and the outer containment shipping cover removed. The dolly moves the cask under the fuel handling cell floor, where it is sealed to the bottom of the cell. An access plug in the floor of the cell is removed by an in-cell crane, and spent
59 fuel assemblies are loaded into the cask. The cask is then decoupled from the FHC, the shipping cover is reinstalled, the cask is removed from the FHC shaft, loaded onto the rail car and checked for radioactive contamination prior to shipment. The cask is then shipped to the reprocessor.

The reprocessed fuel material is provided to the fuel fabricator, who eventually returns new fuel assemblies to the CRBRP, thus completing the fuel cycle.

44 51 There are 385 core special assemblies provided in the CRBRP. They are used for:

1) Initial reactor core loading,

2) Reactor core unloading,

44 3) Lower inlet module replacement (see PSAR Section 4.2.2.1.1.2), and

4) Removing jammed fuel assemblies (see PSAR Section 9.1.4.4.3).

44 Prior to the first reactor startup, the core special assemblies are
59 replaced on a one-for-one basis by new control, blanket, and fuel assemblies.
44 If the core is unloaded, the core special assemblies are inserted for spent core assemblies on a one-for-one basis. During initial core loading the new
44 and clean core special assemblies can be installed using the RCB polar crane since the reactor core is open to the RCB air atmosphere and does not contain sodium at this time. At any other time, they must be handled by the reactor 25
refueling system equipment.
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After removal from the reactor core, the sodium wetted core special assemblies are first brought to the EVST, and are later transferred into the FHC. 162 of these core special assemblies simulate fuel assemblies in the core and have full-flow filters. Some of these assemblies are partially disassembled in the FHC and made ready for sodium removal performed in the large component cleaning vessel. All other core special assemblies are only inspected.

Whenever core special assemblies are handled by refueling equipment, they are accounted for using the same inventory control system as "real" core assemblies. Before entering and after leaving the reactor core lattice they are electromechanically identified by the IVTM using identification notches (see Section 9.1.4.4.2). In the FHC, core assemblies are identified and differentiated both visually and electromechanically. The core special assemblies leave the FHC in a polyfilm wrapped transfer rack. The outer surface of this polyfilm wrap is checked for radioactivity immediately after sealing and leaving the FHC port. The core special assemblies are transferred from the FHC to the Large Component Cleaning Vessel (LCCV) located in the RCB for sodium removal. Cleaned core special assemblies are packaged in polyethylene bags, loaded into holding transfer racks, and transferred to a storage area.

The physical difference of identification marks between special and real core assemblies, the positive identification of core assemblies at two locations, and the radiological monitoring of core special assemblies before cleaning them are regarded as sufficient safeguards to insure that no real fuel assemblies are mistaken as special ones, and stored in a storage facility not designed to receive them.

The maximum pressures and temperatures of fuel handling equipment during normal operations and off-normal design basis events are listed in Table 9.1-2B. The values are within the limits in the same table.

9.1.4.2 Deleted

9.1.4.3 Safety Aspects of the Ex-Vessel Transfer Machine (EVTM)

The primary function of the EVTM is to transfer core assemblies between the reactor, EVST, new fuel unloading station, and FHC. The EVTM is designed to handle both new and irradiated core assemblies in sodium-filled CCPs and bare new core assemblies. The EVTM has the following capabilities:

1) Grapple and release core assemblies, CCPs, and port plugs
2) Raise and lower core assemblies, CCPs and port plugs
3) Provide containment of radioactive cover gas
4) Maintain an argon environment
5) Maintain preheat temperature for new core assemblies
6) Provide up to 20 kw cooling for spent fuel assemblies
7) Provide radiation shielding

The EVTM is a shielded, inerted, single-barrel fuel handling machine. The EVTM is mounted on a trolley, which, in turn, is positioned on rails on top of the gantry. The gantry moves on crane rails between the Reactor Containment Building (RCB) and the Reactor Service Building (RSB). The trolley rails are perpendicular to the gantry rails, allowing complete indexing of the EVTM. The EVTM mounted on its gantry is depicted in Figure 9.1-13.

A perspective cutaway view of the EVTM is shown in Figure 9.1-14. The EVTM is similar in concept to the FFTF Closed Loop Ex-Vessel Machine (CLEM), but is considerably shorter. The EVTM is 35 ft high and weighs 240 tons, including the gantry. The span of the gantry is 30 ft between the rails.

The EVTM is modular in construction to limit the size and weight of the subassemblies for better maintainability and fabricability. Major subassemblies are: grapple drive system, several cask body modules which provide structural support and shielding, service platforms, a view port assembly to permit visual inspection of CCPs, port plugs, or drip pans, a drip pan assembly to collect sodium drippage, an extender which raises and lowers the closure valve, a closure valve which mates with the floor valve and provides the gas tight seal, the cold wall assembly and electrical, instrumentation, and control equipment.

Irradiated core assemblies are handled by the EVTM only in sodium-filled CCPs. New core assemblies are handled both in CCPs (between the reactor and the EVST) and bare (between the new fuel unloading stations and the preheat stations in the EVST and between the preheat stations and the storage stations in the EVST). Different grapples are required for the bare assembly and CCP. Grapple changing is fairly straightforward, and is done in the FHC using the designs and techniques developed for the FFTF CLEM. The CLEM, in turn, is similar to fuel handling machines used successfully at Hallam Nuclear Power Facility and at Fermi-1 (the replacement machine).

The EVTM is connected at each fuel transfer port by lowering the closure valve and mating with the floor valve. The sealing at this interface is accomplished by double seals on the top of the floor valve. The space between the seals is then pressurized with argon buffer gas. The air trapped within the sealed interface space between the valves is purged by pressure-vacuum cycling. First, vacuum is drawn by the plant radioactive vent system, then pressure is applied from the argon supply system. This cycling is repeated a sufficient number of times to reduce the oxygen concentration to an acceptable level. The closure valve and floor valve are then opened, and the drip pan pot is rotated out of the way. The grapple is then driven down to pick up a CCP or a bare core assembly.

59 | The grapple is lowered into the handling socket of the CCP by the chain drive system. The grapple fingers are actuated, engaging the lip on the handling socket. The CCP containing the new or irradiated core assembly is then raised into the EVTM cold-wall assembly. After the CCP is fully raised, the drip pan is closed to prevent sodium drippage on the closure valve or floor valve. The closure valve and floor valve are then shut, and radioactive gas trapped between the valve interface seals is purged by the vacuum-pressure cycling described previously. The extender is then retracted, permitting the EVTM to move to another location. The closure valve and drip pan module are based on the same design and operating concept as the floor valve described in Section 9.1.4.6.

The safety aspects of the EVTM that will be covered in this section are: radiation shielding, prevention of radioactive releases, cooling of spent fuel, and prevention of mechanical damage.

9.1.4.3.1 Design Basis

Adequate shielding for radiation protection is provided in the design of the EVTM to meet the radiation protection requirements of 10CFR20.

The activity released from a damaged or leaking spent fuel assembly while in the EVTM is contained in the EVTM by proper sealing or welding of penetrations and openings. Radioactive leakage and diffusion through seals are well below the limits specified in 10CFR100.

Sufficient cooling capacity is provided in the EVTM to cool spent fuel assemblies with up to 20 kw of decay heat in sodium-filled CCPs and to ensure that fuel cladding temperatures do not exceed the values given in Table 9.1-2.

Mechanical damage to fuel assemblies could potentially be caused by the EVTM due to dropping of a grappled CCP or new fuel assembly or by tipping over of the EVTM. Dropping of new fuel assemblies or of CCPs containing new or spent fuel assemblies is prevented by the design of the grapple special-locking fingers and by suitable interlocks. The EVTM gantry and trolley are designed with anti-lift-off restraints and rail stops to prevent derailing of the gantry and trolley under combined normal operating and SSE loads. Mechanical collision between the EVTM on its gantry and other equipment, especially the control rod drive lines, IVTM and equipment hatch between the RCB and RSB are prevented by a combination of stops, interlocks, and procedures.

If a seismic event occurs while the EVTM is mated to a floor valve at the reactor, or other location, the design limits the transmitted structural loads, such that the reactor head or other mating facility is not damaged. Motion of the EVTM relative to the mated facility is limited to prevent contact with, or damage to a CCP, if a CCP happens to pass through the floor valve or the mating facility and the EVTM at the time of the seismic event. Cover gas release is prevented by maintaining sealing of the EVTM to the mating facility during a seismic event.

9.1.4.3.2 Design Description

Axial and radial shielding is provided in the EVTM to limit the dose rate to less than the criteria given in Sections 12.1.1 and 12.1.2 at the surface of the cask body. Shielding is provided over the entire length of the EVTM and is graduated in thickness, being thinnest at the upper end, where the radiation source from the spend fuel assembly being handled is least. Approximately 11 in. of lead shielding is provided at the lower end of the EVTM.

The pressure boundary of the EVTM is sealed using metallic and elastomer seals. The metallic seals are single seals which serve as backup to two of the pairs of elastomer seals. There are three types of elastomer seals: static, dynamic, and inflatable. All of these seals are provided in redundant pairs and have essentially zero leakage (i.e., leakage is almost entirely due to permeation through the seal material). The dynamic and inflatable seals have slightly larger leakage than the static seals on a comparable basis. All three types of elastomer seals have a buffer space between seal pairs. The buffer space for static seals is used primarily for periodic leak testing.

Effectiveness of the seals does not depend on the presence of a buffer gas. Dynamic and inflatable seals are provided continuously with a buffer pressure between the double seals. The purpose of this buffer pressure is for leak detection and is not required to prevent seal leakage, although it would mitigate an inner seal leak. The inflatable seals are the only ones which depend on a continuous source of electrical power and inflation gas for operation. In case of loss of off-site power, the seal inflation system valves would fail open, providing the seals with a continuous source of inflation gas from the normal supply system. (The valves are closed during normal operation to provide more sensitive seal leak detection.) The gas supply is from two separate gas bottles and is independent of loss of plant gas supply. The gas supply in the bottles is adequate to maintain seal inflation in excess of two hours. Because the supply valves fail open, loss of off-site power would not affect seal inflation. The piping and valves from the gas bottles to the inflation seals are ANSI B31.1. The seal inflation system and controls have been investigated to ensure that there are no common cause failures which would disable both inner and outer seals.

The EVTM is hermetically sealed to a refueling station by lowering the closure valve which mates with a floor valve. The actual sealing at this interface is accomplished by elastomer double seals, which are periodically leak checked.

The space between the closure valve and the floor valve is purged by the plant argon supply and vent system, through hose connections made after the EVTM has been mated to a refueling station.

When the EVTM is mated to a floor valve at the reactor or other location, the large bending moments and shear loads in the combined vertical structure due to a seismic event are relieved by structurally decoupling the EVTM from the floor valve at the joint interface. The joint between the extender and closure valve is designed with two sliding surfaces. One of these can experience limited horizontal motion if horizontal earthquake loads exceed a predetermined value, while retaining its vertical load carrying capability. Similarly, the second surface can experience limited vertical movement during a seismic event but retains horizontal restraint capability. All sliding joint surfaces are sealed against each other to provide cover gas containment under normal and seismic conditions.

Cooling of a CCP within the EVTM is accomplished by heat transfer to a cold wall system consisting of an about 8 in. ID sealed cold wall having an array of axial fins attached to the outside. The cooling concept is illustrated in Figure 9.1-15. Heat from the 3-ft high fueled region of the spent fuel assembly is distributed over the 15-ft length of the CCP by natural convection of the sodium in the CCP. The heat is transferred from the surface of the CCP to the cold wall primarily by thermal radiation and secondarily by conduction across a stagnant argon-filled gap. The cold wall is cooled by forced convection of ambient air circulated past the axial cold wall fins. Forced air convection is provided by a blower with the capacity to circulate sufficient air to maintain fuel cladding temperature to less than the normal limit (see Table 9.1-2). In case of failure of the blower, or complete loss of all power, natural convection of air is initiated by automatic opening of butterfly valves just upstream of the blower. Natural convection air flow is sufficient to maintain fuel cladding temperature to less than the limits for unlikely and extremely unlikely events in Table 9.1-2. The EVTM cold wall is part of the EVTM containment boundary and is designed and fabricated to quality and inspection standards corresponding to Safety Class 3 (see Section 3.2).

Instrumentation is provided to verify adequate cooling of the EVTM. Thermocouples are located along the length of the EVTM cold wall and at the cooling air inlets and outlets. The temperatures measured by these thermocouples will verify adequacy of cooling.

A cooling system for a CCP during normal transfers to or from the EVTM is not necessary. Although there is a region in the transfer path in which the effective cooling is less than needed to prevent fuel assembly heating, the time spent traversing this region is short enough that heating is insignificant. In the event that a CCP became immobilized in this region, however, there would be significant heating of both the fuel assembly and the material surrounding the CCP in the port. Therefore, the capability is provided for removal of heat from an immobilized CCP at each of the spent fuel transfer ports: reactor fuel transfer port (see Section 9.1.4.7), EVST fuel transfer port, and FHC spent fuel transfer port. At each port a blower is attached to a cooling duct in the port to circulate building air as a cooling medium. The blower is normally off but would be turned on if a CCP were immobilized in the port. The blowers are supplied with normal electrical power.

The EVTM grapple drive system instrumentation includes display of the grapple vertical position. The system includes the capability for manual operation with a hand crank to allow raising or lowering a CCP to a region of passive cooling in the event of a power failure. A braking system automatically stops the grapple at its position at the time of power failure. If a CCP is immobilized in a region of reduced effective cooling during a transfer to or from the EVTM, with the transfer port cooling blower inoperable because of the power failure, the CCP could be manually raised or lowered to provide passive cooling.

Electric power to the EVTM can be manually disconnected at either the power interconnecting box mounted on the EVTM gantry or at the substation supplying EVTM power.

The EVTM CCP grapple has an interlocking finger design such that, with the CCP engaged with and supported by the grapple fingers, the fingers cannot be retracted even if the entire weight is supported by the finger-actuating chain. Redundant support chains are utilized to insure component safety in the event of a single chain failure. The EVTM is transported and positioned by a trolley, traveling on a gantry. The gantry, in turn, travels on rail tracks (see Figure 9.1-13) secured to the RSB/RCB floor. Trolley and gantry wheel truck structures incorporate anti-lift-off and overturning restraints. Trolley and gantry rails are equipped with rail stops plus shock absorbers as positive travel limitations. Collisions between the EVTM and other equipment is prevented by a combination of hard stops (with shock absorbers), soft stops (limit switches which interrupt power to the gantry under all conditions), interlocks (which interrupt power unless certain pre-specified conditions are met), administrative procedures, and an operator riding on the trolley-mounted cab. Permanent hard and soft stops are installed at both ends of the gantry rails. Temporary hard and soft stops are installed on the RSB side of the RCB equipment hatch after it is closed following refueling. Interlocks are provided to assure: (1) that the EVTM is centered on its gantry prior to passage through the equipment hatch, (2) that the IVTM is rotated to the farthest position away from the reactor fuel transfer port, (3) the reactor fuel transfer port is rotated over a fuel transfer and storage position, and (4) power to the reactor rotating plugs is off before the EVTM can approach the reactor. The hard and soft stops and interlocks are all backups for, or are backed up by, the operator and normal administrative procedures.

The EVTM containment (pressure) boundary is constructed in accordance with the rules described in Section III of the ASME B&PV Code for Class 3 vessels. The EVTM pressure boundary support is constructed according to the rules described in ASME III NF3 up to the gantry-trolley interface. The gantry-trolley design is governed by CMAA 70 (Crane Manufacturers Association of America, Specification 70) and by the AISC standards, consistent with LWR refueling machine and gantry-crane industry practice. The AISC design rules are essentially identical to those of the ASME Code, Subsection NA, Appendix XVII for ASME III, Classes 2 and 3 linear-type support structures. Both the EVTM, and its gantry-trolley are Seismic Category I structures.

The EVTM drip pan pots are inside the EVTM containment and are at all times surrounded by an inert atmosphere. Three drip pan pots and a throughport are mounted on the EVTM drip pan turntable. It can be rotated by a motor to position a pot under the EVTM cask barrel after a CCP has been hoisted. The EVTM extender module and closure valve are below the drip pan module. Before the EVTM is decoupled from a floor valve over a fuel transfer port, the closure valve will be closed, and no exchange of RSB/RCB atmosphere with the inert EVTM atmosphere can take place. After coupling the EVTM to a floor valve and opening both closure and floor valves, the EVTM inert gas atmosphere is communicating with the cover gas of the reactor, EVST, or FHC. The valve sequencing, inert gas purging of the valve interfaces, and the coupling and decoupling sequences are interlocked in such a way that no air can enter into the EVTM or into the fuel transfer port.

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Each drip pan pot has three level indicator posts mounted on the bottom which indicate the 20, 40, and 60% full positions. The sodium level relative to the posts is remotely viewed by the EVTM operator using an optical system. This optical system consists of a remotely rotatable TV camera, a heated 45° mirror, a heated viewing window, and a light source with fiber optics light guide.

After one drip pan pot is 60% full, another (empty) drip pan pot is rotated into its place. Each drip pan pot is not filled completely to permit pickup by the EVTM grapple and avoid covering the grapple with an excessive amount of sodium. The purpose of providing three drip pan pots in the EVTM is to minimize the refueling time penalty due to drip pan pot exchange in the FHC. It has been estimated that the EVTM requires at the most four trips to the FHC for drip pan pot emptying during one refueling. (This conservatively assumes failure of the CCP siphon; normally there will be only one emptying at the end of refueling.)

When all three drip pan pots in the EVTM have been filled to 60%, the EVTM travels to the FHC and mates to the FHC maintenance port floor valve. After opening the closure and floor valves, the CCP grapple of the EVTM lowers one drip pan pot into an empty, heated position of a rotary table, directly below the maintenance port in the FHC. The rotary table has 6 positions, three of
59 which are heated and three are not heated. The grapple and drip pan pot dwell for a short time in the heated rotary table position until any frozen sodium on the grapple has melted, and the grapple fingers are able to retract. The grapple is released and raised a short distance. Next, the rotary table rotates and brings the unheated position, containing three empty drip pan pots, under the EVTM grapple. The grapple is lowered, engages an empty drip pan pot, and is hoisted into the EVTM. There the empty pot is deposited in the drip pan assembly. The same operation is repeated twice more until the EVTM contains three empty pots, and the FHC three full pots. The EVTM then uncouples from the FHC and resumes its refueling operation.

The three drip pan pots with molten sodium in the FHC are picked up by the powered manipulator, one at a time, and poured into a waste container. The FHC operators observe that each drip pan pot has only a minimum of residual sodium left before returning the pot to the unheated position in the rotary table. The drip pan pots are not decontaminated after each emptying since they are used on a repetitive basis. The container with frozen, possibly contaminated sodium, is later transferred out of the FHC and turned over to the Radioactive Waste System for further processing and disposal. Procedures for handling and disposing of radioactive metallic sodium are discussed in PSAR Section 11.5.3.

9.1.4.3.3 Safety Evaluation

The dose rate from the highest powered spent fuel assembly is limited to less than the limits given in Sections 12.1.1 and 12.1.2 at the surface of the EVTM cask body. A significant dose rate exists only during the time when a spent fuel assembly is located in the machine. Under normal conditions, this amounts to less than 1 hr. per assembly for a maximum of 162 fuel assemblies. In addition, the closest locations where personnel can be exposed to the radiation source are 10 ft. from the cask body for normal operation and 1.5 ft. for infrequent service operations. These distances result in an attenuation of personnel doses by more than a factor of three, so that the
59 integrated dose to personnel is less than the maximum allowable dose.

The EVTM has adequate seals to prevent excessive radioactive emissions to the operating floor of the RSB or RCB. Radioactivity released from the EVTM will not exceed the limits set forth in Section 12.1 when combined with normal releases from all other sources in the RCB or RSB. The RCB and RSB have radioactivity monitors to detect accidental releases and to sound alarms. Leakage through the seals has been evaluated in Section 15.5.2.3 for the case of 100% release to the interior of the EVTM of all fission gas in a high
59 powered spent fuel assembly and there is no hazard to the public.

Assessment of the physical constraints to both horizontal and vertical motion of the EVTM with relation to the floor valve and closure valve indicates adequate assurance for both an OBE and SSE that: (a) the composite component will reseat from much greater than maximum anticipated vertical motion; (b) the clamps will prevent disengagement of the extender from the closure valve; and (c) the lip on the closure valve will limit horizontal motion to one inch. The latter is more than adequate to prevent contact with or damage to a CCP that might be in transit through the plane of the slip joint. The seals between the extender and the closure valve and between the closure valve and the floor valve ensure cover gas containment under normal and seismic conditions.

The EVTM cooling capacity of 20 kW is adequate to provide a substantial margin above the maximum normal heat load expected, which is 15 kW. The active portion of the cooling system, the blower, is capable of providing the specified cooling without exceeding normal temperature limits. In case of failure of the blower or loss of all AC power, completely passive cooling is automatically provided by natural convection. In this case, cladding temperature is maintained to less than the limit for unlikely events. The peak fuel cladding temperatures, shown in Table 9.1-2A, are within the limits given in Table 9.1-2. Therefore, no damage to the fuel assembly will occur.

The temperature of a fuel assembly in a CCP will increase if the CCP becomes immobilized during a transfer to or from the EVTM. The peak fuel cladding temperatures are listed in Table 9.1-2A for immobilization in the EVTM valve stack assemblies (the assemblies below the cask body, see Figure 9.1-14) or the fuel transfer ports for the reactor vessel, EVST, and FHC. In each case, the peak temperature is less than the limits given in Table 9.1-2 for the frequency class of the event and no radioactivity will be released from the fuel assembly.

The design heat removal capability of the EVTM has been experimentally verified in EVTM heat removal tests. These tests were planned early in the CRBRP program; their purpose and outline are described in Section 1.5.2.7 of the PSAR.

The tests have been successfully performed, and the test data have been analyzed. References 7 and 8 of Section 1.6 document test evaluations, test descriptions, and experimental data. Following the review of test data, the EVTM heat transfer computer model was modified to consolidate the model predictions with the experimental data.

The main conclusion from these tests is that the EVTM has heat transfer capability adequate to meet its design conditions for both forced and natural air convection modes.

A summary of the tests and major findings is provided below.

Full-Scale Heat Transfer Tests

Full-scale tests (Reference 7 of Section 1.6) were performed in a HEDL test facility design to simulate the cooling systems of the CLEM for the FFTF and the EVTM for the CRBRP. The fuel assembly was simulated by a full scale, 217-pin, electrically heated "fuel" bundle in a hexagonal duct. The fuel assembly was contained in a sodium-filled core component pot (CCP), surrounded by an inert gas-filled annulus, and cooled by the concentric cold wall. The test facility and test article design assured that accurate extrapolation could be applied to test data for either refueling machine. Major test results showed the following:

1) Maximum heat dissipation rates and corresponding peak cladding
temperatures were measured as presented in columns 2 and 3 of Table
44 9.1-3A. The test data were extrapolated to the actual EVTM
configuration using the Thermal Analyzer Program (TAP), modified to
provide accurate matches with the experimental data. The extrapolated
44 data are given in columns 4 and 5 of Table 9.1-3A.

2) The axial temperature distributions above and below the fueled region gave evidence of additional natural convection sodium loops in the upper part of the fuel assembly, and in the bottom part of the CCP annulus, respectively.

3) A fuel assembly with radially skewed power distribution (from 1.3 times average power on one side to 0.7 times average power on the other side) produces peak pin cladding temperatures insignificantly different from those of a radially uniform power distribution.

<u>Sub-Scale Heat Transfer Tests</u>

Sub-scale tests (Reference 8 of Section 1.6) were performed at AI to
investigate the effects of sodium and sodium frost deposition on the
emissivity of the EVTM heat transfer surfaces. The tests were carried out
using an electrically heated, instrumented CCP surrounded by a cold wall,
filled with inerted gas. The CCP was scaled down in axial length by a factor
44 of 1/3, with a full scale gap between CCP and cold wall. Five yearly EVTM
refueling cycles were performed under simulated EVTM operating conditions with
128 CCP transfers and four different CCP power levels per cycle. Major test
results were as follows:

1) Cold wall and CCP emissivities can be attained to adequately dissipate
44 the heat of a 20 kw fuel assembly in the expected environment of the
EVTM.

2) The emissivities of the CCP and the cold wall did not change essentially during one refueling cycle.

3) No impairment of the mechanical equipment functions occurred even at considerable levels of oxygen and water contamination. CCP emissivity increased somewhat with increased contamination.

4) The emissivity control coating of the cold wall exhibited good durability against exposure to sodium vapor and condensation.

5) Cold wall heaters proved to be an effective means for periodically removing sodium deposits from the cold wall in order to maintain high emissivity.

Prior to these tests, emissivity tests on cold wall specimens, and parametric heat transfer tests on a reduced scale EVTM were carried out to confirm some of the basic assumptions used in the EVTM heat transfer model. Major findings of these tests were:

1) No significant change of cold wall emissivity occurred after continuous exposure times of up to 100 days in 700°F sodium.

2) Coated cold wall specimens could be cleaned by evaporation of sodium deposits at temperatures as low as 400°F.

3) No loss of coating occurred as a result of soaking coated cold wall specimens for 100 days in 700°F sodium.

4) Tests with various annular gaps between a simulated fuel assembly and CCP established the definitive existence and importance of localized annular sodium convection loops in the upper part of the CCP, in addition to the overall sodium convection loop up through a spent fuel assembly and down the annulus between the fuel assembly in the CCP.

5) Tests with varying lengths of unheated fuel assemblies above and below the heated region, varying relative hydraulic resistances, radial thermal resistance, and power level permitted refinement of the heat transfer model to permit prediction of the effect of these variables on EVTM heat transfer.

In addition to the grapple design features intended to prevent dropping of CCP's and fuel assemblies, interlocks in combination with load cells further limit the potential for dropping grapple loads. The interlocks
44 are discussed in Section 7.7.1.9. In the unlikely event that a grapple drops a CCP or new fuel assembly, no serious consequences can occur outside the EVTM, since the EVTM can contain radioactivity releases. The consequences of a limiting accident with 100% fission gas release of a high-powered spent fuel assembly into the EVTM is discussed in Section 15.5.2.3. The anti-lift-off restraints of the gantry and trolley, and the rail stops prevent any undue motions of the EVTM during an SSE. A combination of hard and soft stops, interlocks, normal procedures, and operator presence are used to prevent collision of the EVTM and its gantry with critical equipment whose normal operation or location could permit collision. Administrative procedures and operators are used to prevent collision with equipment which is not normally located between the rails.

If a CCP should drop from the EVTM grapple during transport between refueling stations it would fall about 53 in. to the bottom of the drip pan pot. The drip pan module is designed to absorb this impact load and would support the CCP. Due to the small annular clearance between cold wall inner diameter and CCP outer diameter (nominal diameter difference at the upper end of the CCP is 0.325 in.) the CCP would stand in an almost perfectly vertical 25 position. The EVTM grapple has a conical-shape to provide lead-in for the CCP handling socket and could pick up the dropped CCP.

Even in the dropped position, the entire heat generating region of the spent fuel assembly in the CCP would be <u>within</u> the EVTM cold wall. Conservative heat transfer analysis results have shown that steady state temperatures of the seals near the bottom of the cold wall would be below 320°F. The steady state peak temperatures of the fuel assembly cladding and of the sodium in the CCP would be below 1300°F. The analysis assumed a CCP containing a 20-kw spent fuel assembly with part of the heat generating region of the fuel assembly positioned at a level <u>below</u> the cold wall.

The consequences of an accident involving a CCP dropped within the EVTM can be summarized as follows:

1) The dropped CCP can be picked up again by the EVTM grapple.

2) If pickup were delayed, the cladding of the highest powered spent fuel assembly would reach a steady state peak temperature below 1300°F, which is well below the boiling point of sodium.

3) EVTM seal temperatures at the worst location would be low enough not to affect the integrity of elastomeric seals.

Therefore, the event will not result in any impact on the health 25
and safety of the public.

Dropping of a spent fuel assembly onto the operating floor is precluded by the following design features and interlocks:

44 1) Interlocks prevent release of grapple fingers if load cells indicate that the CCP is supported by the grapple.

2) Grapple design makes release of CCP under load impossible.

44 3) Drip pan prevents a CCP from dropping more than 4.4 feet within the EVTM.

4) Interlocks prevent opening of EVTM closure valve unless EVTM is mated to a floor valve at a fuel transfer port.

Concurrent or sequential failure of all these design features and interlocks is considered to have a hypothetically remote probability, but would be necessary in order to drop a CCP onto the operating floor. However, even in this case the CCP would remain vertical within the EVTM, with only
20 the lower 2 feet exposed, and it would be adequately cooled.

9.1.4.4 Safety Aspects of the In-Vessel Transfer Machine (IVTM)

The IVTM is an under-the-shield refueling machine that is mounted on the Small Rotating Plug (SRP) of the reactor head only when the reactor is shut down for refueling. The primary function of the IVTM is to move core assemblies between their normal positions in the reactor core and the transfer positions outside the core, but within the reactor vessel. In the transfer positions, the core assemblies are accessible for transfer in or out of the reactor vessel by the EVTM. The IVTM executes only vertical movements, all horizontal movements are performed by rotation of the triple rotating plugs in the reactor head. The machine is depicted in Figure 9.1-16. The IVTM is based on a concept which has been under development in the LMFBR Base Program for several years. Major subassemblies (e.g., grapple, core assembly identification system) are very similar to comparable subassemblies of the
FFTF in-vessel handling machine (IVHM). 15

The IVTM implements its function by the following operational capabilities:

1) Grapple and release core assemblies

2) Raise and lower core assemblies

3) Provide holddown of adjacent core assemblies

4) Uniquely identify core assemblies

44 5) Orient and center core assemblies for insertion in the core.

The IVTM is a rising stem type design and consists of two principal
44 subassemblies, the in-vessel section, and the ex-vessel or drive section.
The in-vessel section contains the grapple, centering device, identification
mechanism, the holddown mechanism, and the seals which contain the reactor
20 cover gas. It is installed into the SRP of the reactor vessel head with the
44 auxiliary handling machine. The drive section contains the drive equipment
which powers the in-vessel section and is installed with the polar crane.
The in-vessel section is designed to operate within the sodium filled envi-
44 ronment of the reactor during shutdown, and the drive section transitions the
machine motions to the air filled environment where normal power equipment
such as electric motors and pneumatic cylinders can be located.

44 The grapple and drive mechanism raises or lowers core assemblies
inside the reactor vessel under sodium. By moving the reactor plugs, core
44 assemblies grappled by the IVTM can be translated to and from a core position
and an in-vessel fuel transfer position. Figure 9.1-16A shows the IVTM grap-
ple in a core assembly socket. 14

Safety aspects of the IVTM that will be covered in this section are radiation shielding, prevention of radioactive releases, prevention of
44 mechanical damage, prevention of improper core assembly insertion or removal,
and removal of jammed fuel assemblies.

9.1.4.4.1 Design Basis

Adequate shielding for radiation protection is provided in the design of the IVTM. The shielding meets the radiation protection requirements specified in 10 CFR 20 and ensures an integrated dose below 125 mrem/quarter.

Release of radioactive cover gas through the IVTM during refueling
59 is prevented by buffered seals or welding of penetrations. Radioactive leakage
and diffusion through seals is less than the limits specified in Sections
49 12.1.1 and 12.1.2.

44 Mechanical damage to core assemblies could potentially be caused by
the IVTM due to the following events:

44 1) Exertion of excessive push or pull forces to core assemblies

2) Unscheduled movement of the IVTM grapple or the reactor plugs

44 3) Dropping of a grappled core assembly.

Upward or downward vertical forces on core assemblies during inser-
tion or removal from the core by the IVTM grapple drive mechanism are limited
by the load control system to 1000 LB push or pull loops for normal operations,
44 and up to the maximum allowable push and pull loads on a core assembly of 3000
59 lb and 4300 lb, respectively, for moving a stuck core assembly.

Vertical movement of the grapple is prevented by interlocks until
positive grapple engagement or disengagement is indicated. These interlocks
44 are discussed in Section 7.7.1.9.

Horizontal translation of the grapple by reactor plug rotation is
prevented by interlocks until the grapple, the grappled core assembly and the
44 holddown sleeve are fully raised to clear the upper core internals.

The design of the IVTM grapple, in conjunction with the handling
lip on core assemblies and suitable interlocks, prevents dropping of core
44 assemblies.

Improper core assembly insertion in the core is prevented by limit
switch interlocks and the configuration of the core assembly discrimination
post and its receptacle. Verification of proper seating is accomplished by
satisfaction of all interlocks, and by the IVTM position indication and load
44 control systems.

Improper core assembly insertion is prevented by IVTM design fea-
tures permitting unique identification of each core assembly inside the reac-
tor vessel. Core assemblies are also identified immediately after removal
44 from a core position to ensure that the proper core assembly has been removed.

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9.1.4.4.2 Design Description

The steel shield column of the IVTM has a variable thickness radiation shield (from 2.75 to 1.5 in. above the small rotating plug nozzle) to attenuate gamma radiation from radioactive sodium. The amount of radioactive sodium that will be in the upper part of the IVTM's housing is limited to the surface film remaining on the grapple stem after it is raised from the sodium pool during the transfer operations. The active fuel zone of the core assembly which is under approximately 9 ft of sodium during transfer operations, will also contribute to the external dose rate, but to a lesser degree. There will also be a slight contribution to the dose rate from the radioactive cover gas inside the annulus between the grapple stem and the shield column.

The IVTM is sealed to the SRP by 3 elastomer O-ring seals with buffer region between them. Between each set of seals, a leak detector port is provided to enable connection of a sensor for monitoring the seals integrity before refueling. This arrangement is typical for other seal sets that involve dynamic motion such as the grapple stem and holddown drive shafts. Static seals or dynamic seals with very small displacements, such as those for the identifier pawl shaft are double, with capability for convenient periodic leak testing by pressurizing the buffer region between the seals. The effectiveness of the seals does not depend upon presence of a buffer gas.

The IVTM drive mechanism has been designed to exert an upward or downward load of 5000 lb maximum. Incorporated into the design is a pneumatic load control system and a load cell system that limits the load exerted on a core assembly to 4300 lb pull and 3000 lb push. The load and the grapple vertical position are displayed on the IVTM control console. In the event of a loss of electrical power, a braking system automatically stops the grapple at its position at the time of the power failure. The grapple drive system includes the capability for manual operation to allow raising or lowering a grappled core assembly. Repositioning would not be needed for cooling the assembly. It would be immersed in sodium and thus passively cooled, whenever handled by the IVTM. Electric power to the IVTM can be manually disconnected at either the IVTM control console or the substation supplying IVTM power.

The IVTM is positioned above core assembly locations by rotation of the reactor rotating plugs. The positions of the plugs are displayed on the IVTM control console to define the IVTM horizontal position.

The load in the pneumatic load control system will be set to provide a normal push or pull force on a core assembly of 1000 lb. The pressure of the load control system may be adjusted to provide higher load capability up to the push or pull load limits.

The design provides for the load to be limited in two ways.

1. The primary limitation is provided by the pneumatic load control system. The pressure in the pneumatic load control cylinders limits the load applied by the electromechanical actuator to the driven core assembly. When the core assembly insertion resistance load exceeds the pressure setting in the system, the core assembly stops moving, but the electromechanical actuator continues driving the pistons for about 0.25 inches. This differential travel trips a set of limit

switches which automatically stop the electromechanical drive. Self-contained hydraulic dashpots prevent sudden actuator movements due to sudden changes of the frictional resistance at the load.

2. Load cells are used as backup to the pneumatic system to shut off the electromechanical drive when the preset load limits are exceeded.

Actuation of the grapple fingers for pickup or release of a core assembly is possible only when all the grapple finger actuation interlocks have been satisfied, the grapple is pushing on the core assembly (i.e., core assembly is in full down position), and the load control limit switches that shut off the electromechanical drive are tripped.

59 Grapple vertical motion is possible only when interlock switches indicate either grapple fingers are retracted, or grapple fingers are extended. Schematics of the IVTM interlocks are shown in Figure 9.1-16B and 9.1-16C. 16

The grapple and grappled core assembly must be raised a minimum of 209.8 in., and the holddown sleeve a minimum of 40 in. above the core plane to clear all mechanical interferences before the interlocks permit horizontal translation.

There are three cam operated fingers for picking up a core assembly. The cams are mounted on a rod that is operated by an air cylinder mounted at the top of the grapple stem in the drive section of the IVTM. When the rod is down, the fingers are extended in the pickup mode; when the rod is raised (2.75 in.) the fingers are retracted in the release mode. The design and spacing of the cams on the rod provide for positive engagement or disengagement
59 such that a core assembly cannot be partially grappled.

A mechanical interlock and redundant interlock switches prevent actuation of the IVTM grapple fingers during insertion of a core assembly into the core until the core assembly is 1.32 inches from being fully seated in the core. Within the 1.32-inch distance, proper engagement of core assembly discriminator post to its mating receptable is ensured, and the core assembly can be released after the switches of the load control system are tripped, thereby shutting off the electromechanical drive.

Since the core assembly release can be initiated anywhere within the 1.32-inch distance (core assembly insertion may be terminated in this region as a result of high core friction) assurance that the core assembly is resting at the bottom of its receptable can be verified by the IVTM position indication system within the uncertainty due to the core-assembly-to-core-assembly tolerance, axial core assembly dilation and position indication system inaccuracy. However, the 1000 lb normal push force applied by the grapple is considered to be sufficient to ensure that the core assembly is fully seated in its mating receptacle.

When removing an irradiated core assembly from its reactor core location, the assembly is raised into the IVTM in-vessel section by the IVTM
44 grapple. In this position, the core assembly is rotated by rotating the grapple stem. The rotational position of the grapple stem is checked by an encoder, driven by the rotating stem. Each core assembly has an orientation notch and uniquely defined identification notches on the outside diameter of
44 its handling socket. During rotation of the core assembly these are felt by an identification and orientation pawl, and transmitted by a shaft to an external encoder that supplies electrical signals to the IVTM instrumentation and control system.

44 Each irradiated core assembly is thus uniquely identified immediately following removal from its core location. Prior to any further changes

in core geometry, the identification number of the irradiated core assembly has to match that specified in the refueling computer program (when operating 44 in the automatic mode) or the operator's instruction (when operating in the manual mode). Failure to match will automatically halt refueling when in the automatic mode . The interruption in the manual control mode is accomplished 59 up by the operator.

44 Each new core assembly is likewise identified by the same method, after the IVTM grapple has removed it from one of the in-vessel transfer 44 positions and before it is oriented and inserted into an empty core location.

All transfer operations of the IVTM involving core components are performed entirely under sodium. More than 20 ft of sodium is above the top of the reactor core and the fuel transfer positions, as shown schematically 59 in Figures 9.1-16B and 9.1-16C. The fueled region of the fuel assemblies is covered by approximately 9 ft. of sodium during all transfer operations involving the IVTM, as stated above.

When a spent fuel assembly in a CCP is hoisted from the reactor fuel transfer position into the EVTM cask, it is cooled along its path of vertical travel first by the reactor sodium, then by heat transfer to an air-cooled insert in the reactor fuel transfer port (see Section 9.1.4.7), and finally by heat transfer to the air cooled EVTM cold wall.

There is about 7 ft of travel between the sodium surface in the reactor vessel and the bottom of the cooling insert in the reactor fuel transfer port, and about 11 ft between the top of the cooling insert and the bottom of the EVTM cold wall. A fuel assembly in a CCP traveling through these two sections is only cooled by natural sodium and argon convection, conduction, and radiation. Normally that part of the CCP containing the heat generating part of the spent fuel assembly and the hot CCP sodium above it passes through these two zones of reduced effective cooling in 1 min. and 1.7 min., respectively. In case of a malfunction of the EVTM, the CCP can be either manually lowered or raised at a reduced speed, bringing it into a position where it can be adequately cooled. In such an event the CCP would travel through the two zones in a time conservatively estimated to be 16 min. and 20 min., respectively. Transient heat transfer analyses have shown that the peak cladding temperature of a 20 kw spent fuel assembly in a CCP reaches about 820°F after 20 min. and 940°F after 30 min. when the CCP is immobilized in a location where no forced air cooling or sodium pool cooling is available. The peak metal temperatures in the vicinity of the nearest containment seals were calculated to about 160°F. The initial temperatures of the fuel assembly and the CCP sodium are those of the reactor sodium during reactor shut-44 down for refueling, i.e., about 400°F.

It is concluded that this event will not produce radioactivity release into the EVTM and reactor vessel and/or result in a breach of EVTM/ reactor vessel containment. 25

44 If a core assembly is found to be jammed during an attempt to withdraw it from the core, i.e., the normal IVTM pull load of 1000 lbs is not 44 sufficient to remove it, additional pull load may be exerted with the

approval and cognizance of the refueling supervisor. The pull load can be increased as required by increasing the pressure setting to the load control
44 system up to a limit of 4300 lb. Maximum worst friction loads are currently calculated to be less than 3000 lbs. If the maximum allowable pull load of 4300 lbs is reached and the assembly cannot be removed, the surrounding core assemblies (which may be bowed) will be removed and replaced one at a time,
44 as necessary, with either new core assemblies or core special assemblies (which are straight and have greater clearance than fuel assemblies). This procedure reduces interaction forces from core distortions until withdrawal forces are lowered sufficiently to remove the jammed assembly.

Fuel assembly bowing increases the contact force between assemblies. Because of the resulting inter-assembly frictional force increment, the force required to remove spent fuel assemblies is also increased due to fuel assembly bowing. In general, assembly bowing, and thus assembly withdrawal loads, increase with increasing burnup. Also, withdrawal loads are higher for assemblies on the core perimeter than at the center because assembly bowing is typically larger at the core perimeter.

The spent fuel assemblies can withstand the maximum applied withdrawal pull of the IVTM (4300 lbf) with a margin of safety greater than 4, based upon prevention of yielding in the critical duct welds. The fuel rod bundle is not mechanically attached to the duct. Thus, the force transferred to the fuel rods due to relative rod bundle-duct motion would be due only to frictional forces between the rod bundle and the duct wall. These forces are not applied directly to the fuel rod cladding, but rather to the wire wrap spacer. The magnitude of the rod bundle-duct frictional force depends upon the fuel rod bundle porosity, burnup, and compressibility at end of life. The component of these frictional forces which are applied to the fuel rod cladding, depend on the tension in the wire wrap at end of life. These parameters are currently being analyzed and correlated with experimental results. The actual maximum withdrawal force the fuel rod bundle can withstand at end of life is also currently being analyzed; it is a function of end-of-life cladding materials properties, rod configuration, and rod bundle-duct frictional forces. These parameters will be determined as part of the test program described in the response to NRC Question 001.283. The consequences of extensive fuel rod failure in a spent fuel assembly during refueling were examined in Section 15.5.2.1 in connection with the analysis of a dropped fuel assembly. This analysis concluded that fuel rod cladding rupture within the reactor vessel during refueling can be accommodated from the standpoint of both fission gas release and fuel particle dispersal. 19

9.1.4.4.3 <u>Safety Evaluation</u>

Adequate shielding is provided by the IVTM's shield column to limit the integrated radiation dose to personnel from the IVTM to less than the
59 maximum allowable dose rate.
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The IVTM, when mated to the SRP, has adequate seals to prevent excessive radioactive release due to cover gas leakage into the RCB. Radioactivity released from the IVTM does not exceed the limits set forth in Sections 12.1.1 and 12.1.2. Leak detectors will be used to continuously monitor the effectiveness of the dynamic seals. Periodic leak checks will be carried out for the static seals. In addition, the RCB will have radioactivity monitors to detect accidental releases and sound alarms.

The drive mechanism and grapple design, together with load controls, limit switches, and interlocks will prevent exertion of excessive forces on core assemblies, unscheduled vertical movement of the grapple or disengagement of the grapple fingers except in the full-down position of the grapple. Limit switches and interlocks (see Section 7.6.2) will also prevent inadvertent rotation of the reactor plugs. The extremely unlikely accident of a fuel assembly dropped from the IVTM is discussed in Section 15.5.2.1.

The combination of position interlock switches and the switches of the load control system in conjunction with the design of the discriminator post and matching receptacle ensures proper seating of core assemblies. The fuel and inner blanket assemblies are divided into 8 groups, each having a different configuration of discrimination post inner and outer diameters fitting into corresponding receptacles. This ensures that a fuel assembly of one group can only fit into its corresponding receptacle, see also Section 4.2.1.2.3.

The IVTM grapple and holddown sleeve, when positioned for pickup of a core assembly, partically cover core assembly outlets, reducing coolant flow through affected assemblies. The calculated minimum flow area to ensure a negligible increase in fuel assembly outlet temperature during refueling conditions is 1.2 in.^{2}. The 1.6-in.^{2} grappled assembly flow area provided by the actual grapple provides for adequate coolant flow through the assembly. A substantially greater flow area is maintained for assemblies affected by the simple cylinder of the holddown sleeve.

Core assembly identification by the IVTM identification and orientation system prior to core assembly insertion into the core, and the position indicator system of the reactor rotating plug, will ensure core assembly insertion into a correct core location.

If, however, a core assembly is erroneously inserted into a core location belonging to another core assembly group, the core assembly discriminator post will bottom against the top of the receptable, thus resulting in an improper seating. The length of the core assembly discrimination post is sufficient to preclude tripping of the grapple finger actuation interlock switches, thereby preventing core assembly release even though the preset force of the load control system has been exceeded. This condition and the IVTM grapple position indication system signal will warn the operator to initiate a corrective action.

Immediately after core assembly removal from the core, the IVTM identification system identifies the core assembly serial number. In the automatic control mode, this serial number is compared by the control system with the serial number designated by the refueling program that must be in that particular core location, thus ensuring that the correct core assembly is removed from the core. In the manual control mode, the operator compares the calculated serial number to the serial number designated in the refueling plan.

If, however, a core assembly is erroneously removed from the core, the serial
44 number discrepancy will warn the operator to return the core assembly into the core position last serviced and to initiate a corrective action.

Measures to ensure proper indexing of the IVTM relative to the core are taken prior to refueling start and are as follows:

1) The refueling program is prepared and verified in advance.

2) The refueling program is checked out on a simulator.

An additional safeguard against gross misloading errors is provided by the low
44 neutron monitoring system which will detect any significant deviation from planned changes in reactivity.

The IVTM, in conjunction with the reactor head rotating plugs and new core
44 assemblies or core special assemblies, is capable of safely performing the necessary operations to remove a jammed core assembly without special equipment.

9.1.4.5 <u>Safety Aspects of the Auxiliary Handling Machine (AHM)</u>

The primary function of the auxiliary handling machine (AHM) is to install the in-vessel section of the IVTM in the reactor head prior to refueling, and to remove it afterward. The AHM also serves a plug handling function, in
44 removing and reinstalling the IVTM port plug from the reactor head. Additionally, the AHM supports maintenance functions of other systems, by removing and installing control rod drive lines and core inlet modules using special tools, grapples, adapters, etc., provided by the Reactor Enclosure System and the Reactor System.

The AHM has the following capabilities:

1) Grapple and release components

2) Raise and lower components

3) Transportable by the RCB polar crane (see Section 3.8)

4) Maintain argon environment

5) Provide sealing and containment of the reactor cover gas when mated to the reactor

6) Provide radiation shielding

7) Collect sodium drippage from components handled with surfaces wetted by sodium.

The AHM is a shielded, single-barrel handling machine which is transported by the RCB polar crane. The general arrangement of the machine is depicted in
44 Figure 9.1-17. The machine is about 60 ft high and weighs about 100 tons. The AHM is similar in concept to the EVTM, but it has been simplified, due to

the less demanding requirements of the components it handles (as compared to the irradiated core assemblies in sodium-filled core component pots handled by the EVTM). The major differences are: (1) no radioactive decay heat, (2) lower radiation source strength, (3) less sodium drippage and (4) lower use factor.

The major subassemblies of the AHM consist of the following: the grapple and hoisting system; the cask body assembly, which provides structural support and shielding; the service platforms; a closure valve and drip pan assembly; an extender assembly; a cask containment barrel; a gas service system, which provides gas for process systems and also monitors seal leakage; and lastly, electrical instrumentation and control equipment.

Operations with the AHM begin by connecting the crane hook to the lift structure, then disconnecting the AHM from the AHM storage facility support structure. The polar crane moves the AHM to the proper location, and raises the AHM to place the retracted extender at 9 inches above an AHM floor valve (see 9.1.4.6) installed at the location. The extender lowers a mating flange
59 onto the floor valve. After sealing and purging (see 9.1.4.3), the closure and floor valves are opened. The grapple is driven down, the fingers are actuated, and the component is lifted into the AHM cask. The combination drip pan - closure valve is closed first, then the floor valve is closed. After purging the space between the valves, the extender is raised, and the AHM is moved away by the crane.

The design bases, description, and safety evaluation of the AHM discussed below cover radiation shielding, prevention of radioactivity release, and prevention of mechanical damage to the reactor.

9.1.4.5.1 Design Basis

Shielding for radiation protection is provided in the design of the AHM. The shielding meets the radiation protection requirements specified in Sections 12.1.1 and 12.1.2 and ensures that the integrated dose outside the AHM is less than 125 mrem/quarter.

Radioactive releases from the AHM during installation or removal of the components handled by the AHM are prevented by proper sealing or welding of penetrations. Radioactive leakage and diffusion through seals are limited to meet the radiation requirements of Sections 12.1.1 and 12.1.2.

Mechanical damage to the reactor could potentially be caused by the AHM due to dropping of a grappled component or toppling over or dropping of the AHM. Dropping of components handled by the AHM is prevented by the design of the grapple fingers, the grappling lip of the component grappling sockets, and by suitable interlocks. The AHM handling lugs and lifting eyes are designed to assure positive engagement and accommodate all normal and SSE loads. A single failure of the lifting structure does not lead to a drop of the AHM. The potential drop height of the AHM is limited, and the RCB polar crane lowering speed is limited. The AHM is supported from the polar crane during the entire time when it is in the vicinity of the reactor head and is securely attached to its parking structure when not in use.

If a seismic event occurs while the AHM is mated to the floor valve at the reactor, the design limits the transmitted structural loads, such that the reactor head is not damaged. The ability to operate the AHM floor valve after the seismic event is maintained, and no damage of major equipment located in the head access area is caused by the AHM during or after an OBE or SSE. The AHM is installed on the reactor only when the radioactivity concentration of the reactor cover gas is below the limit given in the Technical Specification, Section 16.3.10.

9.1.4.5.2 <u>Design Description</u>

Figure 9.1-17 shows the AHM shielding arrangement. The cask body is of steel construction with different thicknesses (largest at the bottom) of steel to provide the required shielding. The extender section that mates to the AHM floor valve includes lead shielding. Dose rates at the cask body surface are less than the limits specified in Sections 12.1.1 and 12.1.2. The maximum radiation source is the IVTM port plug.

The AHM is coupled to the floor valve with a mating flange located at the bottom of the AHM extender prior to handling components. The mating surfaces have double seals with leak check capability to prevent radioactive releases. The only source of gaseous activity in the AHM is reactor cover gas containing fission products.

The AHM, like the EVTM, has an interlocking finger design such that, when the handled component is engaged with and supported by the grapple fingers, the fingers cannot be retracted even if the entire weight is supported by the finger-actuating chain. Redundant support chains are used to prevent component dropping in the event of a single chain failure.

The AHM is moved between its storage facility, the reactor head and the RCB plug storage facility by the RCB polar crane at a maximum height of 2 ft. above the operating floor. When the AHM is positioned over a reactor port or plug storage facility port, it is entirely supported by the single failure proof RCB polar crane. The crane raises the AHM to a height of 9 inches above the floor valve and the AHM extender is lowered to mate with the floor valve. Administrative control of this operation prevents accidental lowering of the
59 AHM onto the reactor head. The crane speed is limited to less than 5 fpm. A complete discussion of a lowering event is presented in Section 15.5.2.5.

Handling lugs, crane rigging, and lifting eyes are designed to accommodate all normal dynamic and static loads, in addition to loads caused by SSE, to prevent dropping of the AHM. The handling ball is attached to the polar crane hook by a large diameter clevis pin. Wire ropes and spreader assemblies provide a second, redundant load path from the handling ball to the polar crane hook. When not in use, the AHM is stored at a parking station located in the northeast quadrant of the building.

The parking station is designed for the SSE seismic loads which are carried into the RCB structure.

In the event of a loss of electric power, a braking system automatically stops the grapple at its position at the time of power failure.

Electric power to the AHM can be manually disconnected at either the AHM console or the substation supplying the floor service station from which the AHM is being supplied.

The vertical position of the AHM grapple is displayed on the AHM control console.

When the AHM is in position at the reactor, only the extender mating flange is resting on the floor valve, which in turn is supported from the small rotating plug (SRP) by an adaptor. If the two components were firmly attached to each other, the resulting combined structure, in effect, would represent a tall, vertical cantilever rising from the SRP, attached at its upper end to the polar crane. The large bending moments and shear loads in this combined structure, resulting from horizontal excitation due to an OBE or SSE, are relieved by structurally decoupling the AHM from the floor valve at the extender/floor valve joint interface. At a predetermined horizontal ground acceleration, complete severance of the AHM from the floor valve ("breakaway" concept) eliminates the cantilever beam effect and significantly reduces all seismic loads.

The joint between lower extender flange and floor valve is designed with shear pins which fail upon reaching a predetermined horizontal load. This enables the AHM to separate from the floor valve during a seismic event. The design incorporates a pneumatic reservoir which initiate raising of the AHM extender following the shearing-off of the shear pins. The actuators can raise the extender by about 3 inches in less time than it takes for the extender to clear the floor valve during the horizontal movement due to an OBE or SSE.

9.1.4.5.3 Safety Evaluation

The radial and axial shielding provided by the AHM limits the integrated dose to personnel to less than the maximum allowable dose rate during the installation or removal of the components handled by the AHM. As with the EVTM (see 9.1.4.3) the radiation source in the machine is intermittent and short term.

The AHM has adequate seals to prevent radioactive emissions to the RCB operating floor. Radioactivity released does not exceed the limits as set forth in Sections 12.1.1 and 12.1.2.

The design of the grapple release mechanism, interlocks, and the redundancy of chains will limit the potential for dropping grapple loads. The interlocks are similar to those of the EVTM, and similar to those discussed in Section 7.6.2. The structural support and restraint of the AHM storage facility, and the rigging of the AHM when attached to the polar crane prevent toppling or dropping of the AHM due to an SSE or other loads.

If a seismic event were to occur when the AHM is mated at the reactor SRP, shear pin severance between the AHM and the floor valve would leave the AHM free to swing as a pendulum on the RCB polar crane. The pendulum action of the AHM after decoupling would not induce damage to other equipment, since there is at least 7 inches vertical clearance from any adjacent hardware. The closest item is the space allocation for plumbing and wireways to the control rod drive lines. The AHM is only installed on the reactor when the reactor is shut down, and the control rods are fully inserted and disconnected from their drive lines. The automatic rise of the extender after shear pin severance ensures that the extender cannot drop to its maximum extended position, i.e., to a level of approximately 5 inches below the upper floor valve flange.

If the AHM were to break away during a seismic event coinciding
with an open floor valve, reactor cover gas could be released into the RCB.
The probability for the combination of these events to occur simultaneously
is small, (unlikely incident), since the AHM does not spend more than about
13 hours per year at the reactor and the floor valve is only in the open
position about 9 hours per year. The normal time from reactor shutdown to
reach a configuration with the AHM installed on the floor valve on top of
the reactor SRP, and the floor valve gate being open, is more than 3 days.
An unlikely event discussing accidental cover gas release from the reactor
58 hours after shutdown is discussed in 15.5.2.4. The accident was found to
59 result in a site boundary dose well below the values given in 10 CFR 20 for
unrestricted areas. This dose is considered very low, since the applicable
guide line values for this event are those of 10 CFR 100. The cover gas
radioactivity concentration will be continuously monitored before and after
the AHM is installed. AHM installation on the reactor, and subsequent opening of the floor valve gate, will only be carried out if the cover gas activity is below the limit set by a Technical Specification, see Section
16.3.10. In the unlikely event that all cover gas is released into the RCB,
operators could work in the RCB for a sufficiently long time to manually
close the floor valve, including preparation time, without exceeding the
allowable quarterly dose for restricted areas of 1.25 rem given in 10 CFR 20.

In the extremely unlikely event such that the IVTM port plug, the
IVTM in-vessel section, or a control rod drive line (CRDL) were to be in
transit past the AHM extender/floor valve interface at the moment of an
earthquake, and the AHM broke away, sufficient relative motion could occur
to damage these components. Due to the relatively small diameter of the CRDLs,
damage would not prevent subsequent removal of the grappled components and
closure of the floor valve. With the larger diameter IVTM port plug and the
59 IVTM in-vessel section, the effect could be to key the AHM to the floor valve,
negating the break-away feature. Overstressing of the AHM, adaptor and SRP
could then result, and damage sufficient to jam the IVTM port plug or the IVTM
in-vessel section could be possible. In this event, the floor valve could
not be closed. Cover gas leakage would be much slower than the immediate
release described above. This would permit time to purge the reactor cover
gas, suit-up, and take emergency corrective action with no public safety
44 consequences.

59 The total time interval for all grappled components passing through the AHM extender/floor valve interface is less than one hour per year, the above described events are considered to be extremely unlikely.

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9.1.4.6 Safety Aspects of Floor Valves

The functions of a floor valve are to provide shielding and sealing against radioactivity when the plug is removed from a port in the reactor, EVST, or FHC, thus isolating the contaminated gas and radiation environment of these facilities from the ambient air of the RCB and RSB.

The floor valves are portable rotatable disk-type valves used to seal the reactor, EVST, and FHC ports during various phases of the refueling operations when the port plugs are removed. Two basic sizes of floor valves
44 are used: one for the EVTM for the transfer of core assemblies, and another with a larger opening, used with the AHM for the transfer of the in-vessel section of the IVTM and other components handled by the AHM. Both types have the same basic design as depicted in Figure 9.1-18, except for interior opening size and differences in thickness of the radial shielding surrounding the port.

44 The floor valve design concept is similar to the AHM and EVTM closure valve design. It is also similar to the floor valve developed for the FFTF program. Floor valves are installed by the building crane and bolted down. The AHM or EVTM is moved into position over the floor valve, and the closure valve extender is lowered onto the floor valve at the reactor, the EVST or FHC. After sealing the closure and floor valves are opened. The port plug is removed and the valves are closed. The extender is raised, and
20 the AHM or EVTM is moved away by the crane or the gantry, respectively.

This section covers the safety aspects of the floor valves, i.e., shielding and prevention of radioactive releases.

9.1.4.6.1 Design Basis

The design bases for shielding and radioactive release are the same as for the AHM (see 9.1.4.5.1). In addition the appropriate zone shielding
44 criteria of Section 12.1 apply.

9.1.4.6.2 Design Description

The shielding design of the floor valve which is used with the AHM (23-inch diameter port opening) is an asymmetric cylindrical annulus of steel with a minimum radial shielding thickness of 10 in. The vertical height of the floor valve provides a lead shielding thickness of approximately 9-1/2 in. The floor valve used in conjunction with the EVTM is larger in overall diameter (78 in. as compared to 68 in. of the AHM floor valve), and in height (17 in.
44 20 versus 15 in. for the AHM floor valve). The EVTM floor valve port has an

~11 in. diameter opening. The increased thickness of the EVTM floor valve in radial and axial direction provide the additional shielding required for the much higher radiation source which passes through an EVTM floor valve (spent fuel assembly) compared to an AHM floor valve (IVTM port plug).

The stepped upper and lower steel plates of the floor valves, concentric to the valve port, (see Figure 9.1-18) prevent diffusion and radiation streaming through the minimal mating surface gaps. These design features limit the transient dose rate at the surface to less than 200 mrem/hr during transfer of radioactive components, and 5 mrem/hr when closed over the reactor ports.

The floor valve is sealed to the fuel transfer port adaptor by double seals, and bolted to the adaptor flange. The movable circular disk which closes off the port opening in the valves is also sealed by double seals.

The rotor driveshaft is sealed by dynamic seals. All seals are double buffered and of elastomer material. The discussion of EVTM elastomer seals in Section 9.1.4.3.2 is applicable to the floor valve seals also.

The position of the floor valve gate (open or closed) is displayed on the floor valve control panel and, if the EVTM is mated to the floor valve, on the EVTM control console.

9.1.4.6.3 <u>Safety Evaluation</u>

The radial shielding limits the dose rate on the floor valve surface to less than the criteria in Sections 12.1.1 and 12.1.2 during transfer of the highest powered spent fuel assembly (for the EVTM floor valve). The floor valve is considered a piece of equipment whose main function is to permit transfer of radioactive components, both fueled and non-fueled, between a machine and a facility. The radiation source is transient and short term (less than 1 min per transfer) in nature. Hence, it results in a low integrated dose.

Another function of the floor valves is to provide axial shielding to replace that normally provided by the port plugs. The axial shielding limits the dose rate to personnel to 5 mrem/hr when placed over a reactor port and to 0.2 mrem/hr when placed over EVST or FHC ports. Personnel cannot receive a direct axial dose because of the large diameter of the floor valve. In addition, the valve is covered by a mating machine much of the time. In all cases, sufficient axial and radial shielding for the EVTM and AHM floor valves is provided to limit the integrated dose to less than 125 mrem/quarter, and dose rates to the zone criteria of Section 12.1.

The floor valve has adequate seals to prevent excessive radioactive release to the RCB and RSB operating floors. Accidental cover gas release through inadvertent opening of a floor valve in the absence of a mating fuel handling machine (EVTM, AHM) on top of the floor valve is prevented by interlocks. One interlock prevents energizing the valve operating motor unless a mating machine is on top of the floor valve. (Electrical power to the floor valve motor is supplied by connection to the mating machine.) Other interlocks prevent (1) depressurizing the buffer gas zone, and (2) raising the closure valve extender, unless both the closure valve and the floor valve are in their closed positions.

As discussed in Section 15.5.2.4, an unlikely accident releasing radioactive cover gas from the reactor leads to a site boundary dose well below the guideline value of 10CFR20.

9.1.4.7 Safety Aspects of the Reactor Fuel Transfer Port Adaptor and Fuel Transport Port Cooling Inserts

The reactor fuel transfer port adaptor (see Figure 9.1-19) is positioned on top of the reactor fuel transfer port and extends from the reactor head to the bottom of the floor valve which is located at the elevation of the RCB operating floor. It serves as an extension of the reactor cover gas containment and provides shielding when irradiated core assemblies are removed from the reactor. The adaptor also guides cooling air from an air blower to a cooling insert inside and below the adaptor.

The function of the cooling inserts, located around the EVST and FHC fuel transfer ports as well as the reactor port (see Figure 9.1-19), is to remove decay heat should an irradiated core assembly in a sodium-filled CCP become immobilized in a fuel transfer port during transfer between the reactor vessel, EVST or FHC and the EVTM.

9.1.4.7.1 Design Basis

The design bases for shielding and radioactive release of the fuel transfer port adaptor are the same as for the EVTM (see 9.1.4.3.1). The reactor, EVST, and FHC fuel transfer port cooling inserts have the capacity to remove decay heat from 20 KW irradiated core assemblies in sodium-filled CCPs to prevent exceeding the 1500°F spent fuel cladding temperature limit specified for unlikely or extremely unlikely events (Table 9.1-2).

9.1.4.7.2 Design Description

The reactor fuel transfer port adaptor extends from the upper surface of the fuel transfer port in the reactor head to the operating floor, see Figure 9.1-19. The upper surface of the reactor fuel transfer port adaptor consists of a flange which is bolted to the flange of the cooling insert. Shielding is provided by a thick, annular lead cylinder surrounding the adaptor cover gas containment tube over its entire length to limit the dose rate at the shield surface to less than the limits given in Sections 12.1.1 and 12.1.2. The lower part of the adaptor is bolted to the reactor head permanently, while the upper part is installed only during refueling.

The reactor fuel transfer port cooling insert extends from the top flange of the adaptor to the fuel transfer port nozzle. The cooling insert uses a cold wall cooling concept, similar to the EVTM. The CCP containing a spent fuel assembly is cooled by thermal radiation and conduction across the argon gas gap to the cold wall which forms the confinement barrier for the reactor cover gas. Ambient air is blown down the outside annulus of the cooling insert, and discharges into the reactor head access area. Air flow from the blower is adequate to limit the cladding temperature of a 20 KW fuel assembly to less than 1500°F. Thermocouples are attached at two places (near the cooling air outlet and near the seals) on the reactor fuel transfer port adapter cooling insert. The thermocouples indicate the need for cooling and will verify adequacy of cooling if the adapter blower is in operation.

The EVST adapter contains a thermocouple on its inner wall to serve the same function as the reactor fuel transfer port adapter thermocouples. The FHC spent fuel transfer port does not contain instrumentation. The decay powers or core assemblies transferred are lower than for the other ports.

The cooling insert is sealed to the rotating guide tube and the EVTM floor valve to form a continuous reactor cover gas pressure boundary through the adapter during refueling. Sealing to the rotating guide tube is by three static elastomer seals with a continuously monitored buffer region between the center seal and each outboard seal. Sealing to the floor valve is by two sets of double static elastomer seals for which the floor valve provides the sealing surface. The buffer region of each seal pair is continuously monitored to detect a loss of seal integrity.

9.1.4.7.3 Safety Evaluation

The transient dose rate from the highest powered spent fuel assembly is less than the criteria in Sections 12.1.1 and 12.1.2 at the surface of the adaptor body. This significant dose rate exists only during the short time (a few minutes) when a spent fuel assembly travels through the adaptor and floor valve into the EVTM. The closest location where personnel can be exposed to the radiation source is more than 10 ft. from the adaptor surface for normal operation, and more than 2 ft. from infrequent maintenance operations. Both locations are on the RCB operating floor, above the adaptor. Therefore, integrated exposures are low.

Cooling of spent core assemblies in the reactor or EVST fuel transfer port is adequate to maintain the assembly cladding temperature below the 1250°F limit for normal operations and anticipated events. During normal operations, the transit time of the core assembly through the port is short (a few minutes) so that there is no significant heatup. In the event that an assembly becomes immobilized in the port, design provisions to maintain the cladding temperature below 1250°F will be used. If immobilization of an assembly is the result of a mechanical failure of the EVTM grapple drive system, the backup cooling system for the port may be turned on to provide the necessary cooling. If immobilization is the result of a loss of power (which would also disable the backup cooling system), the EVTM grapple drive system may be operated manually to raise or lower the core assembly to a location (EVTM or sodium pool) where adequate passive cooling is provided to maintain the cladding temperature within the 1250°F limit. However, those operator actions are required only to maintain fuel temperatures within normal limits. In the unlikely event that a core assembly becomes immobilized in the port for a longer time by coincident drive system mechanical failure and loss of power or failure of the operator to respond to this condition the cladding temperature would exceed 1250°F but would remain below the 1500°F limit for unlikely and extremely unlikely events.

The seals of the reactor fuel transfer port cooling insert are adequate to prevent excessive radioactive release due to cover gas leakage into the RCB from the reactor fuel transfer port adapter portion of the equipment stackup at the port. Radioactivity released from the cooling insert does not exceed the limits set forth in Sections 12.1.1 and 12.1.2. The pressure in the seal buffer regions is continuously monitored to ensure continued seal integrity.

In the unlikely immobilized fuel assembly event seal temperatures increase but remain below the seal limit. One of the sets of cooling insert-to-floor valve seals is in a cooler region than the other set and serves as a backup to the inboard pair.

9.1.4.8 Spent Fuel Shipping Cask

The integrity of the SFSC design will ensure sufficient margins to meet all requirements stipulated in the applicable regulations, especially 10 CFR 71. The shipping cask is discussed in this section only to the extent that conditions to which it is subjected inside the RSB are potentially more severe than those design conditions specified in 10 CFR 71. Regulation 10 CFR 71, paragraph 71.36, states that the cask design shall withstand a hypothetical accident characterized by a 30-ft drop onto a flat, essentially unyielding, horizontal surface without exceeding a specified reduction in shielding and containment of radioactive material. The LMFBR spent fuel shipping cask will be designed to withstand, with no release of radioactivity, a maximum deceleration of 123 g if dropped 30 ft onto an unyielding surface. The largest height for a potential SFSC drop in the CRBRP is the 72-ft vertical distance of the SFSC handling shaft.

9.1.4.8.1 Design Basis

The free fall impact energy of the 72 ft SFSC drop to the bottom of the cask handling shaft shall be limited to an amount less than that experienced in a hypothetical cask accident specified in 10 CFR 71.

9.1.4.8.2 Design Description

The SFSC is handled within the RSB and lowered and raised in the cask shaft by the double reeved RSB bridge crane using rigging specially designed and tested for the SFSC. Preliminary analysis indicates that a 72 ft drop of the SFSC onto the concrete floor of the cask shaft would result in a peak deceleration of 92 g, i.e., less than that produced by a 30-ft drop onto an unyielding surface. The reason for the lower deceleration lies in the difference between an unyielding surface and a concrete slab surface.

If the cask corridor transporter is in place at the bottom of the shaft when the SFSC is dropped, it will function as an energy absorber, further reducing the deceleration loads.

9.1.4.8.3 Safety Evaluation

A drop of the SFSC down the shaft onto the concrete slab will produce a deceleration which is smaller than that of a hypothetical 30-ft drop onto an unyielding surface. If the cask falls on the cask corridor transporter, part of the drop energy will be absorbed by the transporter structure and only the remaining energy will be transmitted to the SFSC. In either case the resulting loads will be substantially less than those for which the cask is
59 designed.

An extremely unlikely accident of a SFSC drop with resulting fission gas release is discussed in Section 15.7.3.2. This accident produces a site boundary dose well below the 10CFR20 guidelines.

9.1.4.9 Safety Aspects of the Rotating Guide Tube (RGT)

The function of the Rotating Guide Tube (RGT) is to provide a means for the EVTM to insert a CCP containing a core assembly into one of four fixed in-vessel transfer positions, and to remove a CCP-contained spent core assembly from an adjacent transfer position, all without requiring an EVTM decoupling-recoupling procedure.

The RGT is shown in Figure 9.1-20. It consists of a straight tube with an eccentric lower end. The RGT is permanently mounted on the top of the reactor fuel transfer port nozzle and remains in the reactor during operation. During the refueling sequence, the RGT is rotated 180o to locate its eccentric lower end over either of two adjacent transfer positions. Following refueling, the EVTM is used to insert a plug into the RGT. Sealing and plug holddown are completed by a cap attached to the reactor nozzle. This cap also serves to secure the RGT during reactor operation. For refueling the cap is first removed, and then the reactor fuel transfer port adapter is mated to the RGT prior to removal of the port plug.

The safety aspects of the RGT that will be covered in this section are: radiation shielding, prevention of radioactive releases, and prevention of mechanical damage.

9.1.4.9.1 Design Basis

Adequate shielding for radiation protection is provided in the design of the shield plug to meet the requirements of Sections 12.1.1 and 12.1.2 and to keep the integrated dose at the same level as the remainder of the reactor head (see Chapter 5.2.1.3).

Activity in the reactor cover gas is contained by plug and cap seals during reactor operation and by adapter and floor valve seals during refueling. Under all conditions, radioactive leakage and diffusion through seals are in conformance with the limits listed in Chapter 5.2.1.3.

Mechanical damage to core assemblies is prevented by control interlocks governing RGT positioning during refueling and the RGT cap locking the RGT in place during reactor operation.

9.1.4.9.2 Design Description

The shield plug is so designed as to limit the total radiation dose rate at the upper end of the RGT to less than 2.0 mr/hr at a distance of 3 feet from the closest accessible surface.

Hermetic sealing is provided by both plug seals and seals in the RGT cap. A means to purge the cap-plug interface volume before removal of the cap is also provided. The pressure boundary consists of metalliz seals between the RGT and the reactor fuel transfer port nozzle, elastomer seals for the RGT drive shafts (static during reactor operation, dynamic during RGT movement for refueling), and static elastomer seals in the RGT gear housing which seal to the RGT cap during reactor operation and are not part of the pressure boundary during refueling. The RGT is sealed to the reactor fuel transfer port adapter cooling insert during refueling as described in Section 9.1.4.7.2. All RGT seals are double with a pressurized, continuously monitored buffer region between each pair of seals. The buffer region pressure is for leak detection and is not necessary for seal effectiveness.

Control logic interlocks prevent improper sequences of core assembly-RGT movement whenever the RGT is in use. During reactor operation, the RGT end cap locks the RGT in position and prevents all tube movement. Also, no electrical power is provided to the RGT during reactor operation.

9.1.4.9.3 Safety Evaluation

The RGT, RGT plug, and RGT cap are so designed that refueling and/or operating personnel will never receive a total dose greater than 125 mrem/quarter. (Actual allowed dose and leakage levels are shown in Chapter 5.2.1.3.)

Double seals and a capability of purging the cap-plug interface volume ensure that gaseous radioisotope leakage from all sources to the head area will never cause a dose rate in excess of that given in Section 12.

Control interlocks are designed to prevent mechanical damage to core assemblies contained in core component pots (CCP) and reactor components by preventing the following actions:

1) Inadvertent attempt to insert an assembly in an occupied location.

2) Motion of the RGT with a CCP or grapple extending below the base of the RGT.

3) Any motion of the RGT during reactor operation.

4) Positioning of the RGT over any position except one of the
61 storage/transfer locations.

9.1.4.10 Safety Aspects of Spent Fuel Storage in the Fuel Handling Cell (FHC)

The primary functions performed in the FHC are to: (1) Receive irradiated core assemblies from the EVTM, (2) provide interim storage for these irradiated assemblies during transfer operations, (3) examine selected irradiated assemblies, and (4) load irradiated core assemblies into casks for shipment off site. Other functions, also performed by the FHC, are to provide service and maintenance of radioactive fuel handling equipment (e.g., grapple replacement and drip pan change-out for the ex-vessel transfer machine). The FHC also provides contingency-storage for low-heat producing core assemblies, in the event of a complete core unloading (i.e., blanket assemblies, control assemblies and removable radial shield assemblies that produce little decay power and are coolable by natural circulation in argon).

These functions are implemented by the following features of the FHC:

1) Radiation shielding

2) Inert gas (argon) atmosphere

3) Viewing capabilities

4) Remote manipulation and handling of core assemblies and other components

5) Cooling of spent fuel assemblies (described in Section 9.13)

6) Packaging of liquid and solid radioactive waste.

The FHC (located as shown in Figure 9.1-2) is a shielded, inerted, alpha-tight hot cell facility located between the EVTM gantry rails below the operating floor of the RSB.

The cell design is based on similar facilities used on other programs (e.g., the FFTF inspection, examination, and maintenance (IEM) cell, and the National Reactor Test Station hot fuel examination facility (HFEF) cell).

The main equipment groups of the facility as shown in Figure 9.1-7 are (1) a spent fuel transfer station for interim storage of up to 3 (2 during normal operation) spent fuel assemblies, (2) a gas cooling grapple for handling bare spent fuel assemblies, (3) a maintenance and service station and pit, (4) a spent fuel examination station, (5) waste container set-down space, (6) CCP storage racks (with no fuel), and (7) a spent fuel shipping cask loading station. In addition, provisions are made around the walls of the FHC to store low-heat-producing core assemblies.

1. FHC Normal Fuel Handling

In a typical spent fuel handling sequence, a spent fuel assembly in a core component pot is lowered through the fuel transfer port (see Figure 9.1-7) by the EVTM, into the spent fuel transfer station directly below the port. A lazy susan assembly, with three transfer positions supported by a stainless steel gridwork, provides the storage locations. Each position holds one fuel assembly, in a sodium-filled core component pot. Decay heat is removed by natural convection to the FHC atmosphere.

The spent fuel assembly is removed from the core component pot by the in-cell crane, using a gas-cooling grapple, and allowed to drip dry. If for some reason not identified as a part of normal procedures, it is deemed necessary to remove a sodium film from the exterior surfaces, the exterior surfaces will be wiped with alcohol wetted swabs.

Then the spent fuel assembly is lowered into the spent fuel shipping cask located in a shaft below the cell floor. The sequence is repeated for the number of assemblies necessary to fill the shipping cask. The above functions within the FHC are performed remotely by operators in the adjacent operating gallery, and can be observed through the viewing windows.

Normal core assembly handling operations in the FHC are conducted with assemblies having a decay heat of 6 kW or less. Infrequently, it may be necessary to examine a high-powered core assembly. This will be done only when (1) it is necessary to complete refueling or to commence reactor startup, or (2) use of the FHC is necessary to recover from an EVTM grapple malfunction occurring while grappled to a high-powered core assembly.

During these operations, special precautions shall be observed, including removal of all other spent core assemblies from the FHC prior to introduction of the high-powered core assembly. In addition, only one core assembly greater than 6 kW shall be permitted in the FHC at any time.

2. Spent Fuel Examination

Spent fuel examination in the FHC is limited to inspecting the exterior surfaces of fuel assemblies to determine their geometrical condition before loading into the spent fuel shipping cask. Spent fuel assemblies will not be disassembled or sectioned in the FHC.

It is planned that only a few selected spent fuel assemblies will be examined, after the plant operation has reached its equilibrium. During the first few refuelings, it is expected that more spent fuel assemblies may be inspected.

The extent of the spent fuel examination covers the following operations, all of which will be performed in the fuel examination fixture:

1) Visual inspection of all exterior surfaces

2) Determination of axial and radial dilation of fuel assembly by measuring its length and distances across flats

3) Measurement of the fuel assembly bow

3. FHC Maintenance

In general, standard maintenance techniques will be used to maintain FHC equipment as have been successfully applied at Argonne West's Hot Fuel Examination Facility (HFEF) at the Idaho Nuclear Engineering
59 Laboratory, and at AI's SNAP reactor test facilities.

Most equipment in the FHC is either modularized and sized such that it can be disassembled and positioned for removal from the cell exterior or repaired in the cell utilizing the remote manipulators.

59 Several access modes are available for the FHC including:

1) Removal of all or part of the roof closure gives access to the FHC from the RSB operating floor.

2) 28" dia. Floor Valve

3) 60" dia. Fuel Transfer Port
59

4) An FHC transfer drawer (gas lock) connects the FHC containment to the FHC operating gallery.

Large equipment (e.g., the crane bridge or trolley) can be removed, i.e., bagged out, from the FHC through the FHC roof closure port using the RSB bridge crane. Once outside the cell the equipment can be transferred to the Large Component Cleaning Vessel (LCCV) for cleaning, if required prior to inspection or maintenance. An alternative is to erect a "green house" (plastic tent) over the FHC roof and part of the RSB operating floor. Large equipment could be removed from the FHC through the roof closure port and set down nearby on the RSB operating floor, within the green house. Through an air lock, suited-up personnel with appropriate respiratory protection could enter the green house and perform the required maintenance tasks.

Suited-up personnel could also enter the FHC to repair nonremovable items or to refurbish the cell. If such an occasion would rise, careful plans would be made in advance for all work to be done after entry to minimize occupancy time in the FHC. All fuel would be removed from the FHC, and the cell would be thoroughly cleaned using the powered manipulator with brushes, dustpans, and vacuum cleaners equipped with HEPA filters before personnel could enter. The personnel would wear appropriate respiratory protection and clothing consistent with personnel safety requirements.

Certain consumable material and spare parts (like motors, switches, light bulbs, etc.) will be supplied for permanent storage in the FHC. Small parts can be easily removed through the transfer drawer and can be moved off the facility by bagging.

59 4. Contamination Control

At regular intervals, swipe samples will be taken at various locations in the FHC and particle fallout samples (dishes) will be analyzed to monitor the amount of alpha-contamination. The samples will be compared to established contamination levels and will provide warning of any contamination buildup. If excessive contamination buildup on the FHC liner is indicated, cleaning measures will be initiated. The steel liner on the FHC walls facilitates decontamination.

One method of minimizing the radioactive spread of alpha contamina-
49 tion through the EVTM and SFSC to other facilities will be by continuously filtering the FHC argon atmosphere. Alpha emitting particles suspended in the argon atmosphere will be removed through an HEPA filter bank installed upstream of the FHC argon gas circulation blowers.

5. Grapple Cleaning Operations

The maintenance-service station contains equipment for alcohol cleaning EVTM and FHC grapples. The grapple cleaning equipment is designed to contain and control the supply of alcohol prior to cleaning and the alcohol-sodium products after the cleaning. The EVTM grapple enters the FHC through the maintenance valve in the FHC roof and is inserted in the cleaning tank. Approximately 13 gallons of ethyl alcohol are transferred from the fill tank, located in the operating gallery, to the cleaning tank in the FHC. The alcohol fill tank is a closed tank designed to operate with argon cover gas. Any sodium adhering to the grapple is removed by reaction with the alcohol. Following cleaning, the alcohol is drained from the cleaning
tank and stored in a closed-top drum until ready for removal from the FHC 25
to be treated as liquid radioactive waste.

The grapple cleaning operations take place in the argon atmosphere of the FHC, eliminating flammability problems due to alcohol vapor or hydrogen which is relased during the cleaning process. There is no air in the FHC and oxygen is limited to 75 ppm, which is several orders of magnitude too low to support combustion. The hydrogen generation is very small, about 0.08 lb per grapple cleaning based on removing 2 lb of sodium. The argon circulation rate of 8000 cfm through the FHC and the inlet and exhaust location in the cell assure mixing and dilution of the hydrogen
44 with argon. In addition, a flammable vapor detector is provided in accordance with RDT F5-9T (Sodium Removal Process) requirements. The Inert Gas Receiving and Processing System maintains acceptable hydrogen (and nitrogen) levels in the FHC by purging cell gas to CAPS when required. Part of the hydrogen-argon mixture will react in the CAPS catalytic oxider and be converted to water. The remaining hydrogen will be purged with CAPS-processed cell argon to the HVAC vent system.

59 9.1.4.10.1 Design Bases

Adequate shielding is provided in the FHC containment structure for
49 radiation protection outside the cell, to meet the requirements and radiation zone criteria of Section 12.1.

Radioactive releases and contamination from spent fuel assemblies that are being prepared for shipment in the FHC are contained within the FHC by proper sealing or closure welding of penetrations. Radioactive leakage and diffusion through seals, in the unlikely event of release of the entire fission gas inventory of a fuel assembly, are limited to well below the criteria of 10CFR100.

Criticality of fuel assemblies in the spent fuel transfer station in the FHC is not possible because only three fuel assembly locations are provided.

The spent fuel transfer station design considers all normal loadings in combination with the loads from an SSE in maintaining the necessary physical separation. The FHC roof closure is designed to absorb the load of the heaviest equipment handled by the RSB bridge crane over the FHC: (a) for the main hook, lowered at the maximum crane speed (5 fpm), and (b) for the auxiliary hook, accidentally dropping from the maximum handling height to which it is raised, onto the center of the roof closure without affecting the integrity of the fuel separation lattice. The FHC is located such that heavy equipment not belonging to the fuel handling and storage system is not carried over it by the RSB bridge crane.

The spent fuel transfer station within the FHC is designed so that movement of the lazy susan will not occur while a CCP is being inserted or withdrawn. This design condition prevents mechanical damage to the CCP or its contents.

Monitoring instrumentation is provided for the FHC for conditions that might result in a loss of the capability to remove decay heat, and to detect excessive radiation levels.

9.1.4.10.2 Design Description

The top of the FHC is located at the operating floor of the RSB, as shown in Figure 9.1-7. Sufficient shielding is provided so that the radiation level above the FHC does not exceed the radiation Zone I criteria, see Section 12.1. This shielding is provided by the cell's roof closure assembly, a load-bearing structure which is part of the RSB operating floor. The FHC side wall facing the operating gallery is shielded by high density concrete to protect the operating gallery against radiation dose rates exceeding the radiation Zone I criteria. The other walls and the floor are shielded by conventional concrete to protect the neighboring vaults and the spent fuel shipping cask handling corridor against radiation, see Section 12.1. All windows, and port penetrations through the roof, walls, and floor are stepped to limit radiation streaming in the gaps. The main source of radiation in the FHC is spent fuel assemblies in the spent fuel transfer station.

The pressure boundary of the FHC is sealed using welded seams and elastomer seals. There are three types of elastomer seals used: static, dynamic, and inflatable. All of these seals are provided in redundant pairs and have essentially zero leakage (i.e., leakage is almost entirely due to permeation through the seal material). The dynamic and inflatable seals have slightly larger leakage than the static seals on a comparable basis. All three types of elastomer seals have a buffer space between seal pairs. The buffer space for static seals is used for periodic leak testing. Effectiveness of the seals does not depend on presence of a buffer gas. Dynamic and inflatable seals are provided continuously with a buffer pressure between the double seals. This pressure is monitored continuously for leak detection and is not required to prevent seal leakage, although it would mitigate an inner seal leak. The transfer drawer inflatable seals are the only ones which depend on a continuous source of electrical power and inflation gas for operation. In case of loss of off-site power or gas supply, a check valve in the inflation piping will prevent deflation. Should a failure cause the seals to deflate, additional in-leakage to the IHC would occur. (The valves are closed during normal operation to provide more sensitive seal leak detection.) Since the supply valves fail open, loss of off-site power would not affect seal inflation.

The liner seams on the cell interior walls, roof and floor, and welded penetrations through the FHC walls, roof, and floor are alpha-tight welded and inspected. Fuel transfer ports, the 28-in. maintenance floor valve, window seals, utility penetrations, and slave manipulator penetrations each have double, static elastomeric seals. The 28-in. maintenance port FWDR valve also has double, dynamic elastomeric seals. Sealed cover glasses are provided on the interior side of the window penetrations.

The 28-in. maintenance port floor valve and equipment transfer drawer are sealed by double inflatable seals. The floor valve inflatable seals are active only during periods of valve use. At other times the maintenance port is sealed by a plug with double static seals. The equipment transfer drawer inflatable seals are on the closure doors on each side of the penetration and are pressurized except when a door is open.

The spent fuel transfer station within the FHC is shown in Figure 9.1-8. A maximum of 3 spent fuel assemblies in CCPs can be stored in this interim storage; however, in normal operation, a maximum of 2 fuel assemblies will be stored (1 storage position left empty). The storage positions within the transfer station consist of short, tapered, cylindrical inserts at the bottom, the middle, and the top of a substantial supporting steel grid structure. Each CCP is held in place by the three cylindrical sections. The center-to-center distance of the three storage positions is about 25 in. Each position can hold only one spent fuel assembly in a CCP.

During loading of the spent fuel shipping cask (SFSC), a cask-FHC seal assembly forms a gas-tight extension of the FHC containment to the cask interior. The SFSC is, therefore, connected to the FHC atmosphere and is separated from the air atmosphere of the cask corridor.

The FHC roof closure assembly consists of a large north closure plug (21-ft by 18.5-ft) and a smaller south closure plug (8-ft by 18.5-ft), both joined by a cross beam assembly.

The two closure plugs are composite structures consisting of 34.5-in. thick reinforced concrete, enclosed on four sides by a steel liner, and resting on five 8.5-in. thick steel shield plates. The entire 43-in. thick composite structure provides sufficient radiation shielding and is designed to support normal structural loads as well as the accidental impact loads given in the design bases.

The heaviest load carried over the FHC roof is the EVTM floor valve (9 tons). The lift height of the EVTM floor valve is limited to 2 ft. by administrative controls. All heavy maintenance equipment is transported by the Large Component Transporter (LCT) between the RSB and RCB. Maintenance equipment weighing more than 25 tons and the LCT itself are handled only by the double reeved main hook of the RSB bridge crane. Only maintenance equipment weighing less than 25 tons may be handled by the single reeved auxiliary hook. A load dropped from the auxiliary crane could impact the LCT when it is stationed above the FHC roof. Most or all of the impact energy would be absorbed by the LCT. Seismic restraints prevent equipment loaded on top of the LCT from toppling onto the RSB operating floor or FHC roof during an earthquake. Normal operating procedures require large maintenance loads to be fastened to the LCT seismic restraints before disconnecting them from the RSB bridge crane.

The spent fuel transfer station occupies an approximately 68-in. square corner of the larger FHC maintenance pit. The pit covers an open area approximately 9.5-ft by 12-ft on the FHC floor, and extends 18 feet
44 deep below the FHC floor. The entire pit is lined with a stainless steel liner.

The design and construction of the spent fuel transfer station are in accordance with codes, standards, and specifications listed in Section 3.8
44 for Seismic Category I structures.

Sodium leak detectors monitor the space below the spent fuel
44 transfer station. An argon gas radio activity monitor is provided. An area monitor measures gamma radiation activity on the RSB operating floor above the FHC.

59 59 9.1.4.10.3 Safety Evaluation

Adequate shielding for radiation protection on the operating floor of the RSB, in the operating gallery, and below the FHC floor in the Spent Fuel Shipping Cask (SFSC) handling corridor are provided by the FHC design. Overall dose rates in these locations are less than 0.2 mrem/hr., permitting
49 routinely occupied area access (see radiation zones in Section 12.1).
44

Transient dose rates due to high-powered spent fuel assemblies are less than 200 mrem/hr at the surface of the fuel transfer port in the FHC
49 roof plug. The statements made in 9.1.2.1.3 regarding the acceptability of integrated doses resulting from the allowed transient dose rates are also applicable for the fuel transfer port in the FHC roof plug.

The shielding provided to limit streaming dose rates from gaps around the fuel transfer port plugs in the FHC roof and floor reduce the integrated dose rate to 2 mrem/hr, in accordance with the shielding criteria of
44 20 12.1.2.1. FHC shielding is described in Section 12.1.

The FHC has adequate seals and liners to prevent excessive radioactive emission into other areas of the RSB. Section 15.5.2.4 analyzes a limiting case and shows it to be acceptable. Radioactive releases of fission gas due to diffusion through elastomeric seals do not exceed the criteria of Sections 12.1.1 and 12.1.2.

The small number of fuel assemblies which can be present at any
49 time in the FHC (3 at the most) and the large separation distance of fuel assemblies in the spent fuel transfer station ensures subcriticality in the FHC at all conditions. The penetration of assembly positions through the top grid plate preclude any other spent fuel assembly insertion into
59 44 the spent fuel transfer station, and the position's diameter permits only one fuel assembly per position. More than 30 closely spaced fuel assemblies are required in order to approach criticality, as shown in Figure 4.3-30.

Seismic restraints of the spent fuel transfer station structure prevent any shifting within the vault during an SSE seismic event. Seismic forces will not change the separation distance between spent fuel assemblies due to the gridwork restraints between storage positions. The design prevents movement of the lazy susan sufficient to cause failure of the CCP or
44 damage a spent fuel assembly from either a seismic event or from inadvertent rotation.

44

The RSB design, elevation of the operating floor, and the welded steel liner of the FHC prevent the entry of flood or rainwater into the FHC. No significant quantities of moderating fluids are used in either the FHC or its argon cooling system.

44

In spite of the design consideration (see Section 9.1.2.2.2) a hypothetical heavy weight due to maintenance operations has been postulated to
44 drop on the FHC roof closure. The load limit of the RSB bridge crane auxiliary hook was selected as the hypothetical weight for this "umbrella" event. A drop height of two feet above the FHC roof closure was assumed. (The FHC roof closure is at the same elevation as the operating floor.)

A stress analysis of the composite roof closure structure was performed to determine maximum deflections and stresses due to the postulated impact load. The analysis was based on the following ground rules:

1) The load drops onto the center of the larger of the two closure
44 plugs; i.e., the north closure plug. Deflections and stresses in the north closure plug would be larger than those in the
44 smaller, stiffer, south closure plug.

2) Conventional formulas for a peripherally supported rectangular plate under concentrated load were used to determine elastic deflection and spring constant of the reinforced concrete slab which forms the upper part of the closure plug.

3) Conventional formulas for end supported beams were used to determine elastic deflection and spring constant of the steel shield plates underlying the reinforced concrete slab.

4) The deformations of the reinforced concrete slab and the underlying steel plates were equalized, since it is a composite structure.

The results of these analyses are as follows:

The combined concrete slab and steel shield plates deform plastically due to the impact load. Based on effective elasto-plastic load versus deflection curves for reinforced concrete and ASTM-A.283GRD steel, the impact energy
is absorbed by a 2.15-inch deflection of the composite structure. 20

Fictitious elastic impulse stresses were calculated for the reinforced concrete slab and related to qualitative degrees of concrete damage, using an elastic design index. The relation between elastic design index and degree of damage is based on test data with 104 reinforced concrete beams of varying reinforcement and varying impact velocities. The damage factor scale ranges from 0 to 5, and is defined in Table 9.1-3.

The postulated impact load resulted in a damage factor of 4 for the reinforced concrete. This implies heavy damage but no gross structural failure of the reinforced concrete.

The deflection of the steel shield plates(underlying the reinforced
44 concrete slab)due to the postulated impact load results in a ductility factor
of 2.1. The ductility factor is defined as ratio of maximum dynamic deflection to deflection at the effective elastic limit, and is a measure of the degree of plastic deformation. Reference 2 reports that ductile structures can withstand impact loads with ductility factors in the order of 20 to 30 without failure. A ductility factor of 2.1 means that about one-half of the deflection due to the impact load is in the elastic range, the other half is in the plastic range. This implies some permanent deformation but no loss of structural integrity of the steel shield plates.

From the above considerations, it was concluded that the accidental drop of the heaviest object carried over the FHC roof could not lead to a gross structural failure of the roof plug, and could, therefore, not lead
44 to a change in lattice spacing of the fuel storage positions.

Such a drop load could, however, lead to a failure of the FHC roof closure seal. This could result in some air intrusion into the FHC since the
44 cell is at a slight negative pressure. After pressure equalization, radioactive argon gas could escape from the FHC into the RSB. The radioactivity of the FHC atmosphere is continuously monitored and controlled to a low level,
44 as specified in PSAR Section 16.3.10.3.1. A complete, instantaneous release of all FHC argon gas into the RSB operating area would result in a site boundary dose less than the guideline values of 10CFR20.

No safety consequences would ensue if the postulated impact accident were to occur while a spent fuel assembly was being handled by the FHC crane with cooling grapple. Both the grapple and crane are designed according to Seismic Category I requirements. The FHC crane bridge and trolley have seismic restraints which prevent their "jumping off" from the track due to seismic or transmitted impact loads. A spent fuel assembly held by the cooling grapple will not drop or jump off the grapple during the postulated impact accident.

59 20

Amend. 59
Dec. 1980

References

59

1. "Design of Structures to Resist Nuclear Weapon Effects," ASCE Manual of Engineering Practice No. 42, 1964 Edition. 20

2. Safety Analysis Report for LMFBR Spent Fuel Shipping Cask Model I-(182)-1,
44 Aerojet AMCO Report AMCO-02-R-107, as revised.

TABLE 9.1-1

44 EVST DECAY HEAT LOADS AND SODIUM OUTLET TEMPERATURES

Condition	Heat Load (kw)	Sodium Coolant Outlet Temperature (^{o}F)*
44 20 Design Maximum load	1800	∿510
46 44 20		
Emergency Both normal loops failed	1800	< 775

44 *Sodium inlet temperature for all conditions is 400^{o}F

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TABLE 9.1-2

IRRADIATED FUEL ASSEMBLY STEADY-STATE DESIGN TEMPERATURE LIMITS

Frequency Class or Description of Operation	Cladding or Sodium Temperature Limit (°F)	Comment
Normal operations and anticipated events		
Handling	1250	Additional clad strain ≤0.1% in 8 hours
Short term storage (FHC)	1150	Additional clad strain ≤0.1% in 1 week
Long term storage (EVST)	1000	Additional clad strain insignificant at all times of interest
Unlikely and extremely unlikely events	1500	No sodium boiling
Reactor reinsertion or off-site post irradiation examination	1175	Additional clad strain ≤0.002% in 1.5 hours
	1125	Additional clad strain ≤0.002% in 8 hours
	1000	Additional clad strain insignificant at all times of interest

44

TABLE 9.1-2A

SPENT FUEL ASSEMBLY CLADDING TEMPERATURES
(Sheet 1 of 2)

Location of Fuel Assembly	Frequency Class (3)	Peak Fuel Assembly Cladding Temperature (°F)
CCP in EVST Storage Location (20 kW Assembly)		
Normal operation	Normal	660(1) (510° pool)
Natural convection loop cooling	Unlikely	750(1) (600° pool)
CCP in FHC (15 kWt Assembly)		
Argon circulation system operative	Unlikely	1060(1)
Argon circulation system inoperative	Extremely Unlikely	1265
CCP in FHC (6 kW Assembly)		
ACS operative	Normal	710
ACS inoperative	Unlikely	900°
CCP in EVTM Cold Wall (20 kW Assembly)		
Cold wall blower on	Normal	1240(1)
Cold wall blower off		
<30 min.	Anticipated(4)	935
>30 min.	Unlikely	1350(1)
CCP Immobilized in EVTM Stack Assembly (assemblies below the cask body assembly, see Figure 9.1-14) (20 kW Assembly)		
<30 min.	Anticipated(2)	1230
>30 min.	Unlikely	1415(1)

TABLE 9.1-2A

SPENT FUEL ASSEMBLY CLADDING TEMPERATURES
(Sheet 2 of 2)

Location of Fuel Assembly	Frequency Class (3)	Peak Fuel Assembly Cladding Temperature (°F)
CCP Immobilized in Reactor Fuel Transfer Port (20 kW Assembly)		
Blower on		
<30 min.	Anticipated(2)	950
>30 min.	Unlikely	1444(1)
Blower off (<30 min.)	Anticipated(2)(4)	970
Blower off (>30 min.)	Extremely Unlikely	1482
CCP Immobilized in EVST Fuel Tranfer Port (20 kW Assembly)		
Blower On		
<30 min.	Anticipated(2)	<1250
>30 min.	Unlikely	1389(1)
Blower off (<30 min.)	Anticipated(2)(4)	<1250
Blower off (>30 min.)	Extremely Unlikely	1482
CCP Immobilized in FHC Spent Fuel Transfer Port (15 kW Assembly)		
Blower on		
<30 min.	Anticipated(2)	<1225
>30 min.	Unlikely	1249(1)
Blower off (<30 min.)	Anticipated(2)(4)	1225
Blower off (>30 min.)	Extremely Unlikely	1275 (2 hrs)

Footnotes to Table 9.1-2A

(1) Steady-state temperature.

(2) For a stopped CCP with the fuel transfer port blower inoperative, the operator is required to take action within 30 minutes to raise or lower the CCP.

(3) Loss of off-site electrical power during a fuel handling operation is classified as an unlikely event. This classification is based upon the number of hours per year that this fuel handling equipment is operating and the conditional probability of loss of off-site power at such a time. (For the probability of loss of off-site power, see Question Response 222.24.)

(4) Upon failure of power to the EVTM, the operator is required to connect a temporary power cable to a Floor Service Station.

TABLE 9.1-2B

TEMPERATURE AND PRESSURES FOR FUEL HANDLING EQUIPMENT

	Design Limits Pres. (Psig)	Design Limits Temperature (°F)	Maximum Calculated Values for Normal Operations and Off-Normal Design Events Pressure (Psig) Normal	Pressure (Psig) Off Norm.	Temperature (°F) Normal	Temperature (°F) Off Normal
EVST	15	775 (off normal)	10" WG	8	535	625°
EVTM	15	435	10" WG	11.5	400	435°
FHC	+2.5	225	-3" WG	+1.44	110	185°
Inflatable Seals	50	250	30	50	ambient*	250°**
EVST Cooling Sleeve Seals	-	500	-	-	110	500°
EVST Port Plug Seals	-	250	-	-	142	183°
EVTM Static Seals	-	250	-	-	80	250°
RFTP Seals	-	500	-	-	182	242°
FHC Static Seals	-	250	-	-	110	185°

* Closure valve and floor valve transfer of a 20KW assembly could raise the seal temperature to 150°F.

** Closure valve and floor valve steady-state (more then 12 hours0 270°F.

TABLE 9.1-3

DAMAGE FACTOR SCALE

Damage Factor	Qualitative Degree of Damage to Reinforced Concrete
0	No damage
1	Slight damage
3	Moderate damage
4	Heavy damage
5	4-inch crack widths

20

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April 1978

TABLE 9.1-3A

RESULTS OF EVTM FULL-SCALE HEAT TRANSFER TESTS

Column 1	2	3	4	5
	Experimental Data		Extrapolation to EVTM Configuration	
Cooling Mode	Fuel Assembly Decay Heat (kw)	Peak Clad Temperature (F)	Fuel Assembly Decay Heat (kw)	Peak Clad Temperature (F)
Forced Convection (1260 cfm air)	15.5	1250	20.8	1250
Natural Convection	15.7	1350	20.15	1500

44

46 44 Table 9.1-4 has been intentionally deleted.

44

Table 9.1-5

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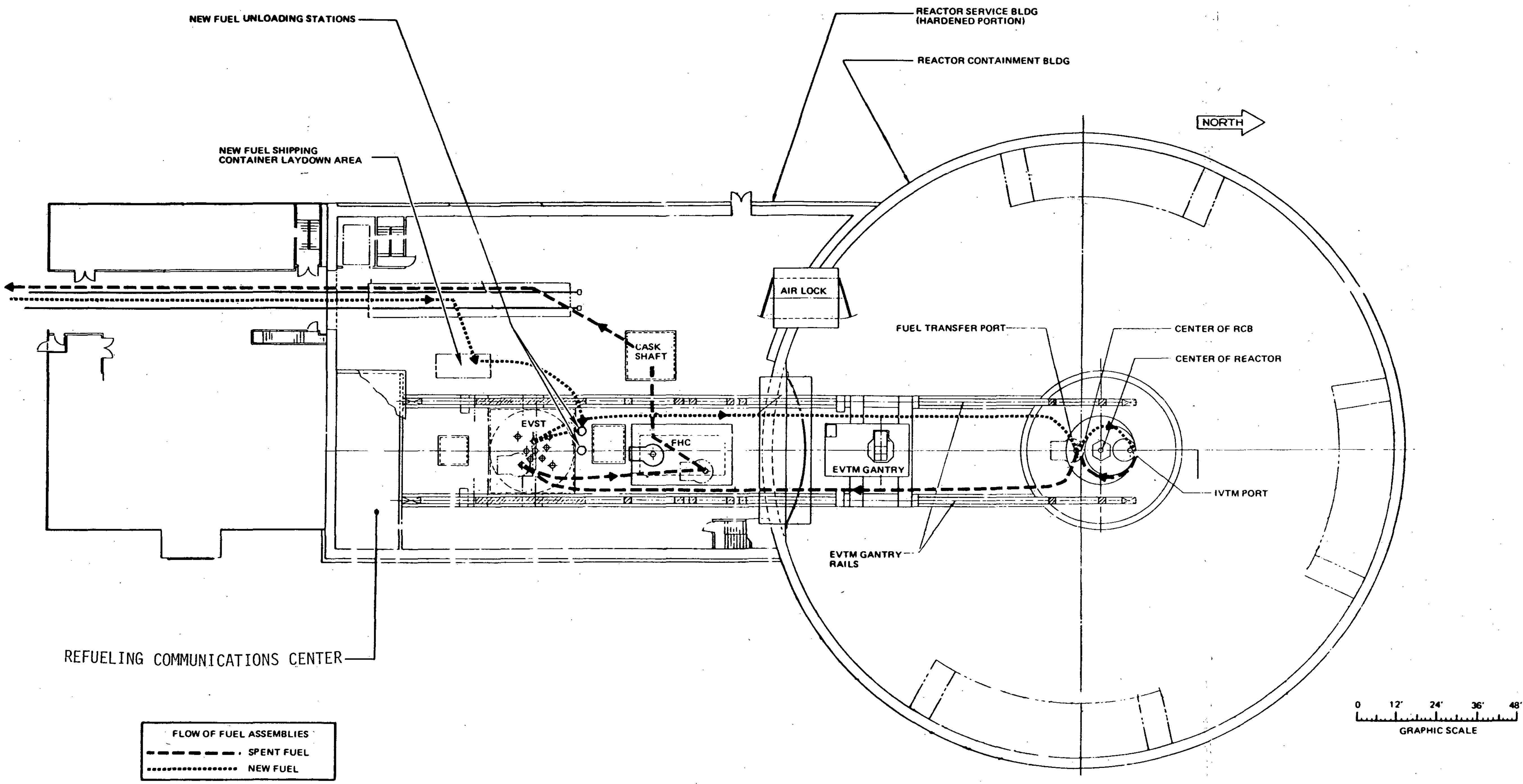

Figure 9.1-1 General Arrangement Plan of Fuel Storage/Handling Equipment and Facilities

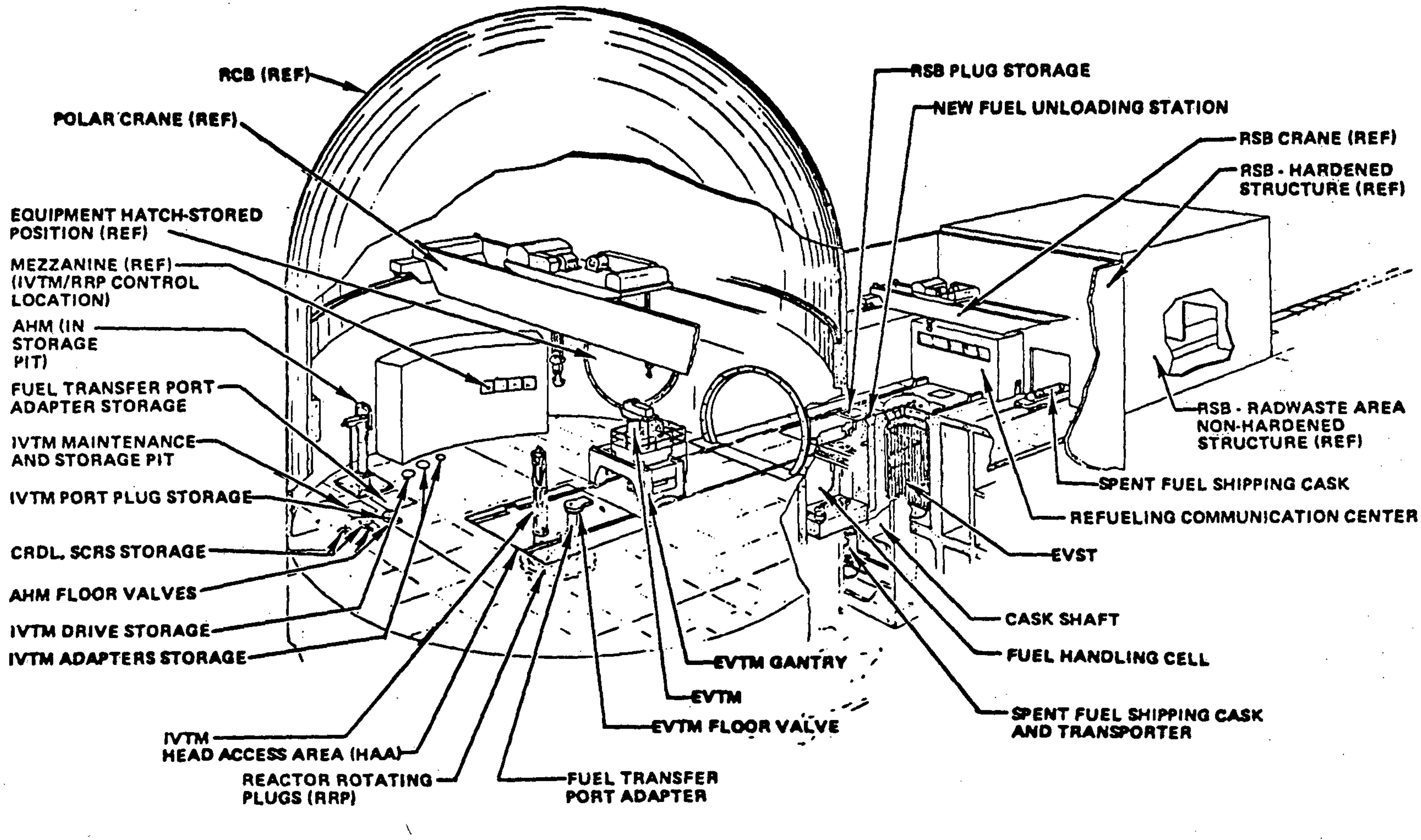

FIGURE 9.1-2 Arrangement of Fuel Storage/Handling Equipment and Facilities (Perspective View)

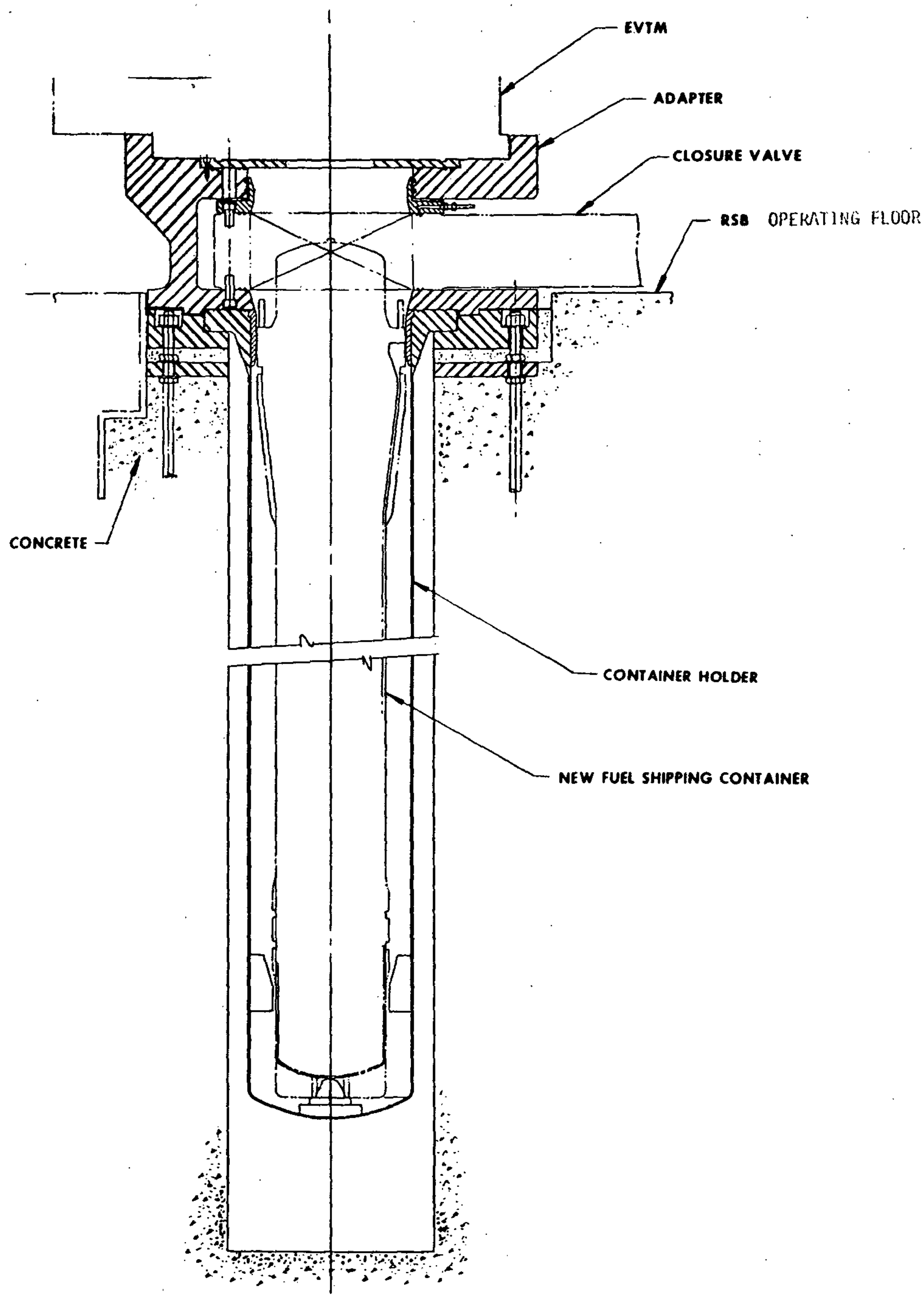

FIGURE 9.1-3. NEW FUEL UNLOADING STATION

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Figure 9.1-4

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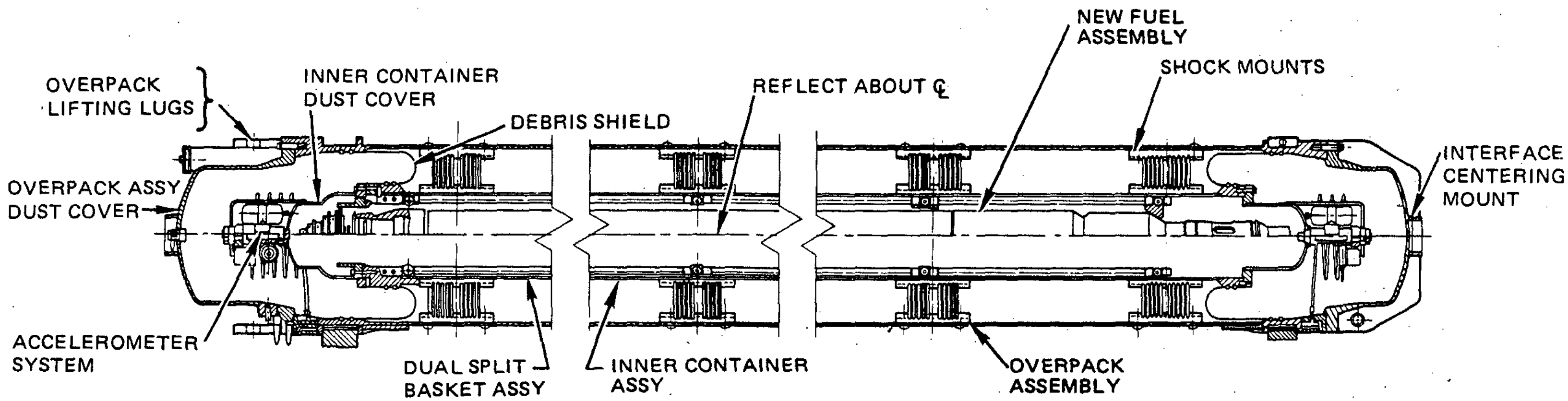


FIGURE 9.1-5 New Fuel Shipping Container Concept

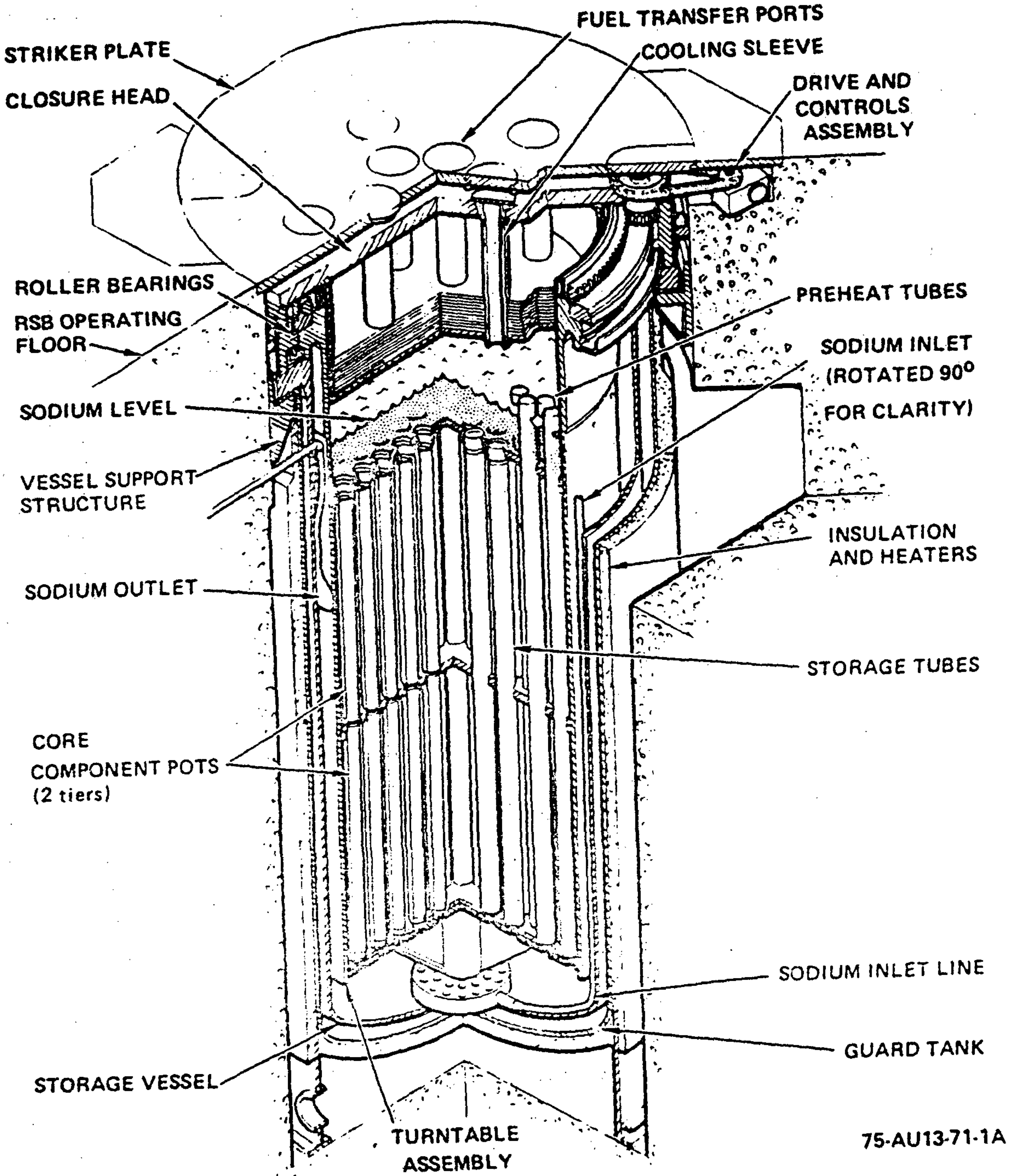


Figure 9.1-6 Ex-Vessel Storage Tank (EVST)

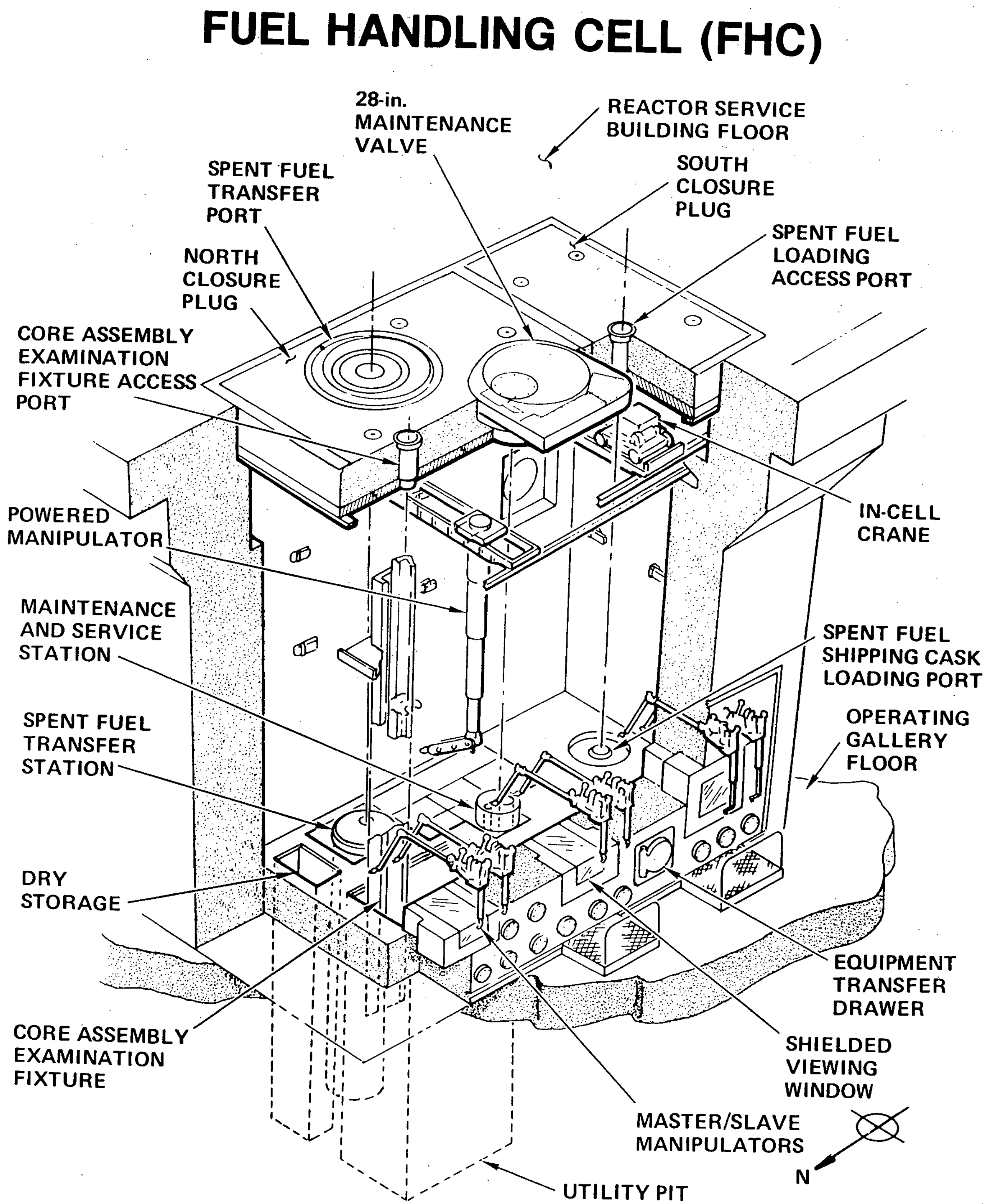


FIGURE 9.1-7 Fuel Handling Cell

Amend. 59
Dec. 1980

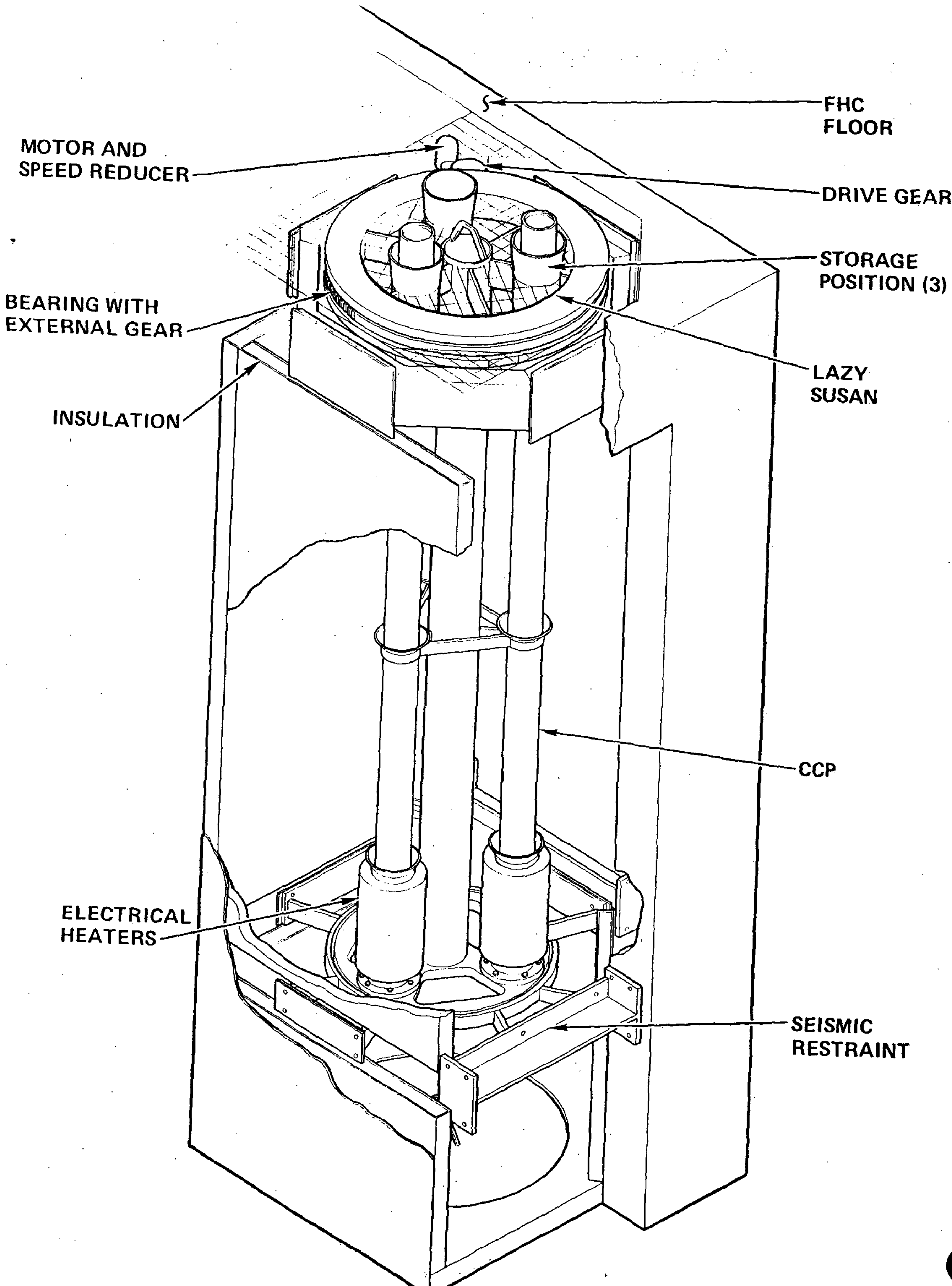


FIGURE 9.1-8. FUEL HANDLING CELL SPENT FUEL TRANSFER STATION

Amend. 44
April 1978

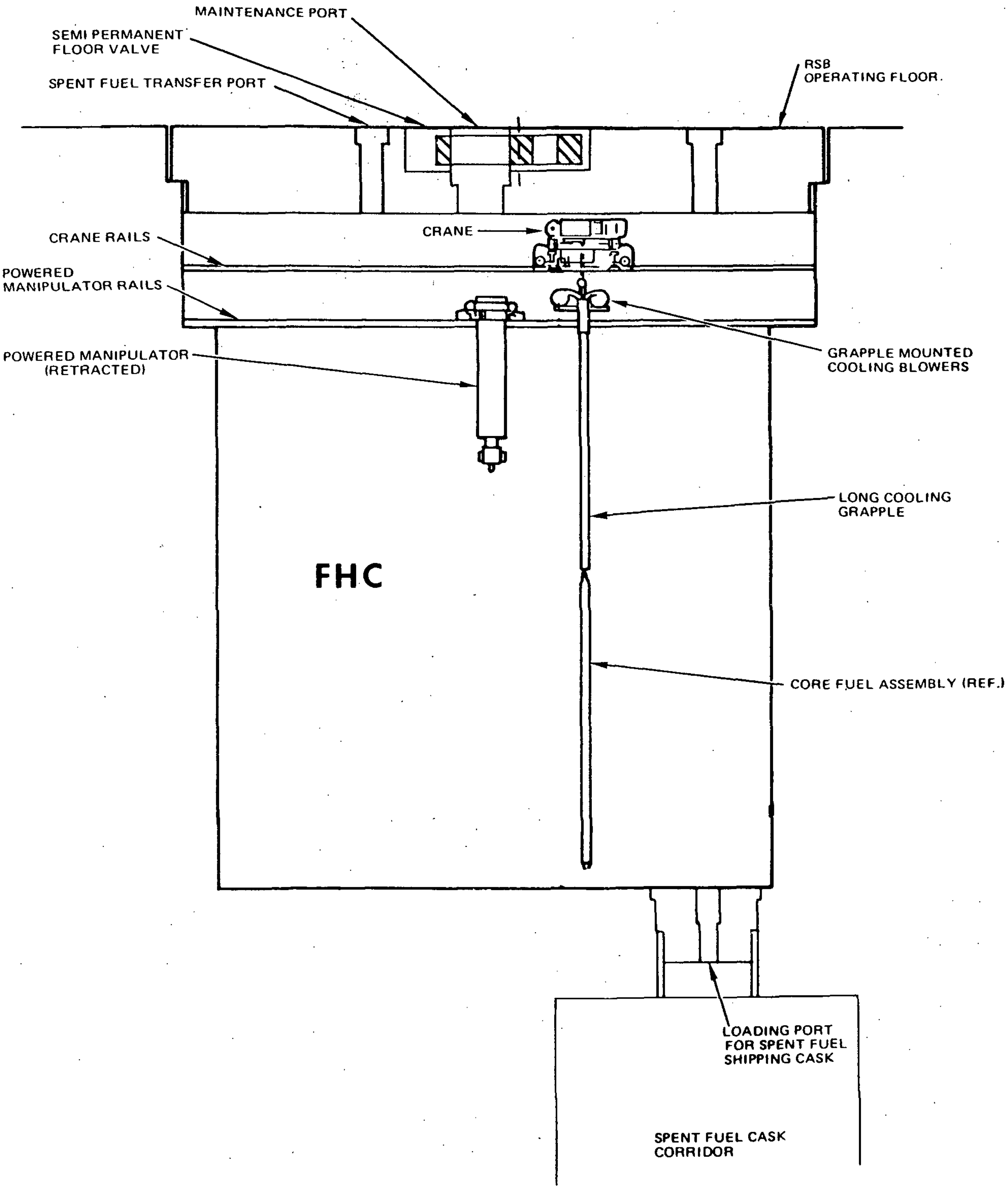


FIGURE 9.1-9. SCHEMATIC OF CRANE-HANDLED GAS COOLING GRAPPLE IN FHC

Amend. 44
April 1978

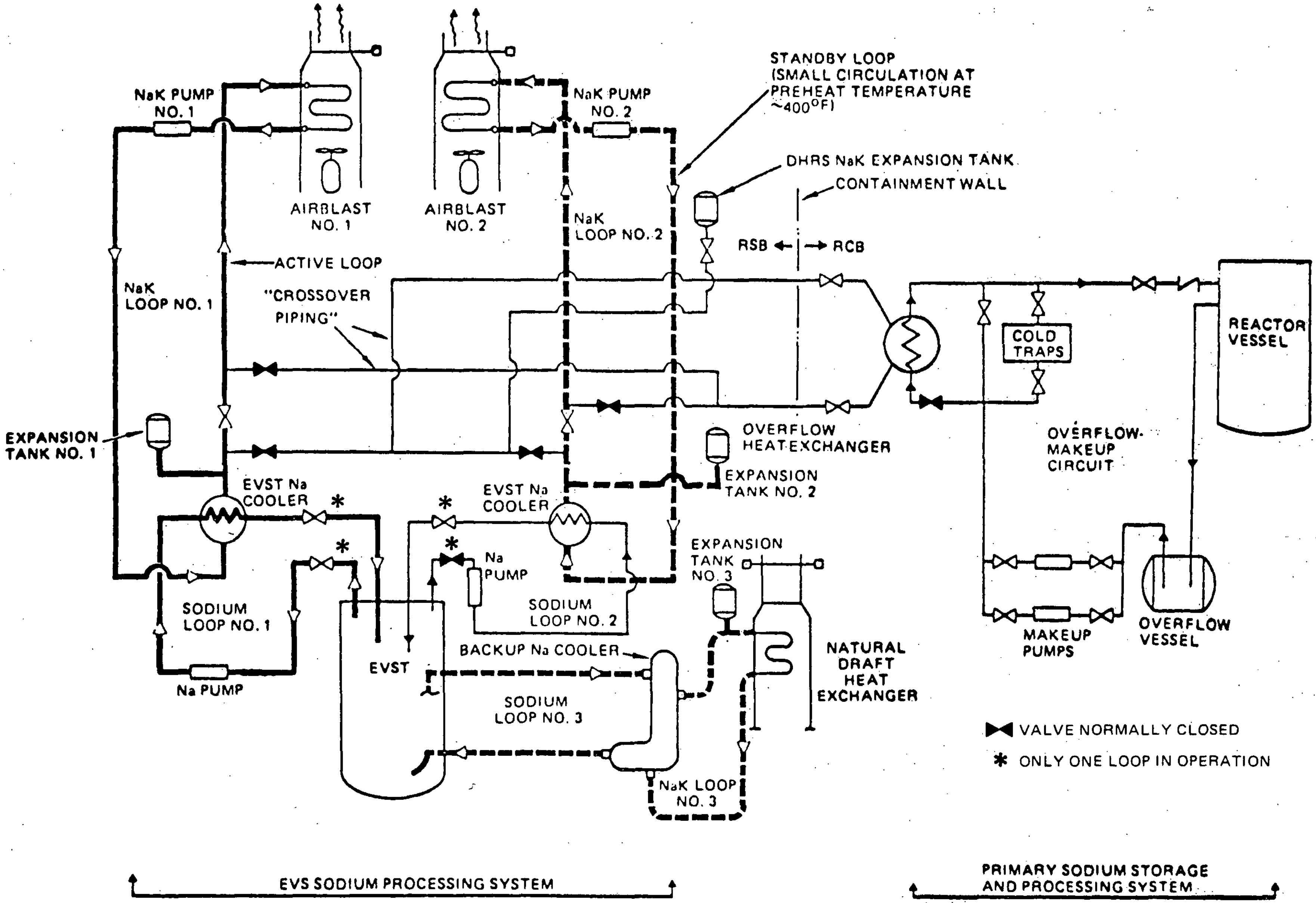


FIGURE 9.1-10. SODIUM-NaK FLOW DURING EVST NORMAL COOLING OPERATION

Amend. 44
April 1978

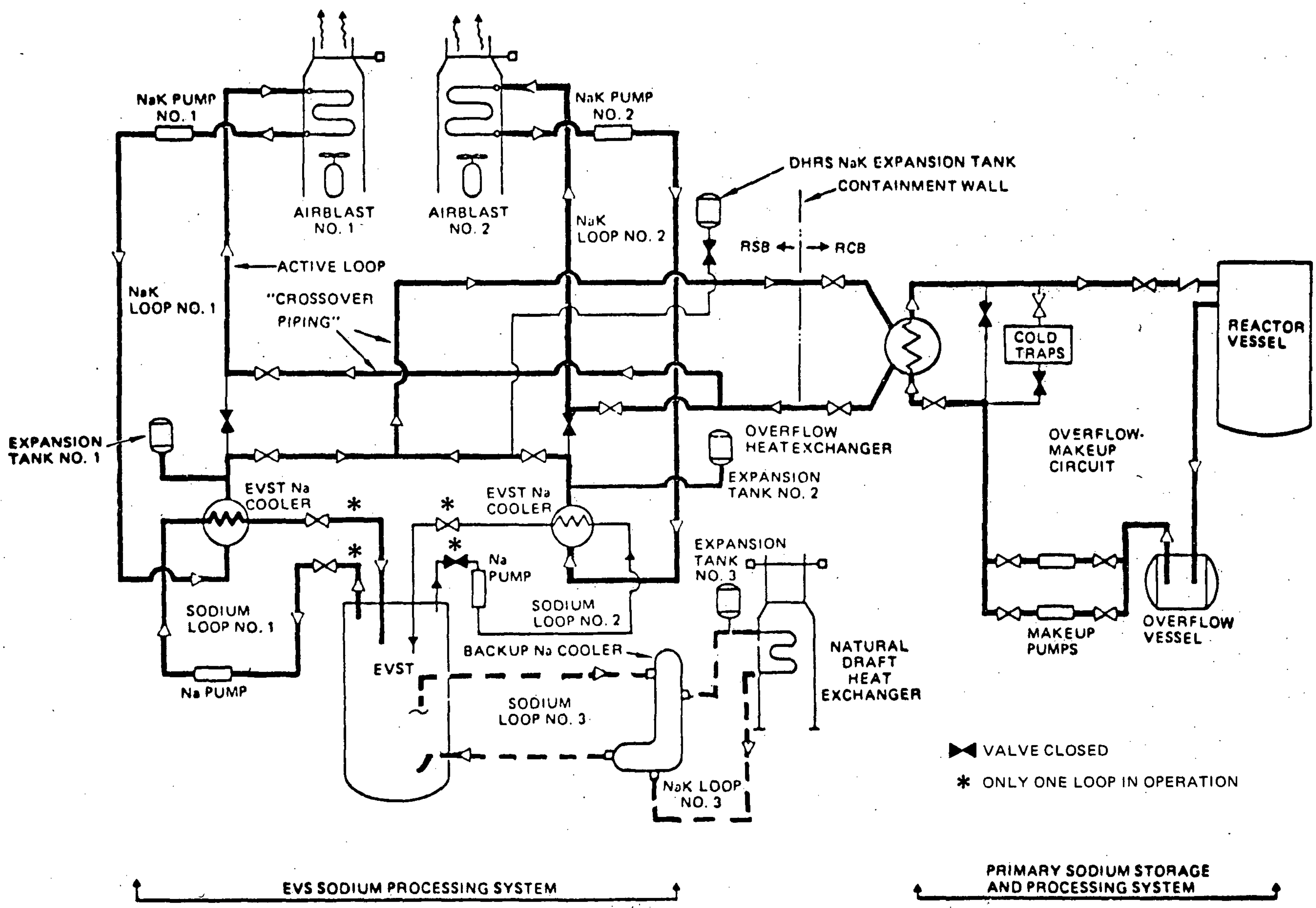


FIGURE 9.1-11. SODIUM-NaK FLOW DURING EVST-PLUS-REACTOR DECAY HEAT REMOVAL

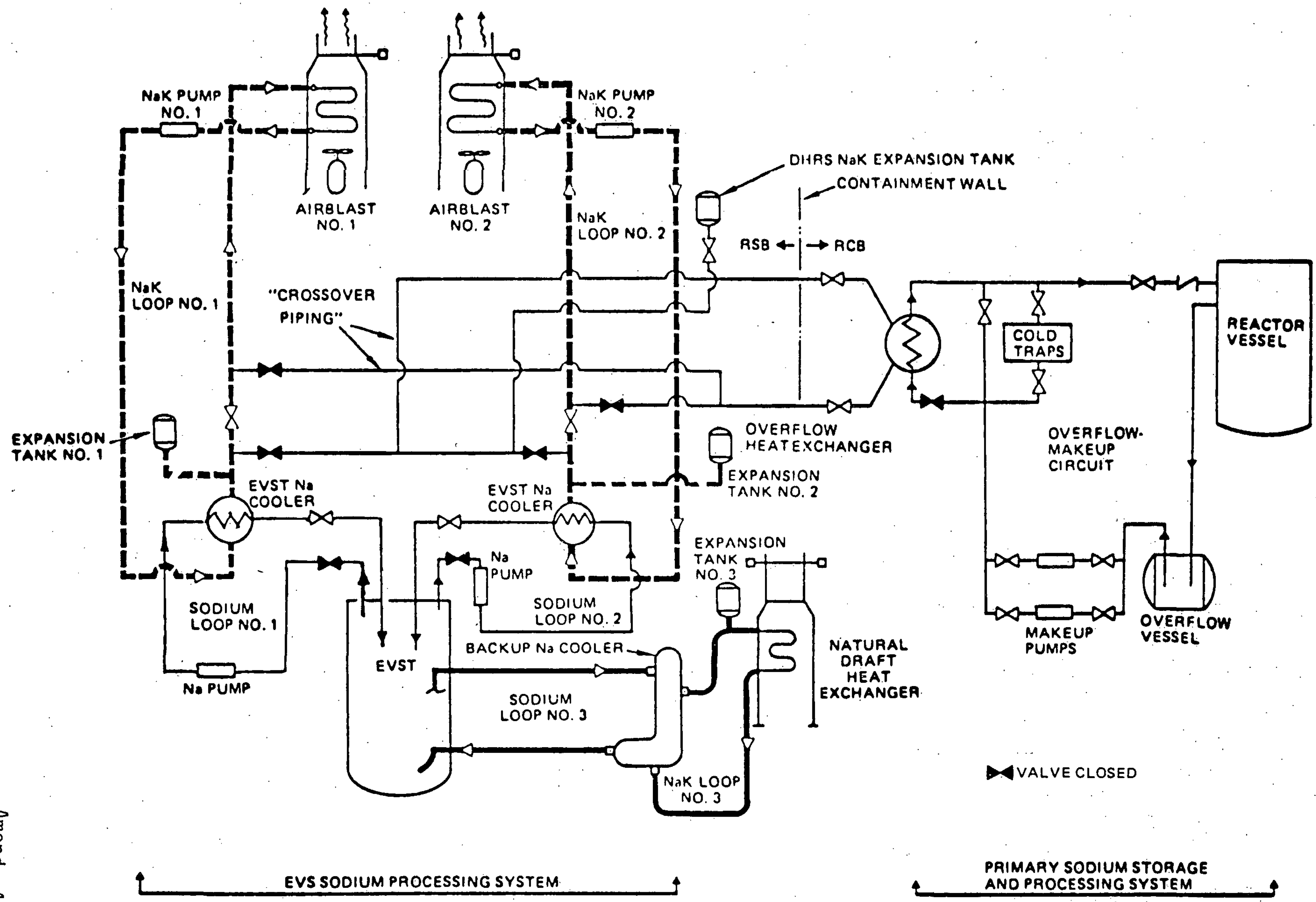


FIGURE 9.1-12. SODIUM-NaK FLOW DURING OPERATION OF THE EVST BACKUP COOLING CIRCUIT

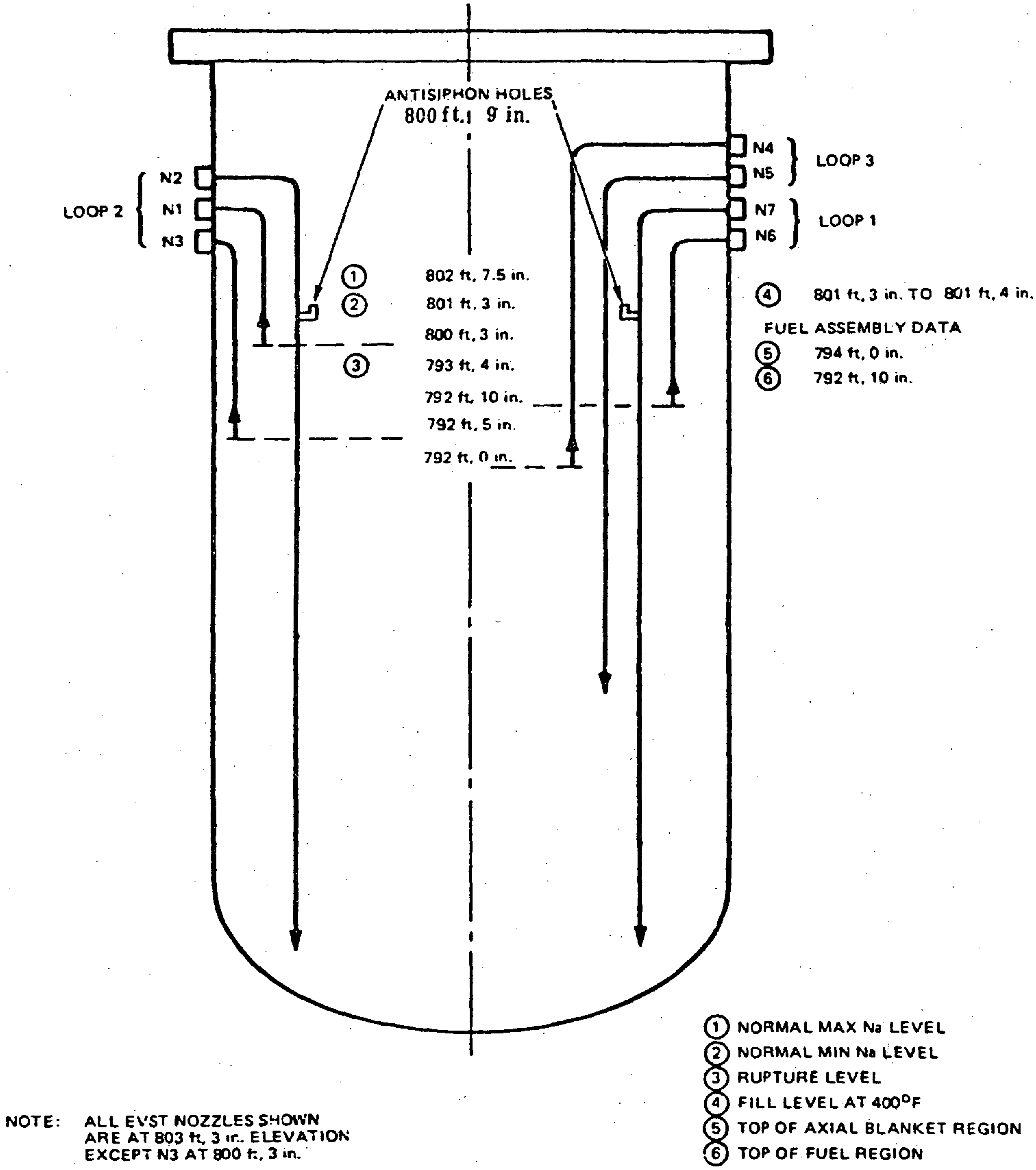
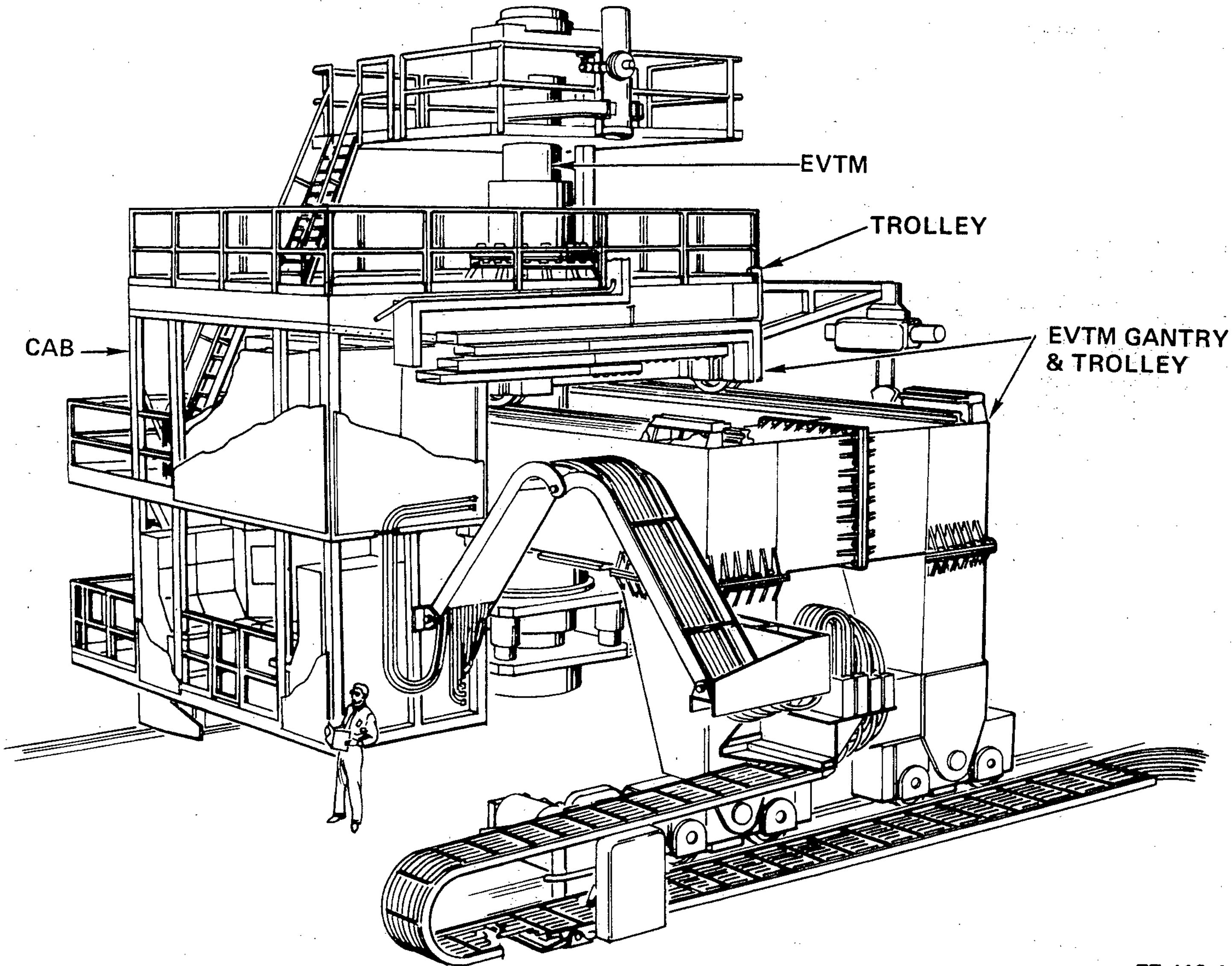


Figure 9.1-12A. EVST Sodium and Nozzle Elevations



77-J19-11-1 A

FIGURE 9.1-13. EX-VESSEL TRANSFER MACHINE ON GANTRY

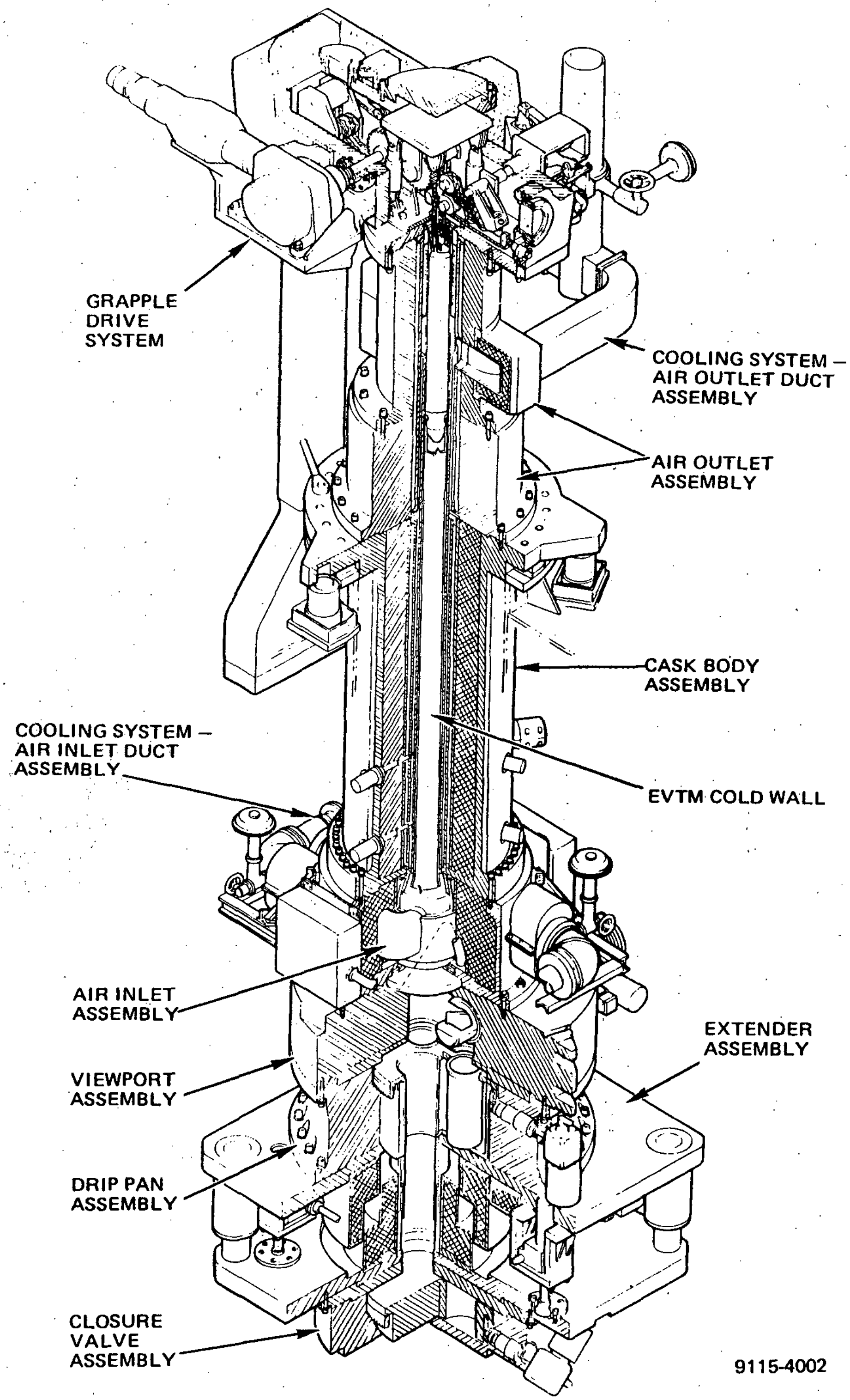


FIGURE 9.1-14. EX-VESSEL TRANSFER MACHINE (EVTM)

Figure 9.1-14a and 14b

44 Deleted

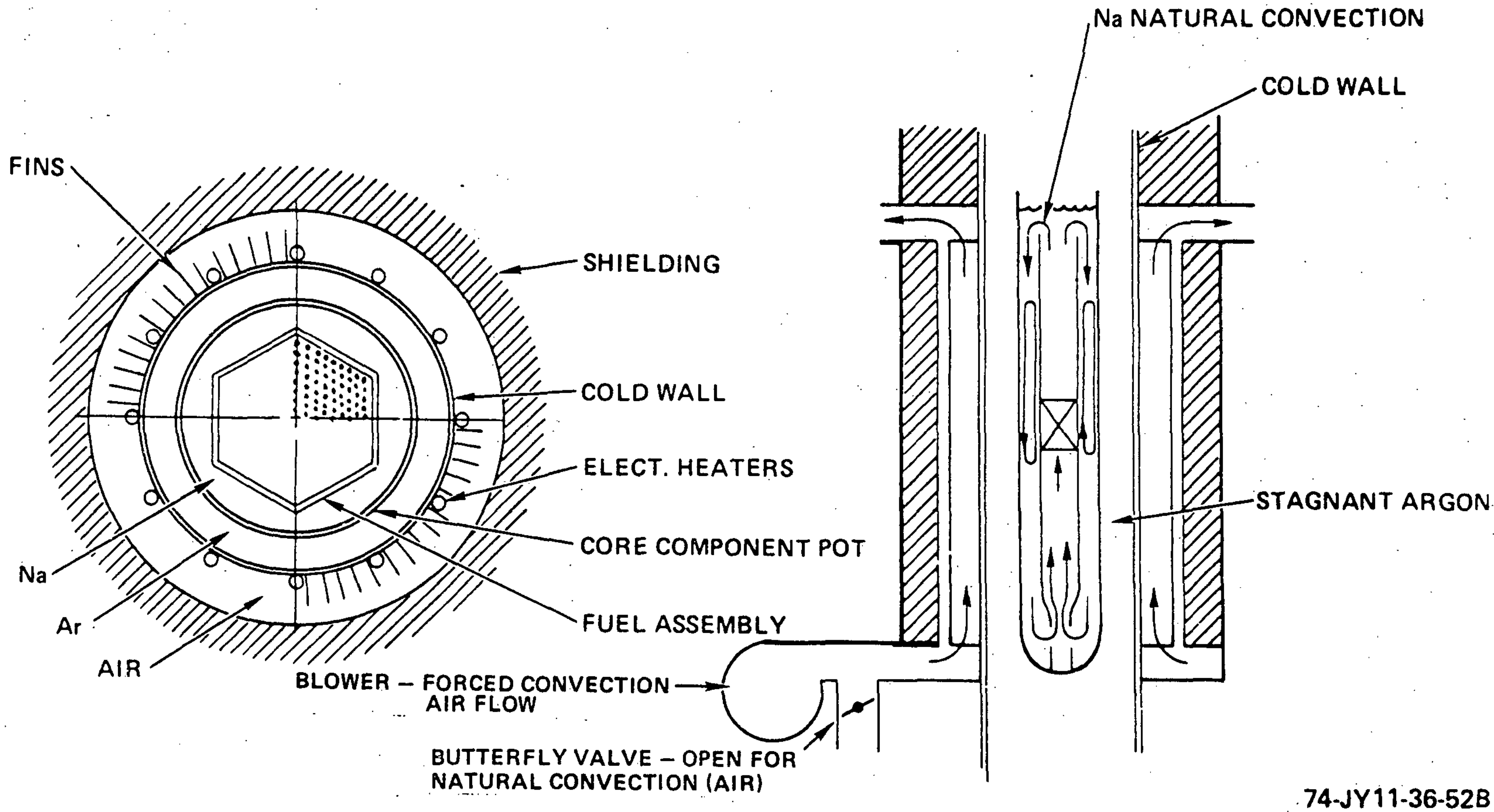


FIGURE 9.1-15. EVTM COLD WALL COOLING CONCEPT

IN-VESSEL TRANSFER MACHINE

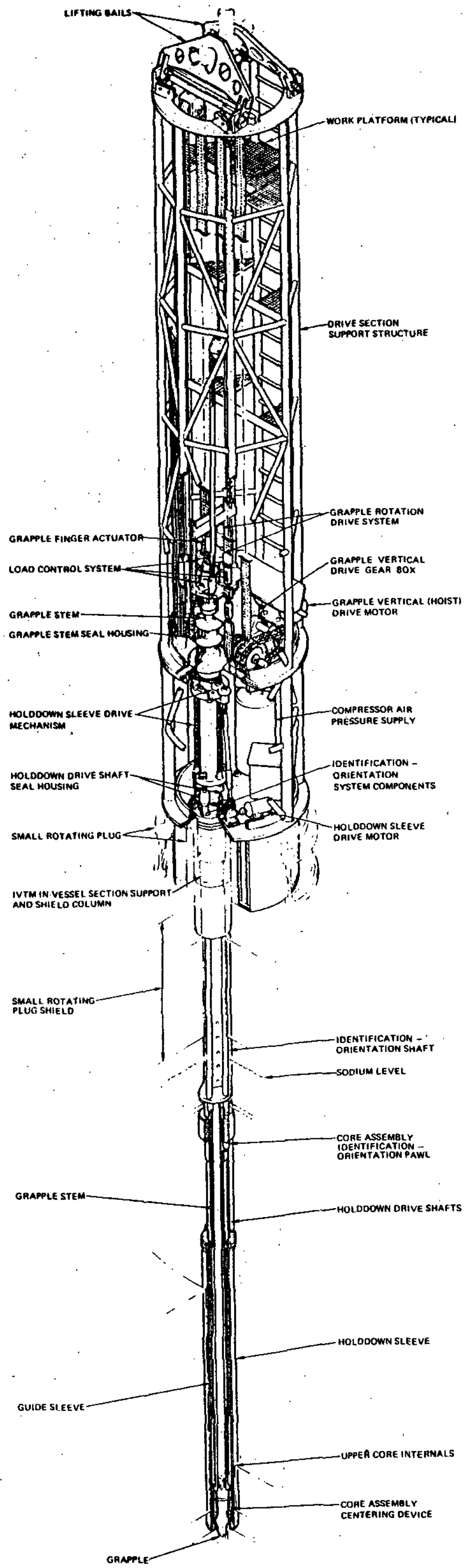


Figure 9.1-16 In-Vessel Transfer Machine (IVTM)

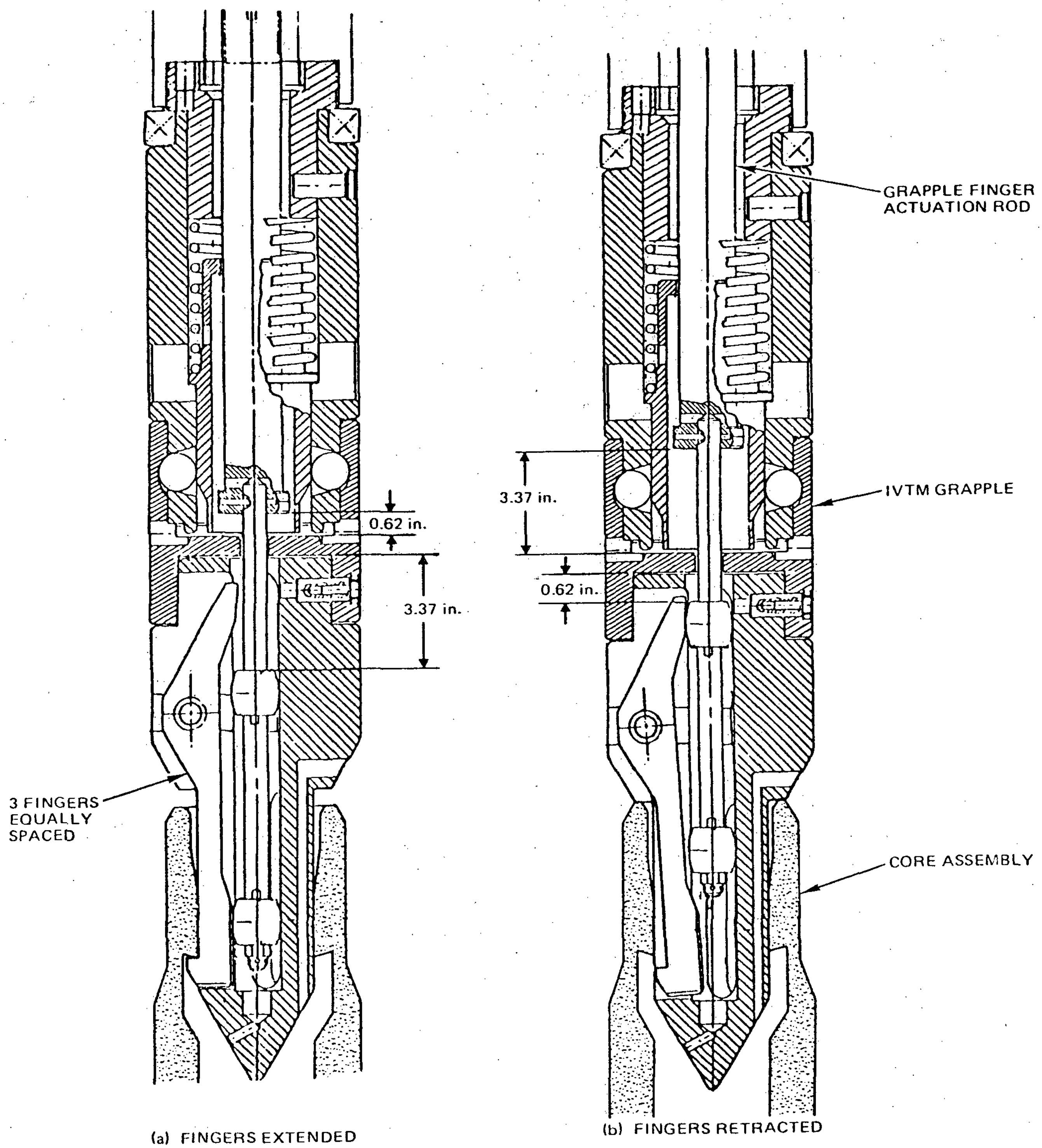
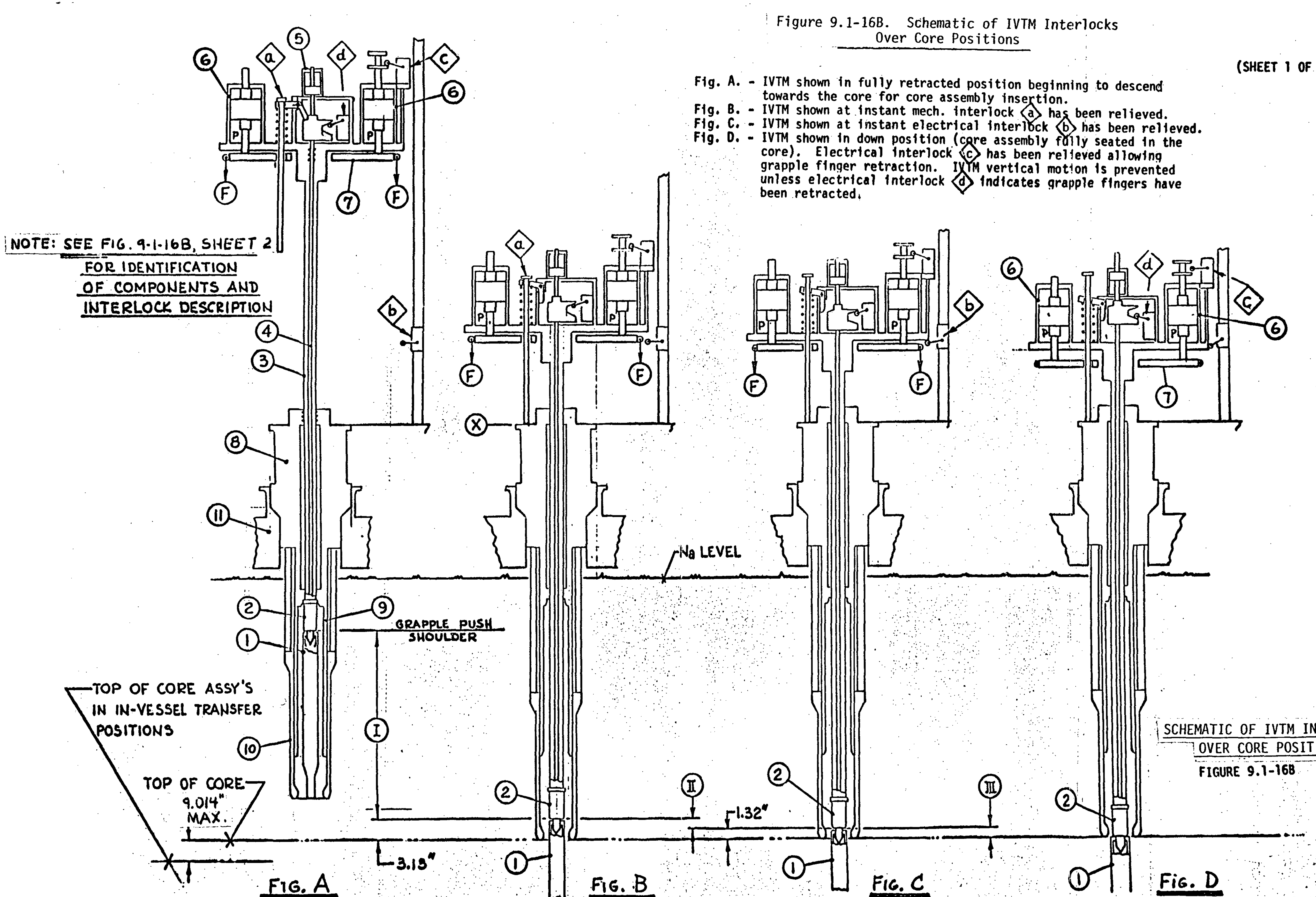


FIGURE 9.1-16A. IVTM Grapple in Core Assembly Socket

Figure 9.1-16B. Schematic of IVTM Interlocks Over Core Positions

(SHEET 1 OF 2)

Fig. A. - IVTM shown in fully retracted position beginning to descend towards the core for core assembly insertion.
Fig. B. - IVTM shown at instant mech. interlock ⟨a⟩ has been relieved.
Fig. C. - IVTM shown at instant electrical interlock ⟨b⟩ has been relieved.
Fig. D. - IVTM shown in down position (core assembly fully seated in the core). Electrical interlock ⟨c⟩ has been relieved allowing grapple finger retraction. IVTM vertical motion is prevented unless electrical interlock ⟨d⟩ indicates grapple fingers have been retracted.



SCHEMATIC OF IVTM INTERLOCKS OVER CORE POSITIONS

FIGURE 9.1-16B

SCHEMATIC OF IVTM INTERLOCKS OVER CORE POSITIONS

DESCRIPTION OF COMPONENTS

(1) Core Assembly
(2) Grapple-IVTM
(3) IVTM Grapple Stem
(4) Grapple Finger Actuation Rod
(5) Grapple Finger Actuator
(6) Load Control System-Pneumatic
(7) Vertical Motion Drive Plate
(8) IVTM Shield Column
(9) IVTM Guide Sleeve
(10) IVTM Holddown Sleeve
(11) Reactor Small Rotating Plug

(F) Vertical Force applied by the Hoist System

GRAPPLE FINGER ACTUATION INTERLOCK DESCRIPTION

a Mechanical interlock consisting of: a weighted rod, bellcrank and bellcrank retraction spring. Interlock prevents Gr. finger retraction in Region (I) see Figure.

b Redundant switches prevent grapple finger actuation in regions (I) and (II) see Figure.

c Redundant switches prevent grapple finger actuation until core assembly has been fully seated within region (III) . During core assembly insertion or withdrawal the core drag forces may exceed the load setting of the load control system (6) . The differential motion of drive plate (7) relieves interlock c and automatically stops the hoist drive system. If grapple motion is terminated within region (III) the grapple fingers can be actuated. It motion stops within region (II) , interlock b prevents finger actuation. If motion stops in region (I) , interlock b prevents finger actuation and interlock a prevents grapple finger retraction.

OPERATIONAL DESCRIPTION

CORE ASSEMBLY INSERTION:

The IVTM is driven downward from full up position, Figure A. After travelling a distance (I) the weighted rod of the mechanical interlock stops moving when it reaches level (X) and the bellcrank spring rotates the bellcrank relieving interlock a , see Figure B. After traveling an additional distance (II) the switches are tripped relieving interlock b , see Figure C. When the core assembly is bottomed in its receptacle within distance (III) the grapple, stem and load control cylinders stop moving, but the drive plate attached to the load control cylinder pistons, continues downward until the actuator on the piston rod moves away from switch c . This action automatically stops down drive motion. With all interlocks satisfied the grapple fingers are retracted. The grapple finger position switch d indicates that grapple fingers have retracted allowing reversal of pressure in the load control cylinders to activate the up drive system, see Figure D.

CORE ASSEMBLY WITHDRAWAL

The operational sequence for core assembly withdrawal is essentially the reverse of the insertion description given above. The basic difference is that the pressure in the load control cylinders is reversed displacing the grapple system and core assembly upwards from drive plate (7) about 1.5 inches. This motion activates switches c that allow the hoist system to drive upwards. On the way up the interlock switch b , and shortly after the mechanical interlock a , is activated. All the interlocks, that prevent grapple finger retraction, remain active until the interlocks are relieved on the way down as described in "core assembly insertion."

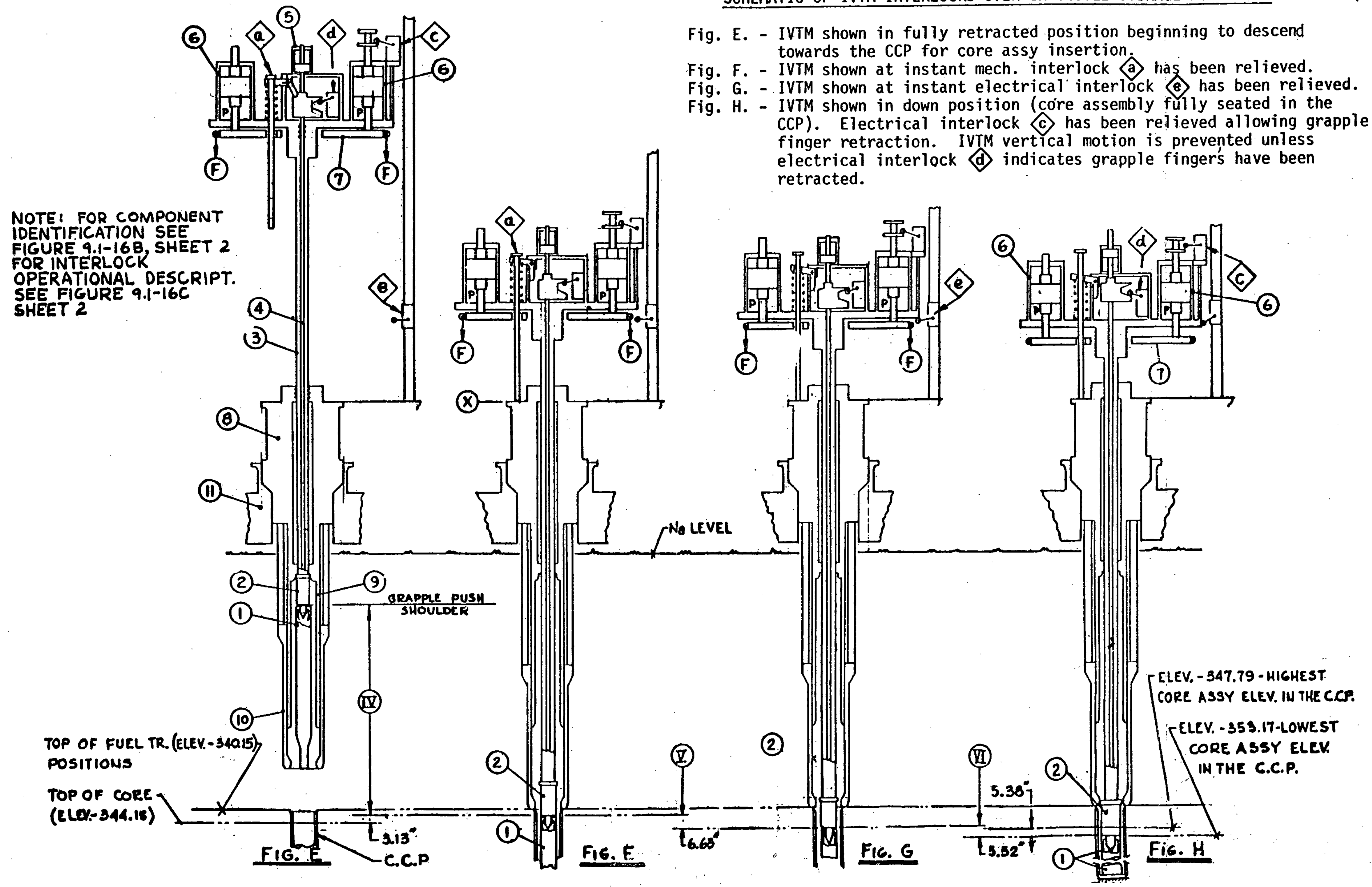
PARTIAL GRAPPLING INTERLOCKS

Redundant switches d automatically preclude hoist vertical motion, unless the switches are tripped either in the finger extended position, or retracted position.

NOTE: Letters in diamonds are also correlated to those in Figure 15.5.2.1.1-1.

FIGURE 9.1-16B - SCHEMATIC OF IVTM INTERLOCKS OVER CORE POSITIONS
(Sheet 2 of 2)

Fig. E. - IVTM shown in fully retracted position beginning to descend towards the CCP for core assy insertion.
Fig. F. - IVTM shown at instant mech. interlock ⟨a⟩ has been relieved.
Fig. G. - IVTM shown at instant electrical interlock ⟨e⟩ has been relieved.
Fig. H. - IVTM shown in down position (core assembly fully seated in the CCP). Electrical interlock ⟨c⟩ has been relieved allowing grapple finger retraction. IVTM vertical motion is prevented unless electrical interlock ⟨d⟩ indicates grapple fingers have been retracted.



SCHEMATIC OF IVTM INTERLOCKS OVER IN-VESSEL STORAGE POSITIONS

FIGURE 9.1-16C

SCHEMATIC OF IVTM INTERLOCKS

OVER INVESSEL STORAGE POSITIONS

OPERATIONAL DESCRIPTION

The IRP and SRP at 0° activates redundant interlock switches ⟨e⟩ and deactivates interlock switches ⟨b⟩.

CORE ASSEMBLY INSERTION:

The IVTM is driven downward from full up position, Figure E. After travelling a distance (IV) the weighted rod of the mechanical interlock stops moving when it reaches level (X) and the bellcrank spring rotates the bellcrank relieving interlock ⟨a⟩, see Figure F. After travelling an additional distance (V) the switches are tripped relieving interlock ⟨e⟩, see Figure G. When the core assembly is bottomed in the CCP within distance (VI) the grapple, stem and load control cylinders stop moving, but the drive plate attached to the load control cylinder pistons, continues downward until the actuator on the piston rod moves away from switch ⟨c⟩. This action automatically stops down drive motion. With all interlocks satisfied the grapple fingers are retracted. The grapple finger position switch ⟨d⟩ indicates that grapple fingers have retracted allowing reversal of pressure in the load control cylinders to activate the up drive system, see Figure H.

CORE ASSEMBLY WITHDRAWAL

The operational sequence for core assembly withdrawal is essentially the reverse of the insertion description given above. The basic difference is that the pressure in the load control cylinders is reversed displacing the grapple system and core assembly upwards from drive plate (7) about 1.5 inches. This motion activates switches ⟨c⟩ that allow the hoist system to drive upwards. On the way up the interlock switch ⟨e⟩ and shortly after the mechanical interlock ⟨a⟩ is activated. All the interlocks, that prevens grapple finger retraction, remain active until the interlocks are relieved on the way down as described in "core assembly insertion."

NOTE: Letters in diamonds are also correlated to those in Figure 15.5.2.1.1-1.

FIGURE 9.1-16C - SCHEMATIC OF IVTM INTERLOCKS
(Sheet 2 of 2) OVER IN-VESSEL STORAGE POSITIONS

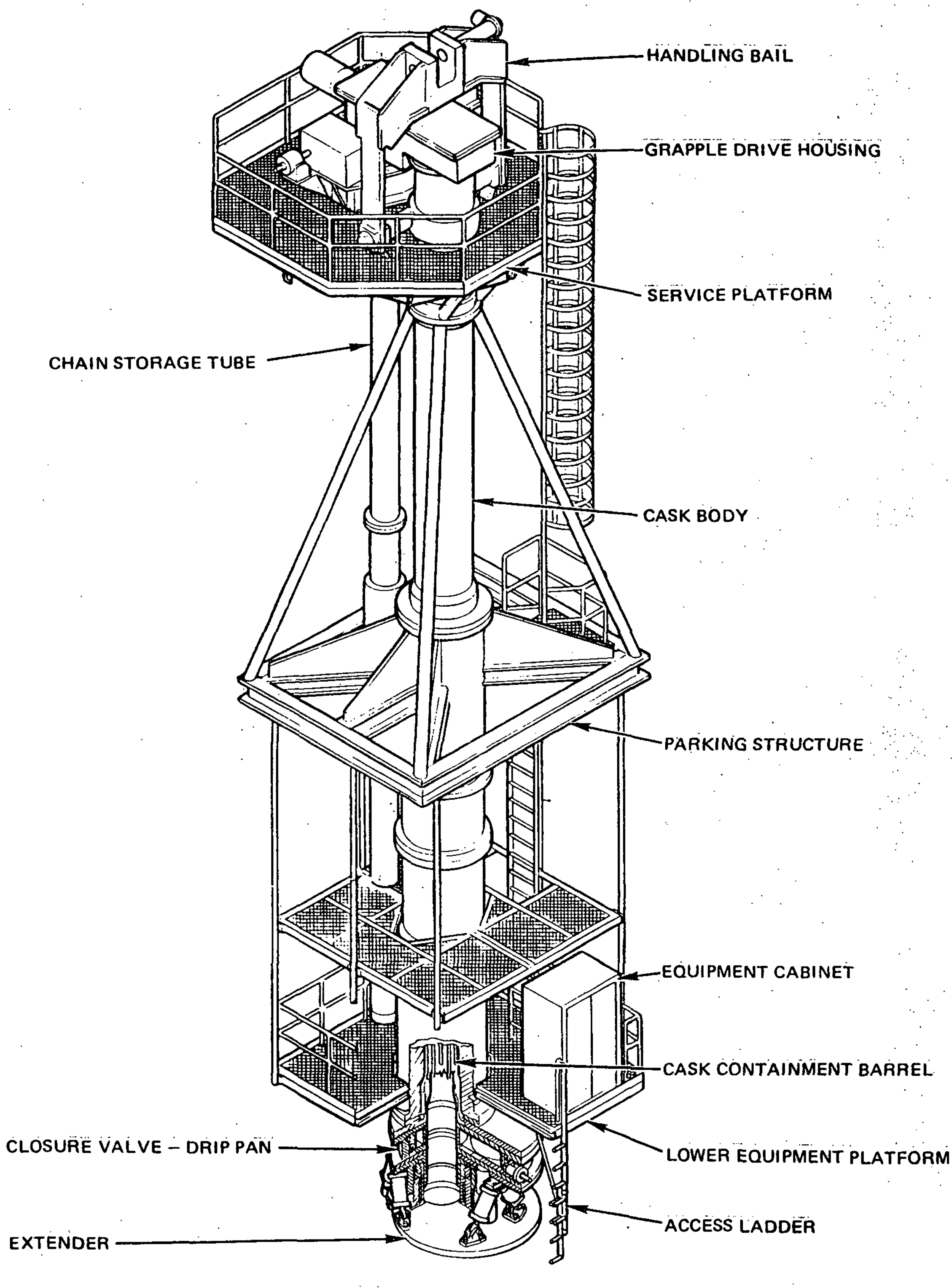


Figure 9.1-17 Auxiliary Handling Machine

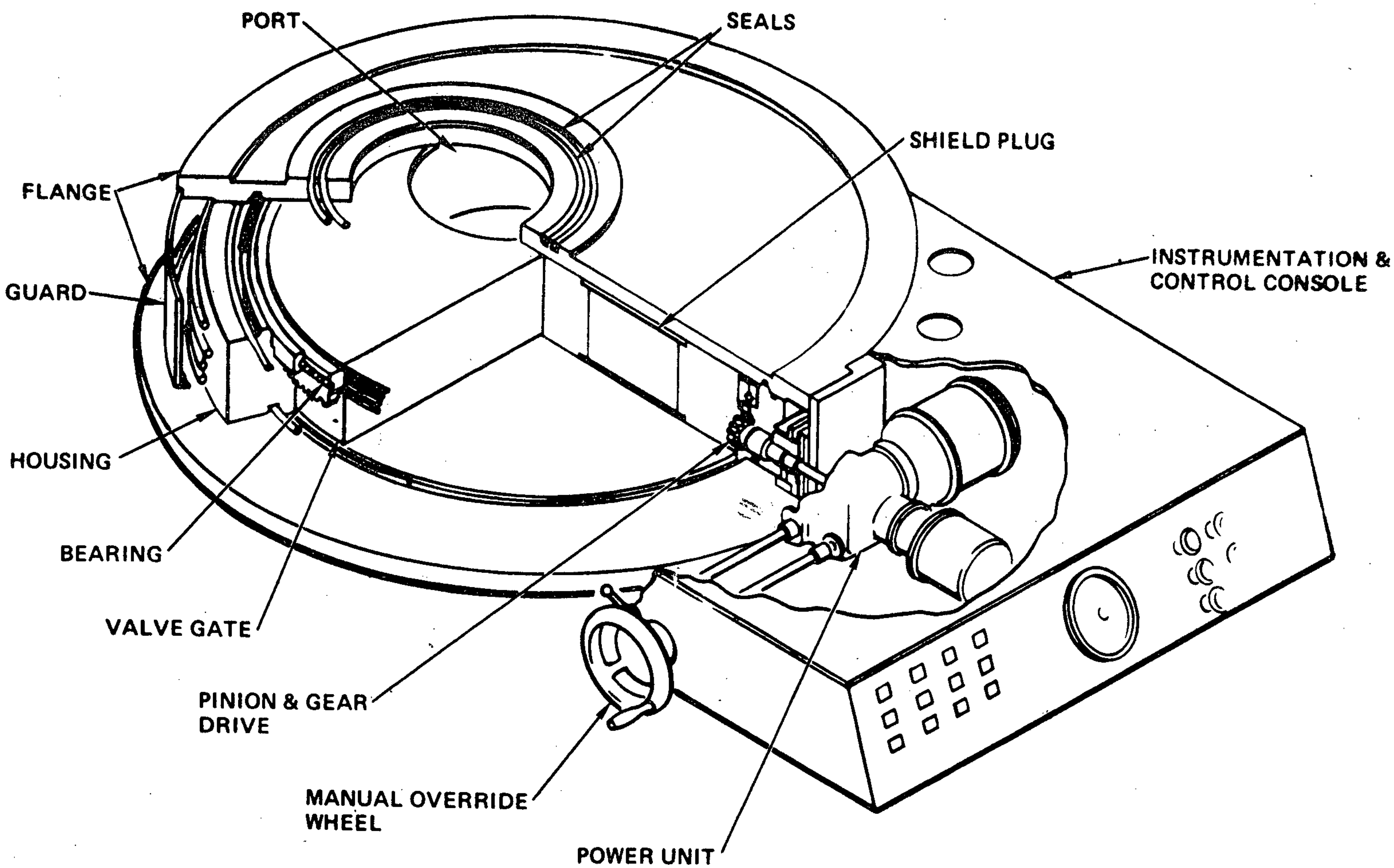
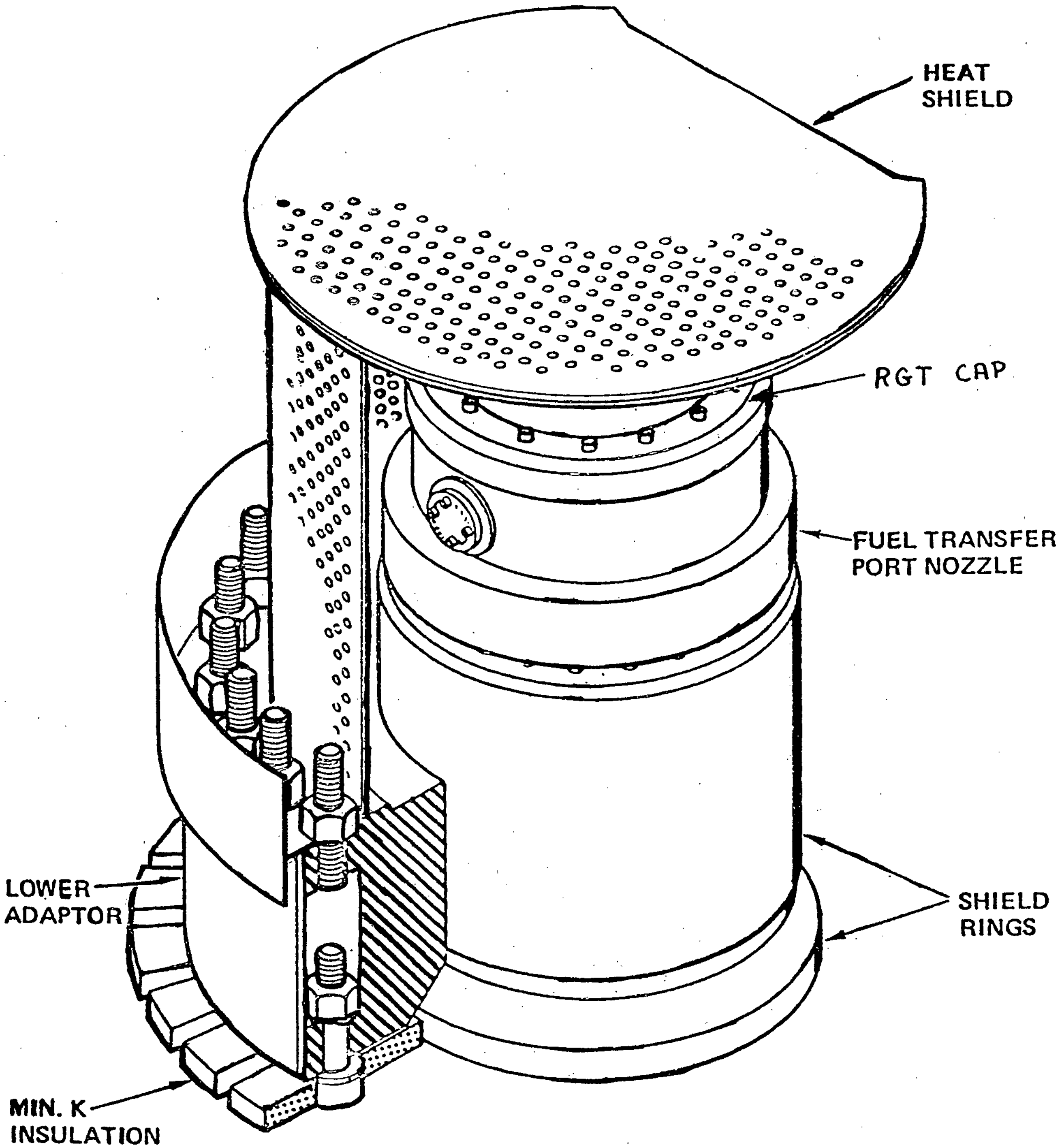


Figure 9.1-18 Floor Valve

Figure 9.1-18a RFTP Lower Adapter (Reactor Operating Mode)



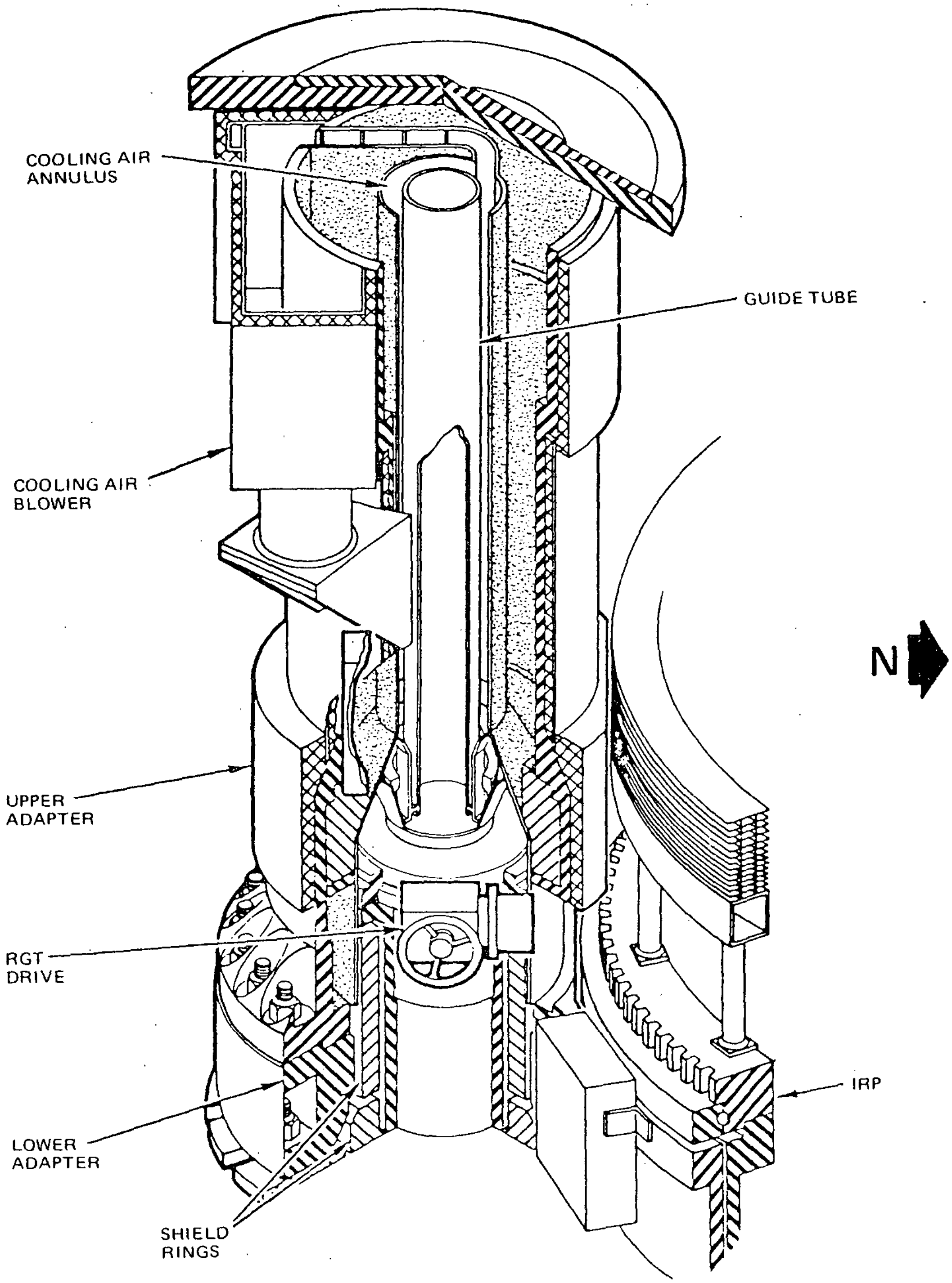


FIGURE 9.1-19 RFTP Adapter Assembly

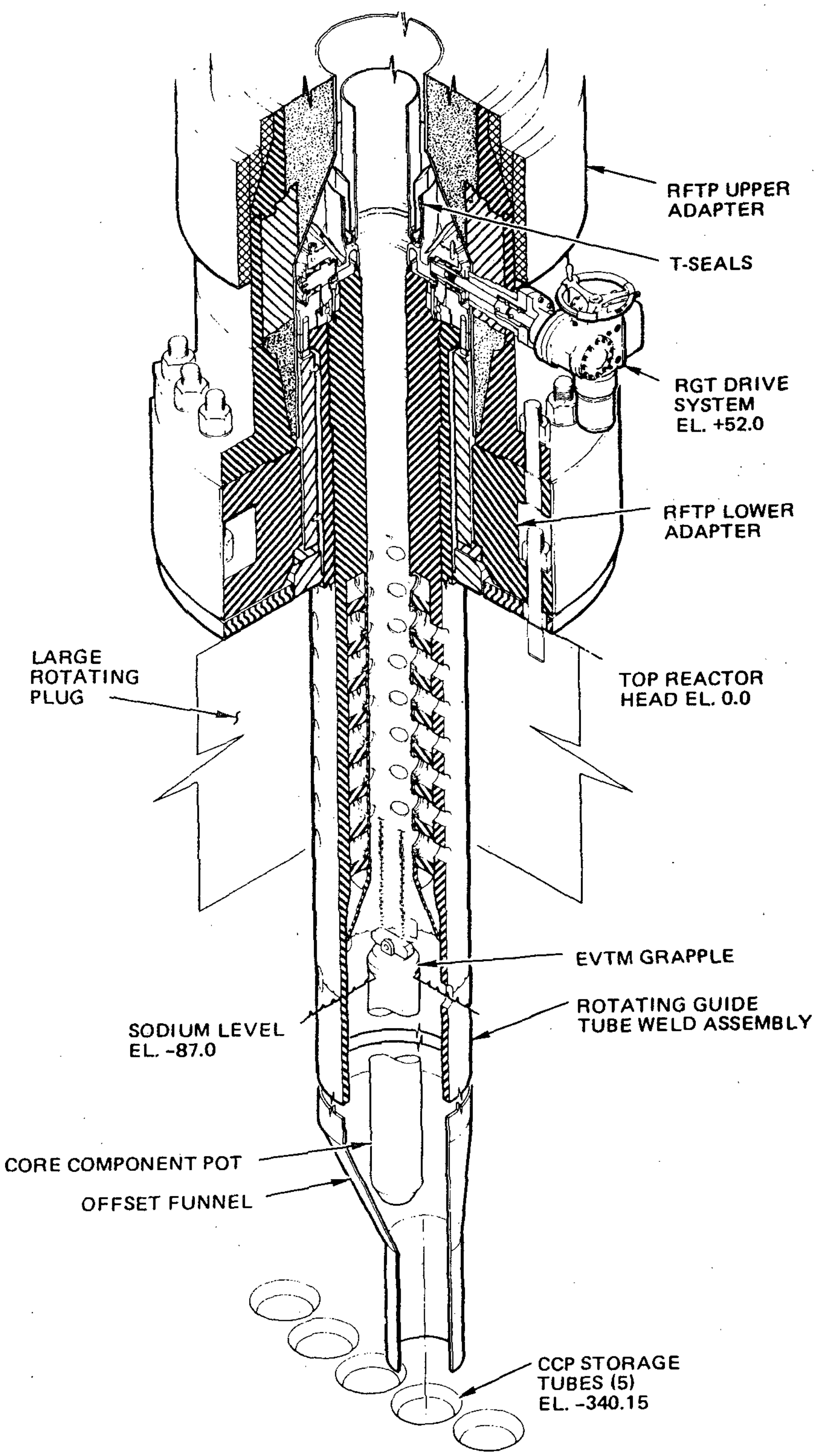
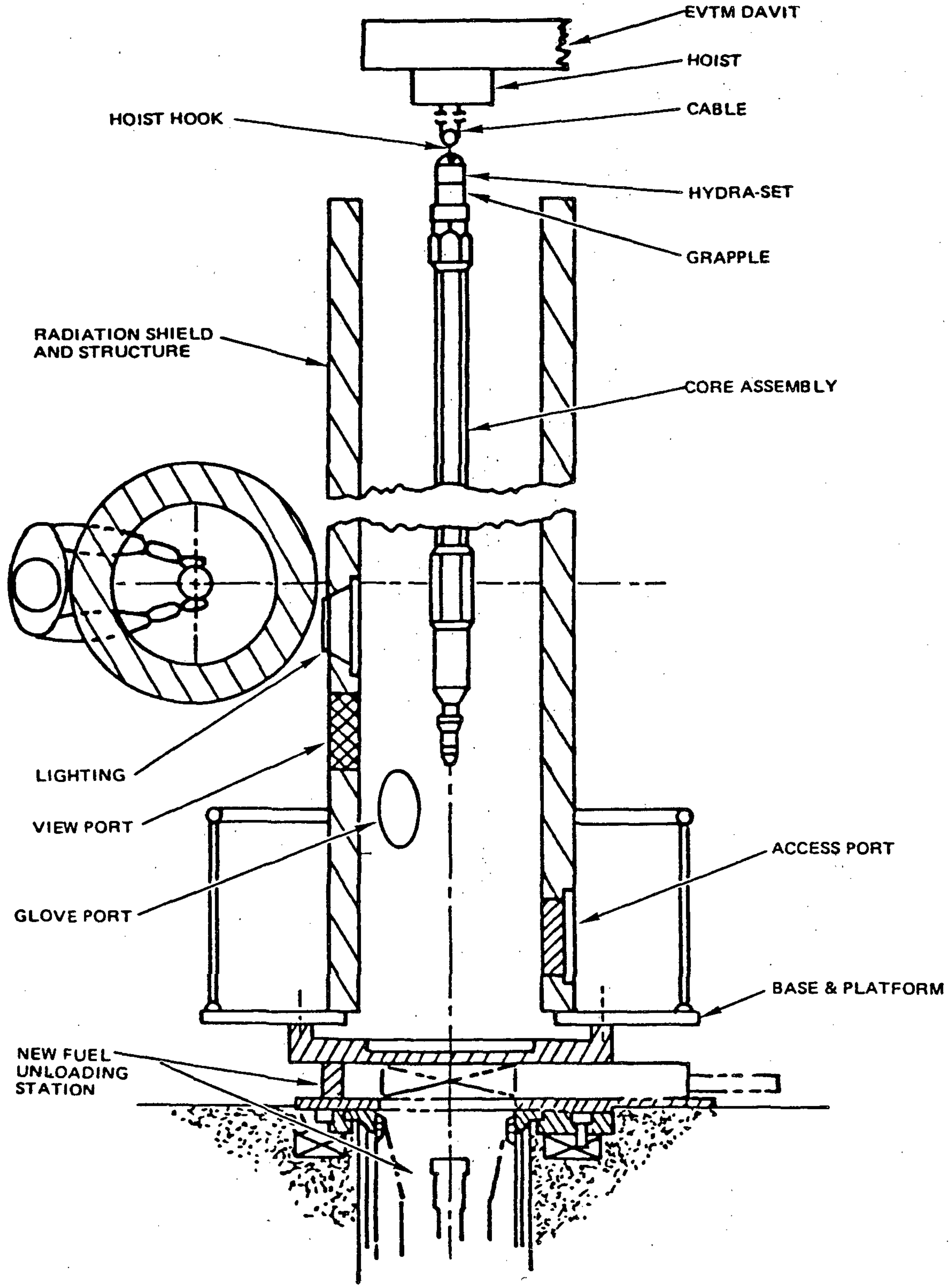


FIGURE 9.1-20 Rotating Guide Tube

Figure 9.1-21 New Core Assembly Inspection Equipment



9.2 NUCLEAR ISLAND GENERAL PURPOSE MAINTENANCE SYSTEM

The Nuclear Island General Purpose Maintenance Equipment System provides general purpose facilities, tools, and fixtures in support of maintenance operations unique to the Nuclear Island. Equipment provided by the Nuclear Island General Purpose Maintenance Equipment System is used in conjunction with equipment as necessary from the Balance of Plant General Purpose Maintenance Equipment System to provide complete maintenance support for the Nuclear Island.

9.2.1 Design Basis

The Nuclear Island General Purpose Maintenance System provides general purpose equipment and facilities to support maintenance activities for visual inspection, handling, cleaning, decontaminating, disassembly, and repair of sodium-wetted and/or radioactive components. The equipment is designed to keep plant personnel radiation exposures as low as reasonably achievable as explained in Appendix 12A. The equipment is designed to support repairs in place wherever possible. The design is also based on maintenance by hands-on, semi-remote, and remote techniques, in that preferential order.

9.2.2 System Description

The Nuclear Island General Purpose Maintenance System consists of equipment, facilities, and subsystems as described below.

a. Facilities

1. The Decontamination Facility, located in the radioactive waste area of the RSB, is a set of negative-pressure, low-leakage rooms in which trace amounts of sodium and low-level radioactive contamination can be removed from components. From this facility, gaseous and liquid effluents and solid wastes are collected and directed to the appropriate radwaste systems for controlled disposal. (See Section 11.2 and 11.3)

2. The Regulated Shop is an area of the Maintenance Shop and Warehouse Building in which machine tools are provided for use on low-level radioactive parts and for which controlled access may be established. From this facility gaseous and liquid effluents and solid wastes are collected and directed to the appropriate radwaste systems for controlled disposal.

b. Sodium Removal and Decontamination Equipment

1. The Primary Sodium Removal and Decontamination System, located in the RCB (See Fig. 9.2-1), consists of the Large Component Cleaning Vessel (LCCV), and associated process equipment. Large components requiring sodium removal may be placed into the LCCV from the RCB operating floor. The system removes surface sodium from the components using the water vapor-nitrogen process, and removes subsurface radioactive contamination using acid etching. The system provides for control and containment of sodium-water reactions and reaction products both during normal operations and in the event of accidental contact between liquid water and bulk sodium during cleaning. Liquid and gas effluents from these processes, which contain radioactive materials from the primary sodium and subsurface contamination, are directed to the appropriate radwaste systems for controlled disposal (see Sections 11.2 and 11.3).

2. The Small Component Autoclave, located in the Decontamination Facility in the RSB, consists of a vessel into which small components requiring sodium removal may be placed and cleaned using the associated water-vapor-nitrogen process equipment. It is similar to the Primary Sodium Removal and Decontamination System although it is smaller and does not provide the capability for removal of subsurface contamination. Liquid and gas effluents from this process, which may contain radioactive materials from the primary sodium, are directed to the appropriate radwaste systems for controlled disposal.

3. Space is provided in the SGB for the possible future addition of sodium removal equipment for the Intermediate Heat Transport System (IHTS) components. This equipment conceptually would not contain decontamination capability as the only source of radioactivity in the IHTS would be from tritium.

c. Handling equipment

1. Handling containers are provided to prevent the spread of radioactivity during removal and transport of radioactive components and to contain solidified surface sodium films within an inert atmosphere during removal and transport of sodium wetted components. The containers include plastic film bags within which components may be transported or transferred between controlled and inerted atmospheres and portable enclosures which provide controlled air atmosphere working spaces. General purpose shielded casks are planned to be provided to reduce direct radiation when handling radioactive or radioactive sodium-wetted components.

2. Handling equipment is provided to move heavy or bulky components. The major equipment is the Large Component Transporter, which provides a platform with tiedown facilities and is moved along the EVTM rails to transport large components between RCB and RSB. The Large Component Transporter (LCT) is also used as a working platform over the reactor for service of the control rod drive mechanism. The Large Component Transporter is fitted with permanent passive seismic restraints (similar to the EVTM) to prevent the LCT from leaving the tracks during a seismic event and also with manually engaged permanent wheel chocks to prevent motion after placement at the desired position along the tracks. Another piece of handling equipment is the Large Lift Fixture, a load-spreading bar for attachment of large equipment to building cranes.

d. Maintenance and Inspection Equipment

1. Maintenance stands are provided to support the primary and intermediate sodium pumps and pump motors during performance and maintenance away from their normal locations.

2. Portable equipment is provided for hands-on and semi-remote cutting, rewelding, and weld inspection of sodium piping.

3. Remote viewing equipment is provided for in-service inspection of components in inerted or radioactive cells.

9.2.3 Safety Evaluation

The Nuclear Island General Purpose Maintenance Equipment System has no nuclear safety function in the operation of the plant. Radioactive fluids or solids resulting from use of the equipment are directed to the waste processing or ventilation systems for cleanup and/or stabilization before discharge to the environment. See Section 11.2 and 11.3 for a discussion of these systems. No direct discharges to the environment are made from the Nuclear Island General Purpose Maintenance Equipment System.

A potential for a significant sodium-water reaction exists in the LCCV. It has been shown that the maximum postulated reaction and its products will be retained within the system and cell and will not perturb the reactor systems during any operating mode. The justification for the LCCV design pressure is discussed in Section 15.7.3.7.

9.2.4 Tests and Inspections

Tests and inspections of Nuclear Island General Purpose Maintenance Equipment are those involved with assuring proper condition and serviceability and, where applicable, calibration prior to use. The equipment itself has no nuclear safety classification . Handling equipment to be used for critical lifts (Class A & B) will be load tested per RDT-F8-6T.

9.2.5 Instrumentation Applications

Instrumentation and process controls are required only for operation of the sodium removal and decontamination equipment. (See Section 15.7.3.7 for a discussion of their potential involvement in the postulated sodium-water reaction in the LCCV.)

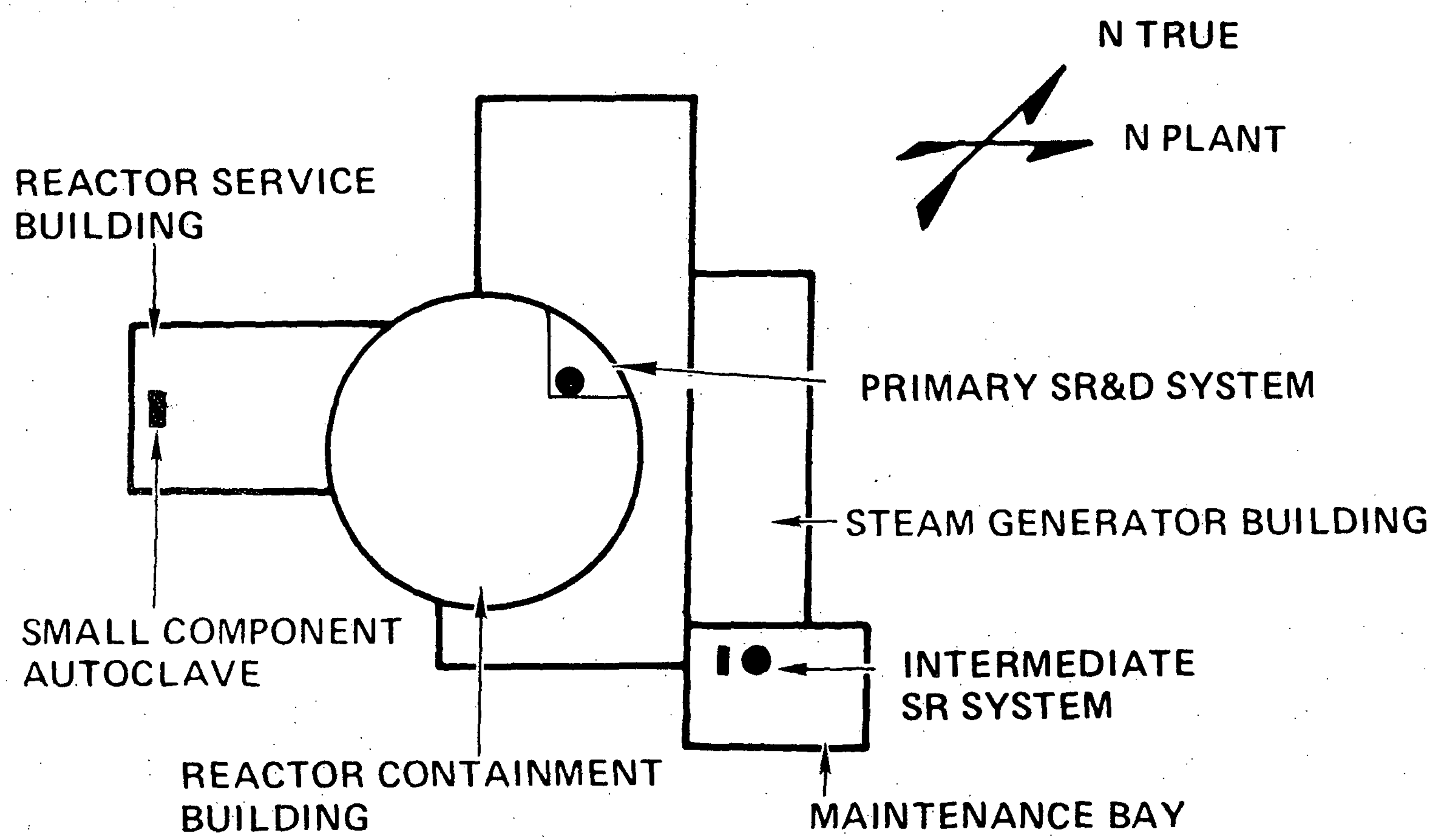


Figure 9.2-1. Location of Sodium Removal & Decontamination Systems in NSSS Buildings

9.3 AUXILIARY LIQUID METAL SYSTEM

The Auxiliary Liquid Metal System is made up of five subsystems. The events considered as design basis accidents are described in Chapter 15 and other comments on safety of individual subsystems are contained in the discussion of those subsystems.

The subsystems comprising the Auxiliary Liquid Metal System are:

	Figure No.
Sodium and NaK Receiving	9.3-1
Primary Sodium Storage and Processing	9.3-2
EVS Processing	9.3-3
Primary Cold Trap NaK Cooling	9.3-4
Intermediate Sodium Processing	9.3-5

All components of the Auxiliary Liquid Metal System are designed for sodium service. Except as noted, the pressure boundary material for all components is Type 304 austenitic stainless steel, conforming to ASME Code requirements, and with the supplemental requirements of RDT E 15-2T imposed for components which are either designed as ASME Section III, Class 1 components or are designed for high temperature (800°F) service. The only exceptions to the use of 304 stainless are (a) the sodium overflow line, between the reactor and overflow vessel, which is type 316 stainless, and (b) the primary sodium storage vessels and the EVST NaK storage vessel which are carbon steel. These items are in the Primary Sodium Storage and Processing Subsystem and the EVS Processing Subsystem.

The active pumps and valves are listed in Table 9.3-6. The pumps and valves must be operable during and after design basis events, such as a Safe Shutdown Earthquake (SSE).

The design, fabrication and testing of supports for piping and associated components are given special attention to assure operability and structural integrity will be maintained during the CRBRP lifetime. The Class 1, 2 and 3 supports all have common basic features, controls and requirements like those defined in Section 3.9.2.6.

After completion of fabrication of each of the carbon steel vessels, the interior is abrasively cleaned. This is followed by brushing or grinding to remove any imbedded material. Additionally, all surfaces are cleaned with a suitable solvent to remove any remaining particulate material. The tank is then inspected for cleanliness. Finally, the interior of the tank is purged with dry inert gas (argon or nitrogen). The vessel is maintained under inert gas at a positive pressure during the period prior to installation.

Austenitic stainless steel has been selected as the principal material of construction for sodium components because of its extensive past use, and successful experience, in liquid metal service. Carbon steel is selected for the primary storage vessels and the EVST NaK for economic reasons and is a suitable material due to the low temperature operating conditions and the fact that the vessels serve a standby purpose only and are not part of the active, flowing, sodium or NaK systems during normal plant operations.

9.3.1 Sodium and NaK Receiving System

9.3.1.1 Design Basis

This system provides the capability to receive and melt fresh, solidified sodium, delivered to the site in tank cars or drums, and transfer the sodium to primary and intermediate storage vessels. System capacity is based on meltout and drain of an 80,000-lb capacity tank car in 16 hours.

The system receives and transfers to storage all NaK used in the plant.

NaK is a eutectic mixture of sodium and potassium. CRBRP plans to use a eutectic mixture which contains 78% (by weight) of potassium and 22% (by weight) of sodium. The solidus temperature of this mixture is $\sim 9^{\circ}F$.

All fresh sodium and NaK will be filtered prior to storage. Components used for sodium transfer will not be used for NaK transfer. The system also provides the capability to remove sodium and NaK from plant systems for off-site disposal.

The design temperatures for the sodium and NaK components are $450^{\circ}F$ and $150^{\circ}F$, respectively. The design pressure for both sodium and NaK components of the receiving station is full vacuum to 20 psig.

9.3.1.2 Design Description

This system consists of a tank car oil heating station for meltout of sodium tank cars, a clam shell heater for melting drums of sodium, transfer piping and valves, and filters for cleanup of fresh sodium or NaK. The piping and filters for the NaK are on a portable rig, and independent of the sodium system. Both systems are shown on Figure 9.3-1. Transfer of sodium and NaK to system storage vessels is by gravity flow. All liquid metal piping, valves, and filters of the sodium and NaK Receiving System are constructed from Type 304 stainless steel.

9.3.1.3 Design Evaluation

The Sodium and NaK Receiving System components are designed to accepted industrial and nuclear standards to insure structural integrity and operational reliability. The components, applicable design code and class, plus their seismic category are listed in Table 9.3-1.

Precautions will be taken during liquid metal loading conditions to limit effects to plant components prior to plant operation. The system will be monitored for external leaks of liquid metal by leak detection devices.

This system for handling the incoming fresh sodium and NaK does not present any radiological hazards, nor is it involved in any way with reactor safety. Portable sodium carbonate fire extinguishers and personnel protective equipment is provided by the Sodium Fire Protection System for protection against potential sodium or NaK fires that can occur during loading and unloading operations. The cells containing system storage vessels are equipped with permanently installed fire protection equipment as described in Section 9.13.2. Prior to receiving sodium on site, appropriate or necessary emergency planning with local officials will be taken to consider an outside fire during loading.

9.3.1.4 Tests and Inspection

Prior to use, leak checks will be made and instrumentation and preheat capability will be checked according to specific procedures. The system filters will be tested to assure that no blockage exists.

9.3.1.5 Instrumentation Requirements

Instrumentation and controls (I&C) are provided for operation, performance evaluation and diagnosis of the Sodium and NaK Receiving System. These functions are required for off-normal, as well as for the full range of normal operation. Details of the I&C for the Na and NaK receiving system are shown in Figure 9.3-1.

Temperatures and flow rates are measured to monitor system status during operation. Leak detection sensors are strategically located to alert the operator of a break in the system in order that correction action may be taken. Table 9.3-4 indicates planned location of leak detection sensors.

No automatic control instrumentation is required for this subsystem as all operations are manual. This subsystem is non-nuclear and operated at peak temperatures of 400°F. Therefore, commercial grade instrumentation is adequate to monitor subsystem performance. After initial transfer of sodium and NaK into the plant, the system will be inactive at ambient temperature except during actual transfer operations.

9.3.2 Primary Na Storage and Processing

9.3.2.1 Design Basis

This system provides primary sodium purification (cold trapping), provides storage for the sodium used in the reactor vessel, the PHTS loops, and the EVST, mitigates the change in reactor vessel sodium level and accommodates thermal expansion and contraction of primary sodium. This system, working in conjunction with the EVS Sodium Processing System, also provides a means of removing reactor decay heat in the event of loss of all the steam generators.

Specific system design basis are as follows:

a. Primary sodium purification - limit the oxygen content to 2.0 gpm and the hydrogen content to 0.2 gpm, and maintain the tritium content within limits which will satisfy plant radiological release criteria.

b. Primary sodium storage - provide on-site storage capacity sufficient to permit anticipated maintenance on plant systems. Total storage capacity is provided to permit complete drainage of the reactor vessel and loop piping to the first high point. This capacity will accommodate the above, the 3 PHTS Loops or the EVST, but not simultaneously.

c. Reactor level control - provide the capability during all plant operating conditions to maintain the reactor vessel sodium level below the reactor head reflector plate.

d. Primary sodium expansion - accommodate thermal expansion of primary sodium from 400°F to PHTS structural design temperatures of 1015°F hot leg and 765°F cold leg.

e. Reactor decay heat removal - system sizing is based on limiting the average bulk primary sodium temperature to approximately 1140°F when the DHRS is initiated one-half hour after reactor shutdown. With DHRS initiated twenty-four hours after reactor shutdown, system size will maintain the average bulk primary sodium temperature below 900°F.

9.3.2.2 Design Description

The Primary Sodium Storage and Processing System consists of the following components:

a. Primary Sodium Overflow Vessel

b. Primary Sodium Makeup Pumps (2)

c. In-Containment Primary Sodium Storage Vessel

d. Ex-Containment Primary Sodium Storage Vessels (2)

e. Primary Sodium Cold Traps (2)

f. Makeup Pump Drain Vessel

g. Interconnecting Piping and Valves

h. Primary Sodium Overflow Heat Exchanger

Refer to Figure 9.3-2 for the P&ID; system arrangement is shown on Figures 1.2-7 through 1.2-14, 1.2-18 and 1.2-19.

9.3.2.2.1 Overflow and Makeup Circuit

a. Circuit Operation During Normal Plant Operation - The overflow vessel, makeup pumps, cold traps, overflow heat exchanger, and associated piping comprise the overflow and makeup circuit. With the exception of the overflow heat exchanger, which is normally by-passed, this circuit operates continuously during reactor operation. This circuit provides the reactor coolant volume control discussed in Section 5.3.3.7. The circuit functions to maintain variations of the sodium level in the reactor vessel within acceptable limits, by accommodating volumetric changes in primary sodium due to sodium temperature variations and primary pump drawdown. During operation, sodium overflows from the reactor vessel by gravity to the primary sodium overflow vessel. The primary sodium makeup pumps continuously pump sodium from the overflow vessel, back to the reactor
46 vessel, at a constant total makeup rate of 150 gpm. The overflow rate varies during primary system transients; during steadystate operation, the
46 rate matches to makeup rate of 150 gpm. The overflow vessel functions, in effect, as an expansion tank for the reactor and PHTS systems, with the overflow vessel level varying, depending on primary system temperature and primary pump flow rate. The 150 gpm makeup rate provides margin to permit
46 maintaining full, rated flow through both of the primary cold traps. The overflow line is sized to permit gravity flow during all anticipated normal and off-normal events.

The overflow vessel is sized to accommodate primary sodium volume expansion from 400°F isothermal to PHTS system design temperatures at 1015°F hot leg and 765°F cold leg, plus the primary pump drawdown volume at 110% rated flow. In addition, the vessel provides sufficient volume to accommodate the expansion of
46 primary sodium during DHRS events.

The overflow and makeup circuit operates continuously during plant operation at
46 a total makeup flow rate of 150 gpm, provided by both of the electromagnetic makeup pumps operated at reduced voltage. The makeup pumps are sized to deliver approximately 280 gpm each (sized by reactor decay heat removal requirement; see Section 9.3.2.2.1.b) consequently either pump alone can provide the normal makeup rate. In the event of a pump failure, the inoperative pump is isolated. Plant operation continues with the remaining
46 pump providing 150 gpm flow. Each pump is located in a separate cell, so that

pump maintenance can be accomplished after shutdown and isolation of the failed pump from the system. The pump is drained, the cell atmosphere changed to air, and the necessary precautions taken with respect to radioactivity, to allow pump repair.

Two liquid-cooled sodium cold traps, arranged in parallel, are included in a bypass on the reactor makeup return line to provide sodium purification. Each trap is rated at 60 gpm at normal system operating temperature, and at 80 gpm and 600°F (reactor hot standby temperature). During normal plant operation, one trap is in use and the second is in standby. Although both traps can be operated, a single trap is sufficient to remove anticipated oxygen inleakage and maintain the oxygen content below 2 ppm.

The concentration of tritium in the primary sodium is also maintained at a low level by cold trap operation. Operation of one primary cold trap, combined with the tritium diffusion through the IHX to the intermediate system maintains the tritium content of the primary sodium at less than 3 Ci T/gm Na.

During a shutdown for fuel handling, both cold traps may be operated, if necessary, to provide a maximum cleanup flow of 160 gpm. The total capacity is designed to provide for removal of potential oxygen inleakage during fuel handling rapidly enough so that no additional plant downtime (over and above that required for fuel handling and normal startup) is required. Flow to the trap(s) is controlled by throttle valves in the outlet from each trap. Electromagnetic flow meters are provided to monitor flow through each pump and each cold trap.

System arrangement also permits independent cold trapping of sodium in the overflow vessel or in the in-containment primary sodium storage vessel during shutdown situations, when makeup to the reactor is not required. The makeup pumps, which can also take suction from these vessels, are used for these operations.

The primary cold trap lifetime is estimated to be 13 years. This trap will be top radioactive to permit hands-on-maintenance. However, steel shielding has been provided around this trap to expedite the replacement operations. The procedure for primary cold trap replacement is as follows. Partially drain the sodium and NaK while maintaining a temperature of 300-400°F, remove all heater power and allow the remaining sodium to freeze, cut and cap weld the sodium and NaK lines, cut the electrical leads, unbolt the supports, pull the trap by crane from the cell, and place it in storage provided in the plant.

b. <u>Circuit Operation During Reactor Decay Heat Removal</u> - The overflow and makeup circuit is designed to provide reactor decay heat removal in the event of loss of all the steam generators in the intermediate heat transfer system. Operation during this mode is referred to as the Direct Heat Removal Service (DHRS). Switchover to this mode of operation is accomplished remotely from the control room.

DHRS components are subjected, during plant service, to numerous reactor scrams. During a scram, primary sodium temperature decreases. The resultant contraction of coolant lowers the sodium level within the reactor, and the sodium overflow is interrupted. The makeup pumps continue to operate, transferring sodium in the overflow vessel back to the reactor vessel until the sodium overflow resumes at a lower temperature - approximately 600oF.

The overflow interruption has no effect on operation of components required for DHRS duty since all components continue to operate in a normal manner throughout the event. Prior to reactor scram, the overflow vessel is approximately one-half full, containing the coolant expansion, from 400oF, of the entire primary system. The makeup pumps continue to transfer sodium back to the reactor, and before the overflow vessel can be emptied, the reactor level is raised high enough to insure resumption of overflow. Consequently, although overflow is temporarily interrupted, flow from the overflow vessel back to the reactor vessel does not stop at any time, and all components continue to function normally.

Following the reactor shutdown, operation in the decay heat removal mode is accomplished by increasing the makeup flow to approximately 560 gpm; 280 gpm supplied by each of the makeup pumps. The cold traps are isolated and the total flow is routed through the overflow heat exchanger. The heat is transferred in the exchanger to NaK circulating in the EVS Processing System. Ultimate heat dissipation is via the EVST airblast heat exchangers.

During DHRS operation, initiated one-half hour after reactor shutdown, the system will limit the average bulk primary sodium temperature to approximately 1140oF. The maximum heat transferred through the overflow heat exchanger is approximately 11 Mw. With DHRS operation initiated twenty-four hours after reactor shutdown, the average bulk sodium temperature is maintained below
26 900oF.

9.3.2.2.2 Sodium Storage, Fill and Drain Operations 12

The primary sodium storage vessels, both inside and outside
containment, plus the overflow vessel, provide a capacity sufficient
35 to permit drainage at 400^{0}F of the reactor vessel. The four vessels
available for sodium storage (overflow, one in-containment, and two
46 ex-containment) provide a minimum gross capacity of ∿190,000 gal.

The primary sodium-storage vessels are also used to provide
46 storage of the sodium for the EVST. Simultaneous drainage of both
the reactor vessel and the EVST is not considered to be a realistic
operating situation.

Of the total tankage provided, enough capacity is provided inside containment to accommodate drainage situations involving a PHTS loop.

46

If complete reactor vessel drainage is required, all storage tanks and the overflow vessel are used. Prior to transfer of sodium to the storage vessel outside of containment, Na^{24} activity will be allowed to decay, and the sodium may be processed through the cold traps on its way to the storage vessels.

46 35

The overflow interruption has no effect on operation of components required for DHRS duty since all components continue to operate in a normal manner throughout the event. Prior to reactor scram, the overflow vessel is approximately one-half full, containing the coolant expansion, from 400°F, of the entire primary system. The makeup pumps continue to transfer sodium back to the reactor, and before the overflow vessel can be emptied, the reactor level is raised high enough to insure resumption of overflow. Consequently, although overflow is temporarily interrupted, flow from the overflow vessel back to the reactor vessel does not stop at any time, and all components continue to function normally.

Following the reactor shutdown, operation in the decay heat removal mode is accomplished by increasing the makeup flow to approximately 560 gpm; 280 gpm supplied by each of the makeup pumps. The cold traps are isolated and the total flow is routed through the overflow heat exchanger. The heat is transferred in the exchanger to NaK circulating in the EVS Processing System. Ultimate heat dissipation is via the EVST airblast heat exchangers.

During DHRS operation, initiated one-half hour after reactor shutdown, the system will limit the average bulk primary sodium temperature to approximately 1140°F. The maximum heat transferred through the overflow heat exchanger is approximately 11 Mw. With DHRS operation initiated twenty-fours after reactor shutdown, the average bulk sodium temperature
26 is maintained below 900°F.

9.3.2.2.2 Sodium Storage, Fill and Drain Operations 12

The primary sodium storage vessels, both inside and outside containment, plus the overflow vessel, provide a capacity sufficient
35 to permit drainage at 400^{o}F of the reactor vessel. The four vessels available for sodium storage (overflow, one in-containment, and two
46 ex-containment) provide a minimum gross capacity of ∿190,000 gal.

The primary sodium-storage vessels are also used to provide
46 storage of the sodium for the EVST. Simultaneous drainage of both the reactor vessel and the EVST is not considered to be a realistic operating situation.

Of the total tankage provided, enough capacity is provided inside containment to accommodate drainage situations involving a PHTS loop.

46

If complete reactor vessel drainage is required, all storage tanks and the overflow vessel are used. Prior to transfer of sodium to the storage vessel outside of containment, Na^{24} activity will be allowed to decay, and the sodium may be processed through the cold traps on its way to the storage vessels.

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All system components within containment are maintained in an inert (nitrogen) atmosphere during plant operation. Since the storage vessels outside containment are not normally used, and are usually empty, these vessels are in an air atmosphere; however, the cell containing the vessels may be in-
44 35 erted prior to the transfer of radioactive sodium to the ex-containment vessels.

Drainage of the PHTS loops is accomplished through drain lines donnected to each of the loops at a low point of the primary pump and IHX. All drain lines are routed to a common header, which is connected to all storage vessels and to the overflow vessel. The drain lines are sized
46 to provide a PHTS drain rate, by gravity, of 200 gpm. There is no permanent drain line provided to drain the reactor vessel.

The two primary sodium makeup pumps, taking suction from the storage vessels, are used for primary system fill. Operation of either or both pumps, to provide a flow of at least 100 gpm, will fill a PHTS loop in a 5-hour period.

44 35 Primary system fill from the storage vessels outside of containment
is accomplished by pressurizing the storage vessel with sufficient gas pressure to assure a positive head to the makeup pumps. The sodium is then pumped to the primary system. The gas pressure required for this type of transfer is ∿5 psig for a 200 gpm flow rate, and 2 psig for a 150 gpm, or lower, flow rate.

If necessary, primary system fill can be accomplished without the use of the makeup pumps, by gas pressurization only. In this situation, a gas pressure of ∿32 psig in the storage vessels is required to provide a fill rate of 200 gpm.

Sodium can be transferred from the ex-containment storage vessels to the EVST tank by applying gas pressure to the storage vessels, sufficient to
44 maintain a positive head to the EVST sodium pump. This would require approximately the same gas pressure in the storage vessels as that required for primary system fill.

In all cases of sodium transfer, involving ex-containment storage, the sodium temperature is $400^{o}F$.

Inadvertent sodium transfer from the primary sodium storage vessels to the reactor coolant system is prevented by a minimum of two isolation valves in series, both locked closed, at each fill and drain connection to the coolant system, as shown on Figure 9.3-2. These are the only connecting lines between the reactor coolant and sodium storage vessels. Another normally closed valve in the drain header piping to the storage vessels provides 2
additional protection against accidental transfer.

Inadvertent sodium transfer between the EVST sodium cooling loops and the sodium storage vessels is prevented by two normally closed isolation valves in series between the storage vessel and each fill and drain connection to the cooling loops, as shown in Figure 9.3-3.

Inadvertent transfer of sodium to and from the EVST itself, via its drain line (from the bottom of the backup sodium cooler), is prevented by a flanged spool piece in the drain line. Once the EVST is filled, the spool piece is removed and the remaining pipe ends sealed by flanges, to prevent accidental drainage. The removed section is physically located high enough so that any sodium leak at the flanged joint cannot drain the EVST below a level which would interfere with the cooling system.

9.3.2.3 Design Evaluation

The system components are designed to accepted industrial and nuclear standards to insure structural integrity and operational reliability. The components, applicable design code and class, plus their seismic category are listed in Table 9.3-1. Design temperatures and pressures are given in Table 9.3-7.

All parts of the system are monitored by leak detection devices, with alarms for detection of external leakage from piping and components.

The instrumentation also will determine the occurrance of internal leakage from one part of the system to another, and alert the operator so the systems can be shut down for maintenance or repair.

Those cells which house the Primary Sodium Processing System will be inerted during system operation and during all periods while a potential Na spill could otherwise result in potential off-site radiological release excess of 10CFR20 limits.

The Primary Sodium Storage and Processing System is connected to the reactor by the overflow and the makeup return lines. The nozzles for these lines are located near the top of the reactor; so that, in the extremely unlikely event of a line failure, there would be very little sodium lost from the reactor, and reactor cooling would not be affected.

All of the system except for the ex-containment storage vessels are located in the RCB, in cells which provide shielding, an inerted atmosphere, and tornado protection.

When it is necessary to store sodium from the Primary Heat Transport System in the ex-containment vessels, it is radioactively decayed for approximately 10 days before it is transferred to the ex-containment vessels.

All components of the overflow and makeup circuit (which may be required for removal of reactor decay heat) are designed as Seismic Class I components. The makeup pumps are connected to the emergency power supply in order to insure operation during decay heat removal.

The worst in-containment accident that can be postulated for this system is rupture of the overflow vessel or the in-containment primary sodium storage tank which could dump ~35,000 gal. of primary sodium into the cell. This accident is discussed in Section 6.2.1.3.

The possible plugging of the overflow line is not considered a credible event, because it is an 8 in. and 6 in. line, normally flowing at 150 gpm, with the line sloped to the overflow vessel at 1/2 to 3/4 in./ft. The sodium is maintained at a low oxygen content (2ppm or less) which, combined with the high temperature, keeps oxides from building up in the line. If somehow the line were plugged, the reactor sodium level would slowly rise, until highlevel alarms initiated operator action to shut down the plant. Shutdown lowers the reactor sodium level, and no safety aspects are involved.

9.3.2.3.1 Analysis of Loss of Cold Trap Cooling

In the event cooling is interrupted on the PHTS cold trap and sodium flow continues, the sodium temperature in the crystallizer will rise rapidly and approach the inlet temperature, which will be up to 880°F. The consequence would be dissolution of the solid sodium-oxide (Na_2O) and solid sodium hydride (NaH).

Dissolution of Na_2O and NaH

If the PHTS were clean (~2 ppm oxygen and ~ 0.2 ppm hydrogen) and the first cold trap were filled with Na_2O and NaH, the interruption of cooling on the cold trap without simultaneously cutting off sodium flow could cause a maximum release of oxygen and hydrogen to the system by dissolution. The hot sodium continuing to flow into the uncooled crystallizer would raise its temperature rapidly and cause the Na_2O and NaH to go back into solution.

The time available to take corrective action (e.g., to shut off sodium valves and flow into the cold trap) depends on the rate of dissolution.

Calculations were made of the time required to dissolve the entire contents of the first PHTS cold trap as a function of the sodium temperature in the crystallizer. Table 9.3-5 shows how this time changes significantly with temperature. It shows that there is some minimum temperature required for all of the Na_2O or NaH to be completely dissolved. Below this temperature, the system would reach saturation with Na_2O or NaH after a long time of recirculation through the trap, and the traps would contain a residue of solid Na_2O or NaH.

The first cold trap has been estimated to contain 13 v/o (volume percent Na_2O and 87 v/o N-H at end-of-life, which is after about 13 years of full-power operation. Table 9.3-5 shows that it would take 2.7 hours with 800°F sodium flowing at 60 gpm to redissolve all the Na_2O while it would take 7.4 hours to redissolve all the hydrogen. If the sodium were 450°F, it would take 138 hours to dissolve all the Na_2O, but only 25.4% of the NaH would be dissolved after a long time (theoretically approaching infinity).

Conclusion

There is on the order of at least an hour available to the operators to take corrective action following loss of cold trap cooling before enough oxide or hydride can be dissolved from the cold trap to raise their concentrations in the PHTS to saturation levels. Assuming that the sodium flow valves were closed within hours after loss of cooling, no potential for plugging any part of the PHTS would exist. Above 440°F (PHTS sodium temperature), the oxygen could not reach saturation even if all Na_2O were redissolved or flushed from the cold trap. Above 555°F, the hydrogen could not reach saturation even if all NaH were redissolved.
Safety related instrumentation will indicate and alarm a cold trap high temperature condition to the operators to assure remote manual closure of the cold trap isolation valves. This will prevent dissolution of enough hydride or oxide to preclude safe cooldown to refueling conditions. Even assuming instrumentation failure and no operator action at all, the dissolved oxide and hydride will remain in solution in the coolant both during reactor operation and following shutdown to hot standby condition. Consequently, operation of the cold trap cooling system is not a safety function and the system should be considered a non-safety class. However, the primary cold traps are connected to the reactor coolant boundary through double automatic isolation valves.

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Feb. 1983

The equipment of the Primary Sodium Storage and Processing System is mounted in cells that have an inert atmosphere, but are accessible after system or component shutdown for inspection after de-inerting cells and radioactivity decay. The equipment is mounted or supported so that inspection of vessels, pumps and piping can be accomplished.

9.3.2.5 Instrumentation Requirements

Instrumentation and controls (I&C) are provided for operation,
performance evaluation and diagnosis of the Primary Na Storage and
Processing System. These functions are required for off-normal as well
as for the full range of normal operation. Details of the I&C for the sub-
system are shown in the piping and instrumentation diagram, Figure 9.3-2.
26 DHRS instrumentation is discussed in Section 5.6.2.1.6. The following
I&C is required to ensure safe operation of and to prevent extensive
damage to the Primary Na Storage and Processing System.

Temperatures at the inlet and outlet of all heat source and
sink components, in conjunction with loop flow measurements are provided
for all systems to monitor their status. Critical temperatures and flows
are alarmed to alert the operator to off-normal operations. All EM pumps
46 are provided with winding temperature measurements and winding coolant
low flow indication. These measurements are alarmed for off-normal
conditions and interlocked to automatically shutdown the pump to prevent
damaging it.

Storage tanks are provided with level measurements, which are
alarmed for abnormal low and/or high level. This information, in con-
junction with leak detection data, is required to diagnose external liquid
metal leaks. The operator is alerted to NaK to sodium leakage by NaK
50 expansion tank high-low pressure alarms. Differential pressure sensors and
46 flow meters are provided to alert the operator to possible plugging of
the cold traps or insufficient cold trap flow. All the bellows seal valves
are provided with leak detectors (Section 7.5.5.1). All valves are
provided with position indicators. Accessories (solenoids, pressure
gauges, I/P converters, etc.) for remotely operated valves installed in
inerted cells are installed off of the valves in adjacent accessible
54 air cells. The stem portion of the sodium valve is monitored and
46 alarmed for low temperature to ensure free operation and protect
the valve sodium seal from damage. To provide for continued
operation and prevent possible system damage resulting from control
system failures, hand controllers are provided for all controllers. The
50 hand controller allows the operator to manually operate the system while
the defect is repaired.

Redundant temperature sensors are provided for each primary cold trap. High temperature conditions in either cold trap are indicated and alarmed in the Control Room to ensure that the cold trap is isolated prior to plant cooldown to refueling temperature. Thus plugging from high impurity content in the PHTS is precluded. 36

9.3.3 EVS Sodium Processing

9.3.3.1 Design Basis

See Section 9.1.3.1.1.

9.3.3.2 Design Description

See Section 9.1.3.1.2.

9.3.3.3 Design Evaluation

See Section 9.1.3.1.3.

9.3.3.4 Tests and Inspections

See Section 9.1.3.1.4.

9.3.3.5 Instrumentation Requirements

See Section 9.1.3.1.5.

9.3.4 Primary Cold Trap NaK Cooling System

9.3.4.1 Design Basis

The system provides the capability to cool the two primary cold traps and transfer the heat (268 kW) to a closed recirculating Dowtherm J loop which in turn transfers the heat to the Normal Chilled Water System (Section 9.7)

9.3.4.2 Design Description

The Primary Cold Trap NaK Cooling System provides capability for cooling the primary sodium cold traps and for transferring the heat to Dowtherm J (as described in 9.3.4.1 and Section 9.7). The system includes facilities for storage, filling and draining of its NaK. The system includes the following components:

a. Primary Cold Trap NaK Storage Vessel

b. Primary Cold Trap NaK Pump

c. Primary Cold Trap NaK Diffusion Cold Trap

d. Primary Cold Trap NaK Cooler

e. Interconnecting Piping and Valves

Refer to Figures 9.3-4 for the P&ID and Figure 1.2-7 for layout and arrangement.

During normal operation, when one primary sodium cold trap is in use, NaK is circulated through the cooling loop at an approximate flow rate of 80 gpm and temperature of $125^{o}F$. The NaK flows from the cold trap NaK pump and up through the jacket of the primary sodium cold trap. From there, it passes down through the primary cold trap NaK cooler, and then back to the pump.

The primary cold trap NaK cooling system is sized to provide the capacity to cool two primary sodium cold traps simultaneously (total flow rate of 160 pgm).

The total maximum heat transferred from two traps is approximately 270 kW. The NaK flow through each trap is regulated by a control valve and a temperature sensing element in each cold trap.

NaK volumetric changes in the system due to temperature variations are accommodated within the primary cold trap NaK storage vessel.

A diffusion cold trap is provided for system cleanup (oxide removal). (See Figure 9.3-4)

The NaK loop is taken up to a near isothermal condition at 600°F and circulated for processing by the NaK diffusion cold trap during initial cleanup, following maintenance operations, or at any other time when maximum cleanup capability is required to reduce the impurities which may have accumulated in the loop. Heat is provided by the pump.

9.3.4.3 Design Evaluation

The Primary Cold Trap NaK Cooling System components are designed to accepted industrial and nuclear standards to insure structural integrity and operational reliability. The components, applicable design code and class, plus their seismic category are listed in Table 9.3-1. Design temperatures and pressures are given in Table 9.3-7.

The NaK Cold Trap Cooling System is a nonradioactive system. The failure of the system causes shutdown of the associated cold trap which would, after some period of time, require an orderly plant shutdown, but would create no safety problems.

The system is monitored by leak detection devices for both internal and external leaks with alarms to alert the operator to take corrective action.

The pressure of the NaK in the cold trap cooling system is maintained higher than that of the sodium in the primary cold traps in order to insure in-leakage of NaK rather than out-leakage of radioactive sodium. The NaK is compatible with the primary coolant and NaK inleakage will have no deleterious effect on reactor operation and safety. The immediate result of a NaK-to-sodium leak will be an abnormal decrease in both NaK level and cover gas pressure in the NaK storage vessel. Both level and pressure are monitored and provided with high and low alarms for leak detection (high level indicates a Dowtherm-to-NaK leak). Due to a low reserve head in the tank, a large leak will stop due to loss of suction head, and this will also be detected by low flow alarms on system flow meters and high temperature alarms on the pump. Upon indication of a leak, the operating cold trap will be immediately isolated to minimize the NaK in-leakage. The cold trap will be solidified, removed, and replaced. The NaK storage vessel is a vertical vessel in order to maximize the change in NaK level for a given change in inventory, thus enhancing the capability of the level indicator and alarm to sense and detect relatively small leaks. A realistic estimate of NaK inleakage, prior to cold trap isolation, would be 25 gallons or less. A more accurate estimate will be made when system arrangement and component designs are finalized.

The equipment is mounted in cells that have an inert atmosphere, but are accessible after system or component shutdown for inspection after de-inerting. The equipment is mounted or supported so that inspection of vessels, pumps and piping for possible deterioration of NaK containment integrity can be accomplished.

9.3.4.5 Instrumentation Requirements

Instrumentation and controls (I&C) are provided for operation, performance evaluation and diagnosis of the NaK cold Trap Cooling System. These functions are required for off-normal as well as for the full range of normal operation. Details of the I&C for the subsystems are shown in the piping and instrumentation diagram, Figure 9.3-4. The following I&C is required to ensure safe operation of, and to prevent extensive damage to the NaK Cold Trap Cooling System.

Temperatures at the inlet and outlet of all heat source and sink
components, in conjunction with loop flow measurements are provided for
all systems to monitor their status. Critical temperatures and flows are
alarmed to alert the operator to off-normal operations. The EM pump is
46 provided with winding temperature measurements and winding coolant low
flow indication. These measurements are alarmed for off-normal conditions,
and interlocked to automatically shutdown the pump to prevent damaging it.

46 The storage tank is provided with a level measurement which is
alarmed for abnormal low and/or high level. This information, in con-
junction with leak detection data, is required to diagnose external liquid
15 metal leaks. The operator is alerted to NaK to sodium or Dowtherm J
46 to Nak leakage by NaK storage tank high-low pressure alarms, in conjunction
with the level measurement mentioned previously (see also Section 9.1.3).
All the bellows seals valves are provided with leak detectors. All valves
are provided with position indicators. Accessories (solenoids, pressure
gauges, I/P converters, etc.) for remotely operated valves installed in
inerted cells are installed off of the valves in adjacent accessible air
54 cells. To provide for continued operation and prevent possible system
damage resulting from control system failures, hand controllers are
provided for all controllers. The hand controller allows the operator
to manually operate the system while the defect is repaired.

9.3.5 Intermediate Na Processing System

9.3.5.1 Design Basis

The system provides the capability to limit the oxygen and hydro-
gen concentration of IHTS sodium to 2.0 ppm and 0.2 ppm, respectively.
50 The system, working in conjunction with the primary cold traps, limits
the tritium content of IHTS sodium.
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Nov. 1980

The system also provides the capability to (1) fill the IHTS loops from the sodium dump tanks (these tanks are part of the Steam Generator System), (2) purify sodium in the dump tanks, independently of the IHTS loops, and (3) permit transfer of sodium from one dump tank to another.

9.3.5.2 Design Description

The Intermediate Sodium Processing System provides purification of the sodium in each of the three IHTS loops. The System does not provide for storage of the IHTS sodium. This capability is provided by the sodium dump tanks, which are part of the Steam Generator System. The Intermediate Sodium Processing System does provide the capability of transferring sodium into the loops from the dump tanks. The same piping network allows the filling of each dump tank with fresh sodium from tank cars or drums at the sodium receiving station. Sodium removal from the tanks into tank cars can be accomplished through the same fill lines.

The system includes the following components:

a. Intermediate Sodium Cold Trap Pumps

b. Intermediate Sodium Cold Traps

c. Interconnecting Piping and Valves

Refer to Figure 9.3-5 for the P&ID and Figures 1.2-8 and 1.2-22 for layout and arrangement.

Each of the three IHTS loops is provided with a separate purification system consisting of a pump, two cold traps, and the necessary valves and piping. A single trap per IHTS loop is sufficient to remove anticipated oxygen and hydrogen inleakage and to limit these impurities to a maximum of 2 and 0.2 ppm, respectively.

The IHTS cold trap lifetime is 3.25 yr unless a regeneration process is adopted. The procedure for replacement of the crystallizer is to partially drain the sodium while maintaining a temperature of 300-400oF, remove all heater power and allow the remaining sodium to freeze, cut and cap the sodium lines, cut the electrical leads, unbolt the supports, pull the crystallizer, and remove it by crane to storage.

The intermediate sodium cold trap pumps are used to pump the sodium from the dump tanks into the loops with a small, ∼ 22 psig, cover gas pressure being maintained on the dump tanks. Sodium can also be transferred from one dump tank to another by gas pressure.

9.3.5.3 Design Evaluation

The Intermediate Sodium Processing System components are designed to accept industrial and nuclear standards to insure structural integrity and operational reliability. The components, applicable design code and class, plus their seismic category are listed in Table 9.3-1. Design temperatures and pressures are given in Table 9.3-7.

The system is monitored by leak detection devices, with alarm for detection of external leakage from piping and components.

The Intermediate Sodium Processing System is a nonradioactive system. A leakage failure in a purification circuit is considered the most significant event, because it ultimately causes loss of the IHTS loop to which it is connected. The loss of fluid would be signaled by the level indicators on the IHTS loop expansion tank, and operators could immediately isolate the purification circuit (remote controlled isolation valves). This event would cause loss of one IHTS loop because of the inability to maintain required purity levels, but the plant could continue to operate on the other two loops at a reduced power level. Power outage to all pumps, or loss of cooling to all cold traps, may cause plant shutdown, but would not constitute a safety problem.

9.3.5.4 Tests and Inspections

Leak checks will be made on the system prior to filling with sodium. Instrumentation and preheat capability will be checked prior to sodium fill, according to specific procedures. Following sodium fill, the system will be operationally tested.

9.3.5.5 Instrumentation Requirements

Instrumentation and controls (I&C) are provided for operation, performance evaluation, and diagnosis of the Intermediate Sodium Processing System. These functions are required for off-normal, as well as for the full range of normal operation. Details of the I&C for the subsystem are shown in the piping and instrumentation diagram, Figure 9.3-5. The following I&C is required to ensure safe operation of, and to prevent extensive damage to, the Intermediate Sodium Processing System.

Temperatures and loop flow measurements are provided for all systems to monitor their status. Critical temperatures and flows are alarmed to alert the operator to off-normal operations. All EM pumps are provided with winding temperature measurements and winding coolant low flow indication. These measurements are alarmed for the off-normal conditions, and interlocked to automatically shutdown the pump to prevent damaging it.

Differential pressure sensors and flow meters are provided to alert the operator to possible plugging of the cold traps or insufficient cold trap flow. All the bellows seal valves are provided with leak detectors, as indicated on Table 9.3-4. All valves are provided with position indicators. The stem

portion of the sodium valve is monitored and alarmed for low temperature, to
ensure free operation and protect the valve sodium seal from damage. To
46 provide for continued operation and prevent possible system damage resulting
from control system failures, hand controllers are provided for all
controllers. The hand controller allows the operator to manually operate the
system while the defect is repaired.

TABLE 9.3-1

AUXILIARY LIQUID METAL SYSTEM COMPONENTS
STRUCTURAL CRITERIA

System	Component	Applicable Design Code: Code	Class	Seismic Category
Sodium and NaK Receiving	Piping*	ANSI B 31.1	-	3
	Fresh Sodium Filters	ASME-VIII Div. 1	-	3
	Fresh NaK Filter	ASME-VIII Div. 1	-	3
Primary Sodium Storage and Processing	Piping*			
	Primary Loop drain lines(1)	ASME III	1	I
	Overflow line	ASME III	1	I
	Makeup and cold trap circuit(1)	ASME III	1	I
	PHTS Drain piping inside containment	ASME III	2	I
	Piping outside containment	ASME III	3	I
	Primary Sodium Overflow Vessel	ASME III	1	I
	Overflow Heat Exchanger	ASME III	1	I
	Primary Sodium Makeup Pumps	ASME III	1	I
	Primary Sodium Cold Traps	ASME III	3	I
	Makeup Pump Drain Vessel	ASME III	3	I
	In-Containment Primary Sodium Storage Vessel	ASME III	3	I
	Ex-Containment Primary Sodium Storage Vessel	ASME III	3	I
EVS Sodium Cooling				
	EVS Sodium Coolers	ASME III	2	I

*Valves will comply to the same standards.

(1) To second isolation valve.

TABLE 9.3-1 (Continued)

	System	Component	Applicable Design Code		Seismic Category
			Code	Class	
	EVS Sodium	EVST Sodium Pumps	ASME III	2	1
44	Cooling (cont)	EVST Sodium Cold Trap	ASME III	3	1
26		Piping*(2)	ASME III	2	1
46	EVS NaK	Piping(5)	ASME III	2	1
	Cooling	EVST NaK Pumps	ASME III	2	1
		EVST NaK Expansion Tanks	ASME III	2	1
		EVST NaK Expansion Tank(4)	ASME III	3	1
26		DHRS NaK Expansion Tank	ASME III	2	1
		EVST Na Coolers	ASME III	2	1
26		EVST Backup Na Cooler(4)	ASME III	2	1
		EVST Air Blast Heat Exchanger	ASME III	2	1
		EVST Natural Draft(4) Heat Exchanger	ASME III	3	1
		EVST NaK Diffusion Cold Traps	ASME III	2	1
		EVS NaK Diffusion Cold(4) Traps	ASME III	3	1
44		EVS NaK Storage Vessel(3)	ASME VIII	-	2
	Primary Cold Trap NaK Cooling (3)	Primary Cold Trap NaK Pump	ASME VIII	-	2
		Primary Cold Trap NaK Cooler	ASME VIII	-	2
		Primary Cold Trap NaK Storage Vessel	ASME VIII	-	2
		Primary Cold Trap NaK Diffusion Cold Trap	ANSI B 31.1	-	2
26		Piping*	ANSI B 31.1	-	2

*Valves will comply to the same standards

44 (2) Drain piping and cold trap interconnecting piping is ASME III-3, Seismic I

(3) Classified ASME VIII or B 31.1 but designed and constructed to ASME III-3 requirements

26 (4) Components of the third (backup) cooling circuit

(5) NaK piping for the third (backup) cooling circuit and NaK drain piping is ASME III-3, Seismic 1. All other NaK piping is Class 2. 46

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TABLE 9.3-1 (Continued)

System	Component	Applicable Design Code		Seismic Category
		Code	Class	
Intermediate Sodium Processing	Intermediate Sodium Traps	ASME III	3	I
	Intermediate Sodium Cold Trap Pumps	ASME III	3	I
	Piping* (4)	ASME III	3	I

26

*Valves will comply to the same standards.

46 26 (4) Piping between IHTS loops and first isolation valves is classified ASME III-2 Seismic 1.

TABLE 9.3-2

TOTAL PRIMARY SODIUM STORAGE REQUIREMENTS
(Sodium Volumes at 400°F)

	Ft^3	Gallons
Reactor Vessel Drain		
Reactor Vessel	13,328	99,960
PHTS Piping which drains to the Reactor Vessel	1,950	14,625
Reduction in PHTS Na Level	2,328	17,460
EVST Reserve	1,000	7,500
Overflow/Makeup Piping	625	4,688
In-Leakage from a Single IHX Leak	1,350	10,125
Total Usable Storage Required	20,581	154,358

46 Table 9.3-3 has been deleted.

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TABLE 9.3-4

PLANNED LOCATION OF LEAK DETECTORS FOR AUXILIARY LIQUID METAL SYSTEM

Building	Cell	Components	Instrument Type	Quantity
SGB	224	Cold Trap Valves	Contact	4
SGB	225	Cold Trap Valves	Contact	5
SGB	226	Cold Trap Valves	Contact	4
SGB	244	Cell	Aerosol	2
SGB	244	Cold Trap Loop 1	Cable	1
SGB	244	Valves	Contact	8
SGB	245	Cell	Aerosol	2
SGB	245	Cold Trap Loop 2	Cable	1
SGB	245	Valves	Contact	8
SGB	246	Cell	Aerosol	2
SGB	246	Cold Trap Loop 3	Cable	1
SGB	246	Valves	Contact	8
SGB	211	Sodium and NaK Receiving System and Primary Sodium Storage Tanks	Cable	4
RCB	121	PHTS Loop No. 1 Drain Valves	Contact	3
RCB	122	PHTS Loop No. 2 Drain Valves	Contact	3
RCB	123	PHTS Loop No. 3 Drain Valves	Contact	3
RCB	102	Cell	Aerosol	2
RCB	102	Overflow & Storage Tanks	Cable	3
RCB	104	Cell	Aerosol	1
RCB	103	Cell	Aerosol	1
RCB	107	Cells	Aerosol	3
RCB	107	Overflow Heat Exchanger	Cable	1
RCB	107	Valves	Contact	23
RCB	157	Primary Na Cold Trap Cells	Aerosol	4
RCB	157	Valves	Contact	8
RCB	157	Pri. Cold Traps	Cable	2
RCB	131	Primary NaK Cold Trap Cell	Aerosol	1
RCB	131	Valves	Contact	3
RCB	131	NaK Cooler and Storage Vessel	Cable	2
RSB	360	EVS Na Loop 1	Aerosol	1
RSB	360	Valves	Contact	7
RSB	360	Components	Cable	1
RSB	357	EVS Na Loop 2	Aerosol	1
RSB	357	Valves	Contact	10
RSB	357	Components	Cable	1
RSB	331	EVST Backup Cooler Cell	Aerosol	1
RSB	331	EVST Backup Cooler Cell	Cable	1
RSB	331	Valves	Contact	2
RSB	332	Natural Draft Heat Exchanger	Aerosol	2
RSB	332	Natural Draft Heat Exchanger	Cable	3
RSB	337	EVST Na Pipe	Aerosol	1

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TABLE 9.3.4

PLANNED LOCATION OF LEAK DETECTORS FOR AUXILIARY LIQUID METAL SYSTEM

(Continued)

Building	Cell	Components	Instrument Type	Quantity
RSB	350	EVST NaK Storage Vessel Cell	Aerosol	1
RSB	350	EVST NaK Storage Vessel Cell	Cable	1
RSB	351	Pipeway	Aerosol	2
RSB	352	Valve	Contact	1
RSB	352A	EVST NaK Loop 1 Cell	Aerosol	2
RSB	352A	EVST NaK Loop 1	Cable	2
RSB	352A	Valves	Contact	4
RSB	353A	EVST NaK Loop 2 Cell	Aerosol	2
RSB	353A	EVST NaK Loop 2	Cable	3
RSB	353A	Valves	Contact	5
RSB	351B	Valves	Contact	2
RSB	354	Pipeway	Aerosol	1
RSB	355	Pipeway	Aerosol	1
RSB	358	Pipeway	Aerosol	3
RSB	361	EVST Cold Trap Cell	Aerosol	1
RSB	361	EVST Cold Trap Cell	Cable	1
RSB	361	EVST Cold Trap Cell	Contact	2
RSB	362	Pipeway	Aerosol	1

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TABLE 9.3-5

DISSOLUTION OF PHTS COLD TRAP
SODIUM OXIDE AND SODIUM HYDRIDE

System Temp. °F.	Time Required to Empty Cold Trap, hours**		Fraction Removed From Cold Trap	
	Oxygen	Hydrogen	Oxygen	Hydrogen
	v/o → 13	87	13	87
350	∞*	∞	.223	.031
400	∞	∞	.572	.099
450	138	∞	1.0	.254
500	45.1	∞	1.0	.525
550	22.9	∞	1.0	.993
600	13.2	63.1	1.0	1.0
650	8.3	31.2	1.0	1.0
700	5.5	18.0	1.0	1.0
800	2.7	7.4	1.0	1.0
880	1.6	4.1	1.0	1.0

*∞ = Infinity implies that all the oxygen or hydrogen could not be removed from the cold trap no matter how long it were flushed at this temperature. The system sodium would become saturated with oxygen (or hydrogen) after a very long time.

** Removal Efficiency = 70%

TABLE 9.3-6

AUXILIARY LIQUID METAL SYSTEM ACTIVE(1) PUMPS AND VALVES

Part No.	Description	System
Pumps		
81PPP001A	Primary Sodium Makeup	Primary Sodium Storage and Processing
81PPP001B	Primary Sodium Makeup	Primary Sodium Storage and Processing
81EPP002A	EVST Sodium	EVS Processing
81EPP002B	EVST Sodium	EVS Processing
81EPP003A	EVST NaK	EVS Processing
81EPP003B	EVST NaK	EVS Processing
Valves		
HV 102	4-in. DHX and Cold Trap Bypass	Primary Sodium Storage and Processing
HV 103	4-in. DHX Inlet	Primary Sodium Storage and Processing
HV 109	3-in. Cold Traps Inlet	Primary Sodium Storage and Processing
HV 149	3-in. Cold Traps Outlet	Primary Sodium Storage and Processing
HV 153	3-in. Overflow Return to Reactor	Primary Sodium Storage and Processing
HV 107	4-in. DHX Inlet	Primary Sodium Storage and Processing
HV 357	4-in. DHX to ABHX (81EPH002A) Connection NaK	EVS Processing
HV 358	4-in. DHX to ABHX (81EPH002A) Connection NaK	EVS Processing
HV 359	4-in. NaK Loop 1 Valve	EVS Processing
HV 383	4-in. Na Loop 1 EVST Return	EVS Processing
HV 384	4-in. Na Loop 1 EVST Outlet	EVS Processing
HV 393	4-in. Na Loop 2 EVST Outlet	EVS Processing
HV 397	4-in. Na Loop 2 EVST Return	EVS Processing
HV 415	4-in. DHX to ABHX (81EPH002B) Connection NaK	EVS Processing
HV 416	4-in. DHX to ABHX (81EPH002B) Connection NaK	EVS Processing
HV 420	NaK Loop 2 Valve	EVS Processing
HV 104	4-in. Makeup Pump Discharge	Primary Sodium Storage and Processing
HV 118	4-in. Makeup Pump Discharge	Primary Sodium Storage and Processing

(1) An active component is one in which mechanical movement must be initiated or electrical power supplied to accomplish the safety function of the component.

TABLE 9.3-7

DESIGN TEMPERATURES AND PRESSURES*

Na and NaK Receiving System

Sodium Piping and Fresh sodium Filters	450°F, 20 psig
NaK Piping and Fresh NaK Filters	150°F, 20 psig

Primary Na Storage and Processing System

Overflow Vessel	900°F, 15 psig
Makeup Pumps, Cold-Trap	900°F, 100 psig
Overflow Heat Exchanger	650°F, 100 psig
IC Vessel, Makeup Pump Drain Vessel	650°F, 50 psig
EC Vessel	450°F, 50 psig
Piping PHTS Drain from 51A/81 Interface to "Spec. Change"	1015°F, 200 psig
PHTS Drain From "Spec. Change" to Second Isolation Valve	650°F, 200 psig
Overflow Line and Makeup Pump Suction Line - from OV to Isolation Valves on Pump Suction Line	900°F, 15 psig
Balance of Makeup Circuit	900°F, 100 psig
All Other Piping	650°F, 100 psig

EVS Processing System

Cold Trap (including piping thereof)	700°F, 100 psig
EVST Backup Sodium Cooler Piping	800°F, 100 psig
EVST NaK Storage Vessel	150°F, 50 psig
NaK Drain Piping from Loop Isolation Valve to Storage Vessel	150°F, 100 psig
All Other Components and Piping	650°F, 100 psig

Primary Cold Trap NaK Cooling System

All Components	650°F, 100 psig

Intermediate Sodium Processing System

Cold Trap Economizer and Pumps	775°F, 225 psig
Cold Trap Crystallizer and Connecting Piping	750°F, 225 psig
Piping, Normally Operating Cold Trap Circuit (incl. stand by trap and drain lines to first isolation valve)	775°F, 225 psig
Between IHTS Loops and First Isolation Valve	775°F, 325 psig
All Other Drain Piping	700°F, 100 psig

* All System 81 components designed for full vaccuum at 450°F

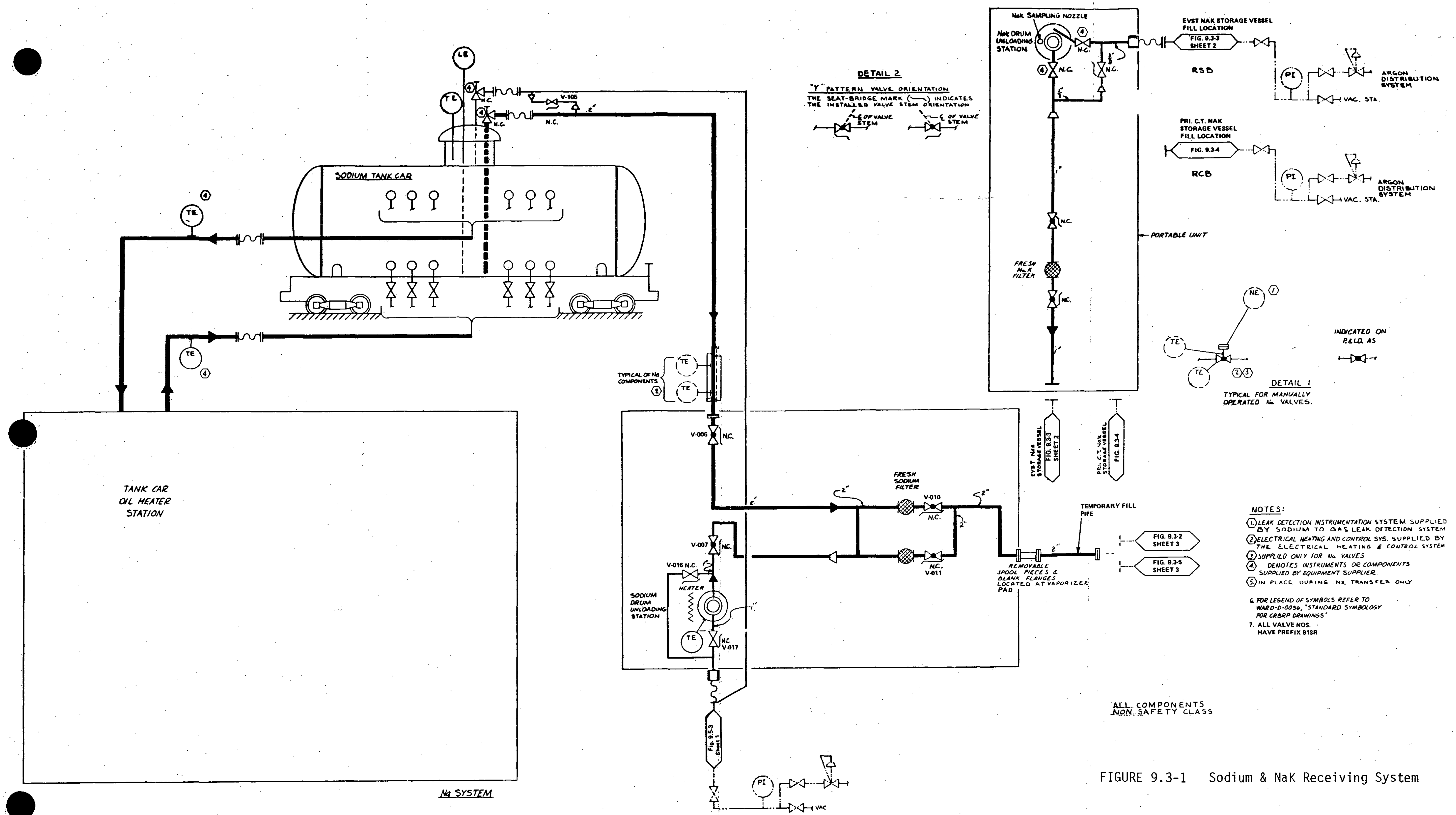

FIGURE 9.3-1 Sodium & NaK Receiving System

N099811021-4

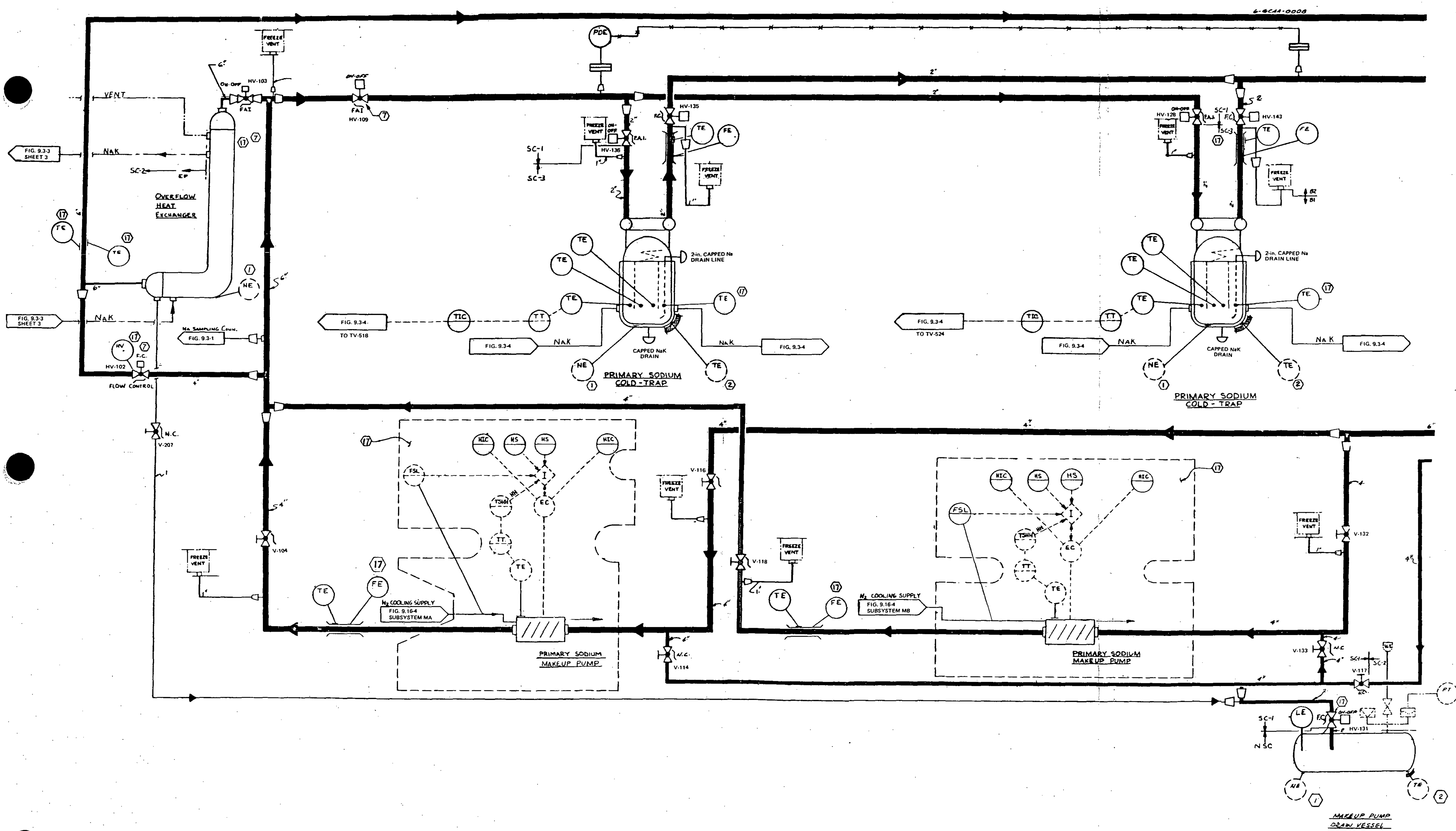

FIGURE 9.3-2 Primary Sodium Processing Storage System (Sheet 1 of 3)

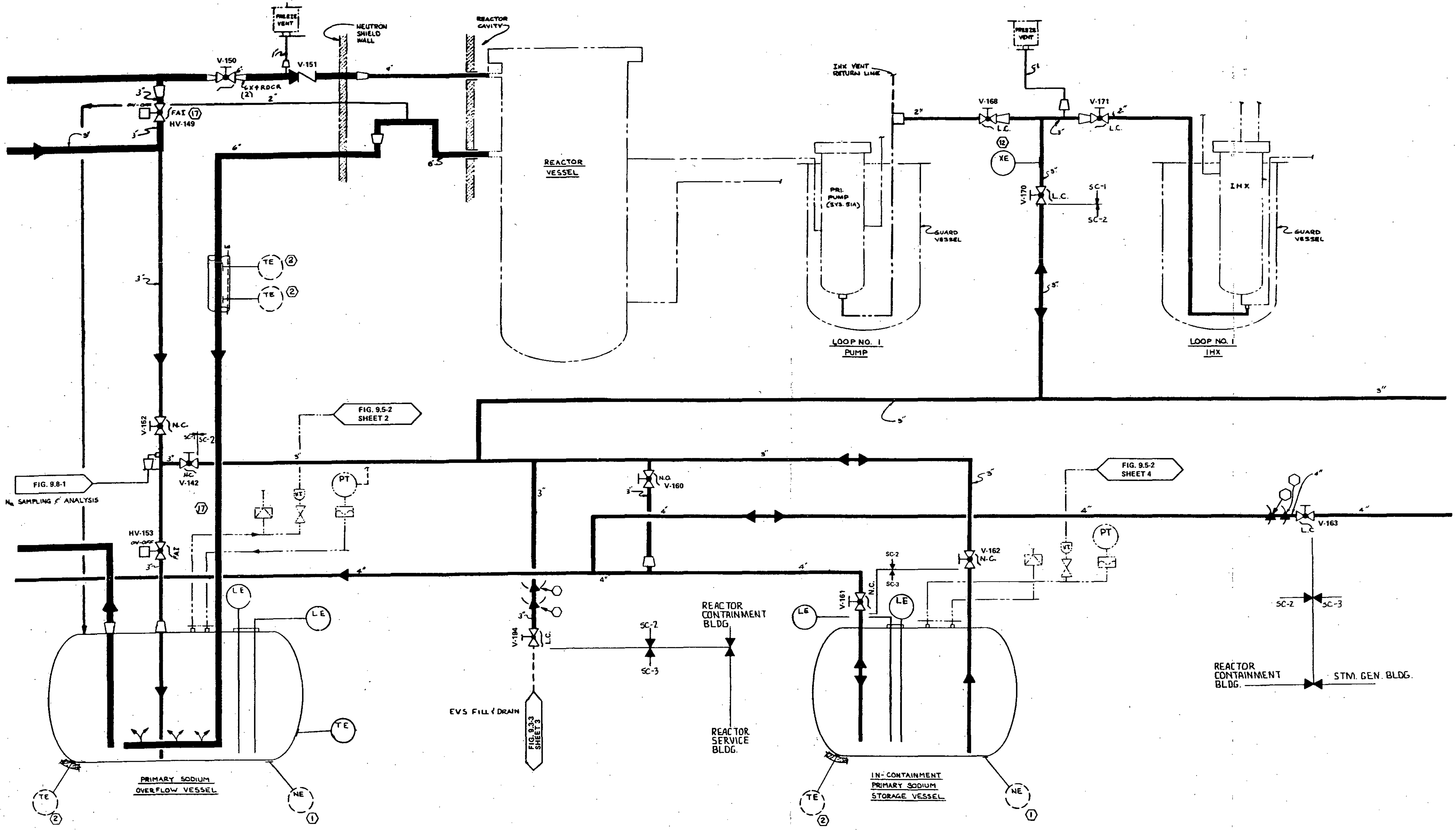

FIGURE 9.3-2 Primary Sodium Processing Storage System (Sheet 2 of 3)

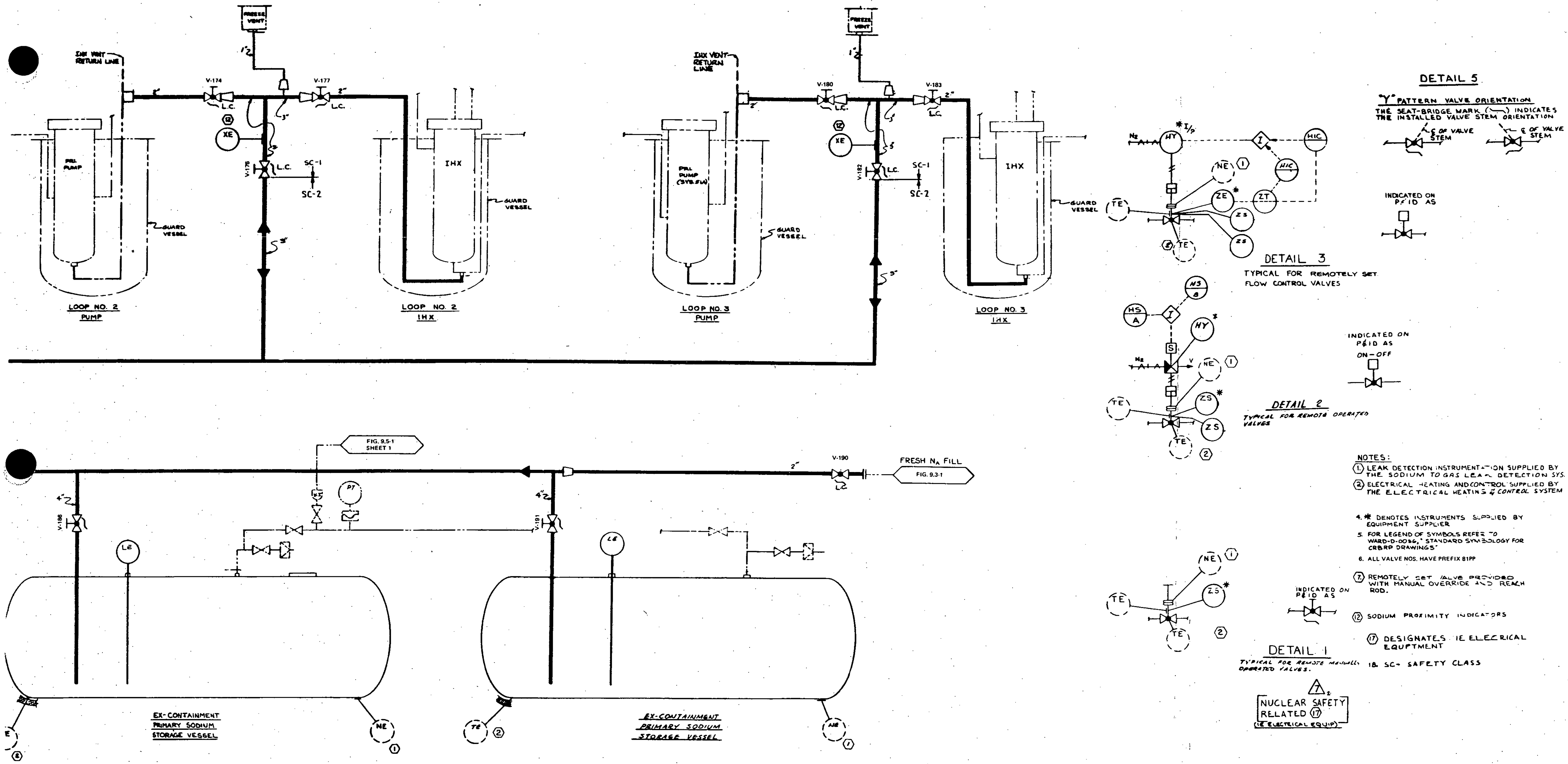

FIGURE 9.3-2 Primary Sodium Processing Storage System (Sheet 3 of 3)

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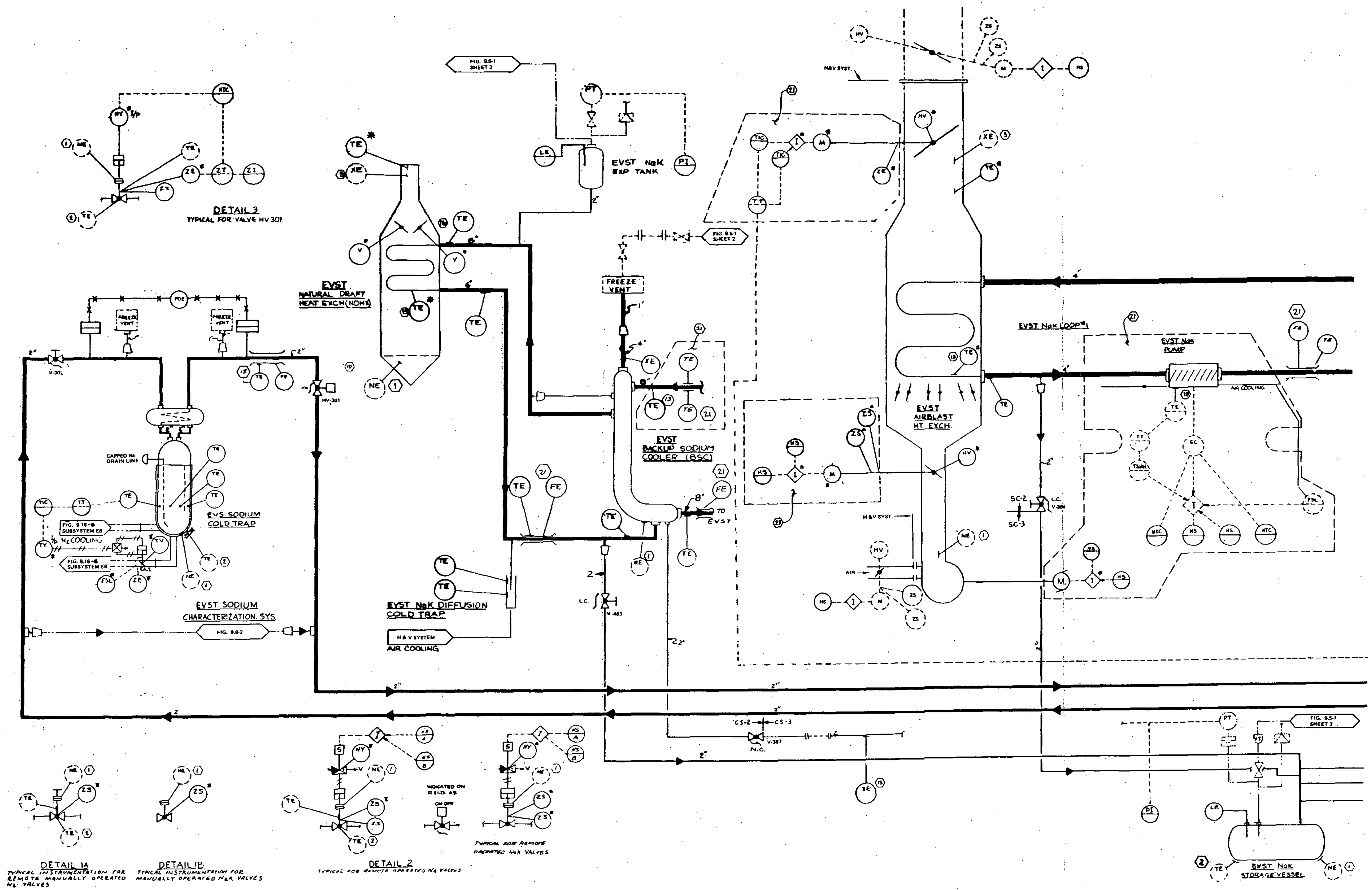

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FIGURE 9.3-3 Ex-Vessel Storage Sodium Processing System (Sheet 1 of 3)

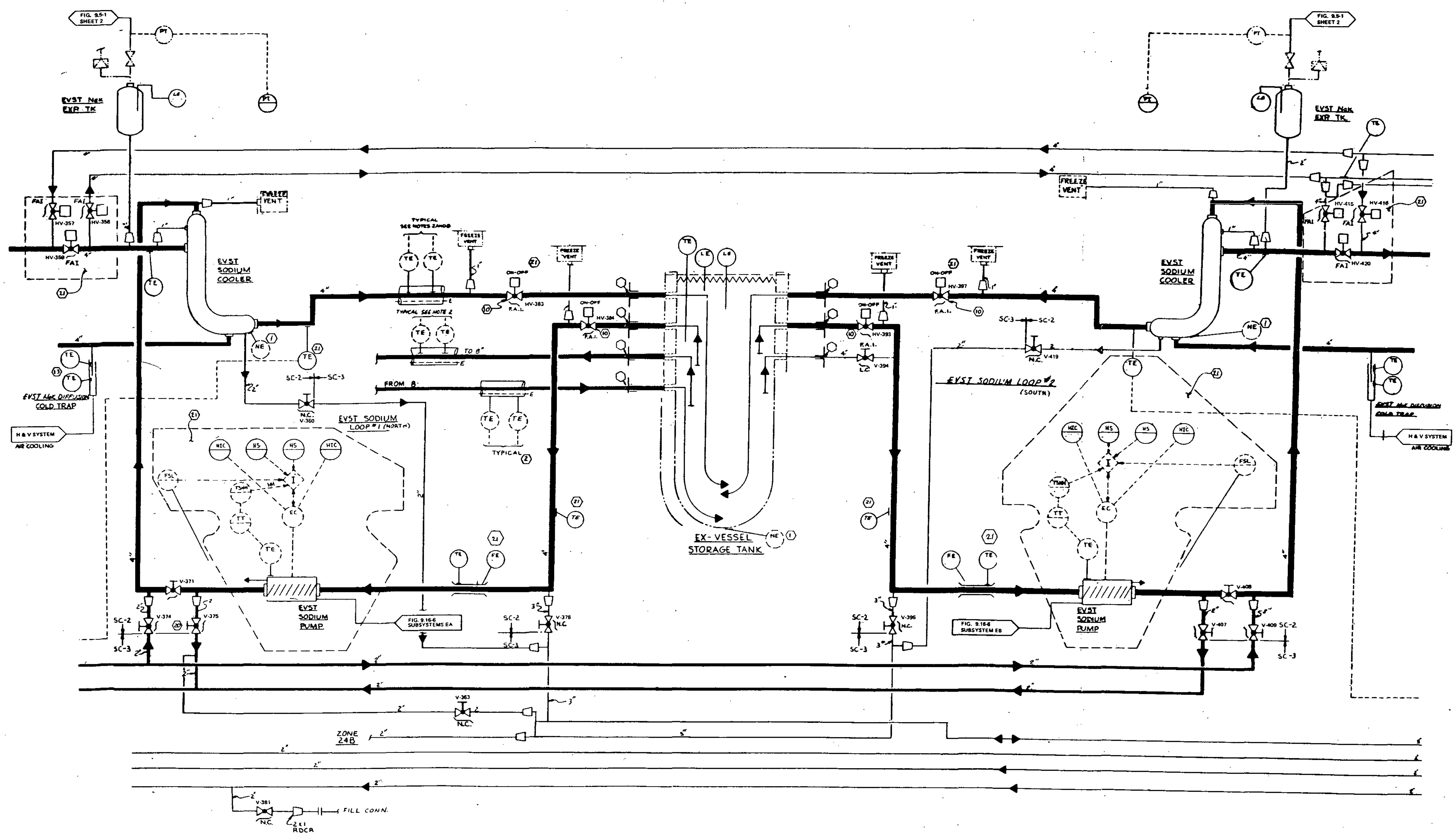

FIGURE 9.3-3 Ex-Vessel Storage Sodium Processing System (Sheet 2 of 3)

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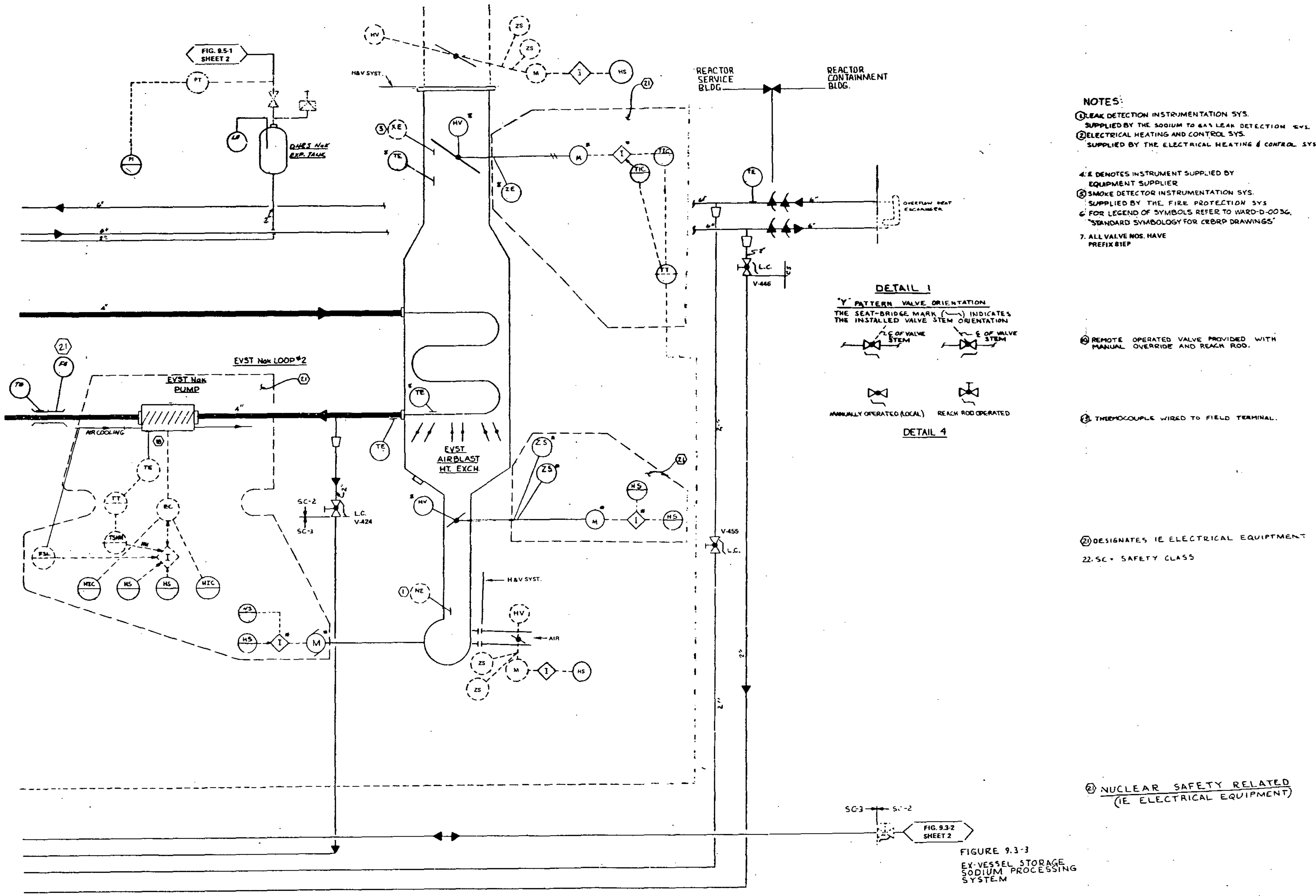

FIGURE 9.3-3 Ex-Vessel Storage Sodium Processing System (Sheet 3 of 3)

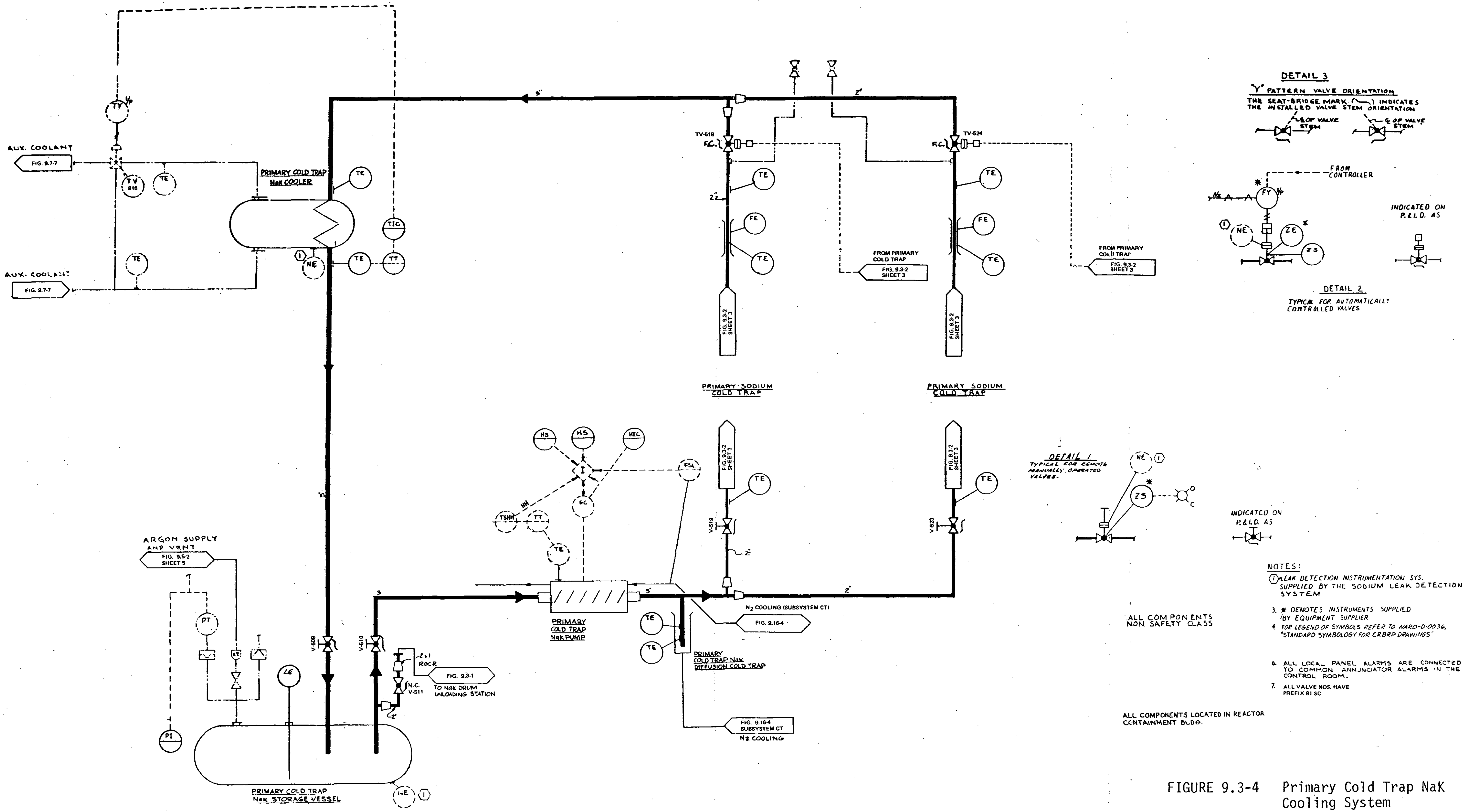

FIGURE 9.3-4 Primary Cold Trap NaK Cooling System

N099811025-10

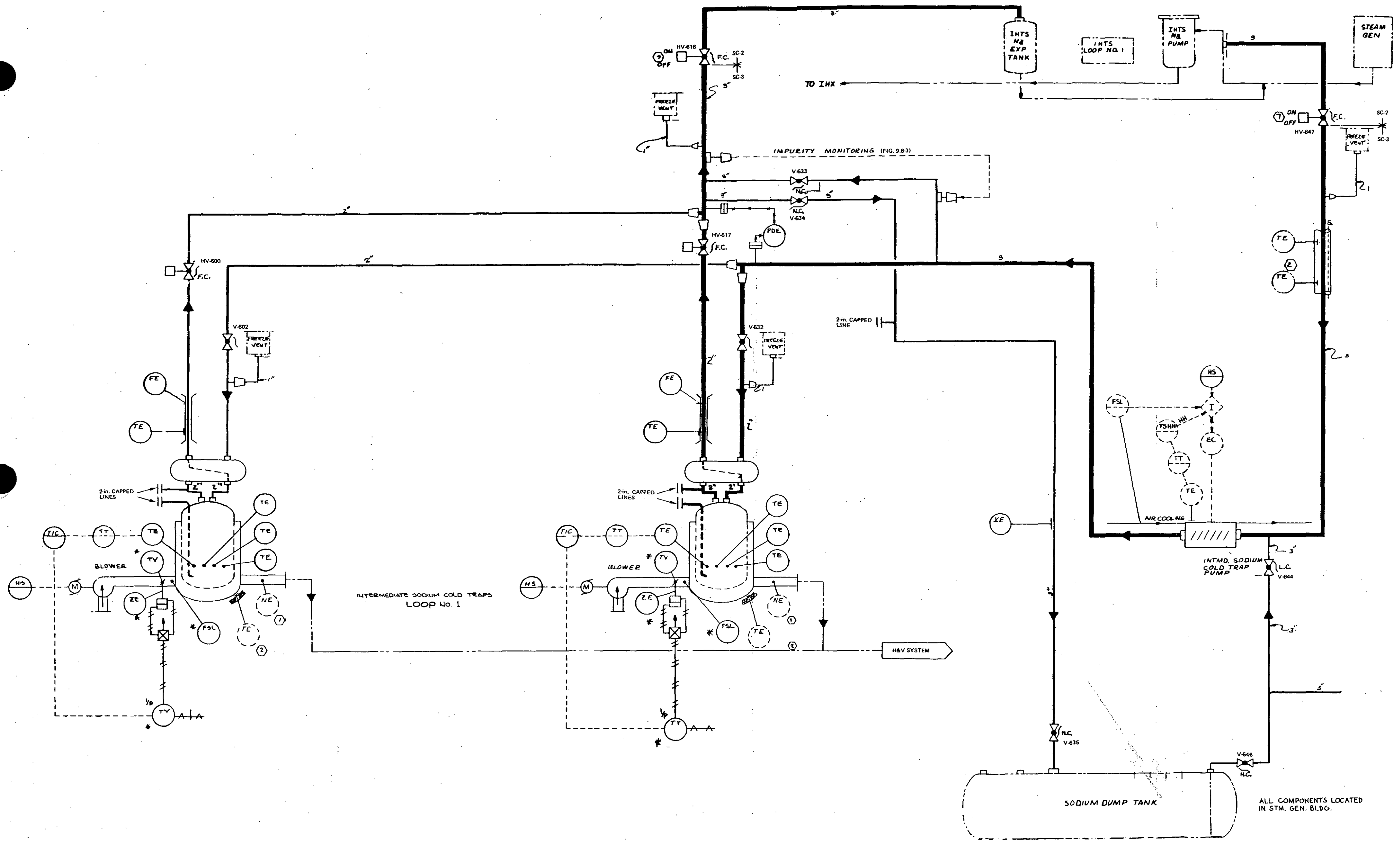

FIGURE 9.3-5 Intermediate Sodium Processing System (Sheet 1 of 3)

N099811026-11

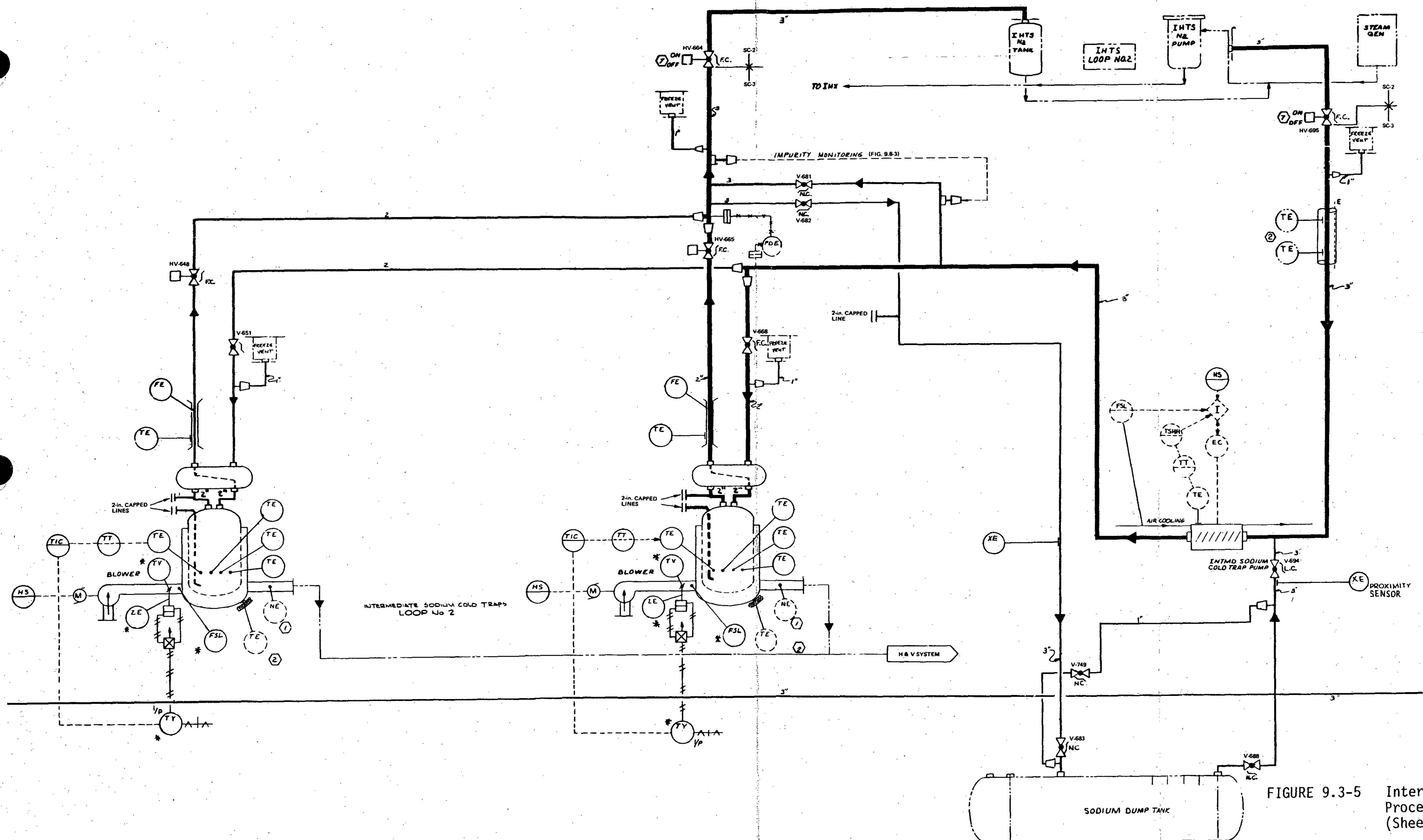

FIGURE 9.3-5 Intermediate Sodium Processing System (Sheet 2 of 3)

N099811026-11

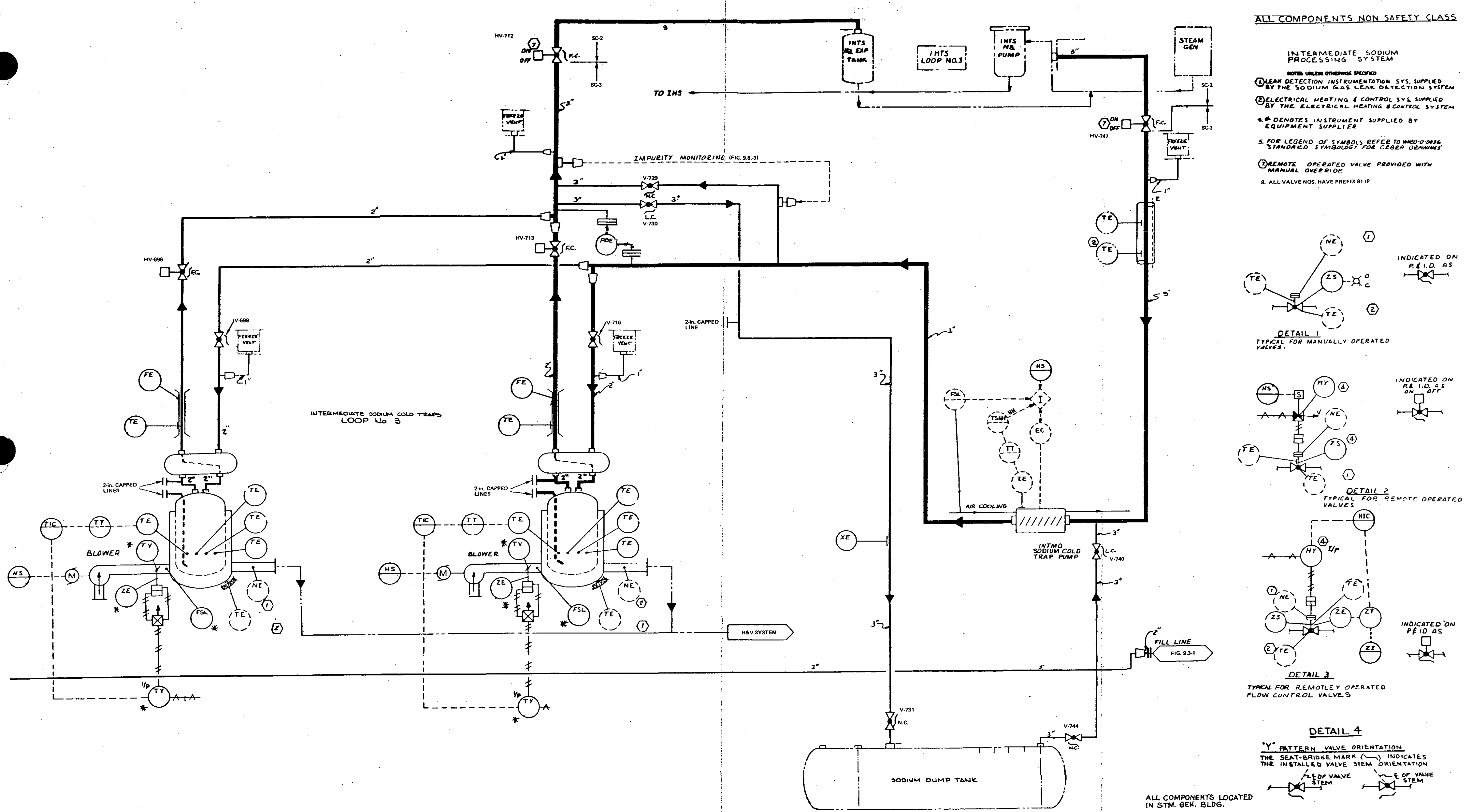

FIGURE 9.3-5 Intermediate Sodium Processing System (Sheet 3 of 3)

9.4 PIPING AND EQUIPMENT ELECTRICAL HEATING

9.4.1 Design Basis

The piping and Equipment Electrical Heating System provides the electrical heaters, electrical heater mounting hardware, heater power controllers and the related temperature measuring and controlling instrumentation and equipment required to heat the following sodium containing process systems and components:

Reactor Enclosure*
Reactor Refueling (Storage Tank)
Reactor Heat Transport (Primary and Intermediate) Systems
Steam Generation System (Dump Tanks and Sodium Water Reaction Product Tanks)
Auxiliary Liquid Metal System
Inert Gas Receiving and Processing System
Sodium Impurity Monitoring System

This heat is required to preheat these sodium process systems prior to fill, to prevent sodium freezing when system heat sources such as reactor decay heat and pumping heat become insufficient, and to maintain pre-established temperature differences in the system.

To perform the dry heat-up function, the Electrical Heating System shall be capable of preheating the sodium process systems from ambient temperature to any temperature between ambient and a maximum of approximately 450°F before the system is filled with sodium. The heating requirements for each trace heated component in the above systems will be determined by the particular sodium process system.

The Electrical Heating System shall also be capable of providing the applicable heatup rate for the particular system or components when filled with sodium and of holding process system temperatures when filled with sodium. Heat provided by this system can be used to melt frozen sodium in piping or components. Freezing of sodium in major systems or components is considered unlikely and is an abnormal event.

The temperatures of all measured points shall be indicated locally and in the Main Control Room. The thermocouple used for monitoring shall be separate from the thermocouple used for control. Thermocouples leads as a matter of good design practice shall use different paths from heater power conductors. The electrical heating, temperature monitoring, and temperature control lines are all non-safety-related. As such, the electrical heating, monitoring, and control lines shall be separated from all Class 1E lines per IEEE Std. 384-1981.

All piping and equipment in inaccessible areas shall be provided with spare heaters and thermocouples. The spares shall be accessible in a junction box in a man-safe location. All heater circuits shall be provided with ground fault protection. All instruments and controls shall be testable in place.

*The Trace Heating System which services the reactor vessel head, control rod drive lines, and vessel support area is discussed in Section 5.2.1.6.

All components of the Piping and Equipment Electrical Heating and Control System shall remain operational for an OBE (i.e., qualified Seismic Category II). In addition, no component of this system shall, as a result of a SEE, impact in any way the performance of the safety function of the piping and equipment heated by this system. The heaters and thermocouples shall withstand the vibration forces of the components to which they are attached without failure or impact on any safety-related function.

The heater physical mounting arrangement and the electrical protection of the heater circuitry shall be designed to preclude damage to the components being heated. The Piping and Equipment Electrical Heating and Control System has been designed and applied such that it is not a system which is important to safety. Safety-related components which require the application of heat from the Piping and Equipment Electrical Heating and Control System have also been designated such that any combination of failures, single or gross, of the non-safety-related Trace Heating System will not compromise the safety-related function of the equipment (e.g., cover gas equalization lines, overflow heat exchangers, ex-vessel storage tank). The location of thermocouples and heaters on components and piping shall be such that leak detection, loose parts monitoring or operability of in-service equipment is not impaired or prevented.

9.4.2 System Description

The Electrical Heating and Control System provides power to the tubular heaters or mineral insulated (MI) heating cable mounted on the piping and/or components of the systems indicated in Section 9.4.1.

The heat rates required by different components are controlled by using thermocouples to monitor piping and component temperatures and to adjust the power supplied to the heaters by means of three mode proportional temperature controllers and solid state relays.

Alarm is provided to the Main Control Room in the event of the following:

1) Setpoint Deviation (Control thermocouple compared to temperature setpoint)

2) High Temperature (Monitor thermocouple compared to a high or low temperature setpoint)

3) Low Temperature

4) Open Thermocouple (Both control and monitor thermocouples)

5) Open Circuit Breakers

6) Load Loss over 10% (Detects a single heater failure)

7) Data Transmission Failure

Tubular heaters apply heat via a spiral wound nickel-chromium alloy resistance wire insulated from its containing metal tubular sheath by tightly packed magnesia (MgO) powder. Several inches on each end of each heater are unheated having a heavy electrical conductor to the electrical termination. In certain cases, i.e., selected piping smaller than eight inches O.D., the heat is applied by mineral insulted heating cable that consist of a metal sheath drawn down over a M_GO insulated single heating element.

Separate chromel-alumel thermocouples are used throughout the systems for the feedback signal to control the operation of the electric heaters and for monitoring the temperature of the metal boundary of the sodium containing piping and equipment. Furthermore, separate signal processing/indication is provided for in the control and monitoring thermocouples. Thermocouple compensation is provided for all thermocouples.

Thermocouples on piping are located at a point on the pipe to enable control of the average temperature of the pipe within specified limits. On equipment, the thermocouples are located in the spaces between heaters for both monitoring and control purposes. The placement of thermocouples, combined with heat transfer characteristics, precludes undetected cold spots.

Control of any heater or bank of heaters is by automatic control. This control provides for continuous and automatic adjustment of heat based on an error signal generated from the difference between the temperature setpoint, as set by the plant operator, and the temperature feedback signal from the thermocouple.

The controller compares the temperature control setting (ramp rate in heat-up mode and setpoint in hold mode) as set by the plant operator to the actual temperature of the sodium process metal as measured by a chromel-alumel thermocouple and generates an error signal. The error signal is converted into a corresponding "on" to "off" ratio of voltage which is applied to a solid state relay which controls the AC power to the heaters.

The required power is controlled by conducting a fraction of the time over a 17 second period. For example, 50 percent power would be conducting for 11.5 seconds and off 11.5 seconds, 90 percent power would be on for 15.3 seconds, and 10 percent power would be conducting for 1.7 seconds.

Heaters are arranged in a particular control circuit according to the uniformity of heating required by a bank of heaters. This type of heat application is called zoning. A heater zone is an area that can be heated with the same unit heat input and can be controlled from a single temperature indicating point that is representative of the zone.

The temperature feedback thermocouple is located in a representative position within the pipe run or area within the heated zone. The monitoring thermocouples are located in different areas of the zone from the feedback thermocouple to provide independent checks on the zone temperature.

All heaters are in operation continuously during dry heat-up, (system completely empty). Some heaters will be in operation continuously for the occasional fill and drain situations in some piping and components such as cold traps, dump tanks, gas equalization lines and other components. For all other normal operations (start-up, hot standby and shut-down) the heaters will be in operation only intermittently to make up for the heat loss through the insulation.

A dedicated, pre-programmed direct digital control system is provided, for the Reactor Containment Building, Steam-Generator Building, and Reactor Service Building. The system is modular, to permit physical distribution of the various functional components and to facilitate future expansion and The upgrading.

The panels and Operator Control Centers are located in these three areas of the plant. An additional Operator Control Center (Master) is located in the Main Control Room. The system arrangement is shown in Figure 9.4-1.

9.4.3 Safety Evaluation

Since the Trace Heating System is not important to safety, the heating system components are non-Class 1E. Therefore, as required by Federal Regulations, the Trace Heating System design features which make operation highly reliable are excluded from safe shutdown scenarios. In fact, the safe shutdown scenarios must assume gross combinations of heating system failures. This is required without consideration to the design reliability of the non-safety electrical trace heating components. Under these extreme conditions, the apparent failure mechanisms of the Trace Heating and Control System are lack of heat when heat is required, heat when none is required, and current flow through piping and other non-wiring components due to shorts concurrent with multiple failures of the over current protection components. The effects of these potential failures on the safe shutdown of the plant is discussed later in this section.

As discussed in PSAR Section 3.2.3, the Piping and Equipment Electrical Heating and Control System is not safety-related. The heating system is not essential for the safe shutdown of the reactor, nor will failure of the system result in a release of radioactive material. However, considerations for trace heating of selected safety-related components have been taken.

A technical specification will be provided to address failures of trace heating in the primary equalization line, the IHTS cover gas equalization line, the argon lines leading to the primary argon relief system, and the overflow heat exchanger. Relief protection for the IHTS is provided by the rupture discs to either the dump tank or the Sodium Water Reactions Products tank. The large eighteen-inch lines downstream of the rupture discs are normally empty and are not trace heated. The eighteen-inch line up to the rupture discs (approximately five foot long) has trace heating installed, but during normal operation the heat transfer from the flowing IHTS sodium is sufficient to maintain the temperature in the line above the heater setpoints. The six-inch gas line between the IHTS expansion tank, with its associated rupture discs, is trace heated to reduce sodium frosting. Should this six-inch line become plugged due to trace heater failure, the effect of a sodium water reaction would be neutralized by blow down of the water side of the SG modules and, if needed by rupture of the large rupture discs in the eighteen-inch line. Thus, failure of the trace heating of the six-inch line will not cause a safety problem.

The EVST is cooled by three cooling circuits, two redundant "normal" forced circulation circuits, and a backup natural circulation circuit. Each of the circuits contains a sodium loop, circulating EVST sodium, and a NaK loop, which transfer the heat to the atmosphere. All of the sodium and NaK loops are electrically preheated. The preheat is segmented into multiple control zones, each zone having at least one control thermocouple. Preheat temperatures higher or lower than the setpoint band are alarmed. The preheaters are not redundant and the preheat system is not on emergency power.

During EVST cooling operation, the forced circulation circuit in operation does not require electrical preheat; the heat is provided by the fuel within the EVST. The natural circulation loop also does not require electrical preheat, as long as a heat source is present in the EVST, since a small flow of sodium and NaK is always maintained in this loop. The standby forced circulation loop does require preheat to maintain temperature. Should a heater circuit, or circuits, fail in the standby cooling circuit (or in any circuit) the cooling circuit can be isolated and the heater repaired while still maintaining two means of EVST cooling. Consequently, failure of individual heater circuits is not considered a significant safety problem. In the event of loss of plant power or a total loss of the preheat system, the normally operating forced cooling circuit and the natural circulation circuit would be unaffected from a standpoint of EVST cooling. In the standby circuit, NaK temperature can be maintained by pump heat (pump on emergency power); only the stagnant sodium loop is susceptible to decreasing temperature (and ultimately freezing) and this situation could be monitored by the safety-related thermocouples on the EVST sodium outlet lines (post-accident monitors). In this situation, plant procedures call for periodically switching EVST cooling from one circuit to another in order to use EVST heat to maintain sodium temperature well above freezing. In any event, loss of power or preheat can cause loss of no more than one of three available cooling circuits, if no operator action is taken, and will result in no loss of cooling circuits if EVST cooling is alternated between circuits. Consequently, neither redundant heater circuits nor safety-related heating is considered required to ensure continued EVST cooling.

The DHRS circuit consists of the normally operating reactor overflow/makeup circuit, the normally stagnant OHX and NaK "crossover" piping, and the EVST NaK forced circulation loops previously discussed. Redundant heaters are currently used on the overflow/makeup circuit and OHX, and a change is being made to add redundancy to the NaK "crossover" piping. Failure of heater circuits on any of these components (including the standby EVST NaK circuit) will be signaled by an alarm. Where redundant heaters are installed, the redundant heater can be connected and energized from accessible areas (outside the inerted cells) within an hours, well within the time when cooldown of an isolated circuit could have significant impact. As previously noted, the preheat on the EVST NaK loop can either be repaired or temperatures maintained by pump heat, at the operators option. Consequently, loss of individual preheat circuits should impact neither reactor operation nor capability to perform the DHRS function. If the <u>entire</u> preheat system (or substantial portions of it) is lost due to some electrical or mechanical failure, then the reactor should be shut down. With the reactor shut down and primary sodium temperature reduced, the DHRS function is not compromised even without preheat, as discussed below.

An evaluation of DHRS capability following loss of plant power was made to ensure that DHRS is not compromised during the plant event. Upon loss of power, the reactor is tripped and sodium temperature automatically reduced to 600°F. Both EVST NaK flow and overflow/makeup sodium flow are continued after a short interruption since these pumps are on emergency power. The stagnant OHX and NaK "crossover" piping slowly begin to cool since the electrical preheat is out of service without normal plant power. This cooling will increase the startup thermal transient stresses, which the preheating is designed to minimize, should DHRS be invoke 1/2 hour or more after scram. The increase in these stresses in the controlling components involved in DHRS (e.g., overflow heat exchanger) was determined to have minor impact on their structural integrity. The worst case thermal condition in the DHRS startup time from 1/2 to 24 hours after scram was analyzed to find that these components are designed to withstand many cycles (>100) of this event. The conclusion is that the additional thermal stresses due to cooldown in this event are not controlling and DHRS can be successfully initiated and carried out. If the loss of power is long enough, ultimately the OHX will freeze; however, calculations indicate that this will take approximately 48 hours, and extremely unlikely situation. If the loss of power does not appear to be extensive, for example, if not restored within 24 hours, then at that time the reactor sodium could be reduced to refueling temperature (400°F) and the makeup sodium stream diverted such that it flows through the OHX. Under these circumstances, freezing would be precluded and DHRS could be initiated at any time, indefinitely, after power loss.

A number of sodium valves are active components, used either for EVST cooling or DHRS initiation. The valves are preheated by separate preheat circuits. The preheat system is not safety-related and is not on 1E power. Safety-related preheat is not considered necessary for these valves, for the same reasons discussed above, since these valves are part of the circuits involved in that discussion. The redundancy noted also applies to the valve heater circuits.

Considering the above capability, a safety-related, 1E-powered preheat system is not considered necessary.

The unwanted additional heating of sodium lines (sensed and controlled normally) due to multiple failures in the Trace Heating and Control System which on trace heaters (which should be off) is less than five percent of the long term subsystem heat removal capability per loop. The unwanted heat is less than one percent of the short term subsystem heat removal capability per loop, and it is less than one-half of one percent of the total plant power removal capability. These percentages are sufficiently small in terms of heat transport system capability that the occurrence of this failure mechanism would not compromise the safe shutdown function.

The third potential failure mechanism is a short to a non-wiring component occurring with concurrent failures of the ground heater sheath, the ground fault detectors, and the over current protective devices. This mechanism would not compromise the safe shutdown of the plant. The smallest pipe where a short could occur is greater than ten times the cross-sectional diameter of the electrical wiring. Therefore, for the smallest pipe, the conductivity of the electrical wiring is one-half the conductivity of the pipe, and the conductivity of the pipe is over forty times higher than the heater wire. In either a short or an arcing situation, the pipe would not fail.

Operationally, the failure mechanism requires the failure of the temperature sensing system and one of the following: (1) excess current application, (2) cross-over in mounting of adjacent heaters, and (3) improper setting or protective devices. For design-related failures, the failure mechanism can be caused by improper heating wire design, fissures in the magnesium oxide, and bends less than the minimum bend radii. The effect of the failure will not cause failure of the sodium containment.

9.4.4 Design Reliability Evaluation

In order to prevent the effects of heater failure from propagating to the piping or equipment to which heaters are attached, the following operational criteria are used:

(1) For normal operation, the heaters are operated at less than 1/2 rated power. For abnormal operation, each heater control circuit is protected against overcurrent by thermal overload circuit breaker and temperature sensors on the heated component. Ground fault interrupters (GFI) will be used for protection against ground currents.

(2) High and low temperature alarms are provided for all control and monitor thermocouples in all heater control zones.

(3) The cold ends of the heaters are bent 90° and brought out from the component. A spacing is maintained between adjacent heaters to prevent crossover of heaters and significant mutual heating by radiation.

(4) The proper setting of the GFI units will be set at installation.

(5) To prevent heater failure from design considerations, the heaters are designed to a high quality standard. The use of the standard requires that heaters be radiographed. In addition, the technical, mechanical, electrical, material, fabrication, and quality assurance requirements are specified.

9.4.5 Tests and Inspections

The design of the Electrical Heating System permits periodic testing to confirm the operation of the Ground Fault Detection System and Heater Control System. The Heater Control System will be tested and inspected at installation and at refueling periods as dictated by application. Inspection of the heaters in accessible areas following shutdown will be performed according to the maintenance requirements of each process system. Redundant heaters* wired to accessible thermal blocks will be provided in inaccessible areas as required. In addition, the Trace Heating System will be tested prior to startup following an earthquake of intensity greater than or equal to the OBE.

9.4.6 Instrumentation Application

Instrumentation application is discussed in Section 9.4.2.

*Redundant heaters are nonoperating installed spares which can be made to operate in place of failed heaters.

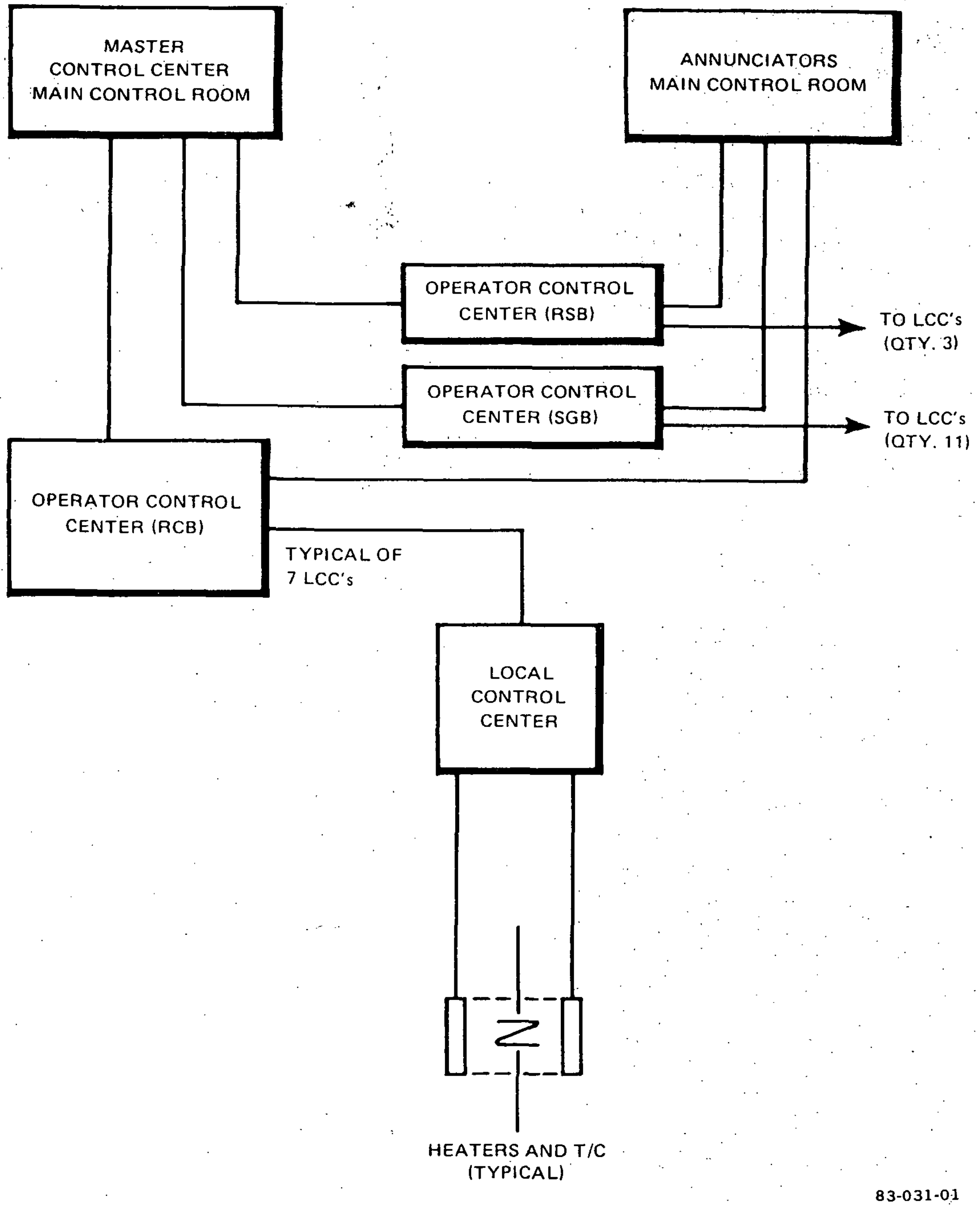

Figure 9.4-1 POWER AND CONTROL SYSTEM ARRANGEMENT