Offshore Power Systems

1st ACCEPTANCE REVIEW

Proj-463 574

3 Plant Design Report

PLANT DESIGN REPORT

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CHAPTER 4

REACTOR

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CHAPTER 4

REACTOR

4.1 SUMMARY DESCRIPTION

This section is presented in RESAR-3, Section 4.1.

RESAR-3 information for the four loop plant without loop stop valves is applicable.

4.2 MECHANICAL DESIGN

This section is presented in RESAR-3, Section 4.2

RESAR-3 information for the four-loop plant without loop stop valves is applicable.

4.3 NUCLEAR DESIGN

This section is presented in RESAR-3, Section 4.3

RESAR-3 information for the four-loop plant without loop stop valves is applicable.

4.4 THERMAL AND HYDRAULIC DESIGN

This section is presented in RESAR-3, Section 4.4

RESAR-3 information for the four-loop plant without loop stop valves is applicable.

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CHAPTER 5

REACTOR COOLANT SYSTEM

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Figures for this chapter are presented in RESAR 3, Chapter 5, except as follows:

CHAPTER 5

REACTOR COOLANT SYSTEM

5.1 SUMMARY DESCRIPTION

This section is presented in RESAR 3, Section 5.1 subject to the following:

1. The information for the four loop plant without loop stop valves is applicable.

2. Modification to information in RESAR 3, Section 5.1 is as follows:

   Figure 5.1-1, Reactor Coolant System Flow Diagram

   Residual Heat Removal System cooldown supply line No. 2 connection to the Reactor Coolant System is located in reactor coolant loop No. 1 in lieu of loop 4. In addition, the reactor coolant loop sample connection to loop 3 is relocated to loop 4. For the Floating Nuclear Plant layout, relocation of the connections results in RHR pipe length reduction. The relocation of the connections has no effect on the functional performance of the Reactor Coolant System, Residual Heat Removal System, or the Sampling System.

   Refer to Figures 5.1-1 (Reactor Coolant System Flow Diagram) and 5.5-2 (Residual Heat Removal System Flow Diagram). Instrumentation nomenclature shown on figures is in accordance with Chapter 7 of this report.

5.2 INTEGRITY OF THE REACTOR COOLANT SYSTEM BOUNDARY

Sections 5.2.1, 5.2.2 and 5.2.3 are presented in RESAR 3, subject to the following:

The information for the four loop plant without loop stop valves is applicable.

5.2.4 REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION SYSTEMS

5.2.4.1 General

The In-Containment Coolant Leak Detection System (CLD) provides the capability to detect the presence of significant radioactive or nonradioactive leakage of reactor coolant to the containment atmosphere during normal operation. Variations in the particulate activity, gaseous activity and specific humidity of the containment atmosphere above a preset level are indicated in the control room.

A leakage estimate is then made from either the variation during the transients or the new steady state. These leakage detection provisions are sufficiently sensitive so that small increases in leakage rates can be detected while the total leakage rate is still below a value consistent with safe operation of the plant.

The particulate and gaseous activity are monitored by the containment air particulate and radiogas monitors. The specific humidity is monitored by the condensate measuring system and the dew point temperature monitoring system.

Positive indications in the control room of leakage of coolant from the Reactor Coolant System to the containment are provided by equipment which permits continuous monitoring of containment air activity and humidity. This equipment provides identification of the normal background level which is indicative of a normal level of leakage from primary systems and components. Any increase in the observed parameters is an indication of change within the containment environmental conditions including air particulate activity, radioactive gas activity, and specific humidity.

The filter paper has a 25-day minimum supply at normal speed. The filter paper feed mechanism and electromagnetic assembly which controls the filter paper movement are provided as an integral part of the detector unit.

The detector assembly is in a completely closed housing. The detector output is amplified and transmitted to the radiation monitoring system cabinet in the control room. Lead shielding is provided to reduce the background radiation level to a level which does not interfere with the detector's sensitivity.

Activity is indicated and recorded in the control room. In addition, high-activity alarm indications are displayed in the control room. Local indicators and alarms provide operational status of supporting equipment such as pumps, motors and flow and pressure controllers.

The sensitivity of the air particulate monitor to an increase in reactor coolant leak rate is dependent upon the magnitude of the normal baseline leakage into the containment. Where containment air particulate activity is below the threshold of detectability, operation of the monitor with stationary filter would increase leak sensitivity to a few cubic centimeters of liquid per minute.

5.2.4.2 System Description

The In-Containment Coolant Leak Detection System consists of the air particulate monitoring subsystem, radiogas monitoring subsystem, condensate measuring subsystem, and the dew point temperature monitoring subsystem.

Portions of the system fluid piping and valves which penetrate the containment and serve as containment isolation are designed to Safety Class 2 requirements. The remainder of the system is classified non-nuclear safety.

5.2.4.2.1 Containment Air Particulate Monitor

This monitor takes continuous flow air samples from the containment atmosphere and measures the air particulate gamma radioactivity. The samples are drawn outside the containment in a sealed system and are monitored by a scintillation counter-filter paper detection assembly. The filter paper collects 99 percent of the particulate matter greater than 1.0 micron in size on its constantly moving surface, which is viewed by a hermetically sealed combination scintillation crystal (NaI) - photomultiplier. The monitor has a measuring range of 10^{-9} to 10^{-6} $\mu c/cc$.

For cases where the baseline reactor coolant leakage falls within the detectable limits of the air particulate monitor, the instrument can be adjusted to alarm on leakage increases from two to five times the baseline value.

5.2.4.2.2 Containment Radioactive Gas Monitor

This monitor measures the gaseous beta and gamma radioactivity in the containment by taking continuous air samples from the containment atmosphere. The samples first pass through the air particulate monitors where particulate matter is removed, and then pass through a sealed system to a gas monitor assembly. After passing through the gas monitor the gas sample is returned to the containment atmosphere.

Each sample is continuously homogenized in fixed, shielded volumes, where it is viewed by Gieger-Mueller tubes. This monitor has a measuring range of 10^{-6} to 10^{-3} $\mu c/cc$.

The detector is in a completely enclosed housing containing a gamma sensitive Geiger-Mueller tube mounted in a constant gas volume container. Lead shielding is provided to reduce the background radiation level to a point which does not interfere with the detector's sensitivity. A preamplifier and impedance matching circuit is mounted at the detector. The detector outputs are transmitted to the radiation monitoring system cabinets in the control room. The activity is indicated by meters and recorded by a multipoint recorder. High-activity alarm indications are displayed on the control board annunciator in addition to the radiation monitoring system cabinets. Local indicators and alarms provide operational status of the equipment.

The air particulate and radiogas monitors have a common pump unit.

The unit consists of:

1. A pump to obtain the air sample.

2. A flowmeter to indicate the flow rate.

3. A flow control valve to provide flow adjustment.

4. A flow alarm assembly to provide low and high flow alarm signals.

5.2.4.2.3 Specific Humidity Monitoring Devices

The containment specific humidity monitoring devices offer further means of detecting leakage into the containment. These devices (the condensate measuring system and the dew point monitor) are not as sensitive as the air particulate and the radiogas monitors, but have the advantage of being sensitive to vapor originating from all sources; the Reactor Coolant System, Main Steam System, Feedwater System, and other auxiliary fluid systems located inside the containment. Thus, these devices are able to detect leakage from non-radioactive sources, as well as the Reactor Coolant System during the period of initial plant operation when the coolant activities are expected to be low.

5.2.4.2.4 Condensate Measuring System

The condensate measuring system permits measurements of the flow rate of liquid run-off from the cooling coils of the containment lower compartment fan cooler units. The system consists of a vertical standpipe, valves and instrumentation installed in the drain piping of the fan cooler units.

Depending on the number of low compartment fan cooler units in operation, the drainage flow rate from each unit due to normal condensation is calculated. With the initiation of an additional or abnormal leak, the containment atmosphere humidity and condensation run-off rate will both begin to increase and the high condensate flow alarm will be actuated.

The containment specific humidity will increase proportionally to time and leakage until the dew point is reached at the fan cooler units cooling coils.

With the increasing specific humidity, the heat removal capacity needed to cool the air-steam mixture to its dew point temperature increases. Therefore, increases in specific humidity and available heat removal capacity from the cooling coils will result in added condensate flow. The condensate flow rate then is a function of specific humidity. Through accurate measurements of condensate flow variation a reliable estimate of leakage rate can be made.

A preliminary estimate of the leakage can be obtained from the condensate flow increase rate during the transient. A better estimate can be determined from the steady state condensate flow when equilibrium has been reached. The device will alarm on a condensate flow rate corresponding to a leak of 1 gpm or larger.

5.2.4.2.5 Dew Point Temperature Monitoring

The dew point measuring system consists of six dewcell elements. One element is located at each of the lower compartment fan cooler units inlet and one each in the upper and lower compartments of the containment. The basis of operation of these elements is the behavior of a hygroscopic solid.

When dry lithium chloride is exposed to the atmosphere under average room conditions, it will absorb moisture and dissolve, forming a salt solution. If this solution is heated, the water tends to escape back to the atmosphere.

A state of equilibrium is reached at a temperature when the tendency of water to escape is equal to the tendency of the salt to absorb moisture. At this equilibrium point, the temperature of the salt and the saturated solution (temperature of the dewcell element) is a measure of the partial pressure of water vapor in the surrounding atmosphere, i.e., dew point temperature. The range of the dew point temperature measuring system is 0 to 100°F. Its accuracy is ± 1°F.

Quantitive information about leakage can be obtained from dew point temperature measurements by considering that every value of leakage corresponds to a maximum value of specific humidity in the containment (steady state value). A plot of the steady state dew point temperature as a function of the leakage flow, gives for every value of dew point temperature in the containment, a minimum value of the leakage likely to be developing.

Due to the slow response of containment atmosphere specific humidity for an abnormal increase in leakage, dew point temperature recordings (specific humidity) may prove useful in establishing the location and history of the leak.

5.2.4.2.6 Gross Leakage

An accident involving gross leakage in the containment would be detected by the following additional methods:

1. <u>Charging Pump Operation</u>

   During normal operation only one charging pump is operating. If a gross loss of reactor coolant occurred which was not detected by the methods previously described, increased flow rate of the charging pump would indicate that the leakage of reactor coolant must be sufficient to cause a decrease in pressurizer level that is within the sensitivity range of the pressurizer level instrumentation.

2. <u>Sump Levels and Pump Operation</u>

   Since a break in the Reactor Coolant System would result in reactor coolant flowing into the containment sump, gross leakage would be indicated by sump water level indication and the frequency of operation of the containment sump pumps.

3. <u>Liquid Inventory</u>

   Gross leaks might be detected by unscheduled increases in the amount of reactor coolant makeup water which is required to maintain the programmed level in the pressurizer. This is inherently a low precision measurement, since makeup water is also required for leakage from systems outside the containment.

5.2.4.3 System Operation

Under normal conditions the reactor coolant pressure boundary leakage detection system monitors and records atmospheric conditions within the containment. For conditions of abnormal leakage into the containment, operator action may be required. The following requirements have been established to specify those limiting conditions for operation of the Reactor Coolant System which must be met to ensure safe reactor operation.

5.2.4.3.1 Requirements

1. Leakage

   a. Detected or suspected leakage from the Reactor Coolant System shall be investigated and evaluated.

   b. If the leakage rate, from other than controlled leakage sources, such as the reactor coolant pump controlled leakage seals, exceeds 1 gpm and the source of the leakage is not identified within 24 hours of leak detection, the reactor shall be brought to hot shutdown. If the source of leakage is not identified within an additional 48 hours, the reactor shall be brought to a cold shutdown condition.

c. If the sources of leakage are identified and the results of the evaluations are that continued operation is safe, operation of the reactor with a total leakage, other than leakage from controlled sources, not exceeding 10 gpm shall be permitted except as specified in 4 below.

d. If it is determined that leakage exists through a non-isolable fault which has developed in a Reactor Coolant System component body, pipe wall, vessel wall, or pipe weld, the reactor shall be brought to a cold shutdown condition and corrective action taken prior to resumption of unit operation.

e. If the total leakage, other than leakage from controlled sources, exceeds 10 gpm the reactor shall be placed in the cold shutdown condition. A 10 gpm leak rate is less than 10 percent of the makeup rate available from the reactor makeup control mode of operation.

A listing of operator actions to be taken in the event of the initiation of an alarm or alarms denoting abnormal leakage is presented in 5.2.4.3.2.

5.2.4.3.2 Containment Radiogas and Air Particulate Monitoring

Upon actuation of a high activity alarm the operator will follow these steps:

1. Record particulate and gaseous activity measurements at equal time intervals.

2. Determine the activity increases during these intervals.

3. Estimate the coolant leakage rate from the radioactivity increases. This is accomplished by comparing the measured increases to a curve showing abnormal reactor coolant leakage as a function of particulate and gaseous activity variation.

The estimated leakage and measured particulate and gaseous activities may then be plotted to test the consistency of the data and estimate the time elapsed since the beginning of the leak.

5.2.4.3.3 Condensate Measuring System

The initiation of an additional or abnormal leak in the containment will result in an increase in the condensate runoff rate. The additional condensation will signal a change in the system and subsequently actuate an alarm.

Upon actuation of the alarm, the operator performs the following:

1. Record condensate measurements at equal time intervals.

2. Determine the condensate flow increase during these intervals.

3. Use available curves indicating leak rate versus condensate flow rate.

5.2.4.3.4 Dew Point Temperature Monitoring

As previously stated, a curve can be generated showing minimum leakage as a function of containment atmosphere dew point temperature. Therefore, if an increase of dew point temperature is observed, the operator refers to a curve to obtain a minimum leakage rate value.

5.2.4.4 Evaluation

The above instrumentation will allow detection of leakages occurring within the containment. The location of specific leaks will in general have to be determined visually, although dew point recordings may aid in this determination. The location of any leak in the Reactor Coolant System would be detected by the presence of boric acid crystals. The leaking fluid transfers boric acid outside the RCS and the process of evaporation deposits crystals.

5.2.4.4.1 Containment Air Particulate Monitor

The containment air particulate monitor is the most sensitive instrument available for detection of reactor coolant leakage into the containment. This instrument is capable of detecting particulate activity in concentration as low as 10^{-9} to 10^{-6} μc/cc of containment air sampled. Leakage rates of about .01 gpm to leaks greater than 10 gpm can be observed for specific values of corrosion product activity and little or no leaky fuel.

5.2.4.4.2 Containment Radioactive Gas Monitor

The containment radioactive gas monitor is inherently less sensitive (threshold at 10^{-6} μc/cc) than the containment air particulate monitor and would function in the event that significant reactor coolant gaseous activity exists due to fuel cladding defects.

5.2.4.4.3 Condensate Measuring System

The condensate measuring system determines leakage losses from water and steam systems within the containment. The system collects and measures the moisture condensed from the containment atmosphere onto the cooling coils of the lower compartment fan cooler units.

5.2.4.4.4 Dew Point Temperature Monitoring

The dew point temperature monitoring provides a backup to the other devices. This instrumentation will be sensitive to incremental increases of leakage to the containment atmosphere on the order of 0.5 gpm per 1°F in dew point.

5.2.4.4.5 Allowable Leakage Rate

The 1 gpm maximum permissible leakage rate from unidentified sources within the reactor coolant pressure boundary is well below the leakage rates calculated for critical through-wall cracks in pipes 3" diameter and larger. The lengths of through-wall cracks that are calculated to leak 0.5 gpm in 2" lines, 1 gpm in 3" lines and 2 gpm in lines 4" diameter and larger are given in WCAP-7503 Revision 1, "Determination of Design Pipe Breaks for the Westinghouse Reactor Coolant System", issued February 1972. Included in this report are the ratios of critical through-wall cracks to computed lengths for these leakage valves as a function of pipe diameter and wall thickness based on the application of the principles of fracture mechanics as well as the mathematical model and data used in the analyses.

Although the 1 gpm maximum permissible unidentified leakage rate is larger than the 0.5 gpm leakage rate analyzed for cracks in 2" lines, core cooling analyses have shown that for "small breaks" that is, for breaks up to the equivalent of the cross-sectional area of a 4" diameter line, no reactor fuel clad damage occurs.

Operating experience from the Robert Emmett Ginna plant indicated that the average total leakage from the Reactor Coolant System, including the charging and letdown portion of the Chemical and Volume Control System, was about 0.5 gpm. Major sources of this leakage were the reciprocating charging pump seals, averaging about 0.2 gpm, and the valves in the pressurizer spray and spray bypass system, which averaged between 0.2 gpm and 0.5 gpm between repackings.

The system design incorporates spray valves of the bellows or packless type, which will eliminate the large valve leakages at the Ginna plant. In addition, the design incorporates a head tank on the reciprocating charging pump to ensure satisfactory operation of pump seals and minimize seal leakage.

With these improvements, the total average leakage rates are much lower than those previously experienced.

The total normal leakage from the Reactor Coolant System is expected to be about 50 lbs per day.

5.2.4.5 Tests and Inspections

Tests and inspections prescribed for the containment air particulate and radiogas monitors are presented in RESAR 3, Section 11.4.

5.2.4.5.1 Condensate Measuring and Dew Point Temperature Monitors

During the period of plant hot functional testing, a reactor coolant leak of known magnitude will be simulated inside the containment, and performance of the humidity detector and condensate measuring system will be observed. The leak will be simulated by introducing steam into one of the loop compartments during a period when containment atmospheric conditions are stable and the fan-cooler units are operating. The increase in containment atmosphere moisture content, as indicated by the humidity detectors, will be recorded as a function of time following initiation of the simulated leak. In addition, the same information will be determined independently using the condensate measuring system. Elapsed time before initiation of condensation run-off from the fan-cooler unit cooling coils will be recorded and compared with the calculated value based on the initial containment humidity. Steam flow will continue, and the performance of the condensate measuring devices in indicating the magnitude of steady cooling coil run-off will be observed.

5.2.5 INSERVICE INSPECTION PROGRAM

During the design phase, careful consideration has been given to provide access for the pre-service baseline and subsequent inservice visual and non-destructive inspections of the reactor coolant primary and associated auxiliary systems and components within the boundaries established in accordance with the requirements of IS-120 Section XI of the ASME Boiler and Pressure Vessel Code.

Specific provisions made for baseline and inservice inspection access in the design of the reactor vessel, system layout and other major primary coolant components are as follows:

1. All reactor internals are completely removable. The tools and storage space required to permit reactor internals removal for these inspections are provided.

2. The reactor vessel shell in the core area is designed with a clean, uncluttered cylindrical inside surface to permit future positioning of inspection equipment without obstructions.

3. The reactor vessel cladding is improved in finish by grinding to the extent necessary to permit meaningful examination of the vessel welds and adjacent base metal in accordance with the Code.

4. The cladding-to-base metal interface is ultrasonically examined to assure satisfactory bonding to allow volumetric inspection of the vessel welds and base metal from the vessel inside surface.

5. The reactor closure head is stored in a dry condition on the operating deck during refueling, allowing direct access for inservice inspection.

6. The insulation on the vessel closure and lower heads is removable, allowing access for the visual examination of head penetrations during inservice inspection.

7. All reactor vessel head bolts, nuts, and washers are removed to dry storage during refueling, allowing inservice inspection in parallel with refueling operations.

8. Access holes are provided in the core barrel flange allowing access for remote visual examination of the clad surface of the vessel without removal of the lower internals assembly.

9. Removable plugs are provided in the primary shield providing access for the surface and visual examination of the external primary nozzle safe-end welds.

10. Manways are provided in the steam generator channel head to provide access for internal inspection.

11. A manway is provided in the pressurizer top head to allow access for internal inspections.

12. The insulation covering all component and piping welds and adjacent base metal is designed for ease of removal and replacement in areas where external inservice inspection is planned.

13. Removable plugs are provided in the concrete structure above the reactor coolant pumps to permit removal of the pump motor to provide internal inspection access to the pump.

14. The primary loop compartments are designed to allow personnel entry during refueling operations, to permit direct inservice inspection access to the external portion of piping and components.

The Inservice Inspection Program described in this section is applicable to the pre-service baseline inspection.

5.2.5.1 Inspection of Primary Loop Components Except Reactor Vessel

The use of conventional non-destructive, direct visual and remote visual test inspection techniques can be applied to the inspection of all primary loop components except for the reactor vessel. The reactor vessel presents special problems because of the radiation levels and remote underwater accessibility to this component.

5.2.5.2 Inspection of Reactor Vessel

As indicated above, the only specialized remote inspection equipment currently required is for the inspection of the reactor vessel. The pre-service baseline inspection will be performed utilizing a remote reactor vessel ultrasonic inspection tool which will perform the code required inspection of the circumferential and longitudinal shell welds, the flange-to-vessel weld, the ligaments between the flange holes, the nozzle-to-vessel welds, and the nozzle-to-safe-end pipe welds.

5.2.5.2.1 Reactor Vessel Inspection Tool Configurations

The vessel inspection tool has two major structural components, the center column assembly which is supported and aligned from the vessel internals support flange by the main carriage and the spider assemblies. These two structures form the rigid base assembly from which all the scanning arm and drive assemblies are mounted and operated. Two configurations of the tool can be arranged depending on the areas of the vessel to be examined. In one configuration the scanning arm and transducer array necessary to examine the vessel shell welds are mounted on the main carriage assembly and can traverse the full length of the vessel up and down and also provides 360 degrees circumferential rotation around the inside of the vessel. In the second configuration, the nozzle scanning assembly is mounted beneath the carriage and can be rotated through 360 degrees to provide alignment with any of the vessel nozzles. The tool, in this configuration, can be installed in the vessel without removal of the lower internals package. The scanning attachment for the examination of the vessel flange weld and ligaments can be installed on the tool in either of the two configurations described above. Figures 5.2-1 and 5.2-2 show the inspection tool in both configurations.

5.2.5.2.2 Reactor Vessel Inspection Tool Operation

For reactor vessel scanning a data acquisition system is used which combines simplicity with operational reliability and adaptability. It employs an electronic system with a receiver or data channel for each transducer unit. The signal received from the transducer is transmitted through an electronic distance amplitude correction device and depending on the amplitude is gated through individual transigates to a printer.

Simultaneously with the gated signal, information is received from the position indication encoder system which provides position coordinates. An audio/visual alarm is also triggered. The resultant indication is printed on a recording device together with its position reference.

Following the completion of the scan of all the areas scheduled for inspection the tool operator(s) will, operating on manual control, return the transducer array to each of the indication position coordinates and fully evaluate the nature of the indication to sufficient extent to be able to determine the size, shape and orientation. Special transducer arrays with variable angle search units are provided to assist in this activity.

For manual ultrasonic scanning, such as examination of a circumferential pipe weld, a set of standard operator procedures will be utilized to ensure that all inspection results are recorded in such a manner as to avoid any ambiguity in interpretation. The procedure specifies how any indication must be recorded with respect to scan direction and weld orientation and how datum points, from which the location of all indications are measured, are to be selected.

The data from the various baseline examinations, collected in accordance with the above procedures, will be collected into a comprehensive report tabulating all the results. The report will describe the scope of the inspection, the procedures utilized, the equipment utilized and names and qualifications of the operators and the examination results including all instrument calibration criteria.

5.2.5.3 Loose Parts Monitoring Program

Programs are under development which might, in the future, prove capable of inservice detection of loose parts and/or excessive vibration. One program involves the use of out-of-core detectors for the collection of nuclear noise signals. Another program involves the study of accelerometer measurements obtained from detectors positioned on the steam generators and the reactor vessel. The accelerometers are sensitive to the range of frequencies expected for small loose parts.

The design of the plant will not preclude the addition of the above described systems, should either prove successful in its intended function. Developments in the technology of loose parts detection will be followed and consideration will be given to the adoption of programs which might prove capable of inservice detection of loose parts.

5.3 THERMAL HYDRAULIC SYSTEM DESIGN

This section is presented in RESAR 3, Section 5.3. The information for the four loop plant without loop stop valves is applicable.

5.4 REACTOR VESSEL AND APPURTENANCES

This section is presented in RESAR 3, Section 5.4. The information for the four loop plant without loop stop valves is applicable.

5.5 COMPONENT AND SUBSYSTEM DESIGN

This section is presented in RESAR 3, Section 5.5 subject to the following:

1. Information for the four loop plant without loop stop valves is applicable.

2. Modifications to RESAR 3, Section 5.5 are as follows:

   a. Subsection 5.5.7, Residual Heat Removal System (RHR)

      The Residual Heat Removal System performs a safety function during only the LOCA injection phase of emergency core cooling and does not operate during the recirculation phase. Emergency core cooling flow during post-LOCA recirculation is provided by the safety injection and high head safety injection pumps as delineated in Section 6.3. The RHR flow diagram is shown on Figure 5.5-2.

   b. Table 5.5-15, Pressurizer Valves Design Parameters

      Three pressurizer power relief valves are provided.

   c. Subsections 5.5.4, 5.5.5, 5.5.9, and 5.5.14, are presented in the following paragraphs.

5.5.4 MAIN STEAM LINE FLOW RESTRICTOR

A flow restrictor is installed in each main steam line inside the containment close to the steam generators.

5.5.4.1 Design Basis

The main steam line flow restrictor is designed to limit steam flow, thereby limiting return to power in the core in the event of a steam line rupture downstream of the flow restrictor. See RESAR, Chapter 15.

Design criteria for the flow restrictors are as follows:

1. Provide plant protection in event of a steam line rupture. In such an event, the flow restrictor reduces steam flow rate from the break, which in turn reduces the cooling rate of the Reactor Coolant System.

2. Reduce thrust forces on the main steam piping in event of a steam line rupture.

3. Provide a portion of the pressure differential necessary for steam flow measurement.

4. Minimize unrecovered pressure loss across the restrictor under normal plant operation to least possible value.

5. Withstand the number of pressure and thermal cycles experienced in the life of the plant.

6. Maintain restrictor integrity in event of double-ended severance of a main steam line.

5.5.4.2 Description

The flow restrictor (Figure 5.5-1) consists of a 16-inch diameter 304 stainless steel venturi nozzle. The flow restrictors, manufactured in accordance with ASME Code, Section III, are welded to the main steam line.

5.5.4.3 Safety Evaluation

The flow restrictor reduces the equivalent area of the main steam line at the restrictor by a factor of 4, which assures adequate flow reduction. The resultant pressure drop through the restrictor at 100 percent steam flow is 3 psi.

5.5.4.4 Tests and Inspections

The flow restrictors are not a part of the Main Steam System pressure boundary. No tests or inspections of the restrictors are anticipated beyond those performed in the fabricator's shop.

5.5.5 MAIN STEAM LINE ISOLATION SYSTEM

The main steam line isolation system is described in Section 10.3, Main Steam System and subsection 6.2.4, Containment Isolation Valves.

5.5.9 MAIN STEAM LINE AND FEEDWATER PIPING

The main steam piping is described in Section 10.3, Main Steam System. The feedwater piping is described in subsection 10.4.6, Condensate and Feedwater System.

5.5.14 COMPONENTS SUPPORTS

5.5.14.1 Design Basis

The Reactor Coolant System component supports are designed to meet the requirements of Safety Class 1.

The supports allow virtually unrestrained, lateral thermal movement of the loop during system heat up and cool down, but provide restraint for all earthquake and accident loading conditions. Support types are linear or plate and shell with integral or non-integral attachment. Material loading combinations and stress limits are provided in Section 5.2. The component supports will employ material selection, fabrication and quality assurance in accordance with the nature of the equipment being supported.

5.5.14.2 Description

The description of the supports for the major components in the Reactor Coolant System will be provided later.

5.5.14.3 Safety Evaluation

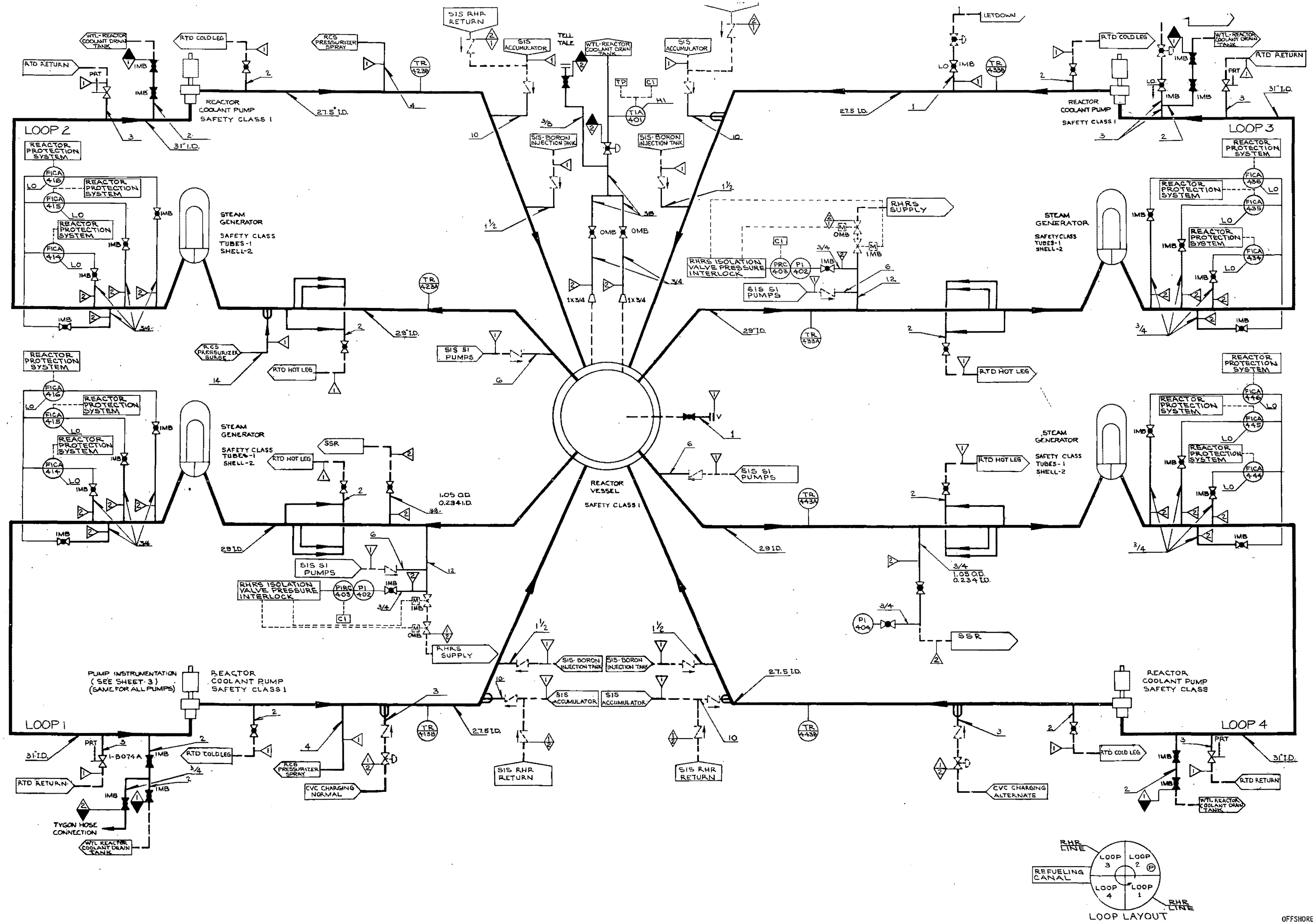
The supports permit expansion and contraction of the supported system and structurally withstand thrusts and moments of imposed loads. Shimming and grouting enable adjustment of the support elements during erection to achieve the correct fit-up and alignment. Welder qualification and procedures are in accordance with ASME Section IX.

5.5.14.4 Testing

All steel material is impact tested in accordance with the provisions of ASTM-A-370. Minimum energy test values are comparable to those specified in ASME Section III, Appendix I. Non-destructive testing is performed in accordance with ASME, Section III.

5.6 INSTRUMENT APPLICATION

This section is presented in RESAR 3, Section 5.6.

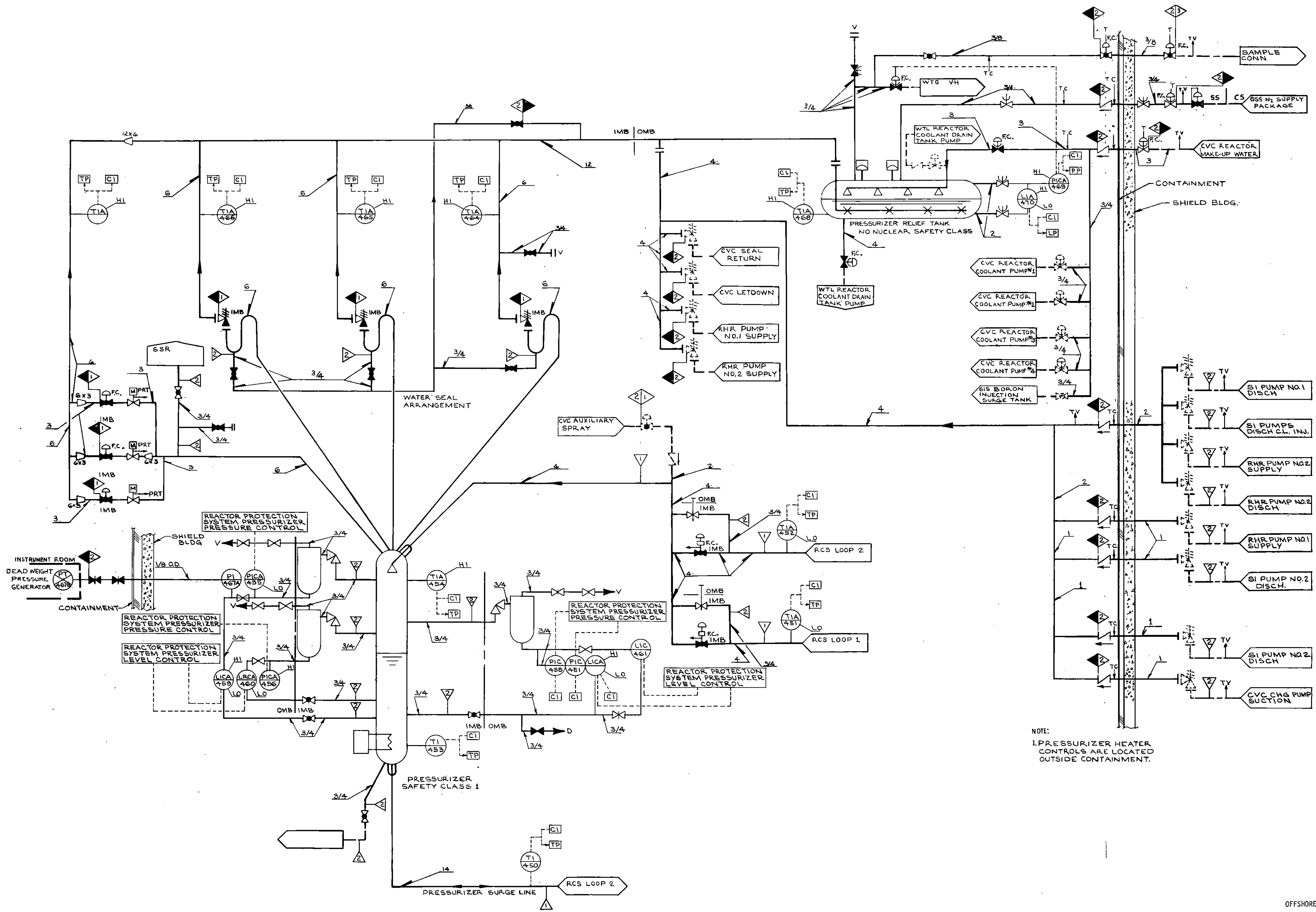


OFFSHORE POWER SYSTEMS

Title: REACTOR COOLANT SYSTEM FLOW DIAGRAM

11/1/72

Figure: 5.1-1
Sheet 1



NOTE:

1. PRESSURIZER HEATER CONTROLS ARE LOCATED OUTSIDE CONTAINMENT.

OFFSHORE POWER SYSTEMS

Title: REACTOR COOLANT SYSTEM FLOW DIAGRAM

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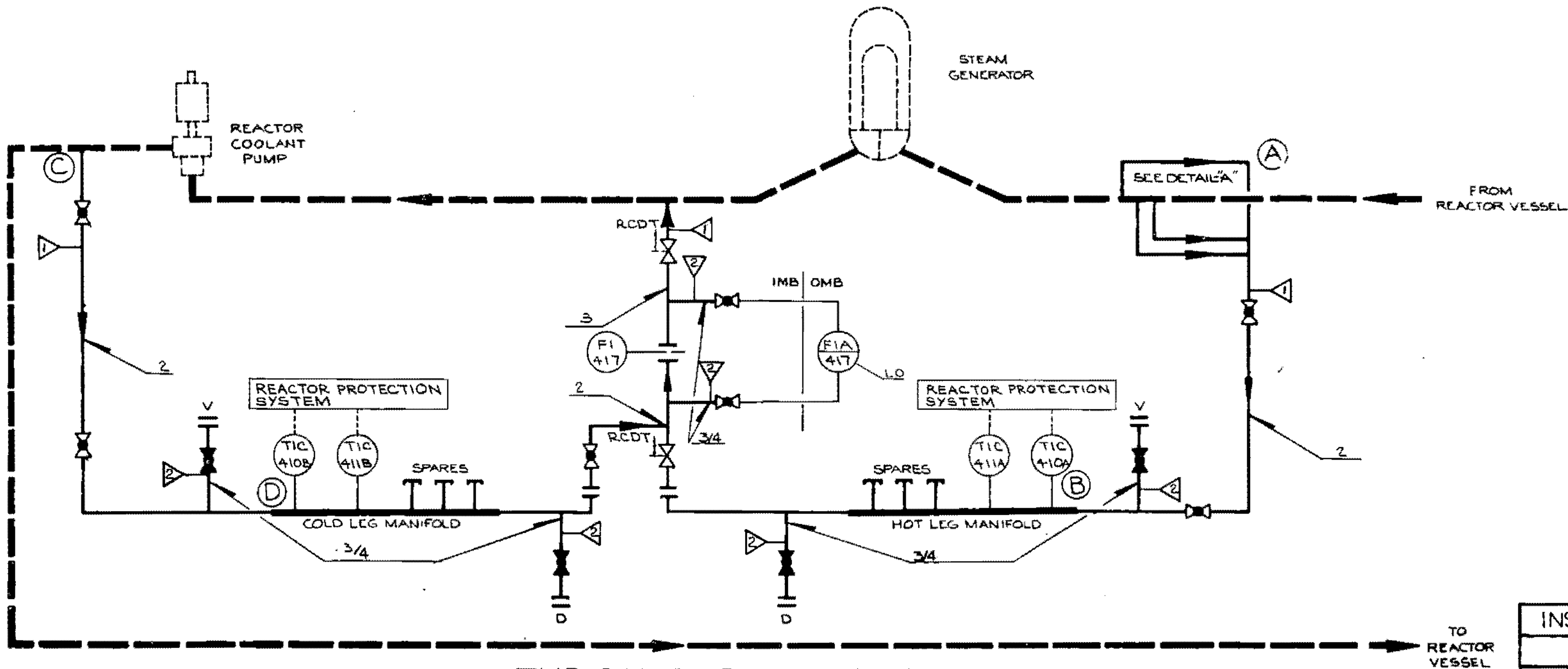
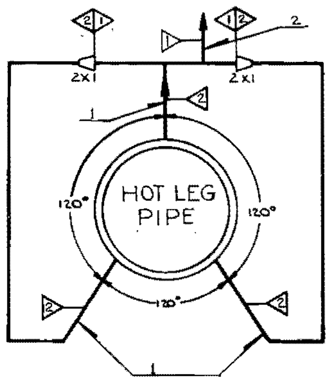
Figure: 5.1-1
Sheet 2



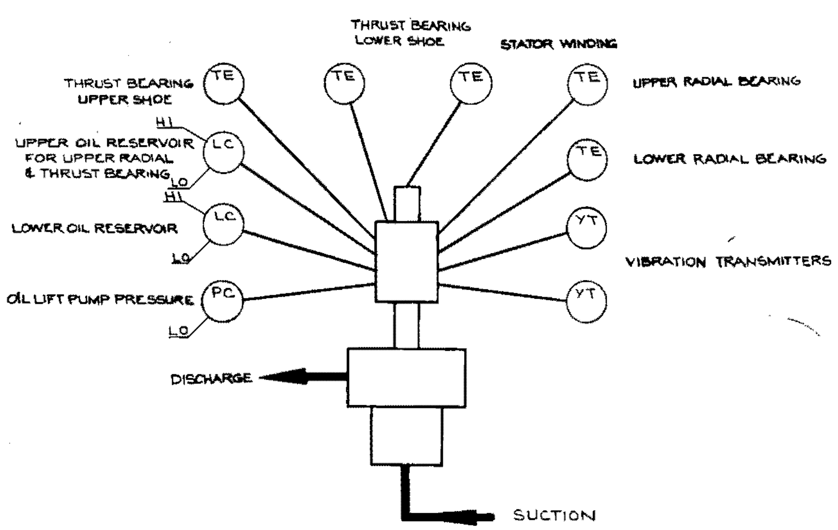
TYPICAL RTD MANIFOLD LOOP



DETAIL 'A'

RTD BYPASS INSTRUMENTATION

INSTRUMENT	LOOP 1	LOOP 2	LOOP 3	LOOP 4
HOT LEG MANIFOLD	TE-410A	TE-420A	TE-430A	TE-440A
1.TEMPERATURE ELEMENT	TE-411A	TE-421A	TE-431A	TE-441A
2.TEMPERATURE ELEMENT				
COLD LEG MANIFOLD				
1.TEMPERATURE ELEMENT	TE-410B			
2.TEMPERATURE ELEMENT	TE-411B	TE-421B	TE-431B	TE-441B
BYPASS RETURN LINE				
FLOW ELEMENT	FE-417	FE-427	FE-437	FE-447
FLOW INDICATOR & ALARM	FICA-417	FICA-427	FICA-437	FICA-447



REACTOR COOLANT PUMP MOTOR INSTRUMENTATION (SAME ON ALL PUMPS)

REACTOR COOLANT PUMP MOTOR INSTRUMENTATION

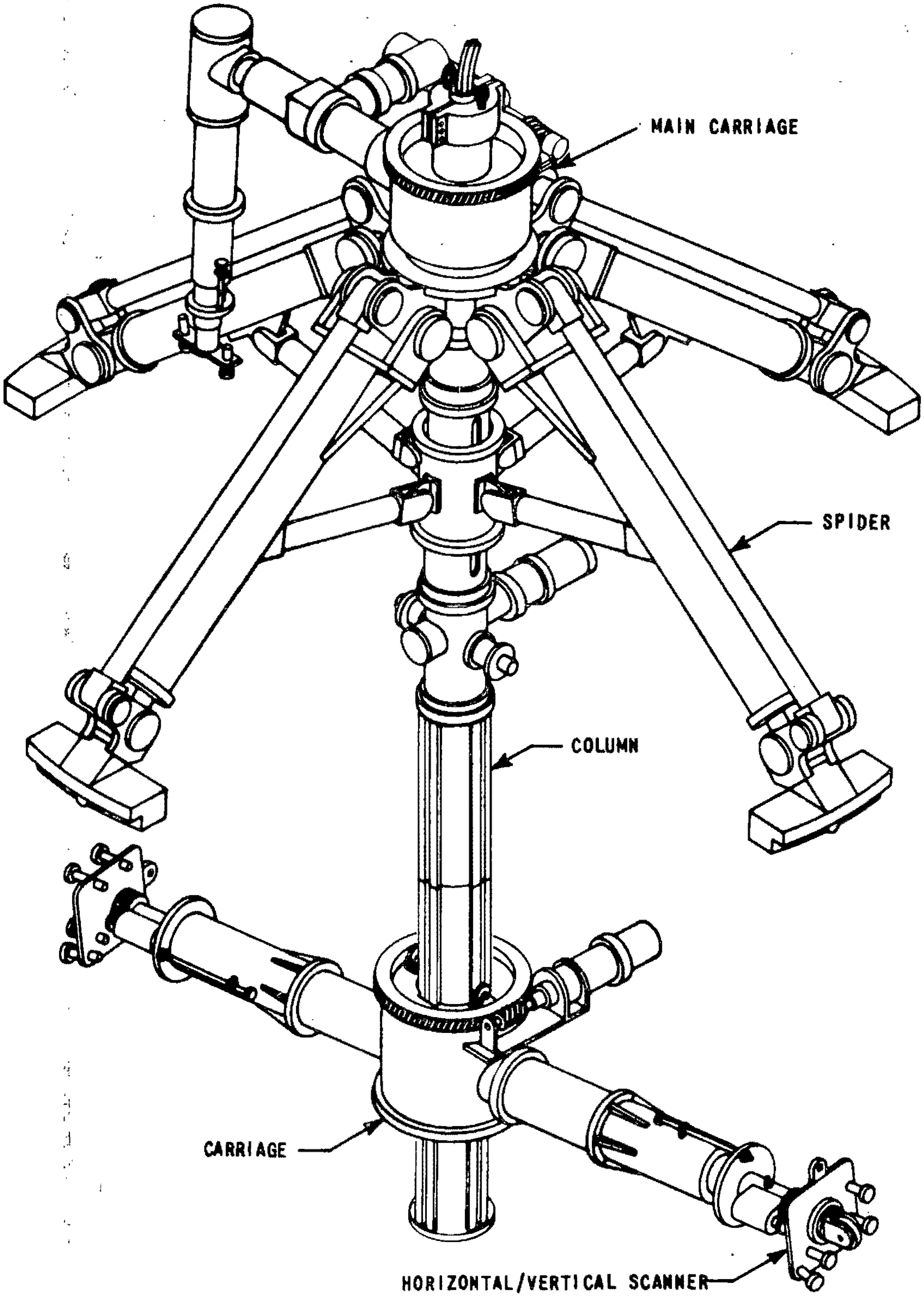
INSTRUMENT	PUMP 1	PUMP 2	PUMP 3	PUMP 4
OIL LIFT PUMP PRESSURE	PC-491	PC-492	PC-493	PC-494
LOWER OIL RESERVOIR	LC-475B	LC-476B	LC-477B	LC-478B
UPPER OIL RESERVOIR FOR UPPER RADIAL & THRUST BEARING	LC-475A	LC-476A	LC-477A	LC-478A
THRUST BEARING UPPER SHOE	TE-479A	TE-480A	TE-481A	TE-482A
THRUST BEARING LOWER SHOE	TE-479B	TE-480B	TE-481B	TE-482B
STATOR WINDING	TE-487	TE-488	TE-489	TE-490
UPPER RADIAL BEARING	TE-483A	TE-484A	TE-485A	TE-486A
LOWER RADIAL BEARING	TE-483B	TE-484B	TE-485B	TE-486B
VIBRATION TRANSMITTER	YT-471A	YT-472A	YT-473A	YT-474A
VIBRATION TRANSMITTER	YT-471B	YT-472B	YT-473B	YT-474B

OFFSHORE POWER SYSTEMS

Title: REACTOR COOLANT SYSTEM FLOW DIAGRAM

11/1/72

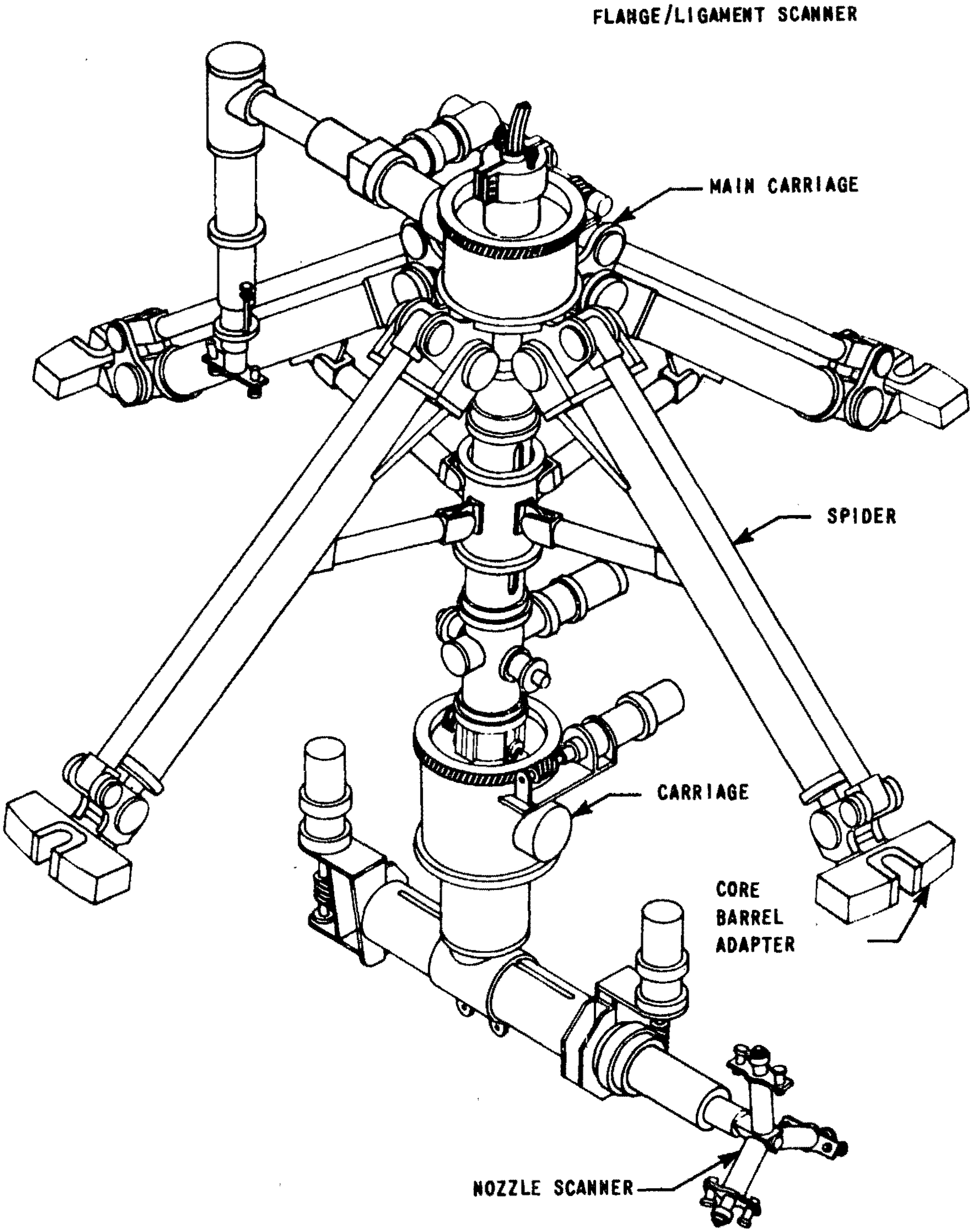
Figure: 5.1-1
Sheet 3



OFFSHORE POWER SYSTEMS

Title: REACTOR VESSEL INSPECTION TOOL WITH VESSEL SCANNER

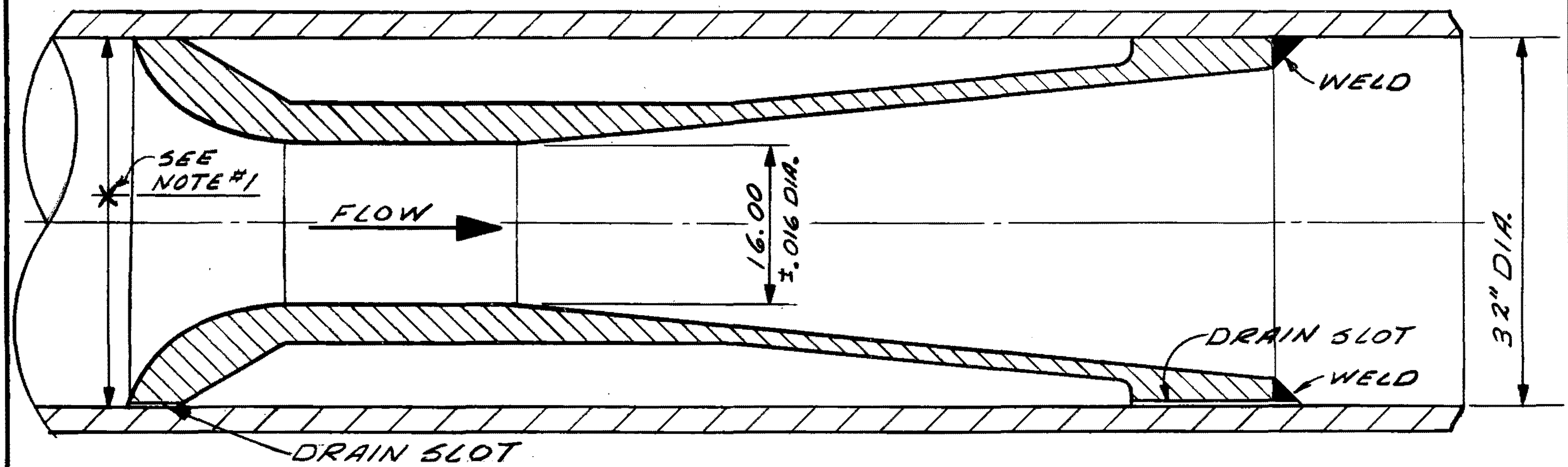
11/1/72 Figure 5.2-1



OFFSHORE POWER SYSTEMS

Title: REACTOR VESSEL INSPECTION TOOL WITH NOZZLE AND FLANGE SCANNER

11/1/72 Figure 5.2-2



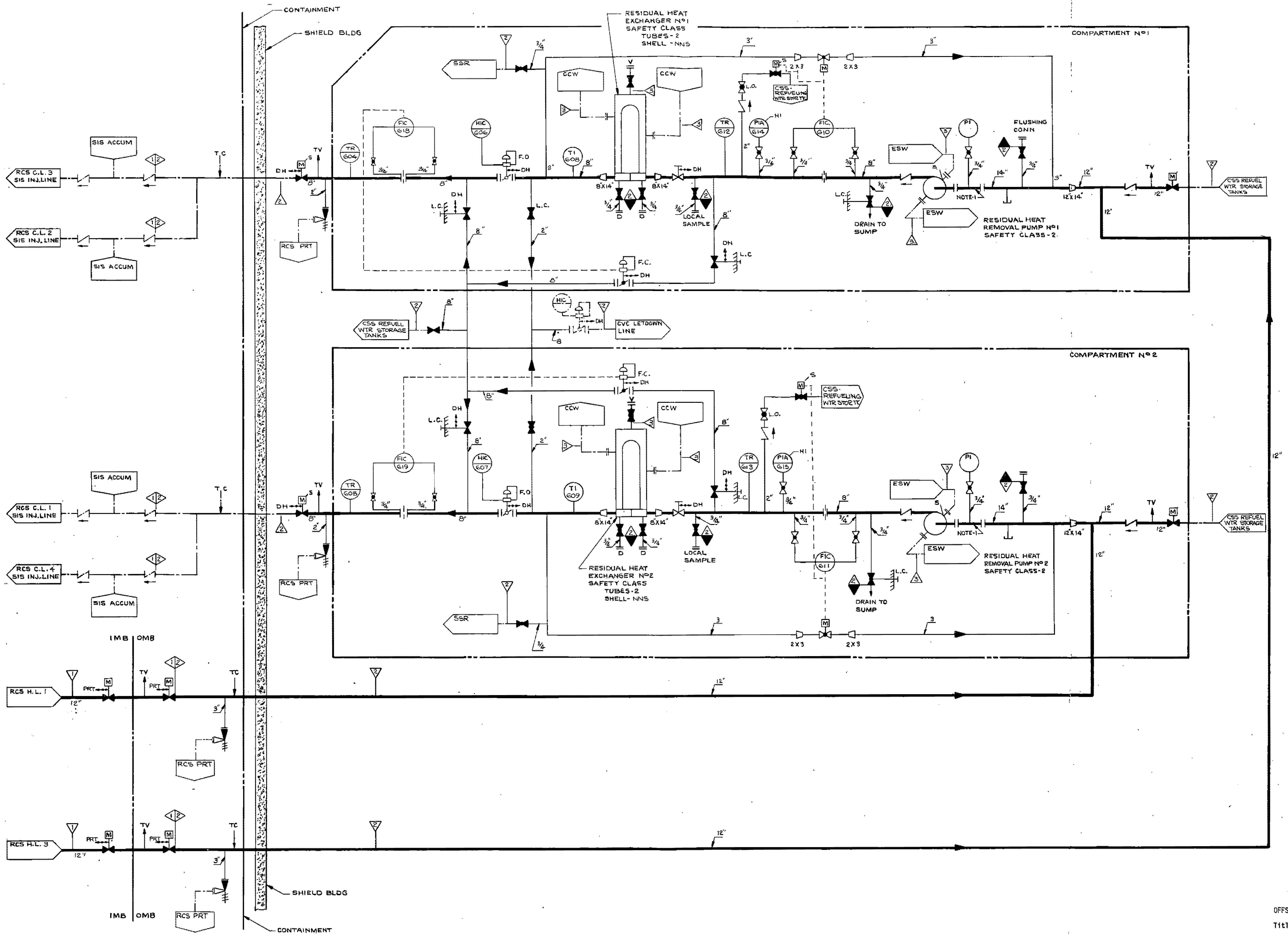
CROSS SECTION VIEW OF STEAM FLOW RESTRICTOR

1. NOZZLE O.D. MACHINED TO FIT PIPE I.D. .065 CLEARANCE MAX. TO SMALLEST DIA. IF PIPE IS OUT OF ROUND.

MAIN STEAM LINE FLOW RESTRICTOR

11-1-72

Figure 5.5-1



OFFSHORE POWER SYSTEM

Title: RESIDUAL HEAT REMOVAL SYSTEM FLOW DIAGRAM

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Figure 5.5-2

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CHAPTER 6

ENGINEERED SAFETY FEATURES

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CHAPTER 6

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CHAPTER 6

ENGINEERED SAFETY FEATURES

6.1 GENERAL

The prime safety objective in reactor design and operation is the control of reactor fission products. The methods used to assure this objective are:

1. Core design to preclude release of fission products from the fuel (Chapter 4).

2. Retention in the reactor coolant system of any fission products which may leak from the fuel (Chapter 5).

3. Retention of fission products by the containment for operational and accidental releases beyond the reactor coolant boundary (Chapter 6).

4. Analysis of fission product dispersal to assure minimum population exposure for an accidental release beyond the containment (Chapters 2 and 15).

5. Provision of adequate cooling of the core under all circumstances (Chapters 5 and 6).

The Engineered Safety Features are the provisions in the plant which embody methods 3 and 5 above to prevent the occurrence or to minimize the effects of any serious accident.

The Engineered Safety Features in this plant are:

1. A free standing steel containment with an outer reinforced concrete Shield Building and an annular space which will be maintained at a lower-than-atmospheric pressure during plant operation and following a loss-of-coolant accident. These structures form a double barrier to the escape of fission products should a loss-of-coolant accident occur. Refer to subsection 6.2.1 for details of the containment.

2. The ice condenser which prevents high pressure in the containment and thus reduces the potential for the escape of fission products from the containment. This low temperature heat sink, located on the inside of the steel containment, consists of a suitable quantity of borated ice in a cold storage compartment. Sodium tetraborate solution, generated during ice meltdown, adsorbs iodine in the containment atmosphere. Refer to details in subsection 6.2.2.

3. The Emergency Core Cooling system which provides borated water to cool the core in the event of an accidental depressurization of the Reactor Coolant System. The combination of the control rods and the boron in the injected water provide the necessary control of reactivity required. Refer to details in Section 6.3.

4. The Containment Spray System which, in conjunction with melting ice, provides long-term cooling of the containment. Refer to details in subsections 6.2.1 and 6.2.2.

5. The Annulus Filtration System which collects containment leakage and filters radio-iodine and particulates prior to discharge. Refer to details in subsection 6.2.3.

6. The air return fans which return air to the lower containment compartment after loss-of-coolant blowdown. Refer to details in subsection 6.4.1.

7. The Containment Isolation System which provides means of isolating the piping which penetrates the containment to prevent the release of radioactivity to the outside environment. Refer to details in subsection 6.2.4.

8. The Combustible Gas Control System which controls the post-LOCA hydrogen concentration in the containment. Refer to details in subsection 6.2.5.

Operability of Engineered Safety Features equipment is assured. Some of the equipment serves a function during normal reactor operation. In those cases where equipment is used for emergency functions only, such as the Containment Spray System, provision for meaningful periodic testing has been incorporated into the design. Quality control procedures are imposed on Engineered Safety Features components. These procedures incorporate the use of accepted codes and standards as well as supplementary test and inspection requirements to assure that all components will perform their intended function under accident conditions. Classification and code criteria applying to Engineered Safety Features are presented in Section 3.2.

General Design Criteria applying specifically to Engineered Safety Features discussed in this section are listed below. The 64 General Design Criteria are discussed in Section 3.1.

Criterion No.	Title
4	Environmental and Missile Design Bases
5	Sharing of Structures, Systems, and Components
16	Containment Design
35	Emergency Core Cooling
36	Inspection of Emergency Core Cooling System
37	Testing of Emergency Core Cooling System
38	Containment Heat Removal
39	Inspection of Containment Heat Removal System
40	Testing of Containment Heat Removal System
41	Containment Atmoshpere Cleanup
42	Inspection of Containment Atmosphere Cleanup Systems
43	Testing of Containment Atmosphere Cleanup Systems
50	Containment Design Basis
52	Capability for Containment Leakage Rate Testing
53	Provisions for Containment Inspection and Testing

Criterion No.	Title
54	Piping System Penetrating Containment
56	Primary Containment Isolation
57	Close Systems Isolation Valves

6.2 CONTAINMENT SYSTEMS

6.2.1 CONTAINMENT FUNCTIONAL DESIGN

6.2.1.1 Design Bases

The containment is designed to assure that radiation loses resulting from leakage of radioactive material following an accident are as low as practicable and do not exceed the limits set forth in 10 CFR 100. For purposes of integrity, the containment may be considered as the Containment Vessel and Containment Isolation System. This structure and system are directly relied upon to maintain containment integrity.

The containment is specifically designed to meet the intent of the applicable General Design Criteria listed in Section 6.1 This subsection (6.2.1), Chapter 3, and other portions of Chapter 6 present information showing conformance of design of the containment and related systems to these criteria.

The functional design of the containment is based upon the following accident input source term assumptions and conditions:

1. The design basis blowdown energies are added to the containment as shown in Figures 6.2-1 through 6.2-4.

2. A reactor power of 3579 MWt is used for decay heat generation.

3. The Engineered Safety Feature performance satisfies the single failure criteria.

4. The combination of loadings on the containment due to an accident condition and a simultaneous environmental effect is discussed in subsection 3.8.2.

Refer to paragraph 6.2.1.3.3 for the containment pressure transient analysis where the above assumptions are discussed and utilized.

Consideration is given to subcompartment differential pressure resulting from a design basis accident in subsection 3.8.2 and paragraph 6.2.1.3.5. If a design basis accident were to occur due to a pipe rupture in these relatively small volumes, the pressure would build up at a faster rate than in the containment, thus imposing a differential pressure across the wall of these structures. The design differential pressures for these subcompartments will be 20 percent higher than the calculated peak differential pressures.

Parameters affecting the assumed capability for post-accident pressure reduction are discussed in paragraph 6.2.1.3.3, Containment Pressure Transient Analysis.

6.2.1.2 System Design

The containment vessel is a freestanding, welded steel structure which consists of a vertical cylinder, a hemispherical dome, and a base plate. The Shield Building is a reinforced concrete structure similar in shape to the containment vessel. The design of these structures is described in subsection 3.8.2.

The design internal pressure for the containment is 15 psig, and the design temperature is 250°F. The net free volume is 1,250,000 ft^3. The design basis accident leakage rate is 0.5 percent/24 hours for the first 24 hours of the accident and 0.25 percent/24 hours thereafter.

The important parameters for the Floating Nuclear Plant ice condenser design compared to the Cook Plant design are shown in Table 6.2.1-1. Basically, the Floating Nuclear Plant ice condenser described in this report is physically identical to current ice condensers as applied to the Cook, Sequoyah, McGuire, and Watts Bar Plants with only two exceptions as follows:

1. The main ice bed sections in the Floating Nuclear Plant design are 36 feet high as compared to 48 feet for previous ice condenser designs.

2. The ice load, per unit length of ice bed, in each ice basket for the Floating Nuclear Plant design has been increased by about 33% as compared to current condenser designs.

These differences result in a shorter ice condenser for the Floating Nuclear Plant but the amount of ice in the ice condenser remains at least the same as in current ice condensers.

The parameters used in the design of the ice condenser for the Floating Nuclear Plant containment are well within the ranges tested in the full-scale

TABLE 6.2.1-1

COMPARISON OF PARAMETERS FOR FLOATING NUCLEAR PLANT AND

COOK PLANT CONTAINMENT DESIGNS

Design Parameter	Cook Plant Design	Floating Nuclear Plant Design
Design Basis Accident		
Blowdown mass release, lb	549,000	513,900
Blowdown energy release, BTU	346.7×10^6	321.3×10^6
Reactor Containment		
Diameter, ft	115	120
Height, ft	172	162
Upper compartment volume, ft^3	745,896	717,000
Ice condenser volume, ft^3	126,940	82,000
Lower compartment volume, ft^3	306,800	349,000
Dead-ended compartment volume, ft^3	61,702	102,400
Total containment volume, ft^3	1,241,338	1,250,000
Air compression ratio at time of final peak pressure	1.41	1.47
Ice condenser volume fraction of air at time of final peak pressure, %	64.5	64.5
Deck bypass area, ft^2	5	5

TABLE 6.2.1-1 (Cont'd)

COMPARISON OF PARAMETERS FOR FLOATING NUCLEAR PLANT AND

COOK PLANT CONTAINMENT DESIGNS

Design Parameter	Cook Plant Design	Floating Nuclear Plant Design
Ice Condenser		
Weight of ice, lb	2.45×10^6	2.45×10^6
Ice packing fraction in baskets, volume %	55	69
Height (floor to top of ice bed), ft	57	45
Main bed height, ft	48	36
Total cross-sectional area, ft^2	2,915	2,915
Ice basket diameter, ft	1.02	1.02
Number of ice basket columns	1,944	1,944
Ice basket column total cross-sectional area, ft^2	1,589	1,589
Channel flow area, ft^2	1,326	1,326
Ice channel surface area (including lower plenum ice columns), ft^2	3.16×10^5	2.43×10^5
Ice channel hydraulic diameter, ft	0.850	0.850
Ice channel ratio		
total surface area/flow area	238	183
main bed length/hydraulic diam.	56.5	42.4
Inlet door flow area, ft^2	1,000	1,000
Upper vent area, ft^2	20	20
Lower drain area, ft^2	15	15

section test facility and well within the range of applicability of the analytical models which were developed. In Table 6.2.1-1 the parameter differences between the Floating Nuclear Plant ice condenser and the Cook Plant ice condenser are clearly defined. The effects of these parameter differences on containment performance are determined in paragraph 6.2.1.3.3.

6.2.1.3 Design Evaluation

The ice condenser containment incorporating forced circulation of the containment atmosphere, together with the Containment Spray System, provides the functional capability of containment for as long as necessary following a loss of fluid from the reactor coolant system or the main steam system. The peak internal design pressure (15.0 psig) of the containment is greater than the peak compression pressure (9.3 psig) occurring as the result of the complete blowdown of the reactor coolant through any rupture of the Reactor Coolant System up to and including the hypothetical double-ended severance of the largest reactor coolant pipe. The design pressure is not exceeded during any subsequent long-term pressure transient.

6.2.1.3.1 Systems Required to Maintain Leaktightness

Following an accident, the containment air may become radioactively contaminated. The isolation of the containment following an accident is therefore mandatory. During normal operation, air flowing intermittently from inside the containment to the outer atmosphere is filtered and monitored by the normal containment purge system. Following an accident, the normal purge system is removed from service. The Annulus Filtration System maintains pressure within the annulus below atmospheric.

Containment leaktightness is accomplished by actuation of the Containment Isolation System. Any leakage from the containment enters the annulus and passes through the annulus filtration system.

6.2.1.3.2 Evaluation of Systems Required to Maintain Leaktightness

The Containment Isolation System is designed to operate during emergency situations. (Refer to subsection 6.2.4.) The containment will be leak-tested using the compressed air system described in subsection 9.3.1.

6.2.1.3.3 Containment Pressure Analyses

The ice bed, which is the heat sink of the ice condenser, will be a completely enclosed annular compartment located around the perimeter of the containment, penetrating the operating deck so that a portion extends into the lower and upper compartments of the containment. The lower portion will have a series of doors exposed to the atmosphere of the lower containment compartment which, for normal plant operation, will be designed to remain closed. At the top of the ice condenser will be another set of doors exposed to the atmosphere of the upper compartment, which remain closed during normal plant operation.

The ice will be held within the ice condenser as ice cylinders arranged to promote heat transfer from steam to ice in case the condenser is required. A refrigeration system will maintain the ice in the solid state.

In the event of a loss-of-coolant accident, the door panels located below the operating deck will open almost immediately due to the pressure rise in the lower compartment caused by the release of reactor coolant and the resulting steam. This allows the air and steam to flow from the lower compartment into the ice condenser. The resulting buildup of pressure in the ice condenser will open the door panels at the top of the condenser and allow the air to flow out of the condenser into the upper compartment. The steam will be condensed in the ice condenser compartment, thus limiting the peak pressure in the containment at the so-called "initial peak pressure." At this stage pressure differentials exist across the operating deck and the crane wall.

A dividing barrier will separate upper and lower compartments and assure that the steam is directed into the bottom of the ice condenser. Only a small amount of steam will be able to bypass the ice condenser through the dividing barrier.

As the transient progresses the pressure in the lower compartment again increases. At the time the pressure is uniform throughout the containment, the so-called "final peak pressure" has been attained. A reduction of the total containment pressure to about 6 psig occurs when the air return fans initiate a rapid return of air to the lower compartment. The containment pressure remains at a low value until all the ice is melted. At the end of ice melt, the total containment pressure again increases to a peak value at which stage the containment spray system is removing from the containment all residual heat generation.

6.2.1.3.3.1 Pressure Differentials Across Internals - Lower Compartment and Ice Condenser

The basic performance of the ice condenser containment has been demonstrated for a wide range of conditions by the ice condenser test program. These results have clearly shown the capability and reliability of the ice condenser concept to limit the containment pressure rise subsequent to a hypothetical loss-of-coolant accident.

To supplement this experimental proof of performance, a mathematical model has been developed to simulate the ice condenser pressure transients. This model, encoded as computer program TMD (Transient Mass Distribution), provides a means for computing pressures, temperatures, heat transfer rates, and mass flow rates as a function of time and location throughout the containment. Thus, it is possible to compute pressure differences on various structures within the containment as well as the distribution of steam flow as the air is displaced from the lower compartment. Further, the model is experimentally verified by comparison with the ice condenser tests. The model includes the effect of entrainment of water in the break flow. An input parameter allows the user to choose entrainment in the range 0-100%.

Description of Analytical Model

The mathematical model is similar to that of the SATAN code in that the entire solution is developed by considering only the conservation equations and the equations of state, together with the control volume technique for simulating spatial variation. The governing equations naturally are somewhat different for the steam-air mixture considered in the TMD analysis, as compared with the single constituent model used in the SATAN code. These equations are, however, quite similar to the equations employed in the COCO computer code. This latter model is used to simulate the pressure and temperature transients in dry containment structures. The TMD analysis is therefore a logical extension in analytical capability achieved by combining the key features of two existing experimentally verified models.

The control volume technique is used to represent spatially the containment. The containment has been divided into 34 elements or compartments as shown in Figures 6.2-5, 6.2-6, 6.2-7, and 6.2-8. The interconnection between containment elements in the TMD code is shown schematically in Figure 6.2-9. Flow resistance and inertia are lumped together in the flow paths connecting the elements shown. The division of the lower compartment into six volumes occurs at the points of greatest flow resistance, i.e., the four steam generators, pressurizer and refueling canal. Each of these lower compartment sections delivers flow through doors into a section of the ice bed. Each section of the ice bed is, in turn, divided into three elements. Thus, a total of eighteen elements (elements 7 through 24) are used to simulate the ice bed. The six elements at the top of the ice bed deliver to element number 25, the upper compartment. Note that cross flow in the ice bed is not accounted for in the analysis; this yields the most conservative results for the particular calculations described herein. The upper reactor cavity (element 29) is connected to the lower compartment volumes and provides cross flow for pressure equalization of the lower compartment.

In each of the elements shown, the detailed calculation of the thermodyanmic state is performed. Also, mass flow rates are calculated for each of the flow paths assigned to an element.

The calculation for an element proceeds in the following manner. At a given instant in time, all properties are known. The rates at which mass and energy are crossing the boundary of an element are evaluated and the mass and internal energy at the new time are calculated via the mass and energy conservation equations. Other thermodynamic quantities of interest (pressure, enthalpy, temperature and density) are found by employing the equations of state and appropriate thermodynamic relationships. Finally, the flows are updated by using the momentum equation. This sequence is repeated for all elements. The heat transfer rate is calculated as a function of time and position in the bed.

The following assumptions are made in the analysis:

1. Discharge mass and energy flow rates through the reactor coolant system break are established from the coolant blowdown and core thermal transient analysis as indicated by the SATAN model. This model predicts conservatively high blowdown rates.

2. At the break point, the discharge flow separates into steam and water phases. The water phase is at the total containment pressure, while the steam phase is at the partial pressure of the steam in the element. The effect of water entrainment can be taken into account. In this case the entrained water is in thermal equilibrium with the steam.

3. Air is taken as an ideal gas; compressed water and steam tables are employed for water and steam thermodynamic properties. Thermodynamic equilibrium is assumed to exist at all times.

4. Because this analysis is concerned with the early part of the transient, accumulation of water in the sump and wall temperature transients are not calculated.

5. The steam and air are thoroughly mixed in each element. Thus, their temperatures are equal and the steam-to-air ratio in the mixture leaving the element is equal to that within the element.

Experimental Verification

To verify the correctness of the mathematical formulation, the code has been applied to certain tests performed in support of the ice condenser concept. The geometrical model used is similar to a section of the model used to simulate the plant geometry in that three volumes are used to represent the ice bed. Techniques for determining the ice bed and door flow resistance, as well as blowdown rate for the test apparatus, were presented in WCAP-7833. These results have been used as input to the present analysis.

Calculated pressure transients when compared with the measured values for tests simulating the reference ice condenser design showed the analysis follows the pressure transient of the experiments quite well but is about one psi higher. This comparison was presented in WCAP-7833 Supplement 2.

Application to Plant Design

The TMD analytical model has been applied to the plant design, using a 34 element configuration to develop design loads for the various structures in the containment. Analyses have been performed to determine the various pressure transients resulting from a reactor coolant pipe break in any one of the six lower compartment elements. The maximum peak pressure and differential pressures were determined for breaks in each element. Initial input pressures for the analyses are:

Ice condenser compartment	Steam partial pressure	0.07 psia
	Air partial pressure	14.93 psia
Other compartments	Steam partial pressure	0.3 psia
	Air partial pressure	14.7 psia

It was found that the highest pressure differentials across the lower portion of the crane wall and across the operating deck occur when 100% entrainment is assumed. However the maximum differential pressure across the upper crane wall (and maximum peak pressure within the ice condenser) occurred when 0% entrainment was assumed.

The maximum differential pressures are summarized on Table 6.2.1-2.

6.2.1.3.3.2 Final Peak Pressure

As the reactor coolant system blowdown continues, the air which was initially in the lower compartment is displaced into the upper compartment and a portion of the ice condenser compartment by the escaping coolant.

As a result, the pressure in the upper compartment rises from the beginning of the blowdown and reaches a peak value at or before the end of the blowdown.

TABLE 6.2.1-2

CALCULATED MAXIMUM PEAK DIFFERENTIAL PRESSURES

Break in element	1	2	3	4	5	6
ΔP (psi) across lower crane wall and operating deck	12.21	9.76	9.34	8.20	10.65	12.51
Maximum ΔP (psi) across upper crane wall	9.59	7.51	7.33	6.40	8.06	9.83

The primary factor in producing the final peak pressure is the displacement of air from the lower compartment into the upper compartment. The ice condenser is very effective in condensing essentially all the steam that is released as a result of the reactor coolant system blowdown, so that the only sources of steam to the upper compartment are from bypass through the deck which separates the upper and lower compartments and from a very small amount of steam leakage through the condenser. This steam bypass adds a small amount to the pressure rise in the upper compartment.

A method of analysis of the final peak pressure was developed based on results of full-scale section tests as described in WCAP-7183 Supplement 2. This method consists of the analysis of the air mass compression process based on the containment compartment volumes, the polytropic exponent for the compression process, the steam bypass through the operating deck, and the steam leakage through the ice condenser. All of these parameters have been obtained based on full-scale section test results.

The final peak pressure for the Floating Nuclear Plant design was calculated to be 9.3 psig (for an initial containment pressure of 15.0 psia) which includes a pressure increase of 0.4 psi from deck steam bypass, a pressure increase of 0.5 psig from condenser steam leakage and also the effects of the dead-ended compartment volumes.

In the following sections a discussion of the major parameters affecting the final peak pressure is presented. Specifically, these parameters are: air compression, steam bypass through the deck, steam leakage through the condenser, blowdown rate and blowdown energy.

Air Compression

As discussed above, during the blowdown resulting from a loss-of-coolant accident, air is displaced from the lower compartment volume and compressed into the ice condenser and into the upper compartment volumes.

A parameter used in describing the air compression effect is the ratio of initial (total) to final (compressed) volumes of air considered in the process. For the purpose of this "compression ratio" definition, initial and final volumes are defined as follows:

1. Initial volume is the total air volume in the containment that actively participates in the compression process during a loss-of-coolant accident.

2. Final volume is the volume that receives the air displaced by steam in such an accident.

The final volume includes the volume of the upper compartment as well as part of the volume of air in the ice condenser. The results of a representative full-scale section test (Figure 6.2-10) show a variation in steam partial pressure from 100% near the bottom of the ice condenser to essentially zero near the top. The thermocouples and pressure detectors confirm that at the time when the final peak pressure is reached steam occupies less than half of the volume of the ice condenser. The analytical model used in defining the containment peak pressure uses upper compartment volume plus 64.5% of the ice condenser air volume as the final volume. This 64.5% value was confirmed by applicable test results as discussed in WCAP-7183 Supplement 2.

There are several dead-ended volumes contained in the lower compartment that are not part of the active volume. In an accident, steam will not force air from these volumes into the upper compartment. Typical dead-ended volumes are the instrument room and accumulator rooms. Further, the results of certain full-scale section tests, which contained dead-ended volumes, showed that some air flowed into these volumes and remained there during blowdown, thus reducing the mass compression ratio for the containment. It was found from tests, as reported in WCAP-7183 Supplement 2 that 18% additional air mass was compressed in the dead-ended volume. This change in air mass when corrected for the lower final peak pressure in the design as compared with the test, indicates that 17% additional air mass will be compressed into the dead-ended volumes of the containment.

The compression ratio for the Floating Nuclear Plant is found from the following:

$$Cr = \frac{V_1 + V_2 + V_3 \; -0.17\; V_4}{V_3 + 0.645\; V_2}$$

where:

V_1 = Lower compartment active volume

V_2 = Ice condenser compartment volume

V_3 = Upper compartment volume

V_4 = Dead-ended compartment volume

A compression ratio of 1.47 is calculated based on the Floating Nuclear Plant compartment volumes shown in Table 6.2.1-3.

Steam Bypass through the Deck

A second factor in determining the final peak pressure is the amount of steam entering the upper compartment. The addition of steam heats the air in the upper compartment resulting in an increase in air partial pressure, as well as an increase in the steam partial pressure.

The mass of steam which leaks through the operating deck into the upper compartment is a function of the relative flow areas and loss coefficients of the deck and inlet doors to the ice condenser, the mass of steam input to the lower compartment at the time of peak pressure, and the mass of steam stored in the lower compartment at the time of peak pressure.

The effect of deck leakage on upper compartment pressure was verified by a series of full-scale section tests. The coefficient, which determines the increase in the final peak pressure from deck leakage, was found to be 0.081 psi increase in containment pressure per ft^2 of deck leakage area for the plant design. These tests and analysis have been reported in WCAP-7183 Supplement 2.

The divider barrier, including the enclosures over the steam generators and reactor vessel, is designed to provide a reasonably tight seal against leakage. Holes are purposely provided in the bottom of the refueling canal to allow water from sprays in the upper compartment to drain to the sump in the lower compartment. Potential leakage paths also exist at all the joints between the operating deck and the pump access hatches and reactor vessel enclosure slabs. The total of all deck leakage flow areas for the containment is assumed conservatively to be 5 square feet as indicated in Table 6.2.1-3. The effect of this potential leakage path on final peak pressure is small and is found to be:

$$\Delta P_{deck} = 5 \times 0.081 = 0.4 \text{ psi}$$

In the event that the reactor coolant system break flow is so small that it would leak through these flow paths without developing sufficient differential pressure (about 1 lb/ft^2) to open the ice condenser doors, the

TABLE 6.2.1-3

SUMMARY OF PARAMETERS FOR CALCULATION OF FINAL PEAK PRESSURE

Upper Compartment Volume, ft^3	717,000
Lower Compartment Volume, ft^3	349,000
Ice Condenser Free Volume, ft^3	82,000
Dead-Ended Volume, ft^3	102,000
Percentage of ice condenser free volume occupied by air at time of peak pressure	64.5%
Compression Ratio	1.47
Compression Pressure, psig	8.4
Deck Leakage Area, ft^2	5
Pressure Increase due to Deck Leakage, psi	.4
Pressure Increase due to Ice Condenser Leakage, psi	.5
Final Peak Pressure, psig	9.3

steam from the break would slowly pressurize the containment. The containment spray system, described in a later section, has sufficient capacity to maintain pressure well below design for this case and for all breaks which result in partial opening of the ice condenser doors.

Steam Leakage through the Condenser

1. Steam Leakage in Applicable Full-Scale Section Tests

The mass of steam which leaks through the ice condenser into the upper compartment is very small and is a function of the condenser channel surface area and flow area, the mass of steam input to the lower compartment at the time of peak pressure and the mass of steam stored in the lower compartment at the time of peak pressure. The condenser steam leakage into the upper compartment and the resulting increase in final peak pressure was determined on the basis of the results of 6 full-scale section tests which fully covered wide variations in all important parameters of interest especially with respect to the differences between the Cook Plant and Floating Nuclear Plant ice condensers. The parameters tested included channel surface area, channel flow area, ice condenser length, and ice bed packing fraction. In Table 6.2.1-4 the parameter values in these 6 tests are compared with the containment equivalent parameters based on a scale factor of 350. All of these 6 tests were conducted at approximately 170% of the containment design blowdown energy and at approximately 110% of the containment design blowdown rate.

TABLE 6.2.1-4

COMPARISON OF TEST AND CONTAINMENT EQUIVALENT PARAMETERS

Parameter	Containment	Test Value Equivalent to Containment	Test Number					
			42	43	44	56	70	71
Channel surface area (ft^2)	2.43×10^5	694	890	890	772	638	655	655
Channel flow area (ft^2)	1326	3.79	3.79	3.79	5.58	3.79	5.39	3.79
Average condenser channel length (ft)	39	39	35.4	35.4	35.4	25.4	35.9	35.9
Number of ice baskets	1944	5.6	8	8	8	8	6	6
Ice bed packing fraction (%)	69	69	57	69	100	57	57	57

The condenser channel surface area in these tests ranged from 92% to 128% of the Floating Nuclear Plant containment equivalent surface area and the channel flow area varied from 100% to 147% of the equivalent flow area for this plant. As indicated in Table 6.2.1-4, tests were performed with condenser channel lengths varying from approximately 25 feet to 36 feet (channel length for Floating Nuclear Plant is 39 feet) and with 6 to 8 ice baskets (Floating Nuclear Plant equivalent value is 5.6). Also, the ice bed packing fraction (within the ice baskets) ranged from 57% to 100% in these tests which compares with the Floating Nuclear Plant ice bed packing fraction of 69%.

Therefore, it is apparent that these 6 tests covered wide variations in the important parameters which determine the amount of steam leakage through the ice condenser for the Floating Nuclear Plant during the reactor coolant system blowdown.

The results of these 6 full-scale section tests (presented in Figure 6.2-11), which cover a wide range of design parameters with ice beds of different lengths, different ice channel surface areas, different channel flow areas, and different ice bed packing fractions, demonstrate the following:

a. The important parameters in determining the amount of steam leakage through the ice condenser, for blowdown times and mass flow rates representative of the Floating Nuclear Plant design, are the ice channel surface area (A_s) and the ice bed flow area (A_f).

b. A change in the total channel surface area resulting from a change in the number of ice basket columns has the same effect on steam leakage as a similar change in total surface area resulting from a change in the condenser channel length, i.e., the total channel surface area is the important parameter.

c. A solid ice bed provides the same condenser heat transfer performance as a cube ice bed of lower packing fraction.

2. Steam Leakage in the Containment

The steam leakage through the ice condenser is calculated based on the results of the tests given in Figure 6.2-11. For the Floating Nuclear Plant ice condenser, the ratio of condenser channel flow area to surface area is 5.4×10^{-3} and for the Cook Plant the ratio is 4.5×10^{-3}. From Figure 6.2-11, the increase in final peak pressure for the Floating Nuclear Plant compared with the Cook Plant (which was essentially 0) as a result of condenser steam leakage ($\Delta P_{i.c.}$) is,

$$\Delta P_{i.c.} = 0.5 \text{ psi}$$

Calculation of Containment Final Peak Pressure

The final peak pressure in the upper compartment for the Floating Nuclear Plant containment design is given by:

$$P_2 = P_1 (Cr)^n + \Delta P_{deck} + \Delta P_{i.c.}$$

where:

P_1	=	Initial containment pressure (15.0 psia)
Cr	=	Volume compression ratio (1.47)
n	=	Polytropic exponent as determined by test results (1.13)
ΔP_{deck}	=	Pressure increase caused by deck bypass (0.4 psi)
$\Delta P_{i.c.}$	=	Pressure increase caused by ice condenser leakage (0.5 psi)

A final peak pressure of 9.3 psig is calculated for the Floating Nuclear Plant containment.

Sensitivity to Blowdown Energy

The sensitivity of the upper compartment final peak pressure to the amount of coolant energy release is shown in Figure 6.2-12. This figure shows the magnitude of the final peak pressure versus the amount of energy released in terms of percentage of equivalent reactor coolant system energy release. These data are based on results of tests which were run at 110% and 200% of the initial blowdown rate equivalent to the maximum containment blowdown rate.

These test results indicate the very large capacity of the ice condenser for additional amounts of energy with only a small effect on final peak pressure. For example, for tests with a 100% energy release a peak pressure of about 6.8 psig was measured while an increase up to 220% energy release resulted in an increase in peak pressure of only about 2 psi. It is also important to note that maldistribution of steam into different sections of the ice condenser would not cause even the small increase in peak pressure that is shown in Figure 6.2-12. For every section of the ice condenser which may receive more energy than that of the average section other sections of the ice condenser would receive less energy than the average section. Thus, the peak pressure in the upper compartment would be indicated by the test performance based on 100% energy release rather than either the maximum energy release section or the minimum energy release section.

Figure 6.2-13 gives some insight as to the very large capacity for energy absorption of the ice condenser. Figure 6.2-13 is a plot of the amount of ice melted versus the amount of energy released based on test results at different energies and blowdown rates. These test results indicate that a 200% energy release melts only about 74% of the ice while 100% energy release melts about 37% of the ice. Thus, even for energy releases considerably in excess of 200% there would still be a substantial amount of ice remaining in the condenser.

Sensitivity to Blowdown Rate

Figure 6.2-14 shows the effects of blowdown rate on the final peak pressure in the upper compartment. Figure 6.2-14 is based on test results with a series of tests all with a condenser configuration closely representative of the Floating Nuclear Plant but with the important difference that all of these tests were run with 175% of the test equivalent reactor coolant system energy release. These tests results indicate that an increase in blowdown rate from 100% to 200% produces a small increase in the magnitude of this peak pressure (about 1 psi).

Sensitivity to Packing Fraction

The sensitivity of the upper compartment final peak pressure to the ice bed packing fraction was determined based on the results of two full-scale section tests. These two tests were run with 175% of the test equivalent reactor coolant energy release and 110% of the test equivalent energy release rate. The results of these tests demonstrated the insensitivity of final peak pressure to ice bed packing fraction. Specificially, the test with a packing fraction of 57% had a final peak pressure of 7.9 psig while the test with a packing fraction of 69% recorded a final peak pressure of 7.4 psig. Therefore, based on these full-scale section test results, it is concluded that ice bed packing fraction has essentially no effect on the containment final peak pressure.

Measurement of Final Peak Pressure in Full-Scale Section Tests

The final peak pressure can also be determined directly from the results of a closely-applicable but still very conservative full-scale 1/350 section test. Table 6.2.1-5 presents a summary of the important parameters for the containment and for Test 71.

As indicated in Table 6.2.1-5, the total coolant energy released in Test 71 was approximately 170% of the equivalent reactor coolant energy release to the containment. Further, the ice weight was substantially less than that equivalent to the containment design. Thus, Test 71 is a very conservative representation of the containment.

Even so, the final peak pressure measured in Test 71 was only 8.8 psig compared to the final peak pressure of 9.3 psig calculated for the containment. Further, the differences in final peak pressures between Test 71 and the containment are directly attributable to the differences between the test and the containment as summarized in Table 6.2.1-6. Therefore, the results of this test are considered to provide an excellent check on the results of the final peak pressure calculations presented in this report.

Small Break and Double-Accident Performance

Analyses and tests of the ice condenser containment performance have demonstrated that the ice condenser alone is capable of limiting containment

TABLE 6.2.1-5

COMPARISON OF CONTAINMENT DESIGN WITH FULL-SCALE SECTION TEST 71

Parameter	Containment Design	Test Value Equivalent to Containment	Test 71
Energy released to containment (BTU)	321.3×10^6	$.92 \times 10^6$	1.69×10^6
Energy blowdown rate (BTU/sec)	56.2×10^6	1.61×10^5	1.77×10^5
Ice weight (lb)	2.45×10^5	7000	5446
Ice bed packing fraction (volume %)	69	69	55
Ice condenser			
- flow area (ft^2)	1326	3.79	3.79
- channel surface area (ft^2)	243,000	694	655
- ice basket diameter (ft)	1.02	1.02	1.02
Compression ratio	1.45	1.45	1.40
Deck bypass area (ft^2)	5	.015	.015

TABLE 6.2.1-6

CONTAINMENT FINAL PEAK PRESSURE
BASED ON TEST 71 TEST RESULTS

	Pressure (psig)
Measured Final Peak Pressure in Test 71	8.8
Correction for Reduced Containment Energy Release[1]	-1.3
Correction for Reduced Containment Blowdown Rate[2]	-0.1
Correction for Increased Initial Containment Pressure and Increased Compression Ratio[3]	1.6
Corrected Test 71 Final Peak Pressure	9.0
Predicted Containment Final Peak Pressure	9.3
Unaccounted for Difference between Predicted Containment and Corrected Test 71 Final Peak Pressures	0.3

Therefore, the corrected Test 71 final peak pressure and predicted containment peak pressure are in close agreement thereby verifying the results of final peak pressure analysis given in this report.

1 Adjusted based on test results given in Figure 6.2.1-12.

2 Adjusted based on test results given in Figure 6.2.1-14.

3 Adjusted based on calculational method for air compression.

pressure for a wide spectrum of reactor coolant system or secondary system blowdowns. However, extremely small blowdown rates would not generate a differential pressure to fully open the ice condenser doors. For these small pipe breaks, a larger than normal fraction of the break flow will pass through the deck and into the upper compartment. Also, for breaks less than approximately 5,000 to 10,000 gpm, only a fraction of the air may be displaced from the lower compartment by the incoming steam. Therefore, for small breaks, the steam bypass through the deck and the performance of the containment spray system become important in determining the containment peak pressure.

Another case has been examined where it is postulated that a small break loss-of-coolant accident precedes a larger break accident which occurs before all of the coolant energy is released by the small break, (i.e., a double-accident). During the small break blowdown, some quantity of steam and air will bypass the ice condenser and enter the upper compartment via leakage in the divider deck. The steam which reaches the upper compartment will then add to the peak pressure for the second part of the accident. Therefore, the containment spray system is used to limit the partial pressure of steam in the upper compartment due to deck bypass.

As discussed previously, for large pipe breaks the major contributor to the containment final peak pressure is the displacement of air from the lower compartment. For this double accident case, any air displaced by the small break will reduce the air displaced by the subsequent larger break, with the result that the air displacement contribution to final peak pressure will be no higher than for the single blowdown accident case.

These small break and double accident analyses were performed previously by Westinghouse for the Cook Plant based on containment volumes which differ somewhat from the volumes which are representative of the Floating Nuclear Plant containment. However, the effects of these parameter differences on the analyses of containment pressures and performance are insignificant.

Small Break Performance

During a small pipe break accident, steam and air will flow through the partially-opened doors into the ice condenser and also through the deck bypass area into the upper compartment. For the containment, this bypass area is composed of two parts: an assumed leakage area around the various hatches in the deck, and a known leakage area through the deck drainage holes located at the bottom of the refueling cavity. Air and steam are assumed to leak through the 5 ft^2 deck leakage area while it is assumed that no air flows through the ice condenser to the upper compartment.

This calculational model is applicable to breaks less than 5,000 to 10,000 gpm where little air is displaced from the lower compartment.

Differential pressure developed across the ice condenser doors and the divider deck and the flow split between the ice condenser and the 5 ft^2 divider deck leakage area were determined. From this analysis, the flow rate of air and steam into the upper compartment were then found. The mass and energy equations were then applied to the upper compartment in order to determine the containment pressure. Note that since the pressure differences between the various compartments are small (less than 1 psf) over the range of break flows of interest in this calculation, the rates of pressure change during this part of the accident were essentially the same in all compartments.

The results of the small break analysis indicated that the increase in steam partial pressure in the upper compartment was 1.35 psi when the containment spray set pressure of 3 psig was reached. Note that this analysis assumed that air and steam flowed through the 5 ft^2 of deck bypass area with no air flow through the ice condenser. If, on the other hand, the mixture were assumed to flow through both the divider barrier and ice condenser, less steam would enter the upper compartment when the containment spray actuation pressure (3 psig) was reached. Thus, for this case the steam partial pressure in the upper compartment would be less.

Double-Accident Performance

Following is a description of the performance of the ice condenser for the case of a double-accident, i.e., an initial small break followed by the maximum size double-ended pipe break. The important design requirement for the case of a double-accident is that the amount of steam leakage into the upper compartment must be limited during the first part (small break) of the accident so that only a small increase in final peak pressure results for the second part (double-ended break) of the postulated accident. The key elements which determine the double-accident performance are the ice condenser lower doors which open at a low differential pressure (less than one psf to admit steam to the ice condenser and limit the bypass flow of steam through the deck, and the upper compartment sprays which condense this bypass flow of steam and limit the partial pressure of steam in the upper compartment to a low value, less than 2 psia. As stated previously the major contributor to the containment final peak pressure for large breaks is the displacement of air out of the lower compartment. For this accident case, any air displaced by the small break would reduce the air displaced by the subsequent larger break, with the result that the air displacement contribution to the final peak pressure would be no higher than for the single large blowdown accident case.

The increase in pressure caused by the steam in the upper compartment was evaluated for the Cook Plant by applying the conservation of mass and energy equations to the upper compartment. It was found that the 1.35 psi partial pressure of steam in the upper compartment at the end of the small break adds only 2.1 psi to the normal final peak pressure of 9.3 psig for the subsequent larger break. The resulting containment final peak pressure of 11.4 psig for the double-accident is still well below the containment design pressure of 15 psig.

6.2.1.3.3.3 Long-Term Containment Performance

Upon completion of blowdown and the initial depressurization of the containment, a large amount (about 60 percent) of the ice remains. This ice together with the containment spray system provides a considerable capacity to accept large post blowdown energy releases.

The primary purpose of the containment spray system is to spray cool water into the containment atmosphere in the event of a loss-of-coolant accident and thereby assure that containment pressure will not exceed the containment peak design pressure. This protection is afforded for all pipe break sizes up to and including the hypothetical instantaneous circumferential rupture of a reactor coolant loop. Description of the containment spray system is in Subsection 6.2.2. The containment spray system is designed based on the conservative assumption that core residual heat is continuously released to the containment as steam, eventually melting all ice. The heat removal capability of the spray system is such as to keep the containment pressure below design after all the ice has melted and residual heat generated steam continues to enter the containment.

In addition to the reactor coolant system blowdown and decay heat generation, the ice condenser and spray system also accommodate the release of stored metal heat and heat removed from the steam generators by the action of the ECCS. The total heat addition to the containment from all of the above sources is shown in Figures 6.2-1 through 6.2-4.

Containment Long Term Transient Model

A long term ice containment computer code (LOTIC) has been developed to analyze the containment transient after completion of the initial blowdown. This code provides the same capability to analyze the ice condenser containment as does the COCO code for the dry containments. The major feature of this code is its capability to describe the several volumes of the ice condenser containment. Not only are the upper, lower, dead ended and ice condenser volumes described, but also the ice condenser is divided into six circumferential sections, each with two vertical divisions. Another significant feature of the new code is the two-sump configuration (active and stagnant sumps) so that the flood level and temperature history of the containment is accurately modeled. The code also describes the performance of the air return fan which blows upper compartment air to the lower compartment. The performance of the spray heat exchangers is accurately modeled in the transients. The basic equations used are the standard transient mass and energy balances and equations of state, appropriately coupled to the multi-volume ice condenser containment. Based on the ice condenser test program the contents of the lower compartment are considered mixed for the long term residual heat boil-off phase. The code also considers accumulator gas added to the containment and the displacement of free volume by the refueling water storage tank volume.

The LOTIC code uses the control volume technique to represent the physical geometry of the system. Fundamental mass and energy equations are applied to the appropriate control volumes and solved by suitable numerical procedures. The initial conditions of the containment before blowdown are specified by compartment. The peak pressure at the end of blowdown is also given as input.

No attempt is made to solve the complex pressure and flow transients in the containment during the reactor coolant system blowdown transient. This is done by the TMD code and the compression ratio calculation discussed previously. Ice melt is calculated for the blowdown period based on the mass and energy released to the containment. The blowdown steam, water and ice melt result in a sump temperature of 175°F at the end of the initial blowdown. Test results showed the 175°F sump temperature to be representative of the sump temperature for this blowdown. For the long term period of decay heat steam boil off and post blowdown energy release an ice melt temperature of 75°F was used. This ice melt temperature is based on test results with very small blowdown rates representative for this period.

The basic LOTIC code assumption is that the total pressure in all containment compartments is uniform at the end of blowdown. This assumption is justified by the fact that after the blowdown of the RCS the remaining mass and energy released from this system into the containment is small and very slowly changing. The resulting flow rates between compartments will also be relatively small. These small flow rates do not create significant pressure differentials between the containment compartments.

Ice condenser tests have been performed to demonstrate the long term performance capability of the ice condenser. Specifically, these tests verified the capability of the ice condenser to reduce the containment pressure within a few minutes following the blowdown and, in addition, tests showed excellent condenser performance even with only a small amount of ice remaining in the ice condenser for tests simulating the long term addition of residual heat. Tests numbered 80 through 87 performed with simulated core heat (as reported in WCAP-7833, Supplement 2) verified the uniform containment compartment pressures that exist during the decay heat addition phase of the accident.

Application to Plant Design

The following are the major input assumptions used for the design basis transients for the containment.

1. Degraded performance of the Safety Injection System is assumed, e.g., residual heat is continuously boiled off as saturated steam.

2. Minimum engineered safety features are employed in all calculations.

3. 2.45×10^6 lbs ice initially in the ice condenser.

4. The maximum blowdown energy and mass releases of 321.3×10^6 BTU and 513,900 lb.

5. The maximum calculated core power of 3579 MWt was used for decay heat generation.

6. Nitrogen from the accumulators in the amount of 5942 lbs included in the calculations.

7. Essential raw water temperature of 85°F was used on the shell side of the spray heat exchanger.

8. Air return fan is actuated 30 seconds after the transient begins.

9. No maldistribution of steam flow to the ice bed was assumed following the initial ice melt. This assumption is conservative, since any early melt out of ice bed sections would allow the sprays to work on the leakage and thus delay complete ice melt out time.

10. No ice condenser bypass was assumed. (This assumption depleted the ice in the shortest time and was thus conservative.)

11. The initial conditions in the containment were pressure 0.3 psig and temperature 100°F in the lower and dead-ended volumes, 15°F ice condenser free volume, and 80°F upper volume.

12. A minimum spray of 7200 gpm was assumed at the initiation of recirculation. The spray pumps take suction from the containment sump when the refueling water storage tanks are emptied (total volume of refueling water is 350,000 gallons).

Figure 6.2.1-15 is a plot of containment total pressure versus time for the design basis accident. The peak pressure of 9.3 psig at the end of the RCS blowdown was input to the code and is based on the compression ratio of containment volumes. The start of an air return fan at 30 seconds assures a rapid reduction in containment pressure. The pressure remains low until continued steam flow due to residual heat melts the ice remaining after blowdown. Subsequent to ice melt the spray system must maintain the pressure below design.

Figure 6.2-1-16 shows the containment upper and lower compartment temperature transients for the design basis accident of the previous figure. The sharp increase in upper and lower compartment temperature occurs at the time of ice bed melt out. The vapor pressure of the upper and lower compartments can be found from the steam saturation pressure at the temperatures shown on the plot.

Figure 6.2.1-17 shows variations of spray temperature and containment sump temperature for the Design Basis Accident.

6.2.1.3.4 Accident Chronology and Energy Balance

For the containment pressure transient analysis of Section 6.2.1.3.3 an accident chronology is presented in Table 6.2.1-7. Table 6.2.1-8 shows the distribution of stored energy before and after accident initiation.

6.2.1.3.5 Subcompartment Differential Pressure Design Basis

Design Basis

Careful consideration is given in the design of the containment internal sub-compartments to local pressure pulses that could occur following a loss-of-coolant accident. If a loss-of-coolant accident were to occur due to a pipe rupture in, or close to, these relatively small volumes, the pressure would build up at a rate faster than the overall containment, thus imposing a differential pressure across the walls of the structures.

The following is an estimate of the design pressure for each of the containment subcompartments based on analyses of previous ice condenser containment designs:

TABLE 6.2.1-7

LOSS-OF-COOLANT ACCIDENT CHRONOLOGY FOR ICE CONDENSER CONTAINMENT EVALUATION

Assumptions used for the LOTIC code for evaluation of the long-term temperatures and pressures inside containment following a LOCA:

Event	Time, seconds
End of Blowdown	0
ECCS Initiation	30
Spray Initiation	30
Air Return Fan Initiation	30
Recirculation Begins	1628
Peak Pressure	12.5

TABLE 6.2.1-8

ENERGY BALANCE

(All quantities in millions of BTU)

I.	Sources		
	Reactor Coolant	305.23	
	Accumulator (3 of 4)	13.879	
	Accumulator (spilling)	4.626	
	Initial Core Stored	41.405	
	Thin metal (Internals)	14.863	
	Thick metal (Reactor Vessel)	17.329	
	Core Power Generation to End of Blowdown	9.313	
	Total Sources		406.645
II.	Blowdown		
	A. Available		
	Reactor Coolant	305.23	
	Transferred into coolant:		
	Accumulator (3 of 4)	4.432	
	Core Stored Heat	31.106	
	Thin Metal Stored Energy	8.918	
	Total Available		349.686
	B. Transferred out:		
	Blowdown	322.283	
	To Steam Generator	-0.963	
			321.320
	C. Remaining in RCS at end of Blowdown	28.366	
	Remaining in Accumulators	9.447	
	Spilling Accumulator	4.626	
	Remaining Core Stored Energy	19.612	
	Remaining Thin Metal Stored	5.945	
	Remaining Thick Metal Stored Energy	17.329	
	Total Remaining		85.325
	Total Accounted For		406.645

1.	Reactor vessel annulus:	100 psig
2.	Pipe sleeve:	900 psig
3.	Compartment above reactor:	41 psig
4.	Compartment below reactor:	32 psig
5.	Steam generator enclosure:	17 psig

These pressures are based upon preliminary compartment vent areas and volumes. Final design pressures will be determined after establishing firm volumes and vent areas for the Floating Nuclear Plant.

Consistent with the design basis of the overall containment, the above pressures are 20 percent higher than the pressure peaks resulting from the design basis blowdown in each compartment.

6.2.1.4 Containment Testing and Inspection

The containment shell and base plate will be structurally tested prior to initial reactor operation in accordance with ASME Code, Section III, Subsection NE. In addition, the containment will be leak tested prior to fuel loading.

Design provisions are incorporated to assure the capability to perform leakage tests during the plant life.

6.2.1.4.1 Types of Tests

The types of tests are as follows:

Test Type A — Tests intended to measure the reactor containment overall integrated leakage rate.

Test Type B — Tests intended to detect or measure local leaks of containment penetrations, hatches, and personnel locks.

Test Type C — Tests intended to detect or measure containment isolation valve leakage.

6.2.1.5 Instrumentation Application

Instrumentation is provided in the containment to remotely monitor the following variables:

Pressure

Sump Level

Radiation level

Humidity

The only containment instrument employed to actuate safety systems is the containment pressure monitor. Dependency of the Engineered Safety Features Actuation System upon this signal is described in RESAR Section 7.3. As shown in RESAR Section 7.5, Safety Related Display Information, the containment pressure instrumentation consists of four channels.

6.2.2 CONTAINMENT HEAT REMOVAL SYSTEMS

The functional performance objective of the containment spray system and the ice condenser as safety systems is to reduce the containment pressure and temperature following a loss-of-coolant accident by removing thermal energy from the containment atmosphere. This function is performed in conjunction with the emergency core cooling system. This cooling function of the containment spray system also helps to limit offsite radiation levels by reducing the pressure differential between the containment and the subatmospheric enclosure building, thereby reducing the driving force for leakage of fission products from the containment atmosphere.

6.2.2.1 Design Bases of Ice Condenser System

The following is a summary of the ice condenser design considerations. The design considerations are presented in two categories: performance criteria and structural and mechanical considerations. Also presented are specific criteria for individual components of the ice condenser.

1. The energy absorption capacity of the ice condenser will be at least twice that required to absorb all energy that can be released during initial blowdown of the Reactor Coolant System for any reactor coolant pipe break size up to and including the double ended rupture of the largest pipe in the Reactor Coolant System, or for any steam system pipe break size up to and including the double ended rupture of the main steam line inside the containment, without exceeding the containment design pressure.

2. Following the initial stages of accidents described in (1), the ice condenser together with the Containment Spray System will have sufficient heat absorption capacity to absorb the subsequent heat loads without exceeding the containment design pressure. The assumed subsequent heat loads include Reactor Coolant System stored heat, residual heat and heat removed from the steam generators by the action of the ECCS.

3. Sufficient heat transfer area will be provided in the ice condenser so that the magnitude of the pressure resulting from accidents described in (1) does not exceed the containment design pressure.

4. The lower (Reactor Coolant System) compartment will be bounded by the divider barrier so that essentially all of the energy released in this compartment is directed through doors at the bottom of the ice condenser.

5. The resistance to flow into the ice condenser will be such that the maximum energy input into any section of the ice condenser does not exceed its design capability.

6. Any energy release of insufficient magnitude to cause venting of energy into the ice condenser will be readily absorbed by the Containment Spray System without exceeding the containment design pressure.

7. The inlet doors of the ice condenser will be designed to open and distribute steam to the ice condenser in accordance with design basis (5) above for any postulated loss-of-coolant accident.

8. Both the inlet and outlet doors of the ice condenser will be designed to fail at a low differential pressure (above the design opening pressure) if for any reason the doors are prevented from opening normally.

9. Ice with a suitable concentration of boron will be used in the ice condenser so that in case of an accident, the water resulting from the melted ice can be used for cooling the core.

10. The addition of sodium tetraborate to the ice will provide for adsorption and retention of iodine released from the core.

Structural and Mechanical Design

11. The ice condenser internal structures will be capable of withstanding loading combinations with the following stress limits:

Loading Combinations	Stress Limits
Normal plus Operating Basis Earthquake (OBE) Loads	Within AISC-69 Part 1 specification allowables.
Normal plus Design Basis Earthquake (DBE)	Within yield after load redistributions (Normal AISC-69 specification allowables.)
Normal plus Accident Loads	Within yield after load redistributions (Normal AISC-69 specification allowables.)

Loading Combinations	Stress Limits
Normal plus DBE plus Accident Loads	Limit Curves; WCAP-5890, Rev. 1, plus normal AISC-69 specification allowables.

In addition to the stated stress limits, structural stability and deformation requirements are determined to assure no loss of function under accident and DBE loads.

12. The structure, equipment mountings, supports and joints will be designed to accommodate the maximum temperature range and gradients which will occur.

13. Structural loads will not be transmitted between the ice support structure and the containment.

14. The internals of the ice condenser will be designed to permit some disassembly and reconstruction to facilitate maintenance, if necessary during the lifetime of the plant.

15. The ice condenser internals will be designed for a lifetime consistent with that of the plant.

16. The materials of construction will be selected to be passive under all conditions of operation of the ice condenser. In particular, corrosion will be prevented by inhibitors and protective coatings where necessary and non-metallic materials which are stable and non-volatile.

17. Materials in the ice condenser will be selected to be compatible with the general environmental conditions inside the containment during normal operation or during accident conditions. In particular, the choice of materials for the ice condenser insulation panels will be compatible with the containment.

18. Sufficient mechanical component redundancy will be incorporated in the ice condenser design to provide a high degree of plant availability.

19. Components forming the boundary of the ice condenser will be sealed to limit the passage of air and vapor, except where specific provision is made for venting.

<u>Specific Design Criteria</u>

<u>Ice Support Structure</u>

20. The structure will be designed to maintain the ice in the required array.

21. The structure will allow loading of the ice cylinders in position and permit lifting of complete ice columns for removal in sections. Selected columns will be designed to be weighed for surveillance purposes.

22. Any section of the ice cylinders will be capable of supporting the total weight of ice above that section.

Insulated Duct Panels and Insulation

23. The insulation will maintain the maximum total heat load on the Ice Condenser Refrigeration System at a level consistent with the installed capacity, including margin.

24. The heat input to the ice bed will be minimized so that the ice bed performance capability will be maintained for a period of at least two weeks if the Refrigeration System is shutdown.

In the region of the ice condenser, increased conductivity due to humidity and compression during an accident will not detract from the performance of the ice condenser. The insulation requirements for the ice condenser (under normal operating conditions) will provide more than the necessary insulation to protect the containment from thermal shock under accident conditions, even allowing for increased conductivity due to compression.

25. The galvanized sheet metal covers on the inner faces of the duct panels will be continuously sealed and the outer sheet metal covers adjacent to the crane wall and the end walls will form a vapor barrier. Under loss-of-coolant accident conditions, the vapor barrier will prevent direct contact of the steam with the containment shell. Under normal operating conditions, the vapor barrier will prevent loss of insulation due to humidity.

26. At the boundaries of the ice condenser where air cooling will not be incorporated, insulation will be provided in a form consistent with the functional requirements of those areas.

27. The insulation can be removed and replaced after disassembly of the ice condenser internals, if necessary during the lifetime of the plant. Precompression of the ice condenser insulation from structural and leakage tests does not detract from its performance capabilities.

28. Performance of the insulation will not be affected by earthquake conditions or platform motion.

29. Under accident conditions the insulation will not affect the overall performance of the ice condenser.

30. The materials used for insulation will be compatible with other areas of the containment and systems or portions of systems in the containment.

31. The method of attachment of the panels and their positions relative to the ice cylinders and support structure will preclude displacement of the panels during towing, normal platform movement or accident conditions.

<u>Ice Condenser Doors</u>

Normal Operation

32. The doors will restrict the leakage of air into and out of the ice condenser during normal operation.

33. The doors will restrict the local heat gain in the ice condenser.

34. The doors will be instrumented to provide indication of their closed position.

35. Provision will be made for adequate means of inspecting the doors during reactor shutdown.

36. The doors will be designed to function under all platform motion conditions while the reactor is operational.

Earthquake Conditions

The doors are to be designed to withstand earthquake and platform motion loadings so as not to affect ice condenser operation for normal and accident conditions. These loads are derived from analysis of the containment interior structure.

Accident Conditions

Lower Inlet Doors

37. The doors will be designed to open in the event of a leak which produces a positive differential pressure across the doors in the opening direction of greater than one pound per square foot.

38. The inlet doors and door ports of the ice condenser will be designed to distribute steam to the condenser to limit maldistribution to less than 150 percent of the flow rate for any loss-of-coolant accident which causes the doors to open.

39. The design of the doors will be such that their inertia is consistent with producing a negligible effect on initial pressure.

40. During blowdown, flow area will be provided for the effluent of condensate and melted ice from the ice condenser, without impeding the distributed input of steam.

Intermediate and Top Deck Doors

41. Venting of the lower compartment for leak rates of less than 70 gmp to 240 gpm will be provided by a permanently open vent of 5 square feet through the operating deck.

42. For larger leak rates, the resulting differential pressure will open a sufficient number of doors to permit air flow into the upper compartment.

Ice Condenser Drains

43. The capacity of the drains will reduce the time that a water level in the ice condenser could cause an additional resistance to door opening and limit the potential increase in containment pressure that could occur in the short time that the doors are closed.

44. Drains will be provided with check valves to allow flow only from the ice condenser compartment into the lower compartment.

45. Drain check valves will be spring or gravity loaded to hold shut against the cold air head (1 psf) in the ice condenser during normal operation.

46. Drains will be located to drain the water level in the condenser to an elevation below the bottom of the doors.

Performance Capability

Because of the static nature of the ice bed, the ice condenser function is not susceptible to failure of active components and the resulting consideration of additional capability to accommodate failures.

The door panels located at the inlet and outlet of the ice condenser will be the only elements required to move during the accident. These items are considered as passive or static elements equivalent to rupture discs rather than active components requiring an external signal and energy source to function.

6.2.2.2 System Design of Ice Condenser

Description of Ice Condenser and Components

Included in this section are descriptions of the general arrangement of the ice condenser, the Refrigeration Cooling System, the door panels at the top and bottom of the ice condenser and the ice condenser internals which form the flow channels and ice beds. Table 6.2.2-1 presents principal design parameters for the ice condenser.

6.2.2.2.1 General Arrangement

The section elevation of the ice condenser is shown in Figure 6.2-19.

The ice condenser will be essentially a well-insulated cold storage room in which ice is maintained in an array of vertical cylindrical columns. The columns are to be formed by ice frozen as cylinders with the space between columns forming the flow chanels for steam and air. The ice

TABLE 6.2.2-1

ICE CONDENSER DESIGN PARAMETERS

Weight of ice required (minimum), lbs.	*
Height of main ice bed, ft.	*
Total cross sectional area, ft^2	*
Ice basket diameter, ft.	*
Number of ice basket columns	*
Channel flow area, ft^2	*
Inlet door flow area, ft^2	*
Equilibrium temperature in ice bed, °F	15
Inlet door-opening pressure, psf	1
Lower drain area, ft^2	*
Upper vent area, ft^2	*
Ice boron concentration, ppm	2000
Ice Condenser Support Structure	
Lower Support Structure Platforms	
Number	24
Weight, each, lb.	2400
Lower Support Portal Frames	
Number	25
Weight, each, lbs.	1750

* Refer to Table 6.2.1-1

TABLE 6.2.2-1 (cont'd)

Lattice Frames	
Number	288
Weight, each, lbs.	550
Floor Lattice Frames	
Number	72
Weight, each, lbs.	220
Lattice Frame Support Columns	
Number	146
Weight, lbs.	550
Ice Basket Section	
12 ft. sections	
Number	5832
Weight, each, lbs.	30
9 ft. sections	
Number	648
Weight, each, lbs.	25
<u>Ice Condenser Doors and Frames</u>	
Lower Inlet Doors and Frames	
Number	24
Weight, lbs.	700
Top Deck and Vent Doors	
Number	24

TABLE 6.2.2-1 (cont'd)

Intermediate Deck and Vent Doors	
Number	24
Upper Plenum Access Doors	
Number	2
Lower Void Access Doors	
Number	2
<u>Air Return Fans and Check Valves</u>	
Number	2
Flow, each, scfm	40,000

condenser will be contained in the annulus formed by the containment and the crane wall circumferentially over a 288 degree arc. The refueling canal and equipment hatch are located in the remaining 72 degree arc.

The total height of the ice condenser compartment will extend from below the operating deck to the top of the crane wall. The main ice bed will be above the operating deck with a partial width section at the bottom of the ice condenser to leave space for opening doors. The uppermost section of the ice condenser compartment will form a plenum to accommodate the air cooling equipment and provide access for ice loading and maintenance. A small bridge crane will be provided at the top of the compartment for construction and maintenance purposes. Below the operating deck, the inner wall of the ice condenser incorporates the inlet doors, through which the air in the lower compartment volume and the discharged fluid will pass to the ice condenser. The top deck of the ice condenser will be formed by doors supported from radial I-beams supported by the top of the crane wall. Another set of doors will form a partition between the upper plenum and the ice compartment.

Under accident conditions there are no necessary insulating qualities that must be maintained. The design of the ice condenser wall panels will assure that the panels remain intact and in place to prevent a significant change in the hydraulic diameter under accident loadings. The wall panel structural encasement provides protection for the insulation during the accident. In the region of the walls and floor of the ice compartment, the insulation will incorporate cooling ducts, by which the heat gained is absorbed and transferred to the coolers in the upper plenum. The temperature of the cooling air in the ducts and temperature of the ice will be maintained between 10°F and 20°F.

The heat absorbed by the coolers will be transferred to the refrigeration units outside the containment by the Ice Condenser Refrigeration System.

6.2.2.2.2 Ice Support Structure

The ice support structure consists of the floor, lower support structure, lattice frames and columns and ice cylinders.

The floor will be arranged to provide structural support of the internals and cooling of the ice bed. Concrete piers will be built up from the ice condenser support slab at three radii, corresponding to the inner and outer rows of lower support structure columns and the extreme outer radius of the slab itself. The concrete surface slab is supported by radial steel beams supported by the concrete piers.

The ice condenser structure and ice will be supported by the floor and lower support structure. The lower support structure will support two-thirds of the ice load. The structure will extend above the floor to provide an access area behind the inlet doors. The lower support structure is to be essentially a lattice of radial I-beams on the centerline of the ice columns. These radial I-beams will be supported by inner and outer main beam members running circumferentially around the ice condenser. The main beam members are to be supported by the columns from the ice condenser floor as shown in Figure 6.2-19. The lower support columns will straddle the ice condenser inlet doors providing a clear area for the inlet doors.

Lateral support against seismic and motion effects is to be provided by structural ties to the crane wall. Clearance will be maintained between the ice support structure and the containment to assure that there are no loads transmitted to the containment.

The lattice frames, of welded steel construction, will locate the ice cylinders in the desired array. The radial length of the frames will be adjustable to accommodate construction clearances and permit vertical alignment of the ice condenser locations within the frames. Cross braced box truss sections will be built into insulated duct panels at the ends and midsection. The vertical spacing of these truss sections is to be such that a reinforced section is provided at each lattice frame elevation. Outrigged brackets with insulating pads at the lattice frame elevation will provide the structural connection between the inner duct panel and the inner lattice frame support column.

Box truss sections of the wall panels will be attached to the panel mounting angles with structural angle brackets. These brackets and the wall panel clamping bolts will provide a vertically pinned connection to the crane wall that will transmit the lateral loading to the wall.

Groups of lattice frames will be connected together with a slip joint between groups to provide for circumferential thermal expansion and contraction. Since there is no connection between the structure and the containment shell, no special provisions will be necessary for radial thermal expansion or contraction.

The ice columns are to be composed of part-length right cylinder sections of ice having a wire mesh periphery. The cylinders will be assembled into the lattice frames to form a continuous column of ice the full height of the condenser. The lattice frames provide lateral ice column support at intervals corresponding to the ends of the ice cylinder sections.

The vertical support of the ice and ice cylinder hardware will be transmitted to the floor and lower support structure. Provision will be made for weighing the entire length of selected columns for surveillance purposes.

Design Loads

The ice bed structure will be designed to meet specific loads. Two load categories are defined for design purposes, with all structures designed for platform motions:

1. Service Load (Normal Conditions)

   Service load (normal) conditions are any conditions encountered during construction and in the normal operation of the nuclear power plant. Included in such conditions are any anticipated transients or test conditions during normal or emergency startup and shutdown of the nuclear power plant, and integrity tests of the containment. Also included in this category wave motion are those severe environmental conditions (Operating Basis Earthquake or OBE) which may be anticipated during the life of the Floating Nuclear Power Plant.

   For normal load conditions the lateral ice load is due to platform motion with minor loading due to misalignment of the ice columns and lattice frames. The lattice frame and column structure will be designed to carry vertical components of load from the ice cylinders as required. Horizontal load components from misalignment of the ice columns and lattice frame locations will be reduced by proper adjustment during installation.

The resultant structure considered for analysis comprises a pin-jointed cylinder laterally supported at the joints and vertically at the base. The loading on the cylinder due to the ice load in the vertical direction is assumed to be applied in uniform shear above any section, combined with a hydrostatic load below that section, thus accounting for the worst conditions of ice support.

The lower support structure will carry a total static load from the inner two thirds of the ice bed uniformly distributed. In addition, the inner circumferential main beams will take the weight of the insulated duct panels on the crane wall and the reactions at the feet of the inner columns supporting the lattice frames.

The normal loads on the surface floor slab are due to the uniformly distributed load from the outer rows of the ice basket columns.

For design purposes the ice columns and structure are subjected to other modes of loading outside the scope of the normal, earthquake and accident conditions. These arise from handling during removal or weighing of the columns when it is necessary to consider dynamic loading of a full column of ice due to lifting by the crane. Consideration is also given to minimizing damage in the event that an ice cylinder is dropped or arrested while being lowered by the crane.

In addition to the seismic effects on the ice condenser structure due to the weight of the ice, seismic effects will be transmitted to the structure through the crane wall and floor. These seismic effects will be compared with platform motions under different loading conditions as imposed by wave motion for determination of the maximum horizontal acceleration. The behavior will then be analyzed using a response spectrum, absolute accelerations, displacements and relative motions of the walls as determined from the dynamic analysis of the containment interior structure. The natural frequencies and vibration modes of the ice internals comprising the columns, framework and structural ties to the crane wall are to be determined by dynamic analysis, the internals being supported transversely by the structural ties incorporated in the insulated duct panels at the crane wall.

2. Extreme Load Conditions

Extreme load conditions are those conditions resulting from a postulated single failure of the Reactor Coolant System or Steam System (Design Basis Accident) and those extreme environmental conditions (Design Basis Earthquake) postulated for the design of the Floating Nuclear Power Plant. Also included in this category are those combinations of single failure of the Reactor Coolant System or Steam System plus severe or extreme environmental conditions considered credible.

Accident loads will be considered for the double ended rupture of the largest pipe in the Reactor Coolant System or double ended rupture of the main steam line. This gives rise to an increase in pressure in the ice condenser together with drag forces on the ice columns and support structure due to the velocity of the steam/air mixture. Consideration is to be given to temperature gradients between the top and bottom of the ice condenser which develop from heat transfer between the structural steelwork and steam condensed by the ice.

The maximum differential pressure between the ice condenser and the upper compartment is effectively applied across the insulated duct panels. The wall panels will be designed and fastened to the walls in a manner that precludes significant steam channeling due to panel deflections, or leakage through the vapor barrier.

The vapor barrier joints will be sealed and designed to withstand the full pressure differential. The pressure differential will act on the sealed joints in a manner such as to increase the sealing pressure applied on the sealants.

Specific loads and load combinations comprising the service (normal) load and extreme load condition categories defined above are as follows:

Dead Load (D)

Weight of structural steel and full ice beds at a maximum ice load. For design purposes, the ice shall be assumed to carry no load.

Live Load (L)

Live load includes any manufacturing and maintenance loads, loads during the filling and weighing operation.

Operational Basis Temperature Load (To)

Normal temperature load includes those loads resulting from normal temperature gradients in the ice compartment during operation plus any loads included by the cooling of ice containment structure from an assumed ambient temperature at the time of installation.

Accident Temperature Load (Ta)

Accident temperature load includes those loads resulting from DBA thermal conditions.

Accident Pressure Load (P)

Accident pressure load includes those loads induced by the relative deformation of the containment structure under accident conditions plus any differential pressure loads across the ice beds.

Design Basis Accident Load (DBA)

This load consists of the effects resulting from a loss of coolant or steam feedwater line-break accident.

Operational Basis Earthquake (OBE)

The operational basis earthquake loads are those inertia and differential displacement induced loads including impact loads determined from the response of the ice bed and supporting structure to the OBE defined for the site.

Design Basis Earthquake (DBE)

The design basis earthquake loads are those inertia and differential displacement induced loads including impact loads determined from the response of the ice bed and supporting structure to the DBE defined for the site.

Stress Criteria

Stress limits are divided into three categories for design purposes. Primary stresses and secondary stresses are as defined in Section NB 3210 of the ASME Section III Code.

Service Load (Normal plus OBE) Conditions

Primary stresses shall be limited to normal AISC-69, Part I Specification allowables (S).

Secondary stresses which are evaluated independently are limited to 3 times the (S) allowables.

Extreme Loads Occurring Singly (Normal plus DBE or Normal plus DBA)

Primary stresses shall be limited to normal AISC-69 Specification allowables (S) increased by 50%. Secondary stresses which are evaluated independently of primary stresses are limited to 3 times the (S) allowables.

Extreme Combined Loads (DBE plus DBA)

Primary stresses will be limited to normal AISC-69 Part I Specification allowables increased in the proportion 1.2 Sy/Sb or 0.7 Su/Sb whichever is smaller where Sy is the specified minimum yield of the material being used and Sb is the allowable stress in bending (open sections) from the

AISC specification and Su is uniform ultimate stress capacity of the material. No evaluation of secondary stress is required.

Alternate Analytical Criteria

As an alternate to the stress criteria, Limit Analysis for primary loads as defined in the AISC Specification Part 2 may be used except that for abnormal or extreme environmental load conditions an overall load factor of 1.3 shall be used and for extreme combined loads, an overall load factor of 1.1 shall be used.

Experimental or Test Verification of Design

In lieu of analysis, experimental verification of design using actual or simulated load conditions may be used.

The safety factor associated with such verification will be 1.4 for dead loads and 1.7 for all other loads in the service load category and 1.3 as an overall safety factor for load combinations in the extreme loads occurring singly category and in the extreme combined load category.

6.2.2.2.3 Ice Condenser Cooling Ducts and Insulation

The insulated cooling ducts for the walls of the ice compartment, shown in Figure 6.2-19 will be installed as prefabricated panels. At the outer wall and end walls of the condenser, the panels will cover the full height of the compartment. At the inner wall the panels will be designed to terminate above the inlet door ports.

Each panel will be an integral duct unit, consisting of a cross braced box section for cooling air flow, insulated with fiberglass insulation. The air flow channels will be fed from the Refrigeration System air handling units by an air duct around the periphery of the upper plenum. The return flow will exhaust directly into the upper plenum. The inner, ice compartment side of the panels will be covered with galvanized steel sheet and the joints between panels sealed to provide a vapor barrier between the ice compartment and the cooling air flow. Galvanized steel sheet will also be provided on the outer surfaces of the panels on the crane wall and end walls and form the vapor barrier for the compartment on those walls.

The panels are to be clamped to the walls by studs, clamp washers and nuts. The weight of the panels will be transmitted to the floor at the outer wall of the ice condenser and to the ice support structure at the inner wall.

The fastening of the insulated duct panels to the walls, the construction incorporating sheet metal faces and the additional constraints provided by the configuration of the ice columns and support structure will obviate any mechanism which allows the insulation material to impede the performance of the ice condenser during accident conditions. The ice condenser wall panels are to be specifically designed to remain in position during an accident condition; however, the consequences of one or more panels being displaced are acceptable. A full-scale section test run by Westinghouse Electric Corporation demonstrated acceptable performance of ice condenser for an equivalent by-pass flow area equal to a condition in which all ice condenser wall panels are displaced.

6.2.2.2.4 Ice Condenser Doors

Inlet Doors

The inlet doors at the bottom of the ice condenser will be insulated and mounted on a frame between the concrete pillars supporting the crane wall. The doors will be provided with a loading device which produces a small force to resist door opening. The magnitude of the force produced when the doors are fully open will be equivalent to a differential pressure of approximately one pound per square foot. The doors will normally be held shut by the static differential pressure due to the higher density air in the ice condenser compartment. With zero differential pressure across the doors (no cold air head), the neutral position of the loading mechanism will be set so that the doors are slightly open. Thus, all doors will open at the same differential pressure (equivalent to the cold air head) and, within the limits of the loading device tolerances, the doors will all open to equal extent.

Provision will be made for a drain area of approximately fifteen square feet at the bottom of the ice condenser to permit water and air to flow from the ice condenser during and after the reactor coolant blowdown. This provision will assure that, if doors reclose after a large break before all water drains out, or for small break or residual heat release cases where doors are not fully open, water from melted ice can drain from the compartment. The total drain area will be provided by several individual drains.

For leaks less than approximately 70 gpm to 240 gpm, a pressure difference sufficient to open the inlet doors would not be developed. This range of leakage would be detected by reactor coolant system instrumentation and plant shutdown would be initiated. For this case, the containment ventilation fan-coolers, although not engineered safety features, will have capacity for removal of additional heat input. In any event, the containment pressure would be limited to a low value by the Containment Spray System.

For slightly greater leak rates, zero differential pressure would be developed across the doors (cold air head balanced) and the neutral position of the loading mechanism is set so that the doors would open slightly. Thus all doors would open at the same differential pressure (cold air head); and, within the limits of spring tolerances, the doors would all open by equal amounts.

For larger break sizes, the doors will open by greater amounts consistent with the spring characteristics of the doors. The one pound per square foot differential pressure required to fully open the lower doors would be developed by the steam flow from an approximate eight inch diameter single-ended pipe break. Above this break size, maldistribution will be limited to a low amount by the size of the door ports.

At the conclusion of the Reactor Coolant System blowdown, the doors would tend to reclose. The provision made for a small drain area at the bottom of the ice condenser would allow for the flow of water and air from the ice condenser during and after the blowdown. If it is assumed that residual heat is released to the containment in the form of steam, the doors would reopen as required to allow the steam to flow into the ice condenser.

All ice condenser doors will be designed with substantial margin, in all members and attachments, under maximum dynamic loading conditions. This margin will provide assurance that ice condenser doors will not become missiles.

Top Deck and Intermediate Doors

The top doors will be supported by the ice condenser bridge crane support structure. The support structure will consist of radial beams spanning the ice condenser annulus at the top of the crane wall.

The doors enclosing the ice compartment and forming the floor of the upper plenum will be similar to the doors described above. These doors will be supported by the lattice frame support columns.

A vent area of approximately 20 square feet will be provided through each set of top doors to equalize pressure between the ice condenser and upper compartment volumes during normal operating pressure fluctuations.

Equipment and personnel access doors will be provided at each end of the upper plenum and personnel access hatches will be included in each end of the lower void space for surveillance of the inlet doors during reactor shutdown. These doors are all to be closed during normal operation.

6.2.2.2.5 Refrigeration System

The Refrigeration System will be designed to initially cool down the ice condenser from the ambient conditions of the containment and to maintain the desired equilibrium temperature in the ice compartment. Cooling of the ice condenser will be achieved by a three-stage system.

First Stage - Refrigerant Cycle

Four freon vapor compression refrigeration units located outside the containment will provide coolant for the plant. Three refrigeration units will have a capacity in excess of the maximum load under normal operating conditions. Cooling water will be provided from the non-essential Service Water System.

Second Stage - Glycol Cycle

The second stage working fluid will be a 50 percent solution of ethylene glycol and water. The cycle will carry the heat absorbed from the ice condenser air handling units to the evaporator/coolers of the refrigeration units.

Two circulating pumps will operate in parallel, each capable of providing 100 percent flow capacity for normal operating conditions. The ethylene glycol will be fed to the containment by single supply and return pipes. The piping will be insulated to minimize heat gain and frost formation.

A surge tank will be provided in the system to accommodate volumetric expansion and contraction of the ethylene glycol solution.

Third Stage - Air Cooling Cycle

Air handling units are to be located around the periphery of the ice condenser plenum, spaced to give an even distribution of air flow to the cooling air ducts in the ice compartment insulation panels.

Air will be drawn from the upper plenum through the cooling coils by the fans and fed to a continuous header around the plenum for distribution to the ducts. This arrangement will assure that the cooling function can be maintained by adjacent units should a fan or cooler be shut down.

The ethylene glycol flow in the cooling coils is controlled by air outlet temperature, and the flow shut off during defrost. The coils defrost automatically by individual electric heaters, and flapper-type check valves prevent reverse air flow through the coil when the fans are shut off during defrost.

6.2.2.2.6 Handling Equipment

The bridge crane will move around the annulus outside the crane wall, passing through the ice condenser equipment access doors into and out of the upper plenum. This crane may be used for manufacturing, ice loading, weighing selected ice columns and general maintenance.

The bridge track will be supported from the beams of the plenum top deck structure and similarly from radial beams supported by the crane wall outside the condenser. The crane will incorporate power operated rotating boom and hoist mechanisms.

6.2.2.2.7 Ice Condenser Materials

Corrosion of ice condenser components will be reduced by the low temperature operation of the ice condenser. To further inhibit corrosion galvanizing or other suitable protection is being considered for all metal panels. All structural members will be protected by corrosion resistant paints. At ice condenser operating temperaures, the corrosion resistant paints are expected to last the life of the ice condenser without maintenance. Any corrosion that might develop over long term operation would not impair the performance of the ice condenser. Any ice condenser equipment whose performance might be affected by corrosion will employ corrosion resistant materials for critical components.

Ice bed structural steel member materials will be impact tested to meet the temperature requirements of NE 2300 of Section III Nuclear Vessel Code of ASME, of at least 30°F lower than the lowest service temperature for all section thicknesses in excess of 5/8 inch. For section thickness equal to or less than 5/8 inch, material will be normalized.

The Ice Condenser Refrigeration System will utilize ethylene glycol as a coolant. According to published data, all ice condenser materials selected have good chemical resistance to ethylene glycol.

Galvanized steel sheet will be used for protecting the ice condenser wall panel insulation. This steel sheet will also serve as the vapor barrier on the crane wall and end walls of the compartment. The containment will serve as the vapor barrier on that wall.

The insulation panels in the ice bed region will be provided with a vapor barrier which will prevent moisture from reaching the containment insulation interface region. A small proportion of the air flow bleeds from the ducts into the annulus between the duct panel and the containment. The inner surface temperature at the boundary of the ice condenser, in particular the containment, will vary with external ambient conditions from a maximum of 80°F down to 0°F. For all boundary temperatures above the duct air temperature, any moisture in the insulation will diffuse to the cooling ducts and in addition will be absorbed by the bleed flow of air. When the containment temperature is below duct air temperature, frost will be formed by moisture transferred to the annulus between the containment and the duct panels. Since the temperature gradients are reversed for this condition, the frost will not detract from the performance of the Cooling System

With the above factors considered, removal of the ice condenser wall panels for containment inspection should not be necessary. Access to the containment in the ice condenser region for inspection will be provided by three special removable section duct panels located approximately at quarter points in the lower section of the ice bed. Access to the containment in the plenum region will be provided by removal of a plenum insulation panel.

6.2.2.2.8 Ice Condenser Operating Considerations

Refrigeration System

The ice condenser and associated systems will provide a reliable, static heat sink which is available for the loss-of-coolant accident. The design will assure that the quantity and configuration of the ice is maintained within the limits acceptable for the accident requirements. The insulation system minimizes the total heat gain into the compartment and air leakage flow through the compartment. Diesel generator power will supply the electricity to operate the Refrigeration System during towing and start-up operations. Long-term ice storage tests have shown that the ice can be stored for long periods of time without significant loss or physical distortion.

The Ice Condenser Refrigeration System will be provided with excess capacity to assure that the ice is maintained below the freezing temperature. The capacity of the refrigeration units will exceed the maximum heat gain to the ice condenser compartment, and standby refrigeration will be available for maintenance shutdowns. The ethylene glycol loop will incorporate two full-capacity circulation pumps. The many air handling units in the ice condenser compartment will also provide excess capacity so that the compartment temperature can easily be maintained if the fans are shut down for defrosting the coils, for maintenance of the equipment, or as a result of equipment failure. The air duct feeding the insulated duct panels will be continuous around the periphery of the upper plenum in the ice condenser compartment, so that air flow is maintained as long as any reasonable number of fans are running.

Ice Bed Loading

The ice condenser compartment will be cooled to operating temperature 10 - 20^{O}F and loaded with ice at the manufacturing facility.

Ice Condenser Maintenance

The ice condenser internals will be designed for a lifetime consistent with that of the plant. Specifically, the design of the ice support structure and insulation duct panels will reduce the need for maintenance. However, if maintenance becomes necessary, the internals can be partially or completely disassembled and reassembled. Individual air handling units, upper and intermediate door panels or instrumentation can be removed and replaced without greatly affecting the ice condenser or the plant operation.

Sensitivity to Small Leaks

Consideration has been given to the possibility that a small leak from the Reactor Coolant System could melt ice without the leak or the ice melt being detected. Temperature detectors in the ice condenser and the inlet door position indicators will provide one indication of this condition.

In order for steam to enter the ice condenser, the door opening differential pressure of one pound per square foot, produced by the colder air density in the ice condenser, must first be overcome. Because of the vent holes in the operating deck, the reactor coolant leakage has to be high enough to generate one pound per square foot differential pressure across the deck before the inlet doors would begin to open.

Calculations of the leakage required to generate this differential pressure through the assumed five square foot operating deck leakage area indicate that the reactor coolant leakage would be between 70 gpm and 240 gpm, depending on the mixture concentration of steam and air passing through the deck. These calculations take no credit for heat removal by structures and by the Containment Ventilation System. This range of leakage would be detected by Reactor Coolant System instrumentation, and shutdown of the unit would be initiated.

Coolant leakages less than the range described above would not affect the ice condenser. Coolant leakages greater than this range would be handled by the ice condenser. Leakage through the deck up to the maximum of this range will be handled by the Containment Spray System.

6.2.2.3 Additional Design Considerations of Ice Condenser System

The development of the Ice Condenser Containment System has been proceeding since 1965. Considerable analytical work has been performed and very extensive test programs involving a total of 87 full-scale section tests have been conducted. Tests have been run covering all parameters of importance to determine their effects on ice condenser performance. The results of these tests confirmed the performance capabilities of the ice condenser, and the adequacy of the analytical methods which were developed. These tests were conducted in a full-scale section test facility which was representative of a full-size ice condenser and containment system. These tests therefore closely reproduced the conditions that would occur in a plant in the event of a loss-of-coolant accident.

Tests were performed using a number of different ice condenser configurations to fully determine the effects of all of the important parameters on condenser performance. These parameters are coolant mass and energy release rates; containment volumes and volume ratios; ice condenser dimensions including condenser length, surface area, flow area and ice bed packing fraction; and total blowdown mass and energy. In parallel, analytical models were developed to correlate the results of these tests and to determine the sensitivity to all important parameters. These analytical models were checked against the full-scale section test results and found to be accurate.

The Ice Condenser Containment concept was first presented to the Division of Reactor Licensing of the Atomic Energy Commission on April 15, 1966, and on March 3, 1967. The information presented in those meetings was documented and formally transmitted to the AEC in Westinghouse reports WCAP-2951 and WCAP-7040. An additional report, WCAP-7079, entitled "Preliminary Design and Evaluation of the Ice Condenser Reactor Containment and Associated Engineered Safeguards," was prepared and issued in July 1967. During their review of these reports and their appraisal of the concept, the Division of Reactor Licensing raised a number of questions concerning design details of the ice condenser, testing and related information. Accordingly, on August 29, 1967, Westinghouse presented further information to the AEC directed at answering these questions. This information was subsequently documented in WCAP-7079, Supplement 1, issued in September 1967. Additional presentations of the ice condenser containment concept were made on October 27, 1967, to the Ice Condenser Subcommittee of the Advisory Committee on Reactor Safeguards, and on November 2, 1967, to the full Committee of the ACRS. The information presented in these meetings was documented in WCAP-7079, Supplement 2, issued in December 1967.

Based on the successful results of this extensive development program, the ice condenser was first applied as a part of the reactor containment system design for the Donald C. Cook Nuclear Plant.

The design and performance of the ice condenser containment used in the Donald C. Cook Nuclear Plant was described in a series of reports submitted to the Atomic Energy Commission's Division of Reactor Licensing. WCAP-7183, submitted in March, 1968, presented a detailed description of the ice condenser, detailed information on the ice condenser performance and sensitivity to parameter variations as applied to the design of the Cook Plant, and detailed information on the ice condenser full-scale section tests performed at the Westinghouse Waltz Mill Facility. WCAP-7183 Supplement 1, submitted in July, 1968, provided a description of the mechanical design of the condenser, information on ice condenser and spray system long-term performances, and detailed information on additional Waltz Mill tests. WCAP-7183 Supplement 2, submitted in August, 1969, contained a description and the results of additional ice condenser full-scale section tests and performance analyses, a summary of the long-term ice storage tests, and a description of a method of calculating containment subcompartment transient differential pressures.

Since these reports were submitted, the same basic ice condenser design has been applied to numerous other reactor containment designs including the Sequoyah, McGuire and Watts Bar Plant containments. However, as a result of continued development efforts, an advanced design ice condenser has now been developed and the applicant has applied this ice condenser design to the Floating Nuclear Plant containment.

The parameters used in the design of the ice condenser for the Floating Nuclear Plant containment are well within the ranges tested in the full-scale section test facility and well within the range of applicability of the analytical models which were developed.

6.2.2.3.1 Ice Condenser Door, Vent and Drain Performance

The door, vent and drain performance analyses described below were performed previously by Westinghouse on the Cook Plant, based on containment parameters which differ somewhat from the parameters which are representative of the Floating Nuclear Plant containment. However, the effect of these parameter differences on the containment pressures and performance is insignificant. These analyses will be revised when the detailed design of the Floating Nuclear Plant containment and the containment design parameters have been finalized.

Door Performance - Large Breaks

The basic performance requirement for the ice condenser doors for large break accident conditions is to open rapidly, to ensure proper venting of released energy into the ice condenser. The opening rate and inertia of the inlet doors is therefore important in determining the pressure buildup in the lower compartment due to the rapid release of energy to that compartment.

Full-scale section tests were conducted to determine the effects of door inertia on the initial peak pressure. In these tests, full-size sections of door panels having inertia representative of the doors for the plant were used both at the bottom and at the top of the condenser. Also, for comparison, a test was run without doors but with the representative door port flow area. These tests were run with a blowdown energy of 175% of the equivalent containment energy and a blowdown rate of approximately 110% of the equivalent containment rate.

The measured initial pressure transients were essentially the same for these four tests, and the doors were found to fully open long before the time that the initial peak pressure was reached. Also, the doors have no apparent effect on the pressure transients during their opening time. Thus, the results of these tests indicate that the ice condenser doors have essentially no effect on the initial peak pressure.

At the conclusion of the reactor coolant system blowdown, the inlet doors would tend to reclose. However, for the case where heat is released to the containment in the form of steam, the doors would reopen as required to allow the steam to flow into the ice condenser.

The upper and intermediate doors are not required either to remain open or to close at the conclusion of the blowdown. As discussed in the following paragraph on vent performance, permanently open vents are provided through both the upper and intermediate decks to accommodate reverse air flow from the upper compartment to the ice condenser following the blowdown.

Door Performance - Small Breaks

For small breaks, less than approximately 200 gpm, a pressure difference sufficient to open the inlet doors would not be developed. However, this range of leakage would be detected by reactor coolant system instrumentation, and plant shutdown would be initiated. In any event, the containment pressure would be limited to less than 3 psig by the containment spray system.

For larger break sizes, the doors will open by various amounts consistent with the spring characteristics of the doors. The one pound per square foot differential pressure required to fully open the lower doors would be developed by the steam flow from approximately an 8 inch diameter single-ended pipe break. Above this break size, maldistribution will be limited to a low amount by the size of the inlet door ports.

For small breaks which generate insufficient pressure difference to open the upper and intermediate doors, vents are provided to accommodate air flow through the ice condenser to the upper compartment. Also, for small breaks, drains are provided in the floor to accommodate the flow of condensate and melted ice and prevent the accumulation of water in the ice condenser. Vent and drain performance are discussed in the following paragraphs.

Vent Performance - General

As stated previously, the upper and intermediate doors are not required to remain open following the reactor coolant system blowdown and also are not required to open for small breaks. For these situations, vents are provided through both the upper and intermediate decks to allow air to flow into or from the ice condenser compartment as required. For normal venting transients, the vent air first passes into the fan cooler plenum at the top of the ice condenser compartment where it mixes with the cooling air. The temperature and humidity of the vent air that passes from this plenum into the ice condenser compartment is therefore about the same as the average temperature and humidity of the air in the condenser compartment.

Accordingly, sublimation or frosting in the ice bed due to this vent flow of air will be limited to a negligible value. Specific performance evaluations for the vents are given below both for large breaks and for small breaks.

Vent Performance - Large Breaks

During the period of pressure decay following the reactor coolant system blowdown, the vents are designed to allow air to return from the upper compartment into the ice condenser without imposing an excessive pressure drop across the upper and intermediate doors. The maximum pressure decay rate and therefore the maximum reverse flow rate of air results from the case where reactor residual heat is not released to the containment following the reactor coolant system blowdown. The pressure decay for this case was measured in a full-scale section test. In this test, the pressure decayed from 6.2 psig to 4.5 psig during the one-minute period immediately following the reactor coolant system blowdown. From this pressure decay rate, the plant equivalent flow rate of air was calculated to be 98 lbs/sec. This flow rate would develop a pressure drop across each of the upper doors of about 0.1 psi, which is well within the structural capability of the upper and intermediate doors and support structures. Further, in this calculation it was conservatively assumed that no air flowed through the deck or through the containment air recirculation fan duct.

Vent Performance - Small Breaks

The full-scale section tests have indicated that for breaks less than approximately 5000 to 10,000 gpm, only a fraction of the air is displaced from the lower compartment through the ice condenser and into the upper compartment. Since the air flow rates resulting from these breaks generate less than the 4 psf pressure difference required to open the upper and intermediate doors, a 20 ft^2 vent area was provided in both the upper and intermediate decks. This vent provides a low resistance flow path for air through the ice condenser to the containment upper compartment, thereby limiting the pressure difference across the deck and the resulting bypass flow of steam to the upper compartment for these break conditions.

Drain Performance - General

Drains are provided at the bottom of the ice condenser compartment to allow the melt/condensate water to flow out of the compartment during a loss-of-coolant accident. These drains are provided with check valves that are counter-weighted to seal the ice condenser during normal plant operation and to prevent steam flow through the drains into the ice condenser during a loss-of-coolant accident. These check valves will remain closed against the cold air head (1 psf) of the ice condenser.

Drain Performance - Large Breaks

A number of tests were performed with the flow proportional-type doors installed at the inlet to the ice condenser and the hinged-type door installed at the top of the condenser. Tests were conducted with and without the water drain area, equivalent to 15 ft^2 for the containment, at the bottom of the condenser compartment.

These tests were performed with various blowdown conditions including the maximum containment-equivalent blowdown rate, an initial low blowdown rate followed by the maximum rate, a low blowdown rate alone, and the maximum containment-equivalent blowdown rate followed by the simulated core residual heat rate.

The results of all of these tests indicated satisfactory performance of the doors, vent, and drain for a wide range of blowdown rates. Also, certain tests demonstrated the insensitivity of the final peak pressure to the water drain area. In particular, the results of these full-scale section tests indicated that, even for the maximum blowdown rate, and with no drain area provided, the drain water did not exert a significant back pressure on the ice condenser lower doors. These results indicated that a major fraction of water drains from the ice condenser compartment by the end of the blowdown and demonstrate that the containment final peak pressure is not affected by drain performance.

Drain Performance - Small Breaks

For small breaks, water will flow through the drains at the same rate that it is produced in the condenser. Therefore, the water on the floor of the compartment will reach a steady height which is dependent only on the energy input rate.

To determine that the 15 ft^2 drain area was satisfactory to prevent water accumulation in the ice condenser, the water height was calculated for various break sizes up to a 30,000 gpm break which fully opens the doors. The maximum height of water required for this break was calculated to be 2 ft above the drain check valve. Since this height corresponds to a water level of approximately 1 ft below the bottom elevation of the inlet doors, it was concluded that water will not accumulate in the ice condenser for small breaks and that a 15 ft^2 drain will give satisfactory performance.

6.2.2.3.2 Flow Distribution

Because the ice condenser inlet doors are located all around the lower compartment and the different reactor coolant loops are located in certain regions of the lower compartment, the flow of steam from a pipe break would not be evenly distributed to all sections of the ice condenser. For a break in one end of the lower compartment, the steam would flow preferentially to nearby sections of the ice condenser. However, the steam flow into any section of the ice condenser is limited by the selected flow resistance of the ice condenser inlet doors. The resistances of the doors are large relative to the resistance to flow around the Reactor Coolant System compartment, thereby limiting maldistribution to an acceptable level.

Flow distribution will be calculated for a number of cases to determine sensitivity to parameter variations. The particular parameters to be studied are:

1. Reference flow distribution with different ice condenser door sizes in different sectors of the ice condenser.

2. Sensitivity of the flow distribution to the break location.

3. Sensitivity of the flow distribution to the variation in flow resistance.

Flow resistances will be calculated so that the maximum steam flow into any section of the ice condenser is limited to less than 150 percent of the average.

Test results indicate that any section of the ice condenser can adequately handle a significantly greater amount of steam than that indicated above.

Also, large variations in both steam flow rate and total quantity of steam into any section of the condenser has little effect on the final peak pressure.

6.2.2.4 Tests and Inspections of Ice Condenser System

A comprehensive program of testing was formulated for all elements vital to the functioning of the ice condenser. The program included performance tests of the ice condenser and doors under actual coolant energy release conditions in the ice condenser full-scale section test complex at the Westinghouse Waltz Mill facility. The program has included inspection and testing of the installed ice condenser before and after the initial ice loading prior to initial plant startup, and inspection and testing throughout the plant operating lifetime. The ice condenser design includes suitable provision for visual inspection of the ice beds, flow channels, door panels and cooling equipment. Samples of the ice are to be taken to check the boron concentration. During periods when the reactor is shut down and door panels are being inspected, the door opening force will be tested and compared with the design force.

In order to assure satisfactory performance in the event of a loss-of-coolant accident, the following inspections and monitoring will be performed:

1. Total weight of ice initially installed in the condenser will be determined by weighing the loads of ice being installed. Total weight of ice is to be at least 2.75×10^6 lbs.

2. Flow passages and ice beds typical of all sections of the condenser will be visually inspected to check that no significant changes are occurring.

3. Selected full-height ice columns are to be weighed to check that no significant changes are occurring. Total weight of installed ice to be at least 2.45×10^6 lbs.

4. Each inlet door assembly will be visually inspected, tested for spring load - door position, for cold head position and door switch operability after installation but prior to ice condenser cooldown. These data will be compared to manufacturer's test data. Doors out of tolerance will be readjusted.

5. Each inlet door assembly will be visually inspected, tested for the required opening force, cold head position, and door switch operability after ice condenser cooldown but prior to ice loading, prior to startup, and during each refueling period.

6. Visual inspection for general condition, frosting, corrosion, and gasket condition. Inspection items are the door panels, spring mechanism, gaskets, hinges, and door position limit switches.

7. The chemical composition of the ice will be checked by chemical analysis.

8. Temperatures in selected locations in the ice condenser are to be monitored continuously to assure that the Cooling System is operating satisfactorily.

9. A Monitoring System will be provided to indicate that the door panels are properly closed.

10. Prior to plant heatup, the hatches in the operating deck will be inspected to assure that they are properly secured.

After the ice condenser has been cooled and loaded with ice, a period of several months will elapse before the Reactor Coolant System is heated to temperature for initial power operation. During this period, a number of inspections will be made of the ice condenser and its systems. After power generation begins, the surveillance program will continue, but at a reduced frequency, depending on the information gained from pre-power inspections. Access to the upper plenum of the ice condenser compartment will be possible while the reactor is at full power; therefore, inspections 2, 3, and 7 (outlined above) will be possible at any time.

6.2.2.5 Ice Condenser Instrumentation System

This section describes the instrumentation employed for the monitoring of the ice condenser during both normal plant operation and post-accident operation.

Ice Bed Temperature Monitoring

Resistance thermometer bulbs will be provided for mounting on ice bed probes throughout the ice bed, with clamps for attachment to lattice frames. These bulbs will tie into one scanner unit suitably enclosed for mounting inside the Containment. A recorder is to be mounted in the main Control Room, and a deviation from the limit of ice bed equilibrium temperature will be annunciated.

Lower Inlet Door Position Indication

Limit switches will be mounted on the lower inlet door frames. The position and movement of the switches will be such that the doors must be effectively sealed before the switches are actuated. A single annunciator window in the Control Room will give a common alarm signal when any door is open.

Equipment and Personnel Access Doors

Limit switches will be provided to monitor the position of the equipment access door and the personnel access door with two switches per door. These switches will be fitted in a single series circuit providing Control Room indication of the position of the doors by an alarm on the annunciator panel.

6.2.2.6 Design Basis of Containment Spray System

Adequate heat removal capability for the containment is provided by four 1/3 capacity separate containment spray system trains whose components operate as described in paragraph 6.2.2.7.

The primary purpose of the spray system is to spray cool water into the containment atmosphere when appropriate in the event of a loss-of-coolant accident (LOCA) and thereby assure that containment pressure does exceed 15 psig. This protection is afforded for all pipe break sizes up to and including the double-ended rupture of the largest pipe in the Reactor Coolant System (RCS). Pressure transients following a loss-of-coolant accident are discussed in paragraph 6.2.1.3. The Containment Spray System (CSS) design is based on the conservative assumption that core residual heat is continuously released to the containment as steam, eventually melting all ice in the ice condenser (See paragraph 6.2.2.2 for ice condenser description). The heat removal capability of the spray system is sized to keep the containment pressure below design pressure after all the ice has melted and steam generated in the core continues to enter the containment. Residual heat generated as steam from the time of the accident through the time that the ice melts is absorbed by the ice condenser. In addition to energy from the RCS blowdown, the ice condenser accommodates the release of stored metal heat and steam generator heat until the end of ice melt.

Containment spray system arises from the AEC imposed requirement to consider a small RCS break occurring prior to a double-ended RCS pipe break. If the break is small enough to preclude opening of the ice condenser inlet doors, a pressure of 3 psig will develop in the containment upper compartment. The containment spray is activated by the "P" signal at this stage and must be able to absorb further steam leakage through the operating deck at a maximum long term deck differential pressure of 1 psf (the ice condenser door opening differential pressure). This ensures that if a large break occurs subsequent to the small break the containment design pressure will not be exceeded. The capability of the containment spray system to perform this secondary function is inherent if the primary design requirement, discussed above, is met.

The CSS is designed to operate over an extended time period following a loss-of-coolant accident while satisfying the single failure criteria.

System active components are redundant. System piping located within the containment is redundant and separated. The piping and components of each spray system train, located outside the containment, are shielded in separate compartments.

6.2.2.7 Description and Operation of Containment Spray System

Adequate containment cooling is provided by the CSS, whose components operate in the following sequential modes:

1. Spraying a portion of the contents of the refueling water storage tanks into the containment atmosphere using the containment spray pumps.

2. Recirculation of water from the containment sump by the containment spray pumps through the containment spray heat exchangers and back to the containment after the refueling water storage tanks have been drained to their low level alarm point.

The CSS is shown on Figure 6.2-20. Each train of the CSS consists of a spray pump, spray heat exchanger, valves, piping and spray headers in the upper containment volume.

The coincidence of two out of four containment high-high pressure signals automatically initiates the operation of the spray system. This signal starts the pumps and opens the discharge valves to the spray headers.

During the injection phase, the Containment Spray System pumps take suction from the two refueling water storage tanks. The Emergency Core Cooling System pumps also take suction from these tanks.

These two 50 percent capacity tanks, each have a total capacity of 200,000 gallons and a usable capacity of 175,000 gallons. The total usable capacity of both tanks (350,000 gallons) is based on the requirement for filling the reactor cavity and refueling canal during refueling. Additional design data are shown on Table 6.2.2-2. The total capacity provides an amount of borated water to assure:

1. A volume sufficient to refill the reactor vessel above the nozzles after a loss-of-coolant accident and for requirements of initial operation of Containment Spray System, and

2. A sufficient volume of water in the containment sump to permit the initiation of post-accident recirculation.

The water in the tanks is borated to a concentration which assures reactor shutdown by approximately ten percent $\Delta k/k$ when all rod assemblies are inserted and when the reactor is cooled down for refueling.

The piping from the refueling water storage tanks consists of interconnecting piping between the tanks and distribution headers to the engineered safeguard compartments which house the Emergency Core Cooling and Containment Spray System pumps.

Motor-operated valves are provided in the Emergency Core Cooling and Containment Spray System for isolation of the lines from the refueling water storage tanks. In addition, check valves are provided to prevent reverse flow to the tanks.

These lines remain in operation until the low level is reached in the refueling water storage tanks, at which time post-accident recirculation from the containment sump is initiated.

The piping from the four containment sumps consists of four separate lines routed in separate safeguard tunnels to the pumps located in the aforementioned safeguard compartments. After the refueling water storage tank water is depleted, these lines are brought into operation to provide suction supply for continued core cooling and spray operation. Separation of the lines provides the capability to isolate a leaking line and assures reliable supply for continued operation of the Emergency Core Cooling and Containment Spray Systems during post-accident recirculation and spray operation. Motor-operated valves are provided in the Emergency Core Cooling and Containment Spray Systems at the suction of the pumps for isolation of the safeguard suction lines from the sumps. A containment sump motor-operated isolation valve is provided in each of the lines. For transfer to post-accident recirculation, these valves are opened by the operator in the control room. Power is supplied to the valves via engineered safety buses.

The passive portions of the spray system located within the containment are designed to withstand, without loss of functional performance, the post-accident containment environment conditions and to operate without benefit of maintenance. All of the active components of the CSS are located outside the containment and are not required to operate in the steam-air environment produced by the accident.

The spray headers which are located in the upper containment volume are separated from the reactor and reactor coolant loops by the operating deck and inner wall of the ice bed. These spray headers are, therefore, protected from missiles which may originate in the lower compartment.

Pumps

The containment spray pumps are of the horizontal, centrifugal type driven by electric motors. These motors are supplied from engineered safeguards buses.

TABLE 6.2.2-2

REFUELING WATER STORAGE TANK DESIGN DATA

Number	2
Material	Stainless steel
Total capacity (each), gallon	200,000
Usable capacity (each), gallon	175,000
Design pressure	Atmospheric
Design temperature, °F	120
Boron concentration (as boric acid), nominal, ppm	2000

The design head of the pumps is sufficient to continue at rated capacity with a minimum level in the refueling water storage tank against a head equivalent to the sum of the design pressure of the containment, the head to the uppermost nozzles, and line and nozzle pressure losses. Pump motors are direct-coupled and large enough for maximum power requirement of the pump. Materials of construction suitable for use in mild boric acid solutions (such as stainless steel or equivalent corrosion resistant material) are used. Design parameters are presented in Table 6.2.2-3.

Heat Exchangers

The containment spray heat exchangers are of the shell and tube type. Borated water from either the refueling water storage tank or the containment sump circulates through the tubes while essential raw water circulates through the shell side. The spray heat exchangers are designed to assure adequate heat removal capacity from the spray water in the recirculation mode. Design parameters are presented in Table 6.2.2-4.

Spray Nozzles

The spray nozzles, of the ramp bottom design, are not subject to clogging by particles less than 1/4 inch in maximum dimension, and are capable of producing a mean drop size of approximately 700 microns in diameter with the spray pump operating at design conditions and the containment at design pressure. Fluid is pumped to the nozzles and screened through a

TABLE 6.2.2-3

CONTAINMENT SPRAY PUMP DESIGN PARAMETERS

Quantity, per unit	4
Design pressure, psig	300
Design temperature, °F	250
Design flow rate, gpm	2400
Design head, ft.	500

NOTES:

1. The pump motors are direct coupled and non-overloading to the end of the pump curve.

2. Adequate NPSH will be supplied to the containment spray pumps by properly locating the pump at a suitable elevation and by minimizing the pressure drop in the suction lines. As specified in Safety Guide 1, adequate NPSH will be available assuming the maximum expected fluid temperatures and assuming no increase in containment pressure.

TABLE 6.2.2-4

CONTAINMENT SPRAY HEAT EXCHANGER DESIGN PARAMETERS

Quantity, per unit		4
Type		Shell and Tube
Heat transfer per unit, Btu/hr.		5.598×10^7
	Shell Side	Tube Side
Flow, gpm	3000	2400
Inlet Temperature, °F	85	165
Outlet Temperature, °F	123.2	117.2
Material	Carbon steel	Stainless steel
Design pressure, psig	150	300
Design temperature, °F	200	250

1/4 inch mesh. The nozzles and headers are so oriented as to assure full coverage of the total containment upper volume.

Material Compatibility

Parts of the system in contact with borated water are stainless steel or an equivalent corrosion resistant material. Exposed surfaces within the containment are not subject to breakdown under exposure to the containment spray and can withstand, without spalling or flaking, the steam pressure and temperature associated with a LOCA. Insulation within the containment will not contain material which could potentially enter the sump and cause plugging of Safety Injection System and containment spray strainers and equipment.

6.2.2.8 Additional Design Considerations of Containment Spray System

The containment spray systems provide four 1/3 capacity heat removal systems for the containment, sized to prevent an increase of the containment pressure above design limits.

The spray system design is based on the spray water being raised to the temperature of the containment in falling through the steam-air mixture within the building. The minimum fall path of the droplets is approximately 75 feet from the spray headers to the operating deck. The actual path is longer due to the trajectory of the droplets sprayed out from the header. Heat transfer calculations, based upon 700 micron droplets, show that essentially 100 percent thermal equilibrium is reached in a distance of a few feet. Thus, the spray water reaches essentially the saturation temperature.

A failure analysis has been made on all active and passive components of the spray systems to show that the failure of any single active component will not prevent fulfilling the design function. This analysis is summarized in Table 6.2.2-5.

6.2.2.9 Tests and Inspections of Containment Spray System

All active components in the CSS are adequately tested both in pre-operational performance tests in the manufacturer's shop and in-place testing after installation.

Permanent test lines for the containment spray trains are located so that all components up to the isolation valves upstream from the spray nozzles may be tested. These isolation valves are checked separately. The air test lines, for checking that spray nozzles are not obstructed, connect downstream of the isolation valves.

TABLE 6.2.2-5

SINGLE FAILURE ANALYSIS

CONTAINMENT SPRAY SYSTEM

Component	Malfunction	Comments & Consequences
Spray Nozzles	Clogged	The large number of nozzles makes the clogging of a significant fraction of nozzles highly unlikely.
Spray Pump	Stops running or fails to start	Four 1/3 capacity pumps provide redundancy.
Heat Exchangers	Tube leak	Four 1/3 capacity heat exchangers provide redundancy.
Valve	Fails to open	Four 1/3 capacity flow paths provide redundancy.

An initial test of the containment spray header system will be made to check the free flow out of the spray nozzles. An air supply will be used to provide the large quantity of relatively low pressure air. It will be connected to a flange on the spray piping. During normal station operation, a blind flange will be placed over the air connection which will be located in such a position in the piping that leakage from the flange will not significantly affect the system operation.

The spray nozzles can be periodically tested with air or smoke using the same method as in the initial test.

6.2.2.10 Instrumentation and Application

This section describes the instrumentation employed for the monitoring and actuation of the CSS. This instrumentation includes that which is used during normal and post-accident operation for measurement of temperature, pressure, and flow.

Containment Spray Heat Exchanger Inlet Temperature

Local indication is provided for each heat exchanger to measure fluid temperature in the CSS inlet.

Containment Spray Heat Exchanger Outlet Temperature

A bulb-type temperature transmitter is provided for each heat exchanger to measure fluid temperature in the CSS outlet. The temperature will be recorded in the main control room.

Containment Pressure

Four pressure transmitters are used to measure containment pressure. These four pressure transmitters through bistables actuate the CSS on high-high containment pressure. High-high pressure alarms and pressure indication will be located in the control room.

Containment Spray Heat Exchanger Discharge Flow

A flow channel is provided for each CSS pump discharge. A flow indicator is provided on the main control board for each channel.

Containment Spray Pumps Inlet Pressure

A local pressure indicator is provided on the inlet side of each pump.

Containment Spray Pumps to Refueling Water Storage Tank Line Flow

A local flow indicator, used for pump testing, is provided to show flow in the line from the pumps to the storage tank.

Refueling Water Storage Tank Level

Two channels are provided. One channel provides a local indication and low level alarm function. The second channel provides indication on the main control board, and two low level alarms (one alarm is a normal operating low level and the other is a low-low level alarm).

6.2.3 CONTAINMENT AIR PURIFICATION AND CLEANUP SYSTEM

The containment air purification and cleanup system consists of the following:

1. Normal containment preaccess filtration and purge system.

2. Post accident containment venting system.

6.2.3.1 Design Bases

Normal Containment Preaccess Filtration and Purge System

This system provides recirculation and filtration of the air throughout all containment areas during normal operation and prior to access. This system is also capable of introducing 100% outside air having sufficient capacity to provide 1.5 changes of the containment air volume in one hour. Air exhausted to the plant vent is filtered, monitored, and diluted to an annual average not to exceed 6×10^{-8} μ Ci/cc at the site boundary.

Post Accident Containment Venting System

This system is designed in compliance with the AEC's Safety Guide 7 (Control of Combustible Gas Concentration in Containment Following a Loss-of-Coolant Accident) and provides the capability for a controlled purge of the containment atmosphere through appropriate fission product removal systems thereby maintaining the hydrogen in the containment atmosphere below maximum specified limits. This system is a backup to the hydrogen recombiners.

6.2.3.2 System Design

Normal Containment Preaccess Filtration and Purge System

This system is shown in Figure 9.4-6. Equipment data are given in Table 6.2.3-1 entitled, "Containment Ventilation System Equipment Design Parameters."

The system consists of two 50% capacity supply fans, each equipped with filter and heating coils and two 50% capacity exhaust fans and associated filter trains, comprising prefilters, high efficiency particulate air filters and activated charcoal filters. The activated charcoal filter trains are aligned with the operation of one 50% capacity exhaust fan. The regular HEPA filter trains are aligned with two exhaust fans. All active components are located outside the containment walls through which supply and exhaust air ducts serve the three major regions, namely, upper compartment, lower compartment, and instrumentation room. Each penetration is isolated from the containment by two normally closed valves.

TABLE 6.2.3-1

CONTAINMENT VENTILATION SYSTEM
EQUIPMENT DESIGN PARAMETERS

	SUBSYSTEM						
	Purge Supply (1)	Purge Exhaust (1)	Upper Compartment	Lower Compartment (2)	Containment Instrumentation (3)	Control Rod Drive Mechanism	Reactor Vessel Support
Number	2	2	2	4	2	4	2
Fan capacity, scfm, each	18,000	18,000	20,000	130,000	8,000	20,000	15,000
Filtering	P	P,A,C	P	P	P	-	-
Design Conditions:							
Source, Summer, °F	-	-	110	120	90	150	150
Source, Winter, °F	-	-	60	60	60		
Outside,Summer, °F	95	-	95	95	95	-	-
Outside,Winter, °F	0	-	0	0	0	-	-
Redundancy:							
Running	2 as required	2 as required	2	3	1	3	1
Standby	-	-	-	1	1	1	1

NOTES:
(1) Purge rate 1.5 air changes/hr. Charcoal filter capacity for each unit is 9000 cfm.
(2) 140°F inside primary shield
(3) Vents to purge system.

LEGEND:
P Prefilter or filter, efficiency 50%, NBS, Dust Spot Test.
A Absolute or high efficiency particulate air filter, efficiency 99.97% on .3 micron DOP aerosol.
C Activated charcoal filter, Type BC 727 efficiency 95% and 99% for methyl iodine & iodine, 70% RH 45 FPM.

Post Accident Containment Venting System

The Post-Accident Containment Venting System Flow Diagram as shown in Figure 6.2-21 consists of one full capacity line through which air can be admitted to the containment by utilizing a compressor, one full capacity exhaust line through which hydrogen bearing gases may be vented from the containment through HEPA and charcoal filters to the atmosphere, and associated valving and instrumentation. Equipment data are given in Table 6.2.3-2. The system is a backup system to the Hydrogen Recombiner System (subsection 6.2.5).

Both supply and exhaust lines in the containment are located in missile protected areas over the total length of the piping run, and are terminated so as to prevent either spray or sump water from entering the pipe. Pipes and valves used for the supply line are carbon steel. Pipes and valves used for the exhaust lines are stainless steel.

6.2.3.3 Design Evaluation

Normal Containment Preaccess Filtration and Purge System

Normal containment and preaccess filtration and purge system duct work inside the containment is ANS Safety Class 3. Ductwork between containment isolation valves is ANS Safety Class 2.

TABLE 6.2.3-2

POST-ACCIDENT CONTAINMENT VENTING SYSTEM EQUIPMENT

Compressor		
Number		1
Air flow at discharge pressure, cfm		250
Discharge pressure, psig		10
Horsepower required, bhp		10.65
Aftercooler		
Number		1
Heat removal, BTU/hr		18,800
	Shell Side	Tube Side
Air, cfm	250	
Inlet air temperature, °F	165.6	
Outlet air temperature, °F	100.0	
Water, gpm		7.5
Inlet Water temperature, °F		95
Outlet Water temperature, °F		100
Materials	Carbon steel	Copper

	Roughing	HEPA	Charcoal
Filters			
Number	1	1	1
Air flow, scfm	1000	1000	1000
Maximum differential pressure (clean), in. w.g	0.25	1.0	1.0
Maximum differential pressure (loaded), in. w.g.	0.5	2.0	1.0

Operation of the system is initiated manually in response to containment radiation level monitor readings taken in the control room. The air flow path is established from the control room to provide the recirculation of air via activated charcoal filters. The system also has the capability of providing a once through air purge function.

Post Accident Containment Venting System

Piping, fittings, and valves in the supply line and in the exhaust line starting inside the containment and proceeding up to and including the isolation valves are ANS Safety Class 2. Equipment and piping beyond the isolation valves do not have ANS Safety Class because the rate of hydrogen accumulation in the containment is low enough to permit repair of faulty equipment either before or during operation of the venting system.

Based on the containment hydrogen concentration and the hydrogen generation rate, the operator will determine the flow rate required to maintain the hydrogen concentration below maximum limits by venting the containment. Normally this will entail about one hour daily. The operator will then determine the containment pressure necessary to obtain the required vent flow. Air will be pumped into the containment until the required containment pressure is reached. The air supply will then be stopped and the supply line isolated.

Because the addition of air to pressurize the containment will also dilute the hydrogen, the containment will remain isolated until analysis of samples indicate that the concentration is again approaching the maximum limit. Venting will be performed when meteorological conditions are favorable by opening the containment exhaust line to the plant vent, and adjusting the hand controlled throttling valve to obtain the required flow. Operation will continue under these conditions for about one hour, and then the system will be shut down. The vent line will be isolated.

6.2.3.4 Test and Inspections

Tests and inspections will be performed to demonstrate the capability of components and the system to perform their assigned functions.

Shop Testing

Tests will be required to demonstrate the following:

1. Carbon filter capabilities for removal of molecular iodine-131 and methyl iodide-131.

2. Carbon filter iodine collection capability

3. Carbon filter cell leak tightness integrity

4. Carbon filter flow resistance.

5. HEPA filter efficiency and resistance test to meet UL label requirements.

Fan performance characteristics tested according to AMCA test procedures.

System Testing and Inspection

Operational testing will be performed in accordance with Section 14.1 prior to initial startup to demonstrate proper functioning of the system. Testing will include the following:

1. Leak tightness tests of components and systems.

2. System functional test

Thereafter, periodic tests and inspections will be performed to demonstrate system readiness and operability. For the purpose of periodically testing the retentive capability of the carbon filter, representative test carbon samples will be placed in the filter housing in locations which allow the samples to be subjected to the same air flows as the carbon bed. The samples will be periodically removed and tested.

6.2.3.5 Instrumentation and Application

Normal Containment Preaccess Filtration and Purge System

The following instrumentation is utilized:

1. Air Flow Alarm - Located in control room.

2. Containment Pressure - Located in control room. Provision is made for locally regulating the air supply fan inlet air dampers.

3. Exhaust Fan Pressure Differential - Located in control room and locally.

4. Fire Detector Alarm - Located in control room and locally, detectors in duct systems with provisions for automatic shutdown of air supply systems.

5. "S" Signal Fan Shutdown - Automatic with manual override in control room.

6. High-Low Temperature Alarm - Located in control room.

7. Filter Pressure Differential - Local indication.

Post Accident Containment Venting System

1. Temperature Indicators

   Temperature indicators located in the supply and exhaust lines give local readout of temperatures of the hydrogen free air being admitted into the containment and of the containment air being exhausted.

2. Flow Indicators

   Flow meters located in the supply and exhaust lines provide readout of process flow rates locally and in the control room.

3. Pressure Indicator

   A pressure indicator with local and control room readout is provided in the exhaust line and used to determine that the containment pressure is suitable for operating the venting system.

6.2.4 CONTAINMENT ISOLATION SYSTEM

6.2.4.1 Design Basis

The Containment Isolation System performs isolation of the various fluid lines which penetrate the containment to prevent release of radioactivity to the environment following a loss-of-coolant accident (LOCA).

In general, all lines which penetrate the containment are provided with isolation valves in compliance with AEC General Design Criteria 54, 55, 56 and 57.

Criterion 54 - Piping Systems Penetrating Containment

Criterion 54 addresses requirements for leak detection, isolation and testing of systems penetrating the containment, with respect to redundancy, reliability and performance capabilities.

The piping systems penetrating the containment are provided with leak detection, isolation and containment capabilities incorporating redundancy, reliability, and performance capabilities that reflect the importance to safety of isolating the systems. A test program provides for periodic testing to determine if valve leakage is within allowable design limits and to verify operability of the isolation valves.

Criterion 55 - Reactor Coolant Pressure Boundary Penetrating Containment

Criterion 55 addresses lines which are a part of the reactor coolant pressure boundary that penetrate the containment.

The reactor coolant pressure boundary, as defined in ANS N18.2, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants", Section 5.4, is that boundary containing reactor coolant under operating temperature and pressure conditions. The entire reactor coolant system pressure boundary, as defined above, is located within the containment. Thus, there are no lines that form a part of the reactor coolant pressure boundary that penetrate the containment.

Criterion 56 - Primary Containment Isolation

Criterion 56 addresses lines that connect directly to the containment atmosphere and penetrate the containment.

These lines are provided with isolation in compliance with Criterion 56 as follows, with exceptions. The exceptions are addressed to demonstrate that the isolation provisions are acceptable.

1. One locked closed isolation valve inside and one locked closed isolation valve outside containment, or

2. One automatic isolation valve inside and one locked closed isolation valve outside containment, or

3. One locked closed isolation valve inside and one automatic isolation valve outside containment, (a simple check valve is not used as an automatic isolation valve outside containment), or

4. One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve is not used as the automatic isolation valve outside containment.

Exceptions to Criterion 56 and the alternate isolation means provided and design basis are as follows:

1. Containment Pressure Sensing Lines.

   These instrument lines serve an engineered safeguards function. Due to the particular importance of the pressure measurements with respect to safety, alternate double barriers are provided which avoid installing isolation valves in the lines as required by Criterion 56. Sealed instrument sensing lines are utilized. Each pressure transmitter, located adjacent to the outside of the containment, is connected to sealed bellows located adjacent to the inside surface of the containment shell by means of a sealed fluid filled tube. This arrangement provides automatic double barrier isolation without operator action and without sacrificing reliability with regard to the safety function of the lines.

2. Containment Sump Recirculation Lines

The containment sump recirculation lines are utilized during the post-accident recirculation phase of emergency core cooling and Containment Spray System operation. The lines from the containment sump are each provided with a single remote manual gate isolation valve. This valve is surrounded by a chamber which is leaktight at containment design pressure. This chamber is located outside of the Shield Building. The portion of the line from the sump to the valve is enclosed in a concentric guard pipe which is also leaktight at containment design pressure. The isolation valve provides a barrier outside the containment to prevent loss of sump water should a leak develop in the recirculation line. Should a leak develop in the valve body or in the line between the sump and the valve, the fluid is contained by either the leaktight chamber or the guard pipe. Since the sump line connection to the sump is submerged under post-accident conditions, there is no requirement for a valve inside the containment. The arrangement meets all safety requirements and any additional isolation valves are unnecessary.

3. Post-Accident Containment Sampling Lines

Two manual isolation valves, locked closed except during post-LOCA containment atmosphere sampling, are provided in each line outside the containment. The double valve barriers provide the necessary isolation to prevent release of radioactivity to the environment.

Criterion 57 - Closed Systems Isolation Valves

Criterion 57 addresses lines that penetrate the containment and are neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere.

With the exceptions noted below, lines that penetrate the containment and are neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere are provided with at least one isolation valve located outside the containment, which is either automatic, locked closed or capable of remote manual operation. A simple check valve is not used as an automatic valve.

1. Residual Heat Removal Supply

The residual heat removal lines from two reactor coolant loops each contain two remote manual motor-operated valves, which are closed during power operation. The valves are interlocked to prevent opening when the Reactor Coolant System pressure is greater than the design pressure of the Residual Heat Removal System. The second valve, which defines the reactor coolant pressure boundary, is considered locked closed during power operation.

2. Auxiliary Feedwater Lines

The four auxiliary feedwater lines, which connect to the main feedwater lines prior to containment penetration by the feedwater lines, each contain a check valve. The auxiliary feedwater lines perform a safety function under accident conditions. The feedwater line comprises part of a closed loop inside the containment which serves as an isolation barrier. The isolation arrangement utilized provides necessary isolation without sacrificing reliability with regard to the safety function performed by the lines.

6.2.4.2 System Design

6.2.4.2.1 Actuation Features

Containment isolation is automatically initiated by signals developed in the engineered safety features system described in Section 7.3. Containment isolation consists of Phase A and Phase B. Phase A isolation signal (T) is actuated from the safety injection signal (S) or manually from the control room. Phase B isolation signal (P) is actuated from HI-HI containment pressure or manually from the control room. The containment isolation signals override all other automatic control signals to the isolation valves. Air operated valves fail to the closed position on loss of air or control signal.

In addition, the containment purge line isolation valves automatically close upon a high radiation signal in the containment.

6.2.4.2.2 Isolation Valve Arrangements

The isolation valve arrangements for lines penetrating containment are described below.

Case A

Isolation valve outside and isolation valve inside (both incoming and outgoing lines).

Case B

Isolation valve outside and check valve inside (incoming lines only).

Case C

Isolation valves outside and closed system inside (both incoming and outgoing lines).

The requirements for a closed system inside containment include the following:

1. Must not communicate with either the Reactor Coolant System or the containment atmosphere.

2. Must be missile protected.

3. Must meet ANS Safety Classification 2.

4. Must withstand external pressure and temperature equal to containment design pressure and temperature.

5. Must withstand accident transient and environment.

A summary of the fluid lines penetrating containment, the associated isolation valves and related design information is presented in Table 6.2.4-1. The lines and isolation valves are shown diagrammatically on the respective system flow diagrams.

6.2.4.2.3 Missile Protection

Isolation valves and associated actuators and controls are protected against damage by missiles. Isolation valves inside the containment are located outside the paths of missiles or provided with missile protection.

6.2.4.2.4 Assurance of Isolation Components Operability

Design features are incorporated to assure isolation component operability. Isolation components are designed to safety class 2 requirements, seismic category 1 criteria and are missile protected. Isolation components located inside the containment are designed to operate for the time period required in the post-LOCA environmental conditions of temperature, pressure, humidity, and also radiation. In addition, testing and inspection are performed to assure operability.

TABLE 6.2.4-1

CONTAINMENT PIPING PENETRATIONS

System	Service	Valve Type		Line Status				Mode of Actuation		Actuation Signal	Flow Direction	Penetrant Line Size
		Inside	Outside	Normal	Shutdown	Blackout	Accident	Primary	Secondary			
RCS	Gas from Pressurizer Relief Tank	Air Operated Globe	Air Operated Globe	Inter-mittent	Inter-mittent	Closed	Closed	Automatic	Remote Manual	T	Out	3/8"
RCS	Nitrogen Supply to Pressurizer Relief Tank	Check	Air Operated Globe	Inter-mittent	Inter-mittent	Closed	Closed	Automatic	Remote Manual	T	In	3/4"
RCS	Relief Lines to Pressurizer Relief Tank	Check	Relief Valve	Inter-mittent	Inter-mittent	Inter-mittent	Inter-mittent	Pressure Actuated	---	---	In	1" & 2"
RCS	Deadweight Testing Line-Out	---	Two Locked Closed Globe Valves	Closed	Closed	Closed	Closed	Manual	---	---	Out	1/8"
RCS	Reactor Makeup Water Supply to Pressurizer Relief Tank	Check	Air Operated Diaphragm	Inter-mittent	Closed	Closed	Closed	Automatic	Remote Manual	T	In	3"
CVC	Charging Line	Check	Motor Operated Gate	Open	Closed	As Is	Closed	Automatic	Remote Manual	S	In	3"
CVC	Letdown Line	Air Operated Globe	Air Operated Globe	Open	Closed	Closed	Closed	Automatic	Remote Manual	T	Out	3"
CVC	Reactor Coolant Pump Seal Supply Water	Check	Motor Operated Globe	Open	Open	As Is	Closed	Automatic	Remote Manual	T	In	2"
CVC	Reactor Coolant Pump Seal Water Return	Motor Operated Gate and Check	Motor Operated Gate		Inter-mittent	As Is	Closed	Automatic	Remote Manual	T	Out	4"

TABLE 6.2.4-1

CONTAINMENT PIPING PENETRATIONS (cont.)

System	Service	Valve Type		Line Status				Mode of Actuation		Actuation	Flow	Penetrant
		Inside	Outside	Normal	Shutdown	Blackout	Accident	Primary	Secondary	Signal	Direction	Line Size
RHR	Residual Heat Removal-In (2)	Check	Motor Operated Globe	Closed	Open	As Is	Open	Automatic	Remote Manual	S	In	8"
RHR	Residual Heat Removal-Out (2)	Motor Operated Gates	---	Closed	Open	As Is	Closed	Automatic	Remote Manual	-	Out	12"
SIS	SIS Test Line-Out	Air Operated Globe	Air Operated Globe	Closed	Inter-mittent	Closed	Closed	Automatic	Remote Manual	T	Out	3/4"
SIS	Boron Injection Surge Tank Drain - Out	Globe	Globe	Closed	Inter-mittent	Closed	Closed	Manual	---	-	Out	1"
SIS	High Head Injection Pumps - In (2)	Motor Operated Gate	Motor Operated Gate	Closed	Closed	Closed	Open	Automatic	Remote Manual	S	In	4"
SIS	High Head Injection Pumps Test Line - In	Globe	Globe	Closed	Closed	Closed	Closed	Manual	---	-	In	1"
SIS	Safety Injection Pumps-In to Cold Legs	Check	Motor Operated Gate	Closed	Closed	Closed	Open	Automatic	Remote	-	In	4"
SIS	Safety Injection Pumps-In to Hot Legs (2)	Check	Motor Operated Gate	Closed	Closed	Closed	As Required	Remote Manual	--	-	In	4"
SIS	Nitrogen Supply to Accumulators	Check	Air Operated Globe	Closed	Inter-mittent	Closed	Closed	Automatic	Remote Manual	T	In	1"

TABLE 6.2.4-1

CONTAINMENT PIPING PENETRATIONS (cont.)

System	Service	Valve Type		Line Status				Mode of Actuation		Actuation Signal	Flow Direction	Penetrant Line Size
		Inside	Outside	Normal	Shutdown	Blackout	Accident	Primary	Secondary			
CSS	Containment Spray-In (4)	Check	Motor Operated Gate	Closed	Closed	Closed	Open	Automatic	Remote Manual	P	In	8"
CSS	Containment Spray System Out From Sumps (4)	Guard Pipe	Motor Operated Gate	Closed	Closed	Closed	Open	Automatic	Remote Manual	---	Out	12"
ESW	Reactor Coolant Pump Thermal Barrier Water Supply (For Blackout)	Motor Operated Butterfly	Motor Operated Butterly	Open	Open	Open	Closed	Automatic	Remote Manual	P	In	4"
ESW	Reactor Coolant Pump Thermal Barrier Water Return (For Blackout)	Air Operated Globe	Air Operated Globe	Open	Open	Open	Closed	Automatic	Remote Manual	P	Out	4"
CAS	Service Air	Check	Locked Closed Globe	Closed	Inter-mittent	Closed	Closed	Manual	---	---	In	3"
CAI	Instrument Air	Check	Air Operated Globe	Open	Open	Open	Closed	Automatic	Remote Manual	P	In	2"
CCW	Component Cooling Water Supply	Check	Motor Operated Butterfly	Open	Open	As Is	Closed	Automatic	Remote Manual	P	In	20"
CCW	Component Cooling Water Return	Motor Operated Butterfly	Motor Operated Butterly	Open	Open	As Is	Closed	Automatic	Remote Manual	P	Out	20"
WTL	Nitrogen to Reactor Coolant Drain Tank	Air Operated Diaphragm	Air Operated Diaphragm	Inter-mittent	Inter-mittent	Closed	Closed	Automatic	Remote Manual	T	In	3/4"
WTL	Reactor Coolant Drain Pump Discharge	Air Operated Gate	Air Operated Diaphragm	Inter-mittent	Inter-mittent	Closed	Closed	Automatic	Remote Manual	T	Out	3"
WTL	Containment Sump Pump Discharge	Air Operated Diaphragm	Air Operated Diaphragm	Open	Open	Closed	Closed	Automatic	Remote Manual	T	Out	2"

TABLE 6.2.4-1

CONTAINMENT PIPING PENETRATIONS (cont.)

System	Service	Valve Type Inside	Valve Type Outside	Line Status Normal	Line Status Shutdown	Line Status Blackout	Line Status Accident	Mode of Actuation Primary	Mode of Actuation Secondary	Actuation Signal	Flow Direction	Penetrant Line Size
NSW	Non-Essential Service Water to Upper Compartment Fan Coolers	Check	Air Operated Diaphragm	Open	Open	Closed	Closed	Automatic	Remote Manual	T	In	3"
NSW	Non-Essential Service Water from Upper Compartment Fan Coolers	Air Operated Diaphragm	Air Operated Diaphragm	Open	Open	Closed	Closed	Automatic	Remote Manual	T	Out	3"
NSW	Non-Essential Service Water to Instrument Room Coolers	Check	Air Operated Diaphragm	Open	Open	Closed	Closed	Automatic	Remote Manual	T	In	2"
NSW	Non-Essential Service Water from Instrument Room Coolers	Air Operated Diaphragm	Air Operated Diaphragm	Open	Open	Closed	Closed	Automatic	Remote Manual	T	Out	2"
NSW	Non-Essential Water Supply to Reactor Coolant Pump Motor Coolers	Motor Operated Butterfly	Motor Operated Butterfly	Open	Inter-mittent	As Is	Closed	Automatic	Remote Manual	T	In	10"
NSW	Non-Essential Water Supply from Reactor Coolant Pump Motor Coolers	Motor Operated Butterfly	Motor Operated Butterfly	Open	Inter-mittent	As Is	Closed	Automatic	Remote Manual	T	Out	10"
PAS	Post Accident Air Recirculation Pumps Return (2)	---	Two Manual Diaphragm Valves	Closed	Closed	Closed	Inter-mittent	Manual	---	---	In	3/4"

TABLE 6.2.4-1

CONTAINMENT ISOLATION PENETRATIONS (cont.)

System	Service	Valve Type		Line Status				Mode of Actuation		Actuation	Flow	Penetrant
		Inside	Outside	Normal	Shutdown	Blackout	Accident	Primary	Secondary	Signal	Direction	Line Size
PAS	Post Accident Air Recirculation Fans Sample Outlet (2)	---	Two Manual Diaphragm Valves	Closed	Closed	Closed	Inter-mittent	Manual	---	---	Out	3/4"
PAS	Sample Point #1 at Ice Condenser Door	---	Two Manual Diaphragm Valves	Closed	Closed	Closed	Inter-mittent	Manual	---	---	Out	3/4"
PAS	Sample Point #2 at Ice Condenser Door	---	Two Manual Diaphragm Valves	Closed	Closed	Closed	Inter-mittent	Manual	---	---	Out	3/4"
PAS	Sample Point #3 at Ice Condenser Door	---	Two Manual Diaphragm Valves	Closed	Closed	Closed	Inter-mittent	Manual	---	---	Out	3/4"
HVC	Post Accident Contain-ment Venting System-In	Motor Operated Diaphragm	Motor Operated Diaphragm	Closed	Closed	Closed	Inter-mittent	Remote Manual	---	---	In	8"
HVC	Post Accident Contain-ment Venting System-Out	Motor Operated Diaphragm	Motor Operated Diaphragm	Closed	Closed	Closed	Inter-mittent	Remote Manual	---	---	Out	3"
HVC	Purge Air-Inlet (Instrument Room)	Air Operated Butterfly	Air Operated Butterfly	Inter-mittent	Open	Closed	Closed	Automatic	Remote Manual	T	In	12"
HVC	Purge Air-Outlet (Instrument Room)	Air Operated Butterfly	Air Operated Butterfly	Inter-mittent	Open	Closed	Closed	Automatic	Remote Manual	T	Out	12"
HVC	Purge Air-Inlet (Lower Compartment)	Air Operated Butterfly	Air Operated Butterfly	Inter-mittent	Open	Closed	Closed	Automatic	Remote Manual	T	In	24"

TABLE 6.2.4-1

CONTAINMENT PIPING PENETRATIONS (cont.)

System	Service	Valve Type		Line Status				Mode of Actuation		Actuation Signal	Flow Direction	Penetrant Line Size
		Inside	Outside	Normal	Shutdown	Blackout	Accident	Primary	Secondary			
HVC	Purge Air-Outlet (Lower Compartment	Air Operated Butterfly	Air Operated Butterfly	Inter-mittent	Open	Closed	Closed	Automatic	Remote Manual	T,HRS (1)	Out	24"
HVC	Purge Air-Inlet (Upper Compartment)	Air Operated Butterfly	Air Operated Butterfly	Inter-mittent	Open	Closed	Closed	Automatic	Remote Manual	T,HRS (1)	In	30"
HVC	Purge Air-Outlet (Upper Compartment)	Air Operated Butterfly	Air Operated Butterfly	Inter-mittent	Open	Closed	Closed	Automatic	Remote Manual	T,HRS (1)	Out	30"
HVC	Containment Vacuum Relief (4)	Air Operated Butterfly	Check	Open	Open	Closed	Closed	Automatic	Remote Manual	T	In	24"
ICR	Glycol Solution to Ice Condenser Air Handling Units	Motor Operated Gate	Motor Operated Gate	Open	Open	As Is	Closed	Automatic	Remote Manual	T	In	4"
ICR	Glycol Solution from Ice Condenser Air Handling Units	Motor Operated Gate	Motor Operated Gate	Open	Open	As Is	Closed	Automatic	Remote Manual	T	Out	4"

(1)HRS= Containment High Radiation Signal

TABLE 6.2.4-1

CONTAINMENT PIPING PENETRATION (cont.)

System	Service	Valve Type		Line Status				Mode of Actuation		Actuation Signal	Flow Direction	Penetrant Line Size
		Inside	Outside	Normal	Shutdown	Blackout	Accident	Primary	Secondary			
SGB	Steam Generator Blowdown Lines (4)	Air Operated Globe	Air Operated Globe	Open	Closed	Closed	Closed	Automatic	Remote Manual	T	Out	2"
SSR	Sample Line from Pressurizer Liquid Space	Air Operated Globe	Air Operated Globe	Inter-mittent	Closed	Inter-mittent	Closed	Automatic	Remote Manual	T	Out	3/8"
SSR	Sample Line from Pressurizer Steam Space	Air Operated Globe	Air Operated Globe	Inter-mittent	Closed	Inter-mittent	Closed	Automatic	Remote Manual	T	Out	3/8"
SSR	Sample Lines from Pressurizer Steam Loops (2)	Air Operated Globe	Air Operated Globe	Inter-mittent	Inter-mittent	Inter-mittent	Closed	Automatic	Remote Manual	T	Out	3/8"
SSR	Sample Lines from Accumulators (4)	Air Operated Globe	Air Operated Globe	Inter-mittent	Closed	Inter-mittent	Closed	Automatic	Remote Manual	T	Out	3/8"
AXS	Auxiliary Steam-In	Locked Closed Gate	Locked Closed Gate	Closed	Open	Closed	Closed	Manual	---	---	In	2"
AXS	Auxiliary Steam-Out	Locked Closed Gate	Locked Closed Gate	Closed	Open	Closed	Closed	Manual	---	---	Out	2"
CFW/ AFW	Main Feedwater/Auxiliary Feedwater-In (4)	---	Check/Check	Open/ Closed	Closed/ Closed	Closed/ Open	Closed/ Open	Automatic/ Automatic	---	---	In/ In	16"
MNS	Main Steam-Out	---	Air Piston Swing Disc Stop	Open	Closed	Open	Closed	Automatic	Remote Manual	S,MSLIS(1)	Out	32"

(1)Main Steam Line Isolation Signal

All air-operated isolation valves fail closed upon loss of air or signal, thereby performing the isolation function.

Power operated valves can be manually operated from the control room.

The integrity of isolation components under the dynamic forces resulting from inadvertent closure is assured by calculational analysis. The time-histories of pressure in the isolation component resulting from sudden inadvertent closure are determined. Analyses are then performed to determine the resulting component displacement stresses and support and restraint reactions. The type and location of component support and restraints are established to accommodate the resultant forces.

6.2.4.3 Design Evaluation

Design features are incorporated to assure performance of containment isolation upon actuation by either the engineered safeguards signal or high radiation signal in the containment. Redundance and independence is incorporated. Testing of the isolation components' operability and leaktightness assure reliable isolation.

6.2.4.4 Tests and Inspections

Tests and inspections performed on containment isolation components are described in the Technical Specifications. The tests and inspections assure the leaktightness integrity of the containment and isolation components and the operatility of automatic and remote manual valves.

6.2.5 COMBUSTIBLE GAS CONTROL IN CONTAINMENT

Control of the post-LOCA hydrogen concentration in the containment is provided by the Containment Hydrogen Recombiner System (CHR). In addition, the Post Accident Containment Venting System described in subsection 6.2.3 is provided as a secondary means of hydrogen control.

6.2.5.1 Design Basis

The design basis of the Containment Hydrogen Recombiner System is removal of hydrogen by the recombination process as required to meet both the Westinghouse-TID hydrogen generation model and the AEC-TID release model. The AEC-TID release model assumes a value of 5 percent zirconium-water reaction and assumes that the hydrogen concentration should not exceed 4 percent by volume as specified in AEC Safety Guide No. 7.

The Westinghouse TID model assumes the fission product activity release specified in TID-14844 and 2 percent zirconium-water reaction. This model is based on the requirement that an average hydrogen concentration of 2 percent by volume is not exceeded. The recombiner units are designed for the containment environment conditions given in Table 6.2.5-1. ANS Safety Classification and ASME Code requirements are shown in Table 3.2-1.

TABLE 6.2.5-1

CONTAINMENT ENVIRONMENT DESIGN DATA

Normal Conditions	
Temperature, ^{o}F	120
Pressure, psig	0
Humidity, per cent (relative humidity)	100
Radiation, rads/hr	5
Post-Accident Conditions	
Temperature, ^{o}F	250 maximum (first day) 155 thereafter
Pressure transient, maximum, psig in 10 seconds	15
Pressure, psig	15 maximum (first day)
Humidity, per cent (relative humidity)	100
Radiation	2×10^8 rads, total dose dose 5×10^6 rads/hr, maximum rate
Containment spray solution	2000 ppm boric acid with NaOH added to pH 10.5

6.2.5.2 System Design

The system consists of two redundant recombiner units located inside the containment in the upper compartment and two associated control panels located outside the containment in the auxiliary area which is accessible following a LOCA. The recombiner design parameters are presented in Table 6.2.5-2.

Following a LOCA, hydrogen is assumed to be generated inside the containment by the mechanisms of radiolysis, zirconium-water reaction and corrosion. The hydrogen recombiner unit recombines the free hydrogen in the containment atmosphere with available oxygen, thereby providing a means to limit the hydrogen level from reaching flammable limits. Basically, the recombiner utilizes conventional electric heaters to heat a continuous flow of containment air above the hydrogen-oxygen reaction temperature causing recombination to occur.

The hydrogen recombiner is shown in Figure 6.2-22. The unit, which has no moving parts, consists of an inlet preheater section, a heater-recombination section, and a mixing chamber. The preheater section consists of a shroud placed around the heater section which functions to increase unit efficiency by reducing heat losses and drying the air before it reaches the heater section. The heater section consists of five assemblies of electric heaters stacked vertically. Each assembly contains 60 individual

TABLE 6.2.5-2

CONTAINMENT HYDROGEN RECOMBINER

DESIGN PARAMETERS

Power, kW (3-phase, 480 volts)	75
Capacity, scfm	100
Heaters:	
Number	5
Surface area/heater, sq. ft.	35
Average heat transfer coefficient Btu/hr. -sq. ft., ^{o}F	2.5
Maximum heat flux, Btu/hr. -sq. ft. or watts/sq. in.	3.3
Maximum sheath temperature, ^{o}F	1550
Gas Temperatures	
Inlet, ^{o}F	80-250
Outlet of heater section, ^{o}F	1150-1400

U-type heaters. The heaters are conventional type electric resistance heaters sheathed with Incoloy-800. The heaters are designed to operate with sheath temperature achieved with certain commercial heaters; however, the recombiner heaters operate at significantly lower power densities than is commercial practice. Protection of the recombiner internals from containment spray water impingement is afforded by the unit outer enclosure.

Containment air is drawn into the recombiner by natural convection passing first through the preheater section where preheating of the air occurs. The warmed air then passes through a fixed orifice plate used to regulate the air flow rate and then enters the electric heater section. In the heater section, the temperature of the air is raised to a level sufficient to cause recombination of free hydrogen and oxygen to occur (approximately 1150^{o}F). The flow then enters the mixing chamber where the hot stream from the heater section is mixed and diluted with cooler containment air for cooling prior to discharge back to the containment atmosphere. Tests have verified that recombination is not a catalytic surface effect associated with the heaters, but occurs due to the increased temperature of the process gases. Since the phenomenon is not a catalytic effect, saturation of the unit by fission products does not occur.

Assured power to the recombiners during an accident is redundantly supplied from the emergency diesel generators with control available at the recombiner control panels. The control panels contain an isolation transformer and a controller to regulate power to the recombiner. The panels, located outside the containment, are not exposed to the post-LOCA containment environment. For recombiner testing, a thermocouple readout indicator is provided in the control room to monitor the recombiner temperature.

Results of recombiner prototype testings are given in WCAP 7820, Supplement 1, "Electrical Hydrogen Recombiner for Water Reactor Containments," May 1972.

Sampling of the containment atmosphere to determine the post-LOCA hydrogen concentration is performed by the Containment Post-Accident Sampling System, subsection 9.3.4.

Mixing of the containment atmosphere is performed to preclude unacceptable localized concentrations of hydrogen within the containment. The containment air return fans are utilized to circulate air via air distribution headers to various containment locations.

6.2.5.3 Design Evaluation

Analysis of hydrogen generation following a LOCA indicates that although the hydrogen production rate decreases with time following the accident, total hydrogen accumulation can exceed the regulatory limit of 4 volume percent. Thus, control measures are necessary to preclude hydrogen accumulation to this limit. Figure 6.2-23 and Figure 6.2-24 show hydrogen accumulation in the containment based on the Westinghouse Model and on the AEC Safety Guide No. 7 model respectively.

The recombiners preclude the hydrogen concentration reaching the regulatory limit. Each recombiner is capable of continually processing a minimum of 100 scfm of containment air. The maximum hydrogen concentration is maintained less than 4 volume percent, if one recombiner is started one day after the accident. Thus the redundancy of the two recombiners is established.

6.2.5.4 Tests and Inspections

Peroperational testing of the recombiners by applying power to the heaters and monitoring the unit temperature is performed prior to reactor operation.

6.2.5.5 Instrumentation Application

Recombiner operation following an accident does not require any instrumentation inside the containment. Proper recombiner operation after an accident is assured by measuring the amount of electrical power to the recombiner which is performed outside the containment at the recombiner control panels. Proper amount of air flow through the recombiner is fixed by an orifice plate requiring no instrumentation.

REFERENCES

"Preliminary Design and Evaluation of the Ice Condenser Reactor Containment and Associated Engineered Safeguards," WCAP-7079, Supplement 2, by S. Weems, W. G. Lyman, N. P. Grimm, and MPR Associates, December 1967.

"Design and Performance Evaluation of the Ice Condenser Reactor Containment System," WCAP-7183L, Supplement 2, by H. W. McCurdy, S. Weems, F. M. Bordelon, N. P. Grimm, E. J. Kilpela, W. G. Lyman and MPR Associates, August 1969.

"Design and Performance Evaluation of Ice Condenser Inlet Doors," WCAP-7689, Supplement 2, by W. G. Taft, March 1971.

"Electric Hydrogen Recombiner for PWR Containment," WCAP-7820, Supplement 1, by J. R. Wilson, May 1972.

"Boiler and Pressure Vessel Code {Nuclear Power Plant Components}," ASME Code Section III, Subsection NE, by The American Society of Mechanical Engineers, July 1971.

"Design and Performance Evaluation of Ice Condenser Reactor Containment System," WCAP-7833, Supplement 1, by H. W. McCurdy, R. Weems, N. P. Grimm, August 1969.

"Alternate Strength Criteria to Insure No Loss of Function of Piping in Vessels Under Earthquake Loading," WCAP-5890, Supplement 2, by R. E. Tome, R. Wiesemann, and R. Salvatori, October 1967.

6.3 EMERGENCY CORE COOLING SYSTEM

This section is presented in RESAR 3, Section 6.3 with the following additional information and modifications:

1. Information for the four loop plant is applicable.

2. System Arrangement

   The Emergency Core Cooling System (ECCS) components and piping outside the shield building are located in safeguard compartments and tunnels as shown on Figures 1.2-1, 1.2-2 and 1.2-3. Redundant system pump and piping trains are located in separate compartments and piping tunnels. Four separate compartments and four associated piping tunnels are provided. Major component location outside the containment is as follows:

   a. Safeguard Compartment No. 1

      High head safety injection pump No. 1
      Residual heat removal pump No. 1

   b. Safeguard Compartment No. 2

      Safety injection pump No. 1

Residual heat removal pump No. 2.

c. Safeguard Compartment No. 3

Safety injection pump No. 2.

d. Safeguard Compartment No. 4

High head safety injection pump No. 2.

The separation of redundant equipment and flow paths ensures continued core cooling, even in the event of a component failure or leakage from a train. The system suction piping from the refueling water storage tanks and the containment sumps are described in Section 6.2.2.7. The suction piping from the four containment sumps consists of four separate lines routed in separate safeguard piping tunnels.

3. Modifications to RESAR 3

The flow diagram of the Emergency Core Cooling System (ECCS) is presented in Figure 6.3-1.

a. High Head Safety Injection Pumps

The centrifugal charging pumps do not comprise part of the Emergency Core Cooling System. The high head safety injection pumps replace and perform the emergency core cooling function of the charging pumps. The pumps are identical in design to the charging pumps and serve no function other than the emergency core cooling function.

The high head safety injection pumps are normally aligned to take suction from the refueling water storage tanks for the injection mode of emergency core cooling. At the termination of the injection mode, suction switchover to the containment sumps is performed by the control room operator. During the recirculation mode, the pumps take suction directly from the containment sumps. The flow in each of the two high head safety pump trains is indicated in the control room. Adequate pump net positive suction head is available.

This modification applies to the following RESAR-3 Sections and Tables.

SECTIONS		TABLES
6.3.2.1.4	6.3.3.1.5	6.3-1 through 6.3-9
6.3.2.2.1	6.3.3.6	6A-1
6.3.2.2.2	6.3.3.7	6A-2
6.3.2.2.3	6.3.3.9	
6.3.2.2.4	6.3.4.2	
6.3.2.2.6	6.3.4.3	
6.3.3.1.3	6.3.5.3	
	6A.3.1	

b. Residual Heat Removal Pumps and Heat Exchangers Functions

The residual heat removal pump-heat exchanger trains do not operate during the recirculation mode. Component cooling water supply to the heat exchangers is isolated during emergency core cooling operation. Emergency core cooling flow during the recirculation mode is supplied by the safety injection and high head safety injection pumps.

A pump minimum flow bypass line is provided on each residual heat removal pump discharge to recirculate flow to the refueling water storage tanks should the Reactor Coolant System pressure be above the pump's shutoff head. A motor operated isolation valve is provided in this line which is interlocked to permit opening only after actuation of a safety injection ("S") signal.

Containment heat removal via containment sump water recirculation is performed by the Containment Spray System, Section 6.2.2.7.

This modification applies to the following RESAR-3 Sections and Tables:

	SECTIONS	TABLES
6.3.2.1.4	6.3.3.1.3	6.3-4
6.3.2.1.5	6.3.3.1.5	6.3-5
6.3.2.2.2	6.3.3.6	6.3-6
6.3.2.2.3	6.3.3.7	6A-2
6.3.2.2.6	6.3.3.7.1	
	6.3.5.3	

c. Boron Injection Tank and Associated Equipment Location

The boron injection tank and associated boron injection surge tank, boron injection recirculation pumps, piping, valves, heat tracing and instrumentation are located inside the containment. The boron injection tank design and associated isolation piping and valves are compatible with the post-accident containment environment for the period they are required to be operable. The surge tank, recirculation pumps, heat tracing, instrumentation and other valves and piping are not required to remain operable during emergency core cooling system operation.

This modification applies to RESAR-3, Sections 6.3.2 and 6.3.3.10.

d. Cooling Water Supply to ECCS Pumps for Seal Cooling

The Essential Service Water System (Section 9.2.2) supplies assured cooling water to the residual heat removal, safety injection and high head safety injection pumps during emergency core cooling operation for pump seal cooling.

This modification applies to RESAR 3, Section 6.3.3.7.1.

e. Refueling Water Storage Tanks

The refueling water storage tanks comprise part of the Containment Spray System, Section 6.2.2.7. Two 175,000 gallon (useable) capacity tanks are provided. The tanks are connected via lines with equal pressure drops to a common discharge header which assures that both tanks are emptied at the same rate. Two water level indicator channels, with read-out in the control room, are provided. Each channel alarms on both low and low-low water levels.

This modification applies to RESAR-3, Section 6.3.5.4 and Table 6.3-1.

f. Containment Sump Water Level Indicator

The containment sump water level instrumentation comprises part of the Containment Spray System, Section 6.2.2.7. Four containment sumps are provided, each with a sump water level indicator channel. Indication is provided in the control room.

This modification applies to RESAR-3, Section 6.3.5.4.

g. Changeover from Injection Mode to Recirculation Mode

The sequence of changeover from the injection mode to the circulation mode is delineated in Table 6.3-4.

This modification applies to RESAR-3, Section 6.3.2.2.2 and Table 6.3-4.

h. Single Failure Analysis

Single failure analyses for the Emergency Core Cooling System is presented in Tables 6A-1 and 6A-2.

This modification applies to RESAR-3, Section 6.3.3.1.5 and Tables 6A-1 and 6A-2.

TABLE 6.3-4

SEQUENCE OF CHANGEOVER FROM INJECTION MODE TO RECIRCULATION MODE

The following is the changeover sequence for terminating the injection mode and initiating the recirculation mode.

1. Stop residual heat removal pump No. 1.

2. Stop residual heat removal pump No. 2.

3. Stop high head safety injection pump No. 1.

4. Close high head safety injection pump No. 1 suction valve in line from the refueling water storage tanks.

5. Open containment sump valve in sump line No. 1.

6. Start high head safety injection pump No. 1.

7. Stop safety injection pump No. 1.

8. Close safety injection pump No. 1 suction valve in line from the refueling water storage tanks.

TABLE 6.3-4 (CONT.)

9. Open containment sump valve in sump line No. 2.

10. Start safety injection pump No. 1.

11. Stop safety injection pump No. 2.

12. Close safety injection pump No. 2 suction valve in line from refueling water storage tanks.

13. Open containment sump valve in sump line No. 3.

14. Start safety injection pump No. 2.

15. Stop high head safety injection pump No. 2.

16. Close high head safety injection pump No. 2 suction valve in line from the refueling water storage tanks.

17. Open containment sump valve in sump line No. 4.

18. Start high head safety injection pump No. 2.

TABLE 6A-1

SINGLE ACTIVE FAILURE ANALYSIS FOR EMERGENCY CORE COOLING SYSTEM COMPONENTS

SHORT TERM PHASE

Component	Malfunction	Comments
A. Accumulator	---	Totally passive system. Not susceptible to active failure.
B. Pump		
1. High head safety injection	Fails to start	Two provided. Evaluation based on operation of one.
2. Safety injection	Fails to start	Two provided. Evaluation based on operation of one.
3. Residual heat removal	Fails to start	Two provided. Evaluation based on operation of one.
C. Automatically Operated Valves		
1. Boron injection tank isolation		
a. Inlet	Fails to open	Two parallel lines; one valve in either line required to open.
b. Outlet	Fails to open	Two parallel lines; one valve in either line required to open.

TABLE 6A-1 (cont'd)

SINGLE ACTIVE FAILURE ANALYSIS FOR EMERGENCY CORE COOLING SYSTEM COMPONENTS

LONG TERM PHASE

A. Valves Operated from Control Room for Recirculation		
1. Containment sump line recirculation isolation	Fails to open	Four parallel sump lines: valves in three lines result in sufficient flow.
2. Safety injection pump suction line from refueling water storage tanks	Fails to close	Check valve in series with gate valve; operation of only one valve required.
3. High head safety injection pump suction line from refueling water storage tanks	Fails to close	Check valve in series with gate valve; operation of only one valve required.
B. Pumps		
1. High head safety injection	Fails to operate	Three of the four pumps deliver adequate flow.
2. Safety Injection	Fails to operate	

TABLE 6A-2

EMERGENCY CORE COOLING SYSTEM RECIRCULATION PIPING PASSIVE FAILURE ANALYSIS

LONG TERM PHASE

Flow Path	Indication of Loss of Flow Path	Alternate Flow Path
Recirculation Mode		
From containment sumps to the high head safety injection and safety injection pumps	Accumulation of water in a safeguard compartment	Via the other three independent ECCS pumps trains located in separate safeguard compartments.

6.4 OTHER ENGINEERED SAFETY FEATURES

6.4.1 AIR RETURN FANS

6.4.1.1 Design Basis

The function of the air return fans is to assure rapid return of air to the lower compartment after the initial loss-of-coolant blowdown. The design basis of the fans is to assure containment pressure reduction to less than 6.0 psig within six minutes for the case where steam generation by residual energy is not terminated by the Emergency Core Cooling System. If the Emergency Core Cooling System functions as designed, the air return fans are not required. Two fans are provided, each with the capability to perform the assigned function.

6.4.1.2 System Design and Operation

Air return fans, located in the upper containment compartment, are provided to blow air from the upper compartment through a portion of the equipment compartments located outside the crane wall and then into the lower compartment. Two full capacity fans are provided, each with the capability to perform the assigned function. Each fan has a capacity of 40,000 cfm.

Both fans are started by the containment spray actuation signal. The fans blow air from the upper compartment to the lower compartment through the equipment annulus outside the crane wall. A non-return valve is provided at the discharge of each fan to prevent reverse flow. Flow enters the lower compartment through ports in the fan room crane wall and is ducted to the Steam Generator enclosures and the reactor vessel cavity. After discharge into the lower compartment, the air flows with the steam produced by residual heat through the ice condenser doors into the ice condenser compartment where the steam portion of the flow will be condensed.

The air return fans have sufficient head to overcome the divider barrier differential pressure resulting from the residual heat steam flow and fan air flow entering the ice condenser through the lower doors. The fans can be powered by either normal or emergency power. Fan design is compatible with operation in a post-accident containment environment.

6.4.1.3 Design Evaluation

The analysis in Section 6.2.1.3.3.3, Long Term Containment Performance, shows the capability of the air return fan to reduce the containment pressure to 6 psig within six minutes.

6.4.1.4 Tests and Inspections

Tests and inspections will be performed to assure the capability of the air return fans to perform the assigned function.

Shop Testing

Air return fans'performance characteristics will be tested according to AMCA test procedures.

Inplace Testing and Inspections

Testing will be performed prior to initial startup to assure proper operation of each air return fan. Thereafter, periodic tests and inspection will be performed to demonstrate system readiness and and operability.

6.4.1.5 Instrumentation and Application

There is no process instrumentation associated with the air return fans.

6.4.2 ANNULUS FILTRATION SYSTEM

6.4.2.1 Design Bases

This system is designed to:

1. Collect and filter radioactive iodine and particulates from containment leakage to the annulus during accident conditions.

2. Relieve the post accident pressure increase due to thermal expansion of air in the annulus between the containment and shield building and pipe penetration areas housing engineered safeguard and mechanical systems piping.

6.4.2.2 System Design

The system consists of two redundant 100% capacity fan filter trains as shown in Figure 6.4-1. Each train consists of a heater, HEPA filter, charcoal filter, fan and associated dampers, ducting and instrumentation. Equipment data are presented in Table 6.4-1.

The heater consists of an electric heating element capable of increasing the temperature of the incoming air by a sufficient amount to assure the air humidity is no greater than 70% relative to saturation prior to entering the filter train. The electric heating elements are supplied from separate engineered safeguards buses.

Charcoal filters are type BC 727 or equivalent having design efficiencies at 70 percent relative humidity of 95.0 percent for removal of methyl iodine and 95.0 percent for removal of elemental iodine. Pressure drop is 1.15 inches of water at 45 fpm superficial air velocity.

Each fan is a full-capacity centrifugal fan. Each fan is supplied from a separate engineered safeguards bus.

TABLE 6.4-1

ANNULUS FILTRATION SYSTEM

Fans	
Number	2
Type	Centrifugal, SWSI
Drive	Multiple
Capacity per fan, scfm	8000
Total static pressure, in/wg	8
Motor size, H.P.	20
Heater	
Number	2
Heating capacity, kw	40

TABLE 6.4-2

CONTAINMENT ANNULUS VENTILATION AND FILTRATION

Materials of Construction:

Air Plenums	Carbon Steel, protective finish
Ductwork	Carbon Steel, protective finish
Dampers	Carbon Steel, protective finish
Fans	All steel construction
Absolute Filters	Same as Table 9.4.2-
Prefilters	Same as Table 9.4 2
Charcoal filters	Same as Table 9.4-2-2

Each filter is sized adequately to accommodate the fission products released to the annulus following any of the postulated accidents.

6.4.2.3 Safety Evaluation

The Annulus Filtration System safety class and code class requirements are presented in Table 3.2-1. A failure analysis of the system is given in Table 6.4-3.

During normal plant operation one exhaust fan operates constantly on by pass of the charcoal filters which are normally isolated from the air stream. At this time the annulus is maintained at a negative pressure of 2 inches water. In the event of a LOCA the "S" signal actuates system operation with flow through both filter trains. Since the post-accident thermal expansion of air in the annulus causes a pressure increase of 1.5 inches water the annulus pressure reaches a negative pressure of approximately 0.5 inch water, but remains negative and thus prevents leakage from the annulus through the shield wall to the atmosphere.

After the thermal transient passes (within 2 minutes) the safeguards piping tunnel and compartments, and piping penetration room are placed in communication with the annulus space via the pipe penetration exhaust dampers which open automatically. The volume of air exhausted is automatically minimized by recirculating part of the air as shown.

TABLE 6.4-3

ANNULUS FILTRATION SYSTEM MALFUNCTION ANALYSIS

Component	Malfunction	Comments & Consequences
1. Annulus Filtration Train	Fan fails to start or stop running and cannot be restarted.	Two full-capacity fans provide redundancy. Fan discharge damper closes by gravity to prevent uncontrolled recirculation of air.
2. Annulus Filtration Filter Train	Filter failure	Two full-capacity filter trains provide redundancy.
3. Annulus Filtration Heater	Heater failure	Two full-capacity heaters provide redundancy.
4. Component Damper	Fails to open	Two full capacity dampers provide redundancy.
5. Filter Train Isolation Dampers	Fails to open	Two full capacity filter trains and fans provide redundancy.
6. Filter Train Bypass Damper	Fails to close	Two dampers in series provide redundancy
7. Recirculation Air Damper	Fails to close	Two dampers in series provide redundancy.
8. Annulus System to Plant Vent Damper	Fails to open	Two full capacity systems provide redundancy.
9. System Controls	Fails to operate	Controlled devices take fail safe positions. Redundant system provides necessary exhaust function.

TABLE 6.4-3 CONT.

	Component	Malfunction	Comments & Consequences
10.	System Exhaust Fan	Fails to continue operating	Crossover damper provides removal of iodine decay heat from charcoal filter.
11.	Crossover Damper	Fails to open	Two full capacity dampers provide redundancy.

Both air flow and pressure differentials across the filter trains are indicated in the control room.

The air in the annulus is not heated or circulated during normal operation. The thermal coupling at the steel containment is adequate to maintain the minimum air temperature above 32°F during normal and shutdown conditions.

6.4.2.4 Tests and Inspections

Tests and inspections will be performed to assure the capability of components and the system to perform their assigned functions.

6.4.2.4.1 Shop Testing

Tests will demonstrate the following:

1. Carbon filter capabilities for removal of molecular iodine-131 and methyl iodide-131.

2. Carbon filter iodine collection capability

3. Carbon filter cell leak tightness integrity

4. Carbon filter flow resistance

5. HEPA filter efficiency and resistance test to meet UL label requirements.

6. Fan performance characteristics tested according to AMCA test procedures.

6.4.2.4.2 System Test and Inspection

Operational testing will be performed in accordance with Section 14.1 prior to initial startup to demonstrate proper functioning of the system. Testing will include the following:

1. Leak tightness tests of components and systems

2. System functional test

Thereafter, periodic tests and inspections will be performed to demonstrate system readiness and operability. For the purpose of periodically testing the retentive capability of the carbon filter, representative test carbon samples will be placed in the filter housing in locations which allow the samples to be subjected to the same air flows as the carbon bed. The samples will be periodically removed and tested.

6.4.2.5 Instrumentation and Application

The following instrumentation is used:

1. Containment Pressure

   Two of the four containment pressure transmitters (see Section 7.3) through bistables are used to actuate the annulus Filtration System on high containment pressure. High pressure alarms and indicators are located in the control room.

2. Annulus Pressure

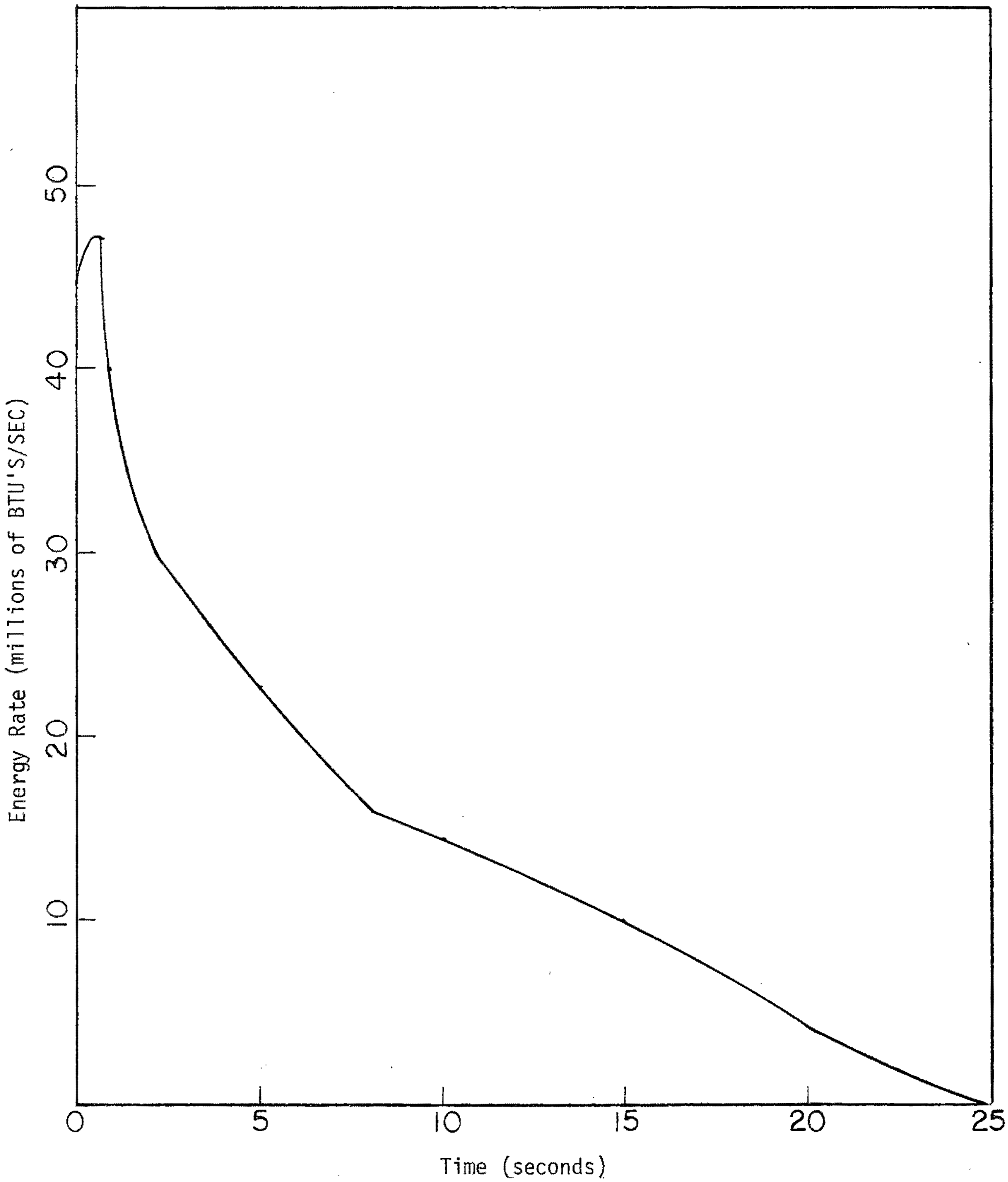
   A pressure indicator is provided in the control room. Additionally, a pressure controller operates the control air dampers to provide the required sequence of operation prior to and following LOCA to maintain the required vacuum condition while minimizing the exhaust volume.

3. Filter Train Differential Pressure

   A pressure transmitter is provided to measure the differential pressure across the filter train. Indication is provided in the control room.

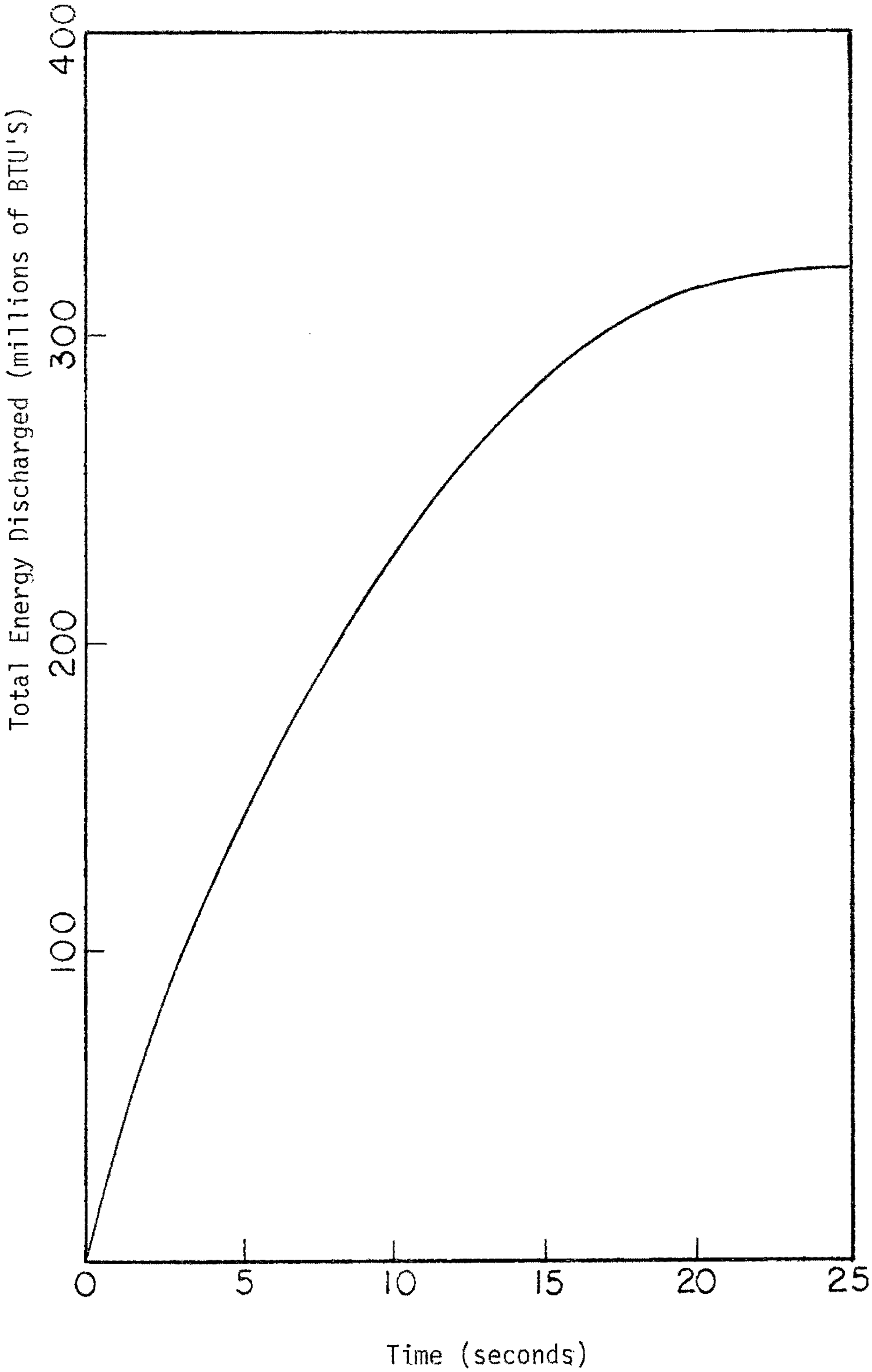
4. Air Flow Meter

   A flow transmitter is provided to measure the flow in the inlet header to the annulus filtration fans. Indication is provided in the control room.



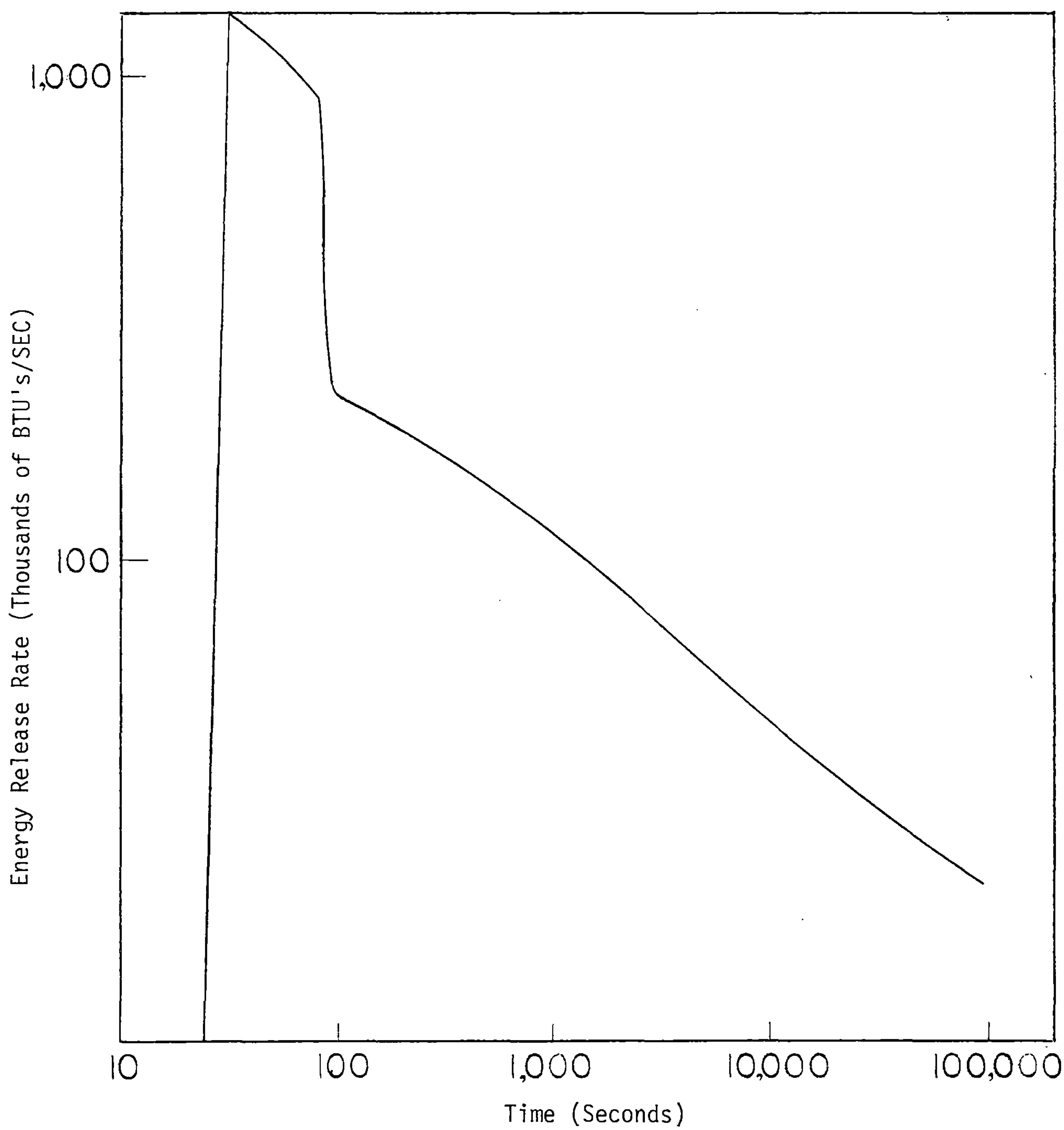
Rate of Energy addition to the containment during blowdown

11/1/72 Figure 6.2-1



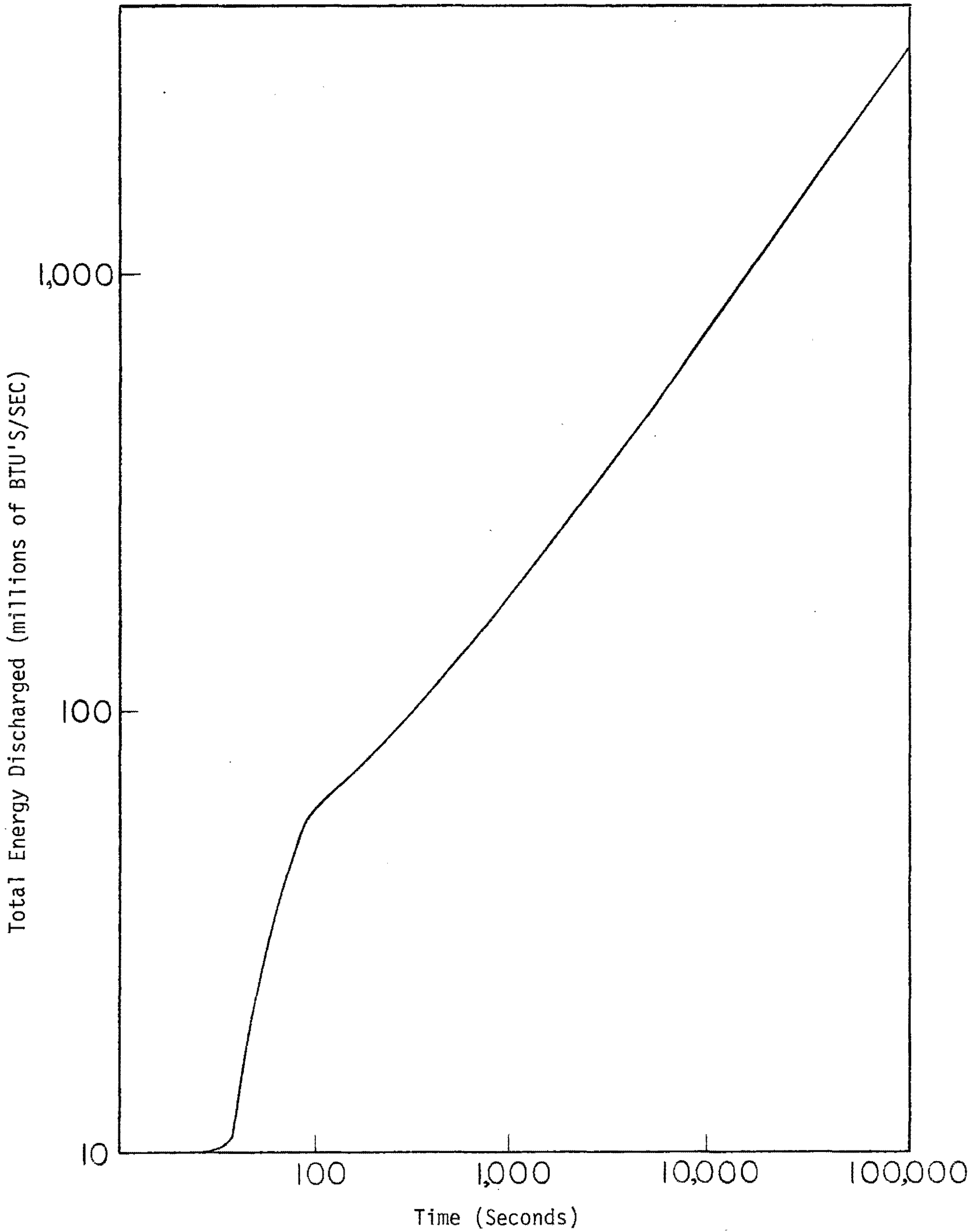
Total energy addition to the containment during blowdown

11/1/72 Figure 6.2-2

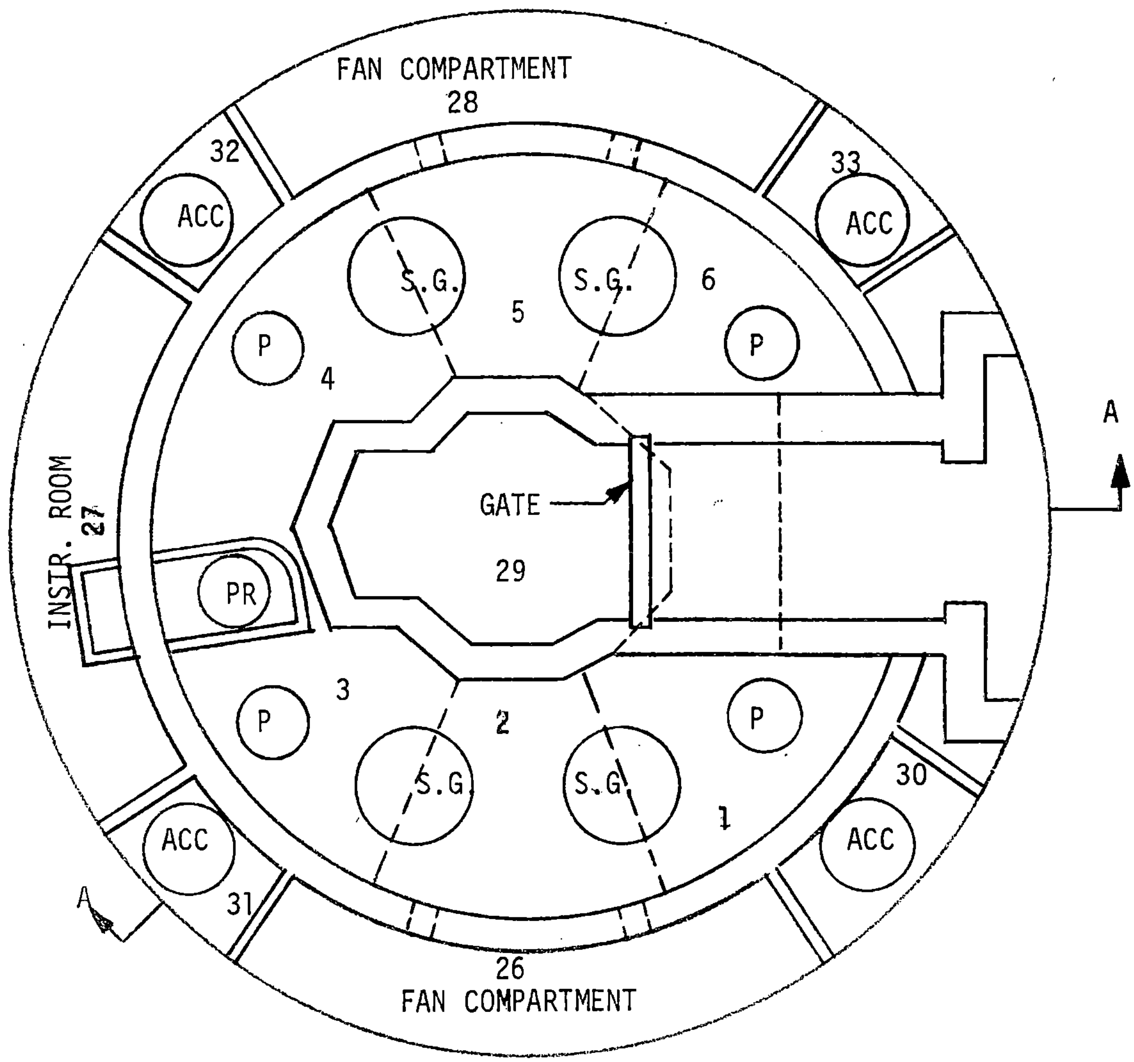


Rate of energy release to the containment, from all sources, during the first day after blowdown

11/1/72 Figure: 6.2-3

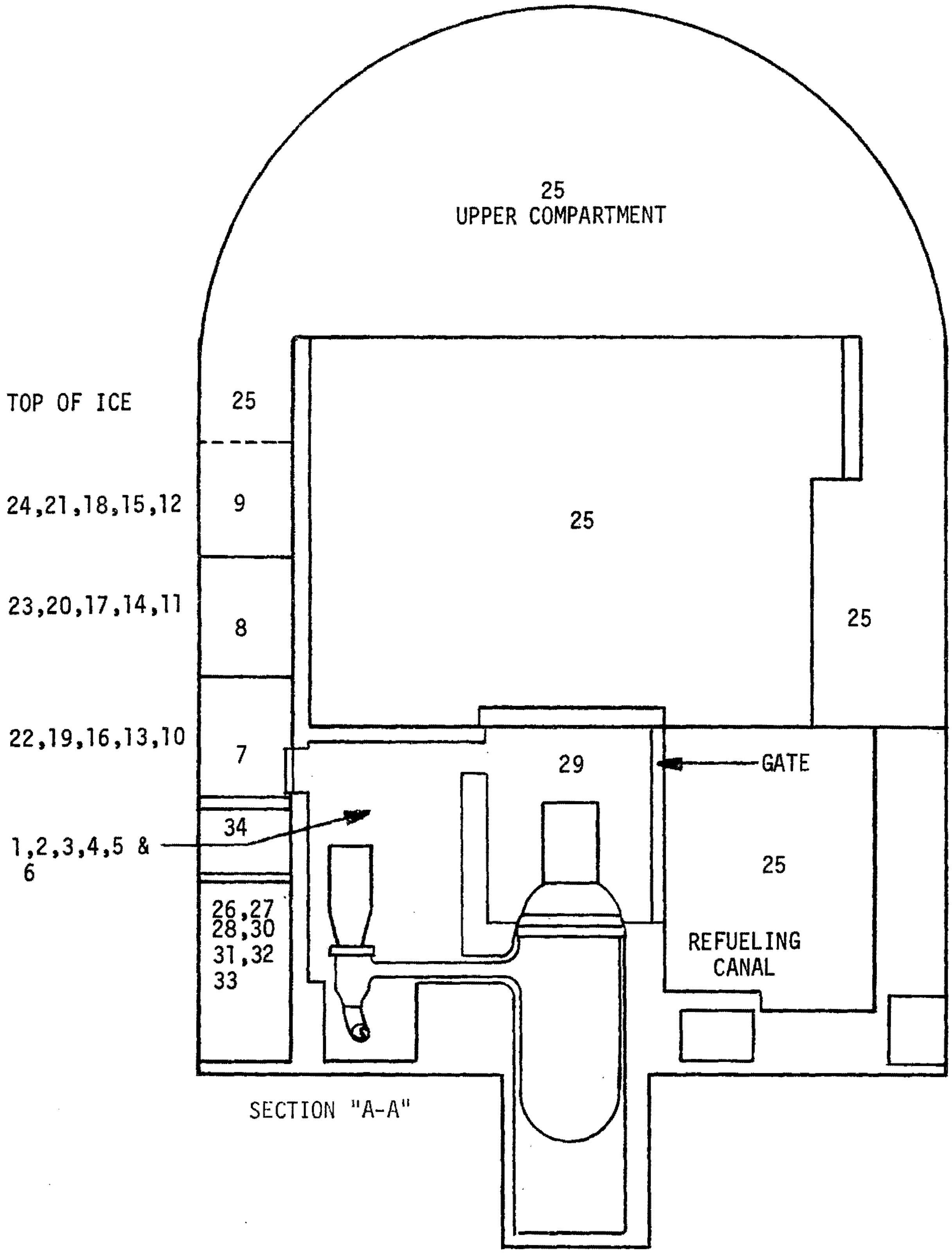


Total energy addition to the containment from all sources, during the first day after blowdown



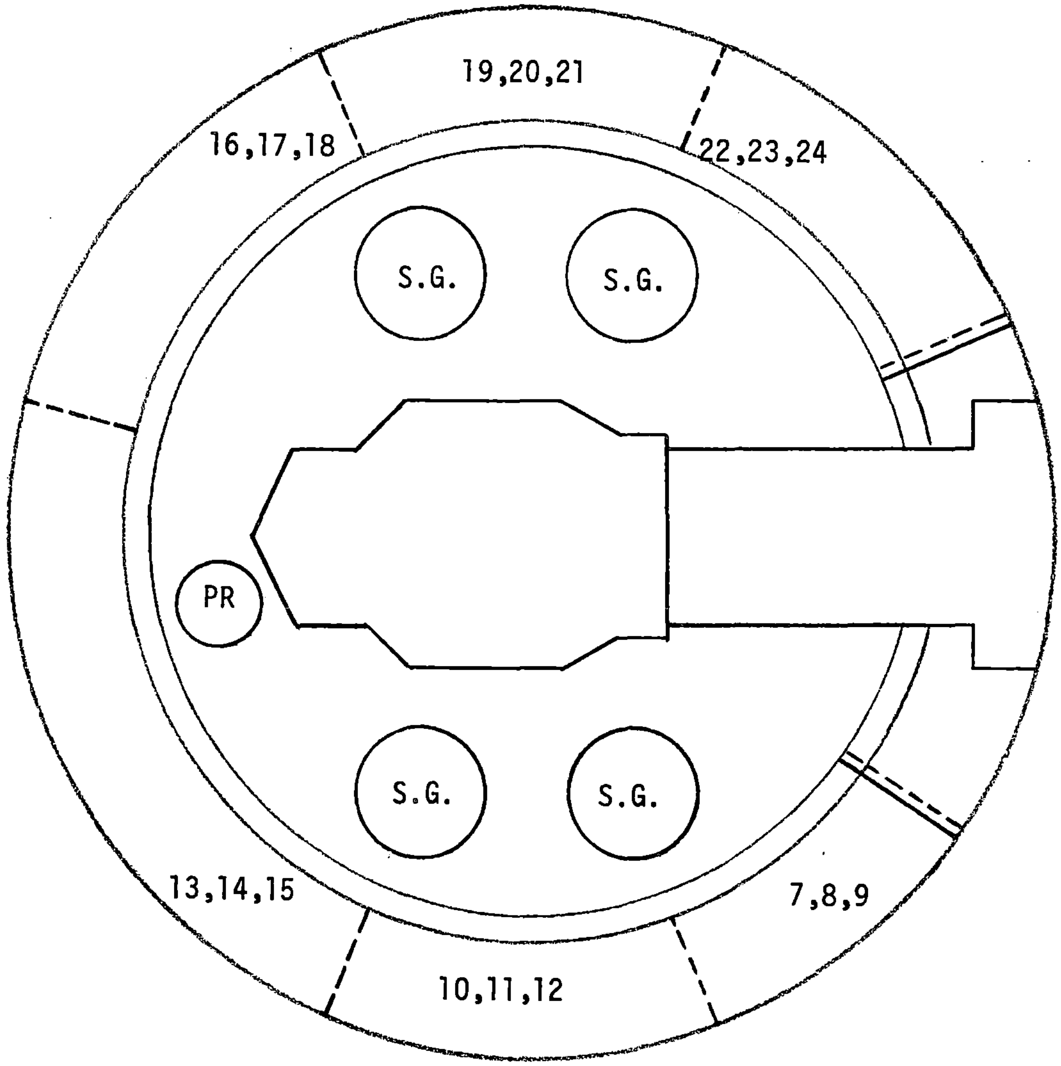
PLAN AT EQUIPMENT ROOMS ELEVATION

11-1-72 Fig. 6.2-5



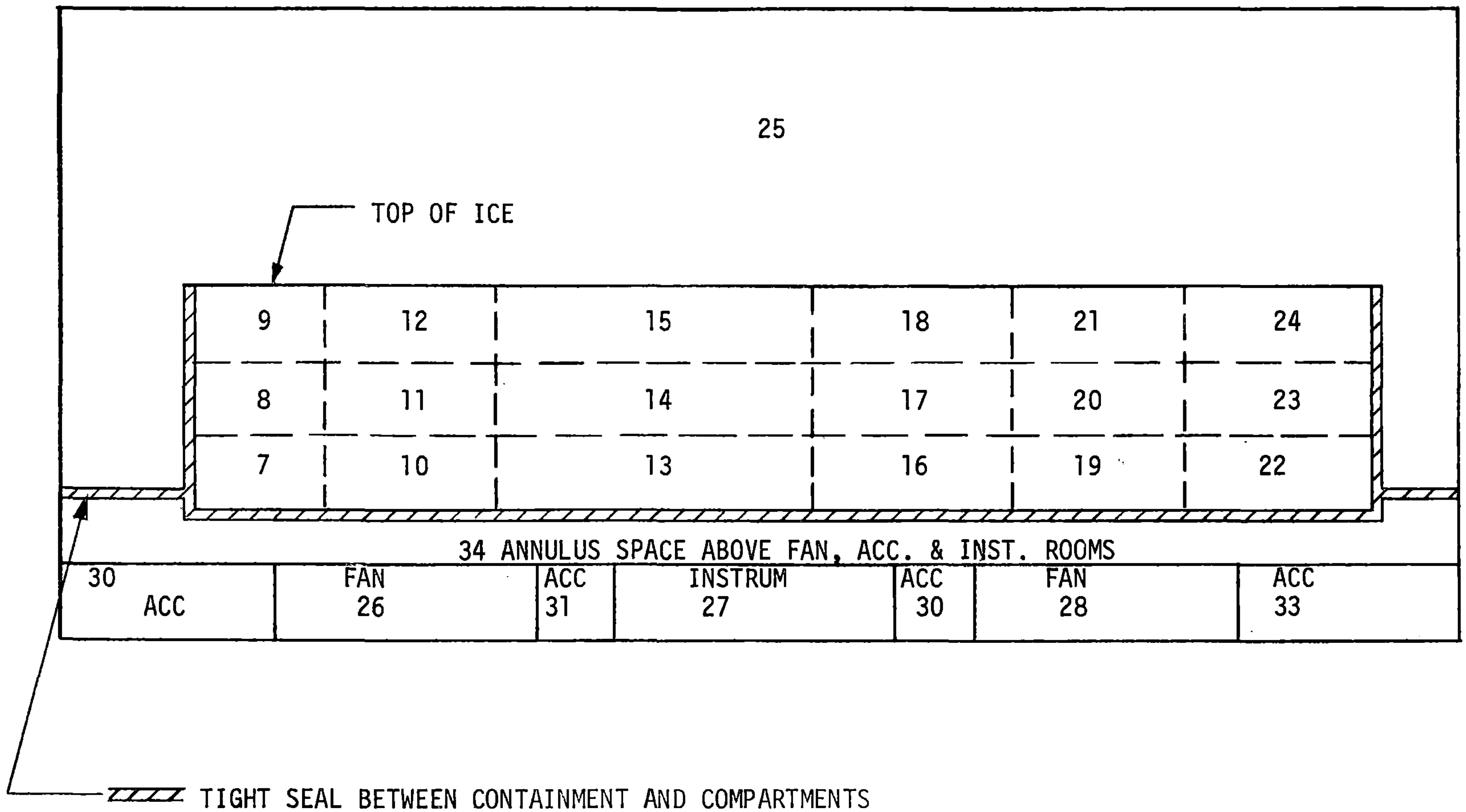
CONTAINMENT SECTION VIEW

11-1-72 Fig. 6.2-6



PLAN VIEW AT ICE CONDENSER ELEVATION-ICE CONDENSER COMPARTMENTS

11-1-72 Fig. 6.2-7

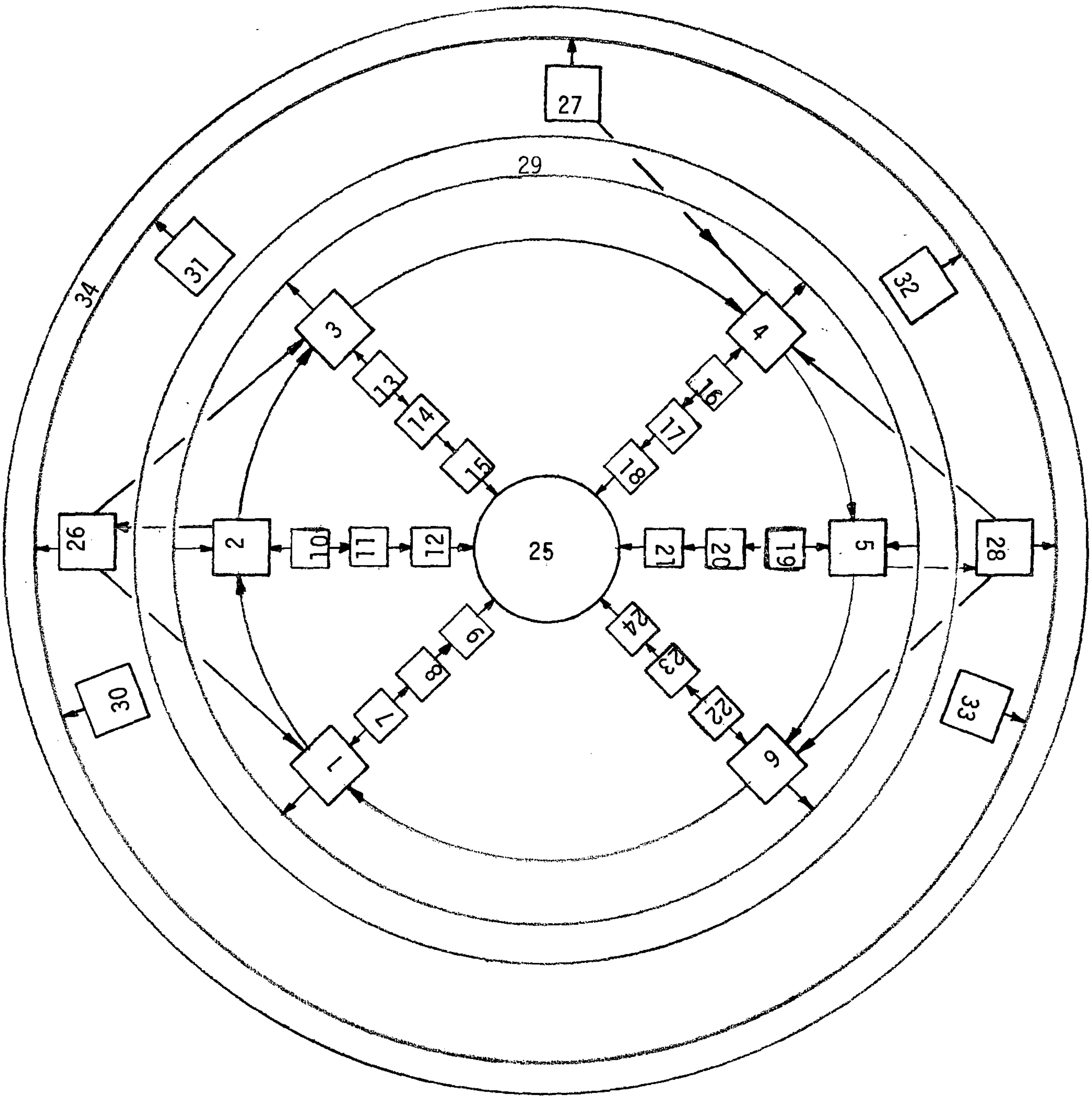


TMD CONTAINMENT VOLUME REPRESENTATION

TMD CONTAINMENT VOLUME REPRESENTATION

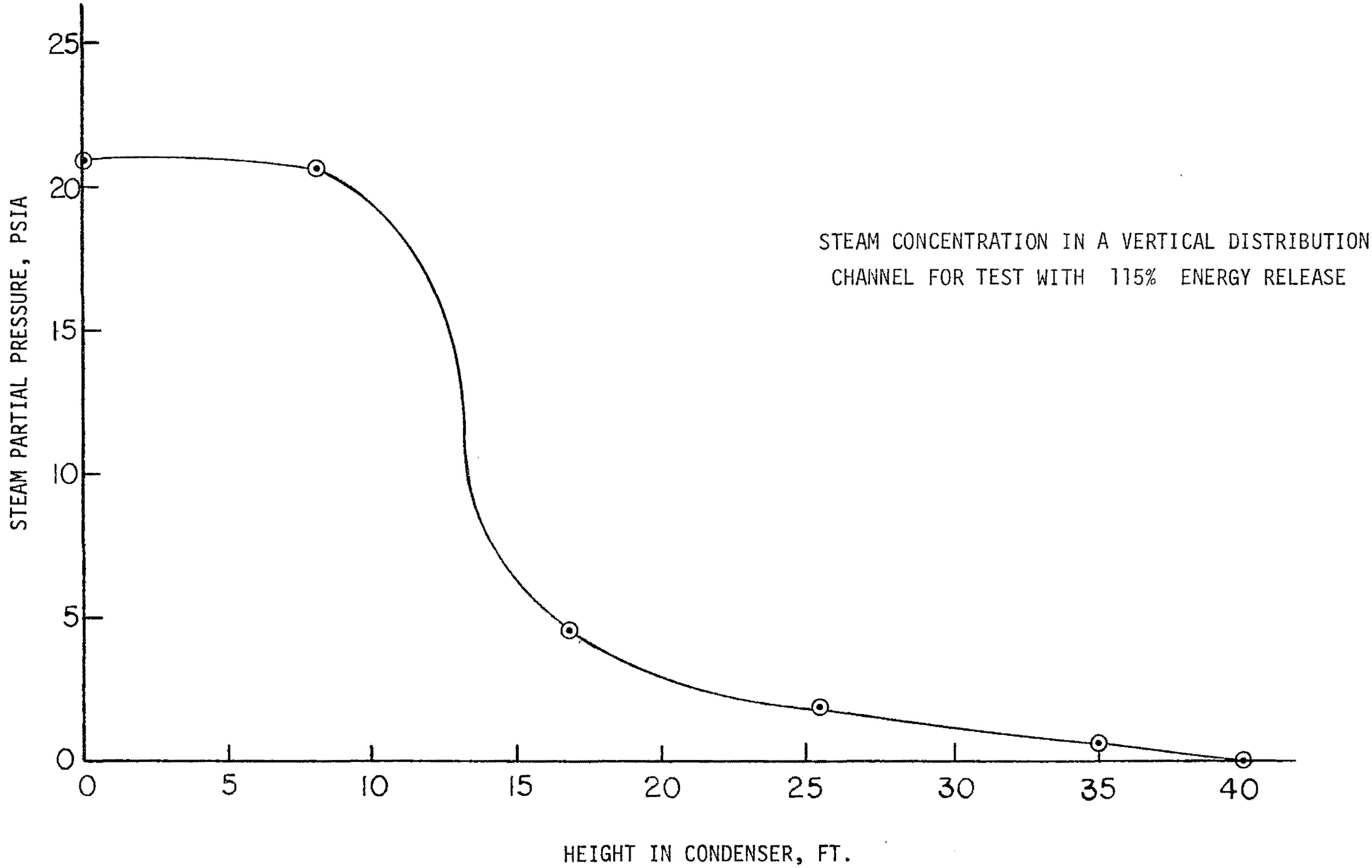
11-1-72

Fig. 6.2-8

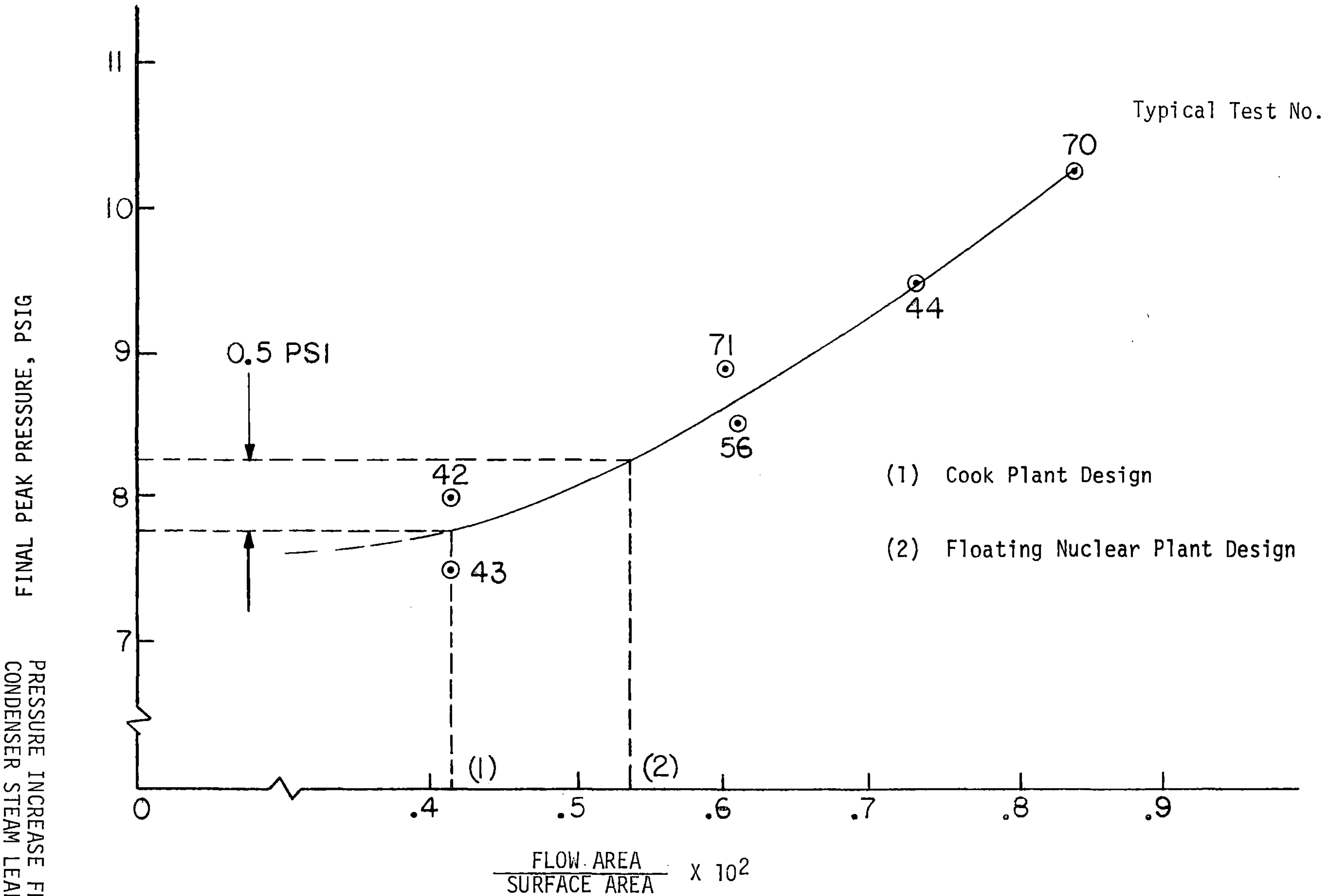


TMD CODE NETWORK

11-1-72 Fig. 6.2-9



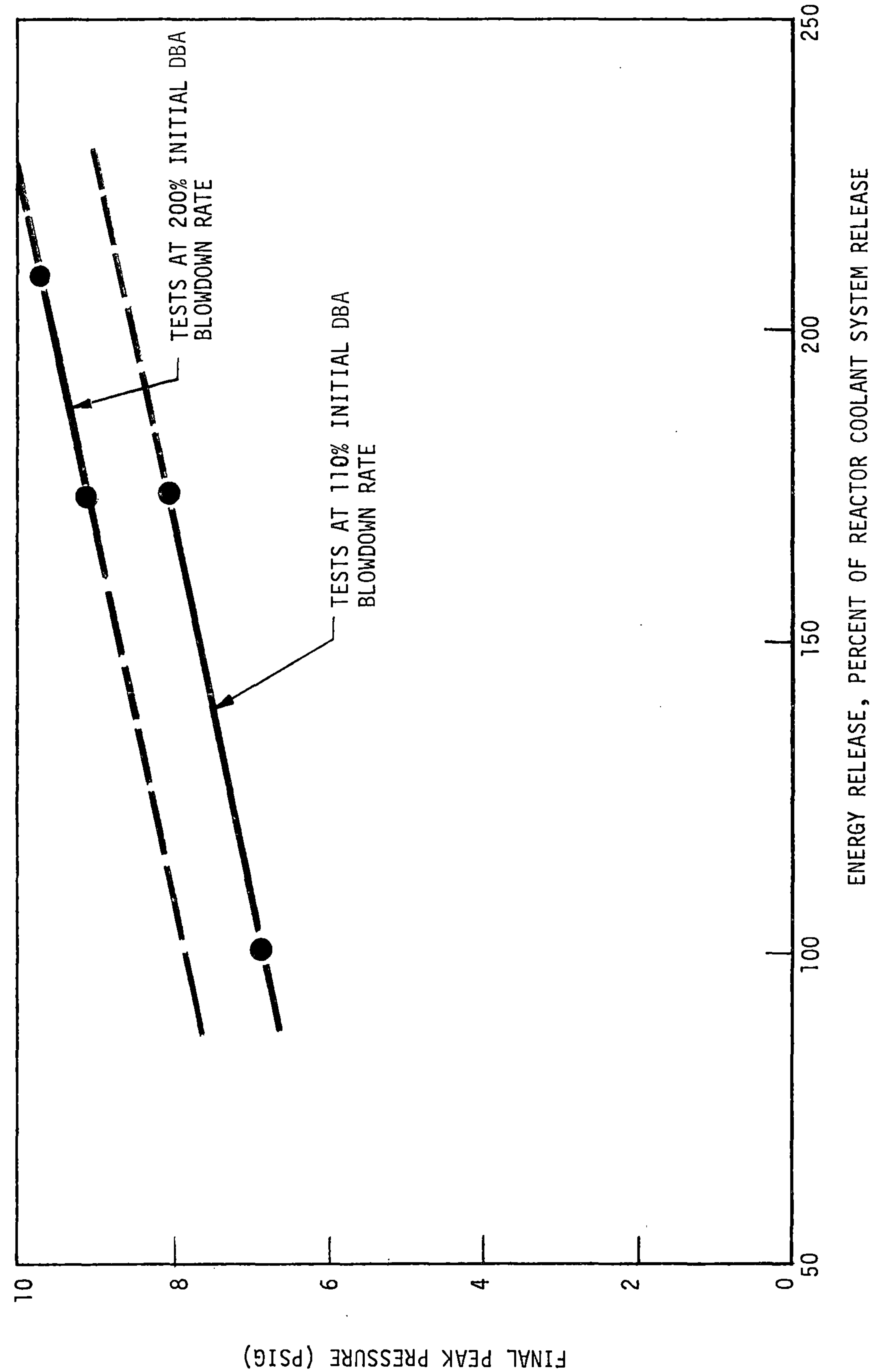
11-1-72 Fig. 6.2-10



PRESSURE INCREASE FROM CONDENSER STEAM LEAKAGE

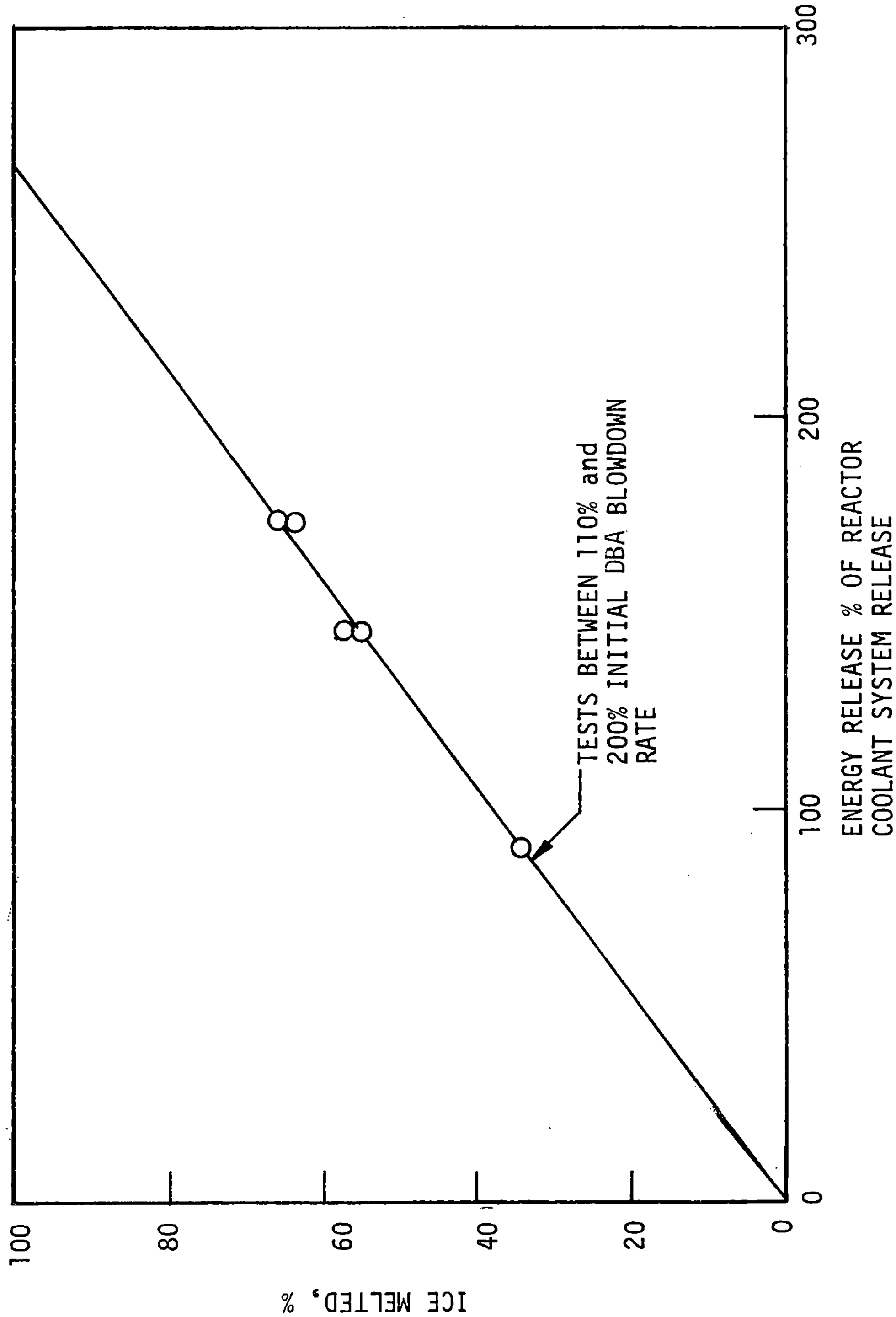
11-1-72

Fig. 6.2-11



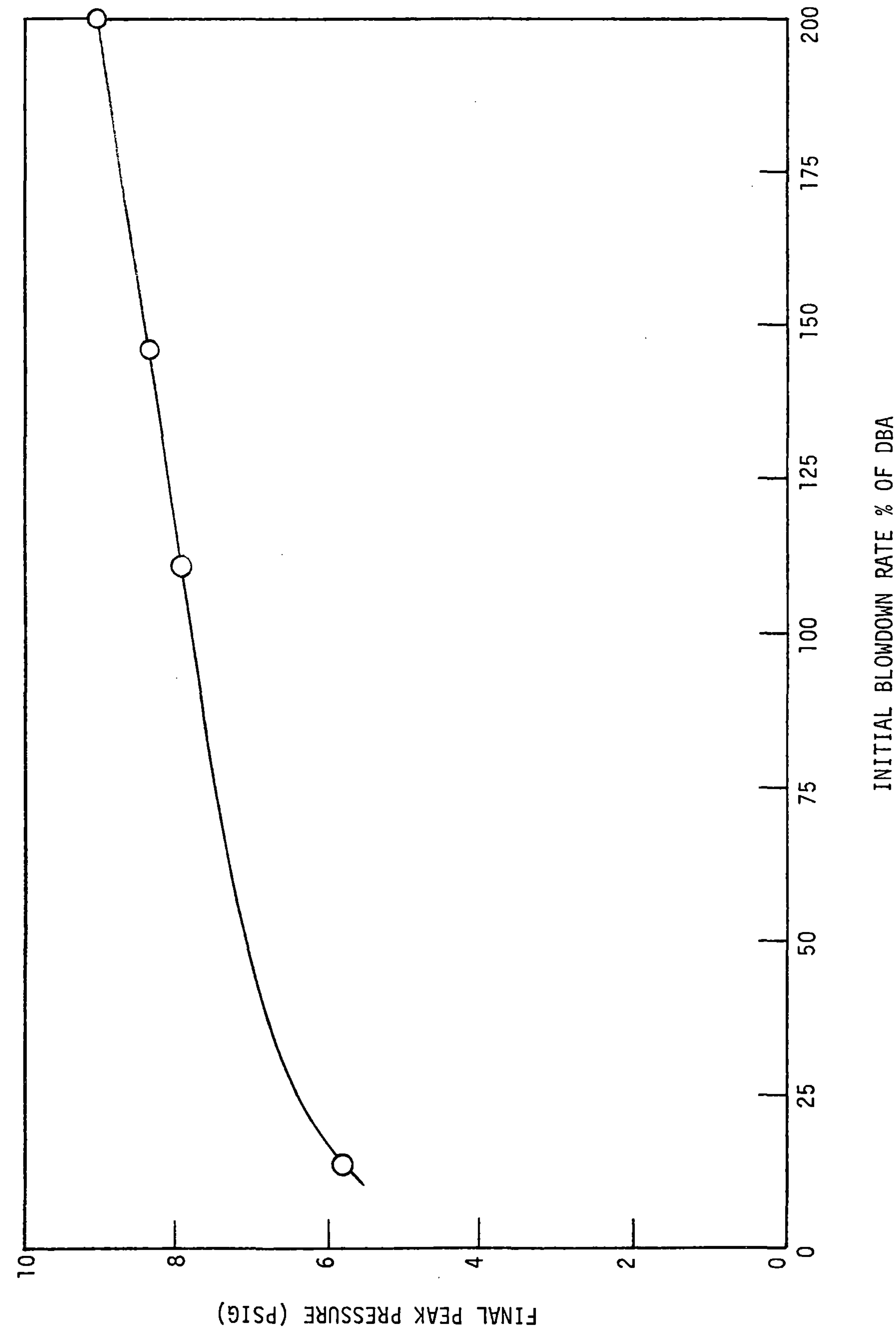
UPPER COMPARTMENT PEAK PRESSURE versus
ENERGY RELEASE FOR TESTS AT 110% and
200% OF INITIAL DBA BLOWDOWN RATE

11-1-72 Fig. 6.2-12

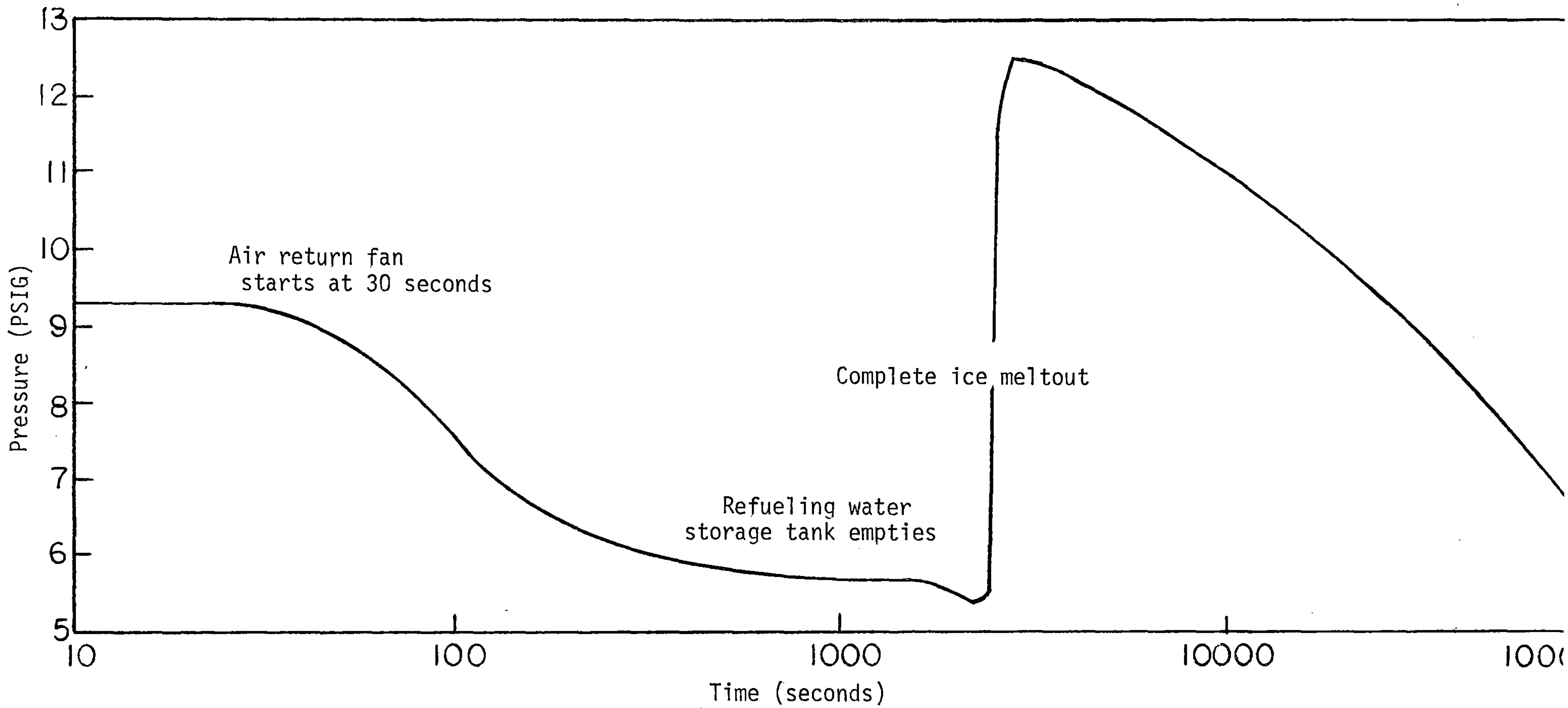


ICE MELTED versus ENERGY RELEASE FOR TESTS AT DIFFERENT BLOWDOWN RATES

11-1-72 Fig. 6.2-13

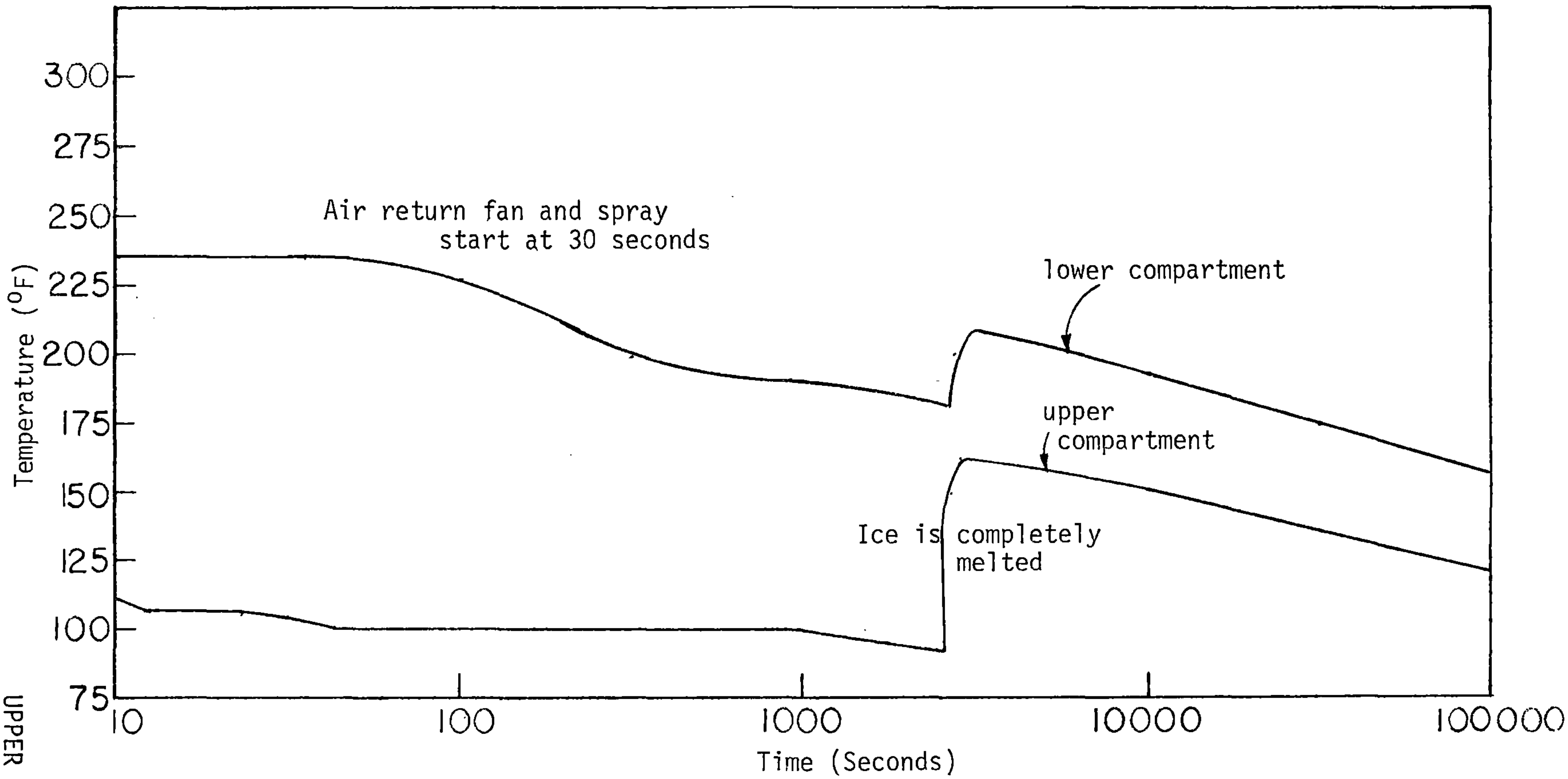


UPPER COMPARTMENT PEAK PRESSURE versus
BLOWDOWN RATE FOR TESTS WITH 175%
ENERGY RELEASE



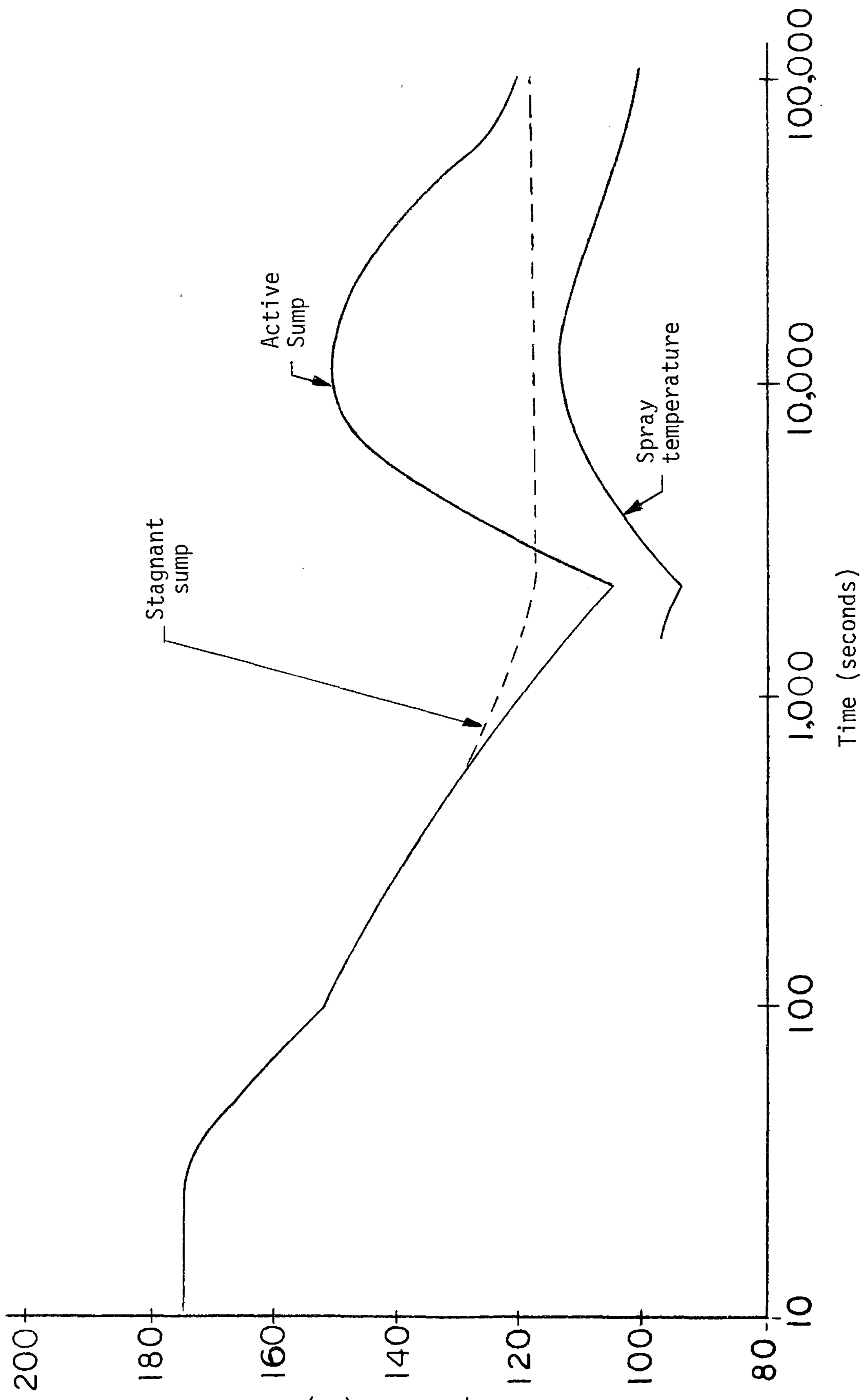
Long Term Containment Pressure Response to a Loss of Coolant Accident

11/1/72 Figure 6.2-15



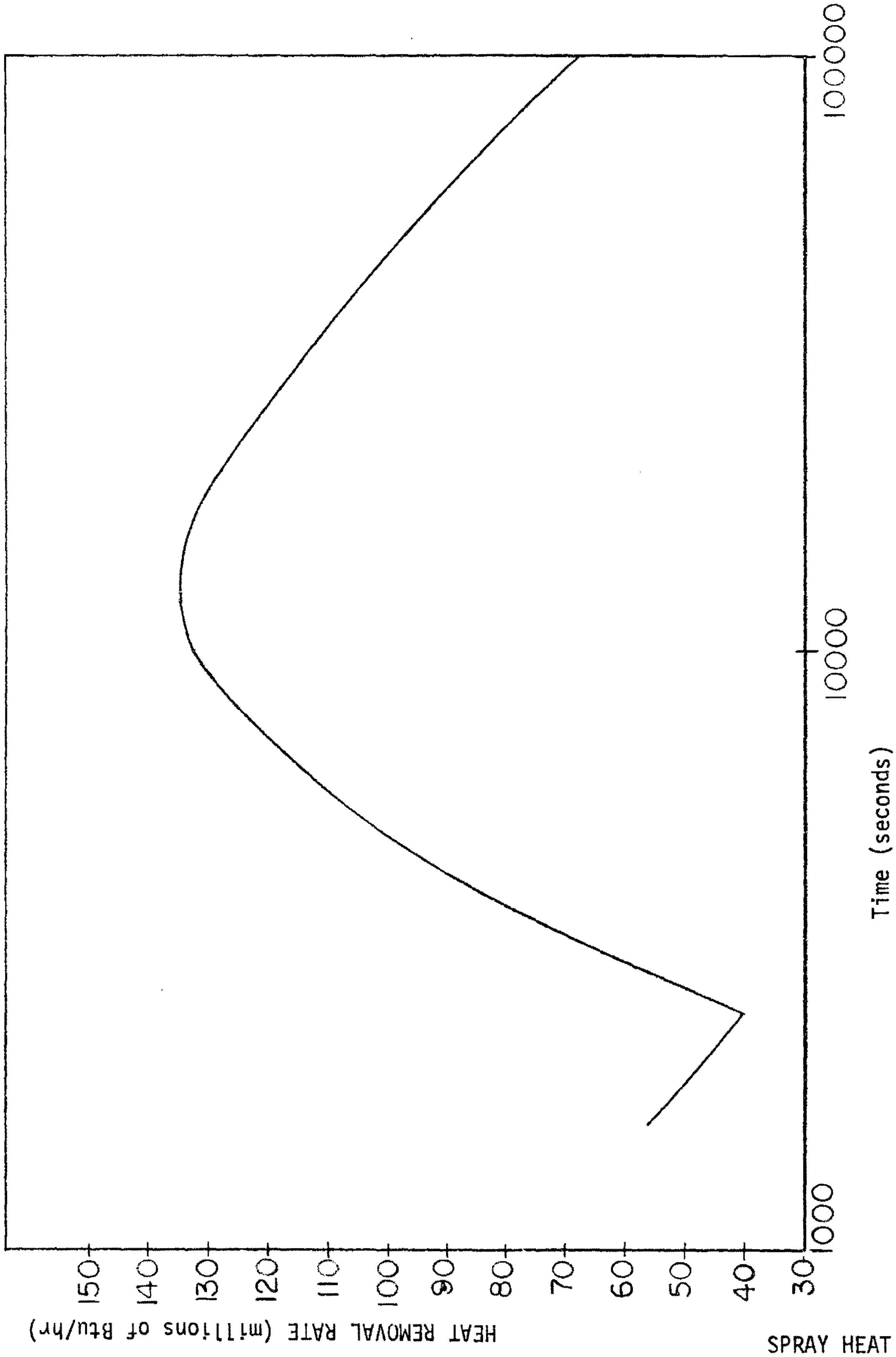
UPPER AND LOWER COMPARTMENT
TEMPERATURES vs. TIME

11/1/72 Figure: 6.2-16



CONTAINMENT SUMP AND SPRAY TEMPERATURES vs. TIME

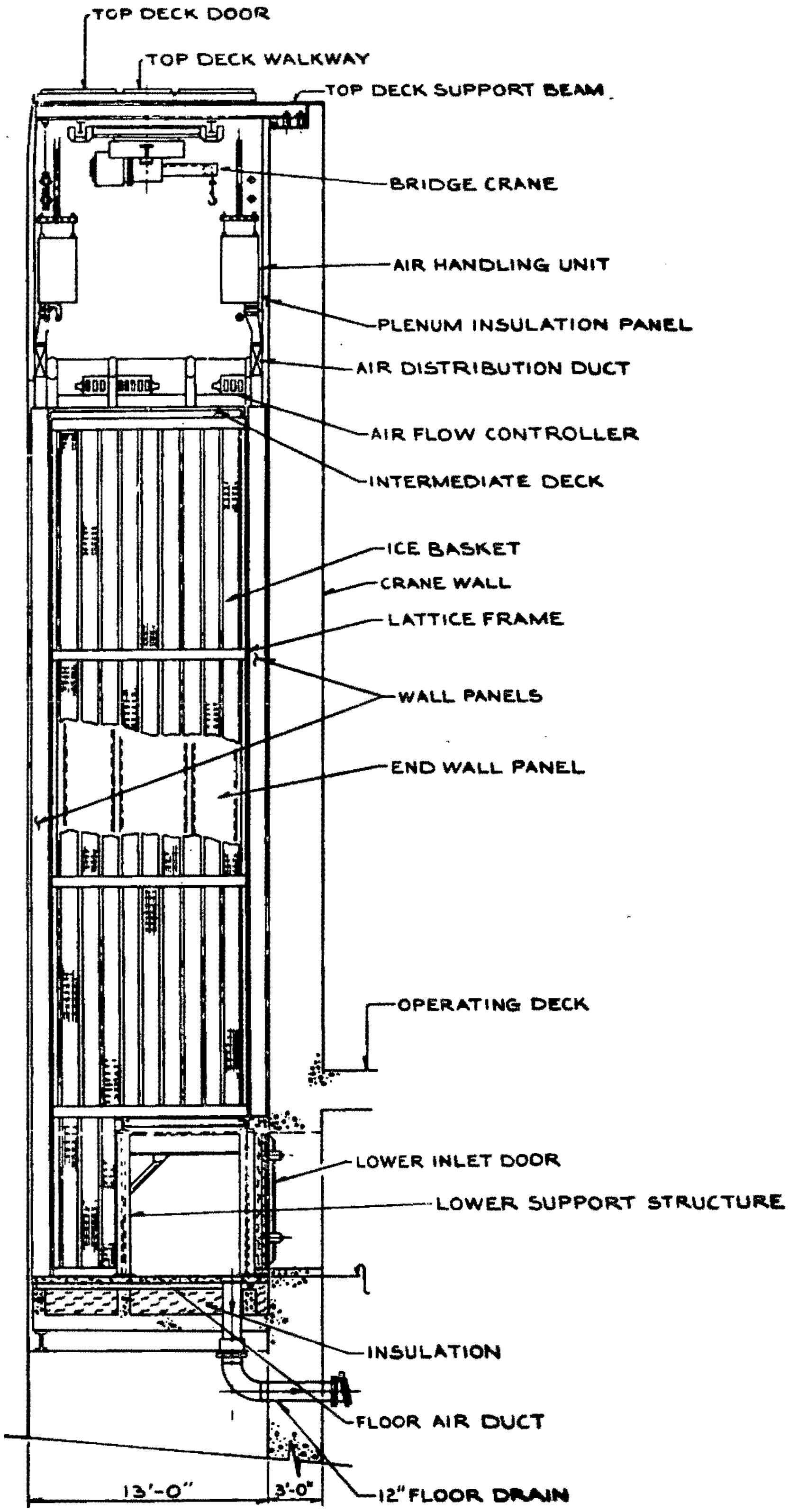
11/1/72 Figure: 6.2-17



SPRAY HEAT EXCHANGER

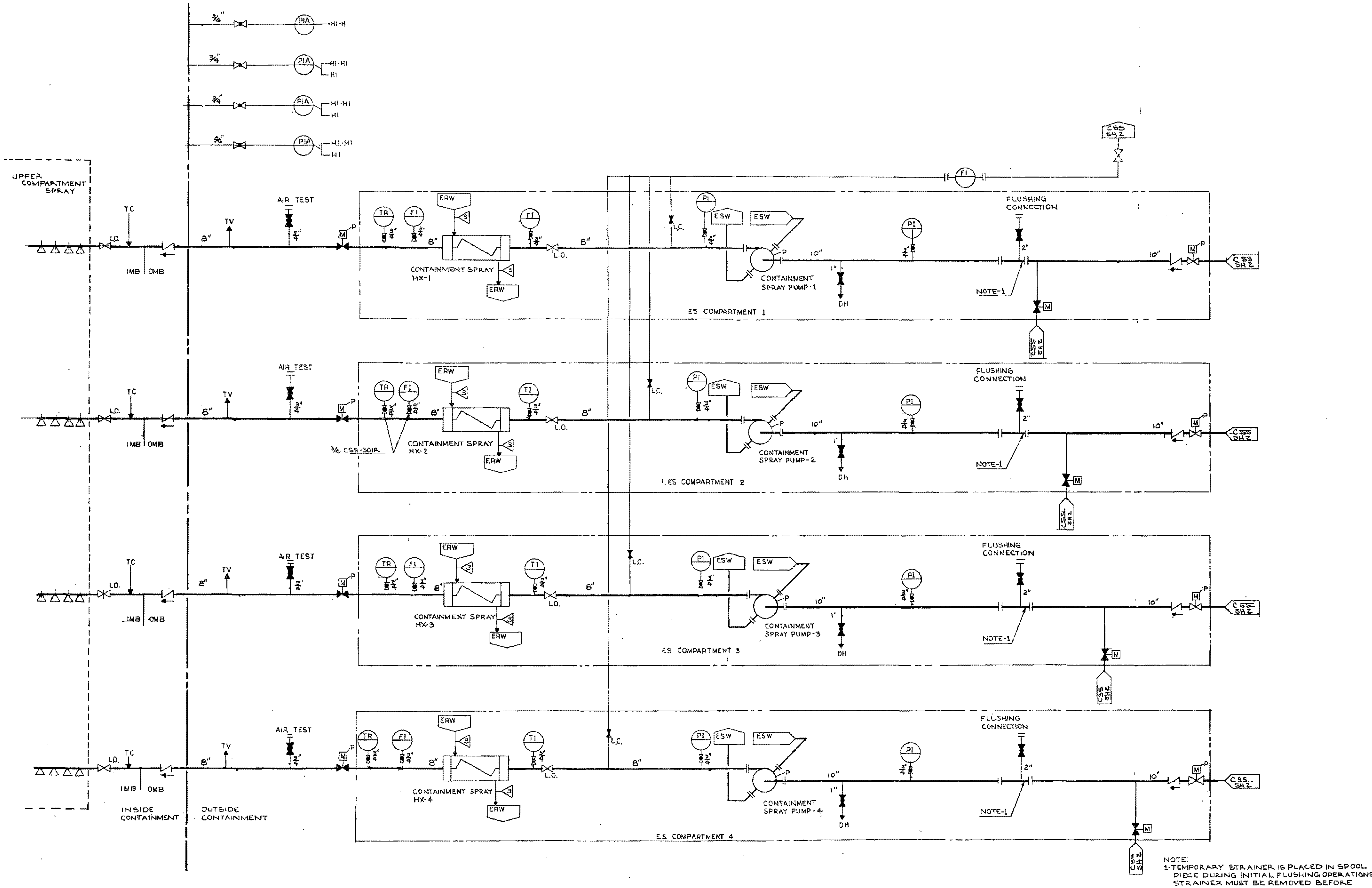
HEAT REMOVAL RATE

11/1/72 Figure 6.2-18



SECTION ELEVATION OF ICE CONDENSER

11/1/72 Figure 6.2-19

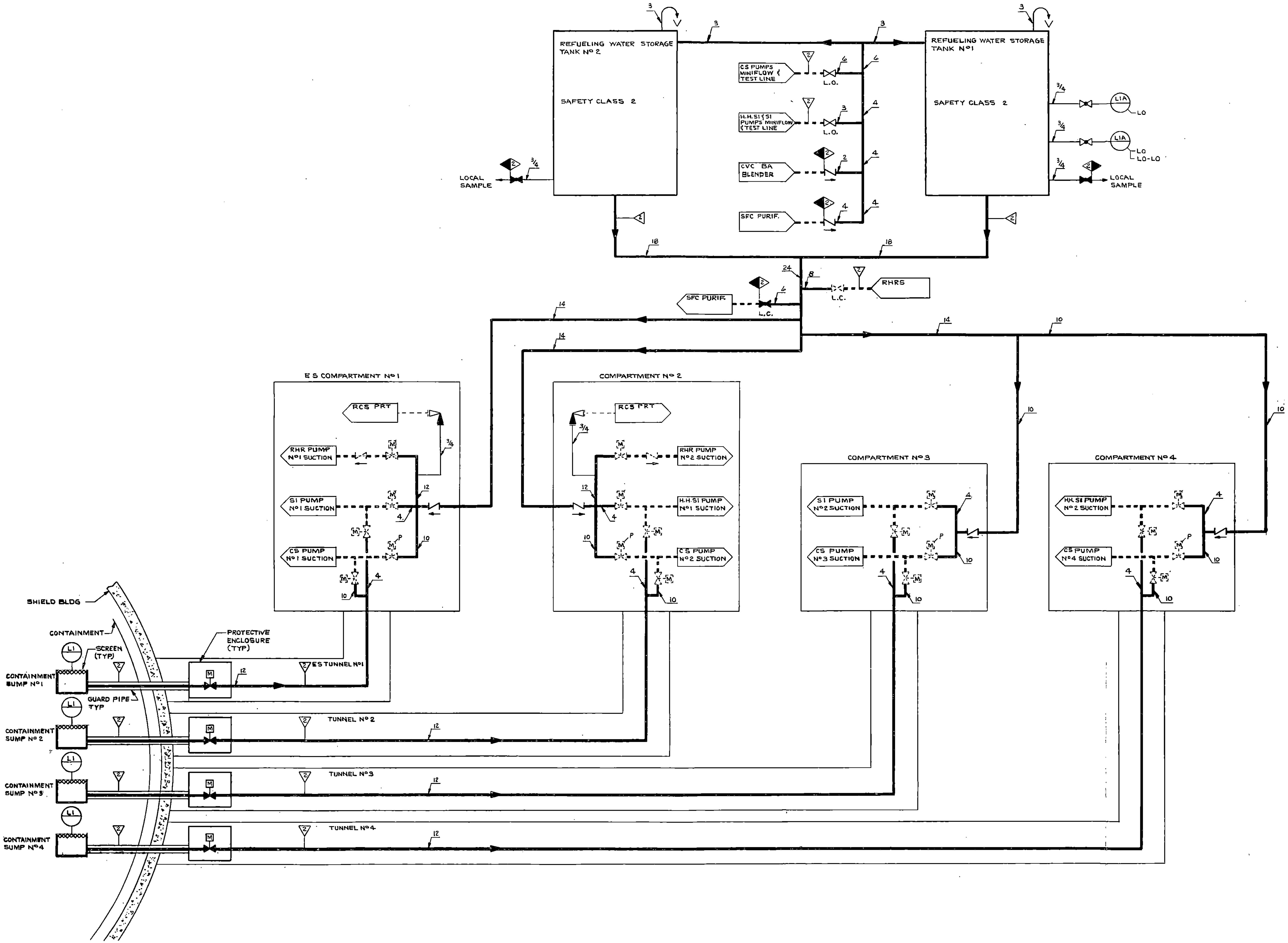


OFFSHORE POWER SYSTEMS

Title: CONTAINMENT SPRAY SYSTEM DIAGRAMS

11/1/72

Figure: 6.2-20
Sheet 1

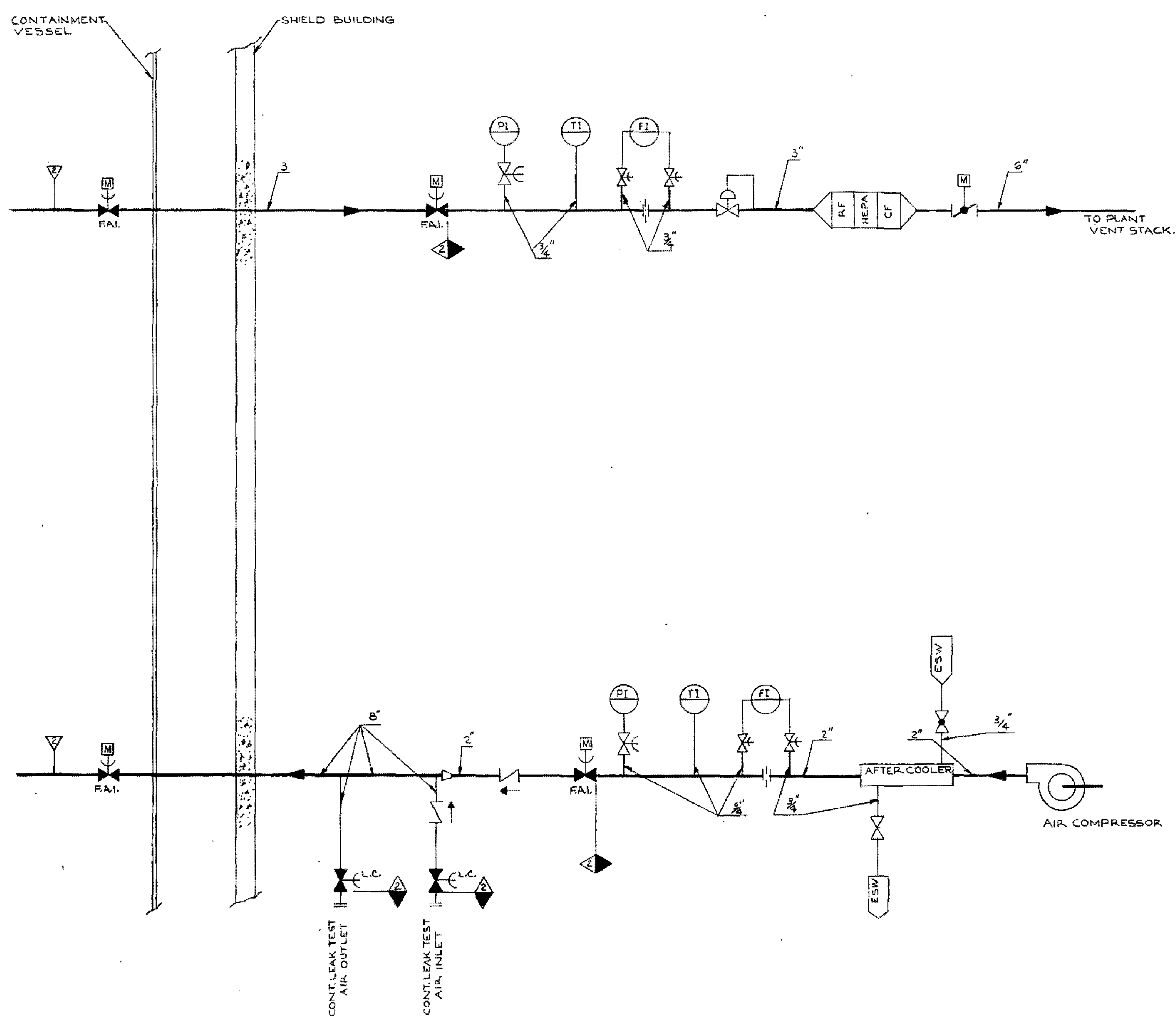


OFFSHORE POWER SYSTEMS

Title: CONTAINMENT SPRAY SYSTEM DIAGRAM

11/1/72

Figure: 6.2-20 Sheet 2

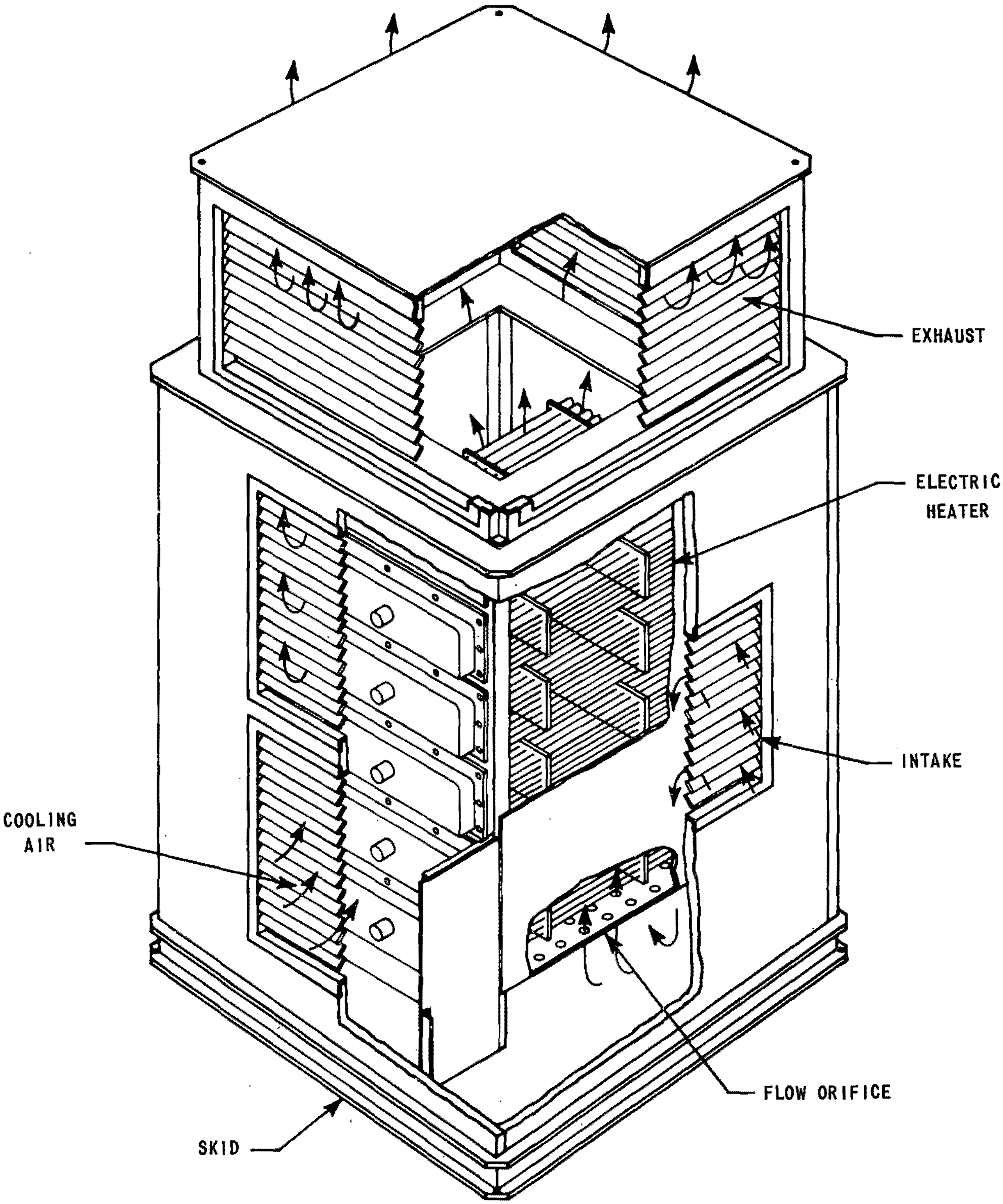


OFFSHORE POWER SYSTEMS

Title: POST ACCIDENT CONTAINMENT VENT SYSTEM

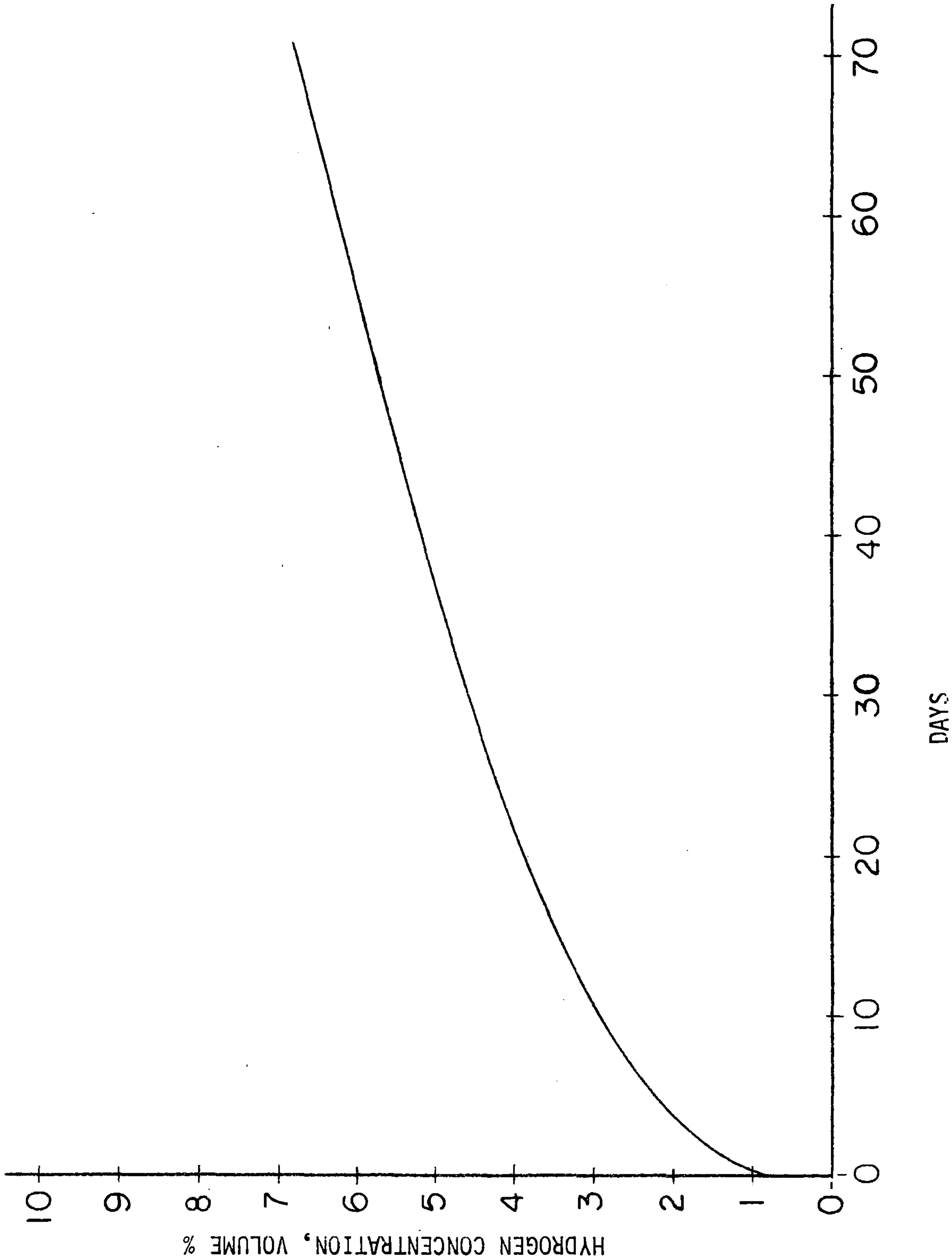
11/1/72

Figure 6.2-21



Electric Hydrogen Recombiner Cutaway

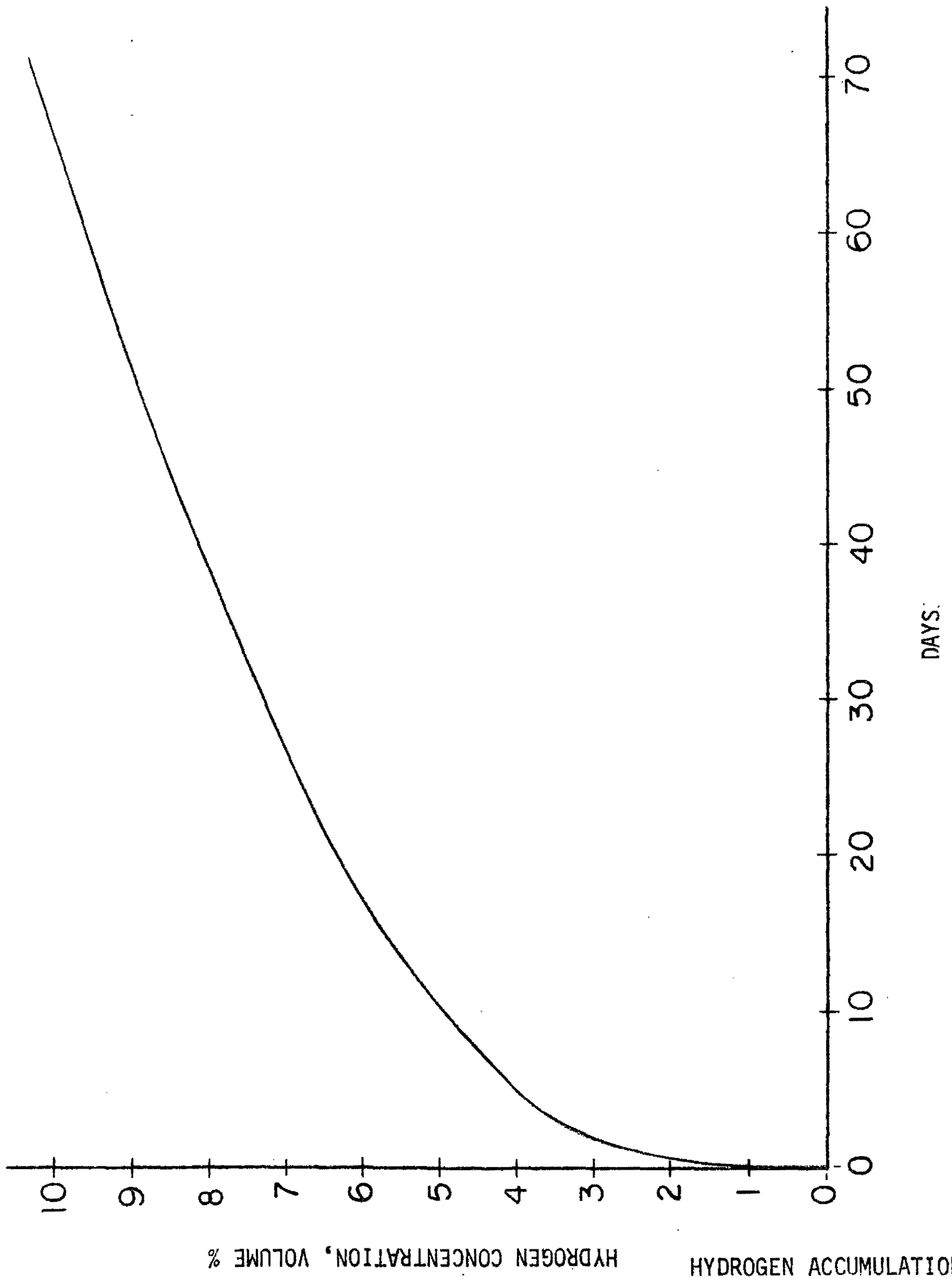
11/1/72 Figure: 6.2-22



DESIGN BASIS HYDROGEN

ACCUMULATION

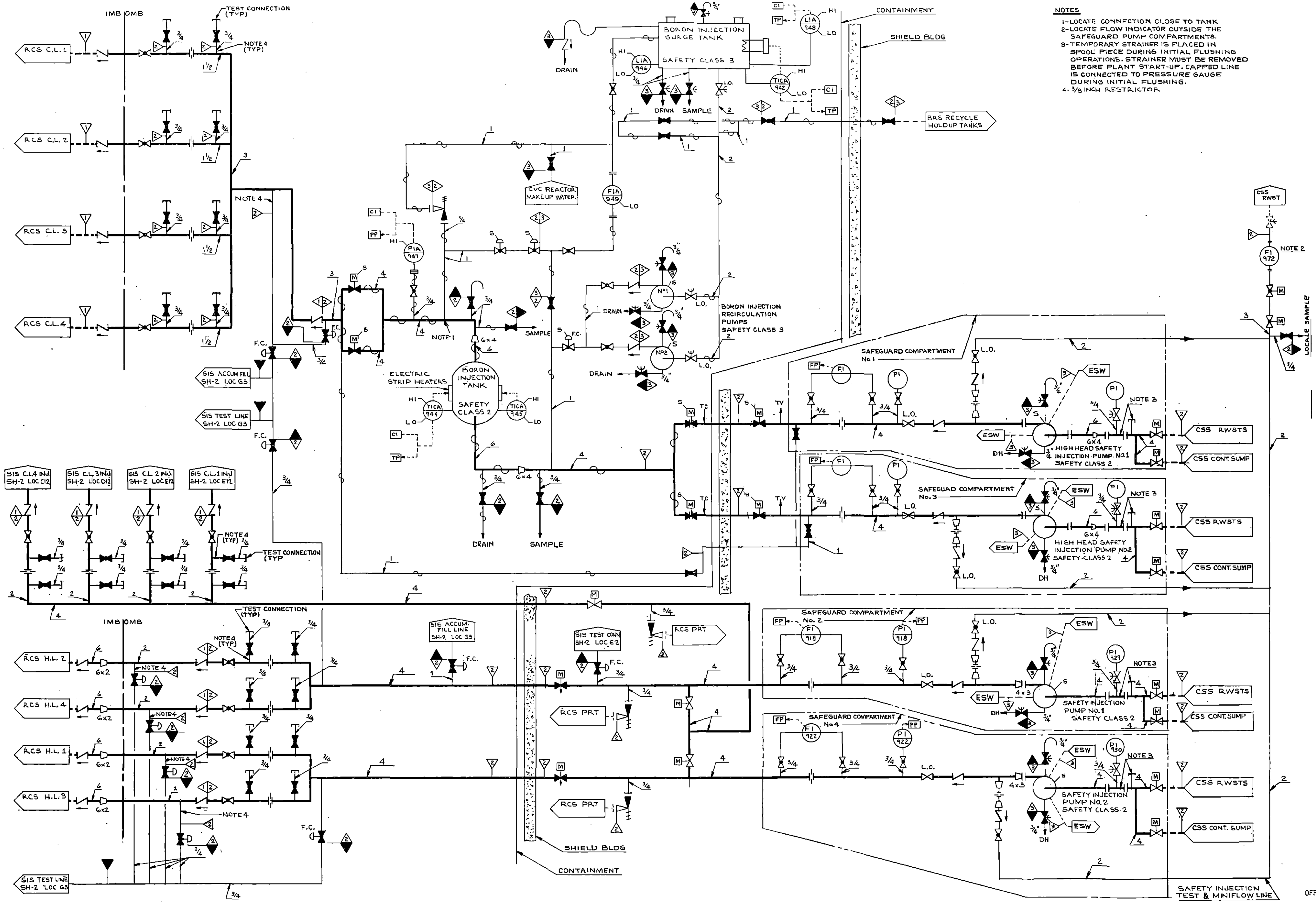
11/1/72 Figure 6.2-23



HYDROGEN ACCUMULATION BASED

ON AEC SAFETY GUIDE NO. 7

11/1/72 Figure 6.2-24



NOTES

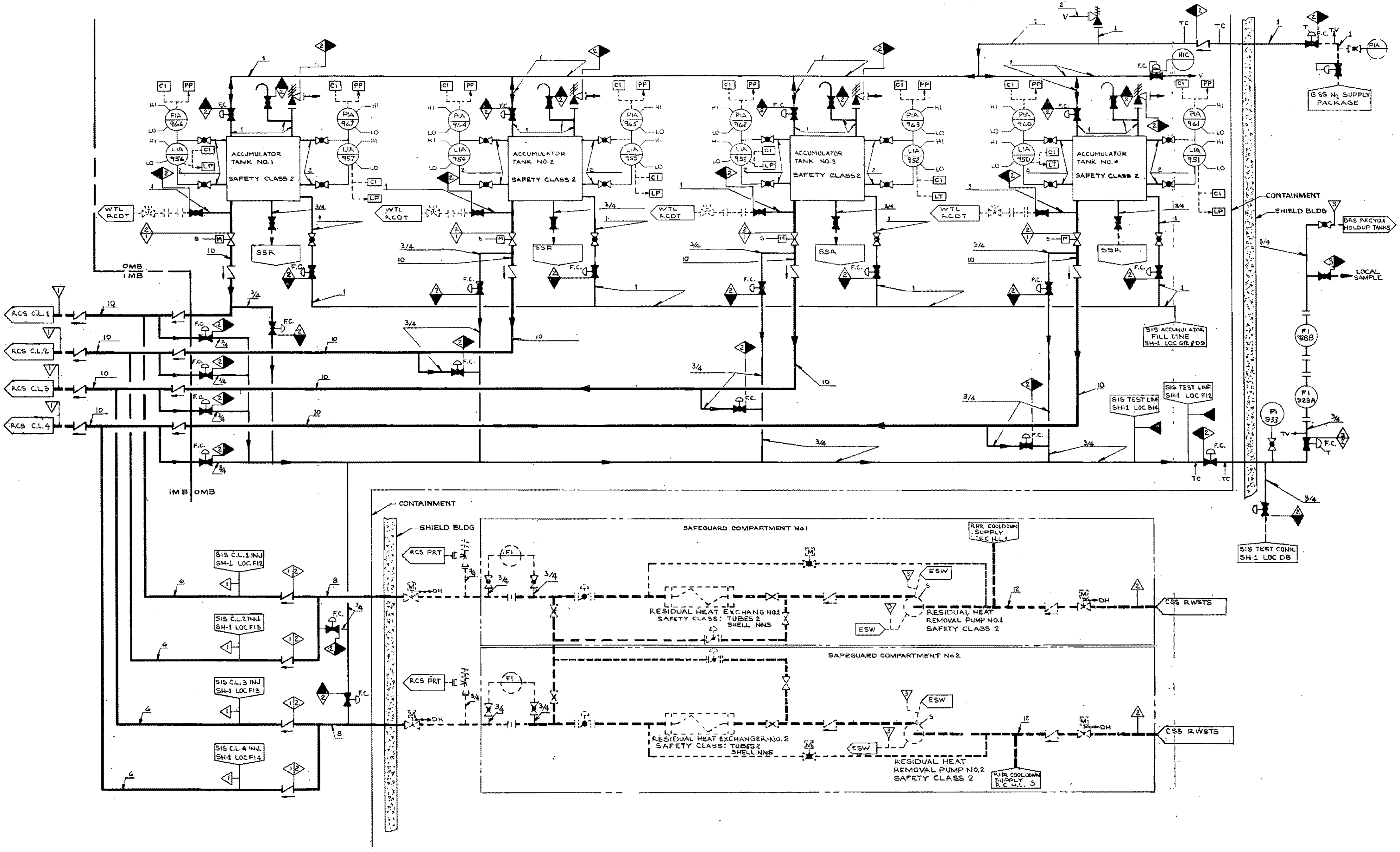
1. LOCATE CONNECTION CLOSE TO TANK
2. LOCATE FLOW INDICATOR OUTSIDE THE SAFEGUARD PUMP COMPARTMENTS.
3. TEMPORARY STRAINER IS PLACED IN SPOOL PIECE DURING INITIAL FLUSHING OPERATIONS. STRAINER MUST BE REMOVED BEFORE PLANT START-UP. CAPPED LINE IS CONNECTED TO PRESSURE GAUGE DURING INITIAL FLUSHING.
4. 3/8 INCH RESTRICTOR

OFFSHORE POWER SYSTEMS

Title: SAFETY INJECTION SYSTEM FLOW DIAGRAM

11/1/72

Figure: 6.3-1 Sheet 1

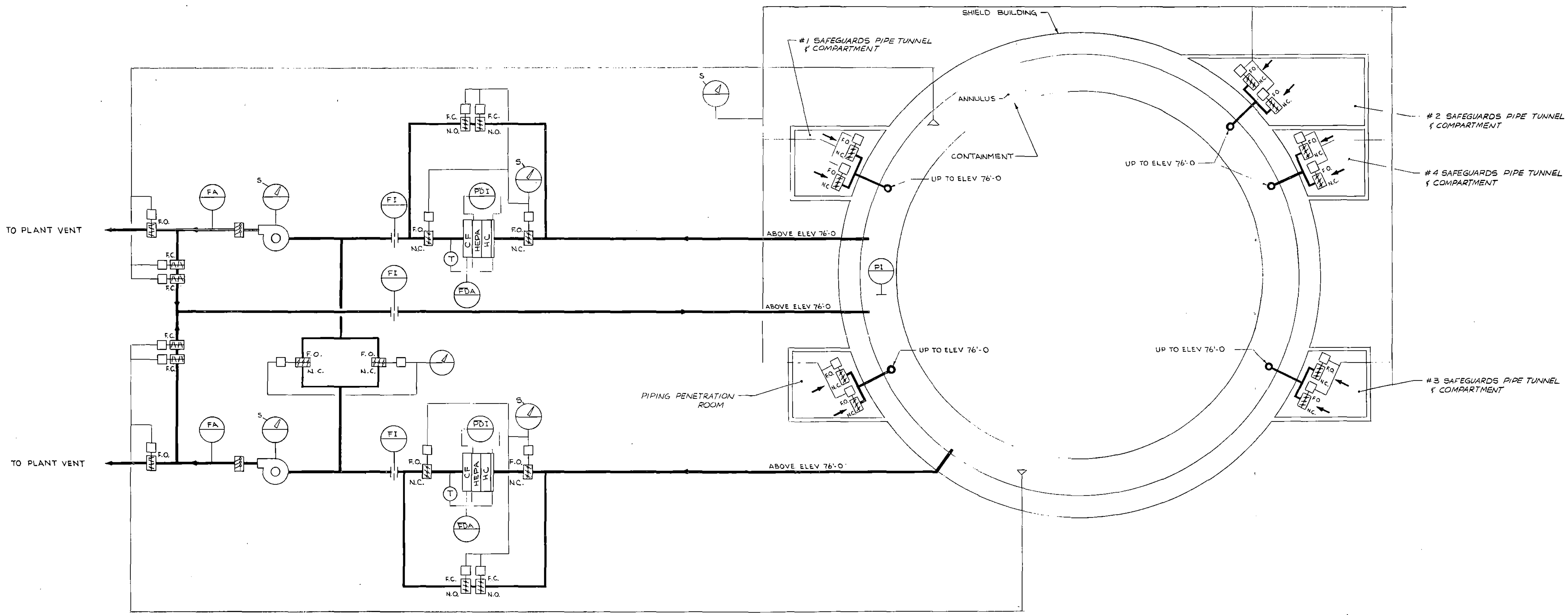


OFFSHORE POWER SYSTEMS

Title: SAFETY INJECTION SYSTEM

11/1/72

Figure: 6.3-1
Sheet 2



OFFSHORE POWER SYSTEMS

Title: CONTAINMENT ANNULUS FILTRATION SYSTEM - FLOW DIAGRAM

1/1/73 Figure: 6.4-1

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CHAPTER 7

INSTRUMENTATION AND CONTROLS

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CHAPTER 7

INSTRUMENTATION AND CONTROLS

7.1 INTRODUCTION

This section is presented in RESAR-3, Section 7.1 with the following exceptions:

1. RESAR-3, Sections 7.1.2.4 and 7.1.2.4.3 are modified and are to be found in Chapter 8.

2. RESAR-3, Section 7.1.2.3.2.b (CVC): Charging pumps are not used for Emergency Core Cooling. The High Head Safety Injection Pumps replace and perform the function of the CVC charging pumps during a LOCA.

3. RESAR-3, Section 7.1.2.3.2.d (ERW) Essential Raw Water Pumps provide cooling water to the containment spray heat exchanger coolers and is thus the heat sink for containment cooling.

4. RESAR-3, Section 7.1.2.3.3, line 19, page 7.1-15.
   Refer to Section 7.3 for a discussion of the Engineered Safety Features Actuation System.

5. RESAR-3, Section 7.1.2.3.4.5. The floating nuclear plant uses an ice condenser.

7.2 REACTOR TRIP SYSTEM

This section is presented in RESAR-3, Section 7.2.

7.3 ENGINEERED SAFETY FEATURES ACTUATION SYSTEM

This system is described in RESAR-3, Section 7.3. The following additional information is applicable to the Floating Nuclear Plant:

Where it is necessary to control four independent engineered safety features actuators using four independent actuating signals (for example, to start four diesels as discussed in Chapter 8 of this report) these signals will be generated by means of Engineered Safety Features Actuation System output circuits currently being designed.

The inputs to these circuits will be obtained from the Engineered Safety Features Actuation System described in RESAR-3, Section 7.3. Each of four mutually independent circuits will generate an independent actuating signal. The circuits accomplish this by generating a "Train A or Train B" output signal; i.e., failure of either train will not prevent generation of any of the four independent actuating signals.

The design of the four independent engineered safety features actuation system output circuits and the implementation of the four independent actuating signals generated by these circuits will be in accordance with IEEE-279 (Criteria for Protection Systems for Nuclear Power Generating Stations").

7.4 SYSTEMS REQUIRED FOR SAFE SHUTDOWN

This section is presented in RESAR-3, Section 7.4 with the following modifications related to implementation of the component cooling, service water, and containment cooling functions in the Floating Nuclear Plant design.

RESAR-3, Section 7.4.1.2.2 Pumps (Controls) is modified as follows:

Item 2. Charging and boric acid transfer pumps. These pumps will start automatically following a loss of normal electrical power. Start/stop motor controls will be provided for these pumps. The controls for the charging and boric acid pumps will be located outside as well as inside the control room.

Item 3. Essential service water pumps. These pumps will start automatically following a loss of normal electrical power. Start/stop motor controls will be located outside as well as inside the control room.

Item 6. Containment lower compartment fan cooler units. These units will start automatically following a loss of normal electrical power. Start/stop motor controls will be located outside as well as inside the control room.

Item 7. <u>Control room ventilation (including the control room air inlet dampers)</u>. These units will start automatically following a loss of normal electrical power. A start/stop switch located just outside the control room will be provided for these units; a control to close the inlet air damper will also be provided. These controls duplicate functions inside the control room.

Item 8. <u>Auxiliary raw water pumps</u>. These pumps will start automatically following a loss of normal electrical power. Start/stop motor controls will be located outside as well as inside the control room.

Item 9. <u>Essential raw water pumps</u>. These pumps will start automatically following a loss of normal electrical power. Start/stop motor controls will be located outside as well as inside the control room.

RESAR-3, Section 7.4.1.4 Equipment and Systems Available for Cold Shutdown.

Add the following items:

10. Essential service water pumps (see Section 9.2.2).

11. Essential raw water pumps (see Section 9.2.2).

12. Auxiliary raw water pumps (see Section 9.2.1).

7.5 SAFETY-RELATED DISPLAY INSTRUMENTATION

This section is presented in RESAR-3, Section 7.5

7.6 ALL OTHER SYSTEMS REQUIRED FOR SAFETY

This section is presented in RESAR-3, section 7.6, with the following exceptions:

Section 7.6.1.1 through 7.6.1.2 is modified and incorporated as part of Chapter 8 of this report.

Section 7.6.3 is not applicable to the Floating Nuclear Plant.

7.7 PLANT CONTROL SYSTEMS

This section is presented in RESAR-3, Section 7.7, with the following exceptions:

1. RESAR-3, Figure 7.7-10 is modified as shown on Figure 7.7-10 of this report.

   RESAR-3, Figure 7.7-15 is replaced by Figure 7.7-15 of this report, which reflects a typical gross gamma detector.

   RESAR-3, Figure 7.7-16 is replaced by Figure 7.7-16 of this report, which reflects a typical gross gamma detector.

2. RESAR-3, Section 7.7.1.12 is modified as follows for the Floating Nuclear Plant:

   The failed fuel detector is supplied as standard equipment, not as an option. The type selected is a gross gamma monitor. Paragraphs 1 and 2, page 7.7-31 RESAR-3 are to be excluded and a rewritten section is included in this report. Exact values of activity as indicated in Paragraphs 2 and 3, Page 7.7-32 are influenced by the specific manufacturers equipment; however, activity values equivalent to those used in Paragraphs 2 and 3 will be used to reduce reactor power and determine the need for chemistry sampling. The equipment reflected in drawing 7.7-16 is representative of a specific supplier and may change significantly once design is finalized and a supplier is selected.

The following are the substitute paragraphs for Section 7.7.1.12, page 7.7-31, RESAR-3, (paragraphs 1 and 2).

7.7.1.12 Gross Failed Fuel Detector

1. The gross failed fuel detector is connected across a portion of the CVC (Figure 7.7-15). The coolant sample (already cooled by CVCS equipment) passes into a fixed volume chamber. A gamma scintillation detector utilizing a sodium iodide crystal optically coupled to a photomultiplier tube counts and amplifies the activity of the sample. The sample continues to flow from the sample chamber back into the CVC system and then to the reactor coolant system. Flow is so regulated that maximum sample time (core to detector) does not exceed 6 minutes.

2. Figure 7.7-16 shows the block diagram of the gross failed fuel detector channel. The detector, preamplifier, sample chamber and associated equipment, are located in the CVC area outside the

containment. The signal processing equipment and readout are mounted on a rack located in the control room. The gamma signal generated is displayed on a recorder located in the rack.

3. Typical control loop symbols are shown in Appendix "A" to this report.

Typical applications of these symbols are illustrated in Figures 7.7-17 and 7.7-18.

7.8 FIRE DETECTION AND ALARMS

A fire and smoke detection alarm system gives immediate audible and visual alarms, as well as locating and identifying the existence of smoke or fire.

The design philosophy used seeks rapid identification of the location of a fire so that corrective measures may be taken to limit damage.

The monitored region of the plant is divided into functional areas. The detection system for each area is independent of every other area, except for a common alarm panel in the control room.

Areas of fire detection include, but are not necessarily limited to:

Cable pull area

Major Electrical containment penetration areas

Diesel generator enclosures

*Safety related switchgear rooms

*Switchyard and Battery rooms

*Note: Some of the detectors located in ventilation facilities (Section 6.2, 6.4, and 9.4) servicing certain areas will be utilized in place of detectors located in the area proper.

7.9 HULL LEAKAGE SURVEILLANCE

7.9.1 BASIS FOR DESIGN

The system provides a positive means for the detection of gross leakage in any of the watertight hull compartments. The units must be rugged, shock and vibration resistant and must actuate at very low water levels.

7.9.2 SYSTEM DESCRIPTION

The water level detector consists of a donut type float, surrounding a hollow shaft. The shaft contains a sealed magnetic switch. The float contains a magnet and actuates the switch when water level raises the float magnet in close proximity to the switch (approximately 2" of movement). Closing of the detector switch sounds a common alarm in the main control room and provides position indication on a local panel. Operations personnel can then check the local panel for location and inspect the indicated alarmed compartment for leakage.

Although one watertight compartment may be divided into two or more sections, drain openings are provided on the dividing bulkheads. This assures a free flow of water within the watertight compartment and permits the use of a single detector to monitor for leakage.

7.9.3 TESTING

The basic detector used has had years of maritime use on both Navy and Merchant ships.

Periodic inspections of the compartment space will be conducted.

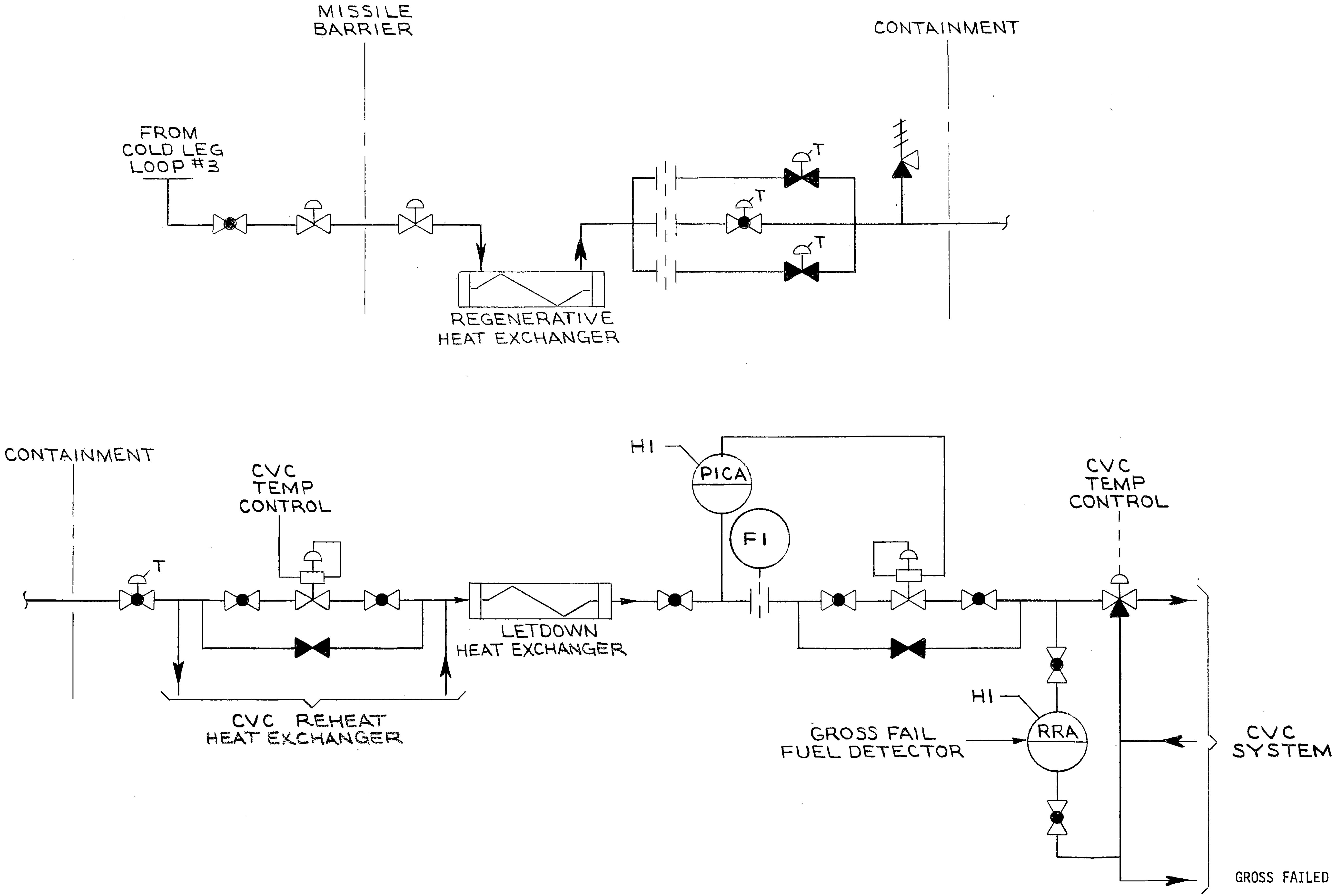
AREA 1	ESSENTIAL RAW WATER	ENGINEERED SAFEGUARDS
AREA 2	ESSENTIAL SERVICE WATER	ENGINEERED SAFEGUARDS
AREA 3	COMPONENT COOLING WATER	ENGINEERED SAFEGUARDS
AREA 4	DIESEL GENERATORS	ENGINEERED SAFEGUARDS
AREA 5	VENTILATION & ISOLATION	ENGINEERED SAFEGUARDS — CONTAINMENT ISOLATION
AREA 6	SPRAY	ENGINEERED SAFEGUARDS — CONTAINMENT ISOLATION
AREA 7	AUXILIARY FEEDWATER	ENGINEERED SAFEGUARDS
AREA 8	RESIDUAL HEAT REMOVAL	ENGINEERED SAFEGUARDS
AREA 9	RECIRCULATION	ENGINEERED SAFEGUARDS — SAFETY INJECTION
AREA 10	INJECTION	ENGINEERED SAFEGUARDS — SAFETY INJECTION
AREA 11	TEST	ENGINEERED SAFEGUARDS — SAFETY INJECTION
AREA 12	LETDOWN SYSTEM	CHEMICAL & VOLUME CONTROL SYSTEM
AREA 13	REACTOR MAKE-UP BORATION	CHEMICAL & VOLUME CONTROL SYSTEM
AREA 14	NUCLEAR INSTRUMENTATION AND ROD CONTROL	
AREA 15	PRESSURIZER AND LOOPS	REACTOR COOLANT SYSTEM
AREA 16	STEAM GENERATOR	REACTOR COOLANT SYSTEM
AREA 17	FEEDWATER AND MAIN STEAM	
AREA 18	TURBINE/GENERATOR	

NOTES:

1. SYSTEMS ARE RELATED TO EACH OTHER TO OPTIMIZE OVERALL PLANT OPERATION WITH AREA 14 ACTING AS THE FOCAL POINT.

TYPICAL LOCATION OF
CONTROL BOARD SYSTEMS

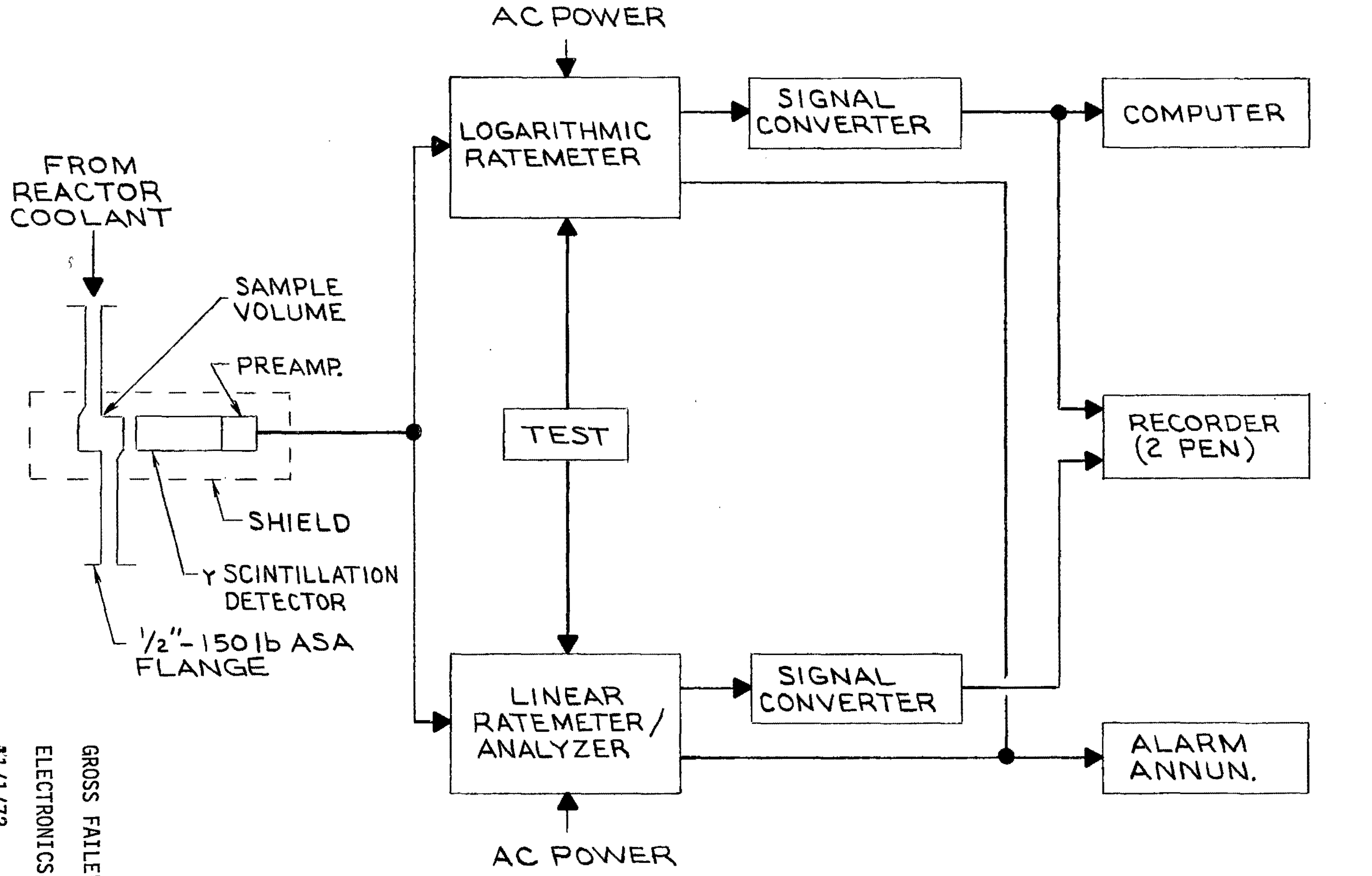
11/1/72 Fig.7.7-10



GROSS FAILED FUEL DETECTOR

FLOW DIAGRAM

11/1/72 Fig. 7.7-15



GROSS FAILED FUEL DETECTOR
ELECTRONICS DIAGRAM

11/1/72 Fig. 7.7-16

<u>TYPICAL METHOD OF REFERENCING INSTRUMENT DRAWINGS ON P & I DIAGRAMS</u>

A rectangular block is shown on the P & I Diagram containing an instrument drawing number. (Simple loops, local pressure gages, or thermometers, do not require this block.) If the instrument has a control function, the final actuator is either referenced or connected as shown in the example below. The referenced instrument drawing shows actual hardware needed to obtain the desired measurements and functions.

Example:

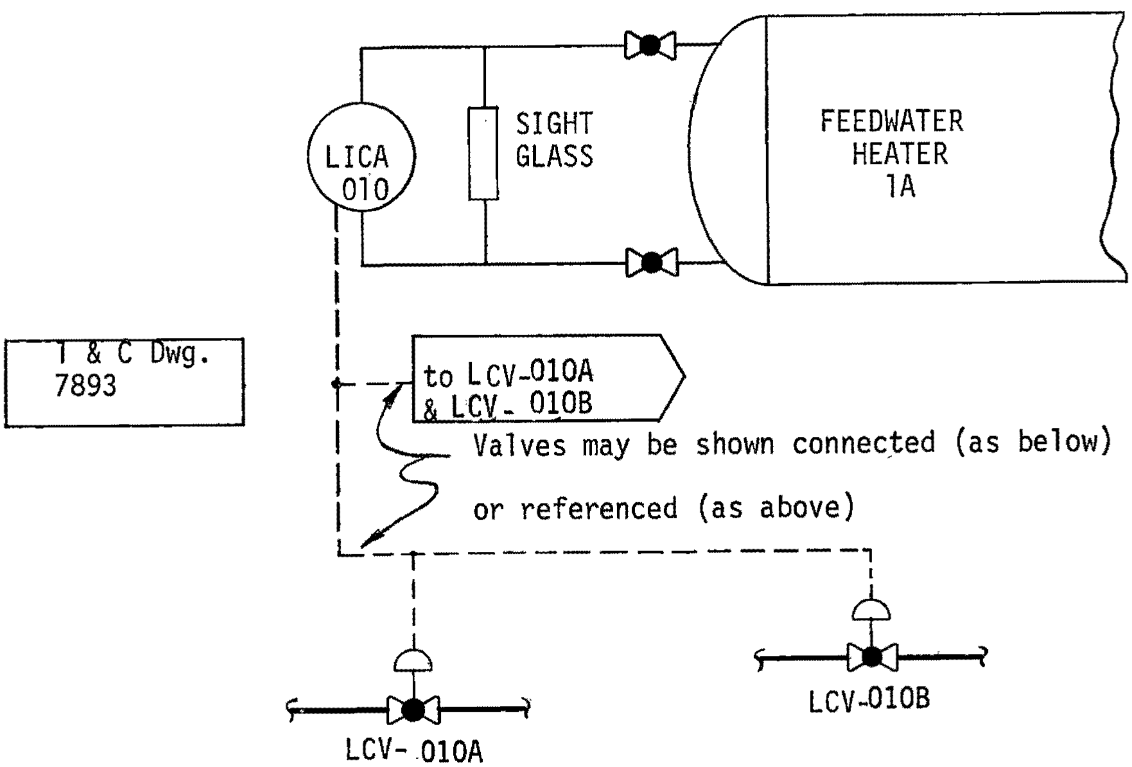
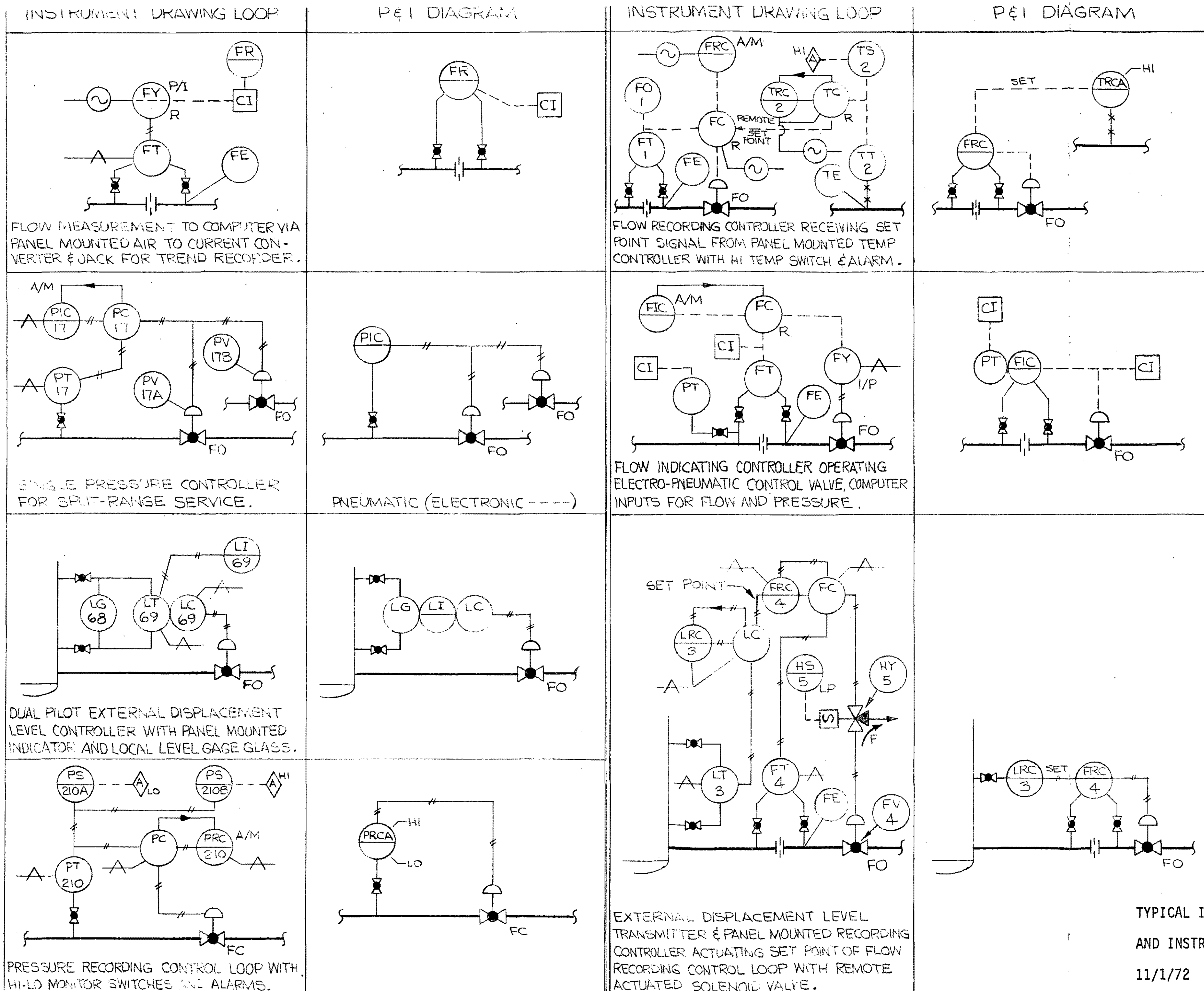


Fig. 7.7-17



FLOW MEASUREMENT TO COMPUTER VIA PANEL MOUNTED AIR TO CURRENT CONVERTER & JACK FOR TREND RECORDER.

FLOW RECORDING CONTROLLER RECEIVING SET POINT SIGNAL FROM PANEL MOUNTED TEMP CONTROLLER WITH HI TEMP SWITCH & ALARM.

SINGLE PRESSURE CONTROLLER FOR SPLIT-RANGE SERVICE.

PNEUMATIC (ELECTRONIC - - - -)

FLOW INDICATING CONTROLLER OPERATING ELECTRO-PNEUMATIC CONTROL VALVE, COMPUTER INPUTS FOR FLOW AND PRESSURE.

DUAL PILOT EXTERNAL DISPLACEMENT LEVEL CONTROLLER WITH PANEL MOUNTED INDICATOR AND LOCAL LEVEL GAGE GLASS.

PRESSURE RECORDING CONTROL LOOP WITH HI-LO MONITOR SWITCHES AND ALARMS.

EXTERNAL DISPLACEMENT LEVEL TRANSMITTER & PANEL MOUNTED RECORDING CONTROLLER ACTUATING SET POINT OF FLOW RECORDING CONTROL LOOP WITH REMOTE ACTUATED SOLENOID VALVE.

NOTES

1. THESE EXAMPLES ARE PRESENTED TO ILLUSTRATE THE USE OF THE BASIC SYMBOLS SHOWN ON STANDARD INDIVIDUAL DRAWINGS.
2. NUMBERS SHOWN ARE TYPICAL ONLY.
3. TYPICAL ILLUSTRATIONS ARE BASED ON ISA STANDARD S5.1-1967.
4. FOR IDENTIFICATION OF P&I SYMBOLS REFER TO APPENDIX "A" TO THIS REPORT.

TYPICAL ILLUSTRATIONS OF P&I DIAGRAM AND INSTRUMENT DRAWING LOOPS

11/1/72 Fig. 7.7-18

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CHAPTER 8

ELECTRICAL POWER

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CHAPTER 8

ELECTRIC POWER

8.1 INTRODUCTION

8.1.1 UTILITY GRID DESCRIPTION

To be provided in owner's application.

8.1.2 ONSITE POWER SUPPLY

The onsite ac electrical system consists of two preferred power supply circuits connected to two shore circuits through station auxiliary transformers. Each station auxiliary transformer has two secondary windings which in turn feed four sections of 6900V switchgear. Each of the preferred circuits also supplies the four 4160V safety related buses through 6900/4160V station auxiliary transformers.

The ac standby power supply consists of four diesel generators which are connected to feed the four 4160V engineered safeguards buses and through them feed other safety related power centers, motor control centers, panels, etc.

The Direct Current System consists of five station batteries along with associated chargers, distribution panels and equipment with one battery and bus for turbine auxiliaries and four other redundant units which supply power to safety related functions.

8.1.3 SAFETY FEATURES

The identification of safety loads, i.e., the systems and devices that require electrical power to perform their safety functions, are included in Table 8.1-1 of this section. This table includes the safety load, function performed, and type of power required (ac or dc). Reference to Chapter 15 is made for additional requirements.

TABLE 8.1-1

SAFETY LOADS AND FUNCTIONS

SAFETY LOADS	FUNCTION	POWER
Component cooling water pumps	Provide cooling water to NSSS equipment during loss of offsite power (LOOP)	ac
Residual heat removal pumps	A. Remove reactor heat during cold shutdown B. Provide emergency core cooling during LOCA (Injection phase only)	ac
Centrifugal charging pumps	Provide make-up water during loss of offsite power (LOOP)	ac
Essential service water pumps	Provide cooling water to NSSS during LOCA or (LOOP)	ac
Safety injection pumps	Provide emergency core cooling during LOCA	ac
Essential raw water pumps	Provide cooling water during LOCA or LOOP	ac
Containment spray pumps	Provide cooling spray in containment during LOCA	ac
Boric acid transfer equipment	Provide boration during LOOP	ac
Lower compartment fan coolers	For cooling containment during LOOP	ac
Containment air return fans	For recirculating containment air during a LOCA	ac
Spent fuel pit cooling pump	Cool spent fuel pit	ac
Containment hydrogen recombiner	Maintain a safe level of hydrogen in containment during LOCA	ac
Emergency air conditioning (control rm)	Maintain a safe air operating temperature	ac
Fire pumps	Provide water to extinguish fires	ac

TABLE 8.1-1 (CONT)

SAFETY LOADS	FUNCTIONS	POWER
Motor control centers	Provide power for M.O.V.'s small motors, fans, heaters, and small pumps associated with safety related equipment	ac
Annulus ventilation system	Minimize offsite dose during LOCA	ac
Diesel auxiliaries	Provide power for diesel support systems	ac
Pressurizer heaters	Provide reactor coolant pressure control during LOOP	ac
Emergency air compressors	Provide air during LOOP/LOCA	ac
Containment hydrogen purge system	Control post-LOCA hydrogen concentration	ac
Solid state/Logic protection system	A. Initiates ESF signals B. Prevents reactor from operating in unsafe conditions. (Reactor Trip)	ac
Instrument power supply inverter	Supplies power to the vital instrument buses	dc
Auxiliary feedwater pumps	Provide feedwater to steam generator during LOCA or LOOP	ac

8.1.4 DESIGN CRITERIA

The design bases, criteria, safety guides, standards, and other documents that are implemented in the design of this unit are:

1. Industry Manufacturing Standards.

   a. ANSI C37 Switchgear

   b. ANSI C50 Rotating Electrical Machinery

   c. ANSI C57 Transformers, Regulators, and Reactors

   d. IPCEA P-46-426 Power Cable Ampacities

   e. NEMA SG3-1965 Low Voltage Power Circuit Breakers

   f. NEMA S64-1968 AC High Voltage Circuit Breakers

   g. NEMA SG5-1967 Power Switching Assemblies

   h. NEMA SG5-1966 Power Switching Equipment

   i. NEMA R12-1971 Battery Chargers - General Purpose & Communications

   j. NEMA IB-1-1971 Defintiions for Lead Acid Storage Batteries

k. NEMA TRI-1968 Transformers, Regulators and Reactors

l. NEMA TR-P3-1970 Guide for Preparation of Specifications for Large Power Transformers

m. NEMA MGI-1967 Motors and Generators

2. The power supply for the reactor protection system and the safety systems will be in accord with Criteria 17 and 18 of the AEC General Design Criteria for Nuclear Power Plant Construction Permits, 10 CFR 50, Appendix A, published in Federal Register, February 1971.

3. Additional design criteria are as follows:

   a. IEEE 279-1971 - Criteria for Nuclear Power Plant Protection Systems.

   b. IEEE 288-1969 - Guide for Induction Motor Protection.

   c. IEEE 308-1971 - Criteria for Class 1E Electrical Systems for Nuclear Power Generating Stations.

   d. IEEE 317-1971 - Electrical Penetration Assembly in Containment Structures for Nuclear Fueled Power Generating Stations.

e. IEEE 323-1971 - General Guide for Qualifying Class 1 Electrical Equipment for Nuclear Power Generating Stations.

f. IEEE 334-1971 - Trial Use Guide for Type Tests of Continuous-Duty Class 1 Motors Installed Inside the Containment of Nuclear Power Generating Stations.

g. IEEE 336-1971 - Standard Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During The Construction of Nuclear Power Generating Stations.

h. IEEE 344-1971 - Guide for Seismic Qualifications of Class 1 Electrical Equipment for Nuclear Power Generating Stations.

i. IEEE 387-1972 - Trial Use Standard Criteria for Diesel Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations.

j. IEEE 379-1972 - Trial Use Guide for the Application of the Single Failure Criterion to Nuclear Power Generating Station Protection Systems.

4. The following documents will be considered:

a. AEC Safety Guide 6 - Independence between Redundant Standby (Onsite) Power Sources and between their Distribution Systems.

b. AEC Safety Guide 9 - Selection of Diesel Generator Set Capacity for Standby Power Supplies*

c. AEC Safety Guide 22 - Periodic Testing of Protection System Actuation Function.

d. AEC Safety Guide 30 - Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment.

e. AEC Safety Guide 32 - Use of IEEE STD 308-1971, "Criteria for Class 1E Electric Systems for Nuclear Power Generating Stations".

*The diesel-electric generating system fully meets the intent of Safety Guide 9 except that calculation of estimated electric loads given in Table 8.3-1 and 8.3-2 of this report is based on a motor efficiency of 92%. This efficiency is based on the average efficiency obtained on similar motors applied on other installations.

8.2 OFFSITE POWER SYSTEM

8.2.1 DESCRIPTION

Two independent offsite power sources with capacity to supply all of the protection and safety systems will be provided. Each offsite power source will be connected to a full capacity station power transformer, thereby providing two immediate access circuits connecting offsite power source engineered safeguards buses.

Details of the plant-to-shore transmission network will be provided in the owner's applciation. As an example, however, buried high voltage cable circuits could be used to provide the connection from the owner's onshore switching station to the Floating Nuclear Plant. The underground cable would be buried independently for their entire length to prevent simultaneous damage to both circuits from dig-ins or anchors. An Edison Electric Institute survey of forced outages of underground transmissions circuits, based on a 5 year average for 1,000 miles of cable, reported an outage rate of 0.00068 outages per mile per year. This average annual rate is about 1/5 the outage rate for overhead line and demonstrate the reliability of underground power transmission.

8.3 ONSITE POWER SYSTEM

8.3.1 AC POWER SYSTEM

8.3.1.1 Description

8.3.1.1.1 Normal Source System

The normal source system consists of two full sized station auxiliary transformers which are connected to the offsite power sources. Each transformer is connected to one of the two circuits and the associated switchgear is so arranged that either transformer can serve one half or all of the plant normal and safety systems.

The engineered safeguards buses are normally supplied through station auxiliary transformers (number 3 and 4), each of which have the capacity, using the ANSI Guide for Short Term Loading, to supply the total safety system loads.

The one line diagram (Figure 8.3-1) illustrates the connections to the normal offsite power sources, onsite standby power sources, and identifies the plant loads to be supplied from each bus. The basis for sizing of all supply equipment is the nameplate rating of the listed motors and other loads. Figures 8.3-3 through -8 illustrate the separation of the systems in the plant and shows physical location of the various electrical system components.

Automatic transfer of the engineered safeguards buses (4.16 kV buses) between the two preferred (offsite) sources on loss of voltage will be furnished. If for any reason neither of these sources is available or the transfer is not completed the buses will be automatically connected to their respective onsite ac standby power sources. Each 4.16 kV bus is an independent redundant system and automatic transfer between ac standby sources will not be furnished.

The safety related buses and equipment served therefrom are divided into four separate groups. Sources and load groups are arranged such that one source, bus, or group can be lost and the system will still have the required capacity to shut the plant down safely. All equipment associated with each safety load group is identified in such a way as to allow easy determination of the redundant load group.

8.3.1.1.2 Standby Power System

The standby power sources consist of four diesel generators, sized, arranged, and connected to supply redundantly the four engineered safeguards buses. Reliability is assured by the use of independent controls and control power sources. Each unit will be housed in a separate seismic Category I Compartment.

Manual and automatic controls for each diesel generator will be located in the vicinity of the engine or switchgear with duplication of such instrumentation in the main Control Room area as required to inform the operator of the load being carried by each engine.

Cooling water for each diesel generator is supplied from the essential service water system described in paragraph 9.2.2.1.

The diesel protective systems (for example, loss of lubricating oil pressure) are set to trip the engine in order to protect the capability of the engine to be returned quickly to service after the trouble is corrected. The various abnormalities are annunciated locally and are repeated as a master single alarm in the Control Room.

Sufficient onsite fuel storage is provided to run the engines at rated load for an extended period. System is described in 9.5.4.

Each diesel generator will be rated for continuous operation and, in addition, will have capability to operate at a higher loading for shorter periods out of every twenty-four hours of operation with increased maintenance. The engine continuous rating is equal to or greater than the expected loads as detailed in Tables 8.3-1 and 8.3-2 and will be 2100 kW each.

8.3.1.1.3 Instrument Power Supply

The four instrument power supply panels will be supplied in accordance with Figure 8.3-9 from four inverters and will be designed as redundant systems. The inverters design will use a non-interrupted transfer. The inverters are normally equipped from the 480 V engineered safeguards control center. In the event voltage is lost on the 480 V source, the inverter transfers to the dc source, the inverter transfers to the dc source without interrupting the ac output. A 120 V maintenance source is available to the instrument bus panels so the inverters can be removed for service if necessary.

8.3.1.1.4 Diesel Generator Starting Air System

This system is designed to provide two fast starts for each diesel generator. Piping, valves, fittings, and other components from the check valve located in the discharge line of each compressor to the flexible connection on each diesel generator are designed to ANS safety class 3 requirements. The rest of the system is designed to ANSI standards.

Each diesel generator has an independent air starting system located in the respective diesel generator room. Each system, as shown on Figure 8.3-2, consists of a starting air tank, an air compressor, and two air motors on the diesel gernerator. The air compressor operates intermittently to maintain the desired pressure in the starting air tank. A pressure switch of the tank starts the compressor on low tank pressure and stops the compressor when a preset higher pressure is reached. An alarm annunciates LO-LO tank pressure. The starting air tank supplies air at the desired pressure to the air starting motors.

Each diesel generator has two air starting motors with independent supply lines from the air tank. These two air motors start the engine simultaneously; a failure of either air motor will not prevent starting the engine. In the event of component failure rendering an air starting system and associated diesel generator inoperable, the three remaining diesel generators will supply power for the minimum required safeguards.

8.3.1.2 Analysis

A failure mode analysis of the safety features auxiliary ac power system verifies that the system satisfies the AEC General Design Critieria for single failure and related safety guides. Table 8.3-3 shows the analysis of each component of the system and the effects of a failure.

Compliance with General Design Criterion 17 is assured by the use of two connections to shore (utility network) which are both normally instantly available without primary switching.

TABLE 8.3-1

Emergency Electrical Loading Requirements

For Loss-of-Coolant Accident

Component	No. Installed Per Unit	HP Each Nameplate Rating	Auto. Sequence Start	Delay time (Seconds) After Safegd Signal Is Actuated
High Head Safety Injection Pumps	2	600	yes	10
Safety Injection Pumps	2	400	yes	15
Residual Heat Removal Pumps	2	400	yes	20
Containment Spray Pumps	4	400	yes	25
Essential Service Water Pumps	4	60	yes	30
Auxiliary Feedwater Pumps (Motor)	4	300	yes	40
Motor Operated Valve Group (MCC)	4	40	yes	10
Annulus Filtration Fans	2	20	yes	20
Annulus Filtration Heaters	2	40KW	yes	20
Emergency Relocation Air Cond. Sys.	2	30	yes	60
Containment Air Return Fans	2	40	yes	25
Control Room Air Cond. Units	2	100	yes	50
Control Room Supply Fans	2	15	yes	50
Control Room Return Fans	2	1 1/2	yes	50
Control Room Aux. Filter Fans	2	7 1/2	yes	50
Diesel Compartments Supply Fans	4	15	yes	10
Essential Raw Water Pumps	4	250	yes	35
Diesel Compartments Exhaust Fans	4	15	yes	10

TABLE 8.3-1 (CONT.)

Emergency Electrical Loading Requirements

For Loss-of-Coolant Accident

Component	No. Installed Per Unit	HP Each Nameplate Rating	Auto. Sequence Start	Delay time (Seconds) After Safegd. Signal Is Actuated
Diesel Oil Transfer Pumps	4	1	yes	10
Safeguards Equip. Compartments Unit Coolers	8	4-15 4-20	yes yes	50 50
Emergency Lighting Panel Board	4	5KVA	yes	Immediately
Battery Chargers	4	40A	yes	10
Emergency lighting	4	10KW	yes	40
Emergency Air Compressors	2	20	yes	40
Battery Rooms Supply Fans	4	3	yes	60
Battery Rooms Exhaust Fans	4	2	yes	60

The following are the projected per train loads based on the Loss-of-Coolant Accident requirements.

Train 1 = 2191

Train 2 = 2253

Train 3 = 2153

Train 4 = 1631

TABLE 8.3-2

Emergency Electrical Loading Requirements

For Loss-of-Offsite Power

Component	No. Installed Per Unit	HP Each Nameplate Rating	Auto. Sequence Start	Delay Time (Seconds) After Blackout
Essential Service Water Pumps	4	60	yes	30
Essential Raw Water Pumps	4	250	yes	35
Auxiliary Feedwater Pumps (Motor)	4	300	yes	40
Motor Operated Valve Groups (MCC)	4	40	yes	10
Control Room Air Cond. Units	2	100	yes	50
Control Room Supply Fans	2	15	yes	50
Control Room Return Fans	2	1 1/2	yes	50
Control Room Aux. Filter Fan	2	7 1/2	yes	50
Diesel Compartments Supply Fans	4	15	yes	10
Diesel Compartments Exhaust Fans	4	15	yes	10
Diesel Oil Transfer Pumps	4	1	yes	10
Safeguards Equip. Compartments Unit Coolers	8	4-15 4-20	yes yes	50 50
Emergency Lighting Panel Board	4	5KVA	yes	Immediately
Battery Chargers	4	40A	yes	10
Emergency Lighting	4	10KW	yes	10
Centrifugal Charging Pumps	2	400	yes	10
Boric Acid Transfer Pumps	2	10	yes	30
Boron Injection Tank Heaters	2	7kw	yes	60

TABLE 8.3-2 (CONT)

Emergency Electrical Loading Requirements

For Loss-of-Offsite Power

Component	No. Installed Per Unit	HP Each Nameplate Rating	Auto. Sequence Start	Delay Time (Seconds) After Blackout
Boron Injection Tank Recirc. Pumps	2	3	yes	60
Associated Heat Tracing	2	40A	yes	60
Emergency Air Compressors	2	20	yes	40
Containment Lower Compartment Fan Coolers	4	175	yes	60
Control Rod Drive Mech. Exhaust Fans	4	40	yes	50
Process Rack Room Supply Fans	2	20	yes	50
Process Rack Room Exhaust Fans	2	3	yes	50
Auxiliary Raw Water Pumps	2	600	yes	35
Component Cooling Water Pumps	2	350	yes	30
Reactor Vessel Support Cooling Fans	2	20	yes	50
Hot Shutdown Room Air Cond. Units	2	2	yes	60
Charging Pump Rooms Unit Coolers	2	5	yes	50
Battery Rooms Supply Fans	4	3	yes	60
Battery Rooms Exhaust Fans	4	2	yes	60

The following are the projected per train loads based on the Loss-of-Offsite Power Loading Requirements.

Train 1 = 2001 HP

Train 2 = 1635 HP

Train 3 = 2510 HP

Train 4 = 976 HP

TABLE 8.3-3

FAILURE MODE AND EFFECT ANALYSIS FOR AUXILIARY AC POWER SYSTEMS

(Reference figure 8.3-1 for Item Location)

ITEM	DESCRIPTION	FUNCTION	FAILURE	CAUSE OF FAILURE	EFFECTS OF FAILURE	HOW FAILURE IS DETECTED	EFFECTS ON SYSTEM
1	Offsite Power Source #1	Supply power to station Transformer #1	Fails to supply power to Station Transformer #1	Failure of shore-to-plant cable, onshore switchyard circuit breaker or SF_6 Bus	Loss of electrical power to Station Transformer #1	Loss of voltage, annunciate in control room	None, load is transferred to Station Transformer #2.
1a	Manual Isolator switch, SF_6 substation	Disconnect shore-to-plant cable from SF_6 substation	Fails Open	Mechanical Failure	Same as 1	Same as 1	Same as 1
1b	Demagnetizing current switch to Station Transformer #2	Disconnect Station Transformer #1	Fails Open	Same as 1a	Same as 1	Same as 1	Same as 1
2	Offsite Power Source #2	Supply power to Station Transformer #2	Fails to supply power to Station Transformer #2	Same as 1	Loss of electrical power to Station Transformer #2	Same as 1	None, load is transferred to Station Power Transformer #1
2a	Manual isolator switch, SF_6 substation	Same as 1a	Same as 1a	Same as 1a	Same as 2	Same as 1	Same as 2
2b	Demagnetizing current switch to Station Transformer #2	Disconnect Station Transformer #2	Same as 1a	Same as 1a	Same as 2	Same as 1	Same as 2
3	Station Transformer #1	Supply power to station and safeguard loads	Fails to supply power	Internal fault, bushing failure, loss of insulating oil	Loss of power to one half of Station and safeguard loads	Same as 1	None, load is transferred to Station Transformer #2.
4	Station Transformer #2	Same as 3	Same as 3	Same as 3	Same as 3	Same as 1	None, load is transferred to Station Transformer #1
5	6.9kV Bus Duct	Same as 3	Short Circuit Open circuit	Mechanical Damage, Fire, or Electrical Failure.	Same as 3	Same as 1	Same as 3

FAILURE MODE AND EFFECT ANALYSIS FOR AUXILIARY AC POWER SYSTEMS

(Reference Figure 8.3-1 for Item Location)

ITEM	DESCRIPTION	FUNCTION	FAILURE	CAUSE OF FAILURE	EFFECTS OF FAILURE	HOW FAILURE IS DETECTED	EFFECTS ON SYSTEM
5a	6.9kV Circuit Breaker	Protect Station Transformer #3	Fails to open Fails to close	Mechanical failure, stuck contacts, relay failure, control power failure or electrical failure	Fails to open: Loss of one power source to safeguards buses. Fails to close: Loss of power to two safeguard buses.	Loss of voltage, annunciate in control room	None: Load is transferred to Station Transformer #4.
5b	6.9kV Bus Duct	Supply power to Station Transformer #3	Short Circuit Open Circuit	Mechanical Damage, Fire, or Electrical Failure.	Loss of power to two safeguards buses	Same as 5a	Same as 5a
6	6.9kV Bus Duct	Supply power to station and safeguard loads.	Same as 5b	Same as 5 b	Loss of power to one half of Station and safeguard loads	Same as 5 a	None, load is transferred to Station Transformer #1
6a	6.9kV Circuit Breaker	Protect Station Transformer #4	Same as 5a	Same as 5a	Same as 5a	Same as 5a	None: Load is transferred to Station Transformer #3.
6b	6.9kV Bus Duct	Supply power to Station Transformer #4	Short Circuit Open Circuit	Mechanical Damage, Fire, or Electrical Failure.	Loss of Power to two safeguards buses.	Loss of Voltage Annunciate in control room	None: Load is transferred to Station Transformer #3.
7	Station Transformer #3	Supply power to safeguards buses.	Fails to deliver power	Internal fault, bushing failure, loss of insulating oil	Loss of power to two safeguards buses.	Same as 6b	None: Load is transferred to Transformer #4.
8	Station Transformer #4	Same as 7	Same as 7	Same as 7	Same as 7	Same as 6b	Same as 6b
9	4.16kV Bus Duct	Supply power to safeguards buses.	Same as 6b	Same as 6b	Same as 7	Same as 6b	Same as 7
10	4.16kV Bus Duct	Same as 9	Same as 6b	Same as 6b	Same as 7	Same as 6b	Same as 6b
11, (18, 25,32, 12, 19 26, 33	4.16kV Main Circuit Breakers	Protect 4kV Buses No. E1, E2, E3, E4	Fails to open, Fails to close	Mechanical failure, stuck contacts, relay failure, control power failure, or electrical failure	Fails to open; Bus is lost Fails to close; Loss of power to 4kV bus	Annunciation, periodic tests for operational readiness	None: Redundant buses or transfer to alternate source.

TABLE 8.3-3 Cont'd.

FAILURE MODE AND EFFECT ANALYSIS FOR AUXILIARY AC POWER SYSTEMS

(Reference Figure 8.3-1 for Item Location)

ITEM	DESCRIPTION	FUNCTION	FAILURE	CAUSE OF FAILURE	EFFECTS OF FAILURE	HOW FAILURE IS DETECTED	EFFECTS ON SYSTEM
13 (20, 27,34)	4.16kV Circuit Breaker	Supply Breaker for diesel Gen #1(2,3&4)	Fails to open Fails to close	Mechanical failure, stuck contacts, relay failure, control power failure or electrical failure	Fails to close: failure to provide Bus with power. Fails to open: loss of bus	Loss of voltage, annunciate in control room	None: redundant engineered safeguards equipment provided.
13a (20a, 27a, 34a)	4.16kV Cable	Connect Diesel to 4.16kV buses	Short Circuit Open Circuit	Mechanical damage, fire, or electrical failure.	Fails to provide power to 4.16kV buses.	Same as 13	Same as 13
13b (20b 27b 34b)	Diesel Generator #1 (2,3, & 4)	Safeguard power for buses E1 (E2 E3, E4)	Fails to deliver power to 4.16kV Bus	Does not start, generator failure	Fails to provide 4.16kV Bus with power	Same as 13	Same as 13
14 (21,28 35)	4.16kV Circuit Breaker	Protect Station Service Transformer E1(E2,E3,E4)	Same as 13	Same as 13	Fails to open; backup breaker opens. Fails to close; loss of power to Station Service Transformer.	Same as 13	Same as 13
14a (21a 28a 35a)	Cable	Connect Station Service Transformer to circuit breaker	Same as 13a	Same as 13a	Fails to provide power to Station Service Transformer	Same as 13	Same as 13
15 (22, 29,36)	Station Service Transformers #E1 (E2, E3, & E4)	Supply power to 480V Bus E1A (E2A, E3A, E4A)	Fails to provide power to 480V Bus	Mechanical damage, short circuit, overheating	Loss of power to 480V Bus	Same as 13	Same as 13
15a (22a, 29a, 36a)	Transformer Throat Connection	Connects transformer to bus	Fails to supply power	Mechanical damage, fire, or electrical failure.	Same as 15	Same as 13	Same as 13

FAILURE MODE AND EFFECT ANLAYSIS FOR AUXILIARY AC POWER SYSTEMS

(Reference Figure 8.3-1 for Item Location)

ITEM	DESCRIPTION	FUNCTION	FAILURE	CAUSE OF FAILURE	EFFECTS OF FAILURE	HOW FAILURE IS DETECTED	EFFECTS ON SYSTEM
16 (23, 30,37	480V Switchgear	Distributes 480V power	Open circuit Short circuit	Mechanical failure, fire or electrical failure	Loss of 480V load	Loss of voltage annunciate in control room	None: Redundant Engineered Safeguards Equipment provided.
16a (23a 30a 37a)	480V Breaker	Protects MCCE1 (MCCE2 MCCE3, MCCE4)	Fails to open Fails to close	Mechanical failure, stuck contacts, relay failure, control power failure or electrical failure	Fails to open: Loss of 480V Bus. Fails to close: Loss of power of MCC.	Same as 16	Same as 16
16b (23b 30b, 37b)	Cable	600 volt cable between 480V switch-gear and MCCE1 (MCCE2, MCCE3, MCCE4)	Same as 16	Mechanical damage, fire, or electrical failure.	Loss of power to MCC	Same as 16	Same as 16
17 (24, 31,38)	MCCE1 (MCCE2, MCCE3, MCCE4)	Supplies 480V loads	Same as 16	Same as 16	Loss of MCC	Same as 16	Same as 16
39 (40, 41, 42)	4.16kV Switchgear	Distributes 4.16kV power	Same as 16	Same as 16	Loss of 4.16kV	Same as 16	Same as 16

8.3.1.2.1 Tests and Inspections

Safety related electric power systems are designed to allow testing of the operability and performance of the Class 1E components under design operational sequence to the fullest extent possible without requiring shutdown of the plant. Where such testing cannot be accomplished without shutdown the complete sequence test will be run during the periodic shutdown time.

The 4160 and 480 volt circuit breakers and associated equipment are tested while individual equipment is shut down. The circuit breakers may be placed in the TEST position and tested functionally. Breaker opening and closing may also be performed. Circuit breakers and contactors for redundant or duplicated circuits may be tested in service without interfering with the plant operation.

The dc system will be ungrounded and will have detectors to indicate when an accidental ground has occurred on any leg of the system. A ground in one leg of the system will not cause trip-out of equipment being served by that leg and the ground may be found and removed in an orderly fashion. If simultaneous grounds on two legs of one system occur, the circuit protective device will interrupt the legs involved and service to all other loads on that system will continue to be served without interruption. The batteries are under continuous automatic charging and are inspected and checked on a routine basis while the unit is in service. Grounds on the dc system, and battery charger failures, are annunciated in the Control Room.

During plant startup testing each diesel will be tested to verify the ability of the unit to perform to the design requirements. The diesel generator units or other units of the same design are given a sufficient number of cycles of tests to certify the adequacy of the units for the intended service. Tests will include starting tests to demonstrate the ability to attain and stabilize frequency and voltage within the rated limits and time, and load acceptance tests to demonstrate the ability to accept loads in the desired sequence and time duration. In addition, operational tests will be run to demonstrate the ability to carry the continuous rated load without exceeding the manufacturer's design limits. Load rejection tests will be run to demonstrate the ability to reject the maximum rated load without exceeding speeds or voltages that cause tripping, mechanical or harmful overstresses.

After being placed in service, the diesel generator unit can be tested periodically to demonstrate the continued ability of the unit to perform to design limits.

The periodic testing program will be defined in the Owner's Application.

8.3.1.2.2 Quality Assurance

All electrical equipment associated with the safety features is listed on table 3.2-2, section 3.2 and is subject to the requirements of the Quality Assurance Program defined in Chapter 17. Suppliers of 1E class equipment will have quality assurance programs that satisfy the requirements of 10 CFR 50, Appendix B and equipment will be inspected "in-shop" for compliance with the specifications. In addition certain operational tests "in-facility" and documentation of such procedures will assure pre-shipment (plant) system compliance and operation integrity equivalent to "on-site" testing.

8.3.1.2.3 Cable Design and Routing of Circuits

The design of cable, electrical penetrations, and circuit routing minimizes failure in electrical cable and penetration systems. Design criteria encompass the areas listed below:

1. Cable construction

2. Circuit protection

3. Sizing of power cable

4. Cable tray loading, construction, and wiring

5. Cable tray separation

6. Cable routing

7. Fire detection and alarms

8.3.1.2.3.1 Cable Construction

To meet stringent service and environmental requirements, all cable complies, where applicable, with the latest standards of the Insulated Power Cable Engineers Association (IPCEA), National Electrical Manufacturers Association (NEMA), Institute of Electrical and Electronic Engineers (IEEE), Instrument Society of America (ISA), and American Society for Testing Materials (ASTM).

All cable is flame-retardant. Power and control cable insulation will be ethylene-propylene rubber-base with an overall jacket of neoprene or hypalon or will be an insulation having properties equal to or better than the above insulation. Instrumentation cable varies with the type of signal conveyed but will meet the insulation properties of power and control cables.

The insulation and jackets described are thermally stable compounds which can perform their designated functions even when subjected to severe loss-of-coolant accident (LOCA) conditions where applicable. Power and control cables have been subjected to Underwriter's Laboratories (UL), NEMA, IPCEA, and other flame tests required to prove cable reliability.

Flame tests have been conducted on two horizontal cable trays, spaced vertically per the criteria for cable tray separation. (See paragraph 8.3.1.2.3.5.) With the lower row of cables subjected to over 900°F, fire was not propagated and the cables in the upper tray were unaffected.

The cable constructions were chosen to offer an optimum balance of electrical, physical, aging, and water absorption characteristics, and flame retardant and mechanical properties. The excellent test and performance record of this type cable was a factor in formulating the cable tray separation criteria.

Several cable manufacturers have conducted tests which demonstrate the insulations to be servicable after having been subjected to a radiation dose of 2.0×10^8 rads and pressure, temperature, and humidity conditions of 50 psig, 300°F and 100% respectively. Actual aging tests simulated a 40 year life by subjecting cable to temperatures in excess of 300°F for 14 days to substantiate that the cable was still serviceable.

6.9 kV system cable is single conductor, 8 kV insulation, strand and insulation shielded, 90°C, copper conductor, ungrounded.

4.16kv system cable is single conductor, 5 kV insulation, strand shielded, 90°C, copper conductor, ungrounded.

480 V system cable is single or three conductor, 600 V insulation, 90°C, copper conductor.

120 VAC, 125 VDC, and 250 VDC system cable is two conductor, 600 V insulation, 90°C, copper conductor.

Generally, control cable will be #16 AWG stranded copper, however, #12 and #14 will be used where required. Power cable will be a minimum #12 AWG.

8.3.1.2.3.2 Circuit Protection

Cables are protected against overloads and faults. All 6.9kV, 4.16kV and 480 V power circuits are breaker protected. All three phases will be simultaneously interrupted by circuit breakers.

All control power feeders are protected by circuit breakers and secondary control circuits are protected by fuses or breakers.

High-speed clearing of faults prevents damage to cables. The sizing of cables prevents the conductor temperature from exceeding the design value for the maximum possible fault current.

8.3.1.2.3.3 Sizing of Power Cables

Power cable ampacity and installation conform to the latest IPCEA requirements. Cables will be derated based on raceway fill to prevent overheating.

8.3.1.2.3.4 Cable Tray Loading, Construction, and Wiring

The percentage of cable tray fill, determined by total cross-sectional area of cables in tray, will be a maximum of 40% for power cables.

Control and instrumentation trays may be filled to the top of the side rails. The allowable load capacity (in pounds per linear foot) will be based on the latest NEMA and IEEE Standards.

Cable trays will be hot-dipped galvanized steel. Either ladder or solid bottom type will be used depending on the protection needed for the cables in the tray.

Cable trays will be supported according to NEMA Standards.

Cable trays, conduits, or other raceways carrying Class 1E circuits will meet seismic Category 1 requirements.

Where cable trays are installed directly above or beneath equipment at which cables terminate, the cables may be run from the tray to the equipment without support provided such cables are supported by other means at intervals not to exceed 42 inches.

Generally, cable will be run without splices. If necessary, splices will be made in a junction, except for cables penetrating the containment which will be spliced at the containment penetration without a junction box.

8.3.1.2.3.5 Cable Tray Separation

Separation criteria will be established based on the location of the trays within the plant so as to preclude any single credible event from rendering inoperative the redundant counterparts of any system. Special consideration will be given to potential hazards, including missiles, in the various areas. The routing of redundant safety related cables in hazardous areas is minimized. Barriers will be provided and maximum practical physical separation will be provided where needed.

In general, cable trays containing safety related power cables will be separated from other redundant power and control trays by a distance of no less than 36 inches vertically or 12 inches horizontally. Redundant control cable tray will be separated by no less than 18 inches vertically or 12 inches horizontally. Where these minimum distances cannot be met, suitable fireproof barriers will be installed, or protection will be furnished by conduits, tray covers, solid bottom trays, or other means.

Cables of redundant systems will be routed separately. Redundant, safety-related cables will be physically separated by such a distance so that damage from potential hazards will not prevent the plant safety systems from fulfilling their intended functions. Routing of one redundant safety-related cable above another redundant cable will be minimized.

8.3.1.2.3.6 Cable Routing

There will be three types of cable raceways:

1. Medium voltage power - 6900 V and 4160 V

2. Low voltage power - 480 VAC, 118 VAC, 125 VDC, and 250 VDC.

3. Instrumentation and Control.

These types will be separated from each other by being installed in different raceways.

An exception to this rule will be that #10 AWG and smaller low voltage power (Type 1) cables may be run in an instrumentation and control (Type 3) tray if a barrier separates the cables. Wherever possible, medium voltage power Type 1) trays will be the uppermost tray with lower voltage power and instrumentation and control trays being successively lower in elevation.

Cables will be run in the appropriate Type and Channel tray. If a cable has no channel classification, it may be run with any one channel. However, once a cable is run with a channel, it may not be run with any other channel.

The arrangement of components on control boards will be in a logical relationship so as to optimize safe operation. This normally dictates that redundant control devices and circuits be located in close proximity to each other on the same panel section. The wiring associated with these redundant circuits will be bundled and routed to maintain maximum air separation between redundant circuit bundles. Each redundant wire bundle will be indentified as to its particular saftety channel or train thereby. The mutally redundant bundles will be kept separate and will be terminated on separate terminal blocks. Mutually redundant cables will enter the enclosure through separate openings of not otherwise isolated.

Fire retardant wiring will be utilized throughout the control boards for both redundant and non-redundant circuits as an additional safety factor.

When it is necessary to preserve the water-tight integrity of a compartment, a multi-cable transit system or the equivalent will be used for cable entry and exit from the containment.

Openings in wall and floors for cable or conduit runs will be sealed with fire-resistant material.

Cable routing records (conduit and cable schedule) will be prepared to establish a permanent record of cable designations, cable types, origins, terminations, and routings. The coding of cables will allow easy identification of safety-related cables. All cables will be identified by a tag affixed at both ends bearing the code and cable designation of the cable.

8.3.1.2.4 Lightning Protection

The uncoated platform steel and building columns, trusses, girders, sheathing, etc., are considered to be a continuous distributed ground grid connecting all parts of the structure to earth through the basin saline water. By this continuous sheath all apparatus contained within the plant is considered to be shielded from the effects of lightning.

The containment shield building will be protected by suitably mounted grounded air terminals.

8.3.2 DC DISTRIBUTION SYSTEMS

8.3.2.1 Description

The 125 volt dc systems provide a source of reliable, uninterruptible, dc power for all normal and emergency control, instrumentation, emergency lighting and power loads. Five independent and separate dc systems are provided, four 125 vdc systems and one 250 vdc system, each with its own battery and battery charger. Only the 125 vdc systems are used for safety related equipment. These 125 vdc systems are separate and redundant so that failure in one system will not affect the other systems.

As shown in Figure 8.3-9, the system consists of 125 vdc distribution panels, 125 vdc battery chargers, and 125 vdc batteries. Each dc bus is supplied from a battery charger with the battery floating on the bus. Power for each 125 vdc distribution bus is supplied through the battery charger from a separate 480 V safety related motor control center, which is automatically powered from a standby diesel generator on loss of the preferred source.

Each charger supplies the normal dc loads and periodically supplies the equalizing charge. One spare battery charger is provided on site and in the unlikely event that a charger fails, it can be connected and placed in service. The basis for sizing each battery charger is to provide capacity to restore the battery to full charge after the battery has been discharged, while also carrying normal load.

To maintain separation in the essential 4160 and 480 Vac systems, separate dc systems are provided for each set of essential buses. Channel independence is maintained in the feeds to the safety related apparatus.

8.3.2.1.1 Station Batteries

The design basis for each of the 125 volt station batteries is to provide capacity to supply vital loads without charger support.

Each battery has adequate storage capacity to carry the following required loads without charger support for a period of two hours:

1. Instrument inverters
2. Engineered safeguards control
3. Emergency lights
4. Circuit breaker tripping and closing
5. Diesel generator control
6. DC solenoids for air-operated valves
7. Miscellaneous controls and alarms

The 125 volt batteries are mounted on racks, with non-combustible spacers between cells to prevent shifting. Each battery room has an independent ventilation system adequate to remove any toxic gases that may be generated.

The batteries are in separate areas in the auxiliary building near the essential dc buses they serve, thus maintaining isolation between the four systems.

8.3.2.2 Analysis

The dc system is so arranged that a single fault within the system will not prevent the reactor protection system and the engineered safeguards systems from performing their safety functions. A failure mode analysis of the safety features dc power system verifies that the system satisfies the single failure criterion. Table 8.3-4 show the analysis of each component of the system and the effect of a failure. The Testing, Quality Assurance, Cable Routing, Separation, Marking and Fire Detection are covered in paragraph 8.3.1.2 for both ac and dc systems.

TABLE 8.3-4

FAILURE MODE AND EFFECT ANALYSIS FOR AUXILIARY DC POWER SYSTEMS

(Reference Figure 8.3-9 for Item Location)

ITEM	DESCRIPTION	FUNCTION	FAILURE	CAUSE OF FAILURE	EFFECTS OF FAILURE	HOW FAILURE IS DETECTED	EFFECTS ON SYSTEM
E1-1	Battery charger (E1)	Supply dc power	Fails to provide power	Component failure, Loses source	Loss of charging capability Loss of dc bus E1 after 2 hours with continuous normal load	Periodic testing, Annunciation	None: Battery E1 supplies power for 2 hours ES equipment provided from busE2, E3, E4. Vital instruments still supplied from inverter.
E1-2	Feeder breaker	Protect bus E1	Fails to open Fails to close	Stuck contacts, Mechanical failure	Fails to open: Loss of bus E1. Fails to close: Loss of charging capability and loss of dc bus E1 after 2 hours with continuous normal load	Annunciation, Periodic Testing	Fails to close: None: Battery EI supplies power for 2 hours with continuous normal load. Fails to open: None: ES Equipment provided from buses E2, E3, and E4. Vital instruments still supplied from inverter.
E1-3	Battery fuse	Protect bus E1 Battery	Fails to open	Bus or cable fault	Fails to open: Loss of bus E1	Annunciation, Periodic Testing	None: ES Equipment provided by buses E2, E3, E4. Vital instruments still supplied from inverter.
E1-4	DC Bus E1	Distributes power	Fails to deliver power	Mechanical failure	Loss of one ES load group	Annunciation	None: ES Equipment provided by buses E2, E3, E4. Vital instruments still supplied from inverter.
E1-5	480V supply feeder	Power feeder from MCCE1 to inverter No. 5	Loses source	Cable fault	Loss of ac power to inverter No. 5	Annunciation, Periodic Testing	None: Inverter will be supplied from dc bus.
E1-6	DC Supply breaker	Protect Inverter No. 5	Fails to open Fails to close	Mechanical failure, Stuck contacts	Fails to Open: Loss of inverter No. 5 Fails to Close: None: Power supplied through regular ac source	Annunciation, Periodic Testing	Fails to open: None: Computer is lost. Not a safety related item. Fails to close: None: Regular supply from MCCE1. NOTE: ES= Engineered Safeguards

FAILURE MODE AND EFFECT ANALYSIS FOR AUXILIARY DC POWER SYSTEMS

(Reference Figure 8.3-9 for Item Location)

ITEM	DESCRIPTION	FUNCTION	FAILURE	CAUSE OF FAILURE	EFFECTS OF FAILURE	HOW FAILURE IS DETECTED	EFFECTS ON SYSTEM
E1-7	DC Supply breaker	Protect inverter No. E1	Same as E1-6	Same as E1-6	Fails to open: Loss of inverter No. E1 Fails to close: None, Power supplied through regular ac source	Same as E1-6	Fails to open: None, ES equipment provided from bus E2, E3, and E4 Fails to close: None, Power supply from MCCE1.
E1-8	Load breaker	Protect distribution panel E1	Same as E1-6	Same as E1-6	Fails to open: Loss of Distribution panel E1 Fails to close: Loss of power to distribution panel E1	Same as E1-6	None: ES equipment provided by bus E 2, E3, and E4. Vital instruments still supplied from inverter.
E1-9	Inverter No. 5	Supplies ac power to computer	Does not convert dc to ac	Component failure	Loss of power to computer	Annunciation, Periodic Testing	None: Computer is shutdown. Not a safety related item
E1-10	DC feeder	Supplies dc power to inverter No.E1	Loses source	Cable fault	Loss of dc power to inverter	Annunciation, Periodic Testing	None: Inverter is supplied through regular ac source.
E1-11	Distribution Panel E1	Distribution power	Fails to deliver power	Mechanical failure	Loss of one ES load group	Annunciation	None: ES Equipment provided by bus E2, E3, E4. Vital instruments still supplied from inverter.
E1-12	480V supply feeder	Power feeder from MCCE1 to inverter No. E1	Loses source	Cable fault	Loss of ac power to inverter	Annunciation, Periodic Testing	None: Inverter will be supplied from alternate dc source
E1-13	Inverter No. E1	Supplies ac Power	Does not convert dc to ac	Component failure	Loss of power to vital instrument bus	Periodic Testing, Annunciation	None: ES Equipment provided by bus E2, E3, E4 NOTE: ES= Engineered Safeguards

TABLE 8.3-4 Continued

FAILURE MODE AND EFFECT ANALYSIS FOR AUXILIARY DC POWER SYSTEMS

(Reference Figure 8.3-9 for Item Location)

ITEM	DESCRIPTION	FUNCTION	FAILURE	CAUSE OF FAILURE	EFFECTS OF FAILURE	HOW FAILURE IS DETECTED	EFFECTS ON SYSTEM
E1-14	Maintenance Tie breaker	Provides alternate supply for maintenance	Fails to close	Mechanical failure	Fails to open: Loss of vital instrument bus Fails to close: Loss of power to vital instrument bus	Periodic testing	Fails to Open: None, ES Equipment provided by bus E2, E3, E4. Fails to Close: None › Vital instrument bus supplied by inverter.
E1-15	Feeder breaker	Protects vital instrument bus	Fails to open Fails to close	Mechanical failure	Fails to open: Loss of vital instrument bus Fails to close: Loss of instrument bus	Periodic Testing	None: ES Equipment provided by bus E2, E3, E4.
E1-16	Vital instrument bus No. 1	Distributes Power	Fails to deliver power	Overload, short circuit	Loss of one ES load group	Annunciation	None: ES Equipment provided by bus E2, E3, E4

NOTE:

Engineered safeguards trains E2, E3, and E4 are identical to Train E1 in Design and Failure Mode and Effect Analysis with the following exceptions:

E1-5 - Computer Inverter Auxiliary Feeder

E1-6 - Computer Inverter Feeder Breaker

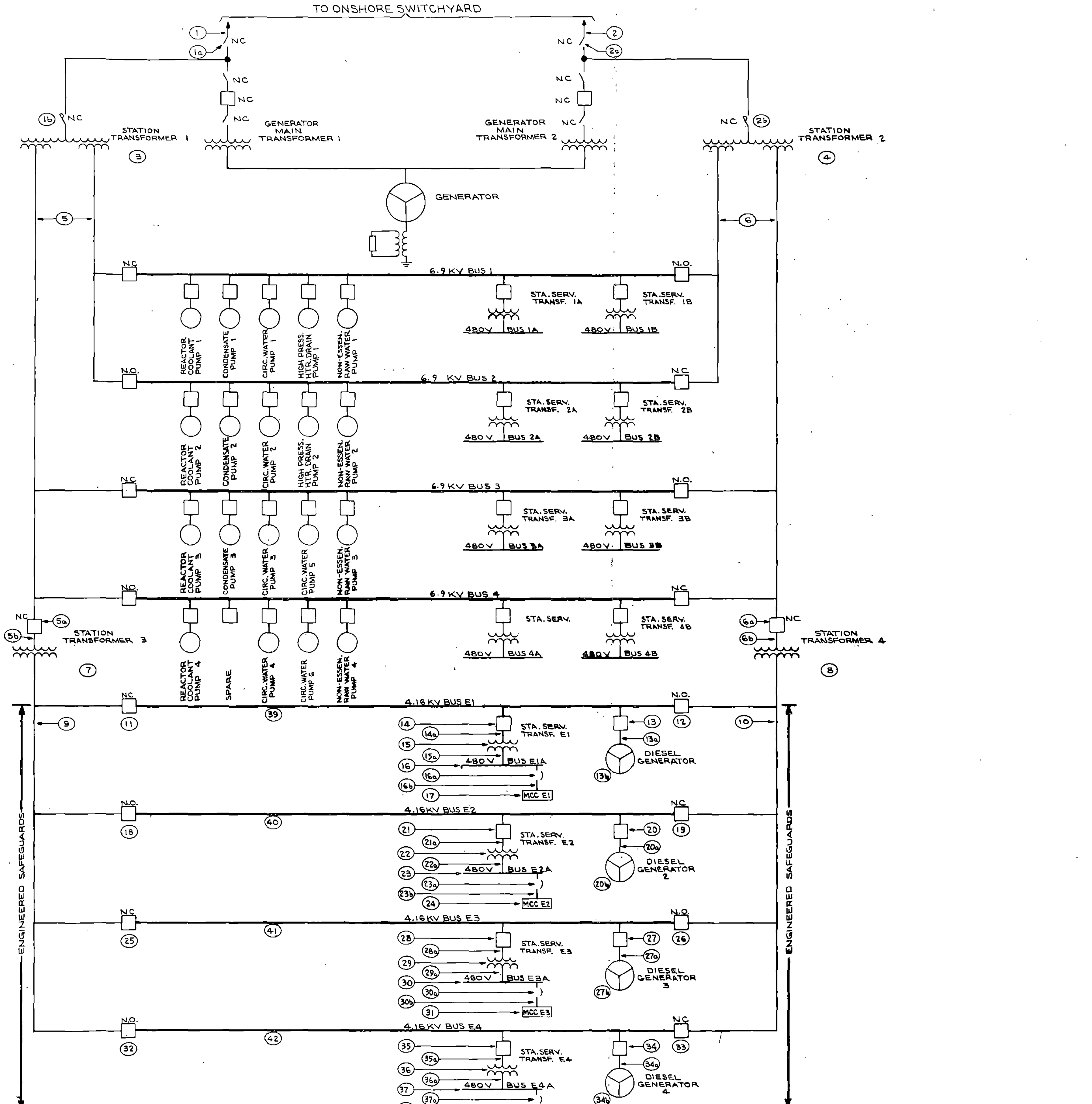
E1-9 - Computer Inverter

NOTE:
ES= Engineered Safeguards

EQUIPMENT LOADING

EQUIPMENT (4.16KV & 6.9KV)	6.9 KV BUS 1	6.9 KV BUS 2	6.9 KV BUS 3	6.9 KV BUS 4	ENGINEERED SAFEGUARDS 4.16 KV BUS E1	BUS E2	BUS E3	BUS E4
REACTOR COOLANT PUMP 1 THRU 4	7000HP	7000HP	7000HP	7000HP				
CONDENSATE PUMP 1 THRU 3	3000HP	3000HP	3000HP					
CIRCULATING WATER PUMP 1 THRU 6	2000 HP	2000HP	2-2000HP	2-2000HP				
NON-ESSENTIAL RAW WATER PUMP 1 THRU 4	700HP	700HP	700HP	700HP				
HIGH PRESSURE HEATER DRAIN PUMP 1 & 2	1250HP	1250HP						
COMPONENT COOLING WATER PUMP 1 THRU 3					350HP	350HP	350HP	
CENTRIFUGAL CHARGING PUMP 1 & 2						400HP	400HP	
FIRE PUMP 1 THRU 3					300HP	300HP	300HP	
AUXILIARY FEEDWATER PUMP 1 THRU 4					300 HP	300 HP	300 HP	300 HP
CONTAINMENT SPRAY PUMP 1 THRU 4					400 HP	400 HP	400 HP	400 HP
HIGH HEAD SAFETY INJECTION PUMP 1 & 2					600 HP		600 HP	
RESIDUAL HEAT REMOVAL PUMP 1 & 2					400HP	400HP		
SAFETY INJECTION PUMP 1 & 2						400HP		400 HP
AUXILIARY RAW WATER PUMP 1 THRU 3					600HP	600 HP	600 HP	

EQUIPMENT (480V)	480V BUS 1A	480V BUS 1B	480V BUS 2A	480V BUS 2B	480V BUS 3A	480V BUS 3B	480V BUS 4A	480V BUS 4B	ENGINEERED SAFEGUARDS 480V BUS E1A	480V BUS E2A	480V BUS E3A	480V BUS E4A
LOW PRESSURE HEATER DRAIN PUMP 1 & 2	350HP		350HP									
SCREEN WASH PUMP 1 & 2				125 HP	125HP							
SERVICE WATER PUMP 1 THRU 4	350HP			350HP		350HP		350HP				
BRINE RECYCLE PUMP 1A, 1B, 2A & 2B		200HP		200HP		200HP		200HP				
CONTROL BLDG. REFRIG. COMPRESSOR 1 & 2	93KW		93KW									
CONTROL ROD DRIVE M-G SET 1 & 2					150HP		150HP					
LIGHTING TRANSFORMER 1 & 4						225 KVA		225KVA				
LIGHTING TRANSFORMER 3			250KVA									
AUX. BLDG. MAIN SUPPLY FAN				150HP								
AUX. BLDG. MAIN EXHAUST FAN 1 & 2					250HP		250HP					
CHARGING PUMP	200HP											
OFF LOAD DERRICK (MAIN HOIST)		125HP										
CONTAINMENT LWR. COMPT. COOLING FANS 1 THRU 4		125HP		125 HP	125HP		125HP					
PRESSURIZER HEATER CONTROL GROUP		416KW										
PLANT VENTILATION SUPPLY FAN 1, 2 & 3				125HP		125HP		125HP				
MOTOR CONTROL CENTERS	300KVA	300KVA	300KVA	300KVA	300KVA	300KVA	300KVA	300KVA				
ADMIN. & SERVICE BLDG. COMPRESSOR 1 & 2			120 KW				120KW					
AUX. BLDG. BASEMENT & SAFEGUARDS EXHAUST 1 & 2			125 HP		125HP							
ESSENTIAL RAW WATER PUMP 1 THRU 4									250HP	250HP	250HP	250HP
MOTOR CONTROL CENTER E1, E2, E3 & E4									300 KVA	300KVA	300 KVA	300KVA
ANNULUS FILTRATION HEATER 1 & 2										40KW	40KW	
ESSENTIAL SERVICE WATER PUMP 1 THRU 4									60 HP	60 HP	60 HP	60 HP
CONTAINMENT AIR RETURN FAN 1 & 2										40HP		40HP
CONTAINMENT HYDROGEN RECOMBINER 1 & 2										75KW		75KW
CONTROL ROOM REFRIG. CONDENSER 1 & 2										100HP	100HP	
SPENT FUEL PIT PUMP										125HP		
CONTAINMENT LWR. COMPT. FAN COOLERS 1 THRU 4									175HP	175HP	175HP	175HP
PRESSURIZER HTR. BACKUP GROUP 1 THRU 4									346 KW	346 KW	346 KW	346 KW



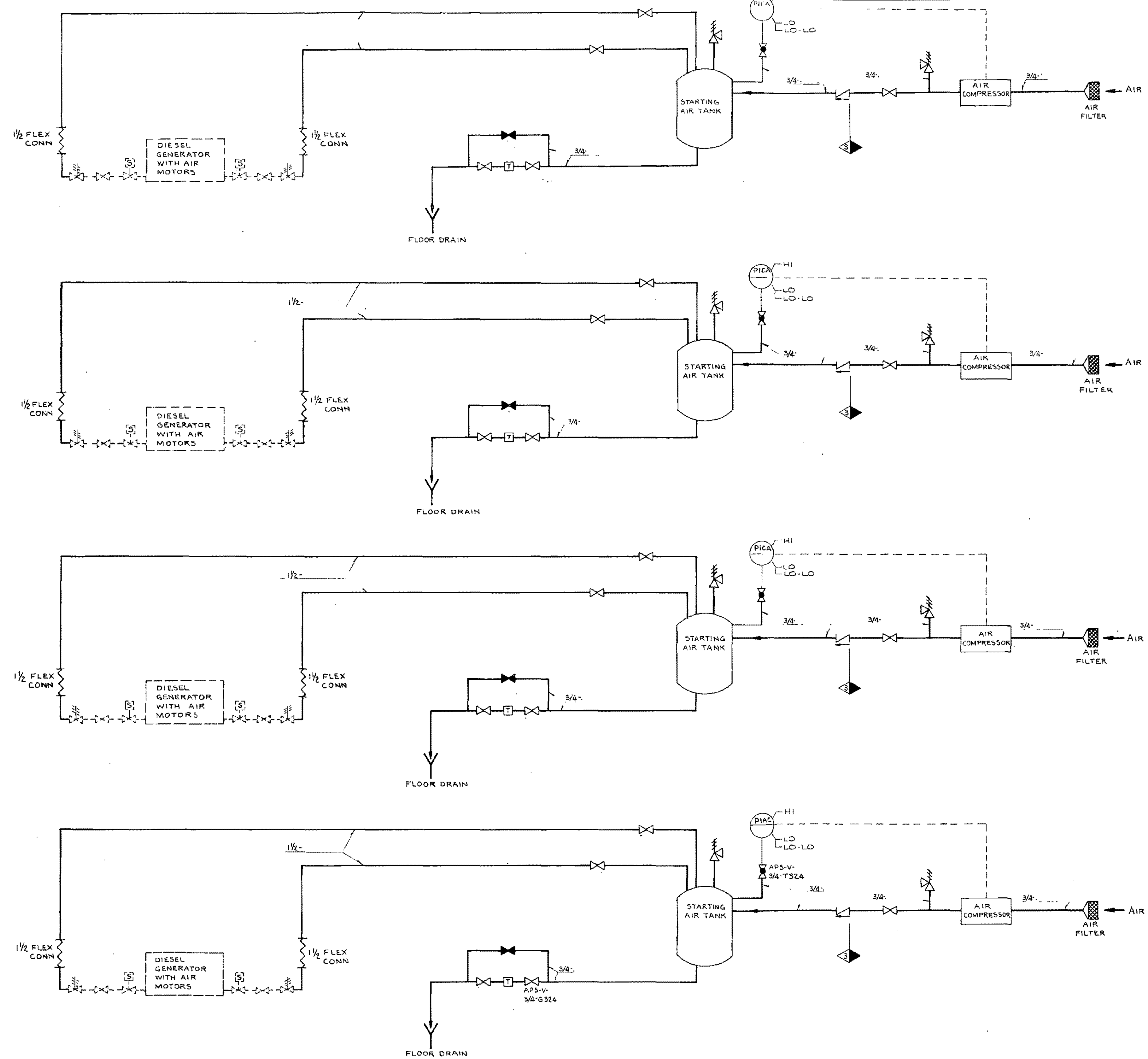
NOTE:
1. ◯ NUMBER TO BE USED WITH FAILURE MODE ANALYSIS.

OFFSHORE POWER SYSTEMS

Title: FLOATING NUCLEAR PLANT MAIN ONE LINE

1/1/73

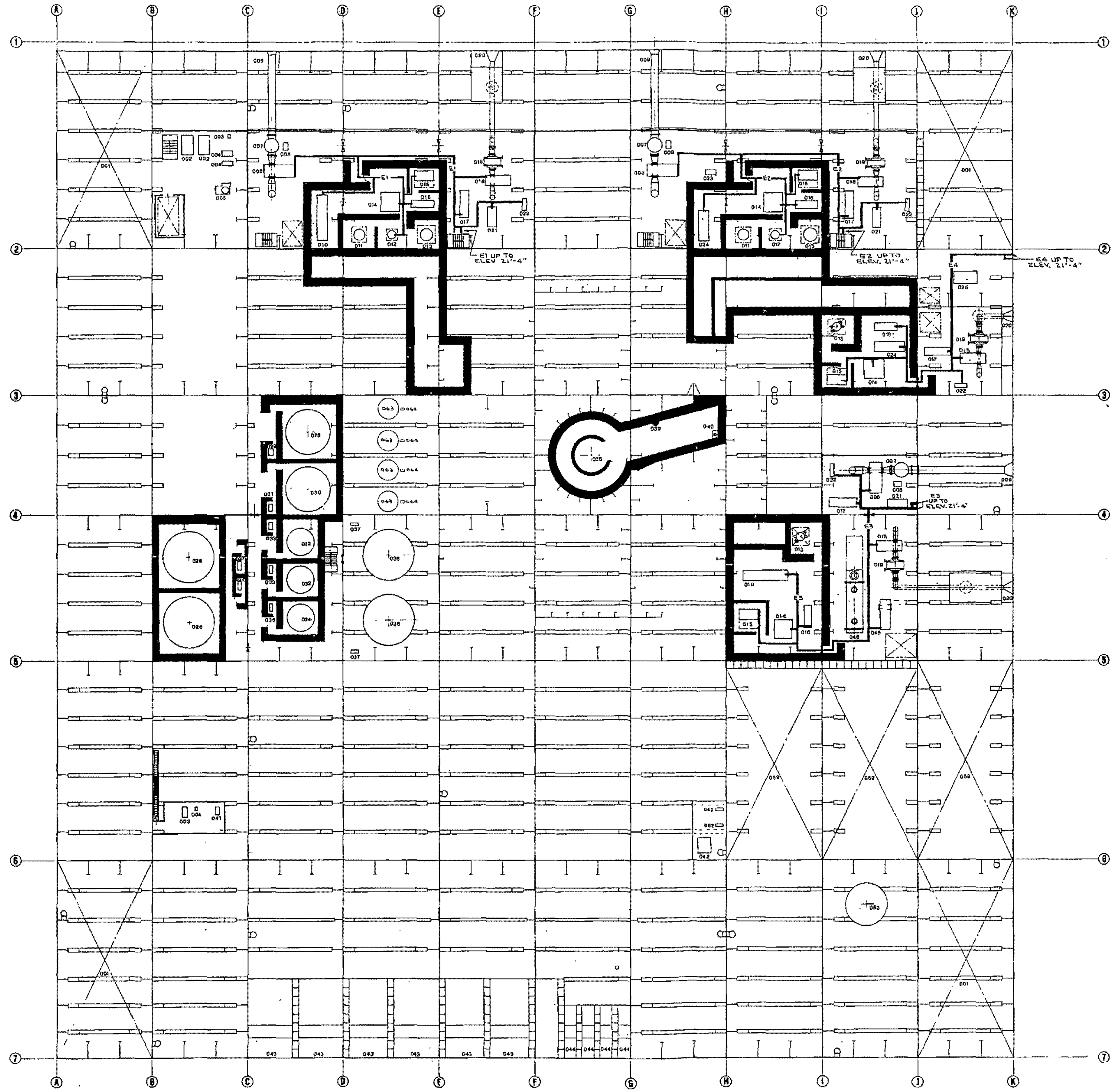
Figure: 8.3-1



OFFSHORE POWER SYSTEMS

Title: DIESEL GENERATOR AIR STARTING SYSTEM FLOW DIAGRAM

1/1/73 Figure: 8.3-2



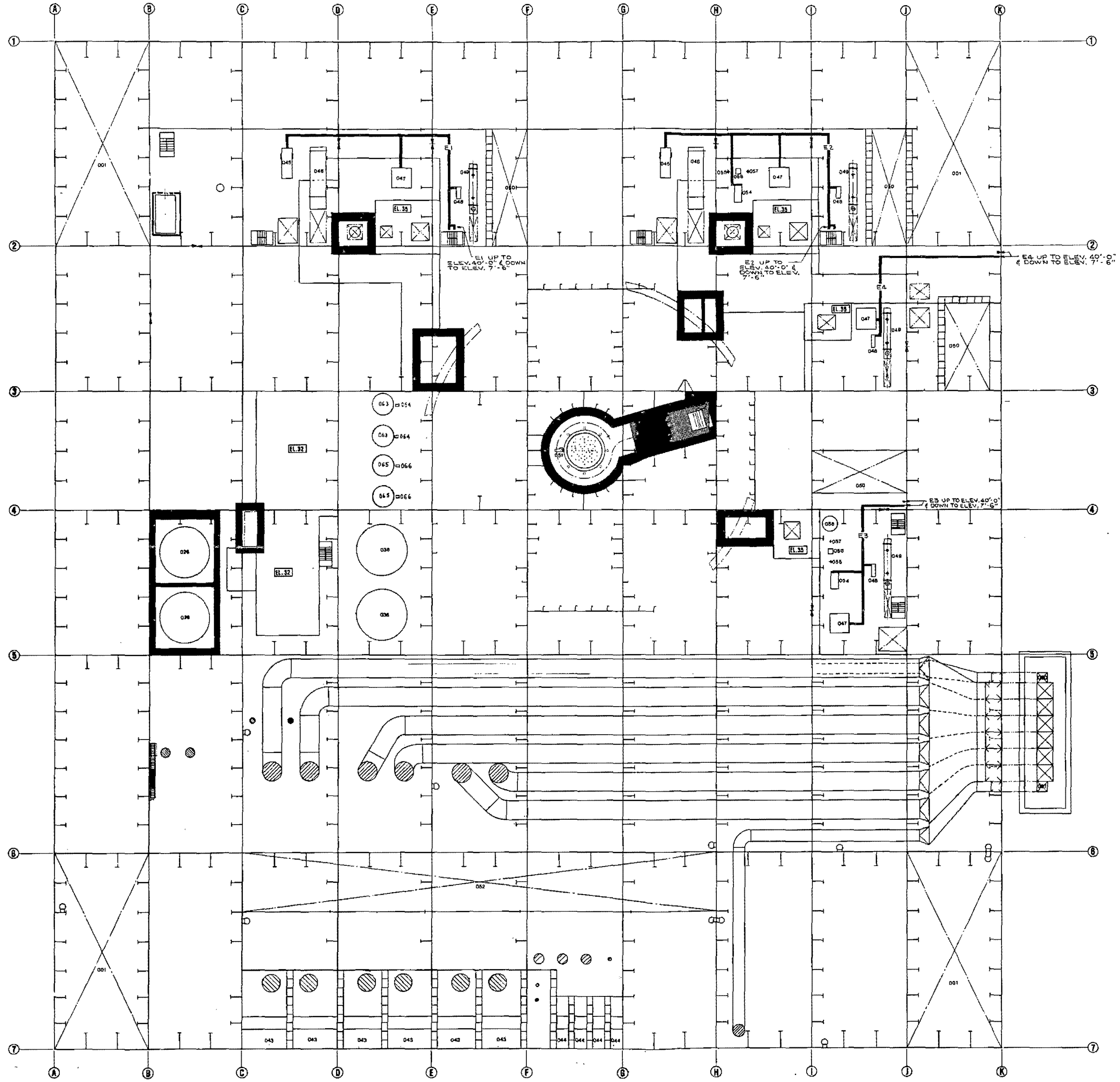
OFFSHORE POWER SYSTEMS

Title: FLOATING NUCLEAR PLANT

ELECTRICAL ARRANGEMENT PLAN

ELEVATION 7' - 6"

11/1/72 Figure 8.3-3

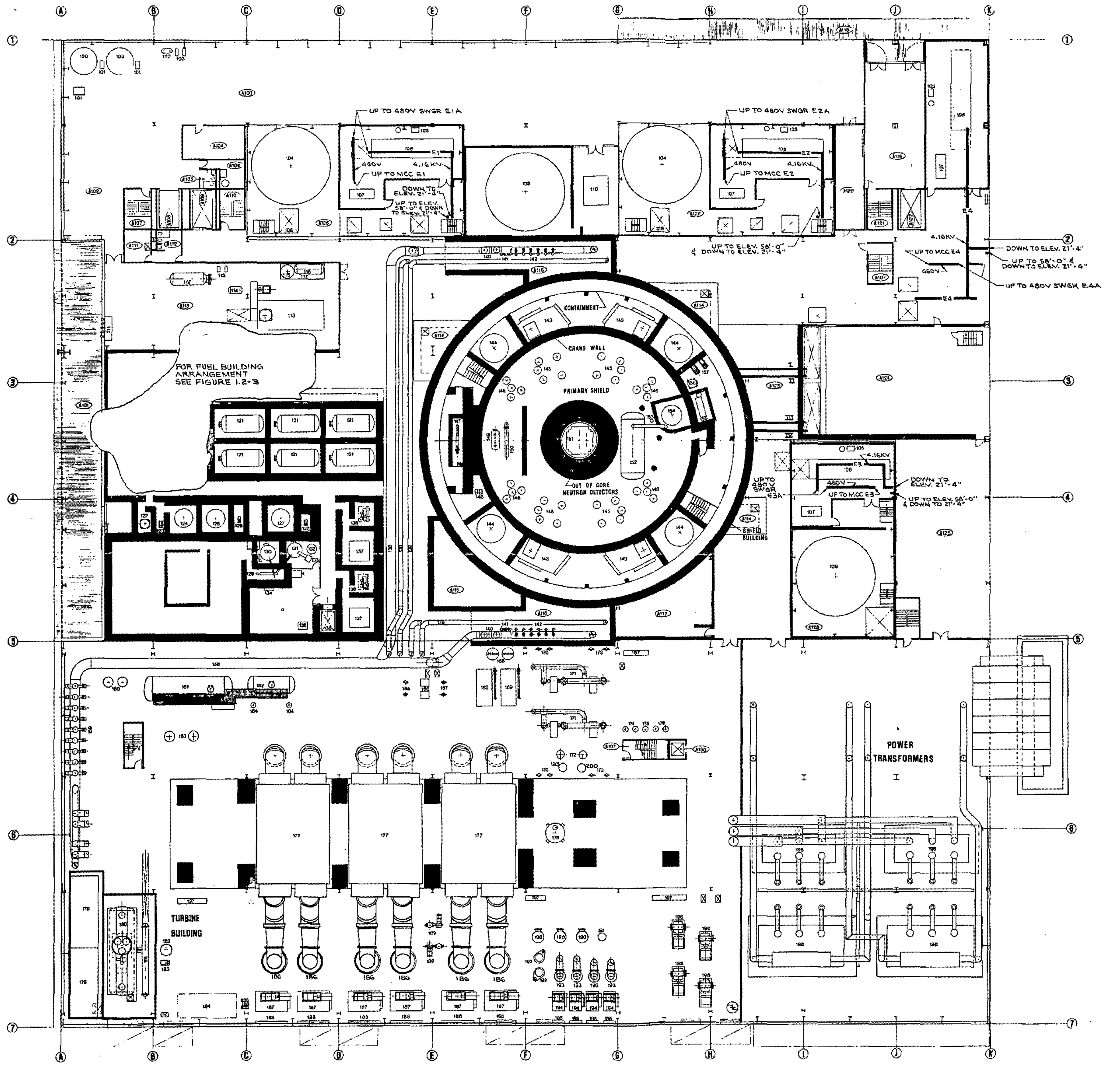


OFFSHORE POWER SYSTEMS

Title: FLOATING NUCLEAR PLANT

ELEVATION 21' - 4"

11/1/72 Figure 8.3-4



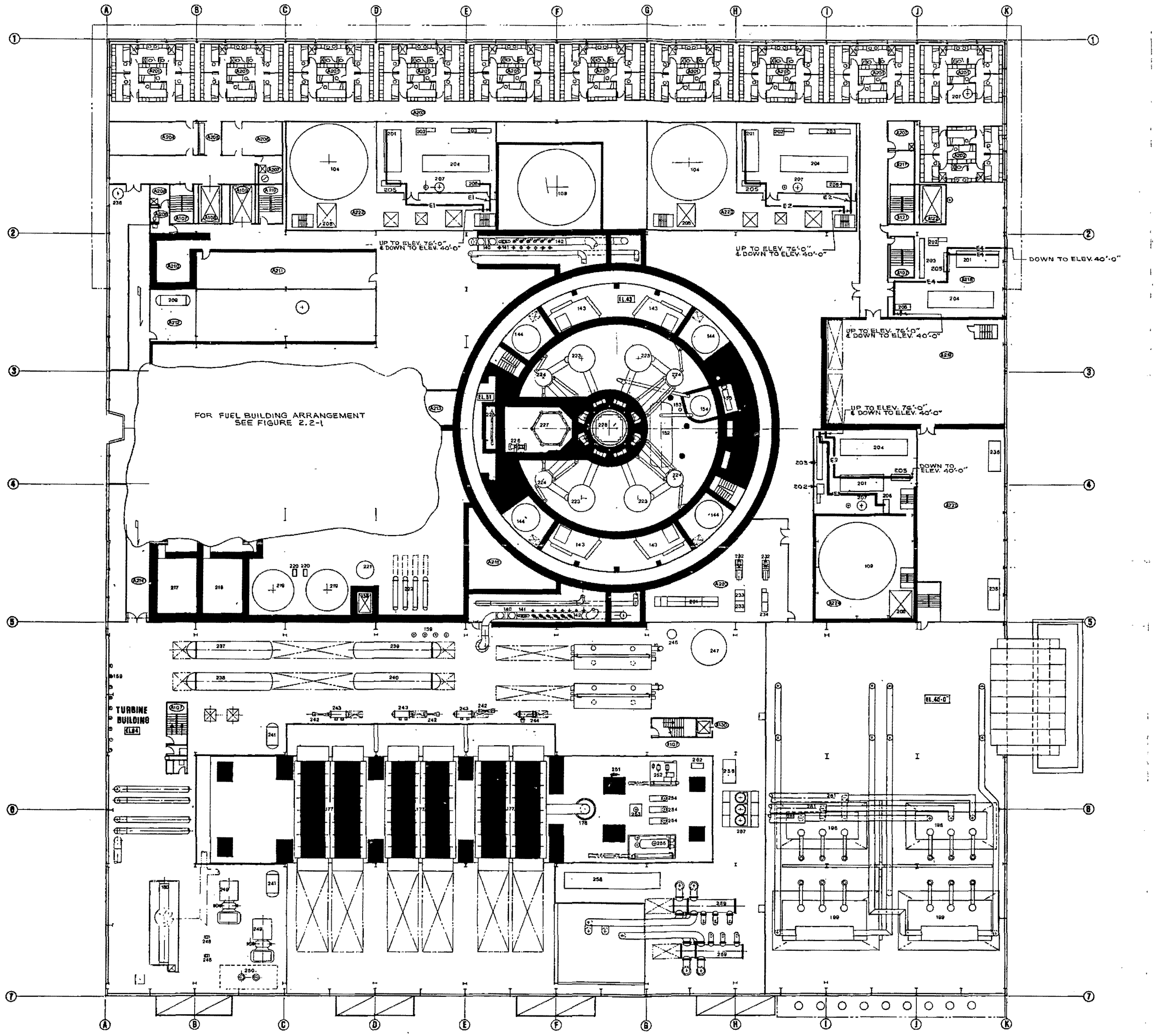
OFFSHORE POWER SYSTEMS

Title: FLOATING NUCLEAR PLANT

ELECTRICAL ARRANGEMENT PLAN

ELEVATION 40' - 0"

11/1/72 Figure 8.3-5



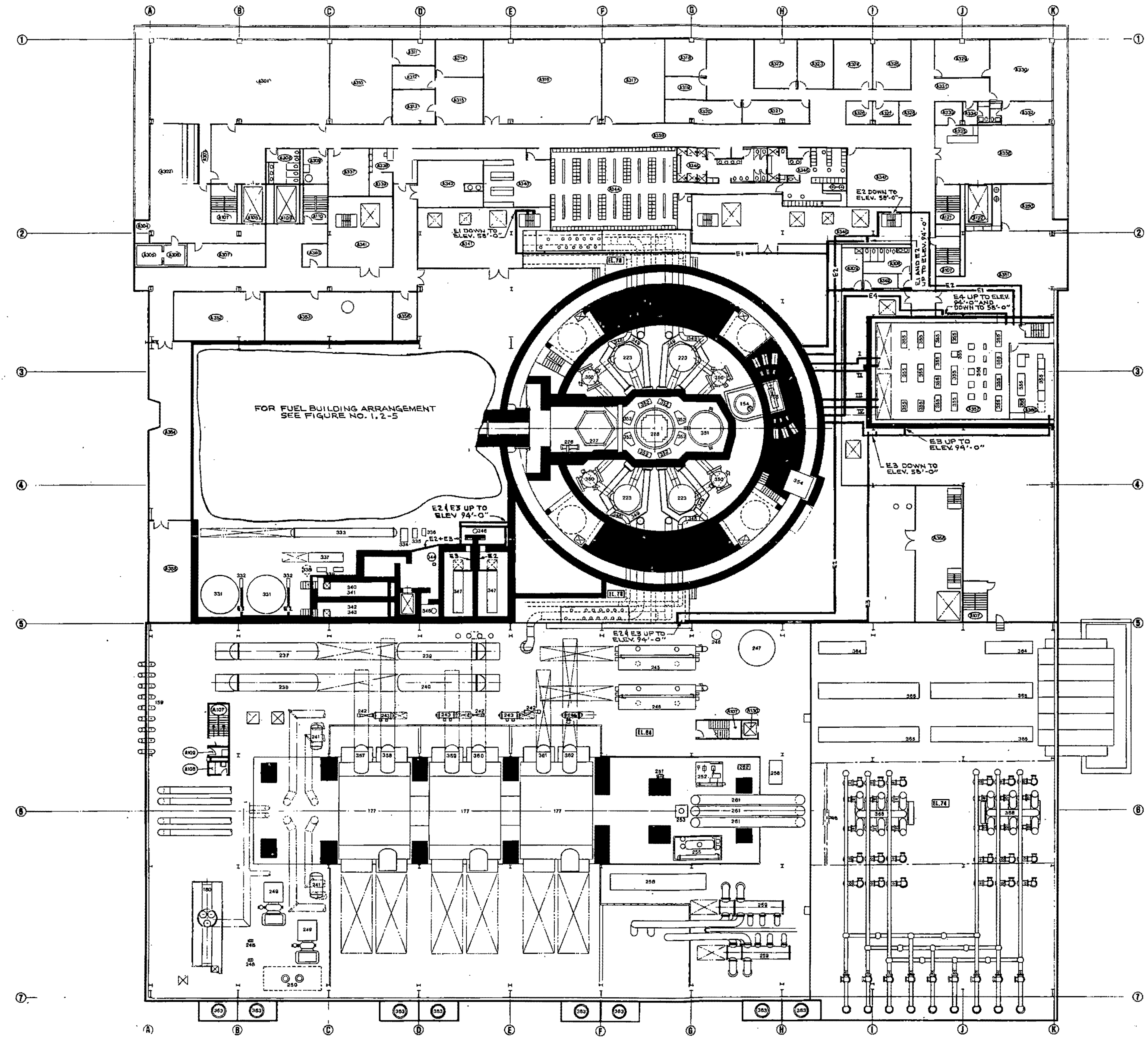
OFFSHORE POWER SYSTEMS

Title: FLOATING NUCLEAR PLANT

ELECTRICAL ARRANGEMENT PLANT

ELEVATION 58' - 0"

11/1/72 Figure 8.3-6



+ INDICATES A SAFEGUARD CHANNEL CROSSING OVER OR UNDER ANOTHER SAFEGUARD CHANNEL WITH REQUIRED SEPARATION DISTANCE PER PARAGRAPH 8.3.1.2.3.5

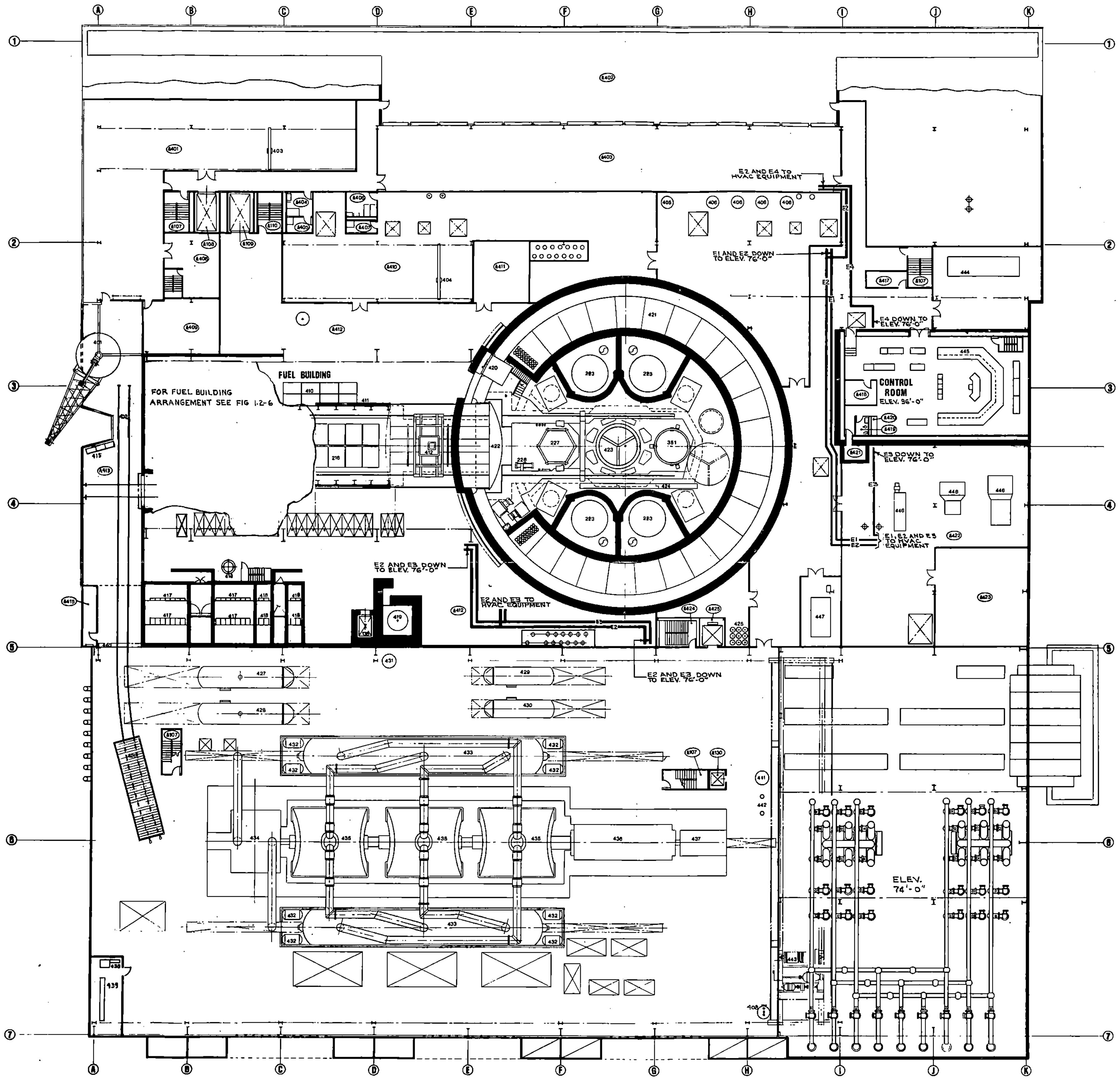
OFFSHORE POWER SYSTEMS

Title: FLOATING NUCLEAR PLANT

ELECTRICAL ARRANGEMENT PLAN

ELEVATION 76' - 0"

11/1/72 Figure 8.3-7



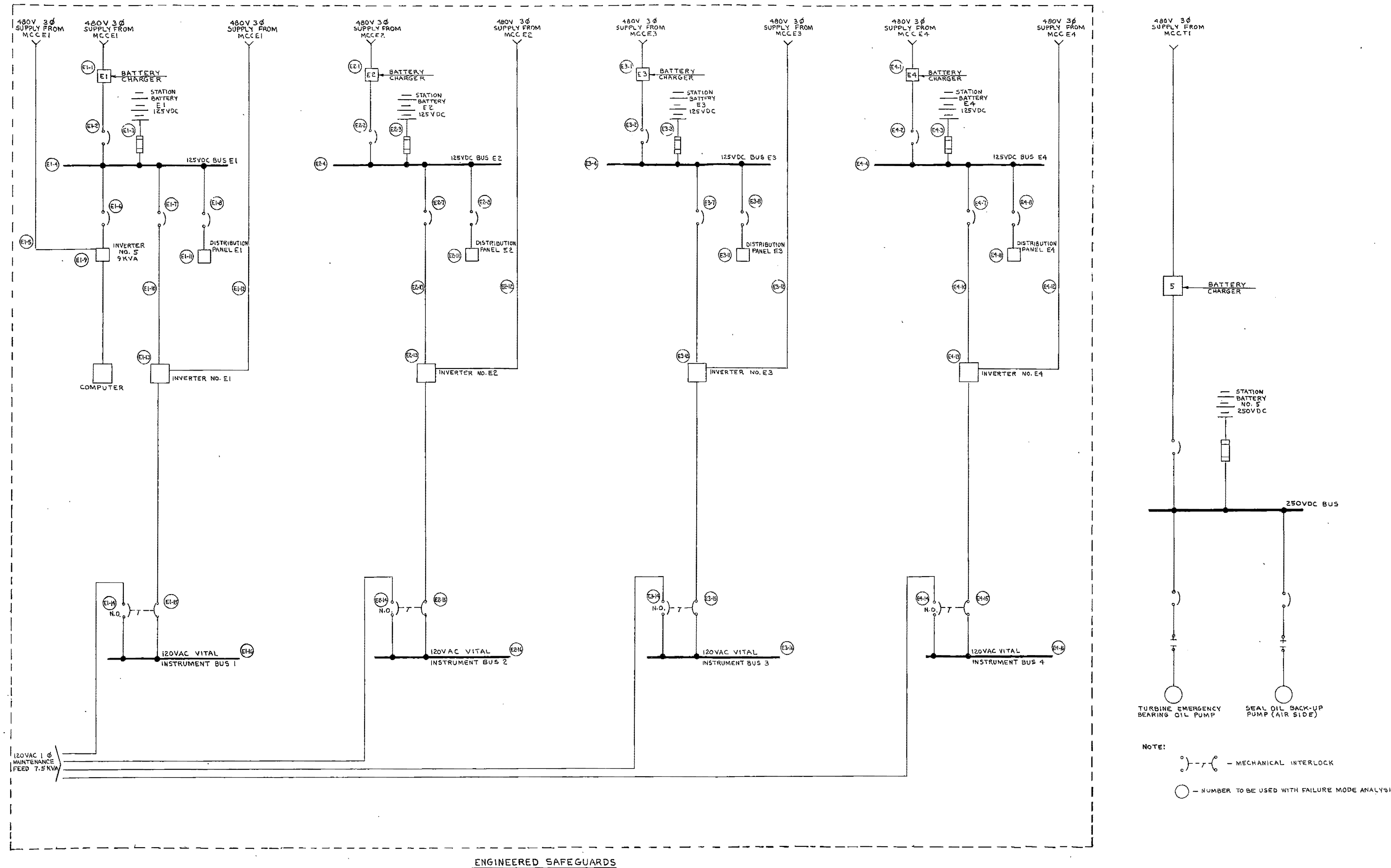
OFFSHORE POWER SYSTEMS

Title: FLOATING NUCLEAR PLANT

ELECTRICAL ARRANGEMENT PLAN

ELEVATION 94' - 0"

11/1/72 Figure 8.3-8



OFFSHORE POWER SYSTEMS

Title: FLOATING NUCLEAR PLANT ONE LINE DIAGRAM (DC)

1/1/73 Figure: 8.3-9